MASTER

CONCEPTUAL DESIGN OF A
DEMONSTRATION TOKAMAK HYBRID REACTOR
(DTHR)

SEPTEMBER, 1978

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(DTHR)

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ABSTRACT

The flexibility of the fusion hybrid reactor to function as a fuel production facility, power plant, waste disposal burner or combinations of all of these, as well as the reactor's ability to use proliferation resistant fuel cycles, has provided the incentive to assess the feasibility of a near-term demonstration plant. The goals for a Demonstration Tokamak Hybrid Reactor (DTHR) were established and an initial conceptual design was selected. Reactor performance and economics were evaluated and key developmental issues were assessed. The study has shown that a DTHR is feasible in the late 1980's, a significant quantity of fissile fuel could be produced from fertile thorium using present day fission reactor blanket technology, and a large number of commercially prototypical components and systems could be developed and operationally verified. The DTHR concept would not only serve as proof-of-principle for hybrid technology, but could be operated in the ignited mode and provide major advancements for pure fusion technology.

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1.0 INTRODUCTION

The main thrust of the FY 78 work was to determine the feasibility of a near-term Demonstration Tokamak Hybrid Reactor (DTHR) which could be the precursor to a viable first generation of commercial hybrid plants. The present study builds on prior hybrid studies, but with some important differences in objectives and ground rules. Much of the prior work in the hybrid field has considered the use of advanced, high performance fuel materials, high temperature coolants and advanced structural materials which would appear to offer the potential of relatively high fissile production rates as well as very efficient thermal-to-electrical power conversion. There is ample evidence from the development history of fission power reactors that a less-than-optimum system in terms of performance may be more viable commercially, however. On this basis an important approach in this study was to make maximum use of proven light water fission reactor technology to define and assess a reference hybrid reactor system for an initial demonstration of fissile production on a significant scale. This approach would not preclude the consideration of advanced concepts in later stages of the study or the testing of advanced fuel and neutron application modules in the DTHR.

The scope of work completed in the hybrid study in FY 78 included the following elements:

- Identification of DTHR design goals.
- Selection of a baseline design approach for each major DTHR subsystem and trade study alternatives.
- Development of a coupled performance and cost systems model to permit sensitivity and trade studies.
- Development of a first-iteration pre-conceptual design and cost estimate for the DTHR.
- Development of reference proliferation-resistant fuel cycle scenarios with particular emphasis on the production of U-233 fissile fuel in the hybrid for use in a conventional LWR (including the performance of an LWR on the thorium fuel cycle).

Much of this work was previously reported in the March 1978 DTHR Status Report⁽¹⁻¹⁾. The principal goals were set at that time for the DTHR. The baseline design approach for each major DTHR subsystem was selected and the technical basis was documented⁽¹⁻¹⁾. A first iteration conceptual design including a reference fuel cycle (ThO_2) and a preliminary analysis of LWR performance on the thorium fuel cycle were also documented⁽¹⁻¹⁾.

This report presents the DTHR design concept as of September, 1978.

The only major subsystem changes were in the bundle divertor and the fertile fuel blanket designs. This report also includes the DTHR economic analysis and an initial assessment of performance and cost trade-offs. Evaluation of the DTHR capabilities as an engineering and materials test reactor is provided as well as the DTHR's projected performance as a commercial hybrid reactor.

REFERENCES:

1. Kelly, J. L., et al., "March 1978 Status Report on the Preliminary Conceptual Design of a Demonstration Tokamak Hybrid Reactor (DTHR)," WFPS-TME-086, June, 1978, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

2.0 SUMMARY AND CONCLUSIONS

The development and deployment of advanced energy technologies has emerged as an urgent factor in meeting the nations's future energy needs. Nuclear energy must assume a significant role in satisfying these needs to avoid excessive dependence on imported fossil fuels, insufficient conservation or undeveloped energy alternatives. The future growth of the nuclear power industry appears to be constrained by two circumventable factors; the projected insufficient quantity of uranium ore that can be recovered at reasonable cost, and the increasing concern regarding possible diversion of fissile materials for weapons use. Recent studies have indicated that the combination of fusion and fission in a hybrid system can supply ample amounts of fissile material from existing fertile materials in a proliferation resistant fuel form. The flexibility of the hybrid to use alternative fuel cycles and its ability to function as a fuel production facility, power plant, waste disposal burner or combinations of all of these, as well as its potential near-term availability, provides a strong incentive for a feasibility assessment of the concept.

2.1 FEASIBILITY ASSESSMENT

The Demonstration Tokamak Hybrid Reactor (DTHR) preliminary conceptual design has been developed to help provide the basis for subsequent assessment of the long-range potential of the hybrid concept. In the study efforts during fiscal year 1978 the following tasks were performed:

- Goals for DTHR were established.
- A baseline design approach for each major DTHR subsystem was selected and scoping analyses were performed.
- A first iteration design concept was completed; second iterations on the bundle divertor and fertile blanket concepts were completed.
- Computer code (COAST) modifications to couple hybrid performance and system costs were modeled; initial sensitivity/trade-off studies were performed.

- A reference thorium uranium 233 fuel cycle scenario was evaluated, including an initial performance estimate for the thorium fuel cycle in an LWR.
- Key development issues were assessed for the fusion driver and the hybrid blanket, and the engineering and materials test capabilities of DTHR were explored.
- Preliminary cost estimates were developed for the DTHR plant and capital equipment, preoperational RD&D effort and annual operations; some capital cost versus operating cost trade-offs were analyzed.
- The DTHR performance and costs were extrapolated to a commercial operation.
- A scoping study of the economics of hybrid enrichment of LWR fuel rods (UO_2) without reprocessing was performed.

The principal goals which were set for the DTHR are listed in Table 2-1, followed by a brief statement of the conclusions of studies to satisfy those goals. The technical basis for each baseline design approach for the major DTHR subsystems was documented in the March Status Report⁽²⁻¹⁾, including the reference fuel cycle (ThO_2) and a preliminary evaluation of LWR performance on the thorium fuel cycle. This report presents the DTHR conceptual design as of September, 1978. The only major subsystem changes were in the bundle divertor and the fertile fuel blanket designs. The report includes the DTHR economic analysis and an initial assessment of performance and cost trade-offs. Projection of the DTHR capabilities as an engineering and materials test reactor are provided and the DTHR's extrapolated performance to a commercial hybrid reactor application has been estimated.

This study has shown that with an aggressive, success oriented program, a DTHR is feasible for the late 1980's, and that a significant quantity of fissile fuel can be produced from fertile thorium using present day fission reactor blanket technology. A projected schedule for DTHR, including the existing and supporting programs as well as a possible path to commercialization is shown in Figure 2-1. This schedule does not include the specific RD&D needed for DTHR which is discussed in section 6.0.

TABLE 2-1

PRINCIPAL GOALS OF THE DTHR AND STUDY RESULTS

INITIAL GOALS (First 3 Years of DTHR Operation)

- Demonstration of the production of a significant amount of fissile fuel (~ 100 kg) in a proliferation-resistant form in a near-term, relatively low cost, fusion hybrid reactor.

Results: The DTHR blanket will produce ~ 165 kg of ^{233}U per year using boiling water cooled, fertile thorium oxide fuel rods. The DTHR plant could be operational before 1990. The reactor equipment cost is $\sim \$700$ M.

- Demonstration of reactor technology, hardware and performance that can serve as proof-of-principle for future commercial applications (hybrid or pure fusion).

Results: The DTHR will provide proof-of-principle for the design, manufacture, and operation of the following major systems:

- Commercial size (6.4 m x 9 m bore) superconducting coils with a 12 tesla peak field.
- Commercial size superconducting PF coils in large, external geometries and large amp-turn ratings (5 to 10 MA-T).
- Commercial size, plasma environment and operation of the bundle divertor plasma exhaust system.
- Typical annual commercial neutron fluence levels (after 3 years in DTHR) on vacuum vessel and blanket materials ($\sim 8 \times 10^{25} \text{ nm}^{-2}$, $E_n > 0.1 \text{ MeV}$).
- Commercial size plasma instrumentation, operation and control.
- Near commercial size, environment and operation of tritium processing, cleanup and fueling systems.
- Blanket structural design and performance data base in a neutron flux of $\sim 4 \times 10^{18} \text{ nm}^{-2} \text{ s}^{-1}$ ($E_n > 0.1 \text{ MeV}$).

TABLE 2-1 (Continued)

PRINCIPAL GOALS OF THE DTHR AND STUDY RESULTS

INITIAL GOALS (Continued)

- Demonstration of long pulse plasma operation to assist in determining meaningful duty cycle factors for commercial hybrid and pure fusion applications.

Results: The DTHR was designed to operate with the pulse "on" for 70 seconds during limited duration test periods (< 1 month) if 15 seconds "off" time is provided (duty cycle = 0.82). The initial test operation was designated as 40 seconds on, 40 seconds off for a 0.5 duty cycle. General testing was assumed to occur on the day shift with data analysis and maintenance during the evening and night shifts.

- Demonstration of blanket-fuel remote handling technology to determine a basis for plant availability projections.

Results: The nominal plant availability for initial test operation was established as 0.4. This will permit blanket refueling tests to establish realistic time periods and techniques using any of the five test stations. Advance blanket fueling concepts can be evaluated in subsequent tests as more sophisticated remote handling equipment is developed.

SUBSEQUENT GOALS (Third Year through 20 Year Life)

- Operation of the plasma in the ignited mode.

Results: The DTHR plasma concept was developed to enable operation in the ignited mode. The required τ_E based on an energy balance assuming $Z_{eff} \simeq 1$ is slightly less than the empirical τ_E .

- Demonstration of tritium fuel breeding.

Results: Blanket test areas are available in five locations for blankets up to 1 m thick on the outer torus. The inner torus shield area could be adapted to test blankets up to 0.35 m thick. The tritium breeding ratio for DTHR is potentially 0.6 to 0.8 without using the test blanket areas.

TABLE 2-1 (Continued)

PRINCIPAL GOALS OF THE DTHR AND STUDY RESULTS

SUBSEQUENT GOALS (Third Year through 20 Year Life) (Continued)

- Evaluation of the potential for breakeven or net electrical power generation and/or process heat capability for future applications.

Results: The DTHR blanket concept has a maximum BOL thermal power of 280 MW using the 27 test blanket bundles. By adding power conversion equipment and operating in the ignited mode, breakeven to ~ 7000 kWe net electrical power generation could be demonstrated.

- Development of alternate fusion blanket concepts.

Results: One entire test station ($\sim 1/8$ of the outer torus periphery) could be designated for advanced blanket concept testing of modules up to 1 m thick and ~ 7 m curved length.

- Extended operation as a materials and engineering test reactor.

Results: A test station comprised of $\sim 1/16$ of the outer torus ($\sim 18 \text{ m}^2$ surface) should be designated for materials testing. An integrated wall loading of 0.4 MW-yr/m^2 would exist at the blanket front wall position. The equivalent neutron fluence of $\sim 3 \times 10^{25} \text{ nm}^{-2}$ ($E_n > 0.1 \text{ MeV}$) after one year of exposure provides radiation damage values of 5 DPA's and He production of 60 appm for a 316 SS material plate. Numerous plasma engineering tests well beyond the TFTR capabilities could be conducted.

Figure 2-1. Key Elements of the Hybrid Reactor Development, Demonstration and Commercialization Program.

The reactor equipment was estimated to cost about \$700 M. These preliminary cost estimates indicate that an independent DTHR plant could be designed and constructed at a total Plant and Capital Equipment (PACE) cost of $\sim \$1.8$ B. Although this represents the largest concept presently perceived necessary to develop and prove the principal goals of the fusion hybrid reactor concept, a concentrated design effort may significantly reduce the reactor size and cost with little sacrifice in system performance. The major radius, $R_0 = 5.2$ m, of this DTHR concept was dictated by the relatively unoptimized (second iteration) bundle divertor space requirements.

2.2 FUSION DRIVER CONCEPTUAL DESIGN AND PERFORMANCE

The DTHR conceptual design consists of a magnetically confined (tokamak) fusion reactor fitted with a fertile thorium blanket. The fusion driver concept is based on a beam driven plasma, but at sufficiently high plasma densities that neutrons originating from the interactions of bulk plasma ions contribute significantly to the fusion neutron wall loading. The concept and parameter selection for the DTHR fusion driver were based on near-term technology and minimizing development requirements. A beam driven hybrid has the advantage of significantly relaxed plasma conditions (nearer-term) relative to an ignition reactor. However, the ignited or near-ignition option was considered in the initial plasma size selections. Considerable effort was made to remove unnecessary system complications to simplify the design within the constraint of effecting a meaningful and convincing demonstration reactor. PLT has already approached most of the DTHR plasma conditions, being within a factor of three of the plasma temperature and a factor of ~ 10 of $n_e \tau_E$. TFTR is expected to serve as a plasma prototype for DTHR with results available in the early 1980's.

The tokamak has a major radius of 5.2 m, a minor radius of 1.2 m, and plasma elongation of 1.6. All of the magnetic coil systems are superconducting Nb_3Sn based on the Large Coil Project (LCP) technology. The toroidal field (TF) coils employ an innovative concept, the "compact D" configuration. Although similar to the constant tension D concept, an hourglass shaped buckling cylinder is used to reduce the overall height of the TF coils. An

engineered bundle divertor concept has been developed based on the bundle divertor design techniques developed for TNS and ISX-B. A thermal power of 150 MW of 175 keV deuterium is injected into the plasma through six ducts of a positive ion, neutral beam injection system (NBIS). A water cooled, 316 stainless steel vacuum vessel concept was developed. The fusile fuel handling system assumed pellet injection and minimal reprocessing equipment. The fusion driver provides a neutron first wall loading of 2 MW/m^2 for fissile fuel production in the blanket. The plasma fusion power is 900 MW (neutron power = 720 MW) from a 17 keV plasma, which has an $n_e \tau_E$ of $\sim 6 \times 10^{19} \text{ m}^{-3} \text{ s}$. Some of the key fusion driver design features are shown in Table 2-2. Other DTHR System features and parameters are highlighted in Table 2-3. A trimetric view of the DTHR is shown in Figure 2-2.

2.3 DTHR BLANKET CONCEPTUAL DESIGN AND PERFORMANCE

A primary objective of the Demonstration Tokamak Hybrid Reactor (DTHR) is the demonstration of the successful integration of a number of major technologies that include a hybrid blanket which has the capability for breeding a substantial amount of fissile fuel. A low temperature, low pressure, water-cooled blanket was selected in March, 1978 as a baseline concept. The blanket was designed for fissile breeding with no potential for electrical power production. It is anticipated that economic Commercial Tokamak Hybrid Reactors (CTHR's) will need to produce not only sufficient electrical power to support the reactor plant, but could require substantial net electrical power to offset high capital cost financing. Consequently, in the latter period of FY '78 blanket design efforts and analyses were carried out with the objective of developing a structurally realistic blanket concept that has the potential for electrical power production. This resulted in a blanket concept that is capable of operating at relatively higher coolant temperatures and pressures. The DTHR blanket concept that evolved and its potential performance in a commercial application are discussed in detail in Reference 2-2.

An extensive assessment of the application of fission technology to the DTHR hybrid reactor blanket was made. Criteria relating to high operational reliability, minimal fabrication development requirements, simple design characteristics and applicability to commercial operation were established. Several

TABLE 2-2
DTHR FUSION-DRIVER DESIGN FEATURES
(September 1978)

Plasma Major Radius	5.2 m
Plasma Minor Radius	1.2 m
Plasma Elongation	1.6
Toroidal Field on Axis	5.5 T
Plasma Current	5.1 MA
Plasma Temperature (Avg.)	17 keV
$n_e \tau_E$	$6 \times 10^{19} \text{ sm}^{-3}$
First Wall Neutron Loading	2.0 MW m^{-2}
Fusion Power	900 MWt
Neutral Beam Energy, Power (Max.)	175 keV, 150 MW
Plasma Pulse Duration, Period (Max.)	70 s, 85 s
Duty Factor x Plant Availability (Annual)	0.20
Vacuum Vessel	Water-cooled 316 SS
Plasma Exhaust	Bundle Divertor
Divertor Particle Collection	On-Line Regenerable ZrAl Getter System & Cryopanel
TF, PF and Divertor Coil Conductors	Superconducting Nb_3Sn
Coil Conductor Cooling	Forced Flow Supercritical Helium
TF Coil Number	16
Peak Field at TF Windings	11.4 T
PF Winding Location	External to TF Array

TABLE 2-3
OTHER DTHR SYSTEM FEATURES AND PARAMETERS
(SEPTEMBER 1978)

<u>PLASMA OPERATION</u>		<u>NEUTRAL BEAM INJECTION LINES</u>	
Plasma Pulse Duration, s	70 Max., 40 nominal	Beam Power, Energy	150 MW, 175 keV
Effective Burn Duration, s	63 Max.	Single Arm Specifications:	
Recycle Time, s	15 Min., 40 nominal	Sources	8-110 A Positive Ion
Design Basis Duty Cycle Factor	0.82	Vacuum Enclosure	8.0 m long x 3.5 m wide x 4.1 m high
Average Duty Cycle Factor	0.50	Number of Arms	6
Design Basis Plant Availability	0.40	Vessel Aperture Per Line	1.4 m x 0.8 m
Design Basis Number of Pulses/Year	10 ⁵		
Pulse Scenario:			
Pulse OH (Bipolar)	2 seconds		
Neutral Beam Injection (Heating to Steady State)	5 seconds		
Burn (with NBI)	63 seconds		
Shutdown	2 seconds		
<u>TOROIDAL FIELD COILS</u>		<u>PLASMA EXHAUST SYSTEM AND IMPURITY CONTROL</u>	
Number of coils	16	Bundle Divertor	Single Toroidal Position at Torus Mid-plane. Superconducting Nb ₃ Sn Circulating Coils, On-line Regenerable Zr Al. Gettering, Cryopanel.
Conductor Type	Nb ₃ Sn	Throughput to Divertor Burial Chamber	3.6 x 10 ²² particles/s (1080 Torr μ /s)
Coil Shape	Compact "D"		
Peak Field at Conductor	11.4 T		
Current per Turn	15 kA		
Average Current Density	4.0 kA/cm ²		
Cooling Mode	Forced Flow Supercritical He		
Structural Material	Stainless Steel		
<u>POLOIDAL FIELD SYSTEM</u>		<u>TRITIUM SYSTEMS</u>	
Conductor Material	Nb ₃ Sn	On-site T ₂ Inventory	One Day
Location	Outside TF	Tritium Storage	Uranium Chip Beds
OH Flux Swing	53 v-s	Fuel Recovery-Recycling Plant	Off-site
Conductor Current	16 kA	Tritium Cleanup Systems	Reactor Torus and Cell
Average Current Density	4.4 kA/cm ²	Fueling	Prefill: Gaseous T Refueling: T Pellets & D-Beams and Pellets
Cooling Mode	Forced Flow Supercritical He		
<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	Wall Thickness (Shell)	0.4 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 H ₂		
Facility Air Conditioning	Fixed Base Load, Reactor Cell Air Conditioning

LEGEND

1 Plasma	7 Toroidal Field Coils
2 Vacuum Vessel	8 Inner Shield
3 Bundle Divertor	9 Outer Shield
4 Neutral Beam Injectors	10 Blanket Module
5 Vacuum Pumps	11 Cooling Headers
6 Poloidal Field Coils	12 Support Structure

Figure 2-2. Trimetric View of the Demonstration Tokamak Hybrid Reactor

concepts were evaluated and the reference concept using fertile thorium (ThO_2) fuel rods clad in Zircaloy 4 based on the LWBR (Shippingport) blanket fuel rods was selected. Even with blanket modules placed only at easily accessible locations, a fissile production rate of 165 kg per year with concentrations of ^{233}U increasing at a rate of 0.12% per year of hybrid irradiation was achieved. The blanket design features are listed in Table 2-4. The reference DTHR blanket concept has vertical modules located around the outer torus with the blanket elevation positioned to align with the maximum neutron wall loading. The rectangular module with a hemispherical front wall consists of vertical, LWBR blanket type fuel rods cooled by two-phase (boiling) water in a cross-flow (radial in-flow) configuration.

The blanket bundles are composed of four individual blanket modules which are secured together by a structural framework so that the complete module can be installed and subsequently removed from the reactor as a unit. The rectangular geometry allows blanket bundle removal by relatively simple, basically vertical lifting (or lowering) between the TF coils. Shield plugs are provided to enable access to the blanket bundles. The module surface closest to the plasma is semi-cylindrical. The outside rectangular dimensions are approximately 23 cm wide, 53 cm deep and the radius of the semi-cylinder front section is ~ 12 cm, which results in a total depth of 65 cm. The vertical height of the individual blanket module is 5.7 m, which is made up of 3.7 m of fuel and the remainder being the fuel rod gas plenum, inlet and outlet manifolds, and the upper and lower support structures. The equivalent poloidal blanket coverage factor b_p , is 43%, i.e., 43% of the fusion neutrons pass through this area. Advanced blanket modules could cover the full curved height of the outer surface of the torus, increasing the poloidal coverage to 67%. When four of the reference DTHR blanket modules are grouped together to form a module bundle, the horizontal cross-sectional space is approximately 81 cm wide by 95 cm deep (this includes the outside structural supporting members). The module bundles will be located in one group of three and four groups of six around the circumference of the torus with one module bundle being located in the clear opening between two toroidal field coils and the two adjacent module bundles being located one on each side and slightly under the toroidal field coils. The resulting blanket toroidal coverage factor, b_t , is 56%.

TABLE 2-4

DTHR BLANKET DESIGN FEATURES

(SEPTEMBER 1978)

BLANKET LAYOUT

MODULE CONSTRUCTION	RECTANGULAR/ROUND-NOSE, 316 STAINLESS STEEL
WIDTH OF BLANKET MODULE, m	0.23
THICKNESS OF BLANKET MODULE, m	0.65
OVERALL THICKNESS OF FUELED ZONES (3), m	0.53
OVERALL HEIGHT OF BLANKET MODULE, m	5.7
HEIGHT OF FUELED REGION, m	3.7
FUEL FORM	ThO ₂ PELLETS
FUEL CLAD MATERIAL	ZIRCALOY 4
CLAD THICKNESS, cm	0.065
FUEL ROD DIAMETER, cm	1.45
PITCH TO DIAMETER RATIO	1.1
BLANKET COMPOSITION, %	
FUEL	45.0
CLAD	9.5
COOLANT	31.0
S.S. - 316	13.5
VOID	1.0

BLANKET OPERATING CONDITIONS

(AT 1.2 MW-Yr/m ² NEUTRON EXPOSURE)	
AVERAGE BLANKET POWER DENSITY, W/cm ³	15
PEAK BLANKET POWER DENSITY, W/cm ³	55
PEAK FUEL ROD POWER DENSITY, W/cm ³	100
MAXIMUM WATER MASS VELOCITY, g/cm ² -s	6
INLET COOLANT PRESSURE, BARS	33.3
INLET COOLANT TEMPERATURE, °C	222
OUTLET COOLANT TEMPERATURE, °C	237
INLET WATER SUBCOOLING, °C	20
HOT ROD CLAD SURFACE TEMPERATURE, °C	250
HOT ROD MAXIMUM CLAD TEMPERATURE, °C	265
HOT ROD MAXIMUM FUEL TEMPERATURE, °C	655
DNB RATIO	3.2
COOLANT PRESSURE DROP THROUGH THE BLANKET, BARS	2.4
COOLANT PUMPING POWER TO BLANKET THERMAL POWER, %	0.07
AVERAGE COOLANT DENSITIES, g/cm ³	
EXIT PLENUM	0.021
FIRST ZONE (PLASMA SIDE)	0.032
SECOND ZONE (MIDDLE)	0.080
THIRD ZONE (SHIELD SIDE)	0.513

BLANKET PERFORMANCE

	AT BOL	AFTER 3 YEARS
EXPOSURE, $\frac{\text{MW-YRS}}{\text{m}^2}$	—	1.2
F, NET NO. ^{233}U ATOMS FUSION NEUTRON	0.62	0.58
M, ENERGY MULTIPLICATION FACTOR	1.6	2.7
ENRICHMENT, w/o ^{233}U IN Th		
FIRST ZONE	—	0.51
SECOND ZONE	—	0.38
THIRD ZONE	—	0.25
BLANKET AVERAGE	—	0.36
SHIELD ABSORPTIONS ^(a) , $\frac{\text{NEUTRONS}}{\text{FUSION NEUTRON}}$	0.60	0.64
G, $\frac{\text{kg } ^{233}\text{U}}{\text{YEAR}}$ PRODUCTION RATE	170	160
CUMULATIVE ^{233}U , kg	—	495 (165/yr)
PEAK BLANKET POWER, MW _t	342	619
AVERAGE BLANKET POWER ^(b) , MW _t	280	500
AVERAGE ELECTRICAL POWER OUTPUT ^(c) , MW _e	84	150

(a) Neutrons available for tritium production; potential tritium breeding ratio without use of multipliers.

(b) Plasma duty cycle averaged at 0.82 for short test periods.

(c) Assuming power conversion equipment added with 0.3 efficiency in relation to note (b).

The blanket module (pressure vessel/fuel bundle assembly) design has three zones, each zone having a radial depth of 14 cm, allowing 10 rows of fuel rods to be accommodated per zone. A triangular pitch of 1.6 cm was used. Overall, each zone has 120 rods with a total of 360 rods per blanket module. Pressurized water (~ 33 bars) at a subcooled inlet temperature ($\sim 220^\circ\text{C}$) is supplied to the back of the module and exits in the inner cylindrical nose of the module at saturation temperature. The exit steam plenum design is constructed from solution annealed 316 stainless steel with the split cylindrical components welded to form the bullet-like cross-section. The three rectangular fuel zone ducts are made from 20% cold worked 316 SS material. The pulsed operation of the hybrid results in cyclic stresses in these module structures. Preliminary analyses indicate that stress and fatigue limits are not exceeded for the planned $1.2 \text{ MW}\cdot\text{Yr}/\text{m}^2$ exposures (3 years maximum irradiation period). Further analysis indicates that design margin will exist for the module structure in a longer life, commercial application ($8.4 \text{ MW}\cdot\text{Yr}/\text{m}^2$), based on extrapolated materials data.

The fuel rod concept is based on the LWBR blanket fuel rod design except that the rod is longer. The overall outside diameter of the fuel rod is 1.45 cm. The clad thickness is 0.065 cm with a helium gap of 0.008 cm between the thorium oxide (ThO_2) fuel and Zircaloy-4 clad. The pitch to diameter ratio is 1.1. The fuel rods are supported at the top, with the vertically spaced grid assemblies (60 cm apart) and flow separators (centered between grids) providing fuel rod stability. The grid assembly concept is similar to the LWBR concept in that it is of light construction to minimize neutron capture. Preliminary structural analysis indicates that the fuel clad margin of safety for crack growth is ~ 100 times larger than the module wall margin and thus the clad is not design limiting during a 3 year test period.

A simple, direct steam cycle system was assumed for the initial evaluation of the DTHR. The overall efficiency of this system for electrical energy generation was estimated to be ~ 0.3 . Extrapolation of the DTHR concept to a commercial application indicated that one hybrid could provide make-up fuel for nearly 5 present day type PWR's (1000 MWe per unit) per year using the chemically separable $\text{Th}-^{233}\text{U}$ fuel cycle and 3 to 4 PWR's (1000 MWe) per year

using the denatured Th- ^{233}U fuel cycle. This was based on a preliminary, first iteration hybrid producing 1200 kg per year of ^{233}U with an average concentration of 0.3 w/o (per year in the hybrid) in the ^{232}Th . The tritium breeding ratio was projected to be ~ 0.6 in the DTHR using areas not designated for blanket testing (inner and outer shields, etc.). It should be noted that these performance values are based on the simpler blanket concepts that do not include use of neutron multipliers. The performance should be improved by such additions.

Core calculations and safety analyses were made using thorium fuel enriched with hybrid bred ^{233}U in a standard pressurized water reactor (PWR) utilizing the oxide fuel form. Highly satisfactory performance is predicted. The make up fissile fuel required is less than for the standard fuel cycle. No safety, performance or capital cost penalties have been identified. An average conversion ratio of 0.74 for the chemically separable fuel cycle and 0.78 for the denatured fuel cycle were determined as compared to the present day PWR designs operating on the U-Pu fuel cycle which average 0.60. The hybrid must supply 255 kg/yr ^{233}U for the PWR on the chemically separable fuel cycle and 340 kg/yr ^{233}U for the denatured fuel cycle. It was determined that by replacing the boron silicated glass burnable poison pins with LiF pins in the PWR's supplied by the hybrid, the PWR's ($\sim 7, 1000 \text{ MWe}$) on these fuel cycles could produce as much as 38% of the tritium burned by the hybrid. This combined with a time averaged (3 years) tritium breeding ratio of 0.62 in the hybrid would result in a nearly adequate tritium fuel supply.

