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Editor: S. Locke Bogart

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FOREWORD

On November 1 and 2, 1977, a meeting was convened in Washington, D. C., to review the status of and prospects for fusion-fission energy systems. These volumes present the papers delivered at this meeting and the questions and answers following each paper.

The agenda of the meeting was developed to address, in turn, the following major areas:

- Problem Characteristics – Specific problem areas in nuclear energy systems for application of fusion-fission concepts.
- Current and Planned Fusion-Fission Energy Systems Activities – Current and proposed fusion-fission programs in response to the identified problem areas.
- Economic Considerations – Target costs and projected benefits associated with fusion-fission energy systems.
- Technical Problem Areas – Technical problems associated with the development of fusion-fission concepts.

The greatest emphasis was placed on the characteristics of and problems associated with fuel producing fusion-fission hybrid reactors. Because of the limited scope of the meeting, the broader issues of advanced nuclear reactors and their fuel cycles received little attention.

Since November of 1977, it has been decided to initiate a broadly based formal assessment of the need for and feasibility of fusion-fission energy systems. This decision resulted from the recognition that fusion may have real opportunities to provide solutions to problems of nuclear energy and also result in a nearer term benefit from an admittedly long term and expensive RD&D program. The assessment program is tentatively scheduled for completion in late 1980.

In addition to the assessment program, fusion-fission energy system design studies will continue with somewhat greater emphasis on the requirements imposed by advanced nuclear reactors and their associated fuel cycles. It is expected that these studies will face the chicken-egg dilemma: Will normally evolving fission reactor designs constrain the fusion-fission characteristics or will the flexibility offered by fusion-fission remove constraints on fission reactor design and deployment? Hybrid reactors and LWBR's could represent such a case.

The next major Fusion-Fission Energy Systems Review Meeting will be held at the completion of the assessment and the design studies presently under way. Interim information will be made available periodically at various national and topical meetings. Your continuing interest in the area of fusion-fission energy is appreciated.

F E Coffman

Franklin E. Coffman
Acting Assistant Director for
Development and Technology
Office of Fusion Energy

AGENDA

SECOND MFE FUSION-FISSION ENERGY SYSTEMS REVIEW MEETING

Wednesday, 2 November 1977

9:00	Opening Remarks - Edwin E. Kintner, Director, Division of Magnetic Fusion Energy	

PROBLEM CHARACTERIZATION

J. M. Williams, Session Chairperson

9:10	Uranium Availability	- J. Boyd (Materials Assoc.)
9:40	Global Proliferation Concerns	- R. Simkins (Dept. of State)
10:10	REFRESHMENTS	
10:30	Nuclear Fuel Cycles	- S. Strauch (Dept. of Energy)
11:00	Utility Perspectives	- P. Bos (Electric Power Research Institute)
11:30	Alternatives to Fusion-Fission Energy	- H. Kouts (Brookhaven National Laboratory)
NOON	LUNCH	

FUSION-FISSION ENERGY SYSTEMS

1:00	Generic Description of Fusion-Fission Energy Systems - S. L. Bogart, Session Chairperson	- L. Lidsky (Massachusetts Institute of Technology)
1:30	Present Status of Fusion-Fission Energy Systems Design - K. G. Moses, Session Chairperson	
	1:30 - Mirror Hybrids	- D. Bender (Lawrence Livermore Laboratory)
	2:00 - Tokamak Hybrids	- R. Rose (Westinghouse Electric Corporation)
	2:30 - Inertial Confinement Hybrids	- J. Maniscalco (Lawrence Livermore Laboratory)
3:00	REFRESHMENTS	

Wednesday, 2 November (Continued)

3:30	New Initiatives in Fusion-Fission Energy System Design - S. L. Bogart, Session Chairperson	
	3:30 - General Atomic Company/ Lawrence Livermore Laboratory	- S. Burnett (General Atomic Company)
	4:00 - Westinghouse Electric Corporation	- T. Varljen (Westinghouse Electric Corporation)
	4:30 - Lawrence Livermore Laboratory (Inertial Confinement)	- J. Maniscalco (Lawrence Livermore Laboratory)
5:00	ADJOURN	

Thursday, 3 November 1977

9:00	Economic Considerations - C. Head, Session Chairperson	
	9:00 - An Examination of Alternative Nuclear Breeding Methods	- B. Augenstein (RAND Corp.)
	9:30 - Economic Regimes	- D. Deonigi (Battelle Pacific Northwest Labs.)
10:00	REFRESHMENTS	
10:30	Technical Problem Areas - J. O. Neff, Session Chairperson	
	10:30 - Hybrid Blanket Design	- K. Schultz (General Atomic Company) J. D. Lee (Lawrence Livermore Laboratory)
	11:00 - Fusion Physics Requirements	- N. Krall (Science Applications, Inc.)
	11:30 - Environment and Safety	- J. Holdren (University of California, Berkeley)
NOON	LUNCH	
	1:00 - Tokamak Technology Requirements	- D. Steiner (Oak Ridge National Laboratory)
	1:30 - Mirror and Other Magnetic Confinement Technology Requirements	- R. Moir (Lawrence Livermore Laboratory)
	2:30 - Inertial Confinement Technology Requirements	- L. Booth (Los Alamos Scientific Laboratory)
3:00	General Discussion (5-10 minute limit)	- GROUP
4:00	ADJOURN	

	NAME	ORGANIZATION	COMMERCIAL TELEPHONE
1.	S. Locke Bogart	DOE/DMFE	301-353-5160
2.	John Clarke	DOE/DMFE	301-353-4940
3.	Mike Lotker	Booz Allen & Hamilton	301-951-2548
4.	Jim Maniscalco	LLL	415-447-1100 x4272
5.	Noel Amherd	EPRI	415-493-4800 x184
6.	Piet B. Bos	EPRI	415-493-4800 x431
7.	Karlheinz Thom	NASA	202-775-3066
8.	Ken Schultz	General Atomic	714-455-4168
9.	Duane Deonigi	PNL	509-946-2475
10.	Ronald Kostoff	DOE	301-353-4203
11.	Charles Maynard	Univ. of Wisconsin	608-263-3285
12.	John Hopps	CSDL/MIT	617-258-4270
13.	L. M. Lidsky	MIT	617-253-3808
14.	Ralph W. Moir	LLL	415-447-1000 x4720
15.	Frank Coffman	DOE/DMFE	301-353-5378
16.	Roy Simpkins	State/OES	202-632-4752
17.	Bob Aronstein	Bechtel Corp.	415-768-7890
18.	William O. Allen	Bechtel Corp.	415-768-7336
19.	J. D. Lee	LLL	415-447-1100 x4681
20.	B. W. Augenstein	Rand Corp.	213-393-0411
21.	David J. Bender	LLL	415-447-1100 x4679
22.	Eugene C. Gritton	Rand Corp.	213-393-0411
23.	Herbert Kouts	BNL	516-345-2918
24.	Fausto Gratton	FEC	609-452-9595
25.	James Treglio	FEC	609-452-9595
26.	Stephen C. Menasian	FEC	609-452-9595
27.	Bogdan Maglich	Fusion Energy Corp.	609-452-9595
28.	William E. Parkins	Atomics International	213-341-1000
29.	Ken Moses	DOE	301-353-4932
30.	Pete Rose	Math. Sciences NW	206-827-0460
31.	J. F. Decker	DOE	301-353-3421
32.	Gunnar Thornton	General Electric Co.	518-385-4327
33.	Roger S. Palmer	General Electric Co.	408-925-1000 x354
34.	Henry Hurwitz	General Electric Co.	518-385-8511
35.	Richard Buthmann	General Electric Co.	202-637-4415
36.	A. E. Toombs	Stone & Webster	301-986-1114
37.	M. A. Rosen	Stone & Webster	301-986-1114
38.	J. O. Neff	DOE/OFE	301-353-4955
39.	Erol Oktay	DOE/OFE	301-353-4928
40.	Donald S. Beard	DOE/OFE	301-353-4959
41.	Samuel D. Harkness	ANL	312-739-7781
42.	John H. Crowley	United Engineers & Construction	215-422-3816
43.	Neil F. Ritchey	United Engineers & Construction	215-422-3813
44.	Robert A. Olstad	General Atomic, Wash., D.C.	202-659-3140
45.	Robert E. Carter	Institute for Resource Management	301-656-1288
46.	Fred H. Tenney	Princeton Univ.	609-452-5650

	NAME	ORGANIZATION	COMMERCIAL TELEPHONE
47.	James R. Powell	Brookhaven National Laboratory	516-345-2440
48.	Meyer Steinberg	Brookhaven National Laboratory	516-345-3036
49.	A. E. Gileadi	KMS-Fusion, Inc.	313-769-8500
50.	Patrick K. Doherty	Combustion Engineering	203-688-1911 x5876
51.	Robert W. Goodrich	Northeast Utilities	203-666-6911 x5159
52.	Alan F. Haught	United Technologies Res. Center	203-565-6789
53.	Gregory A. Moses	Univ. of Wisconsin	608-263-3368
54.	P. Nicholson	CSDL	617-258-1394
55.	Ernie Braunschweig	DOE/DLF	301-353-4938
56.	Richard Schriever	DOE/DLF	301-353-3463
57.	Martin Stickley	DOE/DLF	301-353-3462
58.	L. A. Booth	LASL	505-667-5003
59.	R. A. Krakowski	LASL	505-667-5863
60.	Gene L. Woodruff	U. of Washington	206-543-8374
61.	Fred L. Ribe	U. of Washington	206-543-2754
62.	F. C. Schwenk	NASA	202-755-2488
63.	E. C. Fiss	Duke Power Company	
64.	K. H. Puechl	Combustion Engineering	203-688-1911
65.	Frank Graham	Atomic Industrial	301-654-9260
66.	R. A. Huse	PSE&G Research Corp.	
67.	G. M. Halpern	Exxon Research & Eng. Co.	201-474-2010
68.	D. H. Berwald	Exxon Research & Eng. Co.	201-474-2011
69.	C. E. Taylor	LLL	414-447-1100
70.	C. R. Head	DOE/DMFE	301-353-4953
71.	J. E. Baublitz	DOE/DMFE	301-353-4956
72.	T. C. Varljen	Westinghouse Fusion Power Systems	412-892-5600
73.	D. Green	Westinghouse Fusion Power Systems	412-892-5600
74.	R. P. Rose	Westinghouse Fusion Power Systems	412-892-5600
75.	R. Price	DOE/DMFE	301-353-3287
76.	V. Nardi	Stevens Institute of Technology	201-792-2400
77.	T. J. Kurey	General Electric	518-385-7107
78.	B. R. Leonard, Jr.	Battelle-Pacific Northwest Lab	509-946-2558
79.	W. C. Wolkenhauer	Washington Public Power Supply	509-375-5389
80.	V. L. Teofilo	Battelle-Pacific Northwest Lab	509-946-2159
81.	J. Nelson Grace	DOE/DMFE	301-353-3068
82.	Z. M. Shapiro	Westinghouse Electric Corp.	412-373-4300
83.	D. Klein	Westinghouse Electric Corp.	412-892-5600 x5408
84.	David Green	Westinghouse Electric Corp.	412-892-5600
85.	Saul Strauch	DOE	301-353-4366
86.	Betty K. Jensen	Public Service Electric & Gas	201-430-6633
87.	John G. Gilligan	University of Illinois	217-333-7710
88.	Paul J. Nicholson	C. Stark Draper Laboratory	617-258-1394
89.	T. A. Parish	Univ. of Texas	512-471-5136
90.	John P. Holdren	Univ. of California	415-642-1139
91.	T.V. George	DOE/DMFE	301-353-4926
92.	N. A. Krall	SAI	714-459-0211
93.	George K. Hess,Jr.	DOE/DMFE	301-353-4936

OPENING REMARKS

John F. Clarke
Deputy Director

Office of Fusion Energy

Good morning. Welcome to the meeting. I look forward to hearing the papers that will be presented here because, as I look around the audience, I see unfamiliar faces. Having been in the fusion business for about 17 years, it is always a surprise to attend a fusion meeting and see faces that I don't recognize.

The reason for it is obvious. We have people in the audience that represent a variety of disciplines: people from the fission community, people from the fusion community, even people from a community that is concerned with the social impacts of the technologies of both fission and fusion. I think it is very good to get this combination of experts together, to discuss a subject about which we know very little.

I thought I would give you a little bit of personal historical background on the subject of the meeting, fission-fusion or fusion-fission, depending on which side of the game you come from.

There has always been a certain reluctance in the fusion community to take this kind of a discussion seriously. That is because most of the people in the fusion community look on a discussion of fission-fusion as an example of the two-stone theory.

I am sure that you are all familiar with the two-stone theory. The two-stone theory has its origin in a story about a man who was drowning, and a good soul who was walking along the shore and saw the struggle. The passerby threw the drowning man the only thing at hand, a large rock. This didn't help very much. So another good Samaritan walking along the shore threw the swimmer another rock. And that is the two-stone theory; namely, if the first stone doesn't seem to do any good, maybe a second one will help.

Now, in the past the fusion program has been through many ups and downs. There were times when we seemed to be drowning in a sea of plasma physics difficulties. At the same time we also had a large rock tied to us. That was the rock of the engineering difficulty of trying to implement a fusion system, even if we were successful in the plasma physics. So there we were, struggling and drowning with problems in both of these areas, and there was always a good Samaritan standing on the sideline who was very willing to throw us another rock, fissile material, which was going to solve all of our problems.

Obviously, when the program was in great difficulties with plasma physics and problems of engineering feasibility, this kind of added complication was not welcomed.

This meeting is a historic one because it occurs at a time when the program is not in trouble. The fusion program has never enjoyed more success. The plasma physics, in both inertial and magnetic confinement, seems to be working out as well as we could have expected.

The engineering progress, in terms of satisfying the technical requirements for fusion, has been very impressive. We have made tremendous progress in the reactor designs. We have reduced reactor designs to a size where an engineer can look at them and not begin to tremble.

So the program is successful, and here we are having a meeting, talking about that second rock of fissile material. In the base of this success, the question arises why? Why do we bother with the second rock?

The reason we bother with the second rock is that we are approaching the point where we can begin to generate energy with fusion. We can begin to appreciate, in fact, that in some cases we have in hand the technology that will enable us to produce power.

Now, given this atmosphere of success and progress in the program, it becomes possible for us to take seriously a broader aspect of fusion application in response to national needs. In the case of fissile fuel production, the rock of yesterday may become the cornerstone of a future contribution to the nation.

We are spending a fair amount of money in the coming year, and we will spend more money in the future to look at the broader aspects of fusion. We are attempting to answer the question; given that one can generate fusion power, what does one do with it? Traditionally, we have always thought about generating electricity. This was the drive of the program from the very earliest days. But when you think about it, that is a very primitive use of this marvelous energy source. We throw away two-thirds of the energy generated.

Isn't there something else that can be done with this energy source that would begin to attack some of the broader questions that plague our society today?

These are the questions we are beginning to examine. The potential is there. The preliminary studies in both this area of fission-fusion and some of the other alternate application areas such as the generation of hydrogen from high temperature reactions of water look very promising. In this meeting, we are trying to put some meat on the bones of those conceptual analyses that have been carried out to see what are the real potentials of fusion-fission.

We think, in our hearts, based on preliminary analysis in a number of areas, that fusion can attack a lot of the problems that our society faces, not tomorrow, not ten years from now, but with increasing probability as we advance into the future.

So I wish you luck in your deliberations, and I would request one thing of you. Since people from different fields and different backgrounds are gathered here, and we are discussing a subject about which we have very little detailed knowledge, I ask you for honesty. Let your masks down a little bit to admit ignorance. If you don't understand what the other person is talking about, ask, because you are probably not the only one in the room who does not understand.

With this attitude, this meeting can produce some useful results and, in fact, could set the course of the research over the next few years in this area of alternate applications.

United States Uranium Position **

JAMES BOYD L. T. SILVER

ABSTRACT

For the immediate future the magnitude of uranium reserves are far more important than the ultimate size of the resources which may eventually be found. The rate at which new uranium reserves can be found and equipped to feed the lightwater reactors is a vital factor in the transition to more advanced conversion systems such as the Liquid Metal Breeder. The breeders require substantial accumulations of recycled plutonium and uranium in their inventories before they can sustain their growth from the recycling of their own wastes.

INTRODUCTION

If we follow the most technologically advanced approach to nuclear power development, there could be ample uranium available to meet the world's needs for many decades. If, however, we continue to pursue the current approach, there is little possibility of finding enough uranium, fast enough, to permit nuclear power to make a significant energy contribution in this century.

Background

In any discussion of our uranium position, it is well to recall that the technology of nuclear engineering is only about thirty-five years old, that the peaceful applications of nuclear science are still in their infancy, and that only within the last three decades have our colleges and universities been in a position to supply the nation with scientifically trained nuclear specialists. Furthermore, because nuclear technology had its birth as a destructive and catastropic weapon of war, and because of the inherent problems of safety, development of nuclear plants has been slow and subject to controls that insure high capital construction costs. With this background, we can begin to evaluate the United States uranium position in a realistic manner while recognizing that nuclear science is still only in its infancy as earlier stated.

** Published first by the American Society of Mechanical Engineers (77-JPGC-NE-25)

The economic effectiveness of uranium as an energy resource to the United States is closely related to all available energy sources, and this is true throughout the rest of the world as well. The extent to which uranium will be used, as well as the time involved in its application, is a multifaceted function of technological development in addition to discovery rate for uranium ores, conservation measures, the nature of environmental decisions, other regulatory actions, and international agreements. Basic to all evaluations of uranium utilization is accurate knowledge of existing reserves, and sound estimates of their composition, magnitude, and ultimate availability.

Technological advance is a factor of resource availability, as indicated by the recent development of an improved system to recycle uranium and plutonium, which will extend the U.S. reserves of uranium by about 25%. A fully established modern conversion system, such as the LMBR (liquid metal breeder reactor), can multiply such reserves 60 to 70 fold.

There is, however, a caveat that today cannot be ignored: decisions designed primarily to enhance or preserve the environment retard the rate at which coal will replace oil and gas, for coal production will increase only as fast as the market for it expands.

Political actions based on such decisions further affect the rate by which nuclear energy will supplement the fossil fuels. An example is the present U.S. position on not permitting fuel reprocessing.

THE CONTROLLING ISSUES

Resources and Reserves Defined

Because questions of resource magnitude appear frequently in public debate, we shall start with a discussion of this subject. Those reserves of mineral raw materials that are currently known and economically producible, and those yet to be found, delineated, or made economic by technological advances and relative increases in prices, embrace our field of discussion. Failure to define clearly the terms that we use, only too often causes endless confusion. Under such conditions, it is not surprising that our comprehension of resources problems is incomplete.

A clear definition of the term <u>reserve</u> should initiate any resource discussion because it has been a controversial topic for argument within geological and mining engineering circles for decades. Fortunately we now have a sound basis for such discussion and analysis: Not long ago a memorandum issued jointly by the United States Bureau of Mines and Geological Survey provided a simple definition that establishes a sound starting point for the analysis of any mineral resource, including uranium. Simply stated it says that <u>reserves</u> are those portions of the total resources of any mineral commodity that have been delineated by adequate exposure at the surface, or in drill holes, or excavations available under present technology and current economic and political conditions. The words <u>adequately</u> and <u>economically</u> may be stretched a little at times, but not beyond a realistic or logical limit. All other potential resources then become speculative; no development of sophisticated economic theory can change that. In arriving at sound resource estimates or conclusions for decision making, it may be essential to include some degree of speculation, but the odds themselves must be recognized as speculative also.

Limits to Available Resources

The best data available to us, and derived from our participation in some studies being conducted in the National Academy of Sciences, indicates that U.S. uranium reserves under that definition do not exceed 640,000 short tons of U_3O_8. Production difficulties, environmental obstacles, and safety precautions could result in as little as 500,000 tons of U_3O_8 of those being ultimately produced, a recovery of only 70% in mining terms. Because, however, our present understanding of the way in which past geologic events have concentrated uranium in the earth's crust, these same sources conclude that there is small possibility that the total uranium resources of the U.S. will exceed 3,800,000 tons. Although there are some digressions from this figure, it was deduced by reputable professional geologists in the Federal Government, in universities and/or engineering schools, etc., and confirmed through independent studies by several current uranium producers.

Our figures for the rest of the world were obtained from the International Atomic Energy Association and generally corresponded in magnitude to those of our consultants with world-wide experience. These figures do not differ greatly from the Energy Research and Development Administration's (ERDA) published figures, except as to the limits of speculative resources. ERDA released in April 1977, data from a more recent audit of their figures confirming the order of magnitude of these estimates and these we will use throughout this paper.

Discovery Rate

At this stage in the development of the world-wide nuclear power industry, the ultimate magnitude of total resources is secondary to the rate at which resources can be discovered and then developed into reserves and facilities provided to supply uranium in the form used in the power industry.

It took the U.S. 30 years of intensive effort to get the production of U_3O_8 up to 13,000 tons in 1976, although there were periods of slower development. To satisfy the requirements calculated by ERDA for its case most frequently used for planning (their Mid Case) would require a discovery rate quadruple that achieved over any extended period in the past. The discovery rate involves the number of existing deposits within the resource potential that can be actually found and exposed. But this rate is also closely related to expansion of milling and concentrating facilities to accommodate development and mine production from both new and old sources. It is, therefore, the judgment of our colleagues that if the U.S. follows its present policies of pollution control, safety, and land managment, etc., the rate of U_3O_8 produced will reach the approximate levels of 34,300 tons in the year 2000. In the unlikely event that the most extreme of these policies were modified, and the country dedicated itself to finding and producing uranium, production could be increased to about 81,000 tons in 2000. A few modifications in policy which, will undoubtedly take place when the urgency of the energy situation penetrates the public consciousness, could result in the following schedule, Table I.

Although there are more optimistic figures available, we personally feel that the figures here are more likely to be optimistic than the otherwise. We believe, however, that these figures should be used for prudent policy planning purposes.

These and the reserve figures are necessarily based on the ERDA published data; the basic detailed deposit-by-deposit information is proprietary and not available to anyone but the ERDA staff. The procedures for gathering these data are sound and the results

TABLE I. Annual Uranium Production Projections Under Modified Policies

Year	Convent. Mining Annual Tons	Unconvent.* Sources Annual Tons	Combined Production Tons
1980	17,000	2,600	19,600
1985	26,000	5,100	31,100
1990	38,500	5,500	44,000
1995	53,000	5,500	58,500
2000	61,000	5,500	66,500

* Solution recovery at copper and phosphate mining operations.

should be more accurate than those gathered on any other mineral resource, and it seems wise, therefore, to accept their basic figures derived from this mass of detail. Differences of opinion may still result from variations in precision or from defining the terms used for resources until they are fully identified which, by their very nature, become difficult to interpret or measure. The greatest divergence of opinion embraces reserve discovery rates and the anticipated ultimate recovery to be achieved.

In assembling the above figures, our colleagues depended upon historical data and the experienced judgments of those who must develop reserves, plan for mining, and for facilities to process the ores produced. As an aside, the dangers of misjudgment are too great to depend entirely on theory or speculation, but theoretical modelling is a valuable tool and we are justified in using it to arrive at reasoned judgments.

Supply-Requirement Balance

Only the most optimistic of these production rates approach the calculated requirements of even the present schedule of light water reactors which has progressively declined over the past few years. Private utilities are not likely to commit themselves to finance and construct new plants until they have some measure of assurance that uranium will be available. Recent lawsuits filed in the United States have added caution to expansions, and already some plants have been deferred by this consideration. ERDA is concerned because uranium inventories have been rising. This, and because there is sufficient production to meet the schedules of existing enrichment plants, tends to dull the sense of urgency for uranium exploration.

The early history of delayed development of the atomic electric power industry, concurrent with varying military requirements for uranium, resulted in intermittent requirements for U_3O_8 production, and industry reactions to the uncertainties of the period have carried over to the present day. Now we are faced with a world-wide rapid expansion of the nuclear power industry, a serious decline in the availability of oil and gas in the United States from its own resources, inevitable decline in world oil and gas production in the next few years, and a very large expansion of the coal industry. Exotic sources of energy, such as solar power, fusion, and more effective use of geothermal energy sources are relatively far in the future. Current limitations embrace limits to the availability and utilization of the vast but finite reserves of coal, as well as our technological shortcomings and inhibitions in the use of atomic energy.

This is where the difference between discovered reserves and potential resources is evident because it is all very well to speculate on what may be found within the next few years, but to commit extensive capital for the expansion of the present generation of power plants, without an assured supply of uranium, appears to be another matter. Furthermore, this also applies to world capital centers where there would be no inclination towards extensive financing for such purposes without some assurance that the supply of the basic fuels is available and forthcoming.

THE PHASING INTO THE NEXT GENERATION OF POWER CONVERTERS

There are a number of theories about where the next generation of power converters will come from, but the only major developments currently ready for expansion are the liquid metal breeder reactors and the Canadian Candu system although there are problems involved in Candu which are not likely to satisfy the regulatory authorities in the United States. High temperature gas reactors, which would employ a certain amount of thorium, still require a basic charge of uranium. It does not seem likely that thorium will become a major factor in energy supply for several years. We must still depend upon uranium and the recycling of those wastes which are now being produced and will become available from the present generation of light water reactors.

At the present time, the rest of the world largely depends upon availability of enriched uranium from the gaseous diffusion plants in the United States. These plants in turn depended upon U.S. resources supplemented by imports of raw uranium oxide from outside. There are growing requirements for uranium for power development throughout the world; it can no longer be expected that much surplus uranium will be available to supply the United States nuclear power industry, even though our neighbor Canada will have surpluses to their needs. And the requirements of U.S. industry will leave little to be exported, unless there is uranium raw material imported on toll to be fed into the U.S. enrichment facilities.

Phasing advanced nuclear converters into the United States power industry is therefore one of the most perplexing policy problems facing the United States. The President announced in April 1977 that the United States would depend upon the light water reactor, at least for the time being, and that the United States would not engage in recycling of plutonium from existing power plants. The efficacy of this program is an important consideration in the United States uranium position, and in taking this action, the United States Government has relied upon estimates of uranium resources which are different from the minimum figures we gave you earlier in this paper. The Federal Government, for example, stated that there is ample uranium to run the present schedule of light water reactors (LWRs) for about 75 years, even though the present fuel supplies are not recycled after removal from the power plants following initial burn. Unfortunately, this conclusion is based upon the assumption that some of the reserve figures which have been published by the United States Government are indeed assured reserves, and that the uranium produced from them will be as high as the 640,000 tons which is ERDA's latest figure. This conclusion also depends upon turning more presently undifferentiated resources into reserves and fuel element supplies. As noted previously, it is possible that uranium production will not exceed 450,000 tons.

The U_3O_8 requirements for the present plants now operating, under construction and approved for construction are as shown in Table II.

TABLE II. Annual Natural Uranium Requirements for the Domestic 208,000 MWE Under ERDA Contracts Compared to Projected Domestic Production for 1980, 1985, and 1990

	Thousands of Tons U_3O_8	
Calendar Year	Projected Production	Annual Requirements (Without U or PU Recycle) (Start 0.25% tails assay on October 1, 1978)
1980	19.6	31.25
1985	31.1	43.2
1990	44.0	45.5

We are plotting these annual requirements against the estimates of production from Table I, which are estimates of what could be produced from current reserves, and judgments of discovery rates under present conditions. To the extent that they will come from resources excluded in the generally agreed 640,000 tons of reserves, the production rate includes a degree of speculation increasing with time. After listening to hours of debate and repeated arguments, it is our opinion that these production estimates are reasonable. This would indicate that the uranium requirements for the presently approved programs could not be met for the life of the plants. These plants require from 5,000 to 8,000 tons of U_3O_8 per installed thousand megawatts, or gigawatt, for their lifetime. The present 127 gigawatts of LWR's installed or being installed will therefore require at least 635,000 tons, approaching the limits Table I indicated to be available from recoverable reserves.

Even if the United States would dedicate itself to nuclear power, and remove the serious roadblocks to the search for and production of uranium, there would be barely sufficient time to meet the requirements of present scheduled plants.

These figures have not yet been adopted in governing circles, or at least not publicly. As soon as they have, it will be obviously necessary to reconsider recycling policy and the rate of advanced converter development. It will take many years, certainly well into the next century, to build the initial inventory of uranium and plutonium for LMBRs if they are to be the next generation of nuclear reactors. The initial inventories must come as byproduct from the LWRs until there are a sufficient number of breeders to supply themselves. After that date, there will be large quantities of waste materials to feed the recycling plants required to meet the schedule, and the need for virgin uranium will decline rapidly, a fortunate condition because uranium production will also decline rapidly, following the discovery rate decline by 10 years.

Production Rates

Figure I, discovery rates against time, produces the typical bell-shaped curves of production from a resource of finite size. This nest of three curves shows the potential discovery rate under present regulatory conditions (I), all-out dedication to finding uranium (III), and a more likely condition when the problem is taken more serious (II). As it takes about 6-10 years from discovery to delivery to the processing plants, the availability could be represented by a bell-shaped curve shifted by ten years.

If extensive exploration proved the existence of unimagined sources of uranium, the left side of the discovery curve would not alter much, but the peak would be higher and later as illustrated by the partial dashed curves. We have plotted them only for illustrative purposes. We and our associates believe that the solid curves represent the most likely outcome.

CONCLUSION

The time between the decision to mine and equip a given unit of uranium reserves, to the time that it will be available to a power plant, is from 10 to 15 years. The time in which that uranium is in residence in a LWR before the derived plutonium and U_{235} are recaptured for further use is on the order of 10 more years. It will take several years to reverse the present prohibition on recycling and reactivate or build the required plants. It appears that unless drastic action is taken, there is not sufficient uranium forthcoming to sustain a nuclear industry based upon lightwater reactors. As a consequence, it is imperative that we extend our uranium resources and reserves by developing and commercializing some sort of advanced converter as soon as possible. The present state of the art would indicate that this will have to be the Liquid Metal Breeder Reactor or the advanced system the Canadians have adopted called CANDU.

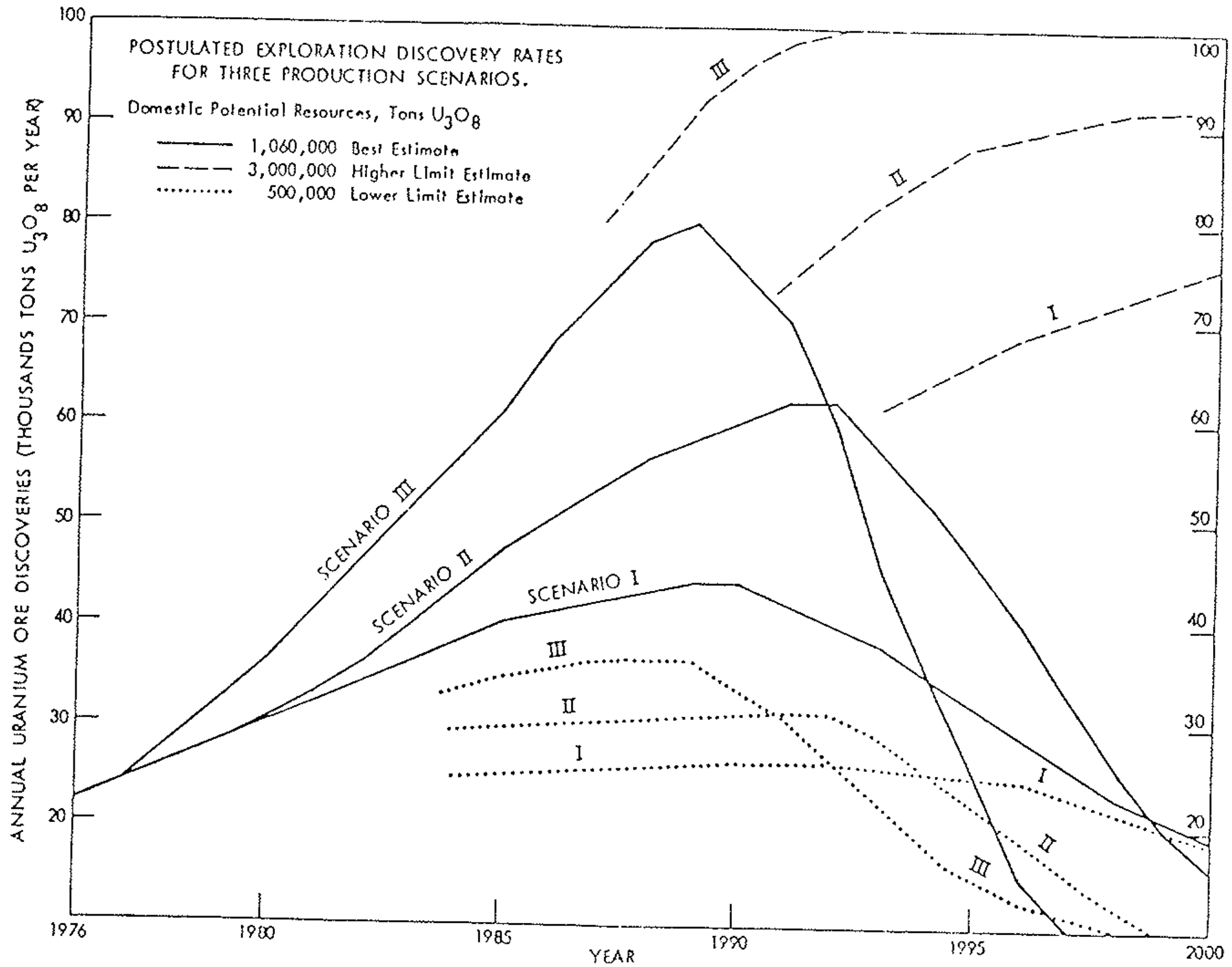

POSTULATED EXPLORATION DISCOVERY RATES
FOR THREE PRODUCTION SCENARIOS.
Domestic Potential Resources, Tons U_3O_8
1,060,000 Best Estimate
3,000,000 Higher Limit Estimate
500,000 Lower Limit Estimate
SCENARIO III
SCENARIO II
SCENARIO I
III
II
I
ANNUAL URANIUM ORE DISCOVERIES (THOUSANDS TONS U_3O_8 PER YEAR)
100
90
80
70
60
50
40
30
20
1976
1980
1985
1990
1995
2000
YEAR

FIGURE 1

MR. WILLIAMS: Thank you very much. Do we have questions?

MR. LOTKER: Mike Lotker, Booz-Allen & Hamilton. This slide in particular, and some of the other comments you made, seems to indicate that there is a near-term uranium production shortage in the 1980-1985 period. It seems that with the 10-year lag you referred to, the problem would be clearing up around the '90's, and then the slide that you described would indicate a problem later on?

DR. BOYD: Yes.

MR. LOTKER: Does this mean that we have this early '80's problem which the LMFBR or any breeding technology can't really help, that in the '90's we seem to have enough uranium even without the breeder, and beyond we need some kind of breeding technology?

DR. BOYD: That is really what I am saying to you. What it means is simply this: That at the moment, DOE has inventory building up in uranium, waiting behind the gas diffusion plants. They have got to build some more plants to take care of it.

The requirements of a light water reactor is about between 6,000 and 8,000 tons for the lifetime of the plant. So there isn't enough uranium in the reserve picture to meet the lifetime requirements of the present light water reactors under construction or already approved.

They will take up all the uranium we can see coming into the reserves at the rate of discovery we talk about. Therefore, it is vital that we get after the next conversion system, and the only one you have got far enough along, really, is the liquid metal breeder reactor.

So that is what I am saying to you. We have to do this if we are going to have a viable atomic energy power industry.

DR. MANISCALCO: Jim Maniscalco, from Lawrence Livermore Laboratory.

Does the situation in the early '80's change much with recycling if you can do it? What is the effect?

DR. BOYD: Recycling will improve that situation by about 25 percent. If you can recycle both plutonium and uranium, the reserves increase by about 25 percent.

DR. MANISCALCO: Jim Maniscalco again. Then, if recycling improves it by about 25 percent and you introduce the fast breeder reactor 10 or 15 years down the road, won't LWR's lose the benefits of recycling?

DR. BOYD: No, no. The liquid metal breeder reactor is a recycling process in itself, and your wastes from that have to be reprocessed to feed into the systems, so it is a recycling.

DR. MANISCALCO: But the recycled fuel is taken away from the existing light water reactor? That is what I am getting at.

DR. BOYD: The wastes from those reactors are now being put in storage. They are not being recycled. Now, you need the recycled fuel to build the inventories in the liquid metal reactors.

DR. MANISCALCO: Okay. But the point is, the LMFBR uses recycled fuel in the same period of time, ('85 to '90) that we are going to need in the light water reactors?

DR. BOYD: That's right.

DR. MANISCALCO: And if you take the recycled fuel away from the LWR, you are back to the near term shortage situation that results from not recycling.

DR. BOYD: I don't think I understand that. You don't take the recycled fuel away if you go to recycling.

DR. MANISCALCO: If you are going to use the recycled fuel, to build enough inventories for your future LMFBRs you will take it away from the LWR's.

DR. BOYD: Well, I am not sufficiently familiar with that changeover to be able to answer the question specifically, but those who have worked with it are working up a schedule showing the rate at which you need to feed back into the light water reactors and their rate at which you are required to build up the inventories for the breeders. I can't give you specific figures on it.

DR. COFFMAN: Frank Coffman, of the Department of Energy.

Regarding the Chattanooga shales, would you comment upon their impact on the resource question and on the feasibility and costs of extraction?

DR. BOYD: Well, I won't try to give you the exact costs because no one really knows. At this moment we are concerned with the environmental concerns, the cost of mining, the disposal and recapturing the land. At the moment it looks as if we would use more energy than we would create by mining even with the breeder reactors. They are a long way off.

MR. WILLIAMS: There is time for one more question.

DR. WOODRUFF: I am Gene Woodruff, University of Washington.

If I understand your numbers, your projections of the ultimate resources are considerably below those predicted in the recent Ford Study which has had quite an impact upon the Administration.

DR. BOYD: That is right.

DR. WOODRUFF: Is there a simple explanation for this?

DR. BOYD: Well, the 10 million tons you are referring to, that was in the Ford Study, was based on economic calculations of experience in discovering materials in other fields, and there is no relationship between uranium and say, copper, lead, or zinc.

We are limited to a resource which has a specific geological history in the surface of the earth and which is concentrated by surface solutions. You cannot apply that same system of projecting resources that you could in metals, that they have had experience with.

We find that there is no geological evidence whatever at the moment. Now, we all hope, as experts and geologists, we will find evidence, but we don't know of any. And we don't think it is safe to count on resources for which there are no existing geological theories. These figures I gave you came from geologists and experienced producers.

MR. WILLIAMS: One last question.

DR. ROSE: Pete Rose, Math Science.

In your numbers, you made the point between reserves and resources, and the reserves were based on known techniques and known deposits, and the resources were unconventional techniques and unconventional--

DR. BOYD: No, undiscovered.

DR. ROSE: Undiscovered and possibly unconventional techniques. Yet, when you put up your projection of production, your conventional techniques tripled between 1980 and the year 2000.

DR. BOYD: Our production figures.

DR. ROSE: And your unconventional techniques at least according to your number, which should have a much larger reserve to dip into, only increase from 2,500 tons to 5000 tons and then stayed constant. I wonder what is hidden behind that.

DR. BOYD: You have got two meanings of "unconventional." The unconventional in that context are not the sandstone deposits which are producing the uranium today, but the by-products from copper production, and phosphate production, and so forth. They depend upon the production of copper and phosphates, and they won't grow that fast.

Now, the unconventional things you have in mind are the vein deposits which seem to be discovered in Africa and some in Canada, and what people hope will be another source of uranium beyond the sandstones.

Those are the two meanings of the word "unconventional," and in these figures, we are referring only to the by-product production and not to those which are the mining of uranium from sandstones. I am sorry, that is a confusing point.

GLOBAL PROLIFERATION CONCERNS

Delivered By Roy Simpkins to the Second DMFE
Fission Fusion Energy Systems Review
Meeting on November 2, 1977

How to make nuclear power available to meet world energy needs without simultaneously accelerating the spread of nuclear weapons capabilities is a global problem. Nuclear technology is no longer--if it ever was--the monopoly of one nation, nor even of a small group of nations.

As you know, President Carter has from the start made nuclear non-proliferation one of his top priorities out of a deep conviction of its importance for present and future generations.

Why does nuclear proliferation deserve such a high priority? Because a multi-proliferated world--a world with many nuclear weapons powers--will be a far less stable world for all nations to live in. We are well aware of the possible outbreaks of war in unstable regions; of overthrown governments and civil wars in unstable countries; and of the potential for damage by terrorist groups. If we imagine easy access to nuclear explosives being added to the existing sources of instability, the picture of the world we envisage is not a pleasant one. Proliferation of nuclear explosive capabilities to an increasing number of countries and transnational terrorist groups would carry with it an inordinate peril to ourselves and to the world. It would reduce our ability to control international crises and have a seriously detrimental effect on our alliances, exposing our nation to grave risks. It would greatly increase the danger of catastrophic nuclear war.

Our goal, then, is to limit the number of nations with nuclear explosive capabilities. How can we do it? First let's examine the international framework. The non-proliferation problem is made more manageable by the existence of two very unique international devices - the Non-Proliferation Treaty (NPT) and the IAEA Safeguards System.

The Non-Proliferation Treaty was signed in 1968 and came into force in 1970. One hundred and two nations have now ratified the Treaty in which non-weapons states agree not to acquire nuclear explosive devices. The NPT has helped to create an international regime in which states agree that their security interests would be better served by avoiding the further spread of the bomb. The Treaty provides reassurances that potential adversaries are confining their nuclear activities to peaceful purposes and that, in the event of diversion to military purposes, the safeguards system provided for by the Treaty would give timely warning of any such cheating. Because it is an indispensable framework for effective non-proliferation efforts, the United States continues to seek the widest possible adherence to the Treaty.

The Non-Proliferation Treaty is a delicate international arrangement. Countries without nuclear weapons have accepted an explicity unequal status in the military area, on the condition that they be treated equally with regard to civil nuclear cooperation.

The basis for the NPT is the system of international safeguards administered by the International Atomic Energy Agency, or "IAEA," an independent United Nations agency that was established in Vienna in 1957. Under the IAEA safeguards systems, countries must file regular detailed reports on their civilian nuclear activities with the Agency, and must allow international inspectors to visit their nuclear facilities to verify the reports and to ensure that there has been no diversion of materials from civilian to military purposes. Underlying the safeguards system is a basic bargain in which we assist other countries in their nuclear energy needs in return for their accepting the intrusion of safeguards into their sovereignty.

But since 1974, we have had doubts about whether this safeguards policy that had worked for two decades would continue to work in the future. This reassessment was triggered partly by the Indian explosion of what they termed a "peaceful nuclear device," and partly by the substantial rise in oil prices. These increased oil prices led to a great increase in the projected demand for nuclear energy which led people to believe that there would be a shortage of uranium and that therefore we would have to move more quickly from a uranium economy to plutonium economy. More countries began to desire their own enrichment and reprocessing facilities and to think in terms of breeder reactors.

The problem with this spread of sensitive enrichment and reprocessing facilities, particularly the change of technological generations--as the world considers moving from low enriched uranium technology to a plutonium technology --is that the new plutonium technology threatens to discredit the political significance of safeguards. In other words, the key aspect of safeguards, the key dimension that has made the system workable for the previous two decades, threatens to be eroded by the change of technology. We will be faced with stockpiles of pure plutonium as well as the flow of fuel from which plutonium is easily chemically separable. Thus, countries would be closer to the threshold of nuclear weapons capability. This evolution would leave less time for diplomacy to work in cases where intentions are volatile.

Technologies often reflect the conditions prevalent at the time of their adoption. For example, the objective of reprocessing some thirty years ago was to derive plutonium in as pure a form as possible in order to make a nuclear weapon. Thus we selected a reprocessing method that was effective considering the overriding security concerns of the period. But the Purex Process is not the only method. Our times and social needs have changed. Today, we are more concerned about non-proliferation, and must look again at alternative technologies that may have been rejected as suboptimal in the past but which may today be preferable.

The nuclear industry has heretofore proceeded on the assumption that reprocessing would begin when there were sufficient light water reactors to justify the large-scale facilities needed for economic operation, and that plutonium would be recycled in light water reactors until fast breeder reactors were introduced. Other nations without our fossil fuel and natural uranium resources are even more strongly wedded to the belief that reprocessing would be needed to reduce long-term risks from nuclear wastes and that plutonium stockpiles would be needed at an early date to achieve energy independence through the use of breeder reactors.

But a plutonium economy based on the spread of national Purex reprocessing plants would challenge the very essence of the international safeguards system that has served us thus far. The "timely warning" function of the present safeguards system would all but vanish in the event of diversion of nuclear materials from peaceful to military purposes.

Let me point out that no one is assuming that the commercial fuel cycle is the only path or even the best path to a nuclear weapon. If a country clearly started out to get a bomb, there are technical reasons why it would be better to build facilities dedicated to military purposes. What we are assuming, however, is that in situations of extreme tension states may turn to second or third best instruments to get their hands on weapons they regard as essential to their security.

If the US were the only country with nuclear technology, this problem of managing the change of nuclear generations would be difficult enough. But there are already some 20 countries with nuclear reactor programs, and at least five other countries with advanced breeder reactor programs. Our strategy cannot rest upon managing our own affairs or merely setting a good example, but will require diplomatic efforts. For one thing, other countries point out that we have the coal and uranium resources that allow us to afford such an example while they do not. This means our example alone is not compelling to them. Neither can our strategy be based on passing domestic laws that prevent nuclear exports from the US, for other countries could quickly step in to fill the order books.

With all this in mind it's time to detail President Carter's program. The President in his statement of April 7 first publicly outlined his non-proliferation strategy - noting that we require:

1) A major change in US Domestic programs and

2) A concentrated effort among all nations to find better answers to the risk/benefit dilemma of nuclear power.

On April 7, he announced specifically:

-- The deferral of commercial reprocessing and recycle

-- A restructuring of the breeder program

-- Funding of R&D to develop alternative fuel cycles not involving direct access to weapons usable material

-- An increase in US uranium enrichment capability, and

-- Legislation to provide for fuel assurance for consumer nations with our proliferation concern

-- The continuation of the embargo of sensitive technology and equipment and lastly,

-- INFCE, the International Nuclear Fuel Cycle Evaluation Program.

Time is short and in my opinion INFCE is the most relevant topic to this technical group. Thus I will discuss in detail the International Nuclear Fuel Cycle Evaluation Program.

The organizing meeting was held, as you probably know, in Washington, October 19 through 21, to kick off the 2-year INFCE study. Some 40 countries and four international organizations with a major interest in nuclear energy attended. We deliberately invited both consumers and suppliers, rich and poor, east and west. The purpose of the 2-year program is to evaluate scientifically various aspects of the fuel cycle, and to develop agreed data based upon which a future consensus might be built. Participation in the program does not commit a country. INFCE is not a permanent international organization. There will be no votes. The approved terms of reference authorized eight multinational work groups, each of which will study an important segment of the problem in our effort to strike a balance between the benefits of nuclear energy and its proliferation risks.

The first two work groups deal with natural resources and enrichment capacity. If the facts support our view that uranium and thorium resources are more plentiful than is commonly believed, we can extend the lifetime of the current generation nuclear reactor. To the extent that adequate uranium and enrichment capacity are available to consumers to meet legitimate energy needs, the less time pressure there will be to move to next generation of fuel cycles before we have examined their proliferation risks. At the same time, we realize that it is not enough merely to prove the existence of sufficient uranium, thorium and enrichment. We must also establish an international system of assured fuel supply. That is why the third working group specifically addresses ways to assure supplies to consumer countries.

The fourth chapter, reprocessing, examines the economic and proliferation implications attendent to various reprocessing alternatives. We in the United States are especially interested in reprocessing techniques that avoid pure plutonium. At the same time, however, the evaluation will also explore the feasibility of technical and international institutional means of increasing the safeguardability of conventional fuel reprocessing. Similarly, the fifth working group, which deals with breeder alternatives, focuses on whether there may be systems which are economical and which minimize the presence of weapons-usable material.

The sixth and seventh work groups examine problems associated with spent fuel and waste disposal. Clearly, the degree to which we can alleviate current storage problems will directly affect the lifetime of current generation reactors. These storage problems are one of the pressures toward reprocessing and plutonium recycle. We also believe that scientific evidence can be brought to bear on the conflicting claims that reprocessing enhances or worsens the environmental risks involved in nuclear waste management. For our part, we are studying both domestic and international solutions which can be of help in dealing with this problem.

The eighth and last work group will look at ways to increase the fuel utilization in present thermal reactors. There is credible evidence that we may be able to double the utilization rate through various techniques. Obviously, this would be like discovering twice as many uranium mines. Again, the longer the lifetime of the current fuel cycle, the more time we have to design more proliferation resistant future fuel cycles. Finally, the eighth chapter will also look at advanced converter reactors and other reactor and fuel cycle concepts, which could increase fuel resources without providing access to weapons-grade material. We will look at alternative concepts not adequately studied in the past, although in many cases substantially developed.

Let me stress that this International Evaluation is not an American enterprise. It will be a truly international effort without results prejudged in advance. The Washington Organizaing Conference established the eight multinational working groups just detailed. The eight groups are scheduled to meet periodically over the next two years the first of such meetings are to be held next week in Vienna. The objective of the United States, as I have indicated before, is to build an international consensus on all the views confronting us. We fully appreciate that we cannot dictate a non-proliferation policy to the rest of the world. We believe that facts will show that recycling plutonium in thermal reactors is a mistake from economic, technical security and ecological points of view, but we accept that our views should be subject to international peer review. We believe that the facts support our view that there is time to examine more proliferation resistant alternatives to conventional reprocessing.

Obviously a more proliferation resistant reprocessing technology than the Purex process is not a panacea, and the claims of its feasibility need careful international scientific study. But this is an example of the type of alternative we believe deserves our careful collective attention. Most important, if we are to develop and coordinate effective policies to reach our mutual goal of nuclear power without nuclear proliferation, we must avoid assuming that there are no alternatives to the technological path upon which we are now embarked. At the very least, we owe to future generations the assurance that we examined real alternatives, and were not simply carried along by the momentum of the past.

Of course, our efforts to develop a consensus about a more proliferation resistant and safeguardable commercial fuel cycle cannot be achieved overnight. Our efforts will require patience and close cooperation among all interested countries.

A very brief summary. I think what we have here is an international political technical system that is continually evolving. It has evolved from the Atoms for Peace Program up to the present. Now, it is faced with its first major transformation - a transformation from a uranium technology to a plutonium technology. This certainly was not unexpected, but the transformation is now thought to be so potentially destabilizing to international relations as to deserve reexamination and INFCE is this reexamination.

I think that the Carter program, in general, and INFCE, in particular, will allow us to examine carefully the global implications and find more acceptable alternatives.

MR. WILLIAMS: A little time for questions.

DR. MANISCALCO: Jim Maniscalco, Lawrence Livermore Laboratory.

Does our present State Department Policy assume that we can be influential in deferring breeder and reprocessing technologies in other nations that are further committed to those paths than we are?

MR. SIMKINS: That is the hard road, obviously. I think we have a much better chance with nations who are as yet uncommitted and who are now making their energy decisions.

DR. MANISCALCO: I am referring to the French, the Germans, the British, and the Japanese.

MR. SIMKINS: All I can say to your question is that they came to our meeting, and they are going to participate in our study. We hope that the results of INFCE will influence their programs.

DR. HURWITZ: I am Henry Hurwitz, General Electric Company.

The evaluation of proliferation resistance in the different fuel cycles seems to rest on a fairly important assumption; namely that no new information or technology having to do with the weapons area will be divulged or developed during the 10 or 20 years that it takes to develop these cycles. Furthermore, it is assumed that the scenarios of major concern are not going to change.

Presently, we seem concerned about a small nation having a low grade A-bomb, but we don't know that this is going to be the concern in 10 or 20 years.

My basic question is what kind of assurance can be given to industry or government participants in programs aimed at developing the so-called proliferation-resistant cycles that when they are finished, the new cycles will actually turn out to be fundamentally superior to what we have now? How can INFCE address this type of problem without having full access to what is now known in the world about weapons technology?

MR. SIMKINS: There we have a pessimistic view of INFCE. I have given you the optimistic view. No one can predict the future; General Electric can't, and the State Department can't.

We are concerned with the nations developing primitive A-bombs and we are concerned with terrorist groups.

I don't think your point about having to divulge weapon secrets is valid. There will be people participating in INFCE with weapons knowledge, and I don't think it is necessary that they divulge the actual techniques involved in order to determine relative proliferation resistance.

The Department of Energy is now actively involved in planning the INFCE participation. But the other points that you made are good points, and I can't predict the future.

DR. BENDER: Dave Bender, from Lawrence Livermore Laboratory.

I would like to clarify a point which you brought up earlier. Is my understanding correct that you concede the point that commercial nuclear power is not the most expedient source or the cheapest source of fissile material, weapons material?

MR. SIMKINS: Yes, but I have reviewed a scenario where a country could turn to its commercial program to develop weapons.

DR. BENDER: Okay. And that, as I understand it, is the situation where it would find itself in a position of international stress.

MR. SIMKINS: That is one possibility.

DR. BENDER: And would turn to developing a nuclear weapon. Doesn't that strike you as rather unusual, or an unusual position in that you say a nation has no weapons technology, but in a period of stress would develop that very quickly? It seems to me that the weapons technology is a much greater body of knowledge than the body of knowledge required to produce fissile material.

MR. SIMKINS: I can only go by what one reads in the newspapers, that various lay people, high school students in Miami for instance have designed credible weapons. That aside, I think the time to develop the weapons technology may in fact be a crucial point in some cases and will of course be considered in INFCE.

MR. GRAHAM: Mr. Simkins, I am Frank Graham, Atomic Industrial Forum.

It is often said--and I think you probably sense it in this audience--that engineers and technicians look to political and perhaps institutional solutions to proliferation, and the politicians and perhaps those in government, at least the Adminstration, are looking to technical solutions for proliferation. And certainly, we have seen a number of the Adminstrations' initiatives in the technical area, including giving up one of our own energy resources.

But we haven't seen, or do we know, I think, of the political initiatives perhaps that are underway, and I wonder if you could describe these to this group.

MR. SIMKINS: I think INFCE is the most important political initiative at this time.

MR. GRAHAM: But isn't that a technical initiative, not a political one?

MR. SIMKINS: Yes, I would say it is a technical/political initiative vis-a-vis, say, the Nonproliferation Treaty which is strictly an international political initiative.

MR. GRAHAM: Are there efforts underway to improve the political or institutional structure that has, in the past, been successful in controlling proliferation?

MR. SIMKINS: Yes, we continue to work with the IAEA to improve their safeguards programs, and we also continue to press for universal adherence to the NPT.

MR. GRAHAM: Do you think technical solutions can solve the proliferation problem?

MR. SIMKINS: I think it is a combined technical/political problem.

MR. GRAHAM: Oh, I do too, but we don't see the political initiatives. At least they are not clear to us.

MR. SIMKINS: What in your view is a political initiative?

MR. GRAHAM: NPT, the IAEA, or perhaps our strengthening of our ties with other countries to reduce their motivation for acquiring weapons.

MR. SIMKINS: Yes, clearly we are doing those on a day-to-day basis, but that is not the sort of initiative that makes the front page.

MR. WILLIAMS: One last question.

MR. DOHERTY: Patrick Doherty, from Combustion Engineering.

Carrying forward on the previous question, assuming that the INFCE program develops information to support the concepts that are involved, don't you, in fact, have a political selling job at least as large as the Nonproliferation Treaty to then get the people to agree?

It is one thing to demonstrate that potentially a workable solution has been described technically; it is another to sell this thing to people.

And isn't this, in fact, going to be the critical issue as to whether this whole thing can work?

MR. SIMKINS: In short, yes.

The Nonproliferation Alternative
Systems Assessment Program
Saul Strauch

This is a time for introspection for the nuclear industry, and it is being studied with the emphasis on proliferation resistance and, consequently, it is also a time when a fresh look is required for all the options. Accordingly, I think this meeting is very timely.

I do welcome this opportunity to tell you about the nuclear Nonproliferation Alternative Systems Assessment Program which is to examine those nuclear alternatives that don't provide a direct access to nuclear weapons materials.

So I would essentially like to give you a little briefing on the Nonproliferation Alternative Systems Assessment Program, commonly called by its acronym of NASAP. I was going to give you some details on the International Nuclear Fuel Cycle Evaluation Program, but that was covered very well by the previous speaker. So I will direct my remarks to the role that fusion-fission systems might play in our NASAP assessment work.

Basically, the concepts shown in Figure 1 are those that we are looking at, with some of their associated fuel cycles. The information around the box in the center give you an indication of how we are evaluating, in a rough sense, these reactor systems.

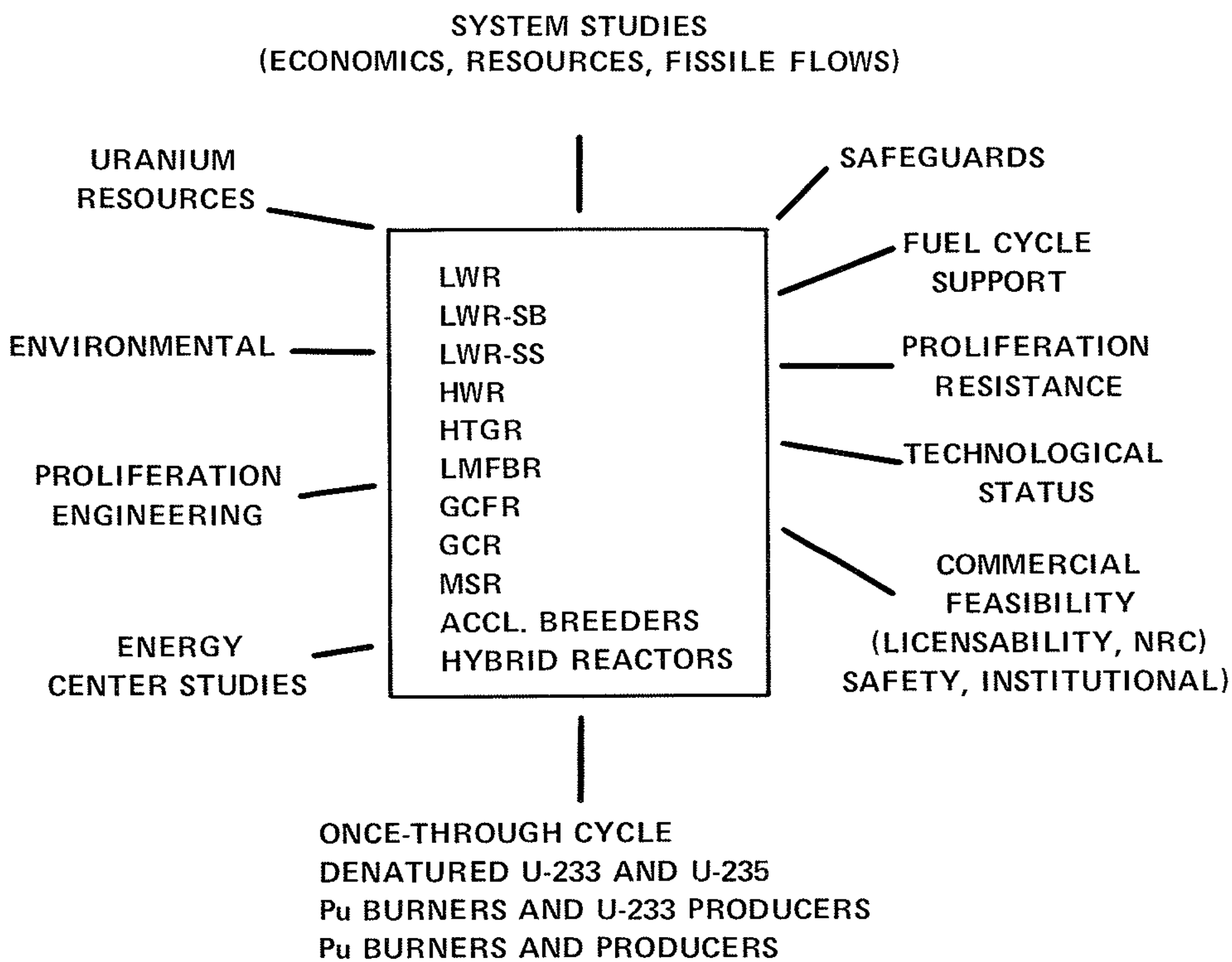
The primary emphasis is on the once-through cycle with its lifetime requirement for the standard light water reactor of 6,000 tons of natural uranium over a 30-year lifetime.

Let me give you some facts to put those numbers in perspective with respect to some of the calculations and work that we have underway.

If you improve a light water reactor by going to a spectral shift mode, which is to use heavy water in combination with light water coolant

FIGURE 1:

CANDIDATE NUCLEAR TECHNOLOGIES AND NASAP INPUTS



to improve the neutronics of the system, you can get about a 20 to 25 percent improvement in the uranium utilization over the plant lifetime. So you can take that 6,000 tons down to about 4,500 tons.

Now, I don't know exactly what number the first speaker utilized for the tails assay in the enrichment process, but if you can get down to a tails assay of about .1 percent (most of the calculations that I have seen have used a tails composition of .25 percent), you can improve the uranium utilization by a factor of about 20 percent.

So you can reduce that 4,500 tons of uranium requirement for the spectral shift reactor down to below 4,000 tons; and these are all once-through systems, not a recycle system.

On the heavy water reactor systems, we are considering slightly enriched designs because they give you the best uranium utilization for the heavy water system and can get, on a once-through cycle, a uranium requirement of about 3,500 tons over its 30-year lifetime.

So you can see by going to systems with improved uranium utilization characteristics, you can decrease the uranium requirements considerably and thereby also decrease the load on uranium mining.

We are looking, as you see on Figure 1, at breeder reactors (liquid metal fast breeders, gas-cooled fast breeders), other gas-cooled thermal reactors, the molten salt reactor, accelerator breeders, and last but not least, fusion-fission hybrids.

We examine these reactor systems for their safeguardability, proliferation resistance--and I will speak more to that point--the technological status of the systems, where are they in terms of their technology, and then what is the route to commercial feasibility. We look at them in terms of their licensability, and we work with the NRC on this aspect. We also examine institutional factors.

We also look at reactors in a recycle mode in the energy center concept. The reactors external to the energy center being operated on a once-through mode, with the spent fuel being sent back to the energy center. The reactors in the energy center are plutonium burners and uranium-233 producers. I will speak of these reactors later.

We also have work underway that examines techniques for improving proliferation resistance of particular cycles; what one might do not only to the reactor but also to the fuel cycle facilities to make it harder to get at fissile material directly.

We have environmental considerations involved there, and we also factor in the uranium resources. Here, we consider the CONAES studies as well as the internal DOE studies on uranium resources.

We also have a good deal of work associated with the systems aspects of these reactor types and the associated fuel cycles. I am sure that most of you know this, but out at Hanford Engineering Development Laboratories, we apply LP codes that use as input the demand for nuclear power, the reactor types with their associated characteristics, uranium resources, coal resources, and then provide through the computer, a projection of the construction schedule for the individual reactor types as a function of time, whether they be coal or nuclear. The computer model then optimizes the system such that the costs are minimal over the set time period. We usually examine these reactor types in this system viewpoint as far out as to the year 2050.

We are getting calculations that indicate that there is a role for these so-called once-through systems that include advance converter concepts. There is also considerable improvement in uranium utilization if you go to recycle modes.

As the first speaker pointed out, you can get considerable savings, perhaps as much as 25 to 30 percent if you recycle both the uranium and the plutonium components, and perhaps 15 to 20 percent if you can just recycle the uranium component.

In the focus on proliferation resistance--and this is the major area that we are examining--we first had to develop the methodology; we have progressed a good way towards that. There has been a report put out by Science Applications Incorporated, that examines the methodology with respect to proliferation resistance attributes: cost, time, difficulty, interruptability, resources at risk, and weapons utility of these particular materials.

We are going to take that information and other associated information to examine and develop some preliminary evaluation criteria: In other words, criteria to evaluate all reactor options and associated fuel cycles with respect to proliferation resistance as well as criteria associated with economics, commercial feasibility, environmental and technological factors.

We also are studying proliferation paths other than the commercial nuclear power route or, as pointed out by earlier speakers, dedicated routes for getting weapons grade material. Somewhere in about the next two months we hope to have a report available on a delineation of these routes and how these particular routes could be utilized by different types of countries.

We are examining the proliferation concerns of the public, the government, and the nuclear industry. We have had Booz, Allen & Hamilton doing a survey of these sectors. There is a draft report that will be finalized shortly.

We are looking at characteristics of potential proliferators and, as I mentioned before, we have a fairly large effort also on proliferation resistance engineering. Considerable work also is being done with respect to technological barriers associated with individual reactor concepts.

We are looking at denaturing of plutonium. If you add not only the isotope plutonium-238 but also the isotope plutonium-236 to the fuel that is obtained from light water reactors or breeder reactors, this contaminant will make the use of the material for weapons much less feasible.

We look at international barriers. Here again we interact with IAEA, the safeguards people, and the employment of sanctions. We have RAND Corporation and others helping us in this particular area. This international analysis is particularly appropriate for the NASAP exercise, and it will be utilized to some degree in the International Nuclear Fuel Cycle Evaluation program (see Figure 2).

Here we are utilizing systems analysis; only we are doing this in an international framework. We are doing the systems modeling with the HEDL staff to examine how these options might be applied on a world-wide basis, on the flows of fissile materials and waste materials associated with these different options when they are applied on a world-wide basis. The principal objectives are to determine the impact on resources and how this impacts the economics of power for particular regions of the world.

Figure 3 presents the milestones for the NASAP exercise. Around March of 1978 we hope to have performed an initial assessment of individual options and screen out those that don't appear to have sufficient promise to warrant detailed analysis. Now, even though they are screened out from the detailed study aspect, they may be studied on a long-term basis in less detail.

