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NUCLEAR ANALYSIS ON THE ENRICHMENT OF PWR
FUEL IN A FUSION HYBRID REACTOR

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ABSTRACT

A preliminary nuclear analysis has been performed on the enrichment of zircaloy clad UO_2 fuel in a tokamak fusion hybrid reactor steam-cooled blanket for direct recycle in a Pressurized Water Reactor (PWR). This direct recycle scheme allows elimination of the chemical reprocessing and fuel refabrication steps between the hybrid and PWR and thus offers potential advantages with regard to both fuel cycle economics and nuclear proliferation concerns. Three fuel compositions on the U-Pu fuel cycle were investigated - the initial enrichment of either depleted (tails) uranium or natural uranium, and the refresh or re-enrichment of spent fuel discharged from a PWR. Results of the calculations indicate that for a tokamak plant factor averaged neutron wall loading of 1.2 MW m^{-2} , fissile enrichments of 3.5-4.0 percent can be achieved in ~ 5 -9 years. For a given wall loading, the time of irradiation in the hybrid depends mainly on the blanket thickness and the initial fissile content of the fuel. During irradiation in the hybrid the blanket fission power increases by a factor of one-half to three, depending on the fuel management scheme chosen, and results in a large amount of thermal power that is available for conversion to electric power. While in the hybrid the fuel undergoes a burnup of $\sim 20,000 \frac{\text{MWd}}{\text{T}}$ which results in significant fission product poison production. Various hybrid blanket fuel management schemes have been evaluated for the different fuels.

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1.0 INTRODUCTION

The fusion-fission hybrid reactor represents a potentially attractive method for producing fissile fuel from the fertile nuclides ^{238}U or ^{232}Th . The principal advantage of this system is based on the combination of the 14.1 MeV D-T fusion neutron energy and the neutron multiplying properties of the fertile materials. In order to maximize this neutron multiplication, most hybrid blanket designs strive to achieve as high a loading of fertile fuel as possible. However, in doing this, the resultant average concentration or enrichment of the bred fissile fuel in the fertile material can be relatively low. As such, most design studies assume the blanket will be removed periodically and then undergo chemical reprocessing to remove the bred fissile material, which would then be mixed and fabricated into a suitable form for use in a fission reactor.

In the study reported in this paper, a preliminary nuclear analysis has been performed on an alternative approach, in which the material in the hybrid blanket is irradiated for a sufficient length of time to achieve a fissile enrichment suitable for direct use in a fission reactor. Elimination of the chemical reprocessing and refabrication steps between the hybrid and the fission reactor offers potential advantages with regard to both fuel cycle economics and nuclear proliferation concerns. However, the direct recycle approach also introduces possible difficulties with regard to material radiation damage, fission-product poisoning of the fuel, and nonuniform enrichment of the fuel.

For this study, a neutronics analysis has been made on the irradiation and enrichment in a hybrid blanket of three compositions of uranium oxide (UO_2) fuel: depleted (tails) uranium with 0.25 w/o ^{235}U ; natural uranium with 0.71 w/o ^{235}U ; and spent fuel discharged from a standard Pressurized Water Reactor (PWR) after a burnup of 33,000 MWd/MTU. The spent fuel has an enrichment of 1.34 w/o fissile U and Pu after removal from the PWR. It was assumed that the fuel would be in the form of ~12 foot (366 cm) long rods. For the tails and natural uranium cases the fuel rods would be fabricated prior to irradiation in the

hybrid, and then bundled in a manner most appropriate for the hybrid blanket lattice. Similarly, the spent fuel rods would not be altered except to rebundle them in a suitable tight lattice configuration in the hybrid blanket. The fuel rod clad material was assumed to be zircaloy-4, and the hybrid blanket coolant was chosen to be steam in order to keep the neutron spectrum fairly hard and reduce the burn-out of the bred fissile fuel. Tritium self-sufficiency was not considered to be an absolute requirement of the hybrid device for this preliminary study, although modifications in the blanket design to breed sufficient tritium were considered.

The U-Pu fuel cycle with the UO_2 fuel form was chosen for this initial study since this is a well-known state-of-the-art technology for which a large data and operating experience basis exists. Present concerns with nuclear proliferation on the U-Pu cycle should be greatly reduced since the hybrid enrichment scheme both eliminates the chemical separation of Pu and also produces very "hot" (radioactive) fuel rods.

In Section 2.0 of this paper the calculational technique, geometric model, and various assumptions made in the study are discussed. Sections 3.0, 4.0, and 5.0 present the results on the enrichment of the three fuel compositions investigated: uranium tails, spent PWR fuel, and natural uranium. A comparison of the results for the three cases is given in Section 6.0 and a summary and conclusions from the study are presented in Section 7.0.

