SIRIUS-T – An Advanced Tritium Production Facility Utilizing Symmetrically Illuminated Inertial Confinement Fusion

I.N. Sviatoslavsky, M.E. Sawan, G.A. Moses, G.L. Kulcinski, R.L. Engelstad, E.M. Larsen, E.G. Lovell, J.J. MacFarlane, E. Mogahed, R.R. Peterson, L.J. Wittenberg, J. Powers

October 1990

UWFDM-835

Presented at the 9th Topical Meeting on the Technology of Fusion Energy, 7–11 October 1990, Oak Brook IL.

FUSION TECHNOLOGY INSTITUTE

UNIVERSITY OF WISCONSIN

MADISON WISCONSIN

SIRIUS-T – An Advanced Tritium Production Facility Utilizing Symmetrically Illuminated Inertial Confinement Fusion

I.N. Sviatoslavsky, M.E. Sawan, G.A. Moses,
G.L. Kulcinski, R.L. Engelstad, E.M. Larsen,
E.G. Lovell, J.J. MacFarlane, E. Mogahed, R.R.
Peterson, L.J. Wittenberg, J. Powers

Fusion Technology Institute
University of Wisconsin
1500 Engineering Drive
Madison, WI 53706

http://fti.neep.wisc.edu

October 1990

UWFDM-835

Presented at the 9th Topical Meeting on the Technology of Fusion Energy, 7–11 October 1990, Oak Brook IL.

SIRIUS-T, AN ADVANCED TRITIUM PRODUCTION FACILITY UTILIZING SYMMETRICALLY ILLUMINATED INERTIAL CONFINEMENT FUSION

I.N. Sviatoslavsky, G.L. Kulcinski, G.A. Moses, M.E. Sawan, R.L. Engelstad, E. Larsen
E. Lovell, J. MacFarlane, E. Mogahed, R.R. Peterson, J.W. Powers, L.J. Wittenberg
Fusion Technology Institute
University of Wisconsin-Madison
1500 Johnson Drive
Madison, WI 53706
(608) 263-6974

ABSTRACT

SIRIUS-T is a study of an advanced tritium production facility which utilizes direct drive symmetric illumination inertial confinement fusion provided by a KrF laser. Symmetrically illuminated reactor systems have some very unique problems which have to do with a large number of beams. In SIRIUS-T, a single shell ICF target is illuminated by 92 symmetrically distributed beams around a spherical cavity of 4 m radius. The driver energy is 2 MJ and the target gain 50. The first wall consists of graphite tiles bonded to an actively cooled vanadium structure. There is a 1.0 torr xenon buffer gas in the cavity. The structural material is the vanadium alloy V-3Ti-1Si, the breeding/cooling material is lithium 90% enriched in Li-6 and the neutron multiplier is Be, giving a tritium breeding ratio of 1.903. The total tritium inventory in the reactor is 184 g. A routine release of 29 Ci/d is estimated and the maximum accidental release is 19.9 g. At 100 MJ yield, a repetition rate of 10 Hz and an availability of 70%, a tritium surplus of 33.3 kg per calendar year is achieved. Using 100% debt financing, and a 30 full power year (FPY) reactor lifetime, the cost of tritium production is \$8,885/g at 5% interest on capital and \$14,611/g at 10% in 1990 dollars.

INTRODUCTION

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 mainline indirect drive targets which do not require symmetric illumination, presently being developed by various national laboratories. As part of this program, the Fusion Technology Institute (FTI) of the University of Wisconsin, 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. An earlier study is a scoping investigation of a materials test facility SIRIUS-M.[1] The present paper gives the final results and conclusions of a similar scoping study of a tritium production facility SIRIUS-T.

Nuclear fusion facilities are ideally suited for T_2 production due to the excess neutrons they provide relative to the thermal energy released. In this respect, they are far superior to fission production reactors. A properly designed fusion reactor will generate tritium ten times faster than a fission reactor of comparable thermal energy optimized for tritium production.

