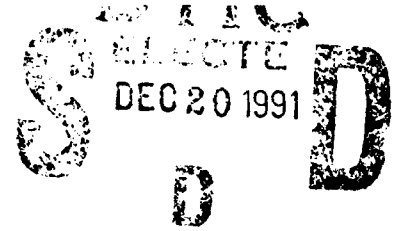


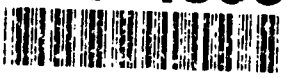
GA-A18658
UCRL 21073
UC-420, UC-424, UC-712

FUSION APPLICATIONS AND MARKET EVALUATION (FAME) STUDY

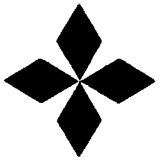
TECHNICAL REPORT

by
R. F. BOURQUE, K. R. SCHULTZ
and PROJECT STAFF

Prepared Under
Subcontract 8236305
for Lawrence Livermore National Laboratory



DATE PUBLISHED: FEBRUARY 1988



GENERAL ATOMICS

DISCLAIMER

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
P.O. Box 62
Oakridge, TN 37831

NTIS Price Codes
Printed Copy A08
Microfiche A01

GA-A18658
UCRL 21073
UC-420, UC-424, UC-712

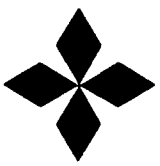
FUSION APPLICATIONS AND MARKET EVALUATION (FAME) STUDY

TECHNICAL REPORT

by
R. F. BOURQUE, K. R. SCHULTZ
and PROJECT STAFF

Prepared Under
Subcontract 8236305
for Lawrence Livermore National Laboratory

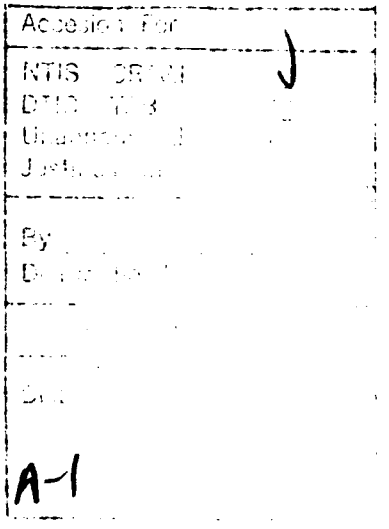
GENERAL ATOMICS PROJECT 3441
DATE PUBLISHED: FEBRUARY 1988



GENERAL ATOMICS

PROJECT STAFF

Program Manager	–	Sam Berk, DOE/OFE
Project Manager	–	Ralph Moir, LLNL
Technical Contributors:		
Robert Bourque	–	General Atomics
Edward Cheng	–	General Atomics
Barney Engholm	–	Consultant to General Atomics
Ralph Moir	–	LLNL
Michael Shaffer	–	General Atomics
Kenneth Schultz	–	General Atomics
Clement Wong	–	General Atomics
Technical Typist:		
Sandy Zepik	–	General Atomics



ABSTRACT

This report discusses the known applications of fusion energy and estimates possible markets. Because fusion reactors can generate surplus neutrons (10 times greater than fission), other valuable products can be made besides thermal energy for electrical power. The purpose of this study was to explore the many other products that could result from neutronic interactions, from the volumetric nature of nuclear heating, and from the utilization of electromagnetic and charged particle energy.

Even with other products being sold, electricity is the major product of the neutron thermal energy and sale of it is generally required for acceptable economics. Other products that are either unique to fusion or can be generated in great quantities that are otherwise unattainable are: (1) fissile fuels; (2) tritium, (3) radioisotopes, especially ^{60}Co, and (4) some rare metals. In particular, the market for ^{60}Co is expected to grow substantially as the food irradiation industry matures. To a limited extent, the fusion neutrons might also be used to transmute fission waste and for radiation testing sources.

The thermal energy from fusion reactors can be used for nonelectrical applications such as synthetic fuel production, industrial process heat, and district heating. Inherently safe reactor designs with low activity materials could be sited near the thermal energy user. Finally, in the long term, the high energy content of fusion fuel makes it an interesting possibility as a power source for deep space missions.

Contents

Figures

Tables

1. INTRODUCTION AND SUMMARY

The Fusion Applications and Market Evaluation (FAME) Study was intended to explore the potential applications of fusion energy, with emphasis on review and evaluation of applications of fusion beyond generation of central station electric power.

1.1. INTRODUCTION

Thirty years of research has gone in to developing fusion energy primarily for central station electricity production and secondarily for fissile fuel for fission electricity producers and other products. Not addressed in any depth has been the possibility of other products being the dominant output from fusion. That is the object of this preliminary study.

Fusion reactions produce thermal energy and neutrons. The thermal energy can be converted to electricity or synthetic fuels, or sold as high-grade, nonpolluting process heat. The neutrons can interact with low-cost materials to breed more valuable substances such as tritium, fissile fuel, ^{60}Co, numerous other radionuclides, and rare metals. In principle at least, the neutrons can transmute highly radioactive fission wastes into more tractable substances. Microwave radiation from a reacting plasma can be guided and used as volumetric process heat. On a more futuristic vein, fusion energy may be the optimum choice for station power and thrusting for long space journeys in large space ships because the mass contribution of fuel is very low and there may be virtually no limit to the specific impulse of a fusion rocket.

The objective of this Fusion Applications and Market Evaluation (FAME) was to investigate, evaluate, and summarize the potential applications of fusion energy, and to identify promising directions for future work on fusion applications. Specific objectives were to examine and evaluate previous work done on fusion applications, explore processes

presently using fission and fossil energy sources for applicability to fusion, and explore new processes that would be unique to fusion. A further objective was to provide an economic assessment, at least parametrically, of fusion reactors producing other items besides electricity. A final objective of the project was to produce a brochure with many illustrations and relatively nontechnical text for general distribution.

1.2. SUMMARY

We considered about 30 applications of fusion. The most promising are:

- Electricity production
- Fissile fuel and tritium breeding
- Radioisotope production for irradiation sterilization
- Other radioisotope production
- Synthetic fuel production
- District and process heat generation
- Rare metals production
- Space propulsion and power

Electricity is likely to remain the dominant product from fusion reactors in terms of revenues per plant and market size. This is followed by fissile fuel production, which is an indirect route to electricity production. Synthetic fuels, either in gaseous or liquid form suitable for transportation, could be a substantial market but would not be competitive with fossil fuels unless the energy conversion efficiency approaches 80% or until fossil fuel prices climb to well over the current (1987) $20/barrel range. Existing synfuel processes are projected to have maximum efficiencies of about 50%. At 50% efficiency, fusion synfuel production could compete when oil prices exceed $50/bbl. For fusion process heat to complete will also require significant increases in fossil fuel price. With cogeneration of electricity or other products, process heat is likely to be competitive with fossil sources

when resource depletion or economic sanctions against pollution raise the cost of oil about $28/bbl.

A surprising result was the potential of ^{60}Co. This radioisotope is currently used in cancer treatment and some product sterilization and is the key element in the burgeoning food irradiation industry. Fusion reactors can produce copious amounts quickly at low cost. In fact, so much could be produced that only a few reactors would be needed and there is the risk that the market might be flooded and prices depressed.

Coproduction sometimes seemed better than producing a single product. For example, the combination of electricity, fissile fuel, and ^{60}Co provided such attractive revenues that each product could be sold at under current market.

These results were very sensitive to the assumptions made of capital costs, interest rates, and recirculating power. In general, because fuel costs are negligible, fusion reactor capital costs can be higher than fission, but only by a small margin. The recirculating power needed for magnets, current drives, etc., cannot exceed about 10% of the fusion thermal power without adversely affecting the economics.

A small but significant application is production of tritium for nuclear weapons. Inertial confinement fusion can also be used to study nuclear weapons physics and weapons effects.

There are numerous other secondary applications that do not appear to have substantial markets at present but may in the future. Examples are volumetric process heat using microwaves guided from the plasma to a heating "oven," space station electrical power and fusion rockets, transmutation of highly radioactive wastes, and radioisotope production besides ^{60}Co.

The notion of a fusion "waste burner," which transmutes fission wastes to more benign forms, does not, after detailed examination, hold much promise. In general, neutron wall loadings required are too high and residence times too long to be practical. The risks

involved in handling large amounts of radioactive wastes for fusion reactor processing are substantially greater than that involved in simple deep burial.

Fusion reactors are capital intensive, and it is expected that 75% of the cost of product will be capital payback charges. One would therefore expect fusion reactor construction to be clustered into periods of low interest rates and high capital availability. The financial stability thus gained is similar to that of buying vs. renting a house and should be considered an inherent advantage of fusion: costs of electricity or other products can be stabilized and forecast well into the future. Variations in fuel cost are not a factor. Because energy costs are a substantial component of many commercial activities, stability and predictability in such costs should provide a stabilizing effect on the entire economy.

2. COGENERATION ECONOMIC SCOPING ANALYSIS

One of the unique features of fusion reactors is the ability to simultaneously produce several products and thereby yield multiple sources of revenue. The thermal energy from fusion reactors can be partitioned to produce electricity, process heat, and synthetic fuels, among other things, while the neutrons can be independently partitioned to produce fissile fuel and ^{60}Co as major products, and other isotopes as secondary products.

2.1. INTRODUCTION

In this chapter, a simple cogeneration model is exercised to show the economic impact of multiple products from fusion reactors. Although other products can be produced, we concentrate on electricity, process heat, synthetic fuel, fissile fuel, and ^{60}Co. Some overlap with this chapter is found in other chapters of this report. The groundrules and assumptions used in this chapter are intended to give a consistent basis for the economic modeling of the various potential applications and may differ somewhat from those of the separate technical description chapters.

2.2. MODELING OF THE FUSION REACTOR

The fusion reactor is modeled generically as a thermal power source with a certain unit capital cost, recirculating power fraction, and blanket neutron and energy multiplication. Table 2-1 shows the particulars. The generic reactor is based on a tokamak or RFP with moderate power density (*e.g.*, 4–8 MW/m^2 neutron wall loading) and conventional power conversion and balance of plant. Any product breeding such as fissile fuel is assumed to occur on the outboard reactor blanket only, which occupies 65% of the first wall area.

A fission-suppressed blanket is assumed for this breeding.[1] The inboard blanket is used for high grade heat and tritium breeding only.

The thermal power shown in Table 2-1 is the power to the coolant. This is a typical value for current plants and was kept invariant here in order to reduce the number of parametric variations. The fusion power is less, depending on the overall blanket energy multiplication. Recirculating power was taken to be a fraction of the fusion power, not thermal power. This allows credit for the smaller fusion reactor when blanket energy multiplication is high.

TABLE 2-1
Generic Fusion Reactor Model

Total Thermal Power	3000 MW
Recirculating power (fraction of fusion power)	0 and 10%
Blanket options	1. Fission-suppressed breeder + pure fusion (65%/35%) 2. Pure fusion (100%)
Neutron utilization	95%
Blanket energy multiplication	
1. Fission-suppressed	2.0
2. Pure fusion	1.2
Neutron multiplication	
1. Fission-suppressed	2.1
2. Pure fusion	1.2
Pure fusion tritium breeding ratio	1.3

The thermal power can be partitioned among electricity generation (42% efficiency), synfuels (50% efficiency — about current capabilities), and process heat (100% efficiency). The useful neutrons in the breeding blanket can be partitioned among fissile fuel, ^{60}Co, excess tritium, and other radionuclides. Because of the expected small market potential for the last two, they were ignored in this analysis.

2.2.1. Costs

Unit capital costs used in the analysis are shown in Table 2-2. They were consolidated from a number of sources[2-3] with considerable intuitive judgement applied. In particular, because the cost of the fusion plant is so speculative, the results below also show the impact of doubling those costs. Values are in constant 1987 dollars.

TABLE 2-2
Cost Elements

Fusion plant cost including BOP but less blanket and power conversion	\$800/kWth blkt
Fission-suppressed blanket	\$100/kWth blkt
Pure fusion blanket	\$50/kWth blkt
Steam power conversion	\$400/kWe gross
Fissile fuel processing	\$2000/(kg/yr)
^{60}Co processing	\$2000/(kg/yr)
Synfuel processing	\$150/(kg/day)
Annual operation and maint.	2% of total \$/yr
Fixed charge rate (constant \$)	.087
Plant factor (power factor × availability)	0.70

2.2.2. Revenues

Because of the uncertainties in estimating the future price for products such as bred fuel, prices charged for products were treated parametrically. Table 2-3 shows the ranges. The units used are those common in the particular industry (note that 1.0 GJ is about 10^6 Btu).

TABLE 2-3
Range of Prices Charged

Electricity	30–90 mills/kWh
Synfuel	\$4–13/10^6 Btu
Process heat	\$3–11/GJ
Fissile fuel	\$10–80/g
Cobalt-60	\$0.25–0.50/Curie

2.2.3. Basic Equations

The analysis of coproduction is basically an accounting activity where one keeps track of neutrons and megawatts. Most of it is self-evident. Elements that are not are briefly discussed below.

The fusion neutron production rate from the plasma is given by

$$\Pi_n = \frac{0.8\ P_{th}}{M_E}\ \frac{6.242 \cdot 10^{18}\ \frac{\text{MeV/Sec}}{\text{MW}}}{14.1\ \text{MeV}}\ (\text{n/sec}) \quad , \tag{1}$$

where Π_n is the 14.1 MeV production rate in neutrons/sec, P_{th} the total reactor thermal power in MW, and M_E the blanket overall energy multiplication (P_{th}/P_{fusion}). The total useful neutrons per fusion neutron in the breeding section of the blanket which, in this study, accounts for 65% of the total blanket, is

$$N_b = M_{nb}\ (1 - f_{pf})\ (1 - f_{ABS}) \quad , \tag{2}$$

where N_b is the number of breeding blanket neutrons per fusion neutron, M_{nb} is the breeding blanket neutron multiplication, f_{pf} is the fraction of total blanket that is for pure fusion (35%), and f_{ABS} is the fraction of neutrons absorbed by structure or otherwise lost (5%).

The pure fusion blanket on the inboard side of the reactor does not breed sufficient tritium for internal use. The number of neutrons per fusion neutron in the outboard breeding blanket that must be devoted to making up this deficit, assuming the desired overall tritium breeding ratio (TBR) is 1.02, is

$$N_{TBR} = \frac{1.02 - f_{pf}\ T_{pf}}{1 - f_{pf}} \quad , \tag{3}$$

where T_{pf} is the tritium breeding ratio of the pure fusion blanket (assumed to be 1.3). The excess neutrons available for fissile fuel breeding, etc., is

$$N_{extra} = N_b - N_{TBR} \quad , \tag{4}$$

The breeding rates, in nuclei/sec, for radionuclide product i is

$$\Pi_i = f_i \, N_{extra} \, \Pi_n \quad , \tag{5}$$

where f_i is the fraction of excess neutrons devoted to that radionuclide product. The annual production of radionuclide product i having atomic number W_i, in kg/yr, is

$$m_i = \Pi_i \, (1.67 \cdot 10^{27} \, W_i) \, (3.16 \cdot 10^7) \, PF \quad , \tag{6}$$

where PF is the plant factor (Table 2-2). The partitioning of reactor thermal energy should appear straightforward except perhaps for synfuel production. The synfuel production rate, in kg/day, is

$$m_{sf} = \frac{f_{sf} \, P_{th} \, \eta_{sf}}{HHV} \, (24 \times 3600 \times PF) \ , \ \text{kg/day} \quad , \tag{7}$$

where f_{sf} is the fraction of thermal power devoted to synfuels, η_{sf} is the synfuel energy conversion efficiency (50%), and HHV is the synfuel higher heating value. This is the energy that can be obtained when the combustion product water condenses. The value assumed here was 53.5 MJ/kg (23,000 Btu/lb – typical of a blend of methane and ethane[4]).

2.3. EXAMPLE DETAILED RESULT

A fast-running code COGEN was written to perform the detailed analysis of the many possible product combinations. The code calculates product output rates, capital costs, annual revenues and costs, and net return (defined as net annual income/total annual expenditures). The rest of this part of the chapter is devoted to the examination of results from code runs. Table 2-4 shows typical detailed results for a single run. Here all the thermal power went to electricity, and the excess neutrons available for breeding were divided 25% for ^{60}Co and 75% for fissile fuel. A net return on expenses of 9.9%

was realized with the electricity sold at 35 mills/kWh, ^{60}Co at \$0.25/Ci, and fissile fuel at \$25/g. Note that the electricity income dominates; the other two are merely supplemental. Nevertheless, this favorable economic result shows the benefits of producing other products besides electricity. It is of course contingent upon keeping capital costs to the level shown in the table and is also dependent upon a financial climate that maintains reasonable fixed charge rates.

2.4. PARAMETRIC RESULTS

The parametric results are divided into two parts: those using a pure fusion blanket that breeds only tritium, and those using a fission-suppressed breeding blanket outboard for breeding fissile fuel and ^{60}Co and a pure fusion blanket inboard.

2.4.1. Pure Fusion Results

Only the thermal power is available for product generation when the fusion neutrons are used just to breed tritium fuel. The three possible products are electricity, synthetic fuel and process heat.

Figure 2-1 shows results for electricity only, Fig. 2-2 for synfuel only, Fig. 2-3 for process heat only, and Fig. 2-4 for a realistic mix of the three. The figure of merit is the percent return on annual expenditures. A value of 10% was deemed acceptable. Realistically, this would probably be whittled to about 5% after transmission, distribution, and administrative costs are included.

Two values are used for electrical recirculating power, zero and 10% of the fusion power. The latter represents 250 MWe, typical perhaps of a resistive coil RFP or a superconducting tokamak with an inefficient current drive. No (*i.e.*, negligible) recirculating power might be representative of a long-burn inductively-driven superconducting tokamak.

TABLE 2-4
Example Fusion Cogeneration Plant

Plant Size and Production Rates	
Blanket Thermal Power, MW	3000.00
Total Fusion Power, MW	1744.00
Gross Electrical Power, MWe	1260.00
Recirculating Power, MWe	174.40
Net Electricity for Sale, MWe	1085.60
Fissile Fuel Output, kg/yr	1705.00
Cobalt Production, kg/yr	146.00
Income in Millions of Dollars per Year	
Electricity at 35 mills/kWh	233.15
Fissile Fuel at $25/g	42.62
Cobalt at $0.25/Curie	41.34
TOTAL ANNUAL INCOME	317.10
Capital Costs and Annual Expenses in Millions	
Fusion Plant less Blanket and Power Conversion	2400.00
Blanket	247.50
Power Conversion	504.00
Fissile Fuel Plant	50.00
Cobalt Plant	50.00
Total Capital Cost	3251.50
Total Annual Cost of Capital	282.88
Operating and Maintenance	5.66
TOTAL ANNUAL COST	288.54
NET INCOME, millions	28.57
RETURN ON EXPENSES, percent	9.90
Neutron Accounting for Breeding Blanket (Per Fusion Neutron)	
Total Neutrons	1.365
Useful Neutrons	1.297
Internal Tritium Breeding	0.869
Excess Neutrons Available	0.428
Neutron for Cobalt	0.107
Neutron for Fissile Fuel	0.321

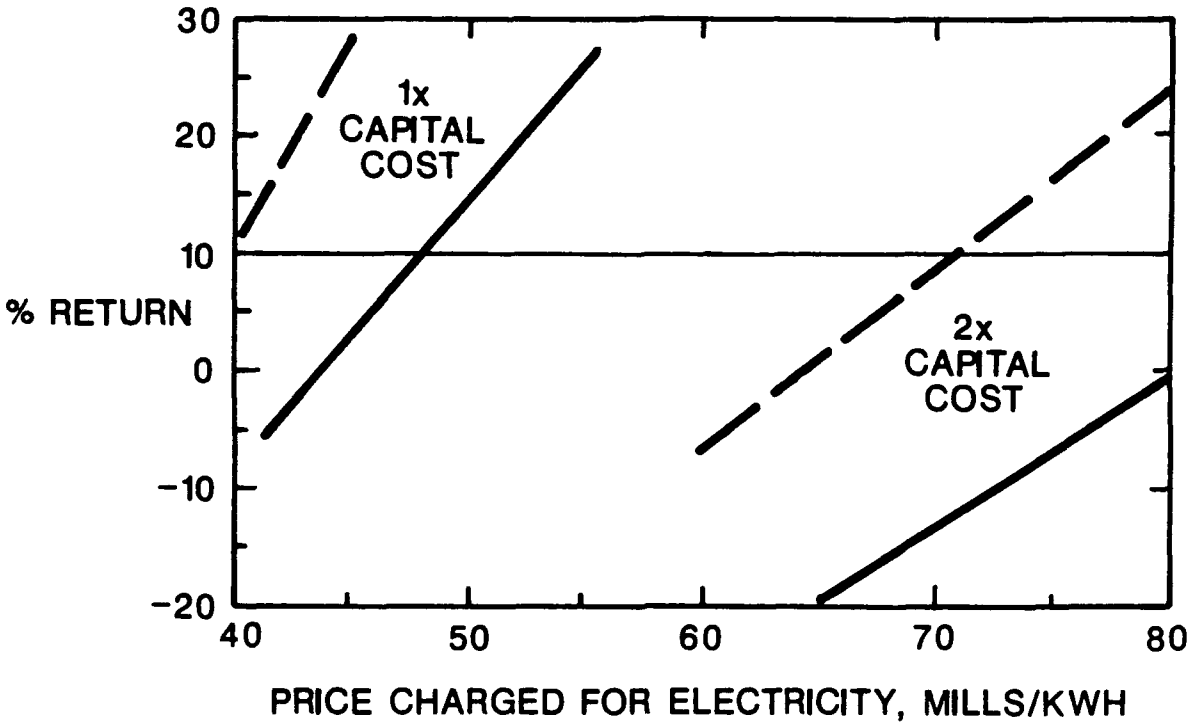


FIG. 2-1. Financial return for an electricity producer for two capital cost assumptions. Solid line – 1010 MWe for sale, 10% recirculating power. Dashed line – 1260 MWe for sale, no recirculating power.

Two values are also used for fusion plant capital costs. The 1× costs are those described earlier. The 2× costs refer to a 2.0 multiplier on the fusion plant and blanket. Because other cost items, such as power conversion and some of the processing plants, are not multiplied, total costs are slightly less than doubled.

The power conversion cost is included in the electricity producer but not in the process heat or synfuel producers. However, the cost of the appropriate conditioning plants are included in the other two, as is the cost of purchasing electricity at 45 mills/kWh to supply the recirculating power.

With 1× capital costs, all three products are nearly competitive with current prices[5] provided no further upgrading of the synfuel is needed and provided non-polluting process heat can be sold at a premium. For example, synthetic fuel at $\$11/10^6$ Btu would provide

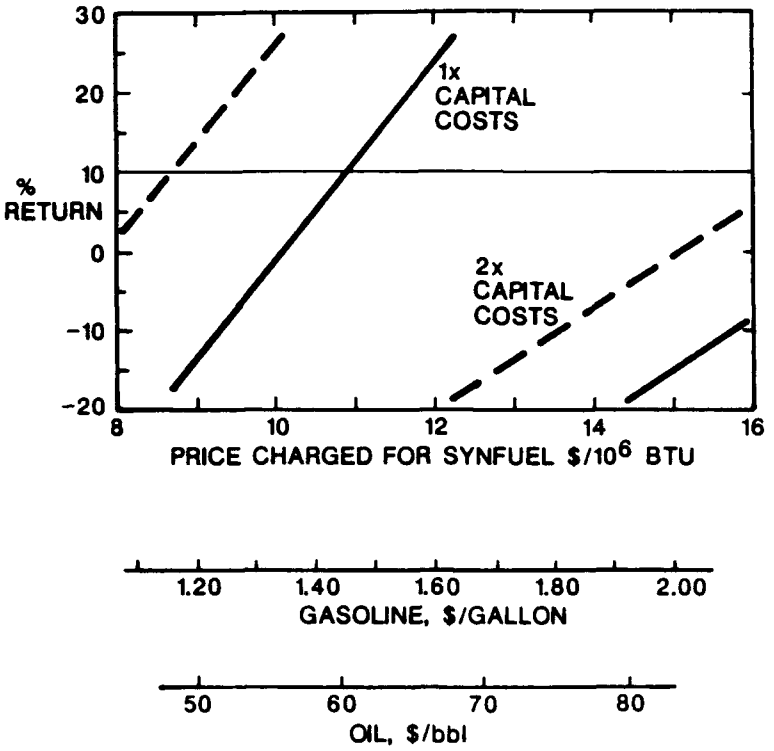


FIG. 2-2. Financial return for a fusion plant producing 1700 tonnes/day of synthetic fuel. Recirculating power (none – dashed line, 250 MWe – solid line) is purchased at 45 mills/kWh. Also shown are equivalent values for gasoline and oil.

a 10% return with 10% recirculating power. For gasoline, this corresponds to \$1.45/gal, a little above current prices at the pump. Similarly, a 10% return can be realized from a process heat plant at \$6/GJ. This corresponds to oil at \$33/bbl, 1.5–2 times today's prices. However, the price charged could be attractive in situations where fossil fuel emissions are not allowed.

The effect of recirculating power at 1× capital costs is about 20% on the price charged for all six cases. With none, the 10% return point drops to 40 mills/kWh for electricity, \$8.70/10^6 Btu for synfuel (\$0.78/gal gasoline equivalent), and \$4.8/GJ for process heat. All three are essentially competitive, again applying an economic benefit of no emissions

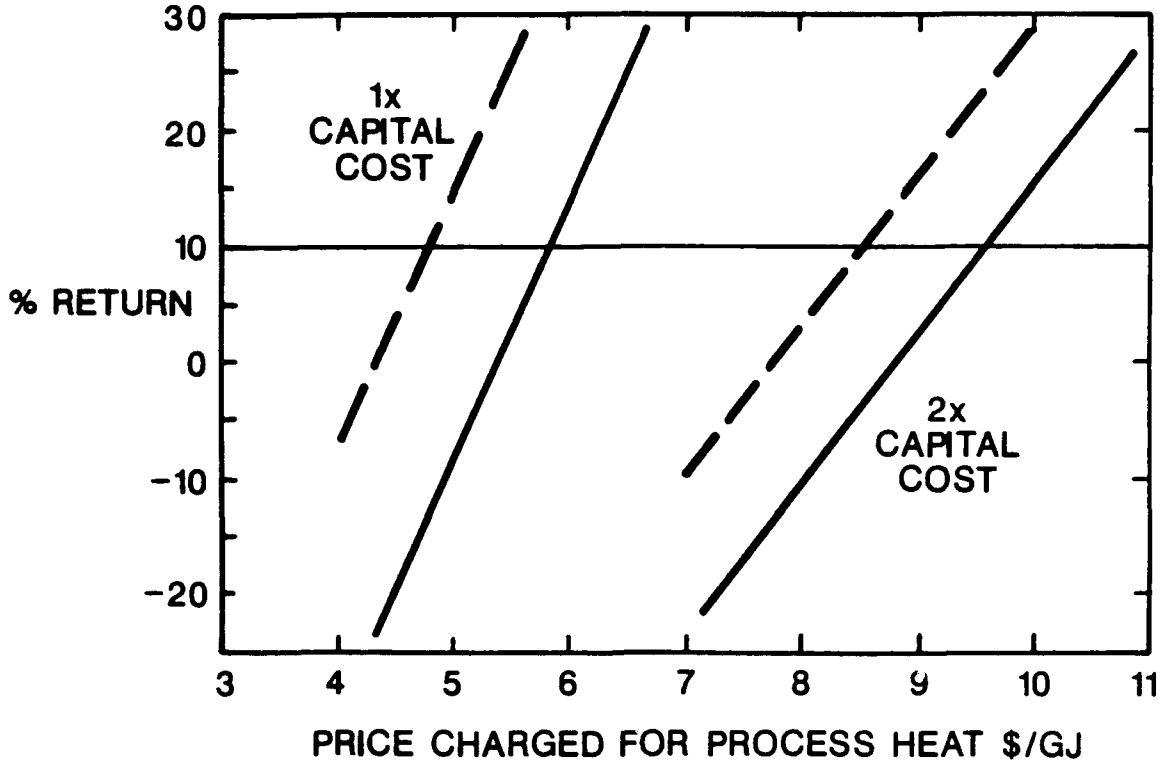


FIG. 2-3. Financial return for a fusion plant producing 3000 MW of thermal power for process heat. Electricity for recirculating power is purchased as in Fig. 2-2 above.

to the last. This points out the importance of recirculating electrical power on costs, a matter that is often underemphasized in fusion reactor studies.

A doubling of the basic fusion plant and blanket costs drives product prices out of the running: almost 90 mills/kWh for electricity, well over $\$16/10^6$ Btu for synfuel ($90/bbl oil equivalent) and $9.50/GJ for process heat. Note that recirculating power effects electricity price more than the other products. It remains a 20% effect at 2× capital cost for electricity because the amount for sale is reduced in that proportion. On the other hand, capital charges become a larger fraction of total expenses with synfuel and process heat, overshadowing the cost of purchased recirculating power.

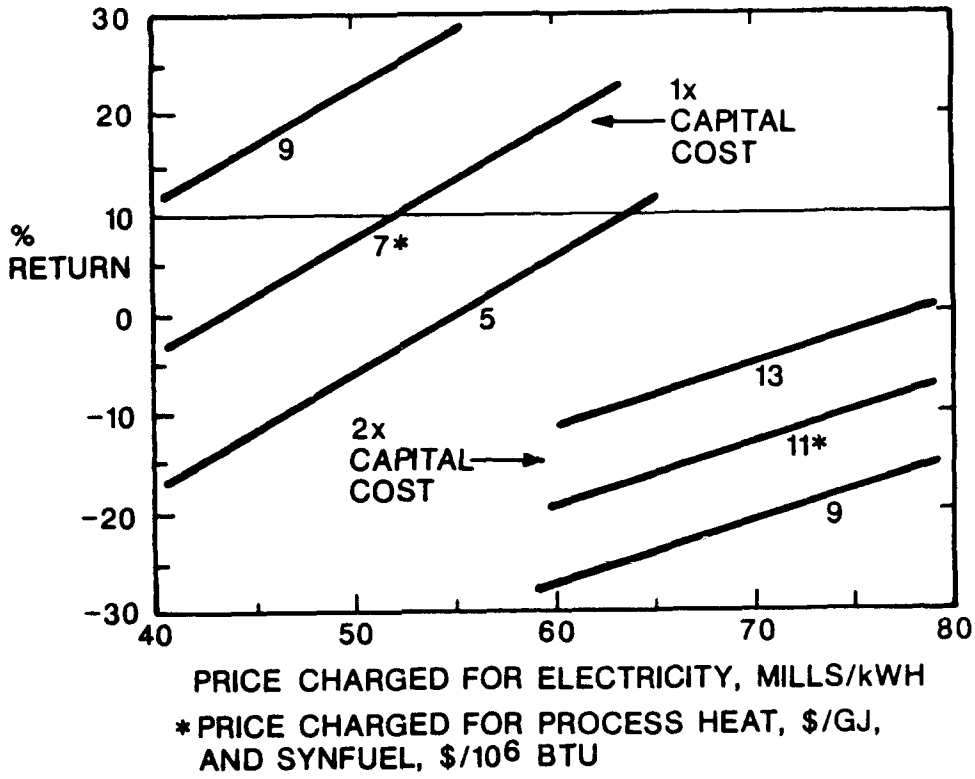


FIG. 2-4. Financial return for a mixed product fusion plant producing 506 MWe electricity, 600 MWth process heat, and 339 tonnes/day synfuel. The 250 MWe recirculating power is supplied internally.

A mix of electricity, process heat, and synfuel shows similar economics as above (Fig. 2-4). With electricity sold at 45 mills/kWh, one must obtain about \$8/GJ or 10^6 Btu for process heat and synfuel, respectively, to get 10% net return. When fusion plant capital costs double, prices charged become excessive, as before.

In summary, pure fusion plants should be able to produce electricity, process heat, and synthetic fuel and sell them at nearly competitive rates even in today's depressed energy market, provided capital costs are not very different from current estimates and presuming a premium can be obtained for non-polluting process heat. If capital costs are, for example, twice that, then the prices that must be charged are too high. Recirculating power to run the plant has a significant effect on prices, and care must to taken to keep it below about 10% of the fusion power.

2.4.2. Radionuclide Breeding Results

When the neutrons are exploited for their breeding potential as well as heat, the economic picture changes from merely acceptable to very competitive. Figure 2-5 shows results for a fissile and synthetic fuel factory where no electricity is produced. The cost of adding processing plants cancels any gain in not having to purchase electricity. This type of plant might be representative of early fusion plants because they can be remotely sited and do not require the reliability of electric generation plants. Because of the higher blanket energy multiplication, the fusion power needed for 3000 MW thermal is reduced from 2500 to 1744 MW. Purchased recirculating power, which is modeled as a fraction of the fusion power, is reduced from 250 to 174 MWe. As seen in the figure, if capital costs do not exceed expectations, one could, for example, sell synfuel at $\$8.60/10^6$ Btu (equivalent to \$1.14/gallon for gasoline) and fissile fuel at \$30/g. These values compete fairly well with today's prices. As before, prices are too high if fusion plant costs are doubled.

The most studied version of the fusion breeder has been the electricity/fissile fuel producer. This is an attractive combination because so much revenue can be obtained from electricity that fissile fuel charges can be kept low. Also, fusion plants can breed fuel quickly and each fusion plant can fuel up to 20 equal-power fission reactors.

Figure 2-6 shows typical results. With nominal capital costs, a 10% return is realized by selling electricity at 40 mills/kWh and fissile fuel at \$20/g. If the more current rate of 45 mills/kWh is charged, then the fissile fuel is almost free, about \$5/g. If, on the other hand, \$30/g is obtained for the fissile fuel, then electricity can be sold at a bargain 36 mills/kWh. These figures clearly show the economic advantage of the hybrid and provides an argument for reinvestigating the engineering and safety issues.

Results improve even further if ^{60}Co is generated as well. Figure 2-7 shows results when electricity production is unchanged but fissile fuel production is reduced to 1105 kg/yr and 146 kg/yr of ^{60}Co is produced with the surplus neutrons. If the cobalt is sold at \$0.25/Ci, well under the current price of ~\$1/Ci,[2] then, if the fissile fuel is sold at

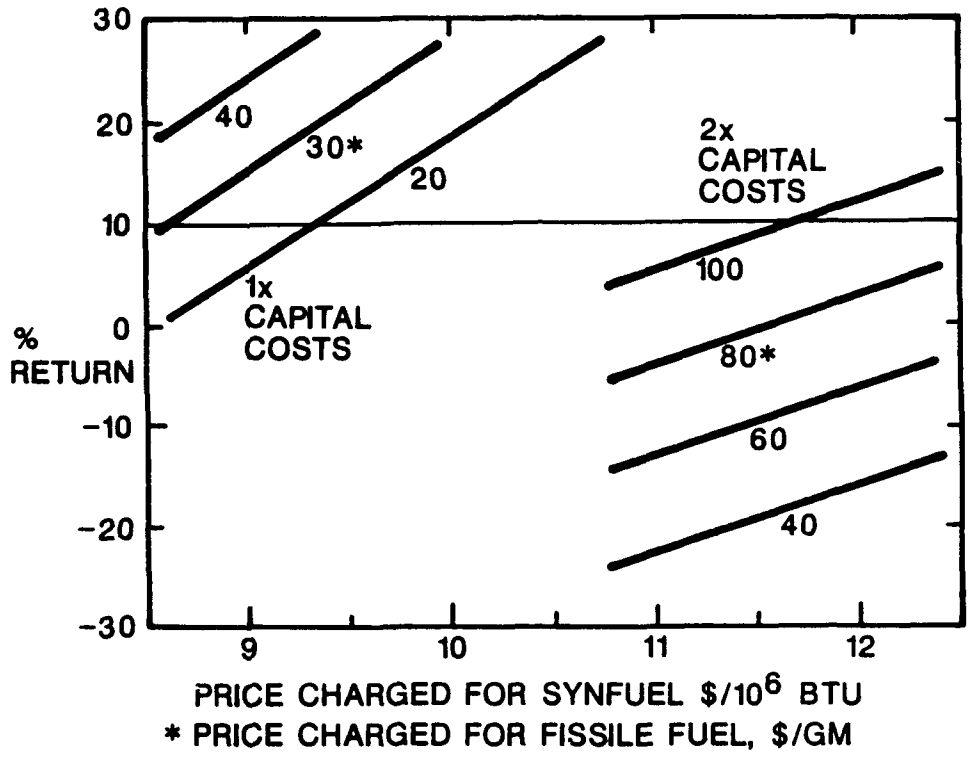


FIG. 2-5. Financial return for a fissile and synthetic fuel factory producing 1696 tonnes/day synthetic fuel and 2273 kg/yr fissile fuel. The 174 MWe electricity to run the plant is purchased at 45 mills/kWh.

$30/g, only 34 mills/kWh need be charged for electricity. If electricity were sold at about 42 mills/kWh, the fuel could be given away. If the cobalt were sold at $0.50/Ci and fissile fuel at $30/g, only 27 mills/kWh need be charged for electricity. Clearly, the economics of fusion reactors depends on the judicious choice of products.

Figure 2-8 shows what happens when the thermal energy is divided between electricity and synfuel with fissile fuel production held constant at 2273 kg/yr as in Fig. 2-6. The synfuel price is fixed at $4/10^6$ Btu, equivalent to $0.53/gallon of gasoline. While a higher price must be charged for electricity than in the previous case, results are still favorable. For example, with fissile fuel sold at $30/g, electricity must be sold at 43 mills/kWh.

