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Fusion Breeder Studies Program

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October 17, 1986

Lawrence
Livermore
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Laboratory

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Fusion Breeder Studies Program

Final Report

October 17, 1986

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3. Technical Papers and Conference Presentations

- o "Updated Reference Design of a Liquid Metal Cooled Tandem Mirror Fusion Breeder", proc. ICENES-86 conf., Madrid (July, 1986)
- o "Overview of Updated Liquid Metal Cooled Fusion Breeder Reference Design", Trans. Am. Nucl. Soc. Fusion Mtg., Reno, Nevada (June, 1986)
- o "Updated Reference Design of a Liquid Metal Cooled Tandem Mirror Fusion Breeder", full paper accepted for publication in Fusion Technology

Characterization of Fusion-Fission Hybrid Reactors

**Presented to:
National Research Council Committee on Fusion Hybrid Reactors**

Contents of the Briefing

Key characteristics

Types of hybrids

Focus on a specific example to evaluate

- **Performance**
- **Economics**
- **Status of the technology**
- **Environmental and safety characteristics**

Comparison with LMFBR and fusion electric

Development requirements and deployment strategy

Fusion Hybrids Can Be Characterized by a Few Key Parameters

Blanket energy multiplication (M)

- **Determines required fusion gain (Q)**
- **Determines fission activation and afterheat level**

Net fissile breeding (F)

- **With M, determines fission client support ratio (R)**
- **Determines actinide activation and afterheat level**

Capital cost relative to fission client

- **Largest impact upon cost of fuel and electricity**
- **Excess cost divided among fission clients**

Fuel cycle cost

- **Also a large system cost impact**
- **Driven by fissile discharge enrichment and cost of reprocessing**

Breakeven price of uranium

- **Best FOM to measure hybrid against alternatives**
- **Basis for projecting possible breakeven date**

There Are Three Major Types of Hybrid Blankets

**Concept 1
Power Amplifier Blanket**

**Basis: Self-generated ^{239}Pu
Thermal Fissions**

**Concept 2
Fast-Fission Blanket**

Basis: Uranium fast-fissions

**Concept 3
Fission-Suppressed Blanket**

Basis: Beryllium $n, 2n$ reactions

The Characteristics of Fusion Hybrid Blankets Vary Greatly

	Power Amplifier	Fast Fission	Fission- Suppressed	Pure Fusion
Blanket energy multiplication (M_{ave})	25	10	2	1.3
Required gain for efficient electricity production (Q)	1	2	10	16
Net fissile breeding per fusion (F)	1	1.2	0.7	—
Fuel production rate (kg/MW-yr)	0.2	0.6	1.9	—
LWR consumption rate (kg/MW-yr)	0.2 (Pu)	0.2 (Pu)	0.14 (U)	—
LWR support ratio (R)	1	3	14	—
Dominant afterheat source	Fission products	Fission products	Actinides	Structure
Dominant mode of operation	Power plant	Power plant	Fuel factory	Power plant
Capital cost goal relative to LWR	1.2	1.4	2.5	1.3
Fuel reprocessing cost impact	Low	Low	High	—

Fission-Suppressed Blankets Require Higher Fusion Performance than Fissioning Blankets

	Minimum Required Fusion Performance		
	Plasma Gain (Q)^A	Fusion Power^B	Neutron Wall Load
Fast Fission Blanket	4	500 MW	1 MW/m²
Fission-Suppressed Blanket	8	2000 MW	2 MW/m²

^A Considers not only power flow, but cost of plasma heating and sale of fissile fuel

^B For ~ 4000 MW_T plant

If These Higher Levels of Fusion Performance Can Be Achieved Fission-Suppressed Blankets Offer Several Advantages

Higher LWR thermal support ratio (~ 15)

- **Higher allowed cost at fixed thermal power**
- **More compatible with fuel cycle center siting and fuel factory mode of operation**
- **Larger impact in earlier time frame**

Improved blanket safety characteristics

- **Lower power density**
- **Lower fission product afterheat levels**
- **Potential for semi-passive response to loss of coolant accident**

Improved proliferation resistance

- **Plutonium can be eliminated from fresh fuel**
- **Denatured uranium fuel cycles (3-20% ^{233}U in ^{238}U)**
- **Spiking of ^{233}U with ^{232}U can be enhanced**

Better fit to fusion-electric R&D program

A Symbiotic Electricity System with Fission-Suppressed Breeders Would Have a Few Fusion-Based Centers

Some Characteristics of the Fuel Cycle Center

Excellent follow-on to once-through LWR

- **Replaces uranium mining and enrichment**
- **Utility continues to operate LWR**

Only a few fuel cycle centers required

- **A fuel cycle center with 3 large (2600 MW_f, 5000 MW_f) fusion breeders can support 80 GW_e of power generation**
- **Remote siting can relax some safety/environmental and safeguards concerns**

Fissile fuel export can provide a needed opportunity to improve US balance of trade and regain leadership in international nuclear markets

- **Encourage LWR development at home and abroad**
- **Enter long-range contracts to build LWRs and/or supply denatured fuel and reprocess spent fuel**
- **Sensitive fuel cycle operations controlled in US**
- **Plutonium economy could be avoided**

Fusion Hybrid Safety Must Be Viewed from the Systems Perspective

In comparison with fusion-electric blankets, fusion hybrids will have higher radioactive inventories and decay afterheat levels

Key requirement for fusion hybrid safety is that it does not degrade the safety of the overall electricity generation system. The hybrid must be as safe as the LWR clients which dominate the system

- **Studies* show that safety goals can be achieved because of subcriticality and lower power density**

In comparison with the LWR, hybrid decay power densities will be reduced, but are a concern-especially considering the added complexity of fusion

*** References**

- 1) "Safety Assessment of the Fusion Breeder," J. Garner and I. Mays, *Nuclear Technology*, July 1985
- 2) Special Topics Report for the Reference Tandem Mirror Fusion Breeder: Reactor Safety Assessment, LLNL Report UCID 20166, September 1984

Fusion Hybrid Safety Must Be Viewed from the Systems Perspective (Continued)

Fusion breeders with low M fission-suppressed blankets offer important advantages

- **Fissioning reduced to the extent that actinides (e.g., ^{233}Pa and ^{233}Th) provide the greatest contribution to both the hazard and the afterheat**
- **Lower afterheat levels allow much longer time before corrective action is required**
- **Mobile fuel provides for semi-passive away from reactor cooling in dump tanks**

Ambitious safety goals have been set for fusion breeders with fission-suppressed blankets

Actinide Afterheat Dominates During Time Periods Relevant to Blanket Heatup

**One Example of a
Fission-Suppressed
Fusion Breeder Which
Operates as a Fuel Center**

The Reference Liquid Metal Cooled Tandem Mirror Fusion Breeder Has Been a Focus of Design Attention Since 1980

The fusion driver

- **A Tandem Mirror based on MARS performance and a 1.7 MW/m² neutron wall load**

The breeding blanket

- **A liquid lithium cooled, packed bed with mobile beryllium/thorium pebbles to multiply neutrons and breed ²³³U**
- **Pyrochemical fuel reprocessing is used to extract ²³³U at low (~ 1%) concentrations**

The fission reactors

- **An LWR with a denatured thorium fuel cycle (3.3% ²³³U, 18% ²³⁸U, and 78.7% ²³²Th)**
- **Thorex fuel reprocessing is used on the spent LWR fuel**
- **LWR generated plutonium burned in LWR plutonium burners (about 12% of the total number of LWRs)**

Many other combinations are feasible, have been investigated, and will be presented to the committee in later sessions

The Reference Liquid Metal Cooled, Tandem Mirror Fusion Breeder

Key Design Specifications

Tandem Mirror Fusion Driver

- MARS performance with lower wall loading

Fusion power

Plasma gain, Q

Neutron wall loading

Performance Parameter

2600 MW

26

1.7 MW/m²

Breeding Blanket

- Average blanket energy multiplication
- Net ²³³U breeding ratio
- Tritium breeding ratio

2.4

0.84

1.06

Overall Performance

- Maximum thermal power
- Average thermal power
- Average net electric power
- Capacity factor
- ²³³U production
- Number of 1 GW_e LWRs fueled

5700 MW

5075 MW

1990 MW_e

70%

6660 kg/yr

25

Fusion Breeders Can Use a Variety of Semi-Passive Heat Removal and Engineering Systems

Encouraging Thermal modeling Results for Loss of Coolant Accident with Fuel in Place

The Fusion Breeder Cost Per Unit Electricity Is Similar to MARS, but It Produces Over 6600 kg/yr of ^{233}U

	MARS	Fusion Breeder
Direct cost, \$M 1985	2235	3615
Indirect and contingency, \$M 1985	3775	1790
Total overnight cost, \$M 1985	3340	5405
Construction time, yr	6	8
Total capitalized cost, \$M 1985	3745	6290
Net electricity, MW_e	1200	1990
Capital cost per unit electrical power, $\\$/\text{kW}_e$	3165	3160
Cost relative to projected LWR ($\\$/\text{kW}_e/\\$/\text{kW}_e$)	2.30	2.35
^{233}U production (kg/yr, 70% capacity)	N/A	6655

Typical Symbiotic Electricity Generation System Based upon Fission-Suppressed Breeder and Denatured Thorium Fuel Cycle

Typical Reprocessing Economics

(1985 \$, First Year of Operation)

	Thorex	Pyro-chemical
Throughput (Mt/yr) ^a	1900	1900
Fissile discharge, assy, atom %	1.05	1.05
Plant capital cost, \$M	2600	1300
Cost of capital, \$M/yr ^b	340	170
Plant operating cost, \$M/yr ^c	170	40
Total annual cost, \$M/yr	510	210
Reprocessing cost, \$/kgHM	270	110
Unit cost, \$/g	26	10.5

^a Three fusion breeders sharing a reprocessing plant

^b Utility ownership (13.1%/yr)

^c First year, increases 3%/yr thereafter

LWR Fuel Cycle Performance Data

	LWR Fuel Cycle Type					
	Pu Burner	Denatured Uranium (DU)	Denatured Thorium (DT)	DU + Pu (a)	DT + Pu (b)	²³⁵ U Burner
Net fissile requirement, g/kW-yr (c)	0.200	0.205	0.144	0.153	0.126	0.194
Excess plutonium production, g/kW _t -yr (c)	None - recycled	0.068	0.028	None - recycled	None - recycled	None - recycled
Fuel burnup, MWD/MT	30,400	33,000	33,400	32,300	33,000	30,400
Equilibrium fissile enrichment, atom %	4.9	2.4	3.3	3.1	3.5	3.2
Core power density, kW _t /kgHM	37.1	37.2	40.2	37.2	39.8	38.4
Equilibrium fissile inventory, g/kW _t	1.83	0.68	1.09	1.07	1.29	1.14
Net thermal-to-electric efficiency, %	33.4	33.4	33.4	33.4	33.4	33.4

a) Mixed system includes 74% denatured uranium fuel cycle LWRs and 26% plutonium fuel cycle LWRs

b) Mixed system includes 88% denatured thorium fuel cycle LWRs and 12% plutonium fuel cycle LWRs

c) at 100% plant capacity factor

Research Requirements to Confirm the Viability of the Reference Fusion Breeder Concept

MHD pressure drops

- **Packed bed and duct flow including insulator development**

Liquid metal compatability

- **Corrosion, mass transfer and coating development**

Beryllium irradiation damage

Pyro-chemical reprocessing

- **Magnesium dissolution process demonstration**

Fissile breeding

- **Integral test assembly with a point**

Development of thorex fuel cycle for the LWR

Simple Average Capital Cost Example

Premise

- Average capital cost of an electricity generation system is a good indicator of overall economics

Assumptions

- Fusion breeder cost per unit of thermal power = 2.4 x cost of fission converter client
- Fusion breeder electrical efficiency = LWR client electrical efficiency
- Fusion breeder provides fissile makeup for 13.5 fission clients of same thermal power

Average capital cost/kW_e (normalized to client cost/kW_e)

$$\frac{(2.4 \times 1) + (1 \times 13.5)}{(1.0 \times 1) + (1 \times 13.5)} = 1.10$$

Some Assumptions of the Economics Analysis

3%/year inflation rate

30 financial lifetime

Fixed charge rates

- **13.1%/yr utility owner**

70% plant capacity factor

8 year construction time

Breeder reprocessing plant owned by breeder owner and financed at same rate

LWR cost = 444 \$/kW_t

LWR fuel cycle costs developed based upon NASAP and other sources

Characteristics of a U_3O_8 Fueled LWR with Reprocessing

LWR capital cost ($\$/\text{kW}_t$)	444
Total fixed charge rate (%/yr)	13.2
Fuel cycle	Full recycle
^{235}U Consumption ($\text{g}/\text{kW}\cdot\text{yr}$)	0.194
Average burnup (MWD/MT)	30,400
Reprocessing cost ($\$/\text{kgHM}$)	417
Fabrication cost ($\$/\text{kgHM}$)	685
Enrichment cost ($\$/\text{kg}$ separative work)	60
Transport and disposal cost ($\$/\text{kgHM}$)	260
Year one price of purchased U_3O_8 ($\$/\text{kg}$)	55
Year one cost of electricity ($\text{mil}/\text{kW}_e\text{H}$)	52
U_3O_8 escalation rate (%/yr)	3
Average PV cost of electricity ($\text{mil}/\text{kW}_e\text{H}$)	33

Economic Results for the Symbiotic System

	<u>Symbiotic System</u>
Breeder	
• Total cost (\$/kW _t)	1120
• Breeder to LWR capital cost ratio	2.35
• Annual capital cost* (\$/kW-yr)	147
• Annual operating cost (\$/kW-yr)	47
• Total annual cost (\$/kW-yr)	195
LWR	
• Total cost (\$/kW _t)	444
• Annual capital cost (\$/kW-yr)	59
• Annual operating cost (\$/kW-yr)	42
• Total annual cost (\$/kW-yr)	101
System Cost	
• Breeder to LWR annual cost ratio	1.93
• Year one cost of electricity (mills/kW-hr)	47
• Average PV cost of electricity (mills/kW-hr)	36
• Average cost of electricity relative to U ₃ O ₈ fueled LWR	1.1
• Average breakeven cost of U ₃ O ₈ (\$/kg)	196

The Fusion Breeder Breakeven Is Much Less Sensitive to Capital Cost than LMFBR Breakeven

Analysis developed from ORNL data using updated hybrid performance, cost, operating, and fuel cycle cost assumptions

Comparison of Fusion Breeders with LMFBRs

This comparison assumes that LWRs will re-emerge as a viable energy option

Fusion breeder can be deployed later and still have an earlier impact

Fusion breeder has the potential to cap the price of uranium at lower levels

- **40 to 100 \$/lb versus 60 to 200 \$/lb**

Market penetration for fusion breeder will be easier and faster

Fusion breeder operation in a safeguarded fuel center can be made more proliferation resistant

Window of opportunity for fusion breeder development, demonstration and deployment is larger

Fusion Technology Development Requirements for Fuel Breeding Are Relaxed in Comparison to Electric Power Production

Because of relaxed economic constraints, performance of proven confinement concepts is acceptable

- **Tokamaks with copper coils and/or low β**
- **Tandem mirrors based on MARS performance**
- **Operation with first wall fluxes less than 2 MW/m²**

Relaxed economic constraints also allow higher costs for magnets and plasma heating systems

Higher blanket energy multiplication allows operation with lower plasma gain, Q

If Fissile Fuel Production Were a Fusion Program Goal more Emphasis Would Be Placed on Nuclear Technology Development

A nuclear R&D program is needed

- **To resolve key blanket technical issues**
- **To develop fuel cycle technologies which are economical and proliferation resistant**
- **To improve safety and environmental characteristics**

This R&D program could serve as a focus for fusion nuclear technology development

- **Provides an excellent opportunity to cooperate with the Russians in fusion development**

Nuclear Technology Development Should Start Now and Proceed Through Three Phases*

Non fusion testing and fuel cycle development through the early 90's

- **In non-nuclear test stands, fission reactors and 14 MeV neutron sources**
- **To address and find solutions to known feasibility issues**
- **To develop promising concepts for integrated testing**

Low fluence integrated tests in the late 90's

- **Reactor relevant components in a reactor relevant fusion environment**
- **An engineering feasibility demonstration**

High fluence tests in a commercial demonstration facility

- **To verify commercial feasibility and establish the breakeven cost of U₃O₈**

* See the FINESSE Study

An Engineering Demonstration of a Fusion Breeder Could Be Completed in the 2000 Timeframe

Conclusions

The fusion breeder - LWR combination has the potential to be the best inexhaustable source of energy

- **Lowest cost**
- **Preserves LWR experience and investment**
- **Can be operated in a safe, proliferation resistant manner**

The fusion breeder technology is less mature than the LMFBR but potential is higher

- **There will be time to develop the technology because hybrid can be deployed later and still have an earlier impact**

A goal to develop the fusion breeder will benefit the fusion program

- **Performance of proven confinement concepts is acceptable**
- **Relaxed economic constraints reduce fusion technology development requirements**
- **Early application of fusion research**
- **Allows utilities to introduce fusion technology gradually**
- **Can serve as a focus for nuclear technology development**

Conclusions (Continued)

If we start now, a fusion breeder demonstration could be operational by 2010

This will be soon enough

- **To ensure an adequate fuel supply for LWRs**
- **To allow the US to leapfrog LMFBR technology**

This demonstration will

- **Stabilize the price of uranium in the 40 - 100 \$/lb range**
- **Allow nuclear power to play a significant role in meeting the energy needs of the US and the world (e.g., 50% of our electricity needs by 2050)**

**SUMMARY OF FISSION-SUPPRESSED HYBRID BLANKET TECHNOLOGY: DESIGN AND
DEVELOPMENT ISSUES**

D. H. BERWALD

OUTLINE

- THREE FISSION-SUPPRESSED BLANKET DESIGN CONCEPTS ARE REVIEWED
  - LIQUID METAL COOLED BLANKET
  - HELIUM COOLED
    - HETEROGENIOUS THORIUM METAL FERTIVE FUEL BLANKET
    - HOMOGENIOUS MOLTEN SALT BLANKET
- FOR EACH BLANKET DESIGN
  - FEATURES AND ADVANTAGES
  - ENGINEERING/DEVELOPMENT ISSUES
  - NEAR TERM DEVELOPMENT LOGIC
- OVERALL PERSPECTIVE

FEATURES

- IT OFFERS HIGH NET ^{233}U BREEDING, (1.7 $\text{Kg/MW}_T\text{-FPY}$)
- IT IS COMPOSED OF REASONABLY CONVENTIONAL MATERIALS (FERRETIC STEEL STRUCTURE, THORIUM METAL FUEL),
- IT PROVIDES EXCELLENT TRITIUM RETENTION,
- IT UTILIZES A LITHIUM COOLANT TECHNOLOGY WHICH IS SIMILAR TO THE LMFBR SODIUM COOLANT TECHNOLOGY. LITHIUM TECHNOLOGY IS ALSO BEING DEVELOPED FOR SPACE POWER APPLICATIONS (E.G., SP-100),
- IT RESULTS IN A LESS COMPLEX INTERNAL STRUCTURE THAN THE PRINCIPLE ALTERNATIVE COOLANTS (HELIUM, WATER).

ISSUES

- MHD PRESSURE DROPS- PRESSURE DROP DUE TO PACKED BED AND DUCT FLOW. PROSPECTS FOR DEVELOPING RADIATION RESISTANT ELECTRICAL INSULATORS WHICH WOULD ELIMINATE CURRENT PATHS AND RESULTING PRESSURE DROPS.
- LIQUID METAL COMPATIBILITY - INTERMETALLIC REACTIONS, MASS TRANSFER, FISSION PRODUCT/ACTINIDE TRANSPORT. PROSPECTS FOR DEVELOPING BERYLLIUM AND/OR THORIUM COATINGS WHICH WOULD LIMIT TRANSPORT.
- BERYLLIUM IRRADIATION DAMAGE - IRRADIATION CREEP, SWELLING, DUCTILITY, AND RESULTING PEBBLE FAILURE MODES.
- PYRO-CHEMICAL REPROCESSING - MAGNESIUM DISSOLUTION PROCESS ATTRACTIVENESS.
- PEBBLE FLOW - ISSUES SUCH AS SELF-WELDING AND THERMAL RACHETING CAUSING POTENTIAL FOR FLOW BLOCKAGE.
- LITHIUM SAFETY - POTENTIAL FOR INTERACTIONS WITH WATER COOLED COMPONENTS.

Near Term Development Program for Liquid Metal Fusion Breeder

CROSS SECTION ELEVATION VIEW OF THE HELIUM COOLED FUSION BREEDER REACTOR CONCEPT WITH A TOP MOUNTED DIVERTOR

FEATURES

- HELIUM DOES NOT INTERACT WITH THE MAGNETIC FIELD,
- HELIUM IS CHEMICALLY AND NEUTRONICALLY INERT,
- A PROPERLY CONSTRUCTED HELIUM SYSTEM CAN LOSE COOLANT PRESSURE, BUT PRESERVE FLOW (AT ~ 1 ATM),
- HELIUM BLANKETS ARE EQUALLY APPLICABLE TO ALL FUSION CONFINEMENT SCHEMES (TOKAMAK OR MIRROR).

TOP VIEW OF HELIUM COOLED TOKAMAK FUSION BREEDER BLANKET

ENLARGED VIEW OF THE FRONT OF A MODULE

ISSUES

- FIRST WALL LIFETIME - FIRST WALL FAILURE MODE SUBJECT TO SURFACE HEAT AND PARTICLE LOADS, NEUTRON IRRADIATION DAMAGE, INTERNAL PRESSURIZATION, DEFORMATIONS WHICH COULD PINCH NARROW FIRST WALL COOLANT CHANNELS.
- TRITIUM BREEDER DEVELOPMENT - ISSUES INVOLVING TRITIUM TRANSPORT, TRITIUM RECOVERY AND THE THERMO-MECHANICAL RESPONSE OF CLAD BREEDING MATERIALS SUBJECT TO NEUTRON FLUENCE (E.G., CLAD RUPTURE, LOSS OF THERMAL CONTACT).
- TRITIUM CONTROL IN HELIUM COOLANT - KINETICS OF IN-LOOP OXIDATION CONVERSION FROM T_2 TO T_2O . TRITIUM PERMEATION BARRIERS AT THE BARRIERS AT THE BREEDER TUBE, FIRST WALL, AND STEAM GENERATOR.
- SIDE SUPPORT OF VERTICLE LOBES - FLAT LOBE SIDE WALL DEFORMATIONS DUE TO INTERNAL HELIUM PRESSURE AT THE BLANKET MODULE ENDS
- BERYLLIUM IRRADIATION DAMAGE - IRRADIATION CREEP, SWELLING, DUCTILITY, AND RESULTING PEBBLE FAILURE MODES.
- PYRO-CHEMICAL REPROCESSING - MAGNESIUM DISSOLUTION PROCESS ATTRACTIVENESS.
- PEBBLE FLOW - PRIMARILY THERMAL RACHETING.

