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STATUS REPORT ON THE
CONCEPTUAL DESIGN OF A
COMMERCIAL TOKAMAK HYBRID REACTOR (CTHR)

MASTER

SEPTEMBER, 1979

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ABSTRACT

COMMERCIAL TOKAMAK HYBRID REACTOR (CTHR)

At a time when the potential benefits of various energy options are being seriously evaluated in many countries throughout the world, it is both timely and important to evaluate the practical application of fusion reactors for their economical production of nuclear fuel. The fusion hybrid reactor represents a concept that could assure the availability of adequate fuel supplies for a proven nuclear technology and have the potential of being an electrical energy source as opposed to an energy consumer as are the present fuel enrichment processes.

Westinghouse Fusion Power Systems Department, under contract with the Department of Energy, Office of Fusion Energy, has developed a preliminary conceptual design for an early twenty-first century fusion hybrid reactor called the Commercial Tokamak Hybrid Reactor (CTHR). This design was developed as a first generation commercial plant producing fissile fuel to support a significant number of "client" Light Water Reactor (LWR) plants. The study has been made in sufficient depth to indicate no insurmountable technical problems exist and has provided a basis for valid cost estimates of the hybrid plants as well as the hybrid/LWR system busbar electricity costs. This energy system can be optimized to have a net cost of busbar electricity that is equivalent to the conventional LWR plant, yet is not dependent on uranium ore prices or standard enrichment costs, since the fusion hybrid can be fueled by numerous fertile fuel resources.

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EXECUTIVE SUMMARY

The likelihood that a growing shortfall in worldwide nuclear fuel resources will develop early in the next century has provided the stimulus for increased interest in the fusion hybrid concept in recent years. The hybrid approach offers some uniquely beneficial characteristics for the production of significantly larger quantities of fissile fuel than that obtainable from fission breeder reactors. This potential for substantial fuel production is attributable to physics characteristics generic to the hybrid concept. The use of a tokamak as a basis for a fusion hybrid appears to provide the nearest term approach to demonstrating this concept. Understanding of the plasma physics of the tokamak is the farthest advanced of any of the fusion confinement concepts currently under development. Operation of a tokamak in a beam injected mode is a promising approach for early hybrid application. However, the tokamak also offers the potential of reaching "ignition" conditions where all beams could be turned off during the pulse. This potential is in contrast to that of the mirror and the ICF confinement concepts where some input energy is always required to sustain the reaction and/or to inhibit leakage of the reacting plasma. These features tend to offset the obvious complexity of the tokamak which has given rise to concerns on the part of some investigators regarding the viability of the tokamak concept as a reactor.

The decision to base the initial application of the tokamak principle in the Tokamak Fusion Test Reactor on the beam driven or "Two Component Torus" led to early utility interest in the TCT concept as a hybrid for fuel production and/or waste disposal. The subsequent trend in tokamak parameters moved toward high β operation. This background in beam driven tokamak hybrids and in approaches to high β operation led to the conceptual design of a Demonstration Tokamak Hybrid Reactor (DTHR) for DOE/OFE. This DTHR size was large enough to offer the potential of ignited operation if favorable physics scaling relationships were found to prevail. A sub-ignition beam driven mode of operation was established as the reference mode to insure that performance objectives could be met under less favorable plasma physics conditions. With

these guidelines, the DTHR ($R = 5.2$ m, $a = 1.2$) is probably the largest such reactor one would consider if the initial objectives were limited to a hybrid (as distinguished from a pure fusion) development program. Furthermore, the DTHR might be a better choice as an immediate predecessor to a Commercial Tokamak Hybrid Reactor rather than an initial hybrid demonstration. In any case, the DTHR formed the basis from which to establish design guidelines and design concept approaches for the CTHR.

The design guidelines established for the CTHR are summarized in Table 1. Many of these guidelines such as the first wall loading, magnet type and peak field, plasma exhaust system and neutral beam criteria are the same as those employed for DTHR. The principal differences are those related to commercial operation: higher plasma duty cycle, higher plant availability, tritium self-sufficiency, electrical power production at least equal to required operating power, full blanket coverage, first wall lifetime compatible with blanket refueling cycle, and access for refueling and reactor maintenance operations.

These guidelines set tritium breeding self-sufficiency as an initial requirement with fissile fuel production as the main design goal. Electricity production was to be at least adequate to meet the hybrid circulating power requirements, although any excess electricity produced could be sold with that produced by the "client" LWR plants supported by the hybrid fissile fuel. The hybrid and LWR spent fuels were both reprocessed by chemical methods and an equilibrium cycle based on the plutonium recycle scenario was assumed. The net system busbar cost of electricity was the "bottom line" economic measurement value.

CTHR CONCEPT DESCRIPTION

The description of the CTHR contained in this status report is a summarization of results of the project during Fiscal Year 1979. Work on this project is scheduled to continue through the 1980 fiscal year.

Selection of a size for the CTHR was influenced by trade-offs involving several factors. On the one hand, there are incentives which favor a larger

TABLE 1
CTHR DESIGN GUIDELINES

- Ignited Plasma (Near Ignition Option); 2 MW/m^2 Average Wall Loading
- Superconducting Nb_3Sn Magnets: 12 T Peak Field
- Positive Ion Neutral Beams ($\leq 200 \text{ keV}$) with Direct Recovery; Negative Ion Option
- Plasma Duty Cycle ≥ 0.9
- Bundle Divertor Plasma Exhaust System
- First Wall Lifetime and Magnet Service Requirements Compatible with Blanket Refueling Cycle
- Provision for Full Blanket Coverage
- U-Pu Fuel Cycle with Chemical Reprocessing
- Fissile Production Maximized (Goal: 3 w/o in 4 Years Residence Time)
- Electrical Power Production $\geq$ Required Operating Power
- Tritium Self-Sufficiency (Breeding Ratio ~ 1.1 at Start of Life)
- Reactor Plant Availability 0.7

size reactor to benefit from economics of scale and to insure adequate space and accessibility for a blanket configuration that covers the toroidal fusion neutron source as completely as possible. On the other hand, increasing the size can lead to the need for extremely large capital investment for construction of a single unit and can result in the production of net electrical energy in amounts that may be difficult to effectively couple to a utility grid. The relationship of size to net electrical energy production is also affected by the type of fissile fuel cycle selected and the fuel management strategies adopted in the hybrid blanket. The uranium/plutonium fuel cycle was selected as the initial reference for the CTHR to provide a design baseline using current fuel cycle technology. This fuel cycle results in the production of significant amounts of energy in the hybrid blanket. Future plans include evaluation of the thorium/uranium-233 fuel cycle for the CTHR. This fuel cycle provides much less blanket energy, primarily due to the lower fast fission cross section of Th-232 as compared to U-238.

With the above considerations in mind, a reference size of 6.0 meters major radius and 1.4 meters minor radius was selected for CTHR. This size provides a net electrical output of 1980 MW_e which is larger than current central station plants, but which may be compatible with electrical grids in the early twenty-first century. Another consideration in establishing the reference size for the CTHR is the trend in economy of scale. Parametric studies using the COAST computer code have indicated a significant improvement in economics up to about 6 meters major radius with less dramatic gains beyond that point.

The CTHR conceptual design consists of a magnetically confined (tokamak) fusion reactor fitted with a fertile uranium blanket. Although two fertile fuels, UC and UO₂, were evaluated in various combinations with two tritium breeding fusile fuels, LiH and Li₂O, the UC-LiH concept was used for most of the economic analyses. The fusion driver concept was based on an ignited plasma, with the near-ignition beam driven option maintained as a viable backup alternative. A comparison of the reference mode plasma parameters to the beam driven backup mode of operation is shown in Table 2. All concepts and parameters were selected on the basis that technical feasibility would be achieved by 1995 to assure a proven commercial operation in the early to mid-21st century. Table 3 summarizes the major CTHR systems features.

TABLE 2
CTHR FUSION DRIVER PLASMA PARAMETERS

	REFERENCE MODE (IGNITED)	BACKUP MODE (BEAM-DRIVEN)
MAJOR RADIUS, m	6.0	6.0
MINOR RADIUS, m	1.4	1.4
ASPECT RATIO	4.3	4.3
ELONGATION	1.6	1.6
PLASMA VOLUME, m ³	370	370
TOROIDAL FIELD ON AXIS, T	5.2	5.2
TOROIDAL FIELD ON COND., T	12.0	12.0
PLASMA CURRENT, MA	6.8	6.8
NEUTRAL BEAM ENERGY, keV (Negative Ion)	200	200
NEUTRAL BEAM POWER, MW _t	80 (STARTUP)	200 (DURING PULSE)
PLASMA SAFETY FACTOR, q	2.8	2.8
PLASMA ENERGY GAIN, Q	IGNITED	5.8
PLASMA TEMPERATURE, keV ($\bar{T}$)	17.3	17.3
PLASMA PROFILE	PARABOLIC	PARABOLIC
ELECTRON DENSITY, 10 ²⁰ m ⁻³ (n _e)	0.97	0.90
PLASMA β , %	6.3	6.3
POLOIDAL β , %	4.3	4.3
WALL LOADING, MW/m ²	2.0	2.0
FUSION POWER, MW _t	1200	1180
REQUIRED τ_E , s	1.38	0.65
EMPIRICAL τ_E , s	2.1	2.0
n _e τ_E , 10 ¹⁹ m ⁻³ s	13.4	5.9
PLASMA PULSE DURATION, s	1200	1200
PLASMA DUTY CYCLE FACTOR	0.9	0.9
PLANT OPERATING POWER CONSUMPTION, MW _e	215	650
HYBRID/LWR SYSTEM BUSBAR ELECTRICITY COST, mills/kWhr	~ 40	~ 43

TABLE 3
CTHR MAJOR SYSTEM FEATURES AND PARAMETERS

<u>PLASMA OPERATION</u>		<u>NEUTRAL BEAM INJECTION LINES</u>	
Plasma Pulse Duration, s	1336 max.	Beam Power, Energy	80 MW, 200 keV
Effective Burn Duration, s	1200	Single Arm Specifications:	
Recycle Time, s	≤ 136	Sources	4-50 A Negative Ion
		Vacuum Enclosure	8.0 m long x 2.5 m wide x 3.4 m high
Plasma Duty Cycle Factor	≥ 0.9	Charge State of Source Ions	Negative
Design Basis Plant Availability	0.7	Number of Arms	4
Design Basis Number of Pulses/Year	23,605	Vessel Aperture Per Line	0.45 m x 0.85 m
Pulse Scenario:			
Pulse OH (Bipolar)	2 seconds		
Neutral Beam Injection (Heating to Steady State)	5 seconds		
Burn	1200 seconds		
Shutdown	2 seconds		
<u>TOROIDAL FIELD COILS</u>		<u>PLASMA EXHAUST SYSTEM AND IMPURITY CONTROL</u>	
Number of coils	14	Bundle Divertor	Single Toroidal Position at Torus Mid-plane. Superconducting Nb ₃ Sn Circular Coils, On-line Regenerable Zr Al Gettering, Cryopanel.
Conductor Type	Nb ₃ Sn	Throughput to Divertor Burial Chamber	~ 4.8 x 10 ²² particles/s (1440 Torr s/s)
Coil Shape	Compact "D"		
Peak Field at Conductor	12 T		
Current per Turn	20 kA		
Average Current Density	3.3 kA/cm ²		
Cooling Mode	Forced Flow Supercritical Helium		
Structural Material	Stainless Steel 310		
<u>POLOIDAL FIELD SYSTEM</u>		<u>TRITIUM SYSTEMS</u>	
Conductor Material	Nb ₃ Sn	On-site T ₂ Inventory	Five Days
Location	Outside TF	Tritium Storage	Uranium Chip Beds
OH Flux Swing	67.5/32.5 v-s	Fuel Recovery-Recycling Plant	Continuous
Conductor Current	10.76/16.24 kA	Tritium Cleanup Systems	Reactor Torus and Shell
Average Current Density	3.0/4.5 kA/cm ²	Fueling	Prefill: Gaseous T Refueling: T Pellets & D-Beams and Pellets
Cooling Mode	Forced Flow Supercritical Helium		
<u>ELECTRICAL SYSTEMS</u>		<u>VACUUM VESSEL</u>	
Facility Feed	Fixed Utility Constraints	Vessel Material	316-SS
S/C TF Supplies	Line-driven Rectifier Supplies	Construction	Rib stiffened shell
Neutral Beam Supplies	Line-driven Rectifier Supplies with Load Dumping Resistors	Vessel Cross Section	D-Shaped
Poloidal Field Power Supplies	AC-MG Storage-Rectifier Conversion	Composite Wall Thickness	0.5 cm
		Toroidal/Poloidal Limiters	Tungsten Rails (Radiation Cooled)
		Maximum Wall Temperature	427°C
		Peak Bulk Wall Temperature	380°C
		Vessel Coolant	Pressurized Water
<u>HEAT DISSIPATION SYSTEMS</u>		<u>VACUUM PUMPING</u>	
Facility Heat Sink	Direct Cycle Steam	Pumping System	Cryosorption Pumps or Panels
Vacuum Vessel Coolant	Pressurized Water	Base Pressure	1 x 10 ⁻⁸ torr
Shield Coolant	Borated Water		
Cryogenic Refrigeration Systems	L He; L N ₂		
Facility Air Conditioning	Fixed Base Load, Reactor Cell Air Conditioning		

Although the reactor was designed to achieve good fissile fuel production, the resulting concepts that evolved were also excellent power producers which significantly improved the economic performance.

With the major radius of 6.0 m and the minor radius of 1.4 m, the elongated plasma has an average temperature of 17.3 keV based on a parabolic profile. The electron density-confinement time product is $13.4 \times 10^{19} \text{ sm}^{-3}$. The fusion power is 1200 MW and the average neutron wall loading is 2 MW/m^2 . The fusion driver was developed with the maximum toroidal field at the conductor restricted to 12 tesla and the neutral beam energy limited to 200 keV for positive ion beam systems. The larger sized concepts studied used negative ion beam systems to enable higher energy beams. The reference concept used a negative ion beam system for economic reasons. A trimetric view of the reactor is shown in Figure 1. The 14 toroidal field (TF) coils are equally spaced with a bundle divertor plasma exhaust system located between one pair. The preliminary sizing of the TF coils was achieved using the COAST⁽⁸⁾ computer code. Detailed sizing and layout work then resulted in a fairly detailed TF coil concept. This concept was used as input data for a final magnetic force analysis to confirm the structural adequacy of the design. The magnetic forces on the poloidal field (PF) coils and TF coil systems were computed by the MAFCO code⁽⁹⁾ in sufficient detail as required for the mechanical design of support and restraining components. It was found that:

- The net overturning moment about the horizontal axis on the TF coils was not as large as expected.
- The twist moments about the vertical axis through the plasma axis were larger than expected.
- The tension along the circumference of the TF coils is not constant, which is consistent with compact D-shaped coils.
- The vertical forces on the SF coils are directed toward the midplane, while hoop stress causes the coils to expand.

The PF coils are located in three general areas: the coils in the bore of the torus; the coils above and below the plasma; and the coils on the outer periphery of the TF coils. The PF coils are all located outside of the TF

Figure 1. Commercial Tokamak Hybrid Reactor Concept

coils and have been grouped and assembled in common dewar systems. This provides several advantages:

- Reduces the space envelope required compared to individual dewars.
- Reduces the manufacturing cost by using common dewar walls, insulation, helium headers, etc.
- Easier handling for installation and removal.
- Better access for blanket refueling and TF coil removal by lifting or lowering a unitized assembly.

The majority of the OH coils (42 of 52) are located in the bore of the device in a vertical stack positioned in slots in the bucking rings. Five SF coils are located in the bore. The Nb_3Sn superconducting cables and cryocoolant lines are carried through slots in the bucking rings. The upper coil cables are fed up the common torus bore dewar chamber to the top where transition to copper busbar occurs. The lower coil cables feed down and across to the center of the central support structure to pass below the reactor to that transition chamber. The cryocoolant lines are fed by manifold lines between the bucking rings and the torsion tube. The four neutral beam injectors are located beneath the reactor to enhance access for blanket refueling; the beams inject $\sim 10^\circ$ off vertical.

The vacuum vessel concept was developed with 28 segments, joined by bolted flanges and sealed with flexible metal O-rings. This concept of assembly is feasible when an external reactor vacuum shell is used since the vacuum vessel must maintain only a partial vacuum around the plasma. An effective wall thickness of 0.5 cm was established for neutronic analysis based on the ~ 1 cm diameter coolant tubes and a 0.4 cm thick vacuum shell.

The blanket concepts operate on the U-Pu cycle and are designed to achieve tritium self-sufficiency while maximizing the fissile fuel production within thermal-hydraulic and mechanical design constraints. The two generic blanket concepts evaluated were: 1) a UC fueled, stainless steel clad and structure, helium cooled blanket (UC-SS-He); and 2) a UO_2 fueled, zircaloy clad, stainless steel structure, boiling water cooled blanket (UO_2 -Zr-BW). The

motivation for investigating the UO_2 -Zr-BW blanket was that it utilized many features of proven light water fission reactor technology and also was similar to the DTHR concept studied previously. The UC-SS-He blanket concept offered the potential for improved neutronic performance relative to the UO_2 concept, but at the expense of employing more advanced, less well developed technology and materials. The two different tritium breeding media, Li_2O and LiH , were evaluated using lead or graphite as a neutron multiplier or reflector.

The neutronic performance was calculated for sufficient irradiation exposure time to achieve about 3% fissile fuel enrichment in the blanket. An important aspect of the analysis was to investigate the neutronic benefits of full blanket wall coverage in a tokamak fusion hybrid reactor. The results of the comparison of full (inner and outer) fertile blanket coverage versus outer fertile blanket coverage only indicated that increases in fissile fuel production of up to about 20% would be possible with the full blanket coverage. The difference occurs because of increased neutron multiplication with the inner fertile zone, but is somewhat smaller than might be expected because of the non-uniform tokamak neutron source distribution which results in about 70% of the D-T fusion neutrons being directed toward the outer blanket region. Also, for an outer fertile blanket only, the inner region can be designed with a lead material which provides some additional neutron multiplication. Another factor in considering the benefits of fertile material in the inner region of a tokamak hybrid is a possible decrease in the plant availability due to increased reactor downtime to service and refuel the inner blanket. Reduced performance can result in lower plant availabilities when vacuum vessel and inner blanket removal are taken into account. If this increased plant outage becomes too large, the neutronic benefit of using full blanket coverage will be negated.

Other conclusions from the neutronic analyses indicated certain advantages in the use of LiH as compared to Li_2O for the tritium breeding material. A blanket using LiH can be thinner than one using Li_2O due to the excellent moderating properties of the hydrogen. Also, the same moderating effect with the LiH decreases parasitic absorption in the structure in the lithium material zone compared to the use of Li_2O . The investigation of the use of

lead in the blanket indicated that significant benefits could result by using it both as a reflector behind the fertile zone and as a neutron multiplier in the inner region for the case of outer fertile blanket coverage only.

For each of the blankets a conceptual mechanical design was developed and the thermal-hydraulic performance was evaluated. High pressure coolants were used in the blankets to obtain high power conversion efficiency and low pumping power requirements. The steam pressure at the turbine throttle valve for the UC blanket is about 8.62 MPa (1250 psia) and that for the UO_2 blanket is about 6.20 MPa (900 psia). With these pressures the present day nuclear turbine technology for economical commercial applications can be utilized. Since the plasma duty cycle for the CTHR is 90% with a proper design of a thermal energy buffer-storage system, the blanket coolant exit temperature averaged over one complete burn cycle can be assumed to be 95% of the value during the plasma-on period. The steady state blanket exit temperature is therefore about 450°C for the helium coolant. This temperature was used as the design value for the power conversion system. Considering six feedwater heaters and an 80% turbine stage efficiency, the estimated gross thermal efficiency for the cycle is 37% and the net cycle thermal efficiency was estimated to be 35%. Five primary loops are included; one for the inner blanket and four for the outer blanket. This maintains reasonably sized coolant headers, provides for better safety through five semi-independent decay heat removal systems, and permits refueling of the blanket by quadrants independent of the other three outer blanket loops. In-situ tritium extraction could be added to the UC-LiH concept coolant system if higher temperature materials were used for the clad and structure to permit "driving-off" the hydrogen from the DT in the LiH fusile fuel, or by changing the fusile fuel to Li_2O to allow continuous tritium removal.

Due to the hydrodynamic limitations for the UO_2 boiling water cooled concepts, the water mixture quality must be kept below 40%. Studies on the variation of the water coolant density indicate decreased fissile fuel production and increased thermal power production with increasing water density. It was determined that pumping power limitations restricted the length of flow in the fertile fueled front zone of the modules to 1.5 m for the UC helium cooled concept. For the UO_2 boiling water cooled concepts,

the channel exit quality restricted the fertile fueled zone coolant path length to about 2.5 m. These factors had significant affects on the initial mechanical concepts.

The blanket mechanical concepts were designed with a separate, independent vacuum vessel-first wall as a part of the CTHR design. The volume fractions used in the neutronic analyses (50% fertile fuel) were set as mechanical design goals, but could not be achieved for the fertile zone on the first iteration (45% fertile fuel). However, the calculated fuel volumes and weights in the mechanical concepts were all within $\sim 4\%$ of the fuel weights used in the neutronic analyses. Thus, the performance of the mechanical concepts approximately achieve the analyzed performance. The initial tritium breeding ratio of 1.1 appeared to be conservative and was reduced slightly (to ~ 1.05 at BOL) to permit increasing the fission events per fusion neutron.

The blanket module was designed so that maintenance and replacement could be accommodated by remote operations. The sectional geometry would allow individual sectors to be removed by simple linear motions to withdraw it from between two TF coils. There are three blanket sectors between each pair of TF coils. Thus, the center outer sector can be directly removed, while the two adjacent outer sectors have to be circumferentially rotated to the unobstructed center area between TF coils to be removed. Two attachments on the back of each sector allow remote handling of the complete blanket sector.

The UC/LiH blanket has a number of advantages and attractive features as follows: 1) The fertile blanket zone has the versatility of being readily detached from the fusile blanket and support structure; 2) The internal fuel assemblies are fabricated as a single unit which slides into the containing canister. Thus, to refuel the assembly, the hemispherical end cap is removed, the bred fuel assembly removed and a new assembly inserted, and 3) It is conceivable that the LiH fusile blanket zone could be designed for in-situ tritium recovery by using higher temperature structural materials from that used in the fertile blanket. A disadvantage of the UC concept is its mechanical complexity and the fuel manufacturing developmental needs. To fully evaluate this blanket concept, it is necessary to continue the design effort with in-depth structural and thermal-hydraulic analysis and development

of serviceability methods to determine if this design is viable. Based on the present level of analysis, the concept appears to be attractive. The performance and cost characteristics of the UC-LiH blanket concept are summarized in Table 4.

The boiling water cooled blanket module appears to be a viable alternative concept based on the level of analysis performed to date. However, the blanket fissile production is extremely sensitive to the water density. Blanket coolant exit conditions close to that required for a commercial power conversion system were developed with reasonable performance still possible. The fertile region requires conventional technology in regards to fuel and fuel form. The fusile and reflector zones, which are helium cooled, are not state-of-the-art technology; however, they do not appear to involve unreasonable development schemes. The refueling required individual sector and module replacement, which directly affect economics because of the plant availability factor.

