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**Interim Report on Core Physics and Fuel
Cycle Analysis of the Pebble Bed
Reactor Power Plant Concept**

D. R. Vondy

OAK RIDGE NATIONAL LABORATORY
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Neutron Physics Division

INTERIM REPORT ON CORE PHYSICS AND FUEL CYCLE ANALYSIS
OF THE PEBBLE BED REACTOR POWER PLANT CONCEPT

D. R. Vondy

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ABSTRACT

Calculations were made to predict the performance of a pebble bed reactor operated in a mode to produce fissile fuel (high conversion or breeding). Both a one pebble design and a design involving large primary feed pebbles and small fertile pebbles were considered. A relatively short residence time of the primary pebbles loaded with U^{233} fuel was found to be necessary to achieve a high breeding ratio, but this leads to relatively high fuel costs. A high fissile inventory is associated with a low C/Th ratio and a high thorium loading, causing the doubling time to be long, even though the breeding ratio is high, and the fuel cost of electrical product to be high. Production of U^{233} fuel from U^{235} feed was studied and performances of the converter and breeder reactor concepts were examined varying the key parameters.

- - - -

Introduction

This is an interim report on the independent review of the potential of the pebble bed reactor¹ for producing fissile fuel. Primary consideration was given to the two pebble concept suggested by the General Electric Company² for achieving a higher breeding ratio than is indicated by German studies of the potential of modifications to the one pebble design of their AVR reactor.

We qualify the information presented herein due to the nature of this exploratory assessment and the low level of effort. Much of this effort was expended in familiarization with the behavior and neutronics characteristics of the reactor, tailoring the available analysis methods for the calculations and establishing a reasonable description of the reactor

implied by the general concept in the form of computer code input data. Specific weaknesses in the calculations include the general areas of modeling of the reactor and the fuel management, calculation of the neutron flux distribution in space and energy for desired conditions, resonance shielding, fission product poisoning, reactor control, and sources of perturbation and their effects. Minor aspects have been ignored.*

We report results of simple calculations made to assess aspects of interest to this review in a way which facilitates understanding, done both by hand and by computer.³ Results are also reported which were obtained from rather sophisticated machine calculations using the code system which applies finite-difference diffusion theory (VENTURE neutronics)⁴ and carries out an exposure calculation (BURNER)⁵. An illustration of application of the continuous fueling, once through model is shown for (R,Z) geometry in Fig. 1. Few group cross sections were generated with the AMPX⁶ code system (NITAWL, XSDRNPM) using inhouse evaluated 123 group cross section data with resonance parameters. In the course of this effort the two computation systems were coupled to allow few group cross sections to be produced and a problem solved in a single computer run. Few group calculations were done with two, four and eight groups (one thermal group).

Herein, simple definitions have been used for conversion ratio and doubling time. For simplicity, Pa^{233} has been identified as a fuel nuclide for the one and two dimensional calculations so conversion (breeding) ratio was calculated as

$$C = \frac{\text{Capture rate in Th}^{232} \text{ and U}^{234}}{\text{Absorption rate in U}^{233}, \text{U}^{235} \text{ and Pa}^{233}} \quad (1)$$

Otherwise the Pa^{233} absorptions were counted as a loss in the numerator, without any nuclide importance weighting. Doubling times have been

*Oxygen and silicon content and hence effects on neutron economy were ignored.

NOTE.

THIS IS AN ILLUSTRATION OF ONE OF THE
MANY POSSIBILITIES.

THREE FLOW PATHS THROUGH AXIAL BLANKET
AND CORE ARE SHOWN.

Fig. 1. The Once-Through, Continuous Fueling Model

calculated for the fissile inventory doubling without considering availability for compounding nor allowing for extra inventory, but considering a plant load factor of 0.9. Residence time is reported as at full power, but may also be used as that at the reduced power associated with a load factor < 1 .

In the results reported, the usually defined fissile nuclides (U^{233} , U^{235} , etc.), those with thermal η values > 1 , were given equal weights in fuel generation and use calculations.

Certain Basic Aspects

Consider that a breeder reactor is to be operated on primarily U^{233} fuel. This fuel must come from a reactor supplied with available fuel, probably highly enriched uranium, primarily U^{235} . There must be concern about where the U^{233} feed material is to come from and how much contamination of the higher uranium isotopes it contains. After a breeder reactor becomes operational it is deemed necessary for it to use its own product in recycle, and the excess used to fuel additional breeder reactors. Thus there are basically three different reactor fueling conditions of interest:

- 1) U^{235} feed, high conversion,
- 2) Converter product as breeder feed, shifting to recycle of discharge, and
- 3) Excess breeder product as breeder feed, shifting to recycle of discharge.

Considering that only a relatively small breeding gain is expected for the reactor concept under study, the possible long term need for converters fed by breeders is not addressed due to its position far in the future, and 3) above is of small concern and not addressed.

Doubling time will be used here, or more usefully its reciprocal in yrs^{-1} when the conversion ratio is less than unity, in the simple sense of

$$\frac{1}{T_d} = \frac{Pu}{I\eta_e} (C-1) \frac{dF}{dE} , \quad (2)$$

where

P is the design power level,

u is the load factor,

I is the associated fuel inventory,

C is the conversion (breeding) ratio,

η_e is the energy conversion efficiency, and

dF/dE is the amount of fuel consumed per unit energy produced,
consistent units required.

Note that for a breeder, the smaller the doubling time the better, or generally the more positive the reciprocal doubling time the better. Decreasing the energy conversion efficiency increases the reciprocal doubling time, but economics demand a high value of η_e generally, except possibly for a converter to produce breeder fuel.

Considering the requirement for a steady state (critical) reactor on the average, the achievable conversion ratio, in terms of effective reaction rates, is given by allowing maximum neutron capture in the fertile material. Using average reaction rates (fraction of total neutron losses over all nuclides of interest, possibly with special weighting),

$$R = (\eta-1) A - 2H - L - Z - Y \quad (3)$$

$$C = F \left[\frac{(\eta-1) A - 2H - L - Z - Y}{A} \right] , \quad (4)$$

where

A is the neutron absorption rate in fuel,

R is the neutron capture rate in fertile,

C is the conversion (breeding) ratio,

F is a factor to account for recovery loss (a form chosen here for simplicity),

H is the loss rate of potential fuel (Pa^{233} absorption removes a neutron and destroys a potential full atom, the effect of U^{234} generation neglected,

L is the leakage loss rate,

Z is a relatively set neutron absorption rate (C, structure, Xe),

Y is the loss rate to fission products, nearly linear with exposure, and

η is the fissile effective $\text{v}\sigma_f/\sigma_a$.

Thus an approximation for the reciprocal doubling time is

$$\frac{1}{T_d} = \frac{\text{PuF}}{I\eta_e} \left[\frac{(\eta-1)A - 2H - L - Z - Y}{A} - 1 \right] \frac{dF}{dE} . \quad (5)$$

Linear perturbations may be directly assessed,

$$\frac{\Delta C}{\Delta x} \approx \frac{F\partial}{\partial x} \left[\frac{(\eta-1)A - 2H - L - Z - Y}{A} \right] . \quad (6)$$

For example, changing the Pa^{233} absorption rate gives

$$\Delta C \approx \frac{-2F}{A} (\Delta H) .$$

Further

$$\frac{\Delta \left[\frac{1}{T_d} \right]}{\Delta x} \approx \frac{\partial}{\partial x} \left\{ \frac{\text{PuF}}{I\eta_e} \left[\frac{(\eta-1)A - 2H - L - Z - Y}{A} \right] \frac{dF}{dE} \right\} . \quad (7)$$

For example

$$\frac{\Delta \left[\frac{1}{T_d} \right]}{\frac{1}{(T_d)}} \approx \frac{-2\Delta H}{[(\eta-1)A - 2H - L - Z - Y]} ,$$

or for a range where $T_d \neq 0$, for two conditions considering only change in H,

$$\frac{T_{d2}}{T_{d1}} = \left[\frac{(\eta-1)A - 2H_1 - L - Z - Y}{(\eta-1)A - 2H_2 - L - Z - Y} \right] .$$

Simple calculations with the base data in use yield the following results of η for U^{233} , U^{235} fuel mixtures as dependent on the fast to thermal flux ratio:

Fraction Absorption		ϕ_1/ϕ_2					
U^{233}	U^{235}	0	1	2.5	5.	7.5	10
1.0	0.	2.243	2.233	2.218	2.197	2.171	2.155
0.9	0.1	2.223	2.208	2.191	2.167	2.138	2.117
0.5	0.5	2.142	2.108	2.084	2.046	2.004	1.966
0.1	0.9	2.060	2.007	1.977	1.925	1.870	1.815
0	1.0	2.040	1.982	1.950	1.895	1.836	1.777

Note that subtracting unity from η places an upper bound on the possible value of the conversion (breeding) ratio. Unavoidable losses reduce this; a one percent loss to fission products reduces the conversion ratio by about 2 percent while a one percent loss to Pa^{233} reduces the conversion ratio by about 4 percent. The flux ratio for a value of C/Th of 80 is estimated at near 5, and a reduction in the conversion ratio of .0027 is estimated for each percent of the absorptions in U^{235} of the total fuel absorptions.

Other factors affecting the conversion ratio are examined later.

Much of the effort in this study was concentrated on a design having a carbon to thorium ratio of 80 in the reactor core and a reactor size of 1,500 Mw_e (3,750 Mw_{th}) with a power density of 6.25 W_{th}/cc . The results obtained for these conditions indicated that a higher C/Th was needed to reduce the fissile inventory, so a value of 110 was then used. Also it appeared that a power density of 5 W_{th}/cc is more reasonable, so the plant power level was reduced to 1,200 Mw_e (3,000 Mw_{th}) for a core volume of 6×10^8 cc.

Notes on Economics

Simple economic analysis was done. Unit costs used here were 200 \$/kgm fabrication, 150\$/kgm processing, shipping and waste disposal, and a plant load factor of 0.9. For a net conversion ratio of unity (after losses) the direct fuel costs would be zero. The direct fabrication and processing costs are given on the basis of one gram of fuel as

$$X \simeq \frac{350L}{7.884 W_e T_r} ,$$

where

L is the heavy metal loading, gm/cc,

W_e is the power density, electrical w/cc, full power,

T_r is the residence time, years at reduced power,

$$7.884 = 8760 \times 0.9 \div 10^3, \text{ and}$$

the result is in units of \$/Mw-hr = mil/Kw-hr electrical.

