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MIRROR HYBRIDS - A STATUS REPORT

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MIRROR HYBRIDS - A STATUS REPORT*

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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.

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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 2XIIB.(6)

Recent Experimental Results From 2XIIB

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 nτ scales with mean plasma energy (E) in the predicted way ($\propto E^{3/2}$) up to 13 keV. The nτ 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.

Proposed Next Mirror Experiment MX

A large mirror experiment using superconducting Yin-Yang coils with neutral beam injection is designed to push the plasma parameters towards the reactor regime (see Table 1). The scaling laws ($n\tau \propto E^{3/2}$, $n \propto B^2/E$, plasma stream requirements, i.e., $n_{warm}/n_{hot} \propto dn/dr$, etc) from 2X will be examined over an extended parameter range and the quasisteady-state nature (0.5 sec and on towards truly steady state) will be tested.

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

Changes In The Plasma Parameters Since The 1975 Study (1)

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 there by increasing the central field. In so doing, the density increased by a factor of 2.2. The penetration of the plasma by the netural 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 2XIIB method of using streams of relatively cold plasma to stablize the plasma can apparently be used to stablize 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 increaseing 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. F CILITY 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 B magnetic well, and is sustained by the injection of beams of energetic neutral atoms (D°, T°) 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 st. cture 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 fusio. .

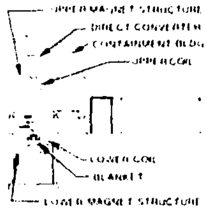

FIGURE 3-1. Plant Layout for Mirror Hybrid.

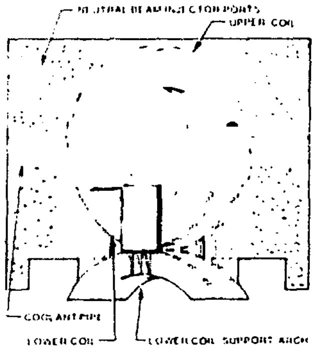

FIGURE 3-2. Magnet Structure

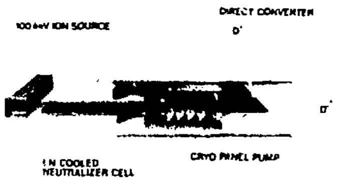

FIGURE 3-3. Neutral Beam Injector

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 presetntly 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) cryopumping 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 accomodate 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.

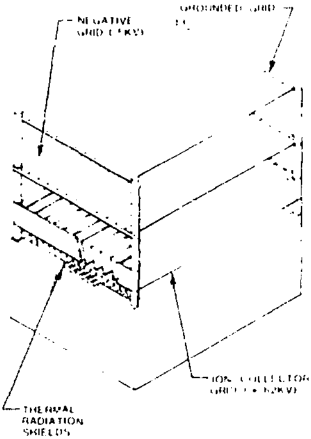

FIGURE 3-4. Direct Converter

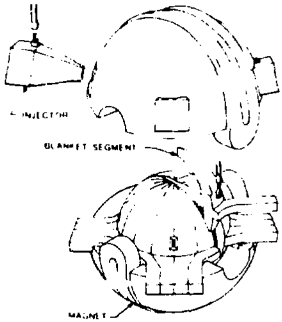

FIGURE 4-1. Blanket Geometry

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

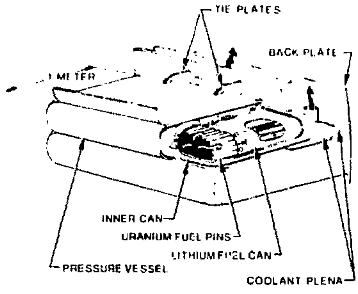

FIGURE 4-2. Blanket Submodule

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 principle 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.

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 $\sim$ 270 MeV per 14 MeV neutron plus $\sim$ four 238-U (n,γ) reactions. Of course, a blanket must a 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/o 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 MeV) neutron flux at the first wall is 5×10^{14} $CM^{-2}\text{-}s^{-1}$ 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 valued are start of life. Fuel pin power density will increase as M increases with exposure.

The final step in the nuclear

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 ~ 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 ~ 2 a/o.

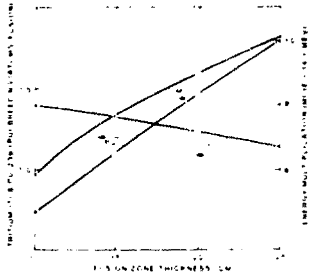

FIGURE 4-3. Nuclear Performance of U-Moly Blanket vs Fission Zone Thickness

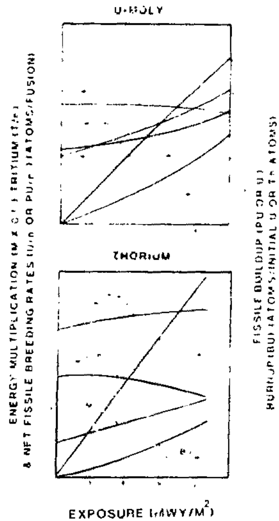

FIGURE 4-4. Blanket Nuclear Performance vs Exposure

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 summerizes the General Atomic Co. blanket study.

