acid dependence curves and the present, this could be attributed to the different conditions in the organic diluent or in the concentration of alkylamines chosen.

Technetium-99m is available as a radioactive tracer after isolating from neutron irradiated molybdenum compound. Figs. 1 and 2 give an isolation method basing on fairly high $K_e$ values of technetium and low values of molybdenum at 1 N HNO$_3$ in the amine extraction. Starting from 2.5 mg ammonium molybdate irradiated in JRR-1 for 5 hr and cooled for 6 days, about 1 mc of technetium-99m was extracted with 5 % Amberlite LA-1 xylene solution from 1 N HNO$_3$. The radiochemical purity was shown to be 99.8% from the decay curve measurement (Fig. 3). The separation process used is represented in Fig. 4.

Figs. 5 and 6 give the exact $K_e$ values for each element. These figures serve an advantage to see easily the difference of $K_e$ values of tow elements.

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黒鉛減速ガス冷却炉におけるプルトニウムリサイクルの研究, (II)

By Kunihiko UEMATSU*

The modified bidirectional fuel management method on plutonium recycle in a gas-cooled and graphite-moderated reactor was investigated. The modified bidirectional fuel management was so designed that the fresher fuel and the more burned fuel existed at the outer and inner regions of the reactor, respectively, to reduce the peak to average ratio of power density.

This fueling method was effective to reduce the peak to average ratio of power density without sacrificing much burn-up of the fuel. The optimum irradiation control factors, $A's$, were at around 0.5 and 0.3 for natural uranium feed and 1.0 % enriched uranium feed, respectively.

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INTRODUCTION

The feasibility of plutonium recycle in a gas-cooled, graphite-moderated reactor was previously investigated in Part I of this report. The advantage of recycling plutonium in this type of reactor was to reduce consumption of enriched uranium, leading to possible use of only natural uranium for fuel make up after several cycles. According to the Part I of this report, the use of bidirectional fueling and plutonium recycle greatly increased the amount of energy that could be liberated from a given amount of uranium fuel of specified enrichment.

Plutonium recycling was also favorable for reducing the peak to average ratio of power density which would eventually lead to the production of the greater amount of energy which was limited by the maximum power density of the reactor. For example, the peak to average power density ratio was reduced from 3.14 to 2.53 by recycling 90% of plutonium produced in the reactor which was fueled with 1.0 % enriched uranium. Since the maximum fuel temperature is one of the limiting design factors of the reactor, a certain limit on the maximum power density should be imposed depending upon the type and material of the fuel element used in the reactor. If the peak to average power density ratio is reduced by fuel management procedures, by changing the composition of fuel, or by other means, the average power density can be accordingly increased without changing the maximum power density. Although the possible heat production increase is not directly proportional to decrement of the peak to average power density ratio, this reactor, from the numbers previously shown, may have a possibility to produce about 3.14/ 2.53 = 1.24 times more heat with recycling plutonium than without it.

Therefore, the reduction of peak to average power density ratio is very essential in the power reactor operation. This can be done by several methods; however, the modified bidirectional fuel management method has been applied for this purpose in this work.

The modified bidirectional fuel management method has the radially distributed discharge flux-time instead of the constant discharge flux-time of the ordinary bidirectional fuel management method which has been investigated in Part I of this report. The modification on the ordinary bidirectional fuel management is such that the radial discharge flux-time distribution has higher flux-time at the inner region and lower flux-time at the outer region of the core. This leads to the presence of fresher fuel at the outer region and of more consumed fuel at the inner region of the core. Therefore, the power density tends to increase at the outer region and to decrease at the inner region of the core.

I. MODIFIED BIDIRECTIONAL FUEL MANAGEMENT

In ordinary bidirectional fuel management, fuel is charged at one end-face of the reactor, moved progressively through the fuel channels parallel to the axis of the reactor, and discharged at the opposite end-face of the reactor when irradiated to a specified discharge flux-time. Fuel in an adjacent channel is moved in opposite directions so that the reduced reactivity of depleted fuel in one channel combines with the high reactivity of the fresh fuel in an adjacent channel to prevent flux and power peaking near one end-face of the reactor and to produce symmetry of these distributions with respect to the axial center plane of the reactor. The discharge flux-time is a constant in radial direction and the neutron flux distribution is very close to the typical flux shape, cosine or $J_0$ curve of the core geometry. Thus the power density distribution also follows cosine or $J_0$ curve. As a result, the maximum power density appears at the center of the core.

It is known from Part I of this report that lowering the peak to average power density ratio can be achieved with plutonium recycling and also with feeding uranium of higher enrichment. The power density distribution is, however, still close to cosine or $J_0$ curve in both plutonium recycling and higher enriched uranium feed, the reduction
of the peak to average power density ratio is somewhat limited in these methods.

