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by

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A GAS-COOLED BLANKET FOR THE TANDEM MIRROR HYBRID REACTOR*

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A helium-cooled blanket design for the LLNL Tandem Mirror Hybrid Reactor has been developed that uses thorium metal as the fertile material, with an average thermal power of ~4000 MW and a production of 2.7 metric tons of ^{233}U per year. It is intended to interface with a thorium/^{233}U fuel cycle economy that is expected to be available in the early part of the 21st century. The design uses fairly conventional gas-cooled reactor technology to minimize the need for extensive development programs. The blanket materials include metallic thorium fuel in HT-9 ferritic steel-clad plates, an Inconel-718 first wall and a Li_2O tritium breeding zone. By using a separate tritium purge flow system, tritium concerns in the fission zone and primary loop are minimized. The cylindrical reactor is separated into 16 replaceable large modules connected by pressure-operated omega joint vacuum seals. The ease of lateral access to the simple cylindrical Tandem Mirror geometry is utilized to provide rapid changeout of entire reactor modules. The reference fast-fission thorium blanket has excellent neutronics performance, with a cycle and blanket average ^{233}U breeding ratio of 0.84 and an average blanket energy multiplication of 5.2. Analysis of the TMHR performance with the fast-fission thorium blanket showed that power and fuel production are quite sensitive to changes in fast-fission zone design, reactor length and fuel residence time. A significant local power swing occurs with blanket exposure as ^{233}U is bred. The fast-fission blanket design was found to be technically viable, but because of concerns about design sensitivity and afterheat cooling, a fission-suppressed blanket alternative was considered by placing a Li_7Pb_2 tritium breeding zone between the plasma and the thorium zone. Preliminary analysis of the gas-cooled fission-suppressed blanket concept was sufficiently encouraging to recommend that it be investigated further for use on the TMHR.

Introduction

During FY 79 a team consisting of Bechtel, General Atomic and General Electric, led by Lawrence Livermore National Laboratory conducted a preliminary conceptual design study to define the engineering characteristics of the Tandem Mirror Hybrid Reactor (TMHR).[1] The resulting reactor configuration is shown on Fig. 1. during this study a wide range of blanket technology design choices was explored, including use of liquid lithium, molten salt, water and gas cooling.

During this scoping study, General Atomic had responsibility for examining gas-cooled blanket concepts for use in the TMHR. The approach used in these scoping study evaluations was to take a broad look at the possible blanket options to make sure that the wide variety of gas-cooled blanket options available was considered and then recommend one concept for more detailed design and investigation during the second half of the year. The molten salt concept and the gas-cooled concept were both carried to the more detailed evaluation phase. General Atomic's work on the gas-cooled TMHR blanket is summarized in this paper. In the final analysis, neither blanket was chosen for the TMHR reference design. The advantages of the "fission-suppressed" hybrid blanket concept were identified and a recommendation made to further explore options for this concept during the continuation of the TMHR design study.[2]

Gas-Cooled Reactor Technology

Gas-cooled reactor technology has been pursued since the very early days of the nuclear program. It is one of only two technologies, the other being water cooling, that has successfully been developed to commercial status. Gas cooling was originally pursued most vigorously in Europe with the development of the French and British CO_2-cooled, graphite moderated, natural uranium-fueled MAGNOX reactors. These led to the CO_2-cooled Advanced Gas Reactor (AGR) program in Great Britain and the helium-cooled High Temperature Gas-Cooled Reactor (HTGR) programs in France, Germany, Japan, and the USA. General Atomic designed and built the 40 MWe prototype Peach Bottom

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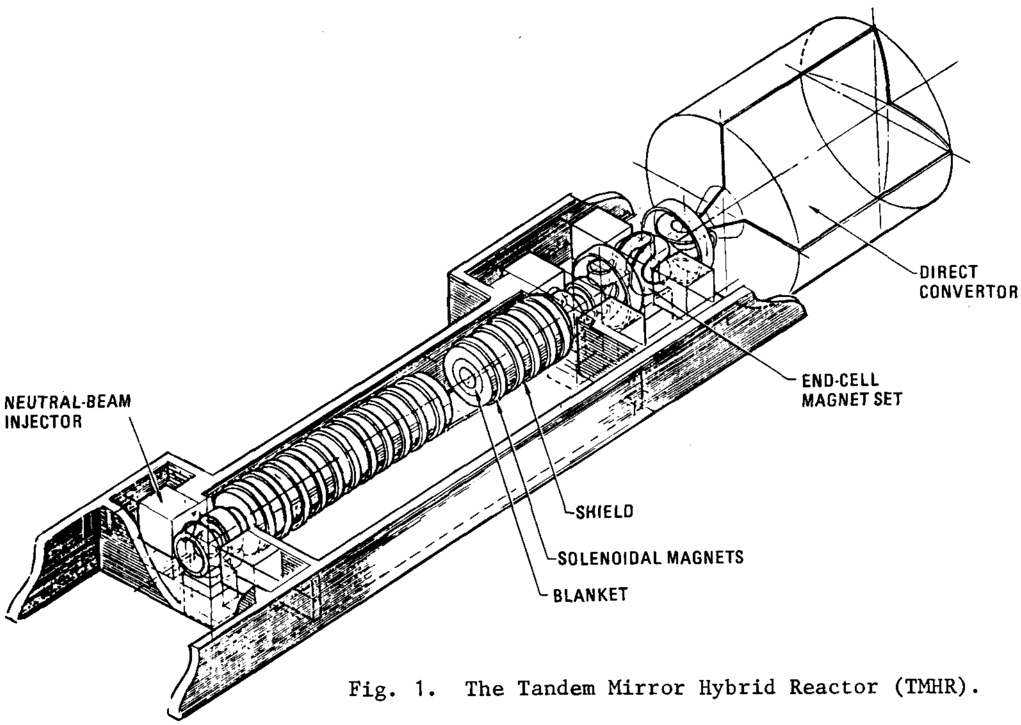

Fig. 1. The Tandem Mirror Hybrid Reactor (TMHR).

HTGR which operated successfully for over five years on the Philadelphia Electric Company grid and also the 330 MWe demonstration Fort St. Vrain HTGR which is now in operation on the Public Service of Colorado grid.

A wide variety of gases have been proposed for use in gas-cooled reactors including CO_2, helium, dry steam, nitrogen and nitrogen oxide. The technical aspects of gas-cooling are reviewed in Ref. 3 and the application of gas-cooled technology to fusion power systems is discussed in Ref. 4. Because of its high heat capacity, high sonic speed and chemical inertness, helium has generally been the first choice of coolant for gas-cooled systems and will be used as a basis for the design of a gas-cooled blanket and power conversion system for the Tandem Mirror Hybrid Reactor.

Advantages of Helium. The principle advantages of helium cooling are summarized in Table 1. Helium is the most inert of all proposed coolants. It has very small nuclear interaction cross sections and is virtually transparent to neutrons. This allows excellent neutron economy and fuel breeding performance to be achieved. Helium is chemically inert; reactions with the fuel, cladding and environment are not of concern. Because helium is a gas, no phase changes are possible, thus the heat transfer regime is stable in both normal, pressurized operation and in the event of depressurization. Because of this, a "loss of coolant" accident cannot occur although loss of coolant pressure and flow accidents must be protected against.

