180
1-19-77

Dr 635
UCRL-50043-1

MIRROR HYBRID REACTOR STUDIES - July 1975 through September 1976

J. D. Lee, Editor

October 8, 1976

Prepared for U. S. Energy Research & Development
Administration under contracts No. W-7405-Eng-48
(LLL) and AT(04-03)-167, Project Agreement 38
(General Atomic Company).

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Energy Research & Development Administration, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately-owned rights.

NOTICE

Reference to a company or product name does not imply approval or recommendation of the product by the University of California or the U.S. Energy Research & Development Administration to the exclusion of others that may be suitable.

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161
Price: Printed Copy \$: Microfiche \$3.00

Page Range	Domestic Price	Page Range	Domestic Price
001-025	\$ 3.50	326-350	10.00
026-050	4.00	351-375	10.50
051-075	4.50	376-400	10.75
076-100	5.00	401-425	11.00
101-125	5.50	426-450	11.75
126-150	6.00	451-475	12.00
151-175	6.75	476-500	12.50
176-200	7.50	501-525	12.75
201-225	7.75	526-550	13.00
226-250	8.00	551-575	13.50
251-275	9.00	576-600	13.75
276-300	9.25	601-625	*
301-325	9.75		

*Add \$2.50 for each additional 100 page increment from 601 to 1,000 pages;
add \$4.50 for each additional 100 page increment over 1,000 pages.

Distribution Category
UC-20

LAWRENCE LIVERMORE LABORATORY
University of California Livermore, California 94550

NOTICE
This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Energy Research and Development Administration nor any of their employees nor any of their contractors, subcontractors or third employees make any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information appearing hereon or for the equipment described or represents that its use would not infringe privately owned rights.

UCRL-50043-1

MIRROR HYBRID REACTOR STUDIES —
July 1975 through September 1976

J. D. Lee, Editor

MS. date: October 8, 1976

MASTER

Preface

This document contains reprints of six principal reports on mirror fusion-fission hybrid reactor design studies performed in FY 1976 at the Lawrence Livermore Laboratory (LLL) and General Atomic Company. The FY 1976 hybrid effort at LLL concentrated on improving the conceptual design of our fissile-fuel producing Mirror Fusion-Fast Fission reactor, upgrading system modeling and costing procedures, and performing optimization studies. The results of this work are encouraging. We estimate the Mirror Hybrid can produce fissile fuel for 45 mills per kilowatt-hour (electric).

In addition to the LLL effort, General Atomic is applying their expertise in helium-cooled reactors to assess and improve the LLL blanket design and perform more detailed design work on the blanket and helium heat transfer loops. General Atomic's preliminary assessment is that such systems look buildable and licensable. A progress report on the General Atomic work has been published under separate cover (K.R. Schultz et al., *Conceptual Design of the Blanket and Power Conversion System for a Mirror Hybrid Fusion-Fission Reactor*, General Atomic Report GA-A14021, July 6, 1976).

In FY 1977 we plan to combine the LLL and General Atomic work and develop a reference design of a fissile-producing hybrid reactor based on the standard mirror.

Contents

Mirror Hybrids — A Status Report (<i>J. M. Lee, D. J. Benier, E. W. Moir, and K. R. Schultz</i>)	1
Mirror Fusion-Fission Reactor Designs (<i>E. W. Moir</i>)	25
Blanket Design for the Mirror Fusion/Fission Hybrid Reactor (<i>J. M. Lee</i>)	39
Reactor Rates in a Uranium Pile Surrounding a 14-MeV Neutron Source: Calculations of the Weale Experiment (<i>P. J. Haiph*, J. M. Lee, and J. A. Manduca</i>)	51
Mirror Hybrid Reactor Blanket and Power Conversion System Conceptual Design (<i>K. R. Schultz, L. A. Pacheco, J. R. Baxi, J. M. Lee, E. A. Estrine, P. H., and A. B. Vega</i>)	61
Mirror Hybrid Reactor Optimization Studies (<i>D. J. Benier</i>)	73

50,058

Mirror Hybrids -- A Status Report

J. D. Lee, D. J. Bender, and R. W. Moir

Lawrence Livermore Laboratory

and

K. R. Schultz

General Atomic Company

Originally issued as Lawrence Livermore Laboratory Preprint UCRL-78079; prepared for inclusion in the Proceedings of the 2nd ANS Topical Meeting of the Technology of Controlled Nuclear Fusion, Richland, Washington, September 21-23, 1976.

100-658

MIRROR HYBRIDS - A STATUS REPORT*

J. D. Lee, D. J. Bender, R. W. Moir, and K. R. Schultz*

LAWRENCE LIVERMORE LABORATORY
GENERAL ATOMIC COMPANY*

ABSTRACT

Studies predict that neutrons produced in a D-T plasma, classically confined in a minimum B mirror field, could be used to economically produce fissile fuel. The conceptual design of such a facility is underway at LLL and GA. This facility, a mirror fusion fast fission hybrid reactor, uses a minimum extrapolation of present fusion technology to produce 2400 kg/y Pu plus 1000 MWe net from a fusion power of 500 MW. The estimated cost of producing the Pu is 55 \$/g or 4.0 mills/kWhr in an LWR fuel cycle.

Summary

In FY 75 we performed our first conceptual design study of a mirror fusion fast fission (hybrid) reactor (Ref. 1). The objective of that study was to develop a plant design that produced both power and fissile fuel from natural uranium while employing the minimum extrapolation of technology.

Based on the experience developed in our initial design study, we have spent FY 76 optimizing, improving and refining the design. We have concentrated our efforts in four main areas: (1) parametric system analysis to optimize the plant, (2) examination of different forms of U fuel to improve blanket neutronic performance, (3) improving the mechanical configuration, and (4) refining costing procedures. The net effect of this work has been a marked improvement in the predicted economics (Pu cost reduced

from 130 \$/g to 55 \$/g) as well as a more practical mechanical design. Preliminary results of this work were reported at the ANS winter meeting (1975) (Ref. 2) and summer meeting (1976) (Ref. 3). A more detailed review of the Mirror Hybrid Program is given in Ref. 4.

While our major emphasis is on uranium blankets we are looking at thorium blankets as well. Initial results show that even though a thorium hybrid produces fissile material at about 3 times the cost of a uranium blanket, its worth is correspondingly more by virtue of the higher conversion ratio of reactors (HWR or HTGR) using the thorium cycle.

In addition to the mirror hybrid work being done at LLL, the General Atomic Co. (GA) is providing industrial support in the area of gas cooled reactor technology.

* Work performed under the auspices of the U.S. Energy Research and Development Administration under Contract No. W-7405-Eng-48.

1. INTRODUCTION

Deuterium-tritium fusion reactions produce 14-MeV neutrons, and it is recognized that these highly energetic neutrons could be used to convert the abundant fertile isotopes ^{232}Th and ^{238}U to the fissile isotopes ^{233}U and ^{239}Pu . The existence of this breeding scheme is quite timely because present predictions on the consumption of fissile fuel by thermal power reactors forecast that the U.S. supply of the one naturally occurring fissile isotope (^{235}U) will be virtually exhausted in the early part of the twenty-first century.⁽⁵⁾ This consideration has been the motivation for the extensive effort devoted to the development of the fast breeder reactor.

A mirror confined plasma appears well suited as the source of D-T neutrons for fissile breeding because of its characteristics of high β (intense neutron source), steady state operation, and spherical geometry. The low power amplification factor, Q , characteristic of a mirror plasma is compensated by the energy release from fast fission of fertile isotopes by the D-T neutrons.

Two years of conceptual design studies of fissile producing mirror fusion - fast fission hybrid reactors providing fuel for converter reactors has shown that this concept may be economically competitive with the fast breeder. Certain design characteristics of the LLL mirror hybrid reactor, when compared to the fast breeder, represent distinct differences between the two systems.

These hybrid differences are:

- . subcritical fission assembly,
- . much lower power density,
- . no initial fissile inventory,
- . separation of fuel and energy production

If fusion and fission technologies can be successfully integrated into a single reactor, the resultant hybrid will offer an alternate or backup system to the fast breeder as a means of assuring a supply of fissile fuel for nuclear power generation.

This paper reviews the status of the Mirror Hybrid Program by summarizing the design and performance of our present conceptual design. The design is discussed in sections 2 - 8:

2. Plasma Physics,
3. Facility Description,
4. Blanket Design and Performance,
5. Blanket Assessment by General Atomic,
6. Power Conversion System,
7. Safety Considerations,
8. Reactor Optimization Studies.

2. PLASMA PHYSICS

The plasma physics parameters for the hybrid are listed in Table 2-1. Shown for comparison are the parameters for the hybrid designed in 1975,⁽¹⁾ a proposed mirror experiment (MX) and the present mirror experiment 2MTR.⁽⁶⁾

DATE	DESCRIPTION	AMOUNT	CHECK NO.	BANK
1954	1/15	100.00		
1954	2/15	200.00		
1954	3/15	150.00		
1954	4/15	100.00		
1954	5/15	100.00		
1954	6/15	100.00		
1954	7/15	100.00		
1954	8/15	100.00		
1954	9/15	100.00		
1954	10/15	100.00		
1954	11/15	100.00		
1954	12/15	100.00		
1954	1/15	100.00		
1954	2/15	100.00		
1954	3/15	100.00		
1954	4/15	100.00		
1954	5/15	100.00		
1954	6/15	100.00		
1954	7/15	100.00		
1954	8/15	100.00		
1954	9/15	100.00		
1954	10/15	100.00		
1954	11/15	100.00		
1954	12/15	100.00		
1954	1/15	100.00		
1954	2/15	100.00		
1954	3/15	100.00		
1954	4/15	100.00		
1954	5/15	100.00		
1954	6/15	100.00		
1954	7/15	100.00		
1954	8/15	100.00		
1954	9/15	100.00		
1954	10/15	100.00		
1954	11/15	100.00		
1954	12/15	100.00		
1954	1/15	100.00		
1954	2/15	100.00		
1954	3/15	100.00		
1954	4/15	100.00		
1954	5/15	100.00		
1954	6/15	100.00		
1954	7/15	100.00		
1954	8/15	100.00		
1954	9/15	100.00		
1954	10/15	100.00		
1954	11/15	100.00		
1954	12/15	100.00		
1954	1/15	100.00		
1954	2/15	100.00		
1954	3/15	100.00		
1954	4/15	100.00		
1954	5/15	100.00		
1954	6/15	100.00		
1954	7/15	100.00		
1954	8/15	100.00		
1954	9/15	100.00		
1954	10/15	100.00		
1954	11/15	100.00		
1954	12/15	100.00		
1954	1/15	100.00		
1954	2/15	100.00		
1954	3/15	100.00		
1954	4/15	100.00		
1954	5/15	100.00		
1954	6/15	100.00		
1954	7/15	100.00		
1954	8/15	100.00		
1954	9/15	100.00		
1954	10/15	100.00		
1954	11/15	100.00		
1954	12/15	100.00		
1954	1/15	100.00		
1954	2/15	100.00		
1954	3/15	100.00		
1954	4/15	100.00		
1954	5/15	100.00		
1954	6/15	100.00		
1954	7/15	100.00		
1954	8/15	100.00		
1954	9/15	100.00		
1954	10/15	100.00		

Recent experiments carried out with the 2XIIB mirror experimental facility have resulted in some major advances in mirror plasma physics. For example, start-up in static magnetic fields on stream plasma has been demonstrated. Steady state behavior has been demonstrated for short times (few msec). Instabilities have been largely suppressed by supplying a stream plasma fed in from the ends. The measured α scales with mean plasma energy (E) in the predicted way ($\propto E^{3/2}$) up to 13 keV. The n_i value is about a third the classical value due to heat loss from the hot plasma to the colder stream plasma. In the hybrid this loss is predicted to be reduced because the stream is needed only on the boundary which is a small proportion of the total volume whereas in 2XIIB the plasma is so small the boundary layer is the total volume.

Most of the important plasma parameters for the hybrid will be achieved if MX is successful, such as the density, n (plasma pressure divided by central vacuum magnetic pressure), injection energy, and magnetic field strength, $(10\text{ }^6\text{ gauss})^2$. The two parameters needing extrapolating are n and plasma dimensions. The hybrid design length, measured in central mean ion orbit radii, is almost 3 times longer than MX. The n -extrapolation is significant, but when account is taken of the higher mirror ratio, β , energy and heavier ion mass (D-T rather than just D-D) the extrapolation is not so great.

A number of changes in the plasma parameters have resulted from the parametric study which resulted in minimum cost of fissile fuel. One of the most notable changes was to raise the fusion power from 210 MW to 470 MW. This was accomplished by lowering the mirror ratio and thereby increasing the central field. In so doing, the density increased by a factor of 2.2. The penetration of the plasma by the natural

beam which previously left a density depression in the center of the plasma will now result in a deeper depression. There are two remedies for this beam penetration difficulty. One is to move the injectors towards the mirrors where the plasma is both thinner and less dense so the beam will penetrate to the center and give a uniform radial density distribution. The other is to allow a density buildup at the plasma edge and rely on the prediction that the plasma will convect inward due to unstable fluting. No deleterious side effects of this process have been found. This inward fluting is the major filling mechanism in the mirror fusion reactor design.

Theory predicts the drift cyclotron loss cone instability will exist and needs stabilization. The 2XIIIB method of using streams of relatively cold plasma to stabilize the plasma can apparently be used to stabilize the edge where the density gradient occurs.

A second instability of possible importance is the convective loss cone mode which is predicted to increase in activity with increasing L/a_i (plasma length/central gyro-orbit radius). Present predictions are that this mode may affect confinement at values of L/a_i greater than ~ 100 , but a thorough understanding of this instability will have to await experimental results from MX.

A more detailed description of mirror plasma physics is given by Post.⁽⁷⁾

3. FACILITY DESCRIPTION

The reactor facility may be roughly divided into two parts, the nuclear island and the thermal transport/conversion equipment. In this section we describe the former and reserve a discussion of the latter for section 6. The fusion power source is a mirror plasma contained in a minimum 8 magnetic well, and is sustained by the injection of beams of energetic neutral atoms (D^0 , T^0) Direct converters are located outboard of the mirrors to receive the plasma leakage, the function of these devices being to convert ion kinetic energy directly into electricity and to provide pumping for the gas load resulting from the plasma flow.

The layout of the containment structure and fusion related components is shown in Fig. 3-1. The dominant characteristics of the design are dictated by the magnet, which is a Yin-Yang coil set with a vertical axis. We are presently proposing the vertical orientation since it permits lowering of the bottom coil with a float-caisson arrangement, as shown in Fig. 3-1, thus exposing the blanket for maintenance operations. The overall blanket geometry is that of a spherical annulus which resides inside the coil and experiences a nearly uniform first wall loading. Further details of the blanket design are presented in sections 4 and 5.

Design constraints that we have placed on the mirror hybrid are (1) to employ fusion components that require the least possible extrapolation of present fusion technology, and (2) to employ existing fission reactor technology. Thus we view the mirror hybrid as a near-term goal in the overall program to commercialize fusion.

FIGURE 3-1. Plant Layout for Mirror Hybrid.

FIGURE 3-2. Magnet Structure

FIGURE 3-3. Neutral Beam Injector

The magnet uses NbTi superconductor, with a maximum field of 8 tesla at the winding. The coil restraint structure is constructed of prestressed concrete. A conceptual design of the magnet developed by Bechtel Corp. is shown in Figure. 3-2. A steady-state NbTi superconducting magnet (1 metre diameter) is presently in operation on the Baseball II experiment at LLL, and preliminary design of a NbTi magnet (3.5 metre diameter) for the MX experiment is underway. It is anticipated that NbTi magnet technology will undergo a progressive development up to coils of 10 - 15 metre diameter required for a commercial hybrid.

