**PROGRESS REPORT
ON FUEL RE-USE, II:
ANALYSIS
OF FOUR FAST-THERMAL
REACTOR COMPLEXES**

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INTRODUCTION

The work being reported herein extends the previous investigation⁽¹⁾ to include several additional types of thermal reactors, to cover stainless steel as well as Zircaloy cladding, and to obtain a more detailed analysis of the thermal-to-fast reactor power ratio. Pressurized and boiling light water reactors, because of their current and near-term commercial usage, were selected as typical of thermal reactors with which one might anticipate fairly large fuel cycle cost savings but low thermal-to-fast reactor power ratios. A D₂O-cooled and -moderated pressure tube reactor and a batch-fueled variable spectrum D₂O reactor were selected as typical of those high-neutron-economy reactors wherein one may expect somewhat lower fuel cycle cost savings, but near maximum thermal-to-fast reactor power ratios.

For both fast and thermal reactors, the normal fuel cycle begins with a complicated and expensive multistage fabrication process. Then, the finished fuel elements (subassemblies) are packaged and shipped to the reactor for irradiation to some desired reactivity limit. On removal from the core, the fuel is stored for decay of short-lived fission products and then shipped in heavily shielded casks to a chemical reprocessing plant for recovery of the valuable fuel materials.

The fuel re-use scheme evaluated herein causes only a small change in the normal fuel cycle of the fast reactor. For the thermal reactor, however, the normal fuel cycle is completely eliminated; the total fuel cycle of the thermal reactor becomes simply a side stream on the fast reactor blanket cycle.

The obvious attractions of such a fuel re-use scheme are that, for the thermal reactor, most fabrication, shipping, and reprocessing expenses and most out-of-reactor inventory costs are completely eliminated or drastically reduced. For the fast reactor, there are no major cost perturbations; the thermal reactor pays standard burnup and inventory charges for the irradiated blanket fuel elements while they reside in the thermal reactor fuel cycle.

SUMMARY OF RESULTS AND CONCLUSIONS

These fuel cycle cost analyses compare U^{235} -enriched uranium fuel with depleted uranium fuel elements which have been enriched with plutonium by irradiation in a fast reactor blanket. By absorbing economic perturbations of the fast blanket cycle in the thermal reactor fuel cycle, it is possible to restrict the major analyses solely to the thermal fuel cycle. The fast reactor selected (the same as in the previous study) is a mixed PuO_2 - UO_2 fuel core with a depleted UO_2 blanket.

The pressurized, boiling, and heavy water reactor types used graded burnup with the reactors operating at an average k_{∞} of 1.04. The variable-spectrum heavy water reactor operated on a batch cycle, with continuous spectrum adjustment and a terminal k_{∞} of 1.03.

As a result of the studies represented in this report, several conclusions can be reached, all of which support the results of the previous document (HW-74549), and some of which present the results in a new light. A summary of these conclusions follows:

- With Zircaloy clad fuel, the heavy water reactors and boiling water reactor achieve burnups of 15 to 20 times the burnup provided in the fast reactor blanket; with stainless steel cladding, this burnup ratio is halved. The pressurized water reactor achieves about one-half to two-thirds the burnup ratio of the other thermal reactors. (See Figures 9 and 10.)

- With Zircaloy clad fuel, thermal-to-fast-reactor power ratios of about 1.6 may be achieved with heavy water and boiling water reactors; this ratio falls to about unity with stainless steel clad. The PWR achieves about two-thirds the power ratio of the other thermal reactors. (See Figures 12 and 13.)
- For comparable economic parameters, fuel costs are always lower for fuel re-use than for the standard fuel cycle.
- For equal fuel cycle cost, all thermal reactor types studied can afford to pay higher prices for fast blanket enriched fuel elements than for normal U^{235} enriched fuel elements. (See Figures 16 through 19.)

NUCLEAR ENGINEERING ANALYSIS

REACTOR MODELS

The 1000 MW_t fast reactor selected is the same one reported in the preceding study.⁽¹⁾ For convenience, the general description of the reactor is repeated here.

To carry out the physics calculations, an equivalent three region spherical reactor model was constructed; its composition and dimensions are given below.

Physics Model for Fast Reactor

<u>Region</u>	<u>Outer Radius, cm</u>	<u>Fuel</u>	<u>Zirconium</u>	<u>Iron</u>	<u>Sodium</u>
Core	81.9	30		15	55
Breeding Blanket	126.9	55	7.5	7.5	30
Reflector	156.9	0		40	60

The core fuel is mixed PuO_2 - UO_2 ; the blanket fuel is depleted (0.38% U^{235}) UO_2 .

The isotopic composition of the core plutonium, given below, was selected as representative of both fast reactors on equilibrium recycle and of light water reactors on high burnup enriched uranium cycles.

Atom Fractions of Total Plutonium in Fast Reactor Core

<u>Pu²³⁹</u>	<u>Pu²⁴⁰</u>	<u>Pu²⁴¹</u>	<u>Pu²⁴²</u>
0.6822	0.1913	0.1029	0.0235

The four thermal reactors are patterned after models which have been in general use in fuel cycle analysis at Pacific Northwest Laboratory.⁽²⁾ The models describing these reactors are set up for use with MELEAGER,⁽³⁾ a thermal reactor point burnup code based on the Wescott cross-section scheme. The four reactor types were chosen to expand the initial examination of Fuel re-use to other reactor types of established national interest in addition to the Heavy Water Moderated type (HWR) previously studied. Thus, the additional types considered are a Boiling Light-Water Reactor (BWR), a Pressurized Light-Water Reactor (PWR), and a Variable Spectrum Heavy Water Reactor (VSHWR). Table I outlines the general characteristics of these four reactor types, and the following paragraphs provide some additional information about these reactor models.

TABLE I. Thermal Reactor MELEAGER Physics Parameters

	<u>HWR</u>	<u>BWR</u>	<u>PWR</u>	<u>VSHWR</u>
1) Slowing down power per unit volume (SDPV)	2.880	2.112	1.3076	Variable
2) Wescott Spectral Parameter (r) ^(a)	0.0975	0.239	0.422	Variable
3) Macroscopic nonfuel absorption cross-section per unit volume				
Zircaloy	0.020	0.0296	0.0344	0.020
Stainless Steel	0.100	0.100	0.100	0.100
4) Average Initial Neutron Temperature, °C	144	300	380	Variable
5) Specific Power, MW/ton	←—————20—————→			
6) Fuel Density, gm/cm ³ fuel	←—————9.256—————→			

(a) This varies, somewhat, with the fuel type and also with burnup, and is intended only to be indicative of the type of spectrum.

The Heavy Water Moderated Reactor Model differs slightly from the model used in the previous study. The present model has a slightly greater slowing down power per unit volume (SDPV) and slightly higher neutron temperature. The new SDPV represents a slightly wider average lattice spacing. This model was chosen as representative of a reactor type with a soft neutron energy spectrum and a high neutron efficiency. The Boiling Water and Pressurized Water type reactor models are characteristic of reactors of somewhat harder neutron energy spectrums. They are chosen as representative of the light-water moderated type of reactors of the current era. Finally, the Variable Spectrum Heavy Water model is a heavy-water moderated reactor with a continuously variable spectrum, the spectrum being changed by varying the fuel-to-moderator ratio. In this model, the neutron energy spectrum is adjusted repeatedly during a burnup to maintain the desired reactivity.

Although the VSHWR is allowed to optimize the neutron energy spectrum, none of the the models can be considered a fully optimized system; and none should be considered as more than representative of a type.

FUEL BURNUP CALCULATIONS

Curves demonstrating plutonium isotope buildup in the fast reactor blanket are shown in Figure 1. Values of isotope concentrations are taken from these curves and utilized as input to the thermal reactor models. The fast reactor blanket's most notable characteristic--a very high breeding ratio (circa 5 to 15)--is portrayed by the curve labeled "Total Enrichment."*

The fuel burnup in the thermal reactors followed a constant power, graded cycle, except for the VSHWR which followed a batch cycle. Each fuel element was assumed to be irradiated at constant specific power of 20 MW/ton** as it passed through the reactor. (In the previous study, the specific power of the HWR

* Includes all odd-numbered isotopes of uranium and plutonium.
** 1 ton = 2000 lb.

was 10 MW/ton.) The average k_{∞} for the graded fuel cycles was 1.04 and for the VSHWR batch cycle was 1.03. The excess reactivity accounts for neutron leakage, high cross-section fission product poisoning, etc.

FIGURE 1. Isotope Buildup in the Fast Reactor Blanket

A sample printout of a MELEAGER burnup calculation is provided in Appendix A; the case shown is the Pressurized Water Reactor on a U^{235} enriched cycle using 2.5% enriched, Zircaloy clad fuel. Figure 2 is a typical plot of isotope concentration versus burnup, as taken from MELEAGER burnup data; the case shown is the Heavy Water Reactor on a fuel re-use cycle using the 2.0% enriched, Zircaloy clad fuel (Figure 1).

