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PRELIMINARY DRAFT

THE CONTROLLED THERMONUCLEAR REACTOR
AS A FISSION PRODUCT BURNER

PRESENTED AT THE
AMERICAN NUCLEAR SOCIETY MEETING

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THE CONTROLLED THERMONUCLEAR REACTOR AS A FISSION PRODUCT BURNER

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INTRODUCTION

The USAEC and other agencies are vitally interested in finding acceptable means of storage or elimination of the large quantities of fission products that are expected to be generated by the nuclear electric generating complex which is expected to exist in the latter part of the twentieth century. Announced plans call for the permanent storage of these radioactive wastes in deep salt mines. Several alternate storage concepts are being explored. Some of these include storage in bedrock or basalt formations under federal reservations. Other long-range storage proposals which are being considered include burial in a nuclear chimney and burial in polar ice caps. Possible long-range alternatives for elimination of the fission products from the biosphere include extra-terrestrial disposal and nuclear transmutation. This paper will concern itself with some aspects of the latter option.

At least three different proposals have been made for transmutation of radioactive wastes into stable daughters. They include transmutation in a fission reactor, transmutation by means of an Intense Neutron Generator (ING) device, and transmutation with a Controlled Thermonuclear Reactor (CTR) if one were available. This paper will be concerned with the CTR transmutation approach although some of the conclusions arrived at in this analysis might be quite applicable to other transmutation concepts as well.

Why is a CTR of particular interest in the transmutation of Radioactive waste? First of all, one should recognize that the fission products which are discharged from a fission reactor can be characterized as having relatively small neutron reaction cross sections. If this were not the case, they would not exist in any quantity in the reactor at discharge.

As will be shown later, some of the most noxious isotopes have a half-life of the order of thirty years. If an assumption is used that, as a minimum, one wishes to transmute at about at least the rate of the nuclear half-life in order shorten the effective half-life, one can perform the following simple calculation. Many of the interesting fission products have neutron reaction cross sections of the order of a barn. By equating the decay rate to the transmutation rate;

$$\lambda \approx \sigma \phi$$

or:

$$\frac{0.693}{t_{1/2}} \approx \sigma \phi$$

One can arrive at an estimate of the minimum required flux for transmutation, at this level.

$$\phi = \frac{0.693}{\sigma t_{1/2}}$$

For a thirty year half-life and a reaction cross section of one barn, the minimum required flux level can be shown to be of the order of $10^{15} \text{ n/cm}^2\text{-sec.}$

The initial justification, then, for analyzing a CTR as a fission product burner is that it is one of a very few devices that one can conceive of that provides this level of neutron flux in appreciable quantities.

SELECTION OF TRANSMUTATION CONCEPTS

Transmutation, particularly if it is a neutron absorbing process, could prove to be relatively expensive. It is, therefore, perhaps worthwhile to try and identify those fission products which could be most usefully transmuted. This identification process could be carried on either

by selecting transmutation targets on the basis of the highest cross section for transmutation or by identifying the most noxious fission products. In this study, the latter approach was used.

To initiate the selection process, a computer calculation was performed on a typical Pressurized Water Reactor (PWR) with the ISOGEN code. In the calculation, the fuel which was 3.4% enriched in ^{235}U was exposed to 30,000 MWd/tonne at 20 MW/tonne. The fuel was then allowed to cool for 10 years and the inventory of fission products at that point in time was calculated. The results of the calculation are displayed in Table 1. Also displayed in this table are the activity level in curies for each fission product, the beta emission in watts for each fission product, and the gamma emission in watts for each fission product.

In order to determine the relative hazard associated with these fission products, those isotopes with appreciable curie activity were arranged in descending order of activity in curies. The result of this ordering is shown in Table 2. In addition, the fission products are shown in Tables 3 and 4 arranged by beta-watts and gamma-watts respectively. In all of these tables (2,3, and 4), the percent of the total activity for each isotope is given along with the percent accumulation as one descends through the table.

Tables 2, 3, and 4 are particularly useful in determining candidate materials for inclusion in transmutation schemes. From Table 2, one finds that ^{90}Sr and ^{137}Cs account for almost 85% of the activity in curies with the next isotope, ^{147}Pm , only providing some 5% more in activity. Table 3 shows that ^{90}Sr and ^{137}Cs account for 95% of the beta activity with the next isotope, ^{85}Kr , providing about 1% additional activity. Table 4 shows that ^{137}Cs provides 75% of the activity from gamma decay. ^{134}Cs provides 17% of the gamma activity and the next isotope, ^{154}Eu , provides 7% of the gamma activity. Therefore, as a working hypothesis, ^{90}Sr and ^{137}Cs can be identified as prime targets for transmutation.