There is another method to reduce this ratio without greatly sacrificing the fuel burn-up which will affect the fuel cycle cost. Since the power density is directly proportional to the product of neutron flux, fission cross section, and nuclide concentration of the fuel, the simple way to reduce the peak to average power density ratio is to reduce those three factors at the inner region and to increase them at the outer region of the core. One way to do this is to arrange the fuel as such that more burned fuel is presented at the inner region and fresher fuel exists at the outer region of the core. This may be done by changing the radial distribution of the flux-time of fuel discharged from each channel by modifying the fuel feed velocity.

In this study, the discharge flux-time is changed in such a way that the discharge flux-time decreases linearly from the center to the edge of the reactor. This discharge flux-time is given as

\[ \Theta\left(\frac{r}{R}\right) = \Theta_0 \left(1 - A\left(\frac{r}{R}\right)\right) \]

where
- \( \Theta_0 \): discharge flux-time along the axis of the core
- \( \Theta\left(\frac{r}{R}\right) \): discharge flux-time at the position \( \left(\frac{r}{R}\right) \)

This is shown in Fig. 1.

In this study, the effect of the fuel irradiation control factor \( A \) on the reactor physics characteristics is investigated. Changes in \( A \) factor have been made ranging \( A=0 \sim 0.7 \) for the case of natural uranium feed and \( A=0 \sim 0.5 \) for the case of 1.0 % enriched uranium feed.

II. RESULTS OF CALCULATION

The calculation of the reactor physics characteristics has been limited to the “recycle steady state” which was defined in the Part I of this report. There is no doubt that the recycle steady state will be achieved in either case by the ordinary bidirectional fuel management or the modified bidirectional fuel management.

The cases studied in this work are the modified bidirectional fuel management with natural uranium or 1.0 % enriched uranium feed blended with 90% of plutonium produced at the recycle steady state of the reactor. The fuel irradiation control factor \( A \) which has been investigated are 0, 0.3, 0.5 and 0.7 for natural uranium feed case and 0, 0.3 and 0.5 for 1.0 % enriched uranium feed case.

The radial and axial variation of the normalized flux are shown in Figs. 2 and 3.
In the ordinary bidirectional fuel management (with $A=0$), the author reported that the neutron flux distribution change due to the higher uranium enrichment occurred only in the axial direction. This is a reasonable result. Since the effect of the enrichment change in uranium feed on the neutron flux along each fuel channel is such that, with the neutron flux at the both ends of the fuel channel kept constant, the peak flux was boosted with increasing $^{235}$U enrichment. However, there were no significant reasons to change the radial neutron flux distribution. Therefore, the axial neutron flux distribution only was flattened with increasing uranium enrichment.

In the modified bidirectional fuel management ($A>0$), the neutron flux change was observed in both directions. The axial neutron flux change with $A$ was not so significant as that of the radial direction. The reason for the axial flux flattening is the same as the aforementioned. The higher the factor $A$, the greater the flattening; this was mainly due to the by-effect of the radial flux shape change.

The radial flux distributions were quite significantly changed and showed strong peaking with increasing $A$. These were, of course, the results of the modified bidirectional fuel management. Since this new fuel management procedure was so designed that fresher fuel existed at the outer region of the core, it resulted in the peaking of the radial flux distribution.

This whole picture will be more clearly understood from three dimensional representation.
tion of the normalized power density of Fig. 4.

From Figs. 2 and 3 which show strong peaking in the neutron flux or power density with increasing irradiation control factor A, it may appear that the modified bidirectional fuel management is no better than the ordinary bidirectional fuel management. However, one should realize that those figures represent only the neutron flux or power density normalized at the mesh point (1, 1).

An example of power density distribution is presented in Fig. 5. The curves show the gradual decrease of the maximum power density with increasing factor A. As can be observed from these curves, the minimum point for maximum power density occurs approximately at A=0.5 for natural uranium feed. With higher numbers than indicated above, the power densities show a strong depression at the central region of the core which is even less than the average power density. Therefore, the existence of the limiting value of A for the most desirable reactor physics characteristics is proven.