2.0 CALCULATIONAL PROCEDURE AND ASSUMPTIONS

The one-dimensional cylindrical geometry model used to assess the nuclear performance of the blanket is shown schematically in Figure 1. This type of representation, with the cylinder axis along the centerline of the torus, was selected as the most appropriate for the assumed tokamak hybrid design in which the UO_2 blanket is placed only on the outer part of the torus. Due mainly to access and maintenance limitations, the inner portion of the torus is used only for tritium breeding and radiation shielding. The isotropic 14 MeV D-T fusion neutron source is distributed uniformly over the plasma region. The plasma is surrounded by a void (scrape-off) layer and then by the vacuum vessel/first wall. For these calculations the tokamak major radius R_0 was 5.2 m, the plasma minor radius a was 1.2 m, the scrape-off layer thickness was 0.2 m, and the first wall was a 0.4 cm thick layer of stainless steel. These dimensions are similar to those chosen for a Demonstration Tokamak Hybrid Reactor (DTHR) pre-conceptual design⁽¹⁾.

The blanket containing the UO_2 fuel rods was selected to be 54 cm thick, based on considerations of minimizing both the back neutron leakage and also the fuel inventory. The blanket was divided into three zones, each 18 cm thick, for calculational purposes. A volume-averaged fissile enrichment was then calculated for each of the three zones. Since it was desired to maintain a hard neutron spectrum in the blanket, wet steam with a density of 0.064 gm cm^{-3} (4 lb ft^{-3}) was used as the blanket coolant. Zircaloy-4 was chosen as the clad and structural material. Based on typical PWR fuel rod sizes, blanket volume fractions for a tight lattice configuration were calculated to be

53 v/o	fuel (UO_2)
24 v/o	coolant (steam)
21 v/o	clad and structure (zircaloy)
2 v/o	gap (void)

Figure 1. Schematic Diagram of the One-Dimensional Cylindrical Geometry Used in the Neutronics Analysis.

For the depleted and natural uranium fueled blankets, the UO_2 pellet density was assumed to be about 94% of the theoretical density or 10.4 gm cm^{-3} .

The neutron transport calculations were performed using the one-dimensional discrete-ordinates code ANISN⁽²⁾ with an S_8 treatment of the angular quadrature and a P_3 approximation to the anisotropic scattering. The burn-up and depletion were analyzed using a combination of the CINDER⁽³⁾ and HIC⁽⁴⁾ codes which were appended⁽⁵⁾ to the ANISN code. For each time step in the calculation, ANISN supplies collapsed five group fluxes and cross sections to the CINDER-HIC module. The burn-up and isotope depletion are then calculated using a library of five group cross sections and decay constants, and new isotope number densities are returned to ANISN for the next time step. In this way, the program cycles through the various time steps requested, computing the isotope number densities at each step. Using a library containing the fission product yield and cross section data, the code also calculates a lumped fission product concentration and cross section which is then used in the ANISN calculation. A number of isotopes and transmutation and decay chains, which are of interest in the U-Pu cycle, are shown in Figure 2.

A 27 neutron group cross section set was used for the ANISN transport calculations. It was prepared by collapsing the 171 group CTR processed multi-group cross section library⁽⁶⁾ over an appropriate flux spectrum. Resonance effects in the ^{238}U were accounted for by using the self-shielding capabilities of the library.

The most important tokamak device parameters with regard to the nuclear analysis of the blanket performance are assumed to have the following values:

Major Radius	5.2 m
Minor Radius	1.2 m
Plasma Elongation	1.6
Neutron Wall Loading	2 MW m^{-2}
Plasma Duty Cycle (DC)	85%
Plant Availability (PA)	70%
Plasma D-T Power	950 MW

Figure 2. Isotopes and Decay Chains of Interest in U-Pu Fuel Cycle.

Hence, the plant factor averaged fusion neutron wall loading, which is directly proportional to the source current of 14 MeV neutrons, is $DC \times PA \times 2 \text{ MW m}^{-2} = 1.2 \text{ MW m}^{-2}$. This value is then used to calculate the neutron source strength for the ANISN-CINDER-HIC calculation.

Assuming a 0.54 m thick blanket located on the outer part of the torus only, plus accounting for major penetrations such as neutral beam injectors and a bundle divertor, the total available blanket volume is about 134 m^3 . Since the fuel volume fraction is assumed to be 0.53, the total fuel volume available for irradiation is about 71 m^3 .

