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US—USSR SYMPOSIUM ON FUSION-FISSION REACTORS

July 13-16, 1976

Hosted by
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PREFACE

Clyde E. Taylor, Symposium Chairman

This document contains presentations given at the *US-USSR Symposium on Fusion-Fission Reactors* held at the Lawrence Livermore Laboratory, Livermore, California, on July 13-16, 1976. Approximately 75 scientists and engineers attended. Nearly all of the work in progress in the US and USSR was reported at the meeting. The *Proceedings* give a good picture of the present state of understanding of fusion-fission hybrid principles.

In the time since the first meeting was held on this topic at ERDA headquarters on December 3 and 4, 1974 (ERDA-4, *DCTR Fusion-Fission Energy Systems Review Meeting* edited by S. Locke Bogart), much work has been done. The attractive fissile production rates, predicted by simple calculations, have proven to remain attractive when complications of engineering structures, cooling, and materials limitations are taken into account. Complex economics questions have been recognized and answered in some cases.

Considerable progress has been made in understanding the many engineering problems of using different types of complex fusion devices as neutron sources for breeding of fissile fuel.

In addition to the prepared presentations, there were four informal discussion sessions on "Fuel Cycles and Economics", "Blanket Designs", "Overall System Designs", and "Fusion-Fission Reactor Safety"; a summary of each of these discussion sessions, prepared by the respective chairman, is included in these *Proceedings*.

Participation by the USSR delegation was active, and the goal of information exchange between the US and USSR programs was achieved.

CONTENTS*

*For the following presentations, no written papers were submitted:

Philosophy of the Fusion-Fission System - A Discussion,
G. E. Shatalov (Kurchatov Institute of Atomic Energy)

Tritium and Inorganic Compounds of Lithium - A Discussion,
V. G. Vasiliev (All-Union Scientific Research Institute of Inorganic Materials). The speaker summarized some of the information contained in a recent literature review titled *Tritium in Inorganic Compounds of Lithium*, by V. G. Vasil'yev and Ye. V. Dmitriyevskaya, All-Union Scientific Research Institute of Inorganic Materials, Moscow (1976); ERDA-TR-218(1976).

MIT Fusion-Fission Studies,
L. Cook (MIT)

Laser Fusion Hybrid Reactor Design,
T. Frank(LASL)

INTRODUCTION AND OVERVIEW

THE ERDA FUSION-FISSION STUDIES PROGRAM

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ABSTRACT

The Energy Research and Development Administration, through the Division of Magnetic Fusion Energy, supports a fusion-fission studies program to evaluate the possible roles that fusion-fission applications may have in future national energy economies. This paper provides a brief synopsis of past studies, describes possible alternate roles for fusion-fission, discusses potential advantages and disadvantages, and describes the nature of and current funding of the ERDA-sponsored studies at the national laboratories and in industry. Discussions also include consideration of the types of information which will be needed to properly assess the viability of fusion-fission applications.

INTRODUCTION

For nearly a quarter of a century, this Nation has supported research and engineering to develop the fusion process as a valuable asset to our society. During the early years, the principal objective of the program was to develop a fusion process to produce fissionable materials for military applications. In 1958, it was determined that this objective was being met adequately by other means. There was also a growing appreciation of the very complex problems associated with harnessing controlled fusion as a neutron source. That year the program was declassified and, at the same time, a new primary objective was selected - the production of electrical power by fusion reactors. This objective is still the primary objective of the ERDA magnetic Fusion Energy Program.

From 1958 to about 1970, fusion-fission as a viable concept lay dormant while program emphasis was placed on the research required to produce controlled thermonuclear reactions on a laboratory scale. Early in this decade, as substantial advances were made in plasma experiments and the improved potential of fusion was recognized, interest in fusion-fission was also revived. Prior to FY 1974, the funds provided for fusion-fission systems studies were modest. In FY 1974, the funding level for these studies rose to about $250K per year. This rate was held constant through FY 1975, but rose to $750K per year in 1976 as substantial scoping studies were begun for tokamaks, mirrors, and theta pinch concepts. The recent growth in the funding level reflects the desire by ERDA to carefully assess the prospects for fusion-fission systems contributing to the solution of the Nation's energy problems. This assessment effort is particularly timely in the context of recent suggestions that fusion-fission systems may provide considerable relief to both the fissile fuel supply and fission product and actinide waste disposal problems. It has also been suggested that fusion-fission electric power plants may be technically and economically practicable and most of the recent ERDA supported studies focus on this particular application.

Figure 1.

POSSIBLE ADVANTAGES AND DISADVANTAGES OF FUSION-FISSION WITH RESPECT TO PURE FUSION AND PURE FISSION

POSSIBLE ADVANTAGES RELATIVE TO		POSSIBLE DISADVANTAGES RELATIVE TO	
FUSION	FISSION	FUSION	FISSION
• RELAXATION OF PHYSICS AND SOME TECHNOLOGY REQUIREMENTS	• CHEAP FUEL NOT REQUIRING ENRICHMENT	• ACQUIRES PROBLEMS ASSOCIATED WITH FISSION	• MACHINE DESIGN IS MUCH MORE COMPLEX
• HIGHER THERMONUCLEAR ISLAND POWER DENSITY	• IMPROVED FISSILE FUEL PRODUCTION RATE	• BLANKET DESIGN IS MORE COMPLICATED	• HANDLING OF LARGE AMOUNTS OF TRITIUM
• EARLIER AND MORE COMPETITIVE OPERATION	• SUBCRITICAL OPERATION	• COST MAY BE HIGHER	• MACHINE MAINTENANCE MAY BE MORE DIFFICULT
	• LOSS OF COOLANT MAY BE LESS PROBLEMATIC		• COST MAY BE HIGHER
	• LOWER FISSION PRODUCT INVENTORIES		

POSSIBLE ROLES OF FUSION-FISSION IN THE ENERGY ECONOMY

Invariably, the question is raised about what are the advantages and disadvantages of fusion-fission systems compared to either pure fusion or fission reactor systems. A definitive answer to this question cannot be given at this time because a detailed assessment of the technical, safety, and economic features of fusion-fission applications is not in hand. However, a few observations are in order on the conditions under which fusion-fission systems may be needed.

The current U.S. fission power economy, based upon low enrichment uranium fueled LWR's, is an economically attractive alternative for meeting a major portion of the Nation's near-term electricity needs. However, rapidly rising uranium prices may change this attractiveness as reasonably priced ores are depleted. Unless an alternate supply of fissile fuel becomes available, the LWR may well have only a transient role in the country's energy economy. Recycle of plutonium generated in LWR's could alleviate the problem somewhat, but not by many decades.

One possible answer to the LWR fueling problem is the fast breeder reactor. Indeed, if the fast breeder is found to be socially and economically acceptable by the U.S., then a major portion of the projected energy needs of the country can be satisfied for many hundreds of years. Numerous problems must be resolved regarding the LMFBR and its fuel cycle. These difficulties have spurred the interest of utilities to look for all possible alternate sources of fissile fuel breeding which could extend the useful lifetime of the already established LWR reactor industry and protect a major national investment in the fission power economy.

Because of the recent progress in fusion, the potential for fusion providing a copious source of neutrons is higher than ever before. This, coupled with the changed economic conditions of fissile fuel supply, has altered the perspective on fusion-fission significantly since the early years when fusion neutron sources were considered as a possible method for uranium and plutonium production for weapons applications.

Figure 2.

POTENTIAL ROLES OF THE FUSION-FISSION APPLICATION

1. FISSILE FUEL PRODUCTION FOR PURE FISSION BURNER AND CONVERTER REACTORS

2. ELECTRIC/THERMAL POWER GENERATION WITH SOME FISSILE FUEL PRODUCTION

3. FISSION REACTOR WASTE DISPOSAL BY FUSION NEUTRON IRRADIATIONS

There is no doubt that fusion-fission has potential for providing an option or backup to the fast breeder. The crucial questions are in what form (i.e., fissile breeding, central electric power stations, waste burners, hybrids) and on what time scale? What are the technical problems which must be solved? Under what conditions is a fusion-fission reactor likely to be a technically and economically attractive option to the breeder? How close is fusion to providing a cheap abundant source of neutrons for this purpose? What are the environmental and safety trade-offs that have to be made? What economic regimes would fusion-fission systems operate in?

As discussed in previous paragraphs, fusion-fission may have application in the areas of actinide and fission product disposal utilizing fusion neutrons, production of fissile fuel with low doubling times and/or electric power production at competitive costs. Some radioactive waste disposal schemes can involve large costs (such as rocketing wastes out of earth's gravitational field) or long-term containment uncertainties (such as storage of wastes in salt mines for several hundred thousand years). The desirability of a fission waste disposal system which could be contiguous to the waste producer in an energy park could be quite high in the future. However, preliminary studies performed to date with fusion applications have not demonstrated any major technical, economic, or safety advantages which indicate that the nation should rethink its near-term objectives for the disposal of fission Rad-wastes.

Production of large quantities of fissile fuel via fusion-fission would be valuable for continuing an LWR energy economy, or for providing the fissile inventory for a rapidly growing LMFBR economy. However, the past studies of fusion-fission for fissile fuel and/or power production have not formed a clear base for assessing this potential since they have not scoped the costs of such a system, nor have they designed a breeding blanket and associated safety systems in a level of detail which allows a judgement of the relative technical and safety merits of fusion-fission reactors.

These and numerous subsidiary questions must be answered before a major commitment can be made to pursuing a specific fusion-fission concept development program. A number of systems studies have been instituted by both ERDA and EPRI to shed light on these questions. In the next couple of days, we will hear papers and discuss results of these studies with a view toward assessing where the most promising avenues of future effort may lie.

INFORMATION BASE NECESSARY TO ASSESS HYBRID VIABILITY

The major objective of ERDA's fusion-fission systems studies program is to provide the technical and economic analysis necessary for assessing the proper role of fusion-fission technology in the ERDA R&D program. For example, it may become apparent at some future time that there is a need to institute a major hybrid development program. However, until sufficient background studies have been performed, and until existing laboratory experiments in the current Magnetic Fusion Energy Program provide the physics and engineering basis for a reliable, compact intense neutron source, it would be premature to initiate a large-scale development activity directed at fusion-fission. As noted in Figure 3, the fusion neutron source requirements are a strong function of the application chosen.

ERDA has adopted the following approach to obtaining the required fusion-fission assessment information. Market penetration studies are being performed at PNL to define the possible roles of fusion-fission devices in the energy economy in the next four decades. These studies assume various roles and energy scenarios for competitive energy suppliers, such as LMFBR's, LWR's, and LWR's with plutonium recycle; specify various energy demand levels; and then examine the share of the market captured by hybrids at discrete points in time as a function of fusion-fission costs. Hybrid parameters varied include specific capital cost, fissile fuel production rate, electricity cost, and electricity production rate. A major utilization of these studies is in the specification of technical and economic targets which the hybrid reactor designers must achieve.

Figure 3.

NEUTRON SOURCE REQUIREMENTS A STRONG FUNCTION OF SELECTED ROLE

- FUEL PRODUCTION → LOW Q, LOW FUSION POWER DENSITY
- POWER PRODUCTION → MODEST Q, MODEST TO HIGH FUSION POWER DENSITY
- FISSION WASTE DISPOSAL → MODEST TO HIGH Q, HIGH FUSION POWER DENSITY, VERY HIGH WALL LOADING

Specific reactor technology studies have also been initiated at PPPL, PNL, LASL, LLL, and MIT. Hopefully, these scoping studies will cover the more promising fusion confinement concepts and examine the most promising combinations for blanket geometries, thermal-hydraulics, neutron energy spectrum conversion, tritium breeding, and fissile fuel breeding systems for fusion-fission reactor design. These scoping studies should provide the basis for a preliminary assessment of the potential feasibility of a particular concept, and identify the need for more detailed reference reactor design studies. These reference reactor designs would examine in some engineering detail those subsystems which are critical to determining the technical, economic, and safety features of fusion-fission and provide the basis for an assessment of whether to proceed with a major development program for hybrids. We hope to accomplish this assessment at the end of FY 1977.

Figure 4.

FUSION-FISSION ASSESSMENT ACTIVITIES

- JANUARY 1974 — ERDA HEADQUARTERS MEETING – ERDA-4
- JULY 1976 — LLL MEETING – AN INTERIM REASSESSMENT
- FALL OF 1977 — A SUBSTANTIAL ASSESSMENT OF THE ROLE OF FUSION-FISSION IN THE NATIONAL ENERGY PROGRAM
- PERIODIC REASSESSMENTS THEREAFTER

PRESENT ERDA PROGRAM

The present DMFE hybrid systems studies program is in the initial stages of the global economics assessment and scoping studies phase. The December 1974 meeting at ERDA Headquarters constituted a first assessment of the possible roles and future directions for fusion-fission and served as a base for initiating the FY 1976 studies. This meeting at LLL will provide the basis for evaluating progress during the last 18 months and for determining the future course of ERDA supported activities regarding fusion-fission.

Figure 5.

PURE FUSION VS. FUSION-FISSION PROGRAM OBJECTIVES

- IN THE NEAR TERM, THE OBJECTIVES ARE THE SAME; WE MUST DEVELOP A COMPACT, INTENSE, RELIABLE, SAFE SOURCE OF NEUTRONS.
- ADOPTING FUSION-FISSION AS A MAJOR OBJECTIVE WOULD NOT CAUSE MAJOR SHIFTS IN PROGRAM ACTIVITIES FOR THE NEXT FEW YEARS.
- EXPLORATORY CONCEPTS ARE NOT EMPHASIZED IN THE ERDA FUSION-FISSION PROGRAM FOR THE SAME REASONS THAT THEY ARE NOT MAIN-LINE CONCEPTS IN THE PURE FUSION PROGRAM – THE PHYSICS, REACTOR ECONOMICS, AND/OR TECHNOLOGY PROBLEMS PRESENTLY APPEAR LESS PROMISING.

As mentioned earlier, the level of funding for fusion-fission is approximately \$750K for FY 1976, and is projected to increase to about \$900K for FY 1977. Present plans are to continue scoping and reference studies until promising design configurations are identified or we find that the technical, economic, or safety problems cast significant doubt on the future potential of fusion-fission. In the meantime, the physics results from PLT, ORMAK-Upgrade, Doublet III, and other experimental devices should be obtained, and will provide the input to the decision-making process regarding a major fusion-fission development activity. It should also be noted that in the near-term, the objectives of the fusion R&D program are consistent with the pursuit of fusion-fission as outlined in Figure 5.

Now, the fusion-fission studies funded in FY 1976 and FY 1977 will be described briefly. Each description contains the study objectives, an assessment of the study status, the group leader, organization, and funding level.

TCT HYBRID - FY 1976 FUNDING = \$285K

This is a scoping/reference design study of a beam-driven tokamak hybrid, with the fusion core effort headed by R. Mills at PPPL and the fusion blanket effort directed by R. Liikala at PNL. The study objectives are feasibility assessments of economical power and fissile fuel production in a small-scale tokamak. By the the end of FY 1977, the fusion core configuration will have been completed; blanket neutronics and thermal hydraulics will have been analyzed in detail; preliminary fuel cycle, safety, and cost analyses will have been performed; and we will have a good idea whether a driven tokamak hybrid has the potential of being technically and economically feasible, and whether there are nearer term applications of tokamak fusion systems to the fissile fuel shortage problem.

MIRROR HYBRID - FY 1976 FUNDING = \$290K

This is a reference reactor study of a classical mirror hybrid, with the fusion core effort directed by R. Moir at LLL and the fission blanket effort directed by C. Baker at GA. The study objectives include feasibility assessments of economic power and fissile fuel production from a hybrid with a classical low-Q mirror fusion driver. By the end of FY 1977, all major subsystems will have been analyzed, a cost-optimized fissile fuel cycle will have been chosen, total reactor costs will have been specified, extensive safety and environmental impact analyses will have been

Figure 6.

FUSION-FISSION STUDIES CURRENTLY SUPPORTED BY DMFE

STUDY TITLE	LABORATORY INVOLVED	LEVEL OF EFFORT (FY-1976)
TCT HYBRID	PPPL/PNL	$285K
MIRROR HYBRID	LLL/GA	$290K
LINEAR THETA PINCH HYBRID	LASL	IN-HOUSE
E-BEAM HEATED SOLENOID HYBRID	PI	$ 80K
LASER FUSION HYBRID	LASL	$ 55K
FUSION-FISSION SYMBIOSIS	MIT	$ 40K
HYBRID ECONOMICS	PNL	$ 90K

performed, and DMFE will be able to ascertain whether the spherical geometry and low-Q mirror device has promise of being an economical hybrid.

LINEAR THETA PINCH HYBRID - INTERNAL FUNDING ≈ $100K

This is a scoping study of a linear theta pinch hybrid directed by R. Krakowski at LASL. The study objectives are feasibility assessments of economic power and fissile fuel production under the conditions of plasma end losses characteristic of quasi-stoppered linear devices. By the end of FY 1977, different end stoppering techniques will have been analyzed, a preliminary cost analysis will have been performed for one fuel cycle, and we should know whether linear theta pinch hybrids of reasonable length have economic potential.

E-BEAM SOLENOID HYBRID - FY 1976 FUNDING = $80K

This is a scoping study of an E-Beam heated solenoid hybrid directed by S. Putnam at Physics International. The study objectives include determination of feasible regions of economical power and fissile fuel production for small linear systems heated by electron beams. By the end of FY 1977, a full parametric variation analysis of all subsystems including preliminary cost analyses should be completed, and should provide a basis for evaluating the need to proceed to a more detailed reference reactor design.

LASER HYBRID - FY 1976 FUNDING = $50K

This is a scoping study of a laser-heated pellet hybrid directed by L. Booth at LASL. The study objectives are feasibility assessments of economic power and fissile fuel production from a laser hybrid utilizing nearer-term low gain pellets. By the end of FY 1976, the preliminary design of a magnetically-protected cylindrical system should be well under way, and we should have some understanding of which of these two concepts, if either, should be carried forward into a more comprehensive reference reactor design. Future ERDA-sponsored systems studies of lasers will be supported by the Division of Laser Fusion Energy.

FUSION-FISSION SYMBIOSIS - FY 1976 FUNDING = $40K

This is a scoping analysis of a fusion-fission symbiote directed by L. Lidsky at MIT. The study objectives are to determine the advantages of a blanket which breeds fissile fuel with essentially zero fission and to ascertain whether a pure fuel breeder can be economical. By the end of 1977, sufficient information should be available to determine whether a reference reactor design would be worthwhile to undertake.

HYBRID ECONOMICS - FY 1976 FUNDING ≈ $80K

This is a market penetration study and cost-benefit analysis of hybrids directed by D. Deonigi at PNL. Its objectives are determinations of the roles, if any, which hybrids could assume in the energy market, with special emphasis placed on analysis of fissile fuel production, fissile fuel costs, power production, power costs, plant capital costs, and LWR and LMFBR economic parameters. By the end of FY 1977, target design parameters which the hybrid must achieve to play a role in the economy will have been assessed, as well as the assessment of the resource required for hybrid construction. This effort is of an ongoing nature, as the target parameters will vary as the prognosis for the future energy economy changes and as new design information for alternative energy systems evolves.

SUMMARY

To summarize, ERDA's fusion-fission efforts are presently concentrating on identifying and evaluating potential roles for hybrids in the future energy economy. Market penetration studies are identifying the economic regimes for hybrids. Potentially desirable fusion-fission confinement concepts are being examined in FY 1976 and FY 1977 through a combination of mainline concept scoping and reference reactor studies, including preliminary analyses of fusion-fission reactor safety features.

Before a major assessment of fusion-fission viability can be completed, scoping studies of non-mainline confinement concepts, comprehensive reference reactor designs of the most promising concepts, more intensive efforts in the areas of safety and reliability analysis, assessments of environmental impact, and a detailed specification of a hybrid development program including supporting technology development will be necessary.

These efforts, along with experimental physics progress toward production of D-T neutrons, should provide the basis for realistic judgement of the promise of fusion-fission. The scoping and reference design studies to be discussed here this week provide the basis for a continuing examination of the engineering and safety problems that must be clearly understood before ERDA, the utilities, and industry can seriously consider fusion-fission as a viable future energy option.

THE EPRI VIEWPOINT ON FUSION-FISSION

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ABSTRACT

The viewpoints of the Electric Power Research Institute (EPRI) and the utility industry on the development of fusion-fission energy systems are discussed. Specific fusion-fission projects supported by EPRI are described. The objective of the EPRI projects is to provide the utility industry a more realistic understanding of the advantages and limitations of the fusion-fission option to fission fuel production, power generation, and actinide waste burning.

Before discussing the viewpoints of the Electric Power Research Institute (EPRI) on fusion-fission energy systems, it would be helpful to review the nature of EPRI. The Electric Power Research Institute is a nonprofit organization that serves as the research arm of the electric utility industry of the United States. The 1977 budget is $200,000,000. R&D planning guidelines for the five year period through 1981 total $1,500,000,000. All funding is provided by voluntary contributions from more than 500 utilities that represent about 80% of the electric power generating capacity of this nation. These utilities are both investor and public owned. They vary tremendously in size. They include private utilities with both large and small systems, and public utilities that range from the mammoth Tennessee Valley Authority to the tiny rural cooperatives. The combination of this diversity within the utility industry, the primitative state of our knowledge on fusion-fission energy systems, and the large number of variations available for such systems mitigate against a single utility viewpoint on fusion-fission systems.

The utilities, however, do have a serious interest in obtaining as rapidly as possible a greater understanding of the potentials and drawbacks of fusion-fission energy systems. The goal of the EPRI work on fusion-fission is to provide the industry with the informational base for wise decision making. This desire on the part of the utilities for greater understanding of fusion-fission is a result of the promising potentials that such systems appear to offer. Specifically, they hold the hope for (1) the long-term availability of nuclear fission fuel with accelerated breeding from fertile materials including the U.S. reserves of thorium, (2) characteristics that might enable such systems to achieve significant environmental advantages over pure fission systems, and (3) the ability to burn actinide wastes more effectively than fission systems. To be yet fully evaluated are the economic tradeoffs. On the one hand, there are the inherent advantages of fusion-fission systems to more effectively use the high grade 14 MeV neutrons from fusion by producing both heat and storable nuclear energy. On the other hand are the possible added complexities from combining two nuclear technologies.

Based upon the potential advantages of fusion-fission systems, some general statements can be made regarding how various fusion-fission systems would be viewed by the utilities. If a fusion-fission system is designed to serve as a base-load electric power producer, utilities will consider it basically another fission breeder reactor. To gain substantial utility executive support, fusion-fission systems prepared for base load application would require <u>significantly</u> more favorable economic, environmental, and resource savings advantages compared to the fission breeders currently under extensive development and would have to be available well before economic pure fusion reactors are possible. This is a very difficult goal to achieve. However, it is a goal which should be explored because of the uncertainties still inherent in the alternative options and the claims by some of the economic and environmental advantages of such fusion-fission power plants. This was the apparent direction of the Soviet program based upon our mutual discussions at EPRI last year.

Nevertheless, most utilities would prefer a fusion-fission device optimized for fuel production that could operate offline. This would allow the utilities to place their base-load requirements upon nuclear fission and fossil fuel systems that have known economic, reliability and safety experience. Such fusion-fission systems (sometimes called symbionts) would provide a backup for the industry to the current uncertainties for an economic fission fuel supply - a need that is more critical than a new base-load breeder system. If a sudden U-235 cost rise or shortage developed for fission light water power plants, utility executive support for fusion-fission systems would certainly expand. The possibility that such plants could be operated by a separate business entity or the government to supply fuel for the utilities has an appeal. The nearer-term potential of fusion-fission systems and the experience they would provide for the operation of large fusion plasma systems could serve an important transition step from fission to pure fusion energy systems.

Workers on the fusion-fission option should give priority to obtaining a better understanding of research directions that can lead to improved environmental, operating and economic characteristics of fusion-fission systems. Such studies should seek ways to minimize the cost of the total development effort to a useful end product. In particular, they should point out how total R&D costs can be reduced by relying on elements that are common to the current pure fusion and fission development programs. In addition, thought should be given on how to penetrate the market by developing confidence on system performance. For example, can subsystem elements be tested separately and can the total system reliability and cost data be obtained through small scale system tests?

Another issue that is discussed in regard to fusion-fission systems is the possible diversion of fissile materials for use in explosive or radiological weapons. This is, however, not a question for the turn of the century. The problem is here today; nuclear proliferation is underway. In addition to the instability of multination possession of nuclear weapons, one must consider extremist groups and warped individuals. The human race must face the issue on a worldwide basis and quickly. Unless the world undertakes a wholly new path of cooperation, the moment of misfortune is coming. It will likely arrive long before fusion-fission systems. In the years ahead, proliferation will be independent of the energy source - fossil or solar energy can provide the electricity to drive new isotope separation techniques or power ion beams aimed at spallation targets to generate neutrons for plutonium production. Diversion of fissionable material is an immediate problem that encompasses far more than any single technology. Hopefully, mankind will give it the priority it deserves and solutions will be available well before fusion-fission energy systems could even become widely available.

In addition to the possibility of producing fuel for nuclear fission systems, fusion neutrons can be used to reduce the fission waste problem through transmutation. Such systems would be another "off-line" application of fusion. The Electric Power Research Institute is exploring this option through a series of projects. The requirements for accelerating the decay of both fission products and actinides are being studied. Those fusion transmutation systems that "burn" the actinides are essentially fusion-fission systems. Two EPRI studies on actinide "burners" have been completed and will be reported upon at this meeting.

Because of the relatively small amount of work underway to provide the utilities guidance on the merits of the fusion-fission options, EPRI has undertaken a number of projects. The goals are to explore the requirements of market penetration, evaluate the possibility of nearer-term systems, assess environmental aspects, define differences in materials and control requirements and in general to insert a greater degree of engineering realism into the subject to determine whether the proposed concepts are truly practical from an overall reactor engineering point of view. Such studies will point the way for future research. A brief description of each of the EPRI projects on fusion-fission energy systems follows:

RP268 - A two part study. The first was an analysis to define ranges of capital cost that allow systems producing fissionable material using fusion neutrons to penetrate the market place. Assumptions based upon utility inputs were used. The second was a study evaluating the long term possibility of actinide transmutation in a fusion reactor. Battelle Pacific Northwest Laboratory (9/74 - 11/75 $100K)

RP374 - As a task under a broad feasibility study of linear laser heated solenoid fusion reactors, self-consistent configurations were developed for fusion-fission designs. The final report is available as EPRI ER-171 dated February 1976. Mathematical Sciences Northwest, Inc. (12/74 - 9/75 ~$50K)

RP473-1 - A preliminary conceptual design of a demonstration fusion actinide burner for operation in the mid to late 1980's based upon the plasma physics of the Tokamak Fusion Test Reactor. Project would set lower bound on the nearer-term possibilities of fusion actinide burning using tokamaks. Westinghouse Electric Corporation (5/75 - 4/76 $450K)

RP473-2 - A preliminary conceptual design of a demonstration fusion device for fission fuel production for operation in the mid to late 1980's based upon the plasma physics of the Tokamak Fusion Test Reactor and the Soviet T-20 devices. Project would set lower bound on the nearer-term possibilities of using fusion neutrons to breed fissionable fuel from tokamaks. Westinghouse Electric Corporation (5/76 - 4/77 $500K)

RP236 - An add-on portion of this project will analyze the safety aspects of fusion-fission energy systems. U. of California, Los Angeles (5/76 - 5/77 $90K)

RP237 - As a portion of a large study on laser initiated inertial confinement fusion a task for the second year will be a scoping study of a fusion-fission energy system using inertial confinement. U. of Wisconsin (1/76 - 6/78 ~$280K)

RP546 - An add-on portion for this project is a study of certain control aspects of fusion-fission systems. This will give insight into whether the combination of two nuclear technologies results in greater complexity in system control. The Charles Stark Draper Laboratory & Massachusetts Institute of Technology (6/76 - 10/77 $60K)

RP472 - As part of a much larger systems analysis of trade-offs on the fusion reactor first wall and blanket, a structural life assessment for fusion-fission systems will be made. This will give insight into the advantages possible from reduced first wall loadings in fusion-fission systems. McDonnell Douglas Astronautics Company-East (12/76 - 9/78 ~$80K)

Planned - As part of a larger project on the definition and conceptual design of small fusion reactors, a design will be carried out to define the possibility of obtaining useful outputs from a mirror fusion-fission system that could be built and operated in the late 1980's. Project would set lower-bound on nearer-term possibilities of mirrors. Lawrence Livermore Laboratory, Pacific Gas & Electric Company, and General Atomic Company (1/76 - 12/76 ~$120K)

Planned - An agreement between the Electric Power Research Institute and the Kurchatov Institute, USSR, leading to the testing on T-20 of fusion-fission modules is under negotiations. The first stage would be the appointment of a Joint Working Group to prepare an overall project plan. Electric Power Research Institute/Soviet Joint Program (1976 - $78K - Joint Working Group)

The availability of a continuous and plentiful energy supply is crucial to provide stability and meet minimum living standards for the people of the world. There are few options - none assured and each with its drawbacks. Fusion-fission energy systems are one important contender among these options. We badly need a more realistic understanding of the advantages and limitations of this option. This understanding should not be blocked by preconceptions of the outcome or apprehensions as to its threat to existing or developing alternative options.

MIRROR
FUSION-FISSION
REACTOR DESIGNS

MIRROR FUSION-FISSION REACTOR DESIGNS*

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ABSTRACT

The mirror fusion-fission (hybrid) reactor is seen as a producer of fissile fuel (2500 kg/yr of ^{233}U or ^{239}Pu). The use of a mirror machine as the source of fusion neutrons has some very desirable features, e.g., steady-state operation, high power density, negligible impurity control problem, and relatively small and simple geometry. The low Q values ($Q = P_{fusion}/P_{injection}$) of a mirror machine are largely overcome by the large energy multiplication in the blanket. Because it uses near-term fusion technologies, the hybrid itself is a near-term reactor, especially when compared to power-producing fusion reactors. The combination of the hybrid fuel producer plus present-day fission reactors constitutes a "long-term" power source just as is claimed for the breeder reactor, but with some very advantageous differences. Present cost estimates suggest that the hybrid reactor could produce Pu for about \$60/g and ^{233}U for about \$130/g.

This paper discusses the role of hybrid reactors and compares the mirror approach with other approaches. Past and present mirror hybrid studies are reviewed. A comparison between the hybrid and fusion reactors is made; the conclusion is that hybrids are technologically less demanding in many ways except in dealing with fission wastes. Finally, the development steps that might lead to a commercial hybrid are discussed.

INTRODUCTORY COMMENTS ON THE MIRROR FUSION-FISSION (HYBRID) REACTOR

Early in the program to develop fusion power, people realized that the 14-MeV neutron resulting from the fusion reaction $D + T \rightarrow {}^4He + n$ could be used to induce fissioning of uranium; the result would be a greatly increased energy yield. Thus began the hybridizing of fusion and fission. The designer of a fission assembly given extra neutrons has two choices:

- If he provides for more fissioning, extra energy is released.
- If he provides for more neutron capture in fertile material (^{238}U, ^{232}Th), extra fissile fuel (^{239}Pu, ^{233}U) is produced.

Thus, hybrid designs can emphasize either power production or fissile-fuel production. We have studied hybrids at both ends of the spectrum of energy production (40 times the fusion energy release and little fuel production) and fuel production (3 times the fusion energy release and 0.7 ^{233}U atoms per fusion event or 10 times the fusion energy releases and 1.7 ^{239}Pu atoms per fusion event). Our studies suggest that fuel production may be more economical than energy production.

A simple but powerful argument will help illustrate why power production is not the best goal for hybrids. The argument goes as follows: The cost of electrical power for very capital-intensive plants (where cost of capital >> fuel-cycle costs) is proportional to the capital cost per unit electrical power, C/P_e. Now, although the cost of a fusion reactor, C, is increased only modestly[1] (by less than a factor of 2) by the addition of a fission blanket, the power is increased by a large factor (5 to 10 or more). Thus, $C/P_{e\ hybrid} \ll C/P_{e\ fusion}$. The hybrid reactor is a fission reactor with the complicated addition to the inside of the fission core of a magnetic field and a vacuum system with all the fusion plasma paraphernalia. The cost must thus be significantly higher for the hybrid than for the fission reactor. Hence,

$$C/P_{e\ fission} \ll C/P_{e\ hybrid} \ll C/P_{e\ fusion}. \tag{1}$$

Our studies, which are based on rather crude models of fusion systems, provided quantitative verification of Inequality (1); we therefore conclude that hybrids can not compete as power producers with fission reactors if fissile fuel is not too costly.

Uranium resource studies predict that the fuel cost for fission reactors will eventually rise significantly compared to interest on capital. If so, hybrids as power producers may compete more favorably. However, fuel-producing hybrids and other fuel-producing methods discussed below will set an upper limit on fuel-cycle costs.

Inequality (1) does not bode well for fusion reactors becoming economical power producers and poses multiple challenges to designers of fusion reactors:

- To raise the power-handling capability of all fusion components;
- To exploit to the fullest the potential for fusion to be inherently less hazardous from a radiological point of view than fission by some orders of magnitude; and
- To exploit the potential of low fuel-cycle costs for fusion.

Present-day studies that have shown the significantly lower radiological hazard associated with fusion reactors have not shown concommitant cost reductions. Also, because the power density in fusion reactor blankets is limited by potential structural radiation damage, power density is not only lower than in fission reactors but also is very low by comparison when averaged over the reactor as a whole (inside the magnet).

We find that hybrids may indeed produce ^{239}Pu and/or ^{233}U at a cost that may be of commercial interest. Alternative sources of the fissile fuel needed to make fission a long-term energy source are

- Fuel produced by mining uranium and separating the ^{235}U isotope,
- Fuel produced in a breeder reactor; and
- Fuel bred either from neutrons produced by accelerated protons or from deuterons striking a target.

The cost of the fuel produced from the first method will increase with time due to depletion of higher-grade ores. Although mining is the only fuel source today, it is only a question of time until this method will become uneconomical. The second source, the fission breeder, is nearing the commercial point, with the main issues being safety and economics. The last source, fuel bred by neutrons, is an option that depends mostly on economics. With a modest extrapolation from existing technology, the accelerator technology would be available. Fuel from a hybrid reactor plus that bred by a fission reactor or a neutron accelerator (if the latter two methods materialize) could lower the price of ^{235}U by isotope separation due to competition.

In comparison to the breeder, the hybrid fuel producer has some significant differences that may be major advantages:

- The hybrid can produce fuel for 5 to 10 fission reactors of the same thermal rating.
- The hybrid can be located away from electrical load centers (the breeder must be located relatively near the load center).
- The fission reactors making up the bulk of the power-producing system are already developed [the light-water reactor (LWR), heavy-water reactor (HWR), and the high-temperature gas reactor (HTGR)], and expansion could be rapid especially because the hybrid needs no enriched fuel.
- Enough fuel can be bred from one hybrid to provide the initial inventory for one fission reactor per year.

The mirror concept for the fusion component of the hybrid offers a number of advantages over alternative concepts. Steady-state operation, high fusion power density, and relatively small size and simple geometry (spherical) are highly desirable features not shared by Tokamaks, θ-pinches, etc. We can make the general comment, however, that any fusion concept that comes close to meeting the Lawson criterion of $n\tau \approx 10^{14}$ cm^{-3} (more precisely $Q \approx 1$, ($Q = P_{fusion}/P_{injection}$) would be a candidate for a hybrid reactor. For example, the beam-driven Tokamak[2] may be a good candidate for a hybrid.

DESIGN PROBLEMS

The problems facing the designer of the hybrid fall into three categories: technological feasibility, safety, and economics.

TECHNOLOGY

Technological feasibility of the hybrid reactor rests on obtaining fusion plasma conditions such that $Q \approx 0.5$. For lower values of Q, recirculating power begins to dominate the economics; for higher values of Q, recirculating power contributes very little to the cost. The same situation occurs for the mirror fusion reactor at Q's about 5 times higher.

SAFETY

Any situation that can lead to the release of radioactive nuclides to the environment is considered a safety problem. Under normal conditions, very little radioactive release is expected from a hybrid reactor. Abnormal or accidental releases form the crux of the safety problem. Such accidents usually involve melting the fuel. However, accidents caused by a nuclear excursion (criticality accidents) are not possible for our hybrid designs because the infinite-medium neutron multiplication constant is less than unity. This is a major advantage over fission reactors, particularly the breeder reactor. Melting of fuel due to a loss of cooling (LOC) is the most serious safety issue with the hybrid. Due to the thin wall between the fuel region in the blanket and the plasma and due to the complex geometry, LOC accidents are more likely for the hybrid than for fission reactors. Studies of the primary cooling systems with their auxiliary backups indicate that the hybrid can be designed safe enough to be licensed.[3] We should point out that the risk to the public can be considerably reduced by placing the hybrid at a location remote from population centers.

The issue of diversion of radioactive elements from the hybrid should not differ from that for the breeder unless the ^{232}Th-^{233}U cycle is employed; in that case, the hybrid would pose significantly less danger of diversion.

Some argue that hybrids should be measureably safer than fission reactors before they become very interesting. This is probably so if the hybrid is viewed primarily as a power producer; however, as a fuel producer the hybrid itself need not be safer than a fission reactor if the hybrid plus the fission reactors that burn its bred fuel are safer as a system than the alternate system (the breeder). Thus, some additional hazard could be incurred to obtain the necessary fuel to supply converter fission reactors which due to fuel scarcity would otherwise be breeders. The relevant hazard question is: Is one (remotely located) hybrid plus half a dozen converter reactors less hazardous than a like number of breeder reactors?

ECONOMICS

The question of economics boils down to the cost of producing Pu or ^{233}U per gram and a comparison of that cost to that of alternative sources. Our studies[4] show production costs of \$60/g for Pu and \$130/g for ^{233}U. These numbers are highly uncertain due to uncertainties in fusion technologies, particularly those not yet demonstrated. Our studies do, however, show the relative importance of the many details of the design. If true, these cost projections would lead to a significant penetration of the market by hybrid reactors if breeder reactors either are not developed successfully or are developed at too high a cost.[5]

Although our studies emphasize fuel production, we find that the economics are strongly improved by converting the heat generated by the nuclear reactions in the blanket to electricity. At the least, this electricity can run the plant; possibly, there would be modest amounts left for sale. (The Pu producer has an overall thermal-to-electrical conversion efficiency of ~20%, whereas the ^{233}U producer falls just short of being a net power producer.) We have investigated the concept of a fusion electric breeder in which electricity is supplied to a driven fusion reactor to produce fusion neutrons for fuel production without generating any electricity. Although this is a significant simplification of the blanket design, it leads to higher fuel costs by about \$50/g for Pu for our low-Q concepts ($Q \leq 1$). Thus, we find that the fusion electric breeding concept makes sense only for high-Q ($Q > 3$) fusion reactors.

Our cost projections are sufficiently low to encourage us to proceed with more studies, knowing full well that the fusion components are being vigorously pursued in the fusion program and likely will be available in one decade (as a spin off) for the hybrid.

REVIEW OF PREVIOUS WORK ON THE MIRROR HYBRID

Work on the mirror fusion-fission reactor concept began in 1972 with the work by Lee[6] on the fast-fission blanket. From his blanket neutronics calculations, which included various structural fractions, uranium thicknesses, and Li thicknesses, Lee concluded that 1.7 atoms of Pu and 140 MeV could be produced per fusion neutron while at the same time tritium was being bred. Hansborough and Werner[7] designed a modular blanket which incorporated a thermal lattice; when fitted together, the modules formed a geodesic shell.

Then a joint[8] Lawrence Livermore Laboratory-Pacific Northwest Laboratory (LLL-PNL) study was carried out in which the modularized blanket design was integrated with the Yin-Yang magnet of the fusion system. Slots in the blanket allowed for plasma leakage, and ports provided for neutral-beam injections. System performance studies showed that the plant efficiency was 32%, with 25% of the gross electrical power recirculated. The blanket was based on the PNL[9] concept of a thin uranium convertor zone (sometimes called a fission plate) followed by a graphite-moderated thermal lattice and tritium-breeding zones. The energy of the 14-MeV neutron was multipled by a factor of 38. Some thermal aspects of design were studied. However, the study had certain deficiencies:

- The blanket at room temperature was near criticality,
- The structural design of the blanket did not allow for containment of the pressurized helium coolant,
- No means was provided for routine removal of the blanket modules,
- No economic analysis was made, and
- The blanket used enriched fuel.

In spite of the deficiencies, the conclusion was that a fission blanket could produce fissile fuel and greatly increase the energy yield of the non-breakeven fusion machine.

Other analyses included some parametric system studies by Moir[10] that showed the quantitative reduction of $n\tau$ allowed by the fission blanket and the importance of low injection energies (100 keV). A study of safety by W. C. Wolkenhauer *et al.*,[11] showed that the radioactivity associated with the hybrid was only slightly less than that of a fission reactor of the same rating and that the large thermal capacity of the low power density (6 $W \cdot cm^{-3}$) graphite thermal lattice greatly reduced the problems usually associated with LOC accidents.

In a point design made in 1975 by Moir *et al.*,[12] a number of the earlier deficiencies were remedied. The design, based this time on the fast-fission blanket, was a conceptual design of each of the fusion and fission components (magnet, injector, direct converter, vacuum system, blanket, cooling system, and the balance of the plant). The design incorporated a removal scheme for the blanket modules in which the blankets were withdrawn in a linear motion along rails through vacuum locks. The mechanical design of the blankets was based on the concept of pressure cylinders with a hemispherically shaped first wall. Both one-stage and multistage plasma direct converters were studied: a one-stage converter was shown to be more economical than no converter, but no additional advantage was found for multi-stage converters. A detailed conceptual design of a 100-keV neutral-beam injector with direct conversion of unneutralized ions was made for the first time (see following discussion of the injector system). A fuel cycle analysis and a study of the plasma physics design were also included. It was determined the reactor would produce 600 MWe at a cost of $2000/kWe and 700 kg of Pu/yr at a cost of $120/g. However, there were several deficiencies in the design:

- The blanket (highly nonspherical and hence not equidistant from the roughly spherical plasma), due to its design requirement to be easily removable, had a large power variation (high ratio of peak to average power),
- The fuel cycle was not optimized,
- The magnet design was not optimal,
- The plasma parameters were not optimal.

Even with these deficiencies, the design was the first hybrid design done in enough detail to justify a cost estimate. Although by our standards the results were not economical, with the design improvements we could see, the potential for economic fuel breeding seemed good. Fuel production, not power production, was seen to be the goal of this hybrid.

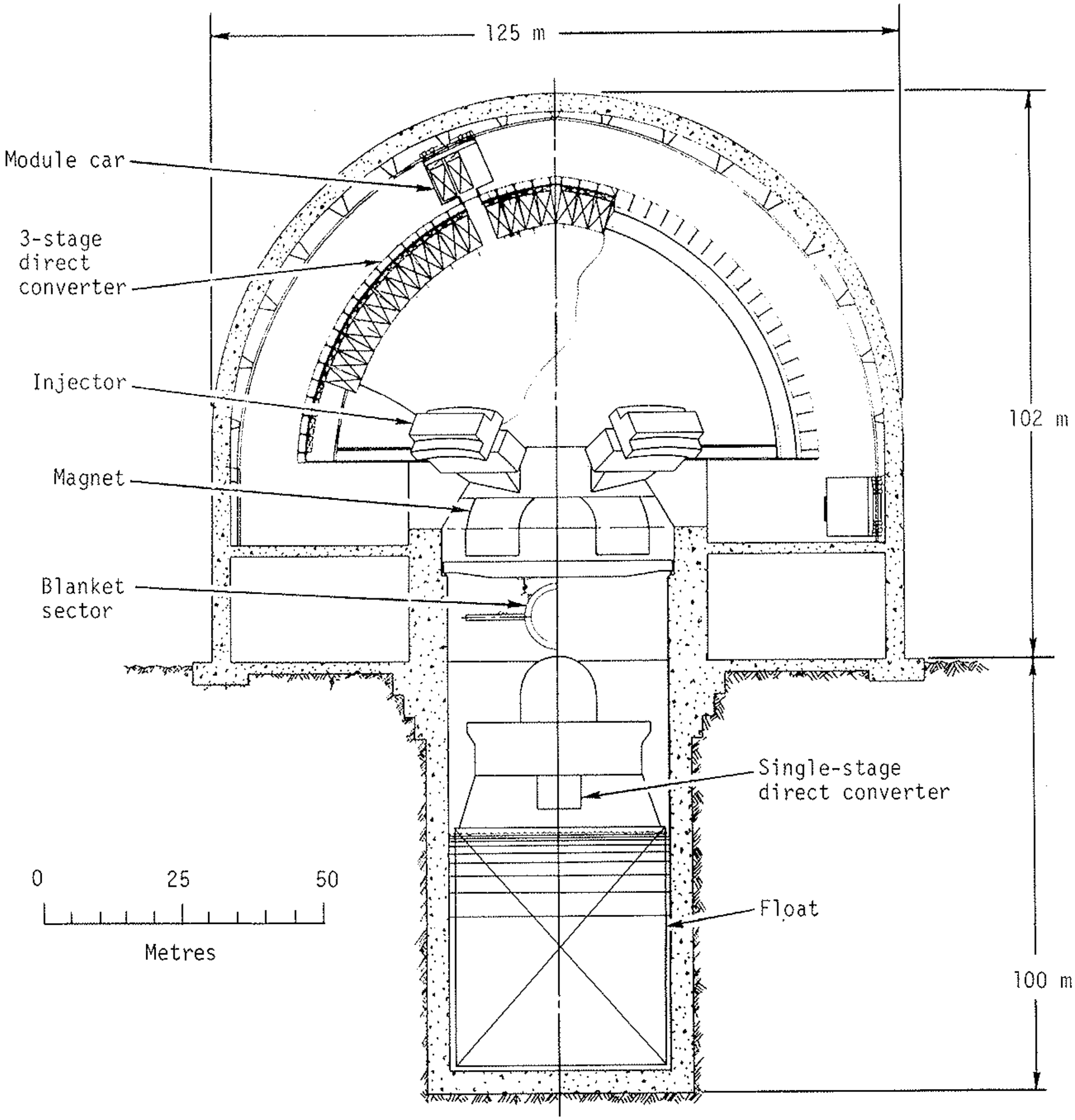

Fig. 1. Mirror fusion reactor.

REVIEW OF PRESENT DESIGN STUDIES

OVERALL DESCRIPTION

Previous design studies[12] had shown a movement towards practicality and near economical operation (within a factor of 3 of a fissile fuel cost that seems economical for fission-reactor fuel). The present studies have capitalized on design optimization and on a new design idea, namely, a design in which the two halves of the magnet are oriented vertically, with one half resting on a float so that the magnet can be opened up to give access to the blanket.[13] As shown in Fig. 1, the reactor is built on a dry dock. The blanket can be handled with an overhead crane and can be made in the spherical shape of previous designs. The blanket is shown in Fig. 2. By greatly reducing the peak-to-average power ratio, the power can be raised by a factor of 4 at the same peak power density as the earlier design. Lee[14] has made more detailed designs of the blanket.

Bender[4] has made an economic analysis of a nuclear park composed of burner reactors consuming the fissile fuel produced by the hybrid reactor. The nuclear park economic model allows optimization of such parameters as mirror ratio (ratio of magnetic field at the mirrors to central vacuum magnetic field) fuel burnup, injection energy, etc.

Table 1 gives some parameters for the mirror hybrid design. Both Pu- and ^{233}U-producing versions are shown.

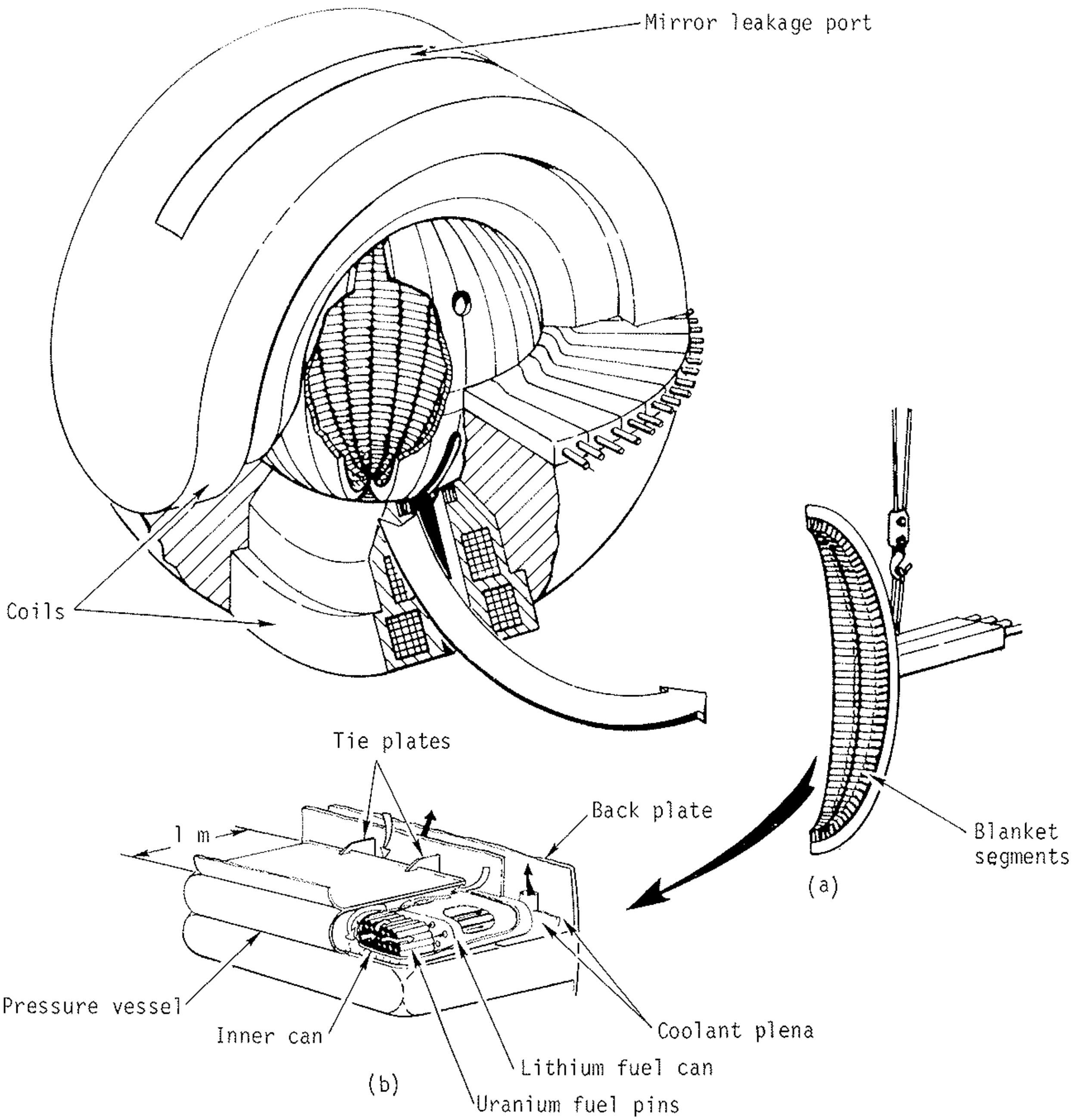

Fig. 2. Blanket, showing (a) segment of a spherical shell or "lune", and (b) detail of the blanket submodule.

Table 1. Fusion-fission reactor parameters.

Parameter	Pu production	^{233}U production
P_{fusion}	470 MW	1500 MW
$P_{thermal}$	4300 MW	3600 MW
$P_{net\ electric}$	1040 MW	-40 MW
Fuel production	2400 kg/yr	2600 kg/yr
Cost	2.3 B$	3.3 B$
Fuel cost	55 $/g	127 $/g

INJECTOR SYSTEM

The injector system for the hybrid[15] is based on 100-keV positive (D^+, T^+), ions and includes direct conversion of unneutralized ions. An experimental program to develop the injector and direct converter is underway at LLL-LBL; 1-MW modules will be tested this year. The efficiency of producing neutral-beam power from electrical power was determined to be 70%.

One bank of injectors producing 60 MW of neutral beam is shown in Fig. 3; a single-beam injector module is shown in Fig. 4. Figure 5 shows detailed injection paths.

In our earlier study, we had determined the plasma density profiles on each of the field lines. The penetration of the plasma by each of three rays representing the neutral beam from the injector system was studied and found to be consistent with the assumed radial density profile but with a noticeable density depression at the center. If the density were slightly higher, the beams would not penetrate the plasma very well. At the higher density of our recent reactor designs (1.7 times the previous design), the injection either must be nearer the mirrors where the plasma is both thinner and less dense or it must rely on inward convection (which is predicted when the density gradient is radially outward) as

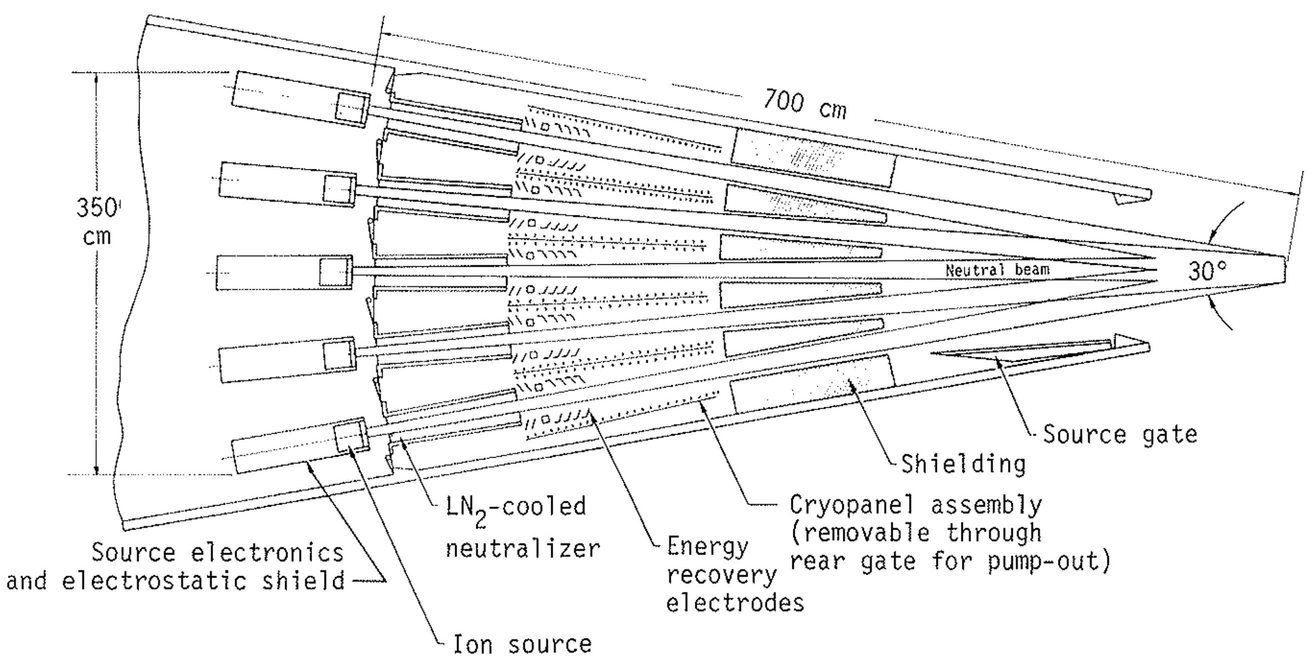

Fig. 3. Neutral-beam injector array.

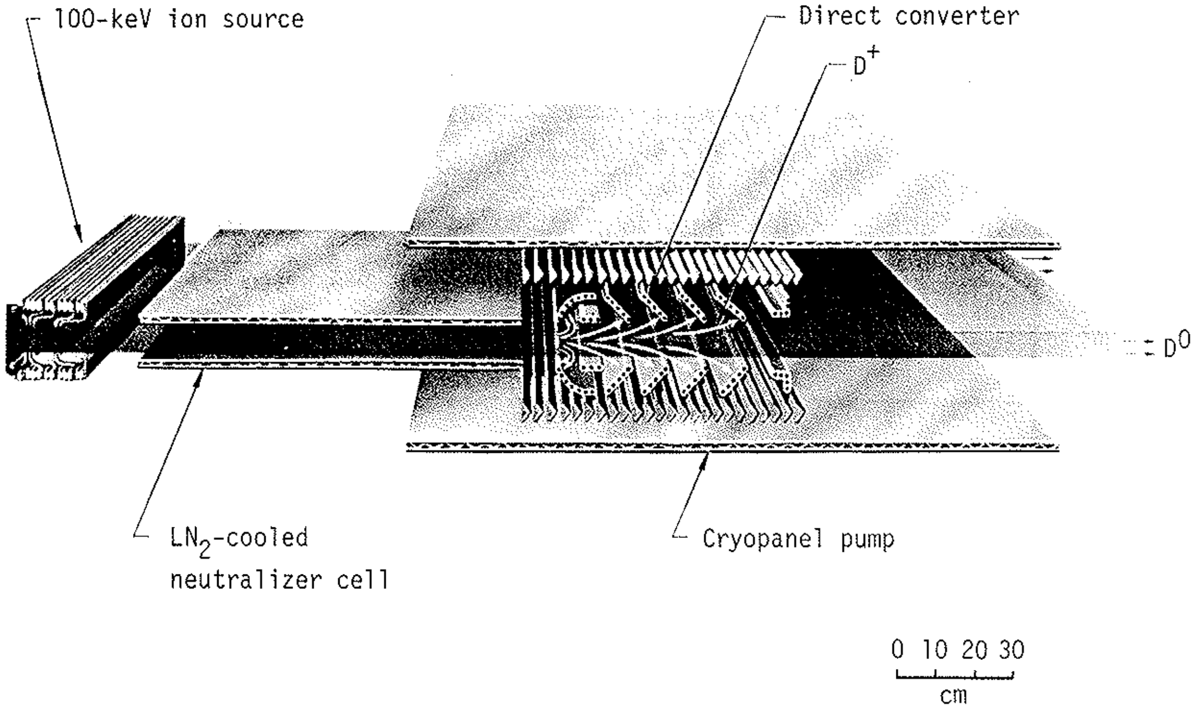

Fig. 4. 100-keV, 10-A neutral-beam injector system.

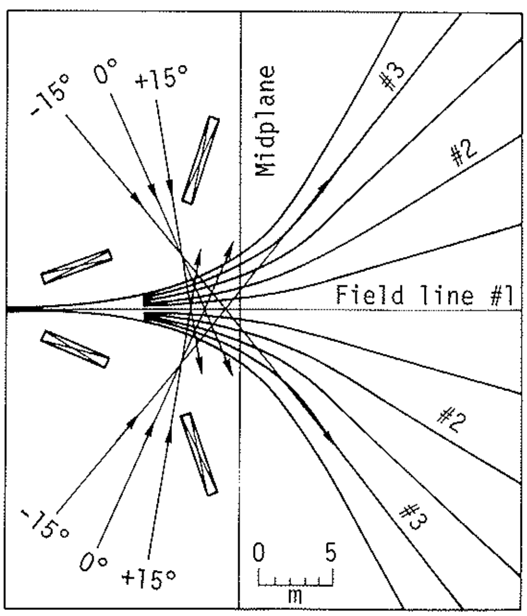

Fig. 5. Injector angles and field lines used in the Fokker-Planck calculations.

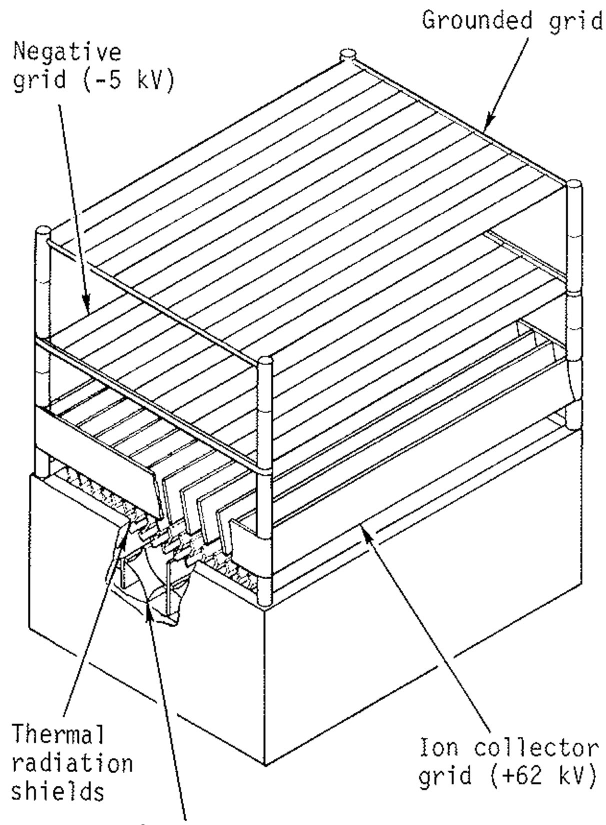

Fig. 6. One-stage, direct energy converter module.

is now being assumed in the mirror fusion reactor design study.

DIRECT ENERGY CONVERTER

The direct converter in the current hybrid reactor design is the one-stage, modular type shown in Fig. 6. The efficiency at 100 keV is expected to be 52%.

TECHNOLOGY ASSESSMENT

After comparing the technological requirements of the mirror hybrids with those of the mirror fusion reactor, we conclude that the technology is much less demanding for the hybrid and (importantly) the time for its development is much shorter. In fact, most of the fusion technologies for the hybrid will be developed for near-term fusion reactor prototypes. The technology requirements discussed below are summarized in Table 2.

MAGNET

For the hybrid design, the magnetic field strength at the conductor is 8 T. There is only modest value in raising this field strength, whereas there is a strong incentive to employ fields up to 16 T for the fusion reactor. The hybrid can employ Nb-Ti conductor. This conductor is already being used for somewhat lower fields and in smaller sizes, and it will be available at full requirements within a decade [on the same time scale as the Fusion Engineering Research Facility (FERF)[16]]. The fusion reactor will need high-field superconductors such as Nb_3Sn, which because of its brittleness is more difficult to work with and will take longer to develop.

BLANKET STRUCTURAL MATERIAL

The hybrid uses stainless steel as a blanket structural material rather than the refractory metals considered in some fusion designs. Developmental costs are greatly reduced because new materials need not be tested. Not only the blanket material but also the fuel (stainless-steel clad uranium molybdenum alloy, for example) are well known. Use of known materials under relatively low neutron fluences will allow the hybrid program to bypass developmental steps such as the Fast Flux Test Facility (FFTF) that is required for the breeder program.

NEUTRAL-BEAM INJECTOR

The injection energy for minimum-cost fuel production by a hybrid reactor is ~100-keV D^0 and T^0. At this energy, a positive-ion technique with a charge-exchange cell giving 50% conversion to neutrals followed by a beam direct converter is expected to convert electrical power to neutral-beam power at 70% efficiency. Such a system[17] (1 MW, 120 keV, and 0.5-s pulse length) will be tested within one year by Lawrence Livermore and Lawrence Berkeley Laboratories for eventual use on the Tokamak Fusion Test Reactor (TFTR), thus giving credence to the claim that such beams are nearly state of the art.

As injection energy is increased to meet the expected requirements for a fusion reactor, the efficiency falls rapidly.

Table 2. Comparision between technology requirements for the mirror hybrid and the mirror fusion reactors.

Requirement	Fusion-fission	Fusion
Magnet material	Nb-Ti(9 T)	Nb_3Sn(16 T)
Blanket structural material	Stainless steel	Niobium
Neutral-beam injector	Positive-ion type (~100 keV)	Negative-ion type (~200 keV)
Plasma direct converter	1 stage or none at all	Multistage
Required Q	0.5-1	>2
Fusion power	470 MW	1300 MW

For energies above 120 keV or so, the negative-ion technique is proposed. This concept is still in the very early stages, and considerable development will be required to produce working systems. Although success seems likely, most of the requirements of the fusion development program for the next 10 years can be achieved with positive ions. Therefore, there will be less development work on negative-ion systems than on positive-ion systems.

PLASMA DIRECT CONVERTER

A relatively simple converter with a one-stage, immersed grid performs the direct conversion of plasma end losses in the hybrid reactor. The direct converter performs several functions that would be needed even if energy recovery were not; e.g., it pumps the neutral gas to prevent excess end heat losses via electron conduction and provides magnetic expansion to reduce the power on the end surfaces. Our studies have shown that a one-stage converter reduces the cost of fissile fuel by only a small amount; there is no benefit from the higher-efficiency, multistage converters.

PLASMA

The value of Q required for the hybrid is between 0.5 and 1.0 (our design uses the classical value of 0.7). If Q should drop by a factor of 2 from the classical value due, for example, to plasma turbulence, the cost of fissile fuel would about double. Thus, we see that this hybrid design is fairly sensitive to Q for $Q \approx 0.7$. By comparison, the mirror fusion reactor is about equally sensitive to changes in Q at $Q \approx 2$.

The hybrid injection energy of 100 keV implies a value of the plasma dielectric constant $(\omega_{pe}/\omega_{ce})^2$ almost twice that for the mirror reactor, which has an injection energy between 150 and 200 keV. High values of $(\omega_{pe}/\omega_{ce})^2$ contribute to reducing microinstabilities. Since high values of Q are not as crucial to the hybrid, the He^{++} ions need not be adiabatically contained for the heating of electrons — as is necessary in the fusion reactor. The implication is that for a given physical plasma length the magnetic field can be reduced (by a factor of 2), thus increasing the orbit size. However, the lower energy somewhat compensates for this gain. Thus, the length of the machine measured in orbit radii, L/a_i, can be reduced; this reduction helps stabilize the convective, high-density, loss-cone microinstability. However, the drift-cyclotron loss-cone microinstability is aggravated by larger orbits.

To compensate for the 10 times energy multiplication in the blanket, the fusion power in the hybrid reactor is several times lower than that in a fusion reactor. This has two advantages:

- Because the power is less, the prototype developmental stages for the hybrid are also smaller than the equivalent fusion reactor stages; hence, they are more quickly and cheaply built.
- Because the plasma density is lower, beam penetration is easier.

The second advantage may be explained as follows. The midplane of the plasma is too thick ($r_{plasma} > 3\lambda_{ionization}$) for the beam to penetrate to the center (as is necessary for uniform filling of the plasma). Hence, we must either inject away from the midplane where the plasma is both thinner (due to field line fanning) and less dense (moving towards the mirrors) or rely on the fluting inward caused by the resultant inverted radial density gradient (as in the mirror fusion reactor). Having lower density tends to somewhat alleviate this problem.

The physics assumptions for the plasma are discussed in Chapter 5 of Ref. 12. Also, a relevant discussion is given in Appendix A of the FERF report.[16] More recent physics developments are discussed in Ref. 18.

DEVELOPMENT STEPS BETWEEN AN MX-SIZE AND A COMMERCIAL-SIZE MIRROR HYBRID REACTOR

We have prepared four mirror reactor designs which can be examined from the point of view of a progression of steps toward a commercial-size mirror hybrid. The first of these is the MX (Mirror Experiment) design.[19] This machine will extend the plasma physics of mirror machines and have limited D-T operation. The next larger machine is FERF,[16] which was designed primarily for testing many small samples for studies of radiation damage effects. Although this machine was not designed for testing the hybrid concept, it would have limited provision for testing one 2-m-deep blanket module that would

Table 3. Parameters for a series of machines leading to a fusion-fission reactor.

Parameter	MX	FERF	DEMO	Commercial
Field strength at conductor (T)	7	9	8	8
Plasma density (cm^{-3})	1×10^{14}	3×10^{14}	1.1×10^{14}	1.3×10^{14}
Plasma diameter (m)	0.6	0.5	2.4	4.8
Plasma length (m)	3.4	4.2	7.5	15
Q	—	0.3	0.7	0.7
Fusion power (MW)	0.3	3.4	44	475
Fissile fuel production rate (kg/yr)	—	3	190	2400
Cost	100 M$	300 M$	540 M$	2.3 B$

have a 50 cm × 50 cm surface facing the plasma.[20] If this module were a hybrid module, plutonium would be bred at the rate of 3 kg/yr.[21] The third machine, a demonstration reactor, DEMO, is the minimum size which can have a reasonable converage blanket between the plasma and shield. Used as a hybrid, its Pu production rate would be above 200 kg/yr. The commercial-size reactor would produce about 2000 kg/yr.

Some of the parameters for this progression of mirror machines are given in Table 3. The costs given in the table are highly uncertain. The plasma physics scaling should be established in the MX so that only a modest extension of parameters would be needed for the DEMO and commercial-size machine. As argued in Ref. 3, the fission technology is already in hand; thus, going from a FERF-size machine to a DEMO-size should be relatively simple.

CONCLUSION

Based on studies of mirror reactors, many generalizations have been made up to this point. Much experimentation and further studies are needed; however, let me conclude by stating that

- Successful development of a fuel-producing hybrid along with conventional fission reactors should constitute a long-term energy option just as is promised by the breeder, and
- The new fusion technology needed will be developed anyway as part of the fusion power development program.

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MIRROR HYBRID REACTOR OPTIMIZATION STUDIES*

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ABSTRACT

A system model of the mirror hybrid reactor has been developed. The major components of the model include (1) the reactor description, (2) a capital cost analysis, (3) various fuel management schemes, and (4) an economic analysis that includes the hybrid plus its associated fission burner reactors. The results presented describe the optimization of the mirror hybrid reactor, the objective being to minimize the cost of electricity from the hybrid fission-burner reactor complex. We have examined hybrid reactors with two types of blankets, one containing natural uranium, the other thorium. The major difference between the two optimized reactors is that the uranium hybrid is a significant net electrical power producer, whereas the thorium hybrid just about breaks even on electrical power. Our projected costs for fissile fuel production are ∿50 \$/g for ^{239}Pu and ∿125 \$/g for ^{233}U.

INTRODUCTION

The primary objectives of the first Lawrence Livermore Laboratory (LLL) fusion-fission hybrid reactor point design[1] were to determine a manner in which all of the necessary system components could be integrated into the reactor, to assess the technological problems, and to obtain a rough cost estimate. The resulting design was not optimized in either an engineering or economic sense, but rather was a reference point design from which further study could proceed.

Based on the point-design cost estimates, it appeared that the requirement of incorporating both fusion and fission components in the hybrid reactor would make the hybrid capital cost (\$/kWe) greater than that for a fission reactor. However, the impressive fissile-breeding performance of the hybrid, as compared to a fast-breeder reactor of comparable thermal rating, indicated that the most promising avenue for commercialization of this reactor concept was as a fissile breeder with electricity production as a by-product. Therefore, the hybrid study at LLL this year has concentrated on optimizing the hybrid for fissile production, employing the technique of parametric system analysis of the plant economics. The optimization was defined as a determination of the reactor parameters which minimize the cost of producing fissile fuel. The optimization thus minimizes the electricity cost component that is attributable to the fissile fuel burned by the fission reactors.

RESULTS AND DISCUSSION

SYSTEM MODEL

A large number of independent parameters that define the fusion component of a mirror hybrid are available to the reactor designer. Variation of these parameters can significantly affect the hybrid performance. In addition, plant economics are influenced by the blanket fissile management scheme and by the characteristics of the fission component. To assess the interplay of these various factors, we have developed a computer model of the mirror hybrid that permits rapid evaluation of the many possible reactor configurations. The components of the system model developed for the parametric analysis are as follows:

- Reactor Description

    Plasma physics
    Magnet design
    Blanket geometry
    Power flow
    Capital cost

*This work was performed under the auspices of the U.S. Energy Research and Development Administration, under contract No. W-7405-Eng-48.

• Fuel Management

Time-dependent mass and energy flows
Capacity factor
Cash-flow accounting techniques

• "Nuclear Park" Economics

Hybrid and fission reactors

The mirror reactor model (essentially the analysis developed by Carlson[2]) includes mirror plasma physics, magnet design, blanket geometry and power flows. Capital costs are a key element in the analysis, and here we have attempted to be as thorough and consistent as possible. However, the costing procedure entails a high degree of uncertainty due to the infancy of fusion technology.

A unique feature of hybrid reactor economics, as compared to those of strictly power-producing fission and fusion reactors, is that fissile fuel does not generate revenues on a continuous basis. Revenue from fissile fuel is only realized when blanket segments are removed from the reactor and reprocessed. In addition, the blanket multiplication increases and the breeding ratio decreases with increasing fuel exposure, as described by Lee.[3] To model these effects, we developed a fuel management package to evaluate the time-dependent production of power and of fissile fuel as functions of specified fuel-management parameters. In this analysis we also determined the timing and magnitude of fuel and blanket fabrication costs as well as spent-fuel shipping and reprocessing costs. The economics of this time-dependent "fuel-cycle" is evaluated using cash-flow accounting techniques.[4]

A second unusual feature of the economic analysis is that the hybrid produces two products, fissile fuel and electricity. To fix the cost of producing these two products, it is necessary to specify a constraint. In the present analysis, we have chosen to fix the cost of hybrid electricity at the same cost as electricity produced by the fission reactors which burn the hybrid fissile fuel. By considering the hybrid and its associated burner reactors as a single entity producing only electricity, we can calculate the electricity cost. Having established the electricity cost, we can then evaluate the cost of the fissile material from the hybrid.

RESULTS OF PARAMETRIC ANALYSIS

In optimization studies to date we have employed variations of the following parameters:

• Plasma Physics

Injection energy
Mirror ratio
Injection angle
Confinement time

• Magnet Design

Conductor field
Mirror-to-mirror length

• Fuel Management

Maximum fertile burnup
Exposure distribution within the blanket

In our experience these quantities have the most important influence on the economics of the hybrid reactor. Two dependent quantities which have been found to strongly influence plant economics are the blanket coverage and the plant capacity factor.

The geometric relationship between the plasma and blanket are shown in Fig. 1. Holes in the blanket must be provided for plasma leakage and neutral injection. We have found that if the power density in the blanket exceeds 100 to 200 W/cm^3, the plenum dimension required to handle the helium flow becomes excessively large, pushing the blanket inward and severely decreasing the blanket coverage. Also, large neutral beam

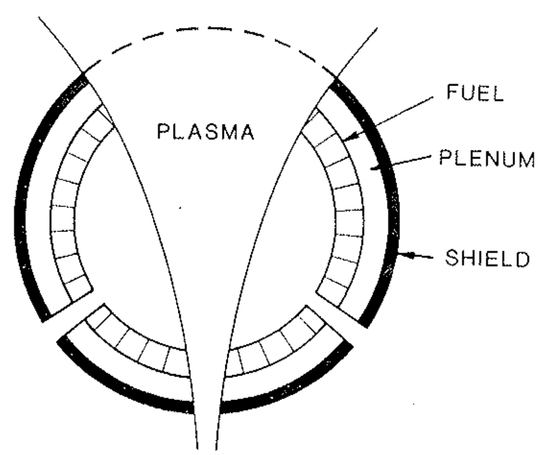

Fig. 1. Blanket coverage.

current requirements demand large injection ports thus reducing the blanket coverage. The equations used to model the plant capacity factor, δ, are:

$$\delta = \frac{t_{op}}{t_{op} + t_{sm} + t_{um} + t_{bc}} ,$$

$$\delta = \frac{1}{1.2 + (t_{bc}/t_{op})} ,$$

$$t_{op} = \frac{\text{exposure (MW-yr/m}^2)}{\text{flux (MW/m}^2)} .$$

Here, t_{op} is the operating time for the reactor, t_{sm} is the time for scheduled maintenance, t_{um} is the time for unscheduled maintenance and t_{bc} is the time required for the blanket change operation. The capacity factor evaluates the trade-off between high first-wall flux (and, therefore, high product generation rates) and the need for shutting down the plant to perform blanket change operations after maximum blanket exposure is reached.

The fission reactors chosen as burners of the hybrid fissile fuel are listed listed in Table 1 along with their fuel

Table 1. Description of thermal converter reactors.

Parameter	Burner ^{239}Pu	Burner ^{233}U
Reactor type	LWR	High-gain HTGR
Fuel cycle		
Fertile feed	Natural U	Th
Fissile feed	Pu	^{233}U
Fissile recycle	Pu	^{233}U
Conversion ratio	0.5	0.8
Fissile feed requirements (kg/yr/MWe)	0.333	0.185

make-up requirements. For a Pu burner, we have used a light water reactor (LWR) on a Pu recycle fuel cycle which is supplemented with hybrid Pu. As a ^{233}U burner, we have used a high-gain high temperature gas-cooled reactor (HTGR) on the thorium-^{233}U fuel cycle. Another possibility as a ^{233}U burner, but not yet examined, is the CANDU reactor.

The representative thermal reactor fuel cycles that are used to couple the fusion and fission reactors (Figs. 2 and 3) have been adapted from fuel cycles based on thermal reactor-produced fissile fuel.[5-7] Actually, the fuel cycles used

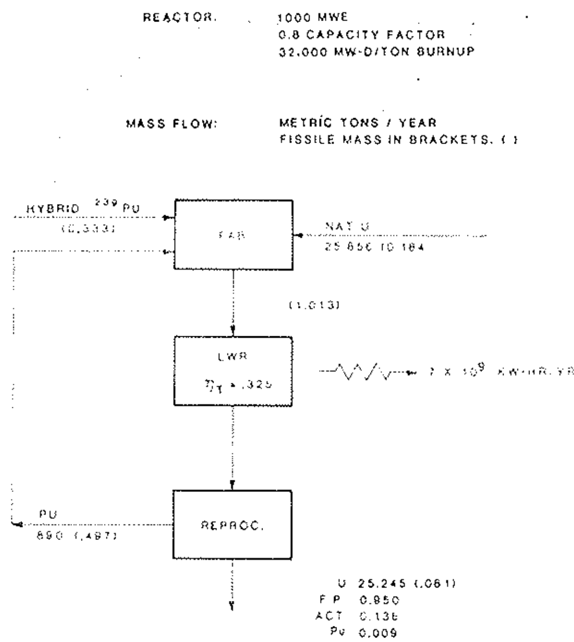

Fig. 2. The LWR fuel cycle.

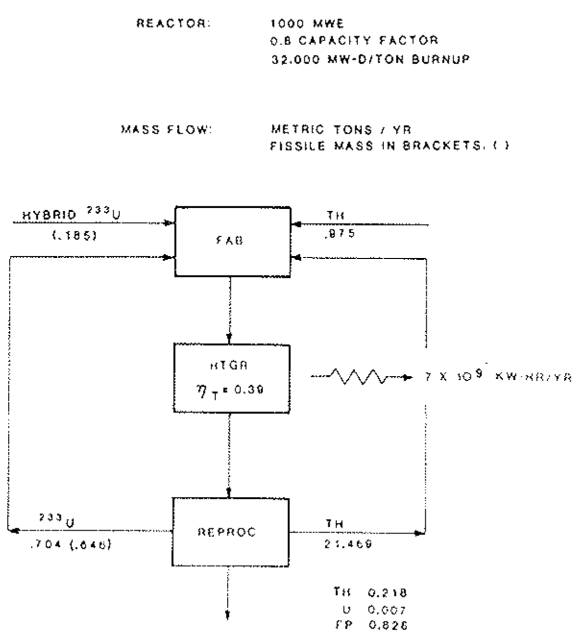

Fig. 3. The HTGR fuel cycle.

should be reevaluated, because the isotopic composition of the hybrid fissile fuel differs from that produced in a thermal fission reactor as a result of differences in flux spectrum and burnup in the two types of reactors.

The LWR fuel cycle[5] (Fig. 2) uses plutonium recycle and has a conversion ratio of 52%. Fissile feed consists of the ^{235}U component of the natural U fertile feed combined with hybrid-produced ^{239}Pu. Here we have assumed that hybrid Pu (>90% fissile) is equivalent to Pu from a uranium-fueled LWR (∿70% fissile). In future studies we plan to examine a variation of this cycle that will permit higher uranium utilization. The uranium will be recycled and the hybrid Pu feed increased to compensate for the absence of ^{235}U. Thus, the hybrid fuel cycle may be designed to ultimately fission the majority of the uranium.

The HTGR fuel cycle[6,7] (Fig. 3) utilizes Th fertile feed, ^{233}U recycle and ^{233}U fissile feed from the hybrid and has a conversion ratio of 0.79. This conversion ratio is considerably higher than that of the ^{235}U HTGR (CR ≈ 0.66) due to the use of ^{233}U, to the higher fertile to fissile ratio and to the lower burnup (32 000 MW-D/T vs 95 000 MW-D/T for the ^{235}U cycle).

Figures 4 to 8 show some of the details of the optimization process, primarily for the uranium blanket. Figure 4 shows that a rather broad optimum exists for various combinations of mirror ratio and injection energy (W_{INJ}).

As shown in Fig. 5, the minimum economical size is about 10 m, mirror-to-mirror. Below this size, the blanket coverage decreases rapidly, strongly degrading the plant economics. A 7.5 m machine appears to be the minimum "demo" size.

The optimization of magnetic field is shown in Fig. 6. For the uranium blanket, a near optimum can be attained at 8 tesla, which yields the optimum blanket power density of ∿100 W/cc. For the thorium blanket, 12 to 14T fields are required to minimize the fissile cost.

The variation of fissile cost with fertile burnup is shown in Fig. 7. At low burnup, high fabrication-reprocessing costs and low capacity factor are in-

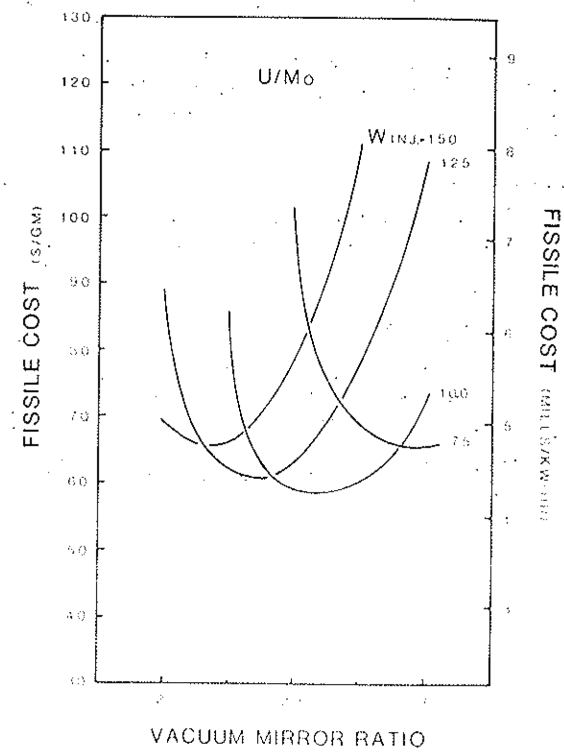

Fig. 4. The plasma physics variation.

curred. As the maximum burnup increases, larger temporal variations occur in the thermal output due to increasing blanket multiplication with burnup. Thus, this situation requires that an increasing fraction of the plant thermal capacity remain idle during periods of non-peak thermal output. Also, the high burnup implies longer delays in the realization of the revenues from fissile breeding. For the uranium blanket, the optimum occurs at a 1% burnup, which is about the maximum tolerable burnup for the U-Mo fuel. For thorium, the 0.5% burnup is well below the maximum obtainable with this fuel.

There is some degree of uncertainty as to the actual plasma Q that will be attained in mirror reactors. It is possible that microinstabilities will limit Q to a value somewhat below the classical value.[8] A second possibility is that Q enhancement techniques under consideration[9] will elevate Q above our presently predicted values. Figure 8 shows that electricity costs are not strongly perturbed even if classical confinement is not attained. A two-fold enhancement of Q improves the economics of the hybrid reactor but, in general, electricity costs are rather insensitive to higher Q.

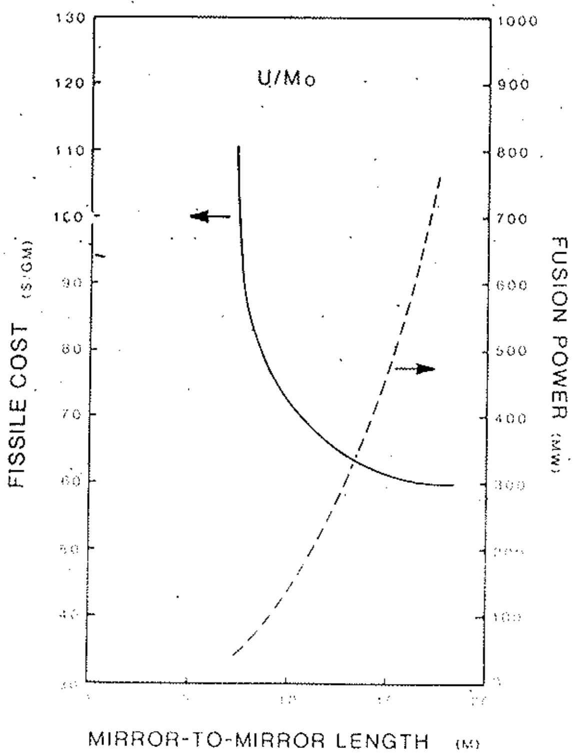

Fig. 5. Variation of fissile cost and fusion power with reactor size.

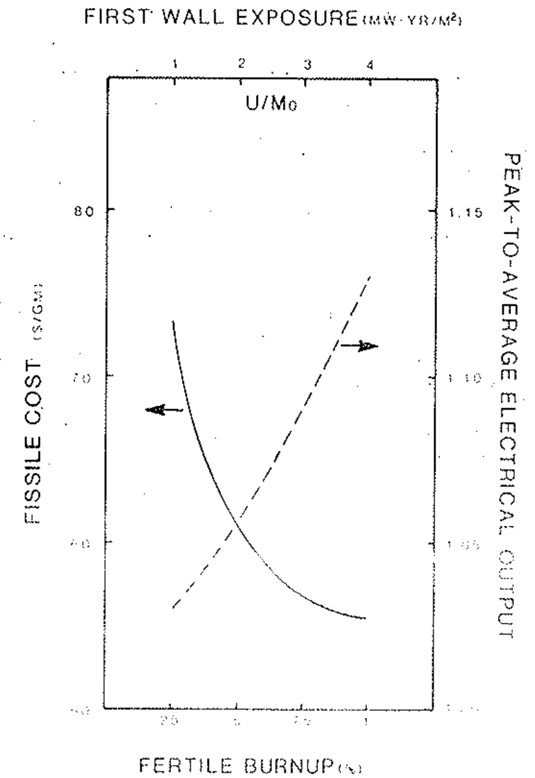

Fig. 7. Variation of fissile cost with fertile burnup.

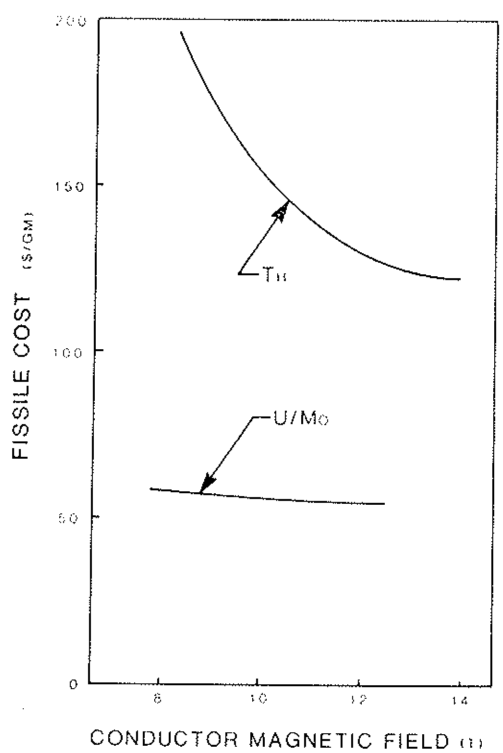

Fig. 6. Variation of fissile cost with magnetic field.

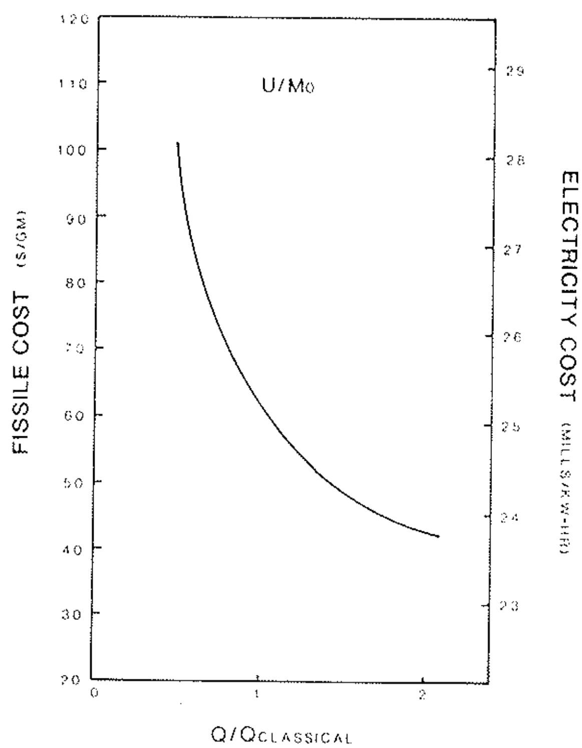

Fig. 8. Variation of fissile value and electricity costs with Q.

Table 2. Parameters for the optimized hybrid reactors.

Parameter	U/Mo	Th
Mirror ratio	2.50	2.75
Injection energy (keV)	100	100
Conductor field (T)	8	12
Q	0.68	0.75
Fusion power (MW)	470	1500
First wall flux (MW/m^2)	1.3	4.2
Blanket thermal power, average (MW)	4220	3340
Electrical output (MW)	1040	-40
Capacity factor	0.75	0.73
Mirror-to-mirror length (m)	15	15

OPTIMIZED REACTOR CONFIGURATIONS

From the reactor parameters for the uranium and thorium blankets (Table 2) several significant differences between the two reactors are evident.

- The uranium blanket, because of its high energy multiplication, results in a plant with a large electrical output. The thorium blanket reactor does not produce net electricity, just fissile fuel.
- Both blankets have about the same thermal rating. This results from the much greater fusion power required for the thorium blanket reactor as compared to the uranium blanket reactor.
- The high fusion power of the thorium blanket reactor is obtained by using a more intense magnetic field than for the uranium blanket reactor. Therefore, the latter may rely on existing NbTi superconductor magnet technology, whereas, the thorium blanket reactor will require the more technologically advanced Nb_3Sn superconductor.
- Optimum economics for the thorium blanket are obtained at high exposure, about 9 $MW\text{-}yr/m^2$. The uranium blanket requires only about 4 $MW\text{-}yr/m^2$. These results assume that the blanket structure is capable of tolerating these exposures.

The reactor size selected, as given by the mirror-to-mirror length of 15 m, is not an optimum. Rather, this size was chosen as being representative of a plant with good economics and with approximately the same thermal rating as present-day nuclear power plants. A point to be emphasized is that the first-wall fluxes for the two blankets were not arbitrarily chosen. In the present analysis, the first-wall flux is a dependent parameter whose value is determined by an economic optimization of the reactor.

Table 3. Blanket parameters for the optimized reactors.

Parameter	U/Mo	Th
Fissile output (kg/yr)	2360	2590
Avg. energy multiplication	11.1	2.8
Blanket coverage	0.86	0.77
Fertile burnup (%)	1.0	0.5
Blanket exposure ($MW\text{-}yr/m^2$)	4.1	9.2
Fuel power density (W/cm^3)	150	110
Peak to average electrical output	1.13	—
Blanket enrichment, average	1.02%	1.06%

The blanket parameters for the optimized reactors are listed in Table 3. Both blankets produce about 2 1/2 metric tons of fissile fuel per year. However, the thorium blanket requires a rather high exposure, and the possibility of the blanket structure being able to attain $\sim$9 $MW\text{-}yr/m^2$ exposure is quite uncertain. For the uranium blanket, the average energy multiplication is higher by about a factor of four than for the thorium blanket; these blanket energy multiplications include the effect of the fractional blanket coverage.

The economic parameters for the hybrid are listed in Table 4. The higher capital cost of the thorium blanket hybrid is associated with the fusion components required to generate the higher fusion power. The ^{233}U cost is more than two times greater than the Pu cost. The lower cost for Pu results from the lower capital cost and from electrical power production revenues. However, because of the lower fissile requirements of the HTGR, as compared to the LWR, the cost of the electricity from the two fission power plants is approximately the same.

Table 4. Economics for the optimized hybrid reactors.

Cost	U/Mo	Th
Capital cost ($10^9$$)	2.3	3.3
($/kWe)	2200	—
Fissile material cost ($/g)	55	127
Capital	80	103
Fuel cycle	13	21
Operation and maintenance	1	1
Electricity revenues	−39	2
Electricity cost (mills/kW-hr)	24.8	25.3

The breakdown of the fissile material costs indicate that they are dominated by capital costs. The fuel-cycle costs account for blanket fabrication, fuel fabrication, reprocessing and spent-fuel shipping. Current (high) estimates for the fuel services have been used,[10] but they are not a dominant cost. For the uranium blanket reactor, approximately 60% of the plant revenues are generated by fissile production. This is in contrast to the thorium hybrid reactor where 100% of plant revenues are generated by fissile material.

The fission reactor economics are listed in Table 5. Here, the category "fuel cycle without fissile material" refers to all normal fuel cycle charges excluding the cost of fissile fuel, i.e., fabrication, reprocessing, spent-fuel shipping and purchase of fertile fuel. The fissile fuel cost is the cost of producing this material in the hybrid reactor. Here, the important result is that the hybrid fissile fuel costs of 4.1 and 5.3 mills/kW-hr are a small fraction of the total cost of the electricity. Based on our current capital cost model, we conclude that the mirror hybrid reactor is capable of converting the world's large fertile resources into fissile fuel at a cost that does not strongly influence the net cost of electricity.

Table 5. Economics for the fission reactors.

Cost	LWR	High-gain HTGR
Capital cost ($/kWe)	750	750
Electricity cost (mills/kW-hr)	24.8	25.3
Capital cost	16.1	16.1
Fuel cycle without fissile material	3.9	3.2
Fissile fuel	4.1	5.3
Operation and maintenance	0.7	0.7

Table 6. Economics for the hybrid/thermal reactor complex.

Cost	U/Mo	Th
Installed capacity (MWe)	8130	14 000
Hybrid	1040	—
Fission reactors	7090	14 000
Capital cost ($/kWe)	935	985
Electricity cost (mills/kW-hr)	24.8	25.3
Capital	19.7	20.5
Fuel cycle	4.3	4.0
Operation and maintenance	0.8	0.8

The overall nuclear park economics are shown in Table 6. Examining the U hybrid/LWR combination, we see the 1040-MWe hybrid supports fissile requirements for about seven 1000-MWe thermal reactors. The combined capital cost of the hybrid and the fission reactors is 935 $/kWe vs 750 $/kWe for the LWR's. The Th hybrid produces enough fissile material to support fourteen 1000-MWe HTGR's, resulting in a capital cost for the nuclear complex of 985 $/kWe vs 750 $/kWe for the HTGR. The fuel cycle costs do not include any cost for fissile fuel since this material is produced entirely within the hybrid-fission reactor complex.

The fact that the hybrid fissile production costs are capital cost dominated (see Table 4) dictates that we regard these fissile costs with a degree of uncertainty, reflecting both the infancy of fusion engineering and our inability to accurately assign costs to reactor components which are merely conceptual designs. However, the presently predicted costs are within the realm of being economically attractive. In our opinion, this conclusion justifies vigorous support of the hybrid concept, with future design efforts continuing to refine the engineering and to incorporate experimental results as they become available. The result could well be an energy option that will ease the transition to a full fusion power economy in the next century and will provide comparatively early benefits from the large R&D investment that will be required to commercialize this energy source.

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MIRROR HYBRID REACTOR BLANKET AND POWER CONVERSION SYSTEM CONCEPTUAL DESIGN*

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ABSTRACT

The conceptual design of the blanket and power conversion system for a gas-cooled mirror hybrid fusion-fission reactor is presented. The designs of the fuel, blanket module and power conversion system are based on existing gas-cooled fission reactor technology that has been developed at General Atomic Company. The uranium silicide fuel is contained in Inconel-clad rods and is cooled by helium gas. The fuel is contained in 16 spherical segment modules which surround the fusion plasma. The hot helium is used to raise steam for a conventional steam cycle turbine generator. The details of the method of support for the massive blanket modules and helium ducts remain to be determined. Nevertheless, the conceptual design appears to be technically feasible with existing gas-cooled technology. A preliminary safety analysis shows that with the development of a satisfactory method of primary coolant circuit containment and support, the hybrid reactor could be licensed under existing Nuclear Regulatory Commission regulations.

INTRODUCTION

This report is the summary of a study performed by General Atomic Company to investigate the feasibility of applying gas-cooled reactor technology to the Mirror Hybrid Fusion-Fission Reactor.[1] The work was specifically directed at design and analysis of the blanket and power conversion system of the Mirror Hybrid Fusion-Fission Reactor (MHFFR). A goal of the study was to use conventional gas-cooled reactor technology so that the conceptual design would have potential for near term construction with a minimum of large scale development programs. The study was done in conjunction with mirror hybrid reactor design efforts at Lawrence Livermore Laboratory.[2,3]

BLANKET DESIGN

FUEL DESIGN

Fuel Materials. The candidate fuel materials that have been evaluated for use in the MHFFR include uranium alloys, uranium oxide, uranium carbide, and uranium nitride. A comparative evaluation was made in terms of the important parameters, such as fuel density, neutron economy, physical, mechanical and corrosion properties, irradiation effects, compatibility with cladding and with impurities in the helium coolant (particularly tritium), and fabrication cost. Two important characteristics of the hybrid reactor are the low burnup[3] ($\leq$ 1%) and the importance of high fuel density.[2]

The operational conditions tend to favor the use of metallic uranium alloys, in particular uranium silicide (U_3Si) fuel which has been under development in Canada for the CANDU reactor program during the past decade.[4] Uranium oxide is less desirable because of the significantly lower fuel density, lower neutron economy, and higher fabrication cost compared to uranium silicide. Uranium carbide and nitride also are less desirable because of similar reasons. However, uranium carbide may be a suitable alternative fuel.

Three primary fuel candidates were selected from those evaluated: uranium-molybdenum alloy, uranium carbide and uranium silicide. The density and maximum operating temperature for these candidate materials are shown on Table 1.

Since any of these materials would be an acceptable choice for the MHFFR, the final selection was based on cost estimates for the system economics that each alternative would dictate. The economics of the

*Work done under United States Energy Research and Development Administration contract number AT(04-03)-167, project agreement 38.

various fuels is affected by four factors:

1. Basic fuel cost.
2. Fabrication cost.
3. Revenue from power produced.
4. Revenue from fissile production.

Table 1. Fuel material candidates.