2.4 DTHR PLANT ECONOMICS

The DTHR conceptual design was developed in sufficient general detail to enable estimation of the Plant and Capital Equipment (PACE) cost using the COAST Code⁽²⁻³⁾. The reactor concept was selected on the basis of minimal equipment and systems to achieve the design and operating goals. However, the DTHR plant was assumed to be a completely independent facility, constructed at a government owned site near a large power source ($\sim 1700 \text{ MW}_e$ maximum). Figure 2-3 shows a typical commercial hybrid reactor plant layout. The DTHR plant would be similar to this layout once the power conversion equipment

Figure 2-3. Typical Commercial Hybrid Reactor Plant Layout

was added. Based on the preliminary level of design detail available, the plant and equipment capital costs were estimated conservatively by assuming facility construction and equipment installation and assembly each at 15% of the total equipment cost, 2% spare parts allowance, 40% EDIA and a 20% contingency on all of the preceeding items. This resulted in a total PACE cost estimate of about \$1,800 M. The reactor equipment was estimated to cost \$702 M. The annual operating cost was determined to be \$269 M based on the beam driven plasma operating mode and procurement of all electricity and tritium needs. These values therefore represent maximum cost projections for the DTHR plant and operation.

The cost advantages of adding power conversion equipment to enable generation of some electrical power and addition of a direct energy conversion (DEC) system to the neutral beam injectors was evaluated. With the partial blanket and ThO_2 fuel DTHR concept, the blanket thermal power was intentionally minimized to reduce cooling tower costs. Thus, the average power that could be produced was only 70 MW_e . This resulted in the capital cost increase (\$37 M) to achieve an operating cost reduction (\$7 M/yr) being economically marginal; hence the decision to add this equipment would have to be made on a technical basis. The DEC equipment capital cost increase of \$155 M would reduce power requirements by $\sim 300 \text{ MW}_e$ and the operating cost would decrease by \$35 M/yr. This equipment could be justified if long term (5 - 20 years) beam driven operation of DTHR was assumed. A more attractive economical alternative would be ignited plasma operation, or at least very near-ignition operation (plasma energy gain > 15).

If the goals for DTHR were modified to take advantage of operation in the ignition mode, the net power consumption could be reduced by a factor greater than 10, the total capital cost reduced $\sim$ \$264 M and the net operating cost reduced $\sim$ \$100 M per year. The capital cost is reduced by \$85 M due to the neutral beam injector quantity reduction and the remaining costs due to power control equipment, cooling and shielding components, and the reduced size of the plasma fuel handling and exhausting systems. In the ignited mode of operation, an appropriately sized power conversion system addition would

increase capital cost by \$47 M and reduce the operating costs by \$7 M. Thus, this equipment would probably have to be justified on a technical basis rather than an economical one. However, an appropriately sized DEC system would increase the capital cost by \$101 M and only reduce the operating cost by \$8 M per year, which could not be readily justified.

The procurement of tritium would represent a significant operating cost for DTHR, since tritium is assumed to have limited availability at that time (early 1990's). Therefore, the addition of a tritium breeding blanket and tritium reprocessing equipment at the DTHR facility appears to have the potential of reducing the fusile fuel cost by $\sim 100\%$. (A rough estimate assuming tritium at 75¢/Ci indicated the operating cost could be reduced from $\sim$ \$72 M/yr to \$36 M/yr if DTHR bred half the required tritium.) Significant changes in operating costs would also be achieved if a high thermal power fuel and blanket concept were selected. Detailed analyses of these options would be beneficial in future studies.

2.5 DTHR KEY DEVELOPMENTAL ISSUES

An assessment of the pre-operational developmental effort required to support a DTHR plant construction was made. The major areas requiring development work not already in progress at a national laboratory or specific to DTHR are:

- Vacuum vessel construction and servicing details.
- Bundle divertor system prototype component testing.
- TF coil prototype fabrication qualification and testing including cryogenic cooling system components.
- PF coil prototype construction and testing under pulsed operations, including power conversion and energy storage equipment.
- Neutral Beam Injection System components and development of a prototype direct energy conversion system.
- Tritium injection prototype and system components.
- Special I & C equipment and systems.

- Blanket module mockups for two-phase flow distribution, fuel rod stability and materials corrosion testing.
- Remote servicing mockups and fuel handling equipment.
- Safety and environmental research (on licensing concerns, etc.)
- Material technology data base preparation.

It is estimated that these RD&D programs will cost \$165 M over a period of 7 to 10 years. Most of the key developmental areas relating to the blanket design will constitute the early post-operational testing in the DTHR and include the following:

- Fuel and clad response to the cyclic fusion neutron pulses: fatigue, thermal stress, structural deformation.
- Radiation effects: radiation-induced creep, swelling, embrittlement, helium gas generation.
- Coolant-clad interactions: corrosion, erosion, fretting, chemical enhancement of fatigue crack initiation, fuel loss.

The engineering and materials testing capabilities of the DTHR are presented in Section 5.0 which discusses how the required data base for future reactor designs could be obtained in the DTHR environment.

2.6 DTHR EXTRAPOLATION TO COMMERCIAL HYBRID REACTOR APPLICATIONS

The feasibility study of the DTHR was conducted with the objective that the concept and performance should allow extrapolation to a Commercial Tokamak Hybrid Reactor (CTHR). This was achieved by a three-step process. First, the reference DTHR concept performance was projected assuming a full outer torus blanket (57% coverage versus 24% for DTHR), a 24 hour operation (plant availability increases from 0.4 to 0.7 since DTHR was assumed to operate about 8 hrs/day), a longer pulse duration (135s minimum vs. 70s max. on DTHR, i.e., duty cycle increases to 0.9), an ignited plasma operating mode (reduces operating power demand by a factor > 2), and a power conversion system. None of these assumptions require scientific or engineering advances that could not be developed and proven with the DTHR facility.

The second step consisted of adopting the more standard UO_2 fuel form. This was done to enable use of existing fuel cycle costing rates and correlation to present day nuclear power plant fuel cycle scenarios. The fuel change also resulted in more than double the thermal power production and, therefore, increased electricity was produced for sale to offset the cost of capital charges in the operating costs. The fissile fuel production was nearly doubled. The economic performance of this concept using PWR fuel rods with no fuel reprocessing was analyzed⁽²⁻⁴⁾. Two basic alternatives which provided maximum coupling between the fusion hybrid reactor and conventional PWR's, yet represent simple, nearer-term applications, were examined. These alternatives were hybrid re-enrichment ("refresh") of spent PWR fuel rods and hybrid enrichment of new fertile (natural U or tails) fuel rods for direct use in a subsequent PWR burn cycle. The results were based on first iteration neutronic and economic parameters and cost normalized to a 1978 startup for the commercial hybrid reactor plant. A levelized PWR busbar electricity cost was developed assuming a typical 1200 MWe, nuclear power plant going on-line in 1978. The levelized PWR capital cost charge (20 mills/kW hr) was based on a \$700 M nuclear plant cost and standard utilities cost levelizing techniques. All of the cases studied including the conventional PWR reference assumed a "throw-away" fuel cycle for the spent PWR rods after a 33 Gwd/THM burnup in the PWR. The fuel was assumed to be owned by another utility with the hybrid owner providing a service at a fixed charge (similar to the present isotopic enrichment costing method). Consideration of irradiation damage effects on the fuel rods was deferred until an economic incentive for adopting this cycle was established.

Hybrid irradiation of spent PWR fuel rods in the "refresh" concept resulted in ~ 50% lower PWR fuel cycle costs and ~ 20% lower PWR power generation costs than those for the conventional PWR enrichment scenario. Hybrid irradiation of new fuel rods manufactured from tails or natural uranium would increase PWR fuel cycle costs by ~ 25% to 50% and power generation costs by ~ 9% to 18% above the reference PWR costs. The results of this analysis were reported in Reference 2-4. The summary is shown on Table 2-5. A reasonable range of parameter variations were considered and the sensitivities developed which showed the hybrid cost scenario varied from competitive to non-competitive compared to the conventional enrichment cost scenario. Figure 2-4 shows a typical plot of these sensitivities. The conclusion of the analysis was that

TABLE 2-5

HYBRID ENRICHMENT COST EFFECT ON PWR BUSBAR ELECTRICITY COST
(SCOPING ANALYSIS)

	Present PWR Cycle	Privately Owned Hybrid		
		U Tails	Natural U	Spent Rods
Plant Capital Cost, 10^6 \$ (1978 Startup)	~700	2200	2250	2325
\$/kWe	580	1080	1020	880
Rated Electrical Power, MWe	1200	2030	2210	2640
Required Enrichment, w/o fissile	3.2	3.9	3.9	5.0
Hybrid Heavy Metal Production Rate/ T/yr	----	62.7	68.5	52.2
Hybrid Enrichment Cost, \$/kg Hm	----	1660	1160	345
Levelized Cost of Delivered PWR Fuel, \$/kg HM	2045	3215	2600	1205
Levelized PWR Fuel Cycle Cost, \$/kg HM	2760	4300	3520	1400
mills/kW hr	12	18	15	6
Levelized PWR Busbar Cost, mills/kW hr	32	38	35	26

Figure 2-4. Typical Hybrid Enrichment Cost Sensitivities to Major Variables — Spent Rod Direct Fuel Cycle.

the PWR busbar cost of electricity using fuel enriched by the hybrid would bracket the busbar cost using conventional nuclear fuel. The fact that these results indicated comparable PWR busbar costs using hybrid enriched fuel was encouraging since many aspects of the study were generally unfavorable to the hybrid option:

- The hybrid blanket lattice was comprised of UO_2 fuel rods designed for PWR operation. More favorable hybrid enrichment performance could be obtained using blanket fuel elements designed specifically for the hybrid (e.g., metal alloys, UC, U_3Si , etc.). Some degree of reprocessing of hybrid blanket fuel elements would be required, of course, with such an approach.
- The buildup of fission products represented a significant penalty to the hybrid enrichment scenarios since appreciably more fissile material was required (5.0 w/o enrichment for the refresh case, 3.9 w/o enrichment for hybrid irradiation of new fuel rods). Here again, some degree of reprocessing would avoid this penalty.
- The combined PWR-Hybrid-PWR burnup for the direct fuel cycle would be nearly three times the normal 33 GWd/T PWR cycle and the fluences seen by the fuel rods were estimated to be an order of magnitude larger than present PWR exposures. This prompted concern regarding fuel rod integrity, particularly during the long (10 year) exposures in the hybrid.
- Private, rather than public, financing of the hybrid was assumed. The comparison penalizes the hybrid in this regard since enrichment costs for the conventional PWR fuel cycle are based on publicly financed isotopic enrichment facilities.

The third step in the extrapolation toward a CTHR included assuming a larger tokamak ($R_0 = 6\text{m}$) with a full blanket (inner and outer torus coverage), a more favorable commercial plasma burn time (20 minutes), and a more advanced fuel (UC). This resulted in significant increases in both electricity and fissile fuel production. Preliminary trade-off studies with the COAST code had indicated a linearizing effect in cost improvements occurring after the major radius exceeded 6.0 m. The design electrical power of the 6 m CTHR concept was 2500 MWe, which was felt to be a practical maximum for an early twenty-first century concept. Superconducting 2000 MWe generators are presently projected to be available by the turn of the century. (Cost improvements projected

for these generators were not considered in this economic analysis.) The extrapolated CTHR concept eliminated most of the unfavorable conditions found in the previous commercial concept. The UO_2 was replaced with UC, which resulted in a 4-year hybrid enrichment period (to reach 3 w/o enrichment) rather than ~ 10 years. This reduced hybrid enrichment charges as well as the fuel carrying charges during the hybrid enrichment period. Reprocessing was included. This removed the undesirable fission product buildup and the concern for the fuel rod integrity. The privately owned hybrid plant is the only unfavorable aspect retained in this analysis. This added conservatism to the cost data, since it is based on the very cursory extrapolation from the DTHR data. The uranium tails case was analyzed since it required a longer hybrid enrichment period than natural uranium would require. The hybrid owner was assumed to own a full fuel load to assure sustained operation of the CTHR plant. This increased capital costs as did the full blanket coverage and tritium breeding self-sufficiency assumptions. The resulting cost of electricity at the nuclear plant busbar using hybrid enrichment was 40 mills/kW hr.

These very preliminary economic analyses show that: (1) the DTHR performance can be extrapolated to commercial operations; (2) the results can be very sensitive to the groundrules assumed for both the technical concept and the economic analysis; (3) the competitiveness of the hybrid enrichment process clearly provides the incentive to develop a detailed CTHR concept to evaluate the economics of the various alternatives, since the hybrid reactor is a net energy producer (e.g., 2500 MWe for the extrapolated CTHR concept) as opposed to an isotopic enrichment facility which is a power consumer (i.e., a Portsmouth Gaseous Diffusion Plant expansion would have consumed 2700 MWe; the gas centrifuge concept 105 MWe)⁽²⁻⁵⁾ and (4) the fusion hybrid reactor versatility continues to offer so many options that it is difficult to select a single representative scenario for studies with the normal limitations of schedule and funding. Therefore a major goal in continuing this effort is to reach a position enabling the selection of a representative point design for a commercial application which can be studied in detail.

The cost and performance data presented for both the DTHR and CTHR must be viewed as preliminary and representative of unoptimized conceptual designs. The CTHR extrapolations particularly are of a gross nature, consisting simply of extending the blanket coverage, duty factor and plant availability. Certainly much work needs to be done to establish a reactor size, blanket configurations and a design point more appropriate for commercial applications. The Westinghouse FY'79 hybrid program will be directed at the design and analysis of an integrated, first generation commercial hybrid plant which will provide a more appropriate basis for the evaluation of economic performance.

REFERENCES - SECTION 2.0

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3.0 DTHR FUSION DRIVER CONCEPTUAL DESIGN

The concept and parameter selection for the DTHR fusion driver were based on near-term technology, attempting to minimize development requirements. Therefore, a beam driven plasma was selected as the reference concept, since it has significantly relaxed plasma conditions relative to an ignited plasma. However, the initial plasma size was selected with consideration of the ignited or near ignition modes of operation. Considerable effort was made to remove unnecessary system complications to simplify the design within the constraint of effecting a meaningful and convincing demonstration reactor. PLT has already approached most of the DTHR plasma conditions, being within a factor of three of the plasma temperature, and a factor of ~ 10 of $n_e \tau_E$. TFTR is expected to serve as a plasma physics prototype for DTHR with results available in the early 1980's. The function of the fusion driver in the DTHR is to generate neutrons that will be used in the transmutation of fertile material to fissile fuel in a blanket surrounding the plasma. For DTHR, a flexible tokamak design has been developed that is based on a beam driven plasma, but at sufficiently high plasma densities that neutrons originating from the interactions of bulk plasma ions contribute significantly to the wall loading. Based on the present empirical transport scaling laws, it is clear that ignition could be achieved with the reference tokamak concept. In this sense, the DTHR could satisfy the goals of TNS in addition to those of the DTHR. Although the electrical power requirements and the amount of gas added to the system via the ion sources could be reduced by turning off the neutral beams if ignition were achieved, there would be a great advantage in being able to meet the neutron production goals without having to reach ignition. The operating conditions of the beam driven plasma would be easily controlled by the neutral beam power, and steady plasmas operating in new regimes could be studied without the restrictive prerequisite of achieving ignition.

3.1 PLASMA PARAMETERS

Two goals of the DTHR fusion driver design are to achieve a stable plasma under the required operating conditions and to produce a neutron wall loading in excess of 2 MW/m^2 . The blanket goal of producing 100 kg of fissile fuel per year could be met with a wall loading of $\sim 1.7 \text{ MW/m}^2$. Some guidelines that were used in defining the reference plasma are given in Table 3-1. The constraints given in Table 3-1 are not rigid, but were simply used as a guide. 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 the plasma minor radius, is 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.

TABLE 3-1
GUIDELINES FOR REFERENCE DTHR PLASMA DEFINITION

	<u>Guideline</u>	<u>Actual</u>
Stable Plasma		
Aspect Ratio	$A \leq 4.5$	4.33
Beta	$5\% \leq \beta \leq 8\%$	6.24
Poloidal Beta	$\beta_p \leq A$	4.33
Safety Factor	$q \geq 2.5$	2.5
α - Confinement	$I_p > 3 \text{ MA}$	5.1
Neutral Beam Penetration	$n_e a \leq 2 \times 10^{20} \text{ m}^{-2}$	1.2×10^{20}
Scaling		
Toroidal Field on Axis	$B_T \geq 5 \text{ T}$	5.5
Plasma Minor Radius	$a \geq 1 \text{ m}$	1.2

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, $B_T > 5T$ and $a \geq 1$ 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 95 cm thick inner shield, 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 3-2 were obtained. The COAST code⁽³⁻¹⁾ was used in establishing most of the design parameters after it was upgraded for inclusion of hybrid reactor considerations (COAST-4 option).

A neutral beam power of 150 MW of 175 keV-D₀ injected into the plasma has been selected. To enhance the wall loading during the intensely beam-driven mode of operation it is desirable to have as large a neutral beam power as possible. However, as a practical matter the neutral beam power must be limited. The greater the number of neutral beam arms, the smaller the blanket coverage which can be achieved. It was decided that for the reference, six beam arms would be used. Based on reasonable extrapolations from present day technology, eight sources can be used per beam arm to provide the 150 MW of injected power.

The required τ_E listed in Table 3-2 is based on an energy balance assuming $Z_{eff} \simeq 1$; the empirical τ_E is obtained from

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

where $s = (1 + \delta^2)/2$ 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 balance, that a higher plasma temperature will be achieved than is given in Table 3-2, or that the neutral beam power can be reduced.

TABLE 3-2
DTHR FUSION DRIVER PARAMETERS

	March 1978 Values Beam Driven	September 1978 Values	
		Beam Driven	Ignited
Major Radius, m (R_0)	5.2	5.2	5.2
Minor Radius, m (a)	1.2	1.2	1.2
Aspect Ratio (A)	4.33	4.33	4.33
Elongation (δ)	1.6	1.6	1.6
Plasma Volume, m ³ (V_p)	236.5	236.5	236.5
Toroidal Field on Axis, T (B_T)	5.5	5.47	5.45
Toroidal Field on Cond., T (B_{max})	12	11.69	11.66
Plasma Current, MA (I_p)	5.1	5.12	5.11
Neutral Beam Energy, keV (W_0)	200	175	185
Neutral Beam Power, MW (P_B)	150*	150*	55**
Plasma Energy Gain (Q)	6	6	∞
Plasma Temperature, keV (Avg., $\bar{T}$)	13	17	17
Plasma Profile	Uniform	Parabolic	Parabolic
Electron Density, 10 ²⁰ m ⁻³ (n_e)	1.5	0.99	1.06
Neutral Beam Penetration, 10 ²⁰ m ⁻² ($n_c a$)	1.8	1.2	1.2
Plasma Beta, % (β)	6	6.24	6.24
Poloidal Beta, % (β_p)	4	4.33	4.33
Wall Loading, MW/m ² (J_w)	2	2	2
Current Density (Source Neutrons) 10 ¹⁸ m ⁻² s ⁻¹	0.9	0.9	0.9
Integrated Wall Loading, MW - yr/m ²	0.4	0.4	0.4
Fusion Power, MW _t (P_f)	950	900	900
Required Confinement Time (τ_E), s	0.5	0.6	1.27
Empirical Confinement Time (τ_E), s	1.85	1.22	1.31
$n_e \tau_E$, 10 ¹⁹ m ⁻³ s	7.5	5.94	13.46
Empirical τ_E /Required τ_E	~ 4	~ 2	~ 1
Plasma Duty Cycle Factor	0.5	0.5	0.5
Plant Availability Factor	0.4	0.4	0.4

* Steady State
** Start Up

Since the collisionality is $\nu^* \sim 1 \times 10^{-3}$ for the reference geometry and plasma conditions, the energy confinement time may be expected to be determined by particle and energy transport due to trapped particle modes⁽³⁻²⁾. In this case $\tau_T \text{ IM} \sim 0.7 \text{ s}$, which is still longer than the confinement time required for energy balance. Furthermore, if profile effects are taken into account, a higher energy confinement time is predicted⁽³⁻²⁾. Table 3-2 shows this effect by comparing the March, 1978 confinement time, which neglected profile effects, to the September, 1978 confinement time ($\tau_E = 1.85$ vs. 1.22).

The DTHR duty cycle times plant availability is calculated to be 0.20 from Table 3-2. This is an average operational factor which is based on a normal duty cycle of about 40 seconds on and 40 off. Operation would normally occur in daytime hours with maintenance and data analysis at night, except for special, continuous test periods. Therefore, the initial plant availability was assumed to be $\sim 40\%$. However, DTHR systems would be designed to achieve a 70 second "on" pulse and 15 seconds "off" so that sustained, three shift operations could be tested for periods up to a month at a time.

3.2 REACTOR MECHANICAL ARRANGEMENTS

The fusion driver mechanical arrangements for the DTHR are typical of tokamak configurations, but with several innovative features. The plan view in Figure 3-1 shows the relative positions of the major components. The 16 TF coils are located on 22° centerlines with respect to each other except for the pair adjacent to the bundle divertor. These two TF coils were opened to 30° to provide sufficient room for the bundle divertor vacuum chamber to protrude to the periphery of the tokamak. The bundle divertor chamber is shown in Figure 3-1 with the cryopanel and outer shielding removed. The outer shield for the TF coils consists of 32 sectors, one under each coil and one between each TF coil pair. The inner shield consists of 16 sectors, one under each TF coil. The TF coils use a 50% wedging and 50% bucking cylinder load removal system. The nose sections of the TF coils are identical, with the extra gap at the 30° position being filled with an assembly wedge. This wedge can be mechanically expanded to remove all clearances between TF nose sections after

Figure 3-1. DTOR Plan View

assembly. An inner shield assembly block is located radially outward from this wedge. Smaller assembly blocks (not shown) can be located at each TF coil opening to permit ease of inner shield installation, since each shield must be slid under the TF coil prior to vacuum vessel assembly. The assembly and disassembly techniques for the reactor components mentioned in subsection 3.3.12 were discussed in more detail in Reference 3-3. Two of the neutral beam injectors (NBI) are located adjacent to the bundle divertor to permit common shielding between these components and minimize loss of blanket module positions. The other four NBI's are located symmetrically around the periphery of the tokamak relative to the bundle divertor's location.

Figure 3-2 shows the cross-sectional view of the DTHR. The NBI duct is shown entering on the horizontal midplane in front of the bundle divertor, which is also positioned at the horizontal midplane. The outer poloidal field (PF) coils are positioned between the bundle divertor and the vertical removal envelope for the blanket modules. The support structures are used to position and hold the inner six PF coils as well as the upper headers and the TF coils. The vacuum vessel "floats" on rails from the shield structures, which also support the blanket modules. The limiters are located at four positions inside the vacuum vessel and are supported by the vacuum vessel structural system. All of the components within the TF coils are supported through the shields, independent of the TF coil support system, through saddle girders and vertical support posts. These girders "saddle" the TF coils without contacting them. The shielding system for the TF coils is shown in Figure 3-3. The inner shield and the vacuum vessel support/shield transmit the inner components loads to the saddle girders and vertical support posts.

The relative locations of the ohmic heating (OH) coils, shaping field (SF) coils and bucking cylinder to the TF coils are shown in Figure 3-4. The TF coil nose section, which wedges near the tokamak bore, is shown by the darker cross-hatching as compared to the outer leg of these compact D coils. The outer legs of the TF coils are each enclosed in a dewar to maintain cryo-cooling of the superconducting coils. Figure 3-5 shows the dewar systems. The inner legs (nose section) of the TF coils have an inner dewar attached to the

Figure 3-2. DTHR Cross-Sectional View

Figure 3-3. DTOR TF Coil Shielding

Figure 3-4. DTHR TF Coils, PF Coils and Bucking Cylinder

Figure 3-5. DTHR Superconducting Coils Cryogenic Cooling System

plasma side of the coil. The OH coils and bore side of the TF nose section are enclosed in a common dewar that maintains the entire tokamak bore area at cryogenic conditions. This simplifies removal of cables and coolant lines from the OH coils in the bore. The superconducting cables are then terminated and the transition to copper bus lines occurs in the cryo-cooled enclosure below the bore area. This reduces the cryo-system complexity in the reactor compartment.

Descriptions of the various fusion driver subsystems such as the magnets, bundle divertor, vacuum vessel, shielding and structure are provided in the following subsections. The radial dimensions of the tokamak are presented in Table 3-3, which also defines the major component and cross-section thicknesses.

3.3 FUSION DRIVER SUBSYSTEM HIGHLIGHTS

The conceptual designs of the DTHR fusion driver subsystems and related supporting subsystems were developed and discussed in detail in Reference 3-3. The following subsections present the highlights of these reactor systems through brief descriptions, key figures and parameter tables. Most of the systems have not changed much except for the redesigned bundle divertor (see subsection 3.3.5), and the blanket modules, which were redesigned for power production as well as fissile breeding. The blanket development work is reported in Section 4.0.

3.3.1 TOROIDAL FIELD (TF) COILS

The toroidal field coils have a "compact D" shape structure with bolted construction and superconducting Nb_3Sn cables similar to that used in the Westinghouse Large Coil Project concept. The "compact D" is a shape that a constant tension (D) coil takes when supported by a bucking cyclinder with flared ends (hour glass) instead of a straight cyclinder. This shape reduces the coil (and reactor) height significantly. The maximum toroidal field, which occurs at the TF coil bore side of the inner leg, has been limited to ≤ 12 tesla. The TF coil parameters are shown in Table 3-4. The cross section of the TF coil shown in Figure 3-6 measures 100 cm long in the radial direction

TABLE 3-3
DTHR RADIAL DIMENSIONS

<u>COMPONENT/LOCATION</u>	<u>THICKNESS, cm</u>	<u>RADIUS, cm</u>
Device Centerline	0	0
Bore		100
OH Coils	25	125
Bucking Cylinder	19	144
TF Coil (Nose)	100	251
Dewar	15	266
Insulation	5	271
Assembly Gap	2	273
Inner Shield	95	368
Assembly Gap	2	370
Vacuum Vessel Wall	1	
Vacuum Vessel Structure	7	
Limiter	2	380
Scrape Off Layer	20	400
Plasma, a	120	
Major Radius, R_0		520
Plasma, a	120	640
Scrape Off Layer	20	660
Limiter Space Allowance	2	
Vacuum Vessel Structure	7	
Vacuum Vessel Wall	1	670
Assembly Gap	10	680
Blanket	65	745
Assembly Gap	35	780
Outer Shield	80	860
Assembly Gap	10	870
Insulation	5	880
Dewar	15	895
TF Coil	100	995
Dewar	15	1010
Insulation	5	1015

TABLE 3-4
DTHR TOROIDAL FIELD COIL PARAMETERS
(SEPTEMBER 1978)

<u>Parameter</u>	<u>Units</u>	<u>Value</u>
Plasma Major Radius	m	5.2
Vertical Bore (Inner)	m	9.1
Radial Bore (Inner)	m	6.4
Central Field	T	5.5
Maximum Field	T	11.7
Number of Coils	—	16
Field Ripple (at Plasma Edge)	%	0.8
Total Stored Energy	MJ	20,650
Stored Energy per Coil	MJ	1,290
Total Ampere Turns	MA-T	142
Ampere Turns per Coil	MA-T	8.9
Coil Cross-Section	m	1.0 x .67
Coil Circumference	m	~ 27
Superconductor/Stabilizer	—	Nb ₃ Sn/Cu
Conductor/Shape	—	Cable/Square
Current per Turn	kA	15
Conductor Area per Coil	cm ²	3000
Structure/Conductor Area Ratio ^(a)	—	1.5
Coil Construction	—	Pancake (plates)
Structural Material	—	Stainless Steel (310)
Cooling	—	Forced Flow Supercritical Helium
Coil Tension	MN	1,265
Centering Force (max. per coil)	MN/m	54
Coil Weight (tonnes per coil)		
Conductor		40
Structure		104
Dewar		27
		171
Operating Power Required	MW _e	23

(a) Over 0.56 x 0.67 x 1.0 cm area

Figure 3-6. TF Coil Nose Cross Section

by 67 cm thick and is made up of 22 stainless steel plates (bobbins). Eighteen of the plates contain ten conductor slots, two contain six conductor slots, and two contain four conductor slots.

