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 problems 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. LWRS
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 LWRSs 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 symbiosis 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 required 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 primary 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, diver-
sion 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 de-
liberate 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 sensi-
tive 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 Fisiverson 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 repro-
cessing. The recovered fissile material is sent to the fuel factory within the secure site
and is fabricated into partially enriched assemblies. These are irradiated in the hybrid un-
til 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-repro-
cessing 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 1. 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 52R 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 accommodate the Na coolant pressure in the blank-
et 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 con-
taining Li. These Li zones both breed tritium and filter thermal neutrons that might oth-
erwise 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: $FM \equiv (UBR)^2/(n,\gamma)_{\text{max}}$ has been used as the criterion for blanket optimization, where UBR is the uranium breeding ratio. The quantity $Th(n,\gamma)_{\text{max}}/UBR$ 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 $\sim$2500 kg of $^{233}$U per year, enough to fuel $\sim$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\textsubscript{e} and the maximum is 2900 MW\textsubscript{e}. 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\textsubscript{e}. The laser requires 225 MW\textsubscript{e} and thus the net output is 700 MW\textsubscript{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 0.55 $^{233}$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\textsubscript{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 45% 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 3300 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

This work was supported by the Electric Power Research Institute under Contract EPP 237.
References


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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