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FUSION TECHNOLOGY INSTITUTE

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# Hybrids for Direct Enrichment and Self-Protected Fissile Fuel Production

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## Abstract

Hybrid reactors to directly enrich LWR fuel bundles with $^{239}Pu$ are described. The fuel distribution across a bundle is found to be more uniform in this case than when $^{233}U$ is produced from thorium. As expected, more fuel is produced from $^{238}U$ than from $^{232}Th$ per fusion event although the fuel production per unit thermal power can be greater in the Th-U cycle. The hybrid can be used to produce fissile fuel at a secure fuel production, reprocessing, and fabrication facility. The high support ratio of the hybrid would then allow 10 to 50 external fission reactors to be supported per secure site, depending upon the conversion ratio of the off-site fission reactors. It is found that fuel to be shipped away from the secure site can be rendered resistant to diversion by irradiation in a low power, on-site fission reactor to a burnup of 0.4 MWd/t.

## I. Introduction

The use of fusion neutrons to produce fissile fuel is one of the earliest proposed applications of fusion and the idea has recently been the subject of renewed attention.[1-9] Most studies have concentrated on using a hybrid fusion reactor as an electrically self-sufficient fissile fuel factory which, together with reprocessing, can assure the fuel supply for a number of fission convertor reactors. The power in fission reactors, $P_{FR}$, which can be supported by a hybrid of fusion power $P_{FUS}$ is given by[10]

$$P \equiv \frac{P_{FR}}{P_{FUS}} = \frac{\varepsilon(C_H-1)}{(1 - CR)(1 + \alpha)} \qquad (1)$$

where CR is the conversion ratio of the fission reactors, $C_H$ is the total breeding capacity of the hybrid per fusion neutron (including captures to make both tritium and fissile fuel), $\alpha$ is the capture to fission ratio in the fission reactors, and $\varepsilon$ is the ratio of the energy released per fusion event (~20 MeV) to the energy released per fission event (~200 MeV). Clearly, P can be a large number typically ranging from 10 to 40 for fission reactors with CR of approximately 0.6-0.75 (typical of light water reactors) to 30 to 80 for advanced convertor reactors with CR approximately 0.9. The range of P is also affected by whether the fissile fuel is $^{233}U$ or $^{239}Pu$.

Most earlier work has concentrated on the fuel production capability of hybrids assuming fuel reprocessing. This work has not in general addressed the proliferation issues associated with the production of $^{233}U$ or $^{239}Pu$ and the formation and shipment of mixed oxide fuels (Th-$^{233}U$ oxide or $^{238}U$-$^{239}Pu$ oxide) to fission reactors. Schultz first discussed the concept of direct fuel irradiation and enrichment in a hybrid reactor followed by shipment and use of such fuel in a high temperature gas cooled reactor (HTGR).[11] The concept is

aimed at avoiding fuel reprocessing although, as noted by Schultz, this implies a rather severe penalty in terms of overall utilization.

More recently, we have examined the idea of direct irradiation and enrichment of light water reactor fuel bundles of $ThO_2$ in a hybrid reactor to produce properly enriched $^{233}UO_2$-$ThO_2$ fuel which has an acceptable enrichment distribution across the bundle.[12,13] It is found that a uniform enrichment distribution could be achieved in which the difference in peak to average enrichment across a bundle is about 10%. Further, it is found that the time to reach 4% enrichment of $^{233}U$ in $ThO_2$ is less than 3 years and that the accumulated fuel burnup in the hybrid is about 4000 MWd/t. After direct enrichment, the shipment of hot fuel, either as "fresh fuel" to a reactor or spent fuel from a reactor, provides a fuel cycle with resistance to proliferation.[14] We summarize these arguments in the next section.

A related concept is to use direct irradiation in the hybrid in conjunction with reprocessing in the context of a secure fuel production and reprocessing site.[15] The original suggestion of a secure site was put forth with the idea of using breeder reactors at the site. We replace the breeder with a hybrid and use direct irradiation rather than denaturing uranium as the approach to proliferation resistance.

In this paper, the direct irradiation concept is developed further by examining hybrid blanket designs to directly produce $^{239}Pu$ in $^{238}UO_2$ light water reactor fuel bundles. We find important differences in blanket design concepts from those developed to produce $^{233}U$. In addition, we develop further the concept of a secure fuel production, fabrication, and reprocessing site where the hybrid serves as the fuel factory and a low power fission reactor

is employed to irradiate the fission power reactor fuel bundles prior to off-secure-site shipment. Fuel shipped to or from the secure site is always highly radioactive. Within the secure site, fuel reprocessing is accomplished and fresh fuel fabrication can be done with a minimum of remote handling. This is in contrast to the recently proposed CIVEX process[16] in which fuel at every stage of reprocessing and fabrication is hot. In such a process, fresh fuel fabrication must be performed remotely. The secure site concept avoids this need (and concomitant economic penalty) while the use of the hybrid allows one to minimize the number of secure sites. This last point follows from the fact that P, the ratio of external fission reactor power to the fusion power of the hybrid, is large.

In the next section, we describe the material flow patterns associated with the concepts of direct irradiation and enrichment without reprocessing, and fuel production and reprocessing followed by fresh fuel irradiation prior to shipment to fission reactors. The direct enrichment of $^{238}UO_2$ LWR fuel using a hybrid is discussed in Section III along with a comparison of blanket designs to produce $^{239}Pu$ from $^{238}U$ versus $^{233}U$ from $^{232}Th$. In Section IV, the irradiation of fuel bundles prior to shipment from a secure site is discussed. Irradiation is possible in either a low power fission reactor or in the hybrid itself. We close the paper with a summary and final discussion.

## II. Once-Through Direct Enrichment and Secure Site Reprocessing

To set the stage for the specific results reported in the next two sections, we describe here two hybrid scenarios. The first involves the direct enrichment of LWR fuel assemblies in a fusion hybrid reactor followed by shipment of this irradiated fuel to the fission reactor.[12] Fuel reprocessing is not included. The second scenario assumes fuel reprocessing is allowed inside a secure site and the hybrid functions as a fuel factory and enrichment facility. Enriched assemblies can be exposed for a short time in the hybrid or a low power fission reactor to render them radioactive and more resistant to diversion. In both scenarios, the fuel shipped to and from the LWR is resistant to diversion by the nature of the hot fuel[10] and by the fact that the fissile material occurs only in hot fuel bundles rather than as fresh fuel pellets.

The details of the first scenario are outlined in Fig. 1. The cycle includes four steps:

1. Fertile fuel, $ThO_2$ or $UO_2$, is fabricated in a form that is directly usable in a LWR. (Other fission reactors could be included but the LWR is used here because it is the workhorse of the U.S. fission reactor industry.)

2. The cold, clean fuel assemblies, containing only fertile fuel, are placed in the hybrid blanket and enriched to a nearly uniform concentration of 3-4% fissile fuel as required by the LWR.

3. The enriched, and now highly radioactive assemblies, are transferred as units directly to the LWRs for burning of the fuel.

4. The spent fuel from the LWR is stored until a decision is made on reprocessing or storing or both. If feasible, the spent fuel can be reinserted into the hybrid to be re-enriched for further burning in the LWR. This