In order to check these selections, maximum permissible concentrations (MPC's) for the fission product isotopes were obtained from 10 CFR 20, Appendix B, Table 1. These MPC values were divided into the activities of each isotope shown in Table 2 and the isotopes were then reordered by (curies/MPC) from the highest value on down. It is hoped in this way to account for the relative biological hazard associated with the form and specific energy of each radioactive emission. The result of this analysis is that the first three isotopes in decending order are ^{90}Sr , ^{137}Cs , and ^{154}Eu . Other potential candidates are ^{134}Cs and ^{127}Sb . On the basis of the above approach, ^{137}Cs and ^{90}Sr were selected as the prime targets for the following analysis. This result is perhaps not surprising to people familiar with waste management problems.

Transmutation will probably be based upon chemical separation of preselected fission products for target material. If this is the case, it is interesting to notice that by selecting ^{90}Sr and ^{137}Cs for transmutation, one would also carry ^{134}Cs , another of the isotopes of interest, along in the process

One should notice that other analyses, based upon different fuel, different cooling times, and other selection criteria, might well result in the selection of different transmutation targets

TABLE 1
POTENTIAL TRANSMUTATION CANDIDATES

Fission Products	GMS Avail- able After Ten Years	Fission Products	GMS Avail- able After Ten Years	Fission Products	GMS Avail- able After Ten Years
^{166}Dy	0.0	^{148}Sm	1.547+2	^{135}Ba	2.375-3
^{164}Dy	1.597-2	^{147}Sm	1.682+2	^{134}Ba	1.970+2
^{163}Dy	6.378-2	^{146}Sm	3.415-5	^{137}Cs	8.867+2
^{162}Dy	1.015-1	^{148}Pm	1.176-28	^{136}Cs	
^{161}Dy	2.379-1	$^{148}\text{M}_{\text{Pm}}$	1.006-26	^{135}Cs	5.587+2
^{160}Dy	1.998-1	^{147}Pm	7.921	^{134}Cs	4.742+0
^{161}Tb		^{146}Pm	2.453-5	^{133}Cs	9.861+2
^{160}Tb	3.244-17	^{150}Nd	1.653+2	^{136}Xe	1.858+3
^{159}Tb	1.785	^{148}Nd	3.422+2	^{134}Xe	1.363+3
^{160}Gd	7.764-1	^{147}Nd		^{133}Xe	
^{158}Gd	1.079+1	^{146}Nd	6.246+2	^{132}Xe	9.797+2
^{157}Gd	6.881-2	^{145}Nd	6.231+2	^{131}Xe	4.181+2
^{156}Gd	5.385+1	^{144}Nd	1.132+3	$^{131}\text{M}_{\text{Xe}}$	
^{155}Gd	4.777+0	^{143}Nd	7.982+2	^{130}Xe	5.618+0
^{154}Gd	1.411+1	^{142}Nd	9.416+0	^{129}Xe	1.043-4
^{152}Gd	9.579-2	^{143}Pr		^{131}I	
^{156}Eu		^{141}Pr	1.071+3	^{129}I	2.002+2
^{155}Eu	1.016-1	^{144}Ce	3.004-2	^{127}I	4.899+1
^{154}Eu	2.241+1	^{142}Ce	1.044+3	^{132}Te	
^{153}Eu	1.247+2	^{141}Ce	4.899-33	^{130}Te	3.784+2
^{152}Eu	2.726-2	^{140}Ce	1.130+3	$^{129}\text{M}_{\text{Te}}$	1.206-33
^{151}Eu	1.291	^{139}La	1.144+3	^{128}Te	1.085+2
^{154}Sm	3.186+1	^{140}Ba		$^{127}\text{M}_{\text{Te}}$	3.061-11
^{152}Sm	9.599+1	^{138}Ba	1.207+3	^{126}Te	3.900-2
^{151}Sm	1.565+1	^{137}Ba	2.904+2	^{125}Te	1.134+1
^{150}Sm	2.816+2	^{136}Ba	2.233+1	$^{125}\text{M}_{\text{Te}}$	7.550-3
^{149}Sm	4.833				

TABLE 1 (contd)