FIGURE 2. *Isotopic Buildup Versus Burnup in Thermal Reactor HWR with 2% Enriched, Zircaloy Clad, Re-Use Fuel*

Figures 3 through 6 show attainable burnup as a function of initial enrichment for (a) U^{235} enriched uranium fuel (U^{235}) and (b) fast reactor blanket enriched fuel (re-use), with both Zircaloy (Zry) and stainless steel (SS) cladding, for all four of the thermal reactors studied. Several observations on these figures may be made:

- (1) Zircaloy clad fuel requires less enrichment for a given burnup than stainless steel clad fuel in all reactors. This commonly reported characteristic derives from the large absorption cross section of stainless steel compared to Zircaloy (See Figure 3 for example).

FIGURE 3. Heavy Water Reactor: Burnup Versus Enrichment for U²³⁵ Enriched Uranium Fuel and Plutonium Enriched Re-Use Fuel

FIGURE 4. Variable Spectrum Heavy Water Reactor: Burnup Versus Enrichment for U²³⁵ Enriched Uranium Fuel and Plutonium Enriched Re-Use Fuel

FIGURE 5. Boiling Water Reactor: Burnup Versus Enrichment for U²³⁵ Enriched Uranium Fuel and Plutonium Enriched Re-Use Fuel

Figure 6. Pressurized Water Reactor: Burnup Versus Enrichment for U²³⁵ Enriched Uranium Fuel and Plutonium Enriched Re-Use Fuel

- (2) Stainless steel clad re-use fuel requires significantly less enrichment than stainless steel clad U^{235} fuel for a given burnup in all reactors. This superior performance of plutonium fuel over U^{235} fuel is a direct result of plutonium's larger absorption cross section by competing more successfully for neutrons, plutonium achieves a higher thermal utilization factor.
- (3) Zircaloy clad re-use fuel and Zircaloy clad U^{235} fuel provide similar performance in D_2O -moderated reactors. In this case, with very little parasitic absorber in the reactors, plutonium's thermal utilization factor is only slightly higher than U^{235} 's, and this advantage is off-set by plutonium's lower η .
- (4) Zircaloy clad re-use fuel slightly outperforms Zircaloy clad U^{235} fuel in H_2O -moderated reactors. This case falls between (2) and (3), above: hydrogen's absorption cross section is just large enough to make plutonium's larger thermal utilization factor outweigh its smaller η .

BURNUP RATIOS

The information in Figure 1 (enrichment buildup as a function of a fast blanket burnup), when combined with the information in Figures 3 through 6 (thermal reactor burnup as a function of fuel enrichment), yields curves of thermal reactor burnup as a function of fast blanket burnup. The resulting curves are shown in Figures 7 and 8 for Zircaloy and stainless steel clad fuel, respectively. For all reactor and clad types, the attainable thermal reactor burnup rises continuously, although at a decreasing rate, with fast blanket burnup (within the range of fast blanket burnup studied). This is, of course, quite expectable since the fuel enrichment is rising continuously (but at a decreasing rate) with fast blanket burnup.

FIGURE 7. Burnup Attainable in Thermal Reactor Versus Burnup Acquired in Fast Reactor Blanket for Zircaloy Clad Fuel

FIGURE 8. Burnup Attainable in Thermal Reactors Versus Burnup Acquired in Fast Reactor Blanket for Stainless Steel Clad Fuel

To emphasize more fully the relative energy output from a unit of fuel irradiated in a fast blanket and then in a thermal reactor, the data in Figures 7 and 8 can be recomputed to form the ratio, β , of burnup in the thermal reactor to burnup in the fast blanket. The resulting burnup ratios (β) are plotted as functions of fast blanket burnup in Figures 9 and 10, for Zircaloy and stainless steel clad fuels, respectively. With Zircaloy clad fuel, the burnup ratio (β) ranges from 16 to 24 in the heavy water reactors and approximates 15 in the BWR and 10 in the PWR. Substituting stainless steel cladding for Zircaloy causes a dramatic ($\sim 50\%$) drop in the burnup ratio: in the heavy water reactors, β drops to about 9, in the BWR to about 8, and in the PWR to about 5.

POWER RATIOS

One important measure of the utility of this fuel re-use concept is the ratio, R , of total energy generated in thermal reactors re-using fast blanket fuel elements to the total energy generated in fast reactors supplying "enriched" blanket fuel elements. For the concept ever to have true value, the total thermal reactor power must be significant compared to total fast reactor power (i.e., $R \gg 0$). Further, it is desirable that R be controllable over a reasonable range to meet the presently uncertain and predictably variable economic requirements of the future. This controllability feature may be viewed thusly: At the present stage of technological development, the cost of power from fast reactors would exceed the cost of power from thermal reactors; therefore, only thermal reactors will be built for central station power production until fast reactor technology is significantly advanced. At some future date, the fast reactor is expected to achieve dominance because its power will cost less than thermal reactor power. Obviously, at some time, there will be a transition period during which power costs will be comparable and both types of reactors will be built and operated. At the beginning

FIGURE 9. Thermal Reactor Burnup-to-Fast Blanket Burnup Ratio Versus Fast Blanket Burnup for Zircaloy Clad Fuel

Figure 10. Thermal Reactor Burnup-to-Fast Blanket Burnup Ratio Versus Fast Blanket Burnup for Stainless Steel Clad Fuel

of the transition period, when thermal reactor power is still somewhat cheaper than fast reactor power, there is an incentive to make R large. Conversely, near the end of the transition period, when thermal reactor power is somewhat more costly than fast reactor power, there exists an incentive for making R small.*

For a steady-state system, the thermal-to-fast reactor power ratio (R) is the product of the fast blanket-to-fast reactor power ratio (α) and the thermal reactor-to-fast blanket burnup ratio (β):

$$\frac{\text{thermal reactor power}}{\text{fast reactor power}} = \frac{\text{blanket power}}{\text{fast reactor power}} \frac{\text{thermal burnup}}{\text{blanket burnup}}$$

$$R = (\alpha)(\beta)**$$

Figure 11 is a plot of α as a function of fast blanket exposure. For the specific fast reactor configuration and composition chosen for this study, α is shown to increase monotonically from 0.06 for a clean blanket to 0.16 at a blanket burnup of ~ 7000 MWd/ton.***

Figures 12 and 13 show curves of R , the thermal-to-fast reactor power ratio, as a function of fast blanket burnup for Zircaloy and stainless steel cladding, respectively. With Zircaloy clad fuel, the heavy water reactors achieve R values of 1.8 and 1.6 over their most interesting burnup range; similarly the BWR achieves about 1.4 to 1.5 and the PWR achieves

* It is fully recognized that fuel re-use must create a net reduction in the cost of (fast + thermal) reactor power or else fuel re-use will not be employed; furthermore, it is recognized that if fuel re-use is employed, the transition period will tend to be lengthened in both directions.

** See page 10 of Reference 1.

*** The blanket burnup calculations used a flux- and volume-weighted, zero-dimensional model; the α values reported here are apparently conservative, being the average of clean and removal burnup powers without correction (increase) of blanket fluxes for blanket source neutrons.

about 0.8 to 1.2. With stainless steel clad fuel, the heavy water reactors achieve R values of about 1.0, the BWR about 0.7 to 1.0, and the PWR from 0.3 to 0.7.

FIGURE 11. *Fast Blanket Power-to-Fast Reactor Power Ratio Versus Fast Blanket Burnup*

FIGURE 12. Thermal Reactor Power-to-Fast Reactor Power Ratio Versus Fast Blanket Burnup for Zircaloy Clad Fuel

FIGURE 13. Thermal Reactor Power-to-Fast Reactor Power Ratio Versus Fast Blanket Burnup for Stainless Steel Clad Fuel

ECONOMIC ANALYSES

BASIS OF EVALUATION

Schematic diagrams of the several fuel cycles are presented in Figure 14. As shown, the core and blanket cycles of the fast reactor and the standard slightly-enriched uranium fuel cycle of the thermal reactor are quite similar; all are complex processes involving many expensive operations. The fuel re-use cycle, contrarily, is shown to be basically very simple from the standpoint of the thermal reactor operator. Comparison of the fuel re-use diagram to that of the standard thermal reactor fuel cycle provides a vivid portrayal of the economic incentives for fuel re-use.

