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RESEARCH
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DEVELOPMENT
REPORT

**PRODUCTION AND INDIFFERENCE PRICING
OF TRANSURANIUM ISOTOPES**

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MARCH, 1966

BATTELLE-NORTHWEST

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PRODUCTION AND INDIFFERENCE PRICING
OF TRANSURANIUM ISOTOPES

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Mathematics Department

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ABSTRACT

This report indicates the computed rates at which Np^{237} , Pu^{238} , Am^{241} , Cm^{242} and Cm^{244} are incidentally produced by simulated power reactors. Also shown is the extent to which some power reactor fueling methods can be altered to increase the production and purity of Pu^{238} and Cm^{244} . An explanation of the indifference price concept is introduced in order to assess the effect of various Pu^{238} and Cm^{244} credits on the equitable price of the precursor, or target, isotopes. Other calculations in this study show how various credits for transuranium isotopes would affect fuel-cycle costs both for normal fueling methods and for some altered fueling methods in power reactors.

TABLE OF CONTENTS

	<u>Page</u>
List of Tables	iv
List of Figures	vi
Summary	vii
Introduction	1
Characteristics of Transuranium Heat Sources	2
Some Computer Code Considerations	4
The Production-Rate Units Used in This Study	6
Formation of Transuranium Isotopes	8
Production of Plutonium-238 via Neptunium-237	16
Production of Plutonium-238 via Curium-242	19
Production of Curium-244	22
Fueling Schemes for Increasing the Production Rate and Purity of Curium-244	25
Recycling Precursor Isotopes to Produce Plutonium-238 and Curium-244 in Power Reactors	28
Indifference Prices for Target Isotopes	33
The Total Cost of Producing Transuranium Isotopes from Target Materials	41
The Impact of Transuranium Isotopes Credits on Fuel- Cycle Costs of Power Reactors	48
The Impact of Cross Section Uncertainties on the Computed Production of Transuranium Isotopes	53
APPENDIX A. Comparison of ALTHAEA Computer Code Results with the Results of Chemical Analyses of YANKEE and VALLECITOS Fuel Rods	56
APPENDIX B. Sample Computer Printout Summary from the Report Entitled <u>Formation of Transuranium Isotopes in Power Reactors, BNWL-140 REV1</u>	60
References.	76

LIST OF TABLES

<u>Table</u>		<u>Page</u>
Summary Table I	The Effect of Transuranium Isotope Credits on the Fuel-Cycle Cost for Normal Power Reactor Fueling Methods	viii
Summary Table II	Production of Transuranium Isotopes in Power Reactors	x
I	Characteristics of Attractive Radioisotopic Heat Sources	5
II	The Computed Production and Purity of Cm^{244} From Plutonium Enriched Natural Uranium Fuel at Constant Power or at Constant Flux	7
III	Some Isotope Properties	14
IV	Relative Power ($\text{W/MW}_e\text{-yr}$) Successively Generated by a Curium Mixture Initially Containing 86 wt% Cm^{244} .	15
V	Recycle of Plutonium and Americium to Form Cm^{242} , Cm^{244} , and Pu^{238} in a PWR Using Slightly Enriched Uranium as Supplemental Enrichment	30
VI	Recycle of U^{236} and Np^{237} to Form Pu^{238} in a PWR Using Slightly Enriched Uranium	31
VII	Recycle of U^{236} and Np^{237} to Form Pu^{238} in a PWR Using Fully Enriched Uranium	32
VIII	Targets and Target Concentrations Used for Determining the Indifference Prices of Isotopes Leading to Cm^{244}	36
IX	Heat Generated by Cm^{242} Produced during the Irradiation of a Target Containing Am^{241}	47
X	The Effect of Transuranium Isotope Credits on the Fuel-Cycle Cost for Normal Power Reactor Fueling Methods	50
XI	The Effect of Transuranium Isotope Credits on Power Reactor Fuel-Cycle Costs When Normal Fueling Methods Have Been Altered to Increase the Production of Transuranium Isotopes	52
XII	The Effect of Varied Nuclear Properties on Isotopic Concentrations at Discharge	54

LIST OF TABLES (Contd.)

<u>Table</u>		<u>Page</u>
A-I	ALTHAEA Burnup Results Compared with Chemical Analysis of Yankee Fuel Rods	57
A-II	ALTHAEA Burnup Results Compared with Chemical Analysis of Vallecitos Fuel Rods	58

LIST OF FIGURES

<u>Figure</u>		<u>Page</u>
1	Heat Production of Four Transuranium Isotopes	3
2	Principal Nuclear Reactions for Higher Isotopes Produced at Typical Power-Reactor Flux Levels	9
3	Isotope Production as a Function of Fuel Exposure in a Simulated Pressurized Water Reactor	12
4	Np^{237} Production for Various Flux Ratios	18
5	Np^{237} Production for Various U^{238} Densities in a Simulated Water Reactor	20
6	Cm^{244} Production for Various Flux Ratios and Neutron Temperatures	23
7	Cm^{244} Production (in a PWR) for Various Plutonium Compositions in Natural Uranium	24
8	Cm^{244} Production From Various Fueling Schemes in a Simulated Water Reactor	27
9	Cm^{244} Purities Obtained From Various Fueling Schemes in a Simulated Water Reactor	29
10	Indifference Price of Np^{237} for a Given Pu^{238} Price	35
11	Target Indifference Prices as a Function of Cm^{244} Credit	38
12	Target Indifference Prices as a Function of Pu^{238} Credit	39
13	Indifference Price of Am^{241} as a Function of Pu^{238} Credit	40
14	The Effect of Processing Costs on the Indif- ference Price of Np^{237} Produced in a Simulated Water Reactor	42
15	The Np^{237} Target Depletion and the Pu^{238} Product Formation in a Simulated Water Reactor at a Flux Level of 2×10^{13} neutrons $\text{cm}^{-2} \text{sec}^{-1}$	45

SUMMARY

Today power reactors incidentally produce separable quantities of the transuranium isotopes Np^{237} , americium, and curium during fuel burnups. These by-products are not being recovered because their value is not enough to reduce fuel-cycle costs. But if a market is created for transuranium heat-source isotopes, such as Pu^{238} or Cm^{244} , recovery of these by-products for sale or for use as target materials would reduce fuel-cycle costs.

The fuel-cycle costs when transuranium isotopes are not recovered are given in Case No. 1 of Summary Table I for four, "normal" power-reactor fueling methods. Some idea of the fuel-cycle cost reductions when transuranium isotopes are recovered can be obtained from the other cases in Summary Table I. In Case No. 3, for example, one can see that if Np^{237} , and it alone, is worth \$100/g net, the fuel-cycle cost reduction both of a Boiling Water Reactor (BWR)* burning slightly enriched uranium and of a Pressurized Water Reactor (PWR)* burning plutonium-enriched natural uranium would be about 0.1 mill/kWhe. For a 1000 MWe power reactor, this 0.1 mill reduction would amount to an annual saving of about \$750,000. Case No. 3 also shows that a PWR burning slightly enriched uranium and a PWR recycling the plutonium formed in a previous uranium cycle would have fuel-cycle cost reductions of 0.141 and 0.161 mills/kWhe respectively, if Np^{237} , and it alone, has a net worth of \$100/g.

Case No. 9 in Summary Table I shows that a PWR burning slightly enriched uranium would have a 0.1 mill/kWhe reduction if Am^{243} is worth \$100/g and Cm^{244} is worth \$500/g. However, if plutonium is recycled, only a Cm^{244} credit of \$250 would be required to obtain a 0.1 mill/kWhe reduction. (See Case No. 10 in the PWR-Pu column of Summary Table I.)

*Although this report presents data for PWR and BWR designs, it should be noted that these designs are continually changing. Consequently, the data in this report apply to a particular stage in the evolution of power reactors (the 1000 MWe reference designs of the year 1963).

SUMMARY TABLE I

THE EFFECT OF TRANSURANIUM ISOTOPE CREDITS ON THE FUEL-CYCLE COST FOR NORMAL POWER REACTOR FUELING METHODS

Case No.	Isotope Credit, \$/g ^(a)									Fuel Cycle Cost Reduction, ^(b) Mills/kWhe			
	Np ²³⁷	Pu ²³⁹	Pu ²⁴⁰	Pu ²⁴¹	Pu ²⁴²	Am ²⁴¹	Am ²⁴³	Cm ²⁴²	Cm ²⁴⁴	BWR ^(c)	PWR ^(c)	PWR ^(d)	PWR ^(e)
												Plutonium	Pu-Nat.
										2.165 ^(g)	1.885 ^(g)	1.885 ^(g)	1.885 ^(g)
1	0	10 ^(f)	0	10 ^(f)	0	0	0	0	0	-0.000	-0.000	-0.000	-0.000
2	150	10	0	10	0	0	0	0	0	-0.147	-0.212	-0.187	-0.144
3	100	10	0	10	0	0	0	0	0	-0.098	-0.141	-0.161	-0.095
4	50	10	0	10	0	0	0	0	0	-0.049	-0.070	-0.081	-0.047
5	0	10	0	10	0	150	0	400	0	-0.042	-0.046	-0.094	-0.300
6	0	10	0	10	0	50	0	200	0	-0.016	-0.019	-0.038	-0.124
7	0	10	0	10	0	0	0	100	0	-0.004	-0.005	-0.011	-0.038
8	0	10	0	10	0	0	300	0	1000	-0.110	-0.217	-0.656	-1.497
9	0	10	0	10	0	0	100	0	500	-0.043	-0.106	-0.285	-0.687
10	0	10	0	10	0	0	0	0	250	-0.011	-0.025	-0.100	-0.290
11	150	10	0	10	0	150	300	400	1000	-0.295	-0.468	-0.980	-1.886
12	50	10	0	10	0	50	100	200	500	-0.108	-0.174	-0.400	-0.845
13	0	10	0	10	0	0	0	100	250	-0.015	-0.030	-0.110	-0.327
14	150	10	0	20	100	150	300	400	1000	-0.428	-0.658	-1.197	-1.777
15	50	10	0	15	50	50	100	200	500	-0.172	-0.251	-0.505	-0.793

- (a) All credits given for the preceding isotopes of a transuranium heat source are target indifference prices. (See Page 33.)
- (b) The fuel exposure has been optimized for each set of isotope credits, with a fabrication cost of \$100/kg uranium, a reprocessing cost of \$24/kg uranium, a shipping cost of \$6.50/kg, a thermal-to-electrical conversion of 31%, and a working capital interest rate of 10%/yr (private ownership assumed).
- (c) Pressurized water reactor fueled with slightly enriched uranium.
- (d) Pressurized water reactor recycling the plutonium formed in a previous uranium cycle. Initial composition is 63% Pu-239, 23% Pu-240, 11% Pu-241, and 3% Pu-242.
- (e) Pressurized water reactor fueled with plutonium enriched natural uranium.
- (f) Value of Pu-239 and Pu-241 as heat producers in thermal reactors.
- (g) This value is in mills/kWh_e and represents the fuel-cycle cost for Case No. 1.

Transuranium isotope production rates—measured in grams per installed MWe capacity per year (g/MWe-yr)*— are given in Summary Table II for seven different fueling schemes. Of these seven, the first three are "normal" fueling schemes; the other four are special fueling methods that would greatly increase the production of transuranium isotopes. But unless transuranium isotopes are worth between \$500 and \$1000 per gram, the fuel-cycle costs of these four fueling schemes will not be competitive with those of the first three.

Of the "normal" fueling schemes, Case 2 has less Np^{237} production than Case 1 because some of the U^{235} enrichment is replaced with plutonium (Pu^{239} , Pu^{240} , Pu^{241} , and Pu^{242}) recycled from Case 1. This plutonium enrichment also accounts for the increased production of Pu^{238} , Am^{243} , and Cm^{244} . Case 3 has a still lower Np^{237} production because the U^{235} is reduced to only that contained in natural uranium. It should be noted that the Np^{237} formed in Case 3 is mostly due to the n-2n reaction of U^{238} . And as in Case 2, so in Case 3, the higher plutonium enrichment accounts for the higher production of Pu^{238} , Am^{243} , and Cm^{244} .

Of the special fueling schemes, Case 7 would produce the most Np^{237} and Pu^{238} , and Case 5 would produce the most Am^{243} and Cm^{244} .

The estimated kilograms of transuranium isotopes to be produced in 1970 and 1975 by each of the seven fueling schemes are also indicated in Summary Table II. These estimated production amounts are based on the AEC revised estimate of a total installed capacity of 6500 Mwe for 1970 and 29,000 Mwe for 1975.** It should be noted that the estimated amounts to be produced in 1970 and 1975 will not be available until 1 to 3 years later.

* If applied to a 1000 MWe reactor, a production rate of g/MWe-yr would equal kg/yr.

** USAEC Analysis of Advanced Converters and Self-Sustaining Breeders, Washington D.C., March 1965.

SUMMARY TABLE II
THE PRODUCTION OF TRANSURANIUM ISOTOPES IN POWER REACTORS
(Fuel Exposure is 24 Mwd/kg)

Case No.	Fueling Scheme	Production Rates, g/MWe-yr ⁽¹⁾				Estimated Annual Production in 1970, Kilograms ⁽²⁾				Estimated Annual Production in 1975, Kilograms ⁽³⁾			
		Np ²³⁷	Pu ²³⁸ ⁽⁴⁾	Am ²⁴³	Cm ²⁴⁴	Np ²³⁷	Pu ²³⁸ ⁽⁴⁾	Am ²⁴³	Cm ²⁴⁴	Np ²³⁷	Pu ²³⁸ ⁽⁴⁾	Am ²⁴³	Cm ²⁴⁴
1	Slightly Enriched Uranium	17.2	0.18	2.25	0.44	111.8	1.2	14.6	2.9	498.8	5.2	70.5	12.8
2	Slightly Enriched Uranium, with Plutonium Recycled ⁽⁵⁾	14.5	0.55	11.40	3.90	94.2	3.6	74.1	25.3	420.5	15.9	330.6	113.1
3	Plutonium in Natural Uranium ⁽⁶⁾	5.6	1.23	24.70	7.59	36.4	8.0	160.5	49.3	162.4	35.7	716.3	220.1
4	Slightly Enriched Uranium, with Plutonium, Americium, and U-236 Recycled ⁽⁷⁾	27.2	0.64	12.20	4.86	176.8	4.2	79.3	31.6	778.8	18.6	353.8	140.9
5	Plutonium in Zirconium	0.4	2.00	65.90	51.73	2.6	13.0	428.3	336.2	11.6	58.0	1911.1	1500.2
6	Fully Enriched U-235 in Zirconium	48.8	24.07 ⁽⁹⁾	1.70	2.30	317.2	156.5	11.1	14.9	1415.2	698.0	43.3	66.7
7	Fully Enriched U-235, with U-236 Recycled ⁽⁸⁾	68.7	32.97 ⁽¹⁰⁾	1.73	1.98	446.5	214.3	11.2	12.9	1992.3	956.1	50.2	57.4

- (1) The load factor is 80%, the thermal-to-electrical efficiency is 0.33, the time period is one year after fuel discharge, and the amount lost in reprocessing is 5%.
- (2) Based on the AEC revised estimate of a total installed capacity of 6500 MWe (USAEC ANALYSIS OF ADVANCED CONVERTERS AND SELF-SUSTAINING BREEDERS, Washington, D. C., March 1966).
- (3) Based on the AEC revised estimate of a total installed capacity of 29,000 MWe (USAEC ANALYSIS OF ADVANCED CONVERTERS AND SELF-SUSTAINING BREEDERS, Washington, D. C., March 1966).
- (4) The Pu-238 is produced by alpha decay of Cm-242, and normally has a purity $[Pu-238/(Pu-238 + Pu-240)]$ greater than 80%.
- (5) The plutonium recycled comes from Case No. 1.
- (6) The plutonium composition is 1.1% Pu-238, 62% Pu-239, 22.8% Pu-240, 11% Pu-241, and 3.1% Pu-242.
- (7) The plutonium, americium, and U-236 recycled come from Case No. 1.
- (8) The U-236 recycled comes from Case No. 6.
- (9) The Pu-238 is 65% pure.
- (10) The Pu-238 is 67% pure.

