MASTER

Conf-771056--9

PREPRINT UCRL- 80076

Lawrence Livermore Laboratory

The Mirror Hybrid (Fusion-Fission) Reactor

D. J. Bender, J. D. Lee, W. S. Neef, R. S. Devoto, T. R. Galloway, J. H. Fink*
K. R. Schultz**, D. Culver**, R. Rao **, S. Rao**

October 1977

* On loan from Westinghouse Research Laboratory, Pittsburgh, Penn. 15235
** General Atomic Company, San Diego, CA 92138

This paper was prepared for publication in the Proceedings of the IAEA Conference and Workshop on Fusion Reactor Design, October 1977, Madison, Wisconsin.

THE MIRROR HYBRID (FUSION-FISSION) REACTOR

D. J. Bender, J. D. Lee, W. S. Neef, R. S. Devoto, T. R. Galloway, J. H. Fink*,
K. R. Schultz**, D. Culver**, R. Rao**, S. Rao**

* On loan from Westinghouse Research Laboratory, Pittsburgh, Penn. 15235
** General Atomic Company, San Diego, CA 92138

INTRODUCTION

At Lawrence Livermore Laboratory, we have been investigating the hybrid reactor concept for about five years. Prior to 1975 the limited hybrid studies concentrated on blanket neutronics, accompanied by some preliminary design work [1]. In 1975 we performed our first conceptual hybrid reactor design [2], and in 1976 we emphasized reactor optimization. In addition, at this time General Atomic Company joined us in a joint study effort to apply their expertise in gas-cooling technology to the mirror hybrid. The results of the 1976 studies were reported at the Livermore Conference [3].

At the present time, we (LLL/GA) are completing a reference design for the mirror hybrid. Our design is intended to be an early commercial facility, illustrating the characteristics of a reactor with near-term physics and technology.

Prepared for publication in the Proceedings of the IAEA Conference and Workshop on Fusion Reactor Design, October 1977, Madison, Wisconsin.

Work performed under the auspices
of the U.S. Department of Energy
under contract No. W-7405-Eng-48.

Characteristics for the reactor are listed in Table 1.

REACTOR CHARACTERISTICS

Table 1

Fusion Power	400 MW
Thermal Power (Avg.)	3600 MW
Injected Neutral Power	625 MW
Net Electric Output Power	525 MW
First Wall 14 MeV Neutron Current	2 MW/m^2
Fissile Production Rate	2700 Kg/yr
Injection Energy D^o	125 keV
T^o	187 keV
β	0.7
Central Ion Density	9×10^{13} cm^{-3}
Q	0.63
$n\tau$	2×10^{13} sec/cm^3

PLASMA PARAMETERS

Table 2

	2XII-B	MFTF	COMM. HYBRID
Plasma Length (m)	1.6	3.4	13
Conductor Field (T)	-	7	8
β	0.4-0.7	0.5	0.7
Injection Energy (keV)	20.0 (40.0)	80.0 (20.0)	125.0
$n\tau$ (s/cm^3)	$7x10^{10}$	$>10^{12}$	$\sim10^{13}$

an outside diameter of about 22 metres, and a distance of 13 metres between mirror points. It is designed with a maximum field at the conductor of 8 Telsa, dictated by the use of NbTi superconductor. The maximum current density is about 10^3 A/cm^2 in the bundle cross-section and the resulting coil-case pressure is about 2000 psi. These conditions imply comparatively modest magnet technology, although the magnet is quite large, about 5500 tonnes for each magnet half (including the stainless steel coil case).

The injector design developed for the reference hybrid is based on the positive ion LBL injector with the following modifications. A hollow cathode ion source is proposed as a means of providing a much longer-lived cathode than the heated tungsten wire cathode now used in the LBL source. A mechanical design for the sources has been developed which will place all high-voltage insulators out of the path of the direct 14 MeV neutron flux. A Hg vapor jet at the end of the neutralizer region strongly impedes gas flow from the source into the downstream regions of the beam line, resulting in much lower background pressure in the beam line and reduced cryopumping requirements. High pressure vacuum pumping in the region of the sources will use Hg diffusion pumps. A cryopanel design has been developed which will permit continuous operation of the cryopumping equipment in the beam line. The design permits defrosting of some of the cryopanels while the remainder continue in operation.

