Concept and Nuclear Performance of Direct-Enrichment Fusion Breeder Blanket Using UO₂ Powder

Yoshiaki OKA*, Takayasu KASAHARA and Shigehiro AN† ‡

Nuclear Engineering Research Laboratory, Faculty of Engineering, University of Tokyo*

Received January 25, 1984

A new concept is presented for direct enrichment of fissile fuel in the blanket of a fusion-fission hybrid reactor. The enriched fuel produced by this means can be used in fission reactors without reprocessing. The outstanding feature of the concept is the powdered form in which UO₂ fuel is placed in the reactor blanket, where it is irradiated to the requisite enrichment for use as fuel in burner reactor, e.g. 3%. After removal from blanket, the powder is mixed to homogenize the enrichment. Fuel pellets and assemblies are then fabricated from the powder without reprocessing. The concept of irradiating UO₂ in powder eliminates the problems of spatial nonuniformity in fissile enrichment, and of radiation damage to fuel clad, encountered in attempting to enrich prefabricated fuel. Powder mixing for homogenization brings the additional benefit of removing volatile fission products. Also burnable poison can be added, as necessary, after irradiation.

An extensive neutronic parameter survey showed that the optimum blanket arrangement for this enrichment concept is one presenting a fission suppressing configuration and with beryllium adopted as moderator. By this arrangement, the average $^{239}$Pu enrichment obtained on the natural UO₂ fuel in the blanket reaches 3% after only 0.56 MW·yr/m² exposure. A conceptual design is presented of the blanket, together with associated fusion breeder, from which, practical application of the concept is shown to be promising.

KEYWORDS: fusion breeder, fusion-fission hybrid reactor, direct enrichment, suppressed fission, fast fission, neutronics, blanket, conceptual design, plutonium, beryllium, uranium dioxide, performance

I. INTRODUCTION

The utilization of fusion neutrons to produce fissile fuel for use in fission burner reactors has received continuing interest in the past several years (1)–(8). These studies have concentrated in most cases on the fuel production capability of hybrids, premised on fuel reprocessing. Reprocessing, however, is expensive, and this adds to the cost of the bred fuel.

Studies envisaging elimination of reprocessing in the manufacture of enriched fuel and its direct application to fission burner have led to two concepts, both based on the use of fusion reactor: (a) Enriching the fissile content of fabricated fuel elements by irradiation in blanket and subsequent application to LWR's and HTGR's (9) (10), and (b) using fuel in the form of UO₂ or ThO₂ particles or pellets, which after enrichment are directly fabricated into fuel elements without reprocessing (11). The forms of fuel adopted in past attempts at realizing the latter concept (b) include UO₂ pellets contained in a graphite ball (11b), and ThO₂ particles coated with SiC suspended in Li₂Pb₈₃ (11c). These ThO₂ particles and UO₂ pellets may prove to involve difficult practical problems in shuffling the numerous fuel

* Tokai-mura, Ibaraki-ken 319-11.
† Present address: Department of Nuclear Engineering, Faculty of Engineering, University of Tokyo, Hongo, Bunkyo-ku, Tokyo 113.
‡ Present address: Tokai University, Tomigaya, Shibuya-ku, Tokyo 151.
pellets or particles in the fusion blanket and other aspects of remote handling. With the first-named concept (a), the main difficulty actually encountered in the case of reuse in HTGR was radiation damage and fission product build-up, which would limit the permissible number of irradiation cycles, while with LWR, the fuel assembly was found to suffer spatial nonuniformity in fissile enrichment, as well as structural radiation damage.

In this paper we present a new direct enrichment concept using fuel in the form of powder\(^{(13)}\). The study has mainly concerned blankets containing \(\text{UO}_2\) powder, which is easily produced from purification and de-oxidization of yellow cake, \(\text{U}_3\text{O}_8\). In current industrial practice, \(\text{UO}_3\) is deoxidized in fluidized bed, to produce \(\text{UO}_2\) already in the form of powder. This powder is irradiated in fusion reactor blanket until the average plutonium enrichment reaches a proper value for use in burner reactor, which would be say 3\%. The temperature of the powder is kept low enough to avoid sintering. After removal from the blanket, the powder is mixed to obtain uniform enrichment, upon which it is used to fabricate the fuel pellets and assemblies.

This method obviates the foregoing difficulties associated with conventional direct enrichment concepts, and in particular that of spatial nonuniformity in fissile enrichment when formed into fuel elements. The method further presents the additional advantages of: (1) Volatile fission products removed from the powder in the course of mixing; (2) possibility of adding poison to the pellets after fusion blanket irradiation; and (3) eventual possibility of refueling with fusion reactor under operation, when the technology of powder transportation is well developed. The disadvantage of the method would be its requiring remote fabrication of the fuel elements, but this difficulty should quickly diminish with the rapid progress seen today in remote-handling technology.