FIGURE 2:

NASAP IN AN INTERNATIONAL FRAMEWORK

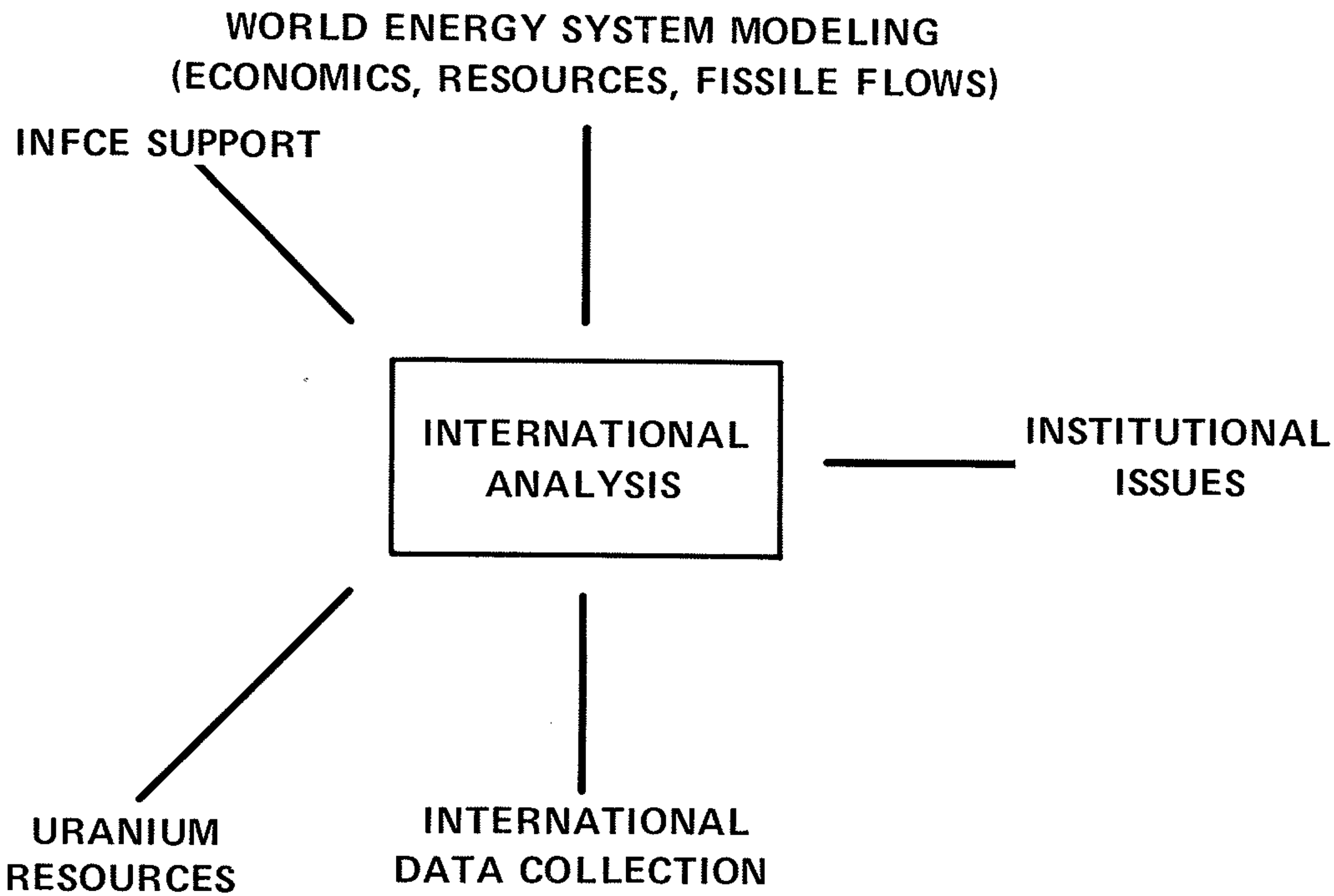


FIGURE 3:

PROPOSED
NASAP MAJOR MILESTONES

• PROLIFERATION METHODOLOGY REPORT	10/77
• SELECT CANDIDATE SYSTEMS	11/77
• PRELIMINARY PROLIFERATION SCREENING CRITERIA FINAL SCREENING CRITERIA OTHER FACTORS	12/77
• PRELIMINARY SCREENING OF CANDIDATE SYSTEMS	12/77
• FINAL PROLIFERATION SCREENING CRITERIA	1/78
• ALL CANDIDATES SCREENED	3/78
• INTERIM TECHNICAL REPORT (SUMMARIZES PRELIMINARY REPORTS ON THE MORE WELL-KNOWN REACTOR AND FUEL CYCLES)	10/78
• ALL SYSTEMS STUDIED COMPLETED	12/78
• SUMMARY TECHNICAL REPORT (SUMMARIZES TECHNICAL AND INSTITUTIONAL DATA)	5/79
• FINAL REPORT AND RECOMMENDATIONS (PROVIDES TRADEOFF ANALYSES, OPTION ANALYSES, AND CONSENSUS REVIEW OF ALL DATA; RECOMMENDS R&D PROGRAM AND PRIORITIES)	8/79

Around October of 1978, we expect to have an interim technical report. This report probably will have an impact on the Fiscal 1980 DOE budget. It will summarize the preliminary reports on the more well-known reactor and fuel cycles. The summary technical report should be published on May of 1979, and the final report in August of 1979. The INFCE study, which is on a two-year schedule, will probably follow with its conclusions about November of 1979. So, we are moving a little ahead of the INFCE study but in coordination with it. Figure 4 identifies the eight INFCE working groups as well as the lead countries for each group. The United States is associated with the last technical area, which is the advanced concepts, once-through, and the alternative reactor concepts.

Figure 1 provides suggestions for possible places where fusion-fission hybrids might fit in the NASAP. I want to stress, as I did before, that the priorities that we are moving with for the moment would give a higher consideration for the once-through cycles. I know, from looking at your agenda, that some of the design concepts that you are looking at in fusion-fission concepts are once-through concepts. But for those concepts, that are associated as fuel producers, we look at them in the context of an energy center where the fuel from the reactors outside the energy centers are returned to the energy centers. These reactors outside of the centers could be fueled with what we call denatured fuel; uranium-233 or uranium-235, mixed with uranium-238. Of course, depending upon how much 238 that you have in the reactor, you are going to generate plutonium.

If it is U-233 denatured with U-238 and you mix that with thorium, both in the outside and the inside reactors, a considerable amount of uranium-233

FIGURE 4:

INTERNATIONAL NUCLEAR FUEL CYCLE EVALUATION (INFCE)
TECHNICAL & ECONOMIC SCOPE

I. FUEL AND HEAVY WATER AVAILABILITY
- LEAD COUNTRIES: CANADA, EGYPT, INDIA

II. ENRICHMENT AVAILABILITY
- LEAD COUNTRIES: FRANCE, FEDERAL REPUBLIC OF GERMANY, IRAN

III. LONG-TERM SUPPLY ASSURANCES
- LEAD COUNTRIES: AUSTRALIA, PHILIPPINES, SWITZERLAND

IV. REPROCESSING, PLUTONIUM HANDLING, RECYCLE
- LEAD COUNTRIES: JAPAN, UNITED KINGDOM

V. FAST BREEDERS
- LEAD COUNTRIES: BELGIUM, ITALY, USSR

VI. SPENT FUEL MANAGEMENT
- LEAD COUNTRIES: ARGENTINA, SPAIN

VII. WASTE MANAGEMENT AND DISPOSAL
- LEAD COUNTRIES: FINLAND, NETHERLANDS, SWEDEN

VIII. ADVANCED CONCEPTS (ONCE-THROUGH/OTHER REACTOR)
- LEAD COUNTRIES: REPUBLIC OF KOREA, ROMANIA, U.S.A.

also will be generated. But the spent fuel returns to the energy center, and it is fed to either fast breeder reactors or to advanced converter reactors. The plutonium fissile material is mixed with thorium, either in the core fuel elements or the blanket fuel elements, depending upon whether it is fed to advanced converters or fast breeders, and U-233 is produced. The U-233 is mixed with U-238 after it is reprocessed and is returned to the outside reactors in the denatured form to produce energy,more 233,and some plutonium.

Now, a fusion-fission hybrid, as well as other fuel producers, also can be inside these energy centers and produce uranium-233. That uranium-233 can be mixed with U-238 and perhaps thorium if it is going to go on the outside in a U-233 producing system as before.

Let me conclude by pointing out that we have a contract with the University of Washington which subcontracts part of the work to Battelle Pacific Northwest Laboratories to look at fusion-fission hybrids. Essentially, this study is to make a survey of all the different fusion-fission hybrid concepts, recommend what is felt would be best in the context that we are examining them, and also give us an option with respect to the proliferation resistance characteristics of these concepts.

MR. STRAUCH: I would be happy to answer any questions that you might have.

DR. COFFMAN: Frank Coffman, with DOE.

Please put up the viewgraph showing the eight INFCE Chapters.

MR. STRAUCH: Yes.

DR. COFFMAN: Okay. In just perusing the lead countries that you have discussed, for instance, Chapter number 1, "Heavy Water Reactors," the lead country is Canada, who is very heavily involved in heavy water sales and production; and India, who used CANDU to develop their weapon system; and Egypt, who is threatened with a nuclear capability in Israel. I wonder how those three lead countries are going to arrive at an embargo, for instance, on CANDU reactor systems. And as I go down the list, in the enrichment area you have France and West Germany, who both would like very much to commercialize and sell enrichment capabilities. And right on down the line, fast breeders, you have got the USSR who, in particular, is very interested in the fast breeder program.

I guess I wonder how these eight committees are going to arrive at the kinds of conclusions that are consistant with the State Department position on proliferation. It seems rather that all the vested interests are placed in the categories which would make them arrive at opposite conclusions. I wonder if you have any feeling how that is going to turn out?

MR. STRAUCH: Well, why don't I take a quick shot at it, and then, Roy (Simkins), if you would like to, you can add more to it. But the work is not limited to the so-called lead countries, and the lead countries were selected to give a wide participation here.

I think a lot of the work, as you point out, reflects the preference associated with the particular countries. In other words, the United Kingdom has a preference for being involved in the reprocessing end of it; and the USSR had a preference for getting involved in fast breeder end of it.

But they will not be solely tackling these tasks by themselves. Each of the countries can get involved in any of the technological areas that they wish to; and it may not be that biased even by the countries that have a vested interest in the areas, themselves. I think it is going to be a pretty unbiased go-around here.

DR. STICKLEY: Martin Stickley, Department of Energy.

One of your block charts showed a study of proliferation paths. Can you comment on what paths are being considered and who is doing that for you? Is there a lead group?

MR. STRAUCH: Yes, the proliferation paths work is being done by a number of groups. Are you referring to the technological paths?

DR. STICKLEY: Yes.

MR. STRAUCH: Yes, we have people out at Savannah River Laboratories and Oak Ridge National Laboratories, in conjunction with the appropriate DOE groups like the Waste Management, that are looking at the individual fuel cycles to point out the so-called easy points of diversion if you could categorize it as that, and what are the routes from those points of diversion towards the accumulation of weapons material. And then we have the people out at Livermore looking at the credibility of that material, from that particular route, in terms of its weapons aspects.

MR. PUECHL: Karl Puechl, of Combustion Engineering.

On these alternate paths, are you also looking at what impact advanced technology might have in defining, at some future date, an alternate path, or are you just looking at current technology? This is sort of the same question that Henry (Hurwitz) asked before. At some point in time there probably will come into being a much easier proliferation path than we now have with the commercial fuel cycle. Are we trying to define that in time?

MR. STRAUCH: Yes, we are looking at it. How well we are able to define that I can't tell you, but we are looking to what are some of the routes to weapons grade materials, not just today's routes, but also possibly tomorrow's routes; and I don't know how well we are going to be able to define tomorrow's routes.

DR. BENDER: Dave Bender, Lawrence Livermore Lab.

With regards to the NASAP study, are you making a strong distinction in your fuel cycle evaluations between proliferation and diversion which, in my mind, are vastly different situations.

MR. STRAUCH: Well, we are looking mostly at national diversion; a particular country choosing to go the nuclear weapons route. We are not spending much time on subnational groups, a terrorist group, trying to get material for a terrorist activity. Am I tracking your question?

DR. BENDER: Yes, yes you are. My reaction to your answer is that technically we have much stronger leverage on the diversion question than on proliferation in that a concerted national effort has many options open to it, whereas a diversion effort could not mount a concerted technical attack, and that an alternate fuel cycle would be much more amenable to addressing a diversion issue as opposed to a proliferation issue.

MR. WILLIAMS: Okay. One more question.

MR. PALMER: Roger Palmer, of General Electric.

Could you tell me a little bit about who, in our country, is participating in the advanced concepts, how is that organized with Korea and Romania, and whether your organization participating in that?

MR. STRAUCH: I don't think that has been established yet, Roger. Perhaps Roy (Simkins) has more on it, but as far as I know, it hasn't been established. I am not aware of the details if it has been established.

MR. WILLIAMS: How would one find out the answer to that question?

MR. STRAUCH: I am sure that when it is established, I will know, and I will be happy to convey the information to whoever wants it.

UTILITY PERSPECTIVE ON FUSION-FISSION ENERGY SYSTEMS

By

Piet Bos

Electric Power Research Institute

It is a pleasure to be here to provide an industry input what may presumptiously be called a utility perspective. As many of you well know, the utility industry is not a homogeneous industry. It is represented by many individual companies. And when I talk about the fusion-fission issue, I am referring to the thinking of certain utilities only. Many utilities are not yet involved in the overall fusion concept since it is a long-range option. Other utilities are not necessarily in agreement with the particular alternative of fusion-fission, which is the subject of this meeting.

So when I say "utility perspective," I am trying to represent a broad spectrum of what we at EPRI consider a position with regard to the hybrid fusion-fission option and also the utility requirements in general as applied to energy systems.

Since the utility industry is a major user of primary resources, we are, of course, very much interested in the energy supply and demand situation, and consequently, our Planning Department has conducted a study showing -- (figure 1) -- showing the electric energy demand requirements, as a function of time, the center vertical line representing the turn of the century-the year 2000. Also on this chart are shown the various supply options to meet this electric energy demand.

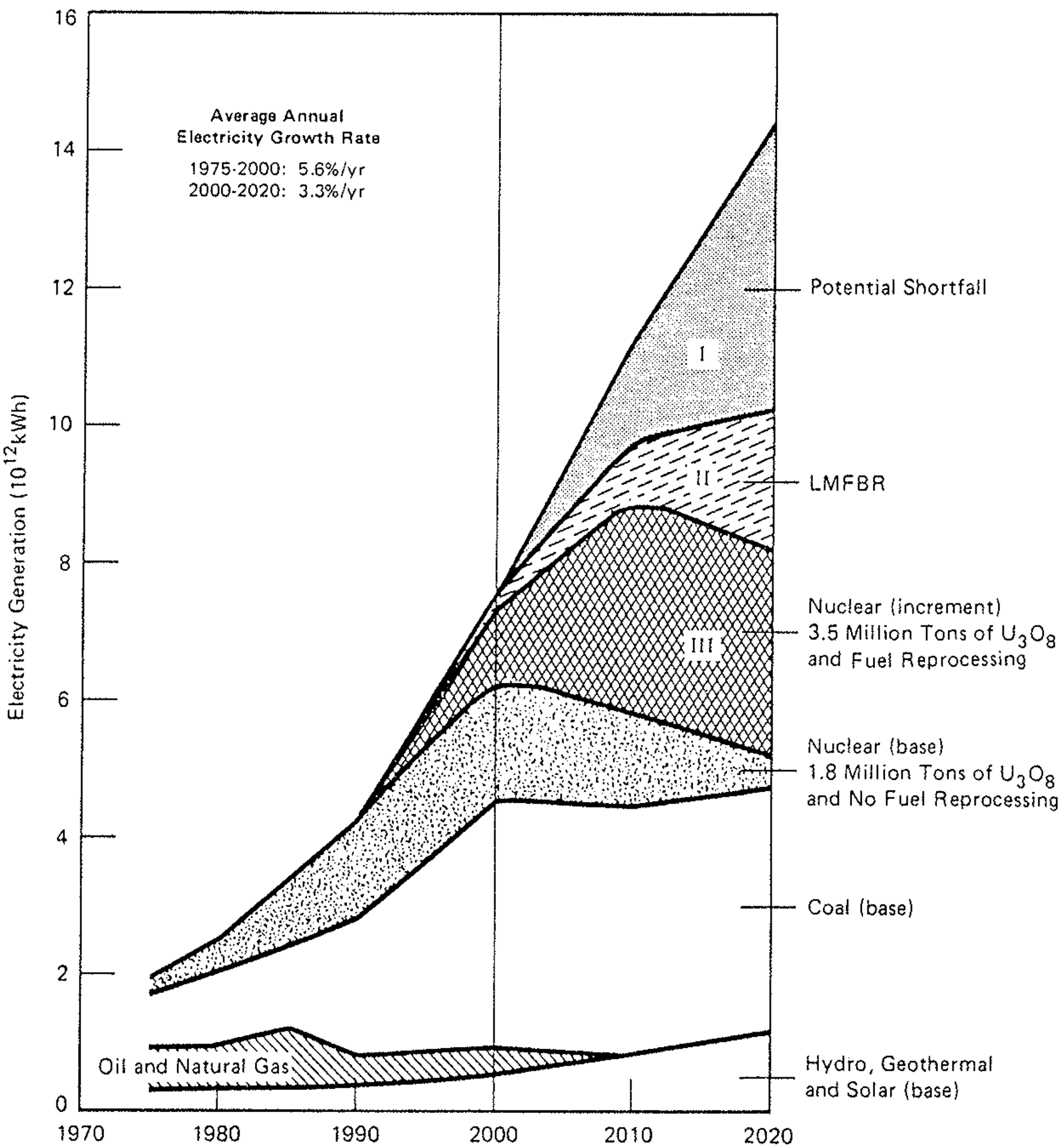
The present rate of electric energy demand growth is about 6½ percent on a national average. This chart represents a much more modest growth scenario which incorporates elements of energy conservation, for example, the application of solar heating and cooling and conservation to buildings to alleviate some of the energy demand.

Specifically, the energy demand shown represents a 5.6 percent growth rate between the years 1975 and 2000, which requires a substantial growth in electricity generation in terms of kilowatt hours, almost reaching 40 to 50 percent utilization of the primary energy resources by the year 2000.

There are some interesting things to be seen from this particular figure. For example, the phasing out of the oil and natural gas supplies somewhere in the beginning of the next century, to be taken up primarily by a number of conventional alternatives such as coal and nuclear.

Coal is shown to be expanded in utilization for base load generation. However, shown is a curtailment in the use of coal starting in the year 2000, which is essentially due to competing alternative needs for these coal resources. Furthermore, as have been mentioned earlier this morning, there may be additional logistics and ecological constraints on the mining and exploration of coal. Consequently, the projections by EPRI show a leveling off in the use of coal for electric power production starting at the year 2000.

The next supply option, the nuclear light water reactor technology, is shown expanding until the year 2000, with subsequent decreasing utilization because of resource limitations of this particular option (especially the 1.8 million tons of uranium ore case) unless another source for fueling these light water reactors can be found.



NOTE: Results plotted here are from sample study cases A and D, Figure 12
Assumptions used to draw this graph are shown in Figure 11

Projected electricity generation mix, 1975 to 2020

Also shown is an expanded LWR resource case using 3.5 million tons of uranium, including fuel reprocessing, which represents about the upper limit discussed earlier this morning.

The LMFBR introduction is shown to commence around the year 2000, but again, after the initial growth rate, a significant limitation occurs because of lack of resource availability with regard to plutonium inventory requirements of the LMFBR.

This leaves a potential electric energy shortfall between demand and supply starting somewhere in the year 2000 and continuing thereafter, which, of course, gives rise to the demand for alternative resources, even though these particular resources may come at a fairly high cost.

Starting at the bottom and going to the top of energy supply options, the cost of energy tends to increase. Of course, the problem is even more severe than indicated in this projection if any of these scenarios do not come to pass. Keep in mind, however, that this is considered to be a moderate growth scenario; even though the chart does not present the low growth nor the high growth situation, it represents a lower growth case than the past or present rates of growth.

In respect to the fusion-fission option, it is important to note the following scenario considerations: (1) the limitation of availability of light water reactors using the natural fissile resources,, (2) the potential of a hybrid fusion-fission device to provide an alternative means of fissile fuel, (3) building upon an existing technology in an evolutionary sense, and (4) using the fusion-fission technology as a backup and a complementary option to the LMFBR to provide for the energy demand of the future.

The options that are available, other than the conventional and fusion options I just referred to, are solar energy and transitionally geothermal energy.

In the geothermal area, hydrothermal deployment might alleviate the energy gap by filling some of the demand as a result of further exploration. We are presently working very hard at EPRI to develop this option. Furthermore, we have just started a major program on geopressure deployment to, again, provide an alternative resource to alleviate the gap between the time that conventional resources will be depleted and the introduction of the new energy options.

In solar energy, the solar thermal and photovoltaics options for electric power generations, based on recent studies, will only provide for intermediate electric demand applications, and will not be applicable to base load. The sun cycle is more suited to provide energy from early morning to evening hours, and it would be too costly, based upon studies performed to date, to conceive them as base load options at the present time. Wind and ocean thermal gradients are further down the line, in that these options are much more limited geographically and economically less attractive. Especially ocean thermal energy conversion is extremely limited, expensive, and consequently, very low in our priorities.

This leaves fusion as a potential base load option in addition to (and complementary to) the only other inexhaustible option: the LMFBR.

Because of uncertainty in developing and deploying any one option, I think it is important that all viable options for each of these alternatives are developed to ensure that viable generation options are vailable to future decision-makers to meet the energy demands of the future. The consequences of not meeting these energy demands, as you well know, will be quite severe for industrialized nations.

Typical lead times for development and market penetration of new concepts are from 30-60 years. As shown in figure 1, a gap between supply and demand starts showing up by the year 2000, which is not too far away -- only about 22 years. Therefore, we must relate the typical lead times required to bring about a new energy technology into a spectrum of significant use with the need for their deployment.

Figure 2 shows the various development phases and associated costs for new energy options. The first phase relates to establishing scientific feasibility or proof of principle -- and here is where fusion is at present. Hopefully, we will be resolving the issue of scientific feasibility for fusion within the next five to ten years. Next, are engineering feasibility and engineering feasibility or pilot plan demonstration; followed by commercial feasibility, consisting of a commercial scale power plant demonstration where we are not only looking at the technical and operational characteristics, but at the economic characteristics required for commercial viability; followed by a phase of utility integration, which represents a period of evaluation after the commercial plant has been introduced to determine the operationl characteristics in addition to the commercial objectives; followed by the time period required for significant market penetration.

The typical overall cycle for achieving significant market penetration of most new technologies in the past has been about 30 to 60 years. Therefore, placing this in perspective to the demand-supply situation, there is not much time to lose.

Related to this are the relative costs for the development phases. Shown on this figure are the costs for each of these phases of RD&D, as well as the cumulative funding, as a function of years from time of the inception from scientific feasibility through utility integration. The scale on the left is logarithmic and consequently this indicates that the costs will rise substantially once you get into the hardware characterized by the engineering and commercial power plant feasibility demonstrations.

We can learn a number of things from this particular figure. Since scientific feasibility studies are relatively inexpensive as compared to hardware deployment, this is the phase where many alternatives or options should be explored.

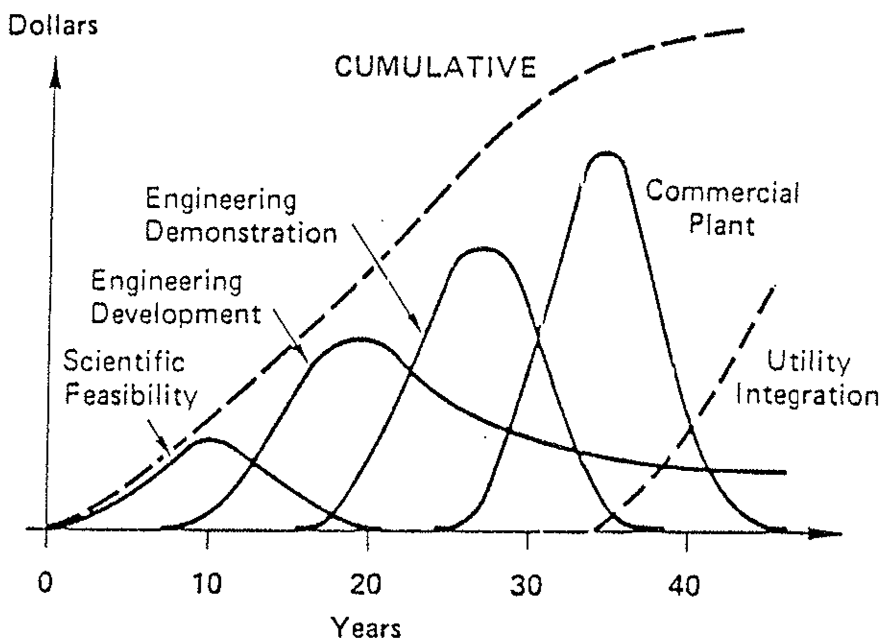
This has been the EPRI's emphasis for fusion to date. Under the leadership previously of Mr. Bill Gough, who has recently rejoined the Department of Energy, and presently of Drs. Bob Scott and Noel Amherd, EPRI has continually emphasized and identified many alternative options and applications for fusion and have pushed for detailed investigation of these options and alternatives to assess their viability.

In the area of alternative concepts, EPRI is looking at the potential of alternative fuels which would better meet many of the utility requirements with regard to power plant availability, such as essentially "neutron-free" reactions (for example: p-B). We are also looking at smaller devices which would be much easier to commercialize because, as you well know, the cumulative costs are very much a function of the size of the demonstration hardware which is related to the eventual size of the commercial plant.

Size is an important aspect not only from economic/financial considerations but from utility operational considerations as well and, consequently, the electric utility industry is very much interested in small fusion devices, which point has adequately been expounded by Clint Ashworth of Pacific Gas & Electric.

Figure 2

PHASES OF R&D



In the area of alternative fusion applications, EPRI is interested in off-line applications such as fuel production (hybrid fusion-fission) and waste transmutation.

It is at this stage of the game where one can do a lot of studies; feasibility studies, as well as engineering development and preliminary design studies. This is the phase where one can evaluate and shake out the various options, to ensure that, once the program goes down stream, all viable options have been identified. It is imperative to do this in the early stages of development, because the nation cannot afford to carry all options forward indefinitely because of the increasing cost of engineering and commercial power plant demonstrations. Eventually only one or two fusion options may survive at best.

From this particular viewpoint one can deduce that small scale systems as well as evolutionary concepts are of interest.

In order to determine which of these many alternative concepts would be attractive and viable options for the future, we have to establish a number of criteria which relate back to the issues of commercialization, and utility requirements for utilization of such power plants. A summary of these criteria are listed in Figure 3.

They incorporate economic and financial considerations, primarily, the cost, which, of course, favors the smaller devices. The smaller the fusion plant size contemplated, the smaller the risk and the cost of investment, making commercialization much more feasible. The other issue relates to the availability of power plants which is related to the reliability of operation, meaning low forced or unscheduled outages in utility jargon, in order to realize capacity credit, which reduces the need for back up of fusion systems by conventional plants in order to meet the reliability criteria imposed by the grid to provide electricity on demand.

Also related to this are the scheduled maintenance requirements, which should not exceed more than about a month. Similar to conventional power plants, building devices that have either high forced outage rates or long scheduled maintenance periods just do not match the grid requirements very well.

One can get around this problem by deployment of a hybrid fusion-fission concept that does not feed electricity directly into the grid, and essentially constitutes a fuel production plant only.

The next issue, resource availability, is very important, and, or course, since we are dealing with inexhaustible resources in the case of fusion, we are satisfying this criterion.

Another criterion deals with the system's capability and flexibility with regard to control and operating characteristics, as well as with regard to environmental and safety issues. The latter directly affects the licensing capability of utilities to install such power plants.

At EPRI we are conducting several studies, which are called RDIA's (requirement definition impact analyses) to determine what these requirements are and how various fusion systems will fit into a power grid. The requirements criteria derived from these studies permit the selection of what we call "preferred system concepts" for future development.

One of the fusion alternatives, as I mentioned, is the hybrid fusion-fission concept. The role and capabilities of fusion-fission energy systems are shown in Figure 4. Studies done by Dr. Jim Maniscalco and Westinghouse Corporation have indicated, for example, the production and consumption relationship of fissile materials using fusion as a fuel production plant.

Figure 3

ECONOMIC

- Cost
- Reliability – Low Forced Outage Rate
- Maintenance – Low Planned Outage Rate

RESOURCE AVAILABILITY

SYSTEM CAPABILITY AND FLEXIBILITY

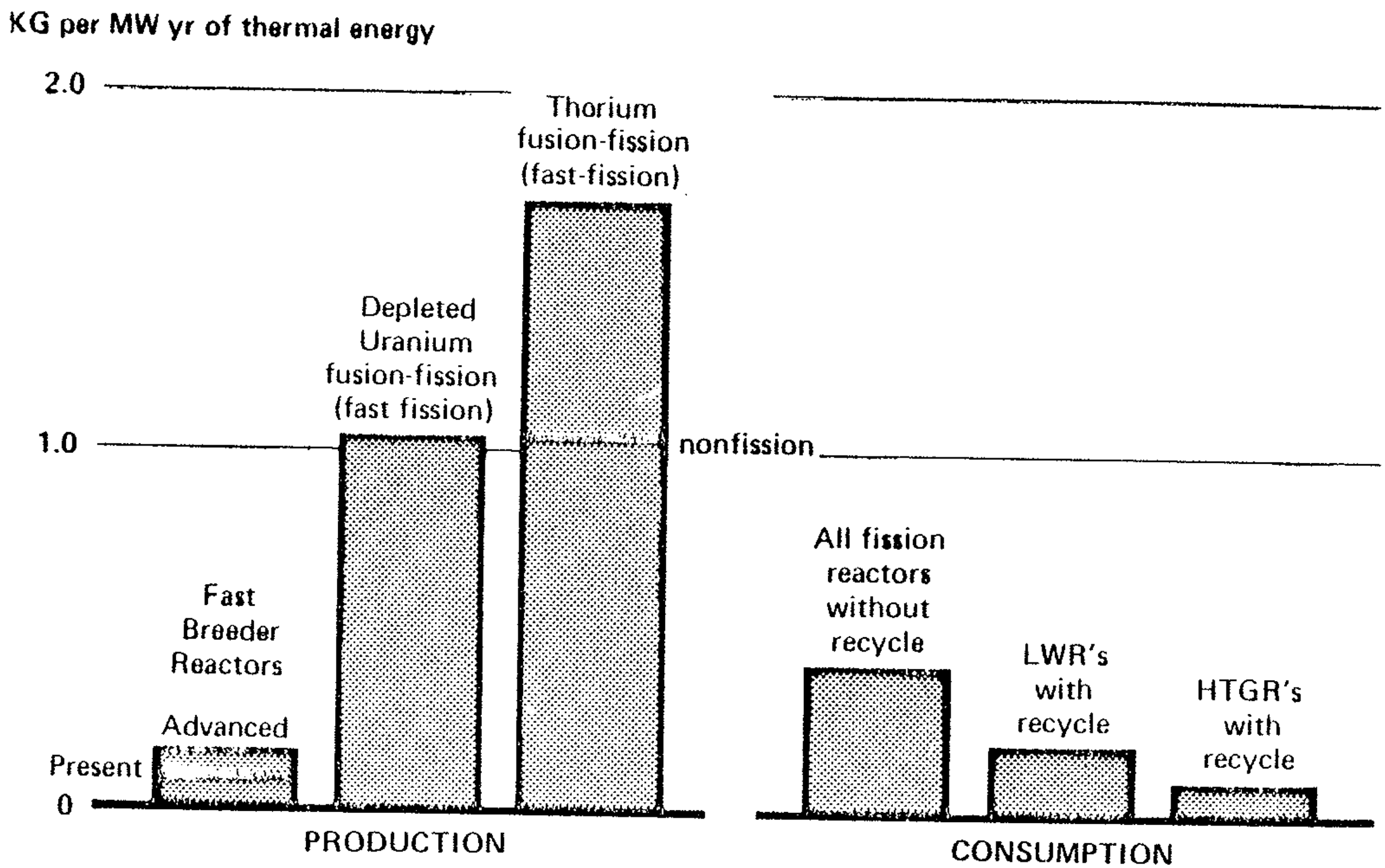
- Control and Operating Characteristics
- Ability to Tolerate Abnormal Events
- Unit Rating
- Environmental and Safety Issues

LICENSING

Utility system criteria for selection of advanced generation options

Figure 4

A ROLE FOR FUSION-FISSION ENERGY SYSTEMS



Such a fusion-fission fuel production plant could either produce electricity and fuel, or just fuel by itself. In either case such a plant has the advantage over an LMFBR as a fuel production plant, in that it is able to generate much more fissile material as compared to the consumption by conventional technology plants such as light water reactors, shown on the right, or even better yet, HTGRs.

Taking the fast fission hybrid, for example, the ratio of fuel production, or the ability to fuel the light water reactors is about 5 to 1. In other words, we could fuel approximately five light water reactors using one fusion-fission hybrid plant of equivalent thermal capacity. If proliferation (perceived or real) is an issue, one can also select the thorium cycle for production of U-233, in which case the ratio of fueling HTGRs can be in excess of 10.

The neutron-rich fusion characteristics gives the fusion-fission concept a significant edge over the LMFBR, since the latter essentially has very low breeding capability. The higher breeding ratio of fusion would permit a later introduction, as Dr. Mike Lotker has shown previously, of the fusion-fission option as compared to the LMFBR, which would essentially be complementary. Even with a later introduction of the fusion-fission option, as a result of the higher multiplier factor involved in breeding, this fusion option presents a viable application for utility use.

I am going to gloss over some of the progress that has been made towards fusion-fission. (Figure 5.) As can be seen, actual experiment data are starting to move very closely to scientific feasibility as indicated by the break-even region, shown as the percent of fuel burned.

Without going into details, the planned experiments will incorporate inertial as well as magnetic confinement systems; potentially small systems such as mirrors as well as Tokamaks.

Some of the advantages of fusion-fission hybrids are listed in Figure 6. Of primary interest are the first number of issues which address the evolutionary development. By producing fissile fuels for existing light water reactor plants, this fusion option can be used to extend the time frame for this technology. In other words, both from a utility point of view and from a vendor point of view, the usefulness of a known and existing technology can be extended. The utility industry is used to the LWR technology, since it represents a proven technology and consequently it will be easier to incorporate fusion in an evolutionary sense rather than developing an entirely new concept. This is a significant advantage for deployment of the fusion-fission hybrid concept.

The other advantage is that it can be utilized off-line eliminating the grid availability requirements or capacity credit issue. Also the concept is less sensitive to cost estimates because of the multiplier effect. One can probably afford up to three times the light water reactor cost or maybe LMFBR cost, since the fusion-fission hybrid costs are spreading over many power plants, including those fueled by this concept, since the sensitivity of fuel cost on busbar energy cost for light water reactors is quite small. All of this suggests a very strong effort to investigate further the fusion-fission hybrid option.

The disadvantages, as shown in Figure 7, include the synergy problem that one encounters, and as is frequently the case synergies incur the 2 plus 2 equal 5 syndrome. In reality, the fusion-fission option will incur a number of technical problems by combining some of the disadvantages of both fusion and fission, but, actually not more than already present in the LMFBR case.

Figure 5

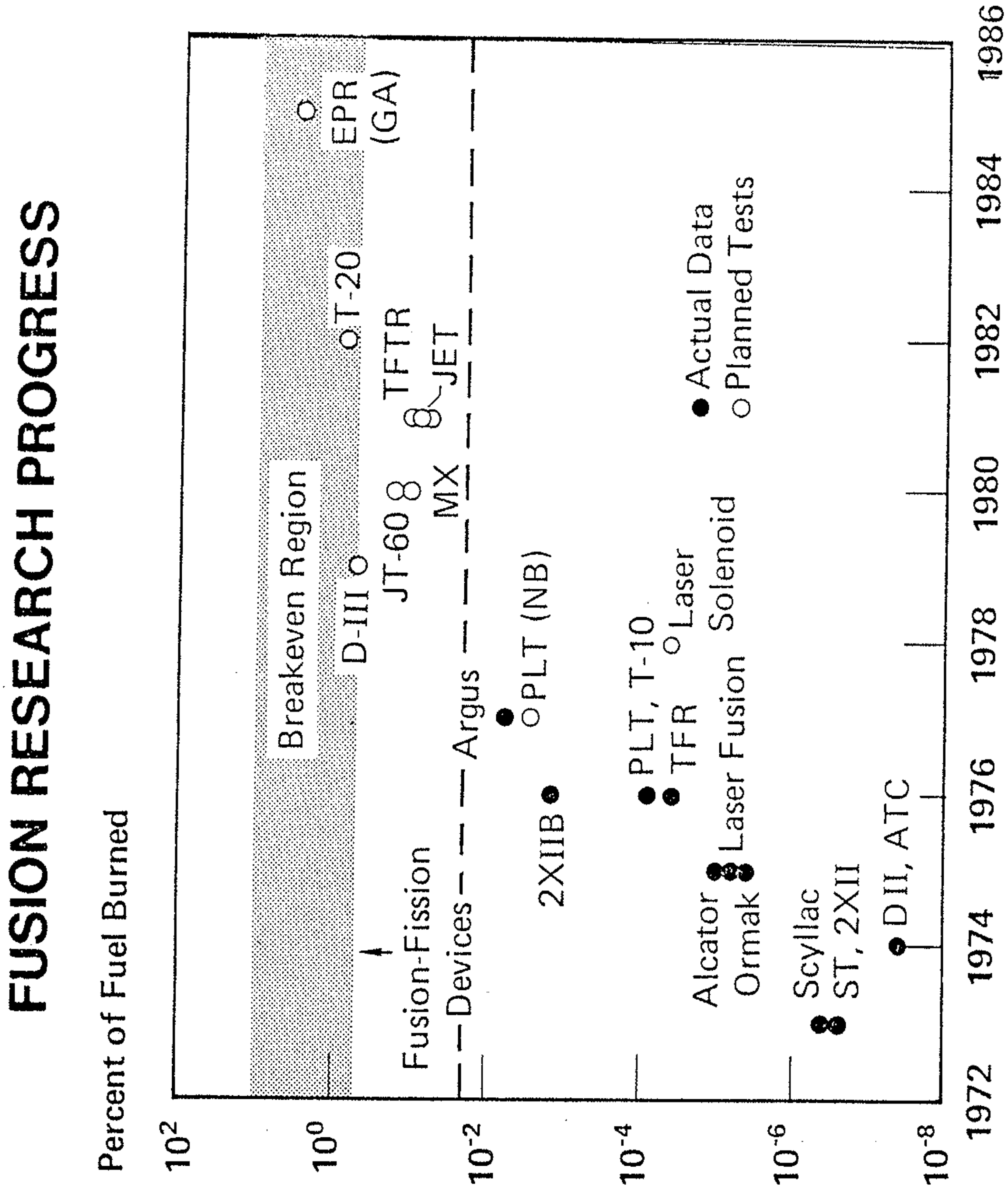

FUSION RESEARCH PROGRESS
Percent of Fuel Burned
10^2
10^0
10^{-2}
10^{-4}
10^{-6}
10^{-8}
1972
1974
1976
1978
1980
1982
1984
1986
Breakeven Region
Fusion-Fission Devices
Argus
2XIIB
PLT (NB)
PLT, T-10
TFR
Laser
Laser Fusion
Solenoid
D-III
JT-60
MX
TFTR
JET
T-20
EPR (GA)
Alcator
Ormak
Scyllac
ST, 2XII
DII, ATC
Actual Data
Planned Tests

Figure 6

FUSION - FISSION HYBRIDS

ADVANTAGES

- ADDRESSES A NATIONAL NEED
  - Fuel For LWR's
- FUSION PHYSICS ALMOST IN HAND
- BASED ON SIGNIFICANT FISSION EXPERIENCE
- UTILITIES CONTINUE TO DEPEND ON LWR's WITH KNOWN
  - Economics
  - Regulation
  - Environmental Issues
  - Safety Consequences
- VENDORS CONTINUE IN LWR BUSINESS
  - Mates With Existing Nuclear Industry
- COULD BE OFFLINE
  - Don't Need Instantaneous Reliability
  - Government Could Be "Enrichment" Supplier
- LESS SENSITIVE TO FUSION COST UNCERTAINTIES
- JUSTIFIES STRONG FUSION PROGRAM
  - Favorable Cost Benefit For Hybrids
  - Acts as Technological "Touchstone" For Fusion

EPRI

FUSION - FISSION HYBRIDS
DISADVANTAGES

- MATING FUSION AND FISSION MAY BE TECHNICALLY HARD
  - BLANKET COOLING
  - MAINTENANCE IN COMPLEX GEOMETRIES
- SOCIOPOLITICAL
  - FISSION WASTES
  - RADIOACTIVITY
  - AFTERHEAT
  - WILL SOCIETY ACCEPT LWR?
- PROLIFERATION CONCERNS
  - PLUTONIUM OR U-233 PRODUCTION
- DIVERT EFFORT AND SUPPORT FROM PURE FUSION
  - PRECLUDES REALIZATION OF FUSION'S ENVIRONMENTAL AND SAFETY BENEFITS

EPRI

With regard to proliferation, this is more or less a political issue at this particular time. However, the fusion-fission concept has the option of using the thorium/U-233 fuel cycle.

Finally, I should mention that the fusion-fission effort should not divert the support of pure fusion. Pure fusion, of course, has many advantages, however I will not elaborate upon them in this talk. I do like to mention here, however, that the hybrid concept is one of many alternatives, and this is the time, as I mentioned before, to consider many alternatives. The pure fission options should continue to be investigated and there is no intention on the part of EPRI at present to abandon this research in favor of the hybrid. We feel, however, that the hybrid fusion-fission is an important concept.

Given the anticipated success of plasma physics, within the next 10 or 15 years, we will continue to scrutinize as many options as are deemed viable. We plan to have a strong complementary interaction with the Department of Energy in this area, both in planning and fiscal support. To date EPRI has supported several programs in fusion-fission area, amounting to approximately \$2.1 million over the last two and a half years, and we plan to continue this effort. DOE has spent about \$1.7 million on this concept over the last two years. Part of the reasons for bringing about this meeting is to establish research criteria, to start a dialogue between interested parties, and to invite the fission community to assist in solving in the many problems that lie ahead for the fusion-fission concept. Many of the technological problems we will be facing in fusion will directly relate to similar problems experienced in the LMFBR program.

My personal opinion is that the country has no choice but to develop all <u>viable</u> options to supply future energy demand. It is not a matter of either/or. The energy demand picture of the future requires that we do develop as many viable options as possible for incorporation into future energy decision-making.

I should mention one additional thing. The next speaker will mention alternatives to fusion for breeding of fissile materials such as the linear accelerator. I support that all alternatives must be investigated before we make a final decision as to how far we will proceed on any particular option.

MR. WILLIAMS: Thank you, Mr. Bos. We have time for one or two questions.

DR. KOSTOFF: Ron Kostoff, Department of Energy.

In your first slide you mention a shortfall after the year 2000. What prevents or what limits the LMFBR from reducing the shortfall?

MR. BOS: The LMFBR is limited: (1) by the time of introduction; and (2) by plutonium availability. A disadvantage of the LMFBR is to initially require a substantial inventory of plutonium, and secondly because the breeding ratio is not nearly as good as the 'neutron-rich' fusion-fission concept. The data shown represents the projection of the EPRI planning staff, which include these considerations with regard to the LMFBR. Consequently, it essentially is a fuel resource limitation.

DR. KOSTOFF: One final question. You mentioned you spent $2 million in the past for hybrids. How much are you spending <u>now</u> to support hybrids?

MR. BOS: During the past year, I think spending averaged about $700,000 or $800,000. We plan to continue at this rate of spending during the next year, depending upon the outcome of the studies. As you well know, you can only do so many paper studies. There is a logical evolution as to how much you can do.

DR. WOODRUFF: Gene Woodruff, the University of Washington.

In that first slide, I didn't understand the size of the nuclear component with reprocessing. If you integrate those curves it is about equal to the nuclear component without reprocessing. Most people seem to agree that if you reprocess both uranium and plutonium, it is about a 25 percent benefit. So that seems to be too large unless you have some spillover for the LMFBR, but based on what you just said, apparently that is not the case either.

MR. BOS: I agree, the data shown represent incremental supply. There are two aspects on the chart. Let me go back one second. (Slide shown.) All of the data shown are additional. The first light water reactor case represent a uranium supply of 1.8 million tons without fuel reprocessing. The next case represents the incremental energy supply for a uranium supply of 3 1/2 million, including reprocessing.

DR. WOODRUFF: It is a double factor then, more ore and reprocessing?

MR. BOS: Yes.

CONTINUED ELECTRICAL POWER FROM FISSION

by

Herbert Kouts

Department of Nuclear Energy

Brookhaven National Laboratory

Upton, New York 11973

March 1978

CONTINUED ELECTRICAL POWER FROM FISSION

As several speakers have already pointed out, it will be necessary for the United States to draw on all feasible methods of providing useful energy in years to come. There really are not alternative ways to supply the energy we need; there are only candidates for the supply. I would like to talk about candidates based on nuclear fission.

Of course, the light-water-moderated and cooled reactor is still our main line for generation of electricity through nuclear fission. The number of light water reactors to be built by the year 2000 still remains uncertain, and estimates made by different people vary by sizeable factors. It is almost without question, however, that the number will be in the hundreds. Associated with such large numbers of central station power plants will be problems related to continuing the necessary supply of nuclear fuel up to the turn of the century, and the ability to provide fuel for nuclear fission reactors in even the more distant future if they should continue to prove as economically attractive as they have in the past. Because of these questions, attention is being given to a number of possible ways of stretching the nuclear fuel supply. The more prominent possibilities are listed in Table 1. I will now discuss these in some detail.

Over the past decade or two, the sodium-cooled fast breeder reactor has been the centerpiece of our national response to the need for long term supply of electrical energy. This early commitment to the sodium-cooled breeder is now in some doubt. The Administration has announced that it is opposed to the construction of the Clinch River Breeder Reactor. This has not absolutely stopped Clinch River, but it is not likely that the project will be continued in the face of the stated opposition. Part of the reason for opposition to continuing Clinch River has been a perception that the construction of this device by the United States at this time represents an endorsement of a fuel cycle choice which may have unfortunate implications with respect to the proliferation of nuclear weapons capability. Some opposition to this fast breeder demonstration plant is also based on its technical features, which many people believe to be out of date. More about this in a moment.

In any case, there is now considerable uncertainty as to timing and type of commercial fast breeder power plants that we may use in the future, if these are, in fact, to be built in this country. At the very least, the fast breeder development program has been substantially set back in schedule. Even further delays are likely for the reasons I have just mentioned, and also because there remains a high level of public wariness with respect to the safety or social acceptability of fast breeder reactors.

I have mentioned the view on the part of some that the Clinch River design is outdated, or that it will not have the features that a breeder reactor must have to meet the long term problems. There are three principal points made concerning the value of the technology proposed for Clinch River. The first is that the choice of fuel in Clinch River leads to a breeding gain so low as to endanger the ability to supply the fissile fuel that would be needed to support additional breeders as these must come on line. A second reason is the high capital cost of this design; it impresses some as uneconomical. A third concern is that steam generators which have sodium on one side and water on the other side of their tubing may not be able to last the lifetime of the plant without failure. Failure of a sodium-to-water steam generator is not a trivial problem. Yet, no one really believes that the design of such a device has been brought to a satisfactory level of performance in any of the sodium-cooled reactors in the world.

An alternate fast breeder design is based on gas cooling. This has many attractive features in principle. However, the question of assuring continued cooling of a gas-cooled fast reactor core after loss of all electrical power continues to bother safety analysts. This presents a major obstacle in the safety review of a gas-cooled fast reactor. In summary, we might say of the fast breeder that in spite of its technical attractiveness, the problems facing it in its different versions are substantial and have led to a growing interest in other forms of solution to the long-term electrical supply problem.

A number of studies have been undertaken to find out how well the thorium fuel cycle might serve as an alternative to the use of the uranium-plutonium cycle. The use of thorium fuel has been proposed in connection with a great many kinds of reactor. The Naval Reactors program has included a strong element directed at development of a light-water breeder reactor, for many years. The Shippingport reactor has now been loaded with an experimental core of this kind. Although almost all financial support of the Molten Salt Breeder Reactor developed at Oak Ridge ended some time ago, there is continued interest in the concept. The CANDU reactor of Canadian design could be fueled with thorium and uranium-233, and this measure has been proposed by some as a means of stretching the useful supply of fissile fuel. In this country, the High-Temperature Gas-cooled Reactor, designed and developed by the General Atomics Corporation, would operate on the thorium cycle, as might the Pebble Bed Reactor developed at Jülich, Germany. All cycles based on use of thorium have the common problem that at least the first reactors using them must be started with other kinds of fissile fuel. The demands for initial uranium-235 would be very high, as would the requirement for separative work in isotope separation plants.

The CANDU reactor has proven to be very successful both in Canada and the international market. It has been pointed out by a number of thoughtful people that if the United States were using reactors of this kind instead of light water reactors that require slightly enriched uranium as fuel, our domestic supply of uranium would last longer. Of course, the light water reactors already exist, and we do not have at present the industrial base for manufacturing heavy water power reactors. To develop the capability needed if we were to build and operate heavy water reactors would take a long time. This loss of time would eliminate much, if not all, of the attractiveness of CANDU reactors for this country. I believe, however, that an up-to-date economic analysis is needed to determine whether there might still be a positive gain from such a change in reactor type. It is true that older analyses have indicated that because of the loss of time there would be no gain. But the rate of reactor construction we are now encountering is much lower than was assumed in these earlier calculations, and we should explore the effect on that conclusion.

Before leaving the subject of heavy-water reactors, we might simply note two of their additional features. They have higher capital costs themselves. On the other hand, the heavy water reactors have the attractive safety feature that no single pipe break could lead to a complete loss of coolant.

A number of years ago, there was an active project in the United States to develop particle accelerators that could be used in making fissile material for weapons purposes. This was called the Materials Testing Accelerator project. High energy protons or deuterons would be directed at targets, in which large numbers of neutrons would be generated through spallation and other nuclear reactions, and those neutrons when captured in uranium-238 would produce plutonium to be used in weapons. This project was dropped when the decision was made to build the Savannah River Reactors instead. Since those early days, substantial advances have been made in linear accelerator design. It is now possible to build linear accelerators with very high currents and with very high efficiency in converting input power to beam power. It is, in fact, now possible to build linear accelerators with beam currents of hundreds of megawatts, and with more than half the input power converted to beam power (efficiency greater than fifty percent). These technological advances have led to renewed interest in use of these so-called electronuclear devices in the fission power cycle. Table 2 lists four variants of electronuclear machines. The first is a driven reactor. This would be a reactor core which, though subcritical, would act as a multiplier of neutrons supplied it from the target of a proton linear accelerator. We have been giving considerable attention at Brookhaven to the possibilities of such driven reactors. We find that their attractiveness will depend most strongly on the ratio of the net electrical power of the reactor to the electrical power used to run the accelerator. Figure 1 shows the kind of reactivity variation with time that might be expected if a light water reactor were to be used in this way. The analysis has assumed that the fuel in the light water reactor is initially natural uranium oxide. As neutrons are fed in, there is a buildup of plutonium which increases the reactivity to a maximum value of k of about 0.9. Past this point, fission product buildup again reduces the reactivity. An average value over the cycle shown might be about 0.8. We see this value as a little low and regard a value of about 0.9 as more desirable. We are exploring other possibilities involving different reactor types, and also see several possible interesting modes of operation. One other version of this machine would not generate useful electrical power directly, but would be directed at the initial objective of years ago of producing plutonium. This would be used in a separate reactor. The concept would have the disadvantage of requiring reprocessing of the spent fuel, and it therefore would not meet criteria which have been tentatively set by the Administration for nonproliferating fuel cycles. However, a variant of this scheme would boost plutonium content of the fuel and then use the fuel in a separate reactor without intervening chemical reprocessing. Boosting could be done over and over, limited in principle only by materials considerations.

Another version which I call a fuel stretcher would take spent fuel from one reactor and then use it as a subcritical multiplying system driven by an accelerator in a second system.

Electronuclear boosting has many attractive features in principle, and the concept must now be carefully assessed to see if it really does present a viable alternative to the fast breeder. A number of technical problems would have to be explored in this connection, principally the need to operate fuel and fuel cladding to levels of burnup not normally encountered with light water reactors.

Analysis to date implies that electronuclear boosting could serve to increase substantially the amount of electrical power per pound of uranium from the ground, but at a relatively high financial cost.

Some stretching of fuel supply for light water reactors could be accomplished through improved isotope separation processes. Almost all separation of isotopes of uranium is done by the gaseous diffusion process at this time. The separation factor in a single gaseous diffusion stage is low, and the consumption of electricity in isotope separation by this process is very high. For these reasons, diffusion plants are operated to leave an appreciable part of the initial U-235 in the low concentration tails. It is interesting to speculate on possibilities of mining the tails for the remaining U-235, by novel isotope separation methods. A number of kinds of isotope separation processes are now under development. Some attention continues to be given to electromagnetic methods, particularly those based on resonances. The possibility of using centrifuges to do the separation was considered during the Manhattan Project. Development of the process was abandoned then, but now the centrifuge is preferred for new installed capacity in this country, in several European countries, and in Japan. Several methods of isotope separation using lasers are under development. The Becker nozzle process, developed in Germany, is soon to go into commercial use in Germany, Brazil, and perhaps South Africa. Of all of these methods, only those based on use of the laser offer at this time some possibility of complete separation of the uranium isotopes at reasonable cost. It is still interesting to review several methods of isotope separation under development, because partial recovery of uranium-235 in the tails could be used to increase the amount of fuel available for light water reactors. Of course, we must recognize that any isotope separation process that works well is likely to be classified. This places limits on the extent of review that can be given.

Figure 2 is a schematic drawing of a short bowl centrifuge of the type developed by Zippe in the United States in the early 1960's. This is a countercurrent design, with a capacity of about 5 kilograms separative work per year and with a separation factor in the range of about 1.1-1.2. A centrifuge of this design might have the characteristics shown in my Table 2. The term "subcritical" in the table refers to the length of the centrifuge. A longer machine might be subject to excitation of higher transverse vibrational harmonics as it spins.

The electrical consumption of a centrifuge plant would be very much less than that of a gaseous diffusion plant. This tends to offset the high capital cost of a plant containing as many centrifuges as the table shows.

One of the schemes for laser isotope separation depends on selective ionization of U-235 relative to U-238 in uranium metal vapor. The scheme depends on use of two lasers, a xenon laser to excite the U-235 atoms to a discrete excited level, and a krypton laser to provide subsequent excitation to the continuum of total ionization. The laser line width is very narrow compared to the line width of excitation probability vs. energy, as Figure 3 shows. This leads to the speculative possibility of nearly complete selection of one isotope over the other by the ionization process. Figure 4 shows in principle how ionized atoms or uranium-235 might be removed from the original metal vapor through use of an electric field.

My next figure (number 5) shows the principle of the Becker nozzle, which closely resembles a centrifuge except that in the Becker process it is the gas that moves and not the nozzle. Gas injected through the entrance orifice of the nozzle under high pressure would be deflected in a curved path in which the fraction of gas near the wall contains less U-235 and more U-238. A knife edge at the exit of the nozzle would provide a physical separation of the stream into the two halves. A single diffuser would have the general appearance shown in my next figure (number 6), with the possibility of several parallel or series stages in one machine. Typical values of operating parameters are shown in Table 3.

The Becker nozzle appears to offer somewhat larger separation factors in a single stage than gaseous diffusion, but plants with comparable capacity would consume comparable amounts of electric power.

In summary, a number of avenues are being explored to determine potential ways of continuing the supply of electricity to the United States in years to come. Many of these ways rely on nuclear fission. The principal technical problem is probably that of assuring use of light water reactors for as long a period as possible, but alternate courses of action are also being studied.

TABLE 1

A L T E R N A T I V E S

1.	FAST BREEDER
2.	THORIUM CYCLE
3.	D_2O REACTORS
4.	BETTER ISOTOPE SEPARATION
5.	ELECTRONUCLEAR BOOSTING

TABLE 2

NOMINAL CHARACTERISTICS, SUBCRITICAL CENTRIFUGE

OPERATING MODE	SUBCRITICAL
ROTOR MATERIAL	ALUMINUM
TEMPERATURE (T)	300° K
ROTOR LENGTH (Z)	51 INCHES
ROTOR DIAMETER (α)	8.5 INCHES
ANGULAR FREQUENCY	3500 RAD/SEC
PERIPHERAL VELOCITY ($\omega\alpha$)	380 M/SEC
IDEALITY EFFICIENCY (Ei)	0.814
CIRCULATION EFFICIENCY (Ec)	0.90
FLOW PATTERN EFFICIENCY (Ef)	0.62
SEPARATIVE CAPACITY ($\delta\mu$)	3.3 kg SWU/YR
SEPARATION FACTOR (α)	1.20

TABLE 3

NOMINAL PARAMETERS, BECKER NOZZLE COMPONENTS

UF_6/H_2 MIXTURE WITH 5 MOLE % UF_6

INITIAL NOZZLE PRESSURE	600 TORR
LIGHT, HEAVY GAS PRESSURES	150 TORR
GAS TEMPERATURE	40° C
UF_6 CUT	1/3
SEPARATION FACTOR (URANIUM)	$\alpha = 1.015$
RADIUS OF DEFL. GROOVE	0.1 MM
WIDTH OF NOZZLE THROAT	0.03 MM
WIDTH OF SKIMMER THROAT	0.02 MM

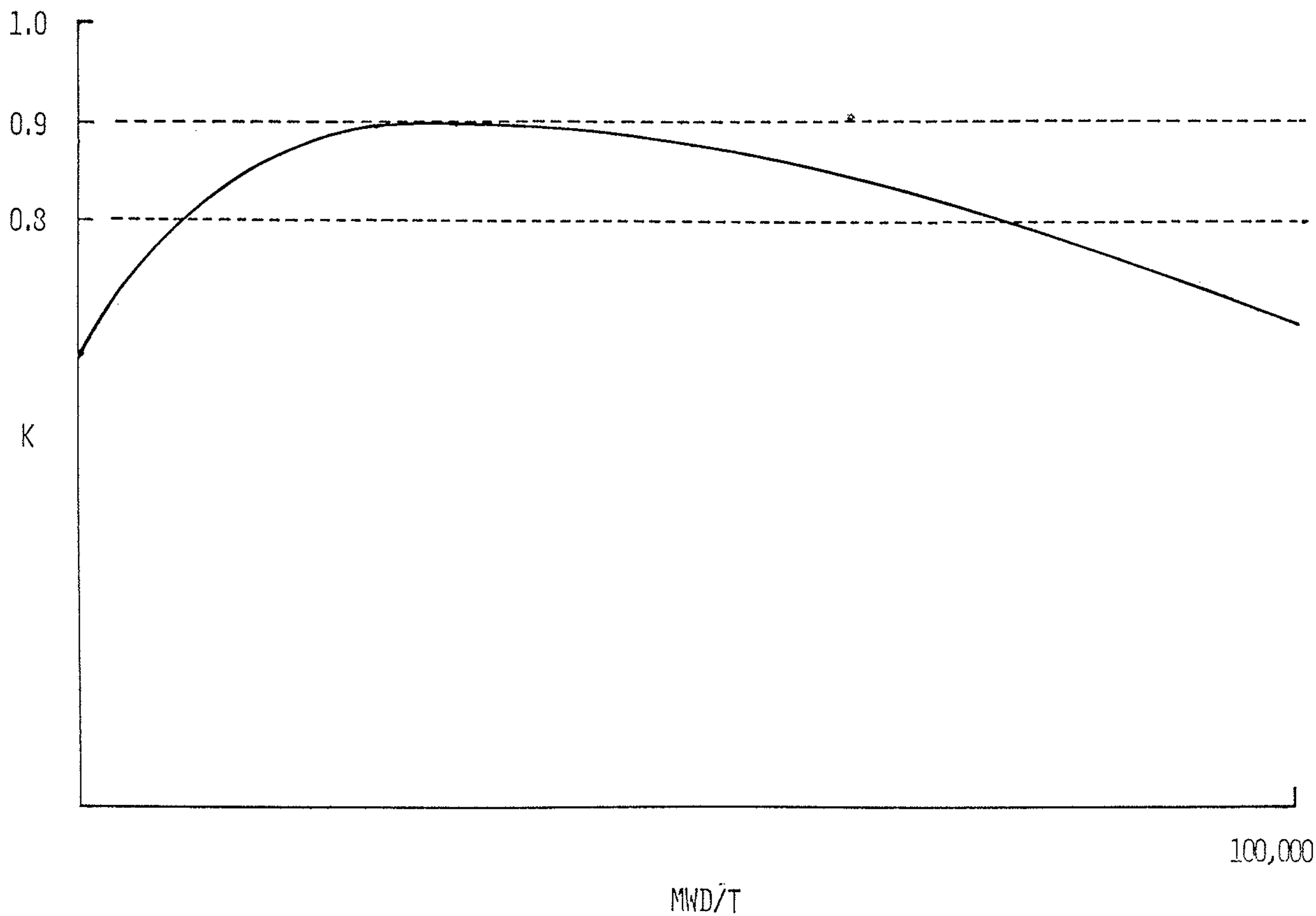


Figure 1 - Reactivity of a Driven Water Reactor

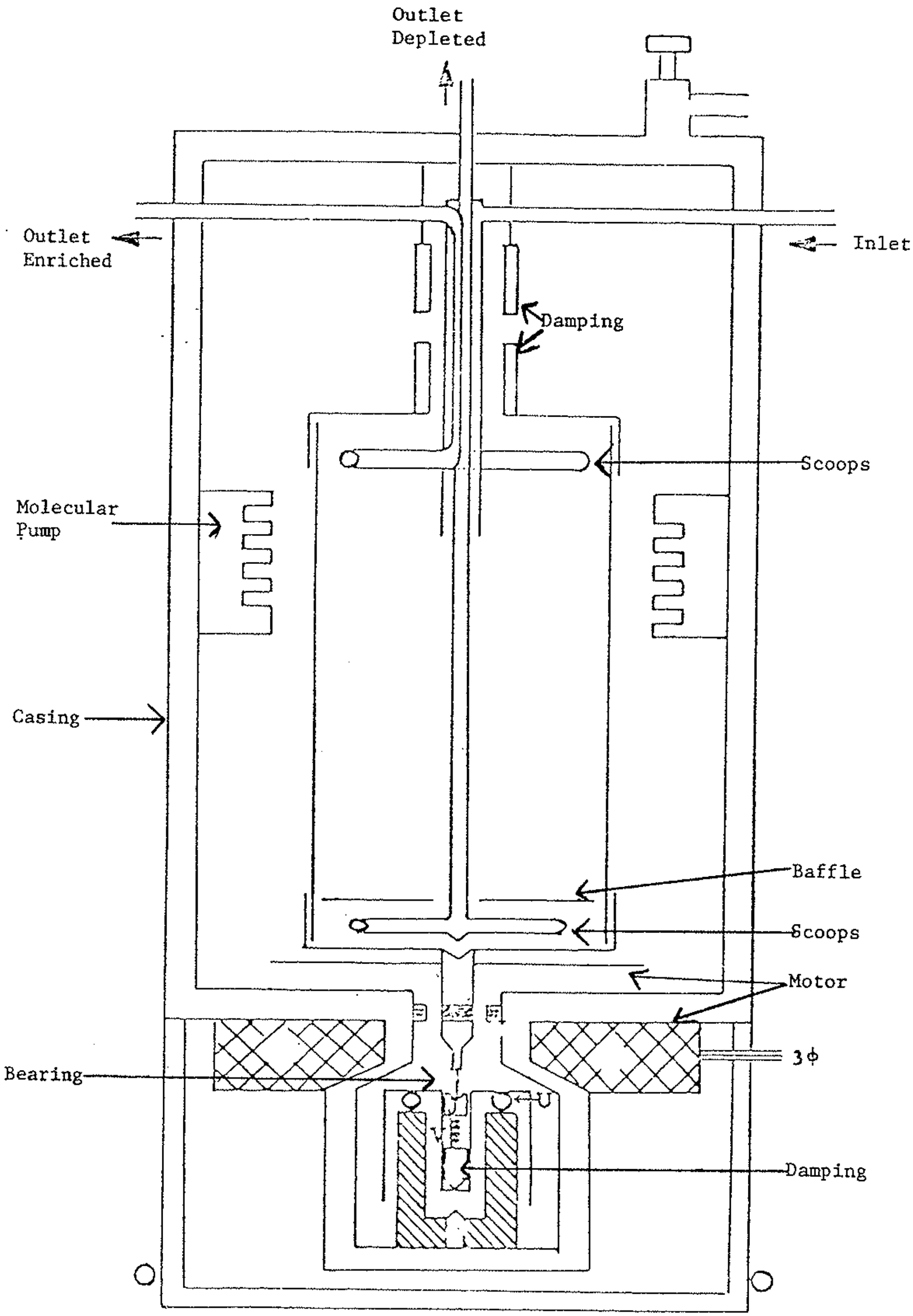
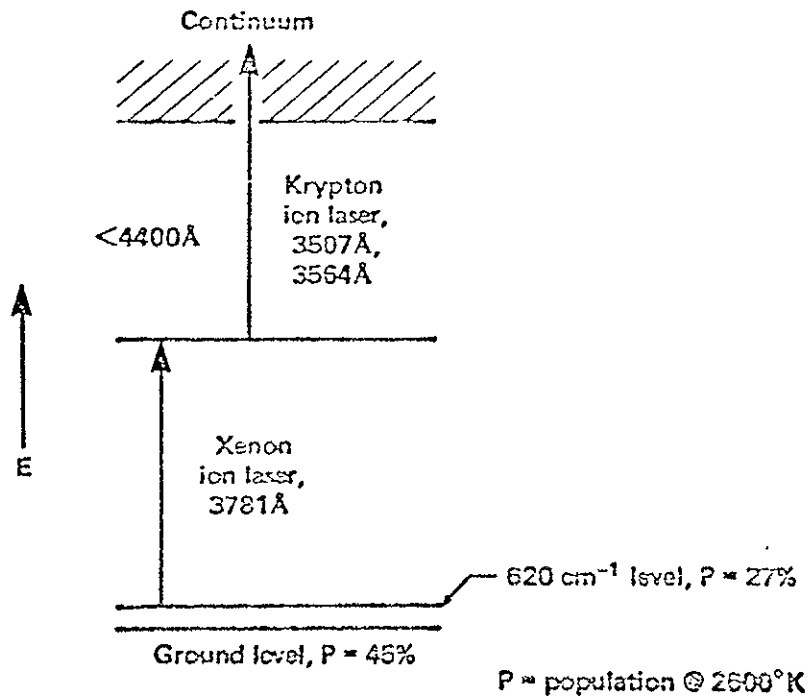