A stainless steel triangular wedge is fitted to each side of the vertical section of the inner leg of the TF coil to complete the tapered nose section of the coil and fill the void between adjacent TF coils. Three bolts (center-lines shown in Figure 3-6) are used to hold the plates and wedges together. The centering force of the TF coil is reacted by the vertical face of the plates in contact with the bucking cylinder and the wedging action between adjacent TF coils. The wedge sections cannot extend around the outer TF coil leg because of space restrictions at the bundle divertor location.

Three Nb_3Sn superconducting cables, each with a square cross-section of 5 cm^2 (2.23 cm on a side) including the stainless steel sheath, will be installed in each plate slot making a total of 600 turns of superconducting cables per TF coil. The cables are cooled by helium coolant flowing through the voids within the conductor. The coolant is fed in and out of the cables at connector joints located at the top or bottom of the "D" shape coil. A liquid nitrogen cooling jacket is carried in the vacuum dewar as a buffer system to minimize refrigeration requirements for the helium cooling system. The preliminary specifications for the superconducting cables and helium coolant are tabulated in Table 3-5.

3.3.2 SHIELDING

Two design criteria were used for the DTHR TF coil shielding:

- a) total radiation heating load in TF coils $\leq 20 \text{ kWt}$ (based on a reasonably sized refrigeration system,
- b) total dose to TF coils $\leq 1 \times 10^8 \text{ rads}$ (based on conservative superconductor insulation damage limits).

A DTHR reactor operating at 0.5 duty factor and 0.4 availability was specified, giving an overall plant factor of 0.2. The shield concept was selected with a simple composition of 65% stainless steel and 35% water loaded with natural

TABLE 3-5

SPECIFICATIONS OF Nb₃Sn-Cu SUPERCONDUCTOR FOR THE TF COILS

Average Current Density*, A/cm ²	3,000
S/C Operating Current Density, A/cm ²	80,000
Area of Conductor + He, cm ²	3.05
Superconductor Area**, cm ²	0.1875
No. of Strands, Active (3 x 3 x 3 x 3 x 6)	486
Non-Active	81
Strand Diameter, cm	0.07
Conductor Hydraulic Diameter, cm	0.06
Conductor Void Fraction	0.3
No. of Slots per Bobbin	10 max.
No. of Conductors per Slot	3
Mean Circumference of Coil, m	27.4
Helium Flow Rate per Conductor, g/s	3.2
Helium Inlet Temperature, K	4.2
Helium Inlet Pressure, Atm	4

* Averaged over slot area (clearance, insulation, jacket, conductor, stabilizer and coolant)

** Copper to noncopper area ratio is 1.37

boron at a ratio of 1 boron atom per 68 water molecules. The calculations indicate that an inner shield of 95 cm thickness and an outer shield of 76 cm on the outside of the fuel zone will reduce the radiation fluxes to the level where 25 years of operation at 0.2 plant factor will be possible before the dose of 1×10^8 rads is reached.

Figure 3-7 shows the arrangement of the shielding and its dimensions. The volumes of the shield segments numbered A through D in this figure are given in Table 3-6 together with the masses of water and stainless steel constituting each.

The resulting shield heating, plotted as a function of distance from the center of the tokamak, is shown in Figure 3-8. The major radius of the tokamak is 5.2 m. From this figure, it will be observed that the ThO_2 fuel zone, assumed to contain 2% ^{233}U , provides about one decade of attenuation of radiation fluxes.

Since the fuel covers only 3.7 m of height along the outside equatorial surface, the outside shield thickness above and below the fuel must be increased by an additional 19 cm of shielding. This extra thickness is shown in Figure 3-7 as two conical shields, E, and its volume is included in Table 3-6. The volumes and masses shown in that table are for uninterrupted shield coverage. In actual practice, they will be reduced by the presence of shield penetrations for the divertor, neutral beam injectors, and vacuum pumping ducts.

The maximum and the volume-average volumetric heating rates in the inner and outer shields are given in Table 3-7. The maximum heating rates on the inner surface of the conical supplementary shield located above and below the fuel as part of the outside shield are nearly the same as on the inside surface of the inner shield.

Shield Composition: 35% B-H₂O, 65% Stainless Steel.
All Dimensions in Meters.

Figure 3-7. Dimensions for TF Coil Shielding in DTHR

TABLE 3-6
VOLUMES AND MASSES OF MAIN SHIELD SEGMENTS

ZONE	VOLUME (m ³)	MASS (TONNES)	
		Water*	SS**
A Outer Shield	370.	130.	1924.
B Upper and Lower Extensions	37.	13.	192.
C Upper and Lower Inner Shield	81.	28.	421.
D Vertical Inner Shield	134.	47.	697.
E Conical Supplement	33.	12.	172.
TOTALS:	655.	230.	3406.

* 35^v/o, 1. g/cm³ density

** 65^v/o, 8.0 g/cm³ density

TABLE 3-7
VOLUMETRIC RADIATION HEATING RATES IN SHIELDS DURING A PULSE

	POWER DENSITY, W/cm ³					
	Water		SS		Shield Total	
	Avg	Max	Avg	Max	Avg	Max
Inner Shield	0.389	4.39	0.900	10.13	1.29	14.52
Outer* Shield	0.062	0.37	0.089	0.85	0.151	1.22

*At inner surface of conical supplementary shield E, the maximum heating rates equal maximum values of inner shield and fall off exponentially with an attenuation constant $\mu = 0.1307 \text{ cm}^{-1}$.

Figure 3-8. Power Density in DTHR TF Coil Shielding

The total power deposited in shield segments A through D is summarized in Table 3-8. The total thermal power in the shields is 412 MW. Thus, based on the total fusion power of 900 MW of which 0.8 is neutron power, an additional 308 MW is deposited in the ThO_2 fuel, over and above the fission power generated in the fuel.

The radiation heating in the dewar of the TF coils is 0.50 kW, in the TF coil itself 1.09 kW, for a total heating load of 1.59 kW per coil. This is well below the 20 kW design limit. Consequently, the total dose criterion is the controlling factor of this design. Additional shielding analyses would be necessary to determine the adequacy of the shield from the point of view of shutdown biological shielding for maintenance purposes and the interaction of the reactor shield and the reactor cell wall thickness design requirements.

For the shield cooling a low pressure water system was used. The shield concept consisted of stainless steel balls which are cooled by borated water in one flow pass with a void fraction of 0.35. Assuming a steady state condition (most conservative assumption for long duty cycle operations), the required water flow rate and pumping power requirements were estimated as shown in Table 3-9. The total water flow rate is about 5×10^6 kg/hr and the total pumping power is about 0.43 MW. A primary loop and a buffer loop may be needed to assure tritium removal has been achieved before the heat is dissipated to the cooling system.

3.3.3 POLOIDAL FIELD (PF) COILS

The poloidal field coils have been located outside the toroidal field coils to provide accessibility, to avoid interference with the blanket, and to prevent radiation heating and damage. Generally, if the shaping field (SF) coils have to be located at distances from the plasma axis which are several times as large as the plasma minor radius, it is very difficult to find SF coil locations which will produce an MHD equilibrium configuration which has an elongation of 1.6 and yet satisfies all the other mechanical constraints, such as sufficient access for neutral beam, blanket, and divertor. The currents and, thus, the energy stored in the SF coil system are large and the equilibrium state exists only in a very narrow range of vertical field strengths. These difficulties were reduced by using the compact D shaped TF coils.

TABLE 3-8

THERMAL POWER DUE TO RADIATION HEATING IN SHIELDS DURING A PULSE

	(MWt)
Inner Shield (95 cm thickness)	276
Outer Shield (76 cm thickness)	56
Supplementary Shield (19 cm thickness)	80
TOTAL:	412

TABLE 3-9

SUMMARY OF SHIELD COOLING SYSTEM PARAMETERS

	<u>Inner Shield</u>	<u>Outer Shield</u>	<u>Supplementary Shield</u>
Heat deposition, MW	276.4	55.8	80.4
Water flow rate, kg/hr	3.35×10^6	0.68×10^6	0.98×10^6
Inlet pressure, MPa	0.7	0.7	0.7
Inlet temperature, °C	74	74	74
Outlet temperature, °C	145	145	145
Pressure drop, MPa	0.138	0.009	0.037
Pumping power (@ 50% Eff.), MW	0.38	0.01	0.04

OHMIC HEATING (OH) COILS (SOLENOID)

The ohmic heating coils within the device bore form a compact solenoid. A cross section of a vertical stack portion of the solenoid is shown in Figure 3-9. The solenoid consists of a total of 23 vertical stacks. Each vertical stack is made up of 5 radial rows of bobbins, each bobbin containing two rows of conductors, 14 conductors in height. The 28 turns within the bobbin result in a coolant path with an average length of 184 m for the forced flow supercritical helium. This is the coolant path length which can be adequately cooled to prevent the superconductors from going normal during operation. This length may have to be shortened pending a detailed cryostability analysis which uses the pulsing profile of the OH coils. There must be a liquid helium inlet and exit for each radial bobbin row and this is accomplished by using the connector shown in Figure 3-10. The connectors will be circumferentially spaced within the vertical space provided by the rings between stacks.

Each stack can be independently assembled, vacuum impregnated with epoxy and then the entire solenoid can be assembled by bolting the stacks to the spacer rings. All final electrical and coolant connections can be made in the solenoid bore. It is proposed that all superconductor leads would exit the solenoid-toroidal field coil common dewar at the bottom end where the transition to a copper bus would occur. The parameters for the OH coils are summarized in Table 3-10. The conductors of the OH coils have been designed to be superconducting to minimize energy requirements. Nb_3Sn is the superconductor material in a cable similar to the TF coil cable except for a thinner jacket. A cross-section of a typical triplex cabled superconductor is shown in Figure 3-11 to illustrate the fabrication technique used to achieve coolant channel voids. Figure 3-12 shows dimensions of the cable used in the PF coil concepts.

SHAPING FIELD (SF) COILS

The shaping field coils and the balance of the OH coils not incorporated within the solenoid are located outside of the toroidal field coils. The upper and lower SF and OH coils within their dewar systems mount directly to the support structure, while the outer SF and OH coils attach through their respective dewar systems to the TF coil dewars as shown in Figure 3-4.

Figure 3-9. Typical OH Coil Cross Section, Inner Bore Coils

Figure 3-10. Conductor Splice Joint

TABLE 3-10

DTHR SUPERCONDUCTING OH COIL PARAMETERS

INBOARD COILS

Design Maximum Field, Tesla	9
Inner Radius, m	1.00
Outer Radius, m	1.25
Total Stored Energy, MJ	645 (Total)
Total Ampere-Turns, MA-T	51.5
Superconductor/Stabilizer	Nb ₃ Sn/Cu
Conductor/Shape	Cable/Square
Current per Turn, kA	16
Average Current Density: Conductor, Stabilizer, and Coolant, A/Cm ² (a)	4400
Structural Material	Stainless Steel
Coils in Vertical Stack	23
Turns per Stack	140
No. of Cooling Channels per Stack	5
Average Length of Cooling Channel, m	184

OUTBOARD COILS

Design Maximum Field, Tesla	3
Total Ampere-Turns, MA-T	10.5
Superconductor/Stabilizer	Nb ₃ Sn/Cu
Current per Turn, kA	16
Average Current Density: Conductor, Stabilizer, and Coolant, A/Cm ² (a)	4400
Structural Material	Stainless Steel
Total Conductor Length, km	18.7

(a) Average current density including jacket is ~ 3700 A/Cm².

CONDUCTOR PRIOR TO COMPACTION
(86% void)

CONDUCTOR FOLLOWING COMPACTION
(30% void)

SUPERCONDUCTOR
 $\text{Nb}_3\text{Sn/Cu}$ (54%)

SS SHEATH (16%)

H_e COOLANT (30%)
(in voids)

Figure 3-11. Cross Sections of Typical Triplex Cabled Superconductor Illustrating the Distribution of Voids.

Figure 3-12. Typical Cable Design Cross Section

The arrangement of the turns in the coil will be similar to that shown in Figure 3-9 for the OH inner bore coils utilizing vertical support bobbins in radial rows. The vertical height of the coil will be adjusted so that the coolant path length will meet the cryostability requirements which prevent the superconductor from going normal during operation. This will mean that, as a minimum, a coolant connector will be required at both ends of each vertical row of conductors. For the large diameter coils, it may be necessary to arrange the coil in stacks with spacer-rings between stacks to accommodate the coolant connectors. The parameters for the SF coils are summarized in Table 3-11. The preliminary specifications for the superconducting cable design parameters for both the OH and SF coils are shown in Table 3-12.

The PF coil locations and currents are listed in Table 3-13. The plasma configuration is shown by Figure 3-13. No inner SF coils, i.e., coils located at the tokamak bore side of the plasma, were used since their omission permits a simpler mechanical design. However, a more stable design can be made if the SF coils are placed in the inner bore area.

3.3.4 VACUUM VESSEL

In view of the experimental mission of the DTHR it was concluded that the vacuum vessel/first wall should be structurally and thermally independent of the blanket. This would permit maximum latitude in the testing of blanket modules with alternate coolants, materials and fuels and would permit blanket inspection and replacement operations with a minimum impact on the plasma vacuum chamber.

The vacuum vessel concept consists of a two component structure. The structural member is a rib stiffened shell that withstands the external static load of one atmosphere. The first wall is a tubular wall which shields the outer structural shell from the high surface heat flux. The thin wall shell under external pressure loading was sized for buckling strength rather than for stress considerations. The vacuum vessel is constructed from 316 stainless steel. The fabricated vessel is made up of 10 identical modules to form the torus shown in Figure 3-14. The structural wall is ~ 0.4 cm thick and is stiffened on the outside periphery at increments of one meter. The stiffeners are designed for minimum mass. Because of the smaller inner equator of the toroidal envelope,

TABLE 3-11
DTHR SUPERCONDUCTING SF COIL PARAMETERS

Design Maximum Field, Tesla	2
Total Stored Energy, MJ	1655
Total Ampere-Turns, MA-T	45.4
Superconductor/Stabilizer	Nb ₃ Sn/Cu
Current per Turn, kA	32
Average Current Density: Conductor, Stabilizer, and Coolant, A/cm ²	8800
Structure Material	Stainless Steel
Total Conductor Length, km	39.2

TABLE 3-12
SPECIFICATIONS OF THE SUPERCONDUCTOR CABLE
FOR THE OH AND SF COILS

	<u>OH</u>	<u>SF</u>
Average current density*, A/cm ²	4400	8800
S/C Operating Current Density, kA/cm ²	185	800
Area of conductor + He, cm ²	3.61	3.61
Superconductor area, cm ²	0.087	0.087
Cu - to - non Cu area ratio	5.08	12.2
No. of strands (3 x 3 x 3 x 3 x 7)	567	567
Strand diameter, cm	0.075	0.075
Conductor hydraulic diameter, cm	0.066	0.066
Conductor void fraction	0.30	0.30
Helium flow rate per conductor, g/s	2.0	2.0
Helium inlet temperature, K	4.2	4.2
Helium inlet pressure, atm	4	4

* Averaged over conductor, stabilizer and coolant.
If averaged including the insulated jacket, it is 3300
and 6500 A/cm² respectively.

TABLE 3-13

DTHR PF COIL SYSTEMS: COIL LOCATIONS AND CURRENTS

Locations ^(a)		No. of Turns	OH Current, MA-T	SF Current, MA-T
Radius R, m	Height Z, m			
Center Bore OH Coils (1.125)	0.35 ea 0.09 Spaces 23 Coils	140 per Stack		- - -
2.2	5.5	369		5.76
2.9	5.7	369		5.76
3.9	6.2	369		5.76
8.9	4.2	116		-1.80
9.6	3.4	116		-1.80
9.7	2.4	116		-1.80
TOTAL OH CURRENTS (AT 50% BIAS)			$I_{OH} = 62.0$	$\Sigma I_i = 45.4$

TOTAL STORED ENERGY

2.3 GJ

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

Figure 3-13. Flux Plot of the DTHR Plasma MHD Equilibrium Configuration

Figure 3-14. DTHR Vacuum Vessel Concept

the stiffening rib will have a varying area moment of inertia of its centroidal axis parallel to the shell, as it traverses the poloidal periphery of the toroidal shell. The stiffener's area moment of inertia varies from 134 cm^4 at the inner equator to a maximum of 250 cm^4 at the outer equator. The stiffeners are designed to support their own weight, the shell's panel weight, and the weight from the tubular first wall. Each rib is supported by a three point suspension structure. The stiffener's supports are also designed to serve as coolant manifolds for the tubular first wall. The stiffener-manifold is illustrated in Figure 3-14.

Pressurized water was determined to be an effective medium for removing the large thermal loads on the vacuum vessel. This cooling is accomplished by circulating pressurized water in the bank of tubes which line the plasma side of the vacuum vessel. The tubes are aligned to run in the toroidal direction with coolant manifolds spaced every four meters on the outer equator to meet hydraulic limitations (pressure drop and temperature rise). However, the manifolds may have to be located every two meters from a fabrication/assembly consideration. The cooling tubes are arranged in the toroidal direction with the manifolds arranged in the poloidal direction. Commercial grade 316 stainless steel tubing with 0.953 cm ($3/8"$) O.D. is used as the cooling tubes. The working temperature limit of the 316 stainless steel is approximately 450° C . The thermal heat flux used in the design calculation was taken to be 2 MW/m^2 and the nuclear heating rate was $\sim 56 \text{ W/cm}^3$. The heat flux (normally taken as one-fourth of the wall loading for a concept using a divertor) is conservative, since the neutron wall loading of the DTHR is 2 MW/m^2 . The results are summarized in Table 3-14.

Reasonable design tolerance is to allow for a hot channel safety margin ($\sim 20^\circ \text{ C}$) between the saturation temperature and the coolant outlet temperature. Nonuniformities in incident flux and heat generation were accounted for in analyzing the maximum tube and shell temperatures. Additional analyses need to be performed to determine the most efficient cross-section for the stiffener rib-coolant manifold. The structural assessment of this design is summarized in Table 3-14.

TABLE 3-14
VACUUM VESSEL (FIRST WALL) PARAMETERS

Structure	Tubular Design With Rib-Stiffened Shell
Material	316 Stainless Steel
Shell Panel Thickness, cm	0.4
Stiffening Ring Spacing, cm	106
Area Moment of Inertia of Stiffener, cm ⁴	250
Coolant Tube OD, cm	0.953
Coolant Tube ID, cm	0.810
Design Surface Heat Flux, MW/m ²	2
Design Neutron Heating Rate, W/cm ³	~ 56
Nominal Coolant Heat Transfer Coefficient, $\frac{W}{cm^2 \cdot ^\circ C}$	2.36
Maximum Tube Temperature, °C	427
Maximum Shell Temperature, °C	394
Maximum Thermal Stress, MPa	
Tube OD	172.3
Tube ID	206.9
Shell Outside Wall	17.2
Shell Inside Wall	41.4
Coolant	Pressurized Water
Inlet Pressure, MPa	6.89 (1000 psi)
Channel Length, m	4
Outlet Pressure, MPa	6.85
Channel ΔP , MPa	0.04
Inlet Temperature, C	74
Outlet Temperature, C	283
Saturation Temperature at Channel Exit, C	284
Total Flow Rate ⁽¹⁾ , kg/hr.	39.8×10^5
Pumping Power ⁽²⁾ , kW	126

Notes:

- (1) Based on 360 m² vessel surface area and a total of 9000 cooling channels.
- (2) Based on 50% pump efficiency.

The vessel first wall cooling system will be independent from the blanket and shield cooling systems because of the pressure difference. The vessel wall cooling system would consist of a closed primary loop to isolate the activated water. This loop would contain the necessary pumps, clean-up and filtering equipment including the system for removal of tritium. A buffer loop would connect the primary loop to the forced draft heat cooling towers.

3.3.5 PLASMA EXHAUST SYSTEM

The initial (March 1978) DTHR divertor concept (Figure 3-15) was based on the existing design analyses for ISX-B in order to quickly establish the feasibility of the bundle divertor for a high field DTHR. To minimize computing time, only critical parameters were analyzed as a check. The rough iterative calculations were done to determine the optimum value of the critical parameters such as coil current, coil position and angle, the stagnation point, and scrape-off layer. This allowed an evaluation of the overall feasibility of the cryogenic, particle collection, and vacuum pumping problems before investing time on the detailed component design. Many iterative processes were carried out and innovative methods were developed for particle collection and vacuum systems to show that the bundle divertor was feasible for the DTHR application. The revised DTHR bundle divertor has been developed with significant improvements over the March concept. Reference 3-4 provides a detailed report of the improved design. The revised parameters are shown in Table 3-15.

The critics of the bundle divertor contend that it is not applicable to a reactor because the divertor current and forces and power consumption are inherently large. However, it has been shown that the current density and forces can be greatly reduced with an improved method for an experimental tokamak device - ISX-B⁽³⁻⁵⁾. The report indicates that such an improved divertor is applicable to a reactor with a toroidal field on the plasma axis as high as 5.5 tesla and an aspect ratio of 4.3. This field is considered to be in the intermediate range.

The most significant achievement is the reduction in current density; the current densities per tesla of field to be nulled are 25 kA/cm² for DITE, 11 kA/cm² for ISX-B and 1 kA/cm² for the present DTHR bundle divertor concept.

Figure 3-15. Bundle Divertor Concept

TABLE 3-15
DTHR BUNDLE DIVERTOR COIL PARAMETERS
(SEPTEMBER 1978)

Maximum Field in Divertor Coils	10 T
Two Circular Divertor Coils Current/Coil	14.0 MA
Two Vertical Auxiliary Coils Current/Coil	6.4 MA
Divertor Coil Angle (α)	51°
Mean Coil Diameter	2.65 m
Overall Width of the Coil	1.3 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	1 kA/cm ²
Coil Structure	Stainless Steel Bobbins Constant Thickness
Minimum Shielding Space Between Conductor and Plasma	.60 m
Helium Inlet Temperature *	4.2 K
Pumping Power (@ 50% Efficiency)	8 MW _e
Coil Inductance	126 mh
Electrical Power Supply (30 min. charging time)	612 kW

* Forced flow supercritical helium cooling at 4 atm inlet pressure.
Conductor void fraction is 0.35.

Besides designing an efficient divertor magnetic system, attention was given to design details which enabled evaluation of an integrated divertor system. This includes the cryogenic design of the divertor coils, mechanical and maintenance design of the divertor coil assembly, the mechanical and thermal design of the divertor collector system, the method of cycling the collector to permit continuous operation of the reactor, and materials and stress analyses. The divertor assembly has been designed for easy removal. Otherwise, a bundle divertor will lose its advantage if it cannot be maintained. An initial concept for the collector system designed for service under the most severe conditions has been completed. The system appears feasible, but is by no means state-of-the-art.

The superconductors for the bundle divertor were selected to be an Nb_3Sn copper composite operating at a constant field of 10 tesla. The conductors are to be cooled by forced flow supercritical helium. The total ampere-turns for one side of the divertor coils was taken to be 14 MA-T. Based on the space available for the divertor coils, the conductor transport current was selected to be 32 kA, the conductor plus helium area 4 cm^2 , and the void fraction 0.35. The number of conductor turns per coil is 438 turns. Assuming a pump efficiency of 75%, the refrigeration load for the pump work was estimated to be about 9 kW at 4.2 K per coil. The nuclear heating was estimated to be 2 kW averaged over a pulsed cycle based on 60 cm shielding surrounding the coils. The dewar thermal loads (although small) would have to be added to obtain the final refrigeration requirement for the divertor coils.

The bundle divertor consists of two circular coils surrounded by a duct that forms an appendage to the side of the reactor vessel. The duct terminates in a combination flange that can be bolted and seal welded to a mating closure. The mating closure which houses the bundle divertor collector system is a rectangular vessel flanged at one end and closed at the other. The rectangular vessel also contains collector discharge compartments located on top and bottom surfaces.

As is shown by Figure 3-16, the two circular divertor coils consist of a total of 37 bobbins; 35 bobbins each containing 12 conductors, one bobbin containing 6 conductors and one special two-piece bobbin that provides for double stacking of the 12 conductors over one-half the circumference of the bobbin. The double stacking arrangement provides for a narrower bobbin in the double stack region to provide space for more shielding. All bobbins are 3.5 cm thick and have a mean circumference of ~ 8.33 m. There are 438 conductor turns per coil and each conductor has a cross-section area of 4 cm^2 excluding the jacket. Each coil enclosed in a separate dewar weighs approximately 18,600 kg.

It was previously concluded that the torques and forces on divertor coils were too large to be constrained for a reactor application, because the current was high and concentrated in a limited area leaving no space for structure. The strength of the conductor had to resist the torque and stress. However, as has been pointed out in some recent ISX-B⁽³⁻⁵⁾ work, the current can be reduced and the space available for conductors can be doubled. This means enough structure can be provided to counter the stress. The DTHR magnetic stresses have been computed. The stresses are largest at the coil edge where the two coils meet at the center of the divertor. The stresses decrease toward the coil bore and change direction from the upper end to the lower end. There are points where the stresses are zero. The maximum stress is $f_p = 300 \text{ N/cm}^2$. Internal structure can easily be designed for the maximum stresses. The internal torques are small. The net outward translational force is 44 MN which can also be restrained.

The two auxiliary divertor coils, each made in two parts, are located vertically above and below the divertor coils as shown in Figure 3-17. The two-part design provides space for the poloidal coils. Each part of the auxiliary coil is made up of ten parallelogram shaped bobbins each 3.0 cm thick and containing five conductor slots. Each conductor slot contains four conductors or 200 conductor turns per coil using the same size and shape conductor used in the divertor coils. The auxiliary coils are superconducting and are designed with individual dewars. The outer part of the auxiliary coil, that is, the part furthest from the divertor coils is mounted to a structure supported off

Figure 3-16. DTHR Bundle Divertor Plan View

Figure 3-17. Bundle Divertor Side Elevation

the adjacent TF coils. The inner part, or part closest to the divertor coils, is provided with a horizontal flange located some distance away from the innermost edge of the coil. The divertor duct directly above and below the divertor coils has rectangular openings to receive that portion of the auxiliary coil extending inward beyond the flange. When the auxiliary coils are installed and the flanges seated on the divertor duct, the portion of the auxiliary coil extending inside the divertor vessel is seated and keyed to the curved center support structure of the two divertor coils. The flange is clamped and seal-welded to the divertor duct.

Included as part of the bundle divertor are two mirror field coils which are mounted on each side of the divertor collector system vessel (see Figure 3-16). The mirror field coils are used to further shape and direct the flux path into the collector panels. Each coil is made up of three circular bobbins with a mean diameter of 2.0 m and contains a total of six conductors of the same size and shape as those used in the divertor coils.

The particle through-put to the divertor is 1080 torr liters/sec which has to be removed. The back-streaming to the plasma has to be kept as low as possible. The thermal power of 280 MW has to be transferred. The projected area of the flux bundle available for particle collection is 6 m^2 . That is, the maximum height of the flux bundle is 2 m and the radial width is 3 m.

For sustaining a long burn time at reasonable high vacuum the collector must be able to trap the particles with a high sticking probability and holding capacity. The materials suitable for this purpose have been surveyed and given in another report⁽³⁻⁶⁾. For DTHR it is assumed that the commercially available alloy Zr/Al, which is deposited on constantan (copper and Zr alloy) and has a trapping efficiency of 100% and holding capacity of $2 \times 10^{19} \text{ cm}^{-2}$ at a temperature of approximately 200°C , is employed. The maximum pulse length of DTHR consists of 70 s operation and 15 s down. The collector should be able to hold the particles for many operating cycles so that the maximum duty cycle can be maintained. It is not possible to regenerate the collector during a short downtime. Therefore, the projected surface was increased by employing 14 V-shaped surfaces which consist of water cooled tubes as shown

by Figure 3-18. The divertor collector assembly consists of a series of panels 1.0 m wide by 2.76 m long. The panels are arranged in a saw-tooth pattern such that the overall arrangement of the collector assembly is accordion shaped. The concept for the construction of the collector panels uses a 2.0 m length of ~ 1 cm diameter copper tube bent at the mid-point to form a Vee with a 1.27 cm bending radius at the apex of the Vee. Two-hundred ninety of the formed tubes are then stacked side by side and stitch-welded to form two panels, one for each side of the Vee. The panels thus formed are of tubular construction for coolant and have a rippled surface to increase the total surface area. This concept provides a surface area capable of a holding time of 17 minutes. The collector would be designed to be regenerated at 10 minute intervals or at $\sim 60\%$ of the full capacity. Therefore, 7 cycles are used based on an 85 s plasma duty cycle.