	U-MOLY	UC	U_3Si
Uranium Density (g/cc)	17.1	13.0	15.5
Maximum Operating Temperatures (°C)	700	2000	900
Approximate Rod Diameter (mm)	10	18	17
Blanket Cost (M$)	154	112	94
Cost (M$/Yr)	--	+17.1	+24.1
Revenues (M$/Yr)	--	-41.7	-17.0
Net Benefit (M$/Yr)	--	-24.6	+7.1

Compared to UO_2 cost as a base, it is estimated that U-Moly will be three times as expensive (due to stringent Q/A requirements), UC will be twice as expensive and U_3Si cost will be on par with UO_2.[5] The fabrication cost of blanket rods is a function of the number of rods in the reactor. Due to the lower peak temperature permissible in the U-Moly, the allowable rod diameter is smallest among the three fuels. The allowable rod size for UC and U_3Si is approximately the same as the cladding temperature is the limiting parameter. Estimated blanket costs are shown in Table 1. These estimates include fuel material and fabrication costs.

Revenues from power and fissile production are roughly proportional to the uranium density in the fertile material. The estimates of changes in power and fissile production with U-Moly as reference, are shown in Table 1. Taking U-Moly as the reference case, UC represents a net loss of $24.6M/year and U_3Si represents a net benefit of $7.1M/year.

In addition to the small economic advantage, U_3Si has the advantages of requiring less stringent quality assurance, of allowing larger fuel rods and of reduced parasitic neutron absorption when compared to U-Moly. U_3Si was chosen as the fuel material for the MHFFR

Fuel Configuration. Five different fuel configurations were investigated for use in the MHFFR blanket as shown in Table 2.

Table 2. Fuel configuration options.

Honeycomb
Plates
Rods:
Cross Flow
Parallel Flow - Solid Pellets
Parallel Flow - Annular Pellets

Honeycomb material has the potential for a high fuel density. Due to the small heat transfer coefficient in the blanket very small holes would be required which would result in high fabrication costs. The technology for fabrication and cladding of honeycomb fuel structure is not developed. A plate configuration has the potential of offering a large heat transfer surface. Bonding technology appears to be difficult to achieve for U_3Si and the preferred cladding material, Inconel. A rod configuration appears to be better suited to application in the hybrid reactor.

In the hybrid reactor, the requirement to keep the first wall cool generally requires that the coolant first cool the first wall and then flow radially outward through the blanket. The fuel rods may be arranged parallel or perpendicular to this flow. The use of cross flow perpendicular to the axes of the rods does not enhance the average heat transfer coefficient but does introduce a large variation in this coefficient and thus temperature around the rod. A radial rod arrangement, on the other hand, results in a uniform heat transfer coefficient around the rod, reducing the potential for hot spots. The only problem expected with the use of radially arranged rods is due to the fuel swelling with burnup. By including a hole in the center of the fuel pellets, this swelling can be greatly reduced.[6] As a consequence, annular fuel pellets arranged in a radial direction were selected for the MHFFR design. To get the maximum possible helium outlet temperature, a high temperature cladding material, Inconel 718, was selected. This choice is discussed further in the section on Module Structural Material below.

Fuel Performance. With the fuel materials selected as U_3Si fuel and Inconel 718 clad, maximum design temperature limits were chosen. The helium inlet and outlet temperature conditions were selected to correspond to the fuel temperature limits although these helium conditions were not optimized. The temperatures are shown on Table 3.

Table 3. MHFFR temperatures (°C)

Maximum Cladding Limit	800
Maximum Fuel Limit	900
Maximum First Wall Limit	500
Coolant Inlet	280
Coolant Outlet	530

The fuel design was based on hot spot temperatures obtained by applying hot spot factors to the nominal temperatures calculated for an average channel. End-of-life conditions were used since these represent the maximum blanket multiplication and power density. Hot spot factors were determined for the MHFFR using a semi-statistical method[7] and are shown on Table 4.

Table 4. Hot spot factors for the MHFFR blanket.

F_b – Coolant Temperature Rise	1.3
F_f – Film ΔT (for clad)	1.8
– Film ΔT (for fuel)	1.25
F_c – Cladding ΔT	1.25
F_g – Fuel-Clad Gap ΔT	1.5
F_r – Fuel ΔT	1.2

The blanket is designed with a flow baffle to keep constant flow velocity over the first wall. The maximum first wall temperature can in this way be limited to 468°C. The fuel rod diameter was selected to be 17 mm which limits the peak hot spot cladding temperature to 800°C and the peak hot spot fuel temperature to 900°C. The power distribution and resulting nominal temperatures along the fuel rod are shown in Fig. 1.

The coolant flow was treated as a one-dimensional incompressible flow. The blanket module was modeled as a flow resistance network and the pressure drops shown on Table 5 were calculated.

Table 5. Helium pressure drop in module.

Blanket Fuel Region	1.2×10^4 Pa (1.8 psi)
Module Total	
-full power operation	3.9×10^4 Pa (5.7 psi)
-depressurized decay heat removal	0.2×10^4 Pa (0.3 psi)

MODULE DESIGN

Structural Material. To insure good neutronics performance of the blanket, it is important that the first wall be kept as thin as possible. It must operate, however, at high temperature under high pressure. Inconel 718 was selected for the blanket structural material and fuel cladding material because it exhibits high strength at elevated temperature. The design stress limit for Inconel at 800°C is 296 MPa (43,000 psi).

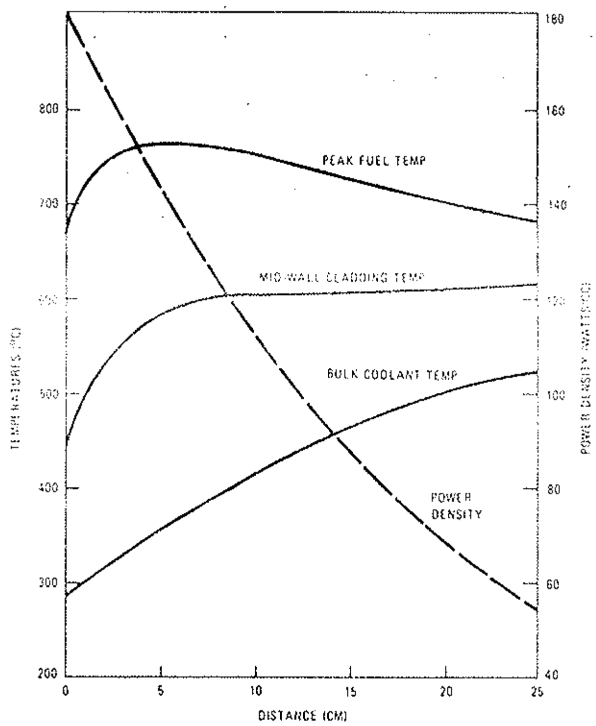

Fig. 1. Power and temperature distribution.

Module Design. The blanket fuel rods are wire wrapped to maintain the proper coolant gap and are grouped into standard sized fuel assemblies on a close-packed triangular pitch. Rectangular assemblies are grouped together with a trapezoidal assembly on each end to form a submodule 25 cm wide with a variable length depending on the position of the submodule in the module. This is shown on Fig. 2. The cool incoming helium flows radially inward along the outside of the submodule, cooling the pressure containing wall, turns at the first wall and flows radially outward through the fuel rods.

Forty-five submodules are arranged to form each of the 16 modules. The module is in the shape of a spherical segment as shown on Fig. 3.

Each module is a separate pressure vessel. With a 25 cm submodule width, and 6.08 MPa helium pressure, the pressure containing first wall can be kept to 0.5 cm

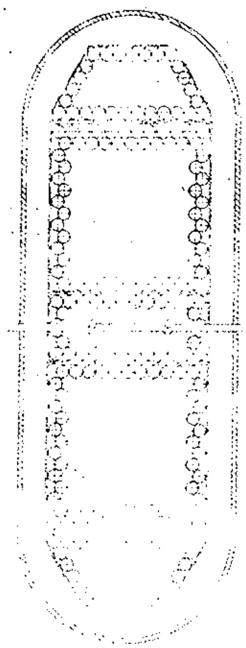

Fig. 2. MHFFR submodule.

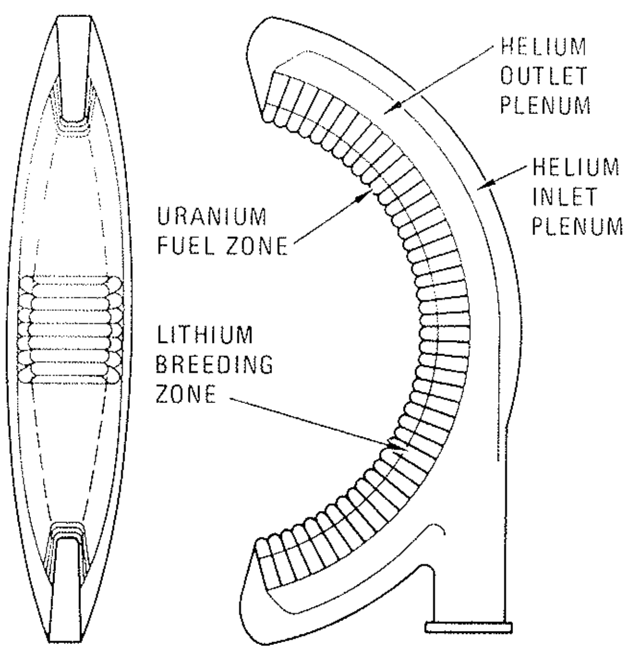

Fig. 3. MHFFR blanket module

thick. A reentrant coolant flow system was chosen which allows the external module walls to be kept near the coolant inlet temperature of 280°C. The helium coolant flow path leads into the inlet plenum at the back of the module. From there the coolant helium flows radially outward through the U_3Si fuel region and the $LiALO_2$ tritium breeding zone. The hot helium is collected in the outlet plenum and flows to the power conversion system.

POWER CONVERSION SYSTEM

SYSTEM PARAMETERS

The system parameter choices influence the reactor component design and the plant economics. The helium temperature and pressure have a significant bearing on the design feasibility, fabricability and size of the power conversion components. The maximum temperature limits are set by material considerations, minimum temperatures are set by steam conditions and heat transfer surface requirements. Pressure has an impact on helium and steam duct sizes and upon the circulator design. Consideration of these effects led to selection of the system parameters shown on Table 6.

Table 6. MHFFR system parameters

Helium		
Average pressure	6.08 MPa	(60 atm)
Blanket inlet temperature	530°C	(986°F)
Steam		
Superheater outlet pressure	8.3 MPa	(1200 psi)
Superheater outlet temperature	445°C	(832°F)
Reheater outlet temperature	504°C	(939°F)

With these system parameters, the power conversion system could be designed. A schematic power flow diagram is shown on Fig. 4 and some of the important reactor power variables are shown on Table 7.

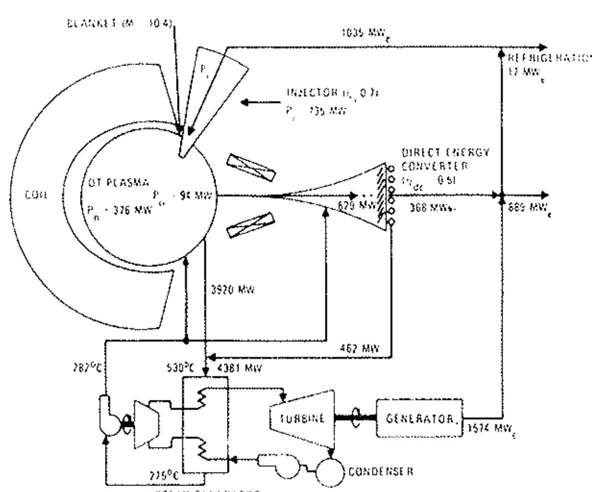

Fig. 4. MHFFR power flow diagram.

Table 7. Reactor power

Fusion Power	470 MWt
Total Nuclear Power	4015 MWt
Direct Converter Output	368 MWe
Turbine Generator Output	1574 MWe
Total Electric Power	1942 MWe
Net Electrical Output	889 MWe
Station Efficiency	22.1%

PRIMARY COOLANT SYSTEM

The reactor blanket consists of 16 modules. These are grouped into four quadrants, each consisting of four interconnected modules. Each quadrant has four steam generators, two helium circulators, two auxiliary heat exchangers and two auxiliary circulators as shown on Fig. 5. Each quadrant is also connected to one quarter of the direct converters. The system components are based on standard gas cooled reactor technology.

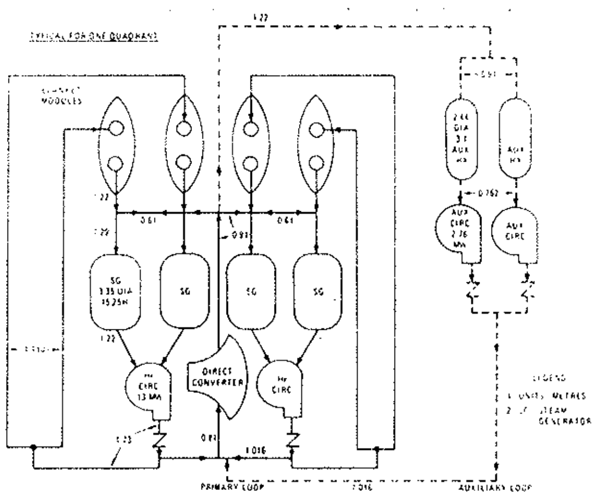

Fig. 5. MHFFR primary loop arrangement.

Helium Circulators. Each quadrant has two circulators to provide cooling system redundancy. Each circulator is rated at 13.2 MW (17,000 HP) and has a single stage axial flow compressor powered by a single stage steam turbine driven by superheated steam from the main loop steam generators. The circulator turbine is located in the main steam loop between the superheater and the resuperheater sections of the steam generator as shown on Fig. 4. This series arrangement provides interent load following characteristics to the main loops.

Steam Generators. Each quadrant has four steam generators, one for each module. Each steam generator is 3.2 m in diameter and 14.6 m high, which is about the largest size that can be rail-shipped without restrictions. The generator uses a helically-wound tube bundle with cross-counter flow of the helium gas.

Auxiliary Cooling Loops. Each quadrant is provided with an auxiliary cooling loop to provide an independent means of cooling the shutdown reactor to remove decay heat. The main loop components are sized so that one main circulator and one steam generator could adequately cool the blanket quadrant even if the helium coolant were depressurized. Nevertheless, the auxiliary cooling loops are required to provide diverse, redundant backup cooling. The auxiliary loops could also be used for long term decay heat removal, if desired.

The auxiliary circulators consist of an electric motor driven centrifugal compressor. The motor is driven from a variable frequency power supply to be able to accommodate the large difference in circulator speed required to operate under both pressurized and depressurized conditions. Each circulator is rated at 2.75 MW (3700 HP).

The auxiliary heat exchanges are axial flow, helically wound and transfer heat to pressurized water at 14.5 MPa (2,100 psi). The heat is transferred to the atmosphere in an air-blast heat exchanger. A control system is provided to regulate both the auxiliary circulator speed and the water flow rate so that both the blanket helium inlet and outlet temperatures remain with acceptable ranges.

SECONDARY COOLANT SYSTEM

Steam Loop Design. The steam loop of the power conversion system is quite conventional. Steam raised in the 16 once-through steam generators is routed through the helium circulator steam turbines, is reheated and then fed to the main turbine generator. The turbine is a tandem-compound machine with an inlet pressure of 5.1 MPa (744 psi) and inlet temperature of 503°C (937°F). The turbine exhausts to a conventional water-cooled condenser. These steam conditions are quite modest; the temperatures are limited by the capabilities of the hybrid blanket fuel and cladding materials. The total thermal power input to the power conversion system from the blanket and direct convertor is 4381 MW, the output from the turbine generator is 1574 MWe and the power conversion system efficiency is 35.9 percent. Due to the large power requirement of the neutral beam injectors the net station efficiency is 22.1 percent.

Power Conversion System Control. The plant control system (PCS) used on the MHFFR power conversion system is quite similar to that developed for gas-cooled fission reactors and allows automatic load following over a wide range. The on-load plant control system compares a set of process variables to the proper load-dependent set points of these variables and generates feedback signals for regulation. In addition to the on-load control system the plant control system also has a decay heat removal control system which provides automatic operation during shutdown cooling, a plant shutdown control system and a startup control system which assist during startup and shutdown. These control systems act to prevent thermal shock by controlling the steam loop to match the helium loop characteristics during startup and shutdown.

The MHFFR is also provided with a plant protection system (PPS) which handles those control fuctions which might be required to protect the health and safety of the public. Various critical plant parameters are monitored by the PPS. Should any of these go beyond the safety limit specified by the plant technical specifications, the PPS automatically actuates reactor trip. In the event of a serious accident the PPS also actuates the plant engineered safety features. The reactor trip parameters are shown on Table 8. Reactor trip is accomplished by stopping the neutral beam injector current which almost instantly reduces reactor power to decay heat production only. The plant control system then acts to shut down the power conversion system in an orderly manner.

POWER CONVERSION SYSTEM LAYOUT

A relatively compact system arrangement can be achieved by dividing the blanket into four quadrants of four modules each. To minimize the length of the large diameter (1.2 m) helium ducts, the four quadrants are not interconnected. Each quadrant comprises one main loop, divided into two interconnected sub-loops, and one auxiliary loop, also divided into two sub-loops. Each main sub-loop has two

Table 8. Reactor trip parameters

Sensed Variable	Indication
Manual	--
Neutron Flux	High
Power-to-Flow Ratio	High
Blanket Coolant Moisture	High
Blanket Coolant Delayed Neutron Activity	High
Blanket Coolant Pressure	High or Low
Containment Pressure	High
Main Steamline Pressure	Low
Main Feedeater Pressure	Low
Plant Electrical System	Power Loss
Main Loop Trouble	Logic Network

blanket modules, two steam generators and one helium circulator. Each auxiliary sub-loop has one auxiliary heat exchanger and one auxiliary helium circulator. These main and auxiliary components and one-quarter of the direct converters are connected together as shown on Fig. 5. The conceptual arrangement of these components around the reactor is shown on Fig. 6.

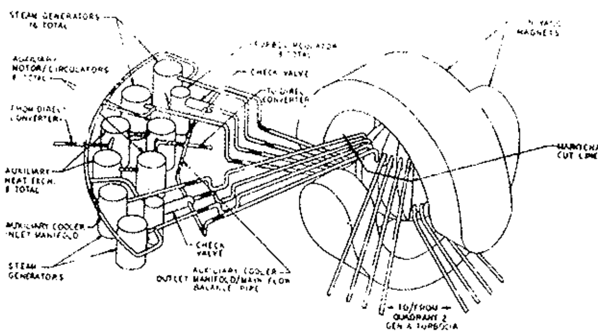

Fig. 6. MHFFR quadrant concept.

SAFETY

MHFFR SAFETY ANALYSIS

The potential routine radioactive releases from the MHFFR will contain both fission products and tritium. The potential fission product releases will be smaller than those of a fission reactor due to the low burn-up and the highly retentive U_3Si fuel. The tritium containment aspects of the MHFFR are expected to be the same as those of a pure fusion reactor. Since both of these potential routine releases can be assumed to constitute only very modest and acceptable public risk, the preliminary safety analysis of the MHFFR has concentrated on the study of potential accidents. The accidents considered are listed on Table 9.

Table 9. Potential accidents

Reactivity Insertion Accidents
Local Flow Blockage
Steam Leakage
Fuel Handling Accidents
Helium Leakage

A preliminary and admittedly conservative estimate was made of fuel heat up rates in the absence of forced cooling by assuming totally adiabatic heating. Starting at normal operating conditions with totally adiabatic heating at full power, the design limits for the U_3Si fuel are met after 6 seconds. If reactor trip accompanies initiation of adiabatic heating so that the power production is from fission product decay heat only, the fuel design limits are reached after about 120 seconds of adiabatic heating. These preliminary results are quite conservative and more detailed analysis may extend these time estimates considerably. Nevertheless, it is expected that forced cooling will have to be assured at all times in the MHFFR. To this end, two redundant helium circulators were provided in each main loop quadrant and these are backed up by two redundant auxiliary circulators of a totally diverse design.

An aspect of the MHFFR that needs further study is the design of the helium coolant ducts. The ducting and module support and containment concept will have to be designed in such a way as to assure reliable maintenance of adequate cooling of the blanket through either the main cooling loops or the auxiliary cooling loops.

Reactivity Insertion Accidents. The MHFFR blanket is designed to be substantially subcritical at all times. There are, however, effects that could potentially increase the blanket multiplication which are outlined below.

Blanket Movement. Since the system will normally be far subcritical, flow or seismic induced vibration and thermal bowing effects are not expected to have any substantial reactivity impact on blanket multiplication.

Steam Ingress. The addition of water to the blanket will soften the neutron spectrum and, due to the low plutonium inventory, will result in reduced reactivity.

Fuel Rearrangement. Major changes in the fuel geometry could potentially be caused by earthquakes or fuel melting. These would be major accidents with possible serious consequences. It is expected, however, that the blanket burnup will be low enough so that criticality will be impossible. Criticality thus will not add to the risks of fuel rearrangement.

Flow Blockage Accidents. Blockage of flow to an entire module will be avoided by adequate redundant and diverse design features. Local flow blockage could conceivably occur and cause local fuel failure. Propagation of such damage must be limited so that cooling geometry is not extensively altered. Based on experience with analysis of the gas-cooled fast reactor, minimum times for fuel damage propagation have been estimated. Assuming full power operation, total coolant blockage and increased multiplication due to fuel melting, it is estimated that the minimum time for fuel damage to propagate from one assembly to the adjacent assembly is 15 seconds. The blanket coolant activity detectors and delayed neutron detectors are estimated to be able to detect cladding failure in less than 5 seconds. Reactor trip in the MHFFR occurs almost instantly upon initiation of the trip signal. Thus it is expected that fuel damage will not propagate in the event of a flow blockage accident.

Steam Leakage Accidents. Rupture of a steam generator tube could introduce water into the blanket coolant stream. This is expected to cause no significant damage to the blanket but could increase the helium loop pressure and ultimately cause the loop pressure relief valve to lift, venting coolant-borne activity to the containment. In the event of a water leak, the loop moisture monitors trip the plant, isolate the loop and dump the steam generator. Even if the wrong steam generator is dumped, the water ingress accident does not constitute a serious risk to the plant or the public. Steam leakage into the containment building or turbine hall is similarly not expected to constitute a significant hazard.

Fuel Handling Accidents. The method of fuel handling during refueling was not addressed in this study. Due to the decay heat, the modules will probably have to be actively cooled during refueling and transport. Adequate cooling will have to be guaranteed. A reliable means of handling the massive modules will have to be developed.

Helium Leakage Accidents. The primary hazard from helium leakage is from coolant borne fission products and tritium. Preliminary estimates indicate that tritium contributes little to this risk and that the release of the circulating activity to the atmosphere would result in off-site doses within 10 CFR 100 guidelines.

A double-ended main helium duct rupture could constitute a significant hazard to the MHFFR and will have to be designed against. The danger from a main duct rupture is the potential for causing further damage that could compromise the ability to cool the blanket. Support and containment of the helium ducts has not been addressed in this study. A number of alternatives are possible including the use of the supports and snubbers as used in LWR's, the use of metallic "rip stops" on the ducts to restrict the maximum size of any duct ruptures and the use of partial or even total prestressed concrete reactor vessel containment of the ducting and primary loop components.

DESIGN BASIS ACCIDENT

The design basis accident (DBA) is a hypothetical, non-mechanistic accident that is not expected to happen but that is postulated for the analysis of engineered safety capabilities. The DBA is initiated by a sudden depressurization of one of the helium loops. It is postulated that one of the two main helium circulators is rendered inoperable by this accident. The capabilities of the reactor component designs are such that occurrence of this DBA and the loss of the remaining main helium circulator, loss of off-site power and occurrence of the design basis earthquake could be tolerated without undue risk to the public. Detailed system reponse analysis to occurrence of such a severe hypothetical accident has not been done for the MHFFR. Nevertheless, the MHFFR conceptual design safety philosophy and component specification are sufficiently similar to those used in the GCFR that the MHFFR could probably be designed to withstand such an accident without undue risk to the public.

LICENSING

On the basis of the discussion above the MHFFR appears to possess no inherent features that could compromise the safety aspects of the reactor. An adequate method for support and containment of the massive blanket modules, helium ducts and primary loop components needs to be developed. If this can be done, the MHFFR should be licensible under existing Nuclear Regulatory Commission regulations.

COST

Some very preliminary cost estimates were made, based on the conceptual design for the hybrid blanket and power conversion system. The blanket must be replaced approximately every four years. Its cost should thus be considered part of the fuel cycle cost rather than a capital expenditure. The blanket cost was estimated to be $94 million. Since one quarter of the blanket is replaced at each reload interval, the reload blanket cost is $23.5 million.

The cost was estimated for the nuclear steam supply system (NSSS) to be $295 million or $332/Kw$_e$ for a net electrical output of 889 MW$_e$. Included in the estimate is the primary coolant system, the auxiliary coolant system, refueling equipment and the instrumentation and controls associated with the primary coolant system. The NSSS cost estimate specifically excludes the nuclear plasma island, the fuel and such equipment that are normally considered in the balance of plant. The estimates include project engineering costs, indirect costs, and an allowance for contractural risk and fees. The cost estimates are based on July 1975 dollars and do not include escallation or interest during construction. The estimates are for an "equilibrium" plant; first-of-a-kind engineering and development costs are not included.

CONCLUSIONS

The purpose of the work summarized here was to assess the feasibility of applying gas-cooled fission reactor technology to the mirror hybrid fusion-fission reactor. The conceptual designs of the blanket and power conversion system presented here are directly derived from General Atomic's design experience with the High Temperature Gas-Cooled Reactor and the Gas Cooled Fast Reactor. From the analysis of the conceptual design we have concluded that the hybrid blanket and power conversion system concepts are technically feasible. Existing gas-cooled fission reactor technology is directly applicable to the mirror hybrid reactor. There are a number of aspects of the present design that may present problems. Nevertheless, in general, the design appears feasible.

There are no inherent features of the hybrid concept that present fundamentally new safety considerations to the reactor design. Further work is needed, however, in the area of the primary loop ducting design, support and containment to assure that the reactor will be adequately cooled under all circumstances. The apparent total absence of potential accidents that could cause the system to become supercritical does offer the hybrid reactor some advantages compared to fission reactor systems. If the maintenance of cooling can be adequately assured, the hybrid reactor could be licensed under current Nuclear Regulatory Commission regulations.

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TOKAMAK
FUSION-FISSION
REACTOR DESIGNS

PLASMA PHYSICS BASIS FOR THE TOKAMAK HYBRID REACTOR WITH INJECTION

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INTRODUCTION

The many years of successful research by the collective of the I.V. Kurchatov Atomic Energy Institute and the studies of many laboratories throughout the world in recent years have advanced the Tokamak-type systems to the forefront among the laboratories working in the field of controlled thermonuclear fusion.

The first experimental results on the T-10 device[1] indicate that the product of the plasma density times the energy lifetime $n\tau_E$ reaches a value of 5×10^{12} cm^{-3}s at an ion temperature of ≈ 0.7 keV and an electron temperature of ≈ 1 keV.

On the other hand, on the TFR device[2] records have been set with respect to the ion temperature when heating a plasma by fast-neutral injection. When working with deuterium in a plasma with a density of $\approx 6 \times 10^{13}$ cm^{-3}, an average ion temperature of ≈ 1 keV was obtained for a temperature on the torus axis of ≈ 1.6 keV.

The results of these studies permit highly reliable extrapolation of the behavior of a plasma in the reactor regime. The Tokamak hybrid reactor with injection, in which the Q-factor of the reactor with respect to the plasma $Q_{pl} \approx 1.7$ is increased to $Q \approx 10$ by placement of uranium-238 in the reactor blanket, can be considered the first step toward the creation of the thermonuclear reactor with DT fuel ignition[3]. As the calculations of Ref. (3) demonstrated, in this case it is possible to conceive of an economically advantageous stationary reactor with total fuel efficiency $\eta \approx 0.3$ and a 14-MeV neutron flux to 6×10^{13} cm^{-2}.s^{-1}. However, in these calculations it is assumed that the value of q characterizing the stability margin in the Tokamak plasma is 1.5. A value of q(a) this small has not been obtained at the boundary of the plasma column in any experiment. Moreover, for a current distribution in the plasma column that is close to parabolic, it is hardly possible to hope for a value of q(a) less than 2.5 without taking additional measures to modify the plasma column. Thus, hereafter we shall consider $q \approx 3.0$.

Three years have passed since the appearance of the first calculations for the stationary hybrid reactor based on the Tokamak,[3] and new experimental and calculated data have been accumulated with respect to certain problems which can influence the overall design as a whole. The possible effect of the new information on the reactor parameters is discussed in this paper.

BASIC ADVANTAGES OF THE TOKAMAK HYBRID REACTOR

The design of the hybrid reactor based on the Tokamak presupposes the realization of a two-component regime in the Tokamak plasma by the injection of deuterons with an energy of $E_0 \approx 200$ keV. To achieve a Q-factor of $Q_{pl} \approx 1.7$, it is sufficient to have a product of the plasma density times the energy lifetime $n_e\tau_E \approx 3 \times 10^{13}$ cm^{-3}·s (see Fig. 1); that is, for the plasma densities already achieved it is necessary to increase the energy lifetime by only 5-6 times and the plasma temperature by about 10 times to $T \approx 10$ keV.

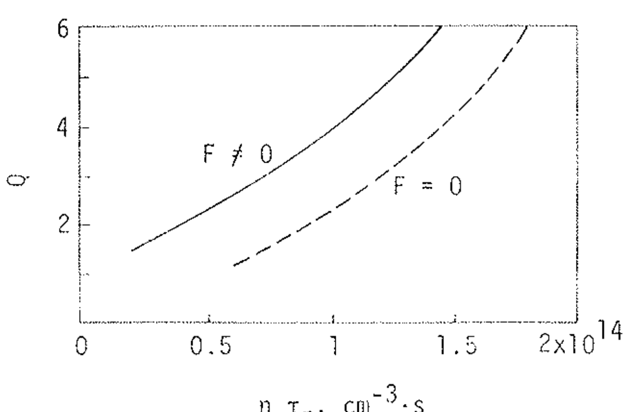

Fig. 1. Q-factor of the reactor Q_{pl} as a function of $n_e\tau_E$ at T = 10 keV, $n_d = n_t$.

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*Translated from the Russian by Addis Translations International, Portola Valley, Ca. 94025

In a reactor with ignition, the thermonuclear reaction is maintained at the expense of the energy of the α-particles which are decelerated on the plasma electrons and ions. Therefore, in addition to satisfying the Lawson criterion for the operation of the reactor, it is necessary that the lifetime of the α-particles in the plasma be no less than their braking time, that is, a value of $\tau \approx 1$ s for $n \approx 10^{14}$ cm^{-3} and $T \approx 15$ keV. This additional condition which is required for a reactor with ignition is unnecessary in the case of a reactor with injection. Fig. 2 shows the Q-factor of the reactor Q_{pl} as a function of $n_e\tau_E$ considering heating of the plasma by the α-particles (the solid curve) and without considering the α-particles in the plasma energy balance (the dotted curve) for the DT-reactor on injection of a deuteron beam with $E_0 = 200$ keV into the plasma. It is obvious that in the two-component Tokamak regime ($n_e\tau_E \approx 3 \times 10^{13}$ cm$^{-3}\cdot$s) the presence of the processes leading to the instantaneous drift of the α-particles from the plasma to the chamber wall has in practice no effect on the Q-factor of the reactor.

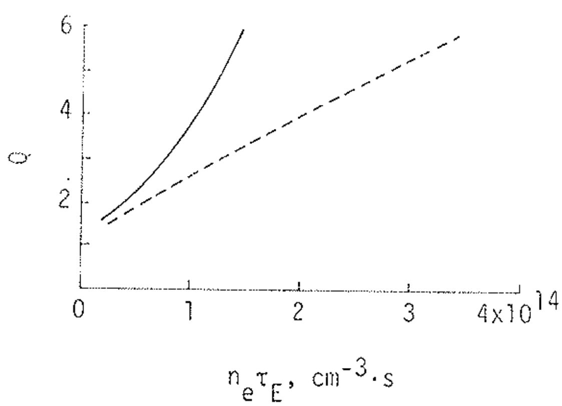

Fig. 2. Q_{pl} as a function of $n_e\tau_E$ considering the α-particle heating (solid curve) and without considering the α-particles (dotted curve).

As is known, the two-component regime of the Tokamak reactor is also attractive in the sense that, in order to realize this regime, it is sufficient for only the plasma electrons to be hot, while the ions can be cold, that is, $T_e > T_i$. A possible reactor operating regime such as this also appears attractive to us, since in this regime one may expect less sputtering of the material of the first wall of the reactor, for the coefficient of desorption of the gas from the metal surface under the effect of electron impact is 10^2-10^3 times less than the sputtering coefficient of the wall material arising from the fast ions and neutrals.[4] This, in turn, must promote less influx of impurities in the plasma in a two-component Tokamak as compared to the reactor with ignition (see Fig. 3).

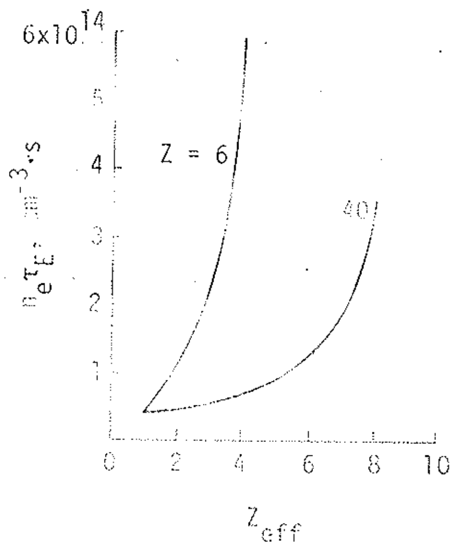

Fig. 3. $n_e\tau_E$ as a function of Z_{eff} for two values of Z: 1 - 6, 2 - 40, for $T = 10$ keV, $n_d = n_t$, $Q = 2$, $E_0 = 200$ keV, $F = 1.1$.

SELECTION OF THE INJECTION ENERGY AND THE IONIZATION OF FAST NEUTRALS IN A PLASMA

As is known,[3] a deuteron energy of $E_0 = 200$-300 keV is most advantageous from the point of view of achieving maximum values of the reactor Q-factor. However, this energy range is disadvantageous from the standpoint of the efficiency of obtaining the atomic fluxes both during charge exchange of the positive ions in the gas and when using the cycle with negative ions. This fact forces us to consider the first demonstration designs with an injection energy of $E_0 < 200$ keV. The use of an injection energy of $E_0 \leq 100$ keV can lead to so-called injection instability connected with intense bombardment by the charge-exchange neutrals of the first reactor wall near the injection point. Fig. 4 illustrates the charge-exchange neutral distribution over the reactor wall with the parameters $\bar{n} = 7 \times 10^{13}\ cm^{-3}$, $\bar{T} = 15$ keV for different values of injection energy. The proportion of neutrals drifting to the chamber wall as a function of the injection energy is presented in Fig. 5. Fig. 6 illustrates the spectrum of the charge exchange neutrals drifting to the chamber wall.

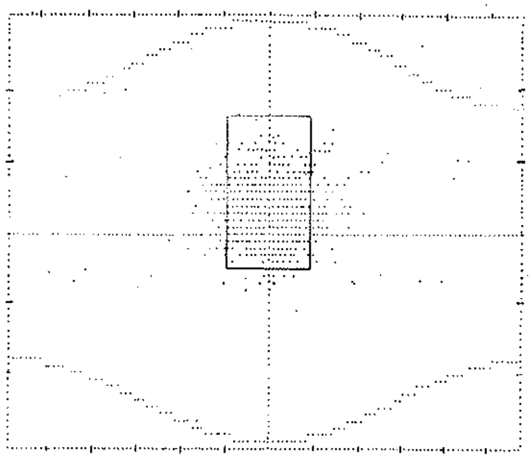

Fig. 4. Charge-exchange neutral distribution over the chamber wall, $E_d = 80$ keV, $T_{max} = 20$ keV, $n_{max} = 10^{14}\ cm^{-3}$.

From the data presented, it follows that in order to decrease the amount of impurities entering the reactor plasma from the wall, the injection energy must be increased above 160 keV. Increasing the injection energy is also useful with regard to maintaining a given density in the reactor, for the corpuscular lifetime of the particles exceeds the energy lifetime.

Recent indication that the presence of impurities in the reactor plasma can lead to a decrease in the depth of penetration of the atomic beam into the plasma also indicates the necessity of choosing a higher injection energy. Thus, the injection energy of $E_0 = 200$ keV selected in the initial version of the design[3] is still sufficiently well substantiated today.

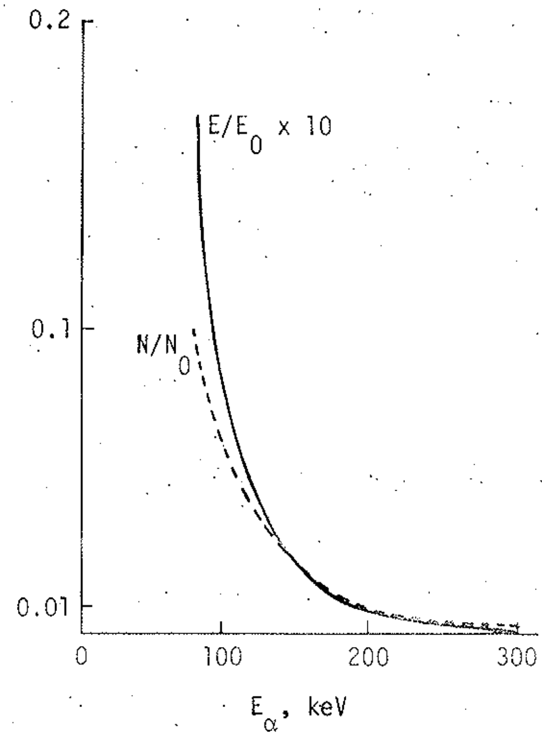

Fig. 5. Proportion of the charge-exchange neutrals drifting to the chamber wall as a function of the injection energy.

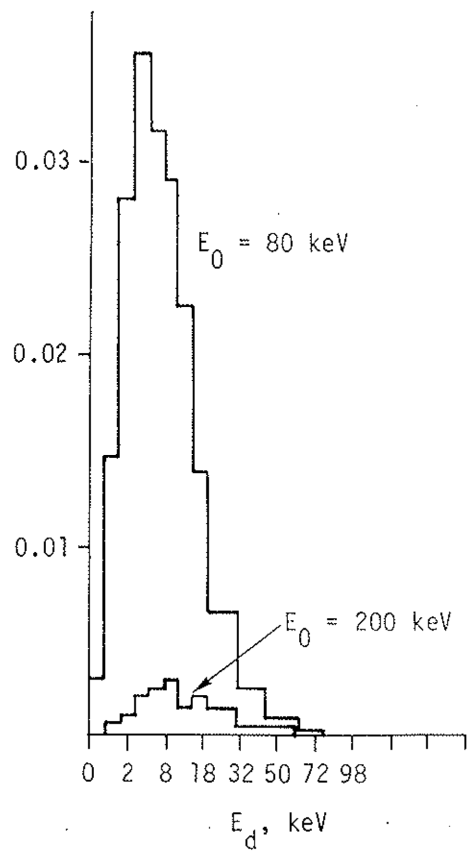

Fig. 6. Spectrum of the charge-exchange neutrals drifting to the first wall of the reactor.

STABILITY OF THE ION BEAM IN A PLASMA

The recent intense theoretical studies of the problem of the stability of the ion beam in the Tokamak plasma[5,6] have revealed that in a isothermal plasma with $T_e \approx T_i$ the most dangerous instability may turn out to be the Alfvén wave instability. However, even in the presence of this instability it is possible to point out regimes in which it will not develop, for example, on injection of the beams at different angles to the magnetic field. In the nonisothermal plasma with $T_e > T_i$, the instability of the oblique ion-sound vibrations is the most dangerous. The theoretical studies of Ref. 7 as well as the model experiments Ref. 8 and 9 indicate that it will apparently be impossible to insure a stable state for the ion beam in the two-component Tokamak plasma at $T_e \gtrsim 3\ T_i$. The ion-sound instability developing here will lead to intense scattering and braking of the ion beam. Unfortunately, this reduces the possible advantages of the two-component regime.

DEVICES FOR PUMPING PARTICLES

In order to create a stationary reactor based on the Tokamak with injection, it is necessary to provide for pumping of the injected particles after their deceleration. The particle fluxes which must be dealt with are such that to remove them from the volume it is necessary to achieve pumping velocities corresponding to particle velocities $\approx 10^7$ cm/s. This leads to the unavoidable use of the devices which are known as divertors. The application of one type of divertor or another complicates the reactor structurally and increases the volume of the magnetic field for the same volume of plasma. At this time it is difficult to give preference to any one type of divertor, but the fact that without such a device the stationary reactor based on the Tokamak is impossible remains unquestioned.

The structural development of the hybrid reactor with a divertor can lead to a reduction in the initial output data of the reactor.

PLASMA PARAMETERS

Recent experiments on the ALCATOR, T-10, T-4, and TFR Tokamaks have demonstrated that from the point of view of the accumulation of impurities in the plasma, the regimes with high density ($n \gtrsim 10^{14}\ cm^{-3}$) are preferable. Under other equal conditions, increasing the plasma density leads to an increase in $\beta_\phi = 8\pi 2nT/H_\varphi^2$ or an increase in the current in the plasma. On the other hand, increasing the stability margin q means using a system with a large magnetic field H_0 and, consequently, implies an increase in the critical value of the magnetic field on the surface of the superconductor.

Thus, the choice of the following reactor parameters appears at the present time to be the most justified: $q \approx 3.0$ instead of 1.5, $\bar{n} = 6\text{-}10^{13}\ cm^{-3}$, $T = 10\text{-}15$ keV, $\beta \approx 2$, $\tau_E = 0.5\text{-}0.3$ s, $a = 1.2\text{-}1.4$ m, $R/a \approx 4$.

These parameters lead to somewhat worse output data for the reactor compared with Ref. 3. However, it is obvious that with the exception of the value of q, from the present standpoint the choice of the remaining parameters in Ref. 3 has proven to be close to optimal.

CALCULATING THE REACTOR Q-FACTOR

The single-particle approximation was used in the first calculations of the reactor Q-factor to calculate the probability of the DT reaction during braking of a fast deuteron. The question of the accuracy of this approximation has been raised. For this reason, calculations were made of the plasma ion distribution function, by means of which it is possible to determine precisely the energy yield of the two-component system. The question of the effect of the collisions of the beam particles with each other on the ion distribution function and, at the same time, on the magnitude of the Q-factor of the two-component system has been clarified. The evolution of the system was considered for plasma velocities uniform in space and isotropic in space: the deuterium beam in the deuterium-tritium plasma, assuming a sufficiently large energy lifetime. The results are presented in Figs. 7, 8, and 9. From the figures it is obvious that, compared with the quasi-linear approximation, a consideration of the collisions of the beam particles with each other leads to little variation in the deuteron distribution function and to an increase of less than 5% in the energy yield.

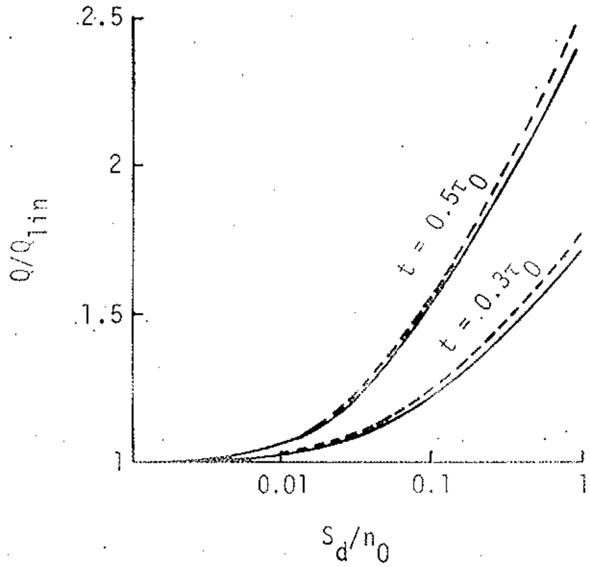

Fig. 7. Reactor Q-factor as a function of beam density.

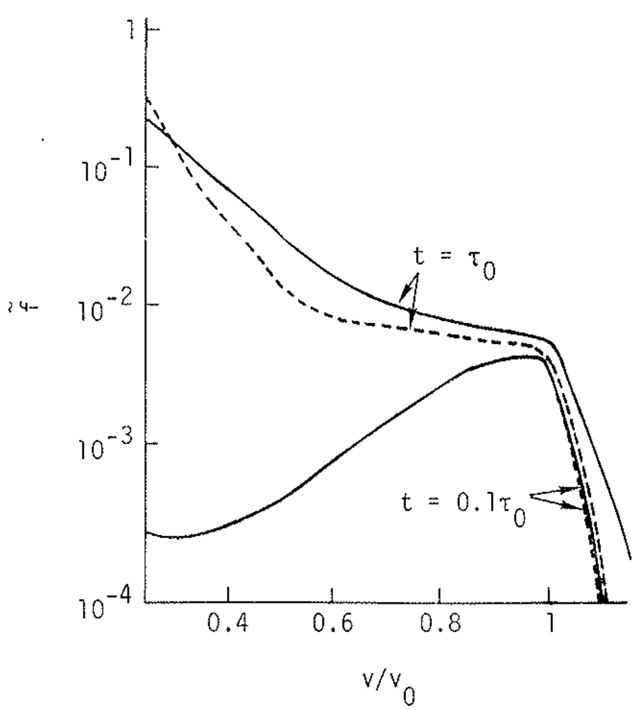

Fig. 8. Plasma deuteron distribution function.

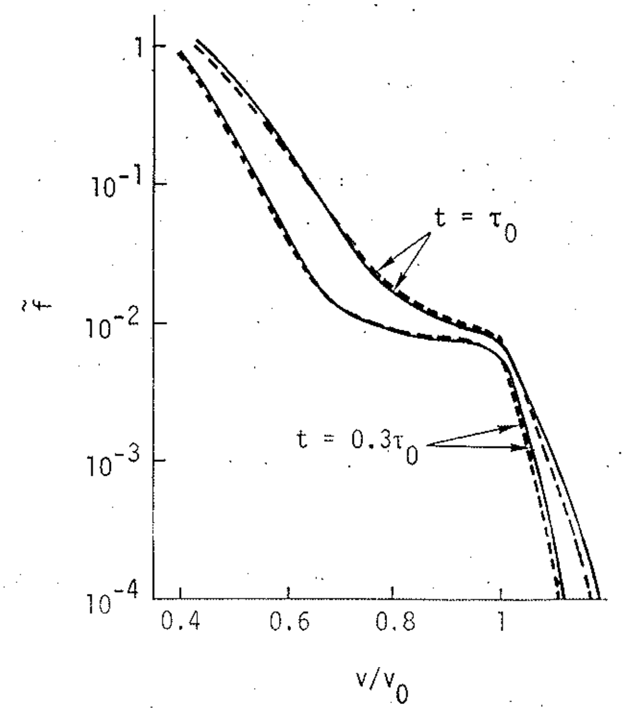

Fig. 9. Plasma deuteron distribution function.

Thus, for calculations of the Q-factor of the model close to the real two-component Tokamak, it is sufficient to solve the quasi-linear problem in which the collisions of the beam particles with each other are not considered, but the effect of the beam on the basic plasma is taken into account. The error in determining the Q-factor does not exceed 5% in comparison with the exact nonlinear problem. This conclusion is quite general in spite of the fact that in the given model the tritium ion temperature was considered to be constant, and the contribution of the tritium ions to the process of relaxation of the deuteron beam was not taken into account.

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OPTIMIZATION OF FUSION-DRIVEN FISSIONING SYSTEMS

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ABSTRACT

Potential advantages of hybrid or fusion/fission systems can be exploited in different ways. With selection of the U^{238}-Pu^{239} fuel cycle, we show that the system has greatest value as a power producer. Numerical examples of relative revenue from power production vs. Pu^{239} production are discussed, and possible plant characteristics described. The analysis tends to show that the hybrid may be more economically attractive than pure fusion systems.

I. Introduction

A fission system has the advantage of generating a large amount of energy (~ 200 MeV) per individual neutron-induced event. A fusion system generates a smaller amount (~ 20 MeV) per individual event, but a larger amount per unit mass of fuel consumed. Either system alone has difficulty with the neutron economy, on the one hand because the fission neutron spectrum is soft (yielding LWR conversion ratios of ~.6), and on the other because the principal fuel-breeding reaction,

$$Li^6 + n \rightarrow T + He^4 \quad , \qquad (1)$$

produces only one fuel nucleus per neutron absorbed, and the fusion reaction,

$$D + T \rightarrow He^4 + n \quad , \qquad (2)$$

makes only one neutron.

However the neutron from (2) is highly energetic (14 MeV), and the interaction of this fast neutron with fissionable materials can remove the difficulty of the neutron economy thereby making possible systems superior to those utilizing fission or fusion alone.

Hypothetical studies of fusion power plants[1,2] indicate that although apparently possible, it is difficult to find an economic fusion system. Should the current research programs prove that high temperature plasma confinement cannot comfortably exceed the traditional Lawson criterion, achievement of practical pure fusion systems may prove very difficult indeed.

Cost estimates of fusion power systems, even those done very carefully as in the cited references, must be viewed with considerable skepticism since these systems have not yet been proved possible and may require systems or ancillaries not anticipated in these studies.

Although any complete evaluation of the economy of a proposed system must include a valid cost estimate of the plant, we seek in this paper to discuss a more restricted question: Is a hybrid fusion-fission system more valuable as a power producer or as a generator of fissile fuel for use in fission reactors? Implicit in our argument is the assumption that the total investment in the final plant will prove relatively insensitive to the design changes necessary to shift the use of the neutrons liberated in the machine between fuel production and power production. Below we describe the neutron balance of the system, and various ways to use the neutrons to generate revenue.

II. Blanket Neutron Balance

The first step in the analysis is to derive an expression for the neutron balance in the hybrid reactor blanket that relates the fissile fuel production to the energy multiplication (fissile fuel burn-up). This may be found by simply equating the neutrons available in the blanket (source plus multiplication) to the neutrons absorbed in the blanket plus leakage from the blanket. The total number of neutrons in the blanket (normalized to one plasma source neutron) is written as:

$$\text{Neutrons available} = 1 + \varepsilon + \hat{\nu}c + \nu s \qquad (3)$$

where

$1 \equiv$ the source neutron

$\varepsilon \equiv$ neutron multiplication through (n,2n) and (n,3n) reactions in the blanket

$c \equiv$ number of (fast) fissions in fertile nuclei (e.g., U^{238} or Th^{232}) per source neutron

$\hat{\nu} \equiv$ average number of neutrons produced per fission of fertile nuclei

$s \equiv$ number of fissions in fissile nuclei per source neutron

$\nu \equiv$ average number of neutrons produced per fission of fissile nuclei.

This equation (3), of course, represents a spectral average for $\hat{\nu}$ and ν since they will be strong functions of energy[3]. It is really a one group average, i.e., we define ν by

$$\nu \equiv \frac{\iint \nu(E)\sigma_f(E)n(\underline{r})\phi(E,\underline{r})d\underline{r}dE}{\iint \sigma_f(E)n(\underline{r})\phi(E,\underline{r})d\underline{r}dE} \qquad (4)$$

where $\sigma_f(E)$ is the fission cross section, $n(\underline{r})$ is the nuclear density, and $\phi(\underline{r},E)$ is the neutron flux. This one group approximation for $\hat{\nu}$ and ν will be satisfactory for our analysis since we are only interested in general trends, and the neutron spectrum is relatively insensitive to the limited variations to be made.

The absorption of the neutrons in the blanket may be written as

$$\text{Neutrons absorbed} = T_6 + p + C + S \qquad (5)$$

where

$T_6 \equiv$ number of neutrons absorbed in Li^6 per source neutron

$p \equiv$ parasitic absorptions (e.g., in structure and shield) and leakage from the blanket per source neutron

$C \equiv$ number of neutrons absorbed in fertile nuclei per source neutron

$S \equiv$ number of neutrons absorbed in fissile nuclei per source neutron.

We assume all absorptions in the fertile and fissile nuclei either produce a fission or radiative capture (n,γ) event, so that

$$C = c + \gamma \qquad (6)$$

where γ is the number of capture events in the fertile nuclei per source neutron. We choose to write S as

$$S = s(1 + \alpha) \quad (7)$$

where α is the capture-to-fission ratio. Again, α is a one-group average to $\alpha(E)$ and is defined in a manner similar to equation (4), i.e.,

$$\alpha \equiv \frac{\iint \sigma_\gamma(E) n(\underline{r}) \phi(E,\underline{r}) d\underline{r} dE}{\iint \sigma_f(E) n(\underline{r}) \phi(E,\underline{r}) d\underline{r} dE} \quad (8)$$

where $\sigma_\gamma(E)$ is the capture cross section.

A parameter of interest in defining the blanket performance is F, the net number of fissile fuel atoms produced per source neutron, defined as

$$F = \gamma - S \quad . \quad (9)$$

By equating (3) and (5) and using (6), (7), and (9), we arrive at an expression for the neutron balance in the blanket:

$$\begin{aligned} &1 + \varepsilon + \hat{\nu}c + \nu s \\ &= T_6 + p + c + F \\ &+ 2s(1 + \alpha) \quad , \end{aligned} \quad (10)$$

which may be rewritten to yield a direct relation between F, the net number of fissile fuel atoms produced in the blanket, and s, the number of fissions of fissile atoms:

$$\begin{aligned} F = &(1 - T_6) + (\varepsilon - p) \\ &+ c(\hat{\nu} - 1) \\ &+ s[\nu - 2(1 + \alpha)] \quad . \end{aligned} \quad (11)$$

We could also write this balance in terms of M, the system energy multiplication defined as:

$$M = \frac{1}{17.6}\,[17.6 + E_f(c+s)] \quad (12)$$

where E_f is the energy release per fission ($\simeq$ 200 MeV), and 17.6 MeV is the fusion reaction energy.

Thus equation (11) is the basic form of the neutron balance that relates the fissile fuel production (F) to the blanket fissile fission rate (s) and will be used in the next section in the calculation of the value of a mole of neutrons. As an interesting aside, equation (11) shows that F will increase with s (i.e., a breeding lattice) if

$$\nu - 2(1 + \alpha) > 0 \quad . \quad (13)$$

This is simply the well-known requirement from fission technology that for a reactor to breed, the quantity η must be greater than 2, where η is defined to be the number of neutrons produced per neutron absorbed in a fissile nucleus. Since

$$\eta = \frac{\nu}{1 + \alpha}$$

the requirement in equation (13) is the same as that of a breeder fission reactor.

III. Revenue from a Mole of Source Neutrons

If we consider the hybrid reactor as an independent device capable of producing large quantities of 14 MeV D-T neutrons, we can design the blanket either to emphasize the production of fissile fuel or the generation of power. The trade-off between these two products can be evaluated by calculating the potential value of a certain quantity of neutrons (say one mole) in terms of revenue from the sale of electricity and from the sale of fissile fuel.

The revenue from the sale of power, R_M, in dollars per mole of source neutrons, may be written as

$$R_M = 4.71 \times 10^5 \, M \, \eta_{sys} \, e \qquad (14)$$

where

$\eta_{sys} \equiv$ the reactor power plant efficiency (discussed in detail below)

$e \equiv$ price of electricity, dollars/kwh.

The revenue from the sale of fissile fuel, R_F, in dollars per mole of source neutrons, may be written as

$$R_F = F \, A \, P \qquad (15)$$

where

$A \equiv$ atomic weight of the fissile fuel

$P \equiv$ price of the fissile fuel, dollars per gram.

Thus, the total revenue R, in dollars per mole of source neutrons, is the sum of R_M and R_F:

$$R = 4.71 \times 10^5 \, M \, \eta_{sys} \, e + F \, A \, P \quad . \qquad (16)$$

System Efficiency

The reactor system efficiency is defined as

$$\eta_{sys} \equiv \frac{\text{Power for Sale}}{\text{Total Power Produced}} = \frac{P_S}{P_T} \qquad (17)$$

For a beam driven reactor with total beam injector power P_I, the total power produced is

$$P_T = P_I \, Q \, M \qquad (18)$$

where Q is the ratio of fusion power to beam power. Typically Q will be on the order of one for a beam-driven system,[4] but can be quite large for an ignition device. Here M is the multiplication of the fusion neutron power in the blanket, defined in equation (12).

The power for sale, P_S, is written as

$$P_S = \eta_{th} \, P_T - \frac{P_I}{\eta_I} = P_I\left(\eta_{th} \, Q \, M - \frac{1}{\eta_I}\right) \qquad (19)$$

where η_{th} is the thermal conversion efficiency and η_I is the beam injector efficiency.

Then from equations (17) - (19) the system efficiency is

$$\eta_{sys} = \eta_{th}\left[1 - \frac{1}{MQ\eta_I\eta_{th}}\right] \quad . \qquad (20)$$

Thus, since η_{sys} is a function of M, we can combine equations (16) and (20) and use equations (11) and (12) (with E_f = 200 MeV) to eliminate F to find an expression for R(M):

$$R = 4.71 \times 10^5 \, \eta_{th} \, e[M - \frac{1}{Q\hat{\eta}_{th}\eta_I}]$$

$$+ AP\Big|(1 - T_6) + (\varepsilon - p) + c(\hat{\nu} - 1)$$

$$+ [\nu - 2(1+\alpha)] \times [0.088(M-1) - c]\Big| \quad . \tag{21}$$

Hence equation (21) is a relation for the value of a mole of source neutrons in terms of a number of variables. Eight of these - A, T_6, ε, p, ν, α, c, and $\hat{\nu}$ - are directly related to the blanket materials and their neutronic performance. Three others - P, e, and η_{sys} - depend on the market the hybrid is serving and on the overall reactor plant design.

Example One

For an example the following values for the neutronic variables: $\hat{\nu} = 3.9$, $\nu = 2.9$, $\alpha = 0.72$, $c = 0.35$, $T_6 = 1$, $\varepsilon = p$, $E_f = 200$ MeV were adopted. These were found to be typical of several blankets.[5] Also assume $\eta_{th} = 0.40$, $\eta_I = 0.50$, $Q = 1$, a price for electricity $e = 0.02$ \$/kWh, and Pu fuel (A = 239). Then equation (21) reduces to

$$R = 3.77 \times 10^3 \, M + P(299.1 - 11.4\, M)$$

$$- 1.88 \times 10^4 \quad . \tag{22}$$

This is plotted in Fig. 1 for four different values of P = \$35, \$100, \$331, and \$400 gm/Pu. The break-even price for this example is \$331 - extremely high in today's market (~ \$35/gm). Hence we conclude it is better to optimize the blanket for power production. However, as the blanket multiplication approaches very high values (M ~ 25 - 30)

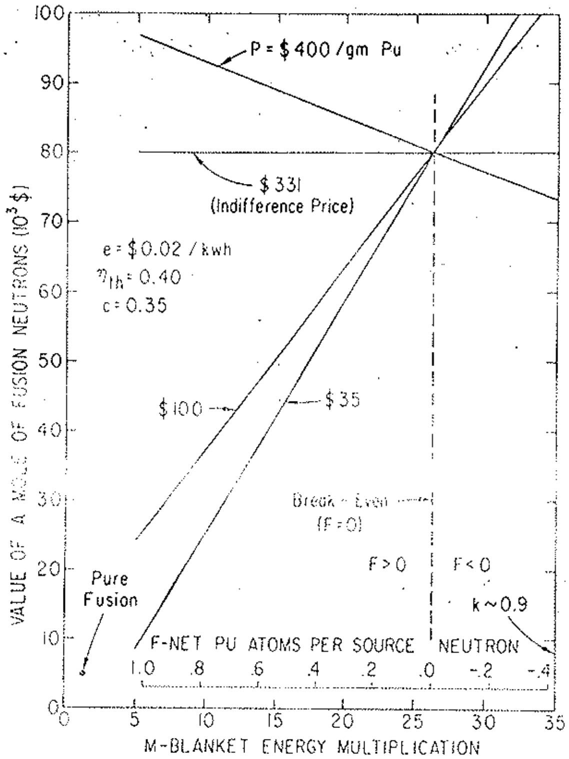

Figure 1. Neutron revenue vs. blanket energy multiplication M for c(fertile fissions per source neutron) = 0.35.

three significant problems arise which tend to limit the desirability of this case. One factor is the very high inventory of Pu needed to achieve these values of M; another is that the blanket may become close to critical. Also, the thermal power output of the reactor could be too large to be practical, even for a low Q beam-driven reactor. Hence the most desirable situation would seem to be one in which M is as large as practical (~ 20) within the inventory, criticality, and total power output limits; there would then still be a substantial production of excess Pu for sale to fission reactors which would generate additional revenue for the hybrid plant. A similar calculation for a pure fusion reactor, assuming a total energy yield of 22 MeV per fusion event (M = 1.25), gives the value of a mole of fusion neutrons in such a system to be \$4700, also plotted on the graph. Hence, the hybrid blanket can significantly increase the value of the fusion neutrons.

To investigate the general case of the break-even price of Pu, we can take the derivative of equation (21) with respect to M and set it to zero to solve for P_c, the Pu indifference price,

$$P_c = 5.35 \times 10^6 \frac{\eta_{th}\, e}{A[2(1+\alpha) - \nu]} \quad . \tag{23}$$

Clearly the price, P_c, will be very sensitive to the neutronic parameters, α and ν, as well as η_{th} and e. However for realistic values we always find P_c is about \$300 - indicating it is more profitable to optimize for power production.

Example Two

In the above example we used a value for c, the number of (fast) fissions in U^{238} per source neutron, of 0.35, which yields a multiplication (M) of 5.0 and fissile production rate (F) of about 1.0 with no Pu fissions in the blanket. The value of c will of course strongly depend on the fast fission (converter) zone thickness and material composition (particularly through the ratio of fertile material to structural material); the value c = 0.35 was found to be typical of several blankets investigated.[5]

However, if a different neutron converter design is utilized, a larger number of fast fissions may occur. As a second example, we will use c = 0.63, which gives $M \simeq 8.2$ and $F \simeq 1.8$ from equations (11) and (12) with s = 0 (no Pu fissions) and assuming $T_6 = 1$, $\varepsilon = p$, and $\hat{\nu} = 3.9$. These values of c, M, and F are approximately those of the blanket design of reference 6.

For the sake of comparison, we can derive an expression for the revenue per mole of neutrons similar to equation (22) except for the larger value of c = 0.63 (with the same values for ν, α, η_{th}, etc. as in example one):

$$R = 3.77 \times 10^3\, M + P(529.3 - 11.4\, M) - 1.88 \times 10^4 \quad . \tag{24}$$

This is plotted in Fig. 2., where we see that the Pu indifference

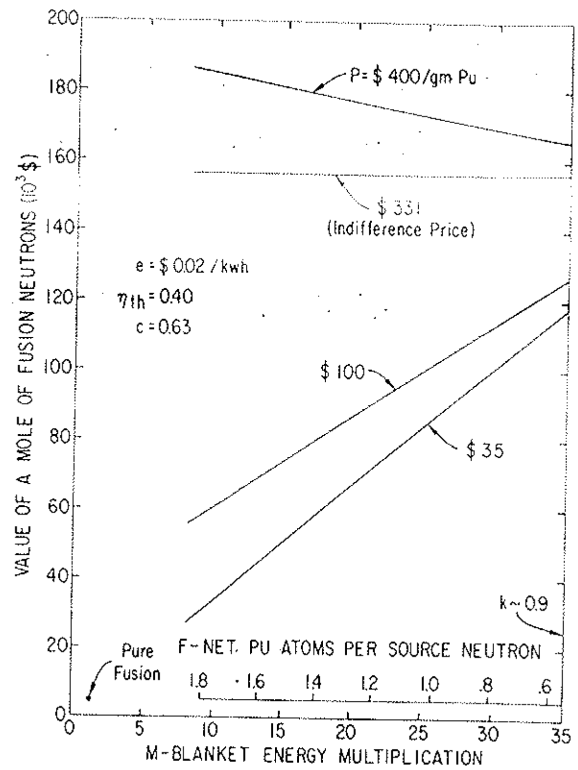

Figure 2. Neutron revenue vs. blanket energy multiplication M for c(fertile fissions per source neutron) = 0.63.

price P_c is still the same as example one, i.e. about \$331/gm Pu. This is shown by equation (23) for P_c, which is independent of c (fertile fissions). However, what the larger c does is to increase the revenue (R) for the same M and P relative to example one. This is due to the fact that the increase in c results in a larger F for the same blanket multiplication.

By comparing Figs. 1 and 2, we can see that for the same blanket power production the revenue from the neutrons increases with c; e.g. for P = \$100/gm and M = 10, blanket 2 yields about a 60% increase in R over blanket 1. Also, if we wish to obtain the same revenue from each blanket, then for blanket 1 some of the Pu must be burned to raise M (and at the same time lowering F), while blanket 2 can operate at a lower M, producing more fuel. Nonetheless, if the Pu price is less than the indifference price then the value of the neutrons in either blanket can be further increased by optimizing for power production.

IV. Possible Plants

To avoid a large initial inventory, it would be possible to begin plant operations as a fuel producer and after a period of time recycle the produced fuel to increase M. The plant power output grows in time.

It would be fairly easy to shift the blanket characteristics from high M to high F, the most obvious method being to control the Pu inventory in the blanket. Another way is to vary the Li^6 atom percent in the Li – a lower Li^6 concentration will tend to raise M by allowing more Pu fissions, and the extra fission neutrons can still provide adequate tritium breeding.[5] Conversely, a higher Li^6 density can quench the fission reaction and raise F. Molten salt type blankets would be particularly amenable to this isotopic concentration control.

As an example, take a beam driven ($Q \simeq 1$) machine with the following parameters: $M = 6$, $F = 1.1$, $P_f = 300$ MW, where P_f is the fusion core power. Such a plant would produce 1800 MWth and about 1500 kg Pu/year.

If we are interested in high M ($\simeq 20$) blankets, the required Pu inventory may be about 8000 kg Pu, so that the low M, high F blanket would need to run about 5.5 years to build up this much Pu.

However, suppose we run the low M blanket for two years, producing about 3000 kg Pu at a power level of 1800 MWth. After this two-year period, the 3000 kg of Pu is transferred to the blanket, raising M to about 12 and lowering F to ~ 0.75. Then the plant would run another five years at about 3600 MWth while producing ~ 1000 kg Pu/year, so that at the end of this period the total plant inventory would be 8000 kg – enough for the large M blanket.

Thus, after this initial seven years of operation in which the Pu inventory is built up to 8000 kg, the plant could run at $M = 20$ ($\simeq$ 6000 MWth) and, with $F \simeq 0.3$, could then sell about 400 kg Pu per year for the balance of the plant operation.

V. Optimization of Revenue vs. Minimizing Cost of Power

In this paper we have addressed the question of the total revenue available from a quantity of fusion-produced neutrons. However, this does not directly specify the most economic system, since that question requires plant cost information. As one shifts plant design towards greater power production at the expense of fuel production, one must add heat exchangers, turbines, etc. as well as modify the blanket to handle the higher energy generation. It is clear that accurate inclusion of these effects in order to minimize the cost of power rather than maximize the revenue would reduce the plutonium indifference price, but it would be surprising if the effect were strong. Hence, it seems likely that the most economic fusion-driven system will turn out to be a hybrid producing power at a competitive cost with a low cost byproduct of fissile material.

Acknowledgements

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A TOKAMAK HYBRID STUDY

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ABSTRACT

A report on one year of study of a tokamak hybrid reactor. The plasma is maintained by both D and T beams. To obtain long burn times a poloidal field divertor is required. Both the single null and the double null style of divertor are considered. The blanket consists of a neutron multiplier region containing natural uranium followed by burner regions of molten salt (flibe) loaded with PuF_3 to enhance the energy multiplication. Economic analysis has been applied only recently to a variety of reactor sizes and plasma conditions. Early indications suggest that the most attractive hybrids will have large plasmas of major radius in excess of 8 meters.

INTRODUCTION

We report here on one year of study at the Princeton Plasma Physics Laboratory on the question: Is there a commercially interesting "TCT style" tokamak hybrid that can be built in the 1990's? Implied in this question is the further query: Can such a device be viewed as a practical spin-off from an eminently successful Toroidal Fusion Test Reactor (TFTR) experiment? The answers to these questions have been sought through a conceptual design. Participating in this study, in addition to PPPL personnel, have been an industrial fellow from Stone & Webster Engineering Corporation and personnel from the Battelle Pacific Northwest Laboratories.

The answer to the first question is not yet clear, but the study has revealed that the implied second question must be answered in the negative, namely: The tokamak hybrid appears to be sufficiently larger ($R > 6$ m, $a \sim 2$ m) than the TFTR ($R \sim 2.7$ m, $a \sim 0.85$ m) and sufficiently more complex, by virtue of the necessity for a divertor, that the TFTR cannot be considered to be a good "test bed" for the fusion part of a commercial hybrid. However, the success of a beam driven plasma in the TFTR has been assumed and serves as the foundation of the analysis of the hybrid plasma reactivity.

A second assumption that has been made is that the hybrid plasma will not, or cannot, satisfy the condition that the product of the electron density, n_e, and the energy replacement time, τ_E, be greater than 10^{14} sec/cm^3. For if the hybrid plasma could attain this condition - akin to the so-called Lawson criterion - then the plasma would be near a state of ignition, and the rationale for a beam driven hybrid operation would be questionable.

The study has fallen into three related but somewhat independent areas: the plasma, the blanket, and economic considerations. The plasma and blanket interact through the geometry, size of divertor ducts, and shielding requirements. Economics - in the form of the estimated cost of electricity produced by the hybrid - is to be used to make design decisions. We shall discuss the blanket first.

BLANKET DESIGN

The choice of blanket has its subtleties such as the desire to analyze something different from what others have done, intuition as to what will be easier to make or what will be of more interest in the future, and so on. The question of how much power versus how much fissile fuel the hybrid should make is still not fully resolved in our minds, but for most of the time we have been persuaded in favor of making power. Therefore we have approached the blanket design with the desire to preserve the option of high energy multiplication. High energy multiplication coupled with the necessity for each 14 MeV neutron to cause the breeding of at least one triton places a premium on the need for neutron multiplication in the blanket. What better neutron multiplier for energetic neutrons than U^{238}? Hence we think of a neutron multiplier region containing uranium. We have also had some experience thinking about the molten salt, flibe, as a source of lithium from which to breed tritium. It is easy to remove tritium from flibe. The molten salt can be slowly pumped across magnetic fields and through almost arbitrary shapes - features that raise the possibility of continuous processing of the toroidal blanket. It has a high heat capacity, but its use as a coolant in magnetic fields is dubious because of possible corrosive electrolytic action between the molten salt and the conduit walls. We chose helium gas at some 50 atmospheres as a coolant. Figure 1 shows our standard blanket and shield model from which further variations will be made. The fissile fuel bred is plutonium.

The blanket and shield occupy 150 cm. The neutron multiplier region is imagined to consist of a bank of long tubes filled with hollow slugs of natural uranium - moly alloy and cooled by helium gas. Here the 14 MeV neutrons are scattered and absorbed, fast fissions multiply the neutrons, and plutonium is bred as a consequence of neutrons scattering back from the burner and moderator regions. The burner regions are mostly flibe spiked with some PuF_3. The proportions of BeF_2 and ^3LiF are adjusted to minimize the melting temperature of the salt. The Li^6 content is depleted to 0.83 a/o in order to reduce the neutron absorption even though Li^6 is the principal source of tritium. The carbon moderator thermalizes an otherwise rather energetic epithermal neutron spectrum thereby reducing the captures and enhancing the fissions in the plutonium.

The scavenger region contains flibe with a small amount of UF_4. The Li^6 is enriched to 10 a/o. This region is designed to utilize the fission-born neutrons from the burner region which would otherwise leak into the shield.

At a first wall loading of 0.5 MW/m^2 of 14 MeV neutrons and a blanket energy multiplication of 23, the power density is 15 MW/m^3 in the U-Mo in the converter region, 5 MW/m^3 in the salt in the burner region, and 35 MW/m^3 in the salt in the moderator region. There is no power produced in the carbon moderator and a negligible amount in the scavenger region. A net of 0.34 atoms of Pu^{239} is produced per 14 MeV neutron. The k_{eff} is 0.7.

The blanket energy multiplication and the net fissile fuel production can be varied in approximately a linear fashion with increases in multiplication accompanying decreases in fuel production. The energy multiplication can be varied from near 33, at which value the net fissile fuel production vanishes, to

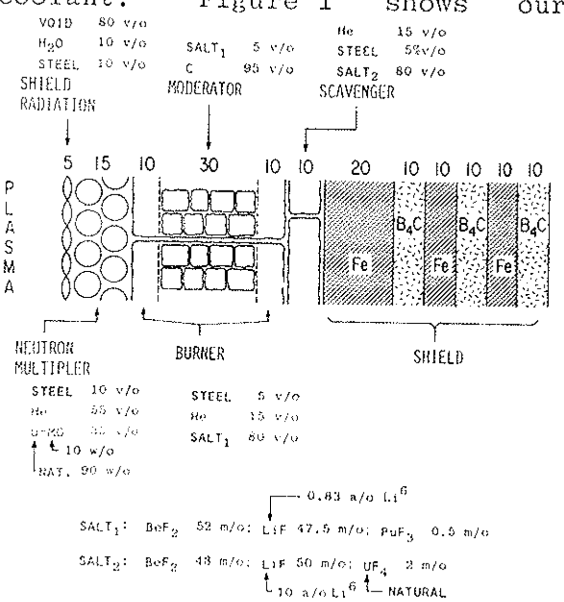

Figure 1. HYBRID SALT BLANKET & SHIELD

below 6. This variation can be produced by changes in: the amount of PuF_3 in the salt, the size of the burner region, or the Li^6 content of the burner salt. The scavenger region can keep the tritium breeding ratio above 1.1 over this range of parameter changes. So much for the blanket.

PLASMA MODEL

A steady state zero dimensional model has been used to describe the plasma. We consider a background plasma consisting of both deuterium and tritium at some bulk temperature common to both the ions and electrons. The composition and the temperature of this plasma is sustained by the injection of energetic neutral beams of tritons and deuterons into the plasma. To permit long pulses without loss of fusion power a divertor is required. The injection energy of the tritons is taken to be 1.5 times the injection energy of the deuterons to ensure equal penetration of the plasma by both beams. The velocity distribution functions for the deuterons and tritons is taken to be isotropic in direction and to be approximated by the sum of a Maxwellian velocity distribution, characterized by the bulk plasma temperature, plus a "hot" distribution of slowing down beam ions. The total reactivity of the plasma is then the sum of four reactivities: that due to the bulk plasma alone, that due to the slowing down deuterons with the background tritons, that due to the slowing down tritons with the background deuterons, and that due to the slowing down deuterons and tritons interacting with each other. Four parameters characterize the reactivity of the plasma. They are: the deuteron beam injection energy, W_0; the background plasma temperature, T_e; the ratio of the densities of the superthermal or "hot" ions to electrons, n_h/n_e; and the ratio of D to T in the background plasma. The reactivity is also proportional to n_e^2. However, the electron density is constrained by considerations of both beam penetration and the maximum plasma pressure that the tokamak discharge can sustain. Some calculated results of this plasma model are presented in Figures 2-5 (D. L. Jassby and H. H. Towner, Fusion Reactivities and Neutron Source Characteristics of Beam-Driven Toroidal Reactors with Both D and T Injection, Princeton Plasma Physics Laboratory Report MATT-1180, 1976).

Figure 2 presents Q, the ratio of total fusion power produced to

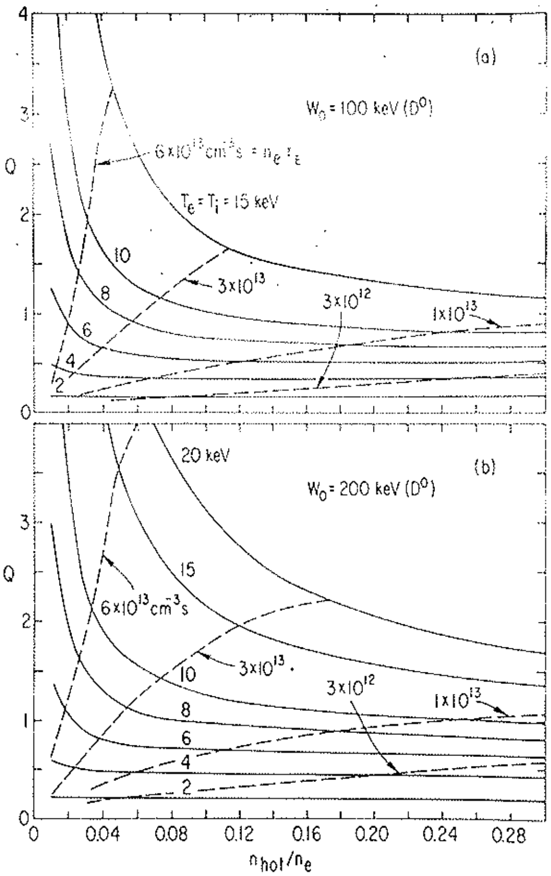

Figure 2.

total beam power deposited in the plasma versus n_h/n_e. Q is insensitive to the absolute value of n_e. Here a 50:50 mixture of D and T in the background plasma has been assumed as well as complete retention of the fusion alpha particles during their slowing down. For plasmas rich in superthermal ions the value of Q tends to be limited to that achievable by the beam-plasma and beam-beam interactions alone. The values of

$n_e\tau_E$, shown in dotted curves, are relatively low in order to accommodate the high beam power. The reverse is true for plasmas that are lean in superthermal ions. The higher values of Q reflect the larger fraction of the plasma reactivity being produced by the thermal background plasma rather than by the beams. Also, with the reduction of plasma heating by the beams it is necessary to reduce the energy loss rate from the plasma, and hence relatively high values of $n_e\tau_E$ are necessary. The value of $n_e\tau_E$ is thus to be viewed as a requirement on the plasma behavior that is determined by the choice of other parameters in the plasma model.

Total beam power and wall loading of an entire reactor can be approximated with the plasma model provided additional assumptions are made regarding the plasma geometry and the strength of the toroidal and poloidal magnetic fields. The results shown in Figures 3 and 4 apply to circular plasmas with: the aspect ratio equal to four; the "safety factor", q, equal to 2.5 at the edge of the plasma; and the plasma pressure being limited by either "beta-poloidal" (the ratio of plasma pressure to the pressure of the poloidal magnetic field) equalling two-thirds the plasma aspect ratio <u>or</u> the e-folding length of beam penetration equalling half the plasma minor radius.

Figure 3 relates Q to the total fusion power developed in different sizes of plasma. For a choice of W_o and n_h/n_e there is a unique relationship among Q, T_e, and $n_e\tau_E$ as indicated in Figure 2. For reactors with plasma minor radii greater than 1 meter it is apparent that a blanket multiplication of 10 can lead to thermal power levels of a few gigawatts and for values of $n_e\tau_E$ in the vicinity of 10^{13} S cm^{-3}. Furthermore there is a maximum fusion power for a fixed plasma size. Conceptually one arrives at this maximum by somehow adjusting the value of $n_e\tau_E$.

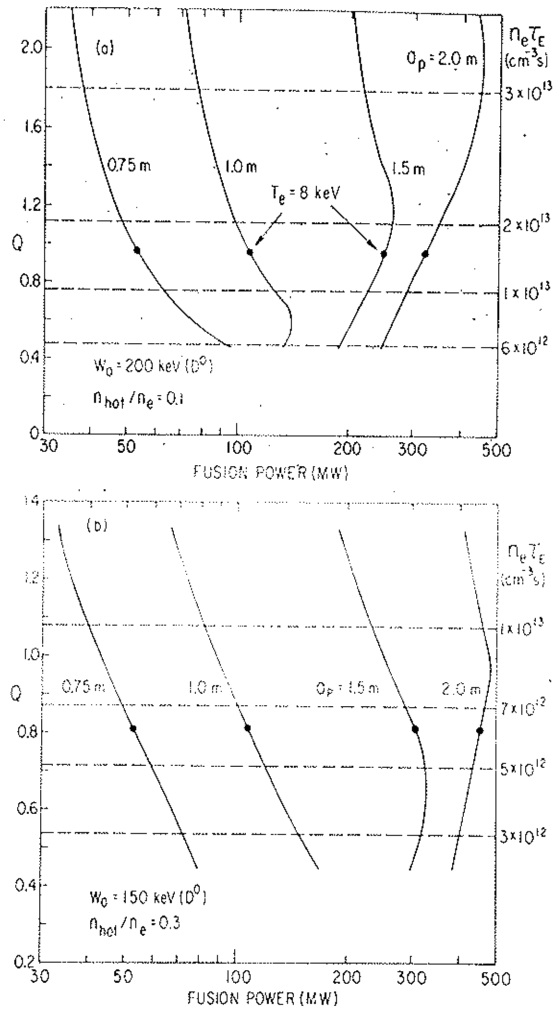

Figure 3.

Figure 4 displays the neutron wall loading versus W_o for various plasma sizes. (The wall is taken to lie 20 cm beyond the plasma.) The maximum in wall loading is created by the limitations placed on the plasma density. For low values of injection energy, the plasma density is being limited by the requirement that the beams penetrate sufficiently deep into the plasma. At high values of injection energy the density is limited by the limit placed on beta-poloidal. The main effect of increasing n_h/n_e seems to be to shift the maxima in wall loading to lower values of injection energy. This shift is caused by the beta-poloidal limit of the plasma being reached at lower injection energies when the relative

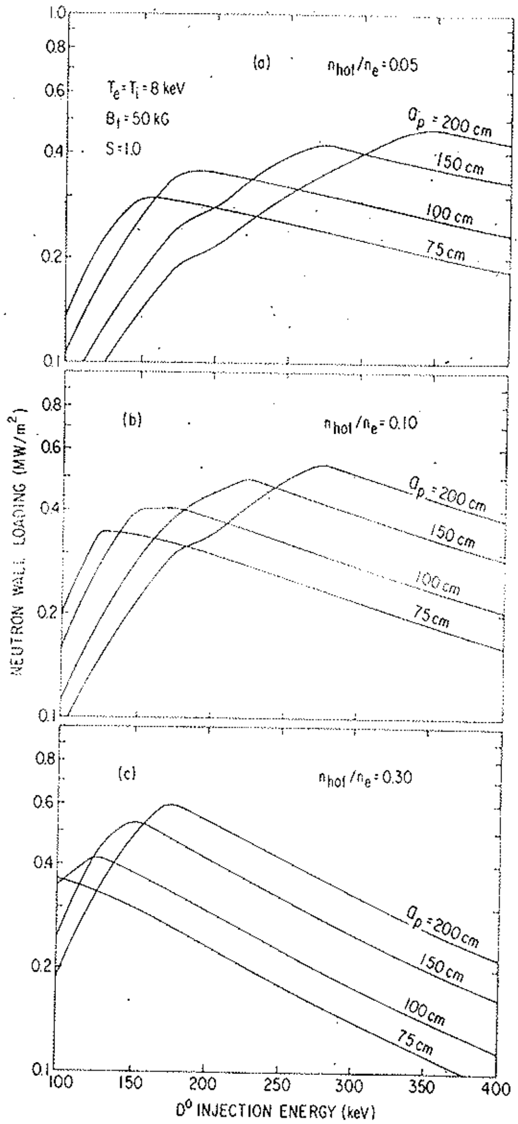

Figure 4.

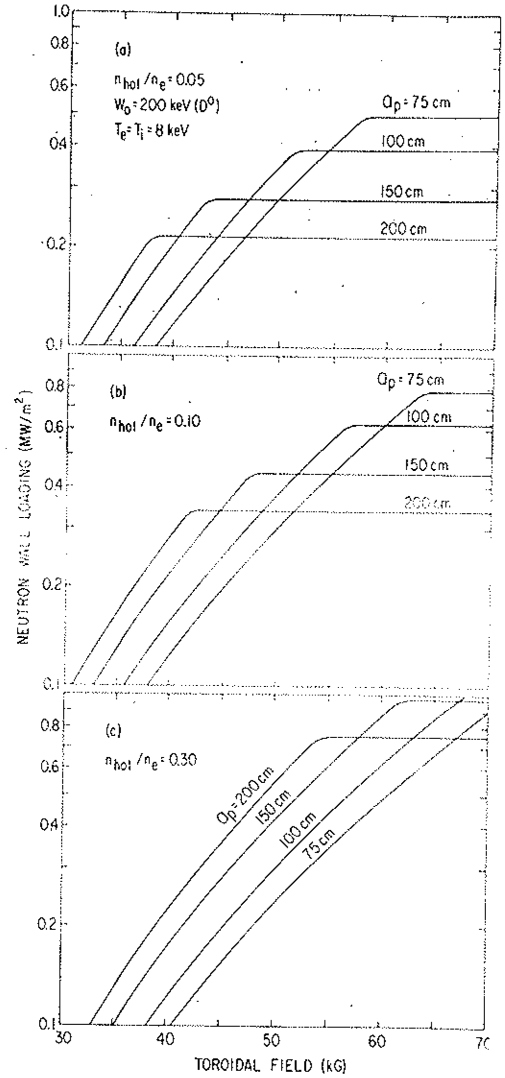

Figure 5.

population of energetic ions is increased. Generally the wall loadings are less than 1 MW/m^2.

Figure 5 displays another consequence of the limitations on plasma density. Here the wall loading is given versus the toroidal magnetic field. The "knee" in the curves occurs when the plasma pressure is limited simultaneously by the beta-poloidal and beam penetration constraints. For magnetic fields below the knee, the plasma pressure is being limited by the value of beta-poloidal. Hence increasing the toroidal magnetic field allows the discharge current to be increased - in order to preserve the assumed value of $q = 2.5$ at the plasma surface - and hence the poloidal field strength and hence the plasma pressure. In this part of the curve the wall loading reflects the plasma pressure and displays a B^4 dependence. However, for magnetic fields above the knee the pressure is limited by the requirement of beam penetration and therefore cannot increase with higher magnetic fields. Thus smaller plasma sizes can capitalize on higher magnetic fields and hence higher densities before becoming beam penetration limited. Another feature of the curves in Fig. 5. is the increase in maximum wall loading at the knee with an increase in the relative population of hot ions. This effect is a consequence of the interplay between the beta-poloidal limitations and the requirement for beam penetration. To see this effect consider that for a beta-poloidal limited plasma, an increase in hot ion population requires a proportionally larger decrease in the cooler background ion population and hence in the electron density. Now the reactivity is dominated by the sum of the beam-plasma and beam-beam interactions and is rather weakly affected by an increase in (n_h/n_e). However the beam penetration depth varies almost inversely with the electron or background ion density and will therefore increase with (n_h/n_e). Thus an increase in n_h/n_e will allow an increase in pressure without violating the beam penetration limit. To reach the beam penetration limit the magnetic field must be increased to allow the increase in pressure, which in turn will increase the wall loading at the knee.

PLASMA SCALING WITH DIVERTOR

Incorporating a poloidal field divertor into the hybrid design requires a significant modification of several of the above assumptions. In the first place the plasma will not be circular. Second, the specification of q = 2.5 at the plasma edge loses its relevance because the plasma will be effectively bounded by a separatrix surface on which q is infinity. Third, the effective wall area will be reduced by the area needed for the divertor channels. In addition there are the questions of stability of the divertor configuration, pressure limitation, and scaling of a separatrix-bounded plasma.

To elucidate these questions we have made numerical explorations of several toroidal equilibria for separatrix-bounded plasmas of both the single-null and double-null variety. The equilibria were characterized by parabolic distributions for both the plasma pressure and poloidal current as functions of the poloidal magnetic flux. In addition the safety factor, q, was required to be only slightly larger than unity on the magnetic axis, and the plasma pressure was kept at a high level such that beta poloidal was in the vicinity of the plasma aspect ratio. Since the plasmas are non-circular, the precise definition of the aspect ratio is not clear. However, a 10% increase in pressure above the values chosen for a given equilibrium will produce a negative toroidal current density in the inside (small major radius) region of the plasma. Intuitively such a toroidal current distribution would seem magnetohydrodynamically unstable - the positive and negative current channels tending to rotate about each other in a poloidal sense. Guided by an experienced intuition a chosen shape of poloidal flux distribution on the midplane was roughly preserved through a variety of changes in toroidal field coil location and vertical field strength at the magnetic axis. For this variety of equilibria we have been able to make a rough analytic correlation between the locations of the magnetic axis, R; the toroidal field coil, h; and the length designated as the minor radius of the plasma, a_p. In addition we have a similarly rough correlation for both the total discharge current and the peak plasma pressure in terms of h and a_p. These correlations are similar, but distinct, for the single-null and double-null configurations and allow us to explore a variety of plasma sizes for their economic consequences without going through a detailed magnetic field coil design for each case. One example of a single-null configuration is shown in Figure 6. This figure shows the minimum size single-null plasma we could obtain and still have: the discharge current at least one megamp, an adequate core size to ensure a 1000-second plasma burn time for $Z_{eff} = 1$, a maximum toroidal field of 160 kilogauss, and 50 cm between the separatrix and the first wall of the blanket. (To obtain the field pattern over such a large area a

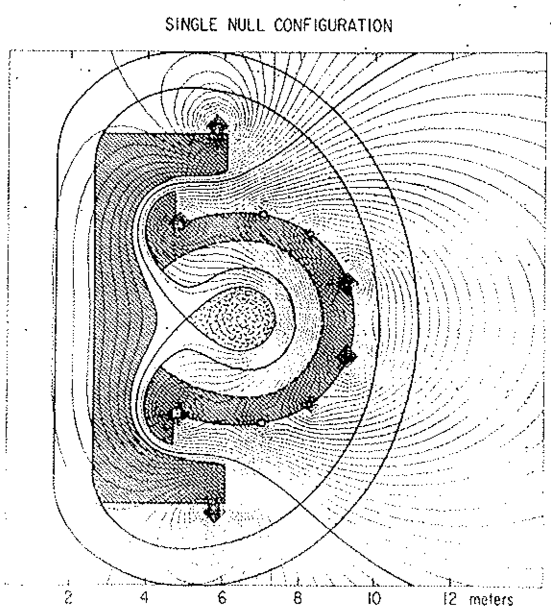

Figure 6.

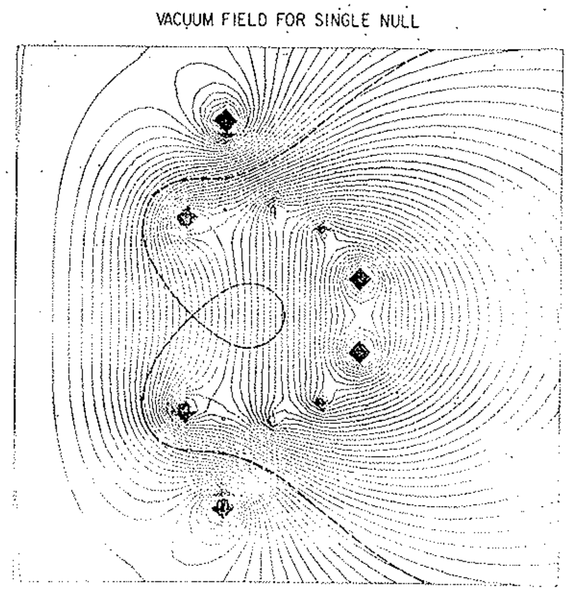

Figure 7.

line current at the magnetic axis was substituted for the equilibrium current distribution. This substitution appears to produce very little change in the field pattern outside the separatrix surface.) The shielding is cross hatched and adequate for superconducting coils. The cross-sectional area allowed for the coils has been calculated on the basis of a current density of 1275 amps/cm^2, which is supposed to allow sufficient room for dynamic stabilization of the superconductor, support structure, and liquid helium Dewars. The economics of this size do not seem attractive.

Figure 7 shows the vacuum poloidal field for the configuration of Figure 6 together with the separatrix surface. There are several points of interest here. First is the "good curvature" of the magnetic field in the region of the plasma. This curvature is expected to produce equilibria stable against rigid body displacements. Second is the rather abrupt increase in field strength in the neighborhood of the null point. This design feature enables one to expand (or contract) that part of the separatrix surface enclosing the discharge while keeping the null point and the remainder of the separatrix surface nearly fixed. The separatrix surface can therefore function as an expanding limiter during start-up. Third is the coupling of the entire poloidal flux that links the discharge in a sense to help induce the discharge current. The demand on the ohmic heating core is thereby reduced.

Figure 8 shows the equilibrium flux surfaces for the single null of Figure 6. The relatively short

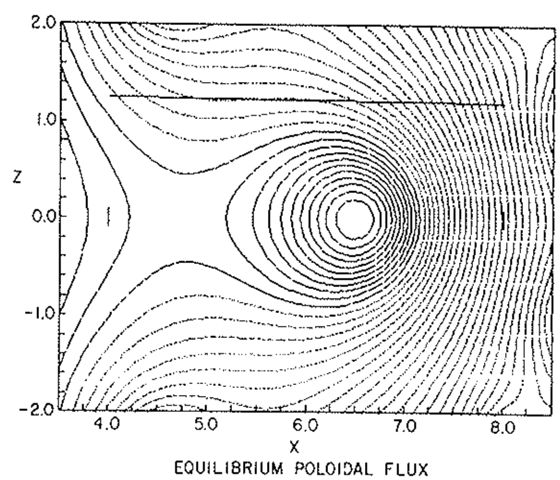

Figure 8.

distance between the magnetic axis and the separatrix surface at the midplane is a consequence of the high value of beta-poloidal and facilitates neutral beam penetration.

An example of a double-null configuration is shown in Figure 9. The toroidal coil has

DOUBLE NULL CONFIGURATION

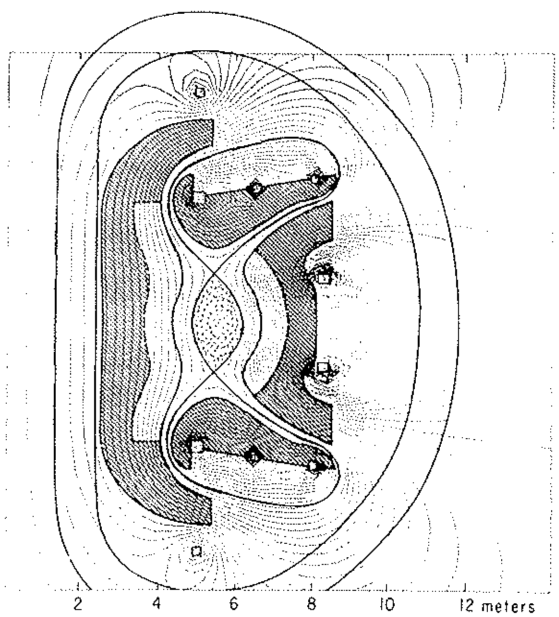

Figure 9.

been placed at the same major radius as in Figure 6. This configuration meets the same specifications as the single-null configuration discussed above with the exception that the core size is too small to provide the necessary volt-seconds. The separatrix passes so close to two of the coils that a narrow divertor channel and at best marginal shielding are produced. These two features seem to be characteristic of double-null configurations. Comparing the two styles of design for a given h and a_p, the toroidal field coil is taller and wider and the blanket subtends a smaller solid angle about the plasma in the double null design than in the single null design. These features argue against the relative performance of the double null configuration. However, the double-null plasma can better utilize the toroidal field because it can be located closer to the toroidal field coil than can the single-null plasma. This effect coupled with the ability for the vertically elongated plasma to hold a higher pressure than the horizontally elongated plasma argues for a better performance of the double-null than the single-null design.

Figure 10 shows the vacuum poloidal field for the configuration

VACUUM FIELD FOR DOUBLE NULL

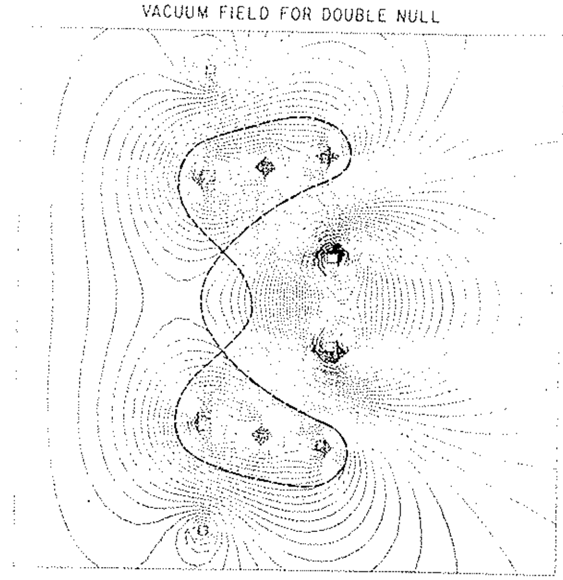

Figure 10.

of Figure 9 together with the separatrix surface. The most striking feature is the "bad curvature" of the field in the region of the plasma. This curvature is expected to produce equilibria unstable against rigid body displacements and is of major concern. Plasmas at larger radii have vacuum fields with both good and bad curvature, but the persistence of bad curvature over a large portion of the plasma is characteristic of double-null designs. We also note the larger fraction of the poloidal flux that links the discharge in a sense that opposes the induction of the discharge current. This feature increases the demand on the ohmic heating core and is the reason why the available core space in the design of Figure 9 is inadequate.

Figure 11 shows the equilibrium flux surfaces for the

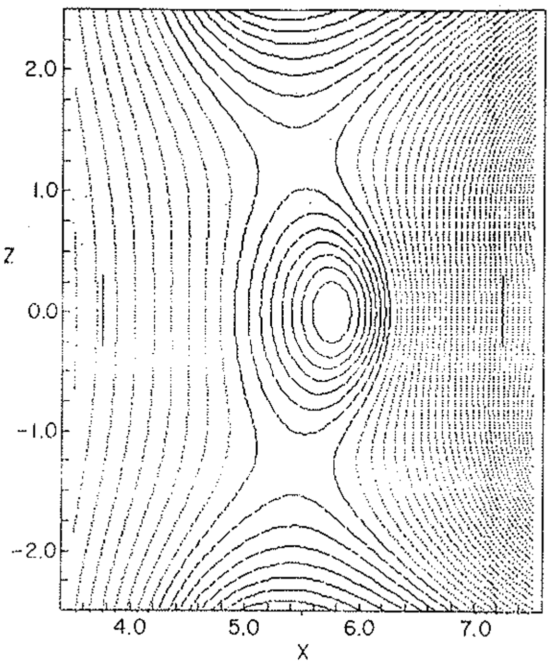

Figure 11.

double-null design of Figure 9. The high values of beta-poloidal produce the same kind of crowding of the magnetic surfaces to larger major radii as is seen in the single-null design.