Figure 2 - SHORT BOWL ULTRACENTRIFUGE

PHOTOIONIZATION, ENERGY LEVELS



Xe EMISSION VS URANIUM ABSORPTION

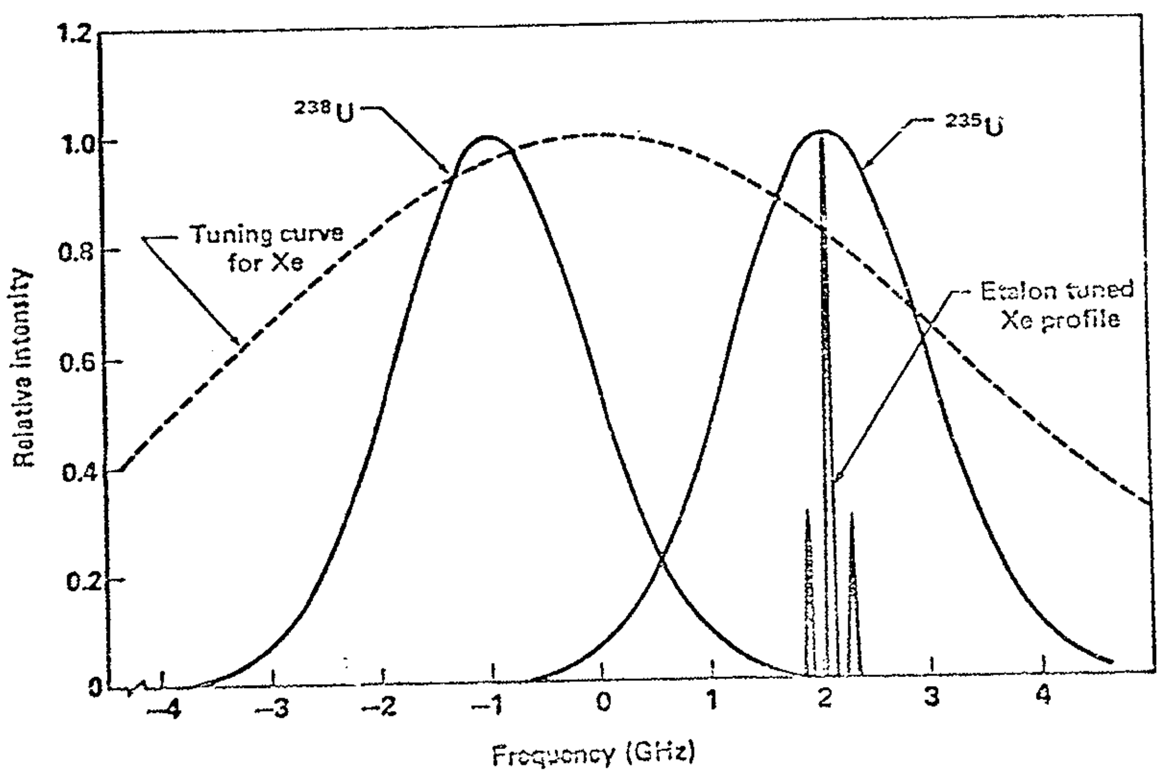


Figure 3 - Basis of Laser Isotope Separation

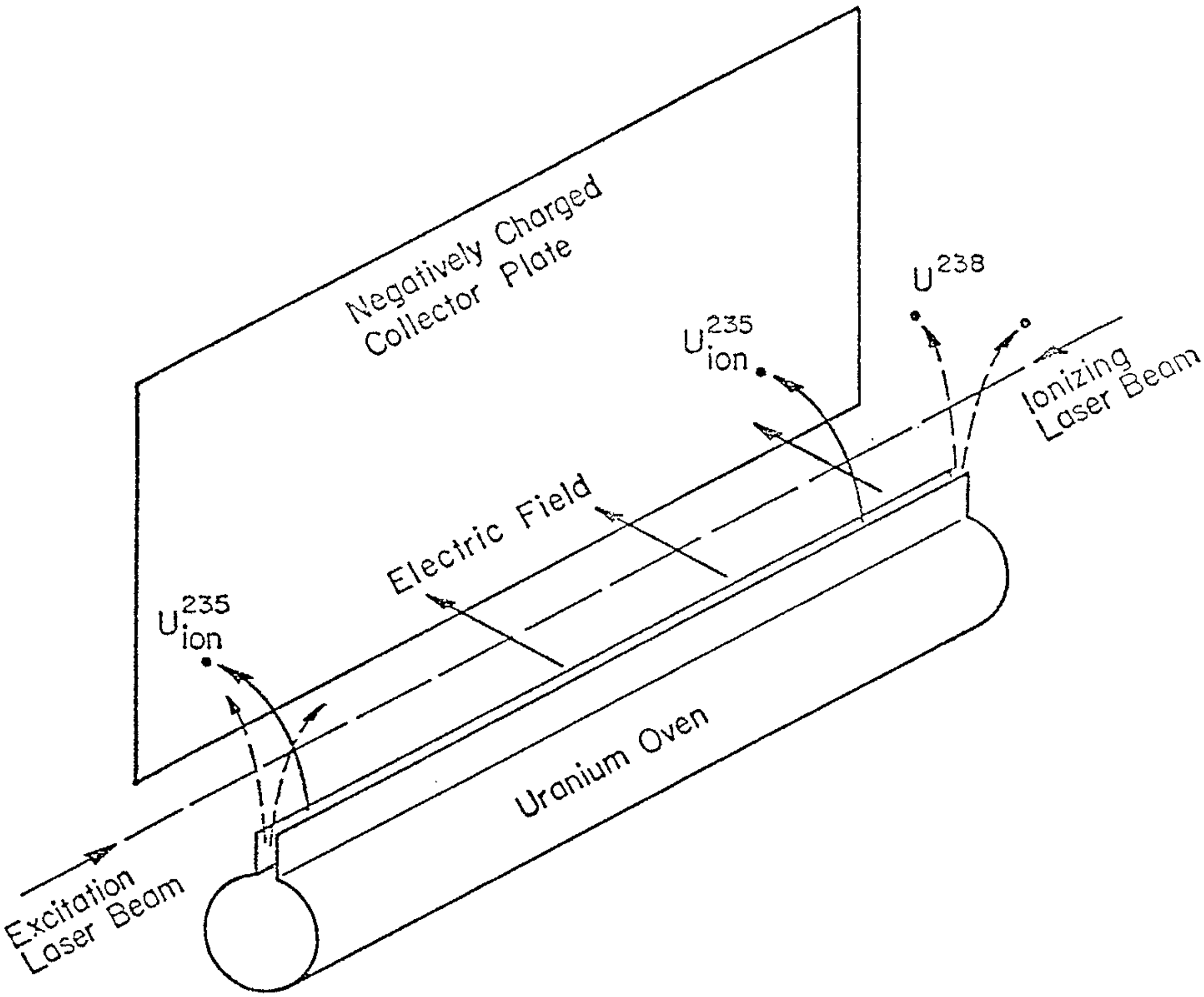


Figure 4 - Method of Separation in Laser Principle

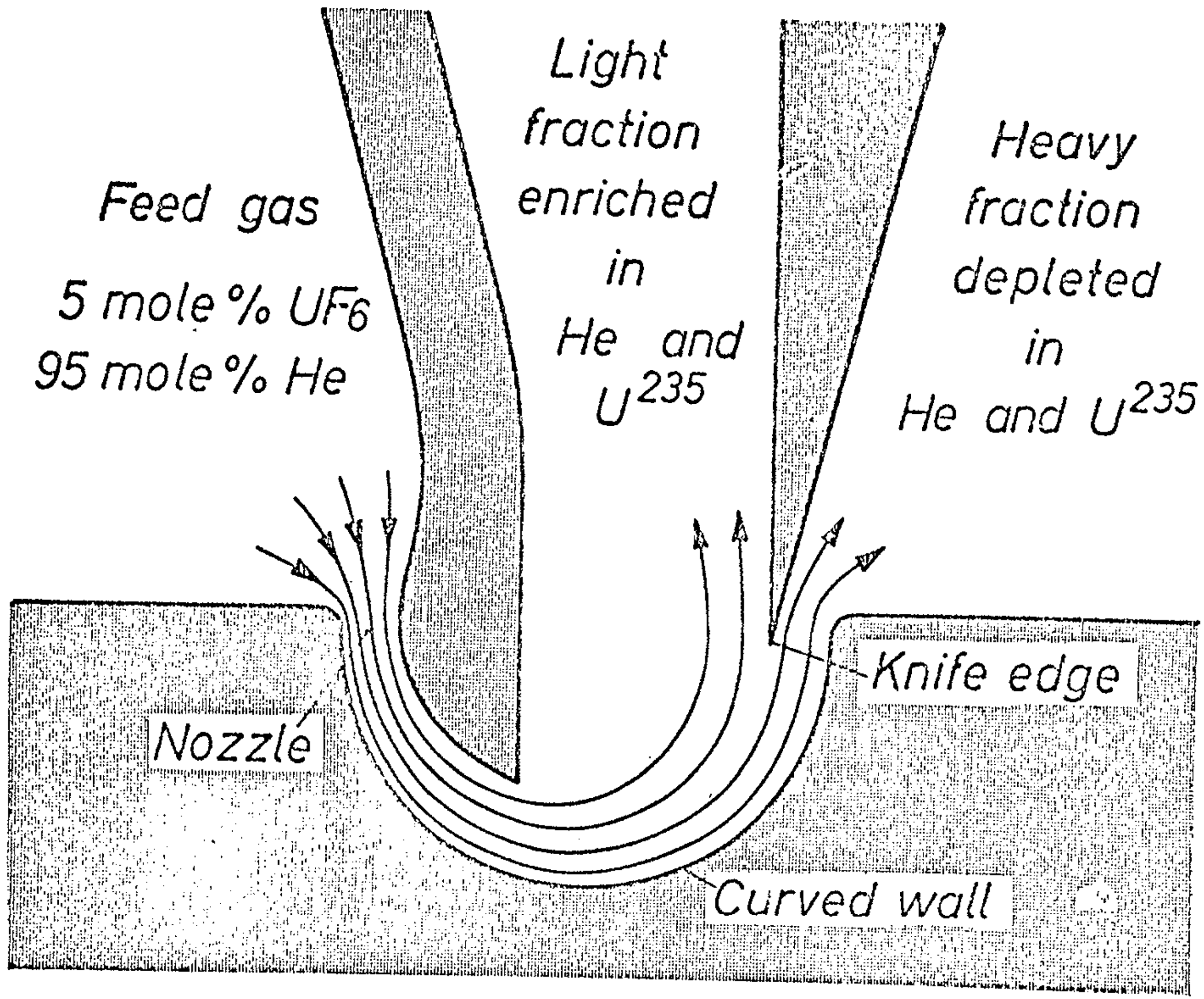


Figure 5 - Basis of Becker Nozzle Method

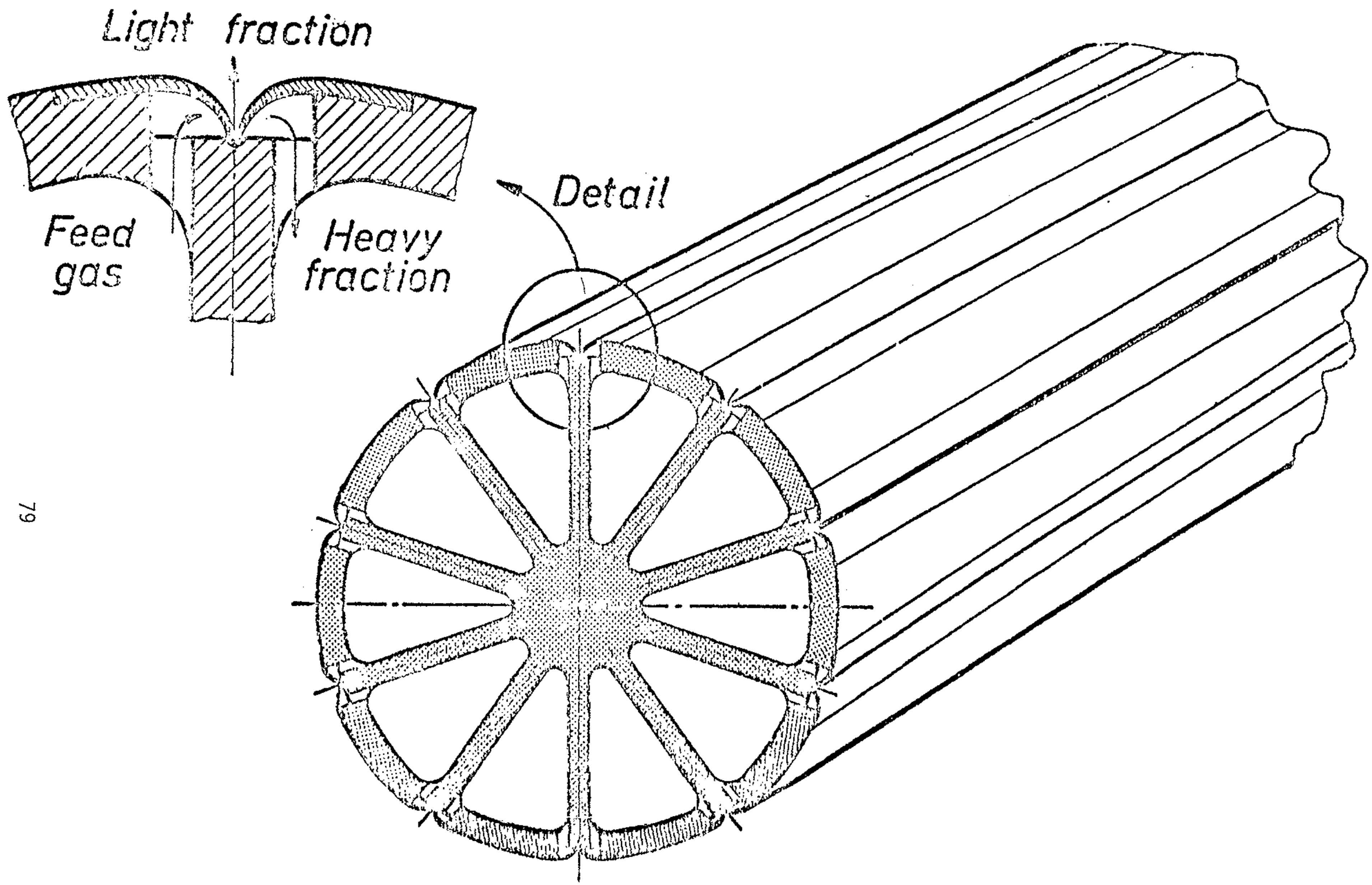


Figure 6 - Becker Diffuser

MR. WILLIAMS: Thank you very much. It is time for questions.

DR. GRACE: Nelson Grace, DOE. Herb, with regard to your natural uranium driven system, were those calculations zero dimensional, or did you consider the problems of non-uniform breeding, nonuniform depletion, taking into account the point of entry of the beam and so forth?

DR. KOUTS: These are zero dimensional calculations, Nelson, and we are in the early stages of this project. We have a one-year project going under Saul Strauch's program. At the end of the year we will give you some three-dimensional calculations, too. Power peaking is certainly going to be a problem.

DR. HURWITZ: Henry Hurwitz, from GE.

I note that the LMFBR is criticized because it has a low breeding gain and a high inventory. Yet the thorium reactors have a still lower breeding gain, possibly negative, and an equally high inventory. Why aren't they considered to be subject to these objections?

DR. KOUTS: Just because of an oversight while I was talking, Henry. One of the principal problems associated with most of the thorium reactors is the large uranium-235 inventory, and the problem of paying back that inventory is one of the real problems in fuel cycle analysis.

To take as an example, the light water breeder estimates I have seen indicate that it would take anywhere from 25 to 50 years to pay back the separative work investment in the light water breeder, that has to be made to get it going in the first place.

DR. LEE: J.D. Lee, Livermore.

I would like to know why you feel that the breeder and the thorium cycle are alternatives to the hybrid, while in our work we assume that these types of reactors are ones that we would supply with the hybrid?

DR. KOUTS: Well, as I said, I was using the word "alternative" very loosely at the beginning. Of course, you could run these devices without a hybrid, and it was in that sense that the word "alternative" was used.

There is, of course, interest in the Administration in how far we can go in avoiding reprocessing; what are the possibilities for avoiding reprocessing and what are their implications.

One of the possibilities for avoiding reprocessing, or at least avoiding it for a substantial period of time, is to operate with a reactor in which you burn the plutonium which is produced in situ without the intervention of a reprocessing step. We have picked up this idea, and have decided to explore it to its logical conclusion to see what the implications are. It may turn out to be a bad idea. It has certainly a number of strikes against it, and these may be enough strikes to shoot it down. But we have decided to follow it up, and there is enough interest in it to follow it up to determine its good features and its bad features. That is what we are doing.

FUSION-FISSION SYSTEMATICS

by

L.M. Lidsky

Department of Nuclear Engineering

Massachusetts Institute of Technology

Cambridge, Massachusetts 02139

In the next day and a half, we are going to see ten to possibly a dozen or more fusion-fission systems of one sort or another. It will become obvious as this meeting goes on that we have not yet converged on the optimal solution. Some will say that we haven't even converged on a statement of the problem. We will be discussing systems that in some cases share many common features and in others, almost none at all. I suspect that will be pointed out by some of the speakers by tomorrow afternoon that the systems we will be looking at in detail at this meeting do not bracket all the possibilities, but that in fact there is room for both extrapolation and interpolation.

Is there any way in which we can begin to think of fusion-fission systems in some systematic fashion? Is there, in other words, some generic classification scheme that will allow us to discuss this broad field in as logical a fashion as possible?

I have been asked by Locke Bogart to describe to you a scheme that we have found useful for classifying the various source of proposed fusion-fission devices. No systematics can be imposed arbitrarily, but must in one way or another reflect the underlying physical systems and, therefore, I will start then by showing you the nuclear physics that underlies our classification scheme.

I will stipulate, ab initio, the use of the DT fusion cycle. Other possibilities exist, some with very significant advantages, but the vastly greater reactivity of the DT reaction at plasma conditions we hope to achieve in the near future far overweighs all its inherent disadvantages. Now it is well known that, because tritium does not occur in nature, it is necessary to close the DT cycle by tritium regeneration through the Li-6(n,α) reaction. Thus, we have the conventional picture of the DT cycle (see Fig. 1a), one in which deuterium Li-6 are consumed with tritium as the effective catalyst for the reaction. The catalyst is regenerated by the neutron shared between the cycles.

The key to the problem is the realization that the neutron that is emitted in the DT reaction is fundamentally different from the one that enters the lithium six reaction. The DT neutron has 14.1 MeV energy which endows it with some rather spectacular properties. The neutron in the lithium six reaction is thermal (fractional electron volt energy) and is a rather common sort of neutron.

The extraordinary properties of the 14.1 MeV neutron are essential to the success of a fusion-fission scheme, indeed to any pure DT fusion scheme because the reactions illustrated in Fig. 1a are a physicist's dream, not an engineer's reality. In fact, of course, some of the neutrons in the reactor are absorbed in materials other than Li-6 (parasitic absorbtions) or simply escape from the system (leakage). Further, it is not true that we can extract every single atom of tritium that we generate, some escape, some decay and some are irretrievably bound into the system.

The standard solution to this problem is illustrated in Fig. 1b. The high energy neutron is capable of undergoing a reaction with Li-7 which generates tritium without loss of neutrons. The fact that two isotopes of the same material allow both for excess tritium production and very efficient neutron capture for thermalized neutrons is a marvelous bit of luck. The Li-7$(n,\alpha)n'T$ reaction then produces the extra tritium necessary to make up for neutron losses and gives us back our original neutron at a cost of only 2.8 MeV.

It was not very long after we had come to the realization that we had the possibility of generating enough extra tritium to account for neutron losses then it began to dawn on us that we had the possibility to in fact overcompensate and generate excess neutrons, even in relatively simple systems. In fact it became quite clear that if one were to go so far as to allow fission reactions then quite extraordinary potential gains could be considered. Let me start, however, by showing you some of the rather simpler things one can do with fusion neutrons.

Figure 2 shows the neutron energy spectrum as a function of position in a fusion reactor blanket. The zero point on the abscissa corresponds to the surface of the blanket facing the plasma. The various materials of construction are illustrated on the figure. The first centimeter or so is composed of niobium or some other refractory structural material. The next half or so is a combination of lithium with sufficient structural material to contain it and to control its flow, followed by a graphite region to serve as moderator and reflector, and finally a thin layer of lithium to absorb those few thermal neutrons that manage to leak out the back end of the reflector.

The 14 MeV flux is, of course, highest at the vacuum interface [note that the ordinate is logarithmic and, therefore, that the 14 MeV flux falls off by almost 4 orders of magnitude in passing the blanket]. The refractory metals that one would like to use for a first-wall have substantial (n,2n) cross-sections at high energies, above 8 MeV or so. At 14 MeV, the cross-section is on the order of about 1 barn. This is equivalent to saying that we generate approximately 5 percent excess neutrons per neutron passing through the first-wall (i.e., o.5 neutrons per fusion event). The thermal neutron absorption cross-section for some of these materials is relatively high but note that the thermal neutrons are generated and absorbed at the rear-end of the blanket. A refractory metal first-wall then can supply a substantial neutron excess, typically 8% or so in realistic blanket designs.

The thermal flux, shown here at the 1.4 eV level, peaks at the back of the blanket. It is in this region that one chooses to have a high concentration of Li-6 in the system for competitive absorption vis-a-vis the structural materials. The average neutron energy in this region is too low to provide appreciable Li-7 reactions which take place preferentially, of course, in the region immediately behind the first-wall. The spatial variation of the average energy of the neutrons in the blanket thus furnishes us with a relatively straight-forward way to design a blanket in which different regions perform different functions.

When realistic blanket designs are investigated in detail with proper attention paid to the detailed behavior of the neutrons fluxes and the most accurate available cross-sections are used to compute reaction rates it can be shown that at least 1.4 tritons per neutron (e.g. 1.4T/n) can be produced. That's a much higher generation rate than we need. The tritium inventory of a fusion reactor is very low and a tritium to neutron ratio of 1.4 implies a fuel doubling time on the order of 2 or 3 days! It's clear, therefore, that we have something left over that a clever man can put to good use. One possibility, of course, is to use the excess tritium to fuel other fusion reactors but that use would saturate almost instantly. How then do we use the neutrons?

We could use them as thermal neutrons; for example, we could make the blanket somewhat thinner and let some thermalized neutrons leak out the back. We could quite simply then absorb these neutrons in either thorium or uranium. This suggests an amusing exercise. A gram of neutrons absorbed in thorium 232 would produce 233 grams of uranium 233 which at $40 a gram, roughly the current price, is worth approximately $9,000. Excess thermal neutrons are thus worth at least 4 million dollars per pound. As we shall see in a little while, excess high energy neutrons are worth substantially more.

We want then to sell these neutrons in one form or another. The economic impact of selling a $4,000,000/lb by produce is substantial. The best way to scale this is the realization that at a T/n-ratio of 1.4 the value of the excess neutrons is almost exactly equal to the worth of the electricity generated by the thermal energy release of the fusion reactor. In other words, one could double the income of a fusion reactor merely by selling its excess neutrons.

The simplest way to sell neutrons is to absorb them in fertile material to produce fuel for shipment to fission reactors. The excess neutrons in fusion cycles can then be used to make up for the deficit of neutrons in the fission reactor cycle. The higher the conversion efficiency of the fission reactor, the fewer excess neutrons are needed and, with high enough T/n ratios and efficient enough fission reactors, it is possible to turn well-explored thermal reactor systems (the HTGR or MSR, for example) into relatively short doubling time breeder systems.

Modern interests in fusion fission systems seems to have been sparked by a paper I presented in 1969 at the Culham Fusion Reactor Workshop. The conclusions of that study are shown in Figure 3. The sketch below the curve shows the fuel cycle for "symbiosis". Essentially the fusion reactors' neutron excess is used to breed fissile fuel for a fission reactor. This scheme is called symbiotic, a title that was particularly appropriate in the past when it was thought that the fission reactor would have to provide substantial amounts of electrical power to operate the fusion reactor. The curve shows the ratio of the fusion reaction rate to the fission reaction rate necessary to achieve a combination system of a given exponential fall growth for various relatively high conversion efficiency fission reactors. For example, if one wanted to achieve a doubling time of seven years in a combined HTGR-fusion system, one would require a fusion/fission reaction rate of 1.2 but because the fusion reaction reduces so much less energy than the fission reaction. The fusion to fission power rate would be only 0.13. In other words, if a 130 megawatt fusion reactor were operated in conjunction with a 1,000 megawatt HTGR fission reactor, then the resulting combination would constitute a breeding system with a doubling time of only 7 years. The net electric power production would depend, of course, on whether or not the fusion machine was a net energy producer or not. Other combinations are seen to be even more intriguing. The symbiotic combination of a fusion reactor with the molten salt reactor is particularly neutron efficient and offers a simple and very inexpensive fuel cycle with, of course, the complete absence of fuel fabrication costs.

The Canadians have looked at symbiosis. Figure 4 shows the reduction in natural uranium mining requirements (in units of grams mined per electric kilowatt year of power produced) for various fusion cycles. The mining requirement would be reduced by a factor of 6.5 if existing candu reactors were operated in conjunction with symbiotic fusion "fuel factories".

Very recently, Blinkiv and Novikov at the Kurchatov Institute have taken the idea of symbiosis yet another step [see Fig. 5]. They point out that if one really wants to minimize fuel costs and also avoid the very difficult problems of generating tritium in a highly complicated fusion machine, all one need do is generate the tritium in the fission reactor, the uranium in the fusion reactor and simply cross connect the fuel cycles of the two systems. They show that it is possible with this at first very highly surprising scheme to build quite reasonable molten salt reactors and, with a fusion/fission ratio of 0.1 to develop a system with significant technological advantages and a fuel doubling time of only 4 1/2 years.

Symbiotic schemes are classified as those in which one uses the excess neutrons gained by the (n,2n) and Li-7 reactions to generate fuel which is then consumed in fission reactors. The conversion ratio tend to lie in the 1.3-1.6 region. It is not economical to complicate this system with smaller conversions, nor is it possible to achieve higher conversions without one form or another of nuclear fission taking place in the fusion reactor blanket. The absence of fission in the fusion reactor blanket is the sole but extremely important advantages of such systems. The spectrum must be thermal if fast fission of the fertile material are to be avoided. The fuel generated in symbiotic systems is, therefore, of very high quality but it is important to emphasize that symbiotic systems are first and foremost systems. <u>The total gain is not very high</u> and the proper match of the fission and fusion fuel cycles and system efficiencies is essential in achieving economic systems.

Symbiosis
C* ∿ 1.3 ∿ 1.6
No In-Situ Fission
High Fuel Quality
"System" must be optimized

The 14 MeV neutron born in the fusion reaction has, I remind you, some extraordinary properties. I call your attention to Figure 6 which shows the fission cross-section as a function of energy for uranium 238 and thorium 232 and point out that it is quite substantial. It corresponds inU-238 to a mean free path for fission of only 10 centimeters. The (n,3n) corss-section for these materials becomes significant above energies of 12 MeV or so. However, let us concentrate for the moment on the fast fissions. The average number of neutrons per fission event rises from just slightly greater than two in the thermal and opithermal regime of fission reactors to values in excess of 4. Furthermore, most of these fission neutrons are themselves above the fast fission threshold. One sees extraordinary possibilities for muliplication of both neutrons and, at 200 MeV per fission, of the fusion reactor energy.

Many hybrids of fission and fusion have been suggested to take advantage of these possibilities. Almost all of them are based at the realization that if the advantages of fusion-fission hybrids are to be maximized then it is necessary to utilize the special properties of 14 MeV neutrons, i.e., one must get them while they are hot. The only place to do that is very close to the plasma interface of the fusion reactor blanket. i.e., at the "first wall".

Figure 7 is a typical high-gain hybrid design. In this case, studied by Su, Woodruff, and McCormick and his colleagues at the University of Washington, the neutrons enter the converter plate from the plasma region where prompt and second generation fast fissions take place. The high energy neutron spectrum is further exploited by a thin beryllium multiplier and then the neutrons are permitted to enter a thermalizing region where additional fission occurs and the resulting neutrons are captured in thorium to breed U233 and in lithium to regenerate the re-

requisite tritium. The multiplication in the coverter region is so high that nearly two U233 nuclei are made available for external use per fusion reaction and the fusion reactor is multiplied 65 fold.

This hybrid reactor utilizes a converter plate and then a region where the neutrons are thermalized. The design feature is almost universally adopted. A properly designed converter plate will be undermoderated because of the desire to use the high energy properties of the neutrons; optimized systems will utilize uranium metal and helium gas cooling. The converterplate regions tend to have very high energy density (much as an LMFBR and for the same reasons). The average power density in the neutron converter plate shown here is 210 kilowatts/liter. The converter plate tends to be the weak point in many hybrid designs because of the very high energy density and associated engineering and safety problems.

Hybrid designs eliminate at least one safety hazard. There is no possibility of a loss of coolant accident [LOCA]. The reason for this is that the source neutron at 14 MeV has much higher "value" in terms of producing secondary neutrons than does the neutrons resulting from fission or (n,3n) reactions. Thus, although the energy gain for source neutrons is very high, the multiplication for secondary neutrons is much smaller and, the system would shut itself down very quickly if the source neutrons were removed. In other words, the multiplication for 14 MeV neutrons is very high even though k_{eff} as ordinarily computed is well below unity. The system is in more common terms very definitely sub-critical. In the system shown here, for example, k_{eff} is only 0.944. Other designs tend to have lower values. Because the source neutrons are so different from the secondary and subsequent generations of neutrons it is claimed often, and with perfect justification, that well-designed hybrid systems operating at very high energy multiplications are nonetheless very definitely sub-critical.

There are many possibilities. The neutron multiplication can take place in uranium or thorium and the subsequent absorption to produce fuel can likewise be made to occur in either uranium or thorium. It seems clear that at least from a neutronic point of view the best material for the converter plate is U-238; the best material for the thermal adsorption is open to question and various workers claim one or another version is superior. Figure 8 shows the results of a Livermore Laboratory study done several years ago to directly compare uranium and thorium based hybrid reactors. This particular blanket was developed for use with a mirror fusion reactor. Note that the better design promises an energy multiplication of 11 and a converter plate power density of 150 watt cms.

Figure 9 is a tabulation of the more important performance parameters of 3 blanket designs. We will see somewhat more modern versions of similar systems in the next day and a half, but I suspect that the numbers won't vary very much from these. One observes total energy multiplications on the order of seven through more than eighty with ten being the value achieved in the most recent more carefully worked out designs. The multiplication is ultimately limited by cooling of the converter plate region although in the University of Washington design, there was substantial energy generation in the fission behind the plate. The inset summarizes the range of performance parameters one can expect in hybrid fusion fission schemes. The inherently high power density in the converter plate is the design limit and raises some very real questions in the event of loss of coolant accident--a particularly likely event in the case of the helium-cooled system.

<u>Hybrid</u>

- Fusion Power Gain
  ($M \sim 3-30$)
- High Specific Fuel Production
  ($C^* \sim 3-10$)
- $k_{eff} < 1$
- Fuel Quality?
- High Power Density (LOCA)

Fuel quality is listed with a question mark. The problem is that the fusion neutron is capable of many inelastic ((n,2n), (n,α)) events. Thus, one has a wide variety of starting nuclei for a series of absorption and decay chains and the resulting witch's brew is very sensitive to spectral shape, flux level, and decay lifetime. This problem is being very actively considered by Bo Leonard and his group at Battelle Northwest Laboratory.

How does one compare symbiotic and hybrid systems? The symbiotic systems are inherently simpler and safer but they have lower outputs and fissile production rates and must be used in a carefully tailored economy, properly matched with their symbiotic partners. The hybrids have much higher output but are saddled with severe engineering difficulties, complicated fuel cycles, and serious perceived political disadvantages. One of my students and I attempted to make such a comparison last year. Figure 10 illustrates one of the results of that study. We plot the maximum allowable capital cost of various hybrid symbiotic and pure fusion systems as a function of the power gain in the fusion reactor core for a given return on investment. If the power gain in the fusion reactor core is high ($Q \gtrsim 10$) then it seems the best solution is to build pure fusion reactors.

If the power gain is very low, less than 0.5 or so, then one has no choice but to build hybrid reactors if one is to introduce fusion neutrons into the power chain. In the intermediate region ($0.5 < Q < 10$) the question is quite complicated, depending on existing fuel cycles, long range plans for reactor mixes, existence of one form or another of fission reactor, questions of proliferation, etc. The detailed economics of this regime is under intensive investigation right now but the answers are not yet in.

There is another use for fusion neutrons that has been seriously considered. This use, mentioned by title this morning, needs be fitted into our systematics. I refer, of course, to the possibility of using fusion neutrons to destroy the waste products of fission reactors. Fuel cycles which use fusion neutrons to destroy fission reactor waste have been named Augean.

There are two possible ways to take advantage of the fusion neutron excess: One can burn out the actinides and thus alleviate the long term waste storage problem, or one can burn out some of the more troublesome fission products and so eliminate much of the short time high activity wastes.

The physics of Augean systems tends to be relatively straight-forward and leads unfortunately to some rather discouraging results. What one wishes to do, of course, is destroy the troublesome waste isotope at a rate much faster than its natural decay rate either by causing them to fission or transmuting them to some rather more benign nucleus. However, as one can guess, this problem is difficult because if the particular nuclei in question were easy to burn out, they would have been destroyed in the fission reactor itself. Therefore, one has to take advantage of the somewhat larger cross-section at very high energy for some of the nuclei or try to achieve very high flux in the fusion reactor. The studies done so far show that in order for the burn out time to be comparable to the natural decay rate requires a very high flux; a higher flux than we know how to get in a fusion reactor. They further show that to avoid an unsupportable waste of neutrons, it is required that the fission product wastes be isotopically partitioned to avoid neutron absorption in nuclei that are not particularly troublesome. This isotopic separation of high level activity is a difficult and expensive project. When finally you realize you must then store the fission reactor wastes in the process of being burned out in the blanket of an operating fusion reactor, we begin to see the difficulties inherent in this scheme.

Augean utilization of fusion neutrons thus, although not impossible, does look to be very difficult indeed. The insert summarizes the essential properties of these systems.

<u>Augean</u>

Actinide, F.P. Burnout

$$\frac{dN}{dt} = -\lambda N - N\sigma\phi$$

very high flux required

iosotopic partioning required

I will conclude with Figure 11 illustrating the Systematics that has been found to be most useful for classification. There are only thermal spectrum symbiotic schemes because fast fissions would be unavoidable in hard spectra and fissions are by definition not allowed in symbiosis. Although both the fast and thermal spectrum hybrids have been listed, it is becoming quite clear that the fast spectrum hybrids have substantial advantages over the thermal branch. The reason for this is quite obvious in retrospect. Anything that one can do with a thermal spectrum in a fusion reactor, one can do equally well with a thermal spectrum in a fission reactor and at least for the foreseeable future, the fission reactor will be a far simpler device. The fast hybrid takes advantage of the special properties of the fusion neutron; the thermal symbiote does a similar thing by using the fusion neutron at 14 MeV for initial multiplication but then rapidly thermalizes it to avoid fissions. Finally, I point out that the behavior of the cross-sections are such that the fast spectrum is most useful for fast fission of actinides whereas both fission products and the actinides can be destroyed by thermal neutrons.

Question: Bo Leonard, Battelle Northwest

With regard to Augean systems and the problem of developing sufficiently high flux in a CTR, one concept that has not been explored that may be able to do that is the laser selenoid where you have multiple line sources in a cylindrical arrangement. This might allow you to build a thermal flux trap similar to those built in fission reactor and this might make Augean systems feasible.

Answer: Dr. Kidsky.

I agree, Bo. In fact your question illustrates the point of trying to develop systematics because it highlights the necessary physical regimes of operations for various end results. I agree with your conjecture that a flux trap system might work and that something like that is certainly necessary.

Question; J.D. Lee, Livermore Laboratories

Larry, you showed a range for the fast systems of total tritium breeding plus fissile production of 3 to 10. I've never seen the design yielding anything nearly that high. Can you describe such designs?

Answer: Dr. Lidsky

You are quite right, J.D. Those designs that have been more carefully worked out have top breeding gains much more nearly equal to 5, the University of Washington design for example. However, one is dealing with a driven subcritical reactor and it is possible to get the criticality coefficient quite nearly equal to 1 in the converter plate for high multiplication and then follow it with a near critical fission lattice. Some of the earlier schemes proposed to do precisely that.

a)

$$D + T \rightarrow He^4 + n + 17.6\ MeV$$

$$Li^6 + n \rightarrow He^4 + T + 4.8\ MeV$$

$$D + Li^6 \rightarrow 2He^4 + 22.4\ MeV$$

$$Li^7 + n \rightarrow He^4 + T + n' - 2.8\ MeV$$

b)

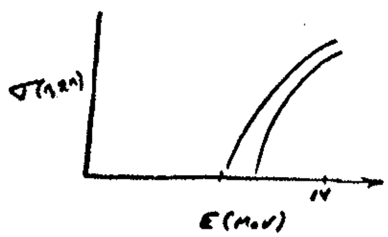


Figure 1

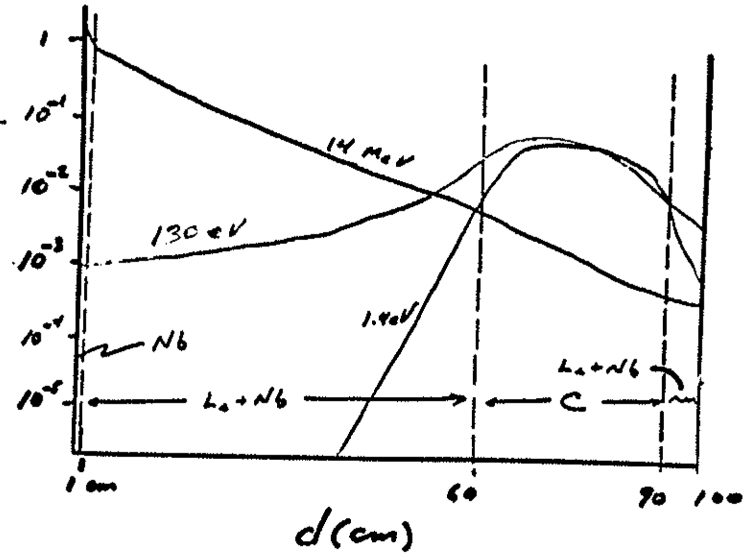
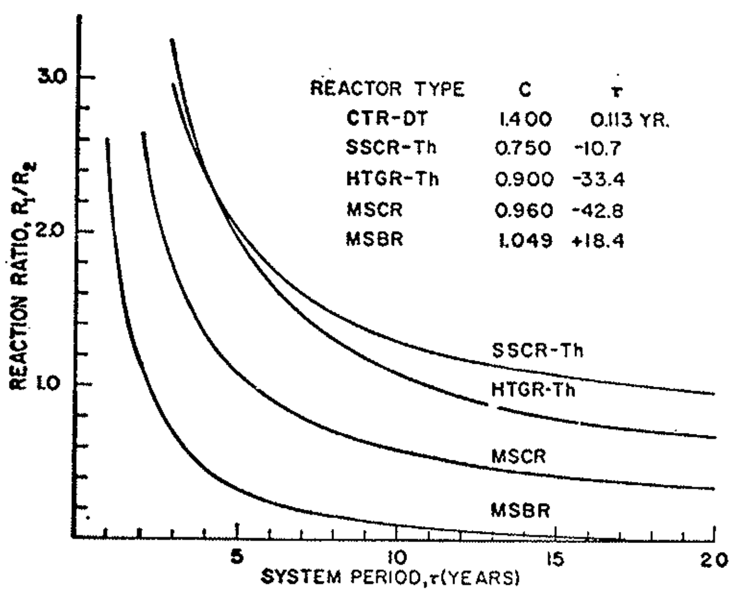


Figure 2



Fusion/Fission ratio in balanced hybrid power plants as a function of system period

Figure 3

SYMBIOSIS - CANDU + CTR

Nat U Mining Requirements (gms/ekw-yr) *

Nat U - once through	130
Nat U - Pu recycle	60
Th - U^{233} recycle-U^{235} makeup	20

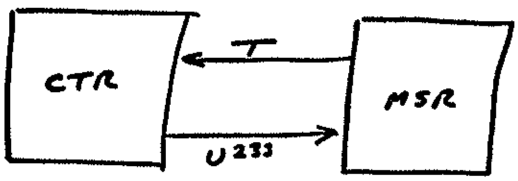
$\frac{W_{FUSION}}{W_{FISSION}}$ for U^{233} ~ 0.1 **

Figure 4

* AECL-4840 (1974)

** Nuc Fusion 15, 151 (1975)

V. L. BLINKIN & V. M. NOVIKOV : KURCHATOV



NaF - 71 %	LiF - 50 %
BeF_2 - 2	BeF_2 - 50 %
ThF_4 - 27	$^{233}UF_4$ - 0.1 %

AT $\frac{W_{FISSION}}{W_{FUSION}} = 10$, $\tau_{1/2} \sim 4.5$ yrs

Figure 5

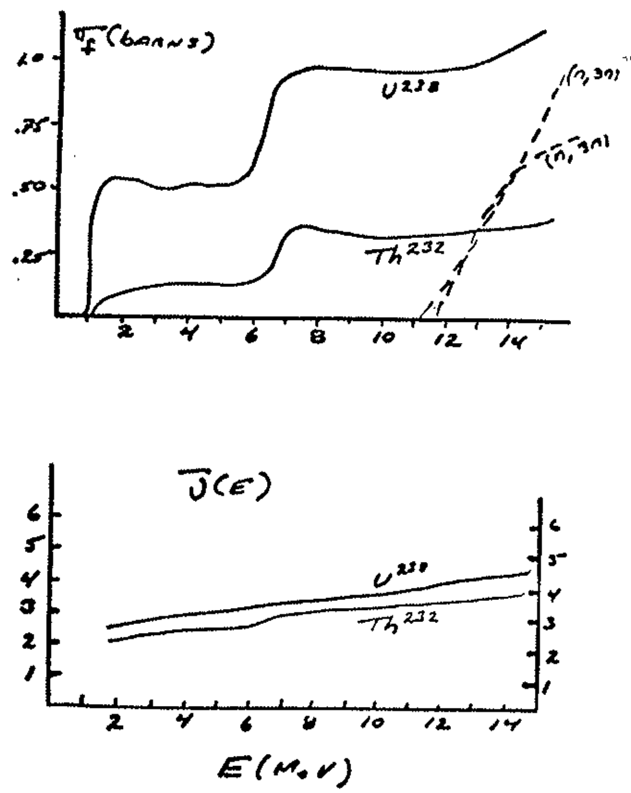
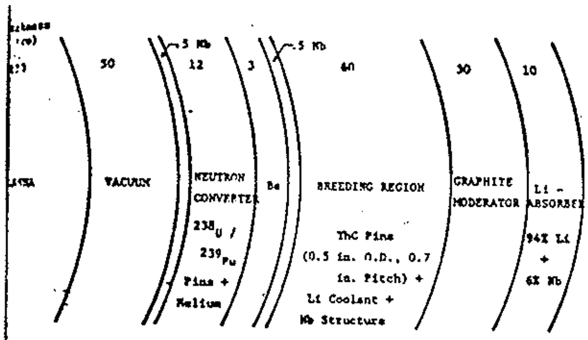


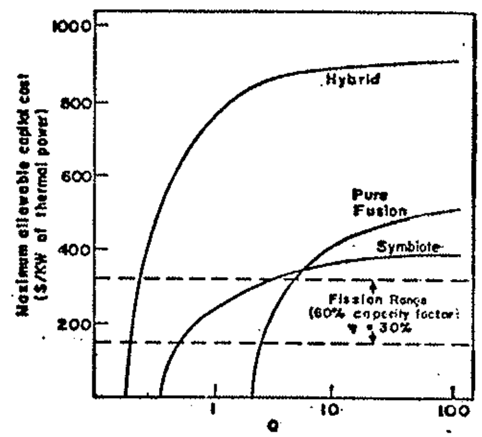
Figure 6



Schematic of the High Gain Blanket Design

Su et al

Figure 7



Assumptions:
1) ^{233}U = $100/g
2) η = 45% = thermal efficiency
3) 100% capacity factor

Maximum allowable capital cost per KW of thermal power versus Q, the fusion reactor gain (current fission reactor costs are also plotted)

Figure 10

HYBRID REACTOR PARAMETERS

	Mo	TH
MIRROR RATIO	2.50	2.75
INJECTION ENERGY (KEV)	100	100
CONDUCTOR FIELD (T)	8	12
Q	0.68	0.76
FUSION POWER (MW)	474	1500
FIRST WALL FLUX (MW/M^2)	1.3	4.2
BLANKET THERMAL POWER (MW)	4220	3340
ELECTRICAL OUTPUT (MW)	1044	-40
DUTY FACTOR	0.76	0.73
MIRROR-TO-MIRROR LENGTH (M)	15	15

HYBRID BLANKET PARAMETERS

	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	0.6
BLANKET EXPOSURE (MW-YR /M^2)	4.1	9.2
FUEL POWER DENSITY (W/CC)	150	110

Figure 8

Performance Parameters of Three Blanket Designs

ITEM	Marin '74	Su, et al	L.L.L.
neutron first wall flux	0.89MW/m^2	0.5MW/m^2	0.84MW/m^2
fertile material	Metallic ^{238}U clad in Nb	ThC	UC
Breeding region coolant	He gas	Li	He gas
Multiplying region material	same as fertile material	92% U - 8% Pu metal	same as fertile material
Multiplying region coolant	He	He	He
average power density in the multiplying region	135 w/cc	210 w/cc	~ 20 w/cc but with peak values up to 300 w/cc
T	1.05	1.052	1.11
^{233}U	---	3.537	---
^{239}Pu	1.38	0.664	1.29
M	7.3	81	12
k_{eff}	<<1	0.94	<<1

Figure 9

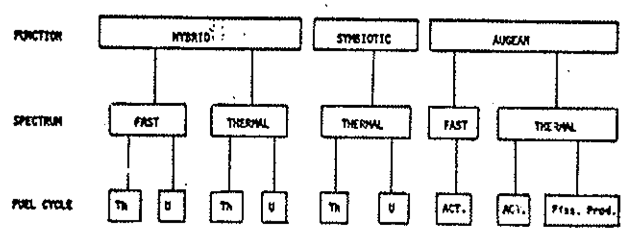


Figure 11

MR. BOGART: Larry, would you take questions from the audience?

DR. LIDSKY: Of course.

DR. LEONARD: Bo Leonard, Battelle Northwest.

With regard to the Augean system and the problem of building a sufficiently high flux in a CTR, one concept that has not been explored, to my knowledge, that may be able to do that, is the laser solenoid concept where you have multiple line sources in a cylinder which could allow you to build a thermal flux trap like we do in fission reactors, and which may make that concept feasible. That's the only one that I've run across.

DR. LIDSKY: I agree, Bo. In fact that, in a way, is the point of trying to develop a systematics because once you know where the physics leads you in Augean systems then, in fact, you are able to begin to decide what system might have potential for that. And I agree, a flux trap system like that may work and is certainly necessary.

DR. LEONARD: I thought about that with the laser solenoid a long time ago but, unfortunately, somebody wrote it down before I did.

DR. LEE: J. D. Lee, Livermore.

Larry, you showed a range for the fast systems of a breeding plus fissile material of 3 to 10. I've never seen one anywhere near 10. Could you describe that very briefly? What is near 10? How do you get that high?

DR. LIDSKY: One can get as high as one pleases, in a way, by going as close as one wants to a fairly high gain in the front, and then putting a fairly substantial near critical fission lattice behind it. And those are the systems that come that close.

DR. LEE: That's gross?

DR. LIDSKY: That's gross, yes.

UCRL 80694

MIRROR HYBRID REACTOR STUDIES

D. J. Bender
Lawrence Livermore Laboratory

Prepared for the Proceedings of the 2nd DMFE Hybrid Reactor Meeting,
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1. Review of Past Studies

The reactor studies group at Livermore has been involved in the conceptual design and analysis of the hybrid reactor, based on mirror confinement of the plasma, for over five years. Prior to 1975, some preliminary engineering analysis was performed [1,2,3], but a great deal of the work was concentrated on blanket neutronic studies, determining interaction of the 14 MeV neutrons with various assemblies of fertile materials, coolant and structural material.[4,5]

In 1975, our first point design study was completed [6]. It was a conceptul design on which we could build and begin to understand the way in which one would perform the different functions that were necessary in the reactor.

We devoted 1976, at Livermore, to optimization of our point design [7,8,9,10,11] and, at that time, General Atomic joined us, applying their expertise in gas-cooled reactor technology to our point design [10,11,12]. In this past year, we have concentrated our effort on a reference design, which will present a good illustration of the capability of the classical mirror hybrid reactor.

Our interest in the mirror hybrid has evolved to optimizing the reactor for fissile fuel production, the hybrid being part of a nuclear power system where it supplies makeup fissile fuel for five to ten fission converter reactors. Our interest in the fissile fuel producer has come from our economic analyses that have indicated that this is the most attractive hybrid system [9].

To summarize the status of our work, we've devoted a good deal of effort to the nuclear analysis of fast spectrum blankets, and at this point, we have established the nuclear performance of these types of assemblies (blanket multiplications and fissile production rates). Our systems studies work has

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evolved a set of "necessary conditions" for an attractive fissile fuel producing hybrid reactor. These conditions are (i) a plasma Q of 1-2, and (ii) a first wall fusion neutron loading of 1-2 MW/m^2. The hybrid reactors we've studied to date have been based on fuel cycles which employ reprocessing, and our optimization studies have indicated to us that these are low burnup fuel cycles (in the hybrid) and we, therefore, have been able to use metallic fuels. And, finally, we've gained some appreciation for the implications of fusion reactor geometry on the mechanical design of the machine.

2. Reference Design Study

We are completing our reference design [13,14] this year, based on classical or minimum-B mirror confinement and a fast fission U-238 blanket. The reactor Q is $\sim$ 0.6, and it achieves a first wall loading of about 1.7 megawatts per square meter of 14 MeV neutrons. The design appears feasible in the sense that we have not encountered any engineering constraints which we have not been able to meet. The economics could be improved in that the rather low plasma Q of this reactor requires recirculation of about 65% of the gross electrical power production to run the injector system, and this is an economic penalty. The spherical geometry of this reactor appears to be workable, but it is more complex than the right circular cylinder geometry of a fission reactor.

The physics demonstration for the plasma that we've used in the reference design will occur with successful operation of a machine called MFTF, the Mirror Fusion Test Facility, which is now under construction at Livermore and will begin operation in 1981. Thus the necessary physics demonstration will occur within the next five years and we can consider a technology development phase after that time.

We have constrained ourselves to a minimum extrapolation of the fusion technology which is now being employed in the experimental plasma physics program. We also used fission reactor technology which is now in use in the fission reactor industry, or is the subject of active development. Our design, then, is best described as an early commercial facility.

Figure 1 is an illustration of classical mirror confinement geometry. We use a Yin-Yang coil which creates a spherical region for plasma confinement inside of the magnet windings. In this confinement scheme, energetic plasma continually streams out through the "ends" of the machine, (the two fans on either end of the figure). To compensate for this loss of fuel and energy, the plasma must be continuously supplied with energetic neutral particles. When the injected neutral particles enter the plasma, they are ionized by collisions and trapped in the magnetic field.

It is these plasma constraints around which the reactor must be designed: the spherical geometry, energetic neutral particle injection into the plasma and the streaming of energetic plasma out the ends of the machine.

2.1 Summary

To summarize the reference mirror hybrid design, we list the major design choices that have been made for the reactor.

- Minimum-B mirror confinement
- Yin-Yang coil design, NbTi superconductor
- positive ion injectors with direct recovery
- fast spectrum blanket neutronics
- single-stage plasma direct converter
- cryocondensation vacuum pumping
- blanket
  - U_3Si fuel (depleted U)
  - LiH tritium breeder (natural Li)
  - Inconel 718 structural material
- He primary heat transfer loop (PHTL)
- Prestressed Concrete Reactor Vessel (PCRV)
  - magnet restraint
  - PHTL restraint
  - blanket support and restraint
- steam thermal conversion system

Characteristics for the reactor are listed in Table 1.

REFERENCE MIRROR HYBRID REACTOR PARAMETERS

TABLE 1

Fusion Power	400 MW
Thermal Power (Avg.)	3350 MW
Injected Neutral Power	625 MW
Net Electric Output Power	600 MW
First Wall 14 MeV Neutron Current	1.7 MW/m^2
Annual Fissile Production	2020 kg
Recirculating Power Fraction	0.65
Q	0.63

2.2 Fusion Core Design

The variation of the basic Yin-Yang magnet, developed for reactor applications, is shown in Figure 2. The magnet has an outside diameter of amout 22 meters, and a distance of 13 meters between mirror points. It is designed with a maximum field at the conductor of 8.5 Telsa, dictated by the use of NbTi superconductor. The maximum current density is about 1000 A/cm^2 in the bundle cross-section and the resulting coil-case pressure is about 2000 psi. These conditions imply comparatively modest magnet technology, although the magnet is quite large, about 3000 tonnes for each magnet half (including the stainless steel coil case).

The injector design developed for the reference hybrid is based on the positive-ion LBL injector. The reactor requires deuterium injectors with acceleration to 125 keV and tritium to 188 keV. When account is taken of the half and third energy components in the beam, the average beam energies are 104 and 156 keV, respectively, for D^0 and T^0. Our analysis predicts an efficiency for the injectors of 60%.

Outboard of the coils, end tanks must be provided to receive the plasma leakage. In the end tanks, we perform direct conversion, converting some of the kinetic energy of the ion flow directly into electricity. The remaining kinetic energy is deposited as thermal energy in the direct converter electrodes and must be removed by active cooling. Upon strikig the direct converter electrodes, the plasma flow is neutralized and the end tank must contain vacuum pumping equipment to remove the resulting gas load.

To provide access to the blanket from outside the machine, it is a convenient design feature to have one of the end tanks as small as possible. We implement the small end tank by designing the magnet such that one of the mirror fields is 5% stronger than the other. This field perturbation causes approximately 90% of the plasma leakage to flow out through the weak mirror and the remaining 10% to exit through the strong mirror. Since the size of the end tank is proportional to the amount of plasma flow, we can use a small end tank on the strong mirror. To keep this tank as simple as possible, we do not perform any direct conversion but design for the plasma energy to be deposited as thermal energy, with provisions for active cooling and vacuum pumping with cryopanels. The large end tank, which receives the 90% end leakage flow, is designed with a simple single-stage direct converter unit, having an efficiency of about 40%. This end tank must also have provisions for active cooling and vacuum pumping.

2.3 Blanket Nuclear Design

In the past, we have examined the use of primarily three fertile fuels in the blanket: UC, U-Mo alloy and thorium [8]. In our present hybrid design we are advocating the use of U_3Si, a fuel being developed in the Canadian nuclear power program for the CANDU reactor. Our reasons for this choice are (1) high uranium density (U_3Si is a metallic alloy), (2) ease of fabricability, and (3) a comparatively high burnup capability (for a metallic fuel), on the order of 3%. Economic optimization of the fuel cycle for this reactor dictates a total fuel exposure of about 6 MW-yr/m^2 of 14 MeV neutron energy through the first wall. In Table 2, the initial (beginning of life) and final (end of life) neutronic parameters for the U_3Si blanket are listed.

TIME-DEPENDENT U_3Si BLANKET NEUTRONIC PARAMETERS

TABLE 2

Exposure ($MW-yr/m^2$)	M	Pu/n	% PU	Burnup %	T/n
0	9.1	1.85	0	0	0.99
6.6	16.2	1.65	2.4	1.0	1.35

In this design we are examining a new approach to tritium breeding, that of holding up all of the bred T_2 in the blanket and recovering it by processing the T_2 pins outside the reactor, in much the same manner as is done to recover the bred fissile material. This scheme has the disadvantages of a large blanket inventory and a large inventory to start the reactor, but the inherent simplicty (which implies good safety characteristics) makes this design option worthy of examination. We are presently considering LiH + Li as a candidate breeding material. With the He coolant temperatures being used in the hybrid (280^{o}C in, 530^{o}C out) this material will have a reasonably low T_2 vapor pressure. By encapsulating this material in pins with a cladding that is a modestly good T_2 diffusion barrier (an Al alloy) we hope to maintain the release rate of T_2 to the coolant below 10 curies/day. The tritium will then be recovered at the end of the blanket life, when the blanket segments are removed from the reactor. Recovery is accomplished by removing the pins from the disassembled blanket and heating them to a high enough temperature in an oven to drive off the T_2. This is basically the procedure that is presently used for T_2 production in fission reactors. The average of the tritium breeding ratios (T/n) quoted in Table 2 is greater than one to compensate for 14 MeV neutrons lost through holes in the blanket and decay of the tritium inventory.

3. Mechanical Design

One of our primary concerns in the mechanical design of the reactor was to provide highly reliable support and containment of the blanket and primary heat transfer loop components. The basis of our concern was the conclusion that the primary safety consideration for the reactor was a loss of flow accident, and the design therefore had to be one in which the maintenance of forced cooling to the blanket could be assured to a high level of confidence.

The design approach we have selected is to mount the magnet, blanket and primary heat transfer loop all within a prestressed concrete reactor vessel (PCRV), of the type developed for gas-cooled fission reactors. This is shown in Figure 3. In the center of the PCRV is the magnet and blanket, and the steam generators and He circulators are located around the periphery. The blanket is a spherical shell inside the magnet. In this way, the blanket and its cooling system are locked together so that no relative motion between them can occur, thus precluding the possibility of rupturing any of the coolant ducts.

The PCRV also serves a second function. It provides the main restraining forces for the magnet. Since the PCRV operates at room temperature, a way had to be found to transmit the forces from the magnet (at 4 ^{o}K) out to the concrete. Our design solution has been to use a high-compressive-strength thermal insulation (Masrock, a silicate refractory), capable of sustaining about 5,000 psi. Our calculations have shown that an insulation thickness of about 50 cm is adequate to reduce the heat leak from the concrete to the magnet to an acceptable level.

The blanket design concept is one which avoids any major disassembly of the reactor during the blanket change operation but instead relies on remote operations to assemble and disassemble the blanket inside the PCRV. The blanket is made up of small cylindrical modules, approximately 50 cm in diameter, with the blanket structure being suspended directly from the inside wall of the PCRV as shown in Figure 4. Removal and replacement of blanket modules is accomplished with refueling machine shown in Figure 5, which consists of a post which is inserted down through the center of the machine and has a pivoting arm to operate on the modules. The maintenance operation consists of a series of manipulations of each of the several hundred modules.

The module, as shown in Figure 6, consists of a cylindrical pressure vessel with a hexagonal base. One of the more challenging aspects of the module design has been to devise a fast, reliable method of making up the seal that isolates the high pressure He coolant from the vacuum region that contains the plasma. We have discarded a welded joint, since remote grinding and welding are time consuming operations and we have serious doubts about the ability to consistently generate remote vacuum-tight welds. We hve therefore adopted a bolted joint using a double knife-edge (Varian type) seal with differential pumping between the two knife-edges. The pressure vessel is bolted in place with 6 bolts, one at each corner of the hex-shaped base. The internals of the module are fabricated as a single unit, containing the U_3Si pins, the tritium breeding pins and internal flow ducting. Thus, to renew a module the pressure vessel is unbolted and removed, the pin assembly is removed, a new pin assembly is inserted and a new pressure vessel is bolted in place. The coolant flow is re-entrant, with the tritium pins being cooled by the inlet flow and the coolant then proceeding down to the first wall, turning, and cooling the uranium pins on its exit path out through the module.

4. Power Conversion Loop

The primary heat transfer loop is designed to operate with helium as the working fluid. The coolant pressure is 60 atm., with an inlet temperature to the blanket of 280°C and an outlet temperature of 530°C. The flow path is designed to maintain the relative pressure drop, $\Delta p/p$, to about 3% through the entire loop (blanket, ducting and steam generator). This combination of conditions permits the use of existing gas-cooled fission reactor technology for the design of the He circulators and steam generators.

The local blanket multiplication and therefore local blanket power density increases by about a factor of almost two over the life of the fuel (See Table 2). By devising an appropriate fuel management scheme for the blanket, we are able to limit the peak-to-average variation in the total blanket thermal power to about 7% (3350 MW average; 3600 MW peak) and the primary heat transfer and power conversion loop capacity are designed to accommodate this power variation. The blanket modules are grouped into four quadrants and at time intervals of one quarter of the blanket life, the reactor is shut down and one quadrant of the blanket is refurbished with new fuel assemblies. In this way we are able to establish an equilibrium fuel cycle where the four quadrants are each at a differenct exposure.

The thermal-hydraulic design for the fuel, on the other hand, must provide adequate cooling of the fuel pins during the lifetime power density variation of 16.2/9.1 = 1.8. Our present design specifies a peak fuel power density (i.e., at the first wall) at beginning-of-life of about 230 watts/cm^3 and an end-of-life value of 410 watts/cm^3. The fuel pins are 0.7 cm in diameter with 0.15 mm thick Inconel 718 clad on a pitch-to-diameter ratio of 1.05. The maximum mid-wall clad temperature (hot channel) is limited to 700°C.

5. Future Directions

In the future, we are going to examine hybrid reactors based on new mirror confinement concepts. As was mentioned previously one area of the mirror hybrid design that could be improved is to use a plasma confinement concept that has a higher Q than the classical mirror. Also, the classical mirror geometry is workable but a cylindrical confinement geometry would be more attractive.

One of the reactors which we're going to examine is based on the tandem mirror confinement concept, which is a modification of classical mirror confinement. There is currently a machine under construction at Livermore, called the Tandem Mirror Experiment, which will begin to investigate this type of confinement concept. Our present understanding of tandem mirror confinement is based on theoretical considerations.

There are several reasons that we find this type of reactor attractive. We have developed a low technology (8T magnet, 125 keV injectors) version of this particular confinement concept which exhibits a plasma Q of about 2 and a first wall neutron loading of 3 megawatts per square meter. The results of a preliminary analysis of the reactor are listed on Table 3. It produces a fusion power of 260 megawatts, would have a net electrical output of 500 megawatts, an acceptable recirculating power faction of about 30 percent and, with a U-233 producing blanket, would generate about one tonne of U-233 per year. A schematic of the reactor is shown in Figure 7. The fusioning plasma is located in the central cylindrical volume 27 meters in length, 7 meters in diameter. Plasma confinement is provided in the end regions by classical mirror plasmas, but no fusioning occurs in these "plugs."

