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ABSTRACT

Production of tritium for weapons programs and for fusion power generation is an important long-range national requirement; in this report we examine the suitability of low-pellet-gain (1-20) Inertial Confinement Fusion (ICF) for that purpose. We assess the technical feasibility and costs of facilities that could be demonstrated near the end of this century. Consideration of ICF for tritium breeding is motivated by (1) the advantages of high-energy (14-MeV) neutron sources over the conventional fission neutron sources, (2) the advantages of relaxed ICF-pellet performance requirements for this application relative to other longer range applications such as commercial power generation, and (3) the potential for incorporating in a production facility the capability to conduct tests of military significance. In addition, the R and D programs leading to a tritium production facility would significantly advance the development of ICF for more demanding applications such as commercial power generation.

Three possible ICF drivers are considered: CO_2 lasers, KrF lasers, and heavy-ion accelerators. Characteristics of reactors and breeding blankets are identified and included in the analysis.

Results indicate that the use of 14-MeV neutrons from ICF pellets for breeding tritium seems technically feasible and economically attractive. At production rates of 6 to 12 kg/yr, projected costs, dependent on ICF technology, vary between $7 and $20 thousand per gram. These costs are less than or competitive withestimated costs for proposed replacement production facilities based on fission technology.

Finally, we outline the essential elements of a program required to develop a demonstration ICF tritium breeding facility by the end of this century.

EXECUTIVE SUMMARY

This study evaluated the technical and engineering feasibility of several combinations of Inertial Confinement Fusion (ICF) drivers and reactor concepts and determined levelized life-cycle unit costs for tritium production. The ICF drivers considered were CO_2 and KrF lasers and heavy-ion-beam accelerators. The reactor cavity concept adopted uses a thin film of flowing liquid metal to protect interior walls from fusion-pellet emissions and is compatible with each of the drivers studied. We evaluated several reactor breeding blankets for tritium production and fusion energy amplification, and we found that the optimum breeding blankets consist of ^{6}Li and ^{238}U. An attractive nonfissioning breeding blanket consists of natural lithium and beryllium.

The results of this preliminary study strongly support the exploitation of relatively near-term ICF technology for the production of tritium and other special nuclear materials. The current and anticipated pace of the ICF program is such that driver and pellet physics will achieve the required levels of development for this application near the end of this century. Thus, although ICF is not a contender for a near-term replacement production reactor, it merits serious consideration for use early in the 21st century. At that time greatly increased requirements for reactor products may exist.

Although incomplete, the results of this study indicate significant advantages of ICF tritium production reactors compared with facilities based on fission reactor technology. These advantages include

- lower cost;
- smaller, economical systems;
- elimination of potential for nuclear criticality accidents;
- design concepts for tritium production that are completely free of fissile materials and fission products; and
- multipurpose facilities that include production reactors and experimental facilities for support of the National Security Programs, the fusion energy program, and other potential applications of ICF.

Tritium production costs were evaluated for driver/reactor systems with energy-conversion equipment that produces sufficient electric power to operate the plant, as well as for systems without cogeneration of electric power. Some general conclusions from these analyses are as follows:

- Costs of tritium production with ICF are competitive with costs of production with fission reactors. Moreover, although the effects of economies of scale result in decreasing production costs with increasing capacity, ICF production facilities are competitive with fission reactor systems at production rates lower than those postulated for fission reactor systems.
- The requirements for driver pulse energy and fusion-pellet gain are modest relative to anticipated goals and achievements of the ICF program during the next two decades. Driver pulse energies of ~ 1 MJ and pellet gains of <5 make these systems appealing.
- Cogeneration of electricity for energy self-sufficiency reduces tritium production costs appreciably (as much as a factor of 2) for systems with low-efficiency drivers (KrF lasers). Energy self-sufficiency is not very important for systems with high-efficiency drivers (heavy-ion-beam accelerators). For systems with very low-gain pellets and/or low-efficiency drivers, energy self-sufficiency can be achieved only if the blanket significantly amplifies the neutron energy. In general, this requires a fissionable neutron multiplier such as ^{238}U. A heavy-ion-beam-driven system with energy self-sufficiency and a nonfissioning neutron mutiplier requires a pellet gain of at least 10.
- Simultaneous production of tritium and a fissile material may be an attractive possibility. For example, a blanket containing ^{6}Li-^{238}UC slurry produces 2.4 atoms of tritium and 0.5 atoms of ^{239}Pu per fusion neutron. Moreover, the plutonium could be almost pure because of the possibility of continuous processing of the flowing fertile materials. No credit was allowed for plutonium production in the economic analyses.
- The cost of producing tritium is insensitive to the change in the tritium breeding ratio, R_b, for values between 2 and 2.4 for a fixed production rate. This insensitivity is due to the effects of the economies of scale assumed for the drivers. However, for a given driver, tritium production costs are proportional to the inverse of $R_b - 1$.
- Tritium production costs vary by as much as 50% for the extremes of pellet gain assumed for systems without self-sufficient energy generation but by only about 25% for energy-self-sufficient systems.

I. MOTIVATION

A. Introduction

Tritium is produced for weapons and other programs by nuclear transmutation of lithium using fission reactor neutron sources. Because tritium has a relatively short half-life (12.3 yr), it must be continuously produced to maintain the existing weapons stockpile. Also, it will be required in significant quantities to supply experimental fusion programs and the initial fuel charges for magnetic-fusion commercial power plants.

We present the results of preliminary assessments of the potential of near-term ICF facilities for the production of tritium and special nuclear materials. Section I.B details advantages of fusion neutrons over fission-spectrum neutrons, and Sec. I.C presents a conceptual design and discussion of the advantages of an ICF tritium breeder reactor. Based on this design and cost scaling relationships[1-3] of three driver systems (CO_2 lasers, KrF lasers, and heavy-ion accelerators), a preliminary economic assessment of tritium breeding has been completed. The analysis is described in Sec. I.D. The ICF target gains assumed in the calculations are modest, from 1 to 20. The results of analyses indicate significantly lower ICF costs compared with fission reactor costs for production rates of approximately 10 kg/yr. The results are particularly interesting because the designs of the low-gain targets required are quite simple structurally. This simplicity greatly reduces the cost of mass producing targets, as compared with more demanding applications, and in addition, values of the design parameters of these targets are in ranges of credible theoretical models.

In Sec. II we describe the program elements and assess the present status of ICF target physics (in particular, verification of low-gain target performance), the sources of focusable energy with ICF drivers having the requisite repetition-rate capabilities to satisfy reactor operating requirements, and the general characterization of an ICF tritium production facility.

Discussions in Secs. I and II are integrated in Sec. III into a recommended program plan for the development of a prototype ICF reactor with tritium production as the main mission. The anticipated time scale for such a project eliminates it as a candidate for a near-term replacement facility but ICF may be appropriate for consideration near the year 2000.

B. Advantages of Fusion Neutrons for Producing Tritium

Natural lithium consists of approximately 93% ^{7}Li and 7% ^{6}Li. The nuclear transmutation cross section of ^{7}Li for the tritium production reaction is significant for high neutron energies, but at ~4.5 MeV, it decreases rapidly with decreasing energy. Thus, fission neutrons (~1 MeV) are much less effective for breeding tritium in this medium than are 14-MeV fusion neutrons.

The nuclear cross section of ^{6}Li for the tritium production reaction increases rapidly with decreasing neutron energy between 10 MeV and 1 keV, and it becomes proportional to the inverse of neutron velocity at lower energies. Therefore, the characteristics of ^{6}Li and ^{7}Li are complementary. This circumstance, combined with the fact that the reaction of a 14-MeV neutron with ^{7}Li produces not only a triton but also a lower energy neutron (which can react with ^{6}Li), makes it possible to obtain substantial tritium breeding rates using natural lithium alone.

Tritium breeding rates can be enhanced significantly by including neutron-multiplying materials in breeding regions. The 14-MeV fusion neutrons can be increased several-fold in neutron-multiplying materials without concern for nuclear criticality, and the resulting lower energy neutrons can be used in ^{6}Li reactions.