The neutral beam sources are positive ion injectors operating at 100 keV. A conceptual design for the injectors has been proposed ⁽¹⁰⁾ and is shown in Fig. 3-3. It features (1) extraction of positive ions from a plasma source, (2) ion acceleration to the desired energy, (3) neutralization (charge exchange) of a fraction of the beam current in the gas cell, (4) direct recovery of ion energy downstream of the gas cell by electrostatic deceleration, and (5) cryo-pumping of the neutral gas. Present positive ion sources operate at up to 40 keV in a pulsed mode (without direct conversion), and 120 keV modules are under development. ⁽¹¹⁾ The development of beam direct conversion technology is also presently proceeding at LLL ⁽¹²⁾

The vacuum system utilizes cryocondensation pumping; the pumping surfaces reside in the neutral beam lines to pump the source gas and in the direct converters to pump the gas resulting from the plasma end-losses. This technology is being developed as part of the high-voltage test stand at LLL, ⁽¹³⁾ and will be utilized in the MX experiment.

The direct converter is a single-stage unit, ⁽¹⁴⁾ a module of which is shown in Fig. 3-4. The performance of this type of device has been verified in small-scale laboratory experiments. ⁽¹⁵⁾ Scale-up of the equipment to reactor conditions has not yet begun. However, we have found with the mirror hybrid system model discussed in section 8 that the plant economics are not strongly sensitive to direct converter efficiency. Thus it is reasonable to propose an introductory version of the mirror hybrid which has not direct converters but merely uses plasma dumps to accommodate the end-losses.

4. BLANKET DESIGN & PERFORMANCE

Design Objectives

The objective of our mirror hybrid design effort is to develop a conceptual design of a mirror fusion fission system that produces, at minimum cost, fissile material (^{239}Pu or ^{233}U) from available fertile materials.

The blankets we are considering to meet this objective are a type we call fast fission blankets, which are designed to maximize fission and other neutron-producing events in ^{238}U or Th.

General Description

The blanket is a spherical annulus surrounding the nominally spherical D-T plasma and surrounded by the Yin-Yang magnet. The blanket has penetrations for neutral beam injection and open slots for plasma leakage. The blanket consists of an assembly of 16 segments fitting together like the segments of an orange. This blanket geometry is portrayed in Figure 4-1.

The spherical geometry is advantageous in that the fusion neutron current is nearly constant over the surface area of the blanket.

FIGURE 3-4. Direct Converter

FIGURE 4-1. Blanket Geometry

Replacement

Blanket replacement is achieved by vertical separation of the two Yin Yang magnet coils to expose the blanket, followed by vertical removal and replacement of individual blanket segments. This blanket replacement procedure is depicted in Figure 4-1. Large hydraulic pistons (or floats, as conceived by Doggett⁽¹⁷⁾) are used to separate the two coil halves. The coils are kept cold during this procedure to minimize reactor down time.

Sub-Module Design

Each blanket segment consists of a collection of sub-modules like those shown in Fig. 4-2. Each sub-module consists of an outer pressure vessel containing two fuel regions. The inner region, that region nearest the plasma, contains fertile material where both fission and capture of extra neutrons to breed fissile material occurs. The outer region contains the tritium breeding material lithium aluminate. Coolant is

supplied to and removed from the sub-modules by plena that makes up the back of a blanket module. The cooler inlet coolant first enters the sub-module and flows down in between the pressure vessel and the inner can which contains the fuel. By this method we can maintain the pressure vessel temperature near inlet conditions. The coolant then reverses direction and flows through the fuel. The coolant is pressurized helium. The principal reason why the blanket modules are segmented into sub-modules is to be able to contain the coolant pressure while still maintaining a rather thin first wall. A thin first wall is important in order to not degrade the fusion neutron energy significantly. With helium as our coolant we can employ gas-cooled fast reactor technology in the design of our blanket.

The thermal-hydraulic design of the sub-module is based on the requirement of maintaining the pressure vessel, the fuel cladding, and the fuel itself below maximum operating temperatures. The design must also provide a high enough coolant temperature to be thermodynamically interesting and have an acceptable pressure drop.

The structural and thermal-hydraulic requirements dictate how much structural material must be in the blanket. In our initial reference design studies (Ref. 1) we calculated a required fuel to structure ratio of 6.29. The fuel zones of the blanket consists of 8.6 volume % stainless steel plus 54% fuel. Fuel is 85% of theoretical density. The remaining 37% of the volume consists of helium coolant plus void. These material volume fractions are used in calculating nuclear performances.

FIGURE 4-2. Blanket Submodule

Nuclear Performance

The principle motivation behind the fast fission blanket is the desire to use 14 MeV fusion neutrons to induce fission in a fertile material such as uranium-238. Both the energy and neutrons produced by fission are important. The additional energy improves the power balance of the system and the excess neutrons produce fissile fuel.

To calculate the performance of fission blankets we use existing neutron transport codes as well as existing nuclear data libraries. At Livermore we have generally used the TART Monte Carlo code and the Livermore Evaluated Nuclear Data Library to perform hybrid blanket calculations.

To get an idea of how well we can calculate hybrid blanket performance we have compared calculated results to experiments. One such experiment is the Weale experiment. (Ref. 16) This experiment consisted of a natural uranium pile approximately a meter in diameter and a meter high with a 14 MeV neutron source in the center. The comparison between experimental results and what we calculated for this pile by using TART with both the old and new ENDF and ENDL libraries indicates the current methods and libraries are capable of calculating reasonably well, both the fission and the capture reaction rates. Based on Weale's results, one can conclude that if you surround a fusion source with natural uranium you could expect ~ 270 MeV per 14 MeV neutron plus $\sim$ four 238-U (n,γ) reactions. Of course, a blanket must have holes and contain structure, coolant, and tritium breeding materials, all of which will reduce performance.

For calculational purposes the blanket is modeled as concentric spherical shells. The shells contain homogenous mixture of materials. The inner most shell is 100% stainless steel half a centimeter thick and approximates the first wall. The inner radius of this first zone is 10 meters. The next shell is the fission zone and the third shell is the tritium breeding zone.

Once the material composition in the zones are specified the nuclear performance of the blanket is determined by calculating tritium breeding energy multiplication and the plutonium breeding versus fission zone thickness. The total blanket thickness is held constant at 1 meter. Results of such calculations are shown in Figure 4-3 for the blanket fueled with U + 7 w/c Mo fuel.

To determine what fission zone thickness is needed one has to include the effects of partial blanket coverage. This is because independent of coverage one must maintain a tritium breeding ratio of slightly greater than unity. As blanket coverage decreases one must decrease the fission zone thickness to maintain the tritium breeding ratio needed. Thus the overall effect of reducing blanket coverage is a marked decrease in performance.

For a blanket coverage of 90% the fission zone thickness is 25 centimeter, resulting in an effective blanket M of 9 and plutonium breeding ratio of 1.5. If blanket coverage were to drop to 75% the fission zone thickness must decrease to 10 cm to maintain T, and M would drop to 4 and the plutonium breeding ratio to 0.6.

At a wall loading of 1 MW/m^2 (fusion neutron energy current) the fast ($> 1 \text{ MeV}$) neutron flux at the first wall is $5 \times 10^{14} \text{ CM}^{-2}\text{-s}^{-1}$ and the peak fuel pin power density is 140 W/cc . Peak to average power density in the 25 cm fission zone is ~ 2 . These values are at start of life. Fuel pin power density will increase as M increases with exposure.

The final step in the nuclear analysis of the blankets is to estimate exposure effects. This was done in a first order manner by iterating with rather large steps and using average reaction rates for the fission zone.

Figure 4-4 shows the blanket performance versus exposure for the two types of fuels we are looking at, namely U-7 w/o Mo, and thorium metal. In each case, the exposure range is limited to exposures of interest. We do not want the U-Mo to go beyond 4 MWy/m^2 or the thorium to go past 10 MWy/m^2 . The limit on the U-Mo fueled blanket is burn up of the fuel itself, since after a burn up of $\sim 2\%$ fuel swelling becomes unacceptable. The limit on the thorium fueled blanket is expected to be structural lifetime of the pressure vessel and fuel cladding. Even without these constraints, economics dictate that the blankets be removed before these limits are reached.

The blankets are to be kept subcritical under all conditions. This is achieved by keeping k_{∞} of the fuel material less than 1.0. Thus far this constraint has not appeared to penalized our design since economics dictate exposures resulting in a fissile build up of only $\sim 2 \text{ a/o}$.

FIGURE 4-3. Nuclear Performance of U-Moly Blanket vs Fission Zone Thickness

FIGURE 4-4. Blanket Nuclear Performance vs Exposure

The evolution of our fast fission blanket concept is documented in references 1-4 and 17-21. For more detailed information on this blanket design, see references 1 and 4.

5. BLANKET ASSESSMENT BY GENERAL ATOMIC

In addition to blanket work at LLL, GA has reviewed the LLL blanket design concept and made recommendations for improvements. This section summarizes the General Atomic Co. blanket study.

Fuel Design

The candidate fuel materials that have been evaluated include uranium alloys, uranium oxide, uranium carbide, uranium nitride and uranium silicide. A comparative evaluation was made in terms of the important design parameters and cost. Two important characteristics of the hybrid reactor are the low burnup⁽³⁾ (~1%) and the importance of high fuel density⁽¹⁾ which tend to favor the use of metallic uranium alloys. Uranium oxide, carbide and nitride are less desirable because of significantly lower fuel density, lower neutron economy, and higher fabrication cost.

Three primary fuel candidates were selected from those evaluated: uranium-molybdenum alloy, uranium carbide and uranium silicide. (22-24) The density and maximum operating temperature for these candidate materials are shown on Table 5-1.

TABLE 5-1
FUEL MATERIAL CANDIDATES

	U-MOLY	UC	U ₃ Si
Uranium Density (g/cc)	17.1	13.0	15.5
Maximum Operating Temperature (°C)	700	2000	900

Since U₃Si has the advantages of requiring less stringent quality assurance, of allowing larger fuel rods and of reduced parasitic neutron absorption when compared to U-Moly, U₃Si is recommended as the fuel material.

Three different fuel configurations were investigated for use in the blanket: honeycomb, plates and rods. A rod configuration appears to be best suited to application in hybrid reactor. The fuel rods may be arranged parallel or perpendicular to the flow. The use of cross flow does not enhance the average heat transfer coefficient but does introduce a large variation in the temperature around the rod. Parallel flow results in a uniform heat transfer coefficient around the rod, reducing the potential for hot spots. By including a hole in the center of the fuel pellets, swelling problems can be greatly reduced.⁽²⁵⁾ As a consequence, annular fuel pellets arranged in a radial direction were selected. To get the maximum possible helium outlet temperature a high temperature cladding material, Inconel 718, is recommended.

The helium inlet and outlet temperature conditions are selected to correspond to the fuel temperature limits as shown on Table 5-2.

The fuel design is based on hot spot temperatures obtained by applying hot spot factors, determined using a semi-statistical method,⁽²⁶⁾ to the nominal temperatures calculated for an average channel. End-of-life conditions were used since these represent the maximum blanket multiplication and power density.

TABLE 5-2
BLANKET TEMPERATURES (°C)

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

The blanket is designed with a flow baffle to keep constant flow velocity over the first wall, keeping the maximum first wall temperature near the inlet temperature. A fuel rod diameter of 17 mm limits the peak hot spot cladding temperature to 800°C and the peak hot spot fuel temperature to 900°C. The power distribution and resulting nominal temperatures along the fuel rod are shown in Fig. 5-1.

FIGURE 5-1. Nominal Temperature Distributions in the Blanket

Module Design

To insure good neutronics performance of the blanket, it is important that the first wall be kept as thin as possible. It must operate, however, at high temperature under high pressure. Inconel 718 is recommended for the blanket structural material and fuel cladding material because it exhibits high strength at elevated temperature.

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

Forty-five submodules are arranged to form each of the sixteen modules. The module is in the shape of a spherical segment.

Each module is a separate pressure vessel. With a 25 cm submodule width, and 6.08 MPa helium pressure, the pressure containing first wall can be kept to 0.5 cm thick. A reentrant coolant flow system was chosen which allows the external module walls to be kept near the coolant inlet temperature of 280°C.

6. POWER CONVERSION SYSTEM

System Parameters

The helium temperature and pressure have a significant bearing on the design feasibility, fabricability, size and cost of the power conversion components. The temperatures are set by consideration of material limits, steam conditions and heat transfer surface requirements. The pressures impact the helium and steam duct sizes and the circulator design. Consideration of these effects led to selection of the parameters shown in Table 3.

TABLE 3
POWER CONVERSION SYSTEM PARAMETERS

Helium

Average pressure	6.08 MPa	(60 atm)
Blanket inlet temp.	280°C	(536°F)
Blanket outlet temp.	530°C	(986°F)

Steam

Superheater outlet press.	8.3 MPa	(1200 psi)
Superheater outlet temp.	445°C	(832°F)
Reheater outlet temp.	504°C	(939°F)

A schematic power flow diagram for the power conversion system is shown in Fig. 6-1. The total thermal power input to the power conversion system from the blanket and direct convertor is 4381 MW, the output from the turbine generator is 1574 MWe and the power conversion system efficiency is 35.9 percent. Due to the large power requirement of neutral beam injectors the net station efficiency is 22.1 percent.

FIGURE 6-1. Power Flow Diagram

FIGURE 6-2. Primary Loop Arrangement

Power Conversion System Layout

A relatively compact arrangement can be achieved by grouping the sixteen blanket modules into four quadrants of four interconnected modules each. To minimize the length of the large diameter (1.2 m) helium ducts, the four quadrants are not interconnected. Each quadrant comprises one main loop, divided into two interconnected sub-loops, and one auxiliary loop, also divided into two sub-loops. These main and auxiliary components are connected together as shown in Fig. 6-2.

Primary Coolant System

Each quadrant has four steam generators, two helium circulators, two auxiliary heat exchangers and two auxiliary circulators. Each quadrant is also connected to one quarter of the direct converters. The system components are based on standard gas-cooled reactor technology.

Main Cooling Loops

To provide cooling system redundancy each quadrant consists of two sub-loops, each with one helium circulator and two steam generators. The components are interconnected and sized so that one steam generator of either sub-loop can adequately cool the quadrant to remove decay heat even if the helium is depressurized. Each circulator is rated at 13.2 MW (17,000 HP) and is powered by main loop steam. The series arrangement provides inherent load following characteristics to the main loops

Auxiliary Cooling Loops

Each quadrant is provided with an auxiliary cooling loop to provide an independent, diverse means of cooling the shutdown reactor to remove decay heat. The auxiliary loops could also be used for long term decay heat removal, if desired.

The auxiliary circulators are driven by electric motors and are each rated at 2.75 MW (3700 HP). The auxiliary heat exchangers dump the blanket decay heat to the atmosphere.

Secondary Coolant System

The steam loop of the power conversion system is quite conventional. Steam raised in the sixteen once-through steam generators is routed through the helium circulator steam turbine generator. The turbine is a tandem-compound machine, exhausting to a water-cooled condenser.

Power Conversion System Control

The plant control system (PCS) used on the power conversion system is quite similar to that developed for gas-cooled fission reactors and allows automatic load following over a wide range. In addition to the on-load control system the plant control system also has a decay heat removal control system, a shutdown control system and a startup control system.

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

7. SAFETY

Safety Analysis

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

Reactivity Insertion Accidents

The blanket is designed to be substantially subcritical at all times and we find no effects that could substantially increase the blanket multiplication.

Flow Blockage Accidents

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

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

Steam Leakage Accidents

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

Fuel Handling Accidents

The method of fuel handling during refueling was not addressed in this study. Due to the

decay heat, the modules will probably have to be actively cooled during refueling and transport. Adequate cooling will have to be guaranteed. A reliable means of handling the massive modules will have to be developed.

Helium Leakage Accidents

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

A double-ended main helium duct rupture could constitute a significant hazard and will have to be designed against. The danger from a main duct rupture is the potential for causing further damage that could compromise the ability to cool the blanket. An aspect of the mirror hybrid that needs further study is the design of the helium coolant ducts. The ducting and module support and containment concept will have to be designed in such a way as to assure reliable maintenance of adequate cooling of the blanket. A number of alternatives are possible, the use of supports and snubbers as used in LWR's, the use of metallic "rip stops" on the ducts to restrict the maximum size of any duct ruptures and the use of partial or even total prestressed concrete reactor vessel containment of the ducting and primary loop components.