The aim of the present study is to design a symmetrically illuminated ICF reactor using conservative physics, near term technology and credible economic analysis, to produce tritium at the lowest possible cost. Many breeding options, including solid ceramic and solid metallic breeders, as well as liquid metals were evaluated. A system utilizing liquid lithium in a vanadium alloy structure, with heavy reliance on neutron multiplication by beryllium metal was selected. Symmetric illumination schemes are very limited in the ways they can protect the first wall (FW). Wetted walls for example would be impossible to incorporate into a chamber with a very large number of beams. We have selected a dry wall scheme utilizing actively cooled graphite tiles and a 1.0 torr pressure of xenon gas which acts to stop x-rays and ion debris. A spherical chamber seems to be most suitable for symmetric illumination from the standpoint of the strategic placement of a large number of beams. Thus, the chamber is spherical of 4 m inner radius, and has 92 beam ports symmetrically distributed. In the next section the chamber and the other systems will be described in more detail.

OVERALL DESCRIPTION

Figure 1 shows the SIRIUS-T spherical chamber at the center of the spherical containment building. The building cutaway shows 15 of the 92 beams which surround the cavity in a symmetric distribution. There are no beam tubes between the chamber and the final focusing mirror. Rather, the final focusing mirrors which are mounted on the wall of the containment building direct the beams through ports in the walls of the chamber. The beams converge in the center of the chamber which is where they illuminate the target. There are several implications to this mode of containment. The chamber shares the same atmosphere as the containment building.

Thus, the gases after the shot are exhausted into the building through the beam ports from where they are pumped out. There is a vacuum barrier in the form of a window at the outer containment building wall in each of the beam tubes. The beam energy density at the window is very low. Unburned T_2 is confined to the building from where it is pumped out. The beams entering through the windows are focused through a shielded cross-over point onto the final focusing mirrors. The function of the cross-over is to minimize neutron streaming through the beam tubes out of the containment building. The containment building which has a 3.2 m thick steel reinforced concrete wall has a biological dose of 2.5 mr/h at the outer surface during reactor operation. The chamber is supported on a hollow steel pedestal as shown in Figure 1. The supports make contact with the frame on a perimeter which traps six of the modules within the pedestal. Holes in the walls of the pedestal make it possible for the beam to reach the beam ports within the pedestal. Rails along the cylindrical portion of the pedestal are for a maintenance machine which can circumnavigate the reactor, reaching all the modules outside the pedestal.

The driver is a 2 MJ KrF laser with a rep-rate of 10 Hz and a 10 ns pulse. A conservative target gain of 50 is used with a direct drive single shell target. This combination yields 1000 MW of fusion power. The Li breeding/cooling material exits the chamber at 550°C and goes to a power cycle which converts the thermal energy to electricity at 36% efficiency. The efficiency of the driver can go down to 5% before the system ceases to become electrically self-sufficient. Lowering the efficiency of the driver down to 5% has a minimal impact on the cost of T_2 production. Table 1 gives the important SIRIUS-T parameters.

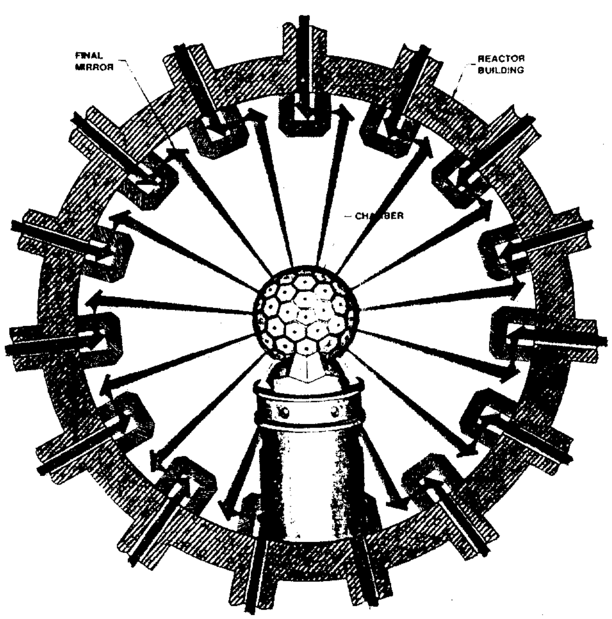

Figure 1. View of SIRIUS-T reactor.

Table 1. SIRIUS-T Parameters.