Although tritium production is the primary concern in this initial investigation, ICF neutron sources could be used to produce fissile materials including ^{239}Pu from ^{238}U and ^{233}U from ^{232}Th.

C. Advantages of an ICF Facility for Manufacture of Reactor Products

Reactor products (special nuclear materials and tritium) are produced by nuclear transmutation using fission reactor neutron sources. Based on preliminary assessments, ICF seems to offer several advantages for this purpose:

- Tritium production using ICF is cheaper than that using fission reactors.
- ICF requires no enriched fissile fuel, so no possibility of a nuclear criticality accident exists.
- In conceptual ICF reactor designs, tritium will be bred in flowing-liquid-lithium blankets that surround the neutron source (the lithium also serves as the reactor coolant); therefore, it can be extracted continuously. In fission reactors, which use solid lithium compounds, tritium can be extracted only periodically after fuel shuffling.
- Simultaneous breeding of fissile materials and tritium could be accomplished by flowing either molten salts or slurries of lithium and either ^{238}U or ^{232}Th through ICF reactor blankets. Continuous processing and extraction of all product species would result in almost pure ^{239}Pu or ^{233}U because of the very short time the fissile breeding material is in the neutron radiation field. Simultaneous breeding of fissile materials and tritium could also be accomplished by including the heavy metal in liquid-lithium-cooled blankets as metal-clad fuel rods.
- Early use of ICF for tritium production would accelerate ICF development for more demanding applications.
- ICF drivers can be designed with pulse-rate

capabilities exceeding the requirements for tritium production. The excess driver capability could be utilized by switching beams to an adjacent test chamber for various experiments including weapons effects simulations and research leading to eventual commercialization of ICF.

ICF is not the only potential source of very high energy neutrons for manufacture of reactor products. The most obvious alternate contender is magnetic fusion. However, ICF has the following advantages over magnetic fusion:

- One of the chief advantages of ICF is the physical separation of the driver from the reactor vessel. This means that an ICF reactor vessel may have a relatively small containment volume; that its operation, maintenance and repair will be relatively simple; and that the most expensive components will not be subject to neutron bombardment and activation.[4]
- A great amount of flexibility is permitted in the design of ICF reactors for special purposes without, for example, the need to accommodate large magnet systems and without restrictions imposed by strong magnetic fields.
- ICF production facilities can be designed in smaller sizes and for lower power levels than magnetic-fusion systems can and thus are more appropriate for this application.

Another potential contender for breeding special nuclear materials is the electronuclear breeder (ENB). Although ENB appears practical, projected tritium production costs are considerably higher for it than costs for ICF (see Sec. I.D).

D. Preliminary Economics

1. Production Rates. One 14-MeV neutron and a 3.5-MeV alpha particle are released by each D-T fusion event. The resulting neutron population can be increased in a suitable neutron multiplier and will interact with lithium to produce tritium at a rate quantified as the tritium breeding ratio, R_b (the number of tritons produced per triton consumed in the fusion reaction). The net amount of tritium produced during a year of continuous facility operation, P_T (g/yr), is

$$P_T = 55.65(R_b - 1)fY \quad , \tag{1}$$

TABLE I. Multiplication of 14-MeV Neutrons

Material	Be	Pb	^{232}Th	^{238}U
Neutron Multiplication	3.03	1.87	2.52	4.12

where f is the frequency of pellet microexplosions (Hz) and Y is the fusion energy release per microexplosion (MJ).

Table I lists multiplications of 14-MeV neutrons achievable in several common neutron multipliers. In the absence of neutron losses caused by parasitic capture in structures and blanket materials and by leakage from the reactor, neutron multiplication is a measure of the potential breeding ratio.

Figure 1 indicates the dependence of P_T on the fusion power output, fY, for a range of values of R_b.

2. Fusion-Pellet Performance. The yield, Y, of an ICF fusion pellet is related to the energy of the driver pulse, E, by the gain function, G(E):

$$Y = E\,G(E) \quad . \tag{2}$$

Gain functions are obtained from numerical simulations of pellet implosions and burns. Curves of pellet gain as functions of driver pulse energy have been published for short-wavelength lasers (~0.25 μm) and heavy-ion-beam accelerator drivers.[5] Both single-shell and double-shell pellets have been studied. Because of potential difficulties associated with mass production of double-shell pellets, we have considered only single-shell pellets. For laser drivers, pellet gain depends on beam uniformity and wavelength; for accelerator drivers, pellet gain depends on focusability and ion kinetic energy. For cases of practical interest, the gain for single-shell pellets is calculated to lie between two curves that can be parameterized by

$$G(E) = kE^{\beta} \quad , \tag{3}$$

where, for the range of driver pulse energy $0.5 < E < 5$ MJ, the values $k = 21.6$ and $\beta = 0.78$ approximate an upper bound on gain estimates and the values $k = 8.4$ and $\beta = 1.03$ provide a lower bound.

Long-wavelength lasers such as CO_2 (10.6-μm wavelength) are considered less effective for driving fusion targets than short-wavelength lasers are, but they

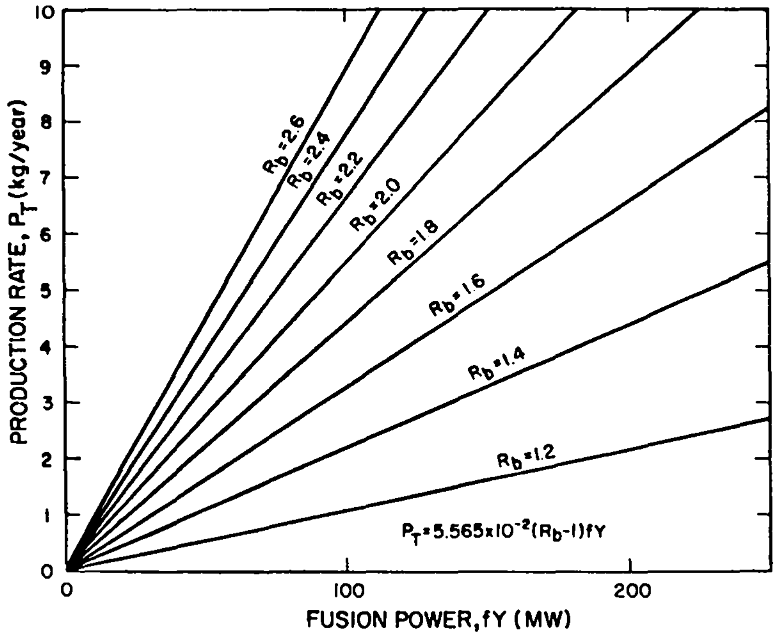

Fig. 1. Tritium production rate.

have the advantage of higher efficiency and lower capital cost. The effectiveness of CO_2 lasers for driving fusion pellets of innovative design is being studied at Los Alamos but has not yet been determined unequivocally. For this study, we assumed that CO_2-laser energy pulses are only half as effective as short-wavelength laser pulses for driving fusion pellets; therefore, for CO_2-laser drivers, Eqs. (2) and (3) become

$$Y(E) = k(\delta E)^{\beta+1} \quad , \tag{2'}$$

where $\delta = 0.5$ and k and β have the values given above. Dividing Eq. (2′) by E results in the following expression for pellet gain:

$$G(E) = (0.5)^{\beta+1}\, kE^{\beta} \quad . \tag{3'}$$

Thus, for a given pulse energy E, the pellet gain for a CO_2 laser is ~25% of the gain for a short-wavelength laser.