Design Basis Accident

The design basis accident (DBA) is a hypothetical, non-mechanistic accident that is not expected to happen but that is postulated for the analysis of engineered safety cap-

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

Licensing

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

8. REACTOR OPTIMIZATION STUDIES

The LLL hybrid study this year has concentrated on optimizing the hybrid for fissile production, employing the technique of parametric system analysis of the plant economics. The optimization was defined to be a determination of the reactor parameters which minimized the cost of producing fissile fuel. The optimization thus minimizes the elec-

tricity cost component attributable to fissile fuel of the fission reactors which burn the hybrid-produced fuel.

8.1 System Model

A large number of independent parameters defining the fusion components of a mirror hybrid are available to the reactor designer, and variation of these parameters can significantly affect the hybrid performance. In addition, the plant economics are influenced by the blanket fissile management scheme and by the neutronic characteristics of the fission assembly. To assess the interplay of these various factors, the approach taken has been to develop a computer model of the mirror hybrid that will permit rapid evaluation of the many possible reactor configurations. The components of the system model⁽²⁷⁾ developed for the parametric analysis are as follows:

- Reactor Description
 - plasma physics
 - magnet design
 - blanket geometry
 - power flow
 - capital cost
- Fuel Management
 - exposure-dependent blanket neutronics
 - time-dependent mass and energy flows
 - capacity factor
 - cash-flow accounting techniques
- "Nuclear Park" Economics
 - hybrid + fission reactors

The mirror reactor model is essentially the analysis developed by Carlson,⁽²⁸⁾ and includes mirror plasma physics, magnet design, blanket geometry, and power flows. The

capital costs are a key element in the analysis, and here we have attempted to be as thorough and consistent as possible. However, the costing is a procedure entailing a high degree of uncertainty due to the infancy of fusion technology.

A unique feature of hybrid economics, as compared to strictly power-producing fission and fusion reactors, is that a principal product of the hybrid, fissile fuel, does not generate revenues on a continuous basis. Revenue from fissile fuel is only realized when blanket segments are removed from the reactor and reprocessed. In addition, the blanket multiplication increases and the breeding ratio decreases with increasing fuel exposure, as described previously. To model these effects, we have developed a fuel management package that evaluates the time-dependent production of power and fissile fuel as functions of specified fuel management parameters. This analysis also evaluates the timing and magnitude of fuel and blanket fabrication costs and spent-fuel shipping and reprocessing costs. The economics of this time-dependent fuel-cycle is evaluated using cash-flow accounting techniques.⁽²⁹⁾

A second unusual feature of the economic analysis is that the hybrid produces two products, fissile fuel and electricity. To fix the individual cost of producing each of the products it is necessary to specify a constraint. In our present analysis we have chosen to fix the cost of hybrid electricity at the same cost as the electricity produced by the fission reactors which burn the hybrid fissile fuel. By considering the hybrid plus its associated burner reactors as a single entity producing just electricity, we are able to calculate the electricity cost.

Having established the electricity cost, the fissile material cost from the hybrid can then be evaluated.

The fission reactors chosen as burners of the hybrid fissile fuel are listed in Table 8-1 along with their requirements for fissile fuel. As a burner of Pu, we have used a light water reactor (LWR) on a Pu recycle fuel cycle. As a ^{233}U burner, we have used a high-gain HTGR using the thorium - ^{233}U fuel cycle. Another possibility as a ^{233}U burner, but not yet examined, is the CANDU reactor.

8.2 Optimized Reactor Configurations

The optimized reactor parameters for both U/Mo and thorium blankets are listed in Table 8-2. There are several significant differences between the two reactors:

- The uranium blanket, because of its high energy multiplication, results in a plant with a large electrical output. The thorium blanket reactor does not produce net electricity - just fissile fuel.
- Both blankets have about the same thermal rating, this being the result of a much larger fusion power from the thorium blanket reactor as compared to the uranium blanket reactor.
- The high fusion power of the thorium blanket reactor is obtained by using a more intense magnetic field than for uranium reactor. The uranium blanket reactor may therefore rely on existing NbTi superconductor magnet technology whereas the thorium blanket reactor optimizes with the more technologically advanced Nb_3Sn superconductor.

The blanket parameters for the optimized reactors are listed in Table 8-3. Both produce between two and three metric tons of fissile fuel per year. However, the thorium blanket requires a rather high exposure, and the possibility of the blanket structure being able to attain $\sim 9 \text{ MW} \cdot \text{yr}/\text{m}^2$ exposure is rather uncertain. The average energy multiplication of the uranium blanket is about a factor of 4 higher than for the thorium blanket; these blanket multiplications include the effect of the fractional blanket coverage.

The hybrid economic parameters are listed in Table 8-4. The higher capital cost of the thorium blanket hybrid is associated with the fusion components required to generate the higher fusion power. The ^{233}U cost is more than a factor of two greater than the Pu cost, the lower cost for Pu being the result of (1) lower capital cost and (2) revenues from electrical power generation. However, the lower fissile make-up requirements of the HTGR as compared to the LWR results in approximately the same electricity value from the two fission power plants. The breakdown of the fissile material costs indicate that they are dominated by capital costs. The fuel cycle costs account for blanket fabrication, fuel fabrication, reprocessing, and spent-fuel shipping. Current (high) estimates for the fuel services have been used,⁽³⁰⁾ but they are not a dominant cost. For the uranium blanket reactor approximately 60% of the plant revenues are generated by fissile production in contrast to the total revenue generation by fissile material for the thorium hybrid.

Our estimates of the fusion power balance parameter Q are based on our current understanding of "classical" confinement in a magnetic mirror. This value of Q is denoted by $Q_{\text{classical}}$. Recognizing that there are uncertainties in this parameter, we have examined the economic implications of variations in Q , as shown in Fig. 8-1. For values of Q less than classical, the Pu cost rises rather sharply, almost doubling for $1/2$ of classical confinement. For Q greater than classical, the fissile cost decreases at a more modest rate. These variations in fissile value with $Q/Q_{\text{classical}}$ are associated with the recirculating power fraction and the revenues realized from net power production. At the classical value of Q the plant has a recirculating power fraction of about 0.5.

FIGURE 8-1. Fissile Cost vs $Q/Q_{\text{Classical}}$

The fission reactor economics are listed in Table 8-5. Here the category "fuel cycle without fissile material" refers to all normal fuel cycles charges excluding the cost of fissile fuel, i.e. fabrication, reprocessing, spent-fuel shipping, and purchase of fertile fuel. The fissile fuel cost is the cost of producing this material with the hybrid reactor. The important result here is that the hybrid fissile fuel costs of 4.1 and 5.3 mills/kw-hr are a modest fraction of the total electricity cost. The conclusion is that the mirror hybrid reactor, based on our current capital cost model, is capable of converting the world's large fertile resources into fissile fuel at a cost that does not strongly influence the net cost of electricity.

The overall nuclear park economics are shown in Table 8-6. Examining the U hybrid/LWR combination, the 1040 MWe hybrid supports fissile make-up requirements for about seven 1000 MWe thermal reactors. The capital cost of the hybrid + fission reactors is 935 \$/kWe vs 750 \$/kWe for the LWR's. The Th hybrid produces enough fissile material to support fourteen 1000 MWe HTGR's resulting in a capital cost for the nuclear park of 985 \$/kWe vs 750 \$/kWe for the HTGR. The fuel cycle costs do not include any cost for fissile fuel since this material is produced entirely within the hybrid/fission reactor complex.

The fact that the hybrid fissile production costs are capital cost dominated (see Table 8-4) dictates that we regard the fissile costs with a degree of uncertainty, reflecting the infancy of fusion engineering and our inability to accurately cost reactor components which are now merely conceptual designs. However, presently predicted

costs are well within the realm of being economically attractive. It is our opinion that this result justifies vigorous support of the hybrid concept, with future design efforts continuing to refine the engineering and incorporating experimental results as they become available. The result could well be an energy option that will ease the transition to a full fusion power economy in the next century with comparatively early benefits from the large R & D investment that will be required to commercialize fusion as an energy source.

Table 8-3
OPTIMIZED HYBRID BLANKET PARAMETERS

	<u>U/Mo</u>	<u>Th</u>
FISSILE OUTPUT (kg/yr)	2360	2590
AVG. ENERGY MULTIPLICATION	11.1	2.8
BLANKET COVERAGE	0.86	0.77
FERTILE BURNUP (%)	1.0	0.5
BLANKET EXPOSURE (MW-yr/m ²)	4.1	9.2
AVG. FUEL POWER DENSITY (W/cc)	150	110
AVG. BLANKET ENRICHMENT AT REMOVAL (%)	1.02	1.06

Table 8-4
OPTIMIZED HYBRID ECONOMICS

	<u>Pu</u>	<u>233U</u>
	<u>Burner</u>	<u>Burner</u>
REACTOR TYPE	LWR	high-gain HTGR
FUEL CYCLE		
FERTILE FEED	Nat. U	Th
FISSILE FEED	Pu	233U
FISSILE RECYCLE	Pu	233U
CONVERSION RATIO	0.5	0.6
FISSILE MAKE-UP REQUIREMENTS (kg/yr/MWe)	0.333	0.185

	<u>U/Mo</u>	<u>Th</u>
CAPITAL COST (10 ⁹ \$) (\$/kWe)	2.3	3.3
FISSILE MATERIAL COST (\$/gm)	55	127
CAPITAL FUEL CYCLE O & M	80	103
ELECTRICITY COST	13	21
ELECTRICITY COST FROM HYBRID AND FISSION REACTORS (mills/kw-hr)	1	1
	-39	2
	24.8	25.3

Table 8-5
FISSION REACTOR ECONOMICS

	<u>LWR</u>	<u>high-gain HTGR</u>
CAPITAL COST (\$/kWe)	750	750
ELECTRICITY COST (mills/kw-hr)	24.8	25.3
CAPITAL COST	16.1	16.1
FUEL CYCLE Without FISSILE MATERIAL O & M	3.9	3.2
FISSILE FUEL (from hybrid)	0.7	0.7
	4.1	5.3

Table 8-6
NUCLEAR PARK ECONOMICS

	<u>U/Mo</u>	<u>Th</u>
INSTALLED CAPACITY (MWe)	8130	14000
HYBRID	1040	-
LWR (HTGR)	7090	(14000)
CAPITAL COST (\$/kWe)	935	985
ELECTRICITY COST (mills/kw-hr)	24.8	25.3
CAPITAL FUEL CYCLE O & M	19.7	20.5
	4.3	4.0
	0.8	0.8

Table 8-2
OPTIMIZED HYBRID REACTOR PARAMETERS

	<u>U/Mo</u>	<u>Th</u>
MIRROR RATIO	2.50	2.75
INJECTION ENERGY (keV)	100	100
CONDUCTOR FIELD (T)	8	12
Q	0.68	0.75
FUSION POWER (MW)	470	1500
FIRST WALL FLUX (MW/m ²)	1.3	4.2
AVG. BLANKET THERMAL POWER (MW)	4220	3340
ELECTRICAL OUTPUT (MW)	1040	-40
CAPACITY FACTOR	0.75	0.73
MIRROR-TO-MIRROR LENGTH (m)	15	15

CONCLUSIONS

This paper summarized the status of the Mirror Hybrid Program by describing the design and performance of our interim conceptual mirror fusion fast-fission hybrid reactor optimized for the conversion of fertile to fissile material, and by discussing the feasibility of, and some planned improvements, to this conceptual design.

The design is relatively conservative. The plasma component performance requirements are less demanding than those that must be developed for non-hybrid mirror fusion reactors. The subcritical, moderate power density, and low burnup characteristics of the blanket might result in less demanding engineering requirements for the fission components than are required for fission reactors.

The conceptual designs of the blanket and power conversion system are derived from General Atomic's design experience with the High Temperature Gas-Cooled Reactor and the Gas-Cooled Fast Reactor. From the analysis of the conceptual design we have concluded that although there are a number of aspects of the present design that require further work, the hybrid blanket and power conversion system concepts appear to be technically feasible. Existing gas-cooled fission reactor technology is directly applicable to the mirror hybrid reactor.

There appear to be no inherent features of the hybrid concept that present fundamentally new safety considerations to the reactor design. Further work is needed, however, in the area of the primary loop

ducting design, support and containment to assure that the reactor will be adequately cooled under all circumstances. The apparent total absence of potential accidents that could cause the system to become supercritical does offer the hybrid reactor some advantages compared to fission reactor systems. Given that maintenance of cooling can be adequately assured, the hybrid reactor could be licenseable under current Nuclear Regulatory Commission regulations.

Our future plans are to apply the methods and findings of both the LLL and General Atomic studies in a joint effort to develop a mirror hybrid reference design. This design will be optimized to produce fissile fuel at minimum cost.

The potential benefit of the fissile-breeding hybrid concept is based on the fact that long-term development of a fission power economy requires utilization of the world's large reserves of fertile isotopes, ^{232}Th and ^{238}U . We see in the fast-fission hybrid a unique facility in its ability to perform fertile-to-fissile conversion. Our studies predict that a hybrid with a uranium fast-fission blanket can support the fissile fuel make-up requirements of 5 light water reactors of comparable thermal rating and that the capital cost (\$/kWe) of the combined hybrid/LWR complex is only ~ 25% greater than the LWR's themselves. Preliminary analysis of a thorium hybrid/HTGR complex gives similar economic results. An additional advantage is that the hybrid requires no initial inventory of fissile material.

The point to be emphasized is that the high breeding rate of the fast fission hybrid is unmatched by any other reactor concept and it is this characteristic that should be exploited. The 5/1 to 10/1 ratio between thermal reactors and their hybrid fuel producers provides the hybrid with very strong economic leverage.

References

1. R. W. Moir, et al, "Progress On The Conceptual Design Of A Mirror Hybrid Fusion-Fission Reactor," UCRL-51797, June 1975.
2. J. D. Lee, et al, "Optimizing The Mirror (Fusion-Fission) Hybrid Reactor For Plutonium Production," UCRL-76986, Nov. 1975. Presented at 1975 Winter ANS Meeting.
3. D. J. Bender and J. D. Lee, "The Potential For Fissile Breeding With The Fusion-Fission Hybrid Reactor," UCRL-76986, June 1976. Presented at the 1976 Summer ANS Meeting.
4. Papers on Mirror Hybrid Program by Moir, Lee, Bender & Schultz; Proceedings of the Joint US-USSR Symposium on Fusion-Fission Reactors, held July 13-16, 1976 at LLL, to be published.
5. Nuclear Power Growth 1974 - 2000 U.S. AEC, Rept. WASH.-1139 (1974).
6. F. H. Coensgen, et al, "2XIIB Plasma Confinement Experiments," Paper CN-35/e1 submitted to proceedings of the International Conference on Plasma Physics and Controlled Nuclear Fusion Research, Berchtesgaden, Federal Republic of Germany, October 6-11, 1976.
7. R. F. Post, "Status of Plasma Physics Of Mirror Devices," these proceedings.
8. C. D. Henning, et al, "Large Superconducting Baseball Magnet", in Advances In Cryogenic Engineering, Vol. 14, Plenum Press, NY, 1969, page 96.
9. F. H. Coensgen, "MX Major Project Proposal", Lawrence Livermore Laboratory, LLL-Prop.-142 (March 1976).
10. J. H. Fink, W. L. Barr, and G. W. Hamilton, "A 225-MW Neutral Injection System For A Mirror Fusion-Fission Hybrid Reactor", Lawrence Livermore Laboratory, UCRL-76704 (1975).
11. K. W. Ehlers, et al, "Conceptual Design Of A Neutral-Beam Injection System for the TFTR", Proc. 6th Symposium on Eng. Problems of Fusion Research, San Diego, 1975, p.855.
12. W. L. Barr, and R. W. Moir, "A Review Of Direct Energy Converters For Fusion Reactors", These proceedings.
13. T. J. Duffy and L. D. Odden, "Beam-Line Cryopump", Lawrence Livermore Laboratory, UCRL-77236 (Nov. 1975).
14. R. W. Moir and W. L. Barr, "Venetian Blind Direct Energy Converter for Fusion Reactors", Nucl. Fusion, 13 35 (1973).
15. W. L. Barr, et al, "Apparatus for Testing Direct Energy Conversion of Plasma Energy to Electricity", Lawrence Livermore Lab., UCRL-75174, (Nov. 1973).