An alternative for the UO_2 concepts is a common coolant system where the helium is replaced with boiling water fed to and from the fusile blanket which preheats the coolant to reach saturation temperature before it enters the fertile blanket. The compatibility of LiH or Li_2O with boiling water coolant must be studied further to assure the safety of this concept. However, the single coolant concept would significantly reduce the coolant manifold and headering space requirements. Performance of the tritium breeding zone is very similar with either coolant when LiH is used. However, using boiling water coolant with Li_2O could improve it's performance with respect to the LiH performance. Neutronic analysis of these alternatives was not conducted.

Both the reference and alternate concepts are self-sufficient in tritium production (TBR ~ 1.1) and produce about the same thermal power (4000 to 5000 MW BOL to 6000 to 8000 MW EOL). However, the UC helium cooled blanket produces about 75% more fissile ^{239}Pu fuel due to both the increased uranium density with the UC compared to the UO_2 and also because of the harder neutron spectrum with the helium compared to the boiling water. In both blankets the power production increases by approximately 79% over the four

TABLE 4
UC-LiH HELIUM COOLED BLANKET CONCEPT SUMMARY

BLANKET SPECIFICATIONS

FERTILE FUEL ZONE	UC	UC
COVERAGE	FULL	OUTER
COOLANT	HELIUM	HELIUM
INNER BLANKET THICKNESS, cm	16	—
OUTER BLANKET THICKNESS, cm	16	24
FUEL/STRUCTURE/COOLANT, %	50/20/30	50/20/30
CLAD/STRUCTURE MATERIAL	316 SST	316 SST
PITCH TO DIAMETER RATIO (PLASMA SIDE/BACK)	1.2/1.6	1.2/1.6
MAX. COOLANT CHANNEL LENGTH, m	1.5	1.5
WIDTH OF BLANKET MODULE, m (OUTER/INNER)	1.26/1.67	1.26/—
THICKNESS OF BLANKET MODULE, m	0.77	0.85
OVERALL HEIGHT OF BLANKET SECTOR, m	8.0	8.0
RADIUS OF SECTOR CURVATURE, m	3.1	3.1

TRITIUM BREEDING FUEL ZONE

COOLANT	HELIUM	HELIUM
INNER BLANKET THICKNESS, cm	34	36
OUTER BLANKET THICKNESS, cm	30	30
FUEL/STRUCTURE/COOLANT, %	70/10/20	70/10/20
CLAD/STRUCTURE MATERIAL	316 SST	316 SST
REFLECTOR OR MULTIPLIER MATERIAL/ LOCATION, THICKNESS INNER/OUTER	Pb/front 10/10	Pb/front 14/10

BLANKET PERFORMANCE CHARACTERISTICS

(BOL - 2 yr. - 4 yr.)

TRITIUM BREEDING RATIO	1.05-1.20-1.34	1.06-1.18-1.29
NEUTRON EXPOSURE, MW-yr/m ²	4.8	4.8
F, NET ²³⁹ Pu ATOMS/FUSION NEUTRON	1.10-0.97-0.85	0.97-0.87-0.78
T + F	2.15-2.17-2.19	2.03-2.05-2.07
M, BLANKET ENERGY MULTIPLICATION	5.9-7.9-9.9	5.1-6.7-8.3
HEAVY METAL INVENTORY, THM	412-410-406	382-380-376
E, FISSILE ENRICHMENT, w/o	0.25-1.64-2.89	0.25-1.50-2.64
CUMMULATIVE BURNUP, MWd/THM	12,700 (4 yr.)	
HEAVY METAL DISCHARGE, THM/yr.	101	92
G, FISSILE PRODUCTION, kg ²³⁹ Pu/yr.	3090-2730-2410	2570-2300-2050
DESIGN FISSILE PRODUCTION, kg/yr.	2868	2413
P, AVERAGE THERMAL POWER, MWt	4500-6100-7620	2760-4920-6080
DESIGN POWER RATING, MWe	1980	1600

(a) Limit due to low film coefficient of coolant and structural material limitations. Effect of plasma-off time included.

(b) Steam pressure at SG-superheater exit for indirect cycle.

(c) Tandem compound units with 6 feedwater heaters and 80% stage efficiency

BLANKET OPERATING CONDITIONS

AVERAGE BLANKET POWER DENSITY, W/cm ³	130
PEAK FERTILE BLANKET POWER DENSITY, W/cm ³	230
PEAK FUEL ROD POWER DENSITY, W/cm ³	420
INLET COOLANT PRESSURE, MPa (psia)	8.62 (1250)
INLET COOLANT TEMPERATURE, °C	280
OUTLET COOLANT TEMPERATURE, °C	478
AVERAGE COOLANT OUTLET TEMPERATURE, °C	450 ^(a)
MAXIMUM STRUCTURAL TEMPERATURE, °C	500
MAXIMUM CLAD TEMPERATURE, °C	550
MAXIMUM FUEL TEMPERATURE, °C	660
COOLANT PUMPING POWER TO BLANKET THERMAL POWER, %	2.0

TURBINE PARAMETERS

STEAM CYCLE	INDIRECT
PRESSURE, psia	1250 ^(b)
TEMPERATURE, °C (SUPERHEATED °C)	420 (120)
FEEDWATER TEMPERATURE, °C	150
SIZE, MWe	1000
NO. OF UNITS IN PARALLEL	2 ^(c)
GROSS THERMAL EFFICIENCY, %	37
ESTIMATED NET CYCLE EFFICIENCY, %	35

CTHR COST CHARACTERISTICS

BLANKET CONCEPT	UC-LiH	UC-LiH
BLANKET COVERAGE	FULL	OUTER
TOTAL CAPITAL COST, \$M	3602	3411
GROSS OPERATING COST, \$M	731	683
ELECTRICITY/FISSILE RATIO, GWhr/kg	4.22	4.06
NUMBER OF 1 Gwe LWR's SUPPORTED	8.0	6.7
CAPITAL COST/ELECTRIC POWER, \$/kWe	1827	2137
GROSS OPTG. COST/ELECTRIC POWER, mills/kWhr	60.5	69.8
GROSS OPERATING COST/FISSILE FUEL, \$/g ²³⁹ Pu	255	283
FUEL TRANSFER VALUE, \$/g ²³⁹ Pu	88	111

CLIENT LWR COSTS

NET FISSILE FUEL VALUE, \$/g fissile Pu	40	51
FUEL CYCLE COST, mills/kWhr	11.7	14.4
CAPITAL COST, mills/kWhr.	28.0	28.0

CTHR/LWR SYSTEM COSTS

BUSBAR ELECTRICITY COST, mills/kWhr	39.7	42.4
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year irradiation cycle. Means of levelizing the thermal power requires further study for the reference concept. The fissile production performance and average blanket thermal power versus irradiation time for four blanket concepts were used in the economic analysis. The thermal power for an equilibrium cycle was estimated for each concept to be approximately the two year average performance.

Economics will be a key factor in the determination as to which of these blankets is ultimately the most desirable for use in the CTHR. The hybrid power production is a much more sensitive factor on busbar electricity costs than is fissile fuel production and therefore, the $\text{UO}_2\text{-Li}_2\text{O}$ concept was less than 2 mills/kWhr higher than the UC-LiH concept, even though fissile production rates were quite different. While most of the analyses were for a four year irradiation period, sufficient to reach ~ 3% enrichment in the blanket, economics studies will be required to determine the optimum exposure time. It should be noted that the results represent an initial assessment. Further iterations between neutronic, thermal-hydraulic, structural, and mechanical design are required to arrive at an optimum design configuration. Additional considerations such as fuel shuffling and operation on an equilibrium cycle will be important.

ECONOMIC ANALYSIS RESULTS

The design of the CTHR was developed in sufficient depth to provide a basis for initial estimates of the capital cost and the plant operating cost. These costs were estimated for three different blanket fuel combinations and one partial coverage blanket (outer fertile blanket only, no fertile inner blanket). The busbar cost of electricity for a conventionally fueled LWR was established for comparison to the busbar electricity cost of these four hybrid blankets and the LWR reactors they fueled. The sensitivity of the busbar electricity cost to the hybrid size was analyzed to determine the hybrid size where equality with the "conventional" LWR electricity cost would occur.

The cost of the hybrid reactor plant was determined using the COAST-4 computer code⁽⁸⁾. Based on the input data for each of the respective blanket concept fuel combinations, the total cost of the CTHR plant was estimated.

The performances of the three full coverage blanket concepts were compared using key parameters and the hybrid cost. These cost factors related key products (fissile fuel and electricity production) to capital or operating costs. It was noted that none of these factors nor the LWR's supported were accurate indicators of which concept would have the lowest busbar electricity cost. The UC-LiH and UO_2 -LiH concepts were basically equal with an electricity cost of 39 to 40 mills/kWhr, while the UO_2 - Li_2O concept was slightly higher at $\sim$ 41 mills/kWhr. The 39 to 41 mills/kWhr values can be compared to the 37 mills/kWhr for a conventional LWR fuel cycle scenario using similar economic analysis techniques. Thus, all three concepts would require further optimization to be competitive as a 6 meter major radius reactor concept. In view of the preliminary nature of the hybrid analysis, however, the differences in busbar costs are not considered to be extremely significant.

A cost comparison between a full blanket, assuming a plant availability (P.A.) of 0.7, to an outer blanket at P.A. = 0.7 should bracket the probable performance range for that blanket concept. For the UC-LiH-Helium cooled concept, the busbar electricity costs fall between 39.7 and 42.4 mills/kWhr, which is a cost increase of $\sim$ 7% for the full blanket concept if the P.A. dropped to $\sim$ 0.6. The outer blanket concept had 16% less fissile production and 19% less power production, which resulted in 7% higher busbar electricity costs when the 70% plant availability was assumed for both concepts. This comparison is shown in Table 4.

The affect of the reactor size on performance and cost was studied parametrically using the COAST and SYMECON⁽⁷⁾ computer codes. The 6.0 m major radius CTHR with a UC-LiH-Helium cooled blanket concept was used as the base design. The major radius was reduced to 5.2 m (the DTHR concept size) and increased to 8.0 m. The cost of electricity and fissile fuel production decreases with increasing major radius. This trend is illustrated in Figure 2. It can be seen that the increase in electric power and fissile fuel production is more rapid than the hybrid capital and operating cost increases and therefore, the busbar cost of electricity decreases with increased reactor size. The typical trend for the tokamak concepts recently studied shows that the production costs drop fairly rapidly until the size reaches a 6 to 7 m radius. Thereafter, the reduction in cost is nearly linear in the range of

CTHR UC/LiH FULL BLANKET CONCEPT
2 MW/m² WALL LOADING

Figure 2. CTHR Performance and Costs Versus Reactor Size

parameters studied (up to 9 m). In comparing the CTHR results to the LWR "conventional" busbar cost of 37 mills/kWhr, a 7.5 m major radius concept would be cost equivalent without further optimization. However, when noting that this concept is a 3000 MW_e plant, it is probable that a more optimum concept in the 6 to 7 m radius range and a 2000 MW_e rating would be more realistic for an early to mid-twenty first century application. Obviously, extensive studies of various fuel management scenarios for a specific design would be beneficial in reducing the busbar cost of electricity for the symbiotic system.

CONCLUSIONS AND RECOMMENDATIONS

The conclusions developed during the FY'79 study of a commercial hybrid reactor and the economics of a symbiotic system of the hybrid and client LWR plants have been summarized in three categories. Table 5 lists the general conclusions relating to the fusion driver and hybrid reactor concept. Table 6 presents the significant conclusions concerning a major reactor component (system), the toroidal field coils. The TF coils have a major impact on reactor size and capital costs as well as the allowable space for the blanket. The remaining conclusions, shown in Table 7, concern the blanket concepts studied. The blanket performance has a direct impact on system economics.

Numerous recommendations for further study have been defined in Section 3.3. All of these studies relate to providing a better technical concept and more in-depth design detail which will enable more accurate cost and performance estimates. The availability of these additional details would contribute to the refinement of the economic analysis tools and data base. Table 8 provides a summary of the recommended refinements and further studies needed to fully develop the economic analysis techniques which are essential to the evaluation of the fusion driven hybrid reactor's potential for commercialization. The specific CTHR follow-on studies recommended for FY'80 are discussed in Section 3.3.

TABLE 5

CONCLUSIONS TO DATE - GENERAL

- FUSION DEVELOPMENT PROGRAMS ARE RAPIDLY APPROACHING PLASMA CONDITIONS NEEDED FOR HYBRID REACTORS.
- FUSION DRIVER CHARACTERISTICS ARE WELL ENOUGH DEFINED TO WARRANT SERIOUS, IN-DEPTH ENGINEERING STUDIES OF THE REACTOR SUBSYSTEMS.
- FUSION DRIVER PARAMETERS WITH LONG PULSE OPERATION (1200 sec.) AND HIGH DUTY CYCLE (0.9) APPEAR FEASIBLE.
- $R_0 = 6$ m APPEARS TO BE A REASONABLE SIZE FOR CTHR, BASED ON PARAMETRIC STUDIES.
- DEVELOPMENT WORK APPLIES DIRECTLY TO PURE FUSION REACTORS.
- DEMONSTRATION OF HYBRID FEASIBILITY IS POSSIBLE IN EARLY 1990's.
- FUSION HYBRIDS HAVE NEARER TERM APPLICATIONS AS FUEL SOURCES (FUEL LWR's, LMFBR's, LWBR's)
- MUCH OF THE FISSION REACTOR TECHNOLOGY IS APPLICABLE TO HYBRID BLANKET CONCEPTS.
- TWO PRODUCTS - FISSION FUEL AND ELECTRICITY - GIVE HYBRIDS UNIQUE ECONOMIC FLEXIBILITY, WHICH APPEARS COMPETITIVE IN EARLY 21st CENTURY.

TABLE 6

CONCLUSIONS TO DATE - TF COIL CONCEPT

- 14 TF COILS ON EQUAL SPACING FEASIBLE WITH BUNDLE DIVERTOR IN $R_0 = 6$ m CONCEPT.
- COMBINED STRESSES DICTATE NOSE CONSTRUCTION.
- WEDGING REDUCES NOSE THICKNESS.
- BUCKING RINGS REQUIRED FOR FAULTED COIL CONDITIONS AND COMPACT "D" COILS.
- OH COILS ADJACENT TO TF COILS FEASIBLE VIA BUCKING RING SLOTS.
- BEAM WEDGES STIFFEN COIL PLATES TO RESIST
 - OUT OF PLANE FORCES
 - TOP TO BOTTOM TWISTS
- CENTRAL TORSION TUBE MAY REDUCE TWIST IN OUTER LEGS.
- THERMALLY RESISTANT MOUNTING STRUCTURES FEASIBLE.
- COMMON EXTERNAL PF COIL DEWAR SYSTEMS ENHANCE TF COIL REMOVAL AND BLANKET REFUELING.
- SHEAR PANELS ABOVE BLANKET ACCESS AREA FEASIBLE.
- INDIVIDUAL OR DUAL TF COIL REMOVAL ACHIEVABLE.

TABLE 7

CONCLUSIONS TO DATE - BLANKET CONCEPTS

- COMMERCIAL SIZED MEDIUM PRESSURE (> 800 psia STEAM PRESSURE) POWER CONVERSION SYSTEMS FEASIBLE FOR TWO PHASED WATER COOLED CONCEPTS.
- STEAM QUALITY AT BLANKET EXIT SHOULD BE ~ 30 TO 40% TO ASSURE FLOW STABILITY, BUT LIMITS FLOW PATH LENGTH.
- HELIUM COOLANT REQUIRES HIGHER OPERATING PRESSURE BUT RESULTS IN HIGHER STEAM PRESSURE FOR POWER CONVERSION SYSTEMS.
- COOLANT PATH LENGTH DICTATES BLANKET MODULE CONCEPTS WITH HELIUM COOLANT.
- THIN FERTILE BLANKETS ATTRACTIVE WITH Pb MULTIPLIER/REFLECTOR AND LiH T-BREEDING FUEL.
- TRITIUM BREEDING SELF-SUFFICIENCY IS FEASIBLE WITH CONCEPTS STUDIED.
- FULL BLANKET COVERAGE HAS 20-25% MORE PERFORMANCE POTENTIAL THAN OUTER COVERAGE ONLY (UC/LiH CONCEPT).
- REFUELING LIMITATIONS MAY NULLIFY FULL BLANKET ADVANTAGES OVER OUTER BLANKET CONCEPT.
- IN-SITU TRITIUM REMOVAL MAY BE FEASIBLE WITH Li_2O , BUT NOT WITH LiH AND STANDARD STRUCTURAL MATERIALS.
- UC/LiH CONCEPT PRODUCES MORE FISSILE FUEL THAN UO_2/LiH OR $\text{UO}_2/\text{Li}_2\text{O}$ CONCEPTS, BUT BUSBAR ELECTRICITY COSTS ARE COMPARABLE (UC/LiH vs. UO_2/LiH) FOR SYMBIOTIC SYSTEMS.

TABLE 8
RECOMMENDED FURTHER STUDY - ECONOMICS

- REFINE COMPUTER CODES
 - COAST
 - SYMECON
- DETERMINE KEY HYBRID SENSITIVITIES
- DETERMINE KEY LWR SENSITIVITIES
- REFINE FUEL CYCLE DATA BASE
 - U-Pu FUEL CYCLE
 - Th-U FUEL CYCLE
 - CHEMICAL REPROCESSING
- DEVELOP SYMBIOTIC HYBRID/LWR SYSTEM SENSITIVITIES
 - TECHNICAL PARAMETERS (FUSILE FUEL OPTIONS, FISSILE FUEL CYCLES, ETC.)
 - ECONOMIC PARAMETERS (FIXED CHARGE RATES, INTEREST RATES, ETC.)
 - ADMINISTRATIVE PARAMETERS (PRIVATE VS. PUBLIC OWNERSHIP, ETC.)

1.0 INTRODUCTION

The purpose of this study is to prepare a scoping design of an integrated first-generation commercial fusion hybrid reactor plant based on the tokamak confinement concept. The work discussed in this report represents the FY'79 progress on the on-going study. The work scope of this study includes the design and analyses of major reactor and facility components in sufficient detail to show the technical credibility and provide the basis for valid cost estimates. The economic analyses includes an initial evaluation of alternative fuel cycles and a very preliminary comparison between fuel versus electricity production. Parametric cost analyses have been made to support a reactor design point selection and evaluate the blanket concept alternatives. Initial consideration was given to reactor maintainability and reliability, safety and environmental impact considerations, and hybrid plant integration into a total fission energy system.

The major objectives of the FY'79 program were to:

- Complete the Demonstration Tokamak Hybrid Reactor (DTHR) feasibility study initiated in FY'78.
- Develop an initial concept of a Commercial Tokamak Hybrid Reactor (CTHR) including scoping analyses of performance and economics.
- Develop an energy system economic analyses code capable of evaluating CTHR/LWR symbiotic systems with various fuel cycles.

The DTHR feasibility study was completed and documented in a summary report⁽¹⁾. Detailed DTHR reports were also prepared covering the neutronic⁽²⁾ and economic⁽³⁾ analyses of several direct fuel cycles, the blanket concept development⁽⁴⁾, the bundle divertor concept⁽⁵⁾ and cost scoping and sensitivity analyses⁽⁶⁾. The results of the DTHR study will not be recapitulated in this report except where a DTHR system has been used as the basis for the CTHR concept.

An energy system economic analysis code was developed in parallel with the DTHR and CTHR conceptual design effort. The general description of this symbiotic energy system economics code has been documented⁽⁷⁾. This code, SYMECON, has been used to determine the preliminary system economic results discussed in this report. This report provides the description of the initial CTHR concept and the scoping analyses performed thus far. Conclusions reached to date are provided as well as the areas requiring further study before meaningful conclusions on the commercialization of fusion hybrid reactors can be made.

2.0 CTHR REACTOR DESIGN AND ANALYSES

The CTHR conceptual design consists of a magnetically confined (tokamak) fusion reactor fitted with a fertile uranium blanket. Two fertile fuels, UC and UO_2 , were evaluated in various combinations with two tritium breeding fusile fuels, LiH and Li_2O . The fusion driver concept was based on an ignited plasma, with the near-ignition beam driven option maintained as a viable backup alternative. All concepts and parameters were selected on the basis that technical feasibility would be achieved by 1995 to assure a proven commercial operation in the early to mid-21st century. The reactor was designed to achieve good fissile fuel production, with electricity production being a second order priority. However, the resulting concepts that evolved were all excellent power producers which significantly improved the economic performance. Many of the subsystems described in the following section were based on the DTHR systems recently developed in the FY 78 studies.

2.1 DESCRIPTION OF REACTOR CONCEPT

The CTHR has a major radius of 6.0 m with a minor radius of 1.4 m. The elongated plasma has an average temperature of 17.3 keV based on a parabolic profile and the electron density-confinement time product is $13.4 \times 10^{19} \text{ sm}^{-3}$. The fusion power is 1200 MW and the average neutron wall loading is 2 MW/m^2 . The significant plasma parameters are listed in Table 2-1. The fusion driver was developed with the maximum toroidal field at the conductor restricted to 12 tesla and the neutral beam energy limited to 200 keV for negative ion beam systems. These were considered to be reasonable parameters for a first generation plant of a mature industry in the 21st century. A comparison of the DTHR and CTHR plasma parameters is presented in Table 2-1.

The plan view of the reactor is shown in Figure 2-1. The 14 TF coils are equally spaced with a bundle divertor plasma exhaust system located between one pair. The 6.0 m device permitted sufficient room for the bundle divertor without using a larger angle between TF coils as had been done on DTHR. The ripple was increased from 1% to 2% peak to peak when the TF coil number was reduced from 16 (DTHR concept) to 14 coils. This was felt to be more

TABLE 2-1
DTHR/CTHR FUSION DRIVER PARAMETERS COMPARISON

	DTHR (September, 1978)		CTHR (September, 1979)	
	Reference Concept Beam Driven	Test Objective Ignited	Reference Mode Ignited	Backup Mode Beam Driven
Major Radius, m (R_0)	5.2	5.2	6.0	6.0
Minor Radius, m (a)	1.2	1.2	1.4	1.4
Aspect Ratio (A)	4.33	4.33	4.3	4.3
Elongation (δ)	1.6	1.6	1.6	1.6
Plasma Volume, m ³ (V_p)	236.5	236.5	370	370
Toroidal Field on Axis, T (B_T)	5.47	5.45	5.2	5.2
Toroidal Field on Cond., T (B_{max})	11.69	11.66	12.0	12.0
Plasma Current, MA (I_p)	5.12	5.11	6.8	6.8
Neutral Beam Energy, keV (W_0)	175	185	200	200
Neutral Beam Power, MW (P_B)	150 ^(a)	55 ^(b)	80 ^(b)	200 ^(a)
Plasma Safety Factor, (q)	2.5	2.5	2.8	2.8
Plasma Energy Gain (Q)	6	Ignited	Ignited	5.8
Plasma Temperature, keV (Avg., $\bar{T}$)	17.3	17.3	17.3	17.3
Plasma Profile	Parabolic	Parabolic	Parabolic	Parabolic
Electron Density, 10 ²⁰ m ⁻³ (n_e)	0.99	1.06	0.97	0.90
Neutral Beam Penetration, 10 ²⁰ m ⁻² ($n_e a$)	1.19	1.27	1.77 ^(c)	1.64 ^(c)
Plasma Beta, % (β)	6.24	6.24	6.3	6.3
Poloidal Beta, % (β_p)	4.33	4.33	4.3	4.3
Wall Loading, MW/m ² (J_w)	2.0	2.0	2.0	2.0
Fusion Power, MW _t (P_f)	900	900	1200	1180
Required Confinement Time (τ_E), s	0.6	1.27	1.38	0.65
Empirical Confinement Time (τ_E), s	1.22	1.31	2.1	2.0
$n_e \tau_E$, 10 ¹⁹ m ⁻³ s	5.94	13.46	13.4	5.9
Plasma Pulse Duration, s	40-70	40-70	1200	1200
Plasma Duty Cycle Factor	0.5	0.5	0.9	0.9

(a) Steady State during pulse

(b) Start Up

(c) Includes 1.3 factor for near vertical path.