3. **ICF Facility Costs.** In this preliminary study of the economics of tritium production using ICF, we used several recently published cost estimates of ICF systems. Independently published estimates by the Electric Power Research Institute (EPRI),[1] the University of Wisconsin,[2] AVCO,[6] and the Los Alamos National Laboratory[7] agreed reasonably well. The capital costs of various ICF drivers have been estimated in terms of the pulse energy, E(MJ), delivered to the pellet. These estimates, adjusted to 1981 dollars, are

CO_2-laser driver	\$1.9 × 10^8 $E^{0.8}$
KrF-laser driver	\$2.5 × 10^8 $E^{0.8}$
Heavy-ion driver	\$5.0 × 10^8 $E^{0.4}$.

Detailed cost estimates of CO_2-laser drivers, including the costs of circulating and cooling the lasing medium, were carried out at Los Alamos National Laboratory;[7] these have quantified the dependence of capital cost on pulse repetition frequency. For the chosen ranges of pulse energy and pulse repetition frequency, this dependence can be approximated by an exponential function included in Eq. (4).

To the capital costs of the driver must be added the capital costs of (1) the reactor cavity and blanket (\$200M), (2) liquid-metal pumps (\$56M), (3) the pellet factory (\$200M), and (4) miscellaneous pipes, dump tanks, and cleanup systems (\$40M), or approximately \$500M total.[2] Finally, we add the capital cost of turbines and generators for an ICF facility intended to cogenerate enough energy for self-sufficiency.

These estimates are summarized by the following relationship:

$$C = a + bE^{\alpha}e^{\gamma f} + d\,\frac{fE}{\eta_d} \quad , \tag{4}$$

TABLE II. Driver Cost Characteristics

	a	b	α	η_d
CO_2 laser	5×10^8	1.9×10^8	0.8	0.10
KrF laser	5×10^8	2.5×10^8	0.8	0.04
HI beam	5×10^8	5.0×10^8	0.4	0.25

where C is the total capital cost, η_d is the driver efficiency, d = \$0.50/W, and the exponent, γ, is 0.026. The coefficients a and b, the economy-of-scale exponent, α, and η_d are listed in Table II for CO_2-laser, KrF-laser, and heavy-ion-beam drivers.

In the absence of electricity cogeneration, we have d = 0, but the operation of the facility requires an annual "fuel" expenditure, F(\$/yr), of

$$F = 8.78 \times 10^3 A_f C_e \frac{fE}{\eta_d} \quad , \qquad (5)$$

where C_e is the electricity cost in mil/kWh and A_f is the fraction of the year during which the plant is operational (availability factor).

The capital and "fuel" costs must be supplemented with an annual operating and maintenance cost, M(\$/yr); it is approximated with[2,3]

$$M = 1.6 \times 10^7 + 0.02C \quad . \qquad (6)$$

4. Unit Production Cost. The levelized life-cycle cost is commonly used in economic comparisons of unit production costs. It is the ratio of annual expenditures to the annual output averaged over the lifetime, L, of the plant. The levelized life-cycle cost, U, is given approximately by the analytic expression

$$U = \frac{1}{A_f P_T} \left(\frac{B}{XQ} C + F + M \right) \quad , \qquad (7)$$

where

$$B = \sum_{j=1}^{X} (1+i)^{X-j} \quad ,$$

$$Q = (1+z)^X \sum_{j=1}^{L} \left(\frac{1+z}{1+i} \right)^j \quad ,$$

X = construction period in years,

i = annual interest rate, and

z = inflation rate.

In the derivation we assumed 100% debt financing and omitted taxes and insurance premiums as befits a government enterprise.

The unit production cost, U (the levelized life-cycle cost), given by Eq. (7), becomes a function of E and f when Eqs. (1)-(6) are substituted into Eq. (7). Figure 2 illustrates the dependence of the cost on these two variables in the ranges of $1 \leqslant E \leqslant 5$ MJ and $5 \leqslant f \leqslant 50$ Hz. These results show that the unit cost is a strong function of the frequency for frequencies less than ~10 Hz but that the cost becomes nearly independent of the frequency beyond ~20 Hz. The input for these calculations is listed in Table III.

To facilitate cost comparisons with alternate production facilities, we plotted the unit costs in Figs. 3-5 as functions of the production rate for CO_2, KrF, and heavy-ion drivers, respectively, with and without electricity cogeneration for energy self-sufficiency. The results show a potential cost advantage of the ICF process over ENB and over the high-temperature gas-cooled replacement production reactor (HTGR). These comparisons do not imply that ICF and alternate facilities may be available at the same time; HTGR can certainly be available at least a decade sooner than any ICF facility can.

To assess the effect of driver efficiency on unit production costs, we made calculations in which only the laser efficiency was varied. The costs do not change by more than 10% when the laser efficiency changes by a factor of 2 with cogeneration of electricity.

II. PROGRAM ELEMENTS

A. Target Requirements and Development

Because tritium can be produced economically with relatively low-gain pellets, we anticipate that the pellets required will be simple and appropriate for mass production. Figs. 3-5 show that ICF tritium production costs are competitive with alternate production methods even at the lower boundaries of pellet-gain estimates (corresponding to the upper boundaries of cost estimates). In this connection we evaluated requirements for pulse

Fig. 2. Unit production cost for a heavy-ion-beam driver.

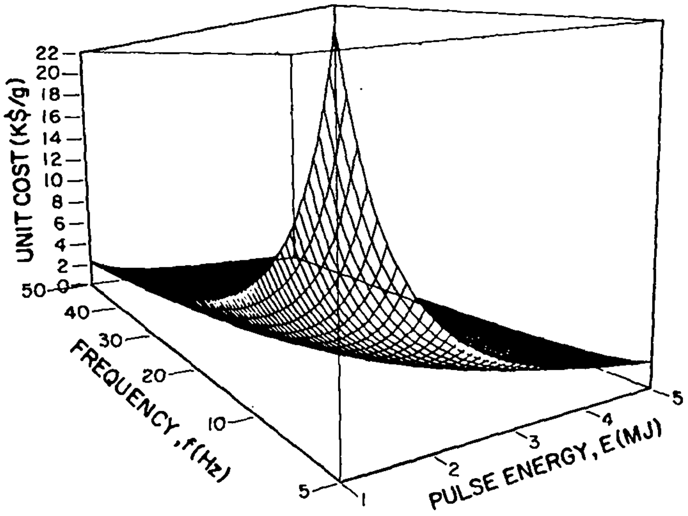

TABLE III. Input to U(E,f) Computation Assuming Heavy-Ion-Beam Driver and High-Gain Pellets

i = 0.15
z = 0.10
X = 8 (years)
L = 30 (years)
R_b = 2.0
k = 21.6
β = 0.78
A_f = 0.95
a = 5×10^8 \$
b = 5×10^8 \$
α = 0.4
d (\$/watt) = 0.5
η_d = 0.25

energy on target and pellet gain corresponding to ICF tritium production rates for which production costs equal the HTGR costs at a rate of 12 kg/yr. From Figs. 3-5 and Eqs. (1)-(3′), the pellet performances listed below are required for an energy-self-sufficient ICF tritium production facility.

Driver	Annual Production for HTGR Cost Equivalence (kg)	Energy on Target (MJ)	Pellet Gain
KrF	7	0.66	5.44
Heavy ion	9.9	0.78	6.54
CO_2	9.8	1.55	3.25

These results indicate that small pellet gains at ~1 MJ of energy on target are economically attractive. In contrast to the production of commercial electric power, the production of tritium does not require high pellet gains. Therefore, the technical risk is much lower than for fusion energy production.

In fact, it is unlikely that a first-generation ICF plant would be designed to produce even 7 kg of tritium per year, so pellet gains of ~1 at <1 MJ of driver pulse energy would permit the start of engineering tests. This could be particularly appealing because of the low capital costs of the driver and reactor and because the ICF-pellet development program credibly can be expected to verify gains of ~1 in the appropriate time frame.

B. Driver Requirements and Development

The driver requirements include

- output energy in the megajoule range;
- repetitively pulsed operation, ultimately exceeding 10 Hz;
- efficiency >4%; and
- affordability.

Three driver candidates exist that may satisfy these requirements: the CO_2 laser, the KrF laser, and the heavy-ion accelerator. Development plans for each of these are discussed in Secs. II.B.1-II.B.3.