16. J. W. Weale, et al, "Measurements of the Reaction Rate Distribution Produced by a Source of 14 MeV Neutrons at the Center of a Uranium Metal Pile", J. Nucl. Energy A/B 14, 91 (1961).
17. J. D. Lee et al, "Mirror Reactor Blankets", UCID 17083, March 1976. To be published in the proceedings of the Blanket/Power Systems for Fusion Reactors Technology Workshop held March 21 - April 12, 1976 at BNL.
18. J. D. Lee, "Neutronics of Sub-Critical Fast Fission Blankets for D-T Fusion Reactors", in Proc. 7th Conf. Intersociety Energy Conversion Engineering, 1972 (American Chemical Society, 1972), p. 1294.
19. J. D. Lee, "Neutronic Analysis of a 2500 MW_{th} Fast Fission Natural Uranium Blanket for a DT Fusion Reactor," in Proc. First Topical Meeting on the Technology of Controlled Nuclear Fusion, 1974 (San Diego, CA, 1974), CONF-740402-P1, Vol. 1 p. 233; also Lawrence Livermore Lab. Rept. UCRL-75304 (1974).
20. R. C. Haight and J. D. Lee, "Calculations Of A Fast Fission Blanket for D-T Fusion Reactors With Two Evaluated Data Libraries", in Proc. First Topical Meeting on the Technology of Controlled Nuclear Fusion, 1974 (San Diego, CA 1974), CONF-740402-P1, Vol. 1 p. 271; also, Lawrence Livermore Lab. Rept. UCRL-75627 (1974).
21. R. C. Haight et al, "Reaction Rates In A Uranium Pile Surrounding a 14-MeV Neutron Source: Calculations of The Weale Experiment", UCRL-77364, Oct. 1975. Also, NS&E, 61, 53 (1976).
22. Schultz, K. R., et al, "Conceptual Design of the Blanket and Power Conversion System for a Mirror Hybrid Fusion-Fission Reactor," GA-A14021, July 6, 1976.
23. Chalder, G. H., et al, "U₃Si as a Nuclear Fuel," AECL-2874 May 1967, Atomic Energy of Canada Limited.
24. Simnad, M. T., "Fuel Element Experience in Nuclear Power Reactors," AEC Monograph, Amer. Nuclear Society, (1971).
25. Leggett, R., et al, "Achieving High Exposure in Metallic Uranium Fuel Elements", Nuclear Appl. Tech., 9, 673 (Nov. 1970).
26. Carelli, M. D., & D. R. Spencer, "CRBRP Assemblies Hot Channel Factors Preliminary Analysis," WARD-D-0050, 1974.
27. D. J. Bender and G. A. Carlson, "System Model for Economic Analysis of the Mirror Fusion/Fission Hybrid Reactor," Lawrence Livermore Lab., report in preparation.
28. G. A. Carlson and R. W. Moir, Mirror Fusion Reactor Study, Lawrence Livermore Lab., Rept. UCRL-76985 (1975).
29. "Guide for Economic Evaluation of Nuclear Reactor Plant Designs", NUS Corporation, Rept. NUS-531, Appendix E (1969).
30. Summary Report, Fuel Cycle Conference 1975, Atomic Industrial Forum, June 1975.

Mirror Fusion-Fission Reactor Designs

R. W. Moir

Originally issued as Lawrence Livermore Laboratory Preprint UCRL-78629 (August 1, 1976); prepared for inclusion in the Proceedings of the US-USSR Symposium on Fusion-Fission Reactors, Livermore, California, July 13-16, 1976.

MIRROR FUSION-FISSION REACTOR DESIGNS*

R. W. Moir
Lawrence Livermore Laboratory, University of California
Livermore, California 94550

ABSTRACT

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

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

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

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

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

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

A simple but powerful argument will help illustrate why power production is not the best goal for hybrids. The argument goes as follows: The cost of electrical power for very capital-intensive plants (where cost of capital $\gg$ fuel-cycle costs) is proportional to the capital cost per unit electrical power, C/P_e . Now, although the cost of a fusion reactor, C , is increased only modestly¹ (by less than a factor of 2) by the addition of a fission blanket, the power is increased by a large factor (5 to 10 or more). Thus, C/P_e hybrid $\ll C/P_e$ fusion. The hybrid reactor is a fission reactor with the complicated addition to the inside of the fission core of a magnetic field and a vacuum system with all the fusion plasma paraphernalia. The cost must thus be significantly higher for the hybrid than for the fission reactor. Hence,
 C/P_e fission $\ll C/P_e$ hybrid $\ll C/P_e$ fusion. (1)

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

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

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

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

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

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

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

The cost of the fuel produced from the first method will increase with time due to depletion of higher-grade ores. Although mining is the only fuel source today, it is only a question of time until this method will become uneconomical. The

second source, the fission breeder, is nearing the commercial point, with the main issues being safety and economics. The last source, fuel bred by neutrons, is an option that depends mostly on economics. With a modest extrapolation from existing technology, the accelerator technology would be available. Fuel from a hybrid reactor plus that bred by a fission reactor or a neutron accelerator (if the latter two methods materialize) could lower the price of ^{235}U by isotope separation due to competition.

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

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

The mirror concept for the fusion component of the hybrid offers a number of advantages over alternative concepts. Steady-state operation, high fusion power density, and relatively small size and simple geometry (spherical) are highly desirable features not shared by Tokamaks, Z-pinches, etc. We can make the general comment, however, that any fusion concept that comes close to meeting the Lawson criterion of $n\tau \approx 10^{14} \text{ cm}^{-3}\text{s}$ (more precisely $Q \approx 1$, $Q = P_{\text{fusion}}/P_{\text{injection}}$) would be a candidate for a hybrid reactor. For example, the beam-driven Tokamak² may be a good candidate for a hybrid.

DESIGN PROBLEMS

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

TECHNOLOGY

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

SAFETY

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

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

Some argue that hybrids should be measurably safer than fission reactors before they become very interesting. This is probably so if the hybrid is viewed primarily as a power producer; however, as a fuel producer the hybrid itself need not be safer than a fission reactor if the hybrid plus the fission reactors that burn its bred fuel are safer as a system than the alternate system (the breeder). Thus, some additional hazard could be incurred to obtain the necessary fuel to supply

converter fission reactors which due to fuel scarcity would otherwise be breeders. The relevant hazard question is: Is one (remotely located) hybrid plus half a dozen converter reactors less hazardous than a like number of breeder reactors?

ECONOMICS

The question of economics boils down to the cost of producing Pu or ^{233}U per gram and a comparison of that cost to that of alternative sources. Our studies⁴ show production costs of \$60/g for Pu and \$130/g for ^{233}U . These numbers are highly uncertain due to uncertainties in fusion technologies, particularly those not yet demonstrated. Our studies do, however, show the relative importance of the many details of the design. If true, these cost projections would lead to a significant penetration of the market by hybrid reactors if breeder reactors either are not developed successfully or are developed at too high a cost.⁵

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

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

REVIEW OF PREVIOUS WORK ON THE MIRROR HYBRID

Work on the mirror fusion-fission reactor concept began in 1972 with the

work by Lee⁶ on the fast-fission blanket. From his blanket neutronics calculations, which included various structural fractions, uranium thicknesses, and Li thicknesses, Lee concluded that 1.7 atoms of Pu and 140 MeV could be produced per fusion neutron while at the same time tritium was being bred. Hansborough and Werner⁷ designed a modular blanket which incorporated a thermal lattice; when fitted together, the modules formed a geodesic shell.

Then a joint⁸ Lawrence Livermore Laboratory-Pacific Northwest Laboratory (LLL-PNL) study was carried out in which the modularized blanket design was integrated with the Yin-Yang magnet of the fusion system. Slots in the blanket allowed for plasma leakage, and ports provided for neutral-beam injections. System performance studies showed that the plant efficiency was 32%, with 25% of the gross electrical power recirculated. The blanket was based on the PNL⁹ concept of a thin uranium converter zone (sometimes called a fission plate) followed by a graphite-moderated thermal lattice and tritium-breeding zones. The energy of the 14-MeV neutron was multiplied by a factor of 38. Some thermal aspects of design were studied. However, the study had certain deficiencies:

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

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

Other analyses included some parametric system studies by Moir¹⁰ that showed the quantitative reduction of $n\tau$ allowed by the fission blanket and the importance of low injection energies (100 keV). A study of safety by W. C. Wolkenhauer *et al.*,¹¹ showed that the radioactivity associated with the hybrid was only slightly less than that of a

fission reactor of the same rating and that the large thermal capacity of the low power density ($6 \text{ W}\cdot\text{cm}^{-3}$) graphite thermal lattice greatly reduced the problems usually associated with LOC accidents.

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

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

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

Fig. 1. Mirror fusion reactor.

REVIEW OF PRESENT DESIGN STUDIES

OVERALL DESCRIPTION

Previous design studies¹² had shown a movement towards practicality and near economical operation (within a factor of 3 of a fissile fuel cost that seems economical for fission-reactor fuel). The present studies have capitalized on design optimization and on a new design idea,

namely, a design in which the two halves of the magnet are oriented vertically, with one half resting on a float so that the magnet can be opened up to give access to the blanket.¹³ As shown in Fig. 1, the reactor is built on a dry dock. The blanket can be handled with an overhead crane and can be made in the spherical shape of previous designs. The blanket is shown in Fig. 2. By greatly reducing the peak-to-average power ratio, the power can be

raised by a factor of 4 at the same peak power density as the earlier design. Lee¹⁴ has made more detailed designs of the blanket.

Bender⁴ has made an economic analysis of a nuclear park composed of burner reactors consuming the fissile fuel produced by the hybrid reactor. The

nuclear park economic model allows optimization of such parameters as mirror ratio (ratio of magnetic field at the mirrors to central vacuum magnetic field) fuel burnup, injection energy, etc.

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

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

Table 1. Fusion-fission reactor parameters.

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

INJECTOR SYSTEM

The injector system for the hybrid¹⁵ is based on 100-keV positive (D^+ , T^+), ions and includes direct conversion of unneutralized ions. An experimental program to develop the injector and direct converter is underway at LLL-LBL; 1-MW modules will be tested this year. The efficiency of producing neutral-beam power from electrical power was determined to be 70%.

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

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

Fig. 3. Neutral-beam injector array.

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

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

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

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

DIRECT ENERGY CONVERTER

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

TECHNOLOGY ASSESSMENT

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

MAGNET

For the hybrid design, the magnetic field strength at the conductor is 8 T. There is only modest value in raising this field strength, whereas there is a strong incentive to employ fields up to 16 T for the fusion reactor. The hybrid can employ Nb-Ti conductor. This conductor is already being used for somewhat lower fields and in smaller sizes, and it will be available at full requirements within a decade [on the same time scale as the Fusion Engineering Research Facility (FERF)¹⁶]. The fusion reactor will need high-field superconductors such as Nb₃Sn, which because of its brittleness is more

difficult to work with and will take longer to develop.

BLANKET STRUCTURAL MATERIAL

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

NEUTRAL-BEAM INJECTOR

The injection energy for minimum-cost fuel production by a hybrid reactor is ~100-keV D⁰ and T⁰. At this energy, a positive-ion technique with a charge-exchange cell giving 50% conversion to neutrals followed by a beam direct converter is expected to convert electrical power to neutral-beam power at 70% efficiency. Such a system¹⁷ (1 MW, 120 keV, and 0.5-s pulse length) will be tested within one year by Lawrence Livermore and Lawrence Berkeley Laboratories for eventual use on the Tokamak Fusion Test Reactor (TFTR), thus giving credence to the claim that such beams are nearly state of the art.

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

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

Requirement	Fusion-fission	Fusion
Magnet material	Nb-Ti(9 T)	Nb ₃ Sn(16 T)
Blanket structural material	Stainless steel	Niobium
Neutral-beam injector	Positive-ion type (~100 keV)	Negative-ion type (~200 keV)
Plasma direct converter	1 stage or none at all	Multistage
Required Q	0.5-1	>2
Fusion power	470 MW	1300 MW

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

ILASMA DIRECT CONVERTER

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

PLASMA

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

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

be reduced; this reduction helps stabilize the convective, high-density, loss-cone microinstability. However, the drift-cyclotron loss-cone microinstability is aggravated by larger orbits.

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

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

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

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

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

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

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

Parameter	MX	FERF	DEMO	Commercial
Field strength at conductor (T)	7	9	8	8
Plasma density (cm^{-3})	1×10^{14}	3×10^{14}	1.1×10^{14}	1.3×10^{14}
Plasma diameter (a)	0.6	0.5	2.4	4.8
Plasma length (m)	3.4	4.2	7.5	15
Q	—	0.3	0.7	0.7
Fusion power (MW)	0.3	3.4	44	475
Fissile fuel production rate (kg/yr)	—	3	190	2400
Cost	100 MS	500 MS	540 MS	2.3 BS

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

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

CONCLUSION

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

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

REFERENCES

1. This part of the arguments seems justified by our technological studies; however, it could be invalidated by social and institutional considerations such as extraordinarily expensive safety systems.
2. D. Jassby and J. D. Lee, "Counter Streaming Ion Tokamak Fissile Breeder," in these Proceedings.
3. K. R. Schultz, G. A. Backus, C. B. Baxi, J. B. Dee, E. A. Estrine, R. Rao, and A. R. Vaca, "Mirror Hybrid Reactor Blanket and Power Conversion System Conceptual Design," in these Proceedings.
4. D. J. Bender, "Mirror Hybrid Reactor Optimization Studies," in these Proceedings.
5. D. E. Dionigi and R. L. Engel, "The Economics of Fusion-Fission Systems," in these Proceedings.
6. J. D. Lee, "Neutronics of Sub-Critical Fast Fission Blankets for D-T Fusion Reactors," Proc. 7th Intersociety Energy Conversion Conf., 1972, p.1294 (unpublished); also J. D. Lee, "Neutronics Analysis of a 2500-MWth Fast Fission Natural Uranium Blanket for a D-T Fusion Reactor," Proc. 1st Topical Meeting on the Technology of Controlled Nuclear Fusion, San Diego, 1974, CONF-740402-Pl, p. 223 (unpublished).

7. L. D. Hansborough and R. W. Werner, "A Modular Fission-Fusion Hybrid Blanket," in Proc. First Topical Meeting on the Technology of Controlled Nuclear Fusion, San Diego, 1974, CONF-740402-P1, p. 211 (unpublished).
8. W. C. Wolkenhauer, C. W. Stewart, R. W. Werner, and J. D. Lee, "Conceptual Design of a Fusion-Fission Hybrid Reactor Based on a Mirror Fusion Reactor with a Subcritical Gas-Cooled Fission Blanket," in Proc. First Topical Meeting on the Technology of Controlled Nuclear Fusion, San Diego, 1974, CONF-740402-P1, p. 238 (unpublished).
9. B. R. Leonard, Jr., "A Review of Fusion-Fission (Hybrid) Concepts," Nucl. Technol. 20, 161 (1973).
10. R. W. Moir, "Conceptual Design Consideration for D-T Mirror Reactors with and without a Fusion Blanket," in Proc. First Topical Meeting on the Technology of Controlled Nuclear Fusion, San Diego, 1974, CONF-740402-P2, p. 373 (unpublished).
11. W. C. Wolkenhauer, C. W. Stewart, R. W. Werner, and J. D. Lee, "Some Safety Considerations of Hybrid Reactors in Comparison with Fission and Fusion Reactors," in Proc. First Topical Meeting on the Technology of Controlled Nuclear Fusion, San Diego, 1974, CONF-740402-P1, p. 588 (unpublished).
12. R. W. Moir, J. D. Lee, R. J. Burleigh, W. L. Barr, J. H. Fink, G. W. Hamilton, D. J. Bender, C. A. Carlson, W. L. Dexter, J. Holdren, C. L. Folkers, M. A. Peterson, M. E. Rensink, H. W. Patterson, R. L. Nelson, and C. E. Taylor, "Progress on the Conceptual Design of a Mirror Hybrid Fusion-Fission Reactor," Lawrence Livermore Laboratory Report No. UCRL-51797, 1975 (unpublished).
13. The ideas for orienting the magnet vertically and for using floats for moving the two halves of the magnet are due to J. N. Doggett.
14. J. D. Lee, "Blanket Design for the Mirror Fusion/Fission Hybrid Reactor," in these Proceedings.
15. J. H. Fink, W. L. Barr, and G. W. Hamilton, "A 225-MW Neutral Injection System for a Mirror Fusion-Fission Hybrid Reactor," Nucl. Fusion 15, 1067 (1975).
16. T. H. Batzer, R. C. Burleigh, G. A. Carlson, W. L. Dexter, G. W. Hamilton, A. R. Harvey, R. C. Hickma, M. A. Hoffman, E. B. Hooper, R. W. Moir, W. J. Silver, B. H. Smith, C. E. Taylor, R. W. Werner, and T. P. Wilcox, Conceptual Design of a Mirror Reactor for A Fusion Engineering Research Facility (FERF), Lawrence Livermore Laboratory Report No. UCRL-51617, 1974 (unpublished).
17. TFTR Neutral Beam Injection System Conceptual Design, Lawrence Berkeley Laboratory Report No. LBL-3296, 1975 (unpublished).
18. D. E. Baldwin, H. L. Berk, J. A. Byers, R. H. Cohen, T. A. Cutler, L. L. Lodestro, N. Maron, L. D. Pearlstein, M. E. Rensink, T. D. Rognlein, J. J. Stewart, D. C. Watson, and M. J. Gerver, Present Status of Mirror Stability Theory, Lawrence Livermore Laboratory Report No. UCRL-17038, Rev. 1, 1976 (unpublished).
19. F. H. Coensgen, MX Major Project Proposal, Lawrence Livermore Laboratory Report No. LLL-Prop-142, 1976 (unpublished).
20. J. N. Doggett, R. R. Vandervoort, and W. L. Barmore, A User's View of FERG, Lawrence Livermore Laboratory Report No. UCRL-76890, 1975 (unpublished).
21. J. D. Lee, private communication.