Fission Products	GMS Avail- able After Ten Years	Fission Products	GMS Avail- able After Ten Years	Fission Products	GMS Avail- able After Ten Years
^{127}Sb		^{111}Ag		^{96}Zr	7.601+2
^{126}Sb		$^{110\text{M}}\text{Ag}$	2.643-6	^{95}Zr	5.257-16
^{125}Sb	5.754-1	^{109}Ag	5.811+1	^{94}Zr	7.188+2
^{123}Sb	5.126+0	^{107}Ag	2.337-4	^{93}Zr	6.650+2
^{121}Sb	3.932+0	^{110}Pd	3.625+1	^{92}Zr	5.895+2
^{126}Sn	2.365+1	^{108}Pd	1.356+2	^{91}Zr	5.454+2
^{125}Sn		^{107}Pd	1.999+2	^{90}Zr	1.352+2
^{124}Sn	7.339	^{106}Pd	3.761+2	^{91}Yo	2.813-18
$^{123\text{M}}\text{Sn}$	1.095-10	^{105}Pd	2.588+2	^{89}Yo	4.057+2
^{122}Sn	4.804+0	^{104}Pd	1.999+2	^{90}Sr	3.730+2
^{120}Sn	3.900+0	^{103}Rh	3.816+2	^{89}Sr	2.427-21
^{119}Sn	3.809+0	^{106}Ru	1.138-1	^{88}Sr	3.111+2
^{118}Sn	3.809+0	^{104}Ru	5.097+2	^{86}Sr	1.183-1
^{117}Sn	3.733+0	^{103}Ru	4.269-27	^{87}Rb	2.115+2
^{116}Sn	2.375+0	^{102}Ru	7.005+2	^{86}Rb	
^{115}Sn	2.584-14	^{101}Ru	7.157+2	^{85}Rb	9.904+1
^{115}In	1.484+0	^{100}Ru	5.061+1	^{86}Kr	1.574+2
^{116}Cd	3.754+0	^{99}Ru	3.127-2	^{85}Kr	1.095+1
$^{115\text{M}}\text{Cd}$	3.316-28	^{99}Tc	7.748+2	^{84}Kr	8.997+1
^{114}Cd	1.082+1	^{100}Mo	9.094+2	^{83}Kr	3.699+1
^{113}Cd	1.471-1	^{98}Mo	7.828+2	^{81}Br	1.688+1
^{112}Cd	9.078+0	^{97}Mo	7.805+2	^{79}Br	5.660-4
^{111}Cd	1.580+1	^{96}Mo	2.949+1	^{82}Se	2.985+1
^{110}Cd	3.000+1	^{95}Mo	7.155+2	^{81}Se	9.271+0
^{109}Cd	1.073-11	^{95}Nb	6.133-16	^{79}Se	4.636+0
^{108}Cd	9.583-7	^{93}Nb	5.925-3	^{77}Se	1.768+0
		$^{93\text{M}}\text{Nb}$	9.314-5		7.159-1

TABLE 1 (contd)

<u>Fission Products</u>	<u>GMS Avail- able After Ten Years</u>
^{075}As	9.661-2
^{076}Ge	2.460-1
^{074}Ge	4.620-2
^{073}Ge	1.904-2
^{072}Ge	6.013-2
^{003}H	1.613-2

TABLE 2
FISSION PRODUCTS ARRANGED BY ACTIVITY IN CURIES

<u>Fission Products</u>	<u>Curies (in order)</u>	<u>% of Total</u>	<u>% Accumulation</u>
¹³⁷ Cs	7.713+4	0.50394	
⁹⁰ Sr	5.292+4	0.34576	0.84970
¹⁴⁷ Pm	7.353+3	0.04804	0.89774
¹³⁴ Cs	6.025+3	0.03937	0.93711
⁸⁵ Kr	4.428+3	0.02893	0.96604
¹⁵⁴ Eu	3.252+3	0.02125	0.98729
¹²⁷ Sb	6.095+2	0.00398	0.99127
¹⁵¹ Sm	4.118+2	0.00269	0.99396
¹⁰⁶ Ru	3.838+2	0.00251	0.99647
⁰⁰³ H	1.568+2	0.00102	0.99749
^{125M} Te	1.360+2	0.00089	0.99838
¹⁵⁵ Eu	1.295+2	0.00085	0.99923
¹⁴⁴ Ce	9.559+1	0.00062	0.99985
⁰⁹⁹ Tc	1.332+1	0.00009	0.99994
¹⁵² Eu	4.932	0.00003	0.99997
⁰⁹³ Zr	2.691	0.00002	0.99999
¹³⁵ Cs	7.398-1		
¹²⁶ Sn	6.709-1		
⁰⁷⁹ Se	2.998-1		
^{093M} Nb	9.677-2		
¹⁰⁷ Pd	9.542-2		
¹²⁹ I	3.468-2		
^{110M} Ag	1.207-2		
¹⁴⁶ Pm	1.086-2		
⁰⁸⁷ Rb	1.849-5		
¹⁴⁷ Sm	3.858-6		
^{123M} Sn	9.297-7		
^{127M} Te	2.997-2		
¹⁴⁸ Sm	3.113-8		