PARAMETERS OF A TANDEM MIRROR HYBRID REACTOR

TABLE 3

Fissile-Fuel-Breeding Central Cell:			
Cylindrical shape	length	=	27 m
	outside Radius	=	3.5 m
Magnetic field strength		=	3 T
Neutal beam injection of D & T		=	1070 A @ 125 keV
First Wall Neutron Loading		=	2.8 MW/m^2
Fusion power		=	260 MW
Blanket thermal power		=	1700 MW
End Plugs:			
Spherical shape	outside coil radius	=	2.0 m
Magnetic field strength		=	8 T
Neutral beam injection of D into each plug		=	42 A @ 125 keV
Performance:			
Overall plasma Q		=	1.8
Recirculating power fraction		=	0.29
Net electrical output		=	500 MWe
Annual fissile production		=	1000 kg ^{233}U

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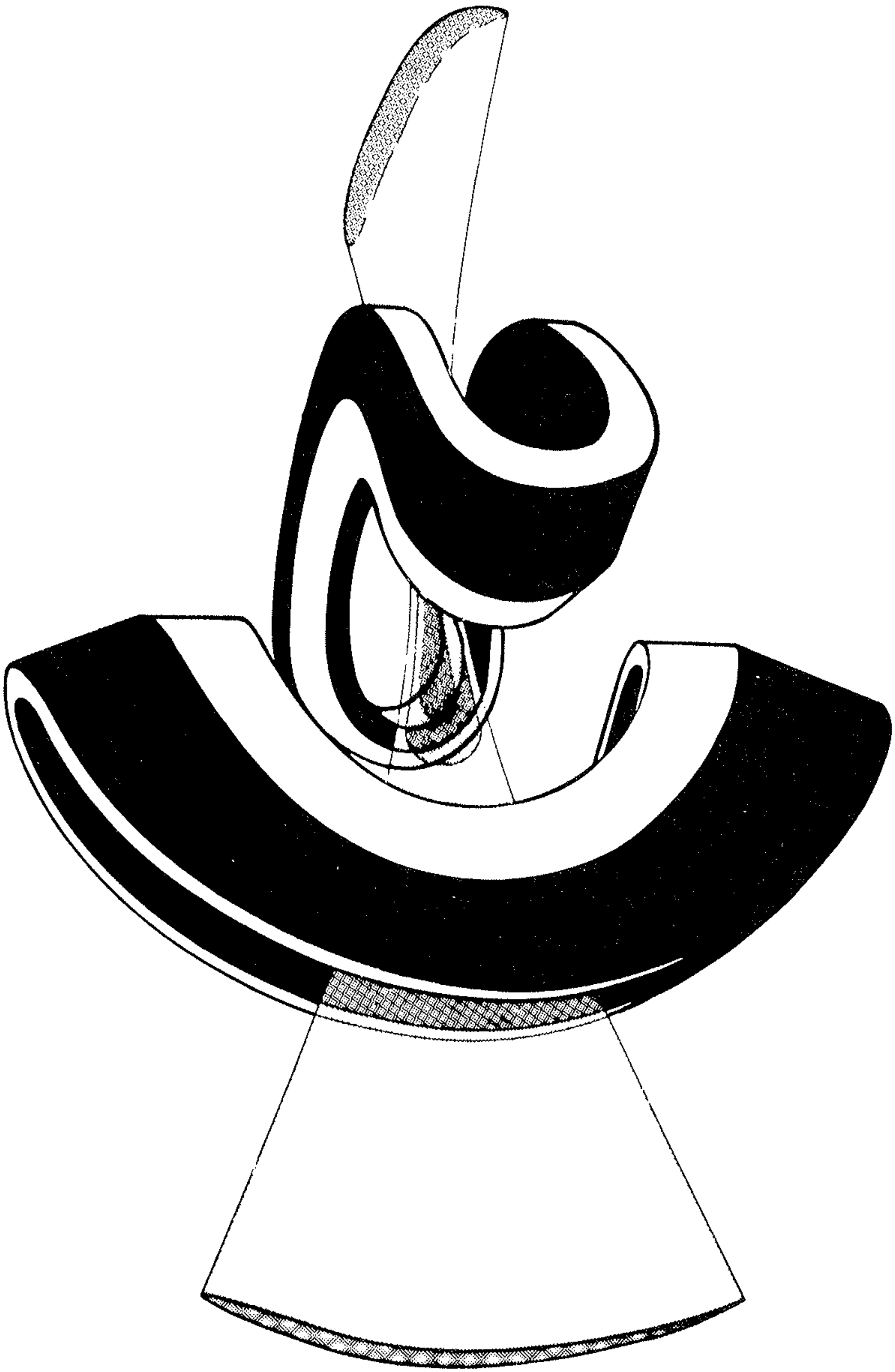


Figure 1

MAGNET CONDUCTOR CONFIGURATION

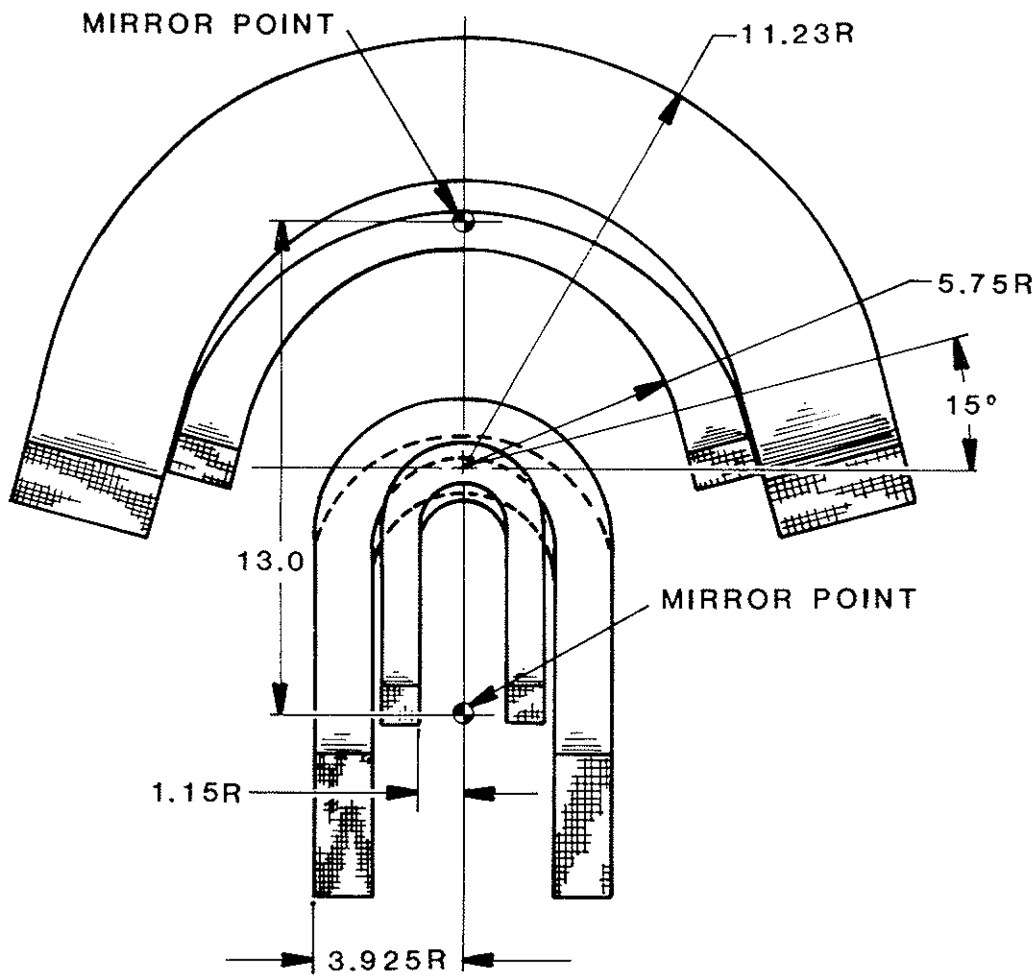


Figure 2

FUSION-FISSION MIRROR HYBRID REACTOR

LAWRENCE LIVERMORE LABORATORY

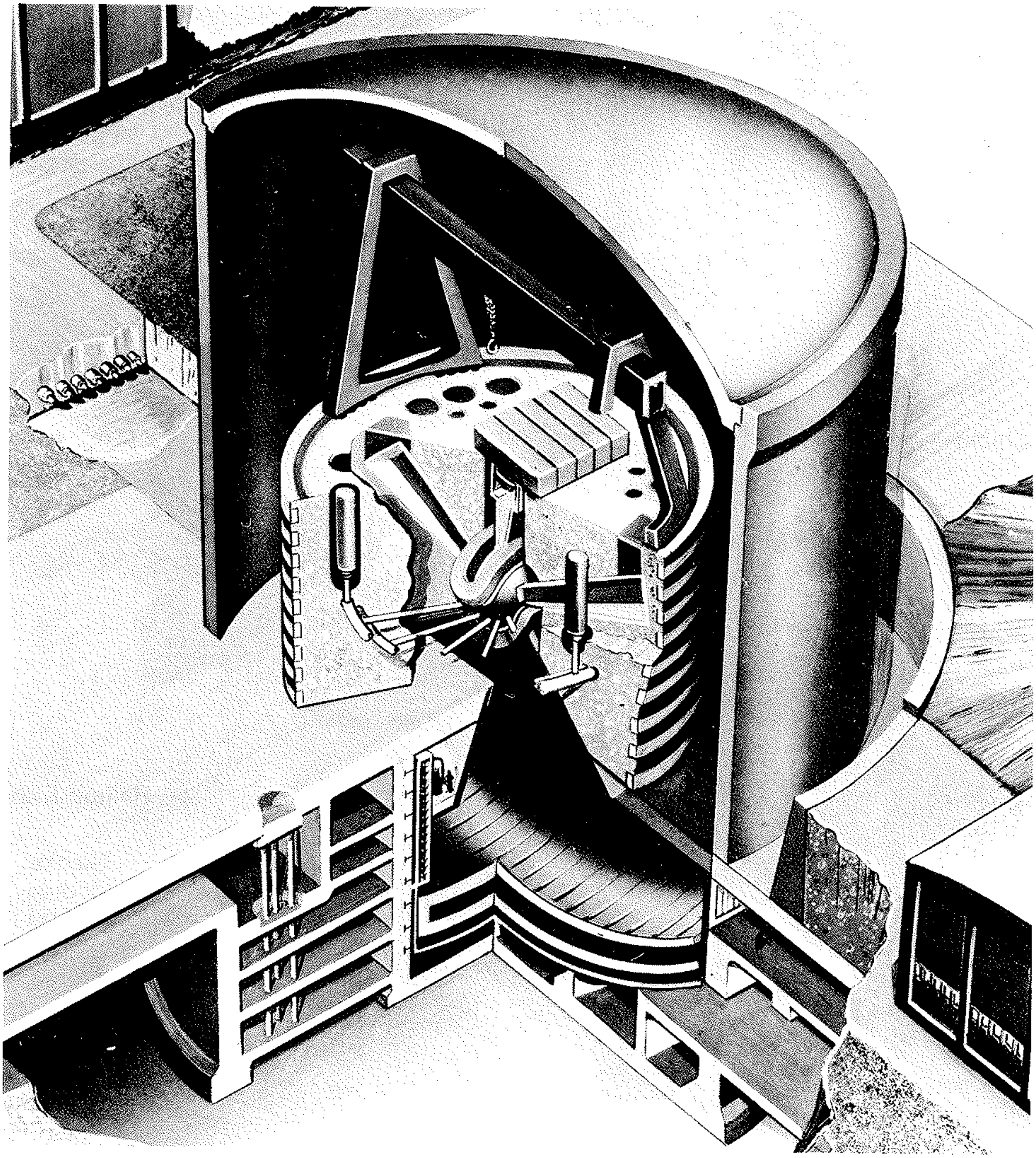


Figure 3

BLANKET/SHIELD CUTAWAY–HYBRID REACTOR

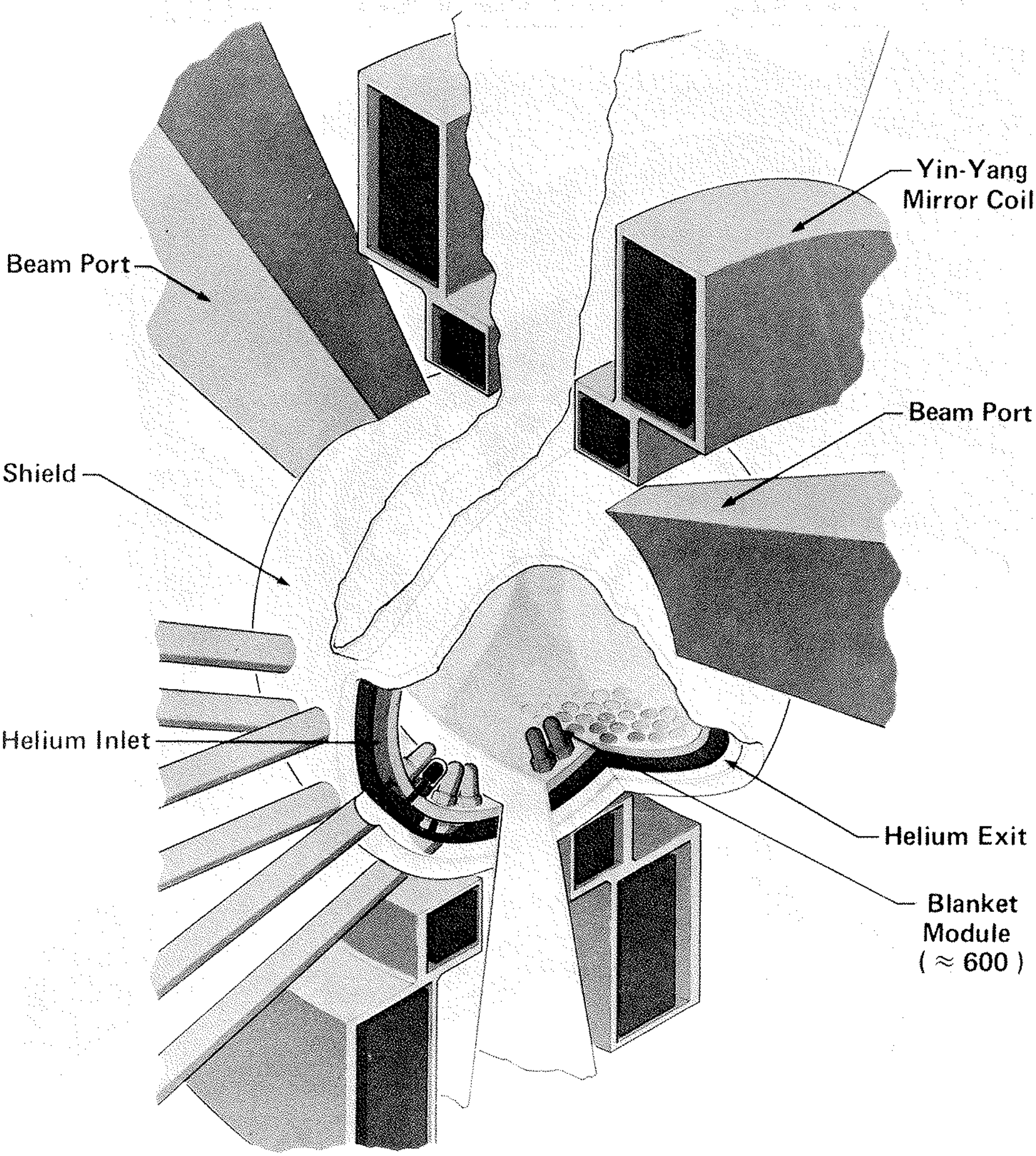
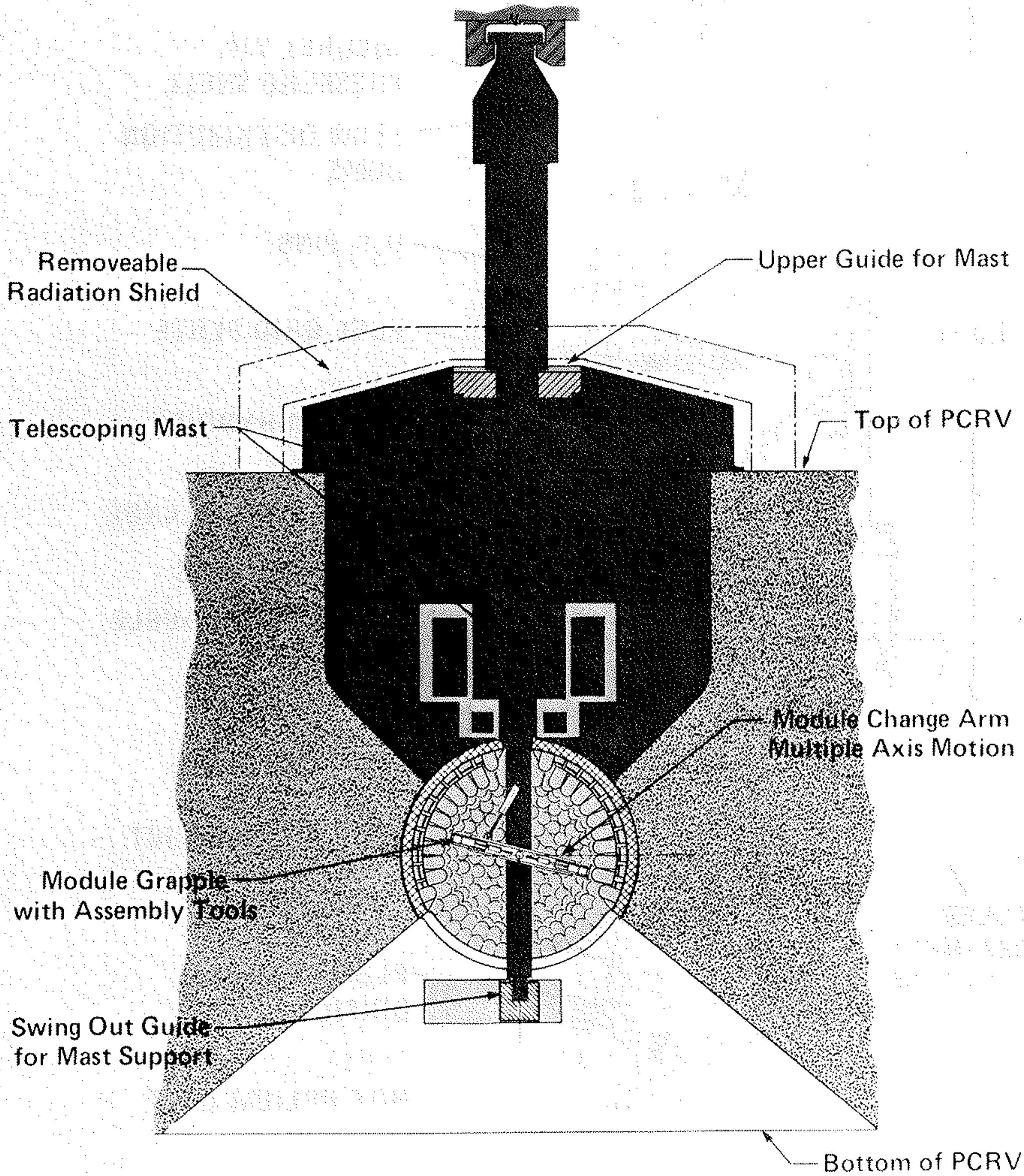


Figure 4



FUEL MODULE CHANGE TOOL

Figure 5

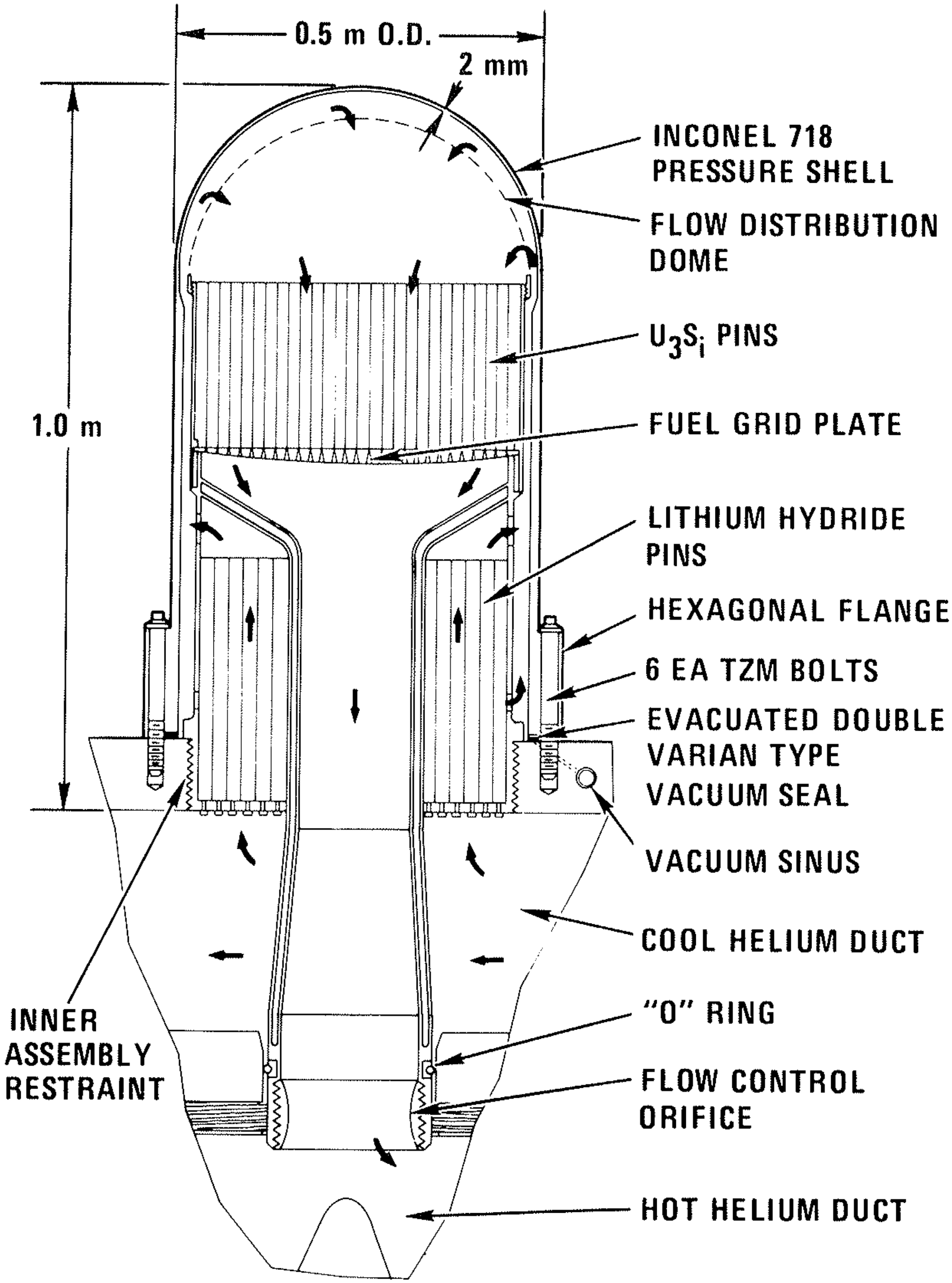

0.5 m O.D.
2 mm
1.0 m
INCONEL 718
PRESSURE SHELL
FLOW DISTRIBUTION
DOME
U_3S_i PINS
FUEL GRID PLATE
LITHIUM HYDRIDE
PINS
HEXAGONAL FLANGE
6 EA TZM BOLTS
EVACUATED DOUBLE
VARIAN TYPE
VACUUM SEAL
VACUUM SINUS
COOL HELIUM DUCT
"O" RING
FLOW CONTROL
ORIFICE
HOT HELIUM DUCT
INNER
ASSEMBLY
RESTRAINT

Figure 6

TANDEM MIRROR HYBRID REACTOR

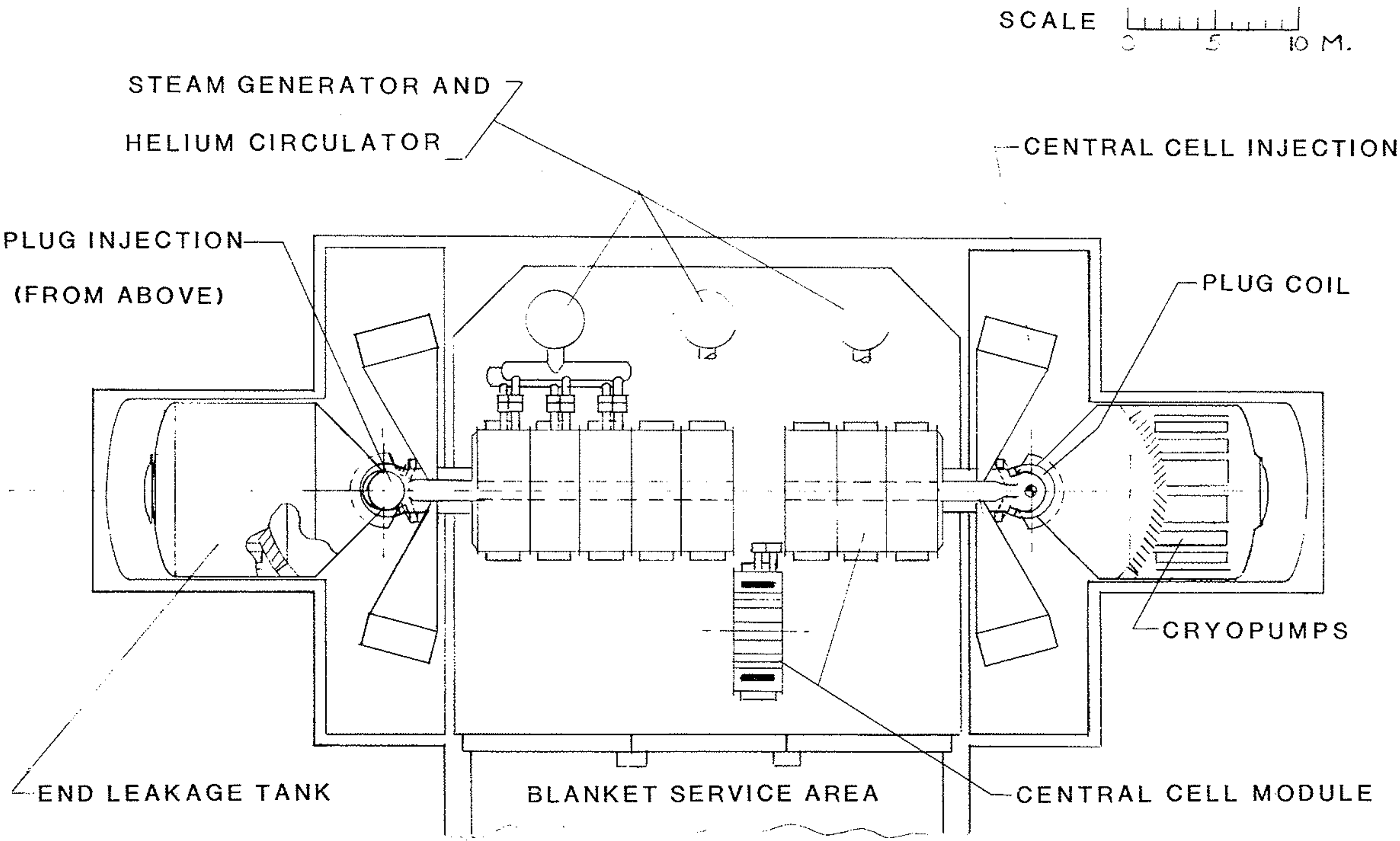


Figure 7

DR. MOSES: This paper is now open for questions.

DR. RIBE: Fred Ribe, University of Washington.

Concerning the 125 kilovolt injection energy, do you consider that to be a positive ion injector?

DR. BENDER: I think it is Fred; yes. I'm hopeful that we might be able to achieve an injector efficiency, at that energy, of somewhere between 60 and 70 percent. That will require the use of direct conversion in the beam line and, almost certainly, optimization of the species mix extracted from the source. I think with development in those areas, we can achieve something in the neighborhood of 60 to 70 percent injector efficiency. That would suffice for this type of machine.

DR. RIBE: One more question. Is this injection only into the end plugs?

DR. BENDER: This configuration, Fred, which we looked at here, is what we call a "driven tandem", and there was 125 kilovolt injection both in the end plugs and in the central cell.

DR. LEONARD: Leonard, Battelle Northwest.

Dave, about five visual aids back, you had a table which showed after 5 megawatt per square meter years, 2.3 percent plutonium and 0.75 percent burnup as I recall.

DR. BENDER: Right.

DR. LEONARD: You said the burnup was the deletion of uranium and it, obviously, isn't that. What is it? Is it the total--

DR. BENDER: That is the fraction of initial uranium atoms which have been fissioned.

DR. LEONARD: Does it include the plutonium atoms that have fissioned?

DR. BENDER: No, I don't believe so. Does it?

DR. LEE: Yes.

DR. KRAKOWSKI: Bob Krakowski, from Los Alamos. If you're going to keep the same fuel production in this machine, the power is going to double over a period of about two years. Is that true?

DR. BENDER: If one batch-loaded the blanket, Bob, that's correct. In other words, this would happen if you took all fresh fuel loaded in the blanket and let the blanket uniformly go to end of life. We don't do that. We have been able to limit the excursions, the peak to average thermal power in the blanket to about 10 to 15 percent. The way we do that is the same way one manages the fuel in a light water reactor core. Even when the reactor core has a life of three years, once a year you shut down and replace a third of the core, and we do the same thing. This blanket has a life of about four years. Once a year we shut down and replace a quarter of the blanket. And so, once we reach equilibrium, we wind up with a blanket that has four different sections at four different exposures. And if one goes through the analysis of this, you find out you can reduce the peak to average power significantly this way.

DR. KRAKOWSKI: So this average multiplication stays at about eight or nine?

DR. BENDER: Right. These are what we call local values, and it's just the value one would obtain for a particular bundle, or a particular module in the reactor.

DR. TEOFILO: Vince Teofilo, Battelle Northwest.

Dave, could you describe, on your module design, how you circumvented the hydrodynamic stagnation point at the dome of the apex--the apex of the dome?

DR. BENDER: What Vince is referring to here is the fact that, in a nice simple clean flow, as the helium comes down around this anulus, one would wind up with a stagnation point in this region here, the hydrodynamic stagnation point.

We anticipate a good deal of turbulent mixing which should give us reasonable heat transfer at this point. Also, with the mirror machine, there is a very low heat transfer rate through this first wall because there are no energetic ions impinging on it. The only heat we have to remove from the first wall is due to the 14 MeV neutrons and that is typically only 5 to 10 watts per square centimeter. We don't need a particularly high heat transfer coefficient to handle that kind of a heat flux.

DR. MOSES: This will be the last question.

DR. GRACE: Nelson Grace, DOE.

Having had to contend with numerous problems over a number of years in the design of pressurized water reactors including the development of fuel elements of high integrity and, also, the problems of LMFBR reactors that I've been associated with, I have been somewhat overawed--overwhelmed, perhaps, is a better word--by the potential problems in the development of a fusion reactor blanket, considering the many functions that have to be performd. And the nonuniform depletion (nonuniform in space and in the different reactants that are involved) all going on in a hostile cyclic environment. Now, then you add to that--

DR. BENDER: That's for Tokamak, not for mirror machine.

(LAUGHTER)

DR. GRACE: Perhaps so far as the cyclic nature is concerned but then, when we add the additional functions of breeding plutonium or U-233, we have all of the problems of breeder reactor development squared.

Now, you used the term "low technology", and I guess that's what throws me. I just wanted to ask you if you have had people with experience in long-range reactor development really look at the blanket and realistically assess the magnitude or the problems and make some estimate of how long it's going to take to develop high integrity blanket elements for a hybrid system?

DR. BENDER: Well, in the last two years we have had a fission reactor contractor, General Atomic, participating with us in our study. Ken, would you mind if I threw that hot potato in your lap?

DR. SCHULTZ: Sure. Ken Schultz, General Atomic Company.

General Atomic has been participating with Livermore in the development of the blanket design for the mirror hybrid reactor. Some of the ideas you see there came from General Atomic Company.

The fuel is uranium silicide, which has been developed in the CANDU reactor program in Canada. They have gotten reasonably extensive burnup runs with uranium silicide; not in a 14 MeV neutron source, of course, but in a thermal reactor spectrum. It's a lot more developed than a brand new concept. And we have some confidence that the data for the uranium silicide from the CANDU program will be applicable.

The cladding and construction material is Inconel 718, which is being pursued as part of the LMFBR program. Under the conditions we are designing for in this system, stainless steel 316, particularly the titanium doped 316, should also be adequate. However, we have chosen Inconel.

The helium cooling technology is one that has received extensive analysis and experimentation as part of General Atomic's gas-cooled fast reactor program.

So, I think it's not a "pie in the sky". To say that we have a working fusion reactor is false, but I think that the state of the blanket development is considerably in excess of the state of the fusion driver development here and would not limit hybrid reactor development.

WFPS-TME-073
DECEMBER, 1977

STATUS OF WESTINGHOUSE TOKAMAK HYBRID STUDIES

RONALD P. ROSE

PRESENTED AT THE
SECOND DMFE FUSION-FISSION ENERGY SYSTEMS
REVIEW MEETING
WASHINGTON, D.C.
NOVEMBER 2, 1977

fusion power
systems department

Westinghouse Electric Corporation
P.O. Box 10864, Pgh. Pa. 15236

WFPS-TME-073
DECEMBER, 1977

STATUS OF WESTINGHOUSE TOKAMAK HYBRID STUDIES

Prepared by: Ronald P. Rose
RONALD P. ROSE, MANAGER
HYBRID FUSION PROJECTS

Westinghouse Electric Corporation
Fusion Power Systems Department,
P.O. Box 10864, Pgh. Pa. 15236

STATUS OF WESTINGHOUSE TOKAMAK HYBRID STUDIES*

RONALD P. ROSE

FUSION POWER SYSTEMS DEPARTMENT
WESTINGHOUSE ELECTRIC CORPORATION
P. O. BOX 10864
PITTSBURGH, PENNSYLVANIA 15236

INTRODUCTION

The first Fusion-Fission Energy Systems Review Meeting was convened in late 1974[1]. Since then, a significant amount of work on hybrids has been done by various organizations. The intent of this recent work has been to place more emphasis on engineering realism and to produce results which represent much more than just neutronic studies. For the first time, it has been possible to characterize the fusion neutron source for a hybrid system with sufficient confidence that engineering-oriented studies can produce meaningful results.

The purpose of this paper is to present a brief summary of the tokamak hybrid design activities in progress at Westinghouse during this period of time. Our principal study programs involving hybrid fusion-fission systems are as follows:

- Fusion-Driven Actinide Burner Design Study
- Tokamak Breeder Design Study (U-Pu Fuel Cycle)
- Laser Fusion Hybrid Study
- T-20 Blanket Test Module Layout
- Tokamak Hybrid Breeder Design Study (Th-U Fuel Cycle)

*Work described in this paper was sponsored by the Electric Power Research Institute under Contract RP473-1 and by member utilities.

The first experience with the tokamak hybrid was provided in the actinide burner work[2]. In that study, an intensely beam-driven tokamak was selected as a fusion neutron source and was surrounded by a blanket containing residual actinides from fission reactor high-level wastes. Following the actinide burner study, work was directed toward breeding fissile material in a tokamak[3]. Initially, the U-Pu fuel cycle was selected for reasons that are discussed later in this paper. Growing interest in the Th-U fuel cycle prompted a re-examination of the use of Th in a hybrid breeder in an attempt to find better ways to utilize this material. Study of a tokamak hybrid utilizing the Th-U fuel cycle is currently in progresss at Westinghouse. Other hybrid activities include a laser fusion hybrid study[4] conducted in cooperation with the Lawrence Livermore Laboratory and a cursory conceptual layout of a hybrid blanket test module for the Soviet Tokamak T-20[5].

Tokamak hybrid studies at Westinghouse have been based on the TCT (or two-energy component tokamak) principle which has been incorporated in the Tokamak Fusion Test Reactor (TFTR) now under construction at the Princeton Plasma Physics Laboratory. The reasons for this choice are as follows:

- Proof of the TCT concept in TFTR affords an early application of fusion technology.
- The high fusion neutron source density in the TCT provides a prolific source of neutrons.
- These fusion neutrons can be coupled to a fission blanket for:
    - fissile fuel production
    - power production
    - radiotoxic waste disposal

The combination of fusion and fission offers the potential for unique performance characteristics, as noted in Dr. Lidsky's paper. The challenge in developing the hybrid concept lies in coupling the advantages of both the fusion and fission systems to realize the best features of each technology.

ACTINIDE BURNER

The high level component of fission reactor wastes can be divided into two categories: fission products and actinides. Accelerated depletion of these wastes by further irradiation has been a topic of interest for a number of years. The relevant characteristics of fission products and actinides are as follows:

Fission Products

- Debris created by fission of heavy nuclei
- Shorter-lived component of waste
- Relatively small neutron cross sections

Actinides

- Created by transmutation of heavy nuclei
- Long-lived component of waste
- Larger neutron cross sections than fission products
- Some actinides are fissile materials

In view of these characteristics, the outlook was one of optimism at the outset that actinide burning would be a viable mission for a near-term tokamak hybrid system. The objective of the actinide burner design was to fission the actinides and, thus, convert these nuclides to much shorter-lived fission products, at a rate sufficient to provide a two or three order of magnitude decrease in their radiotoxic hazard potential during the design life of the hybrid system. If that performance potential could be realized, the residual hazard due to the actinides would then be limited by the ability to separate the actinides from the fission products in the high level waste material.

A cutaway view of the design that evolved for the actinide burner is shown in Figure 1.[6] This view shows the principal components of the tokamak system including field coils, vacuum vessel and liner, blanket and shield, vacuum duct

FUSION DRIVEN ACTINIDE BURNER

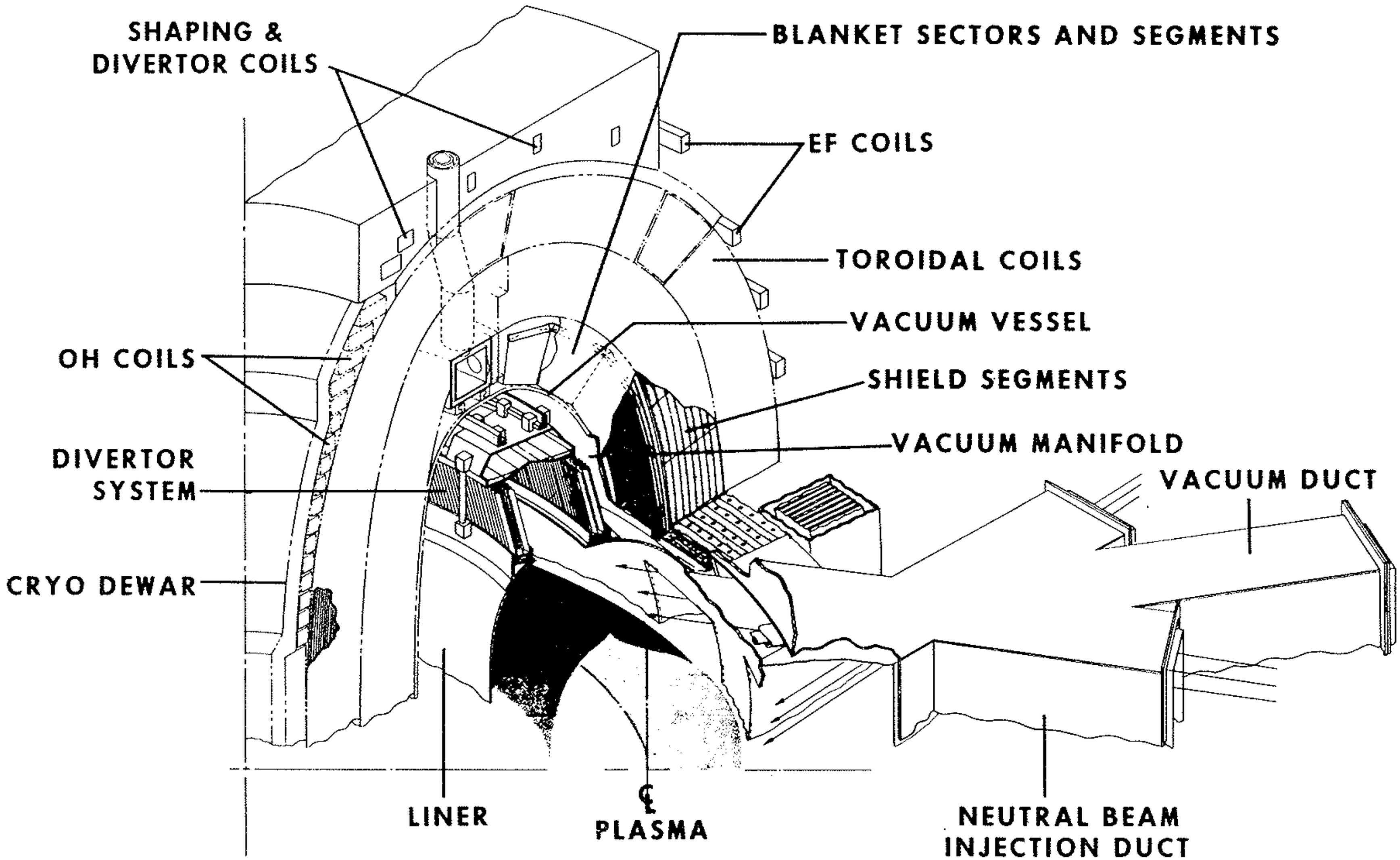
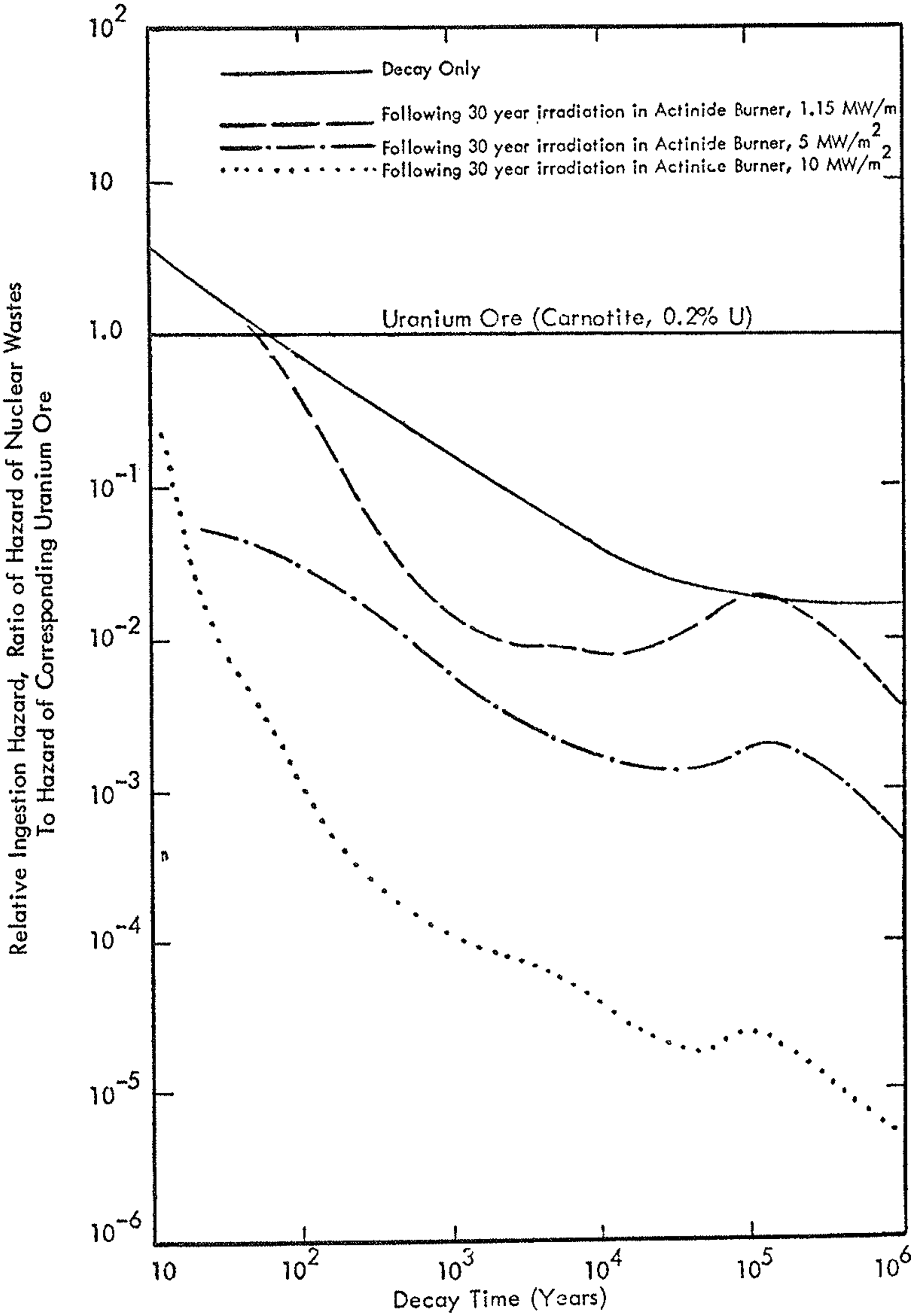


FIGURE 1

and neutral beam injection duct. A poloidal divertor was adopted for the plasma exhaust system. The plasma is driven by intense neutral beam injection and the exhaust is pumped out coaxially through the same vacuum duct penetrations. The blanket containing the actinides is restricted to the outer major radius of the tokamak to insure access for refueling operations.

Performance of the system in depleting the actinides is summarized by the curves in Figure 2. These data present the relative ingestion hazard, normalized to that of the parent ore, as a function of decay time after irradiation. With natural decay (no irradiation), the ingestion hazard drops below that of the parent one after about 100 years. Irradiation in the reference actinide burner at a first wall loading of 1.15 MW/m^2 gave less than an order of magnitude reduction in hazard after a presumed plant lifetime of 30 years. Therefore, calculations were performed to determine how much higher first wall loading would be needed for a reduction in relative ingestion hazard of several orders of magnitude. These results show that 10 MW/m^2 is needed to provide a three order of magnitude reduction in ingestion hazard after a 30 year irradiation. Wall loadings approaching 10 MW/m^2 appear to be greater than one could anticipate based on near-term application of the TCT concept. When it is noted that natural decay affords a reduction in ingestion hazard to that of the parent ore in about 10 years, however, it may be appropriate to reevaluate concerns regarding long term hazards posed by the actinides.

One of the principal reasons that the actinide hazard does not decrease more rapidly with irradiation is the production of Pu-238 by transmutation of Np-237. The peak in ingestion hazard noted after about 10^5 years decay time is caused by Ra-226 which is a daughter product created by the decay of Pu-238. Therefore, the actinide burner is essentially a very effective "breeder" of Pu-238. This characteristic suggested that a TCT hybrid could be an attractive means to produce fissile nuclides from fertile material such as uranium or thorium.



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

FIGURE 2

TOKAMAK BREEDER USING THE U-Pu FUEL CYCLE

Experience in the actinide burner study provided a number of interesting design directions and concepts that were subsequently applied to the tokamak breeder study. Two objectives of the tokamak breeder study were to assess the feasibility of producing fissile fuel with fusion neutrons from a tokamak and, in particular, to determine whether the high yields suggested by earlier neutronic studies could be realized in a viable engineered design.[7]

Candidate blanket lattices for the hybrid breeder are shown in Table 1. Consideration was restricted to present or near term future state-of-the-art technology with the goal of a demonstration of breeding by the late 1980's. These lattices are characterized by fast, intermediate or thermal spectra, and by U or Th as fertile materials. The best performance in terms of total fissile fuel and energy production was obtained using U-carbide in a fast neutron spectrum.[8] Therefore, this blanket concept was selected as the reference design.

A summary of tokamak breeder parameters for the reference design is presented in Table 2. The blanket produces almost 2300 MWt at equilibrium cycle. This energy production is sufficient to offer the potential of break-even in electrical power production.

Net fissile production of 1800 kg/yr Pu is adequate to refuel about 5 light water reactors (LWR'S) operating with Pu recycle and natural uranium makeup. This production rate is about the same whether natural or depleted uranium is employed in the hybrid blanket since the very hard neutron spectrum results in most reactions occurring in the U-238. The minor radius is about the same as TFTR while the major radius is larger than TFTR to afford the increased space needed for shielding with higher fluences. Superconducting Nb_3Sn field coils were chosen with a peak field in the TF coils of 10.5 tesla. A pulse duration of 75 sec on/15 sec off was adopted to provide a high enough duty factor (and thus integrated fluence) to give reasonable fissile production rates.

TABLE 1

CANDIDATE HYBRID BREEDER FUEL LATTICES CONSIDERED

No.	Fuel Form	Spectral Index*σ_p	Neutron Spectrum In Lattice
1	U-Carbide	20	Fast
2	U-Oxide	24	Fast
3	U-Oxide	25	Fast
4	U-Moly Alloy	14	Fast
5	Molten U-Salt	42	Fast
6	U-Carbide	346	Intermediate
7	U-Carbide	3450	Thermal
8	Th-Carbide	23	Fast
9	Th-Oxide	30	Fast
10	Th-Carbide	350	Intermediate
11	Th-Carbide	865	Thermal

*$\sigma_p = \Sigma_s/N$ of U or Th

TABLE 2

TOKAMAK BREEDER PARAMETERS

U-Pu FUEL CYCLE

Blanker Thermal Power Production*	2280 MWt
Net Fissile Production+	1800 Kg/Yr
Blanket Lattice Type	He Cooled U_{nat} C Pins
Major Radius	4.45 m
Minor Radius	0.90 m
Fusion Neutron Loading at First Wall	1.55 MW/m^2
Plasma Q = Fusion Power/Beam Power	1.25
Plasma Exhaust System	Double Null Poloidal Divertor
Field Coil Type	Superconducting Nb_3Sn
Pulse Duration	75 sec. on/15 sec. off

*Nominal Power During Pulse for Equilibrium Fuel Cycle

+Based on 75% Assumed Plant Availability

A cutaway view of the tokamak breeder is shown in Figure 3. A single region blanket restricted to the outer major radius of the tokamak has been incorporated as in the actinide burner. The legend on the figure shows the principal system components. One of four pairs of neutral beam injectors is shown.

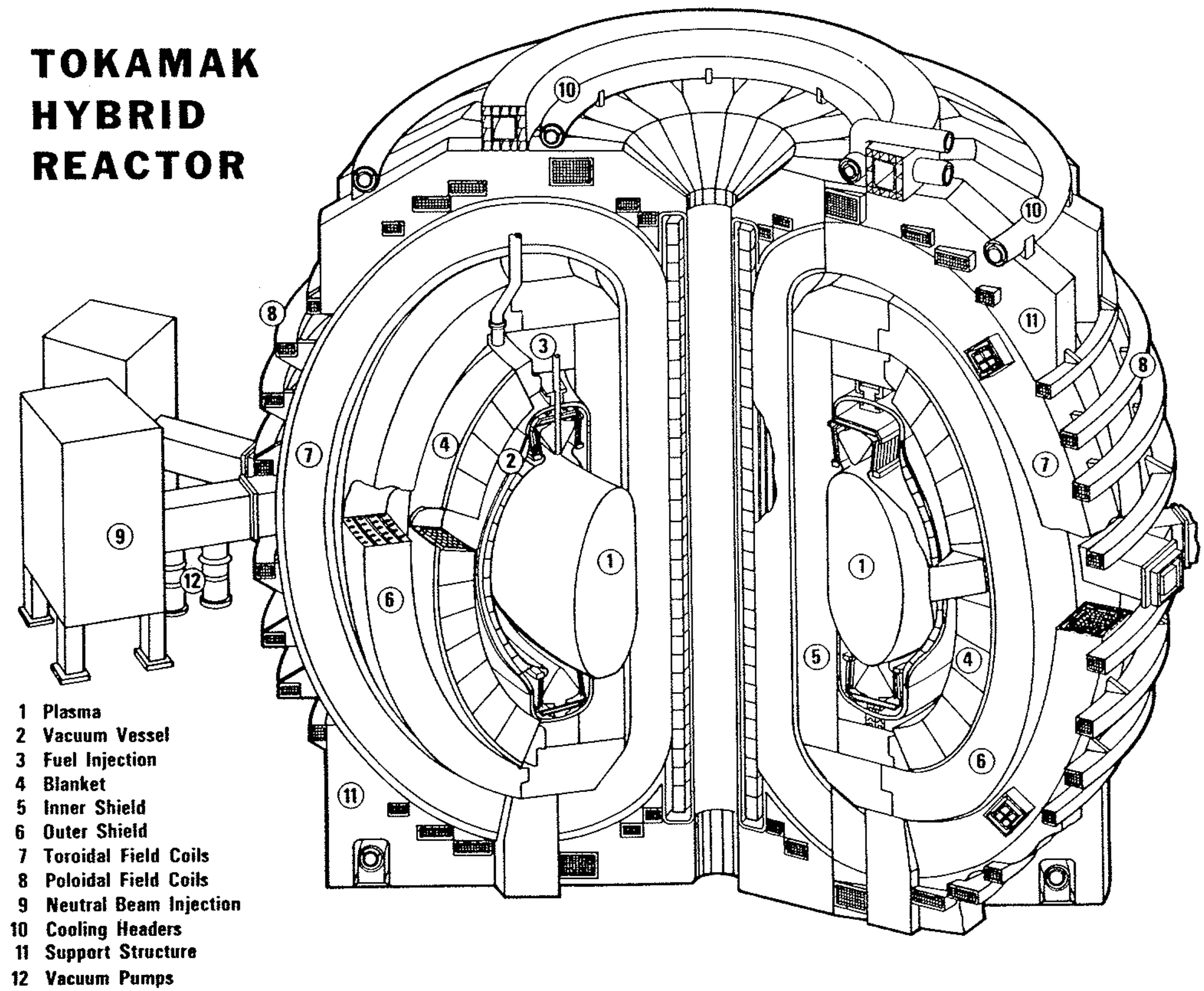
Conclusions from the tokamak breeder study are as follows:

- Fast lattices give the best fissile breeding and power performance
- A tokamak breeder based on TFTR plasma parameters can refuel about 5 LWR's (Pu recycle) with a hybrid blanket of natural or depleted U.
- The tokamak breeder has the potential to at least break even on electrical power generation.
- Uranium gives greater fissile and power production than thorium in a single region blanket.

TOKAMAK BREEDER USING THE Th-U FUEL CYCLE

In view of the growing interest in the Th-U fuel cycle and its perceived advantages with regard to nuclear proliferation, further attention was devoted to identifying approaches by which to achieve better blanket performance from thorium. The neutronic characteristic that affords better performance from U-238 than from Th-232 is the difference in fast fission cross-section as shown in Figure 4. The larger fast fission cross-section of U-238 produces more fission energy and better neutron multiplication characteristics than attainable with Th-232 alone. The concept presently being evaluated to compensate for this effect is a multiple region blanket as shown in Figure 5. The inner region is a fast fission neutron multiplier which also produces a significant amount of thermal power. The U-233 breeding region is located in the outer portion of the blanket.

One possible scenario that might be of interest in the symbiotic relationship of such a hybrid with fission reactors is shown in Figure 6. U-233 produced in the hybrid blanket is fabricated into new fuel assemblies for fission



TOKAMAK HYBRID REACTOR MAJOR COMPONENTS

FIGURE 3

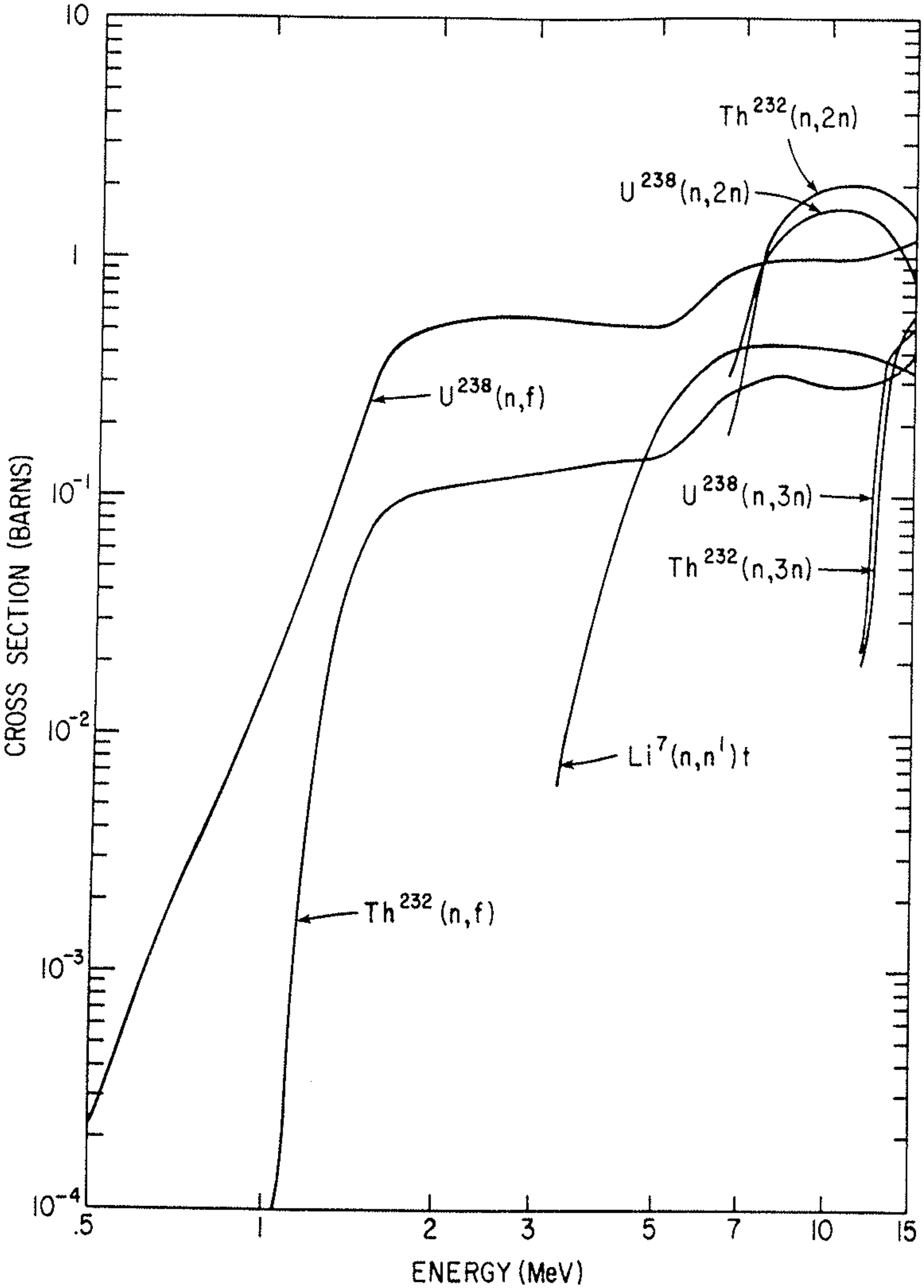

10
1
10^{-1}
10^{-2}
10^{-3}
10^{-4}
CROSS SECTION (BARNS)
.5
1
2
3
5
7
10
15
ENERGY (MeV)
$Th^{232}(n,2n)$
$U^{238}(n,2n)$
$U^{238}(n,f)$
$U^{238}(n,3n)$
$Th^{232}(n,3n)$
$Li^{7}(n,n')t$
$Th^{232}(n,f)$

FIGURE 4

HYBRID BREEDER CONCEPT FOR Th CYCLE

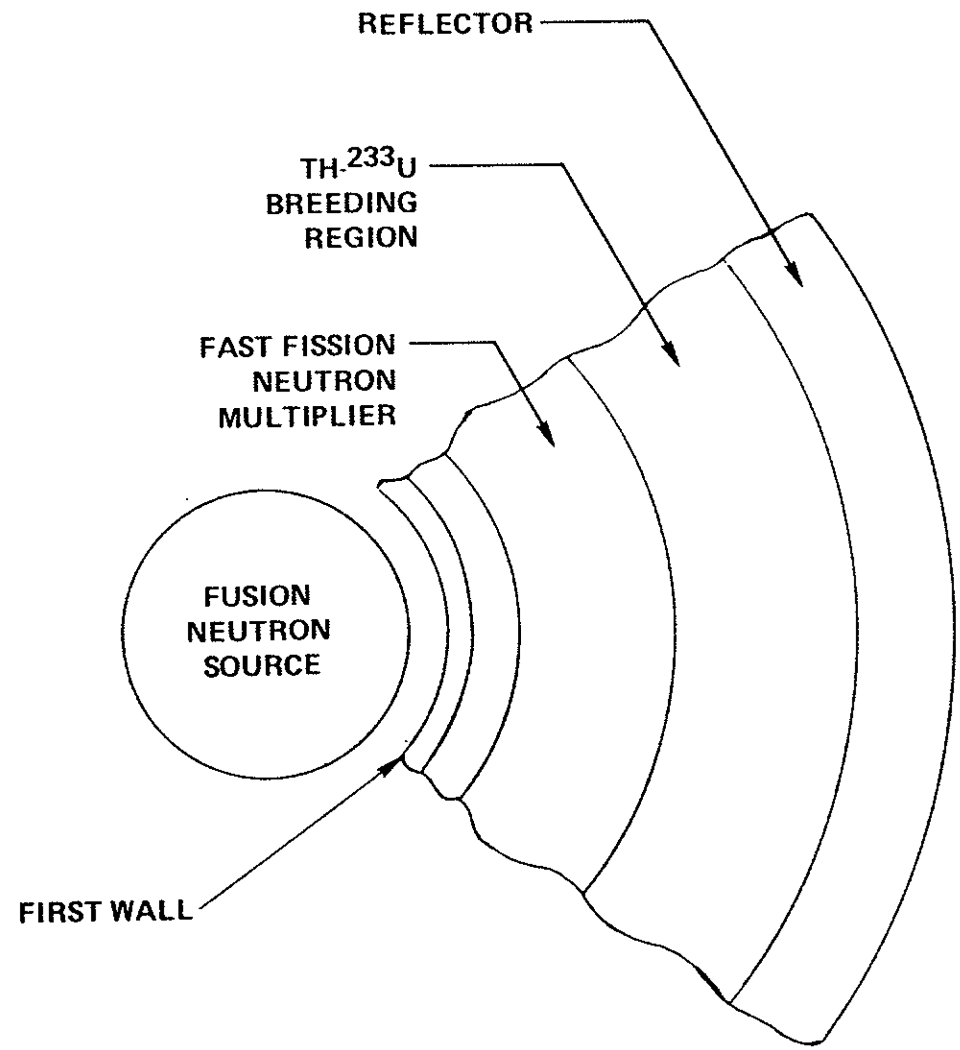


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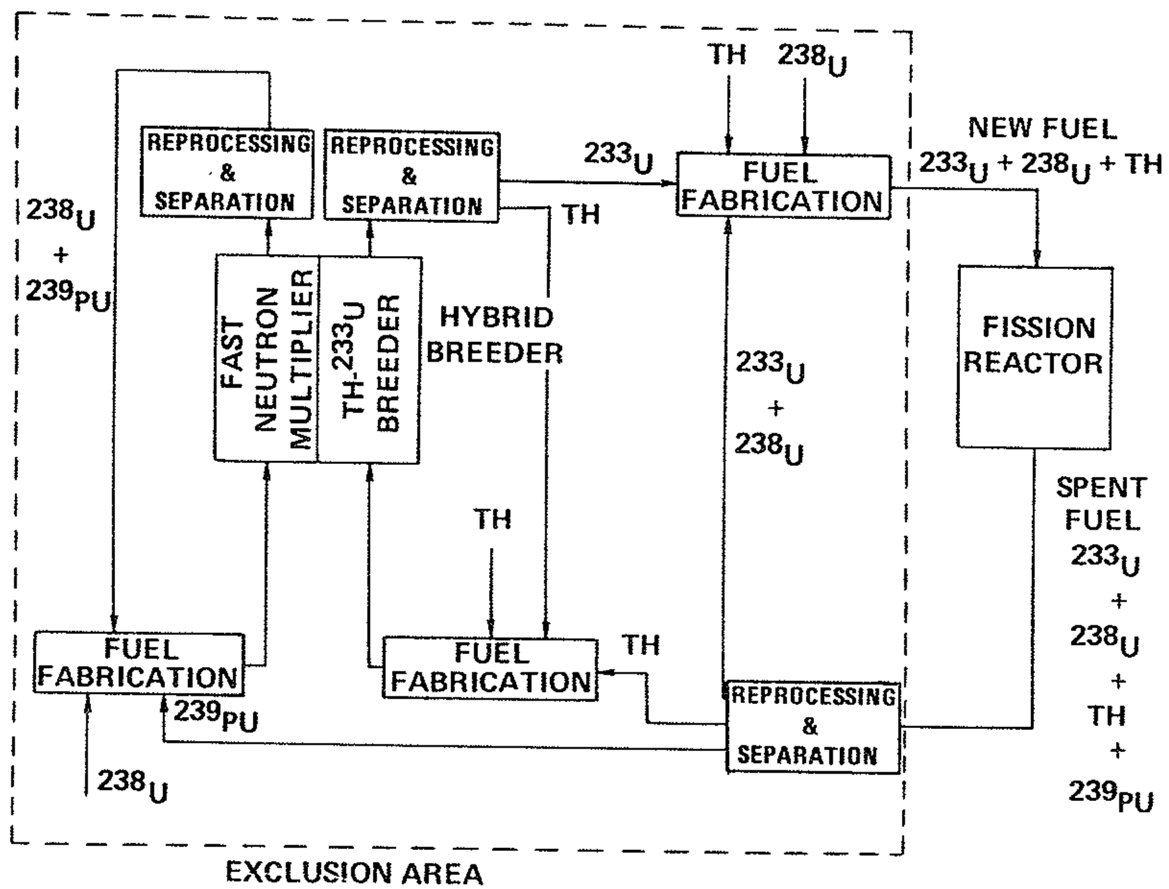
FIGURE 5

HYBRID BREEDER-FISSION REACTOR SCENARIO



S533-91

FIGURE 6

reactors. Addition of sufficient U-238 would permit operation with a "denatured" fuel cycle in which the fissile content of the uranium is insufficient for weapons use without capability for enrichment by isotope separation. The conversion ratio of the fission reactor could be altered by admixture of thorium in the fuel cycle. The spent fuel from the fission reactor would contain Pu-239, of course, in addition to the constituents of the new fuel since U-238 was added in fabrication of the fuel assemblies. One option would be to profitably consume the Pu-239 in the neutron multiplier region of the hybrid blanket, thus enhancing the neutron economy of the symbiotic hybrid/fission reactor system. Other options are possible, of course, ranging from a fission reactor "burner" system in which the 3-3 1/2% fissile content of LWR fuel currently provided by U-235 could be replaced by U-233 to a light water breeder system in which the initial U-233 startup inventory would be supplied by the hybrid.

One of the key aspects of the scenario in Figure 6 is the provision to restrict most of the operations to an exclusion area denoted by the dotted lines. Only natural fertile materials (thorium and natural or depleted uranium) need enter this exclusion area. New fuel destined for fission reactors can be "denatured" as previously noted. Spent fuel assemblies from the fission reactor will be highly radioactive and require appropriate remote handling techniques. Such an approach may be particularly attractive if the hybrid can provide fuel for quite a few fission reactors as suggested by recent studies. In this case, a relatively few fuel production installations such as that shown in the exclusion area of Figure 6 could supply the needs of an extensive array of fission reactors located to serve their respective electrical load centers.

CONCLUSIONS TO DATE REGARDING HYBRID FUEL PRODUCTION

A comparison of fissile fuel production in hybrids with that in fission breeders can be quite instructive in identifying generic differences in the two concepts. The following breeding characteristics appear to be distinctive differences based on assessments performed to date:

- Fission breeders must use one neutron from each fission to sustain the chain reaction (i.e., remain critical)

  - There are 2.5-3 neutrons per fission and > 1 neutron per fission must be captured to breed and overcome parasitic losses.

  - Therefore, breeding potential is limited.

- Hybrid breeders have a subcritical fission blanket which is driven by fusion neutrons

  - Very high energy fusion neutrons can yield 4-5 neutrons per fission, giving a cascading effect of neutrons available for breeding.

  - Breeding potential for a fusion-fission system is significantly greater than that of a fission breeder for this reason.

To capitalize on the potential of the hybrid, the properties of high energy neutrons should be employed with minimum degradation of the neutron spectrum. This requires a structure between the plasma and the blanket which is as transparent as possible with respect to the 14 MeV fusion neutron current. To provide such a structure is one of the challenging engineering problems which must be addressed to realize the full potential of the hybrid concept.

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Northeast Utilities
Pennsylvania Power and Light Company
Public Service Electric and Gas Company

DR. MOSES: The floor is now open for questions.

DR. HURWITZ: Hurwitz, GE. Referring to the cycle with the U-233 and the U-238: you show on the right, going out of the exclusion area, a U-233/U-238 composition. What is the ratio of U-233 to U-238, and what are your grounds for believing that this is not an "overnight" weapons material with suitable weapons design?

DR. ROSE: We haven't really quantified the flow elements on that diagram, as yet. We're in the process of looking at that now. But we would want to stay in some range, like below 10 percent fissile, if we're following the denatured scenario.

DR. HURWITZ: Would not plutonium be produced in the external cycle?

DR. ROSE: In the fission reactor cycle, yes. And of course, one has the choice as to what to do with it. You could run the fission reactor as a once-through fuel cycle, or you could reprocess and burn up the plutonium, conceivably, inside the exclusion area if you could convince yourself that this was a suitable thing to do.

DR. HURWITZ: Would the plutonium concentration in the external reactors be an acceptable criterion in the present cycle if you stay below 10 percent?

DR. ROSE: Yes, that's probably about right.

DR. MOSES: Any other questions?

DR. TEOFILO: Vince Teofilo, Battelle Northwest.

Ron, in your hybrid design work this past year, have you done any computations and design implementation for tritium breeding, and if you have, did you achieve a breeding ratio greater than 1.0?

DR. ROSE: The first time around in looking at actinide disposal, we had a blanket that was much higher in power production and had a K_{eff} of somewhere around .9, or the low .9s. It turns out, in a Tokamak, that if you have high multiplication, you can use the inner magnet shield to breed tritium from the neutrons multiplied in the outer blanket that are transported to the inner magnet shield.

Now, in the breeder scenarios, where we're trying to work with natural fertile material, we don't get that kind of multiplication; so we convinced ourselves, late in the plutonium study, that we are going to have to modify that blanket to breed some additional tritium in the outer blanket. And in the work that we're doing now in the multiple zone blanket, we are including a requirement to do some tritium breeding there, as well.

DR. MOSES: Mike?

MR. LOTKER: Mike Lotker, Booz, Allen & Hamilton.

Ron, there's a point in connection with the denatured thorium cycle that you have discussed in the past but didn't mention today. It relates to the particular advantages of fusion-fission in the context of this cycle.

The key advantage that fusion-fission holds here is the ratio of the number of reactors outside the fence to inside the fence. In the kind of denatured thorium cycle that most people contemplate, you need as many breeders inside the fence, as you have light water reactors outside.

Here one takes advantage of the six to one ratio. In the former case, one requires enormous power parks which could really change the structure of the energy industries. This case, however, like so many aspects of fusion-fission allow the energy industries to continue going on a path that they're already on. And I think it's something that we shouldn't lose sight of.

DR. ROSE: Yes, that's a good point.

PRESENT STATUS OF LASER DRIVEN FUSION-FISSION ENERGY SYSTEMS*

James A. Maniscalco
and
Luisa F. Hansen

Lawrence Livermore Laboratory
University of California
Livermore, California 94550

ABSTRACT

The potential of laser fusion driven hybrids to produce fissile fuel and/or electricity has been investigated in the laser program at the Lawrence Livermore Laboratory (LLL) for several years. Our earlier studies used neutronic methods of analysis to estimate hybrid performance. The results were encouraging, but it was apparent that a more accurate assessment of the hybrid's potential would require studies which treat the engineering, environmental, and economic issues as well as the neutronic aspects. More recently, we have collaborated with Bechtel and Westinghouse Corporations in two engineering design studies of laser fusion driven hybrid power plants. With Bechtel, we have been engaged in a joint effort to design a laser fusion driven hybrid which emphasizes fissile fuel production while the primary objective of our joint effort with Westinghouse has been to design a hybrid which emphasizes power production. The hybrid designs which have resulted from these two studies are briefly described and analyzed by considering their most important operational parameters.

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INTRODUCTION

Interest in fusion-fission hybrid systems, dating back[1,2] to the early 1950's has been motivated by their ability to combine the attractive features of two technologies: the fission system with its power-richness (200 MeV per fission reaction), and the fusion system with its fast-neutron richness. The 14.1 MeV fusion neutrons fission fertile materials such as ^{232}Th or ^{238}U generating large amounts of energy and fissile materials.

The potential for producing fissile fuel and electricity with the laser fusion-fission hybrid reactor has been investigated at Livermore[3,4,5] for the last few years. Our earlier studies primarily used neutronic methods of analysis to identify attractive hybrid concepts and to provide an upper bound estimate on performance. These neutronic studies demonstrated that laser fusion hybrids could be designed to meet a broad spectrum of fissile fuel producing and energy multiplying requirements. The studies also demonstrated that hybrids produce 10 times more fissile fuel (per unit of thermal energy generated) than fission breeder reactors and that laser fusion hybrids produce electricity with laser and target performance requirements that are much lower than for pure 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 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, we engaged Bechtel and Westinghouse to assist in a more realistic assessment of the laser fusion hybrid's potential in a fission power generation economy. With Bechtel,[6,7] we have been engaged in a joint effort to conceptually design a laser fusion hybrid which emphasizes fissile fuel production. The primary objective of our joint effort with Westinghouse[8] has been to conceptually design a laser fusion hybrid which emphasizes power production, but still produces fissile fuel. The hybrid designs which resulted from these two engineering studies are briefly described and analyzed in this paper. The performance of both hybrid designs is evaluated on the basis of operational parameters such as system efficiency, recirculating power fraction, blanket energy multiplication, fissile fuel production, power density, and fuel burnup. Moreover, a detailed cost analysis of the LLL/Bechtel Design has been performed by Bechtel, and the results are presented.

FUEL CYCLES AND BLANKET SELECTION

The energy multiplication and the amount of fissile fuel bred by the hybrid reactor is determined by the fissionable material selected for the blanket and the type of blanket configuration. For example, natural uranium gives an energy multiplication of about 10 in a fast fission blanket design while, with a thermal fission blanket configuration, the multiplication is two to three times larger. However, the fissile fuel bred (per unit of thermal energy) with the fast blanket would be a factor of three to four larger than the quantity obtained from the thermal blanket. For a given recirculating power fraction, the blanket energy multiplication determines the fusion energy gain requirements of the fusion power plant. Fusion energy gain is defined as the product of laser efficiency and pellet gain. As such, it represents the ratio of thermonuclear energy produced to electrical energy input to the laser. The results of our earlier neutronic studies[4,5,6,8] are summarized in Fig. 1 where we have plotted fusion energy gain as a function of blanket energy multiplication and recirculating power fraction for several blanket types. (A thermal efficiency of 35% was assumed in the calculation of the curves.) It can be seen that for given recirculating power fractions, the fusion energy gain requirements decrease significantly as the blanket energy multiplication increases. A breakeven facility could run with fusion gain = 3. The shaded areas depict the regimes (energy multiplication, blanket type, fusion energy gain, and recirculating power fraction) which we have emphasized in our design studies with Bechtel[6,7] and Westinghouse[8].