For magnetically-confined fusion systems the fact that helium is non-magnetic and non-conductive is of advantage. Because of the low

TABLE 1
ADVANTAGES OF HELIUM COOLING

• HELIUM IS INERT:
– CHEMICALLY INERT
– NEUTRONICALLY BENIGN
– NO PHASE CHANGES
– NEGLIGIBLE GRAVITY EFFECTS
– NON-MAGNETIC
– NON-CONDUCTIVE
• DEVELOPED TECHNOLOGY:
– HEAT TRANSFER
– POWER CONVERSION
– PURIFICATION (INCLUDING TRITIUM RECOVERY)
• MAINTENANCE ADVANTAGES
– NO ACTIVATION
– NO ISOLATION
– TRANSPARENT

density, gravitational effects are quite small. As a consequence, the heat transfer in a helium system is not greatly affected by gravitational orientation. This can allow axi-symmetric blanket designs and full coverage of the plasma chamber surface.

Because of the large world gas-cooled reactor program and commercial deployment, gas cooling enjoys a developed technology. The heat transfer and thermal-hydraulic correlations are understood and power conversion equipment (steam generators, circulators, etc.) are developed. Helium purification systems have been successfully developed and include the capability for tritium recovery from the helium stream.

Helium cooling has several advantages from a reactor maintenance point of view. It is transparent and does not activate. Air can be allowed into the helium system ducting during maintenance.

Disadvantages of Helium. The disadvantages of gas cooling are summarized in Table 2. The principle disadvantage of all gas coolants is their low volumetric heat capacity. To achieve adequate heat removal capacity with acceptable coolant pumping power requirements, pressures in the range of 40 to 80 atmospheres appear to be needed. The heat transfer coefficient that may be obtained at reasonable flow velocities in a helium-cooled system is generally smaller than that found in liquid cooled systems and hence the temperature differential is larger, leading to higher fuel and clad temperatures. The power required to operate the helium circulator can be high. With the pressure drops encountered with a steam cycle power conversion system, the pumping power is on the order of 2 to 5% of the reactor thermal power.

TABLE 2
DISADVANTAGES OF HELIUM COOLING

- LOW VOLUMETRIC HEAT CAPACITY
  - 5200 J/kg · K $\cong$ 14 kJ/m^3 · K AT 5 MPa AND 600^{0}C
  - HIGH PRESSURE REQUIRED ($\sim$ 4 TO 8 MPa)
  - MODEST HEAT TRANSFER COEFFICIENTS, LARGER TEMPERATURE DIFFERENTIALS
  - HIGH PUMPING POWER REQUIREMENTS ($\sim$ 2 TO 5% OF THERMAL POWER)
  - LOW NATURAL CONVECTION COOLING
- MODEST REACTOR EXPERIENCE COMPARED TO WATER
- POSSIBLE RESOURCE AVAILABILITY LIMITATIONS

Because of the low volumetric heat capacity of helium, natural convection cooling is difficult to achieve. Although natural convection can provide some shutdown afterheat cooling when the coolant loop is pressurized, it cannot be counted on in the depressurized state.

Despite the fact that gas-cooled reactor technology has been deployed commercially, experience is less extensive than that enjoyed by water-cooled technology, particularly in the USA. A potential concern for helium-cooled systems in the mid 21st century is the possibility of resource limitation. Helium is presently extracted from natural gas which will be in increasingly limited supply in the future. Helium can be extracted from the atmosphere but at higher cost.

Gas-Cooled Blanket Options

Within the general category of helium-cooled hybrid blankets there are a number of further choices that must be made. The fuel cycle may be the uranium/plutonium cycle or the thorium/uranium cycle or both may be used. The fuel structural material may be metal cladding (GCFR technology) or coated particles in graphite (HTGR technology). A set of nine helium-cooled blanket concepts was developed for consideration for the TMHR. On the basis of very simple performance estimates and system energy balance and economic evaluations this list of nine gas-cooled blanket options was rapidly narrowed to four candidates. These are:

1. A metal-clad fast-fission uranium blanket,
2. a metal-clad blanket with a thin uranium fast-fission zone followed by a thorium breeding zone,
3. a metal-clad fast-fission thorium design, and
4. a thermal spectrum graphite-base blanket using ThO_2 fuel (HTGR technology) and lead or beryllium for neutron multiplication by (n,2n) reactions.

Preliminary scoping studies were done on these four blanket options including consideration of the blanket mechanical design, thermal hydraulic aspects, fuel reprocessing concerns and neutronic performance. The neutronic evaluation included consideration of the blanket fuel production, energy multiplication, materials damage effects, and fission product afterheat for safety and residual heat removal system evaluations. Using the results of these scoping calculations the energy balance characteristics of tandem mirror hybrid reactors using these blanket concepts were compared.

Finally, simple system economics models were used to compare the four options. The results of these preliminary evaluations of the four blanket options are summarized on Table 3.

Uranium, Metal Blanket. The uranium blanket is a high performance plutonium producing design that also produces significant amounts of blanket thermal power.[5] Because of the large thermal output, the fuel production per unit thermal power is less (1.0 kg/yr per MWt) than that of the thorium blanket (1.5 kg/yr per MWt). Because of the large energy production, this blanket can be used with a low plasma amplification Q_p and low energy handling efficiences and still attain net power production. The high blanket power results in a high level of blanket afterheat. Assuming no afterheat removal (adiabatic heating) after shutdown, the heat-up rate will result in fuel melting in approximately one minute. This rapid thermal response will require use of a highly reliable, fast-acting residual heat removal system to prevent fuel damage.

Uranium/Thorium, Metal Blanket. The metal-clad uranium fission plate/thorium breeding zone design[6] has many characteristics similar to those of the uranium blanket but enjoys the benefit of producing ^{233}U, which is a more valuable fuel for thermal burner reactors than in plutonium. It produces 0.8 kg of plutonium and 0.8 kg of ^{233}U per MWt·year. The uranium fast fission zone has the same rapid thermal response time and sensitivity to loss of cooling as does the all-uranium blanket. The use of two blanket fertile materials will complicate reprocessing somewhat, with an approximate 20% reprocessing cost penalty.

Thorium, Metal Blanket. The metal-clad thorium blanket has sufficient energy production to be able to break even electrically with even low Q_p and low energy handling efficiencies.[7] It produces only ^{233}U (at 1.5 kg/MWt-yr) and thus reprocessing is simpler than for the U/Th design. Because of the lower blanket power density and higher fuel melting temperature, the thorium blanket is much more tolerant of loss of cooling than are those containing uranium. With totally adiabatic heat-up after plasma shutdown, fuel damage would not occur for over an hour, thus relaxing the requirements for residual heat removal safety systems.

