To E. Elis

ERDA-4

C. II

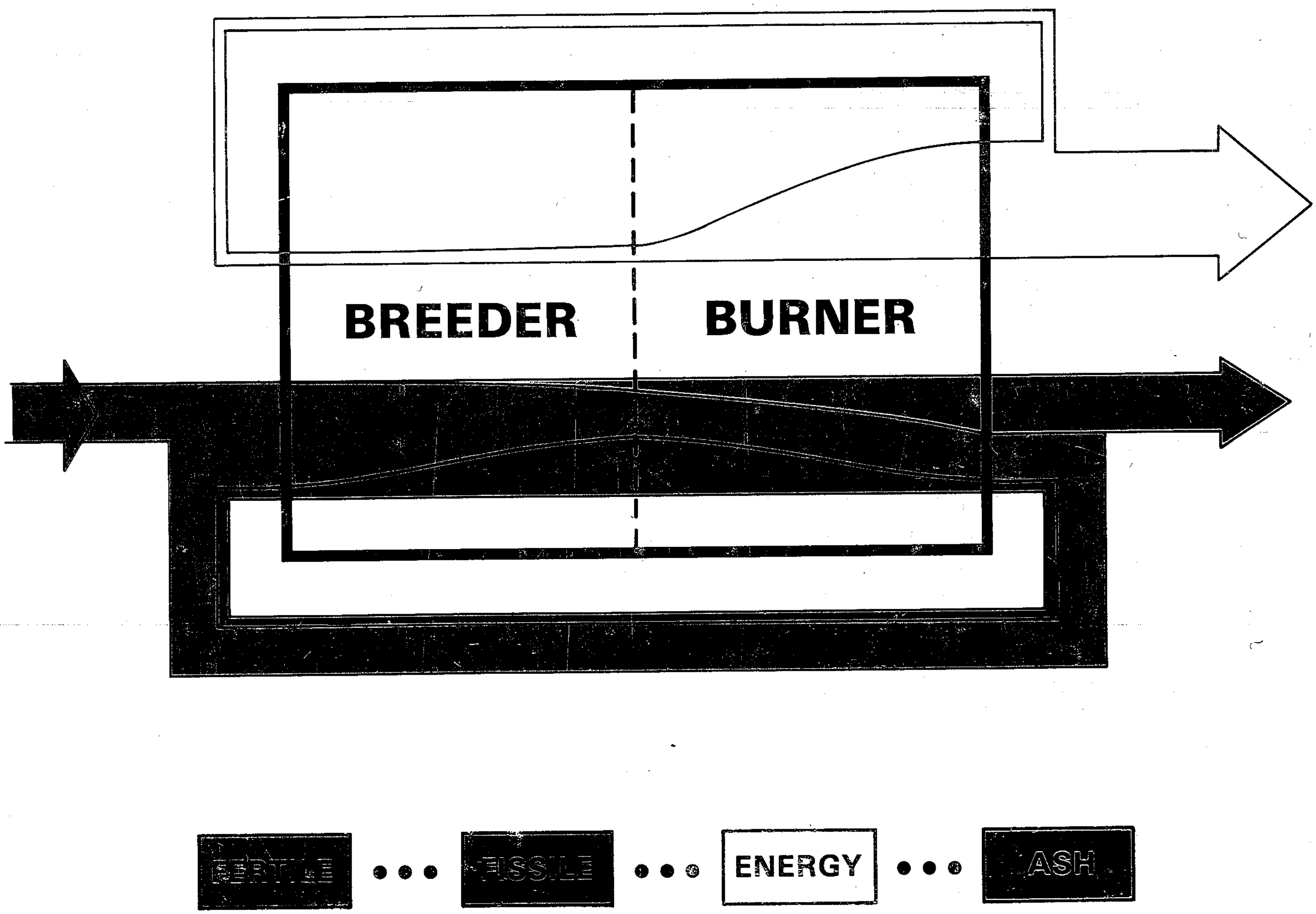
DCTR FUSION-FISSION ENERGY SYSTEMS REVIEW MEETING

CTP

DECEMBER 3 and 4, 1974

ERDA ▪ GERMANTOWN, MARYLAND

CONCEPTUAL FUSION-FISSION ENERGY SYSTEM



U.S. GOVERNMENT PRINTING OFFICE:1975—O-570-664

For sale by the Superintendent of Documents, U.S. Government Printing Office
Washington, D.C. 20402 - Price $4.50

ERDA-4
UC-20

DCTR FUSION—FISSION ENERGY SYSTEMS REVIEW MEETING

DECEMBER 3 and 4, 1974

■

ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
Germantown, Maryland

S. LOCKE BOGART
SYSTEMS AND APPLICATIONS STUDIES BRANCH
DEVELOPMENT AND TECHNOLOGY PROGRAM

There was a man in our town,
And he was wond'rous wise;
He jumped into a bramble bush,
And scratch'd out both his eyes!

And when he saw what he had done,
With all his might and main,
He jumped back in that bramble bush
And scratch'd them in again!

- Randall Garrett

TABLE OF CONTENTS

OVERVIEW

S. Locke Bogart

On December 3 and 4, 1974, meetings were held at ERDA, Germantown to initiate an assessment of fusion-fission energy systems. The meeting on December 3 was devoted to a series of presentations on fusion-fission research efforts being performed at national laboratories as well as presentations on the point of view of utilities and environmental interests. A history and overview of fusion-fission studies also was given. The second day was structured to focus on different perspectives of the status of the following important elements:

- Fusion physics
- Reactor engineering
- Economic justification and role
- Safety, environment, and safeguards.

A guide was prepared for this discussion and is appended to the end of this overview. A conceptual study plan for fusion-fission energy systems also was prepared (appended) but was not discussed because of time limitations.

Questions to the speakers on the first day quickly pointed out many of the inadequacies of the data-base required to assess the prospects for fusion-fission energy systems. However, the perspective of utilities was made quite clear: power companies are very concerned with the matters of nuclear fuel availability and radioactive solid waste disposal. These are long term problems but are quite real to utilities since they must formulate construction plans well into the future. A related presentation, a preliminary cost-benefit analysis, suggested that fusion-fission energy systems appear to be a competitive option in the long term. However, this analysis included the sale of electric power as well as fissile material from hybrid reactors. Utilities may place less emphasis on power than fuel from fusion-fission systems because dependable nuclear reactors are commercially available.

Both fusion physics and reactor engineering were discussed in a series of presentations. Reactor physics considerations appeared in scoping studies performed by the Lawrence Livermore Laboratory (LLL) and the Los Alamos Scientific Laboratory (LASL) and also in a theoretical study performed at the Princeton Plasma Physics Laboratory (PPPL). This latter effort focused on the optimization of a "Two Component Tokamak" for the production of maximum neutron current and indicated that an interesting quantity of fusion neutrons could be produced, in principle, by a fairly small machine ($R_{min} \sim$ 1-1.5 meter). The LASL and LLL presentations focused on linear theta-pinch and mirror confinement hybrid systems, respectively. Both designs produced electrical power as well as fissile materials and each was based on plasma physics requirements regarded as do-able. The theta-pinch hybrid was developed around the ^{232}Th-^{233}U cycle because of perceived environmental and safety advantages. The mirror hybrid principally used the ^{238}U-Pu cycle because of its demonstrated energetic properties and the advanced state of knowledge on this choice. Each design was always subcritical although much more work is required to confirm this conclusion.

An unscheduled but very valuable presentation by Dr. John Holdren (Uv. of California, Berkeley) provided a very convincing argument for early consideration of environmental and safety factors. At the very minimum, any acceptable fusion-fission device must:

- Preclude nuclear criticality accidents by a blanket design that is subcritical in its most reactive configuration,
- Preclude blanket melting in the event of LOCA by a blanket design that remains below its melting temperature by means of passive cooling . . . even in the worst circumstances.

Other characteristics such a minimizing safeguards problems and nuclear waste disposal also were presented as quite desirable. Failure to adhere to these minimum criteria was said to guarantee irretrievable environmental handicaps.

The workshop meeting on December 4 was less structured than the formal presentations of the previous day. Consequently, discussion on the status of physics, engineering, economics, and safety/environment/safeguards was amply sprinkled with digressions, many of which focused on the validity of consideration of fusion-fission energy systems. The gestalt of these digressions is presented as paraphrased statements near the end of this overview.

Opinion on the status of fusion physics is distinctly polarized: either the physics is optimistic and it is realistic to seriously consider applications for fusion neutrons; or it is not, and many years of work and a series of machines are required before fusion-fission is a credible concept. However, it was the consensus that only little near-term modification of the DCTR program would be required to include fusion-fission research. Both the tokamak and mirror programs would not be altered, but in the latter case, could be accelerated somewhat. The theta-pinch program would receive greater emphasis on the linear configurations as the physics of this choice is reasonably well understood and presently more optimistic than for the toroid.

It also is the consensus that the physics requirements for all confinement schemes would be reduced significantly and, in fact, would be somewhat different than for the pure fusion case (optimize neutron current rather than Q). The work at the Princeton Plasma Physics Laboratory resulted in the following findings:

- The $n\tau$ and T requirements for fusion-fission reactors would be reduced significantly from the pure fusion case.
- τ_E need not be long (e.g., 100ms is nominal).
- The impurity problem would be less constraining.
- The Alpha particles need not be confined and, in fact, may not be desired.
- The operating regime is below trapped ion mode (TIM) and the instability problem may be alleviated.

It is observed that similar relaxations of physics requirements exist also for the mirror and theta-pinch configurations although in somewhat different respects.

A new quasi-physics problem did emerge during the discussion. It was stated that the duty cycle of any fusion neutron source must be an important criterion. Although this certainly is evident, it was noted that all experiments to-date have a duty cycle of less than 1 percent which probably is a factor of 70 less than required. If fusion-fission is to be credible as a middle term fusion application, then duty cycle needs early consideration.

In summary, the physics looks reasonably optimistic although a new perspective should be adopted for consideration of fusion neutron sources. Since the physics for pure fusion reactors is regarded as optimistic in 1974, then the physics for a fusion-fission neutron source seems to be even more favorable. However, work now must be initiated to focus on the differences between fusion and fusion-fission neutron sources. Particular attention should be given to neutron current optimization and the cost that is incurred (i.e., maximize Γ/cost and/or Γ/power input)*. Additionally, duty cycles now must be examined for optimization.

The questions of reactor engineering are somewhat less clear than those of physics because distinct blanket configurations lead to different product mixes with different fuel cycles. The only substantive conclusion is that it appears unlikely that a pure fissile breeder will be an economical machine. The production of fissile materials will result normally in the simultaneous production of non-negligible high quality heat, the quantity of which depending on the fusion neutron production rate and the blanket fissions. It must be noted that any reasonable fusion neutron current will give rise to body heating in the blanket regardless of the system Q. This heat will have to be removed and, if in reasonable quantity, should be converted to power or some other energy form.

One statement on reactor engineering that deserves mention is that the reactor wall loading possibly can be reduced. This finding either may or may not extrapolate to economical fusion-fission devices but certainly deserves more careful study. It is evident that there will be trade-offs on wall loading.

There was raised the question of blanket inventory of both fissile and fertile materials. In several designs, very large quantities of enriched fuel was needed for the initial inventory which would require the services of an enrichment plant. Pu was suggested as the initial fissile seed, but this material would be very difficult to obtain in the required quantities, would retard the breeding ratio and would increase significantly the plant capital cost. It is clear that considerable work must be done to examine the choices of fuel cycle, enrichment, thermal or fast spectrum, moderation, etc. At this time, there is insufficient evidence to clearly identify any blanket as superior. In fact, the choice will be influenced strongly by energy marketplace factors during the next two to three decades.

The concept of fusion-fission symbiotic systems (as compared with a hybrid) was discussed at length. The utility point of view prescribed the following advantages:

---

*Γ is defined as either fusion neutron flux or fusion neutron current.

- They separate the fission subsystem from the fusion subsystem.
- They eliminate the requirement for additional power generation reserve.
- Designs for material production and economical dry cooling are plausible.
- There may be greater latitude in sizing constraints.
- They may be easier to design free of LOCA and criticality.
- Other uses for generated heat could be found, e.g., thermochemical production of <u>storable</u> hydrogen and methane (o1).
- Fissile material and waste products could be extracted as they are produced if rapid throughput schemes are used.

Again, it is not clear that symbiosis has an economical advantage over hybrid systems but it is evident that work is needed to compare them.

Fission waste burning was considered briefly. There seem to be two opinions: it is very feasible and it is not feasible. Those who think it not feasible point out that alternative disposal solutions, such as storage in salt mines, should be less expensive by orders of magnitude and, in any event, waste partitioning still would require alternative storage. The proponents point out that only several reactors would be required to handle the rather small volumes of particularly troublesome wastes (e.g., actinides). However, this application of fusion is really a socioeconomic issue and cannot be answered at this time because of the lack of quantification of attitudinal, social, and environmental factors.

The subject of safety, environment, and safeguards (SES) was discussed principally in the context of John Holdren's presentation on December 3. It was not uniformly agreed that actual fusion-fission energy systems could be operated economically while at the same time being always subcritical and free from LOCA's. The premise is that a hybrid would operate more economically near $k_{eff}=1$ although this is less obvious for a symbiotic system. In either event, a fusion-fission system likely will be more complicated than its fission analog and, additionally, will have SES problems unique to CTR's (e.g., tritium, magnets, disruptive instabilities, etc.) The major conclusion regarding SES elements is that little can be stated as fact this early into the consideration of fusion-fission systems. To a large extent, SES constraints are design specific and, with the few preliminary designs now available, no certain conclusions can be made. However, it is a matter of record that at least criticality and LOCA probably will not be acceptable to intervenor interests and future designs should take this into serious consideration.

The economic role of fusion-fission systems also is very uncertain at this time although a preliminary cost benefit analysis did suggest potential advantages. It was made quite clear that the economics will be based on the product mix (fissile fuel versus electric power) and, at this time, pure breeders appear not to be economically feasible. However, this conclusion was drawn from present forecasts of energy prices and could be seriously in error. Similarly, the lack of a reasonable data-base on different fusion-fission systems also could alter the conclusion. It is clear that more designs have to be performed from which additional cost-benefit analyses can be made. Furthermore, the cost-benefit analyses <u>should</u> include the effects of environmental and societal values as well as dollars. More detailed data on forecasted actual machine costs also must be developed. It was suggested that "questionnaires" be developed for the acquisition of data for further cost-benefit analyses.

To conclude the overview of the discussions on December 4, the following paraphrases of statements are presented to suggest the gestalt of the meeting.

- Fission scientists should participate in the assessment of fusion-fission systems since these systems will draw heavily upon fission reactor technologies.
- The public perception of fusion power attributes could be altered.
- EPR-I priorities will be affected by hybrids.
- There is no intrinsic need for a new power producer.
- Utilities are really concerned with saving 1/3 to 1/2 mill. per kwhr.
- Point designs are the only way to establish benchmarks.
- Current and next generation physics experiments will demonstrate scaling of every parameter but not simultaneously.
- Consideration of fusion-fission energy systems will not change the near-term emphasis of the overall DCTR program.
- Fusion-fission energy systems may be beam driven.
- The near term emphasis, regardless of option, is in improving plasma physics.
- The development of a reliable neutron source really is the EPR question.

- Paper studies on fusion-fission are needed to assess advantages and disadvantages.
- The utilities are questioning the CTR program to the core.
- Utilities view fusion-fission as a back-up to the LMFBR program.
- Wall loading is a potent argument for considering hybrids.
- Low wall loading destroys neutron advantages of fusion-fission.
- D-D cycles need consideration for fusion-fission.
- A fast blanket has a strong resemblance to an LMFBR.
- The lack of a need for enrichment is advantageous.
- A system with the lowest heating rate is the purest breeder in a symbiotic sense.
- It would be desirable to use waste heat for process heat.
- Symbiosis is a good scheme for utilities because of no need for power generating reserve.
- The thermal spectrum must be used to avoid actinides and fission products; fast spectrum advantages are lost.
- Electric power is worth much more than fissile materials.
- U-233 is more valuable than Pu-239.
- Hybrid Pu would be better than LMFBR Pu.
- A U-238 system can come on sooner than a thorium system.
- Many burners would not be required to handle fission wastes.
- A hybrid is strongly affected by safety requirements.
- A safe, passive blanket is the key.
- There are some safety advantages in molten salt systems.
- A molten salt blanket would be more hazardous to the fusion neutron source than other choices.

During the several month interval between the meetings of December 3 and 4, 1974, a clearer picture has emerged of the tasks that must be done for an accurate assessment of the prospects for fusion-fission energy systems. In a large measure, these concepts depend on the achievements in the plasma physics and fusion reactor engineering elements of the DCTR program. Work in these areas must continue at the current if not an accelerated rate. In addition to continuation of the basic fusion power research and development program, the DCTR also plans to expand the level of effort in the investigation of fusion-fission systems and their applications.

Near term projects include the design of hybrid and/or symbiotic energy systems with support from parametric, cost-benefit, and marketing analyses. Depending on economics, the initial stress may be for fissile fuel production in response to the perceived needs of the utilities. The heat generated during the production process then would be regarded as a byproduct and would be recovered only if it is economically feasible to do so. These designs also will focus on subcritical, passively safe blankets in response to possible criticism from those who are deeply concerned with nuclear safety. It is expected that this design constraint will result in an economic trade-off, and future work will analyse this effect. Nuclear safeguards, considered to be a problem of nuclear power, will receive attention early in the assessment of fusion-fission energy systems.

These studies are expected to last until the fall of 1977, at which time a fusion community technology assessment is to be made which will include the possible roles that fusion-fission might have in future DCTR fusion power Research & Development activities. It must be made quite clear that the assessment of fusion-fission systems will be based on hard facts: the economic role of fusion-fission systems must be established and be attractive; the plasma physics and fusion reactor engineering must be sufficiently well in hand to lend more than marginal credibility to the design concepts; and the safety, environment, and safeguard characteristics must be benign to the extent that fusion-fission is an attractive alternative to other energy sources. Only if these three criteria are met would it seem reasonable to pursue more costly involvement with fusion-fission energy systems. The potential seems to be there but let us quantify and measure it in a fashion that leaves no doubt of the results.

Near Term Questions

A. Fusion Physics -- Is the state-of-the-art of fusion (plasma) physics really at the point where sensible decisions can be made on the prospects for fusion-fission energy systems? If so, why? If not, when will fusion physics be ready?

1. What fusion drivers appear the most credible in terms of neutron flux (time average).
2. What fusion drivers appear the most credible for the minimum investment of externally supplied energy?
3. What fusion drivers appear the most credible for the minimum investment of dollars?
4. What fusion drivers appear the most credible for the earliest time of demonstration of practicality (as compared with feasibility)?
5. Are there other candidates for fusion drivers and, if so, what?

B. Reactor Engineering -- Evidently, there are a number of blanket concepts one may wish to employ depending on the desired product mix and configuration constraints. Safety, safeguards, and environmental factors also enter in this choice.

1. Independent of presumably solvable configurational and safeguard, environmental, and safety (SES) constraints, what blanket designs would be chosen for:
   - specific product mixes?
   - specific fuel cycles?
   - optimization of energy multiplication?
   - optimization of neutron multiplication?
2. What fuel cycles appear attractive? Why?
3. Are some blanket compositions, fuel cycles, and reactor configurations more desirable than others?
4. What are the interfacial constraints for <u>both</u> the fusion plasma and the balance of fusion thermonuclear requirements (e.g., s/c magnets) for each choice of configuration leading to designs satisfying Question 1.
5. What penalties (or benefits) would accrue for designs that would minimize activation and SES problems?
6. What R&D steps are necessary to determine the technical feasibility of fusion-fission energy systems?

C. Economic -- Is there a need for fusion-fission energy systems, especially those designed for the production of fissile fuel. What is the effect on reactor economics of power production as a by-product (or vice versa).

1. Assess the role of fusion-fission energy systems in different supply/demand forecasts. Should such systems be designed to produce fissile materials, or electric power, or both? If the choice is fissile materials, could the waste heat from such systems be utilized for such purposes as synthetic fuel production or similar by-products? If the choice is electric power, does the by-product of fissile materials mean anything? Is there an economic incentive to use fusion-fission systems for fissile waste product and actinide disposal?

2. If both physics and engineering constraints prescribe specific product mixes, (fissile fuel versus power) how best can these limitations be incorporated into the energy supply projections of the nation? (e.g., given specified product mix, define the optimal energy supply configuration).

3. Given a specific benefit-cost ratio, implementation schedule, product mix, and R&D costs, what is an acceptable capital and operating cost window for fusion-fission systems to be competitive with alternative energy supply systems? What is the sensitivity of this window to different product mixes? What is the sensitivity of this window to different product <u>prices</u>? How should these analyses affect construction plans of utilities seeking to implement fission electric power?

4. What is the sensitivity of the aforementioned simulations to:

- expanded successful uranium discovery rate at acceptable dollar, environmental, and social costs
- massive implementation of coal fired electric plant and conversion plant
- successful development of Laser Isotope separation techniques (at, perhaps, high efficiency low energy cost operation).
- world energy prices controlled by a cartel always set to undercut options.

5. To what extent could fusion-fission energy systems affect national and world economics?

D. Safety, Environment, & Safeguards -- Fusion-fission energy systems will share many of the commonly perceived problems of fission reactors.

1. What are the potential SES problems of fusion-fission energy systems?
2. To what extent can these problems be compared with both pure fusion and pure fission (all types)?

3. Is there something unique about fusion-fission energy systems that gives rise to either greater or lesser potential problems in comparison to the fission option?

4. Can fusion-fission energy systems be designed to minimize the number and extent of potential problems in comparison with pure fission?

   a. If so, how? Treat Safety, Environment, and Safeguards as separate classes.

   b. If so, how would such measures compromise plant economics, fusion physics, and reactor engineering?

Conceptual Study Plan for Fusion-Fission Energy Systems

FY	Activity
75	1. Perform a preliminary assessment of the critical questions of Economics, Fusion Physics, Reactor Engineering, and Safety, Environment and Safeguards, Specifically --
	a. Estimated Role of F-F energy systems
	b. Desired technical performance characteristics of Fusion Physics and Reactor Engineering
	c. Potential SES problem areas needing early resolution.
	d. Major uncertainties in a, b, & c.
	2. Identify high priority fusion and fission subsystems
	a. Establish criteria for ranking each subsystem type
	b. Develop data for each alternative subsystem type to the extent that a common data base exists for all
	c. Devise a ranking scheme to order both fusion and fission subelements in terms of priority. Note, ranking of fusion and fission subelements cannot be performed independently.
	d. Rank fusion-fission concepts.
	3. Prepare a formal study/development plan based on results of Activities 1 & 2.

FY	Activity
76	1. Perform scoping and definitional studies (point designs) of the most promising fusion-fission energy systems. The number of different systems and the level of effort for each will be determined by funding availability.
	2. Initiate studies to formally consider Economic and Safety, Environment, and Safeguard problem areas. The results of these efforts will be used to measure the designs prepared during activity 1.
	3. Perform supportive studies that are essential to but not within the scope of activity 1. Typical efforts could include but are not limited to: • Studies on optimization of the fusion subsystem as an element in a fusion-fission energy system. • Expanded studies in conceptual blankets for future reactor design efforts. • Expansion of the nuclear data files to include fusion-fission requirements. • Characterization of the processes for fuel cycles and production methods. • Characterization of structural alternatives with maintainability being a key factor.

FY	Activity
76 (Cont.)	4. Evaluate the technical status of the point design(s) resulting from scoping and definitional studies. a. Identify essential subsystems for which unacceptable uncertainties exist.
77	1. Initiate preliminary conceptual design studies of fusion-fission energy systems. The selection of these systems will be based on a) Technical maturity of concept b) Economic promise of concept c) Concept adherance to acceptable safety, environment, and safeguard standards. 2. Initiate studies on those essential subsystems that have been found to be incompletely characterized. 3. Continue supportive studies at a level commensurate with program needs. 4. Review status of program with respect to the results of the preliminary conceptual designs, the status of essential subelements, and the economic prospects of fusion-fission energy systems. Decide to proceeed if program status is satisfactory.

FY	Activity
78	1. Initiate conceptual design(s) for fusion-fission energy system(s). Perform the necessary and sufficient analysis to justify construction of a facility should such a decision be made.
79	1. Evaluate conceptual design(s). Decision to construct facility if indicated.

AGENDA

FUSION-FISSION ENERGY SYSTEMS MEETING AT AEC HEADQUARTERS - DECEMBER 3 and 4, 1974

Tuesday	3 December 74
9:00 a.m.	Jim Williams and Locke Bogart (DCTR) -- Opening remarks, Guidelines for presentations and questions will be proposed as well as the desired major objectives of the meeting.
9:10 a.m.	Raymond Huse with James Burger (Public Service Electric & Gas) and Mike Lotker (Northeast Utilities) -- The perspective of Electric Utilities will be presented with particular emphasis on economic and implementation aspects. Utility interest in fusion-fission systems will be explained with respect to alternative electric energy options.
10:00 a.m.	Lawrence Lidsky (Massachusetts Institute of Technology)-- A systems overview of fusion-fission energy systems will be presented. Primary design criteria will be identified for different choices of the product mix or application.
10:50 a.m.	Duane Deonigi with William Wolkenhauer (Battelle Pacific Northwest Laboratory) -- The results of a recent cost-benefit analysis for fusion-fission energy systems will be presented. The plutonium cycle was used because of well-developed baseline data. System cost windows will be defined in terms of product mix, implementation date, and forecast competitive marketplace. The thorium cycle will be discussed briefly in terms of data requirements.
11:40 a.m.	Ronald Liikala with Bowen Leonard (Battelle Pacific Northwest Laboratory) -- BNWL's cumulative efforts in fusion-fission energy systems and an assessment of its future. Engineering requirements will be emphasized.
12:30 - 1:15 p.m.	Lunch
1:30 p.m.	Ralph Moir with J. D. Lee (Lawrence Livermore Laboratory) -- LLL's cumulative history of fusion-fission energy systems with particular emphasis on point design experience.
2:20 p.m.	Daniel Jassby (Princeton Plasma Physics Laboratory) -- Optimization of beam driven tokamaks for high flux neutron production and an assessment of the prospects of this concept.

3:10 p.m.	Robert Krakowski with Donald Dudziak (Los Alamos Scientific Laboratory) -- LASL's recent experience in fusion-fission energy systems. Specific emphasis to be placed on the fusion-fission interface and economies of scale.
4:00 p.m.	Five minute summaries of each presentation followed by open discussion.
<u>Wednesday</u>	<u>4 December 74</u>
9:00 a.m.	Workship Meeting - Discussion to initially focus on the maturity of fusion physics and reactor engineering. Questions of economics (both costs and sales), environment, safety, and safeguards will be examined. Recommendations for studies will be solicited for consideration.
<u>Location</u>	The meetings on both days are scheduled to be held in the Commissioners' Conference Room (A410).

ATTENDANCE -- FUSION FISSION MEETING -- DECEMBER 3, 1974

	Name	Organization	Telephone
1.	Locke Bogart	AEC/DCTR	301-973-3735
2.	Raymond Watts	PNL Engineering Tech.	509-946-2728
3.	Thurman Frank	LASL	505-667-5003
4.	Donald Dudziak	LASL	505-667-6336
5.	Joseph D. Lee	LLL	415-447-1100 ext: 7215
6.	Don Steiner	ORNL	615-483-7995
7.	Larry Killion	HQS AEC - List/DMA	301-973-3345
8.	Michael Lotker	Northeast Utilities	203-666-6911
9.	James Burger	Public Service Electric Gas Company	201-622-7000 ext: 2251
10.	Jim Maniscalco	LLL	415-447-1100 ext: 4272
11.	Robert Bingham	AEC/DCTR	301-973-4097
12.	Michael L. Miller	University of Texas	512-471-3542
13.	Steve Nichols	University of Texas	512-471-5136 or 3542
14.	Franklin E. Coffman	AEC/DCTR	301-973-3734
15.	J. Nelson Grace	AEC/DCTR	301-973-4594
16.	C. W. Maynard	University of Wisconsin	608-263-3285
17.	Charles C. Baker	General Atomic Company	714-455-2831
18.	Daniel Klein	Westinghouse	609-452-5664
19.	G. A. Graves	NSF/OEP (LASL)	202-632-7812
20.	Meyer Steinberg	BNL, Upton, New York	516-345-3036
21.	H. P. Furth	PPPL	609-452-5603
22.	Steve Dean	AEC/DCTR	301-973-4095

	Name	Organization	Telephone
23.	T. K. Fowler	LLL	415-447-1100 ext: 7501
24.	R. F. Post	LLL	415-447-1100 ext: 7153
25.	A. M. Sleeper	AEC/DCTR	301-973-3421
26.	M. M. Cohen	AEC/DCTR	301-973-5514
27.	L. K. Price	AEC/DCTR	301-973-5143
28.	F. R. Scott	AEC/DCTR	301-973-3563
29.	C. E. Taylor	LLL	415-447-1100 ext: 715
30.	J. P. Holdren	LLL/Univ. C. Berkeley	415-642-1139
31.	E. J. Ziurys	AEC/DCTR	301-973-5144
32.	G. W. Kuswa	AEC/List	301-973-3397
33.	M. Murphy	AEC/DCTR	301-973-3304
34.	M. Grunspan	AEC/OPA	301-973-5223
35.	K. G. Moses	AEC/DCTR	301-973-3563
36.	J. D. Hunsuck	AEC/List	301-973-3397
37.	I. Clark	U. S. Congress Staff	201-225-2965
38.	E. Oktay	AEC/DCTR	301-973-3563
39.	G. J. Mischke	AEC/DCTR	301-973-3734
40.	G. H. Miley	Univ. of Illinois	217-333-3772
41.	R. P. Rose	Westinghouse Electric Corp.	609-452-5756
42.	B. R. Leonard, Jr.	Battelle Northwest	509-946-2558
43.	E. Linn Draper, Jr.	Univ. of Texas at Austin	512-471-5136
44.	T. A. Parish	Univ. of Texas at Austin	512-471-5736

	Name	Organization	Telephone
45.	R. M. Rosselli	AEC/DCTR	301-973-5543
46.	J. H. Vanston	Univ. of Texas at Austin	512-471-4049
47.	L. J. Lidofsky	Columbia University	212-280-4460
48.	R. T. Taussig	" "	212-280-2961
49.	M. A. Abdou	Argonne National Lab.	312-739-4098
50.	J. M. Williams	DCTR/USAEC	301-973-3734
51.	W. C. Wolkenhauer	Battelle Northwest Lab.	509-946-2470
52.	A. Hertzberg	Univ. of Washington	206-543-6321
53.	P. H. Rose	Math. Sciences N.W.	206-632-1046
54.	G. M. Halpern	Exxon Res.	201-474-2010
55.	S. D. Putnam	Physics International Co.	415-357-4610
56.	J. H. Huntington	Physics International	415-357-4610
57.	L. M. Lidsky	MIT	617-253-3808
58.	R. A. Huse	P.S.E. & G.	201-622-7000
59.	G. K. Hess, Jr.	DCTR/USAEC	301-973-5378
60.	Lowell Wood	U.C.L.L.L.	415-447-1100 ext: 3444
61.	James Powell	Brookhaven National Lab.	516-345-2440
62.	Howard Greyber	Consultant Potomac, Md.	301-299-8168
63.	Duane Deonigi	Battelle Northwest	509-946-2475
64.	Ron Liikala	Battelle Northwest	509-946-2627

ATTENDANCE -- FUSION--FUSION ENERGY MEETING -- DECEMBER 4, 1974

	Name	Institution	Telephone
1.	Locke Bogart	AEC/DCTR	301-973-3735
2.	R. P. Rose	Westinghouse	609-452-5756
3.	Don Beard	AEC/DCTR	301-973-5143
4.	Donald Dudziak	LASL	505-667-6336
5.	Bob Krakowski	LASL	505-667-7001
6.	Charles C. Baker	General Atomic	714-455-2831
7.	Charles W. Maynard	Univ. of Wisconsin	608-263-3285
8.	James R. Powell	Brookhaven Lab.	516-345-2440
9.	John H. Huntington	Physics Internat'l.	415-357-4610
10.	B. R. Leonard	Battelle	509-946-2558
11.	W. C. Wolkenhauer	"	509-946-2470
12.	R. C. Liikala	"	509-946-2627
13.	M. A. Abdou	ANL	312-739-4098
14.	G. M. Halpern	Exxon Res. & Eng.	201-474-2010
15.	F. R. Scott	AEC/DCTR	301-973-3563
16.	Franklin E. Coffman	AEC/DCTR	301-973-3734
17.	James M. Williams	AEC/DCTR	"
18.	Steve Nichols	Univ. of Texas	512-471-5136 3542
19.	T. A. Parish	Univ. of Texas	512-471-5136
20.	R. W. Moir	LLL	415-447-1100 ext: 715[illegible]
21.	J. D. Lee	LLL	415-447-1100 ext: 72[illegible]

	Name	Institution	Telephone
22.	R. L. Watts	PNL	509-946-2728
23.	Michael Lotker	Northeast Utilities	203-666-6911
24.	L. M. Lidsky	MIT	617-969-9727
25.	T. G. Frank	LASL	505-667-5003
26.	James Burger	PSE&G	201-622-7000 ext: 2251
27.	Duane Deonigi	PNL	509-946-2475
28.	D. L. Jassby	PPPL	609-452-5661
29.	Don Steiner	ORNL	615-483-7995
30.	G. H. Miley	Univ. of Illinois	217-333-3772
31.	Clyde Taylor	LLL	415-447-1100 ext: 7151
32.	Jim Maniscalco	LLL	415-447-1100 ext: 7231
33.	Michael Murphy	DCTR	301-973-3304

FUSION-FISSION ENERGY SYSTEMS
SOME UTILITY PERSPECTIVES

by

Raymond A. Huse
Manager - Research and Development
Public Service Electric and Gas Company
Newark, New Jersey

James M. Burger
Associate Research Physicist
Public Service Electric and Gas Company
Newark, New Jersey

Michael Lotker
Associate Scientist
Advanced Energy Conversion Research
Northeast Utilities
Hartford, Connecticut

paper presented at

United States Atomic Energy Commission Meeting

on

Fusion Driven Fission Systems
Germantown, Maryland
December 3, 4, 1974

I INTRODUCTION

While fusion researchers point to a commercialization date by the year 2000 for thermonuclear reactors, the subsequent introduction of these devices on utility systems will be governed by incentives of reliability, economy, safety, and environmental compatibility demonstrated in earlier research phases. First generation fusion reactors, subject to all the uncertainties of any new energy conversion technology along with the extraordinary challenges of the D-T fuel cycle, exotic structural material requirements, superconducting magnet and/or laser technology, may well be less attractive than fission or coal fired generation options in this time frame.

The principal attraction of fusion-fission energy systems lies in the potential for relaxing some of the engineering requirements of pure fusion systems. Although they may advance the date of commercial demonstration experiments by only some five to ten years, the more limited extension of state-of-the-art required and the enhanced reliance on proven technology may accelerate the utilization of the hybrids by utilities. Furthermore, these systems offer the utilities an interim alternative which should smooth the transition from fission to fusion dominated generation of electricity with a minimum of risk. During this interim period, which may be as long as 50-100 years or more, fusion-fission hybrids can effectively contribute to the solution of the pressing fuel availability and waste disposal problems of the fission economy. If studies currently underway and shortly to begin support the technical and economic promise of these systems' enhanced relevance to our energy future, an increased priority for fusion research should result.

Below we discuss some of the issues that will be important in assessing fusion-fission energy systems from a utility perspective. We begin by making a number of qualitative systems-oriented observations and then attempt to give some economic quantification of the benefits from fusion-fission hybrids and their allowed capital cost.

II ADVANTAGES OF A FUSION-FISSION OPTION

A preliminary economic analysis such as the one presented below suggests that there are strong incentives to make use of the large amount of heat liberated in a fusion-fission hybrid. However, a closer analysis indicates several advantages for a device that breeds fissile fuel and/or burns fission waste products without generating electricity for external sales. In this regard, breeding U-233 from thorium appears preferable to plutonium production from U-238 due to the lower rates of heat generation in the former case and the consequently reduced penalty for not utilizing heat. In any final commercial decisions, economic considerations will be of major importance, but these other issues could also be of substantial significance.

In the following, some of these issues are presented with special emphasis on non-electrical fusion-fission systems.

1. Perhaps the most obvious advantage for a basically non-electrical fusion-fission device results from eliminating the constraint of instantaneous reliability. While total on-line hours per year will still be critical for economic feasibility, the impact of a momentary failure, potentially quite serious on an electrical system, would, for this hybrid, be negligible. Such decoupling from electrical power production supports the practicality of very large sized units that seem to be indicated for many fusion systems, because it eliminates the need for a large amount of back-up generation (spinning reserve) which otherwise would be required. The pulsed nature of the Tokamak causes no problems for this concept(the UWMAK-1 has a burn time of 90 minutes followed by a recharge time of 6½ minutes). Similarly foregoing electricity production may reduce the reliability constraints upon the laser in laser fusion concepts, as impact of missing a few shots every now and again would be minimal. Reduced reliability requirements should lower costs in numerous other areas.

2. Optimization of breeding at the expense of electrical production has advantages from the standpoint of handling the heat produced. In particular:

   (a) if one is not concerned with surrounding the fission blanket with a superconducting magnet, such as in a laser system, the required power density can be made low enough to reduce safety concerns, e.g., a loss of coolant accident.

   (b) Since the principal product is fuel rather than electricity, thermodynamic efficiency has relatively less importance. Therefore a power density and thus a blanket temperature could be selected in either a laser or magnetic confinement system that would permit rejection of heat at temperatures high enough to make dry cooling practical. The independence of a need for cooling water with dry cooling would result in additional flexibility in siting and licensing.

3. There is the strong possibility that a separate business entity would own and operate a fuel producing device. This would be significant in that the price for the fissile fuel produced, or waste products disposed of, would probably not be set by a state regulatory body, although federal regulation as in the case of oil and gas today might still occur. The

company would have a market as broad as the entire fission
economy (i.e., worldwide) rather than one strongly tied to
a narrow operating area. The business operations would
thus be optimized for fuel production and/or waste disposal
leaving the utility to concentrate on electricity production.

4. The non-electrical option allows the utilities to optimize
for the generation of electricity with the (by then) mature
LWR and HTGR nuclear technologies. Increased reliability
and efficiency in utility operations would be major results.
The approach is the one which changes the electrical
utility business least, an advantage for any technology
trying to gain utilization and a strong reason to suspect
that the time scale of introduction (after commercial
demonstration) will be rapid. This, combined with the
shorter research program to demonstration that will be
probable with a simpler device, means that this concept can
be considered to have mid-term (before 2000) relevance.
The case for the high priority of fusion research should
therefore be strengthened even further.

5. The potential for a device to transmute certain fission
waste products, as well as breed fuel, has psychological,
sociological, and philosophical significance. It offers
a way to lessen the burden on future generations to guard

radioactive wastes for thousands of years, requiring potentially unrealistic assumptions about the longevity of our societal institutions. While recycle of radioactive wastes (principally the actinides which are the longest lived) has been discussed in the case of breeder reactors, such fusion-fission devices with their harder neutron spectrum offer some fundamental advantages. The promise of such transmutation, we believe, could have an immediate impact on the current nuclear energy debate.

6. This concept fits well into the nuclear park arrangement which might consist of several fission reactors, a fusion breeder and waste burner, a fuel reprocessing plant, and a short-term storage facility. The safety and security implications are obvious.

7. A device that does not produce electricity for electrical sales may generate power for on-site consumption (i.e. for neutral beams, lasers, etc.). It may therefore be possible to make use of some of the thermal energy generated without sacrificing the above advantages of a non-electrical fusion-fission breeder. The final design configuration will be determined by system optimizations beyond the scope of this paper.

III FISSILE FUEL SAVINGS - R&D IMPLICATIONS

From a national priorities viewpoint, a fuel producer addresses a more critical problem than electrical power production, namely that of fuel supply. In fact, the motivation for developing the LMFBR is fissile fuel supply in future years, not electrical capacity.

To provide some indication of the need for breeding in a nuclear economy, two scenarious were compared for a projection of nuclear power growth to 2020 AD.[1] The nuclear reactor requirements were assumed met either with LWR's or the mix of converters and breeders shown in Figure 1[1]. The breeders shown could either be commercial LMFBR's and advanced fast breeders, or the advanced breeders could be fusion-fission energy systems. The illustrative discussion below is carried out in terms of plutonium bred from U-238 but the production of U-233 from thorium is an alternative option. It should be emphasized that the plutonium or U-233 production characteristics of fusion-fission hybrids and fast breeders may be substantially different.

A comparison of the mined uranium and enrichment requirements for these two scenarios (all LWR's,or converters and breeders) provides a measure of the economic incentive for breeders. The breeder scenario indicates a strongly advancing breeder technology and hence provides a reasonable basis to assess the maximal fuel cycle savings with breeder development. A less rapid rate of breeder introduction appears more likely.

LWR's are seen to be phased out after 2010 and all new capacity thereafter is provided by commercial and advanced breeders. HTGR's are seen to decline to a significant but modest level. The analysis is terminated arbitrarily at 2020 AD.

Figure 2 shows the U_3O_8 (yellowcake) requirement if the nuclear capacity forecast shown in Figure 1 were met with light water reactors (LWR's) employing plutonium recycle or with the mix of converters and breeders shown.[2] In the latter case the initial plutonium requirements for the breeders are met with that produced in LWR's; no plutonium recycle in LWR's is assumed although plutonium would be available for the assumed breeder capacity. The depleted uranium for breeders would be available from stockpiled enrichment plant tailings. The U_3O_8 requirements shown in this time frame thus arise from the predicted LWR and HTGR capacities forecast and are not influenced by breeder requirements. Higher breeder performance permits more plutonium to be recycled, however, and reduces U_3O_8 requirements.

The uranium reserves are shown in Figure 3 as a function of the ore concentration levels and the corresponding price as estimated by the AEC.[1,3] The ore costs at a given concentration are less than those currently being paid today (by a factor of 2) but they do provide an indication of ore costs.

The figures shown are thus conservative. There is considerable uncertainty regarding uranium resources and more uranium may exist at lower prices.[4] Figure 4 shows the cumulative ore savings based on the ore prices of Figure 3 without escalation. While the uncertainties in ore cost and nuclear capacity make any detailed economic analysis uncertain, this capacity and ore cost forecast indicates a cumulative savings on ore alone of almost $300 billion between the years 2000-2020.

The use of LWR's, in contrast to breeders, requires enrichment and the amount of enrichment (in separative work units) for the two reactor scenarios (LWR's only, and breeders and converters) are shown in Figure 5.[2] (A separative work unit is required to make 1 kg of uranium with twice the U^{235} enrichment of natural uranium.) Corresponding savings for enrichment are shown in Figure 6 and indicate cumulative savings through 2020 of $75 billion. Enrichment costs are based on the current AEC charge of $47.80 per separative work unit, which will certainly increase in the future.

The savings calculated by comparing these two scenarios can be translated into today's dollars, i.e., their present worth can be calculated. This provides an estimate of the maximum cost of a fusion-fission hybrid research program (in 1974 dollars) that one can justify on the basis of anticipated fissile fuel cost and enrichment savings. To compute this present worth, one calculates the sum of $(1+i)^{-n}$ times the annual expense where n is the number of years after the reference year, e.g. 1974. For purposes here, let us make the rough assumption that the cumulative savings are all made in the year 2015 AD (c.f., Figures 4,6) and the interest rate (i) is 10%; then the present value of these savings, say in 1974, would be $(300 \times 10^9 + 75 \times 10^9)/(1+0.10)^{41}$ = \$8 billion.

The foregoing two scenarios thus suggest that the ore and enrichment savings from a breeding capability (through 2020) could have a value today on the order of \$8 billion. This could be taken as an upper limit on what the U.S. should be willing to spend for R&D today in this area provided that breeders cost no more per electrical kilowatt than converters. If breeders have higher capital costs than converters or are not electricity producers, the R&D expenditures justified in the above way would be reduced.

The research program would be spread out over a number of years, however, for example, from now until 2000 AD, when (c.f. Figure 1) advanced breeders are indicated as commercial. For this period, the \$8 billion present worth is equivalent to a \$900 million annual R&D expenditure for breeders.

Other portions of the fuel cycle will also have somewhat differing costs depending on whether LWR's continue to be used (through 2020) or whether breeders are introduced. Plutonium is more costly to fabricate than uranium but even with only LWR's, plutonium will still be recycled, entailing extensive plutonium fabrication.

IV PLANT ECONOMICS

Clearly, the cost of a production plant must be directly related to minimum cost for which its product can be sold to break even. The cost of the plant is composed of two portions, namely the initial capital cost, and the operating and maintenance cost, which may be quoted on an annual basis. In the case of nuclear fission or fusion plants, the capital cost dominates and we will confine ourselves to considering this major cost component.

For a non-electrical breeder the minimum allowable price of the fuel produced (dollars per gram) is simply related to the unit capital cost of the plant (dollars per gram per year) through a percentage of the capital cost called the carrying charge. This carrying charge is determined as the percentage of the capital cost that must be earned annually to pay for the plant over its lifetime; the principal components are interest, depreciation, and Federal and local taxes. This relationship is given in Appendix A by setting $e = o$ in the equation.

The relationship for a 15% carrying charge is illustrated in Figure 7. For example, if plutonium (or U-233) can be sold for $30 per gram, the maximum allowed capital cost for the plant is $150 per gram per year (see * on Figure 7). For a capacity of 1½ tonnes/year of plutonium, as has been discussed,[5] the allowed plant cost would be $225 million.

If the plant produces electricity as well as nuclear fuel, revenue is derived from both products. In this case, a question arises as to how much of the costs should be allocated to each of the products. Furthermore, the capital cost is not accurately reflected by the capital cost per unit of one of the products, i.e. by $/gram/year as presented in Figure 7.

To illustrate the interrelationship of the two products, the cost to produce a unit quantity of plutonium (or U-233)* can be related to total capital cost with electrical revenues as a parameter. In Figure 8 both plutonium production without electricity generation and with 1000 MWe at 75% capacity factor in an electrical breeder are illustrated. For the cases of electrical power generation, electrical power credits of 10 and

*This discussion and Figure 8 are given in terms of plutonium. The same discussion and numerical values are also true for U-233 produced from thorium.

20 mills/kWh were taken. The annual revenue requirements were computed from the capital cost using a carrying charge of 15% as above. The equation for the curves is given in Appendix A.

Two plutonium production capacities are illustrated. The lower annual plutonium production rate is the same as that for a 1000 MWe LMFBR (advanced oxide) and the larger for a production rate 10 times as great, which appears readily achievable in a fusion-fission breeder. A capital cost of $500 million would correspond to a 1000 MWe electrical power-plant unit cost of $500 per KWe, which is roughly what is projected for the LMFBR. The value discussed with regard to no electrical production and shown by a * in Figure 7, is similarly indicated on Figure 8.

The curves indicate that, as expected, the unit cost of plutonium is reduced by the concurrent generation of electricity for a given plant capital cost. Increasing plutonium production capacity also reduces the production cost of plutonium.

2008 value enriched uranium ≃ $1400/gm 2008 value ≃ $4000/gm

The current value of plutonium is about $10 per gram based on the cost of the equivalent reactor fuel as uranium. Assuming a future fissile fuel market value of $30 per gram, the following allowed capital costs for the cases considered obtain (see Figure 8 and Appendix A).

TABLE 1

BREAKEVEN PLANT COST
(MILLIONS OF DOLLARS)

	Plutonium Production Capacity (kilograms per year)		
	LMFBR	HYBRID	
	150	1500	3000
Electrical Power Cost (mills per kWh)			
0	22	225	450
10*	460	660	890
20*	900	1100	1330

For these electricity and fuel production capacities and prices, the effect of electricity product and credit substantially alters the allowed capital cost for the plant. An electrical capacity of 1000 MWe corresponds to the largest electrical powerplants being built today; clearly for less electrical power production, the effect is also less.

With little or no electrical production, as in the first case, the allowed capital cost is sensitive to plutonium production capacity and plutonium price. However, for 1000 MWe of base-load output, a major portion of the revenue requirements are met through electrical sales so the effect of plutonium production capacity on breakeven capital cost is less (c.f. Table 1).

*1000 MW of electrical generation at 75% capacity factor is assumed.

One can also express the breakeven capital cost of the plant on the basis of unit thermal (or electrical output) and Figure 9 shows this plotted as a function of the ratio of fissile fuel to heat production, γ. Curves (see Appendix A) are plotted for no electricity production and 1000 MWe generation of electricity (75% capacity factor); the electricity is shown at 10 and 20 mills/kWh, and fissile fuel, e.g. plutonium, is taken at \$30 per gram.

The vertical lines indicate several values of γ discussed in the literature.[5,6,7] The values of γ shown as Reference 7 are for a lithium and U-238 blanket (larger γ), and a lithium, U-238 and 4% plutonium blanket (smaller γ). The capital cost for a beam driven tokamak reactor (TCT) has been estimated to be on the order of \$800 million. For power levels with these two blankets in this reactor of 6000 MW thermal (lower γ) and 2000 MW thermal (higher γ), the unit capital costs would be \$130 per kW thermal and \$400 per kW thermal, respectively.[8] The corresponding production costs for electrical power can be seen from Figure 9 to be modest, indicating that such a system appears to have economic potential.

The question can also be raised as to the allowed capital cost of a fusion-fission plant burning up radioactive wastes (e.g. actinides) either with or without electrical generation. Figures 7 through 9 can be directly applied to this case if the words "plutonium production" are replaced with "radioactive waste consumption", i.e. grams (or kilograms) are interpreted as radioactive wastes consumed. For example, with reference to Figures 7 and 8 or Table 1, consider a plant producing no electricity for external sales and burning radioactive wastes for a fee comparable to the price of fissile fuel. Then for a 1500 kilogram per year waste burning capacity at a fee of $30 per gram of waste, the allowed plant cost would be $225 million. The economics of this application will clearly be determined by the methods and the associated price required by society for the disposal of radioactive wastes. The possibility of significant heat production from the burning of actinides in a fusion-fission system[9] would appear to favorably affect the economics of such an approach through the realization of a significant credit for heat or electricity.

V CONCLUSION

The above discussion indicates a number of systems incentives for the development of fusion-fission energy systems. In addition, a consideration of fissile resource and enrichment requirements for an expanding nuclear economy suggests substantial funds, on the order of billions of 1974 dollars, can be justified over the next half century for the introduction of high performance breeders. The allowed capital cost of the plant will depend strongly on fissile fuel and useful heat production rates and the competitive market price of these products, as well as any credits for waste disposal.

References

1. "Report of the Cornell Workshops on the Major Issues of a National Energy Research and Development Program", College of Engineering, Cornell University, December 1973.

2. "Nuclear Power Growth 1974-2000", United States Atomic Energy Commission, WASH-1139(74), February 1974.

3. "Nuclear Fuel Supply", United States Atomic Energy Commission, WASH-1242, May 1973; "Nuclear Fuel Resources and Requirements", United States Atomic Energy Commission, WASH-1243, April 1973.

4. "Report on Uranium Resources to Meet Long Term Uranium Requirements", Electric Power Research Institute, 1974.

5. R. N. Horoshko, H. Hurwitz, H. Zmora, "Application of Laser Fusion to the Production of Fissile Materials", Annals of Nuclear Science and Engineering, Vol. 1, pp. 223-232, 1974.

6. J. A. Maniscalco and L. L. Wood, "Advanced Concepts in Fusion-Fission Hybrid Reactors", presented at the A.N.S. Winter Meeting, Washington, D.C., October 1974.

7. J. D. Lee, UCRL-73952, June 1972; D. Jassby, private communication.

8. D. Jassby, private communication.

9. J. A. Maniscalco, private communication.

Appendix A

The relationship of product capacities and unit revenue requirements to plant capital cost (neglecting O&M) for the cases of plutonium and electricity production is as follows.

For 15% carrying charges, the annual cost to produce the two products (plutonium and electricity) is 0.15C. If this is set equal to revenues,

$$0.15C = n \cdot P_{th} \cdot 8760g \cdot 10^{-3} \cdot e + P_{pu} \cdot g \cdot f$$

where:

- C - capital cost in \$ (15% carrying charges assumed)
- n - thermal cycle efficiency (1/3 assumed)
- P_{th} - plant thermal output in kW
- g - capacity factor, i.e. percentage of the year (8760 hours) that the plant operates (75% assumed)
- e - electricity price in mills/kWh
- P_{pu} - plutonium production capacity in grams/year
- f - price of plutonium in \$/gram

This equation can be rewritten as

$$f = (0.15C - 6.6P_e e)/0.75P_{pu} = F_{P_e,P_{pu},e}(C)$$

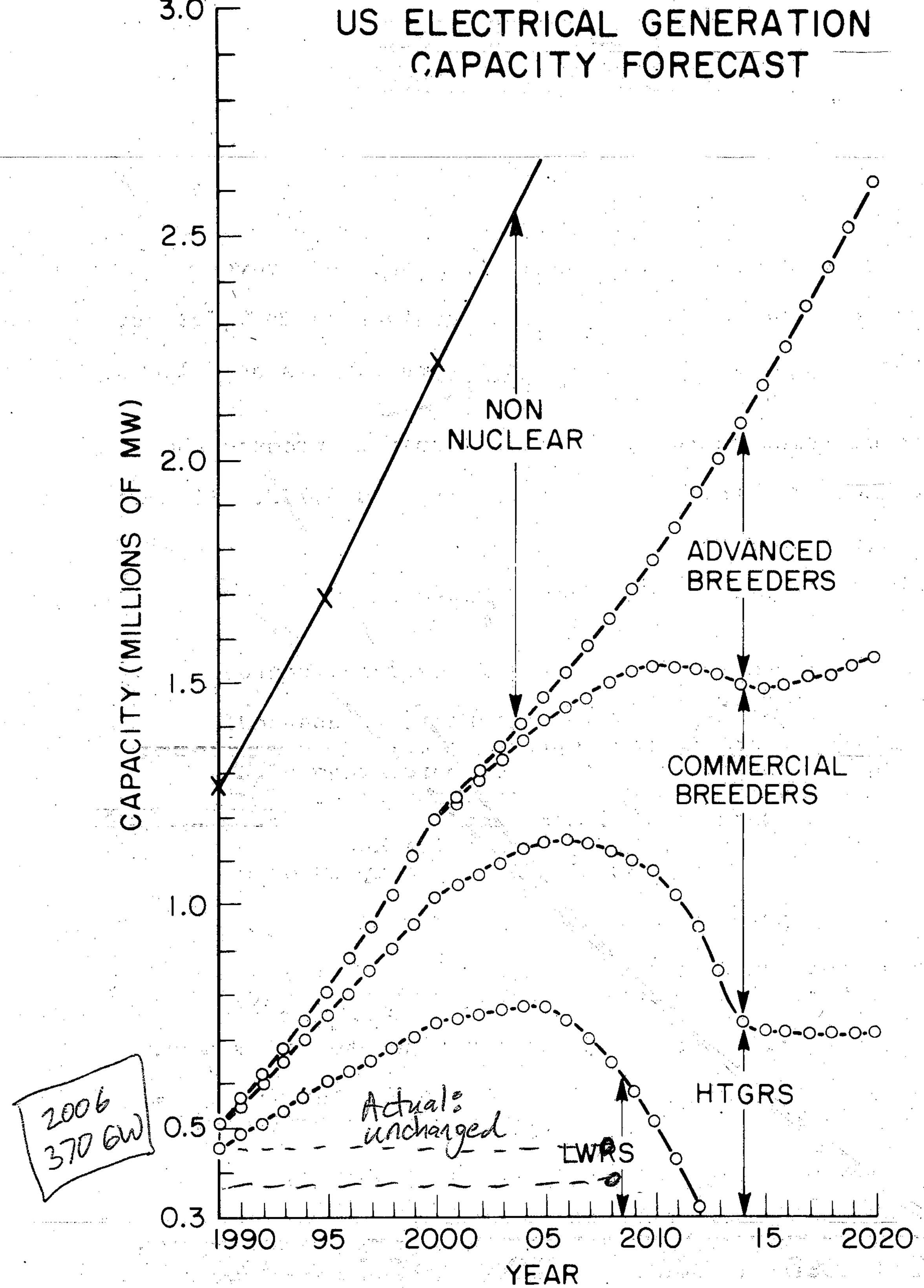
Where $P_e = nP_{th}$ kilowatts and subscripts denote parameters.

This curve is plotted as Figure 8 for $P_e = 0$, 10^6 KWe; $P_{pu} = 0.15 \times 10^6$ g/yr, 1.5×10^6 g/yr; e = 10, 20 mills/kWh

Alternatively, the equation can be rewritten as

$$(C/P_{th}) = 15e + 4.4 \times 10^4 \gamma f = G_{ef}(\gamma)$$

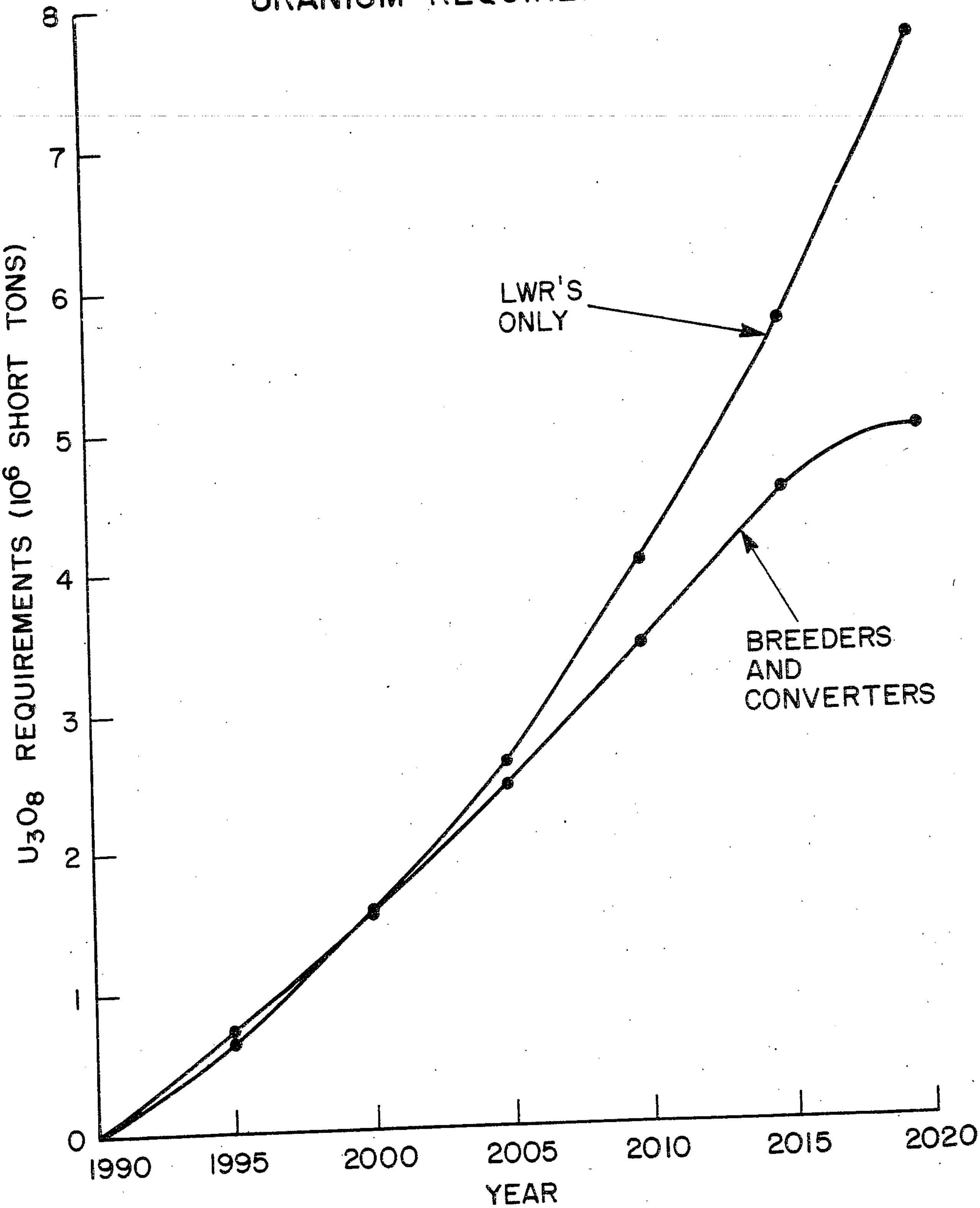
Where $\gamma = P_{pu}/8760P_{th}$ grams/kWh. This curve is plotted as Figure 9 for e=10, 20 mills/kWh and f=30 \$/gram.



ESTIMATED FUTURE NUCLEAR GENERATING CAPACITY, IN MILLIONS OF MEGAWATTS (ELECTRIC). AEC FIGURES UNTIL 2000; AFTER THIS, THE TOTAL ELECTRIC GENERATING CAPACITY IS ASSUMED TO DOUBLE EVERY 25 YEARS. ALL NEW CAPACITY IS ASSUMED TO BE NUCLEAR AFTER 2010. CONTRIBUTION OF VARIOUS TYPES OF REACTORS IS SHOWN SEPARATELY

FIGURE 1

CUMULATIVE MINED
URANIUM REQUIREMENTS



NOTE: CUMULATIVE U_3O_8 REQUIREMENTS TO YEAR 1990 ARE 0.7×10^6 SHORT TONS

FIGURE 2

REASONABLY ASSURED PLUS ESTIMATED URANIUM RESERVES

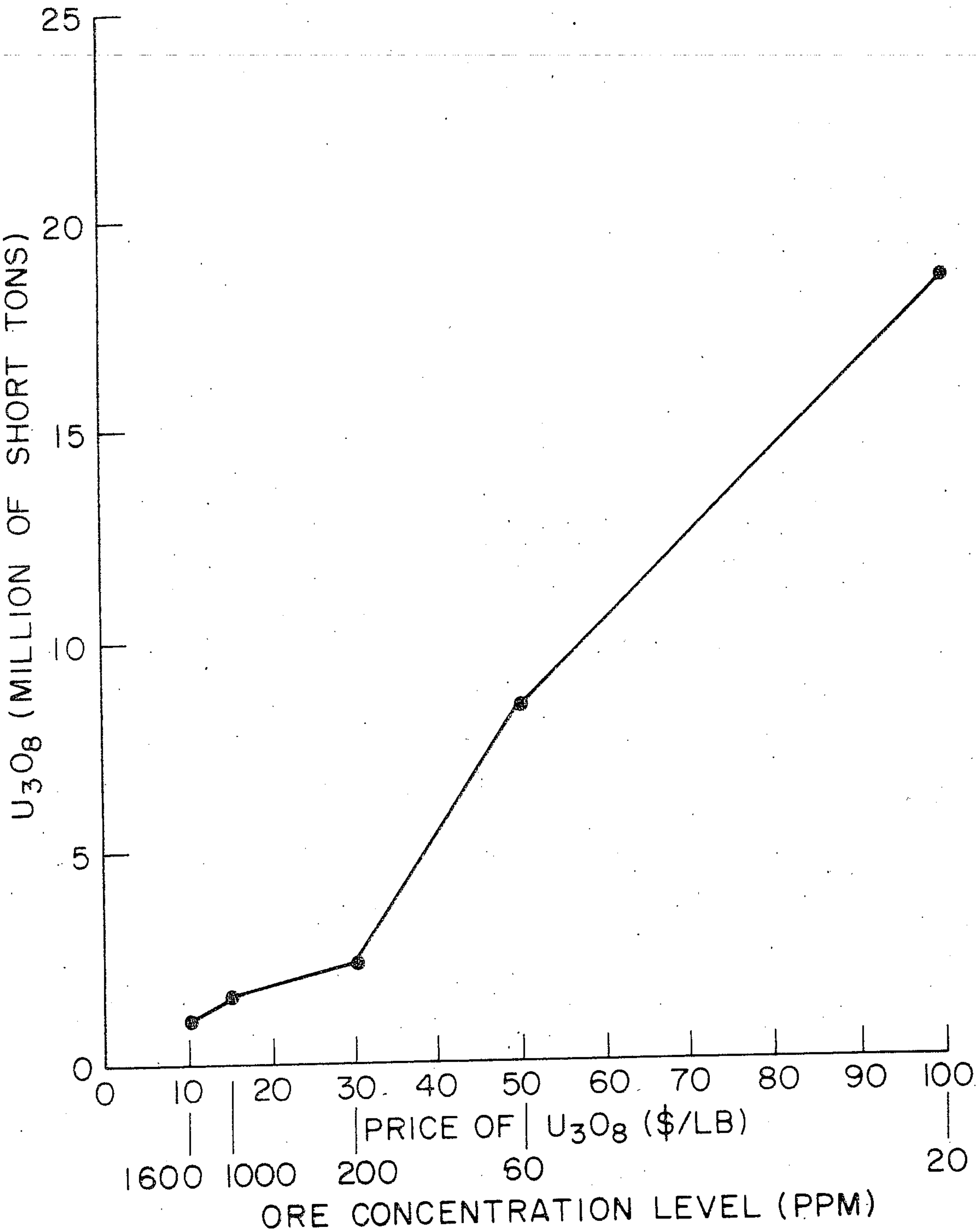


FIGURE 3

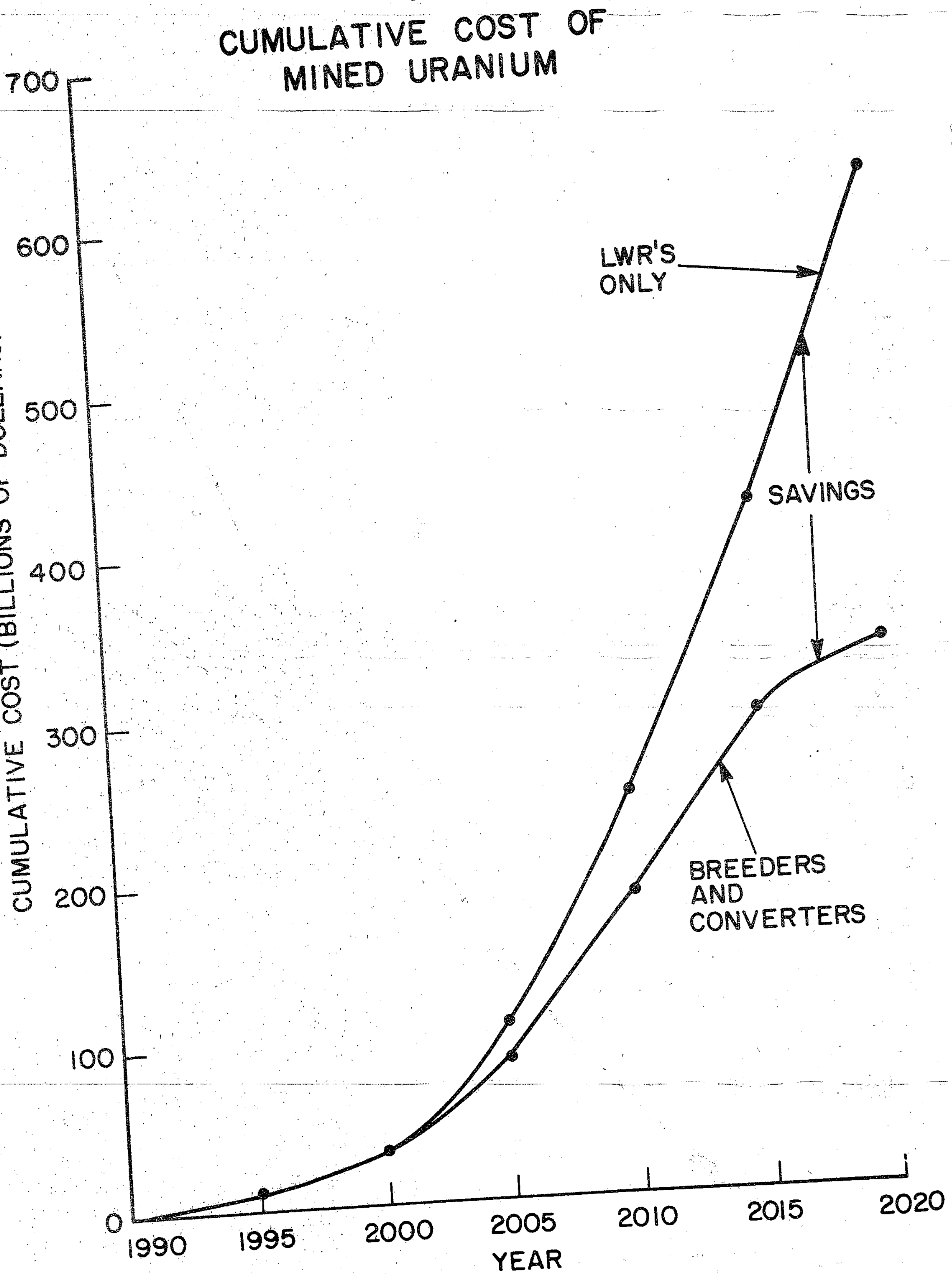

CUMULATIVE COST OF MINED URANIUM
700
600
500
400
300
200
100
0
CUMULATIVE COST (BILLIONS OF DOLLARS)
LWR'S ONLY
SAVINGS
BREEDERS AND CONVERTERS
1990
1995
2000
2005
2010
2015
2020
YEAR

FIGURE 4

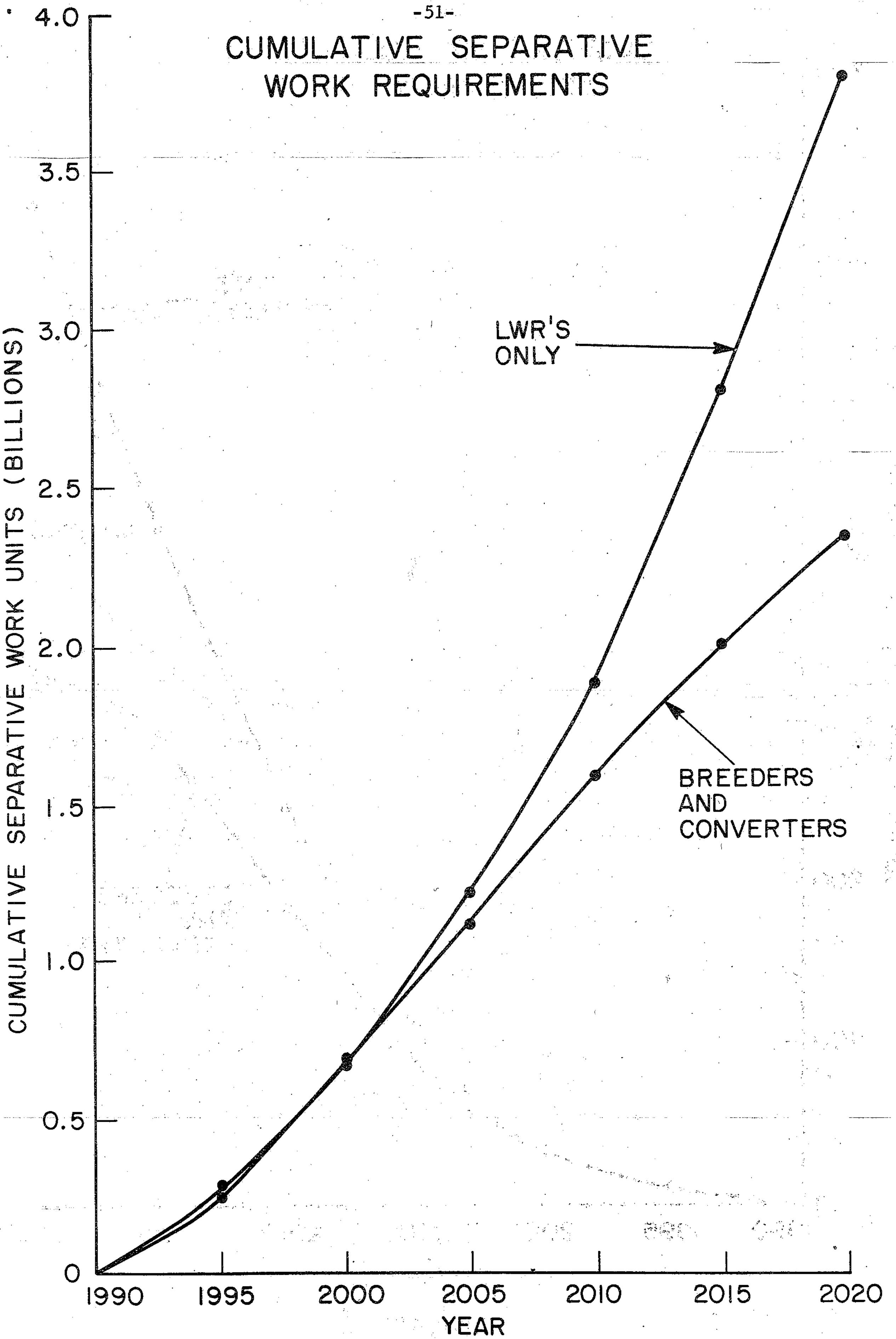

CUMULATIVE SEPARATIVE
WORK REQUIREMENTS
CUMULATIVE SEPARATIVE WORK UNITS (BILLIONS)
4.0
3.5
3.0
2.5
2.0
1.5
1.0
0.5
0
LWR'S
ONLY
BREEDERS
AND
CONVERTERS
1990
1995
2000
2005
2010
2015
2020
YEAR

FIGURE 5

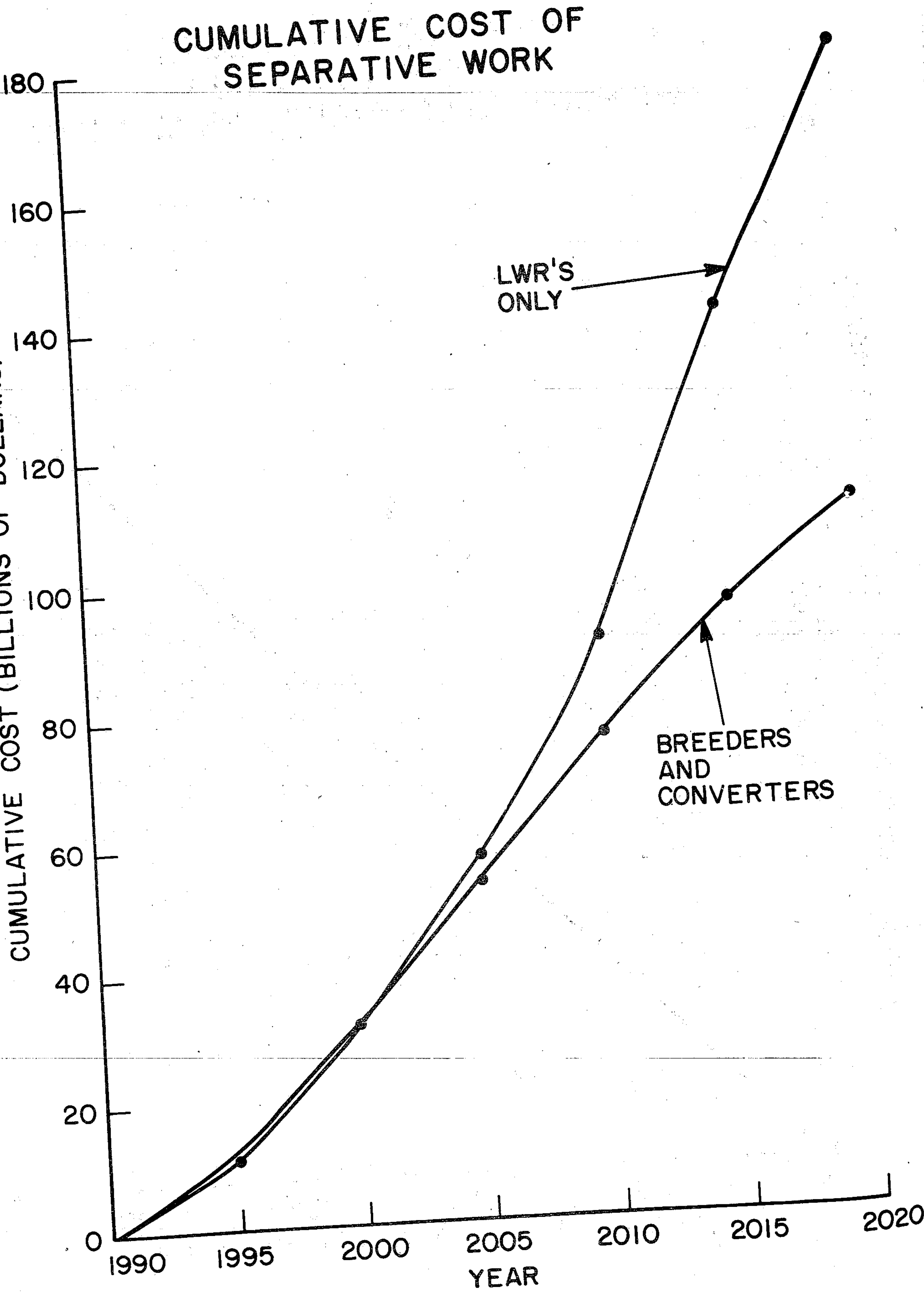

CUMULATIVE COST OF
SEPARATIVE WORK
CUMULATIVE COST (BILLIONS OF DOLLARS)
180
160
140
120
100
80
60
40
20
0
1990
1995
2000
2005
2010
2015
2020
YEAR
LWR'S
ONLY
BREEDERS
AND
CONVERTERS

FIGURE 6

REVENUE REQUIREMENTS FOR PLUTONIUM WITHOUT ELECTRICITY PRODUCTION

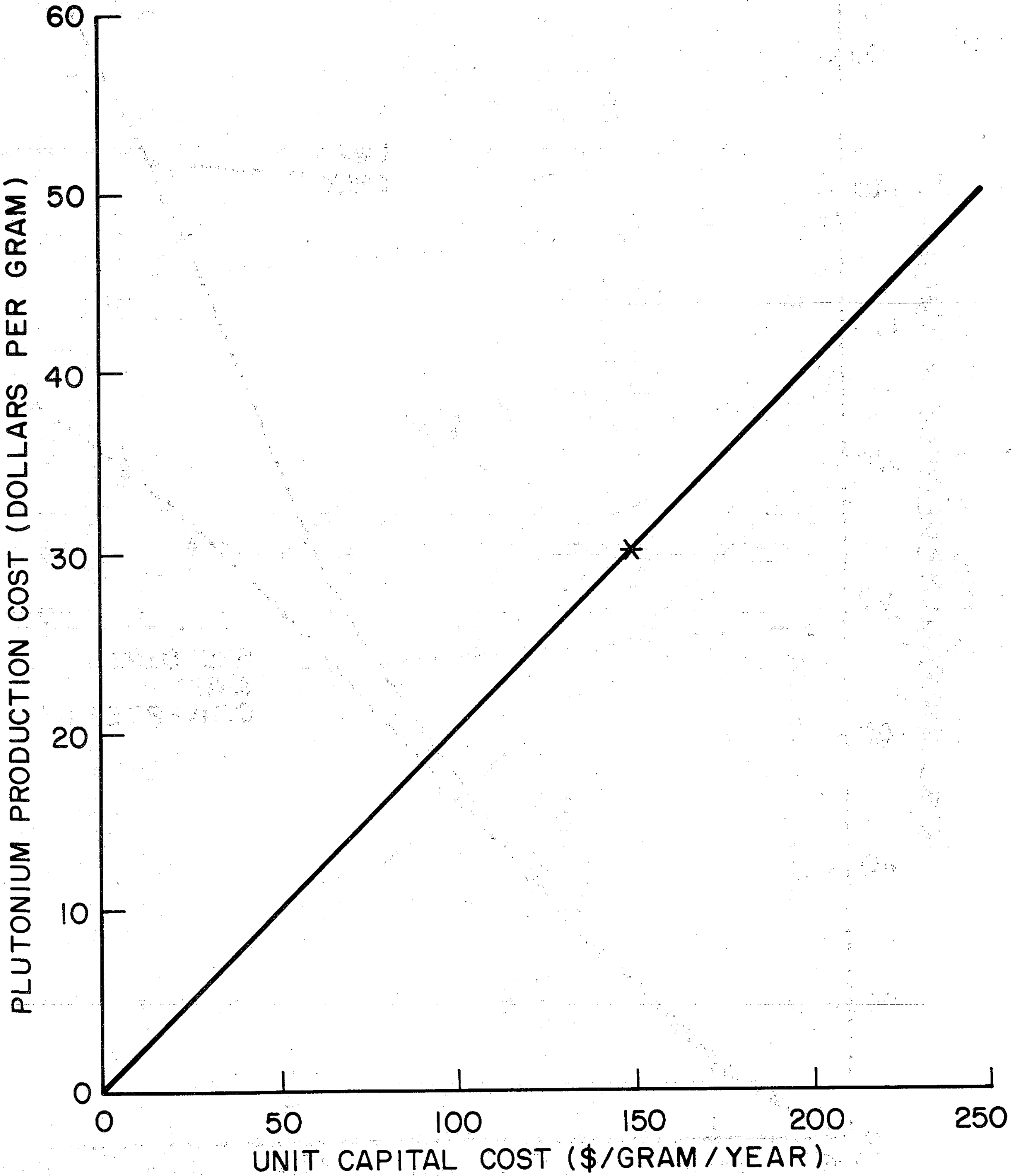


FIGURE 7

PLUTONIUM PRODUCTION COST AS FUNCTION OF PLANT CAPITAL COST

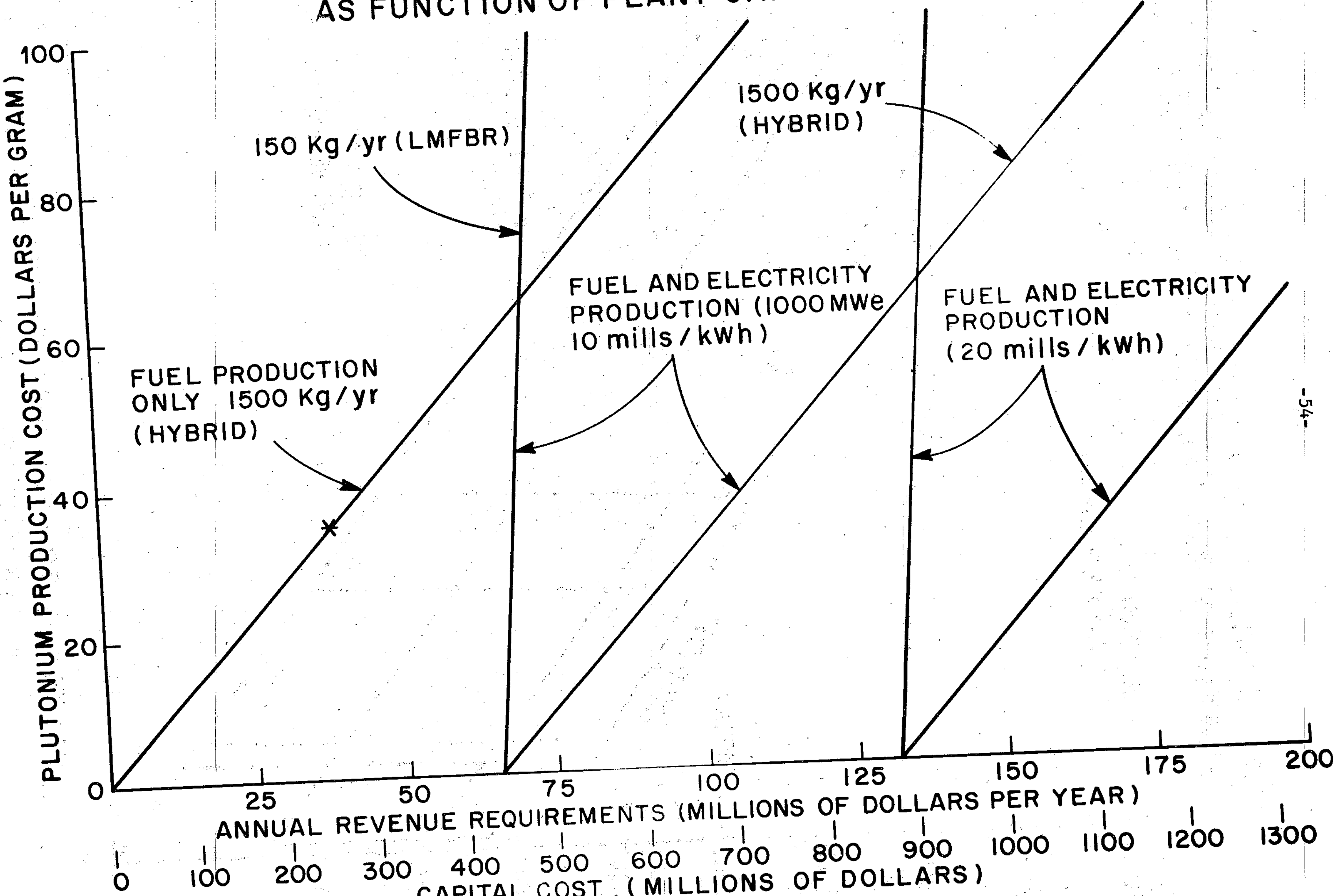


FIGURE 8

BREAKEVEN PLANT COST AS FUNCTION OF RELATIVE HEAT AND PLUTONIUM PRODUCTION

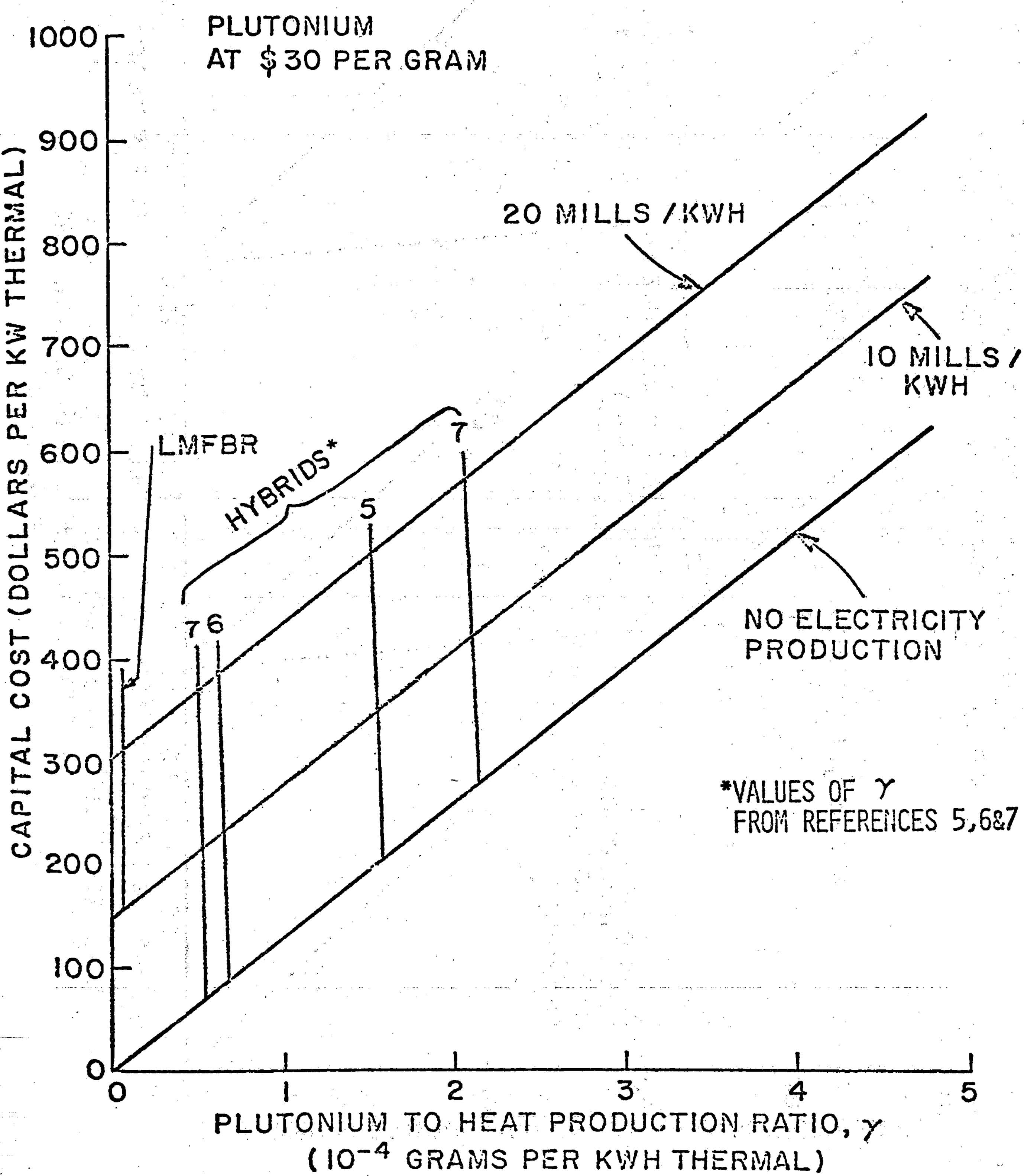


FIGURE 9

QUESTIONS ABOUT FIRST PRESENTATION

Furth: I'd like to make one comment on the basic strategy. You consider two extreme cases if I understand you right; one where you make power and nuclear fuel and one where you just make nuclear fuel. The latter is nice because you don't have to worry about shutdowns; on the other hand, it has more trouble making the grade economically. I think one should consider also a plant which is making nuclear fuel and at the same time using its power production to make some chemical fuel, hydrogen, for example. That would then be a useful intermediate case. It would have the advantage of not being on line for some consumer and, at the same time, it would get a little bit more boost economically than one would get from nuclear fuel production alone, so that the combination might be viable.