## THE HYBRID SYSTEM AS A FUEL FACTORY WITHOUT PROCESSING

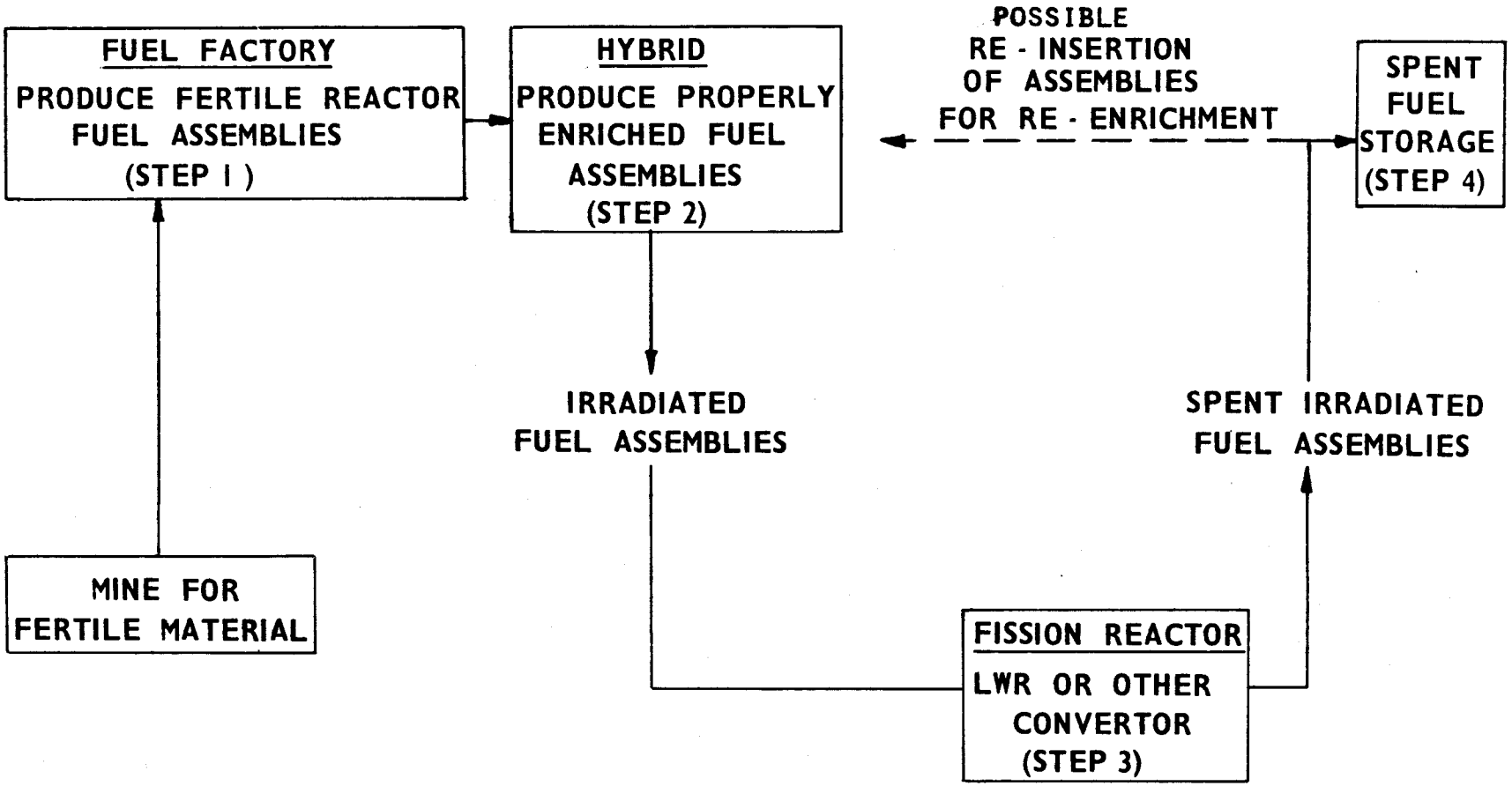



Figure 1

possibility depends on both the importance of fission product buildup to LWR performance and the radiation damage to the fuel and cladding.

One attractive feature of this cycle is that the system is resistant to diversion because fissile material occurs only inside highly radioactive fuel assemblies. Also, the reserves of fissile fuel are extended substantially. If the average LWR fuel enrichment is assumed to be 3%, the fissile fuel reserves are extended by 4.3 × (Thorium Resources + Uranium Resources). Here, if we assume the thorium resources are no larger than the uranium resources, the fissile fuel supply is extended a factor of 8-9 without reprocessing. By extending the fissile fuel supply using the hybrid, additional time becomes available to make deliberate decisions on issues such as internationally controlled, physically secure fuel production and reprocessing centers. Finally, the manufacturing of fresh fertile fuel pellets can proceed without the handling problems inherent in the use of radiation to self-protect the fuel during reprocessing and refabrication, such as in the proposed CIVEX system.[16]

The major disadvantage is that this approach does not take full advantage of the fertile fuel reserves. To achieve a fuel supply measured in thousands of years, fuel reprocessing is essential. Without reprocessing one hybrid reactor is only able to supply fissile fuel to about 2 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 by one hybrid of equivalent power, depending on the conversion ratio of the LWR or other convertor reactors.

The proliferation resistant fuel cycle can be extended to include reprocessing of the spent fuel if one follows a structure such as outlined by Feiveson and Taylor[15] of internationally controlled, physically

secure fuel production and reprocessing sites combined with many national convertor reactors "outside the fence." This process involves the six steps outlined in Fig. 2.

After mining for fertile material or utilizing the stockpile of depleted uranium (Step 1), fertile fuel is fabricated into any form suitable for use in the hybrid reactor (Step 2) and inserted into the hybrid (Step 3) where fissile material is produced. The fuel need not be in a form directly usable in a fission convertor reactor even if this were feasible. The fuel produced in the hybrid is reprocessed (Step 6) and returned to Step 2 for refabrication into properly enriched fission reactor fuel.

This fuel is irradiated in a lower power fission reactor (Step 4) to a burnup level of about 0.4 MWd/t. As discussed in Section IV, for LWR type fuel, this level of burnup is sufficient to render the fuel self-protecting under the guidelines of the Code of Federal Regulations.[22] At this point, the hot fuel is shipped from the secure site to the fission convertor reactors (Step 5). Upon completion of fuel utilization in the fission reactor, the hot spent fuel is returned to the secure site for reprocessing (Step 6) and re-use of both the fertile material (in Step 3) and the fissile material (in Step 4).

The advantages of this approach are several. First, the fuel supply is measured in terms of the fertile material abundance. All estimates show such fuel supplies will last for thousands of years. Second, fuel shipped to and from the convertor reactors is always radioactive and would be resistant to diversion and reprocessing for the reasons described earlier. Third, the convertor reactor need not be restricted to a LWR,