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

50,060

Blanket Design for the Mirror Fusion/Fission Hybrid Reactor

J. D. Lee

Originally issued as Lawrence Livermore Laboratory Preprint UCRL-78515 (July 13, 1976); prepared for inclusion in the Proceedings of the US-USSR Symposium on Fusion-Fission Reactors, Livermore, California, July 13-16, 1976.

BLANKET DESIGN FOR THE MIRROR FUSION/FISSION HYBRID REACTOR^{*}

J. D. Lee
Lawrence Livermore Laboratory
University of California
Livermore, California 94550

ABSTRACT

My purpose here is to summarize the conceptual blanket design that we have developed. I will briefly describe the evolution of our particular blanket concept, then discuss the geometry of our mirror reactor blanket and the replacement methods that we employ. I will describe the sub-module design and its thermal-hydraulic considerations. I will then describe the nuclear model geometry and compositions. The nuclear calculational method will be discussed as well as the data used. I will make a brief reference to calculational versus experimental results, then spend the remainder of my time describing the performance of the blanket fueled with three separate types of fuel.

INTRODUCTION

The evolution of our conceptual blanket design is documented in Refs. 1 - 8. The first reference was exploratory in nature and dealt only in the potential nuclear performance of fast fission blankets driven by fusion neutrons. The second reference deals specifically with a natural uranium-fueled fast fission blanket in which some of the engineering requirements were mocked up. Reference 3 compared results with two different data libraries, namely ENDF/B3 library and the Livermore ENDL library. The fourth reference, Progress On the Conceptual Design of a Mirror Hybrid Fission-Fusion Reactor, was our first attempt at putting together a hybrid system. This was the first time that we did some detailed blanket design resulting in an approximately consistent mechanical-nuclear design. The fifth reference compares calculation versus experimental results of a natural uranium pile driven by a 14-MeV neutron source. The last three references describe our attempts at improving the design discussed in Ref. 4 as well as optimizing it for the production of fissile fuel.

DESIGN OBJECTIVES

The basic blanket design objectives we are attempting to meet are listed below:

MECHANICAL

- Conform to plasma and coil geometries.
- Maximize coverage.
- Maximize fuel-to-structure ratio while maintaining structural integrity.
- Provide for replacement.

NUCLEAR

- Tritium breeding ($T \approx 1.05$).
- Maximize fissile breeding.
- Use fertile material to generate both additional energy and excess neutrons.
- Subcritical under all conditions.

Under mechanical, we have to conform to the plasma and coil geometries as dictated by the plasma physics requirements. At the same time we try to maximize coverage to intercept as many fusion neutrons as possible. The third objective under mechanical is to maximize the fuel-to-structure ratio. This is important from a neutronic standpoint since the presence of structure competes with the fuel and decreases nuclear performance. In this area we have to account for the structural requirements, namely containing the coolant and fuel and provide a thermal-hydraulic design which maintains temperatures of the structure, fuel, and the fuel clad at acceptable levels. The fourth

^{*}Work performed under the auspices of the U.S. Energy Research and Development Administration under contract No. W-7405-Eng-48.

item under mechanical is to provide for replacement of the blanket. This is important and must be considered from the beginning of any blanket design. We feel we have done this at least conceptually.

Under nuclear there are four general objectives. Of course the first is to breed tritium to fuel the DT reaction. We rather arbitrarily pick a desired tritium breeding ratio of 1.05 to make up for losses and have a reasonable doubling time to start up new plants. After meeting this tritium breeding requirement we then try to maximize the fissile fuel breeding in this design. To do this, we maximize the fission of normally fertile materials, U-238 or thorium, to gain energy and excess neutrons. The fourth nuclear design objective is we require the system to be sub-critical under all conditions.

MECHANICAL DESIGN

The basic mirror geometry is displayed in Fig. 1. It consists of two coils in the Yin-Yang configuration confining a plasma whose shape at the center is circular and fan shaped on the ends. Our blanket must be in between the plasma and the coils and intercept as many neutrons as possible. The blanket must also provide openings for the mirror leakage plasma and for injection paths.

Fig. 1. Basic mirror geometry.

Fig. 2. Blanket geometry 1.

To meet the mechanical objectives including geometry requirements imposed by the Yin-Yang configuration, we have developed two basic blanket geometries. The first (Fig. 2) consists of triangular slabs or wedges which surround the plasma while still providing openings for mirror leakage as well as injection ports.

Blanket replacement is accomplished in this geometry by sliding the blanket modules out through the coils; this requires no movement of the coils themselves. I will not dwell on this design since we feel our second and more recent design is superior. The major difficulty we have with this first design is that the wall loading varies by a factor of almost 10; the peak to average wall loading was about 4. This large variation resulted in difficult and economically not very attractive blanket fuel management.

The second blanket geometry we developed, and the one we are presently employing, is displayed in Fig. 3. The principal difference is the blanket is now spherical which leads to almost uniform wall loading over the entire blanket surface. As in the case of the first blanket, slots are provided in this one for leakage, and holes for neutral-beam injection.

Fig. 3. Blanket geometry 2.

Fig. 4. Blanket replacement scheme.

The blanket replacement method developed in Fig. 4. It requires the physical separation of the two magnet coils, which then exposes the blanket and allows the individual blanket segments to be removed and replaced. We believe the added expense of coil separation is more than offset by the improvement in blanket fuel management.

A conceptual layout of the nuclear island of our hybrid reactor is shown in Fig. 5. The coil is shown in the open position exposing the spherical blanket segments. Coil separation is accomplished by vertical movement of the upper coil. The upper coil is mounted on the "A" frame which is in turn mounted on floats. Vertical motion is achieved by flooding and draining the float canals.

Each blanket segment consists of a collection of sub-modules like those shown in Fig. 6. Each sub-module consists of an outer pressure vessel containing two fuel regions. The inner region, that region nearest the plasma, contains fertile material where both fission and capture of extra neutrons to breed fissile material occurs. The outer region contains a tritium breeding material such as lithium aluminate. Coolant is supplied to and removed from the sub-module by plena that make up the back of a blanket module. The

Fig. 5. Nuclear island of mirror hybrid reactor.

cooler inlet coolant first enters the sub-module and flows down in between the pressure vessel and the inner can which contains the fuel. By this method we can maintain the pressure vessel temperature near inlet conditions ($<400^{\circ}\text{C}$). The coolant then reverses direction and flows through the fuel. The coolant is pressurized helium. The principal reason why the blanket modules are segmented into sub-modules is to be able to contain the coolant pressure while still maintaining a rather thin first wall. A thin first wall is important in order to not degrade the fusion neutron energy significantly. With helium as our coolant we can employ gas-cooled fast reactor technology to the design of our blanket.

The thermal-hydraulic design of the sub-module is based on the requirement to maintain the pressure vessel, the fuel cladding, and the fuel itself below maxi-

Fig. 6. Blanket sub-module design.

mum operating temperatures. The design must also provide a high enough coolant temperature to be thermodynamically interesting and have an acceptable pressure drop. The thermal hydraulic design is described in Ref. 4, and is assessed by Schultz (these Proceedings).

NUCLEAR DESIGN

I would now like to discuss the nuclear design analysis and performance of the blanket. The principal motivation behind the fast fission blanket is the desire to use 14-MeV fusion neutrons to induce fission in a fertile material such as U-238. Both the energy and neutrons produced by fission are important. The additional energy improves the power balance of the system and the excess neutrons produce fissile fuel.

To calculate the performance of fission blankets we use existing neutron transport codes as well as existing nuclear data libraries. At Livermore we have generally used the TART Monte Carlo code and the Livermore Evaluated Nuclear Data Library to perform hybrid blanket calculations. To get an idea of how well we can calculate hybrid blanket performance, we have compared calculated results to experiments. One such experiment is the English Weale experiment.

This experiment consisted of a natural uranium pile approximately 1 m in diameter and 1 m high with a 14-MeV neutron source in the center. Table 1 shows the comparison between experimental results and what we calculated for this pile

Table 1. Reaction rates per 14-MeV source neutron integrated over the Weale pile.

Reaction	Calculation				Experiment
	Historical libraries		Current libraries		
	ENDF/B3	ENDL (old)	ENDF/B4	ENDL (new)	
U-235 (n,f)	0.226	0.287	0.228	0.266	0.281 ± 0.017
U-235 (n,γ)	0.055	0.064	0.057	0.061	—
U-238 (n,f)	0.828	1.31	0.949	1.11	1.18 ± 0.06
U-238 (n,γ)	4.03	4.63	4.27	4.36	4.08 ± 0.24
U-238 (n,2n)	0.328	0.417	0.358	0.388	0.277 ± 0.008
U-238 (n,3n)	0.173	0.179	0.176	0.195	0.327 ± 0.052
Leakage	0.284	0.587	0.296	0.504	0.42 ± 0.02

by using TART with both the old and new ENDF and ENDL libraries. Directing your attention to the U-238 fission and U-238 n, γ reaction rates, it appears the current methods and libraries are capable of calculating reasonably well both the fission and the capture reaction rates. Note that out of a total fission of 1.5 fissions per source neutron, only about 20% is from fusion of U-235; the remaining 80% is fission of U-238. Based on these experimental results, one can conclude that if you surround a fusion source with natural uranium you could expect approximately 300 MeV per source neutrons plus four capture reactions in U-238. Of course, a real blanket must have holes in it as I previously discussed, and it must contain structure and coolant materials as well as tritium breeding materials, all of which will degrade performance.

The structural and thermal-hydraulic requirements dictate how much structural material must be in the blanket. In our initial reference design studies (Ref. 4) we calculated a required fuel-to-structure ratio of 6.29. The fuel zones of the blanket consist of 8.6 vol % stainless steel plus 54% fuel. Fuel is 85% of theoretical density. The remaining 37% of the volume consists of helium coolant plus void. For calculational purposes the blanket is modeled as concentric spherical shells. The shells contain a homogeneous mixture of materials.

The innermost shell is 100% stainless steel, 0.5 cm thick, and approximates the first wall. The inner radius of this first zone is 10 m. The next shell is the fission zone consisting of 8.6% stainless steel (by volume), 54% fuel, and 37% void. The third shell is the tritium breeding zone and has the same material volume fractions as the fission zone. The fuel in the third zone consists of 44% graphite plus 10% lithium aluminate. Once the material composition in the zones is specified, the nuclear performance of the blanket is determined by calculating tritium breeding energy multiplication and the plutonium breeding versus fission zone thickness. The total blanket thickness is held constantly at 1 m. Results of such calculations are shown in Fig. 7.

As the fission zone thickness increases, the energy multiplication (M) and the plutonium breeding rate (Pu) increase while the tritium breeding rate (T) decreases. This is because while there are

more neutrons produced by fission, fewer are able to penetrate through the fission zone and enter the tritium breeding zone. To determine what fission zone thickness is needed, one has to have a good idea of what the blanket coverage will be. This is because, independent of coverage, one must maintain a tritium breeding ratio of slightly greater than unity. As blanket coverage decreases one must decrease the fission zone thickness to maintain the tritium breeding ratio needed. Thus, the overall effect of reducing blanket coverage is a marked decrease in performance. This effect is shown in Fig. 8. As you can see, if the blanket coverage is around 90%, then the fission zone thickness is 25 cm, which results in a blanket M of approximately 9 and a plutonium breeding rate of approximately 1.5. On the other hand, if blanket coverage drops to around 75%, the fission zone thickness must decrease to 10 cm to maintain T, and M drops to almost 4 and the plutonium breeding rate to approximately 0.6.

Three types of fission zone fuels for our fast fission blanket have been examined. They are uranium carbide, uranium plus 7 wt % molybdenum, and thorium. The U-Mo blanket is the best performer.

Some of the nuclear characteristics of the U-Mo blanket are shown on the next three figures. They are the neutron flux spectrum at three radial positions (Fig. 9), the total flux profile (Fig. 10), and the heat deposition profile (Fig. 11). At

Fig. 7. Nuclear performance of U-Mo blanket versus fission zone thickness.

Fig. 8. Effective blanket performance versus blanket coverage for U-Mo fuel and a tritium breeding ratio of 1.05.

Fig. 9. Neutron flux spectra in reference U-Mo blanket at three radial distances from first wall.

Fig. 10. Radial profile of total neutron flux in reference U-Mo blanket at a wall loading of 1.15 MW/m^2 .

Fig. 11. Thermal power density in reference U-Mo blanket at a wall loading of 1.15 MW/m^2 .

a wall loading at 1 Mw/m^2 , the first wall sees a flux of approximately 5×10^{14} at energies greater or equal to 1 MeV. Thermal power profile is peaked at the first wall with a peak to average of approximately 2. Since the fuel volume fraction is 0.54, peak fuel power density is $\sim 140 \text{ W/cm}^2$.

The final step in the nuclear analysis of the blankets is to estimate exposure effects. This was done in a first-order manner by iterating in time with rather large time steps and using average reaction rates for the whole fission zone.

Figure 12 shows the effect of long-time exposure of the uranium carbide blanket. Blanket energy multiplication (M) at the start of life is 8 and climbs to 32 after a first-wall exposure of $\sim 20 \text{ Mwyr/m}^2$. This increase in multiplication occurs because of the buildup of the plutonium, represented here by Pu. At start of life, Pu is at 0% and after 25 years it has peaked at almost 10% of the initial U loading. At this point the net plutonium breeding rate has dropped to 0 (its conversion ratio has dropped to 1), since plutonium is being consumed as fast as it is being bred. This blanket is subcritical over its entire life; at time 0, the neutron multiplication or K is approximately 0.3. After 20 years when the peak occurs, K has increased to approximately 0.7. It is possible that the blanket could reconfigure into a critical assembly and therefore, such high integrated wall loading is not attractive. We will see later that economics dictates that the blanket be removed at a relatively low exposure compared to 20 Mwyr/m^2 .

Figure 13 shows the blanket performance versus exposure for the three types of fuels we have looked at, namely uranium carbide, U-Mo, and thorium. In each case, the exposure range is limited to exposures of interest. We do not want the uranium carbide to go past about 6 Mwyr/m^2 , the U-Mo to go beyond 4 Mwyr/m^2 , or the thorium to go past 10 Mwyr/m^2 . The limit on the uranium carbide is criticality in the event of reconfiguration. The limit on the U-Mo is burnup of the fuel itself since after a burnup of 1 to 2%, fuel

swelling gets out of hand. The limit on the thorium is structural lifetime of the pressure vessel and fuel cladding. We will see later that even without these constraints, economics dictates that the blankets be removed before these limits are reached.