Burger: Another thing you might do is generate electricity for internal use in the power plants. That was the example we used in the paper.

Lotker: If I could make just a comment. The strategy which appeals to me so much is that with a fuel producer alone the utility, which is already thinking on a long timescale as Ray (Huse) pointed out, has the option of continuing with what will be by then mature technology, namely light water reactors and HTGR's. So it doesn't have to take the risk of putting a new device on its system; the utility does its business, namely generating electricity with units of increasing reliability, and the fuel producer does its business.

Furth: In addition the fuel producers could make a little something on the side which you could put in your cars. They have to.

Wolkenhauer: I know you didn't directly, but have you looked at the cost of the electricity for running your "fuel production only facility" and its impact on the whole cost picture? Presumably if you are producing fuel only, there is a considerable consumption of electricity in running the plants. What is the order of magnitude here in your calculations?

Lotker: We really didn't address that point except insofar as the case where we produce no fuel. Those charges could be interpreted as having the cost of electricity production as part of the capital cost of the plant. I think it's important to note that in this presentation we considered only capital costs. The costs you are talking about are clearly O&M costs and may turn out to be a substantial fraction of the total production costs. The electricity requirements, I think, will vary considerably from one concept to another. The laser has its own special requirements, as do things like neutral beams, power supplies and what-not. One hopefully would try to get around this, even in a plant that is only producing fuel, by generating electricity for onboard consumption.

Wolkenhauer: You may not know what that is (power consumption), but you can establish an envelope into which it must fall, like you have done for your capital costs.

Burger: Yes, one could do it but we didn't.

Holdren: The cumulative savings on the cost of uranium and on the cost of enrichment are quite sensitive to what you assume for the growth rate of electricity in the future. They are also sensitive to how much uranium you believe there is, and the final comparison on savings is quite sensitive to the incremental value of extra breeding ratio, in terms of how much you can afford to pay in capital costs for the plant compared to a converter or a burner of less efficiency. I wonder if you would comment on all three of these things. What did you assume for the growth rate of electricity? Whose numbers did you use for uranium, and how sensitive did you find the results to be if these numbers prove to be in error, as for example, the recent EPRI study suggests they may be? Have you looked at the incremental value of breeding ratio in comparison to, say, light water reactors and HTGR's?

Burger: The philosophy was to take one sort of mean case and see what types of savings you would have in order to get a feel for the order of magnitude, recognizing that there are a lot of uncertainties that are hard to quantify in detail. We took the Cornell workshops, of the past year, which were organized as an input to the recent five year energy budget planning. We took the projection of the growth rate they used which is roughly an electrical doubling time of 25 years. We took this as a typical case. It shows a very strong introduction of breeders and a very rapid phasing out of converter technology so as to give an upper limit on the kind of benefits you might gain from the introduction of a breeding capability. We didn't make any effort to explore the whole range of parameters with regard to the EPRI study. I think that this is a mean case and not way out on the tails of the probability curve.

Lotker: I think what we were trying to do is to give those of you who are working in the fusion business and in fusion-fission hybrid systems some idea of the way a utility approaches the economics. Jim (Burger) prepared an appendix which gives a broad-brush look at things like what revenue requirements are, what kinds of cost of money we use, and what types of capacity factors we think may be typical. Then I think, using this framework you can go ahead and put in any prediction for uranium supplies or load growth that you care to, and you can come up with an indicated savings.

Burger: What you might do with the curve for the net savings on uranium with a strong breeder scenario versus an all converter scenario is convert those savings which might accumulate over 50 years into what you might be able to spend for a research program over 20 years, by present worth techniques, refering the money to a fixed period of time and then spreading it out over different periods. This could be used to gain an estimate of what you might be willing to spend for a research program. The savings indicated by our curves suggest that you might be able to spend a billion dollars a year between now and the year 2000 for breeder development. That would be sort of an upper limit, the upper limit coming from the assumption that the capital cost per kilowatt electric is the same whether you use fusion-fission hybrid or converter reactors.

Moir: I have been concerned from the point of view of designing one of these hybrids, about the price of plutonium and what it's worth. On some of the slides, you showed \$10 per gram up to \$100 per gram and, being a spread of about 10, it seems that's a very wide margin. I'm wondering what we need to know to pin it down a little more. I notice that it is quite steep for the case of producing 1500 kilograms per year of plutonium. Our example that we have worked out produces 1300 kilograms per year, essentially the same thing, and even if it would go up to \$50 per gram, I think that would change the situation with respect to designing for fuel production rather than for electricity. So I'm wondering what you need to know to pin that worth down more accurately? How do the diffusion plants come into that?

Burger: Well, of course, one reason we showed a range is because, in a way, we're dodging that question. It's about $10 a gram today based on parity with uranium, In some of the examples we assumed that $30 per gram might be a reasonable escalation of this. We didn't really go into the details of what future plutonium prices will be other than to take $10 per gram today and expect to double or triple this, based on discussions and detailed analysis by others of future costs.

Lotker: I think the philosophy you can take today is to take it to the price of alternative fissile isotopes. Uranium prices are going to go up. The cost of the fissile isotopes will go up with the yellow cake prices and with the enrichment cost. The factor of the increase is not unreasonable. It is not a very large economic penalty to fission power in that the fuel costs are so low. So that is something that the industry can easily live with. In the future, of course, the price will be pegged on what the plant producing it will cost; namely what a hybrid or LMFBR will cost, and not an arbitrary price.

Burger: Diffusion plus the cost of the yellow cake is a substantial portion of the net fuel costs.

Jassby: There was a remark that Ray Huse made about the value of actinide transmutation, I think. He said that it was worth per gram the same as the value of plutonium. Is that correct?

Burger: There was some discussion of the possibility of transmuting waste; and not knowing what the value or cost would be to transmute waste, we said, just for one case, suppose that society is willing to pay the same to transmute waste, that it's willing to pay for fissile fuel. Then you could use our production curves by just relabeling the axes as radioactive waste consumption.

Jassby: In fact, do you have any idea how much it would be worth?

Burger: At the moment it would be far below what that curve in our paper indicates. People today argue about a fraction of a mill per kwhr(e) as an acceptable cost.

Coffman: The LMFBR people have projected in their impact statement the value of plutonium per gram, starting about $10 per gram, as showing a peaking up at several 10's of dollars in the late 1980's, and having a very rapid fall off and plateau around the year 2000 to about \$2 per gram. I wonder if you would comment on what that would mean to the feasibility of a hybrid system if, indeed, plutonium turned out to be worth between 1 and 5 dollars per gram.

Lotker: I think from the utility point of view that's great; it reduces our fuel cost, but I think realistically the projected cost of plutonium is going down because there will be lots of breeders making lots of plutonium. If that is not the case or if those breeders cost more than people think they'll cost, then the figure of \$2 will not stand.

Burger: The hybrid system would have to compete on the same basis. However, it has the capability of maybe producing a great deal more plutonium than the breeder so on the plutonium side it might have an edge, although with a low plutonium price, the value of heat or electricity produced would be more crucial to the economics.

Wolkenhauer: I'd like to respond quickly to Dr. Jassby's question. You referred to actinide transmutation. Your answer referred to what the utilities and the people are going to pay in terms of waste management, and I thought your numbers are quite correct. However, the transmutation process for actinides is fission and thus one can view the actinides as a fuel which they are in the peculiar spectrum of the CTR. You can, at least, hope that the actinides in a CTR have a worth close of that of plutonium. It probably won't be the plutonium worth but it might be like a half or a third of that. So your curves might not be too far off.

Burger: Do you have a credit for the heat in a transmutation device?

Wolkenhauer: Yes, and it is substantial.

Williams: Could you say something about the timing problems with the enrichment services, separative work, and how you think that affects the timing of the need for other fuel processing possibilities.

Lotker: I am not up to date on exactly what diffusion plant is being delayed or the so-called constipation problem of reprocessing facilities, but clearly all these things tend to make options like this more necessary. Although we talk about fissile fuel costs being a certain amount, the very fact that you don't have enrichment facilities may make a need for a breeder like this necessary even though there's plenty of cheap yellow cake.

<u>Burger</u>: You can have it if you pay for it today.

<u>Williams</u>: Clearly, the technology of enrichment is known. The question of timing is will enrichment services be available to satisfy the demand from light water reactors until breeders are available.

FISSION-FUSION SYSTEMS: CLASSIFICATION AND CRITIQUE

Lawrence M. Lidsky
Department of Nuclear Engineering
and
Research Laboratory of Electronics
Massachusetts Institute of Technology
Cambridge, Massachusetts

The advent of controlled fusion will make available excess neutrons which can be used in various ways in combined fission-fusion energy schemes. These schemes can be usefully classified as HYBRID - in which heavy element fission occurs in the fusion blanket: SYMBIOTIC - in which fissile material is bred in in the absence of fission events: and AUGEAN - in which the neutron surplus is used to transmute fission reactor waste products. Within these categories the various conceptual schemes can be further differentiated by the neutron spectrum in the blanket (fast or thermal) and by the dominant fuel cycle (thorium-U233 or U238-plutonium).

The possibility of beneficial combination of fission and fusion systems was apparent very early in the fusion program and some of the earliest fusion reactor concepts embodied Hybrid power-producing blankets. These early studies were originally classified Secret and were declassified at a time when the possibilities of combined systems were accorded little interest. The early history of some important recent developments are described below but no attempt is made to present a complete description of recent efforts; several summary articles are available that present a complete survey through June 1974 [1, 2] and this meeting reviews the most recent developments. In general the developmental history shows that the conceptual merit of various fission-fusion schemes (improved energy balance, high breeding gain, profitable fuel generation, etc.) have been extensively analyzed but realistic discussions of the technological implications of combined systems are now, for the first time, under active investigation. In this discussion I will describe a useful classification scheme and point out some common features that the scheme makes apparent. In Section I the classification matrix is defined, in II the early history is reviewed, in III noteworthy examples of various designs are described, in IV the advantages and disadvantages of various subsets are discussed, and in Section V the extension to low and moderate Q devices is noted.

I Definition and Classification

If all the intermediate products are burned, DD cycle fusion reactors will produce 0.5 neutrons per fusion reaction. These neutrons play no role in the fuel cycle and can be used for any desired purpose. The DD reactor has not been extensively considered to date because most fission-fusion studies have concentrated on $Q \gtrsim 1$ systems but merits reconsideration for low Q non-Maxwellian systems. Reactors operating on the DT cycle will require a tritium breeding blanket but calculations based on realistic blanket models predict excess neutron production of 0.1 to 0.5 neutrons per fusion reaction. It will be particularly important to garner the maximum energetic and economic benefit of these excess neutrons if, as appears likely, technological considerations will limit fusion reactors to relatively low power density and high capital-cost-to-power ratio. In any event, the excess neutrons are potentially valuable because the widespread use of fusion reactors will almost certainly follow the deployment of an extensive fission reactor based power system. These fission reactors, whether or not they have been modified to take best advantage of fusion generated fuel, will be potential customers.

Hybrid reactors are those in which both fission and fusion events occur in the same device. Such reactors are based for the most part on DT generated neutrons and are afforded great freedom of design because these high energy neutrons are capable of driving many inelastic neutron multiplying reactions [(n, 2n), (n, n'p), (n, f) and others]. In most cases, the hybrid reactor consists of an energy-multiplying fission blanket surrounding a fusion reactor. Because the 14 MeV neutron by virtue of its higher energy is capable of producing more progeny (has higher Importance) it is possible to design a system that has very high energy multiplication and breeding gain even though the fission lattice remains extremely subcritical. Hybrid reactors are more closely related to fission reactors than to fusion devices and so it is no surprise that they exhibit the same dichotomy between fast and thermal spectrum devices. Fast spectrum hybrids exhibit the most spectacular U238-plutonium conversion ratios but thermal systems can be used with thorium or depleted uranium feedstocks,albeit with lower breeding gain. The choice of the fuel cycle has great influence upon the scale size, power distribution, structural materials, coolant etc. and thus becomes a further classificatory variable.

Symbiotic systems are those in which excess fusion neutrons are used to breed fissile fuel from fertile material in the fusion reactor blanket. Such systems are truly symbiotic only if fission product generation in the blanket is minimized. The minimization of fast fission demands,in essence, that the neutron flux be well thermalized before reaching the fertile material. A thermal spectrum although compatible with both the thorium or uranium cycles, favors operation on the thorium-U233 cycle. It is also necessary, to avoid fissioning the bred fuel in situ, to choose a fueling scheme compatible with rapid fuel processing and short in-core residence time. As with any symbiosis, the symbiotic partner must be modified to make best use of the bred fuel - in the most highly developed examples this emphasizes the use of high conversion efficiency thermal spectrum fission reactors operating on the thorium cycle.

Augean systems are those in which fission reactor waste products are transmuted in fusion reactor blankets to less toxic form. Such systems might or might not have fission events taking place in the blanket (the actinide wastes are best destroyed by fission) but this distinction is of little import because the purposeful inclusion of radioactive wastes nullifies any reason to distinguish between fissioning or non-fissioning systems. The dominant neutron flux may be either thermal or fast. The fast flux is useful even though its intensity is relatively low compared to the thermal because it makes possible transmutation reactions not available in the fission reactor neutron spectrum.

The classification scheme is summarized in Figure 1 (see following page). Specification of these classifying parameters for a specific fission-fusion system yields enough information to predict with reasonable accuracy the purpose, products, scale, size and power density, in a combined power cycle, and major engineering difficulties. It is possible to define additional parameters and subdivide more finely but systems within a given subclass will be similar enough that they can be compared on a rational basis as differing solutions to the same set of problems.

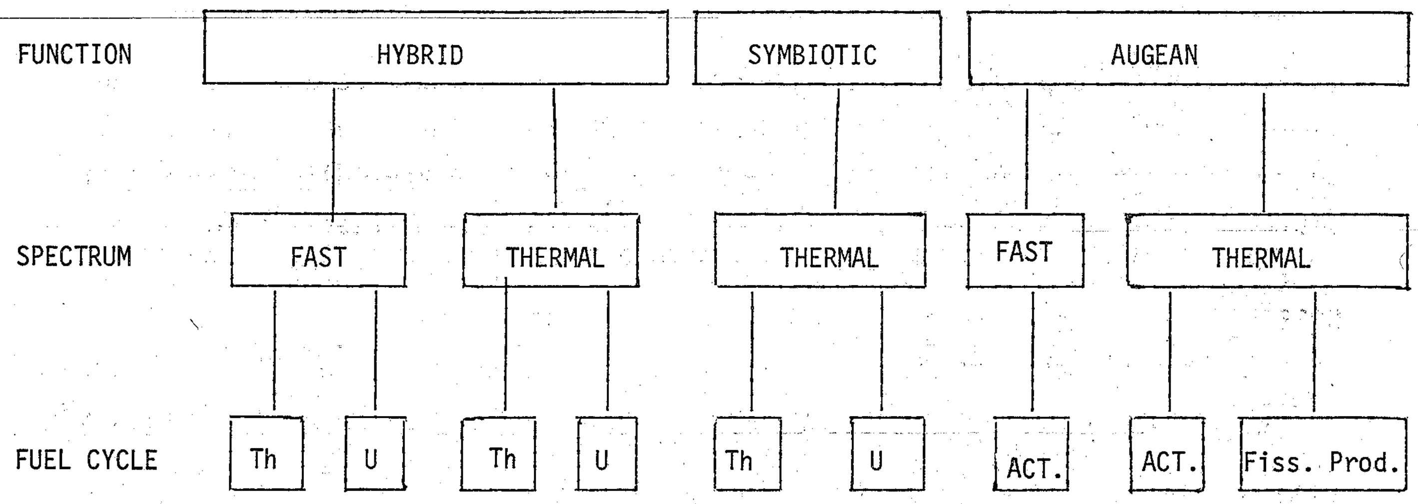


FIGURE 1.

Classification Matrix for Fusion-Fission Systems. Note that some technologically less promising alternatives have been omitted from the figure.

II Paleography

Early fission-fusion reactor concepts were all Hybrid devices, concentrating on the production of fissile materials or tritium. The problem of regenerating sufficient tritium to replenish that consumed in the reactor is now known to be solvable without recourse to heavy element multiplying blankets but the Li^7 (n, n'α)T reaction was not known at the time. The importance of this reaction was first appreciated when an experiment to measure the tritium production by 14 MeV neutrons in a lithium deuteride sphere was performed in Los Alamos Scientific Laboratory during the period 1954-1958. A classified report describing the experiment was issued in 1958 (limited general publication was permitted in 1961 [3]. Designs antedating the 1958 report assumed the necessity of using the fission reaction for sufficient neutron multiplication.

The earliest reasonably well analyzed Hybrid scheme was described in 1953. This proposal [4] called for the use of a pre-ionized high density, mirror-confined plasma as the target for a high energy deuterium or tritium beam. Such a target (with electron temperature of 2-4 keV) is some 20-40 times more efficient than a neutral gas target for production beam-induced reactions. A basic feature of this design was the concept of surrounding the target chamber with a blanket of depleted uranium. The energetic 14 MeV neutrons from the DT reaction would induce fast fission in this blanket with subsequent capture of these neutrons in U238 to form plutonium-239, or in lithium-6 to breed tritium. Using unpublished data for the 14 MeV fast fission cross section, the proposal predicted that sufficient tritium could be regenerated and further that "with the addition of energy produced in capture events, at least 200-400 MeV of energy is released per triton destroyed. This energy will appear as heat in the blanket. The heat generation might be capable of producing considerably more electrical energy than that required for operation of the machine".

The study was soon followed (1954) by a much more detailed version by Imhof, also of California Research Corp. [5]. This report stated explicitly that a multiplying blanket was the key to using the DT reaction and considered toroidal configurations in addition to mirrors. Imhof postulated a plasma device driven by external injection of all components. Using a depleted uranium and lithium blanket of 60 cm thickness, Imhof calculated a resulting total multiplication of 3.0 neutrons per incident 14 MeV neutron after regeneration of tritium. These figures correspond to a total energy release in the blanket of 160 MeV, i. e. an energy multiplication in excess of 10. Imhof pointed out that the low tritium inventory of a fusion reactor coupled with the high neutron gain of a multiplying blanket could yield tritium doubling times of the order of "hours or days". This early paper anticipated many other features of later hybrid studies. For example, there was also considered the use of beryllium moderators, natural uranium fueling, and the possibility of optimizing the design for either fissile material or power production.

Similar studies were carried out in the United Kingdom at about the same time. In a report that was classified at the time of its first publication (1955) J. D. Lawson considered the effect of a U238 fast fission blanket on a DT reactor using lithium-6 for tritium regeneration and also considered an injected beam-target plasma reactor similar to that described above [6]. Lawson calculated that a U238 blanket would increase the power of reactor-scale systems by a factor of approximately five. Lawson did not discuss the use of the excess neutrons for breeding fissile materials, but in a review of a 1957 meeting of the British Association he reported "among the topics raised in the discussion after the talks, was the possibility of using thermonuclear neutrons for breeding fissile material and the point was made that this might be worthwhile even if the thermonuclear reaction cycle itself was not self-sustaining. It is however too early yet to say how this would complete with conventional breeder systems" [7]. There were a few other moderately well documented first generation studies. In 1957 L. G. Barett described a fission-fusion reactor based on the stellator concept with a near-critical or critical aqueous fuel (UO_2SO_4) external fission multiplier and heat transfer medium [8]. There was also filed in 1957 a British patent describing a toroidal vessel with thick metal walls enveloped by a first blanket containing U238 and a second blanket containing lithium-6 [9]. The patent specification pointed out the capability of attaining both power multiplication and enhanced tritium production but no numerical estimates were given.

This early history has some interesing facets. The earliest schemes were in fact "two-component systems" and were discussed as such. The application of this concept to toroidal systems is of course a present reality. Not entirely surprisingly though, linear systems are in many ways better suited for beam target interactions and with the advent of recent technological advances in pulsed beam technology the study of fission-fusion systems is returning to its roots. A second interesting point concerns possible reasons for the 20-year hiatus in the study of non-Maxwellian plasmas and Hybrid reactors. Several strong arguments were made by Lawson in 1955 when he noted that the power gains of 5-20 or so could just as easily be achieved in a reactor by relatively minor increases in density or temperature. There was no reason at that time to doubt that these increases were possible and if they were, there would be no reason to incur the added complexity of hybrid blankets. However, Lawson also pointed out that there was a narrow range where such a factor of several decrease in $n\tau$ might be useful (if reactors were barely submarginal) and also that power multiplication might be important in reducing break-even beam intensity in a target plasma scheme. These cases are precisely those of current interest.

III Recent Developments

In this section I describe some noteworthy examples of fission-fusion systems. A predominant portion of the current research effort is devoted to the development of various Hybrid schemes. The most highly developed of these are the Los Alamos and Livermore studies of theta pinch, mirror and laser Hybrids reported at this meeting. Rather than abstract papers given in this report, I will instead give somewhat earlier examples that illustrate the engineering results of choosing alternative neutron spectra and fuel cycles. I will devote proportionately more space to summarizing important contributions to Symbiotic and Augean systems because these were not presented at length at this meeting.

i) The PNL Thermal Fission Hybrid

B. R. Leonard Jr. and W. C. Wolkenhauer have reported extensive numerical computations of the neutronic performance of hybrid reactors based on subcritical fission lattices. Their most recent results are summarized in Leonard's review article but the earlier laboratory reports give more detail [10]. Leonard and Wolkenhauer considered fuel cycle economics, questions of environmental quality, and resource preservation to be the principal determinants in combined system design. Because the energy equivalence of a neutron is so much higher in a fission fuel cycle than in a fusion fuel cycle, and because Hybrid plant costs could conceivably approach fission breeder costs, they chose to investigate systems capable of very high fissile energy multiplication in subcritical blankets. For similar plant costs, they expect that in addition to the energy gain, important secondary advantages would accrue due to the anticipated relatively low inventory of fissile fuel and reduced criticality hazards. To satisfy the resource preservation criteria the design goal included the requirement that both tritium and fissile conversion ratios exceeded unity.

The reference blanket configuration consisted of a thin neutron converter region followed by a thermal fission lattice followed in turn by a graphite moderator reflector and a natural lithium absorber for thermal neutron leakage. The neutron converter, depleted uranium carbide with natural lithium cooling, serves several functions. It provides nearly unit multiplication of the source neutrons (i. e. $\sim$2 neutrons produced per neutron incident on the convertor) and energy multiplication through fast fission. The uranium and lithium decouple the fission lattice across the plasma cylinder and so help to eliminate extensive power peaking in the vacuum interface. The lithium-6 in the converter provides most of the tritium breeding and shields the plutonium-239 produced in this region from slow neutrons leaking back from the thermal lattice. These neutrons would otherwise make this region an intense fission source and reduce the fissile regeneration ratio. The thermal fission lattice for this model was 150 cm thick and the total thickness of the blanket including the lithium outer absorber was greater than 2.1 meters.

The neutronic behavior of the blanket was analyzed using standard fission reactor analytical methods. The effect of fission products on the neutron balance was calculated on the basis of 60-week irradiation to an average fuel exposure of 90 MWD/T. With 1.35% enriched uranium in the thermal lattice the conversion ratios for both tritium and uranium are greater than unity (1.06), the lattice is subcritical ($k = 0.87$) and the energy deposited in the blanket per source neutron is approximately 500 MeV (i. e. an energy multiplication of 35). Because of the high energy density the 14 MeV neutron energy flux on the first wall is limited to an upper value of 0.05 megawatt/m^2 if the average power density in the thermal lattice is not to exceed that in advanced gas cooled reactors. There are several noteworthy points common to thermal lattice systems. One is that the blanket must be relatively thick to yield adequate thermalization and to minimize leakage. Another is that the energy multiplication is inherently high unless the fuel is considerably enriched. Effectively, this design results in a fission reactor that takes advantage of a source of 14 MeV neutrons to allow high power density in a subcritical system and comparatively high fissile conversion ratio (for a slightly enriched reactor) with a thermal spectrum.

ii) Subcritical Fast Fission Hybrid

In 1970 J. D. Lee of the Livermore Radiation Laboratory reported on calculations of energy generation and fissile breeding reactions for 14.1 Mev neutrons incident on large assemblies of pure thorium, U233, and natural uranium. The results of these calculations indicated that uranium would be an attractive blanket material for both energy generation and fissile breeding. These early results were followed by a detailed study of "plausible" blankets capable of regenerating tritium, breeding fissile materials and releasing fission energy in the blanket [11]. These calculations, employing modern computational techniques to replace experimentally guided approximations, were important connecting links between the simple models of the early 1950s and the fast fission systems discussed at this meeting.

Lee first considered simplified two-zone blanket structures containing uranium, lithium and niobium. He found that it was necessary to use isotopically depleted lithium (depleted in lithium-6) to produce the required tritium. However, he also noted that the U238 (n, α) breeding ratio was very high, raising the potential breeding gain possible for blankets with plutonium recycle. Lee then studied the effect of loading the blanket with plutonium-239 in various concentrations. As expected, the energy generation and tritium production both increased with plutonium loading. The increases were nonlinear with loading because of the effects of fast fission multiplication. Because for a subcritical blanket the energy multiplication scales approximately with $(1 - k)^{-1}$ where k is the "effective reactivity" coefficient, the breeding gain is a very sensitive function of k as the reactivity approaches unity. The optimum value of k must be determined by an economic balance which depends on heat transfer capabilities, specific energy density, fuel cycle costs, etc. Lee somewhat arbitrarily imposed the condition that k was not to exceed 0.9 and concluded that within this limit it was possible to achieve energy multiplication of approximately 30 with respect to nonfission DT reactor blankets. The limiting value of k was apparently based on safety considerations.

To estimate the effect of fission product build-up Lee repeated his calculations with the inclusion of 8 atomic percent fission products. This value of fission product load corresponds to approximately 80,000 MWD/T burnup, consistent with values assumed for fission breeder reactors. This reduced the reaction rate per incident neutron by 30% but the breeding ratio was relatively unaffected. The reaction rate is important because the fuel doubling time depends primarily on the specific energy density in the blanket and thus on the reaction rate per unit of inventory. Lee computed the average blanket power density required to give five year plutonium doubling to be 225 watts/cm^3 and from this value derived the required first wall flux of 14 MeV fusion neutrons; 7.25 Mw/m^2. In Lee's opinion at that time, this did not pose a "nuclear heat transfer problem". It should be pointed out however, that this value is quite high in comparison to the flux levels deemed acceptable in more recent and detailed engineering studies. Lee also investigated several other interesting cases. For example, he appraised the possibility of fast fission systems based on other fuel cycles and more realistic fuel assemblies. He found, for example, that both the breeding ratio and energy multiplication were much lower for thorium-U233 systems compared to the uranium-plutonium-239 systems. This result was not surprising because it is known that the thorium-uranium cycle is particularly badly matched to the spectrum of fast fission reactors and the spectrum of a fusion blanket will be harder still. Lee considered also the effect of replacing uranium metal with uranium oxide as would most certainly be done in a realistic blanket. Although the metal fuel is neutronically superior, various metallurgical considerations rule out its use. With mixed oxide fuels, the average power density required to achieve five year doubling increased to 600watt/cm^3 and because k was low, the first wall neutron flux required to drive the blanket to this level increased proportionally more. This demonstrated out a severe technological problem in fast fission systems that more recent designs have attempted to rectify. Lee concluded that although there were two major advantages of the fission-fusion hybrid (supplementing the fuel production of fission breeder reactors and significantly reducing the plasma containment requirement for a "viable" power system), there was no clear cut neutronic advantage of the subcritical fast fission blanket over a fertile blanket supplementing the breeding of a separate fission reactor.

iii) Fission-Fusion Symbiosis

The potential benefits of fission-fusion symbiosis depend as strongly on the technological aspects of fission reactors and the detailed economics of the fission fuel cycle as they do upon the analogous features of projected fusion reactors. The dominant consideration is that present day reactor technology is based on the existence of naturally occuring U235. The problem of ultimately limited fuel availability could be solved, in principle, by the construction of fast spectrum plutonium cycle or thermal spectrum thorium cycle breeder reactors. However, it appears that the fast reactors are inherently technologically complex, expensive, and require compromises between somewhat conflicting requirements for fuel conversion ratio, cost and safety. Thermal spectrum breeders are potentially simpler and safer but will probably have unacceptably low conversion ratios. Although technological problems mount rapidly as the conversion ratio, C, reaches 1.0, values of C = 0.9 can be relatively easily attained using existing technology. The goal of various symbiotic schemes is, in a combined system, to achieve both power production and fissile breeding while operating the fission component of the system in technologically simple areas. A measure of success is the achievement of these goals at minimum total cost.

Various symbioses based on "electronuclear" neutron sources have been proposed. These sources utilize the nuclear spallation reaction occuring when a high energy beam of charged particles interacts with a high atomic weight target. The neutrons produced in this fashion have a broad energy spectrum and are accompanied by an intense gamma ray flux. Electronuclear sources (based on designs calling for beams of 0.1-2.0 GeV at currents of hundreds of miliamperes) have been proposed for materials testing, transuranic isotope production, and fertile to fissile conversions. It appears that the use of spallation produced neutrons for fertile to fissile conversion is uneconomic at this time but if fusion reactors prove unattainable and fission breeders are incapable of achieving short enough doubling times, then electronuclear generation of neutrons will become essential.

Electronuclear schemes suffer from a fundamentally unfavorable energy balance. From this point of view, fusion reactors are potentially far superior; even submarginal fusion reactors can produce neutrons at much less energy cost per neutron than the theoretically most efficient scheme. Of course if the fusion reactor produces net power also, then the energy production of the fusion reactor in a symbiotic scheme can be viewed as a valuable by-product. Although symbiosis is often alluded to, usually in conjuction with electronuclear breeding, there has appeared only one detailed analysis in the open literature. This analysis by L. M. Lidsky, was first presented at the 1969 Culham Conference of Fusion Reactor Engineering and appears in the proceedings of that conference [12]. A reasonably detailed summary of that paper was given in a recent review and so only the results will be noted here.

Lidsky claimed that symbiotic systems utilizing the strength of fission and fusion systems could achieve properties attainable by neither alone. Properly done, such symbiosis removes constraints from the system (the fission reactor need not breed fuel, the fusion reactor need not generate power) and introduces additional adjustable parameters (relative reactor sizes, breeding ratios, etc.). The components of such a system can be chosen to minimize the total system cost rather than the cost of the individual components. To allow optimization in this sense it is essential that the fusion reactor not be burdened with fission products nor with appreciable power multiplication in the blanket. The analysis contained three segments; first it was shown that it was possible to produce appreciable quantities of a fertile material in a nonfissioning blanket while regenerating tritium; second it was shown that the costs were relatively insensitive to the parameters of the fusion reactor; and third a particular example was chosen to demonstrate that a nonbreeding fission reactor and a fusion reactor with a submarginal energy balance could in symbiosis yield an economically attractive power station with ten years doubling time.

Symbiosis requires the simultaneous production of fissile materials and tritium. It is not essential to do this in a single device because a symbiotic power system might comprise several physically distinct fission and fusion reactors and the production of fissile material could then be concentrated in a single CTR device optimized for that purpose alone. Lidsky presented for purposes of discussion the design of a dual purpose blanket based on molten salt reactor technology. The first wall of the proposed blanket is cooled by liquid lithium, the remainder of the system is cooled by, and the fertile material carried in, a molten salt developed for the two fluid molten salt breeder reactor. This salt is comprised of the fluorides of lithium, beryllium and thorium, depleted in lithium-6 to favor absorption in thorium. The molten salt region is surrounded by an internal graphite moderator to attenuate the high energy neutrons capable of producing fast fission in the thorium, and an outer graphite zone serving as thermal neutron reflector. The tritium and uranium-233 conversion ratios are variable over a wide range; in the design presented there was produced 1.126 tritons and 0.325 uranium-233 atoms per incident 14 MeV neutron. The use of the molten salt is of particular interest because the U233 is easily removed from the salt by fluorination. This eliminates one expensive fuel cycle cost and, most important, ensures that the fission rate in the fusion blanket can be held to very low values.

The costs and relative sizes of the fission and fusion reactors in a symbiotic scheme depend most strongly on the properties of the fission reactor component. This emphasis exists because there is a larger neutron excess per nuclear reaction in the fusion reactor and because the energy released in a fission reaction exceeds that of a fusion reaction by more than a factor of ten. These factors, in combination with the much lower specific inventory in a fusion reactor, result in the conclusion that " for a wide range of reactor types, over an interesting range of system periods, the fusion-fission reaction ratio is near unity". Similarly, "the fuel doubling time of a balanced hybrid system is determined almost entirely by the fusion reactor component". A similar argument can be used to show that for "reasonable" values of Q and reactor core costs, the symbiotic plant's cost is dominated by the fission reactor component.

For a specific example Lidsky considered the design of a central station power plant with net output of 1500 Mw_e and a seven year fuel doubling time. The fission segment was a molten salt reactor producing 4450 Mw(th) with conversion ratio C = 0.96. The fusion reactor was a tokamak with a molten salt blanket capable of a total fuel conversion ratio of 1.40. The expression for system balance showed the required fusion reactor thermal power was only 295 Mw. This reaction rate could be provided by an underated tokamak with 100 Bohm times confinement at a wall loading of 1.0 Mw/m^2. The fusion reactor in this system would be a net consumer of power (Q = 0.57). The net power output of the plant in the example would be, at a thermal equilibrium efficiency of 40%, 1600 Mw_e with a combined thermal efficiency of the system equal to 36%. The net effect of the symbiotic combination is to turn the central station power plant into a breeder reactor for a surcharge in efficiency of less than 4% and a surcharge in cost of approximately 20%, without compromising either reliability or safety.

iv) Augean Systems

The long term biological hazards of fission reactor operation are associated with a relatively small portion of those medium-weight isotopes resulting from the fission process itself and with heavier isotopes produced by nonfissile absorption in heavy elements. It is convenient in considering nuclear waste disposal to consider these two groups separately. The fission product group is composed of medium weight elements resulting from near-symmetric fission. These isotopes have relatively short life (approximately 30 years), high toxicity, and are produced with relatively high probability in the fission process. Most important of these are krypton-85, strontium-90, and cesium-137 which account, with their daughter products, for more than 90% of the total activity of fission reactor waste after a ten year cooling period. The heavy element group, the so-called actinides, are composed of those isotopes formed by successive neutron capture and decay in various heavy elements in the reactor core. The actinides are produced in relatively small quantity compared to the fission products but they are characterized by very long half lives and extreme toxicity. Because the toxicity is dominated by a few isotopes whose absolute production rate per fission event is low, it is possible to consider the economics of using neutrons to burn out the troublesome isotopes. The decay chains following neutron absorption by fission products leads in general to stable or relatively harmless progeny. The actinides can be destroyed by causing them to fission and the resulting fission products then treated much the same as reactor waste. It has been estimated that with a 20% neutron surplus in fusion reactor blankets it would require 1 watt of fusion power generation to burn out the long lived waste products associated with 9 watts of fission power generation.

The process of neutron transmutation, although simple in concept, is quite complex in detail. The complete absorption and decay chains must be considered for each isotope chosen for transmutation because in some cases, the members of the chain are more toxic than the original wastes. Furthermore, the cross sections for the troublesome isotopes tend to be small. If it were otherwise, they would have been consumed in the fission reactor itself. Thus, except for some advantages accruing from a different energy spectrum, the flux required in a fusion reactor to give reasonable transmutation rates must be as high or higher than that in the original power producing fission reactor. Such high fluxes are not easily achieved in fusion reactor blankets.

The most detailed study to date of the neutronics of high level radioactive waste transmutation was carried out by W. C. Wolkenhauer and his colleagues at Battelle Northwest Laboratories in 1973 [13]. The technical approach chosen was the assumption of idealized analysis condition in order to establish upper limits for transmutation rates in a fusion reactor. The particular waste consituents singled out for study included the isotopes krypton-85, strontium-90, cesium-137, iodine-129 and a mixture of strontium and cesium isotopes corresponding to reprocessing plant output. The actinide group included 23 isotopes with particular attention given to the plutonium chain, again at isotope ratios corresponding to reprocessing plant wastes. The study showed the flux levels required to reduce effective life under neutron bombardment to two years was in general in excess of 10^{16} neutron/cm^{2}sec. The exceptions to this were iodine-129 with a required flux of 6×10^{14}/cm^{2}sec and the mixed actinides which required a flux of only 1.6×10^{13}/cm^{2}sec. Although the two year lifetime is very short compared to the natural lifetime of the fission product, it still corresponds to the storage of great quantities of reactor wastes in fusion reactor blankets.

Clearly, very high fluxes are required to achieve transmutation rates high enough to be of value. Wolkenhauer et al. postulated the use of a beryllium loaded blanket to enhance both the fast and thermal fluxes. The transmutation rates were computed using the neutron spectrum and flux level computed for the highest flux region of the blanket, neglecting absorption in radionucleides. No structural material or waste packeting was considered nor was any attempt made to account for self-shielding. All these factors in more detailed treatment would result in a reduction of the transmutation rate.

Wolkenhauer et al. concluded that; i) the transmutation of strontium-90 and cesium-137 is theoretically feasible for source strengths in excess of 1.0 Mw/m^2; ii) transmutation of krypton-85 is not theoretically feasible; iii) transmutation of mixed strontium isotopes which have cooled for ten years is theoretically feasible whereas transmutation of cesium isotopes after ten year cooling period is feasible only for very high reactor surface flux.; iv) That actinide transmutation in controlled fusion reactors appears to be an attractive scheme particularly in view of the fact that the actinides constitute the long term hazard of high level waste.

IV Comments and Conclusions

The study of fission-fusion systems has become fashionable; both quantity and quality of published reports is increasing rapidly. It is encouraging that the recent numerically sophisticated results verify, in large measure, earlier estimates of neutron excess, energy multiplication, fissile element production, and so forth. However, given that from the standpoint of neutronics and simplified engineering studies, such systems could be built, it is far from certain that they should be built. The ultimate worth of any particular means of accomplishing a given end must be judged in the context of other means of accomplishing that end and must also be weighed against the possibility of changing goals. In this section, I will point out some questions raised by consideration of alternatives.

A. Hybrid Systems The most highly developed models are based on subcritical thermal or fast blankets fueled with natural or slightly enriched uranium. The advantages claimed are substantial energy multiplication, absence of criticality hazards, and the ability to produce substantial quantities of fissile material usually Pu-239)

i) Energy Multiplication: If the bulk of the power in a combined system is produced by the fission reaction then the Hybrid device must be judged by fission reactor standards. The engineering design of large fission reactors is dominated by two concerns: The first is the necessity of ensuring that a Loss-Of-Coolant-Accident is highly unlikely and the second is that the fissile fuel be so disposed that it can be transferred safely and efficiently and if possible, while the reactor is operating. It is for these reasons that virtually every reactor (CANDU excepted) is constructed with a vertical axis, and fueled with bundled-parallel linear fuel elements. Even in such simplified geometry the machinery associated with fuel handling is very complex. The irradiated fuel upon removal is so highly radioactive that cooling must be provided during removal and transport to temporary storage. Any accident during this phase of operations, while probably not dangerous to the public at large, would greatly complicate plant operations.

The problem of safe, efficient fuel handling in the complex geometry of a fusion reactor blanket has major implications. For example, one proposed Hybrid reactor would have more than 1.4×10^5 separate fuel pins in 320 modules making up a near spherical lattice. Each pin would be nearly 2 meters long. The manner in which these pins would be replaced is not addressed but the task seems formidable.

It is difficult to point out explicit possibilities of Loss Of Coolant Accidents in Hybrid systems because the engineering has not been done in sufficient detail and because in any event the probability of such an accident is small. However, the Hybrid fission blanket will have a large internal void, the plasma region itself, and will in all probability be traversed by numerous passages for particle injectors, vacuum pumps, electrical leads, etc. that are not needed in a pure fission reactor. Thus, in comparison, the Hybrid system seems more likely to fail than does a pure fission reactor. It should be pointed out that some designs are less sensitive to the effects of coolant loss accidents by virtue of relatively low power density.

ii) Subcritical Operation: The advantage of subcritical operation from the point of view of safety considerations is often overrated because all large power reactors have been designed with large negative power coefficients of reactivity. Therefore, they are in effect passively safe against small reactivity increases. However, excess reactivity is built into the core so that the fuel need not be changed uneconomically often.Usually what is done is to supply enough excess reactivity so that radiation damage, rather than fission product poisoning, determines the core lifetime. This excess reactivity is held down with control rods. The same must be true for Hybrid blankets. Even though the operating reactivity is held constant at some subcritical value, there will almost certainly be enough excess reactivity available so that control rods must be used. The worst possible reactivity accident would be caused by the accidental rapid withdrawal of control rods. Such an accident is, if anything, more likely in the complex environment of a Hybrid device.

Some Hybrid reactor designs, notably those with very high plutonium conversion ratios, actually increase in reactivity with exposure. Such designs have a different problem because the breeding gain is highest in regions of high flux density— and the high flux density regions tend to occur where the local reactivity is highest. Such a system is obviously unstable with respect to nonuniform production of fissile material and to power peaking, and so will require a relatively fine mesh sensing and control system; a control system much more complex than those used in current fission reactors.

Subcritical operation is often cited as a major benefit of Hybrid reactor operation. It does not appear so to people familiar with fission reactor operation because criticality accidents are not the Design Base Accidents. At best subcritical operation offers a small quantitative advantages; at worst the difficulty of control in Hybrid devices actually puts them at a disadvantage.

iii) Fissile Production: The desirability of fissile fuel production in Hybrid reactors poses a most complex economic question. The cost of the fuel produced must be compared with the cost of fuel produced by alternative schemes (the LMFBR, for example) and against the total cost in some larger sense of alternative approaches calling for fuel conservation (for example, a large scale trend toward reliance on U233 fueled, heavy water moderated reactors). The only eventuality in which the Hybrid would be clearly superior to pure fission reactors would require simultaneously an inability to build safe, high gain breeder reactors and an unwillingness to switch to neutron conserving fuel cycles.

B. Symbiotic Systems: Symbiotic systems in general have fewer constraints than Hybrid systems because the blanket is not required to perform three functions at once - power production, tritium generation, fissile fuel production. This freedom is purchased at the cost of reduced fertile conversion and of course, power generation. Because of the relatively low fissile production rate in symbiosis, it is essential that the fission reactor portion of the economy be designed to operate at very high fuel efficiency.

A fusion reactor probably cannot be justified as an "Electric Breeder" if the benefits of electric breeding are computed in the narrow sense (i.e., by balancing total plant costs against fuel production rate).

Instead the total cost of the power producing system must be computed including such considerations as development and deployment of alternatives to the electric breeder, relative costs of various nuclear fuel cycles, etc. It may well turn out that power systems best suited to fission-fusion symbiosis must be based on the Th-U233 fuel cycle. A change to such a cycle, even if justified, would undoubtedly prove to be very difficult.

C. Augean Systems: The Battelle Northwest study of radioactive waste transmutation is the definite work to date. It was undertaken with objective of "surveying the neutron physics characteristics" of CTR transmutation and to "evaluate the potential of CTR as a Source of supplying neutrons" for transmutation. The first of these tasks has been done but the potential utility of fusion radioactive waste burners remains to be proven.

There are severe problems in the use of fusion reactors for high level radioactive transformation. The first of these is logistical. The Battelle Study shows that even at the admittedly unrealistically high flux levels assumed in their study, the Hazard Half-Life for the important fission products is in the range of 5-15 years. The time required for reduction to safe dispersal levels is five to ten times longer than this. It is easy to show, for any reasonable ratio of fusion burners to fission reactors, that the fusion burners would soon contain a much higher radioactive burden than the fission reactors themselves. This is certainly undesirable and probably intolerable.

The problem of inventory build-up does not occur with the actinide wastes. However it is possible to burn out actinides in fission reactors also. The LMFBR is particularly suited to such use because the requisite chemical separation would be reduced (the recycled fuel already contains such wastes) and the incremental increase in radiological hazard of the recycled fuel would be minimal. Even if it should occur that breeder reactors do not have sufficient neutron excess, it might still be preferable to use the excess neutrons produced by the fusion reaction to generate fuel for fission burner reactors. The systematic economics of the various waste disposal and storage schemes remain to be worked out.

V Low Q Systems - The "Electric Breeder"

Recent design excercises for both Hybrid and symbiotic systems have assumed the fusion portion of the system to be one of those developed for use as high Q stand-alone power sources. It is unfortunately true that some of these concepts seem capable only of low Q operation (thus, necessitating Hybrid power multiplication) but the fact remains that the configurations discussed were initially chosen because of properties unrelated to optimization of Hybrid or Symbiotic systems. Jassby's work on the optimization of the two component tokamak as a neutron source, reported in this conference, is consistent with this approach but also illustrates that the conditions in optimized fission-fusion systems may be very different from those of pure fusion systems.

Let us consider a fusion reactor optimized for the production of fissile material. The fuel production cost is composed of the prorated capital amortization charges of the system, the operating costs, and the fuel cycle costs. The first of these is the dominant expense in the systems based on the "standard" fusion schemes. Therefore, there is a possibility of substantial gain if capital expense and fuel handling complications can be traded off for increased operating costs. In essence, one looks for geometrically simple devices of modest scale capable of operation in the range Q = 0.1 - 1.0. Many such systems were proposed as possible fusion reactors but were abandoned when it was realized that they were incapable of achieving Q substantially greater than unity.

The stringent requirements placed on pulsed systems designed for power generation are eased in the low Q regime. Further, if the device is to be used only for fuel production, the average power density can be relatively low without incurring the usual economic penalties because heat transfer and reliability constraints that are reflected in costs are eased. In such a pulsed system, the predominant cost will be that of the energy storage reservoirs and the economic balance struck between their capital cost and the fuel production rate.

One possible example of such a low Q fuel producing system is a linear e-beam heated multiple mirror system [14]. Relativistic e-beams are capable of very efficient energy transfer even to dense plasmas. The plasma contained in the mirror system is to be used as the target for a pulsed ion beam injected along the axis of the system. Note that in this case it would not be necessary to generate a neutral beam so the beam energy can be optimized independently of neutralization restrictions. At a plasma density chosen so that the total length of the device is on the order of 100 meters, and at the temperatures consistent with two component operation, the mean free path is such as to render multiple mirror containment effective. Because of the low plasma temperature the magnetic field need not be inordinately high even at relatively high densities. A highly schematic version of such a mirror target breeder (MTB) is sketched in Figure 2.

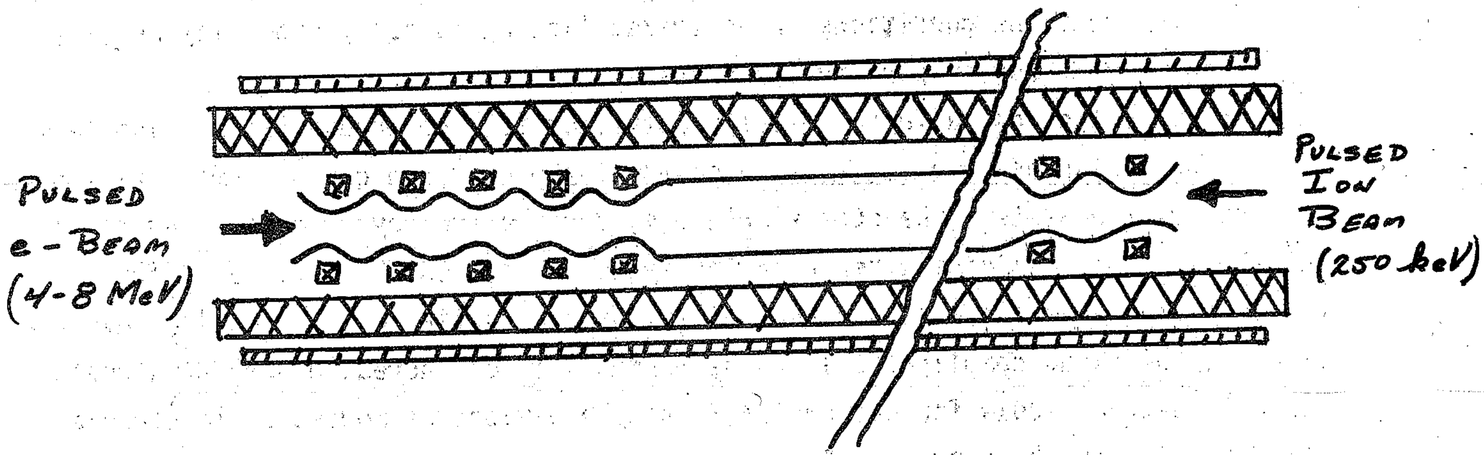


Figure 2.

The Multiple Mirror Target Breeder (plasma length ~100 m., plasma diameter ~1 m.) - one example of a technologically straightforward low Q fissile breeder.

Clearly, there are many other low Q systems that might lend themselves to service as neutron sources for fissile material production. It is possible to think of the MTB or any of these others as possible drivers for Hybrid blankets, producing both power and fissile materials. However, the arguments applied to steady-state Hybrid reactors apply even more strongly when applied to pulsed reactors. A power producer would also be more strongly effected by considerations of availability and siting.

Economic analysis of such low Q electric breeders has barely begun. However, the potential advantages of possible trade offs are obvious. Even in the event that a high Q "stand-alone" fusion reactor is perfected, there will no doubt be a large and possibly still growing fission reactor economy in existence. The electric breeder would operate in symbiosis with fission during the transition from a pure fission to mixed and possibly a pure fusion system.

REFERENCES

1. B. R. Leonard, Jr."A Review of Fusion-Fission (Hybrid) Concepts" USAEC Report BNWL-SA-4706 p. 31. - This report was published in slightly revised form in Nuclear Tech., <u>20</u>, pp. 161-178. (1973).

2. L. M. Lidsky, "Fission-Fusion Systems: Hybrid, Symbiotic and Augean", Nuclear Fusion, (1975).

3. M. E. Wyman, "An Integral Experiment to Measure the Tritium Production from Li-7 by 14 MeV Neutrons in a Lithium Deuteride Sphere", USAEC Report LA-2234 (Rev) Los Alamos Scientific Laboratory (1972).

4. California Research and Development Company, "Proposal for a Driven Thermonuclear Reactor", USAEC, Report LWS-24920 (Rev), (1953).

5. D. H. Imhoff, W. H. Harker, F. E. Bjorklund, H. A. Brammer, and A. E. Stoddard, "A Driven Thermonuclear Power Breeder", California Research Corp. Report CR-6 (1954).

6. J. D. Lawson, "A Survey of Some Suggested Methods of Realizing Fusion Reactors", United Kingdom Atomic Energy Authority, Report A. E. R. E. GP/M 185 (1955).

7. J. D. Lawson, "Power from Nuclear Fusion", Nature <u>180</u> pp. 780-783 (1957).

8. L. G. Barrett, "A Fusion-Fission REactor", USAEC Report KAPL-M-LGB-14 (1957).

9. P. C. Thonemann, M. J. Poole, J. D. Lawson, and J. H. Tait for the United Kingdom Atomic Energy Authority, Patent Application 1920/57, ser. 830, 253 The Patent Office, London. (1957).

10. B. R. Leonard, Jr., "A Hybrid Neutronics Analysis", Pacific Northwest Laboratories Annual Controlled Thermonuclear Technology Report for 1972, USAEC Report BNWL-1685 pp. 18-32 (1972).

and see also:
B. R. Leonard, Jr. and W. C. Wolkenhauer, "Fusion-Fission Hybrids: A Subcritical Fission Lattice for a D-T Fusion Reactor" in <u>Technology of Controlled Thermonuclear Fusion Experiments and the Engineering Aspects of Fusion Reactors</u>, USAEC CONF-721111, pp. 918-930 (1974).

11. J. D. Lee, "Neutronics of Sub-critical Fast Fission Blankets for D-T Fusion Reactors", in Proceedings of the 7th Intersociety Energy Conversion Engineering Conference, also published as USAEC Report UCRL-73952, San Diego, California.

12. L. M. Lidsky, "Fission-Fusion Symbiosis: General Considerations and a Specific Example" in Proceedings of British Nuclear Energy Society Nuclear Fusion Reactor Conference at Culham Laboratory, 1969, pp. 41-53 Culham Laboratory Report CLM-NFR. (1969).

13. W. C. Wolkenhauer, B. R. Leonard Jr., and B. F. Gore, "Transmutation of High-Level Radioactive Waste with a Controlled Thermonuclear Reactor", USAEC Report BNWL-1772 (1973).

14. G. I. Budker, "Thermonuclear Fusion in Installations with a Dense Plasma", Proceedings of Sixth European Conference on Continued Fusion and Plasma Physics, (1974).

ᵔd,

QUESTIONS ABOUT SECOND PRESENTATION

Dean: There was a meeting similar to this about a year ago at Princeton. I am trying to recollect some of the things that were said at that meeting, and I am wondering if you could sort of summarize your view as to anything that has happened in the past year for these systems that may have changed your outlook. In particular, I kind of remember a lot of emphasis placed last year on the engineering complications of fusion-fission designs. Just the geometrical arrangements of things which, from the engineer's point of view, is felt could be very difficult to engineer in a way to compete with fission reactors.

Lidsky: I agree with your comment. I promised that I wouldn't, for the duration of this talk, be a proponent of one scheme or another, but I'm glad you've asked the question now. I agree with you that the hybrid system is subject to the pejorative statement that it can in many ways combine the worst of two systems. And if you're not careful it will. The worst problem is that one has to build a multiplying assembly with a geometry you would not ordinarily choose for a multiplying assembly. Of course a lot of the constraints compete from the point of view of engineering; in cooling, in fuel handling, and in other systems of that sort. One still is subject to the hazards of loss of coolant accidents in such a multiplying assembly, because it is producing copious amounts of power. It is effectively a fission reactor and the most dangerous hazard of fission reactors is the Loss of Coolant Accident. The hybrid is more liable to this failing because the geometry is not of one's choice. It has to be compatible with the fusion machine, tends to have large holes and provisions for access built in and it almost certainly would be penetrated by the many access requirements of the fusion reactor itself. So, there are severe engineering problems; more severe than for the ordinary fission reactor. I think there is no gainsaying that. The question then becomes "are the potential gains that one gets sufficiently large to overcome the engineering difficulties?" I think that this is the question I posed last year and I tend to think that they are not, but that is personal judgement and the engineering really needs to be done to see whether in fact something could be done to alleviate the difficulties.

Maniscalco: I would like to point out that for the two separate systems, which you call the augean and the hybrid systems, that perhaps there may be room to combine the two. A number of the actinides that are put out in the fission economy have fast fission cross-sections which are actually larger than U-238. This may make a hybrid system using some spent waste or actinides a lot easier to engineer.

Lidsky: O.K., except that I would like to take exception to your possibly combining that with the symbiotic systems. I'm trying very hard to maintain at least one "column", the symbiotic one, for listing devices in which no fissions take place at all and therefore in which one need not worry about afterheat accident. In fact the burning of actinides in a hybrid or augean blanket would be perfectly compatible with the definitions of both these schemes.

Maniscalco: O.K., then just one other observation. By definition then, your systems in which no fissions take place would have be a molten salt reactor.

Lidsky: No. It would be quite easy to take advantage of the high temperature gas reactor technology; in fact, it might even be easier.

Maniscalco: Well, once you start building up U-233, how do you prevent fission?

Lidsky: What one has to do, and this is where the molten reactor system works better than the HTGR system, is to move the fuel through at quite a rapid rate. If one employed the HTGR system for the fusion part of the blanket, a pebble bed or equivalent scheme to move the fuel through fairly rapidly would have to be devised. On the other hand, one advantage of symbiotic schemes is that one can run a fusion reactor of one sort, say with a salt blanket, and feed the generated uranium into fission reactors of another sort, say HTGRs. The point is that only the fuel is transfered from one device to another, and so each system can be designed for the advantages you would like to engineer into that system alone.

Baker: Concerning the study you did a few years ago which had a salt-thorium solution, did you flow that solution through the system?

Lidsky: Yes.

Baker: Did you worry about taking out some of the intermediate steps between thorium and U-233?

Lidsky: Yes. It's not necessary to remove the intermediate products, because the flux is so much lower in a fusion blanket than it is in a fission reactor where proctactinium probably must be removed. Here, it makes a very minor difference and the U-233 burnup turns out to be negligible.

Wolkenhauer: I think it's quite appropriate what you're doing, to try to identify one very clean system, and I think that is an appropriate approach. Also, I think it's fair to point out that you're producing something that you ultimately intend to fission so, in effect, you're viewing this as a fuel management problem. You're simply not going to have your fissions here, you're going to have them someplace else. It's the whole system that is important; you want to maximize the safety and whatever of the whole system.

Another point that I thought is well taken was your point on the geometry and that you are, in fact, selecting a geometry which from fission reactor physics you wouldn't have selected. But it also has some slight advantage in that this geometry which is very leaky, at least in the case of the mirror machines we looked at, turns out to have some advantage in the loss of coolant situation.

There is another point I'd like to make which we are probably responsible for perpetrating the literature. You very carefully point out that our systems are very sub-critical which is quite correct, but those are hot values of k_{eff}, which is the regime within which you want to do your calculations. It's fair to point out that as these systems cool down, and our particular system has a very large reactivity defect in it, that one is very close to critical at shut down. That should be recognized.

Lidsky: There are many similar points in this field that I didn't get to discuss. For example, the systems that breed plutonium (especially those that breed it at high gain) tend to breed in hot spots, and the place where the plutonium is, tends to be the place where the plutonium grows. In fact, it may very well be a severe control problem that must be engineered out. Such problems are particularly difficult in high gain systems. However, I don't think that it is fair, until much more engineering has been done, to put these systems down strongly, simply because they are subject to such hazards and problem. However, it is very important to realize that they exist. It's just another example of the problems you can get into if you try combining two technologies without being quite sure how they are going to interact.

Anonymous: Do you have any numbers as to how fission-free a symbiotic system could be in view of the economics of fuel application and exposure requirements?

Lidsky: It's a little bit difficult to work out some of these numbers now. One fact is plain though; one can reduce the U-233 fraction in circulating salt to several parts per million with very simple processing techniques. So at least insofar as the recycle U-233 is concerned, one can keep that down to extraordinarily low values. It is also possible to develop fission products by the fast fission of thorium. One avoids this by ensuring a thermalized neutron spectrum where the thorium is. But I am vague as to how "thermal" the spectrum can be. The fission rate depends on the tail of the neutron distribution and this is hard to calculate.

Anonymous: Are you really talking about exposures of your fuel elements of a few days or a week before they are taken out?

Lidsky: Yes. That is why I prefer to talk about the salt-based fertile system rather than the HTGR system. One has effective exposures that are unmeasurably small and a very small salt recycling time. These considerations can be very successful in keeping U-233 out of the system.

Lee: Larry, I would like to set the record straight. Early in your presentation, you somewhat misquoted my 1970 work.

Lidsky: I apologize.

Lee: The figure you showed for the mixture of fuel, structure, and lithium should have an energy generation of 100 Mev, not 220. The numbers you gave went with the pure uranium and thorium cases.

Lidsky: I think I said that, but if I did misquote you, I apologize.

Coffman: I would like to comment on the subject of transmutation. It would seem that all told, the economic tradeoff is one of what will it cost you to bury 99 percent less actinides versus the cost of getting a 99 percent conversion in a fusion reactor. I guess my point is that the Nation is going to have a federal repository if for nothing else than to dispose of the LSA plutonium wastes, and hence that repository is going to be there to receive actinide waste. Assuming you can burn up 99 percent of your actinides in a reactor, the question is, is it cheaper to mine out 95 or 100 times as much salt and bury all of the actinide wastes in a permanent repository or to burn it up in a fusion reactor. My simple logic tells me that if you look at the costs and safety you will have to decide against introducing actinides into any kind of reactor system. It's extremely cheap to mine salt in terms of kilotons per year.

Lidsky: You raise, I think, a very good point but there is an assumption built into that point. You assume that salt mines will be an acceptable way of getting rid of waste. There is another assumption one tends to make that equally strongly colors one's thinking. For example, it is an assumption that the LMFBR will work. If you believe that, then there is a very strong economic structure that can be erected based on that assumption. On the other hand, if you believe a successful LMFBR is open to question, then a very different situation prevails and many more things are open for discussion. Except for that caveat, I think that the point is very well made.

Coffman: I have one question about the size of the blanket. I got the impression that for a fusion blanket, the blanket size compared to the LMFBR blanket would be quite a bit larger. Is that not true or are they comparable in size? The reason I asked the question is because the fuel reprocessing costs are a very substantial part of your fuel cycle, and if the fusion blanket is required to be an order of magnitude larger, you have an extremely large penalty there and that hasn't been talked about. So my question is, what is the comparative size of the blankets for fusion reactors versus an LMFBR?

Lidsky: They tend to be bigger, several meters thick for hybrid blankets. Furthermore they work at a lower power density. One can hope to minimize some of the fuel processing costs in a hybrid system by going to a continuous processing scheme like molten salt. This is a nontrivial point that has often been discussed, but not at length in the literature.

Wolkenhauer: I have two last big points to modify your answer a bit. The size of the fusion blanket used in the hybrid is pretty much strictly dependent upon the fission technology embraced. It's a matter of how many megawatts per cubic meter you want to pull out of the blanket. Some you can pull more out of than others. HTGR blankets tend to be large, and fast systems tend to be smaller. Another point I'd like to make, for the AEC's benefit, is that I was struck by the fact that for all of the work that Larry reviewed, and it turns out to be an extensive amount of work, my guess is that the integral under the cost curve for all that work is much less than half a million dollars.

<u>Lidsky</u>: Substantially less.

ECONOMIC REGIMES FOR FISSION-FUSION ENERGY SYSTEMS

D. E. Deonigi

January 1975

BATTELLE
PACIFIC NORTHWEST LABORATORIES
RICHLAND, WASHINGTON

ECONOMIC REGIMES FOR FISSION-FUSION ENERGY SYSTEMS

The Objectives

The objectives of this hybrid fusion-fission CTR economic regimes study are to:

- Define the target costs the hybrid must meet.
- Define the optimum fissile/electrical production ratio for hybrid blankets.
- Discover synergistic configurations.
- Define the windows of economic hybrid design having desirable cost/benefit ratios.

The Method

These objectives can be achieved by comparing the CTR and CTR hybrid costs and performance with alternatives. The logical alternatives for this study include the Liquid Metal Cooled Fast Breeder Reactor (LMFBR) which like the hybrid is a fissile fuel producer, fossil plants, and the LWR and HTGR reactors.

The Model

The PNL electrical generation decision model,[1] using linear programming techniques, has the logical alternative already characterized. It was used in the recent LMFBR cost benefit analysis included with the programmatic environmental statement.[2]

Some of the decision model features which make it valuable for such an analysis are:

- A minimum cost objective function is available (the stated long-term electrical utility objective).
- It utilizes 35 two-year time increments covering 1970 to 2030 (a 70-year span).

- Complete fissile accounting is used including a stockpiling feature.
- Fossil fuel prices are determined by demand and supply as encompassed in the linear programing marginal or shadow price system.
- Fuel cycle processing costs are modeled for reasonable market penetration of each alternative technology.
- Fossil fuels are represented on a regional basis. Both base and intermediate power loads are included.
- Limits on the rate of new technology introduction are included in the model.
- Limits are also placed on the rate of old technology phase-out.
- It has capacity to handle CTR and CTR hybrid plants in addition to the existing plants.

Of particular interest to this analysis is that the value of fissile material produced by the hybrid plants is an outcome and need not be treated as a parameter.

The Data

Some of the basic input data strongly affects the benefit-cost results. The discount rate is an important parameter because the costs are being encountered now and the benefits will primarily accrue a generation hence. Thus, introduction dates for various technologies are important as is the demand growth pattern assumed. These and other factors are listed in Table I.

Table I

BASIC INPUTS

Discount Rate	8%/Yr
Electrical Energy Demand	WASH 1139, Case D (5.6% annual growth)
System Capacity Factor	0.57, Fraction
Introduction Dates	
HTGR	1978
Hybrid	1986
FBR	1988
Phase In Capacity Constraints For New Technology	
First Biennium	4 GWe
Second Biennium	8
Third Biennium	16
Fourth Biennium	32
Fifth Biennium	No Limit
Fissile Material Storage Cost	0.75 $/g-Yr

Capital costs are assumed to decline as a function of time due to evolutionary improvements and cost reductions from plant size increases (according to Table II). The base case is assumed not to contain any CTR's. Case H-1 uses an assigned capital cost for the CTR hybrids which would achieve a market penetration but not displace the LMFBR.

Lowered capital costs were assumed for the CTR hybrids for case H-2. It was expected that these lowered costs would result in substantial entry of CTR's which was precisely the result attained when the calculations were carried out.

Table II

CAPITAL COSTS, $/Kw

	1980	1990	2000	2010→
Base Case				
LWR	404	388	374	360
HTGR	405	389	374	360
FBR		458	434	410
Case H-1				
Hybrid Power*		638	614	598
Case H-2				
Hybrid Power*		575	561	548

*Hybrid fuel factory at equivalent cost per kWt.

The decision by a utility to utilize a particular powerplant technology is based on estimates of both capital and spending. These costs may be summed to a single cost equivalent by adding the present worth of future spending costs to the capital costs. The following equation describes this relationship which is used by the decision model to evaluate minimum cost energy systems.

$$30 \text{ yr plant cost} = \frac{\text{Capital} + \sum_{1}^{30 \text{ yr}} \text{PW(OM + IC + Fuel)}}{\sum_{1}^{30 \text{ yr}} \text{PW(Energy Produced)}}$$

The Results

Analysis of the hybrid CTR is more complex than the CTR alone because of the interaction with the HTGR and LWR reactors. The interactions occur because of the use of the fissile species produced in the hybrid reactor to power the HTGR's and LWR's.

Although either ^{233}U or plutonium could be produced in the hybrid blanket, this analysis only considered plutonium because of the lack of complete fuel cycle data of ^{233}U use in LWR and FBR's. In general, it would be expected the ^{233}U would be a superior fissile fuel in thermal reactors. Complete accounting for the fissile and fertile species includes annual flow for each isotope. The mass balance data used in this study is approximately that shown in Table III for the plutonium users and for the LMFBR in Table IV. The hybrid CTR mass balance data is shown in Table V. Negative figures in the annual inventory charge indicate consumption and positive figures indicate generation.

As a result of the interaction between CTR hybrids and LWR's and HTGR's (see the mass flow data), a power producer CTR hybrid of 1000 MWe can support the fuel needs of approximately four LWR's or HTGR's. In the case of the HTGR's the ^{233}U produced more than offsets the greater consumption of ^{239}Pu compared to the LWR. The fuel factory will, on the other hand, support more than eight LWR's or HTGR's.

Table III

MASS BALANCE OF Pu USERS, kg
(75% Capacity Factor, 1000 MWe)

	PWR		HTGR	
Initial Inventory	U^{235}	135	Th^{232}	7,680
	U^{238}	67,530	Pu^{239}	1,975
	Pu^{239}	1,621		
Annual Inventory Charge	U^{235}	- 28	Th^{232}	- 253
	U^{238}	- 1,015	U^{233}	+ 98
	Pu	- 492	Pu^{239}	- 564
Final Inventory	U^{235}	84	Th^{232}	6,916
	U^{238}	64,360	U^{233}	280
	Pu^{239}	1,540	Pu^{239}	328

Table IV

MASS BALANCE OF FBR, kg
(75% Capacity Factor, 1000 MWe)

		Advanced Oxide
Initial Inventory	U^{235}	350
	U^{238}	46,600
	Pu^{239}	1,800
Annual Inventory Charge	U^{235}	- 12
	U^{238}	- 1,560
	Pu^{239}	+ 335
Final Inventory	U^{235}	83
	U^{238}	43,281
	Pu^{239}	2,260

Table V

MASS BALANCE OF HYBRIDS, kg
(75% Capacity Factor - 2500 MWTh)

	Power Producer (1,000 MWe)		Fuel Factory	
Initial Inventory	U^{235}	3,200	U^{235}	800
	U^{238}	457,200	U^{238}	114,300
Annual Inventory Charge	U^{235}	- 100	U^{235}	30
	U^{238}	- 2,600	U^{238}	4,840
	Pu^{239}	+ 2,000	Pu^{239}	4,370
Final Inventory	U^{235}	3,080	U^{235}	800
	U^{238}	452,500	U^{238}	114,100
	Pu^{239}	1,300	Pu^{239}	2,800

Table VI

ELECTRICAL GENERATING CAPACITY BUILT*, GWe

Year	LWR	HTGR	LMFBR	Fossil	Hybrid	Total
			Base Case			
1970-79	78	0.3		167		246
1980-89	224	56	4.4	164		448
1990-99	427	120	217	47		811
2000-09	346	577	849	303		2,075
2010-19	261	496	2,365	1,310		4,432
2020-29	271	699	5,083	3,435		9,488
			Case H-1			
1970-79	78	0.3		167.4		246
1980-89	210	67		159	12	448
1990-99	182	345		25	245	797
2000-09	42	1,448	185		400	2,075
2010-19	42	2,613	778		999	4,432
2020-29	42	4,686	2,971		1,789	9,488
			Case H-2			
1970-79	78	0.3		167		246
1980-89	204	69		163	12	448
1990-99	183	341		23	264	811
2000-09	42	1,604			429	2,075
2010-19	42	3,219			1,171	4,432
2020-29	42	5,243	1,828		2,375	9,488

*Includes replacement of capacity after 30 year plant life time.

The matrix of powerplants selected by the decision model to satisfy the power demand reflects the solution which satisfies the constraints imposed. While stockpiling of ^{239}Pu is allowed, interest must be paid on the stockpiled material and a storage charge is also assessed. The effect is to keep the stockpile to a minimum. Table VI shows the base case with LMFBR's operated at base load and fossil fuel plants at intermediate load supplying a major part of the long-term power needs. HTGR's and LWR's supply only 10% of total demand.

When the hybrid is introduced in case H-1 and case H-2, they generate enough plutonium to allow building a much larger number of HTGR's or LWR's. This could be a sizable benefit in that future fissile fuel supply would be assured for the HTGR and LWR. Thus, capital risk could be reduced. Also, if power cost from the CTR hybrid were to increase 25% above case H-1 values, the power cost from the total power net might increase only by around 5-8%.

The shadow price or implied transfer price for plutonium reflects the dynamics of supply and demand and available technology. During the startup period of the LMFBR's in the base case, the price rises to more than $25/gram fissile by 1990 (see Table VII), and continues to rise because of the relatively low breeding rate (see the mass balances in Table IV). By 2010 the price of plutonium rises to $52.73 in this base case and remains at $20.24 in the year 2030.

Table VII

PLUTONIUM VALUE, $/g

Year	Case H-1	Case H-2	Base
1974	2.08	2.27	3.42
1980	5.90	6.23	8.15
1990	16.97	16.49	25.97
2000	9.76	6.26	45.45
2010	7.75	4.73	52.73
2020	7.24	5.11	40.90
2030	3.86	2.89	20.24

In cases H-1 and H-2 where the CTR hybrid is allowed to enter, the plutonium price rises to only $16.97/gram in 1990 and declines thereafter as CTR hybrids become established in the market place. At the point of initial market penetration the first few plants would enjoy the $50/g price. However, the economic benefit to the nation would be very small. Only when plutonium is sold and produced at lower cost are savings of a magnitude to offset R&D investments.

From the previous discussions, it becomes reasonably obvious that the total overall system must be considered in order to assess benefits. The benefits for the case H-1 are $10 billion after discounting @ 8% and for case H-2 (where the capital costs were assumed to be reduced for the CTR hybrid) they are $14 billion.

The fuel factory was never selected as part of the least cost energy supply system.

The capital costs level necessary for the fuel factory to be selected was found to be about $270/kW. Thus, it appears that a substantial amount of the hybrid energy output must be captured to make the technology useful.

The results of the benefit analysis are summarized in Table VIII. Because of the difference in timing, the benefits are summed after being present worthed to 1974. The costs are also summed by the present worth methods using the same 8% discount rate.

Table VIII

HYBRID BENEFITS, BILLIONS
(@ 8% Discount Rate)

Case	System Costs	Base Case - Hybrid Case
Base	323	
H-1	313	10
H-2	309	14
H-1 Delayed	315	8

Benefit-Cost Ratios of the Hybrid

Substantial benefit-cost ratios (~3÷1) result from the hybrid CTR program. At the assumed capital costs for case H-2, the cost of the hybrid CTR program is approximately $10 billion, which reduces by present worth discounting to about $3 billion (@ 8%).

Case H-2 shows a present worth of future benefits of about $14 billion giving a benefit to cost ratios of about 5. The capital costs of the hybrid were assumed to be $575/KWe at 1990 decreasing to $548/KWe at 2010 and beyond (due to the learning curve).

Case H-1 was recalculated with a 10 year delay (1996) in hybrid technology introduction date. The system costs were increased from 313 to 315 billion which reduced the benefits only 20%. The LMFBR market share expanded to replace the hybrid during the delay period. The plutonium value increased about $10/g thru year 2010 then assumed the case H-1 values.

The information obtained from this analysis is better seen in perspective as shown in Figure 1, where the assumed capital cost in $/KW is shown as a function of the fissile fuel production. The CTR without breeding blanket is shown at the left. This information resulted from the CTR economic regimes study reported in the January-June 1974 PNL "Interim Report on Controlled Thermonuclear Reactor Technology," August 1974.

The fuel factory CTR which would generate about 4,200 kg/yr of Pu is shown on this figure. It reflects the reduced values obtained from cases H-1 and H-2 even though they didn't enter the solution.

The FBR is the natural reference point for the study since it is the alternate means of satisfying energy demands in the future. The $460/KWe capital costs shown as a dotted line yielded satisfactory benefit-cost ratios in previous studies of the LMFBR.(2)

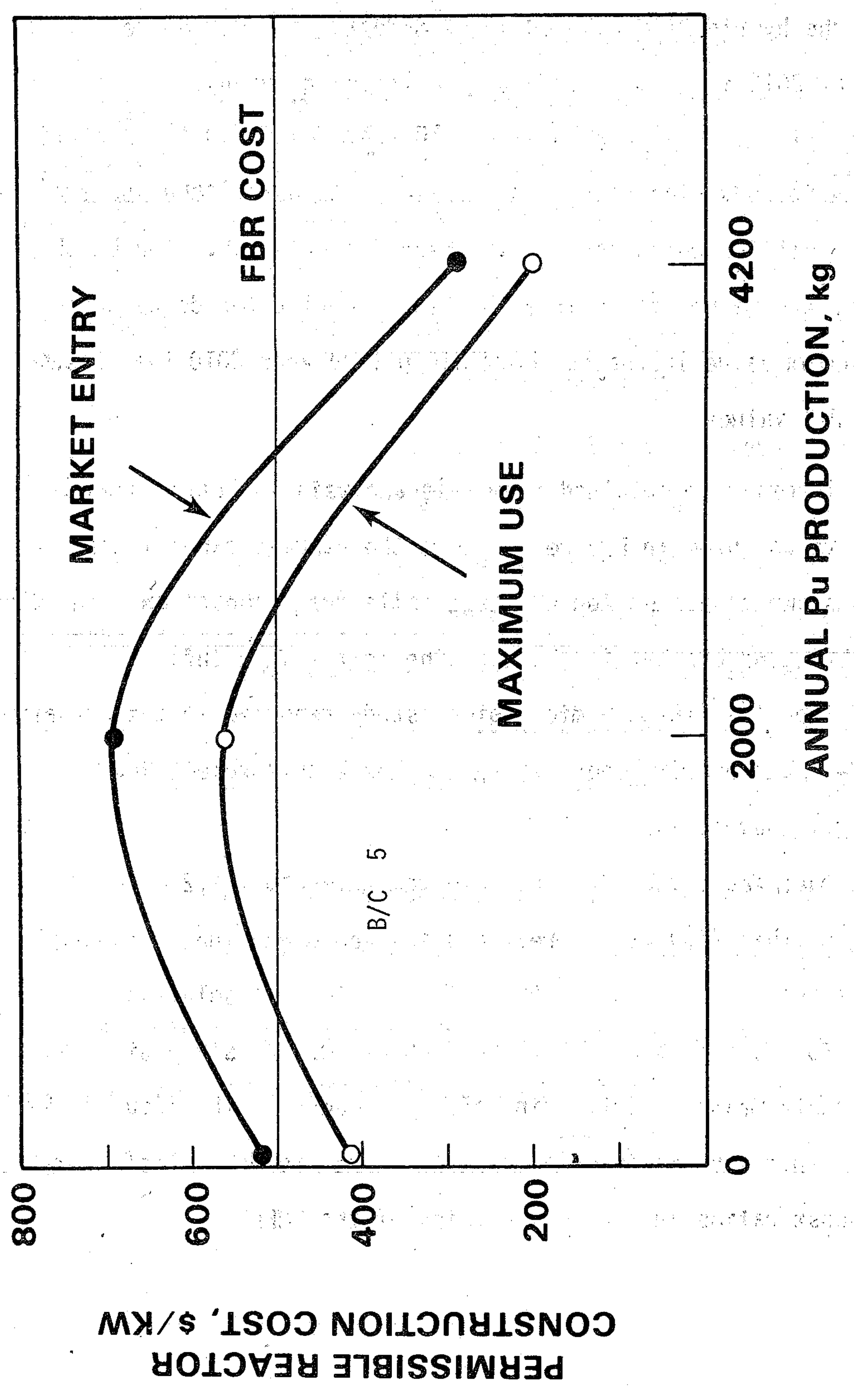

MARKET ENTRY
FBR COST
MAXIMUM USE
B/C 5
800
600
400
200
0
2000
4200
ANNUAL Pu PRODUCTION, kg
PERMISSIBLE REACTOR CONSTRUCTION COST, $/KW

FIGURE : 1

Future work is planned to better define the economic windows of opportunity for the hybrid CTR. Additional values are needed to better define the relationship of allowable capital costs and fuel to energy production. Synergetic combinations and optimum values can result. Important parameters such as introduction date and electrical demand levels will be varied to determine their effect on the benefit-cost ratio obtained.

In summary, the analysis of the economic regime for fission-fusion energy system determines that:

- Target cost of hybrid CTR of 10 to 20% over LMFBR was established.
- The optimum fissile production would be about 1500 kg/yr from a 1000 MWe plant.
- Hybrids designed primarily as fissile material producers are not economical.
- CTR hybrid can synergetically support 3 to 4 thermal reactors.
- For the CTR hybrid to achieve described R&D cost returns it must produce plutonium a $5 to 10/g.

It must be clearly noted that the hybrid could certainly be a step along the path to pure CTR implantation which would add significantly to the overall program benefits.

REFERENCES

1. R. L. Engel, DAEDALUS II: A Computer Code to Generate a Linear Programming Model of a Nuclear Power Economy, BNWL-1459, Pacific Northwest Laboratory, Richland, Washington, June 1970.

2. Environmental Statement. Liquid Metal Fast Breeder Reactor Program, Vol. III, United States Atomic Energy Commission, March 1974 (Draft).

ACKNOWLEDGMENT

The author wishes to thank W. C. Wolkenhauer, R. L. Engel and R. L. Watts for the technical assistance to this research.

QUESTIONS ABOUT THIRD PRESENTATION

Grace: As I recall, the cost-benefit of the LMFBR program over the next 50 years will vary from 60 billion to 0 depending on the set of assumptions used, e.g. power demand growth rate, cost of uranium, dates of introduction of the different machines and the success of the HTGR program. Have you fully exercised your LP programs to determine the sensitivity of your results to all these different assumptions that went into it?

Deonigi: No, we haven't. In fact, we have two data points.

Grace: OK, but for one of the data points you mentioned, did you say the cost-benefit over 70 years was 14 billion and that was the H2 case which assumed a very late introduction of the LMFBR?

Deonigi: No, the introduction dates in both H1 and H2 are the same. There was no shift in introduction dates, we simply lowered the capital costs about $60 a kilowatt to achieve greater market penetration. It appears something on the order of 20 percent more than breeder costs could be sustained by a hybrid system and be a productive element.

Shapiro: If I understand what you said correctly, the value of the introduction of a particular system depends to a large degree on the date of introduction of that system and the cost of the work required to reach that point. As I recall the criteria that you used, you indicated the introduction of hybrid in 1986. Is that correct?

Deonigi: Yes.