Beryllium/ThO_2, Graphite Blanket. The graphite-base Be/ThO_2 blanket is a relatively low performance design. Due to the small energy production, it requires fairly high energy handling efficiencies and a Q_p approaching 2 to be able to break even electrically. Because of the low energy production, the amount of fuel produced per unit thermal power is quite high (1.7 kg/MWt-yr). The low power density, high specific heat and high allowable temperatures of this blanket make it quite tolerant of loss of cooling. Adiabatic heating for about 24 hours could be allowed before fuel damage would occur.

Recommendations. On the basis of the considerations discussed above, the metal-clad thorium blanket was chosen as the gas-cooled blanket candidate for the TMHR. This choice offers the simplest design, good economics and assured positive net electric power while being reasonably tolerant of loss of cooling. This selection should not be construed to indicate that the other three blanket options are not

TABLE 3
GAS-COOLED TMHR
BLANKET CHARACTERISTICS

BLANKET		URANIUM METAL	URANIUM/THORIUM METAL	THORIUM METAL	BERYLLIUM/ThO_2 GRAPHITE
BREEDING RATIO*	Pu	1.8	0.81	--	--
(ATOMS/FUSION NEUTRON)	^{233}U	--		0.74	0.49
(WITH T/n = 1.1)					
ENERGY MULTIPLICATION*		10	5.6	2.6	1.6
FUEL PRODUCTION*	Pu	0.97	0.78	--	--
(kg/MW_t -yr)	^{233}U	--	0.75	1.54	1.65
ADIABATIC TIME TO MELTING FROM AFTERHEAT		1 m	1 m	1.5 h	26 h
APPROXIMATE FUEL LIFETIME ($MW \cdot yr/m^2$)		6	9	9	2
FUEL COST** – ($/gm) (AT 15 MILLS/kWh) ELECTRICITY)		93	77	84	98

*BEGINNING OF BLANKET LIFE VALUES.

** ADJUSTED FOR RELATIVE VALUE OF PLUTONIUM VERSUS ^{233}U.

desirable. In fact, all four options appear to be attractive hybrid blanket alternatives and all four show similar economic estimates of fuel cost and system electricity cost. The metal-clad thorium design selected, however, appears to offer a combination of attributes that is particularly attractive for the TMHR.

Gas-Cooled Blanket Design

The TMHR gas-cooled blanket concept developed by General Atomic is based upon a "large module" configuration. This approach segments the reactor into large axial slices. These slices are moved laterally to a remote service area for off-line maintenance. The reactor blanket consists of 16 of these large modules, two of which are replaced every calendar year.

Design Philosophy and Concept

There are four basic requirements that must be considered in the design of a hybrid reactor blanket. First, it must show good nuclear performance. The reactor average tritium breeding ratio must be greater than 1.1 T/n. The net fuel breeding ratio must be as large as possible. The requirement for good nuclear performance implies the need to minimize the amount of extraneous structural materials in the blanket zone.

Second, the blanket design must exhibit good thermal-hydraulic performance. The pressure drop must be kept to acceptable levels while still allowing adequate flow velocity for good heat transfer. The pumping power requirement should not exceed a few percent of the total thermal power.

Third, the blanket must be mechanically feasible and easily fabricated. The structural design must accommodate a wide variety of environmental constraints including coolant pressure loads, thermal stress, radiation-induced swelling and creep and seismic forces. Of prime importance, the mechanical design must facilitate maintenance while allowing for minimum reactor down time.

Finally, the blanket must exhibit acceptable safety and economic characteristics.

The performance requirements led to the goal to minimize the structural to solid fraction in the blanket design. Because of this, a modularized first wall was developed, composed of 40 submodules each with a semi-circular first-wall 0.5 cm thick which forms the plasma chamber and high pressure helium pressure boundary. The plate geometry blanket fuel design similarly evolved to avoid the geometric restrictions of cladding fraction and packing fraction limits of other geometries, e.g., rods or balls. The plate cladding thickness is restricted only by manufacturing and mechanical design limits.

The maintenance design philosophy is "avoid remote maintenance and repair work on the reactor." This is attempted by making components modular and as large as practical, and removing the entire component to an off-line repair area for maintenance. Welded joints that require cutting and re-welding appear to be particularly time consuming and are to be avoided. Each component must specifically be designed to operate remotely. Given the severity of the fusion blanket design environment and the uncertainties about materials and technology performance in this environment, it is proposed that the entire reactor - first wall, blanket, shield, magnets, coolant piping and support components - be easily and routinely removable.

The tandem mirror appears to offer unique advantages in the maintenance area. There are no interlinked coils and the reactor can be broken into axial modules that are virtually not connected to each other. The only connection between modules is for vacuum sealing.

Mechanical Design

The tandem mirror reactor concept enjoys a simple cylindrical geometry with relatively unobstructed lateral access. The design of the gas-cooled TMHR module concept has emphasized utilization of this geometry to allow rapid maintenance and changeout of the blanket/first wall zone.

Reactor Modularity. A reactor of this size must be broken down into modules, in order to be maintainable with acceptable down-time. The cylindrical reactor was divided by transverse slices into 16 large cylindrical modules. The reactor lay-out is based upon use of a "hot vault" with lateral access to a maintenance area which could be entered when the vault is closed.

Figure 2 shows the module concept and the proposed disassembly pathways. A dedicated wheeled-truck is shown although the final choice, made by Bechtel, was to use an overhead crane to move the modules. Neutronics considerations indicate that only 1% of the heat deposition takes place in the shield which is water-cooled. This has led to the decision to make the blanket and shield separable. The replaceable blanket module is removed from the shield structure in the hot shop and replaced with a fresh module. The entire assembly is returned to the reactor at the next refueling.

Vacuum Seal Configuration. In order to close the vacuum in the plasma chamber it is necessary to seal between the modules. This is

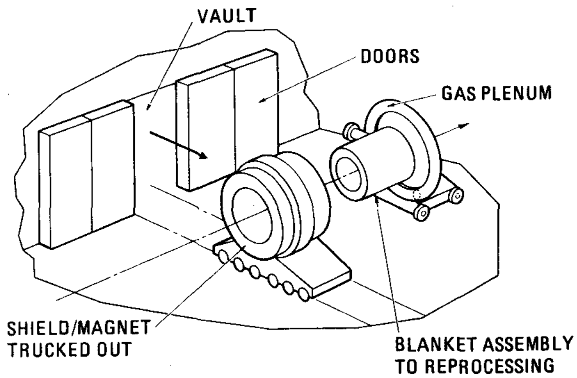

Fig. 2. Blanket module and changeout paths.

achieved with a double knife-edge seal, driven by an inflatable all-metal omega-joint cushion between the shields. Deflation of this cushion provides maneuvering from between the shields. Figure 3 shows the details of this seal arrangement.

A detailed but preliminary estimate was made of the module changeout time requirements. It is expected that the annual change of two of the 16 modules for reprocessing will not in total take more than one month, including time for shutdown and restart of the machine. The afterheat problem has also been given preliminary attention and it is clear that reliable auxiliary cooling system and transport cooling systems will be required to protect the module during the changeout procedure.