The helium concentration is about 1% of the plasma stream or 10.8 torr/sec throughput. A trapping efficiency for helium was pessimistically assumed to be zero. To maintain a helium partial pressure of 10^{-5} torr, a pumping speed of 1.1×10^6 ℓ/s is required. The specific speed of a cryosorption surface is ~ 3.7 $\ell/sec\ cm^2$. Therefore, about 30 m^2 of projected cryopanel area are needed which can be easily designed around the enclosure of the burial chamber. Both the collection and cryosorption panels are dual units. One collection unit is in the lower chamber and is sealed at the top. This unit is warmed up to 600° C to release the trapped particles which are pumped and collected. In the meantime, the other unit is collecting particles, but before reaching capacity, the whole collection system is lifted up slowly until the upper unit enters the chamber on the top which is then sealed for regeneration. This cycle is later reversed when the lower unit reaches its capacity. The system is designed to move at < 1 m/sec. Since the cryopanel are not directly exposed to the energetic particles, they can be isolated from the chamber by gate valves. The modules are further divided into small units so that reasonable component size and moving mechanisms can be designed. The valves are designed to slide back and forth between two neighboring modules. One module can be closed for regeneration while the neighboring one is open for pumping. This dual particle collection unit and cryosorption module system allows for continuous collection and regeneration simultaneously without affecting the plasma duty cycle factor.

Figure 3-18. Collector System for Bundle Divertor

The minimum shielding space to protect against streaming neutrons is between the first wall and the foremost corners (closest to the plasma) of the inner bobbin of the divertor coil (see Figure 3-16). This space is 60 cm. The shielding design has been assessed based on the work done for TNS. The TNS is about the same size as DTHR. The space in TNS between the first wall and superconducting TF coils was filled with 30 cm of shielding material (65% in volume of stainless steel and 35% borated water), 25 cm water-cooled copper SF coils (50% in volume), 5 cm void, 5 cm lead, 15 cm thermal insulation and 60 cm void. The total shielding space excluding void was 60 cm (30 cm S.S. and borated water, 25 cm copper and water, and 5 cm lead), which would fit into the shielding space for the divertor of DTHR. This is obviously not the best choice of shielding design for a divertor but can be used to assess the problem.

The highest nuclear heating rate was $3 \times 10^{-3} \text{ W/cm}^3$ at the corner of the coil closest to the plasma where the insulator will be subjected to the most serious damage. The machine operates $\sim 20\%$ of the time (50% duty factor, 40% availability). The maximum dose on the epoxy insulator is 3×10^9 rads which is about the maximum allowable dose⁽³⁻⁶⁾. This means that the front bobbin has to be repaired once a year. This estimation is conservative. Better shielding materials which can give four decades of attenuation are available. One such shielding design, for example, might be one which consists of W and B_4C . The characteristic length for an e-fold attenuation is $T = 6.5 \text{ cm}$ which is equivalent to 1 decade of attenuation per 15 cm. A 60 cm shielding space would give four decades of attenuation; the lifetime of the insulator would be three times longer. It can be concluded that a 60 cm shielding space for a bundle divertor may be adequate, even for commercial reactors. It can be seen from Figure 3-19, that most areas within the bundle divertor can have significantly more than 60 cm of shielding space.

3.3.6 CRYOGENIC REFRIGERATION SYSTEMS

The cryogenic refrigeration systems for the DTHR consist of liquid helium (LH_e) and liquid nitrogen (LN_2) refrigerators and associated equipment required to cool the superconducting magnetic field coils, the dewars and the cryopanel of the vacuum pumping systems. The major thermal loads for the

Figure 3-19. Bundle Divertor Coil Cut-Away Views

systems were estimated based on the reference reactor design configuration. The LH_e and LN_2 refrigeration loads are summarized as shown in Table 3-16 and Table 3-17. The LN_2 is required mainly to remove the thermal load in the coil dewars. The LN_2 liquifaction required to operate the LH_e refrigeration system, if any, was not included in Table 3-17.

The major portion of the LH_e thermal load was from the superconductor cooling required to provide for cryostable operation. The cryostability of the Nb_3Sn superconductors used in the TF coils was analyzed to establish the thermal load for the DTHR TF coils. The PF coils, unlike the steady state TF coils, are subjected to pulsed current reversal during each "on" cycle of operation. The pulsing induces considerable ac losses in the superconductors. The high refrigeration load for the inboard OH coils was estimated based on a $\dot{B} \sim 100$ T/s for 5 ms field change from a peak field of 9 tesla in one pulse. The refrigeration loads required for other cooling systems were small compared to that of the PF or TF coils. The bundle divertor had the next highest load of ~ 11 kW.

The estimated refrigeration loads are considered conservative, but represent significant amounts of refrigeration. The requirements are larger than any commercial system available today. Even though multiple units can be utilized, extensive research is required to develop units with acceptable efficiency and reliability.

3.3.7 NEUTRAL BEAM INJECTION SYSTEM (NBIS)

A summary of the NBIS design requirements based on a positive-ion system of sources is given in Table 3-18.

Each individual positive-ion source is assumed to be a scaled-up version of those to be utilized on TFTR and being developed at LLL/LBL. Each of eight ion sources can produce 110 A of 175 keV particles (D^+ , D_2^+ , D_3^+) and is sized with dimensions similar to those for TFTR. The width (0.1 m emitters, 0.45 m sources, on 0.67 m centers) is the same as that for TFTR, but the length (0.4 emitter, 0.8 m source in TFTR) was increased to take into account the

TABLE 3-16
SUMMARY OF LH_e REFRIGERATION LOADS

	<u>Heat Load, kW @ 4.2K</u>
Toroidal Field Coils	44
Poloidal Field Coils	
OH (in-board)	185
OH (out-board)	78
SF	56
Bundle Divertor	11
Cryopanel (@ ~ 1 W/m ²)	1
Coil Leads (@ ~ 1 W/1kA)	2
	<hr/>
Total LH _e Refrigeration Load	377
Total Electric Power Load at 300K	133 MWe

TABLE 3-17
SUMMARY OF LN₂ REFRIGERATION LOADS

	<u>Heat Load kW @ 77K</u>
Field Coil Cryostat	
TF Coils	155
PF Coils	47
Bundle Divertor	3
Thermal Shields for Cryopanel (@ ~ 100 W/m ²)	73
	<hr/>
Total LN ₂ Refrigeration Load	278
Total Electric Power Load at 300K	4 MWe

TABLE 3-18
NBIS DESIGN REQUIREMENTS AND ASSUMPTIONS

TOTAL NB POWER TO PLASMA, P_b^*	150 MW _t
INJECTION ENERGY, w_b	175 keV
NB PULSE LENGTH, τ_b	$\lesssim 68$ s
NUMBER OF NB INJECTION ARMS, N_{BI}	6
NB POWER TO PLASMA PER ARM *	25 MW
INJECTION ANGLE AT MIDPLANE, θ_b^{**}	$\gtrsim 16^\circ$
CHARGE STATE OF SOURCE IONS	+ 1

* AT AN ENERGY $w_b = 175$ keV, FRACTIONAL ENERGY COMPONENTS NOT INCLUDED.

** AN ANGLE $\geq 16^\circ$ REDUCES PROBABILITY OF THE RIPPLE TRAPPING OF THE INJECTED BEAM PARTICLES.

increase in source current (from 60 A to 110 A). The DTHR emitters are 0.8 m long and the sources are 1.2 m long with 0.2 m spacings. Each source is characterized as producing 100 A of D^+ current and 10 A of D_2 , i.e., a beam with an 0.9 fraction of atomic particles. A gas-to-ion conversion efficiency of 40% is also assumed, resulting in a required gas feed of approximately 30 torr liter/s. The total design power in the 8 source beams is 150 MW.

A close-coupled D_2 gas neutralizer duct (0.2 m x 0.9 m x 2.0 m) operating at an average pressure of 3.5×10^{-3} torr without any source gas except that streaming through from the ion source is used. The asymptotic value of the neutralization efficiency of a 175-keV D^+ beam is 27% and, after taking the finite target thickness of the neutralizer duct into account, that efficiency is reduced to a value of 22%. For the D_2^+ molecular component the efficiency is 53%. Thus, taking into account an estimated loss due to beam divergence of 3%, the neutral beam power emerging from the eight neutralizers is 25 MW of D^0 and 7 MW of D_2^0 . The power associated with the charged particle portion is 119 MW total for the eight beams.

The ion collector assembly is approximately 2 m long with a 1 m ion deflector and electron deflector assembly held at ~ -35 kV. The load on the deflectors due to secondary electrons and divergence losses in the beam are estimated at 2.2 MW per ion beam (8 beams per arm). The overall efficiency of this system is estimated to be 14%, which includes neutralization efficiency and transport losses.

The vacuum pumping system will include in-line cryopanel or zirconium-aluminum panels for maintenance of acceptable operating pressures in the source region as well as in the collector-drift region. There is approximately 120 m^2 of surface area available in each arm (a chamber with dimensions of 3.5 m x 4.1 m x 8.0 m).

The surface heat loads which size the water cooling system are the neutralizer wall, electron suppressor grids, recovery plates in ion collector systems, and defining plates at the wall defining the two main vacuum pumping regions. The respective heat loads to the surfaces are estimated at 0.3 MW/m^2 , 14 MW/m^2 , 1.1 MW/m^2 , and 1.8 MW/m^2 , where the suppressor and recovery plates are held at -35 kV and 150 kV, respectively. The electron suppressor grids are in the form of tubes with a 2 cm O.D. through which water passes at a velocity of $\sim 6 \text{ m/s}$ and can be operated in steady-state. The defining plates are assumed to be of a double-wall design with a water cooling velocity of 1 m/s .

Each individual ion source is assumed to be powered by an independent supply. The supply design is assumed to be similar to that required for TFTR for an ion source with up to four accel/decel grids. The maximum voltage is 175 kV at a current load of 110 A. In addition to the accel/decel grids, supplies for the filaments and arc are required. For the purpose of the conceptual design, an overall efficiency (beam power to electrical power) of 80% was assumed. A summary of the main design parameters for the NBIS is given in Table 3-19 and a tabulation of the electrical and thermal power loads is given in Table 3-20.

For purposes of sizing the ducting to the torus, the individual ion sources are assumed to be aimed at a cross-over point midway between the plasma edge and the plasma center. A clear opening of 1.37 m by 0.83 m is required. The overall size of the duct cross section using 0.3 m wide radiation shielding is 2.05 m by 1.51 m.

3.3.8 VACUUM PUMPING SYSTEMS

The total vacuum pumping system for the DTHR consists of the neutral beams pumping system, the divertor pumping system, the torus vacuum pumping system, and individual diagnostic pumping systems. The neutral beam vacuum system is designed to pump the gases added to the system via the ion sources and neutralizers. The design goals are to provide an environment for the beams such that beam losses are small due to atomic interactions and to maintain the flow of gas into the torus from the beam lines at an acceptably low level.

TABLE 3-19
NBIS DESIGN PARAMETERS

CURRENT PER ION SOURCE		110 A
ATOMIC FRACTION OF ION CURRENT		0.9
D ₂ GAS FEED RATE PER SOURCE		30 torr ℓ /s
SOURCES PER NB ARM		8
GAS NEUTRALIZER DUCT LENGTH		2 m
AVG D ₂ GAS PRESSURE IN NEUTRALIZER DUCT		3.5×10^{-3} torr
RESULTANT NEUTRALIZER EFFICIENCIES	D ⁺ $\rightarrow$ D ^o	22.1%
	D ₂ ⁺ $\rightarrow$ D ₂ ^o	63.0%
ION COLLECT VOLTAGE ⁽¹⁾		150 kV
ELECTRON SUPPRESS VOLTAGE		-35 kV
OVERALL TRANSMISSION THROUGH ARM		~ 0.88
HIGH VACUUM PUMPING SPEED PER ARM		1.2×10^7 ℓ /s
TOTAL GAS THROUGHPUT TO VACUUM PUMPS IN EACH ARM		400 torr ℓ /s
PUMP PANEL REGENERATION PERIOD ⁽²⁾		~ 1 hour

(1) ASSUMING ION SOURCE AT GROUND

(2) FOR EITHER Zr/Al GETTER PANELS OR CRYOPANELS

TABLE 3-20
SUMMARY OF ELECTRICAL AND THERMAL POWERS FOR NBIS

	<u>Per Source (MW)</u>	<u>Per Beam Arm (MW)</u>	<u>Total System (MW)</u>
Required Electrical Input Power *	23.63	189	1135
Losses in Power Supplies and Ion Generator	4.75	38	228
Losses in Neutralizer Ducts	0.57	4.6	28
Losses in Electron Suppressors	1.6	12.8	77
Losses in Ion Collector Assembly	1.6	13.1	79
Losses in Ion Collector Power Supplies and Resistors	13.8	8.3	498
Losses in Ducting to Torus	0.7	5.6	34
NB Power in D ⁰ Beam	3.1	25	150
NB Power in Molecular Beam	0.85	6.8	41

* An overall power supply electrical efficiency of 80% assumed which includes losses within accelerator assembly.

The Torus Vacuum Pumping System (TVPS) for DTHR will maintain the vacuum vessel at a low base pressure ($\sim 10^{-7}$ to 10^{-8} torr) between plasma pulses and will be used in the vacuum conditioning of the torus during any period in which the divertor pumping system is not used. The divertor pumping system and the TVPS are complementary systems in that the divertor is effective during the pulse and the TVPS is effective between pulses when there is no divertor action to aid the pumping of the divertor burial chamber pumps.

The reference design for the TVPS consists of six high vacuum pumping modules each utilizing 2-3 m² of cryopanel made, in part, of a molecular sieve and, in part, of bare metal surfaces. The pumps will be required to pump all isotopes of hydrogen as well as the inert gas helium. The pump is a compound pump located in the reactor cell basement so that sufficient radiation shielding is provided to avoid any nuclear heating. The pumps communicate with the toroidal vessel via a 0.61 m diameter duct in which is located a large all-metal gate-valve. Taking the various conductances and pumping speeds into account, the effective pumping speed at the torus per pumping unit is estimated at 10^4 liter/s for hydrogen gas. The speed for helium is expected to be somewhat less. The main vessel has approximately 400 m² of surface area and to evacuate it to a starting pressure after shut-down of $\sim 1 \times 10^{-8}$ torr, six pumping units are assumed such that a total effective pumping speed of 6×10^4 l/s is possible. The backing system consists of mechanical pumps as well as Zr/Al getter pumps for use in cryopanel regeneration. The ducting to the torus is via the neutral beam ducts so that the number of penetrations through the blanket is minimized.

There has been no attempt to optimize the design of the TVPS. The design given above represents an initial concept that can be iterated upon to provide the final speeds and pumping capacities required. A summary of the three vacuum pumping systems is given in Table 3-21.

3.3.9 FUSILE FUEL HANDLING SYSTEMS

The initial DTHR fusile fuel handling system is based on a once-through concept wherein the plasma exhaust and the contaminated deuterium generated by the neutral beams would be collected and dispatched to a Government.

TABLE 3-21
VACUUM PUMPING SYSTEMS

TORUS VACUUM SYSTEM

Number of Cryopanel Modules	6
Total Effective Hydrogen Pumping Speed	6×10^4 ℓ/s
Base Pressure in Torus	$\sim 1 \times 10^{-8}$ torr
Pressure at End of 15 s Off Time	$\sim 1 \times 10^{-5}$ torr

DIVERTOR BURIAL CHAMBER PUMPING SYSTEM

Zr/Al Getter Target Pump Area	200 m^2
Cryosorption Pumping Panel Area	40 m^2
Operation Pressure in Burial Chamber	10^{-5} torr
Particle Throughput to Target Pump	$\sim 4 \times 10^{22}$ part/s

NEUTRAL BEAM LINE PUMPING SYSTEM

(On a Per Arm Basis)

Area of Pumping Panel (Cryopanel or Zr/Al Getter Panels)	120 m^2
Pumping Speed	$\sim 1.2 \times 10^6$ ℓ/s
Operating Pressure in Neutralizer Ducts	$\sim 3 \times 10^{-3}$ torr
Operating Pressure in Source Region	$\sim 5 \times 10^{-5}$ torr
Operating Pressure in Drift Region	$\sim 5 \times 10^{-6}$ torr
Total Gas Throughput to Panels (Including Desorption From Walls)	400 torr ℓ/s

facility for cleanup, recovery, separation and recycle back to DTHR. Provisions for on-site reprocessing and recycling have been included in the design of the DTHR facility to permit subsequent addition of these systems.

The beam driven operating mode is the most demanding on required input of deuterium and tritium per fusion event and therefore was used as the basis for sizing the fusile fuel handling system. This higher requirement derives from the following effects: 1) the prefill is 100% T, rather than 1:1 D:T for the ignition or near-ignition modes; 2) the T:D ratio in the plasma, during the steady-state burn portion of a pulse, is higher at $n_T/n_D \sim 2.9$ or $\sim 75\%$ T in the plasma, compared to $n_T/n_D = 1$ for the ignition mode; 3) injection of neutral beam D persists throughout the entire burn period; 4) the fraction of the fuel injected into the plasma that undergoes a fusion reaction (on the order of 0.3-1%) is small, therefore demanding more reprocessing-recycling; 5) due to the substantial D_2^0 content of the neutral beam injected material and the long injection period, this imposes a requirement for larger amounts of refueling T in order to sustain the desired n_T/n_D ratio in the plasma; 6) in the current neutral beam injection devices, very substantial amounts of D have to be fed to the devices in excess of that injected into the plasma, amounting to $\sim 1000\%$ of the D entering the plasma. This material becomes contaminated with T and has to be reprocessed. This contribution to the load of the processing-recovery complex is directly related to the beam-on duration per pulse and the beam power and energy level. For the initial DTHR design, collection of the spent material and shipment to the off-site reprocessing facility would proceed coincidentally with the plant operating schedule. The collected residues would be shipped to the DoE facility as either compressed gas or solidified on getters in special containments.

FUELING

The prefill is gaseous T in amount required to yield a plasma with volume, V_p , and density, n_e , for the reference case. Prefilling is accomplished within 0.2 s after completion of vacuum vessel pumpdown. Capability also exists for prefilling the vessel with gaseous 1:1 D:T.

Refueling is in the form of T solid pellets and neutral beam D for the reference case, in amounts necessary to make up for destruction by fusion and removal by the divertor, and to maintain the desired D^+ and T^+ density in the plasma up to the onset of quench. To provide for operational flexibility, the capability also includes means to inject solid pellets of 1:1 D:T and D.

For the reference case, all of the D refueling is accomplished by neutral beam injection, with a capability to inject with powers up to 150 MW and energies up to 175 keV, using six neutral beam arms. However, for conservatism, a beam energy of 200 keV was assumed for estimating the fusile fueling requirements.

Based on present state-of-the-art for neutral beam devices, their relevant operational characteristics assumed for the fusile fuel handling system sizing are:

- The ratio of monatomic to diatomic D power delivered to the plasma is assumed to be 1.14.
- The ratio of atoms of D injected into the plasma to atoms of D required for the neutralizer and subsequently trapped on the cryopanel of the beam vacuum pumps as T contaminated D is 0.0988 for a 200 keV neutral beam arm.
- Each 200 keV neutral beam arm would contain eight sources with ~ 23 MW supplied by the 200 keV monatomic D and ~ 19 MW supplied by the 100 keV diatomic component.

The magnitude of various fusile fuel handling system quantities for DTHR operating in the beam driven mode were determined based on optimistic plasma parameters (950 MW fusion power, $n_e = 1.5 \times 10^{20} \text{ m}^{-3}$). Table 3-22 presents a summary of the fusile fuel system D and T inventories and fueling methods. The values displayed in Tables 3-22 were sized for a full 24-hour operating day, corresponding to 1016 pulses/day, even though normal operation may only be an 8-hour day, to enable short test periods of continuous operation at above nominal design conditions.

TABLE 3-22

DTHR FUSILE FUEL SYSTEM PARAMETER SUMMARY

TRITIUM INVENTORY

On-Site Standby Inventory	
Form	Solidified, UT ₂
Number of Storage-Generator Units	3
Number of Full Operating Day Worths	1
Quantity	
kg	13
M Ci	126
In-Process Inventory	
Cryogenic Distillation Complex	
kg	0.5
M Ci	4.8
Other Processing Equipment	
kg	1.1
M Ci	10.4
Total On-Site Average Inventory	
kg	15
M Ci	140

DEUTERIUM INVENTORY

On-Site Standby Inventory	
Form	Compressed Gas
No. of Storage Units-On-Site/In-Transit	2/1
No. of Full Operating Day Worths	14
Quantity	
kg	436
STP m ³	2,428
In-Process Inventory	
Cryogenic Distillation Complex, kg	1.2
Other Processing Equipment, kg	2.6
Total On-Site Average Inventory, kg	440

FUELING METHODS

Beam Driven Mode	
Gaseous Prefill	100% T
Deuterium Supply	100% D via neutral beams
Refuel Tritium	100% T pellets
Ignition Mode	
Gaseous Prefill	DT
Deuterium Supply and Refuel	Neutral beam, DT pellets
Refuel Tritium	T, DT pellets

Tritium Containment and Cleanup

All equipment in which processing of T bearing materials is conducted is doubly enclosed and the volumes contained within these jackets are to be continuously flushed by a circulating inert buffer gas, which is treated by a cleanup system to maintain a lower cover gas T level. Pumps for T processing operations, in general, are not to have rotating mechanical seals. If this criterion cannot be met, complete containment of the pump and driver within a separate sealed enclosure is required.

Tritium cleanup systems are provided for process streams, containment sweep streams and atmospheres, with adequate processing rates in closed-cycle mode through the cleanup systems. The He sweep streams through the annuli of the storage-generator units and hot and cold U trapping units need a flow adequate to provide complete volume turnover in one hour or a cleanout time of one day in the event of a major release occurrence. Ar sweep streams through the cubicles of the T storage-generator units and the hot and cold U traps need a flow sufficient to provide 25 complete volume turnovers within two days. The He sweep stream of the T and D-T prefill glove box needs a flow that provides 25 complete volume turnovers within one day. The Ar sweep stream through the vault that houses the T and D-T prefill glove box needs a flow that accomplishes 25 complete volume turnovers within two days. Off-gas stream (principally He and H-D) prior to discharge to the environment, needs a flow capability of ~ 0.12 STP m^3/min . The He sweep-purge stream circulated through the secondary containment system (including pump enclosures, double pipe jackets) needs a flow capability of $4 \text{ m}^3/\text{min}$. Ar back-fill atmosphere inserted in the vacuum vessel to facilitate detritiation prior to opening the vessel for maintenance, needs a flow adequate to achieve ~ 30 complete volume turnovers in one day. Air atmosphere of the reactor cell needs a flow rate that provides ~ 30 complete volume turnovers in five days in connection with detritiating the cell following a major release into the cell. Details of these systems were presented in reference 3-3.

3.3.10 ELECTRICAL POWER SYSTEMS

The electrical systems for the DTHR draw power from two commercial 500-kV transmission lines, transform to lower voltages for distribution and utilization at the loads, and store and convert the energy to meet the DTHR operational requirements. The electrical facilities are divided into the systems illustrated in Figure 3-20 which provide the following functions:

- The Transformer Yard and Distribution System changes the incoming 500-kV power to 220-kV, 69-kV, 13.8-kV, and 4.16 kV, and distributes the power to the various DTHR loads.
- The Toroidal Field Coil Supply and Distribution System takes ac power from the 69-kV bus in the transformer yard and converts it to controlled dc current for the superconducting toroidal coils.
- The Poloidal Field Coil Supply and Distribution System * takes ac power from the 69-kV bus in the transformer yard through rectifier-inverter sets that restore the losses of the four generator-flywheel sets by running them as motors to restore the initial speed over the duration of a reactor cycle.
- The Divertor Field Coil Supply and Distribution System takes ac power from the 69-kV bus in the transformer yard and supplies dc current to the divertor field coils through a rectifier.

* The Energy Storage System is part of the Poloidal Field Coil Supply and Distribution System supplied from the 69-kV bus in the transformer yard and requires special design consideration to provide the large fluctuating load requirements of the poloidal fields to minimize load and voltage fluctuations on the electric utility supply system.

Figure 3-20. DTHR Electrical Facilities Conceptual Design

- The Neutral Beam Power Supply and Distribution System takes ac power from the 220-kV bus in the transformer yard and supplies the neutral beam lines and distribution system. It also supplies the load smoothing resistor assemblies that minimize the neutral beam load swings on the 500-kV electric utility power system.
- The Control Systems also receive power from the transformer yard 69-kV bus through unit substations that provide ac power to the various control systems required at the site.
- The Class IE Electrical Systems provide redundant electrical supplies to safety related loads in the plant. These systems normally operate from the main 500-kV bus and provide power at 4.16-kV for large motors, and 480/240/120 V for other ac loads. The dc loads are supplied at 125 V through rectifiers. If the 500-kV bus is lost, diesel generators and batteries supply the critical loads.

3.3.11 ENERGY STORAGE SYSTEMS

The DTHR requires a substantial supply of both pulsed and steady-state electrical power. The principal goal of an integrated energy storage system design is to define an economical means of presenting a smooth, unpulsed load to a utility line, under all operating conditions.

The peak and continuous electrical power requirements are shown in Table 3-24. No attempt at optimization has been made and the potential for considerable reduction in electrical power requirements is apparent. However, it should be noted that 70% of the energy requirements are associated with the neutral beam supplies. The greatest potential for improvement lies in the poloidal field system. The electrical energy losses associated directly with the poloidal field system and indirectly with its cryogenic refrigeration system are high because of the high field (± 9 T) requirements on the compact OH coil, by the need to provide equilibrium and stability for an elongated plasma with external poloidal field coils, and by the relative frequency (~ 2 /minute) of poloidal field pulses. A combined pulse duty cycle and plant availability of 0.2 was assumed for this table.

TABLE 3-24

TABLE OF ELECTRICAL ENERGY FLOW

<u>SUBSYSTEM</u>	<u>PEAK POWER</u>	<u>AVG. POWER/CYCLE</u>	<u>AVG. POWER/YR</u>
NEUTRAL BEAM SUPPLIES	1,135 MW	568* MW	227 MW
NBIS LOAD SMOOTHING SYSTEM	1,135 MW	568* MW	227 MW
POLOIDAL FIELD SUPPLIES	(4,500) MW	281 MW	112 MW
BUNDLE DIVERTOR AND TF COILS POWER	32 MW	32 MW	13 MW
COOLING SYSTEMS	142 MW	142 MW	57 MW
MISCELLANEOUS	1 MW	<u>1 MW</u>	<u>1 MW</u>
TOTAL		1,592 MW	637 MW

* Load smoothing system plus Neutral Beam Supplies combine to equal 1,135 MW maximum during operation.

3.3.12 REMOTE MAINTENANCE SYSTEMS

The basic concepts for assembly and servicing of the reactor can be divided into three areas of considerations: (1) initial assembly; (2) reactor servicing; and (3) blanket refueling. The initial assembly will be a hands-on operation while the reactor servicing and blanket refueling operations will require remote handling of the components. Processing and storage of the hot fuel in the blankets will be done in water pits adjacent and integral with the reactor compartment of the reactor building. The details of these concepts were presented in Reference 3-3.

Servicing and/or replacement of the vacuum vessel wall, limiters, and neutral beam liners will be coordinated with the blanket module replacements. This will necessitate the need for spare replacement components of these items with nearby storage and maintenance facilities. The periodic blanket replacement with alternate module sets may not require repeated removal and installation of the neutral beam injection systems, but will require removal of some of the outer shield sectors and associated parts. Because of the repetition of remote handling work in this area, the blanket refueling cycle will be of keen interest to plant maintenance and servicing and will require remote handling tools of extreme reliability, repeatability, and simplicity comparable to mass production and assembly line techniques.

3.3.13 RADWASTE HANDLING SYSTEM

The primary function of the Radwaste system and its associated equipment is to serve as a controlled area collection station for receiving and handling contaminated and activated materials generated by DTHR operations, and arranging for their shorter range storage and ultimate disposal.