The reactor requires deuterium injectors with acceleration to 125 keV and tritium to 187 keV. When account is taken of the half and third energy components in the beam, the average beam energies are 94 and 141 keV, respectively, for D^o and T^o. Our analysis predicts an efficiency for the injectors of 55%.

Outboard of the coils, end tanks must be provided to receive the plasma leakage. In the end tanks, we perform direct conversion, converting some of the kinetic energy of the ion flow directly into electricity. The remaining

TIME-DEPENDENT U_3Si BLANKET NEUTRONIC PARAMETERS

Table 3

Exposure ($MW\text{-}yr/m^2$)	M	Pu/n	% Pu	Burnup %	T/n
0	8.8	1.85	0	0	1.05
5	18.4	1.75	2.3	0.75	1.42

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

The average of the tritium breeding ratios (T/n) quoted in Table 3 are greater than one to compensate for 14 MeV neutrons lost through holes in the blanket and decay of the tritium inventory.

MECHANICAL DESIGN

One of our primary concerns in the mechanical design of the reactor was to provide highly reliable support and containment of the blanket and primary heat on the modules. The maintenance operation consists of a series of manipulations of each of the several hundred modules.

The module, as shown in Figure 5, consists of a cylindrical pressure vessel with a hexagonal base. One of the more challenging aspects of the module design has been to devise a fast, reliable method of making up the seal that isolates the high pressure He coolant from the vacuum region that contains the plasma. We have discarded a welded joint, since remote grinding and welding are time consuming operations and we have serious doubts about the ability to consistently generate remote vacuum-tight welds. We have therefore adopted a bolted joint using a double knife-edge (Varian type) seal with differential pumping between the two knife-edges. The pressure vessel is bolted in place with 6 bolts, one at each corner of the hex-shaped base. The internals of the module are fabricated as a single unit, containing the U_3Si pins, the tritium breeding pins and internal flow ducting. Thus, to renew a module the pressure vessel is unbolted and removed, the pin assembly is removed, a new pin assembly is inserted and a new pressure vessel is bolted in place. The coolant flow is re-entrant, with the tritium pins being cooled by the inlet flow and the coolant then proceeding down to the first wall, turning, and cooling the uranium pins on its exit path out through the module.

POWER CONVERSION LOOP

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

REFERENCES:

[1] Proc. First Topical Meeting on the Technology of Controlled Nuclear Fusion, San Diego, (1974), CONF-740402 - P1, Chap. 4.

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

[3] Proc. US-USSR Symposium on Fusion-Fission Reactors, Lawrence Livermore Laboratory, (1976), CONF-760733.

[4] B. McNamara, et al., "Theory of Mirror Machines at High Beta", 6th Int. Conf. Plasma Physics and Controlled Nuclear Fusion Research (Proc. Conf. Berchtesgaden, 1976) IAEA, Vienna, IAEA-CN-35/C3; also, Lawrence Livermore Laboratory Report UCRL-78123 (1976).

[5] Killeen, J., Mirin, A. A., Rensink, M. E., "The Solution of the Kenetic Equations for a Multispecies Beam", Ch. XI, Methods in Computational Physics: Vol. 16, Controlled Fusion, Killeen, J., Ed., Academic Press, New York (1976).

[6] B. W. Stallard, "Radial Plasma Buildup Code for Neutral Beam Injection Into A Mirror Machine", Lawrence Livermore Laboratory Report UCRL-51784 (1975).

[7] B. W. Stallard and R. S. Devoto, "Computation of Plasma Build-Up In Mirror Machines", 13th Int. Conf. on Phenomena in Ionized Gases (Proc. Conf. Berlin, 1977); also, Lawrence Livermore Laboratory Report UCRL-79399 (1977).