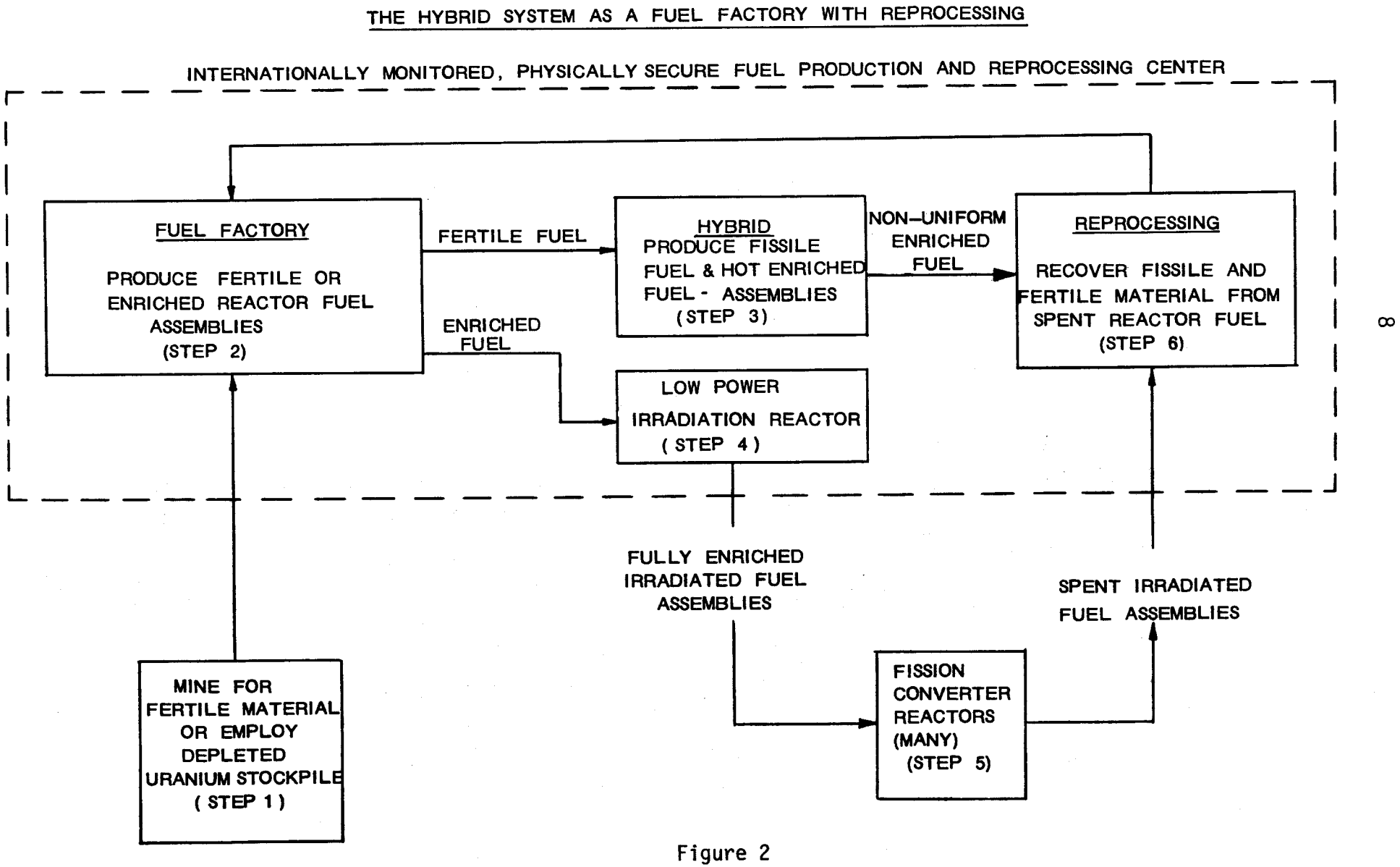

THE HYBRID SYSTEM AS A FUEL FACTORY WITH REPROCESSING
INTERNATIONALLY MONITORED, PHYSICALLY SECURE FUEL PRODUCTION AND REPROCESSING CENTER
FUEL FACTORY
PRODUCE FERTILE OR ENRICHED REACTOR FUEL ASSEMBLIES (STEP 2)
FERTILE FUEL
ENRICHED FUEL
HYBRID
PRODUCE FISSILE FUEL & HOT ENRICHED FUEL - ASSEMBLIES (STEP 3)
NON-UNIFORM ENRICHED FUEL
REPROCESSING
RECOVER FISSILE AND FERTILE MATERIAL FROM SPENT REACTOR FUEL (STEP 6)
LOW POWER IRRADIATION REACTOR (STEP 4)
FULLY ENRICHED IRRADIATED FUEL ASSEMBLIES
SPENT IRRADIATED FUEL ASSEMBLIES
MINE FOR FERTILE MATERIAL OR EMPLOY DEPLETED URANIUM STOCKPILE (STEP 1)
FISSION CONVERTER REACTORS (MANY) (STEP 5)

Figure 2

although using these reactors will minimize the need to develop additional fission reactor technology. Fourth, the ratio of power production outside to inside the secure site is very large. This minimizes the number of required secure sites. Finally, the use of a low power fission reactor to effectively "spike" the fresh fuel assemblies eliminates the concern about the fissile fuel distribution in an assembly that arises when the direct enrichment process is used.

The potential technical success of these hybrid fuel cycle scenarios depends upon two key technical questions: (1) Can the hybrid reactor produce uniformly enriched fuel at an acceptable fusion performance level when the blanket design is constrained to accommodate LWR fuel assemblies? (2) Can a standard LWR burn the irradiated fuel?

We have addressed the first of these questions in earlier work on the Th-$^{233}$U cycle and describe results of analysis for the U-Pu system in the next section in detail. After describing the results on direct enrichment of LWR assemblies in Section III, we analyze in Section IV the irradiation of LWR bundles in a low power fission reactor to achieve self-protection. Detailed analysis of LWR performance with directly enriched fuel bundles has not been undertaken as yet.

## III. Direct Enrichment of $^{238}UO_2$ LWR Fuel

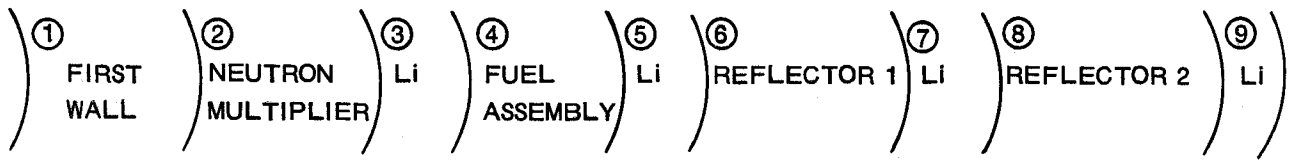
We have studied the design of a fusion hybrid blanket to produce $^{239}Pu$ with an acceptable enrichment distribution in a typical PWR fuel assembly and refer to this approach as "direct enrichment". In order to place the fuel assembly directly in a PWR without reprocessing, the design constraint applied is that the fissile fuel distribution in the fuel assembly is as uniform as possible. The blanket should also produce tritium. However the hybrid for which this blanket is designed[12,13] has regions devoted completely to tritium breeding where the local tritium breeding ratio (TBR) is greater than 1. Therefore the fissile fuel blanket regions can have TBR less than 1. Since the exact TBR needed in the fissile fuel blanket regions will not be known until 3-dimensional calculations are carried out, we have maximized the sum of the fissile breeding ratio (FBR) and TBR with the constraint that $0.6 \leq TBR \leq 0.8$. This approximation is derived from considerations of the reactor geometry. The fissile fuel blanket regions occupy about 50% of the total solid angle. TBR in the remainder of the blanket should be between 1.2 and 1.4. The following is a description of the optimization process used to design the blanket and of the fuel burnup calculations. We close with a comparison of $^{239}Pu$ blankets described here to $^{233}U$ producing blankets described previously.[13]

Calculation of the flux and activities is carried out in spherical geometry using the one-dimensional discrete ordinate neutron transport code ANISN.[17] The energy multiplication is calculated using the ACTEN code,[18] which employs reaction rates and reaction Q values to calculate energy deposition. A 25 neutron group cross section library has been used which was

created from the RSIC DLC-41B/VITAMIN-C AMPX[19] master interface library based upon ENDF/B-IV data.

A general schematic of the blankets considered is shown in Fig. 3 and the purpose of each zone is given. A summary of significant results is given in Table 1. As a starting point (case 1), $ThO_2$ is replaced by $^{238}UO_2$ in the original SOLASE-H blanket design.[12,13] Since $^{238}U$ has a high fast fission cross section, lead is replaced by $^{238}UO_2$ in the neutron multiplier zone in case 2. This results in an 80% increase in the total FBR, but the FBR for the fuel assembly decreases by 25%, leading to the conclusion that the neutron multiplier zone should be eliminated. In case 3 zones 2 and 3 are eliminated and a double row of fuel assemblies is used. This leads to a 9% increase in total captures over case 1. However, a double row of fuel assemblies is undesirable because fuel management is difficult, the total fertile inventory is increased, and the time to reach a specified enrichment is likewise increased.

In case 4, a single row of fuel assemblies is used. A 10 cm lead zone is placed in back of the lithium zone behind the fuel zone to multiply 14 MeV neutrons which pass through the fuel assembly and reflect neutrons back to the fuel and lithium zones. The back reflector is reduced to 40 cm and the lithium zone behind this is eliminated. The results show a significantly more uniform fissile fuel distribution since the fuel assembly is at the front of the blanket and 14 MeV neutrons have a long mean free path. The fissile production rate vs. fuel zone position is shown in Fig. 4 for cases 1, 4, and 6, the final design. In case 4, a large increase in leakage




| Zone | Purpose |
|---|---|
| 1<br>First Wall | Surface heat removal and radiation protection. Combined with Zone 2 for neutronics analysis. |
| 2<br>Neutron Multiplier | Multiplication of 14 MeV neutrons. |
| 3 and 5<br>Lithium | Tritium breeding and thermal neutron filter, decreases depletion of fissile fuel. Use to make flux trap of Zone 4. |
| 4<br>Fuel Assembly | Typical PWR fuel assembly,[5] surrounded by 0.5 cm of stainless steel. |
| 6<br>Reflector 1 | For blankets without Zone 2, multiplies 14 MeV neutrons which pass through fuel assembly and reflects neutrons into fuel zone. |
| 7 and 9<br>Lithium | Tritium breeding. |
| 8<br>Reflector 2 | Neutron reflection and leakage reduction. |

Figure 3 - Schematic of the Blanket

Table 1. Blanket Designs and Neutronics Performance Results

| Blanket Zone | Case Number 1 | 2 | 3 | 4 | 5 | 6 |
|---|---|---|---|---|---|---|
| Zones 1 & 2 | 82.2% Pb<br>9.3% Na coolant<br>8.5% Zirc-2<br>(10 cm) | As #1<br>$^{238}UO_2$<br>replaces Pb | (0 cm) | (0 cm) | (0 cm) | (0 cm) |
| Zone 3 | 95% Nat. Li<br>5% S.S.<br>(1.5 cm) | As #1 | (0 cm) | (0 cm) | As #1<br>(2 cm) | As #1<br>(2 cm) |
| Zone 4 | 30.3% $^{238}UO_2$<br>9.2% Zirc-2<br>59.2% Na Coolant<br>1.3% Void<br>(21.4 cm)<br>0.5 cm SS ea.side | As #1 | As #1<br>(42.8cm) | As #1 | As #1 | As #1 |
| Zone 5 | 95% Nat. Li<br>5% S.S.<br>(8 cm) | As #1 | As #1 | As #1<br>(2 cm) | As #1<br>(4 cm) | As #1<br>(4 cm) |
| Zone 6 | (0 cm) | (0 cm) | (0 cm) | 82.2% Pb<br>9.3% Na cool.<br>8.5% Zirc-2<br>(10 cm) | As #4<br>(40cm) | As #4<br>(40cm) |
| Zone 7 | (0 cm) | (0 cm) | (0 cm) | 95% Nat. Li<br>5% S.S.<br>(10 cm) | As #4 | As #4 |
| Zone 8 | 66.7% C<br>33.3% Pb<br>(60 cm) | As #1 | As #1 | As #1<br>(40 cm) | 100% C<br>(40cm) | As #5<br>(30cm) |
| Zone 9 | 95% Nat. Li<br>5% S.S.<br>(10 cm) | As #1 | As #1 | 0 cm | 0 cm | As #1 |