Driver energy	2 MJ
Target gain	50
Driver efficiency	5-10%
Repetition rate	10 Hz
Pulse length	10 ns
Fusion power	1000 MW
Thermal power	1410 MW
Electrical conversion efficiency	36%
Availability	70%
Chamber radius	4 m
Number of beams	92
Breeding ratio	1.903
Li inlet temperature	350°C
Li outlet temperature	550°C
Maximum structural temperature	650°C
Containment building radius	22 m
Distance to FF mirror	20 m
Annual T_2 production	33.3 kg

REACTOR CHAMBER DESIGN

The reactor chamber is spherical with an inner radius of 4 m and an outer radius of 5 m. The FW consists of a 1 cm thick carbon/carbon composite brazed to a 1 cm thick V-3Ti-1Si vanadium alloy backing which is actively cooled with lithium. The shape of these FW tiles conforms to the shapes of the blanket modules behind them on which they are supported. There are 92 blanket modules, 80 hexagonal and 12 pentagonal all fitting around the spherical chamber within a structural support frame. Figure 2 shows a view of the structural frame. The bodies of the modules and the structural frame are made of V-3Ti-1Si.

Figure 3 is a cross-section of a module showing the internal distribution of the breeding materials. Unclad Be discs of varying thickness are stacked on top of each other with coolant channels machined into the surfaces between the discs. The thickness of the discs and the coolant distribution is optimized to maximize tritium breeding while maintaining the maximum vanadium alloy structural temperature at <650°C. The incoming lithium coolant enters the module from a supply manifold down a tapered hole in the Be discs. The coolant distributes itself radially, then travels through coolant channels circumferentially between the discs, finally emerging through another tapered hole to the return manifold. Some of the incoming lithium is diverted for cooling the FW tile which is an integral part of that module. This lithium travels down one side of the annular space which surrounds the beam port, then flows through the vanadium alloy backing of the FW tile, then returns through the other side of the annular space to rejoin the exiting hot lithium on its way to the return manifold. In this way only one supply and one return connection is needed to hook up the module to the coolant as shown in Figure 3.

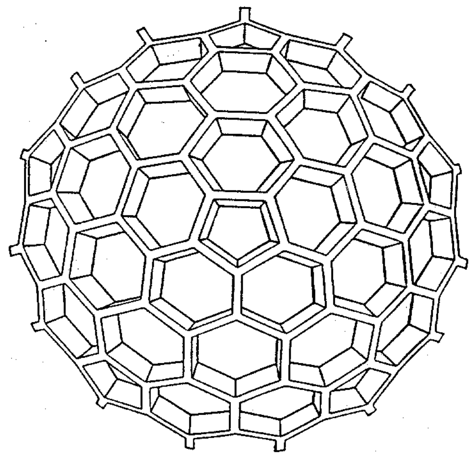

Figure 2. Cavity structural frame.

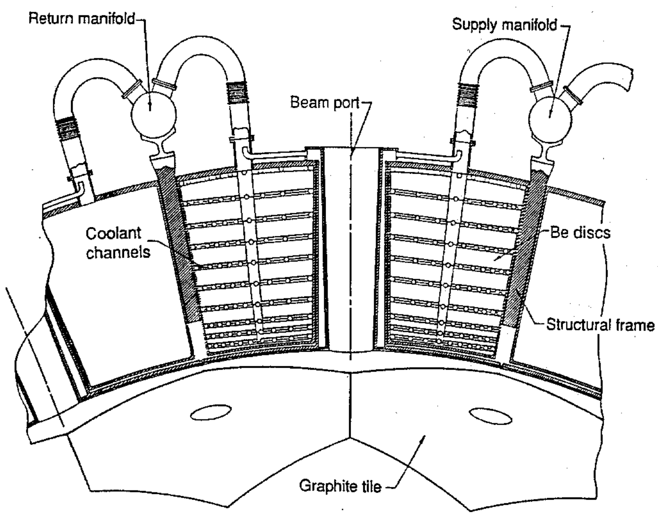

Figure 3. Cross-section of hexagonal module.

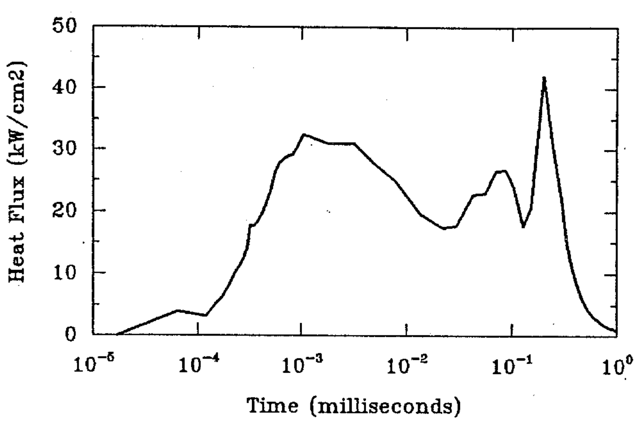

Figure 4. Heat flux on graphite tiles.