At the present time, the design and fueling of power reactors tend to minimize the alpha (σ_c/σ_f) of U^{235} , Pu^{239} , and Pu^{241} . The reason for minimizing these alphas is to maximize the generation of heat and to minimize the production of the two parasites U^{236} and Pu^{242} per gram of U^{235} or fissile plutonium destroyed. However, if there were a market for U^{236} and Pu^{242} as precursors to Pu^{238} and Cm^{244} , respectively, it might be economical to alter the reactor fueling system so as to obtain a higher value of alpha for U^{235} , Pu^{239} , and Pu^{241} . Furthermore, if there were a sudden market for transuranium heat-source isotopes, a short term windfall situation would be created for those holding U^{236} and Pu^{242} , which at present have no market price.

Although a market for a transuranium heat-source isotope would necessarily create a market price for its precursor isotopes, or target material, this price must be known before there can be an incentive to deliberately irradiate target material in power reactors. This report explains how such a price, which is the equitable price of target material, can be adequately determined (as a first approximation). Briefly, this equitable price is defined as the target indifference price—that is, the amount of money that a reactor operator pays for a target whose product yields an income sufficient to make the fuel-cycle cost of slightly enriched uranium with the target equal to the fuel-cycle cost of the same fuel without the target. The target indifference price is also called the target break-even price, because with either type of fueling method a reactor operator breaks even on his fuel-cycle costs.

This report then shows how various market prices for Pu^{238} and Cm^{244} products would change target indifference prices. The effect of processing costs on the indifference price of a special target like Np^{237} is also shown.

All calculations in this report used the zero-dimensional burnup mode of the ALTHAEA computer code.* Appendix A indicates how well the previous results of ALTHAEA agree with the chemical analyses of Yankee and Vallecitos fuel rods. Since transuranium isotopes are produced in amounts that vary with the alpha of Pu^{239} and Pu^{241} , and at rates that vary with the resonance integral of americium, curium, U^{236} , and Np^{237} , an error analysis of these alpha and effective cross section values are included in this report.

This report complements the Battelle Northwest Report BNWL-140 REV1,** which consists mostly of computer printout summaries that indicate the yields of transuranium isotopes from 24 different fueling schemes. Appendix B contains a selection from the Battelle Report BNWL-140 REV1. For data on power reactor fission product yields, see the Battelle Northwest report BNWL-244.+ For an interesting americium management scheme to increase the power reactor yield of decay isotopes, see Lang's report.++ (See also page 21 below.) In his study, Lang used the ALTHAEA code much the same way as it was used in this study.

* E. T. Merrill, ALTHAEA (to be published), BNWL, Richland, Washington

** D. E. Deonigi. Formation of Transuranium Isotopes in Power Reactors, BNWL-140 REV1, January 1966.

+ C. A. Rohrmann, E. T. Merrill, Formation of Selected Fission Products in Power Reactors, BNWL-244, March 1966.

++ L. W. Lang. Management of Americium in Power Reactor Fuels to Optimize Production of Alpha-Emitting Isotopes, BNWL-SA-472, January 1966.

INTRODUCTION

Irradiating reactor fuel produces a wide variety of isotopes besides the classical fuel isotopes (plutonium from U^{238} and U^{235} from Th^{232}). However, after capturing a neutron, only Pu^{239} , Pu^{241} , U^{233} , and U^{235} can readily fission, producing, in turn, a wide variety of fission products, all of which are well known—because of their parasitic neutron absorption in a reactor and because of their radioactivity (mostly β and γ)—but only a few of which have the half-life, specific power, and radioactivity suitable for a practical heat device. On the other hand, Pu^{239} , Pu^{241} , U^{233} , and U^{235} do not necessarily fission after capturing a neutron: they sometimes form a new isotope which, after capturing a neutron, can also form a new isotope. Thus some successively heavier atoms are incidentally produced during the irradiation of reactor fuel, and these progeny with an atomic number greater than that of uranium are known as trans-uranium isotopes.

Some transuranium isotopes have the appropriate combination of half-life, specific power and radioactivity (predominately active alpha particles) to serve as good heat sources. But securing these isotopes from irradiated fuel and, more particularly, arranging fuel cycles in order to increase their production are still very undeveloped technologies.⁽¹⁾ For this reason, the production costs of transuranium heat-source isotopes are high, and only relatively exotic applications (such as in space) can currently justify their use.

In this report no effort has been made to estimate the cost of the special fuel-element encapsulation or separations processes required to produce and recover transuranium heat-source isotopes, the reason being that the future scale of the industry and the likely technology are unknown. In fact, there is still no industry-wide agreement on the fabrication costs of normal power-reactor fuel

elements. It is believed, however, that the associated reactor burnup costs can be computed with some relevance, and this is the task to which the present paper is mainly addressed.

More specifically, this study was undertaken (1) to compute the rate at which Pu^{238} and Cm^{244} (as well as their preceding isotopes Np^{237} , Cm^{242} , Am^{241} , and Am^{243}) are incidentally produced by simulated power reactors, (2) to compute the increase in the production rate and purity of Pu^{238} and Cm^{244} when normal power reactor fueling methods have been altered, (3) to establish equitable price relationships between the product isotopes Pu^{238} and Cm^{244} and their preceding isotopes, or targets, (4) to compute the effect of various transuranium isotope credits on the fuel-cycle costs for normal fueling methods, and (5) to compute the effect of various transuranium isotope credits on the fuel-cycle costs for special fueling methods that increase the power reactor production of transuranium heat sources.

Curium²⁴² and Am²⁴¹ are not the primary isotopes of interest in this study. They enter into the investigation because they are intermediate isotopes to Pu^{238} and, to a less extent, Cm^{244} . Figure 1 illustrates the reason for the special interest in Pu^{238} and Cm^{244} : during the first 5-year period, the heat generated by Pu^{238} drops only 3%, as against 16% for Cm^{244} and 99% for Cm^{242} ; and although the heat produced by Am^{241} remains nearly constant, this constant level is low. It is interesting to note, however, that the high initial heat output of Cm^{242} (see Table I) makes it easily comparable to Po^{210} in short-period applications (6 months to a year).

CHARACTERISTICS OF TRANSURANIUM HEAT SOURCES

In common with other isotopic heat sources, the four transuranium isotopes Pu^{238} , Cm^{244} , Cm^{242} , and Am^{241} generate valuable amounts of heat for useful periods of time. Also, their reliability

Figure 1. HEAT PRODUCTION OF FOUR TRANSURANIUM ISOTOPES

as a power source is unique. Table I indicates the major properties of these four transuranium isotopes and of five other isotopic heat sources.

Plutonium²³⁸ and Cm²⁴⁴ are particularly suitable for use in space or, in other types of remote operation: they both have half-lives that are more than several months but less than a 100 years, their specific heat is not less than 0.1 watt per gram, and the chemical separation of the product and target is economically feasible.⁽²⁾ As for radiation, Cm²⁴⁴ spontaneously emits 50 times as many neutrons as Pu²³⁸, and this may detract from the usefulness of Cm²⁴⁴ as a power source near operating personnel; pure Pu²³⁸* does not have this possible drawback because its radiation is limited mainly to alpha emission, which is easily shielded. But as regards toxicity, Cm²⁴⁴ ranks 20th, whereas Pu²³⁸ ranks 6th.⁽³⁾ Also, Pu²³⁸ has a greater critical mass problem than Cm²⁴⁴: at a density of 10 g/cm³ the critical mass of Pu²³⁸ as an unreflected sphere is about 32 kg, or 18 kW; that of Cm²⁴⁴ is about 58 kg, or 162 kW.⁽⁴⁾

SOME COMPUTER CODE CONSIDERATIONS

The computer used in this study was an IBM 7090; the computer burnup code was ALTHAEA.⁽⁵⁾ The zero-dimensional mode of this code is based on a modified four-group cross section model (employing cross shielding of epithermal resonances) that has been calibrated against many critical and burnup experiments. Appendix A discusses how well the results from ALTHAEA agree with the results of chemical analyses of Yankee and Vallecitos fuels.⁽⁶⁾

Most of the burnup calculations in this study relate to power reactor spectra (water reactors) and to slight deviations from these.

* Pu²³⁸ contaminated with Pu²³⁶, which eventually forms Thallium²⁰⁸, can lead to a gamma radiation problem.

TABLE I⁽²⁾

CHARACTERISTICS OF ATTRACTIVE RADIOISOTOPIC HEAT SOURCES

	Cobalt ⁵⁰	Strontium ⁹⁰	Cesium ¹³⁷	Promethium ¹⁴⁷	Polonium ²¹⁰	Plutonium ²³⁸	Americium ²⁴¹	Curium ²⁴²	Curium ²⁴⁴
Specific Power, Watt/g	17.4	0.95	0.42	0.33	141	0.56	0.11	120	2.8
Half-Life, Yrs.	5.3	28	30	2.7	0.38	89	458	0.45	18
Isotopic Purity, %	10	50	35	95	95	80	90	90	90
Compound Form	Metal	SrTiO ₃	Glass	Pm ₂ O ₃	Metal	PuO ₂	Metal	Cm ₂ O ₃	Cm ₂ O ₃
Density of Compound, g/cm ³	8.9	4.6	3.2	6.6	9.3	10	11.7	11.75	11.75
Active Isotope in Compound, %	10	24	16	82	95	70	90	82	82
Specific Power of Compound, Watt/g	1.7	0.23	0.067	0.27	134	0.39	0.1	98	2.3
Power Density, Watt/cm ³ of Compound	15.5	1.16	0.215	2.22	1210	3.9	1.17	1150	27
Volume for 2 kW Heat, cm ³	129	1840	9300	1120	1.65	513	1710	1.74	74
Availability Annual kW-(1967-(a))	Available	Available (67 kW)	Available (48 kW)	Available (11 kW)	Available	Limited Available	Limited Production	Potentially Available	Potentially Available
Shielding Requirement	Heavy	Heavy	Heavy	Minor	Minor	Minor	Minor	Minor ^(b)	Minor ^(b)
Biological Hazard	3 x 10 ⁻⁹	3 x 10 ⁻¹⁰	5 x 10 ⁻⁹	2 x 10 ⁻⁸	2 x 10 ⁻⁹	7 x 10 ⁻¹³	2 x 10 ⁻¹²	4 x 10 ⁻¹¹	3 x 10 ⁻¹²
Estimated Cost, \$/Watt(a)	33	19(a)	21(a)	91(a)	26,500(c)	894	200	17	357
Estimated Cost, \$/g	570	18(a)	9(a)	30(a)	188	500(c)	1820	2000	1000(c)
Curies/g	1130	142	87	914	4500	17	3.25	3310	84
Curies/watt	65	150	207	2770	32	30	30	28	30
Spontaneous Fission Half-Life, Yrs.						4.9 x 10 ¹⁰	1.4 x 10 ¹³	7.2 x 10 ⁶	1.4 x 10 ⁷

(a) From proposed Hanford Isotope Plant, HW-77770 (kW_t = thermal kilowatts).

(b) Except for shielding against neutrons.

(c) From AEC data; also "Nucleonics," March, 1963, Table I, page 63 ("projected" costs).

As regards fueling schemes, the zero-dimensional mode of the ALTHAEA code simulated graded fuel management at constant power (per kilogram of fuel) throughout the burnup. Since, however, a constant power burnup does not necessarily apply to a graded scheme, where the flux level would be nearly constant, the validity of the ALTHAEA approximation was checked by calculating the production of Cm^{244} from several graded fueling schemes at constant power and at constant flux. Table II shows that for plutonium enriched natural uranium fuel there is almost no discrepancy in the production of Cm^{244} , and only a slight difference in purity [$\text{Cm}^{244}/(\text{Cm}^{244} + \text{Cm}^{242})$ at discharge]. Because of this verification of the ALTHAEA approximation and because ALTHAEA performs burnup calculations much faster for constant power than for constant flux, the data in this paper are based on constant power burnups for graded fueling schemes.

THE PRODUCTION-RATE UNITS USED IN THIS STUDY

The isotopic production rates in this study are given in grams per megawatt electrical per year ($\text{g/MW}_e\text{-yr}$, the capacity factor being 80%) as a function of reactivity limited fuel exposure measured in thermal megawatt days per kilogram (MWd_t/kg), with the thermal-to-electrical efficiency set at 33%. These units are chosen because they seem to be the most generally used and because they can be easily converted into grams or dollars per year for a given reactor or for a total installed water reactor capacity. The unit $\text{g/MW}_e\text{-yr}$ equals kg/yr for a 1000 MW_e reactor. To obtain grams per metric ton (g/MT) or parts per million from a production rate measured in $\text{g/MW}_e\text{-yr}$, use the following formula:

$$\begin{aligned} \text{g/MT} &= \frac{(\text{g/MW}_e\text{-yr})(\text{MWd}_t/\text{kg})(1000 \text{ kg/MT})(0.333 \text{ MW}_e/\text{MW}_t)}{365 \text{ days/yr}} \\ &= (\text{g/MW}_e\text{-yr, isotopic production rate})(\text{MWd}_t/\text{kg, fuel exposure})(0.91). \end{aligned}$$

TABLE II
THE COMPUTED PRODUCTION AND PURITY OF CURIUM²⁴⁴ FROM PLUTONIUM
ENRICHED NATURAL URANIUM FUEL AT CONSTANT POWER OR AT CONSTANT FLUX

Reactivity Limited Fuel Exposure, MWdt/kg	Constant Power		Constant Flux	
	Production, g/MWe-yr	Purity, %	Production, g/MWe-yr	Purity, %
10.0	1.37	49.5	1.38	50.5
18.2	3.66	66.1	3.69	67.8
28.3	7.32	79.2	7.34	80.9

To obtain the curies produced per $\text{MW}_e\text{-yr}$, multiply the isotopic production rate in $\text{g}/\text{MW}_e\text{-yr}$ by the number of curies per gram shown in Table I on page 5.

FORMATION OF TRANSURANIUM ISOTOPES

Plutonium²³⁸, Cm²⁴⁴, Cm²⁴², and Am²⁴¹ are formed by a combination of neutron captures, beta decays, and alpha decays. Figure 2 shows the major sequences used by the computer code ALTHAEA to simulate the production of these isotopes. Examination of Figure 2 will show that not all intermediate isotopes are included. The reason: their half-lives are not long enough to appreciably affect the power reactor production of transuranium heat-source isotopes. A typical example is that of Np²³⁸. Its elimination from the computer code reduces the power reactor production rate of Pu²³⁸ less than 1%* at a normal power reactor flux of 1×10^{13} neutrons $\text{cm}^{-2} \text{sec}^{-1}$. And this 1% error is much smaller than that undoubtedly being caused by the lack of information on the resonance integral of Pu²³⁸. Similarly, the errors due to the lack of information on specific parts of the total resonance data for almost all of the transuranium isotopes will far overshadow the small errors caused by the ALTHAEA code omitting some of the intermediate isotopes produced at power reactor flux levels.

Also omitted in the ALTHAEA computer code sequence are some reactions that significantly affect the production of impurities. The Np²³⁷ n-2n reaction is one example. This reaction produces Pu²³⁶, which always accompanies chemically separated Pu²³⁸; and Pu²³⁶ (2.85 year half-life) alpha-decays into U²³² which, after several alpha decays, forms the hard gamma emitter Thallium²⁰⁸. Thus for long missions (10 years), the gamma activity of Pu²³⁸ would be increased from

* At a flux of 1×10^{15} neutrons $\text{cm}^{-2} \text{sec}^{-1}$, the error would be 34%.

Figure 2. PRINCIPAL NUCLEAR REACTIONS FOR HIGHER ISOTOPES PRODUCED AT TYPICAL POWER-REACTOR FLUX LEVELS

almost nothing to a significant amount. In theory, a computer code should include all those reactions— n - $2n$'s, n - $3n$'s, γ - n 's, etc.,—that significantly affect product purity. But in practice, such computer code coverage can not always be provided, either because too much computer calendar time would be required to account for all the important reactions in the isotopic mishmash of each element under consideration or because the relevant cross section information is lacking. In some cases, therefore, the contamination levels of transuranium isotopes can not be accurately determined except by chemical analysis of discharged fuel.