II. PARAMETRIC SURVEY FOR BLANKET OPTIMIZATION

An extensive parametric survey has been undertaken of the hybrid blanket configuration for the present concept. The calculations were performed by means of a one-dimensional transport and burnup code developed by the present authors, the BISON\(^{(14)}\), with data from the 42-group neutron and 21-group \(\gamma\)-ray coupled cross section from the BISON library\(^{(14)}\) and the GICX40 set\(^{(15)}\). The calculations were performed on the infinite cylindrical model shown in Fig. 1, which embodied the basic elements of: 1-cm thick type 316 stainless steel first wall, 50-cm thick blanket, 30-cm thick reflector/shield. For fertile fuel, \(\text{UO}_2\) was adopted, in consideration of its current wide use in the fission reactor fuel cycle; \(\text{Li}_2\text{O}\), with its high capability of tritium breeding, was chosen as tritium breeder (fusile material), 316 stainless steel as structural material, and helium as coolant.

The parametric survey sought to optimize the parameters of each of the five types of hybrid blanket defined in Table 1, which respectively represent arrangements featuring:

- Fusile behind fertile blanket (Type A)
- Fusile behind fertile blanket, and diluted by moderator (Type B)
- Fertile behind fusile blanket (Type C)
- Fertile and fusile diluted with moderator (Type D)
- Fertile diluted with fusile (Type E).
The first two types are of configuration that promotes fast fission, and the remaining three of configuration tending to suppress fission. Among the blankets of the latter configuration, Type C suppresses fast fission, and Type E thermal fission, while Type D suppresses both fast and thermal fission.

As moderator, beryllium was adopted for the Type B blanket, and for Type D beryllium and graphite, as well as light and heavy water. As neutron multiplier, lead, beryllium, depleted uranium and thorium were considered for all five types of the blanket.

The parameters that were varied for optimizing the performance of the different types of blanket were:

1. Thickness and arrangement of UO₂ and Li₂O zones
2. Volume fraction of UO₂, Li₂O and moderator
3. Thickness and material of neutron multiplier and reflector.

The parameters left unvaried in the initial parametric survey were the total blanket thickness (50 cm) and the volume fractions of helium coolant, of structural stainless steel and of other components and component combinations in the different zones, as given in Table 2.

The quantity adopted for representing the performance to be optimized was the total breeding rate \((T+F)\), the sum of tritium \(T\) and fissile \(F\) breeding rates per fusion neutron. Also taken into account was tritium self-sufficiency \(T>1.0\).

The optimum performances obtained for each of the five types of blanket are summarized in Table 3. In this table, the values given between parentheses indicate the tritium

<table>
<thead>
<tr>
<th>Type</th>
<th>Optimum configuration</th>
<th>Total breeding rate</th>
<th>Blanket energy multiplication</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fast fission configuration</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>A</td>
<td>UO₂ zone 10 cm + Li₂O zone 40 cm</td>
<td>1.71 (1.00)</td>
<td>5.36</td>
</tr>
<tr>
<td>B</td>
<td>UO₂ zone 10 cm + Li₂O(10%)/Be(50%) zone 40 cm</td>
<td>1.74 (1.01)</td>
<td>4.56</td>
</tr>
<tr>
<td>Fission suppressing configuration</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>C</td>
<td>Be neutron multiplier zone 5 cm + Li₂O zone 10 cm + UO₂ zone 12 cm + Li₂O zone 13 cm</td>
<td>1.42 (1.10)</td>
<td>2.32</td>
</tr>
<tr>
<td>D</td>
<td>UO₂(1%)/Li₂O(9%)/Be(50%) zone 50 cm</td>
<td>1.57 (1.09)</td>
<td>1.57</td>
</tr>
<tr>
<td>E</td>
<td>Be neutron multiplier zone 3 cm + UO₂(6%)/Li₂O(54%) zone 47 cm</td>
<td>1.42 (1.15)</td>
<td>1.78</td>
</tr>
</tbody>
</table>

\(\uparrow\) Sum of fissile and tritium breeding rate
\(\uparrow\) Indicated between parentheses is the tritium breeding rate
breeding rate. These latter values show slight differences from type to type of blanket, but the differences in tritium breeding rate tend to be offset by a compensating trend shown by the fissile breeding rate, with the net effect of stabilizing the total breeding rate. The total breeding rate can thus be considered a good quantity on which to base comparisons of blanket performance.