1. CO_2 Lasers. High-energy CO_2 lasers for fusion applications have been under development for more than

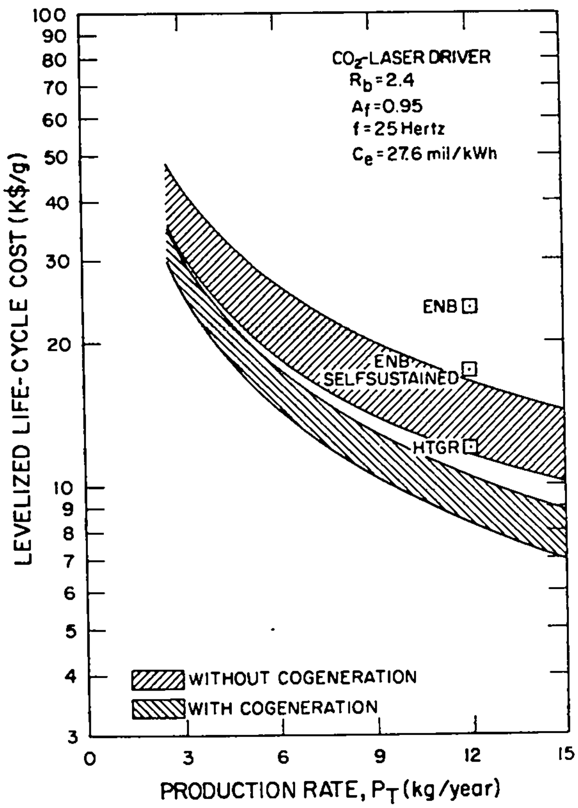

Fig. 3. Unit production cost with the CO_2-laser driver.

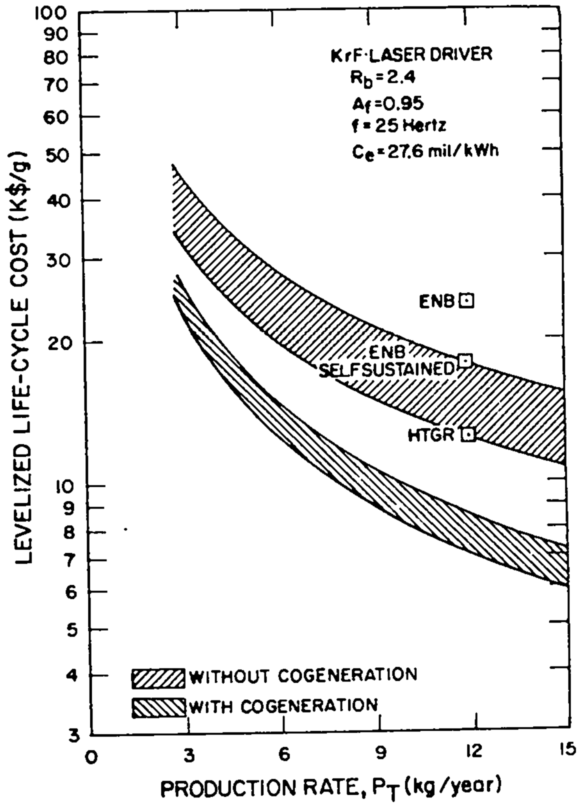

Fig. 4. Unit production cost with the KrF-laser driver.

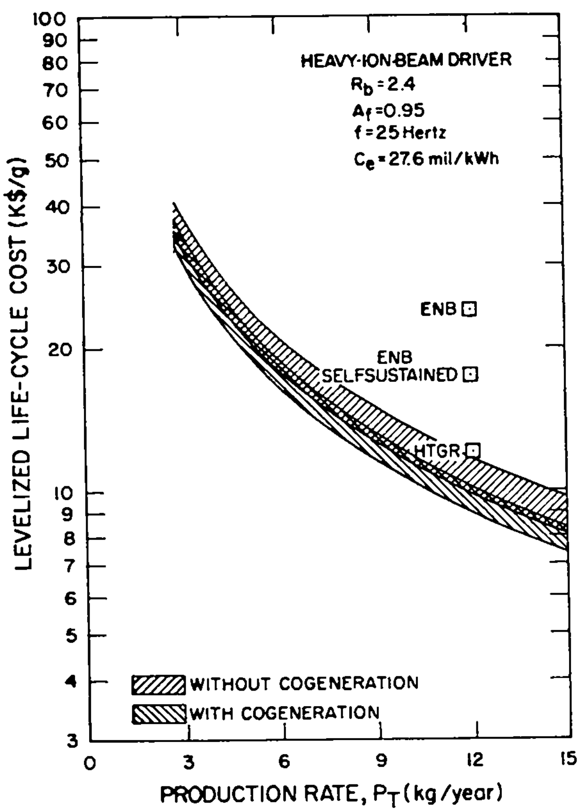

Fig. 5. Unit production cost with the heavy-ion-beam driver.

a decade and now represent a mature technology that we can use confidently to extrapolate performance and costs.

The Antares laser will become operational in mid FY83 and will provide output energies of up to 40 kJ. The Antares upgrade report* indicates that this facility could house a 1.2-MJ CO_2 laser based on current technology. The program plan for tritium breeding using CO_2 lasers would be based on this facility and its advanced development.

Output energy in the 1-MJ range could be available by 1990. With such energy for a CO_2 laser, pellet development would occur in successive stages, with pellet gains exceeding 1 demonstrated early in the experimental program. Fusion-pellet energy releases in the megajoule range would be adequate to confirm the feasibility of the ICF tritium breeding process and to set the stage for an engineering prototype demonstration.

Gas lasers, such as CO_2, can be operated repetitively with the addition of flow systems for circulating and cooling the laser media. Pulsed-power technology for CO_2 lasers has been developed and used at 10 Hz at high energy outputs, that is, with systems comparable in size with the Antares facility.

After successful results from the target physics using an upgrade of the Antares laser to 1 MJ and after verification of calculated tritium breeding rates, the next step would be the design and construction of a repetitively operated prototype demonstration system. Such a system would include, in addition to the repetitively pulsed driver, a reactor cavity surrounded by a tritium breeding blanket, tritium extraction and processing systems, automated pellet-production facilities, and possibly energy-conversion systems for cogeneration of electricity. Conceptual design studies would be conducted to determine whether the demonstration prototype could be accommodated in the Antares facility or whether a completely new facility would be required. Aggressive pursuit of this program goal could result in construction of a prototype demonstration system during the late 1990s. Successful completion of this program would provide the physics and engineering data needed to design a 10-kg/yr tritium production facility that could be on-line early in the 21st century.

2. **KrF Lasers.** Short-wavelength laser-driven ICF offers the potential of a higher target gain for a given laser pulse energy on target than that estimated for CO_2-laser-driven ICF. Pellet gains near 1 are predicted for laser pulse energies between 0.5 and 1.0 MJ. Confidence in these predictions will result from experimental programs during the mid 1980s using frequency-doubled or -tripled glass lasers now being constructed at the Lawrence Livermore National Laboratory. These lasers are valuable experimental tools but do not have the required efficiency or repetitive capability for the applications considered in this study.

An apparently better short-wavelength laser for fusion applications, including tritum breeding, is the KrF system now being studied at Los Alamos. This system may have the potential for an efficiency of 4%, be scalable to high output energies, and have acceptable costs.

The Los Alamos program plan includes development and demonstration of a power amplifier at the 20-kJ level along with the technology required for larger systems. This phase of the program is expected to be completed by 1984.

A successful prototype program by 1984 would allow a design and retrofit for the Antares facility that could provide 100 kJ of laser energy at 2480 Å before 1990. Target experiments at this level would verify target performance at this wavelength and provide a basis for a facility that could provide 0.5-1.0 MJ by 1995. Engineering studies are in progress to determine the largest short-wavelength laser that can be housed in the Antares facility.