Neutronics analysis was performed using the 1-D discrete ordinates code ONEDANT with cross section data based on the ENDF/B-V evaluation. Spherical geometry was used, with the target at the center emitting neutrons at a given energy spectrum. A conservative estimate of the tritium breeding ratio (TBR) is determined using a homogenized composition of each blanket layer taking into account the beam penetrations and the structural frame. The TBR was maximized by varying the Li enrichment, the thicknesses of the alternating layers of Be and Li, and the total blanket thickness. The blanket thickness is 1 m and the Li enrichment 90% ^{6}Li. This yields an overall TBR of 1.903 and an energy multiplication of 1.39. The peak displacement per atom (dpa) rates in the graphite tiles and the vanadium alloy structure are 18 and 30 dpa per full power year (FPY), respectively.

Nuclear heating values obtained from the neutronic analysis are iterated with thermal hydraulic analysis to insure proper temperature distribution and limits. The inlet lithium temperature is 350°C and the outlet 550°C. The steady state, two dimensional thermal hydraulics and stress analysis code ANSYS was used to determine the temperature distributions. Heat transfer between the back of the first wall tile and the front of the module was by radiation. The maximum vanadium structural temperature is 650°C. Details of the neutronics and thermal hydraulics analyses are presented in a companion paper in these proceedings.[2]

The structural frame was analyzed using finite element techniques with the code ANSYS. A worst case scenario was envisaged by assuming that a whole row of blanket modules are removed immediately above the chamber support perimeter. Both in plane and out of plane loads were determined on the frame structure at these points while loaded by the dead weight of the remaining modules. The maximum tensile and compressive stresses were <76 MPa giving a safety factor of ~5.6 relative to the yield strength of V-3Ti-1Si of 429 MPa at 200°C, the temperature of the frame at the time maintenance is performed. The stresses are much lower when all the modules are in place. A maximum deflection of <2 mm was obtained showing that module replacement will not be a problem. Web buckling is also not an issue since these stresses are far below the values for structural instability. These details are also reported in a companion paper in these proceedings.[3]

GRAPHITE TILE TEMPERATURE AND STRESS RESPONSE

Graphite has been chosen for the FW tiles because of its high sublimation temperature and low vapor pressure. The target emits soft x-rays in a sub-nanosecond pulse and debris ions, both of which are stopped and deposit their energy in 4 torr-meters of xenon fill gas between the target and the FW. The gas reradiates the energy to the tiles in the form of low energy thermal photons over a typical time scale of a millisecond, allowing thermal conduction to keep the graphite temperature below the sublimation limit. The graphite tiles have been designed taking into account thermal stress, evaporation and neutron damage induced changes in graphite thermal conductivity.

A computer simulation is used to study the re-radiation of energy from the fill gas to the tiles and their thermal response. We have used the CONRAD[4] computer code to simulate energy deposition in the gas and re-radiation to the tiles. It is found that at least 180 energy groups are required in the multigroup radiation diffusion calculation. The resulting heat flux on the tiles is given in Figure 4 which shows that the heat reaches the FW in three bursts of increasing energy. The temperature profiles in the graphite are calculated by a 1-D finite-difference computer code. Three sets of temperature dependent properties have been used: (1) unirradiated graphite H-451,[5] (2) the same (H-451) irradiated with 10^{22} neutrons/cm^2 and (3) unirradiated woven carbon/carbon composite.[6] We assume the back graphite surface is kept at 960 K, which is consistent with an exit coolant temperature of 773 K, a film rise of 24 K and a 163 K rise across the vanadium. The calculated surface temperatures are shown in Figure 5 as a function of time. The unirradiated graphites show similar behavior, peaking at ~1500 K while the irradiated graphite peaks at ~1950 K due to the degraded thermal conductivity. Figure 6 shows the compressive strength and stress for the irradiated graphite as a function of time. The curves indicate that there is a safety margin of two between the strength and the stress, providing confidence in the viability of the design.

