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TRITIUM AND PLUTONIUM PRODUCTION AS A STEP TOWARD ICF COMMERCIALIZATION

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ABSTRACT

The feasibility of a combined special nuclear materials (SNM) production plant/engineering test facility (ETF) with reduced pellet and driver performance requirements as a step toward commercialization of inertial confinement fusion (ICF) is examined. Blanket design and tritium production cost studies, the status of R&D programs, and the ETF role are emphasized.

I. INTRODUCTION

A. General

Development of high-gain, low-cost targets and efficient, affordable, repetetively-pulsed drivers for (ICF) has been more difficult than originally anticipated. Commercialization of ICF for electric power and process heat will be delayed by these difficulties. Nearer-term, significant, credible, and well-defined goals can justify and focus ICF research and development (R&D) programs. An ICF tritium and plutonium (SNM) production plant that could also serve as an engineering test facility (ETF) for the more technically demanding commercial applications constitutes such a goal. We believe that this goal can be in the national interest, cost effective, and a logical step toward ICF commercialization.

Tritium production is an important national requirement. Tritium is currently produced at high cost for defense programs by transmutation of lithium in fission reactors. Tritium has a short (12.3 years) half-life and must be produced continuously to maintain defense stockpiles. Tritium will also eventually be required in large amounts for fusion R&D programs and startup of pioneering commercial applications plants.

Regardless of ultimate ICF goals, near-term R&D needs are the same--high-gain, low-cost targets, and efficient, affordable drivers--but goals will eventually influence the directions of other ICF R&D efforts, e.g., pellet and reactor design and identification of best driver technology.

B. The ETF Role

An ICF SNM production plant could represent a logical step in an R&D program for ICF commercialization for several progress toward more technically demanding commercial applications. The proposed application would require low-cost, efficient, repetetively-pulsed drivers, rather than the single-pulse capability that is adequate for physics experiments. Also, driver efficiency and low unit capital cost and operating and maintenance cost are not crucial to achievement of experimental aims. Mass (millions/day) production at low cost of simple, robust targets that can survive abrupt injection into hositle reactor cavity environments would be necessary. For single-shot experiments, target chamber conditions can be adjusted to the most favorable, few targets are required, and they can be mounted on stationary supports, so that experimental targets can be complex, fragile, and expensive. On the other hand, competetive SNM production costs can apparently be achieved with lower pellet gains (≤ 15) and driver pulse energies (~1 to 3 MJ) than are required for competetive electric power and process heat production.[1,2]

The proposed SNM production plant/ETF would be intermediate in scale (≤ 1500 MWth for an energy self-sufficient plant to produce ~10 kg/yr of tritium and ~500 kg/yr of plutonium) between proof-of-principle experiments and ICF commercial electric power plants (~3000 MWth and up). A demonstration facility of approximately the same size as the proposed SNM plant/ETF would likely be required during scaleup to ICF commercial electric power. Of course, other intermediate facilities might also be required. As an ETF, the proposed facility could be used to improve ICF technology, integrate complex ICF systems, and qualify materials for long-term service under the demanding conditions of a realistic ICF environment, while providing needed experience under commercial or semi-commercial conditions. The knowledge to be gained from an ETF could be obtained in a facility that was self-supporting, or nearly so, if SNM production were included as part of its function.

C. SNM Production

We believe that ICF offers several advantages relative to magnetic fusion (MF) for the proposed SNM production mission. ICF is projected to be more economical than MF for small plant capacities. In an ICF plant there is less neutron irradiation of complex and expensive subsystems. ICF reactors are simpler and less closely coupled to other fusion subsystems, and accessiblity for maintenance and repair and reprocessing of blanket fertile elements is greater. The total containment volume for a specified plant capacity is less for ICF than for MF. ICF reactor and blanket designs need not accommodate large magnet systems and strong magnetic fields.