Figure 2-1. CTHR Plan View

conservative than 3 or 4% ripple being assumed by other designers which allowed 12 TF coil concepts. The massiveness of many of the 12 coil concepts was felt to be undesirable. The four neutral beam injectors (NBI) were located beneath the reactor to maximize access to the blanket for refueling. The vacuum pumps were located adjacent to the NBI positions.

Figure 2-2 shows the elevation view cross section of the CTHR concept. The major components are identified. The details of the major subsystems are discussed in the following subsections. The reactor layout considered two major servicing operations as the concept developed. The first was a major operation which was the TF coil removal and replacement. The second was the more routine servicing of the blanket sector removal during a refueling operation. The component locations, positionability and access were designed to achieve these operations. The CTHR radial dimensions and component thicknesses are presented in Table 2-2.

REACTOR SERVICING

The servicing of TF coils would normally be done in their operating position. However, if removal for major repairs or maintenance became necessary, a single coil or pair, could be removed from the reactor. The external reactor vacuum shell is paneled to permit removal for access to the TF coils. The PF coils beneath the TF coils are lowered, using hydraulic rams, into recesses in the floor. The outer PF coils and upper PF coils are lowered or raised to clear the TF coils. The shear panels have a large bolt midway between TF coil pairs and a flanged dewar to permit detachment from the adjacent TF coil. A rolling dolly assembly is positioned beneath the TF coil on plates covering the lowered PF coils. The weight of the TF coil is then taken by slightly raising the inner leg of the coil off the central support structure. The outer support post has built in rollers. The reactor shell dome would either be removed or access slots opened to permit attachment of a crane to stabilize the coil as it is rolled out from the reactor for servicing in an adjacent building. Removal of components inside the TF coil (shielding, blanket, vacuum vessel) is required if a single coil is removed. If a coil pair is removed

Figure 2-2. CTHR Elevation View

TABLE 2-2
CTHR RADIAL DIMENSIONS

<u>COMPONENT/LOCATION</u>	<u>THICKNESS, cm</u>	<u>RADIUS, m</u>
Device Centerline	0	0
Bore		1.05
Bucking Ring	65	1.70
OH Coils (Within Rings)	25	1.70
TF Coil (Nose)	100	2.70
Dewar/Insulation	10	2.80
Assembly Gap	10	2.90
Inner Shield	60	3.50
Assembly Gap	10	3.60
Inner Blanket	60	4.20
Assembly Gap	10	4.30
Vacuum Vessel Wall (1 cm)/Structure	10	4.40
Scrape Off Layer	20	4.60
Plasma Minor Radius, a	140	
Major Radius, R_0		6.00
Plasma Minor Radius, a	140	7.40
Scrape Off Layer	20	7.60
Vacuum Vessel Structure/Wall	10	7.70
Assembly Gap	25	7.95
Blanket (UC Concept)	77	8.72
Assembly Gap	43	9.15
Outer Shield	80	9.95
Assembly Gap	20	10.15
Dewar/Insulation	10	10.25
TF Coil	100	11.25
Beam Stiffener	45	11.70
Dewar/Insulation	10	11.80

then only partial disassembly from adjacent components is necessary. The Neutral Beam Injectors are flanged near the shield valve and at the vacuum vessel to permit disassembly from the vacuum vessel.

For blanket removal during refueling operations, a smaller panel is provided in the reactor shell. The outer PF coils are raised or lowered to provide access to the blanket sectors being removed. An external shield is moved up to enclose the area around the TF coil pair between which the blanket sectors will be removed. Radiation levels will not permit pulling the blanket out into the reactor building once the shields are removed. The outer shield is removed, coolant piping disconnected and decay heat removal lines attached. The enriched blanket sector is moved into the external shield chamber and the replacement blanket sector is installed. If the inner blanket area is also refueled, the vacuum vessel segment at that location must be unbolted and removed to allow access to the inner blanket. The TF coils have independent dewar (insulation) systems so that cryogenic conditions are maintained during blanket refueling. It is probable that a quarter or a third of the blanket sectors would be refueled annually. Detailed fuel management studies will be required to determine the optimum strategy.

2.1.1 TOROIDAL FIELD (TF) COIL CONCEPT AND SUPPORT STRUCTURE

From the preceding discussion of the reactor concept and two servicing considerations, it should be apparent that the TF coil design has a major impact on the reactor design and operation. Therefore, considerable design and analysis effort was invested in the TF coil and its support structures to provide a firm foundation for the CTHR concept. This section provides an overview of many of these design considerations and a description of the resulting TF coil concept and parameters that developed. A summary of the TF coil parameters is given in Table 2-3.

The nose cross section of the TF coil, which is a fully wedged concept, is shown in Figure 2-3. Cable placement considerations and the TF coil cross-section thickness were the controlling parameters in this selection.

TABLE 2-3
CTHR TOROIDAL FIELD COIL PARAMETERS

<u>PARAMETER</u>	<u>UNITS</u>	<u>VALUES</u>
Plasma Major Radius	m	6.0
Vertical Bore (Inner)	m	10.7
Radial Bore (Inner)	m	7.5
Central Field	T	5.2
Maximum Field	T	12
Number of Coils	—	14
Field Ripple (at Plasma Edge, Peak/Peak)	%	~ 2
Total Stored Energy	MJ	29,850
Stored Energy per Coil	MJ	2,132
Total Ampere Turns	MA-T	156
Ampere Turns per Coil	MA-T	11.1
Coil Cross-Section	m	1.2 x 0.745 x 1.0
Mean Coil Circumference	m	32
Superconductor/Stabilizer	—	Nb ₃ Sn/Cu
Conductor/Shape	—	Cable/Square
Current Per Turn	kA	20
Conductor Area per Coil (Sheath I.D. Basis)	cm ²	2232
Structure/Conductor Area Ratio	—	2.78
Coil Construction	—	Slotted Plates
Structural Material	—	Stainless Steel (310)
Cooling	—	Forced Flow Supercritical • Helium
Coil Tension (Max. per Coil)	MN	128
Centering Force (Max. per Coil)	MN/m	73
Coil Weight (Tonnes per Coil)		
Conductor		64
Structure		156
Dewar		<u>35</u>
		255
Charging Power Required	MW _e	33.2
He Refrigeration Power (All Coil Systems)	MW _e	31.1
N ₂ Refrigeration Power (All Coil Systems)	MW _e	1.8

Figure 2-3. CTHR TF Coil Nose Cross Section

However, concern for a faulted TF coil and the forces resulting on the remaining coils led to the inclusion of bucking rings in the CTHR concept. The rings are required with the compact "D" shaped TF coils to support the transition from the outer leg radius to the vertical inner leg. The bucking rings were slotted to permit placement of the ohmic heating coils nearer the plasma. The cross-sectional design of the coil plates was dictated by the combined stress from the tension forces and the centering forces. The resulting Nb₃Sn superconducting cable design is specified in Table 2-4.

With the cross section of the TF coil established to provide structural support for the cables, the rigidity of the overall multiplated coil was evaluated for out-of-plane forces. Shear panels were added at the top and bottom of each coil to restrain the overturning moment caused by these out-of-plane forces. The shear panels were located above and below the access area for the blanket sectors. It was then determined that the bending at the mid-plane of the outer TF coil leg was excessive as was the bowing at the top and bottom of each TF coil. Since the plates alone could not resist these deflections, a stiffner beam was added which wedged onto the plate stack, resulting in a rigid cross section. Not only did the beams eliminate the bending and bowing, but they also provided a means of reacting the twisting motion between the top and the bottom of the TF coils. A central torsion tube in the bore of the torus was used to transmit the torque and thus prevent this twisting. A few external supports from TF coils to the reactor floor could also help reduce the twisting moments. Figures 2-1 and 2-2 show the torque lugs attached to the stiffner beams (at the top and bottom) and the torsion tube.

Since the TF coils are cryogenic, it is necessary to minimize heat transfer to the reactor floor at the support positions for the coils. This was done by developing a heat resistant TF coil support post under each coil and a center column support within the cryo-cooled enclosure. The support posts were constructed of telescoping cylinders mounted on roller assemblies as shown in Figure 2-4. The center column support shown in Figure 2-5 minimized heat transfer by carrying the structural loads through slender U-bolts between lugs

TABLE 2-4

SPECIFICATIONS OF Nb_3Sn -Cu SUPERCONDUCTOR FOR THE CTHR TF COILS

Average Current Density*, A/cm^2	3333
S/C Operating Current Density, A/cm^2	62,500
Area of Conductor + He, cm^2	4.00
Superconductor Area**, cm^2	0.32
No. of Strands, Active (3 x 3 x 3 x 3 x 6)	486
Non-Active	81
Strand Diameter, cm	0.08
Conductor Hydraulic Diameter, cm	0.04
Conductor Void Fraction	0.3
No. of Slots per Bobbin	11 max.
No. of Conductors per Slot	2
Mean Circumference of Coil, m	32
Helium Flow Rate per Conductor, g/s	4.2
Helium Inlet Temperature, K	4.2
Helium Inlet Pressure, Atm	6

* Averaged over slot area (clearance, insulation, jacket, conductor, stabilizer and coolant) of 6 cm^2 .

**Copper to noncopper area ratio is 0.83.

Figure 2-4. CTHR TF Coil Support Post

All Dimensions in Meters

Figure 2-5. CTHR Center Column Support

and saddle clamps. The center column support, OH coils, bucking rings and TF coil nose sections were all contained within a common dewar system at the bore of the torus, while the outer TF coil legs, stiffner beams, shear panels and support posts were insulated in individual, form fitting dewar walls. All the upper TF coil cables feed to the common dewar where the transition to the copper busbar occurs. The lower TF coil cables feed through dewar tubes near the posts to a common location under the reactor floor level for transition to the copper busbars.

The preliminary sizing of the TF coils was achieved using the COAST (8) computer code. Detailed sizing and layout work then resulted in a fairly detailed TF coil concept. This concept was used as input data for a final magnetic force analysis to confirm the structural adequacy of the design.

The magnetic forces on the PF coils and TF coil systems were computed by the MAFCO code⁽⁹⁾ in sufficient detail to permit the mechanical design of support and restraining components. It was found that:

- The net overturning moment about the horizontal axis on the TF coils is not large.
- The twist moments about the vertical axis through the plasma axis are large.
- The tension along the circumference of the TF coils is not constant, which is consistent with compact D-shaped coils.
- The vertical forces on the SF coils are directed toward the midplane, while hoop stress causes the coils to expand.

2.1.2 POLOIDAL FIELD (PF) COIL CONCEPTS

The PF coils are located in three general areas: the coils in the bore of the torus; the coils above and below the plasma; and the coils on the outer periphery of the TF coils. The PF coils located outside of the tokamak bore have been grouped and assembled in common dewar systems. This provides several advantages:

- Reduces the space envelope required compared to individual dewars.
- Reduces the manufacturing cost by using common dewar walls, insulation, helium headers, etc.
- Easier handling for installation and removal.
- Better access for blanket refueling and TF coil removal by lifting or lowering a unitized assembly.

The majority of the OH coils (42 of 52) are located in the bore of the device in a vertical stack positioned in slots in the bucking rings as shown in Figure 2-6. The height of the stack "carries" the flux around the corner of the TF coils so that the next OH coil set can be located above and below the TF coils, outboard of the vertical leg. This enables a solid corner construction at the top and bottom of the TF coil vertical leg, which is required to transmit forces from the sharply curved portion of the D-shaped TF coils to the bucking rings. The OH coils in the bore are considerably less expensive to manufacture than the larger diameter outboard coils. Locating the OH coils in slots in the bucking rings kept the coils ~40 cm closer to the plasma and resulted in several million dollars cost savings due to reduced power supply requirements. Each OH coil is constructed of bobbin assemblies stacked radially to achieve the required coil thickness. The five SF coils in the bore are similarly constructed. Cables and cryo-coolant lines are carried through slots in the bucking rings as shown in Figure 2-6. The upper coil cables are fed up to the common dewar chamber where transition to copper busbar(s) occurs. The lower coil cables feed down and across to the center of the central support structure to pass below the reactor to that transition chamber. The cryo-coolant lines are fed by manifold lines between the bucking rings and the torsion tube.

The Nb_3Sn superconducting coil and cable parameters are shown in Tables 2-5 and 2-6. Figure 2-7 shows the typical cable cross-section for PF and TF coil cables. The PF coil locations and respective currents are listed in Table 2-7. A cross-section view of the upper and lower PF coil assembly is shown in Figure 2-8. The peripheral PF coil assemblies would be similar to the construction shown in Figure 2-8. The cross-sectional view of these coil

All Dimensions in Centimeters

Figure 2-6. CTHR Bucking Ring With OH/SF Central Bore Coils

TABLE 2-5
CTHR SUPERCONDUCTING OH COIL PARAMETERS

Design Maximum Field, Telsa	4.9
Inner Radius, m	1.36
Outer Radius, m	1.60
Total Ampere-Turns, MA-T	36.4
Superconductor/Stabilizer	Nb ₃ Sn/Cu
Conductor/Shape	Cable/Square
Current per Turn, kA	10.76
Average Current Density: Conductor Stabilizer, and Coolant, A/Cm ²	3000
Structural Material	Stainless Steel
Coils in Vertical Stack	42
Total Coils	52
Total Turns	3384
No. of Cooling Channels per Stack	10
Average Length of Cooling Channel, m	43 (Solenoid Coils)

TABLE 2-6

CTHR SUPERCONDUCTING SF COIL PARAMETERS

Design Maximum Field, Tesla	2
Total Ampere-Turns, MA-T	49.9
Superconductor/Stabilizer	Nb ₃ Sn/Cu
Current per Turn, kA	16.24
Average Current Density: Conductor, Stabilizer, and Coolant, A/cm ²	4500
Structure Material	Stainless Steel
Total Coils	17
Total Turns	3072

SPECIFICATIONS OF THE SUPERCONDUCTOR CABLE
FOR THE OH AND SF COILS

	<u>OH</u>	<u>SF</u>
Average Current Density*, A/cm ²	3000	4500
S/C Operating Current Density, kA/cm ²	400	720
Area of Conductor + He, cm ²	3.61	3.61
Superconductor Area, cm ²	0.0269	0.0226
Cu - to - non Cu Area Ratio	19.13	22.96
No. of Strands (3 x 3 x 3 x 3 x 7)	567	567
Strand Diameter, cm	0.069	0.069
Conductor Hydraulic Diameter, cm	0.083	0.083
Conductor Void Fraction	0.40	0.40
Helium Flow Rate per Conductor, g/s	2.5	2.5
Helium Inlet Temperature, K	4.2	4.2
Helium Inlet Pressure, atm	4	4

Figure 2-7. CTHR Nb₃Sn Cable Cross Sections

TABLE 2-7

CTHR PF COIL SYSTEMS: COIL LOCATIONS AND CURRENTS

LOCATIONS (a)						
Radius	Height	No.	OH Current,	SF Current,		
<u>R, m</u>	<u>Z, m</u>	<u>of Turns</u>	<u>MA-T</u>	<u>MA-T</u>		
1.148	on .25 centers;
Center Bore OH Coils	0.16 Coil +	60				
	0.09 Ring;					
	21 Coils			per Coil		
1.48	0.0 SF	40		-0.63		
1.48	0.25 SF	40		-0.79		
1.48	0.50 SF	40		-0.79		
2.7	7.0 SF	358	36.4 Total	5.73		
3.2	7.0 OH	96		—		
3.7	7.0 SF	358		5.73		
4.2	7.0 OH	96		—		
4.7	7.0 SF	358		5.73		
10.75	4.5 SF	108		-1.79		
10.75	4.0 OH	80		—		
11.06	3.5 OH	80		—		
11.25	-3.0 SF	121		-1.94		
11.32	-2.5 OH	80		—		
11.40	2.0 SF	133		-2.14		
TOTAL CURRENTS (OH AT 50% BIAS)			$I_{OH} = 36.4$	$\Sigma I_i = 49.9$		

INITIAL STORED ENERGY 580 MJ

TOTAL STORED ENERGY 4.06 GJ

(a) Upper and lower SF/OH coil sets external to the bore are at the same (but inverted) locations.

All Dimensions in Centimeters

Figure 2-8. CTHR Upper and Lower PF Coil Assembly Cross Section

assemblies shown in Figure 2-2 has been exaggerated radially to show the coils within the assemblies. The coils are actually closely adjacent to the outer edge of the TF coil stiffener beam dewars.

2.1.3 BUNDLE DIVERTOR CONCEPT

A reactor application of an improved bundle divertor was developed in detail for the DTHR and previously reported⁽⁵⁾. The divertor concept was fitted into the DTHR ($R_0 = 5.2$ m) by opening the angle between two adjacent TF coils from 22.5° to 30° and placing the remaining 14 coils on 22° spacings. The radius to the inner edge of the outer TF coil leg was 8.9 m. For CTHR it was found that the DTHR bundle divertor could be scaled up to fit the $R_0 = 6.0$ m tokamak and keep uniform spacing (25.7°) between TF coils if 14 TF coils were used instead of 16 as in DTHR; if the radius to the inner edge of the outer TF coil leg was at least 10.25 m; and if the angle of the divertor coils to the divertor centerline was reduced from 51° to 45° . The bundle divertor concept that resulted from these changes is shown in Figure 2-9. The parameters for the CTHR bundle divertor, listed in Table 2-8, were then developed by scale-up from the DTHR concept. The scale-up appears to be a viable concept; however, further analysis is required.

2.1.4 NEUTRAL BEAM INJECTION SYSTEM (NBIS)

The NBIS for DTHR was scaled-up from TFTR and used six beams for a total NB power of 150 MW to the plasma at an injection energy of 175 keV. The DTHR operating mode was beam driven with an ignition option. Since CTHR is an ignited plasma with a beam driven option, the reference NBIS is significantly lower in power. If the beams were injected on the horizontal axis, two NBI would suffice assuming a 200 keV negative ion injection system. However, to provide better access for blanket refueling, the NBIS was designed to inject at a nearly vertical angle from beneath the reactor as shown in Figure 2-2. To reduce the size of duct passing between the TF coils and thus keep the NBI as vertical as possible, four NBI were required. Table 2-9 provides the general parameters. The centerline of the beam intersects the plasma centerline at an angle of 10° off vertical and tilts 10° forward with

Figure 2-9. CTHR Bundle Divertor Plan View

TABLE 2-8
CTHR BUNDLE DIVERTOR COIL PARAMETERS

Maximum Field in Divertor Coils	10 T
Two Circular Divertor Coils* Current/Coil	15 MA
Two Vertical Auxiliary Coils* Current/Coil	6.4 MA
Divertor Coil Angle (α)	45
Mean Coil Diameter	2.65 m
Overall Width of the Coil	1.28 m
Superconductor Cable	2 cm x 2 cm Nb ₃ Sn/Cu
Cable Current	32 kA
Cable Current Density (Inside Sheath)	8 kA/cm ²
Average Current Density/Tesla Field Nulled	0.8 kA/cm ² T
Coil Structure	Stainless Steel Bobbins Constant Thickness
Minimum Shielding Space Between Conductor and Plasma	0.70 m
Helium Inlet Temperature	4.2 K
Helium Inlet Pressure	4.0 atm.
Conductor Void Fraction	0.35
Pumping Power (@ 50% Efficiency)	5 MW _e
Electrical Power Supply (30 min. Charging Time)	~ 600 kW

* Estimated values by scaling up from DTHR

TABLE 2-9
CTHR NBIS GENERAL PARAMETERS AT STARTUP

TOTAL NB POWER TO PLASMA, MW_t	80
INJECTION ENERGY, keV	200
NB PULSE LENGTH, s	$\lesssim 5$
NUMBER OF NB INJECTION ARMS	4
NB POWER TO PLASMA PER ARM, MW	20
NB DUCT DIMENSIONS, m	1.1 x 1.5
NB DUCT LENGTH, m	7
INJECTION ANGLE,	
WITH RESPECT TO PLASMA VERTICAL CENTERLINE	10°
WITH RESPECT TO PLASMA CURRENT PLANE	10°
CHARGE STATE OF SOURCE IONS	- 1
OVERALL SYSTEM EFFICIENCY, %	~ 40

respect to the direction of the plasma current flow. Since the injection path is longer through the plasma than for a horizontal injection path, the beam energy had to be increased from 175 keV in DTHR to 200 keV for CTHR.

The trade-off between two horizontal NBI's and four near-vertical NBI's was evaluated by determining the blanket volume lost at the NBI duct penetrations and the neutron current distribution at the respective locations. Figure 2-10 shows the CTHR neutron current distribution with respect to the inner blanket and outer blanket locations. It can be seen that 70% of the fusion neutrons pass through the outer blanket area. Figure 2-11 shows that at the midplane position (1.6 m from the plasma center) the neutron wall loading peaking factor is ~ 1.5 . Thus, the neutron wall loading is maximum (3 MW/m^2) where the horizontal NBI duct penetration would be. The near-vertical NBI intersects the plasma at $\sim 45 \text{ cm}$ outward from its center and therefore, the wall loading factor is only ~ 0.875 . The neutron wall loading at the near-vertical duct then is only 1.75 MW/m^2 . The lost neutron current is also considered on the wall opposite the NBI duct since protective armor required in these areas would significantly reduce neutron penetration to the fertile blanket behind them. The analysis concluded that the two horizontal NBI's would lose $\sim 1\%$ of the blanket volume and $\sim 0.9\%$ of the neutron current as compared to $\sim 2.5\%$ volume but only $\sim 0.8\%$ neutron current for the four near-vertical NBI's. The larger volume loss for the near-vertical system was attributed to the blanket sector being shortened to end just above the NBI duct and not continued for the small volume left below the duct. Therefore, the general blanket performance was not changed by the near-vertical NBI's, but access to the blanket sectors was greatly enhanced, which would make the assumed plant availability much more realistic for the near-vertical NBIS. In general, the volume available for the outer blanket and the neutron capture potential is reduced very little by the NBIS in comparison to the bundle divertor space envelope, which requires $\sim 12\%$ of the potential blanket volume and sees $\sim 8.4\%$ of the neutron current distribution, since it occupies the entire space between two TF coils (43° included angle-see Figure 2-9).

Figure 2-10. CTHR Neutron Current Distribution

Figure 2-11. Neutron Wall Load Distribution

2.1.5 VACUUM VESSEL CONCEPT

The vacuum vessel concept was developed only in enough detail to establish an effective wall thickness for the neutronic analyses of the blanket. Figure 2-12 shows the 28 segments, joined by bolted flanges and sealed with flexible metal O-rings. This concept of assembly is feasible when an external reactor vacuum shell is used since the vacuum vessel must maintain only a partial vacuum differential between the outside environment and the plasma. The flanges were designed to align with the space between blanket sectors at the inner blanket as shown in Figure 2-13. The thin walls of the vacuum vessel then coincide with the fueled portion of the blanket sector. An effective wall thickness of 0.5 cm was established for neutronic analysis based on the $\sqrt{1}$ cm diameter coolant tubes and a 0.4 cm thick shell. Further detailed design and analysis of the vacuum vessel is required.