MAGNET CONDUCTOR CONFIGURATION

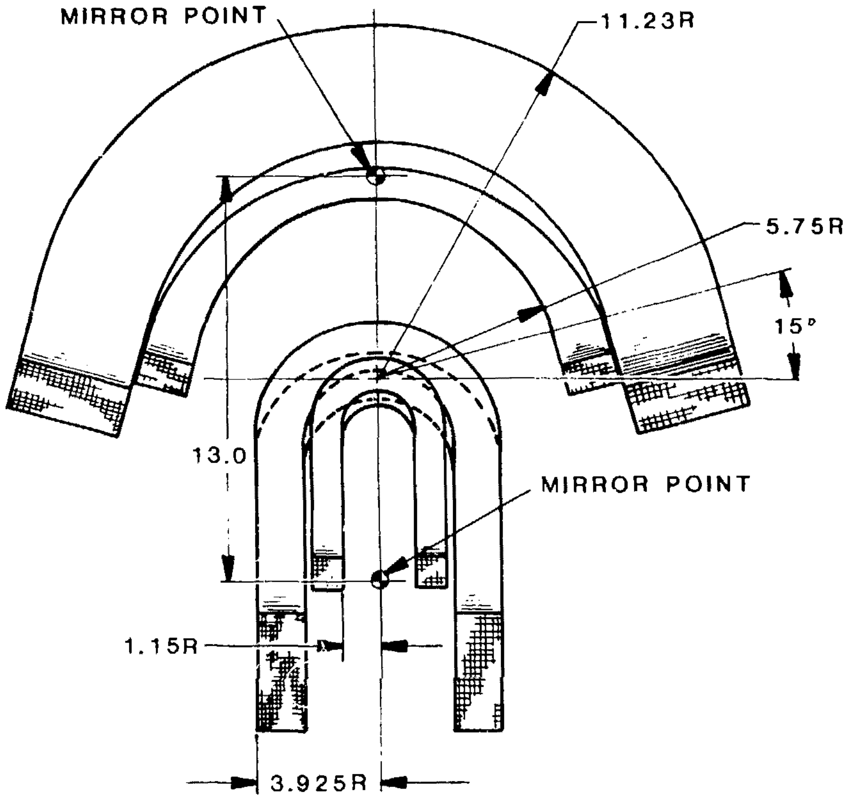


FIGURE 1

FUSION-FISSION MIRROR HYBRID REACTOR