TABLE 2 (contd)

<u>Fission Products</u>	<u>Curies (in Order)</u>	<u>% of Total</u>	<u>% Accumulation</u>
^{109}Cd	2.708-8		
^{144}Nd	1.171-9		
^{146}Sm	6.969-10		
^{142}Ce	5.258-10		
^{149}Sm	2.899-11		
^{95}Nb	2.409-11		
^{95}Zr	1.112-11		
^{115}In	9.226-12		
^{152}Gd	2.048-12		
^{160}Tb	3.627-13		
^{91}Yo	6.959-14		

TABLE 3

FISSION PRODUCTS ARRANGED BY BETA EMISSION

<u>Fission Products</u>	<u>Beta-Watts</u>	<u>% of Total</u>	<u>% Accumulation</u>
⁹⁰ Sr	3.573+2	0.71487	
¹³⁷ Cs	1.189+2	0.23789	0.95275
⁸⁵ Kr	6.562+0	0.01313	0.96588
¹³⁴ Cs	5.357+0	0.01072	0.97660
¹⁵⁴ Eu	4.433+0	0.00887	0.98547
¹⁰⁶ Ru	3.185+0	0.00637	0.99184
¹⁴⁷ Pm	2.650+0	0.00530	0.99715
¹⁴⁴ Ce	7.536-1	0.00151	0.99865
¹²⁷ Sb	4.155-1	0.00083	0.99948
^{125M} Te	1.169-1	0.00023	0.99972
¹⁵⁵ Eu	7.678-2	0.00015	0.99987
¹⁵¹ Sm	4.883-2	0.00010	0.99997
⁰⁹⁹ Tc	6.635-3	0.00001	0.99998
⁰⁰³ H	4.183-3	0.00001	0.99999
¹⁵² Eu	3.800-3		
¹³⁵ Cs	2.543-4		
⁰⁹³ Zr	2.233-4		
⁰⁷⁹ Se	1.066-4		
¹²⁶ Sn	7.954-5		
^{093M} Nb	1.664-5		
¹²⁹ I	1.664-5		
¹⁴⁶ Pm	7.727-6		
^{110M} Ag	6.796-6		
¹⁰⁷ Pd	5.656-6		
⁰⁸⁷ Rb	1.096-8		

TABLE 3 (contd)

<u>Fission Products</u>	<u>Beta-Watts</u>	<u>% of Total</u>	<u>% Accumulation</u>
¹²³ M _{Sn}	2.967-9		
¹²⁷ M _{Te}	5.685-10		
¹⁴⁸ Sm	3.950-10		
¹⁰⁹ Cd	1.412-11		
¹⁴⁴ Nd	1.319-11		
¹⁴⁶ Sm	1.053-11		
¹⁴² Ce	4.675-12		
¹⁴⁹ Sm	3.162-13		
¹⁵² Gd	2.610-14		
¹⁵⁵ In	1.094-14		
⁰⁹⁵ Zr	8.171-15		
⁰⁹⁵ Nb	6.568-15		
¹⁶⁰ Tb	5.804-16		
⁰⁹¹ Yo	2.558-16		
¹⁴⁷ Sm	4.986-18		
⁰⁸⁹ Sr	2.429-19		
¹⁴⁸ M _{Pm}	3.256-25		
¹⁰³ Ru	9.156-26		
¹⁴⁸ Pm	8.089-26		
¹¹⁵ M _{Cd}	3.114-26		
¹⁴¹ Ce	1.490-31		
¹²⁹ M _{Te}	1.315-31		