2.1.6 BLANKET DESIGN CONCEPTS

A detailed report⁽¹⁰⁾ of the results of studies on two generic blanket design concepts for application in a Commercial Tokamak Hybrid Reactor (CTHR) has been prepared. Both blankets operate on the U-Pu cycle and were designed to achieve tritium self-sufficiency while maximizing the fissile fuel production within thermal-hydraulic and mechanical design constraints. The two blanket concepts that were evaluated were: 1) a UC fueled, stainless steel clad and structure, helium cooled blanket (UC-SS-He); and 2) a UO_2 fueled, zircaloy clad, stainless steel structure, boiling water cooled blanket (UO_2 -Zr-BW). The motivation for investigating the UO_2 -Zr-Bw blanket was that it utilized many features of proven light water fission reactor technology and also was similar to the DTHR concept studied previously (1,4). The UC-SS-He blanket concept offered the potential for improved neutronic performance relative to the UO_2 concept, but at the expense of employing more advanced, less well developed technology and materials. Two different tritium breeding media, Li_2O and LiH , were evaluated for use in both blanket concepts. The use of lead or graphite as a neutron multiplier or reflector was also considered.

Figure 2-12. CTHR Vacuum Vessel Concept

Figure 2-13. CTHR Blanket Plan View

A series of neutronics studies was performed to scope the various material zone thicknesses required for different geometric configurations with the constraint that the arrangement be self-sufficient in breeding tritium. The beginning of life tritium breeding ratio was set at ~ 1.1 . The neutronic performance was calculated for sufficient irradiation exposure time to achieve about 3% fissile fuel enrichment in the blanket. An important aspect of the analysis was to investigate the neutronic benefits of full blanket wall coverage in a tokamak fusion hybrid device. The plan view of the CTHR inner and outer blanket arrangements is shown in Figure 2-13. A schematic diagram of the geometry for the analytical model and the volume fractions assumed versus the initial mechanical concept values are shown in Figure 2-14.

The results of the comparison of full (inner and outer) fertile blanket coverage versus outer fertile blanket coverage only indicate that increases in fissile fuel production of up to about 20% would be possible with the full blanket coverage as shown in Figure 2-15. The difference occurs because of increased neutron multiplication with the inner fertile zone, but is somewhat smaller than might be expected because of the non-uniform tokamak neutron source distribution which results in about 70% of the D-T fusion neutrons being directed toward the outer blanket region (see Figure 2-10). Also, for an outer fertile blanket only, the inner region can be designed with a material such as lead which provides some additional neutron multiplication. Another factor in considering the benefits of fertile material in the inner region of a tokamak hybrid is a possible decrease in the plant availability due to increased reactor downtime to service and refuel the inner blanket. Figure 2-15 shows the reduced performance resulting from lower plant availabilities when vacuum vessel and inner blanket removal are taken into account. If this increased plant outage becomes too large, the neutronic benefit of using full blanket coverage will be negated. The lower plant availability values shown in Figure 2-15 are relative values based on preliminary analysis. Additional work on the design of the vacuum vessel, shields and blanket sectors is needed to permit a meaningful value for each scenario.

SCHEMATIC DIAGRAM OF THE ONE-DIMENSIONAL GEOMETRY USED IN THE NEUTRONIC ANALYSES

VOLUME FRACTION COMPARISON BETWEEN ANALYTICAL MODEL AND INITIAL MECHANICAL CONCEPTS

	NEUTRONIC ANALYSES	MECHANICAL CONCEPTS	
		UC	UO ₂
A. FERTILE ZONE			
Fertile Fuel (UC or UO ₂)	0.50	0.45	0.45
Clad/Structural (SST or Zr/SST)	0.20	0.22	0.08/0.11
Coolant (He or Boiling Water)	0.30	0.33	0.344
			0.016 (He in rods)
B. REFLECTOR ZONE			
Reflector (Pb)	0.70	0.77	0.77
Structure (SS-316)	0.10	0.12	0.12
Coolant (He)	0.20	0.11	0.11
C. FUSILE FUEL ZONE			
Fusile Breeding Fuel (LiH or Li ₂ O)	0.70	0.77	0.77
Clad/Structure (SST-316)	0.10	0.12	0.12
Coolant (He)	0.20	0.11	0.11

Figure 2-14. Blanket Analysis Modeling and Volume Fraction Comparison

Figure 2-15. Plant Availability/Blanket Coverage Effects on Net Fissile Production Rate

Other conclusions from the neutronic analyses indicated certain advantages in the use of LiH as compared to Li_2O for the tritium breeding material. A blanket using LiH can be thinner than one using Li_2O due to the excellent moderating properties of the hydrogen. Also, the same moderating effect with the LiH decreases parasitic absorption in the structure in the lithium material zone compared to the use of Li_2O . The characteristics of these two candidate tritium breeding materials are shown in Table 2-10. An investigation of the use of lead in the blanket indicated that significant benefits could result by using it both as a reflector behind the fertile zone and as a neutron multiplier in the inner region for the case of outer fertile blanket coverage only.

For each of the blankets a conceptual mechanical design was developed and the thermal-hydraulic performance was evaluated. High pressure coolants were used in the blankets to obtain high power conversion efficiency and a low pumping power requirement. The steam pressure at the turbine throttle valve for the UO_2 blanket is about 6.20 MPa (900 psia) and that for the UC blanket is about 8.62 MPa (1250 psia). With these pressures the present day nuclear turbine technology for economical commercial applications can be utilized. For the axial flow boiling water cooled blanket, the water density varies along the length of the blanket fuel. This causes uneven axial fissile production. This effect combined with neutron wall loading variations requires further study. Due to the hydrodynamic limitation, the water mixture quality at the blanket exit must be kept below 40%. Studies on the variation of the water coolant density indicate decreased fissile fuel production and increased thermal power production with increasing water density. It was determined that pumping power limitations restricted the length of flow in the fertile fueled front zone of the modules to 1.5 m for the UC helium cooled concept. For the UO_2 boiling water cooled concepts, the channel exit quality somewhat restricted the fertile fueled zone coolant path length to 2.5 m. These factors had a significant affect on the initial mechanical concepts.

The blanket mechanical concepts were designed with a separate, independent vacuum vessel (first wall) as a part of the CTHR design. The volume fractions used in the neutronic analyses were set as mechanical design goals. As Figure 2-14 shows, these volume fractions could not be achieved for the

TABLE 2-10

COMPARISON OF CANDIDATE CTHR TRITIUM
BREEDING COMPOUNDS

<u>COMPOUND</u>	<u>ADVANTAGES</u>	<u>DISADVANTAGES</u>
LiH	<ul style="list-style-type: none"> • THIN BREEDING REGION • NO REFLECTOR NEEDED FOR T BREEDING REGION • Pb EFFECTIVE AS NEUTRON MULTIPLIER OR REFLECTOR FOR FERTILE REGION • SLIGHTLY BETTER NEUTRONIC PERFORMANCE THAN Li_2O • AFTER HEATING TO REMOVE Li, GASEOUS SPECIES CAN BE SEPARATED BY SAME TRITIUM SYSTEM REQUIRED FOR PLASMA EXHAUST 	<ul style="list-style-type: none"> • IN-SITU TRITIUM RECOVERY UNLIKELY • BATCH RECOVERY COMPLICATES REFUELING AND LIMITS RESIDENCE TIME TO AVOID EXCESSIVE TRITIUM DECAY • DISSOCIATION OF LiH MAY BE A PROBLEM UNLESS LOW TEMPERATURES CAN BE ASSURED • HIGH CUMULATED T INVENTORY BEFORE REFUELING
Li_2O	<ul style="list-style-type: none"> • STABLE COMPOUND WITH HIGH TEMPERATURE CAPABILITY (TO 800°C) • HIGH Li NUMBER DENSITY • IN-SITU TRITIUM RECOVERY BY DIFFUSION TO HELIUM COOLANT MAY BE POSSIBLE 	<ul style="list-style-type: none"> • THICK BREEDING REGION • REFLECTOR NEEDED TO MINIMIZE LEAKAGE THRU BLANKET • SOMEWHAT POORER NEUTRONIC PERFORMANCE THAN LiH

fertile zone on the first iteration. However, the calculated fuel volumes and weights in the mechanical concepts were all within 4% of the fuel weights used in the neutronic analyses. Thus, the performance of the mechanical concepts appropriately achieve the analyzed performance. The initial tritium breeding ratio of 1.1 appeared to be conservative and was reduced slightly (to ~ 1.05 at BOL) to permit increasing the fission events per fusion neutron.

UC HELIUM COOLED BLANKET CONCEPT

The UC-LiH-He cooled blanket concept that evolved is shown in Figure 2-16. The individual blanket module consists of a cylinder of UC fuel with stainless steel coolant tubes passing through the fertile fuel. The fuel region located towards the plasma has a tight coolant tube lattice which is necessary to remove the high rate of heat generation associated with that zone. The tube pitch becomes relaxed radially deeper into the UC fuel region due to the exponential power density distribution through the fuel with the peak located at the front wall and rapidly falling off radially from the front zone. The cylindrical UC fuel section is contained in a canister type module that has an individually cooled stainless steel pressure vessel wall - which is necessary to keep the structural temperature below the allowable limit of 500°C . The canisters are oriented in the circumferential direction around the plasma with the inlet and outlet manifolds running in the vertical direction. The canister vessel has hemispherical end caps with inlet/outlet flow channels that are fed from and return to the coolant manifold as shown in Figure 2-17. The intent is to have the fertile fuel canister designed with mechanical seals on the hemispherical end caps. This will provide a non-destructive means of extracting the bred fissile fuel as an assembly. The fuel assemblies will have a variable length for each specific location. This is due to the decreasing circumferential envelope associated with the torus geometry as the major torus radius is decreased. The fertile fuel canister is illustrated in Figure 2-18.

Mechanically the assembled fuel canister manifold section is attached to the Pb-LiH blanket. The Pb-LiH blanket structure is designed to support the fertile blanket. The support is mechanical and does not constrain the thermal

Figure 2-16. CTHR Helium Cooled UC Blanket Concept

A = 1.26 m MAX, OUTER BLANKET
1.67 m MAX, INNER BLANKET

B = 1.06 m MAX, OUTER BLANKET
1.50 m MAX, INNER BLANKET

Figure 2-17. CTHR Helium Cooled UC Blanket Cross Section

Figure 2-18. CTHR Fertile Fuel Canister

growth due to the elevated temperature operation of the blanket zone. This attachment scheme is illustrated in Figure 2-19. This approach allows replacement of the more highly irradiated fertile fuel zone and reuse of the reflector/fusile fuel zone. The fertile/fissile blanket is rotated off the Pb-LiH support structure for removal.

When the complete blanket is assembled the cross section at the upper and lower section of the blanket module is illustrated in Figure 2-17. The power densities associated with the Pb and LiH zones are such that the vertical flow path can be the entire sector length. The maximum temperature for the Pb section is 300°C. The LiH zone can be operated somewhat hotter, but is still cooled well below the structural material temperature limit of 500°C. The stainless steel coolant pressure tubes passing through the Pb and LiH zones are the primary pressure boundary, while the outer containing housing serves the purpose of a clad, rather than as a pressure boundary.

The outer blanket sectors have a radius of curvature of 3.1 m and associated poloidal length of 8 m. Approximately 70% of the neutron population can be intercepted by the outer blanket modules. The inner blanket modules are similar in cross-section design. However, this section has less curvature and can intercept an additional 28% of the plasma neutrons. Thus, the total source neutron capture potential in a full blanket is 98%, losing 1% at the top and bottom for the vacuum vessel support structure. An additional 9 to 10% of the neutron capture potential is lost because of the bundle divertor and NBIS penetrations.

The blanket module is designed so that maintenance and replacement can be accommodated by remote operations. The sectional geometry allows individual sectors to be removed by simple linear motions to withdraw it from between two TF coils. There are three blanket sectors between each pair of TF coils. Thus, the center-outer sector can be directly removed, while the two adjacent outer sectors have to be circumferentially rotated to the unobstructed center area between TF coils to be removed. Figure 2-16 shows the TF coil side of the blanket sector with the remote handling attachments on the outer blanket structure. These two attachments allow remote handling of the complete blanket sector.

Figure 2-19: CTHR Fertile Fuel Assembly Attachment Concept.

The UC/LiH blanket has a number of advantages and attractive features as follows:

- 1) The fertile blanket zone has the versatility of being readily detached from the fusile blanket and support structure.
- 2) The internal fuel assemblies are fabricated as a single unit which slides into the containing canister. Thus, to refuel the assembly, the hemispherical end cap is removed, the bred fuel assembly removed and a new assembly inserted.
- 3) It is conceivable that the LiH fusile blanket zone could be designed for in-situ tritium recovery by using higher temperature structural materials from that used in the fertile blanket.

A disadvantage of the UC concept is its mechanical complexity and the fuel manufacturing developmental needs. To fully evaluate this blanket concept, it is necessary to continue the design effort with in-depth structural and thermal-hydraulic analysis and development of serviceability methods to determine if this design is viable. Based on the present level of analysis, the concept appears to be attractive.

UO₂ BOILING WATER COOLED BLANKET CONCEPT

The basic blanket design is similar in cross section to that of the DTHR blanket design⁽⁴⁾. The structural material is 316 stainless steel. The fertile zone fuel consists of UO₂ pellets clad with Zircaloy-4. The reflector zone is helium cooled Pb using stainless steel tubes passing through the Pb medium. The fusile zone is composed of LiH or Li₂O as the tritium breeding fuel, which is cooled by helium inside stainless steel tubes passing through the medium. Clad rods could be used with a chambered coolant housing as an alternate design with some sacrifice in fuel volume fraction. Scoping neutronic calculations were performed to assess the blanket material thickness combinations necessary to achieve the projected blanket performance.

The blanket cross section is illustrated in Figure 2-20. The rear section is rectangular in cross section to house the Pb reflector and fusile breeding material. The front section of the module is a split cylindrical tube containing vertical fuel rods 2.5 m long, cooled by two-phase boiling water in an axial upward flow arrangement. The reflector and fusile blanket zones are cooled axially with upward flowing helium. These individual zones consist of a block of Pb and LiH or Li_2O respectively with stainless steel coolant tubes passing through the materials. The Pb has a maximum temperature of 300°C , which is below the melting temperature of the material. The Li_2O concept could use vented, clad rods, which would permit in-situ tritium removal via the helium coolant.

Since the Pb zone operates at lower temperatures than the fertile blanket section, it is necessary to provide insulation between the two zones. The proposed method of insulation is a double stainless steel wall. The two walls are rib stiffened and sized to withstand the coolant pressure acting across the surface of the wall. Insulating materials may also be required to fill the space between the two walls. The Pb reflector and fusile fuel zones are contained in a stainless steel housing. The coolant tubes are the pressure boundary for the coolant. Thus the Pb, LiH (or Li_2O) stainless steel housing is not sized to be a primary pressure boundary.

The subcooled water reaching the fertile fuel plenum must be preheated in the inner shield, vacuum vessel or by some other heat source. From there the water passes through an orifice region that lowers the coolant operating pressure so the coolant flashes to a low quality mixture. The wet steam then passes axially through the fertile blanket zone and exits as wet steam having a quality of 30 to 40%. Due to the efficient coolant properties of boiling water, the fuel rods can be effectively cooled in the reference tight lattice. The proposed fuel rod lattice is a triangular array with pitch to diameter ratio of 1.1. A schematic diagram of the coolant flow paths is shown in Figure 2-21.

Four individual modules are bundled together toroidally with three bundled vertical sets secured together by a strongback structural framework as shown in Figure 2-22. This outer blanket occupies the poloidal region which

Figure 2-20. CTHR Dual Coolant UO₂ Blanket Concept Cross Section

Figure 2-21. CTHR Dual Coolant Blanket Concept

Figure 2-22. CTHR Dual Coolant Blanket Concept Sector.

intercepts approximately 70% of the fusion neutron population. These integral module sections are configured such that they can be installed and removed from the reactor as a unit by remote operation. The strongback holding the individual modules together will have a built-in lifting member for handling the complete module sector. The module length is slightly shorter for the three vertical sets of modules bundled together to occupy the inner blanket region. An inner blanket sector has less curvature than the outer blanket sector .

The boiling water cooled blanket module appears to be viable based on the level of analysis performed to date. However, the blanket fissile production is extremely sensitive to the water density. The blanket coolant exit conditions close to that required for a commercial power conversion system were developed with reasonable performance still possible. The fertile region requires conventional technology in regards to fuel and fuel form. The fusile and reflector zones are not state-of-the-art technology; however, they do not appear to involve unreasonable development schemes. The refueling requires individual sector and module replacement, which directly affect economics because of the plant availability factor.

An alternate concept using a common coolant system is shown in Figures 2-23 and 2-24. The helium is replaced with subcooled water fed to and from the fusile blanket which preheats the coolant to reach saturation temperature before it enters the fertile blanket. The compatibility of LiH or Li₂O with water coolant must be studied further to assure the safety of this concept. However, the single coolant concept would significantly reduce the coolant manifold and headering space requirements. Performance of the tritium breeding zone is very similar with either coolant when LiH is used. However, using water coolant with Li₂O could improve it's performance with respect to the LiH performance. Neutronic analysis of these alternatives was not conducted.

BLANKET PERFORMANCE SUMMARY

A list of important performance parameters comparing two blanket concepts (UC-LiH-He cooled to UO₂-boiling water-Li₂O-He cooled) with full blanket

Figure 2-23. CTHR Wet Steam Cooled Blanket Module Concept

Figure 2-24. CTHR Wet Steam Cooled Blanket Sector Concept

coverage using lead reflector regions is presented in Table 2-11. The values given are the average over a four year irradiation period, or 4.8 MW-yr-m^{-2} of exposure which was the approximate time for each concept to achieve ~ 3 w/o fissile enrichment. Both concepts are self-sufficient in tritium production (TBR ~ 1.1) and produce about the same thermal power (4000 to 5000 MW BOL to 6000 to 8000 MW EOL). However, the UC helium cooled blanket produces about 75% more fissile ^{239}Pu fuel due to both the increased uranium density with the UC compared to the UO_2 and also because of the harder neutron spectrum with the helium compared to the boiling water. In both blankets the power production increases by approximately 70% over the four year irradiation cycle. Means of levelizing the thermal power requires further study once a reference concept is selected. The arrangement of the outer blanket in quadrants affords flexibility in fuel management strategies to effect this power leveling. The fissile production performance and average blanket thermal power versus irradiation time for four blanket concepts used in the economic analysis are shown in Figure 2-25 and 2-26. The thermal power for an equilibrium cycle was estimated for each concept to be approximately equal to the two year average performance.

Economics will be a key factor in the determination as to which of these blankets is most desirable for use in the CTHR, i.e., minimizing the cost of electricity for the total system of the hybrid and the fission reactors it supports. As will be noted in the economic analyses in Section 2.4 of this report, the hybrid power production is a much more sensitive factor on busbar electricity costs than is fissile fuel production and therefore, the $\text{UO}_2\text{-Li}_2\text{O}$ concept was less than 2 mills/kWh higher than the UC-LiH concept, even though fissile production rates were quite different. In addition, material lifetime and radiation damage considerations may be important factors. Table 2-12 shows typical radiation rates at the outer surface of the vacuum vessel, adjacent to the blanket front wall. While most of the analyses were for a four year irradiation period, sufficient to reach $\sim 3\%$ enrichment in the blanket, economics studies will be required to determine the optimum exposure time. It should be noted that the results represent an initial assessment. Further iterations between neutronic, thermal-hydraulic, structural, and mechanical design are required to arrive at an optimum design configuration. Additional considerations such as fuel shuffling and operation

TABLE 2-11

SUMMARY OF IMPORTANT PERFORMANCE PARAMETERS FOR
TWO BLANKET CONCEPTS WITH FULL BLANKET WALL COVERAGE

	<u>UO₂ Fueled^(a)</u>	<u>UC Fueled</u>
Fertile Material	UO ₂	UC
Clad Material	Zr-4	SS-316
Structural Material	SS-316 ^(b)	SS-316
Coolant (Fertile/Fusile)	Boiling Water/He	He/He
T-Breeding Material	Li ₂ O	LiH
Reflector Material	Pb	Pb
T, Tritium Breeding Ratio ^(c)	1.13	1.20
F, $\frac{\text{Net } ^{239}\text{Pu Atoms}}{\text{Fusion Neutron}}$ ^(c)	0.54	0.97
M, Blanket Energy Multiplication ^(c)	7.5	7.9
Fissile Production, $\frac{\text{kg } ^{239}\text{Pu}}{\text{yr}}$ ^(c)	1524 (1590) ^(d)	2730 (2868)
Thermal Power, MW _t ^(c)	5840 (5944)	6100 (6266)
Fissile Enrichment After Four Years	2.96%	2.89%
Blanket Inventory, Tonnes U + Pu	255	410
Max. Fuel Temperature, °C	810	660
Max. Clad Temperature, °C	325	550
Max. Pressure Vessel Wall Temperature, °C	500	500
Coolant Inlet Pressure, psia	1000	1250
Coolant Outlet Temperature, °C	280	450
Power Conversion Turbine Inlet Pressure, psia	900	1250
Gross Power Output, MW _e ^(c)	1990 (2021)	2080 (2192)

(a) Values for two phase water at 30-40% exit steam quality

(b) Although Zr was assumed in the neutronic analysis for simplicity.

(c) Average over four year irradiation time.

(d) Bracketed values are from the COAST computer code and were used for economic analyses of these concepts. COAST analyzes the actual blanket geometry while the neutronic data assumed an elliptical blanket shape. COAST also includes high grade heat sources such as the inner shield and reduces electric power (available for sale) by subtracting out the hybrid recirculating power.

Figure 2-25. CTHR Blanket Concepts - Net Fissile Production Comparison

Figure 2-26. CTHR Blanket Concepts - Thermal Power Comparison

TABLE 2-12
RADIATION DAMAGE RATES IN THE SST VACUUM VESSEL WALL OUTER SURFACE

	<u>UC-LiH</u> <u>Blanket Concept</u>		<u>UO₂-Li₂O</u> <u>Blanket Concept</u>	
	<u>At BOL</u>	<u>After 4 Yrs</u>	<u>At BOL</u>	<u>After 4 Yrs</u>
Integrated Wall Loading (MW-yrs/m ²)	---	4.8	---	4.8
Atomic Displacement Rate, 10 ⁻⁷ DPA ^(a) /s	6.90	7.76	6.58	7.32
Cumulative DPA	---	92	---	88
He Production Rate, 10 ⁻⁶ appm ^(b) He/s	7.39	7.40	7.39	7.41
Cumulative appm He	---	930	---	934
Fluence, E > 0.1 MeV (n/cm ²)	---	1.14 x 10 ²³	---	9.60 x 10 ²²

(a) DPA - displacements per atom

(b) appm - atomic parts per million

on an equilibrium cycle will be important. Such blanket fuel management studies will be a key area for future work on the CTHR blanket design. Overall, both of these generic blanket concepts appear feasible and attractive for use in a tokamak fusion hybrid reactor.

2.2 BLANKET COOLING AND POWER CONVERSION SYSTEM CONSIDERATIONS

Two blanket cooling system concepts have been considered for the CTHR; one for the UC fertile fuel blanket is cooled by helium gas and the other for the UO_2 fuel rods is cooled by boiling water. In order to provide sufficient power conversion capability for commercial applications and still maintain a low pumping power requirement, high pressure coolant was utilized. In the helium cooled concept the helium flows inside circular stainless steel pressure tubes with the carbide fuel filling the space between the tubes. Each module consists of coolant tubes which are contained inside a double-wall cylinder. The outer wall cylinders are cooled by helium through the annulus. The cylinder is oriented along the toroidal direction with the inlet and outlet headers running along the poloidal direction. For the water-cooled concept the UO_2 pellets are clad with zircaloy tubes. The rod assembly is arranged in a tight triangular array with the water flowing along the length of the rods. The rods and water are contained inside a semi-circular cylindrical vessel. The lead neutron reflector/multiplier and the fusile fuel (tritium breeding) zones are placed behind the fertile fuel zone. The fusile zones are cooled by helium.