There are a number of different fuel management schemes which might be considered during irradiation of the hybrid blanket to the necessary enrichment. Two possibilities which have been investigated for the three zone blanket configuration used in this study are illustrated in Figure 3. In the rotation scheme, the front and back blanket zones are interchanged periodically (typically every one to two years) while the middle zone is left in place. This could perhaps be achieved by lifting the blanket module out of the tokamak and simply rotating it 180° . Hence, after some length of time the entire blanket is removed with a uniform suitable average fissile enrichment.

In a second scheme, a batch of fuel in each blanket zone would be periodically shuffled inward as enriched fuel is removed from the front zone (adjacent to the plasma) and fresh fuel is loaded into the back zone. In this equilibrium cycle, each batch of fuel would be irradiated for the same length of time in each blanket zone. It could take ~ 10 years to reach this equilibrium cycle if, at the beginning of operation, the front zone is irradiated continuously (~ 5 years) until the required enrichment is achieved. In this scheme one of the blanket zones is removed at each refueling, unlike the rotational scheme where the entire blanket is replaced at one time.

For the irradiation of the three fuel compositions considered, viz, depleted uranium, natural uranium, and spent fuel - it was necessary to make some assumptions on the required final fissile enrichment of the fuel. For the

- a) Rotation Scheme, whereby the Front and Back Blanket Zones are Interchanged Periodically while the Middle Zone is Stationary.

- b) Inward Shuffling Scheme, whereby Batches of Fuel are Shuffled to Inner Blanket Zones Periodically as Fresh Fuel is Loaded in the Back Zone and Enriched Fuel is Removed from the Front Zone.

Figure 3. Two Possible Blanket Fuel Management Schemes.

depleted and natural uranium, a final enrichment of ~ 3.5 w/o fissile in total U + Pu was assumed to be needed. For the spent fuel, the final enrichment was required to be ~ 4.0 w/o, due to the higher residual fission product concentration in the fuel from the PWR burn-up.

3.0 ENRICHMENT OF DEPLETED URANIUM

The first series of calculations was for a blanket composed of depleted (tails) UO_2 fuel with an initial fissile enrichment of 0.25 w/o ^{235}U . The rotation type of fuel management scheme, as illustrated in the upper portion of Figure 3, was assumed in the analysis. That is, the front and back blanket zones were interchanged every two years while the middle zone was left in place. This procedure was continued until a blanket volume averaged fissile enrichment of ~ 3.5 w/o was achieved. The plant factor averaged neutron wall loading was 1.2 MW m^{-2} . The initial fuel density was 9.06 gms U/cm^3 . For the specific tokamak considered with a blanket fuel volume of $\sim 71 \text{ m}^3$, the initial blanket uranium (heavy metal) loading was ~ 643 tonnes of U.

The fissile enrichment in total U + Pu as a function of the blanket irradiation time is shown in Figure 4. This is the blanket volume averaged (over a 54 cm thickness) fissile enrichment under the fuel management scheme discussed previously. It can be seen that an enrichment of ~ 3.5 w/o is achieved after about 8.9 years of irradiation, with fissile content of approximately

0.13 w/o	^{235}U
3.28 w/o	^{239}Pu
0.09 w/o	^{241}Pu

After this time the heavy metal density has decreased due to burn-up to $\sim 8.85 \text{ gms (U+Pu)/cm}^3$, so that the end-of-life blanket inventory is $\sim 628 \text{ T (U+Pu)}$.

The volume averaged burn-up and fluence the fuel will experience, along with the fissile enrichment, are plotted versus irradiation time in Figure 5. The results show that while in the hybrid blanket, the fuel will undergo a burn-up of $\sim 19,000 \text{ MWd/T(U+Pu)}$ and be exposed to a fluence ($E > 0.1 \text{ MeV}$) of $\sim 6 \times 10^{22} \text{ n cm}^{-2}$ while reaching an enrichment of ~ 3.5 w/o. This fairly large burn-up results in significant fission product poison production, which will adversely affect the fuel's performance in a PWR. The fluence, which is about 10 times

Figure 4. Fissile Enrichment vs Time for the Irradiation of Depleted UO_2 Fuel.

Figure 5. Fissile Enrichment, Burnup, and Fluence for Depleted UO_2 Fuel as a Function of Irradiation Time.

larger than what the rod would experience during a standard three year cycle in a PWR core⁽⁷⁾, may cause severe radiation damage in the clad.