Several facility scenarios are potentially interesting--maximum tritium production, maximum plutonium production, maximum tritium plus plutonium production, minimum technical risk, and maximum suitability for the dual production/ETF role. As with fission reactor SNM production, the first three scenarios are mutually exclusive. Although technical risk can be reduced through conservative design based on current practice for conventional plant systems, much of a pioneering ICF plant would involve new technology. Therefore, we view the proposed facility as inherently a high risk enterprise. In addition, a lowest risk design strategy could significantly impact plant economics in an unfavorable way. We have elected to emphasize tritium and plutonium production with some technical risk reduction and production costs competetive with new fission reactor production.

ICF is not a competitor of the near-term fission new production reactor (NPR) for which extensive generic design studies are presently being conducted. The ICF SNM plant/ETF facility horizon lies beyond the year 2000. Full production should not be expected immediately in such facilities because part of the ETF mission would be "bootstrapping" of experimental ICF technology to the reliability and pulse repetition rates that would permit the production rate desired.

However, we feel that ICF can eventually be an important option for production of tritium, plutonium, and other special nuclear materials. An ICF facility could be designed for primary production, for swing production, and/or for production of SNM's with unusual specifications.

We report here on encouraging preliminary SNM blanket design and production cost studies. Ongoing studies will result in detailed driver fuel element, blanket, ICF reactor, and plant designs and more accurate plant capital and production cost estimates.

D. Driver Characteristics and R&D Program Status

Three driver candidates that may satisfy ICF SNM production plant requirements are under development. These requirements include few MJ pulse energies, repetetively pulsed operation at 10 Hz, good absorption of driver pulse energy by targets, and affordability. The three candidates are CO_2 lasers, KrF lasers, and induction or radiofrequency (rf) linac heavy-ion accelerators.

CO_2 lasers have been most thoroughly developed. CO_2 laser technology is relatively mature so that extrapulation of performance and costs can be performed with fair confidence. CO_2 lasers are efficient (~ 10%), but their long-wavelength (10.6 μ) output couples poorly with current pellet designs. Poor coupling (absorption of driver pulse energy to efficiently compress and ignite pellet DT fuel) results in lower pellet gains for a specified driver pulse energy. However, CO_2 lasers are presently projected to be the lowest cost driver candidate at low pulse energies. The circulation of gaseous CO_2 lasing medium for waste host removal permits moderate pulse repetition rates (few 10's of Hz). CO_2 lasers are modular in nature and hence scalable to required pulse energies through duplication of modules. The Antares facility at Los Alamos National Laboratory is operational at 40 kJ; upgrading to 1 MJ by 1990 is possible.

Shorter wavelength (0.248 μ) KrF lasers offer the possibility of better pellet coupling, and hence higher target gains, for a specified driver pulse energy. Maximum practical KrF laser pulse repetition rates are similar to those for CO_2 lasers and high pulse energies would be obtained through duplication of modules, but efficiences may be low ($\lesssim$5%). Very limited studies suggest intermediate costs at low pulse energies and higher costs at high pulse energies than for other driver candidates. More detailed cost/ performance studies have begun at Los Alamos National Laboratory and elsewhere. New single-pulse glass laser experimental facilities at Lawrence Livermore National Laboratories can be used to verify the projected short-wavelength pellet coupling in the 1980s. A 20 kJ KrF prototype should be finished during FY84 at Los Alamos National Laboratory and a 100 kJ facility is planned for the late 1980's.

Heavy-ion accelerators appear to have the greatest potential. Driver-target coupling is better and better understood than for lasers. Very good driver efficiencies (~ 25%) may be possilbe with heavy ion accelerators. High pulse repetition rates (1000 Hz and up) are possible and inherent reliability is projected for heavy ion drivers. Preliminary cost studies suggest the possibility of lower costs for heavy ion fusion drivers than for other driver candidates at high pulse energies. The US heavy-ion program is in an early phase, but aggressive R&D would promote parity with laser technology state of development. The present programmatic emphasis is on induction linac technology, with major experimental facilities proposed for construction in the next few years. The primary induction linac R&D effort has been concentrated at Lawrence Berkeley Laboratory, with Los Alamos National Laboratory responsible for program management.