A program plan for tritium breeding using KrF lasers would be similar to that for the CO_2 program, the steps being

- demonstration of near-one target gain,
- investigation of fundamental parameters of the tritium breeding process,
- development of a repetitively pulsed driver and test chamber for prototype system demonstration, and
- final engineering design for a 10-kg/yr production facility.

3. **Heavy-Ion-Accelerator Drivers.** ICF driven by heavy ions appears advantageous over laser-driven ICF because of the following:

- The physics of beam/target interaction appears to be better understood.
- Accelerator efficiency may be a factor of 3 greater than that for lasers (~25%).
- Accelerators are inherently repetitively pulsed devices that have demonstrated the long-term reliability needed for the breeding application.

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*This information provided by A. C. Saxman, Los Alamos National Laboratory (1982).

- Accelerators appear to be cheaper at the ultimately required energy levels.

Although the ICF accelerator development and target-interaction physics programs are in early phases (the concept was first considered seriously in 1976), a two-phase national program has been developed with the goals of demonstrating the required technology and performing critical target experiments before 1990. The key technologies to be evaluated include

- energy gain and current amplification in heavy-ion linear induction accelerators and
- energy gain in rf linear accelerators and current amplification by means of current-storage rings.

Aggressive development of heavy-ion-beam accelerators could lead to design and construction of a prototype tritium breeding facility on a time scale comparable with that anticipated for laser-driven ICF.

C. ICF Facility Characterization

1. General. The preceding analysis of tritium production using the ICF process provides sufficient information to characterize the facility in which the production potential can be realized. The cost estimates presented in Figs. 3-5 show that the ICF process becomes competitive with alternate production methods at the annual production rate of approximately 10 kg/yr. Figure 1 indicates that such a production rate requires a fusion power output of <200 MW for breeding ratios >2, which are easily attainable. The results presented in Fig. 2 (qualitatively similar for all drivers considered in this study) indicate that f should be >20 to 25 Hz to realize the full potential for the production cost reduction. Therefore, the required pellet yields range from 8 to 10 MJ. The yield requirement and an appropriate gain relation [Eqs. (2) and (3)] determine the necessary value for E.

In the following subsections we characterize the driver/pellet combination, the reactor vessel configuration, and the blanket composition and performance. We conclude the section with a brief discussion of test functions that can be performed in the proposed ICF facility without interference with tritium production.

2. Driver/Pellet Combination. The results presented in Figs. 3-5 indicate that the tritium production cost in ICF facilities designed to cogenerate electricity for self-sufficiency may be a factor of nearly 2 lower than the cost in facilities that buy commercial power. Therefore, we examine the conditions for self-sufficiency.

A useful way to characterize fuel-pellet and driver requirements for energy self-sufficiency is through the expression that relates the average driver beam power, fE, to P_T. The pellet gain required for self-sufficiency is

$$G_s = \frac{1}{(0.7E_x + 0.3)\eta_t\eta_d} \quad , \tag{8}$$

where E_x is the blanket neutron energy multiplication and η_t is the thermal efficiency. We used the fact that for a typical pellet, 70% of the energy yield is contained in neutrons and the remaining 30% in x rays and debris ions. Therefore, the expression for the beam power is

$$fE = \frac{(0.7E_x + 0.3)\eta_t\eta_d\, P_T}{55.65\ (R_b - 1)} \quad . \tag{9}$$

It is plotted in Fig. 6 for $R_b = 2.4$, $\eta_t = 0.30$, and $\eta_d = 0.10$ in the ranges of $1 \leqslant E \leqslant 25$ and $0 \leqslant P_T \leqslant 20$ kg/yr. Neutron-energy multiplications exceeding 10 will require enriched fissile multipliers.

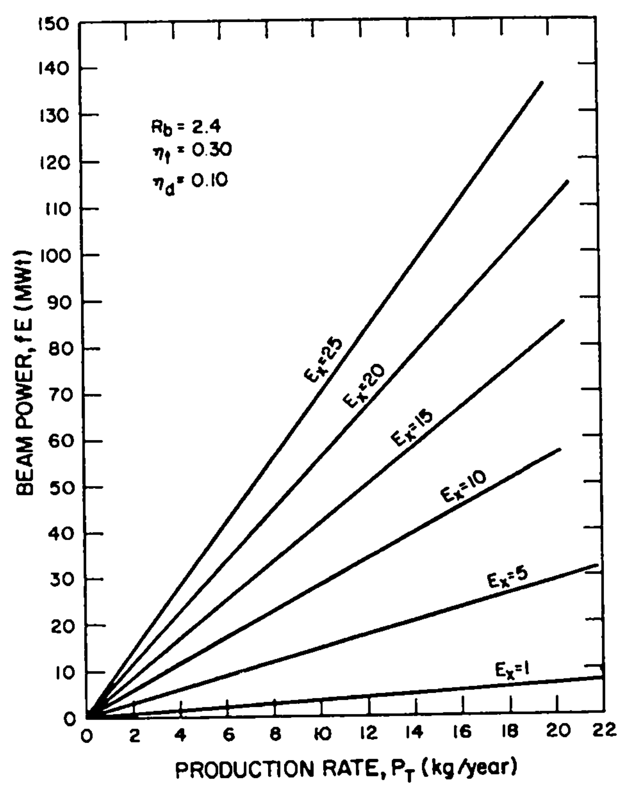

Fig. 6. Driver beam power at energy self-sufficiency.

The relationship between E_x and G_S is

$$E_x = \frac{1}{0.7G_s\ \eta_t\eta_d} - 0.429 \quad . \tag{10}$$

It is plotted in Fig. 7 for $\eta_t = 0.30$, with $\eta_d = 0.10$ and $\eta_d = 0.25$. Neutronic performance calculations show that near the optimum value of the tritium breeding ratio, a ^{238}U-^{6}Li-blanket energy multiplication is 11. The results in Fig. 7 show that for this value of neutron-energy multiplication, self-sufficiency can be achieved with G = 4 and $\eta_d = 0.10$ or with G = 1.5 and $\eta_d = 0.25$.

3. Reactor Vessel Configuration. The reactor vessel of the facility consists of two concentric spherical shells with structural wall thickness of 0.5 to 1.0 cm (Fig. 8). The inner shell radius is 150 cm and the outer is 250 cm.

The 100-cm-thick blanket containing neutron multipliers and liquid lithium for tritium breeding and heat removal is between the spherical shells. Its composition and performance are characterized and discussed in Sec. II.C.4. The microexplosion-facing wall of the inner shell is protected from erosion and excessive temperature excursions by a 1- to 2-cm-thick layer of liquid lithium injected tangentially with an initial velocity of 20 to 50 m/s. The worst case estimate (30% of 8 MJ, or 2.4 MJ, of energy into evaporation of lithium and 1620 K vapor

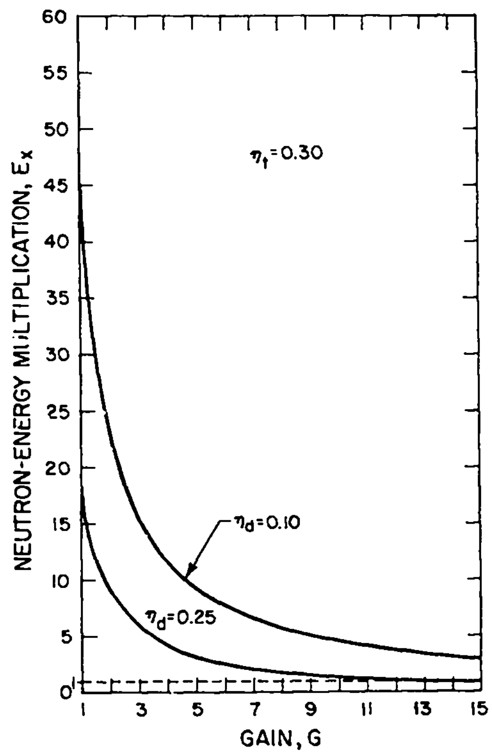

Fig. 7. Neutron-energy multiplication and pellet gain for power self-sufficiency.