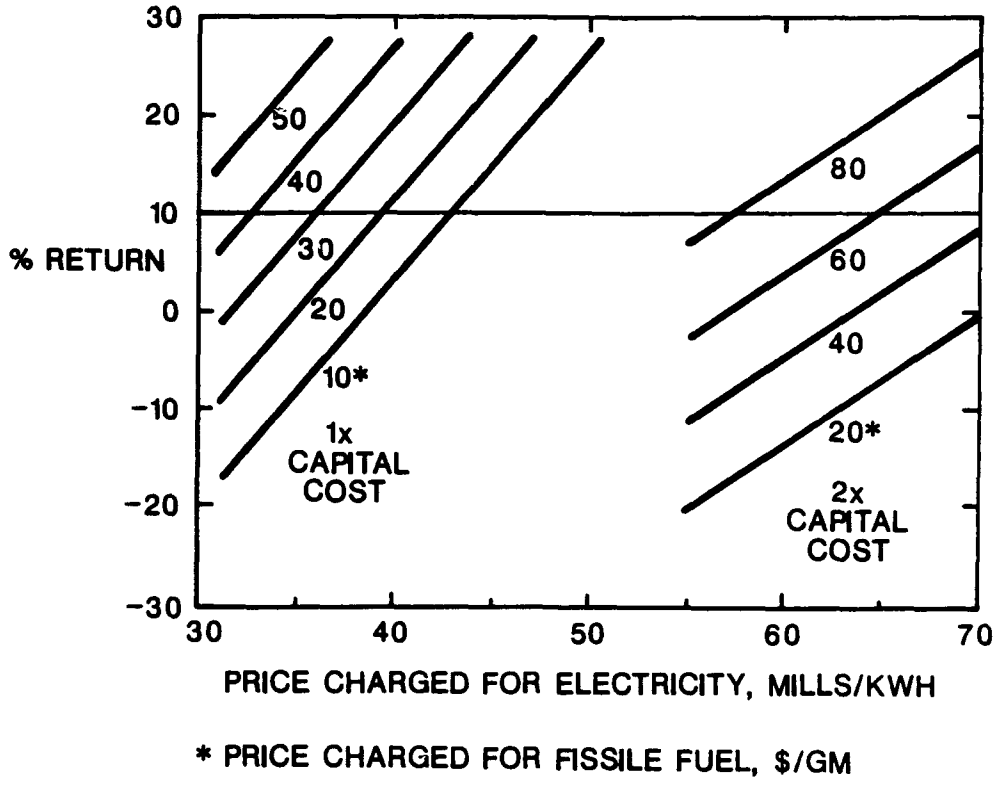


FIG. 2-6. Financial return for a hybrid plant producing 1086 MWe electricity for sale (1260 MWe gross) and 2273 kg/yr fissile fuel.

In summary, exploiting both the thermal power and radionuclide breeding capability of fusion reactors maximizes their economic potential. With a fission-suppressed hybrid having high-temperature blankets, one might simultaneously produce electricity, process heat, synthetic fuel, fissile fuel, and ^{60}Co. The most attractive combination appears to be one that uses all the thermal power for electricity and divides the breeding between fissile fuel and ^{60}Co.

2.5. SUMMARY

Provided capital costs are reasonable, pure fusion reactors can economically produce electricity and synthetic fuel and, if economic allowance is made for lack of pollutants, process heat as well. These products can be produced singly or in combinations. The

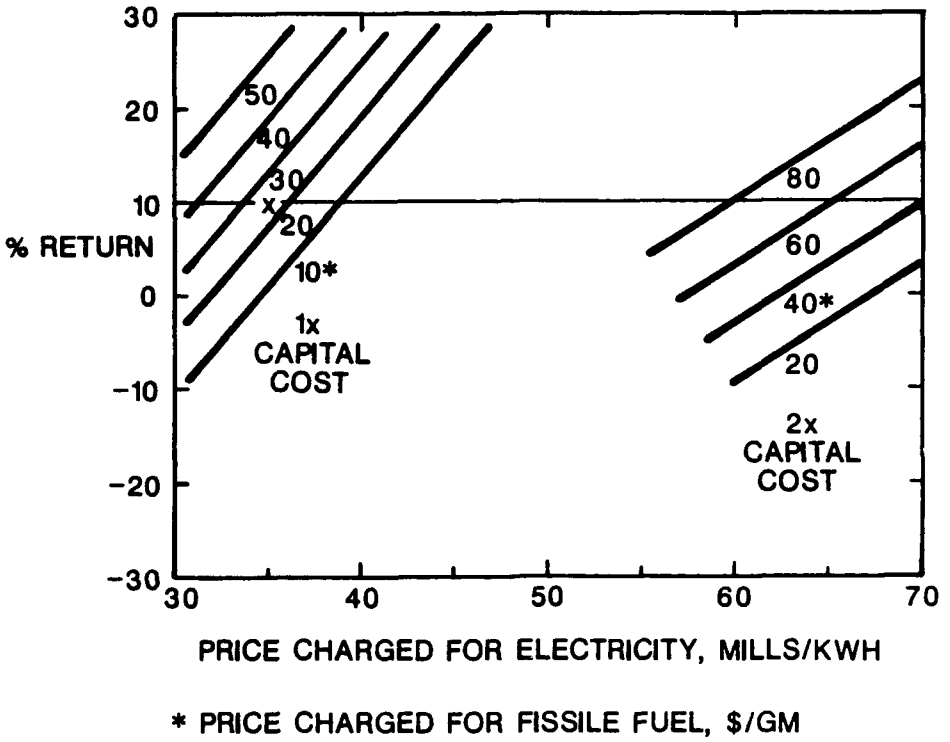


FIG. 2-7. Results for fusion hybrid producing 1086 MWe electricity, 1705 kg/yr fissile fuel, and 146 kg/yr ^{60}Co. The cobalt is sold at \$0.25/Curie. The x-point refers to the detailed example shown in Table 2-4.

economics of synfuel and process heat remain attractive even when the electricity to run the reactor is purchased rather than supplied internally, provided it is no greater than about 10% of the fusion power.

Fission-suppressed hybrids have even more potential because the excess neutrons can be used to breed marketable products such as fissile fuel and ^{60}Co. The best combination is one producing these two along with electricity. Then, with electricity sold at current market rates, fissile fuel and ^{60}Co may be sold below current market.

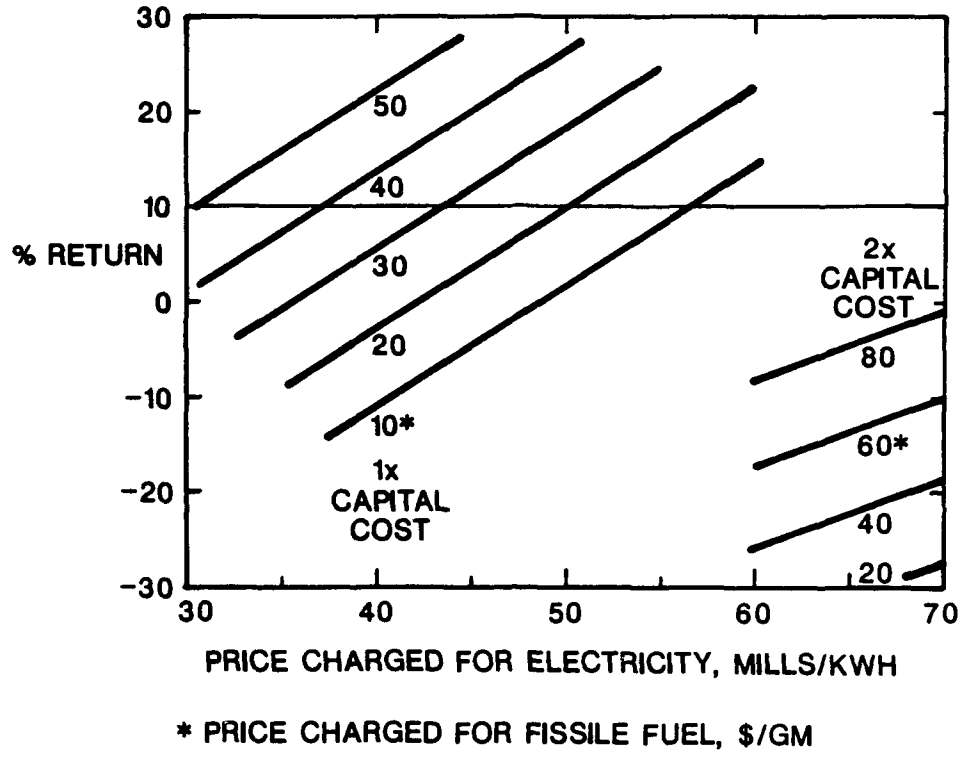


FIG. 2-8. Results for a fusion hybrid producing 582 MWe electricity, 1086 tonnes/day synfuel, and 2273 kg/yr fissile fuel. Recirculating power is 174 MWe. Synfuel is sold at $\$4/10^6$ Btu.

References

[1] E.T. Cheng, GA Technologies, Inc., personal communication, June 1987.

[2] "The Titan Reversed-Field Pinch Fusion Reactor Study," Scoping Phase Report UCLA-PPG-1100, Univ. of California, Los Angeles, January 1987.

[3] D.L. Sonn, GA Technologies Cost Development Dept., personal communication, July 1987.

[4] G.J. Van Wylen and R.E. Sonntag, Fundamentals of Classical Thermodynamics, 2nd ed., Wiley (NY) 1973.

[5] **Power Engineering**, August 1987, p. 11.

3. ELECTRICITY

The production of electricity has historically been the predominant proposed application of fusion energy and is likely to remain so. The production of fissile fuel is of course also very significant; but this is also an indirect route through fission burners to the production of electricity.

3.1. DEMAND FORECAST

The demand for new generation capacity depends on the expected plant lifetime and the capacity growth rate. The worldwide installed capacity[1] in 1983 was about 2000 GWe. Figure 3-1 shows the number of new 1.0 GWe plants that must come on line each year for several assumed capacity growth rates assuming a 40-year plant life. The historical growth rate for electricity production from 1900–1975 has been 7% per year. From 1975 to present, growth has averaged a little under 2% per year. Even with no growth, about 50 new plants must be built each year just to stay even. This alone is a substantial market. With 3% growth, which could not continue for long before reaching some constraint, by 2060 over 700 new plants must come on line annually. Growth in less-developed countries may in the future be much higher than in developed countries. Clearly, there is a substantial market for fusion electricity generators under any circumstance except negative growth. Current attempts at plant life extension are typically aimed at 10–20 year maximum life increase and generally would have only a small effect on new capacity requirements. The exception is the zero growth case where life extension would have a proportionate effect.

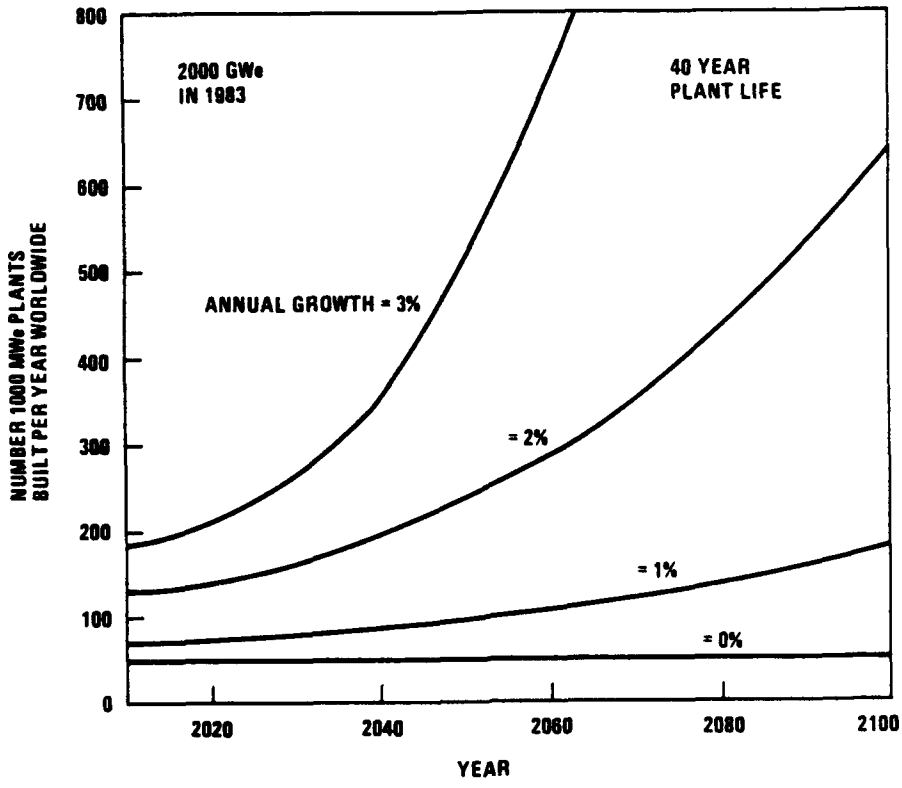


FIG. 3-1. Estimated worldwide annual new plant construction for 40 year life and 0 – 3% capacity growth rate.

3.2. FUSION ELECTRIC PLANT DESIGN STUDIES

Numerous conceptual design studies have been made of tokamak and other fusion electric generating plants. Cost estimates were made generally based on unit costs ($/kg, $/Joule, etc.) scaled from similar systems from other power plants and from existing fusion experiments. Most of these studies had the following two objectives: (1) to provide a very rough assessment of feasibility of a confinement method as an electricity producer, and (2) identify key technology areas where future research should be focused. Unfortunately, statements relating to the second were sometimes construed as indicating lack of feasibility for specific reasons. In fact, none of these studies were of sufficient depth to draw such strong conclusions. All were of the "if/then" variety. That is, if reactor plasmas have betas and confinement as assumed, and if the current drive works as efficiently as expected, etc.,

then a fusion power plant could be built with a certain expected performance and cost. However, the extrapolations required from current experiments were typically orders of magnitude. Therefore, all of these design studies must be approached cautiously and strong conclusions avoided.

Nevertheless, these studies in aggregate do show some interesting trends. Table 3-1 shows some of the more significant design studies performed over the last dozen years. While most are for tokamaks, studies are also shown for the reversed-field pinch, the mirror, and inertial fusion.

The third column in the top half of the table refers to poloidal beta (P), toroidal or total beta (T), or, for ICF, target gain (G), measured as the ratio of the fusion energy release to energy supplied *to* the driver. Driver efficiency is therefore included. The fourth column shows the maximum fields needed. For tokamaks, this is usually the toroidal field. For mirrors, it is one of the end cell coils (the choke coil for MARS, for example). For RFPs, it is the ohmic-heating coil. Note that magnetic fields over 8 tesla are required in each case.

Neutron wall loadings shown are time-averaged values. Actually, for ICF, the energy flux per pulse is more important. The values range over a factor of almost 20. Values below 2–3 MW/m^2 tend to produce large, expensive reactors. Above that, however, these studies show little correlation between wall loading and cost.

Four types of power conversion systems were considered in these studies: saturated steam (300–350°C turbine inlet temperature, 33–36% efficiency), superheated steam (500–550°C, 40–45%), closed cycle gas turbine (1030°C Helium, 55%), and direct conversion (over 60%). The last is for collecting mirror end cell charged particle losses.

Both net and gross electricity are shown in the table, the difference being the recirculating power needed to run the reactor. Values of 200 MWe are typical for the latter. This power is used to run resistive coils, current drives, coolant pumps, ICF drivers, and for pulsed magnetic machines, the startup energy load smeared over the burn. The one

TABLE 3-1
Fusion Electric Plant Studies

NAME	REF	DATE	TYPE	T = Tor Beta P = Pol Beta G = ICF Gain	Bcoil max T	NEUTRON WALL LOAD MW/m^2
Titan	2	87	RFP	.20(P)	8.3(OH)	18.0
Cascade	3	85	ICF/Laser	20(G)	-	8.0
CRFPR	4	84	RFP	.23(P)	9.2(OH)	19.5
MARS	5	82	Mirror	.28(T)	24(Choke)	4.3
OHTE	6	81	RFP	.4(P)	11.2(OH)	19.5
Starfire	7	80	Tokamak	.067(T)	11.1(TF)	3.6
Witamir	8	80	Mirror	.4(T)	15(barrier)	2.4
NUMAK	9	79	Tokamak	.06(T)	12(TF)	4.0
GA Demo	10	78	Tokamak	.10(T)	8.8(TF)	1.1
UWMAK-III	11	76	Tokamak	.058(T)	8.8(TF)	1.9

NAME	PWR CONV	MWe net	MWe gr	REPORTED $B	1986 $K/kWe
Titan	SH Steam	1000	1150	1.92	1.92
Cascade	He CCGT	815	905	1.49	1.90
CRFPR	Steam	1000	1227	1.48	1.55
MARS	Stm/Direct	1200	1464	3.62	3.27
OHTE	SH Steam	904	1390	1.90	2.44
Starfire	Sat Steam	1200	1400	2.12	2.53
Witamir	Stm/Direct	1530	1860	3.26	2.80
NUMAK	Sat Steam	660	725	1.13	2.49
GA Demo	SH Steam	307	374	0.69	3.40
UWMAK-III	He CCGT	1985	2050	5.70	3.82

with the highest recirculating power, OHTE, consumes almost 500 MWe, most of which goes to powering the highly resistive helical coils.

The reported cost is the total capital cost shown in the project report and are taken to be dollars in the year the report was published. They include direct cost plus all the indirects. These are scaled to 1986 using the Handy-Whitman escalation index for steam

and nuclear plants[12] and then normalized to the net electricity. Note the general trend on decreasing costs, reflecting the greater sophistication of the more recent studies.

Figure 3-2 shows 1986 unit capital costs plotted against net electricity. One should discard the mirrors for technical reasons (recent physics results are disappointing), the GA demo as being small demonstration plant, not an equilibrium commercial plant like the others, and UWMAK-III as being an early design, so far out of bounds as to question the design choices (use of large quantities of TZM, for example). One is then left with six reactors in a narrow range of size and cost (see Fig. 3-2). Three are RFP's, two are tokamaks, and one is an ICF reactor. The cost range of \$1500–2500/kWe (1986\$ including indirects) and the power range of 600–1300 MWe are not unlike those found in fission reactors. While it is dangerous to make direct comparisons, this general observation gives hope that fusion may be able to compete with other energy sources for electricity generation.

Four items go into the cost of electricity: (1) capital charges, (2) fuel, (3) operation and maintenance, and (4) payment into a decommissioning account. While the last is insignificant for fossil plants, it cannot be ignored for fusion or fission. Figure 3-3 shows actual breakdowns for an LWR once-through fission plant and a high-sulfur coal plant.[13] Also shown is an estimate for a fusion plant. Because of the greater complexity of the fusion plant, both O&M and decommissioning costs are expected to be higher than a fission plant (here we estimate 30% higher). Yet, because fusion fuel costs are negligible, capital charges can be greater (in this estimate, 20% greater) and still charge the same for electricity. Because the reactor plant equipment cost ranges from 18% of directs (Cascade) to 56% (Starfire), the 20% increment can translate to a doubling of reactor plant equipment over fission for a Cascade-type plant to a 36% increase for a Starfire-type plant.

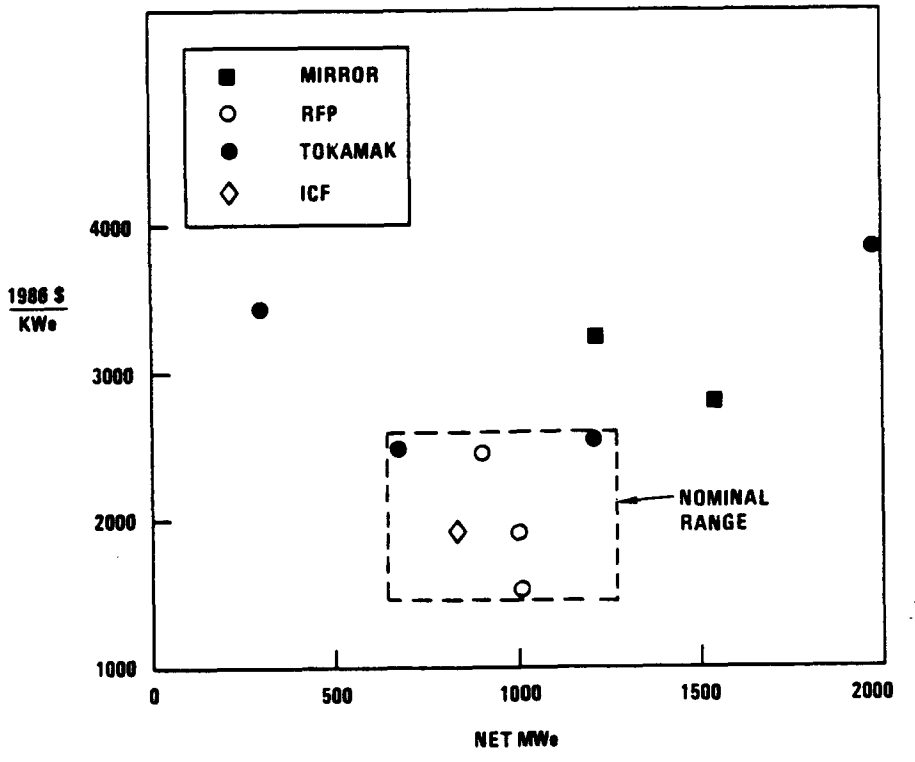


FIG. 3-2. Unit direct costs in 1986 $ vs. net power for the reactor studies listed in Table 3-1.

3.3. SAFETY ISSUES

Fusion reactors have a potential for inherent safety that fission reactors cannot achieve. In the reactor studies discussed above, only the Cascade ICF reactor concept succeeded in pursuasively arguing that inherent safety can effect cost reductions because lower cost non-nuclear grade components can be used. With the ceramic granule blanket and SiC/aluminum structure, both tritium inventory and afterheat were low enough to satisfy 10CFR100 without active controls.

While Cascade was an ICF concept, it is believed that many of its inherent safety/low activation concepts could be adopted by magnetic fusion. Even though, for example, the adoption of a ceramic first wall may require a reduction in wall loading, the savings due to non-nuclear costing may more than offset the cost increase due to the larger reactor.

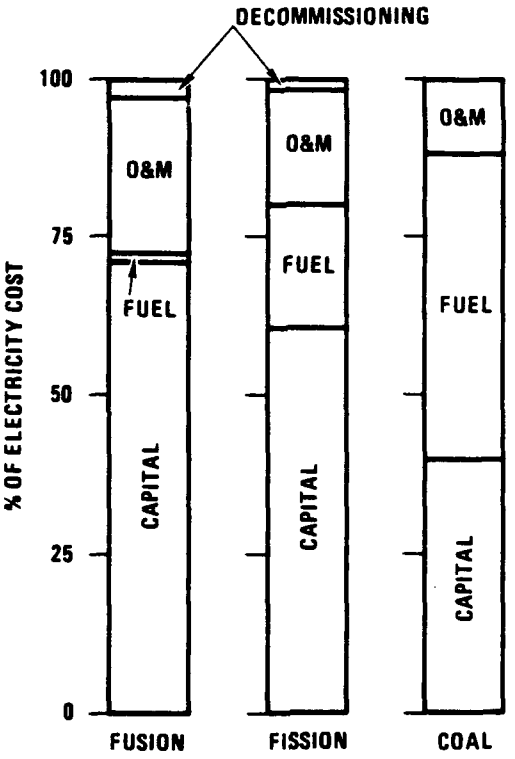


FIG. 3-3. Electricity costs breakdown for fission and coal (Ref. 13) and estimated allowable capital cost fraction for fusion.

3.4. SUMMARY

Electricity is expected to be the primary product from fusion reactors. Projecting even a modest growth rate in electricity demand, there will be a substantial market for new capacity worldwide. Issues of cost, safety, reliability, siting, power level, and construction time should play a prominent role in reactor studies. Even though O&M and decommissioning costs are likely to be be higher for fusion than fission, the negligible fuel cost permits a higher reactor cost than fission for comparable electricity cost. Presuming fusion reactors with acceptable confinement and beta can be developed, costs of electricity should be competitive with fission and fossil plants if attention is paid to compactness through high power density, reductions in recirculating power, and materials and design choices that bring out the advantages of inherent safety.

References

[1] Data compiled by C.J. Hamilton, GA Technologies Inc., April 1987.

[2] "The Titan Reversed-Field Pinch Fusion Reactor Study," Univ. of Calif. (LA) Report UCLA-PPG-1100, Jan. 1987.

[3] I. Maya, K. R. Schultz, *et al.*, "Inertial Confinement Fusion Reaction Chamber and Power Conversion Study," GA Technologies Report GA-A17842, October 1985.

[4] R. L. Hagenson, R. A. Krakowski, *et al.*, "Compact Reversed-Field Pinch Reactors," Los Alamos National Laboratory Report LA-10200 MS, August 1984.

[5] B. G. Logan, C. D. Henning, G. A. Carlson, *et al.*, MARS - Mirror Advanced Reactor Study, Lawrence Livermore National Laboratory Report UCRL-53480, July 1984.

[6] R. F. Bourque, "OHTE Reactor Concepts," Proc. 9th Symp. on Engineering Problems of Fusion Research, Chicago, Oct 26–29, 1981.

[7] C. C. Baker, M. A. Abdou, "Starfire - A Commercial Tokamak Fusion Power Plant Study," Argonne National Laboratory, Argonne, IL, Report ANL/FPP-80-1, Sept 1980.

[8] B. Badger, K. Audenaerde, J. B. Beyer, *et al.*, "Witamir-I, A University of Wisconsin Tandem Mirror Reactor Design," University of Wisconsin (Madison) Report UWFDM-400, September 1980.

[9] B. Badger, K. Audenaerde, H. I. Avci, "NUWMAK - A Tokamak Reactor Design Study," University of Wisconsin (Madison) Report UWFDM-330, March 1979.

[10] D. Kearney, R. F. Bourque, K. Burrell, *et al.*, "Doublet Demonstration Fusion Power Reactor Study," GA Technologies Report GA-A14742, July 1978.

[11] B. Badger, R. W. Conn, G. L. Kulcinski, *et al.,* "UWMAK-III, A Noncircular Tokamak Power Reactor Design," University of Wisconsin (Madison) Report UWFDM-150, July, 1976.

[12] Handy-Whitman Index of Public Utility Construction Cost, Bull. No. 125, Whitman, Requardt & Assoc., Moorestown, NJ, March 17, 1987.

[13] J.G. Delene, "Nuclear Energy Cost Data Base," DOE Report DOE/NE-0044/3, June 1985.

4. NUCLEAR FUELS

Nuclear fuels that can be bred in fusion reactors are fissile fuel (uranium and plutonium) and fusion fuel (tritium). We will discuss each category in turn.

4.1. FISSILE FUEL

An important application of fusion is the production of fissile fuel (^{233}U or ^{239}Pu) for fission plants.[1, 2, 3] Uranium-233 is produced by the neutron transmutation of thorium, while ^{239}Pu results from the transmutation of ^{238}U. The estimated fissile fuel market should justify tens of units of 2500 MW fusion thermal power (*i.e.*, ~1000 MWe) by the year 2050. Well before this time, the market price of U_3O_8 is expected to reach \$100/lb. If fusion power costs twice LWR power, the cost of fusion-produced ^{233}U will be equivalent to \$120/lb of U_3O_8. If fusion power costs $1\frac{1}{2}$ times as much as LWR power, the equivalence will be \$60/lb U_3O_8, cheaper than mining the ore. Plutonium is assumed to be worth two-thirds as much as ^{233}U.

4.1.1. Conceptual Design Studies

Conceptual design studies[1] have been carried out on the three fusion-fission hybrid reactor types. The first is the fission-suppressed hybrid, which maximizes fissile material produced (^{239}Pu or ^{233}U) per unit of total nuclear power. This is done by suppressing the fission process and multiplying neutrons by (n,2n) reactions in materials like beryllium. The second is the fast-fission hybrid, which maximizes fissile material produced per unit of fusion power by maximizing the fissioning of ^{238}U (^{239}Pu is produced) in which twice the fissile atoms per unit of fusion power (but only a third per unit of nuclear power) are made. The third is the power hybrid, which amplifies power in the blanket for power production but does not produce fuel to sell. All three types of blankets must sell electrical power to be economical.

One series of studies led to a reference design of a fission-suppressed breeder that uses liquid-lithium cooling of beryllium balls with thorium snap rings. Another series considered a fission-suppressed, helium-cooled, molten-salt breeder design. Safety improvements for use in fast-fission designs were also identified. Low-burnup metal balls, which can be drained out of the blanket to passively cooled holding tanks, would be used in several of the above design concepts.

The fission-suppressed designs identify as critical issues the need for experimental engineering data on beryllium used as the neutron multiplier and the need for low-cost reprocessing of low fissile in fertile (less than 1%) fuel from the breeder. Pyrochemical reprocessing is identified as having the potential for low cost, but needs development.

The fast-fission designs identify the critical issue of safety and the need for developmental work on afterheat removal systems to prevent serious consequences from loss-of-cooling-type accidents.

4.1.2. Projected Cost of Fissile Fuel from a Fusion Reactor

As the predicted cost of the fission-suppressed fusion plant drops from twice a light-water reactor's (LWR) cost, the calculated breakeven price of uranium from the fusion breeders drops below \$120/lb of U_3O_8. A fusion plant costing 1.5 times a LWR is calculated to produce fuel at an equivalent price of \$60/lb of U_3O_8. Even lower prices are predicted as fusion costs drop. An overall conclusion is that the deployment of fusion technology can easily cap or even depress the price of uranium.

The hybrid fusion plant sells both electricity and fuel whereas the LWR fission plant buys fuel and sells only electricity. When the costs of electricity (COE) and fuel for the two reactors are the same, we say the uranium is at an equivalent breakeven price. The market price for uranium may fluctuate, but will have a tendency in the long term to have a ceiling price determined by this breakeven price. Figures 4-1 and 4-2 illustrate the breakeven price for fission-suppressed designs and fast-fission designs.[4] As the cost of the

hybrid drops, the breakeven price of uranium drops. Design studies have predicted fission-suppressed hybrid costs at 2.7 and 2.9 times a LWR as shown, giving a breakeven uranium price around \$75 per pound. The fast-fission hybrid costs have been predicted to be just under twice the LWR cost, giving breakeven uranium prices just under \$100 per pound. These cost calculations depend on many economic assumptions[2] and on the capital cost of the hybrid which, in turn, is composed mainly of the capital cost of the fusion components (because of the greater electricity output per unit fusion power, the fusion components in fast-fission comprise a lesser fraction of the total than in fission-suppressed designs). Because of fusion cost uncertainties, it is shown as a parameter in the figures.

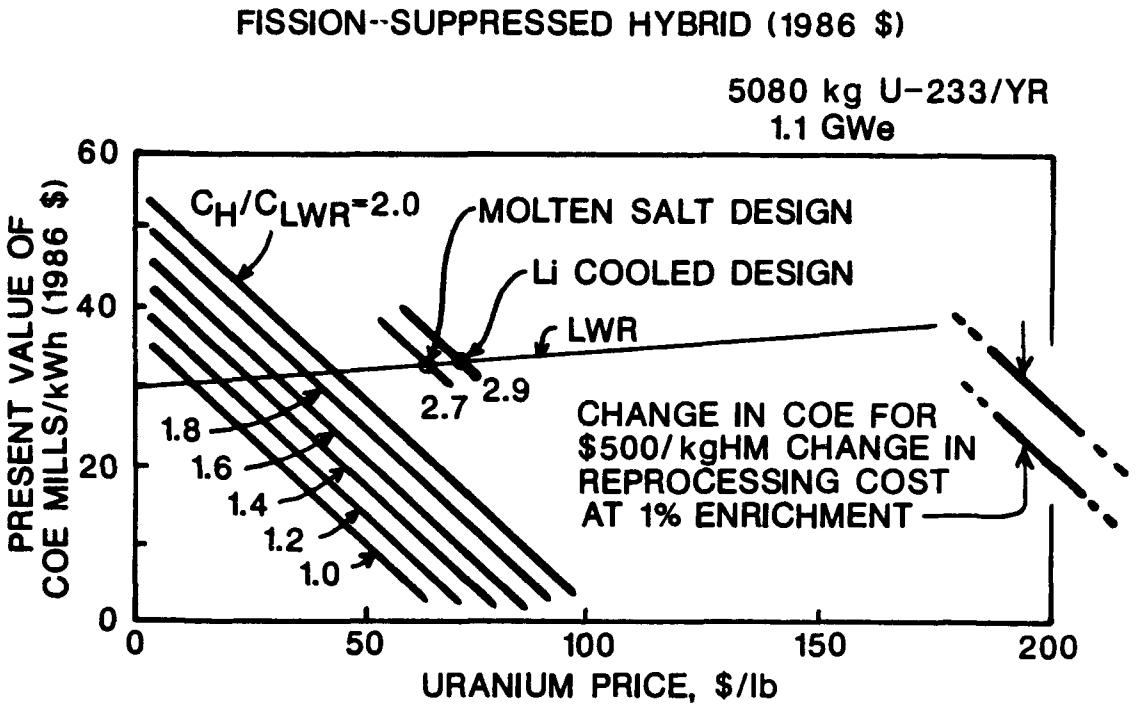


FIG. 4-1. Fission and fusion electricity costs for a fission-suppressed hybrid as a function of bred uranium price and relative cost of hybrid plant.

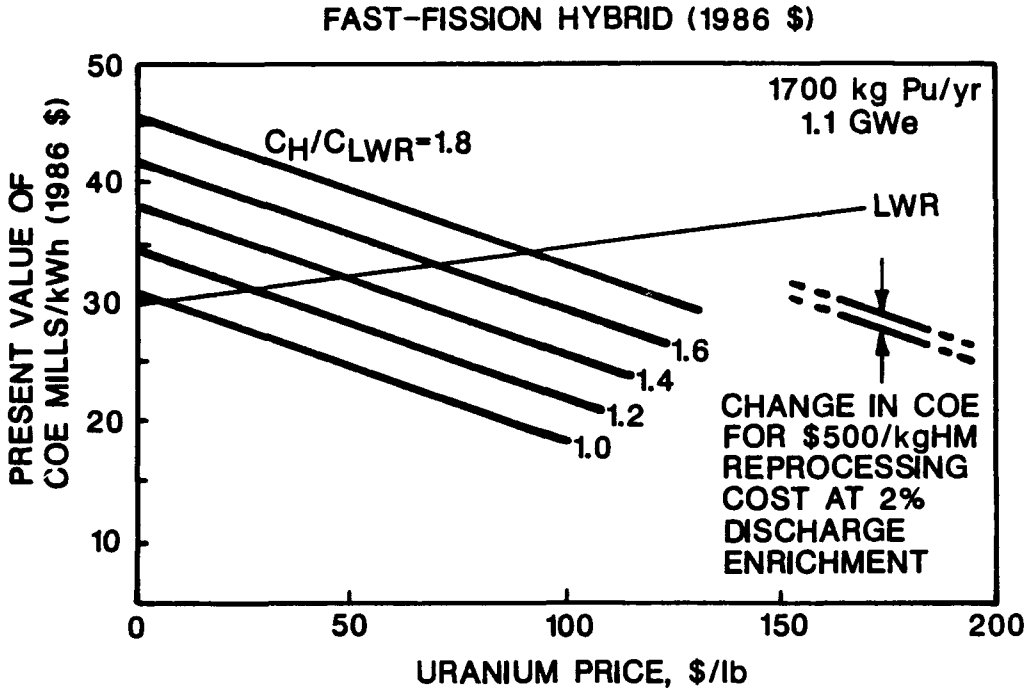


FIG. 4-2. Fission and fusion electricity costs for a fast-fission hybrid as a function of bred uranium price and relative cost of hybrid plant.

4.1.3. Market Projection for Uranium Price

This section discusses speculation on when fusion-produced fissile fuel might fully substitute for mined uranium.

The cutback in projected nuclear plant contracts and the depressed price of uranium may temporarily diminish the need for hybrid development. However, there are two advantages to this delay for fusion. They are: (1) as time goes on, fusion research increases physics confidence; (2) the hiatus also puts off a commitment to development of fission breeding which, in their refined versions, could provide competition for the fusion hybrid.

Figure 4-3 shows the mined uranium price and cost projected into the future. The price rises to \$50/lb in 2007, \$100/lb in 2027 and \$125/lb in 2035, \$206 in 2045, and \$410 in 2055. Some strategies could push these dates into the more distant future (*e.g.,* building fewer LWRs and importing foreign uranium). Some events could also bring these dates into the nearer future (*e.g.,* cartels; embargos; more expensive mining safety requirements; increased electricity use, resulting in higher nuclear fraction to mitigate environmental factors; or new electricity uses such as transportation). For comparison, the figure includes the historic price of uranium in constant 1986 dollars from 1950 to 1982.

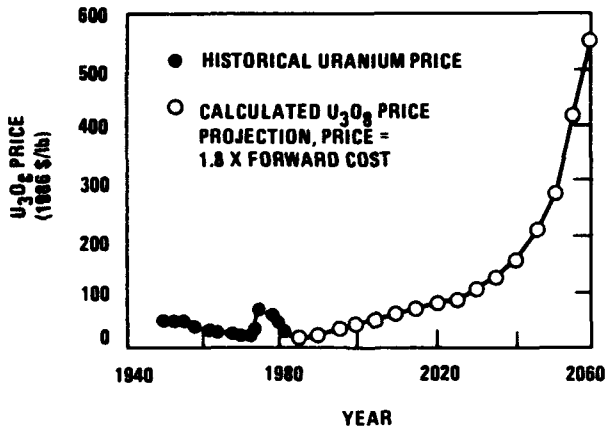


FIG. 4-3. Historical and projected uranium price.

4.2. TRITIUM

This section is based on a study carried out by the Lawrence Livermore National Laboratory on the possibility of production of tritium for the weapons stockpile by magnetic fusion reactors (see Ref. 5).

It is generally assumed that adequate tritium is bred from lithium in D-T fusion reactor blankets to sustain the reactor's fuel supply (the deuterium is obtained from conventional sources). In many blanket designs, however, there is an excess of bred tritium, which could be used for other purposes such as initial fuel loads in new fusion reactors, for

medical and industrial purposes, or for the U.S. nuclear weapons stockpile. This last application is estimated to translate into the need for 1200 MW(th) of fusion power sometime between the years 2000 and 2010. With an estimated plant cost of about $1000/KW(th) ($2500/kWe), typical of many reactor studies, the cost of bred tritium could be as low as $5000/g, half the current official price. This cost margin could allow economic tritium production even with near-term, more expensive fusion reactors.

4.2.1. Background

Today, tritium is produced in the U.S. only at the Savannah River complex. These reactors will be about 50 years old by the year 2000. Thus, planners must be prepared for their replacement about that time. In looking toward this eventuality, the progress in fusion in recent years should be noted.

Magnetic fusion as a source of neutrons to breed tritium has a number of potential advantages compared with fission reactors. Perhaps most important is the fact that fusion is neutron rich and generates about one-sixth of the heat of a fission reactor to breed an equivalent amount of tritium (see Fig. 4-4).

The nation will have a continuing need for an assured supply of tritium as long as nuclear weapons exist. Today's typical weapons require tritium, which decays with a 12-year half-life and must be replaced. While arms control measures may eventually reduce the stockpile, for the foreseeable future stockpile planning must continue to meet today's realities. Moreover, certain changes in the makeup of the stockpile, underway or being considered, could actually increase the need for tritium even if the number of weapons decline.