The major performance parameters of the three blankets are compared in Fig. 14. In each case the fission zone is sized to provide the required tritium breeding. From a strictly blanket performance point of view, the U-Mo is obviously the best with uranium carbide somewhat less and the thorium blanket showing up a poor third. The optimum exposures and the relative economics of the mirror hybrid using these blankets are discussed by Dave Bender (these Proceedings).

Fig. 12. Uranium carbide blanket performance versus exposure.

Fig. 13. Blanket performance versus exposure for (a) uranium carbide, (b) U-Mo, and (c) thorium.

Fig. 14. Comparison of performance versus exposure for blanket fueled with uranium carbide (UC), U-Mo, and thorium.

REFERENCES

1. J. D. Lee, Neutronics of Sub-Critical Fast Fission Blankets for DT Fusion Reactors, Lawrence Livermore Laboratory Rept. UCRL-73952 (1972).
2. J. D. Lee, Neutronic Analysis of a 2500 MWth Fast Fission Natural Uranium Blanket for a DT Fusion Reactor, Lawrence Livermore Laboratory Rept. UCRL-75304 (1974).
3. R. C. Haight and J. D. Lee, Calculations of a Fast Fission Blanket for DT Fusion Reactors with Two Evaluated Data Libraries, Lawrence Livermore Laboratory Rept. UCRL-75627 (1974).
4. R. W. Moir et al., Progress On the Conceptual Design of a Mirror Hybrid Fusion-Fission Reactor, Lawrence Livermore Laboratory Rept. UCRL-51797 (1975).
5. R. C. Haight et al., Reaction Rates in a Uranium Pile Surrounding a 14-MeV Neutron Source: Calculations of the Weal Experiment, Lawrence Livermore Laboratory Rept. UCRL-77364 (1975).
6. J. D. Lee et al., Optimizing the Mirror (Fusion-Fission) Hybrid Reactor for Plutonium Production, Lawrence Livermore Laboratory Rept. UCRL-76986 (1975).
7. J. D. Lee et al., Mirror Reactor Blankets, Lawrence Livermore Laboratory Rept. UCRL-77083 (1976).
8. D. J. Bender and J. D. Lee, The Potential for Fissile Breeding with the Fusion-Fission Hybrid Reactor, Lawrence Livermore Laboratory Rept. UCRL-76986 (1976).

56061

Reaction Rates in a Uranium Pile Surrounding a 14-MeV Neutron Source: Calculations of the Weale Experiment

R. C. Haight, J. D. Lee, and J. A. Maniscalco

Originally issued as Lawrence Livermore Laboratory Preprint UCRL-77364 Rev. 1
(January 19, 1976); published in *Nuclear Science and Engineering*, Vol. 61,
No. 1, pp. 53-59 (September 1976), reprinted by permission.

Reaction Rates in a Uranium Pile Surrounding a 14-MeV Neutron Source: Calculations of the Weale Experiment

R. C. Haight, J. D. Lee, and J. A. Maniscalco

Lawrence Livermore Laboratory, University of California
P. O. Box 808, Livermore, California 94550

Received October 27, 1975

Revised March 3, 1976

To validate the neutronics analysis of hybrid fusion-fission reactor blankets, calculations were made of an experiment by Weale et al., where a 14-MeV neutron source was surrounded by a natural uranium metal pile. The evaluated nuclear data libraries, ENDF-B and ENDF-B, were used. The calculated parameters were found to be in closer agreement for present versions of these libraries than for preceding versions; however, there were still 15% differences in the $^{235}\text{U}(n,f)$ and $^{238}\text{U}(n,f)$ reaction rates. The present version of ENDF gives the results that are the closest to the experimental values for these reactions. For the $^{238}\text{U}(n,\gamma)$ reaction, the calculations with the ENDF-B libraries are closer to the measured values. Both the ENDF and ENDF-B evaluations, however, fail to calculate correctly the neutron leakage or derived values for the $^{238}\text{U}(n,2n)$ and $^{238}\text{U}(n,3n)$ reaction rates. The spatial variations of the $^{235}\text{U}(n,f)$, $^{238}\text{U}(n,\gamma)$, $^{238}\text{U}(n,f)$, and $^{239}\text{Pu}(n,f)$ reaction rates show that the penetration of high-energy neutrons in the pile is better described by the calculation with ENDF, which gives a greater penetration. The effects of resonance self-shielding were investigated and found to require a much smaller correction than the differences between calculations with different data libraries.

1. INTRODUCTION

Fusion-fission hybrid reactors, as they are now conceptualized, consist of a D-T fusion plasma surrounded by a fissionable blanket of thorium

or uranium and usually a tritium-breeding region. The neutronics of many of the conceptual blankets have been calculated both for magnetically confined plasma systems¹⁻¹¹ and for laser-imploded fusion systems.¹² These studies show that the

¹R. W. MOIR et al., "Progress on the Conceptual Design of a Mirror Hybrid Fusion-Fission Reactor," UCRL 51797, Lawrence Livermore Laboratory (1975).

²J. D. LEE, "Neutronic Analysis of a 2500 MW_{th} Fast Fission Natural Uranium Blanket for a DT Fusion Reactor," *Proc. 1st Topl. Mtg. Technology of Controlled Nuclear Fusion*, April 16-18, 1974, San Diego, California, CONF-740402-P1, Vol. 1, p. 233, U.S. Atomic Energy Commission (1974).

³W. C. WOLKENHAUER, B. R. LEONARD, Jr., A. M. SUTLEY, and R. W. MOIR, "Conceptual Design of a Fusion-Fission Hybrid Reactor Based on a Mirror Fusion Reactor with a Subcritical Gas Cooled Fission Blanket," *Proc. 1st Topl. Mtg. Technology of Controlled Nuclear Fusion*, April 16-18, 1974, San Diego, California, CONF-740402-P1, Vol. 1, p. 238, U.S. Atomic Energy Commission (1974).

⁴T. A. PARISH and E. L. DRAPER, "Neutronic and Photonic Analysis of Fusion Reactor Blankets Containing Natural Uranium," *Proc. 1st Topl. Mtg. Technology of Controlled Nuclear Fusion*, April 16-18, 1974, San Diego, California, CONF-740402-P1, Vol. 1, p. 256, U.S. Atomic Energy Commission (1974).

⁵J. A. MANISCALCO and L. L. WOOD, "Advanced Concepts in Fusion-Fission Hybrid Reactors," UCRL 75835, Lawrence Livermore Laboratory (1974).

⁶B. R. LEONARD, *Nucl. Technol.*, **20**, 161 (1973).

⁷B. R. LEONARD, Jr. and W. C. WOLKENHAUER, "Fusion-Fission Hybrids: A Subcritical Thermal Fission Lattice for a DT Reactor," *Proc. Univ. Tex. Conf. Technology of Controlled Thermonuclear Fusion*, November 20-22, 1972, Austin, Texas, also, BNWL-SA-4390, Battelle Pacific Northwest Laboratories (1972).

⁸J. D. LEE, "Neutronics of Subcritical Fast Fission Blankets for D-T Fusion Reactors," *Proc. 7th Conf. Intersociety Energy Conversion Engineering*, p. 1294, American Chemical Society (1972).

⁹L. M. LIDSKEY, "Fission-Fusion Symbiosis: General Considerations and a Specific Example," *Proc. Nuclear Fusion Reactors Conf.*, September 17-19, 1969, Culham, England, paper 12 (1969).

¹⁰L. N. LONTAI, "Study of a Thermonuclear Reactor Blanket with Fissile Nuclides," MS Thesis, Technical Report No. 436, Massachusetts Institute of Technology (1965).

¹¹S. LOCKE BOGART, Ed., *Proc. DCTR Fusion-Fission Energy Systems Review Mtg.*, December 3-4, 1974, Germantown, Maryland, ERDA-4, U.S. Energy Research and Development Administration.

¹²J. A. MANISCALCO, *Nucl. Technol.*, **28**, 98 (1976).

fusion-fission hybrid concept has favorable potential in several respects:

1. The energy produced per fusion reaction can be increased by an order of magnitude or more over that of a pure fusion reactor.
2. Natural uranium or thorium can be used in the blanket, thus obviating the need for expensive isotope separation.
3. Fissile fuel, e.g., ^{235}U or ^{239}Pu , can be bred in the blanket.
4. Tritium breeding can be comparable with breeding in a pure fusion reactor.

The system studies have approached all these aspects and have been able to optimize the designs on one or more of them.

In predicting the neutronic performance of these blankets, it is, of course, crucial to use the most reliable nuclear data in the neutron transport codes. Previous studies have assessed the reliability of these input data in two ways. One approach was to compare the results obtained with two different data libraries.¹³ Although the evaluated data in the libraries are based on the same set of experimental measurements, the evaluations are usually otherwise independent. The comparison, therefore, indicates credible differences in the system parameters if gross errors are assumed to be absent from the evaluated data. The maximum credible differences are not determined in this comparison, however, as there are apt to be compensating discrepancies in the evaluations.

The second approach has been to compare calculation with experiment for relevant integral experiments. We do not know of any experiments for a hybrid reactor with a thorium blanket. For a reactor with a natural uranium blanket, however, there are two types of integral experiments. One type is the Lawrence Livermore Laboratory (LLL) ^{238}U pulsed sphere experiment,¹⁴ in which a pulsed 14-MeV neutron source was surrounded by a ^{238}U spherical shell of either 0.8 or 2.8 mean-free-paths thick. The neutron emission spectra from these relatively thin shells were measured

by time-of-flight techniques. The second type of integral experiment was investigated by Weale et al.,¹⁵ who surrounded a 14-MeV neutron source with a massive natural uranium metal pile. They measured the reaction rates for several nuclear reactions as a function of position. Because the reaction rate is the integral over the neutron energy of the flux spectrum times the energy-dependent cross section, the results give an indication of the neutron flux spectrum. Directly, however, their measurements give the rates for very important reactions in a hybrid reactor blanket, namely the $^{238}\text{U}(n,f)$, $^{238}\text{U}(n,f)$, $^{238}\text{U}(n,\gamma)$, and $^{239}\text{Pu}(n,f)$ reactions.

In this work we have calculated quantities measured by Weale et al. because of the similarity of the experimental pile to conceptual hybrid reactor blankets containing natural uranium. Four data libraries were used: the Evaluated Nuclear Data File ENDF B-IV (Ref. 16), the LLL Evaluated Nuclear Data Library ENDL (Ref. 17), present version dated February 18, 1975 [which will hereafter be denoted as ENDL (new)], and the predecessors of these two libraries, ENDF B-III and ENDL, version dated January 25, 1974 [which will hereafter be called ENDL (old)].

One comparison to make is among the results of the four calculations themselves. Existing discrepancies can be traced directly to differences in the evaluations of the microscopic data. The detection of these relevant differences makes possible their resolution based on experimental data that either already exist or can be measured.

The second and more important comparison is between calculated and experimental results. The adequacy or inadequacy of the individual data library to calculate a 0.1- to 0.5-m-thick natural uranium blanket surrounding a 14-MeV neutron source is indicated. Many of the conceptual hybrid reactor blankets have dimensions in this range.

In addition to making calculations with the four data libraries, we investigated the effects of resonance self-shielding by substituting a corrected cross-section set in the resonance region. Such corrections make little difference in the calculated integral quantities.

Previous calculations of this experiment were

¹³R. C. HAIGHT and J. D. LEE, "Calculation of a Fast Fission Blanket for DT Fusion Reactors with Two Evaluated Data Libraries," *Proc. 1st Topl. Mtg. Technology of Controlled Nuclear Fusion*, April 16-18, 1974, San Diego, California, CONF-740402-P1, Vol. 1, p. 271, U.S. Atomic Energy Commission (1974).

¹⁴K. WONG, J. D. ANDERSON, R. C. HAIGHT, L. F. HANSEN, and T. KOMOTO, " ^{238}U Pulsed Sphere Measurements and CRF Fusion-Fission Blanket Calculations," *Proc. Conf. Nuclear Cross Sections and Technology*, Washington, D.C., Mar. 3-7, 1975, U.S. National Bureau of Standards Special Publication 425, Vol. II, p. 704 (1975).

¹⁵J. WEALE, P. GODFELLOW, M. H. McTAGGART, and M. L. MULLENDER, *J. Nucl. Energy A/B*, **14**, 94 (1961).

¹⁶M. K. DRACK, "Data Formats and Procedures for the ENDF Neutron Cross Section Library," BNL-50274 (ENDF-102, Vol. 1), Brookhaven National Laboratory (1970).

¹⁷R. J. HOWERTON, R. J. DOYAS, T. C. MICHELS, and S. I. PERKINS, "An Integrated System for Production of Neutronics and Photonics Calculational Constants," *Evaluated Nuclear Cross Section Library*, Vol. 4, UCRL-50400, Lawrence Livermore Laboratory (1971).

made by Wolkenhauer et al.¹⁸ and by Newman and Leonard.¹⁹ In both works the ENDF/B-III evaluated library was used in an S_n calculation. Our calculations with this library started from the same evaluated data set but were independent thereafter.

II. CALCULATIONAL MODEL AND METHOD

The experimental apparatus of Weale et al. consisted of a pile of natural uranium metal fuel rods surrounding a 14-MeV neutron source. The rods were each 2.92 cm in diameter and 106.6 cm long. They were oriented parallel to each other in a hexagonal, almost close-packed configuration. The separation of the centers of adjacent rods was 2.98 cm. The rods were held in place by a steel base plate, steel spacing pins, and aluminum plates on a steel framework. These supporting elements were near the ends of the fuel rods. The overall diameter of the cylindrical pile that we used in our calculations was 99 cm. This was the largest of the assemblies in Ref. 15. The experimental results of the smaller assemblies did not seem to differ significantly except that they did not test the deep penetration so well.

The source neutrons in Ref. 15 were produced by a 200-keV deuteron beam incident on a Zr-T target, which was at the position of the center rod. The beam entered the pile 55.9 cm below the top surface of the pile. The drift tube for the beam entered the pile through a channel 7.0 cm deep and 2.2 cm wide, which extended from the outside of the pile to the center.

The experimental data consisted of reaction rates for a variety of nuclear reactions as a function of position in the pile. Some of the rates were measured by activation foils, whereas others were determined by fission chambers. The $^{63}\text{Cu}(n,2n)^{62}\text{Cu}$ reaction rate was interpreted in Ref. 15 to give the penetration of 14-MeV neutrons in the pile. Other measured reaction rates included the $^{235}\text{U}(n,f)$, $^{238}\text{U}(n,f)$, $^{238}\text{U}(n,\gamma)$, and $^{239}\text{Pu}(n,f)$ reactions, all of which are of great importance for a hybrid blanket. The first two reactions determine the power production in the blanket. The third reaction contributes to neutron economy and may justify the hybrid reactor con-

cept economically by breeding ^{239}Pu . The fourth reaction burns up the bred ^{239}Pu and at the same time contributes to energy production.

Our calculational model contained several simplifying assumptions. First, because the supporting structure of the pile was far from the neutron source, it was ignored. The pile was considered to consist wholly of natural uranium (0.72 at.% ^{235}U , 99.28 at.% ^{238}U). Second, because the fuel rods were nearly close-packed, the pile was homogenized in the calculation, the average density being 16.3 g/cm³. Effects of streaming between fuel rods were considered negligible. Third, the void region near the neutron source was ignored. This approximation makes a negligible difference in the total reaction rates and causes a small displacement, roughly equal to the radius of the cylindrical void (i.e., 1.5 cm) in the reaction rates as a function of position.

Two zonings were used, one for k_{eff} reaction rates and the other for reaction rates as a function of position. The zoning for the latter, more complicated case is given in Fig. 1. The geometry is cylindrical.

The evaluated data were processed into energy-averaged group constants with the CLYDE code.²⁰ The group structure has been described elsewhere.²¹ Angular distributions were translated into 16 equally probable intervals of $d(\cos \theta)$. The energy distributions of emitted particles were processed into 32 equally probable energy intervals for all reactions but fission, where 256 intervals were used.

The neutron-transport calculations were done with the TART code,²² which is an analog Monte Carlo code. The source neutrons were taken to be isotropic and monoenergetic, with an energy of 14.06 MeV. Between 4000 and 10 000 source neutrons were followed, depending on the calculation. The reaction rates were determined by TART, which integrates the flux over the reaction cross section in each zone. For the $^{239}\text{Pu}(n,f)$ reaction, a trace amount of ^{239}Pu was entered into the relevant zones.