Blanket Cartridge Design. The blanket cartridge is shown in Fig. 4. The outer shell fits into the shield and is 4.5 cm thick to contain the coolant pressure at 5 m diameter. From one end a radial double passageway connects to an outer torus which serves, by being internally divided, to both deliver and collect the coolant gas.

The body of the blanket cartridge is made nominally about 1% short to allow thermal and neutronic growth. The module ends, having about 0.5 m radial span will be built up structures to allow cooling.

The first wall could not at this diameter be made as a simple cylinder since its thickness would cause a considerable neutronic penalty and a lobed design was adopted in which the lobes are suspended from the outer shell. The suspension webs serve as part of the gas distribution system, but they must be axially broken at intervals to prevent the development of neutron swelling stresses. Materials nearest the plasma are Inconel 718, the first wall being 5 mm thick.

Submodule Configuration. Each of the cusp-shaped azimuthal sectors of the module may be considered a submodule, although they would not be separable from one another. The flow configuration and fuel configuration within these submodules each have several possible options.

Use of radial coolant flow allows the temperature characteristics of the gas coolant to

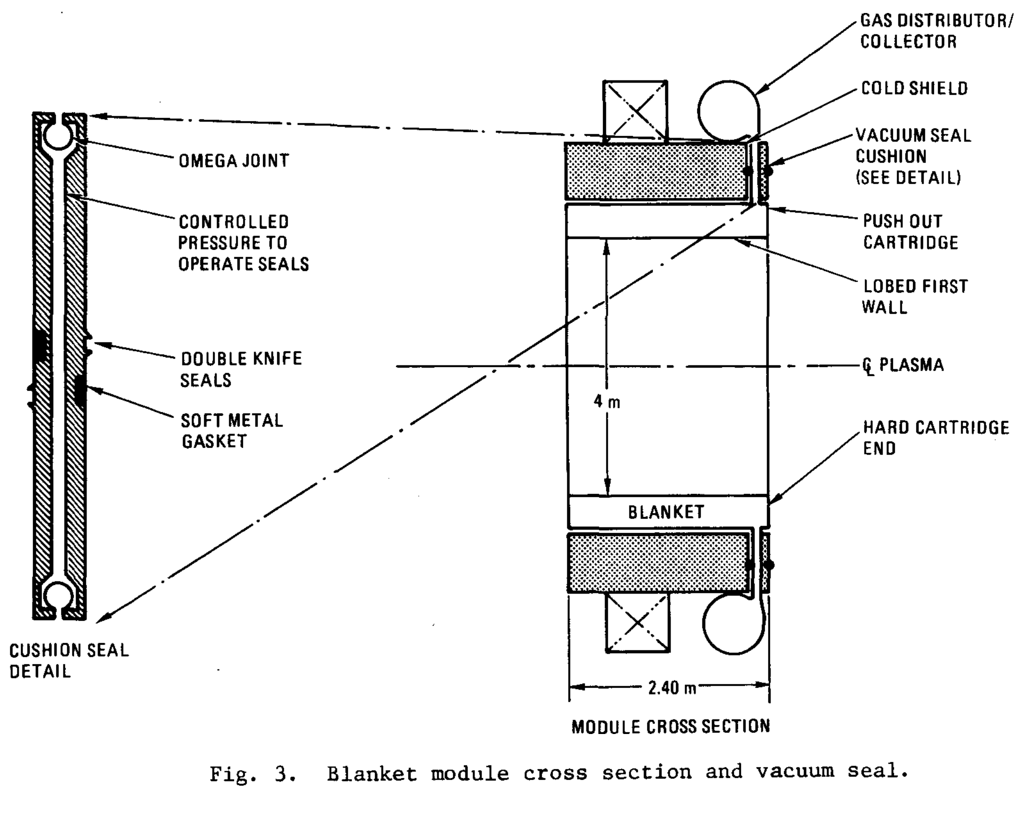

Fig. 3. Blanket module cross section and vacuum seal.

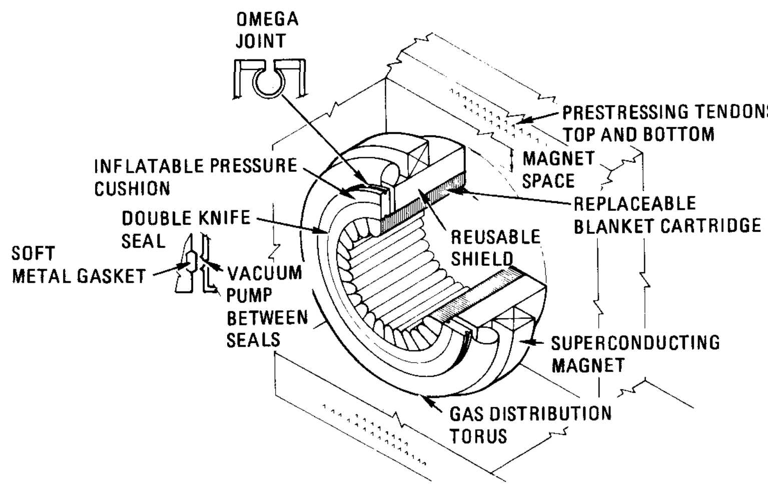

Fig. 4. Helium-cooled blanket modules.

be matched to the very steep radial power density gradient found in a hybrid blanket. By introducing the inlet coolant at the plasma side of the blanket, the coolest coolant is applied at the point of highest power density. The coolant is distributed axially along the module through an inlet plenum at the back of the module. It flows radially inward along the cusp radial tie plates to the first wall region, turns, cooling the first wall, and flows radially outward through the fuel zone. In this manner the entire module structure is maintained near the coolant inlet temperature. This submodule flow path is shown on Fig. 5.

Three fuel configurations were considered for the submodule, radially oriented rods, axially oriented rods, and axially oriented plates. It is felt that the blanket cartridge concept and submodule design are not too sensitive to this choice and could accommodate

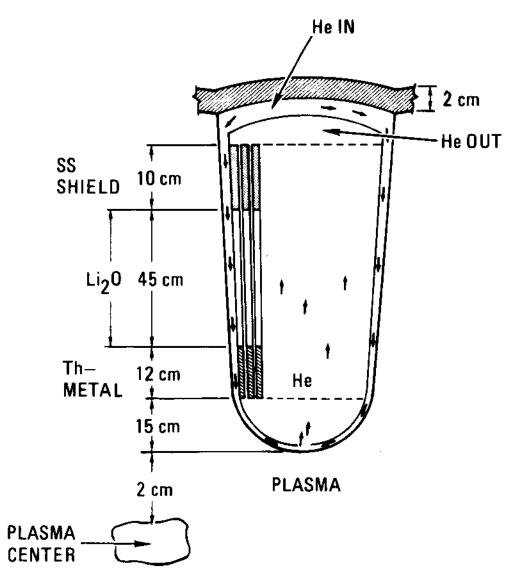

Fig. 5. Helium-cooled TMHR submodule.

any one of the three. The thermal-hydraulic designs of radial rods, axial rods and plates were studied for use in the TMHR gas-cooled blanket and are discussed below.