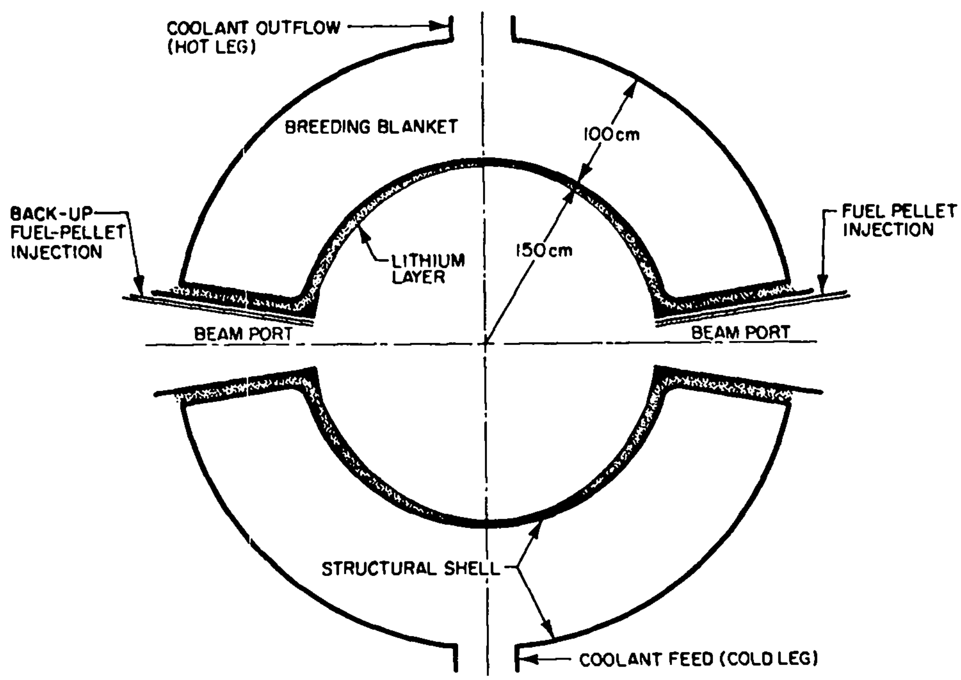

Fig. 8. Reactor vessel (schematic).

temperature) indicates that condensation on the relatively cold (550-650 K) liquid layer[8] will clear the cavity in 32.4 ms to a vacuum sufficiently low to ensure satisfactory propagation of driver beams; therefore, the required 25-Hz operation can be realized. The liquid-lithium coolant in the blanket can be operated at much higher temperatures to ensure satisfactory thermodynamic efficiency.

Figures 9-12 show the results of the one-dimensional modeling of the flow of lithium along the inside wall of a spherical reactor cavity. The velocity variations along the upper and lower hemispheres are shown in Figs. 9 and 11 and the corresponding variations in the layer thickness in Figs. 10 and 12. For the initial injection velocity of 50 m/s, the effect of gravity is negligible and the flow is nearly symmetric (Figs. 9 and 10). For the initial injection velocity of 5 m/s, the effect of gravity is significant (Figs. 11 and 12); however, the quality of the liquid layer remains acceptable for wall protection. These

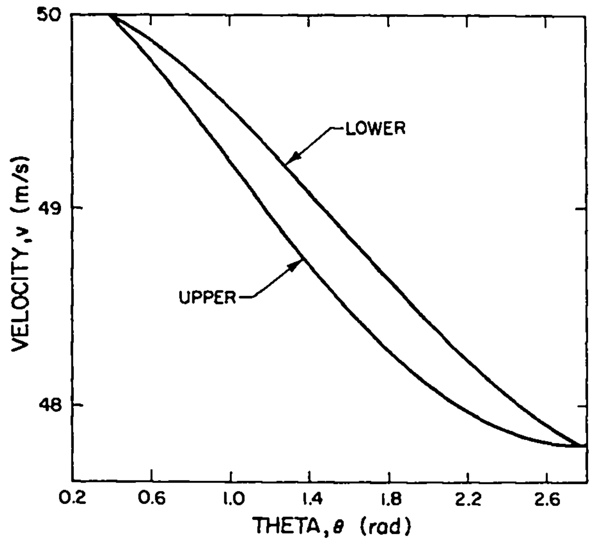

Fig. 9. Protective-layer velocity distribution for the injection velocity of 50 m/s.

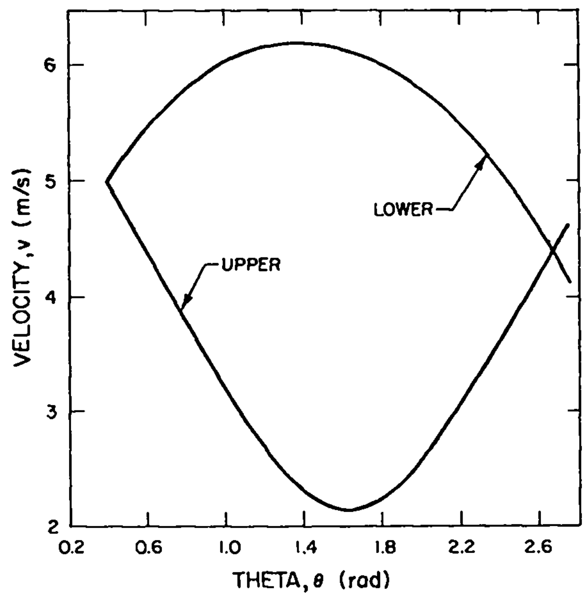

Fig. 11. Protective-layer velocity distribution for the injection velocity of 5 m/s.

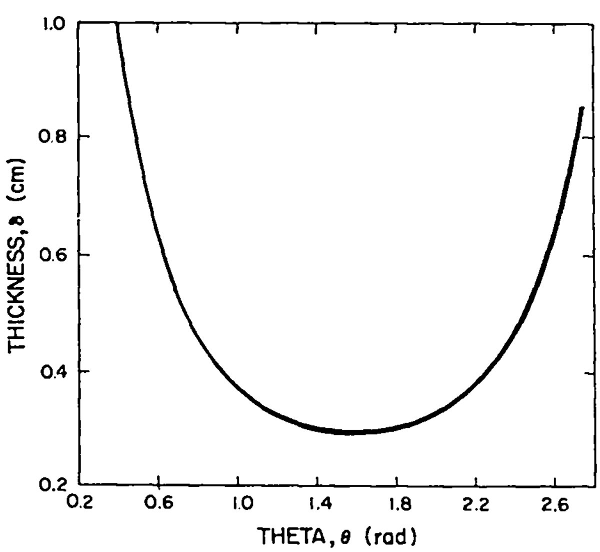

Fig. 10. Protective-layer thickness distribution for the injection velocity of 50 m/s.

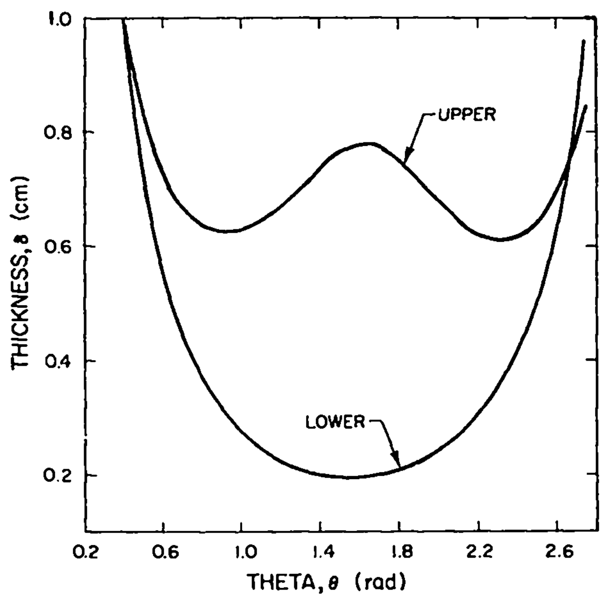

Fig. 12. Protective-layer thickness distribution for the injection velocity of 5 m/s.

results show that the range of admissible initial velocities and layer thicknesses is sufficiently large to allow selection of values needed to satisfy the heat removal requirements. The minimum velocity necessary to hold the layer attached to the wall by the centrifugal force against the force of gravity is 3.83 m/s; therefore, all investigated velocities will suffice.