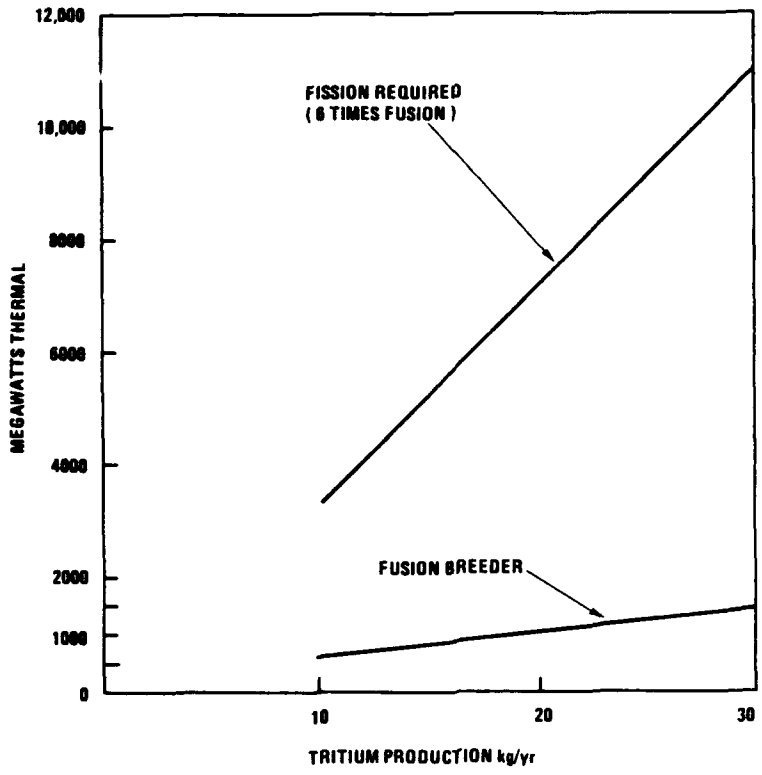


FIG. 4-4. The relationship between tritium production and reactor power for fission and fusion.

4.2.2. Blanket Design

A satisfactory blanket design for tritium production can be based on the use of the lithium-aluminum (Li-Al) alloy used at Savannah River, and beryllium with water cooling below 100°C. Such a design is shown schematically in Fig. 4-5. The design would use hot-pressed beryllium blocks and Li-Al fuel slugs. We expect no major issues with this design other than those due to radiation damage resulting from the hard spectrum of fusion neutrons. Both beryllium and aluminum are predicted to eventually become brittle and crack. Because beryllium is not used as a structural material, careful design practice to accommodate the cracking should be possible. When the aluminum structural material loses its ductility and leaks develop as a result of cracking, the blanket would be replaced with a new one.

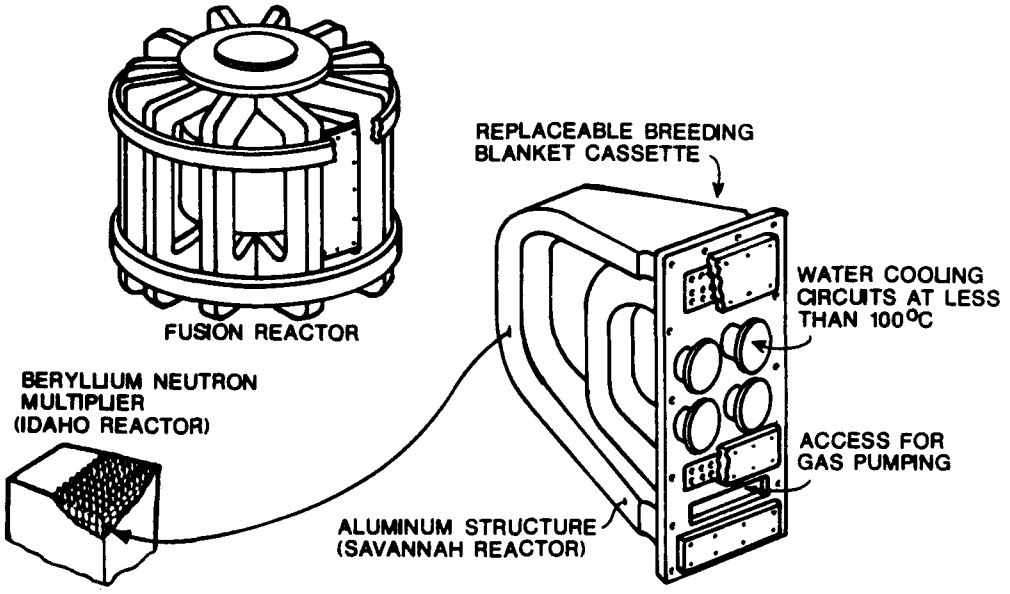


FIG. 4-5. Low-temperature breeder blanket concept.

4.2.3. Fuel-Cycle Description

Lawrence Livermore National Laboratory has studied fuel cycle characteristics of a magnetic fusion production reactor and considered operation at a fusion power of 427 MW. The tritium breeding ratio† for the fusion production reactor is estimated to be 1.56. The blanket energy multiplication‡ is 1.3. A net tritium product of 10.8 kg/yr is generated. The capital and operating costs for fuel-cycle facilities to recover the tritium from the Li-Al fuel slugs are estimated at $124 and $41 million/yr, respectively.

† *Breeding ratio is defined as atoms bred per fusion reaction, including one atom of tritium per fusion required to sustain the fusion reaction.*

‡ *Blanket energy multiplication is defined as blanket energy deposited divided by 14.06 MeV.*

4.2.4. Cost Estimate

The cost of a fusion tritium production reactor is highly uncertain because fusion technology has not reached the demonstration stage. At this time, we can only make cost estimates of preconceptual designs. However, we expect the cost to be lower than that for a fission production reactor for the same tritium production rate for three reasons: (1) the thermal power is four to six times lower; (2) the power conversion and balance of plant systems run cold and do not employ electricity-generation equipment; and (3) fewer fuel-cycle facilities will be needed, since there is no fissile or fertile material.

Operating costs for this magnetic fusion production reactor are high because of the requirement to purchase electricity. The fusion reactor will consume 400 MW-years of electricity annually while the same production fission reactor candidate might sell about 500 MW-years of electricity each year. At a sales price of 23 mill/kWh and a purchase price of 28 Mill/kWh, the fusion case requires a $60 million/year expense and the fission case benefits from revenues of about $100 million/year. On a per-gram basis, the price differential is $16,000/g of tritium. However, the fission reactor cost for fuel purchase, fabrication, and reprocessing is expected to offset the electricity sales advantage. There is no reason, of course, why the fusion reactor could not also sell electricity as well as breed tritium. Under any reasonable circumstances, the added revenue would vastly offset increased costs.

4.2.5. Safety

The main safety issue is the containment of radionuclides. There are no fission products or actinides associated with the fusion tritium breeder. The principal radionuclides will be activated structural material (mostly aluminum) and tritium. Since aluminum is a relatively low-activation material, and since fluence will be low, we expect discarded blankets to be disposed of on-site in shallow burial if long-lived ^{26}Al has not built up to a hazardous level. The bred tritium in the breeding blanket will be well-contained in

aluminum-canned Li-Al slugs similar to the form used successfully in the Savannah River reactors. The average tritium inventory in the fuel slugs in approximately 22 kg.

Although afterheat due to fission products in the fission production reactors could lead to a meltdown and release of fission products and actinides, in fusion blankets, both afterheat and the inventory of radionuclides other than tritium are much smaller. The result is that the safety problem is not only dramatically reduced, but different in quality. One possible safety issue to be investigated is the possibility of a Li-Al and water interaction during a credible accident.

References

[1] R.W. Moir, *et al.*, "Fusion-Fission Hybrid Studies in the United States," submitted to the IAEA 4th Technical Committee Meeting and Workshop on Fusion Reactor Design and Technology, May 26 to June 6, Yalta, USSR, Lawrence Livermore National Laboratory Report UCRL-94306 (1986).

[2] J.G. Delene, "An Economic Analysis of Fusion Breeders," **Fusion Technology**, 8, (1985) 459.

[3] D.H. Berwald, *et al.*, "Updated Reference Design of a Liquid Metal Cooled Tandem Mirror Fusion Breeder," **Fusion Technology**, (1986) (in press).

[4] D.L. Jassby, *et al.*, "Fast-fission Tokamak Breeder Reactors," **J. Fusion Energy** (1986) (in press).

[5] R.W. Moir and R.S. Frost, Editors, "Study of a Magnetic Fusion Production Reactor," to be published in **J. Fusion Energy**; also UCRL-94408 (1986).

5. MATERIALS FOR RADIATION PROCESSING

Fusion reactors can produce copious quantities of hard gamma emitters such as ^{60}Co or ^{56}Mn from neutron activation of parent nuclei. Gamma radiation has worldwide applications and high demand. This section will discuss radiation processing in general and from the standpoint of how fusion reactors can contribute to this expanding field.

5.1. MEDICAL SUPPLIES

Cobalt-60 and ^{137}Cs are used primarily in the medical supply industry for product sterilization. Formerly, about 80% of all disposable medical products were sterilized by ethylene oxide (EtO), a highly toxic gas. As increasingly stringent regulations on this gas were imposed by the Federal government (particularly EPA/OSHA), medical suppliers were forced to look elsewhere for sterilization techniques. In 1964, Johnson & Johnson initiated its program of replacing EtO with ^{60}Co irradiators, and by 1977 was sterilizing two-thirds of qualifying products by irradiation. American Hospital Supply also changed over to the ^{60}Co technique, building a double-cell facility in El Paso in the late seventies, at a cost of $8.5 million. The plant reportedly can treat six million ft^3/yr. Isomedix plants have about 1.2 MCi ^{60}Co per plant, installed; and Johnson & Johnson about 0.7 MCi/plant. The design capacities of these plants are generally well above the installed inventories.

There are 1130 Curies per gram of ^{60}Co. CH_2M-Hill[1] estimates that the total volume of U.S. sterilizable medical products is 125 million ft^3/yr. They also state that 7 MCi of ^{137}Cs sterilizes one million ft^3/yr. If converted to ^{60}Co, using the factor of seven recommended by Les Price of ORNL, then 1 MCi of ^{60}Co will sterilize one million ft^3/yr. Implied usage, if 60% of all gamma sterilizable products employed ^{60}Co, is therefore 75 MCi, plus replenishment.

5.2. FOOD IRRADIATION

Gamma irradiation has been shown to be effective in sterilizing foodstuffs, spices, drinking water, and sewage.[6] In addition to gamma rays, X-rays below 5 MeV and electrons below 10 MeV can also be used.[4] In principle, either could be taken from a fusioning plasma and guided to an irradiation facility.

The impact of irradiation sterilization of foodstuffs could be as revolutionary as canning and freezing were a century ago.[7] Food irradiation for pest control is superior to chemicals such as ethylene dibromide (EDB), the most commonly used substance, because a toxic chemical residue is not left on the food. Also, irradiation preservation can be used on foods already packaged and sealed, so no further handling is needed. In the United States, the incidence of trichinosis, a parasitic disease from infected pork, remains among the highest of any developed country,[8] even though most consumers are aware of the need to cook pork thoroughly. Irradiation of infected pork at moderate gamma doses of 15–30 krad renders the trichina sexually sterile and blocks maturation of the larvae after the pork is eaten.

Other applications of food irradiation are other meats, milk, potatoes, wheat products, citrus, dried fruit and tree nuts, fish and shellfish, and spice/seasonings. If irradiation is performed after final packaging, refrigeration normally is not required, which would be of significant value to Third World countries since at least a quarter of the food supply there is lost to spoilage and, probably, much of the remaining food that is consumed is tainted.

Food irradiation is proceeding at a faster pace in other countries, particularly in the Third World and the Soviet Bloc. Table 5-1 shows irradiated food products cleared for human consumption in selected countries.

Figure 5-1 shows a pallet food irradiator developed by Atomic Energy of Canada[6] that is currently processing large quantities of food on a commercial basis. Pallets offer an attractive processing method because handling fits well with current medium-load handling methods using existing fork lift equipment.

TABLE 5-1
Irradiated Food Products Cleared for Human Consumption

Bangladesh	chicken, papaya, potatoes, wheat products, fish, onions, rice, frog legs, shrimp, mangoes, seeds, spices
Hungary	potatoes, onions, strawberries, spices, mushrooms, grapes, cherries, currant, pears, frozen chicken
Netherlands	poultry, liquid foods, spices, batter mix, endive, onions, potatoes, shrimp, haddock, cod, fresh vegetables, froglegs, rice products, malt, egg powder, dry blood protein
USSR	potatoes, grain, fresh fruits, fresh vegetables, beef, pork, rabbit, dried fruits, poultry, meat products, onions
United Kingdom	any food for consumption by patients requiring a sterile diet.
USA	wheat products, white potatoes, spices, enzyme preparations, pork.

In the U.S., only one of the thirty ^{60}Co irradiation plants is used commercially for food irradiation, and its output is generally for export. Worldwide, about 100 food irradiation plants are in operation.[10] The latest is a 3 MCi-capacity food irradiation plant being built in Marseilles, France.

The effect of irradiation on foods has in fact been more thoroughly studied than any other type of food preservation. While irradiation of living organisms is known to be harmful, particularly mutagenic, similar effects do not occur in irradiated foods.[4] Irradiation

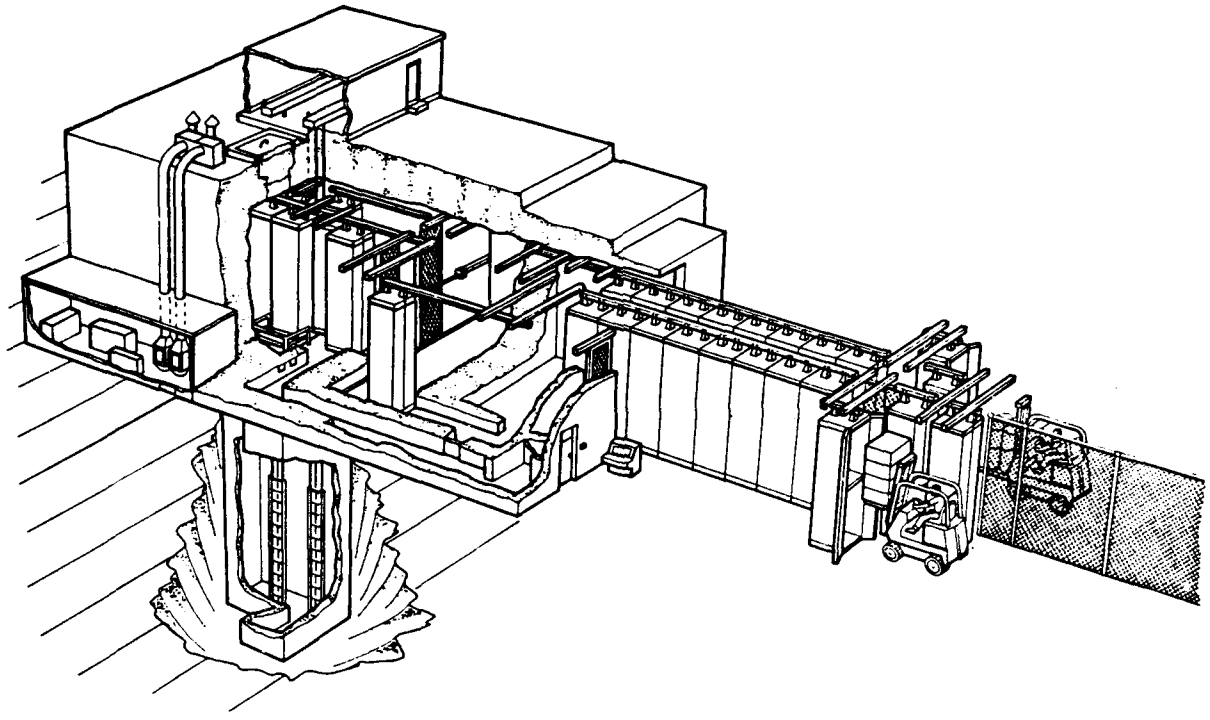


FIG. 5-1. Pallet-type food irradiation facility developed by AECL now in commercial operation.

is far more destructive of large molecules like DNA and proteins than the basic units of food, the amino acids, fatty acids and monosaccharides. These basic units are about a million times smaller than DNA and proteins and, it turns out, about a million times less sensitive to irradiation. In fact, heat sterilization destroys a much larger fraction of these basic units of foods.

For example, most enzymes are easily inactivated by heat, but not by irradiation. Animal feeding studies have shown no effect of any radiolytic products that do occur due to irradiation.

The question remains of public acceptance. It will be necessary to publicize experimental results as well as make the clear distinction between irradiating macromolecules like DNA and food components like amino acids.

The amount of ^{60}Co required for food irradiation is actually quite modest. Cobalt-60 produces 1130 Curies/gram (because only a fraction of the natural ^{59}Co is transmuted, about 5 grams of cobalt source are needed for one gram of ^{60}Co). At 2.5 MeV/disintegration, this corresponds to a gamma power of 16.7 w/gm. Assuming an irradiation facility can achieve 30% utilization of this power level, then 0.4 MJ/day per gram of ^{60}Co are available for food sterilization. On average, about 200 kilorads (2000 J/kg) are needed to sterilize solid foodstuffs and liquids such as milk and juice.[2] Assuming an average daily intake of 1.0 kg solid foods and 2.0 kg liquid food, then the quantity of ^{60}Co required per person is

$$3.0\ kg/day \times 2000\ J/kg//400{,}000\ J/day\ per\ gram = 0.015\ gram\ ^{60}Co$$

Because of natural decay, ten percent of this ^{60}Co must be replaced every year. This corresponds to about 1.6 Ci/person per year. Elsewhere in this report, we estimate the cost of ^{60}Co from fusion reactors to be well under \$1/Ci, which is negligible on a per-person annual basis.

To preserve the entire world's food supply, assuming the same consumption rate and estimating the global population at 5.0 billion at the turn of the millenium, 15 fusion reactors would be needed to provide continuous replacement of the decayed cobalt. This assumes commercial reactors can produce 500 MCi/yr, conservatively under that estimated in Ref. 9. Initial inventory buildup would require about 15 years with all 15 reactors operating. This is clearly a trivial worldwide investment considering the gains achieved and considering that these reactors also can supply electricity, process heat and fissile fuel while they are producing ^{60}Co.

A more speculative approach would be to have foodstuffs or other materials processed by direct fusion reactor radiation instead of by ^{60}Co. The material might be continuously fed through blanket modules, or irradiated externally by beams from large ports pene-

trating to the first wall. However, either method would probably lead to trace-induced radioactivity in the material from threshold and/or (n,γ) neutron reactions. An elegant method to avoid activation is sketched in Fig. 5-2. Manganese in slurry or dissolved form is circulated through the fusion reactor blanket, where much of it is converted to ^{56}Mn, then out of the bulk shield, and thence to a gamma irradiation chamber. Natural manganese has a relatively high cross section and ^{56}Mn a short half-life (2.5 hours) resulting in high gamma intensities, yet low shutdown radioactivity.

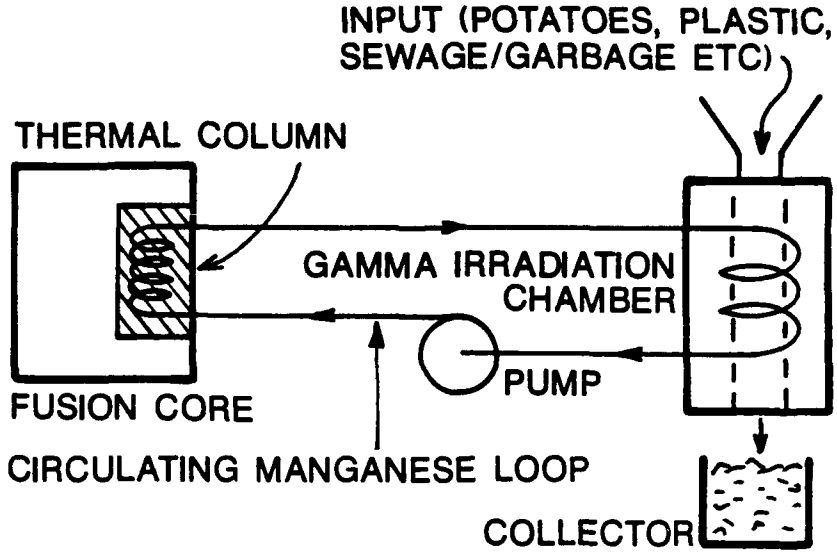


FIG. 5-2. Fusion irradiation loop concept.

5.3. SEWAGE

Irradiation of sewage would provide a large, low-cost source of sterile fertilizer and facilitate recycling of waste water back to the drinking supply. Food waste processing, for example, at large international airports and resorts, would be economically feasible above 10–20 tons/day, according to a study at Toronto International Airport.[6] Large airports such as London and New York accumulate as much as 50 tons of food waste per day.

Sewage sterilization would be of great value in areas where drinking water supplies are uncertain or expensive, or where concern exists about disposal. A demonstration

sterilization facility capable of processing 25 wet tons/day is currently being considered for New Mexico.[5] To estimate costs, we assume the average person requires 10 gal/day of quality water for drinking and cooking (water for bathing and flushing can be lower in quality). No information has been found on the dose level required for waste sterilization, so we assume 500 krad, which is 2–3 times the nominal dose for food preservation. Using the same methods as above, the ^{60}Co requirement for those 10 gal/day (38 kg/day for water) is 0.22 gm/person. This assumes 100% utilization of the ^{60}Co rather than 30%, justified by the fact that it should be easier to envelop the cobalt with liquid waste rather than solid waste. Since the annual replacement rate is one-tenth of the total inventory, the cost of the cobalt at \$0.25/Ci would be about \$6/year per person. The low cost per Curie would be that from a fusion plant that also sells electricity. If, instead of the 10 gal/day, we include all water usage (about 200 gal/day), then the cost of the cobalt rises to \$70/year per person, which is high but still not unreasonable.

5.4. SUMMARY

In summary, medical supply, food and waste sterilization by irradiation with gamma rays from ^{60}Co (or ^{56}Mn) produced in fusion reactors present attractive alternate applications of fusion energy. Provided the cost per Curie can be kept reasonably low, the high transmutation rate in fusion reactors could provide a significant ^{60}Co inventory in a reasonable time. For food irradiation, only about 15 fusion reactors are needed to provide the world's requirements of ^{60}Co.

There are dozens of other applications of ^{60}Co or other hard gamma emitters such as cancer therapy, electronics testing, polymer irradiation, radiography, etc. These applications generally require much lower doses and are expected to generate only modest ^{60}Co demand.

References

[1] "Estimated Cesium-137 Demand," CH_2M-Hill Co, Inc., 1980.

[2] P.S. Elias and A.J. Cohen, "Recent Advances in Food Irradiation," Elsevier Biomedical Press, 1983.

[3] Y. Henon, "Towards Commercial Use of Food Irradiation in France," FAO/IAEA Seminar for Asia and the Pacific on Practical Application of Food Irradiation, IAEA-SR-129, 1985.

[4] A. Brynjolfsson, "Wholesomeness of Irradiated Foods: A Review," **J. Food Safety 7** (1985), 107.

[5] J. Sivinski, "Irradiation of Sewage Sludge using Cs-137," Sandia National Laboratories Report SAND-80-7097, 1980.

[6] F.M. Fraser, "Cobalt-60 Availability for Radiation Processing," 17th Japan Conference on Radiation and Radioisotopes, 1985.

[7] T. Peckinpaugh, "Regulatory and Legislative Progress in Food Irradiation," ANS 1985 Annual Meeting, Boston, Proceedings p. 200.

[8] J.S. Sivinski, "Irradiation of Pork for Trichinosis Control," *ibid.* p. 6.

[9] B.A. Engholm, E.T. Cheng, and K.R. Schultz, "Fusion Reactor Production of ^{60}Co," GA Technologies Report GA-A18070, July 1985.

[10] "Food Preservation by Irradiation: A Fresh Look," **Background INFO**, AIF, October 1983.

6. ISOTOPE PRODUCTION

Because D-T and D-D fusion reactions produce neutrons, an obvious application of fusion energy is to use these neutrons to produce useful isotopes. These can include both radioactive and stable nuclides.

6.1. INTRODUCTION

In a fission reactor, 2.5 neutrons are produced per 200 MeV of total thermal energy. One is needed for fission-producing reabsorption in the fissile nuclide, leaving 1.5 for other absorptions. In a typical LWR, one is generally lost to parasitic absorption in fuel, structure, moderator and control materials, leaving ~0.5 net excess neutron for breeding of additional fuel. Some fraction of the excess neutrons may be diverted to produce other products, such as ^{60}Co, at some penalty to the fuel cycle cost.

In a fusion reactor, one neutron is produced for each 17.6 MeV of fusion energy. Neutron multiplication in the blanket will produce a total thermal energy of 20 to 25 MeV per fusion event. Neutron multiplication by (n,2n) and (n,n't) reactions in the blanket will allow production of from 0.2 to 0.8 excess neutrons per fusion reaction, in addition to production of one tritium atom. Fusion thus has the potential for producing as many as eight excess neutrons per 200 MeV of thermal energy, compared to 0.5 for fission, all of which could be used for producing radioisotopes or stable nuclides. The comparison between fission and fusion is shown pictorially in Fig. 6-1. Incorporation of seed materials into a fusion reactor to utilize these excess neutrons is clearly attractive, whether the seed materials be additional lithium, for excess tritium production; thorium or depleted uranium, to breed fissile fuel; or, in the present case, parent nuclides to produce useful products such as ^{60}Co.

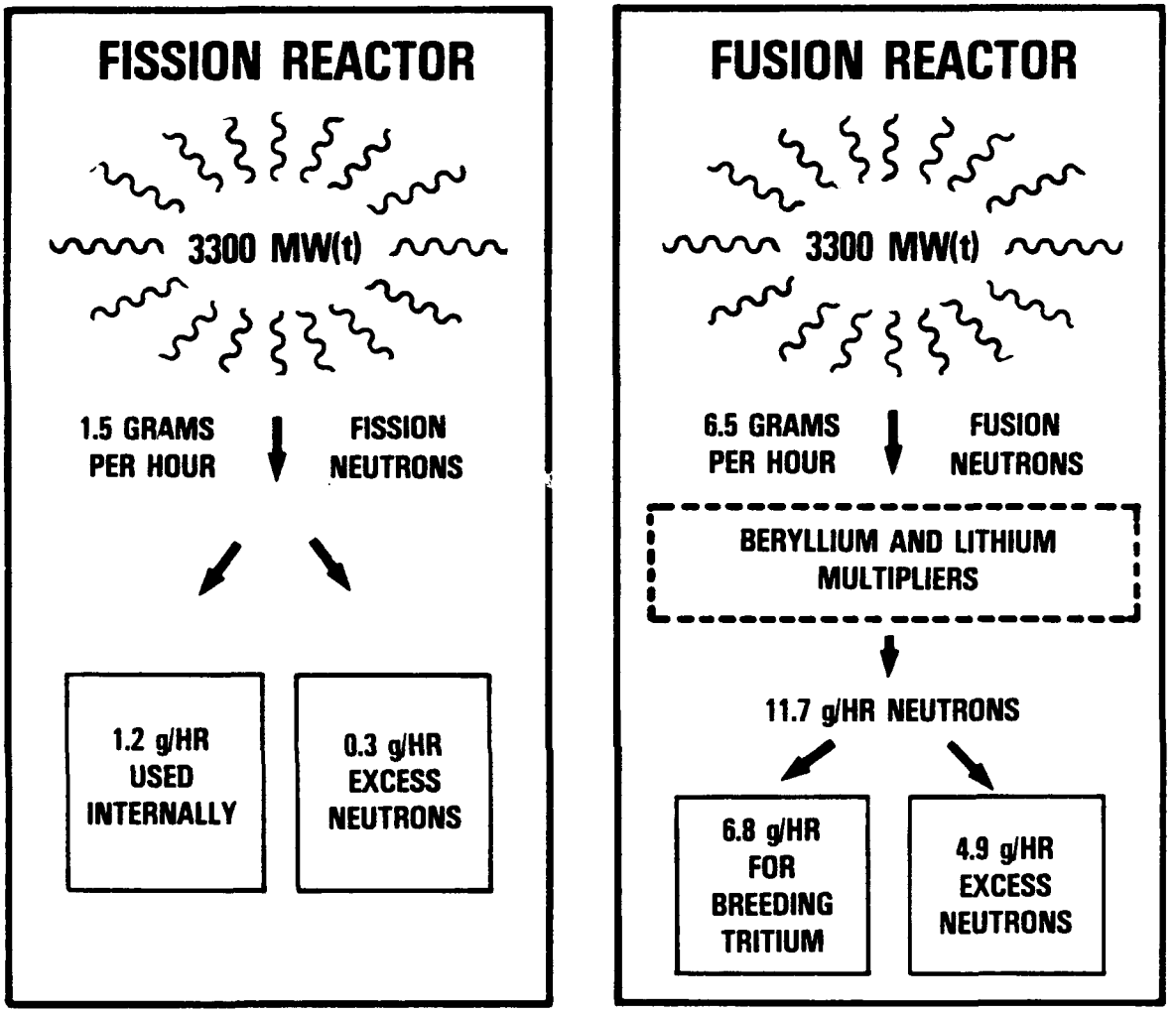


FIG. 6-1. Comparison of excess neutron production in fission and fusion power reactors.

In the following subsections, various radioisotope and stable nuclide products are considered individually. Much of our detailed analysis has addressed ^{60}Co, with production rates for other isotopes generally deduced from the ^{60}Co results.

6.2. RADIOISOTOPE PRODUCTION – COBALT-60

Cobalt-60 is one of the radionuclides used extensively today for radiation processing and sterilization of material, and for medical therapy applications.

6.2.1. Definitions and Data

Some definitions and data for ^{60}Co are shown in Table 6-1. The buildup and decay factors would apply to any radioisotope. Plots of the buildup and decay of ^{60}Co are shown in Fig. 6-2. The decay curve shows the fraction of the ^{60}Co present at zero time that will be left at time t. The buildup curve shows the fractional buildup of ^{60}Co towards the equilibrium value, $(N_{60}/N_{59}) = \sigma\phi/\lambda$, where $\sigma\phi$ is the reaction rate and λ is the decay constant. Application of these curves shows, for instance, that one cobalt rod irradiated for two years and sold to the user at that point, followed by another rod irradiated for two years, actually produces 10% more saleable ^{60}Co than one rod irradiated for four years.

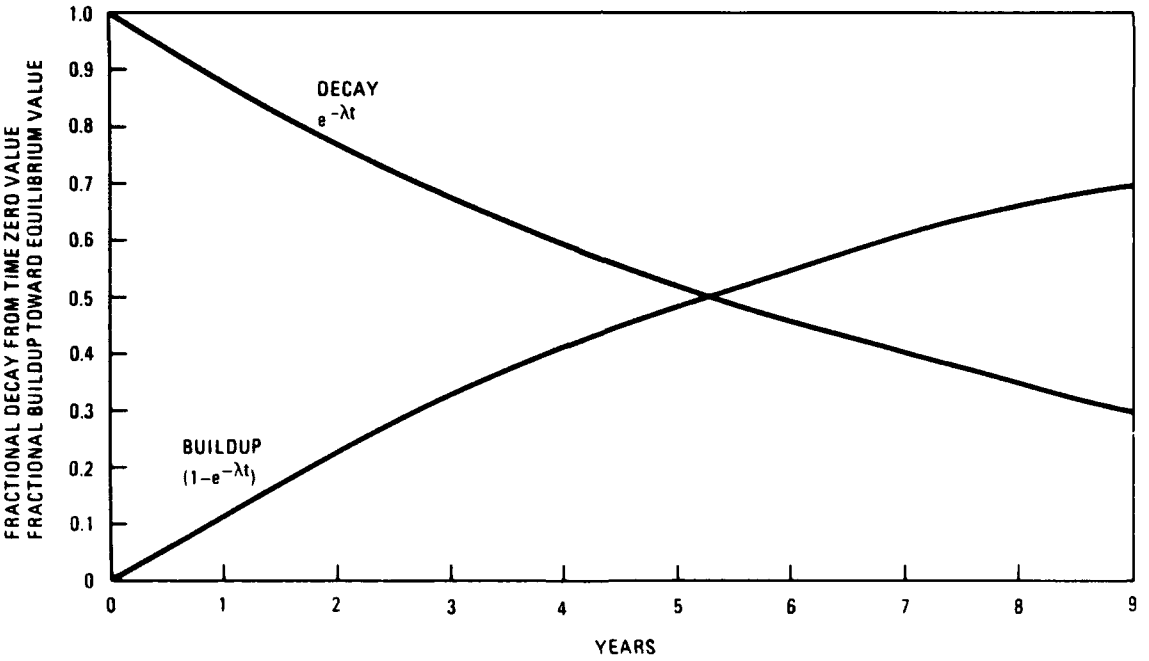


FIG. 6-2. Cobalt-60 buildup and decay.

The equilibrium specific activity in Ci/g is given by $A_{sp}^{\infty} = A\sigma\phi/3.7\times10^{10}M$ where A is Avogadro's number and M is the molecular weight of the material containing the ^{60}Co. The maximum theoretical specific activity of ^{60}Co is 1130 Ci/g.

TABLE 6-1
Definitions and Data

Cobalt metal density	8.9 g/cm^3
Thermal neutron cross section for production of ^{60}Co from cobalt, σ	37 barns
^{60}Co half-life ($\tau_{\frac{1}{2}}$)	5.27 years
^{60}Co decay constant (λ)	4.17×10^{-9} sec^{-1}
Photon emissions per ^{60}Co decay	1.17 MeV (100%) 1.33 MeV (100%)
Disintegrations per sec in 1 curie (Ci)	3.7×10^{10}
Specific activity of 100% ^{60}Co	1130 Ci/g
^{60}Co buildup in constant flux	$(1 - e^{-\lambda t})$
^{60}Co decay after removal	$e^{-\lambda t}$
Appropriate gamma dose rate 1 m from a 1 MCi ^{60}Co point source	4.5 Mrad/hr

One concern is the burnup of ^{60}Co by neutron absorption. For estimating this, MCNP was employed to display absorption cross sections for ^{59}Co and ^{60}Co (the former from RMCSS, the latter from LLLDOS2), as shown in Fig. 6-3. The fact that the ^{60}Co cross section is a factor of 20 below the ^{59}Co cross section indicates that burnup of ^{60}Co will not be a problem.

6.2.2. Cobalt-60 Production from Fission Reactors

For comparison purposes, we summarize the situation of ^{60}Co production from fission reactors.

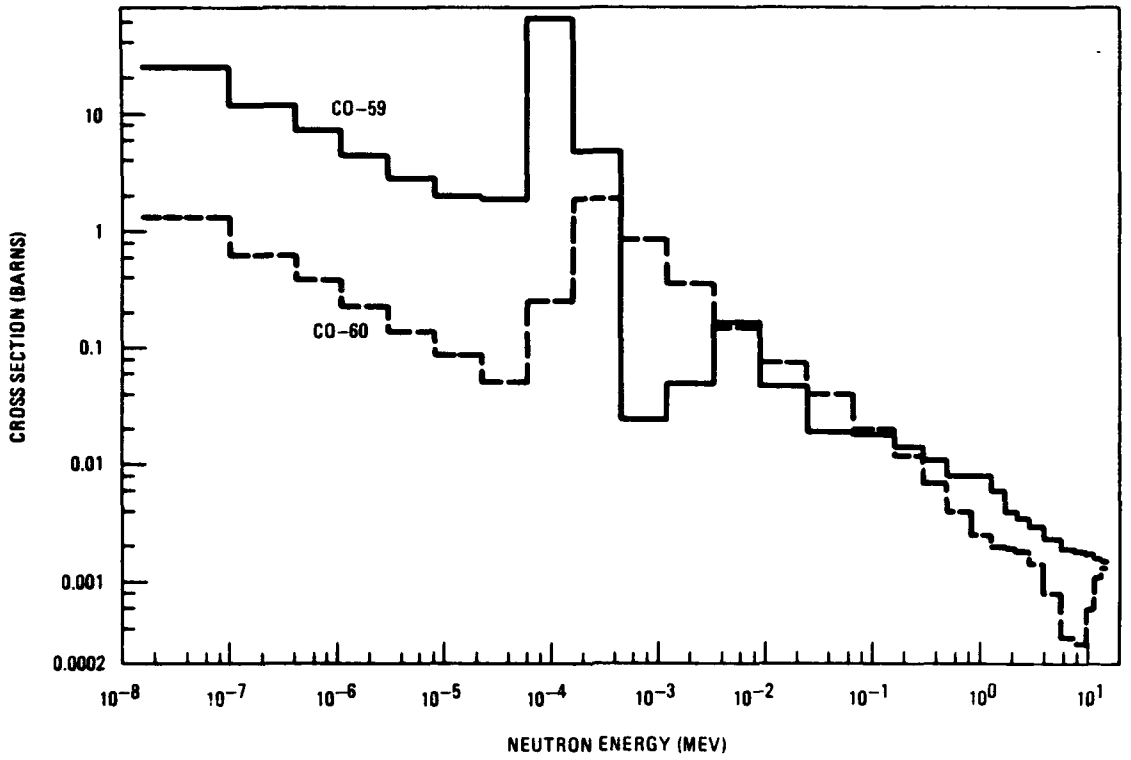


FIG. 6-3. Cobalt-59 and ^{60}Co (n,γ) evaluated cross sections.

The world's leading producer of ^{60}Co is Atomic Energy of Canada Ltd. Utilizing Ontario Hydro's CANDU heavy-water reactors, AECL supplies 77% of the world's ^{60}Co (69 MCi out of 90 MCi), and has designed over half of the ^{60}Co irradiation plants (74 out of 135). Currently, there are 11 CANDU reactors capable of producing a total of 45 MCi ^{60}Co/yr. Cobalt "adjuster rods" accumulate 60 to 100 Ci/g of cobalt (*i.e.*, about 6% of the ^{59}Co is converted to ^{60}Co). If cobalt rods were to be incorporated into fuel channels, the production could reach 6 to 7 MCi/reactor-year.

The average thermal neutron flux in a Pickering Station reactor[1] is 1×10^{14} n/cm^2-sec at an average energy of 0.05 eV. Typically, a rod is exposed 1.5 years. Using these figures, it is easy to confirm the 60–100 Ci/g quoted above.

The latest ^{60}Co cost information currently available is as follows:

- late 1984 price[2] $1.00/Ci
- 1985 price[3] $1.00/Ci installed
- 1985 price[4] $1.10 to $1.25/Ci f.o.b. AECL
- Annual price increase over next five years = 10%

  (although the price went up 21% during the last year)

The U.S.-produced supply of ^{60}Co is controlled by DOE Defense Projects and by DOD. For example, the 2 MCi/yr production from Idaho's ATR is allocated by Oak Ridge, under DOE direction. The level of production at Savannah River, if any, is not known, nor the DOE/DOD "installed capacity" (*i.e.*, for Army food irradiation). It is therefore easy to understand that over 90% of the ^{60}Co used by Johnson & Johnson and Isomedix is supplied by Canada.