Five broad categories of operations will be provided for: short term and long range holdup of contaminated liquid; handling, storing and packaging of contaminated solids; processing to decontaminate radioactivity bearing liquids; solidification and packaging of highly contaminated liquids for off-site disposal; collecting radioactivity bearing gases followed by removing the radioactivity or preparing them for ultimate disposal.

The amount of radioactive liquid and gaseous waste generated by the DTHR operation is expected to be relatively small. No liquids are used directly in the fusion process or come in direct contact with tritium or the thorium-uranium in the fissile fuel breeding material during normal operation of DTHR.

It is anticipated that the concentration of the neutron activated corrosion products (mainly Co-60) in the primary cooling loop streams will be no more than $1 \times 10^{-6} \mu \text{ Ci/ml}$ or ~ 0.02 of the maximum permissible level for discharge to off-site unrestricted areas. Coolers used in the tritium gas prefill supply system are small closed loop water systems in which the heat is removed from the device by package type refrigeration units with room air cooling of the condensers.

The facility is to be provided with a general purpose low level radioactive waste system which collects all potentially contaminated liquids from the drains of the reactor cell, hot cell, and service areas devoted to servicing-maintaining the DTHR device, as shown in the schematic (Figure 3-21). The three tanks are adequate to hold the entire contents of the coolant loops for the vacuum vessel, fissile fuel breeding modules and shield modules in the event of accidental piping ruptures.

The liquids collected in the holdup tanks periodically will be samples to verify that the activity is acceptable, and then discharged. If necessary, additional treatment or holdup to permit activity decay will be employed to ensure that the release levels are as low as practicable, with a target of no higher than 10% of current permissible levels defined in ERDA 0524.

Small amounts of liquid may be generated during the servicing and maintenance of equipment that is neutron activated or comes in contact with tritium. Such liquids may be organic as well as aqueous. The organics will be collected separately and if the activity of the aqueous liquids is unacceptably high to be routed to the low level waste system, this material will be dispatched to the moderate level collection tanks. Subsequently, these liquids will be solidified (concreted) or collected on an absorbent and packaged as solid material for off-site disposal.

Figure 3-21. General Purpose Low Level Liquid Waste System

REFERENCES - SECTION 3.0:

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4. Yang, T. F., et al., "Reactor Application of an Improved Bundle Divertor," WFPS-TME-104, November, 1978.
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4.0 DTHR BLANKET CONCEPTUAL DESIGN AND PERFORMANCE

A primary objective of the Demonstration Tokamak Hybrid Reactor (DTHR) is the demonstration of the successful integration of a number of major technologies that include a hybrid blanket that has the capability for breeding a substantial amount of fissile fuel. A low temperature, low pressure, water-cooled blanket was selected in March, 1978 as a baseline concept. The blanket was designed for fissile breeding with little potential for electrical power production. It is anticipated that economic Commercial Tokamak Hybrid Reactors (CTHR's) will need to produce not only sufficient electrical power to support the reactor plant, but could require substantial net electrical power to offset high capital cost financing. Consequently, in the latter period of FY'78 blanket design efforts and analyses were carried out with the objective of developing a structurally realistic blanket concept that has the potential for electrical power production. This entails blanket concepts that are capable of operating at relatively higher coolant temperatures and pressures. The DTHR blanket concept that evolved and its potential performance in a commercial application are discussed in Reference 4-1. The highlights and conclusions of that report are presented here.

4.1 BLANKET CONCEPTUAL DESIGN

The reference DTHR blanket concept uses fertile fueled (ThO_2) vertical modules located around the outer torus as shown in Figure 4-1 and 4-2. The basic characteristics of the rectangular module with a two channel cylindrical head are listed in Table 4-1. The blanket consists of vertical fuel rods cooled by two-phase (boiling) water in a cross-flow (radial in-flow) configuration. The blanket evaluation was established to align with the maximum neutron field as shown in Figure 4-3. Typical operating parameters are given in Table 4-2 for 1.2 MW-yr/m^2 of neutron exposure.

Figure 4-1. DTHR Blanket Concept Elevation View

Figure 4-2. DTHR Blanket Concept Plan View

TABLE 4-1
DTHR BLANKET SPECIFICATIONS

WIDTH OF BLANKET MODULE, m	0.23
THICKNESS OF BLANKET MODULE, m	0.65
OVERALL THICKNESS OF FUELED ZONES (3), m	0.53
OVERALL HEIGHT OF BLANKET MODULE, m	5.7
HEIGHT OF FUELED REGION, m	3.7
FUELED SECTION VOLUME, m ³	34
FUEL FORM	ThO ₂ PELLETS
FUEL CLAD MATERIAL	ZIRCALOY 4
CLAD THICKNESS, cm	0.065
FUEL ROD DIAMETER, cm	1.45
PITCH TO DIAMETER RATIO	1.1
BLANKET COMPOSITION, v/o	
FUEL	45.0
CLAD	9.5
COOLANT	31.0
S.S. - 316	13.5
VOID	1.0
INITIAL FUEL LOAD, Tonnes (ThO ₂ , 96.5% Theo. density, 9.65 g/cm ³)	149

Figure 4-3. Fraction of Fusion Neutrons Striking DTOR Blanket Modules

TABLE 4-2

DTHR BLANKET OPERATING CONDITIONS
(AT 1.2 MW-Yr/m² NEUTRON EXPOSURE)

AVERAGE BLANKET POWER DENSITY, W/cm ³	15
PEAK BLANKET POWER DENSITY, W/cm ³	55
PEAK FUEL ROD POWER DENSITY, W/cm ³	100
MAXIMUM WATER MASS VELOCITY, g/cm ² -s	6
INLET COOLANT PRESSURE, BARS	33.3
INLET COOLANT TEMPERATURE, °C	222
OUTLET COOLANT TEMPERATURE, °C	237
INLET WATER SUBCOOLING, °C	20
HOT ROD CLAD SURFACE TEMPERATURE, °C	250
HOT ROD MAXIMUM CLAD TEMPERATURE, °C	265
HOT ROD MAXIMUM FUEL TEMPERATURE, °C	655
DNB RATIO	3.2
COOLANT PRESSURE DROP THROUGH THE BLANKET, BARS	2.4
COOLANT PUMPING POWER TO BLANKET THERMAL POWER, %	0.07
FILM TEMPERATURE DROP, °C	13
CLAD TEMPERATURE DROP, °C	15
HELIUM GAP ΔT , °C	126
FILM SURFACE TEMPERATURE, °C	391
RADIAL TEMPERATURE DROP THROUGH ThO ₂ , °C	264

4.1.1 BLANKET MODULE BUNDLES

The blanket is surrounded by shielding, magnetic coils, piping, and structure which makes it difficult to maintain and replace under remote operating conditions. For this reason simplicity was stressed in the overall design of the DTHR module. The rectangular geometry allows for blanket bundle removal by basically vertical lifting and lowering with offset handling tools working between the TF coils.

The DTHR module bundles are composed of four individual blanket modules which are secured together by a structural framework so that the complete module can be installed and subsequently removed from the reactor as a unit. Each of the four blanket modules have a rectangular cross-sectional shape in which the surface closest to the plasma is semi-cylindrical (see Figure 4-4). The outside rectangular dimensions are approximately 23 cm wide, 53 cm deep and the radius of the semi-cylinder front section is ~ 12 cm, which results in a total depth of 65 cm. The vertical height of the individual blanket module is 5.7 m, which is made up of 3.7 m of fuel and the remainder being the fuel rod gas plenum, inlet and outlet manifolds, and the upper and lower support structures. The equivalent poloidal blanket coverage factor, b_p , is 43% as shown in Figure 4-3. If blanket modules covered the entire outer surface of the torus, the poloidal coverage would be 67%. When four of the reference DTHR blanket modules are grouped together to form a module bundle, the horizontal cross-sectional space is approximately 81 cm wide by 95 cm deep (this includes the outside structural supporting members).

The module bundles will be located in one group of three and four groups of six around the circumference of the torus (see Figure 4-2) with a central module bundle being located in the clear opening between two toroidal field coils and the other two module bundles being located one on each side and slightly under the toroidal field coils. The resulting blanket toroidal coverage factor, b_t , is 56%. If optional positions were also used, a $b_t = 65\%$ could be achieved.

BLANKET MODULE BUNDLE

SIMPLIFIED COOLANT FLOW SCHEMATIC

DETAIL "B"
BLANKET MODULE

SECTION "A-A"

Figure 4-4. Reference DTHR Blanket Module Design

4.1.2 BLANKET MODULES

The blanket module (pressure vessel/fuel bundle assembly) design has three zones, each zone having a radial depth of 14 cm, allowing 10 rows of fuel rods to be accommodated per zone (see detail view, Figure 4-4). The pitch and diameter are 1.6 cm and 1.45 cm, respectively. Overall, each zone has 120 rods with a total of 360 rods per blanket module. Pressurized water (~ 33 bars) at a subcooled (20°C) inlet temperature (222°C) is supplied to the back of the module and exits in the inner cylindrical nose of the module at saturation temperature. The exit steam plenum design is constructed from solution annealed 316 stainless steel with the split cylindrical components welded to form the bullet-like cross-section. The nose is a 180° sector of a ~ 23 cm O.D. by 0.46 cm wall circular tube. The three rectangular fuel zone ducts are made from 20% cold-worked 316 SS material. Their outside dimensions are 23.5 cm wide by 16.5 cm deep with 1.3 cm thick walls. The outside surfaces in the radial direction of each rectangular duct are undercut 0.64 cm to leave a weld lip on each edge. The remaining 0.64 cm wall thickness on the radial sides is drilled to form a tube sheet with 1.3 cm diameter holes on a 3 cm triangular pitch. The inlet plenum, consisting of a 2.5 cm thick by 23.5 cm wide 20% CW-316-SS plate, is machined on one side in the same manner as the rectangular ducts. The nose is welded to the lip formed in the first duct, while the other lip is welded to a lip of the second duct. The third duct is similarly attached. The inlet plenum is welded to the remaining lip of the third duct. This type of welded construction is required to limit the heat affected zone to the lip region without affecting the 20% cold worked properties at the critically stressed inside corners of the duct side walls. In this arrangement, three zones with inside dimensions 21 cm wide by 14 cm deep are formed to provide the three fuel rod bundle zones.

4.1.3 FUEL ASSEMBLY DESIGN

Figure 4-5 shows the reference fuel element conceptual design. The fuel rod concept is based on the LWBR blanket fuel rod design except that the rod is longer. The overall outside diameter of the fuel rod is 1.45 cm. The clad thickness is 0.065 cm with a helium gap of 0.008 cm between the thorium oxide (ThO_2) fuel and Zircaloy-4 clad. The pitch to diameter ratio is 1.1. The fuel element concept is identical to that proposed for the low temperature/pressure DTHR design⁽⁴⁻¹⁾.

Figure 4-5. DTHR Fuel Rod Assembly - Grid Assembly Conceptual Design

The fuel rods are supported at the top, with the vertically spaced grid assemblies acting as flow separators and providing fuel rod stability. The grid assembly concept is similar to the LWBR concept in that it is light construction to minimize neutron capture. The top and bottom surfaces are covered to prevent coolant from flow upward or downward between the six controlled passages. The sealing ring at each rod is designed to permit controlled leakage through the grids for cooling the grid components. Use of positioning ball is optional depending on the movement (both vertically and radially) expected for the rods. Simple leaf springs can be used in lieu of these balls. The location of gas plenums at the grids is also optional, depending on top and bottom rod clearance requirements and the fact the fuel pellets at the grids would receive less irradiation than adjacent pellets. The coolant flow path is shown by the dotted arrowhead lines as it enters at the bottom, flows up the outboard inlet plenum, flows radially past the fuel rods and into the steam plenum near the plasma side of the blanket. The front wall cooling flows directly at the bottom (beneath the lowest grid) to the front chamber.

4.2 BLANKET PERFORMANCE

The following subsections discuss the structural analysis, thermal-hydraulic analysis and neutronic performance of the boiling water-cooled DTHR blanket concept.

4.2.1 STRUCTURAL ANALYSIS

The structural evaluation of the DTHR duct sidewalls and fuel rod cladding was made for the reactor operating with a 20% plant availability over a three year period, or 0.6 years of continuous plant operation. Over 0.6 years of continuous reactor operation, there is a total of 236,000 plasma on-off cycles and the EOL fast fusion fluence ($E > 0.1$ MeV) is 1.36×10^{26} n/m² while the nuclear heating in the duct walls and the peak power density in the fuel rod are 13 and 100 watts/cm³, respectively.

The DTHR blanket and fuel rod clad stress analysis derived the stresses over the worst case duty cycle of loading. The pressure and thermal stresses were computed from simple linear classic methods. The use of such criteria is valid, provided that the equivalent stress is less than the associated proportional elastic limit (PEL) stress of the material.

The structural evaluation of the DTHR blanket duct side wall and fuel rod cladding to protect against coolant and fuel leakage was based on hypothetical surface cracks at BOL with a depth which slowly grows through the side wall causing an eventual leakage at EOL. The FBR Core Components Criteria recommends Linear Elastic Fracture Mechanics (LEFM) for irradiated as well as unirradiated structures. This is based on the fact that the elastically calculated stresses were less than the PEL of the structural materials, therefore, LEFM structural evaluation is considered acceptable for the DTHR blanket duct side walls and fuel rod cladding.

The DTHR blanket duct criteria is similar to the criteria developed for the highly irradiated ORNL blanket module⁽⁴⁻²⁾, except that coolant leakage by brittle fracture was neglected. The latter is justified as the EOL plane strain fracture toughness in the 20% CW-316-SS DTHR blanket duct wall is expected to be high because of the relatively low neutron irradiation prior to planned replacement (modules removed every one to three years for destructive examination).

Based on available materials data for the 20% cold-worked 316-SS duct and zircaloy clad, the structural evaluation showed that the DTHR blanket duct wall and fuel rod clad designs are acceptable for a three year test period. However, it should be noted that available creep and fatigue crack growth data in a fast fusion neutron spectra are currently not available. The latter data are required to provide confidence in the current design, or prior to extending the time between planned replacements. A summary of the DTHR blanket duct wall and fuel rod clad margins of safety is given in Table 4-3. It should be noted that oxidation and corrosion effects were not specifically accounted for in this analysis. Since the zircaloy clad is generally considered a viable material for boiling water applications and the stainless steel module walls are relatively thick, corrosion should not be a problem in the short-term DTHR blanket testing. However, commercial applications which intend to reuse the module housings for several refuelings or use higher coolant temperature conditions on the clad will have to be evaluated in detail.

TABLE 4-3
DTHR BLANKET MODULE MARGINS OF SAFETY

<u>COMPONENT</u>	<u>ALLOWABLE CRACK GROWTH (cm)</u>	<u>CALCULATED CRACK GROWTH (cm)</u>	<u>MS*</u>
DUCT WALL	2.5×10^{-2}	1.4×10^{-2}	0.8
FUEL ROD CLAD	1.6×10^{-3}	1.94×10^{-5}	81.0

$$*MS = \frac{\text{ALLOWABLE VALUE}}{\text{CALCULATED VALUE}} - 1$$

4.2.2 THERMAL-HYDRAULICS ANALYSES

The blanket thermal-hydraulic analyses included the calculation of the peak temperatures and temperature distribution in the hot rod, the DNB ratio (departure from nucleate boiling), and the coolant pumping power. Also included in this section are discussions of the coolant selection and front wall cooling.

COOLANT CONSIDERATIONS AND CONCLUSIONS

Scoping design calculations have identified boiling water with radial coolant flow as an attractive method for blanket cooling for a DTHR as well as a CTHR. Neutronic calculations have been performed primarily for radial inflow with single phase water at the blanket inlet and 80% quality steam at the blanket outlet. Although radial outflow is more attractive from the standpoints of thermal hydraulics and maximum power production, the fissile breeding capability becomes reduced. Thus, there is a trade-off between fissile breeding and power production. The selection of the optimum mode of coolant flow depends largely on the overall economics of the hybrid reactor, which is dependent on the net electrical power production, the net fissile breeding rate and the amount of tritium that can be bred in the hybrid reactor. For commercial reactors, equilibrium blanket operating conditions must be established. This can be achieved through fuel shuffling. Neutronic analyses with radial outflow of boiling water as coolant and with different fuel management schemes must be carried out to provide data for a comparative economic analysis.

A change from boiling water with radial inflow to radial outflow would have little effect on the mechanical design of the blanket; however, the thermal hydraulic requirements can be relaxed appreciably, because the peak power density region corresponds to maximum heat transfer coefficients and maximum DNB ratio. Therefore, this mode of blanket cooling could well lead to a significant high power production capability. For a given neutron exposure, the determination of the blanket neutronic performance requires a tandem iterative solution between neutronic and thermal analyses. This is due to the fact that in a two-phase boiling water cooled blanket, the blanket neutronic performance is dependent on the water density distribution. Conversely, the water density distribution is dependent on the power density distribution. The results presented here have been based on nearly-converged solutions; therefore, they can be considered to be only preliminary. The tandem iterative solutions can be complicated appreciably with fuel shuffling, as in the case of a blanket design for a commercial hybrid reactor.

The most critical issues that have been identified for the DTHR blanket concept are: (1) the potential problems due to radial two-phase flow maldistribution, instabilities, and stratifications; (2) the fuel rod vibration and fretting control; and (3) the clad and structural materials corrosion problem with two-phase flow. Although increased steam quality throughout the blanket may alleviate some of these problems, early RD&D using blanket module mockups for flow simulations using electrically heated rods are needed before commitment of the DTHR concept. Further analyses of tritium diffusion and pick-up in the water coolant will also be needed.

EXIT COOLANT QUALITIES AND FILM TEMPERATURE DROPS

For a given mass velocity, the local coolant quality has a significant effect on the heat transfer coefficient and hence the film temperature drop and the clad temperatures. It can be shown that the two phase heat transfer coefficient for forced convective boiling or vaporization increases with decreasing quality. Hence, the lowest heat transfer coefficient is formed at the blanket exit. For radial coolant inflow, the hot rod is also found at the exit end of the blanket. Thus, for this flow scheme, hot rod cooling requirements determine the coolant operating conditions allowed. The hot rod film temperature drop was calculated for a hot rod power density of 100 W/cm^3 over a range of exit fluid qualities and for two pressure levels. The results are presented in Figure 4-6. It is clear that the film temperature drop increases gradually with increasing quality up to a quality of $\sim 80\%$. Beyond this quality, the film temperature drop increases rapidly with increasing quality.

Figure 4-6. Effect of Blanket Exit Steam Quality on Hot Rod Film Temperature Drop.

On the basis of maximum allowed clad temperature of 300° C and a maximum assumed clad temperature drop of 39° C (at end of life), the maximum clad surface temperature allowed is 261° C. The saturation temperatures corresponding to 26.7 and 33.3 bars are 229° C and 242° C respectively. The maximum allowed film temperature drops are therefore 32° C and 19° C respectively. From Figure 4-6 it can be seen that the maximum allowed exit quality is on the order of 90% for a neutron exposure of 1.2 MW-Yr/m² (3 year exposure in DTHR). An exit quality of 0.8 was selected for the DTHR to allow for more flexible blanket operations. However, the oxidation and corrosion effects of boiling water coolants on the clad and module structures must be closely evaluated in a commercial application to determine performance changes and lifetime limitations for potential reuse of components.

FUEL ROD TEMPERATURES

It can be shown that the two-phase heat transfer coefficient increases with decreasing quality, while the fuel rod power density decreases with increasing radial distance from the hot rod. For these reasons, the fuel rod surface temperatures decrease rapidly, radially away from the hot rod. These effects are illustrated in Figure 4-7 for 1.2 MW-yr/m² of neutron exposure. The hot rod fuel centerline temperature at 1.2 MW-yr/m² neutron exposure is only 655° C (1211° F). The temperature distribution through the hot rod is tabulated in Table 4-2.

FRONT WALL COOLING

The objective of this analysis was to determine if the coolant operating characteristics are adequate to keep the operating temperature of the blanket front wall below the maximum allowable temperature of the structural material.

The preliminary thermal hydraulic analysis of the high quality steam flowing past the module nose indicated that the flow velocities and heat transfer coefficients were not adequate to limit the maximum temperature of the structure to the required temperature. Thus, it was necessary to adopt a separate front wall cooling scheme. The flow arrangement suggested was a single phase, superheated steam flowing axially parallel to the split cylindrical nose of the module. The flow path is prescribed by the front wall boundary and a semi-circular flow separator which separates the front wall cooling stream from the coolant exit plenum of the blanket. The operating pressure of the two coolant

Figure 4-7. Radial Fuel Rod Film Temperature Drop and Steam Quality Distributions.

streams on either side of the flow separator will be approximately equal; the temperature difference between the two streams will be low and equal to the magnitude of superheating allowed in the front wall coolant stream. The use of superheated steam as the front wall coolant has a number of design advantages associated with this choice. A primary factor is that a common coolant allows the design to tolerate leakage across the separator boundary without the fear of coolant contamination. Also common coolants allow the use of one common inlet header. The front wall coolant can be bled from the primary inlet manifold. At the collecting or exiting manifold both coolants can utilize the same manifold and piping. The use of a low density superheated coolant for the front wall yields an axially uniform, low density medium for the neutrons to pass through, thus allowing the neutrons to keep their high energy level.

However, the use of a flow separator does increase the structure to fuel volume ratio, degrading the neutronic performance, and increasing the complexity of the mechanical design and fabricating cost of the module.

4.2.3 POWER CONVERSION CONSIDERATIONS

Since one of the major goals of blanket design was to develop a concept that would have the capability of appreciable electrical power production, the effect of power conversion efficiency on blanket operating conditions was analyzed. The basic heat transport and power conversion system considered is a direct conversion steam turbine generator, shown schematically in Figure 4-8. The wet steam produced in the blanket goes to a steam separator and then directly to the steam turbine. The turbine exhaust is condensed in a condenser. The condensate is then preheated by waste heat from other subsystems of the DTHR such as the divertor and the inner shield. The feed water then enters the blanket. A direct steam cycle is possible if the tritium accumulation in the blanket coolant is sufficiently low. This is achieved in the DTHR through the removal of tritium in the separate coolant circuit of the vacuum vessel.

The efficiency of a steam Rankine thermodynamic cycle is Carnot - efficiency limited. This implies that the thermal efficiency is dependent on the maximum turbine inlet temperature. This is in turn a function of the maximum allowable

Figure 4-8. Simplified Flow Diagram of DTHR Power Conversion System

fuel clad and blanket structure temperatures. A steam power plant ideal thermal efficiency is a product of the efficiency of the Carnot cycle, the efficiency of the prime mover, and the fluid efficiency factor. The Carnot efficiency is simply related to the turbine inlet steam temperature and the condenser temperature. The fluid efficiency factor is related to these temperatures and a work factor, which is the ratio of energy available as work in the cycle to the energy available in the Carnot cycle. The performance of typical water reactor steam conversion systems to an empirical equation developed from these considerations is shown in Figure 4-9. It is seen that with the exception of the EBWR (Experimental Boiling Water Reactor), the empirical equation is in good agreement with the gross thermal efficiencies of typical steam power plants. Therefore, the equation was used to extrapolate from DTHR blanket and steam operating conditions and obtain the resulting gross thermal efficiency of $\eta_{th} \sim .3$.

4.2.4 NEUTRONIC PERFORMANCE

The blanket neutronic performance was evaluated using the coupled neutron transport burn-up code system ANISN-CINDER-HIC. The neutron transport was calculated with the one-dimensional discrete ordinates code ANISN⁽⁴⁻³⁾ using an S_8 treatment of the angular quadrature and a P_3 approximation to the anisotropic scattering. The isotope burn-up and depletion were computed using a combination of the CINDER⁽⁴⁻⁴⁾ and HIC⁽⁴⁻⁵⁾ codes which were appended⁽⁴⁻⁶⁾ to the ANISN code. A one-dimensional cylindrical geometry model was used with the cylinder axis along the centerline of the torus. This was selected as the most appropriate for the assumed tokamak hybrid design in which the ThO_2 blanket is placed only on the outer part of the torus. Due mainly to accessibility limitations and maintenance requirements, the inner portion of the torus was assumed to be used only for tritium breeding and radiation shielding. The isotropic 14 MeV D-T fusion neutron source is distributed uniformly over the plasma region. The plasma is surrounded by a void (scrape-off) layer and then by the vacuum vessel/first wall. For these calculations, the tokamak major radius R_0 was 5.2 m, the plasma minor radius was 1.2 m, the scrape-off layer thickness was a 0.2 m, and the first wall was a 0.4 cm thick layer of stainless steel.

Figure 4-9. Effect of Turbine Inlet Temperature on Gross Thermal Efficiency of Steam Conversion Plants.

The nuclear performance of the blanket is characterized by the following parameters:

$$F = \frac{\text{net number of fissile } (^{233}\text{U}) \text{ atoms produced}}{\text{fusion neutron}}$$

$$f = \frac{\text{number of fission reactions in the blanket}}{\text{fusion neutron}}$$

M = blanket D-T fusion neutron energy multiplication factor

P_B = instantaneous blanket thermal power production

G = hybrid fuel production of ^{233}U .

For the vertical, cylindrical geometry used in the neutronics calculation with a blanket on the outer part of the torus only, not all of the fusion neutrons directly strike the blanket. However, by using the blanket coverage factors b_p and b_t , the macroscopic performance parameters such as power and fuel production can be accounted for correctly. The material compositions used to calculate the performance of the reference blanket in the DTHR are shown in Table 4-4. For calculational purposes the blanket was divided into three zones with thicknesses of 10, 14 and 15 cm. An average water coolant density was used in each zone as determined from an overall water density distribution shown in Figure 4-10, which shows the radial distribution of the coolant density. This density distribution is beneficial to the neutronic performance of the blanket since it allows a low density coolant at the front of the blanket to increase the fusion neutron multiplication and a higher density coolant at the rear of the blanket to slow down and reflect the neutrons. A stainless steel vacuum vessel of thickness 0.37 cm was assumed in the calculation for the DTHR blanket. Also, the blanket inlet plenum was modeled as a 10 cm thick zone of low density coolants with a steel volume fraction of 0.083.

TABLE 4-4
REFERENCE DTHR BLANKET MODULE
ASSUMED IN THE NUCLEAR CALCULATIONS

VACUUM VESSEL THICKNESS, cm	0.37
FUEL SECTION THICKNESS, cm	40
ThO ₂ VOLUME FRACTION	0.454
H ₂ O VOLUME FRACTION	0.306
ZIRC VOLUME FRACTION	0.095
SS VOLUME FRACTION	0.135
CAP (VOID) VOLUME FRACTION	0.010
WATER DENSITY, $\frac{\text{gms}}{\text{cm}^3} \left(\frac{1\text{b}}{\text{ft}^3} \right)$	
FIRST ZONE (10 cm) (NEAREST PLASMA)	0.032 (2.0)
SECOND ZONE (15 cm)	0.080 (5.0)
THIRD ZONE (15 cm) (NEAR OUTER SHIELD)	0.513 (32.0)
BLANKET EXIT PLENUM	
THICKNESS, cm	10.0
SS VOLUME FRACTION	0.083
H ₂ O VOLUME FRACTION	0.700
H ₂ O DENSITY, $\frac{\text{gm}}{\text{cm}^3} \left(\frac{1\text{b}}{\text{ft}^3} \right)$	0.021 (1.33)

Figure 4-10. Radial Water Density Distribution for Radial Inflow of Boiling Water.

The performance of this blanket at beginning-of-life (BOL) and after three years of exposure in the DTHR is shown in Table 4-5. It can be seen that the fissile production has decreased by about 6% after three years relative to BOL, for an average production rate of ~ 165 kg ^{233}U per year. The power produced in the blanket has increased by $\sim 80\%$ due to increased ^{233}U fissioning, but the total power including that released in the shields and from alpha particles has increased by only about 25%. Note that after three years the average enrichment of ^{233}U in Th is 0.36% while the peak is 0.51%. The typical enrichment required for PWR startup is $\sim 3\%$. The potential tritium breeding ratio in these blankets, as measured by the number of neutron absorptions in the shield regions is ≈ 0.6 . The performance of a similar two-phase water cooled blanket module under a much longer exposure period is discussed in Section 4.3 of this report where the prospects for extrapolation to commercial reactor performance are examined.