E. Other R&D Requirements

Other R&D requirements for an ICF SNM plant/ETF will become more prominent when required driver and pellet cost and performance objectives are achieved and pellet mass production processes are perfected. Several promising reactor concepts developed by the ICF community require more detailed design studies and feasibility (especially integrated) experiments, with special attention to cavity clearing, pellet injection and tracking, driver/reactor interfaces, driver beam aiming, focusing, propogation, and materials. Tritium migration control and recovery methods must be effective under extreme conditions of low concentrations and high temperatures in the presence of aggressive chemicals. Derated existing pulsed-power technology can be used, but qualification of less-expensive, more-reliable pulsed power technology could significantly impact ICF costs.

II. BLANKET STUDIES

A. Reason for New Blanket Studies

Part of the mission for the proposed facility--production of large amounts of tritium in excess of that required to fuel the plant--is significantly different from the usual commercial-applications goal with respect to tritium--mere plant tritium self-sufficiency. Therefore, extensive blanket conceptual design studies to ensure a good design for the new mission seemed advisable.

B. Fusion Versus Fission Production of Tritium

More energetic fusion neutrons offer significant advantages over fission neutrons for SNM production. Natural lithium is about 7% ^{6}Li, with the remainder being ^{7}Li. However, the isotopic composition can be adjusted relatively inexpensively if required for optimum tritium and plutonium production. ^{7}Li reacts with high-energy neutons (~5 MeV practical threshold) to produce a triton and a lower-energy neutron that can react with ^{6}Li to produce another triton. Fusion neutrons are born at ~14 MeV and fission neutrons at ~ 2 MeV, although there is some spread around these energies. Thus, for practical purposes the ^{7}Li reaction is accessible with fusion neutrons, but not with fission neutrons. Fusion neutrons can also be "multiplied" by (n, 2n) and (n, 3n) threshold reactions not accessible with fission neutrons, e.g., by factors of 3.03 and 4.12 with ^{10}Be and ^{238}U respectively in the absence of competing reactions.

C. Design Parameter Space and Figures of Merit

To reduce the blanket design parameter space to manageable proportions, several preliminary decisions were made early in our blanket design studies,[3] leaving other approaches to be explored as resources permit. In particular, we are presently concentrating on energy-multiplying blankets of basically cylindrical geometry with metallic-uranium fertile elements of relatively conventional fuel pin design and with liquid lithium as both coolant and tritium breeding material.

Preliminary cost studies indicated that plant energy self-sufficiency (or near self-sufficiency) is important for competitive tritium production with low pellet gains--the cost of driver and plant auxiliaries power is burdensome otherwise.[1,2] Large blanket neutron energy multiplication can be achieved only if fissionable materials (200 MeV/fission versus 14 MeV fusion neutron kinetic energy) are included in the blanket. Fissionable isotopes such as ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, and ^{232}Th multiple fusion neutrons by means of (n, 2n) and (n, 3n) reactions and through fission to permit higher tritium breeding ratios.

Metallic uranium was selected as the fissionable material for our reference blanket design. Thorium cycle products would be used if produced, but would not be preferable to uranium cycle products. Difficult isotope separations would be required for some applications of thorium cycle products. Uranium compounds involve neutron-absorbing and moderating diluent elements. The design data base for metallic fuel elements is less extensive than that for oxide fuels but is adequate. Conventional clad fuel-pin designs are being studied first because they may be the only acceptable choice for safety, performance, and cost reasons and the design data base is adequate. Liquid lithium was selected as the blanket tritium-breeding material and coolant. It is necessary for the highest tritium BR and, if also used for first wall protection as in wetted-wall reactors,[2] commonality of liquid metal loops is promoted.

Realistic reactor/blanket concepts were used in our neutronics and burnup studies. We began with the strawman or reference concept depicted in Fig. 1, derived from consideration of preliminary scoping studies[1]. As we proceeded with our nucleanics and burnup calculations, the second strawman or reference conceptual blanket design characterized in Fig. 2 was developed. This second conceptual design, although not completely optimized, offers good performance and is simple. It is also thinner and contains much less heavy metal (HM) than the first reference design. We have emphasized wetted-wall reactors[4] as having greater potential for achieving large tritium breeding ratios with liquid lithium-fissionable blankets.