For a short period of time, ^{60}Co was produced in the U.S. Big Rock Point Reactor, where cobalt rods were used to even out the power distribution. More recently, Duke Power has been evaluating the possibility of ^{60}Co production in LWRs. Their studies[5] indicate that the current ^{235}U enrichment might have to be increased to compensate for the negative reactivity effect of the cobalt pins. A change in fuel enrichment would likely trigger a completely new safety analysis of the reactor to meet NRC requirements. Production efficiency would not be as high as in a CANDU reactor because the LWR thermal flux is lower (5×10^{13}) and harder (0.1 eV).

Spent control rod assemblies removed from LWRs frequently contain parts composed of high-cobalt alloys such as stellite, which have accumulated significant ^{60}Co activity. About 50,000 Ci/yr of ^{60}Co may be available from a typical LWR unit. If the control rod assemblies were intentionally loaded with cobalt, this figure would be substantially increased.

The Advanced Test Reactor at INEL currently produces 2 MCi of ^{60}Co per year at a specific activity of 100 Ci/g. The ^{60}Co production program began in June 1981; irradiation period is about 20 months. A study is underway on the possible recommissioning of the Engineering Test Reactor, which would be capable of producing 7.5 MCi of ^{60}Co per year.

A summary of ^{60}Co production capabilities is presented in Table 6-2.

Because of the CANDU reactor shutdowns in 1983 and 1984, there is an accumulated shortfall of about 20 MCi of ^{60}Co, causing AECL to ration its product in 1985. AECL's own estimate of ^{60}Co production is 45 MCi/yr currently, rising to about 60 MCi/yr in 1991, assuming adjuster rod loading only. If fuel channels were used, 1991 production could reach 100 MCi/yr.

These and other production data are plotted in Fig. 6-4.

TABLE 6-2
Worldwide ^{60}Co Production Capability

		^{60}Co Production (MCi/yr)	
Reactor(s)	**Location**	**Current**	**Potential**
CANDU (11)	Canada	45	70–80
ATR (1)	Idaho	2	2
ETR (1)	Idaho	0	7.5
LWRs (100) (control rod assembly)	U.S.	5	50 (?)
Foreign (?)	Europe, Japan, etc.	7 (?)	50 (?)

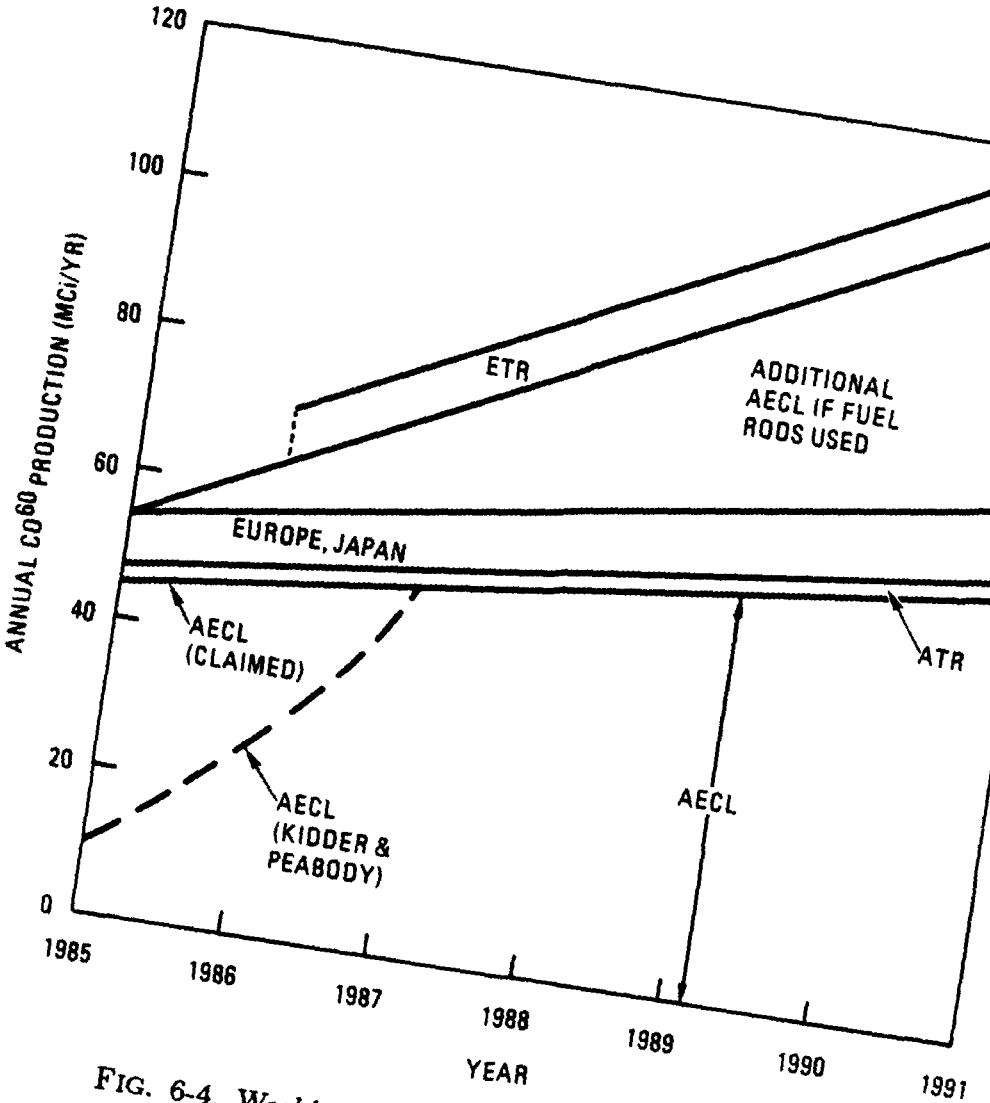


FIG. 6-4. Worldwide ^{60}Co production projections.

Food processing, discussed elsewhere, will create a large new demand for ^{60}Co. Johnson & Johnson, Isomedix, and AECL estimate that the growth of world demand for ^{60}Co is in the range of 6% to 25%/yr, depending upon the assumptions made regarding food irradiation. Figure 6-5 illustrates these growth projections, starting with a worldwide installed inventory of 90 megacuries. Note that since 12% of the ^{60}Co activity decays off in a year, current replenishment requirements are 11 MCi/yr worldwide, even if there is zero growth.

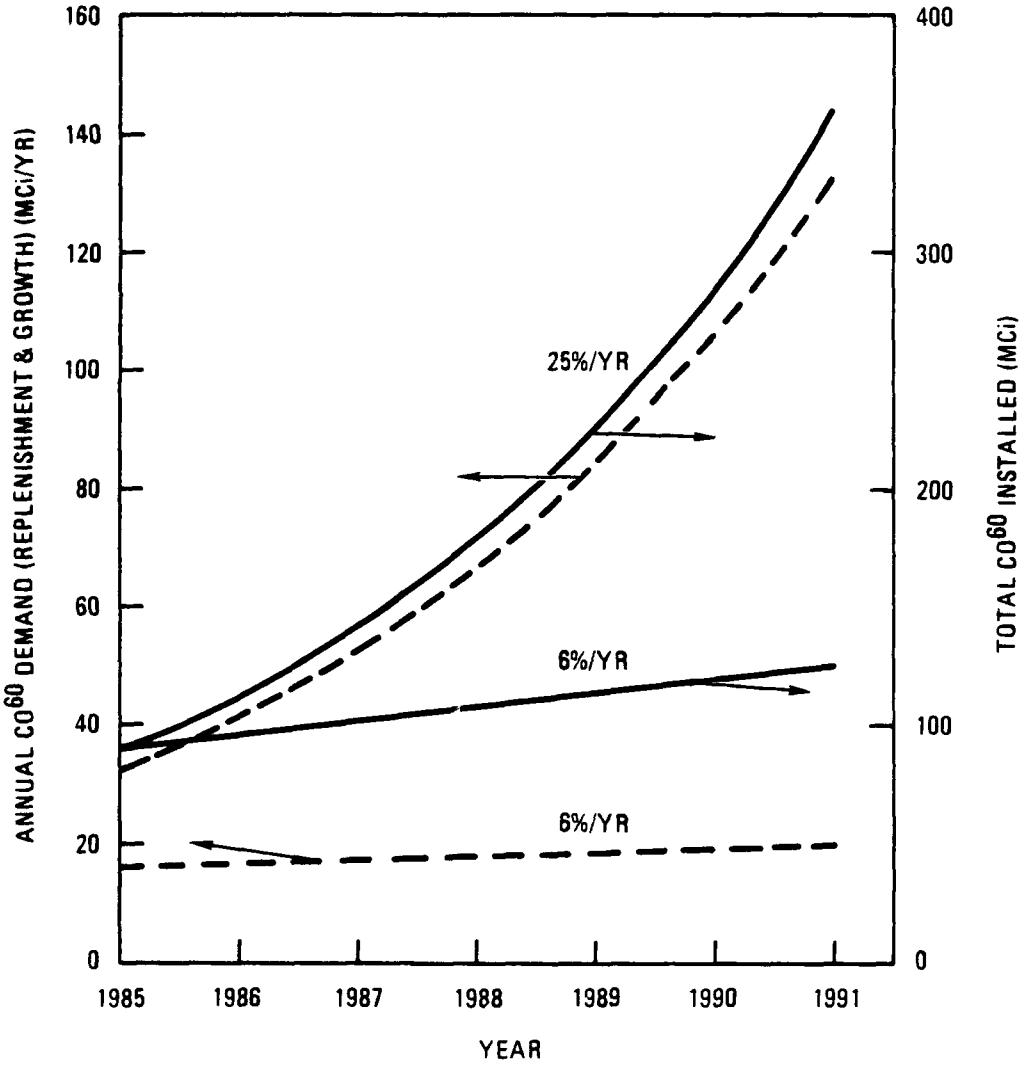


FIG. 6-5. Worldwide ^{60}Co demand projections.

Comparison of Figs. 6-4 and 6-5 shows the following:

- There is a gigantic gap between U.S. ^{60}Co consumption and U.S. ^{60}Co production.
- The worldwide consumption/production gap is much smaller.

- By 1991, in the 25% demand growth scenario, all available production of ^{60}Co (120 MCi/yr) could be used. If CANDU fuel rods were not used and ETR were not recommended, a demand growth of only ~15%/yr could be supported.

Thus, the motivation for fusion production of ^{60}Co is to fill a coming supply gap and hopefully to provide the product at less than $1.00/Ci.

6.2.3. Cobalt-60 Production from Fusion Reactors

An important characteristic of radiation sources is the specific activity, Ci/g. A high specific activity is needed for ^{60}Co irradiators (^{60}Co is currently provided at a specific activity of ~60 Ci/g). High specific activity can be achieved by irradiating long enough to achieve a significant fraction of the equilibrium activation and by irradiating at high flux so that at equilibrium a high specific activity would be achieved.

The brief fusion reactor studies described below illustrate the potential of producing substantial quantities of ^{60}Co, preferably at high specific activities and without penalizing other reactor performance levels. We explore both near-term application for ^{60}Co production in ignition test reactors as well as longer term commercial fusion power reactor application.

6.2.3.1. Cobalt-60 Production in Copper Coils of a Small Ignition Machine. The U.S. DOE/OFE Ignition Studies program is currently investigating designs of compact low cost tokamak ignition test reactors that utilize normal coils. These small machines feature normal-conducting copper coils located close to the plasma. As part of an effort to evaluate the relation between major radius, wall loading, and temperature distribution for the machines, a parametric study was made of the inboard nuclear heating rates vs. major radius, using the ONEDANT 1-D discrete ordinates transport code.[1] The first wall is a simple Inconel structure. Immediately following the first wall is a normal-conducting copper coil. Thus, this model is well suited to evaluate the production of ^{60}Co in copper via the ^{63}Cu (n,α) threshold reaction, in the presence of a relatively unmoderated flux of 14-MeV neutrons. The cross section for this reaction is approximately 40 millibarns.

The absolute flux levels in the ONEDANT calculation were normalized to give a neutron wall loading on the copper coil surface of 1 MW/m^2, and were then folded with the ^{63}Cu(n,α) cross sections. The results were disappointing. ^{60}Co production per full-power year was found to be only 0.84 Ci/cm^3 or ~0.1 Ci/g Cu at the plasma side of the coil, dropping a factor $1/e$ every 7 or 8 cm into the coil. For a more realistic load factor of 25%, the ^{60}Co production would be 0.21 Ci/cm^2-yr or ~0.025 Ci/g. The low production is due to the low cross section.

Discussions were held with GA reprocessing personnel regarding the cost to extract the ^{60}Co from the copper. In the absence of a detailed design study, a comparison with spent fission fuel reprocessing seemed the best approach. Currently, fissile fuel reprocessing runs $1500/kg of metal feed material. Therefore, if the cost is constant per gram, the ^{60}Co extraction cost would be $15/Ci to $60/Ci. Even a factor of five less than this estimate is non-competitive, so one can rule out the production of ^{60}Co from copper in a fusion reactor.

6.2.3.2. Cobalt-60 Production in First Wall of Large Ignition Machine. The ONEDANT and RACC codes were used for calculations of the radioactivity inventories in a typical first wall of a large ignition machine, a water-cooled stainless steel design. Some parameters are shown in Table 6-3.

6.2.3.3. Cobalt-60 Production in Removable Pins in Ignition Machine. MCNP Monte Carlo problems were set up for a TFCX-type machine such as the one considered in Ref. 6. By utilizing a 3-D transport code in the present study, an evaluation could be made of inserting individual cobalt pins into the first wall coolant annulus.

In the cases summarized in Table 6-4, the first wall was backed by water-cooled copper coils (90% Cu, 10% H_2O) and full wall coverage was assumed. We considered one-and two-year irradiation times, at 25% and 50% load factor. Wall loads of 1 and 4 MW/m^2 were used. The heterogeneous first wall model with cobalt pins inserted in the coolant annulus is shown on Fig. 6-6.

TABLE 6-3
Cobalt-60 Produced in a Large Ignition Machine

Reactor power	250 MW
Wall loading	~1 MW/m^2
Operating time	2×10^5 full-power sec in 10 yr
^{59}Co content in SST	2000 ppm
^{60}Co produced in first wall	10,000 Ci (1700 in first year)
Load factor	50%
First wall material	High Co alloy (10%)
^{60}Co produced in first wall	360 MCi in 10 ; rs (60 MCi in first year)
Specific activity	3 Ci/g (in first year)

TABLE 6-4
Cobalt-60 Production from TFCX-type Machine

Description	Heterogeneous array of cobalt pins inserted in the first wall cooling water.							
Irradiation time (Cal years)	1				2			
Atoms ^{60}Co produced per fusion neutron	0.210				0.197			
Load factor	25%		50%		25%		50%	
Wall loading (MW/m^2)	1	4	1	4	1	4	1	4
Specific activity (Ci ^{60}Co/g of cobalt)	2.95	11.8	5.9	23.6	5.55	22.2	11.1	44.4
MCi ^{60}Co produced for 250MW fusion power	16	65	32	130	30	122	61	244

6.2.3.4. Ignition Machine Summary. To summarize ^{60}Co production in ignition reactors:

- Production of ^{60}Co in the copper coils of even an unmoderated configuration does not result in high enough specific activity (Ci ^{60}Co/g Cu) to warrant the expense of extraction.

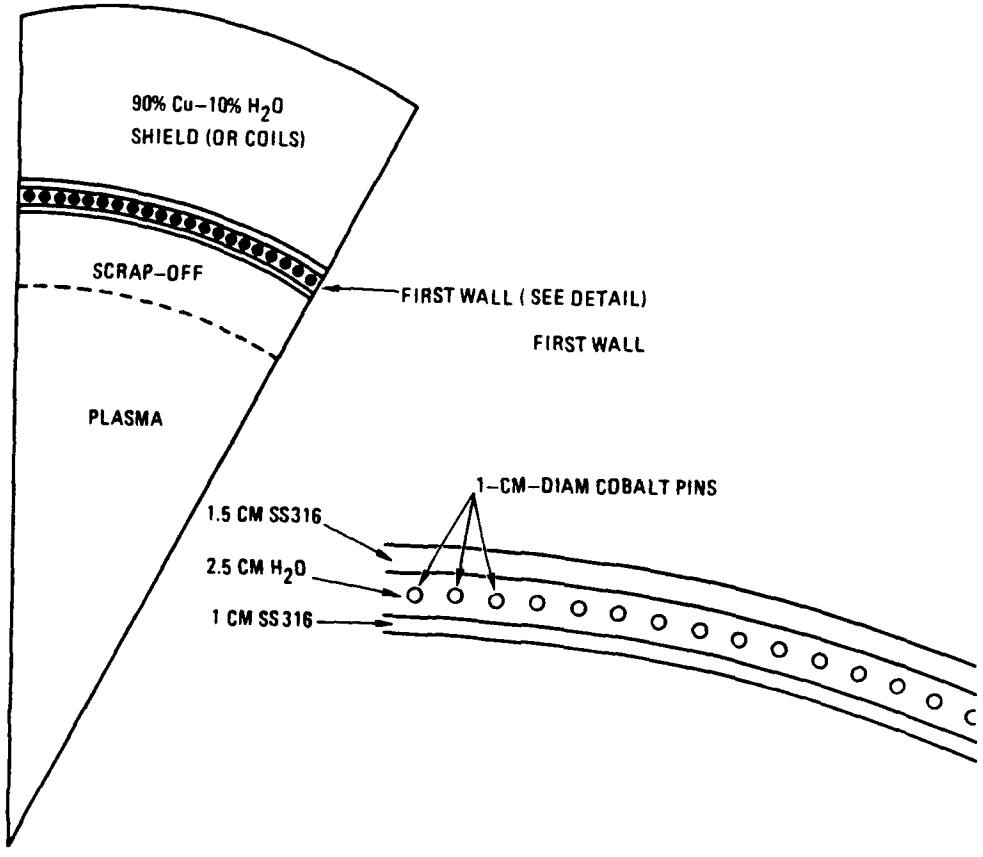


FIG. 6-6. Model for MCNP heterogeneous calculation.

- Production of ^{60}Co in a high-cobalt first-wall alloy would be attractive except, again, the first wall would have to be removed and reprocessed every two or three years. However, this concept might be attractive for a high-wall-load machine where frequent first-wall replacement is planned.

- Production of ^{60}Co in pure cobalt pins (or rods or plates) immersed in the first wall water-cooling annulus appears very attractive. As much as 100 MCi/yr could be produced. Removed rods have a specific activity of up to 45 Ci/g and are essentially ready to use by the medical or food irradiation facility.

We conclude that by using the removable cobalt pin technique, near-term fusion reactors with only modest availability (25% to 50%) could produce a significant quantity of ^{60}Co at high specific activity with minimal impact on the design or operation of the device.

6.2.3.5. Cobalt-60 Production from Power Reactors. The true potential of fusion for the production of such products as ^{60}Co is brought out in the following full-power commercial reactor blanket studies. These studies assume the existence of a commercial fusion reactor and then examine alternative blankets that could be used for ^{60}Co production. The designs are specified on the basis that they produce adequate tritium for reactor self-fueling and that they not compromise electricity production.

Beryllium Multiplier Blanket. Beryllium has the highest neutron multiplier among all nonfissionable materials. Theoretically, it can produce more than two additional neutrons out of each incident 14 MeV neutron in a 1-m-thick beryllium medium with no structure. However, in a practical fusion blanket with structural and breeder materials, a beryllium blanket is capable of producing about 0.8 excess neutrons per D-T neutron in addition to breeding adequate tritium (*i.e.*, tritium breeding ratio $\geqslant$1.1 tritons per D-T neutron). A detailed study of this capability was carried out a few years ago as part of the fusion breeder program led by the Lawrence Livermore National Laboratory.[7] The reference design selected by the LLNL program was adopted in this report for ^{60}Co production with minor modifications.

The beryllium blanket used for ^{60}Co production is described in Table 6-6. The blanket, which has a vanadium alloy structure, consists of a 5 mm first wall, a 0.6 m breeding zone, and a 0.3 m shield. The breeding zone comprises 8% structure, 36.8% liquid lithium, 54.2% beryllium (90% dense), and 1% cobalt, all by volume. The shield zone consists of 80% manganese steel (Fe1422) and 20% H_2O, also by volume. The ^{6}Li content in lithium was adjusted parametrically to obtain the best performance in the blanket in terms of tritium breeding. The optimal design was achieved when the ^{6}Li content is about 1% in lithium. The performance of this blanket is summarized in Table 6-6.

TABLE 6-5
One-Dimensional Neutronic Model of the Beryllium Blanket by Zones and Materials Compositions

Zone	Thickness	Materials Compositions (by volume)
First Wall	5 mm	100% V-15Cr-5Ti
Breeding Zone (T and ^{60}Co)	0.6 m	8% V-15Cr-5Ti+36.8% liquid lithium (1% ^{6}Li+54.2% beryllium (90% dense) +1% cobalt
Shield	0.3 m	80% Fe1422+20% H_2O

The beryllium blanket is optimized to give a ^{60}Co production rate of 0.87 ^{60}Co atoms per D-T neutron, as seen in Table 6-5, in addition to a tritium breeding ratio of 1.14. This blanket also enhances the nuclear heating, which is about 22.2 MeV per D-T neutron. The blanket energy multiplication thus obtained is as high as 1.6. A fusion reactor of 1000 MW fusion power will produce about 950 MCi of ^{60}Co every year assuming continuous operation and full blanket coverage. A total of about 1500 MW thermal power will be available including blanket nuclear heating and alpha power, which is about 20% more than that available from a non-beryllium blanket such as lithium, $Li_{17}Pb_{83}$ and Li_2O blankets. Assuming a net thermal efficiency of 34%, the beryllium reactor will provide an electricity output of about 500 MWe. Taking into account a capacity factor of 0.75, the total income of this reactor from selling ^{60}Co at \$1/Ci would be \$700 M/yr compared with the \$200 M/yr revenues it would earn from sale of electricity at 60 mills per kWh.

One of the important considerations in the production of ^{60}Co is the specific activity. Figure 6-7 depicts the spatial distribution of ^{60}Co specific activity after one year continuous irradiation at 5 MW/m^2 wall load in the beryllium blanket, given in units of Curie per gram cobalt in the blanket. As shown in the figure, the ^{60}Co specific activity is as high as 230 Ci/g near the first wall and drops to about 8 Ci/g near the shield. The average

TABLE 6-6
Neutronic Performance of the Beryllium Blanket

Tritium Breeding (T/D-T Neutron)	
$^{6}Li(n,\alpha)T$	0.8772
$^{7}Li(n,n'\alpha)T$	0.1498
$^{9}Be(n,t)$	0.0153
Tritium Breeding Ratio	1.1423
^{60}Co Production (^{60}Co atoms/D-T Neutron)	
$^{59}Co(n,\gamma)$	0.8670
Nuclear Heating (MeV/D-T Neutron)	
First Wall	0.66
Breeding Zone	21.78
Shield	1.41
Blanket Subtotal (First Wall and Breeding Zone)	22.44
Total	23.85

specific activity after one year irradiation is about 90 Ci/g. The composite pebble fuel concept (beryllium pebbles with cobalt metal snap-rings), which originated in the fusion breeder program, can be adopted for the ^{60}Co production blanket to provide ^{60}Co product management. The cobalt snap-rings are used with the beryllium pebbles according to the proper materials ratio (2% by volume cobalt in the Be-Co pebbles). These pebbles can be circulated out of separate blanket subregions when they reach the desired specific activity in the various subregions.

$Li_{17}Pb_{83}$ Multiplier/Breeder Blanket. Lead is also a neutron multiplier, although the multiplication factor of lead is somewhat less than that of beryllium. A promising fusion blanket employing $Li_{17}Pb_{83}$ as both tritium breeder and coolant was explored by the

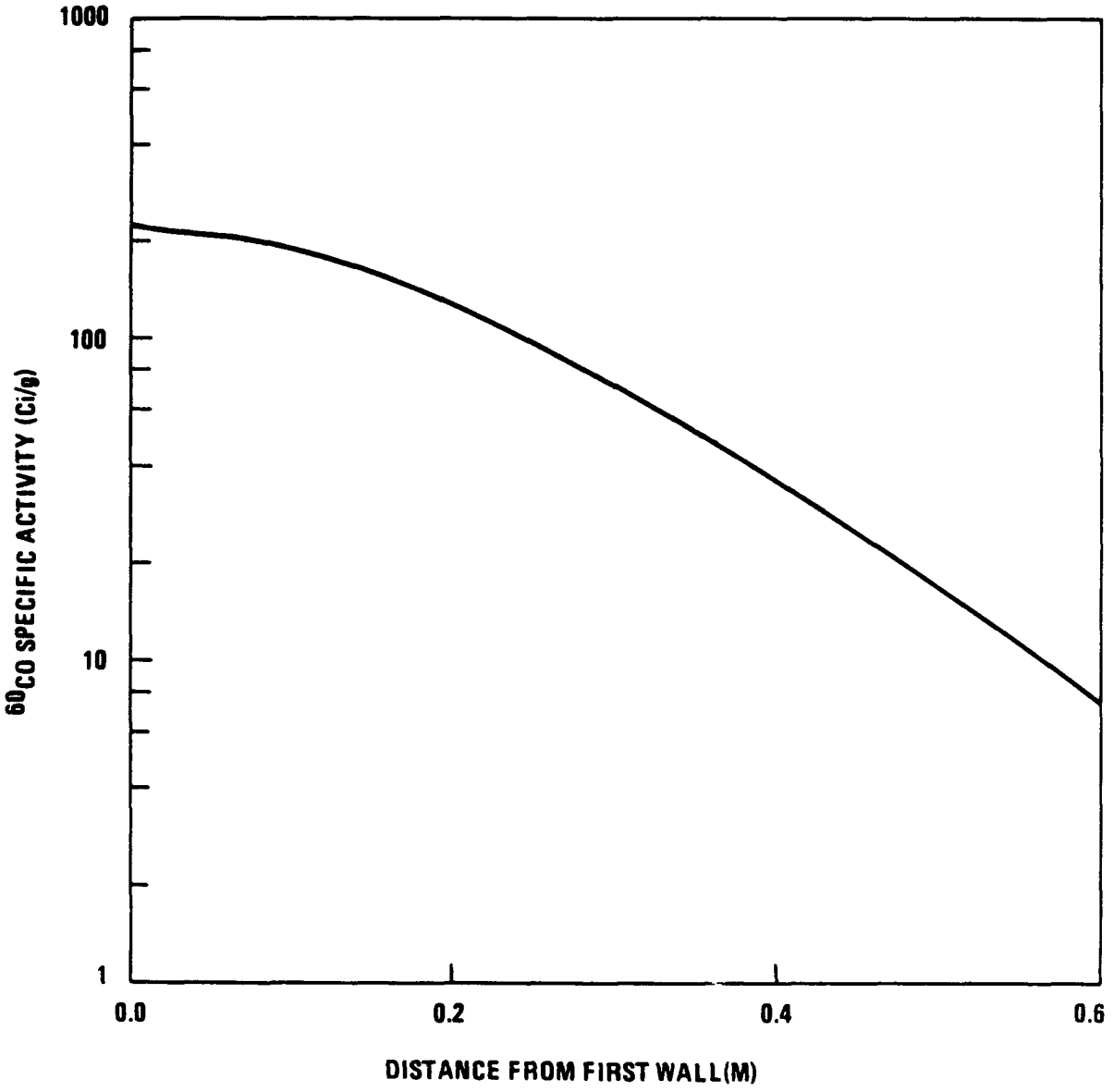


FIG. 6-7. Spatial distribution of ^{60}Co specific activity (Ci/g) in a beryllium blanket (one year continuous irradiation at 5 MW/m^2 wall loading.)

Blanket Comparison and Selection Study (BCSS) program led by the Argonne National Laboratory.[8] The $Li_{17}Pb_{83}$ blanket design from that study has been adopted here and

its compositions modified to include cobalt for ^{60}Co production. We adjusted the cobalt volume fraction in the $Li_{17}Pb_{83}$ breeding zone to optimize the ^{60}Co production rate and the specific activity. Table 6-7 gives the blanket zones and materials composition of the modified designs. The blanket consists of a 5 mm first wall, a 0.6 m $Li_{17}Pb_{83}$-cobalt zone, a 0.4 m $Li_{17}Pb_{83}$ manifold zone, and a 0.3 m shield. The first wall and structural materials are also the vanadium alloy, V-15Cr-5Ti. The $Li_{17}Pb_{83}$-cobalt zone is composed of 7.1% structure, 63.7% $Li_{17}Pb_{83}$, and 10% cobalt, all by volume. The manifold zone consists of 10% structure, 20% $Li_{17}Pb_{83}$, and 70% manganese steel, Fe1422. The ^{6}Li enrichment in lithium is the same as that employed in the BCSS design, 30%. The neutronic results are summarized in Table 6-8.

TABLE 6-7
One-Dimensional Neutronic Model of the $Li_{17}Pb_{83}$ Blanket by Zones and Materials Compositions

Zone	Thickness	Materials Compositions (by volume)
First Wall	5 mm	100% V-15Cr-5Ti
Breeding Zone	0.6 m	7.1% V-15Cr-5Ti+63.7% $Li_{17}Pb_{83}$ (30% ^{6}Li)+10% cobalt
Manifold	0.4 m	10% V-15Cr-5Ti+20% $Li_{17}Pb_{83}$ (30% ^{6}Li)+70% Fe1422
Shield	0.3 m	80% Fe1422+20% H_2O

As shown in the table, the $Li_{17}Pb_{83}$ blanket gives a ^{60}Co production rate of about 0.3 ^{60}Co atoms per D-T neutron, or about 330 MCi per year ^{60}Co for a 1000 MW fusion power reactor assuming continuous operation. The blanket energy multiplication is about 1.29 (18.13 MeV/D-T neutron). Figure 6-8 depicts the spatial distribution of specific activity of ^{60}Co in the breeding zone. Because of the lower production rate and larger volume

TABLE 6-8
Neutronic Performance of the $Li_{17}Pb_{83}$ Blanket

Tritium Breeding (T/D-T Neutron)	
$^{6}Li(n,\alpha)T$	1.0241
$^{7}Li(n,n')T$	0.0136
Tritium Breeding Ratio	1.0377
^{60}Co Production (^{60}Co Atoms/D-T Neutron)	
$^{59}Co(n,\gamma)$	0.3054
Nuclear Heating (MeV/D-T Neutron)	
First Wall	0.33
Breeding Zone	15.46
Manifold	2.34
Shield	0.23
Blanket Subtotal (exclude shield)	18.13
Total	18.36

fraction of cobalt (10%) than in the beryllium blanket, the ^{60}Co specific activity reaches a maximum value of only about 6 Ci/g after one year irradiation at 5 MW/m^2 wall load. The specific activity drops by a factor of 4 to reach the minimum at about 0.55 m from the first wall, and then increases again toward the back of the breeding zone because of neutron backscattering. The average specific activity over the entire blanket after one year irradiation is about 3.1 Ci/g. Compared to the beryllium blanket described above, the $Li_{17}Pb_{83}$ blanket gives a specific activity lower by a factor of 30. The specific activity could be increased by reducing the amount of cobalt placed into the blanket, but at the expense of the total number of Curies produced.

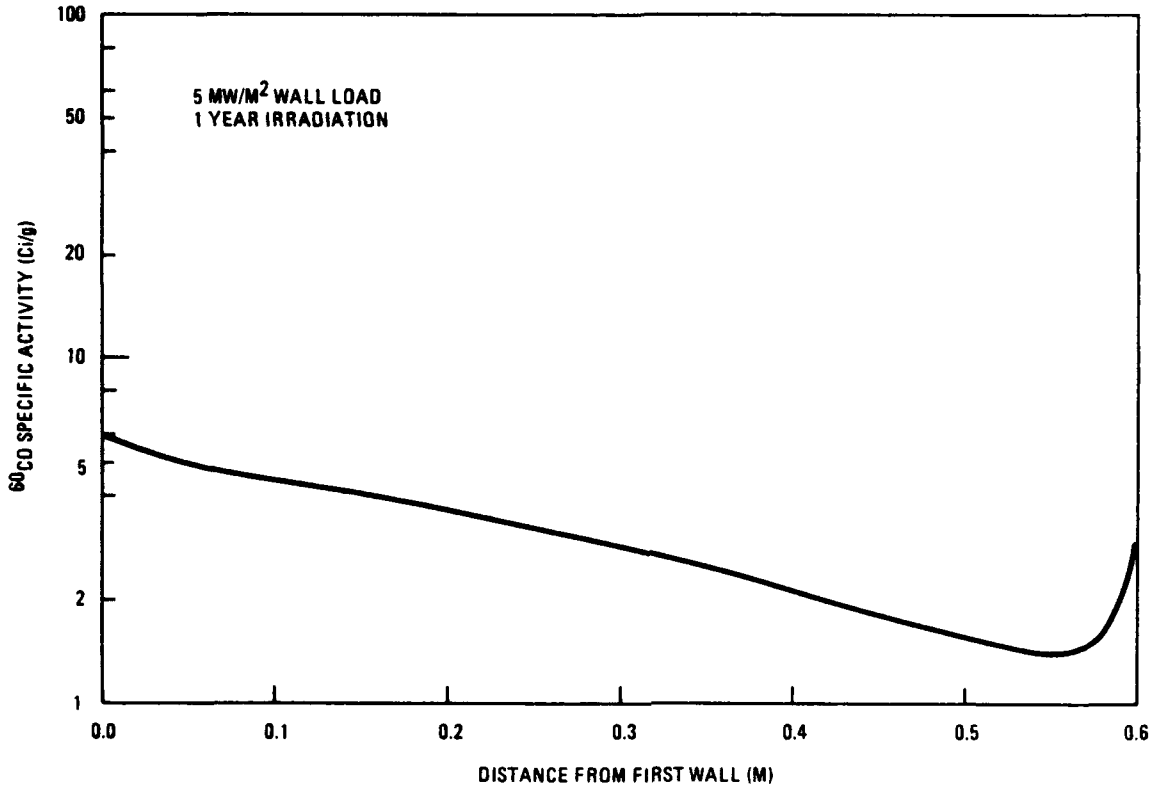


FIG. 6-8. Spatial distribution of ^{60}Co specific activity (Ci/g in a $Li_{17}Pb_{83}$ blanket) (one year continuous operation at 5 MW/m^2 neutron wall loading.)

Liquid Lithium and Li_2O Blankets. Liquid lithium and Li_2O are also promising breeding materials, and blanket designs employing these materials have been studied widely by the fusion community. The liquid lithium blanket was also considered by fusion-fission hybrid studies for breeding of fissile fuels by production of excess neutrons via the $^7Li(n,n'\alpha)t$ reaction.

The performance of a thick liquid lithium blanket can be slightly better than the $Li_{17}Pb_{83}$ blanket. The design modifications for a pure fusion liquid lithium blanket to become a ^{60}Co production reactor are very similar to those for a $Li_{17}Pb_{83}$ blanket. Hence,

we can expect slightly better ^{60}Co production quantity and specific activity in cobalt in a liquid lithium blanket than in a $Li_{17}Pb_{83}$ blanket, but total blanket thickness must be greater.

Lithium oxide (Li_2O) is the only feasible solid tritium breeder that does not require a neutron multiplier. However, the maximum tritium breeding ratio of a helium-cooled Li_2O blanket is only slightly above 1.2. Hence the excess neutrons available for ^{60}Co production in a fusion blanket employing Li_2O breeder is no more than 0.2, which is a factor of 1.5 and 4 lower than in $Li_{17}Pb_{83}$ and beryllium blankets, respectively. The specific activity in cobalt in a Li_2O blanket is also expected to be even lower than that in a $Li_{17}Pb_{83}$ blanket.

6.2.3.6. Cobalt-60 Production Performance Summary. From the results above, it is clear that fusion reactors can produce significant quantities of ^{60}Co. The large difference in ^{60}Co breeding performance of the various blankets, however, points out the importance of blanket design to optimize ^{60}Co production. The water-cooled TFCX first wall and shield are quite effective in producing ^{60}Co due to the high thermal flux. Addition of lithium to the blanket to breed tritium depresses the thermal flux and reduces the total amount and specific activity of ^{60}Co. Use of beryllium gives additional neutron multiplication and moderates the neutron flux, significantly increasing the total amount and the specific activity of the ^{60}Co. A near-term fusion experimental reactor would produce tens of MCi of ^{60}Co per year at ~10 to 40 Ci/g specific activity, assuming ~1 to 4 MW/m^2 wall load and 25% to 50% availability. A commercial fusion reactor with an optimized blanket could produce hundreds of MCi per year at up to 200 Ci/g.

6.3. OTHER RADIOISOTOPE PRODUCTION

Other radioisotopes could also be used for various radiation processing applications.

6.3.1. Manganese-56

Circulating ^{56}Mn is proposed as an irradiator for food preservation and sewage treatment in Section 5. Table 6-9 presents some basic data on this radioisotope.

TABLE 6-9
Manganese-56 Data

Thermal neutron cross section for production of ^{56}Mn from manganese, σ	13.3 barns
^{56}Mn half-life ($\tau_{\frac{1}{2}}$)	2.58 h
^{56}Mn decay constant (λ)	7.46×10^{-5} sec^{-1}
Photon emission per ^{56}Mn decay	0.85 MeV (100%) 1.81 MeV (28%) 2.11 MeV (14%)

The ^{60}Co production rate in the beryllium blanket of a large reactor is calculated to be 950 MCi/yr, or 30 Ci/sec, with ^{59}Co concentration of 10% by volume. Assume that a solution or slurry of manganese is circulated through the blanket in place of the cobalt. Possibly the hollow spheres of manganese could be suspended in the liquid lithium itself, and circulated to the irradiator and back. The neutron absorption rate in the manganese is about the same as the absorption in cobalt, if the manganese volume fraction is ~3%. However, the ^{56}Mn Curie production rate will be much higher in proportion to the decay constants – $\frac{7.46\times10^{-5}}{4.17\times10^{-9}} \times 30 = 5.4\times10^{5}$ Ci/sec. The ^{56}Mn activity will reach equilibrium after 2.58 hours of irradiation, however, at a level of 5000 MCi.

If the holdup fraction (the volume of the ^{56}Mn loop not in the blanket) is 95%, and the irradiator volume is 10%, then the MCi in the irradiator will be 25 MCi (this estimate takes into account that the effective duty factor of ^{56}Mn irradiation is $\frac{1}{20}$ or 5%). This quantity or irradiation is very high compared with most existing irradiators of ~1 MCi ^{60}Co, so the sterilized product throughput would be much greater. Food and/or sewage waste treatment could utilize large irradiator capacities.