For the analysis of the nuclear performance of the DTHR blanket, a 15 cm thick reflector zone with a composition consisting of 35% water and 65% SS was assumed to be directly behind the blanket. In order to optimize the blanket performance, it was of interest to investigate the effect of varying the reflector composition on the ^{233}U production. The results of this analysis are shown in Table 4-6, where 15 cm thick reflectors of all SS or all water as well as a shield of SS and borated water are compared to the reference reflector. It can be seen that the reflector composition has a small ($< 2\%$) effect on the ^{233}U production at beginning-of-life (BOL). This occurs because the blanket is sufficiently thick to absorb most of the neutrons, at least at BOL. However, the reflector would have a greater effect on the blanket performance after exposure when the neutron population has increased and the back blanket leakage is large.

TABLE 4-6
EFFECT OF THE REFLECTOR COMPOSITION ON BOL BLANKET PERFORMANCE

<u>REFLECTOR COMPOSITION</u>	<u>RELATIVE ^{233}U PRODUCTION</u>
65% SS 35% H_2O	1.000
100% SS	1.0157
100% H_2O	1.0093
35% BORATED WATER - 65% SS SHIELD	0.9973

TABLE 4-5

PERFORMANCE OF THE DTHR REFERENCE BOILING WATER COOLED ThO₂ BLANKET

	<u>AT BOL</u>	<u>AFTER 3 YEARS</u>
EXPOSURE, $\frac{\text{MW-YRS}}{\text{m}^2}$	—	1.2
F, $\frac{\text{NET NO. } ^{233}\text{U ATOMS}}{\text{FUSION NEUTRON}}$	0.62	0.58
M, MULTIPLICATION FACTOR	1.63	2.72
ENRICHMENT, w/o ^{233}U IN Th		
FIRST ZONE (10 cm)	—	0.51
SECOND ZONE (15 cm)	—	0.38
THIRD ZONE (15 cm)	—	0.25
BLANKET AVERAGE	—	0.36
SHIELD ABSORPTIONS, $\frac{\text{NEUTRONS}}{\text{FUSION NEUTRON}}$	0.60	0.64
G, $\frac{\text{kg } ^{233}\text{U}}{\text{YEAR}}$ PRODUCTION RATE	170	160
CUMULATIVE ^{233}U , kg	—	495
PEAK BLANKET POWER, MWt	342	619
AVERAGE THERMAL POWER ^(a) , MWt	280	508
AVERAGE ELECTRICAL POWER OUTPUT ^(b) , MW _e	84	150

(a) Assuming a duty cycle factor of 0.82 during short test periods.

(b) Assuming a duty cycle factor of 0.82 and power conversion equipment with 0.3 overall efficiency.

4.2.5 TRITIUM BREEDING POTENTIAL

In view of the limited operating goals of the DTHR, tritium breeding in the initial blanket concepts was not included. As such, the ThO_2 blanket modules were assumed to be placed on the outer portion of the torus only, with shield regions on the inner portion and between the blanket and the toroidal field coils. During the course of operation, a significant number of neutrons are absorbed in both of these shield regions, with a primary source being reflection and back-scattering of neutrons from the outer blanket toward the inner shield region. Later phases of DTHR testing would include tritium breeding blanket modules.

In a commercial hybrid application there is the possibility of placing lithium bearing materials in the front portions of the shield regions and using the available neutrons to breed tritium. Neutronic calculations show that this has the potential for providing from ~60% to 80% of the necessary tritium with the boiling water cooled blanket. The remaining tritium could be produced by making the blanket zone thinner and allowing more neutrons to leak into the shields (in the back of the blanket) or by placing tritium breeding blankets above and below the neutral beam injectors and the bundle divertor (in lieu of fissile breeding blankets).

Tritium self-sufficiency could also be provided for by using some lithium bearing materials in the ThO_2 blanket itself instead of using two separate regions for the Th and Li. This mixing of the Th and Li could have several benefits from a neutronics standpoint, including suppressing the thermal flux and, thus, the fissile burn-out and also lowering the blanket Th inventory, possibly increasing the fissile enrichment. However, it would also decrease the fissile production by allowing neutrons to be absorbed in the Li. This would also occur if the blanket thickness was decreased to increase the back leakage. The use of a mixture of Li and Th also offers the possibility of smoothing the increase in blanket power production with exposure by preventing much of the thermal neutron absorption in fissile material. The use of Li compounds in a steam or water cooled blanket presents potential problems with regard to tritium contamination of the water. The use of a mixture of Li and Th materials offers possible advantages in the blanket design for tritium self-sufficiency. Another alternative is to examine the tritium production potential of the LWR or even design a reactor specifically for tritium production, if this proves to be more economical.

4.3 EXTRAPOLATION OF DTHR TO A COMMERCIAL APPLICATION

4.3.1 PARAMETER DIFFERENCES

The most important tokamak parameters with regard to the nuclear analysis of the blanket performance are assumed to have the values shown in Table 4-7 for either the DTHR or a commercial hybrid. As shown in the table, one of the main differences assumed is in the plant availability and plasma duty cycle. This results in a factor of three increase in the integrated wall loading achievable in one year's time of operation, from 0.4 MW-yr/m^2 in the DTHR to 1.2 MW-yr/m^2 in the commercial hybrid. It is the value of the plant factor averaged wall loading which is needed to calculate the fusion neutron source strength used in the ANISN-CINDER-HIC calculations.

There are also differences in the assumed blanket wall coverage fractions as shown in Table 4-7. The value for the poloidal coverage factor, b_p , of 0.43 in the DTHR represents the approximate fraction of the plasma fusion neutrons intercepted by a vertical blanket module of 3.7 m long fuel rods. For the commercial application, a value of $b_p = 0.67$ is assumed which represents a conservative blanket coverage on only the outer one-half of the vacuum vessel poloidal surface. The toroidal coverage factor, b_t , which represents the fraction of the tokamak toroidal circumference which is covered by blanket modules, is assumed to increase from 0.56 in the DTHR to 0.87 in the commercial hybrid. The value of 0.56 assumes blanket modules are only located between 9 of the 16 TF coils, while the value of 0.87 represents placing blanket modules in all areas except those needed for the neutral beam injectors and the bundle divertor chamber.

The performance of the two phase (boiling) water cooled blanket from BOL to an integrated neutron exposure of 8.4 MW-yr/m^2 is shown in Table 4-8. The fissile production rate decreases by about 27% over this analytical period to a time averaged value of $\sim 1200 \text{ kg } ^{233}\text{U}$ per year, and the blanket volume averaged enrichment is $\sim 2.2\%$. The potential tritium breeding ratio, as measured by the number of neutrons absorbed in the shield regions, increases from ~ 0.6 at BOL to ~ 0.9 at 8.4 MW-yr/m^2 . Of this value, $\sim 79\%$ or 0.79 neutrons are absorbed in the inner shield due to a large reflection of neutrons from the blanket towards the inner shield region. The remaining shield absorptions occur from

TABLE 4-7
TOKAMAK PARAMETERS OF THE DTHR AND A COMMERCIAL HYBRID

	<u>DTHR</u>	<u>COMMERCIAL APPLICATION</u>
MAJOR RADIUS, m	5.2	5.2
MINOR RADIUS, m	1.2	1.2
PLASMA ELONGATION	1.6	1.6
NEUTRON WALL LOADING, MW/m ²	2	2
PLASMA DUTY CYCLE (DC), % ON	50	82
PLANT AVAILABILITY (PA), % AVAIL.	40	70
PLASMA D-T POWER, P _f , MW _t	950	950
PLANT FACTOR, DC x PA	0.20	0.60
INTEGRATED WALL LOADING PER YEAR OF OPERATION, MW-yr/m ²	0.4	1.2
POLOIDAL BLANKET WALL COVERAGE FRACTION, b _p	0.43	0.67
TOROIDAL BLANKET WALL COVERAGE FRACTION, b _t	0.56	0.87

TABLE 4-8
BOILING WATER COOLED ThO₂ FERTILE BLANKET
COMMERCIAL PROJECTION PERFORMANCE

	AT BOL	AFTER EXPOSURE			
EXPOSURE, $\frac{\text{MW-Yrs}}{\text{m}^2}$	—	1.2	2.4	4.8	8.4
F, $\frac{\text{NET NO. } ^{233}\text{U ATOMS}}{\text{FUSION NEUTRON}}$	0.71	0.68	0.65	0.60	0.52
M, ENERGY MULTIPLICATION FACTOR	1.75	3.0	3.8	5.6	8.0
ENRICHMENT, w/o ^{233}U IN Th					
FIRST ZONE (10 cm)	—	0.5	1.0	1.75	3.0
SECOND ZONE (15 cm)	—	—	—	—	2.4
THIRD ZONE (15 cm)	—	—	—	—	1.3
BLANKET AVERAGE	—	—	—	—	2.2
SHIELD ABSORPTIONS, $\frac{\text{NEUTRON}}{\text{FUSION NEUTRON}}$	0.61	0.66	0.69	0.77	0.88
G, $\frac{\text{kg } ^{233}\text{U}}{\text{YEAR}}$ PRODUCTION RATE	1420	1360	1300	1200	1040
CUMULATIVE ^{233}U , TONNES	—	1.2	2.4	4.9	8.6
AVERAGE ^{233}U PRODUCTION, kg/YEAR	—	1390	1360	1280	1200
k _{eff} , BLANKET EIGENVALUE (SOURCE-OFF) ≈	0.02	—	—	—	0.49

neutrons leaking out the back of the blanket. If tritium self-sufficiency is desired, the blanket could be made thinner to increase the back leakage. Notice also from Table 4-8 that a hypothetical blanket is still far from critical even after seven years of exposure, due to the hard spectrum which results from the boiling water coolant.

4.3.2 WATER COOLANT CONDITION CONSIDERATIONS

The performance under long irradiation times of a boiling water cooled blanket module can be compared to that of the low temperature, low pressure water cooled blanket concept developed in early stages of this program and presented in detail in Reference 4-7. The blankets are considered for use in a commercial hybrid with a plant factor averaged wall loading of 1.2 MW m^{-2} and with the commercial blanket coverage parameters (Table 4-7).

The blanket material compositions for the two-phase water and single-phase water cooled designs are shown in Table 4-9. The two-phase water cooled blanket was made 15 cm thicker than the water cooled concept in order to keep the back blanket neutron leakage about the same at BOL. This was necessary because of the decreased moderation with the two-phase water and thus, results in a higher blanket fuel inventory. Another major difference between the two designs is in the blanket module structural material. The water cooled blanket was designed for low temperature and pressure operation, which permitted the use of zircaloy-4 as the structural material. For the higher pressure two-phase water cooled design, it was necessary to use stainless steel as the structural material. This will affect the neutronic performance somewhat since the steel has a higher parasitic absorption cross-section than the zircaloy. However, with the two-phase water coolant the neutron spectrum will be harder and the parasitic losses will not be as important as in the softer water cooled spectrum. The main objective in utilizing steam coolant was to prevent the thermal fissioning of the ^{233}U which was previously found to occur in the water cooled case and at the same time to provide for the possibility of electrical power generation by operating at higher coolant temperatures and pressures.

TABLE 4-9

COMPARISON OF MATERIAL COMPOSITIONS FOR THE WATER COOLED
AND BOILING WATER COOLED BLANKET CONCEPTS

	<u>TWO-PHASE BOILING WATER COOLED</u>	<u>WATER COOLED</u>
BLANKET FUEL SECTION THICKNESS, cm	40	25
STRUCTURAL MATERIAL	STAINLESS STEEL	ZIRCALOY-4
ThO ₂ VOLUME FRACTION	0.454	0.569/0.386*
H ₂ O VOLUME FRACTION	0.306	0.241/0.474*
Zr VOLUME FRACTION	0.095 (Clad Only)	0.716/0.130* (Clad and Structure)
SS VOLUME FRACTION	0.135	—
WATER DENSITY, $\frac{\text{gms}}{\text{cm}^3}$		
FIRST ZONE (10 cm)	0.032	1.00
SECOND ZONE (15 cm)	0.080	1.00
THIRD ZONE (15 cm)	0.513	—

* First Zone/Second Zone

In Figure 4-11 the fissile enrichment in the first 10 cm and the net number of fissile atoms per fusion neutron for the two blanket concepts are shown as a function of the integrated exposure. Note that in the commercial hybrid the annual exposure is 1.2 MW-yr m^{-2} , so that a value of 8.4 MW-yr/m^{-2} would represent seven years of exposure on the blanket module structures. The two-phase water cooled, stainless steel structure blanket reaches an enrichment of $\sim 3\%$ in the first 10 cm in a nearly linear rate of increase, while the water cooled, zircaloy structure blanket approaches a near equilibrium enrichment $\sim 1.86\%$ after four years. Correspondingly, the net ^{233}U production per fusion neutron approaches zero in the water cooled case while it is still fairly large in the two-phase water cooled design.

In Figure 4-12 the blanket energy multiplication and number of neutrons absorbed in the shield regions are shown for the two designs. The energy multiplication (M) for the water cooled case climbs to a high value of ~ 28 , while the two-phase water cooled case only reaches a value of ~ 8 . The number of shield absorptions, which gives an indication of the potential tritium breeding ratio, increases to ~ 1.4 for the water cooled case, but only reaches ~ 0.9 for the two-phase water cooled case. Hence, the water cooled concept offers the potential for tritium self sufficiency without a blanket modification. However, this may not be achievable because of structural and power conversion limitations in the water cooled blanket at higher energy multiplications.

In comparing these two concepts, it is evident from Figures 4-11 and 4-12 that the water cooled concept has higher short term fissile enrichment and production capabilities relative to the two-phase water cooled case, but also undergoes large increases in the blanket multiplication. The reason for this is the increased neutron moderation with the water coolant, which allows the neutrons to be slowed down and captured in a thinner blanket, thereby producing a higher enrichment. However, the increased moderation also results in a large amount of thermal neutron fissions in the ^{233}U , causing the net fissile production to decrease and the blanket energy multiplication to increase fairly rapidly.

Figure 4-11. Comparison of the Water and Boiling Water Cooled Blanket Enrichment and Fissile Atom Production as a Function of Exposure.

Figure 4-12. Comparison of the Energy Multiplication and Number of Shield Absorptions ($\sim$ Tritium Breeding Ratio) in the Boiling Water and Water Cooled Blankets as a Function of Exposure.

The differences in the neutron flux as a function of energy for the two concepts are shown in Figure 4-13 for two different positions in the blanket. As expected, there are dramatic differences of up to an order of magnitude in the thermal flux and the fast flux for the two coolants. As mentioned previously, the main reason for using two-phase water was to achieve a harder neutron spectrum in the blanket and thereby prevent the ^{233}U thermal neutron fissioning. As Figure 4-13 shows, this has been accomplished with two-phase water as blanket coolant.

Another quantity of interest in comparing the two blanket concepts is the spatial variation of the Th capture and ^{233}U absorption reaction rates in the blanket. These variations are shown in Figure 4-14 for the boiling water cooled blanket and in Figure 4-15 for the water cooled blanket. For the water cooled case the ^{233}U absorption rate is nearly equal to the ^{233}U production rate from the Th (n,γ) reactions after seven years of irradiation, so that the net fissile production is about zero. However, in the boiling water cooled case the ^{233}U absorption reactions have been suppressed so that the net fissile production stays fairly large. In both blankets the average ^{233}U capture-to-ratio (α) is in the range 0.12 to 0.13, which indicates that $\sim 88\%$ of the ^{233}U absorption reactions are fission events. The different neutron spectrum and thermal fission rates also result in much different values in the blanket criticality constant for the two concepts. For the boiling water cooled case, $k_{\text{eff}} = 0.487$ after seven years of irradiation while for the water cooled case, $k_{\text{eff}} = 0.187$ after the same length of exposure.

4.3.3 BLANKET OPERATING CONDITIONS

The blanket power increases steadily with neutron exposure. The overall average blanket power density and the average blanket peak power density as functions of neutron exposure are given in Figure 4-16 for the reference case. The peak to average power density ratio at BOL is 4.78 and drops steadily to a value of 2.6 at 8.4 MW-yr-m^2 of neutron exposure. The power density distributions for these levels of neutron exposure are given in Figure 4-17 for the reference case. It is evident that the power density is flattened appreciably with increasing neutron exposure. It should be pointed out that these results

FLUX AT 5 CM DEPTH IN BLANKET

A - BOILING WATER COOLED BLANKET
B - WATER COOLED BLANKET

FLUX AT 18 CM DEPTH IN BLANKET

Figure 4-13. Neutron Flux as a Function of Energy for the Boiling Water and Water Cooled Blankets.

Figure 4-14. Boiling Water Cooled ThO₂ Blanket Reaction Rates

Figure 4-15. Water Cooled ThO₂ Blanket Reaction Rates.

Figure 4-16. Effect of Neutron Exposure on Blanket Power Densities

Figure 4-17. Effect of Neutron Exposure on Blanket Power Density Distributions

do not include any consideration for fuel shuffling, which could further flatten the power density distributions.

With increasing neutron exposure and blanket power, the clad surface temperature remains relatively constant if the blanket exit steam quality is maintained at 80%. This is due to the fact that the mass flowrate and the heat transfer coefficient increase correspondingly. However, the maximum clad and fuel centerline temperatures increase steadily as shown in Figure 4-18. However, the maximum clad temperature increase slowly with neutron exposure. At the end of the 8.4 MW-yr/m^2 neutron exposure period studied, the maximum clad and fuel centerline temperatures reach 324°C and 1778°C respectively. These temperatures are comparable to those of typical light water reactors. Although the peak clad temperature of 324°C is comparable to that expected in typical light water reactors, the clad exposure to fast neutrons in a fusion blanket may limit the maximum clad temperature allowed to a lower value. In any case, a steady fuel rod exposure of 8.4 MW-yr/m^2 without fuel shuffling is highly unlikely from the viewpoint of optimum fuel management and structural life in a high neutron energy exposure.

CRITICAL HEAT FLUX AND DNB RATIO

The critical heat flux was calculated based on the Jens and Lottes correlation⁽⁴⁻⁸⁾ with the assumption of zero subcooling (at the exit of the blanket) and 33.3 bars of pressure. The mass velocity was based on the nominal value for an exit quality of 0.8. The DNB ratios (ratio of critical heat flux to maximum hot rod heat flux) were then calculated for the hot rod. The results (given in Figure 4-19) show that the DNB ratio at the beginning of life is 5.3, dropping steadily to a value of 1.16 at 8.4 MW-yr/m^2 of neutron exposure. The maximum hot rod heat flux varies from 16 W/cm^2 at BOL to 127 W/cm^2 after 8.4 MW yr/m^2 of neutron exposure.

PRESSURE DROP AND COOLANT PUMPING POWER

The total pressure drop through the blanket consists of the friction losses due to boiling two phase flow, acceleration losses, entrance and exit losses, and expansion and contraction losses. By far the largest contribution to the total pressure drop is from the friction losses. On the basis of a constant

Figure 4-18. Effect of Neutron Exposure on Maximum Fuel and Clad Temperatures.

Figure 4-19. Effect of Neutron Exposure on Hot Rod Heat Flux and the DNB Ratio

steam quality of 80% exiting from the blanket, the pressure drop increases steadily with neutron exposure because of the increased water flowrate required. In order to maintain a constant pressure to the steam turbine, the blanket inlet pressure must be increased with time. This is illustrated in Figure 4-20. The coolant pumping power required as a percent of the blanket power is given in Figure 4-21. It is seen from the latter figure that coolant pumping power is extremely modest. At 8.4 MW-yr/m^2 , the required pumping power is only 1.9% of the blanket thermal power.

The increasing blanket power with time suggests that if the blanket were designed to withstand a fixed maximum pressure, say 33.3 bars (500 psia), then the blanket exit pressure must decrease with time, leading to reduced power conversion efficiencies. This does not present a serious problem because the substantial increases in blanket thermal power more than offset the reduced thermal efficiencies. However, for optimum steam turbine operation, constant steam conditions are desired. In such a case, the blanket should be designed based on end-of-life operating conditions.

Figure 4-20. Effect of Neutron Exposure on Blanket Inlet Pressure.

Figure 4-21. Effect of Neutron Exposure on Coolant Pumping Power Requirements.

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5.0 DTHR ENGINEERING AND MATERIALS TESTING CAPABILITIES

Fusion-driven hybrid reactors based on the leading magnetic and inertial confinement concepts have been studied over the past several years. These studies indicate that hybrids are potentially very attractive fissile material producers. The development of confinement systems that will provide viable options for the production of fusion neutrons has been a major thrust of the fusion program. However, the adaptation of fission reactor technology to the blanket and materials of a hybrid has received insufficient attention in the United States. If the potential of the hybrid is to be realized on a timely basis to insure an adequate supply of energy resources, the blanket and materials feasibility issues need to be identified and a program for resolution of these issues needs to be developed. This study indicates that an initial hybrid demonstration reactor could be developed for operation in the late 1980's. Such a reactor would not only enable acquisition of valuable experience with hybrid technology and serve as a proof-of-principle for the hybrid concept, but would also accelerate the work required for the achievement of pure fusion by serving as an engineering and materials test reactor.

It has long been recognized in the development of new fission reactor concepts, that prototypical experiments form an essential part of the design process. This is due both to the uncertainties that are always present in the basic nuclear cross section data and to the approximations that must be made in any calculational model used to describe reactor behavior. These reasons especially hold true for a fusion hybrid reactor. For example, cross-sections of importance to fissile fuel and tritium production rates have significant uncertainties over much of the energy range of neutrons in the hybrid blanket. In addition, the complex geometries characteristic of tokamak blankets require many geometrical approximations in the application of neutronic codes. It is therefore essential that, as soon as prototypical or near prototypical blanket environments become available, neutronic measurements be conducted. The Rotating Target Neutron Source (RTNS-II) is presently being used at the Lawrence Livermore Laboratories to conduct low neutron fluence ($\sim 10^{17}$ n/cm²) irradiations with a point source on small volume test samples. The first larger volume fusion neutron environment appears to be the TFTR test facility.

5.1 TFTR MATERIALS AND ENGINEERING TEST CAPABILITY

The TFTR, scheduled to begin operations during 1981, will provide the first opportunity for obtaining experimental neutronics data in a bulk medium prototypical of a tokamak hybrid blanket. This reactor could provide basic neutronic measurements. The TFTR parameters relevant to the application of TFTR for blanket neutronics are listed in Table 5-1.

TABLE 5-1

TFTR PARAMETERS

Pulse Length	0.5 seconds
Neutron Yield and Power	3.6×10^{18} n/pulse, 20 MW D-T Power
Full Power Pulses (Total, per year, any day)	4000, 1000, 96

The average first wall neutron loading and flux are 0.14 MW/m^2 and $6.3 \times 10^{12} \text{ n/cm}^2\text{-sec}$, respectively, during the 0.5 second full power D-T pulse. A test module could provide measurements of fissile fuel production, power production, and tritium breeding rates. The module design could be flexible enough to permit a variety of fuel arrangements to be tested in the course of TFTR D-T operations. Test results would therefore be available in a short time.

The experimental techniques employed for the measurements would be low risk, i.e., have a high probability for providing useful data. Since there are large and rapidly changing magnetic and electric fields in the vicinity of the test module, the effect of such fields on the neutronic instrumentation could be tested. Foil activation would be the primary measurement technique. This is a well developed method in thermal reactor parameter studies and assures lack of interference from TFTR electromagnetic fields. As a minimum, the following reactions could be measured:

- (a) ^{238}U fission and capture
- (b) ^{232}Th fission and capture
- (c) ^{238}U and ^{232}Th (n, 2n)
- (d) ^6Li (n, α) T + ^7Li (n, n') T
- (e) ^{233}U fission
- (f) ^{239}Pu fission

In addition, multiple foil activation measurements utilizing foils selected on the basis of threshold nuclear cross sections could be performed to specify the energy flux shape. The foils could be counted by standard high resolution x-ray spectroscopy. For tritium production, β counting of thin foils would be necessary.

Estimates have been made of the count rates to be expected from foils in typical hybrid blanket configurations. The concentrations calculated from reaction rates for a single 0.5 sec full power D-T pulse are given in Table 5-2. The estimated count rates, based upon reasonable assumptions as to foil dimensions and detector efficiencies, are listed in Table 5-3. Also listed are the ratios of the minimum to maximum count rates as a result of flux fall-off through the zone. It can be seen that the count rates are adequate, except for tritium. In this case, at least ten full power pulses will be required for good statistics. This does not appear to be an unreasonable requirement.

The low neutron wall loading in TFTR leads to relatively low blanket power densities and low blanket thermal power production. In addition, it is designed to operate with a pulse length of only 0.5 seconds while the time between pulses is relatively long (300 seconds). The relatively low duty cycle, 1.7×10^{-3} , coupled with the low power density, suggest that little useful thermal-hydraulic data could be obtained from a blanket design. Indeed, the anticipated low blanket power densities suggest that relatively low blanket temperatures and pressures can be maintained by a passive means of blanket cooling (such as conduction or natural convection).

One of the most important blanket design issues is thermal stress and cyclic thermal fatigue. Thermal stresses at the levels anticipated in actual reactors cannot be readily simulated when the neutron wall loading and the blanket power densities are as low as those anticipated in the TFTR. Thermal fatigue could be simulated by proper design of the blanket, although the level of fatigue achievable is limited by the relatively low total number of operating cycles planned for the TFTR (1000 full power cycles per year for four years). It appears that meaningful test data could only be obtained by intentionally overstressing the test blanket, trading off the number of pulses for a higher stress level.

TABLE 5-2
PRODUCT CONCENTRATIONS AND DECAY RATES^(a)

<u>Reaction</u>	<u>Product</u>	<u>Concentration (cm⁻³)</u>	<u>Half Life</u>	<u>Decay Rate (sec⁻¹ cm⁻³)</u>
<u>²³⁸UO₂ System</u>				
²³⁸ U (n,γ)	²³⁹ Np	6.98 x 10 ¹⁰	2.35 d	2.38 x 10 ⁵
²³⁸ U (n,f)	¹⁴⁰ Ba	4.34 x 10 ¹⁰	12.8 d	1.63 x 10 ³ ^(b)
²³⁸ U (n,2n)	²³⁷ U	2.26 x 10 ¹⁰	6.7 d	2.71 x 10 ⁴
T Production	T	8.79 x 10 ¹⁰	12.3 y	1.57 x 10 ²
 <u>ThO₂ System</u>				
²³² Th (n, γ)	²³³ Pa	4.50 x 10 ¹⁰	27 d	1.34 x 10 ⁴
²³² Th (n,f)	¹⁴⁰ Ba	1.03 x 10 ¹⁰	12.8 d	3.88 x 10 ² ^(b)
²³² Th (n,2n)	²³¹ Th	3.48 x 10 ¹⁰	25.5 h	2.63 x 10 ⁵
T Production	T	6.94 x 10 ¹⁰	12.3 y	1.24 x 10 ²

(a) Zone averages, zone order and lengths as follows: ²³⁸UO₂ or ThO₂ (20 cm), Li₂O (10 cm), C (35 cm). Single 0.5 sec full power D-T pulse.

(b) Includes 6% fission yield factor.

TABLE 5-3
ESTIMATED DETECTOR COUNT RATES

<u>Product</u>	<u>Decay Rate (sec⁻¹ cm⁻³)</u>	<u>Detector^(a) Efficiency</u>	<u>Count Rate^(b) (sec⁻¹)</u>	<u>Min/Max Count Rate (sec⁻¹)</u>
<u>²³⁸UO₂ System</u>				
²³⁹ Np	2.38 x 10 ⁵	0.007	41	0.68
¹⁴⁰ Ba	1.63 x 10 ³	0.02	0.82	0.11
²³⁷ U	2.71 x 10 ⁴	0.01	6.8	0.08
T	1.57 x 10 ²	0.90	0.042	0.39
<u>ThO₂ System</u>				
²³³ Pa	1.34 x 10 ⁴	0.02	6.7	0.73
¹⁴⁰ Ba	3.88 x 10 ²	0.02	0.19	0.12
²³¹ Th	2.63 x 10 ⁵	0.005	33	0.10
T	1.24 x 10 ²	0.90	0.033	0.24

(a) Ge (Li) detector except for T counting. Includes γ-ray yield estimate.

(b) Foil area 1 cm². Foil thickness 0.025 cm except for T counting where effective thickness β-range limited to 3 x 10⁻⁴ cm.