As can be seen from Figures 4 and 5, the increase in total fissile enrichment is not quite linear with irradiation time. The ^{235}U enrichment decreases with time, while the ^{239}Pu plus ^{241}Pu content increases with irradiation time, although at a slightly less than linear rate. The reason for this behavior is shown in Figure 6, where the net blanket enrichment rate, along with other important blanket parameters, is shown as a function of irradiation time. The net enrichment rate decreases slightly with exposure, from about $0.40 \frac{\text{w/o}}{\text{yr}}$ after one year to about $0.32 \frac{\text{w/o}}{\text{yr}}$ after 10 years. This decrease is reflected in the less than linear enrichment growth shown in Figure 4.

The basic reason behind this decreasing enrichment rate is also shown in Figure 6 by the plots of F and CR. The parameter F is defined as the net number of fissile atoms produced per incident plasma fusion neutron, and is the difference between the per neutron number of neutron captures in fertile nuclides and the number of neutron absorptions in fissile nuclides, i.e.,

$$F \equiv \frac{\text{Captures in } (^{238}\text{U} + ^{240}\text{Pu} + ^{238}\text{Pu}) - \text{Absorptions in } (^{235}\text{U} + ^{239}\text{Pu} + ^{241}\text{Pu})}{\text{Incident Fusion Neutron}}$$

The instantaneous blanket conversion ratio CR during plasma operation is likewise defined as the ratio of the captures in fertile to absorptions in fissile, i.e.,

$$\text{CR} = \frac{\text{Capture Reactions in } (^{238}\text{U} + ^{240}\text{Pu} + ^{238}\text{Pu})}{\text{Absorption Reactions in } (^{235}\text{U} + ^{239}\text{Pu} + ^{241}\text{Pu})}$$

As shown in Figure 6, both F and CR decrease with increasing irradiation time, which is the reason the net enrichment rate decreases. There are two main reasons why F and CR decrease. The first is that with increased exposure more fission neutrons are born within the blanket and the blanket leakage is larger, both out the back of the blanket and also out of the front of the blanket and into the inner torus shielding region. This is evident from the number of neutrons which are absorbed in the two shield regions (inner and outer, as shown in Figure 1) per fusion neutron: 0.70 at beginning-of-life (BOL) to 0.94 after

Figure 6. Time Variation of Important Blanket Neutronics Parameters for Irradiation of Depleted UO_2 Fuel.

10 years. This number also gives an indication of the approximate tritium breeding ratio that would be achievable without any blanket modification if lithium were used in the first part of the shield regions.

A second reason for the decrease in F and CR is increased fission (and thus destruction) of fissile atoms. This is shown in Figure 6 where the blanket energy multiplication factor M is plotted as a function of exposure time. Here, M is the energy multiplication due to blanket fissions and is defined by

$$M \equiv \frac{14.1 + 200 \times (\text{no. fission reactions/fusion neutron})}{14.1}$$

As the figure shows, M increases by nearly a factor of three over 10 years, from a value of 5.3 at BOL to about 15.1 after 10 years. Likewise, the blanket volume averaged power density increases from $\sim 14 \text{ watts/cm}^3$ to $\sim 46 \text{ watts/cm}^3$ over 10 years. The fuel rod linear power density also increases from $\sim 0.53 \frac{\text{kW}}{\text{ft}}$ (17.5 w/cm) to $\sim 1.74 \frac{\text{kW}}{\text{ft}}$ (57.0 w/cm). None of these values seem sufficiently large to present blanket cooling problems, although the increased power generation at end-of-life (EOL) would result in somewhat inefficient use of steam generators, turbines, etc at BOL when the power production is smaller. Certainly a constant power generation over the blanket lifetime would be desirable, and perhaps could be approached by optimization in the blanket fuel management scheme. The equilibrium cycle discussed in Section 4 results in much smaller variations of the blanket power with time.

At BOL, about 83% of the total fissions occur in the ^{238}U while the remainder occur in ^{235}U . As the Pu begins to build up, this distribution changes, so that after about six years of irradiation the ^{239}Pu fission rate is about the same as the ^{238}U fission rate. After 10 years of irradiation, about 57% of the fissions occur in ^{239}Pu , with about 36% in ^{238}U and the rest mostly in ^{235}U and ^{241}Pu .

While the fission rate grows with exposure, the blanket is still far from critical. For example, after 10 years of irradiation, when the enrichment is $\sim 3.9 \text{ w/o fissile}$, the blanket k_{eff} with the low density steam coolant is 0.62. If the coolant density is increased to that for normal water, e.g., if the

blanket was somehow flooded under accident conditions, the blanket k_{eff} increases to 0.87. In calculating these blanket eigenvalues the presence of the fission product poisons was neglected, so that these results on k_{eff} are conservative.