6.3.2. Gadolinium-153

Gadolinium-153 is a recent addition to the growing list of valuable medical radioisotopes. It is the working ingredient in an advanced medical scanning machine that can detect loss of bone materials such as calcium. The radioisotope currently is produced by irradiating Eu_2O_3 in reactors at Oak Ridge and Hanford (FFTF). Production chains for ^{153}Gd were discussed by R. Schenter of HEDL at a U.S. DOE Nuclear Data Committee meeting in Germantown, Md., on May 1–2, 1986. Figure 6-9 shows the chains he presented. Gadolinium-153 is produced from $^{152}Gd(n,\gamma)$, which in turn comes from $^{151}Eu(n,\gamma)$ ^{151m}Eu decaying with a 9-hour half-life. Gadolinium-152 has a thermal absorption cross section of 10 barns, but a resonance absorption integral of 400 barns (peak of 740 barns at 4 eV).

Production rates in a thermal reactor (such as at ORNL) and a fast reactor (FFTF) are compared in Fig. 6-10. After 150 days in a thermal reactor, the ^{153}Gd produced begins to be burned up by thermal neutron absorption, whereas in the FFTF this effect is not observed.

In a fusion reactor blanket, the neutron spectrum could readily be tailored to provide maximum ^{153}Gd production. A one-year irradiation would be near optimal (^{153}Gd half-life is 242 days). Figure 6-9 implies a maximum specific ^{153}Gd production of ~20 Ci/g Eu_2O_3. This may be compared with a specific ^{60}Co activity of ~100 Ci/g in a fusion reactor blanket, or in the ATR at Idaho. However, if the ^{60}Co half-life were 242 days instead of 5.2 years, the specific activity would be roughly a factor of 5 higher (500 Ci/g). The difference between 500 Ci/g and 20 Ci/g must be attributed to the fact that ^{153}Cd production is a two-step process, and not every absorption in the parents leads to ^{153}Gd product; whereas, in the case of ^{60}Co, essentially all absorptions in natural cobalt lead to the desired product. Hence, if 950 MCi/yr of ^{60}Co can be produced in a fusion power reactor, it is likely that the equivalent ^{153}Gd production rate is 40 MCi/yr.

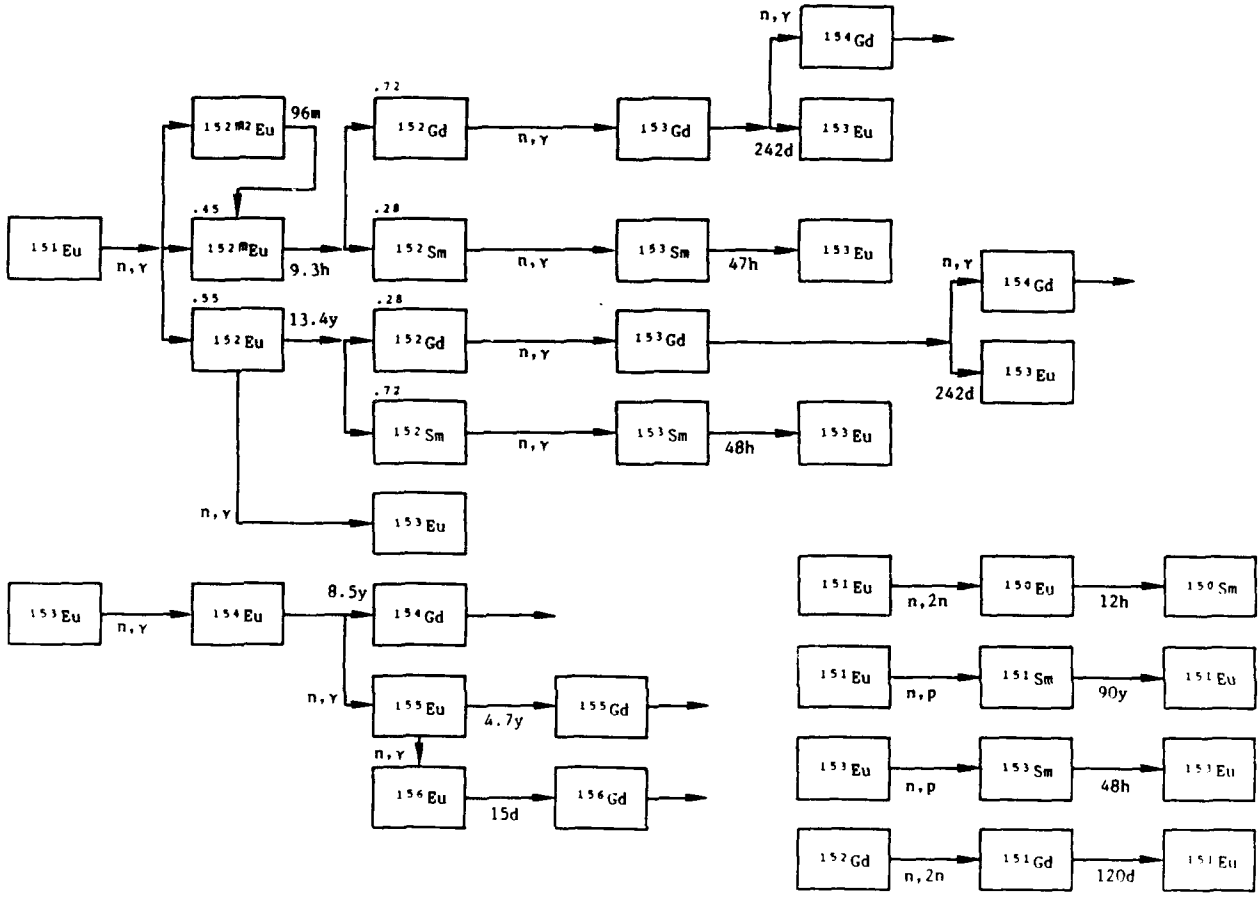


FIG. 6-9. Eu_2O_3 cha

6.3.3. Phosphorus-32 and Phosphorus-33

Phosphorus-32 and -33 are valuable pure beta emitters which migrate to the bone for the treatment of leukemia. Phosphorus-32 has a maximum β energy of 1.71 MeV and a half-life of 14.3 days, whereas ^{33}P has a β energy of only 0.25 MeV but a half-life of 25.3 days. In spite of its low beta energy, ^{33}P has an advantage of more extended irradiation once it reaches the bone.

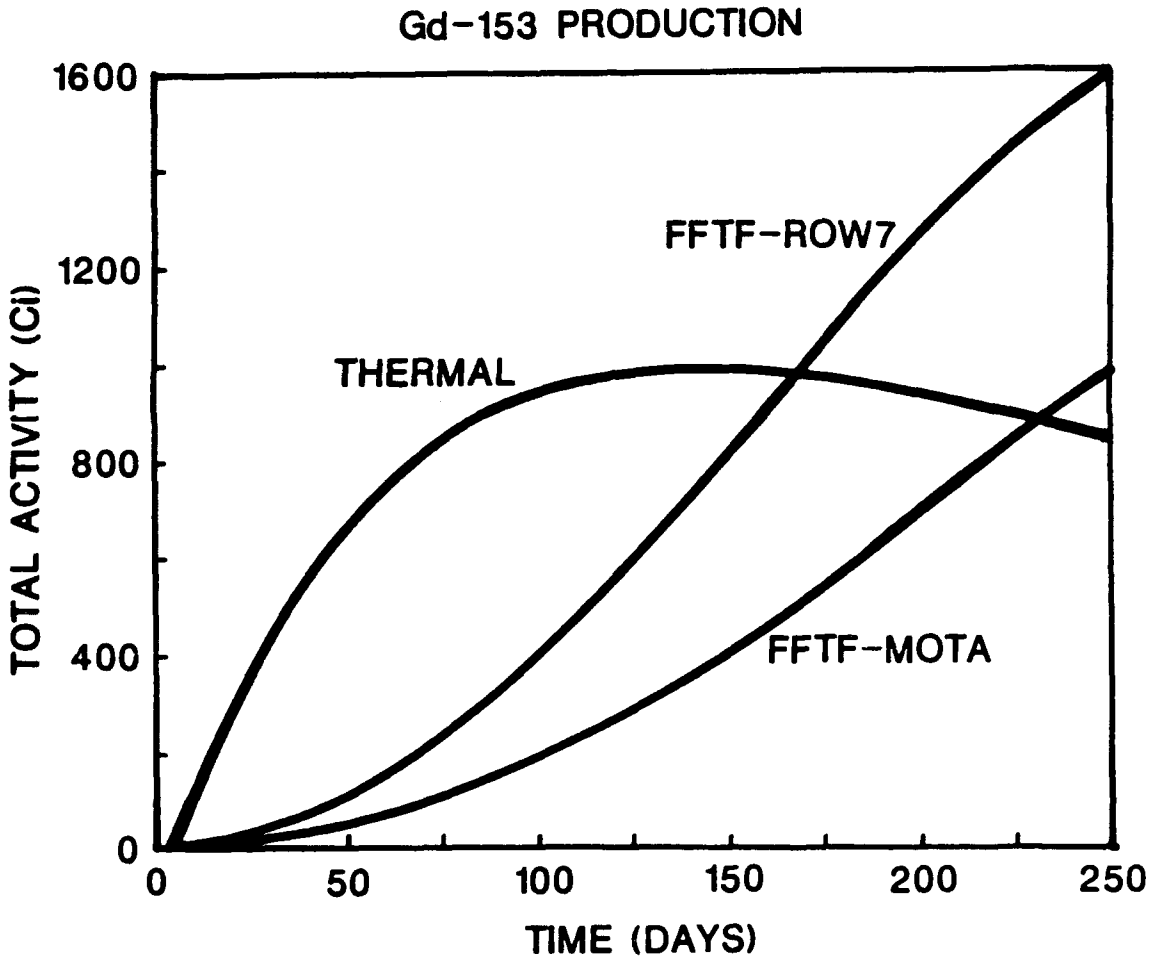


FIG. 6-10. Gadolinium-153 production.

Production of ^{32}P is via the (n,γ) reaction in natural phosphorus (cross section is 0.18 barns). Obviously, irradiation of the target should be brief; some kind of on-line feed system should be used from blanket to hot cell; and shipment of the ^{32}P to the user should be made as promptly as possible. The ^{32}P will be heavily diluted with natural phosphorus. We estimate $\sim$7 Ci of ^{32}P per gram of P (saturated activity), for the same blanket environment as in the earlier ^{60}Co calculations. (One gram of pure ^{32}P would amount to 4 megacuries.)

In a fusion reactor first wall or blanket, ^{32}P could alternatively be produced via the (n,α) reaction in ^{35}Cl. The cross section for this reaction is about 0.1 barn at 14 MeV (threshold is at 3 MeV).

A long exposure of the chlorine would tend to produce the long-lived isotope ^{36}Cl via the (n,γ) reaction in ^{35}Cl (45-barn cross section). The (n,α) reaction in the resulting ^{36}Cl would produce ^{33}P (unknown cross section).

The advantage of using chlorine rather than phosphorus as the target materials is that in the former case the ^{32}P (and ^{33}P) product could be chemically separated from the parent chlorine, resulting in much higher specific activity — perhaps thousands of Curies per gram.

6.3.4. Osmium

Osmium is used to produce very hard alloys for instrument pivots, electrical contacts, etc., and some of its compounds are used in biochemistry. Its April 1986 market price was \$700/oz. Total production and consumption are not known, but with the demise of the fountain pen (which utilized osmium for tips), consumption is probably less than in the 1950's.

Osmium could easily be produced from the relatively cheap element rhenium via two reactions:

$$\begin{array}{lll} {}^{185}Re & (n,\gamma) & {}^{186}Re \xrightarrow{\beta,\ 90\ h} {}^{186}Os \\ (37.4\%) & 170\text{b} & \\ {}^{187}Re & (n,\gamma) & {}^{188}Re \xrightarrow{\beta,\ 17\ h} {}^{188}Os \\ (62.6\%) & 120\text{b} & \end{array}$$

Production is enhanced by large absorption resonances between 1 and 100 eV. If an osmium production rate of 1.0 atoms per D-T neutron is achieved in a 1500-MWT fusion reactor, up to three tonnes per year could be created ($70 M worth at today's price level). Accompanying radionuclides are few, and of half-lives under one year. These would probably be allowed to decay off before chemical separation of the osmium from the rhenium.

6.3.5. Rhodium

Rhodium is a rare metal in the platinum family used as an alloying agent to harden platinum and palladium. Such alloys are used for furnace windings, thermocouple elements, bushings for glass fiber production, electrodes, and crucibles. Rhodium itself is used as an electrical contact material and for optical instruments. Its current price is $1200/oz, and only 2 to 3 tonnes are produced annually worldwide.

The best production route for this rare metal in a fusion blanket is probably

$$\underset{(31.6\%)}{^{102}Ru}\ (n,\gamma)\ ^{103}Ru \xrightarrow{\beta,\ 39\ da}\ ^{103}Rh \quad .$$

Ruthenium is relatively inexpensive, at $80/oz. However, the economic feasibility of producing rhodium from ruthenium in a fusion blanket is almost completely dependent on the relative magnitude of the (n,γ) cross sections of ^{96}Ru, ^{99}Ru, ^{100}Ru, ^{101}Ru, ^{102}Ru, and ^{104}Ru. As shown above, rhodium (^{103}Rh) is actually formed from the (n,γ) reaction in ^{102}Ru. If the macroscopic cross section for this reaction is too small relative to the total macroscopic capture cross section of the natural ruthenium, the parasitic neutron losses will render the process uneconomic.

The Ru cross sections are unavailable in the cross section libraries accompanying the MCNP Monte Carlo code. The only sources of information found so far are:

1. GE "Chart of the Nuclides," Twelfth Edition, revised 1977 by F. William Walker, Dr. George J. Kirouac and Francis M. Rourke, Knolls Atomic Power Laboratory, Schenectady, NY; *copyright 1977 General Electric Company.*

2. "Transmutation of Alloys in MFE Facilities as Calculated by REAC," HEDL-TME 81-37, August 1982

3. R. L. Macklin and J. Halperin, "$^{100, 101, 102, 104}$Ru(n,γ) and ^{103}Rh(n,γ) Cross Sections above 2.6 KeV," Nucl. Sci. & Engr., **73**, 174–185 (1980)

Selected data from the above are presented in Table 6-10.

TABLE 6-10
Ruthenium Cross Sections

Isotope	Abundance (%)	σ_{res} (barns) [from (1)]	σ (n,γ) (2.6 KeV) [from (2)]	σ (n,γ) (2.6 KeV) [from (3)]
^{96}Ru	5.5	5.5	0.5	n.a.
^{99}Ru	12.7	200.0	8.3	n.a.
^{100}Ru	12.6	11.0	0.5	0.85
^{101}Ru	17.0	80.0	8.3	2.8
^{102}Ru	31.6	4.5	0.5	0.75
^{104}Ru	18.7	4.6	0.5	0.7
$\sigma(^{102}\text{Ru})/\sigma(\text{Ru})$		3.2%	5.6%	*22%

* taking $\sigma(^{96}\text{Ru}) = \sigma(^{104}\text{Ru})$ and $\sigma(^{99}\text{Ru}) = \sigma(^{100}\text{Ru})$

Thus we see that based on the cross sections measured by Macklin and Halperin, the production of rhodium ($1200/oz) from ruthenium ($80/oz) might be attractive, but based on the other cross sections sets would be impractical. Clearly, there is a need for more and better values of the (n,γ) cross sections.

6.3.6. Gold

Gold could be made from mercury by the reactions:

$$\underset{(10\%)}{^{198}Hg}\ \underset{(2b)}{(n,2n)}\ ^{197}Hg \xrightarrow{\epsilon,\ 64\ hr}\ ^{197}Au$$

and

$$\underset{(0.13\%)}{^{196}Hg}\ \underset{(3100\text{b})}{(n,\gamma)}\ ^{197}Hg \xrightarrow{\epsilon,\ 64\ hr}\ ^{197}Au$$

Unfortunately, the potential output per reactor is only 200 kg per year, worth ~$2M. Separation of the gold from the mercury might be expensive, not to speak of the carrying charges on the mercury inventory.

References

[1] "Directory of Nuclear Reactors," Vol. VII, IAEA, Vienna, 1968.

[2] R.F. Morrisey, "Radiation Processing at Johnson & Johnson," Internal Report, Jan. 7, 1985.

[3] Isomedix background paper, Jan. 4, 1985.

[4] A.T. McMain, GA, personal communication, June 11, 1985.

[5] Mike Barrett, Duke Power Co., personal communication, June 20, 1985.

[6] B.A. Engholm, "Shutdown Dose Rate Studies for TFCX," 6th Fusion Topical, San Francisco, March 1985.

[7] D.H. Berwald, *et al.*, "Fission-Suppressed Hybrid Reactor – The Fusion Breeder," UCID-19638, December 1982.

[8] "Blanket Comparison and Selection Study – Final Report," ANS/FPP-84-1, Sept. 1984.

7. SYNFUELS

In developed countries, only about 30% of the total energy consumed is used to generate electricity. Seventy percent, in the form of oil, gas and coal is used for transportation and industrial needs. Continuing price and supply fluctuations, environmental problems, and the declining world inventory of these fuels encourage work on alternate, synthetic sources for this huge market.

7.1. BACKGROUND

Accordingly, several studies[1-4] have evaluated methods for converting fusion energy into storable, transportable chemical forms. This would allow utilization of the fusion energy for spatially or temporally remote purposes including load leveling (*e.g.*, fuel cells), as fuel substitutes (hydrogen or methanol) or synthetic hydrocarbons for stationary and mobile applications, and as chemical feedstocks (hydrogen or carbon monoxide) for commercial end-products (plastics, fertilizer).

Conventional generation of electricity uses working fluids at temperatures of 300 to 700°C, depending on proposed design. Temperatures at the upper end of this range, generated for example by employing helium gas cooling, would be useful in supplying energy to a large number of process heat applications. The heat could be transferred across a heat exchanger if necessary to provide for process isolation. However, our approach is to explore processes that utilize some aspect of the unique characteristics of the fusion energy source

The nature of the fusion reaction gives rise to unique processing techniques. Possibilities include utilizing the radiation or energetic particle losses from the plasma, the plasma exhaust itself (as, for example, in mirror reactors), or, in inertial confinement, the target debris. For D-T fusion, 80% of the fusion energy is carried off by high energy neutrons. These neutrons are highly penetrating and pass through the wall of the plasma vacuum chamber and are absorbed in a thick blanket. The neutrons may be employed directly as in radiolytic chemistry or their energy may be used indirectly in the form of heat, generating very high temperatures by extracting the neutron energy in a non-structural refractory ceramic which is thermally insulated from the structural components. Temperatures up to 1000°C should be achievable, which is beyond that possible from contemporary fission reactors.

The neutrons convey these potential advantages but also cause transmutations which produce unwanted radioactive products. The problem is unavoidable for processes which utilize the neutrons directly. For thermal applications, isolating heat exchangers may be employed; but they could severely limit achievable temperatures or heat transfer rates. The magnitude of the problem is specific to the chemical process under consideration and depends on the materials employed, the chemical processing steps, and the possibility for isolation of the activation products downstream. The D-T fuel cycle also requires tritium bred from neutron interactions with lithium. This function requires space in the reactor blanket and control of tritium migration downstream.

Direct process utilization of neutrons may seem simple, but there is a basic difficulty coupling the neutron energy into the reacting medium. Some energy will be deposited in the structure, but for solid or liquid media the majority of the neutron energy can be deposited in the medium. However, if a gas is being reacted, only a small fraction of the neutron energy is likely to be deposited in the gas even if it is at very high pressure. In any event, most of the neutron energy is deposited as thermal energy or heat. Several techniques, including radiolysis and thermal spike chemistry, have been proposed for direct utilization of the neutron energy, but they appear less promising.

7.2. RADIOLYSIS AND THERMAL SPIKE CHEMISTRY

Radiolysis is use of the neutron or secondary gamma ray energy to directly sever chemical bonds, breaking H_2O into H_2 and O or CO_2 into CO and O, for example. Thermal spike chemistry is the use of very energetic knock-on atoms to create microscopic regions of very high temperature where non-equilibrium chemical reactions can occur. In the case of radiolysis, the most energy efficient processes use less than 30% of the deposited energy. The reject energy must therefore be utilized in a co-process or for co-generation. To do so, it must be removed from the radiolysis medium at high temperature, which for systems of interest implies gaseous cooling media and also implies very low capture fractions for the neutron energy.

One of the more interesting radiolytic reactions is the decomposition of carbon dioxide to carbon monoxide.[1] The radiolytic decomposition could be one step of a closed two step thermochemical water splitting cycle.

$$(2CO_2 + \text{Energy} \longrightarrow 2CO + O_2,\ CO + H_2O \longrightarrow CO_2 + H_2)$$

If the reject energy is used for generating additional hydrogen by normal low temperature electrolysis, an upper limit on the estimated overall efficiency would be about 40%. The actual efficiency achieved in practice would probably be only slightly better than the 32% for a reactor plant devoted entirely to conventional electrolysis, which serves as a useful reference base. Radiolysis does not therefore appear to be particularly attractive at present although it might find a role as a topping cycle in special circumstances because of its conceptual simplicity. An additional problem is the production of radioactive carbon, ^{14}C.

Thermal spike chemistry is the interaction of high energy neutrons with materials, which results in transient localized hot spots having sufficient temperature to produce chemical dissociation, but which cool off so quickly that reverse reactions cannot occur. The idea is to yield useful products more valuable than the original chemicals. While this is a novel and unique applications of fusion, calculations show that less than 5% of the neutron energy captured by the reacting medium is funneled into the thermal chemical

reactions. Unless more favorable neutronic interactions are found, the yield would be too small.

7.3. HEAT CYCLES

Several processes that show potential would use the thermal energy carried by the neutrons, with temperatures up to perhaps 2500°C. Among these are water splitting cycles, either pure thermochemical or thermochemical/electrochemical hybrid processes, all of which produce hydrogen. A number of cycles have been invented, but only relatively few can take advantage of the very high temperatures fusion can provide. These are generally faced with a combination of problems relating to materials handling and compatability, and product separation. There are also apparent mismatches between the temperature levels at which heat is likely to be generated in the reactor blankets and those required for the processes.

One interesting cycle employs sulfur and iodine. It has been under development in different variations by three laboratories. Nominally temperatures are somewhat below 900°C, but could benefit if higher temperatures were available. Preliminary estimates suggest increasing temperatures to 1250 to 1400°C could lead to a 10% cost reduction for the product hydrogen, primarily through reduced heat exchanger area. Published engineering analyses[1] suggest the cycle is already about 45% energy efficient so that optimizing capital costs might lead to an attractive cycle.

The high temperature reaction in this cycle is the decomposition of sulfur trioxide. The reverse reaction is exothermic and this chemical system has been proposed as one candidate for a chemical heat pipe. The technology required would be essentially the same as for the water splitting cycle. Instead of shipping the chemicals by pipeline, a useful modification would be to utilize the chemically stored energy (in the decomposed sulfur trioxide) on site for peak shaving, assuming a co-generating system.

7.4. FERTILIZERS

Another attractive thermal application explored[1] is the fixation of nitrogen by the very high temperature reaction of air, ($N_2 + 2O_2 \longrightarrow 2NO_2$). Temperatures of the order of 2100°C are required and the product must be rapidly quenched to prevent back reaction. Calculations of heat and mass transport in a pebble bed and of the chemical kinetics have been carried out. Although the derived overall process efficiency is only about 12–15% this process might be competitive with alternate fertilizer manufacturing processes. The market is small by comparison with the potential market predicted for hydrogen, but is still substantial.

7.5. HYDROGEN

Among different methods evaluated for converting fusion energy into storable, transportable chemical forms, the generation of hydrogen appears to hold the best promise. Being one of the basic building blocks of all hydrocarbon fuels, hydrogen is essential for synthetic fuel production, and has many applications for both energy and chemical feedstock uses.

7.5.1. Applications

Hydrogen can be used directly as a fluid fuel or indirectly as a chemical for production of other fuels. The hydrogen fluid fuel cycle is environmentally very attractive. Hydrogen can be produced from water, so there is no mining involved, and when it is burned, the only ash is water, and, possibly, a small amount of nitrogen oxides. Several experimental hydrogen-fueled ground and air transport vehicles have been explored, with positive results. Hydrogen is a feedstock for other synfuels, such as gasoline, methane, and methanol, and for produce other useful chemicals such as ammonia.

7.5.2. Production Methods

Hydrogen production has been investigated as a product from fusion reactors in three major studies[2-4] performed by Brookhaven National Laboratory (BNL),[2] GA Technologies Inc. (GA)[3] and Lawrence Livermore National Laboratory (LLNL).[4] The following is a summary of their results.

7.5.2.1. BNL Study. The BNL study was a three-year program to examine the commercial and technical feasibility of utilizing fusion power to generate hydrogen synthetic fuel. A tokamak fusion reactor of the STARFIRE design was proposed to generate high-temperature steam (~1700 K) for electrolysis to hydrogen and oxygen gas in a high-temperature electrolysis (HTE) unit. This combination of STARFIRE with an HTE hydrogen production unit is called HYFIRE. The direction of this study was driven by the optimization of high hydrogen thermodynamic conversion efficiencies.

The HYFIRE blankets were designed to provide high-temperature steam for the electrolysis process, to produce thermal energy for efficient generation of electricity to operate the plant, and to breed sufficient tritium to compensate for burnup and process losses. Two types of blanket modules were designed to meet these requirements, a steam-cooled "HTE" module with a tritium breeding zone (Fig. 7-1) and a He-cooled tritium breeding "power" module (Fig. 7-2). In each case, the first wall and blanket structural material is PCA (Prime Candidate Alloy) stainless steel as in STARFIRE; however, in HYFIRE, only the steel shell is cooled by pressurized water, the interior is cooled by either steam or helium. The modules are arranged toroidally since this minimizes differences in overall blanket configuration and associated maintenance procedures between STARFIRE and HYFIRE.

The interior of the HTE steam modules, shown in Fig. 7-1, consists of rods of ZrO_2 which are thermally insulated from the steel shell. The module utilizes a relatively thin tritium breeding layer outboard of the steam-cooled HTE zone. Consistent with the STARFIRE design, the tritium breeding medium is $LiAlO_2$. The design employs at least

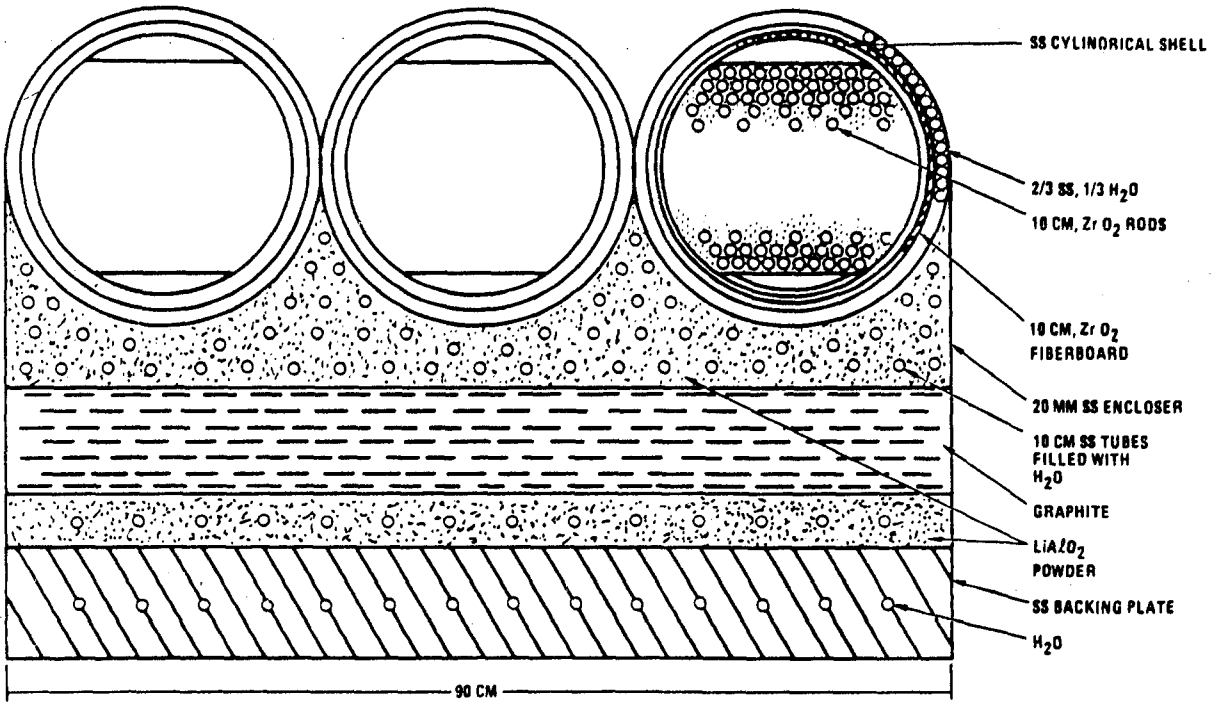


FIG. 7-1. Steam-cooled blanket module.

two structural steel boundaries to minimize the potential of tritium leakage into the HTE steam circuit.

The tritium breeding/power production module, shown in Fig. 7-2, also employs a low-temperature water-cooled shell. The interior region is He-cooled and contains two distinct zones. The inboard zone contains a beryllium multiplier as well as $LiAlO_2$; the outboard zone consists of SiC and $LiAlO_2$. Since interior structural materials are minimized, the blanket may operate at relatively high temperatures, which promotes tritium removal as well as the efficient power conversion. The primary He stream (at ~20 atm) exchanges heat with a secondary He power conversion stream (at ~70 atm) in small heat exchangers outboard of the blanket.

This design, utilizing HTE, has the highest potential efficiency for production of synfuels from fusion; a fusion-to-hydrogen energy efficiency of ~70% is claimed possible

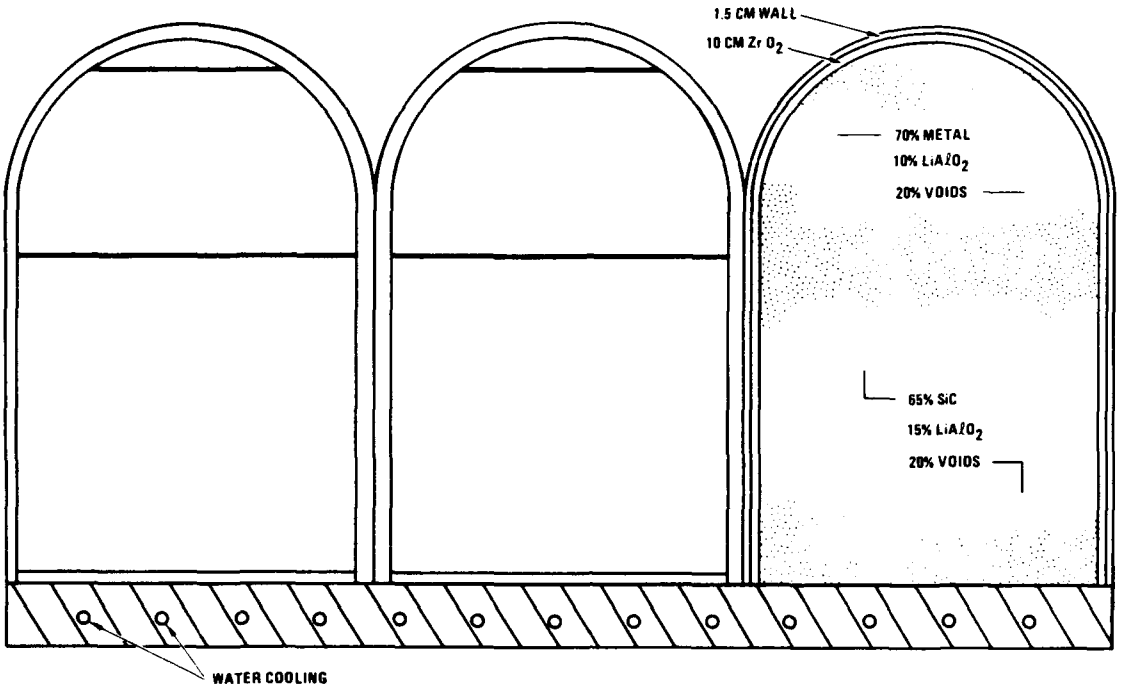


FIG. 7-2. Helium-cooled blanket module.

with 1800°C HTE units and 60% power cycle efficiency. An efficiency of ~50% appears possible with 1400°C HTE units and 40% power cycle efficiency.

7.5.2.2. GA Study. Both the GA and the LLNL studies have based the production of hydrogen on the GA Sulfur-Iodine Thermochemical Water-Splitting cycle.[5] The chemical reagents, sulfur and iodine, are continuously recycled and reused with essentially no loss of materials. The reference process requires 22% of the energy at high temperature (1144 K) and 78% of the energy at a lower temperature. It is a pure (without electrolysis) thermochemical cycle and can be described by the following major reaction steps:

$$2H_2O + SO_2 + xI_2 \rightarrow H_2SO_4 + 2HI_x \qquad (270 - 290\ K) \qquad (1)$$

$$2HI_x \rightarrow H_2 + xI_2 \qquad (393\ K) \qquad (2)$$

$$H_2SO_4 \rightarrow H_2O + SO_2 + \tfrac{1}{2}O_2 \qquad (1144\ K) \qquad (3)$$

Major parts of the process are associated with separation and purification of the reaction products. For example, a critical aspect for the successful operation of the process is in the separation of the aqueous reaction products in reaction (1). GA solved this problem by using an excess of I_2, which leads to separation of the products into a lower density phase, containing H_2SO_4 and H_2O, and a higher density phase containing HI, I_2, and H_2O.

Reaction (2) shows the catalytic decomposition of HI, which is in the purified liquid form (50 atm). Laboratory decompositions are around 30% per pass; therefore, a recycle step is necessary. The unreacted HI is condensed out of the H_2 and I_2 products. Pure H_2 is obtained by scrubbing out I_2 with H_2O.

The equilibrium for reaction (3) lies to the right at temperatures above 1000 K, but catalysts or higher temperatures are needed to attain sufficiently rapid decomposition rates. This is where the high temperature fusion reactor blanket can be utilized.

In the GA study, a conceptual fusion synfuel production system was developed. The blanket design developed in this study is illustrated in Fig. 7-3. This design can be applied to both tokamak and mirror reactors.

This system incorporates a two-zone blanket which can achieve a tritium breeding ratio of 1.1 while delivering a high fraction (30%) of the fusion heat at high temperatures (1250°C). The tritium breeding material is $Li_{17}Pb_{83}$ which is contained in Nb-18 alloy tubes. The high temperature region consists of SiC tiles. The multiple barriers to tritium permeation in the blanket design permit the hydrogen product to meet 10CFR20 regulatory requirements without stringent requirements on the tritium recovery systems. A ceramic heat exchanger, incorporating SiC tubes and headers to contain the process stream and a cooled, Inconel 718 pressure shell to contain the helium, was designed for transferring the heat from the high-temperature coolant to the process. A good heat-line match of the blanket heat-source temperature distribution to the requirements of the thermochemical plant was attained under the dual goal of maximizing process efficiency and minimizing

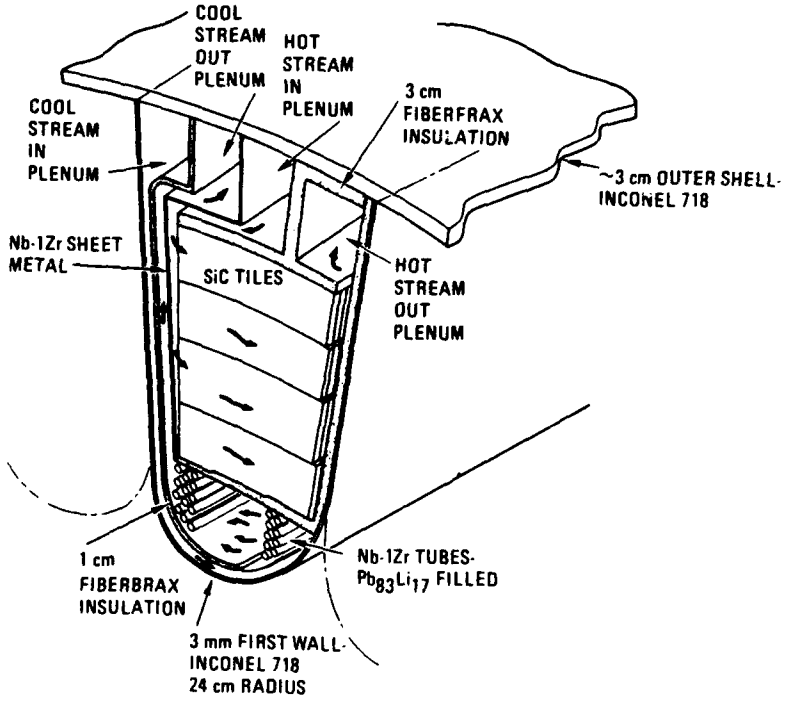


FIG. 7-3. High-temperature blanket lobe using two coolant streams.

the hydrogen cost. The results are a process efficiency of 43%, and an estimated cost of hydrogen of \$12 to \$14 per GJ of hydrogen. By way of comparison, gasoline at \$1.00/gal is \$8/GJ.

7.5.2.3. LLNL Study. In the LLNL study, the fusion driver was the Tandem Mirror Reactor (TMR). Two blanket designs were considered. The first was a modest temperature blanket operating at 825 K to 950 K exit gas temperature. This design relies on electrical heating (a process called *Joule Boosting*) to do the high temperature SO_3 decomposition step in the chemical process. The second design was for a high temperature blanket

operating at 1150–1200 K maximum exit gas temperature wherein the decomposition step is done thermally.

The first design (see Fig. 7-4), which operates at low to medium temperatures, uses a combination of solid lithium oxide as moderator and helium as coolant.

To assure that the hydrogen product is tritium-free, "in-situ" tritium control is used rather than slip stream processing. This isolates the tritium from the main helium flow. Recovery of the tritium is by an independent purge circuit. Tenelon stainless steel was selected as the structural material.