LAWRENCE LIVERMORE LABORATORY

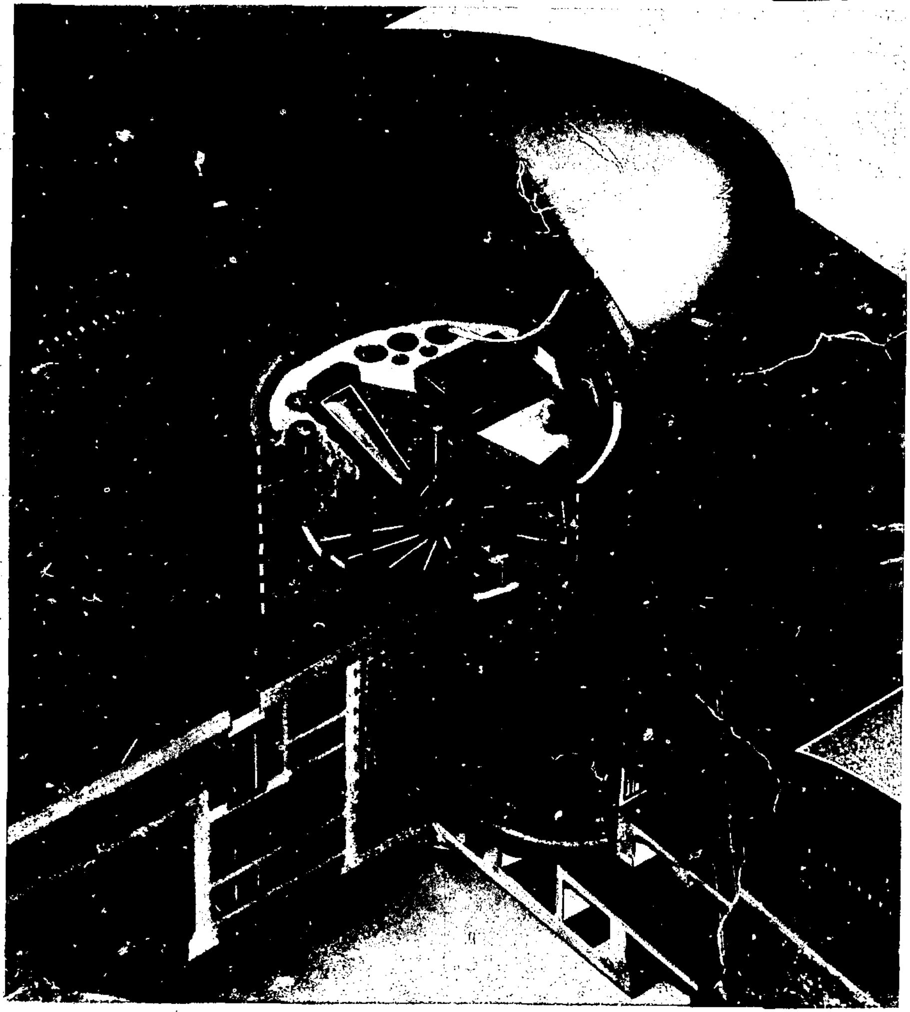


FIGURE 2