A plot of the time variation of the atomic number density of some of the more important nuclides in the blanket is shown in Figure 7. As expected, the density of all of the Pu isotopes increases with time. The ^{235}U density is fairly small at BOL with the depleted UO_2 fuel, and decreases further with exposure since there is no significant source for production of ^{235}U . An important isotope that has a fairly large density and production rate in these hybrid blankets is ^{237}Np , as shown in Figure 7. From the decay and transmutation schemes which are illustrated in Figure 2, it can be seen that the main method of production for ^{237}Np is

This will be an important consideration in the use of hybrid produced fuel in a PWR, since the ^{237}Np is essentially a parasitic nuclide with a thermal absorption cross section $> 100 \text{ b}$. A capture reaction in the ^{237}Np then leads to ^{238}Pu . As Figure 7 shows, the ^{238}Pu concentration is nearly as large as that of ^{241}Pu , which is formed by successive captures in ^{239}Pu and ^{240}Pu .

A plot of the time variation of the isotopic composition of the Pu is presented in Figure 8. The purity or fissile content of the bred Pu decreases from about 97.3 a/o fissile after one year to 88.4 a/o after ten years. The isotope ^{236}Pu , which is important because it decays to ^{232}U with a half-life of 2.9 years, is produced in very small quantities. For example, after one year of irradiation the ^{236}Pu content is $\sim 10 \text{ ppm}$ in Pu, and this increases to $\sim 400 \text{ ppm}$ after ten years. These results are in substantial agreement with those of Leonard⁽⁸⁾ for similar blanket concepts.

Figure 7. Blanket Isotope Number Density Variations vs Time for Irradiation of Depleted UO_2 Fuel.

Figure 8. Plutonium Isotopic Composition as a Function of Time for Irradiation of Depleted UO_2 Fuel.

4.0 REFRESH OF SPENT PWR FUEL

A second series of calculations was performed on the re-enrichment or refresh of spent PWR fuel. The fuel composition was assumed to be that of discharged PWR fuel after undergoing a burn up of $\sim 33,000$ Mwd/T, and to have a fissile enrichment of 1.34 w/o. Other important parameters of the spent fuel are given in Table 1.

TABLE 1
CONTENT AND DENSITY OF THE SPENT PWR FUEL

FISSILE ENRICHMENT	1.34 w/o
^{235}U CONTENT	0.71 w/o
^{239}Pu CONTENT	0.50 w/o
^{241}Pu CONTENT	0.13 w/o
PLUTONIUM ISOTOPICS	55 a/o ^{239}Pu 25 a/o ^{240}Pu 14 a/o ^{241}Pu 6 a/o ^{242}Pu
HEAVY METAL DENSITY	8.59 $\frac{\text{gms (U+Pu)}}{\text{cm}^3}$

The fuel management scheme chosen for the analysis was that of inward batch shuffling, as shown in the lower portion of Figure 3. The blanket was assumed to be kept in place and irradiated until the fuel batch A in the front blanket zone reached an average fissile enrichment over the 18 cm zone of ~ 4 w/o. Batch A was then removed, batch B was moved from the middle to the front zone, batch C was moved from the back to the middle zone, and a fresh batch D was loaded into the back zone. The blanket would then be further irradiated until

batch B in the front zone reached ~ 4 w/o fissile, at which time it would be removed, batches C and D would be shuffled inward, and fresh fuel batch E would be loaded into the back zone. Eventually, an equilibrium cycle would be reached in which a batch of fuel resides in each blanket zone for the same amount of time, beginning in the back zone and then being shuffled inward.

A plot of the enrichment of the spent PWR fuel as a function of irradiation time for the batch shuffling scheme discussed above is shown in Figure 9. During these calculations the plant factor averaged neutron wall loading was kept the same at 1.2 MW m^{-2} . It was found that for batch A in the front zone, an irradiation period of five years was needed to produce an average fissile enrichment of ~ 3.88 w/o. During this time the batch A burn up was $\sim 24,900 \frac{\text{MWd}}{\text{T(U+Pu)}}$. After this five year period, the fuel batch B in the middle zone reached an enrichment of ~ 3.10 w/o and fuel batch C in the back zone reached ~ 2.10 w/o. Recall that all had begun with a fissile content of 1.34 w/o.

After batch B was moved to the front zone and irradiated another two years, it reached an enrichment of ~ 3.95 w/o. Finally, when batch C was shuffled to the front zone and irradiated three more years, it achieved a fissile content of ~ 4.08 w/o. Hence, over a ten year exposure period three batches, or one whole blanket, have been enriched to ~ 4.0 w/o fissile, with two other fuel batches (D and E) enriched to 2.73 w/o and 1.82 w/o. It is clear from this analysis, and by inspection of Figure 9, that eventually some equilibrium shuffling cycle will be reached.