Materials Considerations

In designing a helium-cooled fusion reactor blanket, it is important to select suitable materials for different regions of the blanket in order to optimize the overall design. The material selections for use in the gas-cooled TMHR blanket are fairly conventional and are based on previous fission reactor experience or fusion reactor design studies.

First Wall Material. Inconel 718 is recommended for the first wall material on the basis of its high allowable stresses at the 550°C operating temperature and its excellent potential swelling resistance in the TMHR neutron environment. The ultimate tensile strength of solution treated and aged Inconel 718 is compared to those for 20% cold-worked 316 stainless steel, HT9 ferritic steel, titanium alloy and niobium on Fig. 6. The superior strength of Inconel 718 at elevated temperature is clearly shown.

Approximately 140 appm helium will be produced in Inconel 718 TMHR blanket modules after one year service. There is growing concern about radiation embrittlement of these precipitation - strengthened nickel alloys at temperature above 500°C.[8] The reduction of uniform elongation to 0.5% would control the design lifetimes of the modules. The presence of Nb carbides at the grain boundaries of Inconel 718 may be effective in inhibiting grain boundary sliding and might reduce the rate of helium embrittlement. Compatibility tests of Inconel 718 with impure helium at 650°C indicate little or no corrosion or creep property degradation from impurity-metal interactions. Metallurgical stability of the alloy at 650°C also

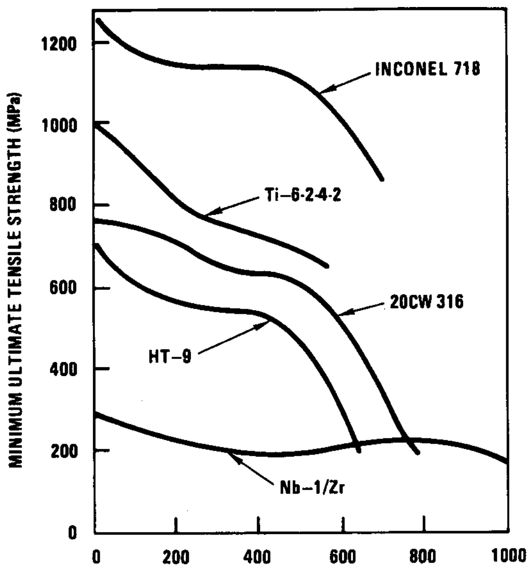

Fig. 6. Structural material strength.

appears acceptable. Although fabrication (machining and welding) of Inconel 718 is inherently more difficult than austenitic stainless steels, the construction of blanket modules from the former alloy appears feasible. The module bodies could be either forged, roll formed or perhaps vacuum shell cast. Weldability of Inconel 718 in section thickness of up to 1 cm using TIG methods is good, and no post weld heat treatment would be required if the welds do not exceed 540°C in service.

Cladding Material for Fuel. Inconel 718 can be used as the cladding material for the fuel plates, yet the ferritic steel, HT-9, may be a better choice because of lower cost, easier fabrication and reduced radiation-induced swelling. The minimum ultimate tensile strength of HT-9 is shown in Fig. 6. Above 550°C HT-9 short term strength decreases rapidly. Since the cladding of the fuel plate would not require very high strength, a possible operating temperature limit extended to 650°C might be feasible.

The primary reason for choosing HT-9 is its apparent resistance to neutron radiation damage. EBR II irradiation data to fluence greater than 1 x 10^{23} n/cm^2 (e > 0.1 MeV) indicate that the maximum swelling ($\Delta V/V$) of HT-9 is less than 1% which is less than half of the swelling of Inconel. This would ease some of the mechanical design requirements for the fuel plates. A concern that is presently unresolved is that of increase in the ductile-to-brittle transition temperature of HT-9 with irradiation. This effect could result in very low fracture toughness values for irradiated HT-9. Research programs presently underway should resolve this concern.

Thorium Fuel Form. Thorium metal fuel has a higher breeding potential than do oxide, carbide, or nitride fuels. Thorium metal has a reasonably high melting point (1755°C), high thermal conductivity and good irradiation stability to the ∿1 to 1.5% burnup levels required in the TMHR. With nominal centerline temperature <850°C, swelling would be <2% which should be acceptable with proper fuel design. Reactor grade thorium metal has been produced commercially. Thorium metal is readily worked by both hot and cold forming.

Tritium Breeding Material. The three most widely considered tritium breeding materials are Li, Li_7Pb_2 and Li_2O. These three were again identified after a review of 20 lithium containing materials. The use of elemental lithium raises several design questions. Methods for tritium recovery from lithium are being developed but will probably require that the lithium be circulated out of the blanket for processing. A major concern with the use of liquid lithium is safety. Lithium reacts with carbon or water violently and reacts slowly with dry air and oxidizes rapidly to form hydroxide in moist air. In the event of a lithium spill, the energy released by reaction of lithium with air, water or concrete could contribute to failure of the containment and release of the tritium inventory. Because of this, material selection for the TMHR has focused on solid lithium breeding compounds.

Li_7Pb_2 has the advantage of build-in neutron multiplication through (n,2n) reactions in the lead. Due to the low melting point of Li-Pb alloys, sintering of Li_7Pb_2 granules is of concern and is expected to occur at temperatures as low as 500 K. This would increase the effective grain size and decrease the tritium release rate. One experiment,[9] however, showed that prolonged annealing at 0.83 T_m (830 K) actually improved the tritium extraction even though this temperature exceeds the expected sintering temperature. Assuming this tritium release behavior is representative, the Li_7Pb_2 blanket material would exhibit a tritium inventory of only about 30 g. The large heat of reaction between Li_7Pb_2 and water indicate that Li_7Pb_2 will share the safety concerns of elemental lithium. This has been dramatically confirmed by recent experiments at ANL.[10]

Li_2O has the highest lithium density of the candidate materials. It has high melting point (∿1700°C) thus allowing high coolant temperature. It releases the bred tritium in the form of T_2O, which allows relative ease in tritium control. A drawback of Li_2O is that it is extremely hygroscopic and forms LiOH or LiOT which are very corrosive to cladding material.

This problem appears to be manageable by the use of a separate purge stream of very dry helium gas, sweeping at low velocity through the packed Li_2O powder in the fuel elements. Concerns have been raised recently about the release of tritium from lithium ceramic materials.[11] Extrapolation of the limited experimental data[10] indicate good release and low tritium inventory. Calculations, however, predict significant inventories.[11] If tritium release proves to be a problem the addition of about 1 ppm H_2 or D_2 to the purge gas would enhance tritium extraction significantly. Because of the softened neutron spectrum in the lithium zone behind the thorium zone, the substitution of $LiAlO_2$ for Li_2O would not significantly degrade the blanket performance.

On the basis of the above discussions Li_2O was selected as the reference tritium breeding material for the TMHR.