This cost is shown below for a loading of 0.2 gm/cc as dependent on the power density and residence time for $\eta_e = 0.4$:

W_{th}/cc	W_e/cc	Residence time (yrs)			
		0.5	1.0	2.0	4.0
3	1.2	14.8	7.4	3.7	1.9
6	2.4	7.4	3.7	1.8	0.9
9	3.6	4.9	2.5	1.2	0.6

Note that for either low power density or short residence time, the direct cost for treating the fuel becomes quite uncompetitive.

Similarly the indirect charges for the fuel are quite dependent on power density and residence time. With the assumptions of a net conversion

(breeding) ratio of unity and one-half year lead time and one year delay in recovery, for a fissile loading of G gm/cc and an interest rate of i annually, the indirect cost of fuel inventory worth 35 \$/gm (U^{233}) amounts to

$$Y \approx \frac{35,000 iG}{7.884 W_e} \left[1 + \frac{3}{2T_r} \right] .$$

Results dependent on power density and core residence time for an interest charge of 0.1 and a fissile loading of 0.006 gm/cc are shown below:

W_{th}/cc	W_e/cc	Residence time (yrs)			
		0.5	1.0	2.0	4.0
3	1.2	8.9	5.5	3.9	3.1
6	2.4	4.4	2.8	1.9	1.5
9	3.6	3.0	1.8	1.3	1.0

Note that this indirect charge is also quite uncompetitive for low power density and short residence time. (Delay in recovery of fuel could also be examined directly since the indirect cost is essentially linear with the delay time.)

The sum of these two contributions is shown below:

W_{th}/cc	W_e/cc	Residence time (yrs)			
		0.5	1.0	2.0	4.0
3	1.2	23.7	12.9	7.6	5.0
6	2.4	11.8	6.4	3.8	2.5
9	3.6	7.9	4.3	2.5	1.7

Note that a required increase in fissile loading with residence time has been ignored.

Operation of the plant at a power level below design does not affect the direct costs (residence time adjusted as the reciprocal of the load factor). However, the smaller the load factor the higher the indirect costs, an effect not considered above.

Increasing the out-of-core inventory makes two primary contributions to the fuel costs. Even if the fuel in this storage is allowed to build up (rather than being loaded with purchased fuel initially), more fuel must be purchased because of delay in recycle, and more of the pebbles must be purchased, even if they contain only fertile material. Indirect costs are increased by delay in recovery of product value and by increase in fuel purchase (which simple economic models may not reflect).

A fifty percent increase in the amount of material which must be fabricated at the start of life increases the direct cost by the relative factor $y = 1 + .5/N$ where N is the number of loadings over the life, usually a small factor for N large. Discounting this to consider the total direct and indirect cost, the cost ratio is approximated by the factor

$$Z = \frac{1.5i + 1 - (1+i)^{1-N}}{i + 1 - (1+i)^{1-N}}, \text{ where } i \text{ is the discount factor. For } i = .07 \text{ and}$$

$N = 15$, $Z = 1.051$, compared with the direct cost factor of $y = 1.033$. These factors would also apply to a fissile inventory increase.

Deferred recovery incurs no direct cost. The effect of deferring return one reference period through the life is given by the relative recovery worth factor of $(1+i)^{-1} = 0.935$. Expressing a one-year delay for the assumptions made above, the fuel cost contribution, relative to that without delay, is given by

$$Z = \left[\frac{2T_r + 5}{2T_r + 3} \right],$$

and for a residence time of 2 years, $Z = 1.286$, quite significant.

A rather involved formulation results when an optimum out-of-core fertile pebble residence time is sought (to reduce Pa losses). It depends on the fissile content of these pebbles, and other factors and generally decreases as core residence time decreases. The economic optimum appears to be less than two Pa^{233} half-lives (<60 days). The decay of $\text{I}^{135} \rightarrow \text{Xe}^{135}$ also makes a significant contribution when the fissile content is significant.

Temperature Coefficient

The temperature coefficients of reactivity were estimated with the AMPX codes considering only a temperature rise in Th^{232} (the Doppler). For a one pebble model $\partial k/k \partial T$ was $-7.5 \times 10^{-5} / ^\circ\text{C}$ and for the two pebble model it was -5.7×10^{-5} with one-fourth of the fertile material in the fertile pebbles, and -7.9×10^{-5} with three-fourths of the fertile material in the fertile pebble. The coefficient would be smaller if part of the material were subjected to a lower temperature increase.

Point Reactor Model Survey Results

The point reactor model is useful for parameter studies. A number of simplifying assumptions were made for these calculations, including the exposure of successive batches of fuel feed to an effective two group flux and the estimate of a neutron balance from conditions at exposure of feed material to half the core residence time (one size pebble), with feed adjusted to satisfy a critical system. Thus advantages from the peaking of the flux toward the inlet of the reactor and hold up of fertile material outside of the reactor are not considered. On the other hand, no axial or radial leakage losses were included in the base calculations

so the result tends to place an upper bound on the achievable fuel breeding (conversion) ratio. It may be of interest that certain seemingly trivial aspects make a contribution. It was found to be desirable to account for the increase in the fast to thermal flux ratio with exposure and enrichment (the increase with recycle contamination), and also the decrease in performance due to generation of U^{234} from Pa^{233} capture.

The fissile consumption rates for the primary fuel nuclides, needed to predict doubling time, are estimated as follows in Kgm/Mw-yr electrical at $\eta_e = 0.4$:

ϕ_1/ϕ_2	U^{233}	U^{235}	Average
0	1.075	1.153	1.114
2.5	1.094	1.200	1.147
5	1.113	1.247	1.180
7.5	1.126	1.298	1.212
10	1.140	1.349	1.224

The reduction in conversion ratio due to absorption in Pa^{233} at equilibrium concentration is estimated as

$$\frac{\Delta C}{C} = \frac{-2}{(1+\frac{\lambda}{g})}$$

where g is the specific absorption rate and λ the decay constant for Pa^{233} . This reduction is found to be relatively insensitive to the fast to thermal flux ratio for U^{233} fuel. Dependence of this loss on the flux level

associated with power density values for U^{233} fuel is shown below:

W_{th}/cc	$\sim W_{th}/gm\ U^{233}$	$\frac{-2}{(1+\lambda)}$ g
3	590	-0.028
6	1130	-0.052
9	1640	-0.075

Thus there is incentive to hold down the flux level when a high breeding ratio is desired. This reduction would be decreased by short core residence time, hold up of fertile balls outside of the reactor before recycle, and flux peaking toward the inlet.

The effect of neutron absorption in fission products on the conversion ratio was estimated for clean U^{233} feed, (ΔC) as shown below for a power density of $6.25\ W_{th}/cc$ as dependent on residence time:

Fission Product	Core Residence Time (yrs)				
	0.5	1.0	1.5	2.0	2.5
Xe^{135}	.0337	.0329	.0322	.0315	.0309
Nd^{143}	.0025	.0046	.0064	.0080	.0095
Pm^{147}	.0029	.0051	.0067	.0079	.0088
Pm^{148}	.0004	.0008	.0010	.0011	.0013
Pm^{148}	.0009	.0015	.0020	.0024	.0026
Sm^{149}	.0082	.0091	.0098	.0104	.0107
Others	.0228	.0429	.0639	.0829	.1010
Sum	.0714	.0969	.1220	.1442	.1648

The above numbers are absorptions relative to absorptions in fissile, some factor of about 2.2 times the fraction of total absorptions. Evidently a short core residence time would be necessary to achieve a high breeding ratio.

Values of reactivity importance for the key nuclides at a one year exposure time for U^{233} feed are shown below using flux, adjoint weighting but excluding neutron transport and scattering contributions:

Nuclide	$\frac{\partial k}{k \partial N}$	$\frac{N \partial k}{k \partial N}$	Fraction Neutron Loss
Th ²³²	-9.59+02	-.4728	.4512
Pa ²³³	-1.72+04	-.0128	.0114
U ²³³	3.21+04	.4590	.4510
U ²³⁴	-1.35+04	-.0128	.0125
U ²³⁵	1.76+04	.0016	.0022
U ²³⁶	-5.88+03	—	—
C	-1.87-01	-.0075	.0062
Xe ¹³⁵	-8.33+07	-.0181	.0143
Nd ¹⁴³	-1.31+04	-.0045	.0036
Pm ¹⁴⁷	-4.80+04	-.0037	.0036
Pm ¹⁴⁸	-1.14+06	-.0005	.0005
Pm ^{148m}	-7.40+05	-.0011	.0011
Sm ¹⁴⁹	-2.73+06	-.0059	.0047
Other fp	-2.01+03	-.0405	.0376

Such data is useful for study of perturbations. For example the relative importance of U^{233} to U^{235} , and hence an estimate of relative economic worth, is 1.82, and for U^{234} relative to U^{235} it is -0.77 (without credit for its fertile potential), and -0.33 for U^{236} .

The above table lists the nuclides treated in most of this study. Not shown are I^{135} and Pm^{149} , required for short exposures, the "other fission products" were treated as two lumped components, and the nuclides U^{238} through the Pu chain to Am^{243} were treated for U^{235} feed.

Shown in Fig. 2 is the estimated conversion (breeding) ratio as dependent on the core residence time through a thirty year history with U^{233} fuel initially and product recycle without delay. A decrease in residence time increases the breeding ratio. The effect of delay in recycle is an increase in the reference time for this data. Thus setting the delay time equal to the residence time would cause two successive batches initially with identical exposure histories, and this would propagate through the history; the thirty year history as shown becomes a sixty year history, or a thirty year history is reached at the reference of fifteen years without delay. Thus the abscissa scale time numbers should be multiplied by the ratio of the exposure plus delay time to the exposure time, different for each case, quite generally increasing the breeding ratio at the expense of more U^{233} feed.

The increase in conversion ratio early in the history is not fully understood; it may be in part a fallacy of the model. However, as Th^{232} absorptions are replaced in part by a small amount of U^{234} absorptions, the Pa^{233} losses decrease. It appears that importance values should be assigned to the fuels to disadvantage the worth of U^{235} in the calculation.

Shown below are estimates of the fuel cost for the thirty year history as dependent on core residence time. Unit values for Th^{232} , U^{233} , U^{234} , U^{235} and U^{236} of .025, 35, -5, 25 and -10 \$/gm were used, \$200/kgm fabrication, \$150/kgm processing, .5 year lead and 1.0 lag times, 0.9 plant factor, and 0.1 interest rate (indirect cost) with discounting at a factor of 0.07. Higher values of unit fabrication and processing costs of course increase the fuel cost and shift the optimum toward longer residence time. The results are shown on Page 17.