| Performance Parameter | | | | | | |
|---|---|---|---|---|---|---|
| Fissile Breeding Ratio | .99 | .74 | 1.50 | .98 | 1.00 | 1.00 |
| Tritium Breeding Ratio | .70 | .51 | .34 | .64 | .70 | .75 |
| Total Captures | 1.69 | 1.25 | 1.84 | 1.62 | 1.70 | 1.75 |
| Leakage | .010 | .008 | .007 | .16 | .06 | .016 |
| Energy Multiplication | 2.6 | 7.2 | 5.2 | 4.2 | 4.0 | 4.0 |
| $\nu\Sigma_f$ | .33 | 1.49 | 1.02 | .80 | .74 | .74 |
| Pb(n,2n) | .54 | .017 | .014 | .11 | .12 | .12 |
| $\frac{^{38}U(n,\gamma)_{max}}{^{238}U(n,\gamma)_{avg}}$ | 1.57 | 1.24 | 1.29 | 1.16 | 1.09 | 1.09 |

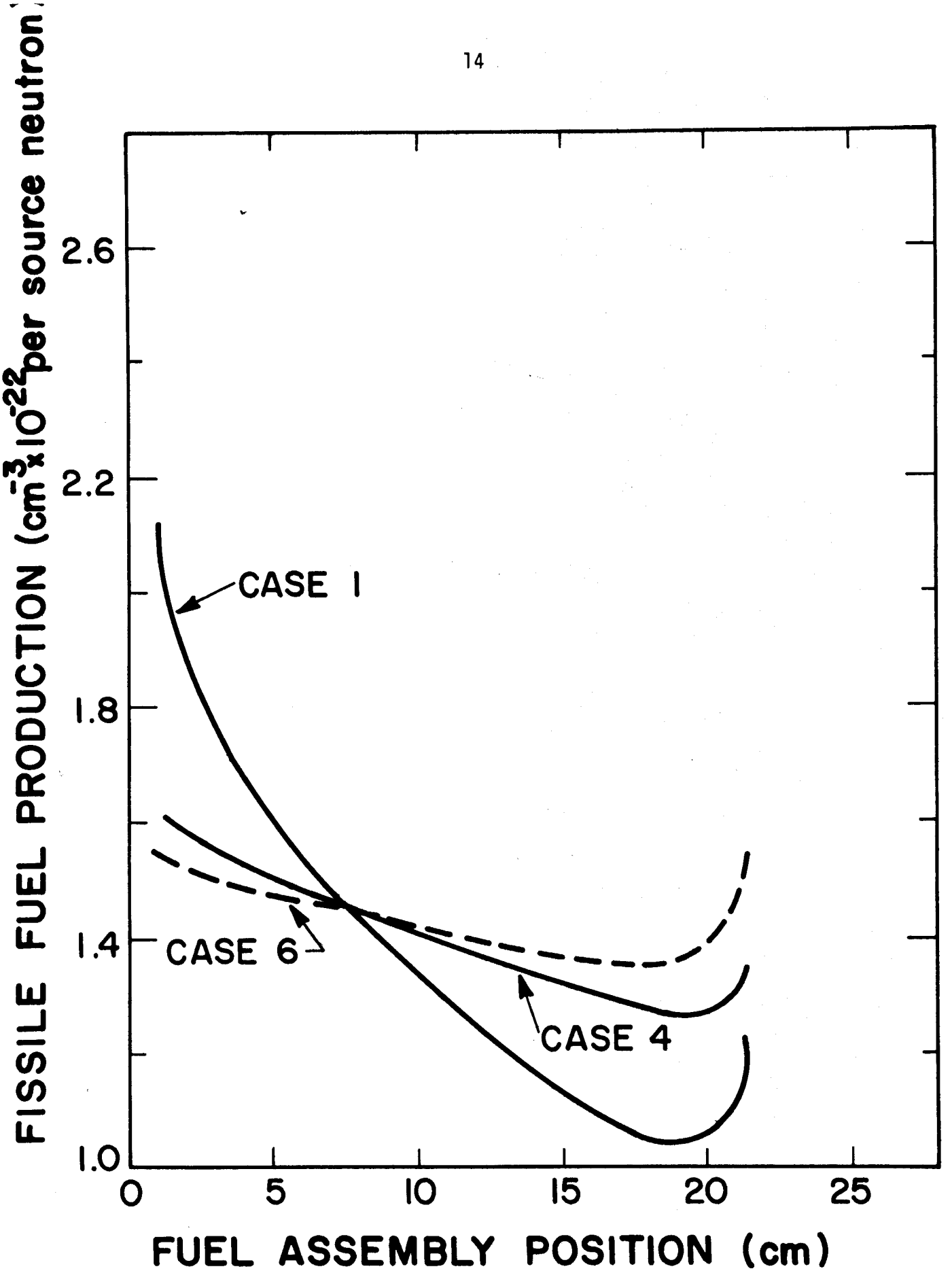



Figure 4 Distribution of the fissile fuel production in a LWR fuel bundle of initially $^{238}UO_2$ fertile fuel exposed in a fusion hybrid. Upon rotation, the peak to average fissile content in the bundle is 10% at the end of exposure when the average enrichment in $^{239}Pu$ is 4%.

causes a decrease in total captures, and the Pb(n,2n) reaction rate in the back reflector is insignificant.

In case 5, a 2 cm lithium zone is placed in front of the fuel assembly, and the lithium zone in back of the fuel assembly is increased to 4 cm. This further flattens the fissile fuel distribution. The lead zone in increased to 40 cm and lead is eliminated from the back reflector. The fissile fuel distribution improves and the total capture is approximately the same as for case 1.

In case 6, the final design, the back reflector is reduced to 30 cm and a 10 cm lithium zone is added to the back of the blanket. This reduces the leakage to an acceptable level. Total captures are the highest of any case with a single row of fuel assemblies.

During the blanket lifetime, significant changes will occur in FBR, TBR, and energy multiplication. The burnup model used[21] considers changes in the density of $^{238}U$ and $^{239}Pu$ only. Fission products and radioactive decay are ignored. In this case, the $^{238}U$ and $^{239}Pu$ densities are given by

$$\frac{dN_2(x,t)}{dt} = \sum_i N_1(x,t)\phi_i(x,t)\sigma_{\gamma_{1,i}} - \sum_i N_2(x,t)\phi_i(x,t)\sigma_{a_{2,i}}$$

$$\frac{dN_1(x,t)}{dt} = -\sum_i N_1(x,t)\phi_i(x,t)\sigma_{a_{1,i}}$$

where $N_2$ is the $^{239}Pu$ atomic density, $N_1$ is the $^{238}U$ atomic density, $\phi_i$ is the neutron flux for group i, $\sigma_{\gamma_{1,i}}$ is the microscopic capture cross section of $^{238}U$ for group i, $\sigma_{a_{1,i}}$ is the microscopic absorption cross section of $^{238}U$ for group i, and $\sigma_{a_{2,i}}$ is the microscopic absorption cross section of $^{239}Pu$ for group i.

The object of the burnup calculations is to find the time to 4% enrichment and the final fissile fuel distribution assuming a 180° rotation of the fuel assembly halfway to full enrichment.

The neutron flux is calculated at 0, 2, and 4% enrichment ($\phi_0$, $\phi_2$, and $\phi_4$ respectively). Depletion of the total fertile plus fissile atomic densities is taken into account. $\phi_0$ is used to calculate the burnup from 0 to 1% enrichment with a 1.92 MW/m$^2$ neutron first wall loading. $\phi_2$ is used from 1 to 2%. The atomic densities are then inverted and $\phi_2$ is used from 2 to 3%. $\phi_4$ is used from 3 to 4% enrichment. The time to enrichment calculated is 2.4 years assuming continuous operation.

Significant changes which occur during the lifetime of the blanket are shown in Table 2. The difference in peak to average enrichment across the fuel assembly is 9.8%. Enrichment versus fuel assembly position, assuming a 180° rotation of the fuel assembly halfway to full enrichment, is shown in Fig. 5.