HELIUM-COOLED MOLTEN-SALT FUSION BREEDER BLANKET FEATURES

- MOLTEN SALT ALLOWS ON-LINE TRITIUM AND ^{233}U REMOVAL;
- MOLTEN SALT PROCESSING BY FLUORINATION IS LOW COST;
- CONTINUOUS PROCESSING PREVENTS POWER SWING AND THEREBY MAXIMIZES THE SUPPORT RATIO (NUMBER OF EQUAL POWER FISSION REACTORS THAT CAN BE RE-FUELED);
- THORIUM FUEL ELEMENT FABRICATION ELIMINATED.
- LOW TRITIUM INVENTORY DUE TO LOW SOLUBILITY OF TRITIUM IN MOLTEN SALT;
- LOW CHEMICAL REACTIVITY OF MOLTEN SALT, WATER AND HELIUM.

ISSUES

- TRITIUM MANAGEMENT - PERMEATION-BARRIER DEVELOPMENT AND DEMONSTRATION.
- BERYLLIUM FEASIBILITY - BERYLLIUM-BERYLLIUM SELF-WELDING AND BERYLLIUM-STEEL WELDING AT THE CONTACT POINTS. ABILITY OF THE BERYLLIUM BALLS TO STAND UP UNDER NEUTRON RADIATION. BERYLLIUM SWELLING--LEADING TO JAMMING, CRACKING, CHIPPING, AND FLYING MISSILES THAT MIGHT DAMAGE CIRCULATORS.
- MATERIAL COMPATIBILITY - STEEL (TYPE 316) LIFETIME IN CONTACT WITH THE SALT (KEPT IN A REDUCING STATE).
- NEUTRON ECONOMY - BREEDING PERFORMANCE OF THE MOLTEN-SALT DESIGN AS AFFECTED BY MODERATION AND PARASITIC ABSORPTION IN STRUCTURAL MATERIAL (PRINCIPALLY IRON).

Near Term Development Program for Molten Salt Fusion Breeder

OVERALL PERSPECTIVE

- ALL OF THESE DESIGNS APPEAR FEASIBLE, BUT AWAIT ANY SERIOUS DEVELOPMENTAL WORK.
- MOST OR ALL OF THE KEY DEVELOPMENTAL ISSUES CAN BE RESOLVED VIA NON-NUCLEAR AND IN-CORE FISSION TESTING BY 2000.
- THE OVERALL COST TO DEVELOP TWO BLANKET CONCEPTS WOULD BE ~\$150M THROUGH 2000.
- AMONG THE BLANKET TYPES,
  - LITHIUM COOLANT IS ATTRACTIVE, BUT MAY NOT BE FEASIBLE.
  - HELIUM COOLANT IS MORE COMPLEX, BUT OFFERS SEVERAL POSSIBILITIES.
  - MOLTEN SALTS OFFER IMPORTANT ADVANTAGES.

Basis for Symbiotic Economics

Zeroth Order Symbiotic Economics

COMMENTS ON JASSBY'S DISCUSSION ON FAST VS SUPPRESSED FISSION

- POINT #1 - JASSBY EXPRESSES MISGIVINGS ABOUT TORTUOUS COST ACCOUNTING METHODS AND ASSERTS THAT USING EXCESS NEUTRONS TO MAKE FISSION POWER IN THE BLANKET IS ENORMOUSLY SUPERIOR TO BREEDING.
MY REPLY:
  - POWER PRODUCTION TO COMPETE WITH LWRS REQUIRES THAT \$/KWE FOR FUSION HYBRID CAN BE LOW ENOUGH TO COMPETE. THIS IS UNLIKELY.
  - FAST-FISSION VERSUS FISSION-SUPPRESSED ECONOMICS WAS EXTENSIVELY STUDIED BY WESTINGHOUSE UNDER EPR1 CONTRACT. THEY CONCLUDED THAT FISSION-SUPPRESSED IS SUPERIOR IF THE PLANT IS THERMALLY LIMITED BECAUSE A LARGER FUSION PLANT (MORE NEUTRONS) ARE ALLOWED. FAST FISSION IS SUPERIOR WHEN THE SAME (SMALL) FUSION DRIVER IS USED.
  - THE "TORTUOUS COST ACCOUNTING METHOD" IS ESSENTIALLY THE STANDARD INDIFFERENCE VALUE TECHNIQUE USED THROUGHOUT THE NUCLEAR INDUSTRY. DELENE'S ANALYSIS SHOWN AT THE FIRST MEETING OF YOUR COMMUNITY NEGLECTS THE SUPPORT RATIO, BUT SHOWS THAT BREEDING CAN BE VERY ATTRACTIVE.

COMMENTS ON JASSBY'S DISCUSSION ON FAST VS
SUPPRESSED FISSION (CONT'D)

- POINT #2 - JASSBY ASSERTS THAT BERYLLIUM MULTIPLICATION IS EXAGGERATED DUE TO LOSS OF COVERAGE AND LOSSES DUE TO THE FIRST WALL. MY REPLY:
  - 2-D MONTE CARLO ANALYSIS OF A TOKAMAK BREEDER WITH A DIVERTOR HAVE BEEN PERFORMED BY LLNL. THE DIVERTOR LOSS IS 15% IN TOTAL BREEDING, OR 34% IN NET FISSION PRODUCTION, GIVING $F_{NET} = 0.5$, SUPPORT RATIO = 10. THIS LOSS IS SIGNIFICANT, BUT NOT CATESTROPHIC.
  - THESE RESULTS CAN IMPROVE IF PUMP LIMITERS (LESS DUCT AREA) OR SELF-PUMPED LIMITERS BEING STUDIED BY ANL, UCLA AND OTHERS ARE VIABLE.
  - THESE RESULTS INCLUDED A 1.05 CM THICK FIRST WALL AND ARE NOT INTOLERABLY SENSITIVE TO INCREASED WALL THICKNESS. THE NEUTRON SPECTRUM FOR TFTR SHOWN BY JASSBY IS A MOSTLY STEEL SYSTEM AND IS IRRELEVANT TO A COMMERCIAL SCALE HYBRID WITH NEARLY COMPLETE BLANKET COVERAGE.

COMMENTS ON JASSBY'S DISCUSSION ON FAST VS
SUPPRESSED FISSION (CONT'D)

- POINT #3 - JASSBY STATES THAT "THE CONCEPT OF SUPPORT RATIO ---SEEMS TO BE AN ARTIFICE TO OBSCURE THE ENORMOUS ABSOLUTE COST OF PRODUCTION FISSION FUEL FROM A HYBRID". HE GOES ON TO INSIST THAT HYBRID FUEL MUST COMPETE ON A \$/G BASIS WITH ENRICHED U_{308} . MY REPLY:
  - EQUIVALENT U_{308} COST IS A BETTER MEASURE OF ECONOMIC VIABILITY THAN \$/G OF ^{233}U OR ^{239}Pu SINCE MARKETS IN THESE DO NOT YET EXIST.
  - THE SUPPORT RATIO CONCEPT IS USED FOR ILLUSTRATIVE PURPOSES ONLY. IT IS NOT INCONSISTENT WITH THE FUEL COST INDIFFERENCE METHODOLOGY WHICH IS ACCEPTED AS A BASIS FOR DETERMINING COMPETATIVE U_{308} COSTS FOR VARIOUS OPTIONS (E.G., LWR, LMFBR, ---).
  - IT IS NOT CLEAR TO ME WHETHER OR NOT U_{308} WILL ESCALATE FASTER THAN OVERALL INFLATION IF AND WHEN URANIUM RESOURCES BECOME SCARCE. HOWEVER, OUR RECENT CALCULATIONS AND THOSE OF ORNL DO NOT RELY UPON INCREASED COST ESCALATION.

**COMMENTS ON JASSBY'S DISCUSSION ON FAST VS
SUPPRESSED FISSION (CONT'D)**

OTHER COMMENTS

- **FAST-FISSION SYSTEMS WILL GENERATE MUCH HIGHER DECAY AFTER-HEAT AND HAZARD LEVELS.**
- **POWER PRODUCING BLANKETS WILL, MOST LIKELY, REQUIRE UTILITY OPERATORS.**
- **POWER PRODUCING BLANKETS WITHOUT REPROCESSING/RECOVERY WILL GENERATE VAST AMOUNTS OF PLUTONIUM.**

Memorandum

☐ Action   ☐ Information Only   ☐ Reply Requested

Date May 2, 1986

No EH-86-74

GAC-FBP-86-017

From

To Professor Thomas Pigford

Subject BREAKEVEN U_3O_8 PRICES FOR ALTERNATIVE FUEL CYCLES

Ralph Moir asked me to gin up and send to you some estimates of the Breakeven U_3O_8 price for the following alternative fuel cycles:

- o A fuel cycle for which plutonium is bred in the fusion breeder, recovered via PUREX, and fully consumed (recycled) in LWRs (also PUREX)
- o A fuel cycle for which ^{233}U is bred in the fusion breeder, recovered via pyro-chemical reprocessing, and fully consumed in LWRs operating on a denatured uranium fuel cycle with plutonium recycle.

In the first case, I assume a net LWR plutonium consumption rate of 0.20 /KW-yr as opposed to 0.126 g/KW-yr for the baseline. This increased fuel consumption is, to first order, equivalent to a lower breeding rate for the fusion breeder, and is accounted for as such. The reprocessing cost (620 \$/Kg) and the other fuel cycle costs are from the Delene report (Table 3.5).

In the second case, a net LWR uranium consumption rate of 0.153 g/KW-yr is assumed and is accounted for as above. The fuel cycle costs (including 1000 \$/Kg LWR fuel fabrication cost) are, again, consistent with Delene's report. The cost of pyro-chemical reprocessing is developed as described in my March 13, memo (attached with minor revisions).

The results of my analysis are provided in the attached table. These were developed, using Delene's sensitivity factors (see his Table 5.1, Figure 5.1, Figure 5.2), in the same manner described in my March 13 memo. I would agree that these results are reasonable, but approximate, given the large deviations from the baseline case. Some additional details for cases II and III (i.e., where they differ from Case I of the March 13 memo) are provided below:

Case IIFuel Production

Decreased from 5178 Kg/yr ^{233}U to 3365 Kg/yr ^{233}U equivalent to account for 0.200/0.126=1.59 increase in LWR fuel consumption on Pu cycle (59 \$/lb penalty).

Hybrid Reprocessing Cost

Decreased from 620 \$/Kg to 390 \$/Kg to shift from THOREX to PUREX (23 \$/lb improvement)

May 1, 1986

IOM: Dave Berwald to Prof. Thomas Pigford
Subj: Breakeven U_3O_8 Costs for Alternative Fuel Cycles

LWR Reprocessing Cost

Also decreased from 620 \$/Kg to 390 \$/Kg (14 \$/lb improvement)

LWR Fuel Fabrication Cost

Decreased from 1000 \$/Kg to 640 \$/Kg to shift from U/Th fuel to Pu/U fuel (27 \$/lb improvement)

Case III

Fuel Production

Decreased from 5178 kg/yr ^{233}U to 4396 kg/yr ^{233}U equivalent to account for $0.153/0.126 = 1.21$ increase in LWR fuel consumption on denatured uranium cycle with plutonium recycle (17 \$/lb penalty)

Hybrid Reprocessing Cost

Same as Case I.

LWR Reprocessing Cost

Same as Case II

LWR Fuel Fabrication Cost

Same as Case I

Please do not hesitate to call me (516-346-3182) if I can be of assistance.

cc: R. Moir

DB/sbd

SUMMARY OF U₃O₈ BREAKEVEN PRICE VERSUS FUEL CYCLE

	Case I (Updated) Breed ²³³ U for Denatured Th Cycle (Pyro + Thorex)	Case II All Plutonium (All Purex)	Case III Breed ²³³ U for Denatured U Cycle (Pyro + Purex)
Delene's Breakeven U ₃ O ₈ Price for Case I	179 \$/lb	179 \$/lb	179 \$/lb
Direct Cost Decrease	-19	-19	-19
Fuel Prod. (Support Ratio) Adjustment	- 4	+59	+17
Capacity Factor Increase	-14	-14	-14
Operating Cost Decrease	- 4	- 4	- 4
Hybrid Repro. Cost Decrease	-50	-23	-50
SWU Cost Decrease	+14	+14	+14
LWR Repro. Cost Decrease	NA	-14	-14
LWR Fab. Cost Decrease	NA	-27	NA
Adjusted Estimated Breakeven U ₃ O ₈ Price (1983 dollars)	102 \$/lb	131 \$/lb	109 \$/lb

Environmental Aspects of
Fusion Breeder
Development

9 June 1986
AES-86-40

GAC-FBP-86-025

Dr. John Simpson
36 E. Beach Lagoon Drive
Hilton Head, S.C. 29928

Dear Dr. Simpson:

The attached draft inputs from our committee on Environmental, Safety and Economic Aspects of Magnetic Fusion Energy may be of interest to the NRC hybrid committee. In this transmittal, I have included our rationale for selection of four fission cases (input prepared by Bob Budnitz and Ron Endicott) and some preliminary economic results (prepared by Jerry Delene).

A conclusion to be drawn from these preliminary results is that the cost of electricity from advanced fission breeder reactors such as PRISM (see description and cost assumptions) is projected to exceed that of the LWR (see description) by about 30%.

If the fusion breeder cost/performance can be as advertised (2-2.5 LWR cost per unit power, 10-15 support ratio), it is likely to get its most difficult challenge from U_3O_8 rather than advanced fission reactors. Thus, a long term economic rationale for fusion breeders supporting LWRs can continue - especially noting that a significant amount of R&D related to fission breeders will be on going.

It should also be noted that a uniform contingency was applied in all cases. If the DOE guideline contingencies are used (15% for LWR, 25% for PRISM, 40% for fusion breeder) the gap would widen, despite the higher fusion breeder cost. This results from the high support ratio.

Dr. Simpson

6 June 1986

Page 2

If you find this material to be useful, please do not hesitate to call for clarification. Please treat the attached as non-referenceable draft material.

Sincerely,

GRUMMAN CORPORATION

Dr. David Berwald

DB:lb

Att.

cc: Professor T. Pigford
Professor M. Abdou
Dr. R.Moir

SECTION 4.3 RATIONALE FOR THE SELECTION OF THE FOUR FISSION CASES

The ESECOM decided that it was desirable to select a few case studies from the fission reactor arena to use as comparison cases for the several fusion cases being studied. There were three considerations that entered into ESECOM's selection process:

- i) ESECOM desired one fission case that represents 'real world data' on costs, environmental impacts, and safety parameters --- such a case would necessarily come from some selection of present U.S. fission experience.
- ii) ESECOM desired one or more fission cases that would be characterized as 'sustainable over the very long range future' in terms of fuel supply (that is, 'breeder' design(s)), because such cases would be the most nearly competitive with fusion's promise of an 'inexhaustible fuel supply'. There are no existing commercial breeder reactors, so such a case or cases would need to be drawn from design studies at a stage of development short of current commercialization.
- iii) ESECOM desired one fission case that would be an attractive 'client' for the fission-fusion hybrid concept. Specifically, such a fission design would be the consumer of the fissionable fuel bred by ~~the fusion hybrid~~ ^{at least} a fusion breeder reactor. Such a case could be one of the current commercial fission designs, or an alternative at a lesser stage of commercialization.

After much discussion, ESECOM picked four fission cases for study and comparison.

These four, each of which will be described in the paragraphs below, are:

- 1) The 'best current practice light water reactor', defined as the LWR of current commercial design that captures the best current commercial practice. There exists a DOE 'best current LWR' that is often used for such discussions, and ESECOM has selected it. ESECOM believes that this selection is a far better benchmark than something like the 'average current LWR', which is believed to be a reactor whose characteristics no utility would ever desire to purchase. ESECOM also believes that this choice is preferable to any of the currently discussed 'advanced LWR' designs, such as the US-Japanese collaborative effort or the PIUS design, since none of these has detailed enough performance specifications in either the financial or the safety arenas.
- 2) The 'large scale prototype breeder reactor' (LSPB), which represents the current centerpiece of the DOE advanced reactor program. This LSPB design, which is significantly more advanced than the decade-old Clinch River design and intends to take advantage of technical lessons learned from the British and European/Super-Phenix programs, has reasonably detailed design information, scoping information about its safety and environmental performance, and a first cut at its financial costs. In selecting the LSPB, the ESECOM decision has decided against any of the other large-scale breeder designs such as the IFR (Integrated Fast Reactor) design or the next generation design that would supersede the European Super-Phenix.

- with superior and safe char.
- 3) The 'PRISM reactor' now under development by General Electric Company with the collaboration of several other US firms under DOE support. The PRISM is a liquid-metal fast reactor of small size (135 MWe core: that takes advantage of smaller size to reduce costs of fabrication and construction significantly and achieves important safety and operational advantages as well. This design is not a true 'breeder' in the same sense as the LSPB, since its breeding ratio is very nearly unity and hence inadequate for competition with fusion over the very long term. However, its features are so different from those of the LSPB in terms of economics and safety that ESECOM believes its inclusion serves as a very useful point of reference. In selecting PRISM, ESECOM has decided not to include a design with most of the same features called SAFR, which could easily have filled the same niche.
- 4) The 'modular high temperature gas-cooled reactor' now under development by GA Technologies and a consortium of other firms under DOE sponsorship. The modular HTGR has been selected as a typical fusion-fission hybrid client reactor, although either PRISM, SAFR, or any of the advanced LWRs might have been selected as well for this niche. Our selection of this reactor design, which is still on the drawing boards, serves to broaden the breadth of our fission group in a way that another LWR or another liquid-metal reactor would not accomplish. This design is a small core (140 MWe) graphite-moderated, helium-cooled design that achieves significant safety and cost advantages by its size and simplicity.

In passing over several other of the possible fission designs, ESECOM realizes that there is a possibility that the fusion/fission comparison may not be as effectively accomplished as it might be if all (or most) of the fission designs had been picked. However, in consideration of the fact that ESECOM's main objective is the study of the fusion systems, picking four fission designs of such diversity as the above group seems a reasonable compromise.

In the following sections, the main features of each of these four fission designs will be covered. It is important to realize that only the first one (the best-present-experience LWR) represents an actual operating reactor data base. Each of the others represents some mixture of data taken from current experience and information still untested, scaled-up from smaller sizes, and/or speculation. In particular, the cost of any reactor that has never been built contains major uncertainties --- for fission reactors less so than for fusion cases, but still so.

4.3.1 The Best-Present-Experience LWR

The Best-Present-Experience LWR (PWR-BE) is a 1139 MW_e (3411 MW_e) pressurized water reactor (PWR) based upon the Westinghouse standardized four-loop single unit nuclear steam supply system (NSSS) design. Data on the PWR-BE was taken from the Energy Economic Data Base (EEDB) report prepared for the U.S. Department of Energy. The purpose of the EEDB is to facilitate technical and cost comparisons of selected power generation alternatives on a consistent basis. The design basis of the LWR's in the EEDB is the Westinghouse Reference Safety Analysis Report (RESAR-3S) with the technical features corresponding to the Seabrook Station circa 1976. The design information has been periodically modified and updated for the EEDB program to reflect current, January 1, 1984, industry practices, experience and response to regulations.

The PWR-BE represents the current best experience plants. The costs associated with the PWR-BE are considered comparable to the average costs anticipated for the "PWR for the 1990's", implying that the improved construction practices plus nuclear regulatory and license reforms anticipated during the next decade would result in significant cost improvements. The PWR-BE is considered an nth-of-a-kind commercial plant.

The reactor vessel is the standard vertical cylindrical stainless steel clad carbon steel vessel. The vessel contains the core supporting structure, control rods, and other parts directly associated with the core. The core's fuel assemblies contain slightly enriched (3%) uranium dioxide pellets which are enclosed in Zircaloy-4 tubes. Pressurized water serves as moderator, coolant, as well as the solvent for the boric acid. Control over the reactor is provided by control rod clusters (rods are of silver-indium-cadmium absorber material) that enter the reactor vessel through the upper head and by a boric acid neutron absorber dissolved in the reactor coolant.

² 4.3.2 The Large Scale Prototype Breeder

The Large Scale Prototype Breeder (LSPB) is a 1320 MW_e (3500 MW_t) conceptual design developed by the Consolidated Management Office for the LMFBR of the Electric Power Research Institute (EPRI/CoMo) and their industrial contractors for the U.S. Department of Energy. The attractiveness of this breeder concept includes the use of current or near term technological developments. Many of the innovations of the LSPB are built upon the technology developed by the DOE's LMFBR Base Program.

A modified revision of the LSPB-Loop Design (EPRI/CoMo also have a LSPB-Pool Design) has been included in the EEDB Program. The modifications were basically adjustments to make the LSPB conform to the EEDB ground rules and assumptions. The most notable adjustment was the addition of a third cooling tower to accommodate the higher wet bulb temperature of the EEDB standard "Middletown" site. ESECOM chose to use the EEDB LSPB technical and cost information since it had already been adjusted to enhance its comparability among other concepts. A notable difference is that the LSPB does represent a 2nd-of-a-kind commercial plant while all other EEDB concepts are nth-of-a-kind.

The heat source for the LSPB is a liquid-sodium cooled fast breeder reactor. The LSPB plant is arranged in a loop configuration with four separate heat transport loops. The four loops transfer thermal energy in parallel from the single reactor to the single turbine generator.

The reactor is a 316 SS vessel. The core consists of four types of assemblies. These are the fuel, breeding blanket, shield, and control assemblies. The fuel is in the form of enriched plutonium/uranium oxide pellets which are enclosed in

stainless steel tubes. The breeding blanket is in the form of depleted uranium oxide pellets which are enclosed in stainless steel tubes. The shield is composed of stainless steel rods. Reactor control is provided by neutron absorbing control rod clusters. In the event of a reactor trip the control rods fall into the core by gravity.

Different breeding ratios are obtainable in the LSPB depending upon core design. One core design allows optimization to achieve minimum breeding gain with minimum busbar cost. Another is designed for a compound system doubling time of less than 25 years.

The cost of the LSPB is significantly reduced by being able to rely on the decay heat removal system to remove heat directly from the sodium coolant in the reactor vessel. This permits the intermediate heat transport system and the steam generation system to have no safety functions. The resulting relaxed requirements i.e. other than ASME Section III requirements, account for a large cost savings.

3 4.3.2 The PRISM Breeder Design

The PRISM breeder reactor design is an attempt to develop a fast reactor using liquid sodium coolant that captures the chief safety and economic advantage that this genre would achieve over the comparable performance of any thermal reactors, at a small scale that can reduce construction times and costs significantly compared to the larger sizes (1000 MWe and up) that have represented the earlier design bases for the liquid-metal program.

As currently configured by its development team led by General Electric Company under DOE sponsorship, the PRISM design consists of a basic unit with a 135 MWe core (425 Mwt). The design links three such cores into a unit driving a single turbine-generator; and a large power station of, say, 1245 MWe could be developed by configuring three such units together (9 cores and 3 turbine-generators).

