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Abstract

A scoping study of a symmetrically illuminated ICF tritium production facility utilizing a KrF laser is presented. A single shell ICF target is illuminated by 92 beams symmetrically distributed around a spherical cavity filled with xenon gas at 1.0 torr. The driver energy and target gain are taken to be 2 MJ and 50 for the optimistic case and 1 MJ and 100 for the conservative case. Based on a graphite dry wall evaporation rate of 0.1 cm/y for a 100 MJ yield, we estimate a cavity radius of 3.5 m for a rep-rate of 10 Hz and 3.0 m for 5 Hz. A spherical structural frame has been scoped out capable of supporting 92 blanket modules, each with a beam port in the center. We have selected liquid lithium in vanadium structure as the primary breeding concept utilizing beryllium as a neutron multiplier. A tritium breeding ratio of 1.83 can be achieved in the 3 m radius cavity which at 10 Hz and an availability of 75% provides an annual tritium surplus of 32.6 kg. Assuming 100% debt financing over a 30 year reactor lifetime, the production cost of T_2 for the 2 MJ driver case is $7,325/g for a 5% interest rate and $12,370/g for a 10% interest rate.

Introduction

Investigation of inertial confinement fusion (ICF) using symmetrically-illuminated direct drive targets has been pursued by the U.S. Department of Energy as a backup option to the main-line indirect drive targets which do not require symmetric illumination, being developed by various national laboratories. As part of this program, the Fusion Technology Institute of the University of Wisconsin (FTI) in collaboration with the University of Rochester's Laboratory for Laser Energetics (LLE) and in consultation with the Naval Research Laboratory (NRL) has been conducting a study of the critical issues related to symmetric illumination ICF systems. One of the recently completed studies is a scoping investigation of a materials test facility SIRIUS-M [1]. In the present paper we report the preliminary results of a similar scoping study of a tritium production facility SIRIUS-T.

It has been known for a long time that a nuclear fusion facility is ideally suited for T_2 production due to the excess neutrons it provides relative to the thermal energy of reaction. In the present investigation we have performed trade studies and system analyses aimed at producing T_2 at the lowest cost using reasonable assumptions. Neutronics analysis has optimized material combinations and structural fractions to maximize the tritium breeding ratio within a realistic engineering design. In the tritium area we have focused on extraction, containment, inventory and release pathways.

An initial screening of candidate breeding schemes has been made with consideration given to liquid metals (Li, LiPb), solid ceramic breeders (Li_2O, $LiAlO_2$, Li_4SiO_4), solid metallic breeders (LiAl), salts (FLIBE, $LiNO_3$, LiOH) and gaseous He-3. The scheme selected is that of liquid Li in vanadium structure, with a backup scheme of using He-3. Lithium in vanadium structure is the number one recommendation of BCSS [2]. Among its advantages are low pressure, excellent heat transfer, relatively simple design, high breeding potential, low T_2 inventory and an established on-line T_2 recovery. The use of liquid metals is usually perceived as a disadvantage by everyone except those who work with it on a daily basis, e.g., the operators of FFTF (Fast Flux Test Facility). Another disadvantage in magnetic fusion is the presence of MHD forces which is not an issue in ICF since there are no magnetic fields.

Two driver scenarios have been considered, a conservative design using a 2 MJ driver and a target gain of 50, and an optimistic design using a 1 MJ driver and a gain of 100. We have assumed a short wavelength KrF laser at 10% efficiency and reasonably well established costing algorithms. Using a 100 MJ yield we have estimated the cavity radius requirement based on an evaporation rate of 1 mm per year of an actively cooled dry graphite first wall in a cavity filled with xenon gas at 1.0 torr. For a rep-rate of 10 Hz the minimum cavity radius is 3.7 m if no redeposition is assumed [3]. If credit is taken for redeposition based on the first wall fraction used up by beam ports, the cavity radius goes down to ~3.1 m. We have thus decided to take an optimistic case cavity radius of 3.0 m and a conservative case of 4.0 m. Table 1 gives the parameters for the two cases.

Cavity Configuration and Maintenance

The reactor cavity is spherical and has 92 beams symmetrically distributed around it. Each beam is located in the center of a blanket module of which there are 80 hexagonally and 12 pentagonally shaped. The first wall is composed of 1.0 cm thick graphite tiles made of a three-dimensional weave composite, bonded to an actively cooled vanadium structure. The tiles are shaped to conform to the blanket module to which they are attached by means of a collar which fits into the beam port. Coolant supply and return