Figure 14 indicates that the fuel re-use cycle is only a small side loop of the normal fast reactor blanket cycle. This small loop, however, can produce major changes in the costs of the normal blanket cycle: normal plutonium revenues are reduced and the time of their receipt is delayed; nonoptimum blanket irradiation times may be required; additional blanket fuel fabrication costs may be necessary. Precise evaluation of fuel re-use would require analysis of both the thermal reactor fuel cycle and the fast blanket fuel cycle by comparing the optimized combinations of the two cycles operating in their standard modes and in the fuel re-use mode. However, the scope of the problem can be reduced to analysis of only the thermal reactor fuel cycle by adopting two simplifying bookkeeping techniques which would not change the fast blanket cycle costs. The first of these is to credit the blanket cycle with the appropriate fuel burnup and use charges of the thermal reactor; this will quite nicely off-set the reductions and delays in receipt of the blanket's plutonium revenue. The second technique is to assess a special service charge for each unit of blanket fuel irradiated in the thermal reactor; the magnitude of this service

FIGURE 14. Schematic Fuel Cycle Diagram

charge will be set to just balance any incremental blanket fuel fabrication costs (and the interest on same) and any costs due to nonoptimum blanket irradiation time.

Evaluation of the economic merits of fuel re-use, then, is based on comparison of the minimum total fuel cycle costs of thermal reactors operating on the fuel re-use cycle and those operating on the standard U^{235} enriched uranium fuel cycle. The minimum total fuel cycle cost is that cost, corresponding to a particular enrichment and burnup, that is lower than any other cost, corresponding to any other particular enrichment and burnup, when all the various economic factors and the manner in which each varies with enrichment and burnup are taken into account. A detailed discussion of the necessity for the use of the minimum total fuel cycle cost as a criterion for evaluating any reactor fuel cycle is provided in Reference 2.

UNIT COSTS

All fuel cycle cost calculations were performed using the Pacific Northwest Laboratory QUICK code.⁽⁴⁾ This code is designed to operate directly on the burnup data provided by the MELEAGER code,⁽³⁾ in conjunction with those economic parameters which describe the fuel cycle. The QUICK code is generally patterned after the techniques and methods of the Nuclear Power Plants Cost Evaluation Handbook;⁽⁵⁾ interest charges are time-based on the weighted average of power sale receipts.

The economic parameters used in calculating the costs of both the "standard" U^{235} enriched uranium fuel cycle and the fuel re-use cycle are listed in Table II. The reactor performance parameters--880 MW_t, a thermal-to-electrical conversion efficiency of 33.3%, specific power of 20 MW/ton, and 80% load factor--are the same for both cycles. Similarly, the fuel

TABLE II. *Economic Parameters*

	<u>"Standard"</u> <u>U²³⁵ Enriched</u> <u>Uranium Cycle</u>	<u>Fuel Re-Use</u> <u>Cycle</u>
Reactor Power, MW _t	880	880
Thermal Efficiency	0.333	0.333
Fuel Loading, kg	40,000	40,000
Fuel Density, g/cm ³	9.25	9.25
Load Factor, %	80	80
<u>Uranium Price, AEC Schedule,</u> <u>Date</u>	July 1, 1963	July 1, 1963
<u>Plutonium Price, \$/g, fissile</u>	10.00	10.00
<u>Interest Rates, %/yr</u>		
Working Capital	12.5	12.5
Fuel Use Charge	4.75, 7.0, 10.0	4.75, 7.0, 10.0
<u>Out-of-Reactor Inventory Time, Days</u>		
Total Time in Fabrication	60	0
Prereactor Shipping Time	40	1
Precharging Inventory ^(a)	100	10
Decay Storage Time	120	0
Postreactor Shipping Time	90	1
Total Time in Reprocessing	75	0
<u>Shipping Charge, \$/kg</u>		
Prereactor	3.30	2.20
Postreactor	7.70	0
<u>Fuel Fabrication Cost, \$/kg</u>	22, 44 66, 88	0
<u>Reprocessing Cost, \$/kg</u>	22	0
<u>Fuel Losses, %</u>		
Fabrication	2.0	0
Reprocessing	1.3 for U 2.0 for Pu	0
<u>Fuel Service Charge, \$/kg</u>	0	2.20, 4.40, 11.00, 15.40, 22, 44, 66, 99, 132, 165, 198, (242, 286, 330) ^(b)

(a) *Including allowance for spares*(b) *242, 286, and 330 \$/kg used for the PWR only*

values of both cycles are based on the uranium price schedule of July 1, 1963, and a plutonium price of \$10.00 per gram fissile.*

Identical interest rates were used for both cycles: working capital was assessed 12.5% per year; fuel use charges were parameterized at 4.75% per year to cover both the present AEC leasing practice and future public power costs, and at 7.0 and 10.0% per year to represent future costs of investor-owned utilities.

Out-of-reactor inventory times are totally different for the two cycles (See Figure 14). The standard cycle involves procurement of UF_6 and shipment to a conversion plant; conversion of fluoride to oxide and transfer to a fabrication plant; preparation, cladding, assembly, inspection, packaging, and shipment to the reactor; precharging inventory and spares; storage for decay of short-lived fission products and accumulation of a reprocessing batch; shipment to a reprocessing plant, separation and recovery of uranium and plutonium, and shipment to central receiving and storage sites. Inventory times for the standard cycle total 200 days preirradiation and 285 days postirradiation.

The fuel re-use cycle is based on close-coupled operation of the fast and thermal reactors with the thermal reactor being located adjacent to the fast reactor. The fuel subassembly discharged from the fast reactor blanket is washed; the fuel element (rod bundle) is removed from the subassembly can, and the fuel element is transferred directly to the core of the thermal reactor. Fuel discharged from the thermal reactor is transferred back to the fast reactor plant for decay storage (or, possibly, further irradiation in the blanket) and further

* Although it is acknowledged that the value of plutonium is affected by isotopic composition, neutron spectrum, and fuel density and geometry,⁽²⁾ and that the price of plutonium will be dependent on its applications--including such as fuel re-use⁽¹⁾--it is believed that a fixed price of \$10.00 per gram fissile plutonium will not produce misleading conclusions in this evaluation.

processing through the normal blanket fuel cycle. Since this evaluation is based on holding the fast blanket costs invariant and since it is assumed that the fast and thermal reactors will be operated in close unison, it is estimated that only 11 days total preirradiation inventory will be required and that only 1 day will be required to return fuel elements discharged from the thermal reactor to the fast reactor storage basin.*

Shipping costs are quite different for the two cycles. For the "standard" cycle, the cost of all fuel shipments from UF_6 to reactor delivery, including insurance, is assumed to total \$3.30/kg of fuel (uranium). Similarly, the total postreactor shipping charge for the standard fuel cycle is assumed to be \$7.70/kg. The fuel re-use cycle is assessed \$2.20/kg of fuel prereactor shipping charge--an arbitrarily selected fee to cover the cost of a fast reactor-to-thermal reactor transfer cask. No postreactor shipping charge is levied, it being assumed that this is merely the return trip of the prereactor transfer. The fuel fabrication cost for the standard U^{235} -enriched uranium cycle is parameterized at \$22/kg, \$44/kg, \$66/kg, and \$88/kg of fuel. The higher charges represent the fairly near-term; the \$22/kg charge is a possible lower limit for a high volume standardized and automated industry. The total fuel reprocessing cost of the "standard" cycle is set at \$22/kg of fuel; this cost is reasonably expectable in the near future. Fuel losses in the "standard" cycle amount to 2% in fabrication; and 1.3 and 2% in reprocessing for uranium and plutonium, respectively. In the fuel re-use cycle (as analyzed here) there is neither fabrication nor reprocessing of fuel: fuel fabrication and reprocessing costs are zero. Furthermore, there can be -- by definition -- no loss of fuel in either fabrication or reprocessing.

* The reader is reminded that the assignment of storage and reprocessing costs, etc., to the blanket cycle is purely one of bookkeeping convenience; the important fact to keep in mind is that these costs are incurred once only for the two cycles.

For the fuel re-use cycle only, a fuel service charge was applied. This charge was initially intended to cover only those incremental fabrication costs and costs due to nonoptimum irradiation times of the fast blanket cycle incurred solely because of the fuel re-use operation. The meaning or significance of the fuel service charge was later extended to cover the full bargaining range of prices that might be charged the thermal reactor operator by the fast blanket operator.

Incremental costs to the fast blanket cycle because of fuel re-use might fall anywhere within a very wide range. At the lower end, the costs may be insignificantly small: the normally optimum irradiation time in the blanket might be just that required to produce the enrichment needed in the thermal reactor, and the fuel design might be ideal for the thermal reactor. On the other hand, the incremental costs might be quite large: blanket irradiation times substantially different from those normally optimum might be required; costly changes in fuel design might be necessary.

Specific incremental costs that might be incurred include costs for substituting Zircaloy-type cladding for stainless steel-type cladding; the incremental cost might be 4 to \$10/kg. Another possible cost is for reduction of fuel rod diameters from the large sizes that might be used in fast reactor blankets to a substantially smaller diameter more suitable for the thermal reactor core; this incremental cost might range up to 10 to \$20/kg, or, in large scale future operations, up to 4 to \$7/kg. Special "take-apart" features and spacing devices may be desired for the thermal reactor; these might add a cost of 2 to \$10/kg.* Incremental fabrication costs, then, might total as much as \$40/kg in early applications and up to \$10/kg in large-scale future operations.