Thermal Hydraulic Design

The thermal hydraulic design of a gas-cooled reactor system should have high thermal efficiency and low pressure losses. The high efficiency requirement dictates a high coolant outlet temperature. The low pressure loss requirement leads to high system operating pressure and a large coolant inlet-to-outlet temperature differential. Experience with helium-cooled nuclear power systems indicates that a helium pressure of 40 to 80 atmospheres will be needed for a good thermal/hydraulic design. Steam-generator design conditions dictate a minimum coolant inlet temperature of about 280°C and a minimum coolant temperature rise of about 100°C. Based on this information, a helium operating pressure of 5.6 MPa (55 atmospheres) and an inlet temperature of 285°C were selected. For the thorium metal-Li_2O blanket, an outlet temperature of 515°C was selected, giving a thermal cycle efficiency of about 38%.

For the mirror confinement concept, where all the charged particles are guided by the axial magnetic field to the direct convertors, there is assumed to be no charged particle or radiation heating deposited onto the first wall. The neutrons deposit only a fraction of their energy (about 5%) as they pass through the first-wall. The rest of the fusion energy is captured in the blanket.

First-Wall Cooling. Thermal-hydraulics calculations were done on the first wall using a maximum temperature criterion of 550°C. To keep the blanket pressure drops within the limit of 124 KPa (18 psi), a limit of 13.8 kPa (2 psi) was set for the pressure drop through the first wall. With reference to Fig. 5, the helium, which enters at 285°C, flows axially along the cold plenum, reaching the module side plenum at 288°C. The coolant temperature increases by about 6 degrees to 294°C as it flows through the module side plenum. It then flows along the first wall in a 0.5 cm thick cooling gap and is heated to about 300°C before it enters the first wall plenum and is distributed to the breeder plates. This simple first wall arrangement gives a 330°C maximum wall temperature. The coolant temperature exiting the module is set at 515°C to give a coolant inlet-outlet temperature differential of 230°C.

Fuel Zone Cooling. A rather detailed comparison was made to determine the best fuel configuration for the TMHR. The detailed thermal hydraulic designs of radially-oriented rods, axially-oriented rods and radially-oriented plates were developed and compared.[12] The characteristics of the three designs are summarized on Table 4. The larger structural fractions required will degrade the uranium breeding ratio and energy multiplicatrion factor of the axial rod design by about 7% and the radial rod design by about 4% compared to the performance expected from the plate design blanket. Because of the superior nuclear performance, the acceptable thermal hydraluic characteristics and the mechanical design feasibility, the plate geometry concept has been chosen for the reference gas-cooled TMHR blanket design. The design parameters of the blanket are shown on Table 5 and the blanket temperature profiles at beginning- and end-of-life are shown on Fig. 7.

One major difficulty in the thermal hydraulics of the fuel design was caused by the varying of the local energy multiplication of the blanket during its lifetime (6.4 full power years) from 2.8 to 8.5. In order to maintain a constant coolant outlet temperature, the coolant mass flow rate has to be changed and the coolant would be operating in the transition flow regime at some time during blanket life. The heat transfer coefficient for transitional flow was obtained by first comparing the friction factors for laminar and turbulent flow at the given Reynolds number, then the higher friction factor, f, was selected for the pressure drop calculation and the corresponding heat transfer coefficient was used for the film drop thermal calculation.[12]

Blanket Coolant Pressure Drops. Simple system pressure drop calculations were performed. The total blanket pressure drop from helium inlet to outlet including turning, expansion, and contraction losses is estimated to be 65 kPa (9.4 psi). With the inclusion of estimated pressure losses from the steam-generator and external ducting of $\sim$110 kPa (16 psi), the total system pressure loss is 170 kPa (24.5 psi). Thus the system $\Delta P/P$ is 3.0%, which is quite acceptable.

TABLE 4
FUEL CONFIGURATION THERMAL-HYDRAULIC CHARACTERISTICS

FUEL CONFIGURATION	PLATE	AXIAL ROD	RADIAL ROD
CHARACTERISTICS DIMENSIONS	THICKNESS: 1.15-1.58 cm COOLANT GAP: 1 mm	O. DIAM.: 1.19 cm P/D: 1.18	O. DIAM.: 1.67 cm P/D: 1.05
$\Delta T = T_{OUT} \cdot T_{IN}$	230°C	230°C	230°C
ΔP (kPa)	20.7	20.7	7.2
$\bar{h}$ (W/m^2°K)	2434	2174 (MINIMUM 1087)	1122
$T_{MAX, CLAD}$	666°C	663°C	662
$T_{MAX, FUEL}$	939°C	775°C	765°C
VOID FRACTION	7.3%	35%	17.8%
CLAD/SOLID	3.6%	8.2%	5.9%
FUEL PERFORMANCE (RELATIVE)	1	0.93	0.96

TABLE 5
BREEDER PLATE DESIGN PARAMETERS

FERTILE MATERIAL	THORIUM
TRITIUM BREEDING MATERIAL	Li_2O
THERMAL POWER OUTPUT	4000 MW_t
MAXIMUM VOLUMETRIC NUCLEAR HEATING	120 MW/m^3
MAXIMUM WALL LOADING	1.5 MW/m^2
NUMBER OF PLATES/SUB-MODULE	24
DIMENSIONS	
WIDTH OF PLATE (NARROW SIDE)	1.15 cm
WIDTH OF PLATE (WIDE SIDE)	1.58 cm
LENGTH OF PLATE	68 cm
HEIGTH OF PLATE	50 cm
CLADDING THICKNESS	0.25 mm
WIDTH OF COOLANT GAP	1 mm
HELIUM PARAMETERS	
PRESSURE	5.6 MPa (55 ATM)
INLET TEMPERATURE TO FUEL ZONE	300°C
OUTLET TEMPERATURE	515°C
REYNOLDS NUMBER*	2161 → 6562
PRESSURE DROP*	3.4 → 20.7 kPa (0.5 → 3.0 psi)
MASS FLOW RATE/MODULE* (3.5 m LONG)	140 → 385 kg/SEC
THERMAL CYCLE EFFICIENCY	~ 38%

* BOL → EOL

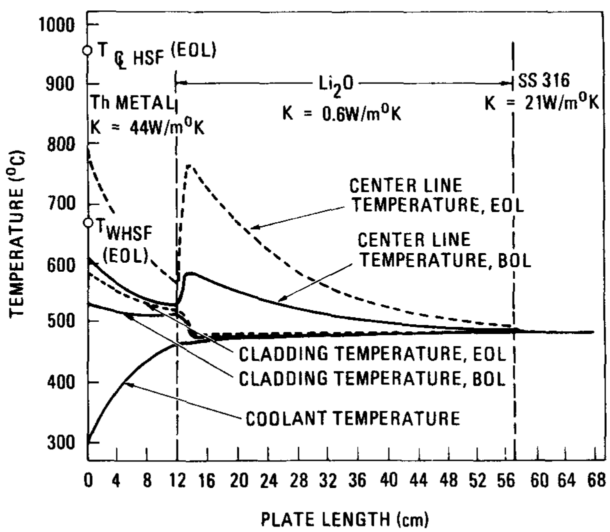

Fig. 7. Nominal temperatures in the blanket.