Fig. 2. Behavior with U^{233} Feed and Product Recycle

Core Residence Time (yr)	Recycle Delay (cycles)	Fissile Data		Effective Conversion Ratio	Effective Reciprocal Doubling Time (yr ⁻¹)	Fuel Cost (mil/Kw _e -hr)
		Feed Rate (gm/Kw _e -yr)	Loading (gm/Kw _e)			
.125	0	18.09	2.25	1.090	.0398	36.93
.25	0	9.51	2.39	1.067	.0283	19.18
.5	0	5.02	2.49	1.041	.0165	10.27
.75	0	3.46	2.57	1.026	.0099	7.31
1.	0	2.66	2.63	1.013	.0050	5.84
1.5	0	1.86	2.76	0.991	-.0033	4.44
2.	0	1.48	2.93	0.969	-.0106	3.80
	1/2	1.42	2.80	0.975	-.0085	3.77
	1	1.37	2.67	0.981	-.0067	3.73
3.	0	1.10	3.10	0.930	-.0216	3.28
4.	0	0.93	3.44	0.891	-.0297	3.18

These cost estimates do not reflect any difference from the delay in recycle due to increase in the inventory because the economic model fixes the lag time and charges unit costs of feed nuclides and credits discharge. Only the effect of differences in unit values for the nuclides, and the amounts, enter the calculation, so these results indicate that the reference unit values of the nuclides are not accurately determined for this situation involving two year exposure; U²³³ probably has a higher value relative to U²³⁵, but the other unit values also play a role.

The above results were recast to approximate the performance with a recycle delay of one year. The time variation through the history is

shown in Figure 3 and data for these histories is shown below:

Core Residence Time (yr)	External U^{233} Feed (gm/Kw _e -yr)	Effective Conversion Ratio	Effective Reciprocal Doubling Time (yr ⁻¹)	Fuel Cost (mil/Kw _r -hr)
.125	0.074	1.112	0.0494	35.87
.25	0.079	1.087	0.0368	19.18
.5	0.086	1.058	0.0233	10.05
.75	0.090	1.040	0.0155	7.16
1.	0.093	1.025	0.0096	5.74
1.5	0.100	0.999	-0.0004	4.37
2.	0.102	0.977	-0.0078	3.75
3.	0.120	0.936	-0.0190	3.26
4.	0.130	0.903	-0.0264	3.15

Note the increase in the amount of U^{233} which must be supplied from an external source as the residence time is increased and as the lag time for recycle is increased.

The effect on performance of neutron losses not accounted for in a simple model may be taken into account. Shown in Fig. 4 is the dependence of the breeding ratio on the residence time and the additional neutron loss fraction, without provision for delay in recycle of the produced uranium calculated for thirty year reactor histories. The range chosen extends to negative values of the additional losses in the event that dimensional calculations show the losses to be overestimated (possibly lower fission product losses due to the skewed flux distribution toward the inlet). Shown in Fig. 5 is the dependence of reciprocal doubling time on these other losses.

Fig. 3. Behavior with U^{233} Feed and Delayed Product Recycle

Fig. 4. Effective Breeding Ratio

Fig. 5. Effective Reciprocal Doubling Time

The dependence of the effective characteristics on power level were estimated. Results for thirty year histories with U^{233} feed are shown below for the idealized conditions:

Residence Time (full power yrs)	Power Level (W_{th}/cc)	Exposure ($Mw_{th}-D/Mgm\ Th$)	Total Fissile Feed (Kgm/Mw_e-yr)	Breeding Ratio	Doubling Time (yr)	Cost (mil/ Kw_e-hr)
<u>No Delay in Recycle</u>						
1	2.5	4,740	5.28	1.100	53.5	13.14
1	3.75	7,110	3.68	1.068	57.1	9.03
1	5.0	9,480	3.10	1.039	79.5	7.01
1	7.5	14,200	2.38	.989	-	5.07
1	10.0	19,000	2.04	.944	-	4.19
2	2.5	9,480	2.71	1.080	69.3	7.75
2	3.75	14,200	2.01	1.039	104.0	5.47
2	5.0	19,000	1.67	1.003	1250.0	4.39
2	7.5	28,400	1.36	.937	-	3.43
2	10.0	37,900	1.25	.877	-	3.06
<u>One Year Recycle Delay</u>						
1	2.5	4,740	5.05	1.107	50.0	12.99
1	3.75	7,110	3.47	1.076	51.1	8.91
1	5.0	9,480	2.94	1.050	62.0	6.91
1	7.5	14,200	2.07	1.002	1700.0	4.98
1	10.0	19,000	1.91	.962	-	4.09
2	2.5	9,480	2.65	1.083	66.8	7.69
2	3.75	19,200	1.96	1.044	92.7	5.41
2	5.0	19,000	1.62	1.009	375.0	4.34
2	7.5	28,400	1.31	.948	-	3.38
2	10.0	37,900	1.19	.895	-	3.01

Production of U^{233} Fuel

Shown in Fig. 6 is the dependence of conversion ratio on core residence time through thirty year reactor histories estimated for a pebble bed reactor with a C/Th of 80, considering U^{235} feed and makeup with product recycle without delay. Performance with a feed mixture of U^{233} and U^{235} can be estimated using this data and results shown earlier (there would be preferential absorption in U^{233} , it having a larger cross section). Delay in recycle adjusts the time scale which reduces the conversion ratio. The need for makeup U^{235} feed reduces the conversion ratio, since inadequate fuel is available from recycle to effect criticality, a requirement which has been accounted for. These results represent an upper limit on performance since no neutron losses were included such as to account for leakage, and recycle was not delayed.

Performance of an efficient converter plant to produce U^{233} breeder fuel was estimated. The following show the dependence on core residence time of the performance and estimated product composition and fuel cost assuming particle separation is admitted and ignoring U^{235} recovery from the primary particles:

Residence Time at Full Power (yr)	Conversion Ratio	Fissile Balance (Kgm feed per Kgm Product)	Real Time to Reproduce Initial Fuel (yr)	Product Composition (atom fraction)			Fuel Cost (mils/ Kw_e -hr)
				U^{233}	U^{234}	U^{235}	
0.5	.779	9.41	5.51	.974	.025	.001	11.90
1.0	.769	5.35	6.50	.960	.038	.002	6.98
2.0	.753	3.43	8.73	.935	.058	.007	4.62
4.0	.712	2.71	14.26	.887	.093	.020	3.64

It is assumed here that clean separation of the product (U^{233} plus exposure products) would be possible, as by particle identification.

Fig. 6. Conversion Ratio with Product Recycle

Fissile mass balances and ore consumption (excludes inventory) follow:

Residence Time (yr)	Primary Fuel (gm/Kw _e -yr)			Fissile Product (gm/Kw _e -yr)	Ore Requirement (kgm/gm Product)		
	U ²³⁵ Feed	Discharge			No Credit	U ²³⁵ -U ²³⁶ Credit	Full U ²³⁵ Credit
		U ²³⁵	U ²³⁶				
0.5	8.59	7.40	0.27	.913	2.00	.34	.27
1.0	4.40	3.29	0.25	.823	1.13	.35	.28
2.	2.31	1.33	0.22	.675	.73	.38	.30
4.	1.35	0.49	0.17	.499	.58	.44	.37

As indicated above, it is possible to generate U²³³ with U²³⁵ feed. However, considerable time elapse is involved, and reducing this by decreasing the residence time, which reduces U²³³ consumption, would incur a severe cost penalty.

More realistic data is obtained by allowing for losses not accounted for in a point model. Results with inclusion of a 0.015 fraction loss for U²³⁵ feed without recycle which are representative for an advanced pebble bed converter with low C/Th are shown below:

Residence Time at Full Power (yr)	Conversion Ratio	U ²³⁵ Feed (gm/Kw _e -yr)	Real Time	Atom	Ore, kgm/gm	Fuel
			To Reproduce Fissile Loading (yr)	Fraction U ²³³ in Product	Product U ²³⁵ -U ²³⁶ credit	Cost (mils/ Kw _e -hr)
1.0	.737	4.64	6.71	.965	.37	7.410
2.0	.719	2.36	8.55	.944	.40	4.919
3.0	.699	1.62	10.89	.926	.44	4.281
4.0	.676	1.28	13.85	.910	.48	4.100

The value of U²³³ so generated is not simple to establish. Reducing the residence time from 4 years to 1 year to increase the U²³³ production has the apparent effect of adding \$37/gm to the product cost, relative to the assumed value of \$35/gm, apparently prohibitive.

Performance of a plant using its own product is shown in Fig. 7 with recycle delayed one year, 0.015 supplemental loss fraction, and low C/Th.

Fig. 7. Performance with U²³⁵ Feed, Product Recycle

Partial refueling of a reactor core appears economically feasible for reactor concepts in which the reactivity excess over the exposure period between fuelings must be compensated for with control rods. Shown below are summaries of the histories for such a reactor concept with a high carbon to thorium ratio and one cycle delay in recycle using U^{235} feed:

Residence Time at Full Power (yr)	Fraction Reactor Refueled	Fissile Data		Effective Conversion Ratio	Ore Kg _m /Kw _e -yr Requirement (No Credit)	Fuel Cost (mil/ Kw _e -hr)
		Feed Rate (gm/Kw _e -yr)	Loading (Kg _m /Mw _e)			
3.	1/3	.986	1.866	.626	.136	2.582
2.	1/4	1.021	1.528	.762	.109	2.428
3.	1/4	.933	1.826	.664	.124	2.423
4.	1/4	.934	2.102	.579	.143	2.575
3.	1/6	.881	1.755	.702	.113	2.267
2.	1/8	.963	1.475	.799	.095	2.268
3.	1/8	.855	1.728	.721	.107	2.191
4.	1/8	.840	2.005	.652	.121	2.274
3.	1/12	.829	1.673	.739	.101	2.115

A U^{233} product may be produced by a reactor partially refueled. Results are shown below for regeneration of the fissile loading in such a reactor (without recycle of the U^{233} product) at quasi-equilibrium:

Residence Time at Full Power (yr)	Fraction Reactor Refueled	Conversion Ratio	U^{235} Feed (gm/ Kw _e -yr)	Real Time To Reproduce Initial Fuel (yr)	Atom Fraction U^{233} in Product	Ore Kg _m /gm Product (No Credit)	Fuel Cost (mils/ Kw _e -hr)
3.	1/3	.679	.976	9.75	.87	.612	2.232
2.	1/4	.740	1.049	5.30	.89	.499	2.031
3.	1/4	.688	.918	9.15	.87	.574	2.041
4.	1/4	.637	.889	13.01	.85	.680	2.118
3.	1/6	.648	.870	8.65	.86	.543	1.861
2.	1/8	.750	.994	4.99	.88	.470	1.840
3.	1/8	.704	.846	8.41	.86	.528	1.775
4.	1/8	.658	.796	12.98	.84	.611	1.795
3.	1/12	.709	.820	8.15	.85	.511	1.692

More sophisticated analysis is possible. The delay in fuel production in the early life amounts to about half the residence time, and account for the early history increases the fuel cost of energy produced and the ore commitment. Ore is also committed for plant inventory.