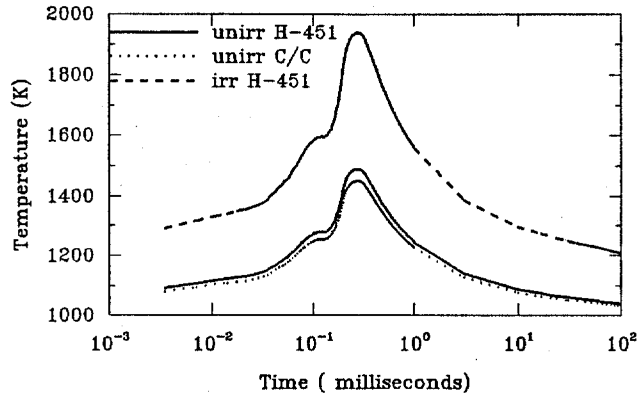

Figure 5. Graphite temperature response.

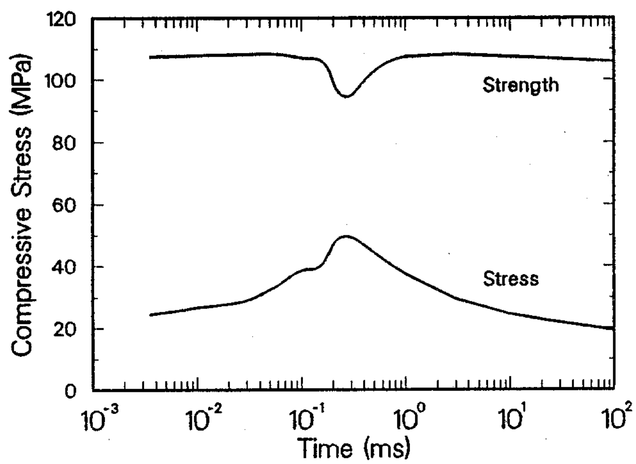

Figure 6. Graphite thermal stress.

MATERIAL ISSUES

The material lifetimes of the vanadium alloy, the graphite tiles and the final focusing mirrors have been estimated from existing data. Graphite lifetime is determined by the damage level at which the graphite passes through the shrinkage phase and crosses the zero dimensional change axis on the way to "runaway" swelling. The peak damage level in the FW graphite is 18 dpa/FPY at an average equilibrated temperature of ~727°C. As the graphite is irradiated its thermal conductivity degrades and the average temperature creeps up to 800°C. Swelling due to radiation damage as a function of temperature varies dramatically for different graphites. Some graphites cross the zero dimensional change axis at ~30 dpa at 800°C.[7] Based on this we have projected a 2 FPY lifetime for the graphite in the FW tiles.

In the vanadium alloy V-3Ti-1Si the need for replacement comes due to loss of ductility rather than swelling. It has been shown that at 420°C with 350 atomic parts per million (appm) of preinjected helium and 82 dpa, this alloy swells 1%.[7] The front part of the module will be operating at an average temperature of ~600°C, and the swelling will be much lower. The peak damage at this point will be ~24 dpa/FPY and helium generation, 70 appm/FPY. Braski[8] has shown that at 420°C and 82 dpa with 350 appm of preinjected He, the total elongation for this alloy goes down to ~3%. At 600°C the elongation will increase to about 3.5-4%. On this basis we have projected a 4 FPY lifetime for the modules. The structural frame which is also made of V-3Ti-1Si starts at a point 35 cm behind the FW where the damage level is 3.5 dpa/FPY and 5 appm of He/FPY, and the average temperature is 600°C. Under these conditions, we project a 30 FPY lifetime for the frame which means that it is a full lifetime component.

The lifetime of the final focusing mirrors is determined by damage to the color centers of the coating with a recommended dose of ~10^{11} rads. Partial recovery of the damage by annealing is achievable, but the degree of recovery has not been established. At 20 m distance from the target, the dose rate for the final focusing mirror is 5×10^{11} rads/FPY and the first annealing will be needed in 0.2 FPY. If an 85% recovery can be achieved, then a 1 FPY lifetime can be projected if the time between anneals is at least one month. Annealing the mirrors can be performed in a very short time by means of resistive elements built into the bodies of the mirrors.