Figure 2 shows that Pu^{238} can be formed in two ways—namely, via Np^{237} or Am^{241} . In one sequence, there are three steps: U^{235} captures a neutron to form U^{236} ; U^{236} captures a neutron to form Np^{237} (Np^{237} is also formed by the U^{238} n - $2n$ reaction); and Np^{237} captures a neutron to form reasonably pure (85%) Pu^{238} , the contamination being due to the new Pu^{238} forming some Pu^{239} during the irradiation of the remaining Np^{237} . In the above sequence, however, there is one assumption—namely, Np^{237} is irradiated only after it is chemically separated from the U^{238} in the slightly enriched uranium fuel; otherwise, the U^{238} will form so much Pu^{239} that the specific heat of the Pu^{238} - Pu^{239} mixture will be tremendously reduced (about 99%) from that of reasonably pure Pu^{238} . The second way that Pu^{238} can be formed requires two steps (but only one neutron)—namely, Am^{241} captures a neutron to form Cm^{242} , and Cm^{242} alpha-decays into Pu^{238} . To obtain Am^{241} , one can allow Pu^{241} to decay either in or out of the reactor. Plutonium²⁴¹ that is separated and allowed to decay outside the reactor will form pure Am^{241} , whereas Pu^{241} decay inside the reactor during irradiation will form a mixture of Am^{241} and Am^{243} , and this mixture will contain less Am^{241} because of the neutron captures by Am^{241} . It should be noted that the rate at which Pu^{238} is formed

from Am^{241} will be limited both by the availability of plutonium with a high content of Pu^{241} and by the Pu^{241} half-life of 13 years.

In contrast to Pu^{238} , there is essentially only one way to form Cm^{244} —namely, by neutron captures in Am^{243} , which is formed by neutron captures in Pu^{242} and to a less extent, in Am^{242} .

It should also be noted, in connection with Figure 2, that the amounts of Pu^{238} and Cm^{244} produced depend on (1) the alpha (σ_c/σ_f) of U^{235} , Pu^{239} , Pu^{241} , and, to a less extent, U^{233} , and (2), also to a less extent, the n-2n reaction of U^{238} . Furthermore, the value of alpha, particularly in the case of U^{235} and Pu^{239} , is sensitive to the energy of the captured neutrons. Because a higher alpha means less fissioning and more production of transuranium isotopes, there is a possible trade-off between the two, depending on the market prices for the heat from fission in the reactor and for the heat from decay out of the reactor. Formerly, when there seemed to be little possibility of a market price for transuranium isotopes, the value of plutonium as a heat producer via fission was calculated on the basis of the upper error limits of alpha so as to avoid overstating the worth of plutonium. Now, however, a more accurate alpha must be used so as to avoid overstating the production of transuranium isotopes. Recent chemical analyses of discharged power-reactor fuels has increased the trustworthiness of the alpha values used in this study. (See Appendix A)

Figure 3 shows the computed production rates of Pu^{238} , Cm^{244} , and their immediately preceding isotopes. The computations were based on a simulated pressurized water reactor patterned after that reported by WCAP.^{*(7)} The Pu^{238} and Np^{237} curves represent the production rates from slightly enriched

* Although this report presents data for PWR and BWR designs, it should be noted that these designs are continually changing. Consequently, the data in this report apply to a particular stage in the evolution of power reactors (the 1000 MW reference designs of the year 1963).

Figure 3. ISOTOPE PRODUCTION AS A FUNCTION OF FUEL EXPOSURE IN A SIMULATED PRESSURIZED WATER REACTOR

uranium fuel, whereas those for americium and curium represent the production rates from fuel enriched with plutonium. Also, the Pu^{238} curve represents the production rate of Pu^{238} mixed with other plutonium isotopes, which, as already noted, greatly reduce the heat output of the mixture.

Curium²⁴⁴ also has a purity problem. For when plutonium (a mixture of Pu^{239} , Pu^{240} , Pu^{241} , and Pu^{242}) is irradiated, Cm^{242} is always produced with the Cm^{244} . And since Cm^{242} has an initial heat output 50 times greater than that of Cm^{244} , the heat output of Cm^{244} , which is relatively stable, will be temporarily masked if there is much Cm^{242} in the curium mixture. This purity problem, however, diminishes with time because the shorter half-life of Cm^{242} causes the percent of Cm^{244} in the mixture to increase. And unless it is needed for an immediate application, Cm^{244} that is 98% pure is considered usable, even though the heat output of the mixture will drop about 35% in the first year. The rates at which Cm^{242} and Cm^{244} are produced depend on the cross section and half-life of their preceding isotopes (Pu^{242} , Am^{241} , and Am^{243}) during the production sequence. How much these cross sections change with a change in the epithermal flux/thermal flux ratio (ϕ_1/ϕ_2) is indicated in Table III. (It should be noted that the cross sections in Table III include resonance data; if this study had used only the 2200 meter cross sections to calculate isotope production rates, the concentrations attained would have been grossly underestimated.) Table IV, on the other hand, shows the relative power successively generated by a curium mixture initially containing 86% wt. Cm^{244} .

The Battelle report titled Formation of Transuranium Isotopes in Power Reactors ⁽⁸⁾ gives detailed calculations for the yields of transuranium isotopes from 24 different power

TABLE III
SOME ISOTOPE PROPERTIES (a)

Isotope	Effective Cross Sections (b)			Half-Life Years	Alpha of Fissile Isotopes			Potential Watts per Gram of U ²³⁵ or Pu ²³⁹ Destroyed, $\phi_1/\phi_2 = 3$
	$\phi_1/\phi_2 = 1^{(c)}$	$\phi_1/\phi_2 = 3$	$\phi_1/\phi_2 = 6$		$\phi_1/\phi_2 = 1$	$\phi_1/\phi_2 = 3$	$\phi_1/\phi_2 = 6$	
Th ²³²	8.4	9.6	11.5	Stable				
U ²³⁵	602	603	597	Stable	0.19	0.22	0.25	
U ²³⁶	23	68	11.8	Stable				
U ²³⁸	5.0	9.3	14.4	Stable				
Np ²³⁷	180	221	260	Stable				
Pu ²³⁸	453	425	397	89				0.101
Pu ²³⁹	1565	1759	1914	Stable	0.50	0.53	0.55	
Pu ²⁴⁰	559	1215	1981	Stable				
Pu ²⁴¹	1602	1601	1573	13	0.37	0.38	0.39	
Pu ²⁴²	88	276	497	Stable				
Am ²⁴¹	874	1248	1605	458				
Am ²⁴²	7220	7201	7140	152				
Am ²⁴³	146	340	553	7650				
Cm ²⁴²	18	18	18	0.45				
Cm ²⁴⁴	14	14	13	18				0.267

(a) Includes resonance data.

(b) The effective cross sections include the effect of thermal flux depression which increases with an increasing flux ratio. The enrichment at each flux ratio was sufficient to attain a fuel exposure of 25 MWd/kg.

(c) ϕ_1/ϕ_2 = epithermal flux/thermal flux = flux ratio.

TABLE IV

RELATIVE POWER (W/MW_e-YEAR) SUCCESSIVELY GENERATED BY A CURIUM MIXTURE
INITIALLY CONTAINING 86% WEIGHT CURIUM-244

<u>Days After Separation</u>	<u>Curium Composition</u>		<u>Relative Power From Mixture</u>			<u>Total Power (W/MW_e-yr)</u>	<u>Number of Cm²⁴² Half-lives After Separation</u>
	<u>% Cm²⁴²</u>	<u>% Cm²⁴⁴</u>	<u>Cm²⁴²</u>	<u>Cm²⁴⁴</u>	<u>Pu²³⁸</u>		
0	13.38	86.12	142.44	20.62	0.00	143.1	0
43	11.38	88.12	118.64	20.52	0.11	139.2	0.4
206	6.42	93.58	59.33	20.17	0.38	79.8	1.4
369	3.38	96.62	29.67	19.81	0.52	50.0	2.4
532	1.75	98.25	14.84	19.47	0.59	35.9	3.4
695	0.90	99.10	7.42	19.13	0.62	26.6	4.4
858	0.46	99.54	3.71	18.80	0.63	22.5	5.4
1021	0.23	99.77	1.86	18.47	0.64	20.9	6.4

reactor fueling schemes. These include:

- (1) Slightly enriched uranium, with plutonium recycled in simulated pressurized water reactors (PWR's) and simulated boiling water reactors (BWR's)
- (2) Slightly enriched uranium, with plutonium, americium, and U^{236} recycled in a PWR
- (3) Plutonium in natural uranium for various neutron spectra
- (4) Plutonium mixed with zirconium
- (5) Fully enriched U^{235} , with U^{236} recycled
- (6) Plutonium and fully enriched U^{235} in a variable neutron spectrum reactor
- (7) Am^{241} and Np^{237} targets in a water reactor.

PRODUCTION OF Pu^{238} VIA Np^{237}

Neptunium²³⁷ is formed directly from U^{236} , which, in turn, is formed directly from U^{235} . Because the alpha of U^{235} and the cross section of U^{236} (see Table III) both become greater with an increase in epithermal flux, there are at least three ways to increase the production of Np^{237} per unit of heat generated:*

- (1) increasing the energy of the neutron spectrum either by

* Resonance Doppler broadening affects almost all isotopes. In general, the higher the isotope temperature, the greater the effective resonance cross section, and the higher the alpha value of a fissile isotope. Since this paper is concerned with power reactor fuels, the effect of temperature increases beyond those normally occurring in power reactors is not considered.

raising the neutron temperature or by tightening the lattice so as to increase the alpha of U^{235} , (2) recycling U^{236} , and (3) increasing the number of U^{235} fissions per megawatt of heat generated. Neptunium²³⁷ production via the U^{238} n-2n reaction can also be increased by increasing the fast flux, which is proportional to the U^{238} fast effect, or ϵ in the classical four factor formula.

As the curves in Figure 4 show, the Np^{237} yield greatly increases as the energy of the neutron spectrum becomes more epithermal. The curve with a fixed initial ϕ_1/ϕ_2 ratio of 1.0 represents the spectrum of a well moderated reactor; the curve with a fixed initial ϕ_1/ϕ_2 ratio of 3.0 (slightly enriched uranium fuel) represents a soft spectrum power reactor; the curve with a fixed initial ϕ_1/ϕ_2 ratio equal to 6.0 represents the harder spectrum power reactor designs.

Figure 4 also shows the Np^{237} yield when U^{236} is recycled at concentrations of 0.5 and 1.0 kg/MT. This recycling, however, usually requires blending fully enriched U^{235} with depleted fuel—a costly technique which, to be worthwhile, must be offset by Np^{237} revenues.

Particular attention must be given to the PWR and BWR* curves in Figure 4. Unlike the other curves in the figure, these curves have a variable initial ϕ_1/ϕ_2 ratio and apply to specific reactor types. The reason the initial flux ratio varies is that the geometry (rod spacing) of these reactors is constant; and, as the initial fuel enrichment in such reactors increases in order to achieve a certain exposure, the resulting increase in thermal blackness increases the initial flux ratio when the exposure changes.

* This curve is based on data in the General Electric report GEAP-4476, entitled 1000 MWe Boiling Water Reactor Plant Feasibility Study (3 vols).

Figure 4. Np^{237} PRODUCTION FOR VARIOUS FLUX RATIOS

The other curves in Figure 4 have fixed initial ϕ_1/ϕ_2 ratios because their purpose is to show how reactor types in general can increase the production rate of Np^{237} . Consequently, care must be taken when applying those curves to an individual reactor: a correction must be made to include the increase in the initial ϕ_1/ϕ_2 ratio; for when the initial fuel enrichment is increased in order to achieve a higher exposure, there is a corresponding increase in thermal blackness, which in turn increases the initial epithermal flux/thermal flux ratio. It should be noted in Figure 4, however, that the flux ratio of each curve varies (from the initial value) when the fuel blackness changes during the burnup.

Because U^{238} always accompanies U^{235} , some Pu^{239} will be produced when U^{235} is irradiated to form U^{236} ; and this Pu^{239} will account for a sizable portion of the fuel fissions. In fact, during the lifetime of a water reactor operating on slightly enriched uranium, 50% of the fissions will occur in plutonium. However, by using more highly enriched uranium, which reduces the amount of U^{238} and plutonium competing with U^{235} and U^{236} for neutrons, the production of Np^{237} can be further increased. (See Figure 5.)

Whether fully (93%)* enriched uranium is an economically attractive fuel for directly producing Pu^{238} depends on the price of Pu^{238} and Np^{237} , as will be explained later. If 65% pure Pu^{238} is acceptable, producing Pu^{238} directly from fully enriched uranium would avoid the necessity of encapsulating and irradiating Np^{237} as a special target.

PRODUCTION OF Pu^{238} VIA Cm^{242}

If plutonium, preferably with a high Pu^{241} content, is separated and allowed to beta-decay outside the reactor, the

* The remaining 7% is mostly U^{238} , from which is formed Pu^{239} at a significant rate.

Figure 5, Np^{237} PRODUCTION FOR VARIOUS U^{238} DENSITIES IN A
SIMULATED WATER REACTOR

Reactivity Limited Fuel Exposure, MWd/kg of Fuel at 90% Theoretical Density

resulting Am^{241} can be chemically separated and irradiated as a target to form Cm^{242} , which will then decay into pure Pu^{238} . This two-decay process is inherently slow, but it may be economically attractive if the price of Pu^{238} is high enough to offset the interest charge on the out-of-reactor plutonium or if the storage of plutonium for sale to fast reactor operators is considered worthwhile. (Americium²⁴¹ and Cm^{242} also have value: Am^{241} in applications where low specific power is required for a long period, and Cm^{242} where high specific power is required for a short period; renting the use of Cm^{242} as a heat source would be one way of defraying at least the inventory costs that Cm^{242} incurs while it decays into Pu^{238} .)

Plutonium²³⁸ will be contaminated if produced by a process that allows Pu^{241} to decay inside the reactor. For if the process produces Pu^{238} from a chemically separated mixture either of curium (Cm^{242} and Cm^{244}) or of americium (Am^{241} and Am^{243}), the alpha decay of Cm^{244} will cause the product to be about 40% contaminated with Pu^{240} .

In his detailed analysis of the problems associated with Pu^{238} formation via Am^{241} and Cm^{242} , Lang⁽⁹⁾ points out that the optimum Am^{241} irradiation time is shorter than the time between water-reactor discharge schedules, and that the production of Pu^{238} from Am^{241} targets would be better accomplished by using either reactors that have on-line charging or production reactors that have frequent discharge schedules. Lang also mentions that, because of losses incurred by Am^{242} fissions, the use of very high fluxes, which may be available in special production reactors, is not necessarily the best technique for irradiating Am^{241} targets.

It should be mentioned here that, unlike Lang's study⁽⁹⁾, the present study did not attempt to maximize Cm^{242} production, because the short half-life of Cm^{242} is incompatible with normal power-reactor discharge schedules.

PRODUCTION OF CURIUM²⁴⁴

Production of Cm²⁴⁴ can be increased by hardening the spectrum—that is, by raising the ϕ_1/ϕ_2 ratio. In Figure 6, the production rate of Cm²⁴⁴ is indicated for three ϕ_1/ϕ_2 ratios at neutron temperatures of 120 °C, 360 °C, and 600 °C when the fuel consists of natural uranium enriched with plutonium (63% Pu²³⁹, 23% Pu²⁴⁰, 11% Pu²⁴¹, and 3% Pu²⁴²). The increased production rate is due to two effects. One effect is the cross section increase of Pu²⁴² as the spectrum hardens. For example, Table III (page 14) shows that the effective neutron absorption cross section of Pu²⁴² is 88 barns for a ϕ_1/ϕ_2 ratio of 1, and 497 barns for a ϕ_1/ϕ_2 ratio of 6. Notice also in Table III that there is a corresponding cross section increase in Am²⁴³ and Pu²⁴⁰, whereas the other cross sections in the Cm²⁴⁴ production sequence remain almost constant. The other effect accounting for an increased Cm²⁴⁴ production rate is the increase in the alpha of Pu²³⁹ throughout this same range (1.0, 3.0, and 6.0) of ϕ_1/ϕ_2 ratios. Because of this second effect, which is due to spectrum hardening and neutron temperature increases, more Pu²³⁹ atoms will escape fission, ultimately forming more Cm²⁴⁴ per gram of plutonium destroyed. Thus not only the production rate but also the amount of Cm²⁴⁴ produced per Pu²³⁹ atom destroyed will be increased.

Observe also in Figure 6 that the higher Cm²⁴⁴ production rates are due much more to higher ϕ_1/ϕ_2 ratios than to higher neutron temperatures. But note, too, that temperature changes at low ϕ_1/ϕ_2 ratios cause a greater percent of increase in the production rate than temperature changes at high ϕ_1/ϕ_2 ratios.

The production rate of Cm²⁴⁴ can also be increased by the obvious step of augmenting the initial concentration of Pu²⁴². Figure 7 indicates that the initial plutonium concentration having the highest percent of Pu²⁴² also has the greatest production rate.