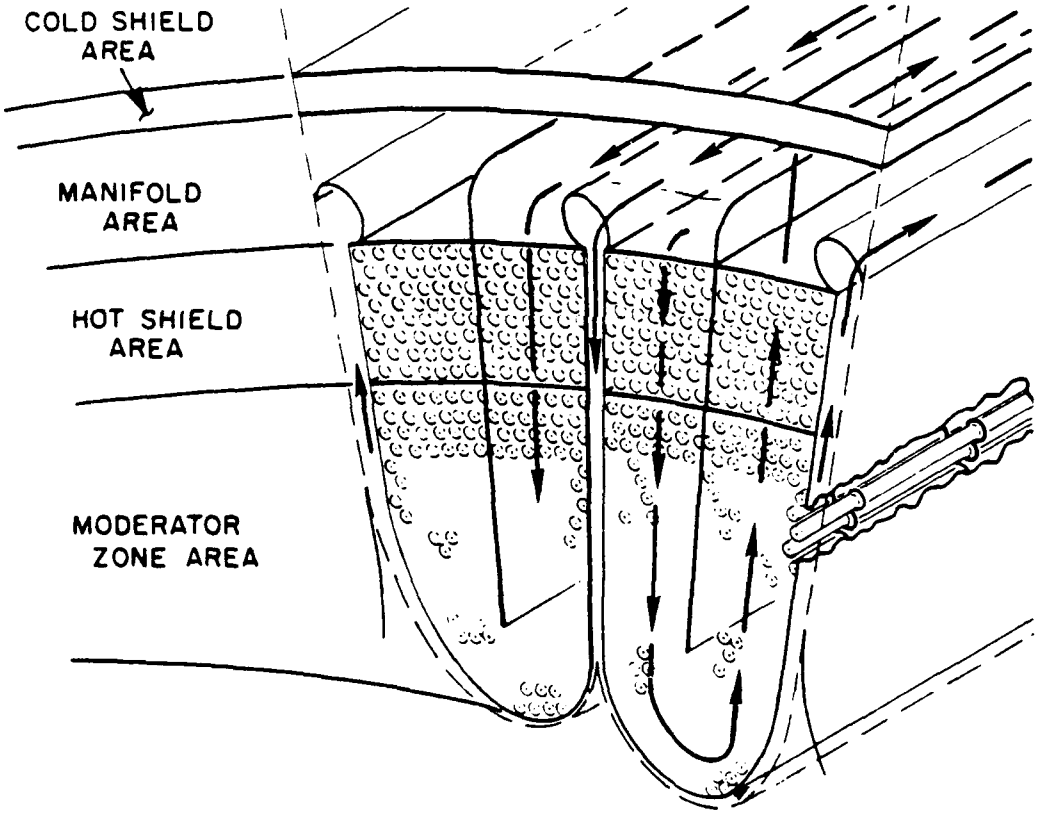


FIG. 7-4. The multiple-pass heat exchanger canister model.

The basic blanket geometry uses a two-dimensional lobe or canister configuration as shown in Fig. 7-4. A subassembly of the canister is one of the long tubes (~2 m) containing the lithium oxide. The tube protects the oxide from the mainstream coolant so that it does not disintegrate due to the high velocity helium flow through the tube bank or due to trace contaminants.

The second design, which operates at high temperature, does not breed tritium but instead relies upon the medium temperature design to provide the tritium breeding. The high temperature energy partition is accomplished by axially zoning the TMR into high and medium temperature zones. The required fraction of axial length for high temperature thermal energy involves a tradeoff between the fraction of high temperature energy supplied to the thermochemical process and the overall tritium breeding ratio. The high temperature medium can be SiC.

Tritium breeding was excluded from the high temperature blanket to allow direct coupling with the SO_3 decomposer (1100 K) without concern for tritium contamination of the thermochemical process. The efficiency of the process using the modest temperature blanket is about 38%, and the efficiency of the process using the high-temperature blanket is approximately 43%. The estimated costs for producing the hydrogen are $12 to $15 per GJ. This should be compared to hydrogen production from natural gas, petroleum or coal, which have estimated costs between 1990 and 2020 in the range of $8–$11 per GJ.[6]

7.5.3. Overall Economics

Gregory and co-workers[7] at the Institute of Gas Technology addressed the question of augmented capacity for hydrogen production by considering three scenarios. They examined the electrical requirement for supplying enough energy to (1) replace the actual 1968 natural gas consumption; (2) replace the projected natural gas consumption of the year 2000, or (3) replace all fossil fuels, other than those used for electricity generation by the year 2000. Based on an advanced electrolysis process, the electricity consumption is 115 kWh/1000 ft^3 of H_2, and the electrical power requirements are 800 GW, 1800 GW,

and 3900 GW for the above scenarios of producing hydrogen at the rate of 60×10^6 ft^3/yr, 135×10^6 ft^3/yr, and 300×10^6 ft^3/yr, respectively. We can see that, in order to support a hydrogen economy, 1 to 5.6 times the present 1986 total U.S. electricity capacity must be added for hydrogen production, beyond the capacity needed to cover other increases in electrical demand.

For the fusion reactor case, reference (3) indicated that the energy consumption for the conversion is 60 kWh/1000 ft^3 of H_2. Based on the three scenarios discussed above, the required number of 1000 MWe equivalent fusion reactors would be 400, 1000 and 2000, respectively. This shows clearly the market potential for fusion reactors in fulfilling the need for production of hydrogen.

References

[1] L. Rovner, *et al.,* "Study of Chemical Production Utilizing Fusion Neutrons," GA Technologies Report, GA-A15371, UC20d, April 1979.

[2] J.A. Fillo and J.R. Powell, eds., "Proceedings of Brookhaven National Laboratory's Fusion Synfuel Workshop," BNL-51188, August 1979.

[3] I. Maya, *et al.,* "Utility Fusion Synfuel Study, Phase III of a Study of Chemical Production Utilizing Fusion Energy," GA Technologies Report, GA-A16813, July 1983.

[4] R.W. Werner, ed., "Synfuels from Fusion Using the Tandem Mirror Reactor and a Thermochemical Cycle to Produce Hydrogen," Lawrence Livermore Laboratory, UCID-19609, November 1, 1982.

[5] G.E. Besenbruch, K.H. McCorkle, J.H. Norman, D.R. O'Keefe, J.R. Schuster, and M. Yoshimoto, "Hydrogen Production by the GA Sulfur-Iodine Process. A Progress Report." in **Hydrogen Energy Progress, Proceedings of the 3rd World Hydrogen Energy Conference,** Tokyo, Japan, 23–26 June, 1980, Vol. 1, edited by T.N. Veziroglu, K. Fueki, and T. Onta, Pergamon Press, Oxford, 1980, p. 243.

[6] K.R. Edman, "Cost of Hydrogen Production from Fossil and Nuclear Fuels," NASA report 5030-461, July 1980.

[7] D.P. Gregory, D.Y.C. Ng and G.M. Long, "The Hydrogen Economy," in **Electrochemistry of Cleaner Environments,** J.O. Bocknis, ed., Plenum Press, New York, 1972, Ch. 8.

8. PROCESS HEAT

More than 20% of energy consumed is used for process heat. This ranges from space heating at 25° C to materials processing at up to 3000° C. Fission reactors are beginning to be used to provide some of this process heat. Fusion could potentially provide such heat as well.

8.1. INTRODUCTION

The thermal power from fusion reactor blankets can be converted into electricity, or sent directly to an industrial process, or a combination of each. In general, because process heat is less transportable than electricity and because few sites require more than a few hundred megawatts of process heat, one would expect to employ a combined reactor producing mainly electricity with a moderate amount of process heat.

Many energy-intensive industrial processes require both steam and electric power.[1] The quality of heat produced from a fusion reactor can be very high, making it suitable, for example, for thermochemical water splitting. When combined with a sensible energy heat transfer salt (such as a sodium nitrate/potassium nitrate mix), a constant heat source can be supplied to meet cyclic loads, and energy can be transferred over distances beyond the range of steam transmission. If the rear portion of the reactor blanket, away from lithium-bearing material, is devoted to process heat, ceramics can be used and temperatures as high as 1250°C are possible with helium coolant. Much of the work on HTGR process heat plants is adaptable to helium-cooled fusion plants.

8.2. ECONOMICS

For purposes of rough analysis, let us consider a generic fusion reactor producing 3000 MW(th) of blanket power. This power plant is assumed to have a total cost of $3 billion. Assuming its net conversion efficiency, after allowance for recirculating power, is 1/3, then 1000 MW(e) could be produced if none of the thermal power were diverted to process heat. Assuming a fixed charge rate of 10 percent, consistent with current lower interest rates, and a plant factor of 0.75, the cost of electricity for a pure electricity producer is

$$\begin{aligned} \text{COE} &= 1000 \times \text{R}/[8766 \times 0.75 \times \text{kW(e)}] \\ &= 45 \text{ mills/kW(e)-hr,} \end{aligned}$$

where R = 0.10 × Total Cost = required annual revenue to meet payments on the construction loan. Note that we have omitted operation and maintenance for simplicity. This introduces little error because costs are capital-intensive in fusion plants.

Suppose now that all of the blanket power is used as process heat instead of electricity. Then, ignoring for the moment the impact this would have on capital costs, the cost of heat (COH) must be

$$\begin{aligned} \text{COH} &= 1000 \times R/[8766 \times 0.75 \times \text{kW(th)}] \\ &= 15 \text{ mills/kW(th)-hr} \\ &= \$4.50/\text{GJ} \\ &= \$4.50/\text{million Btu} \end{aligned}$$

This can be compared with home heating oil, which, at current bulk prices of around $0.50/gallon, costs $3.30/million Btu.

If the price that can be obtained for the electricity exceeds the required COE above, then the price of the process heat can be reduced. How much depends on the surcharge and on the mix of the two. Let M be the ratio of the price of electricity sold to the COE above, and let f be the fraction of electricity sold relative to the maximum of 1000 MW(e). Then, to meet the annual revenue requirement, we must have

$$R = M \cdot \frac{\text{COE}}{1000} \cdot 8766 \cdot \eta \cdot \text{kW(th)} \cdot f \cdot 0.75$$
$$+ \frac{\text{COH}}{1000} \cdot 8766 \cdot \text{kW(th)} \cdot (1 - f) \cdot 0.75$$

or,

$$\text{COH} = \text{COE}\,\eta\,(1 - M\ f)/(1 - f)$$

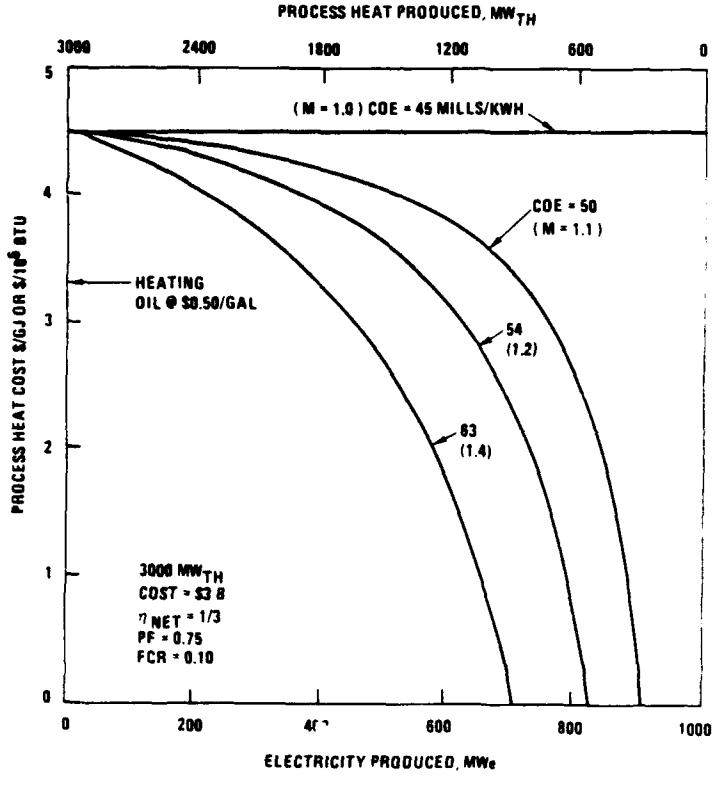


FIG. 8-1. Cost of process heat when subsidized by electricity revenues.

This is plotted in Fig. 8-1 for the 3000 MW(th) example above. Note that overcharging for the electricity can subsidize the process heat to the point where, in some cases, it can be given away. More realistically, process heat costs from fusion reactors can be made competitive with heating oil at current prices with only a modest surcharge on the electricity produced. For example, with a 10% surcharge (M = 1.1), one could produce 450 MW(th) of process heat at less than $2.00/GJ along with 825 MW(e) of electricity at 50 mills/kwh. Few sites have requirements for more than this amount of process heat. The electricity, of course, can be transmitted long distances beyond the site.

In summary, fusion reactors can be used as dual product plants in which both process heat and electricity can be sold at competitive prices. The cost of process heat without electricity production may also be competitive considering the high quality and low environmental impact of the heat compared to fossil burning, and allowing for the fact that capital costs will be somewhat less when electrical generating equipment is eliminated.

References

[1] "Process Heat Reactor Design and Analysis," GA-A15137, Feb. 1979.

9. DEFENSE APPLICATIONS

Defense applications of fusion energy include radiation effects testing, weapons physics experiments using ICF reactors, and tritium production. The first application is discussed in this section. Tritium production was covered in Section 3. The primary mission of the ICF program is development of tools for testing of weapons physics. However, a proper discussion of weapons physics would require introducing classified material. This section is devoted to radiation effects testing.

9.1. RADIATION EFFECTS TESTING

Tokamak or inertial confinement fusion reactors show potential as test reactors for nuclear radiation damage evaluation and hardening programs. Radiation effects testing for satellite, missile, aircraft, and SDI programs (particularly for solid-state control and guidance equipment, optics, etc.) is currently being carried out at the following types of facilities:

(1) Electron-beam (X-ray) facilities such as Aurora, Blackjack, Pithon, Casino, Gamble, Owl, Pulserad, Pocobeam, and MBS. These machines typically emit intense pulses of bremsstrahlung and/or soft X-rays to simulate EMP fields from high-altitude nuclear detonations.

(2) Cobalt-60 irradiators at many locations such as JPL. Used for longer-term cumulative gamma exposure of components, particularly solid-state devices, integrated circuits, and entire subsystems.

(3) Accelerator neutron sources, such as the Rotating Target Neutron Source at Livermore, can provide 14-MeV neutron irradiation of solid-state materials, lubricants, optics, etc., to provide some indication of effects from neutrons emitted from a hydrogen bomb.

(4) Fission reactors are widely used as irradiators, for instance, the TRIGA reactors, the ORNL test reactors, pulsed fast reactors, etc., to determine combined effects of neutrons and gammas on many materials and components. Exposures are usually extended rather than pulsed.

Possible advantages of utilizing fusion machines for the neutron irradiation tasks are:

Tokamak: Much larger test volume

More faithful spectral simulation of thermonuclear test environment.

Integrated exposure (dose)

ICF: Very short pulse, characteristic of thermonuclear weapon.

Dose rate effects can be measured.

Design modifications necessary for a tokamak used as a radiation effects facility include

- Vacuum vessel must be designed for unimpeded fusion-neutron illumination of test modules
- Test modules should be as free as possible of uncontrolled neutron scattering from nearby tokamak components
  - There must be no port nozzle in front of a test module region
  - Depth of the test module region in the bore of the TF magnets should be at least 0.8 m.
- Several adjacent bays should be dedicated to nuclear testing, to facilitate installation of large systems when required

- At least one bay should be suitable for extraction of a fast-neutron beam of 0.3 to 1.0 m or more in diameter for irradiation of large external systems

Exact simulation of gamma-ray fields is of secondary importance in a tokamak radiation effects facility, because many other facilities such as LINACS or ^{60}Co sources offer gamma fields of much higher intensity. However, a substantial gamma field will exist in the tokamak environment, and there may be a requirement to suppress it in order that neutron damage effects can be more completely isolated. The intensity and spectrum of soft and hard X-rays emitted directly from the tokamak plasma can be controlled by varying the type and amount of impurity ions in the plasma.

Figure 9-1 shows the neutron dose rates and integrated doses needed for various radiation effects tests. As shown in Table 9-1, the ranges of interest would be covered by an engineering or demo device such as FED-R. In this facility, operating at 25% duty factor, the weekly neutron fluence at the first wall would be 1 to 3×10^{18} n/cm^2 (uncollided) and the integrated weekly dose would be 1 to 3×10^{10} rads deposited silicon.

TABLE 9-1
Typical Radiation Levels at Front Face of Test Module

	FED-R (1990–95)
Fusion Neutron Current (n/cm^2/s)	$6\times10^{12} - 2\times10^{13}$
Fast Neutron Flux (n/cm^2/s @) $E \geq 0.5$ MeV)	$2\times10^{13} - 6\times10^{13}$
Total Neutron Dose Rate, rad (Si)/s (all energies)	$6\times10^{4} - 2\times10^{5}$
Gamma Dose Rate, rad (Si)/s (mean energy $\sim$ 1 MeV)	$2\times10^{4} - 6\times10^{4}$
Soft X-Ray Flux (W/cm^2)	1 to 5
Pulse Length (s)	$\geq$ 100
Integrated Neutron Dose	Weekly Dose = 1 to 3×10^{10} rad (Si) a 25% Duty Factor

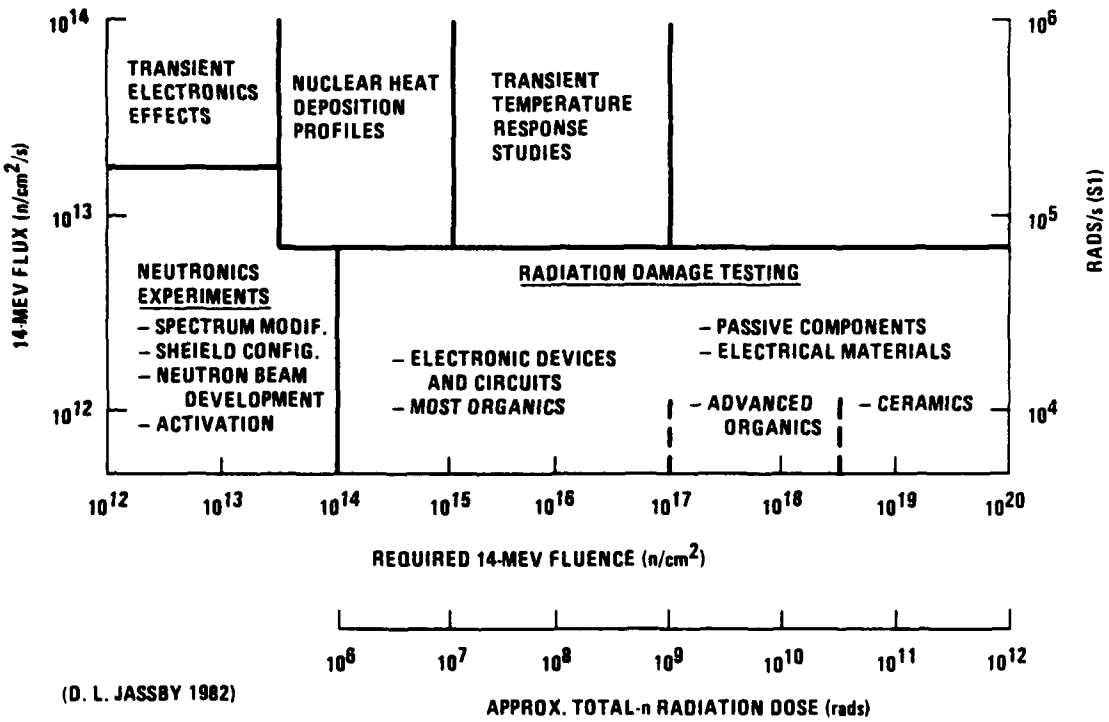


FIG. 9-1. Fusion-Neutron radiation effects.

The relatively low neutron flux but long pulse length (seconds to steady state) and consequently high integrated dose imply that tokamak test reactors are best suited for determining the permanent damage of neutron (and gamma) radiation to dose-dependent devices and subsystems. Nevertheless, transient effects on devices that are damaged by fast-neutron fluxes as low as 2×10^{13} n/cm^2/s or by gamma-ray doses as low as 10^5 rads/s can also be evaluated.

The advantage of a tokamak fusion neutron generator (projected 10^{19}–10^{20} n/s @ 14 MeV) over the most intense accelerator-driven point-neutron sources (projected 10^{14}–10^{15} n/s from a solid or gas target) are the following:

(1) The tokamak has orders of magnitude larger test volume with uniform irradiation flux of $\gtrsim 10^{13}$ n/cm^2/s @ 14 MeV. Thus large subsystems can be irradiated uniformly.

(2) Much more rapid and thorough testing even for small samples is possible in a tokamak facility because a large number of test samples can be irradiated simultaneously to high fluences.

(3) Variation of the ambient neutron spectrum in a point-source facility is difficult without catastrophic loss in neutron density.

The advantages of a tokamak facility over a pulsed fast reactor are

(1) The ambient neutron spectrum in a tokamak originates from a 14-MeV D-T source and can be made characteristic of that of a thermonuclear threat, while damage effects caused by a pulsed fast reactor spectrum must in general be extrapolated.

(2) The number of pulses per day in a pulsed fast reactor is limited by the thermal excursion experienced in each pulse. A tokamak simulation facility can generate a daily integrated neutron dose at least 100 times larger.

References

[1] D.L. Jassby, "Tokamak Fusion Generators for Nuclear Radiation Effects Testing," **IEEE Trans. on Nucl. Sci.**, NS-29, 6, (Dec. 1982)

[2] "FED-R, Fusion Engineering Device Utilizing Fusion Engineering Device Utilizing Resistive Magnets," FEDC report (ORNL, Sept. 1982)

10. FISSION WASTE BURNING

One of the earliest hopes for fusion reactors was that they could be used to "burn" fission wastes by neutron transmutation to more benign isotopes. Study by several authors has shown that, in general, it is not practical to do this. The reasons are discussed in this section.

10.1. INTRODUCTION

The use of fusion reactors to transmute fission reactor wastes to stable species is an attractive concept.

10.1.1. The Nature of Fission Wastes

The fission process produces two or three neutrons and a pair of lighter nuclei, called fission products. For each fissile isotope a spectrum of fission products with a well-defined distribution, rather than a unique pair, is produced. Many fission products are radioactive. Most of these have relatively short half-lives and decay to innocuous levels within a few years after removal from the reactor. Two isotopes, ^{90}Sr and ^{137}Cs, have approximately 30-year half-lives and take roughly 500 years to decay to low levels. Their time-dependent hazards are shown in Fig. 10-1.[1] The hazard index used is the number of cubic meters of water that would be required to dilute the waste to the maximum permissible concentration (MPC) allowed for ingestion (drinking) by the general public, and the basis used in this chapter is a GW(e)-y of fission electrical energy produced. Although two to four times more ^{137}Cs is produced than ^{90}Sr, depending on how much Pu is in the fuel, ^{90}Sr is about 100 times more hazardous biologically. There are also a few very long-lived fission products. Of most concern are ^{129}I (1.7×10^7 y half-life) and ^{99}Tc (2.1×10^5 y half-life), whose hazards are also plotted in Fig. 10-1.

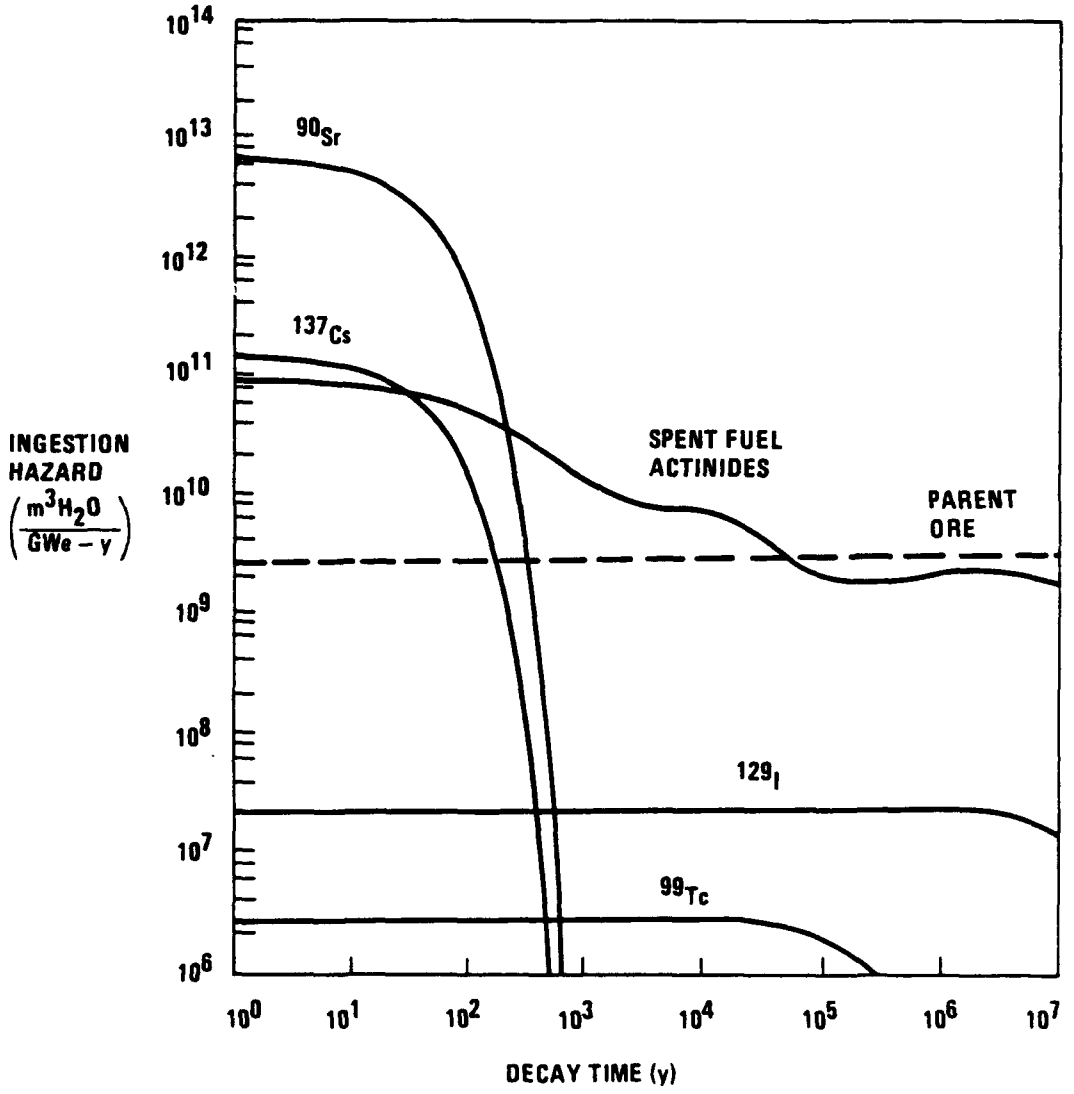


FIG. 10-1. Radioactive waste hazard – LWR once-through fuel cycle.

Actinide series isotopes form another class of fission reactor waste. They are produced mainly by absorption of a neutron by the nucleus of an actinide isotope already present, yielding a heavier nucleus. Initially, ^{238}U and ^{235}U are the only actinides present, but neutron absorption yields ^{239}Np and ^{239}Pu. These in turn can absorb a neutron to produce

still heavier nuclei, and so forth. Alpha decay, fission and (n,2n) reactions reduce nuclear masses. The net result after some time is a distribution of actinide isotopes. All actinide isotopes without exception are radioactive. The actinides formed in a fission reactor decay through about a dozen isotopes along one of four decay chains before terminating in a stable lead or bismuth. All the chains contain both short- and long-lived isotopes. The hazard from the actinides discharged from a typical light water reactor (LWR) is plotted in Fig. 10-1. The hazard is principally ^{244}Cm in the first 100 y, ^{241}Am thereafter to about 1000 y, and the weak peak around 10^6 y comes from the buildup and decay of daughter products ^{229}Th and ^{226}Ra.

The fission product ^{90}Sr, followed by ^{137}Cs, strongly dominates the hazard for the first 500 years or so, and the actinides dominate thereafter. Different kinds of fission reactors and fuel cycles give different mixes of wastes, but all are within a factor of two or three of the case shown. The broad picture is the same for all fission systems.

Spent reactor fuel elements can be either disposed of directly or reprocessed. At present there is no reprocessing of the civilian reactor waste in the U.S.A., and so the high-level waste (HLW) that is the subject of this chapter, consists of all the produced fission products and actinides, including the Pu, and most of the original uranium. The most frequently discussed reprocessing, with or without breeder reactors, would remove 98% or more of the U and Pu and recycle it back to the fission power reactors. Reprocessing is in daily use to remove Pu from "production" reactors for nuclear weapons. Deep burial in geologically stable formations is the principal method proposed for disposal of fission reactor waste.

10.1.2. Waste Hazard and Deep Burial

Any proposed method of dealing with fission reactor high-level waste must be able to show a technical, safety, economic or psychological advantage with respect to deep burial as outlined below.

Characterization of a hazard by the amount of water to dilute it to MPC is appropriate for monitoring the safety of potable water. Similar MPCs apply to water for agricultural uses. However, an MPC is without meaning unless it is coupled in a complete safety analysis to probabilities of escape of the hazardous material by all the relevant mechanisms. For example, more than a year's flow of the Mississippi River is needed to dilute the HLW from one tonne of spent reactor fuel to MPC, but this startling fact ignores the impossibility of dissolving that fuel by natural processes in a year. Similarly, all the water in the world is insufficient to dilute the uranium in the earth's crust to MPC, but the world is not plagued by uranium-contaminated water supplies. The assessment of risk involves consideration of all potential nuclide release mechanisms with their probabilities of occurrence, subsequent movement of the nuclides spatially and environmentally into the biosphere, their uptake by people, and finally, the resultant human health effects.[2]

Safety analyses have been conducted for generic[3, 4] and specific[5, 6] fission waste burial sites or repositories. The principal release mechanisms are either expulsive, by meteor impact or new volcanism, or leaching. The probability of a large meteor impact is about 10^{-13} per year; it is the same everywhere, including natural uranium ore deposits. The probability of new volcanic activity is site-specific, and it is less than 10^{-11} per year for a typical proposed fission waste disposal site. Even if all the waste is expelled, not all of it will be dissolved in water. Leaching is assumed to occur by water entering cracks formed by subsequent faulting. Although burial sites would originally be in dry, unfaulted formations and would be selected for low probability of this event, it is still the geologically most probable release mechanism. But penetration of the barrier does not necessarily mean that humans will be exposed. The rate of release is first limited by finite leach rates. Next, the rate of water percolation through low-porosity rock formations is minimal, typically less than 1 m/y. Repositories should be located sufficiently far from aquifers that the time spent in transit (usually greater than 1000 y/km) allows decay of the most intense radioactive components, ^{90}Sr and ^{137}Cs. In fact, the waste ions migrate much more slowly than the water, because of ion exchange with and retention by the rock. The actinide elements are

strongly retained by this mechanism. Therefore, the most dangerous releases turn out to be ^{129}I and ^{99}Tc, because they are are soluble, are not held up by the rock, and have enough half-life to survive the journey. One safety study for a particular proposed repository found a final hazard equivalent to only 3% of the natural radioactive background[6] resulted from an assumed continuous leaching. Considering all of this, it is difficult to find a significant technical flaw with deep burial.

The favorable conclusions of release analyses of deep burial are supported empirically by studies of the ancient fission reactor that occurred in a rich uranium ore vein some $1.5–2\times10^9$ years ago at Oklo, Gabon Republic.[7] Although the ore was concentrated by water flow, and water was present to moderate the reaction, those fission product elements expected to be retained by the rock are still there today in the proportions corresponding to their generation by fission; they have not migrated anomalously in nearly two billion years. The evidence also shows that the ^{90}Sr, though soluble, all decayed before migrating out of the ore body.

Naturally occurring uranium ore deposits are, on the average, less safely sited than would-be repositories. The ore is also inescapably with us if nuclear power is not developed. The effect of nuclear power is to convert uranium ore into energy and radioactive waste. Therefore, it is appropriate to compare the hazards of waste and the parent ore.[8] (It is not correct, as is sometimes done, to compare the specific activity of waste components and ore, because in principle the waste can always be diluted to low specific activity. Specific activity relates to shielding during treatment and transportation.) The hazard of all the ore that is mined to generate 1 GW(e)-y of electricity in a conventional LWR once-through cycle is also plotted in Fig. 10-1. On this basis, ^{90}Sr is about 2000 times more hazardous initially than the parent ore, but it becomes less hazardous than the ore after about 400 years. The actinides in the spent fuel, with no removal of U or Pu, are initially 40 times more hazardous than the parent ore. Although it takes about 60,000 y to decay to ore level, the actinides are less than 10 times more hazardous than the ore after 400 years. In the very long term, the waste is less hazardous than the ore because some of the original

uranium was fissioned away. Therefore, except for the first few hundred years, the waste hazard is comparable with the hazard of the ore it replaces.

A comparison of the hazard of fission waste and parent ore for a system of LMFBR breeder and LWR burner reactors is shown in Fig. 10-2.[1] The waste now consists of fission products, the heavier actinides and about 0.5% of the U and Pu that remains in the HLW stream. The absolute actinide hazard in this case is a few times less than for the once-through cycle, because most of the Pu is recycled and consumed. The hazard relative to the parent ore is much greater, because uranium is used about 100 times more efficiently to produce energy if the breeder cycle is carried to completion, and it now takes about 40,000 years for the waste to decay to ten times original ore hazard. Provided the selected repository has a long-term release probability of about ten times or more smaller than natural ore bodies, deep burial still can be safer than the parent ore.

To allay doubts about the efficacy of geological containment during the 500 years of greatest radioactivity, additional engineered containment can be added. Egyptian pyramids have already demonstrated good containment for about 3000 years, despite their crude technology. Waste solidification and vitrification is the modern counterpart.

10.1.3. Transmutation of Fission Wastes

The fission products of greatest concern are all transmuted by absorption of a neutron (n,γ) yielding a short-lived nuclide which decays by one or more beta emissions to a stable nuclide. The cross section is greatest for slow or thermal neutrons. It is large enough for the long-lived ^{129}I and ^{99}Tc (31 barns (b) and 22 b, respectively) for transmutation to proceed at technically interesting rates. In the absence of other significant neutron absorbers, the waste can be loaded dilutely and still capture a large fraction of the neutrons. Dilute loading shortens the transmutation half-life. The cross sections of ^{90}Sr (0.8 b) and especially ^{137}Cs (0.11 b) are much smaller. High fluxes of thermal neutrons can be produced in highly moderating blankets of fusion reactors and even in "thermal neutron traps" in breeder fission reactors.

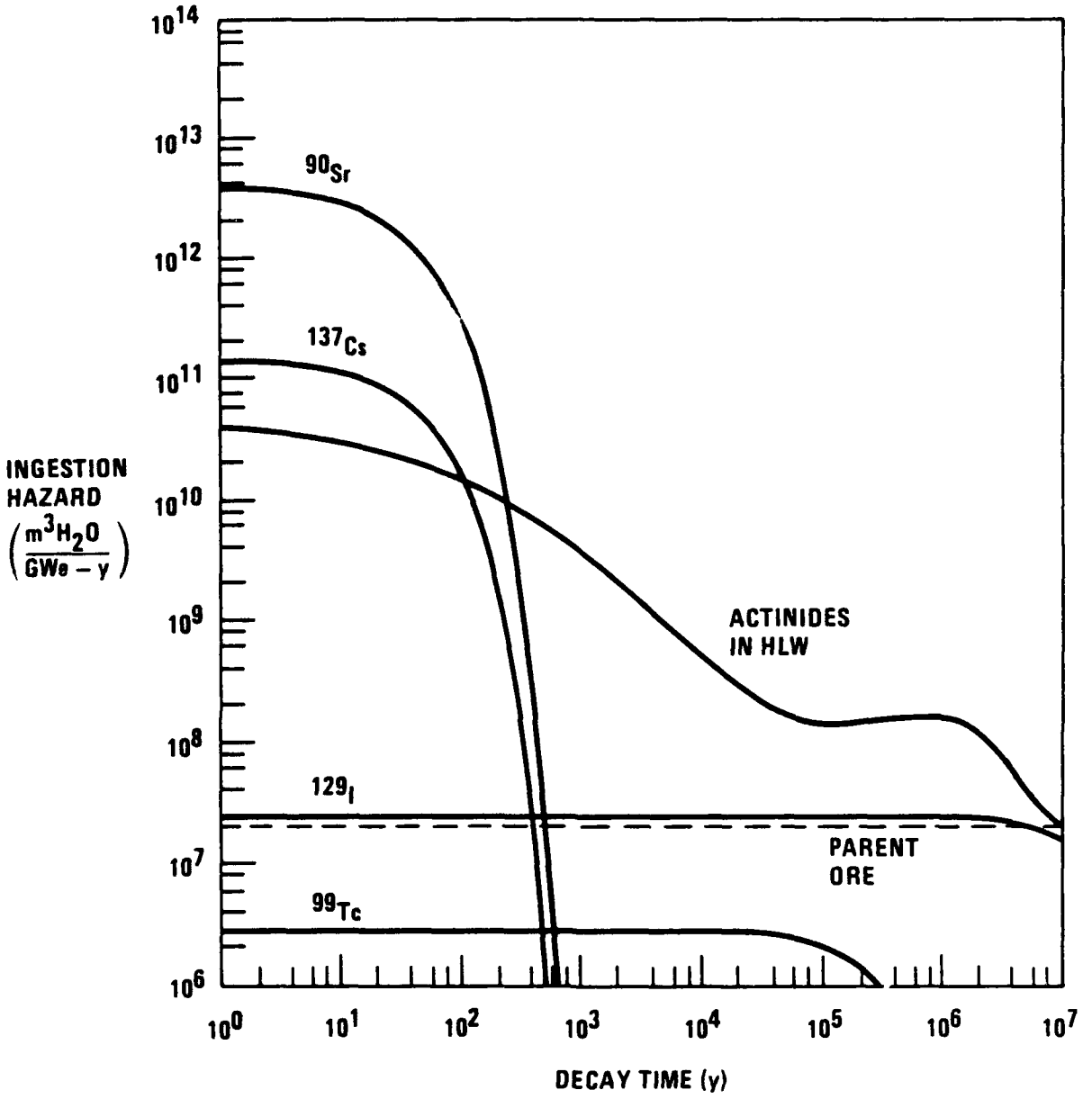


FIG. 10-2. Radioactive waste hazard for an LMFBR breeder-LWR burner system.

