ENGINEERING DESIGN OF THE SOLASE-H LASER FUSION HYBRID REACTOR

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Abstract

The University of Wisconsin fusion reactor design group has undertaken a study of the technological porblems posed by laser fusion hybrids and how they may be integrated into the fission industry. These problems have been examined in the context of a self-consistent reactor design, SOLASE-H.

In this article, the engineering design of SOLASE-H is described. This reactor has been designed so that it can be easily integrated into an economy of light water fission reactors with minimum alterations and technology developments in the present fission industry. LWR fuel bundles containing thorium oxide are irradiated in the hybrid until the proper fissile enrichment (3-4%) is reached. The bundles are then shipped directly to the LWRS where the fuel is burnt.

The merits and demerits of the proposed fuel cycle are discussed. Neutronic, thermal-hydraulic, and mechanical design aspects of SOLASE-H are presented.

1. Introduction

A fusion-fission hybrid reactor utilizes the energetic fusion neutrons for breeding fissile material in the fusion reactor blanket. The fissile material (U-233 or Pu-239) is produced by neutron capture in a fertile fuel (Th-232 or U-238) and is removed periodically from the blanket and burned in fission reactors. The fact that the fission process is "power rich" releasing 200 MeV/fission event, compared to 17.6 MeV/D-T fusion event, but "neutron poor" releasing 2 MeV fission neutrons compared to the 14.1 MeV D-T fusion neutron makes the coupling between these two systems more optimal than either one independently.

The hybrid blanket may be designed to minimize the "in situ" fissioning of the bred fissile material so that the hybrid would be operated primarily as a fuel factory in simbiosis with fission reactors [1-8]. Alternatively, the blanket may be designed to burn most of the bred fissile material in situ with the hybrid operating as a stand-alone power producer [9]. In either case, the fusion performance level rquired for the hybrid is less than that required for pure fusion reactors. The reduced fusion performance is allowable because the fusion neutron energy is multiplied in the hybrid blanket by the fission process. Such reduction in fusion performance requirements may allow the hybrids to be introduced at an earlier date than pure fusion reactors.

While hybrids appear to be a logical avenue for fusion to make an early impact on the world energy problem, they may be considered unattractive inasmuch as they may potentially suffer from the disadvantages of both fission and fusion systems. In an effort to quantify the merits and demerits of hybrids, a conceptual laser-fusion hybrid reactor design, SOLASE-H, has recently been completed [10]. The primarv objectives of the SOLASE-H study have been: (1) To determine how hybrids may alter the necessary laser fusion performance parameters required for an economical plant subject to the constraints imposed by pellet physics and reactor engineering, (2) To determine the ranges of system parameters over which hybrids will most likely operate, (3) To develop a hybrid concept which can be integrated into a mature fission industry, and (4) To self-consistently examine the performance of the main hybrid subsystems, viz., the blanket, cavity, and driver.

The SOLASE-H study covers five main topics: (1) Overall proliferation resistant fuel cycles, (2) Blanket neutronics and mechanical design, (3) Laser fusion performance requirements for hybrids, (4) First wall protection, and (5) Hydrogen fluoride laser design. In the following, only the first two topics will be discussed. Additional details about the SOLASE-H study may be found in ref. [10,11].

2. Fuel Cycle for SOLASE-H

A schematic diagram of the SOLASE-H fuel cycle is shown in Fig. 1. Fertile fuel, ThO_2 or UO_2, is fabricated in a form that is directly usable in a LWR (e.g. 17 x 17 PWR fuel assemblies). The fertile fuel assemblies are placed in the hybrid blanket until they are enriched to 3-4% fissile fuel as required by the LWR. Following enrichment, the bundles, which are now highly radioactive, are shipped directly to the LWRs for burning of the fuel. The spent fuel from the LWR is stored until a decision is made on reprocessing or disposal. If feasible, the spent fuel can be re-inserted into the hybrid to be re-enriched for further burning in the LWR. This possibility depends on both the importance of fission product buildup to LWR performance and the radiation damage to the fuel and cladding.

This concept is attractive inasmuch as it requires minimal modifications to the present LWR industry. It offers potentially tighter proliferation control since the fissile material in transit is contained in the highly radioactive fuel assemblies and is, therefore, diversion resistant [1, 11, 12]. Without reprocessing, it extends the fissile fuel resources by nearly an order of magnitude over the ^{235}U resources. This allows additional time for deliberate decisions to be made on such issues as internationally controlled, physically secure fuel production and reprocessing centers [11-13].