Comparison between the different types of blanket with their performances thus optimized reveals that, of the two types that are of fast fission configuration, Type B shows somewhat better performance. The optimized Type B blanket has its 50 cm thickness divided into 10 cm of UO₂ fuel and 40 cm of mixed Li₂O tritium breeder/Be moderator zone made up of 10%Li₂O and 50% beryllium, the remaining 40% of blanket volume being occupied by the prescribed 10% structural stainless steel and 30% of helium coolant. Among the blankets of the other fission suppressing configuration, the best performance is shown by the Type D, with its mixed fuel/tritium breeder/moderator zone made up of 1%UO₂, 9%Li₂O, 50% beryllium, to constitute the prescribed 60% of blanket volume, the remainder being the prescribed 30% of helium coolant and 10% of structural stainless steel.

Further details of the parametric survey are given in Appendix.

To decide between the Type B and D, an examination was made of the change to be expected of their nuclear performance with progress of exposure, through calculations using BISON. The result is shown in Table 4. The fission suppressing configuration is seen to provide quite rapid buildup of fissile concentration. The average enrichment of UO₂ powder in the blanket would reach 3% after only 0.56 MW·yr·m⁻² of exposure, and 7 months would suffice to attain this level of fissile concentration with a neutron wall loading of 1 MW·m⁻². During this exposure, the blanket power would increase by only 13%.

Table 4  Comparison of burnup performance between fast fission
and fission suppression configurations of blanket

<table>
<thead>
<tr>
<th></th>
<th>Fast fission (Type B)</th>
<th>Fission suppression (Type D)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Exposure required for 3% enrichment</td>
<td>2.9 MW·yr·m⁻²</td>
<td>0.56 MW·yr·m⁻²</td>
</tr>
<tr>
<td>Power increase during irradiation</td>
<td>33%</td>
<td>13%</td>
</tr>
<tr>
<td>Production rate of 3% enriched fuel per unit length of blanket</td>
<td>1.21 t·yr⁻¹</td>
<td>0.91 t·yr⁻¹</td>
</tr>
</tbody>
</table>

† Neutron wall loading: 1 MW·m⁻²

In the fast fission blanket, the average UO₂ powder enrichment increases less rapidly, due to the excessive presence of fertile atoms. The same UO₂ enrichment of 3% would require 2.9 MW·yr·m⁻² to build up. The greater fission product inventory presented by this blanket would add a further demerit on account of the potential hazard.

From the foregoing considerations, the fission suppressing configuration was concluded to be more favorable for the present concept. Consequently, choosing Type D of this configuration as the best blanket type, its thickness was varied around the initially prescribed value of 50 cm, to seek whether its performance might be further enhanced. The result is shown in Fig. 2. The production rate P of 3% enriched fuel ceases its rapid rise beyond 50 cm blanket—i.e. fuel zone—thickness. At about 45 cm thickness, the exposure Y required for obtaining 3% enrichment reaches minimum. Excessive blanket thickness results in reduced average enriching rate over the whole blanket. 50 cm was adopted as blanket thickness.
The best blanket type and thickness being thus determined, the examination was further extended to the effect brought on the total breeding rate by changes in the other parameters that had been prescribed and left unvaried in the initial parametric survey—i.e., type of coolant, volume fraction of structural stainless steel, first wall thickness. The result is presented in Table 5, which reveals high sensitivity of total breeding rate to the structure volume fraction. Increasing presence of liquid lithium—whether as coolant or as tritium breeding material—also contributes to increasing the total breeding rate. With the volume fraction of liquid lithium set at 20%, and that of structure at 10%, the optimal volume fractions of the UO₂ fuel and of the beryllium in the lithium cooled blanket were found to be 5% and 65% respectively.

For such an optimized blanket comprising 20% lithium coolant/tritium breeder, 10% stainless steel structure, 65% beryllium moderator and 5%UO₂ fuel, burnup calculations were performed by BISON, based on one-dimensional cylindrical model. The results indicated 0.8 MW·yr·m⁻² exposure to be required to obtain 3% enrichment and that 1.28 t·yr⁻¹ of enriched UO₂ powder would be produced per unit length of blanket.