TABLE 4
FISSION PRODUCTS ARRANGED BY GAMMA EMISSION

<u>Fission Products</u>	<u>Gamma-Watts</u>	<u>% of Total</u>	<u>% Accumulation</u>
¹³⁷ Cs	2.542+2	0.75451	
¹³⁴ Cs	5.572+1	0.16539	0.91989
¹⁵⁴ Eu	2.467+1	0.07322	0.99312
¹²⁷ Sb	1.564	0.00464	0.99776
¹⁰⁶ Ru	4.619-1	0.00137	0.99913
⁰⁸⁵ Kr	1.312-1	0.00039	0.99952
¹⁴⁴ Ce	8.633-2	0.00026	0.99977
¹⁵⁵ Eu	3.977-2	0.00011	0.99989
¹⁵² Eu	3.400-2	0.00010	0.99999
^{125M} Te	2.015-3		
^{110M} Ag	2.008-4		
¹⁴⁶ Pm	5.022-5		
¹²³ Sn	1.212-10		
^{125M} Te	7.107-12		
⁰⁹⁵ Nb	1.092-13		
⁰⁹⁵ Zr	4.771-14		
¹⁶⁰ Tb	2.408-14		
⁰⁸⁹ Sr	8.377-23		
^{148M} Pm	2.486-24		
¹⁰³ Ru	4.093-25		
¹⁴⁸ Pm	7.098-26		
^{115M} Cd	9.343-28		
¹⁴¹ Ce	5.760-32		
^{129M} Te	6.401-32		

THE TRANSMUTATION CHAIN

Having selected ^{90}Sr and ^{137}Cs as potential transmutation targets, a transmutation chain for each isotope must be selected in order to proceed with the analysis. Along with β^- decay, at least two neutron reactions must be considered. They are the n,γ and $n,2n$ reactions. The former reaction usually has an appreciable cross section for most isotopes and is continuous over the energy range. The latter reaction is a threshold reaction and typically achieves a maximum in the neighborhood of 15 MeV and has a threshold in the neighborhood of 8 MeV. Other potential reactions, such as n,α or N,P , will be ignored in this analysis as being generally of secondary importance.

In the selection of a transmutation chain for neutron transmutation, a certain amount of arbitrariness is in order because, in principle, the chains can be open-ended. However, by comparison of the probable relative transfer rates one can arrive at a reasonable schematic of the chain. The chains selected for this analysis are shown in Figure 1.

By observing the transmutation chains shown in Figure 1 on a nuclide chart, one can see that they end in stable isotopes which are typically above and to the left of the target isotope on a chart of the nuclides. Therefore, one can perhaps say that an isotope which is bordered above and on its left in the nuclide chart is a prime candidate for neutron transmutation. Many of the fission products, being neutron rich, fit this description.

TRANSMUTATION CHAINS FOR Sr-90 AND Cs-137 IN A CONTROLLED THERMONUCLEAR REACTOR

FIGURE 1

THE NUCLEAR DATA

In order to accurately analyze the above transmutation chains, nuclear cross section data is required on all of the isotopes included in the chain. As a minimum, the reaction cross sections are required although a more thorough analysis will require the use of scattering data as well. Unfortunately, none of the above isotopes are represented on the ENDF/B-III library. Therefore, other sources need to be consulted. To perform the desired analysis, 100 group reaction cross sections in the GAM-II format were required. These 100 group cross section sets were obtained in various ways for each of the isotopes.

For the ^{137}Cs transmutation chain, all of the isotopic 100 group cross sections were obtained in the following manner. Data on the n,γ reaction cross was obtained and plotted on log-log paper. For the $n,2n$ reaction cross section, measured values were obtained from various sources at about 15 MeV and then, the cross section shape was calculated by means of the formalism of Pearlstein. The shape was normalized to the measured value and the results were also plotted on log-log paper. From these graphs, 100 group reaction cross sections for both the n,γ and $n,2n$ reaction cross sections were obtained. The sources of data for the data graphs are summarized by isotope in Table 5. The graphs themselves are shown in Figures 2 thru 7. The 100 group cross section sets for these isotopes are shown in Tables 6 thru 11.