Fuel Design

The canadidate 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_3Si
Uraninum Density (g/cc)	17.1	13.0	15.5
Maximum Operating Temperature (°C)	700	2000	900

Since U_3Si has the advantages of requiring less stringent quality assurance, of allowing larger fuel rods and of reduced parasitic neutron absorption when compared to U-Moly, U_3Si 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 [illegible]

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.

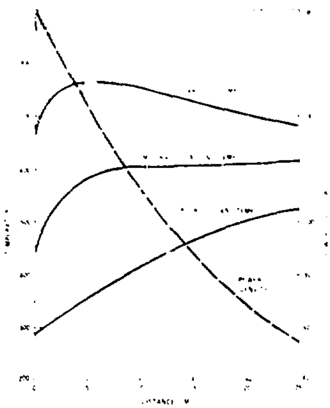

FIGURE 5-1. Nominal Temperature Distributions in the Blanket

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.

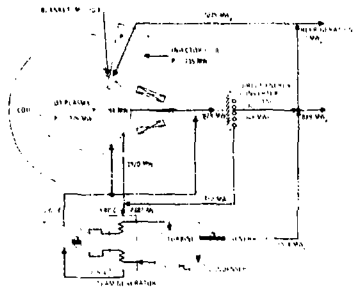

FIGURE 6-1. Power Flow Diagram

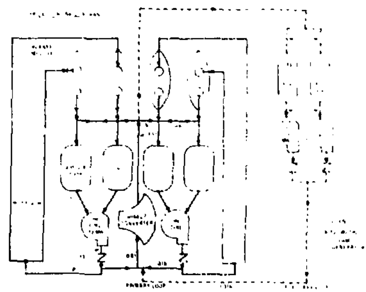

FIGURE 6-2. Primary Loop Arrangement

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 substantial 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 bakced 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 on 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 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 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.

E. REACTOR OPTIMIZATION STUDIES

The LLL hybrid study this year has concentrated on optimizing the hybird 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 electricity cost component attributable to fissile fuel of the fission reactors which the 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 signigicantly affect the hybrid performance. In addition, the plant economics are influenced by the blanket fissile management scheme and be 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 comparted 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 ~9 MW - yr/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 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 $Q_{classical}$. 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 $Q/Q_{classical}$ 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.

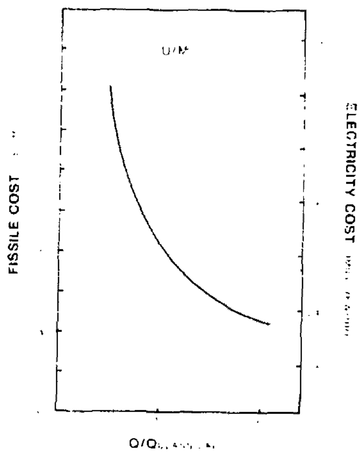

FIGURE 8-1. Fissile Cost vs Q/Q Classical

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 8-1

THERMAL CONVERTER REACTORS

	Pu Burner	^{233}U Burner
REACTOR TYPE	LWR	high-gain HTGR
FUEL CYCLE		
FERTILE FEED	Nat. U	Th
FISSILE FEED	Pu	^{233}U
FISSILE RECYCLE	Pu	^{233}U
CONVERSION RATIO	0.5	0.8
FISSILE MAKE-UP REQUIREMENTS (kg/yr/MWe)	0.333	0.185

Table 8-2
OPTIMIZED HYBRID REACTOR PARAMETERS

	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
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

Table 8-3
OPTIMIZED HYBRID BLANKET PARAMETERS

	U/Mo	Th
FISSILE OUTPUT (kg/yr)	2360	2590
AVG. ENERGY MULTIPLICATION	11.1	2.8
BLANKET COVERAGE	0.86	0.77
FERTILE BURNUP (%)	1.0	0.5
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/Mo	Th
CAPITAL COST (10^9 \$)	2.3	3.3
(\$/kWe)	2200	
FISSILE MATERIAL COST (\$/gm)	55	127
CAPITAL	60	103
FUEL CYCLE	13	21
O & M	1	1
ELECTRICITY COST	-39	2
ELECTRICITY COST FROM HYBRID AND FISSION REACTORS (mills/kw-hr)	24.8	25.3

Table 8-5
FISSION REACTOR ECONOMICS

	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
O & M	0.7	0.7
FISSILE FUEL (from hybrid)	4.1	5.3

Table 8-6
NUCLEAR PARK ECONOMICS

	U/Mo	Th
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	19.7	20.5
FUEL CYCLE	4.3	4.0
O & M	0.8	0.8

CONCLUSIONS

This paper summarized the status of the Mirror Hybrid Program by describing the design and performance of our interium 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 comprable 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/el 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 98.

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. Oddon, "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, el 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, "U_3Si 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.