ECONOMIC ANALYSIS

The economic analysis of the hybrid consists of equating the annual revenue to the sum of the annual operating and maintenance costs plus the product of the annual fixed charge rate and the capital cost of the plant. The revenue is simply the annual production of some product times the price of the product. The plant costs thus determine the unit price, or, in other words, the unit cost of the plant product. The difficulty in analysis is determining what quantities are to be counted as plant costs, how these depend on design parameters, and what are the associated costs. The production capacity of the plant is also a function of the design parameters and subject to uncertainty. In spite of these uncertainties one hopes that design guidance can be obtained through the choice of design parameters that will produce the lowest unit cost of the product.

For the hybrid reactor there is some uncertainty as to what constitutes the product. Is the product electricity or fissile fuel or both? Should one view the hybrid in isolation from the fission reactors that would burn the hybrid's net fissile fuel production? Should the hybrid be judged as one part of an integrated system consisting of a hybrid plus dependent fission reactors where the only system product is electricity? How are the fissile fuel reprocessing and fabricating plants to be accounted for? In view of the fact that there are no plutonium reprocessing plants in commercial operation, it is particularly awkward to assess their costs. Furthermore, it is readily apparent that the capacity of a single fuel reprocessing plant can service more than one hybrid and associated fission reactors.

Our most recent mode of economic analysis is to treat the hybrid and associated fission reactors as a single system or nuclear park. Processing of the molten salt in the hybrid blanket is to be inside the park, but the fabrication and reprocessing of the uranium in the hybrid and the fission reactors is to be done outside of the park. Thus the fuel cycle costs take the form of costs per kilogram of fuel charged against the operation of the park. The only product is electric power. The fission reactors are taken as 1000 MW(e) light water reactors. Their capital and operating costs are taken from present experience. For the hybrid reactor the capital costs are a function of the geometry, the blanket wall area, the wall loading, injected neutral beam power, gross thermal power, and magnetic field strength. The wall loading is connected to the neutral beam power by the plasma parameters. Maintenance costs reflect first wall or blanket replacement costs and schedule. We have taken 5 MW-year/m^2 of 14 MeV neutrons as the first wall capability before requiring replacement and, if current thinking is correct, to achieve such fluence will require limiting first wall temperatures to less than 600°C if a stainless steel is used for structure.

We have only recently begun to exercise this economic model. We are engaged in wandering around in parameter space under certain constraints, namely: plasma equilibrium with the $q > 1$ on the magnetic axis, adequate neutral beam penetration, $n_e\tau_E < 5 \times 10^{13}$ s $\cdot$ cm^{-3}, adequate core size to ensure a plasma burn time of 1000 seconds, total neutral beam power not to exceed 600 MW, and the gross electric power of the hybrid not to exceed 4000 MW(e). We are looking at both single-null and double-null configurations for the plasma. From an earlier analysis we expect that the price of electricity will be most sensitive to the thermal conversion efficiency, the effective wall area, and the blanket energy multiplication. An example of our preliminary findings with single null scaling is shown in Figure 12 and the following table.

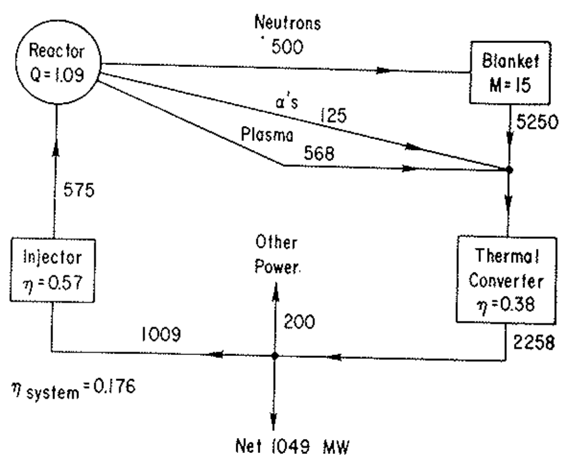

Figure 12.

Major radius, R	8.15 m
Minor radius, a_p	1.6 m
D injection, W_o	180 kV
Plasma temp., T_e	10 KeV
Electron density, n_e	8×10^{13} cm^{-3}
n_h/n_e	0.1
Q	1.09
Beam Power	575 MW
Wall load	0.74 MW/m^2
Blanket Multiplication	15
Thermal Power	5943 MW
Gross electric	2258 MW
Net electric	1049 MW
Net fissile fuel	1186 kg/yr.
No. of LWR's in Park	1.9
Injector source eff.	0.57
Plant eff.	0.176
Hybrid capital cost	≈ \$2 x 10^9
Park cost of electricity	≈ \$0.027/kW-hr.

Acknowledgement

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COUNTERSTREAMING-ION-TOKAMAK FISSILE BREEDER

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ABSTRACT

Tokamak plasmas fueled and heated by energetic neutral-atom beams are characterized by total ion energy greatly exceeding the electron energy. For smaller devices the largest fusion reactivity of energetic-ion plasmas is obtained when oppositely injected D° and T° beams sustain counterstreaming velocity distributions of deuterons and tritons. This scoping study investigates the net fissile and power productions of a tokamak fusion-fission reactor with a counterstreaming-ion fusion driver and a fertile blanket optimized for fissile breeding. The fusion driver has parameters $R_o = 4.7$ m, $a = 1.0$ m, $B_t = 5.6$ T, $W_b = 100$ keV (D°), $n\tau_E = 1.4\times10^{13}$ cm^{-3}s, $Q = 1.5$, 14-MeV neutron production = 175 MW. The blanket contains a fast-fission zone of natural U plus Mo (7%), followed by a Li-bearing zone for T breeding. The reactor produces a net power of 480 MWe and supplies sufficient Pu to support a system of LWR's producing 3800 MWe, with an estimated electrical energy cost for the entire system of 27 mills/kWh.

INTRODUCTION

The potential performance of a tokamak fusion plasma as a copious source of 14-MeV neutrons raises the possibility of employing such a device to produce fissile fuel for distribution to thermal-fission reactors,[1,2] in competition with the usual mining and isotope-separation processes, or with proposed accelerator-based neutron sources. One attractive strategy is to incorporate the tokamak fissile breeder in a so-called nuclear energy center, where the breeder supplies the entire demand for make-up fissile fuel of a group of LWR's, for example. In this role, on-line electrical power production by the breeder itself is a secondary feature of its performance, so that the breeder could be integrated with relative ease into the present fission power economy.

Economic considerations for a tokamak breeder tend to favor the maximization of fusion power density P_f,[2] since the cost of 14-MeV neutrons from the reactor is roughly inversely proportional to P_f. Nevertheless, it is also important that fusion power multiplication Q be greater than about 0.5, since the required investment in neutral-beam injectors, and the quantities of electrical power and tritium that must be circulated, are nearly inversely proportional to Q. A large number of experiments have indicated that the tokamak plasma heated by injected neutral beams is an attractive potential means of achieving high Q-values. It seems probable, however, that a high-Q device would have extremely large unit size and cost, and many unresolved difficulties associated with quasi-stationary operation, such as particle exhaust, fueling, and current maintenance.

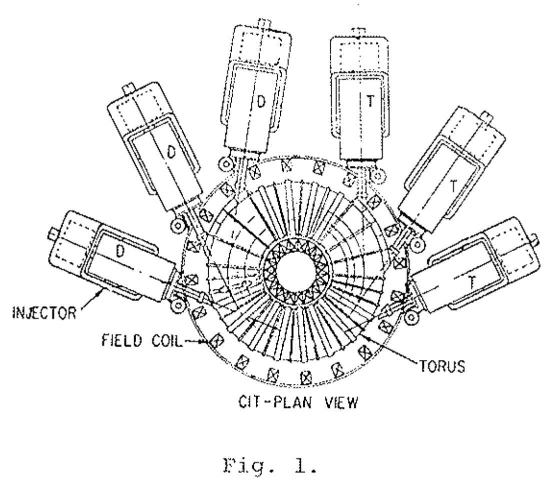

Fig. 1.

Recent neutral-beam injection experiments in tokamaks have demonstrated that it is relatively straightforward to raise the total plasma ion energy well above the electron energy.[3] These results provide support for the concept of energetic-ion tokamak plasmas in which the ion population is supplied wholly by injected neutral beams, and the quantity β_i = (ion energy density)/(electron energy density) >> 1. In small devices with relatively low injection energy, the fusion reactivity and Q of plasmas with $\beta_i >> 1$ are maximized when oppositely directed D and T neutral beams (see Fig. 1) are used to stack large densities of counterstreaming energetic deuterons and tritons.[4,5] An important advantage of this CIT (counterstreaming-ion-tokamak) mode over the TCT (two-component-torus) and thermonuclear modes is that in principle the CIT provides the means for steady-state operation[6]: The beams provide all plasma fueling, and the beam-induced current can be made to provide the tokamak equilibrium. (The problem of high-throughput particle exhaust must still be contended with, however.)

The principal disadvantage of CIT operation is that the product $n_e a_p$ (plasma density × plasma radius) must be restricted to allow adequate penetration by neutral beams in the energy range for which CIT operation is optimal (W_b = 50-100 keV). The fusion power is then limited to a maximum of about 200 MW, and the uncollided 14-MeV neutron wall loading can be at most 0.5 MW/m^2. However, this relatively modest value may be appropriate for near-term applications, in order to take advantage of state-of-the-art materials technology and wall-cooling techniques. The Q-values that are likely to be attained in practice in the CIT mode are in the range 1 to 3, which are adequate for a device whose main purpose is fissile breeding.[1,2]

The present report describes a scoping study of a CIT reactor of maximum size with a blanket containing either U or Th, and designed especially to maximize the breeding rate of ^{239}Pu or ^{233}U, respectively. The size of the tokamak (R_o = 4.7 m) is rather smaller than those contemplated for "pure-fusion" power plants, and the specified 100-keV neutral beams are well within the reach of present technology. The plutonium breeder can supply make-up fissile fuel to support a system of LWR's producing 3800 MWe, and at the same time generate a net power of 480 MWe, with an estimated delivery price for the entire system of 27 mills/kWh.

PLASMA PHYSICS BASIS

IDEAL CIT PERFORMANCE

In steady state the ion population of a CIT plasma makes up two nearly thermal distributions, oppositely displaced in velocity along the magnetic axis. These distributions are maintained by introducing practically all plasma ions by beam injection parallel and antiparallel to the magnetic axis as in Fig. 1, and by minimizing the influx of cold neutrals and plasma into the hot reacting region. The electron temperature T_e is maintained by Coulomb power deposition from the energetic ions. Most of the fusion energy is produced by nearly head-on collisions between oppositely streaming deuterons and tritons. All fueling is carried out by the injected ions, and maintenance of the plasma current by these ions is also possible under certain conditions.[6]

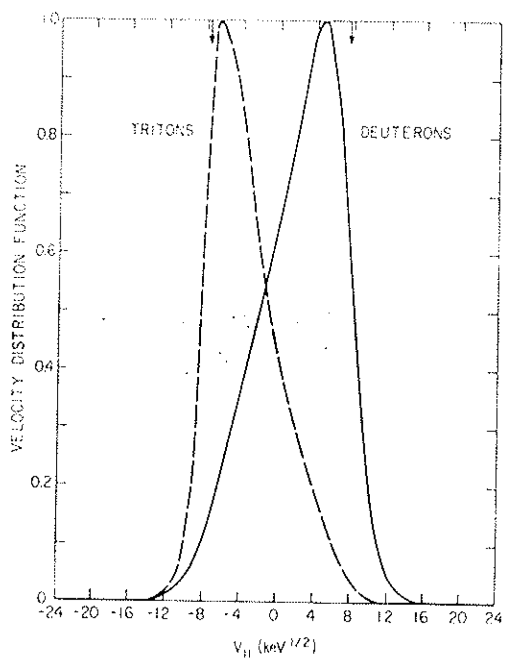

Fig. 2. Sample distribution functions for ion velocity components parallel to the toroidal magnetic field in a CIT plasma when influx of warm ions is significant. Injection energy = 60 keV (D°), 90 keV (T°), indicated by arrows. T_e = 10 keV.

At the beginning of a cycle the injected fast atoms are trapped by an Ohmic-heated low-density ($n_e \sim 10^{13}$ cm^{-3}), low temperature ($T_e \sim 1$ keV) tokamak plasma. During injection for several slowing-down times (a period typically less than 0.5 s),

n_e is built up to the range 3 to 7×10^{13} cm^{-3}, T_e reaches a steady-state value of at least several keV, and the ion population becomes dominated by energetic ions. In the steady state, injected energetic neutrals are trapped by charge exchange and impact ionization with the ions in the energetic distributions. The velocity distributions $f_h(v)$ of Fig. 2 were calculated numerically from the Fokker-Planck equation, assuming that energetic ions are perfectly confined until they decelerate to an energy $W_1 = 3/2\ T_e$, that ions with energy $W \lesssim 3/2\ T_e$ are lost at a rate $\tau_p^{-1} \sim \tau_{Ee}^{-1}$, and that recycling is minimized.

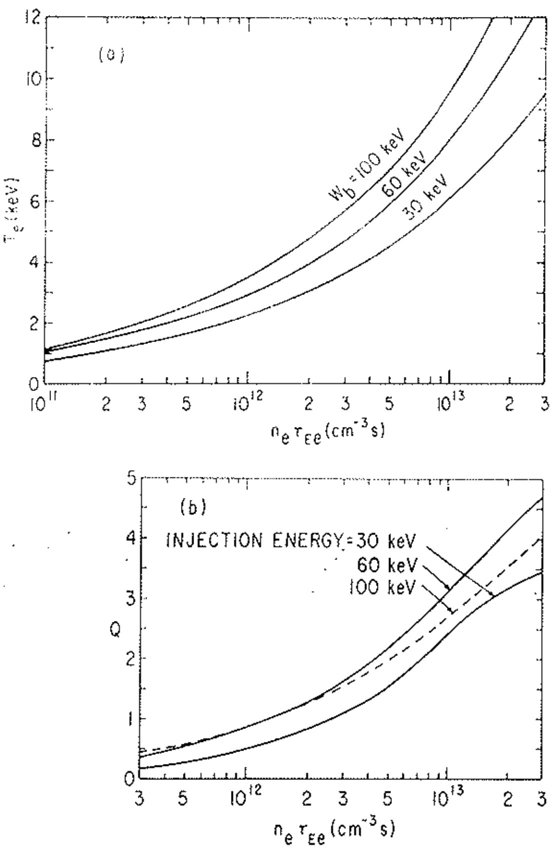

Fig. 3. Dependence of (a) T_e and (b) ideal Q on electron energy confinement parameter for steady-state CIT operation. D° and T° injected at same energy. 17.6 MeV per reaction. Neither charge-exchange loss nor fusion-alpha heating is included.

Figure 3(a) shows the variation of T_e with $n_e\tau_{Ee}$, when fusion alpha particles are not confined. Fusion power multiplication Q is calculated by integrating fusion reactivity over the ion velocity distributions, and dividing by the injected power. Figure 3(b) shows that for injection energy of 60 keV, Q = 1 ("break-even") is attained, ideally, at $n_e\tau_{Ee} \simeq 2\times10^{12}$ $cm^{-3}s$, corresponding to $T_e \simeq 3.5$ keV. If fusion alpha particles are retained in the plasma,[7] the ideal Q reaches 5 when $n_e\tau_{Ee} \sim 3\times10^{13}$ $cm^{-3}s$.

PRACTICAL LIMITATIONS TO PERFORMANCE

The degree to which ideal CIT performance can be approached depends on charge-exchange loss and the size of the warm-ion population. The proportion of warm ions depends first on the residence time τ_p of decelerated ions in the reacting region, and second on the influx of neutrals and cold plasma to this region. For small machines the most severe limitation is expected to be set by cold-particle influx from the scrape-off region surrounding the discharge. In principle recycling can be practically eliminated by means of an "unload" magnetic divertor that exhausts rapidly from the torus those ions diffusing out of the reacting plasma.[8]

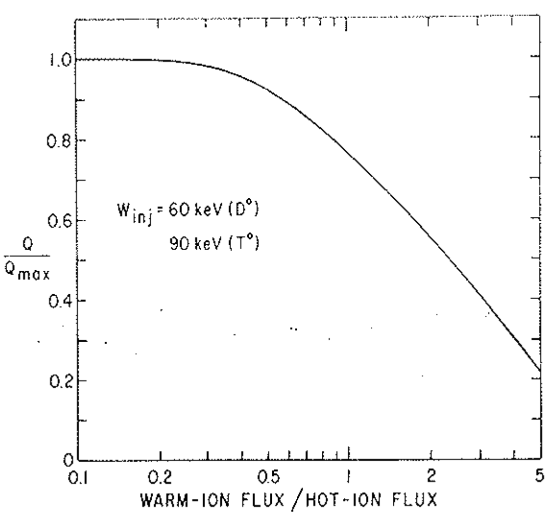

Fig. 4. Reduction of Q by inward flux of warm ions. The hot-ion flux is supplied by neutral beams. Reduction is less severe at W_{inj} = 100 keV (D°).

In order to determine the proximity of a practical operating point to the ideal CIT regime, initial calculations of reactor performance have been made with a coupled Fokker-Planck radial transport code.[7,9] Figure 4 shows how Q decreases with relative cold particle influx into the reacting region. With increasing cold-particle influx, beam-target reactions eventually become the dominant source of neutrons. In the limiting case of interest, the plasma is operated in the TCT regime, driven and fueled by injected D and T beams. At very large $n_e\tau_{Ee}$, which generally implies large

T_e, ions in the counterstreaming distributions tend to thermalize with each other before slowing down. The CIT then evolves into a beam-driven thermonuclear reactor, with ion temperature $T_i > T_e$. In many practical cases, the ion velocity distribution resembles that of Fig. 2.

OPTIMAL INJECTION ENERGY

Maximum Q-values for ideal CIT plasmas are attained with $W_b \sim$ 40-60 keV.[4-6] Other reasons for preferring $W_b \sim$ 50 keV are that (1) neutral-beam injectors are extremely efficient in this range; (2) for a given input power the rate of particle injection by the beams can overwhelm the cold-particle influx; (3) the beam momentum is large, thus facilitating beam maintenance of the plasma current required for tokamak equilibrium.

Nevertheless, there are important reasons for employing a larger W_b: (1) At large W_b one can take better advantage of target-plasma reactions, so that Q-values are not as severely degraded in the event that the warm-ion population is substantial (Fig. 4). (2) The angular scattering rate of fast ions by impurity ions is reduced at increased W_b. (3) The charge-exchange loss that accompanies beam trapping decreases with increasing W_b. (4) Larger W_b permits penetration into plasmas of larger $n_e a_p$. For these practical reasons the optimal energy in CIT operation is probably 80 to 100 keV for D° (120 to 150 keV for T°), where the efficiency of neutral-beam injectors is still high.

The small plasma size permitted for adequate neutral-beam penetration with tangential injection ($\langle n_e \rangle a_p \lesssim 5\times10^{15}$ cm^{-2}) severely restricts the attainable fusion power production $\propto (n_e a_p)^2 \langle\sigma v\rangle$, and corresponding 14-MeV neutron wall loading, ϕ_w. For practical CIT parameters,[6] P_f = 0.5-1.5 W/cm^3 and $\phi_w \approx$ 0.1-0.4 MW/m^2.

A less anisotropic energetic-ion distribution can be established by means of very oblique injection. For $W_b \sim$ 100 keV (D°), $\langle\sigma v\rangle$ would be degraded by relatively little from its value in counterstreaming operation, and the factor of 1.5 or more increase in n_e now permitted for acceptable penetration results in a larger ϕ_w. This approach is used in the present study, where the reacting plasma has a fusion power output of 220 MW — probably the largest practical value for CIT-type operation.

DEVICE PARAMETERS

PLASMA CHARACTERISTICS

The neutral beam energies are 100 keV for D° and 150 keV for T°. (These beams can be produced with equal efficiencies and have equal penetration lengths in the plasma.) With present-day Berkeley-type ion sources, the overall injector efficiency η_I can be as large as 0.74 when direct recovery of unneutralized ions is employed.[10] In this study we use η_I = 0.70 to take into account some transmission loss in the beam ducts.

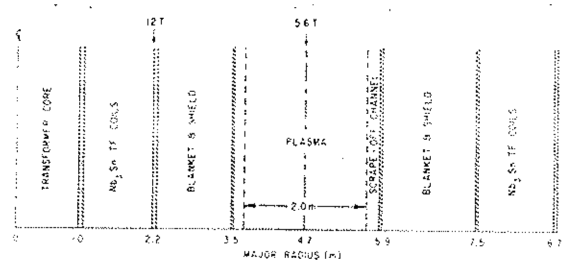

Fig. 5. Schematic layout of the CIT fissile breeder.

Table 1. Plasma parameters of CIT fusion driver.

Geometry	
Transformer core rad.	1.0 m
TF coils	Nb_3Sn
Thickness of TF coils	1.2 m (radial)
TF-coil bore	5.3 m × 8.3 m
B_{max} at TF winding	12 T
Inner shield/blanket	1.3 m
Scrape-off channel	0.2 m
Plasma half-width	1.0 m
Plasma half-height	1.6 m
Circumference/$2\pi a_p$	1.5
Major radius, R_o	4.7 m
Plasma characteristics	
B_t on magnetic axis	5.6 T
Plasma current, I_p	4.8 MA
q at limiter	2.8
$\langle n_e \rangle$	7.0×10^{13} cm^{-3}
$\langle T_e \rangle$	10 keV
$\langle n_e \rangle \tau_{Ee}$	1.4×10^{13} cm^{-3}s
$\beta_p \approx 0.85\ R_o/a_p$	4.0
Neutral-beam energy and power	100 keV, 70 MW (D°) 150 keV, 75 MW (T°)
η_i	5
Power production	
Q	1.50
P_f	1.35 MW/m^3
14-MeV neutron power	175 MW
14-MeV wall loading	0.40 MW/m^2 (avg.) 0.64 MW/m^2 (peak)
Pulse length	$\gtrsim$ 100 s

The device geometry is shown schematically in Fig. 5, and the plasma parameters are given in Table 1. The dimensions have been chosen to give the maximum possible fusion power production compatible with adequate neutral-beam penetration. Under the constraints of maximum plasma pressure and adequate beam penetration, higher neutron wall loading is possible by employing a vertically elongated (noncircular) plasma cross section. Theoretical analysis and experimental results[11] indicate that a vertical elongation of $b/a \lesssim 1.6$ at $q \gtrsim 3$ is feasible. Consequently, our design specifies a D-shaped cross section with $b/a = 1.60$. The plasma half-width, $a_p = 1.0$ m, has been chosen so that beam penetration is just acceptable when the total plasma pressure (including fusion alphas) corresponds to $\beta_k = 0.85\, R_o/a_p$. Each neutral-beam axis crosses the magnetic axis at 45°.

The plasma current $I_p = 4.8$ MA is sufficient to confine all the fusion alphas, which give up the bulk of their energy to plasma electrons, thereby giving a larger T_e (and Q) with the same $n_e\tau_E$. For the conditions of Table 1, the ideal Q-value is 3.9, when charge-exchange loss is neglected.[7] The effects of the slow-neutral influx from the scrape-off region are to increase the proportion of warm ions in the reacting plasma, and to enhance charge-exchange loss of energetic ions. If the warm-ion flux is no more than twice the hot-ion flux injected by the beams, then taking into account all charge-exchange losses, Q is reduced by 60%,[7] so that we have finally Q = 1.5.

(The maximum field at the Nb_3Sn windings is nominally $B_{max} = 12$ T. If the plasma beta (presently $\beta = 5.2\%$) could be increased by a factor of two, as an extreme example, by means of certain techniques now under theoretical investigation,[12] a decrease in B_{max} to 8.5 T would be possible. With higher β it might be feasible to employ normal TF coils (copper or aluminum), since the extra power consumption could well be acceptable in a device whose main purpose is fissile breeding — especially if the substitution of normal coils for superconducting coils results in a substantial decrease in capital cost. However, for steady-state operation a further reduction in B_{max} and corresponding increase in a_p would be required.)

PARTICLE EXHAUST

The vacuum pumping requirement is given by

$$n_p \bar{v}_p \eta_p A_p = F_b + F_g \tag{1}$$

where $F_b = 7.5\times10^{21}\ s^{-1}$ (1200 A-equiv.) is the rate of particle injection by the neutral beams, $F_g \simeq 2\times10^{21}\ s^{-1}$ is the estimated flux of slow gas molecules from the beam ducts, n_p is the particle density outside the discharge, $\bar{v}_p$ is the drift velocity of these particles, A_p is the pumping area, and η_p is the pumping efficiency. If no divertor is used, n_p is a neutral-gas density which must be limited to about $10^{11}\ cm^{-3}$, and $\bar{v}_p$ corresponds to a temperature of order 10 eV. Rapid particle exhaust could be accomplished in principle by use of a very elongated vacuum chamber, with the regions above and below the plasma filled with baffle-protected moving getter assemblies that would capture essentially all impinging neutrals. Then $A_p \sim 90\ m^2$ is about one-third the plasma surface area, and η_p must be at least 1/4. A serious difficulty with this approach is that in order to avoid significant neutron energy loss on the getters, the getter surfaces must be very thin and require cooling.

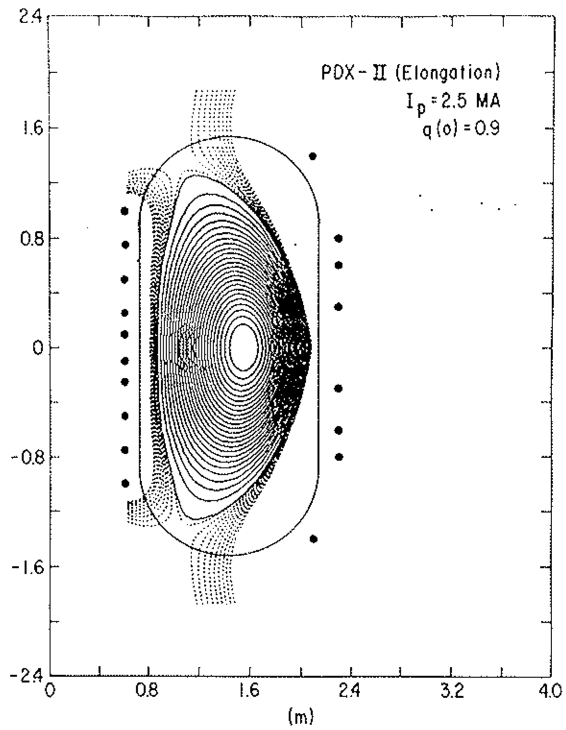

Fig. 6. Flux plot for D-shaped PDX-II plasma with elongation b/a = 1.9. Unclosed flux lines can be used for particle exhaust. Dots show position of low-ampere-turn coils required for gross stability. (Ref. 14)

If a poloidal magnetic divertor is used to exhaust ions diffusing out of the reacting plasma, then $\bar{v}_p$ is very large ($\sim 3\times10^7$ cm/s) and A_p can be relatively small. In employing the divertor technique, it seems most straightforward to use a "natural" divertor, that is, to take advantage of the separatrices that tend to form when the plasma pressure is made very large.[13] In that case the ampere-turn requirements of the divertor-field coils are relatively modest. Figure 6 shows an example of flux surfaces for "natural" divertor operation as calculated for the PDX-II tokamak at Princeton.[14] A suitable collection scheme for the diverted ions is a labyrinth of liquid-lithium covered screens.[15]

NEUTRAL BEAMS AND BURN CYCLE

The neutral beam injector system consists of eight beam lines (of which six are shown in Fig. 1). The injector characteristics are summarized in Table 2. The production of 100-keV D° and 150-keV T° beams with $\eta_I \simeq 0.70$ using essentially state-of-the-art positive-ion technology is discussed in Ref. 10. (Reference 16 discusses the characteristics of 60-keV D° and 90-keV T° injectors for CIT operation, as well as the details of tritium processing.)

Table 2. Neutral beam injection systems.

Beam energy	100 keV D°, 150 keV T°
Beam power	70 MW D°; 75 MW T°
Beam current	700 A-equiv. D°
	500 A-equiv. T°
No. of beam lines	8
Total beam apert.	6.0 m^2 at 0.02 A/cm^2
Fraction wall area	1.4%
Overall effic.	70%
Power consumption	207 MWe
Pulse length	dc

Since all plasma fueling is carried out by the injected beams, the length of the burn period is determined fundamentally by the volt-sec available in the transformer core to drive the toroidal current. While the streaming ions are in fact capable of carrying a large portion of the toroidal current, our design accommodates sufficient volt-sec so that the transformer can by itself support a burn time of at least 100 s, with Z_{eff} up to 2. At the start of each cycle, 3 s are required to form the target plasma, raise I_p to 4.8 MA, and attain the equilibrium plasma. At the end of each cycle, 20 s are required to ramp-down the plasma current, exhaust the torus, and recharge the OH transformer primary. Thus the duty factor is at least 0.81.

The OH (Ohmic-heating) and EF (equilibrium-field) coils are constructed of water-cooled copper. Some of the EF coils must be placed inside the TF coils, in order to reduce power requirements, and to afford better control over plasma shaping. In particular, a number of coils of relatively small ampere-turns must be placed 40 to 50 cm from the vacuum vessel, somewhat as illustrated in Fig. 6, in order to ensure that the elongated configuration has gross stability. The resistive power dissipation in the copper coils during stationary discharge conditions is estimated to be 30 MW. The total circulating power of the reactor is 260 MW, comprised of 207 MW for the injectors, 30 MW for the copper coils, and 23 MW for refrigeration and blanket coolant.

BLANKET

The blanket composition and neutronics analysis are precisely those used in recent mirror-machine hybrid studies.[17,18] As in those studies, the object is to maximize the production of fissile fuel for supply to thermal convertor reactors. Electricity production by the breeder is regarded as a byproduct, and the net thermal efficiency is of secondary concern.

The geometry used in the present blanket calculations is shown in Fig. 7. The blanket coverage is assumed to be 90%, with 10% of the fusion neutrons lost in the beam ducts and in the particle collection areas. The heat from the lost neutrons (18 MW) is not recovered.

The plutonium-breeding blanket has an inner fast-fission zone containing a homogeneous mixture of uranium plus 7% (by weight) of molybdenum. Tritium breeding is performed in the outer zone. (See Ref. 18 for details of the blanket composition.) While natural uranium (0.7% ^{235}U) is used, only about 20% of the fissions are of ^{235}U, so that the use of depleted uranium (0.25% ^{235}U) would give very similar results. The production of plutonium and the blanket power multiplication M increase monotonically with fusion-neutron fluence.[18] The irradiation period after which ^{239}Pu is removed from the blanket is determined by the optimal economics of fuel use in the LWR's (light-water-moderated reactors) and the breeder. In any event, k_{eff} is always substantially less than unity.[18] For this blanket the thermal-to-electric conversion efficiency can be taken as 0.35.

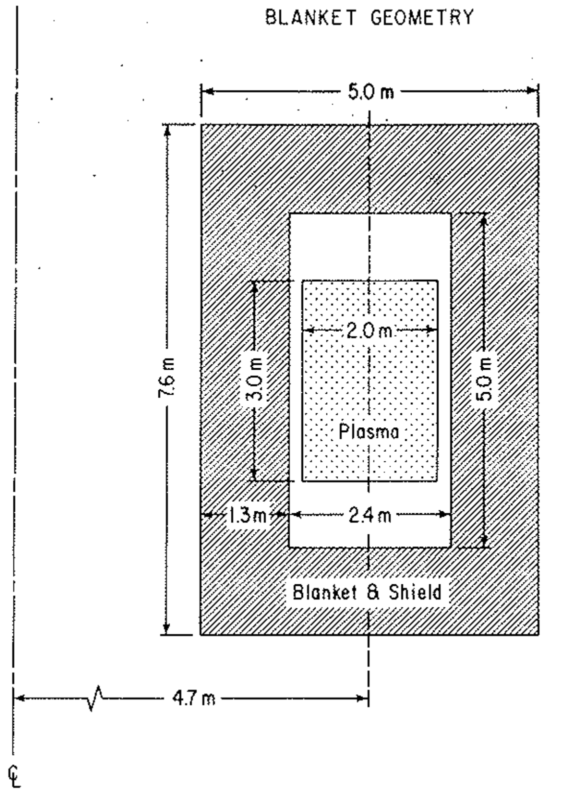

Fig. 7. Geometry for blanket neutronics. The regions above and below the plasma are used for particle collection. (The plasma is actually D-shaped with $b/a \approx 1.60$.)

ECONOMICS

Our economic analysis is identical to that in the mirror hybrid studies,[18] which calculates the cost of electrical energy produced by the breeder together with the system of LWR's that it supports. The ^{239}Pu is removed from the blanket of the breeder after a period that gives the lowest price of electricity for the entire system. This optimization is determined by the capital costs of the CIT breeder and LWR's, the CIT circulating power, the LWR conversion ratio, and fuel-processing costs; these data are given in Table 3. The plant factor for all reactors is taken as 80%.

Figure 8 shows the variation of electrical energy cost with the fractional burn-up of U at the time of blanket removal. The optimal removal time occurs at 0.4% burn-up, or a period of 7.4 years. Evidently the entire fissile fuel production should be removed at that time. Having determined the optimal fuel management period, the blanket performance averaged over its lifetime is readily determined, and is given in Table 4. (The integrated fusion-neutron energy current during this period is 2.4 MW-yr/m^2, averaged over the first wall, so that a stainless steel wall operated at moderate temperature need be replaced only at every second fuel management period.[19])

Table 3. Basic data for economic optimization.

CIT Breeder	
Capital cost[a]	1430 M$ (1976)
Circulating power	260 MWe
Fusion power	218 MW
LWR with Pu recycle	
Capital cost	750 $/kWe (1976)
Conversion ratio	0.5
Fissile requirement @ 80% D.F.	0.35 kg/MWe/yr
Fuel process plus O&M	4.6 mills/kWh

[a]Includes reactor equipment, turbine plant, buildings, and indirect costs (607 M$).

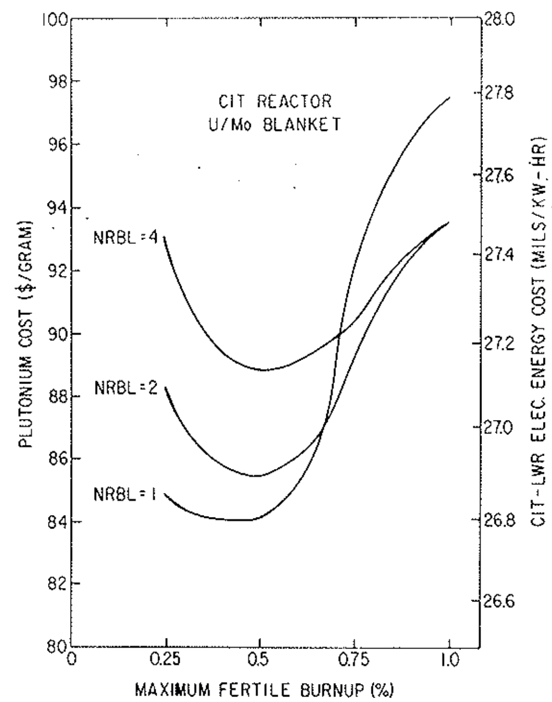

Fig. 8. Variation of cost of electrical energy from CIT-LWR system with per-cent fertile burn-up in CIT blanket at removal. NRBL = number of fuel management regions in the blanket.

Table 4. Blanket performance averaged over 30-yr. lifetime at 0.80 duty factor.

Fuel manag. period	7.37 yr.
Blanket exposure	2.4 MW-yr/m^2
Avg $\dot{M}$	10.94
Avg fertile burn-up	0.41%
Avg enrich. in-core	0.63%
Avg enrich. at removal	1.00%
Avg T breeding ratio	1.09
U consumption	1790 kg/yr
Li consumption	23 kg/yr
Net Pu production	1270 kg/yr
Net T production	0.96 kg/yr

Table 5. Characteristics of the nuclear energy center.

CIT Breeder	
Gross thermal pwr	2110 MWt
Gross elect. pwr	740 MWe
Circulating power	260 MWe
Net elect. pwr	480 MWe
Net plant effic.	0.23
Net Pu product.	1270 kg/yr
CIT-LWR System	
LWR elect. output	3805 MWe
Total elect. output	4285 MWe
Total capital cost[a]	990 \$/kWe
Effective Pu cost[a]	84 \$/g
Elect. power cost[a]	26.8 mills/kWh

[a]1976 prices.

Table 5 summarizes the economics of the CIT-LWR system. The effective cost of Pu ($84/g) is determined from the increase in electrical energy cost of the LWR system that is due to the tokamak breeder, but this Pu cost is meaningful only in comparison with other potential sources of fissile fuel. The important result is the cost of electricity, namely, 26.8 mills/kWh. A second optimization study was made for a CIT device with more modest performance: This example used beam injection of 125 MW into a plasma of identical size as the first, but with a larger vacuum chamber for alleviation of the particle handling problem. The neutron wall loading was reduced to 0.31 MW/m^2. It was found that the effective cost of plutonium increased to $106/g, but the cost of electricity increased only to 28.4 mills/kWh. This result reflects the fact that capital cost rather than fuel cost is the dominant factor determining the price of electricity from fission reactors.

A similar study was made for a CIT breeder with a thorium blanket[18] producing ^{233}U for supply to a system of HTGR's with conversion ratio of 0.85 and fissile make-up requirement of 0.186 kg/MWe/yr. While the effective cost of ^{233}U was found to be extraordinarily high ($380/g), the cost of electrical energy was 36 mills/kWh, or just 33% more than that from the LWR system.

PROSPECTS FOR A CIT-TYPE FUSION DRIVER

The total ion energy (including fast ions) in present injection experiments is already comparable with or even exceeds the electron energy[3] ($\mathbb{H}_i \simeq 1.5$). In larger tokamaks presently under construction or beginning operation (PLT, PDX, DITE), W_b will be sufficiently low so that at least half the beam energy will be transferred to bulk-plasma ions, and the beams themselves will provide significant fueling. Therefore a continuation of the present trend toward higher $\mathbb{H}_i$ seems likely, and with it enhanced interest in hot-ion, warm-electron tokamak reactor plasmas such as the CIT.

The principal plasma-engineering problems are those associated with achieving large fusion power densities in <u>quasi-stationary</u> operation, such as providing adequate means for maintaining a large particle throughput and impurity control. Two new divertor experiments (DITE, PDX) are especially relevant for testing the heavily driven CIT operating mode, since an "unload" divertor should be effective in the rapid pumping of ions that diffuse out of the discharge, thereby reducing the recycling of cold plasma and neutrals into the hot reacting region. In PDX, elaborate getter-pumping surfaces can be installed in large compartments above and below the discharge region,[20] so that even without divertor operation, the influx of cold particles into the discharge should be reduced greatly. It is worth noting that very low recycling is observed in the Alcator tokamak[21] (without a divertor), where the density drops continuously in time unless puffs of gas are injected into the plasma. In the ATC device the same result has been obtained by titanium gettering of the vacuum wall.[22]

The PDX device will be equipped with 4 MW of 40-keV neutral beams, so that the effectiveness of fueling and plasma-current control by the beams can also be explored. Injector systems with characteristics required by the CIT breeder are presently under development in conjunction with various tokamak programs. Thus both the physics and engineering feasibility of the fusion driver described in this paper can be assessed within the next several years.

ACKNOWLEDGMENTS

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TOKAMAK ACTINIDE BURNER DESIGN STUDY*

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ABSTRACT

The principal characteristics of a beam driven tokamak fusion-fission hybrid reactor for actinide depletion have been developed in this design study. Based on mid-to-late 1980's technology, this reactor provides a fusion neutron wall loading of 1 MW/m^2 corresponding to $\sim 10^{14}$ n/cm^2-sec in the 14 MeV energy range. The blanket is fueled with residual actinides contained in the high level waste from the LWR-U cycle. These residual actinides are the only fissionable material contained in the fast helium cooled blanket lattice and provide sufficient neutron multiplication to give neutron fluxes in the range of 10^{15} n/cm^2-sec. A principal result of the study has been the development of a revised criterion relating hazard potential from the wastes to that of the naturally occurring parent uranium ore from which the wastes were produced. Results of the study indicate that the total fluence required for effective depletion is quite high and fusion neutrons do not appear to have distinct neutronic advantages for actinide burning until wall loadings approaching 10 MW/m^2 are considered. Equally significant, some extremely challenging problem areas were encountered in the design area, particularly for the fusion driver. The impact in plasma engineering and technology required for long pulse, high duty cycle operation necessitated the use of superconducting magnets, a divertor system for control of impurities and particles lost from the plasma, a unique concept for a neutralizer system to remove particles swept into the divertor, vacuum and neutral beam systems capable of sustained operation, and a viable means of tritium injection.

INTRODUCTION

The objective of this project was to perform a preliminary design study of a fusion-driven reactor based on coupling a TCT fusion driver with existing fission reactor technology. The goal of such a design was to demonstrate early practical use of TFTR[1] technology to dispose of actinide wastes from fission reactors with a target for operation in the mid-to-late 1980's. The choice of a TCT for this application is attractive for several reasons. Its selection for the TFTR puts the TCT in the mainstream of fusion development. Accordingly, the actinide burner would be able to draw on the copious amount of information expected to be developed with the TFTR and therefore enable such a device to be designed with only comparatively minor extrapolation of the plasma physics data. The TCT does not inherently require the solution of all problems encountered for a true ignition machine. Specific concerns with regard to plasma characteristics, such as confinement time and temperature, are significantly relaxed in the TCT. While some uncertainty exists with regard to the feasibility of a TCT as an economic power producer, it has the potential to supply a substantial number of fusion neutrons. Calculations performed by Princeton,[2,3] have shown that the power density (and, hence the neutron density) in TCT can be an order of magnitude larger than the maximum attainable with a thermal fusion reactor of the same beta at any temperature. Among other applications, these neutrons could be used to transform very long-lived fission reactor waste products into more acceptable shorter-lived radioactive waste products for ultimate disposal.

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** Princeton Plasma Physics Laboratory

Several papers have been published over the past few years in which the transmutation of radioactive wastes by neutron bombardment has been considered. These studies indicate that while transmutation of certain fission products by fusion neutrons may be of interest, actinide disposal in fusion driven devices was potentially very attractive. Although the hazard level represented by actinides is lower than that of fission products, the duration of the hazard is much longer. Therefore, it was suggested that transformation of the actinides by utilizing fusion neutrons offered a means to alleviate the need for very long term storage or other forms of disposition. This could present a good solution to a serious problem now confronting the utilities operating fission power plants.

The design chosen for the actinide burner should utilize a concept that is extrapolatable to an economic commercial actinide burning reactor and which represents significant improvements over pure fission reactors in the areas of safety, environmental impact and safeguards. To achieve these goals, the following design requirements and guidelines were developed:

- Utilize TFTR plasma physics.
- Apply existing fission reactor technology for blanket concepts to the maximum extent possible.
- Minimize technological risks by placing heavy emphasis on projected state-of-the-art technology for the 1980's.
- Minimize use of fissionable materials other than actinides.
- Promote actinide disposal by fission rather than transmutation.
- Accelerate the disposal of actinides relative to natural decay by orders of magnitude.
- Provide ready access for remote refueling and maintenance operations on the blanket.
- Minimize the production of other radioactive by-products.

Although TFTR physics parameters such as plasma temperature, density and confinement time can be utilized, it is necessary to increase the neutron production rate and first wall loading relative to TFTR to accelerate the disposal of actinides. It is also necessary to adopt a high duty cycle mode of operation (long pulses with short time intervals between pulses) to obtain the required neutron fluences. Limitations in the state-of-the-art technology in areas such as materials, however, impose restrictions on how one approaches the burning of actinides in a first demonstration reactor. Therefore, the use of the TCT concept with wall loadings consistent with mid-1980's technology requires neutron multiplication in the blanket to achieve the neutron fluxes necessary to burn actinides at a reasonable rate.

REACTOR LAYOUT

A number of limitations were found in attempting to adapt available tokamak design concepts and fusion driver configurations to the actinide burner. In order to meet the actinide burner driver requirements, several significant departures from the TFTR-TCT fusion driver parameters were identified. These are evident in the parametric data comparison shown in Table 1. Initial fusion driver studies indicated the need for a divertor system to permit long pulse duration (~50 sec) operation. Also, initial blanket nuclear studies indicated that blanket neutron populations and conversion requirement and emphasized multiplication factor requirements and emphasized the need for as high a first wall loading as possible. The near steady-state operating conditions which result from the longer pulse duration and higher plasma power levels (~700 MW) that are required, imposed a major design problem for the liner and vacuum vessel for structural cooling as well as a need for a divertor; a significant departure from the TFTR.

The principal fusion driver design requirements imposed by the actinide burner application are generally common to all long pulse duration tokamak devices, including the power reactors. In addition, the fissioning required to obtain adequate neutron multiplication in the blanket system imposes constraints in blanket configuration and cooling that must be accommodated. The design considerations are listed in Table 2.

Table 1. Fusion Driver Design Parameters for TFTR and the Actinide Burner

	Actinide Burner	TFTR*
R_o (m)	3.9	2.5
a (m)	0.9	0.85
A	4.33	2.9
Elongation	1.50	1.0

Table 1. Fusion Driver Design Parameters for TFTR and the Actinide Burner (Cont'd).

	Actinide Burner	TFTR*
Horizontal wall radius (m)	1.50	1.1
Plasma volume (m^3)	104	35
First wall area (m^2)**	218	110
B_t on axis (T)	3.65	5.2
B_t on coil (T)	8.9 (9.2)	9.5
I_p (MA)	3.8	2.5
q	2.5	3.0
$\bar{n}_e$ (cm^{-3})	1.0×10^{14}	7×10^{13}
$T_e = T_i$ (keV)	6.5	6.0
$\bar{n}_e \tau_E$ ($cm^{-3} \cdot s$)	9×10^{12}	1.5×10^{13}
W_o (keV)	200	120 & 60
Beam mean-free path (m)	0.54	0.60
Beam power (MW)	325	35
Γ	0.92	0.60
β_p	3.4	2.0
Q_b	1.24	1.0
P_f (W/cm^3)	3.36	0.80
Total neutron production (n/s)	1.4×10^{20}	9×10^{18}
Neutron loading at first wall (MW/m^2)	1.47	0.1
Total neutron power (MW)	320	16
Total thermal power (MW)	725	55
TF coil inside dimensions diameter (bore), (m)	5.35 × 9.0	2.8
Magnet type	Superconducting, Nb_3Sn	Water-cooled copper
Impurity control	Liquid Lithium coated nested chevrons and external pumping	Injection of cold neutral gases near the wall
Pulse duration (s)	50	1 to 4 flat top
Time interval (between pulses) (s)	10	284

*Design parameters as of 4/1/76

**Applies to liner for Actinide Burner, vacuum vessel for TFTR

Table 2. Key Design Requirements.

Common to Long Duration Devices

- Large pumping requirements for plasma maintenance/on-line operation.
- Removal of contaminant (scrape-off particles) by a divertor system.
- High beam injection rates and long durations (except for ignition system).
- On-line refueling (D-T) during operation.
- Steady-state structural cooling.
- Provision for first wall* replacement and vacuum vessel maintenance/repair.
- Maximum first wall neutron loadings.
- High electrical impedances in structures (first wall and vacuum vessel).

*The liner is the "first wall" facing the plasma in the actinide burner.

Unique to Actinide Burner

- Blanket neutron multiplication and large thermal power removal from blanket.
- Blanket fuel redistribution and removal system (fuel management).
- Fission product control and management.
- Criticality considerations.

FUSION DRIVER PARAMETERS

The modest $n_e \tau_E$ required for TCT operation indicates that the fusion plasma can be relatively small. Table 1 gives the plasma parameters for the TCT fusion driver of the actinide burner reference design, while Figure 1 shows the vertical cross section. The plasma half-width of 0.9 m insures that the required $n_e \tau_E$ can be attained, is convenient for penetration by 200 keV neutral beams, and results in a satisfactory neutron output.

The blanket on the inside of the torus has been omitted. Including a blanket in this region would result in an unacceptably large device for a given wall loading, while there is serious question concerning the feasibility of servicing inner blanket modules. The major radius is then determined by the thicknesses of the transformer core, the TF coils, and the inner shield.

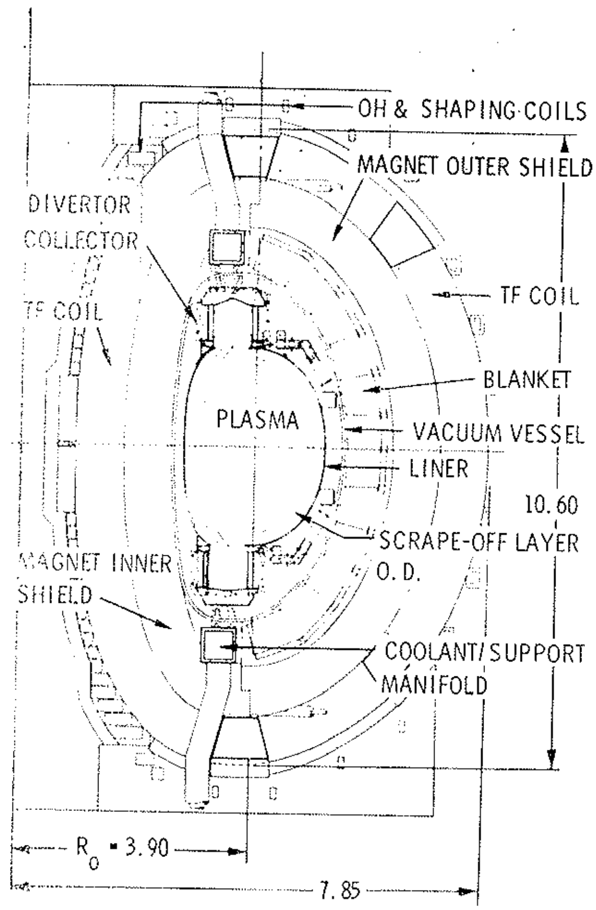

Fig. 1. Vertical Cross Section - Actinide Burner.

The plasma parameters were derived assuming a rectangular cross-section for the plasma; for K = 1.5, the shape factor S = (plasma circumference)/(2 π x half-width) = 1.59. The actual plasma cross-section is more D-shaped with $S \approx 1.45$. Since $P_f \propto (SB_{t\ max})^4$ for a given plasma aspect ratio and q, the plasma pressure (and neutron source strength) can be maintained by increasing $B_{t\ max}$ above the nominal design value of 8.9 T. Note that the design limit for the Nb_3Sn coils that are employed is at least 11 T.

Since the fusion driver for the actinide burner is based on an extension of the TCT concept as employed in TFTR, it is appropriate to discuss fusion driver parameters as they relate to those of the TFTR.

Principal features of the TFTR[4] are listed in Table 1. The TFTR plasma properties are fairly close to those required of the fusion driver for the actinide burner -- that is, plasma current, density, temperature, beam voltage, confinement time are similar. The pulse length in the TFTR will be adequate for studying plasma heating and neutron production by the beams, temperature dependence of confinement time, and in fact all facets of beam-plasma interaction. The principal distinction between TFTR and actinide burner operation is one of duty factor. The TFTR will have a duty factor of at most 1%; it utilizes room-temperature TF coils, short-pulse beam injection, no refueling, and no divertor. The actinide burner, on the other hand, has a steady-state beam injection, continuous refueling, and a magnetic divertor. Thus, the TFTR experiments will provide the necessary plasma physics background for the actinide burner, but great advances in plasma engineering beyond TFTR capabilities are required.

Since the actinide burner study initiated in early 1975, a number of tokamaks have generated experimental information that is especially relevant to the actinide burner design parameters, as well as the design of other beam-driven hybrid reactors. These results, which are summarized here, confirm the essential soundness of the design values.

Confinement Scaling. While scaling of $n_e\tau_E$ from present ohmic heated tokamaks to beam-heated tokamak plasmas of larger size and T_e can be only speculative at the present time, useful estimates can be made. Most ohmic-heated tokamaks show an "Alcator-like" scaling;[5] viz., $\tau_E \propto \bar{n}_e q^{1/2}$ or $n_e\tau_E \propto \bar{n}_e^2 q^{1/2}$. The scaling with plasma radius is not known definitely, but preliminary results from T-10 and PLT indicate that $\tau_E \propto \bar{n}_e q^{1/2} a^2$. The following table shows the expected $n_e\tau_E \propto \bar{n}_e^2 q^{1/2}(a^2 + b^2)$ scaling. If $n_e\tau_E$ increases with size at least as $a^{1.5}$ and does not deteriorate markedly with temperature, then the required $n_e\tau_E$ for TCT reactors such as the actinide burner design can easily be met.

Estimated Confinement Parameters Using

$$\bar{n}_e \tau_E \propto \bar{n}_e^2 \, q^{1/2} \, (a^2 + b^2) \text{ Scaling}$$

Device	a (m)	b/a	q	$\bar{n}_e$ (10^{14} cm^{-3})	$\bar{n}_e \tau_E$ (10^{13} cm^{-3} s)	Required $\bar{n}_e \tau_E$ (10^{13} cm^{-3} s)
Alcator(9)	0.095	1.0	6	4.5^a	0.8^a	---
Actinide Burner	0.90	1.5	2.5	1.0	3.5^b	0.9

^{a}Measured

^{b}Scaled from Alcator

Poloidal Beta. $\beta_p \sim 2$ has been obtained in the Alcator device[5] with ohmic heating alone, and in TM-3 with electron resonant heating.

Neutral-beam injection on the ATC[6] and ORMAK[7] devices has resulted in $\beta_p \sim 1.5$. There appears to be no experimental evidence at this time that would preclude the actinide burner design value of $\beta_p = 3.4$ (0.7 times the aspect ratio).

Beam Pressure. In neutral-beam heating of ATC[6] and ORMAK[7], values of (beam pressure/bulk-plasma pressure) up to 0.6 have been obtained with no deleterious effects. Consequently, the actinide burner design value of 0.92 seems quite feasible.

Perpendicular Injection. Near-perpendicular neutral-beam injection experiments have been successfully performed on the TFR device, without excitation of any deleterious instability[8], and with effective plasma heating. Near-perpendicular injection can often be convenient in beam-driven reactors, because of better accessibility through the blanket, and ease of penetration to the central plasma region.

Vertical Elongation. Elongated plasmas with K = 1.4 on the Doublet IIA device apparently perform according to theoretical expectation.[9] Consequently, a reactor design value of K = 1.5 appears to be feasible.

In summary, these new experimental results confirm the essential feasibility of our plasma physics design values. But there are formidable plasma engineering problems, especially the injection of several hundred megawatts of neutral beams into a relatively small volume, the maintenance of the desired plasma density and composition, and the control of impurities.

DIVERTOR AND VACUUM SYSTEMS CONSIDERATIONS

Disposition of the energy and particles leaking from the plasma during the 50 sec ON pulse constituted a major technological problem that had to be solved in order to establish practical parameters for the fusion driver. A novel solution was conceived as described later. Special attention was given to minimizing both the amount of void in the blanket created by vacuum and neutral beam injection ducts and the amount of space required for the divertor chamber.

The particle removal requirements are listed below:

Total leakage of ions, $N/s = n_i V_p / \tau_p$	6.6×10^{22} D, T/s
Leakage of $D/s = P_B/W_o$	1.0×10^{22} D/s
Leakage of T/s	5.6×10^{22} T/s
Fusion neutron production	1.4×10^{20} n/s
Leakage of α/s	1.4×10^{20} α/s

A double null poloidal divertor is incorporated in the design to remove particles from the plasma chamber which leak from the plasma or sputter off the walls. The total thermal power (725 MW) is composed of neutron power, P_n, alpha power P_α, and neutral beam input power, P_B. Since this is a Q = 1 machine, $P_n \simeq P_B$ = 325 MW and $P_\alpha \sim 80$ MW. The neutrons are not captured by the divertor, the alphas are considerd to give up most of their energy to the plasma, and some 40% of the remaining energy is given up in radiation and charge exchange so the energy carried by the divertor scrape-off-layer to the burial chamber is about 285 MW. Figure 2 illustrates the various mechanisms contributing to the particle load $N \alpha_c c_p (1 - f_{sd})$ in the burial chamber; contributions from impurities and from recoiling D and T atoms are neglected since they are at least 10^2 times smaller. Mercury diffusion pumps or cryopumps in conjunction with necessary cold traps, isolation valves, and pumping ducts can provide a pumping speed of about 3 ℓ/s per cm^2 of duct area. Flowing lithium has been proposed[10] as a surface for the neutralizer plate in the UWMAK design. It has several very attractive features: 1) it has a high sticking coefficient ($f \sim 0.9$) for hydrogenic particles since it combines chemically to form hydrides, 2) it continually presents a fresh surface to the particles so saturation does not occur, and 3) it can efficiently carry heat away from the neutralizer plate. Therefore, flowing Li was selected for the divertor of the actinide burner.

The following table shows the required pumping speed and duct area for reasonable values of α_c, c_p, and f_{sd}, first without a lithium getter and second with a lithium getter.

No Lithium:

N/s	3.65×10^{22} D, T/s
Q_1	950 torr - ℓ/s
p	10^{-5} torr
$S_1 = Q_1/p$	10^8 ℓ/s
A_1	2×10^7 cm^2 = 2000 m^2

With Lithium:

α_c, scrape-off layer capture efficiency		0.9
c_p, fraction of scrape-off layer entering burial chamber		0.9
f_{sd} (flowing Li surface)		0.9
$N \alpha_c c_p (1 - f_{sd})$	5.3×10^{21} D, T/s	
Q_2	74 torr -ℓ/s	
$S_2 = Q_2/p$	$7.4 \times 10^6 \ell$/s	
A_2	2.8×10^6 cm^2 = 280 m^2	

The available surface area is at most 400 m^2; duct area required for pumping out the burial chamber is more than 50% of this.

Two schemes for reducing this void area were investigated: a) use of cryopanels inside the burial chamber and b) manipulation of the parameters α_c, c_p, f_{sd}. Cryopanels have an effective pumping speed of 10ℓ/s/cm^2 so the necessary area will be 74 m^2, which is rather large but can be accommodated. Regeneration of the cryopanels by heating the surfaces would be required once a day; the collected particles when vaporized would generate a pressure of 15 torr requiring about 2×10^4 cm^2 of duct area for mechanical pumps to pump the pressure down to 10^{-2} torr in about 15 minutes. Re-establishing the liquid He temperature on the cryopanel surface would recondense the remainder of the D and T atoms and establish the original low pressure condition. However, the location of these cryopanels in the divertor chamber just above the plasma exposes them to the neutron flux. To maintain the desired temperature against the heating effect of the neutron flux would require some 80 MW of refrigerating capacity. Since shielding for cryopanels would generate similar amounts of heat close to the panels, the refrigeration load would be only slightly reduced. In either case, the refrigeration load is too large and cryopanels turned out not to be feasible.

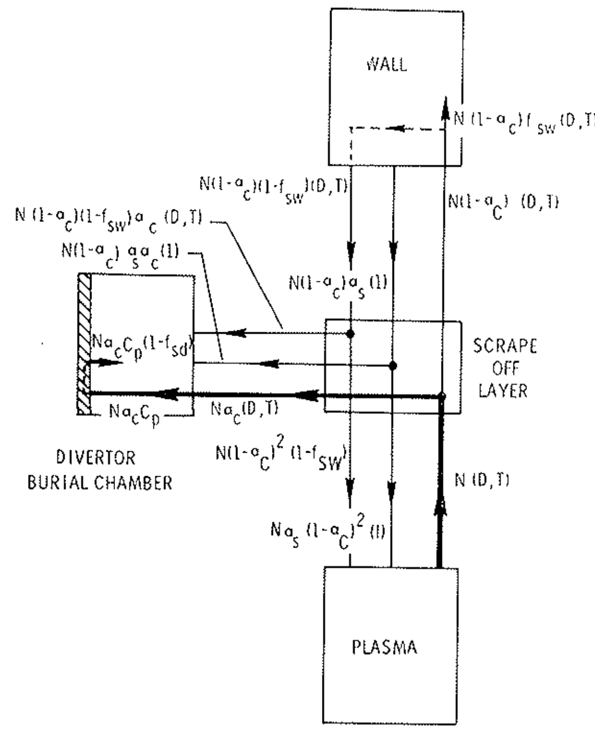

Fig. 2. Mechanisms Involved in Particle Transport

Divertor Capture Efficiency, c_p. The capture efficiency $c_p = 1 - \exp(-W/\lambda)$; the scrape-off layer width $\lambda \simeq 12$ cm is evaluated from the relation

$$\lambda^3 = 4 \times 10^6 \, \alpha D_\perp T^{3/2} \text{ (eV) } \log_{10} M/n_e$$

where $D_\perp \sim 2 \times 10^3$ cm^2/sec is the coefficient describing diffusion of charged particles across the plasma into the scrape-off layer and M = 2 is the mirror confinement factor for the inner divertor configuration. For W = 30 cm, $c_p = 0.92$. Since λ is insensitive to small changes in plasma properties and W cannot be any larger because of space limitations, c_p appears to be fixed at ~0.9.

Scrape-off Layer Capture Coefficient, α_c. Consider making the scrape-off layer partially transparent; (i.e., $\alpha_c < 1$) in order to reduce the particle load in the divertor chamber. This allows emerging D and T ions and neutrals to strike the wall and cause sputtering; because the scrape-off layer is partially transparent, some of these impurities will enter the plasma and cause its Z_{eff} to increase. The concentration of impurities n_I required to make $Z_{eff} = 3$ is shown for three possible wall materials.

Material	Z	$n_{I\,max}$
Li	3	$22 \times 10^{-2}\, n_e$
Fe	26	$4 \times 10^{-3}\, n_e$
W	70	$4 \times 10^{-4}\, n_e$

From Fig. 2 the flux of impurities back into the plasma is $N\, \alpha_s\, (1 - \alpha_c)^2$ particles per second. In 50 seconds the total number of impurity particles in the plasma

$$N_{I\,tot} = 50\, n_i V_p\, \alpha_s\, (1 - \alpha_c)^2/\tau_p$$

Since N_{Itot} must be $\leq n_{Imax} V_p$, we can solve for α_c and get α_c (Li) = 0.88 and α_c (Fe) = 0.93. The capture coefficient, α_c, turns out to be fairly insensitive to choice of wall material or to level of fusion power in the plasma. This is because even though n_{Imax} for Li was 22% n_i and for Fe was only 0.4% n_i, the sputtering coefficients were in the opposite direction.

$$\alpha_s\,(\mathrm{Li}) = 10^{-1},\ \alpha_s\,(\mathrm{Fe}) = 5 \times 10^{-3}$$

The optimum value for α_c appears to be fixed in the range 0.9.

Sticking Coefficient of Neutralizer Plate, f_{sd}. Getters such as freshly evaporated Ti can capture hydrogen particles until they saturate. The PDX design[11] evaporates a new layer of Ti on getter plates in the divertor after each pulse. This is clearly not applicable to the quasi-continuous operation of the actinide burner. However, using a single flowing Li neutralizer plate, as in the UWMAK[10] design, does not reduce the particle load sufficiently as was shown in the previous calculation.

A modified honeycomb[12] design has been conceived[13] which retains all the advantages of flowing Li and increases the overall sticking coefficient of the system to >0.99 by providing several Li covered surfaces so particles are forced to undergo many bounces before they can emerge into the divertor chamber. The configuration for this design consists of nested chevrons as shown in Fig. 3. Since 90% of the incident D and T particles are absorbed at each bounce, only two bounces will reduce the D and T particle load to 1% of the original flux. This concept differs from the original honeycomb concept in that: 1) it is designed for use in a divertor chamber rather than in front of the first wall of the plasma chamber, 2) it uses flowing Li as the surface of contact thereby increasing the sticking factor at each bounce, 3) the peculiar geometry of flux lines in the divertor region requires that there be no side walls that could block particle impingement on Li so an array of plates is used. The total surface area will be determined by the need to remove about 285 MW of power; a conservative thermal limit for liquid lithium is 1 MW/m^2. Thus, 285 m^2 of liquid lithium covered plates will have to be used. This is easily achieved for the multi-layer chevron array within conservative divertor volume designs.

This concept solves the problem of particle removal; furthermore, the powerful gettering action of the multiple bounce array of Li covered plates provides the possibility of completely eliminating the need for cryopanels inside the divertor chamber and substantially reducing the requirements for pumps outside the chamber.

The design approach considered to best accommodate the actinide burner design requirements included the following features:

- Double, inner-major-radius divertor system with divertor coils external to blanket, in conjunction with an elongated plasma and D type TF coils.
- Large liquid metal (lithium) covered chevron "gettering" surfaces for divertor neutralizing plates.

- Full torus annular vacuum manifold internal to vacuum vessel and TF coils.
- Free standing segmented replaceable liner.
- Multiple beam, dispersed angle neutral beam injection at a minimum number of points (four principal locations).
- Omission of inner blanket sectors because of severe access limitations and reactor size considerations.
- Modular blanket sectors to hold fuel elements and control coolant flow.
- Plasma on-line fueling by T injection.

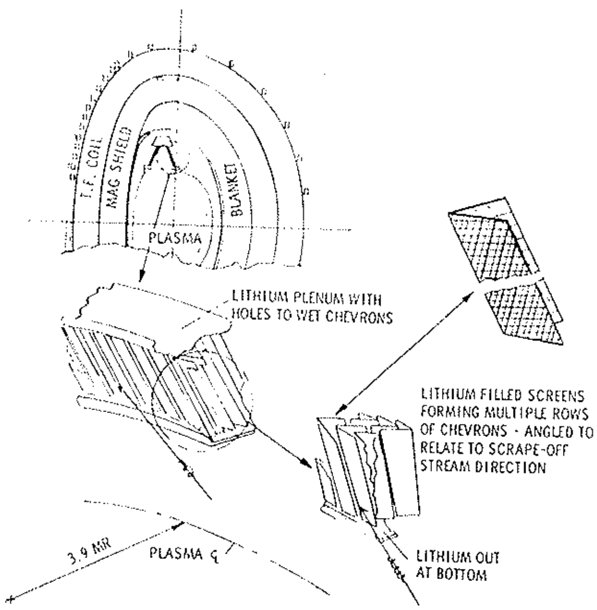

Fig. 3. Divertor Particle Collection System Reference Concept

The principal features developed to accommodate the above requirements are embodied in the reference design depicted in Fig. 4. Tentative arrangements and approximate configuration and sizes are illustrated for the plasma, liner, divertor system, vacuum vessel and blanket. The shield space and coil locations are also shown.

With the exception of local diagnostics, all vacuum vessel systems requiring penetrations are confined to the common beam and vacuum system entry ports located at four azimuthal locations. This minimizes the seal welding requirements. The blanket and vacuum vessel cooling system (entering at the bottom and exiting at the top of the sectors) is fully outside the vacuum system, requiring no penetration. The coolant required for structures within the vacuum vessel is limited to liquid metal (lithium). The general arrangement shown provides the means to remove the liner, divertor system, and coolant plumbing with minimum difficulty. These systems are integral and can be supported on tracks located at the top and bottom of the vacuum vessel. These components are also segmented for ease of handling.

The functions and advantages of providing a vacuum manifold and intercepting the particle flux obtained with the incorporation of a first wall liner have been substantiated during this study. However, the thermal advantages of providing a radiation cooled system, coupled with the high wall loadings that are desired for the actinide burner, leaves an area of large uncertainty and design development that has not been fully resolved.

The incorporation of an effective divertor system, Fig. 3, does reduce the surface heat flux considerably. However, a substantial heat deposition (of the order of half the scrape-off energy, and an internal heat generation due to the passage of the high energy neutrons of nearly 10 watts/cm^3) must be dissipated. Since the identification of a suitable high temperature material with appropriate surface characteristics has not been possible based on the current estimates of divertor system effectiveness, an alternate configuration has subsequently been adopted. This concept involves the augmentation of radiation heat transfer by "conduction paths" to the cooled vacuum vessel structure.

Preliminary sizing, structural requirements, and limited thermal analyses have been conducted on the divertor system. Also, a preliminary experimental assessment of the heat transport capabilities of the lithium concept has been initiated. Preliminary analysis of the divertor field has indicated that a shift in the location of the divertor chevron collectors to be more coincidental with the trajectory of particles swept from the scrape-off layer may be needed.

The design concept of separation of functions permitted the selection of Mo-TZM as the liner material. This easily replaceable, segmented

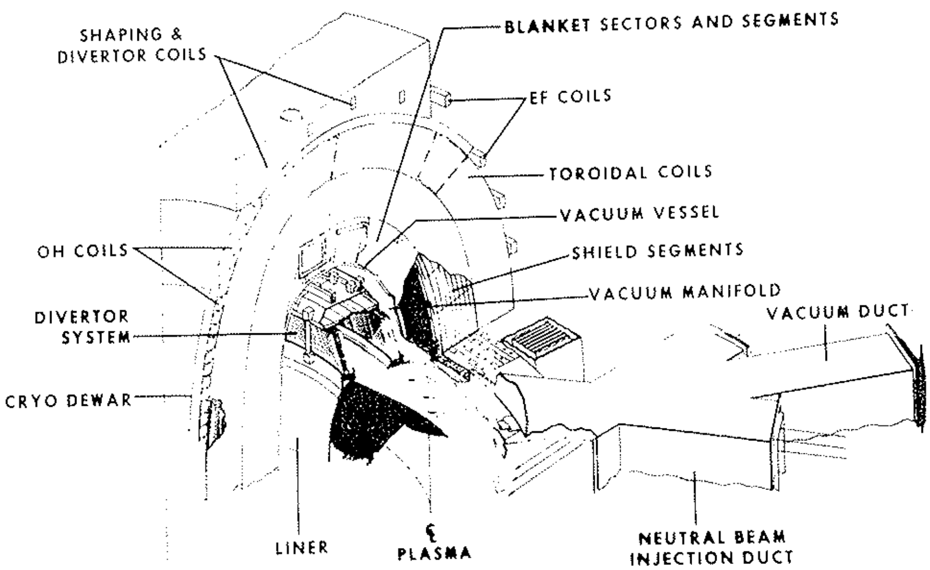

Fig. 4. Reference Configuration, Fusion Driven Actinide Burner

system provides a thermal barrier to the more crucial vacuum vessel. The liner also provides the means of attaining the desired vacuum manifold.

FIELD COIL DESIGN

Figure 5 shows the preliminary layout of the hybrid fusion reactor plan view. The TF coil arrangement and available space between the coils provides reasonable access for the neutral beam system and will permit shielding for the TF coils.

A cryostatically stabilized superconducting magnet configuration has been defined for the TF and OH coils. Figure 6 shows this configuration at the inner torus region. In order to accommodate the small (3.9 meter) major radius, the TF coil nose section and OH coils must be enclosed in a common dewar system. A dewar arrangement has been identified that permits torus separation without dewar wall connections being required. To facilitate separation, a two-way circumferential flow system separated at the mid-plane is provided and the dewars are fitted back-to-back to limit heat leaks into the system.

The principal design criteria established for the conceptual design include:

- Positive positioning and containment of each TF coil conductor in a bobbin structure.

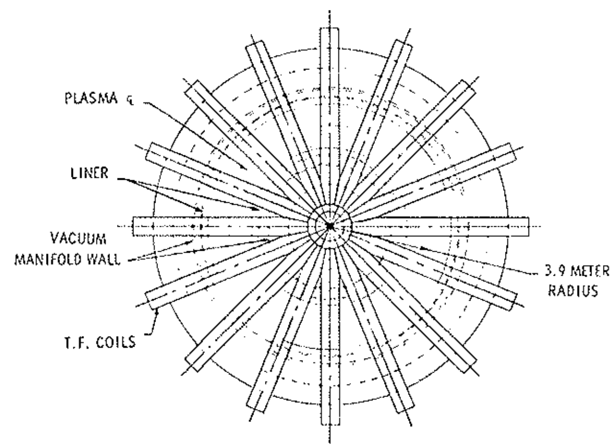

Fig. 5. Reference Concept Plan View

- Positive, controlled liquid helium cooling of each conductor.
- Compressive loads from the wedging action of each bobbin of the coil structure to be reacted by the stainless steel bearing surface and not by the conductor and not exceed a compressive stress of 4.3×10^4 kN/cm^2 (60,000 psi).

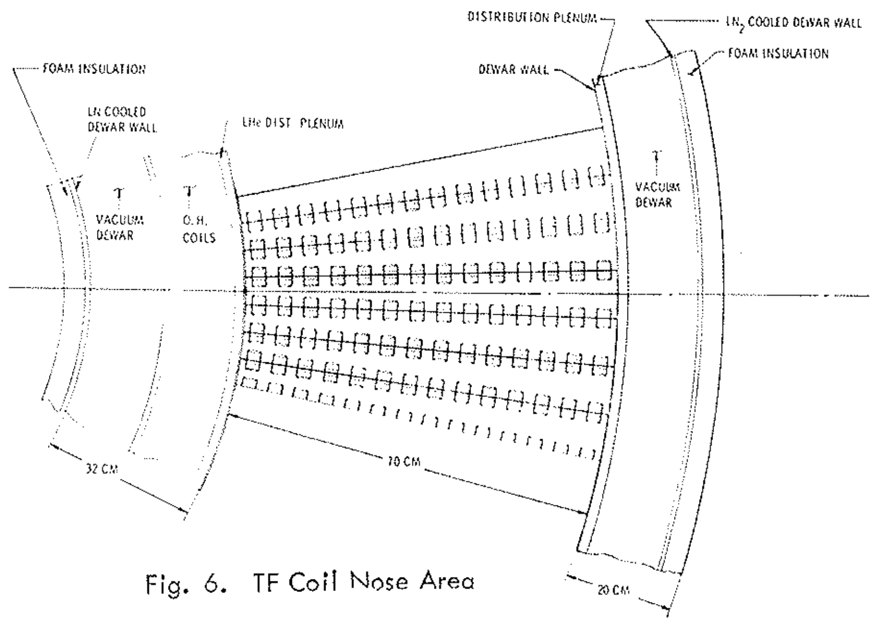

Fig. 6. TF Coil Nose Area

- Cryostatically stabilized design with Nb_3Sn superconductors at 4.2°K.
- Reference fusion driver dimensional constraints.

Scoping calculations were made that indicated the mass relationships needed for conductors of superconductor filaments plus copper cryostatic stabilizing material plus cooling channel geometry were satisfied by the configuration shown in Fig. 6. A total of seven stainless steel bobbins are sufficient to provide the space and structure necessary for the conductors of each TF coil. These bobbins are in direct wedging contact with each other in the nose section of the coil. This minimum space configuration requires the use of a common dewar system for the cryogenic cooling in this region.

The liquid helium used to establish the 4.2°K cryogenic temperature of the conductors and the stainless steel bobbins is provided by two coolant inlet manifolds. One manifold is shown at the inner major radius of the TF coil and the other manifold is at the same horizontal plane on the plasma side of the TF coils. However, a more suitable region for access could be at the top and bottom of the coils. These channels feed liquid helium into the bobbins of the coils where it is in contact with the conductors. Angled feed channels provide a uniform pressure source for flow in all channels of the TF coils. A similar manifolding system exists at the outer major radius position of the TF coils. At this location, the liquid helium is retrieved from the coils and returned to the refrigerating system.

Two coil centering and stiffening rings made of stainless steel are located in a horizontal plane at the inner torus major radius of the TF coils. These rings provide the manifold systems for the coolant of the TF and OH coil systems. They also assure centering so that no individual TF coil bobbin is displaced significantly from its neighbors.

The TF coil bobbin configuration defined as depicted in Fig. 6, assures a maximum physical constraint of each individual conductor, as well as a positive liquid helium cooling capability. The configuration of the liquid helium coolant distribution channel in the TF coil provides for a minimum loss of structure cross sectional area in the region where the fluid entry and exit take place. The preliminary conceptual configuration should prove adequate with only minor modifications once more detailed structural and thermal/fluid analyses are ultimately completed.

VACUUM VESSEL CONCEPT

Attempts to adapt the various vacuum vessel concepts that have been suggested for current EPR and other tokamak reactor designs to the actinide burner have not resulted in an adequate approach compatible with the requirements for blanket fuel management. Some of the principal differences that have produced this situation are:

- The very large, non-symmetrical "D" shaped vacuum vessel cross-section that is required.

- The necessity to have an easily replaceable/movable blanket system that must be periodically shuffled.
- The high neutron wall loadings that are necessary (an order of magnitude greater than EPR designs now being considered).
- The need for incorporation of a quick response leak detection system.
- The patent magnitude of the material mass needed to provide the size vacuum cavity that this device requires.

The very large non-symmetrical cross-section of the vacuum vessel coupled with the leak-tightness requirements of the fusion driver represented a substantial structural design problem. Conventional design of pressure vessels typically results in heavy wall thicknesses. This is untenable for the size pressure vessel needed for the actinide burner, and for the anticipated radiation damage distortions, as well as unacceptable from assembly/disassembly and operation impracticalities.

It appeared prudent to separate the two major functions of sealing and reacting pressure loading. By this approach it was possible to consider the use of less sophisticated (even non-ductile) structural sectors that could be self-supporting and mutually constraining, thus providing a more tractable remote handling arrangement. These sectors could be assembled with no necessity for forming a leak-tight structure. The concept is depicted in Fig. 7. Some of the more pertinent vacuum vessel design assumptions and requirements considered are outlined as follows:

- The vacuum vessel cross-section geometry and key dimensional constraints as shown in Fig. 1.
- A leak-tightness for a 10^{-7} torr initial vacuum condition for startup.
- "Bakeout" capability on the order of 500°C.
- Long life operation of >5 years (contingent on design/replacement ease, but high material activation and huge mass of structure dictates a very long usefulness).
- Remote assembly/disassembly/repair and maintenance capability for entire vacuum vessel.
- Provision for leak detection to identify and locate leaks at $\sim 10^{-7}$ torr.
- Cyclic operation (the order of 2×10^5 per year at 75% utilization).
- Up to 2 MW/meter^2 of 14 MeV neutrons at the inner radius (additional neutron loading will come from the blanket source).
- Structural support that is sufficient to resist a collapsing pressure load of 3 atmospheres.
- High electrical impedance (azimuthally).

The inner surface of the supporting structural sectors must accommodate the heat that is rejected by the liner as well as that which is generated by the high neutron flux. The separation of the liner functions permit the following advantages in meeting these severe requirements:

- Ability to select the best materials for: 1) distributing the heat at the inner interface; 2) structural support and long life in the neutron environment; and 3) leak-tightness requirement where high ductility requirements exist permitting selection of the material that has minimum ductility degradation, but which might not have reasonable strength or structural capacity at high temperature.
- Freedom to redesign for periodic blanket rearrangement and removal without violating integrity of the vacuum vessel - (the separation of the blanket from the vacuum vessel is considered necessary for a tractable design approach).
- Provision for cooling the vacuum vessel without imposing blanket coolant pressure loading (cooling alone presents a formidable problem).

The approach that is suggested for maintaining feasible temperatures for the structural material is shown in Fig. 7. The cooling of the wrapper by use of the helium that would be available from the blanket cooling could be permissible if the high pressure is contained in a header and ducting system within the wrapper. Otherwise, it would present an impossible pressure load against the vacuum vessel structure.

Heat removal from the flanges makes the use of heat pipes appear mandatory in order to maintain reasonable maximum temperatures. However, this does make a more complex design with the uncertainty associated with an unproven technology. Options for the wrapper cooling include the use of a formed coolant sheet for the outer

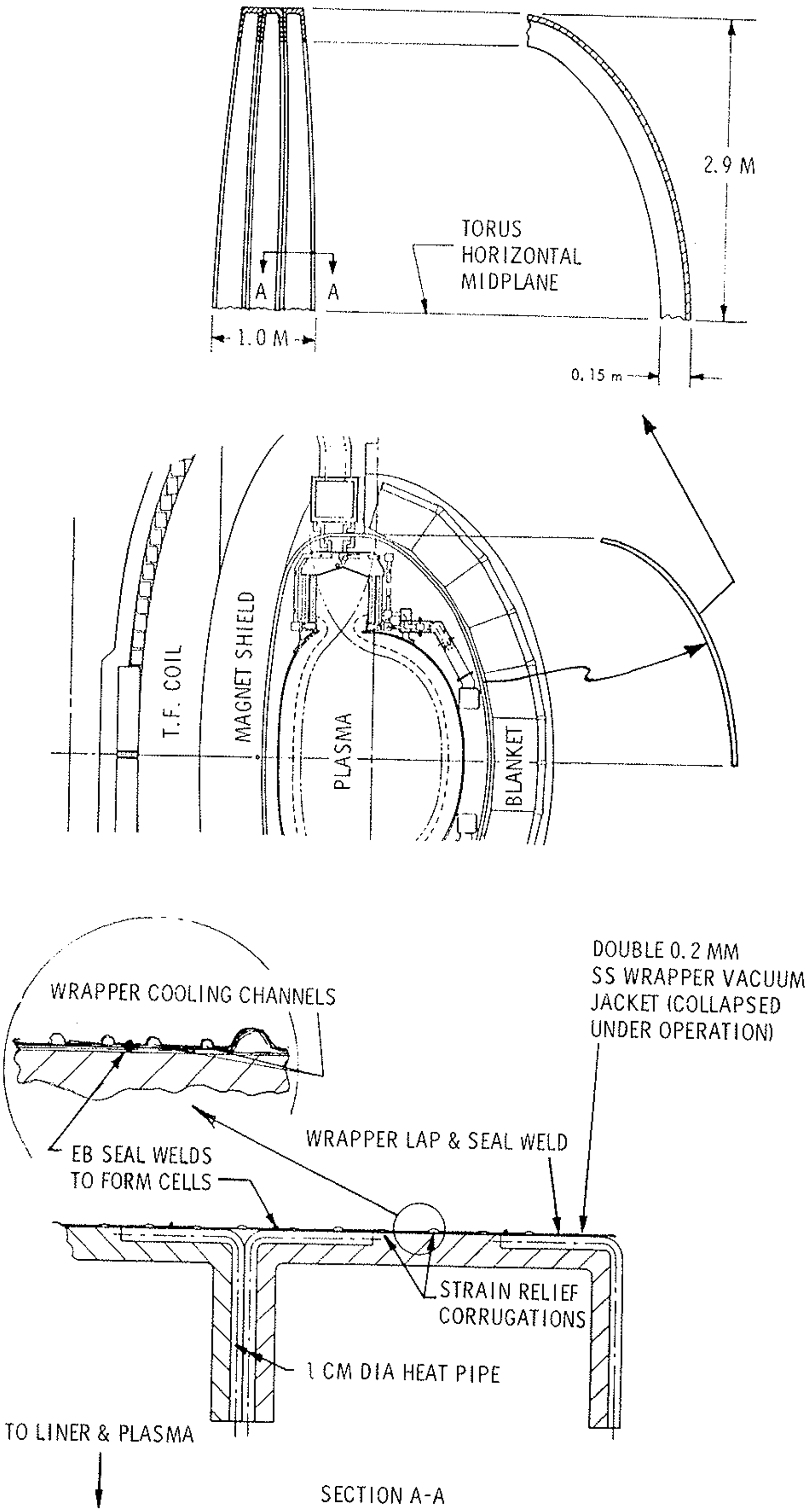

Fig. 7. Self-Supporting Vacuum Vessel Segment Configuration

layer which could be cooled by liquid metal or by another liquid or gas coolant system. If interconnected with the blanket cooling system through leak-tight quick disconnect fittings, it could be part of the blanket cooling system. The candidate approach is to cool the wrapper and the shield by separate cooling systems. This approach permits the blanket to be rearranged and shuffled without interference with the leak-tightness of the vacuum vessel cooling system.

BLANKET DESIGN

Large surface areas exist at the periphery of the tokamak fusion driver vacuum vessel. Therefore, a practical blanket concept requires modular design and the ability to easily and repeatedly attach both modules and cooling systems. In the case of the actinide burner, these modules contain fuel pins that are loaded with the actinides. Cooling system connections are a major concern for attachment of fission hybrid blanket modules. The requirement of the fuel management operations associated with movement of the modules on a periodic basis makes a leak-tight quick-disconnect submodule design important. Features that must be provided include:

- Containment of high pressure, high temperature coolant such as helium,
- accommodation of deformations and deflections associated with cyclic, long life, high temperature operation and remote handling,
- the removal, replacement and repair of modules with very short turn-around times,
- leak-tightness,
- maximum utilization of the available space.

None of the module arrangements or concepts that are currently available from the literature completely satisfy the required functions. Also, desired features such as: 1) rugged, simple attachment with, if possible, a single attachment lever; 2) large module fit-up tolerances to accommodate repeated replacement and variation in surface alignment; 3) provision for repetitive attachment even after deformation is sustained; 4) provision for thermal expansion and manufacturing dimensional differences; 5) provision for loading/unloading one module in a hot bay area while a second module is in operating position in the reactor; and 6) elimination of the requirement for welding at the reactor or in the subassembly to the blanket sector are quite important to a practical design.

BLANKET CONFIGURATION

The conceptual approach suggested for the actinide burner uses the high pressure coolant and a novel attachment device and module configuration to provide the desired features and overcome several of the major problems associated with current module configurations. The conceptual approach for the reference design is shown in Figs. 8 and 9.

Preliminary analyses have been completed on this concept. Compromises were found necessary to obtain desired blanket performance and meet design guidelines. Placement of the coolant duct outboard of the fuel region was necessary to reduce neutron attenuation and keep the modular blanket design and its associated coolant manifold independent of the vacuum vessel. The principal features of this module include the following:

- Standard fabricated fuel pellets of simple geometry.
- Multiple blanket segments permitting flexibility in the re-shuffling and assembly/disassembly of the blanket.
- Radial coolant flow of helium through the blanket segments with common inlet and exit plenums for each segment (48 segments).
- Leak-tight single entry quick-disconnects for the helium connections to the blanket.
- Cooled module containment structure.
- Ruggedness and simplicity of attachment; i.e., the use of a single attachment lever.
- Large tolerances permissible for module fit-up to accommodate repeated replacements and variations in surface alignments.
- Provision for repeated attachments and detachments even after deformation is sustained.
- Allowance for thermal expansion and manufacturing dimensional differences.
- Ability to load/unload one segment in a hot bay area while a second segment is in position in the operating reactor.
- No requirement for welding at the reactor or in the subassembly of the blanket sector.

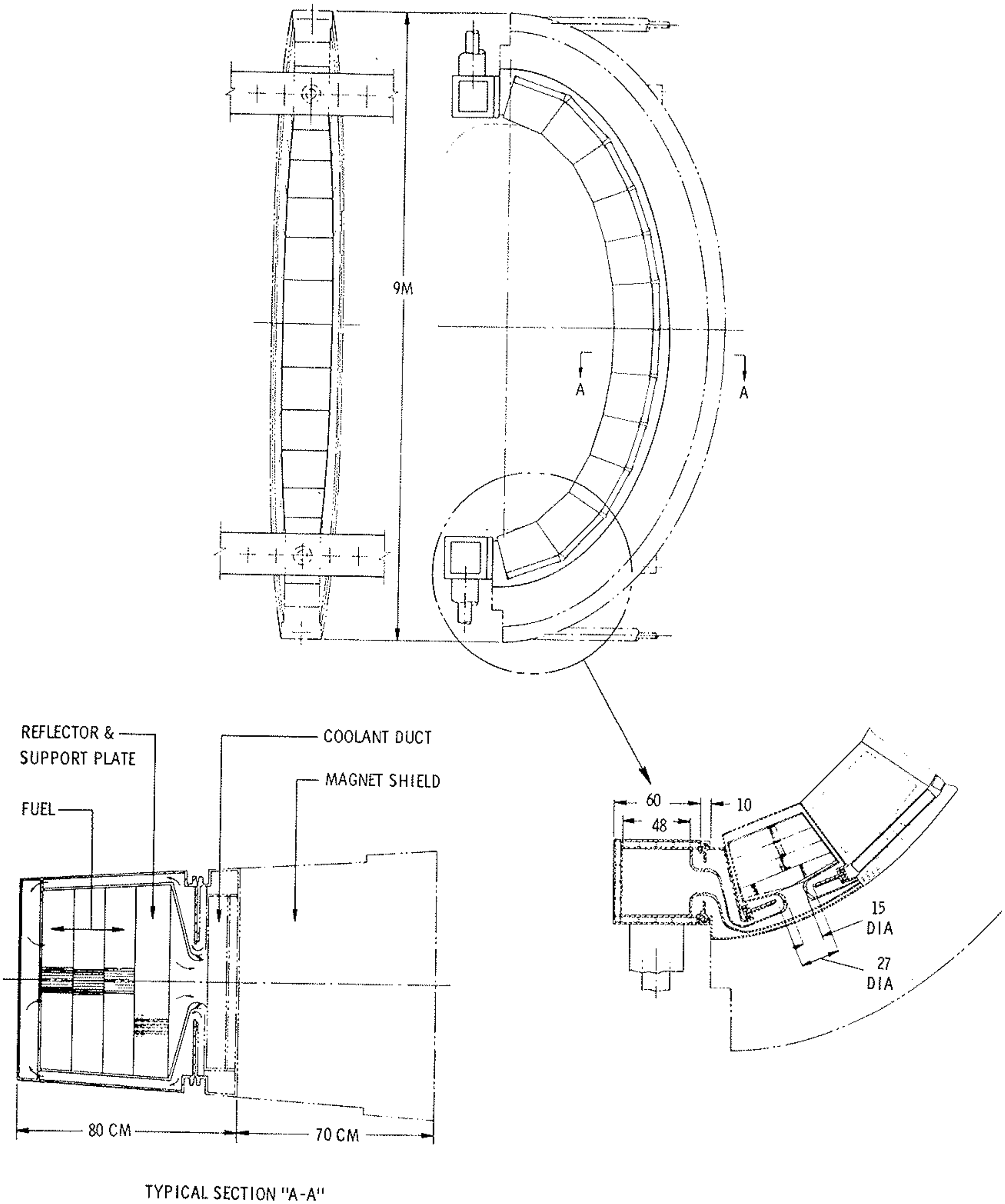

Fig. 8. Reference Blanket and Shield Sector Layout

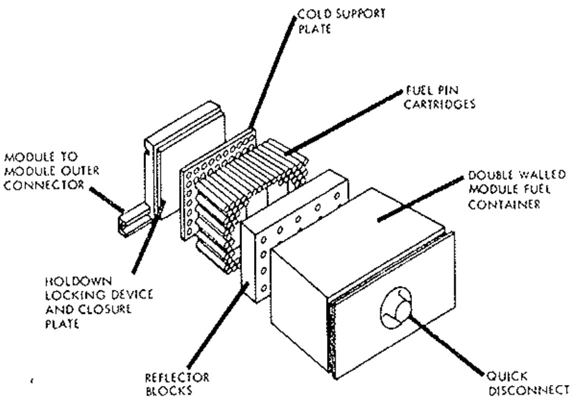

Fig. 9. Module Loading Sequence

A modular configuration offers the flexibility to accommodate different fuel loadings and thus appears to provide a reasonable reference conceptual design approach. The module concept incorporates a quick-disconnect and hermetic sealing feature, is designed to include a double-walled containment, and would be loaded in a suitable hot cell or bay area.

The conceptual design of the reference module/cooling manifold assembly minimizes remote handling assembly/disassembly problems, and maximizes leak-tightness without requiring welding during reactor assembly. It permits the accommodation of relatively large tolerances and permits the substitution of components.

The reference blanket module configuration is shown in Fig. 8. The fuel pellets are modeled after the gas cooled breeder reactor (GCBR) concepts[14] that operate in similar thermal and coolant systems. However, the arrangement of the module is such that it could accommodate other fuel forms such as the modified LMFBR fuel element or other fuel options.

The blanket and shield conceptual design arrangement have been defined to facilitate the removal of the outer shield sectors and blanket sectors without disturbing the fusion driver or TF coil systems. Conceptual design assessment of neutronic performance has shown the desirability of reducing the number of shield sectors to 32 and the need for design optimization to reduce the structural material between the plasma and blanket. Neither of these modifications have required a departure from the assembly/disassembly approach.

COOLING CONSIDERATIONS

Separate cooling systems have been considered desirable for the blanket and outer shield systems of the actinide burner as illustrated in Fig. 10. This provides improved blanket system thermal performance since it permits the maximum possible temperatures in the blanket coolant for maximum power conversion efficiency. Also, the shield sector can operate at lower coolant pressures by virtue of the lower shield power densities. The reference blanket cooling system is based on the gas cooled fast breeder reactor design[15].

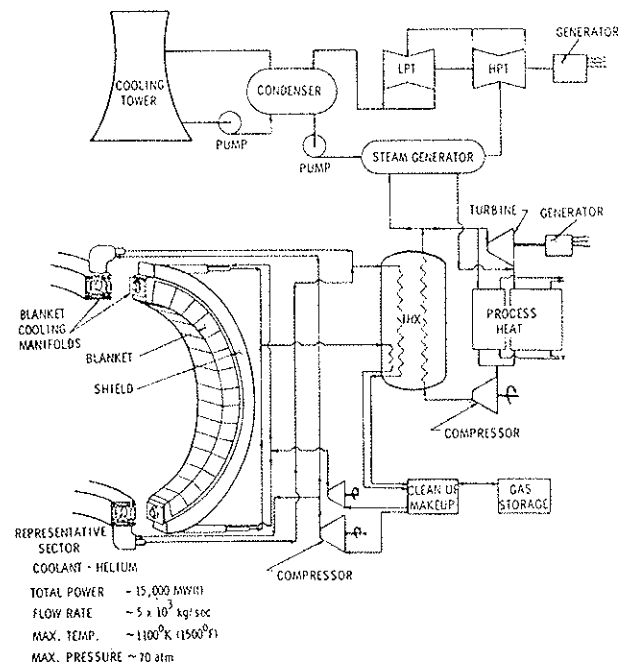

Fig. 10. Primary Coolant Systems: Blanket and Shield.

The configuration of the blanket coolant flow manifold which supports the modules and provides the mating flange and seal surface components of this quick-disconnect module system is shown in the cross-section in Fig. 8. This duct concept permits the high pressures in the rectangular duct sections to resist duct deformation. The double wall coolant inlet channels not only resist a tendency for the duct to deflect but permit duct design to the pressure load of the lower exit pressure of the inner chamber. Since the high pressure gas is cold, the structural containment system operates at low temperatures for maximum strength and minimum temperature variation during normal operation or transients.

This concept uses a non-structural inner hot liner which consists of a thin metal sleeve inside the structural wall. The sleeve contains the flowing hot gas and provides not only an additional interface, but a stagnant region to insulate the structure. Differences between cold cells and the inner liner for the hot gases, which would run almost at hot gas temperatures, require a high thermal expansion allowance for the sleeve; this is readily obtained because the sleeves can be sectioned. Preliminary structural analysis has indicated that the manifold with blanket modules attached is sufficiently stiff to permit transport and assembly to the reactor without undue deformation.

Based on the results of the scoping and parametric analyses as well as mechanical and neutronic considerations, a parallel flow configuration was selected as the reference blanket design. The coolant flow paths around and through the fueled region of the blanket are shown schematically in Fig. 11. A fuel pin diameter of 1.5 centimeters was selected to keep the fuel centerline temperature well below the melting points of the actinide oxides (the melting points are expected to be greater than 2250°C). The maximum pin clad surface temperature calculated is 1130°C. The important temperatures at various regions of the blanket are summarized in Fig. 11.

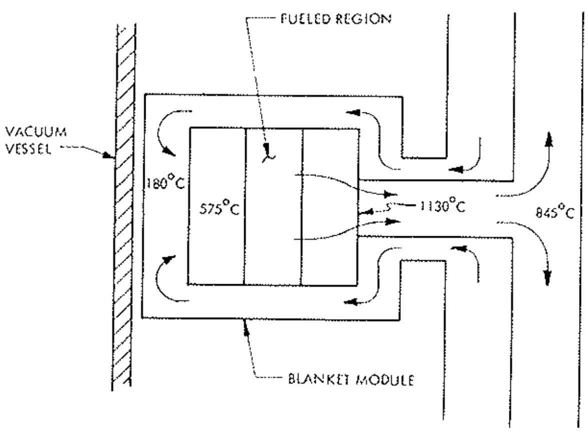

Fig. 11. Schematic Diagram of the Blanket Region Illustrating the Thermal-Hydraulic Mechanisms Involved.

ACTINIDE DEPLETION CHARACTERISTICS

CRITERIA FOR ASSESSING RADIOTOXIC HAZARDS

Several radiotoxic hazard indices have been commonly used in connection with nuclear safety and nuclear health analyses. These are the ingestion hazard index, the inhalation hazard index, and the radiation dose rate. The first two are defined as the volume of water and air, respectively, required to dilute a given amount of radionuclide to the ingestible and inhalable levels established by the maximum permissible concentration (MPC) or the radioactive concentration guide (RCG). Early studies on nuclear waste management problems have generally used the ingestion hazard as the index for evaluating relative toxicities, primarily because ingestion is the most relevant hazard over the long term. However, it should be pointed out that the absolute magnitudes of the ingestion hazard index have little practical meaning when taken out of the context of specific safety analyses that involve waste release mechanisms, including considerations such as specific quantities of the radioactive materials, dilution of the waste within a geologic formation, the leach rates of solidified waste forms, and the absorption of radionuclides in the soil. The waste release mechanisms that must be considered should include the probabilities of waste release under various postulated failure events that cause the wastes to be dispersed through the ecosystem.

Without consideration of these mechanisms, computation of the absolute value of the ingestion hazard becomes somewhat meaningless as pointed out by Claiborne[16] who showed that, for one metric ton of fuel, the dilution of actinides to satisfy the MPC criteria would require "a volume of water equal to the yearly flow of the Mississippi River into the Gulf of Mexico". A relative hazard index was therefore developed for use in the actinide burner study. This index consists of the ratio of the ingestion hazard of a given amount of high level nuclear wastes to the ingestion hazard of the amount of naturally-occurring parent uranium ore (assumed in this case to be carnotite, containing 0.2% uranium) from which the wastes were produced. Formulation of a relative hazard index on this basis is considered to be a more realistic measure of the effects of high level wastes since it depicts the change in hazard potential from that of naturally occurring radioactive substances which existed on earth before primitive life began.

The various processes and the mass balances for the fuel cycle are illustrated in Fig. 12. The ratio of actinide mass to that of the uranium ore was obtained by an overall mass balance. It is significant to note in the figure that many radioactive daughter products associated with the actinides are already present in the ore and in the ore tailings. The mass balance was normalized to one metric ton of enriched uranium feed to the LWR and based on data given in Ref. 17. The enrichment tailings and plutonium from reprocessing are currently being accumulated in storage, pending potential future use in heavy water moderated reactors (enrichment tailings) and in plutonium recycled LWR's, respectively. The uranium from the reprocessing plant is assumed to be recycled as shown. The hazards from enrichment tailings and reprocessed plutonium were not included in the total hazard because these streams were assumed to be recyclable in a totally integrated nuclear power economy of the not too distant future. Even using the normalized hazard index, however, one cannot merely evaluate the change in the present hazard index as a criterion for actinide burning performance because an improvement in the present ingestion hazard does not necessarily result in am improvement in the long term hazard, as was demonstrated by Claiborne and by the résults of this study.