Our blanket design studies have involved determination of the effects of variations of many parameters on blanket performance, including blanket structure/uranium/lithium ratio, distribution of the materials throughout the blanket, blanket lithium isotopic composition, blanket thickness, ^{235}U fraction in blanket uranium, wetted-wall reactor lithium first-wall protection layer thickness and isotopic composition, and fissile and fertile material burnup. A pellet neutron output spectrum (Fig. 3) representative of commercial applications pellets, was used for all nucleonics and burnup calculations. The primary characteristics of the spectrum that was used in our studies were doppler broadening about 14 MeV and a peck at lower energy characteristic of single scattering events in the compressed pellet core. The more realistic spectrum allows tritium breeding at a ratio about 90% of that calculated for pur 14-MeV neutrons. The blanket performance parameters that we have been most interested in include tritium, Pu, and tritium plus Pu breeding ratios (BR's--atoms of isotope(s) bred per fusion neutron), blanket neutron kinetic energy multiplication (blanket thermal energy release divided by fusion neutron kinetic energy or by total thermonuclear energy release as used herein), maximum heating rate (thermal power density), rate of fissionable material burnup, and isotopic composition of bred Pu.

D. Results

a. Lithium/Uranium Volume Ratios. Blanket performance was expected to depend significantly on the ratio of uranium to lithium. Beginning-of-life neutronics calculations for our first strawman blanket (pure ^{238}U uranium, 60% -- ^{6}Li lithium, and 5 v/o structure indicated that as the volume percentage of Li increased from 20 to 95, the Pu and tritium plus BR's decreased monotonically and the tritium BR passed through a fairly flat maximum at about 60 t0 70 v/o Li (Fig. 4). We also found that the blanket energy multiplication factor depends strongly on the blanket v/o Li (Fig. 5). Because of our emphasis on tritium production, we selected 60 v/o blanket Li for our reference blanket design. The flatness of the tritium BR curve near its maximum and the large negative slope of the Pu BR curve results in little loss in tritium BR and a large increase in tritium plus Pu BR compared to the maximum tritium-BR v/o Li.

b. Blanket Lithium Isotopic Composition. Examination of the effects of variation of the isotopic composition of blanket lithium on blanket performance parameters yielded the very interesting results shown in Figs. 6 and 7. We see that over the entire 0 to 100% ^{6}Li range of lithium isotopic compositions the ratio of Pu BR to tritium BR varies from about 3.05 to about 0.16, there is an 6% decrease in tritium plus Pu BR, and an 27% decrease in blanket thermal power. Thus, we conclude that our blankets can be tuned over a wide range of product mixes (Pu versus tritium), with little loss in total breeding capacity and thermal power, by simply changing the blanket lithium isotopic composition. This versatility could permit ready response to changes in SNM demand patterns. However, the full blanket product mix flexibility can be achieved only if heat transport and ultimate rejection systems and electric power generation equipment capacity corresponding to the maximum blanket thermal power is provided. Currently, our reference blanket lithium isotopic composition is 60% ^{6}Li, which gives a near-maximum tritium BR and a higher Pu BR and greater blanket neutron energy multiplication than does 100% ^{6}Li.

c. First-Wall Protection Layers. Important blanket parameters are significantly affected by first wall protection layers in reactor cavities, including breeding ratios, blanket neutron energy multiplication, and heating rates. The thickness of liquid metal first-wall protection layers in wetted wall reactors[4] vary with position. The dry-wall reactor solid sacrifcial layer thickness range is 1 to 5 cm. Different first-wall protection layer compositions have different effects on blanket parameters.

Some computed effects of first wall protection layer thickness and composition variations on blanket performance characteristics are shown in Figs. 8 (tritium BR), 9 (Pu BR), and 10 (thermal power). We see that the influence of protective layer composition is generally more significant than layer thickness.