Effect of Out-of-Core Hold Up on Fission Product Poison

The ORIGEN code⁷ was used to study the effect of out-of-core hold up on fission product poison. Typically cases were run with and without hold up. In these calculations, 100 actinides were treated and 400 fission product nuclides. Representative values of the fraction of total neutron absorptions are shown below for the situation considered (U^{233} , Th^{232} initially, fixed flux), 36 days in-core, no out-of-core and 360 days out-of-core:

Exposure (Days)	No Out-of- Core	360 Day Out-of-Core		
		Start Exposure	Mid Exposure	End Exposure
36	.02015	0	.01846	.02034
72	.02327	.00674	.02231	.02368
108	.02633	.00983	.02519	.02652
144	.02928	.01251	.02778	.02910
180	.03209	.01497	.03018	.03149
216	.03476	.01729	.03245	.03375
252	.03731	.01950	.03461	.03590
288	.03978	.02162	.03669	.03797
324	.09213	.02367	.03871	.03997
360	.04441	.02568	.04067	.04193

A simple linear integration of this data indicates an average for a one year exposure of 0.0311 fraction absorptions in fission products compared with 0.0272 for the year-out-of-core hold up case, a reduction of 12 percent, worth about 0.009 in the effective breeding ratio. (The composition considered was deliberately selected as low enrichment, not core average, 2 percent U^{233} in Th^{232} .)

A number of problems were solved to test the reliability of the results via perturbation. A summary of results is shown below to assess the effect of hold up on the fission product poison level:

Exposure (Days)		Out-of-Core Hold Up (Days)	Fraction Neutron Absorptions In Fission Products				Implied Gain in Breeding Ratio by Fission Product Hold Up
Interval	Total		End of History		Integrated Average		
			No Hold up	With Hold Up	No Hold Up	With Hold Up	
36	360	360	.0444	.0419	.0311	.0272	0.009
72	720	360	.0640	.0608	.0430	.0396	0.008
72	720	180	.0640	.0620	.0430	.0404	0.006
72	720	72	.0640	.0630	.0430	.0411	0.004

A primary contribution to the reduction in fission product losses from external hold up is reduced Xe^{135} absorption. The above exposures result in approximately 270 and 540 $\text{Kw}_{\text{th}}\text{-D/gm U}^{233}$ energy generation ($2.72 \text{ W}_{\text{th}}/\text{cc}$) which have associated fission product poisoning fractions at end of exposure of 0.0444 and 0.0640 respectively.

This data allows a rough estimate to be made in the effect on breeding ratio of neutron loss to the fission products. The higher exposure above is equivalent to about 313 days at $6.25 \text{ W}_{\text{th}}/\text{cc}$. Selected results are displayed below for $\frac{\Delta C}{C}$ and the rate of change with change in exposure time:

Model	$\frac{\Delta C}{C}$	$\frac{\Delta C}{\text{CAT}} (\text{yr}^{-1})$
The ORIGEN results	0.096	0.062
Simple point model results (shown earlier)	0.093	0.051
One-dimensional results (shown later)	0.071	0.055

The fission product nuclides which play a dominating role in the absorption of neutrons are listed below in order of decreasing importance after a long exposure period:

NUCLIDE	WITH HOLD UP		WITHOUT HOLD UP	
	Order of Importance	Fraction Neutron Absorptions	Order of Importance	Fraction Neutron Absorptions
Xe ¹³⁵	1	.0146	1	.0145
Nd ¹⁴³	2	.00550	4	.00537
Cs ¹³³	3	.00524	5	.00513
Xe ¹³¹	4	.00516	6	.00503
Sm ¹⁴⁹	5	.00467	2	.00558
Pm ¹⁴⁷	6	.00364	3	.00537
Rh ¹⁰³	7	.00319	7	.00310
Sm ¹⁵²	8	.00193	9	.00192
Sm ¹⁵¹	9	.00186	10	.00192
Nd ¹⁴⁵	10	.00155	11	.00151
Mo ⁹⁵	11	.00134	13	.00116
Pm ^{148m}	12	.00129	8	.00197
Tc ⁹⁹	13	.00119	12	.00116
Sm ¹⁴⁷	14	.00113	25	.000365
Eu ¹⁵³	15	.00112	14	.00110
Kr ⁸³	16	.00102	15	.000997
Sm ¹⁵⁰	17	.000714	16	.000824
Ru ¹⁰¹	18	.000486	18	.000474
Pm ¹⁴⁶	19	.000452	17	.000675
Eu ¹⁵⁴	20	.000451	20	.000465
Rh ¹⁰⁵	21	.000444	21	.000442
Cs ¹³⁵	22	.000442	22	.000428
Pr ¹⁴¹	23	.000439	23	.000421
En ¹⁵⁵	24	.000434	19	.000470
Zr ⁹³	25	.000422	24	.000412
I ¹²⁹	26	.000354	26	.000345
La ¹³⁹	27	.000298	27	.000291
Nd ¹⁴⁶	28	.000228	29	.000223
I ¹²⁷	29	.000214	31	.000203
Mo ⁹⁷	30	.000210	30	.000205
Cs ¹³⁴	31	.000187	28	.000287
Sum		.0602		.0623

The small difference in the results and the dominant role played by essentially the same nuclides may be noted. Involved chain routes (couplings) complicate the selection of a simplified representation which will be reliable.

The more important actinides are shown below with absorption fractions ordered by importance in neutron absorption (U^{233} fuel):

<u>Nuclides</u>	<u>Fraction Absorptions</u>
Th ²³²	.468
U ²³³	.428
U ²³⁴	.0232
Pa ²³³	.0108
U ²³⁵	.00583
Pa ²³¹	.000164
U ²³⁶	7.93-5
U ²³²	3.65-5
Th ²³³	1.99-5
Np ²³⁷	1.16-5
Th ²³⁰	6.99-6
Pu ²³⁸	1.17-6
Pu ²³⁹	3.28-7
Th ²²⁹	8.30-8
Np ²³⁸	5.98-8
Pu ²⁴⁰	5.27-8
Th ²²⁸	4.33-8
Pu ²⁴¹	1.25-8
U ²³⁷	7.19-9
Pu ²⁴²	2.73-10
Am ²⁴¹	5.02-11
Am ²⁴³	2.51-11
Ra ²²⁴	2.22-11
Pu ²³⁶	3.37-12
Cm ²⁴⁴	1.19-12
Cm ²⁴⁵	8.93-13
Am ²⁴²	1.52-13
Cm ²⁴²	1.27-13
Cm ²⁴³	7.75-14
Am ²⁴⁴	1.09-14
Np ²³⁶	1.64-15
Pu ²⁴³	1.01-15
Cm ²⁴⁶	5.47-16

One Dimensional Survey Results

The reactor core is described quite well with a simple one-dimensional traverse (unless residence time depends on radial location or a radial blanket makes a large contribution). For the reference calculations, 20 axial core zones were treated, plus 3 more with the blanket represented,

in the continuous feed, once-through exposure model. A direct iterative approach was used to adjust the amount of fuel feed to the reactor, and successive neutronics criticality search, exposure calculations effected near critical conditions.

The calculations were done for 6 cm diameter primary (or single) pellets and 1 cm diameter fertile pellets. There is considerable uncertainty regarding the calculation of resonance shielding, so the results are appropriate to some pellet size and some distribution of the carbon and thorium not well identified. Initial calculations were done with pure U^{233} feed in the primary pellets, no fissile material in the fertile pellets, and also pure U^{233} in the fertile pellets to examine performance under this ideal feed condition.

Results are shown in Table 1 for selected cases involving a 2 year residence time for single pebbles and a 1 year residence time for the primary pebbles, 0.1 year residence time for the fertile pebbles with a low ratio of Th in the fertile pebbles to that in the primary pebbles of 1:3. Average power density for the reactor was $2.5 W_e/cc$ (6.25 thermal) and mass balances are for a 1,500 Mw_e plant. The core had a volume of 6×10^6 cc and was 460 cm high. The C/Th ratio was about 80 (slightly higher for the case with a blanket). Note from the neutron balance that the "other losses", which include a radial buckling loss, were small, which assumes relatively effective neutron reflection at the top (entrance), an aspect subject to further analysis.

The fissile mass balance and other data assume full U^{233} recovery (from Pa^{233}) and no processing loss.

Table 1. One Dimensional Results (No Recycle)

Case	B1a1a	B1a2a	B1b2a	B1b2b
Axial Blanket	No	No	No	Yes
Core Residence Time (yr. full power, excludes residence in blanket)				
Primary pebbles	2.0	1.0	1.0	1.0
Fertile pebbles	—	0.1	0.1	0.1
Neutron Energy Groups	2	2	8	8
Conversion Ratio	0.937	1.017	1.016	1.023
Reciprocal Doubling Time (yr ⁻¹)	-0.0354	.0057	.0050	.0078
Fissile Mass Balance (Kgm/day at full power)				
U ²³³ Feed	5.477	18.90	19.81	18.73
Discharge	5.183	18.98	19.89	18.84
Feed Enrichment U ²³³ /Th+U ²³³)				
Primary Balls	.03361	.04995	.05337	.04931
Fertile Balls	—	.00794	.00794	.00794
Fissile Inventory (Kgm)	3,876	4,569	4,900	4,543
Peak Radial Average Power Density				
(W _{th} /cc)	12.42	11.96	11.70	12.06
Neutron Loss Accounting				
Th ²³²	.42794	.46988	.47101	.47235
Pa ²³³	.01290	.00880	.00842	.00903
U ²³³	.45186	.45572	.45610	.45469
U ²³⁴	.01222	.00516	.00500	.00505
U ²³⁵	.00252	.00047	.00042	.00058
U ²³⁶	.00003	.0	.0	.0
Xe ¹³⁵	.01453	.01297	.01217	.01292
Nd ¹⁴³	.00326	.00133	.00121	.00132
Pm ¹⁴⁷	.00323	.00176	.00171	.00173
Pm ¹⁴⁸	.00052	.00027	.00026	.00026
Pm ^{148m}	.00116	.00059	.00056	.00058
Sm ¹⁴⁹	.00467	.00366	.00360	.00364
Other fp	.03574	.01550	.01525	.01542
C	.00563	.00462	.00426	.00491
Other losses	<u>.02379</u>	<u>.01927</u>	<u>.02003</u>	<u>.01752</u>
Total	1.0	1.0	1.0	1.0

These results indicate that an initial conversion ratio of 0.94 is achievable with one pebble and U^{233} feed with a 2 year residence time. With two pebbles, a breeding ratio of 1.02 is achievable initially with residence times of 1 year for the primary pebbles and 0.1 year for the fertile pebbles. The eight group results indicate a slightly lower performance than the two group results (5 percent higher fissile feed). Undoubtedly a better fine group cross section collapse than the point model used would reduce this discrepancy, since it is caused directly by the difference in effective spectrum. Still, the two group results appear adequate for survey analysis, especially considering the close approximation of the conversion ratio and no major discrepancy in the secondary neutron losses which include core leakage. Comparison with the point model results show a gain from the spatial flux distribution in the losses to fission products of fraction 0.10, lower absorptions in C and Pa^{233} and higher U isotopes, but additional 0.02 fraction leakage loss, which is deemed to be low without blankets.