Figure 6. Cm^{244} PRODUCTION FOR VARIOUS FLUX RATIOS AND NEUTRON TEMPERATURES

Figure 7. Cm^{244} PRODUCTION (IN A PWR) FOR VARIOUS PLUTONIUM COMPOSITIONS IN NATURAL URANIUM

Each of the three curves in Figure 7 is based on a different plutonium enrichment of a natural uranium diluent in a power reactor. The plutonium enrichment with 0% Pu^{242} represents the fuel discharge from a fast reactor blanket region; the enrichment with 3% Pu^{242} represents the fuel discharge from a uranium enriched cycle in which the reactivity limited exposure is about 20 MWd/kg; and the enrichment with 10% Pu^{242} represents the fuel discharge after two or three recycles in slightly enriched uranium. If the two curves representing Cm^{244} production from initial Pu^{242} concentrations of 3% and 10% had included the americium from their previous cycle, their slopes would have been much steeper.

FUELING SCHEMES FOR INCREASING THE PRODUCTION RATE
AND PURITY OF CURIUM²⁴⁴

Calculations were made to determine whether the Cm^{244} production rate and purity could be improved by using Th^{232} instead of U^{238} as a fertile fuel. Results show that using U^{238} has a distinct advantage, especially in well moderated reactors, because the absorption cross section of U^{238} is about one-half that of Th^{232} . U^{238} , therefore, requires less enrichment than Th^{232} , and hence the greater flux with U^{238} at the same power level causes a higher Cm^{244} production rate and purity. However, this advantage diminishes if the thorium density is reduced so as to make its soft spectrum cross section equal to that of U^{238} . The advantage of U^{238} over Th^{232} also diminishes when the U^{238} cross section increases with a harder spectrum. But since U^{238} augments the initial plutonium enrichment, whereas Th^{232} does not, U^{238} will still maintain its production advantage in a harder spectrum.

Calculations were also made to determine the effect of different fueling schemes on Cm^{244} production in a simulated water

moderated power reactor. The results are shown in Figure 8. Cases 2 and 3 in the figure are based on the assumptions that plutonium is recycled and that the price of transuranium isotopes is such that there is no incentive to increase production. But if one assumes that the price of curium and americium is sufficiently high to encourage greater Cm^{244} production, Cases 4, 5, and 6 in Figure 8 would be attractive alternatives to consider, because a fuel mixture of plutonium and zirconium would not have any neutron losses to uranium in the plutonium natural-uranium fuel. Notice also in these cases that the production of Cm^{244} increases with an increase in the percent of initial Pu^{242} . Lastly, Cases 7 and 8 in Figure 8 represent fueling schemes in which a small amount of plutonium (with zirconium) is mixed with U^{235} driver fuel. These cases—the one with 3% Pu^{242} , the other with 10% Pu^{242} —are designed to produce very pure Cm^{244} very rapidly. (Plutonium has a higher cross section than U^{235} and will, therefore, burn out more quickly at a high effective flux density.)

All eight fueling schemes in Figure 8* have the same specific power because the exposures for Cases 4 through 8, which have no natural uranium, are based on grams per ton equivalent to those of natural uranium. For this reason, one can compare the eight fueling schemes and draw the following conclusions: (1) slightly enriched uranium produces Cm^{244} at the slowest rate; (2) plutonium mixed with natural uranium increases the rate of production: the more Pu^{242} in the composition, the greater the Cm^{244} production rate; (3) plutonium mixed with zirconium, instead of with natural uranium, increases the Cm^{244} production rate still more: the more Pu^{242} in the composition, the greater the Cm^{244} production rate;

* The incidental production of Cm^{242} from the eight fueling schemes represented in Figure 8 can be calculated by simply multiplying the Cm^{244} g/MW_e-yr values in Figure 8 for a given exposure by 1 minus the corresponding purities in Figure 9.

Figure 8. Cm^{244} PRODUCTION FROM VARIOUS FUELING SCHEMES IN A
SIMULATED WATER REACTOR

(4) a small amount of plutonium (with zirconium) in U^{235} driver fuel rapidly produces more and purer Cm^{244} : the more Pu^{242} in the composition, the more and the purer Cm^{244} obtained.

It is also important to know the Cm^{244} purities produced by different fueling schemes, because the Cm^{244} obtained by irradiating plutonium or americium is always contaminated with Cm^{242} , and the full amount of usable Cm^{244} is not available until almost all the Cm^{242} has decayed into Pu^{238} . (Curium-242, however, has value both because the initial heat output of its decay is very high and recoverable (for some applications) and because it is a precursor to Pu^{238} .) Figure 9 illustrates the different Cm^{244} purities obtained from the eight fueling schemes whose production rates are shown in Figure 8. Not surprisingly, the slightly enriched uranium fuel produces low-purity Cm^{244} ; and when plutonium with no Pu^{242} (Case 2) is used as an enrichment, the flux level is reduced because of the large plutonium fission cross section, the result being that the Cm^{244} purity is almost the same as that from slightly enriched uranium fuel. However, when plutonium with more Pu^{242} is used as an enrichment (Case 3), the effect of the flux-level decrease is overridden, so that the Cm^{244} purity is greater than that from slightly enriched uranium fuel; moreover, by removing the fertile material U^{238} , the purity of Cm^{244} increases still further (Cases 4, 5, and 6); and the purity is greatest in Cases 7 and 8 where only a small amount of plutonium is mixed with U^{235} . The high purities shown for Cases 7 and 8, even for low exposures of less than 10 MWd/kg, suggest that if Cm^{244} were needed for immediate use, the fastest way to obtain large and pure amounts of it would be to irradiate a fuel in which a small amount of plutonium was mixed with U^{235} . This fueling scheme would also produce significant quantities of U^{236} and Np^{237} .

RECYCLING PRECURSOR ISOTOPES TO PRODUCE Pu^{238} AND Cm^{244} IN POWER REACTORS

As shown in Tables V, VI, and VII, the production of trans-uranium heat sources can be increased by recycling the precursor isotopes. Table V shows that recycling plutonium and americium increases the production of Cm^{244} 51.5 times (from 8 to 433 g/MT). (Unlike recycling americium, recycling plutonium is economically

Figure 9. Cm^{244} PURITIES OBTAINED FROM VARIOUS FUELING SCHEMES
IN A SIMULATED WATER REACTOR

TABLE V

RECYCLE OF PLUTONIUM AND AMERICIUM TO FORM CURIUM²⁴², CURIUM²⁴⁴, AND PLUTONIUM²³⁸ IN A PWR
 USING SLIGHTLY ENRICHED URANIUM AS SUPPLEMENTAL ENRICHMENT

(Fuel Exposure of 24 MWd/kg per Cycle)

Cycle No.	Initial Fuel	Enrichment, Wt %	Isotopic Concentration g/MT (One Year After Discharge)									
			Pu ²³⁹	Pu ²⁴⁰	Pu ²⁴¹	Pu ²⁴²	Am ²⁴¹	Am ²⁴³ ^(a)	Cm ²⁴²	Cm ²⁴⁴	Pu ²³⁸ ^(b)	Pu ²³⁸ ^(c) Purity, %
1	SEU ^(d)	2.92	5518	1882	995	255	88	45	2.6	8.4	3.7	95
2	SEU with plutonium and americium recycled from cycle No. 1	3.27	6579	2831	1773	778	256	258	9.6	102.6	13.5	85
3	SEU with plutonium and americium recycled from cycle No. 2	3.57	7136	3223	2249	1217	301	537	14.5	270.1	20.3	77
4	SEU with plutonium and americium recycled from cycle No. 3	3.82	7514	3413	2498	1533	359	797	17.5	433.9	24.4	71

(a) Includes pure Am²⁴¹ formed after separation.

(b) Plutonium²³⁸ formed only from Cm²⁴² decay after fuel reprocessing (120 days after discharge), and is not recycled with other plutonium isotopes and americium.

(c) Plutonium²³⁸ purity = $\text{Pu}^{238} \times 100 / (\text{Pu}^{238} + \text{Pu}^{239} + \text{Pu}^{240} + \text{Pu}^{241} + \text{Pu}^{242})$.

(d) Slightly enriched uranium.

TABLE VI
RECYCLE OF URANIUM²³⁶ AND NEPTUNIUM²³⁷ TO FORM PLUTONIUM²³⁸ IN A PWR USING SLIGHTLY ENRICHED URANIUM
 (Fuel Exposure is 24 MWd/kg per Cycle)

Cycle No	Fuel Enrichment, Wt %	Isotope Concentrations, g/MT					
		U ²³⁶		Np ²³⁷		Pu ²³⁸ (Via Np ²³⁷ Target) (a)	
		Initial	Discharged	Discharged	Target In	Target Out	Pu ²³⁸ Via Cm ²⁴² (b)
1	2.92	0	3288	407	0	0	6.2
2	3.06	3187	6079	665	394	315	5.5
3	3.16	5896	8488	806	950	760	4.8

- (a) Plutonium²³⁸ purity is 85%
 (b) Curium²⁴² is separated 120 days after discharge; Pu²³⁸ is recovered two years after discharge.
 Plutonium²³⁸ purity is 90%. Purity = Pu²³⁸ x 100 / (Pu²³⁸ + Pu²⁴⁰).

TABLE VII

RECYCLE OF URANIUM²³⁶ AND NEPTUNIUM²³⁷ TO FORM PLUTONIUM²³⁸ IN A PWR USING FULLY ENRICHED URANIUM

(Fuel Exposure is 24 MWd/kg per Cycle)

Cycle No.	Fuel Enrich- ment, Wt %	Isotope Concentrations, g/MT						
		U ²³⁶		Np ²³⁷			Pu ²³⁸ Via	
		Initial	Discharged	Discharged	Target In	Target Out	Direct ^(a)	Target ^(b)
1	3.16	0	3863	1150	0	0	559	0
2	3.21	3747	6701	1643	1115	892	777	223
3	3.26	6500	8970	1953	2485	1988	902	497

(a) Plutonium²³⁸ purity is 65%.

(b) Plutonium²³⁸ purity is 85%.

attractive in its own right, even if it is not used to produce Cm^{244} , because plutonium can serve as a thermal reactor fuel, whereas americium is a reactor poison.) Table VI shows how much the production of Pu^{238} is increased if the U^{236} in discharged fuel is recycled and if the Np^{237} formed during the previous cycle is encapsulated and irradiated as a target. Note that recycling U^{236} increases the amount of Np^{237} by 50% (from 525 to 829 g/MT, and from 824 to 1044 g/MT) and that the amount of Pu^{238} produced from encapsulated Np^{237} is simply a function of the flux time during irradiation. Table VI shows that approximately 21% of the original Np^{237} encapsulated as a target is converted into Pu^{238} during a 24 MWd/kg burnup in a rod adjacent to the target capsule. Another way to produce Pu^{238} is to irradiate fully enriched uranium fuel, although the product Pu^{238} will be less pure than that from an Np^{237} target—65% and 85% respectively. However, Table VII shows that the production of 65% pure Pu^{238} is from 3 to 6 times greater than the production of 85% pure Pu^{238} per megawatt day of heat removed from the fuel.

INDIFFERENCE PRICES FOR TARGET ISOTOPES

In discussing Figure 7 (page 24), it was observed that when the initial plutonium composition contains a higher percentage of Pu^{242} , the Cm^{244} production rate will be greater. And, as was pointed out, the production rates in Figure 7 would have been even greater if americium from the previous cycle had been included. Consequently, if a price is assigned to Cm^{244} , a value will also exist for the americium and plutonium isotopes, particularly Pu^{242} ; and this value will reflect both the value they already have as thermal reactor fuels and that due to them as Cm^{244} producers. However, that part of the value due to them as Cm^{244} producers will depend both on their nearness to Cm^{244} in the production sequence and on the difficulty with which they are moved through the sequence.

The equitable price relationship between a product isotope price and the value of a target isotope is adequately expressed (as a first approximation) by the target indifference price. This price is the amount that a reactor operator pays for a target whose product yields a return sufficient to make the fuel-cycle cost* of slightly enriched uranium fuel with the target equal to the fuel-cycle cost* of the same fuel without the target. When a target has such a price, a reactor operator is "indifferent" costwise to the two types of fuel since both fuels have the same fuel-cycle cost; or at such a target price, a reactor operator breaks even with either type of fuel.

Figure 10 shows the target indifference price of Np^{237} for a given Pu^{238} market price. If the market price of Np^{237} is less than the Np^{237} indifference price, a reactor operator could lower his fuel-cycle cost by purchasing the target, irradiating it, and selling the product Pu^{238} . The target indifference price, therefore is the equilibrium point (equitable price) toward which the market price of a target would soon gravitate, eliminating wind-fall situations. But if the product (Pu^{238}) market price changes, so will the equilibrium point, the intersection of the two curves.

A calculation was performed to determine what effect various Cm^{244} market prices would have on the indifference prices of Cm^{244} precursor isotopes produced in a water reactor. In the 20 cases used in the calculation, the number of targets or target concentrations, or both, were varied. (See Table VIII.) The 20 cases were selected on the basis of the statistical requirements necessary to obtain a meaningful set of indifference prices from a minimum number of cases. This explains why some cases in

* The sale of targets or product material formed from slightly enriched uranium must also be included when calculating the fuel-cycle cost.

Figure 10. INDIFFERENCE PRICE OF Np^{237} FOR A GIVEN Pu^{238} PRICE

TABLE VIII*

TARGETS AND TARGET CONCENTRATIONS USED FOR DETERMINING THE
INDIFFERENCE PRICES OF ISOTOPES LEADING TO CURIUM²⁴⁴

Case No.	Isotope Target Concentrations, g/cc				
	<u>Pu²³⁹</u>	<u>Pu²⁴⁰</u>	<u>Pu²⁴¹</u>	<u>Pu²⁴²</u>	<u>Pu²⁴³</u>
1	0.1				
2	0.5				
3		0.1			
4		0.5			
5			0.1		
6			0.5		
7				0.1	
8				0.5	
9					0.1
10					0.5
11	0.05	0.05			
12	0.025	0.025			
13	0.05	0.05	0.05		
14	0.025	0.025	0.025		
15	0.05	0.05	0.05	0.05	
16	0.025	0.025	0.025	0.025	
17	0.05	0.05	0.05	0.05	0.05
18	0.025	0.025	0.025	0.025	0.025
19	0.064	0.022	0.011	0.003	
20	0.128	0.044	0.022	0.006	

*This table is based on the statistical requirements for obtaining a meaningful set of indifference prices from a minimum number of cases. This explains why some cases have concentrations of pure isotopes and some cases have isotope combinations that do not normally occur during irradiation.

Table VIII have concentrations of pure isotopes and some cases have combinations of isotopes that do not normally occur during irradiation. Furthermore, for any given Cm^{244} price in the calculation, the price of each target at each of its locations in the production sequence was adjusted by an iterative procedure until the fuel-cycle costs of all 20 cases were equal to within 0.01 mill/kWh_e of the fuel-cycle cost of slightly enriched uranium without a target. The results of the calculations are shown in Figure 11*. Notice in the figure that the indifference price of Pu^{239} remains almost constant—namely, \$10/g, which is equal to its fuel value in a thermal reactor. However, Pu^{242} , which is closer than Pu^{239} to Cm^{244} in the production sequence, acquires a credit sufficiently high to increase the value of plutonium fuel, in which it is normally a parasite that reduces the plutonium fuel value, as shown in the equation: Plutonium value = A - B (% Pu^{242} in plutonium).⁽¹¹⁾

Another calculation, based on the same method, was performed to determine the effect that different Pu^{238} credits would have on the indifference prices of Np^{237} and U^{236} . But this time, because of the shorter production sequence, only 8 cases were involved. The results of the calculation (see Figure 12) show that Np^{237} is worth from one-third to one-half the product price, and that U^{236} , normally a parasite, is worth from \$4/g to \$10/g, depending on the price assigned to Pu^{238} .

Figure 13 shows the various indifference prices of Am^{241} , another isotope leading to Pu^{238} . This curve is based on the assumptions that Am^{241} is irradiated as a target to form Cm^{242} and that all the Cm^{242} existing in the capsule at discharge decays

* All fuel-cycle costs in this study were calculated by the QUICK cost code.⁽¹⁰⁾

Figure 11. TARGET INDIFFERENCE PRICES AS A FUNCTION OF Cm^{244} CREDIT (ISOTOPES ARE PRODUCED IN A SIMULATED WATER REACTOR)

Figure 12. TARGET INDIFFERENCE PRICES AS A FUNCTION OF Pu^{238}
CREDIT (ISOTOPES ARE PRODUCED IN A SIMULATED WATER REACTOR)

Figure 13. INDIFFERENCE PRICE OF Am^{241} AS A FUNCTION OF Pu^{238} CREDIT

into Pu^{238} . For the sake of simplicity, the indifference prices represented by the curve in Figure 13 do not include any credits for Am^{243} and Cm^{244} formed during the irradiation of Am^{241} . Had these credits been included, more calculations involving interactions would have been necessary in order to relate the indifference price of Am^{241} to the market prices of Am^{243} and Cm^{244} as well.