2.2.1 HELIUM COOLED UC BLANKET CONCEPT

The advantage of having the coolant flowing inside the pressure tubes is that if the thermal conductivity of the fuel material is high, the tubes can be placed far apart without resulting in a high temperature difference between the tube wall and the peak fuel temperatures. This also results in a high fuel volume fraction if the local nuclear heating rate is not extremely high.

In this hybrid blanket application, the stainless steel clad (coolant tube) temperature was limited to 550°C for stress corrosion and fatigue

considerations. This temperature limit is much lower than the fuel material temperature limit (approximately half of its melting temperature). The thermal design objective is therefore, to establish the tube size, length of channels and tubes pitch to tube outer diameter ratio (P/D), so that the tube wall temperature limit is not exceeded and the pumping power requirement is minimized.

A parametric study was performed using a steady state, coupled heat conduction and hydraulic calculation. Assuming a triangular array, the tube wall and peak fuel temperatures and the coolant pressure and temperature distributions along a channel of a given length were computed at various heat generation rates and P/D ratios. Preliminary calculations indicated that a tube OD of 1.0 cm and a channel length of 1.5 m are about the optimum values. The helium coolant inlet conditions selected were: $P_{in} = 8.62$ MPa (1250 psia) and $T_{in} = 280^{\circ}\text{C}$. Since the coolant outlet temperature is limited by the maximum allowable pressure containment wall temperature, the coolant inlet temperature was selected to be as high as practical, so that an efficient power conversion system can be designed. Because of the rapidly decreasing radial power distribution in the blanket zone, the coolant tube spacing could be increased radially as the power density decreases by changing the location of the tube where it welds into the tube sheet. A parametric analysis was performed to determine the pumping power requirements and coolant exit temperatures as a function of power density and tube pitch to diameter ratio. The results are shown in Figure 2-27. With the power density distribution for this design as shown in Figure 2-28, the tubes spacing range would be between 1.2 to 1.8. The helium mixed mean temperature during the plasma-on time is about 478°C . The pumping power requirements are about 2% of the blanket thermal power. In this preliminary design, two pitch to diameter ratios were used: at the plasma side zone, $P/D = 1.2$ and at the outer zone, $P/D = 1.6$.

The solid fuel volume fraction (excluding the space occupied by the coolant tubes) in a unit triangular array at various P/D ratios is shown in Figure 2-29. With a tube OD to the tube wall thickness ratio (α) equal to 25 for this particular canister design, the unit cell fuel volume fraction is 0.53 at $P/D = 1.2$ and 0.70 at $P/D = 1.6$. The average value would be much higher than 0.50. However, when the other structural material and cooling

Figure 2-27. UC-LiH-He Cooled Blanket Concept Coolant Parameters

Figure 2-28. UC-LiH-He Cooled Blanket Concept Power Density Distribution

Figure 2-29. UC-LiH-He Cooled Blanket Fertile Fuel Zone Parametrics

annulus of the canister are included, the overall fuel volume fraction is about 0.45. The peak fuel temperatures at the various heating rates and pitch-to-diameter ratios are shown in Figure 2-30. The peak fuel temperature is about 660°C.

Since the plasma duty cycle for the CTHR is 90%, with a proper design of a thermal energy buffer-storage system, the blanket coolant exit temperature averaged over one complete burn cycle can be assumed to be 95% of the value during the plasma-on period. The steady state blanket exit temperature is therefore about 450°C. This temperature was used as the design value for the power conversion system.

With the helium gas operating between 450°C and 280°C, the steam generator characteristics are as shown in Figure 2-31. In the figure the water/steam temperature is plotted against the water enthalpy showing the regions of the economizer, evaporator and superheater. The helium temperature line in the primary side of the steam generator is also shown in the figure. In this study the pinch point temperature difference was taken to be 26°C (50°F) and the feedwater inlet temperature to the steam generator was taken to be 150°C (300°F). Considering six feedwater heaters and an 80% turbine stage efficiency, the estimated gross thermal efficiency for the cycle is 37% and the net cycle thermal efficiency was estimated to be 35%. The blanket-turbine operating conditions are summarized in Table 2-13. A schematic diagram of the power conversion system is shown in Figure 2-32. Five primary loops are included; one for the inner blanket and four for the outer blanket. This maintains reasonably sized coolant headers, provides for better safety through five semi-independent decay heat removal systems, and permits refueling of the blanket by quadrants independent of the other three outer blanket loops. In-situ tritium extraction could be added to this coolant system if: 1) higher temperature materials were used for the clad and structure to permit "driving-off" the hydrogen to get the T from the DT trapped in the LH fusile fuel; 2) the fusile fuel was Li₂O. More in-depth studies are required to determine the feasibility of these alternatives.

COOLANT TUBE OD = 1.0 cm (.04 cm WALL)
CHANNEL LENGTH = 1.5 m
OPERATING AT MAX. TUBE WALL TEMP. OF 550°C
HELIUM INLET PRESSURE = 8.62 MPA

Figure 2-30. UC-LiH-He Cooled Blanket Concept Peak Fuel Temperature

Figure 2-31. UC-LiH-He Cooled Blanket Steam Generator Characteristics

TABLE 2-13

CTHR BLANKET/TURBINE OPERATING CONDITIONS

BLANKET PARAMETERS

FERTILE FUEL	UO ₂	UC
MAX. FUEL TEMP., °C	810	660
CLAD MATERIAL	Zr-4	316 SST
MAX. CLAD MATERIAL TEMP., °C	325	550
STRUCTURAL MATERIAL	316 SST	316 SST
COOLANT	2 PHASE WATER ^(a)	He
AVERAGE COOLANT INLET PRESSURE, psia	1000	1250
AVERAGE COOLANT OUTLET TEMP., °C	280	450 ^(b)
ΔT THROUGH BLANKET, °C	—	198
MAX. COOLANT CHANNEL LENGTH, m	2.5	1.5
PUMPING POWER TO THERMAL POWER RATIO, %	0.5	2.0
AVERAGE THERMAL POWER, MW _t	4000-6000	5000-7000

TURBINE PARAMETERS

STEAM CYCLE	DIRECT	INDIRECT
PRESSURE, psia	900 ^(c)	1250 ^(c)
TEMPERATURE, °C (SUPERHEATED °C)	278 (o)	420 (120)
FEEDWATER TEMPERATURE, °C	175	150
SIZE, MW _e	900	1000
NUMBER OF UNITS IN PARALLEL	2 ^(d)	2 ^(d)
GROSS THERMAL EFFICIENCY, %	35	37
ESTIMATED NET CYCLE EFFICIENCY, %	34	35

(a) 3.5% inlet quality, ~ 35% exit quality for flow stability.

(b) Limit due to low film coefficient of coolant and structural material limitations. Effect of plasma-off included.

(c) Steam pressure at steam separation exit for direct cycle or at SG-superheater exit for indirect cycle.

(d) Tandem compound units with 6 feedwater heaters and 80% stage efficiency.

Figure 2-32. UC-LiH-He Cooled Blanket Concept Indirect Power Conversion System Schematic

2.2.2 BOILING WATER COOLED UO_2 BLANKET CONCEPT

In a water cooled blanket the mass density of the water has a profound effect on the neutronic performance of the blanket. Dry steam cooling was considered, but had high pumping power (over 50% of the blanket thermal power due to the low operating temperature limit for the zircaloy clad material) and there was difficulty in producing high pressure steam outside the blanket, in spite of the low density of the steam. High quality mixtures are not desirable because of the problems in flow stability and dryout at high heat fluxes. The critical heat flux for two-phase water flow decreases rapidly for a channel exit quality over 40-50%. The coolant exit quality of a typical boiling water reactor fuel channel is less than 40%. Test data^(11, 12) showed that at a pressure of 6.89 MPa (1000 psia) and a channel length to diameter ratio (L/D) of 150, the critical heat flux is about 120-150 W/cm² when the channel exit quality is $\sim$ 35%. The critical heat flux is slightly higher for a given exit quality if the coolant inlet condition is two-phase water rather than subcooled water. Therefore, the thermal design objective using boiling water cooling is to establish an operating condition so that (1) the mixture exit quality is below 40%; (2) the mass weighted coolant average density in the blanket is as low as practical; (3) the clad temperature is within the allowable limit of 325°C for the zircaloy; and (4) the coolant pressure is as high as practical so that a useful power conversion system for commercial application can be utilized. Due to these conflicting requirements the neutronic performance of the blanket would have to be compromised due to the high volume fraction of structural material needed to support the high coolant pressure and the less than ideal density of a low quality water mixture.

Two coolant inlet conditions were investigated. For one, the water at the blanket inlet is subcooled. In the other, the water is two-phase with a quality about 3.5% at the blanket inlet. Because of the 325°C temperature limit for the zircaloy clad, the saturation temperature of the cooling water must be somewhat below this temperature to account for the clad-water film temperature drop. The highest water pressure that can be used would be about 6.89 MPa (1000 psia). This pressure was then selected as the reference system

pressure at the blanket inlet. The thermal performance of the cooling system at the fertile blanket region for the two inlet conditions mentioned above are compared as shown in Table 2-14. For each inlet condition, two heat generation rates were analyzed: 300 W/cm^3 representing a peak power rod in the blanket and 200 W/cm^3 representing an average region of the blanket.

As seen from Table 2-14, the system pressure drop, fuel and clad temperatures and the pumping power requirement are comparable for the two different inlet conditions. The only significant difference is the mass weighted coolant density in the blanket. With subcooling at the inlet, the coolant density is about twice that with two-phase mixture at the inlet. Since neutronic performance requires low water density, the two-phase inlet condition was selected as the reference condition for this blanket design concept. The coolant quality, coolant, clad and fuel temperatures and density distributions at the reference operating conditions are shown in Figures 2-33 and 2-34.

The peak heat flux in this blanket concept is about 88 W/cm^2 as shown in Table 2-14. This is well below the critical heat flux of about 120 and 150 W/cm^2 mentioned previously. The operating conditions selected should therefore satisfy the flow stability and dryout requirements.

In this blanket concept the lead reflector and the fusile breeding zones are behind the fertile blanket region and are cooled by helium. However, water may be feasible to cool the reflector and the breeding zone in this design arrangement, if compatability with Li compounds is shown to be satisfactory. In a helium cooled option, separate water (to cool the fertile zone) and helium (to cool the reflector and breeding zone) cooling circuits are needed. If they are all cooled by water, a single cooling circuit is sufficient. In either case, saturated water is supplied to the fertile zones inlet plenum at approximately 7.9 MPa (1150 psia). The water then flashes to a quality of about 3.5% across an orifice plate before entering the fertile blanket at 6.89 MPa (1000 psia). The pressure loss across the orifice plate is about 1 MPa (150 psia) in order to obtain the required inlet quality.

For power conversion, a direct steam turbine cycle could be utilized. A steam separator is required to separate the moisture and to supply steam for the

TABLE 2-14

THERMAL PERFORMANCE OF THE UO_2
BLANKET COOLED BY BOILING WATER
(Rod OD = 1.45 cm, P/D = 1.10, Zircaloy Clad)

CASE	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>
Water Condition @ Channel Inlet	<u>Subcooled</u>		<u>Two-Phase</u>	
Heat Generation Rate, Q , W/cm^3	300	200	300	200
Heated Length, m	2.5	2.5	2.5	2.5
Inlet Pressure, MPa (psia)	6.89 (1000)		6.89 (1000)	
Inlet Temperature, $^{\circ}\text{C}$	280	280	285	285
Exit Temperature, $^{\circ}\text{C}$	285	285	285	285
Pressure Drop, MPa (psi)	.04 (58.7)	.04 (58.7)	.042(61.5)	.042(61.5)
Peak Heat Flux, W/cm^2	87.6	58.7	87.6	58.7
Peak Clad Temperature, $^{\circ}\text{C}$	320	307	310	302
Peak Fuel Temperature, $^{\circ}\text{C}$	817	618	809	613
Inlet Density, kg/m^3	749.6	749.6	435.4	435.4
Exit Density, kg/m^3	99.8	168.5	83.5	126.7
Mass-Weighted Coolant Average Density, kg/m^3 (lb/ft ³)	372.7 (23.2)	441.4 (27.5)	216.6 (13.5)	252.9 (15.7)
Inlet Quality, Steam Weight %	0.0	0.0	3.5	3.5
Exit Quality, Steam Weight %	30.7	16.0	38.0	23.0
Pumping Power, % of Thermal Power ^(a)	0.23	0.44	0.40	0.70
Coolant Mass Flow Velocity, kg/sec-cm^2	0.38	0.50	0.36	0.43

(a) Twice the Value Through the Heated Length

Figure 2-33. UO_2 Boiling Water Cooled Blanket Concept Temperature and Steam Quality Conditions.

Figure 2-34. UO_2 Boiling Water Cooled Blanket Concept Coolant Density Distribution

turbine from the blanket. With a blanket inlet pressure of 6.89 MPa (1000 psia) and a blanket pressure drop of 0.42 MPa (61.5 psia), a steam pressure of 6.2 MPa (900 psia) at the steam separator exit could be achieved. This steam supply pressure is within the current nuclear steam turbine technology. With a reheat cycle and sufficiently high feedwater temperature, a net cycle efficiency of 34% could be obtained. The blanket and turbine operating conditions were summarized in Table 2-13. A schematic of the power conversion system is shown in Figure 2-35, which includes in-situ recovery of the tritium from the helium cooled fusile fuel zones.

2.3 IDENTIFICATION OF ASSUMPTIONS

The guidelines for the concept and parameter selections for the CTHR fusion driver and blanket were based on relatively near-term technology, attempting to minimize development requirements. Thus, if the proof-of-principle appears feasible by 1995, the concept should be fully developed for a commercial application in 2020 to 2030. The hybrid reactor and balance of plant would be designed for a 30 year life. The plasma size was selected for ignited operation, with the near-ignition mode of operation as an alternative. PLT has already approached most of the CTHR plasma conditions, being within a factor of three or four of the plasma temperature and a factor of 10 to 20 of $n_e \tau_E$. TFTR is expected to serve as a plasma physics prototype for many of the CTHR parameters with results available early in the 1980's decade. Based on the present empirical transport scaling laws, it seems clear that ignition could be achieved with the CTHR concept.

The function of the fusion driver in CTHR is to generate neutrons that will be used in the transmutation of fertile material to fissile fuel in a blanket surrounding the plasma. For CTHR, a flexible tokamak and plant design have been developed based on a tritium breeding blanket and an on-site tritium recovery system which will provide self-sufficiency in the tritium supply. The fissile fuel reprocessing required for the light water reactor (LWR) plant's fuel would be done at a separate, off-site facility. Safety and environmental concerns have been considered to the extent possible in an early conceptual design.

Figure 2-35. UO_2 Boiling Water Cooled, Li_2O Helium Cooled Blanket Concept Direct Power Conversion System Schematic

2.3.1 FUSION DRIVER GUIDELINES AND ASSUMPTIONS

Two goals of the DTHR fusion driver design were to achieve a stable plasma under the required operating conditions and to produce a neutron wall loading of at least 2 MW/m^2 . The CTHR fusion driver was developed from the DTHR concept using similar guidelines. The guidelines that were used in defining the plasma are given in Table 2-15. These constraints were not rigid, but were simply used as a guide, since each quantity will have to be examined and optimized for any final design taking into account the total design and cost impact.

An aspect ratio $A \leq 4.5$ was chosen for a number of reasons. The critical β above which a plasma is unstable, varies inversely as A based on current theories. Thus, the smallest A possible would be desirable. However, small aspect ratios are difficult to achieve in practice, especially when shielding must be provided around the torus for D-T operation. Most tokamak operating experience has been obtained with tokamaks having $A \sim 3$, and TFTR, which will provide proof-of-principle for beam driven plasmas, will have $3 \leq A \leq 4.5$ depending on the mode of operation. Thus, $A \leq 4.5$, has been used as a guide in the selection of the plasma parameters. It is presently accepted by many researchers in the fusion community that stable plasma operation at near-ignition conditions can be achieved for β values in the range of 5-8%. This acceptance is based on current theories, including effects resulting from flux conservation and elongated plasmas, and extrapolations from current experimental results. However, the maximum achievable β will depend on the overall tokamak design. The limitation of β_p , ($\beta_p \leq A$), was chosen to prevent a shrinkage in minor radius of the poloidal field separatrix so that a satisfactory equilibrium could be achieved. The minimum allowable safety factor, q , for MHD stability has been chosen to be $q \geq 2.5$.

As a general guideline, the plasma current was required to satisfy $I_p > 3 \text{ MA}$ in order that the drift surfaces of α -particles, which are born during fusion reactions, do not intercept the vacuum vessel. The product $n_e a$, where n_e is the electron density and a is the plasma minor radius, is

TABLE 2-15
CTHR GUIDELINES FOR PLASMA DEFINITION

	<u>GUIDELINE</u>	<u>ACTUAL</u>
Stable Plasma		
Aspect Ratio	$A \leq 4.5$	4.3
Beta	$5\% \leq \beta \leq 8\%$	6.3
Poloidal Beta	$\beta_p \leq A$	4.3
Safety Factor	$q \geq 2.5$	2.8
α - Confinement		
	$I_p > 3 \text{ MA}$	6.8
Neutral Beam Penetration	$n_e a \leq 2 \times 10^{20} \text{ m}^{-2}$	1.77×10^{20}
Scaling		
Toroidal Field on Axis	$B_T \geq 5 \text{ T}$	5.2
Plasma Minor Radius	$a \geq 1 \text{ m}$	1.4

limited by $n_e a \leq 2 \times 10^{20} \text{ m}^{-2}$ so that adequate neutral beam penetration can be achieved with neutral beam energies $\leq 200 \text{ keV}$, which is considered a practical upper limit for positive ion technology. In CTHR, the near vertical neutral beam injection increases the $n_e a$ product by a fraction of the elongation. This factor, ~ 1.3 , was included in the actual value shown in Table 2-15 for CTHR.

Since the confinement times should increase with increasing a and increasing n (or B_T since n can increase as B_T increases), it seemed desirable to have the capability of achieving a toroidal magnetic field and plasma minor radius comparable to or larger than those achieved in TFTR. Thus, $B_T > 5 \text{ T}$ and $a \geq 1 \text{ m}$ were also used as guidelines.

Within these guidelines and under the engineering constraints such as providing a 20 cm thick plasma scrape-off layer, a 10 cm thick zone for the vacuum vessel, a 60 cm thick inner shield, the 60 cm inner blanket space allowance, the compact D-shaped TF coils, a bucking cylinder for the TF coil, dewar and thermal insulation space, assembly gaps, blanket and outer shield, and space for poloidal field coils, the reference plasma parameters shown in Table 2-1 were obtained. The COAST code⁽⁸⁾ was used in establishing most of the design parameters in that table. Additional design guidelines used in establishing the fusion driver concept are shown in Table 2-16.

The required τ_E listed in Table 2-1 is based on an energy balance assuming $Z_{\text{eff}} \sim 1$; the empirical τ_E is obtained from

$$\tau_E = 3.2 \times 10^{-21} n_e (sa)^2 q^{1/2}$$

where s for a "D" shaped plasma is approximately $0.75\delta + 0.25$ and δ is the elongation.

If the energy confinement time is accurately given by the empirical value it means that a certain amount of impurity can be accounted for in the energy

TABLE 2-16
CTHR FUSION DRIVER DESIGN GUIDELINES

- Ignited Plasma: (Near Ignition Option); 2 MW/m^2 Average Wall Loading.
- Superconducting Cryo-Cooled Magnets (Nb_3Sn).
- Serviceable Magnet System Within Refueling Time Cycle.
- Maximum Field on Coils -- 12 T (Nb_3Sn).
- Maximum Radiation Level on Coil Insulation: 10^9 rads; on Cu: 10^8 rads.
- Positive Ion Sources ($W_b \leq 200 \text{ keV}$) for NBIS With Direct Recovery. Negative Ion ($> 200 \text{ keV}$) Option With Direct Recovery for Larger Devices.
- Maximum First Wall Exposure $\leq 12 \text{ MW yr/m}^2$; Nominal 8.4 MW yr/m^2 . Lifetime Compatible with Blanket Refueling Cycle.
- Plasma Duty Cycle ≥ 0.9 via Longer Pulse Length.
- Refueling D and T by Gas-Gun Pellet Injection.
- Bundle Divertor Plasma Exhaust System.
- Ignited Shutdown in $\sim 1 \text{ s}$ via OH Coil and D/T Injection Shutdown.

balance and that a higher plasma temperature will be achieved than is given in Table 2-1.

2.3.2 BLANKET GUIDELINES AND ASSUMPTIONS

The CTHR blanket design guidelines were established to limit the scope of analysis to a few representative blanket fuel combinations. The versatility of the hybrid enables most fuels to be feasible if the blanket is optimized for it and therefore, to enable a more in-depth analysis only a few combinations were evaluated. The U-Pu fuel cycle was used because of the more accurate cost information available for the reprocessing cycle. The hybrid is assumed to provide fuel for a number of client LWR's. The symbiotic system is subsequently analyzed for its economic performance. The blanket models were developed to maximize fuel production using a good, fast spectrum preforming fuel (UC which requires gas cooling) and a conventional fuel (UO_2) using a conventional coolant (boiling water). Their performances were analyzed for a four year irradiation period to insure sufficient data would be available to evaluate various fuel management scenarios. All fuel combinations were required to be tritium self-sufficient and the fuel zone thickness was adjusted to achieve a tritium breeding ratio of $\sqrt{1.1}$ at beginning of life. A summary of the key blanket design guidelines is presented in Table 2-17. Detailed assumptions used in the neutronics analyses are discussed in the CTHR blanket report.⁽¹⁰⁾

2.3.3 ECONOMIC ANALYSIS GUIDELINES

The general guidelines used for the economic analysis are shown on Table 2-18. These guidelines comply with the PNL-2648 guidelines except that escalation was not applied in estimating the hybrid or the LWR plants capital costs. Both were costed on an equivalent basis and the reference cost of busbar electricity from a conventional LWR system was then established to use in comparing the competitiveness of the hybrid-LWR system.

TABLE 2-17
CTHR BLANKET DESIGN GUIDELINES

- Fissile Production Maximized (Analyze to Reach 3 w/o Fissile Enrichment or Hybrid Residence of 4 Years Maximum).
- Power Production Secondary But $\geq$ Hybrid Plant Required Operating Power.
- Steam Thermal Power Conversion System.
- Tritium Self-Sufficiency (T-BR $\sim$ 1.1 at BOL).
- U-Pu Fuel Cycle With Chemical Reprocessing.*
- Fissile Fuel Concepts*: (1) UC With Helium, (2) UO_2 With Boiling Water
- Tritium Breeding Materials -- Li_2O and LiH (Pb Multiplier/Reflector), Helium Cooled.
- Plant Availability -- 0.7 Max.
- Provision for Full Blanket Coverage

Th-U fuel cycle will be assessed in subsequent studies.

TABLE 2-18

CTHR ECONOMIC ANALYSIS GUIDELINES

- Use General Guidelines of PNL-2648, "Fusion Reactor Studies Standard Accounts For Cost Estimates," 5/78.
- 1979 Equipment and Facilities Cost Basis.
- No Initial Engineering Design Costs, Mature Industry in 21st Century.
- Use 20% Max. Contingency; Use 5% Contingency for State-of-the-Present Day-Art.
- Financing Guidelines:

	<u>PRIVATELY OWNED</u>
Fixed Charge Rate	15%
Interest During Construction	0.4*
Escalation to Plant Startup	None

*Based on 10% cost of capital, 6% escalation, 7 year construction period (PNL-2648).