On the basis of a maximum "during-burn" wall loading of $\sim 0.2 \text{ MW/m}^2$, or approximately $10^{13} \text{ n/cm}^2\text{-second}$ (14 MeV neutrons), the TFTR neutron current density to the first wall will be $\sim 0.5 \times 10^{13} \text{ (14 MeV) n/cm}^2\text{-pulse}$. For a reactor/device lifetime of about 4000 pulses, the maximum accumulated 14 MeV neutron fluence will be approximately $2 \times 10^{16} \text{ n/cm}^2$. Allowing for spectral tailoring, neutron reflection and/or multiplication by the nearby blanket, etc., the total fluence experienced at a TFTR test station just behind the first wall will be less than 10^{17} n/cm^2 ($E_n > 0.1 \text{ MeV}$).

The fluence value calculated above is only on the order of that which is attainable in the LLL RTNS-II (Rotating Target Neutron Source)⁽⁵⁻¹⁾ in somewhat shorter irradiation times. A significant advantage would be the greater volume which could be irradiated in the TFTR. Recent results⁽⁵⁻¹⁾ from RTNS experiments indicate that mechanical property changes, as measured by increases in the room temperature yield strength following 25° C irradiation, follow a "threshold" type behavior for various metals as follows:

<u>Metal</u>	<u>Threshold Fluence for</u> <u>"Y. S. Hardening" n/cm² (14 MeV)</u>
Cu	2×10^{16}
Nb	5×10^{16}
V	5×10^{16}
Nb - 1 Zr	$\sim 10^{17}$

At these fluence levels, damage studies are limited to microstructural characterizations of the very early, and barely resolvable, features/artifacts of the damage process.

Discernible changes in other properties such as dimensional stability (i.e., void formation and swelling), elevated temperature embrittlement (due to helium production and accumulation), or metallurgical stability (e.g., changes in precipitate morphology or structure), require neutron fluences greater than about 10^{21} n/cm^2 . These effects are clearly not amenable to study in a TFTR-level radiation environment.

5.2 DTHR ENGINEERING TEST CAPABILITIES

The TFTR which is to be operated at PPPL will generate a reactor grade plasma operating on D-T fuel. However, the planned D-T operations are much too limited for utilization as an engineering test reactor. The TFTR has been designed primarily as an experimental facility for studying plasmas, to demonstrate $Q = 1$ while operating with D-T, and to address tritium handling and remote maintenance in the context of a tokamak. For the most part, however, TFTR will operate with hydrogen and deuterium fuel.

This is in contrast to the DTHR preliminary design which is based on long pulse (~ 70 s), high duty factor (50 - 82%), and high plant availability (40 - 70%). The fusion power produced by DTHR (~ 1 GW) would be expected to be ~ 50 times larger than that produced in TFTR and the neutron wall loading for DTHR would be an order of magnitude larger than that for TFTR. Therefore, operating experience with a tokamak blanket could be obtained with DTHR for reactor conditions of high power densities associated with high radiation levels. The pulses would be sufficiently long that essentially steady state conditions would be achieved and the product of duty cycle factor and plant availability would be sufficiently large (0.2 - 0.6) that a significant amount of fuel (either fissile or fusile) could be generated in the blanket annually. Power production in a hybrid blanket would also be significant. The fluences and integrated dose would be sufficiently large that radiation damage effects could be studied within blanket test modules. Design features on DTHR that are not available on TFTR that would allow DTHR to serve as an Engineering Test Reactor are (1) adequate radiation shielding for device and facility, (2) adequate cooling, (3) superconducting coils, (4) impurity control, and (5) plasma refueling.

The key DTHR plasma operation engineering test capabilities are as follows:

- Test ignition operation in which the principal means of plasma heating is via plasma-produced alpha-particles.
- Test plasma operation at moderate and high β 's up to $\sim 6\%$ ($\beta \sim 1\%$ for TFTR).

- Study consequences of and improved performance for elongated plasmas ($\delta \leq 1.6$). In TFTR, the plasma is circular.
- Study plasma fueling and profile effects using beam and pellet injection. TFTR injectors are arranged for tangential injection, but for reactor applications radial injection is more feasible.
- Study $\bar{n}_e \tau_E$ scaling in the reactor regime (TFTR should reach $\bar{n}_e \tau_E \sim 1 \times 10^{13} \text{ cm}^3 \text{ s}^{-1}$ which is marginal for reactor operation).
- Study the potential for increasing the plasma temperature beyond $\bar{T}_e \sim 10 \text{ keV}$ (TFTR may operate as high as $\bar{T}_e \sim 5 - 7 \text{ keV}$).
- Study effects due to high plasma currents ($I_p \geq 3 \text{ MA}$) on alpha confinement.
- Study long pulse operations in which fueling and divertor operations are important (TFTR may operate for up to a 4 s pulse).
- Various beam-driven modes of plasma operation can be tested if ignition is not achieved.

Numerous engineering tests can be conducted in the DTHR to establish baseline direction for designers of the first walls, the plasma exhaust systems, and the tritium and fissile breeding blankets. A summary of typical engineering test programs is provided in Table 5-4, which lists the basic test item, controlled variables, in-situ and post-exposure measurements. Many of these tests relate to the determination of a sound engineering data base for all the fusion hybrid reactor materials. The DTHR materials test capabilities are discussed in the following section.

5.3 DTHR MATERIALS TESTING CAPABILITIES

In addition to the demonstration of significant ^{233}U breeding in the boiling water-cooled DTHR blanket modules, it will also be possible to study the effects of radiation damage on structural materials in a prototypical fusion neutron environment. A rather large volume of the blanket region, probably over 20 m^3 , could be dedicated to such materials and engineering test capabilities. On the basis of the nominal neutron wall loading of 2 MW/m^2 and a total projected load factor of 0.20, an effective integrated wall loading of

TABLE 5-4

DTHR ENGINEERING TEST PROGRAMS

<u>ITEM</u>	<u>VARIABLES</u>	<u>IN SITU MEASUREMENT</u>	<u>POST-EXPOSURE MEASUREMENTS</u>
FIRST-WALL TEST STATIONS	ALLOY, GEOMETRY, COOLANT TYPE, PRESSURE, OPERATING TEMPERATURE, WALL LOADING, FLUENCE	FATIGUE, CREEP, BIAXIAL STRESS, CRACK PROPAGATION, HEAT TRANSFER PARAMETERS	FRACTURE TOUGHNESS, TENSILE, SWELLING, RESIDUAL STRESS, CORROSION, HE & DPA PRODUCTION
DIVERTOR COLLECTOR	GETTER, SUBSTRATE, GEOMETRY, OPERATING TEMPERATURE, REGENERATION CYCLE	HEAT TRANSFER PARAMETERS, COLLECTOR EFFICIENCY, REGENERATION EFFICIENCY	MASS LOSS, CRACKING, POWDERING, IMPURITY CONCENTRATION, HE RETENTION
HYBRID BLANKET MATERIALS MODULES (FUEL, CLAD, STRUCTURE AND COOLANT SAMPLES)	ALLOY, GEOMETRY, COOLANT TYPE, PRESSURE, OPERATING TEMPERATURE, WALL LOADING, FLUENCE	FATIGUE, CREEP, BIAXIAL STRESS, CRACK PROPAGATION, (STRUCTURAL MATERIALS)	FRACTURE TOUGHNESS, TENSILE, SWELLING, RESIDUAL STRESS, CORROSION, HE & DPA PRODUCTION, COMPATIBILITY FUEL RETENTION
TRITIUM BREEDING MATERIALS MODULES (SOLID LI MATERIALS, CLAD, COOLANT)	ALLOY, GEOMETRY, COOLANT TYPE, PRESSURE, OPERATING TEMPERATURE, WALL LOADING, FLUENCE	FATIGUE, CREEP, BIAXIAL STRESS, CRACK PROPAGATION, (STRUCTURAL MATERIALS), TRITIUM MIGRATION	FRACTURE TOUGHNESS, TENSILE, SWELLING, RESIDUAL STRESS, CORROSION, COMPATIBILITY, TRITIUM PRODUCTION AND DISTRI- BUTION, TRITIUM RETENTION
LIQUID LITHIUM FLOW MODULE AND LOOP	TEST SECTION MATERIALS AND GEOMETRY, TEMPERATURE, PRESSURE, LOCAL MAGNETIC FIELD STRENGTH	LITHIUM PURITY, HEAT TRANSFER PARAMETERS, TRITIUM MIGRATION	CORROSION, COMPATIBILITY, TRITIUM PRODUCTION AND DISTRIBUTION
HYBRID BLANKET PRODUCTION MODULES	FUEL FORMS, COOLANT, GEOMETRY, OPERATING PRESSURE, OPERATING TEMPERATURE, WALL LOADING, FLUENCE	COOLANT CONTAMINATION, HEAT TRANSFER PARAMETERS, RADIATION TRANSPORT	FUEL RETENTION, MICROSTRUCTURE VOIDS, TRANSMUTATION AND FISSION PRODUCT CONCENTRATION, CORROSION, INTEGRAL NEUTRONICS

0.4 MW-yr/m² could be attained during a typical calendar year of operation. While this value is somewhat less than optimum for performing real-time radiation effects experimentation, it is clearly superior to that available with either current sources or other facilities which might be contemporary with the DTHR. Moreover, once an initial "shakedown" phase is completed with the DTHR, the load factor could be increased substantially, to at least 0.5 or 0.6, thereby increasing the annual accumulated value to well over 1 MW-yr/m².

Tables 5-2 and 5-3 may be easily extrapolated to a DTHR driver instead of TFTR. The relevant quantities to perform this are the first wall currents, which are 0.9×10^{18} and 3.13×10^{16} n/m²-sec for DTHR and TFTR, respectively. Therefore reaction rates are approximately 30 times higher in DTHR. Since the burn time planned for DTHR is 40 s compared to 0.5 s for TFTR, there is an additional factor of 80 in induced activities per burn cycle. All foil count rates are clearly adequate and, in fact, measurements in a second lithium zone beyond the graphite reflector could be added. The advantage of such a zone in an actual blanket is an increase in the tritium production rate of nearly 30% for both the UO₂ and the ThO₂ systems. In addition, sufficient fission heat may be generated in a UO₂ system for measurements to be conducted at elevated temperatures.

Typical values of radiation damage which would occur in a type 316 stainless steel blanket front wall are tabulated in Table 5-5 for test module exposure times of one, two, and five years. This data was derived from a neutronic analysis depicted graphically in Figure 5-1. Even at the conservative value of 0.4 MW-yr/m² integrated wall loading, the damage accumulated during a two year exposure - 10 DPA, 120 appm He — corresponds to approximately a one year damage level anticipated for a commercial fusion reactor. Importantly, the damage will be produced at the appropriate appm He/DPA ratio.

The considerable volume which could be dedicated to materials test positions in the DTHR, as shown in Figure 5-2, offers the opportunity for conducting extensive in-reactor materials property testing. Current emphasis in the Office of Fusion Energy's Alloy Development for Irradiation Performance

TABLE 5-5

TYPICAL DESCRIPTION OF MATERIALS IRRADIATION TESTING CAPABILITIES IN THE DTHR

Fusion Driver Characteristics

Total Fusion Power, MWt	900
Neutron Wall Loading, MW m ⁻²	2
Current Density of Source Neutrons, nm ⁻² s ⁻¹	0.9 x 10 ¹⁸
Plasma Duty Cycle x Plant Availability Factor	0.2
Neutron Flux - First Wall, (a) nm ⁻² s ⁻¹ , (E _n > 0.1 MeV)	4.4 x 10 ¹⁸

Approximate Radiation Damage Values for a Type 316 SS Blanket Front Wall

Operating or Exposure Time, years	1	2	5
Integrated Wall Loading, MW-yr m ⁻²	0.4	0.8	2.0
Equivalent Neutron Fluence, nm ⁻² , (E _n > 0.1 MeV)	2.8 x 10 ²⁵	5.5 x 10 ²⁵	1.4 x 10 ²⁶
Displacement Damage, DPA's	5	10	25
Helium Production, appm He	60	120	300

-
- (a) During operation, this value would gradually increase due to hybrid blanket neutronic activity. This change was ignored for the computations presented for the 316 SS wall.

Figure 5-1. Radiation Damage Parameters as a Function of Exposure for the Boiling Water Cooled Blanket.

Figure 5-2. Materials and Blanket Test Stations in DTHR

Program is on the development of an extensive mechanical property data base for irradiated material ⁽⁵⁻²⁾. It is felt that changes in the mechanical properties, particularly embrittlement due to combined displacement damage and helium effects, may represent the limiting condition for most candidate materials. Particular emphasis is being placed on the effects of irradiation on the mechanical property response under cyclic loading, e.g., fatigue and crack growth properties.

Test modules could be designed to permit extensive mechanical property testing; both in-situ and ex-reactor tests could be accommodated. In-situ tests would focus on those properties which are felt to be most sensitive to the presence of an active energetic neutron environment. For example, properties such as irradiation creep response, thermal creep-rupture, crack propagation behavior, etc., would be performed in-situ as well as in a hot cell facility following irradiation and removal from the DTHR test modules. Current indications are that such properties or response characteristics as tensile behavior, Li corrosion/compatibility as-irradiated, microstructural stability, fracture toughness, etc., can (or, in some cases, must) be determined satisfactorily by post-irradiation examination and/or testing.

For several of the in-reactor tests, procedures or methods and specimens have already been developed; these could be adopted or modified to suit the DTHR module requirements. For example, tensile-loaded creep tests could incorporate the "Belleville Washer" design suggested by the TETR designers ⁽⁵⁻³⁾, and biaxial creep tests are now routinely performed using internally pressurized tubes. Although more difficulty or innovation might be required to implement reliable testing hardware for in-reactor fatigue and crack growth tests, these should be possible.

REFERENCES - SECTION 5.0

1. Mitchell, J. B., "Exploratory Experiments Comparing Damage Effects of High-Energy Neutrons and Fission-Reactor Neutrons in Metals," UCRL-52388, Lawrence Livermore Laboratory, January 12, 1978.
2. The Fusion Reactor Materials Program Plan, Section I, Alloy Development for Irradiation Performance, U. S. Department of Energy, DOE/ET-0032/1, July, 1978.
3. Badger, B., et al., "TETR - A Tokamak Engineering Test Reactor," UWFD-191, University of Wisconsin, December, 1977, (see Section X).

6.0 DTHR ECONOMIC CONSIDERATIONS

The design of DTHR was developed to provide a reactor demonstration plant technically prototypical of projected commercial conditions, yet economically non-prototypical. The intent was to reduce capital cost expenditures on systems that could be developed independently or at a later date as part of an upgrade program. This section provides a scoping estimate of the capital cost of the DTHR and a comparison of the operating cost for the various options had they been included in the DTHR concept. These comparisons provide insight about which systems should be reconsidered for inclusion in the DTHR.

6.1 ASSUMPTIONS

The DTHR is assumed to be a government owned plant constructed on a government site such as Hanford. The plant must be located close to a power source for start-up and operating electrical power needs ($\sim 1700 \text{ MW}_e$ maximum). No time related costs (interest or escalation during construction) were included. The cost estimates are based on current year dollars (1978) and engineering judgment using the preliminary subsystem conceptual designs. The base costs of equipment were determined with the COAST-4 computer code. The spare parts allowance was taken conservatively as 2% of the total equipment cost. The construction costs were taken at 15% of the total equipment cost. The installation and assembly costs were also assumed to be 15% of the equipment cost. The contingency allowance was assumed to be 20% of the total cost after Engineering Design, Inspection and Administration (EDIA, taken at 40%) was included. The standard account numbers were selected in accordance with the DoE guidelines⁽⁶⁻¹⁾ for fusion reactor studies for the reactor equipment cost breakdown accounts.

The DTHR operating costs were established using plant operating and maintenance costs based on the Westinghouse TCT hybrid Reactor Concept developed for EPRI⁽⁶⁻²⁾. These costs (1977 \$) were increased by 8% to reach 1978 \$ and then the annual cost of tritium equipment operation and handling ($\sim \$ 0.8 \text{ M}$) was added. The major maintenance items requiring replacement before end of plant life are

included in O & M for items such as the vacuum vessel wall, limiters and neutral beam injector sources. The cost of electricity required to operate the plant was taken at 28 mills/kW hr based on mid-1980's costs to a government facility⁽⁶⁻³⁾. The fusile fuels (tritium and deuterium) were assumed to be supplied from government facilities. The annual quantities and costs were conservatively estimated to be 10 kgT at 75¢/Ci and 30 kgD at \$3340/kg. For the fertile fuel, thorium was assumed to cost \$30/kg. The blanket material, 316 stainless steel plate at \$4/lb., was selected for reactor grade (RDT specifications) application. A full complement (27) of blanket modules was assumed and none would be used more than three years before removal for destructive examination of material performance. The costs of the fertile fuel and blanket modules were therefore averaged over the three year period as an operating cost for each concept. A total of ten blanket or materials test modules concepts were assumed over the DTHR plant life.

6.2 PERFORMANCE AND COST ESTIMATES

The COAST-4 computer code was used to provide scoping performance and cost estimates for DTHR. The initial operation was assumed to be a proof-of-breeding test during the first three years, coupled with numerous scientific and engineering tests to verify subsystem performance, safety and design margins, and develop operating procedures. Subsequent testing could then be devoted to specific engineering and materials tests to provide a broader data base for guidance in commercial designs. The economic analyses in this section assumed the performance of the reference DTHR concept which uses ThO_2 fertile fuel. ThO_2 fertile fuel has an inherent low thermal power production which minimized the DTHR cooling tower requirements and related costs. If a decision were made to convert thermal power to electricity at the DTHR plant, then reconsideration of the fertile fuel should be made since the performance of UO_2 (for example) could result in > 2 times the thermal power for the blanket concept studied in this report.

6.2.1 CAPITAL COST ESTIMATES

Table 6-1 provides the estimated capital cost breakdown for the major DTHR components. The reference concept described in this document was used to

establish these costs. The level of design detail available and lack of experience with many of the components in the reactor was reflected by the contingency cost (20%) used in the estimation of the Plant and Capital Equipment (PACE) costs shown in Table 6-2. The PACE costs conservatively assumed an independent plant. Considerable savings could be achieved by location at a facility which has many of the support systems available (cooling towers, power substation, administrative offices, maintenance, etc.). The DTHR reactor size is felt to be typical of what a commercial 1000 to 1200 MWe plant would be. The DTHR capital cost is considerably less than the Portsmouth Gaseous Diffusion Plant Expansion costs ⁽⁶⁻³⁾ estimated at 3 to 4 B \$, but somewhat more than the projected costs of a 1000 MWe nuclear power plant in the mid-1980's. An initial study of plasma parameters versus building and equipment cost trade-offs for DTHR has been made.⁽⁶⁻⁴⁾ Figure 6-1 shows a typical effect of the major radius, R_0 , on reactor equipment capital costs. For example, reducing the major radius until the assumed limit of 12 tesla for Nb_3Sn superconducting TF coils is reached would result in $\sim 4\%$ reduction in capital cost and $\sim 11\%$ less fissile production (but still well above the 100 kg/yr goal). A change to this smaller size, however, would require further design iterations of the bundle divertor concept, which is affected by the TF coil field and space envelope, and dictated the DTHR major radius. The engineering design effort for this first-of-a-generic-kind plant design is reflected in the EDIA cost, which was calculated using 40% of the plant cost. The construction, installation and assembly cost was estimated as 30% of the plant equipment and facilities costs. A cost estimate breakdown of the developmental work required for the DTHR is included in Table 6-2. These are pre-operational RD&D tasks specifically oriented to develop and qualify the design, fabrication and performance prior to release of the actual DTHR components.

6.2.2 OPERATING COST ESTIMATE - BEAM DRIVEN MODE

The performance of the DTHR was estimated by COAST-4 and variations in operating conditions studied to determine effects on annual operating costs. The initial operating costs are shown in the first column of Table 6-3. As shown under the performance characteristics, the beam driven operating mode

TABLE 6-1
CAPITAL COST BREAKDOWN FOR DTHR REACTOR EQUIPMENT

<u>Account Number</u>	<u>Account Title</u>	<u>Cost, M \$ (1978)</u>
22	<u>Reactor Plant Equipment</u>	
22.01	Reactor Equipment	
22.01.01	Blanket and First Wall	6.5
.02	Vacuum Vessel	4.5
.04	Limiters	2.0
22.01.02	Shield	68.1
.01	TF Coil Shields (Inner and Outer)	63.0
.02	NBIS Duct Shields	5.1
22.01.03	Magnets	216.6
.01	TF Coils	136.6
.02	SF Coils	32.2
.03	OH Coils	17.6
.04	Winding Equipment	25.9
.05	Bundle Divertor Coils	4.3
22.01.04	Supplemental Heating	153.1
.01	Neutral Beam Injection System	153.1
22.01.05	Primary Structure and Support	20.0
.01	Reactor Structure	14.1
.02	Equipment Support (Bucking Cylinder)	5.9
22.01.06	Reactor Vacuum Systems	8.5
.05	Reactor Vacuum Systems	5.2
.06	Reactor Vacuum Wall	3.3
22.01.07	Power Supply Switching & Energy Storage	85.9
.01	OH Coil Power Conversion Equipment	17.1
.02	SF Coil Power Conversion Equipment	48.5
.03	TF Coil Power Conversion Equipment	1.5
.04	Load Smoothing Circuit	7.1
.05	Pri./Sec Distribution	11.7
22.01.08	Impurity Control	6.9
.01	Bundle Divertor Chamber	6.9

TABLE 6-1 (Cont'd.)

CAPITAL COST BREAKDOWN FOR DTHR REACTOR EQUIPMENT (Cont'd.)

<u>Account Number</u>	<u>Account Title</u>	<u>Cost, M \$ (1978)</u>	
22.02	Main Heat Transfer & Transport System		12.8
.01	Primary Coolant System	7.2	
.05	Clean-up System	5.6	
22.03	Auxiliary Cooling System		61.7
.01	Magnet Cooling	55.0	
.02	Shield and Structure Cooling	1.4	
.03	NBIS Cooling	4.0	
.05	Vacuum Vessel and Divertor	1.3	
22.04	Radioactive Waste Treatment and Disposal		4.2
.01	Liquid Wastes Processing and Equipment	2.7	
.02	Gaseous Wastes and Off Gas Processing Sys.	1.5	
.03	Solid Wastes Processing Equipment	(with .01)	
22.05	Fuel Handling and Storage Systems		24.7
.01	Purification	5.4	
.03	Preparation		
.04	Injection	8.3	
.05	Storage	5.6	
.06	T Recovery (Plasma Exhaust)	5.4	
22.06	Other Reactor Plant Equipment		22.8
.01	Maintenance	22.8	
22.07	Instrumentation and Control		12.2
.01/.02	Reactor I & C/Rad. Monitoring	12.2	
Total Reactor Plant Equipment			702

TABLE 6-2

DTHR PLANT AND CAPITAL EQUIPMENT AND RD&D COST ESTIMATES

<u>DTHR Plant and Capital Equipment (PACE) Costs</u>	<u>Estimated Cost (1978 M \$)</u>
Structures and Site Facilities	83
Reactor Plant Equipment	702
Turbine/Electric Plant Equipment	9
Miscellaneous Plant Equipment	11
Spare Materials and Spare Parts	22
Construction, Installation, and Assembly	248
Sub-Total	1,075
Engineering, Design, Inspection and Administration (EDIA)	430
Contingencies	301
Total (PACE) Cost Estimate	1,806
<u>DTHR Pre-Operational Development (RD&D) Costs</u>	
Vacuum Vessel Construction and Servicing	9
Impurity Control System Components	29
TF Coil Fabrication Qualification and Cryogenic Cooling Tests	15
PF Coil Prototype Coil and Power Conversion Equipment	30
NBI System Direct Energy Conversion	18
Tritium Fueling, Handling, Reprocessing Equipment Components	12
Special I&C Equipment and Systems	13
Blanket Module Mockup Testing	9
Remote Servicing and Fuel Handling Equipment	12
Safety and Environmental Research	12
Materials Technology	6
Total Developmental Cost Estimate	165

Figure 6-1. Tokamak Major Radius Versus Cost and Performance Trade-Offs.

TABLE 6-3

DTHR ECONOMICS - BEAM DRIVEN OPERATION

	<u>Reference Operation</u>	<u>With Pwr. Conv.</u>	<u>With Power Conv. & DEC</u>
<u>Performance Characteristics</u>			
Peak Fusion Power, MW_t	900	900	900
Total High Grade Heat, MW_t (a)	240	240	240
Electricity Production, MW_e (a)	0	70	70
Electricity Consumption, MW_e (a)	1140	1140	845
Net Electricity Consumption, kW-hr/yr ^(b)	4.04×10^9	3.79×10^9	2.77×10^9
Fissile Production, kg ^{233}U /yr ^(b)	165	165	165
<u>Capital Costs (M \$)</u>			
Structures and Site Facilities	83	83	83
Reactor Plant Equipment			
Tokamak & Support Systems	548	548	543
NBIS	154	154	213
Turbine/Electric Plant/Sub-Station	9	26	25
Balance of Plant/Materials/Spare Parts	33	33	34
Construction, Installation & Assembly	248	253	269
Sub-Total	1075	1097	1167
EDIA Allowance	430	439	467
Contingency Allowance	301	307	327
Total Capital (PACE) Cost Estimate	1806	1843	1961
<u>Annual Operating Costs (M \$)</u>			
O & M and Replacement Components	40	40	40
Power	113	106	78
Fusile Fuel	72	72	72
Test Blankets	44	44	44
Total Annual Operating Cost	269	262	234

(a) Averaged values using a plasma duty cycle factor of 0.5.

(b) Averaged values using (a) and a plant availability factor of 0.4.

easily meets the fissile production goal, but would require 1200 to 1700 MW of externally supplied electricity. Although this is significantly less than a Portsmouth Gaseous Diffusion Plant⁽⁶⁻³⁾ power requirements (2700 MW) it does provide incentive to investigate conversion of thermal heat to electricity and incorporating direct energy conversion (DEC) into the neutral beam injection system. These two alternatives were evaluated and the results tabulated in columns 2 and 3 of Table 6-3. The electrical power demands are reduced by $\sim 70 \text{ MW}_e^*$ by the addition of power conversion equipment and power consumption is reduced by $\sim 300 \text{ MW}_e$ by the addition of the DEC system. The results of these changes can be evaluated in terms of capital cost increases and operating cost reductions.

The reference DTHR plant annual operating cost is estimated to be $\sim \$270\text{M}$. This is about one-third of the comparable operating cost of a Portsmouth Gaseous Diffusion Plant⁽⁶⁻³⁾ (O & M $\$22 \text{ M}$ + power $\$658 \text{ M}$ = $\$680 \text{ M}$). However, it is probably two or three times that of a small experimental test reactor's operating cost. The addition of power conversion equipment increases the capital cost about 2% and reduces the purchased electrical power cost by 6%. The net operating cost decreases approximately 3%. Thus, a decision to add power conversion equipment to DTHR would probably be based on technology development rather than economics (for the beam-driven mode of operation) since it would take over five years of operation to justify the increased capital cost.

The addition of direct energy conversion equipment would reduce the operating costs by $\sim \$35 \text{ M}$ per year and thereby could justify the $\$155 \text{ M}$ increase in capital cost if the DTHR were to be operated in the beam driven mode for a period of at least five years. However, a significant advantage of the tokamak concept is that it need not be beam driven as would be required of the mirror concept. Thus, the operation of DTHR in the ignited plasma operating mode was considered.

6.2.3 OPERATING COST ESTIMATE - IGNITED MODE

The DTHR plasma concept was developed with the option of ignited operation. Since ignited operation has not been scientifically verified, it can only be

*DTHR utilizes only 24% of the blanket coverage on the torus.

stated that the required confinement time is approximately equal to the empirical confinement time. If the goals of DTHR were modified to make proof of ignition a higher priority than proof of fissile breeding, then it would make sense to procure only the neutral beam injectors required for start-up (i.e., the NBIS required for beam-driven operation would be purchased only if ignition could not be achieved). The operating cost developed in Table 6-4 represents the "ignited priority" scenario. The first column of Table 6-4 can be compared directly with the first column performance and cost data for the beam driven reference operation on Table 6-3. The net power consumption is reduced over a factor of ten, the capital cost is reduced by ~ \$264 M, and the operating cost is reduced by ~ \$100 M. The capital costs were reduced directly by the neutral beam injector savings of \$85 M, power control equipment savings of ~ \$18 M, cooling and shielding equipment savings of ~ \$10 M, reduced size in the plasma fuel handling and exhausting equipment (~ \$4 M), and other equipment savings (~ \$2 M). The related construction, installation, EDIA and contingency costs are reduced by ~ \$145 M.