Since this report is mainly concerned with Pu^{238} and Cm^{244} , the indifference prices presented thus far have not included various market prices for Cm^{242} as a power source. Including various market prices for this isotope would have increased the indifference prices of those isotopes that are also in the Cm^{242} production sequence.

THE TOTAL COST OF PRODUCING TRANSURANIUM ISOTOPES FROM TARGET MATERIALS

Whether power reactors or special reactors should be used to produce transuranium isotopes is a question beyond the scope of this report. It may be that maximum production would require a combination of the two types. In any event, before a power reactor operator can afford to buy and irradiate a target, the target market price must be equal to the target indifference price. And since the total cost of producing transuranium isotopes from target materials includes the processing costs (fabrication, separations, etc.), the effect of these costs on the target indifference price will now be considered. According to Figure 14, a reactor operator can afford to buy Np^{237} at \$100/g only if the Pu^{238} credit is at least \$215.00 and the Np^{237} processing costs are \$0.00.* But if the processing costs are \$10/g, then Np^{237}

* The Np^{237} curve for \$0.00 processing costs in Figure 14 is the same as the Np^{237} curve in Figure 12.

Figure 14. THE EFFECT OF PROCESSING COSTS ON THE INDIFFERENCE
PRICE OF Np^{237} PRODUCED IN A SIMULATED WATER REACTOR

at \$100/g must have a Pu^{238} credit of at least \$250/g. On the other hand, if the Pu^{238} credit is \$267/g and the processing costs are \$50/g, then the highest price a reactor operator would be willing to pay for Np^{237} is \$100/g, the indifference price of Np^{237} . And if the Pu^{238} credit is \$324/g and the processing costs are \$100/g, the indifference price of Np^{237} is again \$100/g.

The total cost of producing transuranium isotopes from target materials can, therefore, be subdivided into the target indifference price and the processing costs; but the effect of processing costs on the target indifference price must be known before a reactor operator can determine whether he can afford to produce transuranium isotopes from target materials. Apart from the processing costs, the factors that go into determining the target indifference price are (1) the cost of target depletion, (2) the product credit (which may include a product purity correction), (3) the irradiation cost, and (4) the inventory costs of target and product. The processing costs, on the other hand, are made up of (1) the encapsulation cost, (2) fuel displacement cost, (3) shipping and storage costs, (4) recovery costs, (5) out-of-reactor decay costs, and (6) interest on the working capital required. It should be noted that the processing costs in this study are parameterized because both the future scale of target material processing and the likely technology are unknown.

The Cost of Target Depletion. After the target is loaded into a reactor, the target becomes depleted at a rate proportional to its cross section and the reactor flux level. The product formed as a result of target depletion may, in turn, be depleted because of further neutron absorptions or because of radioactive decay (short half-life).

When a product has a short half life or a large cross section, short irradiations are required to maximize the production of that product. For example, Figure 15 shows the depletion of Np^{237} and the formation of Pu^{238} as a function of exposure. Since Pu^{238} has a long half life (89 years) but a large absorption cross section (400 barns), the maximum amount of Pu^{238} recoverable from a single 4-year irradiation at a flux of 2×10^{13} is 28% of the original Np^{237} target. However, if a shorter irradiation time is used—say 1 year—36% of the original Np^{237} target could be converted during four successive irradiations (three recycles) of the target. This simply points out that in producing Pu^{238} from Np^{237} at a flux of 2×10^{13} neutrons $\text{cm}^{-2} \text{sec}^{-1}$, the maximum rate of production occurs at the beginning of the irradiation.

If target material costs are high, short irradiations with a high percentage of target conversion may, with recycle, result in minimum production costs; but, at the same time, the encapsulation and recovery costs associated with recycling the target must be considered. For if these costs are high, it may be more economical to adopt long continuous irradiation. Thus a balance must be struck between the cost involved in recycling and the reduced target conversion associated with a long continuous irradiation.

Product Purity and Credit. Another factor affecting optimum operating conditions is the desired purity of the product. In the case of Np^{237} targets being converted into Pu^{238} (see Figure 15), short irradiations are desirable in order to reduce the formation of the higher plutonium isotopes. But in the production of Cm^{244} , longer irradiations are attractive because Cm^{242} , with its short half life, quickly comes to equilibrium, while the Cm^{244} concentration continually increases.

Irradiation Cost. The cost of irradiation—namely, the cost of neutrons—is essentially the cost of the additional reactor enrichment required to supply neutrons for converting the target into a

Figure 15. THE Np^{237} TARGET DEPLETION AND THE Pu^{238} PRODUCT FORMATION IN A SIMULATED WATER REACTOR AT A FLUX LEVEL OF 2×10^{13} NEUTRONS $\text{Cm}^{-2} \text{Sec}^{-1}$

product, or the cost of diverting available neutrons from one product to another. The calculations used in this study were performed in such a way that the reactor flux level and the neutron spectrum of the power reactors considered were not appreciably altered. But this surely does not represent the optimum situation for very many target-product combinations; special reactors may have more attractive neutron spectra and charge-discharge facilities than power reactors.

Inventory Cost. Both during the irradiation and during the pre- and postirradiation times, interest is paid on the investment in the target material and the product formed. As is generally known, the assessment of inventory charges is complicated by the isotopic transmutation process. In this study, the QUICK⁽¹⁰⁾ cost code accounts for these charges with significant rigor.

Encapsulation Cost. The encapsulation cost per unit of target material varies immensely with the target or product properties.⁽¹²⁾ Encapsulating targets that contain americium is one example of how a product property can affect encapsulation costs. The radiation of such targets results in a heat-transfer difficulty due to the high decay heat flux which Cm²⁴² is capable of producing. Table IX shows the heat which must be removed from a cubic centimeter of the original target containing various quantities of Am²⁴¹. The data in Table IX are based on the assumption that 10% of the americium in the original target will exist as Cm²⁴². Since 144 watts/cm³ (see Table IX) is approximately what a fuel element in a power reactor would produce during full power operation, a reactor cooling problem would exist if the reactor were to shutdown. The encapsulating problem could be solved in one of two ways: by diluting the original Am²⁴¹ concentration to within

TABLE IX

HEAT GENERATED BY CURIUM²⁴² PRODUCED DURING THE IRRADIATION OF A TARGET CONTAINING AMERICIUM²⁴¹

Watts/cm ³ From Cm ²⁴² α Decay at 10% of Original Am ²⁴¹ Target Concentration	Quantity of Am ²⁴¹ in Original Target (g/cm ³)
144	12
96	8
48	4
24	2
12	1
6	0.5

NOTE: Required Heat Removal Capacity = Watts/cm³ = grams of Am²⁴¹/cm³ $\left(\frac{0.1 \text{ g of Cm}^{242}}{\text{g of Am}^{241}} \right) \left(\frac{120 \text{ watts}}{\text{g of Cm}^{242}} \right)$

acceptable heat transfer limits, which would increase encapsulation costs, or by supplying special cooling devices.

Fuel Displacement Cost. The use of a target element that displaces a power-producing fuel rod decreases the total reactor power output by the percent of fuel volume displaced, even though some target materials will produce decay isotopes that have significant quantities of heat. However, target elements may not always require the displacement of power-producing fuel rods; target elements could be placed in existing void areas or around the fringe of the core; they may even be used as control devices if the target cross sections have an appropriate magnitude.

Recovery Costs. The separation of target and product material is a major component of processing costs, and is largely dependent on the amount of target and product processed.

Shipping and Storage Costs. Shipping costs will vary with the amount delivered, the heat removable capability required, and the radioactivity of the target and product.

Out-of-Reactor Target and Product Decay Costs. A cost is involved in the decay losses of the target and product before irradiation and during the postirradiation processing. Cm^{244} , for example, which has a 18-year half-life, decays about 4% a year. At a price of \$1000/g, the cost per kilogram would be \$40,000 a year for the 40 grams of Cm^{244} lost by decay.

THE IMPACT OF TRANSURANIUM ISOTOPE CREDITS ON FUEL-CYCLE COSTS OF POWER REACTORS

If a credit is assigned to a transuranium isotope, the fuel cycle cost of a power reactor will decrease as the production rate of the isotope increases. And if the assigned credit is high enough,

a reactor operator might be willing to alter his fuel cycle to increase the production of the isotope. However, because of the normal workings of a free market, it is unlikely that a very high credit would exist over a long period. Such a credit would probably exist only during such special cases as Government contracts for initial amounts of transuranium isotopes.

Even so, Table X shows that the impact of transuranium isotopes credits on the normal fueling methods of power reactors is not insignificant. For instance, when the market price of Np^{237} is \$100/g (see Case 3 in Table X), the fuel-cycle costs of the PWR burning slightly enriched uranium and of the PWR recycling the plutonium formed in a previous uranium cycle, are reduced by 1.41 and 1.61 mills/kWhe respectively; and the fuel-cycle costs of the other two fueling methods* are reduced by about 0.1 mill/kWhe, which, to a 1000 MW_e power plant, is equal to about \$750,000 per year. Case 9 in Table X shows that to obtain the same reduction in a slightly enriched uranium fueled reactor, Am^{243} and Cm^{244} would have to have credits of \$100/g and \$500/g, respectively. However, if plutonium is recycled (Case 10 under the PWR-Plutonium column in Table X) only a \$250/g credit for Cm^{244} and no credit for Am^{243} are required to obtain the 0.1 mill/kWhe reduction. A credit for Cm^{242} as well as for Cm^{244} would provide even more incentive to recycle plutonium, because plutonium recycle increases the production of both Cm^{242} and Cm^{244} . Taking into account the effective indifference price of a plutonium composition (Case 15 in Table IX), one can see that a \$500/g credit for Cm^{244} implies a \$50/g credit for Pu^{242} , which is now considered a poison in thermal reactors; a credit would also be given to U^{236} , a byproduct of a reactor that burns U^{235} . (See Figure 12 on page 39.)

*The BWR costs are based on data in the General Electric report GEAP-4476, entitled 1000 MWe Boiling Water Reactor Plant Feasibility Study (3 vols).

TABLE X
THE EFFECT OF TRANSURANIUM ISOTOPE CREDITS ON THE FUEL-CYCLE COST FOR NORMAL POWER REACTOR FUELING METHODS

Case No.	Isotope Credit, \$/g (a)										Fuel Cycle Cost Reduction, (b) Mills/kWhe			
	²³⁷ Np	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴¹ Pu	²⁴² Pu	²⁴² Am	²⁴³ Am	²⁴² Cm	²⁴⁴ Cm	BWR (c) (g)	FWR (c) (g)	FWR (d) Pu (g)	FWR (e) Pu-Nat, (g)
1	0	10(f)	0	10(f)	0	0	0	0	0	0	2.165	1.885	1.885	1.885
2	150	10	0	10	0	0	0	0	0	0	-0.000	-0.000	-0.000	-0.000
3	100	10	0	10	0	0	0	0	0	0	-0.147	-0.212	-0.187	-0.144
4	50	10	0	10	0	0	0	0	0	0	-0.098	-0.141	-0.161	-0.095
5	0	10	0	10	0	0	0	0	0	0	-0.049	-0.070	-0.081	-0.047
6	0	10	0	10	150	0	0	0	400	0	-0.042	-0.046	-0.094	-0.300
7	0	10	0	10	50	0	0	0	200	0	-0.016	-0.019	-0.038	-0.124
8	0	10	0	10	0	0	0	0	100	0	-0.004	-0.005	-0.011	-0.038
9	0	10	0	10	0	0	0	300	0	1000	-0.110	-0.217	-0.656	-1.497
10	0	10	0	10	0	0	0	100	0	500	-0.043	-0.106	-0.285	-0.687
11	150	10	0	10	0	0	0	0	0	250	-0.011	-0.025	-0.100	-0.290
12	50	10	0	10	150	0	0	300	400	1000	-0.295	-0.468	-0.980	-1.886
13	0	10	0	10	0	0	50	100	200	500	-0.108	-0.174	-0.400	-0.845
14	150	10	0	10	0	0	0	0	100	250	-0.015	-0.030	-0.110	-0.327
15	50	10	0	20	150	100	50	300	400	1000	-0.428	-0.658	-1.197	-1.777
				15	50	50		100	200	500	-0.172	-0.251	-0.505	-0.793

(a) All credits given for the preceding isotopes of a transuranium heat source are target indifference prices.

(b) The fuel exposure has been optimized for each set of isotope credits, with a fabrication cost of \$100/kg uranium, a reprocessing cost of \$24/kg uranium, a shipping cost of \$6.50/kg, a thermal-to-electrical conversion of 31%, and a working capital interest rate of 10%/yr (private ownership assumed).

(c) Fueled with slightly enriched uranium.

(d) Pressurized water reactor recycling the plutonium formed in a previous uranium cycle. Initial composition is 63% Pu-239, 23% Pu-240, 11% Pu-241, and 3% Pu-242.

(e) Plutonium enriched natural uranium.

(f) Value of Pu-239 and Pu-241 as heat producers in thermal reactors.

(g) This value is in mills/kWhe and represents the fuel-cycle cost for Case No. 1.

Table XI indicates the effect of various transuranium isotope credits on power reactor fuel-cycle costs when "normal" fueling methods have been altered to increase the production of transuranium isotopes. The three fueling schemes listed in Table XI were discussed earlier in connection with both Figure 5 (see page 20), where it was shown that more fully enriched uranium would produce greater amounts of Np^{237} , and Figure 8 (see page 27) where the Cm^{244} production was indicated for various fueling schemes.

It will be noted in Table XI that in Case No. 1 (where there are no credits for transuranium isotopes) the fuel-cycle costs of the three special fueling methods are definitely not competitive with the fuel-cycle costs shown in Case I of Table X for the "normal" fueling methods. And when credits are given for transuranium isotopes, only one fueling method in Table XI—that of a PWR burning plutonium in a zirconium matrix—is competitive with that of the PWR burning slightly enriched uranium indicated in Table X. Furthermore, this special fueling method is competitive in only 6 of the 15 cases listed—namely, Cases 8, 9, 11, 12, 14, and 15. Moreover, in each of these six cases the Cm^{244} credit alone is either \$500 or \$1000 per gram.

The americium and curium credits in Table XI do not apply to the last column, which shows the fuel-cycle cost reductions of a PWR burning fully enriched uranium (93% wt. U^{235}). As mentioned earlier (see page 19), this fueling scheme will produce Pu^{238} that is 65% pure, and will do so directly, without requiring Np^{237} to be encapsulated before irradiation. It should be noted in Table XI that even in Cases 11 and 14, where the Pu^{238} credit is at a maximum of \$250 per gram and causes the greatest fuel-cycle cost reduction, the fuel-cycle costs of this special PWR fueling method are still more than 1.0 mill/kWhe higher than the fuel-cycle costs of the PWR fueling method that burns slightly enriched uranium. (See Cases 11 and 14 of the next-to-last column in Table X.)

TABLE XI
THE EFFECT OF TRANSURANIUM ISOTOPE CREDITS ON POWER REACTOR FUEL-CYCLE COSTS
WHEN NORMAL FUELING METHODS HAVE BEEN ALTERED TO INCREASE THE PRODUCTION OF TRANSURANIUM ISOTOPES

Case No.	Isotope Credit, \$/g (a)										Fuel Cycle Cost Reduction, (b) Mills/kWhe (e)			
	²³⁷ Np	²³⁸ Pu (c)	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu	²⁴¹ Am	²⁴³ Am	²⁴² Cm	²⁴⁴ Cm	Pu-Zr (d)	Pu-U ²³⁵ (e)	PWR	U ²³⁵ (f)
1	0	0	10 (g)	0	10 (g)	0	0	0	0	0	2.396 (h)	3.741 (h)	3.367	3.367
2	150	0	10	0	10	0	0	0	0	0	-0.000	-0.000	-0.000	-0.000
3	100	0	10	0	10	0	0	0	0	0	-0.000	-0.124	-0.366	-0.366
4	50	0	10	0	10	0	0	0	0	0	-0.000	-0.082	-0.238	-0.238
5	0	0	10	0	10	0	0	0	0	0	-0.000	-0.041	-0.116	-0.116
6	0	0	10	0	10	0	150	0	400	0	-0.052	-0.134	-0.000	-0.000
7	0	0	10	0	10	0	50	0	200	0	-0.024	-0.046	-0.000	-0.000
8	0	0	10	0	10	0	0	0	100	0	-0.012	-0.002	-0.000	-0.000
9	0	0	10	0	10	0	0	300	0	1000	-1.560	-0.585	-0.000	-0.000
10	0	0	10	0	10	0	0	100	0	500	-0.729	-0.250	-0.000	-0.000
11	150	0	10	0	10	0	0	0	0	250	-0.335	-0.083	-0.000	-0.000
12	50	250	10	0	10	0	150	300	400	1000	-1.685	-0.852	-0.716	-0.716
13	0	80	10	0	10	0	50	100	200	500	-0.785	-0.337	-0.220	-0.220
14	150	0	10	0	10	0	0	0	100	250	-0.356	-0.085	-0.000	-0.000
15	50	250	10	0	20	100	150	300	400	1000	-3.872	-1.087	-0.716	-0.716
		80	10	0	15	50	50	100	200	500	-1.925	-0.454	-0.220	-0.220

(a) All credits given for the preceding isotopes of a transuranium heat source are target indifference prices.