The fast-reactor design connects through a non-radioactive intermediate sodium loop to a water-based steam-generator loop. Each PRISM core is a compact pool design in which liquid sodium circulation is achieved by submersible pumps. PRISM fuel could be either metallic or mixed-oxide (not yet decided, although the reference design is FFTF-type mixed-oxide fuel). The breeding ratio is just slightly above 1.0, but higher breeding ratios are achievable without great technical modifications. The principal safety advantages achieved result from a combination of the pool-type design, the small total size allowing a secondary guard vessel to surround the primary pool vessel in case of leakage, and design simplicity.

The lower cost of this design arises from factory fabrication of its main nuclear island, allowing much tighter time schedules and quality control; from allowing much of the balance of plant to be non-nuclear-grade since its effect on safety is judged minor; and ^{from} reduction of field construction labor to a minimum. Spent fuel is to be stored within the module for one year, thereby reducing impacts on both the environment and occupational doses; this is a major cost savings as well.

The safety advantages of the traditional liquid-sodium design that arise from lower operating pressures than the LWR are captured by this design as well. Finally, the small modular concept significantly enhances availability/reliability and hence utility economics during operation.

RECOMMENDED COST ADJUSTMENTS (PRISM)

1(A) COMPARISON BETWEEN CRBRP PURCHASED COST AND PRISM COST ESTIMATES FOR COST ACCOUNT 22 SCALED TO PRISM SIZE AND CONVERTED TO 1985 DOLLARS INDICATE PRISM NSSS COST ESTIMATES AS APPROXIMATELY 30% LOW.

(B) SIMILAR COMPARISON BASED ON B&W MANUFACTURING EXPERIENCE PRISM COST ESTIMATES ARE ABOUT 20-30% LOW.

2. CONSTRUCTION SCHEDULE BASED ON 4 X 10 WORK WEEK. THIS IS ASSUMED TO ACCOUNT FOR A 15% PRODUCTIVITY IMPROVEMENT. PRISM ESTIMATES A 17 MONTH CONSTRUCTION SCHEDULE BASED ON EEDB TURBINE BUILDING SCHEDULE. THE FAST-C MODEL ESTIMATES A 34.4 MONTH SCHEDULE AND A \$1,426,400 SCHEDULE PENALTY.

PRISM - OTHER ADJUSTMENTS OR PECULIARITIES

- 1) PRISM USES THE COMMULATIVE AVERAGE LEARNING CURVE. (90%).
VERSUS THE UNIT LEARNING CURVE TO CALCULATE THE NOAK COSTS.
- 2) THE MODULAR PRISM REACTOR CONTAINS MANY MORE COMPONENTS
THAN THE LARGE POOL PLANT, LSPB. TO OVERCOME THE INHERENT
COMPLEXITY DUE TO THE LARGER NUMBER OF COMPONENTS THE
DESIGNERS ARE RELYING UPON FACTORY FABRICATION.
- 3) PRISM EXPECTS THERE WILL BE APPROXIMATELY 35,000 GAL/MONTH
OF LIQUID WASTE, 5,400 CF/MONTH OF COVER GAS, AND 420
GAL/MONTH OF SOLID SPENT RESIN TO BE PROCESSED.

CONTINGENCY

GE'S PRISM HAS A CONTINGENCY ALLOWANCE OF 7%

THE DOE OFFICE OF PROJECT AND FACILITIES MANAGEMENT RECOMMENDS THE FOLLOWING CONTINGENCY PERCENTAGES BASED ON THE COMPLETENESS OF THE DESIGN: PLANNING, 20 TO 30%; CONCEPTUAL, 15 TO 25%; DETAILED, 5 TO 15%. IT IS ALSO RECOMMENDED THAT AN ADDITIONAL 10 TO 15% BE ADDED TO THE ABOVE VALUES IF THE WORK IS OF A HIGH QUALITY LEVEL (E.G. NUCLEAR GRADE). THUS A PROJECT AT THIS STAGE SHOULD BE USING VALUES OF 40% OR 50% FOR COSTING PURPOSES DUE TO THE UNCERTAINTIES INVOLVED IN BRINGING THE RESPECTIVE PROGRAMS TO FRUITION. HOWEVER, THE REVIEW BOARD DID NOT BELIEVE THAT USING SUCH A HIGH VALUE WOULD PERMIT A MEANINGFUL COMPARISON OF THE INNOVATIVE DESIGNS WITH OTHER REACTOR CONCEPTS. THE BOARD SUGGESTED THAT THE 15% VALUE USED FOR THE EEOB LWR ESTIMATE BE THE CONTINGENCY VALUE USED FOR ALL THE LMR PLANTS AS WELL.

4.3.2⁴ The Modular High Temperature Gas-Cooled Reactor

The modular HTGR reactor design is an attempt to develop a gas-cooled reactor using graphite moderator and helium coolant that captures the main safety and operational advantages of the larger and older HTGR designs along with the lower construction costs and higher reliability of a smaller core. The modular HTGR is not a breeder reactor, and hence cannot compete with fusion in the indefinite future without a fuel supply like the breeder's. However, it has been selected for this study because it provides an attractive 'client reactor' for linkage to the fusion-fission hybrid which could be the source of its fissile fuel.

As currently configured by its development team led by GA Technologies under DOE sponsorship, the modular HTGR design has a 350 MWt (135 MWe) core, four of which are linked up to produce a power station module of 560 MWe. The design uses 20%-enriched uranium fuel, or mixed-oxide uranium-plutonium fuel of comparable fissile value. The helium coolant is linked through a heat exchanger to a water-steam turbine generator. Each core is an assemblage of prismatic graphite blocks, with a core annulus surrounded by graphite reflector elements and a central unfueled region. Safety is assured by a separate cooling heat exchanger. The design's intrinsic safety, compared to the LWR, derives from the much longer time constants after loss of coolant and from simpler means of restoring cooling. The small size allows this design to avoid excessive fuel temperatures even after a loss of coolant by heat transfer to the dedicated cavity cooling system.

Economics of construction are much reduced from the larger reactor designs by the small size, shop fabrication, elimination of safety-grade certification for much of the balance of plant, simpler on-site construction and installation, and shorter time schedule for construction. Operating economics are enhanced over the LWR by better fuel economy, shorter refueling outages, and simpler design parameters making operational transients less frequent and less severe.

While a few technical issues still require resolution, this design is much further along than the small-breeder designs such as PRISM and SAFR since it is based on actual operating experience for much of its design and operational input information.

As a client for the fusion-fission hybrid, this reactor is attractive because of its higher burnup efficiency; however, it is fair to point out that any of the fission breeder designs could also serve as a hybrid client.

DRAFT

Table . Fission Reactor Capital Costs
(Millions of 1986 Dollars)

Account	1139 MWe			
	PWR-BE	LSPB ^d	PRISM	MHTGR
20 Land	5.0	5.0	5.0	5.0
21 Structures & Improvements	192.5	261.5	152.0	134.6
22 Reactor Plant Equip.	293.8	634.0	843.7 ^b	264.2
23 Turbine Plant Equip.	215.3	220.2	212.9	123.4
24 Electric Plant Equip.	73.8	84.6	82.8	51.0
25 Misc. Plant Equip.	40.3	55.0	42.9	14.3
26 Main Cond. Ht. Rej. Sys.	45.9	49.1	43.0	20.4
Total Direct Cost	866.6	1309.4	1382.3	612.9
Ind. Costs (37.5%)	325.0	491.0	293.0 ^c	188.7 ^d
Contingency (15%)	178.7	270.1	251.3	120.2
Total Overnight	1370	2071	1927	922
AFUDC ^e	118	178	66	79
Total Capitalized Cost	1488	2249	2093	1001
\$/KWe	1306	1704	1681	1794

^aReported costs reduced 11% for nth of a kind plant

^bReported PRISM reactor plant equip. costs increased by 30%

^c21.2% of total direct costs per PRISM report

^d30.8% of total direct costs per MHTGR report

^eSame 6 year construction schedule assumed for all concepts

PWR-BE - best experience PWR

LSPB - EPRI/C

DRAFT

Power Generation Costs (1986 dollar mills/kWh)

Plant Type	MWe	Capital	O&M	Fuel	Decom ^a	Total
PWR-BE	1139	19.4	7.5	7.5	.5	34.9
LSPB	1320	25.3	7.0	9.0 ^b	.6	41.9
PRISM	1245	24.9	8.3	10.6 ^b	.7	44.5
MHTGR	558	26.6	11.7	11.1	.7	50.1

---

^a15% of direct capital costs accumulated in sinking fund at 8.5% (2.36% real) return over 30 year plant life

^bPreliminary assuming \$50/gram plutonium price

2.5 Methods to Treat Economic Uncertainties Related to Blanket Selection

Because the efficiency, cost, safety, reliability, maintainability, and environmental impact of the fusion energy system are affected most by the selection of a blanket technology, a selection among candidate fusion blankets will be key to developing an attractive fusion energy product. Nevertheless, at the present time it is very difficult to correctly model all of the plant cost implications and cost uncertainties related to these variations. Recognizing this difficulty, a goal of the ESECOM activity has been to identify and tabulate those blanket related differences which can lead to significant uncertainties in the plant cost or performance - especially those differences which are not treated in the economics modeling.

As a first step towards developing an economics perspective relating to such differences, we assume that all such uncertainties can be resolved with a potential cost penalty. Given this assumption, a discussion of the potential cost uncertainties related to blanket technology choices will be provided in Section 5.1. The intent is to familiarize the reader with those issues for which blanket design differences may imply large overall cost differences (eg., safety systems for liquid

metal coolants) and/or those issues for which a design solution (eg., limiting routine tritium releases to allowed levels) may be highly uncertain. Similarly, some blankets may offer safety/environmental and/or performance advantages which translate to significant economic benefits in the overall plant context.

Some of the blanket choice implications can be taken into account in the Generomak model. For example, the blanket energy generation, thermal efficiency, blanket thickness, material costs, fluence lifetimes, the breeding ratio (for fusion breeders) and some power conversion system and primary containment building cost differences have been modeled in the development of the capital and operating cost estimates presented in Section 5.1. Others, such as remote maintenance, have been assigned a common cost target (ie, no difference in target cost among blanket types). Still others are identified as not having been addressed in the economics models.

For cost areas (ie., some reactor sub-systems) which have not been explicitly covered (due to lack of resources within our activity, a non-existent data base, or high uncertainty and/or controversy within the fusion reactor design community), a preliminary evaluation of the potential impact was developed, scored, and presented in Section 5.1. The following levels of impact were used.

- ° HIGH IMPACT (H): Greater than 20% of plant direct cost (nominally 400 \$M for 1000 MWe Plant)

- ° MODERATE IMPACT (M): Greater than 10% of plant direct cost (nominally 200 \$M for 1000 MW_e Plant)
- ° LOW IMPACT (L): Less than 5 % of plant direct cost (nominally 100 \$M for 1000 MW_e Plant)
- ° NOT APPLICABLE (N)

The potential impact is interpreted as a potential additional direct cost penalty relative to the cases which were modeled.

In this activity, the following nominally defined availability, safety, environmental goals are assumed to be satisfied for each blanket type:

- ° Availability - Blanket availability consistent with 75% overall plant availability (eg., blanket system availability greater than 90%).
- ° Safety - Risk contribution which is negligible compared with overall total risk exposure (see Section 3).
- ° Environmental Impact - No significant hazard due to routine release of plant effluents (see Section 3).

These are consistent with the overall ESECOM activity.

2.6 Economic Aspects of Fusion Breeding

2.6.1 Motivation

Among the energy sources which can be considered inexhaustible in a practical sense are solar energy, nuclear fusion energy, and nuclear fission energy with a technology to convert abundant resources of naturally occurring fertile isotopes to fissile isotopes. The fast-fission breeder (e.g., the LMFBR) represents one such breeding technology. A more advanced technology, D-T fusion, could provide a uniquely prolific source of high energy neutrons. More specifically, in addition to producing heat and the required tritium, fusion neutrons could be used to produce fissile materials by the transmutation of abundant fertile materials such as ^{238}U and ^{232}Th .

The neutron rich "fusion breeder" would be a subcritical assembly which would produce an order of magnitude more net excess fuel per unit of thermal power and would not be subject to the criticality and control constraints of conventional fission reactors. In general, the fusion breeders could produce either ^{233}U or ^{239}Pu using metal, oxide, carbide, molten salt or any other fertile fuel form.

The ratio of fissile material to energy production in the fusion breeder can be widely varied according to the type of blanket design. When optimized to emphasize fissile production, the fusion breeder could replace the uranium mining and enrichment segments of the fission fuel cycle and could enable

the continued use of fission converter power reactors for the foreseeable future. Depending upon the type of fission reactor client, fusion breeders could provide fissile fuel to support over 15 times as much fission power as they themselves produce. As shown in Figure 2.6-1, relatively few fusion breeders would be needed to support a nuclear fission capacity which is roughly equivalent to that existing in the U.S. today.

By producing fuel for fission convertor reactors, the fusion breeder can provide important institutional advantages. These result from its primary role as a fissile fuel (rather than power) producer. Fusion breeders and their associated fuel reprocessing facilities could be located together in remote, safeguarded fuel cycle centers and could be functionally equivalent to the existing combination of uranium mines (mills, etc.) and enrichment plants. Therefore, the current institutional framework, consisting of government ownership, control and/or operation of enrichment and reprocessing plants, might be appropriate.

The fusion breeder could accelerate the commercial development of fusion power plants because lower fusion performance and higher fusion component costs than would be acceptable for fusion-electric power plants do not cause the fusion breeder option to become economically unattractive. Indeed, because only a small fraction of the total electricity generation is provided by the fusion breeder, the system electricity cost is relatively insensitive to the cost of the fusion breeder (see Section 5 economics results). Thus the early commercialization of fusion breeders could give the industrial sector experience relevant to the development, operation, and improvement of nearly all fusion component technologies.

2.6.2 System Figures-of-Merit

Three system figures-of-merit are most important in characterizing the performance and economics of fusion-fission electricity generation systems. These are: (1) the nuclear support ratio, (2) the system electricity cost, and the (3) breakeven price of U_3O_8 (i.e., natural uranium). These are briefly defined below.

The nuclear support ratio, R , is defined as the nuclear power of the client fission reactors divided by the nuclear power of the fusion breeder (i.e., fusion power $\times (0.8M + 0.2)$ where M is the blanket energy per source neutron divided by 14.1 MeV). When the support ratio is high, the system electricity cost will be relatively insensitive to uncertainties in the performance and cost of the fusion system and will approach that of the client fission reactor which dominates the electricity generation system. A high support ratio can be economically advantageous if the cost of the client fission reactors can be low compared with the alternatives (eg., coal, fusion-electric, fast-fission breeder).

An important advantage of high support ratio breeders is an increased potential for generation growth. Because a large fraction of the existing nuclear electric grid would be fueled from a small amount of fusion capacity, a few fusion breeders with high support ratios can have larger commercial impacts at earlier dates.

The nuclear support ratio can be put in perspective by considering LWRs fueled by fast-fission breeders (eg., LMFBR). Fission breeders, which are optimized for high breeding (with

associated cost penalties), can produce enough excess fissile fuel to support, at most, 1 LWR of equivalent nuclear power. Even so, the fission breeder must also produce fissile fuel to satisfy the fissile inventory requirement of additional fission breeders. Consequently, LWR support is not a likely mode of fission breeder operation.

System electricity cost is another important figure of merit for a fusion breeder-fission burner system because the only net product of a system consisting of a fusion breeder and its fission clients will be electricity. The estimated system electricity cost is calculated by dividing the total annual cost to operate the system (fusion breeder plus fission clients including fuel cycle costs) by the total electricity production. This cost is most useful when compared to a similar calculation of the cost of electricity produced by an alternative electricity generation candidate such as a coal-fired electricity plant, a fusion-electric generation plant, a current technology LWR fueled with ^{235}U derived from conventional mined uranium, or a fast-fission breeder.

Once the system electricity cost is known, it is possible to determine the bred fuel cost by subtracting the fusion breeder's electricity sales revenues from its operating expense. However, the bred fuel (eg., ^{233}U) is an intermediate product whose cost may be difficult to relate to external market conditions. More importantly, the system electricity cost can be expressed in terms of a breakeven U_3O_8 price. This is the market price for which an LWR, using mined U_3O_8 , would generate electricity at the same levelized cost over its lifetime as the fusion breeder-

fission burner system. Once the breakeven U_3O_8 price is estimated, it can be usefully compared with the breakeven U_3O_8 price for advanced fission reactors, thus providing a consistent comparison. The breakeven U_3O_8 price can also be compared with the expected (predicted) price of mined U_3O_8 to estimate the breakeven year for fusion breeder introduction.

2.6.3 Conditions for Fusion Breeder Viability

Clearly, the need for a fusion breeder is predicated upon the assumption that the nuclear power industry will regain its former vitality to the extent that the fission fuel cycle is closed and future plant capacity (and/or the expectation of such capacity) exerts an upward pressure on the cost of mined uranium. Although the outlook for such a scenario appears bleak from today's perspective, one or more of the following circumstances could motivate a resurgence of fission power and/or enable the fusion breeder to become economically attractive in the 2020 timeframe:

- o An improved framework for the licensing, financing, and operation of fission plants in the U.S.
- o Higher economic and electricity demand growth than experienced during recent years.
- o Economic advantages in Europe (e.g., France) and Japan which can be attributed to less expensive nuclear power generation from plants built during the 80's and 90's.
- o A new real or perceived shortage of fossil fuels.
- o The introduction and public acceptance of an inherently safe fission convertor reactor (e.g., the modular HTGR and/or novel new LWR designs).
- o Continued concern relative to CO_2 , acid rain, and other

adverse environmental conditions resulting from fossil electricity generation.

- o Coupling in the price of uranium as the price of oil, coal and other fuels rise (e.g., the 1973 Arab Oil Embargo experience).

4.2 Fission-Suppressed Fusion Breeder (Hybrid)

The possibility of placing a fissioning blanket around a fusion neutron source to enhance its energy generation and to produce fissile fuel has been recognized since the early beginnings of fusion research. Such devices, termed fusion-fission hybrid reactors, or fusion breeders can be optimized to maximize either the deposited energy or the fissile fuel production (see Section 2.6) per fusion neutron. During the past several years, the fission-suppressed fusion breeder has received the greatest attention in the U.S. because it minimizes the insitu fission rate (thus providing reactor safety advantages) and maximizes the fissile breeding rate, thus maximizing the number of client fission reactors (eg., LWR or HTGR) per fusion reactor. In comparison with fast-fission hybrids, which had been the primary focus of hybrid reactor research through the mid-70's, but were not considered by ESECOM, the fission-suppressed fusion breeder can reduce fissioning by an order of magnitude while at the same time increasing the number of fission clients severel-fold. Institutional advantages can result because a high support ratio fusion breeder would fuel many client fission reactors while requiring only a few fusion breeders which can be remotely sited, if desirable.

The design of a fission-suppressed blanket was first addressed in a 1979 study performed by Lawrence Livermore National Laboratory (LLNL)[4.2.1] As a result of this study, a fission-suppressed blanket concept based upon helium cooling,

beryllium neutron multiplication and a lithium/thorium bearing molten salt coolant was developed. This design has evolved over the past several years and its most recent version [4.2.2] was selected for this study.

A second design, based upon lithium cooling, thorium metal fuel, and beryllium neutron multiplication has also been extensively studied [4.2.3,4.2.4]. The most recent version [4.2.5] of the liquid metal design was also selected for this study. Both blankets are described in the following subsections. The reader is referred to Table 4-1 for a list of blanket parameters for the two fusion breeder blankets. System level parameters are provided in Table 4.2-1.

Table 4.2-1 Fusion Breeder Parameters

	Lithium Cooled <u>Blanket</u>	Helium Cooled <u>Blanket</u>
Blanket Description		
Coolant	Lithium ^a	Helium
Multiplier	Beryllium	Beryllium
Tritium Breeder	Lithium	Molten Salt ^b
Fissile Breeder	Thorium	Molten Salt ^b
Structure	HT-9	PCA
Blanket Performance		

Maximum Energy Multiplicaiton	3.17	2.0 ^c
Ave. Energy Multipliatiion	2.44	2.0 ^c
Ave. Fissile Breeding Ratio	0.70 ^d	0.50 ^d

Typical Fusion Breeder Performance

Fusion Power (MW _f)	2600	2600
Max. Thermal Power (MW _t)	6220	4680
Ave. Thermal Power (MW _t)	5595	4680
Ave. Net Elec. Power (MW _e)	1875 ^e	1715 ^e
Net Fissile Prod. (kg/yr)	5910	4220

Typical System Performance (LWR Clients)

Number of 1 GWe LWRs Supported	20.8 ^f	14.9 ^f
Total Electrical Power (MWe)	22.7	16.6
Percent Elec. Power from	8.3	10.3

Fusion Breeders (%)

- a) Depleted to 3% ⁶Li
- b) 70 LiF - 12BF² - 18THF₄
- c) Constant over cycle
- d) Conservative values anticipating neutron leakage in tokamak configuration
- e) assumes 250 MW_e recirculating power, 38% gross thermal-to-electric efficiency for liquid metal, 42% for helium
- f) 75% average capacity factor

4.2.3 Liquid Metal Cooled Fusion Breeder Blanket Summary Background

A reference liquid metal (lithium) cooled thorium metal fueled fusion breeder blanket design for the tandem mirror was developed during 1982.[4.2.3] Special issues associated with this design were investigated during 1983-1984.[4.2.4] An updated reference design was developed for the 1985 US/USSR Technical Exchange on Fusion-Fission Hybrids. The updated design is described in reference [4.2.5]. The design, shown in Figure 4.2-1, is in many ways similar to liquid metal cooled fusion-electric blanket designs which have been developed during the same timeframe.[4.2.6]

The liquid metal cooled blanket is of interest because:

- o It offers very high net ^{233}U breeding (1.7 Kg/MW_t-full power year),
- o It is composed of reasonably conventional materials (ferretic steel structure, thorium metal fuel),
- o It provides excellent tritium retention,
- o It utilizes a lithium coolant technology which is similar to the LMFBR sodium coolant technology and is currently under development for space power applications (e.g., SP-100),
- o It results in a less complex internal structure than the principle alternative coolants (helium, water).

This design was the first to feature a mobile fuel element in the form of a beryllium/thorium composite pebble. The inner zone pebbles would be circulated through the blanket at a regular interval of about 0.9 full power year (FPY). This circulation

would provide an economically recoverable ^{233}U enrichment of about 1%, while also limiting the blanket module power swing ($\pm 30\%$ over one module, or $\pm 12\%$ over the entire set of modules via the use of fuel management techniques). The design provides for dumping of the entire blanket contents to a passively cooled dump tank should the need arise. This is viewed as a potentially important safety advantage, which was not achievable in earlier fast-fission designs with fixed, conventional, fuel elements.