TRITIUM CONSIDERATIONS

At a 1000 MW of fusion power, an availability of 70%, a TBR of 1.9 and a TBR of 1.05 required to operate the reactor, the tritium surplus is 33.3 kg per calendar year. A He gas intermediate loop is used between the Li and the steam generator.

Tritium recovery from the Li is by means of a fused salt containing LiT.[9] After the T_2 is transferred to the salt it is recovered via electrolysis. The vanadium alloy tubes in the Li/He heat exchanger are coated with 1.6 μ of palladium on the He side. If a 10 Pa partial pressure of O_2 is maintained in the He, the palladium becomes saturated with O_2 and the permeating tritium is oxidized to T_2O. It can then be absorbed on molecular sieves leaving a very small amount of T_2 in the He. In this way T_2 permeation into the steam generator is kept to 23 Ci/d and the total routine release from the reactor, to 29 Ci/d. The total T_2 inventory in the reactor is 184 g of which 132 g is in the lithium as a stable tritide. A maximum accidental T_2 release of 19.9 g produces a dose at the site boundary of <1 Rem thus exempting a site evacuation plan according to NRC guidelines.

REACTOR MAINTENANCE

It was determined that the FW tiles have to be replaced every 2 FPY while the bodies of the blanket modules, every 4 FPY. Thus, every 34 calendar months (2 FPY at 0.70% availability), the reactor will be shut down for a period of time during which one half of the modules and all the FW tiles will be replaced. The Be will be salvaged from the spent modules and used again to fabricate new ones.

Figure 1 shows a set of rails on the support pedestal for a remote maintenance machine which will circumnavigate the reactor 360°. All the blanket modules outside the pedestal can be accessed by this machine for replacement. Figure 7 shows a module being removed from the outside of the reactor. Note that the tile comes out with the module. Two coolant connections have to be uncoupled and the module unfastened from the structural frame. Similarly a FW tile can be removed from the inside of the chamber. For this operation another remote handling machine is inserted through a removed module penetration from within the pedestal. By having the rear closure flange removed from the outside, the tile is then free to be extracted into the chamber for replacement.

Replacing the final focusing mirrors will be done on an 8.5 calendar month schedule when 46 of the mirrors will be replaced. Several special purpose remote handling machines will operate simultaneously aided by the main machine riding on the pedestal tracks. The special purpose machines index onto a turning mirror shield module and

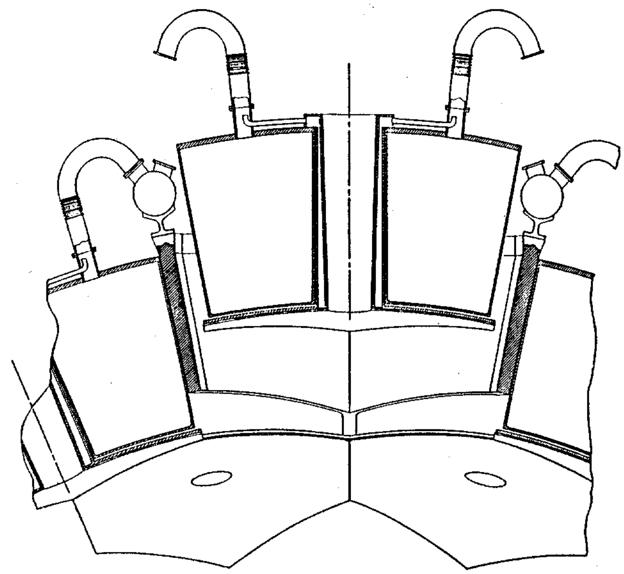

Figure 7. Removal of blanket module.

attach themselves to it. From that perch, they can service several mirrors at a time. Spent mirrors are handed to the main machine which puts them into a retrieval hatch, then takes new mirrors from a supply hatch and hands them to the special purpose machines. Most of the operations will be repetitive and automatic. Recognition of the station being serviced will automatically program the location, distance and orientation needed for the main machine to handle the mirrors.