The salient features and main differences between the Bechtel and Westinghouse designs are listed in Table I. As pointed out earlier, the main product of the LLL/Bechtel design is fissile fuel, while in

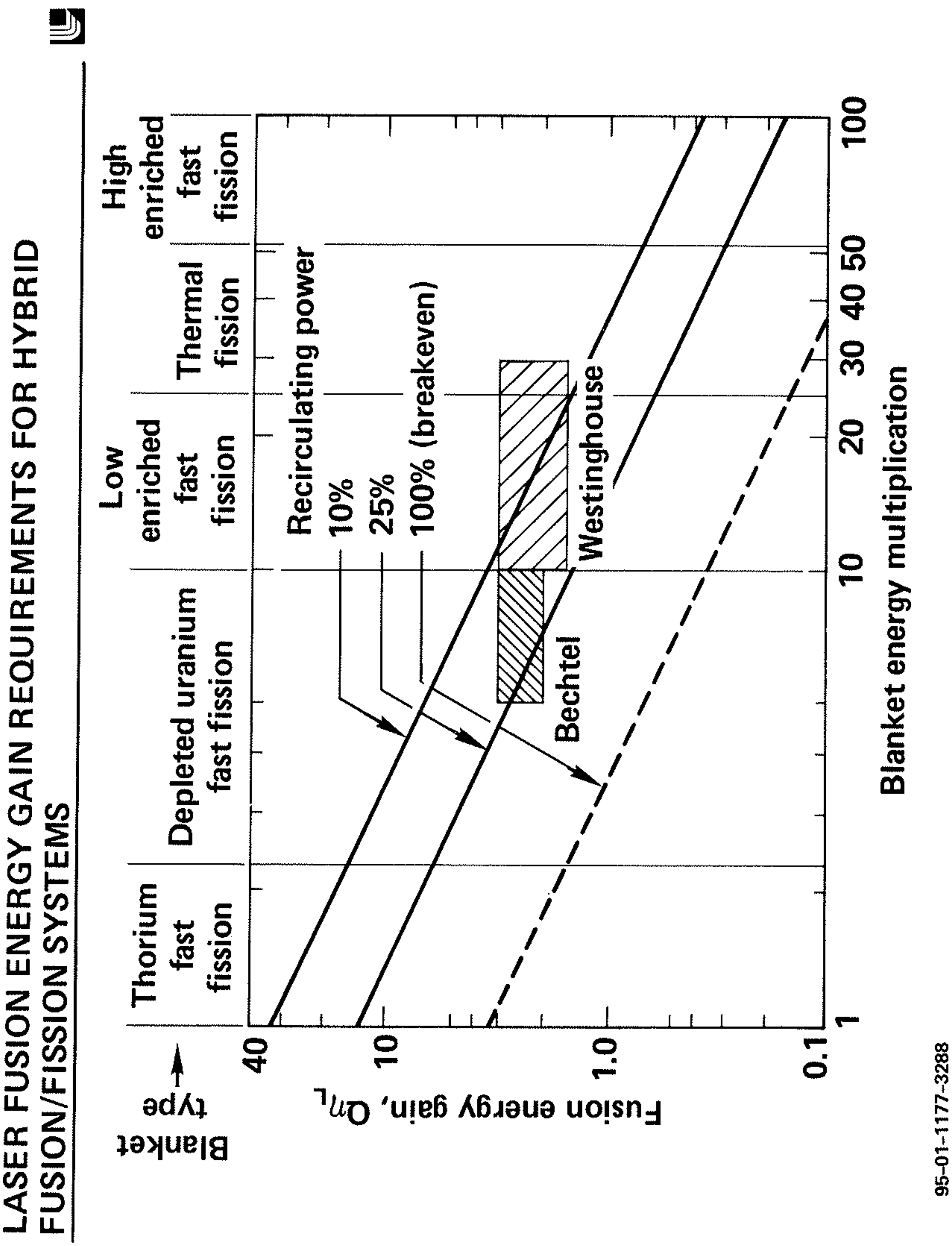
LASER FUSION ENERGY GAIN REQUIREMENTS FOR HYBRID FUSION/FISSION SYSTEMS
Blanket type
Thorium fast fission
Depleted uranium fast fission
Low enriched fast fission
Thermal fission
High enriched fast fission
Recirculating power
10%
25%
100% (breakeven)
Westinghouse
Bechtel
Fusion energy gain, $Q\eta_L$
40
10
1.0
0.1
1
10
20
30
40
50
100
Blanket energy multiplication
95-01-1177-3288

Fig. 1

the LLL/Westinghouse design the major product is electricity. Electricity production in the Westinghouse design was emphasized by designing a fission blanket with a higher first wall flux, energy multiplication, and power density. The resulting smaller power plant has less recirculating power and a higher system efficiency for the same fusion energy gain.

The capital cost of the LLL/Bechtel hybrid has been estimated by Bechtel to be 3 times more expensive than a light water reactor (LWR) of equivalent power. A detailed cost estimate of the LLL/Westinghouse design has not been made; however, because of its reduced fissile fuel performance, it would have to cost somewhat less than the LLL/Bechtel design (approximately 2 times more than an LWR) to be as attractive.

LLL/BECHTEL HYBRID DESIGN STUDY

For the last two years, we have been engaged in a joint effort with Bechtel Corporation to conceptually design[6,7] a laser fusion hybrid reactor. The Lawrence Livermore Laboratory has provided the overall direction, the neutronic data, and the fusion portions of the design while Bechtel has provided the fission section 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. Bechtel has also estimated operating and capital costs. The hybrid concept chosen for this design study is a depleted uranium fueled, fast-fission blanket which produces fissile fuel and electricity. The design maximizes fuel production at the expense of energy multiplication. The selection of depleted uranium limits blanket energy multiplication to less than 10; therefore, fusion energy gains greater than 1 are required to produce electricity with recirculating power fractions less than 0.25. This fusion energy gain requirement is an order of magnitude lower than the requirement for pure laser fusion. Another objective of our design study was to use state-of-the-art fission technology in the design of the hybrid blanket. Accordingly, stainless steel was chosen for the structure and cladding material, and sodium was chosen for the fission blanket coolant.

LASER FUSION HYBRID DESIGN PARAMETER COMPARISON

	LLL/Bechtel	LLL/Westinghouse
Fuel	Depleted uranium	Enriched fuel
Blanket energy multiplication	$M \leqslant 10$	$M \geqslant 10$
Required fusion energy gain	$\geqslant 2$	$\geqslant 1$
Principal product	Fissile fuel	Electricity
Fuel production	6–7 LWRs	3–4 LWRs
First wall	Graphite cylindrical liner	Shperical wet wall
Wall loading	~ 1 MW/m^2	~ 10 MW/m^2
Power density	~ 40 W/cm^3	~ 250 W/cm^3
Reactor configuration	Single cylindrical cavity	Multiple spherical cavities
Capital cost	$3 \times$ LWR	$\sim 2 \times$ LWR

95-01-1177-3289

Table I

The first iteration of the conceptual design was completed last year and reported in the literature.[6,7] A second iteration of this design has been carried out[9] in order to improve the performance and reduce the cost of the reactor. This most recent design is summarized in this paper. The functional shape of the reactor is shown in Fig. 2; it is basically a 10 m diameter cylindrical structure with a height-to-diameter ratio of 1.0. The reactor consists of a cylindrical shell with a removable top cover. The fusion targets are injected from the top, the laser beams enter from the side, and all coolant piping enters and exits at the top. A depleted uranium fueled fission blanket is positioned radially around the fusion chamber. The energy in the fission blanket (amounting to 80% of the total energy) is removed with a sodium coolant system which enters and exits from the upper plenum. This radial blanket is divided into 8 segments which can be individually removed. Liquid 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.

Fission Blanket

In selecting depleted uranium over natural uranium, it was noted that the energy multiplication and net Pu production of natural uranium were 14% and 3% higher respectively. However, this increased performance was not large enough to outweigh the cost and availability advantages of utilizing the billion pound U.S. stockpile of depleted uranium. Fission blankets were not positioned in the top and bottom of the reactor in order to avoid the difficulties of maintaining coolant flow when the top is removed to gain access into the fusion chamber. This decision resulted in a 30% decrease in both fissile fuel production and energy multiplication for the design. However, it was consistent with our desire to use state-of-the-art-fission technology. In the initial design, the fission blanket was 25 cm thick with two sets of fuel elements, front and back. The revised design for this reactor has a thicker fission blanket, 41 cm, with three rows of fuel elements and is divided into eight segments. Figure 3 shows a detailed view of

4000 MWt LASER FUSION HYBRID REACTOR

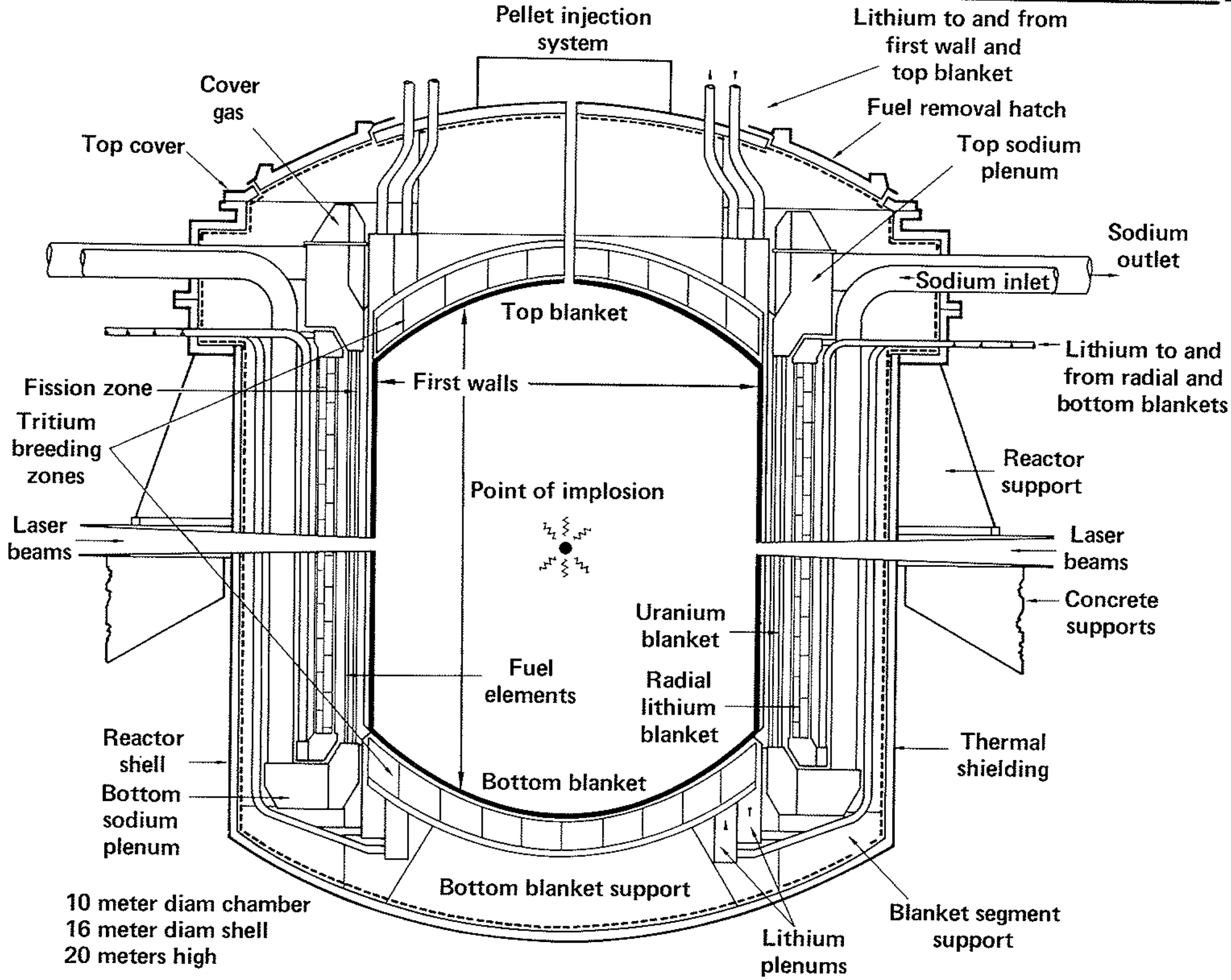

Fig. 2

REACTOR RADIAL BLANKET SEGMENT

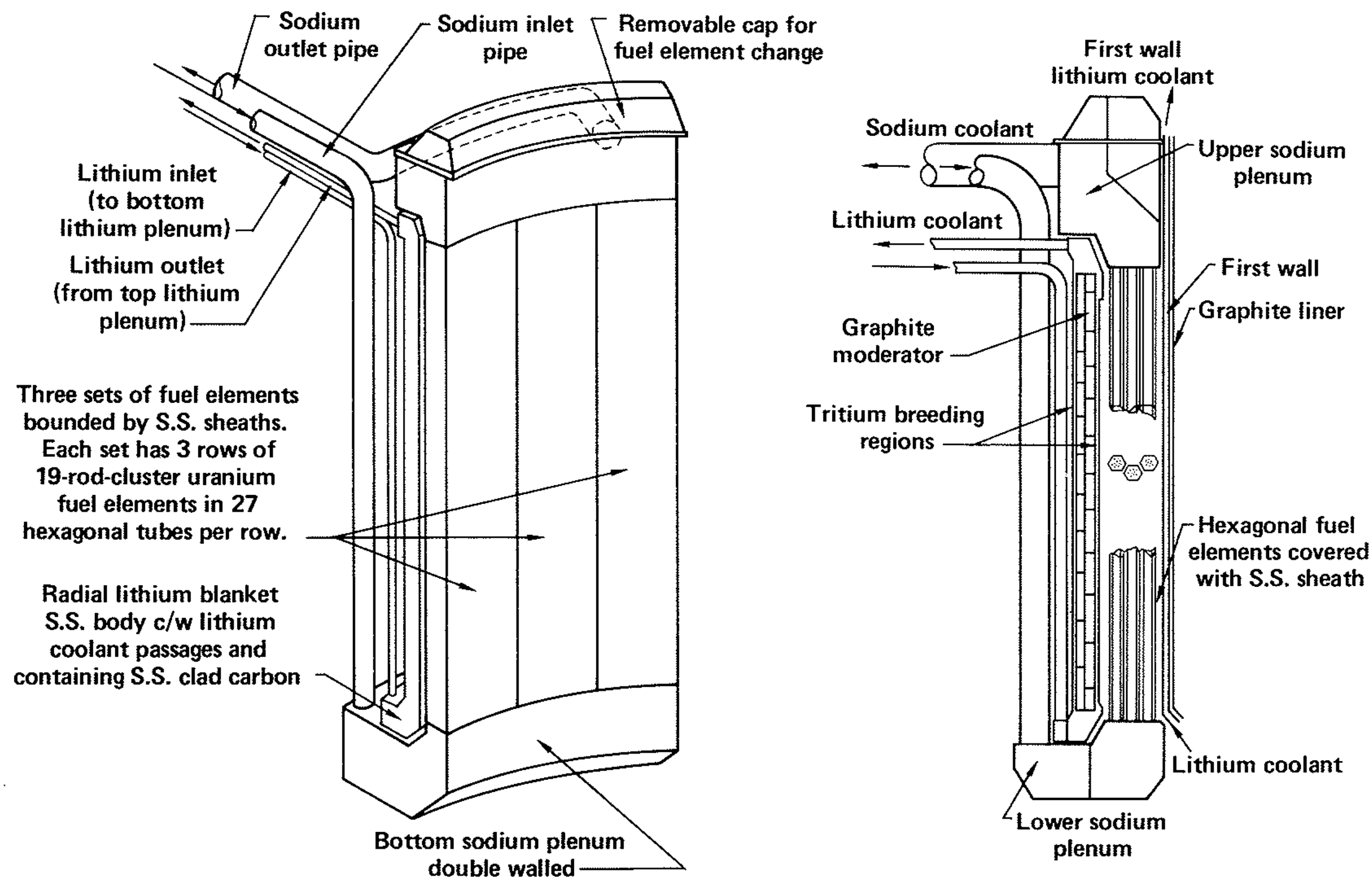

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Fig. 3

one of the eight radial blanket segments, and the fission blanket data is presented in Table II. Each segment has 81 hexagonal process tubes that contain one depleted uranium fuel element. (Half of the outer row process tubes are initially empty to satisfy the tritium breeding requirements.) Each fuel element is a 19-rod cluster of wire wrapped stainless steel clad fuel pins similar to those used in the early sodium-graphite reactors (SGR). The configuration of the fuel pins in the hexagonal process tube and the details of the cladding are shown in Fig. 4 The uranium in these fuel pins could be used interchangeably in low alloy metal form, metal alloy (7% Mo), or as a compound such as UC or UO_2. The low multiplication (approximately 6) of the uranium compounds prohibited their use in spite of their attractive burn-up and high temperature properties (approximately 100,000 MWD/MTU at maximum temperature of 1000°C). The U-7% Mo alloy had a multiplication[6] of approximately 8 and its ^{239}Pu production was about 25% lower than the one obtained from low alloy metal. We selected low alloy metal for several reasons: First, it provided the highest energy multiplication (approximately 10) and fissile fuel production performance. Second, its maximum temperature limit of 600°C was not overly restrictive with a liquid metal coolant. Finally, the disadvantages of its low burnup limit of 6500 MWD/MTU were offset by cheaper fabrication costs and our desire to keep the fissile inventory low.

Tritium Breeding Blankets

There are three tritium breeding zones (TBZs) in the reactor; one is a lithium cooled radial blanket surrounding the fission blanket (Fig. 2), and the other two are positioned at the top and bottom of the reactor. The sections of the TBZ contained in each radial blanket segment (Fig. 3) have a 2-cm thick SS inner wall immediately behind the fission blanket followed by 6 cm of lithium, 50 cm graphite, 2 cm lithium and a 2-cm thick SS outer wall. The lithium flows from an inlet header at the top to a bottom plenum and then upward through the two sections surrounding the graphite. The top and bottom blankets, identical in composition, are positioned at 500 cm from the center of the reactor, and they have a cylindrical curved pancake shape. The first two cm are SS followed by 10 cm of a beryllium region (76% Be, 20% Li, 4% SS), 70 cm of a

DEPLETED URANIUM FISSION BLANKET DATA

Item	Inner row	Middle row	Outer row
Number of fuel elements	216	216	108
Element height — total	9.8 m	9.8 m	9.8 m
— active	8.8 m	8.8 m	8.8 m
Mass of uranium	244 Mg	296 Mg	150 Mg
Number of fuel rods	19	19	19
Fuel rod diameter	26 mm	27 mm	27 mm
Fuel rod pitch	1.15	1.107	1.107
Uranium slug OD/ID	24/11 mm	24/0 mm	24/0 mm
Uranium volume fraction	41	53	31
Sodium volume fraction	47	35	57
Stainless steel volume fraction	12	12	12
Process tube OD — corners	15.94 cm	15.94 cm	15.94 cm
— flats	14.34 cm	14.34 cm	14.34 cm
Process tube wall thickness	3 mm	3 mm	3 mm
Fuel cladding wall thickness	0.4 mm	0.4 mm	0.4 mm
Maximum uranium temperature	600°C		
Maximum sodium velocity	10 m/s		

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TABLE II

FUEL ELEMENT AND PROCESS TUBE CONFIGURATION

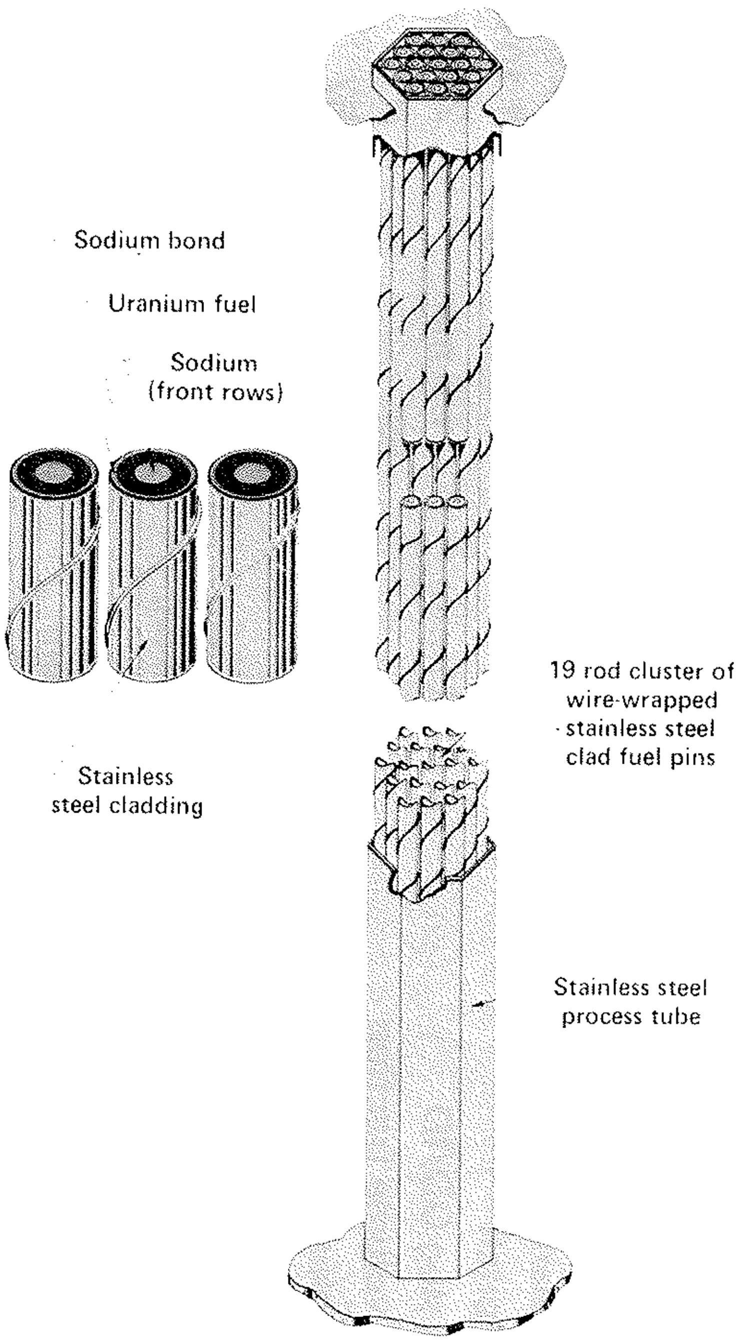
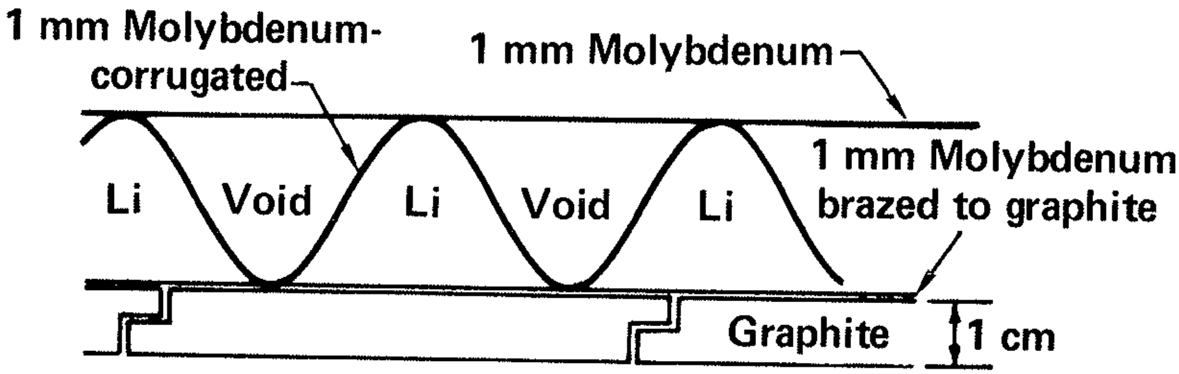


Fig. 4

graphite region (86% C, 10% Li, 4% SS), 10 cm of a lithium region (96% Li - 4% SS), and a 2 cm SS outer wall. The neutron multiplication resulting from beryllium's large (n, 2n) cross section enhances the production of tritium. The TBR for the top and bottom blankets is approximately 1.7, which allows a TBR less than 1 in the side blanket while maintaining an overall TBR of 1.1. The reduced TBR requirement in the side blanket permits a thicker fission blanket which produces more fissile fuel.

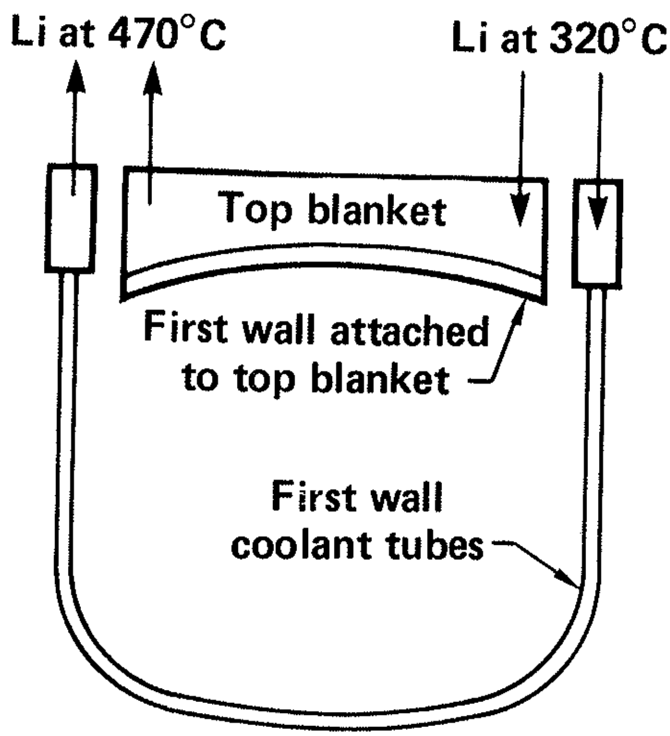
First Wall Design

The first wall has the difficult task of protecting the blanket structure from the effects of x-rays and debris produced by the thermonuclear microexplosion. In the LLL/Bechtel design, we selected a sacrificial liner of graphite to perform this task. The sacrificial liner is designed like a birdcage; it is attached to the top fusion blanket and reactor cover and can be removed by the removal of the top cover. The design and performance of the first wall are of critical importance since its lifetime and replacement time will significantly affect plant availability. The protective first wall also moderates and captures neutrons; therefore, its thickness and composition will influence the performance of the fission blanket. With this in mind, we have tried to design a thin liner which could last at least a year. In the first design[10], the wall consisted of a lithium cooled, 2-cm thick graphite liner with a density of 0.82 g/cm^3, supported by a light stainless steel structure (66.6% C, 9.7% SS, 23.7% void). The effect of this wall on the overall performance of the reactor was to reduce the energy multiplication and tritium breeding by 6 and 12% respectively, with the graphite and SS accounting, more or less, for the same absorption fraction of the neutron flux. In the revised design,[9] the first wall consists of lithium cooled, 1-cm thick graphite blocks brazed onto a 1-mm molybdenum backing (Fig. 5a). The coolant flows through a corrugated structure which is welded to the first wall backing. The brazed connection between the metal structure and the graphite blocks is needed to enhance conduction of heat from the graphite to the coolant. If the graphite is radiatively cooled, the high surface temperature will



A. First wall structure

Fig. 5a



B. First wall outline

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Fig. 5b

accelerate ablation, and it will be difficult to design a thin sacrificial wall which could last a year. On the top of the reactor, the first wall is cooled directly by the lithium in the top fusion blanket (Fig. 5b). The first wall that protects the sides and bottom of the reactor (250 MW energy deposited) is cooled by a separate lithium circuit. The first wall and cooling structure together form an integral cage which can be removed intact from the reactor vessel.

Reactor Performance

Optimization of the reactor performance over the life of the blanket requires a trade-off between power production, fuel management, tritium breeding, and plutonium production. From the beginning of the cycle, the energy generation and tritium breeding increase while the net plutonium production decreases until fresh fuel is added. In order to maintain a constant output power of 4000 MW, the laser pulse repetition rate is decreased from the beginning of the cycle to compensate for increasing energy multiplication as plutonium accumulates in the fission blanket. Reactor performance for this mode of operation is summarized in Table III. The operational parameters presented in Table III were calculated for a fission blanket lifetime of 3 full power years (4.28 calendar years). During this period, the front row fuel elements are alternately rotated and replaced at intervals of 0.75 full power years. All of the fuel elements are replaced at 3 full power years. Blanket energy multiplication increases from 6.0 to 8.3 during the cycle and the first wall neutron flux decreases from 2.0 to 1.3 MW/m^2 as the fusion power is decreased to keep the reactor output power constant. The maximum power density reached during the cycle is 220 W/cm^3, and it occurs in the first cm of the fission blanket. The fuel burnup limit of 0.6% occurs in the front row fuel at 1.5 years and in the second row fuel at 3 years. The total plutonium production by the end of the 3 year cycle is 10,500 kg, sufficient to fuel 6 (4000 MW_t) LWR's with a conversion ratio of 0.6.

Cost Analysis Of The LLL/Bechtel Design

Bechtel has estimated the capital cost of the laser fusion hybrid

OPERATIONAL PERFORMANCE SUMMARY OF THE LLL/BECHTEL HYBRID DESIGN

	Start of Life	End of Life	Cycle Avg.
SYSTEM PERFORMANCE			
Thermal power, MW_t	4000	4000	4000
Fusion Thermal power, MW_t	850	550	700
Gross Electrical Power, MW_e	1520	1520	1520
Net Electrical Power, MW_e	1195	1232	1210
Recirculating Power Fraction	0.22	0.19	0.20
System Efficiency, %	30.0	30.8	30.4
BLANKET PERFORMANCE			
Blanket Energy Multiplication	6.0	8.3	7.15
Tritium Breeding Ratio	0.99	1.07	1.03
Net Fissile Production, Kg/MW_t-yr	1.0	0.84	0.88
Maximum Power Density in Fuel, W/cm^3	189	220	204
Average Power Density in Fuel, W/cm^3	78.4	91.3	84.9
First Wall Flux, MW/m^2	2.0	1.3	1.65

TABLE III

reactor plant from conceptual design information. Since a large portion of the total plant including the thermal energy transfer and conversion, cooling, and auxiliary systems represents conventional technology, the cost estimating is based largely on background experience. The reactor, the laser, and tritium systems are conceptual, and their cost is estimated by unit and component cost methods. The total plant is based upon commercial operation; therefore, costs are assumed to apply to fifth-of-a-kind facilities.

A capital and operating cost summary of the 1200 MW_e laser fusion-fission power plant is presented in Table IV. For comparative purposes, the cost of the hybrid is contrasted with cost estimates for a typical LWR. The LWR cost estimates were made for 1976 price and wage levels while the hybrid estimates have been made for 1977 levels. In both cases no allowance has been made for future escalation.

The indirect costs in Table IV were estimated on the basis of a nine year construction time for the LWR and a 10 year construction for the more complex laser fusion hybrid. As a result, the hybrid's indirect costs account for a larger fraction of its total capital cost of $2,239 million. On a cost-per-kilowatt installed basis, the hybrid is 2.8 times more expensive than the LWR.

The cost of electricity from the hybrid is 56 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 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 its plutonium at $30 per gram.

The major issue concerning a laser fusion hybrid is neither 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 Fig. 6, the cost of electricity has been plotted as a function of the cost of fissile fuel for hybrids with varying capital costs. The intersection points of the

CAPITAL AND OPERATING COST ANALYSIS

Capital cost item (10^6 \$)	LWR 1200 MW(e)	Laser fusion hybrid 1200 MW(e)
Laser system	NA	149
Nuclear steam supply system (NSSS)	78	344
Other mechanical	101	230
Civil and structural	142	165
Piping	77	120
Instrumentation	9	32
Electrical	43	90
Total direct	450	1130
Field costs	79	215
Engineering services	80	215
Contingency	91	390
Owners cost at 8%	56	156
Interest during construction at 8%	197 (9 yr)	615 (10 yr)
Total indirect	503	1591
Total cost	953	2721
Cost per kW installed (\$)	794	2239
Operating cost item (mills/kWh)		
Capital	19.42	55.0
Fuel	6.3	−1.0
Operating and maintenance	1.5	2.0
Total operating	27.22	56.0

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Table IV

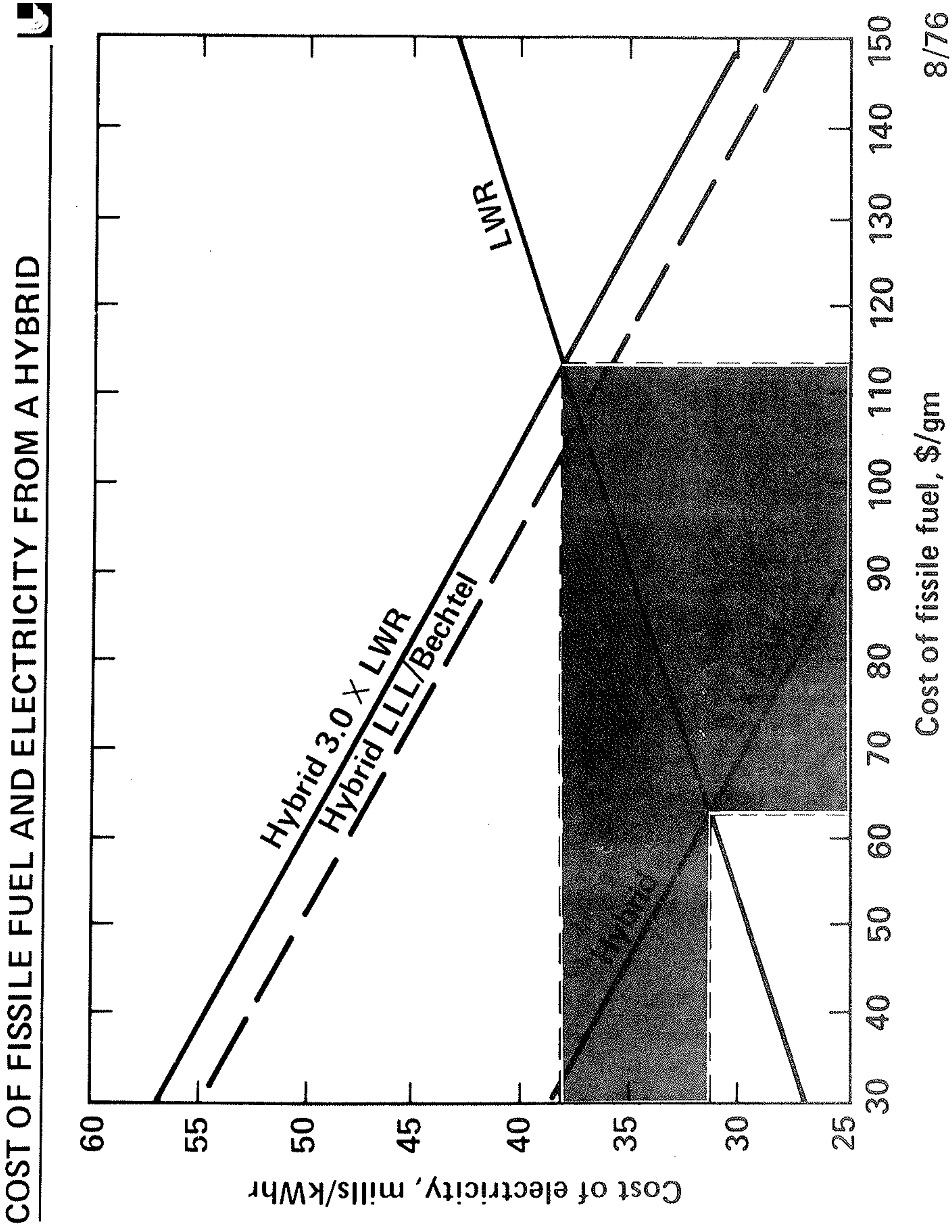
COST OF FISSILE FUEL AND ELECTRICITY FROM A HYBRID
LWR
Hybrid 3.0 × LWR
Hybrid LLL/Bechtel
Hybrid
Cost of electricity, mills/kWhr
60
55
50
45
40
35
30
25
Cost of fissile fuel, $/gm
30
40
50
60
70
80
90
100
110
120
130
140
150
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Fig. 6

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.

LLL/WESTINGHOUSE HYBRID DESIGN STUDY

The main differences between the Bechtel and Westinghouse designs have been listed in Table I. As pointed out earlier, the main product of the Bechtel design is fissile fuel, while in the Westinghouse design the major product is electricity. Electricity production was emphasized by designing a fission blanket with higher energy multiplication ($M \geq 10$) and a higher power density. The resulting smaller power plant has less recirculating power and a higher system efficiency for the same fusion energy gain.

Westinghouse[8] has chosen a very compact structure with a high first wall loading (10 MW/m^2 vs. 1.65 MW/m^2 for the Bechtel design). The reactor consists of a spherical cavity of 1-m radius surrounded by a modular blanket. A modular blanket concept was chosen to facilitate fuel handling and maintenance procedures. Fig. 7 shows a schematic representation of the reactor. Four of these units are located within the reactor building with three of them running simultaneously while one is undergoing maintenance. A lithium wetted wall concept was selected to accomodate the high first-wall loading. In this approach a thin film of liquid lithium is used to protect the first structural wall from x-rays and debris produced by the fusion microexplosion.

The structure of each module is shown in Fig. 8, and a detailed view of the fuel pin assemblies in a module is presented in Fig. 9. The compositions of the module zones are given in Table V. Different fuels were considered: Uranium metal alloys, uranium carbide (UC), uranium nitride (UN), spent LWR fuel $(UPu)_{LWR}$ in metal alloy and carbide form, and metal and uranium carbide fuel with higher enrichments of plutonium (3 and 5%). The most attractive of all these fuels appears to be

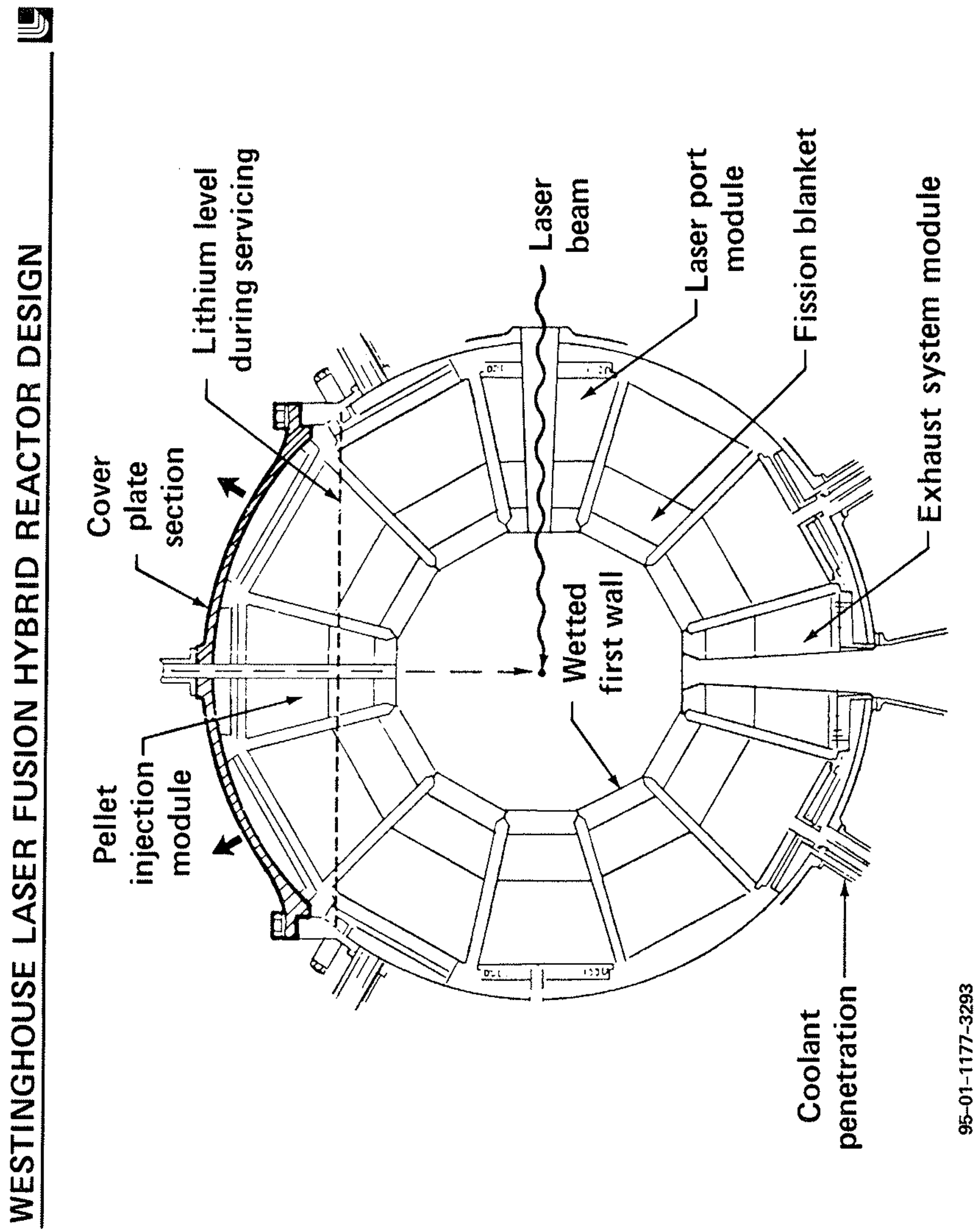
WESTINGHOUSE LASER FUSION HYBRID REACTOR DESIGN
Cover plate section
Lithium level during servicing
Laser beam
Laser port module
Fission blanket
Exhaust system module
Pellet injection module
Wetted first wall
Coolant penetration
95-01-1177-3293

Fig. 7

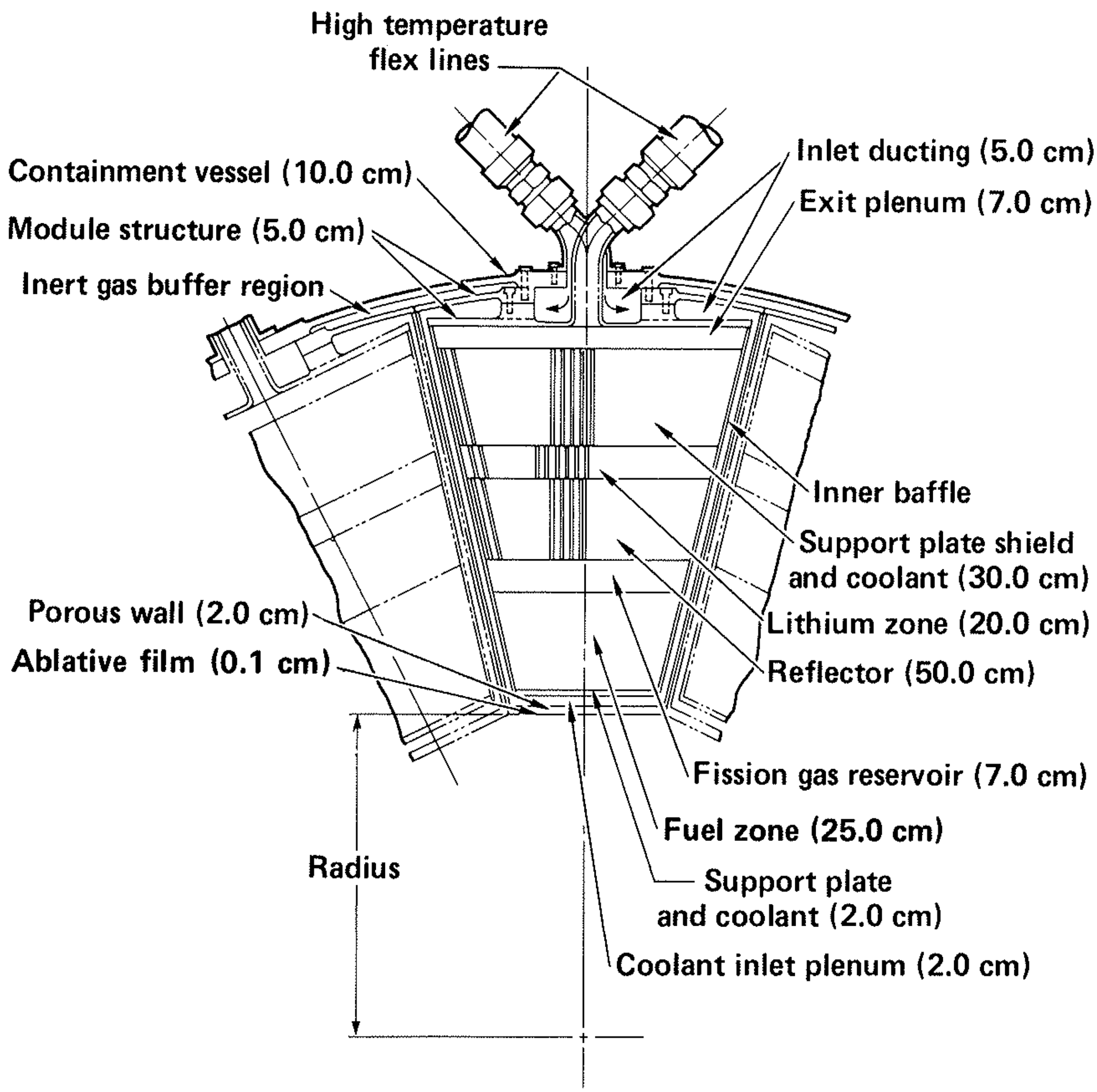

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Fig. 8

MODULE POROUS FIRST WALL AND SUPPORT PLATE

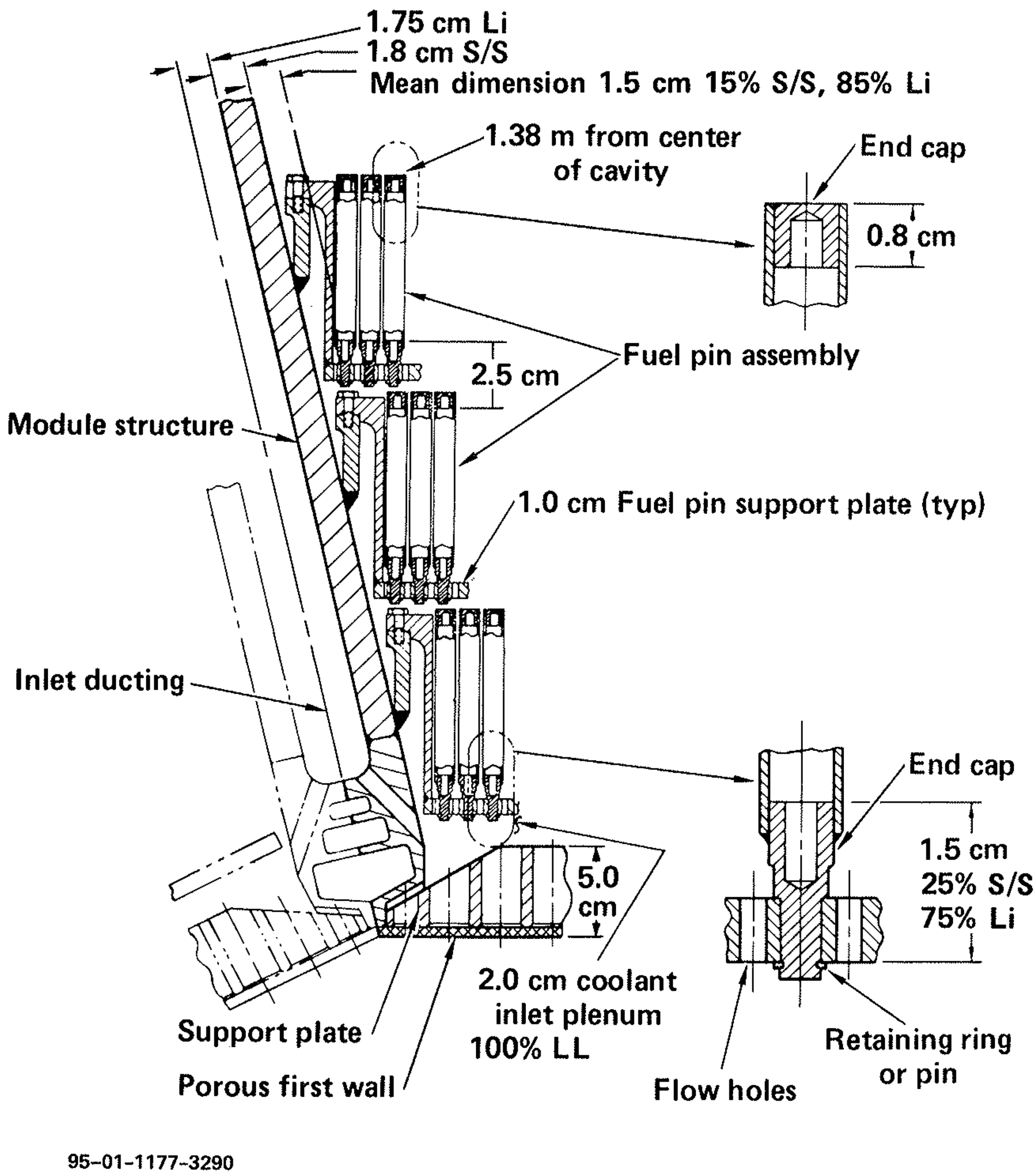

Fig. 9

COMPOSITION BY ZONE FOR FAST REACTORS DESIGNS

Zone	Zone thkns	Zone description	Composition (vol %)			
			Design 1	Design 2	Design 3	Design 4
1		Cavity				
2	0.1 cm	Ablative film	← 100 - Li →			
3	2.0	Porous wall	← 50 - 316 SS; 50 - Li →			
4	2.0	Coolant plenum	← 10 - 316 SS; 90 - Li →			
5	2.0	Support plate and coolant	← 50 - 316 SS; 50 - Li →			
6	25.0	"Fuel zone"	15 - Li 23.1 - 316 SS 9.3 - void 52.6 - U-7Mo	15 - Li 23.1 - 316 SS 9.3 - void 52.6 - UC	15 - Li 23.1 - 316 SS 9.3 - void 52.6 - UN	15 - Li 23.1 - 316 SS 9.3 - void 52.6 - (UPu)C*
7	7.0	Fission gas reservoir	← 75.14 - void; 9.86 - 316 SS; 15 - Li →			
8	50.0	Reflector	← 80 - C; 10 - 316 SS; 10 - Li →			
9	20.0	Lithium zone	← 16 - 316 SS; 84 - Li →			
10	30.0	Support plate and coolant	← 50 - 316 SS; 50 - Li →			
11	7.0	Exit plenum	← 10 - 316 SS; 90 - Li →			
12	5.0	Module struct	← 100 - 316 SS →			
13	5.0	Inlet ducting	← 10 - 316 SS; 90 - Li →			
14	10.0	Containment vessel	← 100 - 316 SS →			

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*U, Pu composition of spent LWR fuel or higher ^{239}Pu enrichments. See text.

Table V

spent fuel from LWRs in carbide form. Spent LWR fuel is attractive because it is cheaper and readily available, and the carbide form is preferred because it allows higher fuel burnup at the high fuel temperatures which result from high power density operation. Table VI lists the composition of the LWR fuel. From an initial ^{235}U enrichment of 3.1%, the fissile concentration drops to approximately 1.6%. It consists of unburnt ^{235}U plus two fissile isotopes of plutonium (^{239}Pu and ^{241}Pu) which are generated from neutron captures in ^{238}U and ^{240}Pu.

The operational parameters calculated for all of the fuels listed above are given in Table VII. It should be noted that a fast-blanket design using carbide fuel would require a Pu enrichment of 5% or larger to achieve an energy multiplication greater than 15. The energy multiplication in the LLL/Westinghouse Design is not as high as in other reported blankets because of the high volume fraction of structure (approximately 25%) in the blanket region and the use of natural lithium as the blanket coolant. Additional blanket structure is required because of the high wall loadings (10 MW/m^2) and power densities selected for the design.

The time dependent performance of the reactor over a two and a half-year operating period was studied for the spent fuel in carbide form. The increase in energy multiplication and tritium breeding are shown in Fig. 10a along with the decrease in the plutonium production per year. The change of fuel burnup and power density (maximum and average) as a function of time are shown in Fig. 10b. The time averaged values for the energy multiplication, power density, and tritium breeding ratio during the 2.5 years are summarized in Table VIII, together with the total amount of Pu produced and the total burnup by the end of this period. It must be pointed out that power density and fuel burnup are parameters that vary in space as well as in time. The values given in Table VIII has been spatially averaged over the dimensions of the fission blanket. Power density and fuel burnup are maximum at the inner edge of the blanket due to the larger fission cross sections for high-energy source neutrons.

TYPICAL COMPOSITION OF SPENT LWR FUEL

Initial U-235 = 3.1%
Burnup 32,000 MWD/MTU

Element	kg/initial MTU	%
U-234	0.14	0.014
U-235	8.6	0.889
U-236	4.0	0.414
U-238	945.0	97.72
Pu-239	5.4	0.558
Pu-240	2.1	0.217
Pu-241	1.3	0.134
Pu-242	0.5	0.052

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TABLE VI

LLL/WESTINGHOUSE FAST REACTOR DESIGNS

Initial Operational Characteristics	U · 7% Mo	UC	UN	(U Pu)-7% Mo LWR	(U Pu) C LWR	(U Pu)$_{3\%}$ C	(U Pu)$_{5\%}$ C
Blanket energy multiplication, M	6.7	5.5	6.0	8.5	6.6	10.4	15.4
*Average power density in the fuel, P_D [W/cm^3]	173	142	153	223	170	275	414
*Net ^{239}Pu production in Kg/MW$_{th}$ per year	1.35	1.44	1.15	1.05	1.15	0.73	0.46
*Number of fissile atoms per source neutron	1.48	1.31	1.13	1.49	0.25	1.27	1.21
*Average burnup in %	0.85	0.90	0.95	1.15	1.10	1.80	2.75
Tritium breeding ratio	0.75	0.78	0.79	0.84	0.80	0.90	1.14

*All values normalized to a first-wall fusion neutron fluence of 10 MW/m^2 per year

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TABLE VII

AVERAGE VALUES OF THE OPERATIONAL
PARAMETERS FOR THE LLL/WESTINGHOUSE HYBRID

	$(UPu)_{LWR}C$
Average Energy Multiplication	11.0
Average Tritium Breeding Ratio	0.98
Maximum Power Density in Fuel, W/cm^3	640.0
Average Power Density in Fuel, W/cm^3	330.0
Total Burnup in Percent	5.8
Net ^{239}Pu Production in Kg/MW_t-year	0.63

TABLE VIII

TIME DEPENDENCE OF THE PARAMETERS FOR THE REACTOR WITH $(UPu)_{LWR}$ C FUEL AND 10 MW/m^2 FIRST WALL LOADING

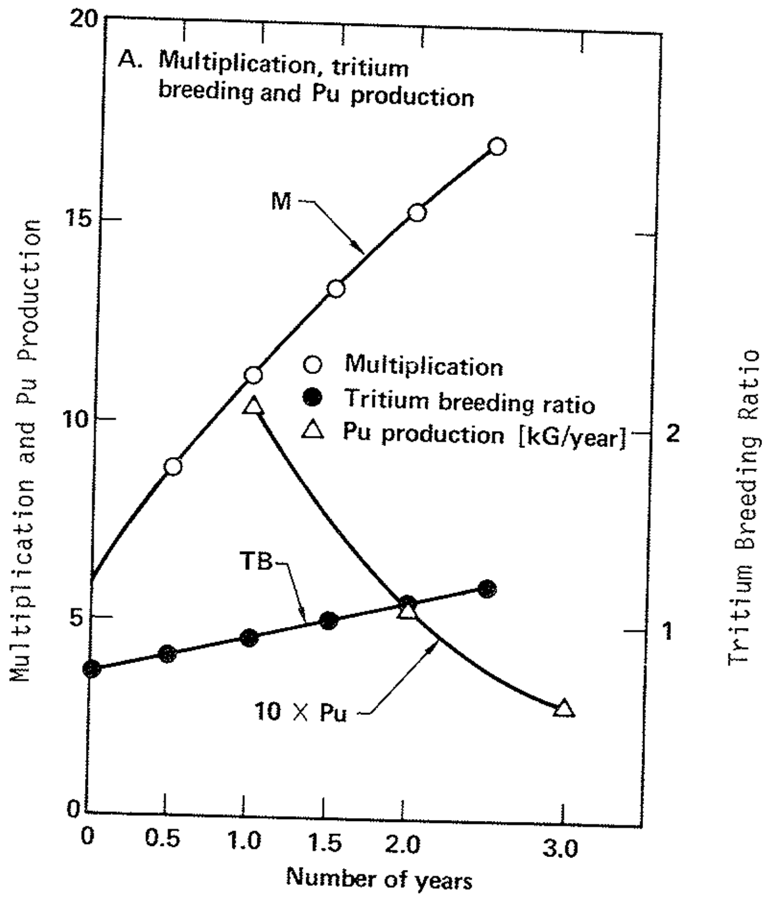

Fig. 10b

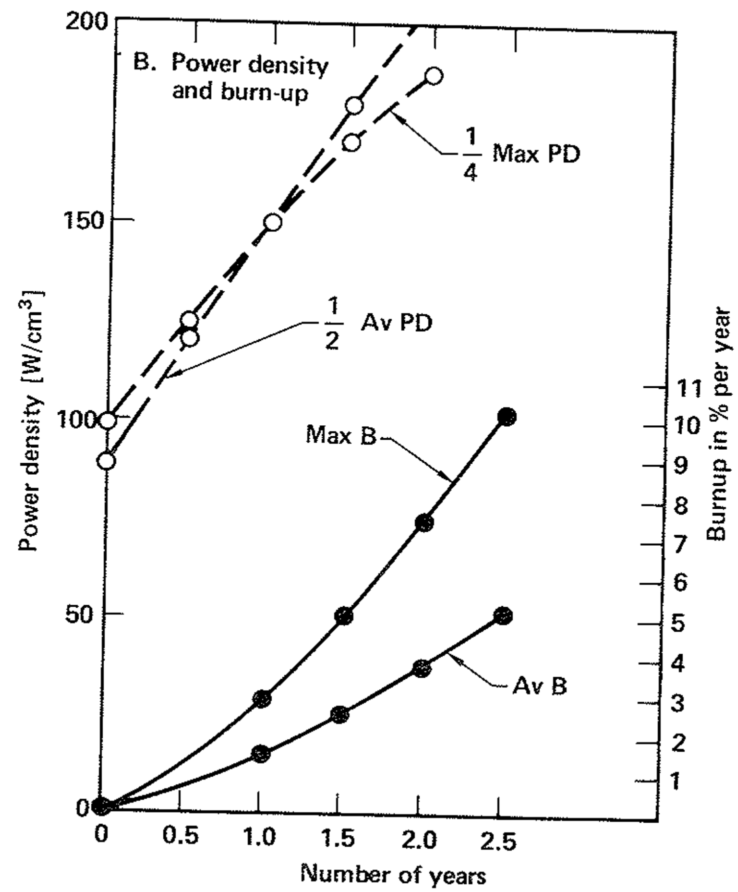

Fig. 10a

The initial tritium breeding ratio of 0.73 reached a value of 1.20 at the end of the 2.5-year period. The initial tritium breeding ratio could be increased by enriching the lithium in ^6Li or by reducing the amount of structural and shielding material used behind the fission blanket.

CONCLUSION

The hybrid reactor design that we completed in collaboration with Bechtel Corporation featured a cylindrical fast-fission blanket fueled with depleted uranium. The inner radius of the blanket is 5 m with a height of 10 m. The average energy multiplication of the fast fission blanket for a three-year period is 7.15 and the fissile production is 0.88 kg/MW_t yr. This is enough fissile material to fuel approximately six LWRs of equivalent thermal power. In contrast, the hybrid design that we completed with Westinghouse is a spherical fission blanket fueled with spent LWR fuel. Four reactor chambers were employed with each vessel having an inner radius of 100 cm. The average blanket energy multiplication for a 2.5 year period is 11 and the fissile production is 0.63 kg/MW_t-yr. This is enough fissile material to fuel four LWRs of equivalent thermal power. The results presented here show that the Westinghouse design provides 50% more energy multiplication than the Bechtel design; however, this enhanced energy multiplication is gained at the expense of a 30% reduction in fissile fuel production.

Our earlier neutronic scoping studies identified several attractive features of laser-driven fusion hybrids: 1) Hybrids can be designed to meet a broad spectrum of energy multiplying and fissile fuel producing requirements; 2) Hybrids can operate in a regime which requires an order of magnitude less laser/pellet performance than pure laser fusion; and 3) Hybrids produce ten times more fissile fuel than breeder reactors. It is encouraging to note that, in general, these attractive features have been retained in this comparative analysis which has been conducted at a higher level of engineering design detail than the earlier studies. In the process of performing these more detailed engineering design studies with Bechtel and Westinghouse, we came to several conclusions: First, laser fusion hybrids should not be designed purely as power producers since they will cost two to three times more than LWRs and be more expensive than fast breeder reactors. Therefore, hybrids that make sense will have to produce fissile fuel for existing burner reactors. (In Fig. 6, we showed that the cost of electricity in an LWR-hybrid scenario is insensitive to the capital cost of the hybrid. It will increase over present prices by only 20-40% when the hybrid cost is two to three times more than an LWR.) Second, hybrids which produce fissile fuel for existing LWR can extend the energy available from those economically proven reactors by two orders of magnitude. Finally, laser-driven hybrids can accommodate a fission blanket in a more straightforward manner than magnetic confinement systems.

The results presented here have led us to conclude that hybrid studies should be a continuing and integral part of the laser fusion technology effort. Future studies should seek to make a closer link between evolving laser fusion performance and the fissile fuel requirements of fission burner reactors. In so doing, it will be possible to maximize the ratio of energy from hybrid-fueled LWR's to hybrid energy, thereby making the cost of electricity in the combined scenario less sensitive to hybrid capital cost.

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DR. MOSES: The paper is now open for questions.

DR. RIBE: Ribe, U.W. What are some of the details of the actual fusion driver? Are they gas lasers (CO_2), or what?

DR. MANISCALCO: In the design study that we did with Bechtel, we designed a photolytically pumped Group 6 laser; however it was just an initial cut and it was performed by assuming that the physics was going well. In costing the laser design, we found that the major portion of the cost wasn't in the laser cavity, but rather it was in the gas-flow and power conditioning equipment. I think that Larry Booth finds similar trends when he does the same thing with the CO_2 laser.

We also took advantage of the fact that the laser system could be built and housed separately from the reactor system. Therefore, we assumed that the laser and its building could be built in five years rather than the ten years it normally takes for the reactor and its building. This cut the indirect cost of the laser system by a factor of two.

DR. RIBE: Can you say anything about the pellet?

DR. MANISCALCO: As I said, we took a laser system with an efficiency of about one percent and we found that we needed pellet gains in the neighborhood of 100. As you know, we're going down a road in the inertial confinement fusion program where we should be able to answer the question of achieving these kinds of pellet gains by about 1983 when the second phase of the SHIVA NOVA laser system comes on line. We expect to demonstrate pellet gains greater than 20.

DR. RIBE: That's the classical pellet gain, as I recall. A few years ago, the laser fusion people used to estimate gains of 100 from just bare DT systems. Do you think the use of a bare DT liquid-solid pellet might work for fusion-fission?

DR. MANISCALCO: No, however, but not for the reason you indicated. A bare DT pellet would require symmetrical illumination. This means that it would be necessary to irradiate the pellet with 10 or 20 beams of low focal length optics which are in as close as the first wall, if not closer. I can't see how we could do this in a fusion-fission system.

For that reason, we've gone to different target designs in laser fusion. To answer your question about the adequacy of pellet gains of 100 which were considered for laser fusion in the past. There still are some scenarios, in which this gain is adequate and fusion-fission one of them.

To provide power with pure laser fusion, and to have a recirculating power fraction of less than 25 percent, the product of laser efficiency in pellet gain must be greater than 10. Therefore, with a pellet gain of 100, you'd need a 10 percent efficiency laser.

If you have a 1 percent efficient laser, for pure laser fusion now, you're going to need a pellet gain of 1,000. Hybrids with an energy multiplication of 10 can use a 1 percent efficient laser with pellet gains of 100. Does that help you.

DR. RIBE: Yes, that helps a great deal.

MR. LOTKER: Mike Lotker, Booz, Allen & Hamilton.

Jim, have you looked at the sensitivities of these costs to laser costs or efficiencies?

MR. MANISCALCO: Yes. I have a vugraph which I didn't show because I ran out of time, but since you ask, I'll show it now. (See figure on next page.)

COST SENSITIVITY FACTORS FOR LASER FUSION—FISSION REACTOR POWER PLANT (1200 MWE)

ITEM		MILLS/KWH
BASE VALUE*		50
CAPITAL CHARGE RATE	(12%)	39
	(15%)	50
	(18%)	61
$100 X 10^6 DIRECT COST LESS		45
100 MWE POWER INCREASE		46
FUSION FUEL (50¢ / PELLET)		58
FISSION FUEL CYCLE (DOUBLE COST)		53
(DOUBLE REVENUE)		38
PLANT CAPACITY FACTOR	(70%)	50
	(75%)	47
	(80%)	43
LASER EFFICIENCY–AVERAGE	(0.75%)	58
	(1.00%)	53
	(1.34%)	50
	(2.00%)	47
URANIUM BLANKET MULTIPLICATION–AVERAGE	(10.34%)	50
	(15%)	46
	(20%)	45

*FISSILE FUEL AT $64/GM, CAPITAL COST AT $2239/KW

The base case of 50 mills/kwhr(e) results in fuel selling for $64 a gram. The captial cost of the LLL/Bechtel hybrid was set at $2200 a kilowatt.

First, we looked at the effects of a change in captial charge rate. The base case was 15 percent, and you can see what happens when it varies from 12 percent to 18 percent.

Next, we asked the question, what happens when the direct costs change by $100 million and you see a 5 mill per kilowatt hour change. And then we looked at a change in the cost of a pellet. In the base case, we assumed a fusion target costs of 10 cents per pellet. You can see what happens if the fusion target cost goes up to 50 cents per pellet.

We also looked at what happens when we double the cost of the fission fuel cycle and then what happens when we double the revenue from the sale of fissile fuel. Here you can see that the cost of electricity is most sensitive to fissile fuel cost. Doubling the fissile fuel revenue decreases the cost of electricity by 12 mills per kilowatt hour.

With regard to plant capacity factor, there is very little change in the cost of electricity as this variable changes from 70 to 80 percent. For laser efficiency, our base case was 1.34 percent. You can see how sensitive the cost of electricity is to a variation the laser efficiency, from .75 to 2 percent.

In the last entry, we were looking at the effects of increasing blanket energy multiplication in the blanket, so these should be just 10.3, 15 and 20. And that would be the multiplication, if the blanket were in spherical geometry covering the four π point.

So in the base case, again, the results show that a 50 percent increase in blanket energy multiplication, decreases the cost of electricity by about 4 mills per kilowatt hour.

DR. MOSES: Time for one more quick question.

MR. DOHERTY: Pat Doherty, Combustion Engineering.

Could you tell me what the breeding ratio was for the LMFBR shown on your economics chart? You show an LMFBR, a light water reactor, and then your various hybrid systems. Would you show where your various capitalization costs are?

DR. MANISCALCO: Yes. That was an LMFBR with a breeding ration of 1.2, uranium oxide.

MR. DOHERTY: I would like to make a point and suggest that if you used 1.4, the line would have fallen at about your 3.5 line for your hybrid system.

DR. MANISCALCO: I haven't looked at that; but I think it does show you where the leverage is in these systems.

MR. MOSES: Is it a quicky?

DR. ROSE: Rose, Math Sciences.