Shapiro: You indicated results at a very optimistic date. What do you consider to be a realistic date for introduction and what would that then mean to your cost-benefit analyses?

Deonigi: The realistic date would certainly have to be in the 90's someplace. We introduced the hybrid at the same time as the breeder was introduced to put them on the same time scale. The effect of a later introduction substantially reduces the benefits because of the present worthing and thus would increase the necessary performance, or reduce the allowable cost, maybe to LMFBR costs or lower.

Shapiro: How realistic do you consider your projection at such a low capital cost, considering the fact that you are combining the technologies of the fusion and fission reactors?

<u>Deonigi</u>: The role here wasn't to define the feasibility of achieving it but what the target is so the designer can look at it from a practical standpoint and say that's not achievable. If it's not, then let's scrap it. Putting the two technologies together may not be a feasible thing to do. If it's going to be more expensive than the LMFBR, maybe, or more expensive than either technology separately, then it looks like a difficult task. I essentially avoided that by trying to define what is needed to make a contribution against these other systems. I'm really not an expert in capital costs myself and it's not for me to say if that's feasible or not. That's up to the designers.

<u>Hertzberg</u>: You have shown, and in fact the first two questions have stressed, that everything depends on a set of assumptions. If you go one step further behind this set of assumptions, there is a technology built up to support a set of assumptions. Now I'm just curious as to your feeling about how valid you think the technology assumptions are that you made in designing the hybrid system. I do believe one of the purposes of our meeting is going to have to be sooner or later to get down to the question about what we are going to have to find out before any of these very elegant numbers become real.

<u>Deonigi</u>: I defer to Wolkenhauer as to the design characteristics because he supplied them to me in this project. I agree with you that this really is the task at hand, trying to establish what the characteristics are and what they are liable to cost. The physical reliability of the design itself, I didn't have much to do with, and Bill Wolkenhauer can speak to that.

<u>Wolkenhauer</u>: I guess, the question is how valid are the assumptions. How easy it is going to be to build a hybrid, hopefully, will be a discussion that will stretch over the next decade. My guess and this is only my guess, is that the problem will go back to the plasma physics. We have attemped to do some of these cost analyses and capital cost calculations and so forth. We have fair confidence that we can price the fission part of the hybrid with some accuracy, and the accuracy depends upon the fission technology one selects. Obviously, if one uses HTGR technology, one has a little more confidence in the numbers than if LMFBR technology is selected. My guess is that the big unknown, whether or not you can make this $540 figure and including the operating costs, is largely based on what the plasma physics portion of the device comes

out to be. We have done some extrapolations based on the cost analyses done for both the UWMAK design and the Princeton design. Crudely taking those reactors and adding fission blankets to them indicates that maybe you could achieve it. We hope to be able to publish some of these very preliminary, speculative kinds of numbers soon.

Deonigi: This morning there have been a number of papers that all indicate possible annual fissile material production in the 2000 kilogram range. I believe each of the papers were in that range at least, 1500 or 1300. I don't know about the 4000 number.

Holdren: I want to take issue on philosophical grounds with one of your premises, and it's related to this question of sensitivity to assumptions. Your premise was that if we had, say, a zero energy growth economy, then none of us would need to be here today. I submit that even in a stablizied energy economy we would want to ask the question, 'what is the best way to meet the stabilized demand in terms of economic costs, environmental costs, and social costs.' We would still be looking at a mix of technologies and trying to find what kind of mix was optimum. There is a part of the analysis which is extremely sensitive to growth rate, and that is the economics of the importance of the breeding ratio, as I mentioned earlier today. If the growth rate is lower than the one you postulated, the importance of breeding ratio enormously diniminishes, and the relative importance of environmental and safety considerations increases by virtue of that. Of course, one of the things that is missing and very difficult to include in this sort of cost benefit analysis is precisely this factor; the environmental and safety considerations. But certainly a system that is unacceptable to the public on these grounds has, in a sense, an infinite cost associated with it; one cannot build it. And, I submit that one is going to have to get around to elevating the priority attached to these environmental and safety considerations, lest we otherwise throw out at an early stage the principal reasons we might want to have a hybrid at all.

Deonigi: I agree with you.

Coffman: I wonder if you could try to scope in some things that bother me. The first thing being the base case for plutonium cost which you projected out to \$20 a gram. I was under the impression that the LMFBR program projects \$2 a gram out in that time range, or at least approximately a order of magnitude lower than your assumed base case. Is that wrong?

Deonigi: No, that is true as far as I know. I think it depends on which case of their's you choose to look at. It is, again, sensitive to the breeding gain. They have a much higher breeding gain for the LMFBR. I assumed a 10 year doubling time and they're running about six.

Coffman: Let me cap up my concern. It looks to me like that in this economic analysis base case, you assumed all the pessimism about the LMFBR. You assumed that you froze the fuel design, you didn't go to carbide fuels, you froze off the LMFBR design, and assumed that you make no technology advancement. You assume an order of magnitude different base case for plutonium fuel costs in the year 2000.

Deonigi: No, that's not an assumption.

Coffman: You take those three together and switch them back to what perhaps is in the LMFBR environmental statement, and I think you will get a totally different answer.

Deonigi: The plutonium is an output not an input. I think if you look at a comparable case in the LMFBR study, you will find comparable prices for plutonium evolved. The initial work under the LMFBR program was at 8 percent discount rate. They later moved it up to 10 percent when they were ordered to by OMB, I understand, but they are now, I guess, going back to the 8 percent again to try to rejustify that level of discount rate. The plutonium value, though, is an output which is a function of those inputs. The breeder that I have in there is a substantial improvement over their early designs that they have. It's about mid-range of what they come up with. It's the best of the oxide designs. I simply didn't go and add the six year doubling carbide design which apparently has some physical core design problems that they are trying to wrestle out, even now, to simply get the heat out, under that higher specific power.

Moses: You gave a realistic date and you also gave a starting date the same as the LMFBR. Under the same assumptions and models that you used, what would be the latest date one could introduce hybrids and remain competitive with the LMFBR?

Deonigi: What would happen basically is that the same benefits at the new introduction date would be achieved. What would happen is that the target costs would drop. If I held the present cost that I assumed and delay the introduction, the benefits would be substantially reduced. The reason for starting with this date was that I really intended to run more than one introduction date and that was the beginning of the series. It just took longer to accomplish than I anticipated. Possibly by the time the proceedings of the meeting are published, we may be able to supply more data points. But right now, the benefits decline fairly rapidly. I think if you look at the breeder numbers and if you look at the rate of decline in benefits with introduction date, it would follow a very similar pattern.

Moses: Then would you say the realistic date you project is probably too late?

Deonigi: No, I don't think that is the case.

Furth: I'd like to make one rather basic point on this cost benefit analysis. And that is, I think it will make a lot of difference whether you think that the fusion-fission hybrid is the end in view or a step on the way of a fusion reactor. I think this is very important. And, I think it would be an interesting expansion of your work if, for example, you assumed that straight fusion reactors would be available at such and such a year and that it is a matter of national policy as we go beyond the year 2000 to infinity, to go to fusion reactors rather than LMFBR plutonium breeders. In that case, I think the cost-benefit picture for introducing hybrids rather than straight LMFBR breeders as an intermediate stage would alter very greatly. In the present analysis it seems that if hybrids were to become available only, say, in the late 80's or 90's, then if we wait a few more years, they lose all interest. But if all the time the long range aim is to run a fusion economy, I think it will come out quite differently, and instead of paying a penalty for beginning to introduce the new fusion technology, you are in fact paying a penalty for going into a LMFBR breeder technology excursion when ultimately you're going back to fusion.

Deonigi: That's right, and the breeder (LMFBR) cost benefit work has this same problem. Usually their first so-called commercial plants aren't really economically viable at the time of introduction and you might be describing that the hybrid might fall into that some category, as a transitionary step to a full CTR. One of the data points on that last slide shows you the "CTR only" entry and it indicates a cost somewhat less than the breeder cost, in order to achieve the same level of benefits. The reason is that the breeder does have some benefit by operating,and supplying excess plutonium to cheaper burners in the system. There are some less expensive burner reactors available in the system we're dealing with, and therefore it's desirable to produce a little bit of fuel for them at the same time you're running CTR's.

BNWL-SA-5304

REVIEW OF BATTELLE-NORTHWEST TECHNICAL STUDIES ON FUSION-FISSION (HYBRID) ENERGY SYSTEMS

R. C. Liikala
B. R. Leonard, Jr.
W. C. Wolkenhauer
D. T. Aase

January 1975

BATTELLE

INTRODUCTION AND BACKGROUND

Battelle-Northwest (BNW) has, over the past four years, been actively involved in the evaluation and development of a nuclear reactor concept which couples fusion and fission technologies. BNW's interest in this concept, generally referred to as fusion-fission (or hybrid) stemmed from the thought that; 1) it may be realizable before pure fusion (CTR)* power plants, because the fusion reaction and plant engineering requirements do not appear to be as stringent; and 2) a hybrid has potential to alleviate some of the constraints in the fission power economy, namely, fuel supply and waste disposal.(1) This paper outlines our perspective of the concept and summarizes technical highlights of our studies.

To provide a brief background, the fission and fusion processes are described along with how these processes combine in the hybrid concept. The fission process is depicted in Figure 1. A heavy element nucleus, such as ^{235}U, is bombarded by a neutron, causing the fission of the nucleus into fragments, yielding between two and three neutrons, and releasing about 200 MeV of energy. The fusion process is depicted in Figure 2. It involves light element nuclei, such as deuterium and tritium, which when confined as a plasma and heated, fuse together to form a helium nucleus, yielding a neutron, and releasing about 17 MeV of energy. Relatively speaking, the fission process can be described as energy rich and neutron poor whereas the fusion process is neutron rich and energy poor.

The hybrid reactor, as illustrated in Figure 3, combines fusion and fission. The concept is based upon interactions between the high energy fusion neutrons ($\sim$14 MeV) and heavy element nuclei placed in the blanket. The choice and arrangement of heavy element nuclei in the blanket depend upon the functional role intended for the hybrid.

The roles envisioned for the hybrid are three: 1) power production, 2) production of nuclear fuels, and 3) destruction of nuclear by-products.

*Hereafter we shall refer to fusion-fission systems as hybrids, or fusion-fission systems whereas we refer to fusion only (i.e., pure) systems as CTRs.

FIGURE 1

FISSION PROCESS

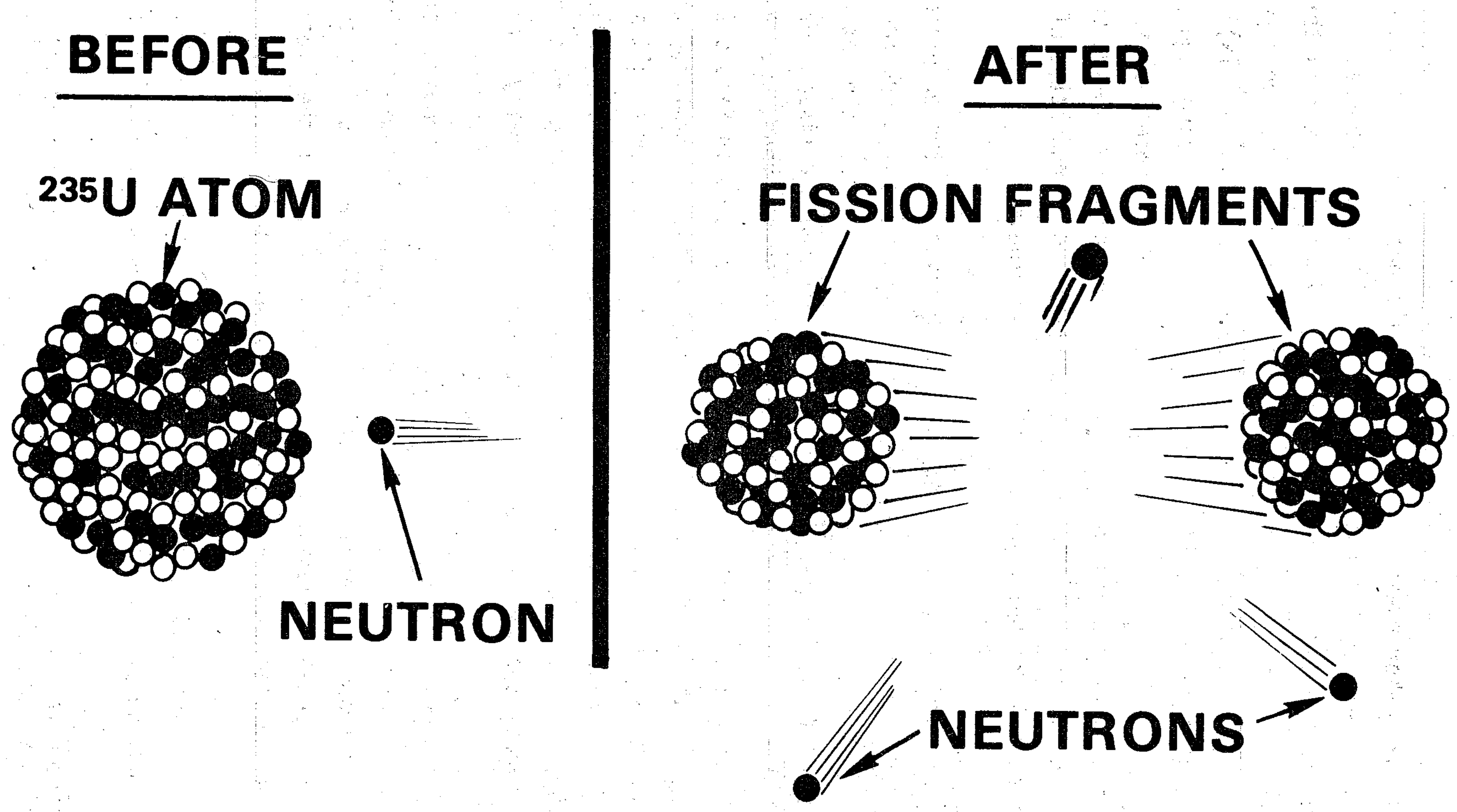


FIGURE 2

FUSION PROCESS

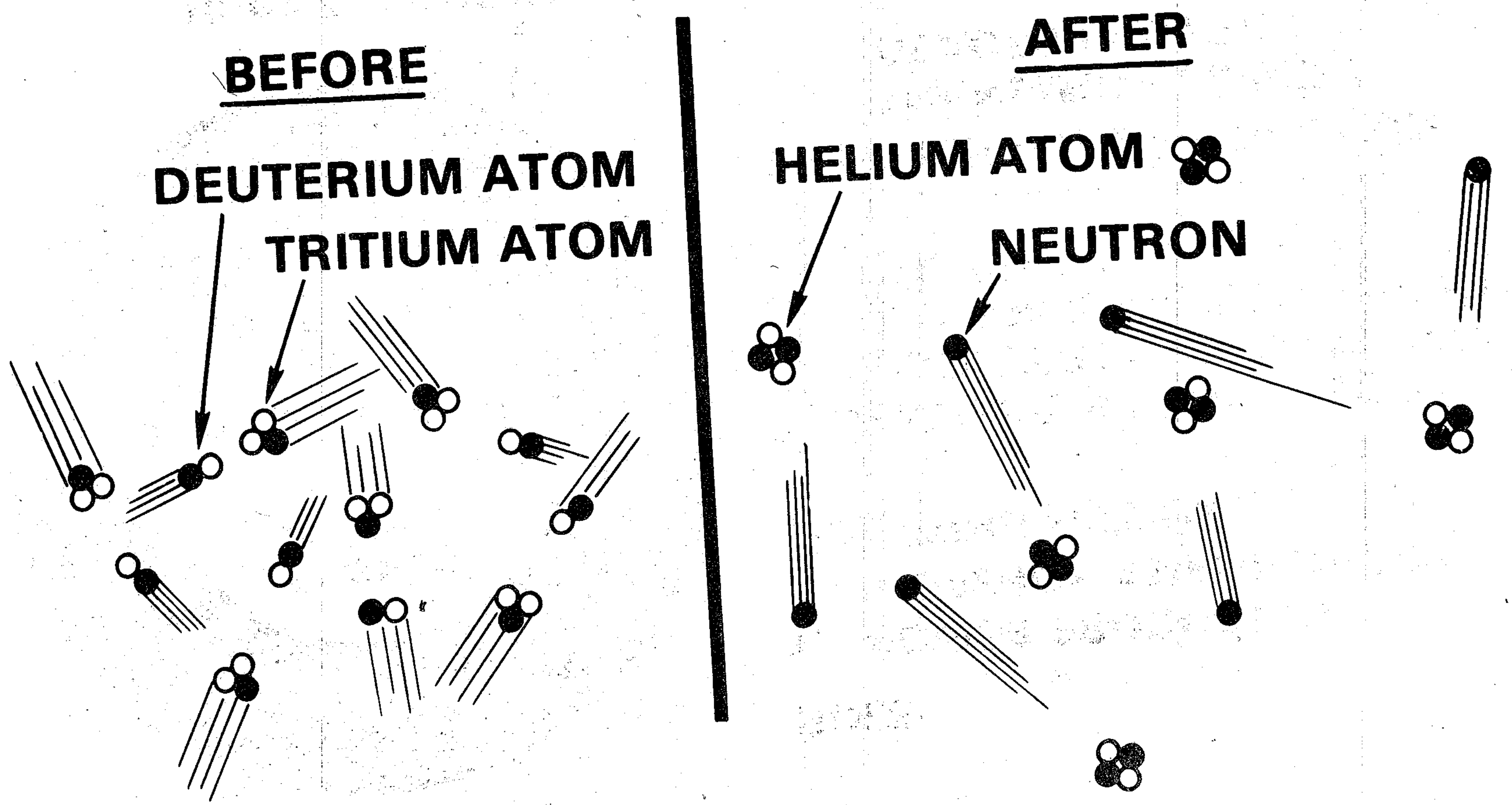
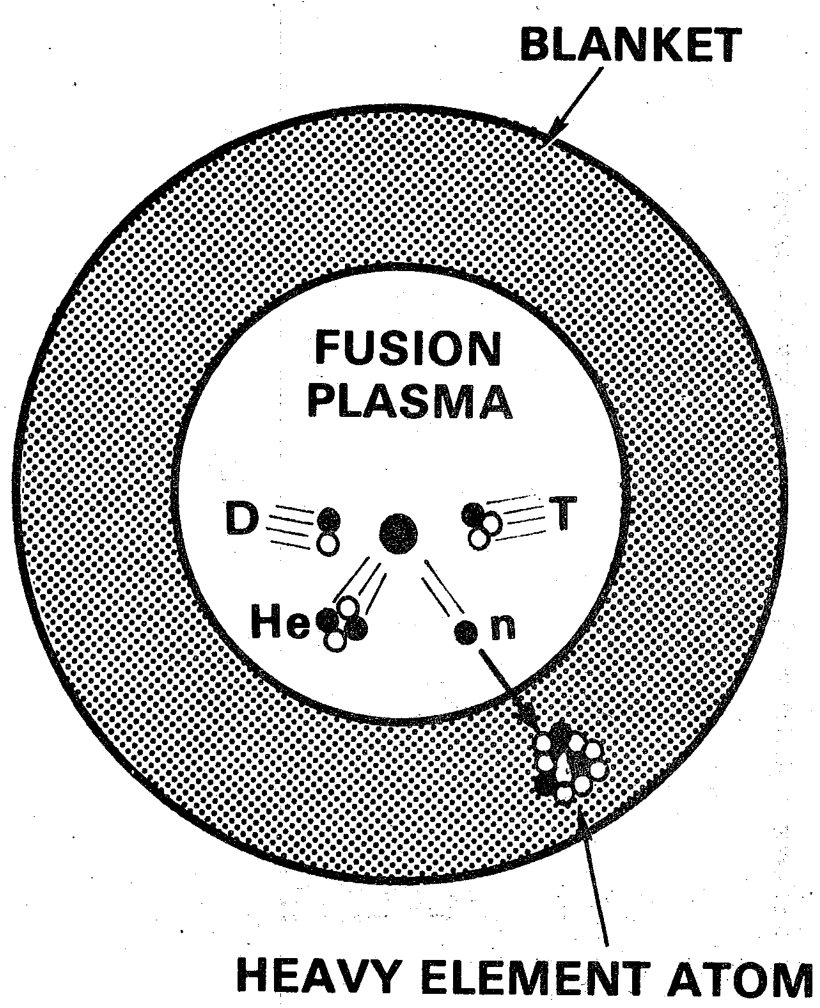


FIGURE 3

HYBRID



ROLE:

1. **POWER PRODUCTION**
   - ENERGY AMPLIFICATION THRU FISSION

2. **MATERIALS PRODUCTION**
   - CONVERT FERTILE MATERIAL TO FISSIONABLE/FUSIONABLE MATERIAL

3. **MATERIALS DESTRUCTION**
   - TRANSMUTE FISSION REACTOR WASTE

The first of these roles seeks to amplify the energy of fusion neutrons ($\sim$14 MeV) through absorption in fissionable nuclei (e.g., ^{235}U), causing fission thereby releasing $\sim$200 MeV of energy. The second role seeks to produce fuels for fission reactors and fusion reactors. This is accomplished by neutrons becoming captured in fertile nuclei (^{232}Th, ^{238}U, ^{6}Li) thereby producing fissionable material (^{233}U from ^{232}Th and ^{239}Pu from ^{238}U) and fusionable material (tritium from ^{6}Li). The third role seeks to eliminate or significantly reduce the radioactive waste created by fission reactors through transmutation. Transmutation is the changing of one nucleus into another nucleus. For the waste management concept, transmutation must result in a product nuclide having a lower toxicity and/or shorter half-life for radioactive decay than its predecessor. The process envisioned in the hybrid, is the transmutation of long-lived nuclei via neutron absorption to produce other nuclei which have shorter half-lives. Thus, the decay rate of certain toxic species in radioactive waste may be effectively accelerated via transmutation. Since all three roles involve use of heavy nuclei (i.e., actinides) in the blanket, a hybrid reactor may be characterized as "any fusion reactor with a blanket containing actinides" (i.e., fissile and/or fertile material).

The concept does not appear to be limited to a singular role. It could be combinations thereof such as producing power and useful nuclear material, or producing power and reducing by-product waste. Further study of the technical and economic feasibility of hybrid reactors designed to fulfill these roles is definitely needed.

TECHNICAL APPROACH

Some general ground rules were established, at the outset, in our study on hybrids. In order that the reader view the findings of our work in the proper context, these ground rules are stated here.

First, the studies are based on technology which has been developed, or which is expected to be developed in the near future. The rationale being, the hybrid should not require substantial R&D investments over and above the investments currently being made in the nations energy plan. Thus, it should be a step along the pathway of achieving pure fusion power, and realizing the benefits of the fission power economy. If substantial new R&D investments are required for the hybrid to become commercially viable, then the concept is less attractive.

Second, idealistic conditions are assumed in initial feasibility assessments. There are two reasons for this: 1) if the concept does not appear technically feasible under idealistic conditions, it will not be feasible under realistic conditions, and 2) the analysis is simplified, thereby requiring less investment of time and money in these studies. However, we certainly realize the need for caution that viable concepts are not initially ruled out on the basis of lack of knowledge.

Lastly, to negate the need for costly operational safety systems, we deliberately selected systems having the greatest potential for avoiding or mitigating the effects of reactor accidents. For example, a safety criterion in our analyses was that the blanket of a hybrid reactor must be subcritical at all times. Likewise, blanket technologies which intuitively seemed to have more potential to withstand loss-of-coolant without fuel melting were favored.

The technical approach described above has been and is being used in the studies conducted at BNW. We feel it is a prudent path in that it identifies the no-go points early in a concept development without investing substantial amounts of time and money.

SUMMARY AND PRINCIPAL FINDINGS

As mentioned in the Introduction section of this report, BNW has been engaged in the study of hybrids over the past several years. The results of our studies to date lead us to conclude that there is no a priori basis to eliminate the hybrid as fulfilling a useful function in meeting the nations energy needs.

We have conducted a variety of studies over the past few years and the principal results of these studies are summarized here. Studies of power producing hybrids, the use of fusion neutrons for transmutation of radioactive waste, and the evaluation of the most likely combinations of fusion and fission technologies are addressed. More details are given in the following section and we refer the reader to the bibliography for more complete discussions. It should be noted that these studies have been primarily technical based and that economic bases are not firmly established. Preliminary economic evaluations of hybrids are given in the paper by Deonigi in this document.

A. Power Producing Hybrids

The initial work on the evaluation of hybrids as power plants began over four years ago. In fiscal year 1974 (July 1973 to July 1974), BNW and Lawrence Livermore Laboratory (LLL) undertook a cooperative study of hybrids based upon a mirror fusion device. The direction taken and conclusions reached on each of these studies are briefly described here.

1. Early BNW Studies - The early efforts were aimed at defining the gross characteristics of a power producing hybrid.[2,3] We chose a graphite moderated helium cooled, uranium fueled blanket, since this type of system has some commonality to the High Temperature Gas Cooled Reactor (HTGR) thermal power systems currently being

operated and constructed. A fusion plasma patterned to the Tokamak machines was selected. The characteristics of such a hybrid power plant were then projected. The principal findings of this early study follow.

- Power multiplications between 30 and 50 over a non-fissile blanket appeared feasible.
- The tritium consumed in the plasma could be reconstituted in the blanket.
- A blanket subcritical from a self sustained fission reaction appeared achievable.

2. <u>BNW/LLL Cooperative Studies</u> - BNW has been involved in a cooperative study with LLL to determine the characteristics of a power producing fissile blanket matched to a mirror magnetic confinement device.(4,5,6,7,8) The basic design objectives were to:

   (1) produce electrical power,
   (2) produce as much tritium as consumed,
   (3) produce more fissionable material than consumed.

   LLL undertook study of a fast fission blanket and BNW undertook study of a thermal blanket. The results of the LLL studies are presented in the paper by Moir and Lee in this document and are summarized in Reference (7). The conceptual designs were basically built around the plasma characteristics as defined by the Livermore group. BNW designed a thermal fission lattice for the hybrid blanket, optimized for power production and fissile-fertile fuel utilization. The principal analytical results of the design analyses for this mirror confinement, thermal blanket system are as follows:

- An electrical power plant utilizing a sub-Lawson plasma to drive a subcritical thermal fission lattice is feasible.
- In this system, more fissionable and fusionable material is produced than is consumed.
- The fission blanket remains subcritical at all times for temperatures ranging from room temperature to those estimated for operating conditions.
- Fuel meltdown is not expected as a consequence of loss-of-coolant.

B. Transmutation

BNW has been studying alternatives for management of high-level radioactive waste.[9,10] Transmutation is one of these alternatives. Possible methods included use of accelerators, fission and thermonuclear explosives, fission reactors, and fusion reactors. One of the conclusions reached in this study was the neutrons produced in CTRs have significant potential for the transmutation of radioactive waste placed in blanket regions of CTRs.[10,11,12] The principal findings of the analytical studies on the use of neutrons produced in CTRs for transmutation are:

- Transmutation of actinides is theoretically feasible.
- Transmutation of selected fission products is theoretically feasible.

However, to accomplish this, the fission products and the actinides in spent fission reactor fuels probably have to be separated (chemically or by other means).

C. Evaluation of Possible Fusion-Fission Technology Combinations

The objective of this study, which is currently being conducted, is to evaluate the technical feasibility of various fusion-fission technology combinations in terms of a self sustained electrical power plant. The combinations will be ranked on the basis of engineering constraints to identify the most promising configurations as candidates for point design studies. Since this study is in progress, no conclusions have been reached as yet. A discussion of the approach being taken in the evaluation is given in Section IV.

Technical details upon which these statements are based are summarized in the following section. For additional information, we refer the reader to the references.

RECOMMENDATIONS

Based upon studies performed to date, the fusion-fission (hybrid) appears to have significant potential for fulfilling useful roles in this nations energy plan. Hybrid plants might be able to: a) produce electricity, b) enhance utilization of natural resources (i.e., breed power producing materials), c) alleviate constraints in management of nuclear by-products. Development of fusion-fission energy systems fits logically on the pathway to development of CTRs because:

- it provides useful information on CTR plasma characteristics,
- it allows early assessment of engineering and economic constraints,
- it provides an avenue for early involvement of private industry.

The plasma physics and materials damage requirements for a hybrid may be less stringent than those for a CTR. Information gained in designing and proof testing a hybrid would add to our understanding of CTR plasmas. Being able to construct and operate a hybrid device, sooner than a CTR, would aid in identifying major engineering constraints and evaluating the economic impacts of these in developing CTRs.

The goal of the CTR program is to develop commercially viable CTR power plants. To achieve this goal, the base technology of CTRs must be developed and this technology must be transferred to private industry so they become capable to design, construct, and operate power plants and related CTR facilities. Since CTRs and fission reactors are congruent being both nuclear based, we speculate that commercial firms currently in the fission reactor business will ultimately become involved in the fusion reactor business. The fusion-fission systems are not as foreign to these companies as CTRs and may prove to entice their venturing into CTR systems sooner than would be the case if hybrids were not in the plans for CTRs.

The results of our studies to date are encouraging. Therefore, we recommend further study of the hybrid concept to crystallize its role in our nations energy plan.

TECHNICAL RESULTS

The principal findings of studies conducted at BNW on hybrids has been summarized above. Additional technical detail is given here.

A. Power Producing Hybrids

The power producing hybrid work at BNW began over four years ago on a very low level funding basis. The early efforts were aimed at roughly scoping the characteristics of a power producing hybrid. These early efforts were focused on roughly defining blanket components and configurations which might achieve:

1) Neutron power multiplications in the order of 25 to 50,

2) Production/destruction ratios for tritium and fissionable material equal to unity or better,

and determining the impact on CTR plasma parameters and neutron wall loadings. In these studies we attempted to stay within the confines of existing or near extant fission reactor technology. Within this restraint, we chose a gas cooled (helium), solid moderator (graphite), uranium fueled fission reactor system as the (near) extant fission reactor technology most obviously suitable for CTR requirements. The criteria for suitability included; the inherent safety of various systems relative to reactivity insertion and loss-of-coolant accidents, and the apparent lack of impact of helium cooling on the CTR magnetic field. We then attempted to define the characteristics of such a system, implicitly tied to a TOKAMAK fusion device. The results of these studies, although most of the analysis was purely neutronic, indicated promise for further analyses of hybrids. Results of these studies and discussion thereof are given in a Nuclear Technology Review Article,[1] Proceedings of the University of Texas Conference,[2] and PNL Annual Reports.[3]

More recently BNW was involved in a cooperative study with LLL to determine the near term characteristics of a power producing fissile blanket matched to a mirror magnetic confinement device, shown pictorially in Figure 4. The conceptual design has been built around the plasma characteristics as defined by the Livermore group. These plasma conditions are expected to be attainable with reasonable extrapolations of present technology. The Yin-Yang confined plasma for this mirror hybrid was fed by four high-energy neutral-beam injectors. The characteristics of the magnetic mirror fusion segment of the hybrid are listed in Table 1.

Battelle-Northwest designed a hybrid lattice which was optimized for power production and fissile-fertile fuel utilization to fit in the design configuration. The blanket consisted of some 320 modules of the type shown in Figure 5. Each module consisted of a convertor region, a thermal fission lattice region, and inner and outer tritium breeder regions. The inner blanket region, called a convertor, consisted of depleted-uranium-dioxide, with the uranium being 0.3% ^{235}U atom percent. The thermal fission lattice region consists of slightly enriched UO_2 fuel (1.35 wt.% ^{235}U), graphite moderator, and helium coolant. The inner and outer breeder regions contained natural lithium.

At initial operating conditions the energy multiplication was calculated to be about 40 times 14.1 MeV. Using this value parametric studies of the system were carried out to define plant parameters. The overall system efficiency, N_S, as a function of the plasma energy multiplication, Q, for various assumed values of the efficiency of the neutral beam injectors (N_I) is shown on Figure 6. In Figure 7, the net electrical power and the overall system efficiency is shown as functions of $N_I Q$. The parameters selected for the magnetic mirror hybrid are shown in Table 2. We note that for the values chosen here, a breakeven power plant $P_{Net} = 0$, would require Q = 0.16.

FIGURE 4

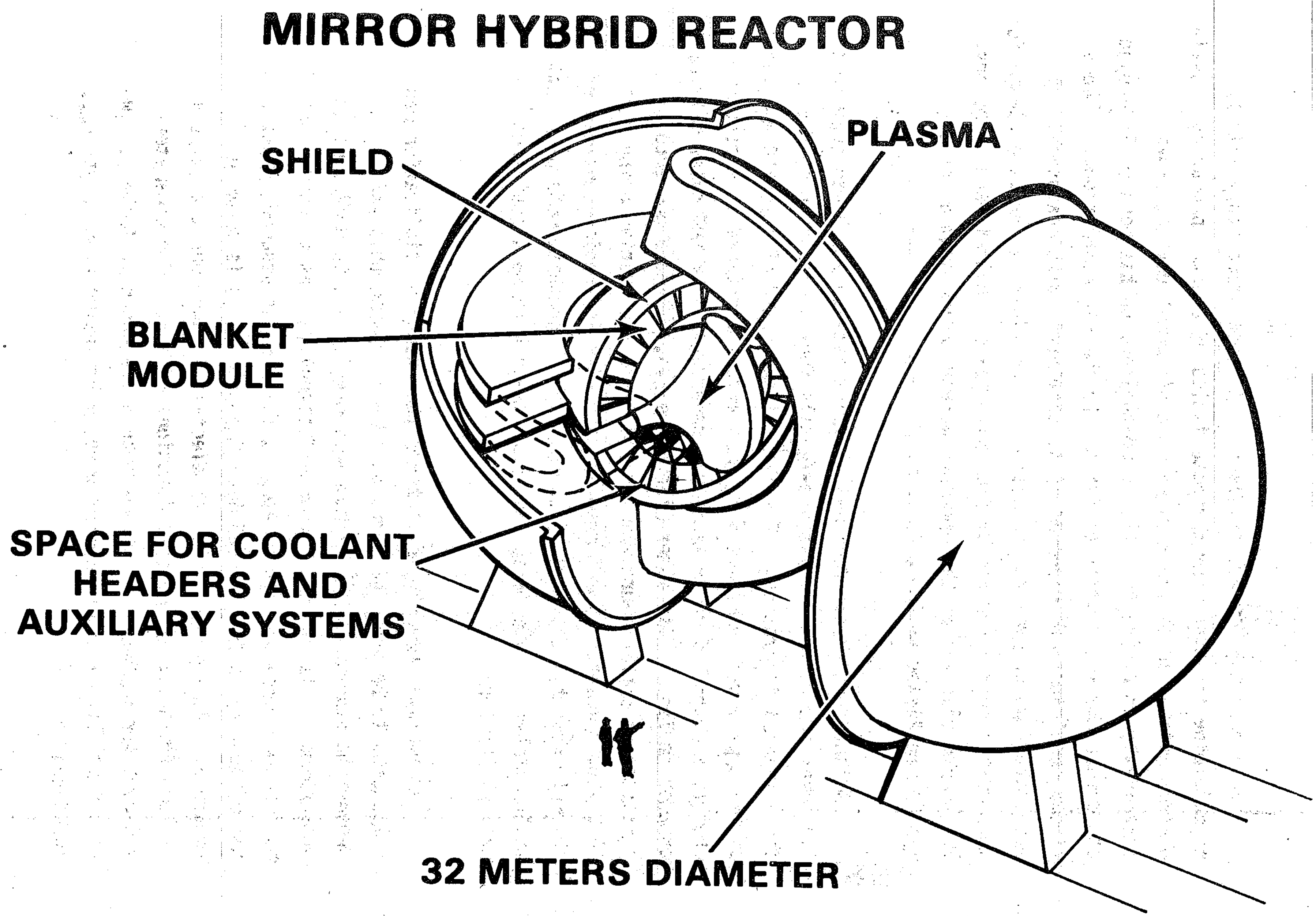

MIRROR HYBRID REACTOR
SHIELD
PLASMA
BLANKET MODULE
SPACE FOR COOLANT HEADERS AND AUXILIARY SYSTEMS
32 METERS DIAMETER

Table 1

MIRROR HYBRID PLASMA CHARACTERISTICS

PLASMA

Ellipsoid
3.5 m radius

NEUTRAL BEAM INJECTION

150 keV
500 Amperes
68 MW(e)

MAGNET

10 m radius
Central Field = 1.9 Tesla
Mirror Field = 7.1 Tesla

FIGURE 5

BLANKET MODULES

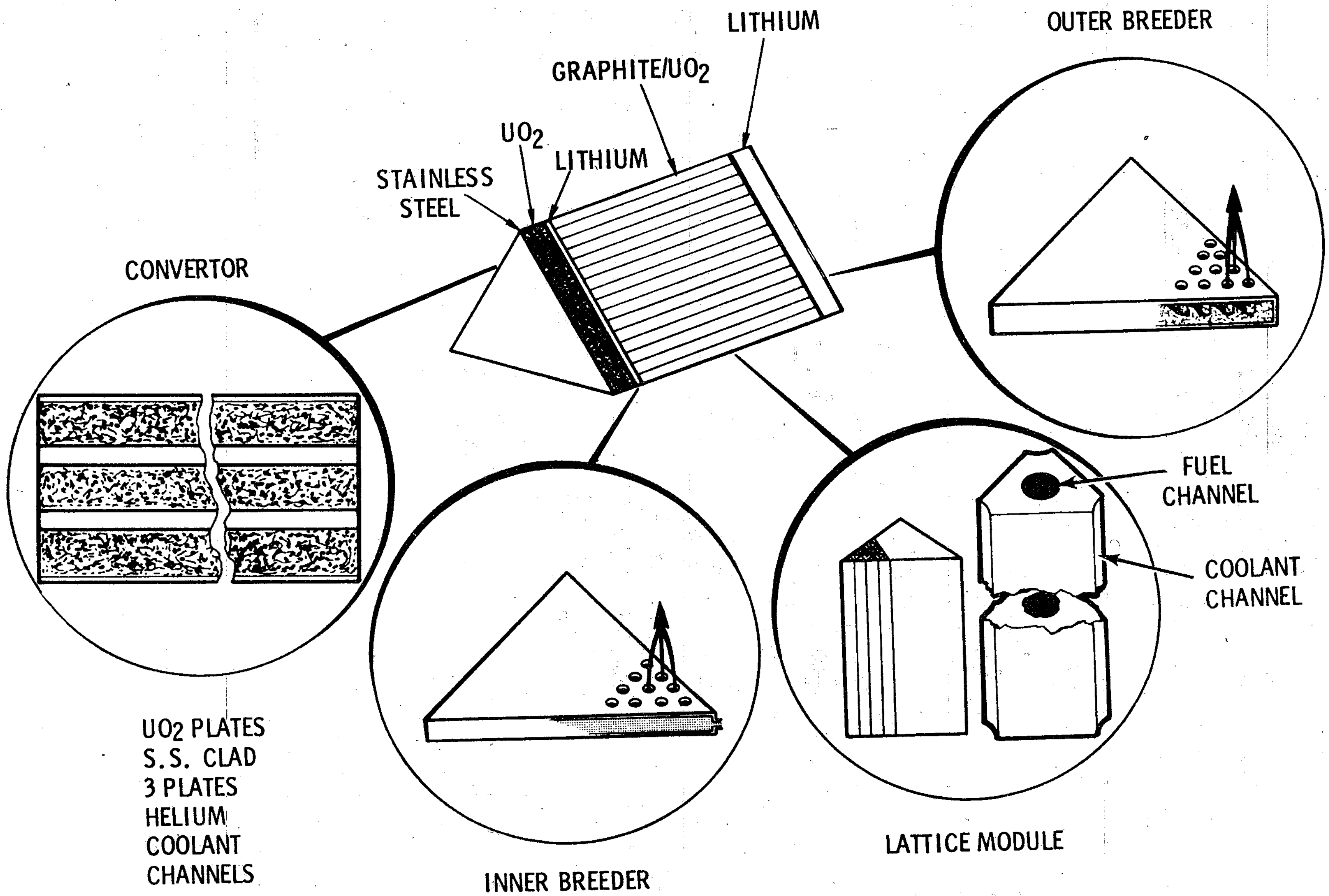


FIGURE 6

SYSTEM EFFICIENCY (N_S) AS A FUNCTION OF INJECTOR EFFICIENCY (N_I) AND Q

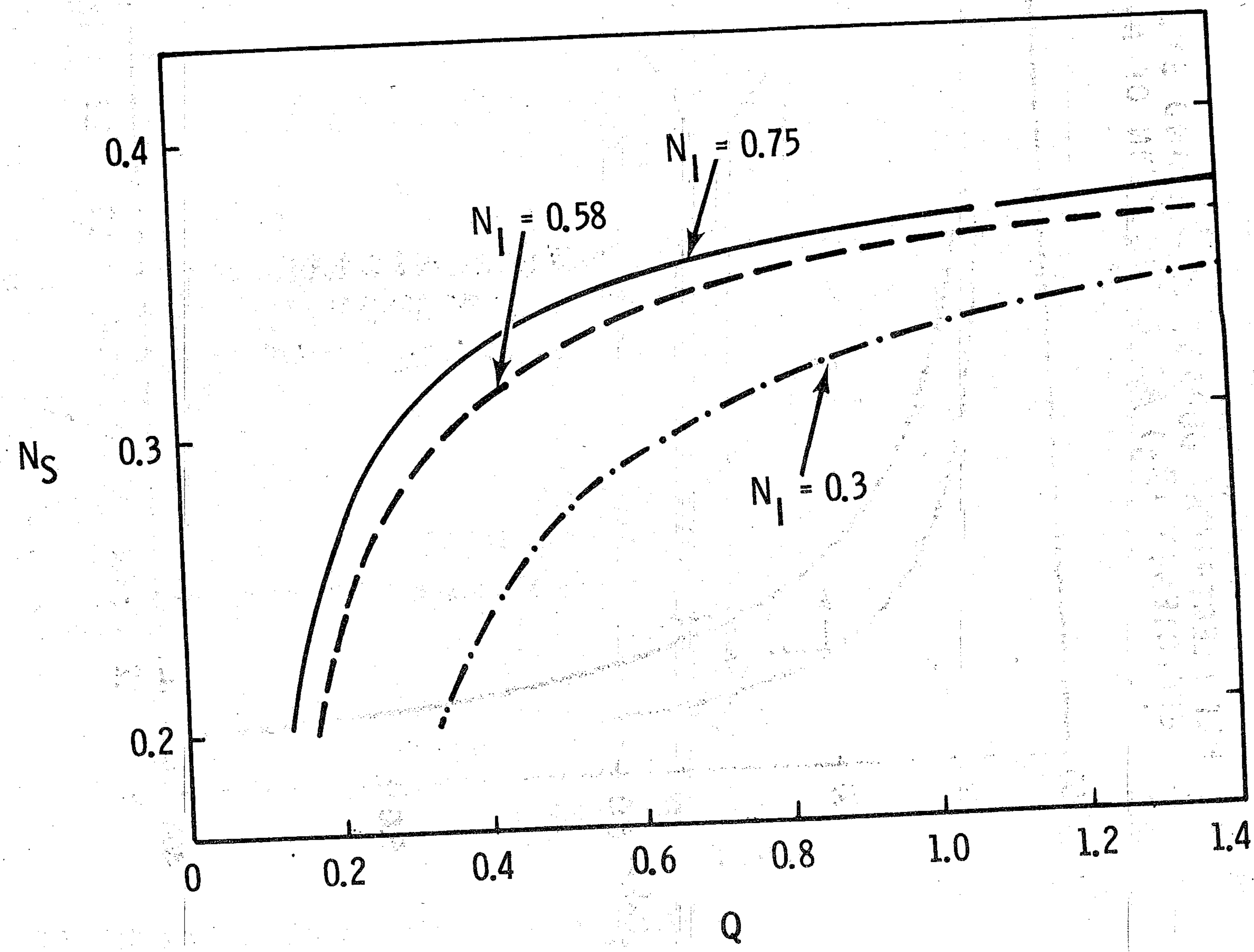


FIGURE 7

NET ELECTRICAL OUTPUT (P_{NET}) AND SYSTEM EFFICIENCY (N_S) AS A FUNCTION OF $N_I Q$

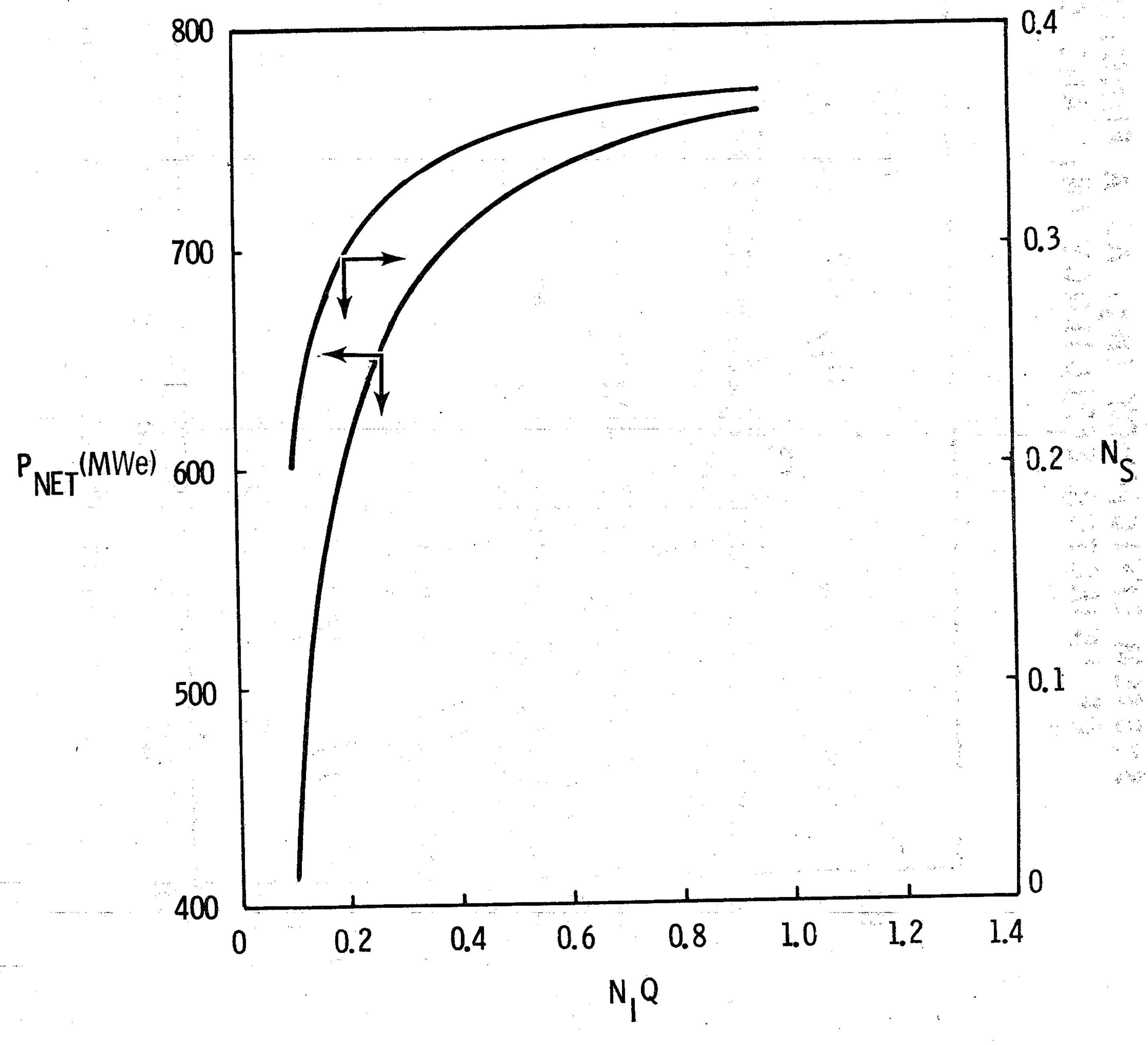


TABLE 2

HYBRID REACTOR DESIGN PARAMETERS - PNL/LLL STUDY

PARAMETER	VALUE
SYSTEM EFFICIENCY	32%
THERMAL TO ELECTRIC CONVERSION EFFICIENCY	39%
PLASMA ENERGY MULTIPLICATION	0.94
FISSILE BLANKET ENERGY MULTIPLICATION	40
INJECTOR EFFICIENCY	30%
NET ELECTRICAL OUTPUT	663 MWe
FISSILE BLANKET THERMAL POWER	2045 MWt
POWER INJECTED INTO PLASMA	68.3 MW
POWER CREATED IN PLASMA	64.2 MWt

Study of the steady state and dynamic thermal hydraulic behavior of the hybrid blanket was also made. In Figure 8 various blanket temperatures calculated assuming different values of average specific power generations are shown. The data on Figure 9 shows calculated temperature profiles for projected operating conditions. The calculated radioactive afterheat following shutdown is shown on Figure 10 for various assumed operating powers and times. Calculations of temperature following loss of gas coolant have also been made and the results are shown on Figure 11. As shown, a peak temperature of approximately 2010°C is predicted at about 55 hours after shutdown. The melting point of UO_2 is 2450°C. Thus there is a 400°C margin between predicted fuel temperatures following loss-of-coolant and the temperature at which the fuel melts. Thus, it tentatively appears that an emergency core cooling system might not be needed in this device.

Some of the more significant accomplishments of the study are outlined on Figure 12. On these bases there would appear to be no technical reasons for eliminating this mirror hybrid as a candidate for further development as a power plant.

The status of the LLL mirror hybrid design results was reported in two papers presented at the First Topical Conference on the Technology of Controlled Nuclear Fusion at San Diego in April[4,5] and in a paper presented at the 8th Symposium on Fusion Technology, Noordwijkerhout, Netherlands, in June.[6] A detailed technical report, BNWL-1835, covering the work has been issued.[13]

Since these initial results, neutronics calculations have been made to extend and clarify these results. Deficiencies were noted in ^{238}U ENDF/B descriptions of secondary neutron energies. These were modified and certain calculations were repeated to determine the impact of these cross section improvements on hybrid performance. We also increased the thickness of the inner lithium region to improve

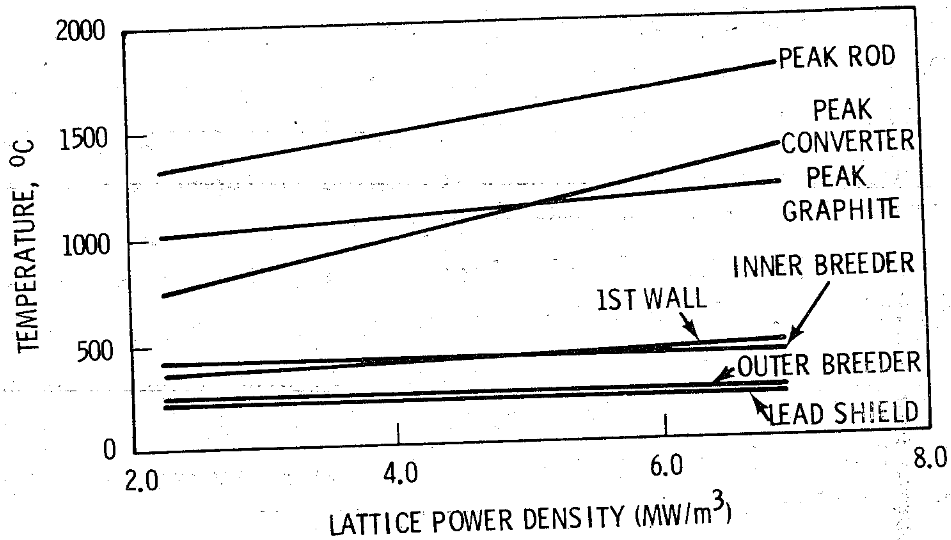


FIGURE 8 Effect of Power on Temperatures with Constant-Coolant Temperature Rise

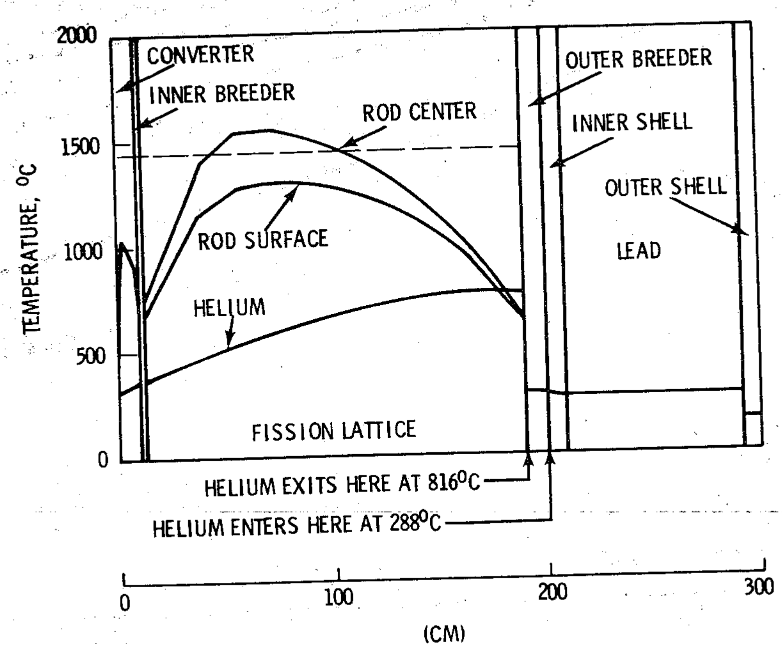


FIGURE 9 Steady State Temperature Distribution

FIGURE 10

RADIOACTIVE AFTERHEAT OF MIRROR HYBRID REACTOR

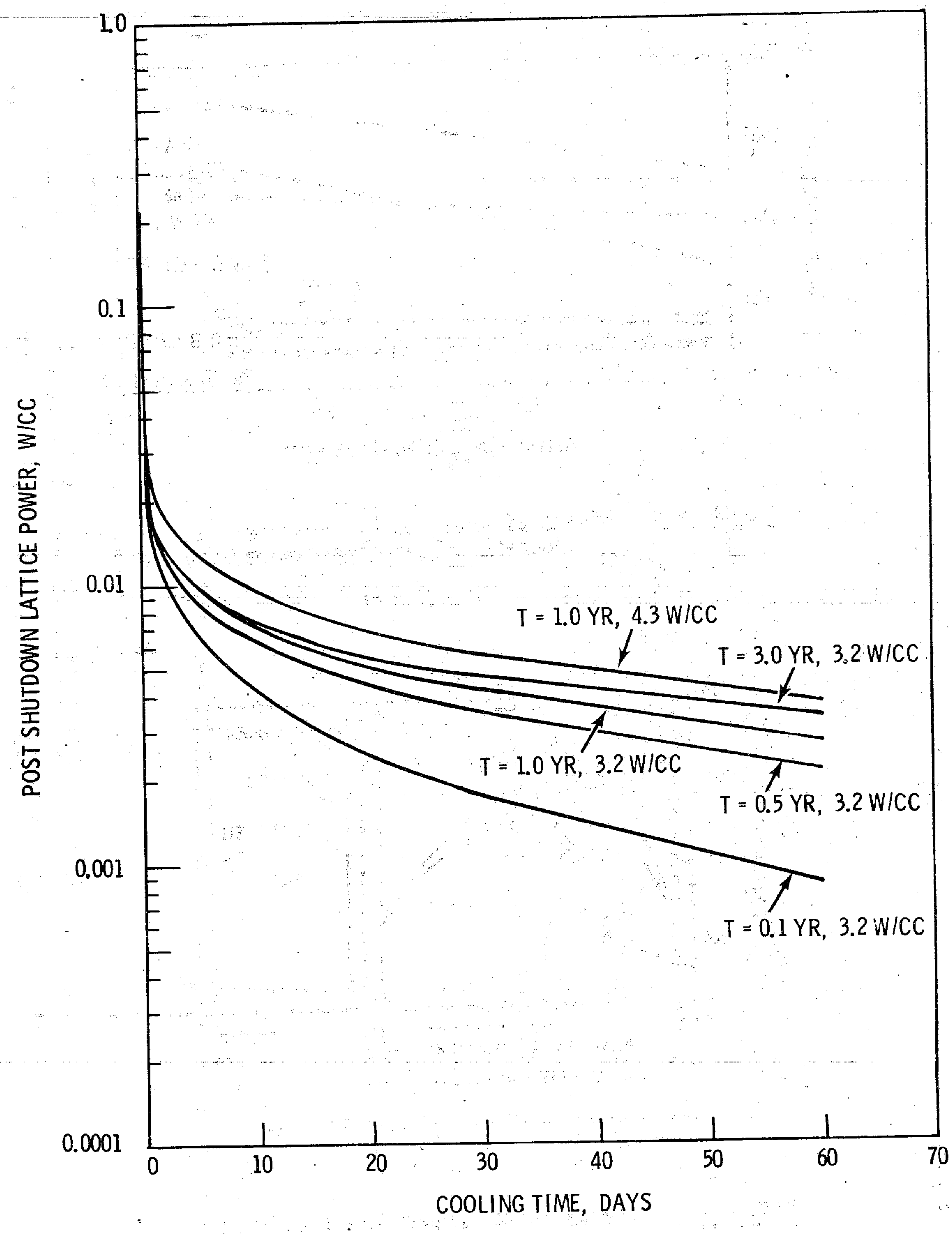


FIGURE 11

LOCA TRANSIENT TEMPERATURES

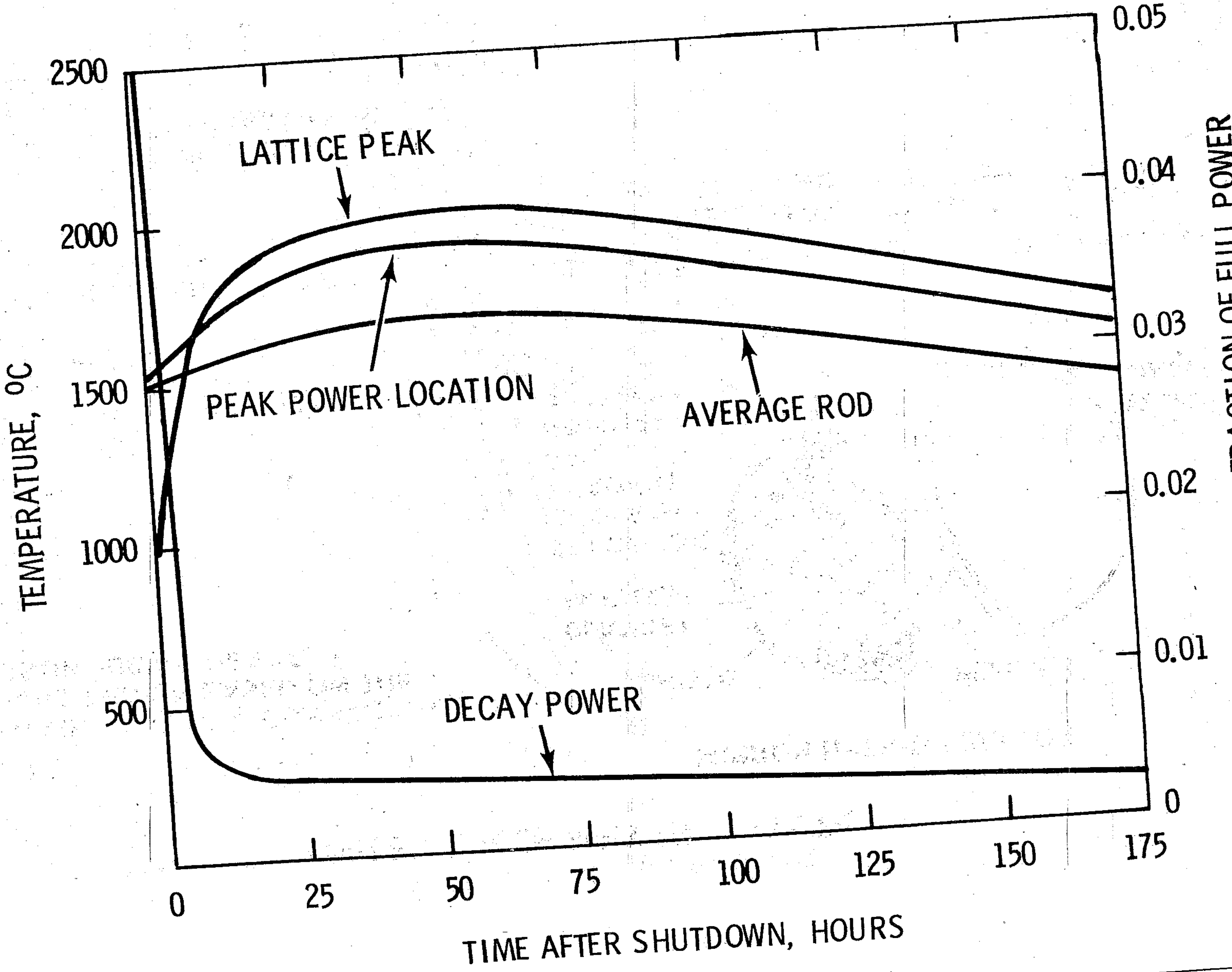
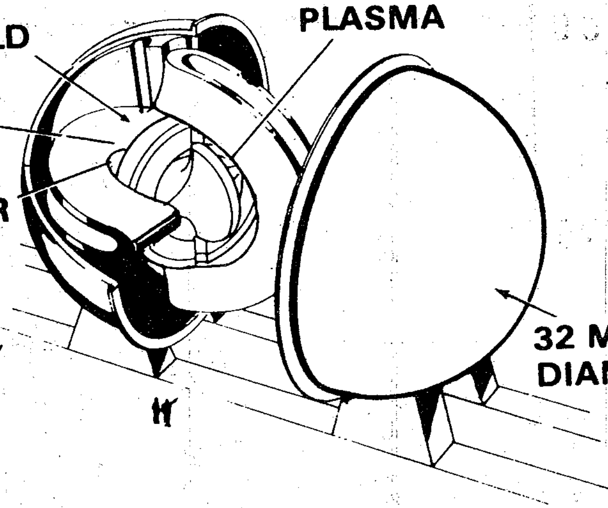


FIGURE 12

MILESTONES MET IN FY-1974

- DEVELOPMENT OF A CONSISTENT HYBRID DESIGN BASED ON THE MIRROR FUSION REACTOR

MIRROR HYBRID REACTOR



- FIRST ASSESSMENT OF HYBRID REACTOR SAFETY
  —FISSION LATTICE ALWAYS SUBCRITICAL
  —NO CORE MELTDOWN WITH LOSS OF COOLANT IN HIGH HEAT CAPACITY SYSTEMS

- ESTABLISHMENT OF A COOPERATIVE EFFORT IN HYBRID REACTOR DEVELOPMENT
  —LLL/PNL

tritium conversion. We calculated the neutron energy source distribution rather than assuming one. Finally, we performed burnup studies to deduce the change in neutronic parameters during operation.

The net effect of these factors on the projected hybrid was not substantial, and the general conclusion is that the previous work is valid. The results of these studies were presented at the Fifth IAEA Conference on Plasma Physics and Controlled Nuclear Fusion Research held in Tokyo on November 11-15, 1974.[8]

The calculated neutron energy source spectrum is worth mentioning further because of its possible significant in other CTR technology areas. In Figure 13 the spectrum calculated essentially exactly from a 100 keV neutral beam ion energy distribution is shown. The distribution shown is very broad, 1.7 MeV, and would be even broader for the 150 keV neutral beams now proposed.

The results of the burnup evaluation are given in Table 3 in terms of initial, final and cycle averaged parameters for a 200 day full power operating cycle. The present projections of tritium and fissile breeding are more favorable than originally predicted and show promise of good fissile-fertile utilization.

B. Transmutation

Studies over the past several years have been concerned with the possible use of CTRs to eliminate or significantly reduce the radioactive waste of the fission reactor economy.[11,12] These studies were integrated in an extensive review of all alternative methods for managing high-level radioactive waste.[9,10] The use of CTRs in this regard has been inspired by the large values of neutron flux which have been projected for CTR blanket systems. Although projected CTR neutron source strengths have been reduced by about an order of magnitude in the recent past due to materials damage considerations, the CTR is still a neutron rich device and some possibilities of useage for neutron-induced transmutation still appear to exist.

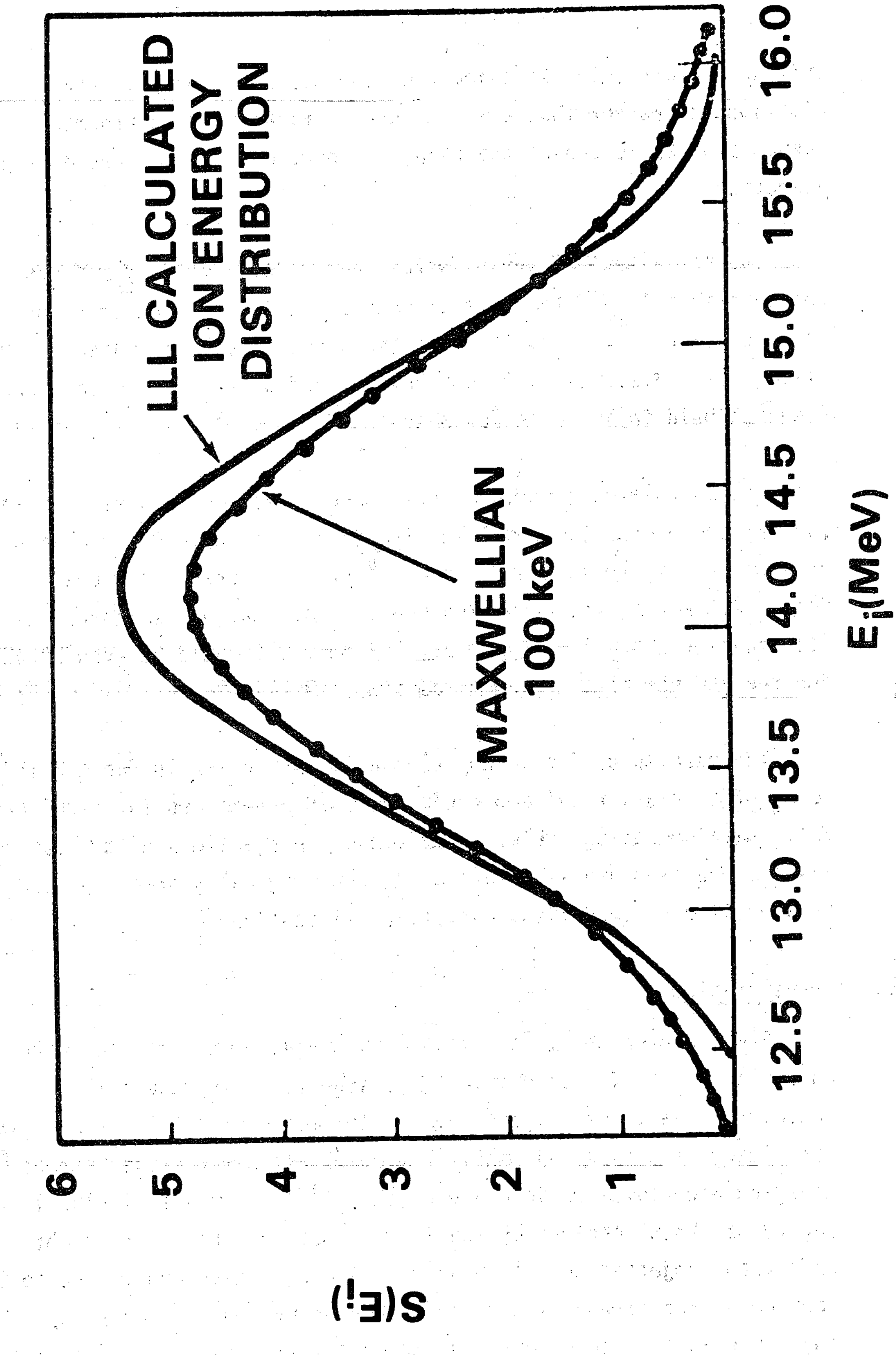

LLL CALCULATED
ION ENERGY
DISTRIBUTION
MAXWELLIAN
100 keV
$S(E_i)$
6
5
4
3
2
1
12.5
13.0
13.5
14.0
14.5
15.0
15.5
16.0
E_i(MeV)

FIGURE 13

TABLE 3

PERFORMANCE PARAMETERS OF HYBRID BLANKET WITH POWER OPERATION

PARAMETER	INITIAL	200 DAYS OPERATION	AVERAGE VALUE
FUEL EXPOSURE IN FISSION LATTICE	0	5000 MWd/MT	25 MWd/MT/day
^{235}U INVENTORY (ENRICHED LATTICE)	1000 kg	685 kg	
TOTAL PLUTONIUM INVENTORY	0	340kg	
BLANKET THERMAL ENERGY $\div$ 14.1 MeV	40	35	45
TRITIUM PRODUCED PER DT EVENT	1.065	0.88	1.07*

*Not corrected for source neutron losses through plasma and beam ports.

As mentioned above, BNW performed a comprehensive overview study of potential alternatives for long-term management of high-level radioactive waste(9,10) for the AEC's Division of Waste Management and Transportation. The logistics of nuclear fuel in the fission power economy is shown in Figure 14. High-level waste is defined in 10CFR50, Appendix F,(14) as the aqueous waste resulting from the first cycle solvent extraction step of reprocessing. It contains the fission products, actinides other than uranium and plutonium, and the unrecovered (i.e., losses of) U and Pu. The concepts reviewed are listed generically in Table 4. Within each there were numerous variations. For example, transmutation was considered on the bases of using accelerators, nuclear and thermonuclear explosives, fission reactors, and CTRs.

Partioning is a chemical separation of waste constituents into two fractions: one which contains the actinide elements and one which contains the fission products. As noted in Table 4, this is needed for the extraterrestial disposal and transmutation concepts.

The methodology employed in the study is outlined in Figure 15. For each variation of every concept, an assessment of technical feasibility was made. If the concept was proven infeasible, relative to predefined feasibility criteria, then it was rejected. If it passed the test, then evaluations were made of safety, environmental impact, research and development requirements (funds and time), the costs for building and operating the needed facilities, conflicts with existing national or international policy, and the attitude of the public and their perception of safety.

For transmutation, the criteria listed in Table 5 were used to determine technical feasibility. First of all, does the concept have a favorable energy balance? Secondly, is the energy consumed in the implementation of the concept sufficiently less than the electrical energy obtained creating the waste? Thirdly, is the predicted rate of elimination sufficient to keep up with the waste

FIGURE 14

NUCLEAR FUEL LOGISTICS

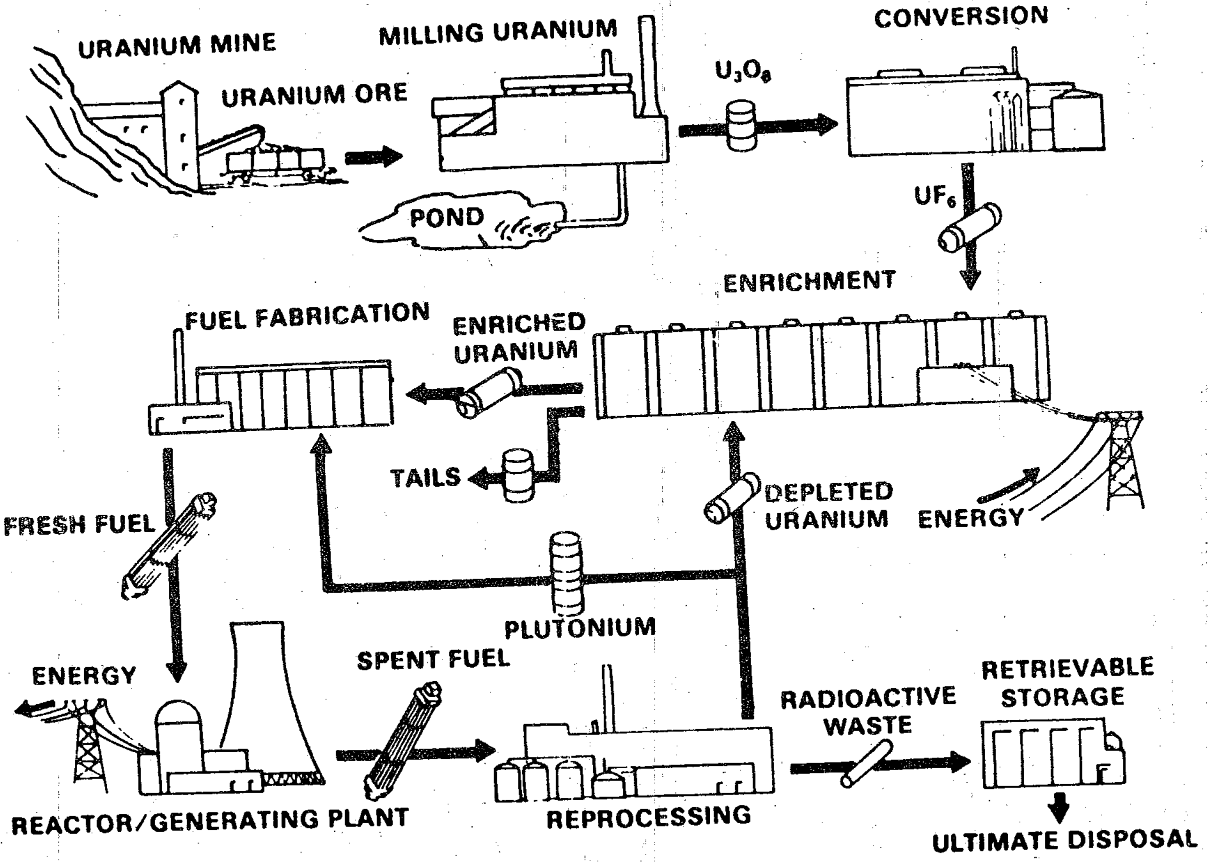


TABLE 4

HIGH LEVEL WASTE MANAGEMENT CONCEPTS

Concept	Variations	
Geologic Storage	6	
Seabed Storage	4	
Ice Sheet Storage	3	
Extraterrestrial Disposal	3	Partitioning
Transmutation	4	

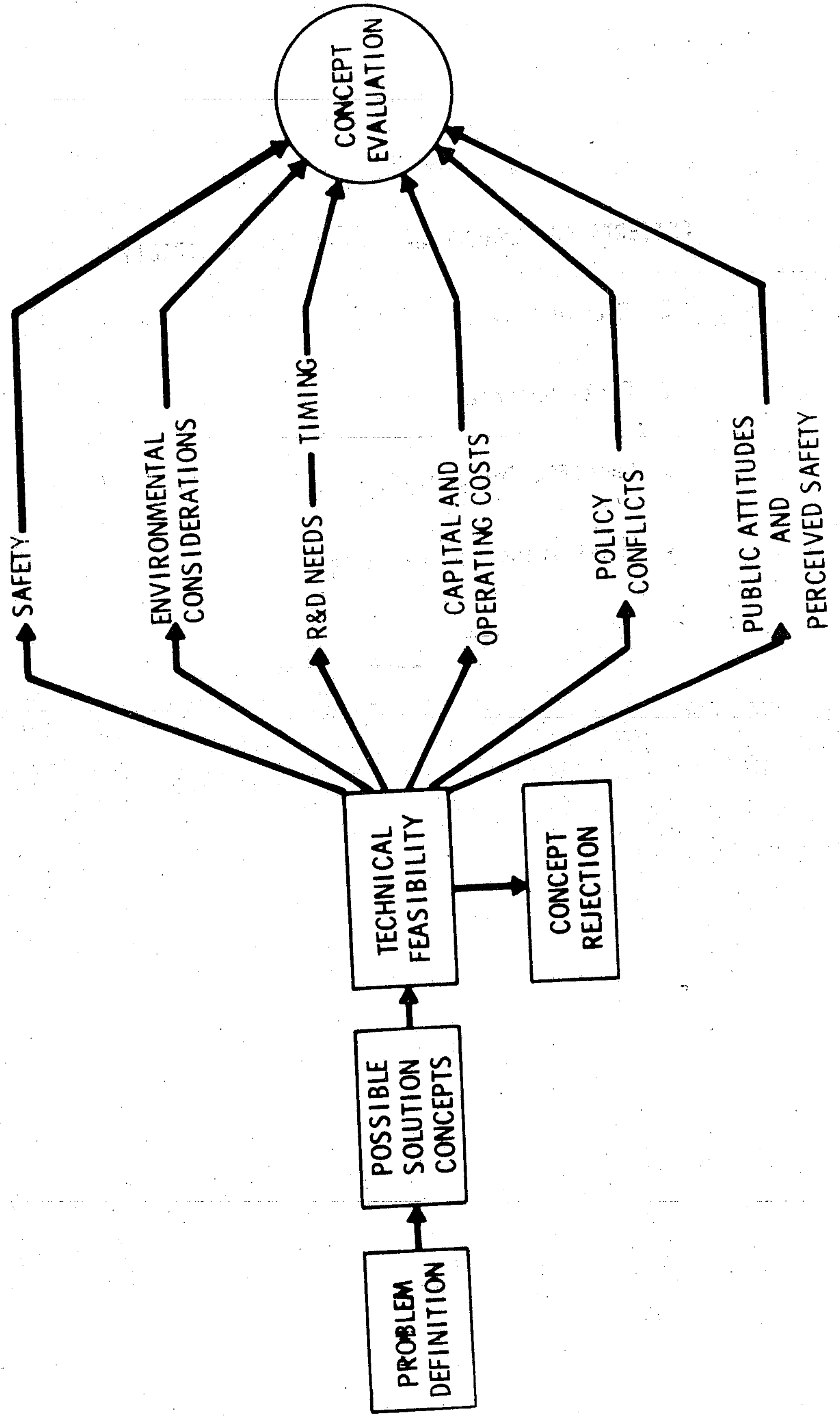

PROBLEM DEFINITION
POSSIBLE SOLUTION CONCEPTS
TECHNICAL FEASIBILITY
CONCEPT REJECTION
SAFETY
ENVIRONMENTAL CONSIDERATIONS
R&D NEEDS
TIMING
CAPITAL AND OPERATING COSTS
POLICY CONFLICTS
PUBLIC ATTITUDES AND PERCEIVED SAFETY
CONCEPT EVALUATION

FIGURE 15

TABLE 5

CRITERIA FOR EVALUATING TECHNICAL FEASIBILITY

- Energy Balance
- Waste Balance
- Specific Transmutation Rate
- Total Transmutation Rate

production rate? Finally, is more waste created by the process than eliminated?

As stated above, the transmutation concepts evaluated included accelerators, thermonuclear explosives, fission reactors and fusion reactors. The conclusions reached in the study are listed in Table 6. The results of the evaluation resulted in elimination of all accelerator schemes except possibly a spallation neutron source for transmutation of long-lived fission products. Likewise, use of neutrons from a thermonuclear explosion does not appear technically feasible. Fusion and fission reactors met the selection criteria for transmutation of actinides. Fusion reactors may also transmute selected fission products. Finally, partitioning will most likely be needed to separate actinides from fission products. Total elimination is not possible, hence some storage for residuals is required.

Results are presented in Table 7 which typify transmutation using CTRs. We note that for fission products, which represent potential short-term hazard, that relative to natural decay, gains in time between factors of 2 and 20 are achieved through transmutation. Substantial gains are achieved for long-lived isotopes such as ^{129}I and the actinides. The long-lived isotopes are the most difficult to manage, thus the continued interest in the use of neutrons to transmute these species to ones which are shorter-lived and/or less toxic.

C. Fusion-Fission Technology Combinations

The studies described above focused on a power producing hybrid system by combining either a Tokamak or Mirror fusion device with a urania-fueled, graphite-moderated, gas-cooled fission blanket to make a self contained electrical power plant. These are just two of the many possible combinations of fusion and fission systems.

The range of possibilities may be seen by first noting that fusion neutrons from plasmas can be generated using injected machines

TABLE 6

TRANSMUTATION STUDY CONCLUSIONS

- Transmutation of Actinides is Technically Feasible in Fission Reactors
- Transmutation of Actinides is Theoretically Feasible in Fusion Reactors
- Transmutation of Certain Fission Products is Theoretically Feasible in Fusion Reactors
- The Transmutation Concept
  - Requires Storage for Residuals
  - Requires Partitioning

TABLE 7

SUMMARY OF TRANSMUTATION IN FUSION REACTORS

	Gain in Time Relative to Natural Decay	
	Low Flux Levels	High Flux Levels
Fission Products		
- Short Term Hazard (e.g., ^{90}Sr)	∿1-4	∿3-25
- Long Term Hazard (e.g., ^{129}I)	$\sim 10^8$	$\sim 10^8$
Actinides		
- ^{241}Am	∿800	∿8000

or ignition machines either of which may operate on either D-D or D-T fuel cycles. To the first order, the blanket must contain an energy multiplier, hence the matrix of possible fuel cycles shown in Figure 16. Another way of looking at the hybrid (Figure 17) is the use of fusion neutrons to alleviate certain constraints in the fission power economy, such as waste transmutation or fissile fuel production. The Electric Power Research Institute (EPRI) is funding BNW and others to evaluate this type hybrid concept. Another way is to view the possible combinations of fusion and fission systems into a hybrid reactor as elements of a matrix having fusion technology as one dimension and fission technology as the other. This is the view being taken in our study for DCTR on evaluating possible fusion-fission technology combinations. Such a matrix is shown in Figure 18, with candidate fusion technologies listed across the top and developed fission technologies listed at the side.

The fission blanket may utilize either thermal neutron systems or fast neutron systems and there exist many combinations of neutron moderator and heat transfer systems for uranium, thorium, or plutonium fuel cycles. In order to reduce the number of possibilities to a manageable set for this study, the fusion and fission systems are considered to operate only on current, or currently planned fuel cycles. Thus, the fusion devices are considered to operate on a D-T cycle and the fission fuel cycles are restricted to current or near-term fuels and materials technology.

The approach being used to evaluate the possible combinations of technologies is illustrated in Figure 19. Current descriptions of CTR and fission reactor systems are being used to define the technology of each. The range of fission technology considered is restricted to

FIGURE 16

POSSIBLE HYBRID FUEL CYCLES

NEUTRON SUPPLY

ENERGY MULTIPLIER	D-T PLASMA		D-D PLASMA	
	INJECTED MACHINES	IGNITION MACHINES	INJECTED MACHINES	IGNITION MACHINES
U	√ PNL	√ PNL/LLL		
Th		√ MIT/U OF TEXAS & CANADIANS		
Pu		√ LLL		
BE		√ LASL/BNL		
Li				

√Some studies have been made in these areas.

FIGURE 17

POSSIBLE PURPOSES OF HYBRID

NEUTRON SUPPLY

NEUTRON UTILIZATION		D-T PLASMA		D-D PLASMA	
		INJECTED MACHINES	IGNITION MACHINES	INJECTED MACHINES	IGNITION MACHINES
REDUCE WASTE	ACTINIDES		√		√
	FISSION PRODUCTS		√		√
PRODUCE FUEL	^{233}U		√		
	^{239}Pu		√		
	T		√		

√Some studies have been made in these areas.