<table>
<thead>
<tr>
<th>Case No.</th>
<th>Manner of parameter change</th>
<th>Resulting change brought to total breeding rate from base value of 1.57</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>He 30%→15% UO₂/Li₂O/Be 60%→75%</td>
<td>+0.09</td>
</tr>
<tr>
<td>2</td>
<td>Blanket S.S. 10%→20% UO₂/Li₂O/Be 60%→50%</td>
<td>-0.34</td>
</tr>
<tr>
<td>3</td>
<td>Blanket S.S. 10%→0% UO₂/Li₂O/Be 60%→70%</td>
<td>+0.38</td>
</tr>
<tr>
<td>4</td>
<td>First wall S.S. 1 cm→0.5 cm</td>
<td>+0.13</td>
</tr>
<tr>
<td>5</td>
<td>He 30%→H₂O 15% UO₂/Li₂O/Be 60%→75%</td>
<td>+0.03</td>
</tr>
<tr>
<td>6</td>
<td>He 30%→D₂O 15% UO₂/Li₂O/Be 60%→75%</td>
<td>+0.04</td>
</tr>
<tr>
<td>7</td>
<td>He 30%→Liq. Li 20% UO₂/Li₂O/Be 60%→UO₂ 5% Be 65%</td>
<td>+0.20</td>
</tr>
</tbody>
</table>

†Type D blanket: Base parameters are 1% UO₂ fuel, 9% Li₂O tritium breeder, 50% Be moderator, 10% stainless steel structure, 30% He coolant, other reactor parameter as indicated in Fig. 1.
Compared with helium-cooling, lithium-cooling should provide a lower enrichment rate, but the fuel production rate would be higher on account of the improved fuel volume fraction. Nonetheless, in the present instance, the helium-cooled and beryllium-moderated blanket (Type D) was adopted, in consideration of some uncertainty still remaining in regard to difficulties that might be encountered in circulating liquid lithium across a magnetic field.

### III. CONCEPTUAL DESIGN OF DIRECT ENRICHMENT FUSION BREEDER BLANKET FOR UO$_2$ POWDER

The practical applicability of the foregoing concept of direct-enrichment fusion breeder blanket was examined through a design study. While this concept can be combined with any type of fusion driver, a tokamak was taken up in this instance, with 1 MW·m$^{-2}$ assumed for neutron wall loading. The blanket consists of triangular beryllium blocks, of cross section as shown in Fig. 3, traversed in axial direction by stainless steel tubes of three different diameters; 45 mm for helium coolant, 19.2 mm for Li$_2$O, and 6 mm for UO$_2$ powder. The beryllium blocks should be broken down into small segments if large blocks would raise difficulties from thermal stress. The pipe for coolant is double-walled. The helium coolant takes the inner tube for incoming, and the outer tube for outgoing pass. The helium also serves to collect tritium for recovery. The beryllium blocks are arrayed as shown in Fig. 4. The first wall is independently cooled by helium.

![Fig. 3 Cross section of beryllium block composing fusion breeder blanket](image)

![Fig. 4 Beryllium block array](image)

A blanket segment has the configuration shown in Fig. 5. The thickness of blanket is 50 cm and also 50 cm thick shield is required for protecting superconducting magnets against excessive radiation damage. The thermal parameters determined by one-dimensional heat transport calculation are summarized in Table 6. This blanket permits the maximum UO$_2$ powder temperature to be kept below 773 K, to avoid sintering. The beryllium blocks and stainless steel structure should also be kept below 773 K to avoid swelling, and also from consideration of compatibility with working fluids.

The fuel production rate of the fusion breeder incorporating this blanket can be calculated using the parameters for an INTOR class tokamak device. Detailed conceptual
Table 6 Thermal parameters of fusion breeder blanket, determined by one-dimensional heat transport calculation

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coolant</td>
<td>3 MPa He</td>
</tr>
<tr>
<td>$T_{in}$</td>
<td>400°C</td>
</tr>
<tr>
<td>$T_{out}$</td>
<td>480°C</td>
</tr>
<tr>
<td>Coolant velocity</td>
<td>40 m/s</td>
</tr>
<tr>
<td>Maximum volumetric power generation</td>
<td>8.1 MW/m³</td>
</tr>
<tr>
<td>Li$_2$O</td>
<td>21.4 MW/m³</td>
</tr>
<tr>
<td>UO$_2$</td>
<td>118.4 MW/m³</td>
</tr>
<tr>
<td>Power conversion system</td>
<td></td>
</tr>
<tr>
<td>Steam efficiency</td>
<td>30%</td>
</tr>
</tbody>
</table>