I think some of your comments are the most important points of this whole meeting and they are not only applicable for laser fusion, but they apply for all fusion systems. You show, that, supplying fuel to five or ten reactors, I think you used six in your cases, then by averaging the fuel cost and the electricity cost from such an ensemble of fuel factories and LWRs, you are no longer talking about electricity cost of more than 50 mills/kWhr, but about the cost at the intercept of the supply and demand curves. These curves show that a hybrid fuel factory costing three times the LWR cost, increases the electricity cost of the system by only 40 percent. There is nothing more important to a new technology than to be able to introduce it when it is not sensitive to the capital cost. That is what is being shown by those slides. I think the most important advantage of a hybrid system is its ability to be introduced into the economy because it has this depressed dependence on capital cost. If you made the same numbers for a pure fusion reactor, the cost of electricity would go one-for-one with capital cost. Here, you are introducing something that is three times as effective, that is, you can go up in capital cost between a factor of two and three and still be competitive in electricity cost. I have used your work widely and I compliment you for it.

Preliminary Evaluation of A U-233 Fusion-Fission Power System Without Reprocessing*

by

K.R. Schultz, R.H. Brogli, G.R. Hopkins
G.W. Shirley and S.C. Burnett
General Atomic Company
P.O. Box 81608, San Diego, CA. 92138

ABSTRACT

A preliminary evaluation was made of the technical feasibility of using a fusion reactor to breed ^{233}U from thorium in high temperature gas cooled fission reactor fuel and of then using this fuel in a fission reactor without reprocessing. Estimates of the neutronic performance of thorium fusion reactor blankets were made. The fuel cycle characteristics of high temperature gas-cooled reactors operating on nonreprocessed fusion-bred ^{233}U fuel were also estimated. The system performance of symbiotic fusion-fission power systems without reprocessing was then determined.

The results of this preliminary study show that the concept of fusion breeding of fissile fuel without reprocessing is technically feasible. An adequate concentration of ^{233}U in thorium can be attained in a fusion reactor blanket to allow operation of a high temperature gas cooled reactor. Estimates of the fuel materials damage indicate that the breeding and subsequent burning of ^{233}U can be accomplished within the currently predicted materials limitations of high temperature gas-cooled reactor fuel. Using this no-reprocessing concept, 1 MW of fusion plasma power could support as much as 27 MW(t) of fission reactor power. Thus preliminary evaluation shows that the concept is technically feasible and warrants more detailed study.

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1. INTRODUCTION

1.1. THE CONCEPT

The basic concept of producing fissile material in the blanket of a fusion reactor is promising because large quantities of high quality fissile fuel could be produced and because the fusion reactor needs no fissile inventory, thus freeing it from doubling time constraints (Refs. 1-3). The idea of using the fusion bred fuel without reprocessing is particularly attractive because of the uncertain state of commercial reprocessing and refabrication technology and because it offers to alleviate concerns about nuclear proliferation. Only fertile material would be fed into a fusion hybrid breeder and the bred fissile material would have a high content of fission products that would never be removed from the fuel, as it would never be reprocessed. These fission products would protect the bred fuel from diversion just as fission products currently protect spent LWR fuel from diversion during shipping and storage.

To take full advantage of the no-reprocessing or "Refresh" cycle concept the thorium-uranium fuel cycle was selected. In this fuel cycle a neutron is absorbed by the thorium fertile material producing uranium-233 which is the best fissile material for use in a thermal spectrum fission reactor. High-Temperature Reactor (HTR) technology, which uses graphite moderator and helium coolant with the thorium-uranium fuel cycle, was chosen for this design evaluation. The HTR fuel is attractive to the refresh cycle concept because it is capable of very high material burnup. Although fuel life is ultimately limited by material fast neutron damage, burnup is limited by fuel inventory considerations and not by materials constraints, and burnups in excess of 70% (700,000 MWD/MT) have been achieved in test capsules. The fuel cycle and reactor technology of the HTR are also fully developed and enjoy strong international interest through the High-Temperature Gas-Cooled Reactor (HTGR) program in the USA and the German High-Temperature Reactor (HTR) program.

The German pebble bed reactor and the HTGR are very similar except for the physical configuration of the fuel elements. The HTGR uses hexagonal prismatic graphite blocks to contain its fuel with the helium coolant flowing through holes drilled in the blocks. The pebble bed reactor uses spherical graphite balls to contain its fuel with the helium coolant flowing through the spaces between the balls. In other respects - fuel cycle, power conversion system and mechanical design - the reactors are quite similar. The choice of HTR fuel technology for this study is also attractive because of the advantages of the HTR design concepts. The refractory graphite base fuel, low power density and stable helium coolant offer significant safety advantages. The high temperature capability offers high thermal efficiency and the possibility for dry cooling and direct cycle gas turbine power conversion. The thorium-uranium fuel cycle with graphite moderator offers a high conversion ratio for effective fuel resource utilization. Because of the high temperature helium coolant the reactors offer the potential for significant future development and improvement; direct cycle gas turbines, binary very high efficiency cycles and high temperature process heat applications are potentially possible.

1.2. TECHNICAL CONCERNS

There are a number of technical considerations that have to be addressed to evaluate the feasibility of the no-reprocessing fuel production concept. In any fission or fusion reactor design the irradiation capabilities of the materials are of prime concern. The capabilities of the fuel materials are particularly critical for the no-reprocessing concept as the fuel must survive irradiation in both the fusion and fission reactors and it is desirable to be able to recycle the fuel several times before the material's limits are reached and it must be discarded. The nuclear performance of the fusion reactor breeding blanket is a critical aspect of the concept and must be analyzed. An adequate quantity of ^{233}U must be bred and the percentage of ^{233}U in thorium (which will be referred to as the "enrichment") that can be reached must be sufficient to operate the burner reactor. The feed requirements of the burner reactor must therefore be investigated to determine how to minimize both the quantity and enrichment of fuel that is required. Since the same fuel element will be used in both fusion breeder and fission burner, it is important that the characteristics of the two reactors be matched together to obtain an optimum symbiotic fusion-fission power system. Fuel production is the prime function of the fusion reactor blanket. Consideration should be given to the fusion reactor requirements imposed by hybrid blanket characteristics. These technical concerns are discussed in the following chapters.

2. MATERIALS CONSIDERATIONS

The fuel element for the HTGR reactor is shown in Fig. 1 and consists of a graphite block into which holes are drilled for the helium coolant and fuel rods. The fuel rod consists of a graphite matrix which acts as a binder to hold the fuel particles together. The pebble bed reactor fuel element is shown in Fig. 2 . It consists of fuel particles dispersed in a graphite matrix which is surrounded by 5 mm of unfueled graphite. (The fuel particles for both fuel elements are identical.) The thorium BISO particle is composed of a spherical kernel in the center, surrounded by a buffer layer of porous carbon and a high density isotropic pyrocarbon shell which provides the primary containment for the fuel and fission products. A TRISO particle has been developed for high burnup which has a thin shell of silicon carbide between two high density pyrocarbon layers around the inner low density buffer layer. This silicon carbide layer was developed to provide a very effective barrier against diffusion of fission products out of the fuel particle.

The present HTGR and pebble bed reactor designs are optimized for a four year life of a fuel element in the reactor. End-of-life of the graphite is determined by radiation damage. Basically, graphite materials exhibit rapid swelling or growth in at least one direction when a certain fluence limit is reached. This fluence limit for end-of-life varies with the specific type of carbon material, its fabrication history, and its operating temperature. The fluence limit for the fuel block material is currently about 1×10^{22} neutrons/cm^2 Equivalent Fission Fluence for Graphite Damage (EFFGD). At this limit, the blocks are beginning to show the rapid increase in dimensional change. Fuel particle lifetime is limited by several criteria. These are thermal migration of the fuel kernel, chemical attack of the coatings by fission products, buildup of fission gas

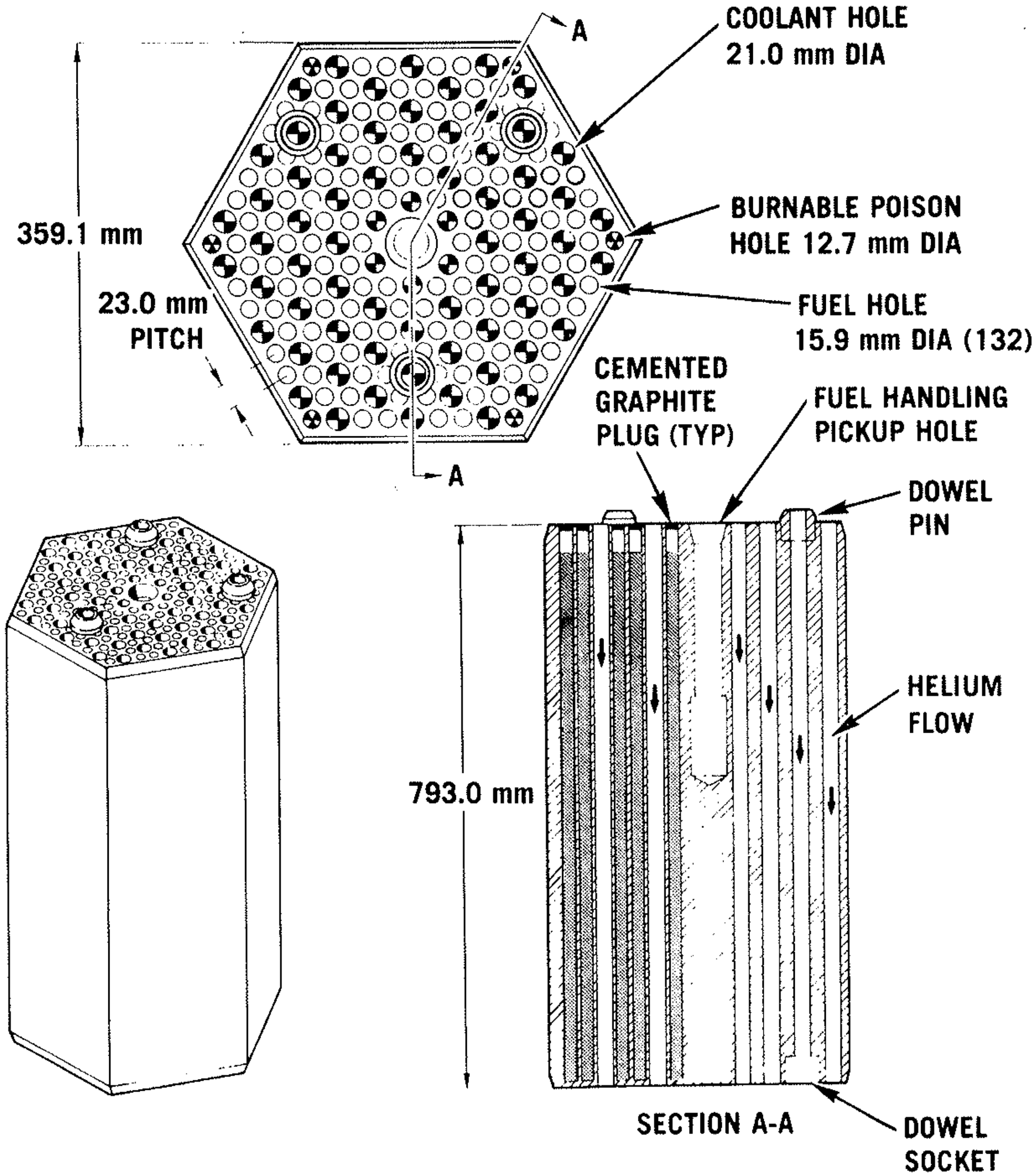

Fig. 1. HTGR fuel element

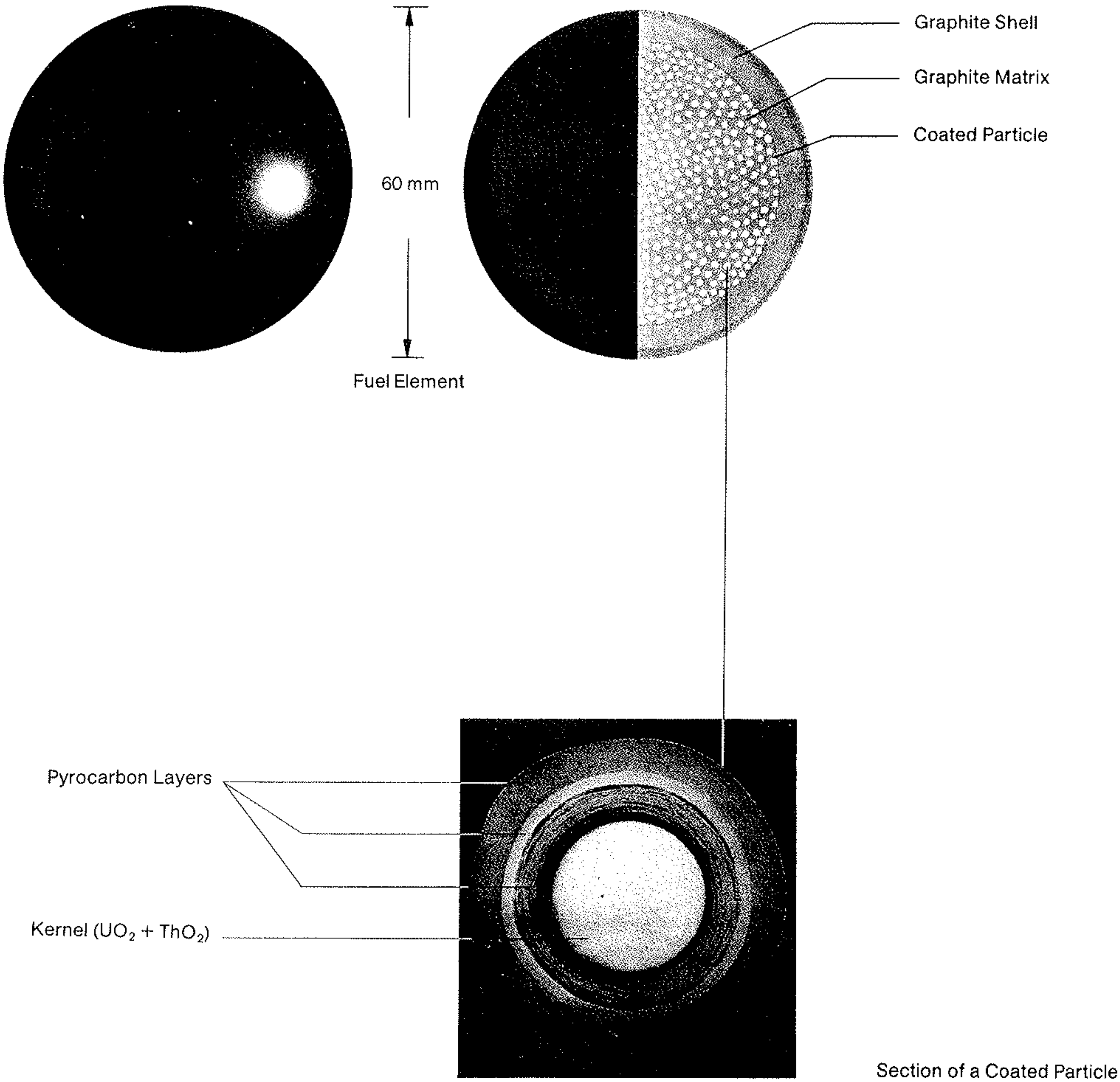

Fig. 2. Spherical pebble bed reactor fuel element

pressures within the coatings and radiation damage of the pyrocarbon coatings and the matrix material surrounding the particles. The first two of these limits are very temperature sensitive and should be extended by irradiation at low temperatures in the fusion reactor. The limit due to fission gas buildup can be adjusted by particle design. Thus, the critical limit on fuel lifetime will be that due to irradiation damage of the pyrocarbon coatings, primarily the outer pyrocarbon coating in a BISO or TRISO particle design, and irradiation damage of the matrix material.

The radiation damage picture for fuel block graphite is shown in more detail in Fig.3 for a typical near-isotropic extruded material - H-451 graphite. Since the material is anisotropic in the planes parallel to the extrusion axis and perpendicular to the extrusion axis, two sets of curves are shown in which the linear dimensional changes are plotted versus neutron fluence with graphite temperature as the parameter. Inspection of these curves shows, for example, at 800°C a limiting fluence of about 1.8×10^{22} n/cm^2, EFFGD, from radial expansion. Above 1000°C the limiting fluence falls to 1×10^{22} (EFFGD). These curves also suggest that the fuel life can be extended in a fission/fusion symbiotic system by operating the material at a lower temperature. As an example, at 400°C, a lifetime of 4.5×10^{22} n/cm^2 EFFGD would be projected. Experimental data on the cumulative effects of irradiation at changing temperatures are meager, however, and confirmation is needed.

It has been further shown that although more isotropic, fine grained graphites have higher thermal expansion and are more expensive, they have smaller neutron damage expansion rates, giving typically 50% more fluence lifetime. Substitution of these materials for the graphite in the present HTGR design would increase their fluence limits accordingly. Some typical values are shown in Table 1 which gives lifetime fluences with a safety factor of approximately 1.3 for two typical graphites. This shows limits up to 4.5×10^{22} n/cm^2 EFFGD.

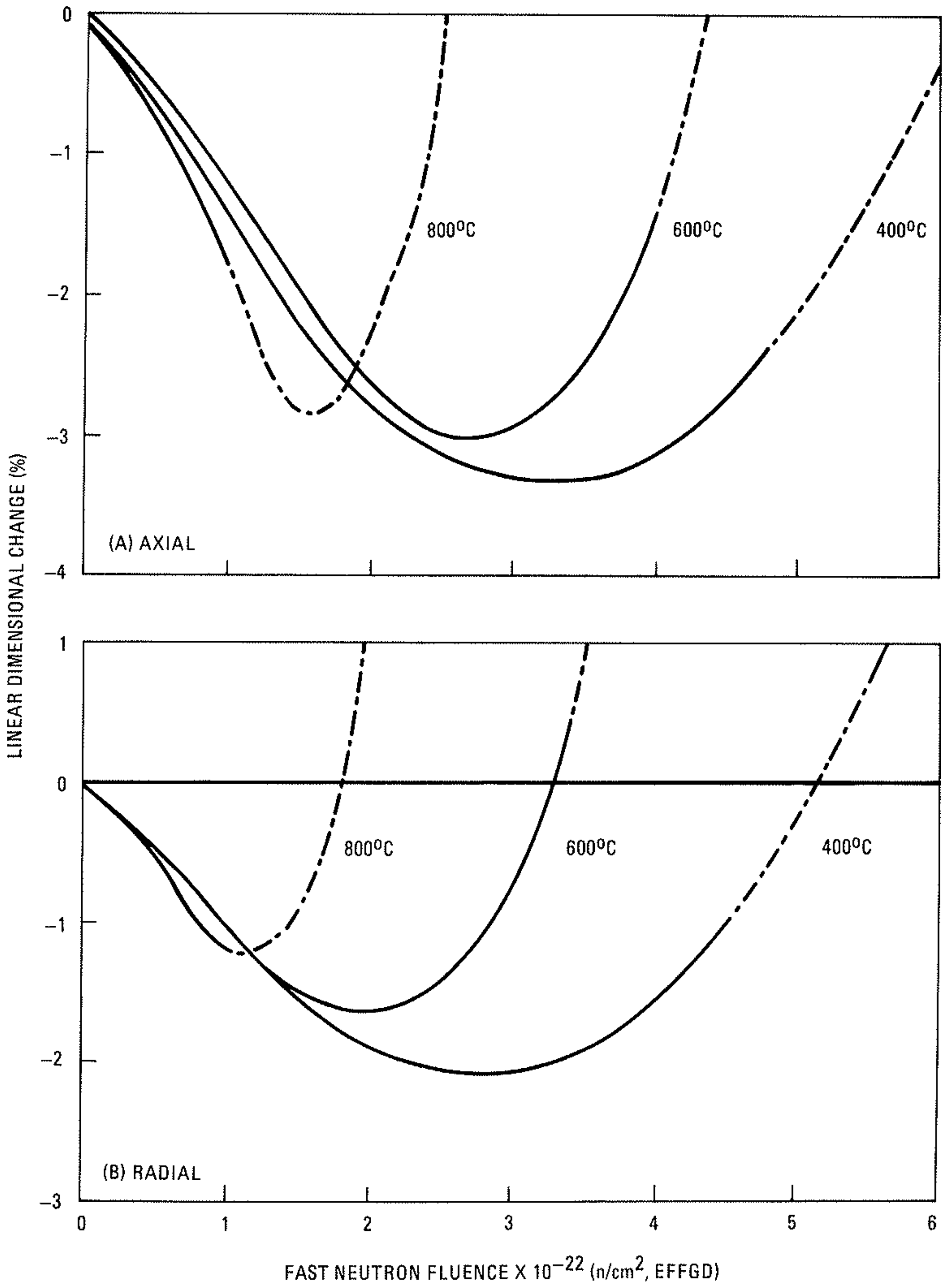

Fig. 3. Dimensional changes in H-451 graphite as a function of fast neutron fluence (a) parallel to extrusion axis, (b) perpendicular to extrusion axis

TABLE 1
ESTIMATED LIFETIME LIMITS FOR GRAPHITES

Graphite Type	Limiting Factor	Lifetime Fluence (10^{22} n/cm^2, EFFGD) At 400°C	At 600°C	At 800°C
Coarse grained near isotropic extruded (e.g., H-451)	Zero radial expansion	3.8	2.5	1.4
Fine grained isotropic (e.g., PoCo AXF-5Q)	1% expansion	4.5	3.0	2.0

The behavioral situation for fuel particle carbon coatings and the fuel rod graphite matrix material appears at first glance to be very similar to that for the fuel block graphite. The fabrication processes and graphite microstructures are quite different, however. Pyrocarbon coatings typically operate at about 1000°C and show maximum lifetime fluences of 1.0×10^{22} n/cm^2 EFFGD. Data are available up to 1350°C with resultant fluences limits of 5-6 x 10^{21} n/cm^2 EFFGD. This again suggests operation at lower temperatures to extend the lifetime either in the fusion reactor breeding cycle and/or the HTGR burnup cycle. In this case however, irradiation behavior data do not exist below 1000°C and increased lifetime projections would need experimental confirmation. The fuel rod matrix material would also need confirmation that it can stand up to increased fast neutron fluences, but it does not appear at present to be as limiting as the particle coatings. Improved matrix materials are possible, if required, by increasing their graphite content.

In contrast to carbon materials, silicon carbide exhibits very stable irradiation behavior at fluences up to approximately 5 x 10^{22} n/cm^2 and at temperatures up to about 1000°C. With this as the basis, several methods of increasing the lifetime fluence of carbon coatings and materials have been proposed. First, silicon carbide alloyed pyrocarbons show irradiation damage resistance intermediate between carbon and silicon carbide, and could be used for particle coatings in either the BISO or TRISO types to increase lifetime. It appears that 1.5×10^{22} could be achieved with SiC alloyed coatings under typical HTR conditions. Alternately, the coating thickness of the silicon carbide layer could be increased to the point where it becomes the structural unit and the outer carbon layer could be dispensed with, increasing the fluence limit to 1×10^{23} n/cm^2. Since SiC coated particles cannot be easily bonded directly into fuel rods of spheres, these particles would be irradiated loose in the fusion reactor where high fast neutron exposures occur. They could then be coated with an outer pyrocarbon layer, bonded into fuel elements and used in the fission reactor. With major fuel redesign to utilize loose coated particles in a graphite matrix, hot refabrication would not be required and only the graphite would need be replaced periodically.

A point worth noting is that neutrons with energies between the fission spectrum and the 14 MeV fusion energy are about equally effective in displacement radiation damage to carbon. Thus the radiation damage rate per neutron in a fusion reactor should be comparable to that in a fission reactor for carbon and silicon carbide. No experimental data are available, however, to project the damage that might occur due to enhanced helium production from (n,α) reactions by the higher energy fusion neutron spectrum.

In summary, the present HTGR fuel design should provide a lifetime, including a 1.3 safety factor, of combined fusion-fission reactor fluence of about 1.4×10^{22} n/cm^2 EFFGD at 800°C. Modest changes through lowering operating temperatures, coating design, and graphite material selection should provide improvement to 2×10^{22} n/cm^2 EFFGD. Improvement by a factor of seven to a lifetime limit of 1×10^{23} n/cm^2 EFFGD appear possible through the extensive use of silicon carbide in place of carbon, but this would require hot refabrication of fuel elements to replace the graphite components or possibly redesign of the fuel system. These present limits and potential future limits are summarized in Table 2.

TABLE 2
HTGR FUEL MATERIALS FLUENCE LIMITS

Component	Present Limit	Potential Limit
Graphite structure	1.4×10^{22} (800°C)	3.8×10^{22} by lower temperature 2.0×10^{22} by improved graphite 4.5×10^{22} by improved graphite and lower temperature
Coated particles	1.0×10^{22} (1000°C)	1.5×10^{22} by use of SiC alloyed pyrocarbon 4.5×10^{22} by lower temperature and possible use of SiC alloyed pyrocarbon 1.0×10^{23} or more by use of all SiC coatings
Matrix material	$>1.0 \times 10^{22}$	4.5×10^{22} by lower temperature and possible use of improved materials

3. FUSION BLANKET ESTIMATES

3.1. DESIGN CONCEPTS

In this study the full range of HTR fuel design possibilities was considered for the fusion blanket, including standard HTGR and pebble bed fuel which is characterized by a carbon to thorium atom ratio (C/Th) of 150, the heaviest thorium load the present HTGR and pebble bed fuel elements will carry (C/Th $\approx$ 80), and the very heavy thorium load that could be achieved by using coated particles of ThO_2 only with no graphite (C/Th = 12). Lithium-6 was added to suppress the thermal flux which would burn out the bred ^{233}U and to breed tritium. The atom densities used are shown in Table 3. Standard HTGR or pebble bed fuel is of interest because it could be utilized directly in the burner reactor without the addition of any extra moderator or other changes, but the highest production rate of ^{233}U is expected from blankets with very little carbon which will require extra carbon to be added before the fuel is used in the fission reactor. Previous studies of hybrid blankets have shown that the highest fuel production rate can be achieved in a blanket with a hard neutron spectrum (Ref. 4). Previous studies have also shown that a blanket of ^{238}U can achieve a much higher multiplication and fuel production rate than can one made of ^{232}Th (Ref. 5). This is due to fast fission of the ^{238}U by the 14 MeV DT fusion neutrons. The potential blanket performance advantages of ^{238}U can be retained in a ^{233}U-producing thorium blanket design by use of a thin fission plate of ^{238}U between the plasma and thorium breeding blanket. The work of Lee at Lawrence Livermore Laboratory (Ref. 6) indicates that a 5 to 10 cm thick fission plate of metallic uranium such as U_3Si will double the number of neutrons incident upon the breeding zone, will multiply the incident fusion neutron energy by about a factor of five and will breed approximately 0.2 ^{239}Pu atoms per incident fusion neutron. While the use of a fission plate can enhance the fusion blanket performance significantly, it is not

TABLE 3
BLANKET FUEL SPECIFICATIONS

Mixture C/Th Ratio	12	80	150
Atom density, atoms/barn-cm:			
Lithium-6	0 to 3.11×10^{-3}	0 to 7.38×10^{-4}	0 to 3.93×10^{-4}
Carbon	3.67×10^{-2}	5.90×10^{-2}	5.90×10^{-2}
Oxygen	6.22×10^{-3} to 7.78×10^{-3}	1.48×10^{-3} to 1.84×10^{-3}	7.86×10^{-4} to 9.83×10^{-4}
Thorium	3.11×10^{-3}	7.38×10^{-4}	3.93×10^{-4}
Uranium-233[a]	0 to 1.87×10^{-4}	0 to 4.43×10^{-5}	0 to 2.3×10^{-5}

[a] Estimated bred uranium isotopic distribution was included in spectrum calculations.

without potential drawbacks. The high power density may complicate blanket design and the fission product decay heat will require reliable emergency and shutdown heat removal systems. Because of the blanket performance improvement possible, however, the use of a fission plate will probably be required in a commercial reactor design.

3.2. BLANKET CONFIGURATION

For this study the fusion blanket was assumed to consist of a 50 cm thick slab for the HTR fuels described above. The blanket was assumed to surround the plasma regions and to receive a neutron wall loading of 1.5 MW/m^2. The mechanical and thermal design details of the blanket were left unspecified; only the nuclear performance was evaluated in this study. The spatial details were similarly neglected and the results presented are average values for the blanket. Most of the results calculated for the various blanket designs are presented on a per fusion neutron basis or are dimensionless ratios. As a consequence, they are independent of the thickness and wall loading assumed for the blanket design and the power level assumed for the fusion reactor. The length of time required to achieve a given blanket-average enrichment does depend upon the blanket thickness and wall load however. This time will be reduced as blanket thickness is reduced, up to the point where a significant fraction of neutrons begins to pass completely through the blanket, and will be increased if the wall load is reduced:

$$t_e \propto \frac{X}{WL} \quad ,$$

where t_e is the time to achieve enrichment,
X is blanket thickness,
WL is neutron wall loading.

The calculational procedure used in this study was to do a detailed neutron spectrum calculation for each of the blanket designs using the General Atomic Company HTGR spectrum code MICROX to obtain accurate two-group microscopic cross sections for each material present: thorium, ^{233}U, lithium-6, oxygen and carbon. These cross sections were then used in two-group zero dimension static diffusion theory calculations to determine the average nuclear performance of the blanket designs. These calculations and the results obtained from them are further described below.

The blanket performance parameters are expected to show a spatial variation of more than a factor of two, peak to average. The calculation of this spatial dependence requires more detailed computations than were done in this study. The spatial variation in blanket performance will have to be dealt with as part of the blanket design by mixing of the fuel materials in the blanket during irradiation or by use of a different irradiation time for different blanket zones.

The use of a fission plate was accounted for by use of an arbitrary source neutron multiplication factor of 2.0 and fusion energy multiplication factor of 5.8 (Ref. 6). This approach neglects the effect of the lower energy of fission source neutrons (2 MeV vs. 14.1 MeV for DT fusion neutrons), but should be consistent with the other approximations made in this preliminary evaluation study. The effect of the fission plate is to significantly increase the total blanket energy production and to double the tritium and ^{233}U production rates which reduces the time needed to achieve a given enrichment by one-half. The fast neutron fluence accrued while achieving that enrichment, however, is not affected.

3.3. SPECTRUM CALCULATIONS

The enrichment of ^{233}U in the fusion blanket is a function of the fluence and the absorption and capture cross sections of ^{232}Th. ^{233}U is produced from the decay of ^{233}Pa, a capture product of ^{232}Th, and removed by the absorption of a neutron which will lead to either fission or capture to ^{234}U. This may be expressed

$$\frac{dN^U}{dt} = \lambda N^{Pa} - N^U \int_o^\infty \phi(E)\sigma_a^U(E)\,dE \quad , \tag{1}$$

$$\frac{dN^{Pa}}{dt} = N^{th} \int_o^\infty \phi(E)\sigma_c^{th}(E)\,dE - \lambda N^{Pa} - N^{Pa} \int_o^\infty \phi(E)\sigma_a^{Pa}(E)\,dE \quad , \tag{2}$$

where N^U = uranium atom density,

N^{Pa} = protactinium atom density,

N^{Th} = thorium atom density,

λ = decay constant for decay of ^{233}Pa to ^{233}U,

$\phi(E)$ = neutron flux,

σ_a^U = ^{233}U absorbtion cross section,

σ_c^{Th} = ^{232}Th capture cross section,

σ_a^{Pa} = ^{233}Pa absorbtion cross section.

The fluxes and cross sections are strongly energy dependent. The fluxes are determined by the cell composition, which varies with time.

Since no analytic solution exists for the above equations, an approximation must be used to rewrite the integral terms. In general, the approach is to divide the energy spectrum into convenient groups, then to define a group flux, ϕ_g, as

$$\phi_g = \int_{E_{g+1}}^{E_g} \phi(E)dE$$

and group cross section, σ_g, as

$$\sigma_g = \frac{\int_{E_{g+1}}^{E_g} \phi(E)\sigma(E)dE}{\int_{E_{g+1}}^{E_g} \phi(E)dE}$$

Such a definition preserves individual reaction rates by energy group.

The selection of the group structure is necessarily a compromise between the accuracy of a detailed analysis and the reduction in computing time using fewer groups. A common procedure is to use a fine group analysis to precisely calculate few-group sets of cross sections at specific compositions of reactor core regions. These broad-group sets then may be used in time dependent or composition variant studies with good accuracy as long as the composition is near that of the reference case.

The General Atomic's unit cell spectrum code MICROX was used to perform these detailed studies using a 200 energy group data set supplemented by 9000 point resonance data at specified compositions, producing both a nine-group and a two-group (fast and thermal) set of cross sections which could be used for fusion blanket ^{233}U production analysis. The grain structure of the coated particle-graphite fuel is specifically accounted for in the spectrum calculations. These few-group sets of cross sections are exact in the sense that at the composition and geometry used in MICROX, the few-group reaction rates are identical with those of the fine group analysis. It should be noted that due to the strong absorption effects of lithium-6, the cross sections produced are significantly different than the values normally quoted.

A significant difference between the fusion blanket analysis and a conventional HTGR calculation using MICROX was the incorporation of a 14 MeV monoenergetic neutron source, supplemented with a smaller ^{233}U fission spectrum source which grew as the ^{233}U loading increased. Along with the high lithium loading, the effect was production of a hard spectrum, with cross sections based strongly on the high energy characteristics, rather than the thermal characteristics which dominate standard HTGR-type fuel spectrum calculations.

3.4. FUSION BLANKET PERFORMANCE ESTIMATES

To estimate the nuclear performance characteristics of the various blanket design options two-group one-region diffusion theory calculations were done. The macroscopic nuclear parameters were calculated for the different blanket designs using the two-group microscopic cross section data described above. These were then used in the following equations to estimate the blanket performance:

$$D_1B_1^2\phi_1 + \Sigma_{a1}\phi_1 + \Sigma_{12}\phi_1 = S + \nu_1\Sigma_{f1}\phi_1 + \nu_2\Sigma_{f2}\phi_2 \quad (3)$$

$$D_2B_2\phi_2 + \Sigma_{a2}\phi_2 = \Sigma_{12}\phi_1 \quad , \quad (4)$$

where D = diffusion coefficient,
B^2 = buckling,
ϕ = neutron flux,
Σ_a = absorbtion cross section,

Σ_{12} = cross section for scattering from group 1 to 2,

S = neutron source term,

Σ_f = fission cross section,

ν = neutrons released per fission,

Subscript 1 = fast neutrons,

Subscript 2 = thermal neutrons.

With a specified source, S, these equations were solved for the fast and thermal neutron fluxes, ϕ_1 and ϕ_2 which were then used to calculate various reaction rates to estimate the blanket performance. The production rate of ^{233}U is given by

$$\frac{dN^U}{dt} = N^{Th}\left(\phi_1\sigma_{c_1}^{Th} + \phi_2\sigma_{c_2}^{Th}\right) - N^U\left(\phi_1\sigma_{a_1} + \phi_2\sigma_{a_2}^{U}\right) \quad , \tag{5}$$

and the buildup of ^{233}U is described by:

$$N^U(t) = N^{Th}\left(\frac{\phi_1\sigma_{c1}^{Th} + \phi_2\sigma_{c2}^{Th}}{\phi_1\sigma_{a1}^{U} + \phi_2\phi_{a2}^{U}}\right)\left\{1 - \exp\left[-\left(\phi_1\sigma_{a1}^{U} + \phi_2\sigma_{a2}^{U}\right)t\right]\right\} \quad . \tag{6}$$

The above equations neglect the change in thorium density which is only a few percent per year. They also neglect the fact that thorium does not become ^{233}U upon capture of a neutron but becomes ^{233}Pa which subsequently decays with a 27.4 day half life to ^{233}U. Calculations show that although this improves blanket performance slightly by reducing the ^{233}U loss rate, the effect is minor and has been neglected.

To solve these equations for a fusion blanket the source, S, was assumed to be a uniform monoenergetic source of 14 MeV DT fusion neutrons. The buckling terms were adjusted to obtain neutron leakage rates consistent with previous detailed spatial calculations of fusion blankets; the fast leakage loss term is 5%, the net thermal leakage is 0 due to backscatter of fast neutrons from the reflector. The following nuclear performance parameters were calculated for each blanket design:

Thermal to fast flux ratio:

$$\delta = \phi_2/\phi_1 = \Sigma_{12}/\left(\Sigma_{a_2} + D_2 B_2^2\right) \tag{7}$$

Fast flux:

$$\phi_1 = S/\left(\Sigma_{a_1} + D_1 B_1^2 + \Sigma_{12} - \nu_1 \Sigma_{f_1} - \delta \nu_2 \Sigma_{f_2}\right) \tag{8}$$

Equilibrium concentration of ^{233}U in thorium which has been referred to in this report as the "enrichment":

$$\varepsilon_\infty = \frac{\sigma_{c_1}^{Th} + \delta \sigma_{c_2}^{Th}}{\sigma_{a_1}^{U} + \delta \sigma_{a_2}^{U}} \tag{9}$$

E-fold time for approach to equilibrium enrichment:

$$\tau = 1/\left[\phi_1\left(\sigma_{a_1}^{U} + \delta \sigma_{a_2}^{U}\right)\right] \tag{10}$$

Tritium breeding ratio:

$$T = N^{Li} \phi_1 \left(\sigma_{a_1}^{Li} + \delta \sigma_{a_2}^{Li}\right) \Big/ S \tag{11}$$

Uranium breeding ratio:

$$U = \phi_1 \left[N^{Th}\left(\sigma_{c_1}^{Th} + \delta \sigma_{c_2}^{Th}\right) - N^{U}\left(\sigma_{a_1}^{U} + \delta \sigma_{a_2}^{U}\right)\right] \Big/ S \tag{12}$$

Blanket energy multiplication:

$$M = 1 + M_d + \frac{\phi_1}{S}\left[N^{Th}\left(\sigma_{f_1}^{Th} + N^U\ \sigma_{f_1}^U + \delta\sigma_{f_2}^U\right)\right] \times 14.2 \tag{13}$$

where M_d = multiplication of fission plate driver region

Time to reach enrichment ε ($\varepsilon/\varepsilon_\infty < 1$):

$$t_\varepsilon = -\tau\ \ell n(1 - \varepsilon/\varepsilon_\infty) \tag{14}$$

Fast fluence accrued to reach enrichment ε:

$$NVT_\varepsilon = f\phi_1 t_\varepsilon \tag{15}$$

where f = fraction of fast flux above 0.18 MeV

Infinite medium neutron multiplication of fuel material:

$$k_\infty = \left(\frac{\nu_1\sigma_{f_1} + \delta\nu_2\sigma_{f_2}}{\sigma_{a_1} + \delta\sigma_{a_2}}\right) \quad . \tag{16}$$

A full set of microscopic cross sections was determined for each blanket design and the above equations were solved for a full range of lithium-6 and uranium densities for each design. The results are summarized in Figs. 4 through 14. The lithium-6 included in the fusion blanket designs serves two functions. It produces tritium to fuel the DT fusion reactor but its main function is to poison down the thermal flux to prevent the bred ^{233}U from being burned out as quickly as it is formed. Figures 4 and 5 show the uranium and tritium breeding ratios as functions of lithium load, as characterized by the Li/Th ratio, with and without uranium. As more lithium-6 is added to the system in the absense of uranium the tritium breeding ratio rises at the expense of uranium breeding, which falls. With a uranium inventory the increased fission at low lithium load helps compensate and the breeding ratios are relatively independent of lithium load.

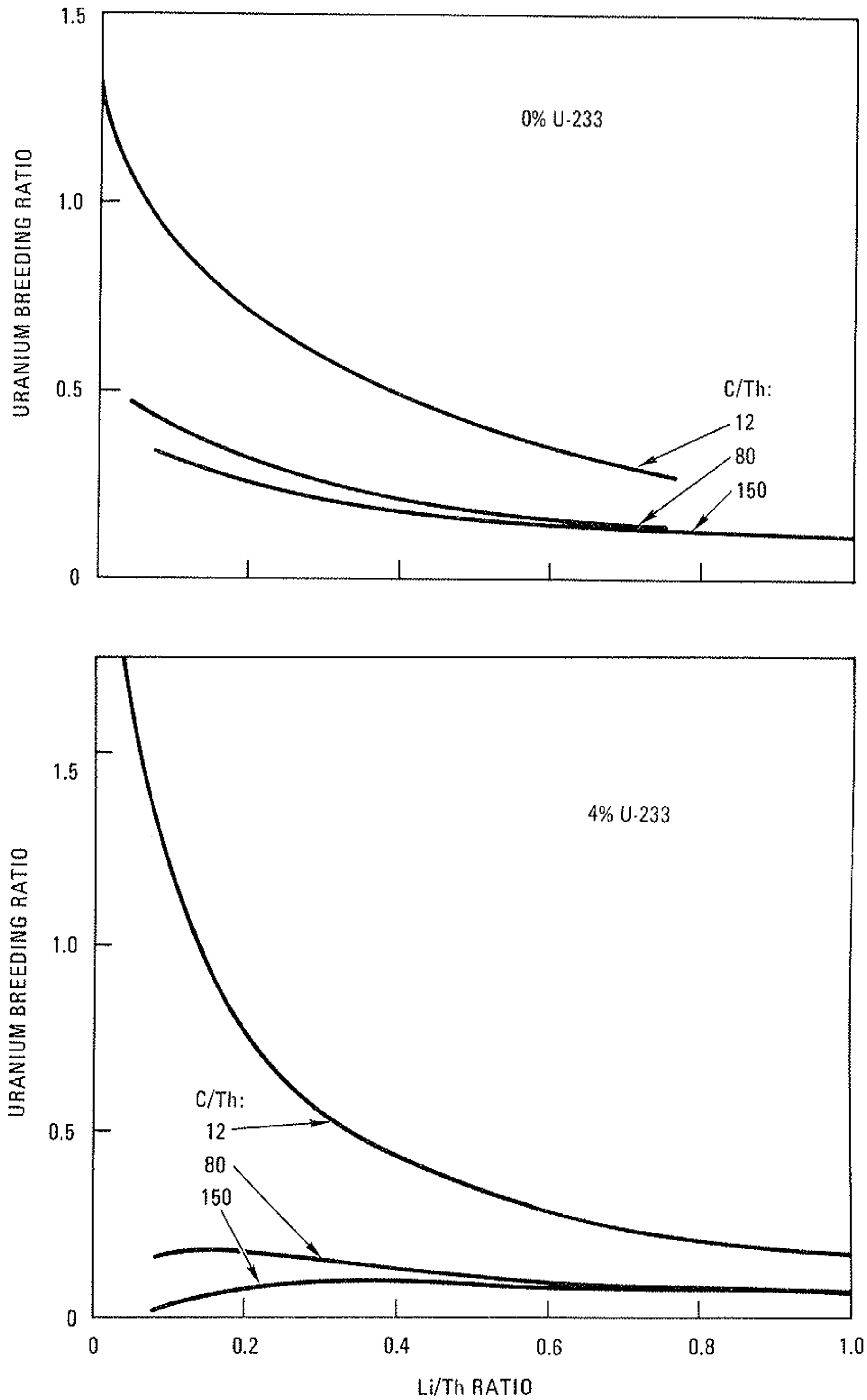

Fig. 4. Uranium breeding ratio

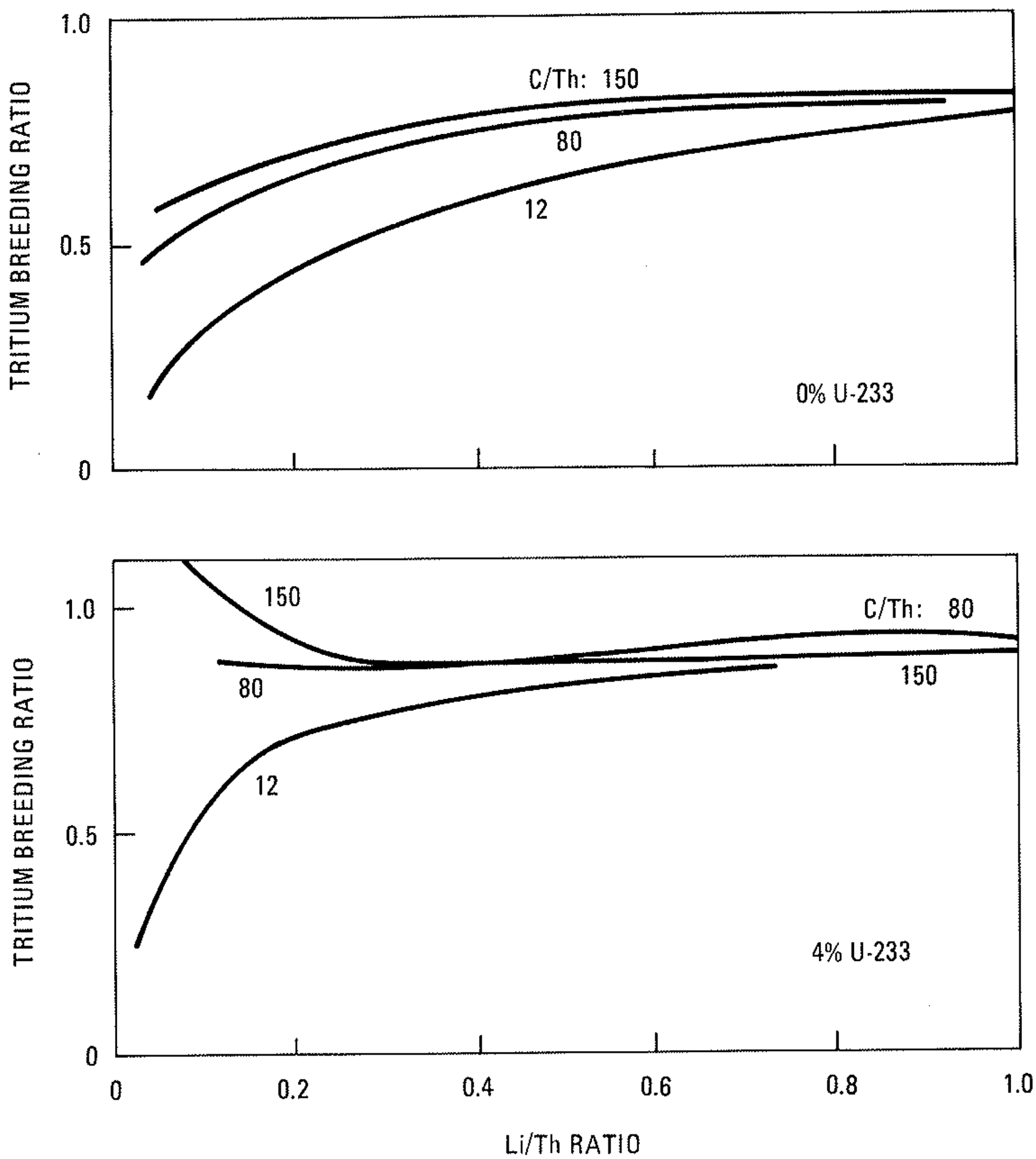

Fig. 5. Tritium breeding ratio

Uranium production in the hard spectrum C/Th = 12 case is a factor of two higher than in the softer spectrum cases.

The concentration of ^{233}U in thorium has been called "enrichment" in this report. In the strictest sense of the word, enrichment is generally taken to denote the concentration of fissile uranium in the total amount of uranium. Using this definition the fissile uranium enrichment in the fusion-bred fuel will be in excess of 80%. The use of an effective enrichment based on the amount of ^{233}U in thorium, however, is a more meaningful measure of the fusion blanket performance. Without reprocessing of the bred fuel, this effective enrichment value is crucial to the utilization of the fuel in fission reactors. The equilibrium enrichment at which uranium production by neutron capture in thorium is balanced by uranium destruction by neutron capture in uranium is shown in Fig. 6. More lithium-6 depresses the thermal flux, allowing a higher enrichment to be reached. The C/Th = 12 case has essentially no thermal flux and hence ε_∞ is independent of lithium-6 load. Note that significant enrichments can be achieved especially in the harder spectrum cases. The equilibrium enrichment value increases somewhat as uranium builds into the blanket which will allow still higher enrichments to be achieved.

Figure 7 shows the time required to reach a 4% blanket average enrichment value with a 1.5 MW/m^2 wall load. Surprisingly the C/Th = 150 case requires the least time despite the much lower uranium breeding ratio shown in Fig. 4. This is due to the fact that the harder spectrum cases have more thorium present. They are producing uranium more rapidly but are actually producing enrichment more slowly. As the lithium-6 load is reduced, the equilibrium enrichment eventually falls below 4%, as was shown in Fig. 6 Below this lithium-6 load 4% enrichment cannot be reached and thus $t_{4\%}$ becomes undefined.

The fast neutron fluence accrued during the time necessary to achieve 4% enrichment, starting at 0%, and that needed to "refresh" the fuel from 3% to 4% are shown in Fig. 8. These enrichment values are used because,

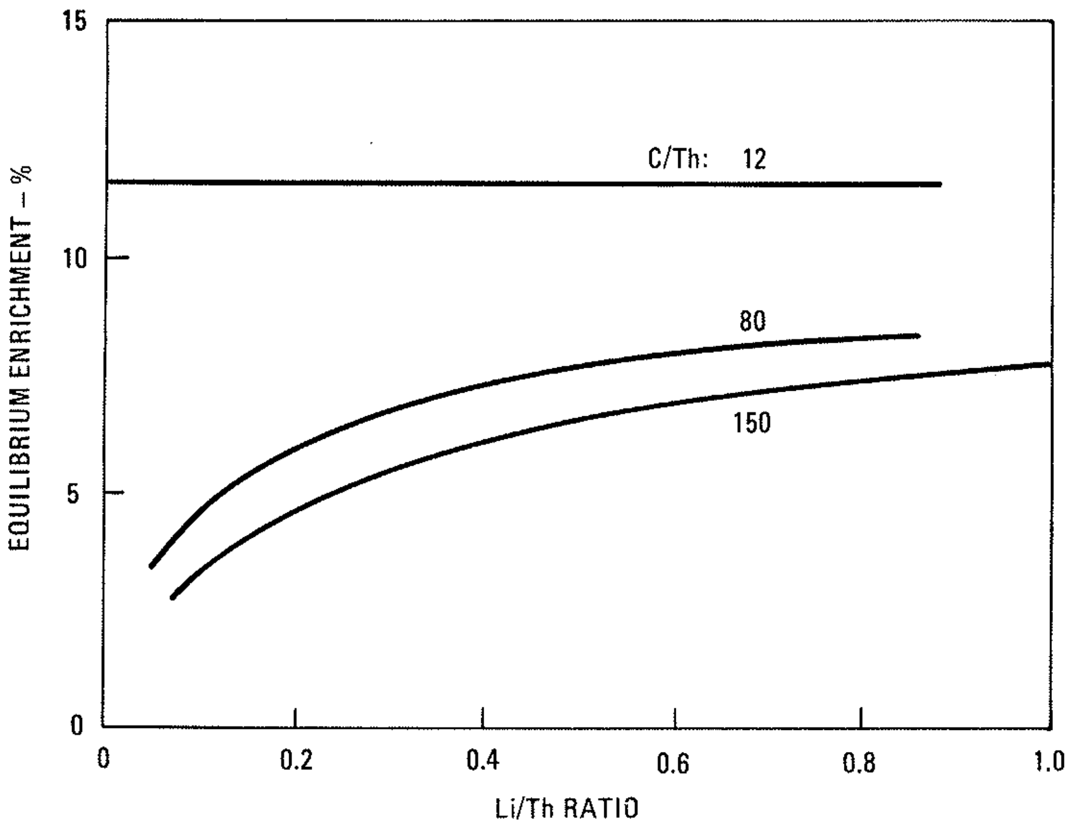

Fig. 6. Equilibrium enrichment

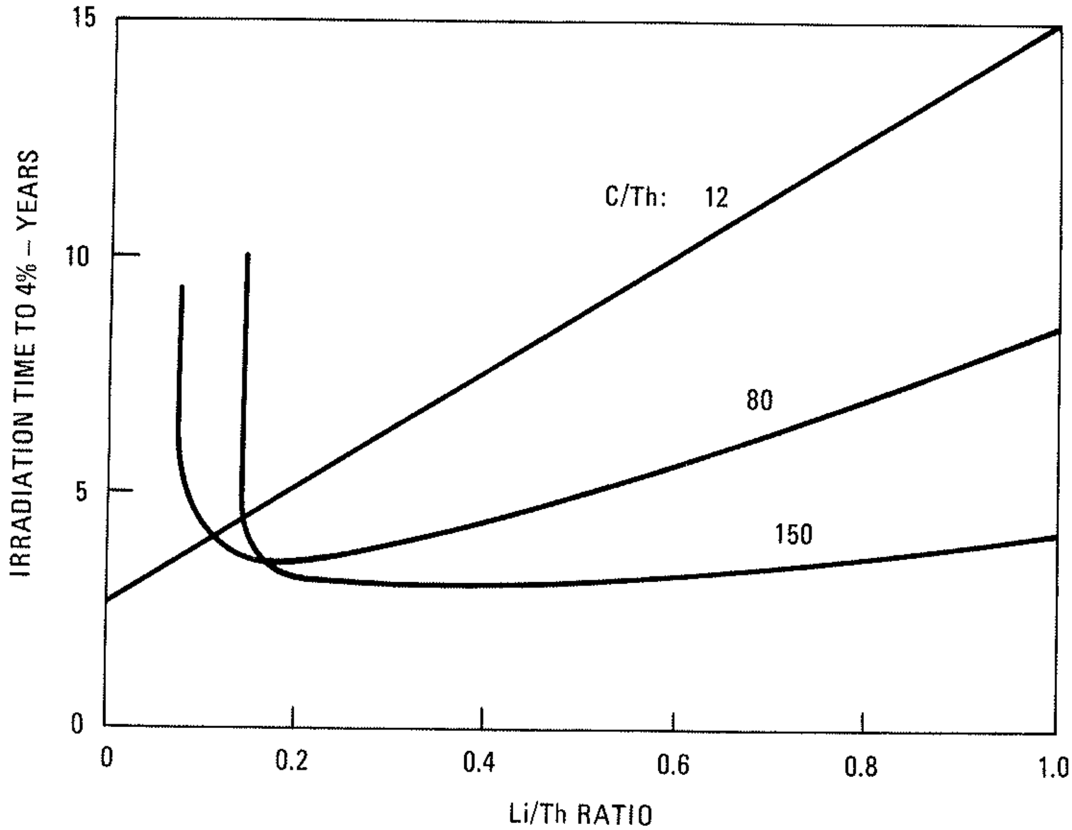

Fig. 7. Irradiation time to 4% enrichment at 1.5 MW/m^2 wall load

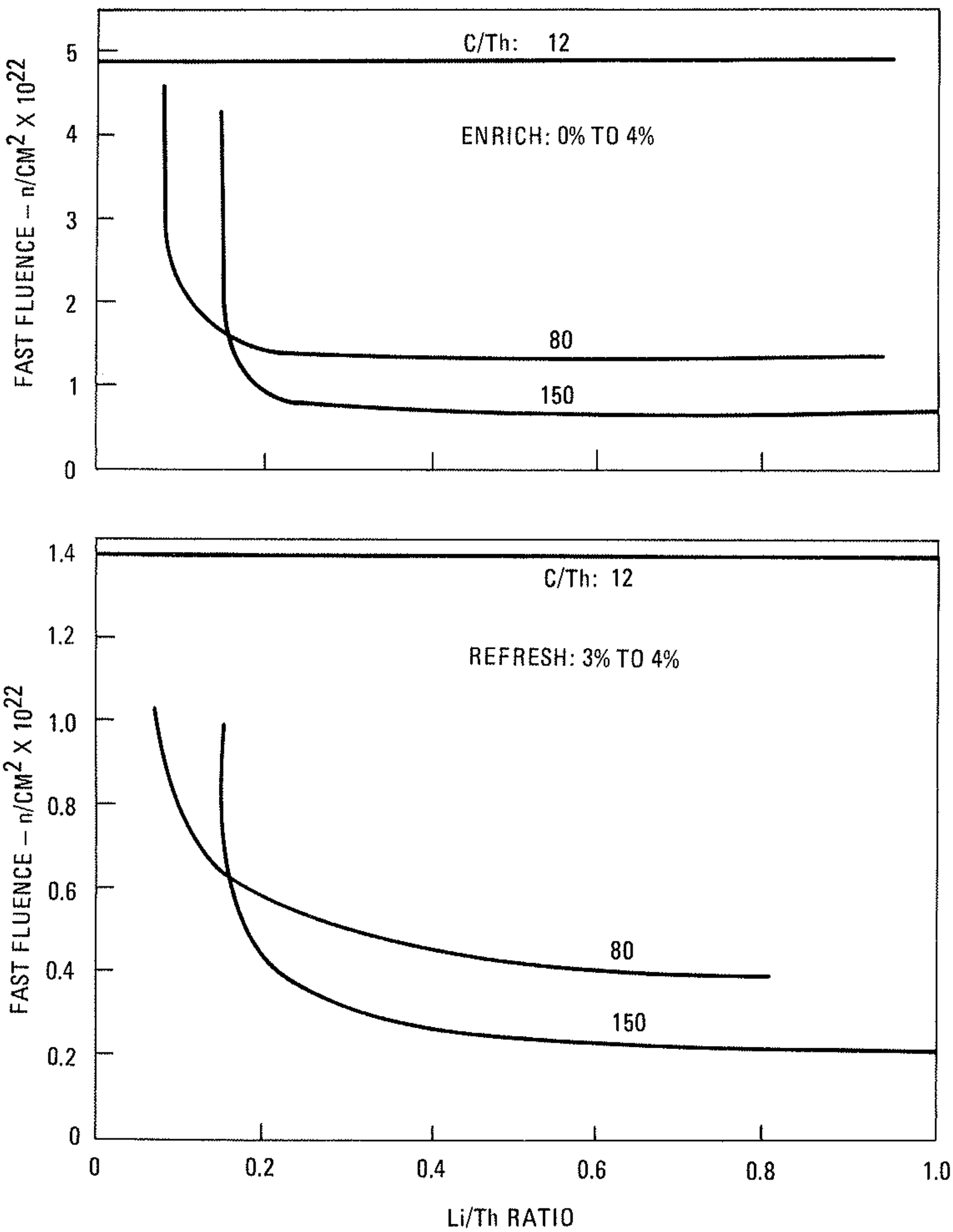

Fig. 8. Neutron fluence (E > 0.18 MeV) to enrich fuel

as will be shown in Section 4, they match the requirements of the HTR burner reactor. Figure 8 shows that the designs with heavier thorium load receive a higher fluence, again due to the simple presence of a greater amount of thorium. The fast fluence is not significantly affected by the lithium load, as long as it is above the load required to allow 4% enrichment to be achieved.

Neither the time nor fluence required to achieve 4% enrichment are significantly affected by uranium as it builds in. Although the breeding ratio falls with increased uranium load, the thermal flux is further depressed by the additional uranium which allows the equilibrium enrichment to rise somewhat and also allows it to be approached more quickly.

From the data presented in Figs. 4 through 8 some conclusions may be drawn concerning the lithium-6 load. The lithium load should be minimized to achieve maximum uranium production. In order to achieve adequate enrichments, however, a ^{6}Li/Th ratio of about 0.2 is required. The blankets with Li/Th > 0.2 all show an infinite medium multiplication factor, K_{∞}, substantially less than one, so criticality of the blanket is not a serious concern. At this 0.2 lithium-6 value, the tritium breeding ratio is only about 0.7. Regardless of the lithium-6 load, a tritium breeding ratio of 1.0 cannot be achieved by the HTR fuel blanket alone. There are several possible ways to breed the additional tritium needed to sustain the fusion reaction. These include breeding it in the associated fission reactors, breeding it in a separate fusion reactor and using a fission plate to improve the blanket performance. Breeding sufficient tritium in fission reactors without significantly disrupting the neutron economy is difficult due to the small number of neutrons per unit power compared to fusion reactors. Breeding the tritium in a fusion reactor would be possible but would require a separate reactor as large or larger than the fissile fuel breeder. The addition of a fission plate, however, would double the tritium breeding ratio and allow a ratio of 1 to be easily reached. It thus appears that adequate tritium production could be most easily achieved by use of a fission plate in the fuel breeder and for this reason it is expected that a commercial symbiotic fusion power system would incorporate a fission plate blanket design.

Selecting the $^{6}Li/Th = 0.2$ point as a reference point, the nuclear performance of the blanket designs is plotted as a function of carbon/thorium ratio in Figs. 9 through 12. The uranium and tritium breeding ratios with 0 and 4% uranium enrichment are shown in Figs. 9 and 10 The harder the neutron spectrum (lower C/Th), the more uranium is bred. Figure 11 shows that the hard spectrum designs are also capable of achieving higher equilibrium enrichment values. Figure 12 shows that the softer spectrum designs, although producing less uranium, can achieve a 4% enrichment sooner and with less fast neutron fluence to the fuel materials than can the low C/Th designs.

From the results presented above it may be seen that the nuclear performance of the blanket cannot be easily optimized. Breeding time considerations and fast fluence limits push the design toward the softer spectrum, high C/Th ratio fuels. The desire to maximize the uranium production rate, however, pushes the design toward the low C/Th, hard spectrum fuels. The assumption of a constant blanket thickness in these calculations also complicates matters and the thickness is a parameter that must be varied in the blanket optimization. Spectral shaping may be possible to help in this optimization. The final design choice will be influenced by economics, materials capabilities, and thermal and mechanical design considerations. It appears, however, that acceptable designs can be achieved between C/Th = 12 and C/Th = 80. The C/Th = 80 design is readily achievable in existing HTGR and pebble bed fuel technologies. Thus it is unlikely that any new fuel technologies will have to be developed for the no-reprocessing refresh cycle concept to work. Achievement of still higher thorium loads (C/Th < 80), however, would allow improved blanket performance and a higher uranium production rate.