The major disadvantages of this approach are: (1) It does not take full advantage of the fertile fuel reserves, and (2) The economic feasibility of this concept is highly sensitive to the hybrid cost since the number of fission reactors to be supported by the hybrid is low. Without reprocessing, one hybrid reactor is only able to supply fissile fuel to about 2.5 LWRs of the same thermal power. This has the economic impact of increasing the effective fuel cost. With reprocessing of the spent LWR fuel, approximately 10 LWRs can be fueled from one hybrid of equivalent power, depending on the conversion ratio of the LWR.

The proliferation resistant fuel cycle can be extended to include reprocessing of the spent LWR fuel if one follows the structure outlined by Feiverson and Taylor of internationally controlled, physically secure fuel production and reprocessing sites combined with many national convertor reactors "outside the fence" as shown in Fig. 2. Fresh ThO_2 or UO_2 fuel is fabricated in assemblies that are directly usable in a LWR or other convertor reactor. The assemblies are irradiated in the hybrid until the proper fissile enrichment is reached. The enriched assemblies are then transferred directly to the fission reactors where the fuel is burnt. The spent fuel assemblies are returned to the physically secure site for reprocessing. The recovered fissile material is sent to the fuel factory within the secure site and is fabricated into <u>partially enriched</u> assemblies. These are irradiated in the hybrid until the proper enrichment is reached before shipment to the fission reactors so that all the fissile material in transit outside the secure site is in a highly radioactive form and is, therefore, diversion resistant. This approach overcomes the disadvantages of the no-reprocessing option while maintaining the proliferation-resistant nature of the fuel cycle.

3. SOLASE-H Cavity and Blanket Design

The main parameters for SOLASE-H are listed in Table I. Schematic diagrams of the cavity and blanket are shown in Figs. 3 and 4 respectively. The reactor cavity is an upright cylinder 6 m in radius and 12 m high with spherical end caps. The cylindrical portion of the cavity is surrounded by the fuel-producing radial blanket while the end caps (axial blankets) are used for tritium production. The fertile material is contained in standard (17 x 17) PWR fuel assemblies stacked in three layers around the cavity; a total of 528 assemblies are placed in the blanket.

The blanket structure is zircaloy to be compatible with the fuel cladding material. The first wall is 0.2 cm thick and is protected from the x-ray and ion debris of the pellet microexplosion by 0.5-1.0 torr of xenon gas circulated through the cavity [10, 11]. The first wall is scalloped as shown in Fig. 4 to accomodate the Na coolant pressure in the blanket and the pressure impulse generated by the microexplosion. Directly behind the first wall are pins of Pb, clad in zircaloy followed by the breeding zone (Fig. 4). The lead serves as a neutron multiplier to enhance the fissile production rate.

The zone containing LWR assemblies is surrounded in the front and rear with pins containing Li. These Li zones both breed tritium and filter thermal neutrons that might otherwise diffuse into the fuel assemblies and induce fission. By poisoning the thermal flux, they enhance the uniformity of enrichment across the LWR assembly. Behind the LWR fuel zone and its Li filter is a Pb and carbon reflector. The fuel zone is therefore surrounded by fast neutron reflecting material and thermal neutron filters. The assemblies behave as a fast neutron flux trap, thus maximizing the fissile fuel breeding rate. The reflector is followed by an outer Li zone to capture any leaking neutrons.

The radial blanket is cooled by sodium which enters at 300°C and leaves at 350°C except for the reflector and outermost tritium breeding zones which are helium cooled. The blanket is divided into four quadrants each with three segments containing twelve columns of the fuel assemblies. Two opposite quadrants contain the beam ports arranged vertically on each side. The axial blanket consists of a 2 mm thick first wall followed by a 50 cm thick lithium zone and lead/carbon reflector. The lithium serves as both a heat transport and tritium breeding medium.

Numerous neutronic calculations using the ANISN neutron transport code were done to optimize the radial blanket so that the fissile breeding rate is maximized with minimum enrichment gradients within the assemblies. For the final design shown in Fig. 4, the maximum to average enrichment is 1.1 with a ^{233}U enrichment of 4.7% at the edge and 3.75% in the middle. The time required to reach this enrichment is 2.7 years of exposure with a neutron wall loading of 2 MW/m^2. The fuel assemblies are rotated 180° at the end of 1.35 years to obtain a symmetric enrichment profile. A penalty is paid in the LWR for large values of maximum-to-average enrichment due to hot channel factors. The profile can be made flatter at the expense of reducing the fissile breeding ratio. Hence, a figure of merit defined as: $FM \equiv (UBR)^2/Th\ (n,\gamma)_{max}$ has been used as the criterion for blanket optimization, where UBR is the uranium breeding ratio. The quantity $Th(n,\gamma)_{max}/UBR$ is proportional to the hot channel factor penalty. For the base case blanket design the breeding ratio is 0.65 ^{233}U/fusion neutron. With this breeding ratio and 1200 MW of fusion power the hybrid produces ~2500 kg of ^{233}U per year, enough to fuel ~2.5 1000 MW_e LWRs without reprocessing.