Key issues which have been identified for the updated reference design fall in the following areas:

- o MHD Pressure Drops
- o Liquid Metal Compatibility
- o Beryllium Irradiation Damage
- o Fuel Reprocessing
- o Pebble Flow
- o Lithium Safety

All of the above issues, with the exception of pyro-chemical reprocessing and beryllium pebble flow must also be addressed (although with slightly different emphasis) in the development of liquid metal cooled fusion-electric blankets.

An advantage of fusion breeder blankets relative to fusion-electric blankets is that substantial relaxations can be obtained due to lower neutron wall loads (1.7 MW/m^2 in the subject design versus about 5 MW/m^2 in recent fusion-electric designs) and lower maximum coolant operating temperatures (425°C versus about 500°C). After 5 FPY of operation, the fusion breeder end-of-life fluence is only 8.5 MW-yr/m^2 , a value exceeded by the fusion-electric blanket in less than 2 FPY.

Importantly, it is believed that a considerable level of confidence in this technology can be obtained prior to the year 2000, without the use of an expensive fusion test platform. [4.2.7] Specifically, it is expected that nearly all nuclear irradiation related aspects of the design can be demonstrated in an integrated, in-core fission loop (10 cm tube in high flux region of a test reactor) containing a lithium coolant and the beryllium/thorium pebbles. Similarly, the MHD flow related aspects of the design can be demonstrated in a prototypical, but non-nuclear blanket mockup constructed from the correct materials. Based upon this results of models derived from previous irradiations, this end-of-life level of displacement damage (about 125 dpa) is believed to be below the swelling threshold in ferritic steels. Thus a high, level of confidence regarding materials performance can be assumed.

4.2.2 Helium Cooled Fusion Breeder Blanket Summary

Recognizing that liquid metal cooled blankets might not be optimal for use in fusion hybrids (excessive MHD pressure drop, chemical reactivity), helium cooled blanket variants have been developed since the mid-70's[4.2.8]. These designs are, in many ways, similar to helium cooled fusion-electric blanket designs which have been developed during the same timeframe[4.2.6].

Helium cooled blankets are of interest because:

- o Helium does not interact with the magnetic field,
- o Helium is chemically and neutronically inert,
- o A properly constructed helium system can lose coolant pressure, but preserve flow (at about 1 atm),

- o Helium blankets are equally applicable to all fusion confinement schemes (tokamak or mirror).

Helium cooled blanket designs for the fusion breeder utilize the same type of beryllium pebble circulation scheme as the liquid metal cooled design and are expected to provide the same safety advantages. Because the tritium breeding capability of the lithium coolant is eliminated in these designs, a separate lithium-bearing material (e.g., liquid lithium, a molten lithium fluoride-beryllium fluoride salt, lithium oxide, lithium aluminate, etc.) must be introduced into the blanket. In addition, a separate tritium breeder flow or purge circuit is required.

An interesting example of this type of design occurs when a thorium bearing molten salt ($\text{LiF-BeF}_2\text{-ThF}_4$) is used as the lithium-bearing material. In this case, tritium breeding issues are largely unaffected, but fissile breeding in the circulating fluid can be obtained and the thorium metal clippings on the beryllium pebbles can be removed. Designs based upon the use of helium cooling and molten salt breeding have evolved since the late 70's[4.2.9] and have evolved towards a blanket design concept similar to the 1983-84 blanket design shown in Figure 4.2-2[4.2.2]. Although the above design was developed for use with a tandem mirror fusion configuration, it could work equally well on a tokamak, provided that the neutron wall load is moderate (about 3 MW/m^2).

In addition to the above advantages of using helium, the molten salt breeder provides the following important advantages:

- o on-line tritium and ^{233}U removal, .
- o breed fuel removal by low cost fluorination process,

- o continuous processing to prevent power swing and thereby increase blanket safety (lower afterheat) and maximize the support ratio (number of equal power fission reactors that can be refueled),
- o elimination of thorium fuel element fabrication,
- o use of beryllium in the form of small spheres (1 cm dia.) which are more durable to radiation damage than larger parts,
- o low tritium inventory due to low solubility of tritium in molten salt,
- o low chemical reactivity of molten salt, water and helium.

Some issues which have been identified for the use of helium coolants and/or molten salts in fusion breeder blanket fall in the following areas:

- o First Wall/Blanket Complexity
- o Tritium Containment
- o Molten Salt/Steel Compatibility
- o Molten Salt Draining and Filling

Other issues for this blanket type are similar to those for the liquid metal cooled blanket described in the previous section. Again, it is believed that the above issues (with the possible exception of the first wall/blanket failure modes and lifetime) can be resolved via fission reactor testing prior to the year 2000 [4.2.7]

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5.1.4 Economic Impact of Blanket Choices

As outlined in Section 2.5, the choice of a blanket technology has a potentially large impact upon the overall economic performance (as well as the environment/ safety characteristics) of a fusion power plant or fusion breeder reactor. Some blanket characteristics, such as the

- ° blanket thermal energy generation,
- ° thermal efficiency,
- ° thickness,
- ° fabricated materials cost,
- ° fluence lifetime, and the
- ° breeding ratio (economic importance for fusion breeder, see Section 2.6)

have been modeled using the Generomak methodology and are presented in the above sections. Other blanket associated costs, such as the primary loop and power conversion system cost and containment building cost, have also been modeled but are more uncertain, lacking design detail. Finally there are cost areas (eg., remote maintenance systems, tritium control systems), for which the requirements and/or design details are so lacking (or controversial) that the cost estimates listed in the code of accounts are, at best, target values.

In this section we first discuss the economic impacts which can derive from differences among blankets. This discussion is followed by a table describing the potential economic impacts of various blankets which have not been fully addressed in the Generomak cost modeling activity.

The costs of containment buildings for fusion plants are somewhat better defined, but fall into the same category. The costs of remote maintenance systems (above) have been considered in previous studies, but are less defined than those for primary loop and power conversion systems.

Finally, for areas such as tritium processing/control from Pb-Li, FLIBE, and some helium cooled solid breeders, the requirements data needed to size plant systems does not exist due to legitimate differences in opinion in the technical community (eg., with regard to permeation rates, tritium oxidation rates, coating performance, etc.).

Potential Economic Impacts of Poorly Modeled Issues/Systems

As discussed earlier (Section 2.5), in this section we make the apriori assumption that any as yet unresolved blanket related issue that impacts blanket availability, reactor safety, or releases of radio-nuclides (or other toxic materials) to the environment can be resolved, at a cost, within acceptable guidelines. These guidelines, repeated below are consistent with the overall ESECOM activity:

- o Availability - Blanket availability consistent with 75% overall plant availability (eg., blanket system availability greater than 90%).
- o Safety - Risk contribution which is negligible compared with overall total risk exposure (see Section 3).
- o Environmental Impact - No significant hazard due to routine release of plant effluents (see Section 3).

The potential cost impacts of issues and blanket characteristics that, in our opinion, have not been adequately covered in our Generomak modeling activity, are provided in Table 5.1.2-1. These potential impacts have been characterized as follows:

Table 6.1.2-1 Potential Cost Impacts Beyond Economic Modeling

Potential Cost Area	Base Case Li/Li/V	Helium Cooled Li ₂ O/He/ HT-9	High Power Pb-Li/Pb-Li /H ₂ O	Low Activity Li ₂ O/He/ SiC/BeO	FLIBE Pool FLIBE/FLIBE /V	Adv. MHD FLIBE/Cd /V/Be	Adv. Fuel NA/H ₂ O/Al	Li Hybrid Li/Li/HT-9 /Be/Th	MS Hybrid MS/He/PCA /Be/MS
Coolant Safety System	M	NA	L	NA	NA	?	NA	M	NA
Coolant (Breeder) Storage/Trace Heating	M	NA	M	NA	L	M	NA	M	M
Remote Maintenance Systems	H	M	H	L	L	M	M	H	H
Seismic Restraints	M	M	M	NA	L	L	L	M	M
Tritium Processing/Containment	L	M	M	M	M	M	NA	L	M
Beryllium/Solid Breeder Recycle	NA	L	NA	L	NA	NA	NA	L	L

Key: T Breeder/Coolant/Structure/Multiplier/Fissile Breeder

- ° HIGH IMPACT (H): Greater than 20% of plant direct cost (nominally 400 \$M for 1000 MW_e Plant).
- ° MODERATE IMPACT(M): Greater than 10% of plant direct cost (nominally 200 \$M for 1000 MW_e Plant)
- ° LOW IMPACT (L): Less than 5 % of plant direct cost (nominally) 100 \$M for 1000 MW_e Plant)
- ° NOT APPLICABLE (N)

In this context, it should be noted that these are potential (rather than actual) impacts - we don't know the actual impacts. Also, the designations H,M,L should be viewed as being broad categories, developed without the benefit of extensive design/costing guidance.

The first potential cost area in the table is coolant safety. For reactive liquid metal coolants, this area involves the potential cost of some combination of :

- o tankage, sumps, steel guard vessels, concrete liners, and sacrificial buffer materials (between the steel and concrete)
- o high heat capacity materials (eg., steel balls) to cool a liquid metal spill
- o active chemical fire fighting equipment
- o remote clean up equipment and instrumentation

all of which have found some use in liquid sodium systems for advanced fission reactors. In this category, lithium cooled blankets are scored as having a potentially moderate (M) impact and the Pb-Li coolant is scored less severely (L) than the lithium coolant because Pb-Li is a less reactive material.

The second potential cost area is coolant (breeder) storage/trace heating. In this case the liquid metal and FLIBE systems are scored as having a potentially moderate impact (M) because of a need to maintain liquid materials above their melting points (for example, by pre-heating the blanket with

electrical resistance heating and/or a inert gas preheat loop).

The FLIBE pool is rated as a lower potential concern (L) than the other cases due to its compact configuration and relatively low surface/volume ratio.

The third potential cost area is remote maintainance. In this case, the liquid metal loops were most penalized and are scored as having a potentially high impact (H). It should be noted that these systems (especially for Pb-Li coolant) may suffer the greatest amount of radioactive mass transfer within the primary loop. The fusion breeder systems were also scored as having a potentially high (H) impact due to the additional in-core radioactive inventory and possibility for mass transfer of gamma emitting radio-nuclides such as ^{228}Th (2.6 MeV gamma). The low activity blanket and the FLIBE pool systems score best (L) in this category.

The fourth potential cost area is seismic restraints. In this area we have, again, penalized the liquid metal and fusion breeder blankets by scoring them as having a moderate (M) potential impact. It should be noted that Pb-Li, with a density of about 10 g/cm^3 , is much heavier than the other coolants and will provide uniquely difficult requirements in the seismic area.

The fifth area is tritium processing/control. In this area, blankets with low tritium solubility and/or helium coolant are most penalized due to the expectation of difficult tritium removal requirements in the primary loop and other parts of the plant and are scored as having moderate (M) potential impact. Even if the tritium in helium cooled blankets (eg., helium cooled Li_2O) can be prevented from entering the primary loop (by using high efficiency permeation barriers), the cost of systems to actively and continuously purge tritium from the blanket internals can be expected to be high and has not been adequately covered in previous studies. Tritium control issues for lithium will be minimal (L).

The final potential cost area is beryllium/solid breeder recycle. The cost of such recycle will depend upon the amount of impurities in the material because neither pure beryllium or Li_2O become activated (although insitu reactions in beryllium do produce about 0.03 tritons per fusion in thick beryllium assemblies). For beryllium, it appears that hands on or glovebox re-fabrication will not be realistic. For Li_2O , hands-on refrabrication may be more feasible. All blankets containing beryllium or solid breeders were scored as having a low (L), but non-negligable potential impact.

6.2.X POTENTIAL OF FUSION-FISSION HYBRIDS

6.2.X.1 General Conclusions

Only a few energy sources can be considered inexhaustible in a practical sense. These include solar energy, nuclear fusion energy, and nuclear fission energy with a technology to convert abundant resources of naturally occurring fertile isotopes to fissile isotopes. The fast-fission breeder (e.g., the LMFBR) represents one such breeding technology. A more advanced technology, D-T fusion, could provide a uniquely prolific source of high energy neutrons so that fusion neutrons could be used to produce fissile materials by the transmutation of abundant fertile materials such as ^{238}U and ^{232}Th .

The neutron rich fusion-fission hybrid, or "fusion breeder", would be a subcritical assembly which would produce an order of magnitude more net excess fuel per unit of thermal power than fission breeders. A wide variety of fuel cycles and fuel forms would be possible. In general, fusion breeders could produce either ^{233}U or ^{239}Pu using metal, oxide, carbide, molten salt or any other fertile fuel form.

When optimized to emphasize fissile fuel production, fusion breeders could provide fissile fuel to support over 15 times as much fission power as they themselves produce. Therefore, as shown in Figure 2.6-1, relatively few fusion breeders would be needed to support a large nuclear fission capacity. Fusion breeders and their associated fuel reprocessing facilities would be located together in a few, remote, safeguarded, fuel cycle centers and would be functionally equivalent to the existing combination of uranium mines (mills, etc.) and enrichment plants.

The environmental, safety, and economic characteristics of the fusion breeder option must be considered in the overall context of a fusion-fission electricity generation system dominated by its many fission client reactors. In this context,

we observe that the "goodness" of the entire system can not be substantially different from the "goodness" of the fission clients unless the fusion breeder has extremely poor characteristics.

In summary, window for fusion-fission electricity generation viability as the preferred intermediate-to-long range option can, therefore, exist if the following three conditions are satisfied:

- o the fission client reactors and their associated fuel cycle activities (including fuel reprocessing) must be acceptable with respect to their safety and environmental characteristics.
- o the safety and environmental characteristics of the fusion breeder (and its fuel cycle) must not cause the symbiotic system to suffer intolerably in these areas.
- o the cost of symbiotic electricity must be significantly less than that of fusion-electric power and fission-electric power.

Regarding the third condition, it is noted that, for a 1000 GWe electric generation capacity (conservative for U.S. in 2020 timeframe), even a 5 mil/KWeH cost advantage translates to a gross national benefit of over 30 billion/year. Thus, small COE differences can be quite significant.

We believe that, although the fusion breeder does not represent the highest safety/environmental potential of fusion, the possibility of meeting the above three conditions is real and provides a large enough potential benefit that this option should be supported in future R&D activities. Because attractive fusion breeding performance can be provided by less advanced fusion systems (such as the point-of-departure tokamak), this R&D activity should be primarily focused upon fusion breeder blanket, safety system, and fuel cycle issues. We further note that an early commercialization of the fusion breeder need not foreclose the long range fusion-electric option. Specifically, an early

introduction of the fusion breeder could give the industrial sector experience relevant to the development, operation, improvement and cost reduction of nearly all fusion component technologies.

6.2.X.2 Specific Conclusions

Some specific ESECOM conclusions relative to the environmental, safety, and economic characteristics of fusion-breeders and fusion-fission electricity generation systems are provided below.

Economics

The economics results presented in Section 5.1 indicate that a fusion breeder costing 2.1 times as much as the "best of current experience PWR" (PWR/BE) might provide fissile fuel to that client with a resulting cost of electricity (COE) of 39.0 mil/KWeH. This cost is only 5.6 mil/KWeH higher than the COE estimate for the PWR/BE operating on the current once-through fuel cycle (no reprocessing or radioactive fuel fabrication) with uranium costing 50\$/lb.

Among the ESECOM case studies, the fusion systems that have potential to compete with the above fusion-fission cost of electricity are:

- o A gas cooled base case tokamak (38-50 mil/KWeH) with extensive inherent safety credits (lower COE range).
- o A high power density Li/V RFQ (31-42 mil/KWeH)
- o A high power density Pb-Li RFQ (29-37 mil/KWeH).
- o An advanced MHD power conversion cycle tokamak (27-35 mil/KWeH), and
- o An advanced fuel tokamak (35-48 mil/KWeH) with inherent safety credits.

The above fusion-electric cases are representative of an encouragingly high potential for fusion economics. Nevertheless,

these cases all benefit from assumptions that require successful physics and/or engineering extrapolation beyond the base case point-of-departure tokamak, whose performance already exceeds that of STARFIRE and several other recent studies.

For example, all of the RFP cases benefit from more favorable, but less demonstrated, plasma scaling than the point-of-departure tokamak. The RFP cases also benefit from higher mass utilization, but are required to operate at 5-fold higher neutron wall loadings. Both of the RFP cases have nominal inherent safety levels of 4 and are not likely to benefit from inherent safety credits.

Similarly, the gas cooled point-of-departure tokamak and the advanced energy conversion tokamak are both nominal level 3 safety, and are likely to receive only part of the potential credit for inherent safety. The advanced energy conversion and advanced fuel tokamaks also benefit from more favorable physics scaling and engineering assumptions than the point-of-departure tokamak.

In contrast, the fusion breeder plasma physics and engineering parameters are nearly identical to those of the point-of-departure tokamak and can be substantially degraded without incurring a significant COE penalty. The stability (or robustness) of the cost of fusion-fission system electricity to the large uncertainties in the capital cost and/or performance of the fusion reactor is shown in Table 6.2.X-1. A 2-fold increase in the fusion breeder cost (assuming that the client fission reactor cost is fixed) represents only a 13% increase in the cost of system electricity. A similar cost increase for the fusion-electric systems results in electricity cost increases of about 65%. A factor of two overall cost uncertainty might seem to be unrealistic, but is supportable based upon the current level of understanding of fusion reactor physics and technology, the lack of detail in existing capital cost estimates, and the cost history of large fusion experiments.

Finally, it should be noted that the fusion-fission system electricity costs (regardless of breeder and/or client type) are potentially lower than those for all the other fission-based systems including the LSPB (43.2 mil/KWeH) and PRISM (43.8 mil/KWeH). The 4.8 mil/KWeH potential COE difference between PRISM and the fusion breeder/LWR system, integrated over a 1000 GWe system (approaching the middle of the next century) represents a \$29 Million/year savings to the national economy with a present value (5% inflation, 50 years) of \$2.6 billion/year! This benefit could decrease if PWR/BE costs can not be realized in the long term, but could increase if initial advanced fission reactor cost estimates exceed those for the PWR/BE by more than assumed in this study.

Safety Characteristics

As mentioned earlier, the fusion-fission electricity generation system will be dominated by the fission reactor clients. Current fission reactors require active, engineered safety systems, however some advanced fission reactor clients, such as the modular HTGR (MHTGR) can be designed to passively survive thermal transients, as long as the geometry stays intact (level of safety assurance = 3).

A fusion-fission electricity generation system based upon the MHTGR would result in an estimated 1 mil/KWeH cost penalty relative to the use of an LWR client, however this investment in passive safety might be worthwhile in the long term. Indeed, it should be noted that advanced converter reactors for electricity or process heat applications (eg., hydrogen production) will not be economically sustainable in the long run without an external source of fuel such as the fusion breeder. Thus, the fusion breeder can open a new range of long term possibilities for fission energy.

To preserve the attractive safety characteristics of the MHTGR, the fusion breeder should provide safety characteristics which are, roughly comparable. The ESECOM results in this regard

are encouraging. We believe that, with the exception of the operator action required to disconnect electrical power from the normally refrigerated dump pipe freeze plug, the lithium-cooled fission-suppressed fusion breeder design would achieve a safety assurance level of 3 (nominally rated level = 4 due to need for operator to de-energize circuit). The fission-suppressed, helium cooled molten salt fusion breeder provides different advantages (lower chemical reactivity and radioactive inventory) and issues (more complex, less favorable heat transfer), but can also achieve a level of passive safety with further design. Ultimately, fusion breeder designs utilizing pool-type geometries (similar to our FLiBe-cooled fusion-electric design concept) might achieve safety assurance levels of 2.

Environmental and Institutional Characteristics

Again, the fusion-fission electricity generation system will be dominated by either light water reactors or advanced converter reactors (eg., modular HTGR, molten salt converter). The environmental characteristics (routine releases of radiation, high level waste disposal) of the fusion breeder part of the system will, with the notable exception of tritium release, be similar to those for fission reactors and will thus preserve the fission-dominated environmental characteristics of the system as a whole.

Regarding tritium, it should be noted that the ability of steady-state tritium releases from fusion reactors can meet current regulatory guidelines (10 Ci/day) without excessive economic penalties is not yet fully resolved. Also, high probability/low consequence tritium releases due to expected operations, maintenance, and minor accident scenarios may lead to larger integrated off-site doses than the very small radiation exposure doses currently encountered in fission power generation. If tritium is difficult to contain, this will be a smaller problem for fusion-fission systems because of fission reactor dominance and location of the fusion breeders in only a few remote sites.

The fusion breeder fuel cycle involves fissile material production, fuel reprocessing and the other technological and institutional elements of a full recycle fission power economy. It is therefore in the same class regarding the potential for weapons material proliferation as fission breeder reactors which have been proposed in the past. Specifically, we note that limiting the proliferation of weapons technologies among national groups and preventing the diversion of weapons materials to sub-national and terrorist groups requires the maintenance of a strong system of institutional safeguards. However, we also note that in time frame of interest, about 40 years hence (75 years after the first practical production of special nuclear materials, there is a strong possibility that breeding, reprocessing, and/or enrichment technologies will be wide spread. Thus, the character of the proliferation issue may be changed in an as yet unknown way.

Assuming that proliferation remains an issue, the fusion breeder can provide fuel cycle options that may be beneficial. Because of its fuel cycle flexibility, it can produce ^{233}U which can be combined at low enrichment with ^{238}U to provide a denatured uranium (DU) fuel that can not be chemically separated and must be enriched to provide weapons grade ^{233}U . This fuel cycle can provide an additional obstacle to diversion (especially if the fresh fuel is spiked with radioactive fission products) and would enable technologically advanced breeder countries, such as the U.S., to continue to provide low enriched fissile fuel for use in burner reactors abroad. In this manner, the burner countries might not feel compelled to acquire their own breeding, reprocessing, and/or enrichment capabilities provided that the supplier could assure the supply of fuel services on a competitive basis. The plutonium produced as a by-product of the DU fuel cycle could be burned within the national breeder country systems.

Table 6.2.X-1
Cost of Electricity as A function of Fusion
Plant Relative Capital Cost

<u>Case</u>	<u>Capital Cost</u>	
	<u>Nominal</u>	<u>Double</u>
Li/V Base Case	52.8	87.7
He Cooled 49.7	85.1	
High Power Density		
Li/V - RFP	41.8	64.8
Pb Li - RFP	37.0	60.7
Low Activation	55.7	97.0
Molten Salt	54.0	86.0
MHD	35.4	54.6
Advanced Fuel	47.6	86.2
Fusion Breeder		
Li/LWR 39.0	44.2	
MS/LWR 38.9	44.0	
MS/MHTGR	39.9	44.0

GAC-FBP 76-039

3 July 1986
DC-86-116

To: Distribution
From: D. Berwald
Subject: Fusion Breeder Radio-nuclide Inventories and Safety

As you know, we have selected two breeder blankets for study as part of ESECOM: (1) a lithium cooled blanket with thorium metal fuel, and (2) a helium cooled blanket with a thorium bearing molten salt fuel. Both use beryllium for neutron multiplication. The molten salt blanket removes ^{233}U at very low concentration (0.21% in thorium versus 1.0% for the lithium blanket) leading to low concentrations of ^{233}U daughters including fission products, ^{232}U , and ^{228}Th . However, the dominant source of activity, ^{233}Pa is similar in both blankets because (1) it is not actively removed in the molten salt loop and (2) its production rate is proportional to the ^{232}Th neutron capture rate, which is always maximized by design. I will discuss each blanket separately.