BLANKET/SHIELD CUTAWAY-HYBRID REACTOR

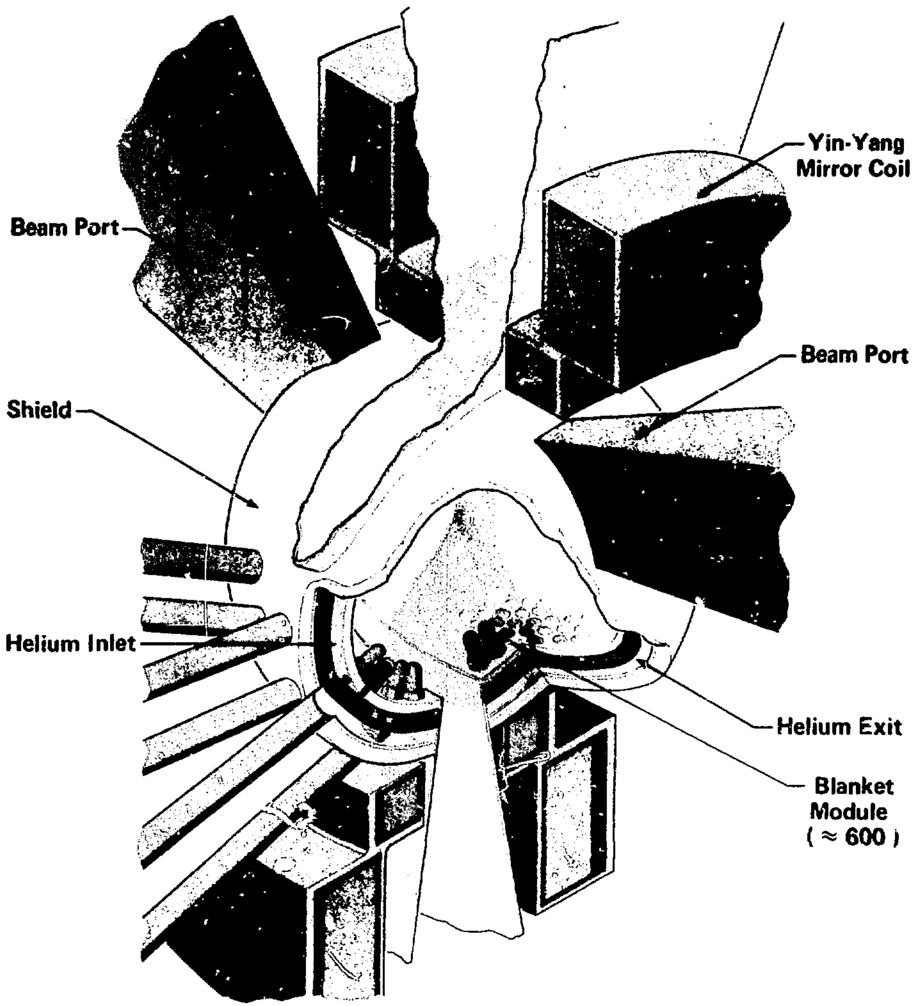
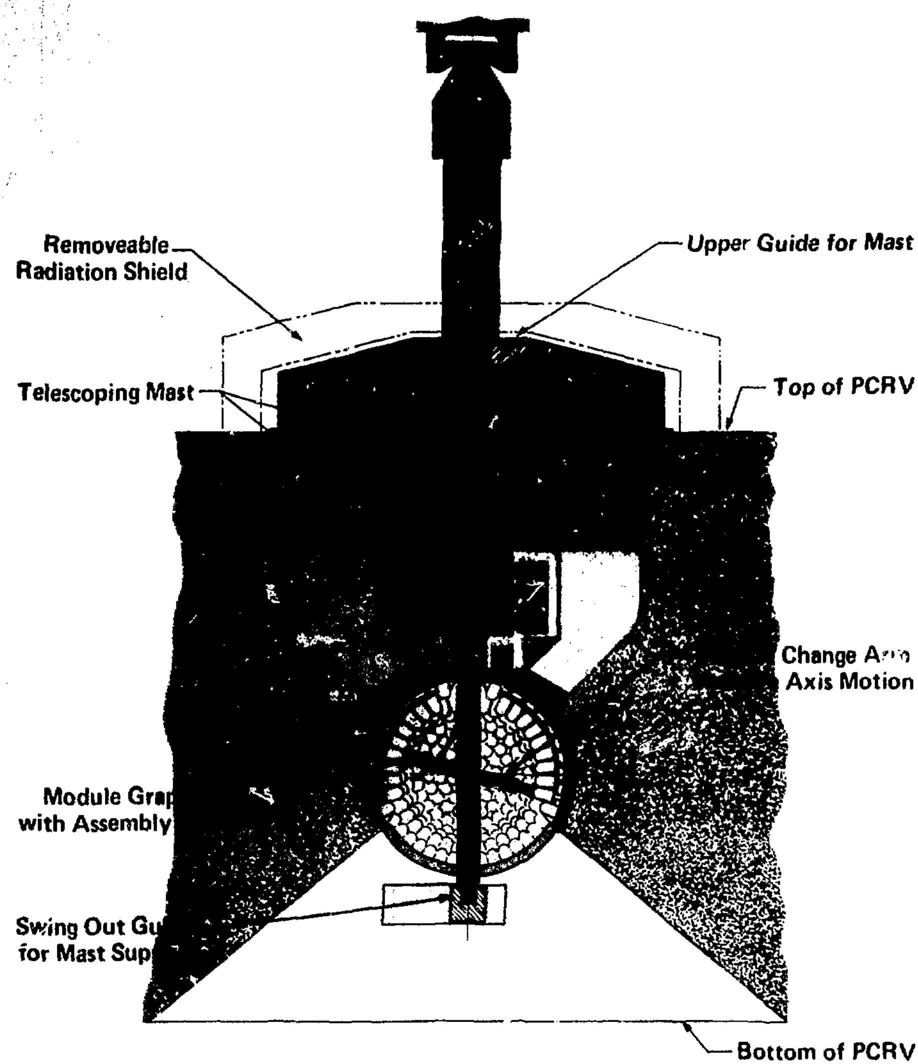