2.4 ECONOMIC ANALYSES

The design of the CTHR was developed in sufficient depth to provide a basis for initial estimates of the capital cost and plant operating cost. These costs were estimated for three different blanket fuel combinations and one partial coverage blanket (outer fertile blanket only, no fertile inner blanket). The busbar cost of electricity for a conventionally fueled LWR was established for comparison to the busbar electricity cost of these four hybrid blankets and the LWR reactors they fueled. The sensitivity of the hybrid size to busbar electricity cost was analyzed to determine the hybrid size where equality with the "conventional" LWR electricity cost would occur. This section defines the costing methods and provides the results of the initial economic analyses of the hybrid-LWR symbiotic system.

2.4.1 ECONOMIC ANALYSES METHODS

The cost of the hybrid reactor plant was determined using the COAST-4 computer code⁽⁸⁾. The application and verification of the COAST-4 code has been previously discussed in a DTHR cost sensitivity study⁽⁶⁾. Based on the input data for each of the respective blanket concept fuel combinations, the total cost of the CTHR plant was estimated. A typical tokamak hybrid reactor plant layout is shown in Figure 2-36. The COAST code uses algorithms based on thermal power requirements to cost many of the BOP components. The general guidelines for the economic analyses were given in Table 2-18. The COAST code was modified to include the first level of standard accounts from PNL-2648⁽¹³⁾. Table 2-19 shows a typical CTHR capital cost account summary based on the UC-LiH-Helium cooled blanket concept. The direct costs include the following average costs applied to the equipment or facility base cost:

- Shipping and Handling 3%
- Assembly and Installation 15%
- Spare Parts Allowance 1%
- Contingency 9%

Figure 2-36. Tokamak Hybrid Reactor Plant Layout

TABLE 2-19

CTHR CAPITAL COST ACCOUNT SUMMARY
UC/LIH FULL BLANKET CONCEPT

<u>COAST Account Number</u>	<u>Account Title</u>	<u>Estimated Cost (1979 M \$)</u>
<u>DIRECT COSTS:</u>		
Depreciating Assets:		
21	Structures and Site Facilities	97
22	Reactor Plant Equipment	1165
23	Turbine Plant Equipment	349
24	Electric Plant Equipment	220
25	Miscellaneous Plant Equipment	75
		<hr/>
	TOTAL DIRECT COST	1906
<u>INDIRECT COSTS (All Depreciating Assets):</u>		
91	Construction Facilities, Equipment and Services	286
92	Engineering and Construction Management Services	286
93	Other Costs	95
		<hr/>
	TOTAL INDIRECT COSTS	667
<u>TIME RELATED COSTS (All Depreciating Assets):</u>		
94	Interest During Construction	1029
		<hr/>
	TOTAL CAPITAL COST	3602

This increases direct costs a total of 28% above the base equipment and facilities cost.

The indirect costs were established per PNL-2648 by using fixed percentages of the direct cost. Although interest during construction was included, escalation of the capital cost to plant startup was not included. This is inconsistent with PNL-2648, but due to the inability to project several decades into the future, it was decided to omit the escalation at this time.

The capital cost breakdown for the CTHR reactor equipment (Account 22) for the UC-LiH-Helium cooled blanket concept is presented in Table 2-20. The cost of one of the several blanket concepts compared in subsequent discussions is shown in account 22.01.01.01. The lead reflector zone cost was included in this account, since the blanket is an integral assembly and the COAST code treats the blanket costs as a single output value.

The hybrid plant operating and maintenance costs were added to COAST as a percentage of the total hybrid plant capital cost and had typical values of ~ 2 mills/kW_ehr. This requires further development to reach a more realistic value of 3 mills/kW_ehr. Other computer code related efforts included development of improved models for eventual incorporation into the COAST code, such as:

- Refined mechanical design model for TF coil nose section sizing.
- Upgraded models for magnetic field coil power supplies.
- Generalized model for fusile fuel handling systems.
- Operating cost models for cryogenic systems.

The hybrid plant performance and cost data developed by neutronic and COAST analyses were used as input data for the SYMECON computer code⁽⁷⁾, which analyzes the hybrid and client LWR's as an integrated energy system. The key input data related to the hybrid plant are:

TABLE 2-20

CAPITAL COST BREAKDOWN FOR CTHR REACTOR EQUIPMENT
UC/LiH FULL BLANKET CONCEPT

<u>PNL-2648</u> <u>Account</u> <u>Number</u>	<u>Account Title</u>	<u>Cost, M \$ (1979)</u>
22	<u>Reactor Plant Equipment</u>	
22.01	Reactor Equipment	
22.01.01	Blanket and First Wall	141.0
.01	Blanket - Inner Blanket	60.4
	Outer Blanket	71.5
.02	Vacuum Vessel (First Wall)	6.3
.03	Reflectors/Multipliers (in Blanket)	
.04	Limiters	2.8
22.01.02	Shield	153.6
.01	TF Coil Shields - Inner	36.3
	Outer	111.7
.02	NBIS Duct Shields	5.6
22.01.03	Magnets	415.0
.01	TF Coils	316.8
.02	SF Coils	51.2
.03	OH Coils	21.3
.04	Winding Equipment	25.7
22.01.04	Supplemental Heating	53.7
.01	Neutral Beam Injection System	53.7
22.01.05	Primary Structure and Support	43.5
.01	Reactor Structure	21.6
.02	Equipment Support (Bucking Cylinder)	21.9
22.01.06	Reactor Vacuum Systems	13.8
.05	Reactor Vacuum Systems	7.4
.06	Reactor Vacuum Wall (Enclosure Shell)	6.4
22.01.07	Power Supply Switching & Energy Storage	105.7
.01	OH Coil Power Conversion Equipment	35.5
.02	SF Coil Power Conversion Equipment	62.1
.03	TF Coil Power Conversion Equipment	2.3
.04	PF Coil MGF & Emergency System	5.8
22.01.08	Impurity Control	15.2
.01	Bundle Divertor and Collector Chamber	5.3
.02	Bundle Divertor Coil	9.9

TABLE 2-20
(CON'T)
CAPITAL COST BREAKDOWN FOR CTHR REACTOR EQUIPMENT (Cont'd.)
UC/LiH FULL BLANKET CONCEPT

<u>Account Number</u>	<u>Account Title</u>	<u>Cost M\$ (1979)</u>	
22.02	Main Heat Transfer & Transport System		39.1
.01	Primary Coolant System	33.4	
.05	Clean-up System	5.7	
22.03	Auxiliary Cooling System		59.7
.01	Magnet Cooling	55.7	
.02	Shield and Structure Cooling	2.1	
.03	NBIS Cooling	0.5	
.05	Vacuum Vessel and Divertor	1.4	
22.04	Radioactive Waste Treatment and Disposal		4.8
.01	Liquid Wastes Processing and Equipment	3.5	
.02	Gaseous Wastes and Off Gas Processing Sys.	1.3	
.03	Solid Wastes Processing Equipment	(with .01)	
22.05	Fuel Handling and Storage Systems		41.7
.01	Purification	8.0	
.03	Preparation		
.04	Injection	11.0	
.05	Storage	3.7	
.06	T Recovery (Plasma Exhaust)	19.0	
22.06	Other Reactor Plant Equipment		24.7
.01	Maintenance	24.7	
22.07	Instrumentation and Control		13.2
.01/.02	Reactor I & C/Rad. Monitoring	13.2	
	Sub Total		1124.7
22.98/99	Spare Parts Allowance, Shipping and Handling, Component Assembly		40.3
	Total Direct Cost		1165.0

- Irradiation Time
- Heavy Metal Discharge Rate
- Discharge Isotopic Composition
- Tritium Breeding Ratio
- Annual Electricity Production
- Capital and Operating Costs
- Fuel Cycle Cost Data
- Financial Factors

The "client" LWR plants are defined as those plants which receive their makeup fissile fuel from the hybrid. The input data for these client converters required by SYMECON includes:

- Core Residence Time
- Burnup Rate
- In-Core Cycles
- Reactivity Worths
- Loading and Discharge Compositions
- Capital and Operating Costs
- Fuel Cycle Cost Data
- Operating and Financial Factors

The SYMECON code analyzes the symbiotic system to establish the busbar cost of the electricity generated by the system. Figure 2-37 shows a simplified schematic of the fuel cycle scenario assumed for this study. The busbar cost for electric energy is set equal for the hybrid and the LWR. The fissile fuel transfer value is then established based on this equality. This method does not reflect an open market price for fissile fuel. It results in a levelized value of the ^{239}Pu that can be used to calculate hybrid revenue. Thus, the hybrid's electricity sales and fissile fuel sales will then offset its gross operating cost. Figure 2-38 shows the generalized materials flow diagram used in the SYMECON analysis. The SYMECON code outputs include the symbiotic system electricity costs, the fuel cycle costs, the fixed and operating costs, the system electrical production, the fission reactor electrical production, and the LWR net fissile fuel value. Since the LWR is assumed to be on a Pu recycle scenario, the fissile fuel in the LWR loop is a mixture of the higher grade hybrid Pu and the lower grade LWR recycle Pu. This net value reflects a cost averaging from the two Pu sources.

2.4.2 CONVENTIONAL LWR BUSBAR ELECTRICITY COST

It is of interest for comparison with the hybrid cases to establish a relative busbar cost for the conventional LWR system using the same financial assumptions applied to the hybrid cost estimates. To achieve this, a capital cost summary was developed for a typical 1 GWe LWR plant using the same general format and economic guidelines used for the hybrid. The time related costs, in particular, were established with the same assumptions. Table 2-21 shows the summary of capital costs for the 1 GWe LWR plant. Using this capital cost and the present day cost of isotopic enrichment (\$89/SWU) with an optimistic cost of yellowcake (\$40/lb), the cost of electricity produced by a typical LWR was determined. The contributing costs resulting in the 37 mills/kWhr value are shown in Table 2-22. The assumptions used in this analysis are shown in the table footnotes. It can be seen from the results of this analysis that the LWR fuel cycle cost contributes about 24% of the busbar cost, while the LWR capital cost contributes about 76% of the cost of busbar electricity.

Figure 2-37. CTHR/LWR Energy System Scenario Used in SYMECON Analysis

Figure 2-38. Generalized Material Flow Diagram Used In SYMECON

TABLE 2-21
TYPICAL 1 GW_e LWR CAPITAL COST SUMMARY

<u>Account Number</u>	<u>ACCOUNT TITLE</u>	<u>Estimated Cost (1979 M\$)</u>
<u>DIRECT COSTS</u>		
21	STRUCTURE AND SITE FACILITIES	60
22	REACTOR PLANT EQUIPMENT	125
23/24	TURBINE-GENERATOR EQUIPMENT	115
25	MISCELLANEOUS PLANT EQUIPMENT	70
	CONSTRUCTION AND INSTALLATION	150
	TOTAL DIRECT COSTS	<u>520</u>
<u>INDIRECT COSTS</u>		
91	CONSTRUCTION FACILITIES AND SERVICES	80
92	PROFESSIONAL SERVICES	80
93	OTHER COSTS	35
	TOTAL INDIRECT COSTS	<u>195</u>
<u>TIME RELATED COSTS</u>		
94	INTEREST DURING CONSTRUCTION ^(a)	285
95	ESCALATION	NONE
	TOTAL TIME RELATED COSTS	<u>285</u>
	TOTAL CAPITAL COST	<u>1000</u>

(a) Based on a 7 year construction period, 6% escalation and 10% cost of capital.

TABLE 2-22
TYPICAL PWR FUEL CYCLE/ELECTRICITY COSTS
USING ISOTOPIC ENRICHMENT

	1979 Cost (\$/kg HM)
<u>Front End Fuel Cycle Costs</u>	
Investment in U_3O_8 (a), (b)	692
Conversion of U_3O_8 to UF_6 (c), (b)	20
Isotopic Enrichment (d), (b)	384
Design and Fabricate Fuel (UO_2) (b)	<u>123</u>
Unlevelized Investment	1219
Levelized Cost of Delivered PWR Fuel (e)	1853
<u>Back End Fuel Cycle Costs</u>	
Levelized Carrying Charge During 3 Year PWR Operation (f)	513
Shipping and Reprocessing (g)	416
U and Pu Credit (Pu Recycle Scenario)	<u>-576</u>
Levelized Back End Fuel Cycle Costs	353
<u>Levelized PWR Fuel Cycle Cost</u>	
	Front End 1853
	Back End <u>353</u>
	Total 2206
<u>Levelized PWR Busbar Cost (h)</u>	
Levelized PWR Fuel Cycle Cost, mills/kWhr.	9 (24%)
Levelized PWR Capital Cost, mills/kWhr.	<u>28 (76%)</u>
Levelized PWR Busbar Cost, mills/kWhr.	<u>37</u>
(a) Based on U_3O_8 @ \$40/lb., 2.6 lb. U_3O_8 /kg U feed, 6.33 kg U feed/kg U product for a PWR requiring 3.17 w/o enrichment, assuming 0.25 w/o tails.	
(b) Preoperation carrying charge rate on investment is 10% per year; 5% de-escalation.	
(c) Based on \$3/kg U in UF_6 and 6.33 kg U feed/kg U product.	
(d) 3.17 w/o enrichment with 0.25 w/o tails requires 4.17 SWU/kg U product; at \$89/SWU.	
(e) Lifetime levelizing factor = 1.52 based on 5% escalation over 30 yr. plant life and discounted at 10% to 1979 present worth.	
(f) Operational carrying charge rate is 17% per year.	
(g) Shipping at \$25/kg, reprocessing at \$200/kg with 5% escalation and 10% carrying charge rate.	
(h) Based on 33 GWd/THM PWR burnup and $\eta = 0.3$ PWR plant efficiency. Assuming a \$1000 M plant (1 GWe) committed in 1979 and 15% fixed charge rate, 30 yr. plant life.	

It can be anticipated that two of the key cost factors effecting LWR busbar electricity costs (the cost of U_3O_8 and isotopic enrichment) may increase significantly beyond the assumed values in the decades ahead. Increases in enrichment costs to over \$93/SWU have already been proposed for October 1, 1979. Figure 2-39 shows the recent trends in the yellowcake and enrichment costs with two arbitrary projections of future costs. The recent lull in demand for U_3O_8 accounts for the possible dip in cost over the next decade. However, rapidly rising enrichment costs will probably offset this effect. Due to the unpredictability of these factors and of escalation over the next several decades, the LWR base was established with the same assumptions as those used on the hybrid capital costs, i.e., present day values and no escalation. Thus, if the hybrid-LWR systems busbar cost of electricity is calculated to be 37 mills/kWhr, that hybrid concept would be economically competitive without further optimization. If the value is slightly higher than 37, then optimization should make the concept economically competitive.

2.4.3 CTHR BLANKET/SYSTEM COST COMPARISONS

The blankets used in this study were intended to provide a representative spectrum of performance and cost. The UC-LiH-Helium cooled, full coverage blanket was the best fissile fuel producer of the three concepts analyzed. The UO_2 -boiling water cooled-LiH-Helium cooled blanket had the highest thermal power output. The UO_2 -boiling water cooled- Li_2O -Helium cooled blanket had the lowest capital and gross operating costs. The costs and performances for these three concepts are compared in Table 2-23.

The outer blankets were more expensive than the inner blankets due to their larger volume. The UO_2 - Li_2O blanket concept was also 2 cm thinner at the fertile fuel section than the other two concepts. The LiH fuel was costed at \$72/kg, while the Li_2O fuel was costed at \$38/kg. These cost estimates were based on the quantities of materials required (~100 tonnes as opposed to 50 to 500 g prices for typical quantities used in laboratories) and the knowledge that the hydrides require significantly more dry, inert atmosphere handling than the oxides require. The UC fuel is more dense than the UO_2 fuel and about 25% more expensive to manufacture. The differences in capital costs for

Figure 2-39. Key Cost Factors Affecting LWR Fuel Cycle Costs

TABLE 2-23
CTHR BLANKET CONCEPTS ECONOMIC COMPARISON

<u>HYBRID COSTS</u>	<u>UC/LiH</u>	<u>UO₂/LiH</u>	<u>UO₂/Li₂O</u>
OUTER BLANKET COST, M\$	72	63	49
INNER BLANKET COST, M\$	60	40	34
POWER CONV. SYS., M\$	273	272	239
REACTOR PLANT EQUIP. COST, M\$	1165	1135	1108
TURBINE PLANT EQUIP. COST, M\$	349	348	305
ELECTRIC PLANT EQUIP. COST, M\$	220	223	217
TOTAL CAPITAL COST, M\$	3602	3549	3406
GROSS OPTG. COST, M\$/yr.	731	662	632
<u>HYBRID PERFORMANCE</u>			
FISSILE PROD., kg ²³⁹ Pu/yr.	2868	1675	1590
HM DISCHARGE THM/yr.	101	69	64
ELECTRIC PROD., TWhr/yr.	12.09	13.47	11.65
DESIGN POWER RATING, MWe	1980	2200	1900
ELECTRICITY/FISSILE RATIO, GWhr/kg ²³⁹ Pu	4.22	8.04	7.33
1 GWe LWR'S SUPPORTED	8	4.7	4.4
<u>HYBRID COST FACTORS</u>			
CAPITAL COST/ELECTRIC POWER, \$/kWe	1827	1616	1793
GROSS OPTG. COST/ELECTRIC POWER, mills/kWhr	60.5	49.1	54.2
GROSS OPTG. COST/FISSILE FUEL, \$/g ²³⁹ Pu	255	395	397
FUEL TRANSFER VALUE, \$/g ²³⁹ Pu	88	79	97
<u>CLIENT LWR COSTS</u>			
NET FISSILE FUEL VALUE, \$/g fissile Pu	40	38	46
FUEL CYCLE COST, mills/kWhr	11.7	11.3	13.0
CAPITAL COST, mills/kWhr	28.0	28.0	28.0
<u>SYSTEM COSTS</u>			
BUSBAR ELECTRICITY COST, mills/kWhr	39.7	39.3	41.1

the three concepts are attributed to the blanket cost differences and the power generation equipment. The lower power of the $\text{UO}_2\text{-Li}_2\text{O}$ concept accounted for nearly \$40 M less in costs for heat exchangers, steam generators, steam turbine-generator, etc., while the blanket was $\sim$ \$20 M less expensive. The helium coolant system components tended to be more expensive than water coolant system components and thus, the $\sim$ 200 MW_e lower rating for the UC-LiH-Helium cooled concept did not reduce costs to any noticable extent. A more in-depth analysis of these costs would be beneficial when more detailed designs are available.

The performances of the three blanket concepts have been discussed in Section 2.1.6. The key performance parameters are listed in Table 2-23 and were used to establish the hybrid cost factors. These factors relate key products (fissile fuel and electricity production) to capital or gross operating costs. The gross operating cost is a levelized value including the capital cost fixed charge depreciation, the O&M cost, the hybrid fertile-fissile fuel cycle cost and the hybrid fusile fuel cycle cost. It was noted that none of these cost factors nor the LWR's supported were accurate independant indicators of which concept would have the lowest busbar electricity cost. As shown in Table 2-23, the UC-LiH and $\text{UO}_2\text{-LiH}$ concepts were basically equal with an electricity cost of 39 to 40 mills/kWhr, while the $\text{UO}_2\text{-Li}_2\text{O}$ concept was slightly higher at $\sim$ 41 mills/kWhr. The client LWR costs shown in Table 2-23 include the net fissile fuel value and resulting fuel cycle costs for the LWR as well as the capital cost depreciation charge. The fuel cycle cost can be compared to the 9 mills/kWhr value in Table 2-22 for a conventional LWR scenario. The 39 to 41 mills/kWhr values can be compared to the 37 mills/kWhr from Table 2-22. Thus, all three concepts would require further optimization to be competitive as a 6 meter major radius reactor concept. In view of the preliminary nature of the hybrid analysis, however, the differences in busbar costs are not considered to be extremely significant.

As previously discussed in Section 2 and illustrated in Figure 2-15, the plant availability has a significant affect on fissile and electricity production. The cost comparison presented in Table 2-23 was based on three full coverage blanket concepts assuming a 70% plant availability (P.A.). This is considered

optimistic for the full blanket concepts, yet may be somewhat conservative for an outer fertile blanket concept that does not require annual refueling of an inner blanket. As indicated in Figure 2-15, the outer blanket performance at P.A. = 0.7 is comparable to the full coverage blanket performance at $\sim 60\%$ plant availability. Therefore, a cost comparison between a full blanket at P.A. = 0.7 to an outer blanket at P.A. = 0.7 should bracket the probable performance range for that blanket concept. This comparison is shown in Table 2-24 for the UC-LiH-Helium cooled concept. It can be seen that the busbar electricity cost would fall between 39.7 and 42.4 mills/kWhr, which is a cost increase of $\sim 7\%$ for the full blanket concept, if the P.A. dropped to ~ 0.6 . The outer blanket concept had 16% less fissile production and 19% less power production, which resulted in 7% higher busbar electricity costs when the 70% plant availability was assumed for both concepts.