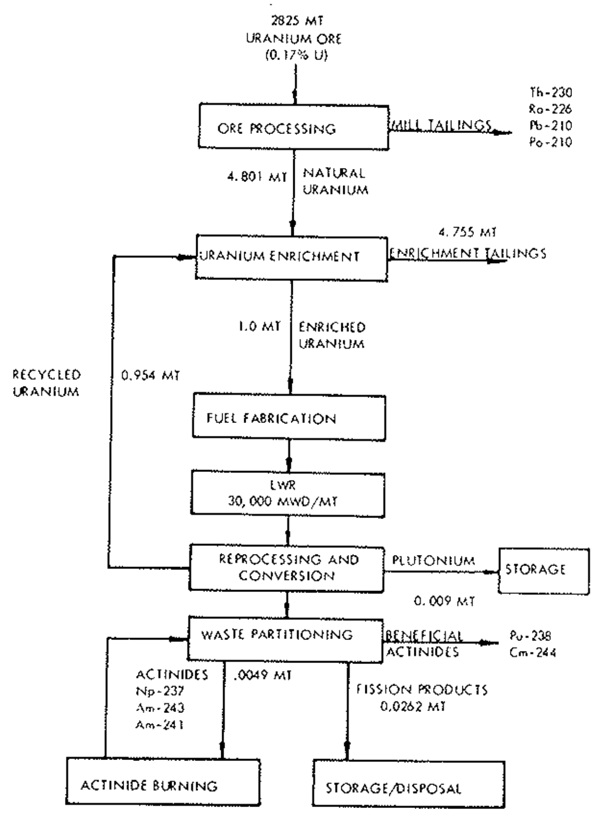

Fig. 12. Typical LWR Fuel Cycle and Nuclear Waste Management with Actinide Burner.

PRINCIPAL DECAY CHAINS OF HEAVY ELEMENTS

The primary radionuclides in the actinide wastes are Np-237, Am-243, Am-241 and Cm-244. The principal decay chains for the actinides are summarized in Figs. 13-16. Uranium 234, 235 238, and Th-232 are all found in nature within natural uranium and thorium ores; consequently, the decay phenomena beyond these elements are naturally occurring, while the higher mass isotopes (actinides) are produced by neutron capture transmutations in either fission reactors or in actinide burners. The chain starting with Cm-244 contains Th-232 as a daughter product. This element has the longest half life (1.4×10^{10} yrs.) among all the daughter products and the half life is significantly greater than those of the other daughter products. For these reasons, Th-232 and its daughter products are found in equilibrium wherever Th-232 is found. The chain starting from Cm-243/Am-243 has U-235 as a daughter product. U-235 also has an extremely long half life - 7×10^8 years. For this reason, its daughter products are also found in equilibrium with it. Because of these extremely long half lives, the radiotoxic hazards of these two chains are relatively minor since the radiotoxic hazards (ingestion or inhalation) are inversely proportional to the half life and inversely proportional to the RCG (or maximum permissible concentration, MPC). It should be noted that the RCG's of the principal radioisotopes are not significantly different when compared with the differences in their half lives.

Although U-233 is found in nature, it exists in trace quantities when compared with U-235 and U-238. Consequently, from an overall nuclear hazard standpoint, the chain containing Np-237 (daughter product U-233) may be considered man-made. Np-237 is a daughter product of Cm-245 or Am-241. It has a half life of some two million years and has highly toxic daughter products in Th-229 and Ra-225. For these reasons, radiotoxic hazards (ingestion and inhalation hazards) are expected to become important at relatively long times ($\sim 10^6$ years) following decay. In the Cm-242 chain, Pu-238 and U-234 are daughter products followed by other highly toxic daughter products, Th-230, Ra-226, Pb-210 and Po-210. U-234 has a half life of 2.4×10^5 years. Consequently, ingestion and inhalation hazards are expected to become important at times on the order of 10^5 to 10^6 years following decay of either Cm-244 or Pu-238. However, it

NUCLIDE	HALF-LIFE	
Cm-244	17.9	y
Pu-240	6540	y
U-236	2.3×10^{7}	y
*Th-232	1.4×10^{10}	y
Ra-228	5.75	y
Ac-228	6.1	h
Th-228	1.9	y
Ra-224	3.6	d
Rn-220	55	s
Po-216	150	ms
Pb-212	10.6	h
Bi-212	60.6	m
Po-212	45	s
Pb-208	Stable	

* Naturally occurring

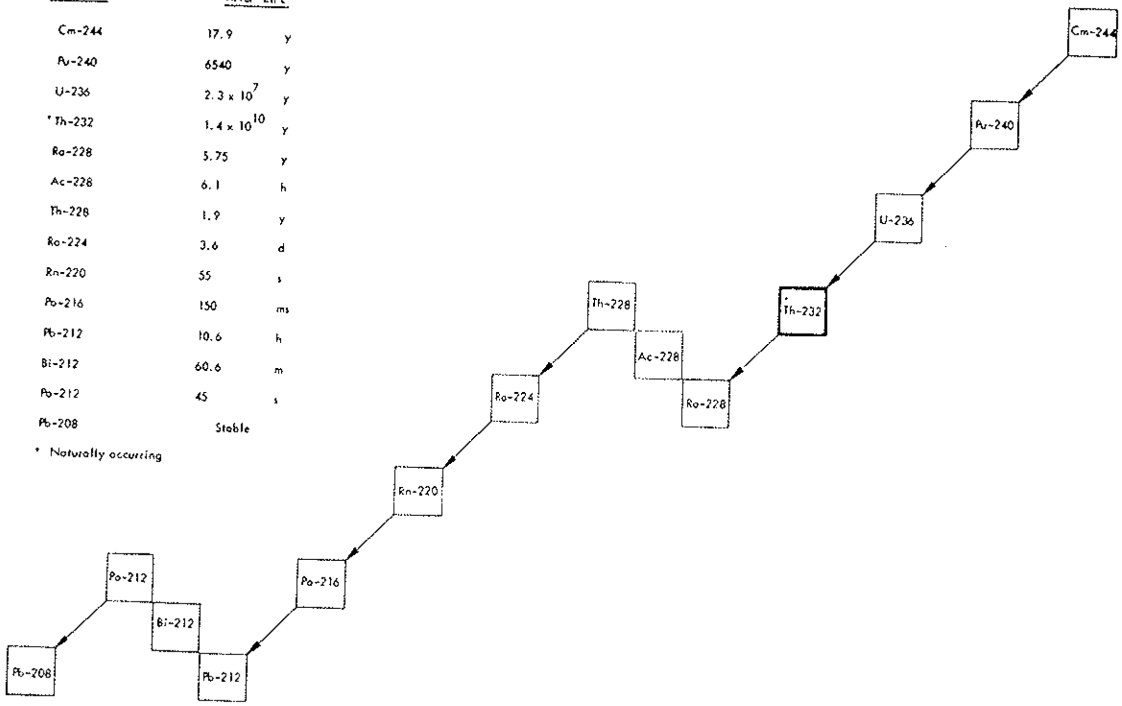

Fig. 13. Decay Chain Starting with Cm-244

NUCLIDE	HALF-LIFE	
Cm-243	28.0	y
Am-243	7370	y
Np-239	2.35	d
Pu-239	2.44×10^{4}	y
*U-235	7.04×10^{8}	y
Th-231	25.5	h
Pa-231	3.25×10^{4}	y
Ac-227	21.8	y
Th-227	18.7	d
Ra-223	11.4	d
Rn-219	3.96	s
Po-215	1.8	ms
Pb-211	36.1	m
Bi-211	2.1	m
Tl-207	4.8	m
Pb-207	Stable	

* Naturally occurring

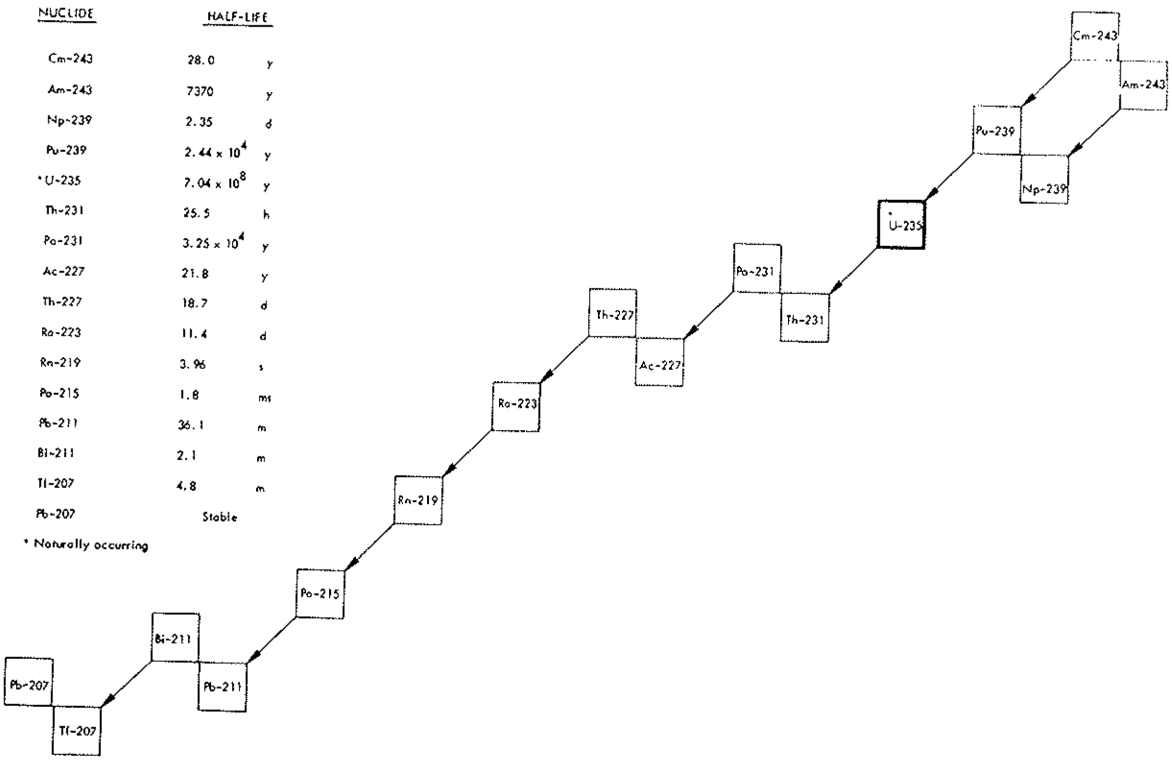

Fig. 14. Decay Chain Starting with Am-243/Cm-243

NUCLIDE	HALF-LIFE	
Cm-245	8.5×10^3	y
Pu-241	15	y
Am-241	433	y
Np-237	2.14×10^6	y
Pa-233	27	d
U-233	1.58×10^5	y
Th-229	7340	y
Ra-225	14.8	d
Ac-225	10	d
Fr-221	4.8	m
At-217	32	ms
Bi-213	4.6	m
Po-213	4	μs
Pb-209	3.3	h
Bi-209	Stable	

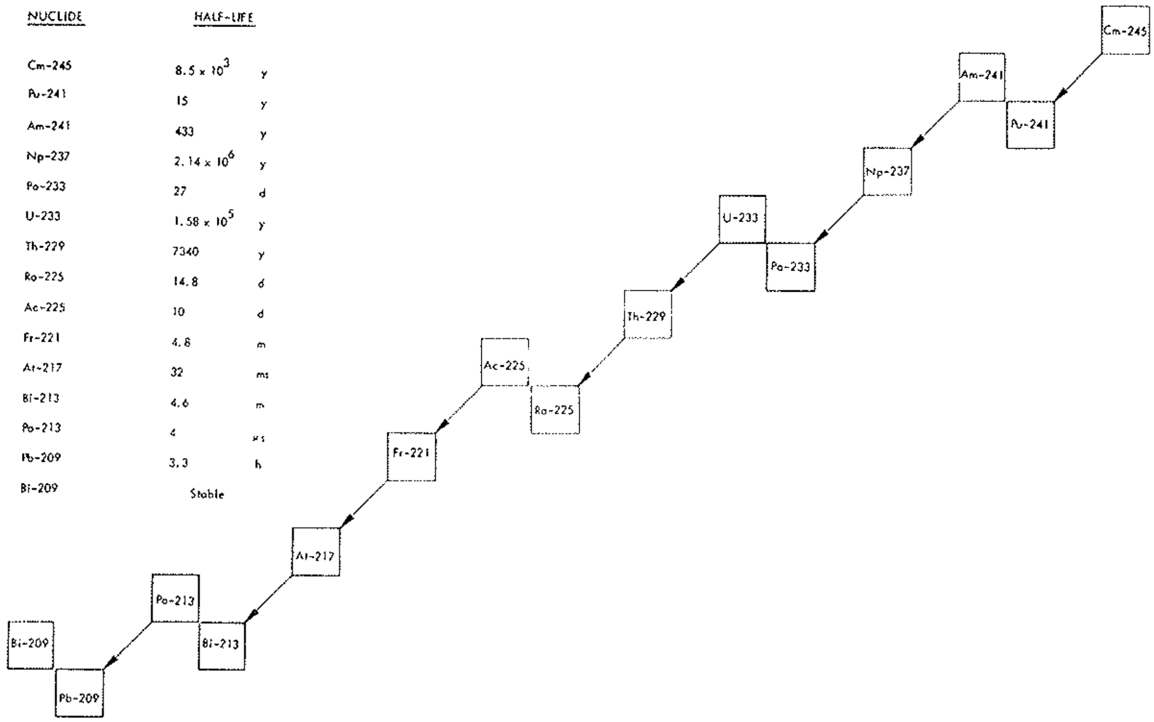

Fig. 15. Decay Chain Starting with Cm-245

NUCLIDE	HALF-LIFE	
Cm-242	163	d
Pu-238	87.8	y
U-234	2.44×10^5	y
Th-230	7.7×10^4	y
Ra-226	1600	y
Rn-222	3.8	d
Po-218	3.05	m
At-218	2	s
Bi-214	19.8	m
Po-214	164	μs
Pb-210	22.3	y
Bi-210	5.0	d
Po-210	138	d
Pb-206	Stable	

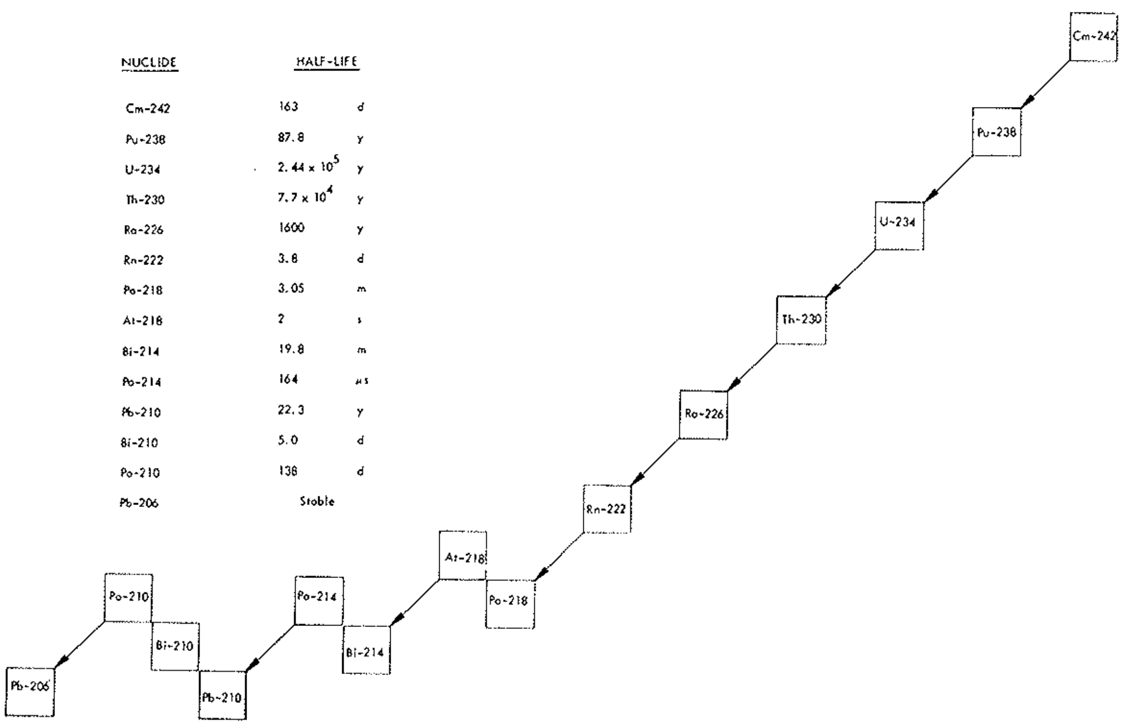

Fig. 16. Decay Chain Starting with Cm-242

should be noted that, since U-234 and all its daughter products are in equilibrium with U-238, these toxic elements already exist in nature.

The above analysis of the decay chains suggests that meaningful evaluations of the effects of actinide burning must involve a study of the long-term effects on radiotoxicity. For these reasons, one must follow the decay chains out for ~10^6 years in order to see the effects of radioactive daughter products on the potential ingestion hazards.

ACTINIDE DEPLETION AND RADIOTOXIC HAZARDS

The evaluation of the effect of neutron wall loading on actinide depletions showed that the principal actinides from fission reactors can be depleted effectively using fusion neutrons. This is illustrated in Fig. 17 for the reference case. It can be seen that the rates of depletion for Np-237, Am-243 and Am-241 are nearly exponential; consequently, a convenient means for comparing the rates of their depletion with the rates of natural decay is to define an "effective half life" as the half life of the isotope over the period of time the actinide is irradiated in the actinide burner. The effective half lives of the three main actinides are compared with their natural half lives in Table 3. The significant depletions of these actinides when irradiated by fusion neutrons are evident by the several orders of magnitude reductions in their effective half lives.

The appreciable depletions of the actinide isotopes initially placed in the blanket lattice occur mainly by transmutation (neutron captures), however, even for the very hard neutron spectrum of the reference actinide burner blanket. As a consequence, significant amounts of Pu-238, Cm-244 and Cm-242 are produced initially but are eventually depleted with extended irradiation, as seen in Fig. 17. The net effect of the production

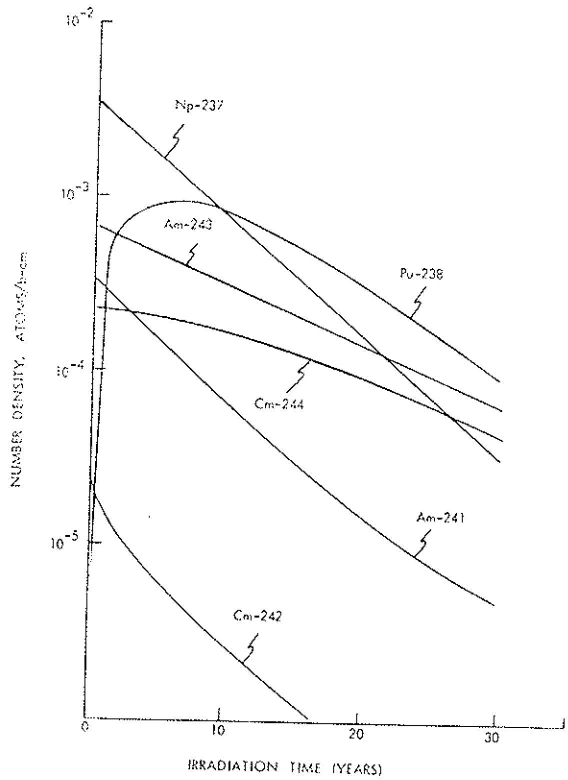

Fig. 17. Actinide Depletions and Transmutations During Irradiation in Reference Actinide Burner (Fast Flux), 1.15 MW/m^2 Neutron Wall Loading

Table 3. Comparison of Effective and Natural Half Lives for Actinide Depletion and Decay

Actinides	Natural Half Life, Years	Effective Half Life, Years — Tokamak Driven, Fusion Actinide Burner: 1.15 MW/m^2 Neutron Wall Loading	5 MW/m^2 Neutron Wall Loading	10 MW/m^2 Neutron Wall Loading	Thermal Flux, PWR Burner
Np-237	2,140,000	4.45	2.40	1.50	3.15
Am-243	7,354	8.90	4.70	2.4	*
Am-241	456	4.20	1.25	0.70	*

*The depletion does not follow an exponential function.

of Pu-238 and Cm-244 is a significant increase in the hazard index over the first few years of irradiation. This results from the relatively shorter half lives of Cm-242, Cm-244 and Pu-238 when compared to the other actinides. Since successive neutron reactions (e.g., transmutation followed by fission) are required to convert the majority of the actinides to suitably short lived fission products, the total fluence necessary for effective actinide depletion is quite high. Thus, the reference case utilizing a fusion neutron wall loading of 1.15MW/m^2 is only marginally attractive in terms of elimination of the long term hazards due to actinides as indicated by the relative ingestion hazard histories summarized in Fig. 18. In fact, irradiation for only 10 years produces a significant increase in hazard index for the reasons discussed above. In particular, it should be noted that the ingestion hazard tends to increase after 10^5 years to a maximum and then decrease with further decay. This phenomenon is due to the appearance of highly toxic daughter products mentioned earlier. Many of these daughter products as well as Np-237 do not have their MPC's listed in the Code of Federal Regulations (10 CFR 20). As a consequence, the default values were used to compute their contributions to the ingestion hazard. The default values are 3×10^{-6} Ci/m^3 for β-decay nuclides with half-lives greater than 2 hr. and 3×10^{-8} Ci/m^3 for alpha-decay or spontaneous fission nuclides. These default values are admittedly conservative and this conservatism may account for the relatively high ingestion hazards of the actinides in general and for the increases in ingestion hazards over the long term in particular.

As shown in Fig. 19, the effect of neutron wall loadings on relative ingestion hazards shows that 5 MW/m^2 of neutron wall loading will reduce the ingestion hazards by one order of magnitude when compared to natural decay, while 10 MW/m^2 of neutron wall loadings can reduce the long term ingestion hazards by 3 orders of magnitude. For the latter case, 10 MW/m^2 provides neutron fluxes ~10^{15} n/cm^2-sec at very high energy

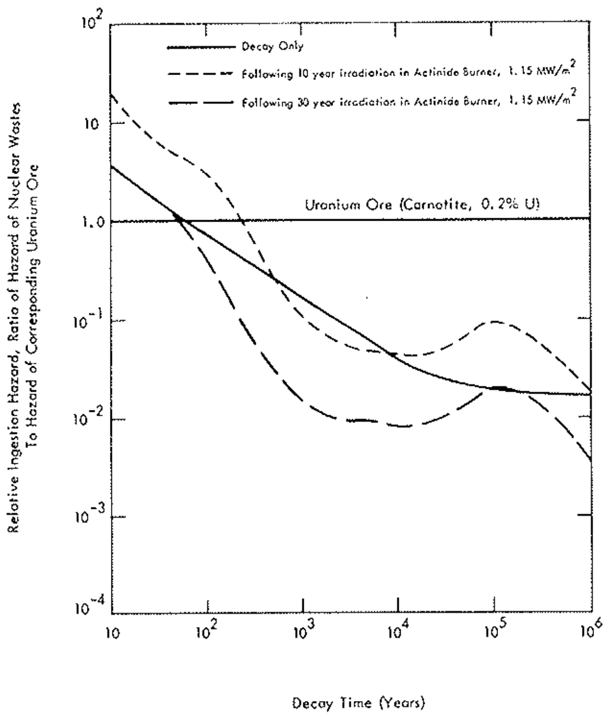

Fig. 18. Comparison of Relative Ingestion Hazards from Actinides as Functions of Decay Time. Bases: One Metric Ton of Uranium Fuel to an LWR.

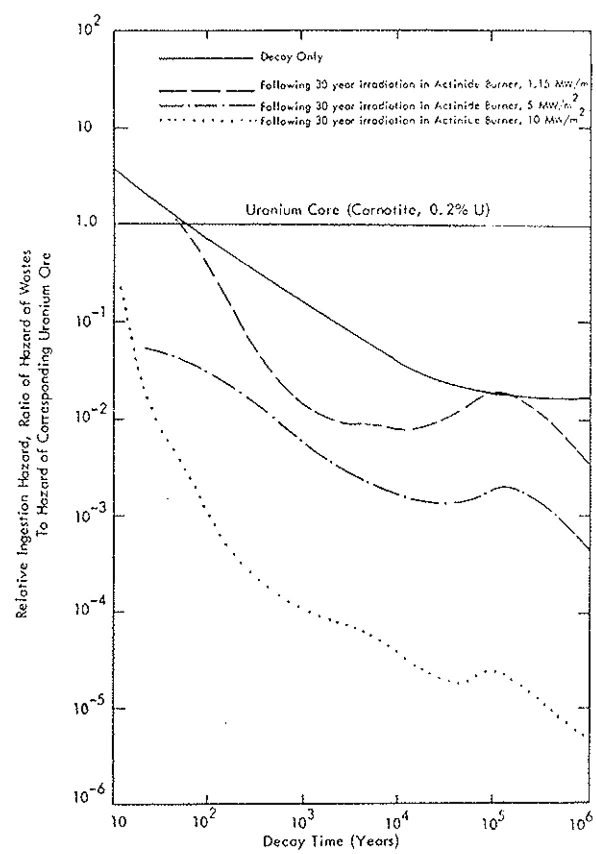

Fig. 19. Comparison of Relative Ingestion Hazards from Actinides as Functions of Decay Time. Bases: One Metric Ton of Uranium Fuel to an LWR.

(10-15 MeV). Enhancement of very high energy neutron flux favors fission rather than transmutation of actinides. For example, the fission cross-section for Np-237 is less than 0.1 barn at 0.1 MeV but increases to ~2.5 barns in the 10-15 MeV energy range while the absorption cross-section for Np-237 is a few tenths of a barn over the energy range 0.1 - 15 MeV. This fact, coupled with the increased value of ν (neutron release per fission) for very high energy neutrons, gives a significant increase in blanket neutron multiplication for a given amount of fuel loading. Therefore, a substantial reduction in actinide loading density (a factor of ~4) can be made while retaining the same blanket power density. Since the power density, P, is held constant in these calculations, the following relationships apply:

$$\left.\begin{aligned} P_{10} &= P_1 \\ N_{10}(\sigma\phi)_{10} &= N_1(\sigma\phi)_1 \\ (\sigma\phi)_{10} &= \frac{N_1}{N_{10}}(\sigma\phi)_1 \end{aligned}\right\} \qquad (1)$$

where N is the number density of the actinide, σ is the cross-section, ϕ is the flux, and the subscripts refer to the 1 MW/m^2 or 10 MW/m^2 case, respectively.

Also, the actinide depletion rate can be expressed as:

$$\left.\begin{aligned} \text{depletion rate} = \frac{d\,(N/N_0)_{10}}{dt} &= -(\sigma\phi)_{10} \\ \frac{d\,(N/N_0)_1}{dt} &= -(\sigma\phi)_1 \end{aligned}\right\} \qquad (2)$$

For a constant power density, the 10 MW/m^2 case requires an initial number density a factor of four less than the 1 MW/m^2 case because of the much higher flux level. Consequently, $(\sigma\phi)_{10}$ is 4 times $(\sigma\phi)_1$ (equation 1). Therefore, the depletion rate (equation 2) for the 10 MW/m^2 case is significantly higher than for the 1 MW/m^2 case resulting in a decrease in the effective half-lives of all the actinides. For example, the effective half-life of Np-237 decreases from 4.5 years to 1.5 years and that of Pu-238 from ~7 years to 2 years for the 1 MW/m^2 case and 10 MW/m^2 case, respectively. Since the irradiation period is 30 years (long compared to the effective half-life), the exponential nature of the decay results in a factor of 10^3 - 10^4 decrease in the residual Np-237 or Pu-238 for the two cases. The hazard at long times (10^5 - 10^6 years) is directly proportional to the residual Np-237 and Pu-238 at the end of the irradiation. Therefore, the 10 MW/m^2 irradiation reduces the long term hazards by 3-4 orders of magnitude over the 1 MW/m^2 irradiation. The ultimate capability of a beam-driven tokamak appears to yield a wall loading on the order of 4 to 5 MW/m^2 if one can achieve an elongated plasma (b/a = 3) and if close to 1000 MW of neutral beam power can be injected into the plasma. At present, laser fusion appears to be the only current identifiable means to attain a 10 MW/m^2 neutron wall loading.

As previously mentioned, examination of the ratio of fission to transmutation for each actinide isotope showed that transmutation is the dominant mechanism for depletion of the actinides originally placed in the blanket lattice even with a very hard neutron spectrum. Thus, the original goal of promoting actinide depletion by fission has not been achieved with the reference fusion neutron wall loading of 1.15 MW/m^2. For comparison purposes, actinide depletion calculations were performed utilizing a neutron spectrum representative of a PWR. While the ratio of fission to transmutation is smaller for the PWR spectrum, cross-sections for both reactions are sufficiently greater that somewhat more favorable actinide depletion characteristics were obtained. Therefore, fusion neutrons do not appear to have a distinct neutronic advantage over recycle in fission reactors until fusion neutral wall loadings approaching 10 MW/m^2 are considered.

The preceding discussion on actinide depletion is based on the premise that all actinide isotopes that contribute to the hazard index are undesirable nuclides. On the other hand, Pu-238, Cm-244 and Cm-242 have identifiable commercial applications as beneficial radioisotopes. Pu-238, for example, is useful for portable and remote power sources that include pacemakers, artificial hearts, and space and terrestrial power stations. The demand for Pu-238 for commercial applications can be readily estimated. Although large numbers of pacemakers will be nuclear-fueled (currently there are 2000 implanted nuclear pacemakers world wide), the actual amount of Pu-238 required is small (~60 kg by the year 2000). By far the application requiring the largest amount of Pu-238 is the power source for artificial hearts. The projected demand for Pu-238 for the year 2000 for various applications could amount to 335 metric tons (335,000 kg). It should be pointed out that the cost of Pu-238 may be substantially reduced from its current price of $750/g by producing large amounts in actinide burners. The projected supply of Pu-238 from actinide waste accumulations is only 50,000 kg. Consequently, there would be a shortfall of some 300,000 kg by the year 2000. Cm-244 has an appreciably shorter natural half life (18 years) compared with that of Pu-238. As a result, it has a higher power density. The higher power density is frequently useful in space and remote terrestrial power systems, particularly systems utilizing thermo-electric generators.

The principal deterrents to the use of these nuclides to date have been their unavailability and high cost. The actinide burner may be a prolific source of these radioisotopes produced by transmutation of actinides. On the other hand, Pu-238 is the principal contributor to increases in the hazard indices during irradiation and its daughter product Ra-226 is responsible for the secondary peak in the hazard index shown in Figure 18 at 10^5 years. Thus, these radioisotopes are a clear example where a value judgment is needed to weight the beneficial and potentially detrimental attributes of nuclear materials.

CONCLUSIONS

In the evaluation of the preliminary conceptual design of the actinide burner, some extremely challenging problem areas were encountered. The fusion driver, in particular, posed some of the principal design problems because of the necessity for advancing the state-of-the-art in plasma engineering and technology required for long pulse, high duty cycle operation. This type of operation necessitates the use of superconducting magnets, a divertor system for control of impurities and particles lost from the plasma, a highly effective neutralizer system to remove particles swept into the divertor, vacuum and neutral beam systems capable of sustained operation, and a viable means of tritium injection. A double null poloidal divertor was selected with the separatrix on the inner major radius side of the plasma. This divertor configuration was combined with a slightly elongated plasma to help compensate for the limitations in field-on-axis. A novel lithium coated nested chevron concept was developed to provide an effective means of removing particles from the divertor, thus minimizing external vacuum system pumping requirements. While this particle collection concept appears to be a viable approach, the need to remove and reprocess tritium from liquid lithium has led to a very large total tritium inventory.

Actinides are produced by transmutation of fuel in fission reactors and represent the longest-lived components in high level radioactive wastes. Results of this design study of a fusion-driven actinide burner with a fusion neutron wall loading of ~1 MW/m^2 indicate that the principal actinides can be depleted with effective half-lives (during irradiation) that are orders of magnitude shorter than those for natural decay. The accelerated depletion of the actinides isotopes initially placed in the blanket lattice occurs primarily by transmutation (neutron capture) to form other heavy nuclides, even for the very hard neutron spectrum of the reference actinide burner blanket. The net effect of this transmutation is actually a significant increase in hazard index due to the actinides over the first few years of irradiation. Since successive neutron reactions (e.g., transmutation followed by fission) are required to convert the majority of the actinides to suitably short-lived fission products, the total fluence necessary for effective actinide depletion is quite high. In contrast to results of earlier neutronic studies, an actinide burner with a fusion neutron wall loading of 1 MW/m^2 and with realistic provision for structure between the plasma and blanket fuel lattice is only marginally attractive in eliminating long term radiotoxic hazards due to the actinides. Fusion neutrons do not appear to be distinctly advantageous in effecting a major reduction (several order of magnitude) in the total long term radiotoxic hazard due to the actinides until wall loadings approaching 10 MW/m^2 are considered.

Although major contributors to the hazard index, some of the isotopes produced by transmutation have commercial applications as beneficial radioisotopes for use in protable and remote power sources. The actinide burner may be a prolific source of these radioisotopes produced by the transmutation of actinides. Depletion of the actinides also produces a significant amount of power. Although the capital cost per kW appears to be significantly higher than that for current fission power plants, it may be acceptable as an incremental cost to dispose o the actinides. A meaningful cost benefit study to assess the merits of actinide depletion will require further work to define processes and costs to partition the actinides from the fission products and to separate the beneficial radio-isotopes from depleted actinides.

Inadequacies in the present concepts of hazard indices prompted the adoption of a revised hazard index in which the ingestion hazard of a given amount of high level nuclear wastes is normalized to the ingestion hazard of the amount of naturally-occurring parent uranium ore from which the wastes were produced. Formulation of a relative hazard index on this basis is considered to be a more realistic measure of the effects of high level wastes since it depicts the change in hazard potential from that of naturally occurring radioactive substances which existed on earth before primitive life began. In applying any type of hazard index to an assessment of long-lived radioactive wastes, it is essential to follow the decay chain out to $\sim 10^6$ years to include the effects of radic active daughter products on radiotoxic hazards.

Comparison of the total hazard from actinide wastes to those of the parent uranium ore indicates that the contribution from the actinides is no longer dominant after several hundred years of natural decay. Furthermore, recent studies of proposed disposal by burial in suitable geological formations at a depth of $\sim$600 meters indicate that the parthway to the ecosystem is much more torturous than that for naturally occurring radioactive materials found in the earth's crust. Conservative projections of waste transport into the ecosystem based on observations of naturally occurring radioactive nuclides indicate that the probability of fatalities during a million years of storage is small. Based on these considerations it would appear that the issue of long term hazards due to the actinides requires serious re-examination to develop a balanced perspective.

Despite the considerations outlined in this discussion, the management of high level radioactive wastes is a genuine societal concern. Therefore, it is deemed advisable to discuss these views on waste management in an appropriate public forum to insure that potential hazards from the wastes, and from the actinides in particular, are assessed in proper perspective.

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USSR
BLANKET STUDIES

EXPERIMENTAL GAS-COOLED HYBRID BLANKET MODULE FOR A TOKAMAK DEMONSTRATION REACTOR*

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INTRODUCTION

Among the various areas in which thermonuclear devices are being developed, the area of Tokamak reactors has received the greatest development.

The economic estimates performed in various countries indicate that the specific capital expenditures on the construction of thermonuclear electric power plants apparently will be about twice as high as those for atomic electric power plants. The economic characteristics of thermonuclear electric power plants can be improved significantly by conversion to hybrid devices using fissionable material in the fusion reactor blanket.[1]

Within the scope of the Soviet program for practical development of thermonuclear power for peaceful purposes, provision has been made for the construction of a demonstration Tokamak thermonuclear reactor in 1983-1984 with the following basic parameters:[3]

- -- major radius of torus 5 m
- -- inside radius of discharge chamber along wall 2.1 m
- -- outside radius of blanket modulus with respect to shielding 3.1 m
- -- volume of discharge chamber 400 m^3
- -- surface area of discharge chamber wall 400 m^2
- -- thermonuclear neutron flux at time of pulse 1.25×10^{13} (neutrons/$cm^2 \cdot s$)
- -- pulse duration to 20 seconds
- -- number of pulses in D-T mixture 10^5

It is proposed that several series of plasma physics, neutron physics, and engineering-technological experiments be performed on the demonstration Tokamak thermonuclear reactor. The engineering-technological testing of the blanket modules, which will be represented by various concepts of thermonuclear power reactors, is to be started in 1985. The theoretical structural diagram of the demonstration Tokamak thermonuclear reactor will permit simultaneous testing of several structural designs of the modules (for example, from 4 to 8 types).

The demonstration Tokamak thermonuclear reactor will be the first Soviet thermonuclear reactor in which significant power will be obtained in the form of energy neutrons and the α-particles of the D-T reaction. Although the neutron flux with an energy of 14 MeV to the first wall of the demonstration Tokamak thermonuclear reactor is several times lower than in the designed thermonuclear power reactors, it is sufficient to study many engineering problems connected with the design and the construction of the blankets for these reactors.

The experimental program for the demonstration Tokamak thermonuclear reactor provides for the following:

- Study of the neutron-physical characteristics of the blanket modules with determination of the breeding factor of tritium and the production factor of plutonium, the energy multiplication factor in the blanket, the energy release distribution among the blanket zones, and so on;
- Study of the radiation resistance of the materials in the presence of fluxes to $2 \cdot 10^{20}$ neutrons/cm^2 in the presence of a significant proportion of neutrons with an energy of 14 MeV in the spectrum;

* Translated from the Russian By Addis Translations International, Portola Valley Ca. 94025.

- Heat engineering testing of the structural design of the blanket modulus under the real temperature conditions of the blankets of future thermonuclear devices;
- Development of methods of extracting tritium from lithium-containing materials and technological process measures for the prevention of tritium leaks from the blanket, extraction of the tritium at operating temperatures, and energy intensities;
- Study of the efficiency of the uranium-containing elements of the hybrid blanket and development of the operations for recharging them;
- Accumulation of experience in the operation and maintenance of the technological loops and equipment of the auxiliary systems, investigation of the degree of activity of the loops;
- Accumulation of experience with respect to the installation and remote dismantling of the blanket modules, determination of the transport operations rules, and the degree of radiation safety during transport operations;
- Testing the effectiveness of the general and local radiation shielding of the blanket module and the entire device as a whole;
- Comparison of the basic characteristics of the alternative versions of the blanket module.

During the course of the experiments, the possibility of obtaining electric power on industrial scales (to 100 kW) as a result of the fusion reaction must be demonstrated, with simultaneous production of significant quantities of tritium and plutonium.

The development presented in this report is one of the possible versions of the theoretical structural design for gas-cooled blanket of a hybrid thermonuclear reactor.

NEUTRON-PHYSICS CHARACTERISTICS OF THE BLANKET MODULE

CALCULATION PROCEDURE

All of the neutron-physical calculations were performed by the BLANK program, permitting us to obtain the spatial energy distribution of the neutrons in the one-dimensional configuration with an external source. The combination of the Monte Carlo method and the P_1 approximation is realized in the program. In the high-energy range ($E > 0.1$ MeV), the integral transport equation is solved by the Monte Carlo method using the 52-group library of nuclear data. In the energy range below 0.1 MeV, the equation is solved in the P_1-approximation with the 21-group library of nuclear data. The Monte Carlo method was used to solve the transport equation only in the first iteration. The subsequent iterations are performed in the P_1 approximation.

The distribution of the gamma-quanta was calculated by the Monte Carlo method in the one-dimensional configuration in the presence of internal sources of gamma radiation, considering the three processes of gamma radiation interaction with matter: the Compton effect, the photo-effect, and the steam formation. The group approach was used in the calculations. The width and the number of the energy groups were determined by the energy scales in which the cross sections of the initial elements are given. The initial data (the gamma radiation sources) were calculated by the BLANK program. From the point of view of their sources, the gamma-quanta were divided into three groups: gamma-quanta formed during the fission reaction, gamma-quanta formed during the (n, γ) reaction in the energy range below 0.1 MeV, and the gamma-quanta formed during inelastic processes. The last component was given in the form of the generation of gamma-quanta during hydrogen captures from the condition of equality of the total energy of the gamma-quanta formed to the neutron energy losses during inelastic collisions.

CALCULATION RESULTS

One of the basic requirements imposed on the blanket of a hybrid thermonuclear reactor is the high energy yield per fission (90-100 MeV). The energy of the fission reaction to 60% of completeness makes the basic contribution to the total energy released in the blanket module; therefore, the primary factor influencing the total energy release in the module is the amount of fissionable material. For the given volumetric content of the materials, the amount of fissionable material will be determined by the size of the zone in which this material is contained. It is obvious that with an increase in the size of the zone, the total energy release per fission will increase. However, the leakage of the neutrons through the outer boundary of the zone will decrease, which is explained by an increase in the absorption in the zone itself. This leads to a reduction in the number of neutrons which can be used for the production of tritium.

In Fig. 6, the material diagram of the blanket module chosen on the basis of various calculations is presented in which uranium carbide (UC) is used as the fissionable (breeding) material, and lithium aluminate ($LiAlO_2$) is used for the breeding of tritium. As the neutron-physical calculations have demonstrated, 75 MeV per fission are generated in the uranium zone, and the leakage through its outer boundary is 1.26 neutrons per fission.

This permits us to obtain a total energy yield on the order of 90 MeV per fission.

The location and sizes of zones 5, 7, and 9 were selected from the condition of maximum neutron absorption in zones 5 and 9 and reduction of the proportion of the neutrons reflected (returned) to zone 3. This explains the choice of the quite thick layer of graphite moderator which serves as the source of the neutrons, which are then absorbed by the lithium in zones 5 and 9. The reaction cross section $^6Li\ (n, \alpha)T$ is appreciably higher than the reaction cross section $^7Li(n, n'\alpha)T$; therefore, with an increase in the concentration of ^6Li in the lithium, the tritium breeding in the blanket module increases. As the various calculation versions demonstrated, the increase in the ^6Li concentration in the natural lithium permits an increase of 5-8 times in the tritium breeding factor.

On the basis of the above discussion, the version was selected (Fig. 6) which insures that the largest possible value of the tritium breeding factor will be obtained which does not result in significant structural complications, the necessity for the application of relatively scarce and expensive materials, or a significant decrease in the energy yield per fission. The spectral characteristics of this version calculated by the BLANK program are presented in Fig. 1 and 2. The energy release in the module materials is presented in Table 1, in which it is obvious that the primary deficiency of the proposed structural design consists in the nonuniformity of the energy release in the uranium zone. Thus, in the fuel elements located at the first wall, six times more energy is released than in the fuel elements located on the outer edge of the uranium zone.

The calculated tritium breeding factor in the proposed version is $K_T = 0.751$, the plutonium production factor $K_{Pu} = 1.492$, and the energy yield per fission $E = 39.8$ MeV.

CHARACTERISTICS OF THE TRITIUM AND PLUTONIUM PRODUCTION PROCESS

One of the physical problems of the experimental program is the investigation of the plutonium production process in the uranium zone and the tritium production in the zones containing lithium.

In connection with the small value of the irradiation integral (Fig. 3), only the determination of the total amount of plutonium produced and the investigation of the isotopic composition of the irradiated fuel elements with low burnups apparently are possible. The calculated plutonium production distribution in the uranium zone is presented in Fig. 4. As expected, the plutonium is formed non-uniformly in the uranium zone. From Fig. 4, it is evident that in the fuel elements located at the first wall, twice as much ^{239}Pu is produced as in the fuel elements located near the outer boundary of the uranium zone.

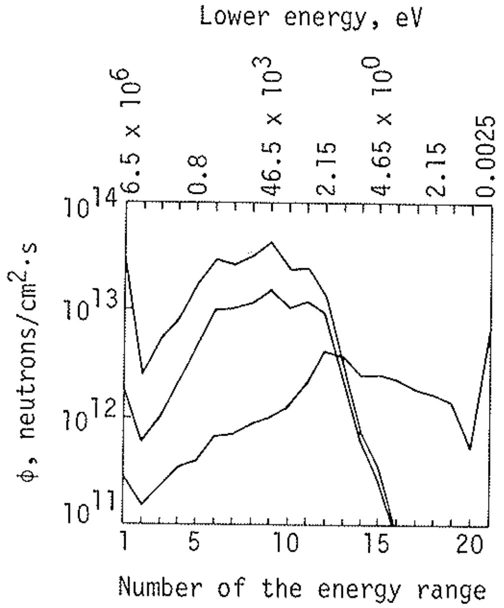

Fig. 1. Neutron spectra: (1) at the first wall, (2) uranium zone, (3) in the moderator.

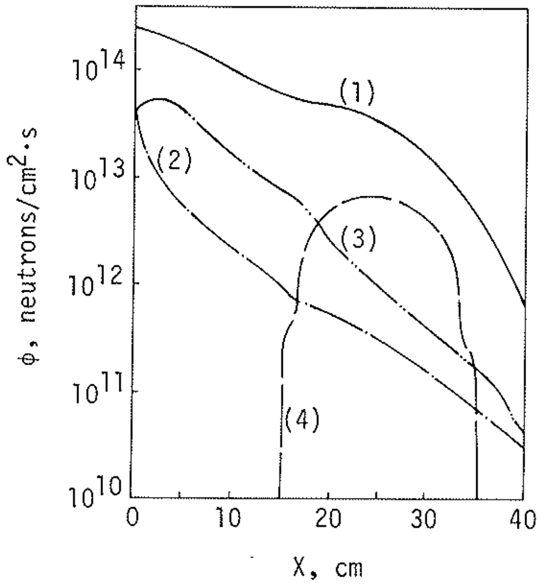

Fig. 2. Distribution of the neutron flux with respect to thickness of the blanket: (1) Total flux (2) Flux with an energy above 6.5 MeV (4) Thermal neutron flux

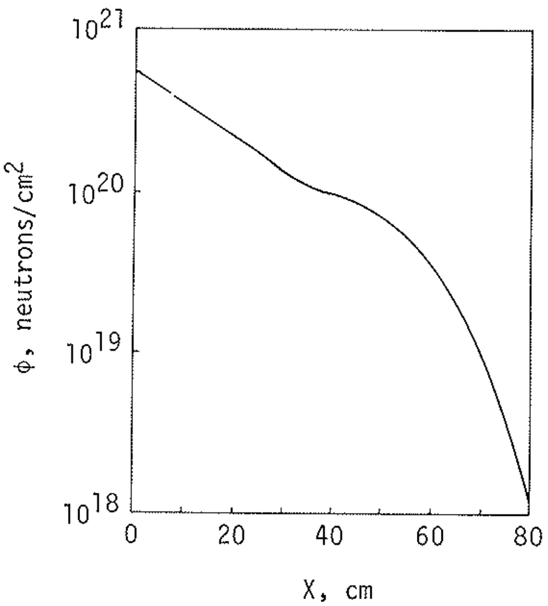

Fig. 3. Distribution of the irradiation interval over the blanket module.

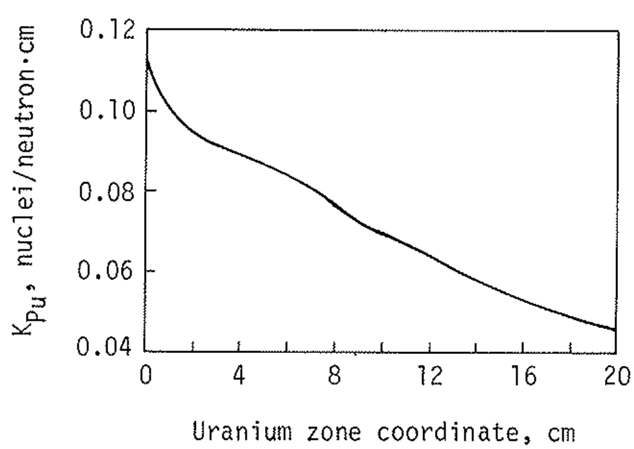

Fig. 4. ^{239}Pu production in the uranium zone.

Table 1. Energy release in the blanket module.

Intervals (cm)	Construction material: Energy release (MeV/neutron) E_{kv}	E_γ	E	W (W cm^3)	Blanket materials: Energy release (Mev/neutron) E_{kv}	E_γ	E_f	E	W (W cm^3)
0-0.5	0.524	0.613	1.137	4.547	Uranium carbide				
0.5-1.6	0.0786	0.143	0.222	4.444					
1.6-3.6	0.198	0.126	0.324	3.976	2.057	2.173	12.98	17.22	30.62
3.6-5.6	0.167	0.0880	0.255	3.089	1.305	1.487	11.22	14.01	24.92
5.6-7.6	0.145	0.0659	0.211	2.595	0.913	1.161	8.28	10.36	18.43
7.6-9.6	0.126	0.0521	0.178	2.171	0.755	0.903	6.43	8.08	14.38
9.6-11.0	0.109	0.0426	0.151	1.862	0.535	0.776	5.34	6.65	11.82
11.6-13.6	0.0964	0.0356	0.132	1.566	0.375	0.647	4.05	5.07	8.930
13.6-15.6	0.0827	0.0309	0.114	1.348	0.295	0.562	3.55	4.41	7.764
15.6.17.6	0.0704	0.0250	0.0954	1.125	0.272	0.457	2.96	3.69	6.499
17.6-19.6	0.0597	0.0229	0.0826	0.979	0.219	0.496	2.47	3.18	5.598
19.6-21.6	0.0500	0.0196	0.0696	0.821	0.203	0.354	2.16	2.72	4.786
21.6-22.9	0.142	0.0398	0.182	0.449	Lithium aluminate				
22.9-24.9	0.0126	0.0616	0.0742	0.332	0.585	0.0347		0.620	1.379
24.9-26.9	0.0100	0.0571	0.0671	0.301	0.539	0.0318		0.570	1.268
26.9-28.9	0.0081	0.0492	0.0573	0.257	0.521	0.0272		0.548	1.216
28.9-30.9	0.0067	0.0388	0.0455	0.204	0.549	0.0211		0.571	1.259
30.9-32.9	0.0057	0.0328	0.0385	0.172	0.993	0.0176		1.011	2.213
32.9-33.6	0.0028	0.0065	0.0094	0.187	Graphite				
33.6-40.6					0.0928	0.00043		0.465	1.110
40.6-47.6					0.0639	0.00034		0.0854	0.204
47.6-54.6					0.0415	0.00027		0.0465	0.112
54.6-61.6					0.0242	0.00021		0.0275	0.066
61.6-68.6					0.0139	0.00016		0.0151	0.037
68.9-69.0	0.00282	0.00061	0.00158	0.0215	Lithium aluminate				
69.0-71.0	0.00033	0.0019	0.0022	0.0325	0.465	0.00043		0.465	1.110
71.0-73.0	0.00030	0.0015	0.0018	0.0270	0.0850	0.00034		0.0854	0.204
73.0-75.0	0.00030	0.0012	0.0015	0.0240	0.0463	0.00027		0.0465	0.112
75.0-77.0	0.00025	0.00087	0.0011	0.0097	0.0273	0.00021		0.0275	0.056
77.0-79.0	0.00020	0.00071	0.0009	0.0079	0.0150	0.00016		0.0151	0.037
79.0-79.8	0.00005	0.00064	0.00072	0.0070					

This corresponds to a mean uranium burnup of 0.011% and to 0.016% burnup in the layer nearest the plasma, which is two orders below the burnup planned for the experimental thermonuclear electric power plant. In the central part of the uranium zone, a burnup depth on the order of 30 MWd/t is achieved, and in the layer located near the first wall, two or three times that figure is reached.

The activity of one spherical element after 10 days of decay will be about 0.09 curies, and after decaying for 100 days it will drop by about three times. Nevertheless, the residual activity of the fuel elements requires the use of the mechanisms for remote separation in the hot cells.

The calculated tritium production distribution in the lithium zones is presented in Fig. 5. From Fig. 6 it is evident that the maximum amount of tritium is formed in the fuel elements located on the outside of zone 5 and the inside of zone 9, i.e., in the layers adjacent to the moderator. This is explained by the fact that the basic contribution to tritium production is made by the thermal neutrons, and the maximum tritium production is in the vicinity of the maximum thermal neutron flux. In the first lithium zone (zone 5) the coefficient of nonuniformity of the tritium production is equal to 4, and in the second lithium zone it is 10. This corresponds to a mean ^6Li burnup of 0.03% for the first lithium zone, and 0.007% for the second lithium zone. The activity of the tritium formed in the first lithium zone is 6.82×10^{-2}, and 1.68×10^{-2} Ci/g in the second zone.

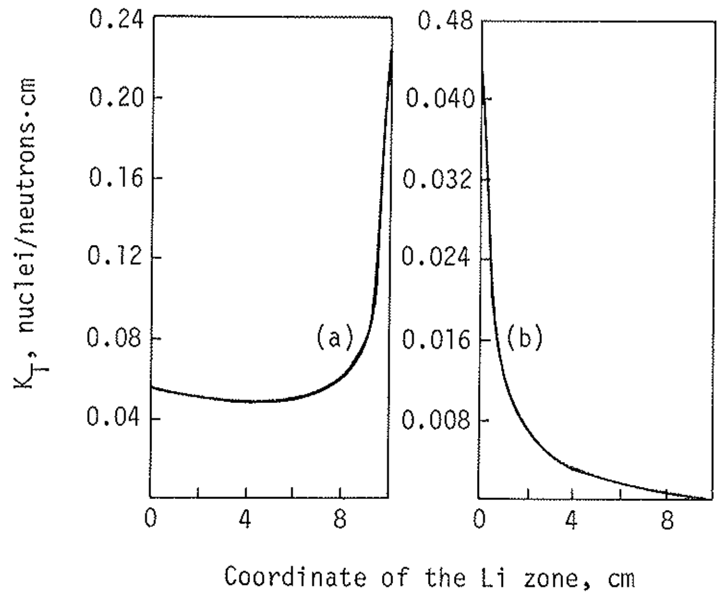

Fig. 5. Tritium production: (a) in the first Li zone (zone 5); (b) in the second Li zone (zone 9).

THERMAL HYDRAULIC CHARACTERISTICS OF THE BLANKET MODULE

The uranium zone of the blanket module is, in thermal respects, a highly stressed structural element. As a result of the significant nonuniformities of the heat release throughout the volume of the blanket, especially with respect to its height, the problem of the cooling of the structural element and the zone materials is highly complex.

Beginning with the admissible thermal operating regimes of the structural materials in the uranium and lithium zones and considering the number of arguments of a heat engineering nature, it was decided that the maximum temperature of the fuel elements in the lithium zone must be no more than 100°C, and for the uranium elements, no more than 800°C.

Considering the pulsed nature of the operation of the demonstration Tokamak thermonuclear reactor, approximate thermal calculations were made of the uranium, graphite, and lithium zones of the blanket module. A study was made of the pulsed heat capacity regime for the heating of the structural elements with subsequent cooling of the module zone (between pulses) as a result of the thermal conductivity of the structural element itself (without forced cooling). The calculations demonstrated that local temperatures close to 100°C are reached in the lithium elements after 10-20 pulses, and in the uranium elements local temperatures of about 800°C are reached only after 30 to 40 pulses. The feed regimes of the heat transfer agent for the various zones of the module were selected on the basis of the results of these calculations: 10-20 helium feed pulses with a flow rate of G = 0.2 kg/s in the cooling system of the lithium zones, and circulation in the cooling system of uranium zone with a flow rate of G = 0.3 kg/s.

The heating of the helium in the lithium zones reaches 50°C, and the heating of the helium in the graphite and uranium zones is 500°C. The helium pressure in the cooling systems was taken as 5-8 kg-force/cm^2.

STRUCTURAL DIAGRAM OF THE HYBRID BLANKET MODULE AND ITS LOCATION IN THE DEMONSTRATION TOKAMAK THERMONUCLEAR REACTOR

STRUCTURAL AND TECHNOLOGICAL PECULIARITIES OF THE HYBRID BLANKET MODULE

During the planning and design of the hybrid thermonuclear reactor, a number of specific design and process problems may arise. In particular, provision must be made for operative and technologically efficient maintenance of the reactor, including the blanket.

One can assume that the blankets of thermonuclear machines will be extremely bulky and heavy, and that the recharging of their active elements by the tradional methods used in present-day reactor technology will be laborious and time-consuming. Accordingly, it is expedient to select a structural form for the uranium and lithium fuel elements which will permit significant simplification of the recharging process and a reduction of its time cycle.

At the same time, the uranium elements of the gas-cooled thermonuclear power reactors must satisfy a number of specific technical requirements, including high burnup of the uranium at high temperature (a maximum fuel element cladding temperature of 1100°C). Requirements are imposed on the uranium elements with respect to strength, radiation resistance (of the basic material and the cladding), fission fragment yield, ratio of the number of uranium nuclei, and construction materials. Therefore the design and the technological development of these elements will be continued on the basis of the latest achievements in the field of high-temperature uranium alloys. As it appears now, in comparison with other types of elements, the spherical element containing uranium carbide clad with a metal jacket, of tungsten, for example, satisfies these requirements for the most part.

The lithium-containing elements based on ceramic compounds of the $LiAlO_2$ or Li_2O type in spherical form have also been selected.

Definite Structural difficulties also arise when organizing the cooling system for the blanket module.

PLACEMENT OF THE HYBRID BLANKET MODULE IN THE DEMONSTRATION TOKAMAK THERMONUCLEAR REACTOR

Beginning with the purpose of the blanket module and considering the class of neutron-physical and technical design problems solved by means of it, it was assumed that the surface of the module adjacent to the first wall of the discharge chamber of the demonstration Tokamak thermonuclear reactor must be no less than 5 m^2. The design developments of the demonstration Tokamak thermonuclear reactor have revealed the possibility of creating a hybrid gas-cooled blanket module of this size. The location and dimensions of the blanket hole in the demonstration Tokamak thermonuclear reactor are determined by the free space in the 3-m zone formed by the coils of the longitudinal field magnets, the coils of the toroidal field magnets, and the discharge chamber itself. The overall dimensions of the blanket hole are 1.8 x 2.8 x 1.0 metres, which consequently also determine the maximum dimensions of the blanket module itself. With these dimensions the module weighs 15 tons.

The general view of the hybrid blanket module and its attachment to the demonstration Tokamak thermonuclear reactor are illustrated in Fig. 6.

DESCRIPTION OF THE STRUCTURAL DIAGRAM OF THE BLANKET MODULE

The blanket module itself is a box structure with sealed separations into the basic zones: uranium, first lithium, graphite, second lithium, and biological shielding.

The zone dimensions of the blanket module are determined on the basis of the neutron-physical calculation.

It is proposed that the graphite zone, like uranium and lithium zones, be made up of spherical elements, whose manufacturing process does not present any difficulties in theory.

Beginning with the convenience of performing the installation and disassembly operations, the blanket module is divided into three autonomous units. It is proposed that one unit be installed in the center and two on the edges of the hole. The two edge units of the blanket module are located under two adjacent coils of the toroidal field magnets, and they are not withdrawn during the operation of the reactor. It is proposed that the central unit be withdrawn during the experiments using a special recharging device and that it be inspected when necessary.

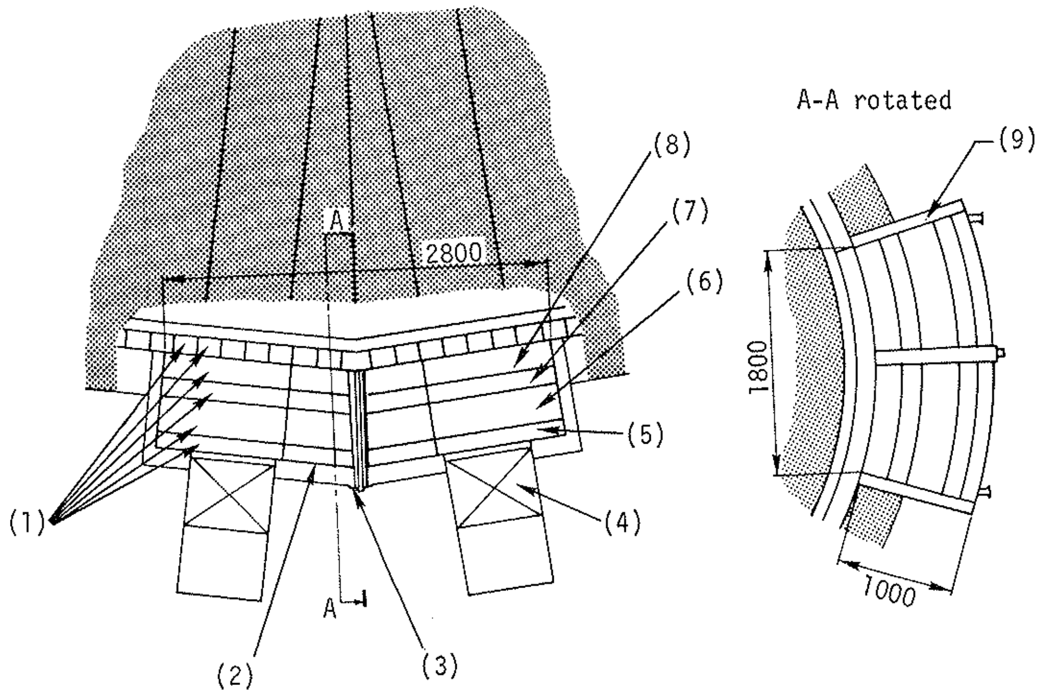

Fig. 6. Demonstration tokamak thermonuclear reactor blanket module: (1) steel, (2) diagnostic channel, (3) measuring channel, (4) toroidal field coil, (5) second Li zone, (6) graphite zone, (7) first Li zone, (8) uranium zone, (9) biological shielding.

Measuring channels are installed to measure the neutron-physical and the heat engineering parameters in the blanket. Each measuring channel has a set of small fission chambers and heat engineering monitoring sensors arranged along the height of the module zones.

BIOLOGICAL SHIELDING OF THE BLANKET MODULE

The biological shielding installed in the blanket module and outside it, between the discharge chamber and the magnetic coil windings, insures normal servicing of the module after the series of pulses in the D-T plasma. Whereas the biological shielding of the reactor installed outside the blanket module basically has the purpose of attenuating the activation gamma radiation, the shielding of the blanket module must also attenuate the residual radiation from the fission fragments of the uranium.

The preliminary instruments demonstrated that the shielding of the blanket module from gamma radiation may be made of lead.

The lithium and graphite zones of the modules serve as shielding from the neutrons leaving the plasma volume (14.1 MeV) and the neutrons with a fission spectrum leaving the uranium zone of the module. The efficiency of the neutron attenuation by these materials is equivalent to the efficiency of the basic shielding located on the surface of the toroidal chamber.

The magnitude of the dose rate on the outside surface of the shielding (equal to 2.5 μR/s, which does not exceed the dosage beyond the biological shielding installed outside the blanket module 10 days after completion of the series of 100 pulses) is taken as the criterion for selecting the blanket module shielding. A study was made of the following gamma radiation sources: fission fragments in the uranium zone of the module, and the sodium, ion, cobalt, nickel, manganese, and chromium isotopes formed upon interaction of fast and thermal neutrons with the construction materials of the module.

The density distributions of the fissions in the uranium zone of the module and the fast and thermal neutron fluxes calculated by the one-dimensional program were used when calculating the intensity of the gamma radiators.

The intensity of the gamma radiation of the fission products was calculated from the condition that a unit current of thermonuclear neutrons with an energy of 14.1 MeV on the inside boundary of the module corresponds to an energy release in the blanket of 90 MeV for a rated neutron current of 1.25×10^{13} neutrons/$(cm^2\ s)$ and a surface area of the blanket module turned toward the neutron source equal to $5 \times 10^4\ cm^2$. Ten days after the series of 100 pulses the total gamma quantum fluxes, considering the density distribution of the fissions with respect to thickness of the module, is 4.7×10^7 gamma quanta/$(cm^2\ s)$ on the outside surface and 2.0×10^8 gamma quanta/$(cm^2\ s)$ on the inside surface of the uranium zone of of the module. The attenuation of the emission of fission fragments in the materials of the lithium and graphite zones of the module leads to the fact that on the outside surface of the module the dose rate of the fragment radiation does not exceed 75 μR/s. When calculating the induced activity of the materials of the blanket module, a study was made of the following isotopes: ^{51}Cr ($T_{1/2}$ = 27.8 days), ^{54}Mn (312 days), ^{57}Ni (36 hours), ^{57}Co (272 days), ^{58}Co (71 days), ^{60}Co (5.24 years), ^{59}Fe (45 days), ^{24}Na(15 hours). The isotopes with a half life of less than 14 hours, the beta-radiator isotopes, and also the isotopes emitting soft gamma radiation were excluded from the investigation. The threshold reaction cross sections were averaged considering the neutron spectra in each zone of the module.

The calculation results demonstrated that for the selected composition of the materials in the blanket module, the activation radiation from the outside lithium zone containing lithium aluminate and construction steel makes the greatest contribution to the dose rate on its outside surface of 230 μR/s.

The activation radiation from the first wall after the column is attenuated by the structural materials of the module.

Thus, the total dose rate on the outside surface of the module facing the process room is 340 μR/s.

The long-lived ^{58}Co (71 days) and ^{54}Mn (312 days) serve as the basic radiator in the tritium breeding zone; therefore, even holding the module for several months does not lead to sufficient reduction in the dose rates in direct proximity to its surface. The necessity for lead shielding on the outer surface of the blanket module is obvious.

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GAS-COOLED BLANKET OF A HYBRID THERMONUCLEAR REACTOR WITH SOLID LITHIUM-CONTAINING MATERIALS*

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INTRODUCTION

The creation of blankets which contain a minimum amount of tritium for controlled thermonuclear reactors has arisen from the requirements for the radiation safety of the reactor under emergency conditions. The solution to this problem appears to be possible by applying compounds based on lithium with low tritium solubility and an efficient system for isolating it. As such compounds, the use of nonmetallic lithium-containing substances may be proposed, such as lithium aluminate ($LiAlO_2$), lithium oxide (Li_2O), lithium carbide (Li_2C_2), lithium hydride (LiH), and so forth. The use of the compounds indicated permits us to reduce the steady-state tritium content in the blanket by more than two times, compared with the blankets using liquid lithium. Helium is used as the heat-exchange agent in this type of blanket.

The designs for fusion reactors which use liquid lithium as the breeding material, the heat transfer agent, and the neutron moderator have disadvantages associated with corrosion of construction materials and MHD losses to pumping. Difficulties caused by the possible formation of gas plugs should also be considered. They sharply reduce the heat exchange and worsen the stagnant zones as a result of the complex configuration of the magnetic fields. All of this results in significant complication of the blanket structure, the system for pumping the heat transfer agent, and the tritium generation system. The use of helium as the heat transfer agent reduces the problem of compatibility; the use of an inert gas excludes the problem of MHD losses and simplifies the tritium generation system.

Thus the use of blankets with gas cooling based on solid lithium-containing compounds in the fusion reactors makes it feasible to simplify significantly the blanket structure as well as to increase the reliability of the entire device as a whole.

STATEMENT OF THE PROBLEM

One of the basic requirements imposed on the blanket of a hybrid thermonuclear reactor is the high energy yield per fission, and the provision for tritium breeding with a breeding factor of K_T = 1.05. In this paper, calculations were made on certain hybrid blanket systems with solid lithium-containing compounds in order to study the effect of different factors on the neutron-physical characteristics of the blanket and obtain the maximum K_T under the condition of energy generation on the 90-100 MeV level per fission. The following materials were elected for the study. Uranium carbide (UC) with 100% uranium-238 content in the uranium was selected as the fissionable material; lithium aluminate, lithium oxide, lithium hydride as well as lithium carbide, lithium, and lithium-aluminum alloy were selected for tritium breeding; graphite and zirconium hydride were selected as the structural material. The 4 basic blanket systems presented in Figs. 1-4 implementing the 11-zone model were compiled on the basis of these materials and adopted for the calculations. In the calculations, the volumetric content of the materials in the blanket as well as the thickness of the first wall and the process clearances were not varied. All of the calculations were performed in plane geometry with respect to the unidimensional BLANK program implementing a combination of the Monte Carlo method and the P_1 approximation.

The preliminary neutron-physical calculations demonstrated that in the adopted model of the blanket it is possible to obtain a high total product yield (the total tritium and plutonium production) on the level of 2.2-2.3 nuclei per fission, which is basically determined by the total number of neutrons participating in the balance, i.e., breeding, parasitic capture on the construction materials, and leakage of the neutrons out of the system. However, the tritium breeding factor in this case is below 1.0. This is

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* Translated from the Russian by Addis Translations International, Portola Valley Ca. 94025.

explained by the high absorption of the neutrons in the uranium zone. The problem was thus formulated concerning the maximum increase in the tritium breeding factor, with the condition that the total product yield as well as the energy generation be maintained (or insignificantly reduced).

CALCULATION RESULTS

The first series of calculations were devoted to the study of the effect of the thickness of the uranium zone (zones 3 and 4) on the tritium breeding and the energy generation in the blanket. The calculations were performed for version A. The thickness of the uranium zone was varied from 10 to 40 cm. The results of the calculation are presented in Fig. 5, from which it is evident that, with an increase in the size of the uranium zone, the total production and the energy yield per fission increase. However, the tritium breeding is reduced in this case. This is explained by a decrease in the neutron leakage through the right-hand boundary of the uranium zone as a result of an increase in the absorption in the zone itself, which leads to a reduction in the number of neutrons which can be used for tritium breeding. Therefore, in order to insure a maximum tritium breeding factor, the size of the uranium zone must be set at 8-10 cm, but this size is unacceptable from the point of view of energy generation in the blanket, for in this case the total energy yield will be 70-75 MeV. In order to insure the energy generation in the blanket at the 90 MeV level, the size of the uranium zone must be set at 20 cm.

Then, in the uranium zone, 75 MeV will be generated per fission, and the neutron leakage through the right-hand boundary will be 1.26 neutrons. For each thermonuclear neutron of the source this permits a total energy yield at the 90-MeV level and the use of 1.26 neutrons for tritium breeding.

One of the acceptable means of increasing the tritium breeding factor is to increase the thickness of the moderator. The tritium breeding factor, the plutonium yield, and the absorption in the moderator are presented in Figs. 6-11 as a function of the moderator thickness. From the functions presented, it is possible to conclude that the basic increase in K_T is observed with an increase in the size of the moderator for version A to 50 cm, for version B to 45 cm, and for version C to 5-6 cm. A further increase in thickness of the moderator will not result in a noticeable increase in K_T; in the case of a moderator made of zirconium hydride (version C), it will result in a sharp decrease. This is connected primarily with an increase in the reflectivity of the moderator, and secondly to an increase in the absorption in the moderator itself. The use of the moderator in version D must be regarded as inexpedient, for the lithium hydride has sufficient moderating capacity.

The next series of calculations was devoted to the investigation of the possibility of an increase in K_T by enrichment of the lithium with the ^6Li isotope. In Figs. 12-15 we have the tritium breeding factor, the plutonium production factor K_{Pu}, and the total product yield K_{Pr} as a function of the lithium enrichment. The relations presented indicate a significant increase in K_T (by 15-40% for various versions) and a reduction in K. Here K_{Pr} increases by several per cent. The significant increase in K_T is explained by the increase in the reaction cross section with the tritium yield, for the ^6Li (n, α)T reaction cross section is appreciably higher than the ^7Li (n, n'α)T reaction cross section. Therefore, with an increase in the lithium enrichment, the probability of the neutron absorption in the lithium increases, and the probability of the "return" of the neutron to the uranium zone decreases. This also explains the reduction in the plutonium production. Thus, in version A using the natural composition of lithium, the absorption in the first five zones is equal to 2.15 neutrons per fission; for lithium with 50% enrichment with respect to ^6Li, it is 1.97 neutrons per fission. The increase in K_T in this case is 25.1%, almost the entire increase in K_T during enrichment is obtained as a result of reducing the uranium-238 capture. Some increase in K_{Pr} is explained by reducing the parasitic capture on the construction materials and decreasing the neutron leakage.

In the next series of calculations, a study was made of the effect of the thickness of the first lithium-containing zone. The calculation results are presented in Figs. 16-19. As should be expected, with an increase in the thickness of the lithium zone, the tritium breeding factor increases along with the total product yield. A significant increase in K_T is observed with an increase in the thickness of the lithium zone for version A to 20-25 cm, for version B to 18-20 cm, for version C to 20-25 cm, and for version D to 55-60 cm. A further increase in the thickness does not lead to a noticeable increase in K_T.

Thus, even an increase in the thickness of the lithium zone and an increase in the thickness of the moderator lead to an increase in the tritium breeding factor. However, the limitation of the total size of the blanket by the structural peculiarities of the specific device does not always permit selection of the optimal dimensions of the lithium zones and the moderator.

Therefore, the next series of calculations was devoted to studying the effect of the ratio of the dimensions of the first lithium zone and the moderator. The total thickness of these zones was taken as equal to 50 cm. The calculations were performed for versions A and B, and the results are presented in Figs. 20 and 21. A quite broad optimum with respect to tritium breeding is observed for both versions, the thickness of the first lithium zone being within the limits of 15-30 cm. It appears difficult to indicate the region of the optimum more precisely, for the effect of the given factor in the indicated region (15-30 cm) is quite weak.

In the last calculation series, a comparison was made of the tritium breeding by various lithium-containing materials. Lithium aluminate ($LiAlO_2$), lithium-aluminum alloy (LiAl), lithium (Li), and lithium oxide (LiO_2), as well as lithium hydride (LiH) and lithium carbide (Li_2C_2) were selected for comparison. Table 1 contains the material composition and the arrangement of six versions of blankets using the indicated materials for tritium breeding. The neutron-physical parameters of these versions are presented in Table 2.

In versions 1-4, the tritium breeding by four different lithium-containing meterials was compared. In the first version, lithium aluminate is used (the composition corresponds to version A); in the second version, lithium aluminum alloy is used; in the third version lithium; and in the fourth version lithium oxide (the composition corresponds to version B). The dimensions of the zones, the material content, and the enrichment of the lithium are assumed to be identical for all versions. As is evident from Table 2, the maximum lithium breeding factor is obtained when using lithium oxide, which is due to the high lithium content in this material.

In versions 4 and 5 the two moderators--graphite and zirconium hydride--are compared. In version 4, a graphite moderator 30 cm thick is used; in version 5, zirconium hydride 5 cm thick is used (corresponding to version C). As is evident from table 2, in the adopted model the zirconium hydride permits a decrease in the blanket thickness by 25-30 cm in comparison with graphite, while keeping the tritium yield on the former level.

The tritium breeding by lithium hydride (version 6) and lithium carbide (version 7) were compared in the last two versions. The model adopted in these versions corresponds to version D, excluding zones 7, 8, and 9. As the calculations demonstrated, the use of these materials makes it possible to exclude the zone with the moderator. However, the absence of test data with respect to radiation resistance, compatibility, and the possibility of generating tritium does not permit any recommendations to be made for their use.

CONCLUSION

The calculations performed demonstrated that, in the hybrid blanket using uranium carbide with 100% uranium-238 content in the uranium as the fissionable material, it is possible to obtain a total product yield at the 2.2-2.3 level of nuclei/neutron, with energy being generated on the 90-100 MeV/neutron level. The effort to increase the total product yield and the energy in the adopted model leads to a reduction in the tritium reproduction coefficient. It should also be noted that the addition of 0.4% uranium-235 to the uranium leads to a 10-15% increase in product yield and energy generation.

In comparison with the graphite moderator, the hydrogen-containing moderators used make it possible to reduce the blanket thickness significantly.

Obviously, in such materials as lithium aluminate and lithium-aluminum alloy, even with great thicknesses and high enrichment, it is not possible to obtain a tritium breeding factor greater than 0.85-0.95. However, when using materials in the proposed system having a high lithium content, such as lithium oxide and lithium hydride, it is possible to obtain a tritium breeding factor on the 1.05-1.1 level, with enrichment of the lithium with lithium-6 within the limits of 50%.

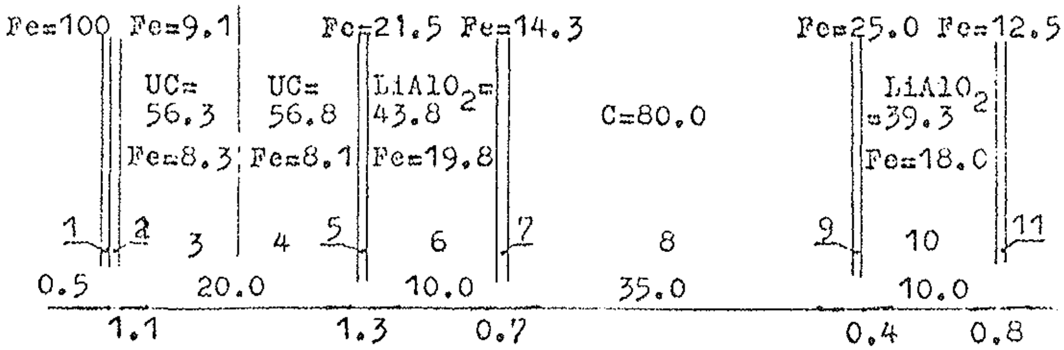

Fig. 1. Blanket diagram -- version A.

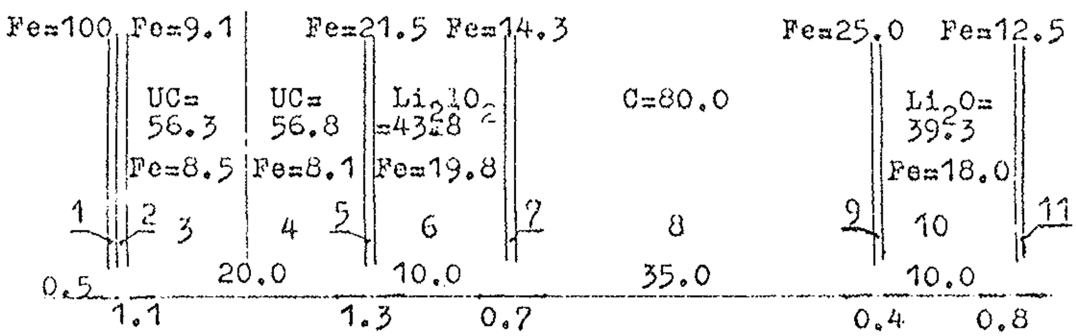

Fig. 2. Blanket diagram -- version B.

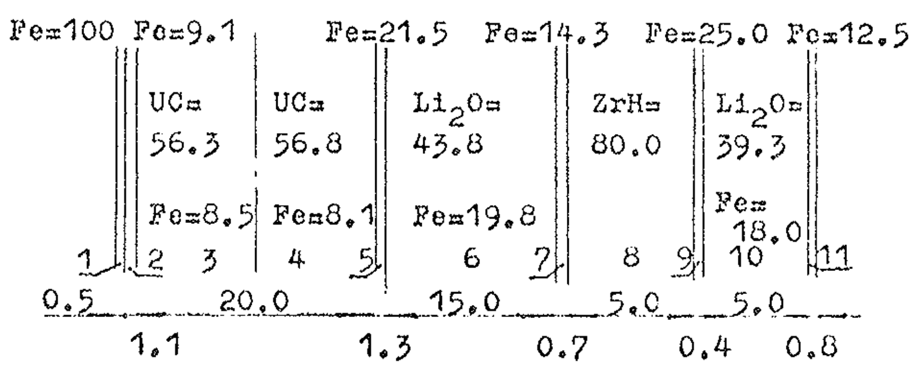

Fig. 3. Blanket diagram -- version C.

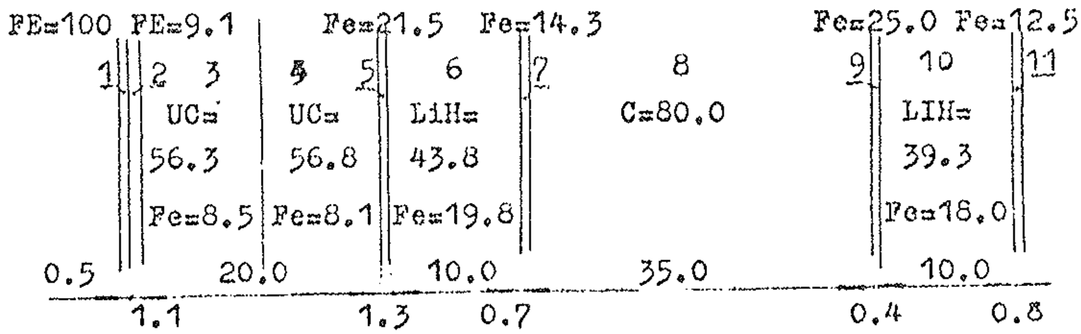

Fig. 4. Blanket diagram -- version D.

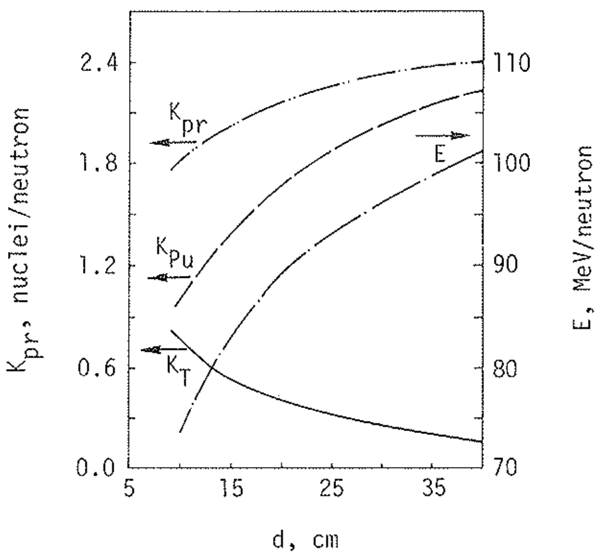

Fig. 5. Neutron-physical parameters as a function of the size of the uranium containing zone (zones 3-4) for version A with natural Li composition. The dimensions of the other zones are in accordance with Fig. 1.

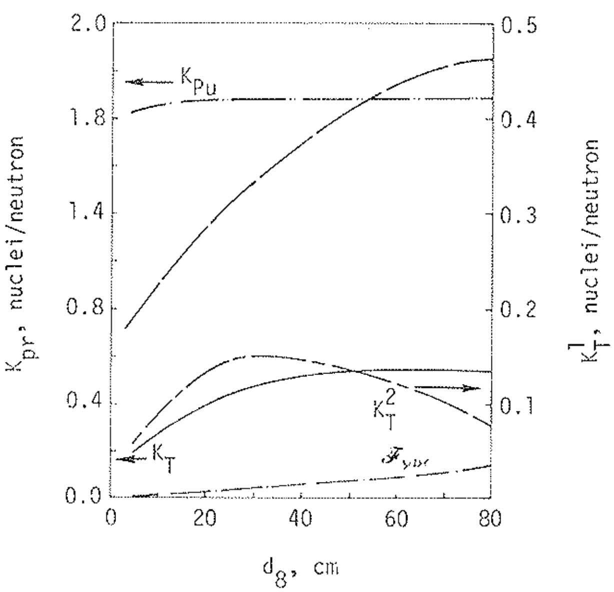

Fig. 6. Tritium breeding and plutonium yield as a function of the size of the moderator (zone 8) for version A with natural Li composition. The dimensions of the other zones are in accordance with Fig. 1.

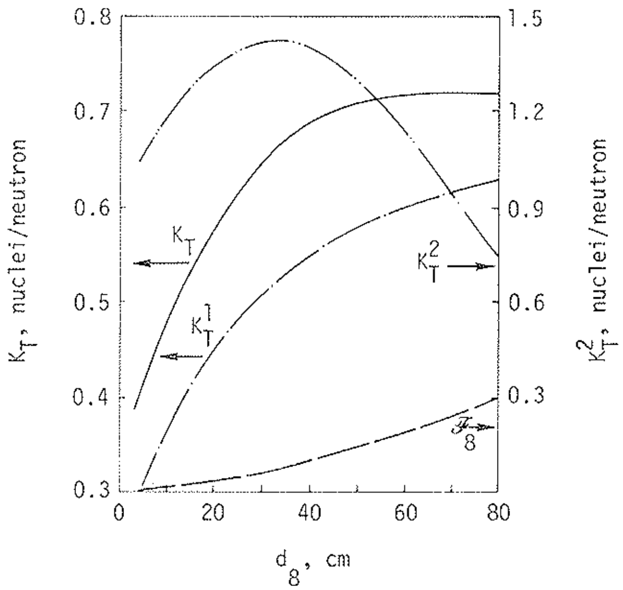

Fig. 7. Tritium breeding and absorption in the moderator as a function of the thickness of the moderator (zone 8) for version B with natural Li composition. The sizes of the other zones are in accordance with Fig. 2.

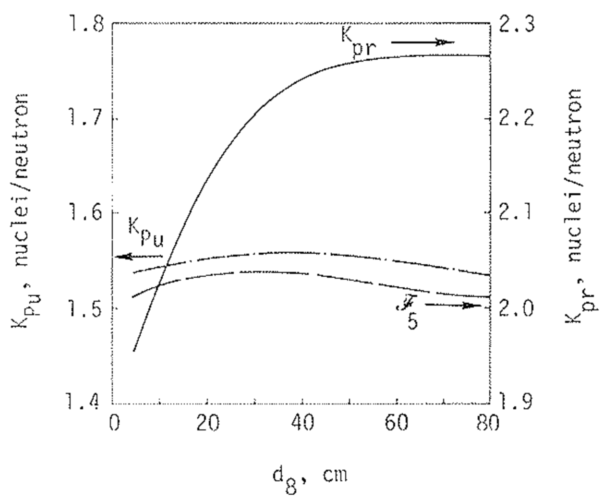

Fig. 8. Product yield and absorption in the first five zones as a function of the thickness of the moderator (zone 8) for version B with natural Li composition. The dimensions of the other zones are in accordance with Fig. 2.

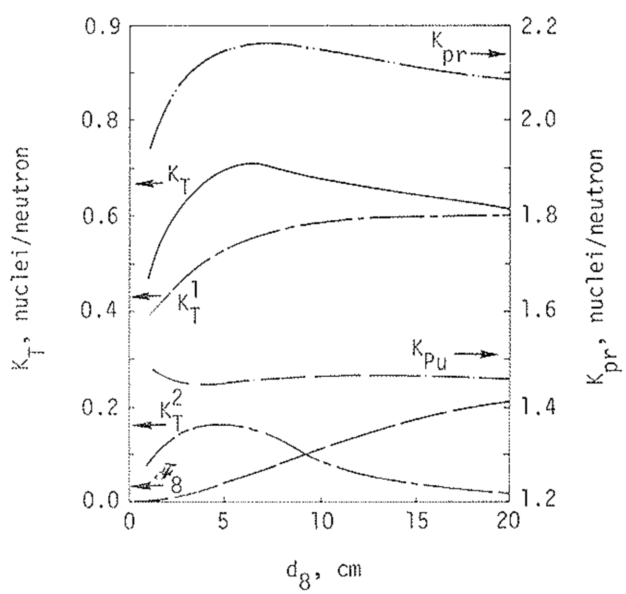

Fig. 9. Neutron-physical parameters as a function of the moderator thickness (zone 8) for version C with natural Li composition. The dimensions of the other zones are not in accordance with Fig. 3.

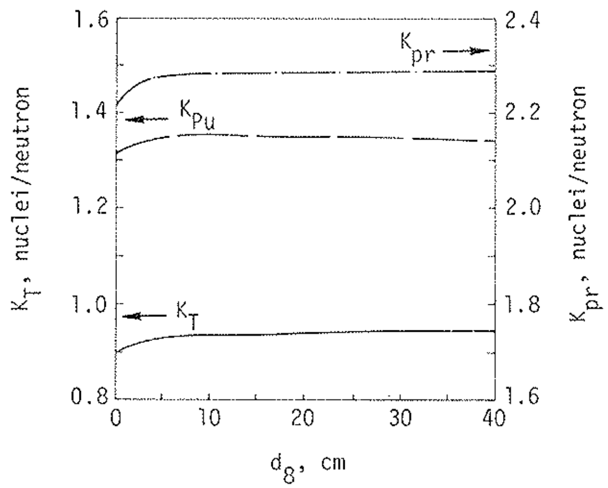

Fig. 10. Neutron-physical parameters as a function of the moderator thickness (zone 8) for version D with natural Li composition. The dimensions of the other zones are in accordance with Fig. 4.

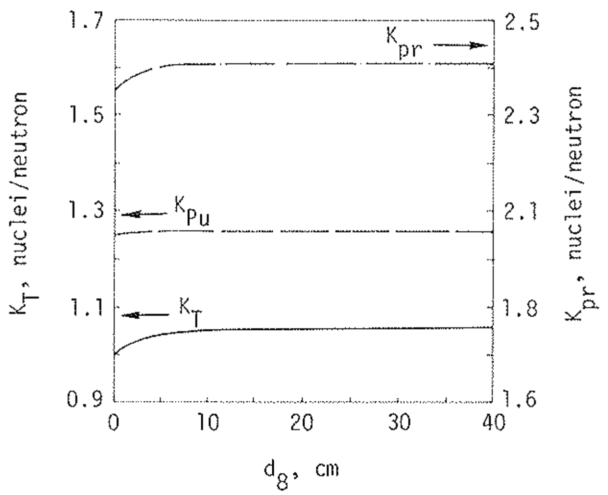

Fig. 11. Neutron-physical parameters as a function of the moderator thickness (zone 8) for version D with 50% Li enrichment. The dimensions of the other zones are in accordance with Fig. 4.

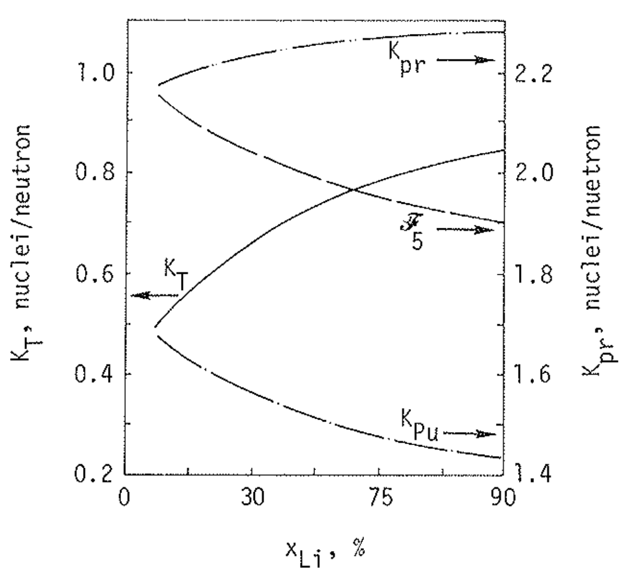

Fig. 12. Product yield and absorption in the first five zones as a function of the Li enrichment for version A. The sizes of the zones are in accordance with Fig. 1.

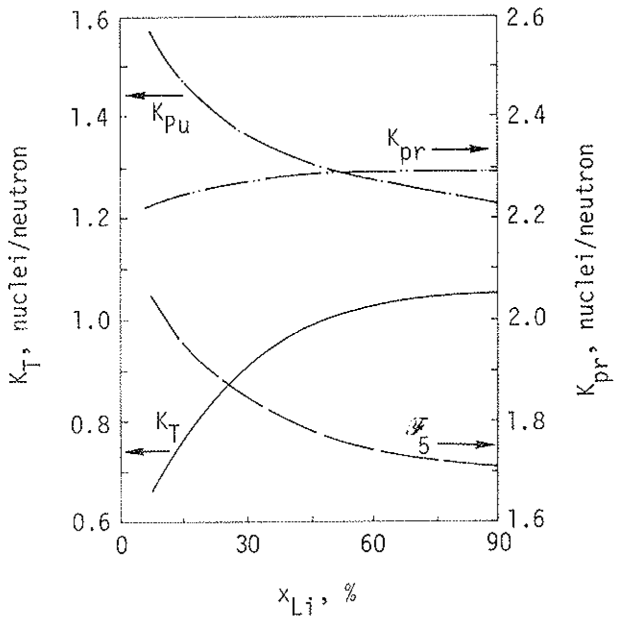

Fig. 13. Product yield and absorption in the first five zones as a function of the Li enrichment for version B. Zone dimensions in accordance with Fig. 2.

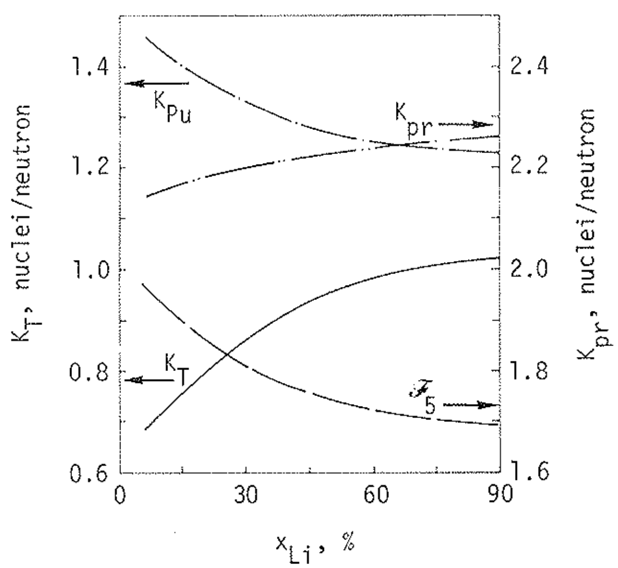

Fig. 14. Product yield and absorption in the first five zones as a function of the Li enrichment for version C. Zone dimensions in accordance with Fig. 3.

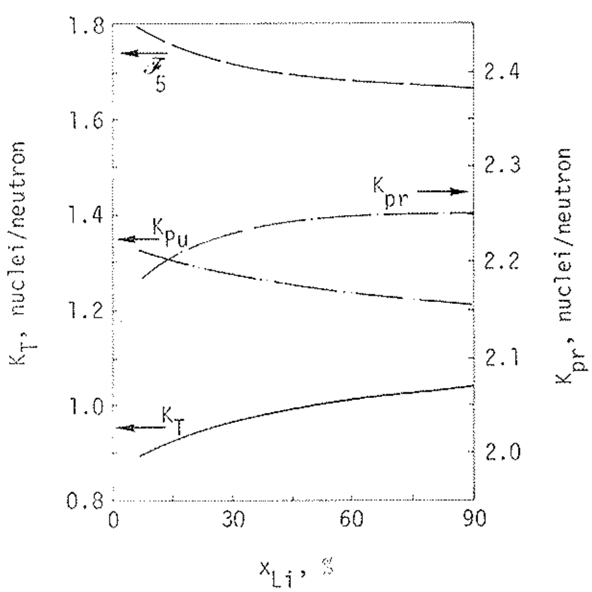

Fig. 15. Product yield and absorption in the first five zones as a function of the Li enrichment for version D. The zone dimensions are in accordance with Fig. 4. The moderator (zone 8) is absent.

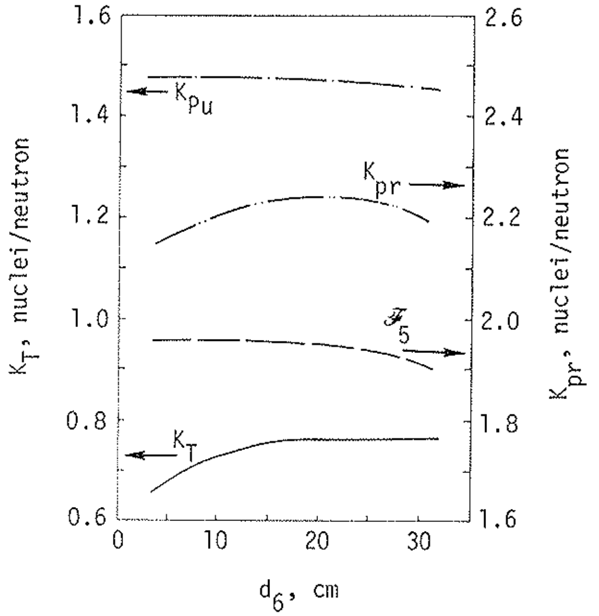

Fig. 16. Neutron-physical parameters as a function of the thickness of the first Li zone (zone 6) for version A with 50% Li enrichment. The thickness of the second Li zone (zone 10) is 2 cm. The dimensions of both zones are in accordance with Fig. 1.

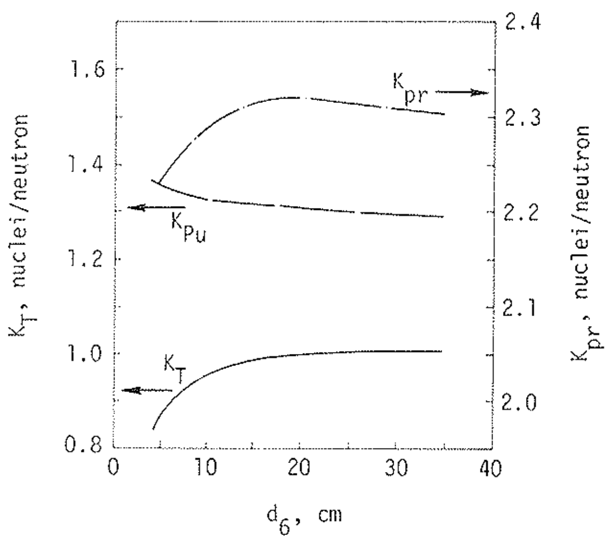

Fig. 17. Neutron-physical parameters as a function of the thickness of the first Li zone (zone 6) for version B (50% ^{6}Li enrichment, thickness of zone 10 2 cm). The dimensions of the other zones are in accordance with Fig. 2.

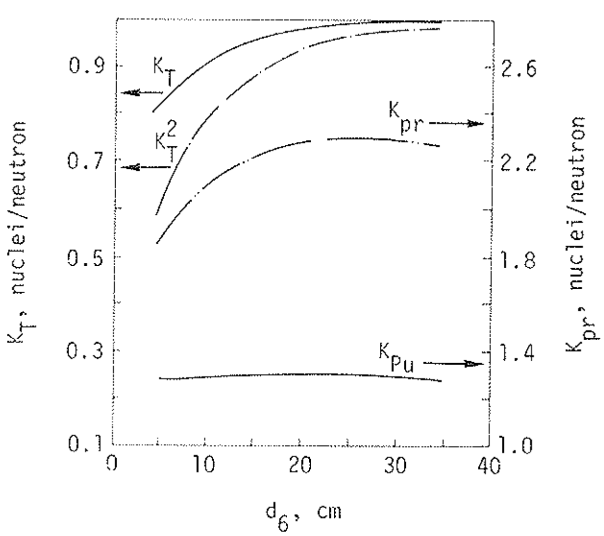

Fig. 18. Neutron-physical parameters as a function of the thickness of the first Li-containing zone (zone 6) for version C (50% ^{6}Li enrichment, thickness of the second Li-containing zone (zone 10) 2 cm). The dimensions of the other zones are in accordance with Fig. 3.