* Because of the very large burnup and enrichment gradients produced in blanket fuel elements, disassembly-rearrangement-reassembly-and-reirradiation may be an economic operation for the blanket cycle, per se; if such a practice were used, there might be no disassembly cost increment attributable to fuel re-use in the thermal reactor.

Cost attributable to nonoptimum irradiation times in the fast reactor blanket are even more nebulous. Obviously, if the enrichment requirements of the thermal reactor happen to coincide with the optimum enrichment produced in the fast blanket, this cost will be zero. Conversely, if the enrichment requirements differ by some factor 'n' from the enrichment normally produced, the total cost of the fast blanket cycle might be increased by some significant fraction of 'n'. The cost for nonoptimum irradiation times is estimated to range as high as 40 to \$70/kg for early applications and as high as 10 to \$20/kg in more advanced systems.* In total, then, the incremental costs in the fast reactor blanket cycle might range from zero to \$110/kg in early usage and from zero to \$30/kg in future large scale applications.*

The secondary purpose of this analysis to help define the bargaining price range for irradiated blanket elements, requires that the fuel service charge range sufficiently high as to establish an indifference value. The term "indifference value" is defined and used herein in meaning and manner similar to its use in plutonium value analysis:** when the fuel service charge equals the indifference value, the thermal reactor operator is indifferent to operating on a standard U^{235} enriched uranium cycle or the fuel re-use cycle, since his total fuel cycle costs are the same for either cycle. As indicated in Table II, this latter purpose for the fuel service charge required unit costs much higher even than the fabrication costs for enriched uranium fuel elements. As will be shown in the next section, it was necessary to carry the fuel service charge values far in excess of reasonably expectable costs--in some cases well in excess of \$200/kg.

* *The ability to reirradiate in the fast blanket restricts the incremental cost to a fairly small fraction of total blanket cycle costs. Furthermore, multiple blanket-thermal reactor fuel re-use cycles have been shown to be economically attractive (HW-74549) when incremental costs are as high as about \$220/kg.*

** *See, for example, pages 13-14 of Reference 2.*

FUEL CYCLE COSTS

For each reactor type, fuel cycle costs were computed for each combination of economic parameters at sufficient different enrichment-exposure combinations (usually 5 to 9) to develop the fuel cycle cost function and to define the minimum fuel cycle cost and optimum enrichment-exposure. Figure 15 is an example of variations with burnup of the total fuel cycle costs and the several major cost components.

FIGURE 15. *Example of the Variation of the Total Fuel Cycle Cost and the Several Cost Components with Fuel Burnup*

Figures B-1 through B-48, separately indexed at the beginning of Appendix B, show total fuel cycle costs as functions of burnup for the various parameters of reactor type, clad material,

use charge, fabrication cost, and fuel service charge. Perusal of these figures shows that the minimum fuel cycle cost occurs at low exposure for low unit fabrication costs and moves to consecutively higher exposures at higher unit costs; the absolute value of the minimum fuel cycle cost, of course, increases as the unit costs increase. Minimum fuel cycle costs, as determined by analyses of the Figures B-1 through B-48, are shown in Appendix C. Figures C-1 through C-24, indexed at the front of Appendix C, show these minimum total fuel cycle costs as functions of the major cost parameters--fuel fabrication cost and fuel service charge--for each reactor type, clad material, and use charge.

Tables III to VI indicate selected minimum fuel costs for four reactor models.

TABLE III. *Selected Minimum Fuel Cycle Costs: Heavy Water Reactor Model (Use Charge = 10%)*

		Fabrication Cost, \$/kg	Minimum Fuel Cycle Cost, mills/kW-hr	
			<u>Zircaloy-Type Clad</u>	<u>SS-Type Clad</u>
Standard Fuel Cycle	}	60	1.12	1.87
		50	1.05	1.82
		40	0.97	1.76
		30	0.89	1.70
		20	0.79	1.64
		Fuel Service Charge, \$/kg		
Fuel Re-Use	}	100	1.23	1.67
		50	0.86	1.38
		30	0.65	1.23
		20	0.51	1.13
		10	0.33	1.01

TABLE IV. *Selected Minimum Fuel Cycle Costs: Variable Spectrum Heavy Water Reactor Model (Use Charge = 10%)*

	Fabrication Cost, \$/kg	Minimum Fuel Cycle Cost, mills/kW-hr	
		Zircaloy-Type Clad	SS-Type Clad
Standard Fuel Cycle	60	1.09	1.69
	50	1.03	1.64
	40	0.96	1.58
	30	0.88	1.51
	20	0.80	1.44
	Fuel Service Charge, \$/kg		
Fuel Re-Use	150	--	1.70
	100	1.19	1.48
	50	0.87	1.22
	30	0.69	1.09
	20	0.57	1.02
	10	0.40	0.93

TABLE V. *Selected Minimum Fuel Cycle Costs: Boiling Water Reactor Model (Use Charge = 10%)*

	Fabrication Cost, \$/kg	Minimum Fuel Cycle Cost, mills/kW-hr	
		Zircaloy-Type Clad	SS-Type Clad
Standard Fuel Cycle	60	1.30	1.93
	50	1.24	1.88
	40	1.17	1.82
	30	1.11	1.77
	20	1.05	1.71
	Fuel Service Charge, \$/kg		
Fuel Re-Use	150	--	1.85
	100	1.31	1.61
	50	1.00	1.32
	30	0.82	1.18
	20	0.71	1.10
	10	0.56	0.99

TABLE VI. *Selected Minimum Fuel Cycle Costs: Pressurized Water Reactor Model (Use Charge = 10%)*

	Fabrication Cost, \$/kg	Minimum Fuel Cycle Cost, mills/kW-hr	
		Zircaloy-Type Clad	SS-Type Clad
Standard	60	1.52	2.21
Fuel	50	1.46	2.16
	40	1.40	2.10
	30	1.35	2.05
Cycle	20	1.27	1.99
Fuel Service Charge, \$/kg			
Fuel	150	1.53	1.83
Re-Use	100	1.28	1.58
	50	0.99	1.32
	30	0.85	1.20
	20	0.76	1.10
	10	0.63	0.99

As indicated in the tables and figures, fuel re-use is expected to provide major fuel cycle cost reductions in thermal reactors. For all reactor types, the fuel cycle cost is less with the fuel re-use cycle than with the standard U^{235} enriched uranium cycle, even when the fuel service charge is as large as the fabrication cost for enriched uranium fuel elements. For this condition (fuel service charge equals fabrication cost), the fuel cycle cost savings vary from 0.1 to 0.2 mills/kW-hr for the heavy water reactor model to 0.8 to 0.9 mills/kW-hr for the pressurized water reactor model. For the condition where the fuel service charge is set at a rate more nearly equal to the incremental cost to the blanket cycle (assumed to be half the fabrication cost for U^{235} -enriched uranium fuel elements), the fuel cycle cost savings with fuel re-use are even larger and more attractive. Under these more expectable

conditions, fuel cycle cost savings range from 0.4 mills/kW-hr with the heavy water reactor model to 1.0 to 1.1 mills/kW-hr with the pressurized water reactor model.

VALUE ANALYSIS

The indifference values of re-usable blanket elements are obtained from figures in Appendix C by reading, at constant minimum fuel cycle cost, paired values of fuel service charge (from the fuel re-use curves) and fabrication cost (from the standard cycle curve). This is demonstrated in Figure C-1. Plots of the indifference values of fuel service charge as functions of fabrication cost are shown in Figures 16 through 19. As indicated in the figures, the indifference values proved totally insensitive to use charge; these figures are valid, therefore, at least for all use charges in the range $4 \frac{3}{4}$ to 10%.

As shown in the figures the indifference value is always substantially larger than the fabrication cost. This means, of course, that the thermal reactor operator is always able to pay more for irradiated blanket fuel elements than for similar U^{235} -enriched fuel elements. Indifference values are higher with stainless steel cladding than with zirconium cladding and higher in light water reactors than in heavy water reactors.

Indifference values in the heavy water reactor vary from 40 to \$100/kg with zirconium cladding to 80 to \$165/kg with stainless steel cladding. In the variable spectrum, heavy water reactor indifference values range from 40 to \$115/kg with zirconium clad and 90 to \$180/kg with stainless steel clad. In the boiling water reactor, values range from 50 to \$130/kg with zirconium clad and 100 to \$175/kg with stainless steel clad. In the pressurized water reactor, values ranged from 90 to \$180/kg with zirconium cladding and 70 to \$270/kg with stainless steel cladding.