Table 1. Sirius-T Cavity Parameters

	3m IR		4m IR	
Driver Energy (MJ)	2	1	2	1
Target Gain	50	100	50	100
Target Yield (MJ)	100	100	100	100
Rep. Rate (Hz)	10		10	
Pulse Width (ns)	10		10	
Driver Efficiency (%)	10		10	
Xenon Pressure (torr)	1.0		1.0	
Blanket Thickness (m)	1.0		1.0	
Fusion Power (MW)	1000		1000	
Thermal Power (MW_{th})	1333		1333	
Power Cycle Eff. (%)	36		36	
Tritium Breeding Ratio	1.826		1.798	
Annual T_2 Surplus (kg/y)	32.6		32.1	

connections are contiguous with the tile connection. Figure 1 shows the cavity located within the reactor containment building which is also spherical. It will be noted that there are no beam tubes connecting the cavity with the final mirror, rather the laser beams are focused by the final mirrors across the vacuum space through the beam ports in the reactor cavity. The cavity is supported on a cylindrical pedestal which has perforations in it to allow laser beams to penetrate to the ports trapped within the cylinder. The 92 blanket modules are supported by a structural frame shown in Fig. 2. The structural beams which make up the frame are tapered toward the cavity center such that the modules, which are inserted from the outside, converge at the first wall. The cavity is supported on the outer perimeter of a cluster of seven hexagonal modules requiring a convoluted structural cone which joins the support cylinder to the cavity frame. We have used several analytical methods to determine the amount of structure needed in the frame to support the weight of the cavity when filled with all the breeding material. Finite element analysis will be performed on the final design. Material fractions have been determined as a function of cavity radius and blanket thickness and these were used in the neutronics analysis.

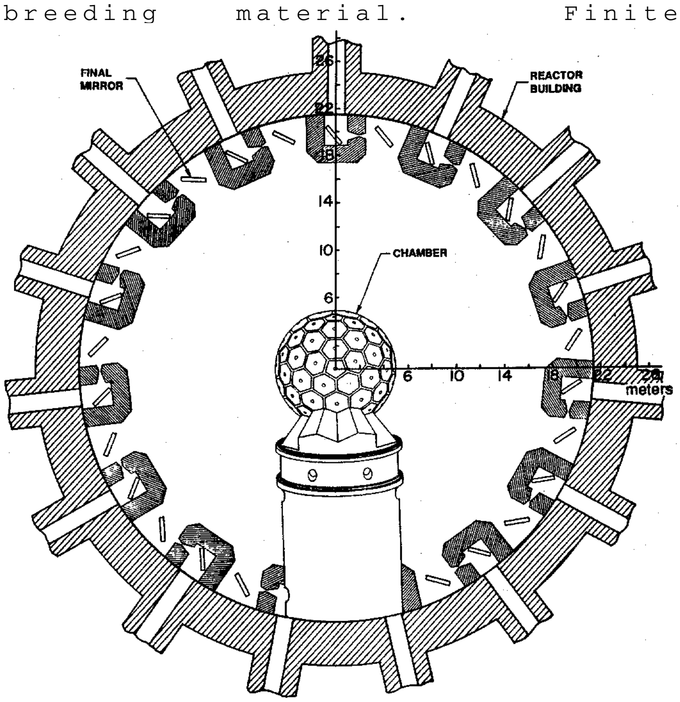

Fig. 1. Cross Section of Reactor Building.

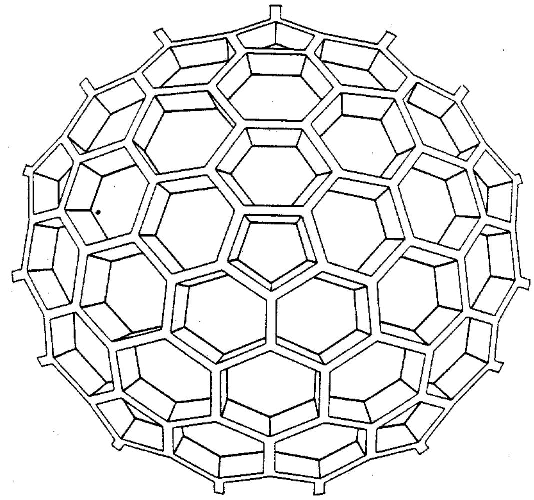

Fig. 2. Cavity Structural Frame.

Figure 3 is a cross section of a hexagonal blanket module. It shows a vanadium canister filled with Be discs separated by spacers and capped in the back by a mounting flange. The first wall consists of graphite tiles brazed to an actively cooled vanadium backing and mounted by means of a collar to the beam port. The Be discs are not continuous but are segmented to avoid problems with swelling and distortion. Each module has a single inlet and a single outlet connection. As shown in the figure, the Be blocks have holes in them which line up with the inlet and outlet connections. The incoming coolant distributes itself between the discs and travels by sheet flow around the module towards the outlet. The thickness of the Be blocks and the spaces between them are adjusted to maintain a uniform coolant temperature. Although not shown in the figure, coolant is also channeled to the vanadium backing for the graphite tile to provide active cooling for the first wall.