Case B1b2b shows the performance with a 31 cm. axial blanket which increased the in-core thorium loading from 116 to 120 Mgms. A gain of 0.005 in the breeding ratio is realized, not all that is possible with a higher thorium loading or thicker blanket.

Additional calculations were made which indicated that only a small decrease in the conversion ratio accompanies increases in the fertile pebble residence time or fissile contents, but the conversion ratio increases substantially as the primary pebble residence time is reduced.

Results from one-dimensional, eight group calculations considering U^{233} feed, a high ratio of Th in the fertile pebbles to that in the primary pebbles of 3:1, with an axial blanket at the entrance of fertile feed pebbles, are shown in Table 2. Although a C to Th ratio of 80 was

Table 2. Results for the Two-Ball Pebble Bed Reactor
(One-Dimensional, Eight Group, Fresh U²³³ Feed, Blanket)

Power Level (W _{th} /cc)	Residence Time ^a (yrs)		U ²³³ Feed Rates (Kg/Day)		In-Reactor Fissile Inventory (Kg)	Peak Average Radial Power Density (W _{th} /cc)	Breeding Ratio	In-Reactor Fissile Doubling Time ^b (yrs at 0.9 load factor)
	Primary	Fertile	Primary	Fertile				
6.25	1.5	0.1	10.85	0	4,740	17.41	1.015	210
6.25	1.0	0.1	13.91	0	4,367	14.77	1.044	72
6.25	0.5	0.1	23.64	0	4,035	12.08	1.072	37
6.25	1.0	0.2	13.94	0	4,466	14.45	1.037	78
6.25	1.0	0.1	12.12	12.72	4,372	14.04	1.046	63
6.25	1.0	0.2	9.59	15.91	4,483	12.77	1.043	68
6.25	1.0	0.1	9.41	31.81	4,373	12.82	1.052	54
6.25	2.0	.0488	7.80	32.48	4,927	18.17	1.003	1080
6.25	1.0	.0488	11.56	32.58	4,305	13.81	1.052	54
6.25	0.5	.0488	19.62	32.58	4,006	11.56	1.079	34
6.25	0.25	.0488	35.76	32.58	3,834	10.49	1.096	24
6.25	2.0	.0488	6.25	65.16	4,820	16.47	1.013	243
6.25	1.5	.0488	7.18	65.16	4,547	14.44	1.035	86
6.25	1.0	.0488	9.23	65.16	4,291	12.65	1.057	49
6.25	0.5	.0488	15.72	65.16	4,039	10.96	1.081	33
6.25	1.0	.0488	4.50	130.3	4,249	10.26	1.073	39
6.25 ^c	1.0	.0488	19.95	32.91	7,320	15.94	1.093	51
12.5	1.0	.0488	16.90	32.58	5,380	37.13	0.974	—
12.5	0.5	.0488	24.93	32.58	4,629	29.17	1.027	56
18.75	1.0	.0488	22.67	32.58	6,480	67.04	0.918	—

^aIn the core proper, excludes blanket time (full power).

^bSimple fissile inventory doubling, out-of-core fissile inventory including that of the fertile balls for hold up not included.

^cAll nuclide densities doubled just to see what happens.

intended and the cross sections were processed for this ratio, these cases were run with a ratio of 88.

Problems were solved to examine the effect of recycle in the early life on performance and the results are shown in Table 3. Material from discharged primary pebbles was used as primary pebble feed, supplemented with U^{233} . Fertile recycle material was considered to be held up one year and primary material chemically processed. Note that a one-year delay in recycle would cause the tenth refueling to occur at about 20 years, or 22 years real time at a load factor of 0.9.

Note that in the results considering recycle, the fissile inventory increases as the higher uranium isotopes, especially U^{234} , build up. It appears that the fertile loading should be reduced to hold down the fissile inventory, but much reduction significantly reduces the conversion ratio and increases the doubling time. (Perhaps the guideline should be minimizing the doubling time. Here the reported results do not take into account the lower worth of U^{235} relative to U^{233} in the reactor.) These results also suggest an effect of uncertainty in the calculations.

Certain calculations were made to test reliability of the results. Shown in Table 4 are results obtained with changes in the axial blanket thickness and a reduced number of depletion zones in the core for residence times in the core proper of 1 year primary and 0.0488 year fertile pebbles.

Unfortunately the original mesh and zone arrangement (20 zones in the core) did not lend itself to consistent reduction in the number of zones in the core. However, even halving the number of depletion zones in the core led to only a small difference in the results. A small increase in the conversion ratio resulted from increase in the entrance blanket thickness and from adding an exit blanket.

Table 3. Results for Recycle

Recycle Pass	Recycle Fraction Primary Fertile		Fissile Rate Balance (Kgm/D)				In-Reactor Fissile Inventory (Kgm)	Breeding Ratio	In-Reactor Fissile Doubling Time (yrs. at 0.9 load factor)
			Feed		Discharge				
			Primary	Fertile	Primary	Fertile			
0	0	0	11.58	32.58	9.10	35.30	4,305	1.0514	53.57
1	1.0	0	12.16	32.58	9.67	35.28	4,524	1.0516	55.87
1	1.0	0	12.56	26.07	9.99	28.88	4,510	1.0525	55.82
1	1.0	1.0	11.95	35.30	9.55	37.95	4,522	1.0528	54.72
1	1.0	0.9	12.19	31.77	9.73	34.48	4,521	1.0524	55.48
1	1.0	0.8	12.44	28.24	9.90	31.02	4,520	1.0520	56.24
2	1.0	0.9	12.78	31.04	10.32	33.74	4,724	1.0519	58.11
3	1.0	0.9	13.29	30.38	10.83	33.08	4,899	1.0523	61.72
4	1.0	0.9	13.87	29.78	11.40	32.47	5,102	1.0494	65.64
5	1.0	0.9	14.32	29.23	11.85	31.92	5,257	1.0479	70.00
6	1.0	0.9	14.74	28.74	12.27	31.42	5,400	1.0462	74.66
7	1.0	0.9	15.12	28.29	12.64	30.97	5,531	1.0444	79.62
8	1.0	0.9	15.47	27.88	12.99	30.56	5,651	1.0426	84.87
9	1.0	0.9	15.80	27.51	13.31	30.18	5,763	1.0406	90.35
10	1.0	0.9	16.09	27.17	13.60	29.84	5,862	1.0391	98.52
10*	1.0	0.9	15.03	27.17	12.62	29.74	5,490	1.0339	104.2
10**	1.0	0.9	14.01	27.17	11.68	29.63	5,135	1.0282	112.5
10***	1.0	0.9	12.04	27.17	9.89	29.39	4,454	1.0180	156.3

*Th²³² feed reduced 5% in fertile pebbles to lower fissile inventory.

**Th²³² feed reduced 10% in fertile pebbles.

***Th²³² feed reduced 20% in fertile pebbles.

Table 4. Further Results From the One-Dimensional Model

Depleting Zones In Core	Axial Blanket Thickness		U ²³³ Feed Rates (Kgm/Day)		In-Reactor Fissile Inventory (Kgm)	Peak Average Radial Power Density (W _{th} /cc)	Breeding Ratio		In-Reactor Fissile Doubling Time (yrs at 0.9 load factor)
	Entrance	Exit	Primary	Fertile			Reaction Rate	Mass Balance	
10	41.3	0	13.86	0	4,296	15.28	1.0550	1.0566	51.62
10	41.3	0	11.54	32.58	4,318	14.17	1.0594	1.0601	47.83
15	27.5	0	11.56	32.57	4,290	13.94	1.0506	1.0517	55.63
20	31.0	0	11.58	32.58	4,305	13.81	1.0527	1.0514	53.57
20	41.3	0	13.86	0	4,302	14.80	1.0538	1.0555	52.53
20	41.3	0	11.55	32.58	4,325	13.80	1.0587	1.0580	48.40
20	41.3	41.3	13.84	0	4,309	14.72	1.0556	1.0575	50.58
20	41.3	41.3	12.67	16.28	4,373	14.21	1.0578	1.0598	49.91
20	41.3	41.3	11.52	32.58	4,443	13.69	1.0608	1.0619	47.94

The convergence properties of the calculational procedure in use are of interest. The initial nuclide concentrations were set uniform as a starting point without any attempt to supply a gradient, but the fuel concentration in an artificial zone for the primary particle feed was made somewhat larger than the value within the core as a technique found to be desirable to accelerate the convergence rate. The iterative behavior in effecting a converged solution for a problem is shown below:

End of Iteration Sweep	Neutronics Problem Outer Iterations	U ²³³ Feed Rate (Kgm/day)	In-Reactor Fissile Inventory (Kgm)	Peak Average Radial Power Density (w _{th} /cc)	Breeding Ratio	
					Reaction Rates	Mass Balance
1	42	44.098	4,213	9.72	1.1700	0.8980
2	40	42.963	3,892	16.63	1.0762	1.1167
3	51	44.503	4,423	13.03	1.0502	1.0262
4	44	44.062	4,275	13.78	1.0540	1.0573
5	27	44.141	4,298	13.91	1.0530	1.0521
6	29	44.174	4,311	13.77	1.0525	1.0503
7	19	44.153	4,304	13.81	1.0528	1.0516
8	20	44.158	4,305	13.82	1.0527	1.0514
9	3	44,158	4,305	13.81	1.0527	1.0514
10	2	44,158	4,305	13.81	1.0527	1.0514

Note that the iterative process is quite stable and the procedure effective in this situation.