FIGURE 18

EVALUATION OF POSSIBLE HYBRID REACTOR COMBINATIONS

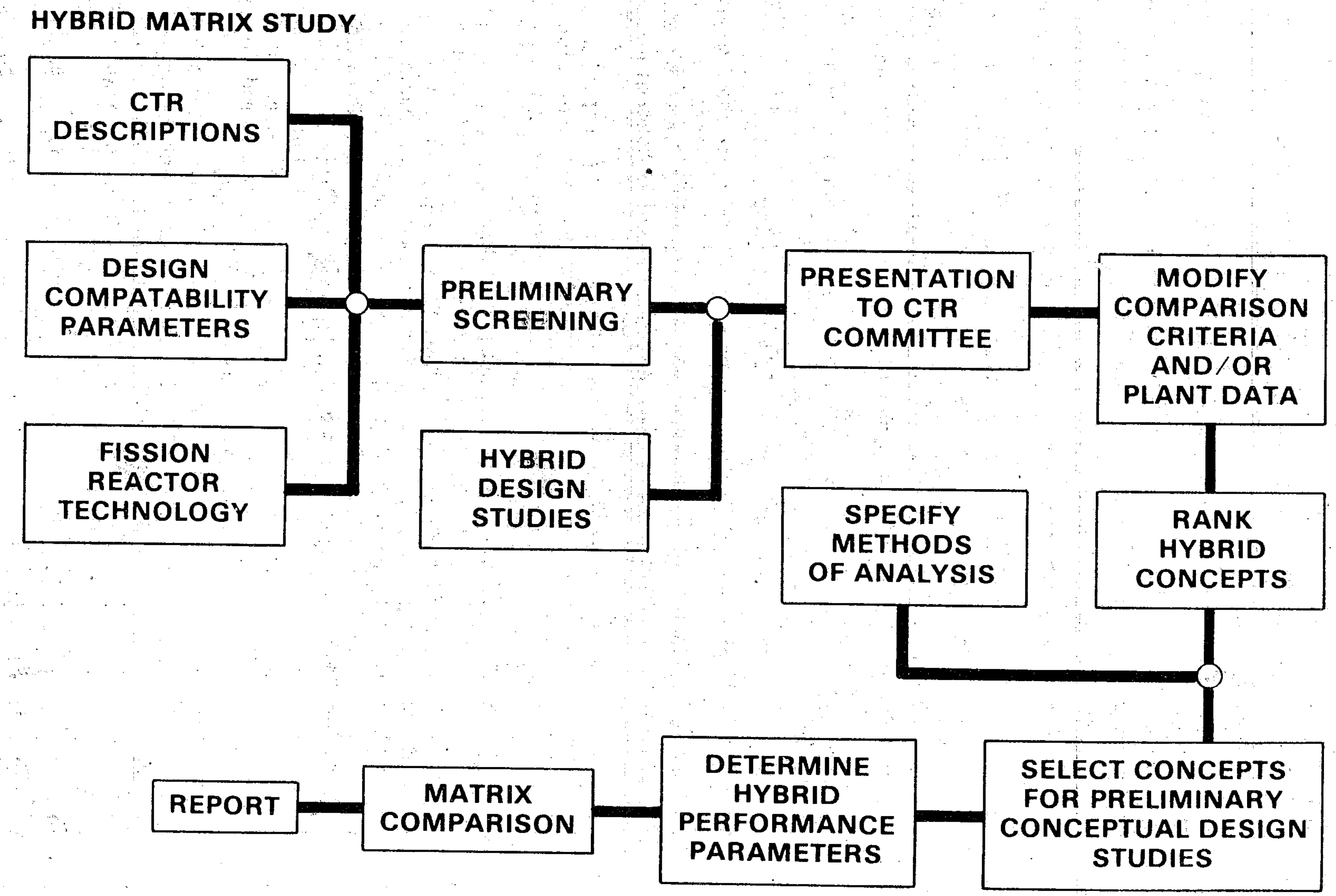
FUSION TECHNOLOGY

FISSION TECHNOLOGY

	TWO COMPONENT TORUS	TOKAMAK	MIRROR	THETA PINCH	LASER
LIGHT WATER MODERATED AND COOLED SYSTEMS					
HEAVY WATER MODERATED AND COOLED SYSTEMS					
GRAPHITE MODERATED GAS COOLED SYSTEMS		✓		✓	
LIQUID METAL COOLED FAST BREEDER SYSTEMS		✓	✓	✓	✓
GAS COOLED FAST BREEDER SYSTEMS					
MOLTEN SALT SYSTEMS			✓		

✓ Some studies have been made in these areas.

FIGURE 19


HYBRID MATRIX STUDY
CTR DESCRIPTIONS
DESIGN COMPATABILITY PARAMETERS
FISSION REACTOR TECHNOLOGY
PRELIMINARY SCREENING
HYBRID DESIGN STUDIES
PRESENTATION TO CTR COMMITTEE
MODIFY COMPARISON CRITERIA AND/OR PLANT DATA
SPECIFY METHODS OF ANALYSIS
RANK HYBRID CONCEPTS
SELECT CONCEPTS FOR PRELIMINARY CONCEPTUAL DESIGN STUDIES
DETERMINE HYBRID PERFORMANCE PARAMETERS
MATRIX COMPARISON
REPORT

that of current manufacturing capability or planned for near term manufacture by industry. This restrictive view is taken to eliminate the need for substantial additional investment in fission technology for hybrid development. Design compatibility parameters for both the plasma and the fission blanket are defined to aid the development of criteria for preliminary screening of the possible combinations. The design compatibility parameters developed to date are listed in Tables 8 and 9 for the plasma and for the fission blanket, respectively. The design compatibility parameters along with other criteria such as illustrated in Table 10, form the bases for making a preliminary screening of the possible combinations. As listed in Table 10, the potential functional capability (i.e., role) can be a criterion for screening. However, for the purposes of this study, the role is limited to an electrical power plant producing more fissionable and fusionable material than it consumes in a given fuel cycle. This view is taken primarily because the design studies performed to date are mostly based upon this premise and it includes elements of fuel production and actinide burning.

As shown, in Figure 19, these data sources are used in making a preliminary ranking of concepts for presentation to a peer CTR committee. When a final ranking is established after peer review and comment, analytical study of key performance parameters for the most likely concepts will be made to firm up the technical basis for the ranking of the systems.

TABLE 8

DESIGN COMPATIBILITY PARAMETERS: PLASMA

- Type of Neutrons and Distributions
- Blanket Volume Characterization
  - Magnet Configuration
  - Structural Requirements
    - Plasma Injection
    - Divertor Design
    - Vacuum System
- Cycle Time and Duration
- Current Performance Range Considering Lawson Criteria, Wall Loadings, etc.
- Subsystem Costs and Sensitivity to Operating Parameters

TABLE 9

DESIGN COMPATIBILITY PARAMETERS: FISSION BLANKET

- Coolant Characteristics
  - Temperatures
    - Pressures
    - Flowrates

- Magnetic Field Characteristics
  - Piping Requirements

- Neutronic Lattice Characteristics
  - Reactivity
  - Heat Generation

- Power Density - Average and Technical Specification Values

- Range of Flexibility in Fission Lattice Characteristics

- Current Subsystems Costs

TABLE 10

PRELIMINARY SCREENING CRITERIA

- Design Compatibility
  - Neutronic Compatibility
  - Structural Compatibility
  - Coolant Compatibility
  - Operational Compatibility
- Range of Flexibility in Plasma and Fission Lattice Technologies
- Potential Functional Capability (i.e., The Role)
- Potential Hybrid Cost Reductions

REFERENCES

1. B. R. Leonard, Jr., "A Review of Fusion-Fission Hybrid Concepts," Nucl. Tech., 20, 161 (1973).

2. B. R. Leonard, Jr., and W. C. Wolkenhauer, Fusion-Fission Hybrids: "A Subcritical Thermal Fission Lattice for a D-T Fusion Reactor," Technology of Controlled Thermonuclear Fusion Experiments and the Engineering Aspects of Fusion Reactors, CONF-721111, pp. 918-930, April 1974, Proceedings of a symposium held at Austin, Texas, November 20-22, 1972.

3. Pacific Northwest Laboratory Annual Reports on Controlled Thermo-nuclear Reactor Technology, 1971 Annual Report - BNWL-1604, 1972 Annual Report - BNWL-1685, 1973 Annual Report - BNWL-1823.

4. W. C. Wolkenhauer - PNL, B. R. Leonard, Jr. - PNL, A. M. Sutey - PNL, R. W. Moir - LLL, "Conceptual Design of a Fusion-Fission Hybrid Reactor Based on a Mirror Fusion Reactor with a Subcritical Gas Cooled Fission Blanket," BNWL-SA-4865, First Topical Meeting on the Technology of Controlled Nuclear Fusion, Vol. 1, pp. 238-255, CONF-740402-pl, April 16-18, 1974.

5. W. C. Wolkenhauer, PNL, C. W. Stewart, PNL, R. W. Werner, LLL, J. D. Lee, LLL, "Some Safety Considerations of Hybrid Reactors in Comparison with Fission and Fusion Reactors," BNWL-SA-4988, First Topical Meeting on the Technology of Controlled Nuclear Fusion, Vol. 1, pp. 588-598, CONF-74042-pl, April 16-18, 1974.

6. W. C. Wolkenhauer, B. R. Leonard, Jr., U. P. Jenquin, "Advances in the Development of the Mirror Hybrid (Fusion-Fission) Reactor," BNWL-SA-4961, 8th Symposium on Fusion Technology, June 17-21, 1974, Noordwigkerhaut, the Netherlands.

7. R. W. Werner, J. D. Lee, R. W. Moir, G. A. Carlson, LLL, W. C. Wolkenhauer, B. R. Leonard, Jr., PNL, "The Interesting Pos-sibilities of Fusion-Fission," IAEA Fifth Conference on Plasma Physics and Controlled Nuclear Fusion Research, IAEA-CN-33/G6-1, November 11-15, 1974, Tokyo, Japan.

8. B. R. Leonard, Jr., W. C. Wolkenhauer, "Progress Toward the Development of a Mirror Hybrid (Fusion-Fission) Reactor," BNWL-SA-4932, IAEA Fifth Conference on Plasma Physics and Controlled Nuclear Fusion Research, IAEA-CN-33/G6-2, November 11-15, 1974, Tokyo, Japan.

9. WASH-1297, "High Level Radioactive Waste Management Alternatives," May 1974.

10. A. M. Platt and K. J. Schneider, Editors, BNWL-1900, "High-Level Radioactive Waste Management Alternatives," Vol. I, II, III, and IV, May 1974.

11. W. C. Wolkenhauer, B. R. Leonard, Jr., and B. F. Gore, BNWL-1772, "Transmutation of High-Level Radioactive Waste with a Controlled Thermonuclear Reactor," September 1973.

12. B. F. Gore and B. R. Leonard, Jr., "Transmutations of Massive Loadings of Cesium-137 in the Blanket of a Controlled Thermonuclear Reactor," Nucl. Sci. and Eng., 53, (319) 1974.

13. W. C. Wolkenhauer, et. al., "Status Report: Mirror Hybrid Reactor Studies," BNWL-1835, November 1974.

14. Code of Federal Regulations, Title 10, Part 50, Appendix F.

QUESTIONS ABOUT FOURTH PRESENTATION

Baker: You stated that one of the reasons or advantages of the hybrid system is to provide information on plasma characteristics. What does that mean?

Liikala: Well, with regards to achieving Lawson conditions, I think in the event you achieve near Lawson conditions that, to me, is along the pathway.

Baker: You may relax conditions, but what does it mean that it's going to provide us sufficient information on plasma characteristics.

Wolkenhauer: On that particular slide, we were talking about those which are critical path items that have to be achieved on the pathway of getting to a hybrid.

Baker: I take issue with that, perhaps, and also with the comment that was made earlier, that the hybrid from the plasma point of view is something that you would do along the road of pure fusion. That may be, but it may not be. It could be a burden.

Wolkenhauer: The point is simply to have a hybrid you need a decent plasma. You don't need a Lawson plasma but, it turns out, it has to be reasonably close. That's the point.

Liikala: I think the point is that we wouldn't feel comfortable going ahead and doing the design, and considering moving on to a hybrid pilot plant with-out first having experimental verification of the plasma characteristics.

Coffman: We have progressed this long, and no one has really addressed the question of having a blanket that was critical. If you are going to take a fusion reactor and put a fission blanket on it, and run it subcritically, then the next logical step to get your energy mulitpli-cation is just run it as a critical blanket and put control rods in it.

Liikala: Well in all our studies, being subcritical basically was a criterion (i.e., that we would indeed be subcritical at all points in time). We have never addressed whether there are advantages to be critical.

Coffman: I guess, my question is why have we not?

Liikala: Again, simply from a safety point of view of the system.

Wolkenhauer: Your energy multiplication is infinity. That's a critical system.

Coffman: Isn't that the design constraint that you're bumping on. It's to get it more than say 20? And if 100 is good, why not run it at 200?

Wolkenhauer: It turns out, and this is a point that has been touched on but only indirectly, that there is a whole spectrum to possible multiplications that you can design a hybrid to, anywhere from essentially zero which was Larry Lidsky's case on up to critical. The selected multiplication is based on tradeoffs in various things, like economics and safety and so forth. One of the things that may be apparent from the studies done to date is that there is an economic advantage to running fairly low multiplication systems, a' la the kinds of things J. D. Lee does. There is an apparent safety advantage to running high multiplication systems like the kind of things we have done. This is simply the realm in which future studies are going to have to be done to nail down what exactly would be done in practice.

Miley: I noticed that in your future plans you have listed DD systems. I was interested in this since as Larry Lidsky pointed out earlier, 14.1 MeV neutrons are best for breeding. Still, I noticed in your earlier studies that 2.5 MeV DD neutrons are quite attractive for fission product burning, provided that we forget about possible plasma problems. Could you comment on this? Why are you looking at DD systems?

Liikala: Our sponsor asked us to. He basically wants to consider all possibilities; we don't want to be constrained with just DT. Let me give you my perspective on what the Electric Power Research Institute is attempting to do. They see the waste management problem with regards to fission reactors as one of the problems utilities face. In the event a technology that is consistent with CTR development can be employed in helping to reduce that problem let's go ahead and evaluate what precisely we can do and develop the idea to the stage that if it's ever needed, we can implement it. If it's needed, then we will have the technical and economic basis established. So that's point number one. Moreover, since we are considering CTR systems, let's not be limited by just DT.

Miley: Could you comment on your previous studies of DD neutrons versus DT?

Liikala: I'll let Bill Wolkenhauer.

Wolkenhauer: That data can be kind of misleading simply because the results were couched in terms of megawatts per square meter on the first wall. Because there is less energy per neutron for the DD reaction, you're getting more neutrons through the wall at the same megawatts per square meter, and so you are getting a faster transmutation rate. Now, if you couch the same data in terms of transmutations per initial neutron, which is another way of looking at the information, it is quite likely that it would turn around, and the DT neutron would be more interesting. Also, there is another point that wasn't factored into that study which since has become very interesting. This is the fact that injected reactors do not have a monoenergetic 14 MeV source; they have a spectrum. You're getting a significant number of neutrons above 14 MeV, and those particular neutrons are very valuable. It simply opens up many more reactions that may be used.

Miley: I guess that to a large extent, it depends on whether or not the wall loading represents a key design limit.

Wolkenhauer: Yes, it's simply the way the data was presented. It can be misleading.

Liilkala: Shown on this slide is what Bill was referring to. Basically, the neutron eenergy distribution is a function of the ion distribution where this shows a Maxwellian distribution and that shows the ion distribution that Livermore calculated.

Dudziak: Regardless of the source you used, do you evaluate the relative safety of the actinides that, for example, tend to transmute for a certain number of years in a reactor versus the number 800 times that out of the reactor? Say, 800 times that many years for whatever other alternatives of disposal. It seems to me you must introduce some qualifying factor for the relative hazard of having the actinides in a reactor subject to release in the event of an accident versus being buried, for example, in a stable geological formation.

Liikala: No, it hasn't been, but that is what needs to be done. That was the reason for putting up that fuel cycle chart. When we're talking about transmutation, we have to keep our eyes wide open. We are keeping these actinides in process, and there are always going to be losses in process. These are likewise going to end up in waste streams, and therefore we have to evaluate what is the relative risk with regards to transmutation in contrast with other disposal schemes. So, really to evaluate what the best scheme is in terms of its safety, we have to come up with a relative risk of all these systems and that gets to be the first hurdle in my way of thinking. The ones that are shown to be less risky are the ones that

I think you would be prone to go ahead and develop in contrast to those that have high risk factors. Now, the next things that alter your thinking are other environmental considerations, costs, policy conflicts and all of the others shown on the slide. Indeed, you cannot just sit there and evaluate the technical feasibility in terms of the gains in time by transmuting these materials because we have to reprocess them, transport them, refabricate them, etc. The judgement of technical feasibility is just the first step. Using these idealistic conditions, if you don't get reasonable gains in time by transmutation of actinides, forget it. You never will achieve these idealistic conditions, and you won't have to evaluate risk and other factors.

Steiner: In one of your slides, you referred to a study of alternative waste disposal methods which include accelerator based schemes. Did you also consider schemes based on deuteron breakup on targets such as lithium and beryllium and, if you did, were any conclusions reached?

Liikala: No, with qualifications from Bo (Leonard). We didn't, did we Bo?

Leonard: No, that type of a source just doesn't produce enough neutrons to do the job. The only accelerator source that has shown some potential of being attractive is a 1 BeV type proton accelerator, where the proton beam is impacted directly on the fission product target that you want to get rid of. You do your transmutation both by internuclear cascades and evaporation process on the incident particles and also by reactions caused by the secondary neutrons. There is a Japanese study that indicates that there is some technical feasibility if an accelerator with the required current at these energies could be built.

Hertzberg: What current do you need at a billion volts?

Leonard: The current? An ampere or more.

Dudziak: Bo, are you considering that as a parasitic use of the accelerator or costing the entire accelerator based upon that use?

Leonard: The Japanese are considering it purely as a transmutation tool because they apparently don't have any salt beds. They have a real storage problem.

Dudziak: You could look at that as a parasitic use of an existing or future accelerator.

Leonard: Yes, as a matter of fact most people haven't even considered recovery of the thermal energy that has been deposited in the target but it's so close to being technically nonfeasible that most people are sitting back waiting for development of the required accelerator technology.

Liikala: I would like to ask one question of our utility friends. What roughly is the capital cost investment in safety systems for, say, the LWR; the ECCS, and control rod systems in terms of percentages? Ten? Thirty? Fifty?

Huse: I don't think we really know the answer. Certainly I don't. The cost changes. We had a unit that was supposed to go on line in 1971 and now we think 1976 will be a good year to look for it. It costs us some $300,000 per day for the outage. Many of these changes are brought about by changes in safety requirements. So how much of this is chargeable directly to safety requirements? I'm not sure, however, it's a large number.

Miley: Along that line, what do you consider to be the most critical, i.e., the maximum credible accident in this type system? You have eliminated some possibilities, but what are left?

Huse: Someone putting a coffee can in a pump, I suppose.

Liikala: I think it would probably have to do with dynamic structural considerations. The reason being is that you have the tendency to design structures on the basis of static considerations for which you have never done a dynamic evaluation. We haven't done any evaluations of the dynamic characteristics for any types of structures. We have done a rough feasibility cut, and, firstly, we know it's not the reactivity insertion accident which needs to be mitigated because we are not critical; secondly, there doesn't seem to be configurations that could lead to a supercritical configuration. As important, we have some studies which illustrate a reasonable margin between fuel melting temperatures and projected fuel temperatures under loss of coolant conditions. However, we haven't done any analyses with regard to large forces on first vacuum walls and counteraction systems. Probably the principal design basis accident is seismic related.

Grace: A lot of comfort is being derived from the assumption that the blanket is going to be made subcritical, but I think no matter how good the arguments are going to be against blanket failure, meltdown, we will still have to contend with the issue. Have any considerations been given or any calculations be made of more critical configurations of the blanket material and whether or not a design criterion would have to be imposed in that regard.

Liikala: We haven't. However, I see that Don Dudziak at LASL has a comment.

Dudziak: Well, we did for our theta-pinch with the LMFBR-type blanket which we studied earlier for a U-238 fuel cycle. We did a loss-of-lithium accident. In fact, I was going to raise another question, which is whether you allow lithium cooled systems in your matrix there?

Liikala: As far as fission technology is concerned?

Dudziak: Yes.

Liikala: No.

Dudziak: But with the liquid lithium there is a fire hazard also.

Liikala: Right, but lithium coolant is not fission technology.

Dudziak: Our particular design did not go critical upon loss of lithium, but it is a problem. One gains a lot of reactivity by losing liquid lithium.

Leonard: Maybe I misunderstood the question on criticality, but yes, for ordinary burnup we did calculate criticality both at room temperature and hot operating conditions. In the various first calculations there were only four time steps involved. We calculated the source of criticality in both hot and cold conditions.

Grace: But how about distorted geometry?

Dudziak: We looked at that also by just filling our plasma chamber with blanket material and then determining reactivity.

Liikala: I would interpret many of the questions with regards to LMFBR technology, as being related to the dynamic behavior of single fuel elements becoming the source term for propagating core meltdown or core destructive accidents, i.e., local events propagating though the core. I wouldn't expect this question being vastly different in a hybrid blanket. So even though you may be subcritical under static conditions, in the event that you may have some local disturbances (i.e., in individual LMFBR fuel channels) you'd probably have to design the system to give sufficient margin against that sort of thing.

UCRL - 76525
PREPRINT

LAWRENCE LIVERMORE LABORATORY
University of California/Livermore, California

MAJOR FEATURES OF A MIRROR FUSION - FAST FISSION HYBRID REACTOR

R. W. Moir, J. D. Lee, R. J. Burleigh, W. L. Barr, J. H. Fink
G. W. Hamilton, D. J. Bender, G. A. Carlson, W. L. Dexter,
C. L. Folkers, and M. A. Peterson

February 4, 1975

This paper was prepared for submission to
Meeting on Fusion-Fission Hybrids

MAJOR FEATURES OF A MIRROR FUSION - FAST FISSION HYBRID REACTOR*

R. W. Moir, J. D. Lee, R. J. Burleigh, W. L. Barr, J. H. Fink,†
G. W. Hamilton, D. J. Bender, G. A. Carlson, W. L. Dexter,
C. L. Folkers, and M. A. Peterson

Lawrence Livermore Laboratory, University of California
Livermore, California 94550

February 4, 1975

ABSTRACT

A conceptual design has been made of a fusion-fission reactor. The fusion component is a D-T plasma confined by a pair of magnetic mirror coils in a Yin-Yang configuration and sustained by hot neutral beam injection. The neutrons from the fusion plasma drive the fission assembly which is composed of natural uranium carbide fuel rods clad with stainless steel and is cooled by helium. We have shown how the reactor can be built using essentially present day construction technology and how the uranium bearing blanket modules can be routinely changed to allow separation of the bred fissile fuel of which ~1200 kG of Plutonium are produced each year along with the ~750 MW of electricity.

*Work performed under the auspices of the U.S. Atomic Energy Commission

†J. H. Fink, on loan from Westinghouse Research Laboratory, Pittsburgh, Pennsylvania 15235.

This report was prepared for inclusion with a collection of works presented at a meeting at AEC Headquarters, Germantown on fusion-fission hybrid reactors, December 3-4, 1974. A complete report is being prepared to which the reader can refer for more complete details.

Introduction

It may be possible and advantageous to obtain energy and/or fissile fuel from the combination of fusion and fission. The major advantage of combining fusion and fission is that the weakness of each are offset by their respective strengths. Fusion is neutron rich but power poor while fission is neutron poor but power rich. A number of combinations of fusion reactors and fission reactors have been suggested. This report describes our work on the conceptual design of one particular fusion-fission hybrid reactor. The fusion reactions takes place in a magnetic mirror confined hot D-T plasma. The 14 MeV neutrons resulting from the fusion reactions drive a subcritical fission blanket which surrounds the plasma. The primary 14 MeV neutrons and secondary neutrons from the fission process and (n, 2n), (n, 3n) reactions produce fission of ^{238}U and ^{235}U (or Pu) and produce ^{239}Pu and Tritium as well as releasing large quantities of energy.

It should be possible to utilize one or several of the essentially existing fission technologies for the blanket. In a previous study[2] we looked at a Thermal lattice blanket based on the General Atomic Graphite moderated, helium cooled reactor technology. For this study we chose a fast spectrum blanket[3] based on helium cooled breeder reactor technology present under development.

In our present study we show conceptual engineering approaches to problems such as routine removal of uranium bearing blanket modules for recovery of bred fissile fuel, replacement of radiation damaged structures, and the steady operation of high powered efficient neutral beam injectors. The present design combines fuel production with electrical power production. We have succeeded in obtaining a design which we think could be built with essentially present day technology, and would produce large amounts of fuel (1200 kG of plutonium per year) and electricity (750 MW_e). The following section gives a general description of the mirror fusion-fission conceptual design.

This paper summarizes the results of our hybrid design study by briefly describing the Major Features of the specific Mirror Fusion-Fast Fission Reactor we developed. The detailed report of this design study is in preparation. The detailed report will include a discussion of safety aspects and a cost estimate.

FUSION-FISSION REACTOR

GENERAL DESCRIPTION OF REACTOR AND FACILITIES

The reactor and its associated facilities consist of the following main components and systems. (See Figs. 1-9)

1) The Main Superconducting Coils which generate the magnetic mirror field for the confinement of the plasma in which the fusion reactions take place.

2) The Coil Clamps which restrain the main coils from distorting under the enormous forces generated by the magnetic field.

3) The Auxiliary Superconducting Coils which control the shape of the leakage flux from the plasma.

4) The Injectors which supply the fuel (deuterium and tritium) to the plasma.

5) The Uranium Blanket which surrounds the plasma and in which the fission reactions take place. The power generated by the fusion reactions in the plasma is multiplied by a factor of about 12 by the fission reactions in the blanket. This blanket also contains lithium for breeding tritium to feed the injectors. The blanket is divided into modules to facilitate removal and recharging.

6) The Helium Cooling System which transfers the heat from the uranium blanket to the steam generators.

7) A Thermal Power System which converts the thermal power from the steam generators into useful electrical power via conventional steam turbines and electrical generators.

8) The Direct Converters which convert the kinetic energy of the leakage flux from the plasma directly into electrical energy.

9) A Vacuum Envelope which surrounds the main coils, the plasma, the uranium blanket, the direct converters, and the injectors.

10) Vacuum Pumping Systems to remove gas from the plasma space, the injectors, and the direct converters.

11) Shielding to:

    a) Reduce neutron heating in the superconducting coils.

    b) Reduce neutron damage to the superconducting coils.

    c) Provide biological protection.

12) Handling Systems to facilitate:

    a) Removal and recharging the uranium blanket modules.

    b) Serviceing the injectors and other elements of the machine.

13) A Tritium Handling System to recover tritium.

14) A Building to house the reactor and its associated facilities.

MAIN SUPERCONDUCTING COILS

The configuration of coils shown here (Fig. 1) is shown as "Yin-Yang." This is really a variation of the familiar "Baseball" design (in which the shape of the coil is similar to the seam of a baseball) which has been used in a number of mirror devices. The Yin-Yang arrangement results if the conductors of a baseball coil are divided into two separated bundles. The two coils are identical, but are rotated 90^{o} to each other. In fact, this is true for the two halves of the whole reactor. The Yin-Yang arrangement provides space between the coils for the four injectors.

The individual superconductors are fine filaments of Nb Ti embedded in copper bars. These bars are separated by suitable spacers to provide passages for the liquid helium coolant. The windings are enclosed in a stout stainless steel coil-tank. This tank in turn must be enclosed in a suitable thermal shield which may consist of a liquid-nitrogen-cooled copper plate and a multilayered reflective heat shield (not shown in drawings).

THE COIL CLAMPS

Enormous forces are produced by the coils. One component of these forces tends to straighten out the arc of the coil into a straight line. This force can be resisted by the coil tank itself.

Another component of these forces tends to open up the coil gap. As it is impractical to make the coil tank strong enough in itself, a massive clamp is provided to resist this force. This clamp is in the form of a picture-fram bent to conform to the contour of the coil. It is fabricated from non-magnetic stainless steel plates bolted together. Because of the way the moments are distributed, the depth of the clamp may be somewhat smaller at the center than at the ends. The coil clamps are cooled to the same temperature as the coils. This means that these members must also be surrounded by the same type of thermal shield as the coil. If the clamps were not cooled along with the coils there would then be the problems of differential thermal contraction, and the transmission of large forces across thermal barriers.

There is also a large force urging the two coils toward each other. This force can be resisted by the internal shielding immediately surrounding the blanket, as the shielding in this area is largely of stainless steel.

THE AUXILIARY SUPERCONDUCTING COILS

With the main coils alone, the magnetic flux lines would start to expand outwardly as soon as they emerge from the coil gap. The leakage plasma would, of course, expand along with the magnetic flux lines. The leakage plasma would then strike the vacuum gates for the blanket modules. To avoid this, and to lead the leakage plasma into the direct converters, it is necessary to add the auxiliary coils. These are superconducting coils of the same general construction as the main coils. It is also necessary to surround these coils with shielding.

THE INJECTORS

The mirror configuration utilizing Yin-Yang coils makes it possible to introduce high energy neutrals into the reactor via four separate injectors. Each of these is roughly pyramidal in shape with a rectangular base and apex angles of 30^{0} and 60^{0}, and holds seventy modified Berkeley sources mounted in five columns of 14 units. Each injector is located in a large pumping chamber serviced by four 48" diffusion pumps. All of the sources are focused, seven meters away, at the apex of the pyramid which is truncated and consists of an opening in the reactor wall roughly 20 by 50 centimeters.

The beams coming from each column of 14 sources go into a common liquid nitrogen cooled neutralizer, about one meter long, and subsequently pass between a set of electrodes which serve to recover the energy of that portion of the beam that was not neutralized. On each side of the column of beams is mounted a cryogenic panel which serves to pump away the unwanted neutral gas and also provides magnetic shielding. With the provision of adequate pumping capability, continuous operation is achieved by sliding each cryopanel back out of service, between the columns of sources into the pumping chamber, where they can be out-gassed and subsequently reintroduced into the injector.

THE URANIUM BLANKET

The gross blanket geometry is defined by the plasma and coil geometries for a minimum-B Yin-Yang magnetic mirror. The blanket design is shown schematically in Figure 3, and is based on a modularization concept where the blanket is constructed from wedge shaped modules whose assembled geometry conforms to the Yin-Yang magnetic field. Each module, in turn, is composed of a collection of domed, cylindrical pressure vessels (submodules) welded to the module base.

The submodule is the basic component of the blanket, and contains an inner fast fission zone and an outer tritium breeding zone. These zones are composed of natural uranium carbide and lithium aluminate contained in wire-wrapped pins in a hexagonal array. Helium coolant is supplied to, and removed from, the submodule by plena in the module base. The helium flow within the submodule is used to first cool the pressure vessel wall and then to remove the heat from the fuel pins and the heat and tritium from the breeding pins.

A summary of blanket parameters is given in Table 1. These parameters are for a new blanket as it starts life with natural uranium fuel. As the blanket is exposed it will enrich itself in plutonium causing Energy Multiplication (M) to increase and the Net Fissile Breeding Ratio to decrease. With the refueling scheme we are presently considering, the life time average Energy Multiplication will be ~11 and the lifetime average Net Fissile Breeding Ratio ~1.2 atoms/fusion.

PRELIMINARY PARAMETERS (INITIAL) FOR THE

FAST FISSION BLANKET SUBMODULE

Table 1

Geometry	
Fission Zone Length	20 cm
Tritium Breeding Zone Length	60 cm
Pressure Vessel Diameter	30 cm
Materials	
Fuel	UC
Tritium Breeding	$LiAlO_2$
Structural	Stainless Steel
Neutronics	
Energy Multiplication	8.1
Tritium Breeding Ratio	1.1
Net Fissile Breeding Ratio (Atoms/DT Neutron)	1.2
Fissile Production/Fissile Consumption	13
Thermal Hydraulics	
Fission Power Density (max.)	100 w/cm^3
Coolant Pressure	20 ATM.
Clad Temperature (max.)	700^{o}C
Pressure Vessel Temperature (max.)	500^{o}C
Coolant Inlet Temperature	300^{o}C
Coolant Outlet Temperature	600^{o}C
Pumping Power/Thermal Power	.01

THE HELIUM COOLING SYSTEM

The heat from the uranium blanket is transferred to the steam generators by helium. To ensure safety, the entire helium system is duplicated. There are two inlets and two outlets on each blanket module. Also, two inlet manifolds and two outlet manifolds are provided for each side of the reactor. Two helium circulators are provided in each circuit. In addition, the helium is everywhere enclosed by two walls. While in the module the helium is enclosed, of course, by the module itself, but it is also enclosed by the vacuum envelope surrounding the entire reactor. All helium lines to and from the reactor are double-walled. Likewise, within the steam generators the helium is again doubly enclosed.

Each of the eight helium circulators (Fig. 5) consists of a helium compressor with a steam turbine main drive and a water turbine (Pelton wheel) auxiliary drive. The latter is used to supply power to the circulators when the steam supply is not available. This arrangement is similar to that used in the Fort St. Vrain, Colorado, helium-cooled nuclear reactor.

An auxiliary after-heat cooling system also using helium, is provided to cool the mdoules while they are being withdrawn and transported for processing. Valves in the main cooling lines are provided at each module so that the module can be made gas-tight during removal and transport.

THE THERMAL POWER SYSTEM

This system converts the thermal power from the steam generators into useful electrical power. We have studied this aspect of the plant only very superficially. We, therefore, assume that this system employs conventional steam generators, steam turbines, and electrical generators.

We are patterning this system after that of the Fort St. Vrain helium-cooled nuclear reactor. We employ 24 vertical type steam generator modules. The hot helium from the reactor flows vertically through the casing giving up its heat to helically wound water and steam tube bundles surrounding a central header. These steam generator modules are shown in the drawings (Fig. 5) in the main reactor room and the turbine and electrical generators in the adjacent generator room.

THE DIRECT CONVERTERS

The power in the inevitable leagage flux from a mirror type reactor may be largely recovered by the use of direct converters. The type of direct converter we propose to use here is similar to that described in UCRL-7436, "A Preliminary Engineering Design of a Venetian Blind Direct Energy Converter for Fusion Reactors."

As previously described, the auxiliary coils guide the leakage flux into the expander tanks and into the direct converters proper. The beam of ions is allowed to expand to a peak power density of about 100 watts per square centimeter. We have shown experimentally that at this power density the elements of the direct converter may be cooled by direct radiation. The converter we have shown on the drawings is a single-stage type employing a pair of wire grids followed by a ribbon grid. The particles see the first wire grid at zero potential then the second wire grid at negative potential. The function of this pair of grids is to reflect the electrons and allow the positively charged ions to continue on through. The ions next see the ribbon grid at a high positive potential. This ribbon grid acts as an ion trap. The wires and ribbons are cooled by radiation to water-cooled plates (not shown) lining the expander tank.

When the ions are neutralized, large quantities of gas are produced. This gas is pumped by cryopumping chambers behind the ribbon grid. The ribbon grid is opaque to the directed ions, but presents only a modest impedance to the flow of the neutral gas.

THE VACUUM ENVELOPE

The entire reactor, including the direct converters, is enclosed in a vacuum envelope. It is of conventional welded construction. Other than the flange joints provided for the removal of the cryopumping chambers and for access to the direct converters, no joints are provided for major dismantling. The tank would be simply welded in place and expected to remain there for the life of the machine. The spherical portion of the tank surrounding the reactor proper is supported internally by the shielding and the magnet structure. The expander tanks, however, require external supports to resist the atmospheric loading. This external stiffening structure also supports the shielding around the expander tanks and also supports the tracks for the blanket module trucks. As proposed, this structure consists of radial members connected at their inner ends to the spherical tank. At their outer ends, each pair of these radial members is supported by a column. At their ends these columns are connected together by a box-beam which is curved in a circular arc conforming to the shape of the expander tank.

VACUUM PUMPING SYSTEMS

The large volume of gas produced at the direct converters is pumped by cryopumping chambers behind the ribbon grids. These cryopumping chambers contain liquid-helium-cooled panels thermally shielded with liquid-nitrogen-cooled panels and multiple reflective heat shields. Any one panel may be isolated in automatic sequence for "defrosting" by means of a "jalousie" type valve. The cryopumping panels may then be warmed up and the evolved gas collected through a mechanically pumped vacuum manifold.

To maintain a good vacuum in the plasma area, additional cryopumping panels (not shown) may be introduced into the intermediate volume formed by the constriction between the coils and the entrance to the expander tank.

Large quantities of gas are also produced in the injectors. This gas is pumped by a combination of cryopumping and conventional diffusion pumps.

SHIELDING

Shielding is required to: a) reduce neutron heating within the superconducting coils, b) to reduce neutron damage to the superconducting coils, and c) to provide biological protection.

A layer of boral (not shown) is placed immediately outside the coil thermal shield to abosrb thermal neutrons. This material has the property of producing alpha particles rather than more penetrating gamma rays. Next is a layer of lead to capture gamma rays. Outside of this is the main bulk of the shielding. Adjacent to the coils and immediately outside the uranium blanket, this main shielding consists of stainless steel and borated water. The outer layer of shielding immediately within the spherical vacuum tank is of concrete encapsulated in stainless steel tanks.

Additional shielding is provided immediately outside the vacuum envelope to prevent the escape of radiation through the injectors and through the direct converters. That part of this shielding which 'sees' the plasma directly may consist of about ~1 meter of 80% stainless steel and 20% borated water followed by ~1 meter of ordinary concrete. This would be used behind the direct converters and behind the injectors. On the sides of the expander tanks and on the sides of the injector enclosures, ordinary concrete may be sufficient.

TRITIUM HANDLING SYSTEM

The tritium handling system is divided into two primary systems: a helium coolant tritium recovery system and a purification and isotopic separation system. Tritium containment in the case of accidental release is accomplished by special design features in the outer walls and by secondary pruifiers which will remove tritium from the room atmosphere.

Helium coolant will carry tritium picked up during passage through the blanket. One percent of the hot helium coolant flow is directed to the tritium recovery system.

Traces of water are removed by a regenerative heat exchanger and cold trap operating at LN temperature. The coolant is reheated in the same heat exchanger and then continues to a vanadium diffuser. Here the helium passes through an array of vanadium tubes, while the tritium diffuses through the vanadium. The tritium is recovered from an outer chamber by a pumping system and can be sent directly to the injectors or stored elsewhere.

Tritium and deuterium recovered from the Vacuum Pumping systems must be purified and isotopically separated before returning to the injectors. The gas is purified by passing it through palladium diffuser. Cryogenic distillation columns then separate the gas into tritium and deuterium streams which are reinjected immediately. These components are similar to those used in the FERF design.

Rooms containing components with large amounts of tritium will be sealed with a nitrogen atmosphere. In case of an accidental release within the room, a purifier will catalytically oxidize the tritium and hold it on molecular sieve beds. This equipment follows the FERF design.

Tritium can also permeate all outer walls, however, slowly, and very large surface areas are involved. We are proposing a new "getter wall" which can be readily adapted to a variety of situations: a structural wall is coated on the outside with a getter forming stable hydrides and, in turn, is sandwiched with a protective outer layer. Operated at ambient temperature, the getter need hold only the small amount of tritium permeating the structural wall.

PLASMA PHYSICS PARAMETERS

The injection energy was chosen to maximize the effiency of the fusion part of the reactor. Injector efficiency decreases while the fusion efficiency increases with increasing energy. Equal velocity of D^o and T^o is necessary for equal penetration. The values of injected energy and other plasma parameters are given in Table 2. The presence of the diamagnetic plasma decreases the magnetic field at the center and therefore, increases the magnetic mirror ratio to $R = 7.8$ from $R_{vac} = 3.5$. Q, the ratio of fusion power to injected power, is approximately equal to 1.0. This means that a large amount of power must be recirculated through the injector which we calculate to be 70% efficient at these energies. The fusion power of 620 MW means that 9×10^{19} 14 MeV fusion neutrons per sec are produced. Most of these enter the blanket and produce fuel and more power. An equal number of 3.5 MeV α-particles are also produced. Their orbits are too large for them to be confined adiabatically by the magnetic field, and they quickly scatter out along field lines. Their energy is recovered in the thermal cycle rather than by direct conversion. The direct converter with its thermal bottoming cycle has a combined efficiency of

64% for ions that reach equilibrium before leaking out of the magnetic field into the direct converter.

PLASMA PARAMETERS

Table 2

n	(ions density at center)	$7 \times 10^{13} cm^{-3}$
W_D	(D^o injection energy)	100 keV
W_T	(T^o injection energy)	150 keV
P_F	(fusion power)	260 MW
P_{inj}	(injected power)	260 MW
Q	(ratio P_F/P_{inj})	1.0
S	(fusion neutrons/sec)	$9 \times 10^{19} sec^{-1}$
β	(particle pressure/magnetic pressure	0.8
$n\tau$	(density times containment time)	$2.5 \times 10^{13} cm^{-3} sec$
r_p	(plasma radius at midplane)	3.5 m
L	(distance between mirrors)	25 m
B_m	(magnetic field at mirrors)	6.0 T
R_{vac}	(mirror ratio without plasma)	3.5

HANDLING SYSTEMS

Means must be provided for the removal and loading of the uranium blanket modules and for servicing the injectors, direct converters, and other elements of the reactor.

The shielding shown provides sufficient biological protection so that the reactor room may be entered as long as all parts of the shielding are in place. However, because of possible tritium contamination, which may result when the reactor has been opened for, say, replacement of ion sources, it will be necessary for operating personnel to wear protective air-tight suits with separate air supplies. When any part of the shielding is removed, the resulting radiation level, at least in the immediate vicinity of the opening, will be such that operating personnel must be excluded. This will be the case even when the reactor is not operating. It will be necessary, therefore, to provide for remote manipulation for all operations involving opening the shielding. This requires far more study.

The general arrangement of the uranium blanket modules are shown in Figs. 1 and 3. The procedure for removing a module is shown in Figs. 4a, 4b, and 4c. As shown, the modules are wedge-shaped and are arranged in a radial array, as in sections of a pie. Each module is provided with a set of stationary tracks which support the module and along which the module may be rolled. The removal paths for opposing pairs of modules cross each other. A port, with a vacuum-tight gate, is provided in the vacuum envelope for each module. A removable plug is provided in the shielding for each port. When a module is withdrawn a set of temporary tracks must be inserted to bridge the gap between the permanent tracks and the port. When a spent module is withdrawn it is, of course, very radioactive. It will, therefore, be necessary to insert the module into a lead casket. The caskets are mounted on carts which in turn run on sets of tracks on either side of each expander tank. A temporary cooling system removes the after-heat from the module during transportation. Twenty-four modules are shown in the drawings. To reduce the weight of the module and the casket it may be necessary to increase the number of modules.

The sequence for the removal of a module is as follows:

1. The shielding plug is removed.
2. The vacuum gate is opened.
3. The temporary tracks are put in place.
4. The temporary cooling lines are attached to the module.
5. The main helium valves at the module are closed.
6. The joints at the module in the main cooling lines are separated.
7. The module is rolled into the casket.
8. The cart is run to a position where the casket may be handled by the overhead crane.
9. The casket is taken by crane to the loading platform (see Fig. 5).

For loading a fresh module into the blanket, the above process is reversed. Remotely operated means (not shown) are provided for servicing the injectors, direct convertors, and other elements of the reactor.

LIST OF FIGURES

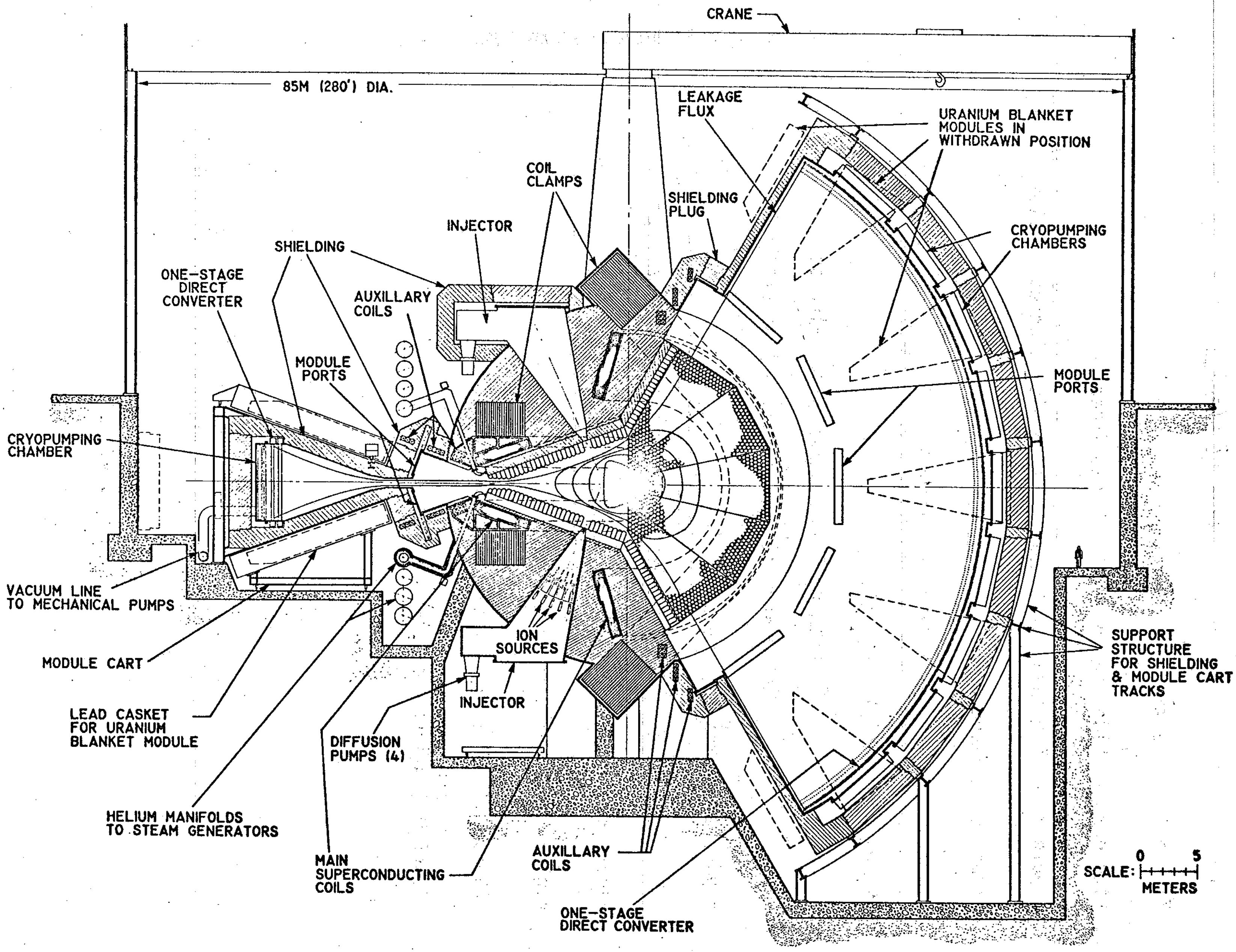


FIG. I, VERTICAL CROSS-SECTION THROUGH REACTOR

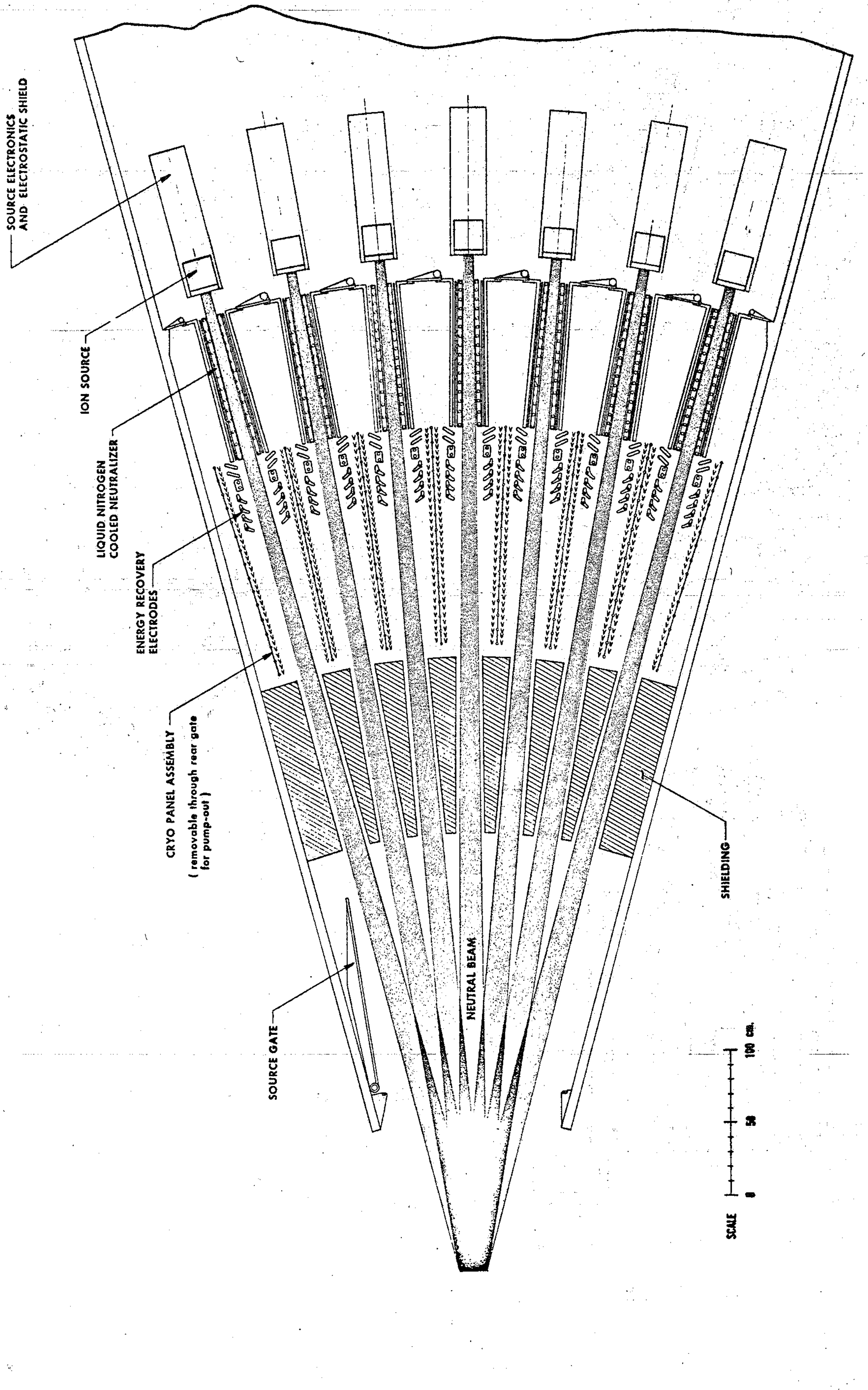
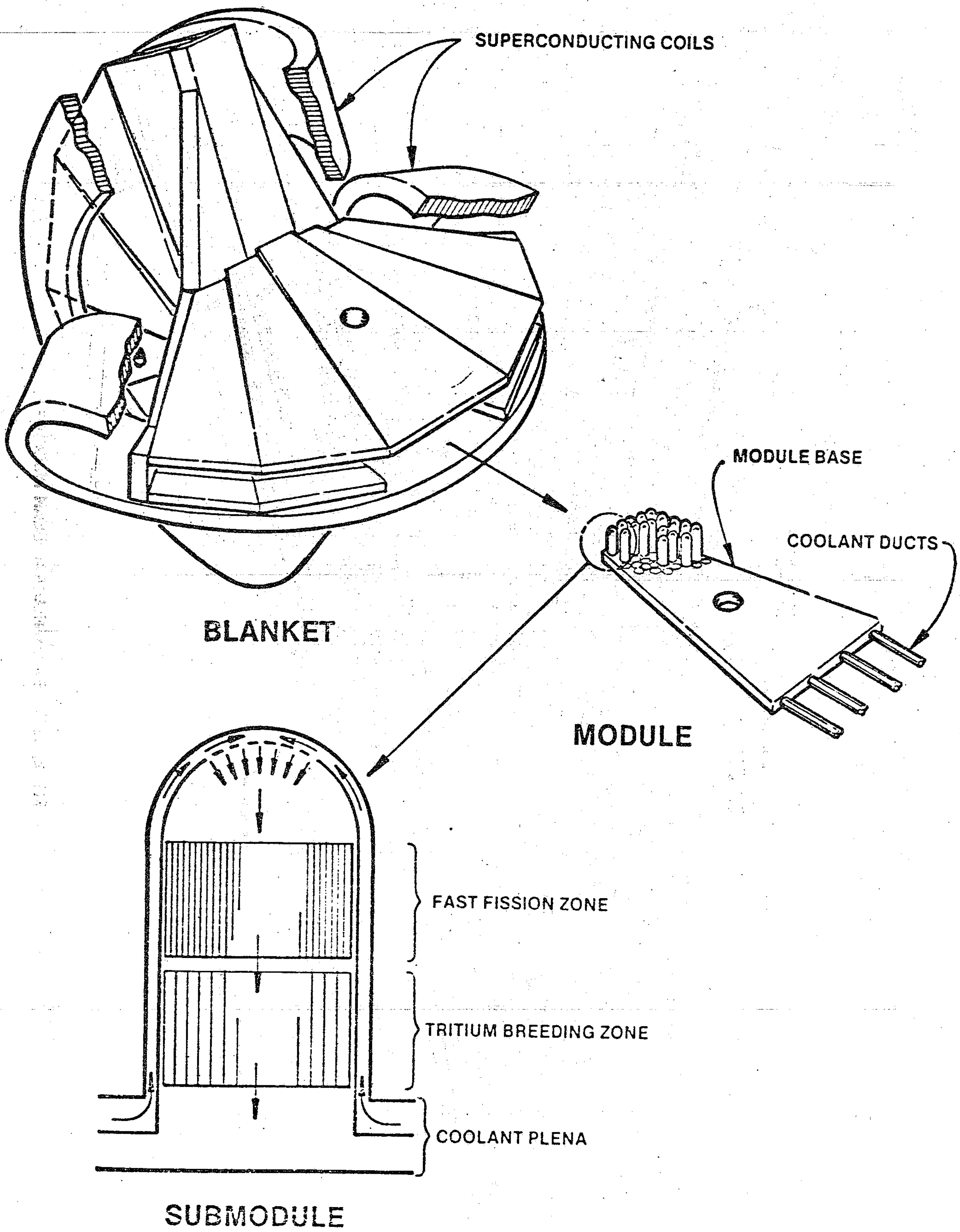


FIG. 2, SIDE VIEW OF NEUTRAL BEAM INJECTOR


SUPERCONDUCTING COILS
BLANKET
MODULE BASE
COOLANT DUCTS
MODULE
FAST FISSION ZONE
TRITIUM BREEDING ZONE
COOLANT PLENA
SUBMODULE

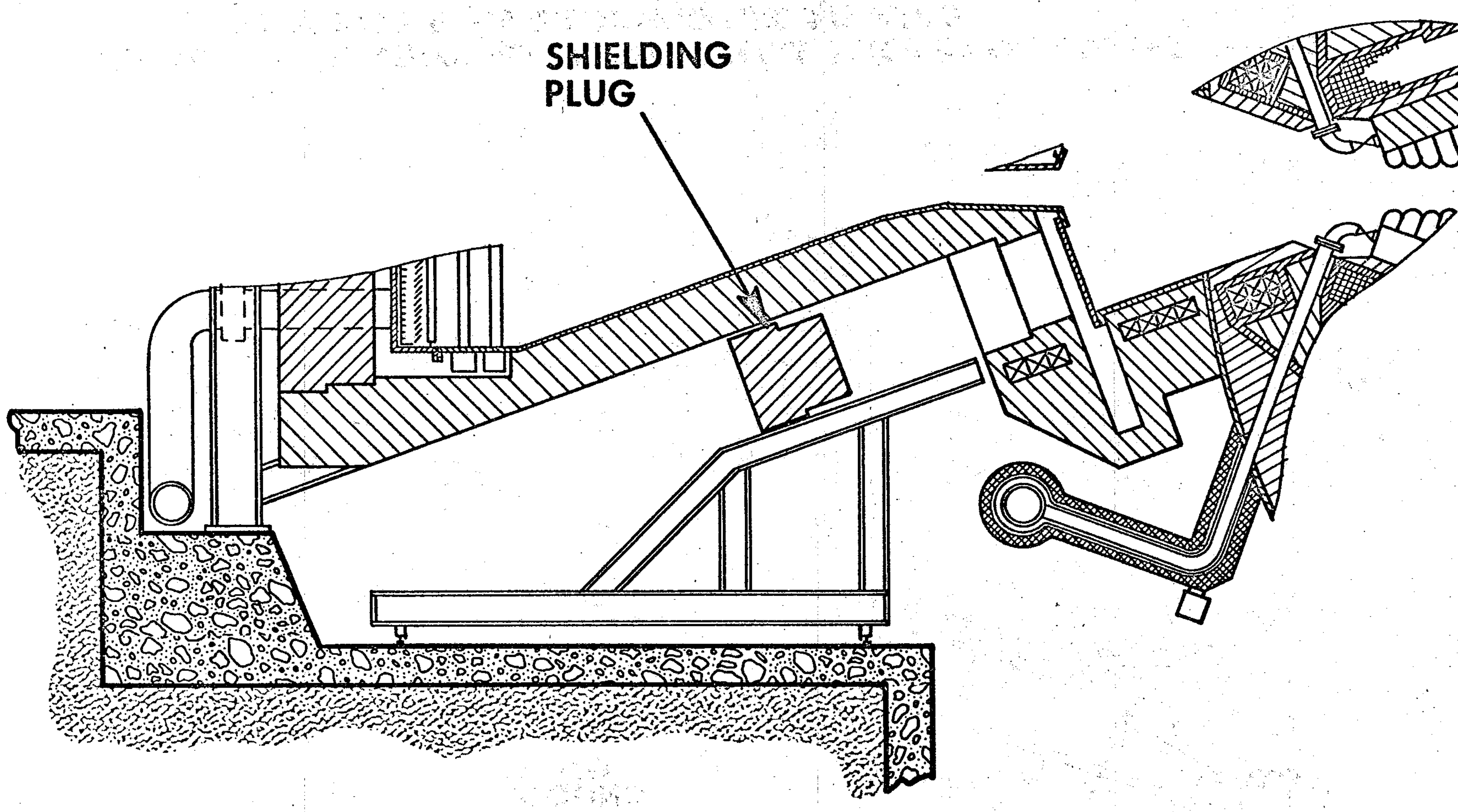


FIG. 4A, MODULE REMOVAL; T SHIELDING PLUG REMOVED

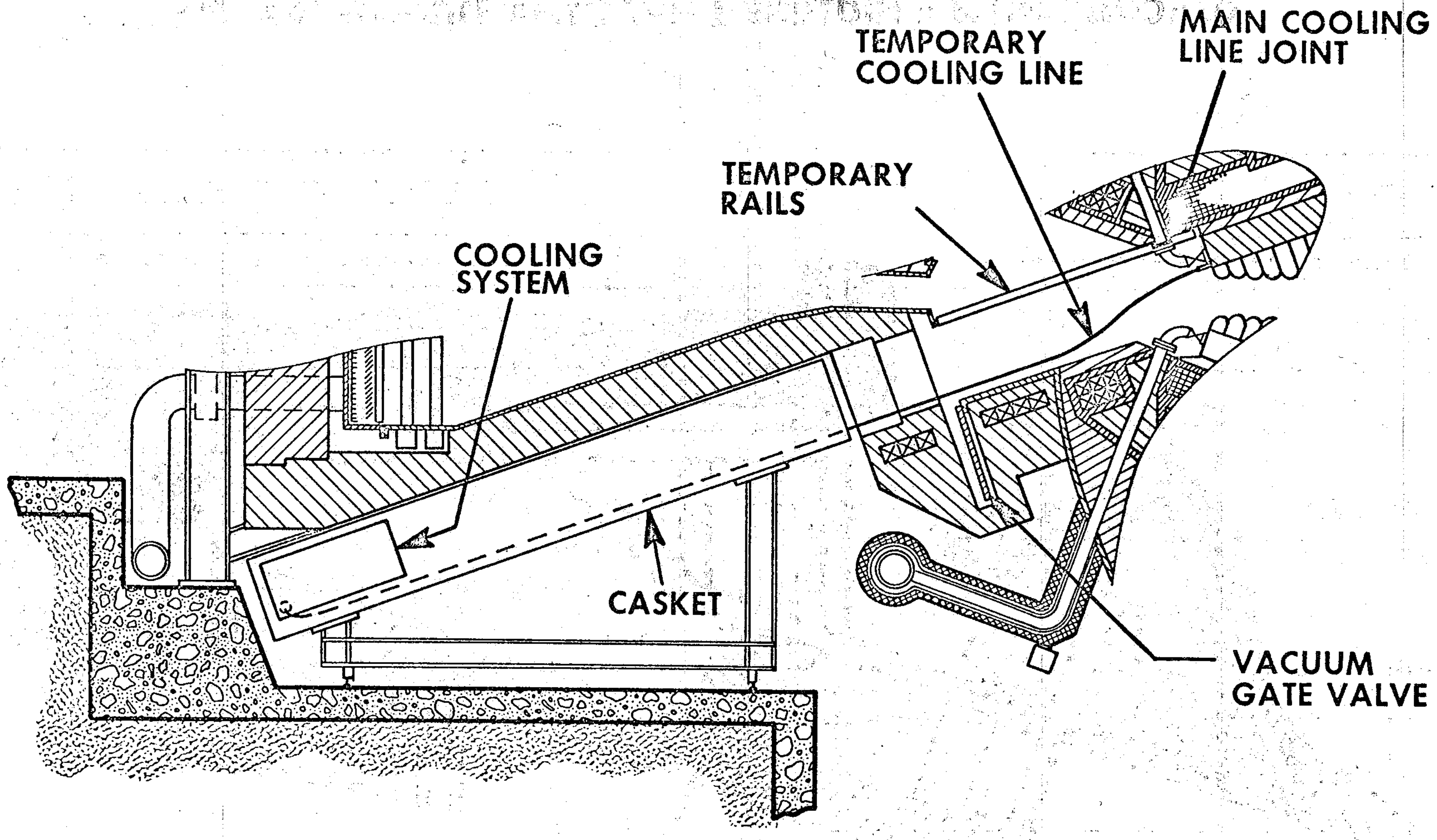


FIG. 4B, MODULE REMOVAL; CASKET IN PLACE, VACUUM GATE VALVE OPENED, TEMPORARY RAILS & COOLING LINE ATTACHED

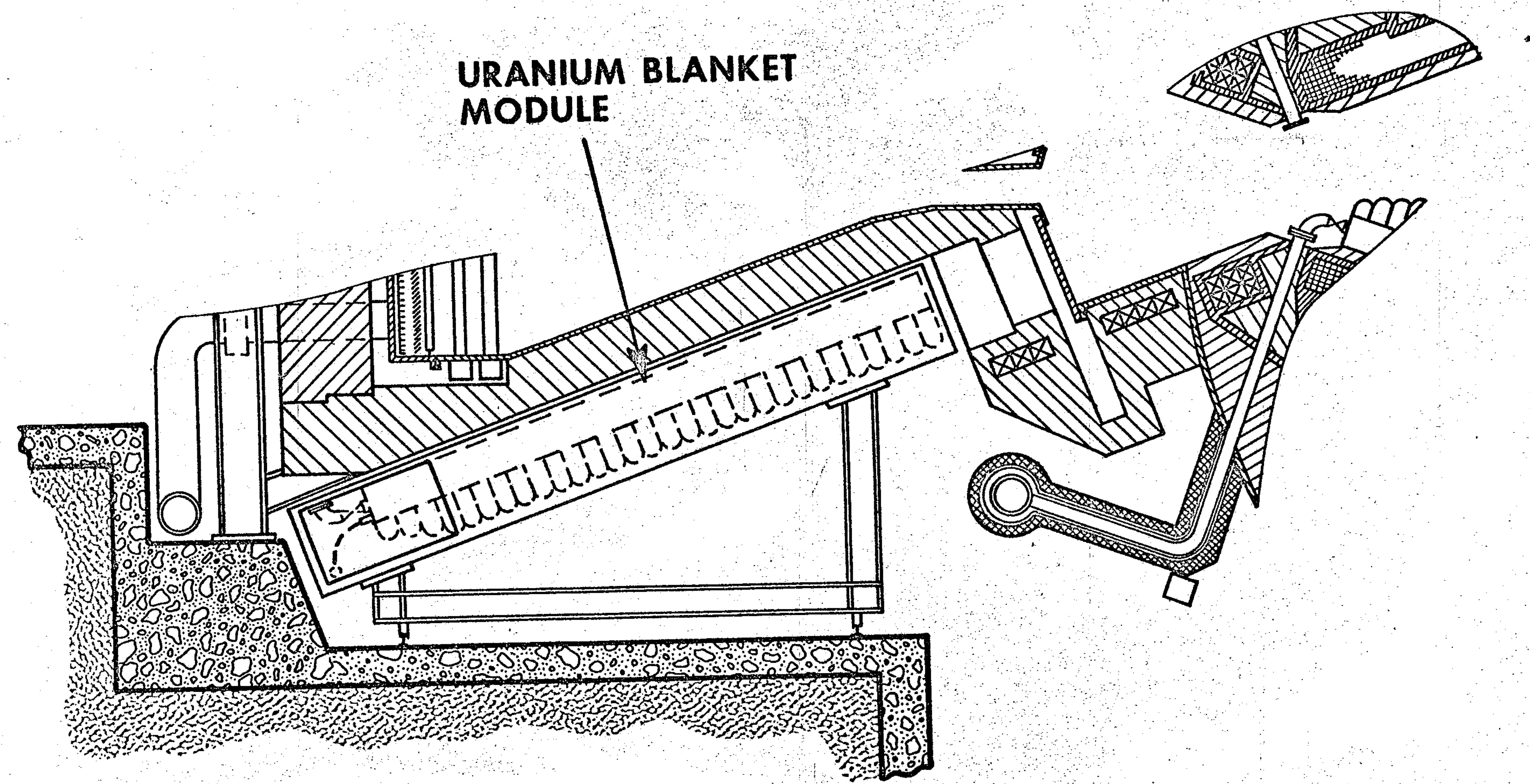


FIG. 4C, MODULE REMOVAL; MODULE IN CASKET

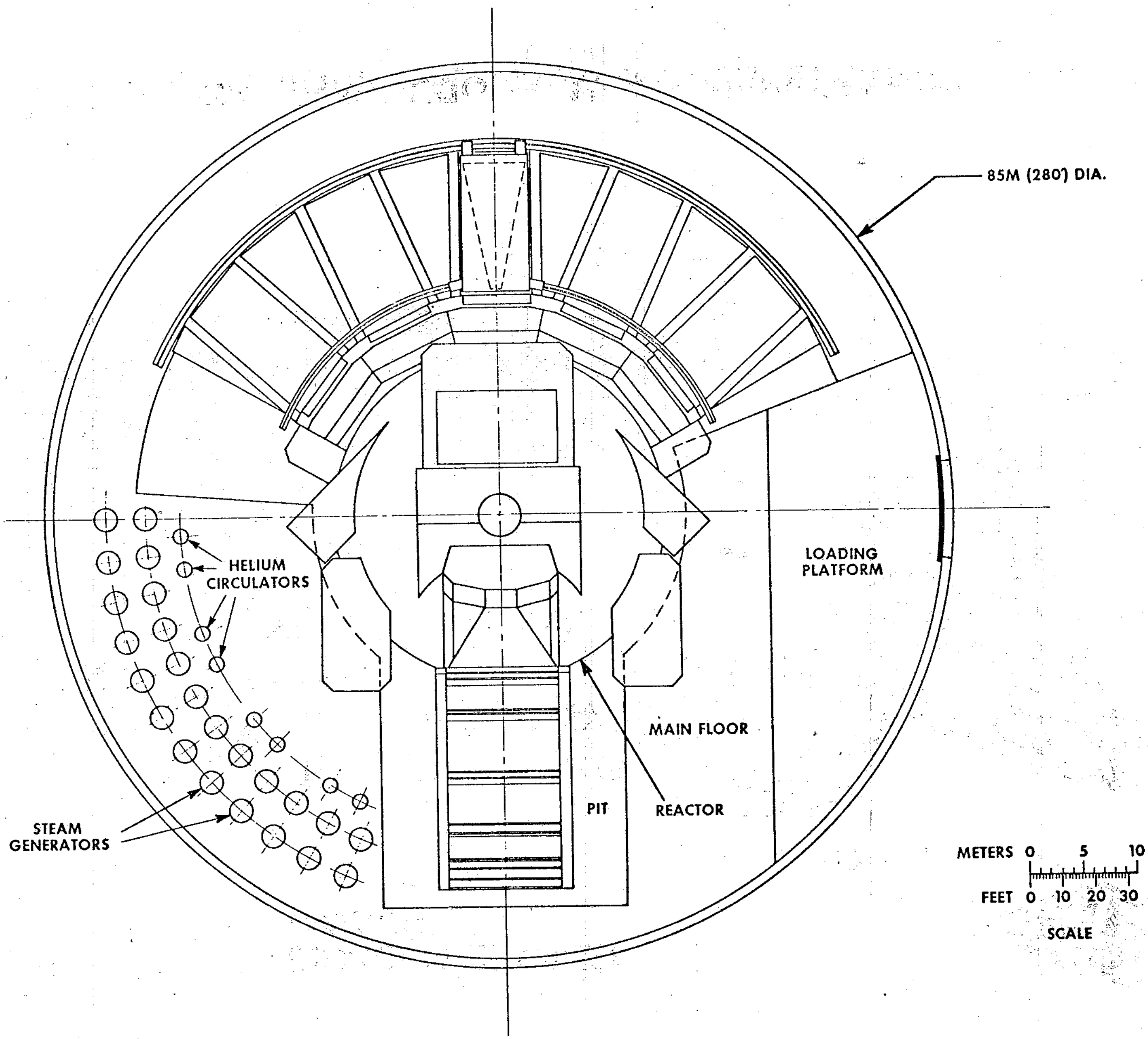


FIG. 5, PLAN VIEW OF REACTOR IN REACTOR ROOM

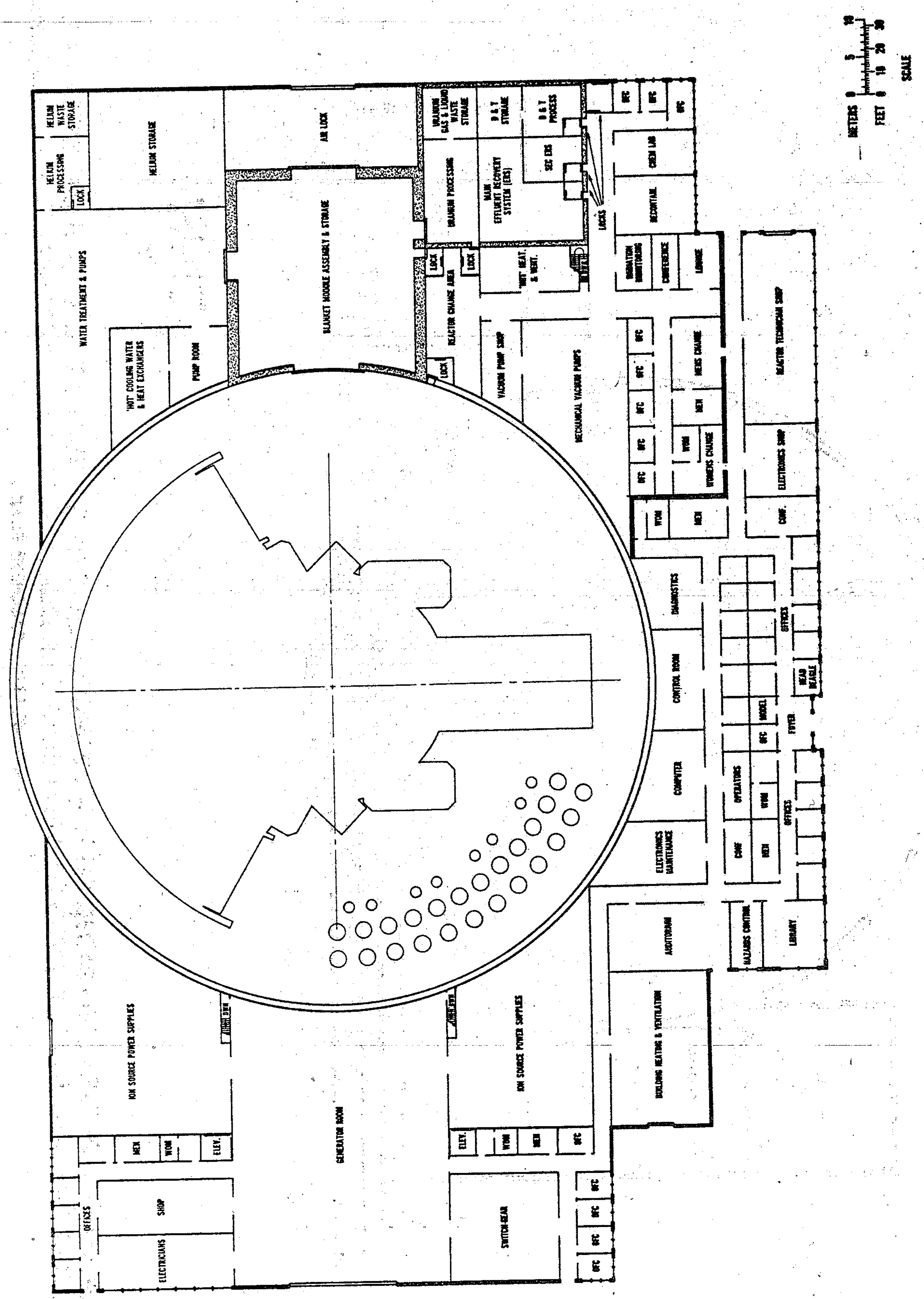


FIG. 6, BUILDING, MAIN FLOOR PLAN

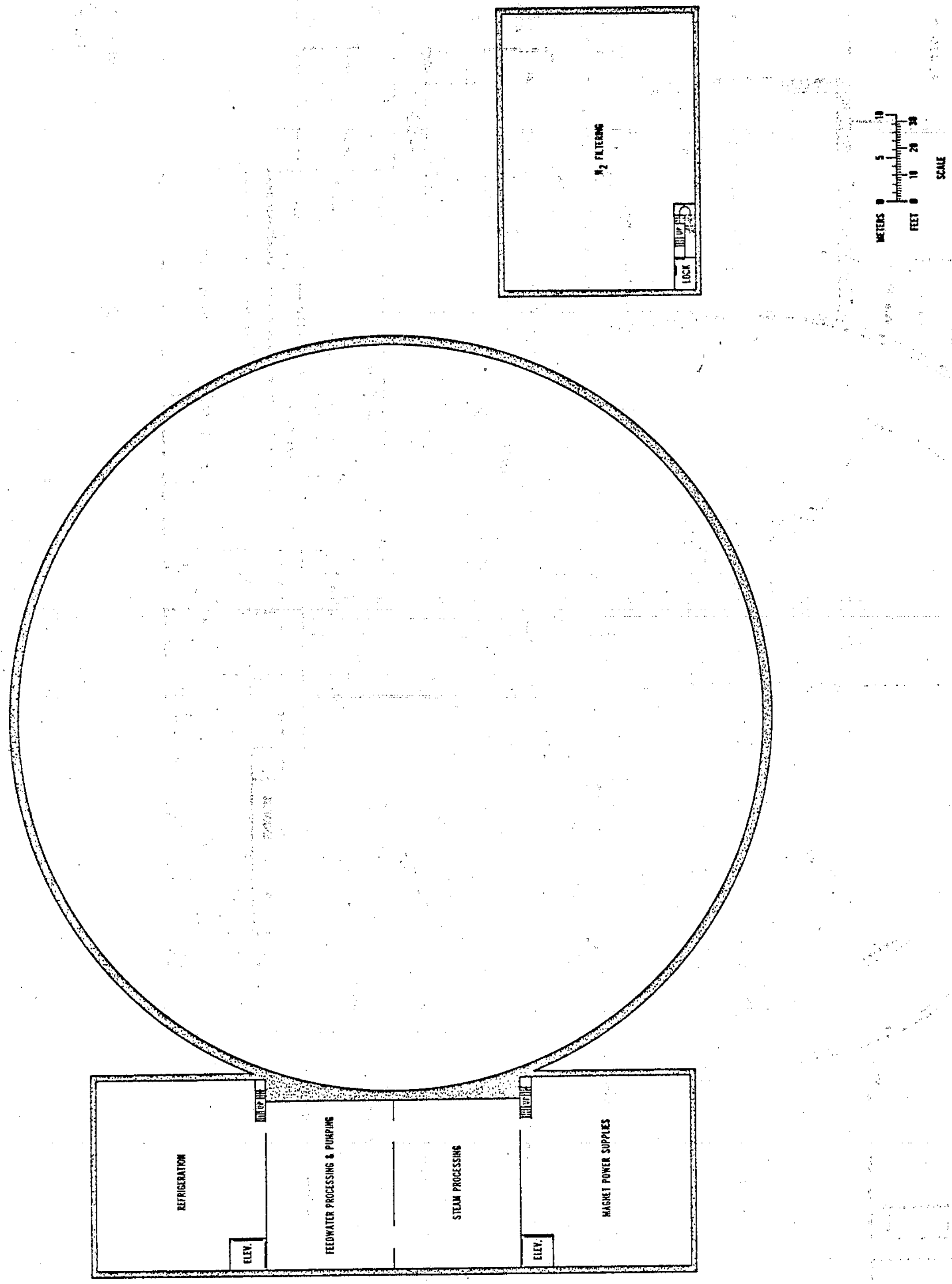


FIG. 7, BUILDING, BASEMENT PLAN

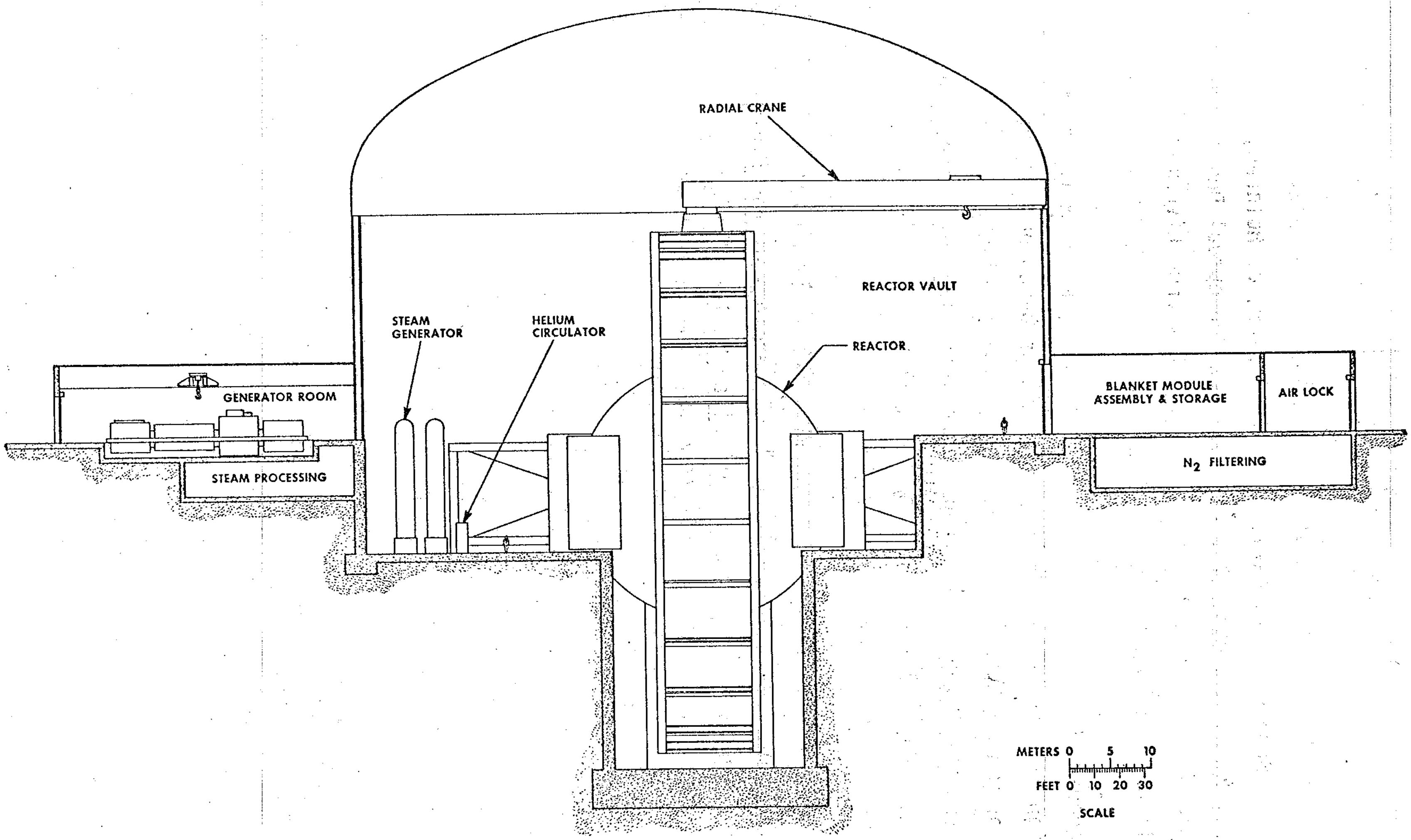


FIG. 8, BUILDING, SECTIONAL ELEVATION

REFERENCES

1) R. W. Moir, J. D. Lee, R. C. Burleigh, W. L. Barr, J. H. Fink, G. W. Hamilton, D. J. Bender, G. A. Carlson, W. L. Dexter, J. Holdren, C. L. Folkers, M. A. Peterson, "Progress on the Conceptual Design of a Mirror Fusion-Fast Fission Hybrid Reactor," Lawrence Livermore Laboratory in preparation.

2) W. C. Wolkenhauer, B. R. Leonard, Jr., A. M. Sutey, R. W. Moir, "Conceptual Design of a Fusion-Fission Hybrid Reactor based on a Mirror Fusion Reactor with a Subcritical Gas Cooled Fission Blanket," Proceedings of the First Topical Meeting on the Technology of Controlled Nuclear Fusion, San Diego, Ca., April 16-18, 1974. CONF-740402-P1, Vol. 1, p. 238.

3) J. D. Lee, "Neutronic Analysis of a 2500 MW_{th} Fast Fission Natural Uranium Blanket for a DT Fusion Reactor." Proceedings of the First Topical Meeting on the Technology of Controlled Nuclear Fusion, San Diego, Ca., April 16-18, 1974. CONF-740402-P1, Vol. 1, p. 223.

LLL Distribution List

W. L. Barr		L-386
D. J. Bender		L-383
R. C. Burleigh		L-383
W. L. Dexter		L-728
J. H. Fink		L-386
C. L. Folkers		L-460
G. W. Hamilton		L-386
J. D. Lee		L-383
R. W. Moir	(25)	L-386
M. A. Peterson		L-383
TID	(15)	L-3

External Distribution List

W. C. Wolkenhauer
Washington Public Power
Supply System
P. O. Box 968
3000 Geo. Washington Way
Richland, Washington 99352

QUESTIONS ABOUT FIFTH PRESENTATION

Lidsky: I have a comment that may be picking on a sore point in an otherwise beautiful design study. It appears that the total efficiency of the entire plant is very low.

Moir: The effeciency of conversion of nuclear energy to electricity was 30 percent. At 30 percent the plant is producing one third of its revenue as fuel, so in that context it is perhaps not too bad.

Lidsky: What was the net electric power output?

Moir: Five hundred megawatts.

Lidsky: I wonder, realistically, what penalty one pays for low thermal efficiency and even more what one will pay in the future. What penalty do you have to pay to raise the efficiency of the plant somehow?

Lee: Or conversely, one can say, how much does it cost to dissipate that heat at $35 per kilowatt or whatever.

Moir: I think it's a well posed question. I don't have anything to contribute to it right now. I think we have to do a cost analysis on this particular one, at that particular efficiency, taking the environmental impact of that thermal efficiency which probably means a wet and maybe a dry cooling tower. There will certainly be a cost tradeoff in raising the efficiency.

Wolkenhauer: Apparently what you have done is drop the multiplication of the blanket from about 30 or so to about 10 and then added a direct converter from your previous design. The numbers then come up about the same.

Moir: The two effects counterbalance pretty much. Raising the injector efficiency from 30% to 70% with a beam direct converter just about compensates for lowering the blanket multiplication from 39 to 9.

Wolkenhauer: And, my question is, are direct converters that cheap or is a blanket with a multiplication of 9 that much cheaper than a blanket multiplication of 30 or so? What is the cost tradeoff?

Moir: I think there are more tradeoffs than that.

Wolkenhauer: Well, your numbers don't show it.

Moir: Let me address one thing. The direct converters don't seem to be very costly. Now, let's take one direct converter at a time. The beam direct converter can hardly be seen on the drawing; the cost is represented by the vacuum chamber and the pumping and the various fairly sophisticated units in the source-power conversion units, electrodes and insulators. The direct converter is kind of a passive element there. We costed that, and that hardly twiddled the total cost of the injector. So we could even employ more sophisticated direct conversion ideas there, if we can think them up. OK, so the direct converter in the beam didn't perturb the injector cost very much. Now, the end loss direct converter required a bigger tank than we might have had, but the tank was sized by the pumping requirement and not by the direct converter. Now our initial design had one direct converter only on one end and on the other end we had a stronger magnetic field to direct the plasma selectively out one end. That cut the pumping area in half and the pumping got into trouble. Those additional elements were not very expensive compared to the vacuum chamber auxiliaries. So that is, I think, my answer on the direct converter. For the economics on the low M, fast fission blanket, the one that makes lots of fuel compared to the high multiplication system, the direct conversion issues do not come in. I think other issues do. I'm not prepared to say which is better just from a cost point of view.

Dean: You have a design for a fusion-fission system, and there is another design for a FERF that you showed on the graph. Is it possible to think of designing one facility which would serve both functions, and what are the difficulties of trying to combine those kinds of functions?

Moir: I presume by that question what you mean is, one physics feasibility experiment that will test the feasibility of both.

Dean: No, I am thinking in terms of one machine which would be useful for both, the things that you described and FERF purposes.