It is difficult to envision thermal neutron fluxes that will transmute ^{137}Cs at useful rates. The (n,2n) cross section for 14 MeV neutrons from the D-T fusion reaction is 2.0 b in ^{137}Cs and 1.5 b in ^{90}Sr, but because there is no enhancement of the high-energy neutron

flux by scattering, as there is during moderation, and because all other materials in the system compete for the same high-energy neutrons with cross sections in the same 1 b to 2 b range, little advantage can be gained over (n,γ) captures at presently foreseeable fusion neutron flux levels.

The actinide nuclides have relatively large cross sections for (n, γ) transmutation. However, since all actinides are hazardous and have hazardous daughters, reduction of the long-term hazard is achieved only when the nuclide is fissioned. About half of the actinide isotopes have large fission cross sections for thermal neutrons. Therefore, actinide waste reduction proceeds typically by one or more neutron captures before a fission event takes place. The ratio of fission to non-fission neutron captures is somewhat greater for fast neutrons than thermal ones.

Waste throughput of a transmuter is maximized, other things being equal, by using as many of the available neutrons as possible. This is accomplished by loading the waste in the transmuter at a relatively high density, to maximize the probability of neutron reactions with waste nuclides.

The transmutation half-life (the time to reduce the target nuclide concentration in the transmuter by half) is made shortest when the waste is loaded at low density, and the available neutrons are absorbed in a smaller total number of waste nuclei. Under these circumstances more source neutrons are wasted, by captures in structural or other material and by escape from the waste-containing region. A short transmutation half-life is important for the successful reduction of ^{90}Sr and ^{137}Cs waste. Because these have natural half-lives of about 30 y, a transmutation half-life of 3 y is needed to eliminate waste ten times faster than nature, or 30 years of transmuter operation will then be needed to equal 300 years of natural decay. The small cross sections make it difficult to project transmutation half-lives as small as 3 y for ^{90}Sr, much less ^{137}Cs.

Waste cannot be simply left in a transmuter as a single batch for a sufficient number of transmutation half-lives until very low hazard levels are met, because after one or two half-lives most of the neutrons would not be utilized in the waste-depleted system. In some cases, the stable transmutation products might also be transmuted into new hazardous substances. Furthermore, after a few ^{90}Sr or ^{137}Cs transmutation half-lives, the waste encapsulation would also be severely radiation-damaged and even transmuted. Therefore, waste must be regularly withdrawn, processed to separate untransmuted waste from transmutation products, and the waste reinserted. Highly selective waste partitioning is a necessary part of any process to reduce waste hazard by transmutation. Needless to say, spent reactor fuel must also be partitioned to separate actinides and possibly some combination of ^{90}Sr, ^{137}Cs, ^{129}I and ^{99}Tc from the remaining waste.

All forms of nuclear waste transmutation were discussed at an international conference in 1980.[9]

10.2. TRANSMUTATION BY FISSION REACTORS

Excess neutrons from fission reactors could also be used to transmute fission reactor wastes.

10.2.1. Partition and Recycle in LWR

It has been proposed since the mid-1960s that high-level waste might be partitioned and selected isotopes recycled for transmutation in a fission reactor.[10] After other early studies indicated the basic feasibility of the concept,[11] more realistic studies were performed. The most complete of these was a broadly-based three-year program of experimental and computational activities to evaluate the technical feasibility and quantitative incentive for transmutation in a closed light water reactor (LWR) cycle. The study was initiated by ORNL, and eight U.S. Department of Energy laboratories and three private companies participated.[12] This is by far the most complete study of any waste transmutation concept to date and must be given due consideration.

The general approach of the ORNL study was to perform an incremental cost-risk/benefit analysis of partition and recycle in LWRs. To do this, two closed LWR fuel cycles were defined. The reference cycle involved reprocessing for the recovery and recycle of uranium and plutonium, whereas the transmutation cycle used additional processes for maximal recovery and recycle for transmutation of all actinides from refabrication and reprocessing plant wastes. The total unrecovered actinides in the enhanced cycle with near-term partitioning technology is about 0.25%, based on available data and explicit flow sheets. This is substantially below the economic optimum with conventional reprocessing. For the first four recycles, the actinide wastes were a net neutron poison, and additional uranium enrichment was needed. After the fifth recycle, the accumulation of readily fissionable higher actinides resulted in a net fissile benefit, and enrichment was decreased accordingly.

It was concluded that LWR transmutation is possible. The high concentration of neutron emitting Cm and Cf isotopes in the recycled fuel would require new shipping casks, but the neutron activity would make illicit diversion more difficult.

The incremental cost of partitioning and transmutation was estimated to be a 5% cost increase in nuclear-generated electricity. The incremental short-term (contemporary) risk to the public comes from the additional processing, larger quantities of actinides in the fuel cycle and a greater number of fuel shipments. This was quantified as only 0.57 additional health effects/GW(e)-y. (A health effect is a premature death or loss of 6000 days of labor.) Of this, only 0.003 health effects/GW(e)-y were of radiological origin. For comparison, the rate from natural background radiation is equivalent to 1.0 health effect/GW(e)-y. The largest non-radiological health effect came from petroleum combustion products for the generation of steam and heat at the waste treatment plants, followed by non-radiological damage during transportation.

The benefit of partition and transmutation is the reduction in the expected long-term health effects through possible release from the geologic repository. Specific calculations[6] were performed for the Waste Isolation Pilot Plant in New Mexico.[5] The probabilistic risk over a one million year period was 5.16 health effects/GW(e)-y for the standard cycle and 5.10 after partitioning and transmutation, for a net benefit of 0.06, or a little more than 1%. The small benefit is explained by the facts that most of the health effects accrue from the risk of leaching iodine and technetium. As pointed out earlier, the actinides are strongly retained by ion exchange and are not carried from this repository to the biosphere. Actinide transmutation therefore yields little benefit. The above absolute health effect numbers depend on the population projection for the vicinity of the repository site. A population-independent measure of the benefit is that it is equivalent to a reduction of less than 10 parts per million in the natural background radiation. The study concludes that the incremental cost and short-term risk of actinide partitioning and transmutation is not justified by the small incremental long-term benefit.

10.2.2. Partition and Transmutation in Breeder Reactors

Attention has also been given to separating the actinides from the high-level waste and recycling them to breeder reactors for transmutation.[13] The concept is the same as for transmutation in LWRs, but the fuel cycle and neutronic details differ. Most attention has been given to LMFBR breeders. The basic conclusion is that actinide recycle does not seriously affect breeding, and safe and stable reactors can be designed. The fast neutron spectrum in the breeder gives a greater fission/capture ratio, so there is less buildup of the higher actinides than in LWR transmuters. Risks and benefits have not been extensively studied in the context of breeder actinide transmuters, but there is no reason to expect them to differ greatly from the findings of the ORNL study of LWR actinide transmuters.

10.3. TRANSMUTATION BY SPALLATION

A limited amount of study has been devoted to the use of nuclear spallation. In spallation a beam of accelerated light particles, typically protons or deuterons at around 1 GeV per nucleon, collide with stationary target nuclei. Many particles per collision are spalled from the target, which can be an intense source of neutrons. The energy of a spallation neutron is intermediate between fission and D-T fusion neutrons. Spallation systems have been proposed as fissile fuel breeders[14] and waste transmuters.[10, 15, 16] Most of this work in the U.S.A. has been done at Brookhaven National Laboratory.

Spallation transmuter concepts usually try to take advantage of the neutron flux to effect waste transmutation. Spallation in lead produces about 35 neutrons per 1 GeV beam proton. The technology of GeV proton accelerators is well developed, and they have an energy efficiency, from electricity to beam, of about 50%. Flowing liquid lead targets have been considered for heat removal in compact sizes. Heat can be recovered from the target and used to supply a small part of the accelerator power. Actinide and/or fission product wastes would be placed in a surrounding blanket, which may also include uranium or thorium for fissile fuel breeding and/or neutron multiplication. While the spallation neutron spectrum is fast, it is not fast enough to produce significant (n,2n) reactions in ^{90}Sr or ^{137}Cs. The small target size, roughly a cylinder 1 m long by 0.1 m radius, permits smaller waste blanket inventories than fusion transmuters, and the intense source is capable of transmuting ^{90}Sr and ^{137}Cs at perhaps 30 and 10 times faster, respectively, than natural decay. Direct spallation of wastes used as the beam target have been considered, but this transmutation is less efficient than by neutron generation. Spallation transmuters could be developed sooner than fusion transmuters. The spallation accelerator draws a large external power, unless the target and blanket contain sufficient fission energy multiplication to make the system self-sustaining. All the cost, risk and benefit considerations discussed for fission transmuters apply equally to spallation systems.

10.4. TRANSMUTATION BY FUSION REACTORS

Fusion, especially D-T fusion, is often cited as being "neutron-rich," meaning that the energy release per neutron generated is low compared with other options. Therefore, it is not surprising that the idea of transmuting fission wastes in fusion transmuters has been independently proposed by various researchers. This line of research in the U.S.A. was first given focus and direction by W.C. Gough of the AEC in 1971. When the complete fusion fuel cycle, including tritium breeding is considered, the number of excess neutrons per unit energy release is considerably less "rich." However, modern fusion blankets have many more excess neutrons than early blanket concepts, and fusion still potentially offers the cheapest source of neutrons in energy terms.

Fusion transmutation was first examined for the reduction of fission products.[17–19] These studies used idealized and simplified blankets and led to a rather optimistic preliminary evaluation of the concept, although many problem areas were also recognized. This motivated the Electric Power Research Institute (EPRI) to commission a series of studies on both fission product and actinide waste transmutation, from 1975 through 1978. It is mainly the results of these more detailed studies that are summarized in the next two subsections.

10.4.1. Fission Product Transmutation

The most detailed series of studies of fusion transmutation of fission wastes was sponsored by the Electric Power Research Institute (EPRI) in response to optimistic preliminary evaluations of the concept. The heart of these studies, neutron analysis of transmutation of selected fission products in semirealistic cylindrical blankets surrounding a D-T fusion plasma, was performed at the Nuclear Reactor Laboratory of the University of Texas at Austin.[20] The more complete cases included structure, coolant, and finite concentrations of waste. No attempt was made to breed tritium needed to sustain the fusion reaction, as it was assumed that enough fusion reactors would be operating to supply the transmuter with their excess tritium. Batch irradiation was considered almost exclusively, in order to

minimize the number of waste processing operations. Calculation of hazard during and after irradiation included all possible transmutation paths starting from the original waste and blanket materials.

Highly moderated idealized blankets are capable of impressive transmutation rates. For example, a meter-thick Be blanket without structure achieved a 0.85 y ^{90}Sr transmutation half-life at 10 MW(n)/m^2 wall loading — when the waste was dilutely loaded only in the first centimeter of the blanket. Neutron utilization was unacceptably poor. The best of the "realistic" blankets, a He-cooled graphite blanket in which no allowance was made for structure to contain the pressurized helium, achieved about a 3 y ^{90}Sr transmutation half-life at the same wall loading with dilute waste concentration. Increasing the concentration to where 30% of the incident neutrons were absorbed in ^{90}Sr increased the transmutation half-life to about 5 y. Transmutation of ^{137}Cs in this blanket proceeded much more slowly. It was found that the long-lived isotopes ^{129}I, ^{99}Tc and/or ^{135}Cs, which all have large thermal capture cross sections, could be transmuted effectively in an outer blanket region by neutrons escaping from the ^{90}Sr region. The generation of minor waste transmutation products and blanket transmutation products can dominate over the reduction of the primary waste hazard for long irradiation of some of the low-hazard fission products.

A simplified ^{90}Sr waste management system was also analyzed in the University of Texas study.[20] Although numerical results were reported for only one particular scenario of fission reactor growth and transmuter introduction, the qualitative finding is generally valid. An optimistic transmutation system might reduce the amount of ^{90}Sr in final disposal by many tenfold, relative to no transmutation. However, the ^{90}Sr inventory in reprocessing facilities and the even greater inventory in the transmuter blankets greatly exceeds the buried amount, although it is still much less than the amount of naturally decaying waste without transmutation. This phenomenon can be expressed simply for a steady-state system, where the rate of generation of waste is constant. Let the waste hazard get transmuted with an effective half-life t_t (including the effect of reprocessing operations) to

a final level η times the input level. Also let t_n be the natural half-life, N_t the inventory in the transmutation and processing system, and N_{st} and N_{sn} the inventories in final burial with and without transmutation, respectively. It is then easy to show that

$$\begin{aligned} \frac{N_t}{N_{st}} &= (1-\eta)\, t_t/\eta\, t_n \quad , \\ &\approx t_t/\eta\, t_n \\ \frac{N_t}{N_{sn}} &= (1-\eta)\, t_t/t_n \\ &\approx t_t/t_n \quad . \end{aligned}$$

The approximate forms are valid when $\eta \ll 1$, corresponding to good transmutation. As an example, consider $t_t/t_n = 6/30 = 0.2$ and $\eta = 100$. Then the transmuter waste inventory is 20% of what the buried inventory would be in the absence of transmutation. It is unlikely that the risk from 20% of the ^{90}Sr waste in blankets and processing operations represents an improvement over 100% of the waste properly buried; it may not be any improvement at all. The 1% of waste buried in the transmutation scenario presents a negligible risk compared to the waste held up in the transmutation system.

The principal conclusions of these studies were as follows:

1. Transmuter neutronic performance is very sensitive to the effects of blanket structure and finite waste concentration.
2. ^{90}Sr can be transmuted at marginally interesting rates by 10 MW(n)/m^2.
3. ^{137}Cs cannot be transmuted at interesting rates by even 20 MW(n)/m^2.
4. The most hazardous long-lived fission products can be transmuted rapidly at readily achievable fusion wall powers.
5. Unless the medium-lived ^{90}Sr and ^{137}Cs can be transmuted as many times faster than natural decay as the hazard reduction desired, transmuter inventory hazard is certain to dominate the risk. There is no way to achieve such rates with any forseeable conventional magnetic fusion reactor.

In summary, magnetic fusion transmutation of long-lived fission products is possible. Medium-lived ^{90}Sr can be transmuted, but not at a rate that produces any clear advantage. Cesium-137, which must be considered if ^{90}Sr is successfully transmuted, is virtually untransmutable.

10.4.2. Actinide Transmutation

The actinide fusion transmuter study commissioned by EPRI and performed by Westinghouse[8] is by far the most complete one to date. The stated goal was to design an actinide transmuter using near-term (c. 1975) fusion technology. The transmuter had a neutron beam-driven, TFTR-like tokamak fusion core and a fast-spectrum, high-multiplication, tritium-breeding, waste-containing blanket. The plasma physics set the basic machine size, R = 3.9 m. Reliability, maintainability and safety were important design considerations. An extensive engineering study was performed. The transmutation performance was found to be substantially less than had been originally projected, which underscores the importance of doing realistic calculations for waste transmuters.

Due to materials technology limitations, the first wall neutron power loading was limited to 1.15 MW(n)/m^2. The neutron flux was boosted, until limited by the 250 W/cm^3 blanket cooling limit, by neutron multiplication ($k_{eff} \approx 0.9$) from the fissioning actinide waste. The resulting transmuter thermal power exceeded 13 GW, of which only 0.4 GW was fusion power. This was actually a fusion-driven actinide fission reactor. 1.0 GW(e) was generated and recirculated to sustain the driven fusion plasma, and the remaining 10 GW(t) was used to make hydrogen, so as not to have an excessively large single unit on the electric power grid.

Despite the above measures, actinide transmutation proceeded slowly. Initially the hazard actually rose, as more dangerous isotopes appeared by neutron captures. Eventually fissioning took over and the the hazard declined. With this blanket and waste load combination, it took more than 30 years of irradiation to effect a ten-fold reduction in hazard. Other calculations showed that at 10 MW(n)/m^2, with a more dilute waste load and

less neutron multiplication but the same 250 W/cm^3 limit, one could achieve a 1000-fold hazard reduction after a 30 y irradiation. The strong nonlinearity came from the much harder neutron spectrum, dominated by fusion neutrons, in the latter case. Although today first wall loadings on the order of 10 $MW(n)/m^2$ are being seriously considered for some fusion reactor concepts, such walls need frequent changes, which is inconvenient when inventories of hazardous waste are present. Finally, the support ratio on a thermal basis was about 5. This number is representative of the fission energy content of the actinide waste of most closed fissile fuel cycles and is not subject to much change. Actinide transmutation, whether by fusion or by recycle in fission reactors, can be viewed as burning fissile and fertile material that would otherwise be wasted.

Some conclusions of the Westinghouse study are as follows:

1. Many difficult and challenging engineering problems must be solved to design a fusion transmuter.

2. Full transmutation calculations must be done to derive any meaningful result. Capture and fission rates depend on the neutron energy spectrum, which depends on the waste load and the blanket structure. The actinide decay chains must be followed and the hazards of decay daughters calculated.

3. Low multiplication, dilute blankets with hard spectrum transmute actinides more effectively for a given blanket power density. This implies high neutron first wall power loadings, which will require technological advances.

4. The hazard of the parent uranium ore as a reference hazard for comparison of different waste management plans was introduced by this study.

10.4.3. Advanced Fusion Transmuters

The EPRI-sponsored transmutation studies showed that high first wall neutron power loading is desirable for both fission product and actinide transmutation in fusion reactor blankets. Liner fusion, where a thick, rotating, liquid vortex (liner) is pneumatically imploded to adiabatically compress an enclosed magnetically confined plasma to fusion ignition,[21] was identified as having the potential for cycle-averaged wall loadings in excess of 100 MW(n)/m^2. EPRI therefore commissioned a background study of liner fusion transmuters.[1]

Liner fusion issues were studied at General Atomic Company (now GA Technologies Inc.), and the neutronics and transmutation calculations were performed at the University of Texas. The plasma was taken as either a field-reversed compact torus or an end-plugged theta pinch, and appropriate physics and engineering constraints were developed. Wastes were dissolved in the liquid blanket, which was stratified in immiscible layers in the optimum cases. Blanket chemistry, immiscibility, and vapor pressure of the surface placing the plasma, which has to be a metallic electrical conductor, severely restricted the ways in which Sr and Cs could be incorporated. Actinides could be dissolved directly in molten lead-lithium liner material in sufficient concentration (~1%) to be transmuted. Even at this low concentration, the large fission power release limited the pulse repetition rate.

Overall, the hybrid field-reversed plasma and liner system was found to make an effective actinide transmuter. Strontium and cesium transmutation was inefficient, with only 5% to 10% of the fusion neutrons effecting transmutations, because the fusion neutrons could not be adequately thermalized in the blanket, and they were rapidly degraded below the (n,2n) threshold energy by the majority blanket components. However, because wall loadings of at least 250 MW(n)/m^2 were calculated to be possible, transmutation half-lives of 3 y to 5 y were predicted.

Inertial confinement fusion systems can transmute waste incorporated in the pellet. There are few published details, because most of the information is classified. One published summary of Monte Carlo neutron transmutation calculations for ^{243}Am, ^{241}Am and ^{237}Np in the target plasma show that 20% to 30% of the initial waste could be transmuted per shot.[22] Quenching of the fusion plasma by the high-Z waste was considered and set an upper bound on transmuted product per shot. Both waste and product must be removed from the reaction chamber, processed, and the untransmuted waste recycled. Although waste will contaminate the coolant, if a liquid wall is used, the total waste inventory in the transmuter can be kept much lower than when it is incorporated in a blanket surrounding a magnetic fusion reactor.

It has also been suggested that transmutation can be effected on isotopes added dilutely to magnetically confined D-^{3}He fusion plasmas.[23] The proton unleashed by this fusion reaction exceeds the (p,n) reaction threshold energy for the waste nuclide. The excellent magnetic confinement of the proton, which would be a corollary of having achieved successful D-^{3}He fusion, lets the fast proton flux build up to about 10^{22} $(cm^2/sec)^{-1}$. This yields transmutation half-lives on the order of 100 sec. In-plasma D-^{3}He transmutation is therefore also a low-inventory concept. Strontium-90 is transmuted to ^{90}Y, which decays by beta emission to stable ^{90}Zr.

10.5. GENERAL EVALUATION

It is incorrect to draw conclusions based on analysis of only one piece of the waste transmutation problem, such as neutronic performance. At such an isolated level, transmutation of specific isotopes is technically feasible in high-power-density fusion reactors, accelerator spallators and fission reactors themselves. But when incremental cost/risk/benefit calculations are done, transmutation offers little quantifiable improvement at a non-negligible cost. This is because geologic sites exist which have been physically and chemically stable for millions of years and have already been proved as effective barriers for isolation of naturally occurring radioactive uranium and thorium deposits from the human environment.

The real impediments to geologic waste disposal are perceptual and political. Multiplication of the risk by a few more negative powers of ten by transmutation will not change attitudes already rigidly formed. It is even possible that serious discussion of such heroic efforts will only serve to reinforce the commonly held lay impression that there is no safe way to dispose of fission waste. It is likely that an education program implemented by psychologists, social and political scientists, and communication specialists would be more cost-effective in the present situation than research on waste transmutation.

References

[1] M.J. Schaffer and T.A. Parrish, "Background Study of Liner Fusion Systems for Transmuting Fission Reactor Wastes," EPRI AP-1644, Electric Power Research Institute (1980).

[2] S.E. Logan and M.C. Berbano, "Development and Application of Risk Assessment Method for Radioactive Waste Management, Vol. I–IV," EPA 520/6-78-005, U.S. Environmental Protection Agency (1978).

[3] K.J. Schneider and A.M. Platt, editors, "Advanced Waste Management Studies: High-Level Waste Disposal Alternatives," BNWL-1900, Batelle Pacific Northwest Laboratory (1974).

[4] U.S. Energy Research and Development Administration, "Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle," ERDA-76-43 (1976).

[5] W.D. Weart, "New Mexico Waste Isolation Pilot Plant Mid-1977 Status Report," SAND 77-0994, Sandia National Laboratories (1977).

[6] S.E. Logan, R.L. Conarty, H.S. Ng, L.J. Rahal and C.G. Shirly, "Actinide Partitioning-Transmutation Program Final Report, Vol. VII, Long-Term Analysis of the Geologic Repository," ORNL/TM-6987, Oak Ridge National Laboratory (1980).

[7] G.A. Cowan, "A Natural Fission Reactor," **Scientific American**, 253, No. 1, **39**, (July 1979).

[8] R.P. Rose, *et al.*, "Fusion-Driven Actinide Burner Design Study," EPRI ER-451, Electric Power Research Institute (1977).

[9] International Conference on Nuclear Waste Transmutation (Proc. Conf. Austin, July 22–24, 1980), The University of Texas at Austin (1981).

[10] M. Steinberg, G. Wotzak and B. Manowitz, "Neutron Burning of Long-Lived Fission Products for Waste Disposal," BNL 8558, Brookhaven National Laboratory (1964).

[11] H.C. Claiborne, "Neutron Induced Transmutation of High-Level Radioactive Waste," ORNL/TM-3964, Oak Ridge National Laboratory (1972).

[12] A.G. Croff, J.O. Blomeke and B.C. Finney, "Actinide Partitioning-Transmutation Program Final Report, Vol. I, Overall Assessment," ORNL-5566, Oak Ridge National Laboratory (1980). Also see detailed companion reports: ORNL/TM-6982 through ORNL/TM-6987 (1980). Summary papers of this study are contained in Ref. 9 above.

[13] S.L. Beaman and E.A. Aitken, "Feasibility Studies of Actinide Recycle in LMFBRs as a Waste Management Alternative," Trans. Am. Nucl. Soc., **23**, (1976) 547.

[14] M. Steinberg, *et al.*, "Electronuclear Fuel Production – The Linear Accelerator Fuel Regenerator and Producer (LAFER and LAFP)," Atom-Kernenergie **32**, (1978), 39.

[15] M.V. Gregory and M. Steinberg, "A Nuclear Transformation System for Disposal of Long-Lived Fission Product Waste in an Expanding Nuclear Power Economy," BNL-11915, Brookhaven National Laboratory (1967).

[16] L.E. Burns, in Ref. 9 above, p. 13. J. Powell in Ref. 9 above, p. 99. H. Takahashi, N. Mizoo and M. Steinberg, in Ref. 9 above, p. 557.

[17] W.C. Wolkenhauer, "The Controlled Thermonuclear Reactor as a Fission Product Burner," **Trans. Am. Nucl. Soc., 15** (1972) 92.

[18] B.F. Gore and B.R. Leonard, Jr., "Transmutation in Quantity of ^{137}Cs in a Controlled Thermonuclear Reactor," **Trans. Am. Nucl. Soc., 17** (1973) 52.

[19] B.F. Gore and B.R. Leonard, Jr., "Transmutation of Massive Loadings of Cesium-137 in the Blanket of a Controlled Thermonuclear Reactor," **Nucl. Sci. and Eng., 53** (1974) 319.

[20] T.A. Parish and E.L. Draper, Jr., "Determination of Procedures for Transmutation of Fission Product Wastes by Fusion Neutrons," EPRI Project 474-2 Final Report, Electric Power Research Institute (1976).

[21] D.L. Book, *et al.,* "Stabilized Imploding Liner Fusion Systems," Plasma Physics and Controlled Nuclear Fusion Research (Proc. Conf. Berchtesgaden, 1976), Vol. III, IAEA, Vienna (1977) 507.

[22] H. Meldner,"Inertial Confinement Fusion and Long-Term Nuclear Waste Management," **Nucl. Sci. and Eng., 69** (1976) 438.

[23] J.M. Dawson, private communication (1986).

11. SPACE POWER AND PROPULSION

Fusion power for electrical generation in space and for thrusting is especially appealing for deep space missions where fuel supplies would otherwise be of concern. For example, for a 100 MW(e) fusion power source used in a five-year deep space mission, only about 200 kg of D-T would be consumed. For thrusting, the specific impulse or thrust per unit mass of propellant used is a key parameter. An alpha-heated plasma at 10 keV could generate specific impulses of order 100,000 sec, over 100 times that available from the best high pressure liquid hydrogen/oxygen system. High thrust and high impulse will be necessary for deep space missions to be accomplished in short times. Figure 11-1 shows the tradeoff between payload fraction at launch and mission duration for chemical, fission, and fusion rockets. The difference is due to the vastly different specific impulses for the three modes. It is clear that fusion can exhibit very high specific impulse and thus would use much less fuel to achieve the same thrust. The primary concern for fusion power in space is therefore not fuel supply but the total weight and bulk of the components.

11.1. BASICS

Specific impulse is defined by

$$I_{sp} = \frac{T}{\dot{m}g} = \frac{u_{ex}}{g}$$

where T is the thrust, $\dot{m}$ is the propellant mass flow, u_{ex} is the rocket exhaust velocity and g is gravitational acceleration (9.8 m/sec^2). The power that must be supplied to the rocket exhaust is

$$P = \frac{1}{2}\,\dot{m}\,u_{ex}^2$$

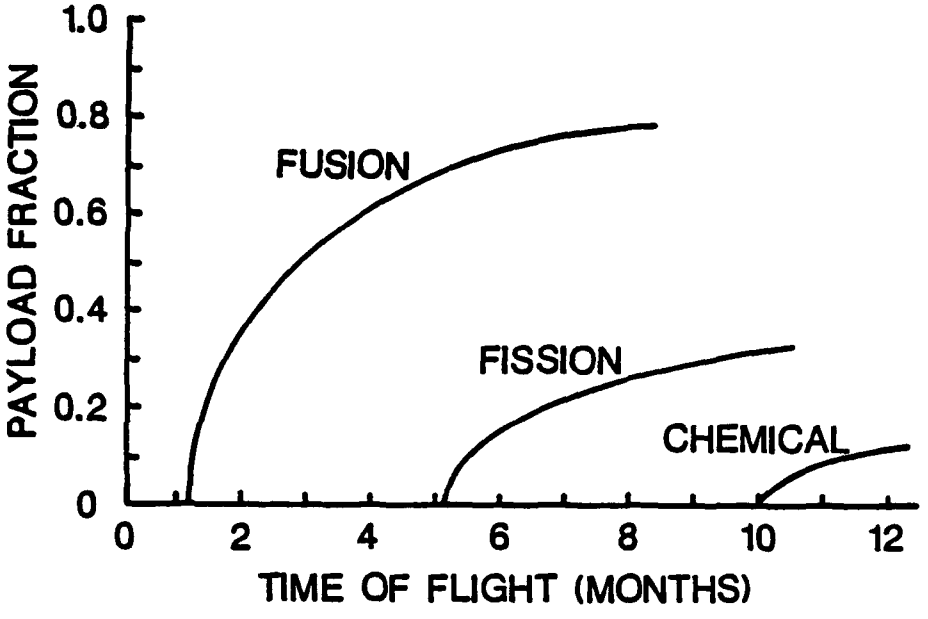


FIG. 11-1. One-way Earth to Jupiter mission.[1]

so that the power-to-thrust ratio is

$$\frac{P}{T} = \frac{u_{ex}}{2} \sim I_{sp}$$

while the mass flow per unit thrust is

$$\frac{\dot{m}}{T} = \frac{1}{u_{ex}} \sim \frac{1}{I_{sp}}$$

At low impulse, the propellant supply is excessive. At high impulse, the power requirements are too high. Clearly, an optimum specific impulse exists that is dependent on the mission requirement's total mass and the power source. An analysis for electric propulsion[3] is directly applicable to fusion. In that analysis, the payload fraction was shown to be

$$f_p \equiv \frac{M_{P.L.}}{M_o} = e^{-\Delta u/u_e}\left(1 + \frac{\alpha\, u_e^2}{2\eta t_B}\right) - \frac{\alpha\, u_e^2}{2\eta t_B}$$

where f_p is the payload fraction, the ratio of payload to launch weight, Δu is the mission velocity increment (*e.g.*, 7.4 km/sec for a one-month earth-moon orbital transfer), t_B is the mission duration, α is the total mass at launch in kg/kW, and η is the thrust chamber efficiency. Total mass for fusion plants can be expected to be 10–100 kg/kW. Thrust chamber efficiencies should be 70–90%. This equation has an optimum exhaust velocity u_e that maximizes payload fraction for a given mission as specified by Δu and t_B and for a given fusion plant as specified by α/η. For example, in Ref. 3, it is shown that, for a lunar mission with $\alpha/\eta = 10$ kg/kW, payload fraction is maximized at a specific impulse of 2000 sec and decreases for higher impulse because of the increased power requirements. Longer missions with higher velocity increments have higher optimum specific impulses.

11.2. FUSION PROPULSION

With the exception of thermonuclear explosives, no fusion power source has been shown to be compact and lightweight when all of the components, including power supplies and energy storage, are included. The only simplifications in space-based fusion power compared to ground-based are the possible simplification of the vacuum system and reduction in shielding. The former is a small weight factor as is. The latter could yield significant weight reductions because there is no atmosphere to scatter neutrons back to living quarters or delicate electronics. Full shielding coverage should therefore not be required.

Early fusion rocket studies[1, 2] assumed D-^{3}He or D-D operation where the bulk of the energy release is in charged particles available for thrusting. Mirror and multipole configurations were adopted for confinement. In the light of today's knowledge, these would probably be the last to be considered. Nevertheless, these idealistic studies did show an order of magnitude improvement over the best forseeable fission-electric system. After allowance for a more realistic evaluation based on current knowledge, fusion might still provide a superior thrust/weight ratio than fission.

Recent work on accelerating of compact toroids[4] suggests that these configurations could provide efficient fusion thrusters. Figure 11-2 shows a representative schematic. One would envision a low-beta D-T toroidal plasma initially formed ohmically and then accelerated to a directed energy corresponding to about 5 keV. The plasma would then be compressed, decelerated, and heated to ignition in a conducting converging channel. Energy from D-T neutrons would be converted to electricity to power the accelerator and service the spacecraft. Alpha power would heat the plasmoid to perhaps 20–40 keV. This could be recovered as thrust by accelerating it through a diverging conducting channel.

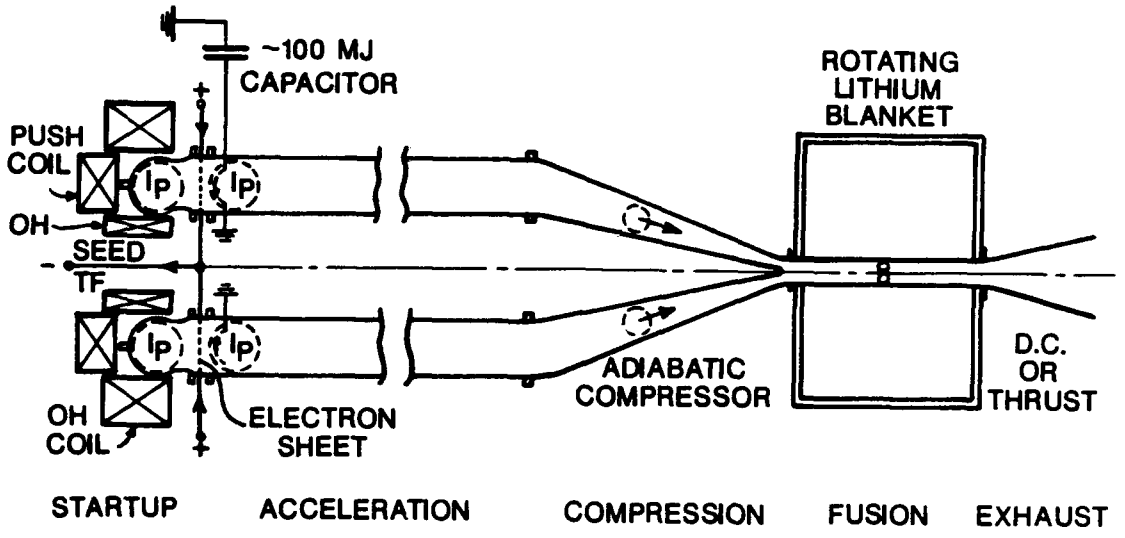


FIG. 11-2. Schematic of compact toroid fusion thruster.

Because all fusion thrusters using magnetic confinement will require copious amounts of electricity to keep them running, D-T may be as good a fusion fuel as any. Compared to D-D or D-^{3}He, confinement is easier, reactors are smaller, and the neutron energy can be used in a thermal conversion system to supply that electricity.

In addition to compact toroids, other magnetic confinement systems deserve consideration for fusion rockets. Among them are the dense plasma focus, the transpiration-cooled,

wall-confined concept, and possibly the unequal mirror where mirror ratio is less at the thrust chamber end than at the other end. In all of these cases, concern exists about excess recirculating power relative to the fusion power produced. With D-T, assuming a thermal conversion efficiency of 25% (*e.g.*, an organic Rankine cycle with high-heat-rejection temperature for radiation into space), the electrical requirements cannot exceed the alpha power produced ($P_e = 0.25P_n = P_\alpha$) without recourse to supplemental electricity.

Inertial confinement fusion has also been explored for space thrusting. Recent studies have been done at LLNL[5] to explore the feasibility of rockets powered by fusion micro-explosions. A conceptual design was developed using slightly tritium-enriched deuterium as the fusion fuel, a high-temperature KrF laser as the driver, and a thrust chamber consisting of a single superconducting current loop protected from the pellet explosion by a radiation shield. According to the study, the power-to-weight ratio turned out to be high enough to consider ambitious projects like cargo runs to Pluto.

In summary, fusion does have promise for supplying electrical power for long periods with negligible fuel consumption or providing high thrust with extraordinary specific impulses. The important issues therefore center around the weight of the fusion plant, and the electrical power required to drive it.

References

[1] J.L. Hilton, **IEEE Trans. Nucl. Sci., NS-10**, 153 (1963)

[2] NASA technical memorandum NASA-TM-X-2106 (1970)

[3] P.G. Hill and C.R. Peterson, **Mechanics and Thermodynamics of Propulsion**, Addison-Wesley, 1965.

[4] C.W. Hartman and J.H. Hammer, "A New Type of Collective Accelerator," LLNL report UCRL-89980, Rev. 1, March 1982.

[5] R.A. Hyde, Paper #IAF-83-396, 34th Congress of the International Astronautical Federation, Budapest (1983)

12. MEDICAL

Some of the ways fusion energy could be used for medical applications indirectly by the production of radioisotopes have already been mentioned. It is possible that fusion could also be used directly for diagnosis and treatment. One treatment possibility is neutron radiography. However, it is extremely limited as a diagnostics application, especially when compared with the well-established X-ray technology, since it cannot distinguish the boundaries between flesh and bone. More research, including additional innovative designs, is needed to improve neutron beam collimation, efficacy, and ease of handling and utilization. It is possible that fusion facilities might be useful in the detection and monitoring of osteoporosis (decalcification of the bones), where a whole-body irradiation by low-dose 14 MeV neutrons would be needed.

Neutron energy in the range of 20 MeV, above that available from fission or fusion, is preferred for the effective treatment of tumors. The most significant potential medical contribution of fusion neutrons, therefore, is in the production of radioactive nuclides that are used as tracers, gamma ray sources and implants. The most important isotope is, of course, ^{60}Co, but others are ^{90}Y, ^{153}Gd, ^{3}H, ^{14}C, ^{32}P, ^{33}P, ^{35}S, ^{45}Ca, ^{55}Fe, ^{59}Fe, ^{137}Cs, ^{24}Na, ^{51}Cr, ^{131}I, ^{90}Sr, ^{198}Au, ^{192}Ir, and ^{182}Ta. An important part of fusion research will be the study of how to produce these and other isotopes for the expanding field of nuclear medicine.

13. APPLICATIONS OF ELECTROMAGNETIC WAVES

High-temperature plasmas emit a variety of electromagnetic waves. This chapter explores the tailoring of these waves to specific frequencies and their utilization.

13.1. ELECTROMAGNETIC EMISSIONS

Fusion plasmas will radiate electromagnetic energy throughout a very broad spectrum, ranging from nearly audio frequencies (drift waves) through penetrating X-rays from runaway electrons. Most of the total radiated power is, however, concentrated in two broad bands. The first consists of synchrotron radiation harmonics, and for most currently envisioned magnetic fusion reactors these will in the sub-millimeter range. The second consists of X-rays at energies on the order of the plasma temperature, from plasma bremsstrahlung and possibly impurity line radiation. Ultraviolet radiation from near the plasma periphery by low-Z impurities might also be important in some cases. X-ray and UV radiation excites and ionizes atoms. Sub-mm photon energies are less than room temperature; they do not ionize directly, but they can excite molecular rotations.