^{137}Cs REACTION CROSS SECTIONS

FIGURE 2

^{136}Cs REACTION CROSS SECTIONS

FIGURE 3

^{135}Ba REACTION CROSS SECTION

FIGURE 4

^{136}Ba REACTION CROSS SECTIONS

FIGURE 5

^{137}Ba REACTION CROSS SECTIONS

FIGURE 6

^{138}Ba REACTION CROSS SECTIONS

FIGURE 7

TABLE 5
NUCLEAR DATA FOR ^{137}Cs TRANSMUTATION CHAIN

Isotope	Thermal n, γ Cross Section (0.025 eV)	Comment	Epithermal n, γ Cross Section	Comment	n,2n Measured Cross Section
^{137}Cs	Item 1, Vol. IIB, p. 55 - 137-1	$\frac{1}{V}$ up to 40 eV	Item 2	From 1000 eV to 10 MeV then $\frac{1}{V}$ back to 40 eV	Item 2, at 15 MeV
^{136}Cs	Item 3	$\frac{1}{V}$ to 50 eV	Item 3	-	-
^{135}Ba	Item 2	$\frac{1}{V}$ to 25 eV	Item 4	Theoretical evaluation from 25 eV	-
^{136}Ba	Item 1, Vol. IIB, p. 56 - 136-1	$\frac{1}{V}$ to 100 eV	Item 4	Theoretical evaluation from 100 eV	Item 1, Vol. IIB, p. 56 - 136-2
^{137}Ba	Item 2	$\frac{1}{V}$ to 400 eV	Item 4	Theoretical evaluation from 400 eV	-
^{138}Ba	Item 1, Vol. IIB, p. 56 - 138-1	$\frac{1}{V}$ to 1 KeV	Item 1, Vol. IIB, p. 56 - 138-4	Resonance Shapes Calculated	Item 1, Vol. IIB, p. 56 - 138-5

Item 1 Neutron Cross Sections BNL-325, Supplement No. 2, May 1966

Item 2 B. R. Leonard, Private Communication

Item 3 J. L. Cook, Fission Product Cross Sections, AAEC TM-549, June 1970

Item 4 Newsletter Bulletin 10, CCND-NW/10, December 1969

TABLE 6

¹³⁷Cs 100 GROUP REACTION CROSS SECTIONS

Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$
1	1.90-03	41	8.00-03	81	2.48-00
2	1.90-03	42	8.70-03	82	3.20-03
3	1.90-03	43	9.60-03	83	3.50-03
4	1.90-03	44	1.07-02	84	3.95-03
5	1.90-03	45	1.18-02	85	4.23-03
6	1.90-03	46	1.32-02	86	5.00-03
7	1.92-03	47	1.45-02	87	5.64-03
8	1.93-03	48	1.62-02	88	6.40-03
9	1.95-03	49	1.79-02	89	7.3 -03
10	1.97-03	50	2.10-02	90	8.2 -03
11	2.00-03	51	2.60-02	91	9.35-03
12	2.01-03	52	3.30-02	92	1.06-02
13	2.05-03	53	3.85-02	93	1.19-02
14	2.10-03	54	4.30-02	94	1.36-02
15	2.13-03	55	5.55-02	95	1.54-02
16	2.18-03	56	6.80-02	96	1.76-02
17	2.25-03	57	8.70-02	97	1.98-02
18	2.30-03	58	9.20-02	98	2.28-02
19	2.38-03	59	1.09-01	99	2.56-02
20	2.45-03	60	1.29-01	100	
21	2.53-03	61	1.53-01		
22	2.63-03	62	1.80-01		
23	2.75-03	63	2.12-01		
24	2.85-03	64	2.50-01		
25	3.00-03	65	2.92-01		
26	3.10-03	66	3.40-01		
27	3.25-03	67	4.00-01		
28	3.40-03	68	4.35-01		
29	3.60-03	69	5.50-01		
30	3.80-03	70	6.20-01		
31	4.00-03	71	7.00-01	Group	$\sigma_{n,2n-}$
32	4.22-03	72	8.00-01	1	1.5
33	4.50-03	73	9.00-01	2	1.38
34	4.80-03	74	1.02-00	3	1.18
35	5.10-03	75	1.08-00	4	0.66
36	5.50-03	76	1.31-00	5	0.27
37	5.90-03	77	1.48-00	6	0.10
38	6.40-03	78	1.70-00		
39	6.80-03	79	1.91-00		
40	7.40-03	80	2.20-00		