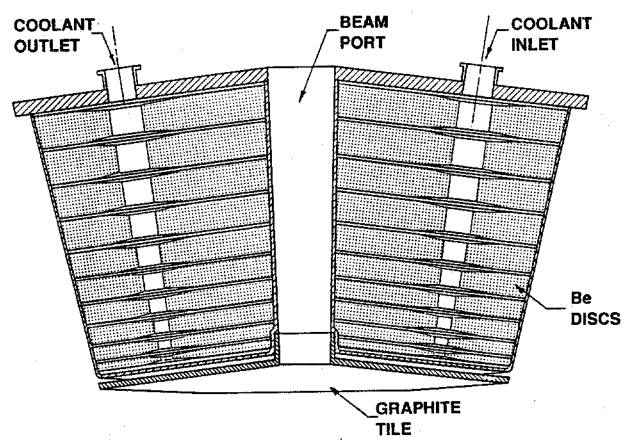

Fig. 3. Cross Section of Hexagonal Module.

Coolant pipes are routed along the outside surface of the cavity following along the structural web layout of the support frame. From there, connections are made to the blanket modules. Blanket modules are removed from the cavity by disconnecting the coolant lines, disengaging the mounting flange from the cavity frame and extracting them out radially. The support cylinder has rails mounted on it as shown in Fig. 1 for a remote maintenance machine which can access any point in the cavity with the exception of modules inside the cylinder. Provision is also made for removing and replacing first wall tiles from inside the cavity. For this, one or more modules are removed from below (within the support cylinder) and a remote maintenance machine inserted through the opening.

Neutronics Analysis

The neutronics analysis performed for SIRIUS-T is aimed at optimizing the blanket design to maximize the tritium breeding ratio (TBR). A self-cooled liquid lithium blanket with the vanadium alloy V-15Cr-5Ti structure is utilized in SIRIUS-T. Beryllium is used as a neutron multiplier. The neutronics calculations have been performed using the one-dimensional discrete ordinates code ONEDANT [4] with neutron cross section data based on the ENDF/B-V evaluation. The chamber is modeled in spherical geometry with the target at the center emitting neutrons with energy distribution given by the SIRIUS target spectrum.

Based on the mechanical design and structural analysis, the structure and void fractions in the blanket and reflector have been determined as a function of the inner radius (IR), outer radius (OR) and web thickness (t). The remaining space in the blanket is assumed to be filled with beryllium plates at a packing fraction of 62% with the liquid lithium flowing in between. The beryllium is assumed to have a 0.9 density factor. Graphite plates are utilized in the reflector. The remaining space in the reflector determined after excluding the structural support webs and void fractions is assumed to consist of 75% graphite, 5% structure, and 20% Li. 1 cm thick graphite tiles are used in the calculations followed by 1 cm back structure that has 50% Li cooling. The blanket has a front wall with one eighth the web thickness and is separated from the tile back structure by a 1 cm thick void region.

The TBR maximizes at a lithium enrichment of 20% ^{6}Li. The first 3 cases in Table 2 give the effect of inner chamber radius (IR) on TBR. The module thickness (OR-IR) is 0.8m, the Li is enriched to 20% ^{6}Li and the reflector thickness is 0.2 m. The TBR is slightly higher for a smaller chamber due to the smaller web thickness that leads to thinner module front. The effect of module thickness on TBR is also given in Table 2. The TBR increases as the module thickness increases with only modest increases beyond 0.8 m. The last two cases indicate that the TBR is the highest with no reflector. For 100 MJ yield and 10 Hz repetition rate, a 0.01 change in TBR corresponds to ~0.55 kg change in tritium production per FPY.

A simple cost tradeoff analysis has been performed to determine whether maximizing TBR results in net economic gain regardless of increased chamber cost. The results indicate that the chamber cost increments required to increase the TBR are several orders of magnitude lower than the achieved increment in the value of tritium produced. Based on this, the reference SIRIUS-T blanket design has an inner radius of 3 m, an outer radius of 4 m, no reflector, 0.06 m web thickness and 20% ^{6}Li. This yields the highest achievable TBR of 1.826.

Table 2. Tritium Breeding Ratio Obtained for Different Blanket Design Options (Li Enriched to 20% ^{6}Li).