Lithium Cooled Blanket

As you know, the four major classes of radionuclides are:

- ° tritium
- ° structure
- ° fission products
- ° actinides

Our previous studies indicate that, based upon activity and BHP, but not release fractions, the hazard presented will vary several orders of magnitude between the tritium (lowest) and the actinides (highest).

Tritium

The tritium in the coolant loop can be maintained at about 1 wppm. A 2600 MW_f reactor with a wall load of 2.2 MW/m² and a wall area of 945 m² would have a total (blanket plus primary loop) lithium inventory of about 600 m³ (300 MT), or about 300 grams of dissolved tritium. Including tritium in process, 600 grams is a reasonable total estimate.

Structure

The structural activation for this blanket is being calculated from pancake diagrams I have supplied to Maninger.

Fission Products

As Steve Fetter correctly observes, although possible, the fission product calculation is difficult to do correctly because the fission product spectrum must be weighted over the fissioning isotopes and their fission energies. Assuming that ²³³U thermal fissions dominate, and that the fission product hazard is proportional to the fission power (a reasonable, but crude approximation to the actual situation immediately following a release), it is possible to scale previous results for similar blankets. Specifically, the calculated (above assumptions) fission product inventories for a 1981 blanket design with helium cooling and a thick beryllium multiplier are shown in Table 1 (Table VI-9 of UCID-19327). Because the average fission power in the present blanket (2430 MW) is a factor of 1.73 higher, the RCM estimate of fission product activation is similarly increased to 2.9×10^3 Ci, or 4.7×10^3 Km³(BHP) for a 2600 MW fusion power. The individual isotopic activities are assumed to be distributed as shown in the table (eg., ⁸⁹Sr activity = $1.73 \times 5.7 \times 10^6 = 9.9 \cdot 10^6$ Ci).

Table 1

TABLE VI-9. Major contributors to fission product BHP for the gas-cooled blanket.

Radionuclide	Activity (Ci)		BHP (km^3)		Half-Life
Sr 89	5.7	+06	1.9	+07	54 d
90	4.4	+04	1.5	+06	28 y
91	2.3	+07	2.5	+06	9.7 h
Y 91	6.8	+06	6.8	+06	58 d
Zr 95	6.6	+06	6.6	+06	65 d
Mo 99	2.4	+07	3.4	+06	67 h
I 131	1.1	+07	1.1	+08	8.1 d
133	2.6	+07	6.4	+07	2.1 h
135	2.2	+07	2.2	+07	6.7 h
Cs 134	1.4	+06	3.5	+06	2.3 y
Ba 140	2.0	+07	2.0	+07	13 d
La 140	2.0	+07	5.1	+06	40 h
Ce 144	1.5	+06	7.7	+06	285 d
TOTAL	1.7	+08	2.7	+08	
TOTAL/module	7.0	+06	1.1	+07	
TOTAL/kg yearly fissile fuel production	2.5	+04	4.1	+04	

Actinides

The best calculation of actinide inventories was also performed for the helium cooled, beryllium multiplier blanket. In this case, previous calculations have followed the chain shown in Figure 1 and have used actual one-group ENDL cross sections according to an ORIGEN-like procedure discussed in Section VII.C of UCID-19327. Although the design we are considering has not had the benefit of these calculations (results for 1981 design shown in Table 2), the previous results indicate that only the ^{233}Th and ^{233}Pa inventories are major contributors to the hazard. These can be easily estimated for fuel residence periods which are several times 27 days (the half-life of ^{233}Pa). Specifically, neglecting small amounts of neutron capture (a good and conservative assumption) the inventories of ^{233}Th and ^{233}Pa should scale only as the ^{232}Th (n,g) rate and the fusion power. The scaling factor is $0.844/0.73 \times 2600/3000 = 1.0$, so the ^{233}Th and ^{233}Pa inventories are expected to be 2.0×10^{10} and 1.3×10^{10} Ci, respectively.

Level of Safety

Figure 2 indicates the many levels of redundancy available for afterheat cooling. A general comment is that the dump tank can provide possible cooling if loss of blanket (eg, first wall) integrity occurs. Because fuel dumping might be inhibited by a large scale loss of structural integrity, this blanket is certainly level 2 or higher, and most likely level 3. As John Holdren correctly points out, the hybrid need only be as safe as its fission client and can be remotely sited. In this context, it appears that acceptable safety can be achieved. Some of the details and similarities of internal heating are provided in UCID-20166.

Figure 1

Figure VII.C-1. Production and decay chain for thorium fuel hybrid blankets.

Table 2

TABLE VI-3. Major contributors to actinide BHP for the gas-cooled blanket.

Radionuclide	Activity (Ci)	BHP (km ³)	Half-Life
Th 228	3.6 +01	1.8 +05	1.91 y
232	1.7 +01	1.7 +04	1.4 x 10 ¹⁰ y
233	2.0 +10	5.2 +08	22 m
Pa 231	1.3 +03	3.3 +07	3.3 x 10 ⁴ y
233	1.3 +10	2.1 +09	27 d
U 232	3.0 +03	3.3 +06	72 d
233	3.1 +03	7.8 +05	1.6 x 10 ⁵ y
TOTAL	3.3 +10	2.7 +09	
TOTAL/module	1.4 +09	1.1 +08	
TOTAL/kg yearly fissile fuel production	5.0 +06	4.0 +05	

Fusion Breeders Can Use a Variety of Semi-Passive Heat Removal and Engineering Systems

Figure 2

Helium Cooled Blanket

The helium cooled molten salt blanket uses the same coolant and Be multiplier as the 1981 helium cooled blanket, but benefits from on-line ^{233}U extraction (not ^{233}Pa or fission product extraction), a chemical process described in some detail for the lithium cooled molten salt blanket considered in 1981 (also UCID-19327).

Tritium

The solubility of tritium in lithium fluoride - beryllium fluoride - thorium fluoride salts is very low (similar to FLISE). Consequently, the tritium inventory in the salt and the primary loop is expected to be less than 100 grams. Added to the assumed 300 gram inventory in process, the total tritium inventory will be about 400 grams.

Structure

The structural activation is being calculated by Maninger and Fetter

Fission Products

I, again, estimate the fission products as proportional to the fission power. Then, based upon a 1520 MW fission power and a 2500 MW fusion power, the factor to scale Table 1 of this memo is 1.1. Thus, the ROM estimate of fission product activation is 1.2×10^8 Ci or $3.0 \cdot 10^8$ Km³ (BHP).

Activides

Since ^{233}Th and ^{233}Pa are not actively removed from the system, their quantities can, again, be reasonably accurately scaled from the ^{232}Th (n,g) rate. The scaling factor is

$0.50/0.73 \times 2500,3000 = 0.6$, so the ^{233}Th and ^{233}Pa inventories are expected to be $1.2 \cdot 10^{10}$ and $0.8 \cdot 10^{10}$ Ci, respectively.

Level of Safety

Assuming that the molten salt can be drained to a dump tank (no detailed design, but no anticipated problems), this design should be more safe than the previous lithium design for several reasons:

- ° lower source term.
- ° salt is not chemically reactive (no highly reactive chemicals in system).
- ° no potential for mechanical jamming or separation of fuel from heat transfer material.

Distribution:

J. Holdren
B. Budnitz
J. Crocker
M. Kazimi
K. Schultz

Lithium Filled Dump Tank

Schematic

**INHERENTLY SAFE LITHIUM FILLED DUMP TANK FOR HYBRID FUEL
OPERATING PARAMETERS**

TANK VOLUME (M ³)	1000
TIME TO COVER 60cm OF DRY FUEL (HRS)	2.3
MAXIMUM URANIUM NITRIDE TEMP. (°C)	<900
COOL DOWN FLUID	AIR
VELOCITY (M/S)	23
PUMPING POWER/MODULE (Kw)	65
COOL DOWN TIME (DAYS)	7

**LITHIUM DUMP TANK USING LATENT HEAT FOR CONTROLLING
 ^{239}Pu HYBRID AFTERHEAT FUEL TEMPERATURES IS FEASIBLE**

- **FOLLOWING LOCA, RELIES ON PASSIVE MELTING OF PLUG AND GRAVITY DUMP TO A LARGE CYLINDRICAL TANK, TANK AND DUMP CHUTE MUST REMAIN INTACT**
- **DUMP TANK CONTAINS FROZEN LITHIUM AT ~ 25 °C**
- **URANIUM NITRIDE TEMPERATURES WILL REMAIN BELOW 900 °C**
- **LITHIUM CAN BE RE-FROZEN USING FORCED AIR IN ABOUT ONE WEEK**
- **LITHIUM RESERVES APPEAR ADEQUATE: INVENTORY FOR 44 FUSION BREEDERS IN KINGS MOUNTAIN**
- **IT IS NOT CHEAP (~ \$600M, OR 10% OF COST OF A LARGE FUSION BREEDER)**

Required Distribution of Uncovered Fuel (Approximately To Scale)

COOLED TANDEM MIRROR FUSION BREEDER

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This paper summarizes the current version of a reference design for a liquid lithium cooled tandem mirror fusion breeder (fusion-fission hybrid reactor). A more detailed description of the design is provided in Reference 1. The updated design (shown in Figure 1) incorporates the results of several recent studies (2) which have attempted to resolve key technical issues which were associated with an earlier reference design completed in 1982.(3,4) The issues which were addressed relate to the following areas of design and performance:

- o nuclear performance
- o Magneto-Hydrodynamic (MHD) pressure loading
- o beryllium multiplier lifetime
- o structural efficiency & lifetime
- o reactor safety
- o corrosion/mass transfer
- o fusion breeder capital cost

A list of overall performance parameters for the updated fusion breeder design is provided in Table 1. The thorium volume fraction in the blanket breeding zones was increased from 3% (original design) to 16% as shown in Figure 2. The blanket is expected to achieve a net fissile breeding ratio (per fusion) of 0.84, with a tritium breeding ratio of 1.06, and an average blanket energy multiplication of 2.44. It would operate at a relatively low neutron wall loading (1.7 MW/m^2) with a low lithium coolant outlet temperature (425 °C). These features provide for low corrosion and very low beryllium swelling ($\sim 0.3\% \Delta V/V$) over the operating cycle. Similarly, the irradiation lifetime of the ferritic steel blanket structure is expected to exceed 10 calendar years (180 dpa).

Despite the increased blanket energy multiplication and reduced lithium coolant outlet temperature, an acceptable first wall MHD pressure of 1.7 MPa is estimated for the reference flow conditions. The updated design provides for a mobile, pebble-shaped, beryllium/thorium fuel element which can be loaded and discharged to a dump tank without removal of the blanket. As shown in Figure 3, the design features redundant reactor safety systems including a dump tank which can be passively cooled.

In addition to the blanket design revisions, a plant concept, cost and fuel cycle economics assessment has been completed. Assuming that the fusion breeder uses the same 2600 MW_e fusion plant design as was developed for the 1983 Mirror Advanced Reactor Study (MARS), the total plant cost and net electrical production are expected to be \$6.3 billion and 1990 MW_e, respectively. In comparison with the MARS fusion-electric plant estimates, these are both ~ 1.7 times higher. However, the fusion breeder also would produce 6660 Kg/yr of ²³³U fuel for consumption in fission burner reactors. As shown in Figure 4, the 6660 Kg/yr would be sufficient to provide makeup for about 25 LWRs operating on a denatured thorium fuel cycle. Economics studies which model the symbiotic electricity generation system, shown in Figure 4, indicate that the reference fusion breeder would be economical if the price of mined uranium were to increase to only about 90 \$/lb.

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and the design continues to promise attractive levels of performance as an economical producer of fissile fuel for many client LWRs.

REFERENCE LIQUID-METAL-COOLED FUSION BREEDER BLANKET

	Performance Parameter
Tandem Mirror Fusion Driver	
• MARS performance with lower wall loading	
Fusion power	2600 MW
Plasma gain, Q	26
Neutron wall loading	1.7 MW/m ²
Breeding Blanket	
• Average blanket energy multiplication	2.4
• Net 233U breeding ratio	0.84
• Tritium breeding ratio	1.06
Overall Performance	
• Maximum thermal power	5700 MW
• Average thermal power	5075 MW
• Average net electric power	1990 MW _e
• Capacity factor	70%
• 233U production	6660 kg/yr
• Number of 1 GW _e LWRs fueled	25

OVERALL DESIGN SPECIFICATIONS

NUCLEAR PERFORMANCE

VERSUS**VERSUS**

THORIUM VOLUME FRACTION

(including self-shielding effects)

**SAFETY SYSTEM SCHEMATIC (OPTIONS) FOR
FUSION BREEDER BLANKET WITH MOBILE FUEL FORM**

OVERVIEW OF UPDATED LIQUID-METAL-COOLED FUSION BREEDER REFERENCE DESIGN

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**Presentation at ANS Fusion Topical Meeting
Reno, NV June 18, 1986**

Outline

- **Review of blanket designs, specifications and expected performance**
- **Overview of Key Design Issues**
  - **Nuclear Performance**
  - **Beryllium Use**
  - **MHD Pressure Drops**
  - **Structural Design**
  - **Beryllium Use**
  - **Reactor Safety**
- **Discussion of Reactor Cost and Plant Economics**
- **Discussion of R&D needs**

CHRONOLOGICAL HISTORY OF FISSION SUPPRESSED BLANKET DESIGN EFFORTS

- a - beryllium multiplier, Li/Th/Be salt breeder, graphite clad beryllium
- b - 2 zone design. Liquid lithium multiplier zone, molten salt breeder zone
- c - beryllium/thorium/sodium packed bed, pipe cooled
- d - beryllium multiplier, Pb-Li/ThO₂ eutectic breeder, pressure tubes
- e - direct cooled packed bed
- f - direct cooled packed bed with tritium breeder in tubes
- g - direct cooled packed bed with molten salt in tubes

A NATIONAL LAB AND INDUSTRY TEAM HAS CONDUCTED THE FUSION BREEDER PROGRAM

Reference Liquid-Metal-Cooled Fusion Breeder Blanket

Coolant Choice

- Lithium was chosen over the ^6Li enriched lithium-lead eutectic (17Li-83Pb) for several reasons.
  - less corrosive in the 400° - 500°C range
  - lighter than the beryllium-thorium pebbles
  - superior tritium solubility
  - less expensive
- Lithium safety and associated cost is a concern.

Structural Material Choice

- Due to several advantages, ferritic/martensitic steel (e.g., HT-9 or 2-1/4 Cr-1 Mo) was selected over austenitic stainless steel (e.g., titanium-modified 316 SS)
  - lower swelling (~150 dpa incubation dose)
  - lower thermal stress (lower thermal expansion)
  - lower irradiation creep
  - lower corrosion in liquid metals
- Potential disadvantages of ferritic/martensitic steels
  - elevation of ductile-to-brittle transition temperature
  - requirement for post-weld heat treatment to re-establish mechanical properties
  - ferro-magnetism

A 425° C Coolant Outlet Temperature Is Expected to Provide Excellent Chemical Compatibility, but Several Concerns Require Experimental Confirmation

- Self-welding of beryllium or thorium (1 - 2 yr)
- Beryllium-thorium reactions (1-2 yr)
- Mass transfer of ferritic steel and/or beryllium in lithium (30 yr)
- Fission product and actinide activation product mass transfer to the coolant and coolant system structure

SOME DESIGN ISSUES WE HAVE BEEN INVESTIGATING

- Nuclear Performance - effects of heterogeneities and resonance self-shielding
- Liquid Metal MHD Pressure Drops - especially flow through the packed bed
- Structural Design - thermal stress
- Beryllium Pebble Issues - irradiation lifetime
- Reactor Safety - accident pathway identification and safety system modeling
- Reactor Costs - a better capital cost assessment

Overall Design Specifications

	Performance Parameter
Tandem Mirror Fusion Driver	
• MARS performance with lower wall loading	
Fusion power	2600 MW
Plasma gain, Q	26
Neutron wall loading	1.7 MW/m ²
Breeding Blanket	
• Average blanket energy multiplication	2.4
• Net ²³³ U breeding ratio	0.84
• Tritium breeding ratio	1.06
Overall Performance	
• Maximum thermal power	5700 MW
• Average thermal power	5075 MW
• Average net electric power	1990 MW _e
• Capacity factor	70%
• ²³³ U production	6660 kg/yr
• Number of 1 GW _e LWRs fueled	25

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Key Design and Performance Parameters for the Reference Blanket

GLOBAL PARAMETERS (70% Capacity factor, average over system and time)

Central cell length	130 m
Number of blanket modules	32
Number of central cell coils	32
Central cell coil B field strength on axis	4.7
Central cell fusion power	2600 MW
Central cell fusion neutron power	2080 MW
Maximum central cell thermal power	5700 MW
Average central cell thermal power	5075 MW
^{233}U fuel production	6656 kg/yr
Average in-core fissile inventory	5500 kg
Thorium throughput	630 MT/yr
Beryllium throughput	275 MT/yr

Key Design and Performance Parameters for the Reference Blanket (Continued)

BLANKET MODULE MECHANICAL DESIGN

Structural material	HT-9 steel
Module length	4.1 m
Fraction of module length used for multiplier/fuel	~95%
First wall radius	1.5 m
Number of multiplier/fuel zones	2
Multiplier/fuel zone volume fractions:	
Beryllium (3.0 cm O. D. pebbles	44%
Lithium	40%
Thorium (including bred fissile)	16%
Ferritic steel	2%
Thickness of each fuel zone	20 cm
Lithium reflector thickness	30 cm
Blanket outer radius	2.34 m
Shield thickness	75 cm
Magnet inner bore	6.7 m
Magnet pitch	4 m
Number of coolant outlet pipes	20 each

Key Design and Performance Parameters for the Reference Blanket (Continued)

HEAT TRANSFER, POWER FLOW, AND THERMAL DESIGN PARAMETERS (at maximum blanket M)

Neutron wall loading	1.72 MW/m ²
Coolant	lithium (3% ⁶ Li)
Maximum thermal power per blanket module	208 MW
Coolant inlet temperature	275°C
Coolant outlet temperature	425°C
Lithium flow rate	0.69 m ³ /sec
Lithium pressure drop	2.6 MPa (370 psi)
Maximum lithium pump power (per module)	1.8 MW
Maximum first wall pressure	1.7 MPa (225 psi)
Minimum first wall temperature	302°C
Maximum first wall temperature	353°C
Maximum structure temperature	425°C
Maximum beryllium surface temperature	410°C
Maximum beryllium internal temperature	445°C
Maximum beryllium ΔT	~70°C

Key Design and Performance Parameters for the Reference Blanket (Continued)

NUCLEAR DESIGN PARAMETERS (at full power and 100% capacity factor)

Net fissile breeding ratio	0.84
Net tritium breeding ratio	1.06
Minimum blanket energy multiplication	1.71
Maximum blanket energy multiplication	3.17
Maximum thorium power density	70 W/cm ³
Maximum beryllium power density	9.1 W/cm ³
Maximum lithium power density	4.5 W/cm ³
Maximum average power density	17 W/cm ³
Zone 1 fuel residence time	295 days
Zone 2 fuel residence time	590 days
Average uranium discharge concentration	1.00%
Average protactinium discharge concentration	0.05%
Average fission rate per fusion	0.09
Average fission burnup at fuel discharge	~1000

Optimal Thorium Concentrations Have Been Increased from 3% to Compansate for Resonance Self-Shielding Effects

The Nuclear Performance for 16% Thorium and 1% ^{233}U Average Discharge Enrichment Is Excellent

	<u>BOC</u>	<u>EOC*</u>	<u>AVE.</u>
Net fissile production	0.834	0.853	0.844
Blanket energy multiplication	1.71	3.17	2.44
Tritium production	1.06	1.06	1.06
Ideal ^6Li enrichment (%)	3.4	2.6	3.0
Thorium fissions	0.033	0.035	0.034
Uranium fissions	0	0.107	0.053

* 1% ^{233}U + 0.05% ^{233}Pa in thorium

The Two Zone Blanket Concept

Maximizes Fissile Enrichment By Allowing

The First Zone To Be Changed Out

Twice As Often As The Second Zone

VERY LOW BLANKET POWER SWINGS CAN BE ACHIEVED

RESULTS FOR EACH 21 FULL
POWER WEEK INTERVAL

Power Swing: $2.14 \leq M \leq 2.74$

Fuel Discharge: $3/8$ Total
Inventory

MHD Flow Issues

- MHD flow issues include:
  - pressure drops (most important for this design)
  - corrosion
  - heat transfer (primarily important for tokamak)
- MHD pressure drop terms must account for:
  - channel flow
  - entrance and exit from the magnetic field
  - flow expansion and contraction
  - turns in the magnetic field
  - flow through the packed bed
- All of these terms are quite uncertain and depend upon:
  - flow development in the blanket
  - eddy current paths in the structure
  - ability to insulate the walls of flow channels

REVISED MHD PRESSURE DROP ESTIMATES ARE MORE OPTIMISTIC BUT HIGHLY UNCERTAIN

	<u>1982</u>	<u>1985</u>
Wall load	1.3	1.7
Peak Blanket M	2.0	3.2
Normalized Blanket Power	1	2.1
	<u>Pressures (PSI)</u>	
Inlet plenum ^a	38	104
First wall	0.1	0.2
Packed bed	262	164
Lithium reflector ^b	1.8	4.8
Outlet plenum and pipes ^a	5.8	16
Turns, contractions, expansions	<u>16</u>	<u>84</u>
Total	323	373
First wall pressure	279	227

a) Insulated pipe - 1 mm thick wall assumed

b) could be 43% lower

c) assumed pressure drop coefficient $k = 0.06$

Structural Design Issues

- Hydrostatic pressure outside of first wall cylinder leads to hoop stress and buckling loads.
- First wall thermal stress is low for the tandem mirror, but 150°C ΔT from front-to-back of blanket causes problems.