Moir: Well now, the FERF was a near-term machine. Its cost analysis was based on near term ideas and it cost about 200 and some million dollars. We made it as small as we could because the object wasn't to get as large a test area as you can but more or less build a reasonably large test area in the smallest size with the lowest cost. The object of the other one was to build something that could be economical on a cost per kilowatt basis and that required getting into about the 500 megawatt class. That meant a big machine and 3 or 4 times the cost. The cost of FERF assumes the present day superconducting technology industry base and the other, a technology basis that we think may exist 15 or 20 years from now. If we would have costed the hybrid machine on the near-term basis, like more or less today's basis as we did FERF, I imagine the cost would be a factor 3, 4, maybe 10 times higher. But they are different machines for quite different purposes. On the FERF, we have considered direct conversion but it doesn't work very well and is not very cost effective because the energy is so low. So it goes with a lot of differences between the two machines. Maybe Clyde Taylor would like to amplify on this.

Taylor: No.

Frank: Have you compared the use of lithium and helium coolants in a hybrid blanket?

Lee: No, we haven't yet. We have just taken this point design and acknowledged that there are other choices one can make, but we have not done a design study of a lithium cooled hybrid.

Dudziak: A related question: In your sub-modules, the 10 percent leakage you quoted, was that within the helium cooling channels of the submodules?

Lee: No, that's out of the mirrors and injector ports where there is no blanket. The neutron leakage through the modules themselves should be quite small.

Dudziak: OK, so that's the coolant channel running in and out, not the pressure drop shown in the slide due to the sizes.

Lee: Yes, but remember that the fission zone is only 30 centimeters deep, and above it you have approximately the same distance for the tritium breeding. So yes, you are talking about kind of on that order, but these are just sketches.

Baker: What was the power density in the plasma?

Moir: It is 2 watts per cc. Since the power goes like the fourth power of the magnetic field, it would be quite easy to raise that by a factor of 10. We could really sizzle the blanket if we wanted to, and at not much additional cost.

Miley: I gather that you picked 500 megawatts as an approximate economic breakeven. How sensitive are the economics to that choice? What if you double the size?

Lee: Well, I think Ralph almost alluded to that just a minute ago. It does not require much change in the magnetic field. We think the blanket as it now sits, this design, can effectively double in power with very little change in the design and very little change in the cost. We have done some preliminary studies to see if that's true. Would you like to address the plasma engineering side of that question?

Moir: No, I think that's for future work.

Graves: Have you done any calculations on the neutron flux that would be at the large direct converter region and what the radiation damage effects of that flux might be?

Lee: No, we haven't.

Moir: The uncollided flux is 7×10^{11} n/cm^2 - sec.

Lee: No, he's speaking of the direct converter. In the direct converters?

Graves: Yes.

Moir: Compared to the damage you get on the first wall, it is very far down. We looked at a design where the plasma is guided by a magnetic field around a bend. It looked like that would be kind of costly but certainly doable and we could cut the neutron flux way down. But I think that the direct converter elements are going to be composed of very rugged materials. There very likely are going to be graphite collectors and hot tungsten wires that can be moved out continuously. I don't think radiation damage is any question here at all.

Lidsky: Let me ask one more technological question. You've presented more details than anyone else, therefore you are more vulnerable to such detailed questions. This system is like many others that breed plutonium in situ and therefore have time-varying multiplication and time increasing tritium generation. All systems that do this are liable to the critisism that they are either underdesigned at the end or overdesigned in the beginning. Have you thought about leveling that off somehow?

Lee: Yes, there are at least three methods with which blanket output could be kept level or nearly level: Use the plasma power level as a control mechanism; Reprocess the fuel frequently to remove Pu; Have the blanket start its life enriched with Pu. These methods could be used separately or in combination.

The Relevance of Environmental Concerns in Contemplating Development of Fission Fusion Hybrids -- A Personal View

John Holdren*

Introduction

What is attractive about the notion of a fission-fusion hybrid? The primary rationale for pursuing this option is rooted in the potential shortcomings of the pure fission and pure-fusion alternatives. The main such shortcomings have been said to be: (a) the breeding ratio of fission breeder reactors may be too low for the expansion of fission-based generating capacity to keep pace with anticipated electricity demand; (b) fission reactors-especially breeders--are expensive; (c) the environmental characteristics of pure fission systems are troublesome; (d) successful pure fusion reactors may elude us for many years to come because of the technical difficulty of the task; (e) the probable need for a high neutron flux through the vacuum wall in a pure fusion system may pose severe materials problems and associated high costs even after the other problems of pure fusion have been solved.

For a fission-fusion hybrid to be an attractive alternative, then, it must not only be easier to attain and/or cheaper to operate than pure fusion, but it must also be significantly better than pure fission with respect to at least one of the following: breeding ratio, economics, environmental characteristics. (Ultimately, of course, the question must be asked in a "systems" context: Is a mix of generating technologies that includes hybrids more attractive than a mix that does not? Still, the answer can be yes only if the hybrid is superior to other ingredients of the mix in at least one of the respects just mentioned.)

Timing and Economics

In comparing hybrids to pure fusion systems with respect to timing and economics, I do not find it likely that first-wall problems alone could delay pure fusion enough or raise its costs enough to make

* Consultant to Controlled Thermonuclear Research at the Lawrence Livermore Laboratory. Dr. Holdren is an Assistant Professor in the Energy and Resources Program at the Berkeley Campus at the University of California.

hybrids interesting; once $n \approx 10^{14}$ has been achieved in a reactor-like configuration, one can surely find a solution to the first-wall situation more straightforward (and more economical) than adding a fission blanket. Moreover, pure fusion would probably have to be significantly more expensive than hybrids before society would choose the latter; that is, if there actually were a choice, the manifest environmental advantages of the pure fusion system would likely be judged worth paying something for.

The more interesting issue is whether a hybrid can be brought to fruition well <u>before</u> the plasma confinement needed for pure fusion is achieved--that is, whether the advantage gained for the hybrid by relaxing plasma-confinement requirements will not be offset by the added difficulties of cramming fission technology into the awkward geometry dictated by the requirements of the fusion core. That the engineering difficulties of the marriage can be solved before the confinement problem for pure fusion is at least plausible enough, in my view, to justify expanded investigation of hybrids--<u>if</u> the potential advantages over pure fission are there.

These potential advantages should be more than marginal if an expensive program of research and development is to be justified. Are they?

First consider the issue of breeding ratio. Preliminary studies of blankets for fission-fusion hybrids do indicate that breeding ratios significantly higher than for pure fission systems can be attained. How important this point is--if more detailed engineering studies confirm it--depends on one's assumptions about the growth of electricity consumption, about the size of uranium resources, and about the cost and availability of enrichment capacity. Some recent studies suggest that the rate of growth of electricity consumption in the U.S. is likely to drop sharply in the next few decades,[2] and that uranium resouces are not as limiting as the technical community has been led to believe.[3] If either of these assertions proves to be valid, the urgency and economic value of high breeding ratios are diminished--the more so if enrichment via centrifuges or lasers succeeds commercially. These issues are still too speculative to support concrete conclusions, and a thorough study of the importance of breeding ratio under alternative conditions is required. It is enough to note here that the matter is in doubt.

The other aspects of the comparative economics of fission and fission-fusion hybrids are hardly clearer. The costs of the U.S. Liquid Metal Fast Breeder Reactor Demonstration Project are much higher than originally anticipated, and the demands that LMFBR technology makes on materials and quality control imply high construction costs even in the longer term. But is there any reason to believe hybrids can be made to be much cheaper than LMFBR's? I think not. The same inherent complexities of a fission-fusion marriage that make it unlikely to be cheaper than pure fusion also apply in the comparison with pure fission. The one way in which a hybrid seems at all likely to be cheaper to build than a pure fission breeder is if the hybrid design is intrinsically so much safer that expenditures on containment and emergency systems can be significantly reduced. This point raises the environmental issue, to which I now want to turn in detail.

Relevance of Environmental Concerns

The thrust of the foregoing discussion is that a clear and compelling case for developing hybrids exists only if these have the potential for significant environmental advantages over pure fission systems. The other widely offered arguments for hybrids rely on supposed advantages that seem unlikely to offset the disadvantages (e.g., relaxed first-wall conditions compared to pure fusion), or that are of questionable importance (higher breeding ratio than pure fission), or that are at all likely to materialize only if the environmental advantages are there (lower construction costs than pure fission). (I believe, incidentally, that the most compelling case for *pure* fusion is also the environmental one, since there is little *a priori* reason to believe fusion will be cheaper than fission, and there is enough uranium and thorium to last for centuries once fission breeders are in business.) It is possible, of course, to argue for hybrid development on the grounds that diversity is simply a good thing, or that hybrids will provide a useful learning step toward pure fusion. The diversity argument, although appealing on philosophical grounds, is unlikely to generate much enthusiasm (or money) unless the technology offers some significant potential advantage. The learning argument is also a bit thin unless hybrids have advantages in themselves; hybrid technology looks so difficult that the things it could teach us about pure fusion could probably be learned in a more straightforward (and cheaper) way.

If so much of the case for hybrids really rests on whether or not they have environmental advantages over pure fission, then environmental considerations should be given great weight in selecting approaches to hybrid design for further investigation. There is little point in pursuing approaches that sacrifice for engineering reasons the main rationale for developing hybrids at all.

Environmental Criteria for Hybrid Designs

The most important environmental issues surrounding today's pure fission fuel cycles are (a) major accidents at reactors or fuel reprocessing plants, leading to large releases of radioactivity to the environment; (b) diversion of fissile materials for use in explosive or radiological weapons; (c) leakage of long-lived radioactive wastes from their storage facilities.

Possible ways that fission-fusion hybrid designs could reduce the probability of these events are of course associated mainly with the treatment of the fission blanket, as follows:

(1) Preclude nuclear criticality accidents by designing a blanket that is subcritical in its most reactive configuration.

(2) Preclude blanket melting in the event of loss-of-coolant accidents, by designing a blanket that remains below its melting temperature by means of passive cooling, (e.g., natural convection, conduction, and radiation) even in the worst circumstances (total loss of coolant just before refueling).

(3) Minimize chance of other types of events that could breach containment by minimizing chemical energy stored in blanket and associated facilities.

(4) Minimize consequences of any breach of containment by minimizing blanket inventory of radioactivity, especially volatile substances.

(5) Reduce effective half-life of radioactive wastes by facilitating separation, rapid reintroduction into the blanket, and subsequent fast fission of plutonium and other actinides.

(6) Reduce number of points in the fuel cycle where fissile materials could be diverted, either by using on-line reprocessing or (at least) by designing to facilitate consolidation of reprocessing and fuel fabrication activities at the reactor site.

It is not obvious that all of these approaches are feasible and some may conflict directly with each other--e.g., minimizing radioactive inventory and consolidating facilities may not turn out to be consistent. I believe, however, that (1) and (2) represent the minimum that must be achieved simultaneously if hybrids are to be widely regarded as significantly better environmentally than pure fission, and that a strong environmental case will require (5) and/or (6) as well.

Applying the Criteria

Hybrid blankets can be characterized by the combinations of moderators and coolants employed. This is done in Table 1 for most of the preliminary designs described in the literature to date. In the table, numbers in parentheses refer to the notes at the end of this paper.

Owing to the very preliminary character of these existing designs, it is difficult to say anything at all about the extent to which they might fit approaches (4) through (6) of the preceding section. It appears that all of them are consistent with approach (1), guaranteed subcriticality. Approach (2), passive coolability following loss of primary coolant, is in general much more difficult to accomplish. The characteristics of the three basic coolants--with respect to stored energy, blanket behavior following loss of coolant, and some other possible advantages and disadvantages--are summarized in Table 2.

Conclusion

Too little information is available to justify any detailed conclusions on the environmental comparison between pure fission systems and fission-fusion hybrids, or on environmental comparisons between different hybrid designs. It does appear that there is some potential in the hybrid concept for significant environmental advantages over pure fission. The case for the desirability of hybrids depends so strongly on whether these possible environmental advantages can actually be realized, that investigating ways to achieve them should be a cornerstone of any hybrid development program. If, on the contrary, the environmental characteristics are treated as secondary concerns to be dealt with <u>after</u> one kind of hybrid or another has been intensively developed, we are quite likely to find that the kind we developed is so irretrievalby handicapped environmentally as to make it not worth having.

Table 1. Blanket Options for Fission-Fusion Hybrids

Moderators / Coolants	none (fast blanket or fast region)	graphite	LiH, LiD
Li metal	Lidsky (4) Lee (5) Maniscalco[A] (6) Braun (7)		Maniscalco[B] (6)
Li salt	Lontai[B] (8)	Lontai[A] (8)	
Li/U-Th salt	Lontai[A] (8)	Lidsky (4) Lontai[B] (8)	
helium	Leonard (9) Hansborough (10)	Leonard (9)	

Table 2. Some Environmental and Engineering Characteristics of Coolants for Hybrid Blankets

liquid lithium:	fire hazard pumping across magnetic field may be expensive low system pressure is an advantage passive cooling after LOCA difficult, although easy if forced flow is lost but lithium stays high energy amplification readily attained, an advantage
molten salt:	confinement of fission products worse than with solid fuel low system pressure prospects for on-line reprocessing and low total fission-product inventory are good energy amplification may be marginal limited experience with this coolant in pure fission systems meltdown is by definition not a problem, but what happens if salt escapes primary system?
helium:	high pressure required passive cooling after LOCA very difficult in fast region, but probably not in graphite-moderated thermal region high energy amplfication

Notes

1. B. R. Leonard, Jr., A review of fusion-fission (hybrid) concepts, Nuclear Technology, Vol. 20, pp. 161-178, December 1973.

2. D. Chapman, T. Tyrell, and T. Mount, Electricity demand growth and the energy crisis, Science, Vol. 178, pp. 703-713, 17 November 1972; Ford Foundation Energy Policy Project, A Time to Choose, Ballinger Publishing Co., Cambridge, Mass., 1974.

3. John P. Holdren, Uranium Availability and the Breeder Decision, Environmental Quality Laboratory Memorandum No. 8, California Institute of Technology, January 1974; Uranium supply: EPRI estimates supplies and requirements, Nuclear News, p. 67, November 1974.

4. L. M. Lidsky, Fission-fusion symbiosis: general considerations and a specific example, Proc. BNES Conf. Nuclear Fusion Reactors, Culham, England, 1969.

5. J. D. Lee, Neutronics of sub-critical fast fission blankets for D-T fusion reactors, Proc. 7th Intersociety Conf. Energy Conversion Engineering, p. 1294, American Chemical Society, 1972.

6. James Maniscalco and Lowell Wood, Advanced Concepts in Fusion-Fission Hybrid Reactors, UCRL 75835, 1974. "A" refers to their "homogeneous" design, "B" to their "heterogeneous" design.

7. G. W. Braun and L. M. Lidsky, Geometry and performance of a Theta-Pinch power breeder, Proc. 1st. Topical Meeting on the Technology of Controlled Nuclear Fusion, San Diego, 1974, U.S.A.E. C. Conf. 74042.

8. L. N. Lontai. See descriptions of Lontai's designs in Reference 1 above. A has U salt in fast region but not in thermal region, B the reverse.

9. See the description of the PNL Thermal Fission Hybrid in Ref. 1 above.

10. L. Hansborough and R. Werner, A Modular Fission-Fusion Hybrid Blanket, U.S. A.E.C. Conf. 740402, 1974 (see note 7).

QUESTIONS ABOUT SIXTH PRESENTATION

Lidsky: I have a comment that I think follows your main line, but I suspect you possibly have left out a very important additional consideration; an environmental consideration. That is, suppose one comes up with a system that allows one to operate safer and environmentally more satisfactory fission reactors. Then, in fact there is a certain "load" taken off the hybrid device and a larger reservoir of "goodness" one can do. For example, suppose you came up with a system that had definite flaws but enabled one to avoid building an extensive LMFBR economy on the way to achieving Nirvana-pure fusion. The point I would like to make is that one has to optimize the whole system; one doesn't optimize any one given machine or type of machine and there, I think, is where the environmental issue lies.

Holdren: OK, let me respond quickly. I agree with you that one has to optimize the whole system. I think we should credit to the hybrid any environmental advantage that it brings about by permitting an environmentally more benign system overall: For example, if one does in fact get a significant environmental advantage from hybrids because they permit the use of HTGR's or LWR's instead of breeders, then one chalks that up for the hybrid as a significant environmental advantage. I think this is proper. It does remain to be seen whether people become convinced that LWR's and HTGR's are, in fact, intrinsically better environmentally than breeders. Right now there is a lot of uproar about whether the potential for a criticality accident in the breeder is worse than the potential for a loss of coolant accident in LWR's. The HTGR certainly looks good in these respects but perhaps because we know less about it. It remains to be seen. But in principle I agree with you - one has to optimize the whole system, and this is a potential environmental advantage that might accrue to hybrids in this sense.

Wolkenhauer: I agree with you wholeheartedly except for one point that you stated that I disagree with. It relates to the significance of the work that Deonigi presented this morning, that hybrids can be more expensive than LMFBRs by about 20 percent or so and be competitive. You said they had to be cheaper.

Holdren: OK, I accept the criticism. I didn't mean to imply that they have to be cheaper than LMFBR's, but rather that they should have some economic advantage across the board. This is not restricted to capital cost. When one talks about cheaper across the board, one is talking about plutonium production and all elements of the fuel cycle.

Wolkenhauer: When one reviews the economics like we have been doing in the past few months, one finds that safety and environmental concerns really don't show up very strongly in the economics. It's an unfortunate truism as you go through the analysis, but this may change and make your comments more pertinent.

There are just a couple of other minor picky points, areas I'd like to see studied. It's maybe not generally recognized that the fission product mixture that you get from the fissions in some of these hybrid systems is quite different from the fission product mixture that you get from, say, thermal reactors. There are possibilities that there are some gains to be found. Somebody ought to look into that. Weinberg has pointed that one of the possibilities for eliminating or helping the safeguard problem is by recycling the actinides into the plutonium, simply to make it very hot stuff so that it can be handled only remotely. Because actinide recycle appears to work a little bit better, perhaps, in hybrid reactors, this is a possible plus that somebody ought to look at.

Holdren: Yes, I agree and, in fact, that last item is in my list of six ways to get at the potential environmental advantages of hybrids. I agree with you that all these things should be looked at in more detail, but my hope is that we not, in pursuit of a fast engineering road to hybrids, demote these considerations to second place and thus throw out the principal merits of the enterprise.

Wolkenhauer: The loss of coolant thing, for instance, is very sensitive to the economics. If you go in the direction the economics says you should go, you very quickly lose the capability to withstand the loss of coolant.

Holdren: Yes, and I think it may well be that when we get a little better at asking this type of question, we may find that the public is prepared to pay something for extra assurances against this kind of event. As you point out, it's very difficult right now to work into a possible cost-benefit analysis this issue of how much the public is prepared to pay for additional safety, or how much money you should spend to reduce any further what presumably is already a small number. But I think we may well find out that in the case of this kind of event, the public is prepared to pay quite a bit. The same thing may hold for pure fusion. We may find out that pure fusion, once it's been achieved, will be considerably more expensive than pure fission for some time. That is not inconceivable by any means. But it may be that people are willing to pay the price for the environmental advantages of the pure fusion system.

Coffman: I'd like to comment on the perspective of the safety guidelines and how we (fusion) compare, for instance, to fission devices. I think the jury is still out on the potential inadequacies of the fission systems whether they are LWR's or breeders. The 10 CFR 50, Appendix I dose limits at the site boundary will give you very nominal or minimal effects from routine releases. I think that the Rasmussen study has shown that a probability factor of 10^{-7} for a major accident gives you a rather nominal risk also. I'm just not sure that it's going to be that easy for fusion reactors to do much better than that.

Holdren: I certainly agree that with respect to routine emissions; fission is already very good indeed. As long as things are routine, you're not going to be able to do much better than that because it's already so close to zero, and that is why I didn't even list routine emissions from operating fission reactors as among the principal issues there. With respect to safety, diversion, and radioactive waste, I agree that the jury is still out. Indeed with respect to accidents, I don't think that Rasmussen has brought the jury in. Rasmussen has made a significant contribution to that debate, but it's by no means over yet. I don't think we have time to go into the details of the discussion now, but it's still a live issue.

Coffman: There is one thing that has to be factored in, that is, we tend to partition out the breeder from the LWR's, and that is really a false logic in that LWR's breed or generate significant quantities of plutonium. If there is not an LMFBR economy to use that plutonium, the plutonium that is being bred right now in the existing 50 LWR plants is going to be factored right back in, just on the basis of economy.

Holdren: I agree, plutonium is already a problem with light water reactors. I don't dispute that at all. There are some characteristics of breeders that are different from light water reactors and, again, I think some of the discussion has already brought it out. It's not really clear whether the breeder overall is better or worse. It may be that the HTGR is better than both of them; it may not. But all I wanted to do is call attention to the importance of these issues in the fission area and argue that, if hybrids cannot do significantly better, then we are going to be very hard pressed to find the motivation to make the significant financial gamble to develop them. A good gambler doesn't wager a lot of money unless there is a big gain. And if the gains over fission are potentially tiny, I don't think we will see hybrids developed. We in the fusion community might want to see them developed for our own parochial reasons as a stepping stone, for example, but I bet we can't sell it on that basis.

BEAM-DRIVEN TOKAMAK FUSION-FISSION HYBRID REACTORS

D. L. Jassby
Princeton University Plasma Physics Laboratory,
Princeton, New Jersey

ABSTRACT

The conditions for maximizing neutron flux from a beam-driven tokamak plasma are derived. The evolution of fusion power density and Q in passing from purely beam-driven operation to purely thermonuclear operation is discussed. With 200-keV deuteron beams injected into a tritium target plasma, one obtains the same power density for $Q \sim 1$ conditions as in a thermal D-T reactor operating at considerably great B_t and temperature, and requiring an $n\tau$ many times larger. Fissile breeding rates and power productions are given for beam-driven fusion-fission hybrid reactors with plasma parameters $B_t = 50$ kG, $T_e = 6$ keV, $n\tau = 10^{13}$ cm^{-3}sec, $a_p = 1.1$ m, $P_f = 4.3$ W/cm^3, $Q = 1.2$. Relevant plasma physics problem areas are discussed.

Paper presented at U.S.A.E.C. meeting on Fusion-Driven Fission Systems, Germantown, Md., December 3-4, 1974.

1.0 INTRODUCTION

While the ultimate goal of controlled fusion research is the development of an economic power plant based on the fusion process alone, it has been recognized from the early days of the fusion program that the 14.1 MeV neutrons from the D-T reaction may find important applications other than direct power production. Some of these applications are the following:

1. Breeding of fissile material;[1] for example, ^{233}U for reactors of high conversion ratio, or ^{239}Pu for startup of fast-breeder reactors.
2. Fusion-fission hybrid reactors,[1] consisting of a D-T fusion core surrounded by a fission blanket of ^{232}Th, ^{238}U, or ^{239}Pu.
3. Transmutation of radioactive wastes,[2] particularly long-lived transuranic radionuclides.

These applications may be combined in a single reactor. Since overall the nuclear processes in the blanket are exothermic, the fusion-driven reactor may possibly be an economic power producer even when the energy multiplication factor of the fusion core itself is $Q \lessapprox 1$. This requirement on Q is most easily met by means of a tritium-target plasma heated by reacting deuteron beams.[3,4]

Another important economic criterion is that the fusion power density P_f be sufficiently large so that the value of the neutron products in the blanket, together with the net power produced, provide a sufficient return on the total capital investment. A beam-driven tokamak reactor that provides fusion energy principally via beam-plasma reactions (called "two-component torus", or TCT), can attain, at moderate plasma temperature and small $n\tau$, a P_f many times larger than that of a thermal (non-beam-heated) reactor of the same total plasma pressure, at any temperature.[5] A small-$n\tau$ TCT plasma surrounded by a high-gain fission blanket can therefore provide a Q characteristic of a large-$n\tau$ thermonuclear reactor, together with the large power density characteristic of the TCT. Similar advantages of beam-driven plasmas over purely thermonuclear plasmas can be demonstrated for other reactor configurations. But since the required plasma parameters for a "breakeven" TCT are not vastly beyond those already attained in tokamaks, it appears that the TCT is the most likely candidate for development as the multi-purpose fusion neutron source of a fission power economy.

In Sections 2 and 3 of this paper we derive the conditions for maximizing neutron power production in a TCT, and demonstrate that the neutron flux at $Q \lesssim 1$ can be much larger than in a thermonuclear reactor, at any Q. More extensive results can be found in Ref. 5. In Sect 4, some practical reactor parameters are considered, while Sect. 5 discusses the plasma physics problems that must be resolved before large neutron fluxes from a TCT can be realized.

2.0 MAXIMIZING FUSION POWER DENSITY

Energetic deuterons injected into a tritium target plasma lose energy by Coulomb collisions with the bulk-plasma electrons and ions. While thermalizing, the deuterons produce Q_b times their injected energy in fusion reactions, where

$$Q_b = \frac{\int_o^{\tau_s} n_T \quad \sigma(v) v E_f dt}{W_o} \tag{1}$$

Here τ_s is the "slowing-down" time of the fast deuterons, $\sigma(v)$ is the fusion cross-section, $E_f = 17.6$ MeV, and n_T is the triton target density. Figure 1 shows Q_b as a function of T_e and W_o, for plasmas with ion temperature $T_i = 0$. In practical cases of interest $T_i \approx T_e$, because roughly half the beam energy is given up to the bulk ions, while the electron-ion equilibration time tends to be comparable to the energy confinement time. For finite T_i the results of Fig. 1 are significantly modified at $W_o \lesssim 100$ keV, but are only slightly altered at $W_o \gtrsim 150$ keV, which is the range of interest in this work.[6] The plasma density enters the calculation of Q_b only through $\ln \Lambda$ factors.

2.1 Optimal Plasma Parameters

In this paper a zero-dimension spatial model is used. Of course, this model is easily extended to include practical radial profiles; the values of density and temperature used here would represent radially-averaged values for those cases. At any rate the average plasma pressure allowed in a tokamak does not depend on the radial profiles, but only on toroidal field B_t, plasma aspect ratio A, and the rotational transform at the limiter, q. In this section we use the illustrative values $B_t = 60$ kG, $A = 3.5$, $q = 2.5$, corresponding to a maximum allowed pressure $p = 0.655$ J/cm^3 (for $\beta_p = A$).

Consider now a plasma maintained at temperature $T_e = T_i$ by neutral-beam-injected energetic ions. The bulk plasma includes a single impurity species of charge Z_I and density n_I. The impurity ions can be expected to exist in good thermal equilibrium with the tritons and electrons. Given T_e and W_o, in order to achieve maximum system Q, there are two conditions that concern the confinement times of the fast deuterons and the bulk plasma. First, the lifetime τ_h of the fast ions must be greater than τ_s, in order that the full power gain Q_b be realized. Second, in order that Q_b define the fusion gain of the entire beam-plasma system, all the energy loss of the bulk plasma must be made up by the injected beams; that is

$$P_b = I_b W_o = \frac{3/2\ (n_T + n_I + n_e)\ T_e}{\tau_E} \tag{2}$$

where I_b is the injection current, and τ_E is the energy confinement time of the bulk plasma. τ_E takes into account all energy loss channels, including heat conduction, particle transport, charge-exchange, and radiation. (We show a posteriori that ohmic heating is insignificant when P_f is maximized. Alpha-particle heating is considered in Ref. 5.)

Since $I_b = n_b/\tau_s$, where n_b is the beam-ion density, Eq. (2) leads to the definition

$$\bar{\Gamma} = \frac{\text{suprathermal-ion energy density}}{\text{bulk-plasma energy density}}$$

$$= \frac{n_b\ \overline{W}_b}{3/2\ (n_T + n_I + n_e)\ T_e} = \frac{\tau_s}{\tau_E}\ \frac{\overline{W}_b}{W_o} \tag{3}$$

Here $\overline{W}_b$ is the average fast-ion energy, which is evaluated in Ref. 5. In principle, τ_E can be made arbitrarily small, with no reduction in T_e, simply by increasing I_b. In the absence of a pressure limitation on the plasma, this increase in I_b would always result in an increase in P_f. However, there are two factors that lead to an optimal value τ_E, and thus a maximum in P_f:

(i) The plasma pressure is limited to the maximum allowed by MHD equilibrium, that is,

$$p = (n_T + n_I + n_e)(1 + \overline{\Gamma}) \leq \frac{1}{8\pi} \frac{B_t^2}{q^2 A} , \tag{4}$$

in the case of a tokamak. For MHD-stable operation, q cannot drop below about 2.5. Hence, an increase in $\overline{\Gamma}$ necessarily leads to a decrease in n_e and target density n_T.

(ii) Charge neutrality requires that

$$n_e = n_T + n_b + Z_I n_I \tag{5}$$

Hence large beam densities appreciably reduce n_T/n_e.

Another restriction on the maximum injection current is that large values of $\overline{\Gamma}$ may lead to the onset of instabilities; however, stability analysis[7] indicates that systems with $\overline{\Gamma} \lesssim 1$ should be stable, for isotropic or tangential injection.

The remainder of this section is devoted to finding the optimal plasma parameters for maximum P_f, in accordance with points (i) and (ii) above. The pressure restriction becomes

$$p = (n_T + n_I + n_e)T_e + \frac{2}{3} n_b \overline{W}_b = \text{constant} \tag{6}$$

where we assume an isotropic steady-state suprathermal ion velocity distribution, such as resulting from injection of beams both tangential and perpendicular to the toroidal current. Using Eqs. (2), (5), and (6), we find that

$$P_f = Q_b P_b \approx \frac{Q_{bo}}{\tau_{so}} \frac{W_o}{n_{eo}} \frac{n_T}{n_e} \left(1 - \frac{n_T}{n_e} - \frac{Z_I n_I}{n_e}\right) n_e^2 \tag{7}$$

where

$$n_e = \frac{p}{(2/3)[1 - (n_T/n_e) - (Z_I n_I/n_e)]\overline{W}_b + [1 + (n_T/n_e) + (n_I/n_e)]T_e} \tag{8}$$

In deriving Eq. (7), we have used $Q_b/\tau_s \approx (Q_{bo}/\tau_{so})(n_T/n_e)$, where Q_{bo} and τ_{so} are calculated for $n_T/n_e = 1$ (as in Fig. 1). τ_{so} is to be calculated at n_{eo}, but note that $\tau_{so} n_{eo}$ is independent of density (except for the weakly varying $\ln \Lambda$ factor). Once W_o, T_e, and n_I/n_e are chosen, the optimal value of n_T/n_e is found by maximizing P_f. Then n_b/n_e, n_e, $\overline{\Gamma}$, and τ_E are found from Eqs. (5), (8), (3), and (2), respectively. Results are shown in Figs. 2 and 3 for $n_I = 0$, $W_o = 200$ keV, and $p = 0.655$ J/cm^3.

Perhaps the most interesting result is that P_f is nearly inversely proportional to T_e. Under the restriction of constant pressure, lower temperatures allow one to increase the plasma and beam densities, and therefore the fusion reaction rate. Furthermore, the beam-plasma reaction rate is relatively insensitive to plasma temperature (in contrast to the thermal reactor case).

Another important result is that for maximum P_f, τ_E must be a factor 1.3-2 times smaller than $\tau_s \lesssim \tau_h$. But P_f is not very sensitive to τ_E/τ_s, as long as $\tau_E < \tau_s$.[5] In view of present experiments, it seems likely that for a device of practical size, i.e., $a \gtrsim 100$ cm, τ_h will be at least as large as the values of τ_s in Fig. 3., viz. 40 - 200 msec at low T_e. Since τ_E may be comparable to τ_h, at least at low T_e, measures will probably have to be taken to decrease τ_E by a significant factor. Possible means of accomplishing this are reduction in q, addition of high-Z impurities, or vigorous injection of tritium pellets.[5]

2.2 Preferred Operating Temperature

The cost-effectiveness of the reactor depends on both P_f and Q_b. If sheer neutron production is the foremost goal, and in particular if long-pulse, high duty-factor operation is precluded by impurity buildup, for example, it might appear that one should operate at small T_e, where $P_{f_{max}}$ is largest. But there are several practical reasons for preferring the range $T_e = 5 - 10$ keV:

(i) In the case of a low-gain blanket, it may be essential to operate at $Q_b > 1$.

(ii) $P_b = P_f/Q_b$ increases extremely rapidly at small T_e, so that a substantial fraction of the vacuum wall area might have to be taken up by neutral-beam apertures.

(iii) The large values of n_e required to maximize P_f at low T_e make beam penetration difficult unless the plasma radius is very small, or unless W_o is considerably larger than 200 keV. But Q_b decreases at larger W_o (Fig. 1).

As a compromise between these disadvantages of low-T_e operation and the need for large P_f, $T_e = 6$ keV seems to be a desirable operating point. For circular cross-section plasmas of arbitrary q, A, β_p and axial magnetic field B_t (in kG), the general expression for P_f is

$$P_f = 3.7 \times 10^{-5} \, P_{fo} \left(\frac{B_t}{qA} \right)^4 \beta_p^{\,2} \tag{9}$$

where P_{fo} is the value given in Fig. 2. In order that Eq. (9) be valid, $\overline{\Gamma}$ and $n_e \tau_E$ must have the values shown in Figs. 2 and 3, respectively. Then Q_b will also have the same value as in Fig. 2. The neutron wall loading is given by $0.40 \, P_f \, a_p^{\,2}/a_w$, where a_p is the plasma radius and a_w is the wall radius.

2.3 Effect of Impurities on P_f and Q_b

The impurity concentration is often expressed in terms of an "effective Z", where

$$Z_{eff} = \frac{n_T + n_b + n_I \, Z_I^{\,2}}{n_e} \tag{10}$$

Hence

$$\frac{n_I}{n_e} = \frac{Z_{eff} - 1}{Z_I (Z_I - 1)} \tag{11}$$

Given Z_I and Z_{eff}, the corresponding n_I/n_e is inserted in Eq. (7), and P_f is maximized by varying n_T. Figure 4 shows the maximum values of P_f and the corresponding values of Q_b for $T_e = T_i = 6.0$ keV and a range of impurity concentration.

Evidently P_{fmax} is reduced by a smaller factor than the reduction in Q_b, for a given impurity content. Stringent impurity control may not be necessary in a TCT whose main purpose is neutron production at low Q. Even for $Z_{eff} = 10$ caused by ions sputtered from the vacuum wall (e.g., steel or niobium), the reduction in P_f is at most 30%. However, the penetration length of 200-keV beams is seriously reduced when Z_{eff} is large.[8]

2.4 Principles of Maximizing Fusion Power Density

The optimal operating conditions of a TCT for obtaining maximum P_f are summarized in the following: Points (4), (5) and (6) are considered in detail in Ref. 5.

(1) The plasma pressure p should be as large as possible for the available B_t, and q may be reduced until the resulting MHD activity lowers $n_e\tau_E$ below the optimal value.

(2) For a given p and T_e, $\overline{\Gamma}$ has an optimal value (Fig. 2) which thereby fixes optimal values of n_e and τ_E (Fig. 3). The optimal τ_E is 0.5 to 0.8 times the fast-ion thermalization time.

(3) The optimal T_e increases with the minimum required Q_b (Fig. 2), but if low Q_b and high n_e can be tolerated, smaller T_e gives the largest P_f.

(4) The pressure of fusion alphas limits the allowed beam-ion density, so that alpha confinement should be reduced as far as possible without losing beam ions in the process.

(5) Energy clamping of injected ions is desirable only if the cost of the electrical components required for clamping is less than that of the injectors eliminated.

(6) Dilution of the tritium bulk plasma by neutral recycling of deuterons should be avoided, either by capture of diffusing ions in a divertor or by vigorous injection of tritium pellets.

2.5 Ohmic Heating

The foregoing analysis has assumed that the power dissipation of the plasma current is negligible. For the machine parameters chosen

(B_t = 60 kG, q = 2.5) and for a = 1.0 m, J = 109 A/cm^2 for a flat current profile and J(0) = 218 A/cm^2 for a parabolic profile. Figure 5 shows the ohmic power dissipation for J = 200 A/cm^2 and Z_{eff} = 5. Evidently this power input is negligible for conditions of maximum P_f.

The bremsstrahlung radiation, also shown in Fig. 5, is a negligible loss channel even for Z_{eff} = 5. Thus only excitation and recombination processes can provide important radiation losses at moderate T_e. (Such losses have been implicitly included in τ_E.)

3.0 COMPARISON OF THE TCT WITH A THERMONUCLEAR REACTOR

Figure 6 compares the maximum P_f in a TCT with P_f for a 50:50 D-T reactor in which all fusion energy is produced by bulk-plasma reactions, and the plasma is not heated by injected beams (i.e., a one-energy-component plasma). The plasma pressure in the thermal reactor is the same as the total pressure in the TCT case. While τ_E for the TCT must be the value shown in Fig. 3, τ_E in the thermal plasma is somewhat arbitrary, unless we specify Q. (If the thermonuclear plasma is heated by injected (nonreacting) beams, its τ_E must be sufficiently large so that the beam pressure is negligible.) For the two-component case, Q_b > 0.8 for the temperature range shown (cf. Fig. 2). For the preferred TCT temperature of T_e = 6 keV, P_f is a factor of 4 larger than the maximum P_f attainable with the one-component plasma at <u>any</u> temperature. In the optimal temperature range for the latter (12 - 15 keV), the required $n_e\tau_E$ for Q = 1.24 (the TCT value at 6 keV) is a factor of 5 larger than in the TCT case. In the event that large plasma temperatures cannot be attained, the superiority of the "pure" TCT as a neutron producer is even more striking.

Let us now consider the evolution of a purely beam-driven reactor (100% tritium plasma) to a purely thermonuclear reactor. The total fusion gain Q_f of a TCT can be raised beyond the values shown in Fig. 1 by increasing $n_e\tau_E$ as well as the proportion of D in the bulk plasma, so that a larger fraction of the fusion energy comes from thermonuclear reactions.[4] The increase in $n_e\tau_E$ allows a reduction in P_b, so that the relatively small thermal fusion power can eventually surpass P_b.

Neglecting alpha particles, we have

$$Q_f = \frac{n_D\, n_T\, \overline{\sigma v}\, E_f + Q_b P_b}{P_b} \tag{12}$$

where P_b is given by Eq. (2), with n_T replaced by $n_i = n_T + n_D$, where n_D is the bulk-deuteron density; Q_b is still given by Eq. (1). For each value of $n_e\tau_E$, there is an optimal value of n_T/n_i for maximizing Q_f. Figure 7 shows maximum Q_f and the corresponding n_T/n_i for $T_e = T_i = 8$ keV. The contribution to P_f from the beam is given by Eq. (7), with n_T again replaced by n_i (and $n_I = 0$). Evidently maximum P_f is attained for an $n_e\tau_E$ that allows only small Q_f. As $n_e\tau_E$ increases and n_D/n_T approaches 1, P_f goes over to the one-component value shown in Fig. 6.

Figure 8 demonstrates the Q_f vs P_f trade-off for a number of plasma temperatures, over all possible operating regimes. Each point on a constant-T_e curve corresponds to a particular value of $n_e\tau_E$, with n_T/n_i optimized to give maximum Q_f. The maximum values of P_f are attained at $n_e\tau_E$ corresponding to the right-hand extremities of the curves. The important features of these curves (and of Fig. 7) are the following:

(i) The largest values of P_f are attained at low values of Q_f, reflecting the greater fusion reactivity of "pure" TCT operation.

(ii) The initial rise of P_f with $n_e\tau_E$ is due to the increase in target density as $\overline{\Gamma}$ is reduced.

(iii) At low T_e, the thermonuclear reaction rate is so small that substantial $n_e\tau_E$ is required before n_T/n_i can be reduced. The reduction in P_f as $n_e\tau_E$ is increased is due to reduction in beam density.

Figures 7 and 8 clearly demonstrate that the plasma conditions for maximum P_f are markedly different from those for maximum Q. For neutron applications such as those listed in Sect. 1, the purely beam-driven regimes on the right-hand side are most attractive. For large fusion power multiplication, on the other hand, reactor operation must be in the nearly pure thermonuclear regimes on the left-hand side.

Although wall loading limitations may restrict the usable power density in TCT operation, one can operate at lower B_t and still obtain the same power density as in a thermal reactor at large B_t. For both two-component and thermal reactors, $P_f \propto B_t^4$. Figure 9 shows the maximum attainable P_f as a function of B_t. For a plasma radius $a_p = 1.0$ m, and a wall radius $a_w = 1.3$ m, the neutron wall loading corresponding to $P_f = 1$ W/cm^3 is 0.31 MW/m^2. For the thermal case, maximum P_f is attained at 15 keV, while P_f for the TCT case can be increased beyond the values shows if $T_e < 6$ keV, albeit at lower Q (cf. Fig. 2). Table 1 compares parameters for the two reactor types, each producing $P_f = 3.5$ W/cm^3.

TABLE 1

Operating Parameters for $P_f = 3.5$ W/cm^3 at $\beta_p = A = 3.5$, $q = 2.5$

	Thermal Plasma	Two-Component Plasma
$T_e = T_i$ (keV)	15	6.0
B_t (kG)	60	43
n_e (cm^{-3})	1.5×10^{14}	1.0×10^{14}
$n_e\tau_E$ for $Q = 1.2$ (sec cm^{-3})	4.6×10^{13}	1.0×10^{13}
$\overline{\Gamma}$ (200 keV)	0	0.84

Operation at lower B_t is desirable from the point of view of superconductor technology. On the other hand, if impurity and refueling problems permit only short-pulse operation, then operation at the largest possible B_t is desirable to obtain the greatest possible instantaneous P_f; the time-averaged wall loading would still be tolerable.

4.0 PRACTICAL REACTOR PARAMETERS

4.1 Machine Parameters

The tokamak parameters must be chosen for maximum fusion power output consistent with tolerable neutron wall loading. Other restrictions include the maximum possible B_t, determined by superconductor technology, and the minimum value of A, determined by the thickness of the blanket and shield. For the latter, 1.3 m would seem to be adequate, particularly for fast-fission blankets. A suitable set of machine parameters is given in Table 2. The large magnetic field required at the coil should be attainable with Nb_3Sn technology. The poloidal beta is chosen to be somewhat less than the MHD limiting value. Note that $\bar{n}_e$ and $\bar{T}_e$ are volume-averaged values. The 1.1-m plasma radius insures that the required $n_e\tau_E$ can be attained, is convenient for penetration by 200-keV neutral beams (unless Z_{eff} is very large), and results in a satisfactory power output for an economic hybrid reactor. The 30-cm space between the plasma edge and the wall can be taken up by a relatively cold plasma, or by a divertor. To admit the required beam injection power, less than 1% of the wall area need be taken up by beam apertures, even for beam current densities as small as 0.1 A/cm^2. With a 75% duty factor, the neutron wall loading should allow satisfactory operation for up to 2 years before wall replacement.[10]

4.2 Fissile Breeding and Power Production

In order to calculate illustrative rates of fissile breeding and power production in the TCT reactor, we have used the neutronics results of Lee[11] for a subcritical fast-fission blanket. Although the blanket geometry of Ref. 11 was a spherical shell of inner and outer radii 2 and 3 m respectively, (except for "infinite" assemblies in the "pure" blanket cases), we assume that these results apply to a toroidal shell of somewhat smaller inner radius. The size of the fissionable inventory is approximately 7×10^5 kg. We use the reactor parameters of Table 2, with a duty factor of 75%.

Table 3 shows the breeding rates and power productions for a number of blanket compositions investigated by Lee.[11] The thermal power production includes the power generated by neutron-induced reactions in the blanket, together with the flow of plasma (360 MW) and fusion-alpha power (90 MW) to the wall. The calculation of the electrical power production assumes a 40% conversion efficiency,

TABLE 2

Parameters for a TCT Hybrid Reactor

Major radius	4.3 m
Minor radius	1.1 m
Plasma aspect ratio	3.9
Wall radius	1.4 m
Blanket + shield	1.3 m
B_t on axis	50 kG
B_t at coil	134 kG
I_p	2.8 MA
$\overline{T}_e = \overline{T}_i$	6.0 keV
$\overline{n}_e$	1.1×10^{14} cm^{-3}
$\overline{n}_e \tau_E$	1.0×10^{13} cm^{-3}sec
β_p	3.5
Injection energy	200 keV (D)
Injection power	360 MW
P_f	4.26 W/cm^3
Q_b	1.24
Neutron output	355 MW (1.6×10^{20} n/sec)
Neutron wall loading	1.47 MW/m^2 peak (6.5×10^{13} n/cm^2/sec)

that the injected neutral beams (360 MW) are generated with 80% efficiency, and that 75% of the reactor electrical input is consumed by the beam injectors. Before the economic implications of the results of Table 3 can be appraised, serious estimates of the future value of fissile material, as well as the reactor cost, must be available. Nevertheless, we can make some qualitative assessments, assuming that the cost of the TCT hybrid reactor is of the order of $800M (1974), excluding the cost of fissile material (if any) in the blanket inventory.

The lack of tritium breeding in the blankets containing only ^{232}Th, ^{238}U, or natural U would seem to make them impractical. But if an inexpensive source of tritium were available, the ^{238}U or natural U cases would probably be viable, because of their large production of both fissile material and power. Consider now the tritium-breeding blankets. The depleted-uranium example would be viable only for very large electricity values. On the other hand, the blankets containing Pu, and probably the Th-^{233}U blanket as well, would seem to be practical even for modest electricity values. In general, breeding of ^{233}U might never be economic unless the reactor also sells considerable electrical power. Note that with a molten salt blanket, the proportion of fission products (FP) could be kept well below 8%, so that the fissile production for a Th-^{233}U blanket would be substantially larger than that indicated in Table 3. The fissile production rates of both the U-Pu and U-Th systems could be significantly increased by reducing the tritium breeding ratio to 1.05.

FISSILE BREEDING RATIOS AND POWER PRODUCTION FOR TCT HYBRID REACTORS

Fast-fission blanket neutronics taken from J. D. Lee, Proceedings Seventh Intersociety Energy Conversion Engineering Conference (American Chemical Society, 1972) p. 1294.

Yield per Fusion Neutron (14.1 MeV)

Blanket Composition:	Th-232	U-238	Natural U	Li (4% Li-6) Depleted U 0.4% U-235	Li (4% Li-6) U-238 Pu-239 (4%)	Li (4% Li-6) U-238 Pu-239 (4%) 8% FP	Li (4% Li-6) Th-232 U-233 (8%) 8% FP
Tritium Breeding	0	0	0	0.99	1.38	1.18	1.29
Fissile Breeding (net)	2.7	4.4	5.0	1.68	2.67	1.66	0.69
Breeding Ratio (Pu breeding/Pu fusion)	—	—	—	—	3.1	3.0	1.3
Energy (MeV)	64	233	309	103	431	306	478
Thermal Power Production* (MWt)	1550	4740	6170	2280	8480	6110	9360
Electrical Power Production** (MWe net)	170	1450	2020	460	2940	1990	3290
Fissile Production (kg/day net)	11.2	18.2	20.8	6.9	11.1	6.9	2.9

TCT Parameters:

$B_t = 50$ kG, $A = 3.9$, $a = 1.1$ m, $I_p = 2.8$ MA, $n_e = 1.1 \times 10^{14}\,\text{cm}^{-3}$, $T_e = T_i = 6.0$ keV, $n_e \tau_E = 1.0 \times 10^{13}\,\text{cm}^{-3}\,\text{sec}$,
$W_o = 200$ keV, $P_f = 4.3\ \text{W/cm}^3$. Duty factor = 75%.

*Average thermal power production = (450 MW + blanket energy generated by neutron flux) $\times$ 0.75 .

**Average electrical power production = 0.40 $\times$ (average thermal power) − 450 MW .

5.0 PLASMA PHYSICS PROBLEM AREAS

5.1 Confinement.

Achieving adequate confinement of the plasma energy and suprathermal ions appears not to be a serious obstacle to the development of a neutron-rich, low-Q TCT reactor. The required energy confinement time of 50-150 msec seems easily realizable in a 1-m radius plasma, even under the most pessimistic extrapolation from present tokamak performance (e.g., $\tau_E \sim 15$ msec at $a_p \sim 20$ cm). The high collisionality of the plasma under the optimal conditions for maximizing P_f (cf. Sect. 3) would make collective trapped-particle effects unimportant. For example, for the parameters of Table 2 trapped-ion scaling[9] for $Z = 1$ predicts $n_e\tau_E \simeq 1.5 \times 10^{14}$ cm^{-3} sec, or 15 times the required value.

Confinement of neutral-beam-injected ions for their thermalizing time of 10-20 msec has been observed in beam heating experiments in existing relatively small tokamaks.[12] The diffusion of energetic ions is apparently less affected by plasma microinstabilities than is that of the bulk plasma.[7] Consequently, the required energetic-ion confinement times of 75-150 msec should be achievable in plasmas of 1-m radius. In the case of a plasma with $Z_{eff} >> 1$, however, energetic-ion confinement may be less favorable, both because the neutral-beam penetration is not as effective, and because of rapid pitch-angle scattering that enhances banana diffusion.

5.2 Slowing-Down of Energetic Ions.

The neutron fluxes calculated herein assume that the beam slowing down is "classical", that is, no effects such as velocity - space instabilities will intervene to decrease the slowing-down time. Optimism in this regard is justified by the classical slowing down that has been observed in all experiments to date,[12] which have operated with $\overline{\Gamma}$ as large as 0.3, as well as by theoretical analyses that indicate the absence of microinstabilities even at $\overline{\Gamma} \sim 1$ for isotropic or tangential injection (provided that the beam velocity is less than the Alfvén speed).[7]

5.3 Plasma Beta.

Reactor operation can be economic only when beta is near the MHD limiting value of $\beta_p = R/a$. To date, tokamak plasmas have operated in quasi-steady-state only at $\beta_p \lesssim 1 \ll R/a$, but this result reflects the basic limitation of ohmic heating in raising T_e. With the application of high-power neutral-beam injection, it is expected that β_p can be raised close to the theoretical limit. Actually, tokamak experiments have been performed[13] in which β_p was increased well above unity by a sharp decrease in the plasma current. Although T_e dropped during a subsequent energy confinement time ($\sim$10 msec), the large beta-values apparently had no adverse effect on the plasma equilibrium.

Larger values of toroidal beta could be obtained with the use of plasmas of noncircular cross-section,[9] provided that the stability of such plasmas can be verified. Rather than going to higher density, the chief advantage of a noncircular cross-section would be viable operation at lower B_t using NbTi coil technology.

5.4 Impurities.

As shown in Fig. 4, an impurity content of the magnitude that occurs in present "good" tokamak discharges[9] will not seriously reduce the attainable power density, although penetration of the neutral beam into the plasma interior would be more difficult.[8] But in large tokamaks of much higher temperature, it is expected that the impurity content may vastly increase because of the sputtering of the vacuum wall by high-energy ions and neutrals. If the impurity buildup is gradual, one could utilize a discharge pulse length of perhaps 10-30 sec, followed by a purge of the vacuum system. A large duty factor should still be feasible.

The influx of neutrals into the discharge leads to charge-exchange loss of beam ions, so that the neutral density must be kept below $\sim 3 \times 10^7$ cm^{-3}. Neutrals produced by desorption of gas from the wall would reach dangerous values in the central region of the plasma only after considerable delay but the chief means of control would again have to be termination of the discharge after 10-30 sec. If the influx of sputtered impurities and neutrals appears to be overwhelming, even for short pulses, then special technological controls such as a divertor would have to be implemented.

5.5 Buildup of Deuterons in the Bulk Plasma; Refueling

For discharge pulses longer than a few energy confinement times, the plasma must be refueled, either by neutral particle recycling or by pellet injection. Even if $\tau_h \approx \tau_s$, so that the deuterons exit from the plasma as soon as they slow down, it is inevitable that if recycling is permitted, D will build up in the plasma - even past the 50% level. Thus, if the bulk plasma is to remain essentially tritium, neutral recycling must be avoided, presumably by means of a divertor that captures all diffusing plasma ions. At the same time, the bulk plasma must be replenished by injection of tritium pellets.

If neutral recycling is unavoidable then one must resort to using both T and D beams, maintaining a plasma composition of 50:50 D-T, and taking advantage of beam-plasma, bulk-plasma, and beam-beam fusion reactions. Figure 11 shows the contribution to fusion power density of the three types of reactions. As before, $n_e\tau_E$ has been chosen to maximize the contribution to P_f from beam-plasma reactions. Evidently contributions from both beam-beam and thermal reactions are negligible for $T_e \lesssim 8$ keV, and even at higher T_e the beam-plasma reactions are dominant. At $T_e = 2$ keV, the total P_f is 49% of the "pure TCT" value, but the ratio increases steadily to 65% at 10 keV and 81% at 16 keV.

5.6 Technological Problems

Technological problems that are not directly concerned with the fusion plasma may well be the most serious obstacles to a TCT reactor, but detailed discussion of these problems is outside the scope of this paper. Many of the required technological developments, such as efficient neutral-beam injectors, durable first walls, and high-field superconducting coils are common to both low-Q TCT reactors and pure fusion ignition reactors. However, a TCT hybrid reactor operating with a large blanket multiplication could conceivably utilize water-cooled copper coils, for the case of short-pulse, large-P_f operation. On the other hand, the maximum thermal stress tolerated by the blanket would probably demand either large duty factors, or fairly long power pulses.

The substantial circuit and power-flow problems involved in discharge current startup may be significantly eased for the 2.8 MA TCT reactor, compared to an ignition device which would require at least 10 MA. On the other hand, the task of fueling and cooling a high-power fission blanket may prove to be much more difficult than operating the relatively simple tritium-breeding blanket of a pure fusion reactor. Even if the myriad technological problems can be overcome, the economic viability of either reactor type cannot yet be assured.

ACKNOWLEDGMENT

The author is grateful to Dr. H. P. Furth for many helpful discussions. This work was supported by Energy Research and Development Administration (Formerly Atomic Energy Commission) Contract AT(11-1)-3073.

REFERENCES

[1]B. R. Leonard, Jr., Nuclear Technology 20, 161 (1973).

[2]W. C. Wolkenhauer, B. R. Leonard, Jr., and B. F. Gore, Battelle Pacific Northwest Laboratories Report BWWL-1772 (1973).

[3]J. M. Dawson, H. P. Furth, and F. H. Tenney, Phys. Rev. Lett. 26, 1156 (1971).

[4]H. P. Furth and D. L. Jassby, Phys. Rev. Lett. 32, 1176 (1974).

[5]D. L. Jassby, Princeton Plasma Physics Lab. Report MATT-1072 (October, 1974).

[6]H. Towner, *et al.*, Bull. Am. Phys. Soc. 19, 877 (1974).

[7]H. L. Berk, *et al.*, Fifth Conf. on Plasma Physics and Controlled Nuclear Fusion Research (IAEA, Tokyo, 1974), paper G2-3.

[8]J. P. Girard, D. Marty, and P. Moriette, *ibid.*, paper A17-2.

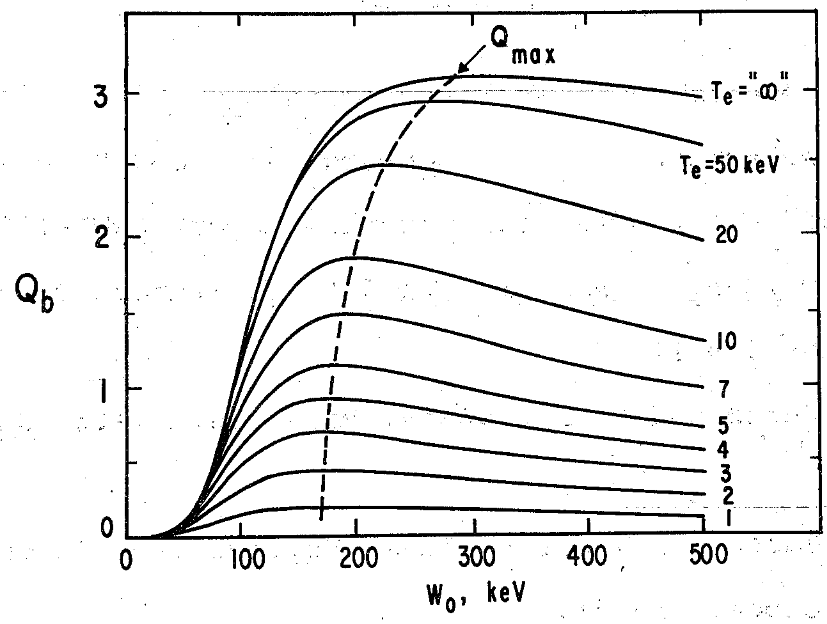
[9]S. O. Dean, *et al.*, Status and Objectives of Tokamak Systems for Fusion Research, USAEC Report WASH-1295 (July, 1974).

[10]R. W. Conn and G. L. Kulcinski, Proc. First Topical Meeting on the Technology of Controlled Nuclear Fusion, (USAEC, San Diego, 1974) Vol. 1, p. 56.

[11]J. D. Lee, Proc. 7th Intersociety Energy Conversion Engineering Conf. (American Chemical Society, 1972), p. 1294.

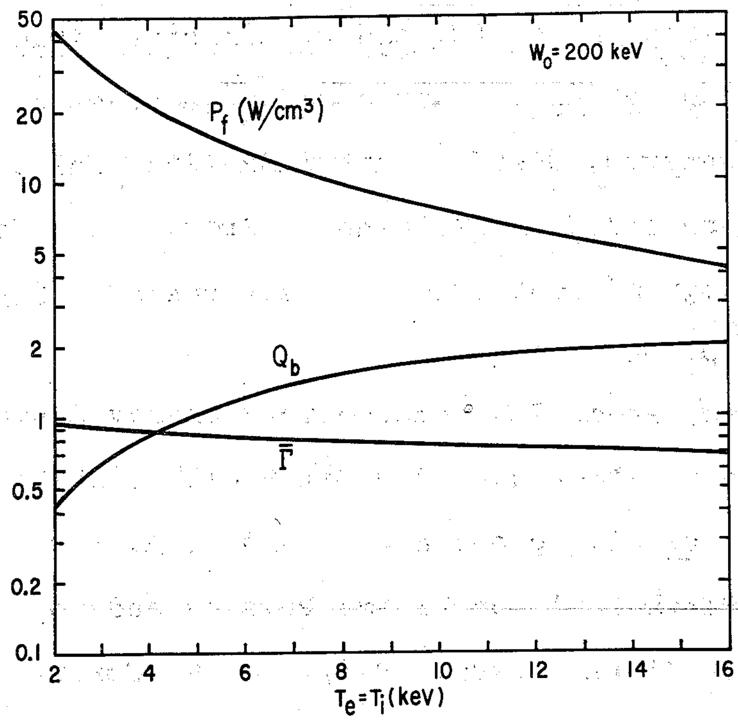
[12]K. Bol, *et al.*, paper A 4-1; and L. A. Berry, *et al.*, paper A 5-2, Fifth Conf. on Plasma Physics and Controlled Nuclear Fusion Research (IAEA, Tokyo, 1974). J. G. Cordey, *et al.*, Nuc. Fus. 14, 441 (1974).

[13]S. V. Mirnov, JETP Lett. 12, 64 (1970); C. Bobeldijk, *et al.*, Bull. Am. Phys. Soc. 16, 1232 (1971).



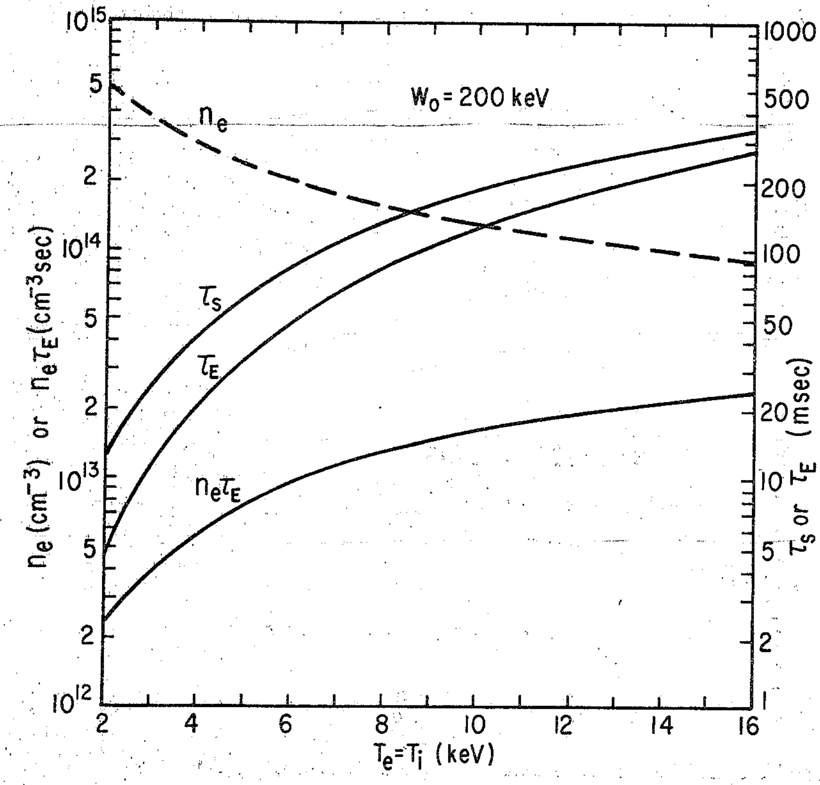
743647

Fig. 1. Fusion power gain Q_b as a function of deuteron injection energy W_o for various values of T_e in a cold-triton target plasma, assuming $n_T = n_e$ (i.e., beam density is neglected). The ln Λ factors are calculated at $n_e = 3 \times 10^{13} cm^{-3}$. (From Ref. 3.)



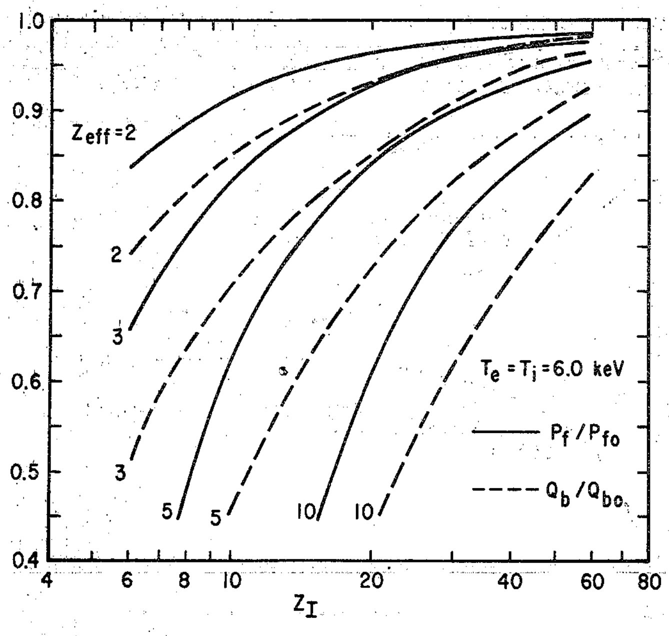
743649

Fig. 2. Maximum attainable fusion power density for 200-keV deuterons injected into a triton-target plasma at T_e, with total plasma pressure = 0.655 J/cm^3, corresponding to B_t = 60 kG, β_p = A = 3.5 ,q = 2.5. $\overline{\Gamma}$ = beam pressure/bulk-plasma pressure and Q_b = effective fusion power multiplication. Alpha-particle effects are neglected.



743650

Fig. 3. Values of n_e, τ_s and τ_E for maximum P_f. Same conditions as Fig. 2. Radiation loss is included in τ_E.



743839

Fig. 4. Maximum attainable fusion power density P_f and the corresponding power multiplication Q_b for 200-keV deuterons injected into a 6-keV triton-target plasma, containing a single impurity species of charge Z_I with concentration Z_{eff}. $p = 0.655$ J/cm^3. P_{fo} and Q_{bo} are values for $Z_{eff} = 1$.

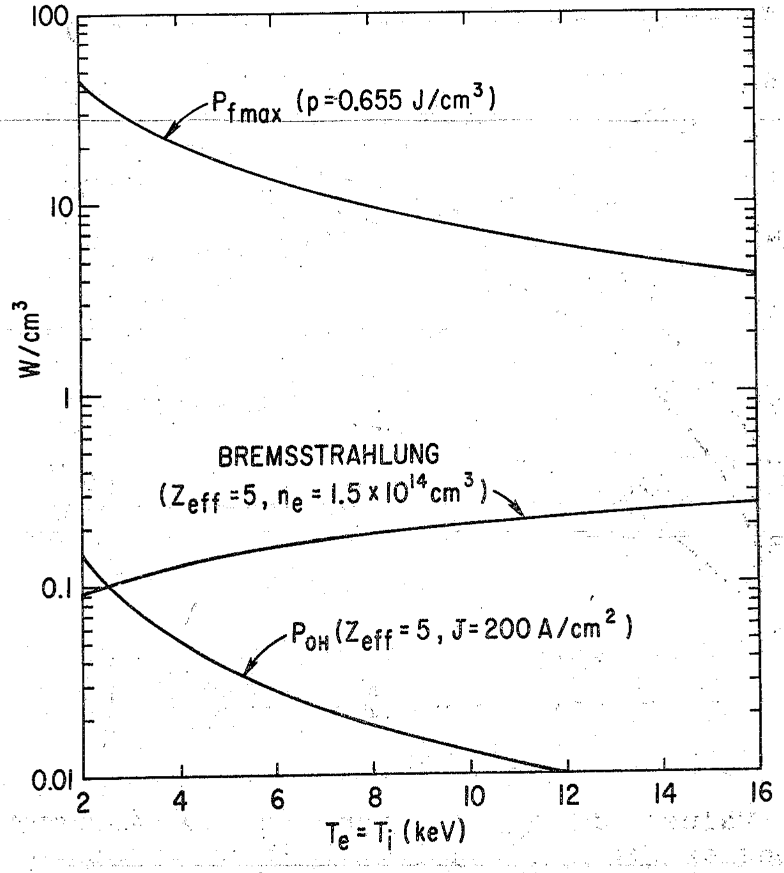


Fig. 5. Ohmic power dissipation (P_{OH}) and bremsstrahlung power density for a $Z_{eff} = 5$ plasma. P_{fmax} is the maximum attainable fusion power density for 200 keV deuterons injected into a triton-target plasma.

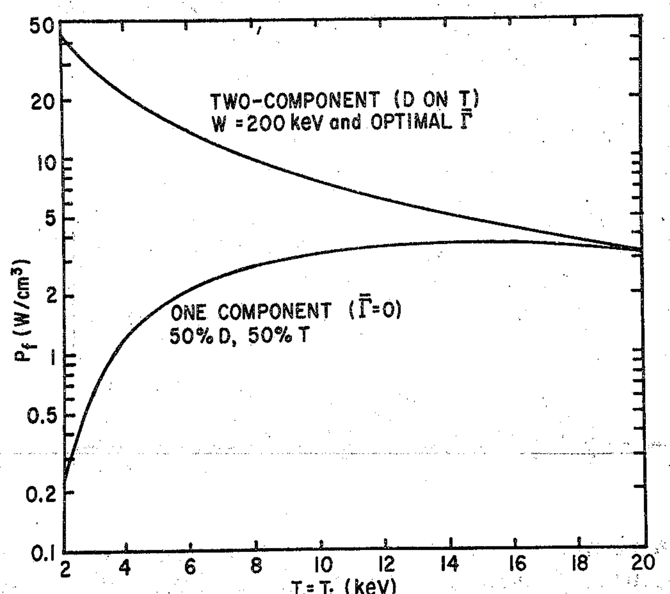
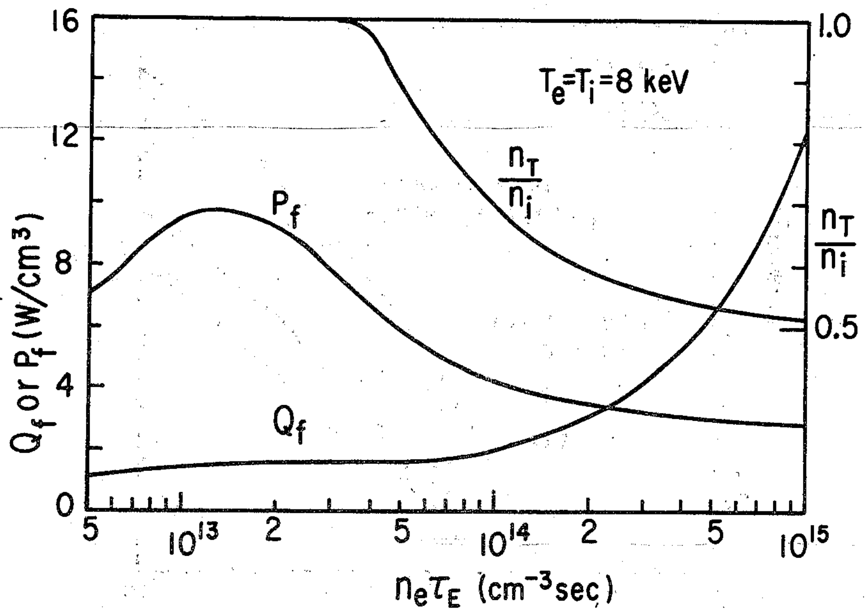
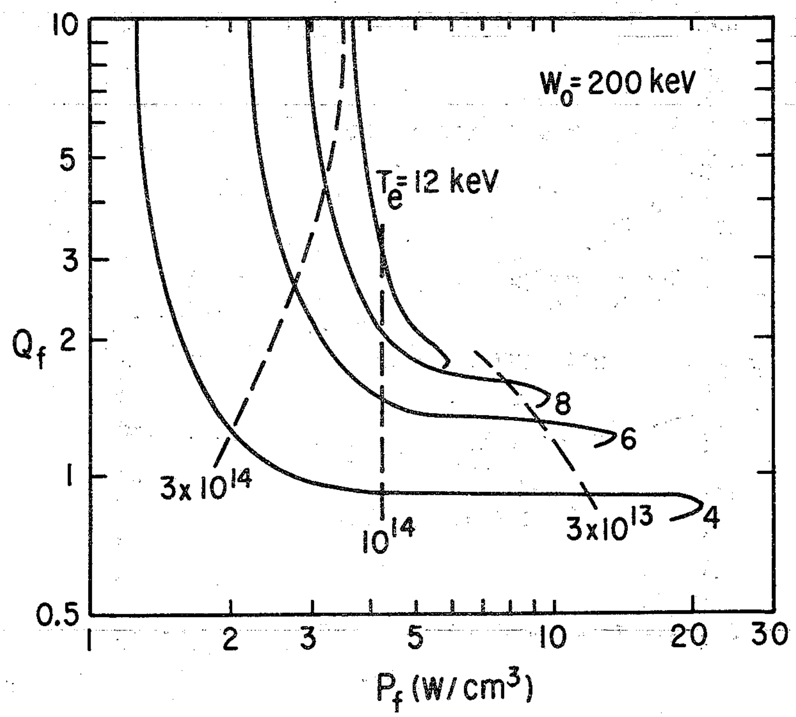


Fig. 6. Comparison of fusion power density for a two-energy-component plasma with optimal $\bar{\Gamma}$ (cf. Fig. 2), and a 50% D, 50% T thermal plasma ($\bar{\Gamma} = 0$). In each case, total plasma pressure = 0.655 J/cm^3, corresponding to B_t = 60 kG, $\beta_p = A = 3.5$, $q = 2.5$. Alpha-particle effects are neglected.



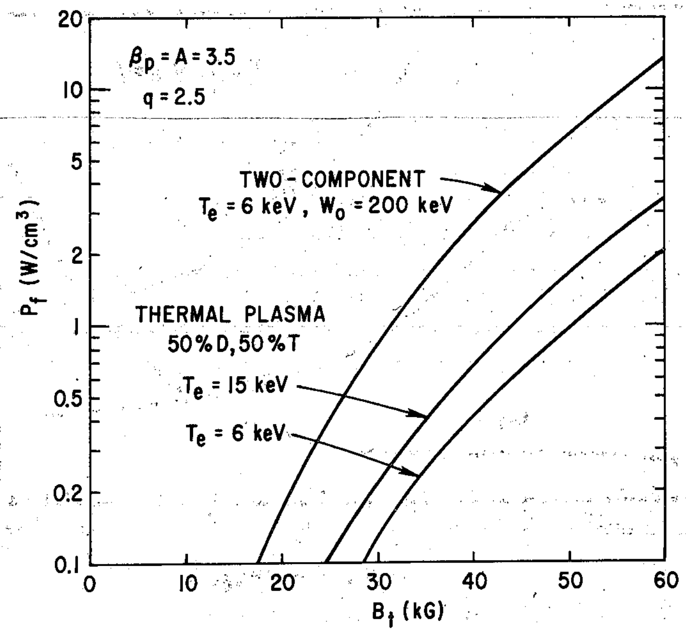
743656

Fig. 7. Dependence of P_f and Q_f on $n_e\tau_E$ for an 8-keV D-T plasma heated by 200-keV D beams. Maximum Q_f is attained for the plasma composition given by n_T/n_i, where n_i is the bulk-ion density. $p = 0.655$ J/cm^3. Radiation loss is included in τ_E. Alpha-particle effects are neglected.



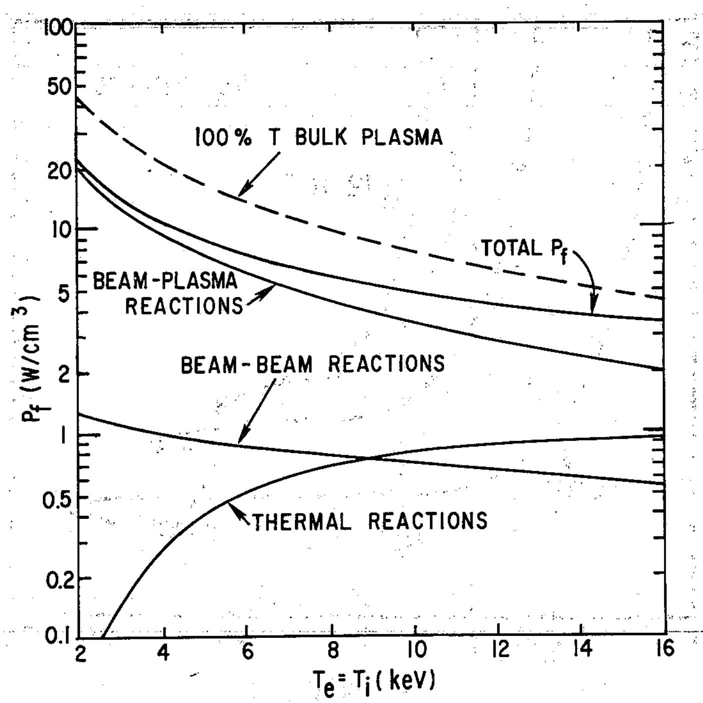
743657

Fig. 8. Q_f versus P_f for a D-T plasma with $T_e = T_i$ heated by 200-keV D beams. For each $n_e\tau_E$, the D-T composition of the background plasma is adjusted for maximum Q_f. For each T_e there is a maximum in P_f, while Q_f increases monotonically with $n_e\tau_E$. The dashed curves are contours of constant $n_e\tau_E$ (cm^{-3}sec). $p = 0.655$ J/cm^3.



753092

Fig. 9. Magnetic field dependence of maximum attainable fusion power densities. $T_e = T_i$. Total plasma pressure $p = 0.655 \ (B_t/60)^2$ J/cm^3. For plasma radius = 1.0 m, wall radius = 1.3 m, P_f = 1 W/cm^3 produces a neutron wall loading 0.31 MW/m^2.



743658

Fig. 10. Contributions to P_f from beam-plasma, beam-beam, and thermal fusion reactions in a 50:50 D-T plasma heated by equal number densities of 200-keV D beams and 300-keV T beams. The dashed curve is for a 100% T bulk plasma heated by 200-keV D beams. In each case, $\overline{\Gamma}$ is chosen to optimize P_f from beam-plasma reactions. Same plasma conditions as in Fig. 2.

QUESTIONS ABOUT SEVENTH PRESENTATION

Moir: This machine will be on 75% of the time, it seems, so about 25% of the time, it won't be generating power. I wonder if you could just tell us what your thinking is about the various ramifications of that second figure?

Jassby: There has been very little thought about that. One would have to have a blanket with a very large specific heat in order to maintain the thermal output. If one was mainly breeding fuel, that would not be so important.

Moir: What is the optimum?

Jassby: We feel that the minimum discharge time has to be about 10 seconds: so typically 10 seconds on and 5 seconds off. Does that strike you as being impractical?

Moir: Well, I can just imagine the power engineer turning himself inside-out if you turned the burners off -- you know, five seconds off and ten on and five off.

Jassby: Well, of course one could buy some storage but that would increase the capital cost.

Moir: Well, I think the issue is probably thermal cycling and structural effects.

Coffman: How many kg's of plutonium did you estimate for your blanket load?

Jassby: In the cases of 4 percent Pu, it would have to be on the order of 20,000 kilograms.

Coffman: At $30,000 a kg ... so that's not in your capital cost?

Jassby: No.

Coffman: The capital cost of the Pu would be in the neighborhood of a billion dollars.

Jassby: That's true, if I use $30 per gram. On the other hand, if one would be willing to put up with a pretty long doubling time at the beginning, in other words forget about making it economic initially, one could start off with a much smaller blanket inventory. One could use actinides as Wolkenhauer just suggested.

Taylor: What fraction of the 800 million is in the magnet cost?

Jassby: In the magnets, including the supports and refrigeration, it's about 13 percent since they are not very large.

Taylor: How do the rest of the costs break down?

Jassby: Well, I don't have time to read it but I'd be happy to show it to you. There are about 20 different items.

Dean: Is that in 1974 dollars?

Jassby: Yes.

Halpern: Could you contrast your beam requirement to those for a beam heated pure fusion tokamak?

Jassby: As far as the beam voltage is concerned, it would mean a larger voltage for a pure fusion tokamak simply because that for a pure fusion tokamak, the plasma would have to be much larger in order to get the required $n\tau$. As far as the powers are concerned, the design that I was talking about uses 370 megawatts. What one would want for a pure fusion device would depend on how fast one wanted to heat it up, and the estimates for that range from 50 MW to 500 MW. You're actually better off with higher power heating because then the energy requirements in the beam are smaller since, for example, the total radiation loss during the heating is greatly reduced. But I would say that the power is comparable although the voltage is smaller in the TCT case.

Baker: The key assumption on whether the approach you presented is viable or not, is whether or not you can really get β_p as a linear power of the aspect ratio which is about 3 or 4. Connected with that, did you consider the effect of alpha particles on the pressure?

Jassby: I assumed that all the alphas escape. One can arrange for that simply by putting a ripple in the magnetic field toward the outside. This is a relatively small device. All of the alphas, no matter where they are produced, are going to have orbits which enter the outer part of the discharge. By putting a ripple in the field at that point, one can make them drift outward. It turns out that, unlike a pure fusion reactor, one does not want to keep alphas since they do apply a pressure and therefore you cannot inject as much beam.

Baker: I would like to make one other comment. I think that this is a good example of how different the plasma systems are if you optimize for the neutrons or if you optimize for the pure fusion reactor. In one case, you maximize the power density, and in the other case you maximize Q. This will lead you to two very different things.

Moir: I wonder if you considered the neutrons streaming up the injectors. In our designs we have looked at this and we say that we need shielding around the injectors so that the source and the pumping inside the neutron region adds quite a lot to the complexity. You will have a lot of injectors on this design.

Jassby: The area is estimated to be 1.3 percent; the neutral beam access is about 1.3 percent.

Moir: So 1.3 percent of the neutrons leak out.

Jassby: Yes. But why can't you shield the beam lines?

Moir: You have to shield it.

Jassby: Yes. That problem exists with the TFTR by the way. It will have to be shielded as well.

Williams: What was the uncollided 14 MeV flux on the first wall in this design?

Jassby: 7×10^{13} neutrons per square centimeter per second. Average: 1.5 MW per square meter.

LA-UR-74-1861-Rev

TITLE: PROSPECTS FOR CONVERTING ^{232}Th to ^{233}U IN A LINEAR THETA-PINCH HYBRID REACTOR (LTPHR)

AUTHOR(S): R. A. Krakowski, D. J. Dudziak, T. A. Oliphant, K. I. Thomassen, G. E. Bosler, and F. L. Ribe

SUBMITTED TO: Fusion-Fission Energy Systems Meeting, AEC-DCTR
Washington, D. C.
December 3-4, 1974

los alamos
scientific laboratory
of the University of California
LOS ALAMOS, NEW MEXICO 87544

Form No. 836
St. No. 2629
73

UNITED STATES
ATOMIC ENERGY COMMISSION
CONTRACT W-7405-ENG. 36

ABSTRACT

The consequences of plasma end losses on the overall energy balance for a linear theta-pinch reactor can be ameliorated by use of a fission-enhanced, energy-multiplying blanket. This study is concerned with the constraints established by plasma physics, neutronics, energy balance, and economics on the design of a Linear Theta-Pinch Hybrid Reactor (LTPHR); although preliminary in nature, this study integrates these four important requirements in a consistent way. Although neutronically inferior to the $^{238}U/^{239}Pu$ fuel cycle, the $^{232}Th/^{233}U$ fuel cycle has been selected by this study on the basis of potential environmental advantages. Only present, well-understood technology for both the fission (HTGR) and fusion (Scylla) aspects of the LTPHR is evoked. On the basis of a generalized energy balance developed for the fusion-fission symbiosis, favorable energy multiplication (Q_E = total electrical power/recirculated power $\sim$ 5-10) results for theta-pinch systems which are below 1 000 m in length and self-sufficient for tritium. Although some ^{233}U-enrichment of the blanket is shown to be necessary, the optimum (minimum inventory for maximum energy multiplication) blanket configuration is yet to be evolved. An electrotechnology based on LC-resonant circuits is proposed, and reversible transfer of magnetic energy with efficiencies > 90% will be required. The mix between fission and fusion energy production is not specified. Preliminary cost estimates of major portions of the "nuclear island" (power supplies, switches, blanket, prorated fission burner) indicate costs in the range of 600 $/kWe for the highly ($^{233}U$) enriched systems. A major cost for the LTPHR (400-500 $/kWe) is associated with the capacitive power supply, and alternative magnetic energy storage schemes will be explored.

Typically, a 1 000-m-long LTPHR which has a 18 kg/m ^{233}U loading (non-optimized) gives an intrinsic doubling time of 11 y (0.79 tonnes/y ^{233}U production) for a first-wall, fusion neutron current of 2.0 MW/m^2. This system has a net tritium breeding

gain, generates $\sim$ 10 times its internal circulating power needs (for reversible transfer efficiency of 95%), and provides 7.38 MWe/m of electrical energy (12.8 MWe/m^3 of enriched seed or 2.44 kg-^{233}U/MWe) for a thermal conversion efficiency of 0.40; approximately 10 GJ of stored energy will be required, which must be switched at a frequency of 2-4 Hz to maintain a 1.0 MW/m^2 fusion-neutron wall loading. The design presented here in many ways is conservative, but these preliminary, favorable results warrant further exploration of the LTPHR concept.

CONTENTS

For the sake of brevity, the authors have omitted from this publication all descriptive and calculation detail contained in appendices. The full text including appendices has been published as USAEC Report LA-UR-74-186/Rev. (February, 1975)

1. INTRODUCTION

The stringent requirement that a DT plasma system meet the energy requirement of a fusion reactor (net electrical output > circulating power input) can be relieved when the plasma system is used to drive a hybrid (fusion-fission) reactor system. Such a fusion system relies on an enhancement of the energy available per neutron by incorporating a subcritical, fissioning blanket. A fusion/fission system can in principle be designed to emphasize either the intrinsic production of electrical energy or fissile fuel.* The former case uses the higher energy yield (~ 200 MeV/ fission) of fission reactions occurring within the blanket to increase the net energy worth of each 14.1-MeV neutron generated by the DT reaction, whereas a hybrid system that is designed to emphasize the production of fissile fuel may be a net consumer of energy. In either case the total energy produced (sensible heat within the blanket plus potential fission energy embodied within the bred, fissile fuel), once converted to electricity, must exceed the power consumed in operating the hybrid reactor by some economical margin.

A majority of studies[1-13] made of fusion/fission symbioses deal with the ^{238}U/ ^{239}Pu fuel cycle and generally emphasize neutronic and blanket design aspects of the problem. Studies which consider particular fusion reactor concepts[1,6,10,13] have been preliminary, do not consider in detail the constraints established by plasma

*No distinction is made in this study between fission reactions occurring within the hybrid blanket per se or within the core of a symbiotic fission burner; as far as the overall energy balance is concerned, energy generated by in situ fission or as potential energy in the form of bred, fissile fuel is not distinguishable. Economic and other considerations will dictate the balance between intrinsic energy vs. fissile fuel production. Quantification of these latter considerations is not within the scope of this paper, although the prognosis of a growing fission economy seems at this time to force the energy vs. fissile fuel "mix" for hybrid concepts more towards that of a fuel producer.

physics, and with one exception[10] present little or no engineering detail. Additionally, most of these hybrid designs rest on plasma physics and technological premises which have yet to be proven.