The blanket performance of C/Th = 80 fuel as ^{233}U builds in is shown in Fig. 13. The characteristics of two blanket designs using this fuel are shown in Table 4. These correspond to the C/Th = 80 design with $^{6}Li/Th = 0.2$ both with and without a ^{238}U fission plate. The performance of a C/Th = 12 blanket is similarly shown in Fig. 14 and Table 4. The C/Th = 80 case corresponds to use of existing HTR fuel technology while the C/Th = 12 case

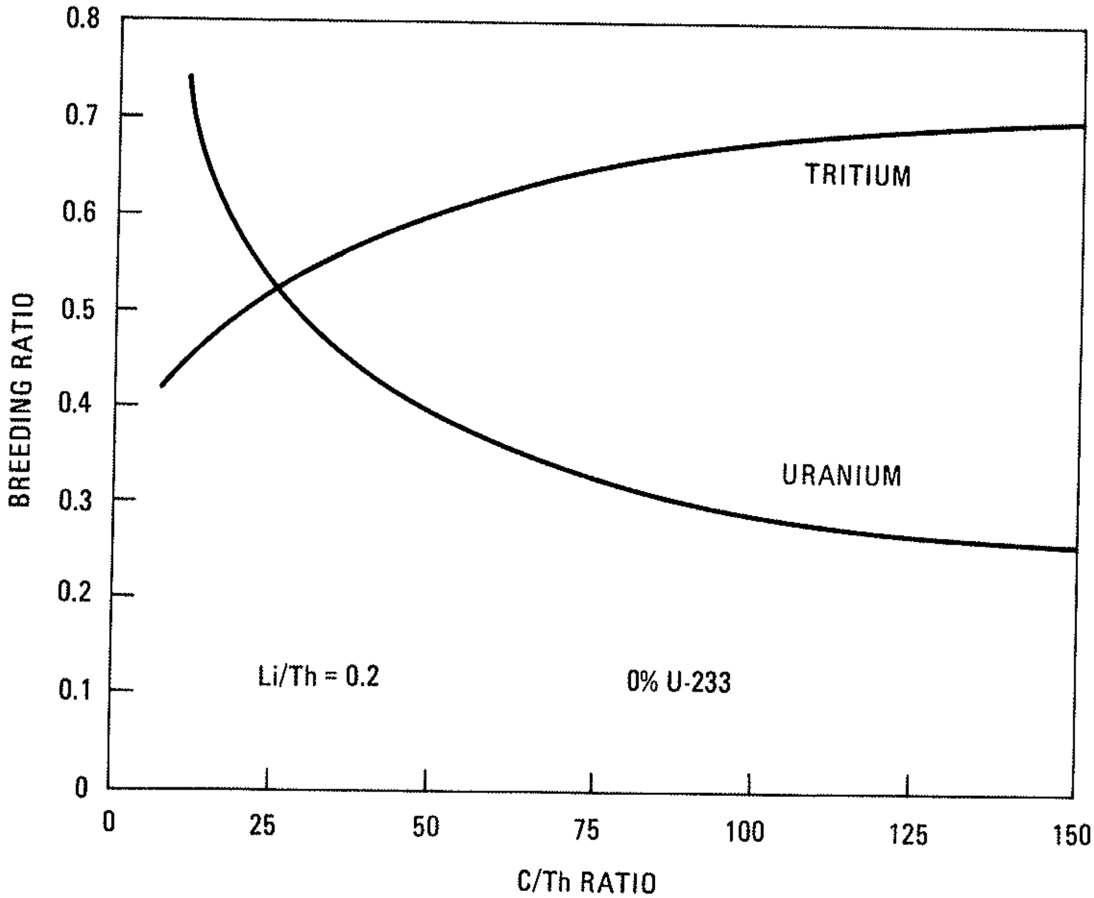

Fig. 9. Uranium and tritium breeding ratios - 0% ^{233}U

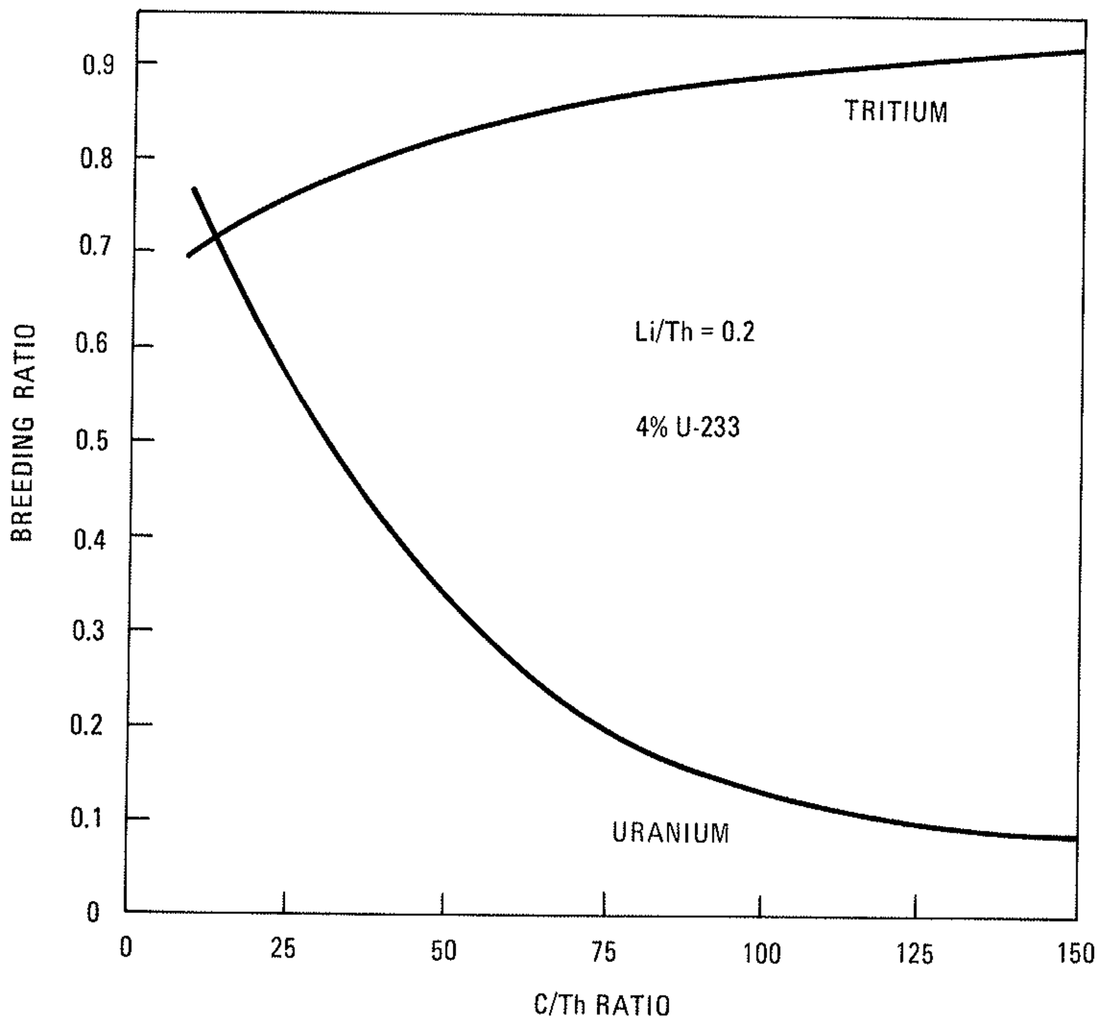

Fig. 10. Uranium and tritium breeding ratios - 4% ^{233}U

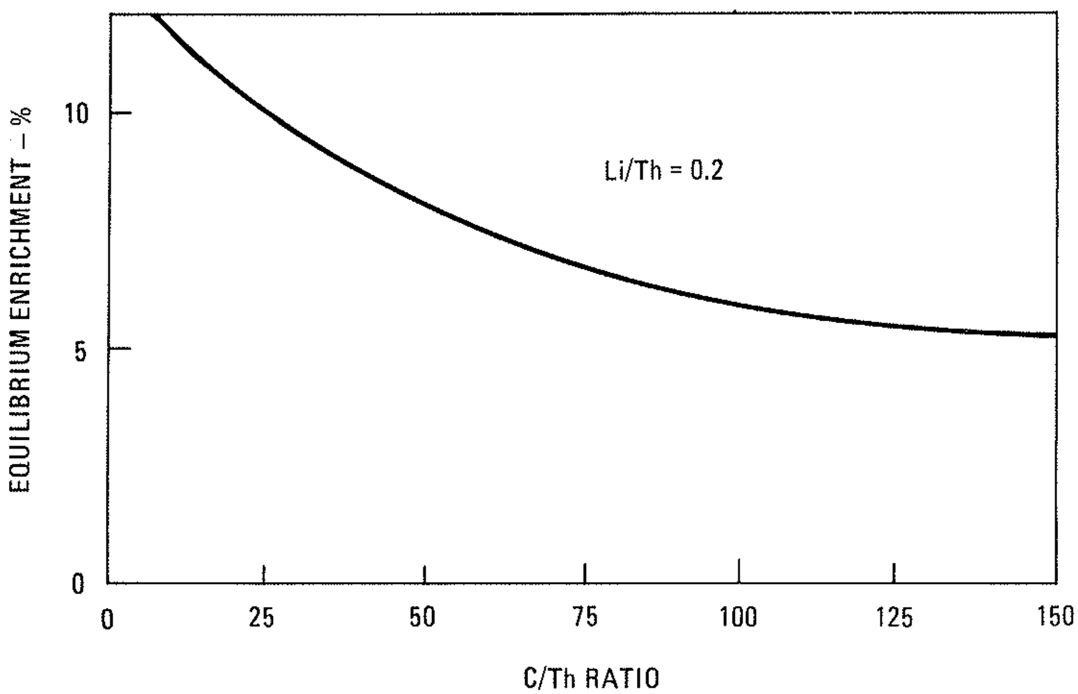

Fig. 11. Equilibrium enrichment

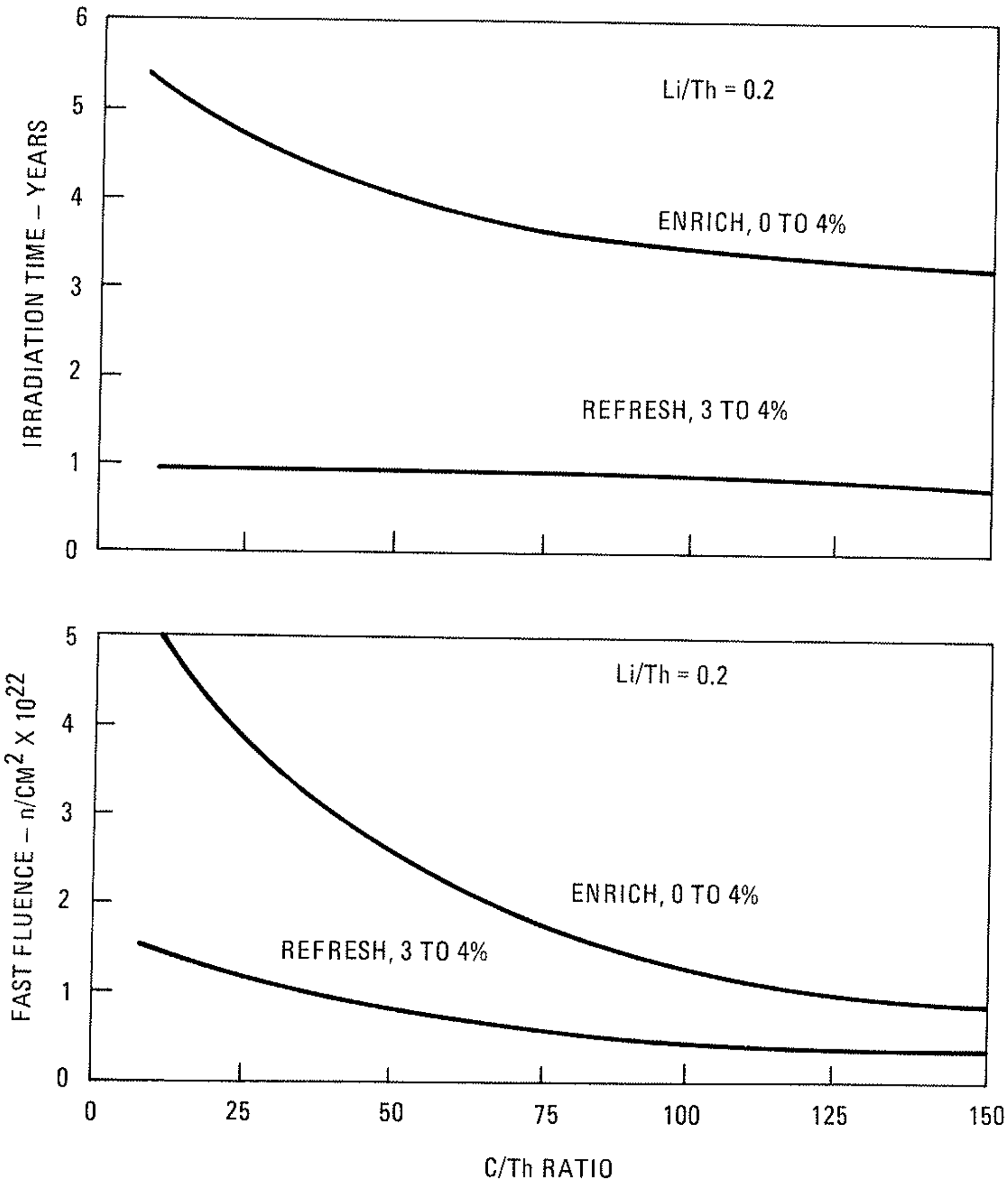

Fig. 12. Irradiation time to enrich fuel

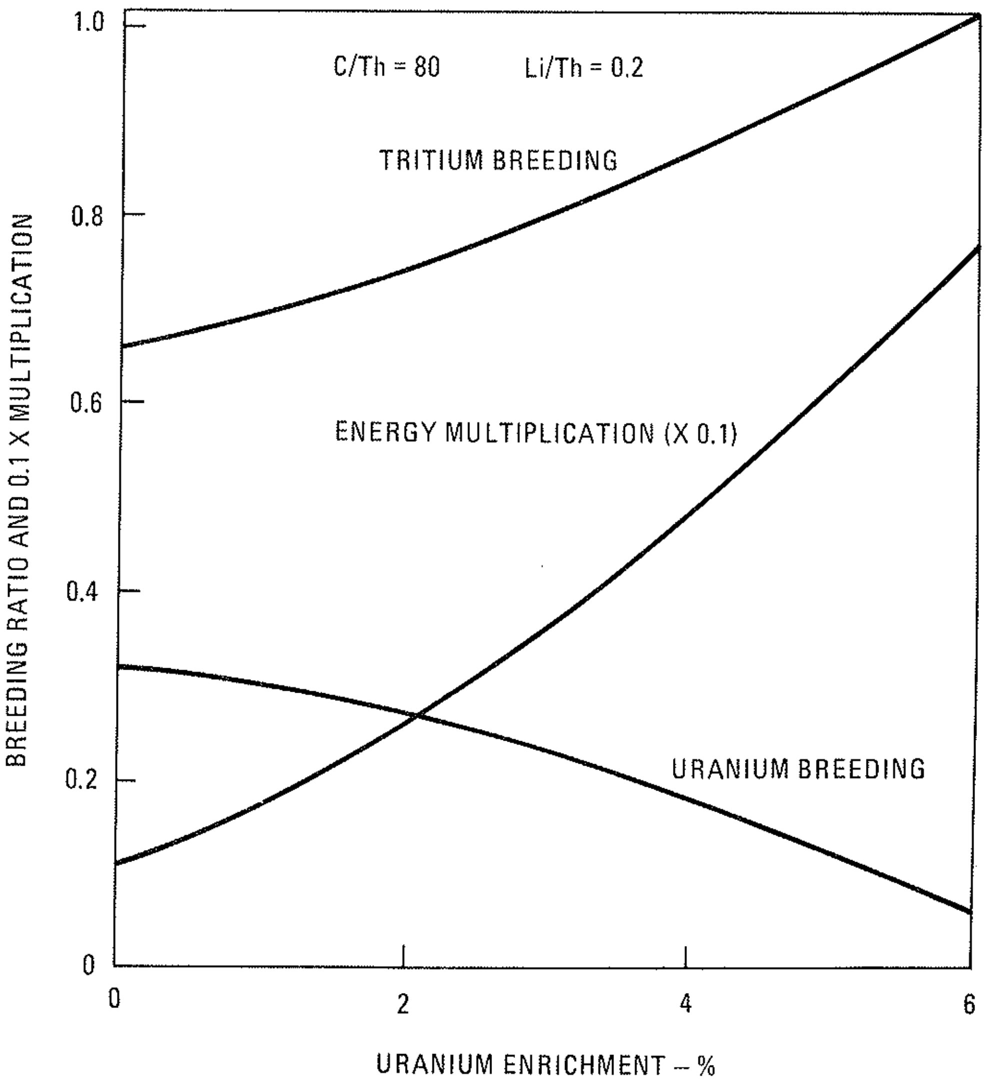

Fig. 13. C/Th = 80 blanket performance

TABLE 4
BLANKET PERFORMANCE CHARACTERISTICS

Parameter	C/Th = 80, Li/Th = 0.20				C/Th = 12, Li/Th = 0.15			
	Without Fission Plate		With Fission Plate		Without Fission Plate		With Fission Plate	
	0% Uranium	4% Uranium	0% Uranium	4% Uranium	0% Uranium	4% Uranium	0% Uranium	4% Uranium
Breeding ratios (atoms/fusion)								
Uranium	0.32	0.18	0.64	0.36	0.78	0.94	1.56	1.88
Tritium	0.66	0.87	1.32	1.74	0.39	0.65	0.78	1.30
Blanket energy multiplication	1.1	4.7	7.0	14.2	2.2	9.0	9.2	22.8
Equilibrium enrichment (%)	6.5		6.5		11.5		11.5	
Years to achieve 4% enrichment	3.7		1.8		4.5		2.25	
Fast fluence at 4% enrichment (n/cm^2)	1.4×10^{22}		1.4×10^{22}		4.8×10^{22}		4.8×10^{22}	
Fast fluence, 3% to 4% enrichment (n/cm^2)	0.7×10^{22}		0.7×10^{22}		1.4×10^{22}		1.4×10^{22}	

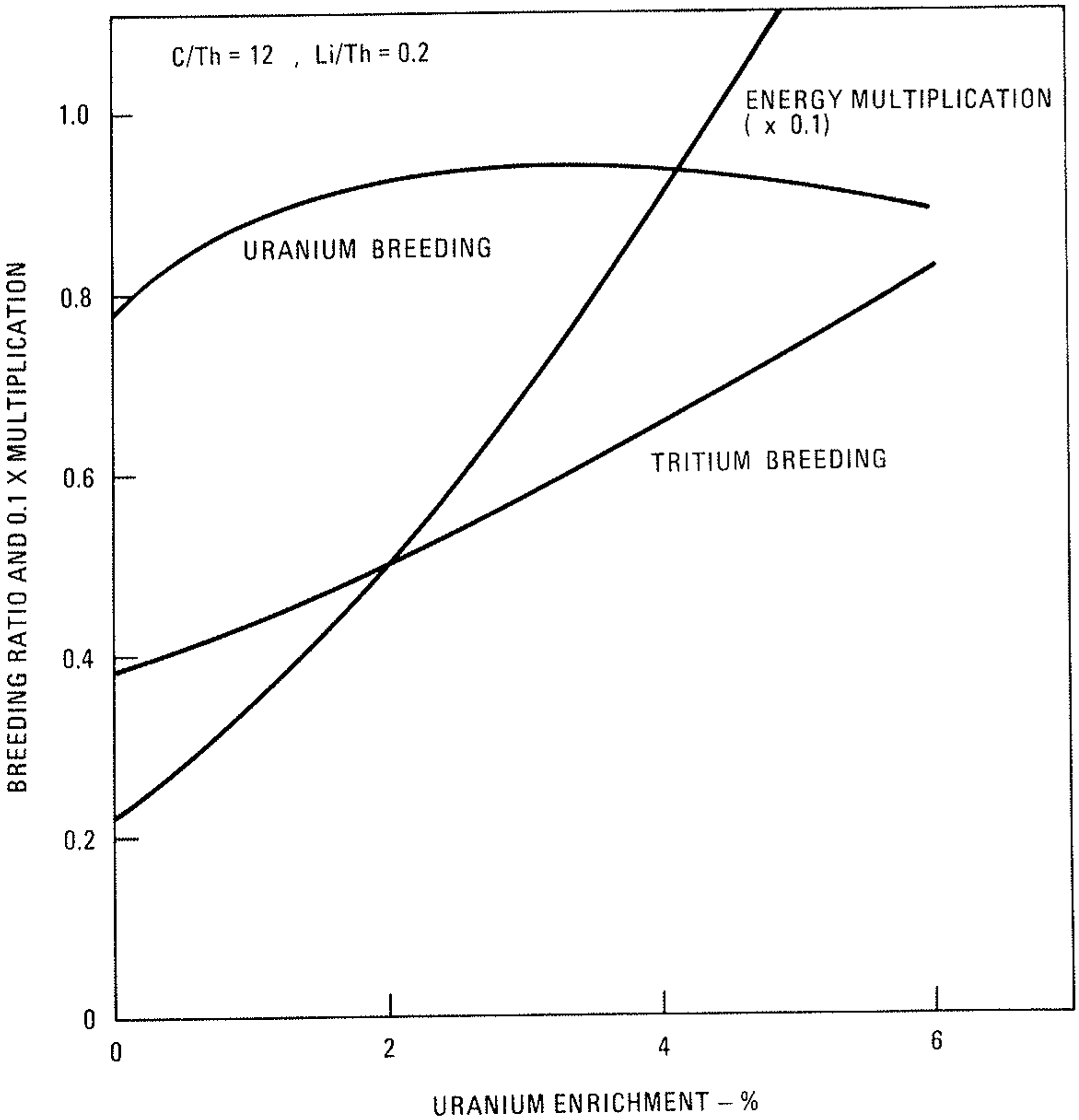

Fig. 14. C/Th = 12 blanket performance

may be thought of as the performance potentially available through use of the coated particles only. These designs will be used in Section 5 to estimate the performance of four symbiotic fusion-fission power systems using the no-reprocessing Refresh fuel cycle .

The model used to estimate the fusion blanket performance characteristics has proven to be a very useful tool. It should be cautioned, however, that it is quite simple. Spatial effects, including neutron spectral shift through the blanket have been homogenized over the entire blanket. These spatial effects and the effect of anisotropy in the neutron distribution, which require space dependent transport theory to calculate correctly, will influence the blanket nuclear performance to some extent. The thermal and mechanical design aspects of the blanket have been neglected in this evaluation. It appears that these considerations will not pose difficult fuel design problems due to the simple, rugged and refractory nature of HTGR and pebble bed fuel. Nevertheless, the thermal and mechanical design requirements will surely influence the nuclear performance of the blanket. The fuel pellet or ball manipulation requirements to achieve the uniform blanket exposure assumed in this model may be difficult to achieve in practice. Despite these limitations, the blanket model should give a reasonably accurate estimate of the performance capability that is possible with an HTR fuel based blanket design. The model has been quite useful in determining the behavior of the nuclear performance parameters as the blanket design specifications are varied.

3.5. CONCLUSIONS

On the basis of the fusion blanket performance estimates described above several conclusions may be drawn. Most importantly, the concept of breeding ^{233}U in a fusion reactor blanket based on HTR technology appears feasible. Concentrations of ^{233}U in thorium ("enrichments") in excess of 4% can be achieved within the expected materials fluence limitations. Blankets based upon present day fuel technology (C/Th > 80) show acceptable uranium breeding performance. This performance could be enhanced by employing high thorium loads (low C/Th ratios) that would require further fuel development. To attain still higher uranium production rates and to reach a tritium breeding ratio of one will require the use of a fission plate to enhance neutron production in the blanket.

4. HTR FUEL CYCLE CALCULATIONS

4.1. FUEL CYCLE PARAMETERS

The HTR is a very flexible reactor in terms of adaptability in its fuel use since it can be fueled with various types and amounts of the fissile fuels, ^{233}U, ^{235}U, or Pu, and fertile fuels, ^{238}U or Th or any combination of them. As the main energy producer of a fusion/fission symbiosis, the HTR has to be optimized to (1) achieve a high fuel utilization with the unreprocessed ^{233}U/Th fissile feed material produced in the fusion blanket, (2) operate on low ^{233}U feed enrichments since it is difficult to breed high fissile enrichments in a fusion blanket without reprocessing.

In order to approach these goals one can vary the following HTR fuel design parameters: the C/Th ratio, the power density and the fuel residence time. The effect of varying those parameters will be described below.

4.1.1. Carbon-to-Thorium Ratio (C/Th)

The ratio of the atomic densities of carbon and thorium in the reactor determines the neutron spectra or the ratio of fast-to-thermal neutron flux. Since the capture of neutrons in thorium and the production of ^{233}U occurs mainly to fast neutrons, while the absorption in ^{233}U occurs mainly to thermal neutrons, this flux ratio and C/Th ratio influence the conversion ratio (ratio of production to destruction of fissile material) and the net fissile ^{233}U requirements. Increasing the thorium loadings, that is, decreasing the C/Th ratio, leads to high conversion ratios and small net fissile fuel consumption. This also results in rather high fissile inventories and such systems may be limited due to the available fuel volume.

4.1.2. Power Density

The power density determines the magnitude of the neutron flux, the active core volume for a given power level, and the energy production or burnup per fuel element. Small power densities lead to small fission product poisoning from the saturating fission products (principally xenon) and therefore, to a high conversion ratio and good fuel utilization. However, these small power density designs have the disadvantage of requiring large cores which cause increased capital costs and, due to the large thorium load, also require an increased fissile inventory implying large fuel inventory costs. Furthermore since part of the handling costs (fabrication, refabrication, shipping, storage and waste disposal) are fuel element related, a small power density or small power per fuel element leads to high energy costs.

4.1.3. Fuel Residence Time

The fuel residence time indicates how long a fuel element remains in the reactor before it is discharged to be retired or to be re-enriched or refreshed in the fusion blanket. The fuel residence time also determines the fraction of the core to be reloaded per year. If the fuel element is not refreshed, smaller residence times mean that larger fractions of the total fuel are discharged and retired. On the other hand long residence times increase the fission product poisoning and reduce the ^{233}U production and the conversion ratio. Furthermore, long residence times lead to large energy production per fuel element and therefore reduced fuel handling costs.

4.2. FUEL CYCLE ESTIMATES

These three important design parameters, C/Th ratio, power density and residence time, can be varied and chosen independently. Combinations of design parameters which most affect the feed enrichment and the annual makeup requirements will now be looked at. In order to simplify the approach, the variation of the C/Th ratio and the power density will be estimated by a single parameter, the conversion ratio (CR). Since the CR depends mainly on the total thorium load which is inversely proportional to the product of power density times C/Th ratio, it is not surprising that the calculated CR for different C/Th ratios and power densities are almost on one line, as shown in Fig. 15. This allows one to represent the necessary feed enrichment and annual uranium requirements as a function of the conversion ratio. This in turn gives an estimate about the feasibility of the project, since the achievable fissile enrichment in the fusion blanket is limited. It also shows how many HTRs could be fed by one fusion reactor.

In order to find some insight and understanding of the relationships between consumption, enrichment, inventory, etc., empirical formulas are developed from a large data base of similar HTR evaluations previously performed. The past experience is, however, not directly applicable since no data are available for use of unreprocessed ^{233}U feed material in HTRs; furthermore, the retiring of substantial quantities of ^{233}U in a no-reprocessing fusion/fission concept leads to selection of only moderate conversion ratios as will be shown. All of the ^{233}U fed HTR systems considered up to now made use of reprocessing and were, therefore, high conversion systems. The empirical relationships derived from that past experience have to be taken as indicative of trends, which means that they are qualitatively correct but may not be quantitatively accurate to better than about ±20%, depending on the particular relationship. These formulas are consequently used to calculate and illustrate graphically the required uranium enrichment as well as the uranium requirements for different conversion ratios and different numbers of fuel life cycles (fusion enrichment or Refresh cycles), but the exact numerical values presented should be used with caution.

4.2.1. Thorium Load

Studies of high conversion ^{235}U fed HTRs (Ref. 7) showed that the conversion ratio can well be parameterized through the total thorium reactor inventory and the irradiation period, the later taking into

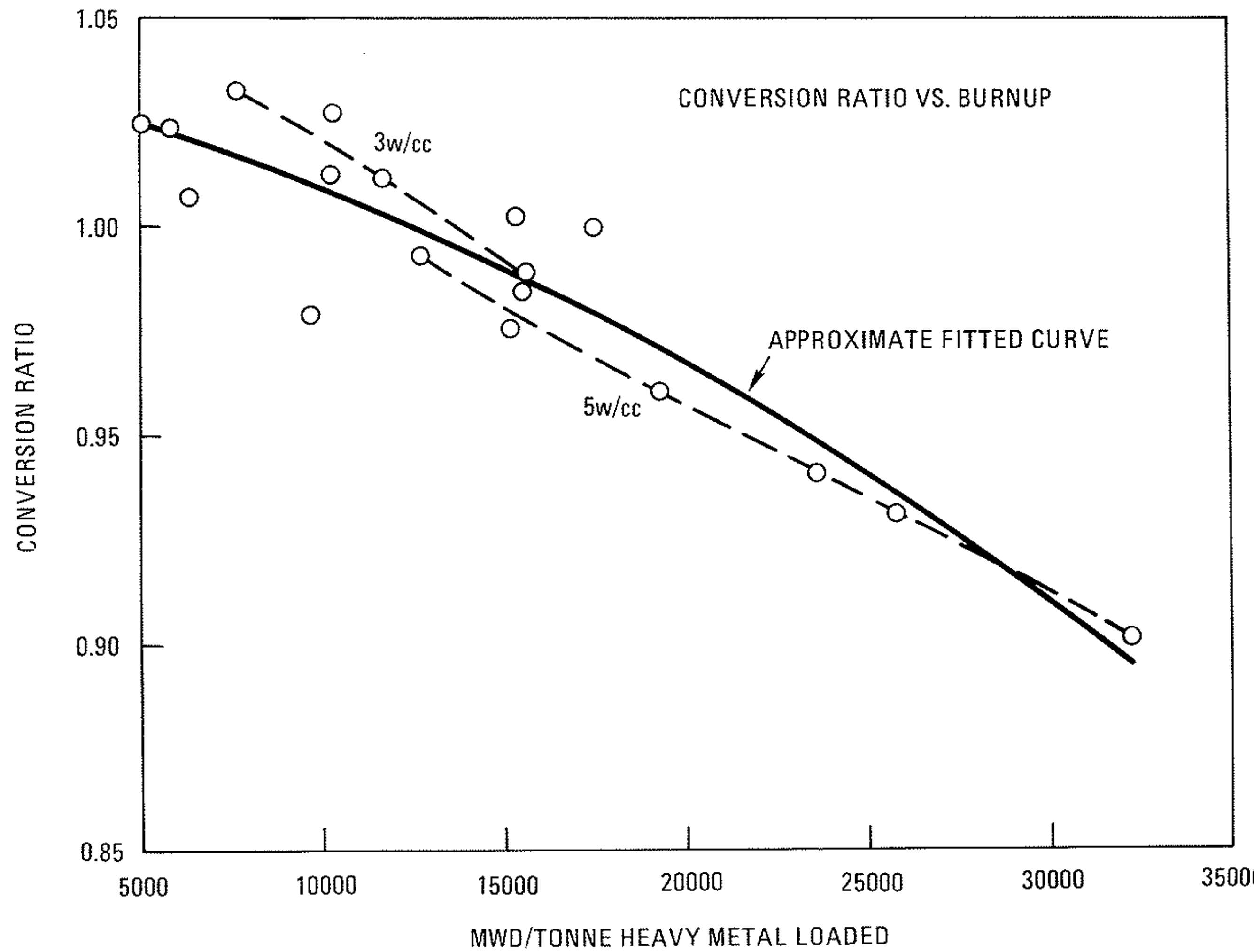

Fig. 15. HTR conversion ratio versus burnup

account the fission product poisoning. The thorium inventory for a 1000 MW(e) reactor is

$$\mathrm{Th} = 122{,}000\ (\mathrm{CR} + 0.02 \cdot \tau)^{3.94}\ \mathrm{kg/MW_e} \quad , \tag{17}$$

where τ stands for the total effective irradiation period of the fuel (HTR and fusion blanket) expressed in years at 7 w/cc fission power density. Figure 16 shows this relation graphically and it points out that with an increased number of fuel life cycles (fusion re-enrichments) the conversion ratio drops due to fission product buildup. High conversion ratios can be achieved through higher thorium load which in turn is limited by the available fuel volume.

4.2.2. Uranium Load

From the same HTR studies mentioned above the relation between uranium and thorium loading is established. The critical fissile uranium inventory can be expressed as a function of the thorium load. The fissile uranium reactor inventory for a 1000 MW(e) HTR is about

$$U_{\mathrm{fissile}} = 400 + \mathrm{Th}/35\ \mathrm{kg/GW_e} \quad , \tag{18}$$

where Th is the thorium inventory for a 1000 MW(e) HTR defined above. Combining the expression for the uranium and thorium inventory allows one to plot the fissile uranium inventory as a function of the conversion ratio and the number of fuel life cycles (Fig. 17). In a graded fuel cycle where only a part of the fuel is discharged each reload, the end of cycle (EOC) discharge uranium load is reduced by half of that depleted each year so that

$$U_{\mathrm{discharge}} = \frac{1}{R}\left[400 + \mathrm{Th}/35 - \frac{866}{2}(1 - \mathrm{CR})\right] \mathrm{kg/GW_e\text{-}yr} \quad , \tag{19}$$

where R = fuel residence time in years.

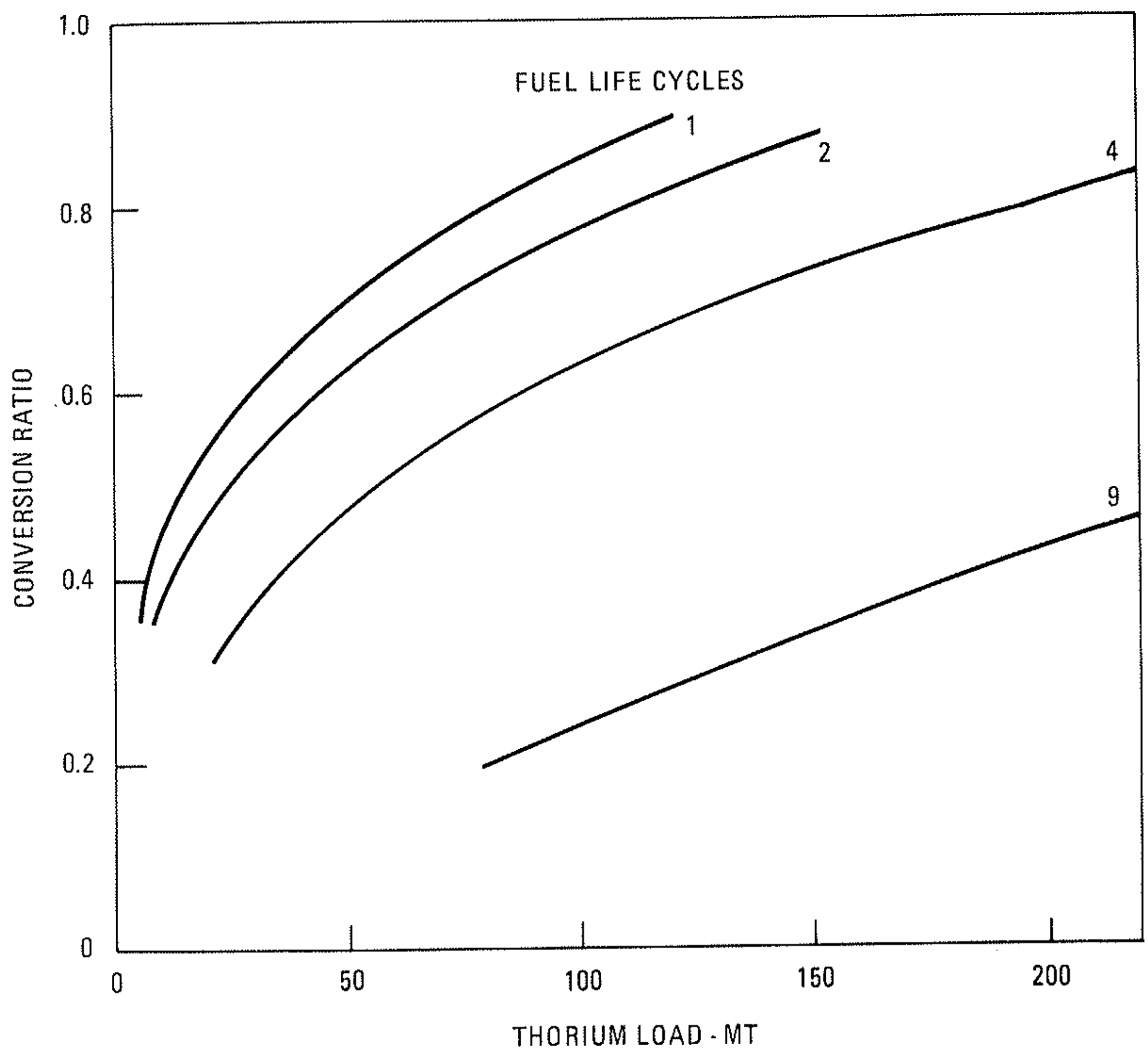

Fig. 16. HTR conversion ratio - effect of number of enrichment-depletion cycles

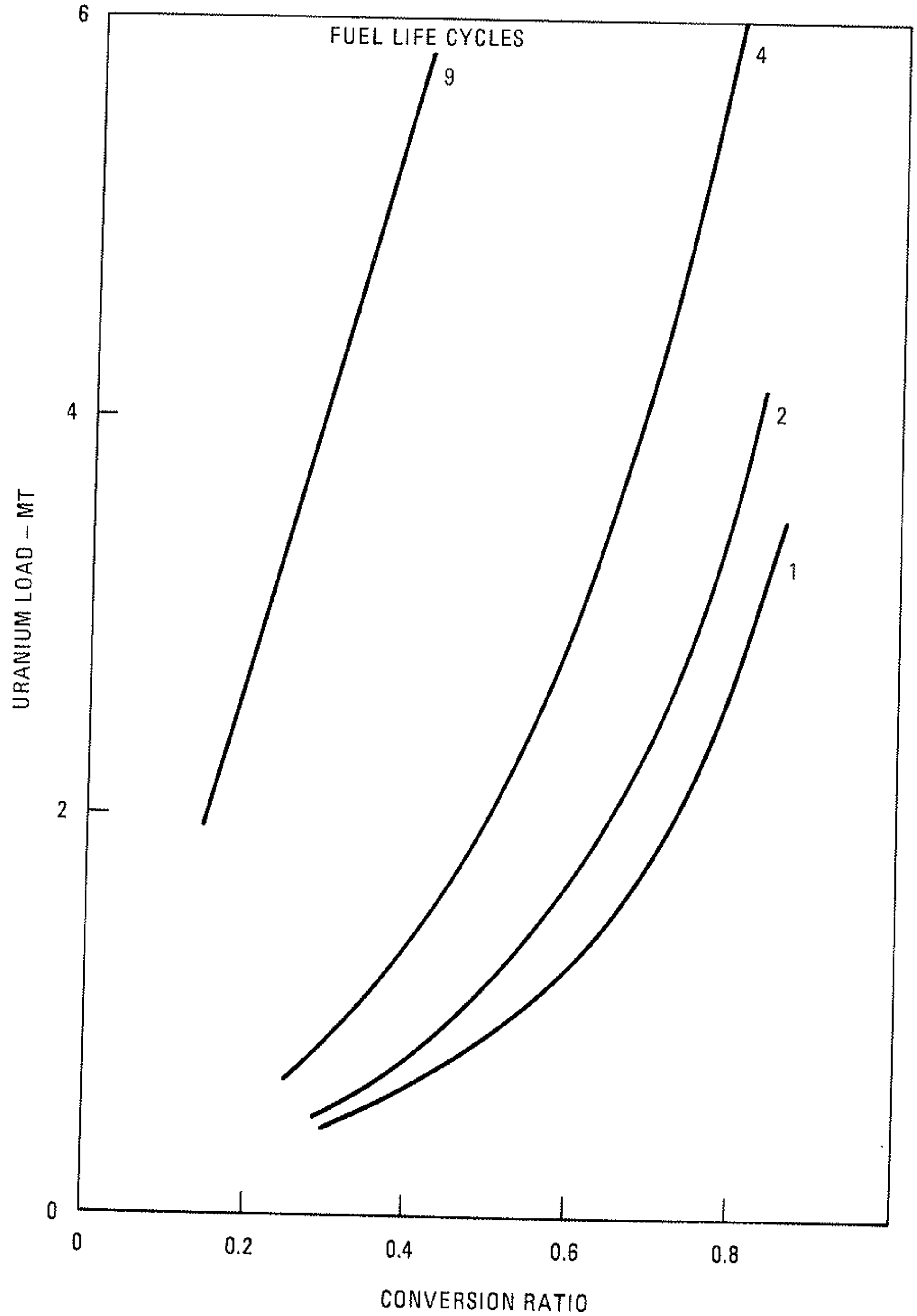

Fig. 17. Uranium inventory

4.2.3. Annual Uranium Requirements

The annual uranium requirements consists of two parts. First, new fuel has to be added to compensate for the fuel being retired every year. The fraction of the fissile inventory retired is inversely proportional to the number of life cycles M, times the HTR fuel residence time R. Second, since more uranium is burned than produced with a conversion ratio smaller than unity, a net fissile uranium makeup of 866 (1 - CR) kg/GW_e is required assuming ^{233}U fuel, 80% capacity factor and 40% thermal efficiency, according to Ref. 8. Therefore, the total annual requirements are

$$U_{consumed} = 866(1 - CR) \quad \text{burned}$$
$$+ \frac{1}{RM}\left[400 + (122{,}000/35)(CR + 0.02 \cdot \tau)^{3.94} - \frac{866}{2}(1 - CR)\right] \quad \text{retired} \qquad (20)$$

The uranium requirements are shown graphically in Fig. 18. The figure illustrates that minimum uranium requirements occur at conversion ratios of about 0.6. Higher conversion ratios reduce the fissile fuel burned but, due to the larger inventories, increase the amount of fissile fuel retired. Re-enriching fuel in the fusion reactor does reduce the uranium requirements. More than three fuel life cycles are, however, not desirable as Fig. 19 indicates since the accumulated fission products reduce the conversion ratio and actually increase the uranium consumption. Figure 19 shows the minimum possible uranium consumption by a fission reactor operating on the no-reprocessing refresh cycle and was obtained by differentiating the above uranium consumption equation, setting the derivative to zero and solving for the optimum.

4.2.4. Uranium Enrichment ($U_{fission}/Th$)

With the formulas so far established one can also calculate the necessary uranium enrichment for the feed material for the HTR. Assuming a linear burnup and on-line refueling then the uranium enrichment is about

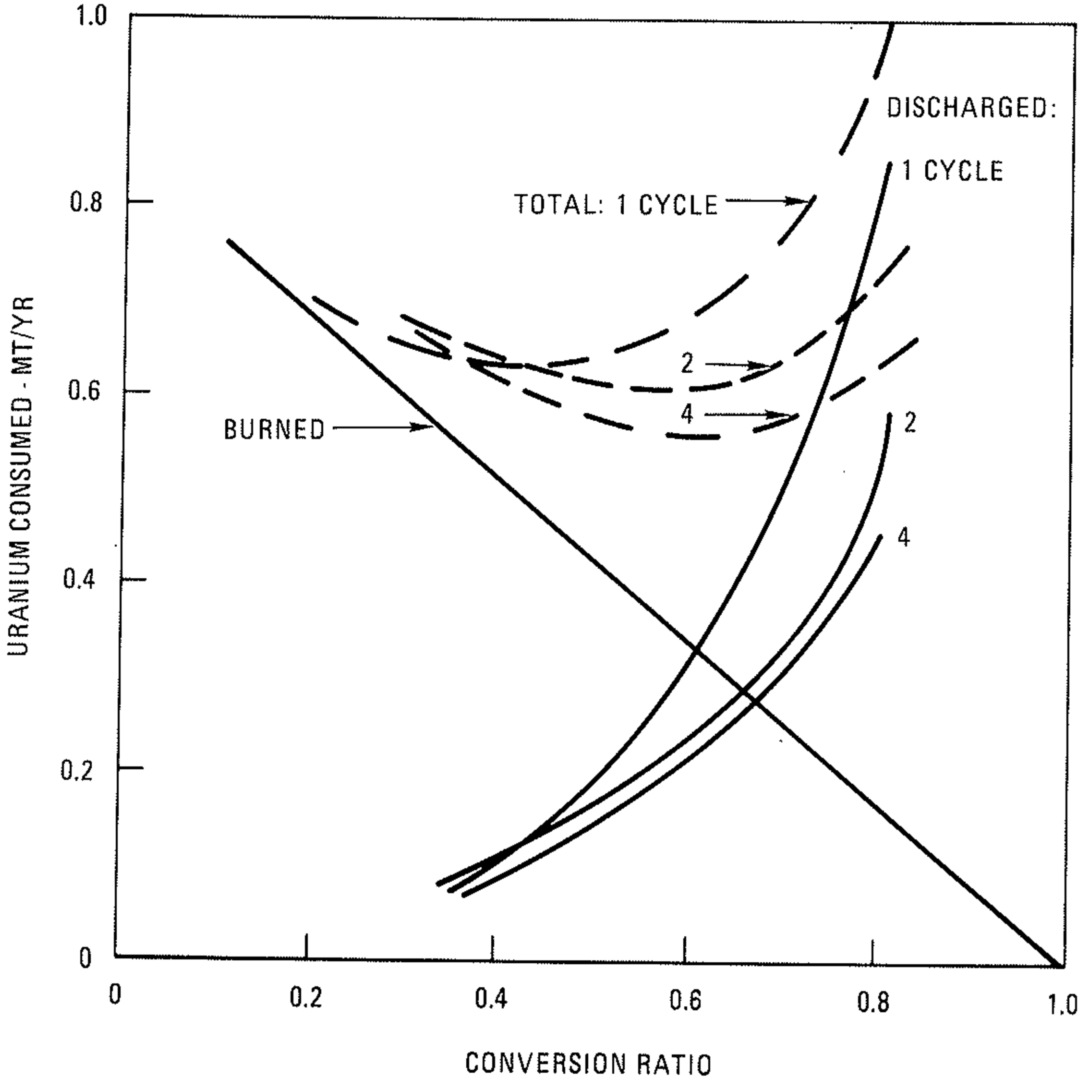

Fig. 18. Uranium consumption

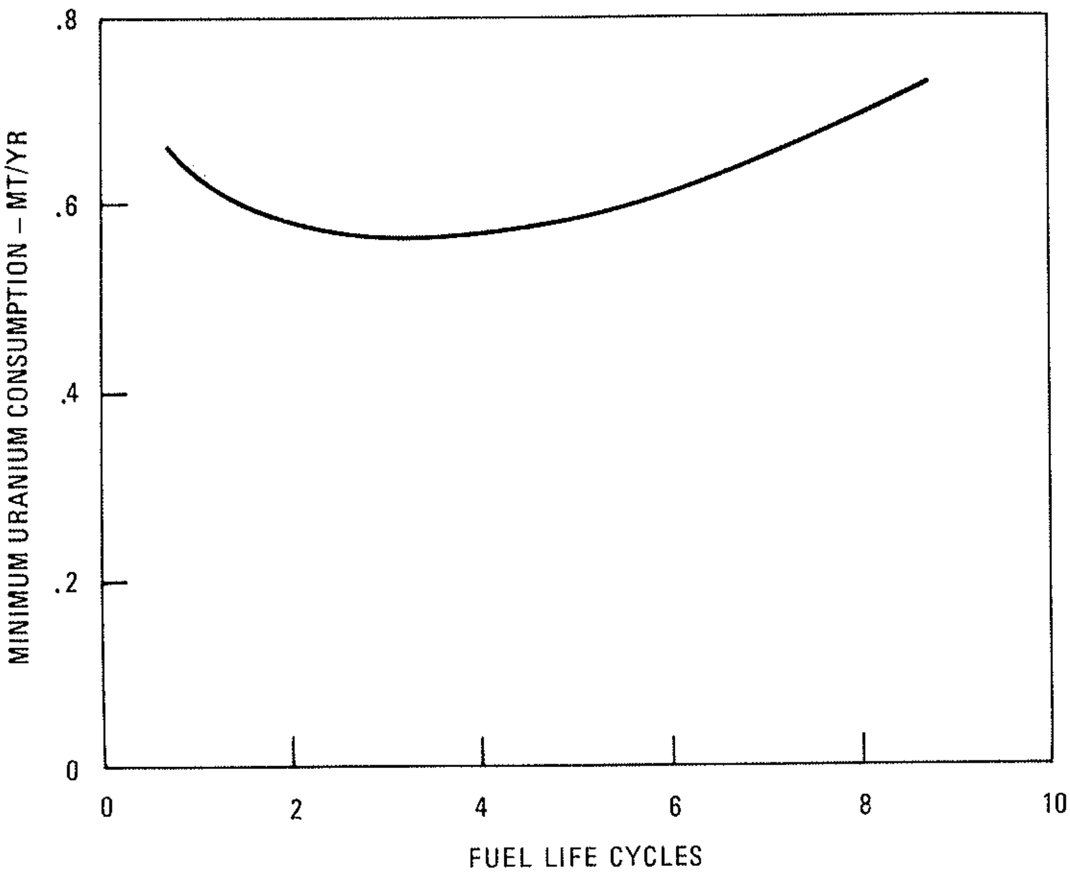

Fig. 19. Minimum uranium consumption

$$U_{fission}/Th = \frac{\frac{866\ (1 - CR)}{2} + 400 + \frac{122{,}000}{35}\ (CR + 0.02\ \tau)^{3.94}}{122{,}000\ (CR + 0.02\ \tau)^{3.94}} \qquad (21)$$

These beginning-of-cycle enrichments as well as the end-of-cycle enrichments are shown in Fig. 20. Two effects are worth noting: lower conversion ratios require higher feed enrichments and secondly, re-enrichment lowers the enrichments mainly due to the larger amounts of thorium required to achieve the same conversion ratio (compare Figs. 16 and 17). It is very encouraging to see that enrichments of around four percent are adequate.

4.3. CONCLUSION

Based on simple models developed from empirical fits to data from a wide variety of previous computations and past experience, not all of which are expected to be strictly applicable, fuel cycle trends are shown which indicate that the fueling of the HTR with unreprocessed fuel bred in a fusion blanket appears feasible. The fissile fuel enrichment achievable in a fusion blanket is sufficient to operate the HTR. It appears possible to operate an HTR on unreprocessed ^{233}U bred in a fusion blanket. "Refreshing" of the HTR fuel after the initial enrichment-depletion cycle reduces fuel consumption somewhat but the gain is small due to the buildup of fission products and there appears to be no advantage of going beyond three cycles before the fuel is retired. The fissile fuel requirements for the HTR are not small, about 500-600 kg ^{233}U per year per GW_e. This reflects an optimum conversion ratio of only about 0.6 due to the impact of retiring the fuel after a given number of cycles which penalizes the heavily loaded high conversion ratio cases. It is clear that the numerical results could shift with better information on explicit calculations but it is unlikely that the basic conclusions and trends will change significantly.

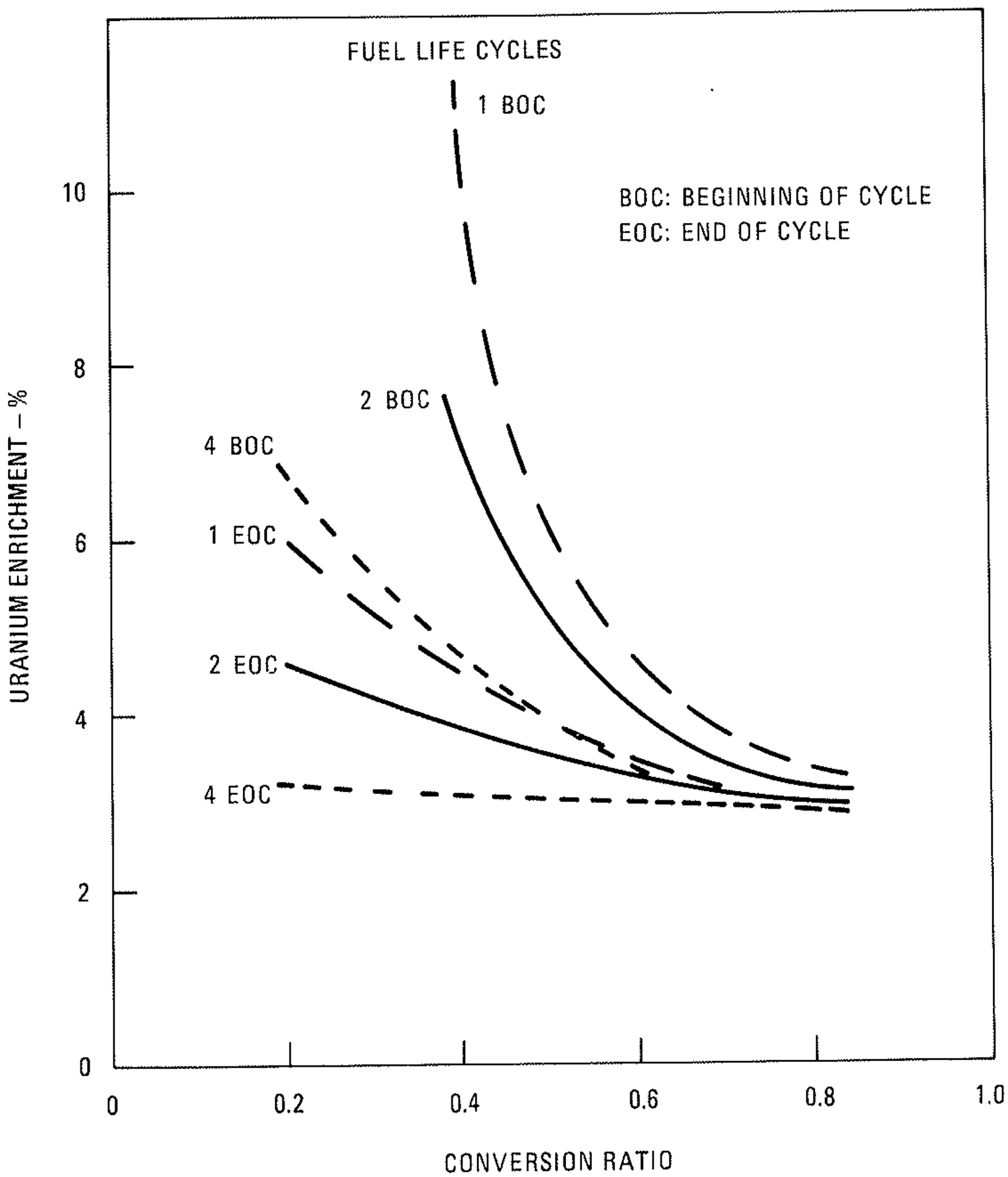

Fig. 20. Required uranium enrichment

5. SYMBIOTIC SYSTEM EVALUATION

5.1. FUSION BREEDER/FISSION BURNER MATCHING

The purpose of this study is to create a symbiotic fusion power system in which the fusion reactor makes fissile fuel from fertile material and the fission reactors convert this fuel into electricity. The fusion reactor may also produce some electricity but its main role is that of a fuel producer. Since both reactors use the same fuel and since the performance of each is strongly influenced by the fuel parameters, it is essential that the characteristics of each reactor be considered in developing the symbiotic fusion power system design concept.

The actual optimization of such a symbiotic power system will depend upon a much more detailed study of the many tradeoffs involved, including the economic considerations, which are not a part of this preliminary evaluation. It may be desirable to alter the C/Th ratio from fusion breeder to fission burner by using more or less graphite in each system. This could be accomplished quite easily if pebble bed reactor fuel geometry is used by using unfueled graphite balls in addition to those containing the thorium particles and the bred uranium. In this way the reactor spectrum could be altered to optimize the nuclear characteristics of the fuel to best suit the particular reactor the fuel is within.

5.2. SYMBIOTIC SYSTEM PERFORMANCE ESTIMATES

Using the data developed above for the fusion breeder blankets and for the fission burner reactor, four conceptual symbiotic fusion power systems have been specified. These four power systems consist of two basic blanket concepts, each used with and without a ^{238}U fission plate. The fission plate significantly enhances the uranium breeding performance of the fusion blanket at the expense of added breeder reactor thermal power and added complexity to the blanket design. The two basic blanket concepts used represent existing technology and potential technology. The "existing technology" blanket is based upon C/Th = 80 HTR fuel and remains below presently expected graphite fluence limits by operating at modest temperature (400°C) during irradiation in the fusion reactor. The "potential technology" blanket is based upon use of coated particles without any graphite in the fusion blanket which are incorporated into a removable graphite structure for subsequent depletion in the fission reactor. By use of SiC coatings and irradiation of the particles only, a fluence limit of 1×10^{23} n/cm^2 may be achieved.

Coupled with these fusion blanket design concepts is a fairly conventional HTR fuel cycle based on a C/Th = 175, which means that graphite will have to be added to the C/Th = 80 present technology fuel as well as to the C/Th = 12 potential technology fuel prior to irradiation in the fission reactor. If the C/Th = 80 fuel is in the form of pebble bed balls this can be accomplished easily by simply adding plain graphite balls to the fueled balls at a ratio of about 1 to 1. The symbiotic power systems formed in this exercise are not expected to be optimal but should give a reasonable estimate of the performance potential of the no-reprocessing concept.

5.2.1. Present Technology Blanket

The present technology blanket is characterized by C/Th = 80 and Li/Th = 0.2 and is given two full enrichment-depletion cycles before being retired. It is irradiated in the fusion reactor at 400°C to increase the fluence limit. To reach 4% enrichment requires that one-third of the 4.5×10^{22} n/cm^2 fluence limit be used. Depletion in the fission reactor at normal HTR temperatures uses 30% of the 1.5×10^{22} limit. "Refreshing" the fuel from 3% to 4% in the fusion reactor, again at low temperature, uses 12% and a final depletion in the burner brings the fuel exposure to 100% of the limit at the given temperatures. There is some uncertainty on the viability of using this temperature control scheme to achieve long fuel life. Although use of low temperature irradiation followed by high temperature irradiation will probably be quite successful, it is not certain that the return to low temperature for the Refresh cycle will reduce graphite damage to the low temperature rate and further investigation is needed.

The average characteristics of the fusion blanket for this cycle, with and without a fission plate are:

Fission plate	No	Yes
Uranium breeding ratio	0.25	0.50
Tritium breeding ratio	0.77	1.54
Energy multiplication	3.0	10.8

Coupling these blankets with a two-cycle fission reactor design with C/Th = 175 and 0.60 conversion ratio with 4.0% enrichment feed requirement and 3.2% discharge enrichment, the system parameters shown in Table 5 are specified.

TABLE 5
SYMBIOTIC SYSTEM POWER

(MW per MW of fusion power)
Blanket C/Th = 80, Burner C/Th = 175

Fission plate	No	Yes
Fusion plasma power, thermal	1.0	1.0
Blanket fission power, thermal	1.60	7.84
Fusion reactor power, thermal	2.60	8.84
Fusion reactor power, gross electric[a]	0.80	2.71
Fission reactor power, thermal	3.55	7.09
Fission reactor power, electric[a]	1.42	2.84
Total fission power, thermal	5.15	14.9
Total power, thermal	6.15	15.9
Total power, gross electric	2.21	5.54

[a] Efficiencies: fusion reactor 31%, fission reactors 40%

The fuel burnup in percent of fissions per initial metal atom (% FIMA) is approximately 7.3%.

5.2.2. Potential Technology Blanket

The potential technology blanket is characterized by C/Th = 12 and Li/Th = 0.15 and can sustain three enrichment-depletion cycles before the SiC coated particle fluence limit of 1×10^{23} n/cm^2 is reached. The coated particle fuel is enriched to 4% uranium in the fusion reactor and then the C/Th ratio is increased to 175 by incorporation of graphite for depletion in the fission reactor where the enrichment is reduced to 3.2%. The fuel will then have the graphite removed and be refreshed to 4% in the fusion blanket and have graphite added and again be depleted twice more before the fluence limit is reached for the SiC coatings and the fuel must be retired. There appears to be no need for reduced temperatures to extend the fuel fluence limits.

The average characteristics of the fusion blanket for this design concept are:

Fission plate	No	Yes
Uranium breeding ratio	0.86	1.72
Tritium breeding ratio	0.50	1.0
Energy multiplication	6.1	17.0

Coupling these blankets to a three-cycle fission reactor design with C/Th = 175 and 0.60 conversion ratio, the system parameters shown in Table 6 are specified. The fuel burnup is approximately 8.2% FIMA when the fuel is retired.

TABLE 6
SYMBIOTIC SYSTEM POWER

(MW per MW of fusion power)

Blanket C/Th = 12, Burner C/Th = 175

Fission plate	No	Yes
Fusion plasma power, thermal	1.0	1.0
Blanket fission power, thermal	4.1	12.8
Fusion reactor power, thermal	5.1	13.8
Fusion reactor power, gross electric[(a)]	2.0	5.5
Fission reactor power, thermal	13.4	26.8
Fission reactor power, electric[(a)]	5.4	10.7
Total fission power, thermal	17.5	39.6
Total power, thermal	18.5	40.6
Total power, gross electric	7.4	16.2

[(a)] Efficiencies: fusion reactor 40%, fission reactor 40%

5.3. CONCLUSIONS

Several conclusions may be drawn from the results presented above. The most important conclusion is that the no-reprocessing concept appears to be technically feasible. Second, the fission plate significantly improves the performance of the blanket. The uranium production, rate is doubled and the reactor can attain a tritium breeding ratio greater than one. Finally, the hard spectrum C/Th = 12 blanket shows a much higher uranium production rate than the softer spectrum C/Th = 80 case. The hot refabrication of the fuel between the two reactors to be able to achieve this better performance may be difficult to accomplish, however. The present technology C/Th = 80 fuel used without a fission plate shows very modest performance but should require virtually no fuel technology development. It is, therefore, feasible to consider a very near term proof of concept experiment based upon this fuel. The potential technology C/Th = 12 fuel, used with a fission plate, on the other hand, shows excellent performance but will require significant fuel technology development. The combination of near term possibility and long range potential is a significant advantage for the no-reprocessing concept.

Some caution must be exercised in the use of the data presented in this preliminary evaluation of the no-reprocessing concept. The models used were quite simple and of necessity incomplete. These models were where necessary extrapolated beyond the range where good accuracy could be expected. The designs did not include the thermal or mechanical design details that can strongly affect the neutronic results. The spatial variation in blanket performance was homogenized over the blanket model and detailed burnup analysis was not carried out. Caution should be exercised in use of the data because of these shortcomings. It is expected, however, that the qualitative results and functional behaviors presented here will be substantiated by more detailed and accurate analysis. It is further expected that more detailed analysis will confirm the technical feasibility of the no-reprocessing concept.

6. IMPROVEMENT POTENTIAL WITH REPROCESSING

This preliminary evaluation of producing ^{233}U in fusion reactors for use without reprocessing in fission reactors has shown that the no-reprocessing Refresh cycle concept appears to be technically feasible. This conclusion is important because it allows fusion power to offer a copious supply of fissile fuel for thermal burner fission reactors without violating any of the fuel cycle constraints that may be imposed due to concerns about nuclear proliferation. The thorium/uranium-233 fuel cycle may be used exclusively; virtually no plutonium need be produced. No initial fabrication or reprocessing of fissile material is necessary. At no time does fissile material exist in a weapons-grade form. That is, fissile material never exists without a substantial inventory of fission products. These features are important in that they allow the Refresh cycle symbiotic fusion power system to provide useful fissile fuel within possible nonproliferation constraints and guidelines.

It should be noted, however, that even with use of multiple refresh cycles, the fuel is discharged from the reactor and ultimately retired at an effective ^{233}U enrichment of about 3%. This means that substantial amounts of valuable fissile material are retired. In addition to retiring useful fuel to storage, the no-reprocessing constraint pushes the optimum burner reactor conversion ratio for minimum net fuel consumption to fairly low values, typically in the range 0.5 to 0.6. Since the HTR using ^{233}U fuel is easily capable of conversion ratios in excess of 0.8 and can, in fact, approach 1.0, this means that the burner reactors are not being used to their full potential. To take advantage of this potential, however, requires chemical reprocessing of the fissile fuel after discharge from the fission burner reactor. Reprocessing allows the discharged uranium to be recovered rather than retired and also allows accumulated fission products to be removed, improving the fuel nuclear performance. Such reprocessing is currently not allowed in the United States due to concerns about nuclear proliferation.

In the future, reprocessing of nuclear fuel may be allowed. If this were to occur, the effective performance of the symbiotic fusion power system would improve markedly. The amount of uranium consumed by a 1 GW_e HTR then becomes simply the fuel burned as none would be retired. From Eq. (20),

$$U_{burned} = 866\ (1 - CR)\ kg/GW_e\text{-}yr \tag{22}$$

Assuming a conversion ratio of 0.85 in the HTR results in a ^{233}U consumption of 133 kg/yr per GW_e. The performance of the symbiotic fusion power systems described above are shown in Table 7 using both the no-reprocessing Refresh cycle and using reprocessing.

Although caution must be exercised in use of the data from estimates based on the simple models developed in this preliminary evaluation, it is clear that the potential exists for substantial future improvement in the performance of the symbiotic fusion power system concepts developed here.

TABLE 7
SYMBIOTIC FUSION POWER SYSTEM PERFORMANCE

MW(t) of Fission Reactors Supported by 1 MW of Fusion Plasma Power

Blanket	Refresh Cycle	Reprocessing Cycle
C/Th = 80:		
No fission plate	3.5	16.3
With fission plate	7.1	32.5
C/Th = 12:		
No fission plate	5.4	22.4
With fission plate	10.7	44.8

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MR. BOGART: Ken, would you take ten minutes worth of questions?

DR. KRAKOWSKI: Two or three years ago, Don Dudziak published the results of very similar sorts of things and concluded that "burning" 4 to 5 percent U-233 into pure thorium would require four to five years. We found during the study it was very difficult to come up with a reasonable tritium breeding ratio without seeding the blanket with uranium 233 initially. Can you address that problem? How are you breeding tritium with substantial tritium breeding ratios?

DR. SCHULTZ: As you see from our present technology blanket design, we do not have an adequate tritium breeding ratio to completely replenish the fuel in a DT fusion reactor. We reach only a tritium breeding ratio of 0.8. Again, this is because of the fact that, in a thorium system, we don't employ the fast fission of uranium 238. By adding a fission plate, we indeed can push that tritium breeding ratio higher.

DR. KRAKOWSKI: Is that ratio 1 or .98?

DR. SCHULTZ: We can easily achieve tritium breeding ratios in excess of 1. In this particular example, what we did is limit it to one and put the rest of the neutrons into producing U-233.

DR. WOODRUFF: Gene Woodruff, University of Washington. Ken, I'm still a little surprised you can do as well as that. Do you happen to remember the wall loading and the net fissile atoms per fusion for those cases?

DR. SCHULTZ: The wall loading in all of these cases is really independent of the performance with the exception of the time curve. All of my numbers are non-dimensional numbers with the exception of the time curve.

DR. WOODRUFF: But will the time required be linear with wall loading?

DR. SCHULTZ: Yes. And the time curve is based upon 1.5 megawatts per square meter of fusion neutron wall loading.

The blanket thickness assumed is 0.5 meters of whatever particular fuel mixture you are talking about. In the heavy thorium load cases, it's principally thorium. In the high carbon to thorium cases, it is principally carbon.

DR. WOODRUFF: Do you happen to remember what the net fissile atoms per fusion turned out to be?

DR. SCHULTZ: The net fissile atoms per fusion neutron, without a fission plate looks like this. (Graph shows range from 0.25 to 0.75.)

DR. WOODRUFF: That says "breeding ratio", but is that what you mean?

DR. SCHULTZ: Perhaps I have been unclear with that. What I am talking about here is net atoms of uranium 233 produced per fusion neutron. This curve corresponds to zero percent U-233 in the blanket.

As we build in U-233, these breeding ratio curves then are reduced as we begin to burn out U-233 as well as produce it.

DR. BURNETT: Before you ask your question, let me address that just a moment. For this preliminary study, we looked at four cases--present materials technology with no fission plate; present materials technology with a fission plate; and then advanced technology with no fission plate, and advanced technology with a fission plate. So it was quite a spectrum. What you see are the two outsides of those four cases. The minimum case and the maximum case. All the cases that we showed you, though, did not include reprocessing.

If you allow reprocessing, you go to the situation which Peter Fortescue has talked about in his Science article last summer (1977) discussing the potential of fusion hybrid breeders and fission burners with reprocessing. Those results are, of course, even more optimistic.

But our goal was to go back to those original three objectives of President Carter's fission energy policy: no plutonium, no reprocessing, no proliferation.

DR. TEOFILO: Vince Teofilo, Battelle Northwest.

Could you make some comments on the licensability of this concept, specifically with regard to the severe restrictions that it might place on refreshing without refabrication?

DR. SCHULTZ: The preliminary evaluation didn't take into account any of those sorts of matters like licensing. I think this is something that will have to be given serious thought by all of us in hybrid design.

My initial reaction here is that the fission product inventory in a fission reactor or in a hybrid blanket is going to be high. The fact that we increase this problem by recycling the fuel without reprocessing through the reactor another time would involve only factors and not than orders of magnitude. As a consequence, I think it would not be a limiting consideration.

DR. TEOFILO: I did not refer to just the fission product buildup but also to materials integrity, the material coolant chemistry, and many other factors.

DR. SCHULTZ: The high temperature reactor fuel is not burnup limited, and, as a consequence, we're talking about burnups considerably below that which coated particle fuels are capable.

The graphite, which is the limiting consideration for exposure is not, per se, a fission product barrier. It is not a structural member essential for the retention of fission products. So again, I do not feel that this will be a limiting consideration for licensing.

DR. TEOFILO: Will you be evaluating it this year?

DR. SCHULTZ: Well, the study we've got this year is fairly modest, and I think that to try to say we're going to do a detailed licensing evaluation would be incorrect. That would be overly optimistic. As in all nuclear systems studies of course, safety considerations will be of major importance.

DR. MOIR: Ralph Moir, Lawrence Livermore Laboratory.

It seems to me, Ken, that you are giving the wrong impression. You say, the conclusions say, that this is a technically feasible procedure; that is, non-proliferation, no plutonium; and the only case that made technical sense was with the fission plate.

DR. SCHULTZ: It depends on what you mean by "technical feasibility". You appear to classify any system that doesn't breed all its own tritium as infeasible. I don't necessarily agree.

DR. BURNETT: As I said, we presented two of four cases. There are intermediate cases particularly the one using advanced materials technology and no fission plate. I don't think it is correct to say that it's not technically feasible. The fusion reactor shown there is producing electricity also. The total electric output of the system is 2.2 megawatts electric for every megawatt of fusion power. So I don't think that you can say the minimum case is not technically feasible. The restrictions that you place upon yourself when you say no reprocessing, no plutonium and a proliferation-free fuel cycle are pretty restrictive, in and of themselves.

What we're saying is that given those restrictions you can still come up with a system that makes you net power and makes you fissile fuel. There are lots of things you can do to go further, and I think that's the case.

DR. MANISCALCO: Do you have a situation that has a tritium breeding ratio of greater than one without fission?

DR. BURNETT: Without a fission plate, no. With a fission plate, yes.

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NEW INITIATIVES IN TOKAMAK HYBRID STUDIES

THOMAS C. VARLJEN

PRESENTED AT THE
SECOND DMFE FUSION-FISSION ENERGY
SYSTEMS REVIEW MEETING
WASHINGTON, D.C.
NOVEMBER, 2, 1977

fusion power
systems department

Westinghouse Electric Corporation
P.O. Box 10864, Pgh. Pa. 15236

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Prepared by: T. C. Varljen

T. C. VARLJEN, MANAGER
ENGINEERING

Approved by: R. P. Rose

R. P. ROSE, MANAGER
HYBRID FUSION PROJECTS

fusion power
systems department

Westinghouse Electric Corporation
P.O. Box 10864, Pgh. Pa. 15236

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NEW INITIATIVES IN TOKAMAK HYBRID REACTOR DESIGN

THOMAS C. VARLJEN

FUSION POWER SYSTEMS DEPARTMENT
WESTINGHOUSE ELECTRIC CORPORATION
P. O. BOX 10864
PITTSBURGH, PENNSYLVANIA 15236

INTRODUCTION

The need for innovative technical solutions and bold planning in the energy field has never been greater than at the present time. One of the most promising of the many optional strategies which have been identified is the fusion-fission hybrid which appears to have uniquely beneficial characteristics with regard to the production of fissile material from fertile material and inherent flexibility with regard to fuel cycle alternatives. In addition to the potentially dramatic impact of the hybrid on the fissile fuel supply question, the hybrid would provide the fusion program with a relatively near-term practical application which would stimulate the development of much of the hardware required for fusion electric power plants.

As a result of these considerations and recent progress which has been made in the design and testing of tokamaks, it is appropriate to consider what steps are necessary to further develop the hybrid concept, take a harder look at both the technical and economic potential of hybrids, and if all goes well, actually embark on a program to demonstrate the technology and prepare the way for commercial application.

A one-year study has been initiated at Westinghouse to develop the preliminary specifications for a near-term Demonstration Tokamak Hybrid Reactor (DTHR) and to assess the long-range potential of the concept. Though this work has only been underway for a brief period of time, significant progress has been made in the identification of design objectives and approach as well as the generation of preliminary performance results.