Nucleonics

The nucleonic design of the blanket and shield for the Tandem Mirror Hybrid Reactor was performed to achieve the goals of breeding adequate tritium, producing a maximum amount of uranium, and producing enough thermal heat to allow electrical breakeven. Alternate thorium fuel materials and alternate fuel arrangements were evaluated.

Methods. The one-dimensional discrete-ordinates transport code, ANISN, was employed for all neutronics calculations with the P_3S_6 approximation in cylindrical geometry. All the nuclear data used were from the DLC-37 or DLC-41 libraries and were collapsed into a coupled 25 neutron and 21 gamma-ray group structure.

Alternate Materials and Arrangements. The thorium materials considered as candidates for the blanket fertile material are thorium metal, ThC, ThC_2 and ThO_2. Several alternate arrangements for the thorium and lithium materials were investigated. The results are summarized on Table 6 for the two most promising concepts, optimized to have a tritium breeding ratio of T/n = 1.1. Design I is a typical fast fission hybrid blanket which consists of a fertile zone immediately behind the first wall followed by a Li_2O tritium breeding zone. The design IV is basically a Li_7Pb_2 "pure fusion" blanket followed by a thorium fertile zone and is called a fission-suppressed hybrid blanket. The thickness of the Li_7Pb_2 zone is 0.5 m in this design in order to have a tritium breeding ratio of 1.10. The U-233 production performance of this fission-suppressed blanket is only 65% of that of the fast fission Design I concept. The Design IV blanket is of interest mainly because of its potential safety advantages. By suppressing fission, the fission product inventory can be kept very low and the after-heat problems of the blanket are expected to be no more than those for pure fusion designs. Although this concept has not been extensively studied, it appears that the fission-suppressed thorium blanket design could offer attractive safety advantages.[13] With increased concern about nuclear safety being evident in the fusion community, this concept appears to deserve further attention in the future.

TABLE 6
NUCLEONICS OF ALTERNATIVE BLANKETS

	DESIGN I				DESIGN IV
THORIUM FUEL	Th	ThC	ThC_2	ThO_2	Th
FERTILE ZONE THICKNESS (mm)	95	70	50	50	125
U-233 PRODUCTION* (U/D-T NEUTRON)	0.63	0.47	0.34	0.32	0.40
BLANKET ENERGY* MULTIPLICATION, M	2.45	2.10	1.85	1.75	1.30

*BOL VALUES

Reference Design. Based on the above discussions, the Design I blanket with thorium metal as fuel material was chosen as the gas-cooled TMHR reference design. The time-averaged tritium breeding ratio and uranium production rate over the blanket lifetime of 9.6 MW-yr/m^2 are 1.1 and 0.84 atoms per D-T neutron, respectively, and the blanket energy multiplication is 5.2. The end-of-life uranium enrichment is about 3.35%. The uranium enrichment in the thorium metal zone is quite uniform, with at most a 10% variation around the average value of 3.35%. The nuclear heating in the thorium zone differs spatially by about a factor of 2 with the peak heating of $\sim$120 MW/m^3 (at a neutron wall loading of 1.5 MW/m^2) occurring at the end of life near the first wall. Owing to the nuclear heating increase at the end of life, the estimated adiabatic afterheat meltdown time of the thorium fuel is reduced from about one hour at the beginning of life to about 11 minutes. A reliable blanket auxiliary cooling system will thus be required.

Analysis of the TMHR performance when the fast-fission thorium metal blanket was used showed that the reactor power production and breeding performance were quite sensitive to changes in the fast-fission blanket design, reactor length and fuel residence time. Because of concerns about this design sensitivity and concern about afterheat cooling under postulated accident conditions, a fission-suppressed blanket alternative was given further consideration. Use of a fission-suppressed thorium blanket option could further increase the adiabatic meltdown time by a factor of 50 or more and might make possible the design of a blanket that could be cooled passively for afterheat removal. Preliminary calculations indicated that fission-suppressed thorium blankets can achieve performance comparable to fast-fission thorium-metal blankets in terms of fissile fuel production per unit thermal power and a large reduction in blanket energy multiplication; the comparison is shown in Fig. 8. Because of its potential safety

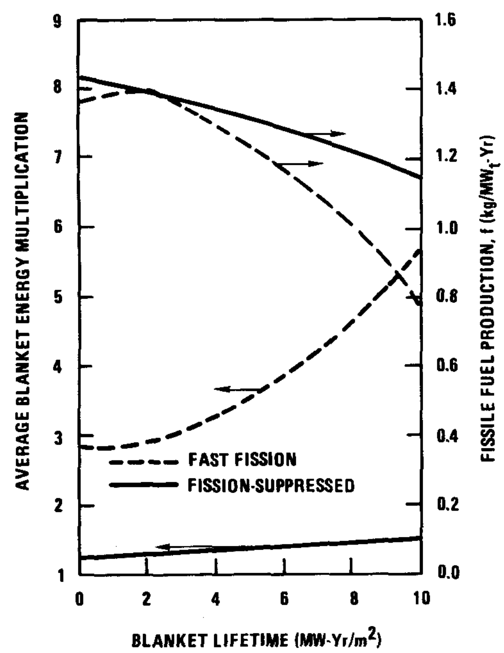

Fig. 8. Performance of fast-fission and fission-suppressed blankets.

advantages and high support ratio, the fission-suppressed design carries the promise of environmental superiority over fast-fission designs.

Shield Design. Changeout of two of the 16 TMHR blanket modules is expected to occur once every year giving a module life of 8 years. During the blanket replacement, the superconducting magnet could be annealed. The 316 SS + B_4C shield thickness required to keep the atomic displacement in the copper stabilizer below an accumulated value of 5 x 10^{-5} dpa in 8 years (6.4 continuous operating years) is estimated to be 0.7 m. The dose on the epoxy insulation material is about 4.4 x 10^8 G for a continuous plant operating lifetime of 30 years which is below the limit of 5 x 10 G.

Power Conversion System

The power conversion system consists of a primary coolant loop, a secondary coolant loop and an auxiliary cooling system. The primary coolant loop consists of 16 blanket modules which are interconnected by ducts and the hot and cold helium manifolds at one side of the reactor. There are 8 monotube helical-coil design steam generators and 8 helium turbo-circulators. Based on previous experience, the helium loop pressure drop through the steam generators is about 54 kPa (8 psi). The helium ducts experience about 55 kPa (8 psi) pressure drop. The pressure rise through the helium circulator is about 170 kPa (25 psi).

The secondary coolant loop consists of 8 once-through steam generators, including re-superheating sections, the turbine generator, steam-turbine drives for the helium circulators, condensers, feed pumps, feedwater heaters and associated piping. The auxiliary cooling system is designed to provide an independent means of cooling the shutdown reactor and removing the decay heat produced by the blanket. This loop consists of 2 heat exchangers and 2 circulators.

Blanket Cost Estimates

A cost estimate for the helium-cooled large blanket module was performed to evaluate the direct cost and the annual blanket replacement cost of the blanket. This estimate is subject to many uncertainties inherent in this early conceptual design stage but should be useful for comparison with other blanket options and for identification of the major cost components of the helium-cooled TMHR blanket design.