Under certain conditions an oscillatory behavior was found to occur, and an acceptable solution to the critical reactor problem was not obtained. The oscillation was found to be driven by a shift between two space, energy flux distributions associated with the successive estimates of the zone nuclide concentrations, not by the criticality search adjustments. Even when a procedure was used to simply determine the multiplication factor in

successive solutions of the neutronics code, the oscillation between two solutions persisted. The exposure module was modified to produce a weighting between start and end exposure nuclide concentrations. In the continuous fueling model, this option weights between successive estimates of the average nuclide concentrations. A simple average between the two was found to stabilize the process, although the rate of convergence was slow. Considering a weighting factor W , the weighted concentrations are given by

$$C(W) = C(E) + W[C(E) - C(S)],$$

where E refers to end and S to start of exposure (successive estimates of the zone average). Results are shown below for a case:

Weighting Factor	0.5	0.3	0.25	0.2	0.1	0
Iteration						
1 search eigenvalue*	-2.18	-2.18	-2.18	-2.18	-2.18	-2.18
2 "	0.216	0.290	0.308	0.326	0.360	0.393
3 "	0.167	0.218	0.228	0.235	0.239	0.222
4 "	0.106	0.060	0.031	-.008	-.120	-.250
5 "	0.073	0.084	0.102	0.129	0.193	0.233
6 "	0.051	0.023	0.004	-.029	-.166	-.305
7 "	0.036	0.032	0.045	0.072	0.182	0.270
8 k_{eff}	0.995	0.9983	1.0003	1.0043	--	1.0529
9 "	0.9895	0.9959	0.9972	0.9986	0.9967	0.9762
End Conversion Ratio Estimate	1.0163	1.0040	1.0018	1.0000	1.0109	1.0392
Cumulated Search Parameter	-1.53	-1.37	-1.46	-1.45	-1.492	-1.617
Fissile Inventory (Kgm/Kw _e)	2.275	2.311	2.317	2.320	2.285	2.197
Conversion Ratio adjusted ($k_{\text{eff}} = 1$)**	1.0060	0.9999	0.9990	0.9986	1.0076	1.0148

* Relative fissile nuclide concentration.

** Estimate for a critical system, converged result 0.9976.

Here the oscillation without weighting is severe. The optimum value of the weighting factor is about 0.2. A larger value decreases the rate of convergence. The worse the oscillation the larger the weighting factor expected to optimize acceleration. Some care was required to identify when incompletely converged results were close to the real solution or not acceptable.

The dependence of the results on the number of meshpoints was examined by doing a calculation with the number of meshpoints doubled. Results are compared below:

Meshpoints	U ²³³ Feed		In-Reactor Fissile Inventory (Kgm)	Peak Radial Average Power Density (W _{th} /cc)	Breeding Ratio		In-Reactor Fissile Doubling Time (yrs at 0.9 loading factor)
	Primary	Fertile			Reaction Rates	Mass Balance	
70	11.56	32.58	4,299	13.82	1.053	1.052	53.5
140	11.56	32.58	4,299	13.84	1.053	1.052	53.4
Extrapolated	11.56	32.58	4,299	13.85	1.053	1.052	53.4

Evidently a sufficient number of meshpoints (70) has been used in these calculations to produce quite adequate accuracy.

Single Pebble Size Results

Calculations were done to examine the performance with a single pellet size but two feed streams. The primary pebbles containing fuel from an external source were introduced at the entrance to the core. Fertile pebbles were introduced at the top of an axial blanket. Residence time in the core was made the same for the two streams, and the residence time of the fertile pebbles in the blanket was made inversely proportional to the volume fraction occupied. Results of these one-dimensional calculations treating four neutron energy groups are shown in Table 5. By "cycle" is meant the pass

Table 5. Results from One-Dimensional Treatment of a Single Pellet Size

Cycle and recycle Fraction Primary/ Fertile	Out of Core Holdup (Days)	C/Th	Fertile Pellet Stream Fraction	Power Density ($\text{W}_{\text{th}}/\text{cc}$)	Residence Time (yr at full power)	Fissile Rates (Kg/D)					In-Reactor Fissile Inventory (Kg/Mw_e)	Peak Average Radial Power Density ($\text{W}_{\text{th}}/\text{cc}$)	Breeding Ratio	In-Reactor Fissile Doubling Time (yr at 0.9 load factor)
						Fresh Primary U_{233} Feed	Recycle Primary Feed	Recycle Fertile Feed	Discharge Primary	Discharge Fertile				
1/1	-	80	.298*	6.25	1.0	9.261	-	-	8.401	0.885	2.261	10.3	1.0054	419.
1/1	-	80	.298	6.25	0.5	17.353	-	-	16.492	1.022	2.125	10.2	1.0346	61.4
1/1	-	80	.298	6.25	0.25	32.898	-	-	32.078	1.107	2.013	10.2	1.0618	32.0
1/1	-	80	.298	5.0	1.0	8.896	-	-	8.253	0.735	2.729	7.98	1.0284	84.4
1/1	-	110	.298	5.0	2.0	3.820	-	-	3.249	0.5621	2.331	9.81	0.9976	-
1/1***	-	110	.298	5.0	1.0	6.949	-	-	6.382	0.7194	2.148	8.34	1.0414	53.7
1/1	-	110	.298	5.0	0.5	13.290	-	-	12.705	0.8328	2.047	8.36	1.0673	30.6
1/1	-	110	.298	5.0	0.25	25.602	-	-	25.028	0.8991	1.963	8.60	1.0891	22.0
1/1	-	110	.5**	5.0	2.0	3.826	-	-	2.620	1.200	2.336	9.81	0.998	-
1/1	-	110	.5	5.0	1.0	6.956	-	-	5.572	1.537	2.150	8.35	1.0417	51.0
1/1	-	110	.5	5.0	0.5	13.317	-	-	11.786	1.776	2.051	8.23	1.0667	30.6
1/1	-	110	.5	5.0	0.25	25.596	-	-	24.006	1.920	1.963	8.59	1.0903	21.7
2,1./2,1.	30	110	.5	5.0	0.25	.891	24.006	1.920	23.505	3.540	2.065	8.19	1.0622	33.2
2,1./2,1.	60	110	.5	5.0	0.25	.753	24.006	1.920	23.378	3.538	2.055	8.37	1.0647	31.7
2,1./2,1.	120	110	.5	5.0	0.25	.650	24.006	1.920	23.283	3.537	2.048	8.49	1.0666	30.7
3,1./3,1.	60	110	.5	5.0	0.25	.696	23.378	3.538	22.858	4.915	2.135	8.32	1.0440	48.6
4,1./4,1.	60	110	.5	5.0	0.25	.746	22.858	4.915	22.514	6.096	2.211	8.28	1.0249	89.3
5 P /5,.85	60	110	.5	5.0	0.25	****	21.240	5.181	20.350	6.358	2.062	8.40	1.0786	26.2
6,1./6,.85	60	110	.5	5.0	0.25	1.678	20.350	5.404	21.107	6.538	2.137	8.28	1.0580	36.9
7,1./7,.85	60	110	.5	5.0	0.25	1.671	21.107	5.587	21.825	6.659	2.204	8.26	1.0406	54.6
8,1./8,.85	60	110	.5	5.0	0.25	1.721	21.825	5.660	22.558	6.739	2.268	8.24	1.0247	92.7
9 P /9,.85	60	110	.5	5.0	0.25	****	21.611	5.728	20.778	6.820	2.135	8.34	1.0707	30.2
Inferred Pseudo Equilibrium with Processed Primary					0.25	-	22.38	6.86	21.67	7.77	2.29	8.21	1.0543	39.
9 P /5,.85	60	100	.5	5.0	0.25	-	21.851	5.181	20.971	6.349	2.109	8.36	1.0785	26.8
9 P /1	60	100	.5	5.0	0.25	-	27.039	-	25.514	1.860	2.073	8.46	1.0915	28.5
8,1./1	60	100	.5	5.0	0.25	-	28.803	-	27.158	1.798	2.198	8.36	1.0426	64.9

* Axial Blanket 41.3 cm thick for this fraction.

** Axial Blanket 45. cm thick for this fraction.

*** This case was also run with the cross sections for a C/Th of 80 giving a breeding ratio of 1.0341.

**** Reprocessed primary pebble uranium mixture from cycle 4 cycles 5-8 as feed material (.943 fraction needed cycle 5).

through the reactor of the pebbles. If the feed for the primary pebbles was processed to remove non-uranium nuclides, a P is shown. The recycle fraction of the discharge from the previous cycle used as feed is also shown. Considerable uncertainty is admitted in the resonance shielding calculation, so the dependence shown on C/Th is at best only approximate. Reduction in the power density from 6.25 to 5 W_{th}/cc reduces the design capacity from 1,500 to 1,200 Mw_e .

The recycle calculations were done by simply recycling a fraction of the fertile pebbles. With discrete steps of fueling and fractional recycle, the effective average residence time is given by

$$\theta = T (1-f) \sum_{n=1}^{\infty} n (f)^{n-1}, \text{ or}$$

approximately $\theta \approx \frac{T}{(1-f)}$ for continuous feed, where f is the fraction recycle and T the residence time. For $T = 90$ days and $f = 0.85$, θ is about 660 days. For a blanket residence time of 90 days and a holdup time out of core of 60 days, the residence time in the system averages about 1173 days, 73 days in the blanket, 660 days in the core and 440 days out of the reactor.

It appears essential to selectively remove highly exposed pebbles. Perfect selectivity reduces the average exposure time to

$$\theta = \frac{T}{(N+1) \left(1 - \frac{Nf}{2}\right)},$$

where $N = \frac{1}{(1-f)}$ truncated to an integer. For this model the above data yields a residence time of 364 days in the core.

If detailed calculations were made for selective discharge of materials, $N + 1$ pebble streams would have to be followed and exposure treated

independently. A linear approximation may be made to estimate the equivalent recycle fraction with selection, for the value of the continuous recycle fraction used, but mass balances must be preserved.

The early history was followed in Table 5 of a plant operated with a 0.25 year core residence time, 60 day hold up, and then recycle of pebbles. This information is used to construct a mass balance history for a 1,200 Mw_e plant which allows for the out of core inventory in the loops and feed material on hand. A ten year history (11.1 years at 0.9 load factor) is estimated as follows:

Year at Full Power	Breeding Ratio	Fissile Inventory (Kgm)			Fissile Feed (Kgm)	Fissile Discharge (Kgm)
		Reactor	Primary Loops	System		
0		2,107	2,107	3,407	3,407	
1	1.061	2,600	4,279	5,580	2,091	
2	1.054	2,650	4,362	6,740	3,249	2,161
3	1.051	2,700	4,444	7,644	3,010	2,174
4	1.049	2,740	4,510	7,710	2,186	2,186
5	1.048	2,770	4,559	7,759	2,181	2,196
6	1.047	2,790	4,592	7,792	2,175	2,205
7	1.046	2,810	4,625	7,825	2,182	2,211
8	1.045	2,830	4,658	7,858	2,188	2,215
9	1.045	2,850	4,691	7,891	2,191	2,218
10	1.044	2,870	4,724	7,924	2,195	2,221

If one considers a one year residence time (1.11 year at .9 load factor), a lead time of 0.5 and lag time of 1.0 real years, the ratio of the associated total fissile inventory to that in the reactor is 2.35. The ratio in the above estimate of mass balances is 2.76.