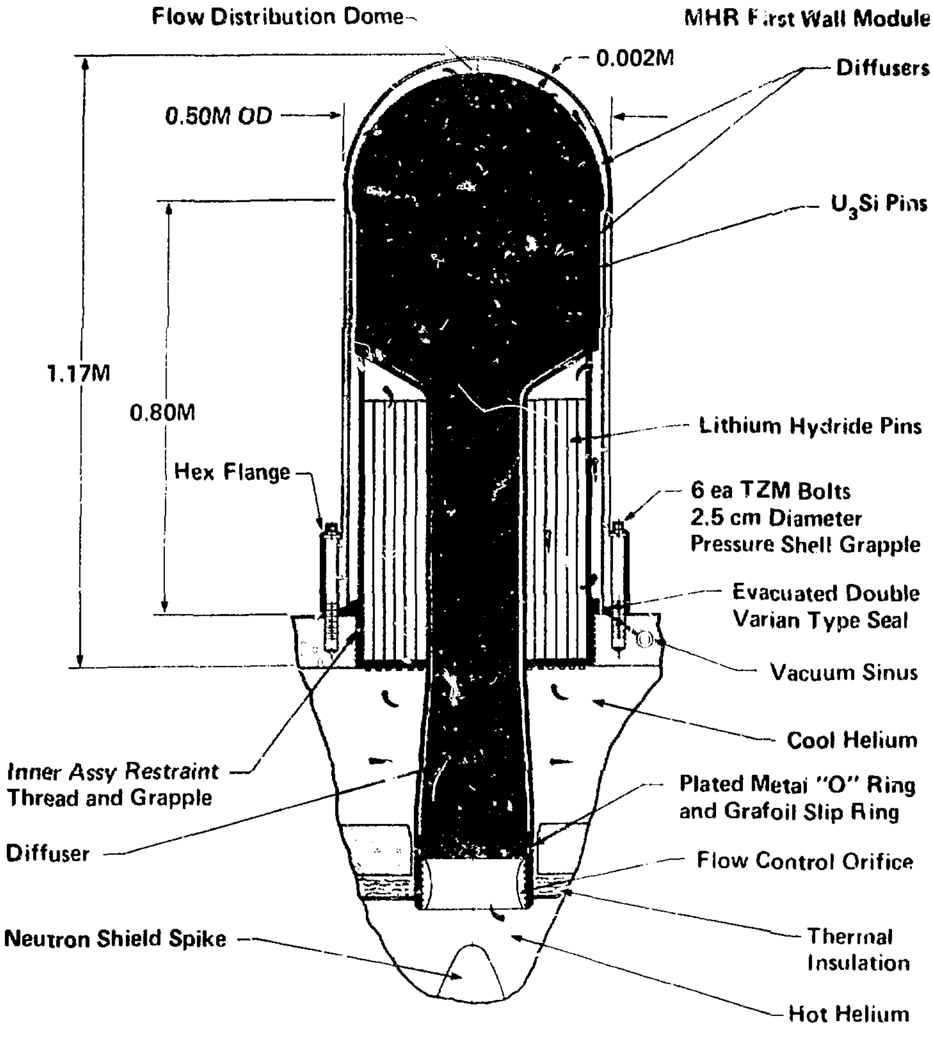
FIGURE 3



FUEL MODULE CHANGE TOOL

FIGURE 4

FUSION-FISSION MIRROR HYBRID BLANKET MODULE



SUMMARY

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

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

FUSION CORE DESIGN

Progress in mirror plasma physics experiments has been rapid within the last few years. The next generation mirror experiment, MFTF, will represent the physics prototype for a reactor-grade plasma. This proposition is illustrated in Table 2, where we list key physics parameters for the present mirror experiment, 2XIIB, for the next experiment, MFTF, and for the commercial hybrid reactor being discussed here. The 2XIIB experiment presently has 20 keV neutral beam injectors which are being upgraded to 40 keV; MFTF will have 20 keV injectors for start-up and 80 keV injectors for sustained (1 sec) operation. The major differences between the MFTF plasma and that required for a commercial reactor are physical size and duty factor; MFTF, however, is a sufficiently versatile experiment that it should establish scaling laws that will allow us to extrapolate to the larger plasma with a high degree of confidence. Also, $n\tau$ for the commercial facility is about a factor of 10 larger than in MFTF, this being a result of higher energy, larger size and the D-T isotopic mixture used in the commercial reactor as compared to MFTF. Thus, achievement of the anticipated plasma performance from MFTF will provide an adequate physics base from which development of a mirror hybrid could proceed.

The plasma modeling effort for the hybrid this year is using the full range of analytical tools that have been developed in support of mirror physics investigations in the 2XIIB experiment. These tools include the MCFUS equilibrium code [4], a two-dimensional (in velocity space) Fokker-Planck code [5], and a radial transport code [7, 8].

The variation of the basic Yin-Yang magnet, developed for reactor applications, is shown in Figure 1. This design uses a large main coil with a small mirror coil inside and has the virtue of locating the mirror point very near the inside surface of the conductor. This location for the mirror point implies a minimum size opening in the blanket for the plasma leakage fans. The magnet has kinetic energy is deposited as thermal energy in the direct converter electrodes and must be removed by active cooling. Upon striking the direct converter electrodes, the plasma flow is neutralized and the end tank must contain vacuum pumping equipment to remove the resulting gas load.