2.4.4 CTHR SIZE/PERFORMANCE/COST COMPARISON

The affect of the reactor size on performance and cost was studied parametrically using the COAST and SYMECON computer codes. The 6.0 m major radius CTHR with a UC-LiH-Helium cooled blanket concept was used as the base case. The major radius was reduced to 5.2 m (the DTHR concept size) and increased to 8.0 m. A comparison of the hybrid plant characteristics over this range of sizes is shown in Table 2-25. The hybrid economic characteristics, client LWR costs and resulting system busbar electricity costs are presented in Table 2-26. As indicated in previous DTHR economic studies⁽¹⁾, the cost of electricity and fissile fuel production decreases with increasing major radius. Figure 2-40 shows a typical trend for the tokamak hybrid, where the costs drop rapidly until the size reaches a 6 to 7 m radius. Thereafter, the reduction in cost is nearly linear over the range of sizes considered. In comparing the CTHR results to the LWR "conventional" busbar cost of 37 mills/kWhr, it can be seen from Table 2-26 that a 7.5 m major radius concept would be cost equivalent without further optimization. However, when noting that this concept is a 3000 MW_e plant, it is probable that a more optimum concept in the 6 to 7 m radius range and a 2000 MW_e rating would be realistic for an early to mid-twenty first century application. Based on results to date, more extensive studies of various fuel

TABLE 2-24
CTHR UC/LiH BLANKET CONCEPTS ECONOMIC COMPARISON

<u>HYBRID COSTS</u>	<u>FULL BLANKET</u>	<u>FERTILE OUTER BLANKET</u>
OUTER BLANKET COST, M\$	72	103
INNER BLANKET COST, M\$	60	11
POWER CONV. SYS., M\$	273	225
REACTOR PLANT EQUIP. COST, M\$	1165	1133
TURBINE PLANT EQUIP. COST, M\$	349	288
ELECTRIC PLANT EQUIP. COST, M\$	220	212
TOTAL CAPITAL COST, M\$	3602	3411
GROSS OPTG. COST, M\$/yr.	731	683
<u>HYBRID PERFORMANCE</u>		
FISSILE PROD., kg ²³⁹ Pu/yr.	2868	2413
HM DISCHARGE, THM/yr.	101	92
ELECTRIC PROD., TWhr/yr.	12.09	9.79
DESIGN POWER RATING, MWe	1980	1600
ELECTRICITY/FISSILE RATIO, GWhr/kg	4.22	4.06
1 GWe LWR'S SUPPORTED	8	6.7
<u>HYBRID COST FACTORS</u>		
CAPITAL COST/ELECTRIC POWER, \$/kWe	1827	2137
GROSS OPTG. COST/ELECTRIC POWER, mills/kWh	60.5	69.8
GROSS OPTG. COST/FISSILE FUEL, \$/g ²³⁹ Pu	255	283
FUEL TRANSFER VALUE, \$/g ²³⁹ Pu	88	111
<u>CLIENT LWR COSTS</u>		
NET FISSILE FUEL VALUE, \$/g fissile Pu	40	51
FUEL CYCLE COST, mills/kWhr	11.7	14.4
CAPITAL COST, mills/kWhr	28.0	28.0
<u>SYSTEM COSTS</u>		
BUSBAR ELECTRICITY COST, mills/kWhr	39.7	42.4

TABLE 2-25

CTHR SIZE VERSUS PERFORMANCE COMPARISON
UC/LiH FULL BLANKET CONCEPT; NEUTRON WALL LOADING AT 2 MW/m^2

HYBRID PLANT CHARACTERISTICS

MAJOR RADIUS, m	5.2	6.0*	7.0	7.5	8.0
MINOR RADIUS, m	1.2	1.4	1.6	1.75	1.9
MAX. TOROIDAL FIELD, T	12.5	12.0	10.5	9.9	9.4
NBI ENERGY, keV	185	197	211	218	224
MAX. PLASMA CURRENT, MA	6.1	6.8	7.6	8.0	8.4
PEAK FUSION POWER, MW_t	910	1200	1600	1840	2070
INITIAL FUEL LOAD, FERTILE TONNES (UC)	330	418	540	600	680
FUSILE TONNES (LiH)	80	104	135	150	165
AVERAGE ELECTRIC CONSUMPTION, MW_e	171	215	278	315	351
DESIGN POWER RATING, MW_e	1500	1980	2665	3060	3450
ELECTRIC POWER (NET), TWhr/yr.	9.51	12.09	16.23	18.72	21.12
FISSILE PRODUCTION (NET), kg ^{239}Pu /yr.	2181	2868	3849	4414	4975
HEAVY METAL DISCHARGE, THM/yr.	77	101	136	155	175

*Reference Case

TABLE 2-26

CTHR SIZE VERSUS ECONOMICS COMPARISON
UC/LiH FULL BLANKET CONCEPT; NEUTRON WALL LOADING AT 2 MW/m²

HYBRID ECONOMIC CHARACTERISTICS

MAJOR RADIUS, m	5.2	6.0*	7.0	7.5	8.0
CAPITAL COSTS, M\$	3129	3602	4285	4676	5085
FIXED CHARGE DEP., M\$/yr.	470	541	643	702	763
O & M COST, M\$/yr.	62	76	100	112	124
GROSS OPTG. COST, (GOC) M\$/yr.	627	731	884	972	1059
CAPITAL COST/ELECTRIC POWER, \$/kWe	2098	1827	1621	1532	1476
GOC/ELECTRIC POWER, mills/kWhr.	65.9	60.5	54.5	51.9	50.1
GOC/FISSILE FUEL, \$/g ²³⁹ Pu	287	255	230	220	213
FUEL TRANSFER VALUE, \$/g ²³⁹ Pu	106	88	70	64	58
1 GWe LWR's SUPPORTED	6	8	11	12	14

CLIENT LWR COSTS

NET FISSILE FUEL VALUE, \$/g fissile Pu	48	40	32	29	26
FUEL CYCLE COST, mills/kWhr	13.6	11.7	9.8	9.0	8.5
CAPITAL COST, mills/kWhr.	28.0	28.0	28.0	28.0	28.0

SYSTEM COSTS

BUSBAR ELECTRICITY COST, mills/kWhr	41.6	39.7	37.8	37.0	36.5
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*Reference Case

Figure 2-40. CTHR Economic Factors Versus Size

management scenarios for a specific design would be beneficial in reducing the busbar cost of electricity for the symbiotic system. Also, increased neutron wall loading for the 6m concept would clearly improve its economic performance.

The LWR busbar electricity costs were shown to be more dependent on the LWR capital cost ($\sim 76\%$) than on the fuel cycle cost ($\sim 24\%$) (see Table 2-22). The hybrid products only affect the fuel cycle costs of the LWR. The sensitivities of the two hybrid products (fissile fuel and net electric power), the hybrid costs, the LWR fuel cycle cost and the resulting system busbar cost of electricity are shown in Figure 2-41. The data is plotted with respect to the reference concept values ($R_0 = 6$ m, UC-LiH-He cooled blanket) and the variation in major radius of the tokamak. Since the hybrid production increases more rapidly than the hybrid costs increase, a significant reduction in LWR fuel cycle cost can be achieved. However, because of the more dominating influence of the LWR capital cost (and hybrid capital cost) on the busbar cost of electricity, the "bottom line" reduction in the system busbar cost of electricity is much less pronounced. It can be concluded from this that further study in reducing the capital cost of the hybrid should be the most cost beneficial effort for concepts using the U-Pu fuel cycle. This requires a significant increase in design detail and component cost analyses than the present level of conceptual design includes. For the Th-U fuel cycle systems, the larger client support capability may have the largest impact, i.e. for a larger number of clients supported, the busbar cost of electricity becomes less sensitive to hybrid capital cost.

Figure 2-41. CTHR Size/Performance/Cost Sensitivities on LWR Fuel Cycle Cost and System Electricity Cost

3.0 CONCLUSIONS AND RECOMMENDATIONS

3.1 GENERAL HYBRID CONCLUSIONS

The likelihood that a growing shortfall in worldwide nuclear fuel resources will develop early in the next century has provided the stimulus for increased interest in the fusion hybrid concept in recent years. The hybrid approach offers some uniquely beneficial characteristics for the production of significantly larger quantities of fissile fuel than that obtainable from fission reactors. This potential for substantial fuel production is attributable to physics characteristics generic to the hybrid concept. The use of a tokamak as a basis for a fusion hybrid appears to provide the nearest term approach to demonstrating this concept. Understanding of the plasma physics of the tokamak is the farthest advanced of any of the fusion confinement concepts currently under development. Operation of a tokamak in a beam injected mode is a promising approach for early hybrid application. However, the tokamak also offers the potential of reaching "ignition" conditions where all beams could be turned off during the pulse. This potential is in contrast to that of the mirror and the ICF confinement concepts where some input energy is always required to sustain the reactor and/or to inhibit leakage of the reacting plasma. These features tend to offset the obvious complexity of the tokamak which has given rise to concerns on the part of some investigators regarding the viability of the tokamak concept as a reactor.

The fusion driver characteristics of tokamaks are now well enough defined to warrant serious, in-depth engineering studies of the reactor and its subsystems. The development work required for hybrids can ultimately be applied directly to pure fusion reactors for most subsystems. Fusion hybrids have comparatively near term applications as fuel sources for LWR's, LMFBF's, and LWBR's. The feasibility of the tokamak hybrid could conceivably be demonstrated by the early 1990's using beam driven plasma technology and subsequently serve as a test facility to explore ignited plasma conditions. Much of the fission reactor technology developed over the past 20 years is applicable to the hybrid blanket concepts. The dual products of the hybrid,

fissile fuel and electricity, give hybrids a unique economic flexibility which has the potential to make the concept competitive in the early twenty-first century to resolve anticipated shortfalls in nuclear fuel resources.

3.2 CTHR CONCLUSIONS

A scoping design of a commercial tokamak hybrid reactor was developed in sufficient detail to show technical credibility and provide the basis for initial cost estimates. Selection of a size for the CTHR was influenced by trade-offs involving several factors. On the one hand, there are incentives which favor a larger size reactor to benefit from economies of scale and to insure adequate space and accessibility for a blanket configuration that covers the toroidal fusion neutron source as completely as possible. On the other hand, increasing the size can lead to the need for extremely large capital investment for construction of a single unit and can result in the production of net electrical energy in amounts that may be difficult to effectively couple to a utility grid. The relationship of size to net electrical energy production is also affected by the type of fissile fuel cycle selected and the fuel management strategies adopted in the hybrid blanket. The uranium/plutonium fuel cycle was selected as the initial reference for the CTHR to provide a design baseline using current fuel cycle technology. This fuel cycle results in the production of significant amounts of energy in the hybrid blanket. Future plans include evaluation of the thorium/uranium-233 fuel cycle for the CTHR. This fuel cycle provides much less blanket energy, primarily due to the lower fast fission cross section of Th-232 as compared to U-238.

With the above considerations in mind, a reference size of 6.0 meters major radius and 1.4 meters minor radius was selected for CTHR. This size provides a net electrical output of 1980 MW_e which is larger than current central station plants, but which may be compatible with electrical grids in the early twenty-first century. Another consideration in establishing the reference size for the CTHR is the trend in economy of scale. Parametric studies using the COAST computer code have indicated a significant improvement in economics up to about 6 meters major radius with less dramatic gains beyond that point.

The CTHR design incorporates an elongated plasma which has an average temperature of 17.3 keV based on a parabolic profile. The electron density-confinement time product is 13.4×10^{19} seconds/m³. The fusion power is 1200 MW and the average neutron wall loading is 2 MW/m². The 6 m size was large enough to incorporate a bundle divertor and inner fertile blanket. The bundle divertor continues to be the preferred concept for the plasma exhaust system. The reactor size was close enough to the projected economical "breakeven" concept to provide a good costing base for further parametric studies. The base concept can be extrapolated to smaller, lower cost reactor concepts designed for predominantly electricity production or to larger reactor concepts dedicated to fissile fuel production. The reactor toroidal field coils were designed with a large enough bore size to permit flexibility in evaluating several blanket concepts with fuel thickness and geometry variations.

3.2.1 TOROIDAL FIELD COIL CONCLUSIONS

The major component in the tokamak concept is the toroidal field (TF) coil. Considerable design effort was applied to evaluating the TF coil serviceability and maintenance, the structural stability and the thermal isolation of the concept. With the inclusion of a bundle divertor in the 6 m concept, it was determined that 14 TF coils on equal spacing were feasible using a 2% peak to peak ripple limit for the plasma. The coils could be designed for individual removal or removal in pairs depending on which blanket coverage concept and refueling scenario was selected. Stresses resulting from the combination of the tension and centering forces dictated the TF coil nose thickness and geometry. Coil wedging was incorporated since it reduced the required nose thickness in the vertical leg of the coil. Bucking rings were required to remove tension from the curved transition between the vertical leg and the semi-circular outer leg. The rings also provide for TF coil stability under a faulted coil condition. The OH coils were maintained close to the plasma by positioning them in slots in the bucking rings.

The out of plane forces were smaller than anticipated and are adequately restrained by shear panels and stiffener beams wedged over the top, outer and bottom cross section of each TF coil. The twist between the upper half of the

14 coil assembly and the lower portion was much larger than anticipated. The stiffener beams were used to restrain these moments by transmitting them to a torsion cylinder located in the torus centerbore. Mounting structures were designed to minimize thermal conduction to the reactor floor. The centerbore area, including the TF coil vertical legs and the OH coils, was enclosed in a common dewar system. The remaining poloidal field (PF) coils were located in four common dewar assemblies to enable their relocation during blanket refueling or TF coil removal operations. The outer leg of the TF coils and shear panels were insulated in individual dewar systems. This enables maintaining cryogenic conditions in the Nb_3Sn superconducting coils during the short-term down-time operations like blanket refueling or routine maintenance. The location of the neutral beam injectors beneath the reactor with the beam ducts entering between the TF coils also eliminates the need for NBI removal during these same operations.

3.2.2 BLANKET CONCEPT CONCLUSIONS

The neutronic performance of two generic blanket concepts were studied parametrically to determine a few representative combinations of fertile and fusile fuels with two different coolants. All the blanket concepts operated on the U-Pu fuel cycle and were designed to achieve tritium self-sufficiency while maximizing the fissile fuel production within thermal and mechanical design constraints. The more promising concepts were analyzed for sufficient irradiation exposure time, to achieve ~ 3 w/o fissile fuel enrichment. The studies included scoping evaluations of fuel zone thickness variations, lead reflector/multiplier advantages, fuel volume fraction sensitivities, boiling water coolant density variations, and blanket coverage factors. Four concepts were selected for economic analyses:

- 1) UC fertile fuel, LiH tritium breeding fuel, helium cooled full blanket concept.
- 2) Same as 1) except outer fertile blanket only; inner blanket is LiH-Pb with no UC.

- 3) UO_2 fertile fuel boiling water cooled, LiH tritium breeding fuel helium cooled, full blanket concept.
- 4) Same as 3) except Li_2O tritium breeding fuel.

The blanket specifications and performance parameters for each concept are shown in Table 3-1. Based on the data shown in this table and the design iterations required to complete these concepts, the following conclusions were drawn thus far in the study.

Tritium breeding self-sufficiency is feasible with all the concepts studied. In-situ tritium removal is feasible with the Li_2O using helium as the transport media. However, LiH would require removal and reprocessing with TSTA type equipment because of the higher temperature ($\sim 600^\circ\text{C}$) required to drive-off the hydrogen to get the tritium. The stainless steel clad and structural materials cannot withstand this temperature. The use of higher temperature material could make in-situ tritium removal from LiH feasible, but it may not be economically practical.

Enrichment to ~ 3 w/o fissile fuel was achieved in a four year irradiation period. The use of LiH as the tritium breeding fuel permitted thinner fertile fuel zones and the use of lead as a reflector (or multiplier for the outer coverage only concept) enhanced the fissile fuel production rate. The full blanket coverage concept had 20 to 25% more fissile fuel production than the outer fertile blanket only coverage concept. However, realistic refueling times to reach the inner blanket area may nullify the full blanket concept advantages if plant availability factors are reduced from 0.7 to ~ 0.6 .

The coolant path length was restricted to 1.5 m long in the UC fertile fuel zone by pumping power limitations. For the UO_2 boiling water cooled concepts, the channel exit quality restricted the fertile fueled zone coolant path length to about 2.5 m. These restrictions had significant effects on the initial mechanical designs. The helium coolant generally required higher operating pressure, but resulted in higher steam pressure for power conversion systems. The boiling water coolant concepts required power conversion systems with greater than 800 psia (steam pressure) to assure commercial turbine

TABLE 3-1
CTHR BLANKET PERFORMANCE COMPARISON

BLANKET SPECIFICATIONS

<u>FERTILE FUEL</u>	UO ₂	UO ₂	UC	UC
COVERAGE	FULL	FULL	FULL	OUTER
COOLANT	BW	BW	He	He
INNER BLANKET THICKNESS, cm	14	16	16	—
OUTER BLANKET THICKNESS, cm	14	16	16	24
FUEL/STRUCTURE/COOLANT, %	50/20/30	50/20/30	50/20/30	50/20/30
CLAD/STRUCTURE MATERIAL	Zr	Zr	SS	SS
<u>TRITIUM BREEDING FUEL</u>	Li ₂ O	LiH	LiH	LiH
COOLANT	He	He	He	He
INNER BLANKET THICKNESS, cm	20	34	34	36
OUTER BLANKET THICKNESS, cm	24	64	30	30
FUEL/STRUCTURE/COOLANT, %	70/10/20	70/10/20	70/10/20	70/10/20
CLAD/STRUCTURE MATERIAL	SS	SS	SS	SS
REFLECTOR OR MULTIPLIER MATERIAL/ LOCATION, THICKNESS INNER/OUTER	C/back 26/52	Pb/front 10/10	Pb/front 10/10	Pb/front 14/10
<u>PERFORMANCE PARAMETERS</u>	(BOL-2 yr.-4 yr.)			
TRITIUM BREEDING RATIO	1.03-1.13-1.21	1.04-1.22-1.37	1.05-1.20-1.34	1.06-1.18-1.29
F, NET ²³⁹ Pu ATOMS/FUSION NEUTRON	0.85-0.54-0.38	1.02-0.57-0.37	1.10-0.97-0.85	0.97-0.87-0.78
T + F	1.88-1.67-1.59	2.06-1.79-1.74	2.15-2.17-2.19	2.03-2.05-2.07
M, BLANKET ENERGY MULTIPLICATION	5.5-7.5-8.7	6.2-8.7-10.0	5.9-7.9-9.9	5.1-6.7-8.3
HEAVY METAL INVENTORY, THM	270-255-235	305-277-244	412-410-406	382-380-376
E, FISSILE ENRICHMENT, w/o	0.25-1.71-2.96	0.25-1.76-3.08	0.25-1.64-2.89	0.25-1.50-2.64
HEAVY METAL DISCHARGE, THM/yr.	64	69	101	92
G, FISSILE PRODUCTION, kg ²³⁹ Pu/yr.	2405-1524-1076	2862-1609-1039	3090-2730-2410	2570-2300-2050
DESIGN FISSILE PRODUCTION, kg/yr.	1590	1675	2868	2413
NUMBER OF 1 GWe LWR's SUPPORTED	4.4	4.7	8.0	6.7
P, AVERAGE THERMAL POWER, MWt	4249-5840-6733	4834-6695-7727	4580-6100-7620	3760-4920-6080
DESIGN POWER RATING, MWe	1900	2200	1980	1600

equipment feasibility. The steam exit quality should be between 30 to 40% to assure flow stability and prevent dryout problems at high heat fluxes for the boiling water cooled concepts.

Equilibrium cycle performances were estimated using average ($\sim$ 2 year performance) data and the rate of change of the fissile fuel production and thermal power output. This technique seemed adequate for the UC concepts, but requires more in-depth study for the UO_2 concepts due to the rapid increase in the Pu burnup rate. This is the result of the soft spectrum caused by moderation of the neutrons in the boiling water coolant. However, the increase in thermal power became an important factor in the economics analysis, since electric power production is a more sensitive parameter than fissile fuel production.

3.2.3 ECONOMIC CONCLUSIONS

The three, full coverage blanket concepts used in the economic analysis were intended to provide a representative spectrum of performance and cost for an initial assessment. The UC-LiH-helium cooled blanket had the best fissile fuel production. The UO_2 boiling water cooled-LiH helium cooled blanket had the largest electricity production. The UO_2 boiling water cooled- Li_2O helium cooled blanket had the lowest capital and gross operating costs, although the fissile fuel and electricity production were also lower. The key performance parameters are listed in Table 3-1. These parameters can be studied to get a general appreciation for how well the performance was optimized for each concept. For example, the $T + F$ parameter is a measure of neutron usage. Since T was the fixed parameter, F became the variable which affected the sum. If the fertile blanket were optimized, both F and the sum $T + F$ would increase. Comparing the $T + F$ value for similar concepts can give a general indication of how well the performance of the concept was optimized. Further study of these concepts could be expected to lead to improved fuel production performance.

For example, a comparison of the UO_2 - Li_2O concept to the similar UO_2 -LiH concept shows that the $T + F$ sum is 10% lower (1.88 vs. 2.06). A review of the blanket specifications reveals that the UO_2 -LiH concept had a 2 cm

thicker fertile blanket and 14 cm thicker tritium breeding blanket. Yet, it was previously concluded that LiH with a Pb reflector should enable a thinner blanket design. Therefore, the $\text{UO}_2\text{-Li}_2\text{O}$ concept could probably be optimized in further studies by using thicker fuel zones and reducing the carbon reflector zone. This should reduce the difference in cost of busbar electricity shown in Table 3-2 (41.1 for $\text{UO}_2\text{-Li}_2\text{O}$ vs. 39.3 for $\text{UO}_2\text{-LiH}$). The conclusion here is that more effort should be concentrated on a single concept to enable more in-depth study, than to pursue the never ending search for the "ultimate" concept.

The hybrid cost characteristics in Table 3-2 can be compared for the three blanket concepts or for the full coverage UC concept to the outer only coverage UC concept. The ratios relating costs to production values do not appear to have any obvious, simple correlation to the resulting busbar cost of electricity. It can be seen that the $\text{UO}_2\text{-LiH}$ concept had the lowest busbar electricity cost, but the UC-LiH full coverage concept is probably about equal within the accuracy of cost estimates, performance scaling, etc. As previously mentioned, further optimization of the $\text{UO}_2\text{-Li}_2\text{O}$ concept would probably reduce its electricity cost value. The plant availability factor assumed also has a significant affect on the full blanket concepts, since the 0.7 value used may be optimistic when inner blanket removal for refueling is fully evaluated. The performance of the outer blanket concept, therefore, represents an upper bracket value for the full blanket UC concept. Thus, it can be concluded that typical busbar electricity costs for the hybrid-LWR system could be in the 39 to 42 mills/kWhr range.

The hybrid-LWR system costs were compared to a "present day" LWR busbar electricity cost of 37 mills/kWhr. The LWR was costed on the same basis as the hybrid and typical yellowcake (\$40/lb) and enrichment (\$89/SWU) charges were used. The hybrid-LWR system scenario can match the "present day" LWR value if the hybrid size is increased from a major radius of 6 meters to 7.5 meters. Other alternatives include cost reducing the hybrid using parametric studies of design details or continued optimization of one of the blanket concepts. The continued increase in isotopic enrichment charges or the cost of uranium ore will also reduce this difference. Therefore, it is concluded that at the present state of design detail, the hybrid has the potential to be

TABLE 3-2
CTHR BLANKET ECONOMIC COMPARISON

HYBRID COST CHARACTERISTICS

BLANKET CONCEPT	UO ₂ -Li ₂ O	UO ₂ -LiH	UC-LiH	UC-LiH
BLANKET COVERAGE	FULL	FULL	FULL	OUTER
TOTAL CAPITAL COST, M\$	3406	3549	3602	3411
GROSS OPERATING COST, M\$	632	662	731	683
ELECTRICITY/FISSILE RATIO, GWhr/kg ²³⁹ Pu	7.33	8.04	4.22	4.06
NUMBER OF 1 GWe LWR's SUPPORTED	4.4	4.7	8.0	6.7
CAPITAL COST/ELECTRIC POWER, \$/kWe	1793	1616	1827	2137
GROSS OPTG. COST/ELECTRIC POWER, mills/kWhr	54.2	49.1	60.5	69.8
GROSS OPERATING COST/FISSILE FUEL, \$/g ²³⁹ Pu	397	395	255	283
FUEL TRANSFER VALUE, \$/g ²³⁹ Pu	97	79	88	111

CLIENT LWR COSTS

NET FISSILE FUEL VALUE, \$/g fissile Pu	46	38	40	51
FUEL CYCLE COST, mills/kWhr	13.0	11.3	11.7	14.4
CAPITAL COST, mills/kWhr	28.0	28.0	28.0	28.0

SYSTEM COSTS

BUSBAR ELECTRICITY COST, mills/kWhr	41.1	39.3	39.7	42.4
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cost competitive when coupled to a system of LWR client reactors. A publicly owned hybrid scenario would undoubtedly be much more cost competitive due to: (1) the reduced capital cost resulting from fewer indirect costs such as interest charges, etc. and (2) the lower operating costs which are reduced because of lower fixed charge rates on the lower capital costs. This scenario may be a more consistent approach in which to compare the hybrid with the conventional LWR fuel cycle since isotopic enrichment facilities are publicly owned.

3.3 RECOMMENDATIONS

Significant effort has thus far been expended in the CTHR program, but many results are quite preliminary in nature. A hybrid system of the type described here is a major facility with capital investment in the several billion dollar range. First time engineering costs for such a facility could amount to 25% or more of the capital cost, i.e., at least several hundred million dollars. An engineering expenditure of this magnitude is even more likely when working at new technologies such as large superconducting coils and neutral beam injectors for which there is only limited experience to date. Viewed in this light, the resources allocated to date for the CTHR study represent a very modest level of effort which can only provide a cursory treatment of the design of many systems and components. Thus, significant additional effort is needed to provide a credible basis for cost and performance data and evaluations.