The deposition of x-ray and debris energy in the lithium layer generates material blow-off and induces impulse at the wall. The stress in a 0.5-cm-thick wall resulting from the deposition of 2.4 MJ of energy is 55.4 $\times 10^6$ dyne/cm^2 (~750 psi);[9] this is an insignificant value. The corresponding radial displacement of the wall is 29.1 μm; this small displacement appears insufficient to shake off the protecting layer of fluid.

We have not yet carried out the detailed analysis of the transient cavity phenomena and of the waves in the liquid layer following rapid energy deposition and their interaction with the wall.

The characteristic thermal response time of a 0.5-cm-thick metal wall exceeds 1 s and the interpulse time at 25 Hz is 0.04 s; therefore, cyclic thermal effects will be negligible and the reactor vessel will operate in an essentially steady state.[10]

Neutron-damage criteria derived by Avci and Kulcinski[11] indicate that at an operating temperature of 300°C, the lifetime of the wall at 150 cm from the 8-MJ fuel pellet will be 4 full-power years. The actual lifetime of the first wall may well be 8 or more calendar years because it is unlikely that the prototype facility described in this report will be operated 24 h/day.

4. **Blanket Composition and Performance.** Tritium production using D-T neutron sources must also satisfy the requirements of the fuel cycle; i.e., one triton per fusion neutron is required to sustain the fusion process. The product tritium is the excess production over fuel-cycle requirements and is given by R_b - 1 (see Sec. I.D).

As indicated above, the simplest reactor concept for producing tritium would consist of a spherical reactor cavity, which would contain the fusion microexplosions, surrounded by lithium. Breeding ratios of about 2 are obtained from fusion neutrons incident on natural lithium without accounting for reactor structures. The introduction of reactor structures reduces the breeding ratio to about 1.7 or less, so in this case, the excess tritium is 0.7 triton per fusion neutron. As described in Sec. I.D, tritium breeding ratios can be increased significantly by including a neutron multiplier in the blanket region. Neutron multipliers can be either fissionable materials such as ^{238}U or nonfissionable materials with large (n,xn) cross sections such as ^9Be.

Scoping calculations indicate breeding ratios as large as 2.75 for mixtures of ^{238}U and ^6Li (Fig. 13). Further increases in breeding ratios can be obtained by enriching ^{238}U with small amounts of ^{239}Pu or ^{235}U. Mixtures of metallic uranium and lithium are probably not acceptable because of chemical incompatibility, so fissionable materials may have to be included as compounds such as carbides or as clad fuel elements similar to fission reactor fuel. Preliminary calculations for a ^{238}UC-^6Li slurry indicate a breeding ratio of 2.41, and the breeding ratio of ^{238}UC clad in zirconium and immersed in ^6Li is 2.32. The use of fissionable neutron multipliers increases the complexity of the system because of fission products; however, significant amounts of fissile material are produced simultaneously with tritium. For example, 0.5 atoms of ^{239}Pu are produced per fusion neutron in the ^{238}UC-^6Li slurry discussed above.

The most attractive nonfissionable neutron multiplier is ^9Be. Breeding ratios as large as 2.7 have been calculated for mixtures of ^9Be and natural lithium. Beryllium neutron multipliers eliminate system complexities required by the presence of fission products in blankets with fissionable multipliers and, therefore, also eliminate the potential for the production of ^{239}Pu. The reactor cost of $200M used in production-cost analyses is representative of estimated costs for lithium-cooled blankets containing metal-clad uranium carbide neutron multipliers; the cost of blankets containing nonfissioning beryllium multipliers may be as much as a factor of 3 lower. Therefore, accurate cost estimates for different blanket designs will require facility specifications that are

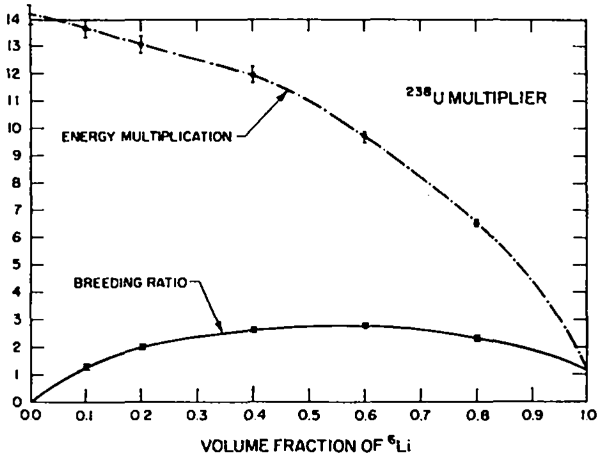

Fig. 13. Characteristics of ^{238}U-^6Li blanket.

more detailed than those available. The higher blanket cost was used in this study to obtain conservative facility cost estimates.

The logical choice of a fertile material for breeding tritium is liquid lithium because of its high atomic density and the prospect of continuous tritium extraction. If liquid-lithium systems are undesirable because of safety and environmental considerations, attractive breeders might be designed with solid-lithium compounds such as Li_3Sb, LiF, LiH, or Li-Pb. Such systems could be gas or water cooled. Other considerations influencing the choice of a solid fertile material would include corrosion, toxicity, suitable cladding materials, and tritium diffusion or other recovery methods.

Optimum ICF reactor designs for tritium production have not been identified or evaluated. Preliminary calculations indicate that breeding ratios considerably in excess of 2 will be possible; we, therefore, used the value of 2.40 in our economic assessments.

A fissionable blanket with neutron-energy multiplication sufficient to ensure facility self-sufficiency requires transport of nearly 1.4 GW of thermal power. If liquid lithium is used for that purpose and a 100 K temperature rise is allowed, then the lithium velocity will be $\sim$1 m/s in the blanket and 35 m/s in a 50-cm-diam feed pipe. These values are not excessive for present-day technologies.

Neutron multipliers may be in the form of approximately 1- to 2-cm-diam rods for which the characteristic thermal response time is approximately 5 s. Therefore, thermal cycling effects will be negligible and the elements will experience only a steady-state thermal load.[10] The steady-state (temporal mean) temperature rise from the cladding to the center of fissionable multipliers made from uranium carbide will not exceed 500 K; therefore, centerline melting will not occur.

5. Test Functions. In addition to its primary function of tritium production, the proposed facility can be used for development testing and weapons effects simulations. Toward this end one or two separate test chambers can be included in the facility at a modest incremental cost, and the driver beam can be directed into these chambers on a predetermined schedule.

The separate test chambers can be utilized to perform nearly all of the functions of a prototype test reactor leading to eventual commercialization of ICF. Of course, construction and operation of an ICF tritium production facility itself will require development of

- high-average-power driver;
- high-energy pulsed-power supplies;
- mass-production techniques for fuel pellets;
- pellet-injection, pellet-tracking, and beam-control systems;
- reactor vessel and primary heat-transport loop; and
- tritium extraction, handling, and processing systems and techniques.

However, the size and reliability requirements for these components are significantly less stringent for tritium production than for commercial power production; therefore, they will automatically serve as prototypes for their larger commercial counterparts.

In addition, existence of an ICF tritium production facility will allow for inexpensive long-term testing and evaluation of

- radiation, thermal fatigue, and corrosion effects on different materials;
- different reactor cavity concepts and their modifications;
- reactor system integration and reliability;
- diagnostics; and
- operating procedures.