The second and third columns in Table 6-4 indicate the effect of the power conversion and direct energy conversion equipment on the capital and operating costs. In the ignited mode of operation, the power conversion equipment increases total capital costs \$47 M, but reduces annual operating costs by \$7 M. Thus, this equipment could "pay" for itself within the first seven years of operation as was the case with beam driven operation. However, the direct energy conversion equipment would not be justified since it increases capital cost by \$101 M, but reduces the operating costs by only \$8 M per year.

It can be seen from Table 6-4 that the fusile fuel charge has now become the largest operating cost. This brings to focus the need for consideration of breeding tritium within the hybrid reactor.

6.2.4 FUSILE FUEL OPERATING COST CONSIDERATIONS

The operating costs for fusile fuel used in the preceding analyses was based on procuring all tritium and deuterium. These annual costs (\$72 M) are dominated by the tritium cost (10 kgT at 75¢/Ci)*. The capital costs related

*An abundant source of tritium would not be available during initial DTHR operation. Therefore, higher costs should be assumed. This value could drop an order of magnitude later.

TABLE 6-4

DTHR ECONOMICS - IGNITED OPERATION

	<u>Ref. With Ignited Operation</u>	<u>Ignited With Pwr. Conv.</u>	<u>Ignited With Pwr. Conv. & DEC</u>
<u>Performance Characteristics</u>			
Peak Fusion Power, MW _t	900	900	900
Total High Grade Heat, MW _t (a)	240	240	240
Electricity Production, MW _e (a)	0	70	70
Electricity Consumption, MW _e (a)	70	70	66
Net Electricity Consumption, kW hr/yr (b)	3.1×10^8	6×10^7	4.5×10^7
Fissile Production, kg ²³³ U/yr (b)	165	165	165
<u>Capital Cost (M \$)</u>			
Structures and Site Facilities	83	83	83
Reactor Plant Equipment			
Tokamak and Support System	521	521	521
NBIS	69	69	97
Turbine/Electric/Sub-Station	2	19	19
Balance of Plant/Materials/Spare Parts	31	31	32
Construct., Install. & Assembly	212	217	226
Sub Total	918	940	978
EDIA Allowance	367	376	391
Contingency Allowance	257	263	274
Total Capital (PACE) Cost Estimate	1542	1589	1643
<u>Annual Operating Cost (M \$)</u>			
O & M and Replacement Components	38	38	38
Power	9	2	1
Fusile Fuel	72	72	72
Test Blankets	44	44	44
Total Annual Operating Cost	163	156	155

(a) Averaged values using a plasma duty cycle factor of 0.5.

(b) Averaged values using (a) and a plant availability factor of 0.4.

to tritium fueling and handling are shown in account 22.02.01.05 (primary coolant system clean-up, \$5.6 M), account 22.04 (waste treatment and disposal, part of the \$4.2 M) and account 22.05 (fuel handling and storage, \$24.7 M) on Table 6-2. These costs represent a once-through tritium system where new fuel is supplied continuously and only safety and environmental control equipment are included. The cost of providing a full tritium reprocessing system is estimated at an additional \$20 M. The maximum breeding potential for the reference DTHR blanket has been estimated to be ~ 0.7 . If the entire inner torus surface area (inner shield in Figure 3-2) and the areas of the outer torus other than the test stations in Figure 5-2 were fitted with tritium breeding blankets, it is estimated that a tritium breeding ratio of ~ 0.5 could be achieved. This would reduce fusile fuel operating cost by $\sim \$36$ M, while increasing the capital cost by $\sim \$60$ M (allowing \$30 M for the tritium reprocessing system and \$30 M for the tritium blanket system). An effective net reduction in annual operating costs would be achieved in the first few years of operation. This is a very cursory estimate and should be evaluated in detail in future studies.

6.3 CONCLUSIONS

The DTHR, designed and constructed with many components prototypical of a commercial application, is estimated to have a reactor equipment cost of \$702M. An independent DTHR plant was estimated to have a PACE cost of $\sim \$1.8$ B (includes EDIA, spare parts and general contingencies). The highest generic component cost is the neutral beam injection system, followed by the TF coil magnet system. The capital costs could be reduced by $\sim \$260$ M if initial operation of the plasma was based on the ignited mode rather than the more conservative beam driven mode presently assumed.

The DTHR annual operating cost is estimated to be $\sim \$270$ M for the beam driven mode, with the cost of purchased electricity representing the largest cost. Including a power conversion system to convert the thermal power to electricity would not significantly reduce operating costs because DTHR uses only 24% blanket coverage and ThO_2 fuel. However, the addition of a direct energy conversion system would reduce the neutral beam injectors power demands and therefore, reduce annual operating costs over \$30 M. A more effective way to reduce the operating cost would be to prove ignited operation feasibility in the initial

testing. Running DTHR in the ignited mode reduces the annual operating cost by over \$100 M. This is due to a reduction in the cost of electricity procurement of \$104 M. Addition of power conversion equipment for the ignited mode would result in an electrical break-even condition (power produced equals power required to operate the plant). The cost of fusile fuel (tritium) now becomes the significant operating cost suggesting that tritium breeding should be considered. A cursory analysis indicates an additional \$30 M/yr could be saved if the DTHR bred half the tritium required for operation.

Thus, it becomes apparent that operation in the ignited plasma mode will reduce capital costs by $\sim 15\%$ and operating costs by $\sim 40\%$. The addition of power conversion or neutral beam direct energy conversion equipment could be an economic advantage after several years of operation in the beam driven mode. The direct energy conversion equipment would not be economically sound for the ignited mode of operation. Therefore, engineering judgment should include economic as well as technical inputs before selecting the initial demonstration systems. Reductions in operating power and tritium procurement requirements significantly reduce operating costs and should be given extensive evaluation before final equipment selections are made.

SECTION 6.0 REFERENCES

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7.0 DTHR EXTRAPOLATION TO COMMERCIAL HYBRID REACTOR APPLICATIONS

The feasibility study of the DTHR was conducted with the underlying objective that the concept and performance should allow extrapolation to a Commercial Tokamak Hybrid Reactor (CTHR). This section presents the results of the extrapolation of DTHR to a commercial application, estimates the performance and costs for an improved commercial reactor concept, and provides a cursory estimate of the economic performance of such a CTHR in terms of cost per product (\$/kg HM for fissile fuel and \$/kW-hr for client PWR produced electricity).

7.1 CTHR PERFORMANCE AND COST ESTIMATES

The DTHR concept presented in this report can be extrapolated to a commercial operation in numerous ways. The most straight-forward approach seems to be merely replacing the partial test blanket (24% coverage) with a full outer torus blanket (57% coverage), changing the one-shift testing type operating mode to three-shift operation (plant availability increases from 0.4 to 0.7), operate the plasma in a longer pulse, ignited mode (duty cycle increases from 0.5 to 0.85 and hybrid operating power demand reduced by a factor of >2), and including the power conversion equipment which enables provision for hybrid operating electricity and transmission of excess electricity produced. None of these changes require scientific or engineering advances that could not be developed and proven with the DTHR facility.

Choosing an ignited versus a beam driven mode of operation for the commercial reactor was based on two considerations. First, the commercial concept is further in the future, thus allowing time for scientific verification of ignited operation. The second point is that the cost per unit of product appears to be significantly less for the ignited mode of operation, due mainly to the reduced electric power needed to operate the hybrid. Figure 7-1 shows typical cost trends based on a trade-study made using the COAST code and the neutronic and

Figure 7-1. Relative CTHR Fuel Production Costs for Beam Driven and Ignited Operation Vs. Reactor Size.

economic analysis methods developed in the study of hybrid enrichment of spent PWR fuel on a direct fuel cycle^(7-1, 7-2). The curves show that the reduced costs for ignited operation are fairly uniform as the tokamak major radius is increased (to further reduce the product costs). Although these curves were developed for a direct fuel cycle, the trends should be the same for fuel cycles using mechanical or chemical reprocessing.

The reference fuel used in the DTHR study was ThO₂. This fuel choice was made with particular consideration toward non-proliferation. Since limited information is available on fuel cycle costs and typical LWR scenarios for enrichment requirements, burnup values, etc., an alternate extrapolation of DTHR to a commercial application is presented using UO₂ (0.25 w/o tails) as the fuel. Table 7-1 shows several key DTHR performance parameters as well as the parameters of the extrapolated performances for the ThO₂ and UO₂ fuel concepts. As can be seen by comparing the latter performances (each are first iteration concepts and far from optimized), the UO₂ concept would provide ~ 3 times more net electric power (for sale) and almost double the fissile fuel production (of Pu versus ²³³U). Both concepts required unnecessarily long (~ 10 years) irradiation periods. Therefore, an additional concept is shown under the heading of CTHR to project a better performing commercial concept which requires only about 4 years to achieve the 3 w/o enrichment required for a typical LWR fueling.

The CTHR concept in Table 7-1 assumes that helium cooled UC fuel is used in a full blanket (inner and outer torus coverage) and a more typical (20 minutes) plasma burn time is applied. This resulted in significant increases in both the electricity and fissile fuel production. The four year irradiation period would also permit a more realistic refueling program, with one-fourth of the reactor being refueled each year once equilibrium was established. The tritium breeding ratio would be adequate to provide a self-sufficient fissile fuel supply. The capital cost for this concept was estimated using the COAST code and summarized on Table 7-2. The spare parts and contingencies were taken as 0.5% and 10% respectively for accounts 21, 23-26. The reactor equipment spare parts and contingencies were assumed to be 2% and 20% respectively. Accounts 91-93 were estimated in accordance with the Fusion Reactor Design Studies-

TABLE 7-1
DTHR EXTRAPOLATED PERFORMANCE TO CTHR

	DTHR	Extrapolated DTHR; $R_0 = 5.2m$		CTHR $R_0 = 6m$
Peak Fusion Power, MWt	900	900	900	1200
Plasma Duty Cycle	0.5	0.85	0.85	0.9
Plant Availability	0.4	0.7	0.7	0.7
Blanket Exposure, MW-yr./m ²	0.4	1.2	1.2	1.2
Blanket Coverage Factors:	Toroidal	0.56	0.85	0.85
	Poloidal	0.43	0.67	0.95
Fertile Fuel	ThO ₂ ^(a)	ThO ₂ ^(a)	UO ₂ ^(a)	UC ^(b)
Average Blanket Multiplication Factor, M	2.4	4.9	10.7 ^(c)	9.4 ^(d)
DCA ^(e) Thermal Power, MWt	240	2400	6100	8200
Design Electrical Power, MWe	0	700	1800	2500
Electric Power Consumption, MWe	1140 ^(f)	200	200	220
Net Electrical Energy, kW-hr./yr.	(4x10 ⁹)	3x10 ⁹	9.8x10 ⁹	1.36x10 ¹⁰
Net Fissile Atoms/Fusion Neutron (Average)	0.6	0.6	1.1	1.2
Initial Fuel Loads, Tonnes Fertile	150	470	600	500
Tonnes Li ₂ O	0	50	50	520
Heavy Metal Discharge Rate, T HM/yr.	-	40	60	123
Fissile Enrichment Rate Avg. w/o/Yr.	0.12	0.3	0.4	0.70
Fissile Production Rate, kg F/Yr.	165	1200	2300	3200
Potential Tritium Breeding Ratio	0.5	0.6	0.8	1.08

(a) Boiling water cooled concept.

(b) He cooled concept.

(c) Average multiplication factor over 10 year irradiation period to achieve ~ 4 w/o enrichment for a direct fuel cycle (7-1).

(d) Average multiplication factor over 4 year irradiation period to achieve 3 w/o enrichment for a chemical reprocessing fuel cycle.

(e) Plasma duty cycle averaged values including high grade heat from all cooling systems.

(f) Beam driven operating mode; all others are ignited operation.

TABLE 7-2
CTHR CAPITAL COST ACCOUNT SUMMARY

<u>Account Number</u>	<u>Account Title</u>	<u>Estimated Cost (1978 M \$)</u>
<u>DIRECT COSTS:</u>		
Nondepreciating Assets:		
20	Land and Land Rights	3
Depreciating Assets:		
21	Structures and Site Facilities	100
22	Reactor Plant Equipment	1578
23	Turbine Plant Equipment	265
24	Electric Plant Equipment	59
25	Miscellaneous Plant Equipment	13
26	Special Materials	50
		<hr/>
	TOTAL DIRECT COST	2063
<u>INDIRECT COSTS (All Depreciating Assets):</u>		
91	Construction Facilities, Equipment and Services	309
92	Engineering and Construction Management Services	309
93	Other Costs	104
		<hr/>
	TOTAL INDIRECT COSTS	722
<u>TIME RELATED COSTS (All Depreciating Assets):</u>		
94	Interest During Construction	1115
		<hr/>
	TOTAL CAPITAL COST	3900

Standard Accounts for Cost Estimates⁽⁷⁻³⁾. Interest during construction was taken at 6% escalation and 10% cost of capital over a 7 year construction period. The cost breakdown for the reactor equipment is shown in Table 7-3. The initial fuel loading of fertile UC, estimated at \$90M, is in account 22.01.01.03 and the Li_2O solid fuel rods, estimated at \$130M, is in account 22.01.01.01. An initial tritium inventory of 40 kg is assumed and costed in the special materials account 26 at the rate of $\$1.2 \times 10^6/\text{kgT}$. These initial fuel load costs are included in the capital costs as suggested by the Fusion Standard Accounts Guidelines⁽⁷⁻³⁾. This method was not used in the previous economic study⁽⁷⁻²⁾.

The operating costs for this CTHR concept are shown on Table 7-4. The O&M cost includes \$38M for replacement parts (vacuum vessel, limiters, neutral beam sources, blanket modules and structures, coils, etc.) and assumed 3 mills/kW hr for the O&M rate⁽⁷⁻²⁾. Cost of capital was calculated at a fixed charge rate of 15%. Electricity sales income was estimated using 33.2 mills/kW hr. The calculation of this value is shown on Table 7-5 and is based on a typical, on-line in 1978 nuclear power plants' cost of electricity to its busbar. A second full load of fusile fuel for tritium breeding was assumed at \$130M and a fixed charge rate of 15% applied. This would permit reprocessing of the Li_2O rods to remove the tritium for subsequent use. The annual cost of deuterium is < \$200,000. The annual operating cost (\$231M) is used to establish the hybrid enrichment charge by dividing by the fissile production rate or the heavy metal discharge rate. The latter value is used in the analysis in Subsection 7.2 since it accounts for the value of fertile fuel in the product and simplifies the LWR fuel cycle cost analysis.

The fertile fuel reloading costs are determined with the LWR fuel cycle cost since it is assumed that the utilities would own the reload fuel throughout the entire process. The hybrid owner would buy the initial fuel load to assure operation of the hybrid (as a power plant or for their own fuel needs) during the life of the plant even if no customer fuel was in process for enrichment.

TABLE 7-3

CAPITAL COST BREAKDOWN FOR CTHR REACTOR EQUIPMENT

<u>Account Number</u>	<u>Account Title</u>	<u>Cost, M \$ (1978)</u>
22	Reactor Plant Equipment	
22.01	Reactor Equipment	
22.01.01	Blanket and First Wall	260.2
.01	T Breeding Fuel (Li_2O)	130
.02	Vacuum Vessel (First Wall)	6.2
.03	Fertile Fuel (UC)	90
.04	Limiters	2.7
.05	Blanket Pressure Vessel - Inner Blanket	13.8
	Outer Blanket	17.5
22.01.02	Shield	84.6
.01	TF Coil Shields (Inner and Outer)	83.2
.02	NBIS Duct Shields	1.4
22.01.03	Magnets	306.5
.01	TF Coils	216.1
.02	SF Coils	36.1
.03	OH Coils	22.0
.04	Winding Equipment	27.0
.05	Bundle Divertor Coils	5.3
22.01.04	Supplemental Heating	58.8
.01	Neutral Beam Injection System	58.8
22.01.05	Primary Structure and Support	44.3
.01	Reactor Structure	24.6
.02	Equipment Support (Bucking Cylinder)	19.7
22.01.06	Reactor Vacuum Systems	11.5
.05	Reactor Vacuum Systems	7.1
.06	Reactor Vacuum Wall (Enclosure Shell)	4.4
22.01.07	Power Supply Switching & Energy Storage	141.7
.01	OH Coil Power Conversion Equipment	39.2
.02	SF Coil Power Conversion Equipment	49.6
.03	TF Coil Power Conversion Equipment	2.1
.04	Load Smoothing Circuit (for Grid Output)	7.1
.05	Pri./Sec Distribution	43.7
22.01.08	Impurity Control	9.2
.01	Bundle Divertor and Collector Chamber	9.2

TABLE 7-3 (Continued)

CAPITAL COST BREAKDOWN FOR DTHR REACTOR EQUIPMENT (Cont'd.)

<u>Account Number</u>	<u>Account Title</u>	<u>Cost, M \$ (1978)</u>
22.02	Main Heat Transfer & Transport System	50.2
.01	Primary Coolant System	44.2
.05	Clean-up System	6.0
22.03	Auxiliary Cooling System	55.0
.01	Magnet Cooling	46.5
.02	Shield and Structure Cooling	3.4
.03	NBIS Cooling	0.5
.05	Vacuum Vessel and Divertor	4.6
22.04	Radioactive Waste Treatment and Disposal	10.1
.01	Liquid Wastes Processing and Equipment	3.4
.02	Gaseous Wastes and Off Gas Processing Sys.	6.7
.03	Solid Wastes Processing Equipment	(with .01)
22.05	Fuel Handling and Storage Systems	43.0
.01	Purification	8.2
.03	Preparation	11.8
.04	Injection	3.7
.05	Storage	19.3
.06	T Recovery (Plasma Exhaust)	
22.06	Other Reactor Plant Equipment	23.8
.01	Maintenance	23.8
22.07	Instrumentation and Control	12.7
.01/.02	Reactor I & C/Rad. Monitoring	12.7
	Sub Total	1111.6
	Shipping & Handling (1%)	11.1
	Installation & Assembly (15%)	166.7
		1289.4
22.98	Spare Parts Allowance (2%)	25.8
		1315.2
22.99	Design Contingency Allowance (20%)	263.0
	Total Direct Cost	1578.2

TABLE 7-4
CTHR OPERATING AND ENRICHMENT COST ESTIMATES

<u>Annual Operating Cost (1978 M\$)</u>	
O&M (Including Replacement Parts)	78
Cost of Capital	585
Fusile Fuel Breeder Reload	20
Sale of Electric Power	(452)
Net Annual Operating Cost	231
<u>Production Cost</u>	
Fissile Fuel, \$/g Pu	72
Heavy Metal, \$/kg HM (3 w/o fissile)	1878

7.2 PWR FUEL CYCLE AND ELECTRIC POWER PRODUCTION COSTS

A preliminary study to develop scoping estimates of hybrid enrichment economics for a number of potentially interesting direct fuel cycle variations was conducted⁽⁷⁻²⁾. Two basic alternatives which provided maximum coupling between the fusion hybrid reactor and conventional PWR's, yet represent simple, nearer-term applications, were focused on. These alternatives were hybrid reenrichment ("refresh") of spent PWR fuel rods and hybrid enrichment of new fertile (natural U or tails) fuel rods for direct use in a subsequent PWR burn cycle. The results were based on first iteration neutronic and economic parameters and cost normalized to a 1978 startup for both the hybrid plant and the reference nuclear plant. All of the cases studied including the conventional PWR reference assumed a "throw-away" fuel cycle for the spent PWR rods after a 33 GWd/THM burnup in the PWR.

Hybrid irradiation of spent PWR fuel rods in the "refresh" concept resulted in ~ 50% lower PWR fuel cycle costs and ~ 20% lower PWR power generation costs than those for the conventional PWR enrichment scenario. Hybrid irradiation of new

TABLE 7-5
TYPICAL PWR FUEL CYCLE/ELECTRICITY COSTS
USING ISOTOPIC ENRICHMENT

<u>Front End Fuel Cycle Costs</u>	Cost 1978 \$/kg HM
Investment in U_3O_8 ^(a)	658
Carrying Charge on U_3O_8 for 13 Months ^(b)	72
Conversion of U_3O_8 to UF_6 ^(c)	19
Carrying Charge on Conversion for 11 Months	2
Isotopic Enrichment ^(d)	371
Carrying Charge on Enrichment for 9 Months	28
Design and Fabricate Fuel (UO_2)	120
Carrying Charge on Fuel for 6 Months	6
Unlevelized Investment	1276
Levelized Cost of Delivered PWR Fuel ^(e)	2271
<u>Back End Fuel Cycle Costs</u>	
Levelized Carrying Charge during 3 Yr. PWR Operation ^(f)	502
Shipping and Disposal ^(g)	300
	802
<u>Levelized PWR Fuel Cycle Cost</u>	
	Front End 2271
	Back End 802
	Total 3073
Levelized PWR Fuel Cycle Cost ^(h) , mills/kW hr.	12.9
Levelized PWR Capital Cost, mills/kW hr.	20.3
Levelized PWR Busbar Cost, mills/kW hr.	33.2

- (a) Based on U_3O_8 @ \$40/lb., 2.6 kg U feed/lb. U_3O_8 , 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.
- (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 = 1.78 based on 6% escalation over 30 yr. plant life and discounted at 10% to 1978 present worth.
- (f) Operational carrying charge rate is 17% per year.
- (g) Shipping at \$25/kg, disposal at \$117/kg with 6% escalation and 10% carrying charge rate.
- (h) Based on 33 Gwd/THM PWR burnup and $\eta=0.3$ PWR plant efficiency.

fuel rods manufactured from tails or natural uranium would increase PWR fuel cycle costs by ~ 25% to 50% and power generation costs by ~ 9% to 18% above the reference PWR costs. A reasonable range of parameter variations were considered and the sensitivities developed which showed the hybrid cost scenario varied from competitive to non-competitive compared to the conventional enrichment cost scenario. Thus, the PWR busbar costs of electricity using fuel enriched by the hybrid were projected to bracket busbar costs using conventional nuclear fuel. The fact that these results indicated comparable PWR busbar costs using hybrid enriched fuel was encouraging since many aspects of the study were generally unfavorable to the hybrid option:

- The hybrid blanket lattice was comprised of UO_2 fuel rods designed for PWR operation. More favorable hybrid enrichment performance could be obtained using blanket fuel elements designed specifically for the hybrid (e.g., metal alloys, UC, U_2Si , etc.). Some degree of reprocessing of hybrid blanket fuel elements would be required, of course, with such an approach.
- The buildup of fission products represented a significant penalty to the hybrid enrichment scenarios since appreciably more fissile material (1.8% for the refresh case, 0.7% for hybrid irradiation of new fuel rods) was required with spent PWR fuel rods. Here again, some degree of reprocessing would be required to avoid this penalty.
- The combined PWR-Hybrid-PWR burnup for the direct fuel cycle would be nearly three times the normal 33 GWd/T PWR cycle and the fluences seen by the fuel rods were estimated to be an order of magnitude larger than present PWR exposures. This prompted concern regarding fuel rod integrity, particularly during the long (10 year) exposures in the hybrid.
- Private, rather than public, financing of the hybrid was assumed. The comparison penalizes the hybrid in this regard since enrichment costs for the conventional PWR fuel cycle are based on publicly financed isotopic enrichment facilities.

The extrapolated CTHR concept described in Subsection 7.1 eliminates most of the unfavorable conditions found in the earlier study. The UO_2 has been replaced with UC, which resulted in a 4 year hybrid enrichment period rather than ~ 10 years. This reduces hybrid enrichment charges as well as the fuel carrying charges during the hybrid enrichment period. Reprocessing will be included in the cost analysis presented in the following paragraphs. This will remove the undesirable fission product buildup and the concern for the

fuel rod integrity. The privately owned hybrid plant is the only unfavorable aspect retained in this analysis. This will add conservatism to the cost data, since it is based on the very cursory extrapolation from DTHR data.

The PWR fuel cycle costs using hybrid enrichment are estimated as shown on Table 7-6 using the methods developed in the previous study⁽⁷⁻²⁾. De-escalation of fuel costs prior to hybrid irradiation was deleted for conservatism and simplicity. The uranium tails case was analyzed since it requires a longer hybrid enrichment period than natural uranium would require. Costing rates and carrying charge periods are shown on the table for each step of the fuel cycle costing. The scenario being costed is shown pictorially in Figure 7-2. The cost of UC fuel rods was assumed to be \$30/kg more than UO₂ fuel rods. Reprocessing of the UC rods was also assumed to be \$30/kg more than UO₂ rods. The front end fuel cycle costs are grouped to show the cost of the delivered PWR fuel. The back end fuel cycle costs were simplified by assuming spent rod disposal rather than additional reprocessing to salvage the U and Pu. The levelized PWR busbar cost was developed assuming a typical 1200 MWe, nuclear power plant going on-line in 1978. The levelized PWR capital cost charge (20.3 mills/kW hr)⁽⁷⁻²⁾ was based on a \$700 M nuclear plant cost and standard utility cost levelizing techniques. The resulting cost of electricity at the nuclear plant busbar using hybrid enrichment was 40.3 mills/kW hr.

To provide a frame of reference for evaluating the hybrid enrichment economics, a similar cost scenario using isotopic enrichment of standard PWR fuel was developed. These costs are shown on Table 7-5. The resulting 33.2 mills/kW hr is still somewhat less than the hybrid enrichment cost scenario. The hybrid enrichment process cost is quite sensitive to its capital cost and the cost assumed for electricity sales. These sensitivities were analyzed and shown in the previous economic study⁽⁷⁻²⁾. The competitiveness of the hybrid enrichment process clearly provides the incentive to develop a more detailed CTHR concept and evaluate the economics of this alternative since the hybrid reactor is a net energy producer (i.e., 2500 MWe for the extrapolated CTHR concept) as opposed to the isotopic enrichment facility which is a power consumer (i.e., a Portsmouth Gaseous Diffusion Plant expansion would have consumed 2700 MWe; the gas centrifuge concept 105 MWe)⁽⁷⁻⁴⁾.

TABLE 7-6

PWR FUEL CYCLE/ELECTRICITY COSTS USING HYBRID ENRICHMENT

<u>Front End Fuel Cycle Costs</u>	<u>Cost 1978 \$/kg HM</u>
Investment in UF ₆ (0.25 w/o tails at \$5/kg U)	5
Carrying Charge on UF ₆ for 6 years (10% carrying charge rate, ccr)	4
Design and Fabrication of Hybrid Fuel Assemblies (UC)	150
Carrying Charge on for Fuel Assemblies for 5.5 Years	104
Shipping Irradiated Hybrid Fuel	25
Carrying Charge on Shipping for 0.7 Years	2
Reprocessing Hybrid Fuel	190
Carrying Charge on Reprocessing for 1 Year	19
PWR Fuel Fabrication (UO ₂)	120
Carrying Charge on Fabrication for 1 Year	12
Unlevelized Sub Total Cost	631
Levelizing (1.78 Factor) ⁽⁷⁻²⁾	1123
Hybrid Enrichment Charge	1878
Carrying Charge for Hybrid Enrichment for 5 Years	622
Levelized Cost of Delivered PWR Fuel	3623
<u>Back End Fuel Cycle Costs</u>	
Levelized Carrying Charges during 3 Years PWR Operation (17% ccr)	847
Shipping and Disposal (25 + 117 \$/kg HM; 6% Escal., 10% ccr)	300
	1147
<u>Levelized PWR Fuel Cycle Cost (FCC)</u>	4770
Levelized PWR FCC (based on 33 GWD/THM and $\eta=0.3$ We/Wt), mills/kW hr.	20.0
Levelized PWR Capital Cost ⁽⁷⁻²⁾ , mills/kW hr.	20.3
Levelized PWR Busbar Cost, mills/kW hr.	40.3

Figure 7-2. Hybrid Enrichment of Uranium Tails for a PWR Fuel Cycle.

REFERENCES - SECTION 7.0

1. Chapin, D.C., "Nuclear Analysis on the Enrichment of PWR Fuel in a Fusion Hybrid Reactor", WFPS-TME-101, September 1978, Westinghouse Electric Corp., Pittsburgh, PA
2. Kelly, J. L. et al, "Preliminary Study of the Economics of Enriching PWR Fuel with a Fusion Hybrid Reactor", WFPS-TME-095, September 1978, Westinghouse Electric Corp., Pittsburgh, PA.
3. Schultz, S. C., Willke, T. L., and Young, J. R., "Fusion Reactor Design Studies - Standard Accounts for Cost Estimates", PNL-2648, UC-20, May 1978.
4. "Final Environmental Statement - Portsmouth Gaseous Diffusion Plant Expansion," ERDA-1549, September, 1977.