Inner Radius IR(m)	Outer Radius OR(m)	Web Thickness t(m)	Reflector Thickness Δ_r(m)	TBR
3	3.8	0.06	0.2	1.778
4	4.8	0.08	0.2	1.751
5	5.8	0.10	0.2	1.725
3	3.5	0.06	0.1	1.554
3	3.6	0.06	0.15	1.663
3	3.7	0.06	0.2	1.728
3	3.9	0.06	0.25	1.802
3	4.0	0.06	0.3	1.818
3	4.0	0.06	0	1.826

Tritium Consideration

SIRIUS-T produces 1000 MW of fusion power requiring 1.8×10^{-3} g/s of tritium. The breeding rate at a TBR of 1.8 is 3.2×10^{-3} g/s yielding a surplus of 1.4×10^{-3} g/s which at a 75% availability is equal to 32.6 kg/y. The tritium is bred in Li (20% ^{6}Li) which exchanges heat with a He gas intermediate loop finally going to a steam generator and a conventional power cycle.

Tritium recovery from Li is by means of a fused salt contactor containing LiT [5]. After the T_2 is transferred to the fused salt, the salt is electrolyzed to recover the T_2. Fifty percent T_2 removal is achieved with a 1:4 Li/fused salt ratio, requiring a processing rate of 0.3% of the total Li flow. Some T_2 permeates the intermediate heat exchanger into the He stream. The vanadium heat exchanger tubes are coated with a 1.6μ coating of palladium. At an O_2 partial pressure of up to 10 Pa, the Pd coating becomes saturated with oxygen and as the tritium permeates through the tubes, it is oxidized to T_2O. The T_2 is then absorbed on a molecular sieve leaving only 0.6 g of T_2 in the He gas. Tritium permeating to the steam generator is <10 Ci/d. Table 3 gives the T_2 inventory and potential release.

Table 3. Tritium Inventory and Release

Location	System	T_2 (g)	Release Routine	Release Accident
Reactor Bldg.	Fuel	5.4		5.4
	Atmos.	0.5	~2 Ci/d	0.5
	Surfaces	<0.1		0.1
Reactor Cavity	Graphite Tiles	0.6		0.6
Breeder System	Liquid Li	55		(drain)
	Li-TRS	0.1	2 Ci/d	0.1
	He Loop	0.6		0.6
	He-TRS	5	2 Ci/d	5.0
Steam Generator		<0.1	10 Ci/d	0.1
TOTAL		67.4	~16 Ci/d	12.4

Max. Accidental Dose at Site Boundary = 0.6 Rem <1 Rem; exempt from NRC Guidelines for site evacuation plan

Economics

The code "FUSCOST" [6], a PC based menu driven program for economic analysis of fusion facilities, was used to evaluate the cost of T_2 production in SIRIUS-T. Costing algorithms were taken from the latest available data presented in the GENEROMAK [7] and SAFIRE [8] codes.

We assume 100% debt financing at an interest rate varying from 5% to 10%, and inflation and escalation rates of 6%. Construction, home office and field office factors are each 15%, owners cost factor 5% and project contingency 10% of the total direct costs. The costs are given in constant 1986 dollars.

Table 4 gives the costs for the 3 m and 4 m radius cavities both for the 1 MJ and 2 MJ drivers at 5% interest rate. As might be expected the target factory and the driver costs dominate. The total direct costs for the 2 MJ driver are 1329 M$ and 1398 M$, and the T_2 production costs are $7325/g and $7794/g for the 3 m and 4 m cavities respectively. The same costs are ~11% lower for the 1 MJ driver. In general production costs are ~70% higher for the 10% interest rate, reaching $13,200/g for the 4 m cavity and 2 MJ driver. It is also interesting to note that the driver efficiency does not affect the T_2 production cost until it falls below ~3.5% after which the cost increases exponentially, due to the fact that electricity has to be purchased to operate the driver.

Summary and Conclusions

A symmetrically illuminated ICF T_2 production facility has been scoped out utilizing a spherical cavity with 92 beams provided by a KrF laser. Lithium in a vanadium structure blanket is used, achieving a TBR of 1.83. Preliminary economic analysis

Table 4. SIRIUS-T Tritium Production Costs (5% Int., 8 yr. Const.)(M$ 1986)

	3m Cavity 2 MJ	3m Cavity 1 MJ	4m Cavity 2 MJ	4m Cavity 1 MJ
Target Factory	394	394	394	394
Driver	318	191	318	191
Reactor Chamber	112	112	180	180
Turbine Plant	111	111	111	111
Electric Plant	96	78	96	78
Buildings	83	83	83	83
Heat Transfer	78	78	78	78
Maintenance Eq.	36	36	36	36
Miscellaneous Plant	32	32	32	32
Heat Rejection	22	20	22	20
Instr. & Control	22	22	22	22
Fuel Handling	20	20	20	20
Land Acquisition	5	5	5	5
Total Direct Costs	1329	1181	1398	1250
Total Capital Costs	2675	2377	2813	2516
Annual O & M	66	58	69	62
Cost of T_2 Prod. ($/g)	7325	6491	7794	6974

shows that such a system can produce tritium at a very competitive cost under reasonable costing assumptions.

Acknowledgment

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