Since we desire to place the fuel assembly directly in an LWR without reprocessing, it is important to know the radiation damage incurred during its lifetime in the hybrid. The quantities important to determining radiation effects are the distribution of the heating, displacement per atom (dpa), and helium production rates across the fuel assembly. These quantities are shown in Figs. 6 and 7. Radiation damage in the first wall is somewhat dependent on the characteristics of the blanket behind it. Calculations of dpa and helium production have been carried out for 316 stainless steel and Zircaloy-2 first walls. For 316 stainless steel, 11.6 dpa/yr per MW/m$^2$ and 120 appm/yr per MW/m$^2$ helium are produced. For Zircaloy-2 11.3 dpa/yr per MW/m$^2$ and 7.8 appm/yr per MW/m$^2$ helium are produced. This very low rate of helium production is due

Table 2

Burnup Results

| | Enrichment Level | | |
|---|---|---|---|
| | 0% | 2% | 4% |
| Fissile Breeding Ratio | 1.00 | 1.06 | 1.14 |
| Tritium Breeding Ratio | .75 | .83 | .93 |
| Total Captures | 1.75 | 1.89 | 2.07 |
| Leakage | .016 | .018 | .019 |
| Energy Multiplication | 4.04 | 6.43 | 7.39 |
| $\Sigma_f$ | .736 | 1.05 | 1.41 |
| Pb(n,2n) | .122 | .123 | .123 |

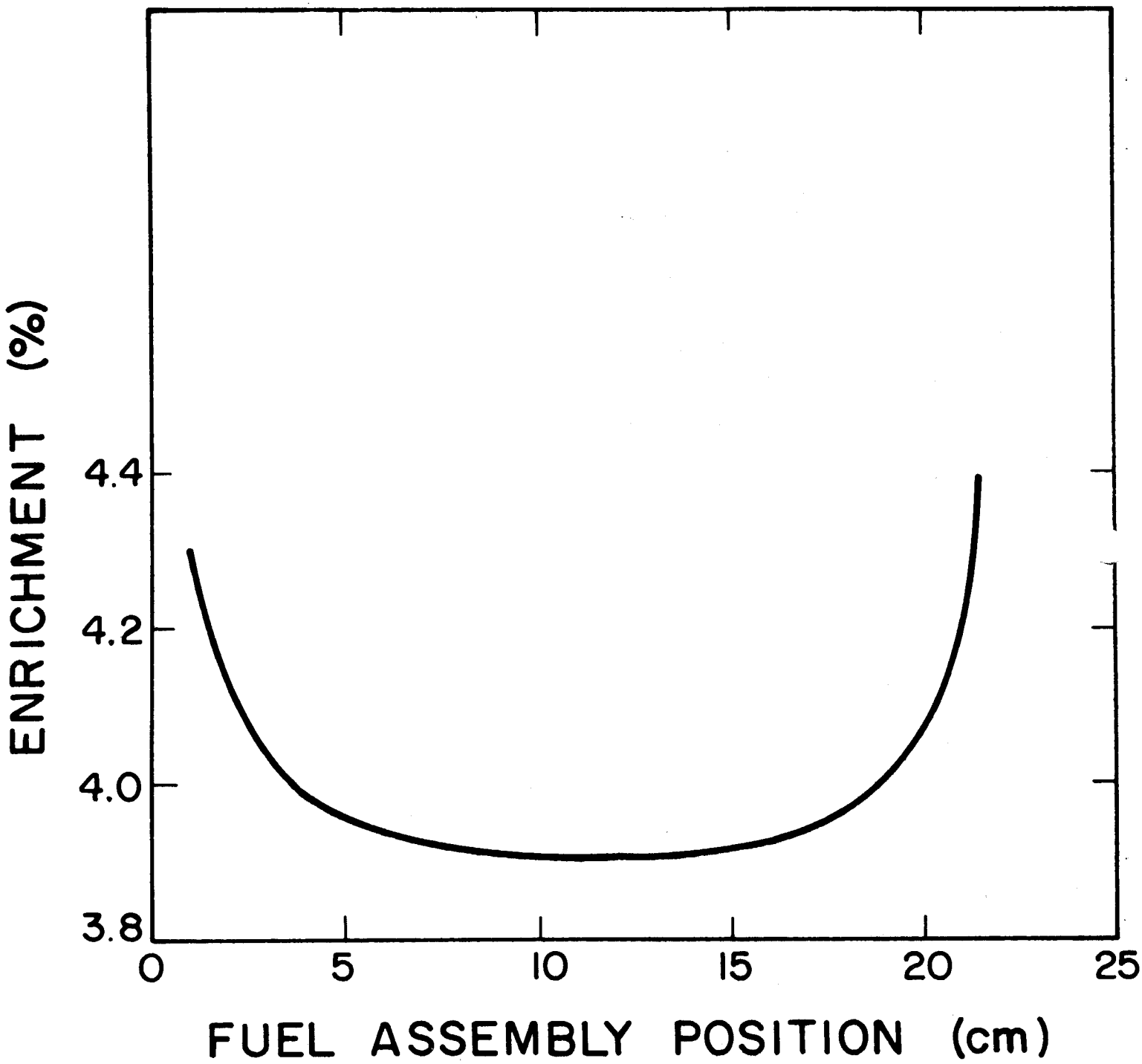



Figure 5 Enrichment vs. fuel assembly position assuming 180° rotation of fuel assembly halfway to full enrichment.

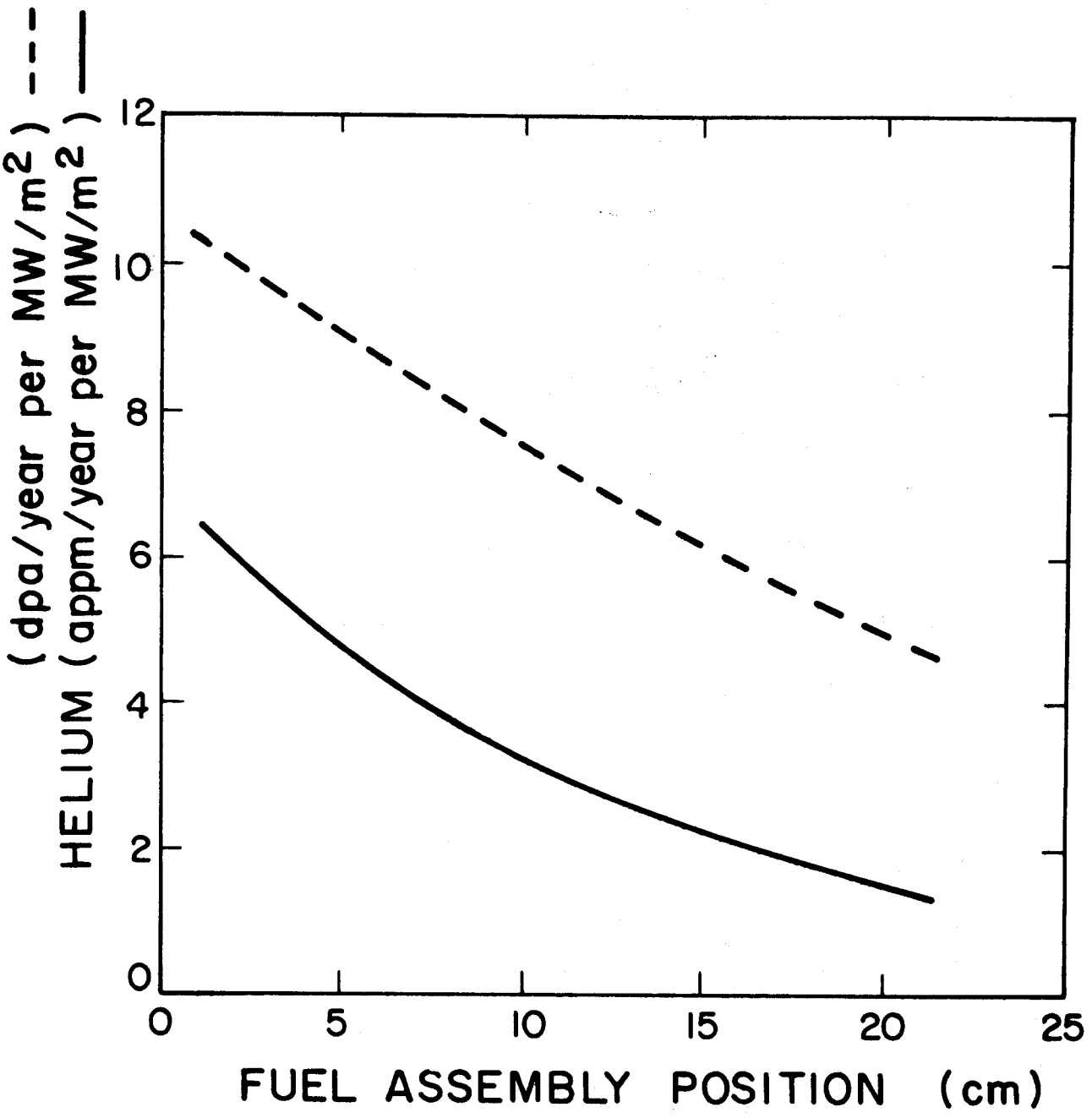



Figure 6 Helium production and dpa rates in cladding vs. fuel assembly position.