This study considers a high-density, linear theta pinch as a source of DT fusion neutrons for a fissioning/breeding blanket and is based on the $^{232}Th/^{233}U$ fuel cycle. Although the results presented herein are also preliminary and lack engineering detail, a concerted attempt has been made to couple closely the plasma physics, electrotechnology, and blanket neutronic aspects of the hybrid design. The basic premises used in this study are:

* Only the present, well-understood technology for both fusion and fission aspects of the Linear Theta-Pinch Hybrid Reactor (LTPHR) is to be used.
* The LTPHR is to be a net breeder of tritium to the extent that the system is self-sufficient for tritium fuel on the basis of a reasonably short ($\sim$ 1 y) doubling time.
* The LTPHR is to be a net producer of energy, although the mix between fissile burning vs fissile breeding is not specified.
* A thermal fission blanket (i.e., $^{232}Th/^{233}U$) is to be employed to minimize production of radioactive actinides.
* The LTPHR must yield a favorable energy balance without exceeding $\sim$ 1 000 m in length and without using unrealistically high magnetic fields ($\leq$ 30 T).

The linear theta pinch promises to generate DT fusion neutrons using a plasma physics and electrotechnology base which has been developed over many years of experimental, developmental, and theoretical experience at LASL. Hence, the DT plasma is heated and contained using well-established implosion/compression techniques, and experimentally confirmed scaling laws on end-loss behavior serve as the basis for the plasma models used herein. Although not yet specified in design detail, the LTPHR blanket proposes the direct adaptation of designs, materials constraints, and neutronic responses developed for the High-Temperature Gas-Cooled Reactor[14] (HTGR).

The major uncertainties in the proposed LTPHR are the long-term reliability of high-voltage components in the fast-pulse heating/confinement system (capacitors, switches, etc.) and the mechanical design of the theta-pinch coil. These problems, however, appear soluble with present experience and are not treated in this scoping study. These difficulties notwithstanding, the attractiveness of the LTPHR concept rests in its intrinsic simplicity as well as the large quantity of experimental and theoretical information that testify to its potential as a near-term source of DT neutrons. Integration of these advantages with a fissioning blanket to compensate for the effects of plasma end-losses is the primary purpose of this study.

A generalized energy balance is formulated that relates the number of fusion neutrons yielded per unit of invested energy, α^*(n/MeV), to the total energy delivered by those neutrons to the blanket, $E + E^*$ (MeV/n), as sensible heat E or bred fissile fuel E^*. The plasma physics is developed to evaluate numerically the range of attainable values of α^*; generally, reversible LC circuits are considered. The magnitude of the blanket parameter $E + E^*$ is determined for both ^{233}U-enriched blankets and pure thorium (breeder) blankets. The energy transfer efficiencies and the LTPHR lengths required for an economical system are bracketed. The economies of scale with respect to capacitor and switching cost are estimated, and a preliminary cost estimate of the "nuclear island" is presented.

2. ENERGY BALANCE

To evaluate the LTPHR energy balance, the idealized energy flow diagram depicted on Fig. 1 has been developed. Within the LTPHR blanket is developed the thermal power P_{TH} (MWt/m). The sensible energy worth of the fusion neutron is E(MeV/n),+ the (net) number of ^{233}U atoms produced per fusion neutron is [CV], the potential energy of bred fuel per fusion neutron is E^*(MeV/n), and the net power recirculated to the LTPHR is P_c(MWe/m). The fraction η_H of recirculating power P_c actually enters the plasma. Figure 2 gives a simplified electrical circuit, and major energy sinks are identified. Other quantities relevant to the energy balance are identified on Fig. 1, Fig. 2 and Table I.

+The energy E includes the 3.5 MeV/fusion associated with the alpha-particle reaction product.

The ^{233}U is bred at a rate $R(kg/m\ y) = I_{23}/[DT]$, where $I_{23}(kg/m)$ is the ^{233}U inventory in the LTPHR, and [DT] (y) is the associated fissile fuel doubling time. The bred ^{233}U is assumed to be continually burned in the associated fission reactor, which generates a thermal power $P^{*}_{TH}(MWt/m)$.+

The assessment of fusion energy systems has generally been presented in terms of defined "Q-parameters", which represent in one form or another the amount of energy generated per unit of energy invested. For the system depicted in Fig. 1, two "Q-parameters" are defined:

$$Q_P = (P_{TH} + P^{*}_{TH})/P_I \tag{1A}$$

$$Q_E = (P_E + P^{*}_E)/P_c \tag{1B}$$

The quantity Q_p is a plasma-oriented quantity, whereas Q_E represents a parameter of more engineering interest and characterizes the whole reactor power plant. As seen from Fig. 1 the potential for reversible transfer of energy to the plasma core is represented by the recovery efficiency $\eta_B = (1-\eta_I-f_{BLK}-f_L f_{EX})$. The total rate of energy input to the LTPHR, P_B, divides according to:

i) the fraction η_I which actually enters the plasma, ii) the fraction f_{BLK} deposited into the coil/blanket region as recoverable thermal energy, and iii) the fraction f_{EX} is either irretrievably lost as joule losses in electrical connections and switches, or is transiently stored in external inductances; the quantity $1-f_{EX}$ is an LCR transfer efficiency, and f_L is the fraction of external losses that is irretrievably lost. The remaining fraction η_B of P_B is assumed to be reversibly recoverable by the driving capacitor bank or other energy store.

If $<J>$ $(n/m^2\ s)$ is the average 14.1-MeV neutron current at the first wall (i.e., coil) of radius b(m), then, the thermal power developed respectively by the LTPHR and

+ The power developed by the symbiotic fission burner is expressed per unit length of LTPHR. An alternative approach is to consider the net production of fissile fuel by a hybrid reactor as a source of revenue to be factored into an economic assessment. In view of the uncertainty in the price for ^{233}U (or ^{239}Pu for that matter), the tack taken by this study is to acknowledge only the well-known energy worth of bred fuel and thereby decouple the analysis from uncertainties of the future fuels market.

the fission burner is given by,

$$P_{TH} = 2\pi b \langle J\rangle Ee + \eta_I P_B = f_{BLK} P_B \tag{2A}$$

$$P^*_{TH} = 2\pi b \langle J\rangle Ee^* \tag{2B}$$

where e equals 1.603 x 10^{-19} J/eV. Using Eq. (2), and the definitions given on Fig. 1, Eq. (1) becomes

$$Q_p = 1 + \alpha(E+E^*) + f_{BLK}/\eta_I \tag{3A}$$

$$Q_E = \left\{Q_p + [\eta^*_{TH}/\eta_{TH} - 1] \quad \alpha E^*\right\}\eta_I\eta_{TH}/(1-\eta_B).$$

For $\eta_{TH} = \eta^*_{TH}$:

$$Q_E = \eta_{TH}\,\eta_H\,Q_P = \frac{\eta_{TH}}{1-\eta_B}\,[\eta_I + f_{BLK} + \eta_I \quad \alpha(E+E^*)] \quad , \tag{3B}$$

where the relationship $\eta_H = \eta_I/(1-\eta_B)$ is used; η_H and η_I are respectively the fractions of P_c and P_B that are absorbed by the plasma. The parameter α(n/MeV) is the number of neutrons created for every unit of energy invested in the plasma <u>per se</u>,

$$\alpha = 2\pi b \langle J\rangle e/P_I \quad . \tag{4}$$

The quantity α is a plasma physics parameter, $(E+E^*)$ is dependent entirely upon the neutronics of the blanket design, and Eqs. (3) join both plasma and blanket responses into an evaluation of energy utilization <u>via</u> Q_p (physics) or Q_E(engineering).

In addition to an acceptable energy balance, the investment I_{23} (kg/m) of ^{233}U and the need to supply a symbiotic fission reactor with fissile fuel at some rate puts restrictions on the fissile fuel doubling time [DT] (y). The definition of [DT] given on Fig. 1 pertains only to the fissile inventory I_{23} (kg/m) associated with the fusion part of the symbiosis. In actuality, an inventory I^*_{23} (kg/m) is associated with the fission burner, and a more realistic definition of [DT] is given by $R = (I_{23} + I^*_{23})/[DT]$. Using this definition, equating the latter ratio to $2\pi b \langle J\rangle [CV]/(8.21 \times 10^{16})$,+ and defining the total specific inventory $[SI]_T = (I_{23} + I^*_{23})/(P_E + P^*_E)$, the following expression for [DT] results.

+The constant 8.21 x 10^{16} converts atoms/s to kg/y.

$$[DT] = 0.013316 \frac{Q_E [SI]_T}{\alpha \eta_H [CV]} \quad . \tag{5}$$

If Q_E is assumed to be ~ 10, as will be required for low electrical recirculating power, Eq. (3B) predicts $\eta_H \alpha \sim 0.05$ for $(E+E^*) \sim 500$ MeV/n and $\eta_{TH} = 0.4$. Eq. (5) predicts a doubling time of the order of 8 y if $[SI]_T$ is equal to a value that is common to an HTGR ($\sim$ 3.2 kg/MWe) for [CV] = 1.0. Recalling that η_H equals $\eta_I/(1-\eta_B)$ and that η_I is not expected to be a large number (η_I for theta pinches is on the order of the square of the compression ratio x_f), energy transfer efficiencies η_B of the order of 0.9 will be required for the LTPHR.

3. THERMONUCLEAR BURN AND CIRCUIT MODELS

To estimate the plasma parameter α(n/MeV) used in the simple energy balance for the LTPHR, a detailed thermonuclear burn calculation must be integrated with a realistic dynamic model of the circuit and implosion process. In this section is first obtained an order-of-magnitude estimate of α and η_I from a simple analytical model. The results of a more accurate computer calculation of the implosion/compression is then discussed.

3.1. Simplified Analytical Estimates

3.1.1. Estimate of Heating Efficiency

According to Fig. 1, η_I represents the fraction of circulating power P_B that is eventually recoverable from the plasma by the fusion thermal cycle.+ For example, if the LTPHR is assumed to be driven by a capacitive energy store of capacitance C(f/m), then energy $(1/2)CV_o^2$ is delivered to the LTPHR once every τ_c(s), where V_o is the bank voltage and τ_c is the cycle time. Of this energy the quantity $\sim (B^2/2\mu_o)\,\pi b^2$ enters the coil region as electromagnetic energy plus plasma energy. The plasma energy equals $\sim (B^2/2\mu_o)\pi b^2 x_f^2$, where x_f is the minimum plasma-to-wall radius ratio, a/b. If the theta-pinch coil is located at the first wall and operates at the blanket temperature, all of the plasma internal energy can be recovered by the thermal cycle. If, for instance, the coil resistance is negligible compared to the external resistance, then, $f_{BLK} \simeq 0$,

---

+The majority of the power $\eta_I P_B$ will exit the LTPHR _via_ the ends and in principle can be recovered by a direct-conversion cycle. For the purposes of this analysis, however, the end losses are assumed recoverable by the thermal cycle. In any case $\eta_I P_B$ is small relative to the fusion energy.

and $1-f_{EX} = (B^2/2\mu_o)\ \pi b^2/(CV_o^2/2)$ is similar to a vacuum transfer efficiency, then $\eta_I \sim (1-f_{EX})\ x_f^2$. The transfer efficiency $1-f_{EX}$ is governed primarily by external inductances and resistances and in practice can be made to exceed 0.9; the injection efficiency is determined almost entirely by x_f, which in turn is governed by the physics of implosion and compression. Compression ratios obtainable by simultaneous implosion/ compression techniques used in the Scyllac experiments at LASL[15] are $\sim$ 0.1, and the corresponding values of η_I are therefore about one percent. Although η_I has no impact on Q_p, from an engineering viewpoint [Eq. (3B)] the desirability for large values of x_f and hence η_I is obvious. The attainment of larger values of x_f is possible through staging techniques.[16,17] A portion of P_B can be recovered by i) the thermal cycle (joule heating within the compression coil), and ii) reversible transfer of magnetic energy to the compression coil and back to the capacitor bank.

3.1.2. Estimate of Neutron Yield

The conditions for pressure equilibrium and the production of thermonuclear energy for a plasma of radius a, density n and electron-to-ion temperature ratio $\lambda = T_e/T_i$, are given by,

$$nkT_i\ (1+\lambda) = \beta B^2/2\mu_o \tag{6}$$

$$1/4\ \pi a^2 n^2 \langle\sigma v\rangle = 2\pi b \langle J\rangle\ . \tag{7}$$

Use of Eq. (4) for the number of neutrons generated per unit energy invested into the plasma, α(n/MeV), gives

$$\alpha(\text{n/MeV}) = \frac{1}{6e}\ \frac{\beta}{(1+\lambda)^2}\ \frac{B^2}{2\mu_o}\ \frac{\langle\sigma v\rangle}{T^2}\ \tau_B\ , \tag{8A}$$

where T is in keV and SI units are otherwise used, and the time τ_B is equal to the burn time. For the purposes of this study the time when the magnetic field is applied τ and the end loss time τ_{EL} are taken to be the same, i.e., the machine length ℓ is selected to make both times compatible. The burn time τ_B, therefore will equal the end-loss time reduced by a form factor f_τ that in turn depends on the magnetic field waveform.

The fraction of P_B ($\sim 1/2\ CV_o^2\ \tau_c$) that actually enters the plasma as internal energy is given by

$$\eta_I = \frac{3}{2}\beta\,(1-f_{EX})\,x_f^2/(1+\beta\,x_f^2/2) \simeq (1-f_{EX})\,x_f^2\ . \tag{8B}$$

A simple expression for the end-loss time is given by[18]

$$\tau_{EL} = \sqrt{\frac{m_i}{2kT}}\,(\ell/2)\,\eta_{EL} \tag{9}$$

where η_{EL} equals $4\sqrt{\pi}\,R/(1+\sqrt{1-\beta})$ and R is the applied mirror ratio (assumed here to be unity). A more recent numerical calculation[19] gives τ_{EL} in the form of Eq. (9) and η_{EL} in numerical form. For $\beta = 0.8$, η_{EL} equals 4.9 for the old theory compared to 3.9 based on the recent numerical evaluation. Experimental evidence for R = 1 indicates that both theories render predictions which are $\sim$ 60% too high. Since experimental evidence is far from conclusive[19], however, the more recent numerical evaluation will be used to compute the theta-pinch length, ℓ. For ℓ = 500 m, T = 15 keV and $\beta = 0.8$, a self-mirrored theta pinch will have an end-loss time of 912 μs (1 146 μs based on the old theory). For this condition $\langle\sigma v\rangle \sim 3 \times 10^{-22}\ m^3/s$, with B = 30 T, $f_\tau = 1.0$, $\lambda = 0.5$ and $\alpha = 0.161$ n/MeV. Hence, approximately 6 MeV must be invested in the plasma for every neutron delivered; by increasing the LTPHR length to 1 km, this investment is reduced to $\sim$ 3 MeV.

The expression for the simple implosion-compression model[16,17] ($x_{SH} = 0.632$) without ion-electron equilibration can be used to relate x_f to the implosion electric field E_θ(kV/cm) for DT ions:

$$E_\theta(\text{kV/cm}) = 6.32\ x_f^{7/3}\ T^{1/2}\ B\ . \tag{10}$$

It is useful to define the quantity

$$\alpha^* = \eta_I\alpha = (1-f_{EX})x_f^2\ \alpha\ . \tag{11}$$

Whereas α is the number of fusion neutrons generated for each MeV unit of energy invested into the <u>plasma</u>, α^* represents the neutron yield for each MeV unit of energy

---

+ The neutron yield α can be evaluated for other fusion devices, such as a beam-driven tokamak. For instance, with ion temperatures of $\sim$ 15 keV and slowing-down times of $\sim$ 100 ms, the ratio of fusion energy (at 17.6 MeV)n) to injected beam energy is $\sim$ 2. Hence, $\alpha \sim 0.11$ n/MeV for proposed beam-driven tokamaks.

delivered to the LTPHR, i.e., $\alpha^* = 2\pi b \langle J\rangle e/P_B$. Hence, α^* is the parameter having more engineering and economic implications. Substituting x_f from Eq. (10), the following expression for α^* results,

$$\alpha^* = 0.852 \times 10^{23} \frac{\beta(1-f_{EX})}{(1+\lambda)^2} \frac{\langle\sigma v\rangle}{T^{17/7}} E_\theta^{6/7} B^{8/7} \tau_B \quad , \tag{12}$$

where SI units are generally used, but T is expressed in keV and E_θ in kV/cm.

Numerical evaluation of Eq. (12) requires the selection of appropriately time-averaged values for T and B when τ_B is related to the circuit LC time constant. More conveniently, use of maximum values and the assumption that $\langle\sigma v\rangle/T^2$ in Eq. (8) is slowly varying results in the introduction of a "form factor" f_τ, that reduces the circuit pulse duration τ to the approximately correct burn time. For instance, $\tau = \tau_{1/2} + \tau_{FT}$ for an LC circuit with half period $\tau_{1/2} = \pi\sqrt{LC}$ and a period where the maximum field is held flat-topped for a time τ_{FT}. In this case,

$$f_\tau = [1 + \frac{3}{8} (\tau_{1/2}/\tau_{FT})]/(1+\tau_{1/2}/\tau_{FT}) \quad . \tag{13}$$

This expression for f_τ will be used when comparing numerical results from Eq. (12) with values of α^* obtained from computer calculation. Equating τ_{EL} [Eq. (9)] with τ_B/f_τ, an alternate expression for α^* results:

$$\alpha^* = 0.514 \times 10^{18} \frac{\beta(1-f_{EX})}{(1+\lambda)^2} \frac{\langle\sigma v\rangle}{T^{41/14}} E_\theta^{6/7} B^{8/7} \eta_{EL} \ell f_\tau . \tag{14}$$

As an example consider the previous values for T, B, β, and ℓ. For $f_{EX} = 0.05$, $f_\tau = 1.0$ and $E_\theta = 3$ kV/cm, Eq. (14) yields $\alpha^* = 0.00125$ n/MeV. This is to be compared to the value derived for an economical power plant ($Q_E = 10$) at a reasonable value of $E + E^*$ (500 MeV/n). Neglecting η_I and f_{BLK} and setting $\eta_{TH} = 0.4$ in Eq. (3B), $\alpha^*/(1-\eta_B)$ equals 0.05. Hence these preliminary estimates indicate that $\eta_B \sim 0.97$ will be required. More detailed computer calculations will explore the engineering breakeven condition in Sec. 3.2. below.

When a capacitor bank of capacitance C(F/m) and voltage V_o is used to power the adiabatic compression, the stored energy per metre length of plasma is expressed as

$$W_c \text{ (MJ/m)} = \frac{1}{2} \times 10^{-6} \, CV_o^2 = \pi b^2 (B^2/2\mu_o) \, 10^{-6}/(1-f_{EX}) \quad , \tag{15}$$

where $(1-f_{EX})$ accounts primarily for magnetic energy stored in the parasitic inductance L_{EX} between the capacitor bank and the compression coils, as well as external resistive losses. Therefore, using SI units,

$$W_c \text{ (MJ/m)} = 1.25 \, b^2 B^2/(1-f_{EX}) \quad . \tag{16}$$

In the example given above and for b = 0.1 m and $1-f_{EX} = 0.95$, W_c equals 8.3 MJ/m and the total capacitor energy is 4 121 MJ. Substituting Eq. (16) into Eq. (14) yields:

$$\alpha^* = 0.136 \times 10^{18} \, \frac{\beta(1-f_{EX})^{11/7} f_\tau}{(1+\lambda)^2} \, \frac{\langle\sigma v\rangle}{T^{41/14}} \, \frac{E_\theta^{6/7} W_c^{4/7} \eta_{EL} \, \ell}{b^{8/7}} \quad . \tag{17}$$

In summary, the present model indicates in order of magnitude that a power plant with acceptable energy balance ($Q_E = 10$) is achievable for a 500 metre system with a compression field of 30 T and a shock-heating field of 3 kV/cm. For a coil inner radius equal approximately to the first-wall radius of 0.10 m, the total stored energy would be 4.1 GJ. Such a large value for $W_c \ell$ implies the use of high-energy density, low-voltage capacitors of high reliability. At this level of energy storage the alternatives of magnetic and inertial storage must also be examined. The assumption of $\sim$ 95% efficiency in transferring energy from the energy store to the compression coil and back implies a Q value of the LCR circuit of 20. A more exact treatment of the plasma burn and estimation of α^* is given in Sec. 3.2.

3.2. Numerical Calculations of Thermonuclear Yield

The primary objective of these numerical calculations is to study parametrically α(n/MeV), η_I, and α^*(n/MeV) = $\eta_I \alpha$ with a dynamic thermonuclear burn model used in conjunction with a realistic representation of the non-linear circuit. A number of schemes were investigated for programming the magnetic field to test the validity of the simple implosion theory [Eq. (10)] and to investigate the sensitivity of α^*(n/MeV). The idealized magnetic field waveforms considered by this study are summarized in Fig. 3 and are described as follows:

Curve 1: Simple, staged implosion, sinusoidal rise of compression field, and crowbar at the quarter period. (Appendix 9.1)

Curve 2: Same as Curve 1 except a free-expansion implosion is used.

Curve 3: Same as Curve 1 except the crowbar is removed and the LC circuit is allowed to ring down to zero.

Curve 4: Implosion and compression of the plasma derived from the same, high-voltage capacitor bank, followed by a crowbar. This circuit is not realistic from an engineering viewpoint, but was used only to model realistically the implosion process. (Appendix 9.1)

Curve 5: "Stretched" wave form: Same as Curve 3 except the peak magnetic field is held flat for a fixed time prior to ringing down.

Because of the lack of realism associated with standard crowbar techniques and/or the use of high-voltage compression field capacitors, the calculational results pertaining to Curves 1 and 4 are discussed in Appendix 9.1. The free-expansion model for implosion heating is yet to be proven technologically and therefore is not considered further. Hence, the computational results considered herein pertain only to Curves 3 and 5: an LC-ringing circuit with or without a "flattop" at the peak magnetic field using simple implosion heating (x_{SH} = 0.63).

3.2.1. Calculational Model

The computation of the neutron yield α^* involves the solution of a non-linear, LC circuit equation in conjunction with the thermonuclear burn equations. The (β=1) plasma balance determines the time-dependent inductance of the compression coil, alpha-particle heating is not taken into account, and end losses are not modeled into the computation. A detailed description of the thermonuclear burn model is given in Appendix 9.2.

strates in schematic detail the staged LC circuit being proposed to 3 on Fig. 3. Appendix 9.3 describes a circuit which can extend maximum compression field (waveform 5); both the ringing LC circuit stretched" LC circuit models are the basis of the α^* computations reported ein. Referring to Fig. 4, the low voltage ($\sim$ 15 kV) compression-field capacitor bank is discharged through the SCR switch 1 after the plasma has been imploded. SCR switch 1 closes, and the energy within the theta-pinch coil flows back into the capacitor. During the off period, the compression field energy store is slowly charged to make up the loss $(1-\eta_B)\ W_B$. On the next half cycle SCR 2 passes the current from the capacitor bank which is charged to reverse voltage. The stretched LC circuit results in more efficient use of the magnetic energy in generating fusion neutrons; however, it may be more difficult to achieve. In the case of the simple LC circuit the half-period is assumed to be extended by using an N-turn compression coil whose inductance is $\mu_o N^2 \pi b^2$ (H m). The reactor length will then scale as $C^{1/2}N$.

3.2.2. Calculational Results

The computational results summarized herein are based on the LC circuit model depicted on Fig. 4. In Appendix 9.1 a comparison is given between a numerical implosion model and the simple implosion theory [Eq. (10)]. Appendix 9.2 describes the computational method used to determine the thermonuclear burn parameters. For all calculations E_θ = 3 or 4 kV/cm, x_{SH} = 0.623, b = 0.10 m (vacuum inductance = 40 nH m), and L_{EX} = 4 nH/m.

Table II summarizes the dependence of α^* (n/MeV) on C, V_o and N. For a fixed set of parameters, α^* shows a maximum as a function of the initial filling density n_o. Figure 5 illustrates this behavior (which is discussed in Appendix 9.1) for C = 180 mF/m and V_o = 10 kV. The maximum values of α^* are given in Table II, and the time dependence of B(T), T_i(keV), x, J(n/cm^{2}s) and $\int_0^t$ Jdt for a typical case is shown in Fig. 6.

Using the numerically computed results given in Table II, the validity of Eq. (12) can be tested. Figure 7 compares the numerically computed values of α^* with those predicted by the analytical expression; the analytical prediction is surprisingly good, reproducing the numerical results to within a factor of 0.8. Hence, one can use Eq. (12) or a variation thereof with some degree of confidence for scaling purposes.

Since the stored energy $W_c = 1/2\ CV_o^2$ is limited by i) cost considerations and ii) the upper limit on B to which the theta-pinch coil can be subjected, the desired increases in α^* have been achieved by increased burn times (i.e., increased N and reactor length ℓ). As observed from Fig. 6, a sinusoidally varying compression field is not very efficient, in that the thermonuclear burn occurs for only a small portion of the period. This is improved when the compression field is held constant for a flat-top period τ_{FT} after reaching its maximum. Computer runs 25 to 35 listed in Table II summarize the burn results for various values of τ_{FT}, N = 1, C = 90 mF/m, and V_o = 15 kV. Computer runs 18 to 24 model the identical case except τ_{FT} = 0 and N is allowed to vary. The dependence of α^* on $\tau = \tau_{FT} + \tau_{1/2}$ for both cases is illustrated in Figs. 8 and 9 for E_θ equal to 3 and 4 kV/cm. This series of four cases is characterized in Table III for ℓ = 1 000 m and will be used for numerical evaluation in Sec. 5 once the blanket neutronics are resolved (Sec. 4). The required reactor lengths ℓ(m) were calculated from Eq. (9) on the basis of the time $\tau_{1/2} + \tau_{FT}$ and the maximum ion temperature (as determined from computer calculations), and therefore are conservative. The 25% increase in E_θ (from 3 to 4 kV/cm) generally results in a 20% increase in α^* which is in accord with the predictions of Eq. (12). As is seen from these results and Eq. (3A), favorable energy balance (Q_E = 5-10) appears to be possible for $\eta_B \sim 0.95$ and $E + E^* \sim 500$ MeV/n. The question of $E + E^*$ is addressed in the following Sec. 4, whereas Appendix 9.4 addresses the question of η_B; typically values of $E + E^*$ and η_B in the range 400-500 MeV/n and 0.95, respectively, appear feasible.

4. THE FISSIONING BLANKET

4.1 Rationale for the $^{232}Th/^{233}U$ Fuel Cycle

The $^{232}Th/^{233}U$ fuel cycle was chosen for study in the LTPHR for several reasons, paramount of which is the potential for a lower radiological hazard compared to the $^{238}U/^{239}Pu$ cycle. The lower fissile inventory per unit of power (i.e., higher specific power, MWt/kg fissile fuel) and the larger neutron yield per fissile absorption, η, inherent to ^{233}U in a thermal neutron spectrum are other advantages. The radiological advantage stems from at least three factors:

i) The specific activity (Ci/g) of ^{233}U compared to ^{239}Pu scales inversely as the relative half-lives, 2.439×10^4 y/1.58×10^5 y = 0.154.

ii) The maximum permissible concentration (MPC) for ^{233}U is above that for ^{239}Pu by a factor of 6.7×10^{-3} for air or 6.0×10^{-2} for water.

iii) For a given power level, the thermal-spectrum ^{233}U blanket requires less fissile fuel than a fast-spectrum ^{239}Pu blanket.

These radiological considerations do not account for secondary isotopes such as ^{231}Pa or ^{232}U, which could considerably degrade the radiological advantage if thorium is exposed to a high-energy neutron flux. As seen from the activation/decay scheme for Th/Pa/U depicted in Fig. 10, the ^{231}Pa, ^{230}Th, and ^{232}U isotopes, all radiologically undesirable, are produced by high-energy neutron absorptions on ^{232}Th or ^{233}U.

In addition to the potential radiological advantages, the following advantages can also be identified for the $^{232}Th/^{233}U$ cycle.

* A fuel technology can be used that is well established for the HTGR.
* Thorium resources are used, rather than uranium.
* Fuel separation is entirely by chemical means (i.e., lower separative work), as compared to hybrid systems which must be driven by ^{235}U.
* Bred fuel can be provided to the HTGR, a fission reactor concept considered to be of low environmental impact (i.e., higher temperatures and less reject heat per unit of generated electrical power).

The primary reason why the $^{232}Th/^{233}U$ fuel cycle has not been seriously considered for fusion/fission systems is the low fast-fission cross section for ^{232}Th relative to ^{238}U; this shortcoming represents a distinct disadvantage. These and other neutronic considerations are reviewed in Appendix 9.5.

4.2 Neutronic Model

All survey neutronics calculations reported by this study have been made with one-dimensional cylindrical models. In all cases the neutron transport calculations employ a discrete-ordinates multigroup code, DTF-IV,[20] in an S_4-P_3 mode.

Cross sections have been derived from the LASL Nuclear Data File. A 19-group multigroup structure, with energy bounds as shown in Table IV, was chosen for analysis of thermal "hybrid" systems. This library is a collapsed subset of the 25-group library used for "fast" hybrids[13] employing the ^{238}U-^{239}Pu cycle, except for the use of four thermal groups. Provision is then made for thermal upscatter in these last four groups. Resonance self-shielding and thermal spectra effects were included for a temperature of 1373 K.

Multigroup cross section sets for carbon, ^{232}Th, and ^{233}U were generated directly from ENDF/B basic data files with the use of a number of available computer codes. The LASL calculating procedure uses the MC^2 system for generating all above-thermal broad-group cross sections. For thermal groups, the neutron spectrum generated by MC^2 is incorrect, and the generation of broad-group cross sections requires the support of the FLANGE and GLEN codes, which generate the correct neutron spectrum for a second pass through MC^2.

Transport calculations were performed for a unit 14.1-MeV, fixed neutron source uniformly distributed in the plasma region. The converged fluxes were then used for reaction-rate edits to determine quantities such as tritium production by $^6Li(n,\alpha)$ and $^7Li(n,n'\alpha)$, ^{233}U production via $^{232}Th(n,\gamma)$, $^{230,231}Th$ production via $^{230}Th(n,Nn)$, and fission rates.

The development of the neutronic blanket model has been evolutionary as both neutron and plasma physics considerations became more apparent. Preliminary scoping calculations used a 0.05-m-radius coil and a very simple, two-region blanket design. The physics requirements necessitated an increase of the coil radius to 0.10 m, and preliminary neutronics results pointed to the need for refinements in the material arrangements within the blanket. Although major conclusions are based on the evolved models, the results of preliminary computations, inconsistencies notwithstanding, are briefly summarized in order to describe clearly the development and rationale of the neutronics model used. After a brief discussion of these preliminary results, more detailed analyses of a pure breeder (i.e., no ^{233}U enrichment) and a breeder/burner (i.e., considerable ^{233}U enrichment) models are presented.

4.3 Neutronic Results

As discussed earlier, the fusion/fission system in extreme cases can be designed to emphasize either the breeding of fissile fuel or the generation of in situ energy. Because of the lower fast fission cross section for ^{232}Th, a $^{232}Th/^{233}U$ hybrid reactor suffers a distinct disadvantage relative to the $^{238}U/^{239}Pu$ fuel cycle; Lee[21] has reported that the pure thorium system multiplies energy and converts fertile to fissile fuel by factors of 0.207 and 0.540, respectively, less than an equivalent uranium (natural) system. Similarly, thorium energy multiplication and conversion ratios are, respectively, factors of 0.275 and 0.614 less than for pure ^{238}U, according to Lee.[21]

In Appendix 9.6 are discussed the results of a similar calculation, in which fusion neutrons impinge upon a pure Th (metal) blanket containing appropriate quantities of lithium for tritium breeding. For this highly idealized case, $E = 49$ MeV/n, $E^* = 175$ MeV/n, and [CV] = 1.0 when the tritium breeding ratio, [BR] = 1.0. For $I_w = 1.0$ MW/m^2, the fissile fuel production rate amounts to $R(kg/m\,y) = 3.39\ I_w\ [CV] = 3.39\ kg/m\,y$. At $\sim$ 200 MeV/fission and $\eta_{TH} \sim 0.4$, the energy worth of ^{233}U is $\sim$ 0.95 MWe y/kg . Hence, the pure thorium breeder described in Appendix 9.6 is capable of satisfying the fuel needs of two 1 780 MWe fission burners per year if such an LTPHR is 1 000 m in length. If the ^{233}U fission reactor has itself a breeding deficit of 0.05 (i.e., $[CV]^* = 0.95$), then the LTPHR is capable of refueling $\sim$ 40 such burners per year. Because of the relatively low value of $E + E^*$ (224 MeV/n), however, a 1 000-m LTPHR would give $Q_E = 2.31$ or 2.67 for the LTPHR cases A and B (Table III, $\eta_{TH} = 0.4$, $\eta_B = 0.95$). The ratio P_c/P_E is given approximately by $(1 + E^*/E)Q_E$ and equals 1.98 and 1.75 for these two cases; the purely breeding LTPHR of 1 000-m length cannot generate enough intrinsic power to satisfy the total recirculating power requirements. For P_c/P_E equal to unity, Q_E would have to equal 4.69 and α^* would have to exceed 25.5×10^{-4} n/MeV for the pure breeder. The length for either case A or B would exceed 1 km under these conditions. Since the blanket model used for these estimates (Appendix 9.6) is neutronically optimistic (i.e., pure metal, little structure), it is apparent that a LTPHR driving a pure ^{232}Th blanket is not practical, and acceptable values of Q_E can be achieved only _via_ a ^{233}U-fission boost.+ Evoking the stretched LC circuit for $\ell \sim 1\,000$ m increases α^* (and therefore Q_E) by a factor of 1.6 (Figs. 8 and 9). As shown, however, in Appendix 9.6, use of a realistic blanket structure++ results in values of $E + E^*$ which are below 100 MeV/n. Hence, the major portion of the neutronic studies focuses on ^{233}U-enriched blanket concepts to achieve simultaneously tritium breeding ($[BR] \geq 1.0$) and high values of $E + E^*$ ($\sim$ 400-500 MeV/n). The following two sections address, respectively, preliminary scoping calculations and more detailed parameter studies on ^{233}U-enriched blanket systems.

+This statement is made on the basis of simple end-loss theory. Decreasing the rate of plasma end loss by a factor of 2 to 4 significantly improves the prognosis for a pure-thorium LTPHR.

++Neutronic calculations on a carbide/graphite blanket have been made and are reported in Appendix 9.6.

4.3.1 Preliminary Scoping Calculations

Initial attempts to establish an optimum blanket design (i.e., maximum $E + E^*$, minimum fissile inventory I_{23}, and net tritium breeding) used the simple blanket model depicted in Fig. 11. A Be or BeO region containing Li_2O was placed immediately adjacent to the compression-coil/first-wall region in order to multiply fusion neutrons and to breed tritium. A two-region ^{233}U-^{232}Th enriched seed/^{232}Th fertile-seed region containing $^6Li_2O$ was stationed outside the beryllium multiplier region. Some stainless steel structural material was also included in the blanket. This simple model yielded inadequate energy multiplication and tritium breeding. Neutron multiplication via the Be(n,2n) reaction was not sufficient to boost the neutron population to adequate levels for tritium breeding and significant energy production.

Table V summarizes preliminary neutronics results that are based on this simple model. For the calculations summarized in Table V the lithium was removed, since the compression coil degraded the neutron energy to the point where the $^7Li(n,n'\alpha)T$ reaction was unimportant and attempts to breed tritium via the $^6Li(n,\alpha)T$ reaction seriously reduced the neutron population and energy multiplication.

Although replacement of BeO with Be increased the energy multiplication by 7.4%, substitution of Th (metal) for Be resulted in a 58% increase in energy multiplication over the BeO case. Recognizing that the use of ^{232}Th as a neutron multiplier in place of Be leads to radiologically undesirable side reactions, [$^{232}Th(n,Nn)$, c.f. (Sec. 4.2, Fig. 10)], this choice was made in favor of the greater energy multiplication. The stainless steel structural material was also removed, which resulted in an additional 58% increase in energy multiplication.

4.3.2 Refined Blanket Model with ^{233}U-Enrichment (Breeder/Burner)

To give some flexibility to the task of maximizing energy multiplication and minimizing the fissile ^{233}U inventory, a two-zone loading scheme within the enriched-seed region was adopted. Hence, the blanket design evolved with a Th multiplier, a two-zone enriched-seed region, and a totally graphite structure. Preliminary calculations on the basis of this model were made to optimize energy multiplication and to determine neutron leakage from the blanket. The neutron leakage from the enriched seed was sufficient to allow incorporation of an outer Th region (to enhance E^*) as well as an enriched (95 a/o) $^6Li_2O$ region (to breed tritium). In this way tritium breeding was re-introduced into the blanket model.

Figure 12 schematically illustrates the blanket model which evolved from the preliminary studies. Five series of parametric neutronic studies, using the model depicted in Fig. 12 as a "point case," were made to investigate the influence of composition and geometry on tritium breeding [BR], I_{23} (kg/m), neutron leakage, E(MeV/n), E^*(MeV/n), specific power E/I_{23} (MJ/kg per n/m), and [CV]. The conditions, variables and results of these parameter studies are summarized below.

4.3.2.1 Parameter Study 1: Variation of ^{233}U Concentration in Enriched Seed

Using the dimensions indicated in Fig. 12, the ^{233}U loading was varied uniformly in both inner and outer enriched seed regions. Figure 13 shows the results of this variation. Tritium breeding is possible for $C/^{233}U$ ratios below 200 ($I_{23} \geq 50$ kg/m), $E + E^* =$ 415 MeV/n, and $k_{eff} \sim 0.69$. The neutron leakage from the blanket portends an increase in tritium breeding or ^{232}Th conversion by either the addition of more ^{232}Th to the reflector region or more $^6Li_2O$. The specific power $(E^* + E)/I_{23}$ is a minimum at this point and equals 1.3×10^{-18} (MJ/kg)/(n/m), or $E/I_{23} = 0.95 \times 10^{-18}$ (MJ/kg)/(n/m).

The ^{233}U enrichment where [BR] = 1.0 on Fig. 13 is selected for a blanket model with which to evaluate further other blanket parameters. Given on Fig. 14 is the spatial distribution of [CV], total fission rate, and [BR] for this particular blanket configuration; these distributions can be considered as typical of the other blanket parameter studies reported herein. The tritium breeding is maintained exclusively by the outer $^6Li_2O$ region, and [BR] is seen to fall-off rapidly with $^6Li_2O$ thickness. Within the enriched seed region [CV] is negative, i.e., the ^{233}U is being consumed faster than it is being produced in this region. Fast fission in the multiplier region experiences a four-fold decrease across 7 centimeters of thorium and is a direct indication of the spectral shift to lower neutron energies. A similar behavior is indicated in the outer fertile seed region. In summary, therefore, the majority of $^{232}Th/^{233}U$ conversion occurs in the thorium multiplier region, and most of the *in situ* energy multiplication takes place in the enriched seed region.

4.3.2.2 Parameter Study 2: Variation of Th Reflector and Multiplier Thicknesses

Figure 15 illustrates the effect of transferring Th from the reflector region to the multiplier region; Reference Case 1 corresponds to Parameter Study 1 (Fig. 13). Transfer of thorium from the reflector to the multiplier decreases E (i.e., in situ fission) and increases $E + E^*$ (i.e., conversion, [CV]). Although only a marginal decrease in leakage results, the radiological undesirable Th(n,Nn) reactions are enhanced by adding more Th to the multiplier region. Tritium breeding will not occur for multiplier thickness above 11 cm, although conversion of all leakage neutrons theoretically allows multiplier thickness up to 15.5 cm. In essence, the breeder/burner character of the LTPHR blanket can be adjusted by variations in the thickness of the Th multiplier thickness, ΔR_m.

4.3.2.3 Parameter Study 3: Variation of Coil Thickness

Using Reference Case 2 (Fig. 15) as a baseline, the compression coil thickness was varied keeping the blanket at a fixed thickness (89 cm). A disadvantage of a theta-pinch operating with high magnetic fields is the need to place the compression coil at the first wall. As seen from Fig. 16, the effects on E and E^* of the neutron moderating/absorbing coil is significant. Generally, tritium breeding for coil thickness much above 1 cm in thickness is difficult for this design. The ^{233}U loading, I_{23}(kg/m), could be increased to improve energy multiplication and tritium breeding at the expense of increased ^{233}U inventories, doubling times, and neutron multiplication (criticality). Criticality, however, is never a problem for any of the designs considered.

4.3.2.4 Parameter Study 4: Vary Inner Enriched Seed Thickness With a Pure Th Outer Seed of Fixed Thickness

Some indication of the effects of the enriched seed thickness is indicated in Fig. 17. As expected, tritium breeding, fertile conversion, and total energy multiplication $(E + E^*)$ show dramatic increases with increased ΔR_{IS}. The in situ fission rate, as indicated by E(MeV/n), particularly shows a dramatic increase as ΔR_{IS} is increased. Although 1.6 to 1.8 ^{233}U atoms are created for each fusion neutron, this scheme shows little promise for tritium breeding. Tradeoff of [CV] for [BR] could, of course, be achieved by thinning the Th reflector.

4.3.2.5 Parameter Study 5: Vary Pure Thorium Outer Seed Thickness

In order to ascertain the effects of the outer thorium region a parameter variation was performed as shown in Fig. 18. In this study the thickness of the pure thorium outer seed was varied and the enhancement of [CV] and E^* with increased thickness can be seen in the figure. However, this occurs at the expense of decreased [BR] and specific power. Because of the high fissile loading and inadequate tritium breeding, this blanket configuration was not considered further.

4.3.3 Reference Blanket Design

After analysis of the parameter studies discussed above, along with further unparameterized attempts at optimization, a reference blanket was chosen for preliminary evaluation of the LTPHR. Although the blanket is still unoptimized with respect to maximizing $(E + E^*)/I_{23}$, it does have the requisite properties of tritium breeding and adequate energy multiplication. Table VI presents a summary of the reference blanket configuration and performance characteristics. The chosen design has the virtue of preserving the radiological advantages of the $^{232}Th-^{233}U$ cycle to a large degree, by avoiding the thorium multiplier concept. Upon analysis of the data from the foregoing parameter studies, as well as those discussed in Appendix 9.6, it appeared that the relatively small fission rate in ^{232}Th did not justify use of a multiplier. For example, in the reference core 1 (Fig. 15) 0.94 fissions occur in ^{232}Th per DT neutron. Thus, the tritium breeding was accomplished in a 99 a/o-^{7} Li enriched region immediately outboard of the coil, while at the same time degrading the neutron energy spectrum impinging on the inner enriched seed. Using enriched ^{7} Li rather than natural Li also decreases the low-energy neutron capture in the tritium-breeding region, thereby enhancing the thermal flux in the inner enriched seed. The inner and outer enriched seeds have 10 a/o and 4 a/o-^{233}U, respectively, with a carbon to heavy-metal ratio of 214 (typical of an HTGR). Thus, ^{232}Th conversion occurs in the enriched seeds as well as the fertile seed. Most of the energy produced by the blanket is *in situ*, yet the doubling times for the blanket are reasonable; vis, 5.6 to 11.2 y, respectively, for I_w = 4.0 to 2.0 MW/m^2. The tritium breeding ratio, [BR] = 1.35, is more than adequate to provide a rapid doubling time while conservatively accounting for blanket streaming paths and additional structures which may be required in a detailed engineering design. Criticality accidents do not appear credible under conceivable geometry changes as the unperturbed effective multiplication factor, k_{eff}, is 0.80. The unoptimized, reference blanket summarized on Table VI is used to numerically evaluate the energy balance and economic picture for a number of LTPHR cases (Table III).

5. SUMMARY AND CONCLUSIONS

Throughout the presentation of physics (Sec. 3) and neutronic (Sec. 4) results, indications were given of the overall energy balance for the LTPHR. A specific, optimized design cannot be given on the basis of these results. However, numerical parameters for the blanket design given in Table VI are presented.

The four LTPHR DT burn cases described in Table III are considered. Table VII summarizes physics, blanket and energy-balance parameters for these four cases. In addition, the DT burn case D has been combined with a pure Th (metal) and a Th/C blanket (Appendix 9.6), corresponding to cases E and F, respectively, in Table VII.

On the basis of the sample results presented in Table VII, favorable (Q_E = 4-12) energy balances can be achieved with ^{233}U-enriched blankets. The more efficient use of circulating energy made by the "stretched" LC circuit (cases C and D) renders a 1000-m system that generates 790 kg/y of ^{233}U (at $I_w = 1.0\ MW/m^2$) and enough intrinsic electrical energy to satisfy $\sim$ 10 times its recirculating power needs (for $\eta_B = 0.95$). The large fissile fuel inventory (I_{23} = 18 kg/m), however, will necessitate a first-wall neutron current, $I_w > 4.0\ MW/m^2$ in order to yield intrinsic doubling times for ^{233}U below 5 y. The purely breeding LTPHR (cases E and F, no ^{233}U in the blanket) gives a poor-to-acceptable energy balance, depending on the optimism evoked by the neutronic design (case E vs case F); fissile fuel breeding rates and values of Q_E decrease by a factor of 2.5 when thorium metal is replaced by the more realistic Th/C HTGR blanket materials. Some ^{233}U enrichment of a pure breeder seems inevitable, therefore, and further computation is needed to determine an optimum value for the ^{233}U inventory, I_{23} (kg/m).

The capital cost anticipated for capacitors and SCR switches has been estimated in a preliminary way in Appendix 9.7 on the basis of 0.25 $/J for long-life capacitors and $25 per SCR component (rated at 1.2 kV and 2 kA). Cost estimates for the systems described on Table VII range around 400-500 \$/kWe for capacitors and switches. The cost of an HTGR reactor core is approximately 340 k\$/m^3, on the basis of which the LTPHR blanket is expected to cost in the 120 \$/kWe range for cases A-F. Assuming 500 \$/kWe for the nuclear portion of the symbiotic fission burner, the prorated capital charge for the fission burner portion of the symbiosis is expected to be $\sim$ 60 \$/kWe (see Appendix 9.7). Hence, the major components of the fusion/fission symbiosis are estimated to cost $\sim$ 600 \$/kWe for cases A-D, more than half of which is associated with the LTPHR power supply. Since the power supply is presently a major cost factor, further investigation will be done on the possibility of replacing the simple capacitor-SCR drive by magnetic energy storage at reduced cost. The costs associated with the pure breeder cases (E and F) are greater than the ^{233}U-enriched cases because of the low energy multiplication, $E + E^*$.

In summary, therefore, these preliminary estimates indicate that the LTPHR operating on a ^{232}Th/^{233}U fuel cycle can show an economical energy balance while simultaneously breeding tritium and maintaining a length below 1 000 m. Enrichment of the fissioning blanket with ^{233}U appears necessary, although an optimized design which minimized the fissile fuel inventory I_{23} (kg/m) must await further computation. In addition to these neutronic parameter studies, time-dependent burn-up calculations, cross-section sensitivity estimates, and blanket engineering evaluation (heat transfer, assembly, operations, etc.) must proceed before the LTPHR concept can be assessed further. To some extent the required engineering effort has begun; a preliminary sketch of a gas (He)-cooled LTPHR module is shown in Fig. 19. A two feed-slot arrangement is shown and a fuel element technology that is readily extrapolated from the HTGR experience is envisioned. As noted previously, the electrotechnology is well understood in principle, but questions of scale and associated transfer efficiencies must be further resolved.

Further refinements to the computational model used to determine the thermonuclear burn parameter α^* will also be made in future studies. These refinements include a more realistic treatment of the spatial/time decay of ion density during the burn as a result of end loss. Methods to "stopper" the ends of the theta pinch; in particular, simple and multiple mirrors, cusped ends, end-feeding by plasma (Marshall) guns and beams are being considered to reduce the particle and energy end losses. A less phenomenological approach to the question of ion/electron equilibration must also be developed. These refinements in plasma physics models as well as more profound engineering of the LTPHR concept is warranted by the preliminary but encouraging results presented herein.

Aside from the details of the plasma system being considered to drive a fusion/fission reactor, the goal of a pure fuel producer (i.e. no intrinsic energy production) does not seem realistic for the blanket systems considered herein. For the ^{233}U-enriched blankets (Cases A-D, Table VII), the ratio $E/E^* \sim 10$ shows that for every unit of energy delivered to the fission burner, 10 units will be generated by the hybrid reactor; this particular hybrid system will indeed be a significant energy generator in its own right. The un-enriched blankets (Cases E and F, Table VII) have E/E^* ratios of ~ 0.3. Clearly, rejection of $\sim 30\%$ of the potential energy of bred fuel is unwise, and although not a <u>net</u> energy producer, a substantial power conversion facility can be anticipated for this "pure breeder".

These conclusions are based on the specific blanket design considered by this study. A theoretical lower limit to E/E^*, however, can be postulated by an idealized blanket which breeds tritium by the neutron-conservative $> {}^{7}Li(n,n'\alpha)T$

reaction. The secondary neutron is assumed absorbed by ^{232}Th to yield ^{233}U, and for this idealized case both [BR] and [CV] are unity. Taking 14.1 MeV as the fusion neutron energy, subtracting 2.5 MeV for the endothermic ^{7}Li reaction, and adding ~ 6 MeV for the binding and decay energy released by the ^{232}Th absorption gives 17.6 MeV/n. To this must be added the 3.5 MeV fusion alpha particle and approximately $1/\alpha \sim 5.0$ MeV/n associated with plasma internal energy, both of which can be thermally converted. Hence, a total of $E = 26.1$ MeV/n is deposited into the blanket for every atom of bred ^{233}U. Taking 200 MeV/fission and the capture-to-fission ratio equal to 0.15, $E^{*} = 175$ MeV/n for this idealized, "pure breeder"; the ratio E/E^{*}, therefore, equals ~ 0.15 as a theoretically minimum value. For instance, if this "pure breeder" produced 1000 kg-^{233}U/y, which was subsequently burned in a fission reactor at 40% thermal efficiency (1.01 MWe y/kg), the idealized hybrid reactor would produce ~ 150 MWe. Clearly, substantial intrinsic production of electrical energy can be anticipated for "pure breeder", hybrid reactors.

6. REFERENCES

1. L. M. Lidsky, "Fission-Fusion Symbiosis: General Considerations and a Specific Example", BNES Nuclear Fusion Reactor Conference, Culham Laboratory, September (1969).

2. E. L. Draper, Jr., S. J. Gage, "The Fusion-Fission Breeder: Its Potential in a Fuel-Starved Thermal Reactor Economy", Technology of Controlled Thermonuclear Fusion Experiments and the Engineering Aspects of Fusion Reactors, Austin, Texas, November 20-22, 1972. Proc. publ. April 1974.

3. B. R. Leonard, Jr., W. C. Wolkenhauer, "Fusion-Fission Hybrids: A Substantial Thermal Fission Lattice For a D-T Fusion Reactor", ibid.

4. J. D. Lee, "Neutronics of Sub-critical Fast Fission Blankets for D-T Fusion Reactors", Proc. 7th Intersociety Energy Conversion Engineering Conference, San Diego, California, September 25-29, 1972. (Also USAEC Report UCRL-73952).

5. B. R. Leonard, Jr., "A Review of Fusion-Fission (Hybrid) Concepts", Nucl. Tech., 20, 161 (1973).

6. G. W. Brown, L. M. Lidsky, "A Blanket for a Breakeven Theta-Pinch Power Plant", First Topical Meeting on the Technology of Controlled Nuclear Fusion, April 16-18, 1974. San Diego, California.

7. L. M. Lidsky, "Optimal Use of Fusion Neutrons", ibid.

8. L. D. Hansborough, R. W. Werner, "A Modular Fission-Fusion Blanket", ibid.

9. J. D. Lee, "Neutronic Analysis of a 2500-MWt Fast Fission Natural Uranium Blanket for a D T Fusion Reactor", ibid.

10. W. C. Wolkenhauer, B. R. Leonard, Jr., R. W. Moir, R. W. Werner, "Conceptual Design of a Fusion-Fission Hybrid Reactor Based on a Mirror Fusion Reactor With a Sub-critical Gas-Cooled Fission Blanket", ibid.

11. T. A. Parish, E. L. Draper, "Neutronic and Photonic Analysis of Fusion Reactor Blankets Containing Natural Uranium", ibid.

12. R. C. Haight, J. D. Lee, "Calculations of a Fast Fission Blanket for D-T Fusion Reactors With Two Evaluated Data Libraries," ibid.

13. G. E. Bosler, D. J. Dudziak, W. R. Ellis, Jr., "A Preliminary Appraisal of a Fusion/Fission (Hybrid) Reactor Based on the Linear Theta Pinch", 9th Intersociety Energy Conversion Engineering Conference, August 26-30, 1974, San Francisco, California (also USAEC Report LA-UR-74-805).

14. Proc. IAEA Symposium on Advanced and High-Temperature Gas-Cooled Reactors, Jülich, 21-25 October 1968.

15. W. E. Quinn, personal communication, 1974.

16. F. L. Ribe (ed.), "Proposed Experiments on Heating, Staging, and Stabilization of Theta Pinches", USAEC Rept. LA-5026-P, Feb. 1973.

17. F. L. Ribe, R. L. Gribble, USAEC Rept. LA-4194-MS.

18. W. R. Ellis, "Scaling Laws for the Linear Theta Pinch, II; Circulating Power in High-Field Reactors", USAEC Rept. LA-5499-MS, Feb. 1974.

19. J. P. Freidberg, personal communication, 1974.

20. K. D. Lathrop, "DTF-IV, A Fortran-IV Program for Solving the Multi-Group Transport Equation with Anistropic Scattering", USAEC Rept. LA-3373 (1965).

21. J. D. Lee, "Neutronics of Sub-critical Fast Fission Blankets for DT Fusion Reactors" USAEC Rept. UCRL-73952,June 1972.

22. T. A. Oliphant, "A Mixed Snowplow Bounce Model for Shock Heating in a Staged Theta Pinch", Nucl. Fus., 14, 377 (1974) (Also, LA-UR-73-1689, 1973).

23. R. A. Krakowski, F. L. Ribe, T. A. Coultas, A. J. Hatch, "An Engineering Design Study of a Reference Theta-Pinch Reactor (RTPR)" USAEC Rept. LA-5336/ANL-8019, (1973).

24. J. P. Freidberg, R. L. Morse, F. L. Ribe, "Staged Theta Pinches With Implosion Heating", Tech. of Cont. Therm. Fusion Exp. and Eng. Aspects of Fusion Reactors, p. 812, Nov. 20-22, 1972, Austin, Texas.

25. L. Spitzer, Jr., Physics of Fully Ionized Gases, Interscience Publ., Inc. N.Y. (1956).

S. Glasstone, "Principles of Nuclear Reactor Engineering," 1st Ed. pp. 373-374, D. Van Nostrand Company, Inc., New York, July 1955.

E. D. Arnold, "Radiation Limitation on Recycle of Power Reactor Fuels", Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958, Vol. 13, p. 237, United Nations, New York, 1958.

S. Yiftah, D. Okrent, P. A. Moldauer, "Fast Reactor Cross Sections", Pergamon Press, New York, (1960).

M. K. Drake and Donald J. Dudziak (eds.), USAEC report ENDF-102, Vols. I and II (1970, 1971).

I. Langner, J. J. Schmidt, D. Wall, "Tables of Evaluated Neutron Cross Sections for Fast Reactor Materials", KFK-750, (Jan. 1968).

S. Yiftah and D. Okrent, "Some Physics Calculations on the Performance of Large Fast Breeder Power Reactors", USAEC Report ANL-6212, Argonne National Laboratory, (December 1960).

S. Yiftah, D. Okrent, and P. A. Moldauer, "Fast Reactor Cross Sections", pp. 29-34, Pergamon Press, New York, (1960).

D. T. Goldman and J. R. Stehn , "Chart of the Nuclides" 6th Ed., General Electric Company, revised December 1961.

Yevick, John G. (ed.) Fast Reactor Technology: Plant Design, MIT Press, 1966.

D. Okrent and W. Loewenstein, "The Physics of Fast Power Reactors", Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958, Vol. 12, p. 16, United Nations, New York (1958).

D. J. Hughes, B. A. Magurno, and M. K. Brussel "Neutron Cross Sections", USAEC Report ANL-325 (2nd Ed.) (Suppl. 1), Brookhaven National Laboratory, Jan. 1, 1960.

37. J. S. Fraser, C. R. J. Hoffman, P. R. Tunnicliffe, "The Role of Electrically Produced Neutrons in Nuclear Power Generation". Paper presented to the Summer Study Group on Practical Appl. of Accelerators, Los Alamos, N. M., Aug. 27-31, 1973, also, AECL-4658.

38. M. W. Rosenthal, P. P. Kasten, R. B. Briggs, "Molten-Salt Reactors-History, Status, and Potential", Nucl. Appl. and Tech. 8, 107, (1970).

TABLE I. NOTATION FOR FIG. 1 AND TERMS COMMONLY USED IN LTPHR ENERGY BALANCE

P_c	Recirculating power (MWe/m)
P_B	Capacitive power (MWe/m) ~ $(\frac{1}{2}CV_o^2)/\tau$
P_I	Power actually absorbed by plasma (MWt/m)
P_{TH}	Recoverable thermal power from LTPHR (MWt/m)
P^*_{TH}	Recoverable thermal power from fission burner (MWt/m)
P_E	Gross electrical power from LTPHR (MWe/m)
P^*_E	Gross electrical power from fission burner (MWe/m)
$\langle P_E \rangle$	Net electrical power from LTPHR/fission burner (MWe/m)
η_{TH}	Thermal conversion efficiency for LTPHR
η^*_{TH}	Thermal conversion efficiency for fission burner
η_I	Fraction of P_B which is absorbed by plasma
η_B	Fraction of P_B which is recoverable directly = $1 - \eta_I - f_{BLK} - f_L f_{EX}$
η_H	Net heating efficiency = $\eta_I/(1 - \eta_B)$
f_{BLK}	Fraction of P_B which is recoverable by thermal cycle
f_{EX}	Fraction of P_B which does not enter into compression field
f_L	Fraction of f_{EX} P_B which is irrecoverably lost
Q_p	Physics Q-value = $(P_{TH} + P^*_{TH})/P_I$
Q_E	Engineering Q-value = $(P_E + P^*_E)/P_c$
ε	Recirculating power fraction = $1/Q_E$
E	Energy deposited into LTPHR blanket per fusion neutron (MeV/n)
E^*	Potential energy per fusion neutron of converted ^{232}Th (MeV/n)
α	Linear fusion neutron flux per unit of P_I (n/MeV)

TABLE I. (Cont)

α^*	Linear fusion neutron flux per unit of P_B (n/MeV) $= \eta_I\alpha$
[CV]	Net ^{233}U created per fusion neutron
I_{23}	^{233}U inventory in LTPHR (kg/m)
I^*_{23}	^{233}U inventory in fussion burner (kg/m)
$[SI]_T$	Total specific inventory (kg/MWe) $= (I_{23} + I^*_{23})/(P_E + P^*_E)$
[DT]	^{233}U doubling time (y)

TABLE II. SUMMARY OF NUMERICAL THERMONUCLEAR BURN CALCULATIONS

Run	C(mF/m)	V_o(kV)	W_c(MJ/m) $= 1/2\ CV_o^2$	N	$\tau(\mu s)$	B(T) (max)	T_i(keV) (max)	α*(n/MeV) $\times 10^4$
1	180	10	9.0	1	284	24.69	16.62	2.24
2	180	10	9.0	2	541	26.22	15.72	4.21
3	180	10	9.0	4	1054	26.65	14.43	7.38
4	180	10	9.0	8	2107	26.76	13.95	13.54
5	180	20	36.0	1	291	49.38	14.73	3.51
6	180	20	36.0	2	541	52.45	13.21	7.23
7	180	20	36.0	4	1081	53.34	12.17	14.77
8	180	20	36.0	8	2107	53.54	12.21	29.76
9	360	10	18.0	1	402	34.92	15.6	3.61
10	360	15	40.5	1	412	52.36	13.26	4.93
11	360	20	72.0	1	412	69.82	11.05	6.49
12	180	15	20.3	1	284	37.03	13.93	2.89
13	180	20	36.0	1	291	49.37	14.73	3.51
14	90	10	4.5	1	201	17.47	13.23	1.37
15	90	10	4.5	2	382	18.54	12.71	2.55
16	90	10	4.5	4	745	18.83	17.09	4.47
17	90	10	4.5	8	1862	18.92	15.30	9.08
18	90	15	10.1	1	202	26.19	13.86	1.80
19	90	15	10.1	2	397	26.84	16.81	3.18
20	90	15	10.1	4	794	27.02	15.10	5.44
21	90	15	10.1	8	1568	27.06	14.10	9.68
22	90	15	10.1	10	1960	27.10	11.49	11.91
23	90	15	10.1	12	2352	27.07	11.49	14.24
24	90	15	10.1	14	2744	27.07	11.49	16.58
25	90	15	10.1	1	251	26.18	13.87	2.64
26	90	15	10.1	1	302	26.18	13.88	3.42
27	90	15	10.1	1	404	26.18	13.90	4.88
28	90	15	10.1	1	703	26.15	13.95	8.85
29	90	15	10.1	1	810	25.60	14.09	9.73
30	90	15	10.1	1	1002	26.16	13.89	12.72
31	90	15	10.1	1	1402	26.18	13.88	17.93
32	90	15	10.1	1	1600	26.18	13.70	20.53
33	90	15	10.1	1	1703	26.17	13.87	21.85
34	90	15	10.1	1	2003	26.18	13.70	25.77
35	90	15	10.1	1	2202	26.15	13.80	28.27
36	90	15	10.1	1	203	26.19	14.24	2.37
37	90	15	10.1	2	397	26.85	16.19	4.11
38	90	15	10.1	4	784	27.02	14.55	7.00
39	90	15	10.1	8	1568	27.06	13.75	12.48
40	90	15	10.1	10	1960	27.06	13.69	15.31
41	90	15	10.1	12	2352	27.07	13.59	18.30
42	90	15	10.1	14	2744	27.07	13.62	21.31
43	90	15	10.1	1	251	26.20	14.18	3.37
44	90	15	10.1	1	300	26.20	14.20	4.37
45	90	15	10.1	1	399	26.19	14.20	6.23
46	90	15	10.1	1	700	26.19	14.20	11.40
47	90	15	10.1	1	800	26.19	14.20	13.03

TABLE II. SUMMARY OF NUMERICAL THERMONUCLEAR BURN CALCULATIONS (continued)

Run	C (mF/m)	V_o (kV)	W_c (MJ/m) $= 1/2\ CV_o^2$	N	τ (μs)	B(T) (max)	T_i (keV) (max)	α^* (n/MeV) $\times 10^4$
48	90	15	10.1	1	1000	26.19	14.20	16.40
49	90	15	10.1	1	1400	26.19	14.20	23.00
50	90	15	10.1	1	1700	26.19	14.20	28.03
51	90	15	10.1	1	2000	26.19	14.20	33.04
52	90	15	10.1	1	2200	26.19	14.20	36.37

Runs 1-17 Vary C, V_o and N with $\tau_{FT} = 0$

Runs 18-24 Vary N with E_θ = 3 kV/cm and $\tau_{FT} = 0.0$

Runs 25-35 Vary τ_{FT} with E_θ = 3 kV/cm and N = 1

Runs 36-42 Vary N with E_θ = 4 kV/cm and $\tau_{FT} = 0.0$

Runs 43-52 Vary τ_{FT} with E_θ = 4 kV/cm and N = 1

TABLE III.

IDENTIFICATION OF ELECTRICAL, BURN, AND ENERGY PARAMETERS FOR THE FOUR LTPHR CASES SELECTED FOR NUMERICAL EVALUATION ($\tau_{1/2}$ = 202 μs)

Case	E_θ(kV/cm)	τ_{FT}(μs)	N	$\alpha^*(\ell = 1000$ m) (n/MeV)
A	3	0.0	variable	11.5×10^{-4}
B	4	0.0	variable	14.9×10^{-4}
C	3	variable	1	24.4×10^{-4}
D	4	variable	1	31.4×10^{-4}

TABLE IV. MULTIGROUP STRUCTURE

Group	E_{max}		Group	E_{max}	
1	15.00	MeV	11	302.0	keV
2	13.50	MeV	12	183.2	keV
3	12.21	MeV	13	1.234	keV
4	10.00	MeV	14	22.60	eV
5	6.065	MeV	15	3.059	eV
6	3.679	MeV	16	1.125	eV
7	2.231	MeV	17	0.414	eV
8	1.353	MeV	18	0.100	eV
9	820.8	keV	19	0.040	eV
10	497.9	keV		0.00758	eV

TABLE V. PRELIMINARY NEUTRONIC RESULTS USING THE SIMPLE BLANKET MODEL DEPICTED IN FIG. 15

Case	I_{23} (kg/m)	Be or BeO	Be Thickness (cm)	Coil Thickness (cm)	Fe in Seed	E (MeV/n)	E^* (MeV/n)	$E + E^*$ (MeV/n)	[CV]
1	4.08	Be	5.0	3.0	No	30.3	99.8	130.1	0.571
2	16.3	Be	5.0	3.0	No	66.6	84.3	150.9	0.482
3	17.4	Be	5.0	5.0	No	62.6	75.7	138.3	0.432
4	4.74	Be	10.0	3.0	No	30.7	96.2	126.9	0.550
5	4.74	BeO	10.0	3.0	No	29.2	88.3	117.5	0.504
6	10.2	BeO	10.0	3.0	No	32.8	67.8	100.6	0.387
7	8.69	BeO	5.0	5.0	No	38.8	86.8	125.6	0.496
8	10.2	BeO	10.0	5.0	No	36.1	71.4	107.6	0.408
9	9.80	BeO	10.0	5.0	Yes	30.2	22.3	52.4	0.128
10 (a)	13.9	BeO	10.0	5.0	Yes	26.2	25.1	51.2	0.143
11	4.78	BeO	10.0	5.0	Yes	25.1	26.1	51.2	0.149
12	15.2	Be	5.0	1.0	No	66.4	87.3	153.7	0.499
13	16.3	Th	5.0	3.0	No	65.9	172.0	237.9	0.983

(a) For case 10 the ^{233}U is enriched throughout the seed region.

TABLE VI

REFERENCE BLANKET CONFIGURATION AND PERFORMANCE

Region	Inner Radius (m)	Outer Radius (m)	Material
Plasma Chamber	0	0.100	Vacuum
Coil	0.100	0.115	Mo, 30 v/o He
Tritium Breeding	0.115	0.415	^{7}Li(99 a/o), 25 v/o He
Inner Enriched Seed	0.415	0.665	10 a/0 ^{233}U, C/HM = 214, 25 v/o
Outer Enriched Seed	0.665	0.865	4 a/o ^{233}U, C/HM = 214, 25 v/o H
Fertile Seed	0.865	1.215	C/^{232}Th = 214, 25 v/o He

PERFORMANCE CHARACTERISTICS

[BR] = 1.35

[CV] = 0.233 (including leakage)

E = 413.5 MeV/n

E + E* = 454.4 MeV/n

^{232}Th (n, Nn) = 0.0087/DT neutron

I_{23} = 18.2 kg/m

$E/I_{23} = 3.64 \times 10^{-18}$ MJ/kg per n/m

$E + E^*/I_{23} = 4.00 \times 10^{-18}$ MJ/kg per n/m

R = 0.79 (kg/m y) [^{233}U production at 1 MW/m^2 wall loading]$^+$

$^+$1 MW/m^2 of 14.1-MeV neutrons is equivalent to $<J> = 4.44 \times 10^{17}$ (n/m^2 s)

TABLE VIII

SUMMARY OF OPERATING PARAMETERS FOR CASES A THROUGH D OPERATING WITH A BLANKET DESIGN DEPICTED ON TABLE VII ($C/^{233}U$ = 214, [BR] = 1.35

	CASE A (Simple LC Circuit, E_θ = 3 kV/cm)	CASE B (Simple LC Circuit, E_θ = 4 kV/cm)	CASE C (Stretched LC Circuit, E_θ = 3 kV/cm)	Case D (Stretched LC Circuit, E_θ = 4 kV/cm)	CASE E (Same as Case D but with Th (metal) breeder blanket)	CASE F (Same as Case D but with Th/C breeder blanket)
REACTOR PHYSICS PARAMETERS						
(a) Reactor length, ℓ(m)	1000	1000	1000	1000	1000	1000
(a) Recovery efficiency, η_B	0.95	0.95	0.95	0.95	0.95	0.95
(b) Capacitive energy store, W_c (MJ/m)	10.1	10.1	10.1	10.1	10.1	10.1
Number of turns on compression field, N	10	10	1	1	1	1
Confinement time, $\tau_{1/2} + \tau_{FT}$ (μs)	1900	1900	1900	1900	1900	1900
Maximum compression field, B(T)	27.1	27.1	25.3	26.2	26.2	26.2
Maximum plasma temperature, T_i (keV)	11.5	13.6	14.1	14.5	14.5	14.5
(c) Transfer efficiency, $1 - f_{EX}$	0.91	0.91	0.85	0.85	0.85	0.85
Neutron yield, α^*(n/MeV) x 10^4	11.5	14.9	24.4	31.4	31.4	31.4
BLANKET PARAMETERS						
Energy worth of fusion neutron, $E + E^*$ (MeV/n)	454	454	454	454	224	87
In situ energy worth of fusion neutron, E (MeV/n)	414	414	414	414	49	20
Ratio of intrinsic to bred-fuel energy E/E^*	10.4	10.4	10.4	10.4	0.28	0.30
Cycle time to maintain I_W = 1.0 MW/m^2, τ_c (s)	0.25	0.32	0.52	0.68	0.68	0.68
Fissile fuel inventory I_{23}, (kg/m)	18	18	18	18	0.0	0.0
Conversion ratio [CV]	0.233	0.233	0.233	0.233	1.0	0.38
P_E (MWe/m) ~ $2\pi b I_W (E/14.1)\eta_{TH}$, ($I_W$ = 1.0 MW/m^2, η_{TH} = 0.4)	7.38	7.38	7.38	7.38	0.9	0.4
Specific inventory [SI] = I_{23}/P_E(kg/MWe) (I_W = 1.0 MW/m^2, η_{TH} = 0.4)	2.44	2.44	2.44	2.44	0.0	0.0
Fissile breeding rate at I_W = 1.0 MW/m^2, R(kg/y m)	0.79	0.79	0.79	0.79	3.39	1.29
ENERGY BALANCE PARAMETERS						
Engineering Q value, $Q_E = (P_E + P_E^*)/P_c$	4.16	5.39	8.99	11.6	5.63	2.19
Neutron yield for breakeven (Q_E = 1), α^*(n/MeV) x 10^4	2.7	2.7	2.7	2.7	5.56	14.31
(d) LTPHR length for overall breakeven (Q_E = 1) ℓ(m)	194	120	150	110	210	580
Internal recirculation, P_c/P_E	0.244	0.203	0.122	0.0948	.81	2.0
Q_E value for $P_E = P_c$ ($Q_E \sim 1 + E^*/E$)	1.096	1.096	1.096	1.096	4.57	4.35
α^* value for $P_E = P_c$ (n/MeV) x 10^4	3.0	3.0	3.0	3.0	25.50	62.50
LTPHR length for internal breakeven ($P_E = P_c$) ℓ(m)	226	150	160	120	820	1980
PRELIMINARY COST ESTIMATES (e)						
Cost of capacitors and switches ($/kWe)	415.	302.	476.	467.	1073.	4175.
Cost of LTPHR blanket ($/kWe)	123.	122.	120.	119.	122.	133.
(f) Prorated cost of fission burner ($/kWe)	66.	61.	56.	55.	474.	709.
Total cost of capacitor/switches, blanket, fission burner ($/kWe)	604.	585.	652.	641.	1669.	5017.

(a) Assumed for purposes of estimations

(b) corresponding to C = 90 mF/m and V_o = 15 kV

(c) Most of energy $f_{EX} W_c$ stored reversibly in external inductances

(d) Based on the pessimistic assumption of uninhibited end loss

(e) Costs are expressed in terms of dollars per electrical kilowatt of salable power (P_E) (see Fig. 1) and are based on I_W = 1.0 MW/m^2.

(f) Based on 500 $/kWe for fission nuclear island.

8. FIGURES

Fig. 1. Schematic diagram of the LTPHR energy balance.

Fig. 2. Simplified circuit diagram of the LTPHR indicating major energy sinks

Fig. 3. Various compression field waveforms considered to drive the LTPHR or to test analytical expressions (see text for identity of curves).

Fig. 4. Schematic diagram of simple LC circuit and associated current waveform

Fig. 5. Dependence of neutron yield, α^*, on initial filling density, n_o for capacitance C = 180 mF/m and voltage V_o = 10 kV (computer runs 1-4 on Table II.)

Fig. 6. Time dependence of ion temperature, plasma compression ratio, fusion neutron flux, and integrated flux for a typical LTPHR burn using a simple LC circuit (computer run 1 on Table II).

Fig. 7. Comparison of α^* (n/MeV) computed numerically with the predictions of Eq. (12).

Fig. 8. Dependence of α^* (n/MeV) and ℓ(m) on $\tau = \tau_{1/2} + \tau_{FT}$ for C = 90 mF/m, V_o = 15 kV (W_B = 10.1 MJ/m), and E_θ = 3 kV/cm.

Fig. 9. Dependence of α^* (n/MeV) and ℓ(m) on $\tau = \tau_{1/2} + \tau_{FT}$ for C = 90 mF/m, V_o = 15 kV (W_B = 10.1 MJ/m) and E_θ = 4 kV/m.

Fig. 10. Detailed isotopic chain for the $^{232}Th/^{233}U$ fuel cycle (high-energy cross sections are given for ~ 14 MeV neutrons).

Fig. 11. Neutronic blanket model used in preliminary scoping calculations.

Fig. 12. Neutronic model used in blanket parameter studies (Figs. 13-18).

Fig. 13. Variation of blanket parameters with ^{233}U enrichment (Parameter study

Fig. 14. Radial dependence of [BR], [CV] and fission rate for the [BR] = 1.0 case depicted on Fig. 13.

Fig. 15. Variation of blanket parameters with thorium multiplier thickness (Parameter study 2).

Fig. A-9. Energy dependence of neutron yield per fission[28].

Fig. A-10. Energy dependence of neutron yeild per absorption, η[37], and typical neutron spectra for thermal fission, fast fission and fusion reactors.

a) Ref. 23

Fig. A-11. Neutronic model for LTPHR idealized breeder blanket.

Fig. A-12. Dependence of blanket parameters on thorium thickness for purely breeding LTPHR using metallic fuel.

Fig. A-13. Dependence of blanket parameters on thorium/carbon thickness for purely breeding LTPHR using HTGR fuel element technology.

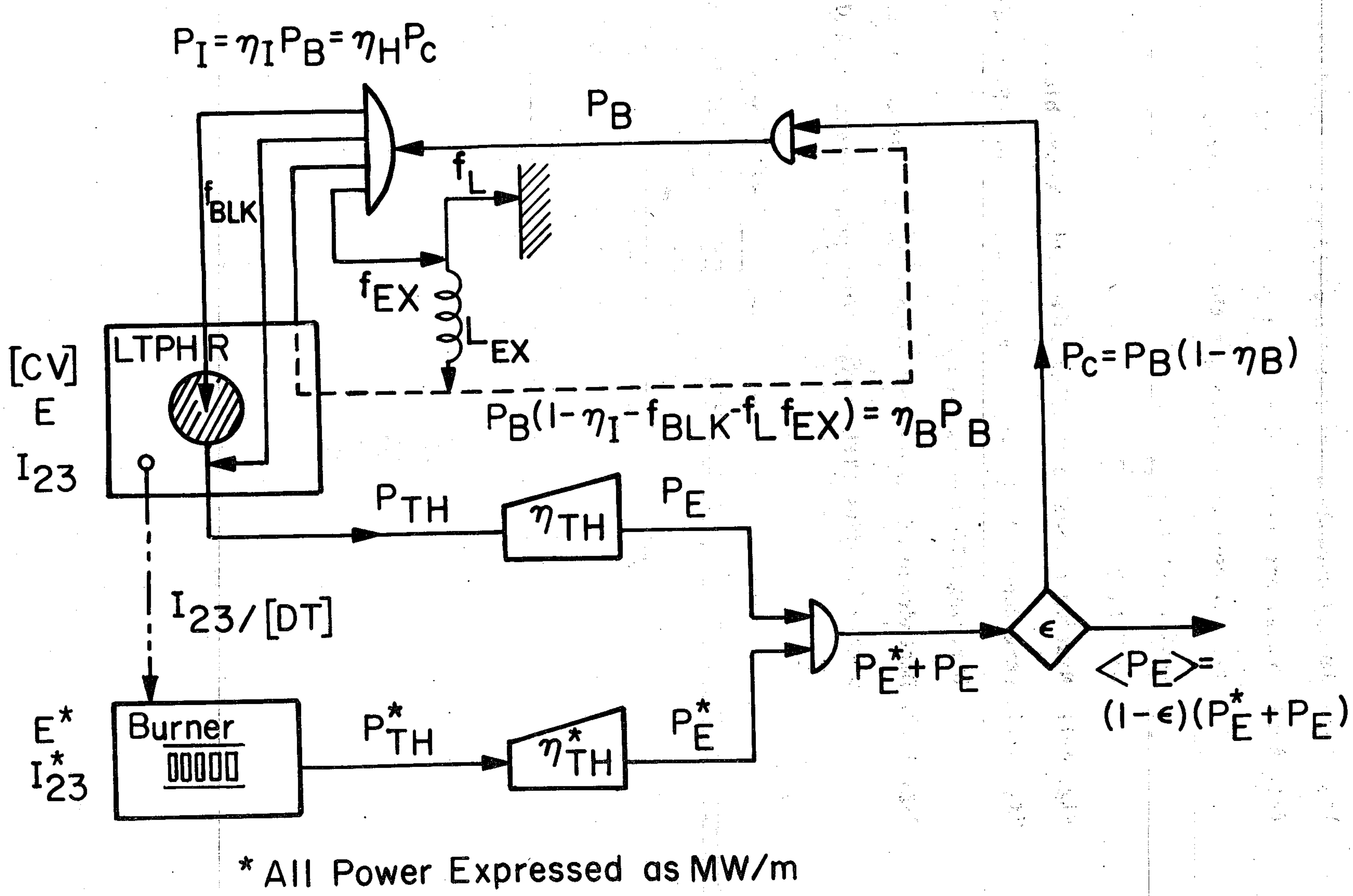


Fig. 1. Schematic diagram of the LTPHR energy balance.

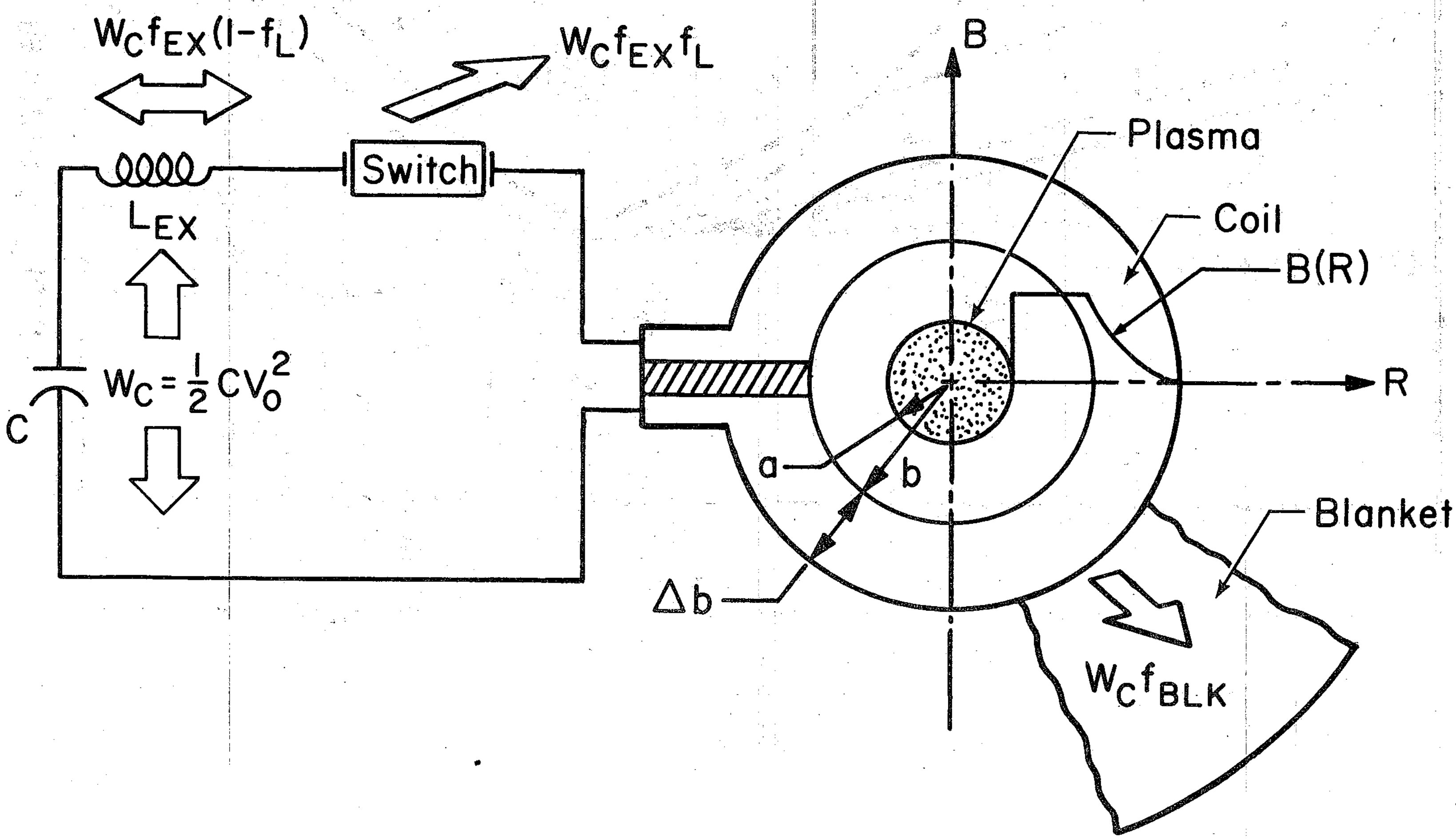


Fig. 2. Simplified circuit diagram of the LTPHR indicating major energy sinks.