Using a fissile consumption rate of 1.112 Kgm/ Mw_e yr, the plant consumption is 1334 Kgm/yr. For a breeding ratio of 1.049, the annual

production is 65.4 Kgm fissile. The time required to produce enough fuel to inventory a second plant and associated system with an inventory of 7,924 Kgm is 135 years, at a load factor of 0.9. Note that the annual fuel production is less than one percent of the system inventory. Note also that of the required inventory of 7.924 Kgm after 11 years, only 654 Kgm were produced, requiring an external feed of U^{233} of 7,274 Kgm. There is breeding potential, but U^{235} fuel will have to be used for a long time.

Processing losses were ignored in the analysis above. The effect of fraction processing loss for the ten year history is shown below considering that all discharged fuel and the final on hand inventory would be processed:

Processing Loss (Percent)	Net Ten Year Production (Kgm)	Fraction of System Inventory	Effective Breeding Ratio
0	654	.082	1.049
0.5	541	.068	1.041
1.0	429	.054	1.032
1.5	316	.040	1.024
2.0	203	.025	1.015
2.5	90	.011	1.007

The data presented allows an estimate to be made of the fuel cost. Considering a thirty year operation at 0.9 load factor, fissile value of \$35/gm, 0.07 discount factor, and other parameters used earlier in this study, the following results are obtained in mils/ Kw_e hr using 0.07 discount factor compounded quarterly:

<u>Cost</u>	<u>Direct</u>	<u>Indirect</u>	<u>Total</u>
Fabrication	2.43	.29	2.72
Processing	1.82	-.34	1.48
Fuel	<u>-.07</u>	<u>3.21</u>	<u>3.14</u>
Total	4.18	3.16	7.34

Data were presented in Table 5 for a second cycle considering out of reactor hold up periods of 30, 60 and 120 days. Coarse economic analysis was done for the mass balances. Although the breeding ratio can indeed be increased by such holdup, the fuel cost was significantly increased due to the increase in pebble and fissile inventories. No economic incentive was found to justify such holdup.

The dependence of fuel cost on residence time is estimated for a first cycle as follows (mils/Kw_e hr):

Residence Time (Yr)	0.25	0.5	1.0	2.0
Fabrication	8.81	4.41	2.20	1.10
Processing	6.55	3.27	1.63	.80
Fuel	.17	.05	.02	.12
Indirect	<u>5.55</u>	<u>3.38</u>	<u>2.27</u>	<u>1.75</u>
Total	21.08	11.11	6.12	3.77

Note that the mode of operation is forced to a long residence time to achieve reasonably low cost of electrical energy.

Early Operation

One-dimensional problems were solved to examine the behavior of the single pebble concept with two pebble streams using U²³⁵ fuel. The primary pebbles containing fuel were introduced at the top of the core while fertile pebbles were introduced at the top of an inlet axial blanket 45 cm. thick. Both fully enriched U²³⁵ and low enriched (high security quality) U²³⁵ feed were considered. The results are summarized in Table 6 for a power density of 5 W_{th}/cc in a 1,200 Mw_e plant. A special qualification is placed on these calculations because additional effort should be invested in improving the accuracy of the resonance shielding and its dependence on the nuclide concentrations and carbon to heavy metal ratio. The reference C/HM was 160.

Table 6. Results for One Pebble Size and U^{235} Fuel
(One Dimensional, Four Group Calculations)

Case	C1	C2	C3	C4	C5	C6
Feed U Enrichment	0.18	0.17	0.20	0.95	0.95	0.95
Core Residence Time (Yr)	2.	2.	3.	2.	3.	3.
Cycle	1	2	1	1	1	2*
Peak Radial Average Power Density (W_{th}/cc)	10.9	12.5	13.9	10.0	13.3	14.9
Conversion Ratio	0.767	0.800	0.735	0.774	0.755	0.714
Fissile Inventory (kgm/Mw_e)	2.258	2.043	2.398	1.813	1.867	1.986
Actinide Mass Balance (kgm/Mw_e -Yr)						
Feed Th 232	28.521	28.521	19.014	32.721	21.814	21.373
U 233	-	0.43386	-	-	-	0.19173
U 234	-	0.03993	-	-	-	0.03346
U 235	1.3273	0.76279	1.0447	1.0890	0.85151	0.71729
U 236	-	0.14512	-	-	-	0.00066
U 238	6.0867	5.7251	4.0578	0.05316	0.04211	0.03792
Discharge Th 232	27.837	27.818	18.370	31.816	20.960	20.584
Pa 233	0.00337	0.00260	0.00078	0.00607	0.00129	0.00077
U 233	0.43049	0.57718	0.34784	0.49104	0.37977	0.39883
U 234	0.03993	0.09818	0.04269	0.06149	0.06472	0.09065
U 235	0.53703	0.31112	0.31811	0.32000	0.16796	0.17379
U 236	0.14512	0.21397	0.13194	0.13548	0.11837	0.10021
U 238	5.8723	5.5228	3.8536	0.04684	0.03505	0.03185
Np 239	0.00004	0.00003	0.00001	-	-	-
Pu 239	0.05100	0.04578	0.03662	0.00100	0.00079	0.00075
Pu 240	0.01889	0.01718	0.01454	0.00040	0.00033	0.00031
Pu 241	0.01786	0.01672	0.01521	0.00047	0.00040	0.00038
Pu 242	0.00561	0.00566	0.00697	0.00026	0.00034	0.00028
Am 243	0.00065	0.00066	0.00119	0.00003	0.00007	0.00005
Neutron Accounting (Fraction Absorptions)						
Fissile	0.50936	0.49631	0.50706	0.49811	0.49331	0.49080
Fertile	0.39408	0.39946	0.37654	0.38878	0.37048	0.35255
Other Actinides	0.00485	0.01002	0.00688	0.00484	0.00652	0.00513
C	0.01341	0.01399	0.01202	0.01838	0.01685	0.01545
Fission Products	0.06714	0.06660	0.08288	0.07834	0.09734	0.12006
Other	<u>0.01116</u>	<u>0.01362</u>	<u>0.01462</u>	<u>0.01155</u>	<u>0.01530</u>	<u>0.01601</u>
Total	1.0	1.0	1.0	1.0	1.0	1.0

*Only fertile pebble recycled without reprocessing.

With recycle of high U^{238} content material, the amount of fissile material in the total reprocessed stream decreases with a conversion ratio well below unity. If makeup material is added which also has a high U^{238} content, the feed uranium enrichment decreases with time. Either this decreased enrichment must be used and the thorium feed reduced, or highly enriched U^{235} feed be added as makeup. The former would reduce the U^{233} production and increase plutonium generation, reducing the conversion ratio.

Continuing recycle calculations were done with fixed U^{238} feed (two cycles shown in Table 5), and the calculated conversion ratio increases with increase in U^{233} feed for the cycles starting with the first cycle as follows: .767, .800, .829, .834, .837.

Fuel costs determined for the case of the early history with U^{235} feed with Pu throwaway are as follows (mils/Kw_e hr):

<u>Case</u>	<u>C1</u>	<u>C2</u>	<u>C3</u>	<u>C4</u>	<u>C5</u>	<u>C6</u>
Cycle	1	2	1	1	1	2
Residence time (yr)	2	2	3	2	3	3
- Basis -						
Reference	3.22	3.44	2.93	2.84	2.50	2.78
Devalued fuel	3.33	3.66	3.05	2.99	2.63	2.89
Throwaway fuel*	5.65	5.77	4.48	4.56	3.57	3.81
Credit only fertile pebble fuel	4.86	5.00	3.92	3.78	3.00	-

Ore requirements are estimated from first cycle in Mgm U_3O_8 /Mw_e capacity at 0.9 load factor, and approximated for recycle as follows:

<u>Case</u>	<u>C1</u>	<u>C3</u>	<u>C4</u>	<u>C5</u>
Reactor inventory	.476	.506	.382	.398
System inventory	.832	.758	.669	.597
Full fissile credit				
30 year use	2.120	2.217	1.626	1.798
30 year total	2.952	2.975	2.295	2.395
No fissile credit				
30 year use	7.550	5.943	6.249	4.927
30 year total	8.382	6.701	6.918	5.524
Fertile pebble fuel credit only				
30 year use	6.365	4.983	3.396	3.833
30 year total	7.197	5.741	4.065	4.430
Approximated for delayed recycle, full fissile credit				
30 year use	1.456	2.077	1.067	1.234
30 year total	2.288	2.835	1.736	1.831

*No charge for shipping and long-term storage.

Using the data on page 48 for once-recycle of unprocessed fertile pebbles with fully enriched U^{235} feed, and fissile throwaway, the fuel cost is estimated at 3.25 mils/kw_e hr. The 30-year ore requirement at a plant factor of 0.9 is estimated at 5.0 Mgm U_3O_8 /Mw_e capacity for inventory and make up, a 10 percent reduction over full throwaway. This could evidently be improved by loading the fertile pebbles with more than half of the thorium, and further assessment is needed considering the decrease in conversion ratio and the possible gain by increasing the C/Th ratio.

The dependence of the fuel cost of produced electricity on the unit charges and indirect cost rate is shown below for case C4; note that the handling cost should include fabrication, processing, shipping, disposal, etc., and a total reference cost of 350 \$/kgm has been used in this study:

Indirect Charge Rate Unit Handling Charge (\$/kgm)	0.07	0.10	0.13	0.16
150	1.67	2.02	2.37	2.71
300	2.30	2.64	2.99	3.33
450	2.92	3.26	3.61	3.95
600	3.53	3.88	4.23	4.57
750	4.15	4.50	4.85	5.19
1000	5.19	5.54	5.89	6.23
1250	6.23	6.58	6.93	7.27

Our economic assessment indicates that the fissile and fertile loading should be reduced to minimize the fuel cost. Likely, the optimum C/Th is well above 110 for U^{233} feed is needed and above 160 for U^{235} feed.

Further, to minimize ore consumption, a very long residence time of U^{235} feed is needed and a high C/Th. It appears that the calculations done toward the end of this study may overestimate the thorium resonance shielding for relatively small pebbles causing the fissile inventory to be underestimated.