In the area of weapons research the ICF tritium production facility will allow for experimental studies of

- vulnerability to neutron fluxes;
- x-ray effects (lethality and vulnerability);
- fireball phenomena; and
- general weapons physics (D-T burn, implosion stability, radiation energy transport, etc.).

In addition, a 1-MJ-class driver may be used without D-T fuel pellets to gather data pertaining to the equation-of-state studies using ablative sample acceleration and impact and for the directed energy weapons vulnerability studies.

III. PROGRAM PLANNING

It is important that the ICF program have a well-defined goal that can be achieved by the year 2000. The manufacture of reactor products (special nuclear materials and tritium) represents such a goal, one that is an integral part of the National Security Programs, is in the national interest, appears to be cost effective, and supports other, more technically demanding applications.

Regardless of the ultimate goals of the ICF program, the near-term R and D goals and milestones will be the same; i.e., the program will be paced by driver development and construction and by progress in pellet design and development. Ultimate goals will, however, affect required supporting research and may influence driver selection, assuming there is comparable success in more than one driver type.

In addition to the driver- and pellet-development research described here, considerable supporting research must be undertaken on a timely basis if the goal of demonstrating the production of reactor products is to be accomplished during the next two decades. Some of the more important areas of required supporting research are briefly discussed in the following sections.

A. Automated Target Production

Preliminary projections of fusion-pellet requirements for the production of ~ 10 kg of tritium per year indicate a daily requirement of about 2.2×10^6 pellets. Moreover, fusion pellets for this application must be reasonably cheap, and the materials of construction must be compatible with the reactor cavity first-wall protection schemes, pellet debris removal, etc. These requirements are not new; they have been established in the past for other applications of ICF. Nevertheless, a concentrated effort in this area would have high leverage in providing credibility for proposals such as tritium production.

B. Reactor Development

The results of preliminary designs and evaluations of reactor cavity concepts have identified appropriate design features and materials. Much work remains to be done to design these systems in detail and to optimize them for tritium and/or fissile-material production. This effort will require continuity and close collaboration between Los Alamos and other DOE personnel to establish production and product quality requirements, as well as guidelines relating to safety and environment.

Essentially no experimental work has been performed to establish the engineering feasibility and operating conditions of the several reactor cavity concepts that are now being seriously proposed and studied by ICF systems study groups throughout the nation. EPRI has taken the lead in initiating such an experimental program directed ultimately at the production of commercial electric power using ICF. This effort will be of considerable value in establishing the feasibility of the manufacture of reactor products; however, it will not be sufficient, so a joint effort involving the national laboratories and EPRI would be appropriate. Areas requiring development for which there are no current plans for research include pellet-injection, pellet-tracking, and beam-targeting systems; lithium pumps and heat exchangers; and radiation, fatigue, and corrosion effects on materials. Integral tests of reactor designs will also ultimately have to be performed.

Note, however, that the operating conditions proposed for the manufacture of reactor products are much less severe than for the production of commercial electric power. Materials endurance in radiation fields, for example, can almost be considered state of the art for the proposed application. Indeed, production facilities for reactor products would serve as test vehicles for the production of commercial electric power.

C. Tritium Extraction and Processing

An essential component of any fusion production facility is a tritium recovery and processing system. Considerable theoretical and engineering design effort has been expended in this area, and several potential processes have been identified; however, no process for extracting tritium from liquid lithium has been demonstrated superior. Such a selection will require extensive experimentation. The only experimental work in this area was carried out at the Argonne National Laboratory and involved a molten-salt extraction process. This effort was supported by the Magnetic Fusion Energy Program but was terminated before definitive results were obtained.

It is also possible that the tritium breeding material will not be liquid lithium. If a solid-lithium compound is used for this purpose, for example, totally different tritium recovery processes will be required.

D. Pulsed-Power Systems

Repetitive-pulse power systems are a concern for applications such as the one proposed in this report. It would be possible to satisfy the pulsed-power requirement using derated existing systems; however, component lifetimes for continuous, extended operation would have to be established by an experimental program. Eventually, new pulsed-power concepts will be required to reduce costs and improve reliability. An R and D program in this area would be very worthwhile.

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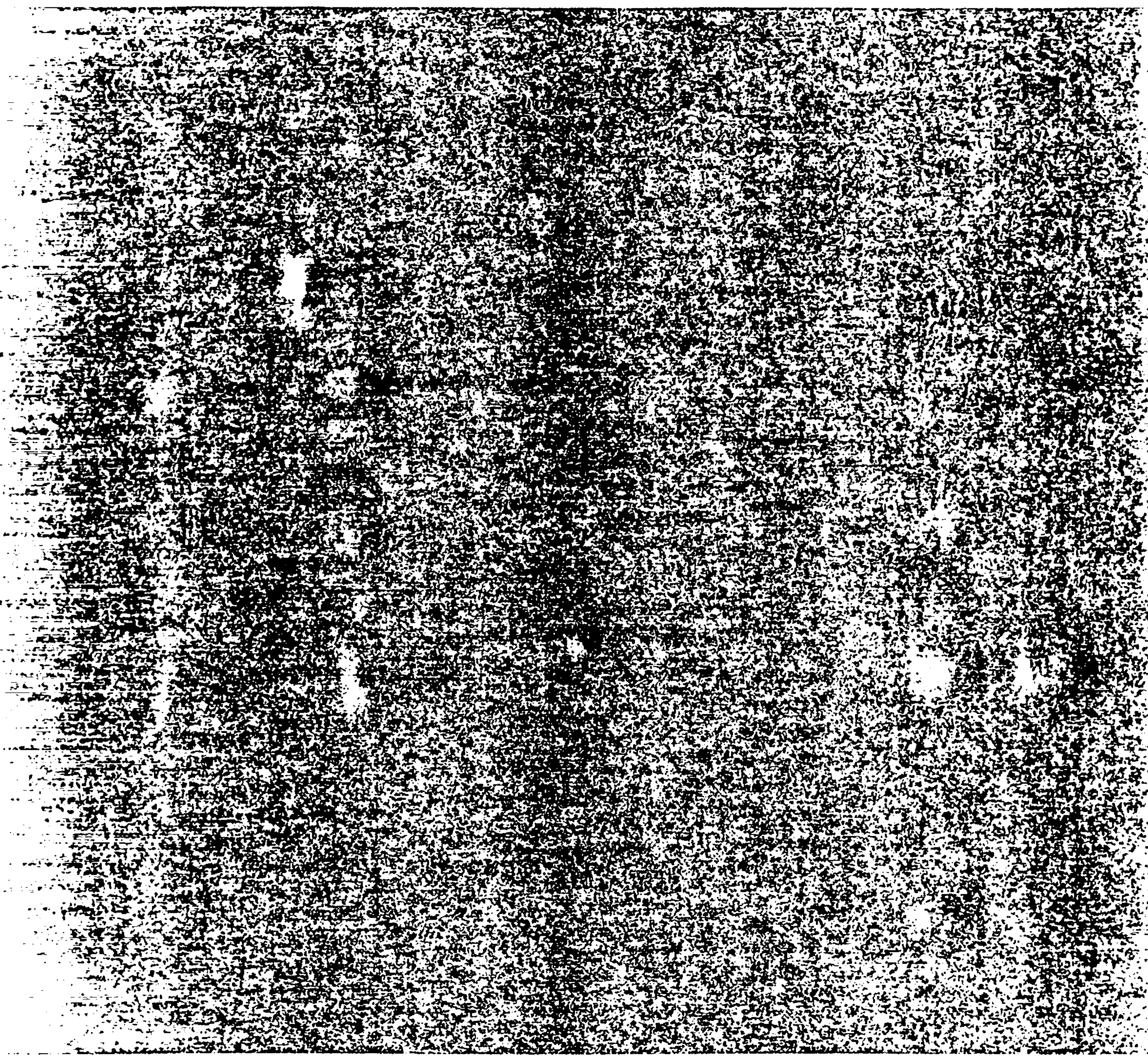

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