TABLE 7

¹³⁶Cs 100 GROUP REACTION CROSS SECTIONS

Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$
1	0.006	41	0.0623	81	0.0350
2	0.0025	42	0.0630	82	0.05098
3	0.0017	43	0.0635	83	0.05777
4	0.0012	44	0.0640	84	0.06546
5	0.00098	45	0.0650	85	0.07418
6	0.00095	46	0.0660	86	0.08405
7	0.00096	47	0.0662	87	0.09525
8	0.00101	48	0.0675	88	0.1079
9	0.00114	49	0.0690	89	0.1223
10	0.00163	50	0.0720	90	0.1386
11	0.0027	51	0.0870	91	0.1570
12	0.0042	52	0.103	92	0.1779
13	0.0062	53	0.125	93	0.2016
14	0.0085	54	0.148	94	0.2285
15	0.0110	55	0.168	95	0.2589
16	0.0136	56	0.217	96	0.2934
17	0.0162	57	0.247	97	0.3324
18	0.0186	58	0.285	98	0.3767
19	0.0214	59	0.330	99	0.4269
20	0.0229	60	0.380	100	1.8563
21	0.0238	61	0.440		
22	0.0242	62	0.510		
23	0.0245	63	0.700		
24	0.0245	64	0.860		
25	0.0240	65	0.930		
26	0.0227	66	0.680		
27	0.0218	67	0.200		
28	0.0218	68	0.192		
29	0.0220	69	0.096		
30	0.0250	70	0.360		
31	0.0360	71	0.1039		
32	0.0490	72	0.740		
33	0.0558	73	0.0152		
34	0.0595	74	0.0380		
35	0.0615	75	0.0360		
36	0.0615	76	0.0560		
37	0.0618	77	0.0660		
38	0.0620	78	0.0400		
39	0.0620	79	0.0350		
40	0.0620	80	0.0350		

TABLE 8

¹³⁵Ba 100 GROUP REACTION CROSS SECTIONS

Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$
1	0.800	41	0.133	81	51.0
2	0.270	42	0.140	82	63.0
3	0.102	43	0.145	83	77.0
4	0.050	44	0.150	84	0.307
5	0.014	45	0.155	85	0.347
6	0.004	46	0.161	86	0.394
7	0.0019	47	0.167	87	0.446
8	0.0011	48	0.174	88	0.506
9	0.00108	49	0.183	89	0.573
10	0.00132	50	0.199	90	0.649
11	0.00162	51	0.245	91	0.736
12	0.00215	52	0.251	92	0.834
13	0.00277	53	0.290	93	0.945
14	0.00340	54	0.335	94	1.071
15	0.00440	55	0.390	95	1.213
16	0.00560	56	0.465	96	1.375
17	0.00700	57	0.530	97	1.558
18	0.00850	58	0.620	98	1.766
19	0.01220	59	0.720	99	2.001
20	0.01250	60	0.850	100	8.7
21	0.01500	61	1.01		
22	0.0180	62	1.22		
23	0.0212	63	1.46		
24	0.0245	64	1.77		
25	0.0285	65	2.15		
26	0.0315	66	2.60		
27	0.0350	67	3.18		
28	0.0380	68	3.90		
29	0.0400	69	4.70		
30	0.0430	70	5.80		
31	0.0460	71	7.00		
32	0.0520	72	8.60		
33	0.0610	73	10.4		
34	0.0700	74	12.8		
35	0.0810	75	15.5		
36	0.0900	76	19.0		
37	0.10	77	23.0		
38	0.11	78	28.0		
39	0.118	79	34.5		
40	0.127	80	42.0		

TABLE 9

¹³⁶Ba 100 GROUP REACTION CROSS SECTIONS

Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$
1	0.012	41	0.04	81	0.000293
2	0.006	42	0.0402	82	0.000330
3	0.0032	43	0.0410	83	0.000373
4	0.0016	44	0.0420	84	0.000423
5	0.00095	45	0.0425	85	0.000480
6	0.000535	46	0.0435	86	0.000543
7	0.000495	47	0.0440	87	0.000616
8	0.0006	48	0.0450	88	0.000698
9	0.00082	49	0.0460	89	0.000791
10	0.00124	50	0.0478	90	0.000896
11	0.00184	51	0.054	91	0.00102
12	0.00276	52	0.052	92	0.00115
13	0.0038	53	0.061	93	0.00130
14	0.0052	54	0.070	94	0.00148
15	0.0070	55	0.0815	95	0.00167
16	0.0090	56	0.1000	96	0.00190
17	0.0112	57	0.116	97	0.00215
18	0.0133	58	0.136	98	0.00244
19	0.0152	59	0.157	99	0.00276
20	0.0154	60	0.178	100	0.012
21	0.015	61	0.200		
22	0.0135	62	0.228		
23	0.0123	63	0.250		
24	0.0121	64	0.274		
25	0.0122	65	0.295		
26	0.0125	66	0.340		
27	0.0131	67	0.390		
28	0.0155	68	0.460		
29	0.0235	69	0.520		
30	0.0481	70	0.580		
31	0.0460	71	0.660	Group	$\sigma_{n,2n}$
32	0.0440	72	0.760	1	0.68
33	0.0422	73	0.845	2	0.58
34	0.0406	74	0.960	3	0.46
35	0.04	75	1.080	4	0.17
36	0.0398	76	1.200	5	0.0025
37	0.0394	77	1.330	6	-
38	0.0391	78	0.0002		
39	0.0394	79	0.00023		
40	0.0395	80	0.000258		