Geometry and Coordinate System of Blanket Structural Model

A Design Solution Which Decouples the Entire Interior Structure Is Expected to Provide Both Low Stress and Design Simplicity

New Models Have Been Developed to Predict Beryllium Swelling

This Data Indicates that Beryllium Swelling Will Be Negligible

- Maximum helium production = 3800 appm/MW-yr/m²
- Beryllium temperature = 400°C
- Service Life = 0.87 FPY at 1.7 MW/m² (0.87 = 1.02 x .85)
- Swelling = 0.32% ($\Delta V/V$)

With Beryllium Recycle, A Large Symbiotic Electrical Generation Economy Is Possible

Basis

- Be avail. = 50% of post-2000 domestic resource = 71,400 MT
- 200 year fusion breeder economy

Fusion Hybrid Safety Must Be Viewed from the Systems Perspective

In comparison with fusion-electric blankets, fusion hybrids will have higher radioactive inventories and decay afterheat levels

Key requirement for fusion hybrid safety is that it does not degrade the safety of the overall electricity generation system. The hybrid must be as safe as the LWR clients which dominate the system

- **Studies* show that safety goals can be achieved because of subcriticality and lower power density**

In comparison with the LWR, hybrid decay power densities will be reduced, but are a concern-especially considering the added complexity of fusion

*** References**

- 1) **"Safety Assessment of the Fusion Breeder," J. Garner and I. Maya, *Nuclear Technology*, July 1985**
- 2) **Special Topics Report for the Reference Tandem Mirror Fusion Breeder: Reactor Safety Assessment, LLNL Report UCID 20166, September 1984**

Fusion Hybrid Safety Must Be Viewed from the Systems Perspective (Continued)

Fusion breeders with low M fission-suppressed blankets offer important advantages

- **Fissioning reduced to the extent that actinides (e.g., ^{233}Pa and ^{233}Th) provide the greatest contribution to both the hazard and the afterheat**
- **Lower afterheat levels allow much longer time before corrective action is required**
- **Mobile fuel provides for semi-passive away from reactor cooling in dump tanks**

Ambitious safety goals have been set for fusion breeders with fission-suppressed blankets

Actinide Afterheat Dominates During Time Periods Relevant to Blanket Heatup

Fusion Breeders Can Use a Variety of Semi-Passive Heat Removal and Engineering Systems

The Following Safety System Capabilities for Various Accident Scenarios Result

	<u>Shield Cooling Alone</u>	<u>Auxiliary First Wall Cooling Alone</u>	<u>Both Systems</u>
• Loss of coolant flow, fuel not dumped	Adequate	Not adequate	---
• Loss of coolant flow, fuel dumped	Adequate	Adequate	---
• Loss of coolant, fuel not dumped	Not adequate	Not adequate	Marginal*
• Loss of coolant, fuel dumped	Adequate	Adequate	---

- * exceeds 730°C limit for reuse in ~2 hr. blanket intervals. May exceed 900°C limit for blanket structure integrity if steel emissivity is less than 0.5. Further analysis required.

Encouraging Thermal modeling Results for Loss of Coolant Accident with Fuel in Place

**The Fusion Breeder Cost Per Unit Electricity Is Similar to
MARS, but It Produces Over 6600 kg/yr of ^{233}U**

	MARS	Fusion Breeder
Direct cost, \$M 1985	2235	3615
Indirect and contingency, \$M 1985	3775	1790
Total overnight cost, \$M 1985	3340	5405
Construction time, yr	6	8
Total capitalized cost, \$M 1985	3745	6290
Net electricity, MW_e	1200	1990
Capital cost per unit electrical power, $\\$/\text{kW}_e$	3165	3160
Cost relative to projected LWR ($\\$/\text{kW}_e/\\$/\text{kW}_e$)	2.30	2.35
^{233}U production (kg/yr, 70% capacity)	N/A	6655

Simple Average Capital Cost Example

Premise

- Average capital cost of an electricity generation system is a good indicator of overall economics

Assumptions

- Fusion breeder cost per unit of thermal power = 2.4 x cost of fission converter client
- Fusion breeder electrical efficiency = LWR client electrical efficiency
- Fusion breeder provides fissile makeup for 13.5 fission clients of same thermal power

Average capital cost/kW_e (normalized to client cost/kW_e)

$$\frac{(2.4 \times 1) + (1 \times 13.5)}{(1.0 \times 1) + (1 \times 13.5)} = 1.10$$

Typical Symbiotic Electricity Generation System Based upon Fission-Suppressed Breeder and Denatured Thorium Fuel Cycle

Some Characteristics of the Fuel Cycle Center

Excellent follow-on to once-through LWR

- **Replaces uranium mining and enrichment**
- **Utility continues to operate LWR**

Only a few fuel cycle centers required

- **A fuel cycle center with 3 large (2600 MW_f, 5000 MW_f) fusion breeders can support 80 GW_e of power generation**
- **Remote siting can relax some safety/environmental and safeguards concerns**

Fissile fuel export can provide a needed opportunity to improve US balance of trade and regain leadership in international nuclear markets

- **Encourage LWR development at home and abroad**
- **Enter long-range contracts to build LWRs and/or supply denatured fuel and reprocess spent fuel**
- **Sensitive fuel cycle operations controlled in US**
- **Plutonium economy could be avoided**

Typical Reprocessing Economics

(1985 \$, First Year of Operation)

	Thorex	Pyro-chemical
Throughput (Mt/yr) ^a	1900	1900
Fissile discharge, assy, atom %	1.05	1.05
Plant capital cost, \$M	2600	1300
Cost of capital, \$M/yr ^b	340	170
Plant operating cost, \$M/yr ^c	170	40
Total annual cost, \$M/yr	510	210
Reprocessing cost, \$/kgHM	270	110
Unit cost, \$/g	26	10.5

^a Three fusion breeders sharing a reprocessing plant

^b Utility ownership (13.1%/yr)

^c First year, increases 3%/yr thereafter

Economic Results for the Symbiotic System

	Symbiotic System
Breeder	
• Total cost (\$/kW ₁)	1120
• Breeder to LWR capital cost ratio	2.35
• Annual capital cost* (\$/kW-yr)	147
• Annual operating cost (\$/kW-yr)	47
• Total annual cost (\$/kW-yr)	195
LWR	
• Total cost (\$/kW ₁)	444
• Annual capital cost (\$/kW-yr)	59
• Annual operating cost (\$/kW-yr)	42
• Total annual cost (\$/kW-yr)	101
System Cost	
• Breeder to LWR annual cost ratio	1.93
• Year one cost of electricity (mills/kW-hr)	47
• Average PV cost of electricity (mills/kW-hr)	36
• Average cost of electricity relative to U ₃ O ₈ fueled LWR	1.1
• Average breakeven cost of U ₃ O ₈ (\$/kg)	196

The Fusion Breeder Breakeven Is Much Less Sensitive to Capital Cost than LMFBR Breakeven

Analysis developed from ORNL data using updated hybrid performance, cost, operating, and fuel cycle cost assumptions

Research Requirements to Confirm the Viability of the Reference Fusion Breeder Concept

MHD pressure drops

- **Packed bed and duct flow including insulator development**

Liquid metal compatability

- **Corrosion, mass transfer and coating development**

Beryllium irradiation damage

Pyro-chemical reprocessing

- **Magnesium dissolution process demonstration**

Fissile breeding

- **Integral test assembly with a point**

Development of thorex fuel cycle for the LWR

Near Term Development Program for Liquid Metal Fusion Breeder

A 425° C Coolant Outlet Temperature Is Expected to Provide Excellent Chemical Compatibility, but Several Concerns Require Experimental Confirmation

- Self-welding of beryllium or thorium (1 - 2 yr)
- Beryllium-thorium reactions (1-2 yr)
- Mass transfer of ferritic steel and/or beryllium in lithium (30 yr)
- Fission product and actinide activation product mass transfer to the coolant and coolant system structure

Summary

- **An attractive liquid metal blanket fusion hybrid has been developed for the tandem mirror**
- **A fully developed version of this technology costing 2.4 x LWR could be economical in the U.S. at mined uranium prices of about 100 \$/lb**
- **If government owned, the breakeven price of uranium can decrease to about 55 \$/lb**
- **This blanket concept will be applicable to large tokamaks with low ($\sim 2 \text{ Mw/m}^2$) wall loads if the magnetic fields can be reduced (e.g., by achieving a plasma beta of 20%)**
- **Liquid metal safety is a concern – especially for tokamaks**
- **Tremendous progress can be made via non-fusion testing of fusion breeder blankets. Relatively more than for fusion-electric blankets**

UPDATED REFERENCE DESIGN OF A LIQUID METAL COOLED
TANDEM MIRROR FUSION BREEDER

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Abstract

~~UPDATED REFERENCE DESIGN OF A LIQUID METAL COOLED TANDEM MIRROR FUSION BREEDER~~

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This paper summarizes the current version of a reference design for a liquid metal cooled tandem mirror fusion breeder (fusion-fission hybrid reactor). The design update incorporates the results of several recent studies which have attempted to resolve key technical issues which were associated with an earlier reference design completed in 1982. The issues which were addressed relate to the following areas of design and performance: nuclear performance, magneto-hydrodynamic (MHD) pressure loading, beryllium multiplier lifetime, structural efficiency and lifetime, reactor safety, corrosion/mass transfer, and fusion breeder capital cost.

The updated blanket design provides increased performance and reduced technological risk in comparison with earlier fission-suppressed hybrid blanket designs. Specifically, ~~as shown in Table 1~~, the blanket is expected to achieve a net fissile breeding ratio (per fusion) of 0.34, with a tritium breeding ratio of 1.06, and an average blanket energy multiplication of 2.44. It would operate at a relatively low neutron wall loading (1.7 MW/m^2) with a low lithium coolant outlet temperature (425°C). These features provide for a very low beryllium swelling ($\sim 0.3\% \Delta V/V$) over the operating cycle. Similarly, the irradiation lifetime of the ferritic steel blanket structure is expected to exceed 10 calendar years (180 dpa). Despite the increased blanket energy multiplication and reduced lithium coolant outlet temperature, an acceptable first wall MHD pressure of 1.7 MPa is estimated for the reference flow conditions. The updated design ~~continues to~~ provides for a mobile, pebble-shaped, beryllium/thorium fuel element which can be loaded and discharged to a dump tank without removal of the blanket. The dump tank can be passively cooled to provide attractive reactor safety features.

In addition to the blanket design revisions, a plant concept, cost, and fuel cycle economics assessment has been completed. Assuming that the fusion breeder uses the same 2600 MW_e fusion plant design as was developed for the 1983 Mirror Advanced Reactor Study (MARS), the total plant cost and net electrical production are expected to be 6.3 billion and 1990 MW_e, respectively. In comparison with the MARS fusion-electric plant estimates, these are both ~ 1.7 times higher. However, the fusion breeder also would produce 6660 kg/yr of ^{233}U fuel for consumption in fission burner reactors. Specifically, $\dots$ 6660 kg/yr would be sufficient to provide makeup for about 25 LWRs operating on a denatured thorium fuel cycle. Economics studies which reflect this high level of market leverage indicate that the reference fusion breeder would be economical if the price of mined uranium were to increase to only about ~~100~~ \$/lb. 90 \$/lb

In summary, an updated liquid metal cooled blanket design for a tandem mirror fusion breeder has been completed. Several prior feasibility issues have been addressed and the design continues to promise attractive levels of performance as an economical producer of fissile fuel for many client LWRs.

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I. INTRODUCTION

I.A. Motivation

The growth of fission power generation has slowed dramatically during the past decade. However, fission remains one of few future energy options with potential to provide an economically and environmentally acceptable alternative to the continued use of diminishing fossil fuel resources for central station electrical power generation. Its principal competitor, coal-fired electricity generation, has its own environmental and economic shortfalls. Consequently, there is a strong possibility that nuclear fission power may, again, be projected to provide an increasing fraction of U.S. and world electricity generation. If so, it is also possible that a shortfall in uranium resources will occur by the middle of the next century⁽¹⁾. Fusion could alleviate this shortfall by producing fissile fuels via the transmutation of abundant fertile materials such as ^{238}U and ^{232}Th . Fusion reactors which utilize fusion neutrons in this manner are designated "fusion breeders." The often used term "fusion-fission hybrid" is roughly synonymous, but also includes very high blanket energy multiplication systems, which would be optimized for in-situ power production.

Fusion breeders optimized to emphasize fissile fuel production would replace the uranium mining and enrichment segments of the fission fuel cycle and would enable the continued use of fission converter reactors such as those currently in use. Studies indicate that each fusion breeder can provide an economical source of fissile fuel to support 10 to 15 times as fission reactors of the same rated (thermal) power ^(2,3,4). Because relatively few fusion breeders would be needed to support a rather large nuclear fission capacity, it is not likely that a fusion breeder would be owned by an

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Beryllium Performance

In addition, investigators from the University of California at Los Angeles (structural mechanics and ferritic steel irradiation damage), the University of Wisconsin (beryllium irradiation damage) and the Energy Technology Engineering Center (liquid metals and materials) participated in the study.

Several detailed laboratory reports relating to the evolution of a reference liquid metal cooled tandem mirror fusion breeder design concept have been issued. (2,5,6,7,8,9) Reference 2 is the most complete description of the design, but is updated by Reference 5, which is devoted to more detailed studies of key engineering issues. References 8 and 9 primarily relate to the use of beryllium in fusion applications. In this report, the results of these prior studies have been synthesized to best represent a "reference"

Table I.
Major Parameters for the MARS Fusion Driver

Central cell length	130 m
Plasma radius	0.49 m
First wall radius	0.6 m
Fusion power	2600 MW
Plasma power gain, Q	26
Average central beta	0.28
Peak central density	$3.3 \times 10^{20} \text{ m}^{-3}$
Ion temperature	24 keV
Electron temperature	24 keV
Central cell field	4.7 T
Peak choke coil field	24 T ^a
Yin-yang mirror field	7.5 T ^b
First wall loading	1.72 MW/m ²
Anchor ICRH power, absorbed/ injected	5.7/6.7 MW each anchor
Plug neutral beam power, absorbed/injected	2.84/4.43 MW each plug
ECRH power, absorbed/injected	42/42 MW each plug
Copper coil and drift pumping power	50 MWe each end
Total recirculating power	~350 MWe

---

generate excess neutrons beyond that required to sustain tritium self-sufficiency. This class of blanket has been emphasized in recent studies because the decay afterheat is lower and the fissile fuel production per unit of thermal energy is higher than that of "fast-fission" blankets that utilize ^{238}U or ^{232}Th fissions induced by the 14.1 MeV neutrons to generate the excess neutrons for breeding. The lower afterheat leads to simpler, less risky, blanket designs, while the higher fuel production results in fewer fusion breeders and more attractive deployment scenarios. The reference fission-suppressed, fusion breeder can provide fissile makeup to fuel 13.5 LWRs* of equal thermal power (thermal support ratio) while a typical uranium fast-fission, fusion breeder can fuel fewer than 5 LWRs. (14,15)

---

*The LWRs referred to in this paper are conventional designs in use today. Some evolution in core design, resulting in higher fuel utilization is possible.

II. DESIGN OVERVIEW

II.A. Design Description

After an extensive scoping phase^(6,7) during 1981-1982 (see Section II.B), a reference blanket concept, based upon flowing liquid lithium coolant radially through a two-zone packed bed of beryllium pebbles with thorium snap rings, was selected. This concept, shown in Figure 1, has been designed in accordance with the specifications and performance levels provided in Table II.

The reference tandem mirror blanket design would be constructed from a low nickel ferritic steel (e.g., HT-9) and would be liquid lithium cooled. The blanket would operate at a moderate maximum temperature (425°C) and neutron wall loading (1.7 MW/m^2). In terms of performance, technological development requirements and risk, this design can be classified as "moderate technology." For comparison, a "low technology" blanket could be developed using a low temperature (approximately 100°C) water-cooled design producing fuel but no power, while a "high technology" high performance blanket might be based upon helium cooling and Molten Salt Breeder Reactor technologies.⁽³⁾

The coolant flow in the blanket resembles that of a conventional oil filter. Specifically, coolant flows radially inward to the first wall plenum through a thin coolant annulus and is distributed to the packed bed through perforations in an intermediate wall which, in combination with the first wall and radial stiffening rings, provides a very stiff cylindrical structure. Having passed through the intermediate wall into the blanket, the coolant flows radially outward through two fuel zones (separated by another perforated wall), exits the bed through a third perforated wall outside of the second fuel zone, and exits the blanket through 20 large outlet pipes. The composite

Table II.
Key Design Specifications and Performance Parameters
for the Reference Fusion Breeder Blanket

<u>Global Parameters</u> (70 percent capacity factor)	
Central cell length	130 m
Number of blanket modules	32
Number of central cell coils	32
Central cell coil B field strength on axis	4.7
Central cell fusion power	2600 MW
Central cell fusion neutron power	2080 MW
Maximum central cell thermal power	5700 MW
Average central cell thermal power	5075 MW
^{233}U fuel production	6656 kg/yr
Average in-core fissile inventory	5500 kg
Thorium throughput	630 MT/yr
Beryllium throughput	275 MT/yr
<u>Blanket Module Mechanical Design</u>	
Structural material	HT-9 steel
Module length	4.1 m
Fraction of module length used for breeding ^{233}U	~95%
First wall radius	1.5 m

Table II (continued)

Maximum first wall pressure	1.7 MPa (225 psi)
Minimum first wall temperature	302°C
Maximum first wall temperature	353°C
Maximum structure temperature	425°C
Maximum beryllium surface temperature	~~400°C~~ 400°C
Maximum beryllium internal temperature	445°C
Maximum beryllium ΔT	~40°C
<u>Nuclear Design Parameters (at full power and 100 percent capacity factor)</u>	
Net fissile breeding ratio	0.84
Net tritium breeding ratio	1.06
Minimum blanket energy multiplication	1.71
Maximum blanket energy multiplication	3.17
Maximum thorium power density	70 W/cm ³
Maximum beryllium power density	9.1 W/cm ³
Maximum lithium power density	4.5 W/cm ³
Zone 1 fuel residence time	316 full power days
Zone 2 fuel residence time	632 full power days
Average uranium discharge concentration	1.00%
Average protactinium discharge concentration	0.05%
Average fission rate per fusion	0.09
Average fission burnup at fuel discharge	~1000 MWD/MT

Figure 2. *Fission -*
Contributions to the Afterheat in Suppressed
Fission Blankets

Figure 3.
Schematic Diagram of Fusion Breeder Fuel
Handling Circuit

Table III.

Summary of Structural Materials Assessment

(from Reference 16)

Candidate Alloys	Austenitic Steel PCA-CW	Ferritic Steel HT-9	Vanadium V-15 Cr - 5 Ti
Thermal stress factor, $\text{MW/m}^2\text{-mm}$ (500°C)	3.2	4.8	9.8
Maximum surface heat flux ^a , MW/m^2	0.3	0.4	1.3
Design stress limit, MPa (Smt) ^b	100	155	165
Maximum allowable temperature, °C ($\sim 0.5 T_m$)	550	550	720
Lithium Corrosion rate $\text{mg/m}^2\text{-h}$ (500 °C) ^c	60	2	<0.01
Radiation lifetime, DPA (5% swelling)	100 (500°C)	190 ^d	220 ^d
Critical design issues	• Limited lifetime (swelling) • High-thermal stress • Liquid-metal • Radiation Creep • Operating temperature limit		
	• Weld procedure (postweld heat treatment) • Ductile-to-brittle transition temperature (DBTT) above room temperature • Operating temperature limit • Liquid-metal embrittlement • Ferromagnetic properties		
	• R&D requirements • Weld procedure (inert environment) • Oxidation characteristics • High-tritium permeation rates • Costs		

^aIdealized flat plate 5 mm thick with 50°C film coefficient, $T_{\text{out}} = 400^\circ\text{C}$.^b $2 \cdot 10^4$ hr, 100 dpa, 500 °C.^cPredicted for 1.5 m/s flow rate.^dNot well defined, may be higher.

Figure 4a.
Reference Blanket First Wall DBTT vs. Axial Position
and Irradiation Time

lead-lithium systems and will have the advantage of lower normal tritium releases. An acceptable level of lithium safety appears to be achievable based upon the development of liquid sodium coolant safety systems in the LMFBR program.^(2,5) A recognition that fusion breeder reactors would not, most likely, be sited near population centers (but, rather, in remote safeguarded fuel cycle centers) provides additional motivation for the choice of a liquid lithium coolant.

The choice of thorium metal as a fertile fuel form rather than thorium dioxide (thoria) or another thorium form is primarily based upon fuel cycle considerations. Specifically, thorium metal is less expensive to reprocess using either conventional aqueous or the less developed, but potentially less expensive pyrochemical techniques.^(2,20) There is, however, a concern regarding the potential for solid-solid chemical reactions (Be_{13}Th inter-metallic, ^{compound formation} Th-Th self welding) and/or fission product release in the liquid metal bath.^(2,5) If these effects are substantial, then the use of a thin coating (Mo, ThO_2 , ThC, or TiC) should be investigated.

Similarly, there is a concern relative to Be-Be self-welding which might result in pebble sticking. In this case, the use of a thin beryllium coating (Be_{12}Fe , Be_{12}Cr , Be_{12}Mo) can be investigated as a means of mitigating both Be and Be-Th interactions. Although some experimental work has been conducted to determine the extent of chemical interactions for the reference fusion breeder blanket,⁽²⁾ more work in this area is needed. Our choice of 425°C as the maximum blanket temperature should be conservative with respect to materials compatibility issues.

III. KEY DESIGN AREAS

Several features of the reference fusion breeder design are discussed in more detail in this section. Each of these was addressed as a "special topic" during 1983-1984^(5,8,9) after the initial design concept was developed during 1982.⁽²⁾ The intent of this section is to introduce the reader to the unique design and analysis issues associated with the reference fusion breeder blanket described in the previous Section.

III.A. Nuclear Design and Fuel Management

The economic attractiveness of the fusion breeder is strongly dependent upon its ability to supply excess neutrons for fissile breeding above and beyond the requirement for tritium self-sufficiency. Consequently, the nuclear performance of the reference fusion breeder blanket continues to be a primary focus in design studies.

Beryllium, the principal constituent in the reference blanket, is a very strong neutron energy moderating material. Consequently, local heterogeneities and resonance self-shielding in the thorium have been recognized as being very important to the development of a realistic estimate of the net breeding performance. ^{As Taczenowski:} Taczenowski⁽²¹⁾ was the first to point out that the fusion breeder blanket, as originally defined in 1982 [55 volume percent Be, 3 % Th (including bred ^{233}U), 40 % Li (0.2 a/o ^6Li)] would produce substantially more tritium, less ^{233}U and more energy than had been expected prior to consideration of resonance self-shielding effects. Later, it was shown^(5,22,23)^{~24} that by increasing the thorium volume fraction in the bed, it would be possible to both increase fissile production relative to tritium

Figure 5.
 Net Fissile Production and Blanket Energy Multiplication
 versus Thorium Concentration for Fixed Tritium
 Breeding ($T = 1.06$)

Table IV.
Nuclear Performance for 16 Percent Thorium and
1 Percent ^{233}U Average Discharge Enrichment

	BOC	EOC ^a	Average
Net fissile breeding per fusion	0.834	0.853	0.844
Blanket energy multiplication ^b	1.71	3.17	2.44
Tritium breeding	1.06	1.06	1.06
Ideal ^6Li enrichment (%)	3.4	2.6	3.0
Thorium fissions	0.033	0.035	0.034
Uranium fissions	0	0.107	0.053

^a1 percent ^{233}U plus 0.05 percent ^{233}Pa in thorium.