Table 7 Principal parameters of fusion breeders

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major radius</td>
<td>5.0 m</td>
</tr>
<tr>
<td>Neutronic load on first wall</td>
<td>1.0 MW/m³</td>
</tr>
<tr>
<td>Fusion power</td>
<td>370 MWt</td>
</tr>
<tr>
<td>Total thermal power</td>
<td>490 MWt</td>
</tr>
<tr>
<td>Gross electric power</td>
<td>147 MWe</td>
</tr>
<tr>
<td>Power required for RF heating, pumping and system support</td>
<td>147 MWe</td>
</tr>
<tr>
<td>Net electric power</td>
<td>0 MWe</td>
</tr>
<tr>
<td>Production rate of 3% enriched U-PuO$_2$ fuel</td>
<td>23 t/yr</td>
</tr>
</tbody>
</table>

Blanket coverage: 0.9  Plant factor: 0.9

design of the fusion breeder is out of the scope of the present report, and only the resulting parameters are summarized in Table 7. The reactor will produce 23 tons of 3% plutonium-enriched UO$_2$ powder every year. The thermal support ratio (client LWR power to fusion breeder power) would be about 13, assuming 142 kg·yr$^{-1}$ of plutonium to be required for supporting a 1 GWt LWR. The power generated by the fusion breeder would be wholly consumed in the plant for supporting radiofrequency heating, for coolant pumping and for other purposes, leaving no net power available for out-of-plant consumption.

IV. SUMMARY CONCLUSION

(1) The concept is presented of a direct-enrichment fusion breeder using fertile powder.

(2) Parametric survey of alternative types of blanket indicated promise for the fission suppressing configuration, with UO$_2$ powder as fuel, Li$_2$O as tritium breeder and beryllium as moderator, for the rapid buildup of fissile concentration that could be expected of this type of blanket. The optimized volume fractions of the blanket are
1%UO₂, 9%Li₂O and 50% beryllium, given the fractions of 10% for structural stainless steel and 30% for helium coolant. The blanket can be constituted of beryllium blocks traversed by small tubes providing separate channels for UO₂ and Li₂O.

(3) The design study of the blanket indicated practical application of the concept to be quite promising.

ACKNOWLEDGMENT

The authors wish to thank A. Suzuki for his suggestion to have the enriched powder mixed outside the blanket, which would also serve in removing volatile fission products. They also express their appreciation of the valuable discussions provided by the members of the research group organized by Y. Fujii-e. Grateful acknowledgment is also due to K. Furuta, M. Akiyama and H. Hashikura for assistance in the calculations, and S. Onuki for typing the manuscript.

(Text edited grammatically by Mr. N. Yoshida.)

REFERENCES


[APPENDIX]

Particulars of the results obtained from the parametric survey for optimizing blanket performance are summarized below:

Type A blanket

(1) An additional neutron multiplier zone should serve no purpose since uranium fuel itself is a good neutron multiplier.

(2) An additional graphite reflector should serve no purpose, since the minimum total thickness of the blanket has been set to be 50 cm.

(3) The upper limit of total breeding rate would be 1.77 if no limit is set on blanket thickness, the thickness of the front UO₂ zone being otherwise limited from considerations of tritium self-sufficiency, and in view of the premises adopted in the present study of fixing the component volume fraction of each zone and of ⁴Li enrichment.

Type B blanket

The total breeding rate is maximized when the volume ratio between beryllium and Li₂O is 5.

Type C blanket

(1) Use of neutron multiplier increases total breeding. If uranium is used for this purpose, the optimum configuration becomes similar to that of the type A blanket.
(2) Beryllium is superior to lead and thorium as neutron multiplier.
(3) The optimal thickness of the beryllium neutron multiplying zone is 5 cm.

**Type D blanket**

1. Replacement of beryllium by graphite, heavy water or light water as moderator in the UO$_2$ and Li$_2$O blanket diminishes the total breeding rate.
2. Mixing beryllium with UO$_2$ and Li$_2$O increases the total breeding rate. With the aggregate volume set at 60%, the optimal volume fractions are roughly UO$_2$/Li$_2$O/Be = 1%/9%/50%, as indicated from Fig. A1, the volume ratio of 9:1 between Li$_2$O and UO$_2$ being determined from the tritium self-sufficiency requirement.

**Type E blanket**

1. Use of neutron multiplier increases the total breeding ratio. Beryllium is superior to lead as neutron multiplier. The optimum thickness of beryllium is about 3 cm.

---

**Fig. A1** Change of nuclear performance brought to helium-cooled beryllium-moderated UO$_2$ blanket with variation of beryllium volume fraction (sum of UO$_2$, Li$_2$O and beryllium volume fractions held fixed at 60%, the remaining fraction divided into 30% helium coolant and 10% structural stainless steel, and the ratio between UO$_2$ and Li$_2$O maintained at 1/9.)