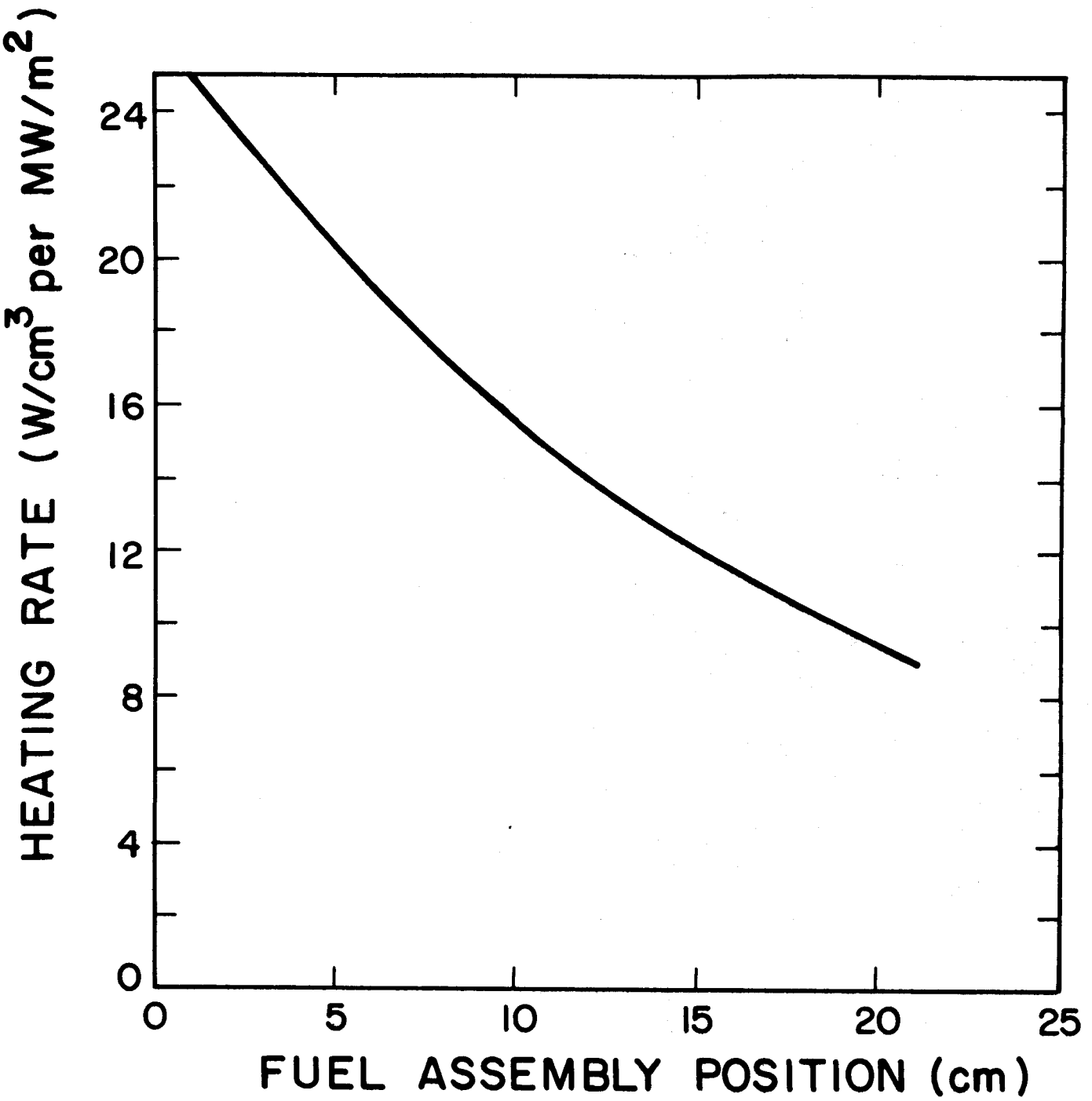



Figure 7 Heating rate in fuel assembly vs. assembly position.

to the high threshold (10 MeV) of the $(n,\alpha)$ cross section of zirconium. This cross section rises to only $3.9 \times 10^{-3}$ barns at 14 MeV.

Eventually a choice must be made between producing $^{233}U$ and $^{239}Pu$ in the hybrid. Although $^{233}U$ is a better LWR fuel due to its lower capture to fission ratio, more plutonium can be produced in the hybrid because $^{238}U$ has a higher fast fission cross section than $^{232}Th$. We find significant differences in design between $^{233}U$ and $^{239}Pu$ producing blankets. The most important difference is the absence of a neutron multiplier zone in the $^{239}Pu$ producer. This leads to a much flatter distribution of fissile production in the fuel zone. The $^{233}U$ producing blanket designed for the same hybrid [12,13] has a distribution similar to that in case 1 (Fig. 4). A comparison between $^{239}Pu$ and $^{233}U$ producing blankets is shown in Table 3 and Figs. 8 and 9.

Table 3

Comparison Between $^{233}U$ and $^{239}Pu$ Producing Blankets

| | $^{233}U$ | $^{239}Pu$ |
|---|---|---|
| Fissile Breeding Ratio | 0.94 | 1.00 |
| Tritium Breeding Ratio | 0.62 | 0.75 |
| Total Captures | 1.56 | 1.75 |
| Leakage | 0.010 | 0.016 |
| Energy Multiplication | 1.5→5.0 | 4.0→7.4 |
| $\nu \Sigma_f$ | 0.080 | 0.736 |
| Pb(n,2n) | 0.538 | 0.122 |
| Blanket Thickness | 112 cm | 118 cm |
| Time to 4% Enrichment | 2.75 yrs. | 2.40 yrs. |
| Max. to Ave. Fissile Enrichment | 1.10 | 1.09 |