FIGURE 16. Indifference Values of Fuel Re-Use Service Charge as Functions of U^{235} -Enriched Uranium Fabrication Cost: PWR Reactor Model

FIGURE 17. Indifference Values of Fuel Re-Use Service Charge as Functions of U^{235} -Enriched Uranium Fabrication Cost: BWR Reactor Model

FIGURE 18. Indifference Values of Fuel Re-Use Service Charge as Functions of U^{235} -Enriched Uranium Fabrication Cost: HWR Reactor Model

FIGURE 19. Indifference Values of Fuel Re-Use Service Charge as Functions of U^{235} -Enriched Uranium Fabrication Cost: VSHWR Reactor Model

The indifference value of irradiated blanket elements then is seen to vary from the minimum with the most neutronically efficient reactors and clads to a maximum with the least efficient neutronically efficient reactor types and clad types. In all cases except the very lowest cost in most neutronically efficient reactors, the thermal reactor operator will be able to pay a fuel service charge for irradiated blanket elements which is in excess of the probable full cost of fabricating and reprocessing those blanket elements.

It is worth noting that the highest indifference values occur in those cases where plutonium is more advantageous compared to U^{235} ; namely, in cases where there is a large parasitic absorption cross section (stainless steel cladding in light water) and in which the neutron spectrum is hardened. In both cases plutonium's large total cross section makes it a better competitor for neutrons. Since the indifference values computed are well in excess of the expected incremental costs of the fast reactor blanket cycle, it is apparent that there is a very large bargaining range for price of re-usable blanket elements. Presumably, this large spread between cost and value will result in a price at which both the thermal reactor operator and the fast blanket operator can achieve benefits and profits. Furthermore, it appears as though the largest incentive, that is, the largest spread between fuel element value and expected fuel element cost, will occur in light water reactors which are expected to be in large scale usage in the near future. Smaller, though still substantial, incentives exist for the heavy water high neutron efficiency reactor which may be introduced into the economy before the fast reactor.

FUTURE WORK

SYSTEM ANALYSES

Although the analyses included in this report, based on the thermal reactor fuel cycle, provide a reasonable assessment of incentives for fuel re-use, it is considered desirable to extend these to include the fuel cycle economics of the fast reactor blanket and of the fast-thermal reactor complex as a whole. The exploration of other applications, such as thorium fuel in the fast blanket, and combinations of thermal reactors of differing enrichment, should be considered.

It is expected that analyses of the blanket cycle will require analysis of the whole fast reactor fuel system including core, although partial analyses of the blanket cycle only may be of some value. Because of the close neutronic interrelationship between core and blanket, it seems unlikely that any perturbation in the blanket can be analyzed as a thing by itself because of its substantial feedback and effect on the core fuel cycle. This would provide a basis for comparing fuel re-use with an optimized fast reactor fuel cycle.

The next step in the analyses would be to analyze the fast reactor and thermal reactor together as a composite system, to determine fuel re-use cycles which would represent a reasonable economic benefit to both the thermal and fast reactor operator as compared with their separately optimized fuel cycles. The effect of fuel re-use on uranium mining requirements and the ability to supply fuel inventory for an expanding fast breeder reactor complex should also be studied.

Also of interest are combinations of thermal reactors operating in a fuel cascade. As an example, the boiling water reactor model of this present study fuel could be considered to use enriched uranium burned up to a nominal 15,000 to 20,000 MWd/ton then transferring this irradiated fuel to a

heavy water reactor of the type used in this study, and burning an addition 10,000 to 15,000 MWd/ton (as before, without intermediate reprocessing or fuel refabrication).

REACTOR DESIGN ANALYSES

Concurrent with the system analyses, it would be desirable to do more detailed fuel and core design studies for a specific fuel re-use scheme, to evaluate power distribution and hot-channel factors as a function of fuel segmentation, and to estimate the incremental fuel fabrication cost associated with the segmentation.

Following this, study would be extended to such items as fuel transfer systems, plant layouts, and siting of the reactors (integrated versus separate siting). Particular attention would be paid to fuel washing, disassembly and reassembly facilities, reactor transfer machines, and fuel storage area and to the cost associated with these items.

EXPERIMENT

Some consideration has been given to a demonstration experiment using a fast reactor, such as the Enrico Fermi Fast Breeder Reactor, and a thermal testing reactor such as the Plutonium Recycle Test Reactor. A preliminary experimental program outline has been prepared, feasibility investigated, and the approximate cost estimated. It was concluded that the experiment plan could be readily implemented.

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APPENDIX A
DESCRIPTION OF NUCLEAR CALCULATIONS

APPENDIX A

DESCRIPTION OF NUCLEAR CALCULATIONS

THE FAST REACTOR IRRADIATION CYCLE

For the fast reactor cycle, an initially clean fast reactor facility was postulated. The reactor core and its associated blanket and reflector were idealized to an equivalent three-region, spherical model as indicated in the text. Static, 16-group calculations were carried out for this reactor model using the HFN code^(A.1) and Yiftah, Okrent, Moldauer (YOM) cross-section set.^(A.2) The fission product absorption cross sections were taken from Reference^(A.3). The fission product nonabsorption cross sections were assumed to be adequately simulated by the nuclear properties of molybdenum. In the calculation of fission product buildup in the fast reactor blanket, it was assumed that fission products did not burn out.

Fast reactor burnup calculations were carried out by means of a zero-dimensional, one-group matrix diagonalization routine developed by B. H. Duane.^(A.4) One-group cross sections for the burnup calculations were deduced from the 16-group set. These average cross sections were characterized by the spectrum of the clean reactor, and no modifications of these cross sections during burnup was made. This constant spectrum assumption seems to be reasonably valid since the exposures in the fast reactor blanket are not very high. The change in cross sections with burnup was examined for selected exposures and the results are tabulated in Table A-1.

THE THERMAL REACTOR IRRADIATION CYCLE

For the thermal irradiation cycles, the Pacific Northwest Laboratory MELEAGER code^(A.5) was utilized. This burnout code generates cross sections based on the

Wescott routine^(A.5) [See Reference (A.7) for a partial evaluation of MELEAGER]. For the reactors under consideration in this study, a range of spectra are provided. In the HWR, the spectra is relatively soft, i.e., Wescott's spectral parameter is not unduly high for practical exposure range of interest. The BWR model exhibits a somewhat harder spectrum (r is higher), and the PWR an even harder spectrum. For the Zoned Spectrum reactor, the spectrum is allowed to vary from very hard, initially to a reasonably soft spectrum near the end of the burnup. Following is a sample printout of a MELEAGER run for the Pressurized Water Reactor model operating with Zircaloy-clad, 2.5% U^{235} enriched fuel.

TABLE A-I. *Average Microscopic Cross Sections in Barns as a Function of Fast Blanket Exposure*

The isotopic buildup in the fast reactor blanket is shown in Figure 1, page 6.

		MWd/ton		
		0	434	1359
U^{235}	σ_a	3.111	3.054	2.977
	σ_f	2.375	2.336	2.285
U^{238}	σ_a	0.405	0.397	0.388
	σ_f	0.019	0.020	0.023
Pu^{239}	σ_a	2.853	2.818	2.772
	σ_f	2.116	2.100	2.080
Pu^{240}	σ_a	0.835	0.831	0.826
	σ_f	0.167	0.178	0.194
Pu^{241}	σ_a	3.839	3.776	3.691
	σ_f	3.348	3.297	3.230
Pu^{242}	σ_a	0.832	0.828	0.823
	σ_f	0.164	0.175	0.191

KEY DEFINITIONS

The following definitions are provided as a key to the meaning of some of the important MELEAGER parameters shown in the computer printout.