These results suggest that if first-wall protection is to be provided by liquid lithium, it should be the natural mixture or be enriched in ^{7}Li for the ICF SNM production mission. Separate liquid metal loops would then be required for the blanket and for first-wall protection. However, because of pellet debris deposition in the first-wall protection loop, segregation of the two loops is desirable in any event. The computed effects of first-wall protection layers on blanket performance are due to a combination of (n, 2n) neutron multiplying reactions, the ^{7}Li tritium breeding reaction, and neutron-moderating (scattering) interactions.

d. ^{235}U Enrichment of Blanket Uranium. The effects on initial blanket performance of several ^{235}U enrichments have been studied. Pure ^{238}U was used in our first scoping studies. In addition, we considered the use of diffusion plant tails (~0.25% ^{235}U and available for the cost of conversion from UF_6 to the metal), the natural mixture (~0.71% ^{235}U), a typical light water reactor enrichment (~3.2% ^{235}U) and higher enrichments all the way up to criticality (~30% ^{235}U at beginning of life). Large enrichments may be of interest, but cost effectiveness and the acceptability of approach to criticality must be stablished. Isotopic enrichment is costly for heavy elements and is not required because the fusion neutron source is supposed to substitute for the criticality required for fission reactor operation and because bred fissile isotopes serve the same purpose. One potential safety advantage for ICF hybrid facility, as opposed to a pure-fission plant, is that criticality and its inherent risks are avoided. Fertile element fabrication becomes more difficult and expensive at high enrichments. Certainly higher enrichment means better blanket performance insofar as fusion breeding ratios and blanket power density are concerned, but the question "at what point does the fusion neutron source become superfluous as enrichment is increased?" must be addressed.

We find at low ^{235}U concentrations in ^{238}U that tritium and plutonium breeding ratios increase roughly in proportion to ^{235}U enrichment (illustrated in Fig. 11). Significant departures from proportionality occur at large ^{235}U enrichments (see Fig. 12). Our burnup studies have necessarily been limited in scope because of the computational effort required. However, as we discussed below, the principal effects of burnup of fissionable materials in our blankets result from Pu bred into them and appear from our limited burnup study to roughly parallel the effects of variation of blanket initial ^{235}U enrichment on blanket performance. Thus, Figs. 11 and 12 give an indication of the magnitude of the effects of burnups other than those for which calculations were performed.

e. Effects of Blanket Structure and Reactor Cavity Size. Our calculations indicate that tritium and Pu FR's decrease roughly in proportion to increase in amount of blanket structure for modest amounts of steel structure with the uranium/lithium volue percent ratio held constant. This dependence is illustrated by Fig. 13. As expected, blanket tritium and Pu BR's (Table) are essentially independent of reactor cavity size, whereas power densities (Fig. 14) are approximately inversely proportional to cavity radius squared. Fertile material burnup rates, which determine times between reprocessing, also will vary approximately inversely as the cavity radius squared.

f. Burnups, Production Rates, and Product Specifications. Burnup calculations indicate that our blankets breed very good plutonium at a rate of 500 kg/yr in a blanket assigned to breed 10 kg/yr of tritium. The quality of the bred Pu is indicated by the computed results in Figs. 15 and 16. We also find (Fig. 17) that our blankets can be irradiated for several years before reprocessing of inner blanket elements, which are the most intensely irradiated, is required when a conservative maximum permissable burnup of 5% is assumed. Exchange of more highly irradiated fuel pins with lower-burnup pins could eliminate any need for reprocessing for the entire plant lifetime if desired, provided that other fuel pin service lifetime limiting factors do not become operative.

Our burnup calculations also reveal only modest increases in BR's and blanket thermal power for long irradiation periods (Table I). Heat tranport and ultimate rejection systems and power generation equipment must be sized for maximum blanket thermal power; the modest increase in heating rate for substantial amounts of bred Pu means that considerable over design of these costly plant systems is not necessary. Approximate interchange schedules for fuel elements with different irradiation histories can reduce even the modest thermal power variations listed in Table I.