Most of the blanket neutronics analyses were done using ANISN and assuming a one-dimensional, spherical blanket. A solid angle weighting of 70% is then applied to the results to account for the fact that the fissile fuel is only in the circumferential blanket in the cylindrical reactor. Once a near optimum detailed blanket configuration is determined, three-dimensional Monte Carlo calculations are performed on the entire blanket including the upper and lower tritium breeding regions. These calculations are done to determine the enrichment profile in the axial direction and to test the solid angle weighting approximation. This analysis shows that the upper and lower blankets can be strongly neutronically coupled to the circumferential blanket and hence the simple solid angle weighting technique must be cautiously applied. However, the total number of absorptions per fusion neutron is almost constant at 1.65. Therefore, a proper three-dimensional design can be established that will give the same results as the one-dimensional designs with solid angle weighting. Furthermore, alternate fuel assemblies can be replaced with scattering material and a thermal neutron filter so that the remaining assemblies are reduced in number and are surrounded by scattering material and thermal neutron filters. The three-dimensional analysis shows that this does not seriously reduce the total number of absorptions per fusion neutron but significantly reduces the fuel inventory. The fuel in the blanket is enriched more quickly, reducing the associated carrying charges [14].

Burnup calculations show that approximately 13% of the total fuel generated is consumed before it is removed from the blanket. This burnup is equivalent to 4300 MWd/MT. The power swing, due to changes in the blanket multiplication during enrichment, is 19%. The minimum power is 2400 MW_t and the maximum is 2900 MW_t. These values are for the equilibrium cycle where there are 4 different batches of fuel in the blanket. Therefore, the blanket contains fuel that is fresh, 1/4 enriched, 1/2 enriched, and 3/4 enriched. A thermal efficiency of 35% results in a gross electrical output of 925 MW_e. The laser requires 225 MW_e and thus the net output is 700 MW_e.

4. Conclusions

The potential feasibility of using hybrid reactors as a source of fissile fuel for LWRs in a nuclear future that does not allow reprocessing due to proliferation concerns has been established. This involves direct irradiation of fertile fuel assemblies in the hybrid blanket until the fissile enrichment required by the LWRs (3-4%) is reached. In doing so, the fuel becomes highly radioactive and is rendered diversion-resistant. Such a hybrid reactor produces ~0.65 ^{233}U atoms/fusion event while achieving a tritium breeding ratio of unity so that one hybrid with a fusion power of 1200 MW can fuel approximately 2.5 1000 MW_e LWRs in a once-through cycle.

The hybrid can also be incorporated into a scenario where the spent LWR fuel is sent to an internationally monitored, physically secure center containing the hybrid, the reprocessing facility, and the fuel manufacturing plant. This would allow the hybrid to fuel 10 LWRs with a conversion ratio of 0.75 for ^{233}U fuel.

Using careful blanket design, LWR fuel pins can be nearly uniformly enriched to 4% fissile concentration in approximately 2.7 years. The spectrum of neutrons incident on the assemblies must be carefully tailored to provide uniform enrichment. A hard spectrum is desired and this favors Pb, rather than Be, as a nonfissionable neutron multiplying material in the blanket. The optimum blanket design is not necessarily that which produces the flattest enrichment distribution since a flat profile can only be achieved at the expense of the ^{233}U breeding ratio.

Three-dimensional neutronics calculations show that some of the fuel can be replaced by neutron scattering material and that the remaining fuel still has the same production rate, 0.6 ^{233}U/fusion neutron with a tritium breeding ratio of one. This reduction of fuel inventory shortens the time to 4% enrichment from 2.7 years to 1.4 years of exposure. These 3-D calculations also show that the axial and radial blankets can be strongly coupled neutronically. This suggests that blanket design using simple 1-D calculations with solid angle weighting must be carefully evaluated for validity. Elaborate fuel management schemes are required to reduce the radial and axial nonuniformities in the fissile enrichment distribution.