²⁰R. J. DOYAS, T. C. MICHELS, S. T. PERKINS, and R. J. HOWERTON, "An Integrated System for the Production of Neutronics and Photonics Calculational Constants," CLYDE, a Code for the Production of Calculational Constants from Nuclear Data, Vol. 5, UCRL-50400, Lawrence Livermore Laboratory (1971).

²¹E. F. PLECHATY, D. F. CULLEN, R. J. HOWERTON, and J. R. KIMLINGER, "An Integrated System for the Production of Neutronics and Photonics Calculational Constants," Tabular and Graphical Representation of 175 Neutron Group Constants Derived from the LLL Evaluated Nuclear Data Library (ENDL) Vol. 1b, UCRL-50400, Lawrence Livermore Laboratory (1975).

²²E. F. PLECHATY and J. R. KIMLINGER, Private Communication (1971).

¹⁸W. C. WOLKENHAUER, B. R. LEONARD, Jr., U. P. JENQUIN, D. F. NEWMAN, A. M. SUTLEY, C. W. STEWART, D. L. PREZBINDOWSKI, R. W. MOIR, J. D. LEE, and R. W. WERNER, "Status Report: Mirror Hybrid Reactor Studies," Appendix B, BNWL-1835, Battelle Pacific Northwest Laboratories (1974).

¹⁹D. F. NEWMAN and B. R. LEONARD, Jr., "Correlation of Reaction Rates in Uranium Metal Using a 14 MeV Neutron Source," BNWL-SA-5288, Battelle Pacific Northwest Laboratories (1975).

Fig. 1. Zoning used in the calculation of reaction rates as a function of position. Numbers are displacements from the center-line or midplane in centimeters. Note that the figure is not to scale.

Statistical errors of the Monte Carlo calculation of the individual reaction rates are not computed by TART. We can, however, estimate their magnitude from the number of source neutrons and the neutron spectrum as a function of position. For the total reaction rates, we estimate that the statistical errors are $<5\%$, which is slightly better than the experimental errors quoted in Ref. 15. For the reaction rates as a function of position, the errors are larger. Near the outer edge of the pile, errors of 30% are possible. We estimate that the errors, toward the center of the pile, are no worse than 10%.

Resonance self-shielding in the ^{238}U of the pile was investigated by substituting self-shielding-corrected group cross sections for the flat-flux-weighted cross sections in the new version of ENDL. The absorption cross section for ^{238}U was corrected to account for Doppler-broadened resonance self-shielding with General Atomic's²³

GGC-4 multigroup cross-section code.^{24,25} The Doppler-broadened corrections were made for a temperature of 300 K.

In performing the resonance calculations, GGC-4 uses a nuclear data library tape²⁶ that was also generated at General Atomic. The microscopic cross-section data and the resonance parameters on the nuclear data tape were obtained and processed from ENDF/B-III. Consequently, the input to TART in this case was a combination of corrected ENDF/B-III cross sections below a certain energy and flat-flux-weighted ENDL (new) cross sections above that energy. Two dividing energies were chosen, 47 and 182 keV, to investigate the dependence of the results on the region corrected for self-shielding.

III. RESULTS AND DISCUSSION

III.A. Total Reaction Rates

The results for the total reaction rates in the pile are given in Table I. The values calculated from the four different evaluated libraries can be compared with each other and with the experimental values.

The comparison of the calculated quantities indicates that the current versions of the ENDL and ENDF/B libraries give results that do not differ as widely as those calculated with preceding versions of these libraries. For example, the important $^{238}\text{U}(n,f)$ reaction rate differed by 58% between ENDF/B-III and ENDL (old), whereas it now differs by only 17%. The $^{238}\text{U}(n,\gamma)$ reaction rate differed by 15%, whereas now only a 2% difference is calculated. Only the $^{238}\text{U}(n,n)$ / ^{235}U reaction rate shows a wider discrepancy in calculations with the present libraries, but this discrepancy is not huge.

Nevertheless, the present versions of ENDL and ENDF/B do give results that, although less discrepant than previously, still differ significantly. The 17% difference in the $^{238}\text{U}(n,f)$ rate and the 17% difference in the $^{235}\text{U}(n,f)$ rate are two particularly striking examples. The 70% difference in the leakage is the worst disagreement. Clearly, for a hybrid reactor design, such uncertainties in the calculated system parameters are unacceptable.

²⁴J. ADIR and K. D. LATHROP, "Theory of Methods Used in the GGC-4 Multigroup Cross Section Code," GA-0921, General Atomic Company (1968).

²⁵J. ADIR, S. S. CLARK, P. FROELICH, and L. J. TODT, "Users and Programmers Manual for GGC-3 Multigroup Cross Section Code," parts 1 and 2, GA-7157, General Atomic Company (1967).

²⁶M. K. DRAKL, "Description of the Auxiliary Codes Used in the Preparation of Data for the GGC-3 Code," GA-7158, General Atomic Company (1967).

²³Reference to a company or product name does not imply approval or recommendation of the product by the University of California or the U.S. Energy Research and Development Administration to the exclusion of others that may be suitable.

TABLE I
Reactions per 14-MeV Source Neutron Integrated over the Pile

Reaction	Calculation				Experiment ^a
	Historical Libraries		Current Libraries		
	ENDF/B-III	ENDL (old) ^b	ENDF/B-IV	ENDL (new) ^c	
²³⁵ U(n,f)	0.226	0.287	0.228	0.266	0.281 ± 0.017
²³⁵ U(n,γ)	0.055	0.064	0.057	0.061	---
²³⁸ U(n,f)	0.828	1.31	0.949	1.11	1.18 ± 0.06
²³⁸ U(n,γ)	4.03	4.63	4.27	4.36	4.08 ± 0.24
²³⁸ U(n,2n)	0.328	0.417	0.358	0.388	0.277 ± 0.008 ^d
²³⁸ U(n,3n)	0.173	0.179	0.176	0.195	0.327 ± 0.052 ^d
Leakage	0.284	0.587	0.296	0.504	0.42 ± 0.02

^aIntegrated reaction rates were determined by drawing smooth curves through points measured at various positions in the pile and then integrating. The errors are those assigned in Ref. 15.

^bENDL (old) is the version of the LLL Evaluated Nuclear Data Library dated January 25, 1974.

^cENDL (new) is the version of ENDL dated February 18, 1975.

^dExperimental values were actually derived based on the 14-MeV flux as determined by the ⁶³Cu(n,2n)⁶²Cu results and on an assumed value of the ²³⁸U(n,2n)²³⁷U or ²³⁸U(n,3n)²³⁶U cross section at 14 MeV.

When compared with experiment, the quantities calculated with the new libraries generally show better agreement than those calculated with the previous versions. Serious discrepancies with experiment exist for all the ²³⁸U(n,2n) and ²³⁸U(n,3n) reaction rate calculations. These quantities were not directly measured in the experiment; instead, they were deduced from the cross section near 14 MeV and the 14-MeV flux, which was itself deduced from the measured ⁶³Cu(n,2n) reaction rate. This deductive path could introduce further errors because the ⁶³Cu(n,2n) reaction (threshold = 11.5 MeV) and the ²³⁸U(n,2n) reaction (threshold = 6.2 MeV) are particularly sensitive to neutrons below 14 MeV. It should be noted that the sum of the deduced ²³⁸U(n,2n) and ²³⁸U(n,3n) reactions is quite close to the sum of the reactions calculated with each library. Another serious discrepancy for all the calculations is in the neutron leakage. This quantity was measured directly, and hence the discrepancies appear to be real. It is of some comfort that the results given by the current libraries bracket the experimental leakage rate.

Of the current libraries, ENDL is seen in Table I to do better on calculations of the ²³⁸U(n,f) and ²³⁵U(n,f) reaction rates. ENDF/B-IV, on the other hand, is somewhat more successful on the ²³⁸U(n,γ) reaction rate. The conclusions for the ²³⁸U reactions are consistent with comparisons between the two libraries when they were used to calculate the ²³⁸U pulsed spheres.¹⁴ In that work, it was concluded from analysis of the neutron emission spectrum that ENDL (new) did better on

the high-energy (>2.5 MeV) neutrons, which determine the ²³⁸U(n,f) rate, and ENDF/B-IV did better on the low energy (<1 MeV) neutrons that are most important for the ²³⁸U(n,γ) rate.

The effects of resonance self-shielding on this system are given in Table II. The uncorrected results are those calculated with ENDL (new). The results with the correction for self-shielding were obtained with the corrected group-averaged cross sections from ENDF/B-III substituted for the ENDL (new) cross section below the indicated energy as described in Sec. II.

Differences between these results are seen to be significantly smaller than the differences in Table I between calculations based on completely different evaluations. Until the evaluations become more consistent, therefore, it does not seem worthwhile to worry about self-shielding corrections from this system. The differences existing in Table II may be statistical to a large extent. For example, the ²³⁸U(n,2n) and ²³⁸U(n,3n) reaction rates should not be affected by self-shielding because the corrections are made well below their thresholds and the only up-scattering mechanism is fission, which puts very few neutrons above their thresholds. The relatively large change in ²³⁵U(n,f) with the self-shielding correction seems statistically significant, but may simply be caused by differences in the ENDF/B-III and ENDL evaluations below 182 keV. It is also conceivable that the lack of large change in the ²³⁸U(n,γ) and ²³⁸U(n,γ) reaction rates could be due to the compensating effects of self-shielding and the differences in the evaluated data.

TABLE II

Calculated Reaction Rates per 14-MeV Source Neutron Without and With Resonance Self-Shielding Corrections: ENDL (new) Evaluated Library Corrected as Described in Text

Reaction	Without Resonance Correction	With Resonance Correction	
	ENDL (new)	Corrected to 47 keV	Corrected to 182 keV
$^{235}\text{U}(n,f)$	0.266	0.264	0.238
$^{235}\text{U}(n,\gamma)$	0.061	0.061	0.053
$^{238}\text{U}(n,f)$	1.11	1.083	1.065
$^{238}\text{U}(n,\gamma)$	4.36	4.20	4.18
$^{238}\text{U}(n,2n)$	0.388	0.381	0.379
$^{238}\text{U}(n,3n)$	0.195	0.192	0.191
Leakage	0.504	0.495	0.429
Energy deposited (MeV)	269	263.2	255.4

III.B. Positional Distribution of Reaction Rates

The positional distribution of the reaction rates is given in Figs. 2 through 5. In the figures, the experimental data are denoted by points and the calculations based on ENDL (new) and ENDF/B-IV by the solid and dashed lines, respectively. The calculations were made at the positions of the experimental data points, and smooth curves were drawn through them. Except near the edge of the assembly, the calculations gave the same results for either the horizontal or vertical displacements from the source for a given source-to-detector separation. Near the edges, only the horizontal results are given.

In Ref. 15 the experimental results are separated into the reaction rates induced by 14-MeV neutrons and those induced by secondary (non-14-

MeV) neutrons. This separation depends on the determination of the 14-MeV flux through the $^{63}\text{Cu}(n,2n)$ measurements and on the 14-MeV cross section of the measured reaction. In this work, the total reaction rate is reconstructed by summing the two parts. The total rate is the quantity measured directly.

The largest difference in the distributions is for the $^{238}\text{U}(n,f)$ reaction rate. The calculation using the ENDL (new) evaluation comes much closer to the experimental values than does that with ENDF/B-IV. This result was expected because the high-energy part of the neutron emission spectrum from the first few interactions of a 14-MeV neutron with natural uranium determines the probability of further fissioning of ^{238}U , which has an (n,f) threshold of ~ 1.5 MeV. These high-energy neutrons were investigated in the 14-MeV pulsed sphere experiments¹⁹ and were found to be better described by the ENDL (new) evaluation than by the ENDF/B-IV evaluation.

Fig. 2. Distribution of the $^{238}\text{U}(n,f)$ reaction rate. Here R is the distance to the 14-MeV neutron source. The experimental values are the horizontal scan data of Ref. 15.

Fig. 3. Distribution of the $^{238}\text{U}(n,\gamma)$ reaction rate. Here R is the distance to the 14-MeV neutron source. The experimental data that were taken from Ref. 15 are for the horizontal scan (circles) and the vertical scan at a horizontal distance of 10.5 cm from the center of the pile (squares).

Fig. 4. Distribution of the $^{235}\text{U}(n,f)$ reaction rate. Here R is the distance to the 14-MeV neutron source. The experimental values are the horizontal scan data of Ref. 15.

Smaller differences are found for the $^{235}\text{U}(n,f)$, $^{238}\text{Pu}(n,f)$, and $^{238}\text{U}(n,\gamma)$ reactions. All these reactions are particularly sensitive to neutrons below 1 MeV. The more correct treatment of neutrons in this range by ENDF/B-IV (Ref. 14) compensates for the deficiencies in the higher energy neutrons. The magnitudes of these reaction rates calculated with ENDF/B-IV are less than those calculated with ENDL (new) because the former calculates fewer ^{238}U fissions and hence fewer neutrons in the assembly.

Calculations with ENDF/B-III (not shown in the figures) were compared with the results of Wolkenhauer et al.¹⁸ for the positional dependence of reaction rates. The agreement was quite good. The only significant discrepancy was that the present Monte Carlo calculation gave a slightly greater penetration of MeV neutrons than did ANISN. Newman and Leonard¹⁹ noted previously that the P_3 S_4 approximation used in their work and in Ref. 18 may give an anomalously low penetration for high-energy neutrons.

IV. SUMMARY

Because of the similarity to a fusion-fission hybrid reactor blanket, the experiment of Weale et al. has been calculated with four evaluated data libraries. The current versions of the ENDF/B and ENDL libraries give total reaction rates differing less than those of the previous versions. Nevertheless, there still remain significant differences, especially in the ^{238}U fission rate and in the neutron leakage. These differences result from differences in the neutron emission spectrum from 14-MeV neutrons incident on natural uranium.

Fig. 5. Distribution of the $^{239}\text{Pu}(n,f)$ reaction rate. Here R is the distance to the 14-MeV neutron source. The experimental values are the horizontal scan data of Ref. 15.

It should be emphasized that the disagreements between the calculations with the two current evaluated data libraries imply significant credible errors in the calculated parameters of hybrid reactors with natural uranium blankets. The maximum credible errors may be even larger due to the possibility that compensating errors exist in the two libraries.

In comparison with experiment, the results using ENDL (new) give better agreement for the $^{238}\text{U}(n,f)$ and $^{235}\text{U}(n,f)$ reaction rates both for the positional distributions and the integrated values. The ENDL library also reproduces more successfully the $^{239}\text{Pu}(n,f)$ reaction rate distribution. On the other hand, the results with ENDF/B-IV are somewhat closer to the measured $^{238}\text{U}(n,\gamma)$ reaction rate. These results are expected on the basis of a previous study of 14-MeV pulsed ^{238}U spheres. Neither library is successful in calculating the neutron leakage or derived values of the $^{238}\text{U}(n,2n)$ and $^{238}\text{U}(n,3n)$ reaction rates.

Finally, the effects of self-shielding show that these corrections are considerably less than the differences between results calculated with the current versions of the two data libraries. Consequently, the evaluations must be brought into line with each other and with experiment before much effort is expended in the self-shielding correction for hybrid reactor blankets.

ACKNOWLEDGMENTS

We wish to thank M. H. McTaggart and E. Pendlebury for sending further details of the experiment. The TART calculations were facilitated thanks to help from R. Dye and F. Piechaty.

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

50-62

Mirror Hybrid Reactor Blanket and Power Conversion System Conceptual Design

K. R. Schultz, G. A. Backus, C. B. Baxi, J. B. Dee, E. A. Estrine,
R. Rao, and A. R. Veca

Originally issued as General Atomic Company Report GA-A14008 (June 29, 1976); prepared for inclusion in the Proceedings of the US-USSR Symposium on Fusion-Fission reactors, Livermore, California, July 13-16, 1976.