The following discussion presents an overview of areas in which significant additional efforts are needed, followed by a more limited outline of specific areas in which work can be anticipated during the coming year. The CTHR program for FY 80 has been formulated with guidelines that involve strict limitations on the availability of resources.

3.3.1 REACTOR DESIGN AND ANALYSES

The major components in the tokamak have been defined in sufficient depth to assure that viable design approaches exist. However, many components which directly affect the plasma, TF coils or blanket performance require more

detailed design effort to enable better performance parametric analyses and cost trade-off studies. Table 3-3 lists several reactor components and supporting systems requiring more detailed design work to permit refinement of hybrid performance projections and economic assessments. For example, the vacuum vessel design has a direct affect on blanket neutronics performance. Ideally, the vacuum vessel should be as transparent as possible to fusion neutrons to ensure that their energy is degraded by a minimal amount before reaching the fertile material in the blanket. Even a difference of 1 to 2 cm in equivalent vacuum vessel wall coolant can also be an important factor, especially if water coolant is selected. Another important aspect of vacuum vessel design is the impact on downtime needed for removal, either to replace the vessel or to gain access to the inner blanket. These considerations require significant in-depth effort to develop an approach to vacuum vessel design which adequately serves the function for this component without downgrading the performance of the system.

The other items in Table 3-3 have similar impacts on adjacent components, plasma operating conditions, blanket space availability, and component and operating costs. Further, in-depth studies would define any significant problems with these systems or confirm that present assumptions are appropriate for the performance and cost conclusions that have been reached.

Specific areas of effort anticipated during the coming year include the following: The vacuum vessel should be designed in sufficient detail to confirm structural adequacy and thermo-hydraulic feasibility of the first wall cooling system. The assembly and removal method should be defined. The TF coil dewar system should be designed in enough detail to include space envelope requirements for the refrigeration coolant system feed lines and the electrical cable arrangements up to the transition to copper busbars. The PF coil system should be designed and analyzed to confirm magnetic requirements and plasma control, establish physical requirements for space and scope initial structural support configurations. The bundle divertor system should be sufficiently developed to define all the mechanical requirements to ensure adequate space is available for the superconducting coils, the gettering collector, and the recharging chambers. The fusile fuel injection system

TABLE 3-3
COMPONENT/SYSTEMS REQUIRING FURTHER STUDY - REACTOR DESIGN AND ANALYSES

- VACUUM VESSEL DESIGN
- VACUUM SHELL CONCEPT
- TF COIL SHIELDS AND SUPPORT STRUCTURE
- PF COIL DESIGN
- VERTICAL NBI SYSTEM
- CRYOGENIC SYSTEMS
 - TF COILS
 - PF COILS
 - BUNDLE DIVERTOR COILS
 - NBIS
- BUNDLE DIVERTOR SYSTEM
- FUSILE FUELING SYSTEM
- PRIMARY COOLANT SYSTEM CONTAINMENT
- BALANCE OF PLANT

conceptual design should be developed and system requirements defined. The reactor cross-section and plan-view layouts should be upgraded to incorporate all of the upgraded concepts.

3.3.2 BLANKET DESIGN AND ANALYSES

Several key issues requiring further study have already become apparent during the CTHR blanket design and analyses. The present level of blanket design, plant performance and system costing is not of sufficient depth to clearly distinguish the "best" solution to many design and analysis questions. The ideal approach would be to fully develop several blanket concepts to enable detailed comparison of the "bottom-line" costs. For example, in-depth development of the three concepts shown in Table 3-4 could yield the following comparisons:

- Uranium carbide versus uranium oxide fertile fuel economics.
- Helium versus boiling water primary coolant system costs.
- U-Pu fuel cycle versus Th-U fuel cycle scenario costs.
- Costs for in-situ tritium breeding with Li_2O versus a removal and reprocessing tritium breeding scenario with LiH .
- Plant availability differences for the preceding concepts.

The further study of the key issues listed in Table 3-4 would aid the comparisons above as well as provide for a better definition of the power conversion systems and the overall plant efficiencies. Safety and reliability issues related directly to the blanket require good, in-depth conceptual designs to ensure that realistic fault analyses can be developed.

A much more restrictive blanket design effort than that outlined above is all that can be anticipated during the coming year. Development of a detailed blanket concept using a single fertile fuel and a tritium breeding fuel should be achieved. The mechanical design of the modules, sectors, coolant system, structural support system, and refueling system should be detailed to the

TABLE 3-4

KEY ISSUES REQUIRING FURTHER STUDY - BLANKET DESIGN AND ANALYSES

- DETAILED BLANKET DESIGNS
 - UC/LiH HELIUM COOLED CONCEPT
 - $\text{UO}_2/\text{Li}_2\text{O}$ WET STEAM/He COOLED CONCEPT
 - Th-U/TRITIUM BREEDING CONCEPT
- BLANKET REFUELING SYSTEM CONCEPT
 - FERTILE INNER BLANKET
 - TRITIUM BREEDING BLANKETS T RECOVERY
 - REFUELING EQUIPMENT CONCEPTS
 - REFUELING SAFETY & ENVIRONMENTAL LIMITATIONS
- BLANKET FUEL MANAGEMENT STRATEGIES
 - EQUILIBRIUM CYCLE
 - POWER FLATTENING
 - BLANKET COVERAGE TRADE-OFFS
 - USE OF Pb/Li IN FERTILE ZONES
- POWER CONVERSION SYSTEMS AND OVERALL PLANT EFFICIENCIES
- RELIABILITY AND SAFETY ISSUES
 - EFFECTS OF FIRST WALL COOLANT LEAKS INTO PLASMA
 - EFFECTS OF WATER COOLANTS ON LITHIUM BLANKETS
 - MATERIALS AND STRUCTURAL LIMITATIONS AND RD&D NEEDS

extent required to perform structural and thermo-hydraulic analyses. An equilibrium cycle and fuel management scheme should be developed for the reference concept's operation.

Refueling considerations should include an in-depth study of the tritium recovery cycle to confirm technical feasibility and establish cost algorithms for the process. A scoping analysis of the radiation levels for components requiring handling and removal should be made. The fissile recovery system and related processes should be re-examined and upgraded to present day technology and costs. An analysis of the plant availability factor should be conducted to establish a "most probable" factor for the resulting CTHR concept.

The thorium-uranium fuel cycle should be evaluated by performing a scoping neutronic analysis to establish a representative blanket model. An initial preconceptual design should be developed.

3.3.3 ECONOMICS

In the area of economic analyses, further refinement of the computer codes, COAST and SYMECON, would greatly enhance the cost analyses. Present methods require constant benchmark testing with in-depth neutronic analyses to project blanket performance and numerous mechanical layouts and analyses to confirm mechanical design feasibility of the overall reactor concept. Key sensitivities in the hybrid and LWR plants requires better defined models for cost estimating. More detailed models for the fuel cycle alternatives are needed for incorporation into the SYMECON code to provide versatility in trade-off studies of the numerous options possible. An expansion of the SYMECON models would also add depth in the technical and economic parameter variations to permit better sensitivity studies of the combined system. Flexibility in plant ownership and financing arrangements would expand the administrative analysis ability of the SYMECON code.

In the coming year, the uranium-plutonium fuel cycle cost comparisons made in this report should be upgraded as a result of the additional design and analysis effort just defined. Tritium fueling and reprocessing scenarios should be developed including representative costing rates and the parametric

sensitivities established. The thorium-uranium fuel cycle costing parameters should be incorporated in the computer codes and a preliminary parametric study conducted. The SYMECON code should be upgraded for fusile fueling material and cost analysis. Parameter sensitivity analyses should be made using COAST and SYMECON on the reference CTHR concept.

3.3.4 SAFETY AND ENVIRONMENTAL CONCERNS

As more in-depth reactor concepts are developed, safety and environmental concerns can begin to be addressed with some degree of realism. Numerous key issues requiring further study are listed in Table 3-5. Although many of these can be studied in general terms to provide broad guidelines, the specific acceptability of a system or component cannot be fully confirmed until the interrelationship with all the reactor/plant systems is integrated into the analysis. Based on the limited resources of the program, however, no specific effort is anticipated in these areas in the coming year, except those associated with the blanket refueling considerations.

TABLE 3-5
KEY ISSUES REQUIRING FURTHER STUDY - SAFETY AND ENVIRONMENTAL CONCERNS

- REACTOR CONTROL
 - STARTUP AND SHUTDOWN
 - PLASMA STABILITY
 - MAGNET SYSTEM FAILURES
 - LOSS OF POWER
 - LOSS OF BLANKET COOLANTS
 - LOSS OF CRYOGENIC COOLANTS
- RADIATION CONTROL
 - TRITIUM CONTAINMENT
 - BLANKET CRITICALITY ANALYSES
 - BLANKET REFUELING
 - ATMOSPHERIC RELEASE PREVENTION
 - WASTE DISPOSAL
 - BIOLOGICAL SHIELDING
- ENVIRONMENTAL CONTROL
 - WASTE HEAT REJECTION
 - MAGNETIC FIELD EFFECTS
 - POWER TRANSMISSION
 - FUEL TRANSPORTATION
 - ACCIDENT CONTROL SYSTEMS

4.0 TABLE OF CTHR DESIGN PARAMETERS

For ease in referring to data characterizing the CTHR concept, the following table has been completed in accordance with DOE guidelines. Entries which do not apply to the CTHR concept are labeled "N/A". Entries which are applicable, but which are beyond the scope of the FY'79 work are labeled "later".

Additional entries will be made during the FY'80 effort to more thoroughly characterize the CTHR concept. Thus, the table represents the interum (September, 1979) CTHR concept parameters and are subject to change during the fiscal year.

TABLE OF CTHR DESIGN PARAMETERS

September, 1979

		<u>Unit</u>	<u>Value</u>
<u>1. Characteristic Machine Dimensions</u>			
1.1	Reactor Envelope		
1.1.1	Height	m	27
1.1.2	Width	} Reactor Vacuum Shell Diameter	~ 30
1.1.3	Length		
1.2	Vacuum Vessel		
1.2.1	Major Radius	m	2.7*
1.2.2	Minor Radius	m	1.7*
1.2.3	Volume	m ³	370
1.2.4	Inner Surface Area	m ²	600
* Vacuum Vessel is "D" Shaped			
<u>2. Plasma Parameters</u>			
2.1	Plasma Dimensions		
2.1.1	Major Radius, R	m	6.0
2.1.2	Minor Radius, a	m	1.4
2.1.3	Plasma Elongation		1.6
2.2	n_0 , centerline density	m ⁻³	1.94×10^{20}
2.3	$\bar{n}$, average density	m ⁻³	0.97×10^{20}
2.4	τ_E , energy confinement time	sec	1.38
2.5	τ_e , electron confinement time	sec	1.38
2.6	τ_i , ion confinement time	sec	1.38
2.7	$n\tau_E$ (averaged through plasma)	sec/m ³	13.4×10^{19}
2.8	β_0 , centerline toroidal beta	%	18.9
2.9	$\langle\beta\rangle$, average toroidal beta	%	6.3
2.10	β_{p0} , centerline poloidal beta		4.3
2.11	$\langle\beta_p\rangle$, average poloidal beta		4.3

		<u>Unit</u>	<u>Value</u>
2.	<u>Plasma Parameters (Continued)</u>		
2.12	I_p , plasma current	MA	6.8
2.13	T_{io} , centerline ion temperature	keV	26.0
2.14	$\bar{T}_i$, average ion temperature	keV	17.3
2.15	T_{eo} , centerline electron temperature	keV	26.0
2.16	$\bar{T}_e$, average electron temperature	keV	17.3
2.17	Z_{eff} , effective plasma ion charge		~ 1.0
2.18	q , plasma safety factor		2.8
2.19	Volt-seconds	Volt-Sec	100
2.20	Reactor Cycle	Steady State/ Pulsed	Pulsed
2.20.1	Burn Pulse Length	sec.	1200
2.20.2	Total Pulse Length	sec.	1336
2.21	Fuel Cycle (i.e., D-T, D-D, etc.)		D-T
2.22	Plasma Heating Method		Ohmic/Neutral Beam
2.23	Plasma Heating Power	MW	80
2.24	Plasma Heating Energy	keV	200
2.25	Plasma Energy Gain, Q_p		∞ (Ignited)

3. Power Output

3.1	Plasma Power (Peak)	MW (th)	1200
3.2	Plasma Power (Total Cycle Time Average)	MW (th)	1080
3.3	Power to Blanket (Peak)	MW (th)	960
3.4	Power to Blanket (Total Cycle Time Average)	MW (th)	864
3.5	Blanket Power Amplification Factor	(Avg.)	8.14
3.6	Power to Direct Convertor (Peak)	MW (th)	N/A
3.7	Power to Direct Convertor (Total Cycle Time Average)	MW (th)	N/A
3.8	Power to Divertor (Peak)	MW (th)	139
3.9	Power to Divertor (Total Cycle Time Average)	MW (th)	125
3.10	Plasma Chamber Power Density (Total Cycle Time Average)	MW/m ³	2.9

		<u>Unit</u>	<u>Value</u>
3.	<u>Power Output (Continued)</u>		
3.11	Nuclear Island Power Density (Total Cycle Time Average)	MW/m ³	N/A
3.12	Plant Gross Electrical Output	MW _e	2195
3.13	Plant Net Electrical Output	MW _e	1980
3.14	Thermal Cycle Efficiency	%	37
3.15	Direct Convertor Efficiency	%	N/A
3.16	Net Plant Efficiency	%	35
4.	<u>Reactor Coolant System</u>		
4.1	Blanket Coolant Type		Helium
4.2	Blanket Outlet Temperature (Hot Leg) - Peak/Average	°C	478/450
4.3	Blanket Inlet Temperature (Cold leg) - Peak/Average	°C	280
4.4	Blanket Outlet Pressure - Peak/Average	MPa	8.48
4.5	Blanket Inlet Pressure - Peak/Average	MPa	8.62
4.6	Blanket Coolant Flow Rate	kg/s	842
4.7	Blanket Coolant Pipe Material		SST
4.8	First Wall Coolant Type		Pressurized Water
4.9	First Wall Outlet Temperature - Peak/Average	°C	Later
4.10	First Wall Inlet Temperature - Peak/Average	°C	Later
4.11	First Wall Outlet Pressure - Peak/Average	MPa	Later
4.12	First Wall Inlet Pressure - Peak/Average	MPa	Later
4.13	First Wall Coolant Flow Rate	kg/s	Later
4.14	Total Number of Blanket Coolant Loops	Inner Outer	1 4
4.15	Type of Blanket Coolant Circulator		Compressors
4.16	Power Input to Each Circulator	MW _e	~ 26
4.17	Peak First Wall/Blanket Temperature in Case of Loss of Coolant Flow	°C	Later
4.18	Energy Storage	J	Later

		<u>Unit</u>	<u>Value</u>
5.	<u>Intermediate Coolant System</u>		
5.1	Coolant Type		N/A
5.2	IHX Outlet Temperature (Hot Leg)	°C	N/A
5.3	IHX Inlet Temperature (Cold Leg)	PC	N/A
5.4	IHX Outlet Pressure	MPa	N/A
5.5	IHX Inlet Pressure	MPa	N/A
5.6	Coolant Flow Rate	kg/s	N/A
5.7	Coolant Pipe Material		N/A
5.8	Total Number of Coolant Loops		N/A
5.9	Type of Coolant Circulator		N/A
5.10	Power Input to Each Circulator	MW _e	N/A
5.11	Number of IHX Per Loop		N/A
5.12	IHX Material - Shell/Tube		N/A
5.13	Energy Storage	J	N/A
6.	<u>Steam Generation System</u>		
6.1	Steam Outlet Temperature	°C (°F)	420
6.2	Steam Outlet Pressure	MPa (psia)	8.62
6.3	Steam Flow Rate	kg/s (lb/hr)	Later
6.4	Feedwater Temperature	°C (°F)	150
6.5	Number of Steam Generators Per Loop		1
6.6	Number of Modules Per SG		Later
6.7	SG Materials, Shell/Tube		Later
7.	<u>Shield Coolant System</u>		
7.1	Total Energy Deposited in the Shield	MW _t	350
7.2	Shield Coolant Type		Borated Water
7.3	Shield Outlet Temperature - Peak/Average	°C	Later
7.4	Shield Inlet Temperature - Peak/Average	°C	Later
7.5	Coolant Outlet Pressure - Peak/Average	MPa	Later
7.6	Coolant Inlet Pressure - Peak/Average	MPa	Later
7.7	Coolant Flow Rate	kg/s	Later

	<u>Unit</u>	<u>Value</u>
8. <u>Reactor Auxiliary Systems</u>		
8.1 <u>Vacuum Pumping System</u>		
8.1.1 Plasma Chamber Pressure	torr	1×10^{-8}
8.1.2 Plasma Chamber Volume	m^3	370
8.1.3 Number of Pumps		6
8.1.4 Capacity of Each Pump	torr-litre/s	19.4
8.2 <u>Magnet Cooling System</u> - Cooling Load	kW_t	33,000
8.3 <u>Plasma Heating System</u> - Cooling Load	kW_t	(in 8.2)
8.4 <u>Plasma Fueling System</u>		
8.4.1 Type		Gas Gun
8.4.2 Fuel Composition	%D	50
	%T	50
8.4.3 Fueling Rate	kg/s	1.1×10^{-4}
8.4.4 Pellet Size, Radius	m	0.004
8.4.5 Pellet Injection Frequency	sec^{-1}	1.4
8.5 <u>Tritium Processing and Recovery System</u>		
8.5.1 Total Tritium Inventory	kg	50
8.6 <u>Impurity Control System</u>		
8.6.1 Type		Bundle Divertor
9. <u>Reactor Components</u>		
9.1 <u>Blanket/First Wall</u>		
9.1.1 Structural Material		316 SST
9.1.2 Breeding Material		UC/LiH
9.1.3 Breeding Ratio	Avg.	1.0/1.2
9.1.4 Number of Modules (Sectors)		37 Outer 13 Inner
9.1.5 Weight of Module (Sectors)	Tons (Metric)	25.1
9.1.6 Weight of Largest Single Component	Tons (Metric)	25.1
9.1.7 Dimensions of Largest Component	m x m x m	$0.77 \times 1.26 \times \sim 7$
9.1.8 First Wall Loading (Peak/Average)		1.5
9.1.8.1 14.1 MeV neutrons	MW/m^2	2.0
9.1.8.2 Alpha Particle Flux	MW/m^2	0.4

	<u>Unit</u>	<u>Value</u>
9. <u>Reactor Components</u>		
9.1.8.3 Electro-Magnetic Radiation ^(a)	MW/m ²	0.04
9.1.8.4 Plasma Thermal Conduction	MW/m ²	0.4
9.1.8.5 Wall Life	MW/yrs/m ²	4.8-9.6
9.2 <u>Shielding</u>		
9.2.1 Material		Borated Water and SST
9.2.2 Number of Modules (50 Sectors)		~ 65 Inner ~ 260 Outer
9.2.3 Weight of Each Module	Tons (Metric)	Later
9.2.4 Weight of Largest Single Component	Tons (Metric)	Later
9.2.5 Dimensions of Largest Component	m x m x m	Later
9.3 TF Coil Magnet		
9.3.1 Superconducting	yes/no.	yes
9.3.2 Conductor Material		Nb ₃ Sn
9.3.3 Structural Material		310 SST
9.3.4 Operating Temperature	K	4.2
9.3.5 Coolant		Helium/N ₂ Guard
9.3.6 Maximum Stress in Coil	N/m ²	561 x 10 ⁶
9.3.7 Maximum Force Transmitted to Building	N/m ²	later
9.3.8 Maximum Field	T	12
9.3.9 Field on Axis	T	5.2
9.3.10 Number of Magnets		14 TF Coils 52 PF Coils
9.3.11 Field Ripple-Edge/Center	%	~ 1.0
9.3.12 Stored Energy	MJ	~ 30,000
9.4 <u>Energy Transfer and Storage</u>		
9.4.1 Plasma Preparation		
9.4.1.1 Type	MJ	Later
9.4.1.2 Energy Per Unit	MJ	Later
9.4.1.3 Total Energy	MJ	Later
9.4.1.4 Peak Power Transfer Rate	W	Later
9.4.1.5 Transfer Time	sec.	Later

(a) Include electro-magnetic radiation components from synchrotron, bremsstrahlung, and impurity inputs.

	<u>Unit</u>	<u>Value</u>
9. <u>Reactor Components</u> (Continued)		
9.4.1.6 Depth of Discharge	%	Later
9.4.1.7 Recharge Time	sec.	Later
9.4.1.8 Pulse Frequency	sec. ⁻¹	Later
9.4.1.9 Switching Requirements		
9.4.1.9.1 Current	A	Later
9.4.1.9.2 Volts	V	Later
9.4.1.9.3 Number		Later
9.4.2 Primary Power Supply		
9.4.2.1 Type		Later
9.4.2.2 Energy Per Unit	MJ	Later
9.4.2.3 Total Energy	MJ	Later
9.4.2.4 Transfer Time	sec.	Later
9.4.2.5 Recharge Time	sec.	Later
9.4.2.6 Pulse Frequency	sec. ⁻¹	Later
9.4.2.7 Switching Requirements		
9.4.2.7.1 Current	A	Later
9.4.2.7.2 Volts	V	Later
9.4.2.7.3 Number		

10. Electrical Power Requirements

10.1	Cold Start Power from Grid	MW _e vs. sec.	~ 800 for < 10
10.2	Auxiliary Power Requirements (Normal Operation)	MW _e	215
10.2.1	Electrical Energy Storage	MW _e	N/A
10.2.2	Magnet Power Supply (Other than Energy Storage)	MW _e	0.2
10.2.3	Blanket Circulators	MW _e	114.0
10.2.4	First Wall Coolant Circulators	MW _e	~ 1.4
10.2.5	Shield Coolant Circulators	MW _e	1.5
10.2.6	Refrigeration System	MW _e	32.9
10.2.7	Vacuum System	MW _e	3.6
10.2.8	Plasma Heating System	MW _e	0.2
10.2.9	Miscellaneous Reactor Plant Auxiliaries	MW _e	61.2

		<u>Unit</u>	<u>Value</u>
10.	<u>Electrical Power Requirements (Continued)</u>		
10.2.10	Feed Pump System	MW _e	(in 10.2.9)
10.2.11	Condensing System	MW _e	(in 10.2.9)
10.2.12	Heat Rejection System	MW _e	(in 10.2.9)
10.2.13	Misc. BOP Auxiliaries	MW _e	(in 10.2.9)

11. Buildings

11.1	Reactor Building		
11.1.1	Characteristic Dimensions	m x m x m	later
11.1.2	Enclosed Volume	m ³	later
11.1.3	Minimum Wall Thickness for Shielding	m	later
11.1.4	Internal Pressure, Normal/Accident	MPa	later
11.1.5	Containment Atmosphere		later
11.2	Electrical Energy Storage Building		later
11.2.1	Characteristics Dimensions	m x m x m	later
11.2.2	Wall Thickness for Shielding	m	later
11.2.3	Internal Pressure Normal/Accident	MPa	later
11.2.4	Safety Related or Not	Yes/No	later
11.3	Reactor Service Building		
11.3.1	Characteristic Dimensions	m x m x m	later
11.3.2	Special Functions (i.e., hot cells, blanket processing equipment, etc.)		later

12. Reactor Maintenance

12.1	Blanket/First Wall Replacement	% Surface Area/Yr. Tons/yr.	25 later
12.2	Radioactive Material Storage Requirement, Years/Volume	Yr./m ³	later

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