Burnup calculations show that approximately 13% of the fissile fuel generated is burned in situ. This burnup is equivalent to 4300 MWd/MT. The power swing due to changes in the blanket multiplication during enrichment is 19%. The maximum damage rate to the zircaloy clad during exposure in the hybrid is about 20 dpa over the total 2.7 year period which is a tolerable level.

Acknowledgements

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Table I

SOLASE-H PARAMETERS

Cavity Shape	Cylindrical
Cavity Radius	6 m
Cavity Height	12 m
Structure - Blanket	Zircaloy
- First Wall	2 mm Zircaloy
First Wall Protection	0.5 - 1.0 torr Xenon Gas
Fusion Power	1200 MW
Pellet Yield	300 MJ
Pellet Gain	200
Pulse Repetition Frequency	4 Hz
Laser Energy on Target	1.5 MJ
Laser Type	Hydrogen Fluoride
Laser Energy	2 MJ
Maximum Power	300 TW
Wavelength	2.7-3.5 m
Pulse length (Multiplexed)	3 ns
Net Efficiency	2.6%
Electrical Efficiency	24%
Number of Final Amplifiers	20
Number of Last Mirrors	56
Last Mirror Position	22 m
Fertile Material	ThO_2
U^{233} Production Rate	0.65/Fusion Neutron
	2.5 Tonnes/yr
Fuel Form	(17 x 17) PWR Assemblies
Number of Assemblies	528
Time to 4% Enrichment	2.7 yr
Max/Ave. Enrichment	1.1
Neutron Multiplier	Pb
Blanket Power Multiplication	1.5 - 5.0
Average Thermal Power	2650 MW_t
Thermal Power Range	2400-2900 MW_t
% Variation	19%
Gross Elecrical Output	925 MW_e
Recirculating Power Fraction	26%
Neutron Wall Loading (Max)	2 MW/m^2
Coolant	Na
Coolant Temperatures	300-350°C
Tritium Breeding Ratio	1.0

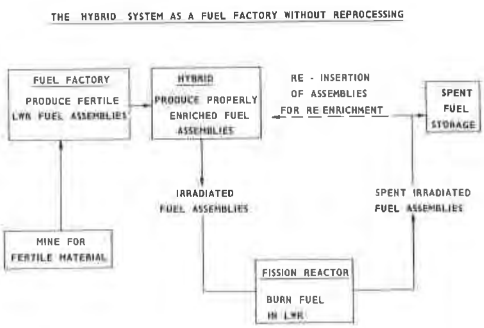

Fig. 1 - Fusion-fission hybrid fuel cycle without reprocessing

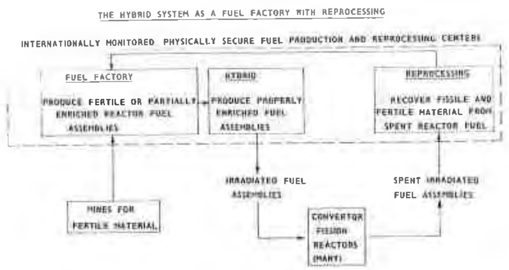

Fig. 2 - Fusion-fission fuel cycle with reprocessing

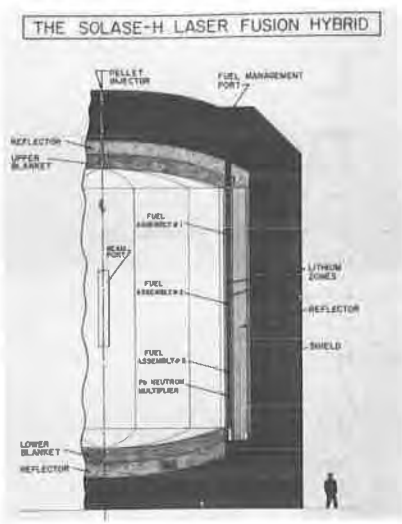

Fig. 3 - Cutaway views of the SOLASE-H reactor cavity and blanket

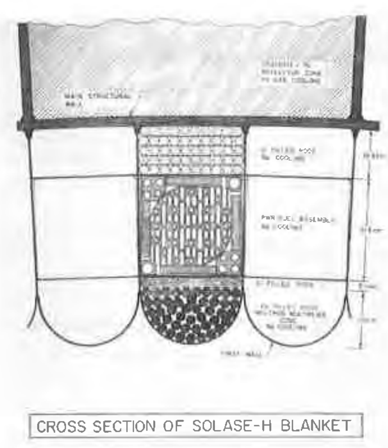

Fig. 4 - Cross sectional view of the radial blanket in SOLASE-H