MIRROR HYBRID REACTOR BLANKET AND POWER CONVERSION SYSTEM CONCEPTUAL DESIGN*

K. R. Schultz, G. A. Backus, C. B. Baxi, J. B. Dee,
E. A. Estrine, R. Rao, and A. R. Veca
General Atomic Company
P.O. Box 81608, San Diego, CA. 92138

ABSTRACT

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

INTRODUCTION

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

BLANKET DESIGN

FUEL DESIGN

Fuel Materials. The candidate fuel materials that have been evaluated for use in the MHFFR include uranium alloys, uranium oxide, uranium carbide, and uranium nitride. A comparative evaluation was made in terms of the important parameters, such as fuel density, neutron economy, physical,

mechanical and corrosion properties, irradiation effects, compatibility with cladding and with impurities in the helium coolant (particularly tritium), and fabrication cost. Two important characteristics of the hybrid reactor are the low burnup³ ($\leq 1\%$) and the importance of high fuel density.²

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

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

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

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

various fuels is affected by four factors:

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

Table 1. Fuel material candidates.

	U-MOLY	UC	U ₃ Si
Uranium Density (g/cc)	17.1	13.0	15.5
Maximum Operating Temperatures (°C)	700	2000	900
Approximate Rod Diameter (mm)	10	18	17
Blanket Cost (M\$)	154	112	94
Cost (M\$/Yr)	--	+17.1	+24.1
Revenues (M\$/Yr)	--	-41.7	-17.0
Net Benefit (M\$/Yr)	--	-24.6	+7.1

Compared to UO₂ cost as a base, it is estimated that U-Moly will be three times as expensive (due to stringent Q/A requirements), UC will be twice as expensive and U₃Si cost will be on par with UO₂.⁵ The fabrication cost of blanket rods is a function of the number of rods in the reactor. Due to the lower peak temperature permissible in the U-Moly, the allowable rod diameter is smallest among the three fuels. The allowable rod size for UC and U₃Si is approximately the same as the cladding temperature is the limiting parameter. Estimated blanket costs are shown in Table 1. These estimates include fuel material and fabrication costs.

Revenues from power and fissile production are roughly proportional to the uranium density in the fertile material. The estimates of changes in power and fissile production with U-Moly as reference, are shown in Table 1. Taking U-Moly as the reference case, UC represents a net loss of \$24.6M/year and U₃Si represents a net benefit of \$7.1M/year.

In addition to the small economic advantage, U₃Si has the advantages of requiring less stringent quality assurance, of allowing larger fuel rods and of reduced parasitic neutron absorption when compared to U-Moly. U₃Si was chosen as the fuel material for the MHFFR

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

Table 2. Fuel configuration options.

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

Honeycomb material has the potential for a high fuel density. Due to the small heat transfer coefficient in the blanket very small holes would be required which would result in high fabrication costs. The technology for fabrication and cladding of honeycomb fuel structure is not developed. A plate configuration has the potential of offering a large heat transfer surface. Bonding technology appears to be difficult to achieve for U₃Si and the preferred cladding material, Inconel. A rod configuration appears to be better suited to application in the hybrid reactor.

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

Fuel Performance. With the fuel materials selected as U₃Si fuel and Inconel 718 clad, maximum design temperature limits were chosen. The helium inlet and outlet temperature conditions were selected to correspond to the fuel temperature limits although these helium conditions were not optimized. The temperatures are shown on Table 3.

Table 3. MHFTR temperatures ($^{\circ}\text{C}$)

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

The fuel design was based on hot spot temperatures obtained by applying hot spot factors to the nominal temperatures calculated for an average channel. End-of-life conditions were used since these represent the maximum blanket multiplication and power density. Hot spot factors were determined for the MHFTR using a semi-statistical method⁷ and are shown on Table 4.

Table 4. Hot spot factors for the MHFTR blanket.

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

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

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

Table 5. Helium pressure drop in module.

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

MODULE DESIGN

Structural Material. To insure good neutronics performance of the blanket, it is important that the first wall be kept

as thin as possible. It must operate, however, at high temperature under high pressure. Inconel 718 was selected for the blanket structural material and fuel cladding material because it exhibits high strength at elevated temperature. The design stress limit for Inconel at 800°C is 296 MPa (43,000 psi).

Fig. 1. Power and temperature distribution.

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

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

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

Fig. 2. MHFFR submodule.

Fig. 3. MHFFR blanket module

chick. A reentrant coolant flow system was chosen which allows the external module walls to be kept near the coolant inlet temperature of 280°C. The helium coolant flow path leads into the inlet plenum at the back of the module. From there the coolant helium flows radially outward through the U₃Si fuel region and the LiAlO₂ tritium breeding zone. The hot

helium is collected in the outlet plenum and flows to the power conversion system.

POWER CONVERSION SYSTEM

SYSTEM PARAMETERS

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

Table 6. MHFFR system parameters

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

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

Fig. 4. MHFFR power flow diagram.

Table 7. Reactor power

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

PRIMARY COOLANT SYSTEM

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

Fig. 5. MHHFR primary loop arrangement.

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

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

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

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

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

SECONDARY COOLANT SYSTEM

Steam Loop Design. The steam loop of the power conversion system is quite conventional. Steam raised in the 16 once-through steam generators is routed through the helium circulator steam turbines, is reheated and then fed to the main turbine generator. The turbine is a tandem-compound machine with an inlet pressure of 5.1 MPa (744 psi) and inlet temperature of 503°C (937°F). The turbine exhausts to a conventional water-cooled condenser. These steam conditions are quite modest; the temperatures are limited by the capabilities of the hybrid blanket fuel and cladding materials. The total thermal power input to the power conversion

system from the blanket and direct converter is 4381 MW, the output from the turbine generator is 1574 MWe and the power conversion system efficiency is 35.9 percent. Due to the large power requirement of the neutral beam injectors the net station efficiency is 22.1 percent.

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

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

POWER CONVERSION SYSTEM LAYOUT

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

Table 8. Reactor trip parameters

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

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

Fig. 6. MHFFR quadrant concept.

SAFETY

MHFFR SAFETY ANALYSIS

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

Table 9. Potential accidents

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

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

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

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

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

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

Fuel Rearrangement. Major changes in the fuel geometry could potentially be

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

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

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

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

Helium Leakage Accidents. The primary hazard from helium leakage is from coolant

borne fission products and tritium. Preliminary estimates indicate that tritium contributes little to this risk and that the release of the circulating activity to the atmosphere would result in off-site doses within 10 CFR 100 guidelines.

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

DESIGN BASIS ACCIDENT

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

LICENSING

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

COST

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

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

CONCLUSIONS

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

There are no inherent features of the hybrid concept that present fundamentally new safety considerations to the reactor design. Further work is needed, however, in the area of the primary loop ducting

design, support and containment to assure that the reactor will be adequately cooled under all circumstances. The apparent total absence of potential accidents that could cause the system to become supercritical does offer the hybrid reactor some advantages compared to fission reactor systems. If the maintenance of cooling can be adequately assured, the hybrid reactor could be licensed under current Nuclear Regulatory Commission regulations.

REFERENCES

1. K. R. Schultz, et al., "Conceptual Design of the Blanket and Power Conversion System for a Mirror Hybrid Fusion-Fission Reactor," GA-A14021, to be published.
2. R. W. Moir, et al., "Progress on the Conceptual Design of a Mirror Hybrid Fusion-Fission Reactor," UCRL-51797 (June 25, 1975).
3. J. D. Lee, D. J. Bender, and R. W. Moir, "Optimizing the Mirror (Fusion-Fission) Hybrid Reactor for Plutonium Production," UCRL-76986 (Nov. 17, 1975).
4. G. H. Chalder, et al., "U₃Si as a Nuclear Fuel," AECL-2874 (May 1967), Atomic Energy of Canada Limited.
5. M. T. Simnad, "Fuel Element Experience in Nuclear Power Reactors," AEC Monograph, Amer. Nuclear Society, (1971).
6. R. Leggett, et al., "Achieving High Exposure in Metallic Uranium Fuel Elements," Nuclear Appl. Tech., 9, 673 (Nov. 1970).
7. M. D. Carelli, and D. R. Spencer, "CRBRP Assemblies Hot Channel Factors Preliminary Analysis," WARD-D-0050, 1974.

50,063

Mirror Hybrid Reactor Optimization Studies

D. J. Bender

**Originally issued as Lawrence Livermore Laboratory Preprint UCRL-78614
(September 9, 1976); prepared for inclusion in the Proceedings of the US-USSR
Symposium on Fusion-Fission Reactors, Livermore, California, July 13-16, 1976.**

MIRROR HYBRID REACTOR OPTIMIZATION STUDIES*

D. J. Bender

Lawrence Livermore Laboratory, University of California
Livermore, California 94550

ABSTRACT

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

INTRODUCTION

The primary objectives of the first Lawrence Livermore Laboratory (LLL) fusion-fission hybrid reactor point design¹ were to determine a manner in which all of the necessary system components could be integrated into the reactor, to assess the technological problems, and to obtain a rough cost estimate. The resulting design was not optimized in either an engineering or economic sense, but rather was a reference point design from which further study could proceed.

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

a determination of the reactor parameters which minimize the cost of producing fissile fuel. The optimization thus minimizes the electricity cost component that is attributable to the fissile fuel burned by the fission reactors.

RESULTS AND DISCUSSION

SYSTEM MODEL

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

• Reactor Description

- Plasma physics
- Magnet design
- Blanket geometry
- Power flow
- Capital cost

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

- Fuel Management

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

- "Nuclear Park" Economics

- Hybrid and fission reactors

The mirror reactor model (essentially the analysis developed by Carlson²) includes mirror plasma physics, magnet design, blanket geometry and power flows. Capital costs are a key element in the analysis, and here we have attempted to be as thorough and consistent as possible. However, the costing procedure entails a high degree of uncertainty due to the infancy of fusion technology.

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

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

RESULTS OF PARAMETRIC ANALYSIS

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

- Plasma Physics

- Injection energy
- Mirror ratio
- Injection angle
- Confinement time

- Magnet Design

- Conductor field
- Mirror-to-mirror length

- Fuel Management

- Maximum fertile burnup
- Exposure distribution within the blanket

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

The geometric relationship between the plasma and blanket are shown in Fig. 1. Holes in the blanket must be provided for plasma leakage and neutral injection. We have found that if the power density in the blanket exceeds 100 to 200 W/cm³, the plenum dimension required to handle the helium flow becomes excessively large, pushing the blanket inward and severely decreasing the blanket coverage. Also, large neutral beam

Fig. 1. Blanket coverage.

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

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

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

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

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

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

Table 1. Description of thermal converter reactors.

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

make-up requirements. For a Pu burner, we have used a light water reactor (LWR) on a Pu recycle fuel cycle which is supplemented with hybrid Pu. As a ^{233}U burner, we have used a high-gain high temperature gas-cooled reactor (HTGR) on the thorium- ^{233}U fuel cycle. Another possi-

bility as a ^{233}U burner, but not yet examined, is the CANDU reactor.

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

Fig. 2. The LWR fuel cycle.

Fig. 3. The HTGR fuel cycle.

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

The LWR fuel cycle⁵ (Fig. 2) uses plutonium recycle and has a conversion ratio of 52%. Fissile feed consists of the ^{235}U component of the natural U fertile feed combined with hybrid-produced ^{239}Pu . Here we have assumed that hybrid Pu (>90% fissile) is equivalent to Pu from a uranium-fueled LWR (~70% fissile). In future studies we plan to examine a variation of this cycle that will permit higher uranium utilization. The uranium will be recycled and the hybrid Pu feed increased to compensate for the absence of ^{235}U . Thus, the hybrid fuel cycle may be designed to ultimately fission the majority of the uranium.

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

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

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

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

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

Fig. 4. The plasma physics variation.

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

There is some degree of uncertainty as to the actual plasma Q that will be attained in mirror reactors. It is possible that microinstabilities will limit Q to a value somewhat below the classical value.⁸ A second possibility is that Q enhancement techniques under consideration⁹ will elevate Q above our presently predicted values. Figure 8 shows that electricity costs are not strongly perturbed even if classical confinement is not attained. A two-fold enhancement of Q improves the economics of the hybrid reactor but, in general, electricity costs are rather insensitive to higher Q .

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

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

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

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

Table 2. Parameters for the optimized hybrid reactors.

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

OPTIMIZED REACTOR CONFIGURATIONS

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

- The uranium blanket, because of its high energy multiplication, results in a plant with a large electrical output. The thorium blanket reactor does not produce net electricity, just fissile fuel.
- Both blankets have about the same thermal rating. This results from the much greater fusion power required for the thorium blanket reactor as compared to the uranium blanket reactor.
- The high fusion power of the thorium blanket reactor is obtained by using a more intense magnetic field than for the uranium blanket reactor. Therefore, the latter may rely on existing NbTi superconductor magnet technology, whereas, the thorium blanket reactor will require the more technologically advanced Nb₃Sn superconductor.
- Optimum economics for the thorium blanket are obtained at high exposure, about 9 MW-yr/m². The uranium blanket requires only about 4 MW-yr/m². These results assume that the blanket structure is capable of tolerating these exposures.

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

Table 3. Blanket parameters for the optimized reactors.

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

The blanket parameters for the optimized reactors are listed in Table 3. Both blankets produce about 2 1/2 metric tons of fissile fuel per year. However, the thorium blanket requires a rather high exposure, and the possibility of the blanket structure being able to attain ~9 MW-yr/m² exposure is quite uncertain. For the uranium blanket, the average energy multiplication is higher by about a factor of four than for the thorium blanket; these blanket energy multiplications include the effect of the fractional blanket coverage.

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

Table 4. Economics for the optimized hybrid reactors.

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

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

The fission reactor economics are listed in Table 5. Here, the category "fuel cycle without fissile material" refers to all normal fuel cycle charges excluding the cost of fissile fuel, i.e., fabrication, reprocessing, spent-fuel shipping and purchase of fertile fuel. The fissile fuel cost is the cost of producing this material in the hybrid re-

Table 5. Economics for the fission reactors.

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

actor. Here, the important result is that the hybrid fissile fuel costs of 4.1 and 5.3 mills/kW-hr are a small fraction of the total cost of the electricity. Based on our current capital cost model, we conclude that the mirror hybrid reactor is capable of converting the world's large fertile resources into fissile fuel at a cost that does not strongly influence the net cost of electricity.

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

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

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

The fact that the hybrid fissile production costs are capital cost dominated (see Table 4) dictates that we regard these fissile costs with a degree of uncertainty, reflecting both the infancy of fusion engineering and our inability to accurately assign costs to reactor components which are merely conceptual designs. However, the presently predicted costs are within the realm of being economically attractive. In our opinion, this conclusion justifies vigorous support of the hybrid concept, with future design efforts continuing to refine the engineering and to incorporate

experimental results as they become available. The result could well be an energy option that will ease the transition to a full fusion power economy in the next century and will provide comparatively early benefits from the large R&D investment that will be required to commercialize this energy source.

REFERENCES

1. R. W. Moir, et al., "Progress on the Conceptual Design of a Mirror Hybrid Fusion-Fission Reactor," Lawrence Livermore Laboratory Report, UCRL-51797, 1975 (unpublished).
2. G. A. Carlson and R. W. Moir, "Mirror Fusion Reactor Study," Lawrence Livermore Laboratory Report, UCRL-76985, 1975 (unpublished).
3. J. D. Lee, "Blanket Design for the Mirror Fusion/Fission Hybrid Reactor," these proceedings.
4. Guide for Economic Evaluation of Nu-

- clear Reactor Plant Designs, NUS Corporation Report, NUS-531, Appendix E, 1959 (unpublished).
5. T. H. Pigford and K. P. Ang, "The Plutonium Fuel Cycles," Health Physics, 29, 451 (1975).
6. R. H. Brogli and K. R. Schultz, "Thorium Utilization in an FBR/HTGR Power System," presented at American Power Conference, Chicago, Ill., April 1974.
7. R. H. Brogli, General Atomic Co., (private communication) 1976.
8. M. E. Rensink, et al., "Theoretical Studies of Plasma Confinement in Magnetic Mirrors," Lawrence Livermore Laboratory Report, UCRL-75993, 1974 (unpublished).
9. W. C. Condit, T. K. Fowler, and R. F. Post, Status Report on Mirror Alternatives, Lawrence Livermore Laboratory Report, UCRL-52008, 1976 (unpublished).
10. Summary Report, Fuel Cycle Conference '75, Atomic Industrial Forum, 1975 (unpublished).