^bblanket energy per fusion divided by 14.1 MeV

Figure 6.
The Blanket Power Swing Would be Minimized by Dividing
the 32 Central Cell Modules into Four Groups
of Differing Fuel Zone Maturity

the hydrostatic pressure at the first wall of the blanket. In the reference fission-suppressed tandem mirror fusion breeder design concept, the surface heat flux is very low and heat transfer limitations due to flow profile modification are a relatively minor concern, but the required first wall structure thickness is directly related to the MHD pressure drop in flowing the liquid lithium coolant. As such, it is a major concern which directly impacts fissile breeding efficiency.

One equation that blanket designers can use with confidence to calculate liquid metal MHD pressure drops is for Hartmann flow in a simple circular or rectangular channel. However, large uncertainties still exist for the following, more complicated, flow configurations which occur in the reference fusion breeder blanket:

- Flow through a packed pebble bed.
- Flow through bends (for one or both legs perpendicular to the B-field).
- Flow through a B-field gradient.
- Flow through channels with varying conducting wall thickness.
- Flow at an angle to the magnetic field.
- Flow through contractions, expansions, and distribution plena.
- Flow through ducts which incorporate electrically insulating materials.

Although we are reasonably confident regarding the ability to flow liquid metals in this design (i.e., the magnetic field of 4.7 T and neutron wall load of 1.72 MW/m^2 are low compared with those expected in a tokamak), it is clear that an experimental confirmation of pressure drop scaling in the relevant flow regime will be required.

Figure 7.
The Coordinates, Geometry, and Current Paths
in the Reference Blanket.

$$\nabla \times \mathbf{B} = \mu \mathbf{J}$$

(6)

where P is the pressure in the fluid and J is the current density in the fluid. Averaged over the blanket, the pressure gradient is radial and the currents in the fluid and pebbles are azimuthal. The average azimuthal current in the fluid follows from Ohm's law:

$$\langle J \rangle_f = \sigma \langle U \rangle B (K-1), \quad (7)$$

where K is defined as the ratio of the average electric field in the fluid divided by $\langle U \rangle B$. The quantity K is an intrinsic property of only the pebble-to-fluid conductivity ratio, S , the volume fraction, ϵ , and the boundary conditions on either the average azimuthal current or electric field. This constant can be evaluated precisely if the local flow field can be precisely described. However, because of the complex nature of the local flow field, such evaluation is difficult. Rather, it is observed that the value of K must be between zero and unity.

The magnetic induction, B , follows from an integration of Ampere's law. Before the integration can take place, the boundary condition on the azimuthal electric field, $\langle E \rangle$, averaged over the blanket must be specified. The boundary condition of $\langle E \rangle = 0$ leads to

$$B(r) = B_0 \left(\frac{r}{R_0} \right)^{Rm} [1 + K(S-1)] \quad (8)$$

Having specified the average azimuthal current in the fluid and the magnetic induction, B , the momentum equation can be integrated for the average

Table VI.

Summary of Overall MHD Pressure Drop Estimate

	Pa
Inlet plenum and pipes ^a	$7.3 \cdot 10^5$
First wall	$1.1 \cdot 10^3$
Packed bed	$1.2 \cdot 10^6$
Lithium reflector	$3.4 \cdot 10^4$
Outlet plenum and pipes ^a	$1.1 \cdot 10^5$
Turns, contractions, expansions ^b	$5.9 \cdot 10^5$
Total	$2.6 \cdot 10^6$
First wall pressure	$1.5 \cdot 10^6$
Pumping power (peak)	1.9 MW/mod.
Pump/thermal power ^c	0.9%
Average pump/thermal power ^d	0.7%

^aInsulated pipe - 1 mm thick wall assumed.^bAssumed pressure drop coefficient $K = 0.06$ (see Reference 2 discussion).^cAt peak power.^dOver entire plant.

mobile fuel.

The structural loading of the reference blanket results from the sum of three contributions:

- The MHD flow induced coolant pressure loading.
- Thermally induced loads due to the radial variations in the coolant temperature (i.e., the hotter back wall expands more than the cooler first wall).
- Dead weight gravity loads.

In response to these loads, the blanket will deform elastically (and possibly plastically in localized areas) and will undergo volumetric swelling and irradiation creep, all of which will determine the structural lifetime.

In order to increase stiffness and withstand the compressive buckling pressure due to the coolant, the first wall is connected to a thicker intermediate wall which separates the first wall coolant annulus from the fertile fueled region. The connections between these two walls consist of 90 radial ribs, equally spaced between the first and/intermediate wall assemblies, and extending the full length of the module (see Figure 8). By corrugating the first wall circumferentially, the first wall thickness of 0.36 cm is capable of withstanding the bending stresses. The 0.64 cm perforated intermediate wall is also corrugated to increase the stiffness. A feature of the original design, shown in Figure 9, is the connection of the double shell first wall/intermediate wall assembly to the thick outer wall of the module via several radial support plates. These plates extend from the intermediate wall, through the fuel zone separators and outer lithium plenum, to the outer wall of the module. The ~1 cm thick plates are similar to

Figure 9.

Fusion Breeder Blanket Module

Cross Section - Top Portion of Module

Figure 10.
 Trimetric of Fusion Breeder Blanket Module Showing
 Radial Zoning and Fuel Sphere Movement
 Through the Blanket

	MODELED VALUES	PRESENT VALUES
	420°C	
RIGID BOUNDARY		425°C
LITHIUM EXIT ZONE	0.3 MPa	0.4 MPa
FOURTH SHELL	417°C	415°C
FUEL ZONE	0.7 MPa	1.0 MPa
THIRD SHELL	412°C	392°C
FUEL ZONE	1.0 MPa	1.5 MPa
SECOND SHELL	394°C	332°C
FIRST WALL	398°C	337°C
	PLASMA (VACUUM)	

Figure 12.

Comparison of the Pressure and Thermal Loads Used in the Structural Model with Those of the Updated Reference Design.

HT-9 (approximately 175 MPa for operation up to 425°C. However, it was found that two features of the initial model led to the high stress levels:

- The first and intermediate wall were modeled as two independent walls rather than as a composite.
- The radial support plates were rigidly connected to the outer wall of the blanket.

The composite wall was not included in the initial stress model, but has been anticipated in the design and does not imply a significant design change. The effect of rigid connections to the outer wall was not anticipated in the earlier design and does imply a significant change in the current mechanical design.

As shown in Figure 14, with a composite first wall/intermediate wall and with a decoupled back wall, the stress levels for the earlier design can be reduced to below the allowables in all locations^a except the module side wall. The side wall was modeled as a 4 cm thick flat plate, but would be replaced by an ~1 cm thick elliptical end closure which would translate most of the bending stress to membrane stress.

These encouraging results for the earlier design can be translated to the current design, despite its higher loading. Specifically, the thermal stresses can be almost eliminated by decoupling the back wall, and the pressure stresses are expected to be approximately 1.5 times as large, still within the allowable limit of 175 MPa. A potential design solution which

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^aIt should be noted that the intermediate wall stresses, which fall between 110 and 160 MPa on Figure 14, are near the lower value at the first wall.

accomplishes the goal of decoupling the interior structure of the blanket is shown in Figure 15. This modification preserves the blanket fuel zoning, but features three nested mechanical assemblies: (1) a first wall/second wall composite with a split (to reduce thermal stress) diaphragm defining the first fuel zone, (2) a third wall (fuel zone separator) with a split concentric diaphragm (defining the second fuel zone), and (3) a fourth wall and split diaphragm (defining lithium outlet plenum). The entire internal assembly is supported by nesting the circular diaphragms within the third, fourth, and outer walls of the assembly (close tolerance not required). Fabrication of the modular blanket should be much easier than for the original design. This improved option (or a similar variant) will replace the original blanket internal structure of the reference design.

A possible design approach that has not been considered here is to support the first wall structure in tension from the back wall. This approach would be attractive because it results in containment of the coolant pressure with the minimum amount of structure. For example, most helium cooled blanket concepts would use a pod type structure in which the pressure on the first wall lobes is taken in hoop stress (3, 4, 16). The use of such pods raises two issues. If the pods are horizontally oriented, additional complexities arise because the fuel pebbles must flow vertically, causing interruptions in the structural load paths (3). If the pods are vertically oriented, supporting (i.e., balancing) the coolant pressure induced bending loads on the side wall of the last pod at the end of the module requires a special design solution(⁴).

III.D. Beryllium/Thorium Pebble Fuel Element

In this section, issues related to the use of beryllium/thorium fuel elements are reviewed. These issues include beryllium resources, fuel element

design options, lifetime considerations, and fabrication technologies. These topics are described in more detail in References 2, 5, 8, 9, and 29.

III.D.1 Beryllium Resources

Recent estimates of economically recoverable beryllium resources are shown in Table VII.⁽²⁹⁾ When compared with an expected cumulative U.S. beryllium usage by the year 2000 of only 8000 MT, it is clear that a large fraction of the beryllium resource will be available for fusion applications.

The beryllium requirements for the reference fusion breeder are given in Table VIII and are compared with the resource estimates in Table IX. Without recycle, the beryllium throughput is so large that beryllium makeup requirements can seriously impact resource availability. Beryllium will have to be recycled and losses cannot exceed a few percent for long term service (e.g., over 200 years).

Estimates of the electric power generation capacity which can be supported by half of the nominal domestic resources are shown in Table X. These estimates indicate that, with efficient recycle, fusion breeders using beryllium multiplier blankets can easily support over 1000 GW_e of electric power production (including their own) for over 200 years into the future - a period roughly corresponding to the period from the beginning of the industrial revolution until today.

III.D.2 Pebble Design Considerations

Three beryllium/thorium composite fuel candidates are shown in Figure 16. Each has advantages and disadvantages. Candidate A, a solid thorium pin, which is nested in the core of the beryllium pebble using alloy steel clips, has the advantage of separating the beryllium and thorium (small lithium filled gap), thus limiting possible intermetallic compound (Be_{13}Th)

Table IX.
Summary of Contributions to Beryllium Requirements
for the Reference Fusion Breeder

	Annual (MT/yr)	30-Year Life Cycle (MT)	As Fraction of Nominal Domestic Resource (%)
Initial inventory	--	450	0.30
Burnup	0.4	11	0.01
Makeup (1.65 yr life)			
No recycle	265	7,950	5.30
Recycle (7% loss)	19	560	0.37
Recycle (1% loss)	2.7	52	0.05

Table X.
Results of the Beryllium Resource Assessment
for Fusion Breeder Applications
[U.S. Electrical Capacity Which Can Be Supported by
Domestic Resources (GW_e)]

Beryllium Lifetime (yr)	Fusion-Fission Economy Duration (yr)	Without Recycle	With Recycle Loss of		
			7%	1%	0%
1	30	131	1026	1824	2095
	60	69	679	1065	2079
	200	21	262	1030	2013
2	30	255	1645	2535	2785
	60	133	1164	2307	2760
	200	41	490	1624	2640
5	30	587	2583	3312	3475
	60	320	2040	3128	3434
	200	104	1030	2486	3252
10	30	1030	3186	3687	3786
	60	593	2723	3549	3738
	200	200	1624	3020	3524

	Volume	Mass
Beryllium	10.4	19.1 (at 1.84 g/cm ³)
Thorium	3.7	43.3 (at 11.7 g/cm ³)
Total	14.1 cm ³	62.4 g

Figure 18.

The Beryllium/Thorium Pebble Modified to Accommodate the
Increased Thorium Fraction of the Current Reference Design.

formation between the two metals.⁽²⁾ However, the large heat generation in the thorium pin drives a temperature gradient from the center to the outer surface of the beryllium pebble, possibly shortening the beryllium lifetime by causing differential swelling induced stresses.

The scroll pin fuel form (candidate B) has the advantage that the steel clips (which raises mass transfer issues of their own) are eliminated, but does not seek to inhibit the Re-Th interaction and continues to drive heat through the beryllium pebble.

The snap-ring fuel form (candidate C), which was selected as a baseline, also places beryllium in contact with thorium, but maintains a very uniform temperature profile in the beryllium pebble (see Figure 17). If Re-Th interactions are shown to be excessive, then a diffusion barrier (e.g., molybdenum on thorium or TiC on beryllium) should be investigated. It would also be possible to adapt the alloy spring to an outside thorium ring. The beryllium/thorium fuel element, sized for current reference conditions, is shown in Figure 18. The pebble consists of 27 volume percent thorium in a 3.5 mm thick ring. As shown, the maximum temperature difference for a pebble at the front of the blanket is expected to be 40°C.

III.D.3 Beryllium Lifetime

In a functional sense, the beryllium lifetime must exceed the fuel residence time (315 full power days in inner fuel zone, 630 full power days in outer zone). These minimum requirements imply a radioactive beryllium recycle rate of 265 MT/yr, or 14 million 3 cm O.D. pebbles per year or 0.5 pebbles/second. This requirement is within the existing technological capability for automated fabrication of low tolerance parts⁽⁹⁾ and is not expected to be prohibitively expensive (roughly \$40 million in direct capital

	Volume	Mass
Beryllium	10.4	19.1 (at 1.84 g/cm ³)
Thorium	3.7	43.3 (at 11.7 g/cm ³)
Total	14.1 cm ³	62.4 g

Figure 18.

The Beryllium/Thorium Pebble Modified to Accommodate the
Increased Thorium Fraction of the Current Reference Design.

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n be amenable to
result to pebbles

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will be 40ppm and

C. The volumetric
($\Delta V/V$) and will be
a thermally induced
is conclusion also

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er and the helium

haps, 5 full power
ally, for the above
ppm and temperature
ortant above about

Figure 19.

New Theoretical Models of Beryllium Swelling
versus Temperature and Fluence.

350°C. In such cases, it is believed that the core of the pebble will swell more, expanding the outer skin until it cracks due to fracture toughness limits. Very conservative models of this mechanism have been used to estimate an average-over-blanket beryllium lifetime in excess of 3.4 full power years (over twice assumed lifetime), assuming a 50 percent stress relaxation due to thermal and/or irradiation creep. Higher fracture toughnesses than assumed ($12 \text{ MPa} \cdot \text{m}^{1/2}$), or stress relief due to shallow cracking are additional mechanisms which could further extend the beryllium lifetime. This failure mechanism is highly interactive and requires in-core fission testing to better understand the failure mode.

III.E. Reactor Safety

In this section, reactor safety considerations for the reference fusion breeder blanket are reviewed. Safety goals and objectives are discussed, the features of several reactor safety systems are defined, and the results of a preliminary probabilistic risk assessment are discussed. It should be noted that the later activity was performed with an eye towards identifying those accident scenarios which deserve increased attention rather than as an exercise to provide an absolute estimate of the integrated risk. Safety topics are described in more detail in References 2, 5, and 6.

III.E.1 Safety Requirements and Goals

In comparison with fusion-electric power generation, fusion breeders are characterized by higher radioactivity inventories and higher decay afterheat levels. However, fusion breeder safety must be considered in the context of the overall fusion-fission electricity generation. Specifically, the following system level features should be noted:

- The system would be dominated by fission reactors.
- The fusion breeder could be located far from population centers in dedicated fuel cycle centers.

Thus, a logical overall requirement for fusion breeder safety is that it not detract from the safety of the entire fission reactor dominated system. In practice, this implies that the integrated risk (measured in Rem/kWh_e-H) should be roughly equal to, or less than that of the client fission reactors (e.g., LWRs). In contrast, fusion-electric reactors, which are likely to be located near population centers and are likely to be more expensive than fission reactors, must provide demonstrated safety and environmental advantages. These advantages can include some combination of passive (or inherent) safety, low radio-nuclide inventory, low activation structural materials, etc.

Although safety requirements are relaxed for the fusion breeder relative to the fusion-electric application, ambitious safety goals have been established. These include the use of multiple, redundant safety systems and the use of one or more passively cooled fuel dump tanks as an ultimate safety system.

III.E.2 Safety Issues, Design Features, and Subsystems

The leading causes for concern regarding blanket safety are those mechanisms by which "stored energy" can result in the release of radioactive products into the environment. Four potential sources of stored energy have been identified.

- Nuclear energy which might be released if the reactivity of ²³³U in the thorium fuel were to increase.

III.E.2.h Pebble Expansion

A rapid increase in the blanket temperature could lead to a rapid increase in beryllium volume which could conceivably lead to a fuel blockage and/or rupture of the contents of one or more blanket modules. As indicated in Figure 19, this concern may not be important for the reference conditions (approximately 5600 appm (maximum)) because the net linear expansion for a blanket temperature excursion to 900°C (for which structural integrity is assumed to be lost in any case) would result in a linear growth of less than 0.5 percent, or only 0.2 cm across the entire blanket zone. Pebble expansion would become a serious issue if an ~5 yr beryllium lifetime were specified.

III.E.2.c Lithium Fire

The consequences of a lithium fire could be the failure of an overheated structure, with the consequent release of volatile oxides and other radioactive constituents. Although the potential for lithium fires is a concern, the design philosophy is to borrow techniques for active and passive liquid metal fire protection which have been developed for the sodium cooled LMFBR. A number of engineered safety features can be invoked, including the following:

- Use of an inert gas environment.
- Use of steel-lined concrete chambers with sacrificial material between the steel liner and the concrete.
- Use of deep, narrow sumps with sloped surfaces leading to dump tanks which collect any spilled lithium in the reactor building.
- Use of steel balls and hollow graphite microspheres in spillage areas to rapidly cool and choke any spilled lithium.
- Active chemical fire fighting techniques.

* (REQUIRED NUMBER OF COMPONENTS/NUMBER OF COMPONENTS AVAILABLE)

Figure 20.

Fusion Breeder Primary Loop Logic.

- Heat conduction to a cooled shield.
- Heat conduction through the first wall using an auxiliary first wall cooling system.
- Heat removal following semi-passive fuel dump to a cooled tank below the blanket.

The first and second of the above heat flow paths are available in both LOCF and LOCA situations. Decay heat removal using the shield involves the use of an independent shield cooling system (lithium or other coolant compatible with blanket coolant) which is maintained near the blanket coolant outlet temperature of 425°C. In this case, as the blanket temperature increases beyond the outlet temperature, heat is conducted and (if the fuel and/or coolant are absent) radiated to the cooler shield. An independent capability to discharge the shield heat to the environment must be provided.

The auxiliary first wall cooling system is assumed to consist of two large fans (i.e., for redundancy) located near the ends of the central cell. In the event of an accident, these fans (which are comparable in power rating to small airplane engines), would circulate the reactor building cover gas through the central cell to cool the first wall (maximum required velocity approximately 50 m/s).

The third mechanism for removing decay afterheat is to dump the contents of the blanket to a separately (and probably passively) cooled dump tank located below the blanket. As shown in Figure 3, one such tank might be provided for each five blanket modules. The same tank would also be used for fuel management operations. The act of dumping can be performed semi-passively. That is, a modest refrigerator coil under each module can be used in normal operation to freeze a solid lithium plug (m.p. = 190°C) in the dump line. If the fuel is to be dumped, a loss of refrigerator power will cause

Table XI.

Safety System Capabilities for Various Accident Scenarios

	Shield Cooling Alone	Auxiliary First Wall Cooling Alone	Both Systems
Loss of coolant flow, fuel not dumped	Adequate	Not adequate	--
Loss of coolant flow, fuel dumped	Adequate	Adequate	--
Loss of coolant, fuel not dumped	Not adequate	Not adequate	Marginal ^a
Loss of coolant, fuel dumped	Adequate	Adequate	--

^aExceeds 730°C limit for reuse in approximately 2 hours. Blanket intervals temperature may exceed 900°C limit for structure integrity if steel emissivity is less than 0.5. Further analysis required.

- Capabilities of safety systems.
- Component failure probabilities (e.g., valves).
- Event trees leading to releases.
- Radiological consequences of release.

Previous fission reactor experience and analysis were used to develop many of the expected probabilities and consequences.

One event tree, starting with a first wall failure, is shown in Figure 23. As shown, if the primary safety systems (dump valve, module isolation valves, shield coolant) function, the consequence is small, but the probability (1%/year) is relatively large because the first wall is expected to be a weak link in the system. The branches with the highest consequence are D8 and D10, but these require secondary containment failure, have probabilities in the range of $c < 10^{-9}\text{yr}^{-1}$ per year and are not considered to be credible. The branches with the next highest consequences are D7 and D9. In both cases the failed module is isolated, but the dump valve fails, and either the shield cooling system or the auxiliary first wall cooling system fails. Consequently, the radioactivity associated with one module would be released to the containment building.

Three other event initiators were assessed: (1) module inlet/outlet distribution plenum failures, (2) failures of all primary coolant pumps or intermediate heat exchangers, and (3) failures of the inlet/outlet piping or the cold or hot manifolds. The overall results are shown in Figure 24. The first wall failure (branch D1) discussed above is expected to provide the highest overall risk (probability x consequence). Other high risk branches involve an inlet/outlet piping failure (R1) and a complete coolant pump/IHX failure (A7). The piping failure is similar to the first wall failure but lower probabilities and higher consequences are assumed. The pump/IHX failure is a low probability-high consequence event that might be made to be inconceivable via use of emergency diesel backup generators.

It is important to emphasize that the PRA technique is viewed as a useful method to identify weak links in the system (e.g., the first wall), to set goal requirements (e.g., failure probabilities), and to establish a preferred safety system logic. For example, the severity of branches 07 and 09 of Figure 23 is greatly reduced by adding the shield and first wall coolant systems. This is indicated in Figure 24. The PRA is not expected to provide

IV. FUSION BREEDER PLANT CONCEPT

Although the fusion breeder program has emphasized the design of breeding blankets and associated technologies, there has also been an attempt to describe and cost the entire plant. This integrated description of the plant cost and power flow can be used to provide a best estimate of the ultimate commercial merit of the fusion breeder application.

The plant design information developed in the section has been used to generate the systems and economics analysis described in Section VI.

IV.A Plant Overview

In this section, a top level description of the fusion breeder plant is given. The reader is reminded that the fusion reactor, which is central to the plant description, is nearly identical to that of the Mirror Advanced Reactor Study (MARS) described in detail in Reference 10. The MARS end plug, shown in Figure 25, has not been changed, although it is clear that the more recent octupole end-plug designs currently being pursued as part of the MINIMARS effort at LLNL (12), would reduce the capital cost of the fusion breeder.

The principal difference between the MARS and the reference fusion breeder plant (2) involve the nuclear, power conversion, and reactor safety systems of the plant. Although the central cell length and magnetic field strength are preserved, the use of the lithium cooled fusion breeder blanket design (MARS would be cooled using the Pb-Li eutectic) leads to changes in the following areas:

- Blankets would be replaced by a horizontal, rather than vertical, translation. Thus, as indicated in Figure 26, all heat exchangers, pumps, valves, etc. would be located on one side of the central cell, preserving the other side for maintenance operations.
- Recognizing the potential for the lithium-water reaction, water would be eliminated from the central cell building and a reasonably fast acting isolation barrier between the central cell and plug would be provided.
- As shown in Figure 26 and discussed in Reference 2, Li/Na intermediate heat exchangers and a sodium intermediate loop would be provided.
- A nuclear grade containment building and the safety/fuel management systems discussed in Sections II.A, III.A, and III.E would be provided. These would include provision to convey spent thorium and beryllium parts directly to adjoining process plants for reprocessing and refabrication, respectively. The latter facilities are also considered to be part of the fusion breeder plant (see Section V), but could be shared among more than one fusion breeder.