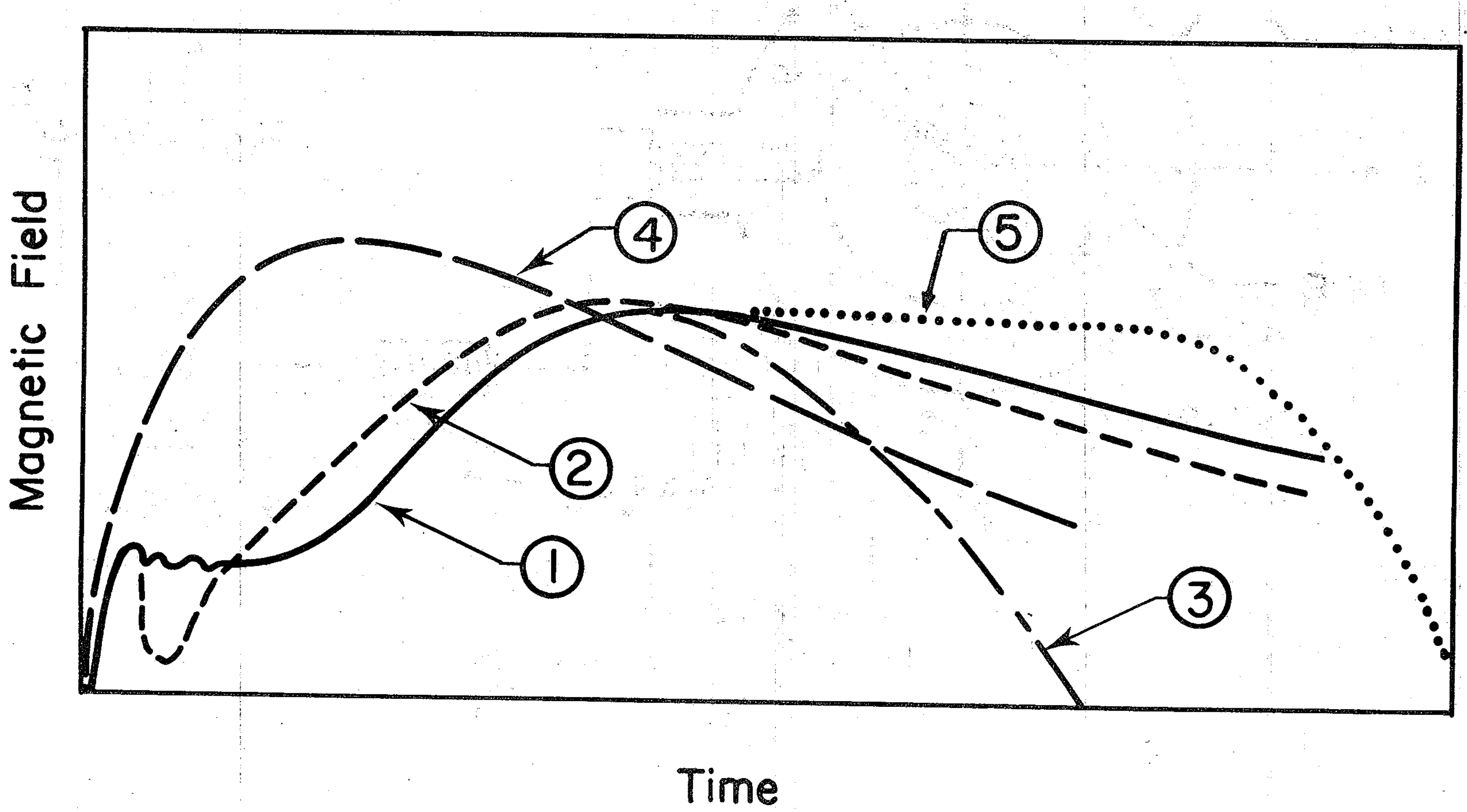
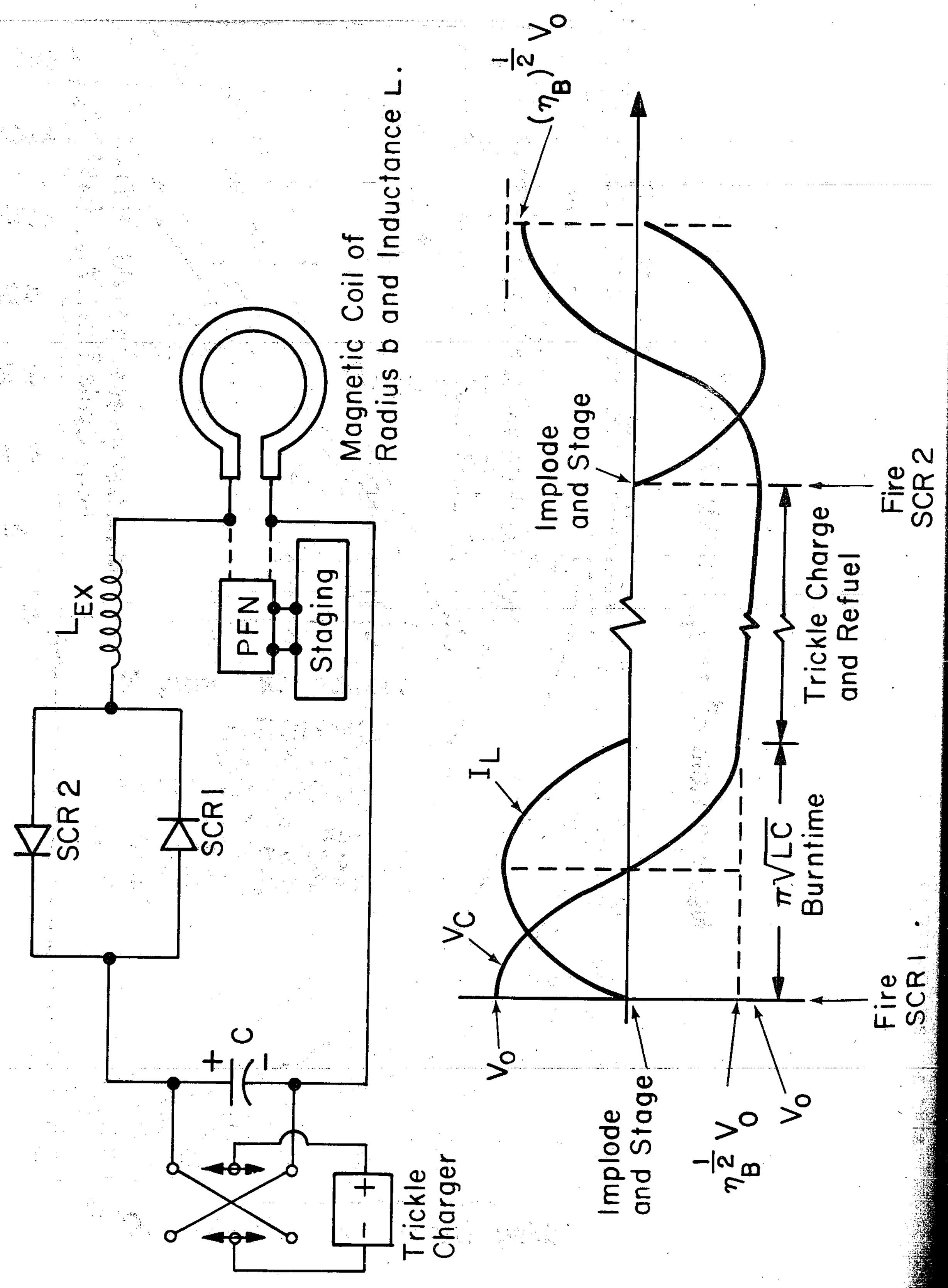


Fig. 3. Various compression field waveforms considered to drive the LTPHR or to test analytical expressions (see text for identity of curves).


Magnetic Coil of
Radius b and Inductance L.
L_{EX}
PFN
Staging
SCR 2
SCR I
+ C -
+ -
Trickle
Charger
$(\eta_B)^{\frac{1}{2}} V_0$
Implode
and Stage
Fire
SCR 2
Trickle Charge
and Refuel
I_L
V_C
V_0
$\pi\sqrt{LC}$
Burntime
Fire
SCR I
Implode
and Stage
$\eta_B^{\frac{1}{2}} V_0$
V_0

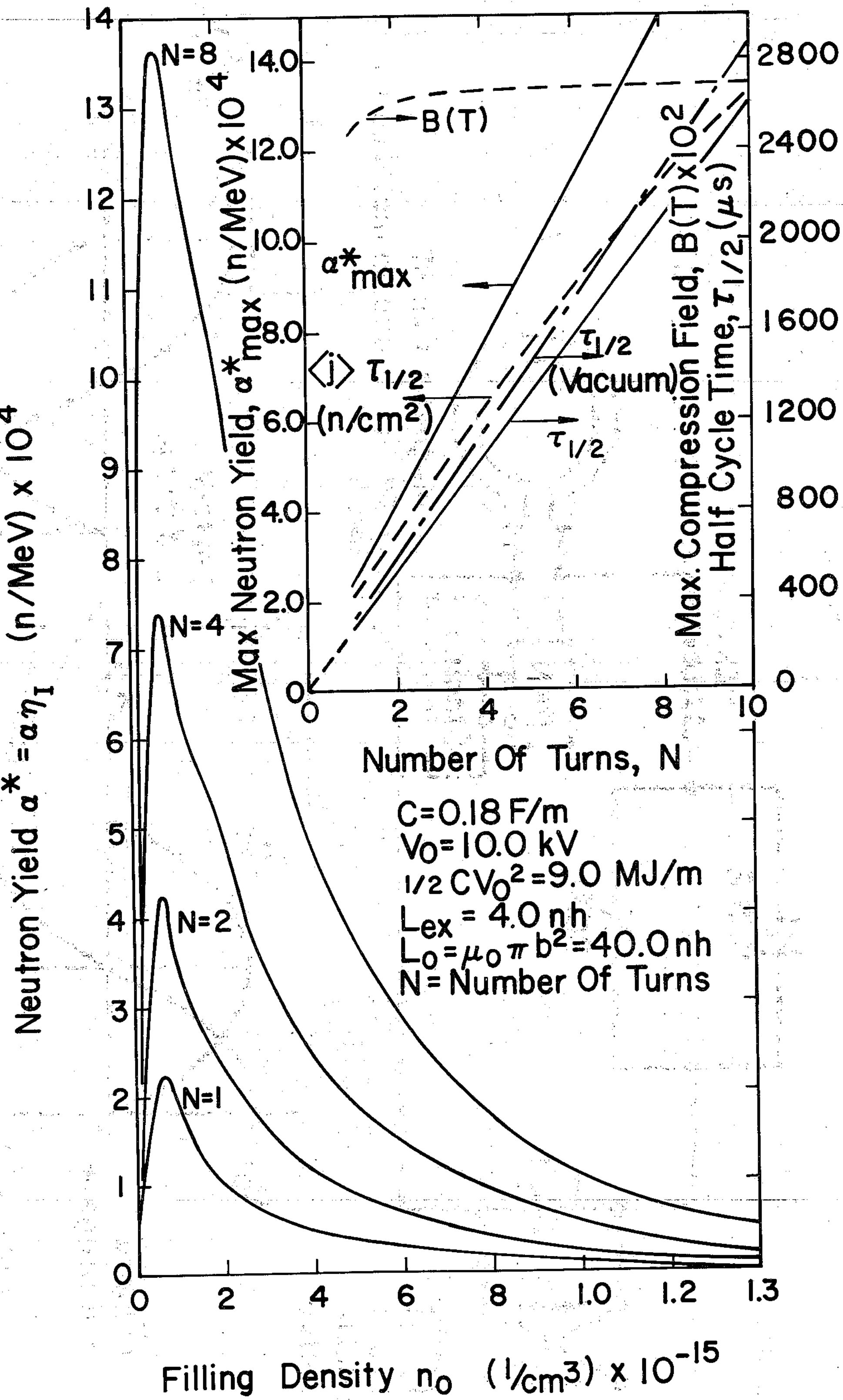


Fig. 5. Dependence of neutron yield, α^*, on initial filling density, n_o for capacitance C = 180 mF/m and voltage V_o = 10 kV (computer runs 1-4 on Table II.)

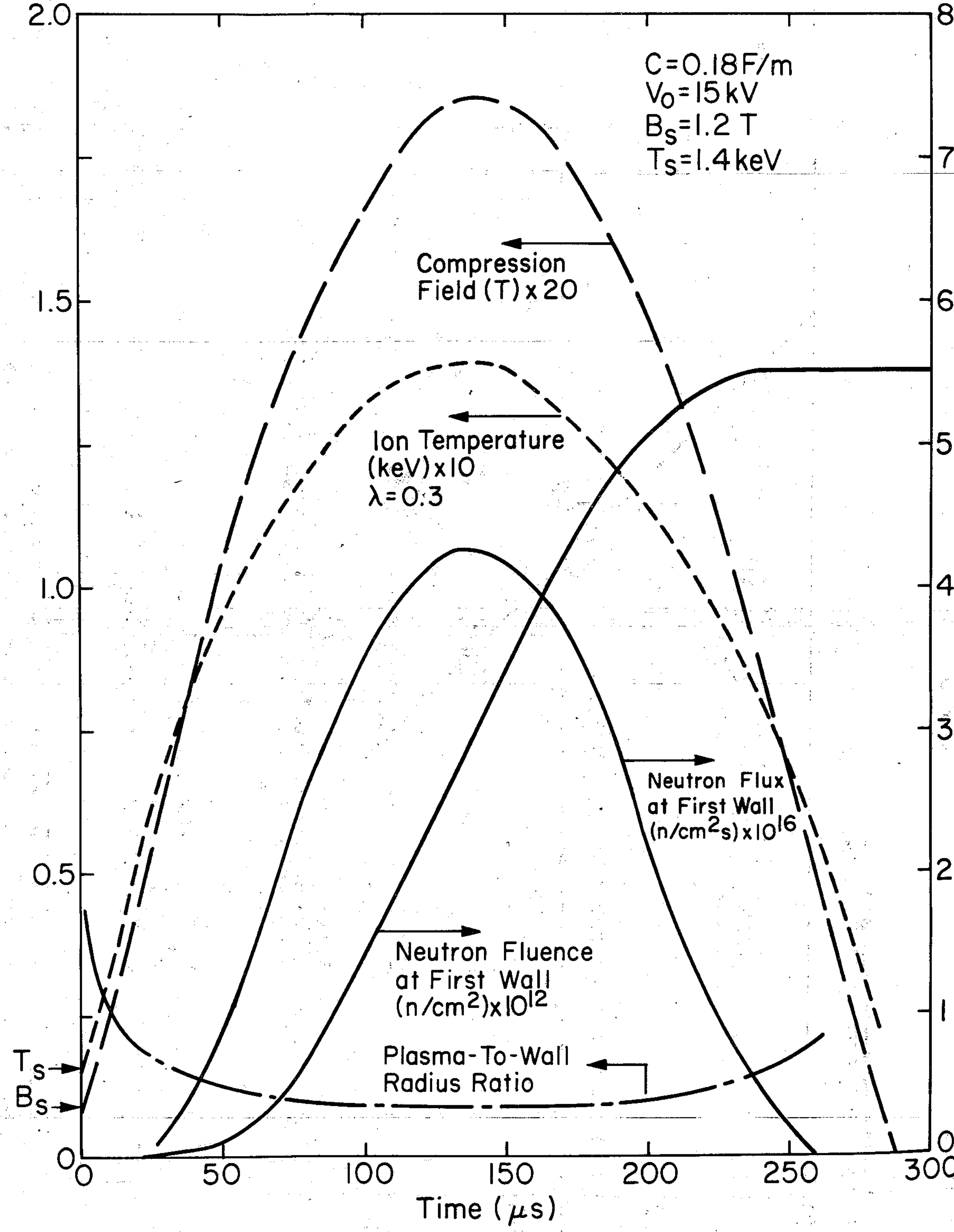


Fig. 6. Time dependence of ion temperature, plasma compression ratio, fusion neutron flux and integrated flux for a typical LTPHR burn using a simple LC circuit (computer run 1 on Table II).

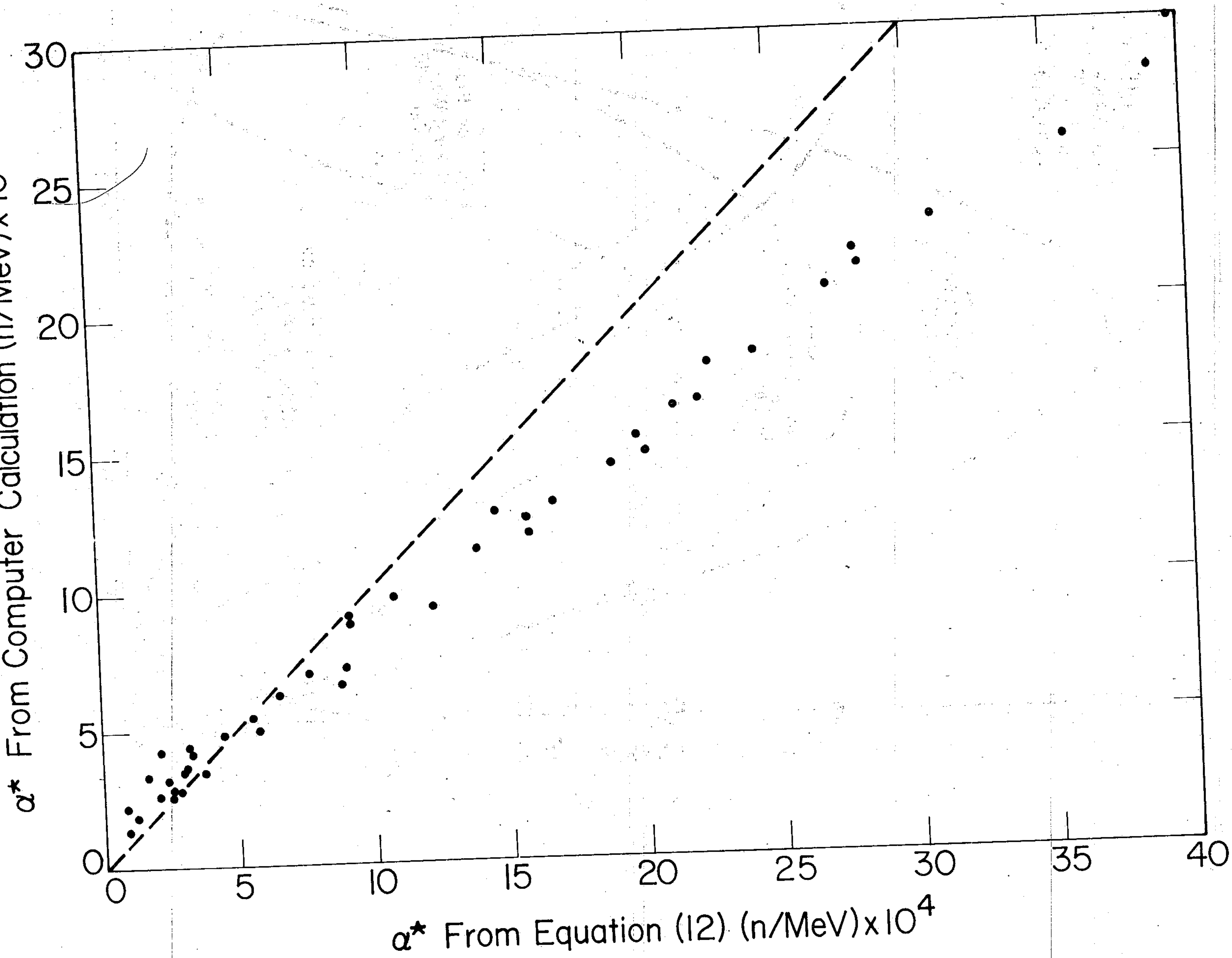


Fig. 7. Comparison of α^* (n/MeV) computed numerically with the predictions of Eq. (12).

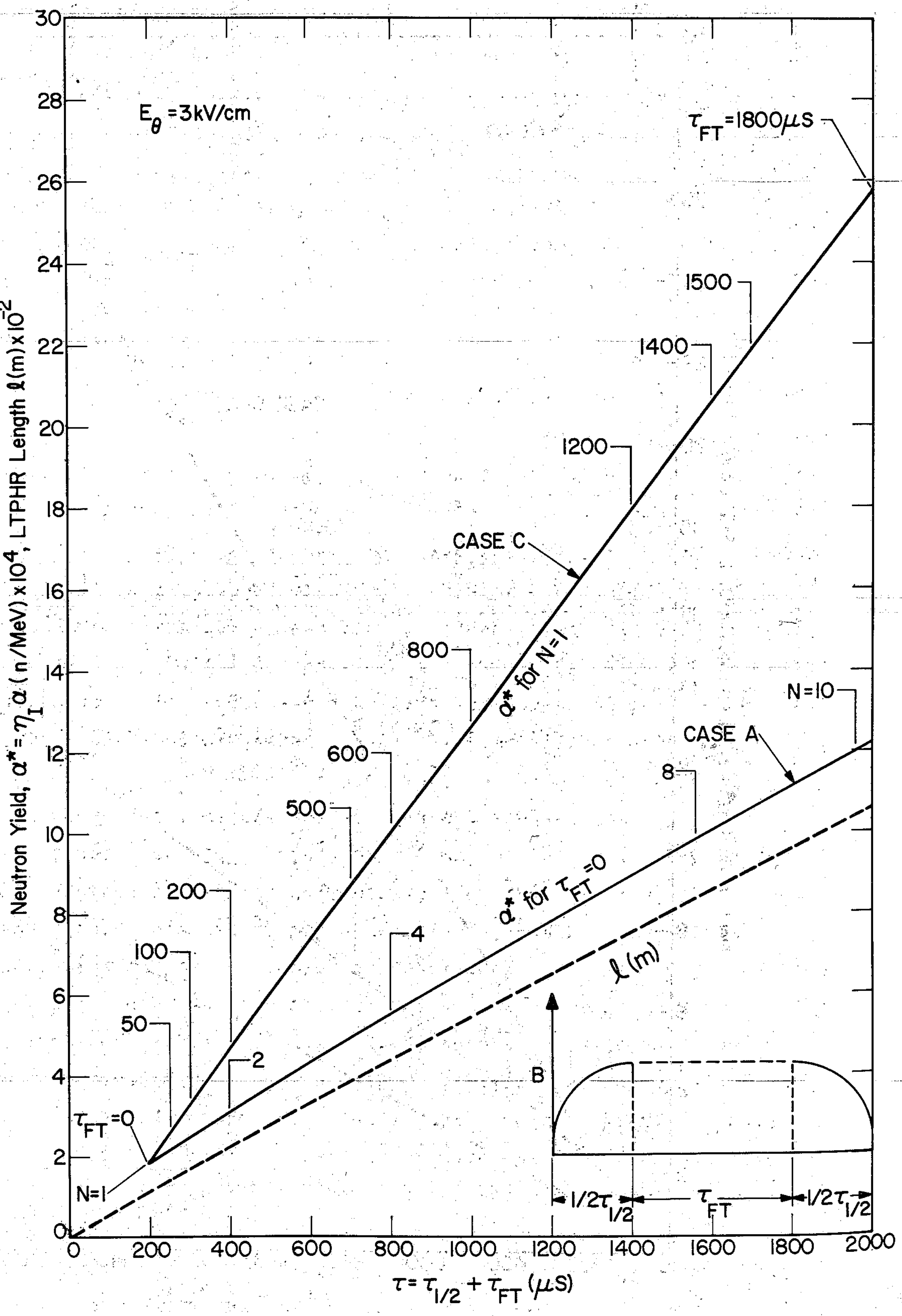


Fig. 8. Dependence of α^* (n/MeV) and ℓ(m) on $\tau = \tau_{1/2} + \tau_{FT}$ for C = 90 mF/m, V_o = 15 kV (W_B = 10.1 MJ/m), and E_θ = 3 kV/cm.

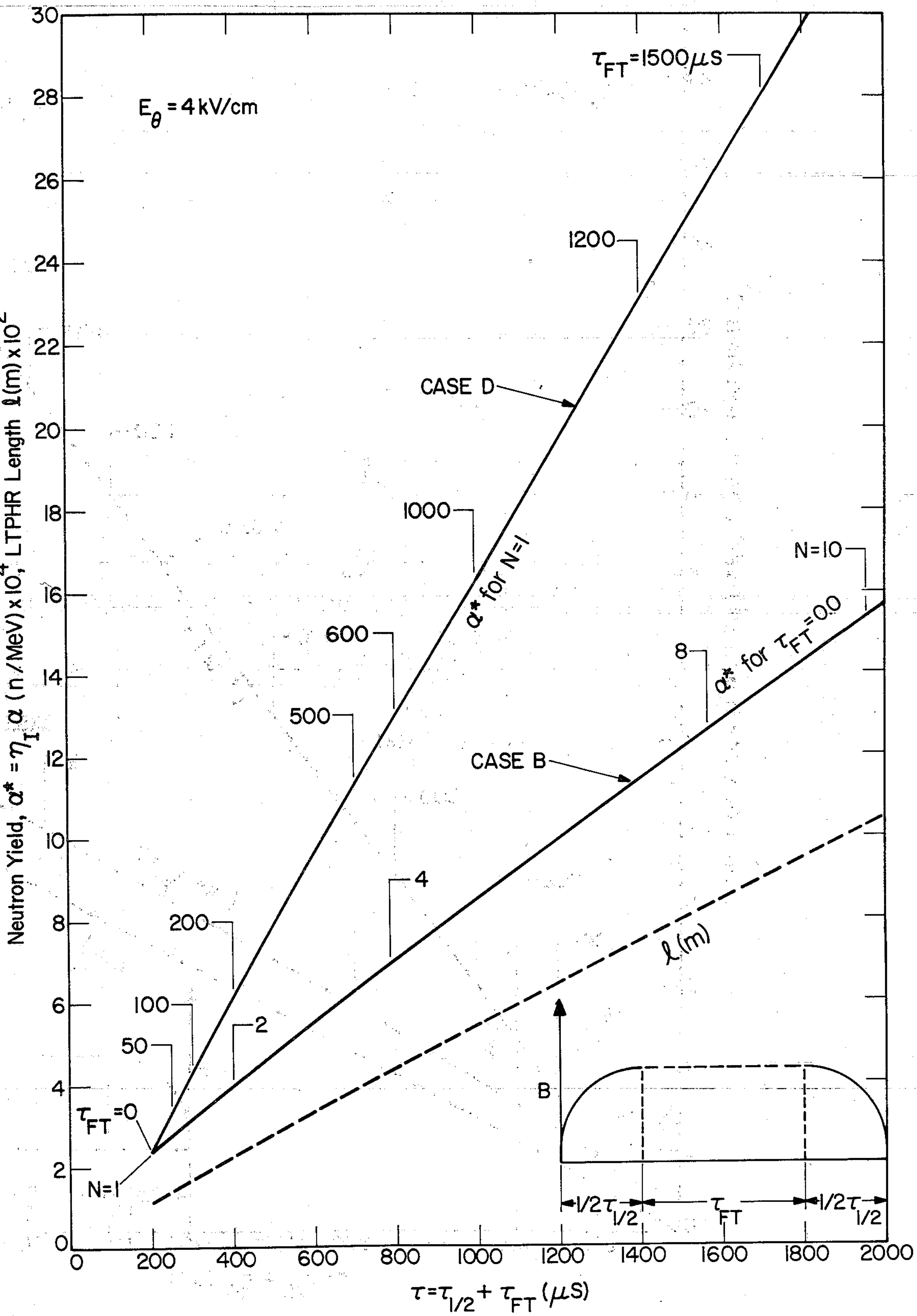


Fig. 9. Dependence of α^* (n/MeV) and ℓ(m) on $\tau = \tau_{1/2} + \tau_{FT}$ for C = 90 mF/m, V_o = 15 kV (W_B = 10.1 MJ/m) and E_θ = 4 kV/m.

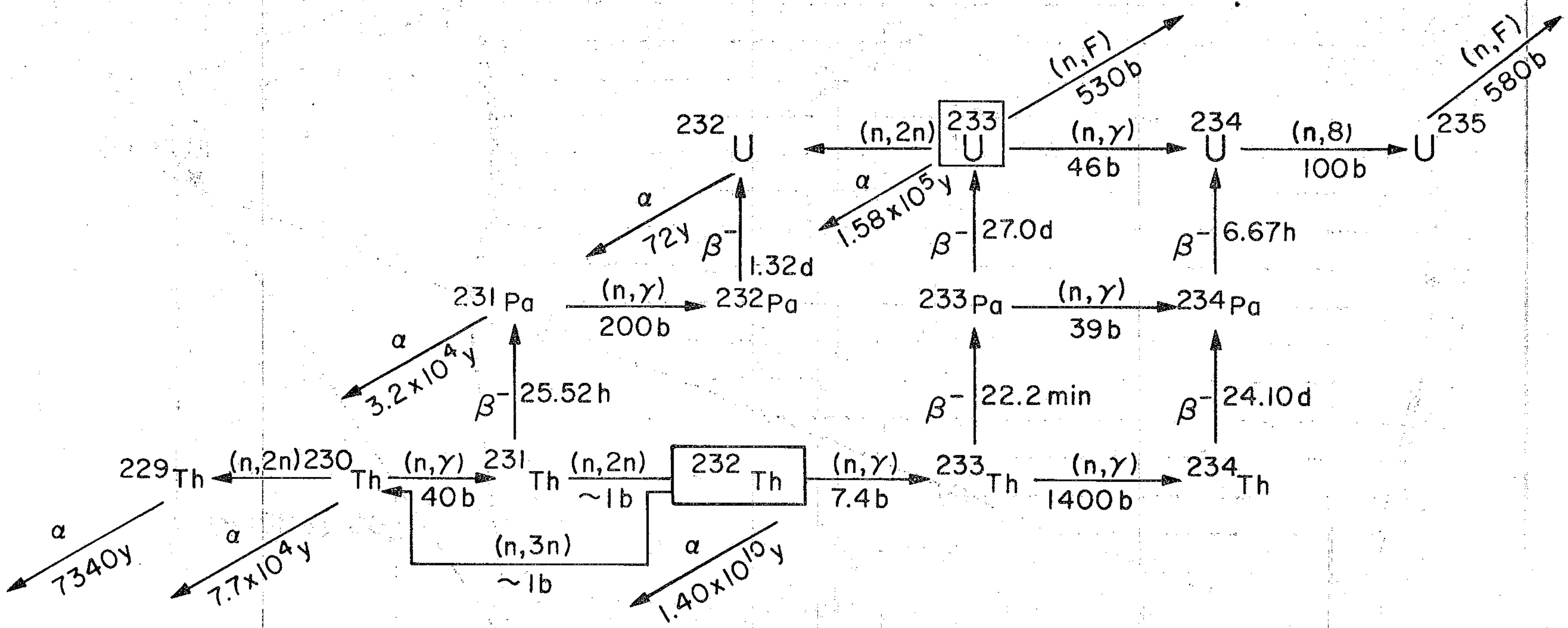


Fig. 10 Detailed isotopic chain for the ^{232}Th/^{233}U fuel cycle (high-energy cross sections are given for ~ 14 MeV neutrons).

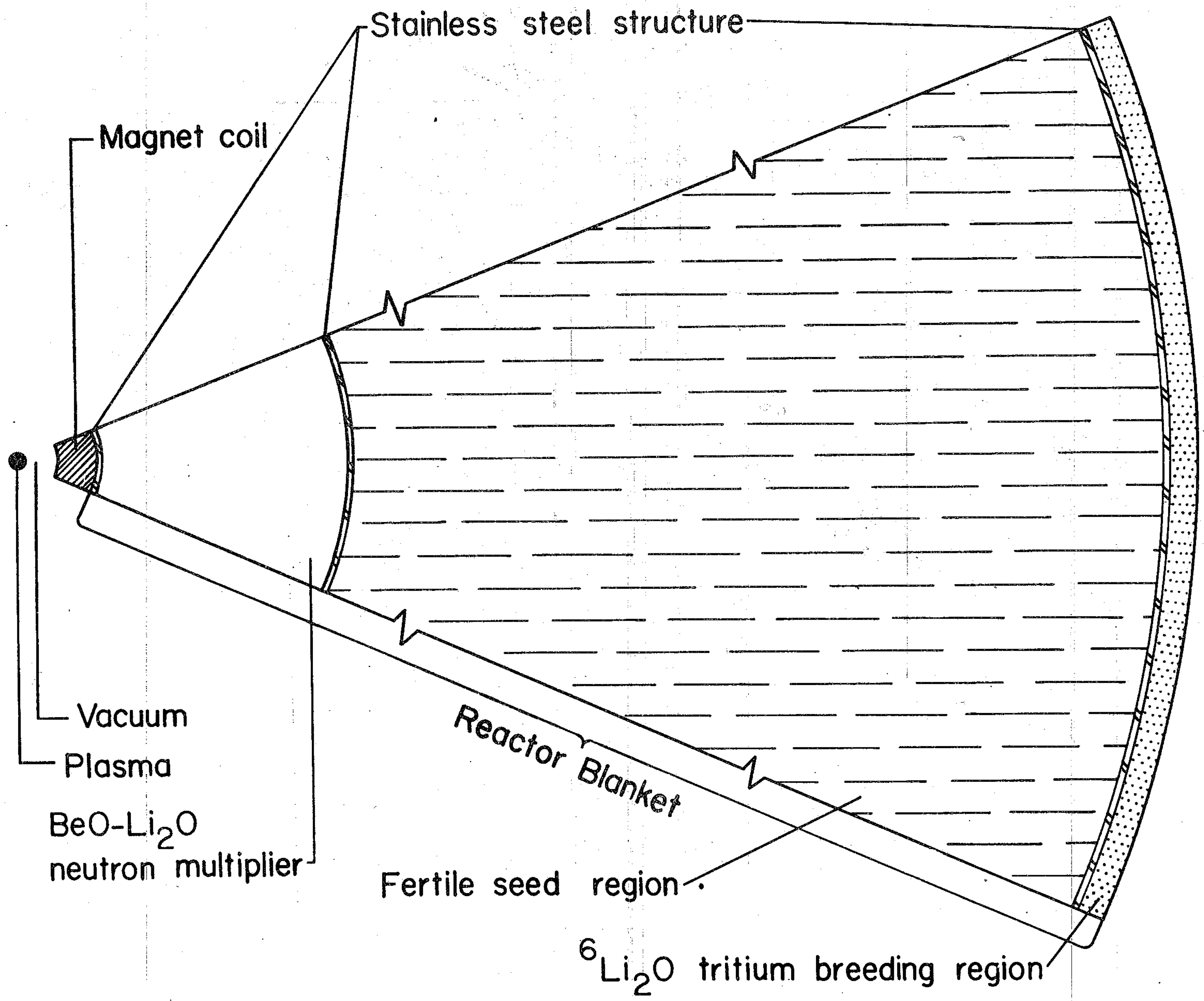


Fig. 11. Neutronic blanket model used in preliminary scoping calculations.

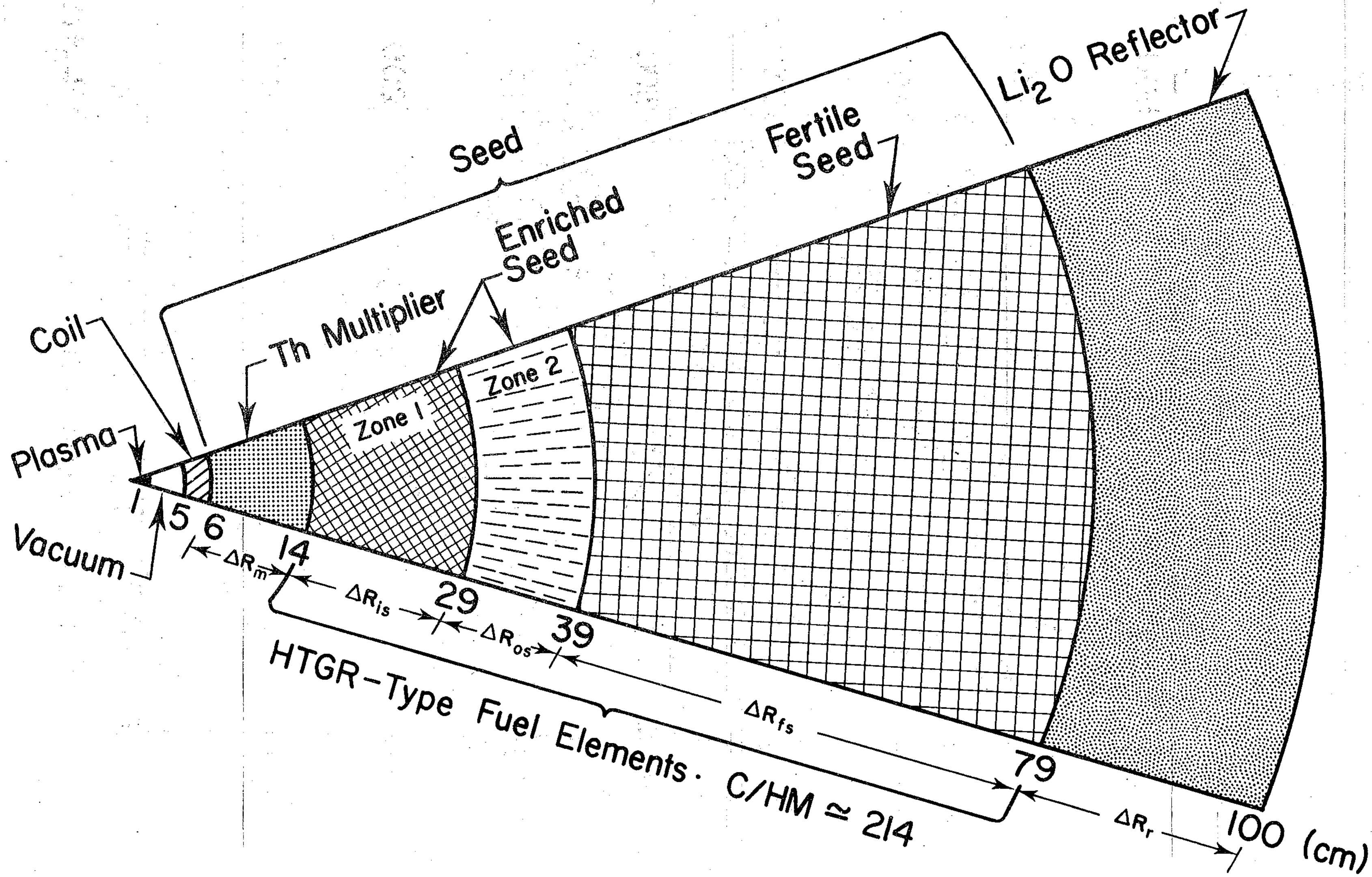


Fig. 12. Neutronic model used in blanket parameter studies (Figs. 13-18).

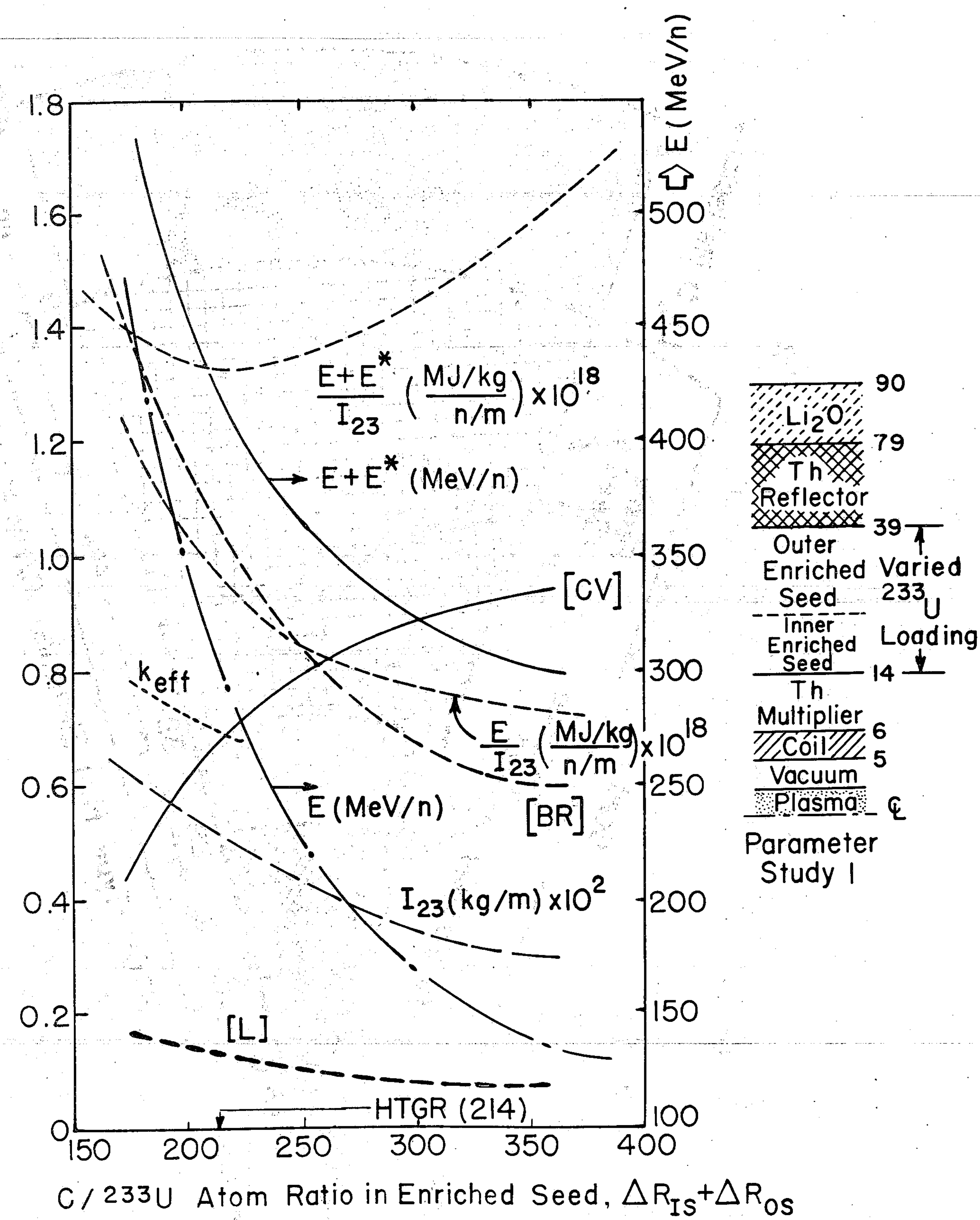


Fig. 13. Variation of blanket parameters with ^{233}U enrichment (Parameter study 1).

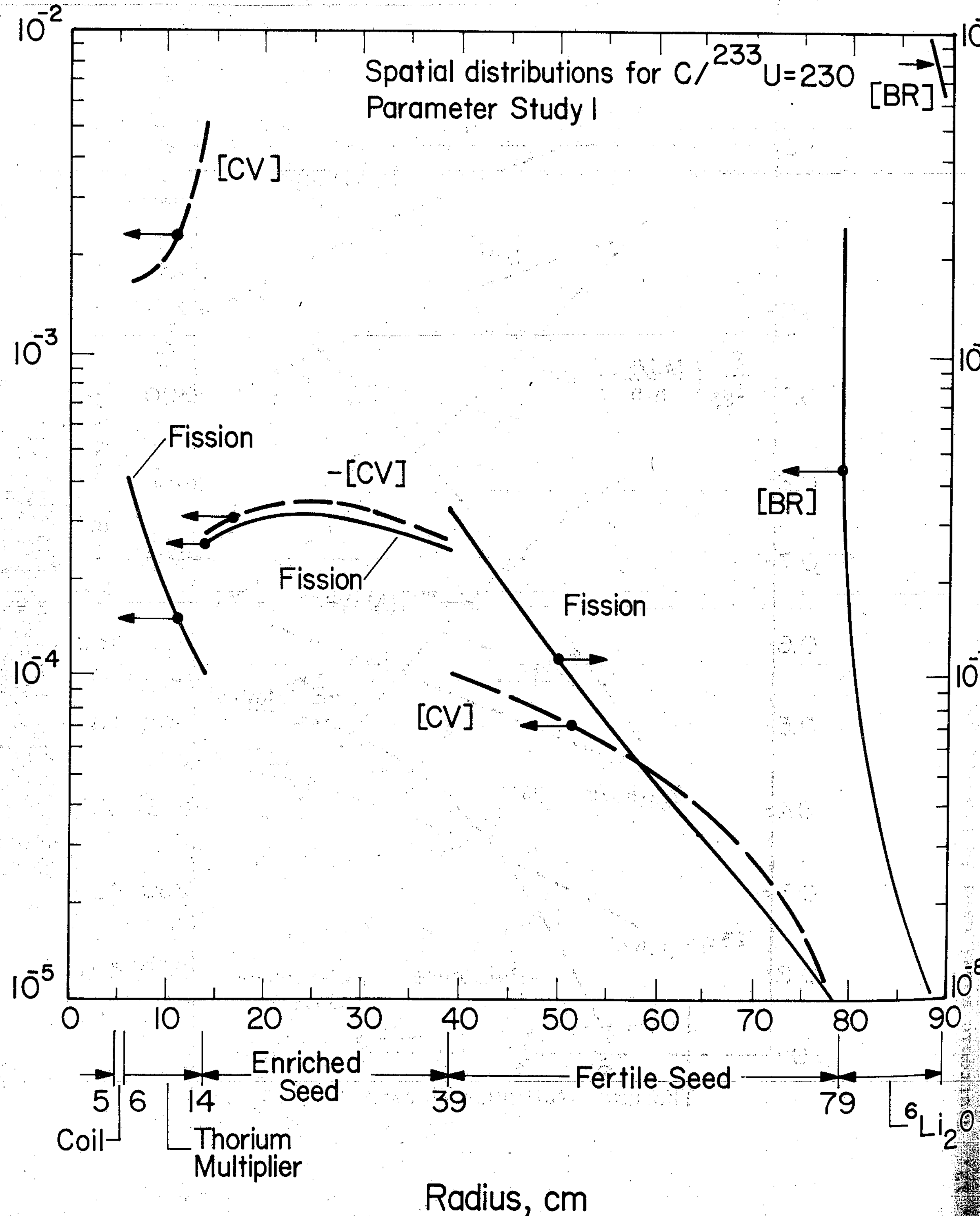


Fig. 14. Radial dependence of [BR], [CV] and fission rate for the [BR] = 1.0 case depicted on Fig. 13.

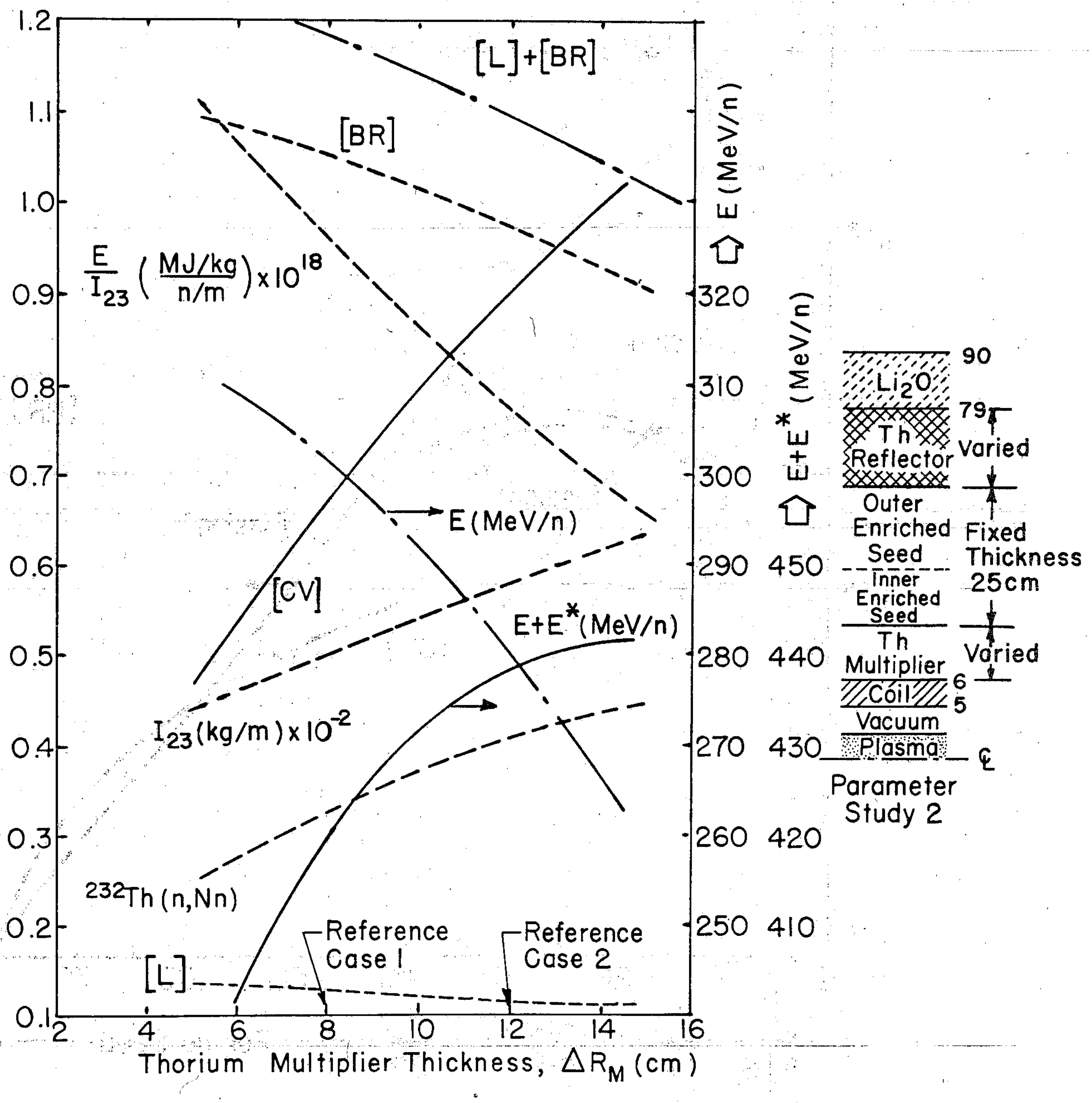


Fig. 15. Variation of blanket parameters with thorium multiplier thickness (Parameter study 2).

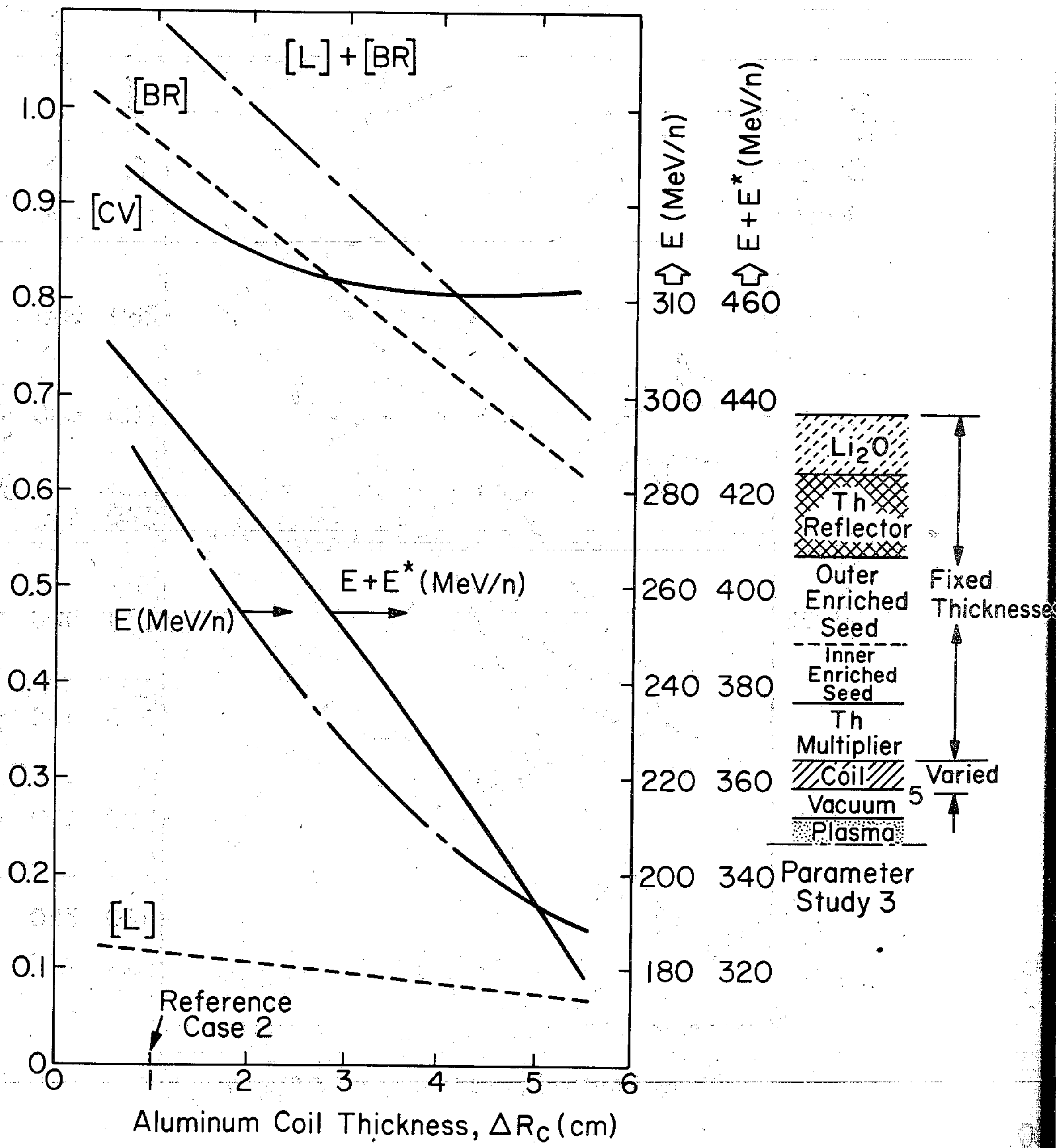


Fig. 16. Variation of blanket parameters with (Al) coil thickness (Parameter study 3)

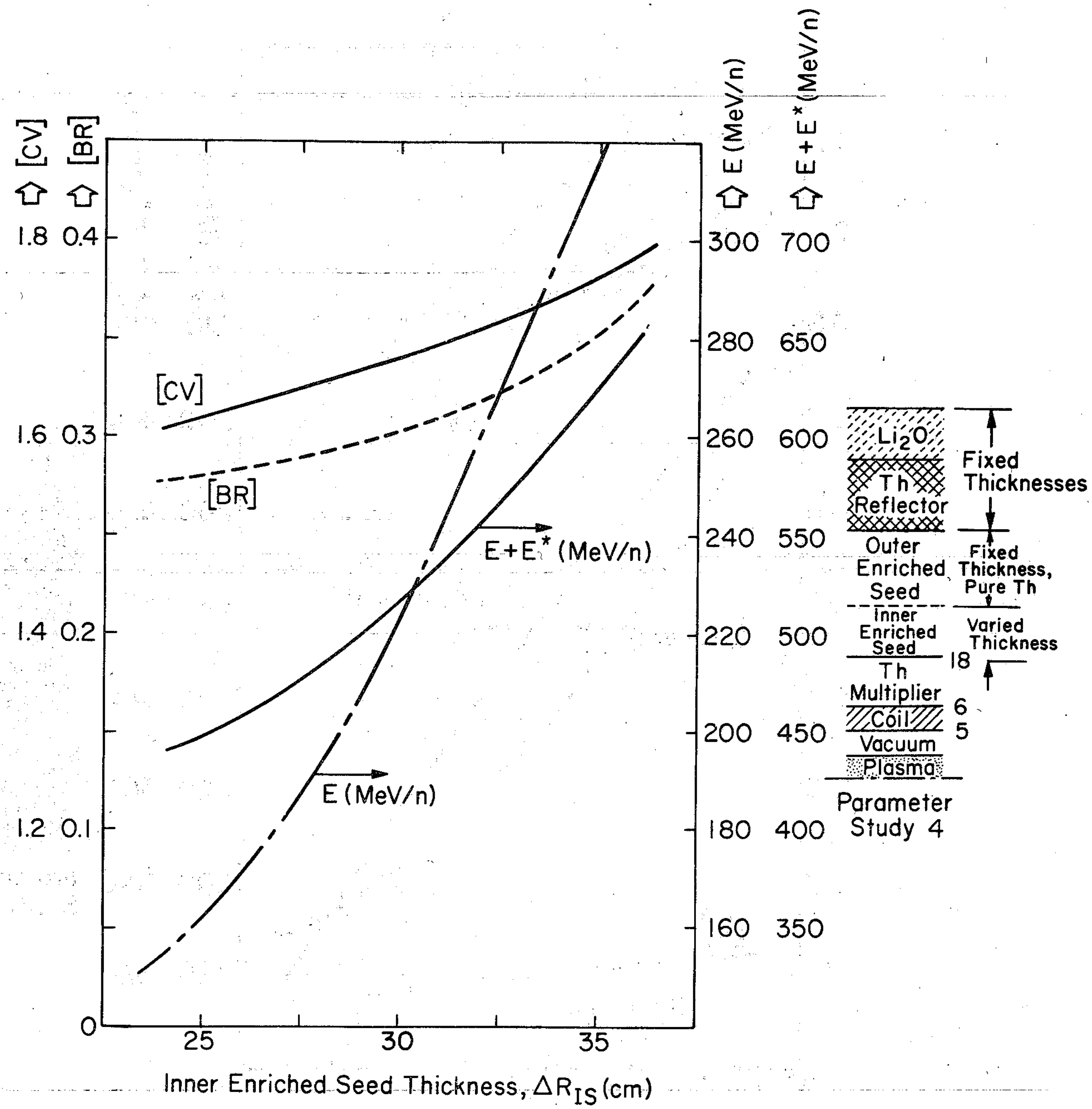


Fig. 17. Variation of blanket parameters with thickness of inner enriched seed region (Parameter study 4).

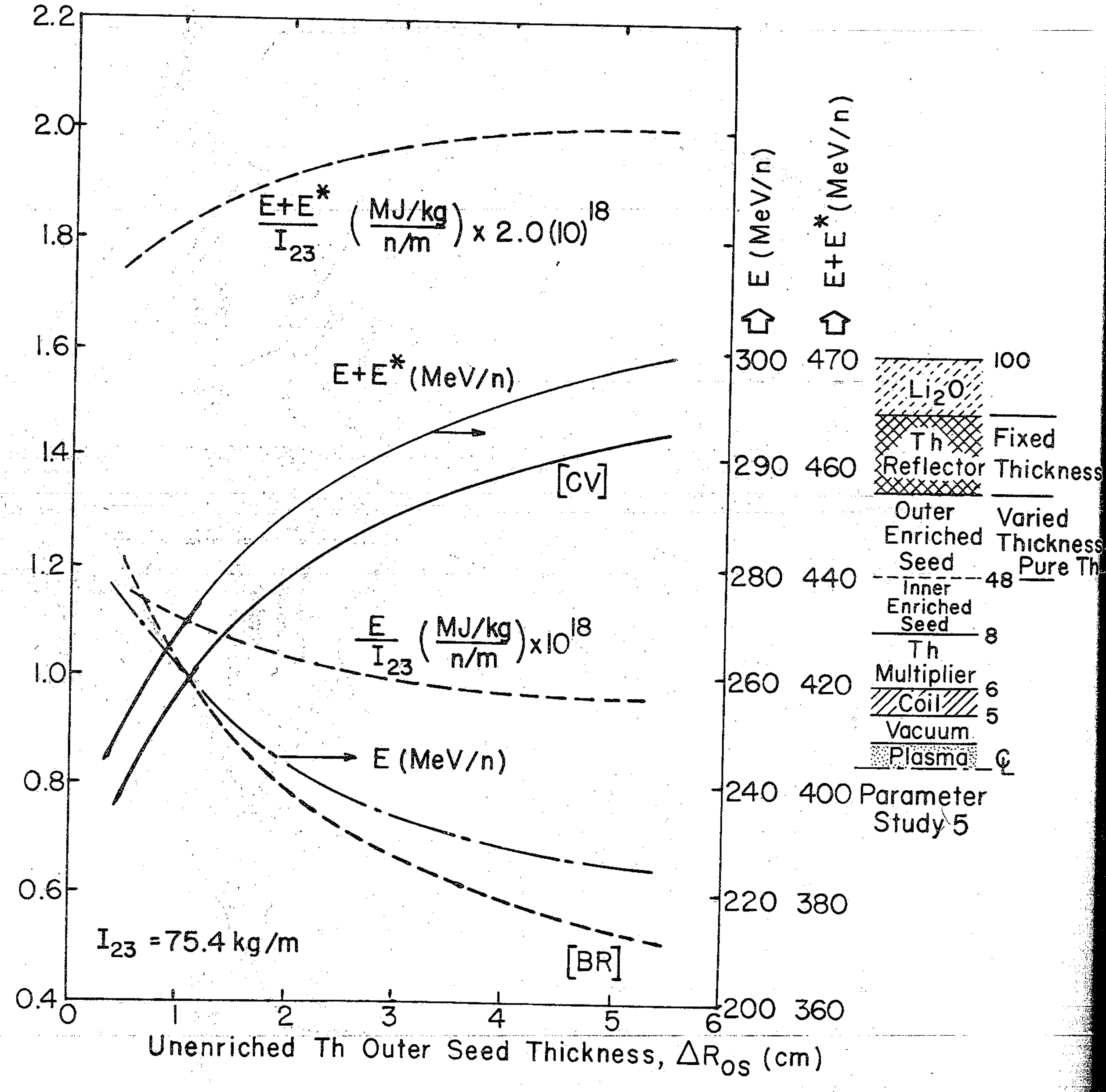


Fig. 18. Variation of blanket parameters with thickness of outer Th seed region (Parameter study 5).

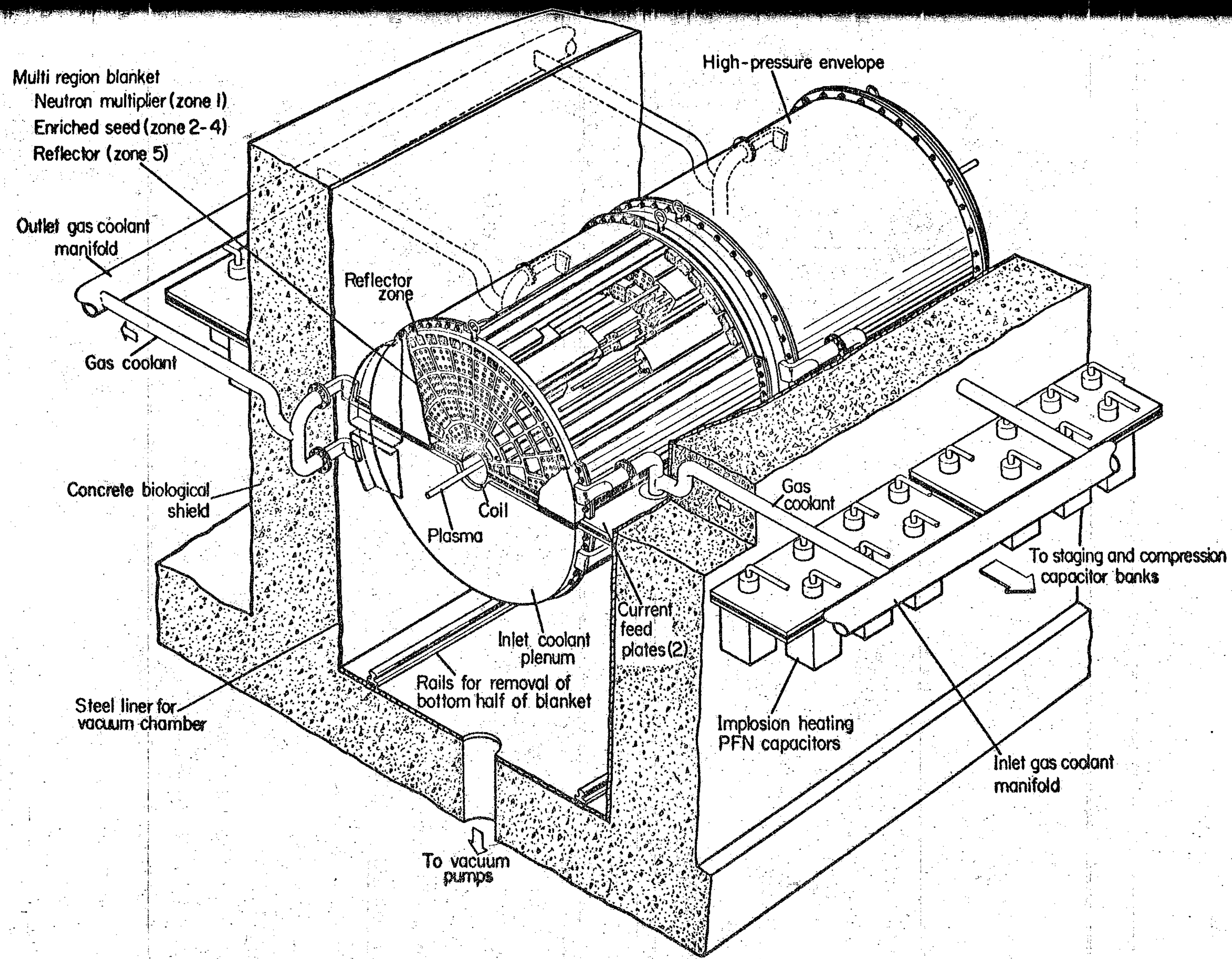


Fig. 19. Isometric diagram of gas-cooled LTPHR module (conceptual).

QUESTIONS ABOUT EIGHTH PRESENTATION

Lidsky: You raised the point that you put a hole in the place where the neutron flux was the highest. Despite the fact this reduces activation problems, I suggest that you would lose some 14 MeV neutrons and these neutrons by virtue of their energy and position within the reactor are precisely those that, using the reactor physicist's definition, have the highest importance.

Krakowski: That is a very good point because the 14 MeV neutrons are much more important than the ones that would be there if there was fissionable fuel.

Post: What was the effective multiplication factor, say on a typical design, that you achieve from the addition of the fissionable material? In other words, how much sub Q = 1 more than if it had not had that?

Krakowski: Oh, if it had not had that?

Post: Yes, What was the multiplication factor effectively?

Krakowski: For the reference case we had like 415 Mev per neutron. So the multiplication without fissile material would be on the order of one and probably nearer to one-half.

Post: So, Q would be on the order of a tenth considering the system as a pure fusion device.

Krakowski: That's right.

Post: OK, the effect of the multiplication was to reduce the length to a certain extent as compared to a simple pure fusion device.

Krakowski: That's right. For pure fusion that machine will have to be at least 10 kilometers long, and now we squeezed it down by the addition of this blanket to something below one kilometer.

Post: My comment had to do with the length. That is, I realize it would introduce a new set of problems which might be just as difficult as the ones you're talking about, but a still further reduction in length might be achieved in this linear geometry which is, of course, a very good geometry. This could be accomplished by operating in the "two component" mode that we have discussed for mirror systems, introducing the second component by neutral beams. There are beam access problems and other problems but there does result, at least in our calculations, a considerable further reduction in length as compared to the conventional straight thermonuclear burn mode.

Krakowski: There is also another point concerning length. Jeff Freidberg has just written a paper summarizing the end loss problem. This paper considers a number of theories of end loss. A parameter, η, that goes into the end loss formula as a function of the beta of the plasma is compared. There are two experimental points and there isn't good agreement between either experimental point or the five theories that were considered. I think the stoppering of the ends really ought to be looked at more seriously. A factor of two on the end loss means a lot when you're going from a kilometer down to five hundred meters.

Post: Right. Some form of stoppering action, either arising from the plasma beta itself or coming from extra stoppers will be required for the high energy trapped ions.

Coffman: Would you comment on the feasibility of obtaining the fissile seed, whether it's U-233 or plutonium?

Krakowski: The feasibility of obtaining 50 kg/meter (of the theta-pinch length) of uranium-233 is not promising. We are in the process of looking at the time scale require to "burn-in" this kind of inventory. The curves I showed here were based on one point out of a parameter space (50 kg per meter of fuel). Don Dudziak has recently looked at ways of chipping away at that inventory and, by suitable rearrangement of materials, has reduced the fissionable fuel inventory to about 18 kilograms per meter. In addition, a bonus results by increasing E + E* from 414 to about 480 Mev. A lot of optimization remains to be done on the blanket design. The main task is to push down the fuel inventory and to increase the value of E + E*.

Coffman: I suspect that your inventory problems are probably in the same ballpark whether you use plutonium or U-233 in terms of both costs and resource problem. In addition, if you look at what it would take to run a fusion reactor with a plutonium blanket using LWR's as a source, you would have to run the entire Nation's LWR reactor system (i.e. about 50 reactors) for 20 to 50 years jto get one fuel load.

Krakowski: Once there, however, you can take this thousand meter device and support 25 to 30 convertor reactors that have conversion ratios on the order of .9.

Dudziak: Even with the large inventory case which was very unoptimized, once you get going you have a doubling time of about five years at 4 megawatts per square meter and you're fine.

Krakowski: The intrinsic doubling time was five years for the reference case.

Dudziak: We will be looking at some boot-strapping to achieve a reasonable loading.

Lee: Why did you go to the thorium multiplier as opposed to some other option?

Krakowski: Working with the thorium system in terms of energy balance is like tying one hand behind your back.

Lee: What is the advantage that you see over uranium?

Krakowski: Well, there is supposedly an environmental advantage. The specific activity for U-233 is a factor of ten below that for Pu-239 and the BHP is lower by a factor of a hundred relative to Pu-239.

Lee: U-233 versus plutonium?

Krakowski: That's right.

Dudziak: Yes. There is another hand you tie behind your back which makes it hard to work. And that is you have to keep the thorium out of the high energy neutron spectrum if you want to achieve this great radiological advantage. This is one reason we are now looking at, among other things, a lithium blanket preceeding the thorium/graphite in order to minimize the production of Th-231 and other actinide chains.

Moir: I think that this is very interesting to look at these systems and give information on it from different perspectives. And yet, there may be a lot of other systems that don't have to be as long to get that high an $n\tau$ to make a smaller section useful. But it sure helped me a lot to have somebody put in persepctive whether this design is reasonable and needs incremental work on several aspects or whether this design is really way out in terms of the total amount of power or the total capital cost.

Krakowski: You would like to know the total amount of power? Four gigawatt's electrical at one megawatt per square meter wall loading.

Moir: This seems not in the same ballpark as power plants that we are thinking about now.

Krakowski: Well, it probably is, in the near term. It's not in the 500 MW(e) ballpark but, then again, that is not in today's ballpark as far as any power plant is concerned. A 500 MW(e) plant today is considered a small to average plant size.

Powell: The Rasmussen report estimates that you can go something like several hundred years before you get a core meltdown. Have you given any thought to how long one of these things could operate without one of the coils breaking up and spreading part of the blanket over the environment?

Krakowski: No, we haven't given any thought to it.

Powell: There is something like 20,000 psi in that field.

Krakowski: There is one other point I'd like to make. There has been some talk about carefully defining a boundary between a symbiosis reactor and a hybrid power generating reactor. From our experience at LASL, try as you may to develop a blanket which just breeds fuel, a lot of energy comes along with it. So if one is trying to design a "pure breeder reactor," I think that one is still talking about hundreds of megawatts of thermal energy that is going to have to be converted. In a sense, the idea of a sort of room temperature breeder blanket just sitting there shoveling out U-233 really isn't realistic.

Lidsky: I would like to point out that there are counter-examples to that conclusion which exist in the literature. They are not easy to find and they are not prevalent but counter-examples exist.

Baker: Your thorium is static, isn't it, and not flowing through the blanket?

Krakowski: Yes.

Baker: I think you mentioned before that with the TCT, the power density in the plasma would be much higher and one could breed a lot better. I don't agree with the statement that pure breeding hybrid systems are out of the question. We need to increase the power density of the plasma, and one technique is to use the two component torus.

Krakowski: Yes, but each neutron that is coming into the blanket will eventually be multiplied and, at some point, will dump 14 Mev of its own thermal energy into the blanket which you're certainly not going to throw away. When you look at TCT or the mirror concept or this concept in its overall context, that represents megawatts of energy.

Baker: It doesn't have to be that large. You don't have to have gigawatts; you can have a few hundred megawatts.

Krakowski: That is not negligible.

Leonard: You have consider the limit on the amount of fissionable material you are going to produce. You count the number of neutrons that it takes and multiply it by 15 or 20 Mev and you are going to have a considerable amount heat if you are talking about producing materials for thousand megawatt thermal systems.

Krakowski: That's exactly the point I'm trying to make.

Baker: That also assumes it's DT.

Krakowski: That's implicit.

GENERAL QUESTIONS TO SPEAKERS

Lee: I have just a quick comment about the idea of a "breeding only" system. It's very obvious, but if you have a system in which you dump the heat, you have to buy power to run it. If you're running at a Q of one and a breeding ratio of one, you're talking about $50.00 or more of electric power for each gram of plutonium you produce. So there doesn't seem to be much room there to throw away your heat. If you do, you might not be economic.

Bogart: Has anybody here, besides some of us, calculated the heating value of a gram of plutonium in terms of present day heating value prices? Is it realistic to expect that the prices of plutonium, U-233 and U-235 will remain stable if there is a significant difference between that heating value price and the heating value price of fossile fuels?

Lidsky: I think that is a good question to ask, but you also have to ask what is the cost to burn it. The example you come to is solar power. There, in effect, the energy is free but it costs you a lot to collect it. So you have to be a little bit careful about putting something that far removed from its ultimate useable form on an energy basis. Can I also make a response to J. D. Lee's comment about pure breeder systems. Apparently, there is some confusion between pure breeder systems and gadgets in which you only breed and throw away the heat. I don't think that there is anyone who can claim that there is an advantage to throwing away the heat if there is a substantial heat output in any system. You convert as much as you can economically. The distinction should be made between pure breeders and gadgets that produce power by fission reactions which opens up a whole bunch of problems. To compare pure breeders with gadgets that can do no more than breed while throwing away the byproduct heat is essentially begging the question.

Coffman: I'd like to have the speakers, if they are willing, to comment briefly on the technical and economic feasibility of generating an adequate fissile seed for either Pu-239 or U-233 production. It seems to me that this is an initial hurdle that needs to be looked at very early-on. If it takes 1 to 5 thousand reactor years of LWR's to generate one fusion hybrid reactor fuel load, the question is sort of academic. I'd like to see them focus in and talk about that problem.

Wolkenhauer: I was going to comment on this before. You focussed on a couple of designs. There are maybe 8, 10, or 12 different designs wandering around in the literature if you go look for them. If you start looking at all of these designs on some sort of economic grounds, you very quickly conclude that, yes, the initial load is a problem. In fact, I have about arrived at the conclusion that for hybrids to really be interesting, they are going to use natural or depleted fuel. There are a lot of attractive designs that aren't enriched, and then you just don't have this problem at all. None of these, I think, are optimized designs. I think that this is another way of putting it.

Moir: I have one relevant point. I think our design would not be arranged if it didn't contain fissile fuel to begin with and produce 1300 kg per year. Now that would be enough to start up two HTGR's of the type at Fort St. Vrain per year.

Coffman: And it bred the tritium?

Moir: And it bred an adequate amount of tritium.

Maniscalco: As you pointed out these systems, especially fast fission blankets, operate at lower power densities than commercial fission reactors. Because of this, they exhibit lower fuel burn-up rates and poorer fuel utilization factors in terms of kilograms of fuel per megawatt electrical. At these lower power densities and fuel burn-up rates, it may be possible to significantly improve the energy amplification and fissile breeding performance by utilizing uranium metal as the fertile material.

Moir: I just have a general thing I'd like to throw out to everybody here. We have heard some advantages of hybrids and I would like to go over them. They can be built employing a sub-lawson criteria by a order of magnitude and some are what we are seeing in laboratories or are what people are willing to project. So maybe the fusion might come along quite simple. The fission technology: some of these are quasi-existing technology so we don't have to build experiments to see if they work. I think we ought to address ourselves to the question that if all things fall in line, how could we employ or bring to commercialization the results on a rapid time scale. Ray (Huse), this morning, said that there is no way we can get anything before about 1990 and probably you can't gear up enough even though it would be nice to go as rapidly as possible from the initial phases.

Bogart: Well Ralph (Moir), as a precursor exercise even to that, we ought to define very precisely whether or not the idea makes any sense at all economically, environmentally and all of the other factors before we spend a lot of money in the technical arena.

Moir: I think that probably there are hybrids that aren't so sophisticated and some of them might be developed much more rapidly.

Williams: Do you have some specific examples in mind Ralph (Moir)?

Moir: Well interim experiments, as far as the fission technology is concerned, could begin without the fusion driver. When we were in Japan, we saw the Japanese doing a hybrid experiment.

Williams: Your're talking about laying the technical basis for the fissile blanket should the fusion driver prove to be technically and economically viable?

Moir: Yes, and if one of our goals was: which could be the quickest but not quite the best, we might take a somewhat different tack. We would probably go for a smaller system so that pilot plants could be built more rapidly.

Lotker: I think one way of heading in this direction is to go to a "fuel production only" option. The problem with the utility is it has to commit generating facilities. It's got to say, OK., we are going to use your new-fangled generator on our system, and then we have to meet some dependability criterion. You could justify this thing on a "fuel production only" basis, and by that I mean, no external sales of electricity. You might generate electricity too, as I said before, to run the laser or run the neutral beam injectors, but once you cut it off from the system for all practical purposes, it represents a much lower risk to the utilities. Private ventures could be created to produce this thing and sell the fuel without having to get a utility to commit its next thousand megawatts of generation.

APPENDIXES

January 29, 1975

Appendix A

THE ROLE OF A HYBRID (FUSION-FISSION) REACTOR IN A NUCLEAR ECONOMY

by

W. C. Wolkenhauer

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

RICHLAND, WA 99352

INTRODUCTION

The Hybrid (fusion-fission) reactor is defined as any fusion reactor with a fissile or fertile blanket. The concept is one that has been proposed since the advent of the fusion reactor program. Recently, two major roles have been identified for the hybrid reactor.[1] One role is that of a breeder of fissile fuel only and the other role is that of a self-contained power plant which also breeds fissile fuel. Investigations have been and are being carried out to determine the characteristics of the hybrid in both of these roles.

Much of the recent work at Pacific Northwest Laboratories (PNL) has concentrated on the role of the hybrid reactor as a power plant.[2] One of the original hybrid proposals envisioned the use of a fusion reactor as a breeder of fissile fuel.[3] More recently, Lidsky proposed using a fusion reactor as a source of bred fissile material to feed a molten salt reactor.[4] In another variation, Lubin and Hurwitz propose to use a laser fusion device to enrich light water reactor fuel either prior to or after its exposure in a fission reactor.[5,6] Finally, one of the apparent long-term goals of the two component torus (TCT) experiment is the development of a hybrid fissile fuel factor.[7] Thus, there seems to be growing interest in the use of a hybrid reactor as a fissile fuel factory. This interest is apparently brought about by its promise of achieving a relatively near term payoff for investment in fusion research and development. A fissile fuel factory is defined as a hybrid reactor employed solely for the breeding of fissile fuel from uranium or thorium. Any heat produced in the process is released to the atmosphere with no electrical generation.

In this paper, some of the hybrid concepts which have been proposed to date will be separated into groups according to their apparent neutronic characteristics. Then, an attempt will be made to identify these hybrid groups with two possible roles in a nuclear economy (i.e., fuel breeding and direct power production). An analysis will be made to see if there is enough current information to determine which hybrid concepts appear most promising when assigned to these roles. In this way, specific Hybrid concepts may be identified with certain missions. It should be emphasized that, because of the lack of hybrid design definition, any conclusions arrived at in this analysis can only be considered informal speculation.

The analysis will begin by first comparing the hybrid reactor concepts available on the basis that they are operated as fissile fuel factories. This is done by comparing their income producing potential for producing fissile fuel to that of the original fuel factory; i.e., the diffusion plant. In addition, for comparison purposes, the characteristics of a fast fission breeder reactor operating as a fuel factory are developed. Then one can rank the hybrid concepts as fuel factories and perhaps determine which, if any, of these concepts could compete with the diffusion plant. After this is done, a measure of the additional income which is available to each of the concepts from the sale of electrical power is computed in order that the various concepts be evaluated as power plants. Finally, an estimate of the capital costs of each of the concepts both as fuel factories and as power plants is made. A comparison of income production with capital cost allows assignment of the various concepts to specific roles.

THE DIFFUSION PLANT AS A SOURCE OF FISSILE FUEL

If one is considering operation of a hybrid reactor as a factory for fissile fuel only, (i.e., release of thermal heat to the environs), one logical place to start assessing the viability of this concept is to review the available alternatives for fissile fuel. In this way, economic goals can be identified which must be met by the fuel factory hybrid in order that it be competitive. The prime source of enriched fissile fuel is the diffusion plant.

Currently, the cost of 93% enriched ^{235}U is \$15.28/gm-fissile from a diffusion plant. This value is based on a value of \$44.25/separative work unit (SWU), a tails assay of 0.3% and a uranium cost of \$8/lb-U_3O_8.[8] Plutonium bred in thermal reactors has an estimated value in the range from .5 to .7 times 93% enriched uranium in thermal reactors due to fabrication penalties and higher isotope concentrations. Uranium-233 bred from thorium currently has an estimated value in the range from 1 to 1.1 times 93% enriched uranium in thermal fission reactors.[9] The specific value of both bred plutonium and uranium-233 is dependent on the reactor specified and relates in part of the fabrication cost. If the price of U_3O_8 should rise to \$16/lb, the price of 93% enriched uranium becomes about \$19/gm. At \$24/lb for U_3O_8, the price becomes about \$24/gm. If the price of separative work should increase by 50% with the price of uranium remaining stationary at \$8/lb, then the price of 93% uranium becomes about \$20/gm.[8]

These parameters constitute an approximate economic goal which any competing fuel factory concept must meet. That is, in order for the fuel factory hybrid to be competitive, it must produce fissile fuel at prices which are competitive with the primary source of fissile fuel at that point in time which the hybrid fuel factory is proposed to be introduced. In this analysis, differences in the value of the bred fuel are ignored. This is due primarily to the lack of knowledge of the composition of the bred fuel. There is some evidence that fuel bred in a hybrid is of higher value than reactor bred fuel.[2] The state of the art of hybrid technology does not at this time allow for such fine definition.

THE FAST BREEDER FISSION REACTOR AS A FUEL FACTORY

To initiate the analysis of the fuel factory concept, an analysis of a high gain fast breeder fission reactor as a fissile fuel factory is performed. The fast breeder fission reactor is a useful place to initiate the discussion because, among other things, much more is known about this concept than about the hybrid reactor.

Using an approach suggested by Fortesque,[10] assume that one wishes to operate a high gain fast breeder reactor which produces fissile fuel at the rate of E_R gms/kW(t)-hr and that any heat generated in the process is dumped ultimately to the atmosphere, (i.e., no electrical generation, thus the discussion here is in thermal kilowatts). Assume that the capital charge of the reactor (Cc) in units of MILLS/Kw(t)-hr is known. The capital charge is defined as the rate of return required from the reactor based on a given unit capital cost, interest rate, and reactor load factor. Then, if as a minimum, one wishes to sell the bred fuel at a rate such that only the capital costs are recovered, one can relate this rate (Ec) in \$/gm to the capital charge (Cc) and production rate (E_R) by:

$$Ec = \frac{Cc \times 10^{-3}}{E_R} \qquad (1)$$

The net production rate of the reactor, allowing for replenishment of the breeder reactor fuel, can be related to the fission rate, (F_R) (in units of gms fissioned/Kw(t)-hr) and the reactor conversion ratio (C_R) by the relationship

$$E_R = F_R\ (C_R-1) \qquad (2)$$

where the conversion ratio is defined here at the ratio of the rate of production of fissile fuel divided by the rate of loss of fissile fuel.

Combining the above two relationships results in:

$$Ec = \frac{Cc \times 10^{-3}}{F_R\ (C_R-1)} \qquad (3)$$

The fission rate can be approximated by the relationship that the fission of one gram of material results in the release of 1 megawatt day (MWD) of thermal energy. Thus, we take F_R to be equal to:

$$F_R = \frac{1 \text{ gm/MWD}}{10^3 \frac{\text{kW(t)}}{\text{MW(t)}} \times 24 \text{ HRS/DAY}} = 4.167 \times 10^{-5} \text{ gms/kW(t)-hr.}$$

The capital charge of the <u>reactor</u> (Cc) (no thermal conversion system is required) in MILLS/Kw(t)-hr can be calculated from the capital cost (Cc') (in \$/kW(t) installed) of the reactor by the relationship:

$$Cc = Cc' \times 10^3 \times \frac{I}{F} \times \frac{1}{24 \text{ HRS/DAY} \times 365 \text{ DAYS/YEAR}} \tag{4}$$

where, F = reactor load factor

I = Interest charge rate on invested capital

If a reactor load factor of 80% and an interest charge rate of 14% are assumed, then

$$Cc = Cc' \; 0.01998 \text{ MILLS/kW(t)-hr} \tag{5}$$

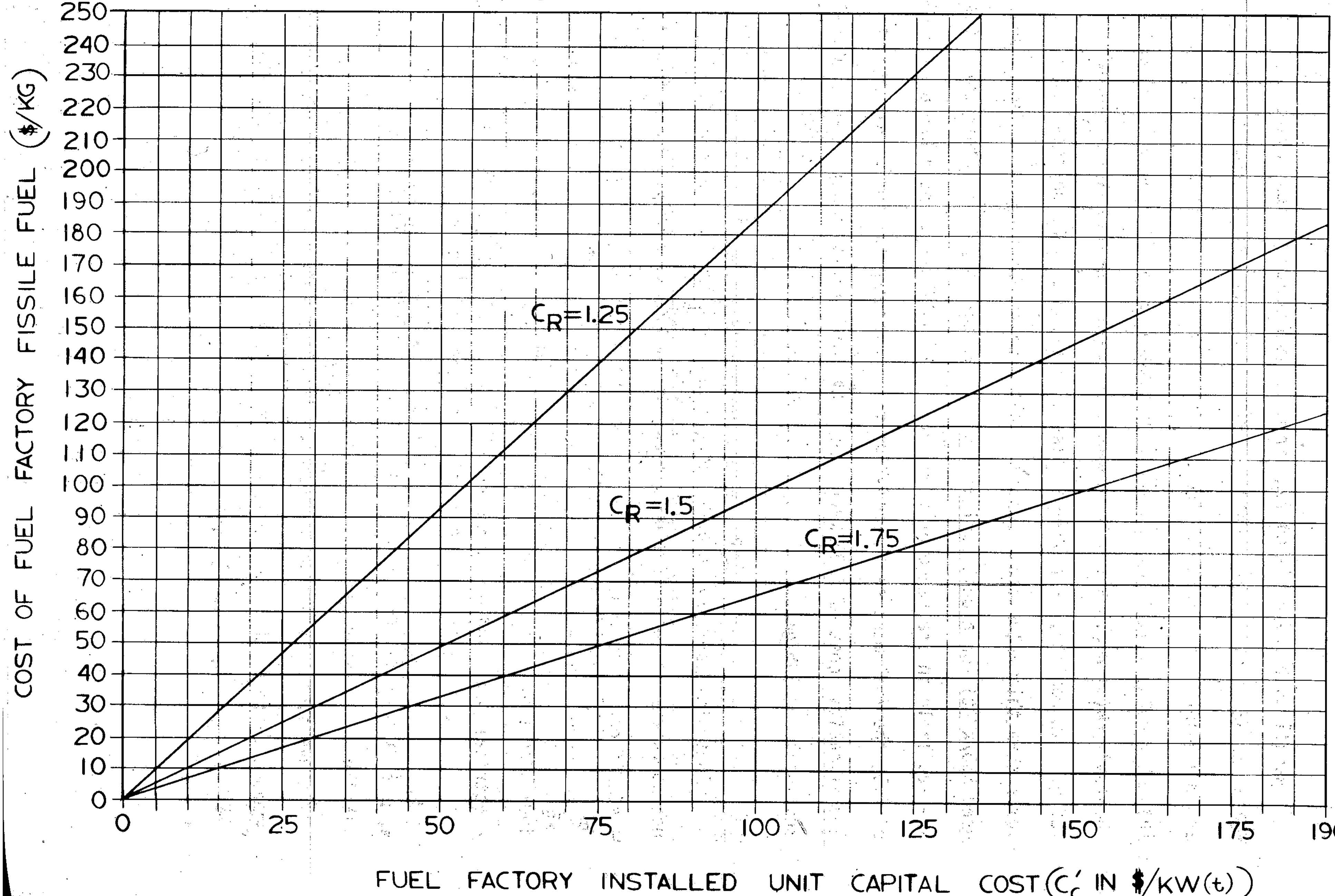
Substitution of the above fission rate (F_R) and capital charge (Cc) into equation (e) results in:

$$Ec = \frac{Cc' \; 0.4794}{(C_R - 1)} \tag{6}$$

The above equation relates the cost at which fissile fuel from a fast breeder fission reactor must be sold (in units of \$/gm) in order to recover the capital charge only (no consideration being given to fuel or other operating charges) as a function of the installed capital unit cost of the reactor (in units of \$/kW(t)) and fissile fuel conversion ratio.