(b) The fuel exposure has been optimized for each set of isotope credits, with a fabrication cost of \$100/kg uranium, a reprocessing cost of \$24/kg uranium, a shipping cost of \$6.50/kg, a thermal-to-electrical conversion of 31%, and a working capital interest rate of 10%/yr (private ownership assumed).

(c) Plutonium-238 credit applies only to the fully enriched U-235 cases in the last column.

(d) Plutonium in a zirconium matrix in a PWR.

(e) Plutonium in a zirconium matrix with fully enriched U-235 driver fuel.

(f) Fully enriched U-235 in a zirconium matrix, the americium and curium not being recovered.

(g) Value as a heat producer in thermal reactors.

(h) This value is in mills/kWhe and represents the fuel-cycle cost for Case No. 1.

THE IMPACT OF CROSS SECTION UNCERTAINTIES
ON THE COMPUTED PRODUCTION
OF TRANSURANIUM ISOTOPES

The greatest uncertainties affecting the computed production of transuranium isotopes are the alpha values of Pu^{239} and Pu^{241} as well as the resonance integrals of americium, curium, U^{236} , and Np^{237} . (The other nuclear properties of the isotopes considered in the ALTHAEA computational sequence are well enough known that their perturbation analysis can be dispensed with.) Table XII shows how variations in nuclear properties affect isotopic concentrations when the reactivity limited fuel exposures are 18 MWd/kg. Because the fuel exposures are reactivity limited, the amount of supplementary U^{235} enrichment varies in Cases 2 and 3, causing slight changes in the amount of plutonium burned. In all cases, the initial plutonium enrichment is 1.25% and consists of 63% Pu^{239} , 23% Pu^{240} , 11% Pu^{241} , and 3% Pu^{242} .

The first three cases in Table XII indicate that a variation in the alpha of plutonium has only a moderate effect on the estimated Cm^{244} produced. The reason for this is that if the estimated plutonium alpha is higher than it should be, the required reduction in U^{235} enrichment will cause more plutonium reactions, which form more Cm^{244} . On the other hand, if the estimated plutonium alpha is lower than it should be, the required increase in U^{235} enrichment will cause fewer plutonium reactions, which form less Cm^{244} .

Except for having a constant U^{235} enrichment, Cases 4 and 5 are the same as Cases 3 and 2, respectively. Comparing the Cm^{244} concentrations in Cases 4 and 5 with those in Cases 3 and 2, one finds that a constant U^{235} enrichment amplifies the effect of a 10% change in the alpha of plutonium.

TABLE XII

THE EFFECT OF VARIED NUCLEAR PROPERTIES ON ISOTOPIC CONCENTRATIONS AT DISCHARGE

(Fuel Exposure is 18 MWd/kg)

Case No.	Nuclear Property Varied	U ²³⁵ Enrichment, Wt %	Isotope Concentrations, g/MT										
			U ²³⁶	Np ²³⁷	Pu ²³⁸	Pu ²³⁹	Pu ²⁴⁰	Pu ²⁴¹	Pu ²⁴²	Am ²⁴¹	Am ²⁴³	Cm ²⁴²	Cm ²⁴⁴
1	None	2.94	1279	239	191	7109	3991	2397	968	150	245	36	62
2	Alpha of Pu ²³⁹ and Pu ²⁴¹ decreased 10% (a)	2.34	903	226	189	6670	3819	2263	960	138	253	37	67
3	Alpha of Pu ²³⁹ and Pu ²⁴¹ increased 10% (a)	3.53	1651	261	198	7421	4140	2519	1002	163	252	37	64
4	Alpha of Pu ²³⁹ and Pu ²⁴¹ increased 10% (b)	2.94	1306	247	193	9070	4237	2471	1067	150	267	37	68
5	Alpha of Pu ²³⁹ and Pu ²⁴¹ decreased 10% (b)	2.94	1253	232	190	7152	3749	2324	877	149	226	35	56
6	Cross section of Am ²⁴¹ and Am ²⁴³ increased 20%	2.94	1279	239	201	7113	3991	2398	967	135	235	42	71
7	Cross section of Am ²⁴¹ and Am ²⁴³ increased 50%	2.94	1279	239	214	7117	3991	2399	965	116	219	49	86
8	Cross section of Cm ²⁴² and Cm ²⁴⁴ increased 100%	2.94	1279	239	191	7109	3991	2397	968	150	245	36	61
9	Cross section of Cm ²⁴² and Cm ²⁴⁴ increased 400%	2.94	1279	239	191	7109	3991	2397	968	150	245	36	61
10	Cross section of U ²³⁶ increased by 20%	2.94	1262	256	192	7113	3991	2398	968	150	246	36	62
11	Cross section of Np ²³⁷ increased by 20%	2.94	1279	234	196	7113	3991	2398	968	150	246	36	62

- (a) With U²³⁵ enrichment adjusted to the desired reactivity limited exposure.
 (b) With U²³⁵ enrichment held constant.

The remaining cases in Table XII show how the production of isotopes would be affected if additional resonances were present. Cases 6 and 7 show that when the americium cross sections are increased by 20% and 50%, the curium concentrations increase about 16% and 37%, respectively. In Cases 8 and 9, both the Cm^{242} and the Cm^{244} cross sections are arbitrarily increased 100% and 400% so as to reflect possible uncertainties in the resonance data. Case 8 shows that a 100% increase in the curium cross sections reduces the Cm^{244} concentration only 2% and the Cm^{242} concentration even less. Case 9 shows that a 400% increase in the curium cross sections has the same effect as the 100% increase in Case 8. In the final two cases are shown the changes in the estimated Np^{237} concentrations when the U^{236} or Np^{237} cross sections are increased by 20%. In Case 10, the effect of the larger U^{236} cross section is to increase the Np^{237} concentration by 7%. But in Case 11, the effect of the larger Np^{237} cross section is to reduce the Np^{237} concentration by 3%.

APPENDIX A

COMPARISON OF ALTHAEA COMPUTER CODE RESULTS
WITH THE RESULTS OF CHEMICAL ANALYSES
OF YANKEE AND VALLECITOS FUEL RODS

APPENDIX ACOMPARISON OF ALTHAEA COMPUTER CODE RESULTS
WITH THE RESULTS OF CHEMICAL ANALYSES
OF YANKEE AND VALLECITOS FUEL RODS

Tables A-I and A-II show how well the results from the ALTHAEA computer code agree with the chemical analyses of Yankee and Vallecitos fuel rods.⁽⁶⁾ The Yankee data in Table A-I relate to four separate rods, two of which were in the center of a fuel assembly and two near a water gap. The Vallecitos data in Table A-II relate to three different points along a single rod. Since the exact neutron spectrum was not known for these data, the following procedure was used for calculating the neutron spectrum in the ALTHAEA code:

(1) the burnup was terminated when the U^{235} concentration in ALTHAEA agreed with the U^{235} concentration in the chemical analyses, and (2) the coolant density was adjusted so as to produce a plutonium concentration that agreed with the chemical analyses. (Adjusting the coolant density in this way allowed the resonance absorption in U^{238} to be used as a spectrum indicator.)

As shown in Table A-I, the fuel exposures in ALTHAEA fall within the error limits of the exposures determined by cesium analysis. Also, the relative plutonium isotopic composition agrees to within 1%. Of the higher isotopes, Cm^{244} shows better agreement than Am^{241} and Cm^{242} , both of which are strongly affected by the power level during the burnup history—which is not well known. As for Np^{237} production, the ALTHAEA results substantially agree with chemical analysis of the Yankee rods only when the U^{236} concentration is in substantial agreement. According to calculations both by Pacific Northwest Laboratory and by Westinghouse,

TABLE A-I

ALITHAEA BURNUP RESULTS COMPARED WITH CHEMICAL ANALYSIS OF YANKEE FUEL RODS

Sample (rod)	Y-5	ALITHAEA	Y-2	ALITHAEA	Y-3	ALITHAEA	Y-1	ALITHAEA
	13,700	13,200	17,700	16,000	24,300	22,400	31,800	31,000
	(1)	(2)	(1)	(2)	(1)	(2)	(1)	(2)
	Burnup, MWd/MT							
	Concentration, g/MT							
U ²³⁵	21,800	22,083	19,900	19,795	16,100	15,900	11,100	11,175
U ²³⁶	2,530	2,546	2,230	2,678	3,650	3,653	3,690	4,049
Np ²³⁷	242	218	205	233	455	460	473	542
Pu ²³⁸	33	24	32	26	90	81	113	117
Pu ²³⁹	5,740	5,687	5,620	5,517	7,490	7,380	6,350	6,306
Pu ²⁴⁰	927	977	1,240	1,183	1,700	1,840	2,160	2,307
Pu ²⁴¹	468	473	599	556	1,110	1,156	1,300	1,280
Pu ²⁴²	51	53	96	85	200	200	390	384
Am ²⁴¹	69	82	85	93	119	122	139	141
Am ²⁴³	--	5.8	--	8.9	---	39	---	81
Cm ²⁴²	0.03	0.03	0.03	0.04	1.0	1.2	1.6	2.0
Cm ²⁴⁴	0.55	0.55	0.89	0.50	7.9	7.6	16.0	17.0

(1) By cesium analysis
 (2) By matching U²³⁵ depletion (with MeV per fission of 194 for U²³⁶, 201 for Pu²³⁹, and 205 for Pu²⁴¹)

TABLE A-II
AIITHAEA BURNUP RESULTS COMPARED WITH CHEMICAL ANALYSIS OF VALLECITOS FUEL RODS

Sample (rod)	V-55	AIITHAEA	V-57	AIITHAEA	V-60	AIITHAEA
	Burnup, MWd/MT					
17, 400(1)		16, 070(2)	9, 900(1)	9, 500(2)	6, 500(1)	6, 026(2)
Concentration, g/MT						
U ²³⁵	19,000.0	18,432	25,000.0	24,500.0	28,500.0	28,077.0
U ²³⁶	2,890.0	2,736	1,830.0	1,767.0	1,230.0	1,167.0
Np ²³⁷	140.0	123	47.0	63.0	22.0	32.0
Pu ²³⁸	17.5	17	5.1	4.1	1.42	1.2
Pu ²³⁹	3,330.0	3,330	2,370.0	2,721.0	1,961.0	2,002.0
Pu ²⁴⁰	880.0	880	420.0	420.0	182.0	201.0
Pu ²⁴¹	329.0	~277	121.0	96.0	35.0	35.0
Pu ²⁴²	57.0	~49	7.6	8.6	1.9	1.8
Am ²⁴¹	45.0	43	15.0	14.9	5.6	4.2
Am ²⁴³	-----	--	-----	0.29	---	0.04
Cm ²⁴²	0.39	0.29	0.066	0.07	0.18	0.01
Cm ²⁴⁴	0.48	0.12	0.72	0.01	0.012	0.00

(1) By cesium analysis

(2) By matching U²³⁵ depletion (with MeV per fission of 194 for U²³⁶, 201 for Pu²³⁹, and 205 for Pu²⁴¹)

the U^{236} concentration found by chemical analysis should be higher when the ALTHAEA results differ with the Yankee data. Analysis of the three isotopes U^{236} , Np^{237} , and Pu^{238} , each by a different method, has also verified that the alpha for U^{235} was abnormally low in the Y1 and Y2 fuel rod samples. This phenomenon has been observed in previous Yankee rod analyses, and is possibly due to local shielding of the U^{235} resonance.

The reason the ALTHAEA values in Table A-II show a lower production of transuranium isotopes than the Vallecitos data is that these data are for a much softer spectrum than those either of Yankee or of recent BWR designs. On the other hand, the production of Np^{237} indicated by ALTHAEA for low exposures is higher than that found by chemical analysis. The reason may be that in the ALTHAEA code the production of Np^{237} is primarily due to n-2n reactions (60%), which may be too high for the Vallecitos spectrum. At higher exposures, however, where the n-2n reactions are not of much account in Np^{237} production, the difference between ALTHAEA and the chemical analysis is small. Lastly, the discrepancy in curium values is due to the fact that the chemical analysis was in error for the BWR cases.⁽¹³⁾

APPENDIX B

SAMPLE COMPUTER PRINTOUT SUMMARY
FROM THE REPORT ENTITLED
FORMATION OF TRANSURANIUM ISOTOPES IN POWER REACTORS,
BNWL-140 REV1

GRADED BURNUP

*** FUEL PROPERTIES ***

CASE NUMBER 1000

PLUTONIUM RECYCLE IN PRESSURIZED WATER REACTOR

EXPOSURE, MWD/Y	24686.0	INITIAL FLUX, N/SQ CM*SEC	1.63E 13	FISSILE RATIO	0.60	U235 IN TAILS/U235 ENR	0.42				
POWER DENS, MW/T	25.00	AVG FLUX, NEUT/SQ CM*SEC	1.56E 13	INITIAL CONV RATIO	0.59	KG NATURAL/KG PRODUCED	5.883				
RESIDENCE TIME, YRS	3.38	MAX FLUX/MIN FLUX	1.09	NONFUEL X-SECT 1/CM	0.065	KG NAT U/MWE-YR	100.251				
TOTAL ENRICHMENT	2.92	AVG NEUTRON TEMP, C	387.	FUEL DENSITY	9.26	AVERAGE R	0.251				
LOAD FACTOR	0.80	THERMAL TO ELECTRICAL EFF.	0.33								
KW/GRAMS FISSILE	0.69	WATTS/CC	185.12	DT (NO LOSSES)	0.	DT (AEC LOSSES)	0.				
INITIAL REACTIVITY	1.204	FINAL INTEGRATED REACTIVITY	1.083	FINAL REACTIVITY	1.000	REACTIVITY CHANGE/MWD, MK	0.0083				
ISOTOPES	TH232	U233	PA233	U234	U235	U236	U238	PU239	PU240	PU241	PU242
MICROSCOPIC CROSS SECTION											
FISSION-INITIAL	0.	681	0	0	476	0	0.	1398	0	1210	0
AVERAGE	0.	701	0	0	483	0	0.	1199	0	1137	0
FINAL	0.	706	0	0	492	0	0.	1154	0	1116	0
ABSORPTION-INITIAL	42.19	767	276	182	587	130	12.09	2133	2909	1706	414
AVERAGE	46.60	792	305	196	602	113	13.28	1875	1838	1575	458
FINAL	47.57	798	312	200	617	101	13.49	1797	1271	1542	456
MACROSCOPIC CROSS SECTION											
FISSION-INITIAL	0.	0.	0.	0.	0.329	0.	0.	0.	0.	0.	0.
AVERAGE	0.	0.000	0.	0.	0.213	0.	0.	0.112	0.	0.012	0.
FINAL	0.	0.000	0.	0.	0.125	0.	0.	0.159	0.	0.030	0.
ABSORPTION-INITIAL	0.	0.	0.	0.	0.406	0.	0.275	0.	0.	0.	0.
AVERAGE	0.000	0.000	0.000	0.000	0.266	0.005	0.299	0.175	0.034	0.017	0.001
FINAL	0.000	0.000	0.000	0.000	0.156	0.008	0.300	0.248	0.062	0.041	0.003
ALPHA	0.	0.126	0.	0.	0.233	0.	0.	0.594	0.	0.403	0.
AVERAGE	0.	0.130	0.	0.	0.246	0.	0.	0.564	0.	0.384	0.
FINAL	0.	0.130	0.	0.	0.254	0.	0.	0.557	0.	0.382	0.
PERCENT OF TOTAL FISSION	0.	0.0	0.	0.	56.5	0.	10.2	30.0	0.	3.3	0.
ETA	0.	2.21	0.	0.	1.95	0.	0.	1.85	0.	2.19	0.
MWD/GRAM DESTROYED	0.	0.	0.	0.	1.47	0.	1.27	0.	0.	0.	0.
INITIAL CONVERSION RATIO	0.	0.	0.	0.	0.	0.	0.	0.591	0.	0.	0.