The above changes increase the plant cost relative to MARS and are reflected in the cost estimates of Section IV.C

IV.B Integrated Power Flow Summary

The integrated power flow associated with the reference fusion breeder is summarized in Figure 27. In this figure, the plasma heating systems (RF heating, neutral beams) are lumped into one system providing 100 MW of power

Figure 27.

Fusion Breeder Average Integrated Power Flow Summary [pph-plasma heating wall plug efficiency, pph-plasma heating RF power, Q-plasma energy gain, PL thermal-power loss to system, PBL-blanket thermal power, η_t - gross thermal conversion efficiency, PTC-electrical power from thermal conversion, PDR power to direct energy recovery system, η_{ER} -direct energy recovery efficiency, PER-direct energy recovery electrical power output, PERT-recovered thermal power from direct energy recovery system, PREC-recirculating electrical power, PAUX-auxiliary (other) systems electrical power requirement, PPH-plasma heating electrical power, Pg-gross electrical power, PNET-net electrical power (ave. over cycle)].

and n , the scaling exponent varied between 0.6 and 0.7, depending upon the particular subsystem.

A comparison of the MARS and fusion breeder direct cost estimates by engineering cost account is provided in Table 12. Some notable differences include the following:

- Reactor Building (Acct 21.01)- Cost increased by 55 \$M to provide nuclear grade containment
- Miscellaneous Building (Acct 21.06)- Cost increased by 48 \$M to provide beryllium/thorium fabrication buildings
- Fusion Systems (Acct 22.01)- First wall/blanket/shield and magnet costs increased by 505 \$M to provide larger first wall radius and 253 m³ of beryllium pebbles at a unit cost of $8.9 \cdot 10^5$ \$/m³ (480 \$/Kg)
- Main Heat Transfer Components (Acct 22.02)- Cost increased by 223 \$M to provide higher power and intermediate sodium loop
- Other Plant Equipment (Acct. 22.06)- Cost increased by 442 \$M to provide large coolant dump tanks, dump tank safety systems, the Re/Th fuel handling system, beryllium and thorium fabrication equipment, and a pyro-chemical fuel reprocessing plant.
- Turbine Generator Plant (Acct. 23)- Cost increased by 128 \$M to provide higher power level

Table XII. System Cost Summary (Continued)

NUMBER	ACCOUNT TITLE	MARS	FUSION BREEDER
22.01.09	Direct Converter	23	23
22.02	Main Heat Transfer (incl. Primary loop, pumps, heat exchangers)	237	460
22.03	Cryogenic and Auxiliary Cooling	50	47
22.04	Structure and Coolant Radioactive Waste Treatment	11	12
22.05	Tritium Handling and Storage	46	27
22.06	Other Plant Equipment	30	472
22.06.01	Maintenance Equipment	26	37
22.06.02	Liquid Metal Heating	1	3
22.06.03	Coolant Dump Tanks	3	11
22.06.04	Dump Tank Safety System	--	80
22.06.05	Be/Th Fuel Handling System	--	30
22.06.06	Be Fabrication	--	38
22.06.07	Th Fabrication	--	38
22.06.08	Pyro-Chemical Uranium/Thorium Reprocessing (incl. building)	--	235
22.07	Instrumentation and Control	25	25
23	Turbine Generator Plant	236	364
24	Electric Plant Equipment	130	138
25	Miscellaneous Plant Equipment	20	28
26	Special Materials	126	72
26.01	Liquid Metal Coolant	124	18

- Special Materials (Acct. 26)- Cost decreased by 106 \$M because highly enriched ^6Li required for Pb-Li coolant in MARS is not used in fusion breeder

More accurate cost estimates for the various subsystems can be developed, but will require a considerable effort to increase the design definition at the subsystem level.

As shown in Table 13, the overall direct cost increase for the fusion breeder relative to MARS is estimated to be about 1380 \$M in mid-1985 dollars. Adding 30% for indirect construction costs, 15% for contingency, and assuming 6 and 8 year construction periods for MARS and the fusion breeder, respectively, results in total cost estimates of about 3560 and 6290 \$M for the two plants. Thus, the fusion breeder is estimated to cost about 70% more than the MARS fusion-electric plant, but also would produce about 70% more electricity on average. In addition, the fusion breeder would produce about 6660 kg/yr of ^{233}U (70% capacity factor) - enough fissile fuel to provide makeup for 25 1 GWe LWR reactors!

V. FUEL CYCLE TECHNOLOGIES

The principal role of the fusion breeder reactor is to provide fissile fuel to support a fission power reactor economy composed of light water reactors (LWRs) or other fission reactors. In this role, the fusion breeder is operationally similar to a fissile enrichment plant which requires no fissile feed stream and is an electricity producer rather than a consumer. In contrast with fission breeder reactors (i.e., liquid metal fast breeder reactor (LMFBR) and light water breeder reactor (LWBR), the neutron rich fusion breeder is a subcritical assembly, produces an order of magnitude more net excess fuel per unit of thermal power, and is not subject to the neutron balance constraints of conventional fission reactors. As a result, a wide variety of fuel cycles and fuel forms are possible.

In this section, several candidate fuel cycle and technology options for fusion breeders and LWR's are briefly reviewed. This review is followed by a more specific discussion relating to the reference fusion breeder fuel cycle. Although not discussed here, a process description and cost estimate for beryllium pebble manufacture/recycling is provide in Reference 2. Conceptual plant designs and costing analysis relating to aqueous (PUREX-/THOREX), pyro-chemical and molten salt fuel reprocessing facilities for the fusion breeder discharge fuel are presented in References 2,6, and 20. Cost estimates for other fuel cycle operations (e.g., tritium removal from lithium) are also provided in Reference 2.

V.A. Fusion Breeder Fuel Reprocessing Issues

A general schematic of the fusion breeder/fission burner reactor fuel cycle is shown in Figure 28. Several features are deserving of note. Most importantly, the overall fuel cycle is separable into two distinct fuel cycles

which are coupled by the flow of bred fissile material from the fusion breeder to the fission converters. With the exception of issues which bear upon the type of converter fuel fabrication plant required (e.g., the amount of ^{232}U in the bred ^{233}U), the breeder and burner fuel cycles are entirely separable.

Also, both the fusion breeder and fission client fuel cycles are closed by reprocessing and recovery of fissile materials. Although direct enrichment (or "refresh") fuel cycles, which do not employ reprocessing, have been examined in the past, these lead to inefficient fissile production in the fusion breeder as well as the disposal and loss of large quantities of valuable fissile resources. Our results indicate that, like the LMFBR, the fusion breeder requires a closed fuel cycle to achieve adequate economic performance.

In developing the reference fusion breeder concept, two candidate thorium fuel reprocessing technologies have been considered. A key concern regarding the use of a conventional aqueous chemistry (THOREX) fuel reprocessing plant to extract ^{233}U from thorium metal fuel is cost. In the fission-suppressed fusion breeder fuel cycle, fissile fuel is discharged at very low fission burnup ($\sim 800 \text{ MWd/MTM}$) and at low concentration ($\sim 1\%$ fissile material in thorium). Low burnup, an advantage, leads to lower radioactivity in the discharged fuel with favorable impact on fuel reprocessing and fabrication processes. The low discharge concentration, leads to high unit costs (i.e., $\$/\text{gm}$) to recover the bred fuel. This situation is illustrated in Table 14, which shows typical reprocessing cost data allowing a comparison of the cost to recover fissile fuel using the THOREX and a pyro-chemical reprocessing options. It is important to note that a typical ^{233}U production cost not including reprocessing is $50 \text{ } \$/\text{g}$. Therefore, THOREX reprocessing could account for as much as $1/3$ of the cost of bred fuel. It should also be noted

that the thorium throughputs shown in Table 14 are applicable to full scale reprocessing plants that each process output of three fusion breeders.

In comparison with the aqueous fuel reprocessing technologies, the more compact pyro-chemical fuel reprocessing technology for thorium metal fuels is expected to result in a more tolerable contribution to the cost of bred fissile fuel. Although the development program required to assure the feasibility of this technology is more of a concern, the pyro-chemical reprocessing technology offers an economic incentive in the range of 100 \$M/yr-a strong motivation for its development.

V.B. Overview of LWR Fuel Cycles and Issues

Light water moderated power reactors are expected to dominate nuclear power production well into the future. Consequently, the LWR is considered as the principal type of fission client reactor, but the discussion is also applicable to more advanced fission converters (modular HTGR, etc.). The cores of these reactors would be modified to provide passive safety or increased fuel economy, but fissile makeup and inventories will continue to be required.

If the fuel bred in the fusion breeder is ^{233}U , three fuel cycle options (30,31) are available:

- The thorium fuel cycle (typically 3.4% ^{233}U , 96.6% ^{232}Th)
- The denatured thorium fuel cycle (typically 3.3% ^{233}U , 18% ^{238}U , 79.7% ^{232}Th)
- The denatured uranium fuel cycle (typically 2.4% ^{233}U , 97.6% ^{238}U)

Among these, the thorium fuel cycle is most efficient with respect to fissile

Table XV. LWR Fuel Cycle Performance Data

	LWR FUEL CYCLE TYPE					
	Pu Burner	Denatured Uranium (DU)	Denatured Thorium (DT)	DU + Pu (a)	DT + Pu (b)	²³⁵ U Burner
Net Fissile Requirement, g/KW-yr (c)	0.200	0.205	0.144	0.153	0.126	0.194
Excess Plutonium Production, g/KW _e -yr(c)	none- recycled	0.068	0.028	none- recycled	none- recycled	none- recycled
Fuel Burnup, MWD/MTHM	30,400	33,000	33,400	32,300	33,000	30,400
Equilibrium Fissile Enrichment, atom %	4.9	2.4	3.3	3.1	3.5	3.2
Core Power Density, kW _t /kgHM	37.1	37.2	40.2	37.2	39.8	38.4
Equilibrium Fissile Inventory, g/KW _e	1.83	0.68	1.09	1.07	1.29	1.14
Net Thermal-to-Electric Efficiency, %	33.4	33.4	33.4	33.4	33.4	33.4

- a) mixed system includes 74% denatured uranium fuel cycle LWRs and 26% plutonium fuel cycle LWRs
- b) mixed system includes 88% denatured thorium fuel cycle LWRs and 12% plutonium fuel cycle LWRs
- c) at 100% plant capacity factor

VI. ECONOMICS

In this section, the results of two models of the economic performance of the symbiotic shown in Figure 29 are reviewed. The symbiotic system is also compared with a conventional LWR fuel cycle (235) Burner of Table 15) to determine the "indifference", or "breakeven" cost of U_3O_8 . When the market cost of mined U_3O_8 reaches the breakeven cost, the fusion breeder is expected to provide an economic benefit.

VI.A System Average Capital Cost Model

All of the economic models of symbiotic electricity generation systems share a common feature: they recognize that including the fusion breeders and all of the client LWRs, only one product, electricity, is produced. The first and most simple model attempts to estimate and compare the average capital cost (per unit electricity generation) of the symbiotic system with that of an LWR alone. This model ignores all fuel cycle facilities and fissile inventory charges, but provides an excellent measure of the dominant features of the system. In this model, the average capital cost, in LWR units is

$$ACC = \frac{(1 \times C^F) + (R_t \times 1)}{(1 \times \eta_{rel}) + (R_t \times 1)} = \frac{C^F + R_t}{\eta_{rel} + R_t}$$

where C^F is the fusion breeder cost relative to the LWR cost (basis: \$/KW_t), R_t is the fusion breeder thermal support ratio and η_{rel} is the relative thermal efficiency of the fusion breeder relative to the LWR.

For example, the fusion breeder cost (not including its pyro-chemical reprocessing plant or other fuel cycle facilities) is 5596 \$M and its average thermal power is 5600 MW_t. Its cost relative to a 1 GWe, 1330 \$M LWR of 3000 MWt power, C^F , is $(5596/5600) / (1330/3000) = 2.25$. The fusion breeder

of electricity which reflects the overall capital, operating, and fissile inventory charges of the symbiotic system.

A second feature of the PERFEC model is that it allows a year-by-year comparison with an alternative LWR fuel cycle in which the fissile feed comes from mined U_3O_8 . Three types of comparison are of interest:

- year one breakeven price of U_3O_8 .
- 30 year breakeven price of U_3O_8 .
- 30 year integrated benefit.

In the first comparison, the required price of U_3O_8 (in 1985 dollars) such that the fusion-fission system provides electricity at the same cost as the U_3O_8 - fueled LWR in the first year operation is calculated. When the actual market price of U_3O_8 (again, in 1985 dollars) reaches this level, the fusion breeder becomes economical in the first year.

If, in later years, the cost of U_3O_8 increases with inflation, then the comparison becomes increasingly favorable to the symbiotic system during the second and later years. The second of the above comparisons, determines a lower year one cost of U_3O_8 that, inflated in each of 30 years, provides a zero net benefit over the entire period. Thus, the 30 year breakeven cost of U_3O_8 does not provide breakeven in the first year (the fusion breeder loses revenue), but the discounted present value of excess revenues exceed breakeven during the second half of the thirty year period by enough to compensate for early losses.

The third type of comparison is intended to evaluate the present value of a thirty year net benefit (or loss) starting with a user input cost of U_3O_8 in year one and assuming a U_3O_8 escalation rate (e.g., 55 \$/kg and 3%/yr). If

Table XVI. Economics Results for a U_{308} Fueled LMR with Reprocessing

LMR OWNER	UTILITY
LMR Capital Cost (\$/kW _e)	444
Total Fixed Charge Rate (%/yr)	13.2
Fuel Cycle	Full Recycle
^{235}U Consumption (g/kW _e -yr)	0.194
Average Burnup (MWD/MTUW)	30,400
Reprocessing Cost (\$/kgU)	417
Fabrication Cost (\$/kgUW)	685
Enrichment Cost (\$/kg separative work)	60
Transport and Disposal Cost (\$/kgUW)	260
Year One Price of Purchased U_{308} (\$/kg)	55
Year One Cost of Electricity (mil/kW _e h)	52
U_{308} Escalation Rate (%/yr)	3
Average PV Cost of Electricity (mil/kW _e h)	33

Table XVII. Economics Results for the Symbiotic System

	Case 1	Case 2
Fusion Breeder Owner/Operator	Govt	Utility
Fusion Breeder direct cost (\$/kW _t)	604	604
Fusion Breeder total cost (\$/kW _t)	1015	1121
Total fixed charge rate on fusion breeder plant capital (%/yr)	6.3	13.1
Fusion Breeder annual capital cost (\$/kW _t -yr)	64.2	147
Fusion Breeder annual operating cost (\$/kW _t -yr)	34.0	34.0
Fusion Breeder annual fissile inventory cost (\$/kW _t -yr)	0.2	13.6
Fusion Breeder total annual cost (\$/kW _t -yr)	98	195
LWR Owner/Operator	Utility	Utility
LWR direct cost (\$/kW _t)	250	250
LWR total cost (\$/kW _t)	444	444
Total fixed charge rate on LWR plant capital (%/yr)	13.2	13.2
LWR annual capital cost (\$/kW _t -yr)	58.6	58.6
LWR annual operating cost (\$/kW _t -yr)	30.1	30.1
LWR annual fissile inventory cost (\$/kW _t -yr)	0.5	11.9
LWR Total Annual Cost (\$/kW _t -yr)	89.2	100.6

As shown in the table, for government ownership, the thirty year breakeven price of U_3O_8 is only about 13 \$/Kg, or about 1/3 of the current price. Indeed assuming that the mined uranium price is 55 \$/Kg, the government owned fusion breeder would breakeven in its first year of operation and would provide an integrated benefit of about 1230 \$/KWh_e, or about 7 billion dollars over 30 years! For utility ownership, the thirty year breakeven price of about 200 \$/Kg is considerably higher than the currently depressed price of uranium. However, this price is not inconsistent with prices based upon projected mined uranium forward costs for the 2020-2040 timeframe (30), which assume that nuclear power develops an increasing share of the electricity generation market and that the uranium price/cost ratio exceeds about 1.5 (historically closer to 2). Finally, it should be noted that the difference in year one ^{233}U costs between the government and utility ownership cases, about 56 \$/g, translates to only a 2.3 mil/KWe LWR electricity cost difference! Thus, even a utility owned fusion breeder can place a very reasonable upper bound on the cost of LWR electricity, as well as on the cost of mined U_3O_8 and the allowed cost of competitive technologies (such as fusion-electric power generation and the LMFBF).

VII. SUMMARY, CONCLUSIONS, AND RESEARCH NEEDS

VII.A. Background

A reference liquid metal (lithium) cooled fusion breeder blanket design for the tandem mirror was developed during 1982.⁽²⁾ Special issues associated with this design were investigated during 1983-1984⁽⁵⁾. Subsequently, an updated reference design was developed and is described in the previous sections. The design is in many ways similar to liquid metal cooled fusion-electric blanket designs which have been developed during the same timeframe⁽¹⁶⁾.

VII.B. Advantages/Disadvantages

The liquid metal cooled blanket is of interest because

- It offers high net ^{233}U breeding, (1.7 Kg/MW_t-FPY),
- It is composed of reasonably conventional materials (ferretic steel structure, thorium metal fuel),
- It provides excellent tritium retention,
- It utilizes a lithium coolant technology which is similar to the LMFBR Sodium coolant technology. Lithium technology is also being developed for space power applications (e.g., SP-100).
- It results in a less complex internal structure than the principle alternative coolants (helium, water).

This design was the first to feature a mobile fuel element in the form of a beryllium/thorium composite pebble. The inner zone pebbles would be circulated through the blanket at a regular interval of about 0.9 full power year (FPY). This circulation would provide an economically recoverable ^{233}U enrichment of about 1%, while also limiting the blanket module power swing to a tolerable variation (+30% over one module, or +12% over the entire set of modules via the

use of fuel management techniques). The design provides for dumping of the entire blanket contents to a passively cooled dump tank should the need arise. This is viewed as a potentially important safety advantage, which was not achievable in earlier designs with fixed, conventional, fuel elements.

Several potential disadvantages result from the use of the liquid metal coolant. Specifically, a pivotal issue for this design is the MHD pressure drop. It is conceivable that this type of design can be acceptable for use in tandem mirrors, but not in tokamaks, where the magnetic fields and neutron wall loads are expected to be higher. Also, isolating the lithium coolant from water cooled systems appears possible in tandem mirrors, but may not be possible in tokamaks, which require high heat flux components in close proximity to the blanket.

VII.C. Technical Issues

Several key issues which have been identified for the updated reference design fall in the following areas:

- MHD Pressure Drops- Pressure drop due to packed bed and duct flow. Prospects for developing radiation resistant electrical insulators which would eliminate current paths and resulting pressure drops,
- Liquid Metal Compatibility- Intermetallic reactions, mass transfer, fission product/actinide transport. Prospects for developing beryllium and/or thorium coatings which would limit transport,
- Beryllium Irradiation Damage- Irradiation creep, swelling, ductility, and resulting pebble failure modes,
- Pyro-chemical Reprocessing- Reduction to practice of magnesium dissolution process,

- Pebble Flow- Issues such as self-welding and thermal ratcheting causing potential for flow blockage,
- Lithium Safety- Potential for interactions with water cooled components.

All of the above issues, with the exception of pyro-chemical reprocessing and beryllium pebble flow must also be addressed (although with slightly different emphasis) in the development of liquid metal cooled fusion-electric blankets⁽³⁵⁾. In the case of fusion breeder blankets, substantial relaxations can be obtained due to lower neutron wall loads and neutron fluences (1.7 MW/m^2 in the subject design versus 5 MW/m^2 in recent fusion-electric designs)⁽¹⁶⁾ and lower maximum coolant operating temperatures (425°C versus 500°).

Importantly, it is believed that a considerable level of confidence in this technology can be obtained prior to the year 2000⁽³⁵⁾, without the use of an expensive fusion test platform. Specifically, it is expected that nearly all nuclear irradiation related aspects of the design can be demonstrated in an integrated, in-core fission loop (10 cm tube in high flux region of a test reactor) containing a lithium coolant and the beryllium/thorium pebbles. Similarly, the MHD flow and pebble flow related aspects of the design can be demonstrated in a prototypical, but non-nuclear blanket mockup constructed from the correct materials.

Because the neutron wall load and operating temperature are relatively low, fusion irradiation damage may not be a critical issue. A 5 calendar year blanket lifetime in the 1.7 MW/m^2 fusion breeder blanket would produce a first wall displacement damage level of less than 100 dpa. Based upon the results of models derived from previous irradiations, this end-of-life level of displacement damage

(and helium production) is believed to be below the swelling threshold in either ferritic (HT-9) or austenitic steels (PCA). Thus a high, level of confidence regarding materials performance can be assumed.

VII.D. Development Plan

A development plan for the lithium cooled fusion breeder blanket developed for the DOE - sponsored Technical Planning Activity (TPA)⁽³⁵⁾ is shown in Figure 30. The shaded areas of the plan indicated those issues which are unique to the fusion breeder blanket. The unshaded areas have a large degree of commonality with the required developmental activities for liquid metal cooled fusion-electric blankets. The emphasis in the plan is to use two partially integrated test facilities (PITF) to demonstrate critical technology areas prior to fully integrated testing in a high duty cycle fusion test reactor.

The first PITF, designated TMIF for Thermo-mechanical Integration Facility, would provide a full scale, prototypical, non-nuclear environment with respect to MHD flow; mass transfer in a heated forced convection loop; and packed bed pebble flow.

The second PITF, designated NIF for Nuclear Integration Facility, would provide a prototypical nuclear environment via an incore lithium loop in a fission test reactor (e.g., ETR, FFTF). This reactor loop could be used to simulate an actual production environment including fissile breeding, hot fuel handling operations, tritium removal, radioactive mass transfer, and on-site pyrochemical reprocessing (similar to EBR-II).

Other facilities shown in the diagram are as follows:

- Liquid Metal Flow Facility (LMF) - Smaller scale facility required to develop understanding of liquid metal MHD flow and heat transfer in complex, but controlled geometries,

Figure 30

Near Term Development Program for Liquid Metal Fusion Breeder

- ALEX - Existing ANL facility used to develop instrumentation and study MHD flow at low field in simple geometries,
- MHD Mass Transfer Facility (MHDM) - Facility to explore MHD heat and mass transfer issues in a forced convection loop,
- Pebble Lifetime Simulation (PLS) - In-core fission capsule test to simulate end-of-life beryllium/thorium pebble failure modes.

VII.E. Overall Evaluation

The lithium cooled fusion breeder blanket is a reasonable attractive and conservative design whose key issues can be resolved over the next 15 years via a non-fusion test program. It is conceivable that this type of design can be acceptable for use in tandem mirrors, but not in tokamaks.

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