The D-T fusion reaction releases 80% of its energy to neutrons and only 20% to ions. Only the latter, which are confined by the magnetic field, heat the plasma, which in turn powers electromagnetic radiation. Therefore, electromagnetic power will always be a minor portion of D-T reactor output. D-^{3}He releases most of its energy to ions. By running with a helium-rich mixture, neutron-producing side reaction power can be held to 5% or less, then electromagnetic power can be a majority of the reactor output. The possibility of D-^{3}He reactors can be considered more seriously now that it appears that exploitable amounts of ^{3}He are present in lunar soils.[1]

The mix between X-ray, sub-mm and plasma transport powers can be varied to a considerable extent by the choice of operating plasma electron temperature T_e, electron beta β_e and effective ionic charge Z_{eff}, according to the scaling laws for bremsstrahlung power,

$$P_b \sim n_e^2 Z_{eff} T_e^{\frac{1}{2}} \quad , \tag{1}$$

synchrotron power in all harmonics above the plasma frequency,

$$P_s \sim n_e^2 T_e^2/\beta_e \quad , \tag{2}$$

and charged particle fusion power,

$$P_f \sim n_i^2 Q_c \langle \sigma v \rangle \quad . \tag{3}$$

Here n_e and n_i are electron and ion particle densities, Q_c is the charged fusion product energy and $\langle \sigma v \rangle$ is the fusion reactivity. Weak parametric dependences have been ignored in Eq. (2). These dependences are illustrated qualitatively in Fig. 13-1. Ratioing (1) and (2),

$$\frac{P_s}{P_b} \sim \frac{Te^{3/2}}{Z_{eff}\,\beta_e} \quad . \tag{4}$$

Synchrotron radiation is enhanced over bremsstrahlung by increased electron temperature and decreased Z_{eff} and β_e, and vice-versa. The bremsstrahlung fraction can also be increased by reflecting as much synchrotron radiation as possible from the reactor first wall back into the plasma. Particle transport is determined implicitly as the difference between fusion power and the radiated powers. To make the radiated power fraction large, it is necessary to reduce particle transport through its dependence on plasma parameters and, especially, by increasing the physical size of the plasma. It may be possible to make P_s/P_f as large as 2/3 in large D-^{3}He reactors,[2] although 1/2 is probably a more realistic value. Some possible applications of sub-mm and X-ray power are discussed in the following two sections.

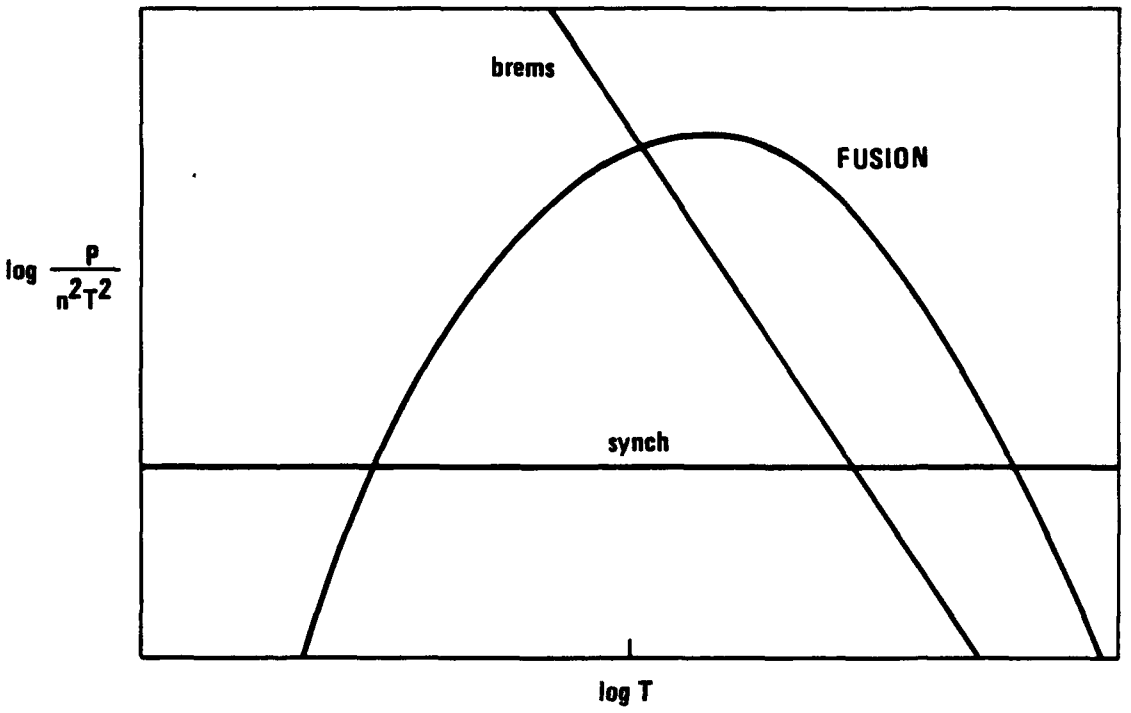


FIG. 13-1. Qualitative dependences with temperature of total fusion, bremsstrahlung and synchrotron powers. Powers are normalized to n^2T^2.

13.2. SUB-MILLIMETER WAVES

The fundamental electron cyclotron frequency in a 10 T magnetic field is about 300 GHz, corresponding to a wavelength of 1 mm and a photon energy of 0.00124 eV. The plasma frequency is about 100 GHz for $n_e = 10^{20}$ m^{-3} and 300 GHz for $n_e = 10^{21}$ m^{-3}. Therefore, the fundamental and its harmonics can be made to escape from a plasma with typical expected parameters. In a hot plasma, *e.g.*, 50 keV to 150 keV for a D-^{3}He reactor, many synchrotron harmonics radiate, and significant power[3] extends down to at least 100 μm.

Sub-mm waves can be guided tens of meters and around bends without excessive loss in hollow metal waveguides. The high electrical conductivity metals are best. The waves can be transmitted over longer distances by using parabolic mirrors and large diameter, evacuated tubes. Guiding removes the point of utilization from plasma bombardment, neutrons and other ionizing radiation, and fusion blanket constraints. The total waveguide aperture can be less than 10% of the reactor first wall area and still have good coupling out, if the wall is highly reflective.[2] When the application is incompatible with high vacuum, the waveguides must contain low-loss solid windows. Most substances have absorption bands in the wavelength range 0.1–1 mm and are unsatisfactory window materials. Crystal quartz transmits well at wavelengths above 0.25 mm, but it is essentially opaque in the range 4–100 μm. Any window will require cooling to remove heat from residual absorption.

The potential applications of sub-mm waves are heating and electrical power. Synchrotron emission is far too incoherent to be of much use for communications, and besides the atmosphere is strongly absorbing at the wavelengths in question. The photon energy is too low for industrially interesting photochemistry. Until tunable sub-mm lasers are developed, the fusion reactor might be a brighter source of tunable radiation for scientific research than those presently available.

Heating applications can be divided into low- and high-temperature. Industrial microwave heating to date has been almost exclusively low-temperature. The applications have been primarily drying, especially for food processing; curing polymers, including laminated wood products; and vulcanization of rubber.[4] Important to all these applications is the fact that the usual greater-than-10-cm wavelength industrial microwaves penetrate the product and heat more or less uniformly throughout its volume. Sub-mm waves are strongly absorbed in the first millimeter of most materials and produce surface heating; interior and shadowed regions are heated only indirectly. Therefore, fusion synchrotron radiation is not suitable for typical bulk drying and low-temperature applications.

Synchrotron radiation could be uniquely useful in the production of high-temperature heat. Conventional attempts to obtain high-temperature fluids from fusion reactor blankets are limited by severe materials problems in the blanket itself, where neutronic and radiation damage considerations dominate, and because a high-temperature heat exchanger is needed to isolate the radioactive primary fluid from the application process. A particular concern at high temperatures is diffusion of tritium into the heat transfer fluid and thence into the final product. Synchrotron radiation power can play the role of a nonradioactive primary fluid. (Redundant barrier windows should be provided.) The sub-mm waves are well absorbed in most molecular gases by a variety of processes, including rotational and pressure-induced rotational absorption.[5] The latter process is quadratic in pressure and is the main absorption process in this range by nonpolar molecules, such as CO_2 and N_2. For example, the absorption coefficient in CO_2 is 1×10^{-3} m^{-1} atm^{-2} at λ=1 mm and reaches a high of 6.5×10^{-3} m^{-1} atm^{-2} at λ=0.2 mm. Because the waves propagate along metal pipes until absorbed, a pressure of about 10 atm suffices for absorption in a 10 m piping run when a configuration similar to Fig. 13-2 is used. Steam absorbs more and nitrogen less. The modest gas pressure reduces mechanical stresses and facilitates very high-temperature operation, at the cost of a greater volumetric flow. Multiple waveguides and gas pipes are required in a full-power system. This concept can yield uniquely high-temperature, unactivated process heat from a nuclear system. A more near-term application is to use synchrotron radiation, from a D-T plasma adjusted for high-temperature operation, to superheat steam in an otherwise conventional power conversion cycle, and thereby to place less demand for high temperature on the blanket. It has also been recently proposed that synchrotron radiation be used to increase the ionization of the gas in an MHD generator, in order to convert the heat from a D-T fusion reactor more efficiently.[6]

Another way to produce high-temperature process heat is to use synchrotron radiation to heat either a fixed or fluidized bed of ceramic beads, through which a non-absorbing inert heat transfer gas like helium flows.[7] Because it may be difficult to find a ceramic sufficiently transparent to sub-mm waves to insure volumetric heating, fluidized beds, with

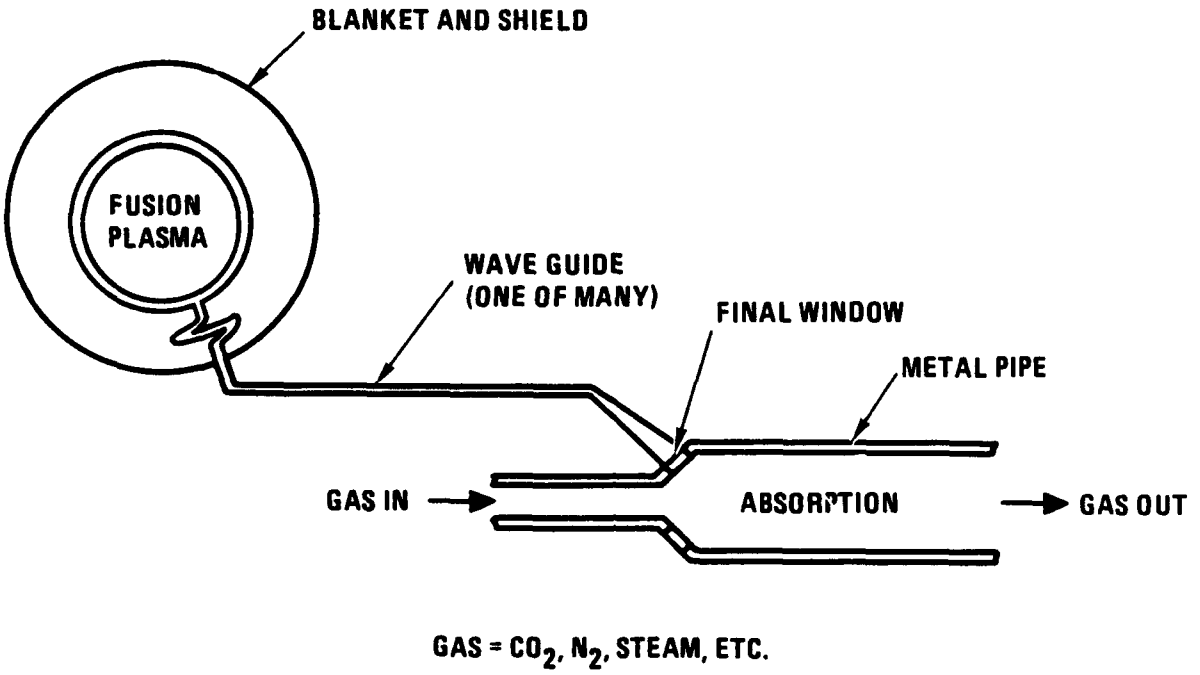


FIG. 13-2. Illustrating the concept of heating a molecular working gas inside a pipe by synchrotron radiation.

rapid bead mixing from surface to interior of the bed, are probably more practical than fixed beds.

It may also be possible to efficiently convert sub-mm waves directly into electric power. This would be particularly advantageous for a D-^{3}He reactor. The waves would be directed to a cavity or cavities lined with detector elements, which rectify the incident wave power to dc electricity. This is then inverted to line frequency ac. To keep costs low, the detectors should be mass-produced using integrated circuit techniques. The detectors need not be broad-band. Provided they reflect out-of-band energy, they can be tuned and

optimized for individual, overlapping frequency bands, and a wave will reflect around the cavity until it hits a detector tuned to its frequency.[2] Integrated antenna-diode detectors, called rectennas, have been built on a limited basis and have been studied for conversion of solar-generated microwave power from space, but these were for lower frequencies. The state of the art for efficient solid-state microwave detectors is about 1 mm wavelength, and this limit appears to be fundamental unless unforeseen new materials appear.[2] A field emission tip-to-plane gap has a rectifying characteristic, and, if made with a radius and spacing of about 0.1 μm, the complex impedance would be appropriate for sub-mm rectenna power converters.[2] Mass production of devices with 0.1 μm dimensions will require X-ray lithography or other advanced techniques. However, these are already being actively developed, driven by the demand for higher density integrated circuits, and it is a reasonable expectation that the technology will be available long before D-^{3}He reactors. Energy conversion efficiencies on the order of 70% are calculated.[2]

13.3. X-RAYS

We have not been able to identify any practical direct applications for the diffuse X-rays emitted by magnetic fusion plasmas, although several applications as research tools may exist. Most X-ray applications do not require large power, hence there is no incentive to move applications to a fusion reactor site. Large scale X-ray radiochemistry is ruled out, because radiochemical product yields are too low per unit energy invested to be of commercial interest. Direct conversion of X-rays to electrical power is possible in γ cells and in semiconductor junctions operating in the photovoltaic mode, but energy efficiencies are low.

The recent availability of high spectral brilliance ultraviolet and X-ray synchrotron radiation at dedicated electron storage ring facilities is rapidly revolutionizing many fields of scientific research,[8] and practical applications may follow. As illustrated in Fig. 13-3, photon energies currently available or projected range from 10 eV to 30 keV. Synchrotron

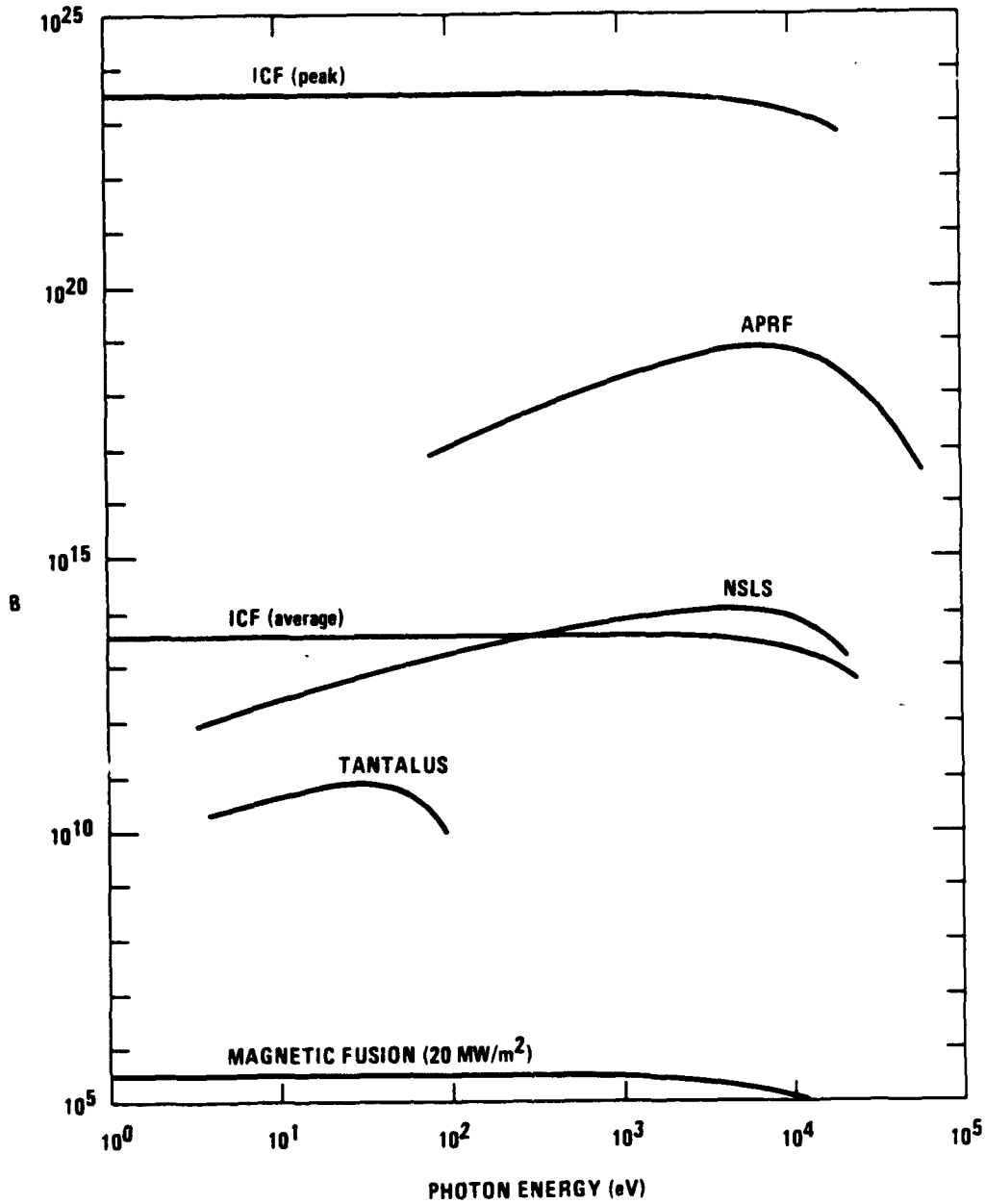


FIG. 13-3. Spectral brilliance B (photon/s) (mm·mrad)^2 (0.1% bandwidth) of some existing (Tantalus, National Synchrotron Light Source [NSLS]) and a proposed (Advanced Photon Research Facility [APRF]) synchrotron radiation sources. Also shown are estimated fusion bremsstrahlung at Z_{eff}=10 for ICF (peak and average) and magnetic fusion at 20 MW/m^2 wall power.

radiation from fusion plasmas is far below this range, but fusion bremsstrahlung, recombination and impurity line radiation cover it well. However, because of their extended size and low plasma density, the familiar magnetic fusion concepts fall many orders of magnitude short of the spectral brilliance (photons/sec [mm/·mrad]^2) of existing sources. On the other hand, laser-driven inertial fusion (ICF) plasmas are extremely small ($\sim$100 μm) and dense ($\sim 10^{32}$ m^{-3}), and the radiation at the reactor chamber wall is highly collimated. Time-average spectral brilliance for a 1 GW(t) D-T reactor can be 10^{12} to 10^{13} (photon/s) (mm mrad)^2 (0.1% bandwidth) at 20 keV from hydrogen bremsstrahlung alone, and this can be increased at least 10-fold by the addition of small amounts of higher-Z elements. The bremsstrahlung spectrum has an advantage in that its brilliance does not decrease at lower photon energies, whereas storage ring synchrotron brilliance increases roughly as the square root of the photon energy. Therefore, an ICF source has its greatest advantage for applications requiring 10 to 100 eV photons. Another advantage of the inertial fusion source is that, because its photons are released in 10–100 ps, the peak brilliance of a pulse greatly exceeds any likely to be achieved using storage ring techniques. The shortness of the pulse also allows time of flight separation from the direct neutron pulse. The neutrons will activate the experimental apparatus, which will have to be maintained remotely. The scientific demand for such a facility is not likely to exceed one or two units.

It may be possible and practical to efficiently convert X-ray power to narrow band continuum UV fluorescence radiation near 7.5 eV (165 nm) by absorbtion and excimer production in high-pressure Xenon gas. The processes are well understood for electron beam excitation of rare gases at high pressure[9]: The electrons ionize and excite gas atoms, these make excited Xe_2^* molecules (excimers) with a high yield and little energy loss, and almost all emission is due to transitions from the excimer to the repulsive ground state. This ground state does not accumulate, and the gas is highly transparent to this radiation. Up to 54% of the initial electron energy emerges from Xe in the narrow UV band, if the gas pressure is high enough so that the necessary three-body processes occur faster than competing processes, and if the excitation rate does not saturate the processes.[9] Both

conditions will be readily satisfied in a magnetic fusion blanket. Similar effects occur in the other rare gases, but Xe is a better absorber of X-rays. Below a few MeV, photons interact with matter mainly by Compton scattering, which yields a high energy electron and a lower energy photon, and at lower energy by photoionization, which ejects an electron originally bound in the atom with an energy somewhat less than that of the photon. Compton and photoionization cross sections vary approximately as (target Z)^{1} (photon energy)$^{-0.4}$ and (target Z)^{5} (photon energy)$^{-3}$, respectively. In either case, the target atom emits additional photons as its electrons cascade down to fill the vacancy left by the missing electron.[10] The end result in a thick target is a distribution of energetic electrons, which generate excimers as before. Just 4 atm-m density of Xe gas scatters 90% of 100 keV photons (characteristic of a D-^{3}He fusion plasma; this energy is still in the photoionization regime for Xe), so reasonable pressures suffice. The wall separating the plasma from the Xe must be thin and low-Z for maximum transparency. Beryllium and fiber-reinforced graphite honeycombs are candidate wall materials. Wall area mass must be 0.6 g/cm^{2} or less to have 90% or greater transparency to 100 keV photons. The Xe gas must circulate to remove the 50% or so of the incident energy that appears as heat in the gas and the first wall. The Xe must be kept pure, since most contaminants provide alternate excitation decay pathways and seriously reduce the desired fluorescence.[11] Ultraviolet production by X-rays has not received much quantitative study.

Large-scale applications of UV radiation are limited. Although 4 eV radiation from low-pressure mercury lamps, which convert about 55% of their electrical input into UV, has been cheap for decades, only modest applications have appeared.[12] Ultraviolet is germicidal, but chlorination is cheaper. In principle, it is possible to efficiently convert nearly monochromatic UV, such as that produced by the Xe blanket described above, directly to electricity by specially tailored photovoltaic cells. The cells would cover the outer wall of the Xe gas chamber, and the inner wall would have a reflective aluminum coating. No such UV photovoltaic device yet exists. Diamond, with an energy band gap of about 6 eV and carrier mobilities at least as high as silicon is a possible material, but there is no guarantee

that the requisite technologies could be developed nor that the final product would be operable in the reactor environment. Even with the virtually monochromatic incident light, photovoltaic energy conversion efficiency is unlikely to exceed 80%, and there is also loss in the dc-to-ac converter. Therefore, the overall efficiency, from plasma bremsstrahlung to electric output will be less than 40%. Although this is no better than thermal conversion, there would be an economic advantage if the non-moving components of the direct converter could be made reliable and long-lived at lower cost. It was suggested that knock-on recoil ions from D-T neutrons be used to generate excimer UV radiation, to be converted by specially tailored photovoltaics.[6] The neutron knock-on involves additional losses, relative to the photon processes discussed above, but it offers a possible alternate way to harness the large neutron power fraction in D-T fusion. The long neutron stopping distance in gas and neutron absorbtion in structural materials are concerns for this application.

X-rays can also be converted directly into visible photons, but it might be difficult to obtain interesting energy efficiencies. The very strong resonant transitions of cesium vapor at 852 nm and 894 nm are conveniently matched to existing silicon photovoltaics, but these transitions are to a ground state, and the visible radiation is strongly reabsorbed and dissipated when the vapor density is high enough to absorb most of the X-rays. The same situation holds for mercury vapor and its efficient resonant fluorescence at 254 nm in the UV. Conventional scintillator crystals, such as CsI(Tl), have quantum efficiencies of virtually 100% for 100 keV photons, but energy conversion never exceeds 15% due to competitive nonradiative pathways.[10] Doped ZnS and ZnCdS CRT phosphors used in color television convert between 15% and 22% of the received electron energy into light, but because these cannot be made into large crystals, the visible light is scattered and absorbed when such phosphors are made thick enough to absorb X-rays. X-ray fluoroscope screens are much like CRT phosphors and also suffer from low efficiencies. Although all these existing converters are optimized differently than a fusion X-ray power converter would be, the absence of higher efficiencies in these mature technologies is not encouraging. We did not have the expertise to evaluate conceptual new phosphors from scratch, but rare earth

ions in solid crystals appear to have interesting properties. Pr^{+3} is somewhat analogous to rare gas excimers: visible light excitation is efficiently channeled (quantum efficiency near 100%) into a few closely spaced 4f states, which decay to a vacant state a few kT above the ground state. There is almost no self-absorbtion, and optically pumped lasing of $PrCl^3$ and $PrBr_3$ at low threshold has been reported.[13] The Ce^{+3} ion has a lone 4f electron and an empty 5d shell. The lowest excited states all place the electron in the 5d shell, and they decay nonradiatively to the lowest 5d state, which decays radiatively back to the split 4d ground state. (The radiation is in two lines in the near UV, at approximately 300 nm and 320 nm.) The system might be sufficiently transparent to its own radiation.[14] Because X-rays initially excite and ionize mostly inner electrons, there are many ways that the primary energy can be lost to phonons (heat), rather than ending up in the few-V excitations desired here. We did not find data on X-ray-excited visible fluorescence of rare earth ions to evaluate the efficiency of the conversions outlined above.

Assuming for the moment that a large visible light output could be obtained from a fusion reactor, applications are still limited. Consider a central fusion "sun lamp," consisting of a large tower-mounted reflector system above a fusion reactor with a 1 GW luminous output. Ideally this gigawatt might illuminate 10 (km^2) of land at 100 W/m^2. With an optimized spectrum, this might be enough in some cases to improve crop yields at high latitudes or permit night growth. A power cost of $0.05 per luminous kW-h translates into $440,000 per year per hectare, which far exceeds the full market value of the produce, even with multiple crops per year. It is difficult to beat the sun for widespread, diffuse power when no collecting and storage structures are required. However, electricity could be produced from visible light by the addition of photovoltaic cells optimized for monochromatic light, as already discussed above for UV light. The potential overall efficiency must be high to justify the phosphor development that would be required. Compared with the rare gas excimer system, solid phosphors, if they could face the plasma on the first wall, can convert both incident particle and X-ray energy into photons.

13.4. EVALUATION

Synchrotron radiation has unique advantages for the production of very high temperature process heat if the reactor is optimized for high-temperature (~30–50 keV) plasma operation. Extraction of the synchrotron radiation minimally perturbs other aspects of the fusion reactor. It is decoupled from ionizing radiation. Sub-mm power is efficiently absorbed in gaseous working fluids at modest pressures. This is probably the highest temperature and most attractive source of process heat from a nuclear system. Only the microwave windows appear to present possible development problems. In D-T reactors the process heat would only be a minority product or would be used for a topping cycle. In D-^{3}He reactors it could be over half the gross fusion power. With some development, the synchrotron radiation can also be directly converted to electricity. This is an important option for D-^{3}He reactors, but it would have little impact on mainline D-T reactor development.

X-radiation is more difficult to harness, other than as heat. The radiation pulse from ICF microexplosions has uniquely high spectral brilliance, which undoubtedly will be useful for research. It is already a major mission of the ICF program to produce pulsed radiation for military development applications. X-rays can be converted into UV at about 50% energy efficiency, but there are no practical direct uses for large amounts of UV. Monochromatic UV or visible light could be converted to electricity by spectrally optimized photovoltaic cells. Overall efficiency, from X-ray to electricity, is not likely to be any higher than for conventional thermal cycles, but the direct conversion might be advantageous if its conceptual simplicity led to a more reliable, lower cost system. No efficient method to convert X-rays to visible light was identified in the present work.

References

[1] L.J. Wittenberg, J.F. Santarius and G. Kulcinski, "Lunar Source of ^{3}He for Commercial Fusion Power," **Fusion Technology**, Vol. 10, No. 2, Sept. 1986.

[2] B.G. Logan, private communication (1986).

[3] J.L. Hirschfield, D.E. Baldwin and S.C. Brown, **Phys. Fluids 4** (1961) 198.

[4] A.C. Metaxas and R.J. Meredith, **Industrial Microwave Heating**, (Peter Peregrinus, Ltd., London, 1983).

[5] G.W. Chantry, **Submillimetre Spectroscopy**, (Academic Press, London, 1971).

[6] L.J. Perkins, B.G. Logan, C.D. Henning and G.H. Miley, "A Rationale for Novel Neutron Energy Conversion Schemes for D-T Fusion Reactors," (to be published) Lawrence Livermore National Laboratory (1986).

[7] G.E. Besenbruch, private communication (1986).

[8] P. Eisenberger, **Science**, (1986) 687.

[9] D.C. Lorents, "The Physics of Electron Beam Excited Rare Gases at High Densities," **Physica, 82C** (1976) 19.

[10] J.B. Birks, **The Theory and Practice of Scintillation Counting**, (Pergamon Press, Oxford, 1964)

[11] T.E. Stewart, *et al.*, "Proton Excitation of Continuous Emission in the Nobel Gases," **J. Opt. Soc. Am. 60** (1970) 1290.

[12] L.R. Koller, **Ultraviolet Radiation**, (Wiley, New York, 1965).

14. CONCLUSIONS AND RECOMMENDATIONS

In the Fusion Applications and Market Evaluation (FAME) Study we have surveyed the many potential applications of fusion energy. We reviewed previous fusion applications work, searched for new applications ideas, and evaluated the various concepts for technical soundness and market size. A number of useful conclusions emerged. They are listed below, followed by recommendations for future work on fusion applications and for the fusion program in general.

14.1. CONCLUSIONS

The conclusions reached for each of the potential applications described in the chapters above are summarized below.

14.1.1. Electricity Production

The revenue from the sale of electricity can be expected to dominate over that from other products. The thrust of the reactor studies program has therefore generally been in the right direction. Even with the unlikely assumption of zero growth, there will be a substantial demand for new capacity just to replace those power plants being retired. However, one would expect significant growth in electricity demand in less-developed countries because historically increases in living standard have been accompanied by increases in per capita electricity consumption. Even with fixed per capita consumption, population growth can be expected to fuel demand for decades to come. In developed countries, efforts to curb consumption through conservation measures are now at a point of diminishing returns. Therefore, even with fixed population, electricity growth will continue as indicated by the trend toward an increasing fraction of total energy consumption being in the form of electricity.

Many fusion reactor studies have shown that fusion has the potential to be roughly competitive with other sources of electricity. As we proceed toward commercialization, it will be necessary to effect cost reductions to offset the inevitable cost escalation that occurs as design details are filled in. The two possible means for reducing costs are higher power density fusion cores and making reactors inherently safe so that non-nuclear costing can be used throughout. In the first, if the reactor core cost is reduced, then cost estimates become more reliable because they are dominated by well-known balance-of-plant elements. In the second, most of the cost reductions tend to occur in the indirect costs because all of the licensing and qualifications testing costs are eliminated or greatly reduced.

Capital charges dominate electricity costs in fusion plants more than fission or fossil, both of with have substantial fuel costs. Once fusion is developed, its attractiveness will in part be dependent on the financial climate, being more attractive when there is ample capital money available at low interest rates. Because electricity costs are essentially fixed by capital payback charges, a fusion-dominated grid should have a stabilizing effect on the economy as a whole.

14.1.2. Nuclear Fuels

Fusion reactors can be prolific fissile fuel and tritium producers. One fusion reactor can produce enough fissile fuel to service up to 20 fission reactors. Holdup time for breeding is fairly short so that fissile fuel can be sold for income typically within a year. Prices are very competitive whether or not electricity is sold as well. When electricity is generated, the added revenue from fissile fuel substantially supplements income from electricity sales. When it is not, capital costs are reduced by eliminating the electric generating plant.

The high energy multiplication from fissile fuel breeding blankets may permit the use of confinement methods which would otherwise have excess recirculating power as a pure fusion reactor. Also, the total fusion power is reduced for a given total thermal power, thereby reducing the cost uncertainty contributed by the fusion reactor core.

The fissile blanket is subcritical during operation, thereby eliminating the need for control rods. However, loss of flow or coolant incidents can still have serious consequences because of the high levels of afterheat. It is this safety issue, along with the issue of handling a large inventory of highly radioactive materials, that has caused hybrids, and nuclear power in general, to be less favored in recent years. As the growing world demand for electricity causes a resurgence in nuclear power demand, and this in turn increases demand for nuclear fuel, interest in the fusion breeder should revive.

14.1.3. Materials for Radiation Processing

The production of ^{60}Co turned out to be one of the most interesting and marketable near-term products from fusion reactors. Market potential hinges on the growing food irradiation industry. Although not very visible in this country, it is rapidly growing elsewhere, particularly in the Third World. When combined with other products like electricity and fissile fuel, one obtains perhaps the most economically desirable fusion plant system. A problem may occur because production rates can be so high that excess availability will depress prices. Only a few fusion reactors would be needed to supply all of the world's ^{60}Co needs. Nevertheless, it could be a useful product from near-term experimental reactors.

Another radioisotope for food preservation is ^{56}Mn. Because of its short (2.6 h) half-life, it would be limited to use at the reactor. Its main advantage is that its radioactivity decays quickly after shutdown.

14.1.4. Other Isotope Production

Many other products can be bred using fusion neutrons. Among these are ^{153}Gd, used in medical scanning, ^{32}P and ^{33}P, which are beta emitters that are useful in the treatment of leukemia, stable osmium and rhodium for alloying, and, of course, gold, the alchemists dream. Whether or not fusion reactors can produce these economically will require a far more extensive analysis than was possible here. It does appear, however, that none of these products would be economical without the co-production of electricity.

14.1.5. Synthetic Fuels

The thermal energy from fusion reactor blankets can be partitioned to generate electricity, or make synthetic fuels and process heat, or a combination. Since a large fraction of society's energy consumption is in the form of transportable liquid fuels, the market potential for synthetic fuels is vast. A key issue in synthetic fuel production is the efficiency of energy utilization and recovery of waste heat. If one can eventually achieve 80% energy utilization, synthetic fuels will become competitive with fossil fuels, even at present low costs of these fuels. Current synfuel generation methods have low-to-moderate efficiencies. Low-temperature electrolysis has an efficiency of about 32%. Fusion radiolysis has only 32–40% projected efficiency including using reject energy for low-temperature electrolysis. High-temperature electrolysis claims an efficiency of up to 70%, but this requires temperatures of 1800°C and 60% power cycle efficiency. The GA sulfur-iodine cycle has 43% efficiency. Other cycles are in the same range. With these efficiencies and the capital costs projected for fusion, fusion synfuel will not be able to compete until fossil fuels escalate in cost to ~$12–15 GJ.

14.1.6. Process Heat

The process heat from a fusion plant could be very high quality with no pollution, and therefore one could sell it at a premium price. With this caveat, process heat production is viable if electricity is produced as well. Typically, only a small portion of the blanket thermal energy would be devoted to process heat because no single site would be expected to be able to utilize more than a few hundred megawatts.

If fusion plants could more than compete with other electricity generating plants, then a small surcharge in electricity would be possible, and this would be so highly leveraged that it could lower the price charged for process heat to very competitive levels.

14.1.7. Defense Applications

A limited but useful fusion application is for defense activities. Surplus tritium can be used to replenish the constantly decaying supply in the nation's nuclear weapons. Magnetic

and especially inertial confinement reactors could possibly be used for weapons' effects testing and component hardening. In addition, inertial confinement fusion experiments could be used to study nuclear weapons physics.

14.1.8. Nuclear Waste Burning

Because fission waste burning has long been viewed as a possible valuable application of fusion, a considerable effort was made in this study to examine it in detail. It was found that, while the most hazardous long-lived fission wastes, ^{129}I and ^{99}Tc, can be rapidly transmuted in fusion reactors with moderate neutron wall loadings, the most hazardous fission products, ^{90}Sr and ^{137}Cs cannot, nor can the long-lived actinide wastes. The hazards associated with processing large inventories of these intensely radioactive wastes renders them hard to compete with simple deep burial.

14.1.9. Space Power and Propulsion

While fusion has the potential of very high specific impulses, too little is known at this point about an equally important figure of merit: the ratio of thrust to total system weight. The total system weight includes not only the fusion power core but those components needed to operate the core. These include power supplies, drivers (for ICF), heating systems, current drives, energy storage, fueling systems, shielding, cryogenic systems, vacuum systems, and heat rejection. As much if not more effort is needed to reduce the mass and bulk of these components as that of the core itself. Given the potential for very high specific impulse and the benefit this would have for deep space missions, more detailed design studies to assess the total system characteristics are needed.

The total system weight is closely tied to the choice of fusion fuel. Aneutronic fuels such as D-^{3}He are appealing because essentially all of the reaction products are charged particles, available for thrust or direct conversion to electricity. Yet the fact remains that, for a given plasma pressure, the charged particle power is at best 1/10 that of D-T and the total power is 1/50 as much. Because the purpose of supplying all of that expensive and weighty hardware is to create a desired plasma pressure, the fusion power per unit

pressure should be as high as possible. While D-T does produce neutrons, this is not totally negative because that thermal energy can supply heat and electricity to the space ship.

One constraint that is relaxed for fusion power in space is cost: mills per kW-hr are not the dominant criteria as on earth. Weight and mass dominate as figures of merit. Therefore, one would probably choose very expensive, high-temperature Brayton cycles, for example, over Rankine cycles because of their compactness.

14.1.10. Electromagnetic Waves

Synchrotron radiation has unique advantages for the production of very high temperature process heat. The power can be transmitted to points of utilization some distance from the reactor boundary. Sub-mm power is efficiently absorbed in gaseous working fluids at modest pressures, making it an attractive source of process heat. In D-T reactors, it would be only a small fraction of the fusion power; however, in D-^{3}He reactors, it could be as much as half the power. X-rays from fusion reactors are more difficult to harness, other than as heat. However, the high spectral brilliance of X-rays from ICF explosions may provide a useful research tool.

14.2. RECOMMENDATIONS

The most marketable products from fusion reactors are (1) electricity, (2) fissile fuel for fission reactor fuel and tritium for defense applications, (3) ^{60}Co, and (4) small amounts of high-grade nonpolluting process heat. It would be well worth performing a reactor conceptual design study in which all of these products are produced simultaneously. The study should be in sufficient detail to explore critical engineering and economic issues. The suggested confinement concept would be either a high-beta tokamak with 5–10 MW/m^2 neutron wall loading, or a reversed-field pinch with 10–15 MW/m^2. Emphasis should be placed on minimizing the volatile and radioactive product inventories and out-of-reactor processing steps. Continuous versus batch processing issues could also be addressed.

The long-term potential for fusion to provide a source of power and thrust for deep space missions is exciting. Further work is needed to perform design studies of alternate fusion concepts for these applications. If the total system mass can be kept to reasonable levels, possibly by using advanced fusion fuels and advanced confinement concepts, these applications could provide an attractive long-term mission for fusion.