GRADED BURNUP *** FUEL PROPERTIES *** CASE NUMBER 1000

ISOTOPES	TH232	U233	PA233	U234	U235	U236	U238	PU239	PU240	PU241	PU242
PERCENT OF ENRICHMENT											
INITIAL	0.	-0.	0.	0.	2.92	0.	0.	-0.	0.	-0.	0.
FINAL	0.	0.00	0.	0.	1.07	0.	0.	0.59	0.	0.12	0.
PERCENT OF ELEMENT COMPOSITION											
INITIAL	0.	-0.	-0.	-0.	2.9	-0.	97.1	0.	0.	0.	0.
FINAL	100.0	0.0	100.0	0.0	1.1	0.3	98.5	62.6	22.2	12.2	3.0
ISOTOPIC DENSITY GM/CC											
INITIAL	-0.	-0.	-0.	-0.	0.2698	-0.	8.9897	-0.	-0.	-0.	-0.
FINAL	0.0000	0.0000	0.0000	0.0000	0.0989	0.0304	8.7921	0.0547	0.0194	0.0107	0.0026
NET ISOTOPIIC PRODUCTION											
GM/TON-YR	0.0	0.0	0.0	0.1	-5460.2	972.5	-6312.7	1747.1	620.9	342.8	84.6
GM/MWE-YR	0.0	0.0	0.0	0.0	-819.0	145.9	-946.9	262.1	93.1	51.4	12.7
GM/MWH	0.0000	0.0000	0.0000	0.0000	-0.0935	0.0167	-0.1081	0.0299	0.0106	0.0059	0.0014
GM/TON	0.0	0.0	0.0	0.5	-18463.9	3288.5	-21346.6	5907.9	2099.6	1159.0	286.0
GROSS RECOVERABLE ISOTOPES											
GM/TON-YR	0.	0.0	0.0	0.1	3160.2	972.5	280916.	1747.1	620.9	342.8	84.6
GM/MWE-YR	0.	0.0	0.0	0.0	474.0	145.9	42136.	262.	93.	51.	13.
GM/TON	0.	0.0	0.0	0.5	10686.3	3288.5	949925.	5907.9	2099.6	1159.0	286.0

GRADED BURNUP		NEUTRON BALANCE					CASE NO, 1000	
PLUTONIUM RECYCLE IN PRESSURIZED WATER REACTOR								
EXPOSURE, MWD/TON	24686.0	INITIAL ENRICH.	2.92	POWER DENS, MW/T	25.0	AV. NEUTRON TEMPERATURE, C	387,	
RESIDENCE TIME, YRS	3.38	AVERAGE R	0.251	KW/GRAMS FISSION	0.69	AVERAGE FLUX, NEUT/SQ CM-SEC	1.56E 13	
BASIS-100 NEUTRONS ABSORBED IN FISSION MATERIAL								
FISSION NEUTRONS FORMED		FUEL INITIAL	REACTOR AVERAGE	FUEL FINAL	NON-FISSION CAPTURE	FUEL INITIAL	REACTOR AVERAGE	FUEL FINAL
FROM U-233		-0.	0.00	0.00	FERTILE CAPTURE IN TH-232 IN U-234 IN U-238 IN PU-240 TOTAL	-0.	0.00	0.00
FROM U-235		197.37	113.42	68.16		-0.	0.00	0.00
FROM PU-239		-0.	70.61	103.01		59.98	59.43	52.21
FROM PU-241		-0.	8.03	20.36		-0.	6.72	10.78
TOTAL		197.37	192.06	191.52		59.98	66.15	62.99
NET FROM U-238		14.15	13.67	13.53				
NET FROM U-238		0.	0.	0.				
TOTAL		211.53	205.73	205.06				
NEUTRONS ABSORBED IN FISSION ISOTOPES								
TO CAUSE FISSION								
IN U-233		-0.	0.00	0.00	NON-FERTILE CAPTURE IN U-236 IN PU-242 IN SPECIAL PRODUCTS TOTAL	-0.	0.98	1.36
IN U-235		81.09	46.65	28.00		-0.	0.17	0.52
IN PU-239		-0.	24.51	35.70		-0.	0.38	0.96
IN PU-241		-0.	2.65	6.72		-0.	1.53	2.84
TOTAL		81.09	73.82	70.43				
AS CAPTURE FORMING FERTILE ISOTOPES								
IN U-233		-0.	0.00	0.00	NON-FUEL ABSORPTION + LEAKAGE IN SM + XE IN OTHER F.P. IN STRUCT, + MOD. + IMPUR. CONTROL + LEAKAGE TOTAL	4.60	4.20	3.67
IN PU-239		-0.	13.76	19.89		0.	9.42	12.28
IN PU-241		-0.	13.76	19.89		9.51	8.67	7.58
TOTAL		-0.	13.76	19.89		16.21	15.77	15.72
AS CAPTURE FORMING NON-FERTILE ISOTOPES								
IN U-235		18.91	11.41	7.11	NEUTRONS INTO OR FROM OTHER FUEL IN PA-233 NON-FERTILE CAPTURE IN NP-239 NON-FERTILE CAPTURE TOTAL ABSORPTION + LEAKAGE	21.23	0.	-0.03
IN PU-241		-0.	1.01	2.56		-0.	0.00	0.00
TOTAL		18.91	12.43	9.68		0.	0.	0.
TOTAL FISSION ABSORPTION		100.00	100.00	100.00		211.53	205.73	205.06
TOTAL FISSION ABSORPTION		100.00	100.00	100.00	FERTILE CAPTURE + (PA-233 + NP-239) NON-FERTILE CAPTURE		59.98	66.15
TOTAL FISSION ABSORPTION		100.00	100.00	100.00			59.98	62.99

EXPOSURE		5000.0 MHD/T		*** PROPERTIES OF SPECIAL PRODUCTS ***										CASE NUMBER 1000	
ISOTOPES		NP237	AM241	AM243	AM242	CM242	CM243	CM244	PU238	FP	-0				
HALF LIFE		STABLE	458 YR	6760YR	152 YR	163 DY	32 YR	17.6YR	89 YR						
MICROSCOPIC CROSS SECTION															
ABSORPTION-INITIAL		245	1553	476	7172	18	1255	14	406						
AVERAGE		248	1507	497	7148	18	1251	13	403						
FINAL		251	1507	518	7130	18	1248	13	401						
MACROSCOPIC CROSS SECTION															
ABSORPTION-INITIAL		-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0		
AVERAGE		0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000		
FINAL		0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000		
ISOTOPIC DENSITY MILLIGRAM/CC															
INITIAL		-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0		
FINAL		0.4242	0.0066	0.0009	0.0000	0.0005	0.0000	0.0000	0.0145	0.0	0.0	0.0	0.0		
PERCENT OF ELEMENT COMPOSITION															
INITIAL		-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0	-0.0		
FINAL		100.0	87.0	12.4	0.5	94.1	0.1	5.8	100.0	0.0	0.0	0.0	0.0		
HEAT OUTPUT, WATTS/CC															
INITIAL		0.0	0.0000	0.0000	0.0000	0.0001	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000		
FINAL		0.0	0.0000	0.0000	0.0000	0.0001	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000		
NET ISOTOPIC PRODUCTION															
GM/TON-YR		66.9	1.0	0.1	0.0	0.1	0.0	0.0	2.3	0.0	0.0	0.0	0.0		
GM/MHE-YR		10.04	0.16	0.02	0.00	0.01	0.00	0.00	0.34	0.0	0.0	0.0	0.0		
GM/MMH		0.0011	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000		
GM/TON		45.8	0.7	0.1	0.0	0.1	0.0	0.0	1.6	0.0	0.0	0.0	0.0		
WATTS/MHE-YR		0.0	0.0	0.0	0.0	1.3	0.0	0.0	0.2	0.0	0.0	0.0	0.0		

EXPOSURE 10000.0 MWD/T		*** PROPERTIES OF SPECIAL PRODUCTS ***					CASE NUMBER 1000			
ISOTOPES		NP237	AM241	AM243	AM242	CM242	CM243	CM244	PU238	FP 50
HALF LIFE		STABLE	458 YR	6760YR	152 YR	163 DY	32 YR	17.6YR	89 YR	
MICROSCOPIC CROSS SECTION										
ABSORPTION-INITIAL		245	1553	476	7172	18	1255	14	406	0
AVERAGE		249	1507	512	7136	18	1249	13	401	0
FINAL		252	1509	536	7120	18	1247	13	400	0
MACROSCOPIC CROSS SECTION										
ABSORPTION-INITIAL		-0.	-0.	-0.	-0.	-0.	-0.	-0.	-0.	-0.
AVERAGE		0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.
FINAL		0.001	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.
ISOTOPIC DENSITY MILLIGRAM/CC										
INITIAL		-0.	-0.	-0.	-0.	-0.	-0.	-0.	-0.	0.
FINAL		1.1010	0.0533	0.0174	0.0005	0.0070	0.0000	0.0012	0.0722	-0.
PERCENT OF ELEMENT COMPOSITION										
INITIAL		-0.	-0.	-0.	-0.	-0.	-0.	-0.	-0.	0.
FINAL		100.0	75.0	24.3	0.7	85.6	0.2	14.2	100.0	0.
HEAT OUTPUT, WATTS/CC		0.	0.0000	0.0000	0.0000	0.0008	0.0000	0.0000	0.0000	0.
NET ISOTOPIC PRODUCTION										
GM/TON-YR		86.8	4.2	1.4	0.0	0.6	0.0	0.1	5.7	-0.
GM/MWE-YR		13.03	0.63	0.21	0.01	0.08	0.00	0.01	0.85	-0.
GM/MWH		0.0015	0.0001	0.0000	0.0000	0.0000	0.0000	0.0000	0.0001	-0.
GM/TON		119.0	5.8	1.9	0.1	0.8	0.0	0.1	7.8	-0.
WATTS/MWE-YR		0.	0.1	0.0	0.0	10.0	0.0	0.0	0.5	0.

EXPOSURE 15000.0 MWD/Y		*** PROPERTIES OF SPECIAL PRODUCTS ***						CASE NUMBER 1000		
ISOTOPES		NP237	AM241	AM243	AM242	CM242	CM243	CM244	PU238	FP 00
HALF LIFE		STABLE	458 YR	6760YR	152 YR	163 DY	32 YR	17.6YR	89 YR	
MICROSCOPIC CROSS SECTION										
ABSORPTION-INITIAL		245	1553	476	7172	18	1259	14	406	0
AVERAGE		250	1507	521	7130	18	1248	13	401	0
FINAL		252	1506	543	7122	18	1247	13	400	0
MACROSCOPIC CROSS SECTION										
ABSORPTION-INITIAL		-0.	-0.	-0.	-0.	-0.	-0.	-0.	-0.	-0.
AVERAGE		0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.
FINAL		0.001	0.001	0.000	0.000	0.000	0.000	0.000	0.000	0.
ISOTOPIC DENSITY MILLIGRAM/CC										
INITIAL		-0.	-0.	-0.	-0.	-0.	-0.	-0.	-0.	0.
FINAL		1.9297	0.1519	0.0837	0.0018	0.0275	0.0001	0.0091	0.1940	-0.
PERCENT OF ELEMENT COMPOSITION										
INITIAL		-0.	-0.	-0.	-0.	-0.	-0.	-0.	-0.	0.
FINAL		100.0	64.2	35.1	0.8	75.1	0.2	24.7	100.0	0.
HEAT OUTPUT, WATTS/CC										
INITIAL		0.	0.0000	0.0000	0.0000	0.0033	0.0000	0.0000	0.0001	0.
FINAL										
NET ISOTOPIC PRODUCTION										
GM/TON-YR		101.5	8.0	4.4	0.1	1.4	0.0	0.5	10.2	-0.
GM/MWE-YR		15.22	1.20	0.66	0.01	0.22	0.00	0.07	1.53	-0.
GM/MWH		0.0017	0.0001	0.0001	0.0000	0.0000	0.0000	0.0000	0.0002	-0.
WATTS/MWH		208.5	16.4	9.0	0.2	3.0	0.0	1.0	23.0	-0.
FINAL		0	0.1	0.0	0.0	26.0	0.0	0.2	0.9	0.

EXPOSURE 24686.0 MWD/Y

*** PROPERTIES OF SPECIAL PRODUCTS ***

CASE NUMBER 1000

ISOTOPES	NP237	AM241	AM243	AM242	CM242	CM243	CM244	PU238	FP 20
HALF LIFE	STABLE	458 YR	6760 YR	152 YR	163 DY	32 YR	17.6 YR	89 YR	
MICROSCOPIC CROSS SECTION									
ABSORPTION=INITIAL	245	1553	476	7172	18	1255	14	406	0
AVERAGE	250	1504	530	7130	18	1248	13	400	0
FINAL	251	1491	541	7137	18	1249	13	401	0
MACROSCOPIC CROSS SECTION									
ABSORPTION=INITIAL	-0.	-0.	-0.	-0.	-0.	-0.	-0.	-0.	-0.
AVERAGE	0.001	0.001	0.000	0.000	0.000	0.000	0.000	0.000	0.
FINAL	0.002	0.002	0.001	0.000	0.000	0.000	0.000	0.001	0.
ISOTOPIC DENSITY MILLIGRAM/CC									
INITIAL	-0.	-0.	-0.	-0.	-0.	-0.	-0.	-0.	0.
FINAL	3.7669	0.4353	0.4934	0.0062	0.1126	0.0006	0.0991	0.6797	-0.
PERCENT OF ELEMENT COMPOSITION									
INITIAL	-0.	-0.	-0.	-0.	-0.	-0.	-0.	-0.	0.
FINAL	100.0	46.8	52.6	0.7	53.2	0.3	46.5	100.0	0.
HEAT OUTPUT, WATTS/CC	0.	0.0000	0.0000	0.0000	0.0135	0.0000	0.0003	0.0004	0.
NET ISOTOPIC PRODUCTION									
GM/TON-YR	120.4	13.9	15.8	0.2	3.6	0.0	3.2	21.7	-0.
GM/MWE-YR	18.05	2.09	2.36	0.03	0.54	0.00	0.48	3.26	-0.
GM/MWH	0.0021	0.0002	0.0003	0.0000	0.0001	0.0000	0.0001	0.0004	-0.
GM/TON	407.0	47.0	53.3	0.7	12.2	0.1	10.7	73.4	-0.
WATTS/MWE-YR	0.	0.2	0.0	0.0	64.7	0.0	1.3	1.8	0.