The logic plan for the hybrid study in FY'78 includes the following elements:

- Identification of DTHR goals and objectives
- Selection of a baseline design approach for each major DTHR subsystem and trade study alternatives
- Development of a coupled performance and cost systems model to permit sensitivity and trade studies
- Development of a first-iteration pre-conceptual design and cost estimate for the DTHR
- Assessment of the critical development issues associated with the fusion driver and the hybrid blanket
- Development of reference proliferation-resistant fuel cycle scenarios with particular emphasis on the production of U-233 fissile fuel in the hybrid for use in a conventional LWR (including the performance of an LWR on the thorium fuel cycle)

DTHR OBJECTIVES AND DESIGN PHILOSOPHY

The present study builds on prior hybrid studies, but with some important differences in objectives and ground rules. Much of the prior work in the hybrid field has considered the use of advanced, high performance fuel materials, high temperature coolants and advanced structural materials which would appear to offer the potential of relatively high fissile production rates as well as very efficient thermal-to-electrical power conversion. There is ample evidence from the development history of fission power reactors that a less-than-optimum system in terms of performance may be more viable commercially, however. On this basis an important approach in the present study is to make maximum use of proven light water fission reactor technology to define and assess a reference hybrid reactor system for an initial demonstration of fissile production on a significant scale. This approach would not preclude the consideration of advanced concepts in later stages of the study or the testing of advanced fuel and neutron application modules in the DTHR.

The principal goals which have been set for the DTHR include:

- Demonstration of the production of a significant amount of fissile fuel in a proliferation-resistant form in a near-term, relatively low cost fusion-fission reactor
- Demonstration of the integration of reactor technology, hardware and performance that can serve as proof-of-principal for future commercial applications
- Demonstration of long pulse plasma operation to assist in determining meaningful duty cycle factors for commercial applications
- Demonstration of blanket-fuel remote handling technology to determine a basis for plant availability projections

The DTHR is thus envisioned as an engineering and technology demonstration vehicle which would prepare the groundwork for a commercial prototype which would have demonstration goals related to economics. A "significant" production of fissile fuel is interpreted to mean a production rate of at least 100 kg/yr.

In addition to the demonstration of the breeding of a significant amount of fissile fuel, the DTHR would provide an urgently needed capability for testing materials, components, systems and concepts for both advanced hybrids and pure fusion reactors. Such testing would include: 1) alternate converter blanket concepts, 2) tritium fuel breeding technology, 3) the evaluation of the potential for breakeven or net electrical power generation and/or process heat capability for future applications, and 4) extended operation as a materials and engineering test reactor.

It should be noted that many of the engineering goals for DTHR are quite similar to those which have been identified for the various versions of TNS/ITR which are now under study. In this respect, the DTHR can provide an important near-term demonstration of fusion driver technology which is common to pure fusion, fissile production and synthesis fuel production applications.

SUMMARY OF DTHR DESIGN INITIATIVES

A number of preliminary decisions have been made with respect to specific system parameters and features to permit the rapid development of a reference point design to serve as the starting point for subsequent system trade studies. Table 1 outlines the reference system geometry and features which have been selected.

In some respects this first trial design point is pessimistic. For instance, detailed profiling effects have not been accounted for in the neutral beam requirements and relatively inefficient positive ion source neutral beam injection has been specified without the benefit of direct energy recovery. The toroidal field capability provided is quite generous to provide maximum operational latitude in view of uncertainty with respect to plasma performance scaling. In many respects the design is reminiscent of one of the TNS options (TNS-4)[1] developed during the FY'77 joint ORNL/Westinghouse TNS design studies. The cost of this particular TNS design was estimated to be $388M (buildings and equipment only, 1977 dollars) providing some basis for the hope that the hardware cost of DTHR will be reasonable.

Another important consideration reflected in Table 1 is the provision, at least in the early design stages, for the phased implementation of certain features which would enhance the ultimate contribution of the DTHR to the overall fusion program. Tritium breeding and processing, materials and component testing, alternate fissile fuel production and synthetic fuel production modules, as well as electrical power generation could be planned for and accommodated in later stages of the program, thus reducing the initial investment and providing for program review checkpoints prior to hardware implementation.

There has been recent interest in the possible use of fusion power to generate synthetic fuels. The DTHR would provide the opportunity to test some of these concepts in situ. Of particular interest are the high-temperature electrolysis techniques such as the Westinghouse Sulphur Cycle Process[2,3] for hydrogen production which would not directly compete for fusion neutrons.

TABLE 1

PRELIMINARY DTHR DESIGN FEATURES

Plasma Major Radius	5.2 m
Plasma Minor Radius	1.2 m
Plasma Elongation	1.6
Toroidal Field on Axis	5.5 T
Plasma Current	5.1 MA
First Wall Neutron Load	2.0 MW m^{-2}
Fusion Power	950 MWt
Neutral Beam Energy, Power	200 keV, 150 MW
Plasma Pulse Duration, Period	70 s, 85 s
Duty Factor x Plant Availability (Annual)	0.20
Vacuum Vessel	Water-cooled 316 SS
Blanket Fuel Form	Thorium Oxide or Metal
Blanket Coolant	Low Temperature Water
Fuel Assemblies	LWBR Technology
Plasma Exhaust	Bundle Divertor
Divertor Particle Collection	ZrAl and Cryopanels
TF, PF and Divertor Coil Conductors	Superconducting Nb_3Sn
Coil Conductor Cooling	Forced Flow Liquid Helium
TF Coil Number	16
Peak Field at TF Windings	12 T
SF Winding Location	External to TF Array
Tritium Breeding Complete Tritium Recycle System Materials and Component Testing Alternate Fuel Modules Electrical Power Generation Synthetic Fuel Test Modules	Provided for in a Phased Program

Figure 1 is a plan view of the initial DTHR concept showing in particular the provisions made to integrate the bundle divertor. Note that a pair of adjacent TF coils has been located to subtend an angle of 30° to accommodate the divertor structure with the remaining coils located at 22° intervals. Figure 2 is a cross-section elevation of the reactor showing the initial location of the poloidal field windings, shielding and the LWBR-type blanket module assemblies. This configuration provides for vertical access to the modules for fuel shuffling or refueling.

The particular rationale for the tentative selection of some of the major DTHR parameters and features shown in Table 1 will now be described.

FUSION DRIVER CONCEPT

The function of the fusion driver is to generate neutrons that will be used in the transmutation of fertile material to fissile fuel in a hybrid blanket. The quantity of fuel produced is dependent on system size and geometry as well as on the neutron wall loading. The present design is aimed at achieving a wall loading of at least 1 MW/m^2 from beam-plasma fusion reactors and, by taking advantage of plasma-plasma reactions, to drive the wall loading to $\gtrsim 2$ MW/m^2. This should be accomplished with the least costly machine feasible, under the constraint that a stable plasma be achieved under normal operating conditions. An important ground rule for the DTHR is that it should achieve the stated goals with a neutral beam driven plasma. Flexibility has been provided, however, for several modes of operation.

Typical plasma parameters for the reference geometry are presented in Table 2 for three modes of operation: pure TCT, beam-driven and ignition. For each mode of operation $\beta_p < A$, $\beta < 6\%$, $n_e a \leq 2 \times 10^{20}$ m^{-2} (200 keV D° injection) and the required energy confinement times (τ_E's) required to achieve an energy balance (consistent with the plasma temperatures) are less than the τ_E's that would be expected based on an empirical formula. This latter consideration may be interpreted as an indication of the margin for success.

FIGURE 1
DTHR PLAN VIEW

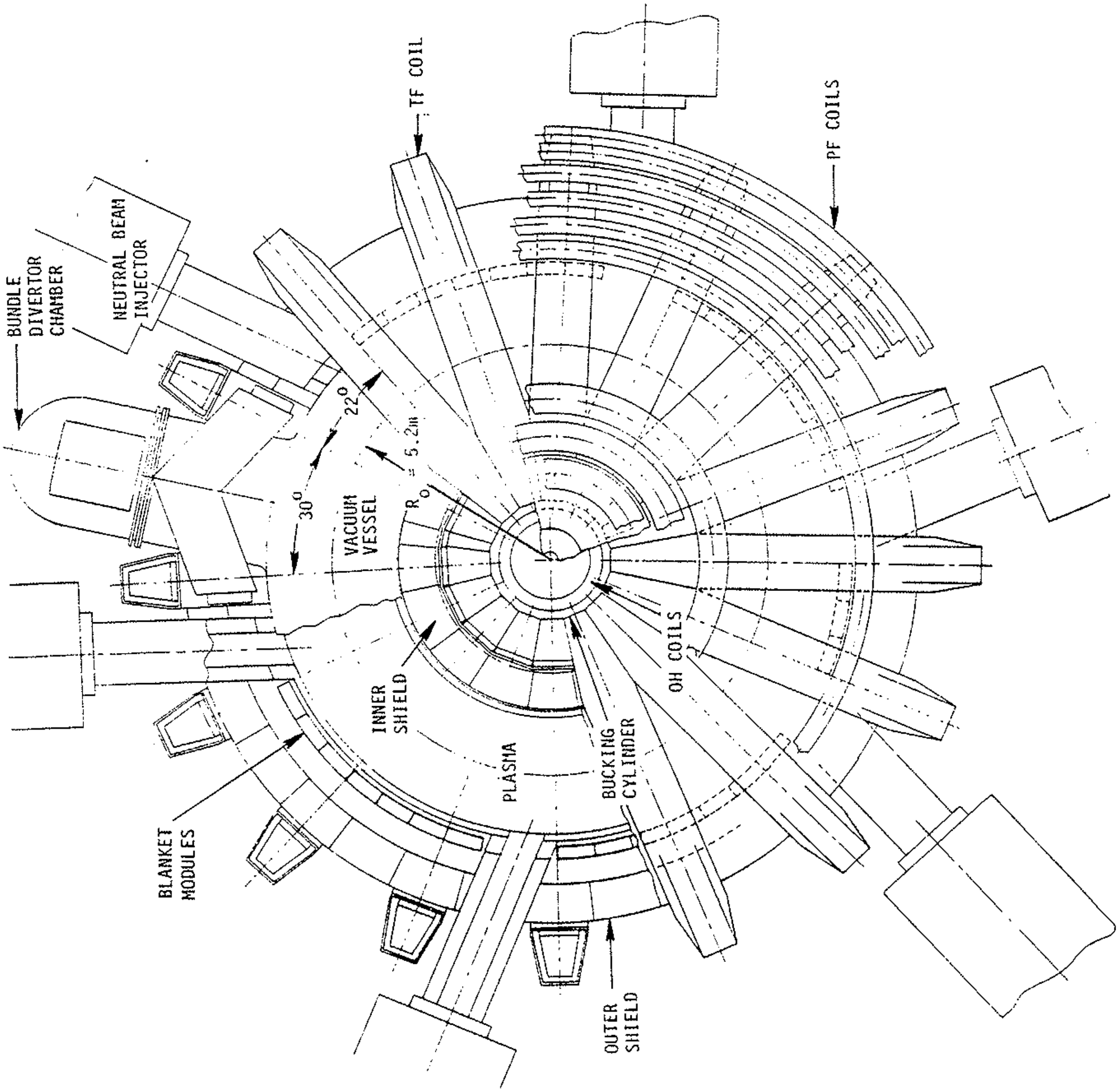

FIGURE 2
DTHR CROSS SECTION

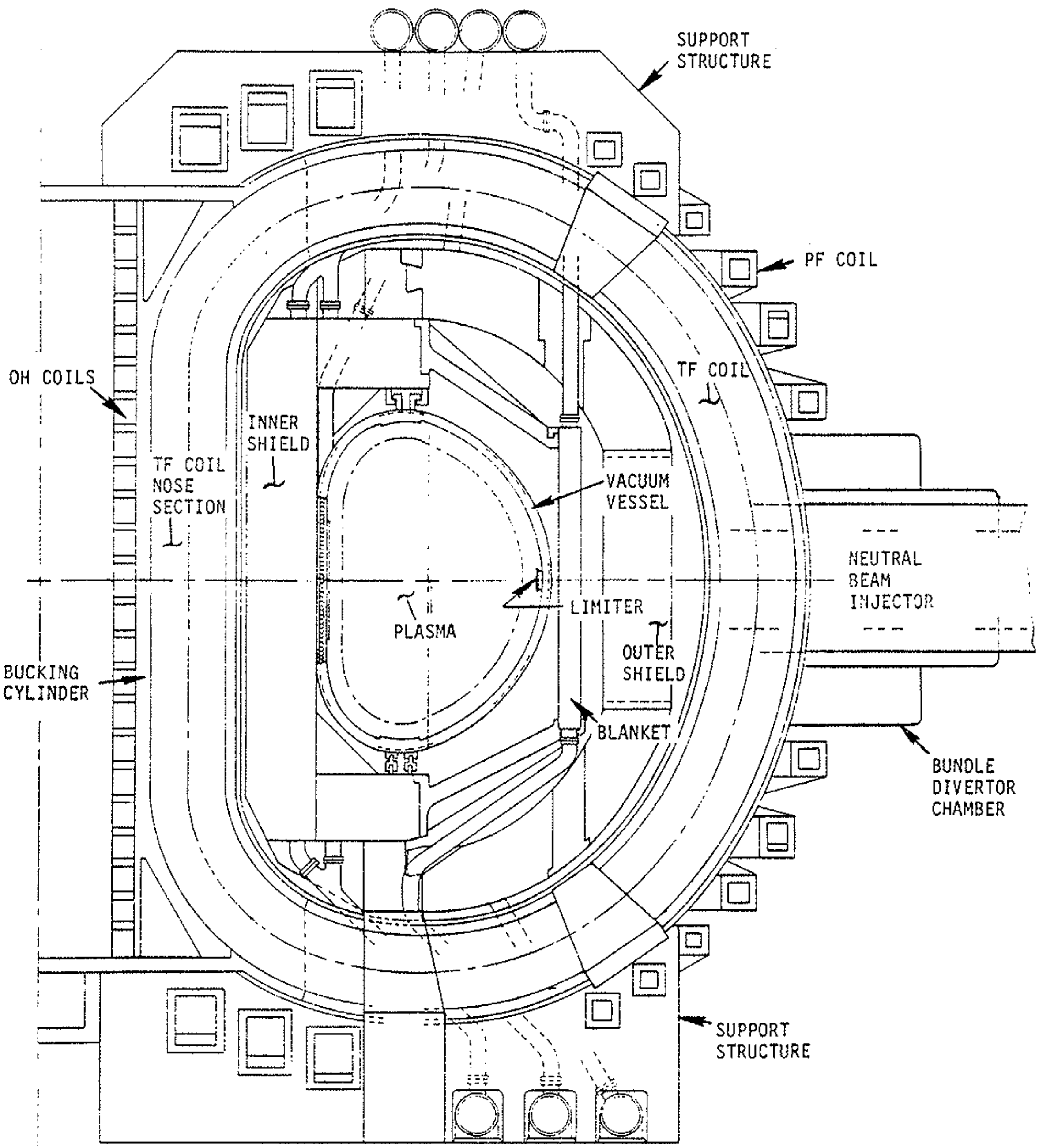

TABLE 2

TYPICAL PLASMA PARAMETERS FOR THE REFERENCE DTHR GEOMETRY

OPERATING MODE	IGNITION		BEAM DRIVEN		PURE TCT†	
T (keV)	13	13	13 **	6	13	6
n_e (10^{20} m^{-3})	1.67	1.33	1.50	1.50	0.50	0.50
n_T (10^{20} m^{-3})	0.83	0.67	0.75	0.75	0.37	0.37
n_D (10^{20} m^{-3})	0.83	0.67	0.75	0.75	0.13	0.13
$n_e a$ (10^{20} m^{-2})	2.0	1.6	1.8	1.8	0.6	0.6
β (%)	6.0	4.8	6.0	2.7	2.3	1.2
β_p	4.0	3.2	4.0	1.8	1.6	0.8
Q (beam)	-	-	1.15	0.725	1.7	1.1
Q (total)	-	-	6.3	1.43	2.1	1.1
WALL LOAD (MW m^{-2})	2.1	1.3	2.0	0.45	0.68	0.37
FUSION POWER (MW)	950	610	950	210	320	170
REQUIRED τ_E (s)	1.4	1.7	0.50	0.50	0.35	0.18
EMPIRICAL τ_E* (s)	2.2	1.7	2.0	2.0	0.65	0.65
$n_e\tau_E$ (10^{20} m^{-3} s)	2.3	2.3	0.75	0.75	0.17	0.09

†All deuterium is added via neutral beams

* $\tau_E = 3.2 \times 10^{-21}\, n_e\, (sa)^2\, q^{1/2}$, $s = \sqrt{(1 + \delta^2)/2}$

** Reference operating mode for neutronic calculations

The reference parameters presented represent a trial design point which is being used as a basis for performance sensitivity studies. Alternative designs are being considered, for instance, which are somewhat smaller than the reference and which are more strongly beam driven.

BLANKET-SHIELD DESIGN CONCEPTS

Scoping design studies for the DTHR blanket have been initiated to survey a number of alternative fuels, fuel forms coolants, and geometries to establish reference designs meeting the preliminary performance and economic objectives which have been formulated. In the long run it will be advantageous to provide for the simultaneous testing of several types of hybrid blanket modules in DTHR; however, the cost of carrying a large number of fuel options through the qualification stage may be prohibitive. The initial fuel screening activities for DTHR are, therefore, aimed at identifying a relatively small number of options, each of which represents a relatively mature technology from the point of view of prior materials characterization, fabrication and fission reactor irradiation experience.

Based on the above considerations the most obvious candidates for evaluation would be the fuel-clad-coolant systems currently being used in commercial light water power reactors and the various fuels used in the Hanford and Savannah River production reactors. If any of these can be shown to be viable choices for both the near term DTHR application and first-generation commercial deployment, it will be possible to take advantage of the vast experience which exists for these fuel systems and consequently enhance initial acceptance of the hybrid concept by the technical community, regulating agencies and ultimately the electrical utilities.

An important additional consideration at this time is the current national interest in the identification of nuclear fuel systems which appear more favorable, from a diversion or proliferation point of view, than systems involving unrestricted recycle of plutonium. DOE has instituted a major activity called the Nonproliferation Alternative Systems Assessment Program (NASAP), to evaluate a large number of overall fuel cycles and consider possible redirection of current fuel technology RD&D and commercial power reactor fuel utilization scenarios. The results of the NASAP study will probably not be available until the end of the current fiscal year; however, it may be anticipated that thorium-based fuel cycles will turn out to be favorable from both the proliferation and resource utilization points of view, with the possible qualification that sufficient start-up inventories of U-233 will be difficult to accumulate and will thereby limit the rate at which LWR's and advanced coverters or breeders can be deployed based on this fuel cycle.

As a consequence the DTHR studies during FY'78 will emphasize the use of the Th-U fuel cycle as a reference, with some U-Pu evaluations performed for comparison purposes. While a large number of fuel-clad-coolant options are still under evaluation, those shown in Table 3 have received the greatest attention. The reference system which has been chosen for the DTHR consists of thorium oxide pellets, zircaloy clad rods, with water coolant. The prime alternate is water-cooled, zircaloy clad metallic thorium rods. These two options represent the most mature fuel technologies available for consideration in a near term application. The use of clad balls is a strong alternate in view of the possible ease of refueling and reduction of structural concerns as compared to the clad rod systems. The use of clad balls, however, would involve significant development and qualification from the fabrication point of view and generally will not achieve the same level of neutronic performance as the rod geometries.

Technical feasibility issues related to the application of these fuel systems to a tokamak operating environment are being examined. The initial concern has been the impact of cyclic operation on fuel integrity. Preliminary results indicate that the reference geometry identified for the two leading candidate fuel systems will be adequate from a structural and thermal point of view for the DTHR application.

TABLE 3

CANDIDATE FUEL SYSTEMS FOR A NEAR-TERM DTHR

(THORIUM OR URANIUM BASED FUELS)

Fuel Material	Clad Material	Fuel Form	Coolant
Oxide Pellets*	Aluminum	Rods*(Solid or	Water*(Steam)
Oxide Microspheres	Zircaloy*	Annular)	Helium
Metal Alloy	Stainless Steel	Balls	Heat Transfer Salt
Carbide	TZM	Liquid	Liquid Metal
Molton Salt	Graphite		

* Reference DTHR Selection

DTHR BLANKET PERFORMANCE

A series of scoping neutronic studies have been initiated to investigate the range of neutronic performance attainable in the DTHR reference design and to indicate trends which might be used to optimize system designs. The base case fuel module is 25 cm thick (in the radial direction) and is divided into two radial zones with different water/metal ratios. Table 4 presents some results of the initial survey for the reference system with oxide and metallic thorium based fuels. These results are based on the following assumptions:

- Plasma Fusion Neutron Power = 760 MWt
- Fusion Neutron Wall Loading = 2 MW m^{-2}
- Blanket Wall Coverage Fraction = 0.24
- Plant Availability = 0.4
- Plasma Duty Cycle = 0.5

The calculations were done for beginning of life (BOL) for the same blanket thickness (25 cm) and the same zone compositions. It can be seen that the ^{233}U production rate is about 20% higher with the metallic alloy fuel consistent with the higher thorium atom density and relative heavy metal inventory. The total power production is essentially the same in both cases.

The last three assumptions set for the neutronic calculations dealing with blanket coverage, plant availability and duty cycle are considered very conservative, but consistent with the initial minimum cost goals of DTHR. The estimated fissile production rates are well above the target production rate of 100 kg/yr which has been set, thus providing confidence that the goal of significant fertile/fissile transmutation can be met with relative ease in the DTHR.

TABLE 4

RELATIVE BOL PERFORMANCE OF ThO_2 and Th ALLOY FUEL SYSTEMS IN THE REFERENCE DTHR SYSTEM

	ThO_2	Th (5% Zr) Alloy
U^{233} Production Rate (kg/yr)	201	240
Total (Fusion + Fission) Power (MWt)	1230	1280
Relative Heavy Metal Inventory	1.00	1.20
Average U^{233} Concentration (1 yr.) (1st Zone/2nd Zone)	0.30%/0.19%	0.27%/0.18%

DTHR MATERIALS TESTING CAPABILITY

In addition to demonstrating the fissile manufacturing capability of tokamaks, the DTHR will provide an irradiation test bed for materials and component testing. Its capabilities are generally indicated in Table 5 for the reference system geometry and operating scenario. It is obvious that the DTHR should be capable of providing large fluences over relatively large test volumes, with modest irradiation duty factors.

PLASMA EXHAUST SYSTEM

The DTHR will be designed to demonstrate the technology for the achievement of a moderately long ($\sim$ 70 s) plasma pulse duration. It can be shown that when the device is operated in the beam driven mode, energetic particles will be injected at such a rate as to double the plasma density in a time that is very short compared to the burn time. Although there is some doubt as to the precise magnitude of the particle confinement time which may be expected, there is no doubt that it is many times smaller than the reference discharge pulse length. Therefore, recycling of the exhaust gas must be minimized and must be removed essentially continuously during a long pulse. The most likely means for accomplishing this is by magnetically diverting the ions escaping from the plasma into a burial chamber where they may be effectively gettered or pumped. Important subsidiary benefits anticipated with the use of a divertor is the reduction of the alpha particle load on the first wall and the minimization of plasma-wall interactions which would permit maintenance of a relatively low Z_{eff} in the plasma.

A bundle divertor has been selected as the plasma exhaust reference concept for initial DTHR studies, based in part on the bundle divertor concept developed for TNS[4].

TABLE 5

DTHR MATERIALS TESTING CAPABILITY

TOTAL FUSION POWER	950 MWt
NEUTRON WALL LOADING	2 MW m^{-2}
CURRENT DENSITY OF SOURCE NEUTRONS	1×10^{18} n m^{-2} s^{-1}
DUTY FACTOR x PLANT AVAILABILITY	0.20

AFTER THREE YEARS OF OPERATION, WILL ACHIEVE:

INTEGRATED WALL LOADING	1.2 MW yr m^{-2}
SS-316 FIRST WALL DAMAGE	
- FAST NEUTRON FLUENCE, E > 0.1 MeV	1.04×10^{26} n m^{-2}
- ATOM DISPLACEMENTS	18.3 DPA's
- He PRODUCTION	246 appm He
- H PRODUCTION	862 appm H
MAXIMUM ThO_2 BLANKET DOSES	
- TOTAL NEUTRON FLUX, E > 0.1 MeV	5.85×10^{18} n m^{-2} s^{-1}
- NEUTRON FLUENCE, E > 10 MeV	2.65×10^{25} n m^{-2}
- NEUTRON FLUENCE, E > 1 MeV	5.36×10^{25} n m^{-2}
- NEUTRON FLUENCE, E > 0.1 MeV	9.25×10^{25} n m^{-2}

The essential features of the concept under development are:

- D-shaped Superconducting Nb_3Sn Divertor Coils
- Forced Flow Liquid Helium Cooling at 4.2 k
- Nuclear Shielding of at Least 0.6 m Provided for the Coils
- Solid-state Particle Collection:
    - Water-cooled Zr/Al Coated Tubes and Cryopanels
- On-line Collector Regeneration

Extensive iterations have been initiated between the bundle divertor and the toroidal field coil system. Initial MHD equilibrium for the configuration has been established and initial force calculations indicate that the structural design is feasible.

TOROIDAL FIELD AND POLOIDAL FIELD COIL SYSTEMS

The initial conductor selections for these systems involve the use of forced flow, liquid helium cooled, superconducting Nb_3Sn for the toroidal field (TF), ohmic heating (OH) and shaping field (SF) coils. This selection is particularly necessary in view of the peak field (12 T) specified for the TF coils and current uncertainties in the design margins required to assure the stability of rapidly pulsed superconducting poloidal field windings.

PROGRAM RECOMMENDATIONS

Key experimental programs related to the tokamak fusion driver technology are in-place and should provide answers to many of the critical physics and technology questions which have been identified for the hybrid driver within the next five years. These programs include the Large Coil Program (LCP), PDX, ISX-B, the high power neutral beam program and TFTR. Therefore, it is felt that the technology base will exist for a hybrid demonstration in the mid-to-late 1980's, provided care is taken to utilize known technology to the maximum extent and appropriate design and RD&D effort is mounted to take advantage of the results of the main-line experimental program in turn provide those programs a focussed application.

Figure 3 outlines the key elements of a program to complete the evaluation of the potential of the hybrid concept and, given that favorable results are obtained from the evaluation process, a schedule for the design and construction of the DTHR and its follow-on, a Prototype Tokamak Hybrid Reactor (PTHR). As envisioned the DTHR would provide the engineering and technology demonstration function while the PTHR would provide the commercial technical and economic demonstration. The DTHR schedule as defined represents a minimum duration base case and assumes success in related programs and no funding constraints. Another important point to note is that the initiation of DTHR Title I is keyed to initial operation of TFTR in 1981 and DTHR hardware procurement and construction (Title III) is predicated on successful D-T operation in TFTR in 1983.

The emphasis on early and parallel commercialization assessment studies as shown in the plan is crucial at this time to place a proper perspective on the economic incentives and benefits which might be derived from the development and deployment of hybrid technology, as well as to assure that the DTHR design reflects the requirements of a future commercial prototype. It is essential in the context of realistic cost benefit and risk assessment analyses to assure that it meets the required tests before large sums are committed to its development. On this basis, it is imperative that every effort be made to define a feasible commercial reference point with credible cost estimates.

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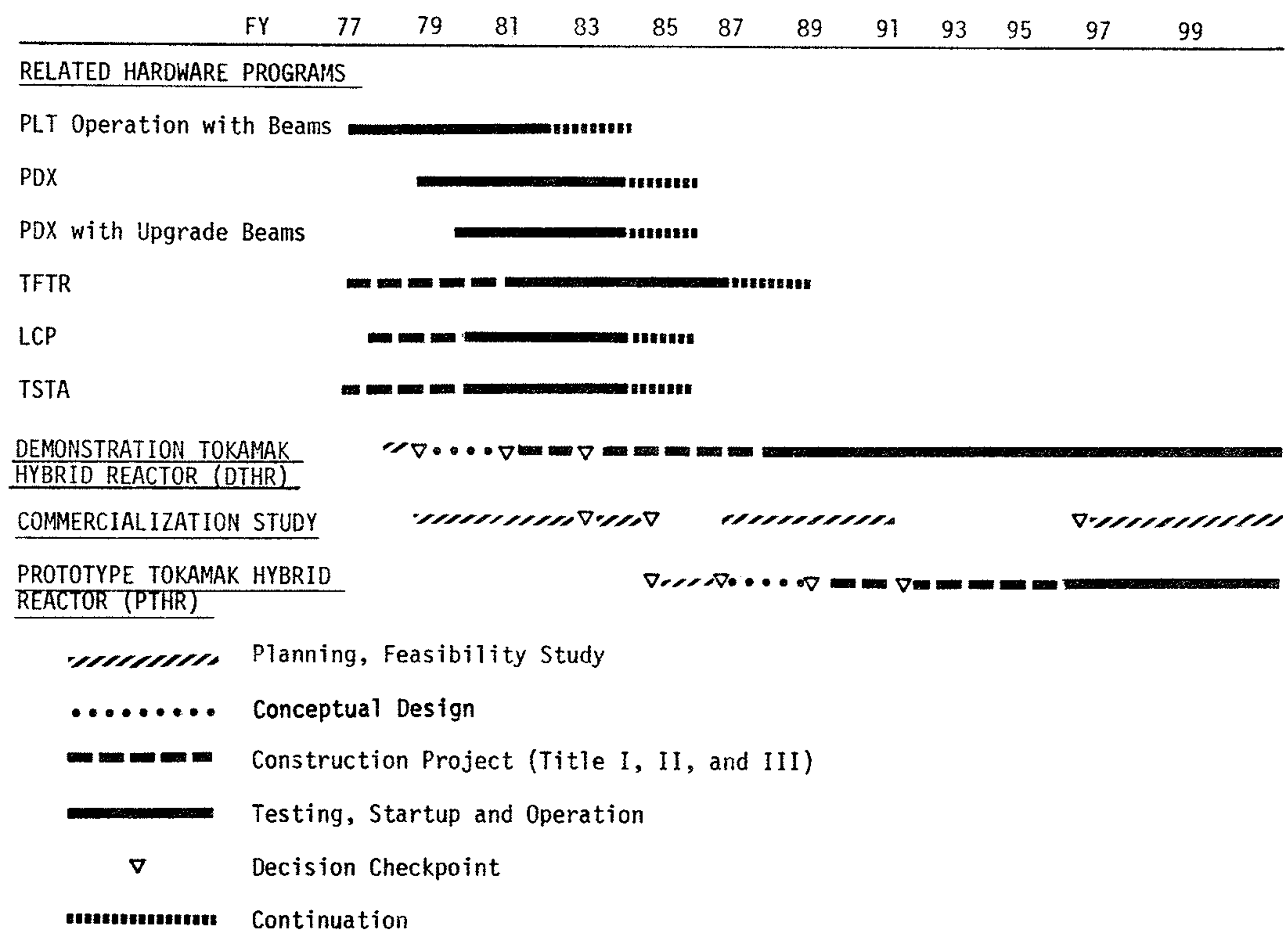

FIGURE 3. Key Elements of the Tokamak Hybrid Reactor Development, Demonstration and Commercialization Program

DR. TEOFILO: Vince Teofilo, Battelle Northwest.

You mentioned that the bundle divertor would be a significant technological breakthrough for Tokamaks, and you mentioned that Ted Yang had shown MHD equilibrium of this concept.

DR. VARLJEN: Not yet. Probably it will be done in the context of ISX-B if it is adopted. ISX-B is dickering with the Culham people to use, perhaps, a DITE-type divertor.

I am not sure where that stands. We're working with Dave Swane, at least, to show how that concept, Ted's concept, can be fitted to the ISX-B. If that goes forward, I am sure that these very detailed calculations will occur.

We've done at least preliminary calculations above the level of a back-of-the-envelope calculation, but nothing like a very detailed stability calculation.

DR. TEOFILO: I'd also like to make a comment on that last slide, that recommendation of a new initiative to go forward with Tokamak Demonstration Hybrid Reactor.

Bob Conn of the University of Wisconsin and myself completed a study last February with that same goal, to operate a Tokamak Demonstration Hybrid Reactor in the late 1980s, based upon the Tokamak Engineering Test Reactor study that the University of Wisconsin had performed.

DR. HURWITZ: I note that you item schedule for the demo was 1990, probably allowing a little slippage. How do you recommend factoring in other decisions that presumably have to be made earlier in view of the limitations of nuclear resources that were discussed this morning.

DR. VARLJEN: What other such decisions?

DR. HURWITZ: Support of other approaches like the LMFBR.

DR. VARLJEN: What we are doing, at least at this time, is carrying forward a specific point design for a standard reference scenario, a fuel cycle scenario. We also are continuing to do trade studies which will pick--we're using the reference thorium U-233 cycle, the reference uranium oxide-plutonium fuel cycle--two or three of the NASAP scenarios and carry those forward in the trade study mode to see if we have any engineering feasibility problems associated with a blanket which could accommodate any one of those fuel cycles.

And then one does not have to commit himself until conceptual design which could--until the end of conceptual design or the start of Title I construction, be as early as three years from now or further on, depending on the support and the funding.

DR. HURWITZ: What about the other independent pure fission programs, or don't you consider that?

DR. VARLJEN: Well, the pure fission programs really have not defined what they are going to do--in other words, TNS is still in the air. The next step after TFTR is up in the air.

DR. HURWITZ: Pure fission?

DR. VARLJEN: I'm sorry. That word has bounced around so much today. How does that question go again?

DR. HURWITZ: Well, I gather from some of the speakers this morning that some fundamental decisions and advances have to be made in the pure fission area long before the fusion prototype demonstrations will be available.

I was wondering how you imagined such decisions to be impacted by what you're doing.

DR. VARLJEN: I think that really it should be the other way around. They should not consider what we're doing. In other words, the fission reactor policy ought to be set based on what they know, the bird-in-the-hand kind of idea.

What we're doing is still in the technology development area, and we want to do a demonstration which can encompass any one of these fuel cycle scenarios. I don't think we're going to be in a position to tell them what is best from the economic point of view until they settle their political problems and so on, and until we are able to do more hard engineering so that we know exactly what it costs to build a divertor; we know exactly what it costs to fuel these tokamaks.

Right now, those are important quesitons which we can only come to grips with doing some hardnosed engineering. Thank you very much.

NEW INITIATIVES IN LASER DRIVEN FUSION-FISSION ENERGY SYSTEMS*

James A. Maniscalco

and

Luisa F. Hansen

Lawrence Livermore Laboratory
University of California
Livermore, California 94550

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INTRODUCTION

We have been investigating the potential of laser fusion driven hybrids at LLL for several years. Our earlier studies[1,2] primarily used neutronic methods of analysis to identify attractive hybrid configurations and to provide an upper bound estimate on their performance. The most significant features which emerged from our earlier neutronic studies were:

1) Hybrids can be designed to meet a broad spectrum of energy multiplying and fissile fuel producing requirements.
2) Hybrids can operate with fusion energy gains* that are an order of magnitude lower than required for pure laser fusion.
3) Laser fusion hybrids produce ten times more fissile fuel per unit of thermal energy generated than fission breeder reactors. Our results show that a depleted uranium fueled hybrid could fuel six light water reactors (LWR's) of equivalent thermal power and that one thorium fueled hybrid could fuel more than 20 advanced convertor reactors (conversion ratio = 0.8) of equivalent thermal power.

More recent hybrid studies dealing with the engineering, environmental, and economic issues as well as the neutronic aspects have been carried out in collaboration with Bechtel[3,4] and Westinghouse[5] corporations. The results of these studies were reported in an earlier session of this meeting. It was encouraging to note that, in general, the attractive features which emerged from our neutronic studies were retained in both of the engineering designs. Several other important features of laser fusion driven hybrids emerged from the process of performing the more detailed engineering design studies. First, laser repetition rate was found to be an excellent control variable to

*Fusion energy gain is defined as the product of laser efficiency and target gain; as such, it represents the ratio of fusion energy to electrical energy input to the laser.

compensate for changes in blanket energy multiplication so that a constant output power level could always be maintained. Second, we found that laser fusion driven hybrids will cost at least twice as much as LWR's and that they probably will be more expensive than fission breeder reactors. Finally, and most importantly, the cost of electricity in a scenario with hybrids providing fissile fuel for LWR's was shown to be insensitive to the capital cost of the hybrid. It increased over present prices by only 20 to 40% when the hybrid's cost ranged from two to three times more than an LWR.

The results presented here have led us to conclude that fissile fuel production is a much more attractive role for a fusion-fission hybrid than was previously realized. Electricity should be produced in the hybrid only to the extent that it makes the fuel product cheaper. Further, as is shown in the section on Timetables for New Sources of Energy, a fissile fuel producing hybrid could impact our energy needs much faster than a power producer.

We have performed engineering design studies on plutonium producing hybrids. In future studies, we would like to perform an engineering design study of equivalent depth on a laser fusion driven hybrid which produces ^{233}U for modified LWR's and advanced converter reactors. The anticipated performance of a ^{233}U producing hybrid which appears attractive will be presented in the Blanket Design section

<u>Proliferation Issues</u>

There has been a significant change in the U. S. Government policy on reprocessing spent reactor fuel. Upon taking office, President Carter put forth an energy program which called for indefinite deferral of reprocessing and recycling of plutonium, and for cancellation of the Clinch River Breeder Reactor. To compensate for these deferrals, the President's program calls for acceleration of research into alternative reactors and nuclear fuel cycles which do not allow direct access to materials useable in nuclear weapons. The purpose of this policy is to reduce the risk of additional countries obtaining the materials, technology, and equipment necessary for the manufacture of nuclear weapons.

Fusion-fission hybrids do not require plutonium or for that matter any fissile enrichment; therefore, they do offer the prospects of more fully utilizing our thorium and uranium resources without full scale reprocessing and recycling of plutonium. The ^{233}U/thorium fuel cycle has been proposed as a less proliferating nuclear fission option. In this approach, thermal fission reactors could be supplied with ^{233}U that has been diluted with ^{238}U to a point where it is no longer a nuclear weapon material.

Laser fusion driven hybrids and reprocessing facilities located in secure areas could provide denatured uranium to thermal burner reactors located outside the secure areas. However, with ^{238}U in their fuel, the thermal burner reactors will produce plutonium; therefore, their spent fuel must be removed and shipped back to secure areas where the plutonium could be removed and either burned in the hybrids or stored. The larger the number of thermal burner reactors which can be fueled from a ^{233}U producing hybrid of equivalent thermal power, the smaller the fraction of energy generation which must be located in secure areas. For example, 10% of the total energy generation would have to be located in secure areas if a thorium fueled hybrid can produce enough denatured ^{233}U to fuel 9 thermal burner reactors. The ^{233}U producing hybrid described in the Blanket Design section could be operated in this less proliferating fashion.

Going further towards less proliferating fuel cycles, one can consider hybrid schemes which don't require fuel reprocessing. However, all of the nonreprocessing options that we have considered exhibit reduced fissile fuel production performance and result in more costly electricity than the reprocessing options. One nonreprocessing scheme proposed by Westinghouse[6] involves the fissile enriching of depleted uranium fuel by irradiation in a hybrid for subsequent burnup in an LWR. The feasibility of this approach would primarily rest with the development of high-burnup fuel elements that could operate in both the hybrid and LWR environments. Alternatively, the enriched hybrid fuel could be paritally reprocessed without removing the fission products.In this approach, the hybrid fuel could be declad, ground up, homogenized and compacted into LWR fuel pins. All of these operations would have to be conducted at high cost in shielded hot cells. Fissile fuel producing hybrids utilizing either the partial or non-reprocessing scheme could be expected to provide enough fuel for two LWR's of equivalent output power. This represents a factor-of-three reduction in the fissile fueling capability that would be available with reprocessing.

The most straightforward nonreprocessing approach would be to use hybrids strictly as power producers with throw away fuel cycles. These hybrid power plants could be fueled with either thorium or depleted uranium, thus taking advantage of the fusion energy multiplying characteristics of fuels that are abundant and cheap. Hybrid systems will be significantly more expensive than LWR's; therefore, electricity generated from power producing hybrids without fissile fuel revenues can be expected to cost substantially more than it does now. In the Blanket Design section, we describe a thermal fission blanket which could be used to produce power in this nonreprocessing mode. By selecting this approach, we are implying that proliferation issues will be important enough to steer us to less economical modes of generating electricity. Therefore, we will attempt to estimate how much more expensive electricity will be from a power producing hybrid which uses a throw away fuel cycle.

TIMETABLES FOR NEW SOURCES OF ENERGY

To understand the role that a fissile-fuel producing hybrid could play as an electrical energy supplier, one must project the energy supply-demand situation into the 21st century. Coal fired plants and uranium fueled light water reactors (LWR's) will be the only significant sources of electrical energy going into the 21st century, with the mix being something like 70% coal and 30% LWR. The exact split will be determined by factors such as air pollution, effects of CO_2 in the atmosphere, proliferation issues, uranium supply, and economics. Somewhere near the turn of the century (give or take 10 years), the lack of an assured uranium fuel supply will begin to curtail LWR deployment. Specifically, the ratio of the uranium rate-of-find to the rate-of-consumption will not be large enough to reasonably assure 30 to 40 years of fuel for newly committed LWR's. At this point, there will be a need for a new source of electrical energy--not just a source which can be deployed by the turn of the century, but one which can pick up a significant fraction of the national grid in the first 25 years of the next century. This need will be accentuated in regions of the country where coal can not be burned because of air pollution.

The problem common to any advanced power producer, be it fission breeder, solar, or fusion is that the new source of energy does not contribute something approaching 10% of the total electric demand until more than twenty years after commercial demonstration. At one point, the LMFBR had the potential to meet our energy needs in the 2000 to 2025 time frame because it would have been available for commercial deployment before 1990.

Energy planners have looked to advanced converter reactors on the thorium-^{233}U cycle (HTGRs, LWBRs and CANDU reactors) to help us fill the anticipated energy gap. Unfortunately, without a breeder of ^{233}U, these reactors can not significantly improve a fissile fuel shortage problem in the 2000-2025 time frame. To illustrate this, consider the HTGR because it is closest to commercial deployment in the U.S. Before a utility would buy an HTGR, a commercial size plant must be built and the back end of the fuel cycle must be closed. This will take at least 10 years. Here we assume that the poor start-up performance of Fort St. Vrain can not be expected to stimulate utility orders. Ten years later (about 2000), we would have at most 10 HTGRs commercially deployed. Without a breeder of ^{233}U, these systems must burn ^{235}U and/or Pu, resulting in a conversion ratio closer to 0.6 than 0.8. Therefore, in the 2000 to 2025 time frame, they will not significantly replace or supplement LWR fuels. Our major contention here is that a hybrid which produces fissile fuel for existing fission burners could impact the grid much faster than any new power producer, and if the hybrid were available for commercial deployment by 2000, it could meet our energy needs in the 2000 to 2025 time frame. The fraction of the electric grid which a hybrid that fuels existing burner reactors could impact is (N+1) times the fraction the hybrid supplies alone (N is the number of fission burner reactors of equivalent thermal power which can be fueled by one hybrid).

One study which illustrates the potential speed of hybrid impact on the grid was performed by Lotker[7]. He calculated the time scale that will allow hybrid fueled capacity to catch-up to LMFBR capacity for various delay times between the commercialization of the two technologies. For example, Fig. 1 shows that for hybrid introduction beginning 10 years after LMFBR introduction, hybrid-fueled capacity at the lower rate of hybrid installation (4.3 GW_e per year) will surpass LMFBR capacity in two to four years depending on the LMFBR construction rate. The implicit assumption in this calculation is that an LMFBR will only refuel itself while a hybrid will provide fuel for six existing LWRs of equivalent capacity. This results in a 7:1 energy generation ratio between a hybrid and an LMFBR.

HYBRID BLANKET DESIGNS

Several different fusion-fission hybrid blanket designs have been investigated and reported in the literature. They include uranium fueled fast-fission blankets, thorium fast-fission blankets, thermal fission blankets, and plutonium and ^{233}U enriched fast-fission blankets. To add to the confusion, some of these blankets have been designed to emphasize fissile fuel production while others emphasize power production. At LLL, we have analyzed all of these different blankets, and we believe that they can be categorized into one generic fusion-fission hybrid with several "after-blanket" options as shown in Fig. 2. Our major contention here is that all of the blanket systems described above perform best when they are preceeded by a uranium-fueled fast-fission blanket. To emphasize this, Table I shows the energy multiplication and neutron leakage characteristics for several depleted-uranium fast-fission blankets of varying thickness. The fast fission blanket in this example contains by weight 53% depleted uranium metal, 35% sodium, and 12% stainless steel. This corresponds to the composition used in the LLL/Bechtel hybrid design reported earlier. The results show that a region of depleted uranium from 4 to 15 cm thick will multiply the fusion neutron energy by a factor of three to seven and more significantly, the number of neutrons leaving the blanket will be 1.6 to 1.9 times larger than the number of incident fusion neutrons.

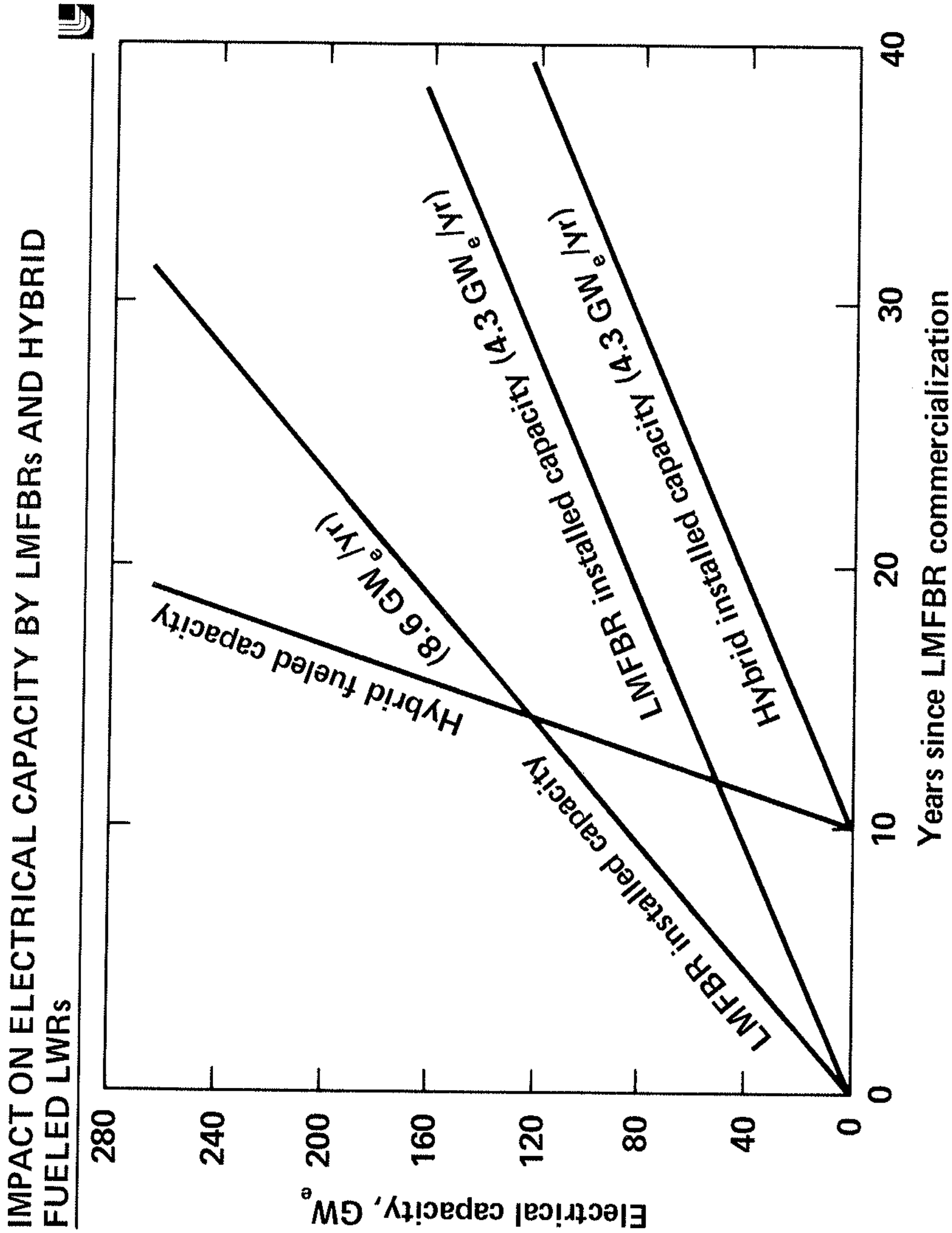
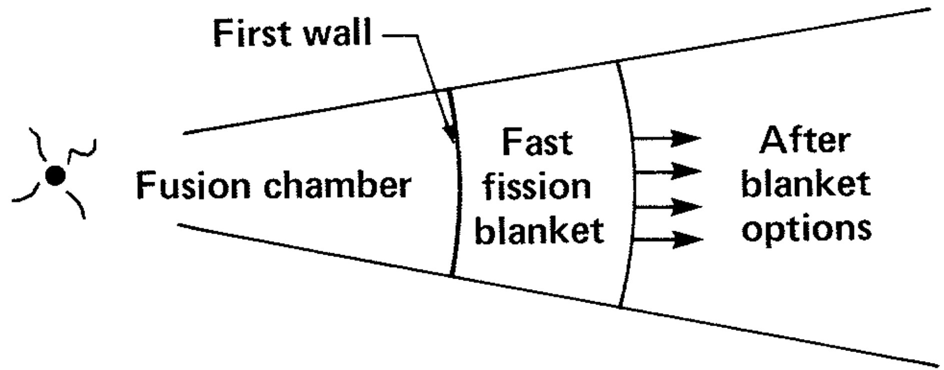

IMPACT ON ELECTRICAL CAPACITY BY LMFBRs AND HYBRID FUELED LWRs
Electrical capacity, GW$_e$
0
40
80
120
160
200
240
280
Years since LMFBR commercialization
0
10
20
30
40
LMFBR installed capacity
Hybrid fueled capacity
(8.6 GW$_e$/yr)
LMFBR installed capacity (4.3 GW$_e$/yr)
Hybrid installed capacity (4.3 GW$_e$/yr)

Fig. 1

WE SEE ONLY ONE GENERIC FUSION–FISSION HYBRID WITH SEVERAL AFTER BLANKET OPTIONS



Fast fission blanket characteristics

- Thickness – 6 to 12 cm
- Fuel – depleted uranium
- Energy multiplication – 3 to 6
- Neutrons (per source neutron) leaving the back of the blanket – 1.7 to 1.9

After blanket options

- Uranium to produce Pu
- Thermal lattice to produce power
- Thorium to produce U^{233}
- Lithium to produce excess tritium

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Fig. 2

ENERGY MULTIPLICATION AND NEUTRON LEAKAGE FROM A
DEPLETED-URANIUM FAST-FISSION BLANKET

URANIUM THICKNESS, cm	ENERGY MULTIPLICATION	NEUTRON LEAKAGE
4	1.95	1.55
6	3.5	1.71
8	3.9	1.82
10	5.1	1.85
12	5.8	1.78
15	6.7	1.68

a) Thermal energy deposited in the blanket per 14.1 MeV neutron.

b) The number of neutrons leaking out of the blanket per source neutron.

TABLE I

The increased performance resulting from the fast-fission blanket can be illustrated by considering a thorium blanket with and without a fast-fission plate. Earlier neutronic results reported in the literature[2] demonstrated that a 20 cm region of thorium could multiply the fusion neutron energy by a factor of 2 and could produce 0.8 ^{233}U atoms per source neutron while maintaining a tritium breeding ratio greater than one. In contrast, a hybrid system with 7 cm of depleted uranium followed by 14 cm of thorium multiplied the fusion neutron energy by 6.6 and produced 0.85 ^{233}U and 0.70 ^{239}Pu atoms while maintaining a tritium breeding ratio greater than one.

At the bottom of Fig. 2, we list four after-blanket options. The uranium-to-produce-plutonium option includes the fast fission blankets that we investigated in our design studies with Bechtel and Westinghouse. This option has been fairly well scoped out and reported in the literature.[3,4,5] In the next two options, a thermal lattice behind the fast fission blanket can make power, or a thorium after-blanket area can produce ^{233}U. Both of these after-blanket options are more fully developed later in this section. The final option, lithium behind the fast fission region, can breed excess tritium for the startup of a pure fusion economy.

When fusion-fission hybrid designs are reported in the literature, there is a tendency to describe them by and therefore emphasize their after-blanket options rather than the inner blanket regions. We maintain that the most crucial issues affecting the technical and economic feasibility of fusion-fission hybrid systems occurs in the blanket region closest to the fusion source. It is this region that is exposed to the harshest and least known environment (x-rays, debris, 14 MeV neutrons, and cyclical stresses from the fusion microexplosion). Therefore, important performance indicators such as first wall flux, blanket power density, and blanket lifetime will be determined by the conditions in the region closest to the fusion source.

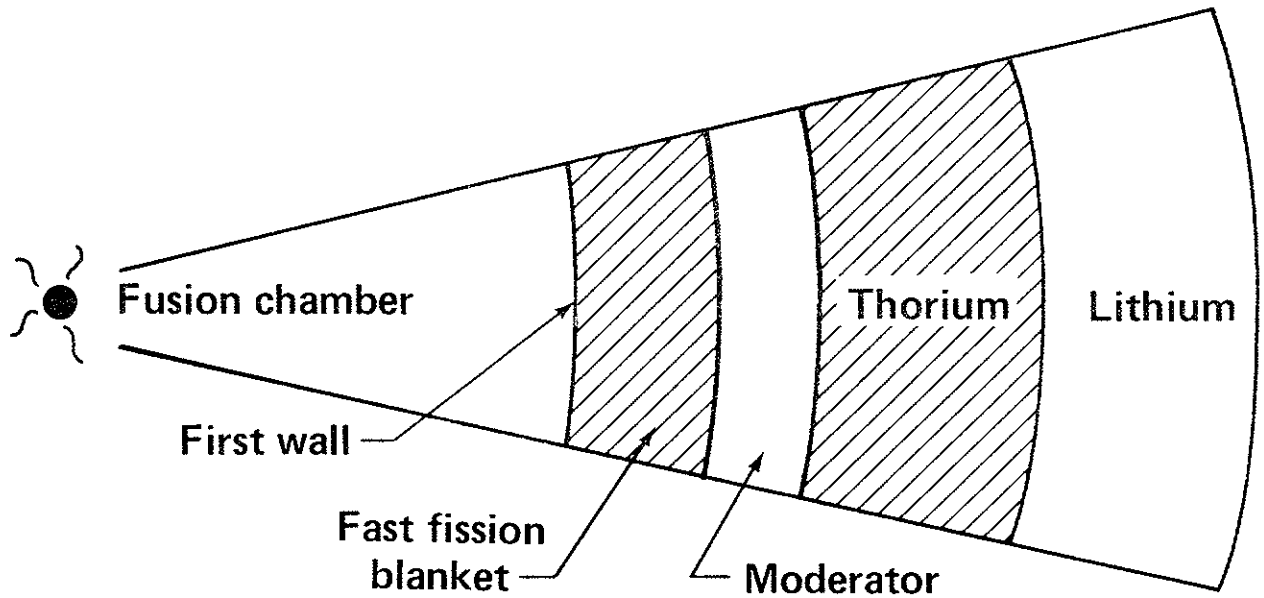
In addition, our results have shown that the performance of a fast fission blanket is always increased by maximizing the uranium atom densities in the regions closest to the fusion source. The presence of moderating material in the form of coolant, uranium compounds (UO_2, UC_2, or UC), and structure degrades both the energy and neutron multiplying performance of a fast fission blanket. To complicate matters, the highest power densities and hardest neutron spectra occur in this inner region where we are seeking to maximize uranium atom density.

A Hybrid Blanket Which Produces ^{233}U

In fig. 3, we show the generic configuration of a hybrid blanket which produces ^{233}U. It consists of a depleted uranium fueled fast-fission blanket positioned immediately behind the first wall, followed by a moderator zone, a thorium blanket, and a lithium-containing tritium-breeding blanket. The fission blanket multiplies the fusion energy and more importantly, the number of fusion neutrons. The composition and design of the thorium blanket fuel is chosen with the end use of the fissile fuel in mind. For example, the thorium could be incorporated in the blanket as a clad fuel if the ^{233}U produced is intended for light or heavy water reactors. It could be manufactured into carbon clad spheres if high temperature gas (HTGR) or pebble bed fuel reactors are to be fueled, and finally, it could be incorporated in a fluid fuel form (molten salt or aqueous) to allow rapid removal of the ^{233}U. The thickness of the moderator region between the front uranium blanket and the thorium blanket can be chosen to serve two functions: A thick moderator will minimize the concentration of ^{232}U in the thorium blanket, thereby producing ^{233}U fuel that is easier to handle (^{232}U is a hard gamma emitter produced by (n,2n) reactions induced in ^{233}U by high energy neutrons). Conversely, a thin moderator zone will result in a higher concentration of ^{232}U making the fuel more difficult to handle, and therefore providing a less proliferating fuel option.

Neutronic scoping calculations have been carried out for the design shown in Fig. 3 in order to investigate the effects of fast fission blanket parameters such as thickness, composition, and fuel type (metal, carbide or oxide form) on the production of ^{233}U. The fast region contains by weight 53% uranium fuel, 35% lithium coolant and 12% stainless steel structures. These proportions were used in an earlier

A HYBRID WHICH PRODUCES U^{233}



Thorium zone options

- Clad fuel for LWR's or HWR's
- HTGR or pebble bed fuels
- Molten salt and other fluid fuels

Moderator zone options

- Thick to produce fuel that is easier to reprocess
- Thin to produce more proliferation resistant fuel

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Fig. 3

hybrid design study with Bechtel,[3] although the Bechtel design used sodium rather than lithium as the coolant. In these scoping calculations, the lithium is 50% enriched in ^6Li to increase tritium production at the expense of Pu production. The moderator region contains carbon, and the thorium region has the same proportions of fuel, coolant, and stainless steel as the fast blanket with the fuel being thorium metal, and the coolant being sodium. The production of ^{233}U and ^{239}Pu, the blanket multiplication, and the tritium breeding ratio are shown in Table II. The results are shown for three different fuels in the fast blanket: uranium metal, uranium carbide, and LWR spent fuel in carbide form. Production of ^{233}U in kg per MW_t-yr and blanket energy multiplication are the output variables to be maximized in this blanket system. Blanket energy multiplication is important because it allows the hybrid to operate with lower fusion energy gain performance. By maximizing ^{233}U production per unit of thermal energy, the number of thermal burner reactors which can be fueled from the hybrid blanket is maximized.

To put our results for fissile fuel production in perspective, a hybrid which produces 1 kg of ^{233}U per MW_t-yr can fuel nine to thirteen thermal burner reactors with conversion ratios of 0.7 to 0.8. In contrast, a hybrid which produces 1 Kg/MW_t-yr of ^{239}Pu can fuel five to seven LWRs with conversion ratios from 0.5 to 0.6. In a realistic design of a hybrid reactor, a tritium breeding ratio larger than 1.0 is required. This was accomplished in our earlier hybrid design studies by dedicating the top and bottom of the reactor to tritium breeding.

The neutronic results presented here lead us to believe that we can design a laser fusion driven hybrid blanket which multiplies the fusion neutron energy by a factor of five and produces enough ^{233}U to fuel more than ten fission burner reactors of equivalent thermal power. With blanket energy multiplications in the range of five, a hybrid can operate with fusion energy gains that are three times lower than required for pure fusion power production. From our earlier hybrid designs studies with Bechtel, we estimate that the capital cost of this ^{233}U producing laser fusion driven hybrid would be about three times as much as an LWR.

TABLE II: NEUTRONIC PERFORMANCE OF A HYBRID WHICH PRODUCES ^{233}U

Case	U Zone(cm)	C Zone(cm)	Th Zone(cm)	Blanket Multiplication (M)	Tritium Breeding Ratio	Pu Production Kg/MW-yr	^{233}U Production Kg/MW-yr
1	6^{a}	13	20	4.89	.83	.622	1.05
2	8^{a}	13	20	5.66	.88	.659	.864
3	8^{b}	13	20	4.06	.88	.549	1.08
4	8^{b*}	13	20	4.08	.49	1.21	1.38
5	8^{c}	13	20	4.46	.49	.535	1.00
6	8^{c*}	13	20	4.78	.50	.96	1.24

a) depleted uranium metal

b) depleted uranium carbide

c) spent LWR fuel in carbide form

* Natural lithium instead of lithium enriched to 50% ^{6}Li has been used as the coolant in the uranium fast fission blanket.

Therefore, in a combined scenario with this type of hybrid producing ^{233}U for thermal burner reactors, the cost of electricity can be expected to increase over present prices by less than 30%.

<u>A Hybrid Power Producer Which Does Not Require Fuel Reprocessing</u>

Since fusion-fission hybrids don't require fissile fuel to generate energy, they can be designed to produce electricity with a nonreprocessing or throw away fuel cycle. An example of such a hybrid design is shown in Fig. 4. In this design, a thermal fission lattice has been placed behind the fast fission blanket.

Depleted uranium blanket elements in the fast fission region will breed 0.5 to 1.5 % plutonium by weight and will then be shifted to the thermal lattice where the plutonium will be burned. Considerable flexibility in the fuel cycle is possible. The rate and extent to which plutonium is burned is a function of the thermal lattice spacing (degree of moderation) and thickness of the front fast-fission blanket. Either graphite or beryllium would be a satisfactory moderator. Instead of one cycle in front and one or more cycles in back before being discarded, fuel elements could be cycled twice (front, back, front, back). This approach involves peak burnups of 20,000 to 30,000 MWD/Mg and would require U-7 % Mo (by weight) or UC. It is obviously desirable to design the fuel cycle so that the maximum power density and maximum fuel temperature in the two regions are similar.

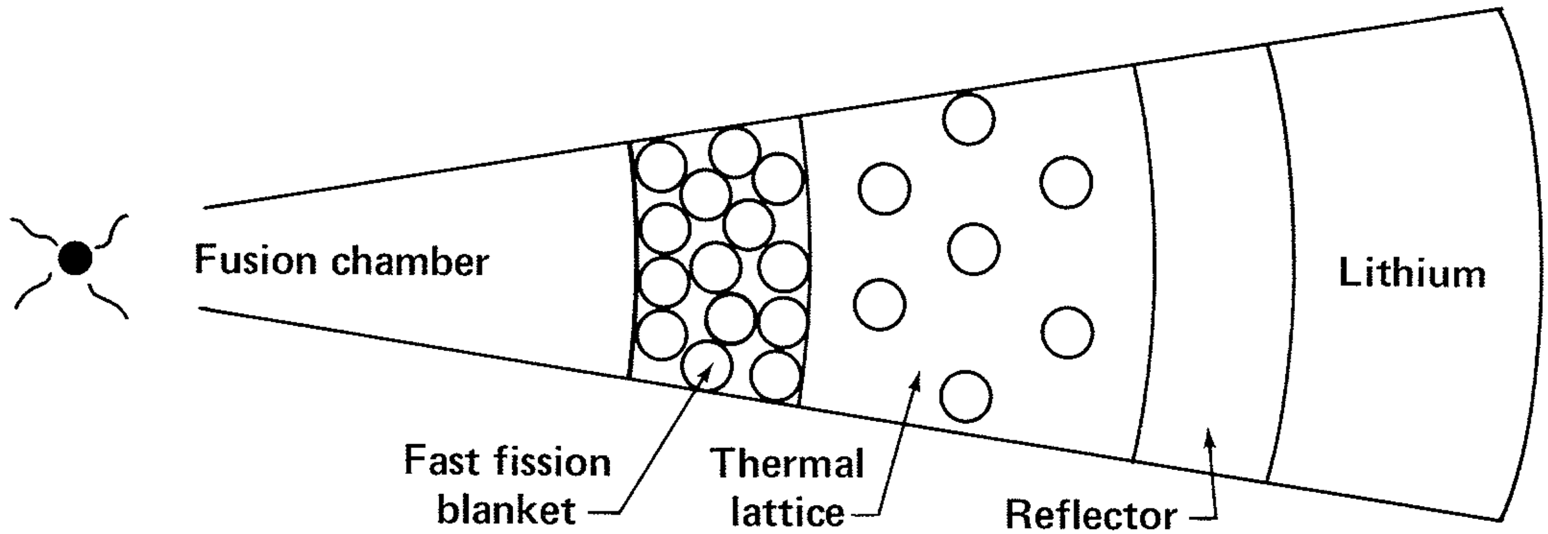
The nonproliferation and environmental advantages of the proposed concept are probably unique:

No uranium mining and milling is required,
No uranium enrichment is required,
No reprocessing is required.

Enough depleted uranium tailings will exist by the year 2000 to fuel hundreds of the proposed reactors throughout their lifetimes. If reprocessing were allowed, all the uranium-238 economically available could be burned without the need for fast breeder reactor.

From the results of earlier studies, we estimate that the blanket design shown in Fig. 4 could multiply the fusion neutron energy by more

A HYBRID POWER PRODUCER WHICH DOES NOT REQUIRE FUEL REPROCESSING



- SS-clad, dep-U fuel bundles are enriched in plutonium in the fast fission blanket
- The enriched fuel bundles are then loaded in the thermal lattice where the plutonium is burned

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Fig. 4

than 15. Therefore, this power producing hybrid could operate with fusion energy gains that are an order-of-magnitude lower than required for pure fusion power production. The capital cost of this hybrid system in $/kW can be expected to be cheaper than lower energy multiplying fissile fuel producers. Nevertheless, we would expect it to still cost at least twice as much as an LWR. Therefore, the cost of electricity from this hybrid power producer is expected to be at least 50% more than present price for LWR-produced electricity.* If fuel burnups of 30,000 MWD/Mg can be achieved, this power producing hybrid will allow us to extend the energy available from uranium in a throw away fuel cycle by a factor of five.

CONCLUSIONS

Our earlier studies showed that fissile fuel production will be a very attractive role for a fusion-fission hybrid. Since laser fusion driven hybrids will be at least twice as expensive as LWRs and most likely more expensive than fast breeder reactors, they should not be designed as pure power producers. To be feasible, they must be designed to:

- Produce fissile fuel for existing thermal burner reactors.
- Operate in a less proliferating fashion than plutonium fueled fast breeder reactors.

It is desirable for the hybrid to meet both of these objectives; however, the pursuit of less proliferating options almost always increases the number of hybrids required to fuel the existing grid of thermal burner reactors and the price of electricity in the combined scenario.

We have demonstrated that fusion-fission hybrids exhibit the best energy multiplication and fissile fuel production performance when the

*For example, assume LWR costs are 20 mills/kWh capital, 6 mills/kWh fuel cycle and 2 mills/kWh O&M. Then, if the corresponding hybrid costs are 40, 2, and 2 mills/kWh, the total cost increases from 28 to 44 mills/kWh (57% increase).

region closest to the fusion source is a uranium fast-fission blanket, and we have explored the potential of this type of blanket with two different after-blanket options: A ^{233}U-producing thorium after-blanket, and a power-producing thermal lattice. Our neutronic analysis of the thorium after-blanket option leads us to conclude that we can design a laser fusion driven hybrid which multiplies the fusion neutron energy by a factor of five and produces enough ^{233}U to fuel more than ten thermal burner reactors of equivalent thermal power. With capital costs in the neighborhood of three times an LWR, the cost of electricity in the combined hybrid-burner reactor scenario can be expected to increase less than 30% over present prices.

We have estimated that the thermal lattice after-blanket option would lead to a power producing hybrid which could operate with fusion energy gains that are an order of magnitude lower than required for pure fusion power production. The nonproliferation and environmental advantages of this hybrid power producer result from the fact that it doesn't require fuel reprocessing, uranium enrichment, or urnaium mining and milling.

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