![Power density distribution](image)

Fig. 5 Radial power density distribution at axial mesh point 1 (Natural U, 0.90 Pu-recycle, bidirectional)

Table 1 Maximum power density (W/cm³) and maximum conductivity integral \( \int kdT \) (W/cm) of fuel

<table>
<thead>
<tr>
<th>A</th>
<th>Natural U</th>
<th>1.0 % U</th>
<th>1.3 % U</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Core</td>
<td>Fuel</td>
<td>Core</td>
</tr>
<tr>
<td>0</td>
<td>Max. power density</td>
<td>2.685</td>
<td>162.4</td>
</tr>
<tr>
<td></td>
<td>Max. conductivity integral</td>
<td>—</td>
<td>87.9</td>
</tr>
<tr>
<td>0.3</td>
<td>Max. power density</td>
<td>1.933</td>
<td>117.4</td>
</tr>
<tr>
<td></td>
<td>Max. conductivity integral</td>
<td>—</td>
<td>63.5</td>
</tr>
<tr>
<td>0.5</td>
<td>Max. power density</td>
<td>1.760</td>
<td>106.8</td>
</tr>
<tr>
<td></td>
<td>Max. conductivity integral</td>
<td>—</td>
<td>57.8</td>
</tr>
<tr>
<td>0.7</td>
<td>Max. power density</td>
<td>1.788</td>
<td>108.6</td>
</tr>
<tr>
<td></td>
<td>Max. conductivity integral</td>
<td>—</td>
<td>58.8</td>
</tr>
</tbody>
</table>

Table 1 is the summary of the maximum power density of the core and the fuel and the conductivity integral, \( \int kdT \), of the fuel. The maximum power density of the fuel element, which occurs at the same point as the maximum power density of the core, is computed by assuming that 92% of the total heat would be produced in the fuel element\(^{(2)}\). Using this maximum power density of the fuel, the conductivity integral is computed from the following equation.

\[
\int_{T_s}^{T_e} kdT = \frac{H_{\text{max}}}{2} \int_0^{R_1} r dr,
\]

where

- \( k \): thermal conductivity
- \( T_s \): fuel centerline temperature
- \( T_e \): fuel rod surface temperature
- \( H_{\text{max}} \): maximum power density
- \( R_1 \): radius of the fuel rod.
According to the recent experimental studies of uranium oxide fuel on its thermal performance, the design limit assigned in terms of the conductivity integral is $65 \sim 70$ W/cm for melting of fuel and 40 W/cm for preventing the release of fission products. The maximum conductivity integrals shown in Table 1 exceed the design limit for uranium oxide. However, the thermal conductivity of uranium carbide which is the fuel used in this reference reactor is about six times greater than that of uranium oxide. Therefore, even the highest number, 87.9 W/cm, appeared in Table 1, may well be within the reasonable design limit of uranium carbide, which is not available at present.

The low values for the maximum conductivity integral are 57.8 W/cm at $A = 0.5$ for natural feed, and 50.9 W/cm at $A = 0.3$ for 1.0 % enriched uranium feed. From the design standpoint of the power reactor, it is reasonable to push these fuel performances up to 87.9 W/cm which was the design maximum for the reactor in consideration. This considerably increases the thermal output of the entire system.

The average and center burn-up and the peak to average power density ratio are shown in Figs. 6 and 7. Although the center burn-up increases with increasing $A$, the average burn-up conversely decreases with increasing $A$. This behavior may be explained by the fact that some of the fuel at the outer region of the core is discharged before the local reactivity of the fuel reaches zero. The decrease of neutron non-leakage probability, which is related to the neutron flux flattening, is also responsible for this behavior (refer to Table 2).

The peak to average power density ratio shows the minimum point at $A = 0.5$ for natural uranium feed and at $A = 0.3$ for 1.0 % enriched uranium feed.

Fig. 8 shows plutonium output concentrations at the recycle steady state and also shows the quality of plutonium, which is defined as the ratio of the fissionable plutonium to the total plutonium. The quality of plutonium improves slightly with increasing $A$. This improvement is mainly due to the lower

$k_{UC} = 0.06 \text{ cal/(sec)(cm)(}^\circ \text{C)}$ at $100^\circ \sim 700^\circ \text{C}^{(4)}$

$k_{UO_2} = 0.011 \text{ cal/(sec)(cm)(}^\circ \text{C)}$ at $600^\circ \text{C}^{(5)}$
average burn-up of the fuel. The lower burn-up leaves more of $^{239}\text{Pu}$, and this leads subsequently to less of higher plutonium isotopes. The total amount of plutonium does not show any significant change over the range of $A$ in both cases of uranium enrichments.