67

BNWL-223

GRADED	FINAL K 1.083	***	AFTER DISCHARGE BEFORE SEPARATION	***	CASE NUMBER	1000					
EXPOSURE THERMAL TO ELECTRICAL EFF. 0.33	24686.0 MWD/T	POWER DENS 25.00 MW/T	RESIDENCE TIME 3.38 YRS,	LOAD FACTOR 0.80							
TIME AFTER REACTOR DISCHARGE UNITS OF 0.45 YEAR HALF-LIFE	U236	NP237	PU238	PU239	PU240	PU241	PU242	AM241	AM243	CM242	CM244
			*** GM/TON ***								
0.	3288.54	406.99	73.43	5907.9	2099.6	1159.0	286.0	47.04	53.31	12.16	10.71
0.09	3288.54	406.99	74.16	5907.9	2099.7	1156.5	286.0	49.57	53.31	11.41	10.69
0.18	3288.54	406.99	74.84	5907.9	2099.7	1154.0	286.0	52.09	53.31	10.71	10.68
0.28	3288.54	406.99	75.48	5907.9	2099.7	1151.4	286.0	54.61	53.31	10.05	10.66
0.37	3288.54	407.00	76.08	5907.9	2099.7	1148.9	286.0	57.13	53.31	9.42	10.64
0.55	3288.54	407.01	77.16	5907.9	2099.8	1143.9	286.0	62.14	53.31	8.30	10.61
0.74	3288.54	407.02	78.10	5907.9	2099.8	1138.9	286.0	67.13	53.31	7.30	10.57
0.92	3288.54	407.03	78.92	5907.9	2099.8	1133.9	286.0	72.10	53.31	6.43	10.54
1.10	3288.54	407.10	79.64	5907.9	2099.9	1129.0	286.0	77.04	53.31	5.66	10.50
2.21	3288.54	407.30	82.36	5907.9	2100.1	1099.7	286.0	106.25	53.31	2.63	10.30
4.41	3288.54	407.51	83.78	5907.9	2100.4	1043.4	286.0	162.33	53.30	0.57	9.91
6.13	3288.54	407.51	83.68	5907.9	2100.7	1001.6	286.0	203.90	53.30	0.17	9.62
			*** GM/MWE-YEAR ***								
0.	146.116	18.083	3.263	262.50	93.29	51.50	12.71	2.090	2.369	0.540	0.476
0.09	146.116	18.083	3.295	262.50	93.29	51.39	12.71	2.202	2.369	0.507	0.475
0.18	146.116	18.083	3.325	262.50	93.29	51.27	12.71	2.315	2.369	0.476	0.474
0.28	146.116	18.084	3.354	262.50	93.29	51.16	12.71	2.427	2.369	0.446	0.474
0.37	146.116	18.084	3.380	262.50	93.29	51.05	12.71	2.538	2.369	0.419	0.473
0.55	146.116	18.084	3.428	262.50	93.30	50.83	12.71	2.761	2.369	0.369	0.471
0.74	146.116	18.084	3.470	262.50	93.30	50.60	12.71	2.983	2.369	0.324	0.470
0.92	146.116	18.085	3.507	262.50	93.30	50.38	12.71	3.203	2.369	0.286	0.468
1.10	146.116	18.085	3.539	262.50	93.30	50.16	12.71	3.423	2.369	0.251	0.467
2.21	146.116	18.088	3.659	262.50	93.31	48.86	12.71	4.721	2.368	0.117	0.458
4.41	146.116	18.097	3.723	262.50	93.33	46.36	12.71	7.213	2.368	0.025	0.440
6.13	146.116	18.107	3.718	262.50	93.34	44.50	12.71	9.060	2.368	0.008	0.427

GRADED	FINAL K 1.083	*** AFTER DISCHARGE BEFORE SEPARATION ***	CASE NUMBER 1000									
TIME AFTER REACTOR DISCHARGE UNITS OF 0.45 YEAR HALF-LIFE	DAYS	U236	NP237	PU238	PU239	PU240	PU241	PU242	AM241	AM243	CM242	CM244
				*** GM/MWH ***								
0.	0.	0.0167	0.0021	0.0004	0.0299	0.0106	0.0059	0.0014	0.0002	0.0003	0.0001	0.0001
0.09	15.0	0.0167	0.0021	0.0004	0.0299	0.0106	0.0059	0.0014	0.0003	0.0003	0.0001	0.0001
0.18	30.0	0.0167	0.0021	0.0004	0.0299	0.0106	0.0058	0.0014	0.0003	0.0003	0.0001	0.0001
0.28	45.0	0.0167	0.0021	0.0004	0.0299	0.0106	0.0058	0.0014	0.0003	0.0003	0.0001	0.0001
0.37	60.0	0.0167	0.0021	0.0004	0.0299	0.0106	0.0058	0.0014	0.0003	0.0003	0.0000	0.0001
0.55	90.0	0.0167	0.0021	0.0004	0.0299	0.0106	0.0058	0.0014	0.0003	0.0003	0.0000	0.0001
0.74	120.0	0.0167	0.0021	0.0004	0.0299	0.0106	0.0058	0.0014	0.0003	0.0003	0.0000	0.0001
0.92	150.0	0.0167	0.0021	0.0004	0.0299	0.0106	0.0057	0.0014	0.0004	0.0003	0.0000	0.0001
1.10	180.0	0.0167	0.0021	0.0004	0.0299	0.0106	0.0057	0.0014	0.0004	0.0003	0.0000	0.0001
2.21	360.0	0.0167	0.0021	0.0004	0.0299	0.0106	0.0056	0.0014	0.0005	0.0003	0.0000	0.0001
4.41	720.0	0.0167	0.0021	0.0004	0.0299	0.0106	0.0053	0.0014	0.0008	0.0003	0.0000	0.0001
6.13	1000.0	0.0167	0.0021	0.0004	0.0299	0.0106	0.0051	0.0014	0.0010	0.0003	0.0000	0.0000
				*** WATTS/MWE-YEAR ***								
0.	0.	0.	0.	1.8271	0.	0.	0.	0.	0.023	0.	64.85	1.332
0.09	15.0	0.	0.	1.8452	0.	0.	0.	0.	0.024	0.	60.85	1.330
0.18	30.0	0.	0.	1.8622	0.	0.	0.	0.	0.025	0.	57.09	1.328
0.28	45.0	0.	0.	1.8781	0.	0.	0.	0.	0.027	0.	53.56	1.326
0.37	60.0	0.	0.	1.8929	0.	0.	0.	0.	0.028	0.	50.25	1.324
0.55	90.0	0.	0.	1.9198	0.	0.	0.	0.	0.030	0.	44.23	1.320
0.74	120.0	0.	0.	1.9432	0.	0.	0.	0.	0.033	0.	38.94	1.315
0.92	150.0	0.	0.	1.9638	0.	0.	0.	0.	0.035	0.	34.28	1.311
1.10	180.0	0.	0.	1.9816	0.	0.	0.	0.	0.038	0.	30.17	1.307
2.21	360.0	0.	0.	2.0492	0.	0.	0.	0.	0.052	0.	14.04	1.282
4.41	720.0	0.	0.	2.0846	0.	0.	0.	0.	0.079	0.	3.04	1.233
6.13	1000.0	0.	0.	2.0820	0.	0.	0.	0.	0.100	0.	0.92	1.196

GRADED FINAL K 1.083

*** AFTER DISCHARGE,BEFORE SEPARATION ***

CASE NUMBER 1000

TIME AFTER REACTOR DISCHARGE
UNITS OF 0.45
YEAR HALF-LIFE DAYS

U236	NP237	PU238	PU239	PU240	PU241	PU242	AM241	AM243	CM242	CM244
*** PERCENT OF ELEMENT ***										
0.34	100.00	0.77	62.02	22.04	12.17	3.00	46.87	53.13	53.18	46.82
0.34	100.00	0.78	62.03	22.05	12.14	3.00	48.18	51.82	51.63	48.37
0.34	100.00	0.79	62.04	22.05	12.12	3.00	49.42	50.58	50.07	49.93
0.34	100.00	0.79	62.05	22.05	12.09	3.00	50.60	49.40	48.52	51.48
0.34	100.00	0.80	62.07	22.06	12.07	3.00	51.73	48.27	46.97	53.03
0.34	100.00	0.81	62.09	22.07	12.02	3.01	53.82	46.18	43.89	56.11
0.34	100.00	0.82	62.12	22.08	11.98	3.01	55.74	44.26	40.85	59.15
0.34	100.00	0.83	62.14	22.09	11.93	3.01	57.49	42.51	37.89	62.11
0.34	100.00	0.84	62.17	22.10	11.88	3.01	59.10	40.90	35.01	64.99
0.34	100.00	0.87	62.35	22.16	11.61	3.02	66.59	33.41	20.35	79.65
0.34	100.00	0.89	62.71	22.29	11.07	3.04	75.28	24.72	5.44	94.56
0.34	100.00	0.89	62.98	22.40	10.68	3.05	79.28	20.72	1.77	98.23

70

BNWL-223

GRADED FINAL K 1,083

*** AFTER SEPARATION ***
*** GM/TON ***

CASE NUMBER 1000

EXPOSURE 24686.0 MWD/T POWER DENS 25.00 MW/T RESIDENCE TIME 3.38 YRS, LOAD FACTOR 0.80
THERMAL TO ELECTRICAL EFF. 0.33

TIME AFTER DISCHARGE		TIME AFTER	URANIUM		NEPTUNIUM	PLUTONIUM						
0.45 YR		SEPARATION	..RECOVERED...		RECOVERED	RECOVERED.....						
HALF-LIVES	DAYS	DAYS	U236	OTHER U	NP237	PU238	PU239	PU240	PU241	PU242	NP237	AM241
		YIELD										
		FACTOR	0.970	0.970	0.950	0.970	0.970	0.970	0.970	0.970	0.	0.
0.74	120.0	0.	3189.89	931794.1	386.66	75.76	5730.6	2036.8	1104.7	277.5	0.	0.
0.83	135.0	15.0	3189.89	931794.1	386.66	75.73	5730.6	2036.8	1102.3	277.5	0.00	2.42
1.00	163.0	43.0	3189.89	931794.1	386.66	75.69	5730.6	2036.8	1097.8	277.5	0.00	6.91
2.00	326.0	206.0	3189.89	931794.1	386.66	75.42	5730.6	2036.8	1072.0	277.5	0.01	32.71
3.00	489.0	369.0	3189.89	931794.1	386.66	75.16	5730.6	2036.8	1046.8	277.5	0.04	57.88
4.00	652.0	532.0	3189.89	931794.1	386.66	74.90	5730.6	2036.8	1022.2	277.5	0.09	82.44
5.00	815.0	695.0	3189.89	931794.1	386.66	74.64	5730.6	2036.8	998.2	277.5	0.16	106.41
6.00	978.0	858.0	3189.89	931794.1	386.66	74.38	5730.6	2036.8	974.7	277.5	0.24	129.79
7.00	1141.0	1021.0	3189.89	931794.1	386.66	74.12	5730.6	2036.8	951.8	277.5	0.33	152.61

TIME AFTER DISCHARGE		TIME AFTER	AMERICIUM				CURIUM			
0.45 YR		SEPARATION	RECOVERED.....				RECOVERED.....			
HALF-LIVES	DAYS	DAYS	AM241	AM243	NP237	PU239	CM242	CM244	PU238	PU240
		YIELD								
		FACTOR	0.950	0.950	0.	0.	0.950	0.950	0.	0.
0.74	120.0	0.	63.77	50.64	0.	0.	6.94	10.04	0.	0.
0.83	135.0	15.0	63.77	50.64	0.00	0.00	6.51	10.03	0.43	0.02
1.00	163.0	43.0	63.76	50.64	0.01	0.00	5.78	10.00	1.16	0.05
2.00	326.0	206.0	63.72	50.64	0.05	0.00	2.89	9.82	4.04	0.22
3.00	489.0	369.0	63.68	50.64	0.10	0.00	1.45	9.65	5.47	0.39
4.00	652.0	532.0	63.63	50.64	0.14	0.01	0.72	9.48	6.17	0.56
5.00	815.0	695.0	63.59	50.64	0.18	0.01	0.36	9.32	6.51	0.73
6.00	978.0	858.0	63.55	50.63	0.23	0.01	0.18	9.16	6.66	0.89
7.00	1141.0	1021.0	63.50	50.63	0.27	0.01	0.09	9.00	6.73	1.05

GRADED	FINAL K 1.083	*** AFTER SEPARATION ***		CASE NUMBER 1000	
		*** GM/MWE-YEAR ***			
TIME AFTER DISCHARGE	TIME AFTER SEPARATION	URANIUM	NEPTUNIUM	PLUTONIUM	
0.45 YR	SEPARATION	..RECOVERED...	RECOVERED	..RECOVERED...	
HALF-LIVES	DAYS	U236 OTHER U	NP237	PU238 PU239 PU240 PU241 PU242 NP237 AM241	
0.74	120.0	141.73 41401.42	17.18	3.37 254.62 90.50 49.09 12.33	0.
0.83	135.0	141.73 41401.42	17.18	3.36 254.62 90.50 48.98 12.33	0.11
1.00	163.0	141.73 41401.42	17.18	3.36 254.62 90.50 48.78 12.33	0.31
2.00	326.0	141.73 41401.42	17.18	3.35 254.62 90.50 47.63 12.33	1.45
3.00	489.0	141.73 41401.42	17.18	3.34 254.62 90.50 46.51 12.33	2.97
4.00	652.0	141.73 41401.42	17.18	3.33 254.62 90.50 45.42 12.33	3.66
5.00	815.0	141.73 41401.42	17.18	3.32 254.62 90.50 44.35 12.33	4.73
6.00	978.0	141.73 41401.42	17.18	3.30 254.62 90.50 43.31 12.33	5.77
7.00	1141.0	141.73 41401.42	17.18	3.29 254.62 90.50 42.29 12.33	6.78

TIME AFTER DISCHARGE	TIME AFTER SEPARATION	AMERICIUM	CURIUM	..RECOVERED...	..RECOVERED...
0.45 YR	SEPARATION	AM241 AM243 NP237 PU239	CM242 CM244 PU238 PU240		
HALF-LIVES	DAYS				
0.74	120.0	2.83 2.25 0.	0.31 0.45 0.	0.31 0.45 0.	0.
0.83	135.0	2.83 2.25 0.00	0.29 0.45 0.02	0.29 0.45 0.02	0.00
1.00	163.0	2.83 2.25 0.00	0.26 0.44 0.05	0.26 0.44 0.05	0.00
2.00	326.0	2.83 2.25 0.00	0.13 0.44 0.18	0.13 0.44 0.18	0.01
3.00	489.0	2.83 2.25 0.00	0.06 0.43 0.24	0.06 0.43 0.24	0.02
4.00	652.0	2.83 2.25 0.01	0.03 0.42 0.27	0.03 0.42 0.27	0.02
5.00	815.0	2.83 2.25 0.01	0.02 0.41 0.29	0.02 0.41 0.29	0.03
6.00	978.0	2.82 2.25 0.01	0.01 0.41 0.30	0.01 0.41 0.30	0.04
7.00	1141.0	2.82 2.25 0.01	0.00 0.40 0.30	0.00 0.40 0.30	0.05

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*** AFTER SEPARATION ***
*** PERCENT OF ELEMENT ***

CASE NUMBER 1000

TIME AFTER DISCHARGE		TIME AFTER SEPARATION DAYS	URANIUM		NEPTUNIUM RECOVERED NP237	PLUTONIUM RECOVERED						
0.45 YR HALF-LIVES	DAYS		..RECOVERED.. U236	OTHER U		 PU238	 PU239	 PU240	 PU241	 PU242	 NP237	 AM241
0.74	120.0	0.	0.34	99.66	100.00	0.82	62.12	22.08	11.98	3.01	100.00	100.00
0.83	135.0	15.0	0.34	99.66	100.00	0.82	62.13	22.08	11.95	3.01	100.00	100.00
1.00	163.0	43.0	0.34	99.66	100.00	0.82	62.17	22.09	11.91	3.01	100.00	100.00
2.00	326.0	206.0	0.34	99.66	100.00	0.82	62.34	22.16	11.66	3.02	100.00	100.00
3.00	489.0	369.0	0.34	99.66	100.00	0.82	62.51	22.22	11.42	3.03	100.00	100.00
4.00	652.0	532.0	0.34	99.66	100.00	0.82	62.68	22.28	11.18	3.03	100.00	100.00
5.00	815.0	695.0	0.34	99.66	100.00	0.82	62.85	22.34	10.95	3.04	100.00	100.00
6.00	978.0	858.0	0.34	99.66	100.00	0.82	63.02	22.40	10.72	3.05	100.00	100.00
7.00	1141.0	1021.0	0.34	99.66	100.00	0.82	63.18	22.45	10.49	3.06	100.00	100.00

TIME AFTER DISCHARGE		TIME AFTER SEPARATION DAYS	AMERICIUM				CURIUM			
0.45 YR HALF-LIVES	DAYS		RECOVERED..... AM241	RECOVERED..... AM243	RECOVERED..... NP237	RECOVERED..... PU239	RECOVERED..... CM242	RECOVERED..... CM244	RECOVERED..... PU238	RECOVERED..... PU240
0.74	120.0	0.	55.74	44.26	100.00	100.00	40.85	59.15	0.	0.
0.83	135.0	15.0	55.74	44.26	100.00	100.00	39.36	60.64	96.35	3.65
1.00	163.0	43.0	55.73	44.27	100.00	100.00	36.63	63.37	96.15	3.85
2.00	326.0	206.0	55.72	44.28	100.00	100.00	22.73	77.27	94.82	5.18
3.00	489.0	369.0	55.70	44.30	100.00	100.00	13.02	86.98	93.31	6.69
4.00	652.0	532.0	55.69	44.31	100.00	100.00	7.08	92.92	91.68	8.32
5.00	815.0	695.0	55.67	44.33	100.00	100.00	3.73	96.27	89.97	10.03
6.00	978.0	858.0	55.66	44.34	100.00	100.00	1.94	98.06	88.25	11.75
7.00	1141.0	1021.0	55.64	44.36	100.00	100.00	0.99	99.01	86.54	13.46

75

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