The equilibrium cycle for the enrichment of spent PWR fuel to ~ 4 w/o fissile content is shown in Figure 10. Under this scheme, a fuel batch is loaded into the back zone and irradiated 2.75 years. It is then moved to the middle zone for another 2.75 years of exposure, and then to the front zone for the final 2.75 years. Hence, the total residence time in the blanket for each batch of fuel is $3 \times 2.75 = 8.25$ years. The enrichment growth rates for each isotope in each blanket zone are shown as being linear in Figure 10. As discussed in the previous section, the enrichment is actually slightly less than linear with time; however, for convenience an average linear approximation was used in making Figure 10. As would be expected, the slopes are different in each

Figure 9. Fissile Enrichment by Fuel Batch vs Time for the Irradiation of Spent PWR Fuel.

Figure 10. Equilibrium Cycle Enrichment of Spent PWR Fuel as a Function of Irradiation Time.

blanket zone, with the steepest slope or largest enrichment rate occurring in the front blanket zone.

After the 8.25 year irradiation cycle, the fissile content of the fuel was approximately

0.42 w/o	^{235}U
3.40 w/o	^{239}Pu
0.18 w/o	^{241}Pu

The average burn up experienced in the hybrid blanket was about $21,600 \frac{\text{MWd}}{\text{T(U+Pu)}}$. This value, in addition to the $\sim 33,000 \frac{\text{MWd}}{\text{T}}$ burn up already on the fuel, results in the fuel having undergone $\sim 55,000 \frac{\text{MWd}}{\text{T}}$ burn up even before further reuse in the PWR. Upon removal from the hybrid, the fuel heavy metal density has decreased to $\sim 8.34 \frac{\text{gms (U+Pu)}}{\text{cm}^3}$. The blanket energy multiplication factor, M, has increased by about 60% during the 2.75 year equilibrium irradiation cycle, from a value of ~ 8 to ~ 13 .

The isotopic composition of the Pu in the hybrid blanket is shown as a function of irradiation time in Figure 11. At BOL the composition, as also shown in Table 1, is ~ 69 a/o fissile. After the equilibrium 8.25 year cycle the fissile content has increased to ~ 87 a/o fissile. The ^{240}Pu levels out at ~ 10 a/o, while the other ~ 3 a/o is ^{238}Pu and ^{242}Pu . Note that this isotopic distribution is not greatly different from that shown in Figure 8 for irradiation of depleted uranium.

Figure 11. Plutonium Isotopic Composition as a Function of Time for Irradiation of Spent PWR Fuel.

5.0 ENRICHMENT OF NATURAL URANIUM

For the third series of calculations the irradiation and enrichment of UO_2 rods composed of natural (0.71 w/o ^{235}U) was analyzed. The batch shuffling scheme where fuel batches are moved radially inward in the blanket, as discussed in the previous section, was assumed for the fuel management program. The desired final fissile enrichment was ~ 3.5 w/o. During the calculations the plant factor averaged neutron wall loading was kept at the same value of 1.2 MW m^{-2} .

A plot of the equilibrium cycle enrichment of the natural uranium fuel is shown in Figure 12. Also shown for comparison are the cycles for the depleted uranium and the spent PWR fuel. To reach ~ 3.5 w/o fissile, the natural uranium fuel is irradiated ~ 2.67 years in each of the three blanket zones, for a total residence time of $3 \times 2.67 = 8.0$ years. The fissile content of the fuel after this cycle is approximately

0.40 w/o	^{235}U
3.02 w/o	^{239}Pu
0.08 w/o	^{241}Pu

The burn up that this fuel experiences during irradiation in the hybrid is $\sim 17,400 \frac{\text{MWd}}{\text{T}(\text{U+Pu})}$. The fuel heavy metal density after discharge from the hybrid is $\sim 8.81 \frac{\text{gms}(\text{U+Pu})}{\text{cm}^3}$.

Figure 12. Equilibrium Cycle Enrichment vs Time for Irradiation of Depleted Uranium, Natural Uranium, and Spent PWR Fuel.