INPUT DATA

I	Name of isotope
M	Sequence number of isotope formed by neutron capture
Y	Concentration of isotope, nuclei/barn-cm
SPEC	All averaged slowing down power volume ratio
TNL	Thermal nonleakage probability
TEM	Moderator temperature, °C
SNF	Nonfuel thermal absorption cross section
SCA	Epithermal effective macroscopic potential scattering cross section in the fuel
PWI	Reactor specific power adjusted by load factor, MWd/ton
DENS	Density of fuel as metal (i.e., excluding oxygen in oxides, etc.), gm/cm ³
E	(-1) times the fast effect

OUTPUTGroups at Top of Pages

SIGMAF	Fission cross section
SIGMAA	Absorption cross section
G,S	Parameters in Wescott formula
	$\sigma = \sigma_0(G + rs)$
CONCENTRATION	Nuclei/barn-cm of isotope
DESTROYED	Nuclei/barn-cm of isotope destroyed
ABS	Macroscopic cross section

ALPHA	Ratio of capture to fission cross section
HEAT	Time average percent of heat produced (not to be confused with HEAT at the bottom of pages)
POWER	Instantaneous percent of heat production
CONV	Isotope conversion ratio, percent

Groups at Bottom of Pages

TIME	Total irradiation time in seconds
HEAT	Exposure, MWd/ton
PAV	Average flux for the burnup, $\times 10^{-24}$
PHI	Flux at TIME, $\times 10^{-24}$
PWR	Reactor specific power, MW/ton
TNEUT	Neutron temperature, °C
SIGSUM	Reactor total macroscopic cross section
R	Wescott spectral index
ABSF	Total macroscopic fission cross section
ABSR	Total macroscopic absorption cross section
XMLT	K_{∞}
AMLT	Reactor Average K_{∞}
ZTA	Neutrons produced per neutron absorption in the fuel
J	Thermal utilization
EPS	Fast effect
Q	Resonance escape probability
CR	Conversion ratio

ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF
U233	7.218E 02	0.9739	1.6899	U235	5.245E 02	0.8894	0.0695	PU239	1.562E 03	1.6484	1.7969
PU241	1.531E 03	1.3862	0.4233								
ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA
U233	9.629E 02	0.9681	2.9976	U235	6.633E 02	0.9060	0.3067	PU239	2.455E 03	1.8680	2.4072
PU241	2.008E 03	1.3733	0.3906	TH232	4.385E 01	0.9672	20.7019	U238	1.186E 01	0.9870	12.2457
U234	2.383E 02	1.0177	5.3118	PU240	3.669E 03	1.1132	50.4694	PA233	3.982E 02	0.9672	28.9581
U236	2.014E 02	1.0096	138.0875	PU242	4.358E 02	1.0115	93.0222	NP237	4.746E 02	1.0524	7.2620
AM241	1.794E 03	1.3685	6.1697	AM243	6.449E 02	0.5603	12.5091	AM242	7.810E 03	0.9672	0.0386
CM242	1.953E 01	0.9672	0.0386	CM243	1.367E 03	0.9672	0.0386	CM244	1.474E 01	0.9672	0.0386
PU238	4.071E 02	0.9672	0.0386	FPA	4.881E 01	0.9672	0.0386	FPB	2.929E 02	0.9672	0.0386
FPC	7.810E 02	0.9672	0.0386	XE135	3.354E 00	1.2216	0.0487	SM149	6.193E 04	2.0215	-2.1364
SM151	6.476E 03	0.7476	-0.4242								

ISOTOPE	DESCENDANT	CONCENTRATION	DESTROYED	ABS	ABSAV	ALPHA	HEAT	POWER	CONV.
U233	U234	-0.	0.	-0.	-0.	0.33405	-0.	-0.	-0.
U235	U236	5.79425E-04	0.	0.38433	0.38433	0.26469	92.3746	92.3746	-0.
PU239	PU240	-0.	0.	-0.	-0.	0.57138	-0.	-0.	63.1576
PU241	PU242	-0.	0.	-0.	-0.	0.31167	-0.	-0.	-0.
TH232	PA233	-0.	0.	-0.	-0.				
U238	PU239	2.25976E-02	0.	0.26808	0.26808	9.62618	7.6254	7.6254	
U234	U235	-0.	0.	-0.	-0.				
PU240	PU241	-0.	0.	-0.	-0.				
PA233	U234	-0.	0.	-0.	-0.				
U236	NP237	-0.	0.	-0.	-0.				
PU242	AM243	-0.	0.	-0.	-0.				
NP237	PU238	-0.	0.	-0.	-0.				
AM241	CM242	-0.	0.	-0.	-0.				
AM243	CM244	-0.	0.	-0.	-0.				
AM242	AM243	-0.	0.	-0.	-0.				
CM242	CM243	-0.	0.	-0.	-0.				
CM243	CM244	-0.	0.	-0.	-0.				
CM244	-0.	-0.	0.	-0.	-0.				
PU238	PU239	-0.	0.	-0.	-0.				
FISSION PRODUCTS				0.02189	0.02189				

FPA =-0. FPB =-0. FPC =-0. XE135 = 4.626E-09 SM149 = 7.277E-08 SM151 = 2.884E-07
 PU239 =-0. PU240 =-0. PU241 =-0. PU242 =-0. AM241 =-0. AM242 =-0.
 AM243 =-0. CM242 =-0. CM243 =-0. CM244 =-0.

TIME/HEAT	PAV/PHI	PWR/WRAT	TNEUT/WA	PMXV/PMNV	SIGSUM/R	ABSF/ABSR	XMLT/AMLT	ZTA/J	EPS/Q	CR/FISBU	CAB/FRAB
0.	1.99587E+11	2.00000E 01	397.272	1.00000	0.44013	0.41660	1.15018	1.73907	1.05543	63.1576	0.
0.	1.99587E+11	0.	63.821	1.00000	0.23570	0.67873	1.15018	0.87785	0.71384	0.3843	35.7625

ISOTOPE	SIGMAF	SF	ISOTOPE	SIGMAF	SF	ISOTOPE	SIGMAF	SF
U233	7.770E 02	0.9657	U235	5.287E 02	0.8837	PU239	1.473E 03	1.5945
PU241	1.539E 03	1.3583						1.1665
ISOTOPE	SIGMAA	SA	ISOTOPE	SIGMAA	GA	ISOTOPE	SIGMAA	GA
U233	1.075E 03	0.9600	U235	6.800E 02	0.8997	PU239	2.321E 03	1.8018
PU241	2.010E 03	1.3447	TH232	5.353E 01	0.9594	U238	1.452E 01	0.9787
U234	2.755E 02	1.0098	PU240	2.605E 03	1.0983	PA233	4.982E 02	0.9594
U236	2.519E 02	1.0005	PU242	5.324E 02	1.0024	NP237	5.546E 02	1.0355
AM241	2.038E 03	1.3187	AM243	7.994E 02	0.5624	AM242	7.179E 03	0.9594
CM242	1.948E 01	0.9594	CM243	1.364E 03	0.9594	CM244	1.471E 01	0.9594
PU238	4.062E 02	0.9594	FPA	4.871E 01	0.9594	FP8	2.922E 02	0.9594
FPC	7.793E 02	0.9594	XE135	3.346E 06	1.2118	SM149	5.526E 04	2.0052
SM151	6.134E 03	0.7416						-2.1173

ISOTOPE	DESCENDANT	CONCENTRATION	DESTROYED	ABS	ABSAV	ALPHA	HEAT	POWER	CONV.
U233	U234	1.04722E-12	3.13488E-13	0.00000	0.00000	0.38304	0.0000	0.0000	0.0000
U235	U236	2.85305E-04	2.95008E-04	0.19401	0.27967	0.28621	59.0404	39.4096	0.0014
PU239	PU240	1.09413E-04	1.75826E-04	0.25393	0.16668	0.57608	29.5461	43.6150	59.4660
PU241	PU242	2.08290E-05	1.76244E-05	0.04200	0.01593	0.30998	3.4642	8.8511	15.9421
TH232	PA233	1.76294E-12	3.50774E-14	0.00000	0.00000				
U238	PU239	2.22801E-02	3.21037E-04	0.32343	0.30435	9.34374	7.9493	8.1243	
U234	U235	4.98322E-09	3.03290E-10	0.00000	0.00000				
PU240	PU241	2.63354E-05	3.72547E-05	0.06860	0.03532				
PA233	U234	2.17768E-13	1.40081E-12	0.00000	0.00000				
U236	NP237	5.54939E-05	8.18206E-06	0.01398	0.00776				
PU242	AM243	3.46061E-04	5.27777E-07	0.00184	0.00050				
NP237	PU238	6.81663E-04	1.44707E-06	0.00378	0.00137				
AM241	CM242	5.04200E-07	3.12681E-07	0.00103	0.00030				
AM243	CM244	4.42819E-07	7.87820E-08	0.00035	0.00007				
AM242	AM243	2.47231E-06	5.27451E-09	0.00002	0.00000				
CM242	CM243	1.61660E-07	1.16645E-07	0.00000	0.00000				
CM243	CM244	6.05424E-10	1.74272E-10	0.00000	0.00000				
CM244	PU238	7.48949E-06	1.20497E-09	0.00000	0.00000				
FISSION PRODUCTS				0.00055	0.00014				
				0.05108	0.03724				

FPA	= 2.790E-04	FP3	= 4.729E-05	FPC	= 3.852E-06	XE135	= 5.042E-09	SM149	= 1.148E-07	SM151	= 5.632E-07
PU239	= -0.	PU240	= -0.	PU241	= -0.	PU242	= -0.	AM241	= -0.	AM242	= -0.
AM243	= -0.	CM242	= -0.	CM243	= -0.	CM244	= -0.				
TIME/HEAT	PAV/PHI	PWR/KHAT	TNEUT/MA	PXV/PMW	SIGSUM/R	ABSF/ABSR	XMLT/AMLT	ZTA/J	EPS/Q	CR/FISRU	CAB/FRAB
5.95919E 07	1.77010E-11	2.00000E 01	380.498	1.12755	0.3546	0.48123	1.03033	1.72044	1.05465	73.4095	8.8722
1.37944E 04	1.71553E-11	-2.37358E-08	74.251	0.96811	0.30733	0.95185	1.08872	0.84441	0.67247	0.4920	36.5213

ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF
U233	7.873E 02	0.9645	1.6516	U235	5.310E 02	0.8833	0.1051	PU239	1.459E 03	1.5771	1.1098
PU241	1.533E 03	1.3503	0.4285								

ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA
U233	1.097E 03	0.9588	2.9429	U235	6.854E 02	0.8990	0.3473	PU239	2.303E 03	1.7840	1.5541
PU241	2.008E 03	1.3377	0.3976	U238	5.641E 01	0.9584	20.3691	U238	1.501E 01	0.9776	12.2194
U234	2.829E 02	1.0074	5.2348	PU240	2.317E 03	1.0952	22.1018	PA233	5.180E 02	0.9584	28.4845
U236	2.613E 02	0.9991	152.0186	PU242	5.022E 02	1.0009	78.9246	NP237	5.667E 02	1.0319	7.0415
AM241	2.083E 03	1.3055	6.1245	AM243	8.216E 02	0.5639	12.1240	AM242	7.793E 03	0.9584	0.0489
CM242	1.948E 01	0.9584	0.0489	CM243	1.364E 03	0.9584	0.0489	CM244	1.471E 01	0.9584	0.0489
PU238	4.062E 02	0.9584	0.0489	FPA	4.871E 01	0.9584	0.0489	FPB	2.923E 02	0.9584	0.0489
FPC	7.793E 02	0.9584	0.0489	XE135	3.347E 06	1.2105	0.0618	SM149	5.393E 04	2.0031	-2.1148
SM151	6.067E 03	0.7408	-0.4162								

ISOTOPE	DESCENDANT	CONCENTRATION	DESTROYED	ABS	ABSAV	ALPHA	HEAT	POWER	CONV.
U233	U234	4.21698E-12	2.74530E-12	0.00000	0.00000	0.39288	0.0000	0.0000	0.0000
U235	U236	1.55949E-04	4.24575E-04	0.10689	0.21884	0.29071	45.7747	22.5273	0.0040
PU239	PU240	1.26962E-04	4.21393E-04	0.29216	0.21719	0.57873	38.2215	52.1571	62.8742
PU241	PU242	3.87773E-05	7.47859E-05	0.07787	0.03678	0.30992	7.9455	17.0873	19.8794
U238	PA233	4.60288E-12	1.90585E-13	0.00000	0.00000				
U238	PU239	2.19907E-02	6.11319E-04	0.33015	0.31509	9.85393	8.0582	8.2283	
U234	U235	4.48677E-09	5.42681E-09	0.00001	0.00000				
PU240	PU241	4.12348E-05	1.11692E-04	0.09554	0.05756				
PA233	U234	5.34878E-13	7.16525E-12	0.00000	0.00000				
U236	NP237	6.98148E-05	2.28550E-05	0.01824	0.01178				
PU242	AM243	1.28709E-05	4.13395E-06	0.00646	0.00213				
NP237	PU238	1.58868E-05	7.14367E-06	0.00900	0.00368				
AM241	CM242	1.37336E-06	2.06674E-06	0.00286	0.00106				
AM243	CM244	2.97052E-06	1.17733E-06	0.00244	0.00061				
AM242	AM243	8.08440E-09	4.19992E-08	0.00006	0.00002				
CM242	CM243	6.98465E-07	1.15961E-06	0.00001	0.00000				
CM243	CM244	4.83755E-09	3.13806E-09	0.00001	0.00000				
CM244		1.13374E-06	3.60801E-08	0.00002	0.00000				
PU238	PU239	6.56223E-06	1.51089E-06	0.00267	0.00075				
FISSION PRODUCTS				0.07039	0.04808				

FPA = 5.128E-04 FPB = 8.167E-05 FPC = -6.105E-06 XE135 = 4.846E-09 SM149 = 1.185E-07 SM151 = 6.100E-07
 PU239 = -0. PU240 = -0. PU241 = -0. PU242 = -0. AM241 = -0. AM242 = -0.
 AM243 = -0. CM242 = -0. CM243 = -0. CM244 = -0.

TIME/HFAT PAV/PHI PWR/WRAT TNEUT/WA PMXV/PMNV SIGSUM/R ABSF/ABSR XMLT/AMLT ZTA/J EPS/Q CR/FISBU CAB/FRAB
 1.10316E 08 1.75872E-11 2.00000E 01 373.972 1.13484 0.54664 0.47311 0.95351 1.68577 1.05394 82.7575 3.9648
 2.55360E 04 1.78607E-11 -6.77813E-08 71.318 0.97437 0.32220 1.01306 1.03965 0.81419 0.65915 0.4806 37.5289

ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF
U233	7.896E-02	0.9643	1.6484	U235	5.318E-02	0.8833	0.1080	PU239	1.457E-03	1.5730	1.1019
PU241	1.531E-03	1.3484	0.4236								
ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA
U233	1.101E-03	0.9586	2.9376	U235	6.869E-02	0.8990	0.3506	PU239	2.300E-03	1.7780	1.5455
PU241	2.005E-03	1.3354	0.3939	TH232	5.687E-01	0.9582	20.3397	U238	1.512E-01	0.9773	12.2434
U234	2.1844E-02	1.0071	5.2253	PU240	2.260E-03	1.0944	21.2476	PA233	5.223E-02	0.9582	28.4430
U236	2.634E-02	0.9987	131.7765	PU242	4.829E-02	1.0006	74.9779	NP237	5.686E-02	1.0308	7.0030
AM241	2.092E-03	1.3010	6.1166	AM243	8.220E-02	0.5646	11.9990	AM242	7.794E-03	0.9582	0.0491
CM242	1.948E-01	0.9582	0.0491	CM243	1.364E-03	0.9582	0.0491	CM244	1.471E-01	0.9582	0.0491
PU238	4.063E-02	0.9582	0.0491	FPA	4.871E-01	0.9582	0.0491	FPB	2.923E-02	0.9582	0.0491
FPC	7.794E-02	0.9582	0.0491	XE135	3.347E-06	1.2102	0.0620	SM149	5.363E-04	2.0027	-2.1143
SM151	6.051E-03	0.7406	-0.4160								
ISOTOPE	DESCENDANT	CONCENTRATION	DESTROYED	ABS	ABSAV	ALPHA	HEAT	POWER	CONV.		
U233	U234	5.9833E-12	5.14497E-12	0.00000	0.0000	0.39472	0.0000	0.0000	0.0000		
U235	U236	1.16339E-04	4.64252E-04	0.07991	0.19008	0.29174	41.1906	17.2515	0.0065		
PU239	PU240	1.29391E-04	5.47799E-04	0.29869	0.23136	0.57869	40.9092	54.6640	64.6772		
PU241	PU242	4.39823E-05	1.12064E-04	0.08820	0.04518	0.30982	9.8105	19.8417	21.6543		
TH232	PA233	6.00371E-12	3.19181E-13	0.00000	0.00000						
U238	PU239	2.18497E-02	7.52532E-04	0.33041	0.31784	10.11579	8.0890	8.2428			
U234	U235	8.55681E-08	1.32051E-08	0.00002	0.00001						
PU240	PU241	4.51200E-05	1.53997E-04	0.10197	0.06504						
PA233	U234	6.6620E-13	1.14656E-11	0.00000	0.00000						
U236	NP237	7.08484E-05	3.07695E-05	0.01867	0.01300						
PU242	PU243	1.81167E-05	7.39082E-06	0.00875	0.00312						
NP237	PU238	1.95376E-05	1.14560E-05	0.01111	0.00484						
AM241	CM242	1.66126E-06	3.43262E-06	0.00349	0.00145						
AM243	CM244	4.87741E-04	2.54890E-06	0.00401	0.00108						
AM242	AM243	1.01592E-07	7.25871E-08	0.00008	0.00003						
CM242	CM243	9.57769E-07	2.13222E-06	0.00002	0.00001						
CM243	CM244	7.94474E-09	6.95593E-09	0.00001	0.00000						
CM244		2.44310E-04	0.84208E-08	0.00004	0.00001						
PU238	PU239	1.02315E-05	3.01074E-06	0.00415	0.00123						
FISSION PRODUCTS				0.07863	0.05287						
FPA	= 6.191E-04	FPC	= 0.325E-05	XE135	= 4.741E-09	SM149	= 1.181E-07	SM151	= 6.160E-07		
PU239	= 0.0.	PU240	= 0.0.	PU241	= 0.0.	PU242	= 0.0.	AM241	= 0.0.	AM242	= 0.0.
AM243	= 0.0.	CM242	= 0.0.	CM243	= 0.0.	CM244	= 0.0.				
TIME/HEAT	PAV/PHI	PMW/MRAT	TNEUT/WA	PMXV/PMNV	SIGSUM/R	ABSF/ABSR	XMLT/AMLT	ZTA/J	ERS/O	CR/FISRN	CAB/FRAB
1.33955E-08	1.76750E-11	2.00000E-01	372.372	1.12921	0.54818	0.46652	0.92453	1.66864	1.05360	86.3381	1.0677
3.10080E-04	1.83095E-11	-6.66752E-08	69.570	0.96953	0.32566	1.02724	1.01968	0.80073	0.65674	0.4709	37.8414
THE AVERAGE R IS 3.03422E-01											

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APPENDIX B

PLOTS OF FUEL CYCLE COSTS AS FUNCTIONS OF FUEL BURNUP

APPENDIX B

PLOTS OF FUEL CYCLE COSTS AS FUNCTIONS OF FUEL BURNUP

Appendix B contains the plots of total fuel cycle costs versus exposure which were used in determining the minimum fuel cycle costs for all of the variables including the fueling scheme (U^{235} or fuel re-use), reactor type, clad type, and fuel interest charge rate. The minimums from the curves are then used to plot the curves of minimum total fuel costs versus fuel service charge (fuel re-use) or fabrication charge (U^{235} fuel) shown in Appendix C. Tables B-I is a guide to the figures showing what combination of variables is represented on each figure.