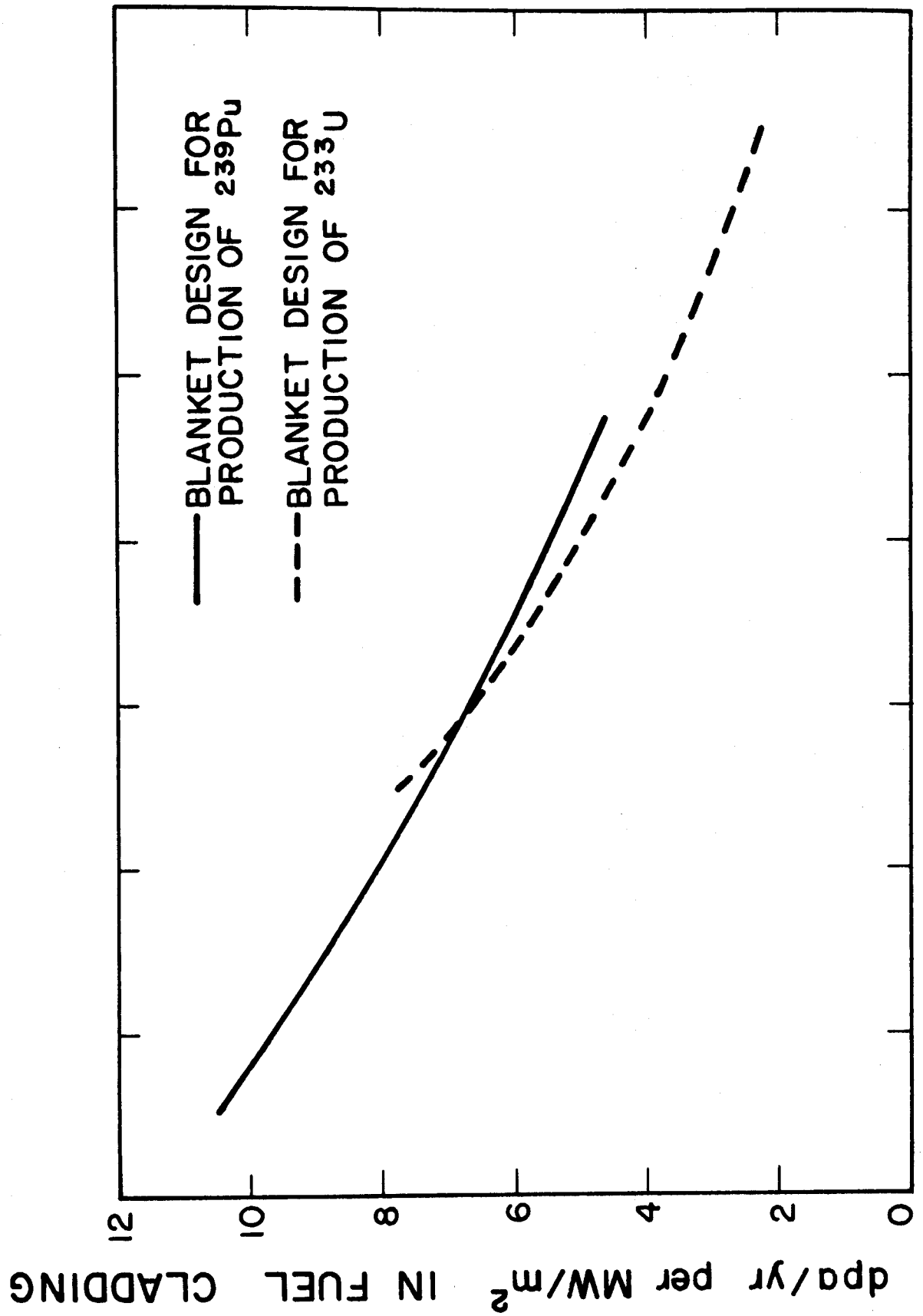



Figure 8 Comparison of dpa rate in fuel assembly vs. distance from first wall for $^{239}Pu$ and $^{233}U$ producting blankets.

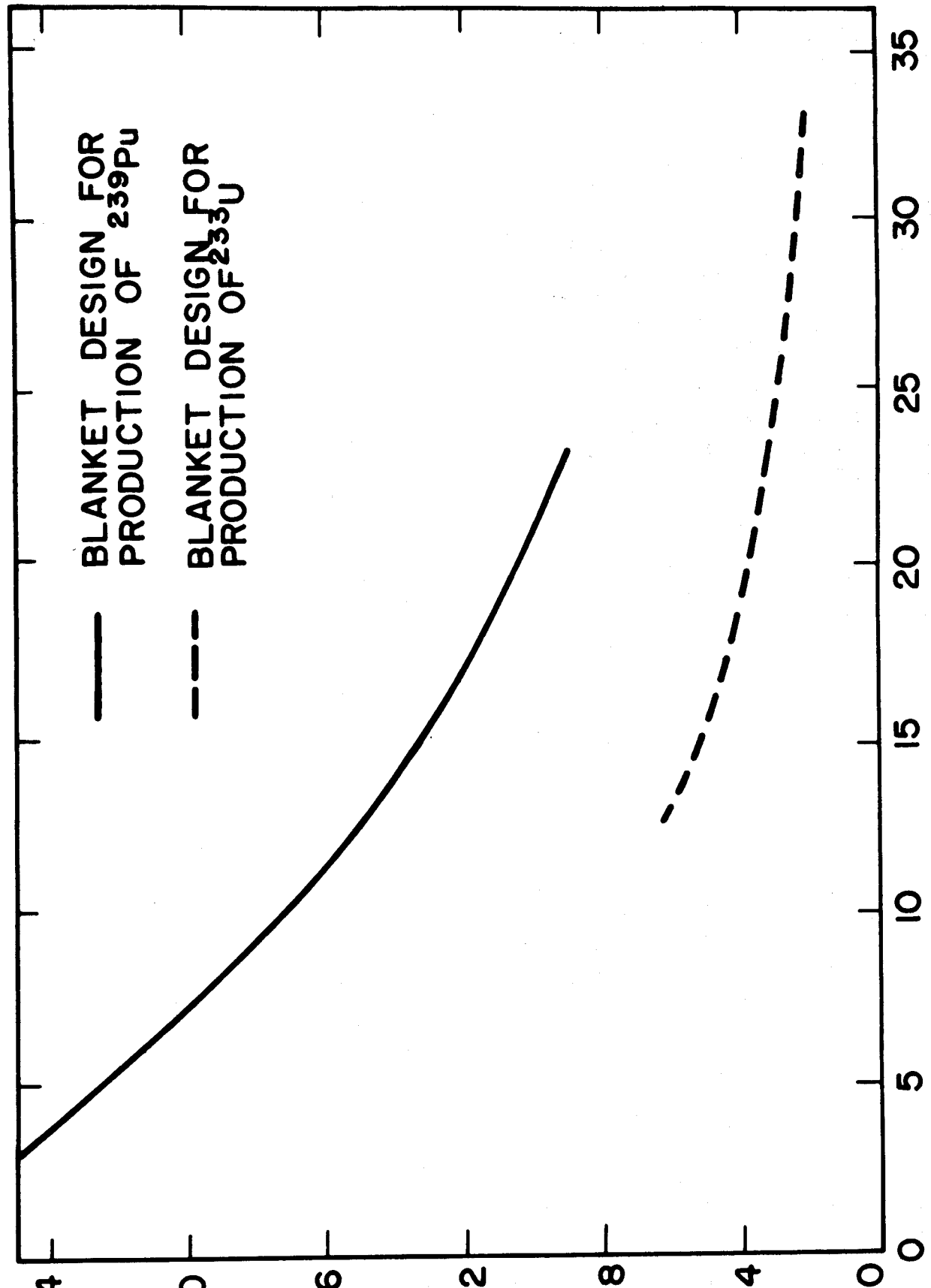



Fig. 9 Comparison of heating rate in fuel assembly as a function of the distance from first wall for $^{239}Pu$ and $^{233}U$ producing blankets.

## IV. Direct Fuel Assembly Irradiation

An important aspect of the secure site concept is the irradiation of the fuel bundles prior to their transport beyond the site boundary. One approach is to irradiate these bundles in the hybrid itself or to use a low power fission reactor to produce self-protected fuel assemblies. Self protection is defined as the bundle exposure required to produce a 100 rem/hr radiation field three feet from a bundle after a cooling time of 30 days. This follows the guidelines as stated by 10 CFR 73.[22] We find that after just 0.4 MWd/t of fuel exposure in a fission reactor, adequate self-protection for a typical fuel bundle is achieved.

To obtain this estimate of the exposure time required for self-protection, a simple model is used. The fuel bundle is assumed to be cylindrical and the calculations are performed using the method of Foderaro.[23] Further assumptions include the neglect of the shielding properties of both air and the bundle itself and of the buildup of radioactivity in the surrounding air. The calculated dose rate is underestimated and thus conservative.

The total dose rate expression of Foderaro[23] modified for this case is

$$D_i = \frac{k(E_i)E_i S_{vi} R_o^2}{2a} K(\theta,b) \quad .$$

$k(E_i)$ is the energy flux to dose rate conversion factor for the $i^{th}$ energy group (R/hr per MeV/cm$^2$-sec), $E_i$ is the mean energy of the $i^{th}$ group (MeV), $S_{vi}$ is the volumetric isotropic source strength for the $i^{th}$ energy group ($cm^{-3}sec^{-1}$), $R_o$ is the cylinder radius (cm), a is the distance from the

cylinder (cm), $K(\theta,b)$ is defined as $K(\theta,b) = e^{b} \int_{0}^{\theta} e^{-b\sec\theta'} d\theta'$, and is tabulated by Foderaro, and $\theta$ is the angle subtended by the fuel bundle at the point of interest. We have taken $R_0$ as 10 cm, a as 90 cm and the fuel bundle height as 380 cm. It is then found that $\theta$ is 65° and $K(\theta,0)$ is 1.13. The values for $k(E_i)$ are tabulated by Foderaro.

The source strength is found using the one group ORIGEN isotope generation and depletion code.[24] A typical light water reactor fuel element composition is chosen, namely, 3.3% enriched uranium $^{235}U$ in $^{238}U$. Similar results are expected for $^{239}Pu$ or $^{233}U$ fuel. The fuel bundle exposure time is varied and the radiation dose rate from the bundle after 30 days of cooling is calculated. In this way, the burnup to achieve a given dose rate is obtained. As a part of the ORIGEN output, photon release rates in photons per second per tonne are obtained for the fission products and the actinides in the fuel. As expected, the latter add little to the total dose rate and can be neglected.

Of the twelve photon energy groups for the fission products, only six are found to be significant contributors to the total dose rate and are listed in Table 4 along with their respective parameters and results. The source strength values are given after a 30 day post-irradiation cooling time. These values are given as photons per second per tonne of heavy metal and must be multiplied by 0.42 tonnes of heavy metal per fuel bundle and divided by the bundle volume of $1.66 \times 10^{5} cm^{3}$ to arrive at the isotropic volume source strength. The results in Table 4 are for 0.3 and 0.6 MWd/t and vary linearly with burnup. Thus, on the basis of 10 CFR 73[22], we find adequate bundle self-protection after a burnup of 0.4 MWd/t.