The direct costs for the complete 39 m long TMHR blanket was estimated to be $186 M in July 1979 dollars. The annual replacement and fuel recovery cost is $58 M, or $21 per gram of bred ^{233}U.

Conclusions

General Atomic Company has designed a helium-cooled blanket for the TMHR which generates an average thermal power of ∿4000 MW with ^{233}U production of 2.7 metric tons per year. The characteristics of this blanket are summarized on Table 7. This design appears to be technically sound and uses fairly conventional gas-cooled reactor technology to minimize the need for extensive development programs. The thorium fuel cycle is intended to interface with a thorium/^{233}U fuel cycle economy that is expected to be available in the early part of the 21st century. Although new head-end facilities will be needed for TMHR fuel, the reprocessing and recycle facilities expected to be in existence can be used directly. Thus fuel cycle development will not be needed to implement the TMHR.

TABLE 7
HELIUM-COOLED TANDEM MIRROR HYBRID
BLANKET CHARACTERISTICS

REACTOR POWER:[a]	
NUCLEAR POWER	4000 MW (t)
BLANKET POWER	3817 MW (t)
^{233}U PRODUCTION[b]	2647 kg/YR
BLANKET DIMENSIONS:	
THORIUM ZONE THICKNESS	0.12 m
Li_2O ZONE THICKNESS	0.45 m
SS-316 REFLECTOR THICKNESS	0.10 m
SS-316/B_4C SHIELD THICKNESS	0.70 m
BLANKET NEUTRONICS:[c]	
TRITIUM BREEDING RATIO	0.93/1.24
^{233}U BREEDING RATIO	0.92/0.75
ENERGY MULTIPLICATION	2.7/8.1
BLANKET EXPOSURE	0/9.6 MW YR/m^2
BURNUP	0/1.7%
HELIUM CONDITIONS:[c]	
PRESSURE	5.6 MPa
PRESSURE DROP IN BLANKET	3.4/20.7 kPa
INLET TEMPERATURE	285°C
OUTLET TEMPERATURE	515°C

[a] REACTOR TIME- AND SPACE-AVERAGED VALUES.
[b] 80% CAPACITY FACTOR ASSUMED.
[c] LOCAL BLANKET VALUES AT BEGINNING OF LIFE/END OF LIFE.

The mechanical design of the blanket system attempts to maximize the topological advantages of the simple cylindrical Tandem Mirror geometry. The ease of lateral access is utilized to provide the capability for rapid changeout of entire reactor modules. The entire blanket and solenoid section of the reactor is thus replaceable. By using spare modules, the reactor can be rapidly put back into operation while refurbishment or repair of spent modules is accomplished off-line in the hot shop. The design is within the imposed temperature and pressure drop limits from thermal-hydraulic considerations, with careful integration of the mechanical and neutronics designs.

The reference fast fission thorium blanket has excellent neutronic performance, producing significant amounts of bred fuel and blanket energy. The cycle- and blanket-averaged ^{233}U breeding ratio is 0.84 ^{233}U atoms per D-T neutron and the average blanket multiplication is 5.2. The local blanket energy multiplication varies from 2.8 at BOL to 8.5 at EOL. By changing out one-eighth of the blanket (two modules) per year the power swing of the entire reactor can be kept to a tolerable ±6%. The power swing for an individual module, however, is more than a factor of three from BOL to EOL, which complicates the thermal-hydraulic design. The peak blanket fuel power density at EOL is 120 W/cc, 96% of which is due to fission power. This results in a high level of afterheat and an adiabatic meltdown time of only 11 minutes, which necessitates the provision of highly reliable auxiliary cooling and decay heat removal systems.

The neutronics performance of the blanket is quite sensitive to the details of the design. The average blanket multiplication is a strong function of blanket lifetime, first wall thickness and thorium zone design. The reactor power for a given length (or length for a given power) is directly proportional to average multiplication. Thus, the blanket design, reactor length and fuel lifetime must be carefully coordinated to achieve the simultaneous design goals of a given reactor power level, adequate tritium breeding, acceptable fuel irradiation lifetime and favorable fuel handling economics. This sensitivity will make final design specifications important, and difficult. Because of these concerns about the neutronic design and performance of the reference fast-fission thorium blanket, a closer look should be taken at the relative merits of the fission-suppressed blanket concept. The economic tradeoffs inherent with use of a relatively low performance but environmentally attractive hybrid blanket of this type should be evaluated.

References

1. R.W. Moir, et al., "Tandem Mirror Hybrid Reactor Design Study Annual Report," Lawrence Livermore National Laboratory report UCRL-52875, September 1980.

2. J.D. Lee, "Parametric Systems Analysis for Tandem Mirror Hybrid REactors," Proceedings 4th Top. Mtg. Tech. Cont. Nucl. Fusion, October 14-17, 1980, King of Prussia, PA.

3. G.M. d'Hospital and P. Fortescue, "On the Choice of Gas Coolant for Nuclear Reactors," General Atomic Company report GA-7344, August 1966.

4. G.M. d'Hospital and G.R. Hopkins, "Gas Cooling for Fusion Reactor Blankets," General Atomic Company report GA-10004, May 1970.

5. D.J. Bender, et al., "Reference Design for the Standard Mirror Hybrid Reactor, joint LLL/GA report UCRL-52478/GA-A19796, May 1978.

6. J.A. Maniscalco, "Fusion-Fission Hybrid Concepts for Laser Induced Fusion," Nucl. Tech. 28, 98 (1976).

7. D.J. Bender and J.D. Lee, "The Potential for Fissile Breeding with the Fusion-Fission Hybrid Reactor," LLL report UCRL-77887, June 1976.

8. S.N. Rosenwasser, et al., "The Application of Martensitic Stainless Steels in Long Lifetime Fusion First Wall/Blankets," First Topical Meeting on Fusion Reactor Materials, Miami Beach, Florida, January 1979.

9. R. Wiswall and E. Wirsing, "The Removal of Tritium from Fusion Reactor Blankets," Brookhaven National Laboratory report BNL 50748 (1977).

10. R.G. Clemmer, D.R. Armstrong and N.E. Parker, "An Experimental Study of the Reaction of Li_7Pb_2 with Water," Trans ANS 32, June 1979, pp. 71-72.

11. K. Okula and D.K. Sze, "Tritium Recovery from Solid Breeders: Implications of the Existing Data," Proc. Conf. on Tritium Tech. in Fission, Fusion and Isotopic Applications, April 29-May 1, 1980, Dayton, Ohio.

12. C.P.C. Wong, et al., "Thermal-Hydraulics Design Considerations for the Tandem Mirror Hybrid Reactor Blanket," Proc. 4th Top. Mtg. Tech. of Cont. Nucl. Fusion, October 14-17, 1980, King of Prussia, PA.

13. I. Maya, K.R. Schultz, R. Katz and F.S. Dombek, "Safety Evaluation of Hybrid Blanket Concepts," Proc. 4th Top. Mtg. Tech. of Cont. Nuc. Fusion, October 14-17, 1980, King of Prussia, PA.

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