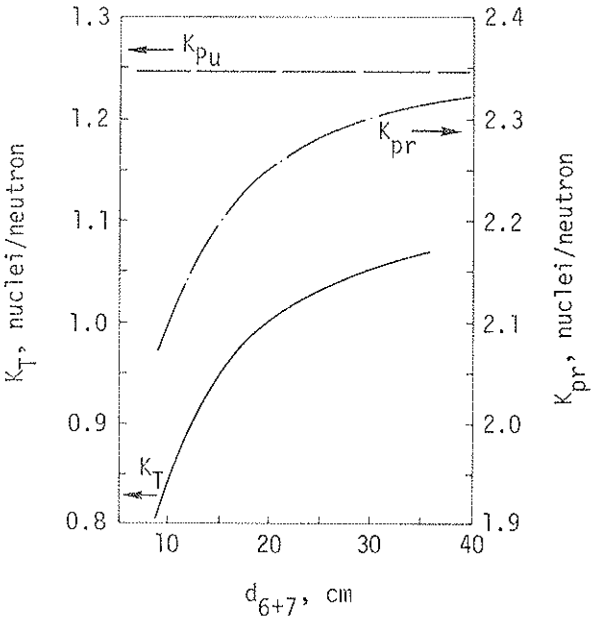

Fig. 19. Product yield as a function of the thickness of the Li-containing zone for version D. The ^{6}Li enrichment is 50%, the moderator is absent. The dimensions of the other zones are in accordance with Fig. 4.

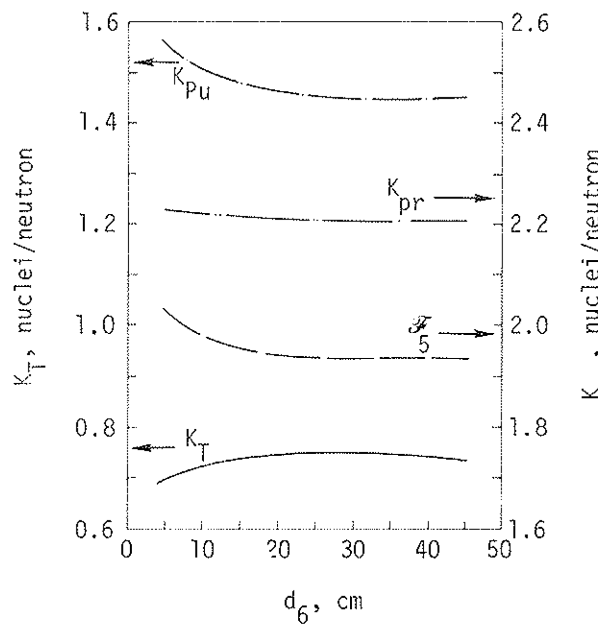

Fig. 20. Product yield as a function of the ratio of the dimensions of the first Li-containing zone (zone 6) and the moderator (zone 8) for version A. ($d_6 + d_8 = 50$ cm). Thickness of zone 10 5 cm. The dimensions of the other zones are in accordance with Fig. 1.

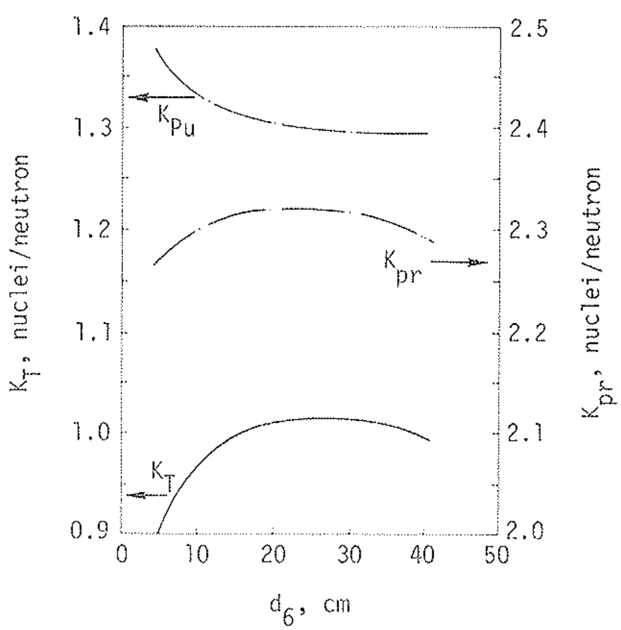

Fig. 21. Product yield as a function of the ratio of the dimensions of the first Li-containing zone (zone 6) and the moderator (zone 8) for version B.($d_6 + d_8 = 50$ cm).

Thickness of zone 10 5 cm. The dimensions of the other zones are in accordance with Fig. 2.

Table 1. Material composition and arrangement of 6 versions of the blanket.

Version	Zones	Thickness of the zone (cm)	Elements and nuclear densities (10^{24} nuclei/cm^3)
	1	0.5	FE 0.0848
	2	1.1	FE 0.00772
	3	20.0	U-238 0.01625 C 0.01625 FE 0.00720
	4	20.0	U-238 0.01640 C 0.01640 FE 0.00686
	5	1.3	FE 0.01820
	6	20.0	LI-6 0.005266 LI-7 0.005266
1			AL 0.010533 O 0.021066 FE 0.016748
	7	0.7	FE 0.01210
	8	30.0	C 0.06420
($LiAlO_2$)	9	0.4	FE 0.02120
	10	5.0	LI-6 0.004581 LI-7 0.004581
			AL 0.009162 O 0.018324 FE 0.016451
	11	0.8	FE 0001060
	1	0.5	FE 0.0848
	2	1.1	FE 0.00772
	3	20.0	U-238 0.01625 C 0.01625 FE 0.00720
	4	20.0	U-238 0.01640 C 0.01640 FE 0.00686
	5	1.3	FE 0.01820
	6	20.0	LI-6 0.005913 LI-7 0.005913
			AL 0.011826 FE 0.016748
2	7	0.7	FE 0.01210
	8	30.0	C 0.06420
(LiAl)	9	0.4	FE 0.02120
	10	5.0	LI-6 0.005306 LI-7 0.005306
			AL 0.010611 FE 0.016451
	11	0.8	FE 0.01060
	1	0.5	FE 0.0848
	2	1.1	FE 0.00772
	3	20.0	U-238 0.01625 C 0.01625 FE 0.00720
	4	20.0	U-238 0.01640 C 0.01640 FE 0.00686
	5	1.3	FE 0.01820
	6	20.0	LI-6 0.009198 LI-7 0.009198
3			FE 0.016748
	7	0.7	FE 0.01210
(Li)	8	30.0	C 0.06420
	9	0.4	FE 0.02120
	10	5.0	LI-6 0.008253 LI-7 0.008253
			FE 0.016451
	11	0.8	FE 0.01060

	1	0.5	FE	0.0848				
	2	1.1	FE	0.00772				
	3	20.0	U-238	0.01625	C	0.01625	FE	0.00720
	4	20.0	U-238	0.01640	C	0.01640	FE	0.00686
	5	1.3	FE	0.01820				
	6	20.0	LI-6	0.018774	LI-7	0.018774		
			O	0.014140	FE	0.016748		
4	7	0.7	FE	0.01210				
	8	30.0	C	0.06420				
(Li_2O-C)	9	0.4	FE	0.02120				
	10	5.0	LI-6	0.016332	LI-7	0.016332		
			O	0.012300	FE	0.016451		
	11	0.8	FE	0.01060				
	1	0.5	FE	0.0848				
	2	1.1	FE	0.00772				
	3	20.0	U-238	0.01625	C	0.01625	FE	0.00720
	4	20.0	U-238	0.01640	C	0.01640	FE	0.00686
	5	1.3	FE	0.01820				
	6	15.0	LI-6	0.018774	LI-7	0.018774		
			O	0.014140	FE	0.016748		
5	7	0.7	FE	0.01210				
	8	5.0	ZR	0.02749	H	0.04673		
(Li_2O-ZrH)	9	0.4	FE	0.02120				
	10	5.0	LI-6	0.016332	LI-7	0.016332		
			O	0.012300	FE	0.016451		
	11	0.8	FE	0.01060				
	1	0.5	FE	0.0848				
	2	1.1	FE	0.00772				
	3	20.0	U-238	0.01625	C	0.01625	FE	0.00720
	4	20.0	U-238	0.01640	C	0.01640	FE	0.00686
6	5	1.3	FE	0.01820				
	6	20.0	LI-6	0.013509	LI-7	0.013509		
(LiH)			H	0.027017	FE	0.016748		
	7	20.0	LI-6	0.011752	LI-7	0.011752		
			H	0.023503	FE	0.016451		
	8	0.8	FE	0.01060				
	1	0.5	FE	0.0848				
	2	1.1	FE	0.00772				
	3	20.0	U-238	0.01625	C	0.01625	FE	0.00720
	4	20.0	U-238	0.01640	C	0.01640	FE	0.00686
7	5	1.3	FE	0.01820				
	6	20.0	LI-6	0.008979	LI-7	0.008979		
(Li_2C_2)			C	0.017958	FE	0.016748		
	7	20.0	LI-6	0.008056	LI-7	0.008056		
			C	0.016113	FE	0.016451		
	8	0.8	FE	0.01060				

Table 2. Neutron-physical parameters of six versions of the blanket.

Version	K_T	K_{Pu}	I	n,2n	n,3n	n,f	E
	nuclei/neutron		number of reactions/neutron				MeV/neutron
1 $LiAlO_2$	0.747	1.454	0.034	0.251	0.111	0.347	89.8
2 LiAl	0.816	1.427	0.056	0.257	0.118	0.352	89.3
3 Li	0.914	1.306	0.054	0.251	0.108	0.350	90.1
4 Li_2O-C	1.020	1.299	0.016	0.253	0.117	0.353	91.6
5 Li_2O-ZrH	0.959	1.274	0.047	0.259	0.117	0.342	89.6
6 LiH	1.079	1.248	0.009	0.248	0.122	0.351	89.9
7 Li_2C_2	0.845	1.382	0.093	0.265	0.114	0.349	90.0

PROVISIONAL NOTATION

K_T - tritium breeding factor in the blanket.

K_T^1 - tritium breeding factor in the first lithium zone.

K_T^2 - tritium breeding factor in the second lithium zone.

K_{Pu} - plutonium-239 production coefficient.

I - neutron leakage through the right-hand boundary of the blanket.

K_{pr} - total product yield in the blanket ($K_T + K_{Pu} = K_{pr}$).

$\mathcal{F}_5$ - neutron absorption in the first five zones.

$\mathcal{F}_8$ - neutron absorption in the moderator.

E - energy yield for one thermonuclear neutron.

THORIUM IN THE BLANKET OF A HYBRID THERMONUCLEAR REACTOR*

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ABSTRACT

A study is made of the possibility of using thermonuclear neutrons with an energy of 14.1 MeV for the conversion of natural thorium raw material to nuclear fuel for fission reactors. The possible versions of the hybrid blanket systems for thermonuclear reactors were investigated.

INTRODUCTION

The modern era is characterized by an accelerated growth of industrial production throughout the entire world. Nuclear and thermonuclear power engineering are being developed intensely at this time to satisfy the demands of industry for electric power.

The nuclear fuel reserves for fission reactors are determined by the amount of uranium-235: one gram of natural uranium contains 0.007 grams of uranium-235. The data on the uranium reserves are presented in Table 1 as a function of the cost of uranium extraction. The low uranium-235 content in natural uranium means that there are limited fuel reserves for fission reactors, but there is a possibility for expanding this reserve by processing thorium raw material into nuclear fuel.

Thorium is in about thirty-fifth place among the elements with respect to the frequency of its occurrence in nature.[1] There is five times a much thorium in the earth's crust as uranium; according to the latest data the thorium content in the earth's crust is $1.2\text{-}1.3 \times 10^{-13}\%$.[2]

In contrast to uranium-235, thorium-232 is not in itself a fissionable material, but when natural thorium is irradiated with slow neutrons, a number of nuclear conversions take place which result in the formation of the long-lived fissionable isotope uranium-233. Herein lies the basic value of thorium-232 as a source for supplementing the world reserves of fissionable materials.[3]

*Translated from Russian by Addis Translations International, Portola Valley, Ca. 94025.

The nuclear reaction occuring on irradiation of thorium-232 can be represented as follows:

Thorium-232 is contained in the hybrid reactor blankets in the form of Th, ThO_2, and ThC, as well as in the form of molten lithium salts. When selecting the construction materials, it is necessary to consider their reserves in the earth's crust, in addition to the neutron-physical characteristics. Table 2 (from ref. 4) gives some idea of the reserves of the materials used in a blanket.

The study of the systems for the conversion of thorium-232 to uranium-233 was started in ref. 5, 6, 7, and 8.

In the hybrid reactor that burns thorium-232, three systems are possible. In the first, the symbiosis system, studied in detail in ref. 6, the thorium-232 is processed into uranium-233 with subsequent use of the latter in the nuclear power reactors. As shown in this paper, the primary purpose of this system can be to supply the nuclear reactors with additional fuel required for expanded breeding. The proportion of this fuel in the energy balance will be 10% of the total fuel available for nuclear power engineering, and it will have no significant effect on the balance.

The second system, the fuel blanket system, involves the hybrid in the breeder regime burning uranium-233 directly in the blanket. Two means of implementing it are possible--in the reactors in molten salts with continuous removal of the fission products and part of the uranium-233 formed, and in the reactor with solid fuel elements based on a mixture of thorium dioxide or carbides with regular processing of uranium-233. The system of this type was investigated in ref. 8. It operates efficiently for a slightly subcritical blanket, the structural design and the composition of which are similar to the nuclear reactor. A number of advantages essential to the hybrid systems are lost in this system: the absence of the doubling time and the increased yield of fissionable material per unit power.

The third possible system is based on the fast blanket comprising the natural or impoverished uranium-238 or thorium-232 raw material. Some of its possibilities are investigated in Ref. 8. When burning only thorium-232, a defiency of this system is the relatively low energy released in the blanket per fission ($\sim$40 MeV). This energy is insufficient for an acceptable efficiency of the device, especially in the case of open traps, pulse systems or Tokamaks without ignition.

A number of calculations have been made of blankets which could improve the reactive parameters in the third system, burning thorium-232 as the raw material.

COMBINED URANIUM-THORIUM SYSTEMS

The hybrid reactors can offer power engineering a possibility not realizable in nuclear systems, namely, the burning of a significant portion of the raw material without processing the irradiated fuel. For uranium-238, the degree of burnup can correspond to loads of 100-150 GWd/t at the present time, which amounts to 10-15% of all the fuel. Theoretically analogous degrees of burnup could also be obtained with thorium-232, placing it directly under the 14-MeV neutron flux. In this case, at the beginning of the run the energy yield would be small ($M \approx 2$),* but it would increase with time to a value of $M \approx 10$-20.

The output of the power plant would vary with time by approximately 10 times (while maintaining plasma parameters), which is quite disadvantageous from the economic point of view. The blanket comprised of uranium-238 and thorium-232 at the same time would be more advantageous. Its first layer contains uranium-238 to increase both the number of fissions in the 14-MeV neutrons and the value of M at the beginning of the run. Thorium-232 and lithium-containing materials must be available in the subsequent zones. It may be hoped that in this type of system the value of M will vary during the run by 2-3 times (from 7-10 to 20-30).

Inasmuch as the uranium zone is located in a higher neutron flux than the tritium zone, in the absence of fuel recharging, the degree of burnup in it will be higher. The replacement of several uranium zones for burning thorium alone is disadvantageous, for in this case the total amount of fuel used will approximate the amount of uranium used. The possibility of irradiating the thorium fuel elements until approx. 3-5% of the uranium-233 is accumulated in them, then recharging the irradiated fuel elements in the nuclear reactor or in a separate section of the thermonuclear reactor, and continuing the irradiation there to burnup determined by their mechanical strength, appears to be theoretically promising. The idea of recharging is fraught with numerous technical difficulties, but successful implementation of it will permit a significant expansion of the fuel reserves, thus avoiding the difficulties associated with the processing and refabrication of the fuel elements.

A number of neutron-physical calculations of uranium-thorium blankets are presented below. The choice of the optimal versions requires a knowledge of both the properties of the different thorium fuel elements and the parameters of the special nuclear reactors in which it is possible to arrange their complete burnup.

HYBRID REACTORS IN THE SYMBIOSIS SCHEME

The first of the schemes discussed above appears to be a pure producer in which, when irradiating thorium-232 with thermonuclear neutrons, the production of uranium-233 takes place. It is assumed that the number of fissions in it is not

*M is the ratio of the energy released per fission in the blanket to the energy of the 14-MeV neutron.

large. In order to obtain uranium-233 production in the hybrid reactor while maintaining a tritium breeding factor greater than one, it is necessary either to make efficient use of the reaction of inelastic scattering on the lithium-7 (with tritium yield) or to breed thermonuclear neutrons using the (n, 2n) and (n, 3n) reactions in various materials. The second part of the paper has been aimed at studying the possible neutron breeders and selecting the optimal ones.

CALCULATION PROCEDURE

All of the neutron-physical characteristics of the hybrid blankets were obtained by the BLANK computation program.[9]

In the fusion reactor the initial energy of the neutrons is quite high, and to calculate the neutron flux in the blanket, a detailed consideration of all of the reactions taking place, of the anisotrophy of the elastic scattering of the neutrons, and of the corresponding energy yields is required. The procedure for the neutron-physical calculation of the thermonuclear reactor blanket consists in solving two successive problems: determination of the space-energy distribution of the neutron flux and calculation of some of the flux functionals. The neutron flux distribution in the thermonuclear reactor blanket with the D-T reaction is described by the nonuniform Boltzmann kinetic equation using a monochromatic neutron source with an energy of 14.1 MeV.

The energy range from 14 MeV to 0 is divided into two intervals, each of which has its own systems of constants and method of solving the transport equation. The boundary of the intervals was selected at an energy of 0.01 MeV. In the upper energy range, the equation was solved by the Monte Carlo method; the energy variation was considered to be continuous, and the process cross section was averaged in 52 energy groups. As a result of the necessity for exact consideration of the angular and energy distributions of the secondary neutrons, these processes were described with maximum Q-factor. The scattering process was worked out separately for each element. The description of the process cross sections was produced on the basis of the approximate data from the UKNDL, ENDF/B and the KEDAK bibliographies and Soviet authors, in accordance with which the 52-group system of constants designed for calculations by the Monte Carlo method in the 0.1-15 MeV range was compiled. The external source was given as isotropic on the left-handed boundary for plane geometry. In the energy range of 0 to 0.1 MeV, the equation was solved in the P_1- approximation by the method of numerical integration. The 21-group system of constants was used, which is designed to calculate the fission reactors. The solutions were joined with respect to the moderation density for an energy of 0.1 MeV. The procedure can also be used to calculate neutron fluxes in the blanket with the fissionable material. In this case, the Monte Carlo method is used only in the first integration, and the remaining integrations are calculated in the P_1-approximation. It must be noted that the final selection of the group cross sections and the secondary distributions of the scattered neutrons has not been made at this time. The maximum indeterminancies exist in the uranium-238 constants. Therefore, the authors permit the indeterminancy of the result on a 7-10% scale in the total neutron balance in the majority of the versions investigated.

CALCULATION RESULTS

The calculations were performed on the BLANK program in plane geometry; the number of iterations in the Monte-Carlo method is $\sim 10^4$. The calculation results are presented in Table 3 and they are normalized for one thermonuclear neutron. The gaseous helium is proposed as the heat transfer agent in the majority of the versions. Version 1 is a combined uranium-thorium blanket. Metallic uranium-238 with a volumetric content of 70% is used as the zone breeding thermonuclear neutrons. The breeding zone contains lithium dioxide with natural enrichment. Uranium-233 is obtained when thermal neutrons are absorbed on the metallic thorium-232. The intermediate graphite layer is used to moderate the 14.1-MeV neutrons, and the outer layer decreases the neutron leakage from the system.

In version 2 the thickness of the uranium zone was made thinner so that the uranium-233 yield would increase to the maximum. The obtained yield of the uranium-233 is 0.75, close to the maximum for such blankets.

Versions 3 and 4 illustrate the transition to uranium and thorium carbide in order to increase the radiation resistance of the blanket zone. The replacement of uranium by its carbide leads to a sharp decrease in the production of uranium-233.

The results of version 4 are analogous.

In ref. 6 a study was made for the first time of the symbiosis blanket system with molten salt. In accordance with various arguments, the use of the molten salts can be advantageous. Version 5 indicates approximate characteristics of the blanket with the molten salt. The possibilities of the salt systems still have been insufficiently studied.

Versions 6 and 7 constitute the thorium converter with quite high (0.6) uranium-233 yield. The low energy yield can be compensated by using the thorium zone enriched to 3-5% with respect to uranium-233. The zone enriched with respect to uranium-233 can be obtained by the scheme of version 1, which opens up the possibilities of direct burning of significant thorium-232 reserves without intermediate processing. Thorium carbide is used in version 7.

Versions 8 and 9 constitute the hybrid reactors of the symbiosis group similar to those proposed in ref. 6. Liquid lead, which is a good neutron breeder by the (n, 2n) and (n, 3n) reactions, was selected as the neutron breeder for economic reasons.

CONCLUSIONS

When constructing the combined uranium-thorium blankets it is expedient to use metallic uranium and thorium, alloyed 5-10% with molybdenum, tungsten, vanadium or zirconium. The uranium-233 and plutonium-239 enriched zones can be used after some processing or without it in the LWR or HTGR type reactors. The enriched thorium zones can be placed directly under the thermonuclear neutron flux, which makes it possible to burn thorium without intermediate processing.

The symbiosis schemes with breeders have still not been adequately investigated, but from a comparison of versions 6 and 9 it is clear that the thorium zone placed under the thermonuclear neutron flux yields similar neutron breeding.

BURNUP ANALYSIS

During the operation of the blanket in its various zones, plutonium-239 and uranium-233 are accumulated, which leads to significant variation of the blanket characteristics. In order to test the possibility of producing uranium-233 in the fuel elements of the thorium zone and for subsequent use of these fuel elements in the nuclear reactor or an individual section of the thermonuclear reactor, a calculation was made of the version 1 blanket run. The calculation was performed with a step size of 2 MWYE/m^2 by the neutron load of the first wall, assuming nonrechargeable uranium and thorium zones. The results are presented in Fig. 1. The plutonium-239 and uranium-233 concentrations are averaged over the zone. The coefficients of nonuniformity of the production of fissionable materials by zones are presented in Table 4. For times of 4-6 MWYE/m^2, the uranium burnup is not too great; the average uranium-233 concentration in the elements of the thorium zone is 0.7-1.0%. The maximum concentration reaches 3-4%. Elements with this concentration can be used for further burnup in the nuclear reactors of the water-cooled, water-moderated type. It should also be noted that on irradiation to 4 MWYE/m^2, the integral neutron flux with an energy of $E > 0.1$ MeV in the uranium zone is $4 \cdot 10^{22}$ n/cm^2; therefore, the efficiency of the thorium elements must not become significantly worse. The version investigated is not optimized, and obviously it is possible to create a blanket having the best characteristics from the standpoint of uranium-233 production in the thorium fuel element.

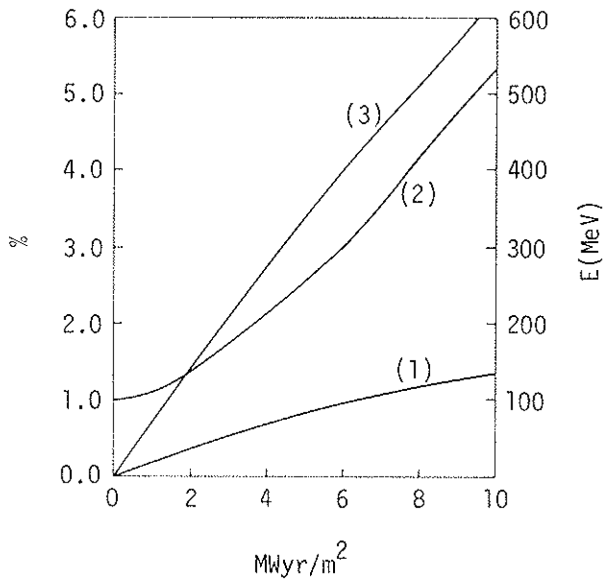

Fig. 1. Energy and isotopic composition as a function of time: (1) accumulation of U-233 in percentages throughout the zone, (2) energy release on fission of the heavy nuclei, (3) accumulation of Pu-239 in percentages with respect to the zone.

Table 1

Cost of ore in dollars/kg	22	33	66	110	200	500
Reserves, discovered and estimated in thousands of tons	2320	4020	6160	1400	$5 \cdot 10^5$	$5 \cdot 10^6$

Table 2

Element	Reserves 10^6 tons	Demand tons/MW(E)
Iron	180,000	10.0
Aluminum	3,000	0.6
Chromium	370	2.0
Lithium	180	0.9
Titanium	134	0.8
Copper	310	2.0
Boron	66	0.8
Vanadium	26	0.5
Nickel	24	1.5
Niobium	8.0	0.8
Lead	85.0	11.0
Helium	1.2	0.3
Beryllium	0.4	0.1

Table 3

N	Leak.	(n,Nn)	T_6	T_7	T	Pu^{239}	U^{233}	(n,f)	E	$K_{pr.}$
1	0.020	0.649	0.911	0.029	0.940	1.69	0.44	0.546	98.0	3.02
2	0.040	0.661	1.078	0.012	1.090	0.30	0.75	0.200	36.0	2.14
3	0.009	0.485	0.974	0.018	0.993	1.19	0.11	0.335	60.5	2.31
4	0.005	0.585	0.226	0.001	0.227	1.37	0.71	0.336	60.5	2.31
5	0.010	0.651	0.861	0.023	0.884	1.90	0.23	0.529	95.4	3.02
6	0.056	0.656	1.056	0.020	1.076	0.00	0.59	0.089	16.2	1.70
7	0.017	0.549	0.691	0.021	0.713	0.00	0.84	0.078	14.2	1.55
8	0.056	0.759	0.375	0.016	0.391	0.00	1.17	0.062	11.2	1.62
9	0.010	0.686	0.558	0.011	0.570	0.00	0.93	0.003	6.3	1.50

Notation:

"Leak." - total number of neutrons leaving the system.

(n,Nn) - additional number of neutrons as a result of (n, 2n) and (n, 3n).

T_6 - tritium yield for lithium-6.

T_7 - tritium yield for lithium-7.

T - total tritrium yield.

Pu^{239} - plutonium yield per neutron.

U^{233} - uranium yield per neutron.

(n, f) - number of fissions of heavy isotopes in the blanket.

E - energy released in the blanket.

$K_{pr.} = T + Pu^{239} + U^{233}$.

Table 4

Integral neutron load, MWYE/m^2	2	4	6	8	10
Uranium zone	1.6	1.56	1.45	1.45	1.28
Thorium zone	8.7	8.5	8.4	7.0	6.3

70%U^{238} 10%Fe 20%He	70%Li_2O 5%Fe 25%He	80%C 20%He	70%Th^{232} 10%Fe 20%He	80%C 20%He	70%Li_2O 5%Fe 25%He	(1)
10.	10.	20.	20.	10.	5.	

1	U-238	0.0335	Fe	0.0084				
2	Fe	0.0040	Li-6	0.0042	Li-7	0.0528	O	0.0235
3	C	0.0645						
4	Fe	0.0084	Th-232	0.0205				
5	C	0.0645						
6	Fe	0.0040	Li-6	0.0042	Li-7	0.0528	O	0.0235

Fe						
	60%U^{238} 10%Fe 30%He	60%Th^{232} 10%Fe 30%He	50%Li_2O 15%Fe 35%He	80%C 20%He	50%Li_2O 15%Fe 35%He	(2)
0.5	3.	15.	20.	20.	5.	

1	Fe	0.0848						
2	Fe	0.0085	U-238	0.0285				
3	Fe	0.0085	Th-232	0.0175				
4	Fe	0.0127	Li-6	0.0202	Li-7	0.0202	O	0.0168
5	C	0.0645						
6	Fe	0.0127	Li-6	0.0202	Li-7	0.0202	O	0.0168

Fe	Fe*	Fe*		Fe*		Fe*		Fe*		Fe	
	60%UC 10%Fe 30%He		70%Li_2O 10%Fe 20%He		80%C 20%He		70%Th^{232}C 10%Fe 20%He		80%C 20%He		(3)
0.5	15.	1.	15.	1.	10.	1.	15.	1.	20.	1.	

1	Fe	0.0848						
2	Fe	0.0170						
3	Fe	0.0084	U-238	0.0195	C	0.0195		
4	Fe	0.0170						
5	Fe	0.0084	Li-6	0.0171	Li-7	0.03490	O	0.0236
6	Fe	0.0170						
7	C	0.0645						
8	Fe	0.0170						
9	Fe	0.0170	C	0.0200	Th-232	0.0200		
0	Fe	0.0170						
11	C	0.0645						
12	Fe	0.0848						

Zr	Zr*		Zr		Zr*		Zr*		Zr	
		60%UN 10%Zr 30%He		70%ThN 10%Zr 20%He		70%Li_2O 10%Zr 20%He		80%C 20%He		(4)
1.	1.	15.	1.	20.	1.	20.	1.	25.	1	

1	Zr	0.0423					
2	Zr	0.0042					
3	Zr	0.0042	U-238	0.0204	N	0.0204	
4	Zr	0.0042					
5	Zr	0.0042	N	0.0196	Th-232	0.0196	
6	Zr	0.0042					
7	Zr	0.0042	O	0.0235	Li-6	0.0285	Li-7 0.0285
8	Zr	0.0042					
9	C	0.0645					
10	Zr	0.0423					

Fe					
	100%U^{238}	70%Li_2O 30%He	$LiF+BeF_2+ThF_4$ 71% 2% 27%	100%C	(5)
0.5	10.	20.	30.	15.	

1	Fe	0.0848									
2	U-238	0.0478									
3	Li-6	0.0042	Li-7	0.0528	O	0.0235					
4	Li-6	0.0017	Li-7	0.0214	Be	0.0051	Th-232	0.0038	F	0.048	
5	C	0.0803									

Fe					
	60%Th^{232} 10%Fe 30%He	50%Li_2O 15%Fe 35%He	80%C 20%He	50%Li_2O 15%Fe 35%He	(6)
0.5	15.	20.	20.	5.	

1	Fe	0.0848						
2	Fe	0.0085	Th-232	0.0175				
3	Fe	0.0127	Li-6	0.0202	Li-7	0.0202	O	0.0168
4	C	0.0645						
5	Fe	0.0127	Li-6	0.0202	Li-7	0.0202	O	0.0168

Fe		Fe*		Fe*		Fe*		Fe*	
	70%ThC 10%Fe 20%He		$70\%Li_2O$ 10%Fe 20%He		80%C 20%He		$70\%Li_2O$ 10%Fe 20%He		(7)
1.	15.	1.	20.	1.	15.	1.	10.	1.	

1 Fe	0.0848						
2 Fe	0.0084	Th-232	0.0200	C	0.0200		
3 Fe	0.0085						
4 Fe	0.0085	Li-6	0.0170	Li-7	0.0400	O	0.0235
5 Fe	0.0085						
6 C	0.0645						
7 Fe	0.0085	Li-6	0.0170	Li-7	0.0400	O	0.0235
8 Fe	0.0085						

Fe		Fe					
	100%Pb		70%ThC 10%Fe 20%He	$70\%Li_2O$ 10%Fe 20%He	80%C 20%He	$70\%Li_2O$ 10%Fe 20%He	(8)
0.5	5.	0.5	20.	20.	20.	10.	

1 Fe	0.0848						
2 Pb	0.0330						
3 Fe	0.0848						
4 Th-232	0.0200	C	0.0200				
5 Fe	0.0084	Li-6	0.0042	Li-7	0.0528	O	0.0235
6 C	0.0565						
7 Fe	0.0084	Li-6	0.0042	Li-7	0.0528	O	0.0235

Fe		Fe		Fe		Fe	
	100%Pb		$LiF+BeF_2+ThF_4$ 71.7% 16% 12.3%		C		(9)
1.	15.	1.	50.	1.	10.	1.	

1 Fe	0.0848								
2 Pb	0.0330								
3 Fe	0.0848								
4 Li-6	0.0017	Li-7	0.0214	Be	0.0051	Th-232	0.0038	F	0.047
5 Fe	0.0848								
6 C	0.0803								
7 Fe	0.0848								

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RAPIDLY PULSED
FUSION-FISSION REACTORS

ENGINEERING AND PHYSICS CONSIDERATIONS FOR A LINEAR THETA-PINCH HYBRID REACTOR (LTPHR)

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ABSTRACT

A fusion-fission hybrid reactor based on pulsed, high-β, linear theta-pinch magnetic confinement is considered. A preliminary design which incorporates key physics, engineering and economic considerations is presented. An extensive presentation of the system energy balance is made, and this energy balance is evaluated parametrically. The feasibility of end-loss reduction is addressed.

1. INTRODUCTION

The possibility of reducing the energy requirements of a fusion reactor by the introduction of a fissioning blanket around a thermonuclear D-T plasma has been discussed at varying levels of engineering detail.[1] Reference 1 in particular addresses the questions of blanket neutronics and energy balance for a pulsed, high-β Linear Theta-Pinch Hybrid Reactor (LTPHR). This machine is sized to be appropriate for a "first commercial" plant. Although the present study does not give a detailed engineering design of the LTPHR, key physics, engineering and economic uncertainties are addressed here in a systems analysis context. Specifically, a revised and more realistic energy balance for the LTPHR is developed and evaluated parametrically. This energy balance is conservatively based on free-streaming end loss of the plasma, although the feasibility of end stoppering is considered. A simple economic model is developed, by which the ergonic performance of the LTPHR is evaluated. The major intent of this study is the identification of an economically attractive, albeit unoptimized, design point for the LTPHR, as well as the identification of key R/D requirements for the economic production of fissile fuel and net electric power from a theta-pinch hybrid reactor. To this end an approximate but very flexible time-averaged model has been developed to examine a wide range of LTPHR parameters. Given an interim operating point based upon the time-average model, a global time-dependent burn computation is used both to examine the validity of the time-average model and to explore in more detail the effect of particle and/or energy end loss.

2. DESCRIPTION OF LTPHR ENERGY BALANCE

2.1. GENERAL FORMALISM

To evaluate the overall energy balance for the LTPHR the generalized energy flow diagram depicted in Fig. 1 has been developed.[1] The energy worth of a fusion neutron as sensible heat deposited within the LTPHR blanket is E(MeV/n), and the net number of fissile atoms produced per fusion neutron is [CV]. Hence, if E_F(MeV/fission) is the energy released per fissile isotope being considered, the potential energy of bred fission fuel per fusion neutron is $E^*(\text{MeV/n}) = [CV]E_F/(1+\alpha_{CF})$. The energy multiplication of a given blanket configuration is measured by the parameter $M = (E + E^*)/14.08$.

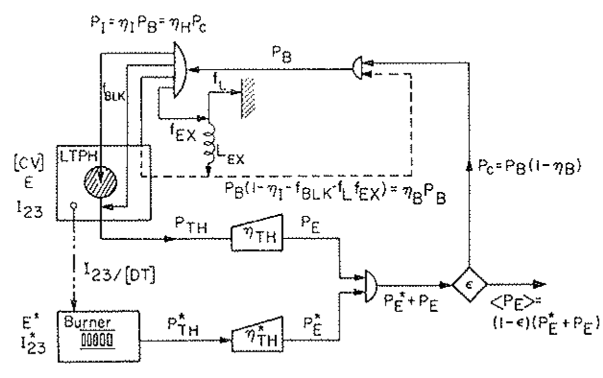


Fig. 1. Generalized energy-flow diagram for the Linear Theta-Pinch Hybrid Reactor (LTPHR). Refer to text for notation.

The LTPHR and the associated fissile fuel burner produce, respectively, a gross electrical power P_E(MWe/m) and P_E^*(MWe/m), where all quantities are expressed on the basis of a unit length of LTPHR. The production of $P_E + P_E^*$ is accomplished at the expense of a net power P_c(MWe/m) circulated to the LTPHR. The engineering Q-value, Q_E, is a measure of the return on the invested power P_c,

$$Q_E = (P_E + P_E^*)/P_c \quad . \tag{2-1}$$

A plasma-related Q-value, Q_p, can also be defined and is a measure of the total thermal power generated by the fusion neutron, $P_{TH} + P_{TH}^*$, per unit of power invested directly into the D-T plasma, P_I. Hence,

$$Q_p = (P_{TH} + P_{TH}^*)/P_I \quad . \tag{2-2}$$

P_I is the rate of energy injection into the theta-pinch plasma in order to initiate and sustain the thermonuclear burn. Engineering and plasma Q-values for the LTPHR *per se* can be obtained from Eqs. (2-1) and (2-2) by division with $(1 + E^*/E)$, i.e. $q_E = Q_E/(1 + E^*/E)$ and $q_p = Q_p/(1 + E^*/E)$.

The performance of the LTPHR is evaluated on the basis of the figure of merit Q_E in Sec. 4. In Sec. 2.2 and 2.3, respectively, analytical expressions for the circulating power and the fusion power in terms of system and plasma parameters are developed.

2.2 Circulating Power Losses

The circulating or make-up power P_c is considerably less than the total power P_B required to initiate and sustain the thermonuclear burn. The fraction η_B of P_B is recovered and re-used by the pulsed theta pinch. This section describes the derivation of an expression for η_B in terms of key system parameters.

As seen from Fig. 1 the fraction $(1 - \eta_B)$ lost from the circulating power loop is a frequency-dependent (i.e. dependent upon the pulse duration) quantity and is comprised of three major components: plasma heating η_I, joule losses in the blanket region (including the first-wall theta-pinch coil) f_{BLK}, and resistive losses incurred external to the theta pinch, $f_{EX}f_L$. The transfer efficiency f_{EX} gives that fraction of P_B that must be supplied to portions of the circuit external to the reactor, and f_L gives that fraction of $f_{EX}P_B$ irretrievably lost.

Because of the "snap-shot" nature of the time-averaged model, some degree of arbitrariness is associated with the description of the time sequence of energy transfer to and from the LTPHR; only a more complicated dynamical description will resolve this arbitrariness. The procedure adopted here fixes the compression field B_o at its maximum value. From the maximum field the total energy stored within the compression field can be computed. In order to sustain the compression field against resistive losses for an average burn period τ_B, the joule-loss energy is added to the energy stored in the compression field. The energy given to the plasma by the field is also included. Division of this sum by $(1 - f_{EX})$ and an appropriate cycle time τ_c gives P_B. By this artifice the joule losses are supplied and used at the beginning of the burn. The energy requirement of implosion heating is partially taken into account by starting the compression field from zero. Implosion energy stored externally to the coil bore is assumed to be part of the external loss, as are irreversible losses in the Energy Transfer and Storage System (ETS). Bremsstrahlung radiation losses should be small relative to end losses in a short-pulsed system and are not explicitly taken into account.

2.2.1 Implosion-Heating Energy.

Although the implosion heating energy requirement is not explicitly taken into account, a formalism is developed that monitors this quantity for a given implosion heating coil (IHC) configuration. An end-fed fractional-turn implosion heating coil of low inductance is envisaged for the LTPHR. Figure 2 depicts schematically the IHC and associated electrical-circuit equivalent. As seen from Fig. 2, the IHC is represented by a series of feed-plate L_{FP}, coaxial L_{cox}, and vacuum L_θ, inductances with the external return-flux inductance L_{EX} placed in parallel with L_θ. Referring to Fig. 2 for notation, the following expressions describe the series inductances,

$$L_{FP} = \frac{\mu_o}{2\pi} d_1 \ln\left(\frac{b_3}{b_1}\right) \tag{2-3}$$

$$L_{cox} = \frac{\mu_o}{2\pi} \ell \ln\left(\frac{b_2}{b_1}\right) \tag{2-4}$$

$$L_\theta = \frac{\mu_o N^2 \pi}{\ell} \left[\left(\frac{b_1 + b_2}{2}\right)^2 - a^2\right] \quad , \tag{2-5}$$

where a is the plasma radius ($\beta = 1$), $\mu_o = 4\pi \times 10^{-7}$ h/m and N is the IHC turns ratio,

$$N = \frac{2\ell\tan\alpha}{\pi(b_1 + b_2)} \tag{2-6}$$

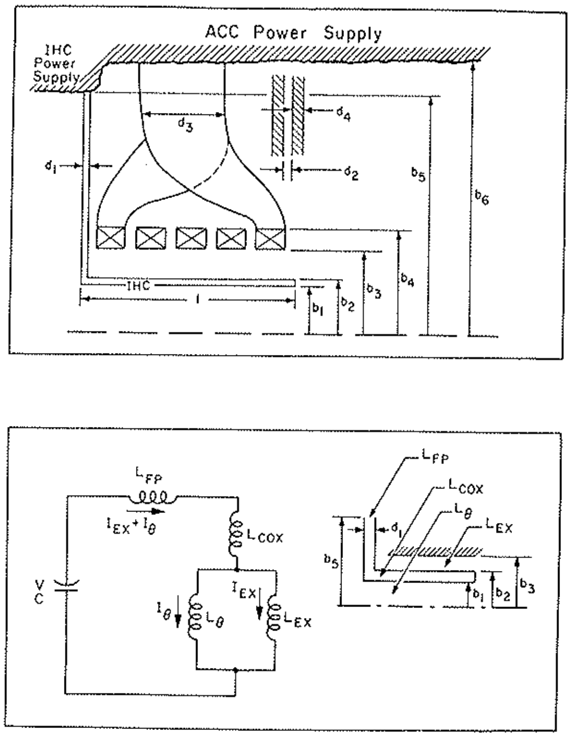


Fig. 2. Schematic diagram of Implosion Heating Coil (IHC) and equivalent electrical circuit, this notation used in text.

The current that is diverted into L_{EX} is related to the axial flux density in the return-flux path B_{SHO} by

$$I_{SHO} = B_{SHO}\ell/(\mu_o\mu_F N) \quad , \tag{2-7}$$

where μ_F is the permeability (in units of μ_o) of the material placed in the return-flux path. Hence, the energy stored in L_{EX} is given by,

$$W_{SHO} = \frac{L_{EX}}{2\ell}\left[\frac{B_{SHO}\ell}{\mu_o\mu_F N}\right]^2 = \frac{1}{2}\,\frac{B^2_{SHO}}{2\mu_o}\,\pi\left(b_3^2 - b_2^2\right) \quad . \tag{2-8}$$

Solving Eq. (2-8) for L_{EX} yields

$$L_{EX} = \frac{\mu_o\mu_F}{\ell}\,\pi\left(b_3^2 - b_2^2\right)N^2 \quad . \tag{2-9}$$

Defining L^{∞}_{IHC} as $L_{FP} + L_{cox} + L_\theta$ and letting B_{SHi} be the magnetic flux density inside the IHC, the total implosion heating energy requirement is given by

$$W_{SH} = \frac{L^{\infty}_{IHC}}{2\ell}\left(\frac{B_{SHi}\ell}{N\mu_o}\right)^2 + \frac{L_{EX}}{2\ell}\left(\frac{B_{SHO}\ell}{N\mu_o}\right)^2 \quad . \tag{2-10}$$

The requirement of flux conservation leads to

$$B_{SHO} = B_{SHi}\,\frac{b_1^2}{b_3^2 - b_2^2} \quad , \tag{2-11}$$

Hence,

$$W_{SH} = \frac{B^2_{SH}}{2\mu_o}\,\pi b_1^2\, f_{SH} \quad , \tag{2-12}$$

where B_{SH} refers to the implosion field at the plasma (i.e., B_{SHi}) and f_{SH} is given by

$$f_{SH} = \frac{\ell}{\mu_o\pi b_1^2 N^2}\left[L^{\infty}_{IHC} + \frac{L_{EX}}{\mu_F^2}\left(\frac{b_1^2}{b_3^2 - b_2^2}\right)^2\right] . \tag{2-13}$$

Expressed in terms of geometric quantities (Fig. 2), f_{SH} becomes

$$f_{SH} = \left(\frac{b_1 + b_2}{b_1}\right)^2 \frac{(d_1/\ell)\ln(b_5/b_1) + \ln(b_2/b_1)}{8\tan^2\alpha} + \left(b/b_1\right)^2\left[\left(\frac{b_1 + b_2}{2b}\right)^2 - x_s^2\right] + \frac{b_1^2}{\mu_F(b_3^2 - b_2^2)} \quad , \tag{2-14}$$

where x_s equals a/b, and b is the first-wall radius (not necessarily equal to b_1). The quantity f_{SH} gives the amount of inductive energy required to activate the IHC circuit per unit of inductive energy stored within the IHC inner radius, $W^o_{SH}=(B^2_{SH}/2\mu_o)\pi b_1^2$. In addition, the IHC power supply must provide the plasma internal energy after the implosion, $W^o_{INT}=W^o_{SH}\,\frac{3}{2}\,x_s^2(b/b_1)^2$ (for $\beta = 1$). Adding the latter requirement to the IHC energy leads to the following expressions for W_{SH} and f_{SH}*

$$W_{SH} = W^o_{SH}\, f_{SH} \tag{2-15}$$

$$f_{SH} = \left(\frac{b_1 + b_2}{b_1}\right)^2 \frac{(d_1/\ell)\ln(b_5/b_1) + \ln(b_2/b_1)}{8\tan^2\alpha} + \left(b/b_1\right)^2\left[\left(\frac{b_1 + b_2}{2b}\right)^2 + \frac{1}{2}\,\beta x_s^2\right] + \frac{b_1^2 - b^2}{b^2} + \frac{b_1^2}{\mu_F(b_3^2 - b_2^2)} \quad , \tag{2-16}$$

*In general, a plasma pre-heating efficiency η_{IH} can be defined as the ratio of plasma internal energy after preheating to the total energy transferred to the theta pinch. For the case of pre-heating by implosion to a plasma radius $x_s b$,

$$\eta_{IH} = \tfrac{3}{2}(b/b_1)^2 x_s^2\beta/f_{SH}$$

where, again $W_{SH}^{O} \equiv (B_{SH}^2/2\mu_o)\pi b_1^2$. Equations (2-15) and (2-16) are used to determine the implosion heating requirements in terms of given or optimized IHC geometric parameters.

2.2.2 Energy Stored in ACC and Leads. The adiabatic compression coil (ACC) is assumed to be fully transposed and lietzed, i.e., the current density within the ACC windings is uniform. Referring to Fig. 2, the compression field will have a radial profile described by

$$B(r) = B_o (r < b_3)$$
$$B(r) = B_o \frac{\delta_c - r/b_3}{\delta_c - 1} \quad , \tag{2-17}$$

where $\delta_c \equiv b_4/b_3$. Integrating $B^2(r)/2\mu_o$ over the ACC volume gives the maximum stored energy during the compression such that,

$$W_{BC} = \int_o^{b_4} (B^2(r)/2\mu_o) 2\pi r dr = W_{BCO} f_C$$
$$W_{BCO} = (B_o^2/2\mu_o)\pi b_3^2 \tag{2-18}$$
$$f_C = 1 + (\delta_c - 1)(\delta_c + 3)/6 \quad .$$

The compression field energy stored in the ACC leads (Fig. 2) is obtained by assuming a linear flux density in the leads [thickness d_4, width d_3 and length $(b_6 - b_4)$]. The magnetic energy stored within the leads is given by

$$W_{BL} = W_{BCO} f_L$$
$$f_L = \frac{(b_6 - b_4)d_4}{\lambda_{LZ}\pi b_3^2 N_{ACC}^2}\left[1 + \frac{2}{3}\frac{d_4}{d_3}\right] \quad , \tag{2-19}$$

where λ_{LZ} is the axial conductor filling density (d_3/ℓ) and N_{ACC} is the number of turns in the ACC. Combining Eqs. (2-18) and (2-19) gives the following expression for the energy required by the ACC.

$$W_{BC} = W_{BCO} f_{BC}$$
$$W_{BCO} = (B_o^2/2\mu_o)\pi b_3^2$$
$$f_{BC} = f_c(b_4/b_3) + \frac{(b_6 - b_4)d_4}{\lambda_{LZ}\pi b_3^2 N_{ACC}^2} f_L\left(\frac{d_3 + 2d_4}{d_3}\right) \tag{2-20}$$
$$f_C(y) = 1 + (y - 1)(y + 3)/6$$
$$f_L(y) = 1 + (y - 1)/3$$

Equation (2-20) gives the total compression energy under the assumption that compression starts from a null field. In actuality, the adiabatic compression will start with a field B_i, where $B_i = B_{SH}$ if implosion preheating is used. For the latter case B_o^2 should be replaced by $\left(B_o^2 - B_{SH}^2\right)$ in the expression for W_{BCO} (Eq. 2-20) and W_{SH} (Eq. 2-15) should be explicitly taken into account by the energy balance. Rather than introduce the complexities of two power supplies into the energy balance, W_{BC} accounts for the inductance actually added to the vacuum chamber by the implosion heating supply, but the remaining implosion-heating energy, $\sim W_{SHO}(f_{SH} - 1)$, is considered as part of the external loss (f_{EX} in Fig. 1). This aspect of the LTPHR energy balance is discussed in more detail in Sec. 2.2.4.

2.2.3 Joule Losses Incurred During Adiabatic Compression. A major loss incurred during operation of the LTPHR is expected to occur as resistive transport within the ACC and associated leads. Under the assumption of a fully transposed and lietzed ACC (i.e., uniform current density) the rate of joule transport loss is given by I^2R/ℓ where the current I is given by

$$I = \frac{B_o \ell}{\mu_o N_{ACC}} \quad , \tag{2-21}$$

and the resistances of the ACC and leads are given, respectively, by (Fig. 2),

$$R_{BC} = \eta N_{ACC}^2 \frac{\pi}{\lambda_c}\left(\frac{b_4 + b_3}{b_4 - b_3}\right) f_{AC}$$
$$R_{BL} = \eta \frac{2}{\lambda_L}\left(\frac{b_6 - b_4}{d_3 d_4}\right) f_{AC} \quad . \tag{2-22}$$

The quantities λ_C and λ_L are, respectively, the conductor filling fractions of the ACC and associated leads, and f_{AC} is the ratio of ac to dc resistance, which for a range of ACC dimensions, transpositions, and risetimes is shown in Fig. 3. Hence, the total rate of joule transport-current losses incurred during plasma compression is given by,

$$P_T = W_{BCO} f_T'$$
$$W_{BCO} = (B_o^2/2\mu_o)\pi b_3^2 \tag{2-23}$$
$$f_T' = \frac{\kappa}{\pi b_3^2}\left[\frac{2\pi(b_4 + b_3)}{\lambda_C(b_4 - b_3)} + \frac{4(b_6 - b_4)}{\lambda_{LZ}\lambda_L N_{ACC}^2 d_4}\right] f_{AC} \quad ,$$

where $\kappa = \eta/\mu_o$ is the electrical diffusivity of the ACC and conductor leads (for copper $\kappa = 0.014 m^2/s$ at room temperature). The total joule loss W_T is obtained by integrating P_T (i.e., B_o^2) over the burn period. For instance, if the compression-field waveform is described by a pure sine

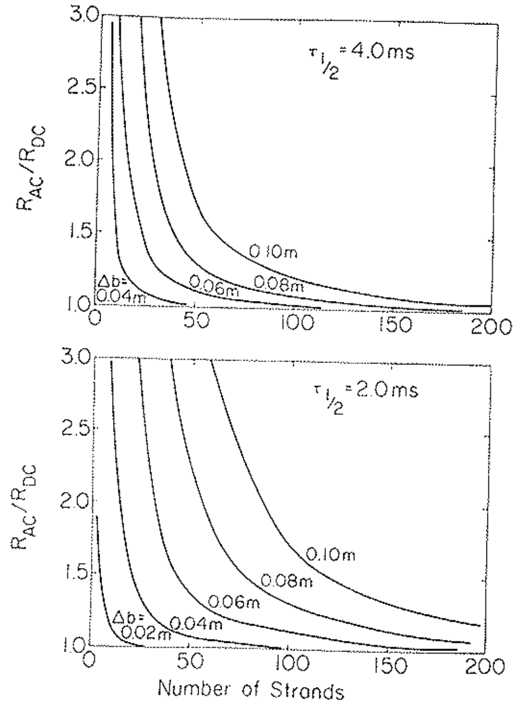


Fig. 3. Ratio of ac to dc resistances as a function of winding transposition, coil geometry and rise time.

of half-period $\tau_{1/2}$, the transport current loss W_T is given by,

$$W_T = W_{BCO}\, f_T \, , \qquad (2\text{-}24)$$

where $f_T = f_T^{*}(\tau_{1/2}/2)$.

2.2.4 Analytical Expression for η_B and External Losses. As seen from Fig. 1, η_B represents the fraction of the total stored energy $W_B = P_B\tau_c$ recovered by the energy transfer and storage (ETS) system. Of the fraction $(1 - \eta_B)$ irretrievably lost the fractions η_I, f_{BLK} and $f_{EX}f_L$ are distributed as plasma heating, blanket losses (mainly joule transport current losses in the ACC) and external losses (that part of the implosion heating energy stored externally to the coil bore and irreversible ETS losses). As noted previously, the non-dynamic nature of this model forces somewhat *ad hoc* assumptions to be made, although the resulting energy balance gives a reasonably accurate representation of the LTPHR (when compared to time-dependent burn calculations) and is much more flexible for parametric systems studies.

The energy transferred to the theta pinch *per se*, $W_B(1 - f_{EX})$, must be sufficient to satisfy the requirements of the ACC system [Eq.(2.20)], the transport current losses [Eq. (2-24)] and the plasma internal energy $[3\pi b^2x^2\beta(B_o^2/2\mu_o) = 3W_{BCO}(b/b_3)^2\beta x^2]$. Hence,

$$W_B(1 - f_{EX}) = W_{BCO}(f_{BC} + f_T + f_I/3) \, , \qquad (2\text{-}25)$$

where $f_I = 3(b/b_3)^2\beta x^2$ is the ratio of plasma internal energy to energy stored in a coil bore of radius b_3. The losses W_L that must be supplied to maintain the store W_B equals the sum of transport current losses, plasma losses, and irreversible external losses. Hence,

$$W_L = W_{BCO}(f_T + f_I) + f_{EX}f_LW_B \qquad (2\text{-}26)$$

Combining Eqs. (2-25) and (2-26) leads to the following expression for η_B, where $1 - \eta_B \equiv W_L/W_B$

$$1 - \eta_B = \frac{(f_I + f_T)(1 - f_{EX})}{f_{BC} + f_T + f_I/3} + f_{EX}f_L$$
$$= \eta_I + f_{BLK} + f_{EX}f_L \, . \qquad (2\text{-}27)$$

The previously defined quantities $\eta_I = f_I(1 - f_{EX})/(f_{BC} + f_T + f_I/3)$ and $f_{BLK} = f_T(1 - f_{EX})/(f_{BC} + f_T + f_I/3)$ (Fig. 1) represent the fraction of total energy switched to the theta pinch W_B that appears as plasma energy and blanket losses (joule losses in the ACC), respectively.

The energy balance thus far accounts only for W_{SHO} of the total implosion energy $W_{SHO}\,f_{SH}$ [Eq. (2-15)] transferred to the theta pinch. The energy $W_{SHO}(f_{SH}-1)$ residing outside the IHC bore is considered as a portion of the external loss, $f_{EX}f_LW_B$ (Fig. 1). In addition, the ETS system will have an inherent transfer efficiency η_{ETS}, and the loss $(1-\eta_{ETS})\,W_B$ is also considered to contribute to external losses, $f_{EX}f_L$. Under these approximations and definitions it follows that,

$$f_{EX}f_L = 1-\eta_{ETS} + \frac{W_{SHO}}{W_B}\,(f_{SH} - 1)$$
$$= 1-\eta_{ETS} + \frac{W_{SHO}}{W_{BCO}}\,\frac{(f_{SH} - 1)\,(1 - f_{EX})}{f_{BC} + f_T + f_I/3} \qquad (2\text{-}28)$$
$$= \frac{(1-\eta_{ETS})\,(f_{BC} + f_T + f_I/3) + (W_{SHO}/W_{BCO})\,(f_{SH} - 1)}{f_L\,(f_{BC} + f_T + f_I/3) + (W_{SHO}/W_{BCO})}$$

Although this formulation allows for external but reversible energy sinks, typically f_L is set equal to unity and f_{EX} can then be considered the total irreversivle external loss fraction. For most case being considered here $W_{SHO}/W_{BCO} \sim 10^{-3}$ and, hence, $f_{EX} \sim 1-\eta_{ETS}$.

2.3 Engineering Q-Value The previously described efficiency η_B gives all energy losses incurred in the transfer of energy $W_B = P_B\tau_c$, where τ_c is the cycle time (inverse of the pulse rate). Assuming for the moment that the 3.52 MeV alpha-particle energy is embedded in E and, therefore, is recoverable by the thermal cycle, the recoverable energy per pulse, $(P_E + P_E^*)\tau_c$, can be considered to be comprised of four major components:

$$\eta_I W_B \eta_{TH}$$

= plasma internal energy recovered either from bremsstrahlung radiation or plasma streaming from the ends of the theta pinch.

$$f_{BLK} W_B \eta_{TH}$$

= recovered joule losses occurring within the theta-pinch blanket (compression coil). If the theta-pinch coil is operated near room temperature, these joule losses cannot be recovered by the thermal cycle.

$$\langle J \rangle 2\pi b E e \eta_{TH} \tau_c$$

= energy deposited within the theta-pinch blanket by fusion neutrons.

$$\langle J \rangle 2\pi b E^* e \eta_{TH}^* \tau_c$$

= energy recovered from bred fissile fuel that is subsequently burned in a symbiotic fission burner.

Here, e is the electronic charge (1.602×10^{-19} J/eV), the time-averaged current of 14.08 MeV neutrons is $\langle J \rangle$, and the thermal-to-electric conversion efficiencies of the LTPHR and fission burner are respectively η_{TH} and η_{TH}^*. Alpha-particle thermalization is assumed not to occur in the burn times being considered. This approximation is examined in Sec. 2.3.2.

2.3.1 Neutron Yield Without Alpha-Particle Direct Conversion. If τ_B is the thermonuclear burn time for a plasma of radius $a = xb$ and density n, $\langle J \rangle$ is given by,

$$\langle J \rangle (n/m^2 s) = \frac{1}{8} \frac{a^2}{b} n^2 \langle \sigma v \rangle_{DT} \; \tau_B/\tau_c \quad , \tag{2-29}$$

where the ratio τ_B/τ_c is the duty cycle factor, and $\langle \sigma v \rangle_{DT}$ is the D-T fusion cross section averaged over the first moment of the Maxwellian distribution. The following approximate expression is used for $\langle \sigma v \rangle_{DT}$.[2]

$$\langle \sigma v \rangle_{DT} \; (m^3/s) = \frac{5.0 \times 10^{-16}}{T^{2/3}} e^{-19.94/T^{1/3}} \quad , \tag{2-30}$$

where the ion temperature T is given in keV units. The following pressure balance for an electron-to-ion temperature ratio λ relates the density n to compression field B_o.

$$n \, 10^3 e T(1 + \lambda) = \beta B_o^2/2\mu_o \quad , \tag{2-31}$$

where again temperature is expressed in keV.

For the purpose of examining the dependence of the engineering Q-value, Q_E, on system parameters, the "frequency-dependent" efficiency η_B has been introduced. In order to couple the system parameters to the plasma parameters, the neutron yield α(n/MeV) is defined as the number of fusion neutrons generated, $2\pi b \langle J \rangle \tau_c$, per unit of plasma internal energy, W_{INT}. Hence,

$$\alpha (n/MeV) = \frac{1}{6e} \frac{\beta}{(1 + \lambda)^2} \frac{B^2}{2\mu_o} \frac{\langle \sigma v \rangle_{DT}}{T^2} \tau_B \quad . \tag{2-32}$$

The parameter α is related to the neutron yield per unit of total invested energy, W_B, by

$$\alpha^* (n/MeV) = \eta_I \alpha \tag{2-33}$$

The following expression for Q_E results from these defined and calculable quantities,

$$Q_E = \frac{\eta_{TH}}{1-\eta_B} \left[\eta_I + f_{BLK}\Delta + \alpha^*(E + E^*) \right] \quad , \tag{2-34}$$

where $\eta_{TH} = \eta_{TH}^*$ has been assumed and $\Delta = 0$ or 1 depending if joule losses in the compression coil are recoverable by the thermal cycle.

2.3.2 Alpha-Particle Direct Conversion. The *ad hoc* assumption that the alpha-particle energy, E_α(3.52 MeV), is recovered totally by the thermal cycle (i.e., no plasma heating or direct-conversion plasma expansion, as for the RTPR[3]) is justified on the basis of slowing-down distances and times required for the plasma densities being considered [$(dE/dx) \sim 3.8 \times 10^{-21} n_e$ (keV/km)$\sim$38 keV/km for $n_e = 10^{22}$electrons/m^3].[4] In view of the benefits of alpha-particle heating, however, it seems prudent to examine this possibility insofar as Q_E [Eq. (2-34)] is concerned.

Using the previously defined quantities η_I [Eq. (2-27)] and α^* [Eq. (2-33)], the total plasma and alpha-particle energy equals $W_B(\eta_I + \alpha^* E_\alpha)$. If the fraction η_{DC} of the total alpha-particle energy $\alpha^* E_\alpha W_B$ is used to expand the plasma against the confining magnetic field, then the direct-conversion energy $\eta_{DC} E_\alpha \alpha^* W_B$ can be subtracted directly from the circulating energy requirement. The remaining energy $W_B[\eta_I + (1 - \eta_{DC})\alpha^* E_\alpha]$ streams from the ends of the theta pinch as end loss energy

W_{EL}, and is assumed converted to electrical energy with an efficiency η_{EL}. The energy $\eta_{EL}W_{EL}$ is added to the total electrical output, and the energy $(1 - \eta_{EL})W_{EL}$ is assumed to be sufficiently high in temperature to be used by the thermal-conversion system at an efficiency η_{TH}. Hence incorporating both plasma-expansion (η_{DC}) and end-loss (η_{EL}) direct conversion into the definition of Q_E,

$$Q_E = \frac{\eta_{TH}}{1-\eta_B}\left\{\left(1-\eta_{EL}+\eta_{EL}/\eta_{TH}\right)\left[\eta_I + \alpha^* E_\alpha\left(1-\eta_{DC}\right)\right] + f_{BLK}\Delta + \alpha^*(E+E^*)\right\} , \tag{2-35}$$

where

$$1-\eta_B = \eta_I + f_{BLK} + f_{EX}f_L - \eta_{DC}\alpha^* E_\alpha . \tag{2-36}$$

When both η_{DC} and η_{EL} equal zero,

$$Q_E = \frac{\eta_{TH}}{1-\eta_B}\left\{\eta_I + f_{BLK}\Delta + \alpha^*\left(E + E^* + E_\alpha\right)\right\} , \tag{2-37}$$

which is identical to Eq. (2-34) (where E_α is incorporated into E).

The major unknown at this point is the magnitude of η_{DC}, which in the absence of plasma end loss approaches a thermodynamically ideal value of ~66%. For the case where $\eta_{EL} = 0$ (i.e., the end loss energy is converted to electrical energy with efficiency η_{TH}), direct-conversion *via* plasma expansion increases Q_E by reducing $(1 - \eta_B)$, as seen from Eqn. (2-36). Since the other terms that comprise $(1-\eta_B)$ are of the order of 0.01-0.05, α^* will have to exceed 0.004 n/MeV before direct conversion of alpha-particle energy has any impact upon the energy balance (i.e., Q_E), For "injection" efficiencies η_I on the order 0.01, the number of neutrons per MeV of plasma energy, α, must exceed 0.4 (n/MeV) i.e., 2.5 x 10^{12} fusion reactions for each joule of plasma internal energy at peak compression). This yield is on the order of that computed for the RTPR (2.83 x 10^{12} n/J at peak field) and implies either commensurate burn times or higher compression fields (densities) for the linear theta pinch.

An accurate estimate of η_{DC} in the presence of end loss requires a two-dimensional, non-equilibrium, time-dependent model for the thermonuclear burn. Since such a computation is beyond the scope of this study, a highly approximate estimate must be made. The burn is assumed to occur over one end loss time τ_{EL}, which defines the assumed exponential decay of the particle line density N (particles/m) within the pinch. During this time the alpha-particle energy $\alpha^* E_\alpha W_B$ is assumed deposited in the plasma without either plasma expansion or end loss. The increased plasma temperature is then given by $T(1 + \alpha E_\alpha)$, where T is the average plasma temperature at which the burn was assumed to occur. The end loss of plasma energy then is assumed to occur for one end-loss time, η_{EL}, at a temperature $T(1 + \alpha E_\alpha)$. The plasma subsequently will expand or contract at a temperature $T(1 + \alpha E_\alpha)$, depleted line density N/e, and constant (maximum) compression field. The difference between initial and final plasma volumes, as determined by the pressure balance, when multiplied by the (constant) field pressure, gives the pressure-volume work, W_{DC}. This "direct-conversion" work may be positive or negative depending upon whether the plasma expands or contracts, which in turn depends upon the degree of alpha-particle heating and end loss. Finally, the plasma internal energy remaining after the plasma expansion is added to the initial estimate of the end-loss energy to give the total end-loss energy, W_{EL}. Given below are expressions for W_{DC}, W_{EL} and $\eta_{DC} = W_{DC}/(W_{DC} + W_{EL})$ based upon this simple model.

$$W_{EL}/(\eta_I W_B) = 1 + \tfrac{2}{3}(1 - e^{-1}) + (1 - \tfrac{2}{3}e^{-1})\alpha E_\alpha \tag{2-38}$$

$$W_{DC}/(\eta_I W_B) = \tfrac{2}{3}\left[\alpha E_\alpha - (1 - e^{-1})\right] \tag{2-39}$$

$$\eta_{DC} = \left(\tfrac{2}{3}\right) e^{-1}\,\frac{\alpha^* E_\alpha - (e-1)\eta_I}{\alpha^* E_\alpha + \eta_I} = \tfrac{2}{3}e^{-1}\,\frac{\alpha E_\alpha - (e-1)}{\alpha E_\alpha + 1} \tag{2-40}$$

Hence, before plasma expansion can occur (i.e., $\eta_{DC} > 0$), the neutron yield per unit of plasma internal energy must exceed 0.5 (n/MeV) or 3.1 x 10^{12} n/J. For much higher yields η_{DC} approach the ideal value of (2/3) reduced by e^{-1} = 0.3679. On the basis of the foregoing analysis, showing the severe degradation of plasma-expansion direct conversion by end losses, as well as the previously noted difficulties in thermalizing alpha particles in linear plasmas of reasonable length and density, the energy E_α will be incorporated into E and will be assumed converted to electrical energy with an efficiency η_{TH}; plasma heating and expansion *via* alpha-particle energy seems highly unlikely, unless much denser plasmas are considered. In essence, the computations reported pertain to a non-ignition device, where both alpha-particle direct conversion and plasma heating are assumed not to occur.

2.3.3 Average Plasma Temperature and Burn Time. Equation (2-34) is used to evaluate the engineering Q-value for a given set of plasma and system parameters. On the basis of a simple implosion/compression model,[3,5] the following relationships between B_o, x, T, initial filling pressure P_A (mTorr), and implosion field E_θ(kV/cm) can be derived

$$T(\mathrm{keV}) = 3.9848\, P_A^{-7/10}\, E_\theta^{3/5}\, B_o^{4/5} \tag{2-41}$$

$$x = 0.3351\, P_A^{3/20}\, E_\theta^{3/10}\, B_o^{-3/5} \ . \tag{2-42}$$

The equilibrium heat capacity ratio γ = 5/3 has been used.

The remaining relationship between burn time τ_B and $\tau_{1/2}$ is yet to be specified. First, τ_B is equated to $\tau_{1/2}$ reduced by a form factor f_T which reflects the fact that the thermonuclear burn does not occur as a function of time with equal vigor for a given magnetic field waveform; for a purely sinusoidal waveform $f_\tau \sim 3/8$. Secondly, $\tau_{1/2}$ is equated to an end loss time, τ_{EL}, as determined by a free-streaming model

$$\tau_{EL} = \sqrt{\frac{m_i}{2kT_i}}\ \frac{\ell}{2}\ \eta_{ELS} = \sqrt{\frac{m_i/e}{2\times 10^3 T}}\ \frac{\ell}{2}\ \eta_{ELS} \tag{2-43}$$

The length of the LTPHR is ℓ, and η_{ELS} is an end-loss parameter equal to $4\sqrt{\pi}\, R/(1-\sqrt{1-\beta})$, where R is the applied mirror ratio (assumed to be unity). A more recent numerical evaluation[6] gives η_{ELS} as a function of β. For $\beta = 0.8$, $\eta_{ELS} = 4.9$ for the old theory compared to 3.9 for the more recent numerical evaluation. Experimental evidence from short theta pinches for R = 1 indicates that both theories overpredict by $\sim$ 60%. Since experimental evidence is based on short theta pinches and, therefore, may not be applicable to the longer devices being considered here, the η_{ELS} values derived from the more recent numerical evaluation are used.

The foregoing relationships render a complete set of equations by which the LTPHR energy balance can be evaluated. For given values of P_A, E_θ and B_o, the plasma radius a = xb and temperature can be determined. Given f_{EX}, f_L, and β the efficiency η_B can be determined for a given theta-pinch coil design, once the burn time τ_B is specified. The length of the theta pinch specifies τ_{EL}, and hence τ_B for a given form factor f_T. Finally, α^* and hence Q_E can be determined for a given burn time, i.e., length ℓ.

The foregoing analysis will yield the LTPHR energy balance on a "per pulse" basis. To determine the pulse rate, the first-wall neutron loading I_W(MW/m^2) must be specified. The following relationship exists between I_W and cycle time τ_C.

$$I_w(\mathrm{MW/m^2}) = \alpha^*(14.08)\, \frac{W_B}{2\pi b \tau_c} \simeq \frac{(1-\eta_B)}{M Q_{TH}}\, Q_E\, \frac{W_B}{2\pi b \tau_c}\ , \tag{2-44}$$

where the energy multiplication M is given by (E + E*)/14.08. Hence, given I_W the cycle time τ_c can be determined for a given LTPHR design.

3. DESCRIPTION OF PARAMETRIC ANALYSIS AND MAJOR CONSTRAINTS

In view of the formidable list of variables which impact on the overall performance and design of the LTPHR, a logic sequence was adopted for the system analysis. Table I summarizes the more important relationships between LTPHR variables, and Table II lists the definitions, units, and status of key parameters. The logic flow diagram shown in Fig. 4 illustrates the computational procedure, wherein given or assumed parameters are used to evaluate the LTPHR energy balances.

TABLE I

SUMMARY OF LTPHR DESIGN RELATIONSHIPS

Final plasma to first-wall radius	x = 0.3351 [illegible]
Final ion temperature (keV)	[illegible]
End-loss time (s)	[illegible]
Half-cycle time (s)	[illegible]
Average time during which currents are applied (s)	[illegible]
Total energy switched from [illegible] power supply (J/m)	[illegible]
Energy stored in [illegible] bank under [illegible] conditions (J/m)	[illegible]
Total energy in [illegible] at peak compression (J/m)	[illegible]
Total Joule loss in [illegible] (J/m)	[illegible]
Total energy in plasma at peak compression	[illegible]
Fraction of W_B that enters the plasma	[illegible]
Fraction of W_B lost as Joule heating within blanket	[illegible]
Fraction of transferred but externally stored energy	[illegible]
Fraction of transferred but externally stored energy lost per cycle	[illegible]
Fraction of W_B recovered each cycle	[illegible]
Neutron yield per unit of plasma energy (n/MeV)	[illegible]
Neutron yield per unit of switched energy (n/MeV)	[illegible]
Engineering Q-value [illegible]	[illegible]
First-wall, 14.1-MeV neutron wall loading (MW/m²)	[illegible]

The values for the fixed parameters on Table II and Fig. 4 were selected either on the basis of values known or estimated to be achievable in the near future. The electrical properties of the compression coil were assumed those of pure copper at room temperature (most optimum); the first-wall radius is large enough to assume effective implosion heating, but small enough to minimize stored energy requirements; and a thermal neutron blanket characterized by high E + E* values, but low fissile fuel production was used.[1]

TABLE II

SUMMARY OF KEY LTPHR TIME-AVERAGED PARAMETERS

Designation	Definition	Units	Value
P_A	filling density	mTorr	variable
E_θ	azimuthal implosion electric field	kV/cm	variable
B_o	maximum compression magnetic field	T	variable
λ	electron-to-ion temperature ratio	--	~ 1.0
β	plasma beta	--	0.8
f_{EX}	fraction W_B stored externally	--	variable
b_1	first-wall radius	m	variable
d_5	IHC flux return path $(b_3 - b_2)$	m	variable
d_6	ACC thickness, $(b_4 - b_3)$	m	variable
κ	electrical diffusivity of coil conductor	m^2/s	0.0127
f_{ST}	ACC structural volume fraction	--	variable
ℓ	LTPHR length	m	variable
ℓ^*	effective LTPHR length (w/end-stoppering)	m	variable
η_{TH}	thermal conversion efficiency	--	0.4
η_{ETS}	ETS efficiency	--	0.95
E+E*	total energy worth of fusion neutron (MeV/n) for $(b_4 - b_3) = 0$	--	447
E/ (E*+E)	fraction of total energy released as sensible heat to LTPHR, $(b_4 - b_3) = 0$	--	0.906

The major variables are P_A, E_θ, B_o, b_1, (b_3-b_2), (b_4-b_3) and ℓ.

For given values of E_θ, B_o, b_1, and ℓ, Q_E is determined as a function of P_A, and the maximum value of Q_E is obtained. The reason for this maximum is clearly illustrated from Eqs. (2-41) and (2-42), which indicate that x increases with P_A but T decreases with increasing P_A. Once the optimum Q_E (with respect to filling pressure) is determined, Q_E is then evaluated as a function of coil dimensions,

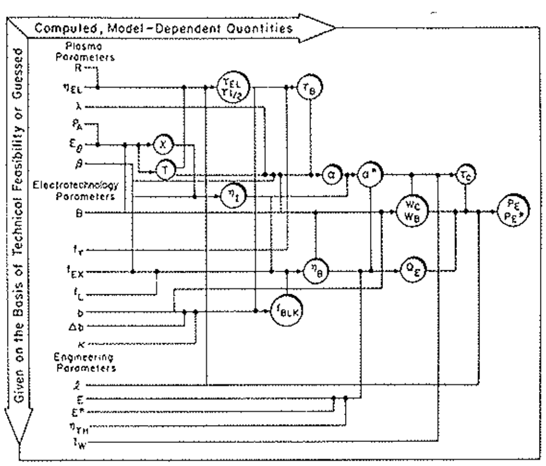


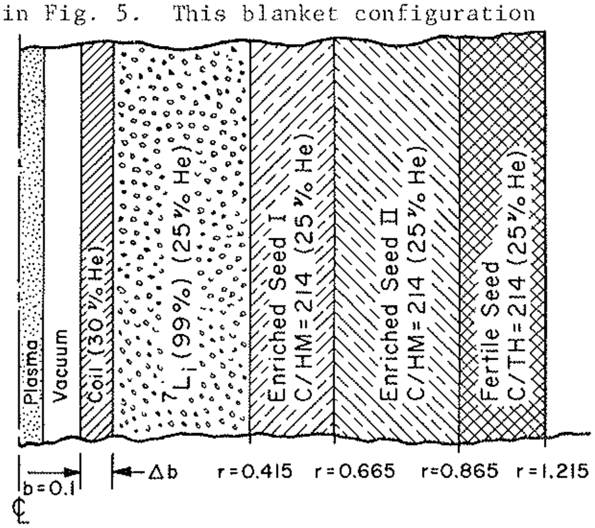
Fig. 4. Logic diagram of LTPHR parameter systems study. Refer to text for notation.

E_θ (2 - 4 kV/cm), B(5-30T) and ℓ(1-10 km). Given Q_E and I_W, the cycle time τ_c and the power levels P_E and P_E^* can be computed (Fig. 4). It should be noted that the optimum Q_E *vs* P_A values will differ from that for the neutron yield α. Referring to Eq. (2-23) α(n/MeV) optimizes at T = 15.48 keV for a fixed compression field. Hence, Eq. (2-45) predicts $E_\theta^6 B^8 / P_A^7 = 1.54 \times 10^6$ for the α(n/MeV) optimum. For E_θ = 2 kV/cm and B = 20 T, P_A equals 5.92 mtorr, and the corresponding value of x [Eq. (2-42)] is 0.086. Because of the importance of $\alpha^* = \alpha\eta_I$ in determining Q_E, the Q_E *vs* P_A optimum will demand higher values of x and hence higher values of P_A. End-loss reduction (achievable by mirroring, cusps or some other method) is modeled by considering an effective length $\ell^* > \ell$ for the device such that the end-loss time is increased to $\tau_{EL}^* = (\ell^*/\ell)\tau_{EL}$, where τ_{EL} is given by Eq. (2-43).

The complexity of the Q_E optimization necessitates a numerical evaluation that is subject to neutronic and mechanical constraints. Lastly, a simple economic model is described which is used in conjunction with the overall parametric study. These latter constraints are described below.

3.1. NEUTRONIC CONSTRAINTS

A thorough parametric analysis of the neutronic design for the LTPHR blanket is not intended by this study. The blanket model used in these analyses is similar to a high-multiplying system that was previously reported and is illustrated in Fig. 5. This blanket configuration



Thermal Blanket Model (Series 400)

Fig. 5. Diagram of neutronic model used for LTPHR blanket.

generates a highly thermalized neutron spectrum (thick ^{7}Li region immediately adjacent to the first-wall ACC) and is used to model important neutronic requirements (high energy multiplication, $(E + E^*)_0 \sim 477$ MeV/n), and tritium breeding, ($[BR] \geq 1.$) as well as the interaction of these requirements with the ACC design. Since $E^*/E \sim 0.1$, this particular design would be a poor choice for pure fissile-fuel production (fuel factory) with a minimum fissile-fuel ^{233}U inventory. Neutronic optimization to enhance E^*/E (fuel production) is beyond the scope of this study.

The dependence of $E + E^*$ on ACC thickness $\Delta b = (b_4 - b_3)$ for three coil materials is shown in Fig. 6, and Fig. 7 illustrates the decrease in the tritium breeding ratio [BR] and ^{233}U conversion [CV] with increased Δb. The total energy worth of the fusion neutron can be decribed by the empirical relationship,

$$(E + E^*)/(E + E^*)_0 = 1.0 - 1.95\Delta b, \tag{3-1}$$

for the range of ACC thickness of interest, where $(E + E^*)_0 = 477$ MeV/n. This decrease in $E + E^*$ with increased Δb is included in the evaluation of Q_E. As with the joule losses, neutron energy deposited into the ACC is assumed lost from the system if $\Delta = 0$ in Eq. (2-34).

3.2 STRUCTURAL CONSTRAINTS

In addition to the aforementioned neutronic [Eq. (3-1)] and joule-loss [Eq. (2-24)] constraints, obvious structural limitations must be imposed upon the ACC and, therefore, on the overall energy balance. A detailed structural analysis of

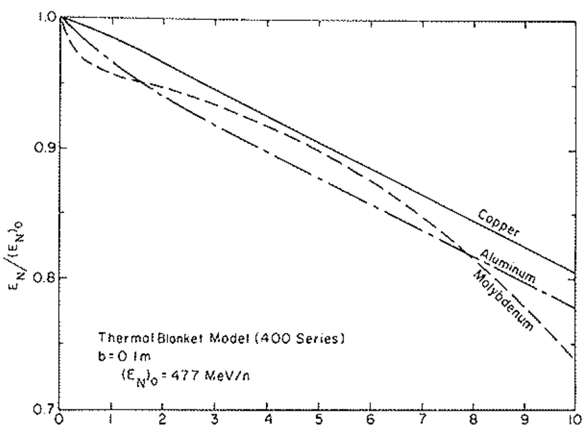


Fig. 6. Dependence of total (real E and virtual E*) energy multiplication, E+E*, on the first-wall ACC thickness for various coil materials (Ref: Fig. 5).

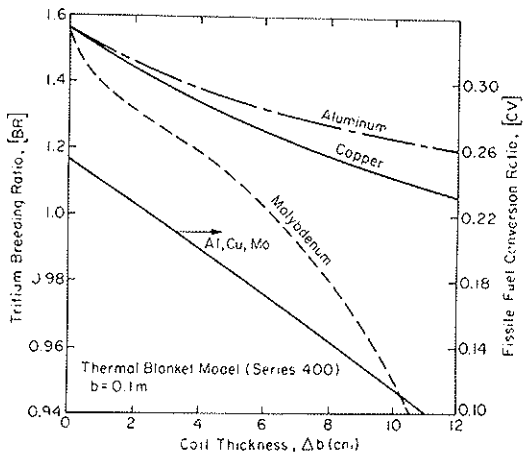


Fig. 7. Dependence of tritium breeding ratio, [BR], and fissile conversion ratio, [CV], on the first-wall ACC thickness for various coil materials (Ref: Fig. 5).

the ACC has been reported,[7,8] and only essential elements are summarized here. Utilizing the elasticity theory solutions for a thick-walled cylinder,[9] the radial and tangential (hoop) stresses are given, respectively, by

$$\sigma_r = P^* \frac{1 - (b_1/r)^2\delta^2}{\delta^2 - 1}$$
$$\sigma_\theta = P^* \frac{1 + (b_1/r)^2\delta^2}{\delta^2 - 1} \quad , \tag{3-2}$$

where $\delta = 1 + \Delta b/b_3 = b_4/b_3$ (Fig. 2), and P^* is an effective pressure that depends on the magnetic field and the dynamic loading. After determining the strain energy for a given coil of length ℓ, modulus E and Poisson's ratio ν, the energy principle was used[7] to derive an equivalent stiffness k_r for motion at the outer coil radius b_4

$$k_r = \pi\ell E \frac{(\delta^2 - 1)}{2(\delta^2 + 1)} \left[(1 - \nu) + (1 + \nu)\delta^2\right] . \tag{3-3}$$

A force balance in the absence of damping gave the following dynamic equation for radial displacement,

$$\left(\frac{\tau_M}{2\pi}\right)\ddot{u} + u = \xi(t) \quad , \tag{3-4}$$

where the mechanical period $\tau_M = 2\pi\sqrt{M_t/k_r}$, M_t is the displaced mass, and $\xi(t)$ is a (sinusoidal) driving function. The mass M_t can be that of the blanket plus coil (coupled) or that of the coil alone (uncoupled). Using shock-spectrum techniques,[10] the maximum dynamic amplification/attenuation factor $\gamma(\tau_{1/2}/\tau_M)$ can be derived[7] and used to determine the effective pressure, $P^* = (B_0^2/2\mu_0)\gamma$. The maximum hoop stress is evaluated at $r = b_1$ from Eq. (3-2),

$$\hat{\sigma}_\theta = \left(B_o^2/2\mu_o\right)\gamma\,\frac{\delta^2+1}{\delta^2-1} \quad . \tag{3-5}$$

The dependence of $\hat{\sigma}_\theta$ on pulse duration (and hence length of LTPHR) is reflected through the parameter $\gamma(\tau_{1/2}/\tau_M)$.

Numerical evaluation of k_r, τ_M, and subsequently $\gamma(\tau_{1/2}/\tau_M)$ is based upon a composit ACC comprised of custom 455 stainless steel (structural component) and and Cu/Be (conductor). Reference 7 describes the method used to arrive at average mechanical properties. The volume fraction of structure in the ACC is designated f_{ST}, and the following relationship was used to compute the composite electrical resistivity for the ACC.

$$\langle\eta\rangle = \frac{\eta_{ST}\,\eta}{f_{ST}\eta + \left(1 - f_{ST}\right)\eta_{ST}} \quad , \tag{3-6}$$

where the electrical resistivities for copper and structure are respectively η and η_{ST}. Table III summarizes the electrical and mechanical properties of the ACC, the yield, ultimate and endurance ($\geq 10^7$ cycles) strengths being designated by σ_{YLD} σ_{ULT}, and σ_{END} respectively. The custom

TABLE III

MECHANICAL AND ELECTRICAL PROPERTIES OF ACC

Property	Custom 455 Stainless Steel	Cu/Be Alloy
E (MPa)	2×10^5	1.3×10^5
ν	0.29	0.355
σ_{YLD} (MPa)	1634.	172-241 (annealed) 896-1034 (hardened
σ_{ULT} (MPa)	1689.	414-552 (annealed) 1136-1276 (hardened)
σ_{END} (MPa)	750 (tension-compression) 965 (tension-tension)	---
η ($\mu\Omega$-m)	0.75	0.0482 - 0.0582
ρ (kg/m^3)	7.9×10^3	8.9×10^3

455 stainless steel was used to clad the multifilamentary Cu/Be electrical conductor and was assumed to support entirely the mechanical load. An optimum amount of structural material, as reflected by f_{ST} is expected. Small amounts of structure will minimize joule losses, but the maximum allowable field will be severely limited by strength considerations, whereas increased values of f_{ST} will permit higher compression fields at the expense of increased joule losses. This tradeoff as well as those associated with pulse duration [i.e., stress attenuation ($\tau_{1/2} << \tau_M$) *vs* stress amplification ($\tau_{1/2} \sim \tau_M$) *vs* steady-state stresses ($\tau_{1/2} >> \tau_M$)] and coil geometry (strength *vs* neutronics *vs* joule losses) is examined.

3.3 ECONOMIC CONSTRAINTS

The value of Q_E where economic breakeven occurs must be determined by a detailed cost analysis of a particular system. The design of the LTPHR is not sufficiently advanced for such an analysis, but a preliminary assessment of the Q_E *vs* cost relationship can be made and used to provide an indication of the magnitude of Q_E required for economic breakdown.

As previously noted[1] a significant quantity of intrinsic energy is expected to be generated by a hybrid reactor, even for blanket designs that minimize *in situ* fissioning and enhance fissile fuel breeding (i.e., $E/E^* \sim 0$). Preliminary estimates[1] indicate that as a minimum for every unit of potential energy associated with bred fissile fuel E^*, 0.15 units of energy E will be deposited into the blanket of the hybrid reactor, i.e., $E/E^* \geq 0.15$. Hence, in addition to the relationship between Q_E and economic considerations, a question arises as to the economic incentive for converting this energy E versus rejection as waste heat. The results of the following analysis give the relationship between Q_E and $(\text{REVENUES/COSTS})_\Delta$, where $\Delta = 0$ or 1 gives the waste *vs* convert option; this simple analysis generally can be applied to any fission/fusion reactor concept.

The capital cost of the hybrid reactor is divided into a cost of the nuclear island C_{NI}, a cost of the power conversion equipment C_{PC}, and cost of non-nuclear/non-conversion components C_{NN}. These costs are expressed in units of \$/kWe. Since the hybrid reactor will produce significant quantities of fissile fuel, a fuel handling cost C_{FH} (\$/kg/y) is also envisaged. A significant capital cost may be incurred for the energy storage required for both the implosion heating and adiabatic compression. The cost of the IHC energy store is C_{IH}(\$/J), and the corresponding cost for the ACC store is C_{AC} (\$/J). Lastly, certain LTPHR blanket designs envision a fissile fuel (^{233}U) inventory for which carrying charges will be incurred. The above-described costs are summarized below:

$$\text{POWER-RELATED COST (\$)} = (C_{NI}+C_{NN}+C_{PC}W)\,P_E(10)^3 \tag{3-7A}$$

$$\text{FUEL HANDLING COSTS (\$)} = C_{FH}R \tag{3-7B}$$

$$\text{IHC ENERGY STORAGE COSTS (\$)} = C_{IH}W_{SH}(10)^6 \tag{3-7C}$$

$$\text{ACC ENERGY STORAGE COSTS (\$)} = C_{AC}W_B(10)^6 = \frac{C_{AC}(10)^6(1+E^*/E)P_E\tau_c}{Q_E\;1-\eta_B\Theta} \tag{3-7D}$$

$$\text{COST OF FUEL INVENTORY (\$)} = I_{23}c_f = [DT]\,Rc_f \quad , \tag{3-7E}$$

where I_{23}(kg) is the ^{233}U fuel inventory,

[DT] (y) is the intrinsic fissile-fuel doubling time, and R(kg/y) is the fissile fuel production rate. Table IV summarizes the cost definitions and numerical values used in this analysis.

TABLE IV

SUMMARY OF COST TERMS AND VALUES USED IN ANALYSIS

Symbol	Definition	Value
	Cost of nuclear core and primary coolant system	250 \$/kWe
	Cost of secondary coolant system[(a)]	50 \$/kWe
C_{NI}	Cost of the nuclear island	300 \$/kWe
	Cost of turbine set	100 \$/kWe
	Cost of turbine building	100 \$/kWe
C_{PC}	Cost of power conversion[(b)]	200 \$/kWe
	Cost of miscellaneous buildings	40 \$/kWe
	Cost of site and miscellaneous equipment	10 \$/kWe
C_{NN}	Cost of non-nuclear equipment[(c)]	50 \$/kWe
C_{FH}	Cost of fuel handling[(d)]	50 \$/kg/y
I	Annual interest rate	0.15/y
T	Debt payoff time	10 y
ϵ	Escalation factor	0.3
c_p	Cost of power (30 mill/kWe h)	2.62×10^5 \$/MWe y
c_F	Cost to fission burner of fuel[(e)]	$\sim 0.2\ c_p$ [(11)]
C_{AC}	Cost of ACC energy store	0.01 \$/J
C_{IH}	Cost of IHC energy store	0.25 \$/J

[(a)] A secondary coolant system will be required whether the sensible heat is converted to electrical energy or deposited as waste heat to the environment. For the former case the cost of the secondary coolant system will primarily be associated with the steam generator, whereas for the latter case a substantial cooling-tower cost will be incurred. In any event the cost of the nuclear island C_{NI} is assumed to be the sum of items (1) and (2).

[(b)] Taken to be the sum of Items (4) and (5).

[(c)] Taken to be the sum of Items (7) and (8).

[(d)] Cost associated with the storage manipulation, and processing of radioactive nuclear fuels. This represents a base cost for the hybrid reactor which generates no *in situ* power. Part of the core and shielding costs are included in C_{FH}, as well as in C_{NI}. The numerical value selected for C_{FH} is ~15% that incurred by the fuel reprocessing industry and is assumed approximately equal to the anticipated cost for a zero-power hybrid reactor. A fuel reprocessing plant at Morris, Illinois, was to have had a capacity of 520 Tonne/y and will cost ~380 \$/kg/y, whereas a larger plant at Barnwell, South Carolina, will treat 1560 Tonne/y and is expected to cost ~260 \$/kg/y.

[(e)] The units of c_F used herein are \$/kg. Since, for $\eta_{TH} \sim 0.4$, 1 kg of fissile fuel will generate 1.01 MWe y of electrical energy. The units \$/kg and \$/MWe y are used interchangeably.

The major annual cost anticipated for a hybrid reactor will be associated with the power cost, the cost of energy storage and the cost of the capital investment. On the basis of the definitions given in Table IV, the annual cost of investment is given by

$$\text{ANNUAL COST OF INVESTMENT (\$/y)} = (1 + IT + \epsilon)\Big[\left(C_{NI} + C_{NN} + C_{PC}\xi\right)P_E \times 10^3 + C_{FH}R + \left(C_{AC}W_B + C_{IH}W_{SH}\right)(10)^6 + [DT]Rc_f\Big]/T \quad . \qquad (3\text{-}8)$$

Since P_E(MWe) can be expressed in terms of the fissile fuel production rate R as follows ($\eta_{TH} = 0.4$),

$$P_E(\text{MWe}) = 1.01(\text{MWe y/kg})\ R(E/E^*) \quad , \qquad (3\text{-}9)$$

the annual cost of investment becomes,

$$\text{ANNUAL COST OF INVESTMENT (\$/y)} = \frac{R}{T}(1 + IT + \epsilon)\left[\left(C_{NI} + C_{NN}\right)\left(1 + \frac{C_{PC}\xi}{C_{NI} + C_{NN}}\right)10^3(E/E^*) + C_{FH} + \frac{10^6(1 + E/E^*)\xi}{\left(1 - \eta_B\right)Q_E}\left(C_{AC} + \xi C_{IH}\right) + [DT]c_f\right] \quad . \qquad (3\text{-}10)$$

where ξ is the ratio of implosion to compression stored energy. Likewise the circulating power cost is given by,

$$\text{POWER COST (\$/y)} = 1.01(\text{MWe y/kg})\ R(E/E^*)c_p/q_E = \left(Rc_p/Q_E\right)(1 + E/E^*) \quad . \qquad (3\text{-}11)$$

where q_E is the intrinsic Q-value for the LTPHR ($q_E = P_E/P_c$). The total cost, therefore, equals the sum of expressions (3-10) and (3-11).

The revenue generated by the hybrid reactor equals the sum of the fuel revenue, Rc_f, and the power revenue, $P_E c_p$. With the aid of Eq. (3-9), the costs and revenues are given by

$$[\text{COST}]_\Delta = Rc_p\frac{E}{E^*}\left\{\frac{1 + E^*/E}{Q_E} + \frac{1 + IT + \epsilon}{c_p T}\left[\left(C_{NN} + C_{NI}\right)\left(1 + \frac{C_{PC}\xi}{C_{NI} + C_{NN}}\right)10^3 + C_{FH}\frac{E^*}{E} + \frac{10^6(1 + E^*/E)\xi}{\left(1 - \eta_B\right)Q_E}\left(C_{AC} + \xi C_{IH}\right) + \frac{E^*}{E}[DT]c_f\right]\right\} \qquad (3\text{-}12A)$$

$$[\text{REV}]_\Delta = Rc_p\frac{E}{E^*}\left[\frac{c_F}{c_p}\frac{E^*}{E} + \Delta\right] \quad , \qquad (3\text{-}12B)$$

where again $\Delta = 0, 1$ represents the waste *vs* convert option for the *in situ* energy generated by the hybrid reactor. Equations (3-12) are evaluated in conjunction with the LTPHR energy balance. In addition to the "economic Q-value," $[\text{REVENUE/COST}]_\Delta$, capital costs on a per kWe basis are also estimated. It should be noted that [COST] represents a lower bound in that the costs of contingencies, operating/maintenance, etc. have not been included.

4. RESULTS OF PARAMETRIC SYSTEMS STUDIES

4.1 TIME-INDEPENDENT PARAMETER SURVEY

The approach employed to explore the LTPHR parameter space (Fig. 4) and to identify a near-optimum design point was to vary systematically the several independent geometric parameters with a view toward achieving a favorable energy balance (i.e., maximizing Q_E) subject to the economic and structural constraints described in previous sections. Results are reported for both the nominal implosion electric field value of $E_\theta = 2$ kV/cm and a more opmistic value of 4 kV/cm.

The energy transfer and storage efficiency is fixed at $\eta_{ETS} = 0.95$, and an effective length ℓ^* was varied, where ℓ^* equals the true LTPHR length ℓ multiplied by the endloss rate reduction relative to that predicted by free-streaming particle loss (τ_{EL}, Eq. (2-43)). The calculational approach adopted here is summarized below.

a) For $B_o = 20$ T and $E_\theta = 2$ or 4 kV/cm, the optimum first-wall radius b_1 was determined *vs* ℓ^* for a fixed IHC flux return thickness $d_5 = b_3 - b_2$, fixed ACC coil winding thickness $d_6 = b_4 - b_3$ and fixed amount of support structure f_{ST} incorporated into the ACC.
b) For the value of b_1 selected in a) the optimum value of $d_5 = b_3 - b_2$ was determined.
c) Given the optimum values of b_1 and d_5, the optimum value of ACC coil thickness d_6 was determined.
d) Once b_1, d_5 and d_6 were determined, an optimum value of ACC structural fraction f_{ST} was computed.
e) Given these optimal choices for key geometric parameters, Q_E *vs* ℓ^* and B_o was determined and the structural (endurange limit) and economic (REV/COST) constraints were imposed.

For all computations the filling density p_A selected was that which maximized Q_E, this p_A value not necessarily resulting in the maximum neutron yield.