The one-dimensional results reported above assume effective reflection of neutrons at the edge of the core. To achieve the low radial leakage which was assumed, a radial blanket of thorium pellets of some form would be required.

Two-Dimensional Results

The whole reactor was described with a two-dimensional (R-Z) model. Calculations were done treating 2, 4 and 8 energy groups with the once-through fuel movement followed in five regions of equal volume (axial flow in radial annuli). Results are shown in Table 7, which were obtained early in the effort treating 8 energy groups with a C to Th ratio of 88. A discrepancy in the assigned C density to the reflector caused the effective radial blanket thickness to be less than intended in the cases run initially.

Two-dimensional survey calculations were done treating four neutron energy groups. The residence time was varied over five flow paths (as above), the axial blanket was 51.6 cm thick at the entrance, and a fixed 36 cm radial blanket was included. The thorium loading was fixed and the carbon varied, and somewhat different resonance shielding parameters were used in this assessment. Uncertainty in the resonance shielding treatment causes inaccuracy in the intended parameter changes. Results are shown in Tables 8 and 9.

The fuel cost dependence on residence time is estimated as follows assuming an average residence time of 5 full power years for the fertile pellets and blanket for a power density of $5 W_{th}/cc$:

Residence Time (yr)	.5	1.	2.
Cost Component			
Fabrication	1.93	1.24	.92
Processing	1.47	.95	.71
Indirect	<u>5.42</u>	<u>4.14</u>	<u>3.75</u>
Total	8.82	6.33	5.38

Table 7. Results for the Two-Ball Pebble Bed Reactor
(Two-Dimensional, Eight Group, Fresh U^{235} Feed, Axial Blanket)

Power Level (w_{th}/cc)	Radial Reflector (cm)	Blankets (cm)		Residence Time (yr)		Fissile Feed Rates (Kgm/day)		In-Reactor Fissile Inventory (Kgm)	Peak Power Density (w_{th}/cc)	Breeding Ratio	In-Reactor Fissile Doubling Time (yr at 0.9 load factor)
		Axial	Radial	Primary	Fertile	Primary	Fertile				
6.25	4.2	31	0	1	.1	12.29	31.71	5,325	23.85	0.974	-
6.25	4.2	31	0	1	.1**	12.34	34.66	5,346	23.63	0.972	-
6.25	8.4	31	0	1*	.1*	12.11	31.71	5,263	23.06	0.978	-
6.25	8.4	31	0	1*	.1*	12.11	34.66	5,475	25.42	0.995	-
6.25	16.8	31	0	1	.1	11.84	31.71	5,176	22.22	0.981	-
6.25	16.8	31	0	1	.1*	11.78	34.66	5,147	22.33	0.983	-
6.25	16.8	31	0	1*	.1*	11.91	34.66	4,919	24.27	0.997	-
6.25	90	31	0	1*	.1*	10.33	34.66	4,378	16.16	1.007	360
5.0	90	31	0	1*	.1*	9.471	34.66	4,198	12.13	1.025	136
6.25	12.6	31	24***	1	.1	11.73	31.71	5,701	21.54	1.0086	435
6.25	12.6	31	24***	1*	.1*	11.80	34.66	5,576	23.65	1.0201	182
6.25	12.6	31	36***	1*	.1*	11.80	34.66	5,415	23.62	1.0333	107
6.25	12.6	31	36	1*	.1*	11.91	34.66	4,950	24.27	1.0380	85
6.25	12.6	52	36	1*	.1*	11.89	34.66	5,061	24.18	1.0471	66
6.25****	90	52	36	1*	.1*	14.32	34.66	5,862	22.44	1.0575	60

*Residence time varied in proportions 0.6, 0.8, 1, 1.2, 1.4 in the five paths of equal volumes radially outward

**Residence time inverted from *.

***These cases have a few years exposure products in the radial blanket.

****C/Th 80 in core instead of 88.

Table 8. Two-Dimensional Results for the Two Pebble Reactor

C/Th	Residence Times*		Power Density (W _{th} /cc)	Fissile Mass Balance (Rates, Kgm/Day)				In-Reactor Fissile Inventory (Kgm/Mw _e)	Peak Average Power Density (Kw _{th} /cc)	Breeding Ratio	In-Reactor Fissile Doubling Time (yr. at 0.9 load factor)
	(yrs)			Feed		Discharge					
	Primary	Fertile		Primary	Fertile	Primary	Fertile				
80	1.	.1	6.25	12.991	34.66	10.868	36.79	3.618	23.13	1.0565	62.4
110	1.	.1	6.25	7.961	34.66	6.336	36.38	2.517	28.75	1.0600	41.3
140	1.	.1	6.25	5.730	34.66	4.471	36.07	2.077	35.37	1.0630	32.9
110	1.	.1	5.0**	12.020	0	9.495	2.624	2.926	26.37	1.0670	43.0
110	1.	.1	5.0**	7.292	34.66	6.064	36.03	3.037	20.68	1.0783	38.4
110	2.	.1	5.0**	4.565	34.66	3.201	36.05	3.239	26.52	1.0470	68.2
110	.5	.1	5.0**	20.990	0	18.507	2.681	2.771	20.91	1.0925	29.7
110	.5	.1	5.0**	12.461	34.66	11.346	36.15	2.889	18.34	1.1020	28.1

* Residence time distribution: 0.6, 0.8, 1, 1.2, 1.4 in five flow paths.

** Reduced power density means lower plant design capacity, a reduction from 1,500 Mw_e to 1,200 Mw_e .

Table 9. Results for the Two Pebble Concept
(Two-Dimensional, Four-Group Calculations)

Case	L1	L2	L3	L4	L5	L6
C/Th	80	110	140	110	110	110
Power Density (W_{th}/cc)	6.25	6.25	6.25	5.0	5.0	5.0
Core Residence Time (yr)	1./1	1./1	1./1	.5/.1	1./1	2./1
Primary/Fertile						
Breeding Ratio	1.0565	1.0600	1.0630	1.1020	1.0783	1.0470
Doubling Time (yr)	62.4	41.3	32.9	28.1	38.4	68.2
Actinide Mass Balance (Kgm/ Mw_e -yr, including radial blanket)						
Feed Th^{232}	694.0	694.0	694.0	893.7	867.5	854.4
U^{233}	11.595	10.371	9.828	14.332	12.760	11.931
Discharge Th^{232}	692.7	692.7	692.7	892.3	866.1	853.1
Pa^{233}	.490	.447	.466	.560	.495	.444
U^{233}	11.16	9.954	9.425	13.883	12.347	11.529
U^{234}	.151	.140	.164	.136	.138	.135
U^{235}	.008	.007	.006	.003	.006	.010
Fissile Inventory (Kgm/ Mw_e)						
	3.618	2.517	2.077	2.889	3.037	3.239
Neutron Loss Accounting (Fraction Absorptions by Nuclide)						
Th^{232}	.48833	.48563	.48298	.50195	.49279	.47694
Pa^{233}	.00588	.00739	.00823	.00505	.00607	.00658
U^{233}	.45648	.45109	.44660	.44913	.45058	.45028
U^{234}	.00368	.00333	.00298	.00137	.00263	.00502
U^{235}	.00028	.00038	.00043	.00007	.00025	.00087
Xe^{135}	.01247	.01641	.01809	.01604	.01548	.01569
Nd^{143}	.00089	.00135	.00162	.00065	.00112	.00198
Pm^{147}	.00133	.00126	.00114	.00060	.00103	.00171
Pm^{148m}	.00043	.00044	.00041	.00019	.00034	.00060
Pm^{148}	.00022	.00020	.00018	.00008	.00014	.00027
Sm^{149}	.00312	.00321	.00318	.00272	.00299	.00339
Other fp	.01138	.01085	.01005	.00463	.00871	.01666
C	.00462	.00910	.01482	.01041	.00946	.00865
Other losses	.01089	.00936	.00929	.00542	.00841	.01136
Total	1.0	1.0	1.0	1.0	1.0	1.0

The requirement to inventory the volume associated with hold up of the fertile streams places a high demand on fuel feed in the early history. For a 120 day hold up of a feed rate of $.03 \text{ kgm/D /Mw}_e$, the amount would be 3.6 kgm/Mw_e , which adds an indirect charge of $1.1 \text{ mils/kw}_e \text{ hr}$ to the thirty year fuel cost.

Further Discussion

It appears possible to provide for trim and safety shutdown with control rods, although not simply. Rod tubes may be required with associated structure. Trim control is likely necessary to compensate for the tendency of the flux to roll around the core, and perhaps for load following. Also trim can be expected to be necessary to compensate for fuel addition and removal which could be increased with small pebbles flowing between large ones. Control rod absorptions have not been included in the calculations; performance would be degraded.

It may not be practical to achieve the fertile loadings considered, especially that in the small fertile pellets. Further assessment is needed of the inner structure and outer coating required on the pebbles. At the low rate of fissile production achievable with this reactor concept with U^{233} fuel without serious economic penalty by high fuel cost, the availability of U^{233} to inventory plants would lie far in the future. Thus it appears desirable to place primary emphasis on the performance with U^{235} feed and uranium recycle in continuing analysis. More attention is needed in the areas of resonance shielding, fission product treatment, axial leakage, economic optimum configuration and modeling of the reactor with selective recycle. There is special need to calculate a quasi-equilibrium condition involving a snapshot of the reactor under some defined state of

partial recycle after hold up of discharged material with reprocessing and make up feed. There is some difficulty fabricating an accurate reactor history from snapshots of continuous fueling conditions, which is not simply resolved; only an approximation can be made of the reactor history which becomes less accurate as impacted by hold up and more than one residence condition.

Possible alternatives should be considered which will reduce the cost of the electricity product. This implies reducing the associated system fissile inventory. While it is true that fertile pebbles can be used as blanket material prior to insertion in the core, the economic optimum configuration and mode of optimum has not been identified for a practical design. Increases in system inventory for blankets and out-of-core hold up add significant indirect costs. The cost of the reactor should be considered (capital charges) and acceptable rates used for all indirect costs, costs for fabrication and processing, and fuel worth. More attention should be paid to establishing the maximum allowable local power density, limiting exposure, and minimizing the power peaking to minimize the core size and cost.

Realistic economic projection requires better estimates of unit costs for fabrication, reprocessing and disposal. The data used herein are undoubtedly low estimates, possibly applicable to a very large industry. Complications which should be taken into account are the projected throughput requirements, dependence on pebble design and loading, and higher fabrication load than processing load in an expanding industry.

There are also technical questions to be addressed which involve practical considerations. It is not clear that feed pebbles can easily

be distributed and then discharged adequately as well as selectively removed for reprocessing.

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