Table 2 Neutron non-leakage probability

<table>
<thead>
<tr>
<th>Enrichment</th>
<th>$A$</th>
<th>$P_1$</th>
<th>$DB^p$ 1/cm of the core</th>
</tr>
</thead>
<tbody>
<tr>
<td>Natural U</td>
<td>0</td>
<td>0.98910</td>
<td>2.7145 × 10^{-5}</td>
</tr>
<tr>
<td></td>
<td>0.3</td>
<td>0.98730</td>
<td>3.1490 × 10^{-5}</td>
</tr>
<tr>
<td></td>
<td>0.5</td>
<td>0.98564</td>
<td>3.5503 × 10^{-5}</td>
</tr>
<tr>
<td></td>
<td>0.7</td>
<td>0.98327</td>
<td>4.1211 × 10^{-5}</td>
</tr>
<tr>
<td>1.0% enriched U</td>
<td>0</td>
<td>0.98799</td>
<td>2.9843 × 10^{-5}</td>
</tr>
<tr>
<td></td>
<td>0.3</td>
<td>0.98496</td>
<td>3.7131 × 10^{-5}</td>
</tr>
<tr>
<td></td>
<td>0.5</td>
<td>0.98232</td>
<td>4.3462 × 10^{-5}</td>
</tr>
<tr>
<td>1.3% enriched U</td>
<td>0</td>
<td>0.98566</td>
<td>3.5266 × 10^{-5}</td>
</tr>
</tbody>
</table>

Initial values: $P_1 = 0.9899$  
$DB^p = 2.455 \times 10^{-5}$ 1/cm of the core

Normalized fuel feeding velocities which have been calculated by using Eq. (2) are given in Fig. 9. Comparing Fig. 9 with Fig. 2 (radial variation of normalized flux), there is close similarity in the characteristics of the curves in both figures. This can be easily explained from Eq. (2). $\theta(r/R)$ and $\theta_0$ are linearly proportional from the definition of the modified bidirectional fuel management; then from Eq. (2)

$$
\frac{v(r/R)}{v_0} = \alpha \int \Phi(r/R, z)dz/\int \Phi(0, z)dz,
$$

where $\alpha = 1/(1 - A(r/R))$.  
Now, if the flux $\Phi(r/R, z)$ can be separated into the product of two variables $\Phi(r/R)$ and $\Psi(z)$, which is approximately true, Eq. (4) becomes to Eq. (5)

$$
\frac{v(r/R)}{v_0} = \alpha \frac{\Phi(r/R) \int \Psi(z)dz}{\Phi(0) \int \Psi(z)dz},
$$

Therefore, one may find that the fuel feeding
velocity follows closely the flux shape in the radial direction. In other words, we might be able to control the radial flux shape by controlling the fuel feeding velocity so that the peak to average power density ratio gives the lowest value.

From the operational stand point of the power reactor, the constant fuel feeding velocity is the most preferable, since this will not confuse the reactor operator who should feed the fuel into several hundreds of fuel channels from time to time. In such a case, the flux shape in radial direction would be close to the most favorable flux distribution, since the radial flux distribution must be close to the flat flux.

III. SUMMARY

Fuel feed cases studied consist of natural uranium or 1.0% enriched uranium blended with recycle of 90% of the plutonium produced in the reactor. The fuel irradiation control factor, A, which characterizes the modified bidirectional fueling, investigated in this paper were 0, 0.3 and 0.5 for the case of 1.0% enriched uranium feed. The principal aim of the modified bidirectional fuel management was to improve the peak to average power density ratio over the improvement which could be expected from the recycle of plutonium and the uranium feed of higher enrichment.

The best peak to average power density ratios obtained with the modified bidirectional fuel management were:

1.93 at $A = 0.5$ for natural uranium feed blended with 90% plutonium produced in the reactor;

1.70 at $A = 0.3$ for 1.0% enriched uranium feed blended with 90% plutonium produced in the reactor.

The best numbers obtained with the ordinary bidirectional fuel management were:

2.94 for natural uranium feed blended with 90% plutonium produced in the reactor;

2.53 for 1.0% enriched uranium feed blended with 90% plutonium produced in the reactor;

3.14 for 1.0% enriched uranium feed without plutonium recycle.

Therefore, the modified bidirectional management could achieve substantial improvement in the peak to average power density ratio. However, the average burn-up experienced by the fuel showed a slight decrease with the modified bidirectional fuel management. For example, the average burn-up changes were as follows:

- 9086 MWD/t of fuel for the ordinary bidirectional fuel management with natural uranium feed blended with 90% plutonium produced in the reactor;
- 8340 MWD/t of fuel for the modified bidirectional fuel management ($A = 0.5$) with natural uranium feed blended with 90% plutonium produced in the reactor;
- 15167 MWD/t of fuel for the ordinary bidirectional fuel management with 1.0% enriched uranium feed blended with 90% plutonium produced in the reactor;
- 14538 MWD/t of fuel for the modified bidirectional fuel management ($A = 0.3$) with 1.0% enriched uranium feed blended with 90% plutonium produced in the reactor;
- 11352 MWD/t of fuel for the ordinary bidirectional fuel management with 1.0% enriched uranium feed which was not blended with plutonium.

(Received August 10, 1962)

--- Reference ---

3. WAPD-228, (1960).

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