As a starting point the first-wall radius b_1 was varied for $B_o = 20$ T, $d_5 = 0.05$ m, $d_6 = 0.05$ m and $f_{ST} = 0.5$. As seen from Figs. 8a-b, Q_E increases with b_1 and ℓ^* until saturation is reached, the saturation occurring at higher values of b_1 and Q_E as ℓ^* increases. The approach towards a limiting value of Q_E is a result of joule losses [f_{BLK} or f_T, Eq. (2-27)] becoming a smaller fraction of the total losses, with the external losses (mainly η_{ETS}) eventually dominating for large values of b_1. The rate at wich the external losses overcome the joule transport losses as b_1 increases depends upon the time during which the ACC is activated (i.e., burn time or ℓ^*). Consideration of the ACC stress constraint will inhibit

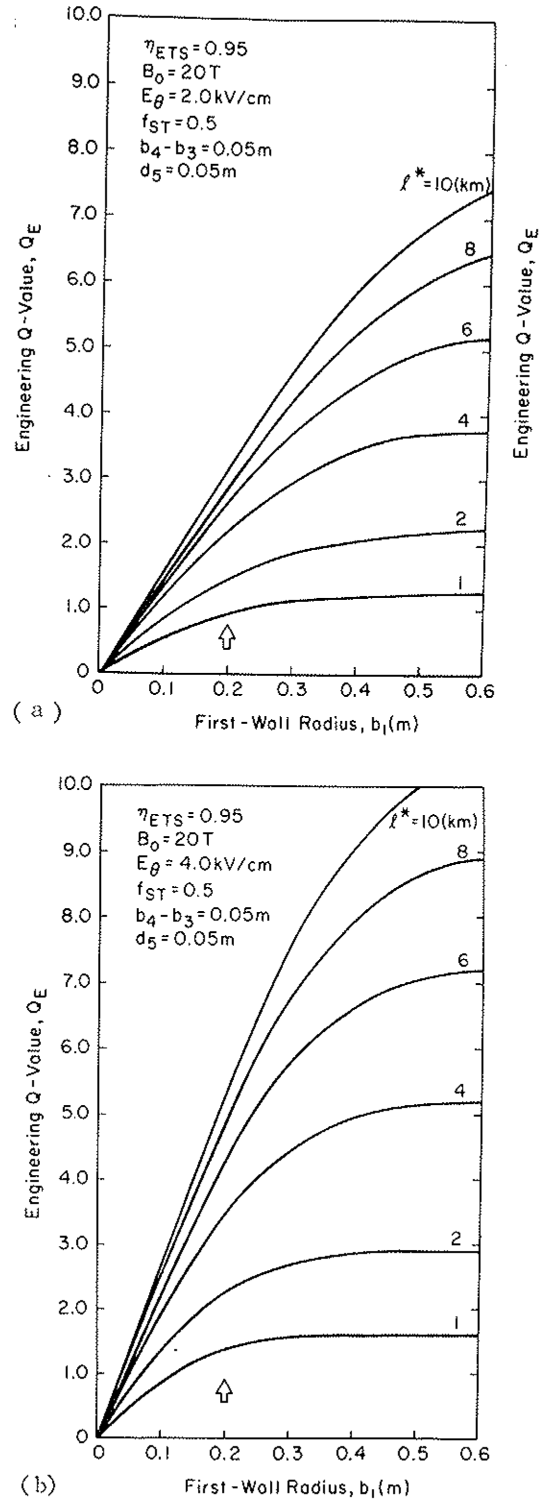


Fig. 8. Dependence of Q_E on first-wall (IHC) radius b_1 and effective LTPHR length ℓ^* for implosion fields E_θ=2 and 4 kV/cm. Values of other parameters represent a first guess. Arrow indicates choice of b_1.

operation at the higher b_1 values, however, so a conservative value of $b_1 = 0.2$ m was selected for subsequent calculations. It may also be noted from Figs. 8a-b that if a detailed structural design permits larger values of b_1, gains in Q_E can be realized, particularly for the larger ℓ^* values. Using this choice for b_1, the thickness of the IHC flux return path ($d_5 = b_3 - b_2$) was varied as indicated in Figs. 9a-b. Above a minimum value of ~ 0.01 m, Q_E drops with increasing values of d_5, suggesting a choice of $d_5 = 0.02$ m. However, to allow for modest shifts in the peak values, d_5 was set equal to 0.03 m

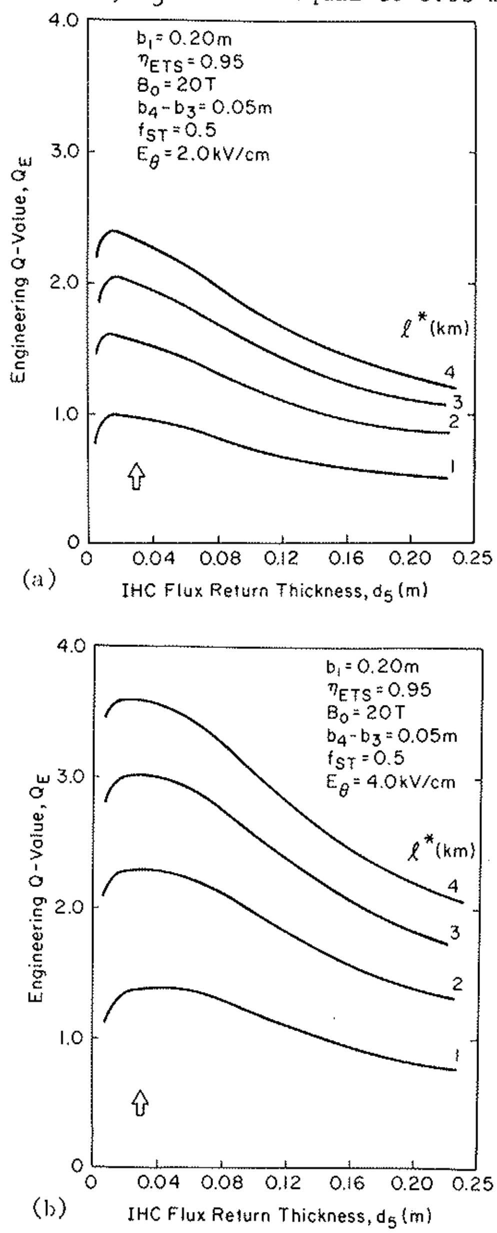


Fig. 9. Dependence of Q_E on IHC flux return path thickness (b_3-b_2). Arrow indicates choice of (b_3-b_2).

for subsequent calculations. The maxima indicated in Figs. 9a-b can be explained as follows. For very small IHC flux return area, the return flux density is very high (compared to that in the IHC bore), the magnetic material in the flux-return path saturates, and considerable implosion energy must be stored external to the IHC to generate a given implosion field. This problem is resolved by increasing $d_5 = b_3 - b_2$ at the expense of increased ACC bore b_3 and associated stored energy. Hence, the optima given in Fig. 9a-b are generated.

With $b_1 = 0.2$ m, $d_5 = 0.03$ m the effects of ACC thickness $d_6 = b_4 - b_3$ were investigated. As seen from Figs. 10a-b, Q_E is insensitive to the ACC thickness beyond $d_6 = 0.03$ m. For small values of d_6 Q_E is reduced because of the increased joule losses, whereas increasing d_6 results in a degradation of the neutron energy worth, $E+E^*$ (blanket is located radially outward from the ACC). Both the sharpness and the location of the maxima shown on Figs. 10a-b will depend upon the maximum compression field B_o and ACC bore b_3. The locus of optimum values of

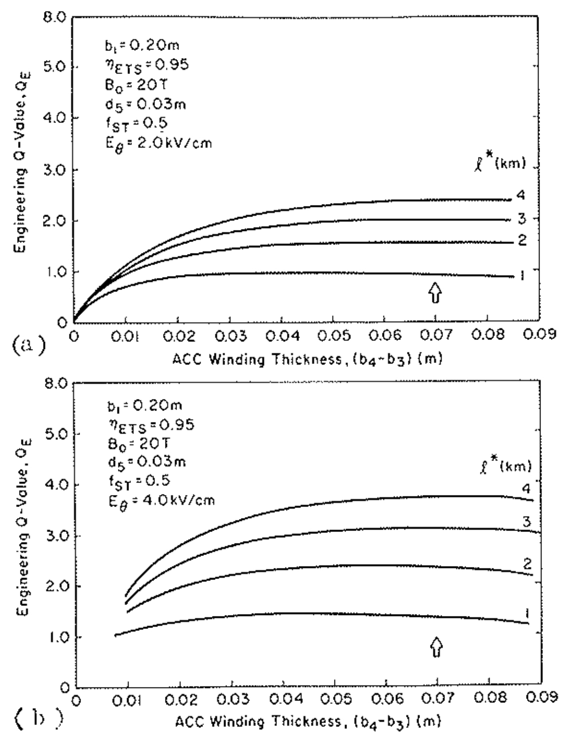


Fig. 10. Dependence of Q_E on ACC thickness (b_4-b_3) and effective length ℓ^*. $E_\theta = 2$ and 4 kV/cm for implosion fields; $b_1 = 0.20$ m, and (b_3-b_2) = 0.03 m. Arrow indicates choice of (b_4-b_3).

Q_E, ACC strength considerations, and neutronic constraints (Fig. 6 and 7) suggest a choice of $d_6 = d_4 - b_3 \gtrsim 0.07$ m for ℓ^* greater than 4 km.

As discussed in Sec. 3.2, a tradeoff exists between the ACC strength requirements and the joule losses as reflected by the fraction of structural material f_{ST} needed for a given ACC thickness d_6 and inner radius b_3. For a given f_{ST} a maximum field B_E will be permitted on the basis of structural endurance limits. Small values of f_{ST} for a given ACC geometry will limit the value of B_E and hence Q_E. Higher compressions will be permitted as f_{ST} increases, but the joule losses will increase as more structure is added to the fixed geometry ACC. Defining Q_E^* as a maximum value attainable for $B_o = B_E$, Fig. 11a gives the dependence of this "stress-limited Q-value," Q_E^*, on f_{ST} for E_θ = 2 kv/cm. The previously assumed value of f_{ST} = 0.5 is increased to $\sim$.6 on the basis of this calculation. It is reasonable to expect that

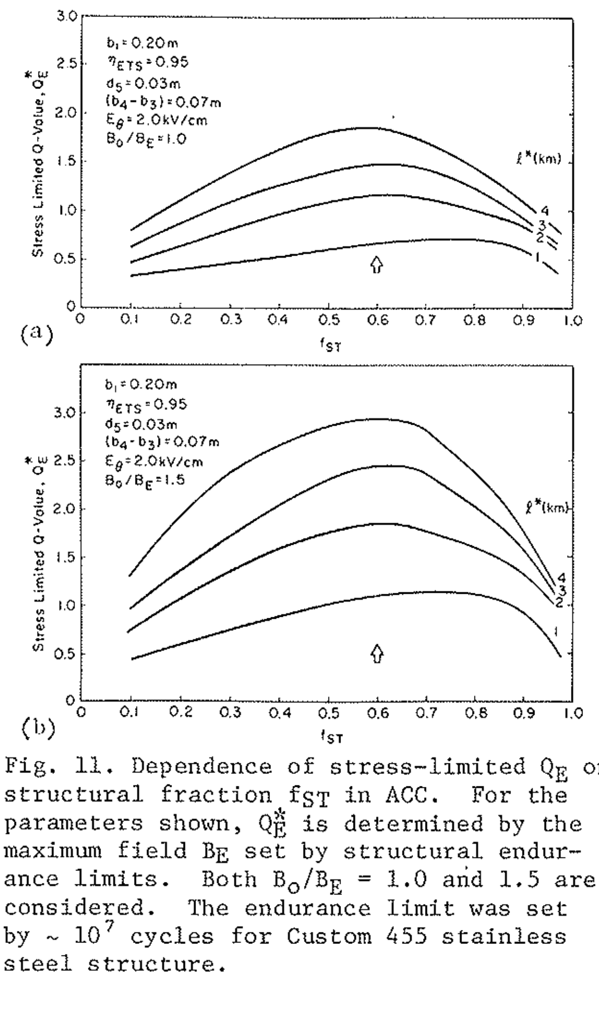


Fig. 11. Dependence of stress-limited Q_E on structural fraction f_{ST} in ACC. For the parameters shown, Q_E^* is determined by the maximum field B_E set by structural endurance limits. Both B_o/B_E = 1.0 and 1.5 are considered. The endurance limit was set by ~ 10^7 cycles for Custom 455 stainless steel structure.

ACC redesign could allow 50% improvement in this B_o/B_E constraint. As seen in Fig. 11b, this does not alter the choice of optimum f_{ST} = 0.6 but does improve the resultant Q_E^* values.

In addition to the system dimensions, an optimum Q_E depends on an appropriate choice of the initial D-T gas filling pressure, P_A. As noted previously, all data presented here are based upon this P_A optimum. The dependence of Q_E on P_A for the geometric values selected thus far is shown in Figs.12a-b to illustrate this behavior. Low values of P_A result in efficient heating but the neutron yield per unit of pinch length is reduced. Increasing P_A, on the other hand, leads to lower plasma temperatures, and without substantial compression the neutron yield is again reduced. It should be noted that the Q_E - P_A optimum occurs at P_A values in excess of those which merely maximize the plasma temperature *per se*.

Using the previously selected geometric values, Figs. 13a-b illustrates the influence of compression field B_o on the length or end-stoppering dependence of Q_E. Shown also are the stress constraints, as reflected by $B_o/B_E \leq 1.0$ or 1.5, and the economic constraint, [REV/COST] $\geq$ 1.0. The choice of B_o = 20 T used in the parameter survey up to this point is seen to violate the B_o/B_E = 1.0 constraint of Fig. 11a for the range of effective lengths shown but satisfies the more liberal $B_o/B_E \leq$ 1.5 limit. In order for economic breakeven to be reached, an effective length ℓ^* = 4 km must be achieved (e.g. a factor of 4 reduction of a free-streaming 1 km device). Admittance of higher B_o values reduces the effective length required until the stress limit is reached. The attainment of more effective end stoppering relaxes the field requirements and hence improves the stress picture.

Further iteration of the foregoing parameters could be expected to improve the LTPHR operating characteristics somewhat, and as technology improves, the rather severe limitations imposed on this study may be relaxed. This exercise has, however produced an LTPHR design point, summarized in Table V, which simultaneously satisfies the imposed economic and structural constraints. The resultant LTPHR energy balance characteristics are presented in Table V.

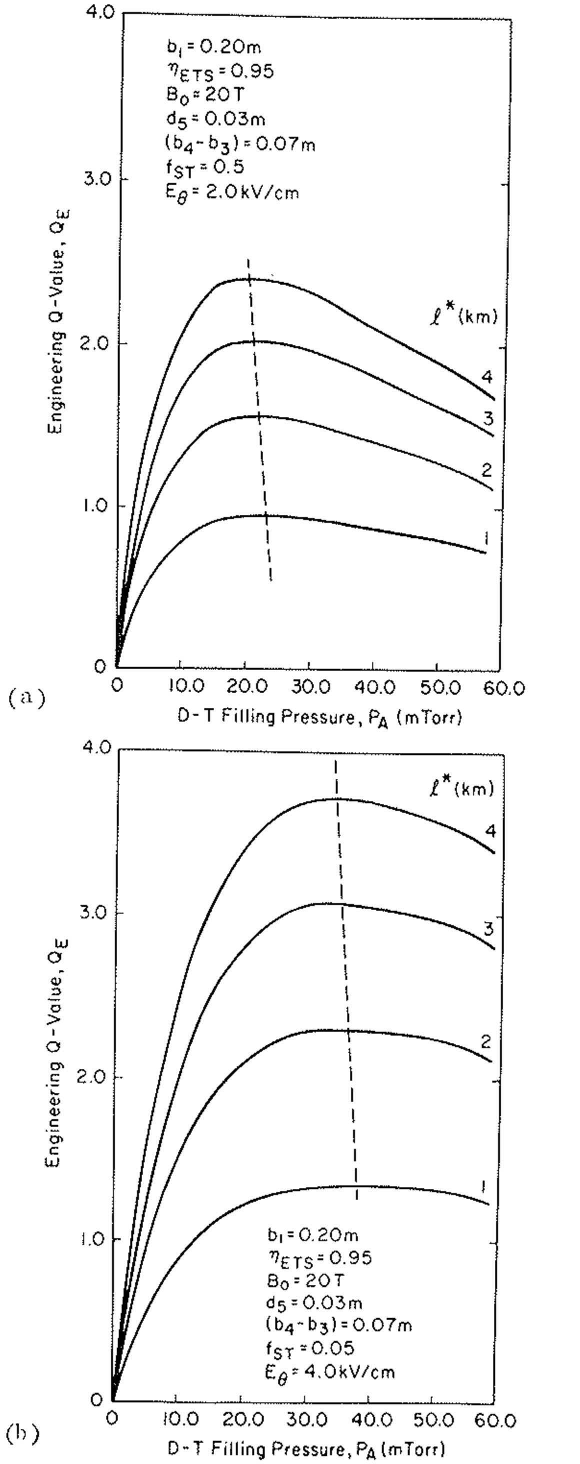


Fig. 12. Dependence of Q_E on D-T filling pressure P_A for the optimum parameters and E_θ = 2 and 4 kV/cm. All Q_E values reported herein are based upon the optimum filling pressure.

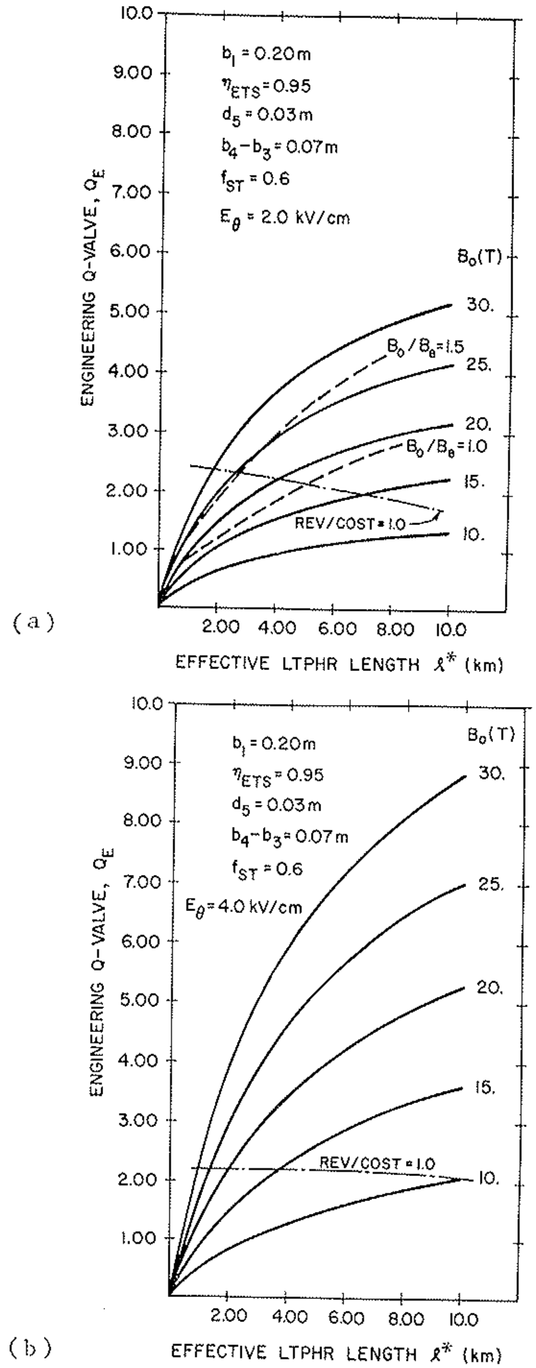


Fig. 13. Dependence of Q_E on effective length ℓ^* and compression field B_0. The stress limits (B_0/B_E = 1.0 or 1.5) and economic limits [REV/COST] are also shown. Structural limits dictates operation below the B_0/B_E curves and economic considerations dictate operation above the REV/COST curves.

4.2 TIME-DEPENDENT REFERENCE CASE RESULTS

The model used in the time-averaged parameter survey necessarily sacrificed a detailed treatment of the LTPHR pulsed burn dynamics for ease in treating the many variables involved. Once a nominal operating point is identified, however, it is instructive to examine the time history of

TABLE V

INTERIM LTPHR DESIGN POINT VALUES

First wall radius, b_1 (m)	0.200
IHC outer radius, b_2 (m)	0.204
ACC inner radius, b_3 (m)	0.234
ACC outer radius, b_4 (m)	0.304
Outer radius of IHC feedplates, b_5 (m)	3.0
Distance from centerline of LTPHR to ACC power supply, b_6 (m)	4.0
Number of turns in the ACC fed by each lead set	10
Number of turns in the IHC	0.3
ACC structural volume fraction, f_{ST}	0.6
Implosion electric field, E_θ (kV/cm)	2.0
Initial D-T filling pressure, P_A (mTorr)	19.0
Initial D-T filling density, n_o (ions/m^3)	1.35×10^{21}
Shock magnetic field, B_{SH} (T)	1.3
Shock radius ratio, x_{SH}	0.632
Burn time, τ_B (msec)	10.0
Flat top time, τ_{FT} (msec)	0.0
Maximum compression field, B_o (T)	20.0
Maximum ion temperature, T_i^* (keV)	6.6
Maximum electron temperature, T_e^* (keV)	6.4
Minimum radius ratio, x	0.114
Energy worth of fusion neutron, $(E + E^*)$ (MeV/n)	412.4
Effective LTPHR length, ℓ^* (km)	4.0
Nominal LTPHR length, ℓ (km)	1.0
Lawson parameter, $n\tau$, (s/cm^3)	4.13×10^{14}

crucial physics parameters. This study is done using the time dependent, global thermonuclear burn code DTBURN.[12] The effects of end loss have been incorporated into DTBURN by artificially decreasing the ion and alpha-particle density with time according to an exponential decay which has as a decay constant τ_{EL}(Eq. 2-43); a truly axial-dependent burn calculation has not been made to date.

In order to test the validity of the simple model used in this report, Fig. 14 compares the value of the neutron yield parameter α(n/MeV) [Eq. 2-32] computed from the survey calculations to those generated by DTBURN. The agreement is seen to be fairly good, with the survey calculation giving more conservative values for α (and hence Q_E) for the higher ℓ^* values. The correspondence of the two models increases confidence in the ability of the time-averaged model to accurately reflect LTPHR operating conditions.

The time histories of several LTPHR burn parameters are presented in Fig. 15 for the operating point summarized in Table V and VI. Subject to the sinusoidal compression field waveform, the plasma radius ratio x decreases from the

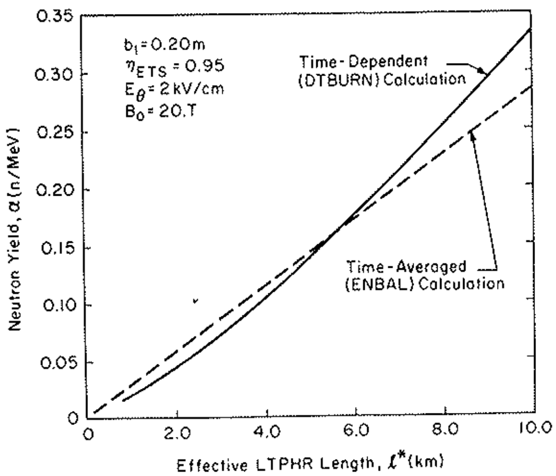


Fig. 14. Comparison of neutron yields α(n/MeV) predicted by ENBAL and DTBURN.

TABLE VI

INTERIM LTPHR ENERGY BALANCE

Total shock energy, W_{SH} (MJ/m)	0.370
Initial plasma internal energy, W^i_{INT} (MJ/m)	0.051
Final plasma internal energy, W^f_{INT} (MJ/m)	0.031
ETS efficiency, η_{ETS}	0.95
Thermal energy conversion efficiency, η_{TH}	0.40
Fusion neutron energy, W_N (MJ/m)	17.195
Fusion alpha energy, W_α (MJ/m)	0.145
Direct conversion energy, W_{DC} (MJ/m)	-0.082
Transport losses in ACC, W_T (MJ/m)	0.916
External ETS losses, W_{SC} (MJ/m)	1.671
End loss energy, W_{EL} (MJ/m)	0.194
Maximum magnetic energy stored in LTPHR, W_{BCM} (MJ/m)	33.427
Total recoverable thermal energy, W_{TH} (MJ/m)	17.344
Total circulating electrical energy, W_C (MJ/m)	3.042
Total electrical energy out of LTPHR, W_{ET} (MJ/m)	6.937
Net electrical energy out of LTPHR, W_E (MJ/m)	3.896
Recirculating power fraction, ε	0.438
Engineering Q-value, Q_E	2.28
Thermal first wall loading, I_w (MW/m^2)	1.0

implosion-heated value x_{SH} = 0.632 to aproximately 0.11, which is maintained until the plasma re-expands and quenches the burn. Under compression the plasma density and temperature increase, producing the thermonuclear burn and consequent neutron pulse. Streaming from the ends of the device depletes the ion density by approximately 50% of its initial value when the pulse ends at 10 ms. In Fig. 16 are depicted several of the energy balance quantities. Of particular interest is the direct conversion term W_{DC}. The inability to adequately confine the alpha-particle energy in the presence

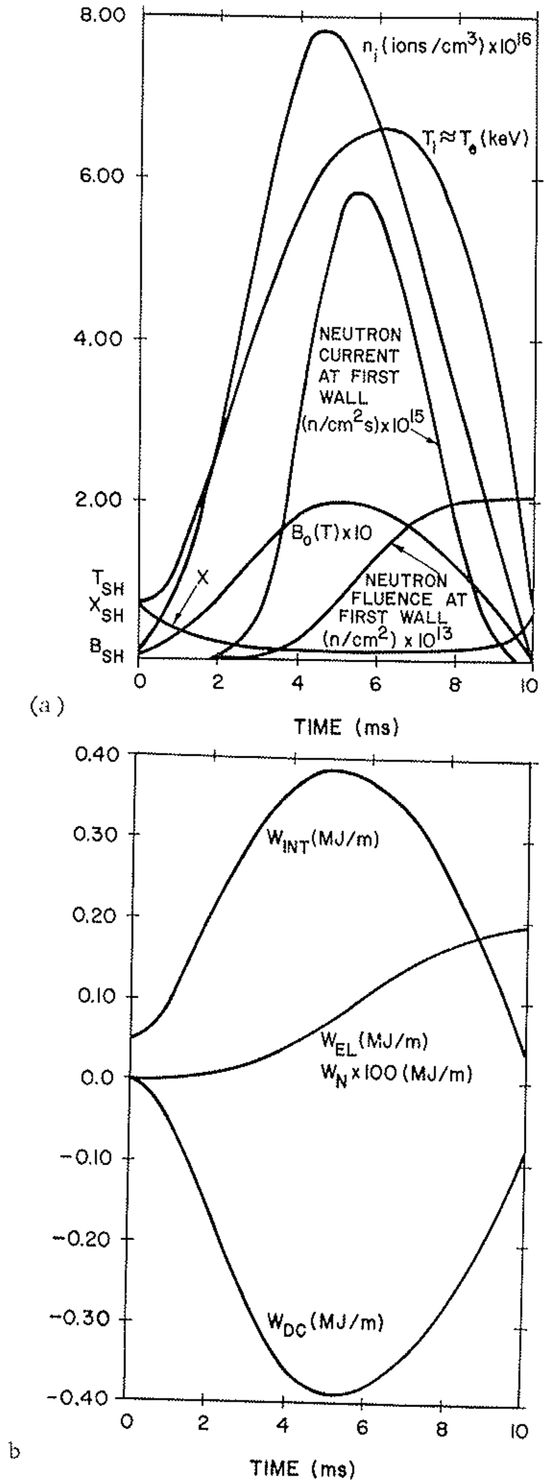


Fig. 15. Time dependence of key plasma and energy quantities predicted by DTBURN for the interim operating point selected by ENBAL. W_{INT} = plasma internal energy, W_{EL} = end-loss energy (free streaming), W_N = thermonuclear neutron energy, and W_{DC} = alpha-particle-direct-conversion energy.

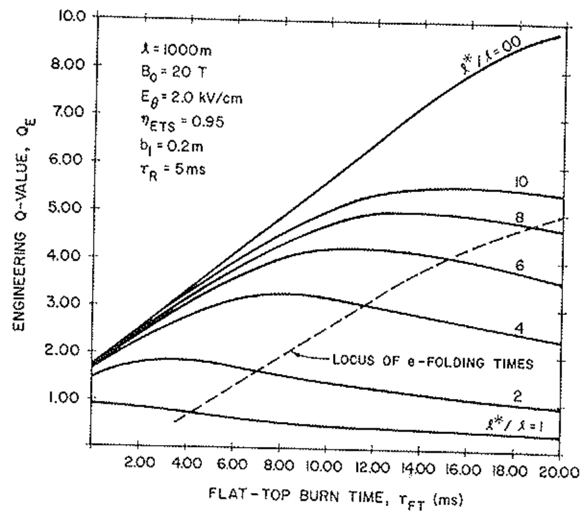


Fig. 16. Dependence of Q_E on flat-top time τ_{FT} and effective length $\ell^*/\ell = \infty$ corresponds to the case of no particle and energy loss.

of end loss keeps $W_{DC} < 0$ at all times. For longer effective length devices, this term can be made positive at the end of the burn and thus contribute favorably to the overall energy balance. Generally, plasma ignition does not occur for the densities (i.e., compression fields) and $\ell^* < 4$ km for the particle-loss end condition. Increasing the compression fields above 20 T, increasing the burn time, or increasing the pre-heating (E_θ = 4 kV/cm) leads to ignition conditions and substantially improved values of Q_E.

4.3 EXAMINATION OF THE EFFECTS OF REDUCING PARTICLE AND/OR ENERGY END-LOSS

Several theories which quantitatively describe the particle and energy loss from the ends of a linear theta pinch have been published[13,15] and have recently been subjected to evaluation.[6] Particle streaming end loss has been represented by an exponentiation time τ_{EL} given by Eq. (2-43), where, depending upon the end loss factor η_{ELS} (Eq. 2-43 and subsequent discussion), τ_{EL} can be represented by uninhibited free-streaming end loss ($\eta_{ELS} \sim 3.9$) or an inhibited (i.e., multiple mirrors or cusped ends) particle loss ($\eta_{ELS} > 3.9$). To examine the dependence of inhibited particle loss from the ends of the theta pinch, the previously defined effective length ℓ^* ($\ell^*/\ell = \eta_{ELS}/3.9$) presents a convenient parameter. The global thermonuclear burn code DTBURN was used to examine the dependence of Q_E on ℓ^*/ℓ and burn time for the intermin design point given in Table V as determined by ENBAL. The burn time was varied by fixing the compression-field sinusoidal rise and fall time τ_R at 5 ms and interposing a period of constant (maximum) field for a "flat-top" time τ_{FT}. For ℓ = 1 km, B_o = 20 T and E_θ = 2 kV/cm the dependence of Q_E on τ_{FT} and $\ell^*/\ell \equiv \eta_{ELS}/3.9$ is depicted in Fig. 16. The case where τ_{FT} = 0 corresponds to a pure sinusoidal field pulse of half period $2\tau_R$ = 10 ms (i.e. the reference case illustrated in Fig. 15). The $\ell^*/\ell = \infty$ case shown in Fig. 16 corresponds to a fully stopped theta pinch and, therefore represents an ideal maximum limit for the interim optimum summarized on Table V. Shown also in Fig. 16 is the locus of line-density e-folding points, which generally do not occur at the optimum values of Q_E. As seen from Fig. 16, a factor of $\ell^*/\ell \sim 4$ reduction in particle loss relative to the free-streaming case (ℓ^*/ℓ=1) has a dramatic effect on the overall attractiveness of the LTPHR as measured by Q_E. Both the physics and engineering aspects of achieving $\ell^*/\ell \geq 4$ are not resolved and, in view of the predictions of Fig. 16, should be pursued intensively on both experimental and calculational levels.

The use of solid end plugs represents a method to eliminate particle end loss. Given a theta-pinch end plug that is capable of maintaining a constant plasma density throughout the pinch, axial therman conduction *via* electrons will represent a major heat sink. To examine the effect of purely axial heat loss on a constant line-density pinch, the time-dependent code DTBURN was modified to include an additional electron energy sink in accordance to the simple axial conduction model proposed by Morse.[15] Using the following expression for electron thermal conductivity k parallel to the axial field lines,[4]

$$k\ (W/eV\ m) = 3.10 \times 10^4\ T_e^{5/2}/\ell n\Lambda, \tag{4-1}$$

the rate of axial conduction loss is given by,

$$\dot{Q}_{END}\ (W/m^2) = -3.10 \times 10^4\ T_e^{5/2}/\ell n\Lambda\ (dTe/dz). \tag{4-2}$$

Following Morse, $\dot{Q}_{END}$ is assumed constant, Eq. (4-2) is solved for T_e, $\dot{Q}_{END}$ is divided by one-half the plasma volume, and the resulting expression is equated to the time rate of change of electron energy. The resulting rate of decrease of T_e as a result of axial heat conduction is

$$(dT_e/dt)_{COND} = -T_e/\tau_{COND} \tag{4-3}$$

$$\tau_{COND} = 2.7 \times 10^{-11}(n/T_e^{5/2})\ell^2(\ell_g/\ell)$$

Here, T_e is in eV, T_e at the pinch ends is zero and mks units are otherwise used. For a given total pinch length ℓ and gradient lengths ℓ_g, Eq. (4-3) is used in DTBURN to model the additional decrease in T_e at each time step. The associated increased ion cooling *via* electron-ion collisions and decreased fusion rate will tend to decrease Q_E, even though the ion density is not decreased by free-streaming particle loss. In addition, Q_E should be decreased because of the ineffectiveness of that portion of the pinch ℓ_g where the thermal gradient exists. Assuming $T_e/T_i \sim 1$, (as is the case for the pinch lengths and associated burn-times being considered), the effective pinch length ℓ_{EFF} where T_e is in excess of a minimum value T_e^* is approximately given by $\ell_{EFF} = \ell\,[1-2(T_e^*/T_e)(\ell_g/\ell)]$. Consequently, Q_E must be reduced by ℓ_{EFF}/ℓ, which for $T_e^* \sim 1$ keV and the range of $\ell g/\ell$ values being considered results in a 10-20% reduction in Q_E.

The dependence of Q_E (corrected by ℓ_{EFF}/ℓ) on τ_{FT} and $\ell g/\ell$ is shown in Fig. 17 for τ_R = 5 ms, ℓ = 1000 m, B_o = 20 T, and E_θ = 2 kV/cm. Shown by dashed lines are representative Q_E *vs* τ_{FT} curves for the particle-streaming end condition (Fig. 16).

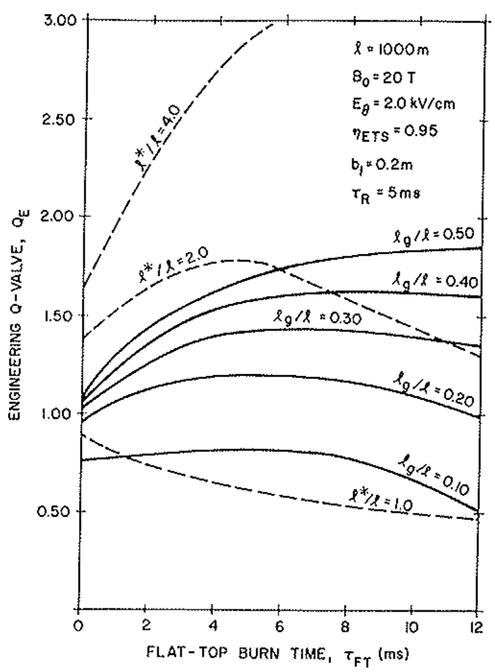


Fig. 17. Dependence of Q_E on flat-top time τ_{FT} and ℓ_g/ℓ for axial electron heat conduction only (end-plugged case). Curves from Fig. 16 for ℓ^*/ℓ = 1, 2 and 4 are shown for comparison. The electron thermal gradient occurs over a length ℓ_g at each end of the pinch.

Generally, for $\ell \leq 1000$ m, axial heat conduction effectively cools the ions, reduces the fusion reaction rate and decreases the Q_E considerably relative to the case of moderately inhibited particle loss ($\ell^*/\ell \lesssim 4$). Since $\tau_{COND} \sim \ell^2$ for a given ℓ_g/ℓ, small increases in pinch length should considerably reduce the conduction loss mechanism and permit the plasma to approach ignition (i.e. alpha-particle heated) conditions. For ℓ_g/ℓ fixed at 0.1, Fig. 18 illustrates this dependence on length ℓ for both the end-plug case (axial electron conduction only) and the inhibited particle loss case (for free-streaming end loss, $\ell^* = \ell$). For the end-plug case, $\ell \sim 4000$ m approaches the ideal limit of no particle or conduction loss ($\ell^* = \infty$). The results presented in Fig. 18 clearly show the advantages of end plugging (axial electron conduction only) relative to free-streaming particle loss ($\ell^* = \ell$), although this advantage can be realized only for pinch lengths on the order of 2 km or greater.

5. SUMMARY AND CONCLUSIONS

The time-averaged plasma model has been used to examine the dependence of Q_E on a wide range of physics and engineering parameters. Realistic mechanical, electrotechnological, physics and economic constraints have been imposed. Although the hybrid blanket model was not altered from the thermal neutron, high fission-rate design,[1] the blanket response *vis a vis* E + E* was coupled to the energy

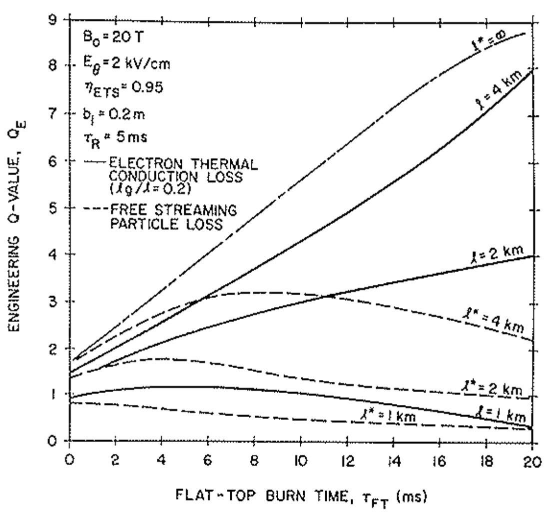


Fig. 18. Dependence of Q_E on flat-top time τ_{FT} for $\ell_g/\ell = 0.2$ for axial electron heat conduction only and comparison with inhibited particle loss case.

balance *via* the thickness of the first-wall ACC. The following qualitative and quantative constraints have lead to the ultimate choice of key LTPHR parameters:

a. First-wall Radius, b_1: The desire for high neutron yield per unit pinch length leads to increased values of b_1, although considerations of stress, energy storage and joule loss lead to diminished gains in Q_E as b_1 increases. For $B_0 = 20$ T, a value of 0.20 m for b_1 represents a compromise choice.

b. IHC Flux-Return Path (b_3-b_2): If the IHC return flux area is too small, the associated high flux density leads to saturation of the high permeability (μ_F=10) material contained therein; the associated stored energy is high and Q_E subsequently is reduced. Large values of (b_3-b_2) result in increased ACC radius and energy stored therein; Q_E therefore, is also reduced. The optimum value of (b_3-b_2) for a range of implosion fields is 0.03 m for b_1 = 0.20 m.

c. ACC Winding Thickness (b_4-b_3): Although increased ACC winding thicknesses decrease joule losses, the stored energy increases and both the neutron energy worth (E+E*) and the tritium breeding ratio [BR] decrease. For $\eta_{ETS} = 0.95$, the optimum is broad and a value of 0.07 m was chosen. A value in excess of ~0.15 m will force [BR] below unity.

d. ACC Structural Fracture f_{ST}: The trade-off between increased joule losses and higher neutron yield (increased allowable compressions) as f_{ST} increases leads to an optimum value of 0.6 for the previously selected parameters.

On the basis of these parameter variations and the desire for economic break-even [REV/COST] ≥ 1 , an effective length $\ell^* = 4$ km was selected. Comparison of the time-averaged results with the predictions of an end-loss modified DTBURN computation showed excellent agreement and gives confidence in subsequent time-dependent DTBURN computations which use "optimum" parameters determined from the more flexible time-averaging model. The dynamic modeling of particle and energy end loss *via* DTBURN showed the advantages of extended burn periods ($\tau_{FT} > 0$) for effective lengths $\ell^* \geq 2$ km (Fig. 16), although the added complexities of switching associated with $\tau_{FT} > 0$ must be examined. The advantages of end plugging (no particle loss, axial electron heat conduction) over inhibited particle loss

($\ell^*/\ell > 1$) was shown for pinch lengths in excess of 2 km for τ_R = 5 ms and $\tau_{FT} \gtrsim 10$ ms. The axial heat conduction model used, however, simply assumes constant axial heat flux[15] and should be supplemented with a truly time-dependent model. Furthermore, the global DTBURN model should be replaced with a truly one-dimensional calculation of the plasma (and alpha-particle) burn dynamics. Both aspects of the plasma behavior will be treated by more representative models in the near future, which provide a more physical picture of both axial particle and energy losses.

Although the energy-balance model is adequate for the intent of the analysis, the neutronic constraint has focused onto a single blanket design, the economic constraint is based upon a very simple model, and a number of alternative ACC structural schemes were not considered. The benefits associated with higher compression fields and thinner first-wall compression coils (i.e. coil support within or external to the blanket) are tremendous (Q_E vs B_0) and will be examined in more detail. Such external or blanket support coils will influence the neutronic response as well as the mechanical design of the LTPHR blanket. Because of the complex nature of the neutronic analysis *per se*, this aspect of the study has purposely been decoupled as much as possible from the present analysis, although further studies will consider newer blanket concepts. Lastly, a constant η_{ETS} = 0.95 was assumed throughout this analysis. In actuality, η_{ETS} is a frequency dependent quantity,[15] and the dependence of η_{ETS} on τ_R should be taken into account. Although the extension of burn time by use of flat-top times $\tau_{FT} \neq 0$ leads to considerable enhancement of Q_E, the associated switching and current interruptor requirements will be severe. This switching requirement and the need for high efficiency ETS systems represent the major technological hurdles for the LTPHR.

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A CONCEPTUAL DESIGN STUDY FOR A LASER FUSION HYBRID

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ABSTRACT

Lawrence Livermore Laboratory and Bechtel Corporation have been involved in a joint effort to conceptually design a laser fusion hybrid reactor. The design which has evolved is a depleted-uranium fueled fast-fission blanket which produces fissile plutonium and electricity. A major objective of the design study was to evaluate the feasibility of producing fissile fuel with laser fusion. This feasibility evaluation was carried out by analyzing the integrated engineering performance of the complete conceptual design and by identifying the required laser/pellet performance. The performance of the laser fusion hybrid has also been compared to a typical fast breeder reactor. The results show that the laser fusion hybrid produces enough fissile material to fuel more than six light water reactors (LWR's) of equivalent thermal power while operating in a regime which requires an order of magnitude less laser and pellet performance than pure laser fusion. In comparison to a fast breeder reactor the hybrid produces 10 times more fissile fuel. An economic analysis of the design shows that the cost of electricity in a combined hybrid - LWR scenario is insensitive to the capital cost of the hybrid, increasing by only 20 to 40% when the capital cost of the hybrid ranges from 2 to 3 times more than an LWR.

INTRODUCTION

The possibility of producing fissile fuel and electricity by placing a subcritical fission blanket around a fusion chamber has emerged as a promising application of fusion. Fusion-fission hybrid systems naturally combine the "power richness" of fission with the "neutron richness" of fusion. In system studies for laser fusion at Lawrence Livermore Laboratory we have evaluated the potential of fusion-fission hybrids that make sense as part of the evolution of a fusion power economy. Earlier studies(1,2,3) primarily utilized neutronic methods of analysis to identify more attractive hybrid systems and to provide an upper bound estimate on performance. These earlier studies identified several promising concepts. They also demonstrated that fusion-fission hybrids could be designed to meet a broad spectrum of fissile fuel producing and energy multiplying requirements. The two most significant features which emerged from our neutronic scoping studies were:

1. Laser fusion hybrids produce 10 times more fissile fuel (per unit of thermal energy generated) than fission breeder reactors.
2. Laser fusion hybrids produce electricity with much lower laser efficiencies and pellet gains than required for pure laser fusion.

The neutronic results were encouraging but it was apparent that a more accurate assessment of the hybrid's potential and a definitive ranking of the more promising concepts would require studies which deal with the engineering safety and economic issues as well as the neutronic aspects. With this in mind Bechtel Corporation was engaged to assist us in a conceptual design study of a fusion hybrid. The scope of the study was defined by the requirement to provide enough design detail to realistically gauge the value of a laser fusion hybrid in a fission power generation economy. The joint effort has been underway since July 1975. This paper will describe and analyze the laser fusion hybrid design which has evolved.

The hybrid concept chosen for this design study is a depleted-uranium fueled fast-fission blanket which produces fissile plutonium and electricity. It emphasizes fissile material generation by maximizing for fuel production at the expense of energy multiplication. This blanket selection was based on reported neutronic results[4,5] which indicated that a depleted uranium fueled fast fission blanket could provide enough fissile fuel to extend the energy available from economically proven light water reactors (LWR's) by as much as two orders of magnitude.

A comparative analysis[4] of the neutronic properties of several hybrid concepts has shown that depleted uranium fast fission blankets provide the largest amount of fissile fuel (per unit of thermal energy) with the lowest laser efficiency and pellet gain requirements. The depleted uranium blanket selected for our conceptual design produces enough fissile material to fuel more than six LWR's of equivalent thermal power. Thorium fueled hybrids produce more fissile fuel per unit of thermal energy but their fusion energy multiplying capabilities are much lower. Hence, they require a higher performing laser fusion system. There are blanket concepts which have higher energy multiplication capabilities than depleted uranium blankets. These blankets could efficiently produce electricity with lower fusion energy gains; however, their enhanced energy multiplication is gained at the expense of decreased fissile production.

Light water reactors will be the major and most likely the only, source of commercial nuclear electric power for the remainder of this century. Their dominance over coal fired plants as base load electrical generators will be strongly dependent on the adequacy of their long term fissile fuel supply. By converting the U^{238} in natural uranium to fissile plutonium, the hybrid could extend the fissile fuel supply for economically proven LWR's by two orders of magnitude. Fast breeder reactors also offer the prospect of more fully utilizing the uranium resources but they will not provide fissile fuel for LWR's. Therefore, the usefulness of fast breeder reactors will be entirely dependent on their economic competitiveness as power plants.

THE LASER FUSION HYBRID DESIGN

Work in the joint laser fusion hybrid design study was apportioned as follows: Lawrence Livermore Laboratory provided the overall direction, the neutronics data, and the fusion portions of the design. Bechtel Corporation provided the fission portion of the hybrid, the design of the thermal energy transport and conversion system, the tritium recovery system, and the layout of the complete power plant. They also analyzed the fuel cycle, capital, and operating cost. Bechtel's contribution to the laser fusion hybrid design is fully documented in their final report to Lawrence Livermore Laboratory[6].

We set for ourselves four major objectives by which to gauge our success in the design study:

1. Identify the laser/pellet performance required to economically produce fissile fuel and power with a hybrid.
2. Evaluate the integrated engineering performance of a complete conceptual design.
3. Compare a laser fusion hybrid to existing fission breeder options (LMFBR, GCFBR, LWBR).
4. Identify major technological problems associated with a laser fusion hybrid.

Achievement of these objectives completely defined the level of design detail and costing analysis for the study.

In addition to these objectives there were a few philosophical points of view which significantly affected our design choices. First, we wanted to operate the laser fusion hybrid in a regime which required an order-of-magnitude less laser/pellet performance, i.e., fusion energy gains in the neighborhood of 1.0. This implied blanket energy multiplications approaching 10.0. Second, we wanted to utilize state-of-the-art fission technology in the design of the hybrid blanket. In keeping with this principal, we chose stainless steel as the structural and cladding material instead of higher performing refractory metals. Finally, we believed that a hybrid reactor which produces fissile fuel should be designed to be as safe as and with the same environmental impact as the fissile burning reactors which it is providing fuel for. Here we note that a negligible improvement in the overall environmental impact results from making the hybrid environmentally more attractive than the larger number of light water reactors it is supplying fuel for.

HYBRID REACTOR DESIGN

The functional shape of the laser fusion hybrid chosen for final evaluation is shown in Fig. 1. In its simplest form it is a cylinder with a height-to-diameter ratio of 1.0. The center of the fusion chamber is the focal point for a six beam, 100 KJ laser system which irradia-

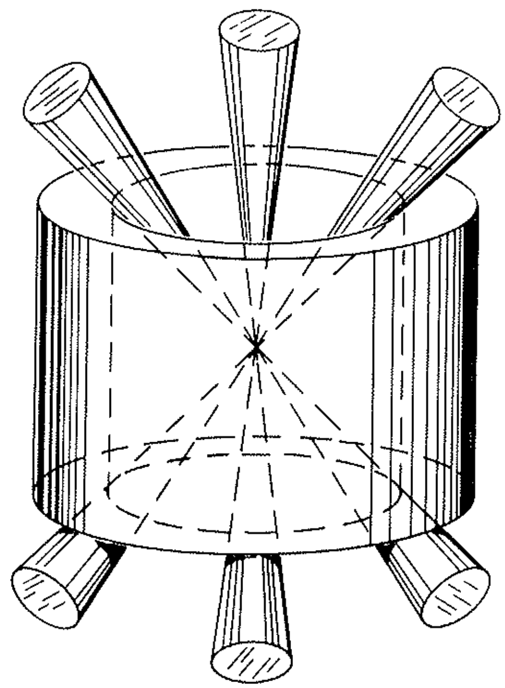

FIGURE 1. Geometry Used in the Laser Fusion Hybrid

tes the fusion target from the top and bottom of the cylinder.

The basic features of the hybrid reactor are displayed in Fig. 2. A depleted uranium fueled fast-fission blanket has been positioned radially around the fusion chamber. The energy in the fission zone (amounting to 90% of the total energy) is removed with a sodium coolant system. The liquid sodium enters into the fission zone from the lower plenum and flows to the upper plenum through hexogonally shaped process tubes.

Lithium-cooled graphite-moderated blankets are positioned in the top and bottom of the reactor and behind the fission zone. These lithium blankets moderate and capture neutrons and breed tritium. All penetrations for the laser beams and pellet injectors are made through the top and bottom blankets thereby leaving the radial fission blanket unencumbered. We had originally intended to use fission blankets in the top and bottom regions; however, the difficulties of maintaining coolant flow while the top blanket was being removed for access into the fusion chamber along with the requirements for unconventional fission blanket design led to our choice of lower performing nonfissioning blankets for these regions. The decision not to use fissionable fuel in these regions resulted in a 30% reduction in both fissile fuel and energy production; nevertheless, the decision was consistent with our desire to utilize state-of-the-art fission technology in the hybrid design.

As shown in Figure 3 the entire blanket system is enclosed within a spherically shaped stainless steel vacuum vessel which has a removable top. The final focusing mirrors are placed in beam tubes outside the vacuum vessel to minimize damage caused by the fusion microexplosion and provide for easier replacement.

FISSION BLANKET DESIGN

An expanded side view and a top view of the radial fission blanket are shown in Figures 4 and 5. As shown, the fission zone is made up of two rows of hexagonally shaped process tubes which contain the depleted uranium in the form of stainless steel clad fuel pins. The process tubes in the innder row are protected from the fusion cavity environment by a stainless steel supported graphite liner.

Our neutronics calculations indicated that energy multiplication and fissile fuel production are maximized by using uranium metal fuel instead of oxides or carbides and by maximizing the ratio of the volume fraction of uranium to structural material. Increasing the residence time of the fuel also increases the average energy multiplication since more energy is produced as plutonium is bred, accumulates, and fissions in the blanket.

Fast neutron damage may limit the useful life of stainless steel and other structural metals in a fusion reactor. A common design criteria for both laser and magnetic fusion reactors is a first wall neutron flux limitation of 1 MW/m^2 and an expected lifetime of less than five full-power years. Neutronics calculations at a first wall loading of 1 MW/m^2 show a maximum power density in a depleted uranium blanket of about eight watts per gram or about 150 watts per cubic centimeter of uranium metal. The average power density in a blanket with a uranium thickness of 250 grams per square centimeter is less than four watts per gram. Five full-power years of blanket operation would result in an average burn-up of less than 7,000 MWD/MTU. This low burn-up limitation favors the choice of metallic fuel.

The low power density and the low burn-up capability of a depleted uranium (or a natural uranium) blanket demand care in minimizing fuel cycle costs, including fabrication cost. Large fuel rods would thus be favored over small rods, and long fuel elements would have some advantage over short elements. The basic configuration of a laser fusion reactor--a vacuum chamber with laser beams converging from several angles--introduces great difficulties in the mechanical design of the blanket. Neutronic calculations show a severe reduction in performance if a pressure vessel wall is introduced between the blanket and the fusion core. The fuel element must therefore operate and be cooled within a surrounding vacuum. Thin process tubes and low pressure coolants would appear to be the most reasonable design approach, but coolant leaks and process tube reliability will always be potential problems. It is expected that melting of the fuel due to loss of coolant will be the most serious safety issue with the hybrid.

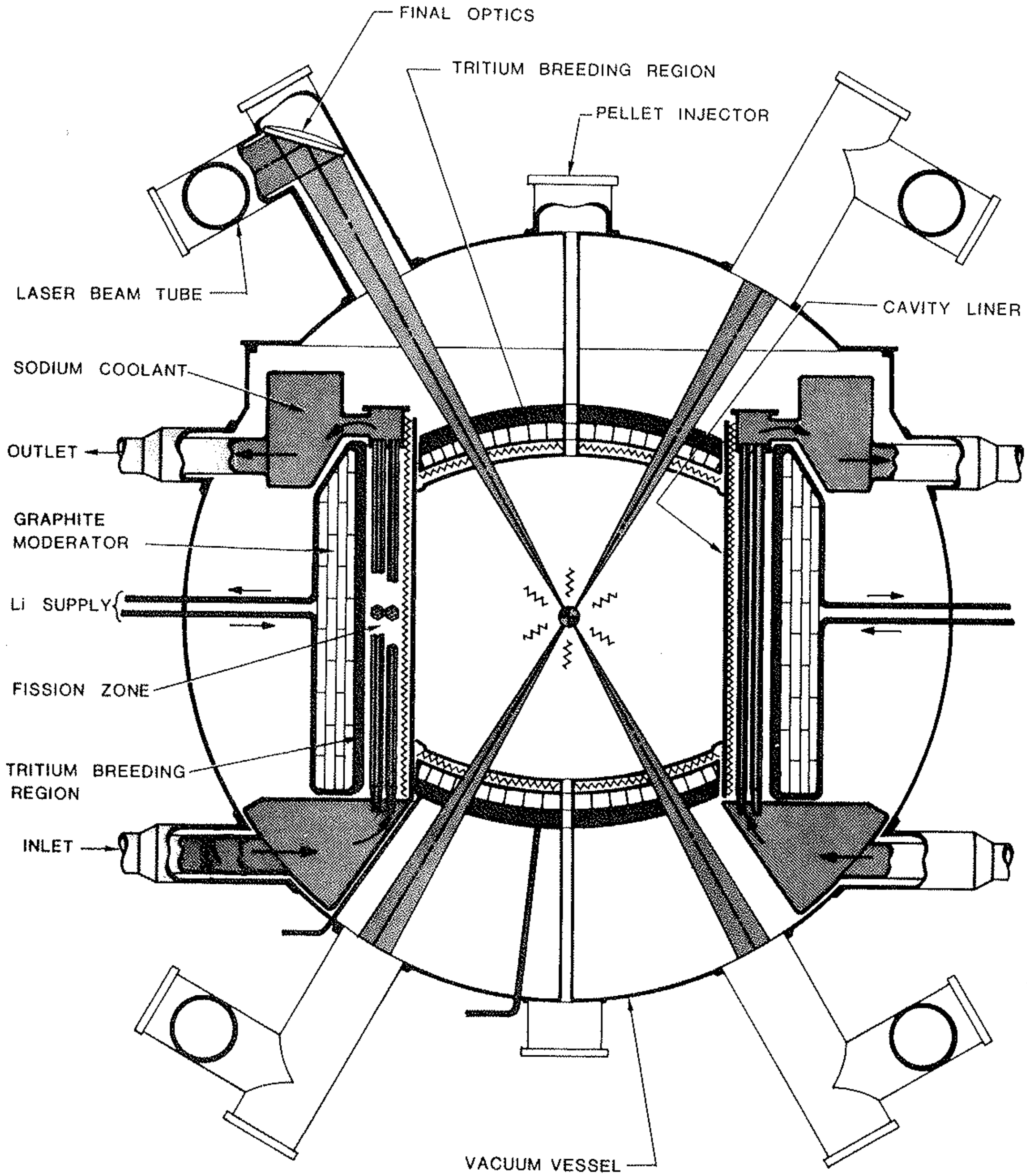

FIGURE 2. Side View of a Conceptual Laser Fusion Hybrid Reactor

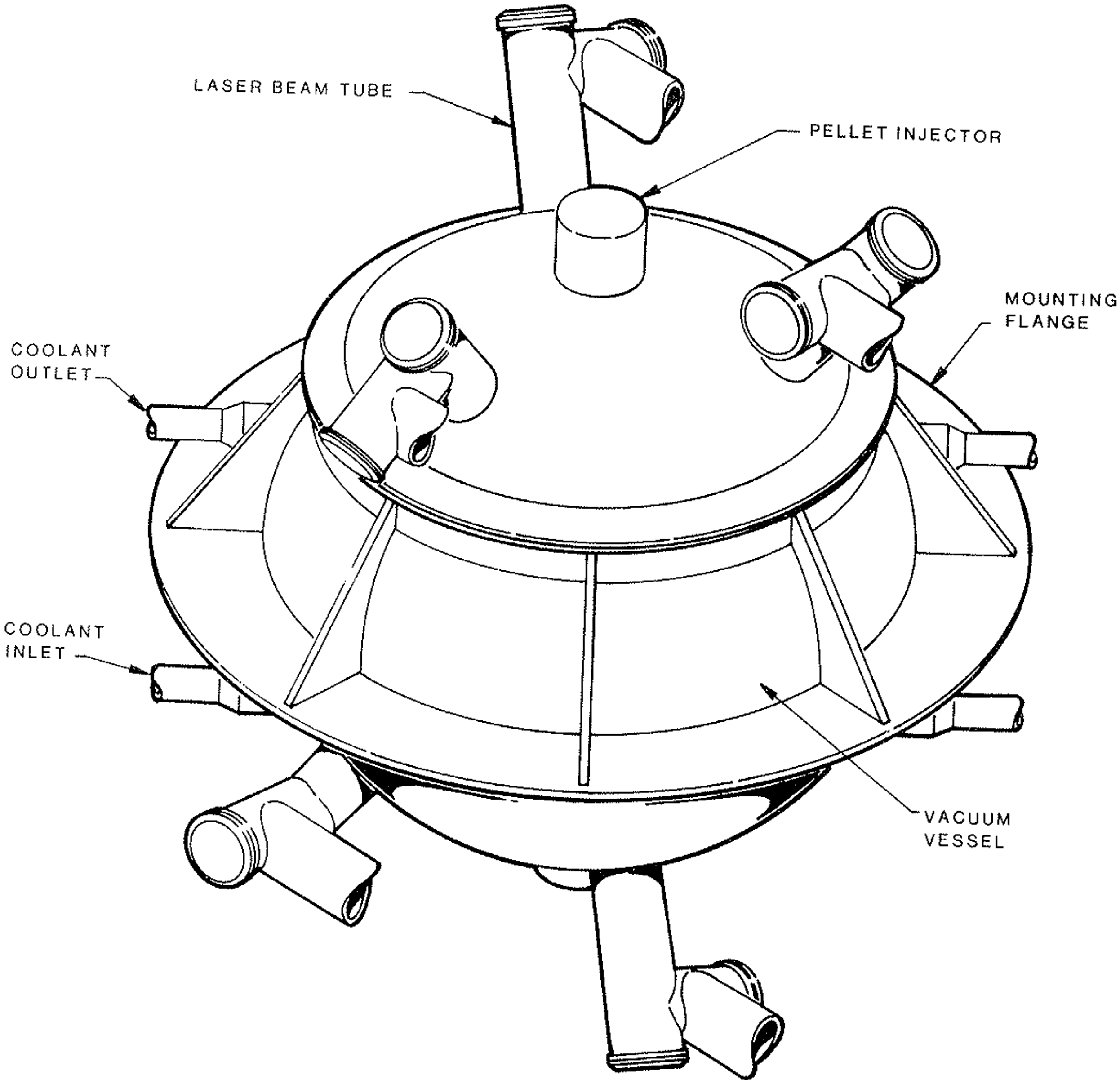

FIGURE 3. Vacuum Vessel for the Laser Fusion Hybrid Reactor

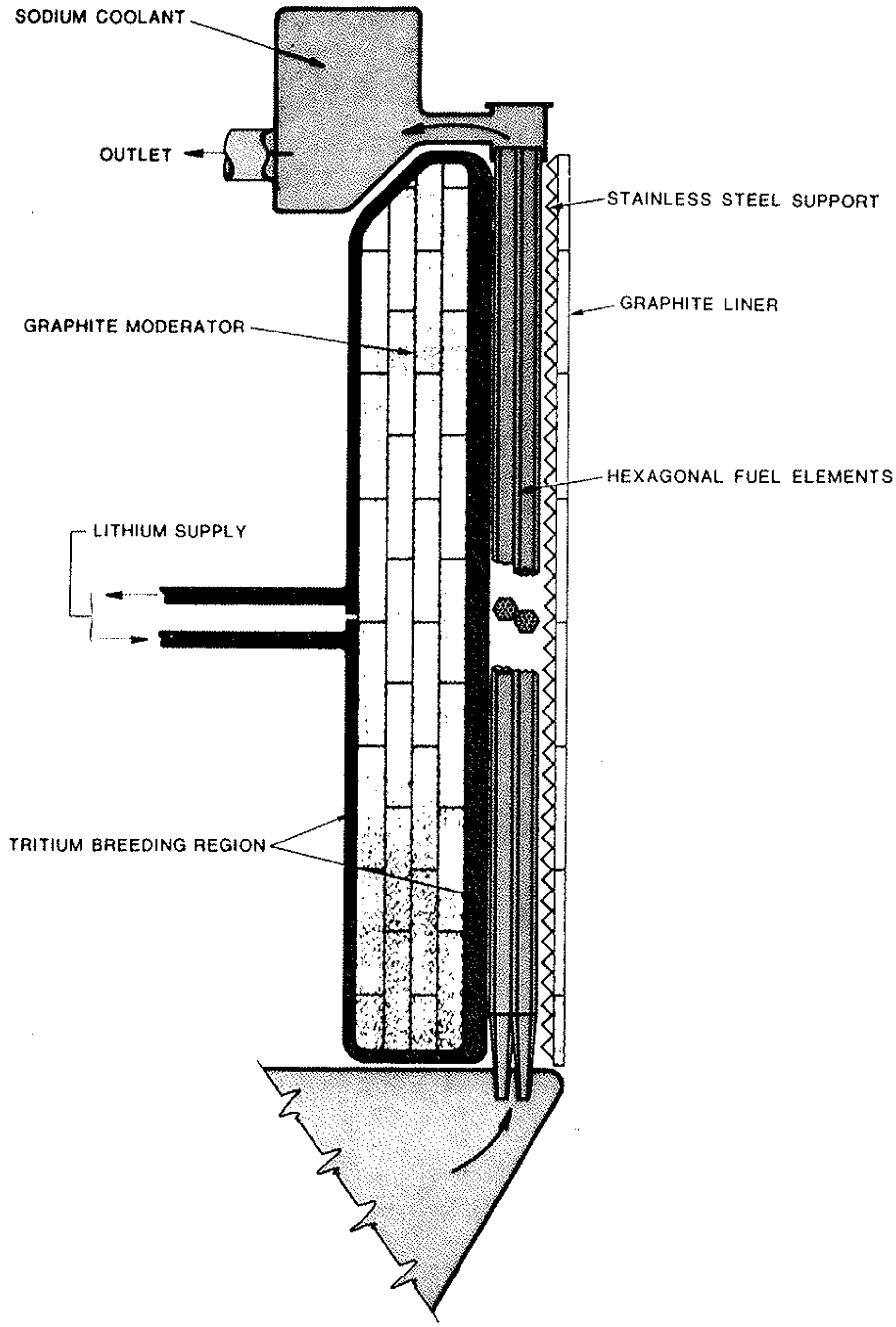

FIGURE 4. Side View of Fission Blanket in Laser Fusion Hybrid Reactor

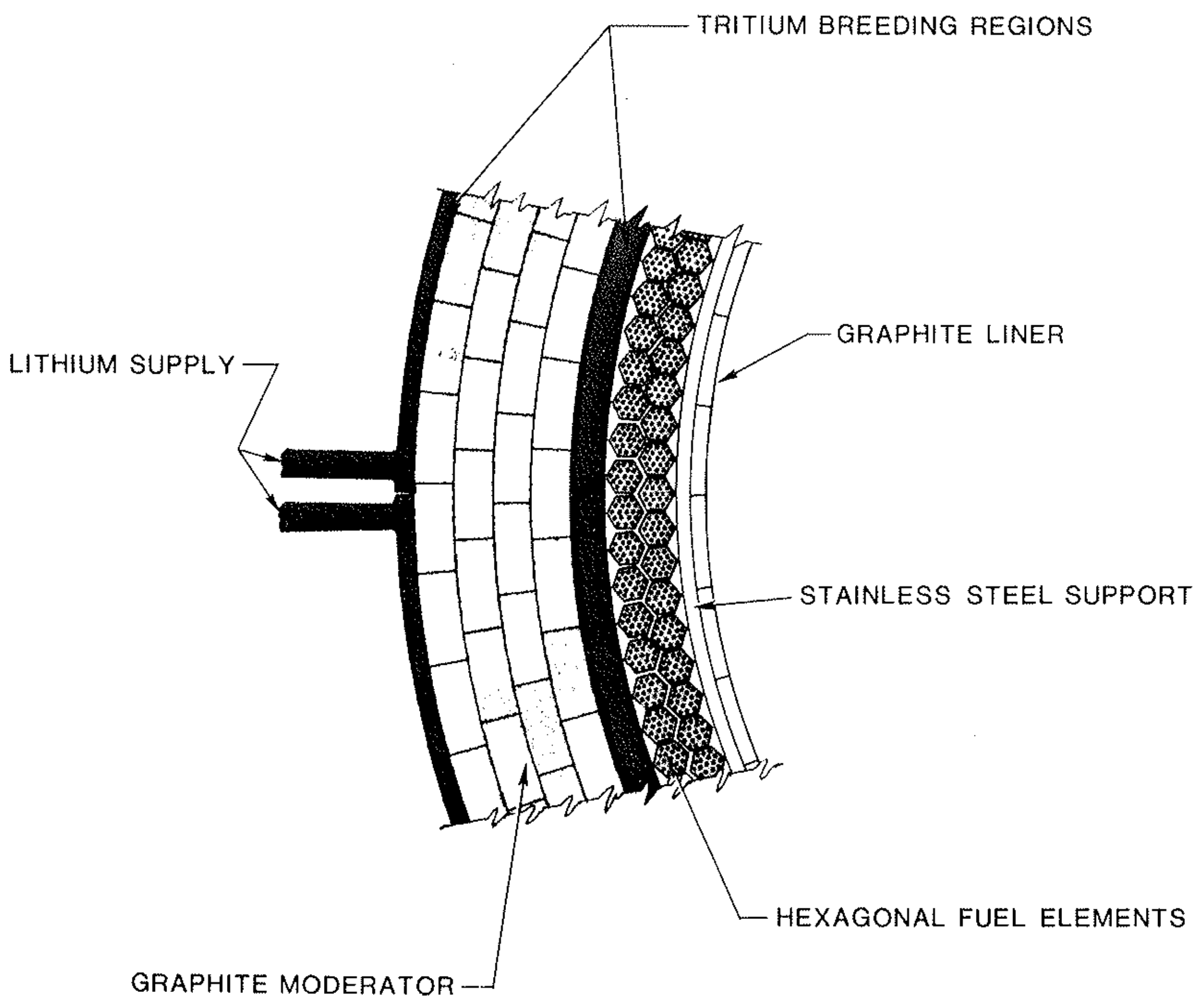

FIGURE 5. Top View of Fission Blanket in Laser Fusion Hybrid Reactor

SELECTION OF A FUEL MATERIAL

The general design considerations of maximizing neutron energy multiplication, minimizing fuel cycle costs, and developing a concept that could be licensed by the Nuclear Regulatory Commission led to the selection of uranium metal fuel elements with sodium as the coolant. LMFBR technology and balance-of-plant design concepts were used to the maximum extent possible. The selected fuel element is a 19-rod cluster similar to those developed during the early 1960's for use in sodium-graphite reactors (SGR). Fuel rods 18-feet long were developed at that time and required care in handling because of the flexibility of the rods and the tendency to buckle if they were not handled vertically. Adapting this experience to a blanket design with larger diameter fuel rods and a thicker cladding, it was estimated that 7-meter (23-ft) long fuel rods and fuel assemblies were feasible; this determined the reference height of the fusion reactor core. A full size cross section and a three dimensional view of the fuel assembly are shown in Figures 6 and 7.

Uranium metal, uranium with 7 wt% molybdenum alloy (U-7 Mo) and uranium carbide (UC) were all considered fuel materials for the sodium cooled blanket, and all are satisfactory. The reference fuel element can accept 30-millimeter diameter fuel slugs of each of these materials interchangeably. The U-7 Mo fuel should be capable of burn-ups to 20,000 MWD/MTU at maximum center temperatures of 650°C. This burn-up would require at least 8 full-power years to achieve, which the cladding probably could not tolerate. The U-7 Mo alloy was the reference fuel for the Dounreay fast reactor and performed satisfactorily at the conditions noted. This alloy has a reduced energy multiplication and plutonium production rate compared to uranium metal, and is also more expensive to fabricate and reprocess. Uranium carbide fuel would be capable of more than 100,000 MWD/MTU burn-up at maximum temperatures of 1,000°C (with a sodium bond) if the cladding were adequately strong. However, its energy multiplication is 30% less than uranium metal. In the low-power-density configuration of the reference blanket, the higher burn-up capabilities of U-7 Mo and UC cannot be used effectively. If higher power densities were possible, by using first-wall fluxes of 3 to 4 megawatts per square meter, or by using fissile enriched uranium fuel, or if gas cooling were chosen UC would probably be the fuel material of choice.

Uranium metal, "adjusted" with minor alloying additions in order to control swelling, was chosen as the reference fuel material for several reasons: first, because its multiplication and breeding performance were superior, second because its burn-up capability was judged adequate and a good fit to the blanket's low power density, and finally because it was cheaper to fabricate and reprocess.

Pure uranium metal begins to swell disasterously at temperatures greater than 400° C and burn-ups greater than 1,000 MWD/MTU (about 0.1 atom percent fissions). The British development of "adjusted" uranium metal as a fuel for their gas-cooled Magnox reactors during the early 1960's demonstrated that additions of 800 to 1,000 ppm Al, 350 to 500 ppm Fe, and approximately 500 ppm carbon could control this swelling and allow burn-ups of 5,000 to 6,500 MWD/MTU at temperatures of 600°C. Extensive development of similar alloys using Al, Fe, and Si (and sometimes Mo) have taken place at the Savannah River Laboratory and at Battelle Northwest Laboratory confirming the satisfactory performance of these fuels. A maximum fuel temperature of 600°C and a maximum burn-up of 6,000 MWD/MTU were chosen as a design basis for the blanket. A volume increase (swelling) of 4 percent is expected at this burn-up. However, a volume increase of 8 to 10 percent is easily tolerated by the sodium-bonded fuel rod. A maximum burn-up of 6,000 to 6,500 MWD/MTU is probably a reasonable expectation for this fuel. Sodium bonding was chosen over contact bonding of the cladding to the fuel rod in order to accommodate fuel swelling without straining the cladding and in order to use a thin cladding.

LITHIUM-COOLED RADIAL BLANKET

The lithium-cooled radial blanket located behind the fission blanket extends from the top to the bottom sodium plenums. The blanket consists of a cylindrical stainless steel container 62 centimeters wide and 750 centimeters high. From the inside face of the blanket there is first a 2-centimeter thick stainless steel inner wall, then 6 centimeters of lithium, 50 centimeters of stainless steel-clad graphite, 2 centimeters of lithium, and a 2-centimeter thick stainless steel outer wall. The lithium which is enriched to 50% li^6 enters the reactor from an inlet header and is fed into a plenum at the bottom of the radial blanket. Lithium flows upward through the two channels provided on either side of the clad graphite

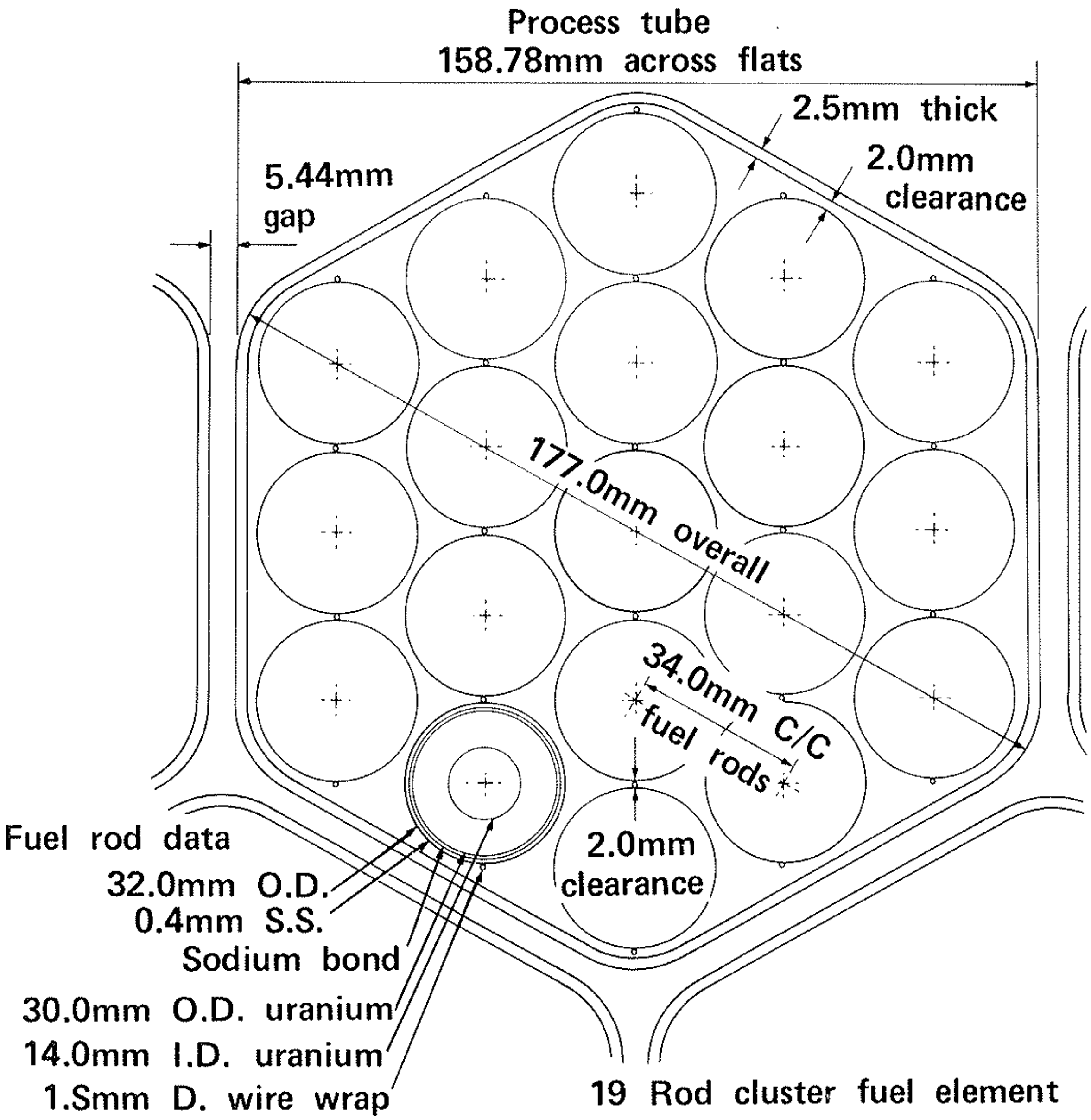

FIGURE 6. Fuel Assembly Cross Section (Full Size)

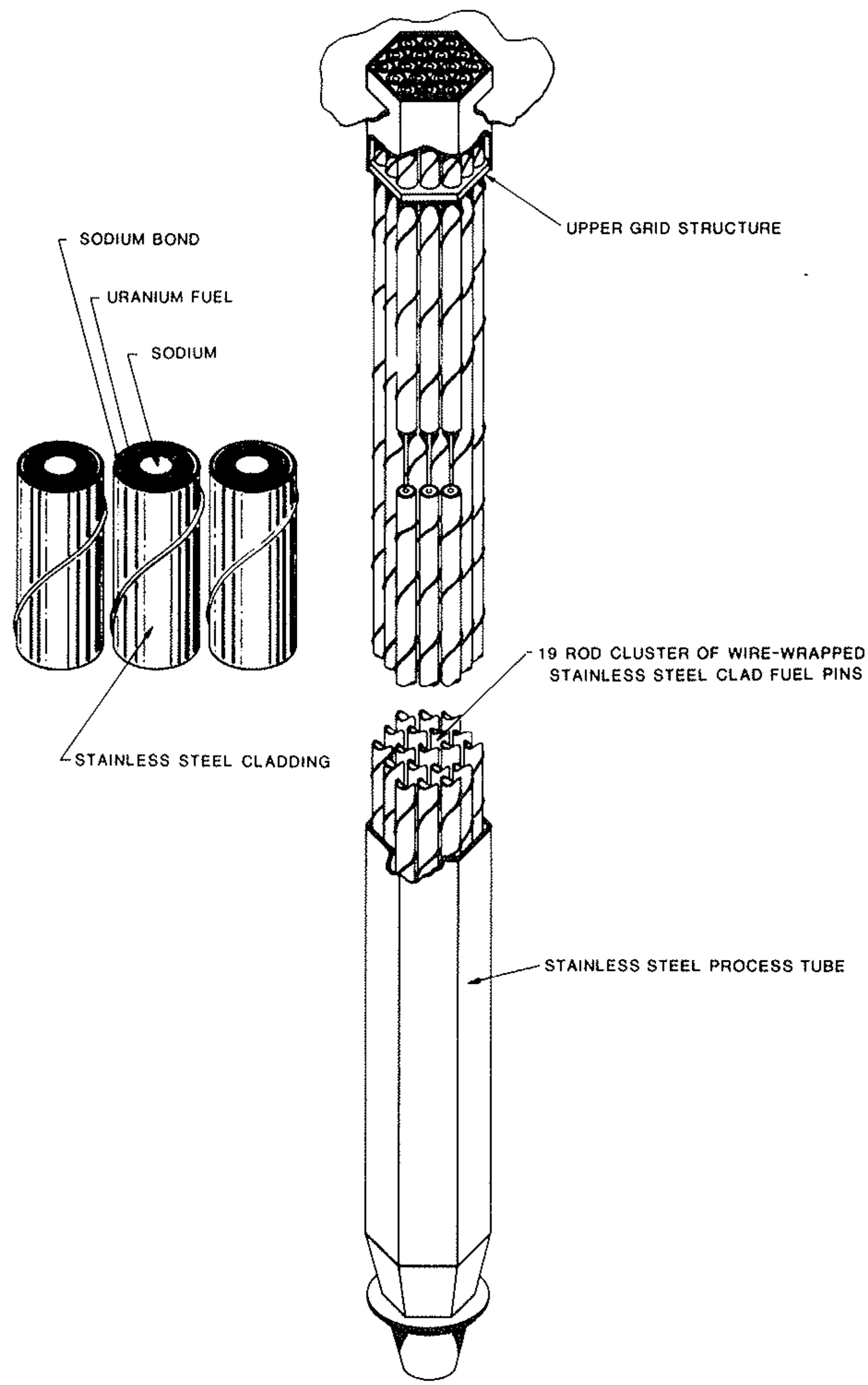

FIGURE 7. Fuel Element and Process Tube Configuration

and into the top plenum. The radial blanket is stiffened internally with stainless steel plates and is primarily supported from the bottom sodium plenum.

TOP AND BOTTOM BLANKETS

The top and bottom blankets are cylindrical curved pancake shaped plenums. The shell is 2 centimeters thick, and from the center of the reactor each blanket has the identical construction of 2 centimeters of stainless steel, 10 centimeters of beryllium, 70 centimeters of graphite, 10 centimeters of lithium, and 2 centimeters of stainless steel shell. The blankets are located 350 centimeters from the center of the reactor. The use of beryllium in these blankets provides a tritium breeding ratio of 1.5. This allows the radial blanket to operate with a tritium breeding ratio of 0.95 and still maintain an overall tritium breeding ratio of 1.1. Reducing the tritium breeding requirements in the radial blanket make it possible to use a thicker fission zone, thereby increasing the fissile fuel production capability of the reactor.

FINAL LASER OPTICAL SYSTEM

Six laser beams approach the reactor from one side of the reactor containment facility. A radiation shield is provided for each beam to prevent radiological hazard in the laser facility. The horizontal beams enter the reactor through a double mirror system located on the outside of the reactor. The laser light is focused on the fusion target by six metal mirrors located at the top and bottom of the reactor. The focal length of the mirrors is 12 meters, and the final mirror is designed to accommodate a beam diameter of 1.2 meters, and a maximum energy flux of 1.5 joules per square centimeter.

To be good reflecting and focusing elements, mirrors must be smooth to approximately one-fourth of the wavelength of the laser light impinging on them. Highly polished metal surfaces are susceptable to all types of radiation damage and we are currently investigating the effects of the neutrons, x-rays, and energetic debris from the fusion microexplosion. The damage caused by this loading has been lessened somewhat by placing the mirror at a point where the radiation fluxes emmenating from the fusion target are more than an order of magnitude lower than the first wall fluxes. An additional problem uncovered by our preliminary optical studies involves damage to the mirrors from the laser light as debris from the fusion chamber accumulates on the mirror surface. A mechanism for removing the debris between shots will have to be devised.

Removing and replacing mirrors will be expensive because of the handling required. Replacing the mirrors at the bottom of the reactor will be difficult without special equipment. The lifetime of the mirrors will determine whether it is necessary for special designs to be used such as rotating mirrors, gas windows, magnetic field director and special remote handling for replacing mirrors. No consideration has been made within this report for optical system design except for layout and basic laser system functional purposes surrounding the reactor.

PELLET APPARATUS

The pellets of deuterium-tritium must be injected into the reactor at a nominal 20 times a second and must reach an exact location without error. The apparatus must be insulated from the reactor (if frozen pellets are to be used) and must remain correctly aligned with laser beams during the expansion of the reactor caused by internal heat load. The apparatus is also subject to thermonuclear blast and the normal reactor vacuum of 0.1 torr. No apparatus is available today for this purpose.

FIRST WALL CONSIDERATIONS

The technological and economic feasibility of the laser fusion hybrid are critically dependent on the design and performance of the first wall because its radius and lifetime determine both the size of the reactor for a given output power and the availability of the power plant. We have performed calculations to determine the radiation exposure capabilities of several first wall materials. The results of these calculations have led us to choose a 2 cm thick graphite liner which is supported by stainless steel and cooled with lithium. The graphite first wall is designed for an operational lifetime of one year with a neutron wall loading of one megawatt per square meter and a repetition rate of 20 Hz. The nominal charged particle loading is then 0.25 MW/m^2 or 12.5 KJ/m^2 per pulse. The charged particle energy is carried in the form of pellet debris and alpha particles which escape the pellet.

Calculations have been performed to determine the temperature rises, the stresses and the amount of material vaporized by the interaction of x-rays, charged particles, and reflected light with the first wall material. The analytical methods used for these estimations have been developed by Hovingh and they are presented in Reference 7. The rate and depth of energy deposition in the first wall from the thermonuclear burn products and the reflected laser light is dependent on several parameters. They include:

- laser wavelength
- laser energy and power
- thermonuclear yield
- pellet mass and composition
- gas pressure in the fusion cavity
- first wall composition and configuration

A computer code called LASNEX has been developed at LLL to explore this complex parameter space[8]. LASNEX is a Lagrangian hydrodynamic code which incorporates the principal physical processes that occur in laser produced plasmas and computes the time evolution of the basic physical characteristics of the plasma. By using LASNEX, it is possible to calculate the transport and interaction of laser photons, electrons, ions, x-rays, and fusion reaction products along with the induced magnetic and electric fields and the hydrodynamic behavior of the pellet.

We have selected a laser target which LASNEX predicts will yield 10 MJ of thermonuclear energy from an implosion caused by a few hundred kilojoules of 1 μm laser light. One percent of the fusion energy is released in the form of x-rays, 23% in charged particles and 76% in 14 MeV neutrons. The energy spectra and pulse widths occurring at the first wall have been determined by continuing the LASNEX calculation long enough for the x-rays and charged particle to interact with the cavity gas at 0.1 torr. The resulting spectra and pulse width have then been input into the first wall calculations and the results indicate that the temperature rise in a graphite wall 3.5 meters from the microexplosion can be kept below the vaporization temperature of 3500°K. Spallation of the graphite caused by the temperature induced stresses has not been considered in these first wall calculations.

With the graphite held below its vaporization temperature and spallation not considered, the major factor limiting first wall lifetime will be erosion from the formation of hydrocarbons.

For a DT pellet with five percent burn, the graphite will erode at a rate of less than one centimeter per full-power year, assuming that all the hydrogen reacts with the curtain to form acetylene.

This graphite curtain must be flexible to withstand the thermal stress caused by reflected light and x-ray loadings, as well as by charged particle loading. A weave of graphite fibers has been proposed for this purpose, but the transmission of heat through such a cloth is uncertain. It may be preferable to use a one-layer, two-dimensional weave that is continuously replaceable as it erodes.

In summary, the first wall is assumed to be a 2-centimeter thick graphite curtain supported on a stainless steel backing. The design of a cooling system for this structure is not included, and the structural design for sufficient flexibility has not been considered in detail.

ANALYSIS OF THE DESIGN

SYSTEM PERFORMANCE

The overall performance and the more significant design parameters of the laser fusion hybrid are summarized in Table 1. A thermal output of 1400 MW was chosen to emphasize that a laser fusion driven hybrid could operate as a relatively small power unit. The fusion targets are irradiated by a 6 beam, 100 KJ, 20 hertz laser with an overall efficiency of 2%. The fusion energy gain (i.e. the product of laser system efficiency and pellet gain) for this reactor is 2.0. This results in a plant recirculating power fraction of 25% and a net system efficiency of 29%. If the fusion energy gain were increased to 4.0 the recirculating power would decrease to 16% and the net system efficiency would increase to 32%.

The performance and the design parameters presented in Table 1 can be placed in perspective by comparing the laser fusion hybrid to a typical fast breeder reactor. This comparison is shown in Table 2 where both systems have been normalized to a thermal output power of 2500 MW. The fast breeder reactor used in this comparison is an LMFBR with a breeding ratio of 1.2. As shown, the laser fusion hybrid generates 30% less electrical power because it is being driven by a

TABLE 1

LASER FUSION HYBRID DESIGN PARAMETERS

SYSTEM PERFORMANCE

Thermal Power, MW_t	1400
Fusion Thermal Power, MW_t	200
Gross Electrical Power, MW_e	535
Net Electrical Power, MW_e	400
Recirculating Power Fraction	0.25
System Efficiency, %	0.29
Average Blanket Energy Multiplication	8.7
Net Plutonium Production, Kg/yr	1300
Total Tritium Production, Kg/yr	8.0
Laser Energy, KJ	100
Laser System Efficiency, %	2.0
Power Supply Energy, MJ	5.0
Pulse Repetition Rate, sec^{-1}	20
Pellet Gain, Q	100
Fusion Energy Gain	2.0

OPTICAL TRANSPORT SYSTEM

Number of Beams	6
Maximum Energy Flux, J/cm^2	1.5
Beam Diameter, m	1.2
Focal Length of Final Mirrors, m	12.0

TABLE 2

COMPARISON OF LASER FUSION HYBRID AND FISSION BREEDER PERFORMANCE

	HYBRID	FISSION BREEDER
Thermal Power, MW_t	2500	2500
Net Electrical Power, MW_e	725	1000
System Efficiency	0.29	0.40
Net Fissile Production, Kg/yr	2300	260
Fissile Fuel Loading	0.0	2500
Maximum Power Density in Fuel, W/cm^3	150	1500
Average Power Density, W/cm^3	30	$\sim$ 300

The fission breeder used in this comparison is an LMFBR with a breeding ratio of 1.2.

laser which requires 19% of the gross power. This inferior performance in power generation results from design choices which were influenced by our desire to emphasize fissile fuel production at the expense of energy multiplication. The advantages of the laser fusion hybrid over the LMFBR are readily apparent from Table 2. Specifically, the hybrid produces 10 times more fissile material, requires no initial fissile fuel loading, and operates at one-tenth the power density. With no initial fissile inventory it becomes possible to operate the hybrid in a regime were both criticality accidents and core disruptive accidents are impossible. Moreover, control rods are not required. The lower power densities make it possible to design a hybrid blanket which provides much more time to recover from a loss of coolant accident. In fact, it is technologically feasible. and it may be economically feasible. to design a hybrid blanket which passively copes with a loss of coolant accident.

ECONOMIC ANALYSIS

The capital and operating costs of the laser fusion hybrid in this conceptual design were estimated by Bechtel Corporation. Their preliminary economic analysis of the reference 1400 MW(t) design revealed that severe economic penalties resulted from some of the design choices. A survey of the high cost items indicated that the reactor containment structure and several of the other buildings had been sized much too large for the nominal output power of 400 MW(e). In addition, there were several other balance of plant items whose costs were relatively independent of output power, thereby implying that a larger plant output power would be more economical. These results agree with scaling factors for other nuclear power reactors in that an electrical power plant is more economical in the 1200 MW(e) range and, where possible, in twin units.

The results presented above led us to perform our more detailed cost analysis on a hybrid with a larger output power. We scaled our conceptual design to a size which had a gross yield of 1300 MW(e) and a net yield of 950 MW(e). This output was obtained from the original design by increasing the laser energy from 100 KJ to 200 KJ increasing the average pulse repetition rate from 20 to 25 Hz and increasing the inner radius of the blanket from 3.5 to 6.0 m.

CAPITAL AND OPERATING COSTS

The capital cost of the laser fusion hybrid reactor plant has been estimated from conceptual design and engineering information. A large portion of the power plant consists of conventional technology such as thermal energy transfer, electrical generation, cooling systems, and auxiliary systems; therefore, cost estimating can be based on background experience. The fusion reactor and the laser interface fusion fuel cycle being conceptual have been estimated on a first-of-a-kind cost basis. The operating costs of the laser fusion hybrid reactor plant are based upon nuclear fuel cycle and equipment replacement costs of this reactor, capital charge rates, and general operating and maintenance costs similar to those of LMFBR reactors.

CAPITAL COSTS

In general, the power conversion system of the laser fusion facility is similar in function to a liquid metal fast breeder reactor (LMFBR). The plants differ mainly in design of the reactor, the reactor building, and the lithium and tritium systems. The approach adopted in estimating the overall plant cost has been to use light water reactor (LWR) experience where appropriate; e.g., in determining the extent and cost of electrical, piping, and instrumentation systems. With the exception of the reactor containment and steam generation buildings, the civil and structural costs were derived in the same manner.

Systems unique to the fusion-fission heat source were evaluated differently. The reactor cost was determined on a unit weight basis and the tritium system cost was established on the basis of its component equipment costs. The estimates are based on the conceptual design and engineering drawings, outline specifications, and equipment lists.

The results of the cost analysis are summarized in Table 3. For comparative purposes the costs of the laser fusion hybrid have been presented along with cost estimates for a typical LWR. All of the cost estimates have been made at first quarter, 1976, price and wage levels and no allowance has been made for future escalation. The Nuclear Steam Supply System (NSSS) category for the hybrid primarily consists of the reactor vessel with its internals, and the primary coolant loop with its associated pumps, motors, heat exchangers, and steam generators. Major items included in

TABLE 3

CAPITAL AND OPERATING COST ANALYSIS

Capital Cost Item (10^6 \$)	LWR 1200 MW(e)	Laser Fusion Hybrid 950 MW(e)
Nuclear Steam Supply System (NSSS)	78	268
Other Mechanical	101	201
Civil and Structural	142	158
Piping	77	105
Instrumentation	9	11
Electrical	43	72
Total Direct	450	815
Field Costs	79	171
Engineering Services	80	197
Contingency	91	272
Owners Cost at 7%	56	116
Interest During Construction at 8%	197 (9 yr)	487 (10.5 yr)
Total Indirect	503	1243
TOTAL COST	953	2058
Cost per kW installed (\$)	794	2166
Operating Cost Item (mills/kWh)		
Capital	19.42	55.77
Fuel	6.3	(-3.17)
Operating and Maintenance	1.5	2.40
Total Operating	27.22	55.00

the category "other mechanical" are the turbine generators, the vacuum system, the tritium system, and the cooling towers. Site improvements, the reactor containment structure and all the buildings make up the civil and structural category.

The indirect costs in Table 3 were estimated on the basis of a nine year construction time for the LWR and a 10.5 year construction for the more complex laser fusion hybrid. As a result of this, the indirect costs for the hybrid account for a larger fraction of the total capital cost. Field costs are those items of construction cost which cannot be ascribed to the direct portions of the facility. They include temporary construction facilities, supply and maintenance of construction equipment and tools, field office operation, acceptance testing, project insurance, Taxes and permits. The engineering services include all engineering costs and home office costs and fees. Included in the indirect cost is a contingency allowance for the uncertainty that exists within the conceptual design in quantity, pricing, or productivity.

The total capital cost of the laser fusion hybrid is estimated to by $2,058 million. Thus on a cost-per-kilowatt installed basis, the hybrid is 2.7 times more expensive than the LWR. It should be noted that this cost estimate does not include the laser system or the pellet manufacturing facility. If $200 million dollars are allowed for these omitted facilities the laser fusion hybrid would cost approximately three times more than a typical LWR.

OPERATING COST

The cost of electricity from the hybrid is 55 mills/KW hr. This is approximately twice as much as the cost of electricity from the LWR. The capital portion of the operating cost is by far the dominant factor in the cost of electricity. It has been estimated for both reactors on the basis of a 15% rate of return on the capital invested. The fuel cycle cost for the laser fusion hybrid is negative because of revenues obtained from the sale of the plutonium it produces. The cost bases used to estimate the fuel cycle cost for both the LWR and the hybrid are presented in Table 4. The fabrication cost for the hybrid is cheaper because the cladding material is stainless steel and the cross sectional area of the fuel pin is much larger. Both the spent fuel shipping and the reprocessing costs are less for the hybrid because its fuel is being operated at lower average burn-ups (6000 vs 33,000 MWD/MTU).

The major issue concerning a laser fusion hybrid is not how much it will cost nor the price at which it can generate electricity, but rather the cost of electricity in a scenario with hybrids providing fissile fuel for existing burner reactors. In Figure 8 the cost of electricity has been plotted as a function of the cost of fissile fuel for an LWR and hybrids with varying capital costs. The intersection points of the curves determine the cost of electricity and fissile fuel in the hybrid-LWR scenario. These results indicate that the cost of electricity is quite insensitive to the capital cost of the laser fusion hybrid. Specifically, the cost of electricity increases by only 20 to 40% when the capital cost of the hybrid ranges from 2 to 3 times more than the LWR.

CONCLUSIONS

The production of fissile fuel by a hybrid is a promising step in the development of fusion. This study has disclosed a number of advantages resulting from the addition of a depleted uranium fission blanket to a laser fusion system. These include:

1. The hybrid operates in a regime which requires an order of magnitude less laser/pellet performance than a pure laser fusion system.
2. First wall requirements and 14 MeV neutron damage are less severe in a laser fusion system with a fission blanket.
3. The laser fusion hybrid produces a large amount of fissile material - enough to fuel more than six LWR's of equivalent size.
4. In a scenario with laser fusion hybrids producing fuel for existing reactors the cost of electricity is insensitive to the capital cost of the hybrid.
5. The laser fusion hybrid would extend the total energy available from economically proven light water reactors by two orders of magnitude.

The feasibility of the laser fusion hybrid should be evaluated from three points of view: Scientific, technological and economic. The scientific feasibility of the laser fusion hybrid is dependent on (1) achieving pellet gains in the neighborhood of 100, and (2) developing suitable lasers with overall system efficiencies greater than 1%.

TABLE 4

FUEL CYCLE COST BASES

PWR/BWR REACTORS

Uranium	\$40/lb U_3O_8
Conversion	\$4.50/kg U
Enrichment	\$100/SWU
Fabrication	\$100/kg U (PWR)
	\$ 80/kg U (BWR)
MO Fabrication	\$250/kg U (PWR)
	\$180/kg U (BWR)
Spent Fuel Shipping	\$ 20/kg U
Reprocessing	\$225/kg U
Plutonium Credit	\$34.25/g Pu_f (PWR)
	\$26.95/g Pu_f (BWR)

70% Plant Capacity Factor

17.4% Working Capital Charge Rate

Process Losses:	Conversion	0.2%
	Fabrication	0.5%
	Reprocessing	0.5%

Laser Fusion Hybrid Reactor

Fabrication	\$30/kg U
Spent Fuel Shipping	\$10/kg U
Reprocessing	\$125/kg U
Plutonium Credit	\$30.00/g Pu_f

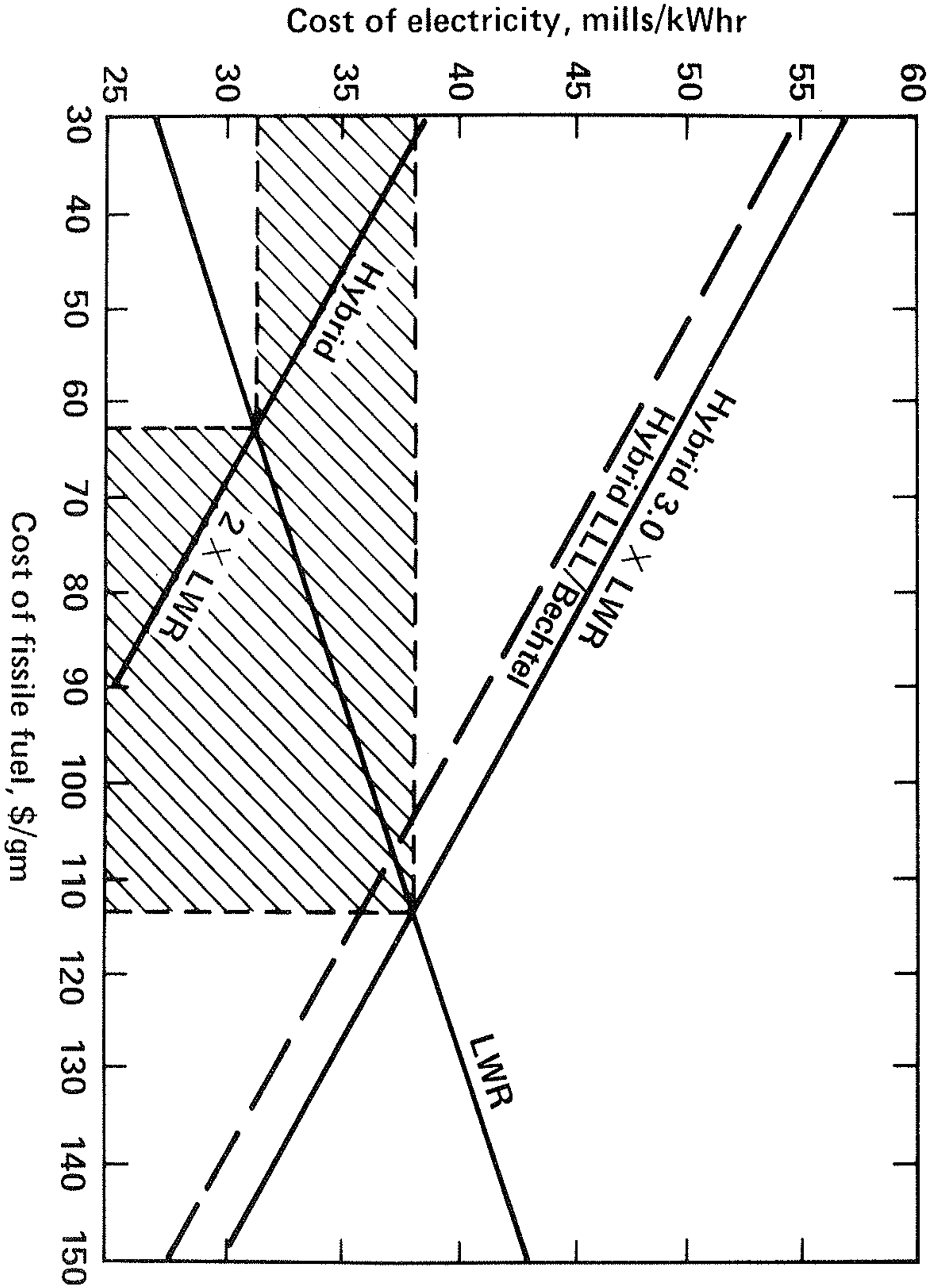

FIGURE 8. Cost of Fuel and Electricity from a Hybrid

The fission blanket surrounding the fusion chamber was designed with state-of-the-art fission technology. This was done to facilitate a straight forward assessment of the technological feasibility of the laser fusion hybrid, but a definitive statement regarding the feasibility of fissile fuel production with laser fusion cannot be made without further study. The laser and optical systems need to be conceptually designed. The pellet manufacturing and injection systems need to be considered. A more detailed analysis of the first wall design should be carried out. Finally, a safety analysis of the design is required with particular attention being given to system failures which could result in a release of radioactive nuclides to the environment. The most obvious release mechanism being melting of the fuel in a loss of coolant accident.