TABLE B-I. Plots of Fuel Cycle Costs as Functions of Fuel Burnup

Figure Number	Reactor Model	Fuel Cladding Material	Fuel Charge, %	Fuel Cycle	Page Number
B-1	Pressurized Water Reactor	Zircaloy	4.75	U ²³⁵	B-4
B-2				Re-Use	
B-3				U ²³⁵	B-5
B-4				Re-Use	
B-5		Stainless Steel	10.0	U ²³⁵	B-6
B-6				Re-Use	
B-7				U ²³⁵	B-7
B-8				Re-Use	
B-9		Stainless Steel	7.0	U ²³⁵	B-8
B-10				Re-Use	
B-11				U ²³⁵	B-9
B-12				Re-Use	
B-13	Boiling Water Reactor	Zircaloy	4.75	U ²³⁵	B-10
B-14				Re-Use	
B-15				U ²³⁵	B-11
B-16				Re-Use	
B-17		Stainless Steel	10.0	U ²³⁵	B-12
B-18				Re-Use	
B-19				U ²³⁵	B-13
B-20				Re-Use	
B-21		Stainless Steel	7.0	U ²³⁵	B-14
B-22				Re-Use	
B-23				U ²³⁵	B-15
B-24				Re-Use	
B-25	Heavy Water Reactor	Zircaloy	4.75	U ²³⁵	B-16
B-26				Re-Use	
B-27				U ²³⁵	B-17
B-28				Re-Use	
B-29		Stainless Steel	10.0	U ²³⁵	B-18
B-30				Re-Use	
B-31				U ²³⁵	B-19
B-32				Re-Use	
B-33		Stainless Steel	7.0	U ²³⁵	B-20
B-34				Re-Use	
B-35				U ²³⁵	B-21
B-36				Re-Use	
B-37	Variable-Spectrum Heavy Water Reactor	Zircaloy	4.75	U ²³⁵	B-22
B-38				Re-Use	
B-39				U ²³⁵	B-23
B-40				Re-Use	
B-41		Stainless Steel	10.0	U ²³⁵	B-24
B-42				Re-Use	
B-43				U ²³⁵	B-25
B-44				Re-Use	
B-45		Stainless Steel	7.0	U ²³⁵	B-26
B-46				Re-Use	
B-47				U ²³⁵	B-27
B-48				Re-Use	

KEY TO ECONOMIC SYMBOLS AND NOTATIONS ON GRAPHS

FEFJ	The fuel element fabrication charge. This includes all charges for fabricating the fuel elements, including jacketing materials.
Δ FEFJ	A dummy fabrication charge used here to account for the fuel re-use service charge in the fuel cost calculations.
AEC or Fuel Interest	The interest on the initial value of the fuel bought from the Atomic Energy Commission, or privately, whichever the situation may be.
EXP	Fuel Exposure in thousands of MWd/ton
TFC	Total fuel cycle costs, mills/kW-hr

FIGURE B-1. U^{235} -PWR-Zr-4.75%

FIGURE B-2. Re-Use-PWR-Zr-4.75%

FIGURE B-3. U^{235} -PWR-SS-4.75%

FIGURE B-4. Re-Use-PWR-SS-4.75%

FIGURE B-5. U^{235} -PWR-SS-7.0%

FIGURE B-6. Re-Use-PWR-SS-7.0%

FIGURE B-7. U^{235} -PWR-Zr-7.0%

FIGURE B-8. Re-Use PWR-Zr-7.0%

FIGURE B-9. U^{235} -PWR-SS-10.0%

FIGURE B-10. Re-Use-PWR-SS-10.0%

FIGURE B-11. U^{235} -PWR-Zr-10.0%

FIGURE 12. Re-Use-PWR-Zr-10.0%

FIGURE B-13. U^{235} -BWR-SS-4.75%

FIGURE B-14. Re-Use-BWR-SS-4.75%

FIGURE B-15. U^{235} -BWR-Zr-4.75%

FIGURE B-16. Re-Use-BWR-Zr-4.75%

FIGURE B-17. U^{235} -BWR-SS-7.0%

FIGURE B-18. Re-Use-BWR-SS-7.0%

FIGURE B-19. U^{235} -BWR-Zr-7.0%

FIGURE B-20. Re-Use-BWR-Zr-7.0%

FIGURE B-21. U^{235} -BWR-SS-10.0%

FIGURE B-22. Re-Use BWR-SS-10.0%

FIGURE B-23. U^{235} -BWR-Zr-10.0%

FIGURE B-24. Re-Use-BWR-Zr-10.0%

FIGURE B-25. U^{235} -HWR-SS-4.75%

FIGURE B-26. Re-Use-HWR-SS-4.75%

FIGURE B-27. U^{235} -HWR-Zr-4.75%

FIGURE B-28. Re-Use-HWR-Zr-4.75%

FIGURE B-29. U^{235} -HWR-SS-7.0%

FIGURE B-30. Re-Use-HWR-SS-7.0%

FIGURE B-31. U^{235} -HWR-Zr-7.0%

FIGURE B-32. Re-Use-BWR-Zr-7.0%

FIGURE B-33. U^{235} -HWR-SS-10.0%

FIGURE B-34. Re-Use-HWR-SS-10%

FIGURE B-35. U^{235} -HWR-Zr-10.0%

FIGURE B-36. Re-Use-HWR-Zr-10%

FIGURE B-37. U^{235} -VSHWR-SS-4.75%

FIGURE B-38. Re-Use-VSHWR-SS-4.75%

FIGURE B-39. U^{235} -VSHWR-Zr-4.75%

FIGURE B-40. Re-Use-Zone-Zr-4.75%

FIGURE B-41. U^{235} -VSHWR-SS-7.0%

FIGURE B-42. Re-Use-VSHWR-SS-7.0%

FIGURE B-43. U^{235} -VSHWR-Zr-7.0%

FIGURE B-44. Re-Use-VSHWR-Zr-7.0%

FIGURE B-45. U^{235} -VSHWR-SS-10%

FIGURE B-46. Re-Use VSHWR-SS-10%

FIGURE B-47. U^{235} -VSHWR-Zr-10%

FIGURE B-48. Re-Use-VSHWR-Zr-10%

APPENDIX C
PLOTS OF MINIMUM TOTAL FUEL CYCLE COSTS
AS FUNCTIONS OF FUEL FABRICATION COST
AND FUEL SERVICE COSTS

FIGURE C-1. HWR-SS-4.75%

FIGURE C-2. HWR-Zr-4.75%

FIGURE C-3. HWR-SS-7%

FIGURE C-4. HWR-Zr-7%

FIGURE C-5. HWR-SS-10%

FIGURE C-6. HWR-Zr-10%

FIGURE C-7. PWR-Zr-4.75%

FIGURE C-8. PWR-SS-4.75%

FIGURE C-9. PWR-SS-7%

FIGURE C-10. PWR-Zr-7.0%

FIGURE C-11. PWR-SS-10%

FIGURE C-12. PWR-Zr-10%

FIGURE C-13. BWR-Zr-4.75%

FIGURE C-14. BWR-Zr-7%

FIGURE C-15. BWR-Zr-10%

FIGURE C-16. BWR-SS-4.75%

FIGURE C-17. BWR-SS-7%

FIGURE C-18. BWR-SS-10%

FIGURE C-19. Zone-Zr-4.75%

FIGURE C-20. Zone-SS-4.75%

FIGURE C-21. Zone-SS-7%

FIGURE C-22. Zone-Zr-7%

FIGURE C-23. Zone-SS-10%

FIGURE C-24. Zone-Zr-10%

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