In Figure 1, Equation 6 is graphed as a function of various fast breeder reactor conversion ratios and installed unit capital costs for a capital charge rate of 14% and a reactor load factor of 80%.

FIGURE 1
COST OF FUEL FACTORY FISSILE FUEL AS A FUNCTION OF INSTALLED CAPITAL COST AND FUEL CONVERSION RATIOS



C-2-N-001

Recall that 93% enriched fuel from the diffusion process currently sells for about $15/gm and could be expected to rise to perhaps $25/gm at some point in the future. The relationship shown in Figure 1 indicates that at a conversion ratio of 1.5, for example, a maximum installed cost of about $26/kW(t) would allow for capital cost recovery at a sale price of $25/gm. In reality, fast breeder fission reactor conversion ratios are lower than this value and installed capital costs are higher.[11] Fortesque used $50/kW(t) for a unit capital cost in his analysis.[10] Other estimates are available in the range of $100/kW(t). At $50/kW(t), and a breeding ratio of 1.5, the fuel must sell for $48/gm to recover capital costs only and at a unit capital cost of $100/kW(t) and a breeding ratio of 1.5, the bred fuel must sell for $96/gm.

Thus, this analysis shows that the fast breeder fission reactor cannot compete with the diffusion plant as a fuel factory only. It must sell a significant amount of electrical power in order to justify the capital expenditure. Indeed, this is the case. Review of fast breeder reactors indicates that most of the revenue from the plant comes from the sale of electrical power and a minor portion from the sale of fissile fuel.[12]

A fundamental reason that this reactor is not competitive as a fissile fuel factory apparently is that the diffusion plant profits from the conversion of relatively cheap uranium into a much more valuable end product while at the same time, the breeder gains only fractionally because its product is no more valuable than its feed.[10]

Considering the hybrid fuel factory, it would appear that any hybrid concept based on enriched fuel would suffer the same difficulty. Analysis of the PNL/LLL Mirror Hybrid Reactor, as presently designed,[2] by this method indicates that the reactor can cost no more than \$3.1/kW(t) in order that it be competitive with the diffusion plant because of its low current conversion ratio. While hybrid cost numbers are virtually unknown, this appears to be a ridiculously low capital cost. Thus, it appears that a successful fuel factory must have a high conversion ratio and operate on non-enriched (i.e., cheap) fuel.

THE HYBRID (FUSION-FISSION) REACTOR AS A FUEL FACTORY

The next step is to develop an analysis model for the CTR similar to that developed for enriched high gain fast breeder reactors in order to determine the probable characteristics of a hybrid fuel factory CTR. If, from such a model, one could determine the characteristics of a hybrid fuel factory, then one could perhaps make a judgment as to whether a hybrid fuel factory is indeed a practical concept.

Remember that Equation (1) related to the plant capital charge (Cc), fissile fuel production rate (E_R) and the minimum fissile fuel cost (E_c) by

$$Ec = \frac{Cc}{E_R} \times 10^{-3} \frac{\$}{gm} \qquad (1)$$

For a non-enriched fuel factory, the term E_R must be redefined because there is no requirement to reinvest enriched fissile fuel into the reactor if non-enriched Hybrids only are considered. In addition, low energy multiplication hybrids would have a significant amount of their energy produced by the fusion process. Thus, one cannot relate fissile fuel production to power output in a linear manner.

If one defines F_c^{238} as the rate of ^{238}U capture reactions, F_F^{235} as the rate of ^{235}U fission reactions, F_F^{238} as the rate of ^{238}U fission reactions and F_u as the fusion reaction rate in a hybrid reactor, one can write:

$$E_R = \frac{F_c^{238}}{200(F_F^{235} + F_F^{238}) + 17.6\ F_u} \times \frac{K_1 \text{ gms}}{K_2 \text{ Kw(t)-hr}} \qquad (7)$$

The above relationship assumes a fission energy release of 200 MeV per event and a fusion energy reaction release of 17.6 MeV (i.e., the deuterium-tritium reaction). K_1 is a conversion factor from atoms produced to grams produced and K_2 is a conversion factor from MeV to kW(t)-hr. Thus, the units of E_R become gms of fissile fuel produced per kilowatt-hour of thermal energy produced just as in the previous analysis.

Assuming a value of 3.95×10^{-22} grams of ^{238}U per atom of ^{238}U (K_1) and a value of 4.44×10^{-20} kW(t)-hrs per MeV (K_2), one can then write:

$$E_R = 0.889 \times 10^{-2} \frac{F_c^{238}}{200(F_F^{238} + F_F^{235}) + 17.6\ F_u} \qquad (8)$$

One can substitute this relationship for E_R into Equation 1 and obtain:

$$Ec = \frac{Cc \quad 200(F_F^{238} + F_F^{235}) + 17.6\ F_u}{F_c^{238} \times 0.889 \times 10^{-2}} \times 10^{-3} \qquad (9)$$

If, as before, one uses a reactor load factor of 80% and an interest charge rate of 14%, then:

$$E_c = \frac{C_c' \quad 200(F_F^{238} + F_F^{235}) + 17.6\ F_u}{F_c^{238}}\ 2.24 \times 10^{-3} \qquad (10)$$

In the analysis of a system designed for ^{233}U fissile production, F_c^{238} is replaced by F_c^{233} (the capture rate for ultimate formation of ^{233}U), F_F^{238} is taken as zero, and F_F^{235} is replaced by F_F^{233} which is the fission reaction rate for ^{233}U.

Equation (10) is the model which will be used in the following analysis. Notice that if $F_F^{238} = F_u = 0.0$ and if one defines $C_R - 1 \approx \frac{F_c^{238}}{F_F^{235}}$, then Equation (10) becomes consistent with Equation (6). The minor numerical discrepancy comes about because of the use of 1 gm fissioned per megawatt-day, on the one hand, and 200 MeV per fission, on the other hand, neither of which are strictly correct.

In developing Equation (10), parasitic (n,γ) absorptions in ^{235}U have been ignored. This will turn out to be a minor factor in the hybrids which will be analyzed here. Also, notice that provision has been made in the equation for ^{238}U fission. This is a reaction which can potentially be much more important in hybrid reactors than in fission reactors.

ANALYSIS OF THE INCOME POTENTIAL OF SOME SPECIFIC HYBRID CONCEPTS

In using the model as developed in Equation (10) in order to determine the income potential of hybrid reactors as fuel factories, one is faced with one of the facts of the current state of hybrid reactor development. There has been relatively little analysis of possible hybrid reactor designs to date. Moreover, all of the calculations that have been performed are deficient in one or more ways. Common deficiencies include use of inadequate data, performing calculations at room rather than at operating temperature, and lack of engineering reality in blanket designs. The session on hybrid reactors included in the First ANS Topical Conference on Fusion Reactors[13] constitutes an acceptable summary of the technical analyses which have been performed to date on this concept. In general the hybrid concept characteristics reported are taken uncritically in this analysis.

Table I summarizes the characteristics of the hybrid designs which have been selected for analysis here. For purposes of this analysis, we will identify the designs by the name of the lead author. In the case of duplicate lead authors, the designs are also assigned a number.

G. W. Braun and L. M. Lidsky[14] have proposed a hybrid reactor based on the theta pinch fusion reactor concept. The blanket is fueled with natural uranium and is cooled with lithium. The blanket is cylindrical in shape. The blanket is 40 cm thick and has an inner radius of approximately 20 cm. It has a 20 cm graphite reflector surrounding the blanket. The lithium is enriched to 30% in Li^6. The device operates with a $10/MW/M^2$ neutron wall loading and a blanket power density of 60 kW/liter. The theta pinch device employed is running at breakeven conditions and is 1000 meters in length. It has an apparent thermal output of 60,000 MWt.

TABLE I. Hybrid Reactors Analyzed

AUTHORS	FUEL CYCLE	COOLANT	STRUCTURE	MODERATOR	BLANKET POWER DENSITY (kW(t)/liter)	FERTILE CAPTURES PER FUSION NEUTRON	THERMAL FISSIONS PER FUSION NUETRON	FAST FISSIONS PER FUSION NEUTRON	FRACTION OF ENERGY FISSION
BRAUN AND LIDSKY(14)	URANIUM	LITHIUM	STAINLESS STEEL (?)	----	60	.883 (D)	.053 (D)	.300 (D)	.800
LEE (15)	URANIUM	He	STAINLESS STEEL	----	39	1.5	.074	.536	;874
PARISH AND DRAPER (16)	URANIUM	LITHIUM	NIOBIUM	----	300-30	.6654	.0315	.1986	.723
PARISH AND DRAPER (17)	THORIUM	LITHIUM	NIOBIUM	----	203-56	.3118*	----	.0472+	.316
LONTAI (18)	URANIUM (DEPLETED)	Li_T;BEF_2; UF_4	MOLYBDENUM	SAME AS COOLANT	----	,185	----	,095	,519
LIDSKY(19)	THORIUM	Li_T; BEF_2 ThF_7	TZM	GRAPHITE PLUS COOLANT	.90	.3995*	----	----	0.0
MANISCALCO AND WOOD(20)	URANIUM	LITHIUM(7)	NIOBIUM	Li^7H	87.3	3.51	1.655	5.69	.962

D - DERIVED   * - Capture rate for ^{232}Th + - Fission rate for ^{232}Th

J. D. Lee[15] has proposed a hybrid reactor based on the mirror (Yin-Yang) fusion reactor concept. The blanket is fueled with natural uranium and is cooled with helium. The blanket is spherical in shape, has an inner radius of 4.8 meters and is 1 meter thick. The blanket is divided into an inner fissile region and an outer tritium breeding region. The fissile region is 20 cm thick. The fissile region has a power density of 39 kW/liter. Uranium carbide fuel is used. It has a nominal thermal power output of 2500 MWt.

Parish and Draper[16] have designed a hybrid reactor apparently based on TOKAMAK geometry. The blanket has a toroidal shape. The blanket is fueled with natural uranium and cooled with lithium. The blanket has an inner radius of 200 cm and an outer radius of 300 cm. The fissile region of the blanket has an inner radius of 204 cm and an outer radius of 230 cm. Regions beyond the fissile region include lithium and graphite regions. The device has a neutron wall loading of 10 MW/m^2 and the fissile blanket has a power density which ranges from 300 kW/liter to about 30 kW/liter.

Parish and Draper[17] have also conducted a comparison study to the one reported above with the emphasis on thorium fuel. The fission fuel consists of ThO_2 rods clad in niobium. Natural liquid lithium was chosen to be the coolant for the fissioning regions of the blanket. The fissile region is 13 cm thick. The total blanket, which is toroidal in shape, has a 200 cm inner radius and 300 cm outer radius. The fissile region has an inner radius of 200.5 cm. The fissile blanket has a power density which ranges from 203 kW/liter to 56 kW/liter.

Lontai[18] designed a hybrid reactor fueled with UF_4 (depleted) in a fused salt coolant. The fusion device is unspecified. The total blanket is 51.0 cm thick with a 49 cm thick uranium fueled region outside of a 1.5 cm thick coolant region. The vacuum wall is 1 cm thick molybdenum with a 5 MW/m^2 neutron wall loading. The lithium is 50% Li^6.

Lidsky[19] has proposed a true fuel factory concept based on a thorium bearing Li-Be salt fuel as developed in the molten salt reactor program. The fusion device has a 125 cm plasma radius. The blanket is 85.75 cm thick with a 30 cm thick thorium fueled region. The fusion reactor has a neutron wall loading of 1 MW/m^2.

Maniscalco and Wood[20] describe a hybrid design based on a natural enriched uranium Li^7H moderated blanket. The blanket uses the convertor/lattice concept. The design is intended for use in conjunction with a D-T laser fusion device. The blanket is spherical with an inner radius of 220 cm and an outer radius of 320 cm. The convertor is 8.5 cm thick and the lattice is 42 cm thick. The device has a neutron wall loading of 1 MW/m^2.

An analysis of the hybrid concepts developed in Table I is given in Table II. In Table II the fissile production rate of the various concepts is given in units of gms/kW(t)-hr (E_R). The maximum allowable capital cost of the reactor as a function of several assumed market values for fissile fuel for the sale of fissile fuel only and for the sale of both fissile fuel and electric power is given based on equation (10). This latter parameter is calculated by assuming that the energy created in operating the fuel factory is not dumped, but rather is converted to electricity at an efficiency of 40% and sold at 7.96 mills/kW(e)[21] while the bred fissile fuel is sold at 15 $/gm. These values correspond to the approximate current values for these items. Because inflationary forces which would increase these values in future years would tend to increase other parameters (such as plant capital cost), a relative comparison at current price levels is felt to be useful in drawing preliminary conclusions about the possible future development of the hybrid reactor.

In Table II, the maximum possible capital cost or potential income of the device when used only as a fuel factory is calculated for three possible fissile fuel prices. The 15 $/gm cost is approximately equivalent to present day costs. The 20 $/gm and 24 $/gm values are used both to depict possible future events and to indicate the impact of future price changes. One should be aware that the maximum possible capital cost or potential income given here is just that. No allowance has been made for fuel and operating costs, as this is merely a speculative upper bound analysis. Actually, to be economically competitive, the capital costs would have to be lower in order that the other costs (i.e., operational) could also be recovered. That is, the values given in Table II can be viewed as either the maximum capital cost at which a given concept can be competitive or they can be viewed as the maximum potential income from the design. This approach will allow one to make relative comparisons between designs. Determination of the absolute costs would require more detailed analysis.

In reviewing the results of the analysis shown in Table II, certain aspects become apparent. The three fast uranium blankets (Braun and Lidsky, Lee, Parish and Draper[1]) exhibit remarkably similar characteristics under this analysis model even though they contain different structural materials, different coolants, etc. and perhaps as importantly, have been analyzed neutronically with different methods and data. Apparently, the fact that they are all fast spectrum devices results in differences in detailed design having relatively little impact on their characteristics when analyzed in this matter. This similarity allows them to be discussed as one group.

TABLE II. Summary of the Potential Income for Various Hybrid Concepts

AUTHORS	GMS-FISSILE PRODUCED PER kW(t)-HR(E_R)	MAXIMUM POTENTIAL INCOME AT $15/gm in $/kW(t)	MAXIMUM POTENTIAL INCOME AT $20/gm in $/kW(t)	MAXIMUM POTENTIAL INCOME AT $24/gm in $kW(t)	MAXIMUM POTENTIAL INCOME $15/gm FOR FISSILE FUEL AND 7.96 MILLS/kW(e)-HR. FOR ELECTRIC POWER IN $/kW(t)
BRAUN AND LIDSKY	8.89×10^{-5}	67.0	89.3	107.0	226
LEE	9.51×10^{-5}	71.7	95.6	115.0	231
PARISH AND DRAPER-1	9.33×10^{-5}	70.3	93.7	112.0	229
PARISH AND DRAPER-2	1.025×10^{-4}	77.1	103.0	123.3	236
LONTAI	4.49×10^{-5}	33.8	45.1	54.1	193
LIDSKY	2.02×10^{-4}	152.0	203.0	243.0	311
MANISCALCO AND WOOD	6.75×10^{-5}	50.8	67.7	81.3	210

The fast thorium system of Parish and Draper has a higher fissile production rate per unit of thermal power when compared to the uranium system of Parish and Draper. This results in higher potential income for this design per unit of thermal power for all cases considered.

The high multiplication and thermal system of Maniscalo and Wood apparently must have a lower potential income than either the low multiplication fast uranium systems or the design of Lidsky. The reason for this is that it takes relatively little advantage from ^{238}U fast fission. As ^{238}U is only fissionable in special spectrums, one is not penalized (economically) for its use compared to ^{235}U which is fissionable in many spectra.

Lontai's design apparently has the lowest potential income. This is due primarily to the relatively low ^{238}U absorption in his particular concept.

Lidsky's design, according to this analysis, has the highest potential income. This comes about because of his much higher ratio of fertile absorptions per unit of energy. This, in turn, comes about because he has designed for zero fissions and depends solely on fusion for primary energy creation. Thus, he releases many more neutrons per unit of energy. This results in a much higher allowable capital cost.

In general, the fast uranium systems are apparently equivalent to a fast breeder reactor, analyzed in the same way, with a conversion ratio of 3.24. (Using a capital cost of $70/kW(t)-hr, a representative value for the fast system at 15 $/gm.) Lidsky's device is comparable to a fast breeder reactor with a breeding ratio of 5.86. All machines analyzed here are apparently more attractive as fuel factories than the fast breeder reactor at credible fast reactor conversion rates (i.e., $C_R \leq 1.5$). In summary, the higher the conversion ratio, the higher the permissable capital cost.)

What are the chances of the hybrid reaching these capital cost values? The Princeton design, a pure fusion reactor, is estimated to cost \$230/kW(t) with no electrical generating equipment.[22] The maximum competitive CTR capital cost for CTR fuel factory was estimated based on Deonigi's[23] analysis. It must be less than \$134/kW(t) in 1974 dollars as a mature design based on a LMFBR capital cost of \$460/kW(e). Thus, apparently a fuel factory of Lidsky's design has an opportunity to be economically attractive if the capital cost goal of the fusion program can be met. If this goal cannot be met, none of the fuel factory concepts appear attractive.

The income from the sale of both fissile fuel and power shown in Table II is useful in the following way. Assuming that one wishes to achieve additional income from a fuel factory by selling power directly, one can develop the impact of this additional source of revenue on the allowable capital cost. Thus, Lidsky's design has the highest potential income but the allowable capital cost increase is only a factor of two from selling power, much of which could be cancelled by having to buy electrical generating equipment, (traditionally, about 40% of the cost of a power plant). In addition, with no energy multiplication in the Lidsky design, it seems probable that the installed cost per kilowatt might be higher than the multiplying systems.

COST ANALYSIS OF HYBRID CONCEPTS

To complement the potential income analysis developed previously, preliminary cost estimates of the various hybrid concepts both as fuel factories and as combined fuel factory power producers are developed. For the magnetically confined devices, great reliance is placed on the cost analysis developed by Mills, et.al.[22] for the Princeton fusion power plant. For the laser approach, the values of Maniscalco[20] are used.

Certain costs generally would be invariant for a fusion device whether or not it is a hybrid machine. These costs are developed in Tables III and IV using the Atomic Energy Commission Guide for Economic Evaluation of Nuclear Reactor Plant Design and are taken in large part from reference 23.

Table III lists the invariant cost for the fuel factory approach where as Table IV lists the invariant costs for the power plant. The major difference between the two costs is the absence of the turbine generator system from the fuel factory concept.

The area of major variable costs apparently exists in Account No. 22, the reactor plant equipment. This cost is determined, in a crude sense, by a trade-off between the more costly blanket of the Hybrid and the possibly cheaper plasma confinement system.

Table V gives an estimate of the cost of the reactor plant equipment as a function of hybrid energy multiplication. The cost of the magnetic confinement system is a function of the square of the magnetic field and therefore is a strong function of the energy multiplication of the design. On the other hand, the cost of the fissile blanket is assumed to be a very weak function of the design energy multiplication. This results in the capital cost of the reactor plant appearing to approach a asymptotic limit at an energy multiplication of about 10.

TABLE III. Invariant Capital Costs for Hybrid as a Fuel Factory

Account No.		$/kW(t)
20	Land and Privilege Acquisition	.19
+21	Structures and Facilities	18.09
*23	Turbine Plant Equipment	4.46
24	Electric Plant Equipment	1.90
25	Miscellaneous Plant Equipment	3.05
+91	Construction Facilities, Equipment and Services	10.40
+92	Engineering Services	5.60
93	Other Costs	5.10
	Total	48.79

+Not truly invariant

*No electrical generation, but some necessary functions still performed here.

TABLE IV. Invariant Capital Costs for Hybrid as a Power Plant

Account No.		$/kW(t)
20	Land and Privilege Acquisition	.19
*21	Structures and Facilities	18.90
23	Turbine Plant Equipment	23.52
24	Electrical Plant Equipment	6.01
25	Miscellaneous Plant Equipment	3.05
*91	Construction Facilities, Equipment and Services	11.56
*92	Engineering Services	6.22
93	Other Costs	5.66
	Total	75.11

*Not truly invariant

TABLE V. Cost of the Reactor Plant Equipment as a Function of Hybrid Energy Multiplication in $/kW(t)

Account No.		Energy Multiplication Equals 1	Energy Multiplication Equals 2	Energy Multiplication Equals 5	Energy Multiplication Equals 10
22	Reactor Plant Equipment				
221	Nuclear Island				
221.1	Reactor Supports and Foundations	6.41	4.27	3.21	2.14
221.2	Magnet Systems				
221.21	Toroidal Field Magnets	11.65	3.0	.47	
221.22	Divertor Magnets	.5	.12	.02	
221.23	Vertical Field Magnets	1.24	.30	.05	
221.24	Control Field Magnets	.40	.1	.02	
221.26	Magnet Protection System	.80	.21	.03	
221.27	Power Supplies for Magnets and Protection Devices	6.33	.40	.1	
SUBTOTAL	ACCOUNT 221.2	20.97	4.13	.69	.2
221.3	Dewar and Refrigeration System	15.58	3.90	.62	.15
221.4	Coolant Inventory				
221.41	Cryogenic Coolant	.36	.09	.01	
221.42	Reactor Coolant	.08	.08	.08	
SUBTOTAL	ACCOUNT 221.4	.44	.17	.09	.09
221.5	Divertor and Vacuum Vessel	5.66	3.77	2.83	1.89
221.6	Shields, Blankets and Cooling				
221.61	Breeding Blanket	26.89	26.89	26.89	26.89
221.62	Shielding Blanket	3.51	3.51	3.51	3.51
221.63	Other Biological Shields	.14	.14	.14	.14
221.64	Blanket Cooling System	16.08	16.08	16.08	16.08
SUBTOTAL	ACCOUNT 221.6	46.62	46.62	46.62	46.62
221.7	Fuel Injection System				
222	Main Heat Transfer and Transport Equipment	33.49	33.49	33.49	33.49
225	Deuterium and Tritium Handling System and Inventory	1.49	.75	.30	.14
226	Other Plant Equipment	1.24	1.24	1.24	1.24
227	Instrumentation and Controls	2.00	2.00	2.00	2.00
TOTAL ($/kW(t))	ACCOUNT 22	133.90	100.34	91.09	87.96

In Table VI, a capital cost summary for each of the design concepts is given. This summary is given in present dollars and includes the cost of interest during construction. For each design, the capital cost summary is developed based on the neutronic properties of the device and the values in Tables III, IV, and V. The exception is the design of Maniscalco and Wood. In this instance, the capital cost figures were directly available from Reference 20 and only needed to be placed in the required format.

SUMMARY

The results shown in Table II permits relative comparisons of the possible income which can come from variously proposed hybrid reactor concepts both as fissile fuel factories and as producers of both fissile fuel and power. Both as fuel factories and as power plants, the Lidsky design has the highest potential income per installed unit of power. Then as a group come the fast lattice designs. Next comes the thorium design of Parish and Draper. The high multiplication device of Maniscalco comes fourth and the lowest is Lontai's design. Apparently, Lidsky's design has an opportunity to be successful in the role of a fuel factory if the unit cost goal for fusion reactor proposed by Deonigi can be met. A hybrid fuel factory apparently should be designed for a few fissions as possible, which by extension implies the use of the thorium cycle, an extremely thermal lattice, and short residence time for the fuel in the blanket.

TABLE VI. Capital Cost Estimates in Hybrid Designs in $/Kw(t)

AUTHORS	FUEL FACTORY	POWER PLANT
BRAUN AND LIDSKY	170	200
LEE	170	200
PARISH AND DRAPER-1	170	200
PARISH AND DRAPER-2	190.00	230
LONTAI	180	210
LIDSKY	220	250
MANISCALCO AND WOOD	190	240

When considering Hybrid design for production of both power and fissile fuel (i.e., low multiplication hybrid), the assumption of constant fissile blanket cost may be incorrect due to power density considerations. Considering the available fission technologies, the fast neutron lattices (both gas and liquid metal fueled) tend to allow for higher blanket power densities than the thermal systems (both gas and molten salt cooled).

A gas cooled fast breeder reactor lattice can have a power density of 258 kW/liter(24). A liquid metal fast breeder reactor can have a power density of 215 kW/liter(25). A molten salt breeder reactor can have a power density of 22.5 kW/liter(26). In Table I, the fast gas lattice designs and the liquid metal designs have power densities ranging from 300 to 30 kW/liter. The molten salt concept has a much lower power density. Thus, one can develop from Table II, last column, the maximum possible capital cost of each concept, for the sale of both power and fissile fuel, per liter of active core. If this is done, one finds that there is an apparent preference for the high power density concepts.

Considering Table VI, one finds that Lidsky's design is most expensive and that the fast systems are apparently the least expensive. If Maniscalco's design is eliminated on the grounds that its costs were developed in a manner which is possible inconsistent with the others, one finds a decreasing unit capital cost for energy multiplications out to about 10 and then a leveling of capital cost. On the basis of these speculative estimates, none of the fuel factories appears attractive. This is particularly true if one considers the cost of electrical power at 7.96 mills/kW (e) required for fuel factory operation. When considered as power plants, both the fast system and Lidsky's system appear attractive.

On the basis of the preliminary speculative analysis given, a number of conclusions have been made:

- A fissile fuel factory appears most attractive when designed for zero fissions (not truly possible) and which, therefore, utilizes the thorium cycle.
- None of the fuel factory concepts appear to be attractive based on present cost estimates.
- The justification for high multiplication systems apparently rests on the possible need for these systems if a Lawson plasma is never achieved and thus energy is required for input to the plasma device.
- The most attractive hybrid is apparently a combined fuel factory and power plant which is designed for no fissions, and, if that proves not to be possible in a practical sense, then the most attractive hybrid is the fast system optimized for ^{238}U fissions.

The last conclusion points out that there is strong economic incentive to build the hybrid which produces a high neutron population per unit of installed power. In other words, the economic potential for a hybrid apparently rests on its possible ability to approach a fast breeder reactor as a producer of power while at the same time surpassing it as a breeder of fissile fuel.

In performing this study, a number of things became apparent which, while not constituting technical conclusions, could prove instructive in future studies. First of all, the results given here must be called speculative because of the need to accept all results uncritically. Thus, more design studies need to be performed by a number of different investigators in order that points of commonality, such as occurred in the fast systems, can be identified. Secondly, the apparent ability to analyze diverse designs on a common basis by identifying various designs with their energy multiplication is most interesting. While the absolute values of both the income potential and capital cost will likely change with more detailed future analyses, the relative variations of performance with design energy multiplication might prove to be relatively fixed. If this is confirmed by future work, it will prove to be useful in directing the development of the hybrid reactor.

REFERENCES

1. B. R. Leonard, Jr., "A Review of Fusion-Fission (Hybrid) Concepts," Nucl. Tech., vol. 20, pp. 161-178, December 1973.

2. W. C. Wolkenhauer, B. R. Leonard, Jr., A. M. Sutey, PNL; R. W. Moir, LLL, "Conceptual Design of a Fusion-Fission Hybrid Reactor based on a Mirror Fusion Reactor with a Subcritical Gas Cooled Fission Blanket," Presented at the First Topical Meeting on the Technology of Controlled Fusion, San Diego, California, April 16-18, 1974, BNWL-SA-4865, Battelle-Northwest, Richland, WA, 1974.

3. F. Powell, Proposal for a Driven Thermonuclear Reactor, LWS-24920, California Research and Development Co., October 6, 1953.

4. L. M. Lidsky, "Fission-Fusion Symbiosis: General Considerations and a Specific Example," Presented at the BNES Nuclear Fusion Reactor Conference, Culham Laboratory, September 1969.

5. M. Lubin, Newsweek Magazine, May 27, 1974.

6. R. Horoshko and H. Hurtwitz, Application of Laser Fusion to the Production of Fissile Materials, Report 19, Laboratory for Laser Energetics, University of Rochester, 1974.

7. Two Component Torus Joint Conceptual Design Study, Vol. I, p. 14; Vol. II, pp. 1-5, 1-71, 2-2, Plasma Physics Laboratory, Princeton, New Jersey, and Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, June, 1974.

8. AEC Gaseous Diffusion Plant Operations, ORO-658, February 1968.

9. D. E. Deonigi, R. L. Watts, W. J. Bailey, and D. F. Newman, The Value of 233U in High Temperature Gas Reactors; 1980-2000, BNWL-1854, September 1974.

10. P. Fortesque, Fast Breeders and High-Temperature Gas Cooled Reactors: A Profitable Partnership, Gulf-GA-A10966, February 1972.

11. Liquid Metal Fast Breeder Reactor Program Plan, WASH-1101, August 1968.

12. Reactor Fuel Cycle Costs for Nuclear Power Evaluation, WASH-1099, December 1971.

13. Proceedings of the First Topical Meeting on the Technology of Controlled Nuclear Fusion, Volumes I and II, CONF-74042-P1, CONF-740402-P2, San Diego, California, April 16-18, 1974.

14. G. W. Braun and L. M. Lidsky, "Geometry and Performance of a Theta Pinch Power Breeder," Presented at the first Topical Meeting on the Technology of Controlled Nuclear Fusion, San Diego, California, April 16-18, 1974, Vol. I, CONF-74042-P1, pp. 199-210.

15. J. D. Lee, "Neutronic Analysis of a 2500 MW_{th} Fast Fission Natural Uranium Blanket for a DT Fusion Reactor," Presented at the First Topical Meeting on the Technology of Controlled Nuclear Fusion, San Diego, California, April 16-18, 1974, Vol. I, CONF-74042-P1, pp. 223-237.

16. Theodore A. Parish and E. Lynn Draper, Jr., "Neutronic and Photonic Analyses of Fusion Reactor Blankets Containing Natural Uranium," Presented at the First Topical Meeting on the Technology of Controlled Nuclear Fusion, San Diego, California, April 16-18, 1974, Vol. I, CONF-74042-P1, pp. 256-270.

17. Theodore A. Parish and E. Lynn Draper, Jr., "Neutronic and Photonic Analysis of Fusion Reactor Blankets Containing Thorium," Proceedings of the Fifth Symposium of Engineering Problems of Fusion Research, IEEE Pub. No. 73 CH0843-3-NPS, Princeton University, November 5-9, 1973.

18. Laszlo N. Lontai, Study of a Thermonuclear Reactor Blanket with Fissile Nuclides, MS Thesis, Technical Report No. 436, Massachusetts Institute of Technology, 1965.

19. L. M. Lidsky, "Fission-Fusion Symbiosis: General Consideration and a Specific Example," Presented at the BNES Nuclear Fusion Reactor Conference, Culham Laboratory, September 1969.

20. J. A. Maniscalco and L. L. Wood, Advanced Concepts in Fusion-Fission Hybrid Reactors, UCRL-75835 (Preprint) Lawrence Livermore Laboratory, July 1974.

21. 1973 Annual Report, Jersey Central Power and Light Company, Morristown, N. J.

22. R. G. Mills, editor, A Fusion Power Plant, MATT-1050, Princeton Plasma Physics Laboratory, October 1974.

23. D. E. Deonigi and J. R. Young, CTR Economic Regimes, Pacific Northwest Laboratory Interim Controlled Thermonuclear Reactor Technology Report - January-June 1974, August 1974.

24. Presentation to ACRS Subcommittee and DRL Staff of GCFBR Demonstration Plant, Gulf General Atomic Company, July 21, 1971.

25. Westinghouse Letter WZ00Z1, to W. T. Sha and D. S. Rowe, PNL, April 15, 1974.

26. Conceptual Design Study of a Single-Fluid Molten-Salt Breeder Reactor, ORNL-4541, June 1971.

COO-2218-25

Appendix B

ENERGY BALANCES FOR FUSION-FISSION HYBRIDS

by

G. H. Miley

Nuclear Engineering Program
University of Illinois
Urbana, Illinois 61801

December 27, 1974

This report was prepared as a contribution to the Proceedings of the Meeting on Fusion-Fission Hybrids held at AEC headquarters, December 3-4, 1974.

Abstract

Graphs are presented to evaluate the effect of blanket energy multiplication on fusion plasma requirements and overall plant efficiency. An example is given where a fusion plasma characterized with a Q_p-value corresponding to about 1/6th of the Lawson breakeven value and about 1/7th of the classical confinement value is able to drive a well-subcritical uranium blanket and still provide reasonable overall plant efficiencies for electrical production.

ENERGY BALANCES FOR FUSION-FISSION HYBRIDS

Introduction

An inherent feature ascribed to fission-fusion hybrids is that the fusion plasma can operate below the classical Lawson breakeven limit. This has the important implication that a hybrid reactor could conceivably be constructed before technology has developed to the point where a net power output from a pure fusion reactor is possible. This advantage alone does not seem to be sufficient reason to pursue hybrid development, however. Rather it is imperative that the hybrid be economically competitive within its own right, and this hinges on the success of applications such as breeding fissional material or burning fission waste products. Still, the interrelation between fusion plasma requirements and blanket multiplication is an important factor in system performance. Thus the present report is concerned with the derivation of some simple equations and graphs that describe these relations for use in survey calculations. For this purpose, we would like to relate the fusion-plasma energy-multiplication factor (Q_p) to the blanket multiplication (M_B) and overall plant efficiency (η_o).*

I. Overall Energy Balances For Fusion Systems

Before considering hybrids, we will first review some general balances for a pure fusion reactor. This provides a basis for comparison and establishes the notation.

The energy flow in a fusion plant are illustrated in Figure 1. Both the fusion and injected energies ($E_f + E_I$) are converted to electricity at

---

*It should be noted that in some concepts the hybrid would be used for breeding (or waste burning) without a net electrical output. This would correspond to an overall efficiency η_o of zero which is a subcase of the following analysis. On the other hand, if simultaneous production of electricity is desired, it is important to evaluate the overall plant efficiency as well as fusion plasma requirements.

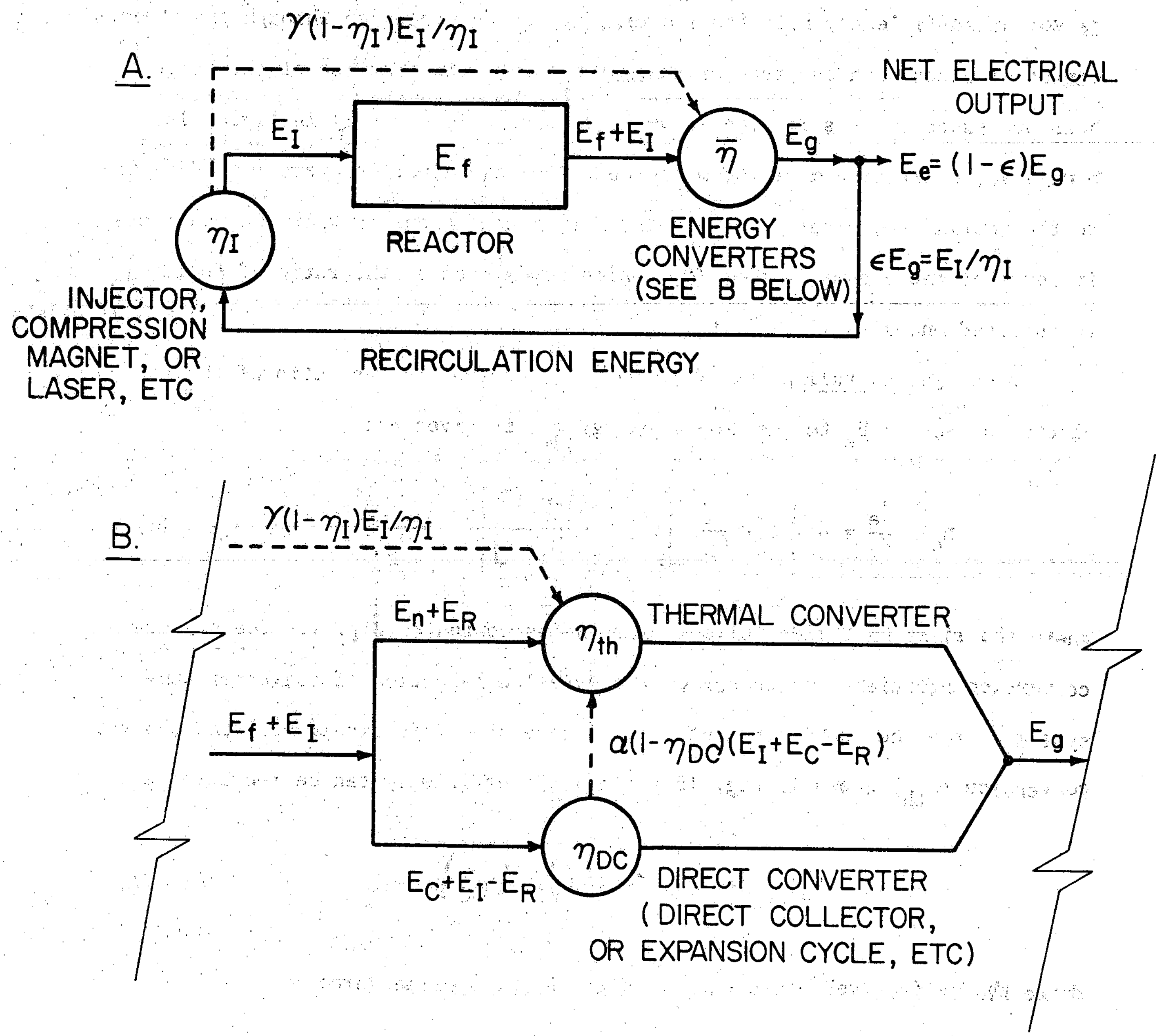


Fig. 1 General Energy Flow Diagram. Sketch A shows overall flows while B indicates possible use of a direct converter for charged particles coupled with a thermal converter for neutron and radiation energy.

an average efficiency $\bar{\eta}$, and a fraction of the gross output E_g is recirculated through an "injector" of efficiency η_I.* If the waste heat from the injector leaves at sufficiently high temperature, it can be processed through the thermal converter along with the reactor thermal output. The fraction of the waste heat processed in this fashion is designated as γ ($o \lesssim \gamma \lesssim 1$) in Figure 1A. Similarly, a fraction α of the waste heat from the direct converter is directed to the thermal converter in Figure 1B. It is convenient to write the balances in terms of the plasma Q-value (Q_p) which is defined as the ratio of fusion to injected energies, i.e. E_f/E_I.

Then, the overall plant efficiency η_o, defined as the ratio of the net electrical output E_e to the fusion energy E_f, is given as:

$$\eta_o \equiv \frac{E_e}{E_f} = \bar{\eta}\left\{1 + \frac{1}{Q_p}\left[1 + \gamma\frac{(1-\eta_I)}{\eta_I}\right]\right\} - \frac{1}{\eta_I Q_p} \qquad (1)$$

where the right hand side follows from an inspection of Fig. 1. The average conversion efficiency $\bar{\eta}$ depends on the detailed selection of converter subsystems. For the combination of direct conversion (efficiency η_{DC}) and thermal conversion (η_{th}) shown in Fig. 1B, the overall efficiency can be rewritten as:

$$\eta_o = (1 - \Psi_R)\eta_{th} + \Psi_R\eta_{DC}' - \frac{1}{Q_p}\left(\frac{1}{\eta_I'} - \eta_{DC}'\right) \qquad (2)$$

where the "effective" efficiencies indicated with primes are:

$$\eta_{DC}' \equiv \eta_{DC} + \alpha\ (1 - \eta_{DC})\ \eta_{th};\ o \lesssim \alpha \lesssim 1 \qquad (3a)$$

---

*The term injector is used here in a general sense to represent any subsystems that return energy to the reactor.

$$\eta_I' \equiv \eta_I \Big/ \left[1 - \gamma\,(1 - \eta_I)\eta_{th}\right] \; ; \; 0 \lessapprox \gamma \lessapprox 1 \tag{3b}$$

Also, Ψ_R, termed the "radiation parameter", is defined as

$$\Psi_R \equiv (1 - \chi_R) f_c \tag{3c}$$

where f_c is the fraction of fusion energy going into charged particles, i.e., E_c/E_f, and χ_R is the ratio of the energy E_R going into radiation to the original fusion energy E_c going into charged particles.*

For convenience some typical ranges of Ψ_R for various fuels are indicated in Table 1. For negative Ψ_R a limit occurs on Q_p because the radiation energy cannot exceed the sum of the charged-particle plus injection energies, i.e.

$$Q_p \lessapprox \frac{1}{|\Psi_R|} \qquad\qquad \Psi_R < 0 \tag{4}$$

which simply states that sufficient injection energy must be supplied to make up for radiation losses.

A plot of η_o vs. Q_p is shown in Figs. 2A and B for various values of the radiation parameter Ψ_R and for some "typical" values of the thermal, direct conversion, and injection efficiencies.

II. Blanket Energy Multiplication and Fusion-Fission Hybrids

Nuclear reactions induced by neutrons in the blanket of a fusion reactor can represent a significant additional energy source. Depending on the materials

---

*Note that $\Psi_R = 0$ corresponds to ideal ignition such that radiation losses from the plasma are just made up by charged-particle input. To achieve actual ignition, energy losses by plasma leakage must also be supplied by charged-particle input, thus $\Psi_R > 0$. On the other hand, with injection it is possible to convert some of the injected energy to radiation energy along with the charged-particle energy. In that case $\Psi_R < 0$, but since continuous injection is required to maintain operation, such a reactor is often dubbed a "Wet-Wood Burner" (WWB).

Table 1

Range for the Radiation Parameter ψ_R

Fuel		f_c from Table 2.1	Approximate Range of ψ_R *
D-T (Li Blanket)		.159	-.08 to 0.159
catalyzed D-D		.617	-.31 to 0.617
$D\text{-}He^3$		.97	-.48 to 0.97
$p\text{-}Li^6$, $p\text{-}B^{11}$	exotic fuels	1.0	-.50 to 1.0

* Assumes E_R/E_f ranges from 0 to 1.5. In the latter case all of the fusion energy plus additional injected energy is radiated away.

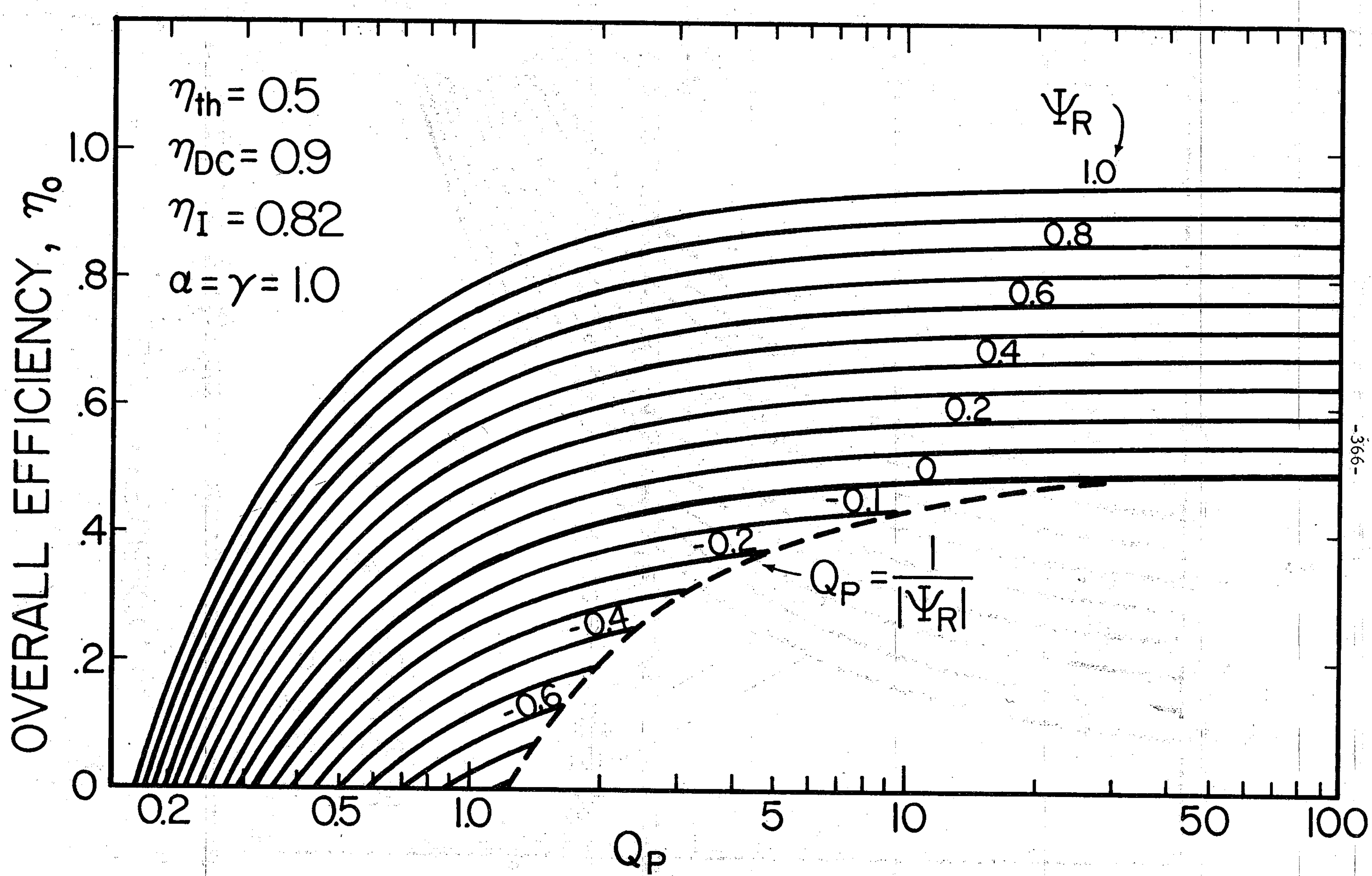


Fig. 2A Overall efficiency vs. the plasma Q_p-value for various levels of radiation emission indicated by Ψ_R.

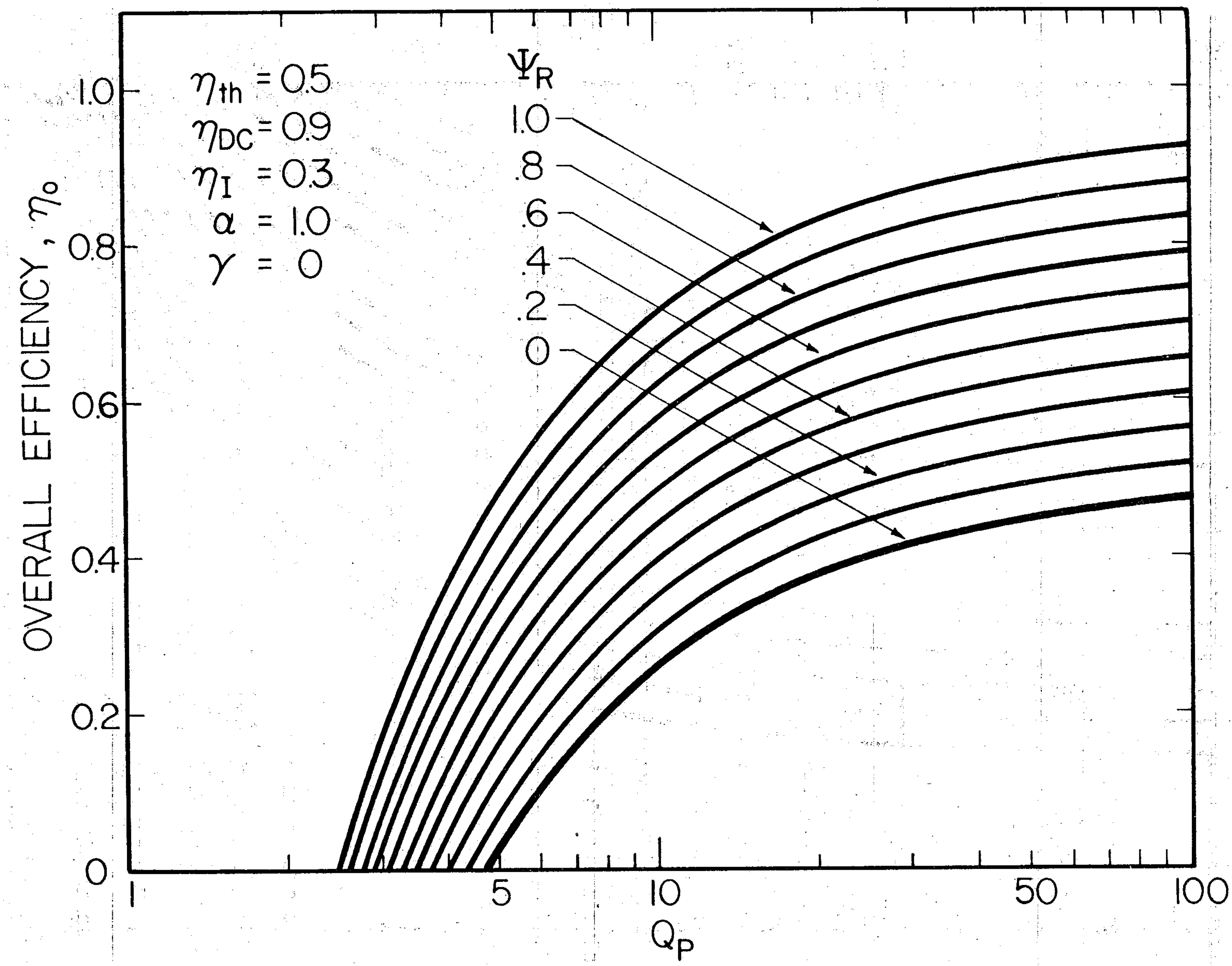


Fig. 2B Overall Efficiency vs. Q_p with a Lower Injection Efficiency Characteristic of a Laser-Pellet System. Since it may be difficult to reclaim the low-temperature heat rejected from the laser, γ is assumed to be zero.

and coolant employed, fusion neutrons entering the blanket can produce additional neutrons through various n,2n reactions or fission. These neutrons, as well as the virgin fusion neutrons, also provide energy multiplication by undergoing exothermic nuclear reactions.

For example, in the conventional D-T reactor some neutron multiplication occurs via n,2n reactions in the niobium first wall. Tritium breeding occurs by both $Li^6(n,\alpha)T$ and $Li^7(n,n',\alpha)T$ reactions and the former, being exothermic, also provides some energy multiplication. To account for this in a crude fashion, it has been traditional to use an energy release value of $\sim$ 22.4 MeV/fusion rather than the 17.6 MeV actually involved in D-T fusion. Where additional neutron multiplication is desired, some conceptual blanket designs have incorporated beryllium, either in solid form or in a coolant such as the molten salt "Flibe", (Li_2BeF_4). For designs where breeding is not critical, materials such as sodium or aluminum might be added.

For generality we will incorporate the added blanket energy through an explicit multiplication factor for the blanket, M_B, defined as:

$$M_B \equiv \frac{\text{total energy released in blanket}}{\text{fusion neutron energy entering}} = \frac{E_B}{E_n} \tag{5}$$

Conventional blankets may achieve M_B-values ranging from 1.3 to 1.8, depending on the design. Much larger multiplications are possible using fissionable materials in the blanket, and such reactors are termed fusion-fission hybrids.

For non-enriched blankets, maximum M_B-values ranging from 4 to 22 can be obtained as indicated in Table 2 where results are given for infinite size assemblies of pure thorium, U-238, and natural uranium.(1) However, with some enrichment, higher M_B-values are easily obtained even in a well subcritical blanket.

Table 2

Energy Multiplication and Fissile Breeding in Infinite Blankets*

Blanket Material	M_B	Breeding Reactions†/Fusion Neutron
Thorium	4.6	2.7 [$Th^{232}(n,\gamma)$]
U-238	16.6	4.4[$U^{238}(n,\gamma)$]
Natural Uranium	22.1	5.0 [$U^{238}(n,\gamma)$]

*From J. D. Lee, Ref. 1.

†Breeding reaction indicated in brackets.

For example, recent studies by Lee[1] and by workers at the Pacific Northwest Laboratories[2,3] have indicated M_B-values over 30 could be achieved with $k_{eff} \lesssim 0.9$ while also retaining attractive fissile fuel and tritium production.

A convenient method of estimating M_B is possible if k_{eff}, the effective neutron multiplication factor for the blanket, is known. [This follows normal fission reactor convention where $k_{eff} = 1$ defines criticality so that $k_{eff} < 1$ represents a subcritical assembly, the case of interest here.*] Each fusion neutron entering the blanket will produce, on the average, $(1 - k_{eff})^{-1}$ neutrons. We will assume that lithium is the main material other than the fissionable inventory that contributes to energy multiplication. Then, defining β_L as the fraction of all blanket neutrons absorbed in lithium, β_A as the fraction absorbed without significant energy multiplication, and Q_L and Q_f as the average energies (MeV) released per absorption in lithium and fission, respectively, we have:

$$M_B \sim [\beta_L G_L + (1-\beta_L-\beta_A)G_f]/(1 - k_{eff}) \tag{6}$$

The energy gains G_L and G_f associated with the lithium and fissionable materials are [Based on an average value of Q_L of 3 MeV to account for both Li^6 and Li^7 reactions and assuming $Q_f \sim 180$ MeV.]

$$G_L \equiv 1 + Q_L/14.1 \approx 1.2 \tag{6a}$$

$$G_f \equiv 1 + Q_f/14.1 \approx 13.9 \tag{6b}$$

For example, for the case considered by Lee in Ref. 1, $\beta_L \sim 0.19$, $\beta_A \sim 0.53$, and $k_{eff} = 0.84$, giving $M_B \sim 26$, slightly lower than the value of 30.6 he obtains from computer calculations.

---

*As seen from Eq. (6), M_B can be made arbitrarily large by letting $k_{eff} \to 1$, but this violates the safety feature of a well-subcritical assembly.

The effect of M_b on the energy flow in a fusion plant is illustrated in Fig. 3. The overall plant efficiency η_o is now defined as the ratio of the net electrical output divided by the sum of the fusion plus fission energies. Based on this figure the equivalent of Eq. (2) is found to be:

$$\eta_o \equiv \frac{E_e}{E_c + M_b E_n} = \frac{(1+\psi_B-\psi_R)\eta_{th} + \psi_R \eta'_{DC} - \frac{1}{Q_p}\left(\frac{1}{\eta'_I} - \eta'_{DC}\right)}{1 + \psi_B} \tag{7}$$

where all symbols are as before but now the blanket parameter ψ_B is introduced such that

$$\psi_B \equiv f_n(M_b - 1) \tag{7a}$$

For reference some typical ranges of ψ_B are indicated in Table 3, and plots of η_o for several values of ψ_B are shown in Fig. 4 where the other parameters are held the same as in Fig. 2.

It is observed that for smaller values of Q_p the overall efficiency is increased with the addition of a multiplying blanket. This is simply because the added fission power reduces the role of the fusion power and in particular reduces the detrimental effect of the large recirculation required in that part of the plant. With ψ_B very large, the fission energy so dominates that the plant is essentially equivalent to a conventional fission reactor. Whether or not an optimum plant lies somewhere between this extreme and a pure fusion reactor ($\psi_B \sim 0$) is still a subject provoking much controversy.

III. Plasma Q_p-value Requirements

If Eq. (2) is solved for Q_p, we obtain:

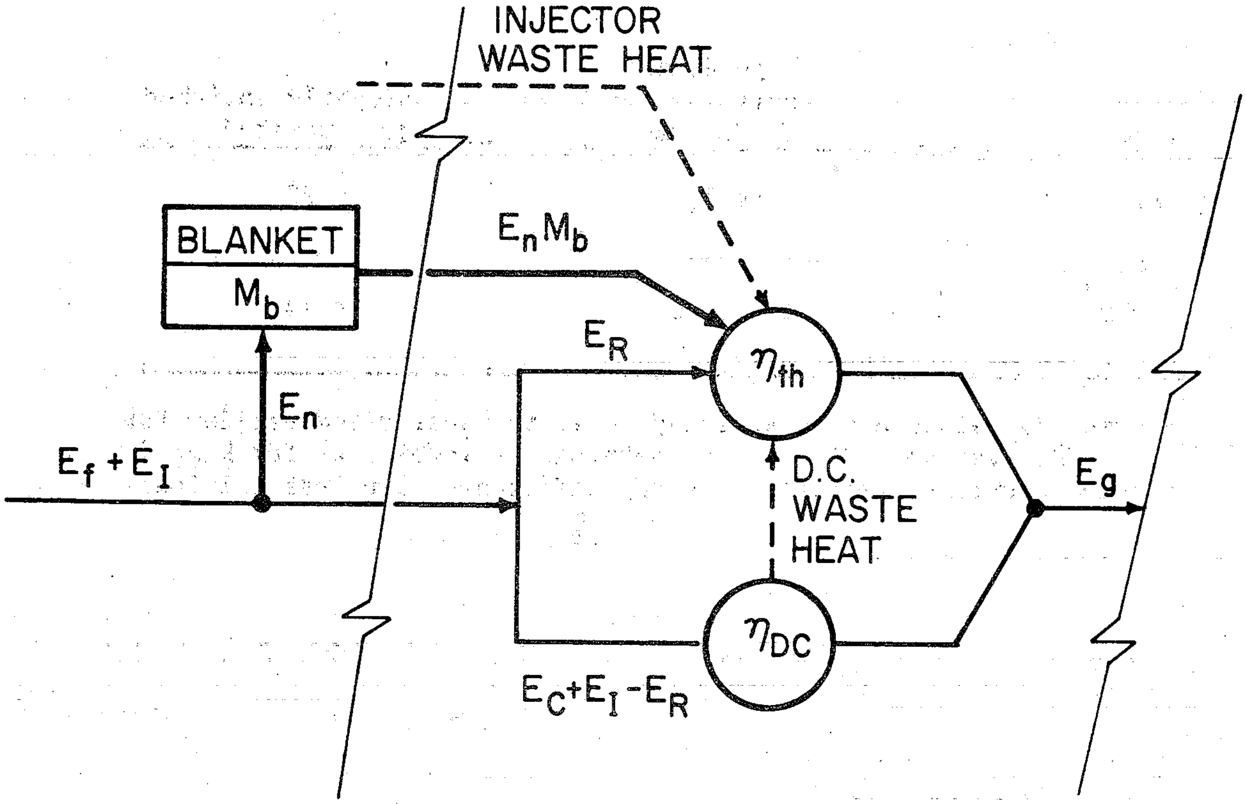


Fig. 3 Inclusion of Energy Multiplication M_b in the Blanket in the Energy Flow Diagram. The flow shown here replaces sketch B in Fig. 1 when blanket multiplication is employed.

Table 3

Approximate Ranges for the Blanket Parameter Ψ_B

Plasma	Type Blanket Pure Th, U-238, Nat-U	Slightly enriched or Moderated
DT	3 to 18	$\lesssim$ 30
Catalyzed DD*	1.5 to 9	$\lesssim$ 14

*Assumes M_B values are essentially equal to those cited earlier for D-T. The average energy of D-D neutrons is lower, but for $k_{eff} \gtrsim 0.7$, fission neutrons dominate so energy differences are less critical.

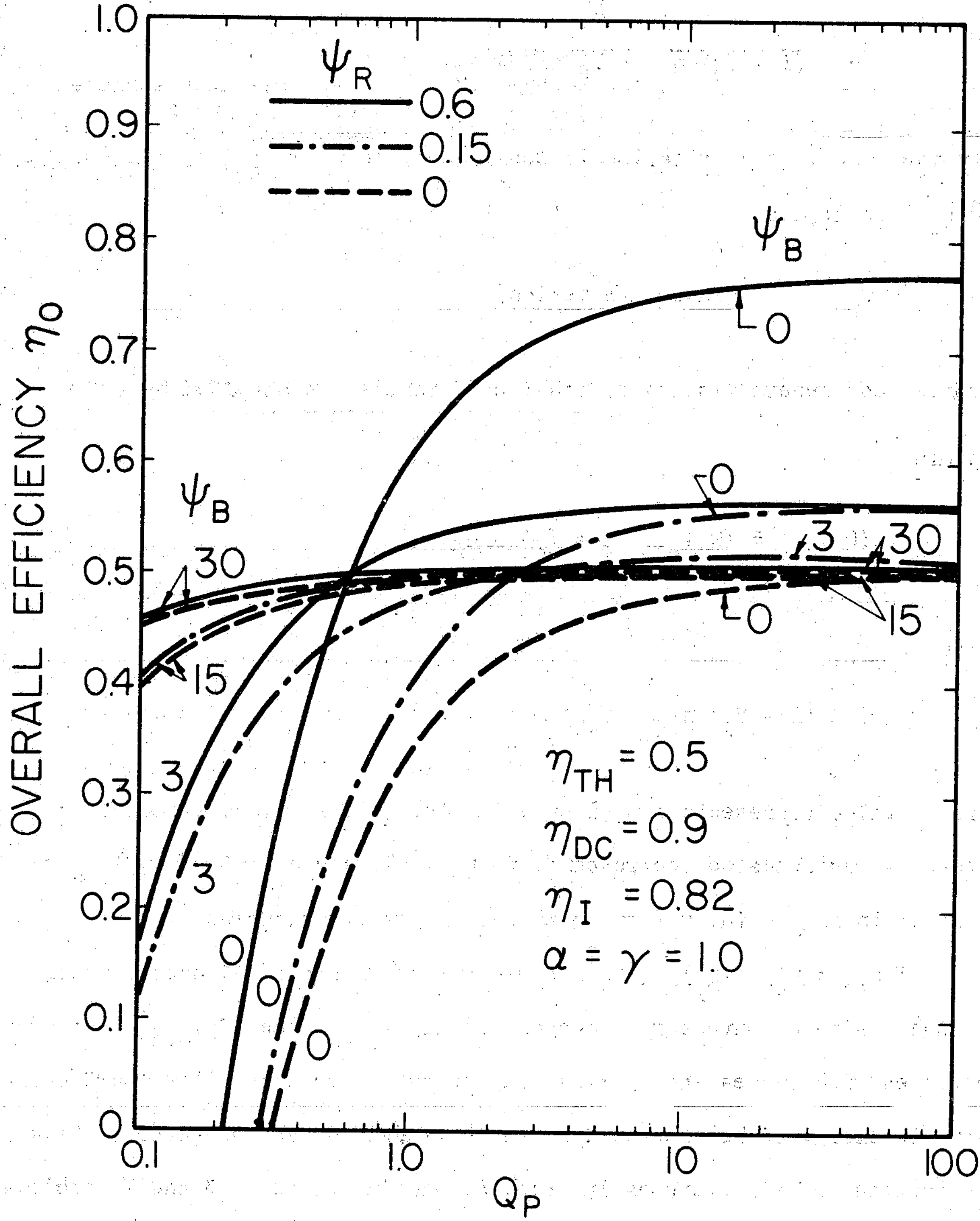


Fig. 4 Effect of Blanket Energy Multiplication on the Overall Efficiency of a Fusion-Fission Hybrid with $\eta_{th} = 0.5$, $\eta_{DC} = 0.95$, and $\eta_I = .9$ as in Fig. 2. The values of Ψ_R and Ψ_B were selected to roughly represent extremes for D-T and catalyzed D-D reactors.

$$Q_p = \frac{\frac{1}{\eta_I'} - \eta_{DC}'}{(1 - \Psi_R)\,\eta_{th} + \Psi_R\,\eta_{DC}' - \eta_o} \tag{8}$$

Note that the Lawson criterion is obtained by letting $\eta_o = 0$, $\eta_{DC}' = \eta_{th} = 1/3$, and $\eta_I' = 1$, giving

$$Q_p = 2 \quad \text{(Lawson Criterion)} \tag{9}$$

A convenient generalization to other efficiencies is obtained by defining $(Q_p)_{min}$ as

$$(Q_p)_{min} \equiv (Q_p)_{\eta_o = 0} = \left(\frac{1}{\eta_I'} - \eta_{DC}'\right)/\Delta \tag{10}$$

where

$$\Delta \equiv (1 - \Psi_R)\,\eta_{th} + \Psi_R \eta_{DC}' \tag{11}$$

This Q_p-value represents a minimum value for a given plant since all of the output is recirculated (corresponds to $\eta_o = 0$), and a plot of $(Q_p)_{min}$ vs. Ψ_R is shown in Fig. 5 for various combinations of efficiencies.

If $\eta_{th} = \eta_{DC}'$, $(Q_p)_{min}$ is independent of Ψ_R since all output is processed through a single converter. However, if $\eta_{DC}' \neq \eta_{th}$, the $(Q_p)_{min}$ requirement decreases with increasing Ψ_R (less radiation) since this allows more charged-particle energy to be processed by the efficient direct converter. Several combinations of efficiencies in Fig. 5, namely curves 2, 3 and 7, achieve more favorable (lower) $(Q_p)_{min}$-values than the classical Lawson criterion.

For a net power output ($\eta_o > 0$), we require $Q_p > (Q_p)_{min}$. Division of Eq. (8) by (10) gives:

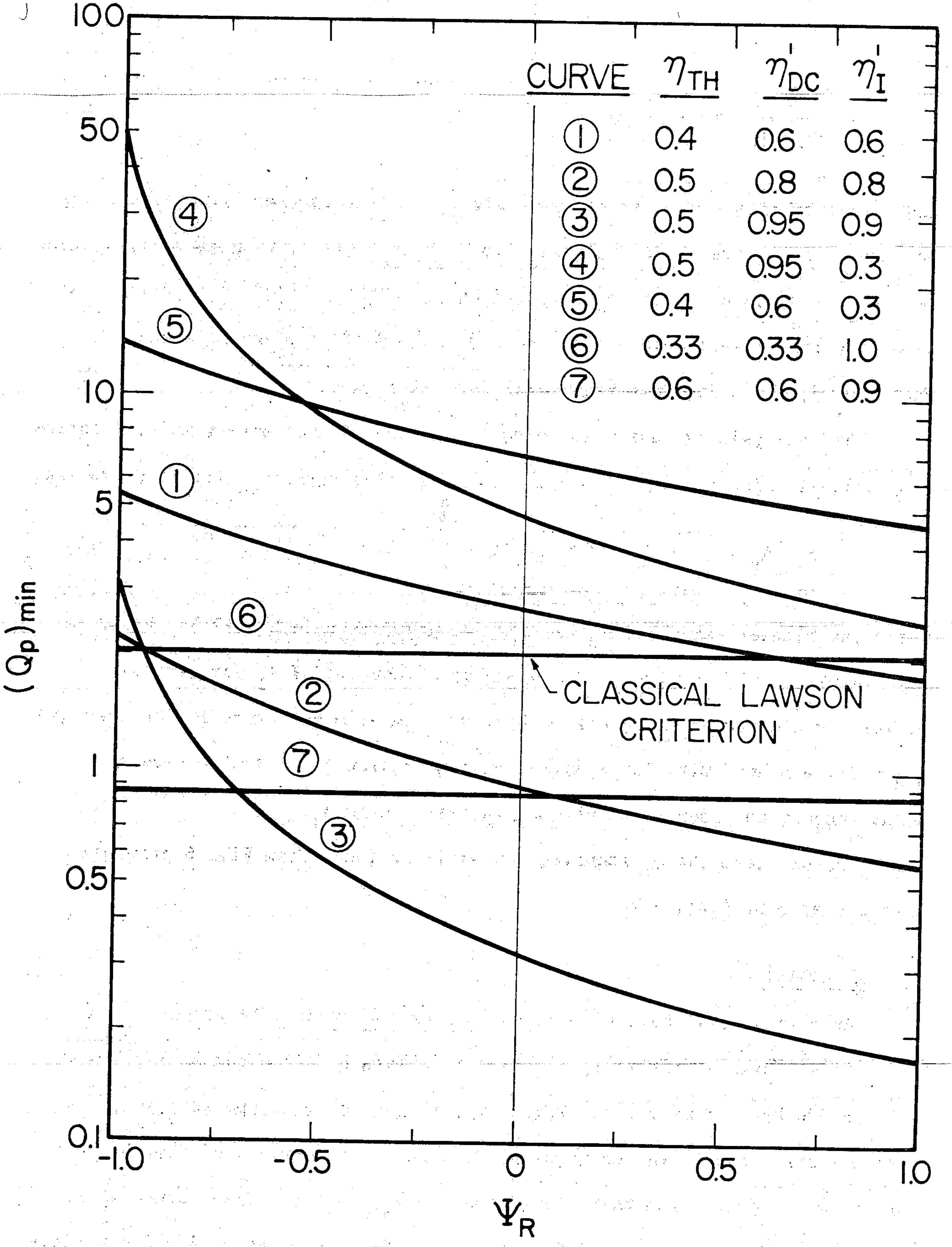


Figure 5 The Minimum Q_p-value vs. the Radiation Parameter Ψ_R.

$$\frac{Q_p}{(Q_p)_{min}} = \frac{1}{1 - \eta_o/\Delta} \tag{12}$$

and a plot of this result is given in Fig. 6. The parameter Δ represents the maximum possible overall efficiency, i.e., the efficiency with no recirculation ($\varepsilon = 0$). Thus in order for η_o to approach Δ, an infinite Q_p is required. As seen from the figure, unless the actual Q_p exceeds $(Q_p)_{min}$ by more than a factor of 5, the efficiency ratio will fall below 0.8.

This analysis is easily extended to include blanket energy multiplication and fusion-fission hybrids. In this case, Eq. (12) remains valid if we define:

$$\Delta \rightarrow \Delta_B = (1 + \Psi_B - \Psi_R)\eta_{th} + \Psi_R\, \eta'_{DC} \tag{13}$$

where the blanket parameter Ψ_B was defined previously in Eq. (7a). Based on this result, a plot of $(Q_p)_{min}$ is shown as a function of Ψ_B for typical efficiency and radiation levels in Fig. 7. Compared to a pure fusion reactor ($\Psi_B = 0$), a hybrid with large energy multiplication ($\Psi_B = 30$) is seen to allow roughly an order to magnitude reduction in $(Q_p)_{min}$.

In this case the Q_p required can still be found from Fig. 6 provided Δ is replaced by $\Delta_B/(1 + \Psi_B)$.

IV. An Example

An example will help clarify the use of the preceeding graphs.

First, assume that it is possible to obtain $Q_p \sim 2.4$ and $\Psi_R \sim 0.17$ using a 300 keV D-T mirror reactor. If this reactor were operated as a pure fusion system, and if the conversion efficiencies of curve 2 of Fig. 5 are assumed, we see from that figure that $(Q_p)_{min} \sim 0.9$. Then, since Q_p is 2.4, the $Q_p/(Q_p)_{min}$ ratio is 2.6, which, from Fig. 6, gives $\eta_o/\Delta = 0.6$. Since the efficiencies of curve 2 correspond to $\Delta = 0.55$, [see Eq. (11)] the overall

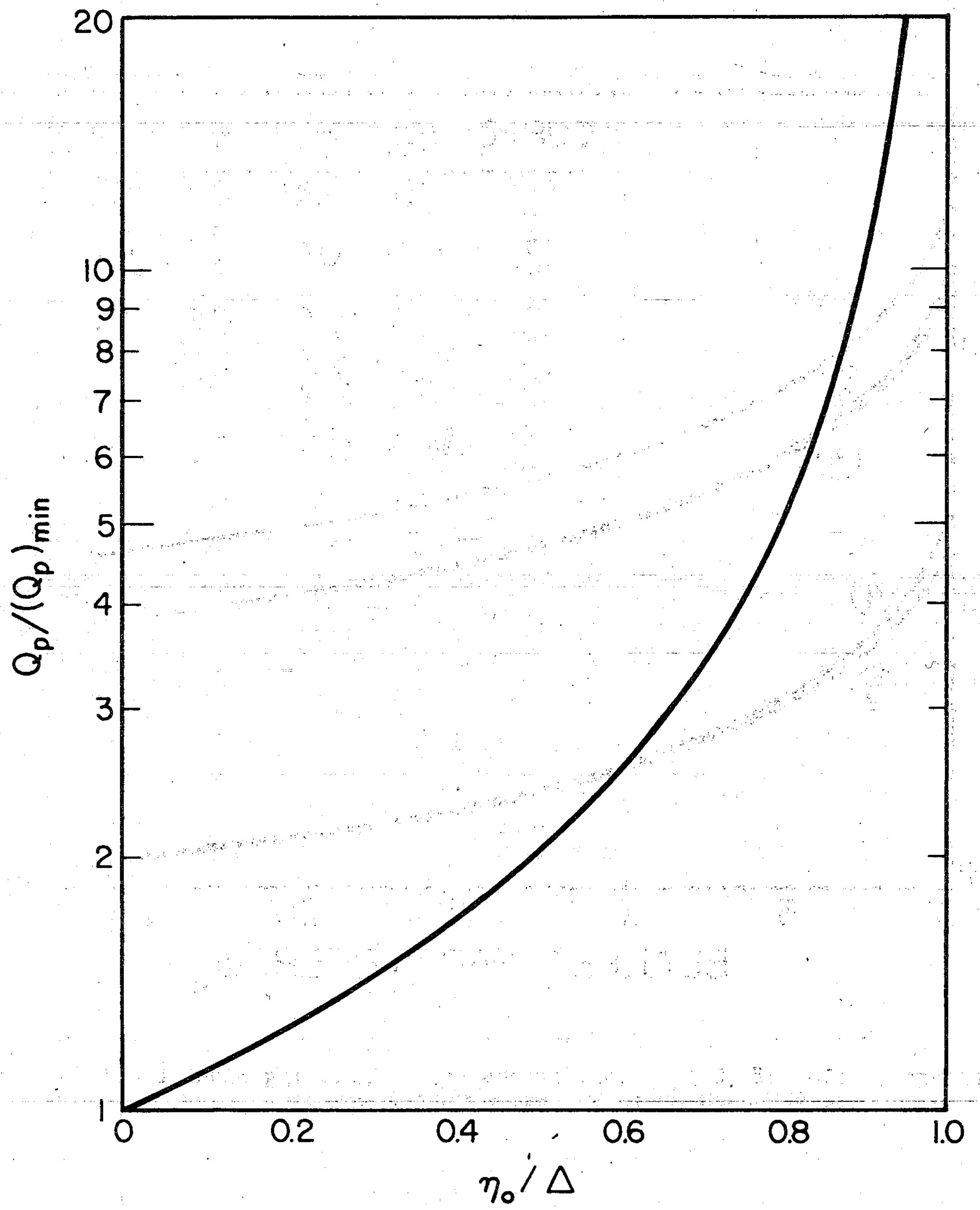


Figure 6 Ratio of the Required to Minimum Q_p to Achieve a Given Overall Efficiency η_o. For blanket multiplication the abscissa becomes $\eta_o(1 + \Psi_B)/\Delta_B^o$.

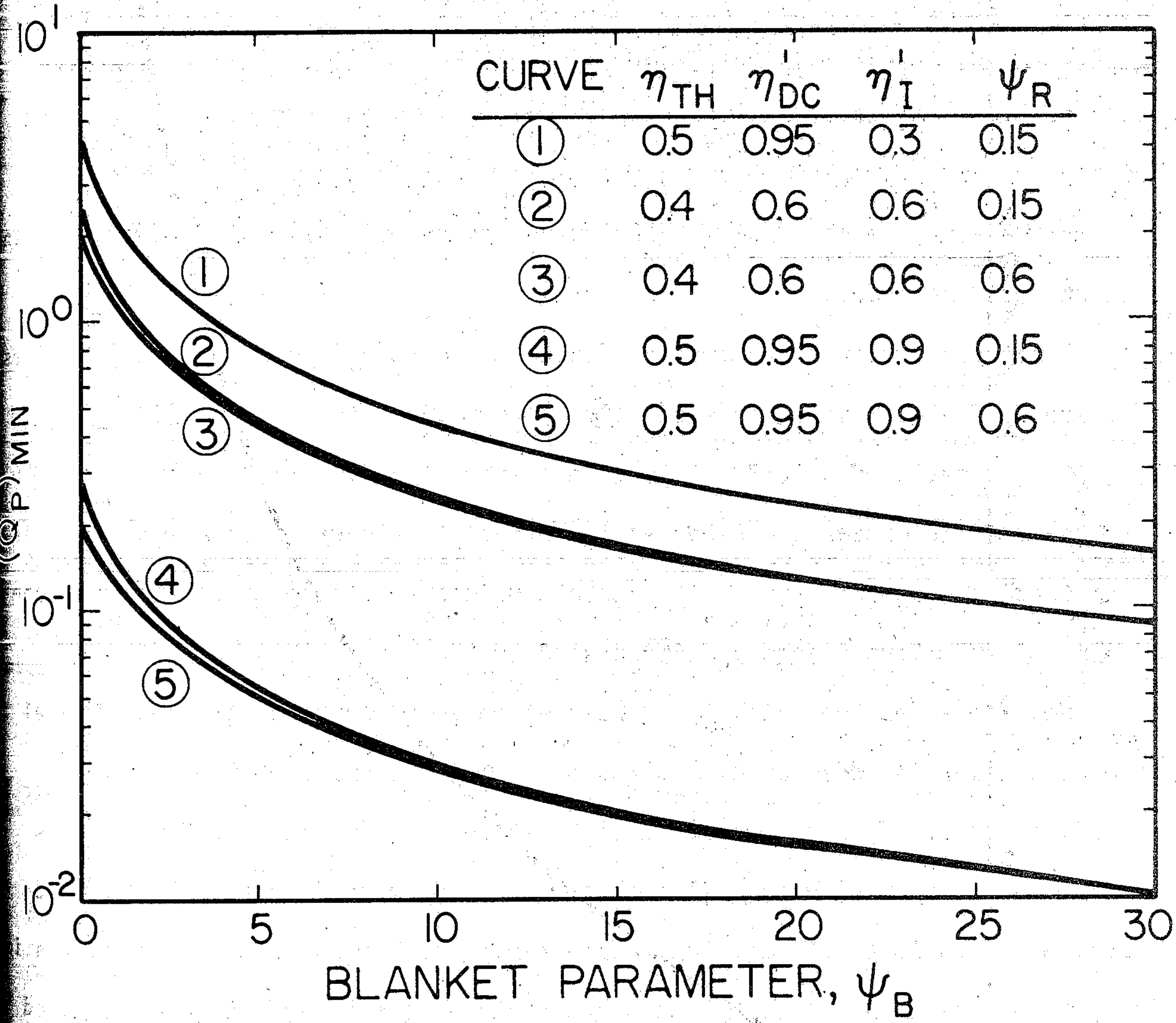


Figure 7 Plot of $(Q_p)_{min}$ for Various Cases Including Blanket Multiplication as Encountered in a Fission-Fusion Hybrid.

plant efficiency is 33%.

This represents a reasonable power plant; however, the Q_p value assumed is probably optimistic and could only be achieved with nearly classical plasma behavior. Assume for the moment that instabilities and anomalous losses cause Q_p to drop to 1/7th of this value or 0.34. Since this is below the $(Q_p)_{min}$ value of 0.9, the preceeding reactor could no longer operate with a net power production. Two courses of action are conceivable to attempt to avoid this dilemma; namely, the conversion efficiencies might be increased or a multiplying blanket could be added.

First consider the conversion efficiency route. If the quite high values of curve 3 of Fig. 5 were achieved, $(Q_p)_{min}$ would be lowered to ~ 0.3, giving a $Q_p/(Q_p)_{min}$ ratio of ~ 1.13. This gives, from Fig. 6, a value of η_o/Δ of ~ 0.1. With these efficiencies $\Delta = 0.58$, so the plant efficiency is only $\sim 6\%$. It is clear that this approach is unsatisfactory unless something could also be done to control anomalous losses and increase Q_p.

Next consider the multiplying blanket route. Assume that a subcritical uranium blanket is added with $k_{eff} = 0.8$, $\beta_L \sim 0.15$, and $\beta_A \sim 0.6$. Then, according to Eq. (6), the energy multiplication M_B is roughly 18.3. If this blanket is employed with the D-T mirror reactor discussed above, the blanket parameter Ψ_B is 13.8 [cf. Eq. 7a]. With the added blanket cost and complication, it might not be possible to maintain the high conversion efficiencies assumed in the pure fusion system. Thus selecting the more modest values of curve 2 in Fig. 7, we find $(Q_p)_{min} \sim 0.2$. Then, $Q_p/(Q_p)_{min} = 0.34/0.2 = 1.7$, giving, from Fig. 6, $\eta_o(1 + \Psi_B)/\Delta_B \sim 0.35$. For these efficiencies, Δ_B [from Eq. (13)] is 5.95, so the overall efficiency η_o is $\sim 15\%$. While this is low compared to normal plant operation, it could still be attractive when credit is taken for breeding of fissile materials etc. Thus operation might be possible despite the very sub-Lawson plasma [$Q_p \sim 0.34$ here vs. $(Q_p)_{Lawson} = 2.0$]. Further, several options would remain open for improving plant performance. As in the pure fusion plant, any increase in Q_p would have a dramatic effect. However, improvements are possible even with the poor Q_p. For example, if the ultra-high efficiencies assume earlier in the pure fusion plant were used here (cf. curve 4 of Fig. 7), the overall efficiency jumps to $\sim$ 46%. It would probably not be possible to go this high, but some increase would be quite feasible.

Some improvement would also be achieved by increasing k_{eff} of the blanket. It would be quite feasible (and probably desirable) to design for $k_{eff} \sim 0.9$ to 0.95. However, care must be exerted to keep the system well subcritical so that the safety advantage is retained. Actually, as seen from Fig. 7, from an energy balance point of view there is not much to be gained from increasing Ψ_B much above $\sim$ 15.* For example, assume k_{eff} is increased to 0.90 in the preceeding example. This gives $M_B \sim 37$ and $\Psi_B \sim 28.8$. Retaining the original modest conversion efficiencies (cf. curve 2 of Fig. 7), we find that $(Q_p)_{min} \sim 0.1$. Then from Fig. 6, $\eta_o(1 + \Psi_B)/\Delta_B$ is $\sim$ 0.7, and since $\Delta_B \sim 12$, we obtain $\eta_o \sim 28\%$.

This represents an attractive gain in efficiency, but it is clear that further increases in k_{eff} would result in only small increases in efficiency. Of course, here we are only considering one aspect involved in the selection of k_{eff}. The optimum selection would involve a balance between conversion efficiency, breeding ratio (or burning rate if the hybrid involved fission waste disposal), and safety.

In summary we have developed a series of interrelated graphs that can be used to investigate the performance and efficiency of a variety of reactor approaches and conditions. This should be most useful for survey calculations. Final precision evaluations will, however, require numerical solutions for the specific cases of interest.

*Similarily, Fig. 4 shows that η_o saturates at $\sim \eta_{th}$ as ψ_B increases above $\sim$15.

References

1. J. D. Lee, "Neutronics of Sub-Critical Fast Fission Blankets for D-T Fusion Reactors", UCRL-73952, Lawrence Livermore Laboratory, Livermore, California (June 1972).

2. B. R. Leonard, Jr., "A Review of Fusion-Fission (Hybrid) Concepts," Nuclear Tech., 20, 161-178 (1973).

3. B. R. Leonard, Jr. and W. C. Wolkenhauer, "Fusion-Fission Hybrids: A Subcritical Thermal Fission Lattice for a DT Reactor." Proc., Conf. on Technology of Fusion, Austin, Texas, Nov. 20-22 (1972).