The economic analysis shows that the cost of electricity in a hybrid-LWR scenario is insensitive to the capital cost of the hybrid. The laser-fusion is estimated to be three times more expensive than an LWR. The cost of electricity is shown to be only 40% more than the present price. Nevertheless, substantial economic gains would be realized if the laser fusion hybrid's cost could be decreased to twice that of an LWR. Possibilities for reducing capital cost which should be explored in future studies include:

1. Replacement of the reference coolant and tritium breeding systems with helium cooling and a solid lithium blanket.
2. Investigation of fission blankets which enhance energy multiplication, and
3. Consideration of blanket geometries which more efficiently utilize the point source from laser fusion.

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LASER SOLENOID FUSION-FISSION DESIGN

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ABSTRACT

The dependence of breeding performance on system engineering parameters is examined for laser solenoid fusion-fission reactors. Reactor performance is found to be relatively insensitive to most of the engineering parameters, and compact designs can be built based on reasonable technologies. Point designs are described for the prototype series of reactors (mid-term technologies) and for second generation systems (advanced technologies). It is concluded that the laser solenoid has a good probability of timely application to fuel breeding needs.

INTRODUCTION

The laser solenoid (Fig. 1) is a linear high-beta fusion concept in which plasma heating is achieved by absorption of an axially directed laser beam together with a moderate amount of magnetic compression; confinement is supplied by the magnetic field produced by a large bore steady superconductor (outside the blanket and shield) and several small bore pulsed coils (spaced at intervals in the blanket). It is a multiple tube pulsed system; each tube is fired in sequence, and the laser beam is directed to the proper tube by the optical transfer system.

The obvious question is, why is the laser solenoid being considered as a fusion-fission option, and how does it stand out from other exploratory concepts. There are a number of unique aspects of the laser solenoid which give it one attractive aspect or another, but two stand out in particular, the small bore magnet, and the laser. With a small bore magnet, very high magnetic fields (300-500 kG) can be achieved for reasonable stress levels. Since the length of linear systems generally scales as the inverse square of the magnetic field, relatively modest length linear systems

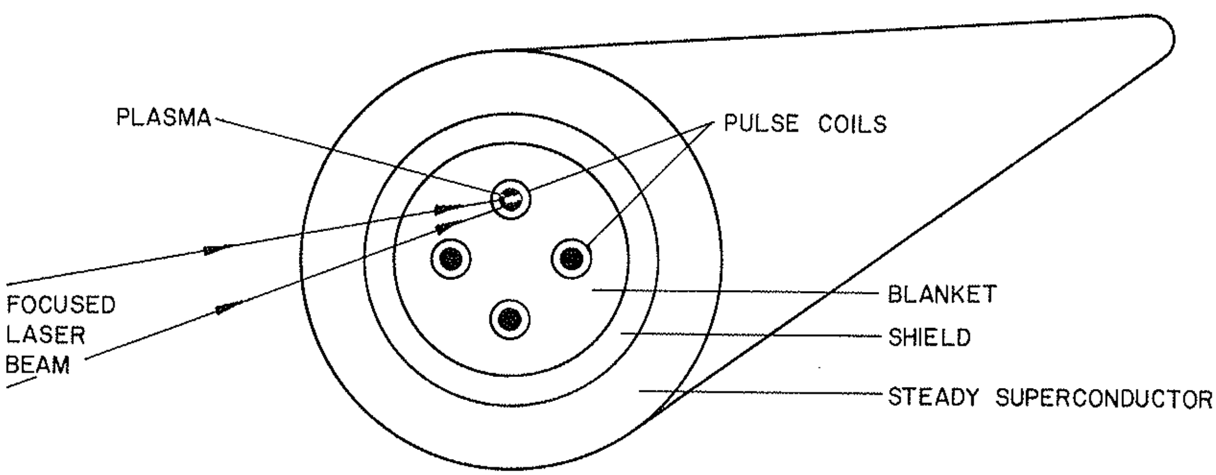

Fig. 1. Schematic diagram of laser solenoid reactor elements.

(100-300 m) can be conceived compared to larger bore, lower field devices. Moreover one need not invoke unknown or uncertain end-stoppering techniques to achieve a compact reactor design. The laser, which supplies a crucial part of the plasma heating, is another distinctive feature of the laser solenoid. The long pulse CO_2 laser is an efficient well-developed source of light which has demonstrated scalability to large sizes. As a plasma heater, the laser energy can be focused into a slender plasma, and the energy coupling to the plasma is at least fair (based on classical absorption) for short systems. Moreover, the laser heater can be isolated from the fusion core and located some distance away, due to the superior focusability of the beam: thus there is little difficulty in protecting the heater from streaming plasma or radiation, nor is it necessary to have large open channels passing through the blanket in order to transmit energy from the heater to the plasma.

Other advantages are worth mentioning as well. There is the inherent simplicity of the straight solenoid configuration. This lends itself to sectioned construction with a large number of small size modular units, and the consequent suitability for factory production methods. Indeed, the engineering and construction of a superconductor in simple solenoid section is vastly more feasible and straightforward than for noncircular and toroidal elements. Moreover, there is great potential for access in the linear system, from both sides and from the ends, although the presence of the superconductor sections around the blanket is a complication. The redundancy of the multitube system can lead to more reliable operation, since the malfunction of one tube does not necessarily force shutdown of the others. Because of the small plasma chambers, there is little likelihood of the assembly of a critical mass by collapse of part of the blanket into the void, an important safety advantage. Finally, as will become clear in the present paper, the laser solenoid scales very nicely to compact size devices for fusion-fission applications. This attractive feature is a consequence of the difference in laser solenoid geometry and heating methods compared to other fusion concepts. Stated another way, the fusion-fission reactor is much smaller than its pure fusion counterpart, an advantage which can have great economic benefits.

The results presented in this paper are an outgrowth of a recent laser solenoid reactor feasibility study.[1] This study covered a broad range of crucial reactor aspects. The plasma physics of both the heating phase and the ensuing fusion burn were treated. Time-dependent plasma characteristics were analyzed accounting for end-loss of particles and energy, alpha particle effects, and transverse diffusion. These studies determined the laser and magnetic field energies required to create the plasma, and the fusion energy generated.

Laser solenoid neutronics were studied using the neutron transport code ANISN. In particular these calculations studied the effects of (1) a moderately thick resistive magnet between the plasma and blanket, and (2) other magnets nearby in the multi-tube design. The latter required modeling assumptions in order to apply a cylindrically symmetric code to a nonsymmetric geometry. The dependences of tritium breeding ratio (and fissile breeding in fusion-fission blankets) on magnet thickness, magnet spacing, blanket thickness and blanket composition were examined. Also calculated were radiation damage indices; stainless steel displacement rates, helium and hydrogen implantation rates, and neutron leakage from the shield to the superconductor.

Accompanying the analyses of plasma physics and neutronics were studies of reactor subsystems, the most extensive being the magnet feasibility study.[2] The superconducting magnet design was based on a support scheme in which strain of the windings is essentially eliminated by using a controlled pressure bag between each layer of windings and the surrounding layer of support material. The pressure bag concept allows a significantly reduced thickness of windings and support, compared to conventional designs. Design details were determined for a superconductor employing layers of NbTi, Nb_3Sn, and (if fields over 150 kG are assumed) V_3Ga. The pulsed coil design was based on strip-wound layers incorporating laminated conductor and support materials. The stored energy, resistive loss, and time-dependent stresses were determined. A power supply design was proposed incorporating electrostatic storage for fast current rise in conjunction with a homopolar generator for pulse flat-topping. The pulsed coil first excludes the field from the plasma tube (to permit creation of a nearly field-free plasma), and then raises it to approximately double the superconductor field. Materials problems of both magnets were assessed including, in particular, difficulties due to high cyclic stress in the presence of intense radiation (pulsed coil). The advantages of cooled (near room temperature) operation were demonstrated, for reducing both resistive losses, and mechanical degradation due to helium implantation.

Other reactor subsystems were studied including plasma chamber (first wall) design, laser module design, and tritium recovery and handling. The reactor component studies were integrated with the plasma and neutronics calculations to form self-consistent conceptual reactor designs.

In the following discussion, a simple parameter study is presented which explores the dependence of breeder performance on various scale and engineering parameters, such as reactor length, confinement model, magnet stress, superconductor field, etc. Out of the parameter study, three conceptual point designs are developed, including two prototype series reactors which have less stringent engineering parameters such as might be achieved with mid-term technology in first generation systems. The third point design is a second generation reactor incorporating engineering parameters consistent with more advanced technologies.

PARAMETER STUDY

The baseline quantities taken in the parameter study are shown in Table 1: this is the condition about which individual parameters are varied. These parameters are consistent with reasonable expectations of the technology that would be available for first generation systems. The plasma end-loss is free-streaming with self-mirroring, accounting for the effects of time-dependent beta.[3] The plasma temperature is determined by requiring that the classical absorption length be 10 times the device length at the end of the laser heating phase, which is consistent with 10 x classical absorption, or with multiple passing of the beam through the plasma (10 times). Single pass, classical absorption systems were also studied but resulted in longer devices and larger lasers in order to achieve comparable breeding performance. The plasma temperature is assumed to be roughly constant during the fusion reaction phase. Resistive loss computations assume that each pulsed coil section is turned off slightly after passage of the area wave front

Table 1. Assumed baseline quantities for parameter study.

Item	Parameter
Magnet:	
inner diameter	6 cm
thickness	3 cm
length	200 m
maximum tensile stress	
support material	120,000 psi
conductor material	45,000 psi
volume fraction	
support material	0.5
conductor material	0.45
temperature	100 °C
conductor resistivity	3.5 μΩ-cm
superconductor field	150 kG
total pulsed field (derived quantity)	338 kG
Laser:	
pulse energy	10 MJ
absorption coefficient	10 x classical
initial plasma radius (laser heating phase)	~2.5 cm
Plasma Confinement	free-streaming (self-mirroring)
First Wall Radius	2.5 cm
Efficiencies:	
thermal conversion	0.35
electrical switching	0.95
laser electrical	0.25
Power:	
thermal power	4000 MW_t
circulating power fraction	1.0

(self-mirrors), which can be done without increasing the end-loss rate.[3] The blankets considered are based on depleted uranium (metal) with an without plutonium enrichment. Details of blanket construction are described in Ref. 4. Both blanket and streaming plasma energies are converted to electricity at the thermal conversion efficiency.

Given the reactor parameters, the fusion energy produced and the required circulated energy per pulse can be determined. Therefore, to achieve a circulated energy fraction of unity, a certain fusion energy multiplication is required. This is achieved in the fusion-fission blanket by enrichment with an appropriate concentration of fissile atoms: the higher the required energy multiplication, the higher the required energy multiplication, the higher the enrichment fraction. In summary then, the logic taken for this

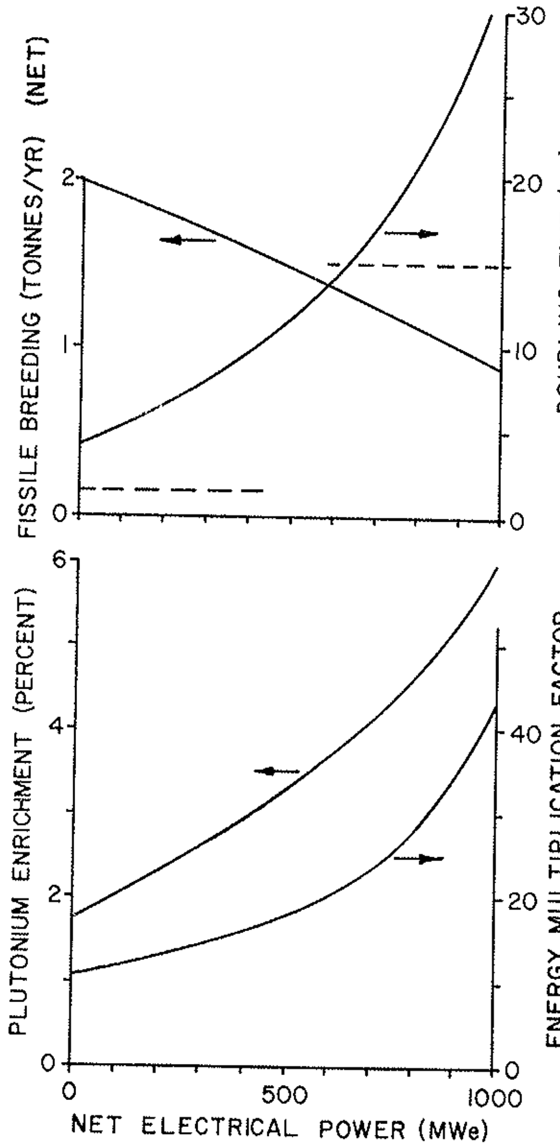

Fig. 2. Dependence of breeding performance on net electric power.

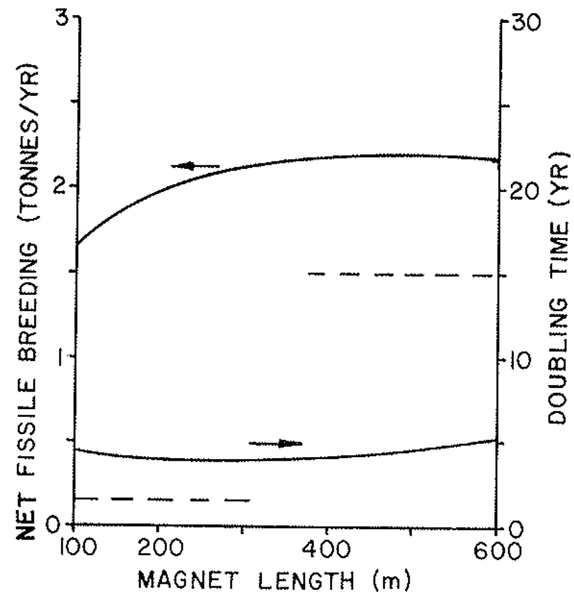

Fig. 3. Dependence of breeding performance on magnet length.

particular parameter study is that deficiencies in system energy balance are made up by choosing an appropriate enrichment fraction of fissile isotope in the blanket. This, of course, is one of a number of strategies that could be taken.

NET ELECTRICAL POWER

One of the important unresolved questions regarding fusion-fission is whether reactors should be devoted to fissile fuel breeding alone or to net electrical power production plus a modest amount of breeding. No attempt is made to resolve this issue here. Along this line, it is of interest to examine the relation between breeding characteristics and net electrical power production. This is shown in Fig. 2, where all the assumed quantities in Table 1 are held constant except the circulating power fraction. The dashed lines in the upper figure represent a projected net breeding rate (0.15 tonnes/yr) and doubling time (15 yrs) for a liquid metal fast breeder reactor (LMFBR). The laser solenoid reactor is a very good breeder of plutonium, but may have some difficulties with long doubling time if too much net power is demanded. Figure 2 represents only a trend though, and specific designs can be conceived (at the expense of larger sizes or stiffer engineering requirements) which produce net power with no initial charge of plutonium and hence zero doubling time.

PLASMA LENGTH

The most serious problem with linear fusion systems is plasma end-loss, which necessitates uncomfortably long systems to achieve adequate confinement. One of the objectives of linear fusion-fission reactor design is to get a much shorter system, since confinement time need not be as long: a definite economic benefit accompanies the more compact device.

Figure 3 portrays the dependence of breeding performance on magnet length. One thing is immediately clear, the laser

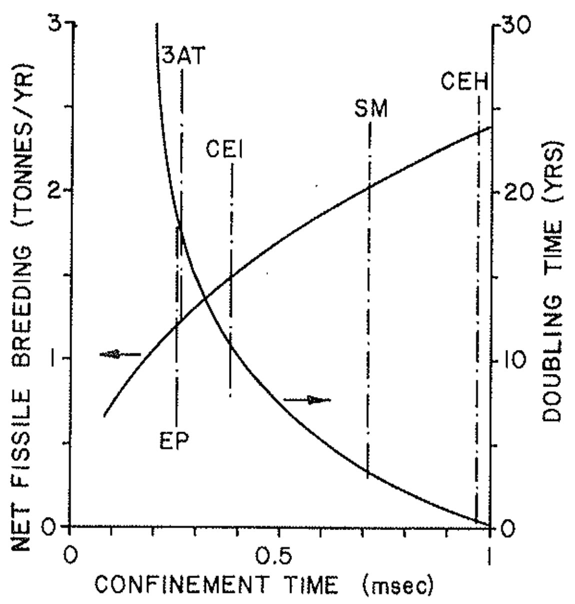

Fig. 4. Dependence of breeding performance on confinement model. EP = end-plugs; 3AT = three acoustic transit times; CEI = cusp-ended solenoid with ion larmor radius hole size; SM = self-mirroring (varying beta); and CEH = cusp-ended solenoid with hybrid radius hole size.

solenoid scales very nicely to short systems. Indeed it appears that lengths as short as 100 m or less are possible without seriously degrading breeding performance. Moreover, these results are based on end-loss retarded only by self-mirroring: unknown or uncertain end-stoppering techniques are not invoked. An unusual effect is apparent in Fig. 3; the doubling time is nearly independent of length. This effect follows because doubling time equals the plutonium change (tonnes) divided by the net breeding rate (tonnes per year), connected with the facts: (1) plutonium charge is proportional to length as well as plutonium enrichment fraction, and the former grows slightly faster than latter decreases in this case; and (2) at fixed thermal power, the breeding rate does not change markedly. For other conditions (e.g. larger laser), the doubling time drops to zero with increasing length because the enrichment fraction goes to zero, as will be seen later.

CONFINEMENT MECHANISM

The dependence of breeding performance on confinement mechanism is illustrated in Figure 4. Several models are shown: (1) a simple three acoustic transit times model (time for a sound wave to traverse half the plasma length); (2) free-streaming end-loss with self-mirroring and accounting for variations in beta; (3) solid end-plugs (no particle loss but conductive energy loss to the plugs); and (4) cusp-ended solenoid accounting for variations in beta (minimum hole size based on ion larmor radius, and hybrid ion-electron larmor radius). The end-plugged plasma does not appear to perform well in this application because of too rapid energy losses to the cold plugs. The hybrid radius cusp-ended solenoid has the longest confinement, about 1 msec. It only exceeds the ion larmor case by about a factor of 2.5 rather than by the ratio of hole sizes, about 8. This is a consequence of varying beta effects which cause the hole size to be important only early in the confinement period. Although the free-streaming plasma performs adequately, a clear benefit accrues from having longer confinement times, allowing improved breeding performance and/or production of net electrical power. Some preliminary analytic modeling has been performed on plasma end-stoppering to give the results discussed above, but a much more detailed analysis is required to determine the best end-stoppering technique for each application. At present there is virtually no experimental data directly relevant to these theories.

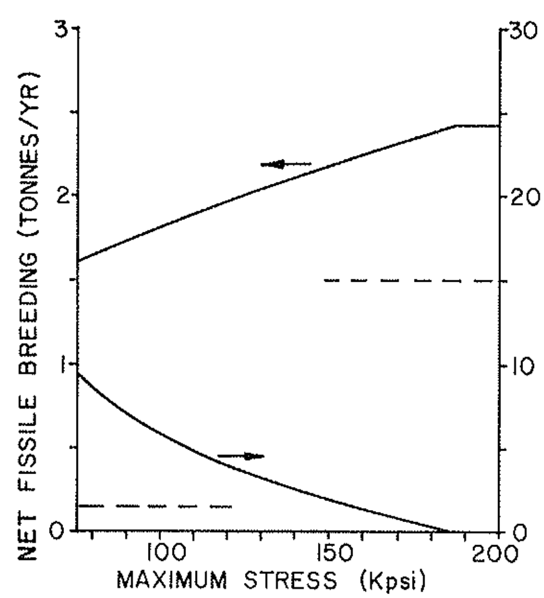

Fig. 5. Dependence of breeder performance on maximum stress in the pulse coil strength material.

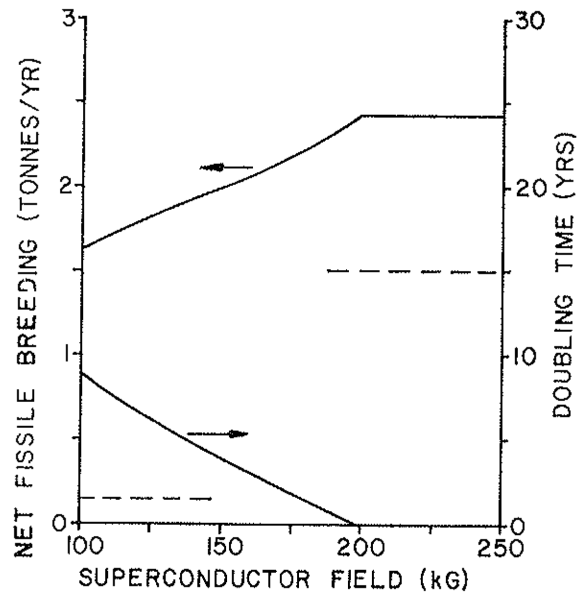

Fig. 6. Dependence of breeder performance on superconductor field.

PULSE COIL STRESS

Rather high stresses are generated in the pulse coil, necessitating that it be composed of layers of conductor and high strength support materials. It was suggested in the laser solenoid reactor feasibility study[1] that Zr-Cu be used for the conductor, and 2.5% Be-Cu or high strength stainless steel for the support. The stresses required for high magnetic field operation (over 300 kG) are generally below the ultimate tensile strength of the materials considered, but above the fatigue strength. However Fig. 5 indicates that reasonable breeder performance can be achieved for stresses which are approximately within the fatigue strength of 2.5% Be-Cu, i.e. 70-80,000 psi and well within the fatigue strength of high strength stainless steels. The bend in the curves at 188,000 psi corresponds to reaching zero plutonium enrichment in the blanket. The stress at the point of zero enrichment could be lowered, for example, by going to longer systems.

SUPERCONDUCTOR FIELD

Figure 6 describes breeder performance as a function of superconductor field. Reasonable performance can be achieved for fields as low as 100 kG or less, and very good improvements accompany higher field operation. The bend in the curves at 200 kG corresponds to reaching a zero plutonium blanket.

LASER ABSORPTION ENHANCEMENT

The baseline condition assumes a factor of 10 enhancement of the laser absorption coefficient, which could be achieved if an anomalous absorption mechanism applies or by multiple passing of the beam, or if a suitable longer wavelength laser is available. Figure 7 describes the effect on breeding performance if less than a factor of 10 is achieved. It appears that the performance is seriously degraded if only classical (single pass) CO_2 laser absorption is assumed for short devices. It is possible to achieve reasonable performance (e.g. doubling time less than 15 yrs) if two passes and a larger laser (20 MJ) are allowed, or if longer lengths are allowed.

In summary, breeding performance is relatively insensitive to reactor length (for lengths of at least 100 m), stress in the pulse coil support material and superconductor field. Breeding performance is moderately sensitive to the net electrical power desired, and the plasma confinement model. Performance is most sensitive to the laser absorption enhancement factor. Improved absorption could be achieved by multiple pass heating which has been demonstrated in a single experiment[5]

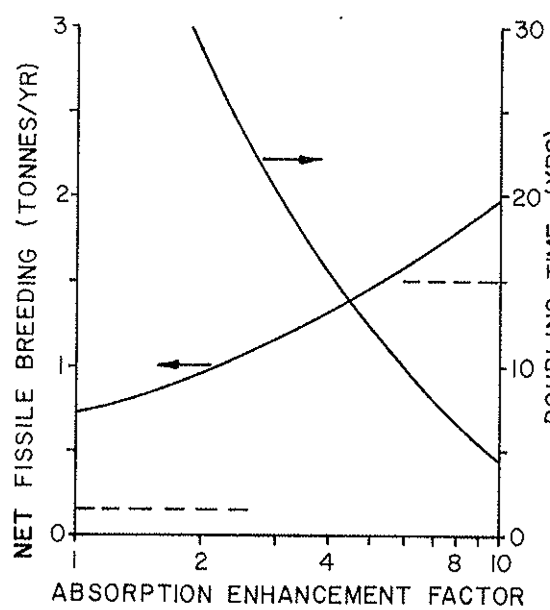

Fig. 7. Dependence of breeding performance on laser absorption enhancement factor.

and is the subject of another currently underway at the University of Washington. Moreover the laser intensities envisioned for reactor devices (10^{11} to 10^{12} watt/cm^2) are well above the thresholds for excitation of backscattering instabilities which could, in effect, lead to enhanced absorption by reflecting the beam back toward the laser. Clearly, the physics and engineering of improved absorption needs to be established more firmly.

CONCEPTUAL REACTOR DESIGNS

Three point design fusion-fission reactors are described. These designs should be construed as illustrative but not optimal or unique. A conceptual point design must be interpreted with caution since it depends on many parameters: a measure of uncertainty necessarily exists about each parameter, related to the future course of technological advance. It may be possible to project the advance of one or two technologies with a degree of confidence, but the multitude of subsystems in the complete reactor defies this kind of exercise. Along this line, the laser solenoid has a significant advantage in that breeding performance is relatively insensitive to most of the important engineering parameters, within certain bounds.

PROTOTYPE SERIES FUSION-FISSION REACTORS

In order to achieve a practical fusion-fission reactor in the mid-term (before the year 2000) projected technology extrapolations from current state-of-the-art must be limited somewhat. This is essential even if it compromises the level of performance and economic benefit that is ultimately desired. Therefore the two first generation systems presented here are based on engineering parameters that can probably be attained in the mid-term with reasonable development programs. Thus, for example, the superconductor field is restricted to 150 kG, based on NbTi (at low field regions) and Nb_3Sn (at high field regions); the pulse coil stress is limited to 120,000 psi; and the reactor is a pure breeder, producing only enough electricity for its own circulating energy needs. The latter consideration relates to the question of producing fuel plus power versus fuel alone. The issue of course has not been settled, but there are two very strong arguments for opting for pure breeding at least in first generation systems: (1) the reliability of a new class of reactors is generally lower during the early years, making it undesirable to have them in the power grid; and (2) the technology is easier for pure breeders (e.g. see Fig. 2).

The two prototype series point designs are presented in Table 2 and represent reactors with no initial charge of plutonium, and with 10 tonnes initial charge, respectively. The zero charge design has lower energy multiplication in the blanket and thus the fusion source must produce more energy. Consequently it is larger both in length, 300 m, and in the required laser energy, 12 MJ. It is a very good breeder of plutonium however, with a net fissile breeding rate of 2.3 tonnes/yr. The rapid breeding rate without the requirement of an initial charge makes this design very attractive for supplying the fuel needs of thermal fission reactors at once, without having to devote a significant period of time simply generating fuel to charge up other breeders.

The 10 tonne charge design has a distinct advantage in size, being only 100 m long, with a 6 MJ laser. Because the blanket multiplication is higher (23), it does not breed fuel as well (1.3 tonnes/yr), but still has a moderately short doubling time of 8 years for the fissile inventory.

SECOND GENERATION FUSION-FISSION REACTOR

A second generation point design, also shown in Table 2, is characterized by more advanced engineering parameters. For example, the superconductor field is 200 kG which would require use of V_3Ga or another high current density conductor in the high field region: the maximum pulse coil stress is 160,000 psi, and the reactor produces 500 MW_e of net power. This design requires no initial charge of plutonium, and breeds plutonium at the attractive rate of 2.4 tonnes/yr.

CONCLUSIONS

Summarizing, conceptual laser solenoid fusion-fission reactors have certain outstanding characteristics.

(1) They are compact. In general linear systems are plagued with rapid end-loss and long lengths; however, laser solenoid systems can be built at lengths of 100 to 300 m. This can be accomplished moreover without invoking unknown or

Table 2. Conceptual Fusion-Fission Reactor Designs

Item	Prototype Series Designs: Zero Pu Charge	Prototype Series Designs: 10 tonnes Pu Charge	Second Generation: Zero Pu Charge
Size of Primary Elements			
length of magnet (m)	300	100	100
superconductor outer diameter (m)	3.5	3.5	4.8
thermal power (MW_t)	- - - - - - - - 4000 - - - - - - - - -		
Magnet System			
pulse coil inner diameter (cm)	5.8	6.2	6.0
pulse coil thickness (cm)	4.0	3.0	4.0
maximum tensile stress:			
support material (psi)	- - - 120,000 - - -		160,000
conductor material (psi)	- - - 45,000 - - -		60,000
volume fraction:			
support material	0.45	0.65	0.5
conductor material	0.5	0.3	0.45
average temperature (°C)	- - - - - - - - -100- - - - - - - - - -		
conductor resistivity (μΩ-cm)	- - - - - - - - -3.5- - - - - - - - - -		
superconductor inner diameter (m)	- - - - 2.8 - - - -		3.0
superconductor field (kG)	- - - - 150 - - - -		200
total pulsed field (kG)	361	353	433
Laser System			
pulse energy (MJ)	12.0	5.6	6.0
effective absorption	- - - - - -10 x classical - - - - - -		
repetition rate (Hz)	9.3	23	20
First Wall			
inner radius (cm)	2.4	2.6	2.5
number of tubes	- - - - - -4- - - - -		10
first wall loading (MW/m^2)	2.4	2.1	2.8
Plasma			
radius (cm)	0.84	0.98	0.85
temperature (keV)	5.08	3.77	4.50
confinement x free-streaming	- - - - - -1- - - - -		1.5
confinement time (msec)	0.81	0.45	0.62
"Q" (fusion energy/plasma energy)	1.15	0.35	1.11
Blanket			
average power density (kW/ℓ)	30	91	91
energy multiplication factor	7.2	22.9	7.2
plutonium enrichment (percent)	0	4	0
K value	<0.3	0.78	<0.3
Efficiencies			
thermal conversion	- - - -0.35- - - - -		0.45
electrical switching	- - - - - - - - 0.95 - - - - - - - - -		
laser electrical	- - - - - - - - 0.25 - - - - - - - - -		
Performance			
net fissile production (tonnes/yr)	2.3	1.3	2.4
fissile doubling time (yrs)	0	8	0
net electrical power (MW_e)	- - - - - -0- - - - -		500

uncertain end-stoppering schemes.

(2) They breed large quantities of plutonium.

(3) No initial plutonium charge is essential.

(4) The design is flexible. The breeding performance level is not sensitive in a serious way to most of the engineering parameters. Therefore the time-table for the construction of a successful fusion-fission reactor would not be seriously delayed if a subsystem technology fails to progress fully to the level projected for a reactor.

Certain key developmental programs are needed which are peculiar to the laser solenoid, relating to both the physics and engineering. The greatest area of uncertainty at present regards the efficient coupling of laser energy into the plasma, at higher than classical absorption. The basic coupling processes (beam trapping and absorption) are being examined carefully under the ERDA-funded proof-of-concept experiment at Mathematical Sciences Northwest. Enhanced absorption by multiple passing of the beam through the plasma is being studied in an ERDA funded experiment at the University of Washington. Anomalous reflectivity processes (which may lead effectively to enhanced absorption) are also under study at the University of Washington.[6] Another area of the physics requiring both experimental and theoretical study is the end-loss problem. This area is less uncertain since projected fusion-fission reactor performance is less sensitive to confinement time than it is to laser absorption enhancement.

Three areas of necessary engineering development stand out; pulse coil and power supply design; large bore, high field superconducting solenoid design; and repetitively pulsed large laser module design. There is a high probability of success in each of these areas due to existing advances, e.g. 150 kG Nb_3Sn superconductors of moderate bore, and high energy, single pulse N_2-CO_2 lasers; and due to the availability of undeveloped but highly promising design schemes conceived during the magnet feasibility study, e.g. the strain-free superconductor design based on a pressure bag support technique, and the laminated strip wound pulse coil design.

In conclusion, laser solenoid fusion-fission has a good probability of success on two accounts: (1) engineering; practical breeder reactors could be built with reasonable extrapolations of current technology, and at a technology level much less advanced than required for pure fusion; (2) economic; compact breeder reactors can be built, at a size considerably less than required for pure fusion. Both of these accounts stand in contrast to the mainline fusion concepts, for which fusion-fission designs are generally comparable to pure fusion designs in size and technical difficulty. Therefore the laser solenoid, because of compactness and moderate technology, has potential for relatively early application to fuel breeding needs.

ACKNOWLEDGMENTS

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ELECTRON BEAM HEATED SOLENOID REACTORS FOR BREEDING FISSILE FUELS

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ABSTRACT

A study of electron beam heated linear solenoid fusion reactors showed that they could economically breed fissile fuel without producing excess electricity for sale. By using a molten salt breeding medium, ^{233}U could be bred in a 300 meter long, self-powered reactor with continuous processing and negligible fission (< 1% of the fusion) in the device. By using a depleted uranium fission plate for neutron and energy multiplication, the length could be reduced to below 100 meters. The reactors used multiple magnetic mirrors for end loss reduction and superconducting solenoids for radial confinement. The blankets required no fissile material. Plasma heating was accomplished solely by electron beams.

INTRODUCTION

Fission reactors are likely to be a major source of power in the coming decades with about 2000 reactors projected worldwide[1] by 2000 C.E., most being light water reactors (LWR's). These use fuel slighly enriched to about 3 percent in ^{235}U, but could use ^{239}Pu or ^{233}U as the fissile species. The more abundant ^{238}U is not used effectively in the LWR cycle. Fission breeders can burn ^{238}U efficiently but produce relatively little excess ^{239}Pu for use in LWR's. Estimates of uranium resources have generally indicated sufficient low cost reserves up to about 2000 C.E.,[2] but possible shortages or higher costs beyond that date and more recent studies[3] indicate a potentially severe shortage, either sooner or under slower reactor growth assumptions. Thus an alternative source of fissile material appears imperative to maintain the output from LWR and other non-breeding systems.

We have examined a new fusion reactor concept, the electron beam heated, linear solenoid contained plasma reactor, applied to the breeding of fissile fuel for external use as its main product.

CONCEPTS AND OPTIONS

ELECTRON BEAM HEATED SOLENOID (EBHS)

The EBHS[4] (Fig. 1) is a column of DT plasma heated to fusion conditions by an electron beam injected from the end along a guiding axial magnetic field. Within this general concept are many options relating to pulsed or steady state operation, radial and axial confinement methods, etc. Our baseline concept uses microsecond pulsed beams, a superconducting solenoid field for beam guidance and for radial confinement, and multiple magnetic mirrors to reduce axial loss rates. The plasma is moved away from the tube wall at the start ("Isolated Plasma Case") by a pulsed solenoid, but very little compressional energy is added to the plasma, in contrast to the theta pinch. Alternatively, this coil may be eliminated, leaving an ionized gas blanket between the hot plasma and wall ("Gas Blanket Case"). The latter is simpler but requires larger diameter plasmas for the same radial loss rate. A limiting case is wall confinement which is being considered by Soviet scientists for an e-beam heated plasma. There are several other options for end loss reduction. Free streaming leads to long reactors but solid or gas end

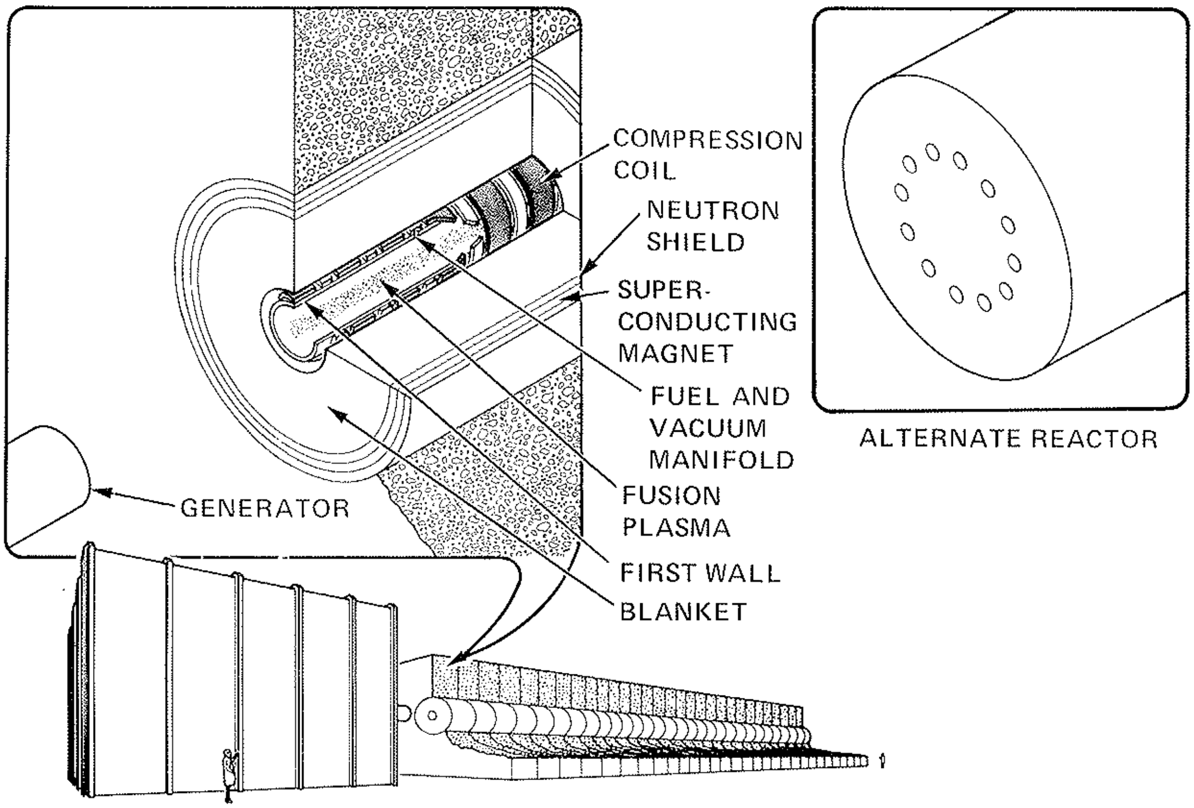

Fig. 1. Electron beam heated linear solenoid.

plugs and turbulent reduction of thermal conductivity or ionic streaming are possibilities.

The system operates at high density (10^{16} to 10^{17} cm^{-3}), modest fields (150 kG), high β (~0.9), and burns for tens of milliseconds. Plasma and wall radii are a few centimeters, and lengths of 100 to 500 meters are of interest.

In a reactor the power is limited by the wall area and it may be desirable to have a number of plasma tubes inside a single solenoid (Fig. 1a), with the beam fired sequentially into the tubes. Pulse repetition rates are of the order of 1 to 10 pps for production plants. The electron beam generators for the baseline designs are a straightforward scale-up of existing large single pulse megavolt, megajoule devices, with repetitive switching being the main technological advance required.

Long pulse duration beam diodes (plasma or thermionic) could greatly relax the electrical system requirements (to millisecond or longer duration circuitry), or even allow steady state operation. In the baseline design, plasma heating is possible under some conditions, such as by reflecting the e-beam from the tube ends ("reflex diode").

BREEDING CYCLE

The two breeding cycles are ^{238}U - ^{239}Pu and ^{232}Th - ^{233}U, with many possible media, fuel forms, neutronic spectra, etc. Although there are many options to examine we have selected breeding ^{233}U in a molten fluoride (LiF-BeF_2-ThF_4) salt medium[5] for our baseline. Extensive data on this system has been developed under ORNL's Molten Salt Breeder Reactor Program (MSBR) including processing technology and cost estimates. An attractive feature is the continuous processing to remove bred fuel and thus minimize fission, fission product inventory and decay heat in the entire breeding plant. Lidsky[6] and

Cook[7] have examined the neutronics, economics, and other aspects of the Th-molten salt breeding system applied to fusion reactors with favorable conclusions.

A blanket design for this cycle (Fig. 2) shows a moderating region to reduce the fast fission of thorium, and a shield to reduce the heating and radiation damage of the super-conducting magnet. The plasma tube is cooled by lithium that also breeds tritium. An option which yields greatly improved performance at the cost of introducing large amounts of fission is a depleted ^{238}U plate surrounding the plasma tube for the purpose of multiplying the neutron and thermal output. The large neutron flux from n,2n and fission triples the ^{233}U yield and the extra energy allows the use of much lower gain fusion sources. Although plutonium is bred into this plate, we do not consider its value and endeavor to keep the plate in place as long as possible. It is then removed with the value of plutonium offset by assumed processing costs, although this plutonium could be a valuable product.

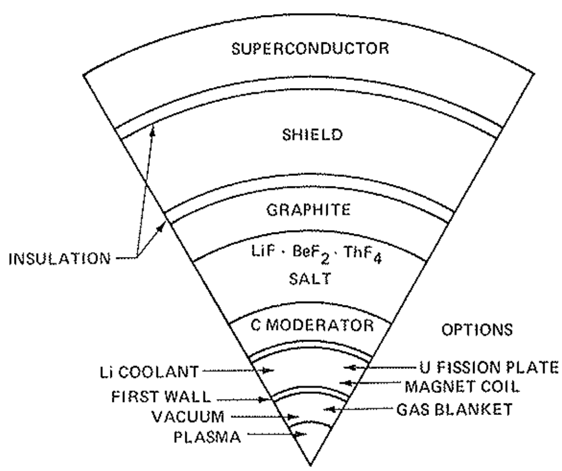

Fig. 2. Breeding blanket.

Based on the MSBR technology and studies of fusion plants,[8] a complete processing system for U, T, heat transport, steam generation and electric power production (Fig. 3) has been defined and costed.

REPRESENTATIVE DESIGNS

In this section we present some typical designs which are complete and consistent but not optimized. Several parameters, particularly the wall flux and the superconducting magnet cost, may vary by factors of 2 to 5 and lead to greatly different product costs from the values presented in this section. Nevertheless, these give a semiquantitative idea of the size, physical parameters, and cost of the various components. The parametric study will show how these can vary over the operating range.

THE EBHS SYSTEM

Typical parameter values for single plasma tubes of 300 and 75 meters length (Table 1) show plasma (and e-beam) energies of 8 to 70 MJ and fusion output ranging from 8 to over 1000 MJ. The losses, dominated by magnetic mirror joule heating (non-optimized), are several times the e-beam energy. The plasma isolation coils are back biased against the superconducting fields and thus do not add to the total stresses on the superconducting magnets. The isolation coils have modest energies (0.7 to 1.3 MJ/m) compared to theta pinch compression coils (10 MJ/m) and their switching and joule losses are small compared to the mirror coil joule losses. The superconducting magnets are about 150 kG. At 300 meters the fusion gain (fusion thermal energy/electrical input) is more than sufficient to power the breeder (80 percent circulating power).

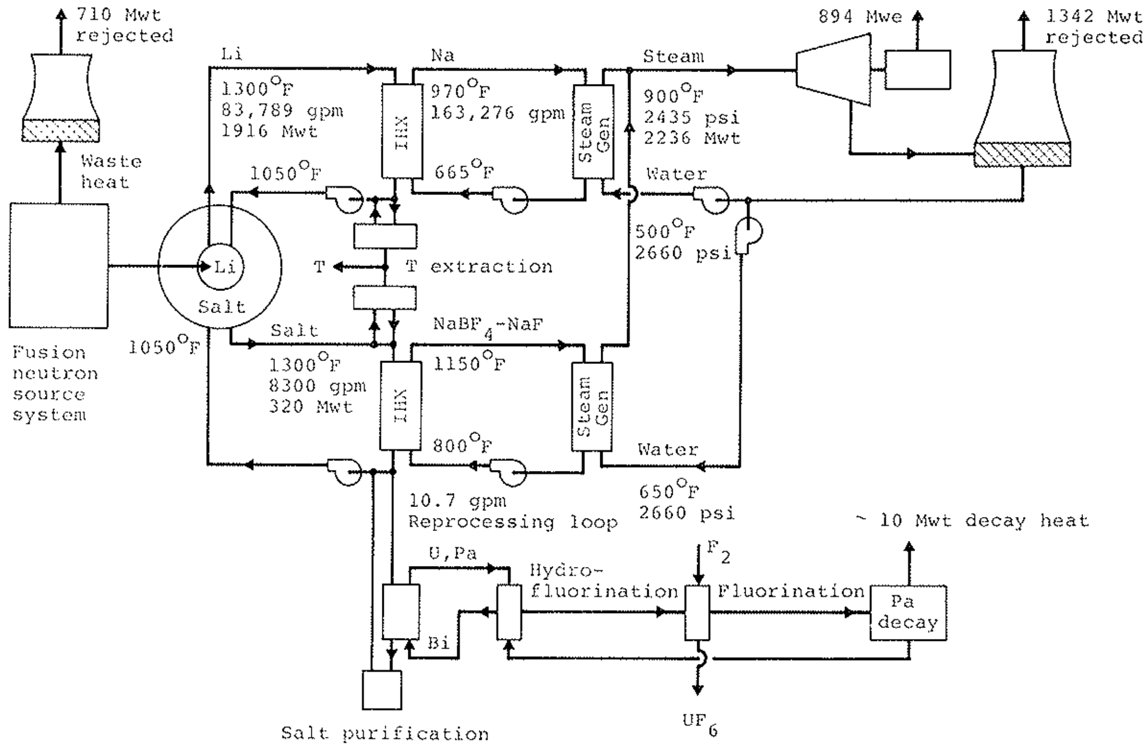

Fig. 3. Complete proceeding system for U,T, heat transport, steam generation, and electric power production.

Table 1. Physics parameters of representative designs.

	300 meters	75 meters
Tube length	300	75 m
Plasma radius	2.6	1.67 cm
Wall radius	6.17	5.28 cm
S/C magnet radius	2	2 m
Mirror ratio	2	2
Density	10^{17}	$10^{17} cm^{-3}$
Temperature	5	5 keV
β	0.83	0.90
Superconductor field	156	149 kG
Burn time	63	4 ms
Plasma energy	69	7.8 MJ
Fusion energy	1078	8.2 MJ
*Losses (magnetic fields)	244	6.4 MJ
Fusion gain	3.26	0.5 MJ

*Mainly joule heating in mirror magnets.

Based on these plasma devices we have calculated two breeder reactors, assuming 10 tubes operating at a first wall neutron flux of 1 MW/m^2 (Table 2).

The first is a pure fusion device, 300 meters long, that is self sustaining with a small excess electric power output (527 MW circulating, 137 MW output). This plant produces 2000 kg/yr of ^{233}U compared to only 150 to 200 kg/yr for a 1000 MWe fission breeder. Thus a self-powered fusion breeder can supply ten times as many LWR's as a 1000 MWe LMFBR, whose main product is electric power rather than bred fuel. Because a 1000 MWe LWR will burn 500 to 1000 kg/yr of fissile fuel (depending upon conversion and recycle rates), the LMFBR is not very effective in supplying the LWR needs until there are many more fission breeders than burners.

Table 2. EBHS breeder reactors: N_T = 10 tubes, wall flux = 1 MW/m^2

	Pure Fusion	Fusion-Fission
Length	300 m	75 m
Fusion power	1836	397 MWt
Fission power	10	1668 MWt
Atoms bred/fusion	0.4	1.1
Fissions/fusion	0	0.5
Cycle time per tube	6.9	0.2 s
Total electric power	664	820 MWe
Net electric power	137	42 MWe
Circulating power fraction	0.8	0.95
Bred fuel (80 percent load)	2000	1175 kG/yr

If one can use fission in the fusion breeder blanket in the form of a depleted uranium plate, then lower fusion gain and much shorter reactors are allowable. The 75 meter device (Table 2) is self sustaining, while still breeding 1175 kG/yr of ^{233}U. With a somewhat greater length or larger diameter, this 75 meter plant would be comparable in output to the 300 meter pure fusion plant. This fusion-fission plant requires no fissile material for startup and has no criticality concerns. Eventually plutonium and fission products build up in the plates, but Cook (Reference 8) has found that loss-of-coolant accidents can be handled in the specific design used here.

COMPONENT AND SYSTEM MODELING

EBHS PHYSICS AND DESIGN

A program has been written which takes specified input parameters for the plasma, beam, and operating conditions and calculates the thermonuclear yield per pulse per tube. In addition it calculates the kinetic energy of the e-beam, magnetic field strengths, plasma radius, tube radius, energy loss times, beam pulse length, diode and e-beam generator requirements, beam time, plasma Q, and cycle time. It allows two beam-plasma interaction models, two radial plasma modes, and four axial end loss mechanisms.

A brief flow chart of the code is given in Fig. 4 along with the subroutines which it calls directly. It first computes the kinetic energy of the injected e-beam which will give the specified energy absorption length. The beam-plasma interaction models are those of T.S.T. Young at PI[4] and L. Thode of LASL.[9]

Using the value of the kinetic energy of the injected beam and the specified input beam current density, the magnetic field (B_i) on the inside of the heated plasma column is chosen large enough so that the beam filamentation instability is stabilized. The magnetic field (B_o) on the outside of the heated column is then determined from pressure equilibrium.

Requiring that the m = 1 plasma instability be stabilized[10] and that the heated plasma column be at least one Larmor diameter, of the 3.5 MeV alpha particle, away from the wall gives the inequality

$$r_p + \frac{540}{B_o(\mathrm{kG})} \leqslant r_W \leqslant \frac{r_p}{(1-\beta_p)^{\frac{1}{2}}} \qquad (1)$$

where β_p is the plasma beta, and r_p and r_W are the radii in centimeters of the plasma column and conducting wall, respectively. This inequality determines the minimum plasma radius

$$r_p^{min} = \frac{\frac{540}{B_o(kG)}}{\left[\frac{1}{(1-\beta_p)^{1/2}} - 1\right]} \quad (2)$$

The axial energy loss time is calculated for one of four different end loss mechanisms: (1) free streaming without self mirroring, (2) free streaming with self mirroring,[11] (3) end plugs (i.e., the energy loss is governed by thermal diffusion,[12]) and (4) multiple mirrors (i.e., the energy loss is governed by diffusion of ions through the mirrors.[13]) The most promising results have been obtained for the multiple mirror end loss mechanism which has an axial energy confinement time of

$$\tau_{EL}(s) = 9.8x10^{-27} m^2 n_p \ell^2 T_i^{-5/2} \quad (3)$$

where m is the mirror ratio, n_p is the plasma number density per cubic centimeter, ℓ is the length of the reactor in centimeters, and T_i is the ion temperature in keV.

The input value of τ_{RL}/τ_{EL} (ratio of radial loss time to axial loss time) and plasma parameters are used to compute the plasma radius which will give the specified ratio τ_{RL}/τ_{EL}. If the plasma radius (r_p) which is determined in this manner is less than r_p^{min} (Eq. 2) the plasma radius is set equal to r_p^{min} and a new radial energy loss time is calculated. The subroutine treats either the isolated plasma mode or the gas blanket mode. The appropriate diffusivities for each mode are taken from Spitzer.[14]

The beam pulse length is determined by requiring that the pulse length be long enough to provide the energy required to heat the plasma column. For the gas blanket mode where the expanding heated plasma does work in compressing the exterior magnetic field and cool plasma, the code increases the beam pulse length to provide the additional beam energy required for this compression.[4]

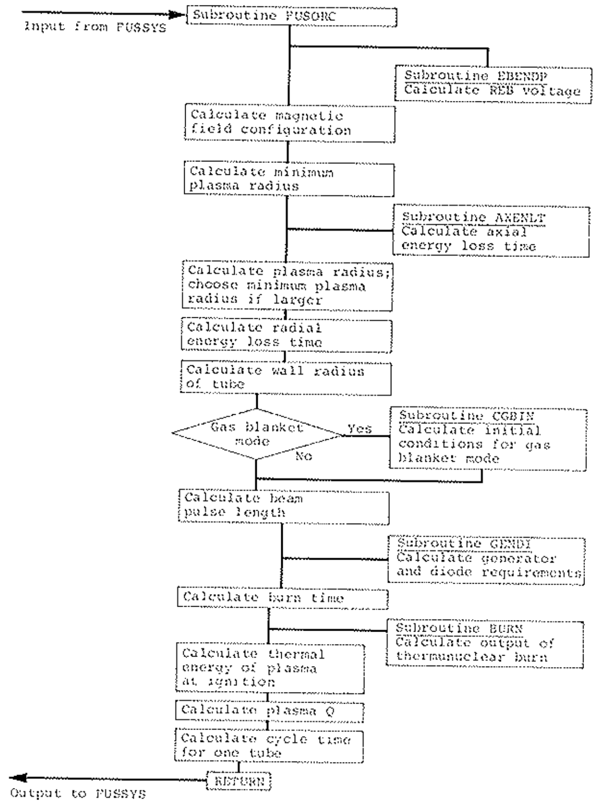

Fig. 4. Calculational schematic flow chart of the fusion source subroutine FUSORC.

The thermonuclear burn time is taken to be the minimum of the radial and axial confinement times. The total energy of the neutrons, alphas, and the energy due to breeding reactions in the tritium in the blanket per tube per pulse are summed to give the total thermonuclear yield. At present no alpha energy deposition is assumed and the burn calculation is space and time independent.

The plasma gain Q_p is then the ratio of the total thermonuclear yield to the energy of the plasma at ignition, and the fusion gain Q_F is the total thermonuclear yield divided by all the input energy and losses.

The cycle time for each tube is determined by limiting the 14 MeV neutron energy flux through the tube wall to the value specified by the user.

SYSTEM ANALYSIS

The EBHS breeder was modeled in terms of energy flow among its components (Fig. 5). Thirty-four independent parameters characterized the system, including dimensions, efficiencies, confinement parameters, neutronic characteristics, and plasma and beam variables. The blanket performance is given in terms of fissile atoms bred per fusion and fissions per fusion. These were obtained from the nuclear properties and compositions either by simple approximations in terms of nuclear cross sections, or from detailed multigroup neutron transport codes. The fraction of heat available for electrical generation is separately specifiable for the fusion energy, the fission energy, and the energy deposited in the plasma by the e-beam. No magnet joule heat, switching losses, or e-beam generator losses are assumed recoverable for electrical generation. The useful heat is converted to electricity and subtracted from the fusion source requirement to obtain net power.

The system costs have been broken into eight elements:

- Superconducting magnet
- E-beam generator
- Blanket
- Heat transport system
- Nuclear fuel handling
- Electric generators, structure, site
- Special materials (salt, lithium)
- Pulsed magnet power supplies

The magnet costs are a major factor, depending upon the source of the estimates. We have used values generated by L. Lidsky from Lubell[15] and estimates from Powell[16] and Marston.[17] Many of the remaining components were based on MSBR and LMFBR cost estimates adjusted to 1976 dollars and with scaling for size, power level, etc. The program computes total capital cost and debits or credits the equivalent power cost at 820$/kWe (~20 mills/kWh). The bred fuel cost is computed in $/gm at a given load factor (~80 percent) and carrying charge (16 percent). Contingency, interest during construction, A/E fees and operating costs are neglected or assumed included in the carrying charge.

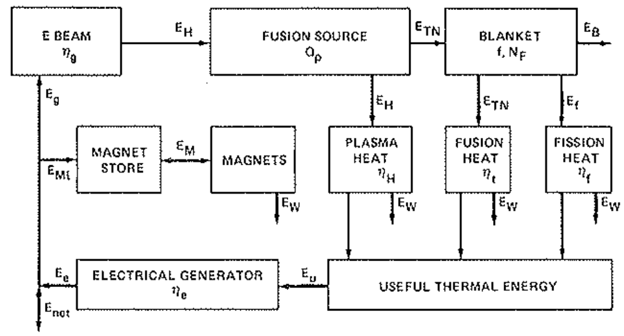

Fig. 5. Energy flow schematic.

PARAMETER STUDY RESULTS

There are about ten important but interrelated parameters which determine the bred fuel cost. The most important are reactor length, the use of fission in the blanket and the allowable wall flux.

Results of a limited parameter survey (Fig. 6) show that pure fusion reactors of 250 to 300 meters can be competitive breeders even at current fuel prices, and that these can be self-sustaining, i.e., independent of electric power costs or the need to be on the power grid. If excess power is sold at 20 mills/kWh, then longer reactors can make fuel at zero or negative cost, while if the excess power is not sold, the bred fuel cost drops slowly with reactor length. The self-sustained reactor length for a fissioning blanket reactor is 75 meters because lower fusion gain is offset by the fission blanket energy multiplication. The capital cost equivalent of the net power is shown for a typical plant (Fig. 7) as a function of length.

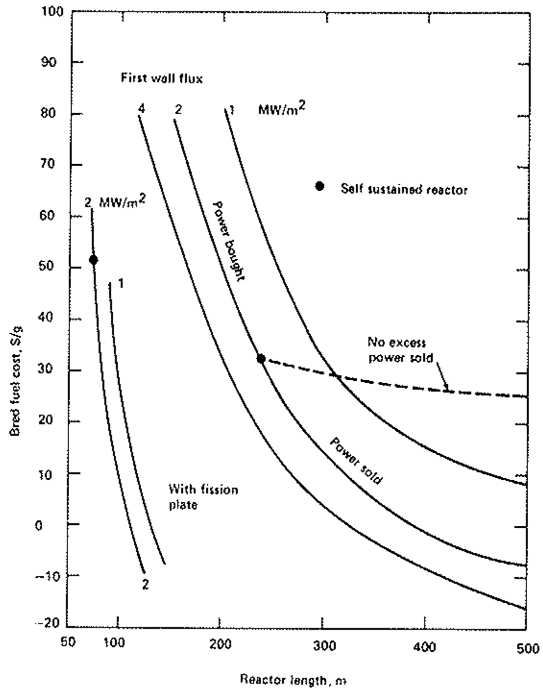

Fig. 6. Bred ^{233}U costs for electron beam heated linear solenoid reactors.

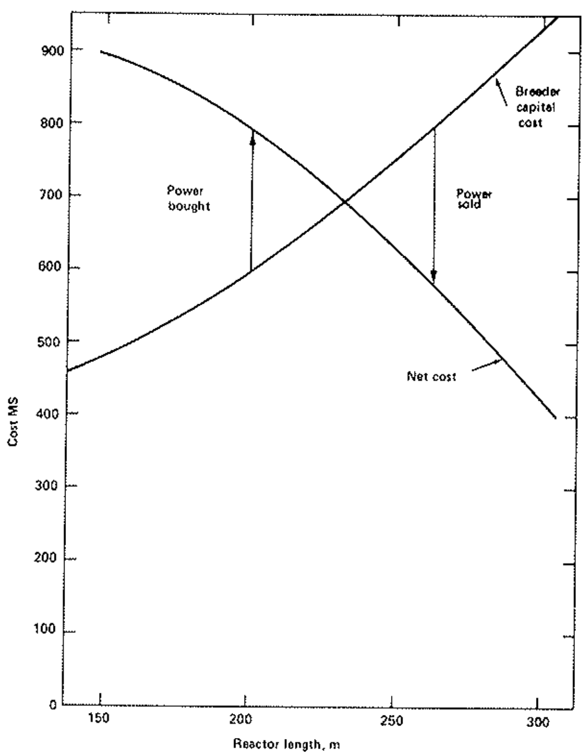

Fig. 7. Electron beam heated reactor capital costs and the effect of purchase or sale of the net electric power produced.

Many other variables are of interest, but care must be taken in varying single parameters which cause others to change. Typically the plasma radius and e-beam pulse energy will vary, and the bred fuel costs are sensitive to these, usually decreasing with increasing radius or beam energy. It would be desirable to reduce the magnetic field by lowering the density, n_p, but because of the necessity for a field to guide the beam, the containment field is only weakly dependent upon n_p (Fig. 8). As n_p is reduced below 8×10^{16}, the radius and e-beam energy increase rapidly while the plasma β declines. This is dependent upon the beam stability limit and the interaction model assumed, and the use of lower beam voltages would allow significantly lower magnetic fields.

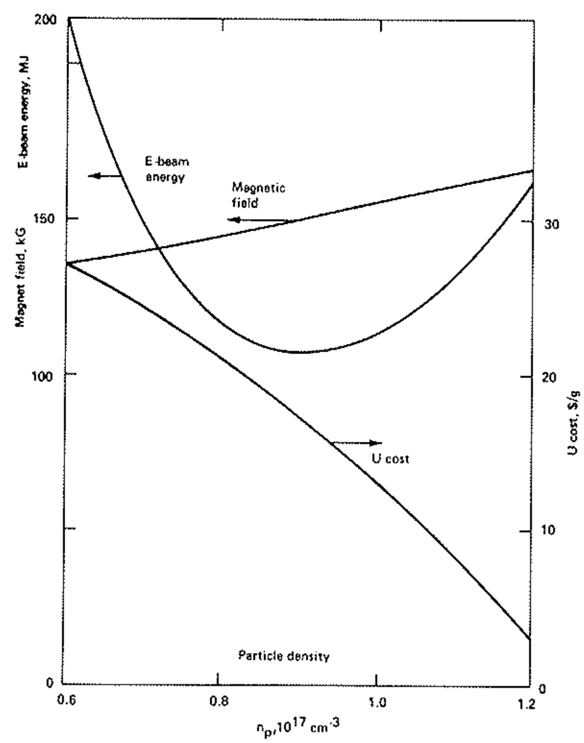

Fig. 8. Effects of varying the plasma density upon some system variables. For lower value of beam voltage, the magnetic field would decrease more sharply with n_p.

CONCEPT EVALUATION

There are three main aspects to the evaluation of a concept in its early stages: the accuracy of the physical principles, the technological development required, and the eventual economics. This is complicated by the existence of many options, some of which are clearly superior but more speculative in nature. In our evaluation here we shall emphasize the nearer term, better understood options, which shall be denoted as "conventional" technology. We emphasize the anticipated problem and development areas rather than the advantages of the concept.

E-BEAM HEATED SOLENOID

Physics. The crucial question for achieving short linear reactors is end loss reduction. Subsidiary to this is confirmation of the plasma heating and propagation of electron beams at reactor conditions of density, temperature, length, field, etc. Radial transport phenomena are important in the gas blanket mode. There are also a number of advanced options which rely on physical processes in the e-beam diode that require further study.

Thus far we have considered free streaming, end plugs, and multiple magnetic mirrors for the end loss modes. Of these only the last leads to useful reactors of 500 meters or less. The nominal case used mirror ratios of 2.0, but values as low as 1.25 give acceptable results. The key problem is stabilization of the field and plasma under reactor conditions.

Experiments by Lieberman and Lichtenberg at UCB[18] and by Budker and co-workers at Novosibirsk[19] have demonstrated basic multiple mirror confinement in agreement with theory. The principal problems are stabilization and enhanced diffusion in regions of distorted flux surfaces. Linked quadrupoles are being used to achieve average minimum-B stabilization of flute modes in experiments at UCB which will extend this technique to moderate β.[20] Here the question is whether higher will allow local modes to grow in a system which has average stable properties.

Other methods of stabilizing magnetic mirrors are line-tying, wall currents, and feedback. Line tying appears to work in short systems, but the question remain as to the availability of sufficient electrons in the end region and whether these electrons will increase the axial heat loss. This method is difficult to test in a small device.

Suppression of flute modes by wall stabilization, which requires a net current in the plasma column that returns through a conducting wall, is suited to the e-beam since some residual net current will remain after passage of the beam down the channel. Only a few kiloamperes current is required for stability.[21] No experiments on this stabilization method are known to us, but calculations show that modest size devices ($\leqslant$10 meters) could be used to test the theory. Dynamic feedback stabilization is inherently simpler for linear geometry as compared to toroidal because fewer modes need be stabilized.[22] Power flow seems favorable: systems analysis of toroidal reactors have shown that power requirements of the feedback system can be reduced to low levels. Experiments at UCB have investigated feedback on small-scale mirrors, but much more should be done.[23]

There are several other possible ways to reduce end loss including turbulent reduction of electron thermal conductivity, turbulent reduction of ion streaming, field reversal, etc.

Turbulence can extend confinement time by reducing mean free paths. The timescale for electron thermal conduction to reduce the temperature at the midplane of a linear device is proportional to the electron collision frequency. Turbulence increases this frequency; in a beam heating experiment at PI we have observed enhancements of $\sim 10^5$, resulting in energy containment during the turbulent period.[24] Both ion and electron collisions are enhanced, so both are contained. For reactor operation, turbulence must be maintained after the beam pulse by injection of a weak beam into the low-density end region (where turbulence is easier to excite) or by RF excitation of turbulence. The influence of turbulence is yet to be assessed.

Field reversal which results in an average minimum-B well, can be considered for energy loss reduction. The closed field lines are produced by injection of a rotating beam, which can be separate from the heating beam. Major questions remain concerning the stability of long, thin reversed field layers. The processes which determine the lifetime of the layer are not understood and are under study in the LINUS program at NRL.

E-beam induced plasma heating by anomalous conductivity has been extensively demonstrated at low plasma densities ($\leqslant 10^{14}$ cm^{-3}) and high electron temperatures. Extension of these results to the high densities (10^{16} to 10^{17} cm^{-3}) by all present theories leads to adequate heating and efficient beam utilization. The main requirement is confirmation at reactor densities, with intense beams and over plasma lengths which minimize end effects on the beam plasma interaction. This appears quite straightforward with existing e-beam devices.

The propagation of e-beams in the plasma over long distances must also be confirmed, although experiments to date indicate this is not a problem. Radial diffusion of the beam by scattering from the individual plasma particles is negligible but there may be interactions between the beam and waves in the plasma. The beam spreading could be observed with sufficient precision over a 10 meter path for extrapolation to reactor lengths.

Cross-field diffusion of the plasma is the chief radial loss mechanism for the isolated plasma mode, and this is significantly less than the axial loss rate for most reactor designs. However for the gas blanket option the loss rate for the same plasma radius by thermal conduction is much higher. Thus larger radii are needed and the effects of cooling the outer regions of the plasma become more important. In this regard the heating of the cooler plasma regions by alpha particles (neglected thus far) should be included, based on preliminary computations. Thus more detailed physics calculations are needed to determine the minimum plasma radii (and thus the minimum e-beam pulse energy) for the gas blanket mode.

Technology Development. The e-beam solenoid reactor shares many technological problems with other fusion concepts but differs quantitatively in some areas and has some unique features. The major unique development required is the repetitively pulsed intense electron beam. Performance will also depend upon such factors as first wall flux, plasma handling, superconducting magnet technology, wall replacement, and the development of the fissile breeding cycle process.

The e-beams required range from 8 to 200 MJ/pulse, with average powers of the order of 100 MW. The largest existing device, AURORA[25] has an output of several megajoules per pulse from a 5 MJ, 8 MV Marx circuit capacitor bank. An analysis of the losses in this bank show that energy transfer efficiencies of over 90 percent can be achieved.[26] Much more important is repetitive operation, and a study of this for the AURORA bank shows that rates up to 200 pps can be achieved with small loss of efficiency and tolerable heating in the capacitors and switches. Inductive rather than resistive components would be required for rapid charging and discharging with low loss (~1 percent). The existing switches have an inherent recovery time of the order of 1 msec. The crucial problem is the lifetime of the components, particularly switches. Gas spark gaps might be developed to survive 10^6 to 10^8 pulses at high energy and power. The actual lifetime and reliability can only be determined by a serious experimental effort but there are no fundamental reasons for excluding gas gaps for this application. A recent development is the use of laser triggered silicon rectifiers as switches. These might be designed to handle high voltage and have long lifetime. This is a promising but quite new concept.

E-beam diodes have operated at the peak power levels required, but need extension of their pulse duration from the present 1.5 μsec to 10 μsec or longer.

The magnetic fields required from the superconductors are in the range of 120 to 160 kG. The mirror fields are localized and can be smaller than the superconducting field in magnitude.

The Th-^{233}U cycle would require additional development, but the processing is simpler than the Molten Salt Reactor case as there is very little uranium and fission products present in the salt.

Economics. The cost of any undeveloped concept is somewhat speculative and quite uncertain and we have approached this problem in several ways. First, we have estimated the component costs and have computed the unit product cost. This total can be compared to the existing equivalent product cost (e.g., ^{235}U in reactor grade feed). However, much higher future costs may still be economically attractive. Another way to estimate breeder costs is in terms of \$/kWt of the total thermal energy handled by the plant components, because much of the cost is in heat transport equipment. Both approaches indicate a wide range of unit product costs with a large portion being economically attractive assuming the reactor physics is correct. The conclusions drawn will depend heavily upon the assumptions concerning the market price of fuel as well as the breeder cost and performance. For example, the magnet cost uncertainties can represent 15 \$/g in the product price, but this may be either a major cost or an acceptable differential.

CONCLUSIONS

This study has shown that the e-beam heated solenoid reactor can breed large amounts of fissile fuel at competitive costs in reactors of 75 to 300 meters length. The crucial physics assumption is the stabilization of multiple magnetic mirrors or the development of another end loss reduction technique. The main engineering development required is repetitive pulsing of the intense e-beams that are already a mature technology on single shot devices. Most other physics and engineering questions such as the heating interaction, superconducting magnets, etc., appear to be soluble or are under development in the existing fusion program.

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UNIVERSITY
FUSION-FISSION STUDIES

FUSION FISSION STUDIES AT THE UNIVERSITY OF TEXAS AT AUSTIN

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ABSTRACT

Fusion-fission studies at The University of Texas at Austin are divided into two distinct phases. These are (1) neutronic and photonic analyses of fast fission blankets and (2) economic feasibility investigations of fusion based energy storage systems. The fast fission blanket studies included both thorium and natural uranium fuels. Detailed results for liquid lithium cooled blankets are presented. The economic feasibility studies have only recently begun. The system currently being considered runs only in off-peak periods and accomplishes energy storage by producing hydrogen from a thermochemical water-splitting process. Utility system operation is simulated using the ORSIM program to determine the optimum capacity factor for the hybrid system. Income is derived by sale of both hydrogen and fissile fuel.

INTRODUCTION

Fusion-fission studies at The University of Texas at Austin began in 1972. The initial investigations of hybrid systems were spurred by their potential as breeders of fissile fuel. This potential is truly significant if one gives credibility to a scenario which continues thermal reactor deployment and growth in electricity demand while delaying the introduction of the LMFBR[1]. It was also recognized that such hybrid systems might generate net power with considerably lessened plasma requirements. Later studies have concentrated on the early introduction of fusion in utility systems. Specifically, small to medium size near break-even fusion systems which can breed fissile fuel and produce hydrogen while being driven with large base load thermal fission plants are currently under investigation. Such systems offer the potential to reduce power-peaking costs, simplify fission reactor refuelling decisions, and improve overall utility system reliability while introducing fusion technology at an early date.

The accumulation of data and formulation of methods to perform detailed analyses of fissioning blankets was completed in 1973. The methods and data were employed to compute neutron transport, photon production and transport, fission energy production, fissile fuel breeding, and energy deposition profiles for both thorium and natural uranium blankets. The techniques and data for analyzing the early introduction of fusion reactors in utility systems have been under formulation since 1975. Evaluations of this type are just beginning but preliminary results should be available in the next six months.

EARLY FAST-FISSION HYBRID STUDIES

The main objective of The University of Texas at Austin work was to quantify the ability of the 14.1 MeV fusion neutrons for breeding fuel and initiating fissions directly in U^{238} or Th^{232} while maintaining adequate tritium breeding and energy deposition profiles in the fusion reactor blanket. This direct fissioning approach was taken because it presented a capability for the use of these materials which is unique to the D-T fusion blanket system. The approach also has the advantages of allowing the blankets to remain far subcritical and makes the most use of neutron multiplying reactions to increase fissile fuel production. Direct fission in Th^{232} and U^{238} is also advantageous, since these nuclides are plentiful and any dependence on breeding of fissile materials for fuel is eliminated. The fission cross sections of Th^{232} and U^{238} are .36 barns and 1.2 barns, respectively, at 14 MeV. In addition to the fission reactions, Th^{232} and U^{238} have substantial (n,2n) and (n,3n) cross sections which provide neutron multiplication. The basic fusion reactor blanket to which the thorium or natural uranium fuel was added employed liquid lithium metal coolant, niobium vacuum wall, and graphite moderator-reflector regions with the same overall dimensions as the standard blanket model[2]. Since this approach sought to use as fuel nuclides which do not readily fission, the large energy releases in the blanket per fusion neutron of some other studies[3,4] could not be achieved.

The fission fuel was assumed to be in the form of either ThO_2 or UO_2 rods clad in niobium. Natural liquid lithium was chosen to also be the coolant for the fissioning regions of the blankets. The volume fractions of fuel, clad, and coolant were specified to be .45, .15, and .40, respectively. These values are typical of the volume fractions employed for liquid metal cooled fast reactors which are designed to operate at average power densities greater than 500 w/cm^3. The density of the ThO_2 and UO_2 fuel was assumed to be 9.35 and 10.5 gm/cm^3, respectively.

A description of the hybrid blankets studied is presented in Figure 1. The thickness of the region containing the fission materials was increased until the tritium breeding ratio dropped below 1.1.

CALCULATIONAL METHODS

The transport calculations were performed using the ANISN computer program[5]. Cylindrical geometry, 100 neutron groups, 21 gamma ray groups, an S_4 angular segmentation, 84 spatial mesh intervals, and a P_3 expansion of the cross sections were employed. The neutron cross sections were taken from the DLC-2D[6] library. Activity cross sections were generated using SUPERTOG[7] and the ENDF/B version III library. Secondary photon production cross sections for Li^6, Li^7, and C were taken from the data of Ritts, et. al.[8] POPOP4[9] and POPLIB were used with neutron cross sections from SUPERTOG to generate the secondary gamma ray production cross sections for Nb^{93}, O^{16}, Th^{232}, U^{235} and U^{238}. The neutron kerma factors for these materials were calculated with a code written for this purpose (KRMCAL). The neutron kerma factors for thorium and uranium are dominated by the fission reaction which was assumed to impart 172 MeV per fission to charged particles. Gamma ray kerma factors and cross sections were calculated using MUG[11] with the photoelectric effect and pair production cross sections from Howerton's Photon Interaction Cross Section Library. The (n,3n) reaction was included in the DLC-2D cross section matrices through a computer program (SECN3N) written for this purpose. A block diagram of the computational scheme is shown in Figure 2. In the figure, the codes are indicated along with their source.

THORIUM BLANKET RESULTS

Blanket performances for thorium regions 6 and 13 cm thick were calculated. Tritium breeding summaries for the 6 and 13 cm cases are presented in Table 1. The total tritium breeding ratios for these cases were 1.30 and 1.10, respectively. Table 2 contains the (n,fission) and (n,γ) fissile fuel breeding reactions by region for the two thorium blankets. The .311 Th^{232} (n,γ) reactions per fusion neutron for the 13 cm thick thorium region blanket was near the .325 value of Lidsky's thorium blanket[12]. The total tritium breeding ratios for these blankets were also nearly the same (1.10 for the 13 cm thick blanket and 1.13 for Lidsky's blanket). In addition, the 13 cm thick blanket studied here produced 9.25 MeV per fusion reaction due to fissions in Th^{232}. The fission reactions took place primarily at 14 MeV. For the 13 cm blanket, the first neutron group (14.9-13.5 MeV) accounted for 73, 63, and 55 per cent of the fission reactions which took place in regions 4, 6, and 7, respectively. Resonance self-shielding was neglected in these calculations. Another later study[13] showed that accounting for resonance self-shielding results in increases in the tritium breeding ratio of $\sim$.04 and $\sim$.12 for shielding niobium and thorium, respectively. These increases in tritium breeding ratio for a given thorium region thickness are of course accompanied by similar decreases in niobium absorption and U^{233} production. Thus, the optimum actual Th thickness will be larger than the 13 cm arrived at in the calculations. Increases in the region thickness do not result in large increases in fission power so that the main effect is to increase the required reactor inventory of fission material and the results of the un-self-shielded calculations remain basically valid.

The heating rates by region for the 6 cm thick and 13 cm thick thorium blankets are presented in Table 3. The local heating rates in these tables include the kinetic energy imparted to the fission products. If the first wall loading of 14 MeV neutrons is assumed to be 10 MW/m^2, the peak power density in both blankets is about 200 w/cm^3. This value is reasonable for the lattice, but an analysis to assess whether adequate cooling of the niobium walls could be maintained was not performed.

The heat deposition due to the local and gamma ray interactions in the niobium walls was reduced for both the thorium cases relative to the standard blanket. The gamma ray contribution to the heating of the first wall was reduced from 4.6 per cent of the fusion neutron energy for the standard blanket to 4.0 per cent of the fusion neutron energy for the 13 cm thick thorium blanket. A similar reduction is observed in the second niobium wall. The energy leakage from the thorium blankets was also decreased relative to the standard blanket case. The total energy leakage from the 6 cm and 13 cm cases were .0096 and .0075 of the incident fusion neutron energy. About 36 per cent of this energy was carried away by the neutrons. The energy leakage from the standard blanket was .012 of fusion neutron energy.

URANIUM BLANKET RESULTS

A 13 cm thick natural uranium blanket was also considered. The total tritium breeding ratio for this blanket was 1.27. The fission energy generation in this blanket was 40.3 MeV per fusion neutron. If a first wall loading of 10 MW/m^2 of fusion neutrons is assumed, the maximum power density in the uranium region is over 500 w/cm^3. This occurs near the first vacuum wall. Since this wall will be very difficult to cool, the uranium was removed from the space between the niobium walls (region 4) for the remaining natural uranium blankets studied. Blankets were studied with uranium absent in this region but with thicknesses of 10, 13, 20, and 26 cm of natural uranium[14].

The tritium breeding results for the four uranium blankets are presented in Table 4. The number of fissions and fissile fuel producing capture reactions per fusion neutron for the four uranium blankets are shown in Table 5. The 20 cm thick blanket achieved a tritium breeding ratio of 1.09 and .53 fissile fuel nuclei per fusion neutron. Since resonance self-shielding was neglected, the 26 cm case is probably more accurate for estimating the fission energy production. This blanket generated an additional 45 MeV per fusion neutron from fission (86 per cent due to U^{238} fissions).

Figure 3 contains a plot of the local and gamma ray energy deposition for the 26 cm thick uranium blanket assuming a fusion neutron wall loading of 10 MW/m^2. The gamma ray energy deposition in the first niobium wall for this case increased from .046 for the standard blanket to .052 of the incident fusion neutron energy. Another significant result was that the $Nb(n,\gamma)$ reaction rate in the niobium walls for the 20 cm uranium blanket was over twice the value for the standard blanket. The total energy leakage from the 20 cm uranium blanket was calculated to be .0055 of the incident fusion neutron energy. This is less than one-half the value for the standard blanket.

CONCLUSIONS ABOUT THE FAST FISSION BLANKETS

The natural uranium blankets studied were found to be capable of producing as much as 45 MeV of fission energy and .53 fissile nuclei per fusion reaction. The first wall heating load for such a blanket was not found to be greatly different from the heating load calculated for the standard blanket. A reasonable maximum power density of 203 w/cm^3 was calculated for such a blanket with a first wall fusion neutron energy flux of 10 MW/m^2. For thorium oxide instead of uranium oxide in a similar blanket configuration, a maximum power of 9-12 MeV and .31 fissile nuclei per fusion neutron can be achieved while adequate tritium breeding is maintained. The natural uranium blanket therefore is capable of producing about four times the fission energy and 1.7 times the fissile material production of the thorium blanket. About 85 per cent of the fission power from the natural uranium blanket is due to fissions in U^{238}.

COMMERCIALIZATION OF FUSION REACTORS

After the methods and data to perform neutronic evaluations of fission-fusion blankets was assembled, our attention turned to concepts which might ease the introduction of fusion to utility systems. Hybrids seem to have some particular advantage in this respect. The introduction of fusion reactors will depend largely on their reliability, economy, safety, and environmental impact. First-generation, base-load, pure fusion reactors, subject to all the uncertainties of any new energy production technology, may well be less attractive to utilities than the fission or coal generation options. Intermediate fusion-fission systems offer the potential to relax some of the engineering requirements of pure fusion systems. There are even some advantages to a system that breeds fissile fuel without generating electricity, and it is this type of system which we have decided to evaluate.

It is the purpose of the present study to investigate the economic feasibility of an elementary fusion device which would produce fissile material and possibly fission energy as well. Since utility system reliability is considered to be of overriding concern when evaluating the hybrid, it will be assumed that the proposed device may only be run in off-peak periods. A preliminary analysis has shown that water splitting rather than electric generation is possibly more economical for such a hybrid.

HYBRID ENERGY STORAGE SYSTEMS

The advantages of a fissile fuel and hydrogen producing system which operates only in off-peak periods are numerous. One of the most important is that it may allow early introduction of fusion systems which are smaller and which achieve less stringent plasma conditions than base-load pure fusion reactors. Such a concept also minimizes the effects of forced outages of the fusion device on the utility system operation. Another advantage is that the fusion hybrid system can be driven with cheap excess fission reactor capacity to produce more fissile fuel than is consumed and copious quantities of hydrogen for displacing the expensive fuel of peaking units and for sale. Disadvantages of such a system are that it requires the introduction of both fusion and water splitting technology and such an energy storage system is probably more expensive and less reliable than a pumped storage unit.

A schematic diagram of the system we are considering is shown in Figure 4. The important components of the device are (1) the plasma heating device, (2) the fusion plasma, (3) the fusion reactor blanket, (4) a direct conversion device, and (5) the water-splitting equipment. Other components not directly part of the fusion reactor system are the storage tanks and hydrogen-oxygen turbine. The economic feasibility of such a system is under investigation for various performance assumptions for each of the components. In addition, a hypothetical utility system is required for the analysis.

A modified version of the ORSIM[15] program is being employed to determine the optimum capacity factor for the hybrid system for each month of a multiyear operating period. The ORSIM program computes the energy generation from each station of the utility system such that the total fuel costs over the simulation period are minimized. Forced outages are accounted for using a probabilistic model. For the hybrid system credit is taken for sale of fissile fuel and excess hydrogen production. Although comprehensive parametric studies have not been completed, and example which illustrates the results follows.

The hybrid system parameters chosen are presented in Table 6. Several cases were run and all of the values used are listed. The reactor performance is based on the TCT design of Jassby[16]. The thermochemical water splitting process was the one proposed by De Beni[17] which was designated as the ISPRA Mark I process. This water splitting process was chosen since the maximum temperature it requires is 750°C., which should be attainable in the fusion reactor blanket. The blanket fissile fuel and fission power production were based on the calculational results presented in the first part of this paper. Cases were run to examine the effects of changing the hybrid reactor system size and performance.

PRELIMINARY ECONOMIC RESULTS

The utility system chosen for the simulations is described in Table 7. It was derived from the APPLE system which was included with the ORSIM computer code. A two-year simulation period was utilized. The fissile material and hydrogen values employed were \$25/g and \$1/MBTU, respectively.

Figure 5 summarizes the results of fusion reactor operation simulation for three different hybrid systems. These are: (1) thorium blanket; $Q_p = 1.2$, $C = .4$, $M_B = 1.7$; (2) thorium blanket; $Q_p = 1.8$, $C = .4$, $M_B = 1.7$; (3) uranium blanket; $Q_p = 1.2$, $C = .6$, $M_b = 5.4$. Each of the three systems was studied for power consumptions of 200, 400, 600, 800, and 1000 MW_e. To illustrate the relative economics of these systems, the yearly average value of the produced fissile material and hydrogen is plotted in Figure 5. The broken line in the figure indicates the required revenue for a 4.5% annual return if the hybrid systems costs are \$1000/kw consumed. For the 400 MW system, the total income receives about the same amounts from hydrogen and fissile material. The capacity factors for the hybrid systems vary from about .25 - .40.

The results just presented are preliminary and final conclusions regarding the economic feasibility of such a system will require more detailed studies. Even a marginally economic system could be viable, since it would represent an important intermediate step toward a pure fusion and hydrogen economy.

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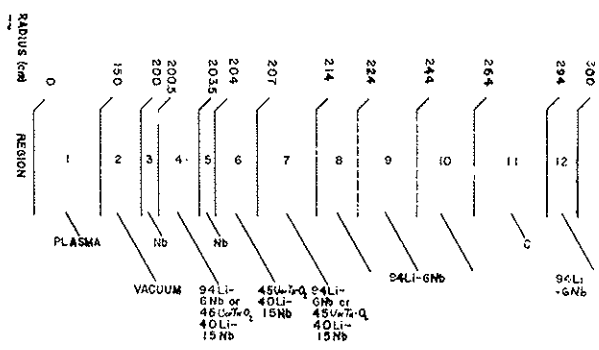


FIGURE 1. BLANKET REGION DESIGNATIONS

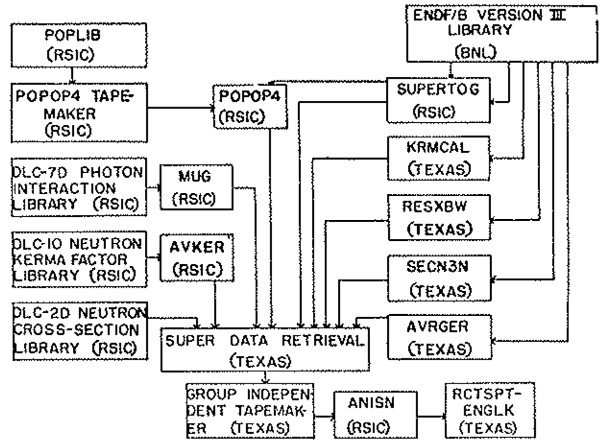


FIGURE 2 CALCULATIONAL FLOW CHART: NEUTRONIC AND PHOTONIC ANALYSES OF FUSION-FISSION BLANKETS

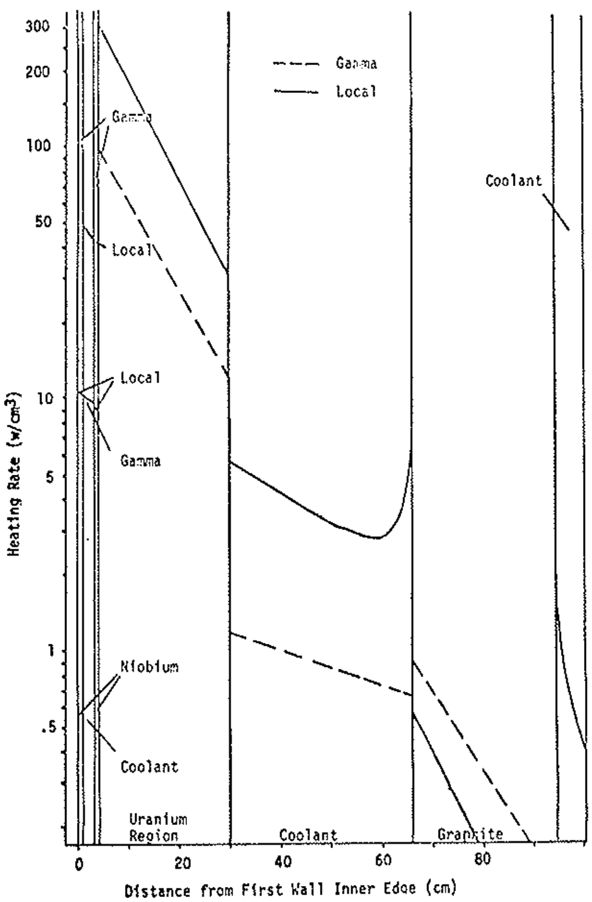
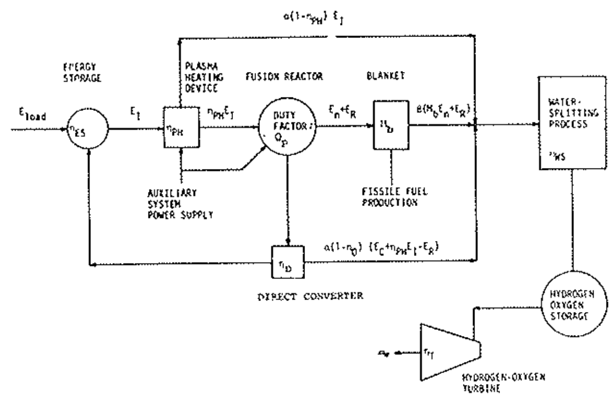


Figure 3. 26 cm Thick Uranium Blanket Heating Rates



SCHEMATIC DIAGRAM OF THE FUSION ENERGY STORAGE SYSTEM PRESENTLY CONSIDERED

FIGURE 4

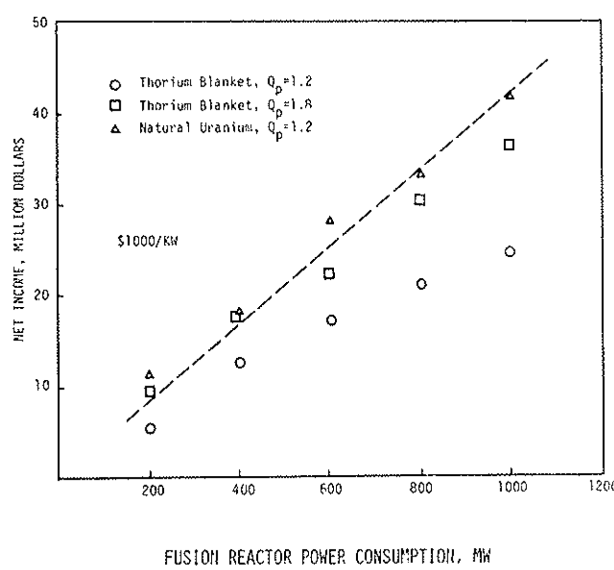


FIGURE 5

TABLE 1

TRITIUM BREEDING SUMMARY

BLKT RGN	$Li^6(n,\alpha)T$			$Li^7(n,\alpha)T$			Total		
	STD BLKT	6 cm BLKT	13 cm BLKT	STD BLKT	6 cm BLKT	13 cm BLKT	STD BLKT	6 cm BLKT	13 cm BLKT
3	..	..	..	..	..	..	..	..	..
4	.0480	.0299	.0355	.0806	.0280	.0275	.1291	.0579	.0630
5	..	..	..	..	..	..	..	..	..
6	..	.0270	.0329	..	.0179	.0170	..	.0449	.0499
7	.2912	.1296	.0633	.2012	.0742	.0243	.6707	.2038	.0876
8	..	.1616	.1588	..	.0744	.0505	..	.2360	.2093
9	.2364	.2627	.2502	.1098	.0799	.0576	.3434	.3426	.3078
10	.2944	.3165	.2942	.0458	.0339	.0251	.3387	.3504	.3193
11	..	..	..	..	..	..	..	..	..
12	.0634	.0650	.0590	.0009	.0006	.0005	.0634	.0656	.0595
TOTAL	.9334	.9923	.8939	.5163	.3089	.2025	1.4453	1.3012	1.0964

TABLE 2

FISSION AND FISSILE BREEDING REACTION RATES FOR THE THORIUM BLANKETS

Th^{232} Blanket	Blanket, Region	$Th^{232}(n,f)$	$Th^{232}(n,\gamma)$
	Reactions Per Fusion Neutron		
6 cm	4	.0189	.0681
6 cm	6	.0121	.0645
6 cm	4 + 6	.0310	.1326
13 cm	4	.0187	.0803
13 cm	6	.0118	.0763
13 cm	7	.0167	.1552
13 cm	4 + 6 + 7	.0472	.3118

TABLE 3

HEATING RATES BY REGION

Fractional Heating Per 14.1 MeV Fusion Neutron

BLKT RGN	Local Heating			Gamma Ray Heating			Fission Heating			Total Heating		
	STD BLKT	6 cm BLKT	13 cm BLKT	STD BLKT	6 cm BLKT	13 cm BLKT	STD BLKT	6 cm BLKT	13 cm BLKT	STD BLKT	6 cm BLKT	13 cm BLKT
3	.0047	.0048	.0049	.0455	.0388	.0403	..	..	..	.0502	.0436	.0452
4	.1177	.3136	.3143	.0270	.1847	.1926	..	.2624	.2587	.1447	.4983	.5069
5	.0035	.0030	.0031	.0422	.0282	.0295	..	..	..	.0457	.0312	.0326
6	..	.2042	.2041	..	.1373	.1359	..	.1676	.1637	..	.3115	.3400
7	.4459	.1475	.2586	.1314	.0311	.2213	.	..	.8323	.5773	.1786	.5199
8	..	.1570	.1310	..	.0413	.0292	..	..	..	..	.1983	.1602
9	.2195	.2019	.1725	.0804	.0647	.0506	..	..	..	.2999	.2666	.2231
10	.1634	.1604	.1420	.0352	.0433	.0386	..	..	..	.1986	.2037	.1806
11	.0234	.0188	.0145	.0483	.0305	.0249	..	..	..	.0717	.0493	.0394
12	.0228	.0234	.0212	.0017	.0014	.0011	..	..	..	.0245	.0248	.0223
TOTAL	1.0009	1.2346	1.3042	.4117	.6013	.7660	..	.4300	.6537	1.4126	1.8359	2.0742

Table 4

TOTAL TRITIUM BREEDING FOR THE NATURAL URANIUM BLANKETS

Blanket	$Li^6(n,\alpha)t$	$Li^7(n,n'\alpha)t$	Total Tritium Breeding
	Reactions Per Fusion Neutron		
10 cm	1.0530	.2722	1.3252
13 cm	1.0781	.1913	1.2694
20 cm	.8902	.1963	1.0865
26 cm	.7852	.1762	.9614

Table 5

FISSION AND U^{238} (n,γ) REACTION RATES FOR THE NATURAL URANIUM BLANKETS

Natural Uranium Blanket	U^{238} Fissions	U^{235} Fissions	Total Fissions	$U^{238}(n,\gamma)$
	Reactions Per Fusion Neutron			
10 cm	.1330	.0133	.1463	.2487
13 cm	.1863	.0161	.2024	.3818
20 cm	.1837	.0259	.2096	.5320
26 cm	.1986	.0315	.2301	.6654

TABLE 6

Hybrid System Performance

Fusion Reactor Model - TCT

η-injector = .64

Q-plasma = 1.2, 1.8

Duty factor = .8

η-direct convertor = 0.0

α = .5

γ = .95

Blanket Parameters

M-blanket = 1.7 (no fission), 5.4 (nat. U)

Fissile Fuel Conversion Ratio C = .4, .6

β = .99

Water Splitting Process - ISPRA Mark I

η-water splitting = .4

T_{MAX} = 750°C

η-hydrogen-oxygen turbine = .65

TABLE 7

UTILITY SYSTEM CONSIDERED

MONTHLY PEAKING LOAD VARIATION: 11850 - 21210 MW

GENERATION STATION DATA

	NO. OF STATION	CAPACITY (MW)	AVERAGE FUEL COST (MILS/KWHR)
NUCLEAR	11	200 - 1089	1.85 - 2.13
BASIC FOS.	21	222 - 840	5.89 - 7.03
MID-RANGE FOS.	20	22 - 270	6.17 - 10.90
PEAKER	11	22 - 600	15.70 - 20.00
INTERCHANGE			22.00 - 25.00

ENERGY STORAGE UNIT

1. PUMPED STORAGE

   PUMP CAPACITY : 624 MW

   GENERATOR CAPACITY: 624 MW

2. FUSION ENERGY STORAGE

   FUSION REACTOR POWER CONSUMPTION: 200-1000 MW

   HYDROGEN-OXYGEN TURBINE CAPACITY: 624 MW

ECONOMICS

THE ECONOMICS OF FUSION-FISSION SYSTEMS

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ABSTRACT

To determine if a fusion-fission (hybrid) system will be an economic energy source in the future, a computer simulation is used to predict future requirements, given a demand and cost factor scenario. A linear programming model can evaluate new generation alternatives by estimating costs of new technologies and comparing results to a base case without the new alternative. It was determined that the allowable capital cost for a successful hybrid system ranges from \$450 to \$680/kWt. This is 1.0 to 1.5 times current estimated costs of LMFBR and 1.8 to 2.7 times the present cost of LWRs. Many plants might be built ranging from 200 to 1000 gigawatts installed capacity. If the cost targets can be met, the hybrid power plant would be an economically justified research and development program in the U.S.

INTRODUCTION

Fusion-fission (hybrid) systems may become economically desirable for one or more of the following reasons: (1) the cost of fissile material becomes very high, (2) the hybrids become technological stepping stones to fusion systems and (3) electric production costs are lower than alternative systems. The development of fusion-fission systems will require substantial investment in research and development over the next 20 to 30 years in the United States. It is expected that such an investment will result in a net financial benefit. As a point of reference, the most likely return from the estimated \$8 billion Liquid Metal Fast Breeder Program[1] (LMFBR) in the United States is expected to be \$52 billion over a 50-year time period.

The decision to construct an electrical generating plant of any technology is ultimately made by the individual electric generating utility. Each utility, in turn, is expected to provide electricity to its customers at the lowest possible unit price. Thus, the only way that hybrid power plants will be built in significant numbers is if the long-term economics appears favorable as viewed by the utilities.

UTILITY DECISION PROCESS

Fusion-fission system, by their very nature, present the designer with greater problems than either of the separate systems. Besides the normal tradeoffs between capital and operating cost factors, there is the added problem of choosing the right level of electric production relative to fissile material production. Designs have ranged from those which consume electricity and produce large amounts of fissile material, called "electric breeders", to designs where only small amounts of fissile material are produced and the dominant product is electricity. To assist designers in making these decisions, the situation which may exist at the turn of the century has been simulated so that the relative value of producing fissile material or electricity can be more accurately appraised. This appraisal requires identifying the potential alternatives that electric utilities will have to choose from in selecting the next power plant addition to their system.

The generally accepted technique, stated as the criteria for utility decision process, is to sum all plant costs over a 20 to 30-year period into a single figure of merit through a process known as present worthing. This process weights the relative importance of expenditures in future years to the initial capital expenditure in proportion to the cost of borrowing money. The mathematical expression for this term, which we will call the 30-year cost, is as follows:

30-YEAR COST

$$C = P + \sum_{i=1}^{L} (1+r)^{-i} (F_i + R_i + OM_i + NM_i) \quad (1)$$

WHERE

C = CAPITALIZED COST

P = PLANT COST

r = DISCOUNT RATE

L = PLANT LIFETIME

F_i = FUEL CYCLE EXPENSE FOR YEAR i EXCLUDING NUCLEAR MATERIAL PURCHASE OR CREDITS

R_i = INTERIM CAPITAL REPLACEMENTS

OM_i = OPERATION AND MAINTENANCE (INCLUDING TAXES)

NM_i = NUCLEAR MATERIALS (U_3O_8 AND PLUTONIUM) PURCHASED

POWER PRODUCTION ALTERNATIVES

The principal alternatives for electric power production likely to be available in year 2000 are coal-fired plants, light-water reactors, and LMFBRs, in addition to the potential availability of fusion-fission systems. Table 1 describes the general characteristics of each system. Other advanced sources such as solar electric may also be attractive by year 2000 but were not included in this analysis. The characteristics of coal, LWR and LMFBR are based on a recent study of the U.S. breeder program. The hybrid power plants, on the other hand, are presented as a range of fissile material and electrical production levels and the associated allowable cost that would let the hybrid be economically effective in the estimated situation near year 2000. The role of the coal-fired plant in this analysis is to provide an upper limit on the cost of nuclear systems before a system of well-known performance becomes a less costly alternative. The light-water systems also have well-known performance characteristics but consume a resource (uranium) which is considered to be in much shorter supply than coal and would experience rapid increases in costs even before year 2000. The performance characteristics of the LMFBR are less certain and are represented in this analysis with a design yield of a 12-year doubling time of plutonium inventory.