6.0 COMPARISON OF DEPLETED, NATURAL, AND SPENT FUEL ENRICHMENT

In this section a brief comparison is made on the enrichment of the three UO_2 fuel compositions investigated: depleted (tails) uranium, natural uranium, and spent PWR fuel. In order to permit easy comparisons, the same fuel management scheme of inward batch shuffling was assumed for each fuel type. Also, for the comparison an assumption was made that the final enrichment for irradiation of the depleted and natural uranium would be 3.5 w/o, while for the spent fuel the enrichment would be 4.0 w/o. Both of these values are larger than the present day PWR fuel enrichment of ~ 3.2 w/o because of the fission product build up during exposure in the hybrid. These values of 3.5 w/o and 4.0 w/o are based on estimates of the required enrichment for use in a PWR, and are used here only to compare the different fuel performances. If other final enrichment values are thought to be required, similar comparisons can be made by extrapolating the results presented in this and previous sections.

Based on recent studies⁽⁹⁾ of utilizing hybrid enriched fuel in a standard PWR, it has been determined that because of the isotopic composition of the fuel and the concentration of fission products that the following enrichments are necessary in order to achieve a burn up of $\sim 34,000 \frac{\text{MWd}}{\text{T}}$ in the PWR⁽⁹⁾:

- a) refresh of spent fuel - 5.0 w/o
- b) enrichment of depleted (tails) uranium - 3.9 w/o
- c) enrichment of natural uranium - 3.9 w/o

For the fuel enriched to 3.5 w/o from irradiation of depleted or natural uranium, the achievable fuel burn up in the PWR is⁽⁹⁾ about $29,700 \frac{\text{MWd}}{\text{T}}$. For the refresh of spent fuel to 4.0 w/o, the achievable fuel burn up in the PWR is⁽⁹⁾ $\sim 18,300 \frac{\text{MWd}}{\text{T}}$. All of these PWR burn up values assume, of course, that no fission products are extracted between irradiation in the hybrid and the PWR.

The growths in enrichment with irradiation time in the hybrid for the three fuel compositions are shown in Figure 12. For the assumed plant factor averaged wall loading of 1.2 MW m^{-2} , the natural uranium case takes about eight years to reach 3.5 w/o fissile content, averaged over the entire blanket volume, while the depleted uranium takes about 8.9 years. Likewise, the spent fuel case is irradiated for about 8.25 years to achieve 4.0 w/o enrichment. As discussed earlier, the enrichment growth rates shown in Figure 12 are average linear approximations, since the actual rate is slightly less than linear with time. By examination of Figure 12, it is evident that these average enrichment rates are different for each zone and are also slightly different for each fuel composition in the same zone.

These different enrichment rates for the three fuels and blanket zones are shown along with a summary of other important parameters in Table 2. It is interesting to note that for each of the fuels the initial ^{235}U enrichment has decreased by about a factor of two during the irradiation cycle. For the enrichment rates by zone, the highest values over the radiation cycle are for the depleted uranium case, while the lowest occur for the spent fuel case. This is because of the decrease in the net blanket fuel production with exposure time, and hence with higher fissile content, as discussed in Section 3.0 and Figure 6 for the depleted blanket case. Fairly large burn ups have been experienced by each of the fuels. The natural uranium has a slightly lower burn up than the depleted uranium due to its smaller residence time in the hybrid. If the depleted uranium fuel was in the blanket for the same length of time (eight years) as the natural uranium fuel, the burn up as shown in Figure 5 would be $\sim 16,000 \frac{\text{MWd}}{\text{T}}$. The spent fuel case has the largest burn up because of increased fissioning associated with its higher initial fissile content.

The enrichment rates by zone for the various fuels shown in Table 2 can be used to estimate irradiation times for the front blanket zone only, with thickness of 18 cm. For example, for the natural uranium case the front zone rate of $0.515 \frac{\text{w/o}}{\text{yr}}$ means that in ~ 5.4 years an average fissile content of 3.5 w/o will be reached in the front blanket zone if left in place. This then translates to

TABLE 2

SUMMARY OF ENRICHMENT PARAMETERS FOR THE THREE FUEL COMPOSITIONS
UNDER THE INWARD BATCH SHUFFLING FUEL MANAGEMENT SCHEME
IN A THREE ZONE BLANKET