For burnups 5%, fission products do not seem to substantially affect the neutronics of our blanket and changes in ^{238}U concentration are small. Most of the changes in blanket performance are therefore largely attributable to bred Pu and are approximately proportional to the bred Pu concentration.

g. Required Gains. Only small gains are required for plant energy self-sufficiency with energy-multiplying blankets. Energy self-sufficiency gains for fissioning blankets for several drivers are compared with corresponding gains required with pure-fusion systems in Table II. Fig. 18 is a comparison of gains required to produce 10 kg/yr of tritium as a function of driver pulse energy and repetion rate. Tritium production rate and plant fusion power are related in Fig. 19 with tritium BR as a parameter.

III. PRELIMINARY TRITIUM PRODUCTION COST STUDIES

Although scarcity of reliable ICF cost data makes accurate estimation of absolute costs difficult, we believe that our preliminary tritium production cost studies clearly delineate trends. We have emphasized tritium production costs by treating electric power production in excess of that required for plant energy self-sufficiency, as well as, plutonium as by products, with all profits derived from their sale credited against tritium production costs. Standard life-cycle cost analysis methods were employed with representative economic parameters and simple, but widely used cost models for major subsystems. The analyses were performed with conservative and optimistic pellet gain-driver pulse energy relationships. We compare our ICF tritium production cost estimates with recent estimates for new fission reactor production costs, with approximate costs claimed for existing production reactors, and with estimates prepared for electronuclear breeding.

We note first that production costs decrease rapidly with increase in driver pulse energy and repetition rate at low values of these parameters, with less rapid decreases for large values of pulse energy and repetition rate. We find that credits for bred plutonium, even without claiming extra value for the superior quality of ICF plutonium, are substantial (up to \$1000's/g of tritium, with exact values depending on reprocessing and fuel element fabrication cost assumptions). The significance of export power credits depends, of course, on the amounts of power available for export and the cost of electricity, but large credits are possible. The cost penalties associated with discard of all thermal energy and import of all electric power for driver and plant auxiliaries operation are generally substantial. These penalties are generally larger for less efficient drivers. More optimistic pellet gains result in lower production cost estimates; the relative effect is greater with less efficient drivers. Plant scale also has a significant effect on tritium production cost estimates. However, all competetive processes exhibit similar cost scaling with production rate.

Our tritium production cost estimates are less sensitive to BR for BR $\gtrsim$ 2. They differ by ~25% for optimistic and conservative pellet gains for energy self-sufficient plants and by ~50% when all thermal energy is discarded rather than used to generate electricity to power the plant.

We obtain ICF tritium production cost estimates that for many cases are substantially lower than for new fission reactors and for all cases can be considered competetative with new fission reactors when the accuracy of both sets of estimates are considered. The lower estimated costs for ICF tritium production are competetive with current production reactor costs which are primarily operating and maintenance costs. The high driver efficiencies, optimistic pellet gains, optimistic reprocessing and fuel element cost assumptions, high plutonium and export electric power values, and so forth. Electronuclear breeding is not competetive under any circumstances.

IV. SUMMARY

A significant, credible, well-defined, near-term application of ICF has been identified--a combined SNM production plant/ETF that could be a logical step toward commercialization of ICF. This application is less demanding of driver and target performance than commercial electric power generation and process heat, but would require development of much technology required for commercial applications. The required driver and target performance is within the scope of anticipated near-term ICF R&D program achievements.

Our tritium-producing blanket studies are yielding encouraging results. We have concentrated on fissionable blankets for plant energy self-sufficiency which our cost studies indicate is important for economical tritium production. We find that realistic maximum tritium and Pu BR's are respectively ~2.0 and ~0.6 in uranium/lithium blankets. The ratio tritium BR/Pu BR can be adjusted over a wide range, with little effect on blanket thermal power, by simply adjusting blanket lithium isotopic composition. About 500 kg/yr of ~99% ^{239}Pu plutonium, with reprocessing required only after several years irradiation, can be produced in a blanket designed to produce 10 kg of tritium per year. The effects of variation of numerous design parameters on blanket performance were investigated.

The results of preliminary ICF tritium production cost studies are encouraging. We find that ICF can compete successfully in producing tritium at small rates with fission reactors producing at larger rates and with magnetic fusion. The effects of various influences on tritum production costs, including production rate, plutonium credits, net electric power production, gain, driver pulse energy and repetition rate, and choice of driver technology, have been examined.

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