The demand for electricity can markedly affect not only the number of hybrids which may be built but, in the case where introduction of the hybrid is keyed to high cost of fissile material, may control the date of hybrid introduction. Figure 1 shows the total electric energy requirements projected through year 2040 along with the estimated operating capacity of various technologies, called the reference case. The reference case includes the introduction of an LMFBR in 1993 and the eventual introduction of a pure CTR in year 2010. It is into this environment that we estimate the characteristic properties of a hybrid system which would be economically acceptable in the U.S. electric generating system near year 2000.

Table 1. Characteristics of Alternative Energy Systems Potentially Available in 2000

	LWR URANIUM	LWR PLUTONIUM	LMFBR	CTR	HYBRID	COAL
ANNUAL PLUTONIUM YIELD, kg/MWt	0.069	-0.128	0.142	0	0.5→1.5	0
INITIAL PLUTONIUM REQUIREMENTS, kg/MWt	0	0.403	0.684	0		
ANNUAL U_3O_8 REQUIRED, TONS/KWt	0.075	0.012	0.001		0	
ANNUAL SEPARATIVE WORK REQUIRED, kg/MWt	48	2.3	0.080		0	
^{6}Li (ANNUAL), kg	0	0	0	140	~50	0
ANNUAL NUCLEAR FUEL PROCESSED, kg/MWt	12	12	9.5	0	~20	0
TONS COAL (LIFETIME), TONS/MWt						18 THOUSAND
ELECTRICAL-TO-THERMAL EFFICIENCY	0.325	0.325	0.38	0.40	-0.20→0.40	0.40
CAPITAL COST, $/KW (1975 DOLLARS)	640	640	740	770	VARIABLE	550 +$5 MILLION/YR OPERATING COST

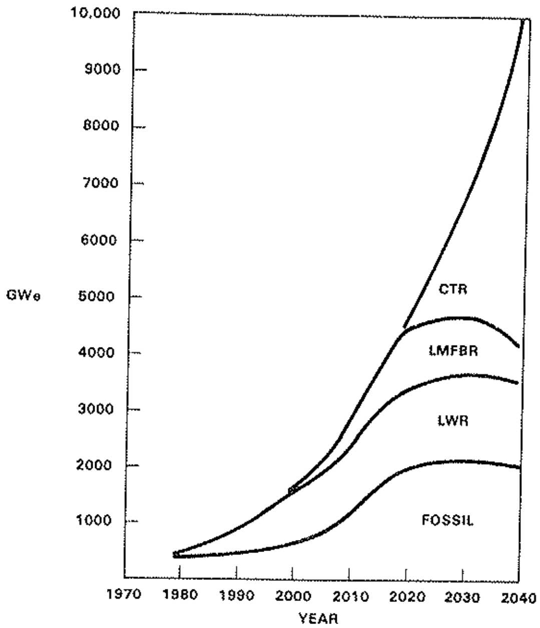

Fig. 1. Forecasted Operating Electric Generation Capacity for the Reference Case.

ECONOMIC ANALYSIS MODEL

A computer simulation system[2] that has been developed for the United States Energy Research and Development Administration (ERDA) was used for this study. The simulation estimates the optimum pattern of future electrical supply and is centered around a linear programming (LP) model. The model simulates the industry's process of determining what types of plants to build and when to build them. The decision criterion is the minimization of present worth system cost, subject to many decision limiting constraints. Many of the constraints make the decision process more closely resemble the utility decision process. The system cost includes both construction and operating costs.

In this model, each decision variable represents the number of plants of a certain type to be built in a fixed time interval. Each fueling scheme considered is represented as a unique variable. For instance, several variations of which years an LWR uses plutonium recycle are specified. This allows a simulation of the utility decision process to build an LWR plant without knowing what the lifetime fueling scheme would be. At some future date, if plutonium becomes available economically, the utility would change over to plutonium recycle. The LP model optimizes over the whole study horizon and selects the variable that represents switching over to plutonium recycle at that later date. This model, of course, is an aggregation of all utilities and does not attempt to model an individual utility. In this study, a model plant was generated for each combination of the performance characteristics for the hybrid. Then by changing the capital cost of the model plant until the desired effect is obtained, one can estimate the allowable cost for the hybrid.

A very strong feature of this model is that the cost of uranium and fissile material do not have to be specified as a function of time. Rather they are determined by the solution of the model, and thus effectively have continuous feedback loops. As the plant mix changes during solution, the cost of uranium and fissile material changes. Uranium cost is specified as an increasing function with the cumulative amount consumed (Table 2). These values were reviewed by a group of interested utility representatives serving on the EPRI Fusion Committee.

Table 2. U_3O_8 Cost/Availability (1975 Dollars)

THOUSAND TONS AVAILABLE	U_3O_8 COST, ($/lb) REFERENCE CASE	LOW COST CASE
400	18	16
500	20	16
500	25	20
500	40	20
1,000	60	24
1,000	80	40
1,000	100	55
1,000	110	70
1,000	120	80
3,000	125	95

SEPARATIVE WORK COST

YEAR	$/SW UNIT
1980	76
1990	75
2000	70
2010	65

A second important factor affecting fissile material price is the enriching process known as separative work. The unit cost (shown in Table 2) of this process is estimated to increase over the next 10 to 15 years but then decline as the result of new technologies that are presently being developed. A fissile material stockpile is constrained to never go negative: the fissile material must be produced before it can be used. The cost of the plants and other factors in the solution thus determines the price of the fissile material.

In each of these unit cost tables, as with the other costs that have been presented, it is assumed that inflation can be neglected in that it will have similar impacts on each of the alternative technologies and would also be extremely difficult to estimate over the time horizons treated in this analysis.

To accumulate the impacts of decisions by utilities beginning in 1975 and extending through year 2040, the cost of operating the entire United States electric generating system is summed where the weighting placed on each year's cost is modified by the present worth factor based on a 7.5% annual discount rate. These expenditures can be compared with R&D costs which are accumulated from 1975 in an identical fashion.

Figure 2 shows the unit cost over time for uranium and fissile material for the reference case. The leveling off of these costs is the result of the introduction of the breeder reactors and eventually the CTRs after the turn of the century.

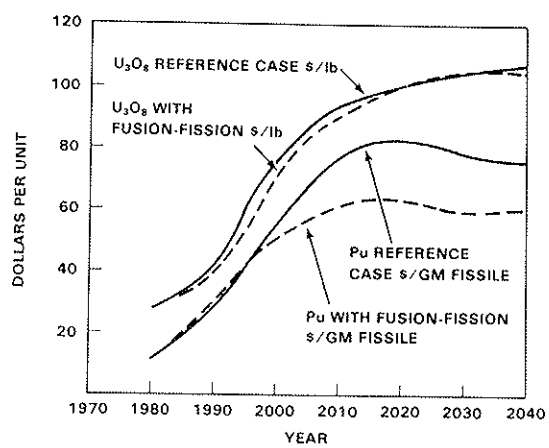

Fig. 2. Fissile Plutonium and U_3O_8 Prices as Affected By the Introduction of Fusion-Fission Systems in Year 2000.

DESIGN TRADEOFFS

Design tradeoffs can be established for fusion-fission systems based on the reference case conditions. These conditions simulate the availability of a hypothetical hybrid system which is characterized in terms of the fissile production/MWt, net electric efficiency and the 30-year cost exclusive of nuclear materials. To bracket the possible range of design characteristics, a high thermal-to-electrical efficiency of 0.4 was used, and a low value of −0.2 representing the electric breeder concept was selected. To make another data point available for interpolation, a third case was included in which the net electric production was zero. The range of plutonium production allowed was zero to 1.5 [illegible]/MWt. The point where zero plutonium oc[illegible]s is equivalent to the performance char[illegible]teristics of a pure fusion system.

By introducing all hybrid model plants into the model at high cost, the cost that makes each hybrid design just competitive can be determined from the reduced cost, a standard sensitivity analysis feature of linear programming. By further reducing the 30-year costs, the hybrid progressively captures a larger and larger market share and the resulting increase in system benefits is determined. Figure 3 illustrates the

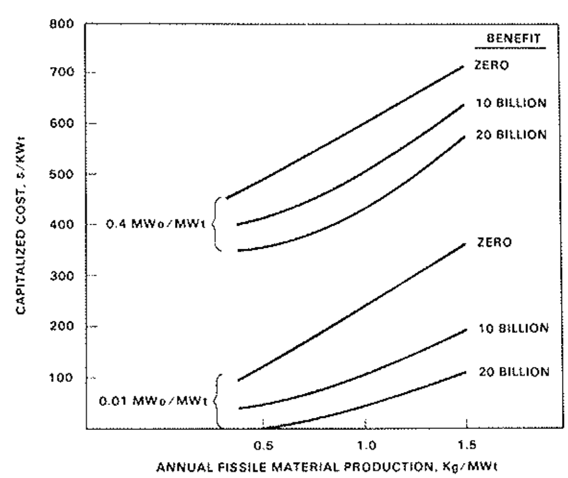

Fig. 3. Allowable Capitalized Cost For Fusion-Fission System (2000) With LMFBR (1993)—With CTR (2010).

result of this procedure for the reference case condition. From this figure, design tradeoffs between initial system cost, operating costs, plutonium production rate, and net electric efficiency can be evaluated. Experiments with additional design points have verified that simple interpolation between data points is quite satisfactory.

Figure 4 applies this concept to a mirror plasma confinement concept reported by Moir.[3] Point A on Fig. 4 shows the design location using Moir's cost estimate with A' indicating the allowable cost for fusion-fission concept application in the reference case situation. An alternative design point where more fissile material is produced is indicated by Point B and its associated cost requirement by B'. As can be seen, the design with lower plutonium production is very nearly at an acceptable level, whereas the electric breeder design is substantially off the target value and would require considerable cost reduction to be chosen by the simulation model.

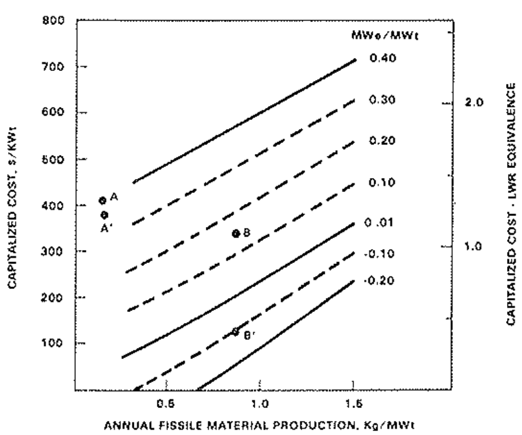

Fig. 4. Allowable Capitalized Cost For Fusion-Fission System (2000) That Yield Zero Benefit With LMFBR (1992)—With CTR (2010).

FUSION-FISSION ROLE WITH NONREFERENCE CASES

The fusion-fission systems could capture a substantially greater share of the electric producing system in cases where the LMFBR or other fission alternatives were found to be impractical or were delayed substantially. Of course, it is also possible that the other factors such as lower total energy demand, lower uranium costs and improved performance by the LMFBR would lead to a smaller share of the market for fusion-fission systems.

In the reference case, a window in time exists for the hybrid power plant between its approximate time of introduction in year 2000, when plutonium supplies are still short, and the time of introduction of the pure fusion technology. Although after the pure fusion alternative is available, there is still a strong continuing need for fissile material supplied to the large number of light-water reactors in operation by year 2000. Figure 5 shows the impact on the reference case of introducing hybrids in year 2000 with a relatively high electrical efficiency (0.4 MWe/MWt) and an annual yield of 1.0 kilograms of fissile plutonium/MWt. The number of hybrids built was quite small due to the availability of the pure CTR only 10 years after the availability of the hybrid. However, the effect is significant; 30% more LWRs operating in year 2020 using hybrid-produced plutonium yielding a $10 billion electrical system benefit. The impact on uranium and plutonium prices is shown in Fig. 2 for comparison with the reference case.

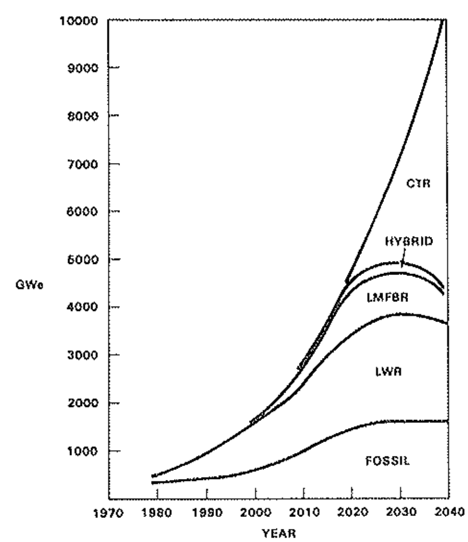

Fig. 5. Forecasted Operating Electric Generation Capacity With Hybrids.

The first nonreference situation examined involves the elimination of the pure CTR as a future energy source. Table 3 summarizes this analysis, comparing it with the case when the CTR is available. Note that the allowable 30-year cost has gone up only slightly expressed in dollars/kW. The 1.5 "LWR equivalence" means that a utility could afford to pay one and a half times the cost of building the LWR for a hybrid having the previously described performance characteristics. The uranium consumption is basically unchanged as is the value of plutonium. The result is that many more fusion-fission plants are required; these operate in a symbiotic fashion with the expansion of light-water capacity as the preferred substitute for the CTR market share. At the allowable 30-year cost shown, the hybrid-LWR combination is preferred over the LMFBR.

The next competitive situation, Table 4, considered involves the elimination of the LMFBR from the available alternatives to a utility. In this case the optimal solution in the absence of hybrid availability involves a considerable expansion of the energy supplied by fossil-fired power plants. As a result, the addition of hybrid technology into this situation with very high plutonium values requires only half as many plants to yield the same $10 billion in system cost reduction. In this case, the allowable cost of the hybrid is up to 2.25 in LWR equivalence units. Uranium consumption stops about 2020, but use prior to hybrid introduction still exceeds that used with LMFBR available.

The importance of the date when fusion-fission systems become available is shown in Table 5 for the 1990 and 2000 dates. The earlier introduction results in accumulative savings at an earlier date with higher present worth importance. Also, fewer light-water plants which consume plutonium from hybrids are in operation. The higher allowed capital cost ($510/kW) eventually limits the total market available to hybrids and LWRs in competition with the LMFBR technology. Due to the earlier introduction, less total uranium is consumed through year 2030.

The cost of uranium, which is a ready substitute for plutonium produced by fusion-fission systems, has a minor affect on allowable cost. When the schedules of cost and availability are changed from the reference case shown in Table 2 and replaced by the low schedule, a $30/kWt reduction in the allowable cost results. This is shown in Table 6 for the situation with LMFBRs available. The uranium usage is considerably higher as expected at the lower prices and LWR capacity also expands along with a slight reduction in the need for fusion-fission systems.

The demand for electricity is another important factor which is uncertain today and considerably more uncertain in the time period examined in this analysis. A lower energy demand reduces demand for uranium as shown in Table 7 by nearly a million tons through 2030. This reduced demand reduces plutonium values and necessarily lowers target cost for hybrids to produce a competitive product.

CONCLUSION

The economics of fusion-fission systems is more complex than any other reactor system due to its hybrid qualities. Design tradeoffs must be made between fissile material production, electrical production and capital costs. The proper choice of tradeoffs depends on the situation which the designer thinks is most probable some 25 years from now.

From the reference case and the series of alternative situations which have been presented, the allowable capital cost for a successful hybrid system ($10 million benefit) ranges from $450 to $680/kW. This is approximately 1 to 2 times the presently estimated cost of LMFBRs and 1.4 to 2.2 times the present cost of LWRs. These cost targets appear to leave sufficient room for the fusion-fission system to realistically operate. The number of plants that may be built could be large, as noted in each of the cases examined, ranging from 200 to 1000 gigawatts installed. Thus it appears that the hybrid power plant, <u>if</u> the described cost targets can be met, would be economically justified as an R&D program in the United States.

Table 3. Effects of CTR Availability (LMFBR - 1993, CTR - 2010)

	CTR	No CTR	Change Magnitude
Fusion-Fission 30-Year Cost ($/kWt)	450	465	15
LWR Equivalence	1.45	1.5	0.05
U_3O_8, Required (Million Tons) By 2030	4.0	3.9	0.1
Plutonium Value, 2020 ($/g)	38	38	0
Fusion-Fission Capacity, 2030 (GWe)	190	1036	846
LWR Capacity, 2030 (GWe)	2210	2830	620

Table 4. Effect of LMFBR Availability (LMFBR - 1993, No CTR)

	No LMFBR	LMFBR	Change Magnitude
Fusion-Fission 30-Year Cost ($/kWt)	680	465	215
LWR Equivalence	2.2	1.5	0.7
U_3O_8, Required (Million Tons) By 2030	4.7	3.9	0.8
Plutonium Value, 2020 ($/g)	55	38	17
Fusion-Fission Capacity, 2030 (GWe)	600	1036	434
LWR Capacity	3300	2830	470

Table 5. Effects of Fusion-Fission Availability Date (LMFBR - 1993, No CTR)

	1990	2000	Change Magnitude
Fusion-Fission 30-Year Cost ($/kWt)	510	465	45
LWR Equivalence	1.6	1.5	0.1
U_3O_8, Required (Million Tons) By 2030	2.9	3.9	1.0
Plutonium Value, 2020 ($/g)	37	38	1
Fusion-Fission Capacity, 2030 (GWe)	320	1036	716
LWR Capacity, 2030 (GWe)	2110	2830	720

Table 6. Effects of Uranium Cost Schedule (LMFBR - 1993, No CTR)

	Reference Schedule	Low-Cost Schedule	Change Magnitude
Fusion-Fission 30-Year Cost ($/kWt)	465	435	30
LWR Equivalence	1.5	1.4	0.1
U_3O_8, Required (Million Tons) By 2030	3.9	5.9	2
Plutonium Value, 2020 ($/g)	38	37	1
Fusion-Fission Capacity, 2030 (GWe)	1036	995	41
LWR Capacity, 2030 (GWe)	2830	4580	1750

Table 7. Effects of Energy Demand (LMFBR - 1993, No CTR)

	Moderate High	Low	Change Magnitude
Fusion-Fission 30-Year Cost ($/kWt)	465	435	30
LWR Equivalence	1.5	1.4	0.1
U_3O_8, Required (Million Tons) By 2030	3.9	3.1	0.8
Plutonium Value, 2020 ($/g)	38	36	2
Fusion-Fission Capacity, 2030 (GWe)	1036	650	386
LWR Capacity, 2030 (GWe)	2830	2640	190

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2. R. L. Engel, The Regional Analysis of the U. S. Electric Power Industry. Volume 2. Linear Programming Solution, BNWL-B-415 V2, Battelle Pacific Northwest Laboratories, June 1975.

3. R. W. Moir, et al., Progress On the Conceptual Design of a Mirror Hybrid Fusion-Fission Reactor, UCRL-51797, Lawrence Livermore Laboratory, June 25, 1975.

INFORMAL DISCUSSIONS

US - USSR SYMPOSIUM ON FUSION-FISSION REACTORS

DISCUSSION OF FUEL CYCLES AND ECONOMICS

Thursday, July 15, 1976
1055-1230
Bowen R. Leonard, Jr., Session Chairman
Battelle-Pacific Northwest Laboratory

SUMMARY

I. INTRODUCTION

In this discussion session, formal presentations on selected topics were not requested in advance. As a result most of the session consisted of far-ranging discussion comments from the participants with, at times, rapid changes in discussion topics. Although this procedure maximized the opportunity for individual expressions it also minimized the probability of successfully summarizing the discussions into a coherent topical breakdown as has been attempted.

The day before this session, the chairman circulated a list of topics for discussion. Some of these received little or no attention in the dicsussion session apparently because of lack of previous consideration to the topic or that prior considerations had not allowed reasonable conclusions to be drawn. These topics are itemized here for future consideration.

1. The question was raised as to whether conclusions could now be drawn on whether the different magnetic confinement schemes, laser fusion, or other concepts have inherently different hybrid roles.

2. Many hybrid studies are based on high conversion ratio fission power reactors. In these cases alternative fuel sources, e.g., enrichment, need to be included in assessing the value of hybrid produced fuel.

3. Hybrid-produced plutonium can have quite different characteristics than LWR or LMFBR produced plutonium. It can be very highly enriched in ^{239}Pu and can also contain large quantities of potentially troublesome lighter isotopes ^{236}Pu and ^{238}Pu. These characteristics have not been included in determining economic and social impacts in the fuel cycle including: fuel form, reprocessing, refabrication, fission reactor utilization, radioactive waste management and safeguards.

Much of the discussion session consisted of comments on the topic of the desirability of hybrids which produce fissile fuel and no export power (Symbionts) versus the results of many economic studies which showed that fissile fuel could be produced at (near) reasonable cost only when subsidized by the sale of electricity. It seems quite clear that the Symbiont is that desired by the customer (utilities). It seems also clear, at least in the absence of high-conversion ratio fission power reactors, that the sale of electricity from the hybrid reduces the cost of the fissile fuel produced in the hybrid in the majority of the studies which have been made. In view of this dichotomy, the chairman has not attempted to summarize the discussions separately in this area.

In the summary which follows, the chairman has attempted to organize the discussions into five topical areas. In the first of these, identified as II. General Economic Evaluations, some continuous presentations were made and these have been summarized separately. In the remainder of the topical areas, the authors of the comments are identified in the narrative summary.

II. GENERAL ECONOMIC EVALUATIONS

A. C. P. Ashworth, Pac. Gas & Elec. Co.

Mr. Ashworth presented the results of some estimated values of hybrids as interpreted by his utility. Comparisons were made for a hybrid which produced 1200 MWe of net power and for non-power producing hybrids (Symbiots) which utilized either the DT or DDD fusion cycle. The assumed performance characteristics of the hybrids were not presented except that they each were based on 4,000 MW of thermal energy. The first presentations were for hybrids producing Pu to fuel either LWRs or HTGRs compared with obtaining the same power from LMFBRs. These comparisons have been summarized at the top of Table I. The improved benefits for the non-power-producing DT symbiont over those of the power-producing hybrid appear to be dominated by the assumption that the assumed capital cost and carrying charges of the power-producer are 50% greater than those of the symbiont but they are not. The projections of overall plant performance are affected only to a minor extent by these assumptions. On the bases used, the DT symbiont appears to have some promise, particularly with the HTGR power source.

The comparison of the hybrids with the projections of a mixed GCFR and HTGR power system is unfavorable for any of the hybrid assumptions as shown in the second half of Table I. The overall conclusion of these comparisons then was that the pursuit of the attainment of the GCFR was much more favorable than the development of fusion hybrids.

B. I. V. Pistunovich, Kurchatov Institute, USSR

There are additional factors which should be considered in economic analyses of hybrids. A value factor was proposed which would compare the probable value of different types of hybrid reactors with each other and with other types of energy sources. This value factor would conceptually consist of the product of the probability coefficients. The first coefficient would be the probability that the type of fusion device would prove to be achievable in a physical sense. The second coefficient would be the probability that an economically

Table I. PG&E Estimated Annual Net Benefit Per Hybrid

		LWR	HTGR
LMFBR Comparison			
Power-Producing Hybrid	DT	\$ 1.2 B	\$ 0.1 B
Non-Power-Producing Hybrid Symbiont	DT	\$ 3.7 B	\$13.0 B
	DDD	\$23.0 B	\$55.0 B
GCFR + 3HTGR Comparison			
Power-Producing Hybrid	DT	\$-2.6 B	\$-2.8 B
Non-Power-Producing Hybrid Symbiont	DT	\$-0.8 B	\$ 0.2 B
	DDD	\$ 1.5 B	\$ 0.3 B

Bases: Hybrid Capital Cost \$3.0 B @ 19%
Symbiont Capital Cost \$2.0 B @ 13%

profitable system would be realized. The third coefficient would be the probability that the system would play a significant role in the future economy.

Dr. Pistunovich speculated on the probability of physical success, particularly for Tokamaks and Mirrors. Since the plasma parameters required for a successful Tokamak appear to need extrapolation of only about a factor of five from present values, the value of the first coefficient must be approaching unity. For the mirror system, however, the value of the successful coefficient may turn out to be zero.

Dr. Pistunovich did not speculate on the probability of fusion or hybrids but did point out that as opposed to thermal fission reactors, the fast breeder reactor could not yet be said to have a coefficient of unity.

The third coefficient was said to contain factors such as environment and safety which would be difficult to evaluate even in the comparison of thermal fission and coal power.

C. Ron Kostoff, DMFE-ERDA

The basic objective in the DMFE fusion program is to evaluate the various concepts for development. The whole point of having the economic studies is, through a cost/benefit type of methodology, to see which concept will offer the greatest net benefit and, thus, should experience the greatest development. Although a number of concepts have been proposed which appear to be attractive, there is a different probability for each to be successfully developed by a certain point in time at a certain cost. This must be taken into account in any rational cost/benefit analysis. Many of the critiques of the direction of DMFE programs have suggested different forms of development including advanced cycles and non-mainline concepts. The element of increased risk of successful development has not been considered in these suggestions. For example, the DT and DDD comparisons shown earlier by Mr. Ashworth should not contain the same 19 percent cost of capital because of the far greater risk of development of the DDD system.

One of the factors which must be included in future economic analysis is, thus, an accounting for differences in probability of development of different schemes at different times and costs as discussed by Dr. Pistunovich. DMFE has a study underway by Steve Nichols and John Vanceton at the University of Texas. The objective is to try to develop a methodology to assess different probabilities of success of development. This includes relations to different program development costs and in this way including it in the cost part of the cost/benefit analysis.

D. D. E. Deonigi, PNL

In response to questions from the floor it was explained that his economic projections to study hybrid market penetration were based on the assumption of introduction of a 20-year doubling time LMFBR in 1993 and a 12-year doubling time LMFBR in 1998. The 20-year doubler had no significant impact on the projections but the 12-year doubler was a critical item. Given a viable hybrid and only a 20-year doubler LMFBR, it was speculated that the 20-year doubler LMFBR would "fall by the wayside." With the 12-year doubler LMFBR, however, the introduction of a viable hybrid has only a small effect on the projected role of the LMFBR economy. The significant impact of the hybrid is to increase the relative contribution of the LWR primarily at the expense of reduced fossil energy. Even in this case, however, only a relatively small number of hybrids are projected. For this reason it seemed that the introduction of advanced fissile fuel systems into hybrids would introduce economic penalties which would outweigh their apparent advantages.

III. FUEL CYCLE DYNAMICS AND BURNUP

Duane Deonigi raised the question of the impact on fuel cycle dynamics on economic factors. In particular, in the economic analyses,all studies seem to be based on a common assumption that all of the components of the plant are operational at the same point in time. For example, the LLL-mirror hybrid study, the output thermal power would increase by a factor of two over the initial five year operation. How does this affect plant costs and can it be optimized?

D. J. Bender reported that in their studies they addressed these problems. The parameters which affect the economics are the capital penalty for unused thermal capacity, fabrication and processing fuel costs, and inventory of fissile material. All of these considered can be overall optimized in terms of fuel burnup. In the case of their particular plant the optimum burnup came out to be one percent but would expect to be dependent on plant concept. James Maniscalco, LLL, said they recognized the large economic penalty probably associated with a large swing power over the lifetime of the system. Their solution was to reduce the swing by proposing a changing laser repetition rate and also by adjusting to a shorter cycle through lower burnup.

W. H. Bohlke commented that in the PPPL hybrid design, molten-salt fuel was chosen, in part, in order to maintain a constant power level during plant life by controlling the fissile content of the salt.

Ken Schultz pointed out the correlation of fissile inventories in fission reactors and symbiots. Fuel dynamics may be an important economic consideration for symbiots.

Bill Allen pointed out that in their studies the burnup of the fuel was based on the anticipated use of stainless steel process tubes and they were assumed to have a lifetime of perhaps 5 MW-yr/m^2.

IV. STANDARD FACTORS FOR ECONOMIC ANALYSES

Ken Schultz commented that economic ground rules can make a big difference in the results of any study. Bookkeeping method, discount rate, lead times, etc. all will affect the final cost. A set of benchmark economic parameters would be useful to form a common basis for all analyses. Lack of such a set of parameters seems to be basic to the apparent diametrically opposed conclusions of some studies.

W. H. Bohlke explained in detail the economic factors considered in the analysis of the PPPL hybrid design. They concluded that increased electrical power output led to the economic optimum. The difference in this conclusion with that of the LLL mirror hybrid study he concluded must be dominated by differing economic assumptions rather than the physical parameters of wall load and converter power density.

Bill Allen reported that the analyses of the LLL laser-fusion hybrid showed that a large factor in the determination of capital cost resulted from the sheer size of the nuclear plant. He compared the volume as six times that of a PWR nuclear plant and emphasized that this was a factor which needed study to see if a significant size reduction could be accomplished. Jim Maniscalco, LLL, reinforced this viewpoint. Dan Cook, MIT, also emphasized that they are studying E-beam or laser-heater solenoid symbiots that are scaled to small size, as there appears to be economic advantages of size.

Ron Kostoff reported that DMFE-ERDA has been considering a two-day workshop in the next few months for the purpose of achieving a common costing basis for hybrids and CTRs.

V. MOLTEN SALT TECHNOLOGY

The chairman had noted that several hybrid reactor studies have projected the use of fission reactor molten salt technology in the hybrid blanket. The question was raised as to whether such studies were fruitful in view of the long lead time required in the development of the molten salt technology. It was noted that the final report of the Molten Salt Reactor Experiment concluded that engineering scale proof does not exist for the following key elements of the system; chemical steps, continuous fluorination for uranium removal, UF_6 reduction, bismuth isolation, and short-time cooled chemistry. In particular the bismuth isolation was noted to be a key problem.

W. H. Bohlke commented on the advantages of the molten salt hybrid blanket which had led to the choice of flibe for the Princeton reference design. A major advantage resulted from the ability to adjust the fissile concentration of the salt to maintain a constant thermal output to match that of the installed generating capacity. They received encouragement from chemical engineers at Princeton in their projected use of flibe. However, it was recognized that this choice would undoubtedly involve developmental costs and probability of success of the development would be difficult to assess.

R. S. Cooper argued that the fission technology (molten salt) should be viewed relative to fusion technology. Many millions of dollars have been invested in molten salt technology and undoubtedly many more millions are required for commercial development. However, all of the fusion systems are really in the research stage and not the development stage. Since there has not yet been a fusion equivalent of the Fermi pile, his opinion was that there will be sufficient time and funds available to develop appropriate fission technology as fusion-fission systems are desired.

Andrew Cook responded that they had to look into molten salt problems, of course, to justify their design. The first point made is that the bismuth extraction system for removal of protactinium is really not necessary for a hybrid. In particular, it was stated that you can afford to pay for extra inventory and hold up the salt for 21 or 23 days in order to allow the protactinium to decay into ^{233}U. (The chairman notes, however, that 90 days would be required for 90% of the 27-day ^{233}Pu to decay to ^{233}U.) The second point was that the MSRE did run successfully for two years at Oak Ridge with a few corrosion problems. Thus, although it is not a cut and dried proven technology, at least it is something you know can work.

In summary, it is clear that at some stage a decision will be required on the use of a molten-salt hybrid. The decision must be based on the projected cost:benefit of the technological development and the probability of success. The decision must be made at a time which will allow development concurrent with that of the hybrid fusion device.

VI. ^{233}U-Th FUEL CYCLE

The chairman had noted that the hybrid ^{233}U-Th fuel cycle, analagous to the hybrid ^{239}Pu cycle, had not been adequately assessed for economic and social impacts on: fuel form, reprocessing, refabrication, fission reactor utilization, radioactive waste management and safeguards. In particular, a study with U. P. Jenquin, BNW, to be reported at the 2nd Topical Conference on Fusion Technology had revealed several aspects which were indicated to require much more extensive study and evaluation. These included 1) the impact on fission reactor utilization and safeguards of the high ^{233}U content which can be achieved, 2) the impact of the ^{232}U content which could be 10-10^2 times that of HTGR produced ^{233}U, and 3) the implication of the enhanced toxicity of the ^{233}U at intermediate and long times.

Ken Schultz did not feel that an order of magnitude more ^{232}U would necessarily impact fuel reprocessing costs significantly since remote fabrication was already required. In addition, the high fissile content could substantially increase the fuel value above HTGR produced ^{233}U.

S. Gerstel expressed the greatest concern about the biological hazard potential of the hybrid produced ^{233}U due to the lower mass uranium isotopes. If this led to a higher environmental impact from a radioactive waste viewpoint it could be an important consideration in whether there were advantages in the thorium cycle.

The chairman agreed that this was an important consideration that needed much more evaluation than it had presently received. It was further pointed out that ^{234}U was enhanced greatly in its usage in the fission reactor thorium cycle. Since the ^{234}U radioactive decay daughters dominate the long-term toxicity of the uranium cycle, this should be an important factor to be considered in the thorium cycle, independent of the hybrid except that the hybrid is viewed as the attractive source of ^{233}U.

SUMMARY OF DISCUSSION SESSION ON HYBRID BLANKET DESIGNS

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INTRODUCTION

The discussion session on hybrid blanket design covered a wide range of topics, including neutronic analysis, spatial distribution and criticality, maintainability, etc. In preparing a summary of this session, excerpts from the discussion were employed to highlight the key points. Comments and questions from the participants have been paraphrased from the transcript of the meeting. Speakers and their affiliations have been identified, where possible. The apologies of the chairman are extended to speakers who may have been misquoted in the process.

OPENING REMARKS BY CHAIRMAN

I had some reactions to the blanket aspects of the studies we have heard described this week that might be worth sharing with you and which might provide a format for the organization of questions about designs in this session. One observation was that we have several different approaches to blanket design, in part because of different perceptions of what the design requirements are, and in part because people have adopted different design solutions to satisfy those requirements. It would be useful for us to try to separate those two effects, especially when we come to dealing with different sources of fusion neutrons. One is tempted to say that since we are all dealing with 14 MeV fusion sources, there ought to be a great deal of similarity in what you do with those neutrons in the blanket. However, I think there really isn't all that much similarity, after all. From the point of view of what the blanket designer perceives as his constraints then, the first table lists some considerations that I had seen manifested in one way or another in the various blanket designs.

Table 1. Blanket Design Considerations

- Mission - Breeding vs Power Production
- Duty Cycle - Fast Pulsed to Steady-State
- Geometric Constraints (e.g., toroidal vs mirror configuration)
- Fuel Management - Frequency of Reload
- Neutron Spectrum (e.g., Fast vs Thermal)
- Neutron Multiplication Required
- Constraints on Structure Between Plasma and Blanket
- Remote Handling Constraints
- Tritium Breeding
- Power Density Implications
- Constraints on Coolant Selection

First of all, from the point of view of what one is trying to accomplish, what is the mission? There has been a different emphasis in various studies on whether one should emphasize breeding or power production, or, in the case of our study, an even more diverse mission in terms of burning actinides. So the design of the blanket that the designer comes up with then has to be influenced by his perception of what the mission is.

The next item is the duty cycle. Even though we all work with a source of 14 MeV neutrons, the distinction between getting them in a very fast-pulsed mode from implosion of a laser pellet at the one end of the spectrum to essentially a steady-state duty cycle which one would get from a mirror-type concept on the other end has to have an impact. I'd be interested in the various designers' reactions to that design consideration. How did the pulse characteristics affect your approach to blanket design? In the case of the actinides, for example, we decided we have to have a relatively short off-pulse as well as a high-duty cycle, since the magnitude of the cool-down you get depends on the absolute value of off-time.

The third consideration involves geometric constraints. There have to be differences one is faced with in blanket design in trying to cope with various geometries, such as the toroidal shape of the tokamak and the mirror configuration, for example.

Next, fuel management. The frequency of reload that one anticipated having to deal with and how many of the blanket elements you have to replace have an impact on the design.

The neutron spectrum required is another area of consideration. In many instances, these are interactive considerations that depend on the mission you are trying to accomplish; the neutron multiplication you require, for example. For the actinides, we found we couldn't go it on fusion neutrons alone. We needed to have quite a considerable amount of neutron multiplication. We have a K-effective of about 0.9 in the reference actinide burner blanket. In fact, that consideration alone excluded quite a few blanket concepts that were otherwise attractive.

The next item deals with constraints on the structure between the plasma and the blanket. I'd like to stimulate some comments on that point. We found for our application you couldn't stand much attenuation of the hard neutron spectrum by the structure and, in fact had to take the whole blanket configuration and turn it inside out so the coolant manifold was on the outside and the fissile lattice was as close as possible to the 14 MeV neutron source.

The next deals with remote handling constraints. In a tokamak, for example, I think you have to be able to simply get at the blanket elements without taking half of the rest of the device apart, and from that point of view, that has an impact on how you approach the blanket design.

With regard to tritium breeding, one can try to do this in an integral sense with blanket modules or by some separate modules designed solely to produce tritium.

The next item deals with power density implications. We have seen a range of considerations there with the actinides. We see that we have to push the power density pretty high to accomplish the mission and to live with some of the implications that entails, in terms of cooling system requirements and in terms of consideration of faulted conditions, where you would postulate a loss of coolant.

The final item deals with constraints on coolant selection. Again, the effect of that constraint will vary depending upon what kind of a fusion source is involved. The magnetic confinement systems, for example, are faced with constraints due to magnetic fields. The complicated geometries have constraints due to the inability to use thick-walled pressure vessels that would be needed for very high pressure coolants.

That was one set of thoughts that occurred to me in response to the discussion of blanket designs this week. It might be appropriate for the people involved in some of the other studies to comment on how they perceived some of these considerations and what impact they had on the design.

The other topic I would suggest for discussion has to do with the fact that fissile technology has been considered to be "here-today" to the extent that we use essentially present-day fissile technology in several of the studies. There is a development and testing phase that has to be faced up to, however, and it would be worth stimulating some discussion on what that might involve. To start that off, even if one assumes that the fuel lattice type is utterly conventional, there are still some "environmental" considerations on the second table that one has to adapt to in getting the system to work.

First of all, one has to determine the neutronic response in a 14 MeV neutron environment. Development and testing would be required to neutronically qualify a given blanket design. We might be looking at some point, hopefully in the near future, to commit these designs to an actual full-size reactor test in determining their neutronic response.

We have geometric constraints that are different than those for conventional fission reactors with respect to coolant flow connections, access, and so on. In this case, one might very well find that the flow distribution and parameters such as

Table 2. Blanket Development & Test Considerations

Elements Unique to CTR Environment:
• Neutronic Response to 14 MEV Neutrons
• Geometric Constraints with Respect to Coolant Flow Connections, Access, etc.
• Materials Behavior for High Fluence Hard Spectrum Conditions
• Pulsed Operation Effects
• Magnetic Field Environment

unrecoverable loss coefficients for these geometries will require some testing; perhaps some of that can be done largely out of pile.

Materials behavior for high fluence, hard spectrum conditions is another such case. The thinking today is, at least in the U.S. program, that one would like to have something on the order of D-Li sources to look at materials behavior itself per se. However, there is an interactive behavior between materials and other aspects of the module design that can sometimes be rather a painful surprise. One example that comes to mind is that, even though we knew that fuel could swell or grow early in the days of the fission program, it turned out that some of the module design had an interactive effect. When the fuel swelled relative to the restraining structure, it effectively put the fuel elements into compression. They, in turn, tended to go into a buckling mode, which interacted with the distance between elements in a way that the local flow velocities would increase where the elements tended to come together, dropping the static pressure, and pulling the elements even closer together. Thus, even with a reasonable knowledge of materials behavior, one can still have a need to get at some of those kinds of effects by an integral test.

Pulsed operation effects are something we don't normally see in power fission reactors. These would require some plan to address how elements are qualified to behave in that kind of an environment.

And finally, for the magnetic confinement concepts, operation in the appropriate magnetic field environment would be needed to see what kind of surprises might come there.

So, I'd like to invite you to give some reactions where you think they are appropriate in terms of what you think the development and qualification problems are on some of these blanket modules concepts.

NEUTRONIC ANALYSIS

Ron Kostoff - (ERDA - DMFE):

We have about half a dozen different types of designs going, roughly half a dozen different fusion confinement concepts, and one of the things that I tend to worry about is that each designer is doing his own blanket design in terms of neutronic analysis. I wonder what the major differences are in terms of blanket design, and specifically neutronic analysis, with regard to the geometry of the blanket and to the fusion design concept and why for the purpose of conceptual designs we couldn't have just one or two organizations spending full time doing a different blanket neutronic analysis for each configuration, having the other system designers essentially listing the results of the neutronic design from specialist organizations. In the conceptual design phase, do we really need each system designer doing his own blanket design and blanket neutronic analysis? The second part of the question I wonder about, in terms of future work, is what remains to be done; one of the main things which remains to be done is the neutronic analysis of blankets regarding breeding and region sizes.

Chairman - Ron Rose:

Our own reaction on neutronics is that it has to be very closely coupled with all of the aspects of the design, and I foresee difficulties in integrating that because of possible communications problems with another group.

Ken Schultz - (GA):

I think the diversity we have seen in the blanket design here today supports your point of view there, Ron, that indeed the neutronics and the mechanical and thermal designs are very closely tied together. Each individual is going to come up with a different configuration, depending on their particular mechanical, thermal, and nuclear points of view. Perhaps one of the points that you're questioning, Ron - Ron Kostoff this time - is the expense involved with a detailed neutronics analysis. It is indeed expensive. In the past, much of this analysis expense has been associated with data handling and processing; 90 percent of all the work that I do anyway generically seems to be shuffling cross-section data, acquiring them, sorting them, and so on. This is an area where cooperation among the labs in a formal manner would be very beneficial; but in terms of actually doing the neutronic calculations, I think we would be stifling ingenuity, and it would not be productive at all.

Luisa Hansen - (LLL):

Outside of the differences in technology and in the size of the blanket, many of the presentations that we have seen here use different cross-sections, different codes, different ways to calculate neutronics.

Ken Schultz - (GA):

I think you have to be careful about saying we are going to work out a uniform approach, because again then you are closing doors, closing avenues. The benefit to be gained by the fusion community would be in having a standard means of analysis, a standard set of data, a standard set of tools available. If you want to use them, fine, use them; if you want to go out and rediscover the well yourself, that's your business also.

Paul Nicholson - (Draper Lab):

I just wanted to mention that some of the suggestions being made by Ron are, in fact, being pursued. Our current program is developing a standard methodology which then could be used by the individual designer suitably tailored to their needs. It will strive for some consistency in terms of data base, and also in terms of models and appropriate methods. These will be made available when we complete the job, which started about nine months ago.

Chairman - Ron Rose:

Compared to fission reactors, our experience is that neutronic analysis of hybrids is a good bit more complicated in the geometric respect. One is trying to get the neutrons to travel from one place to another; namely, from the plasma to the blanket. Depending upon the details of the structure that occur in between, an explicit representation can show that some of the smearing techniques that would be acceptable in fission reactors do not really give you the same kinds of satisfactory result.

Bo Leonard - (Battelle - PNL):

There is a special group cross-section tape distributed by Oak Ridge for CTR calculations, so you do have a standard data base to use for somebody to calculate their own system using standard cross-sections. One of these days that standard cross-section set will actually come up with all the errors fixed and be useful, I think.

But there are complications if you try and do benchmark calculations. It's very obvious that certain biases exist between calculational experiments and these biases are dependent on how important they are in a particular design. In a case where I now am getting a bias, I would want to go in and make my own data adjustments to minimize the biases that were important to that specific design. I wouldn't want J. D. Lee to calculate it for me, because I don't know where his biases are.

I do not think that the survey type calculations really take that much money. The important thing in the long run, when you try and really optimize and understand a design in detail, then we are going to want to use the CTR computers with Monte Carlo calculations. But, again, I think the person that is mentally and morally responsible for the design wants to make the calculation. The important part of it, the standardization of the cost intensive calculations, it seems to me will be taken care of by the development of the Monte Carlo program on the CTR computer.

Ron Miller - (LASL):

I tend to support this opinion. First you have to leave the designer of a blanket the freedom to calculate his own blanket parameters simply because he has to go through a lot of interactions which depend on blanket calculations. It must be iterative interaction between the design of the blanket and the neutronic results.

We try to issue computer codes only in a form that they are easily accessible on other machines. We have a CDC machine. There are IBM machines. There are UNIVAC machines. This has been a big problem for blanket designers to get some computer codes going on their machines that have been developed on other machines. So the interfacing of computer codes so they are easily run on other machines is a big and growing task, and we feel it should be undertaken on a broad scale.

The same holds for nuclear data, as Bo Leonard has said. There is a code being developed right now - it's being tested - which should have applicability to all major CTR blanket designs.

Ron Kostoff - (ERDA - DMFE):

Let me respond to that comment. What set me off on thinking about this possibility was a number of months ago when I was talking with a few designers about blankets which have as their main objectives power production and breeding with thorium. Talking to a wide-range of people, it appeared that you had a U-238 fission zone right next to the neutron source; you had a tritium breeding zone or two tritium breeding zones next to that going into the blanket using helium cooling -- you had a carbon-moderating region, and then the salt where you bred U-233. It seemed a number of different designers would work around and arrive at this result, this type of blanket; maybe a slightly different thickness and regions, differences in details but mainly ending up with the same type of blanket, given the same type of objectives. I am just wondering, while it's nice for everybody to compute their own blanket; is it really worth the cost at this stage in hybrid designs?

Luisa Hansen - (LLL):

I would add only one comment to that. We all agree that the person would like the freedom of calculating his own parameters. However, I do not agree that the person has the freedom of changing cross-sections. A benchmark, as he called it, is only valid as long as it's able to reproduce a benchmark measurement. Only at that point will I trust the cross-sections that I use as standard.

From the Floor:

In a quick response to Ron Kostoff, I don't think we are at the point yet where everybody comes out with the same blanket design even with the same objective. I think we still have to learn a lot. I would like to see the flexibility where everybody can do these calculations themselves.

SPATIAL NON-UNIFORMITY AND CRITICALITY CONSIDERATIONS

From the Floor:

I observed several things from these presentations here. Some are more questions than answers, but maybe this is the forum to raise the questions. One is, can we quantify what additional design complexities are required if we have a fusion-fission blanket? By additional design complexities, I mean, first, those dictated by geometry due to the plasma requirements. Several calculations presented at this meeting have shown that fissile breeding is highly nonuniform with solid blankets. This is not true for the molten salt. In the molten salt blanket, you can circulate the fissile material. Now the question is, knowing these spatial non-uniformities, which cover maybe two or three orders of magnitude, is it really meaningful for us to say that a blanket is always and in all conditions subcritical? We have much higher concentrations of fuel bred close to the plasma than further away from it. So, perhaps what we mean by keeping the blanket always subcritical should be quantified.

J. D. Lee - (LLL):

You were right, I think, in pointing out that the capability of our methods at present and our depth of calculations have only scratched the surface. We have to look at the spatial build-up in blanket zones. In our case, we looked at the time effects only by smearing the isotopic buildup throughout the whole zone. Obviously the peaking factor will be greater than we calculate, and the enrichment will be greater than the value we calculate. But I think to say we are going to reach criticality is probably overstating the problem. Obviously it should be designed so it won't be a problem, and the economics tend to show that you want to remove that fuel long before you ever run into the criticality problem.

Ralph Cooper - (Physics International):

Many other design considerations which you discussed are matters of engineering. However, in terms of complexity, I think we can have a qualitative measure of geometric complexity in the sense that linear systems such as fuel rods in fission reactors are simple in the sense of being one-dimensional. When one has a singularly curved surface, like a cylinder, you now have greater geometric complexity. Doubly curved surfaces, like spheres, I claim are more complex. I am not a topologist, but multiply connected figures, like toroids, are still more complex. I leave the audience to make their own judgment about complexity.

Dave Chapin - (Princeton PPL):

Part of all this is the plasma, and tokamaks and mirrors, although more complex, are more stable. I think you have to take that into consideration as well.

J. D. Lee - (LLL):

In the blanket that we showed, our fission rate across the fission zone varies by only a factor of four. I would be very surprised if the isotope production throughout the fission zone varies much greater than the fission rate varies.

From the Floor:

I agree with that. I'd like to carry it one step further in the sense you can get in there every year and reload your blanket. Perhaps the quantity that is important in this is the breeding rate relative to the burnup rate from the front to the back of the blanket. Surprisingly enough, that ratio stays fairly constant from the front to the back of the blanket. That, I think is a much more important parameter than just a breeding rate.

Chairman - Ron Rose:

In any case, that does establish a design requirement that you have capability to reshuffle fuel, say on an annual basis, without an undue shutdown of the reactor.

From the Floor:

That's right. And it allows you to talk about, on the average, an amount of breeding or percent of fissile material that you can get in the blanket for an average amount of burnup that's fairly constant.

Chairman - Ron Rose:

Do our guests have any specific comments you'd like to address to the issue of blanket design at this point?

Dr. Shatalov via Russian Interpreter:

We agree that these questions are important for further studies. One of the most interesting questions is the question of uniform fuel production in the blanket. We are also interested in obtaining a unified system of data. To obtain these data correctly, we have to carry out further experiments for a more complicated system. If we carry out such experiments, then we have a greater degree of reliability in predicting blanket performance.

MAINTAINABILITY

Bruce Twining - (ERDA - DMFE):

I wonder how many of the designers have been involved in the question of replaceability/maintainability? It seems to me that one of the things that's really going to prove the feasibility or unfeasibility of hybrids is the ability to get in there and get the fuel in and out in a reasonable time. I think Jim Maniscalco has gone into this question, in quite a bit of detail, in his laser fusion study. I wonder if some of the other designers have had a chance to estimate how long it takes for maintenance over the period of a year or what the availability of these plants might be rather than just assuming a number of, say, 80 percent.

Bob Holman - (Westinghouse):

In the actinide burner study, we did take this into consideration. Pushing as far as we could in this direction, we only ended up with a 66 percent capacity factor.

I think one of the things that has to be considered here is that there are major compromises in regard to the blanket. It would be very desirable to get the blanket in closer to the plasma, which means it would be integral with the vacuum vessel itself. With respect to maintenance, repair and management of the fuel, it almost becomes imperative that you separate these two functions. The design therefore has to be compromised with respect to the amount of material between the plasma and the blanket. We went through this with some real soul searching in the design of the actinide burner.

Dave Bender - (LLL):

We also try to assess these same considerations in our mirror hybrid work. Basically we know as we go to higher first-wall loadings, we are going to have to shut the machine down more frequently and pull the blanket out. Therefore, the higher wall loading design will have a lower capacity factor, and we have taken, I would probably imagine, a rather optimistic time of one month to change out the blanket in the reactor. At this point one could certainly question how we quantified this effect; at least at this point we are including it in our methodology. As we look at it further, we hopefully will have more confidence in the numbers we are using. But it's an extremely important economic concern, this trade-off on wall loading vs capacity factor.

Jim Maniscalco - (LLL):

Just perhaps a quick point on maintenance; perhaps we should make a distinction. It is one thing to go in there and shift fuel around. It's another thing to talk about going in there for a major change of the structural material. Perhaps both procedures would take a much longer period of time than shown by present estimates. In our design study we looked at the procedure to shift around the fuel as something we felt we could live without replacing the structure, for example, every year. A reasonable time, to us, seemed to be about five years to replace structure.

J. D. Lee - (LLL):

With respect to the system that we showed Tuesday, the mirror system with the reference module, the design of the reactor was based primarily on the requirement to remove and replace the blanket while still maintaining the geometry required for the inner plasma. In that reference case I think our duty factor was something like 76 percent after accounting for blanket removal and replacement time.

I'd like to make one other comment to what you said, Bruce. I hope you didn't give the impression that we have done all the neutronics work that's necessary. A lot of the new data are probably needed in higher energy ranges. I think we have only scratched the surface in the type of calculations and in the detail that is required.

I think one of the biggest uncertainties we have is how long the structure will last. We are assuming that the structural material will not last much longer than what we are predicting as economic times for the fuel to be removed. So, unfortunately, we not only have to remove fuel; we have to remove the structure that sees the primary neutron fluence; in other words, the first wall. In fact, for some of our cases, like the thorium cases, that is what can limit our time in core.

Chairman - Ron Rose:

That's interesting to hear that your time interval is about the same for the first wall and the module. If so, it makes sense to replace them concurrently and have them part of one unit. In the actinide burner, our goal was to be able to leave the vacuum vessel intact and not have to breach its integrity whenever the outer magnet shield and the fuel modules had to be removed. Bob, could you tell us how long it took typically to remove and replace the module?

Bob Holman - (Westinghouse):

First, let me just back up and mention one other item. The separation of the vacuum vessel from the module had another factor which must not be forgotten, and this is safety. We have a tremendous inventory of fissionable material in the blanket. We cannot afford to have anything happen that would change that geometry. Having the blanket as an integral part of the vacuum vessel gives you a possible concern with regard to geometry changes, at least locally.

Getting back to replacement, there was no real problem timewise. We elected to use a replacement scheme that permitted us to have blanket sectors or segments already assembled, so it was a matter of taking one out and putting one back in with a minimum amount of effort, as far as time was concerned. An estimate of roughly four to five weeks, which is much the same as indicated here, would permit us to do a complete change of the blanket. It we wanted to change only part of it, it would take correspondingly less time. However, to replace a first-wall or something pertaining to the vacuum vessel, on a five-year basis, we are talking 12 weeks at least and probably much longer than that.

Dave Kearney - (GA):

I wanted to comment very briefly on a concern that we have had at General Atomic on a pure fusion device that I think is relevant. In our designs, recognizing that blanket maintenance is a difficult problem, one again which will have a very significant effect on the viability of the reactor, we have taken an approach to minimize the remote handling problem by trying to shield the reactor such that one can have hands-on access at the external boundary of the reactor; in other words, at the magnets. I think the same solution could be applied to a hybrid device, even though the activity levels within the blankets are different than they are in a pure fusion device. By using this approach, we have felt that having access at least to the exterior shield, one can cut down significantly the time it will take to maintain the reactor even in maintenance problems where you have to go inside.

Chairman - Ron Rose:

What types of operations would you envision doing on a hands-on basis?

Dave Kearney - (GA):

I think any operations dealing with the diagnostics and auxiliary systems that are around the reactor as opposed to having to maintain everything within the containment vessel remotely. You know there are a great many systems that will be within the containment vessel and outside the external shield or in the vicinity of the toroidal coils, and there are bound to be many things that have to be done in that region with hands-on access. If we can, the time involved will be considerably shorter.

From the Floor:

The problem we found in the laser-fusion reactor design was the fact that tritium is going to walk through most parts of the reactor, in view of the high temperatures. If you are going to keep the tritium inside, you may be able to do that by double-walling the reactor and running the coolant inside of the reactor. But as soon as you start maintaining the reactor, you have to get inside, take the top off, open it up. In our case, we have three beams coming in the top, three in the bottom. Most reactors are serviced from the top with remote control equipment. Our reactor is basically designed so that to remove the first wall, one would open up the top, take the first wall out every year, and the second year one would remove the first wall again, take out fuel elements and put other fuel elements in. We were limited in the design to inputting fuel elements, top and bottom, because we had to break the coolant lines to remove the top and bottom blankets. We basically looked at the reactor from the point of view of geometry. We redesigned the reactor to minimize the neutrons which get to the main structure, and that would hopefully last 30 years.

From the Floor:

We have used low activity materials like carbon and iron carbide for shielding under the coils and there's a possibility that we would have to use, let's say, aluminum in the coils as opposed to stainless steel for the structure. However, at the stage at which our analysis has been carried out so far, we haven't had to go to that. We feel that we could have stainless steel as the structural material for the coils.

Chairman - Ron Rose:

If you had to go to aluminum, have you looked at what you have to do in the way of alloying it to get strength?

From the Floor:

I only mention aluminum because I believe the UWMAK III design has gone to aluminum in the toroidal coils for the purposes of keeping the activity level down. We have not looked into it.

Dave Kearney - (GA):

A question that's in my own mind because of the question Bruce asked involves the timing requirement for maintenance in the Solenoid Design that was described this week. I would expect one of the distinct advantages of the linear approach would be maintenance ability. Have there been some time estimates made of what it would require for maintenance in that concept?

Ralph Cooper - (Physics International):

Not quantitative ones. I think that we will only know how long they really take to maintain when we build systems. People, I am sure, have made many calculations or estimates in fission reactors of how long it takes to change fuel. There was a series of articles in Nuclear News on the real world difficulties of even working outside the pressure vessel. It's our hope that in a linear system one can replace linear elements relatively easily, but that may mean it would take three months in our case and a year in another case. I certainly hope that toroids can work, but they are much more difficult in a qualitative sense to provide for wall replacement. Statements were made, for example, that with ten tubes, if one tube fails, we will leave it there and use the other nine. Efficient reactors have hundreds of tubes. When one tube fails, you do not leave it there. I can sit down and calculate that it will take us eleven weeks to simply pull out a rod and put in a fresh one. But it's not that simple. These are vacuum systems. One has to have systems to bring in the tritium; one has to have the material to cool a wall, and can treat this either as a single integral system which might be pulled out of the center or one might have modules. I would like to believe one can have a module like the reference case of the Scyllac design. It showed a several-meter module that you pull out. But I am still faced with the problem of having a junction for the plasma tube, and that junction has to be broken; it has to be replaced in a vacuum-tight system. There are a hundred million questions in this area and you don't get the answers for a one hundred thousand dollar study.

ADDITIONAL ANALYSIS REQUIREMENTS

From the Floor:

The hybrid studies that we have to date have been funded on a relatively modest scale. For many of the blanket designs, there were neutronic analyses done, there were some structural analyses done, and there were analyses that were not done because of the shortage of funds. What I'd like to know from the specific designers is what analyses in the blanket weren't done because of lack of funds, but which they considered relatively high priority and they would do if they had the funds?

Chairman - Ron Rose:

One of the concerns we had in that category was the cyclic effect of pulsed operation on the blanket. There was an attempt made to design to accommodate that problem in the sense that we tried to keep the helium at the inlet temperature in contact with all of the main structure and coolant manifolds. However, the concern still arises with regard to the fact that, with helium flowing through the fuel elements at a flow rate that will extract 250 watts per c.c., all of a sudden you have a 10-second off-interval pulse to pump out and get ready for the next pulse, reverse the field coils, and so on. The impact of that on the fuel rods is something that we identified as a conern but didn't really have an opportunity to evaluate.

From the Floor:

Cyclic stresses in the fuel rods?

Chairman - Ron Rose:

Yes. In fact, I was especially interested in the laser system where you have a different pulse duration but have a similar type of concern.

Jim Maniscalco - (LLL):

I would respond on that question by telling you that, after spending a year and a few hundred thousand dollars, there are areas that we know we need to look at and do some more work on.

I think we indicated that a lot of the cost of the plant was in the buildings and the pipes, and that that portion of our design was certainly not optimized or even considered in enough detail. So we would like to do that. We would also like to do some more work on accident analysis in the hybrid design. Something that haunts me to a certain degree is that, in terms of low-level radioactivity release in any device that has a lot of tritium in it, I'm not sure whether with reasonable technology we can meet the release rates that are being met by fission reactors today. I think we need to take a look at that. We also need to take a look at accident analyses such as loss of coolant accidents in our respective systems. We make statements to the effect that we are operating at power densities in the same neighborhood as a light water reactor, and our accidents are therefore going to be of the same type. However, I note that a lot of the designs I have seen don't have one big pressure vessel. We have a lot of small tubing or piping or small vessels. So, therefore, the probability of any one of these smaller vessels rupturing is probably higher than it is in a fission reactor. Granted, the accident would be a lot less severe, but we need to analyze that. So, these are some things that I felt need to be done where we should spend some effort and some money after we have finished this first go-around.

From the Floor:

In terms of making repairs to the blanket, what maintenance procedures are involved? What kinds of things in that category are worthy of study?

Jim Maniscalco - (LLL):

I think for the hybrid it probably isn't as important as it is for the pure fusion device, because we know we have similar radioactivity problems to those that people have in fission reactors. We are going to have to use the same kind of remote handling equipment, the same types of shielding. Much more important than the type of equipment, we need to know just how long we are going to be down. I don't think we have really analyzed that point.

Chairman - Ron Rose:

There are some differences, however, when you are not handling fuel modules submerged under water which provides both cooling and shielding, particularly in using present-day fuel handling equipment. So there have to be some different approaches, I think, to incorporate provisions to both shield and cool the elements while you are moving them around.

Jim Maniscalco - (LLL):

I think, as you pointed out, there is a problem in moving these fuel elements around and maintaining some kind of coolant while you are doing that which perhaps hasn't been addressed. If it's going to require flexible coolant piping, that certainly provides more opportunity for an accident to happen during the procedure. We haven't looked at it in our design in any detail. We feel that, using liquid sodium as our coolant, we could sit with the coolant inside the fuel pin assembly process tube as we move it. The temperature would not go up enough to be a problem, provided we could complete that moving process in something in the neighborhood of an hour or half an hour. But again, that's a very rough calculation. This point needs to be examined in better detail.

Chairman - Ron Rose:

Is this an element that typically operates at 60 watts per cc?

Jim Maniscalco - (LLL):

The average power density would be about 60 watts per cc.

Bob Holman - (Westinghouse):

One of the things we ran into, both in laser CTR work and with the actinide burner, was lack of design data. When you try to find the capability of the structure, you have to have some properties for the material you are using. With stainless steel at perhaps 700° C maximum, there is no code-approved information which we require to do design work. One wonders when is it going to be available. I would say the fundamental problem that exists right now has to do with the lack of irradiated property data, the lack of data for cyclic operation, and the need for methods to apply this to the design.

Chairman - Ron Rose:

Particularly where you know that one of your principal radiation damage mechanisms is embrittlement of the material, so you end up trying to design for brittle behavior.

Dave Chapin - (Princeton):

I have a question on further calculations. One thing we would like to do is two or three dimensional blanket calculations, one reason being to more accurately predict the breeding rates. Another reason is to accurately predict the radiation damage at the magnets. What distance are we going to need for adequate shielding? It's difficult to fit the vacuum vessel with adequate shielding and coils all in a limited space.

J. D. Lee - (LLL):

I suggest when it comes out, you will find a lot of this type of information in the blanket workshop proceedings. We spent a week doing exactly this sort of thing at Berkeley a month or so ago, and came out with a docket that you would spend a few nights reading, listing all the things that need to be done. Shielding is one of the topics. Materials is an even larger topic. I think you would find a lot of answers to your questions in that docket.

Dave Kearney - (GA):

I want to comment on that, too. In regard to the blanket workshop, there are workshops on neutronics, et cetera, but there was one particularly on engineering. I note the similarity between the problems of the hybrid and pure fusion devices. In fact, there are very few differences. The questions which have been brought up now, particularly in terms of maintenance and cooling, were addressed there. The same source of concerns were addressed, and many comments will be relevant to the concerns of hybrid designers. I think those proceedings will be coming out in a few months.

ACTINIDE FUEL COMPOSITION

Luisa Hansen - (LLL):

This question refers to the actinide burning blanket you discussed. You assumed that the only impurity in your blanket composition was one percent plutonium, which seems to assume that you already have the technology for almost one hundred percent separation of the actinides from uranium. What would be the effect if impurity in blanket composition cannot be controlled this well?

Chairman - Ron Rose:

If we got too high an impurity level, it might run us up against the loading density limitation in the fuel matrix we selected, and it might force us to go, for example, from an oxide to a nitride form, just to ensure enough latitude over composition of the fuel pellet to get all the neutron multiplication characteristics required. Of course, it would pose an additional complication in that some uranium is now in a neutron flux environment as well. So one would get some additional fission and some further transmutation from that fact. We intentionally wanted to see if one could operate that kind of a depletion scheme based on actinides alone, however, assuming technology for a relatively high partitioning efficiency was going to be forthcoming. With this approach, one would not have to incur the additional transmutation that would occur if other fissionable material were also included.

Luisa Hansen - (LLL):

However, assuming that you succeeded in this separation, would you also have problems surviving thirty-year bombardment at high wall loadings of 10 megawatts per square meter?

Chairman - Ron Rose:

The wall loading of 10 megawatts per square meter was a level we identified as a goal that could change the complexion of what one could do with actinides. Getting to 10 megawatts per square meter has a number of problems associated with it, however. The approach we looked at was a combination of going to higher field on axis and greater elongation, from 1.5 up to 3.0. As a consequence, a higher neutral beam injection energy is required. The energy increased from two hundred keV up to about five or six hundred keV, which then raises a lot of concern about designing a neutral beam injection system with any reasonable efficiency. Last, but not least, among those complications is the question of the lifetime of the first wall. It was our assessment that at around one megawatt per square meter, one has a reasonable hope with materials we could foresee today in being able to look at a five-year vacuum vessel replacement cycle and perhaps a two-year liner replacement cycle. If one expects to get anything in that same range of time interval with mugh higher wall loadings, that's a question that just can't be answered today. It requires a lot more work in alloy development and in trying to see if one can understand the damage mechanisms in materials and find ways of developing them for fusion applications.

BREEDING BLANKET COMPOSITION

Sig Gerstl - (LASL):

It may be appropriate here to amplify an idea that was presented this morning by Dr. Shatalov, by asking the appropriate questions with respect to it. As you remember, he had presented some computational results which indicate that from a fusion point of view it may be possible to enrich thorium fuel elements enough in uranium 233 that you can put them right into a fission reactor without going through any reprocessing. The concept, I think, is very interesting. The question is, even if it were technically possible to do this, is it ecomically feasible? That is, how much do you save if you do not have to reprocess? Second, if it's technically possible, one could conceive of taking spent fuel elements from other reactors and enriching them in a fusion-fission reactor to a high enough degree of fissile material so that you can place them back into that fission reactor again. My question is whether this is feasible.

Dr. Shatalov via Russian Interpreter:

I have not worked out the idea at present, as far as economics are concerned. There might be some saving since, along with the uranium 238, we obtain uranium 233. This can be compared in a convertor with an economical gain of a factor or two. As far as a strategy for reloading is concerned, this is a related technical problem for which I cannot give you any answer.

Chairman - Ron Rose:

Would you have a concern about the durability of the cladding with this type of scheme?

Dr. Shatalov via Russian Interpreter:

According to the data we have now, the fluence is about 10^{20}. This fluence is admissible for a number of cladding materials, and certainly for stainless steel.

Dave Bender - (LLL):

On the basis of our economic studies, I would estimate that at most you might save 10 percent on the total cost of electricity by eliminating the one fabrication and reprocessing step between the hybrid and the burner reactor. It's still a plant capital intensive process in breeding.

Chairman - Ron Rose:

Are you quoting 10 percent of the gross cost or just of the fuel cycle?

Dave Bender - (LLL):

Ten percent of the total cost of the electricity. That would be the very maximum I would estimate you would realize, and probably less than that.

J. D. Lee - (LLL):

In response to your question, based on our thorium blanket calculations, it looks like you would have to have approximately six megawatt years of exposure to reach about three years atom fraction in the innermost zones of the thorium blanket, and it would probably be shot by then, according to present estimates of radiation damage.

From the Floor:

That was one of my concerns, although the fact that the initial U-233 will not be uniformly distributed in the pins is also a concern.

J. D. Lee - (LLL):

I think the blanket that Dr. Shatalov described has pins oriented parallel to the first wall, so they don't have that effect.

Jim Maniscalco - (LLL):

I understand in the design described by Dr. Shatalov you have a uranium blanket in front so you get some moderation of 14 MeV.

neutrons and in addition you have a thorium blanket. A design concept to carry further would be to also put some moderator between the back end of the fast fission blanket and the thorium blanket. You know if you get lower energy neutrons they are going to be less damaging to the materials. Our calculations have shown that in the region where the material is thorium, the burnup is going to be very low. A burnup limit, if it were a thorium oxide or thorium carbide, for example, assuming that thorium dioxide would have similar burner ability to uranium dioxide, would be 32,000 megawatt days per ton. In ten years, we predict a burnup of 10,000 megawatt days per ton. So the main limit will be the flux, and it will be the high energy neutron flux that will do the damage. I think low energy neutron flux just won't be as damaging to a scheme like that. So a scheme like that is worthy of analysis. You will pay a neutronic penalty by putting a moderator between the two zones. Some of the neutrons are going to be parasitically captured in the back end of the uranium blanket now, because you put moderator there. On the other hand, you could come up with a design where you feel fairly confident you are not damaging the material and the fuel elements could then go into a reactor.

From the Floor:

Since our main concern seems to be radiation damage, and Dr. Shatalov has answered this with total fluence numbers, may I ask what energy spectrum this is?

Dr. Shatalov via Russian Interpreter:

I do not carry these data with me. As far as I can remember, the spectrum is not very much different than in a fission reactor; very close.

From the Floor:

But the radiation damage function, that is, the function of radiation damage versus energy, is a very steep function toward high energy. This is why the fraction of high energy neutrons is so important.

Dr. Shatalov via Russian Interpreter:

There is a difference of a factor of two in this spectrum region we are talking about, and I would allow for a factor of two or even more. In any case, I show a probable thickness of moderator in choosing the zone thickness.

US - USSR SYMPOSIUM ON FUSION-FISSION REACTORS

DISCUSSION OF OVERALL SYSTEM DESIGNS

Friday, July 16, 1976
0900-1045

PROGRAM

I. Plasma Characteristics
Moir, Miller, Tenney, Pistunovich

II. First Wall and Coolant
Maniscalco, Rose, Schultz

III. Divertor and Pumping
Rose, Tenney, Jassby

IV. Systems Aspects of Blanket and Shield
Schultz, Powell, Lee

Session Summary Prepared by R. G. Mills

SUMMARY

I. Plasma Characteristics

A. R. W. Moir

The major issue for magnetic mirrors has always been and still is the enhanced losses due to microinstabilities. Experimental operations with the 2XII-B machine have demonstrated MHD stability, high density, and high ion temperature. An $n\tau$ product of almost 10^{11} has been achieved with τ scaling as $E_o^{3/2}$ up to 13 keV. For a hybrid reactor an $n\tau$ of 3×10^{13} is needed. Neutral beam injection has been demonstrated at a power of 3 MW and 20 kV. Start-up has been demonstrated and a quasi-steady state achieved for about four particle lifetimes in contrast to exponentially decaying plasmas previously achieved. The experimentally achieved $n\tau$ is about one-third of the classical value.

The major requirements needed to achieve a fusion-fission hybrid reactor are that $n\tau$ continue to scale with $E_o^{3/2}$ from 13 keV to 100 keV and that it approach the classical value. In other words the ion confinement time must be increased from 5 msec to 300 msec.

The main issue is enhanced losses caused by microinstabilities. Some have been identified and ways found to cure them.

In response to a question from the floor, the Q for a mirror hybrid was given as 0.7 at 100 keV.

B. R. Miller

The technological requirements of the linear theta pinch seem to be a relatively straightforward extrapolation of current experiments, and the engineering requirements of compression heating are well understood. Extrapolation of heating results on 10 cm plasmas to hypothetical 20 cm reactor plasmas is not extreme. Experiments on the Scylla 4P device, 5 meters in length, are already giving data on long systems. An extension of technology to the generation of 20 tesla fields seems within reach. In short, it seems that the ability to create a thermonuclear plasma in these linear experiments is possible, and we are reasonably confident that this can be achieved.

In response to questions and discussion from the floor, it was stated that Harry Dreicer will soon publish new studies on particle loss and thermal conductivity heat loss from linear pinch machines.

C. F. H. Tenney

The problem of estimating the reaction rate to be expected from a tokamak hybrid machine's plasma depends critically on the question of how high a pressure can be achieved in a tokamak configuration. The Q is relatively insensitive to plasma density, but the reaction rate is proportional to its square. The pressure will be limited by two effects: a) The ability of the beams to penetrate the plasma, requiring a high energy of the order of 100 keV and b) the plasma equilibrium limits of the tokamak configuration. A commonly-used rule-of-thumb is a limiting poloidal beta equal to the plasma aspect ratio.

However efforts to fix this value require assumptions about the distribution of pressure and poloidal current (θ-component of the total current) in terms of the poloidal flux function. Maximum beta-theta may be approximately the aspect ratio, but it is not at all clear yet what the limit really is.

Since the reaction rate varies as the square of beta-theta, model parameters such as tons per year of fissile fuel probably cannot yet be predicted within a factor of two.

D. V. Pistunovich

Optimization of tokamak plasma parameters leads to values not far from existing experimental parameters. Examples are plasma transport coefficients, stability margin and magnetic fields. The magnetic field does not offer a large region of parameter space within which to optimize. Our flux will vary with the fourth power of the magnetic field. Technical limitations will lead to a plasma density of about 10^{14} cm^{-3}. We must be precise in selecting transport coefficients and q. There may be reasons for selecting low q's, but on the basis of experimental data that exist now, it would seen reasonable to assume q greater than 3.

Finally, the question of high beta is a very interesting question to be pursued in the near future. At present a β of $\sim 14\%$ seems reasonable, but this may be high.

Comparison with current data shows that a break-even regime is possible, but this does not imply that it will be easy to take the next step to economic possibility. The experimental work is at the beginning.

During the floor discussion that followed the above section on plasma characteristics, it was stated (Moir) that the MX experiment anticipated an energy of 80 keV (100 needed for hybrid) and an $n\tau$ of $\sim 10^{12}$ (3×10^{13} needed for hybrids). For the mirror hybrid reactor, the neutral atom wall bombardment will be very low, about one watt per square centimeter.

II. First Wall and Coolant

A. J. Maniscalco

Consideration of the first wall shows problems that are unique to laser ignited fusion and problems that apply to all fusion systems. For laser ignited systems, neutrons, alphas, x-rays and debris will all react with the first wall. Neutrons are common to all approaches. However the alphas, the debris, and the x-rays to a certain degree represent something unique to laser fusion. Sputtering will lead to an increased vacuum pumping load; the temperature limits of the wall must be considered; and finally the stresses in the first wall are important; in particular, cyclical stresses and the heat they generate. Internal combustion engine data provide information about 10^7 to 10^8 cycle lifetimes at low stresses, but few data are available in the 5-10 kpsi range.

Three ideas for solving the first wall problem in laser systems are: dry wall, wet wall, and magnetically-protected wall. In the dry wall concept an uncooled, ablating sacrificial graphite liner is used. The wet wall utilizes a continually regenerated liquid metal coating. This method implies high stresses and a large pumping load from evaporated metal. The magnetically protected wall diverts charged particles and reduces the load to that due to x-rays, neutrons, and neutrals.

If any solution of the first wall involves a significant amount of moderator such as would be produced by five or ten centimeters of graphite, or if it involves more than a few centimeters of stainless steel, the performance of a fast fission blanket can be so seriously degraded to make it unattractive.

Cooling seems to be well done by liquid metals in the absence of a magnetic field.

In response to questions and discussion from the floor, it was explained that optical damage would result from a subsequent pulse encountering debris on the mirror surface and that an air blast may be adequate for interpulse cleaning. The hybrid scheme would simplify laser-ignited fusion problems by an order of magnitude.

B. <u>R. P. Rose</u>

In the Westinghouse actinide burner design, the first wall sees a fusion neutron energy flux of a little over a megawatt per square meter, with a system Q only a little greater than one. This implies a large amount of energy must be removed from the vacuum chamber. In order to avoid high pressure coolant passages, an attempt was first made to design an internal liner to be cooled by radiation. However, with a heat load of 0.4 MW/m^2 on the surface plus 15 w/cm^3 within the liner, a surface temperature of 1700° C resulted plus a 600° C rise through it. Consequently this approach was abandoned, and heat pipes were adopted. They require about 0.25 out of 200 m^2 of wall area.

In response to questions it was indicated that the first wall in the design was equivalent to about 2 cm of graphite and two centimeters of stainless steel. Deterioration of neutronic performance will make it advisable to utilize holes to reduce this amount.

C. <u>K. Schultz</u>

Fusion systems, whether pure or hybrid, have identical first wall problems of high stress and high thermal loads. Work at General Atomic indicates the use of a low Z ceramic material as a liner may be necessary. As an example a 7 mm silicon carbide plate should last about 4 years at 1 MW/m^2. It would be cooled by radiation to the wall.

Cooling of the structure would be done by helium, a medium preferred to liquid metals, salts, or water. The advantages of helium are: good heat transfer capability, chemically inert, transparent to neutrons and light, has no phase changes, and can rely on existing fission reactor technology.

Its disadvantages are lower heat transfer coefficients than for liquids, and a high cost.

III. Divertor and Pumping

A. <u>R. P. Rose</u>

The suggested particle-collecting arrangement consisting of flowing lithium guided by a wetted screen has had preliminary studies carried out in the laboratory. Photographs were shown of the liquid lithium, heated by heat lamps. The experiment allowed control of the lithium inlet temperature, inlet slot width, flow rate, heat flux, vapor pressure, and temperature.

Good wetting and rewetting were achieved, and the screen angle with the vertical could be varied widely without losing the lithium from the screen. This would allow close alignment of the flow path to the magnetic field in a real machine to minimize MHD effects.

B. F. H. Tenney

Consider the divertor as a pump. Although it would seem attractive to have a gas blanket around the plasmas, all the system implications are not yet well explored or understood. On the supposition that one is going to require low gas pressure around the tokamak plasma, the particle throughput requirement of 10^{22} per second looms large. If one wants to pump this quantity at low pressure, he needs a large pumping area of the order of the size of the first wall of the reactor, this doesn't seem very practical.

To pump the same quantity through a small area would require so high an $n\bar{v}/4$ as to imply a plasma flow (from high temperature) rather than a neutral gas flow. To guide a plasma flow through a small opening, one requires a magnetic field. Such a device is called a divertor. Pumping 10^{22} sec^{-1} at low pressure implies a divertor.

Current tokamak designs feature poloidal divertors, but there are other types. Stellarators, toroidal devices of large aspect ratio, have been built with toroidal divertors. The English are experimenting with bundle divertors.

Divertor questions complicate machine design and increase the complexity, cost, and maintenance problems. Coil designs and their shielding from neutrons usually force the size of the machine beyond that of the initial concept.

C. D. L. Jassby

The use of a divertor forces a large increase in size of the reactor, and for many reasons one should like to avoid its use.

This might be done by providing a pumping area equal to one-third of the wall area with a pumping efficiency of about one-quarter. A neutral temperature a little more than 10 volts could provide a gas density of only 10^{11} cm^{-3} and still allow pumping 10^{22} per second.

This might be accomplished by having large getter panels protected by chevrons allowing some mechanical means for moving them out physically for regeneration. Illustrations were shown of conceivable arrangements.

The PDX device, under construction, might be a suitable test-bed for development of such a system.

IV. Systems Aspects of Blanket and Shield

A. K. Schultz

All of the designs we have seen at this conference are low Q devices, many because they have to be. The tokamak is perhaps unique in its potential for high Q. Since I firmly believe that the most useful fissile material is U^{233}, almost twice as good a thermal reactor fuel as plutonium, I am interested in high Q systems.

Low Q systems require the adoption of high blanket multiplications thus forcing us away from thorium, and this may be closing an option to produce excellent fissile material for the existing light water and HTGR thermal reactor economy.

B. J. R. Powell

There is little incentive for low reactivity blanket designs for hybrid systems, unless one wants easy servicing of the vacuum liner when the fissioning modules are removed. Aluminum is rather highly transparent to neutrons and would help increase the blanket multiplication. Brookhaven studies have led to double cooling systems to keep aluminum alloys at low temperature, but only about 30% of the energy needs to be collected at low temperature, about 70% can be collected at high temperature.

This might be a benefit to fission-fusion systems since the high multiplication factors should give even lower fractions in the first wall. Evidently damage characteristics are better at lower temperatures. Lower temperatures would also allow thinner sections.

A principal safety problem will be the integrity of the fast fission zone. The consequences of a rupture in that zone should be carefully considered. In this connection it might be argued that helium would be a better coolant, since a rupture would not be so disastrous as with liquids.

C. J. D. Lee

The blanket, plenum, and shield occupy the space between the coils and the plasma. The reason we have such a thick plenum is to accommodate helium cooling. Liquid metals would be nice, but in a magnetic field it appears that even though the pressure drop may not be an overall problem, the structural material requirements to contain the pressure head with a circulating system would be unacceptable for a conducting coolant. The use of helium, requiring a large plenum, reduces the blanket coverage, a disadvantage since blanket performance drops precipitously with coverage, more than proportionally.

The answer to the questions of producing power or fissile fuel is system related. The systems we have looked at tend to favor fissile fuel. Minimizing the cost of the final product, electricity, suggests producing fissile fuel in a hybrid and burning it in a fission reactor. With blankets of energy multiplication in the approximate range of 10-20, system optimization calls for relatively low burnup which implies favoring net plutonium production over power production.

SUMMARY OF THE DISCUSSION SESSION ON FUSION-FISSION REACTOR SAFETY

John P. Holdren, Session Chairman
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Berkeley, California 94720

ABSTRACT

The session consisted of opening remarks by the Chairman to outline his views of the role of safety and environment considerations in the overall rationale for hybrids, followed by seven five-minute presentations from various perspectives and audience responses to these. The presentations were by: V. G. Vasiliev, USSR (engineering aspects of hybrid safety); G. E. Shatalov, USSR (impact of choice of fuel and coolant); K. Schultz, General Atomic (a vendor's view of safety, features of mirror systems); L. Steinhauer, Math Sciences Northwest (features of linear systems); J. Maniscalco, LLL (hybrid rationale, features of laser systems); F. Tenney, Princeton (hybrid rationale); R. Rose, Westinghouse (use of hybrids for transmutation of wastes). Presented here are the Chairman's synopses of the presentations and audience comments (based on notes taken at the time and on a stenographer's transcript), followed by an overview of the main issues raised and directions for further work.

J. HOLDREN

The rationale for developing hybrids, or any other new energy source, presumably consists of potential advantages over alternative systems with respect to one or more of the following characteristics: abundance of fuel, cost of delivered energy, timing, and environmental and social characteristics. Because the LMFBR already solves the fuel-abundance problem for thousands of years, there is little <u>practical</u> advantage in this respect for either pure fusion or fusion-fission hybrids. With respect to cost of delivered energy (not just fuel cost), it is also hard to make a compelling argument for fusion or fusion-fission: the capital costs of these systems, which will dominate the energy cost, are quite uncertain but likely for basic engineering reasons to be higher than those of pure fission. With respect to timing, hybrids have the potential advantage over pure fusion of being deployable <u>sooner</u>, and the potential advantage over LMFBRs of permitting a <u>more rapid expansion</u> of nuclear capacity (owing to higher breeding ratio). The possible early availability of hybrids and the related "stepping stone" argument (that learning from hybrids will bring about pure fusion sooner) are weak rationales for hybrid development unless hybrids are better than alternative "interim" technologies in other respects; the hybrid's advantage for rapid expansion of nuclear capacity is weakened by the increasing likelihood of slower electricity growth in the industrial nations where the main market for this sophisticated technology lies, and by the accumulating inventory of plutonium from light-water reactors (in the U.S., at least), which makes the LMFBR's low breeding ratio less of a liability. If, as suggested here, the rationale for hybrids on grounds of fuel supply, cost, and timing is relatively weak, then the environmental and social characteristics in comparison to alternative technologies take on increased importance. (A more provocative statement of this point of view, with which hybrid proponents should be prepared to contend, is: the technical/economic rationale for developing the hybrid is so marginal that it should be built only if a design with <u>significant</u> social/environmental advantages over pure-fission alternatives can be found).

In evaluating environmental as well as other characteristics of hybrids, it is essential to determine the appropriate "yardsticks" against which hybrids should be compared. The answer depends on which of the possible roles of hybrids is under discussion. In the electricity-production role, the appropriate comparisons are with pure fission and pure fusion reactors (and in a broader context with non-nuclear generating technologies); in the fuel-production role the comparison is with mining and enrichment or with pure-fission breeders; and in the waste-transmutation role the comparison is with other waste-management schemes.

Four sets of environmental issues generic to nuclear power provide a framework for a first cut at evaluating the environmental characteristics of hybrids: (1) routine emissions and exposures (public and occupational); (2) accidents and sabotage (reactors and other fuel-cycle facilities); (3) international and intranational spread of nuclear weapons (explosive and radiological); (4) long-term management of radioactive wastes. Characteristics of hybrids that are relevant to evaluating the risks in these categories are: (a) radioactive inventories, including tritium, activation products, fission products, and actinides (including fissile isotopes); (b) pathways for release of the inventories, relevant to which are criticality behavior, response to loss of coolant or coolant flow, other stored energy forms, "geometrical" aspects (seams, welds, valves, lengths of pipes), and amount of transport and handling of radioactive materials; (c) systems aspects, including impact of hybrid technology on fuel choices and fission-reactor mix within the nuclear system as whole.

V. VASILIEV

Some of the most complicated safety problems in hybrid systems arise from the necessity of a tritium-production zone in close proximity to the core of the reactor. The environmental and safety characteristics of hybrids will depend critically on the choice of materials for coolant, structural elements, and breeding of fissile isotopes and tritium. Liquid-metal and molten-salt coolants will be troubled by high corrosion-product activity, and the fire/explosion hazard of liquid metals must be considered. Soviet work at this point seems to favor tritium breeding in solid (ceramic) lithium compounds, with gas cooling. It is important to develop much additional basic information about the properties of candidate materials under the neutronic, thermal, and chemical conditions that would prevail in hybrid reactors, in order to be able to iterate intelligently on early designs and to re-evaluate economic costs and safety characteristics.

G. SHATALOV

Although economic costs have been central in choosing energy systems, environmental and safety aspects cannot always be evaluated in terms of money. The correct choice of systems depends on these nonmonetary aspects, too, and it is not too soon to start considering them now in order to develop the necessary information to make intelligent choices.

With respect to criticality in hybrid-reactor blankets, it is important to note that some systems designed to be subcritical could become critical in the extreme circumstances of certain low-probability accidents. The systems with the best criticality characteristics are those that use natural or depleted uranium with little moderator in the reaction zone. One should try first to develop hybrids along these lines, although keeping in mind the economic considerations. As far as coolant is concerned, helium is one of the best choices; if the energy production is not too high (i.e., fuel production is emphasized) the helium pressure need not be very high, alleviating the only safety issue with this coolant. Molten salt coolants are more dangerous, both in terms of probability of accident and in terms of the consequence of an accident.

K. SCHULTZ

It is not too soon to start thinking carefully about the safety aspects of hybrid systems, but it is not useful to think in terms of one system being "safer" than another. One can assume, with some faith in our regulatory agencies, that any system that is permitted to be built will be safe; the question is how the safety is achieved and what it costs. The hybrid's potential advantage is in being able to replace some of the expensive, engineered safety systems of pure-fission reactors with inherent (passive) safety features for which one doesn't pay extra. (To try for a "super-safe" system by adding engineered safety features to an inherently safe-enough design is to waste society's money).

Although mirror hybrids - perhaps all hybrids - seem to look more like fission reactors than fusion reactors, achieving a design with guaranteed subcriticality at all times as an inherent characteristic may be possible and is worth striving for. To achieve this goal requires considering fuel reconfiguration and cooling; not only must k at operating configuration and temperature be less than unity, but (more difficult) k_∞ with no coolant and at room temperature (perhaps even cryogenic temperatures) should be less than unity. Concerning choice of coolant, General Atomic agrees strongly with the view expressed by the Soviets here that helium is preferable to lithium systems on safety grounds.

Completely passive cooling of the blanket in loss-of-coolant accidents is hard to guarantee (except perhaps for pure fusion systems); reliance on natural convection requires maintaining geometry within certain limits, and the energy stored in magnet systems and vacuum systems may generate forces that can reconfigure the fuel. (Magnet-system and vacuum-system energy represent safety disadvantages compared to pure fission).

An important disadvantage of hybrids (and pure fusion) compared to pure fission is that the activation products produced in such abundance by fusion neutrons cannot readily be concentrated into small volumes as can the fission products and actinides that dominate fission wastes. (This is so because many activation products are the same chemical element as the parent material, making separation from large volumes of structure extraordinarily difficult). Another important problem may be doses received by workers in routine maintenance or major repairs; some fission-reactor repairs have "burned out" (in terms of allowable radiation dose) a significant fraction of the available skilled workers, and the pool of people who could repair the sophisticated equipment in hybrids might be much smaller.

L. STEINHAUER

An advantage arising from the geometrical simplicity of linear hybrid systems is ability to construct the blanket from a large number of identical modules, easing mass production and quality control. Another consequence of geometrical simplicity is permitting a design with relatively few welds, a safety/environment advantage. Linear systems with small plasma chambers minimize the potential for accidental reconfigurations of fuel that depend on falling into the void to increase criticality, a point of particular concern if highly enriched plutonium blankets are considered. Possible disadvantages of linear systems are a high surface-to-volume ratio (although rough estimates seem to come out not much different than for mirrors and tokamaks), and the use of magnets adjacent to the first wall. The latter arrangement puts large magnetic forces in a place where reconfiguration is dangerous, and it introduces a different class of materials (e.g., copper) into a region where neutron activation will be intense.

J. MANISCALCO

The rationale for the hybrid on grounds of economics and timing is stronger than suggested by J. Holdren's opening remarks, if one considers the fuel-producing rather than the power-producing role. This has to do with the economics of the nuclear sector as a whole: a hybrid that can supply fuel to eight LWRs is economically attractive at 2.2 times the cost of an LWR, while the LMFBR can only compete at 1.25 times the cost of an LWR (all according to the work of D. Deonigi, PNL). And although the LMFBR has a head start on the hybrid of 10 or 20 years in terms of first deployment, the hybrid's ability to fuel 8 converter reactors per hybrid plant means that, once deployment does begin, the impact on the whole energy system can increase very rapidly. From the safety/environment viewpoint, the pertinent question is whether a system of hybrids feeding LWRs is preferable to a system consisting mainly of fission breeder reactors.

The philosophy in our laser-hybrid design was to achieve essentially the same environmental characteristics as an LWR. Inasmuch as hybrids would represent less than 20 percent of the installations in a mix of hybrids and LWRs, there would not be a major improvement in the safety of the system to be gained from making hybrids much safer than LWRs.

Our hybrid design with liquid lithium coolant can handle a loss of coolant flow accident, but we haven't looked at actual loss of the coolant from the fission zone.

F. TENNEY

If the public doesn't accept fission reactors for safety reasons, it is unlikely to accept hybrids. If fission reactors are accepted, on the other hand, then the main requirement for hybrids is that they not be worse. It seems unlikely that they will be, in view of the potential advantages in respect to criticality accidents and loss of coolant. A way to look at the economic rationale for the hybrid is as protection for the country's investment in LWRs.

R. ROSE

One environmental consideration that is different in the hybrid's actinide-burning mode than in the power-producing or fuel-producing modes is the large quantity of actinides assembled in one place; one must think both about the potential problems in transporting all this material around and about the safety aspect of having it all together in one spot. Another safety consideration is that the technology of the actinide-burning mode pushes one in the direction of high power density, making a loss-of-coolant accident more difficult to cope with.

Concerning the philosophy of safety, it is not useful to postulate non-mechanistic accidents - those for which no physically realistic mechanism of occurence can be identified. The useful approach is to do failure analyses of the system's components. Safety does need to be addressed early; this will provide incentive for early definition of systems in detail, to permit characterization of the failure potentials.

Of particular concern to us (at Westinghouse) is wrapping large quantities of actinides in a system that has a great deal of stored energy on the surface (in the magnets). The determination both experimentally and theoretically of the potential failure modes of these magnets needs much more attention. Our own heat-transfer analyses suggest that trying to cool a postulated situation in which the whole magnet goes normal is out of the question. Needed is a combination of magnet design and cooling system that provides adequate protection against release of the magnet energy in a catastrophic way.

COMMENTS FROM THE FLOOR

ON THE ISSUE OF HYBRID RATIONALE AS IT RELATES TO SAFETY - J.D. Lee (LLL)

If the doubling time of the LMFBR turns out (perhaps because of safety factors imposed on the design) to be 20 or 30 years, a useful role for the hybrid that hasn't yet been mentioned here might be as a supplier of fuel to the LMFBR. R. F. Post (LLL): The rationale for the hybrid as a stepping-stone to earlier deployment of pure fusion has a strong environmental component, as follows - if environmental and social characteristics are indeed becoming more and more important, and if pure fusion is the only viable long-term source that is good enough in these respects, then the running start toward pure fusion that the hybrid can provide is very valuable. C. Ashworth (PG&E): Building power plants these days has become such a burdensome proposition from the standpoint of public acceptance that any all-new concept (such as the hybrid) will have to be perceived by utility executives as having major public-acceptance advantages over existing concepts (such as the LWR and LMFBR), or they will not be interested. R. Moir (LLL): This may not be so big a problem since the hybrid as a fuel producer does not represent an all new system; only one hybrid would have to be added for every five or so fission reactors of existing, familiar types. Also, one could move the fuel-producing hybrids to locations very remote from population centers, alleviating the public-acceptability problem. J. Holdren (LLL): Separating the fuel-producers from the power plants aggravates what many people think is the most serious problem, namely transporting fissile material and thus increasing vulnerability to theft. There seems to be a growing feeling that clustering fuel-cycle facilities and power plants is better.

ON TECHNICAL ASPECTS OF SAFETY - W. Allen (Bechtel):

Passive cooling in the primary loop is not too hard to achieve, but usually an active means of removing heat from the secondary loop must be used to provide the ultimate heat sink necessary to prevent an eventual meltdown; in LMFBRs, if the heat cannot be dumped to the condensors via the steam generators, then a system of dump heat exchangers fed by fans goes into action. Thus the system is not passively cooled all down the line, and this is likely to be true of hybrids also. Liquid-metal cooling in a fusion or hybrid system has a liability not present in the LMFBR, in that the combination of a large vacuum system and the containment of the coolant in process tubes (as opposed to a "pot" or vessel) permits loss of coolant away from the region that must be cooled. Putting a thick vessel between core and blanket as a back-up to catch coolant leaking from process-tubes would impair the neutronics; putting such a vessel outside both blanket and core would entail a very large volume to be filled before cooling was assured, meaning the system would have to contain a great deal of liquid metal. If a melt-down does occur, incidentally, it's easier to cope with in a metallic-fuel system because the melting point of the uranium metal is lower than that of steel; thus it is possible to cool from outside a steel vessel which catches the molten fuel, or to cool the steel liner of the inner containment if that's where the molten fuel ends up.

OVERVIEW: ISSUES AND DIRECTIONS FOR FURTHER WORK

Concerns about safety, environmental, and social impacts seem to be growing in importance relative to other criteria for choosing among energy-supply options; this is true in at least part of the technical community, it seems to be true in terms of public acceptance of new energy facilities) it is true with respect to utility executives. Hybrid proponents must consider the possibility that this technology might not be developed, and if developed might not be deployed, unless it is perceived to have significant safety/ environment advantages over pure-fission alternatives.

Of course, the likelihood that the safety/environment factor will be this critical and, if it is, the margin by which the hybrid will have to be better, both depend on the strength of the incentive for hybrids in strictly technical and economic terms. It appears from analyses done to date that only two such incentives are likely to be very significant: (1) by serving as a stepping-stone, development (and possibly deployment) of hybrids will accelerate the time when much more desirable pure-fusion systems will be available; (2) in a primarily fuel-producing (as opposed to power-producing) mode, hybrids could supplement long-doubling-time LMFBRs or replace them as suppliers of fissile fuel for pure-fission convertor reactors, permitting a bigger fission economy and/or getting there faster.

The first incentive may be found very persuasive in the fusion community, but it is hard to imagine its impact being very substantial among hard-pressed utility executives or government policy makers dividing up finite development funds. The second incentive is potentially more important. What needs early quantitative investigation is the sensitivity of this economic rationale for the hybrid to (a) assumed electricity growth rate, (b) assumed nuclear fraction of electricity genrating capacity, (c) LMFBR doubling time, and (d) cost of mining enriching uranium.

The apparent unattractiveness of hybrids as straight power producers means the central question about safety and environment is not "How does the hybrid compare to an LWR or LMFBR as an individual power-producing unit?" but rather "What are the environmental characteristics of a mix of nuclear systems that includes hybrids compared to those of a mix that does not?" Some of the principal uncertainties that must be resolved to answer the latter question have to do with pure fission: the hybrid might permit LMFBRs in the fission part of the system to be replaced with HTGRs, but one needs to know more about both fission reactor types to know if this would really be a safety/ environment improvement; the hybrid might permit the nuclear system to run on the thorium/uranium-233 cycle instead of the uranium-238/plutonium cycle, but one needs to know more about both cycles to know if this would be an improvement. (The "conventional wisdom" says yes in both cases, but there remain too many uncertainties to promote confidence that it's really so).

Both from the point of view of understanding the environmental characteristics of the system and in order to favorably influence hybrid design at an early stage of development, of course, it is necessary to look in detail at the environmental and safety characteristics of individual conceptual designs. Points which the discussions in this symposium identified as particularly worthy of attention are (in no implied order of priority):

(1) Guaranteeing blanket subcriticality at room temperature or even cryogenic temperature, in the absence of coolant, and considering improbable reconfigurations such as collapse into the vacuum chamber. Are there combinations of

blanket composition and geometry that can meet this condition?

(2) Designing adequate safeguards against the destructive release of magnet energy. How much could be released, how fast, and what could be tolerated?

(3) Investigating the details of the apparent trade-off between breeding ratio and passive coolability of the blanket in the event of loss of coolant flow. How passive do we really mean or want when we say "passive coolability"? Passive in the primary loop only, or more?

(4) Developing more detailed, disaggregated quantitative measures of hazard potentials in various designs: curies, biological hazard potentials (curies divided by MPCs), integrated biological hazard potentials (BHPs times mean lives), integrated biological potentials times physically realizable release fractions in accidents, extent and frequency of handling of materials needed (expressed for radiological hazards in BHP or IBHP per year and for fissile materials in grams fissile per year, and also in number of batches per year). How do those figures vary with burn-up and fuel turnover time?

(5) Investigation of the sensitivity of answers to the foregoing questions to choice of coolant, tritium-breeding material, fertile material, metallic or ceramic fuel, cladding, structural materials. This requires acquisition of a great deal more basic materials information.

All seemed to agree that it is not too soon to begin in earnest to tackle these complicated and important environmental aspects of hybrid design.

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