	<u>DEPLETED URANIUM</u>	<u>NATURAL URANIUM</u>	<u>SPENT PWR FUEL</u>
INITIAL FISSILE ENRICHMENT, w/o	0.25	0.71	1.34
²³⁵ U CONTENT, w/o	0.25	0.71	0.71
²³⁹ Pu CONTENT, w/o	-	-	0.50
²⁴¹ Pu CONTENT, w/o	-	-	0.13
FINAL FISSILE ENRICHMENT, w/o	3.50	3.50	4.00
²³⁵ U CONTENT, w/o	0.13	0.40	0.42
²³⁹ Pu CONTENT, w/o	3.28	3.02	3.40
²⁴¹ Pu CONTENT, w/o	0.09	0.08	0.18
BATCH TOTAL IRRADIATION TIME, YEARS	8.9	8.0	8.25
BATCH IRRADIATION TIME PER ZONE (3 ZONES), YEARS	2.96	2.67	2.75
NET FISSILE ENRICHMENT RATE BY ZONE, w/o PER YEAR			
BLANKET AVERAGED	0.365	0.349	0.323
FRONT ZONE	0.527	0.515	0.462
MIDDLE ZONE	0.386	0.375	0.353
BACK ZONE	0.182	0.156	0.155
HYBRID DISCHARGE BURN UP, $\frac{\text{Mwd}}{\text{T(U+Pu)}}$	19,100	17,400	21,600

a fuel management scheme whereby the front zone batch of fuel is removed approximately every 5.4 years, while the middle and back zones are at lesser enrichments. This is in contrast to the reference inward batch shuffling scheme for the natural uranium, where one fuel batch is removed every 2.67 years after having spent a total of 8.0 years in the hybrid. The relative merits of these different fuel management schemes can best be resolved from a systems economic analyses, which is beyond the scope of this report.

As previously noted, the values of 3.5 or 4.0 w/o final enrichment were chosen simply to illustrate the relative performance trends of the various fuels; and that enrichments of ~ 3.9 w/o for the depleted or natural fuel and ~ 5.0 w/o for the spent fuel would be required to allow the fuel to achieve a burn-up of $\sim 34,000 \frac{\text{MWd}}{\text{T}}$ in the PWR. By use of the enrichment rates shown in Table 2, the irradiation time needed to reach these higher values can be estimated. For example, for the spent fuel the time to reach 5% would be

$$\frac{5.0 - 1.34}{0.323} \simeq 11.3 \text{ years.}$$

Similarly, to enrich the depleted and natural uranium to ~ 3.9 w/o fissile would require ~ 10 years and ~ 9.1 years, respectively. These irradiation times necessary to achieve the same energy output of the enriched fuel in the PWR are important considerations in the economics of the direct recycle process, as discussed in Reference 10.

7.0 SUMMARY AND CONCLUSIONS

A series of nuclear calculations has been performed to study the enrichment of UO_2 fuel in a hybrid fusion-fission reactor blanket for direct recycle in a PWR. Three fuel compositions were investigated: depleted (tails) uranium, natural uranium, and spent PWR fuel. All fuel was assumed to be in the form of rods which would be directly used in the PWR after removal from the hybrid, thereby avoiding the chemical separation and reprocessing and fuel refabrication steps in the PWR fuel cycle. This scenario offers potential advantages with regard to fuel cycle economics and nuclear proliferation concerns.

Results of the calculations indicate that for a tokamak plant factor averaged neutron wall loading of 1.2 MW m^{-2} , fissile enrichments of 3.5-4.0 w/o can be achieved in ~ 5 -9 years. The time of irradiation in the hybrid depends mainly on the blanket thickness and the initial fissile content of the fuel. For thinner blankets the irradiation time is shorter, although the annual production of enriched fuel is also smaller. Selection of the optimum situation will depend upon economic and fuel management considerations which were not addressed in this report. Preliminary calculations on the economics of direct recycle after hybrid enrichment are contained in a companion report⁽¹⁰⁾.

The average rate of enrichment or net fissile fuel production over the irradiation cycle was found to be lower for the spent fuel case than for the depleted uranium case. This is because as the fissile content of the fuel increases, the net fissile production rate decreases due to both increased blanket neutron leakage and increased fissile atom destruction (fission). The increased fissioning with irradiation time results in about a factor of one-half to three increase in the blanket power over the cycle time, depending on the fuel management scheme. This power production offers the possibility of significant electric power generation in the hybrid, which has been shown to be an important factor in the economics of the direct recycle scheme⁽¹⁰⁾. The fuel also undergoes a burn up of $\sim 20,000 \frac{\text{MWd}}{\text{T}}$ and a fluence of $\sim 5 \times 10^{22} \text{ n cm}^{-2}$ ($E > 0.1 \text{ MeV}$)

during irradiation in the hybrid blanket. The cycle average tritium breeding ratio is ~ 0.8 . The necessary tritium make up could be obtained by a blanket modification, by production in the PWR's being fueled by the hybrid, or a combination of these methods.

Overall, this nuclear analysis has tended to show that enrichment of PWR fuel in a hybrid blanket looks potentially attractive. However, further work remains to be done on ways to reduce the irradiation time in the hybrid, on different blanket fuel management schemes, and on economic analyses of the hybrid-PWR symbiosis.

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