To provide access to the blanket from outside the machine, it is a convenient design feature to have one of the end tanks as small as possible. We implement the small end tank by designing the magnet such that one of the mirror fields is 5% stronger than the other. This field perturbation causes approximately 90% of the plasma leakage to flow out through the weak mirror and the remaining 10% to exit through the strong mirror. Since the size of the end tank is proportional to the amount of plasma flow, we can use a small end tank on the strong mirror. To keep this tank as simple as possible, we do not perform any direct conversion but design for the plasma energy to be deposited as thermal energy, with provisions for active cooling and vacuum pumping with cryopanels. The large end tank, which receives the 90% end leakage flow, is designed with a simple single-stage direct converter unit, having an upper limit on its efficiency of about 50%. This end tank must also have provisions for active cooling and vacuum pumping.

BLANKET NUCLEAR DESIGN

In the past, we have examined the use of primarily three fertile fuels in the blanket: UC, U-Mo alloy and thorium [3]. In our present hybrid design we are advocating the use of U_3Si, a fuel being developed in the Canadian nuclear power program for the CANDU reactor. Our reasons for this choice are (1) high uranium density (U_3Si is a metallic alloy), (2) ease of fabricability, and (3) a comparatively high burnup capability (for a metallic fuel), on the order of 2-3%. Economic optimization of the fuel cycle for this reactor dictates a total fuel exposure of about 5 MW-yr/m^2 of 14 MeV neutron energy through the first wall. In table 3, the initial (beginning of life) and final (end of life) neutronic parameters for the U_3Si blanket are listed.

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

The design approach we have selected is to mount the magnet, blanket and primary heat transfer loop all within a prestressed concrete reactor vessel (PCRV), of the type developed for gas-cooled fission reactors. This is shown in Figure 2. In the center of the PCRV is the magnet and blanket, and the steam generators and He circulators are located around the periphery. The blanket is a spherical shell inside the magnet. In this way, the blanket and its cooling system are locked together so that no relative motion between them can occur, thus precluding the possibility of rupturing any of the coolant ducts.

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

The blanket design concept is one which avoids any major disassembly of the reactor during the blanket change operation but instead relies on remote operations to assemble and disassemble the blanket inside the PCRV. The blanket is made up of small cylindrical modules, approximately 50 cm in diameter, with the blanket structure being suspended directly from the inside wall of the PCRV as shown in Figure 3. Removal and replacement of blanket modules is accomplished with the refueling machine shown in Figure 4, which consists of a post which is inserted down through the center of the machine and has a pivoting arm to operate

The local blanket multiplication and therefore local blanket power density increases by about a factor of two over the life of the fuel (see Table 3). By devising an appropriate fuel management scheme for the blanket, we are able to limit the peak-to-average variation in the total blanket thermal power to about 10% (3,600 MW average; 4,000 MW peak) and the primary heat transfer and power conversion loop capacity are designed to accommodate this power variation. The blanket modules are grouped into four quadrants and at time intervals of one quarter of the blanket life, the reactor is shut down and one quadrant of the blanket is refurbished with new fuel assemblies. In this way we are able to establish an equilibrium fuel cycle where the four quadrants are each at a different exposure.

The thermal-hydraulic design for the fuel, on the other hand, must provide adequate cooling of the fuel pins during the life-time power density variation of 18.4/8.8 = 2.1. Our present design specifies a peak fuel power density (i.e., at the first wall) at beginning-of-life of about 240 watts/cm^3 and an end-of-life value of 500 watts/cm^3. The fuel pins are 0.7 cm in diameter with 0.15 mm thick Inconel 718 clad on a pitch-to-diameter ratio of 1.05. The maximum mid-wall clad temperature (hot channel) is limited to 700°C.

CONCLUDING REMARKS

In this paper, we have summarized the reference mirror hybrid ractor design performed by LLL and General Atomic. The reactor parameters have been chosen to minimize the cost of producing fissile fuel for consumption in fission power reactors. As in the past, we have emphasized the use of existing technology where possible and a minimum extrapolation of technology otherwise. The resulting reactor may thus be viewed as a comparatively near-term goal of the fusion program, and we project improved performance for the hybrid in the future as more advanced technology becomes available.