ECONOMIC ANALYSIS

Tritium production cost analysis has been performed with the code "FUSCOST",[10] a PC based, menu driven program for economic evaluation of fusion facilities. Costing algorithms are the latest available for use in reactor evaluation. We have assumed a 100% debt financing at interest rates between 5-10%. Construction time is 8 years and plant amortization is 30 FPY at an availability of 70%. Construction factor, home office factor and field office factor are each 15%, owner's cost factor 5% and project contingency 10% of total direct costs. Targets are priced at $0.15 each, which is consistent with projections for mass produced single shell targets without a high Z coating on the outside. Operation and maintenance is taken as 3% of total overnight costs. Annual component replacement cost is based on a 2 FPY life for the tiles, 4 FPY life for the module bodies and 1 FPY year life for the final focusing mirrors. A 50% additional Be stock was included in the original purchase to account for the initial replacement of one half of the modules. Annual replacement of the remaining reactor components is taken as 1% of total direct costs. Table 2 summarizes the costs in 1990 dollars.

The largest cost item is the reactor chamber followed closely by the driver. The total direct cost for both 5 and 10% interest rate is 1413 M$. Total overnight costs have 50% of the construction cost factor, home office and field office, and owners cost added. A 10% contingency is then added to the total sum to round out the overnight costs. The total capital costs include interest during construction and here the values differ for the two interest rates. The bottom line is that the cost of T_2 production in constant 1990 dollars is $8,885/g and $14,611/g for the 5% and 10% interest rates respectively, based on 33.3 kg of T_2 produced per calendar year.

Table 2. SIRIUS-T Costs

Interest Rate	Costs in M$ (1990) 5%	10%
Reactor chamber	411	411
Driver	375	375
Turbine plant	136	136
Heat transfer equipment	115	115
Electric plant	115	115
Buildings	98	98
Maintenance equipment	43	43
Miscellaneous plant	39	39
Heat rejection	26	26
Instrumentation and control	26	26
Fuel handling	24	24
Land acquisition	6	6
Total direct costs	1413	1413
Total overnight costs	2332	2332
Total capital costs	2838	3465
Annual operation and maintenance	70	70
Annual component replacement	30	30
Annual target costs	35	35
Annual principal and interest	161	352
Total annual payment	296	487
Cost of T_2 production $/g (1990)	8,885	14,611

Sensitivity to target cost and driver efficiency is low. If the target cost is increased to 20¢, T_2 cost increases by 4% while reducing the driver efficiency from 10-5% results in an increase of ~2%.

SUMMARY

A tritium production facility based on symmetric illumination direct drive inertial confinement fusion is a very attractive option producing T at ~9000-15,000 $/g depending on interest on capital.

ACKNOWLEDGEMENT

Support for this research has been provided by the Inertial Fusion Division, Office of Research and Advanced Technology, U.S. Department of Energy.

REFERENCES

1. S. ABDEL-KHALIK et al., "Design Considerations for the SIRIUS-M ICF Materials Test Facility," 12th Symposium on Fusion Engineering, Monterey, CA (1987).

2. M. SAWAN and E. MOGAHED, "Neutronics and Thermal Analyses for the Breeding Blanket of the ICF Tritium Production Reactor SIRIUS-T," these proceedings.

3. J. POWERS et al., "SIRIUS-T Structural System Design and Analysis," these proceedings.

4. R.R. PETERSON, J.J. MacFARLANE and G.A. MOSES, "CONRAD - A Combined Hydrodynamics-Condensation/Vaporization Computer Code," University of Wisconsin Fusion Technology Institute UWFDM-670, July 1988.

5. G.R. HOPKINS et al., "Carbon and Silicon Carbide As First Wall Materials in ICF Fusion Reactors," GA-A14894, General Atomics (1978).

6. CLIFF BAKER, Program Manager, Fiber Materials Inc., Biddeford, Maine, personal communication (1990)

7. M. BIRCH and J. BROCKLEHURST, "A Review of the Behaviour of Graphite Under the Conditions Appropriate for Protection of the First Wall of a Fusion Reactor," ND-R-1434(S), UKAEA (1987).

8. D.N. BRASKI, "The Post-Irradiated Tensile Properties and Microstructure of Several Vanadium Alloys," ASTM-STP-1047 (1990).

9. W.F. CALAWAY, "Electrochemical Extraction of H_2 From Molten LiF, LiCl, LiBr and Its Application to Liquid Li Fusion Reactor Blanket Processing," Nucl. Tech., 39, 63 (1978).

10. Z. MUSICKI, "FUSCOST: A PC Based Menu-Driven Program for Economic Analysis of Fusion Facilities," 12th Symposium on Fusion Engineering, Monterey, CA, (1987).