Table 4

Dose Rate for a Typical LWR Fuel Bundle After 30 Days of Cooling

| Mean Gamma Energy (MeV) | $k(E_i)$ (R/hr per MeV/cm$^2$sec) | $S_o$ @ 0.3 MWd/t (photons/sec/tonne) | Dose Rate @ 0.3MWd/t(R/hr) | $S_o$ @ 0.6 MWd/t (photons/sec/tonne) | Dose Rate @ 0.6 MWd/t(R/hr) |
|---|---|---|---|---|---|
| 0.30 | 2.06-06 | 5.71+12 | 5.6 | 1.14+13 | 11.2 |
| 0.63 | 1.94-06 | 2.03+13 | 39.4 | 4.06+13 | 78.9 |
| 1.10 | 1.81-06 | 7.13+11 | 2.3 | 1.42+12 | 4.6 |
| 1.55 | 1.68-06 | 7.24+12 | 30.0 | 1.45+13 | 60.0 |
| 1.99 | 1.56-06 | 1.75+10 | 0.1 | 3.51+10 | 0.2 |
| 2.38 | 1.49-06 | 2.34+11 | 1.6 | 4.67+11 | 3.2 |
| | | Total | 79.0 | Total | 158.1 |

In summary, we have determined the burnup of fresh fuel bundles to achieve self-protection to be modest. Since each fuel bundle contains about 0.42 tonnes of heavy metal, the irradiation required per bundle is 0.17 MWd, corresponding to about $4.58 \times 10^{21}$ fissions per bundle. If a low power reactor could be designed to generate 1 MW per bundle of loaded fuel, then self-protection is achieved in an irradiation time of four hours. For a reactor with less power per bundle, a proportionately larger time is needed. Thus, the technique appears favorable and should apply regardless of the type of fissile fuel used.

## V. Summary

The direct enrichment of LWR fuel bundles of initially fertile $^{238}UO_2$ fuel can be carried out in a fusion hybrid reactor and a relatively uniform fuel distribution (maximum to average enrichment less than 10%) can be achieved. The blanket design to produce $^{239}Pu$ directly in such bundles is quite different from earlier designs[13] developed for direct enrichment of $ThO_2$ in $^{233}U$. Such fuel is highly radioactive upon removal from the hybrid. It could be shipped directly to a LWR and be considered self-protecting from diversion, as spent fuel is considered today.

Direct enrichment would extend fissile fuel supplies and provide diversion protection but there is an obvious and serious economic penalty. Without reprocessing, fuel utilization is poor in that one hybrid can provide fuel for no more than two LWRs. The hybrid cost must be included in the total fuel cost. This low support ratio of fission reactors per hybrid means the fuel produced will likely be too expensive. Nevertheless, effective fissile fuel reserves could be extended a factor of 4 to 10, depending on the size of the thorium reserves relative to those of uranium and the size of the depleted uranium stockpile.

A more desirable alternative is to use the hybrid as a fissile fuel factory within a secure fuel production, reprocessing, and fabrication site. A single hybrid will now support 10 to 50 off-site LWRs, depending on their conversion ratio. Thus, the number of secure sites is smaller when the fuel producer at the site is a hybrid instead of a fast breeder reactor.

A low power fission reactor at the secure site can be used to render LWR bundles self-protecting before shipment. We have determined the burnup of fresh fuel bundles to achieve self-protection to be only 0.4 MWd/t. Since each LWR fuel bundle contains about 0.42 tonnes of heavy metal, the irradiation required per bundle is 0.17 MWd, corresponding to about $4.58 \times 10^{21}$ fissions per bundle. If a lower power reactor is designed to generate 1 MW per bundle of loaded fuel, self-protection is achieved in an irradiation time of four hours. For a reactor with less power per bundle, a proportionately larger time is needed. Thus, this technique appears favorable and should apply regardless of the fissile fuel.

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## References


1. F. Powell, "Proposal for a Driven Thermonuclear Reactor", Calif. Research and Development Co., Report LWS-24920 (Rev.), (1953).

2. B. R. Leonard, Jr., Nucl. Technol. 20, 161 (1973).

3. L. M. Lidsky, Nucl. Fusion 15, 151 (1975).

4. R. P. Rose, et al., "Fusion Driven Breeder Reactor Design" Westinghouse Electric Co. Report, WFPS-TME-043 (May 1977).

5. J. A. Maniscalco, "A Conceptual Design Study for a Laser Fusion Hybrid", in Proc. Second ANS Top. Meeting on the Technology of Controlled Nuclear Fusion, ERDA-CONF-760935-P2 (1976) p. 676.

6. J. D. Lee, "Nuclear Design of the LLL-GA $U_3Si$ Blanket" in Proc. Third ANS Topical Meeting on the Technology of Controlled Nuclear Fusion, CONF-780508, Sante Fe, NM, 219 (1978).

7. D. J. Bender, K. R. Schultz, R. H. Brogli, and G. R. Hopkins, "Preliminary Evaluation of the $^{233}U$ Refresh Cycle Hybrid Power System Concept", ibid., p. 149.

8. W.O. Allen and S.L. Thomson, "Electron Beam Fusion-Fission Reactor Studies", Joint US-USSR Conference on Fusion-Fission Hybrids, Princeton, NJ (1979).

9. M. O. Saltmarsh, W. R. Grimes, R. T. Santoro, "An Optimization of the Fission Fusion Hybrid Concept", Oak Ridge National Lab. Report, ORNL/PRA-79/3 (April 1979).

10. M. Z. Youssef, R. W. Conn, and W. F. Vogelsang, "Tritium and Fissile Fuel Exchange Between Hybrids, Fission Power Reactors, and Tritium Production Reactors", University of Wisconsin Fusion Engineering Program Report UWFDM-279, University of Wisconsin (1979).

11. K. R. Schultz, R. H. Brogli, G. R. Hopkins, M. Janzen and G. W. Shirley, "A $^{233}U$ Fusion-Fission Power System Without Reprocessing", General Atomic Co. Report GA-A14635-UC-20 (1977).

12. R. W. Conn et al., "SOLASE-H, A Laser Fusion Hybrid Study", Trans. Am. Nucl. Soc. 30, 58 (1978); also University of Wisconsin Fusion Engineering Program Report UWFDM-270, University of Wisconsin (1978).

13. M. Z. Youssef, R. W. Conn, and G. A. Moses, "Blanket Neutronics Studies for the SOLASE-H Hybrid Reactor," Trans. Am. Nucl. Soc., 32, 35 (1979); also University of Wisconsin Fusion Engineering Program Report UWFDM-263, University of Wisconsin (Oct. 1978).

References (cont.)

14. R. W. Conn, G. A. Moses, and S. I. Abdel-Khalik, "Notes on Fusion Hybrid Reactors", University of Wisconsin Fusion Engineering Program Report UWFDM-240, University of Wisconsin (1978).

15. H. A. Feiveson and T. B. Taylor, "Alternative Strategies for International Control of Nuclear Power", Report Prepared for the 1980's Project of the Council on Foreign Relations (1976).

16. M. Levenson, E. Zebrowski, "A Fast Breeder System Concept - A Diversion Resistant Fuel Cycle", 5th Energy Technology Conference, Washington, D.C., (Feb. 27, 1978).

17. W. W. Engle, Jr., "A Users Manual for ANISN", Oak Ridge National Laboratory Report RISC-CC-82, Oak Ridge National Laboratory (1967).

18. M. Z. Youssef, "ACTEN: A Computer Program to Calculate the Reaction Rate, the Rate of Energy Deposited and the Power Multiplication in a Fusion Blanket Using the Reaction Q Values", University of Wisconsin Fusion Engineering Program Report UWFDM-276, University of Wisconsin (Dec. 1978).

19. DLC-41/VITAMIN-C, RSIC Data Library Collection, Radiation Shielding Information Center, Oak Ridge National Laboratory (1977).

20. J. J. Duderstadt, Nuclear Reactor Analysis , John Wiley & Sons, Inc., p. 635 (1976).

21. M. Z. Youssef, R. W. Conn and G. A. Moses, "Burnup Calculations for the Laser Driven Fusion-Fission Fuel Factory, SOLASE-H", University of Wisconsin Fusion Engineering Program Report UWFDM-264, University of Wisconsin (Oct. 1978).

22. Code of Federal Regulations, Title 10, part 73.

23. L. Foderaro, "The Photon Shielding Manual", University Park, PA (1976).

24. M. J. Bell, "ORIGEN - The ORNL Isotope Generation and Depletion Code", Oak Ridge National Lab. Report ORNL-4628 (May 1973).