TABLE 10

¹³⁷Ba 100 GROUP REACTION CROSS SECTIONS

Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$
1	0.0290	41	0.0339	81	0.00969
2	0.0130	42	0.0350	82	0.0110
3	0.0056	43	0.0360	83	0.0124
4	0.0025	44	0.0375	84	0.0141
5	0.00105	45	0.0390	85	0.0160
6	0.00044	46	0.0404	86	0.0181
7	0.00027	47	0.0412	87	0.0205
8	0.00027	48	0.0425	88	0.0233
9	0.00033	49	0.0433	89	0.0264
10	0.00046	50	0.0451	90	0.0299
11	0.00063	51	0.0575	91	0.0338
12	0.00082	52	0.0660	92	0.0383
13	0.00103	53	0.0760	93	0.0434
14	0.00134	54	0.0920	94	0.0492
15	0.00165	55	0.1060	95	0.0558
16	0.00200	56	0.131	96	0.0632
17	0.00255	57	0.150	97	0.0716
18	0.00480	58	0.177	98	0.0812
19	0.00575	59	0.210	99	0.0920
20	0.00615	60	0.246	100	0.400
21	0.00670	61	0.290		
22	0.00740	62	0.340		
23	0.00840	63	0.395		
24	0.0108	64	0.470		
25	0.0155	65	0.535		
26	0.0183	66	0.640		
27	0.0205	67	0.780		
28	0.0219	68	0.970		
29	0.0225	69	1.120		
30	0.0240	70	1.300		
31	0.0256	71	1.500		
32	0.0290	72	1.720		
33	0.0320	73	0.00357		
34	0.0320	74	0.00404		
35	0.0319	75	0.00458		
36	0.0320	76	0.00519		
37	0.0321	77	0.00588		
38	0.0322	78	0.00666		
39	0.0323	79	0.00755		
40	0.0330	80	0.00856		

TABLE 11

¹³⁸Ba 100 GROUP REACTION CROSS SECTIONS

Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$
1	0.0	41	0.00180	81	0.00848
2	0.0	42	0.00188	82	0.00961
3	0.0	43	0.00196	83	0.0109
4	0.0	44	0.00210	84	0.0123
5	0.0	45	0.00224	85	0.0140
6	0.0	46	0.00240	86	0.0158
7	0.0	47	0.00252	87	0.0180
8	0.0	48	0.00270	88	0.0203
9	0.0	49	0.00290	89	0.0231
10	0.0	50	0.00322	90	0.0261
11	0.000127	51	0.00395	91	0.0296
12	0.00018	52	0.00480	92	0.0336
13	0.00024	53	0.00580	93	0.0380
14	0.00035	54	0.00710	94	0.0431
15	0.00049	55	0.00860	95	0.0488
16	0.00067	56	0.01010	96	0.0553
17	0.00093	57	0.01210	97	0.0627
18	0.00135	58	0.0145	98	0.0710
19	0.002	59	0.0175	99	0.0805
20	0.00275	60	0.0210	100	0.35
21	0.0036	61	0.0250		
22	0.00395	62	0.0300		
23	0.00410	63	0.0360		
24	0.00405	64	0.0430		
25	0.00385	65	0.0520		
26	0.00370	66	0.062		
27	0.00348	67	0.00147		
28	0.0032	68	0.00167		
29	0.00295	69	0.00189		
30	0.00270	70	0.00214		
31	0.00244	71	0.00243	Group	$\sigma_{n,2n}$
32	0.00220	72	0.00275	1	1.23
33	0.00200	73	0.00312	2	1.08
34	0.00190	74	0.00354	3	0.86
35	0.00179	75	0.00401	4	0.58
36	0.00173	76	0.00454	5	0.056
37	0.00171	77	0.00515	6	0.008
38	0.00170	78	0.00583	7	-
39	0.00171	79	0.00661		
40	0.00176	80	0.00749		

With regard to the ^{90}Sr transmutation chain, the sources for the data are summarized in Table 12. ^{90}Sr and ^{89}Y n, γ cross sections were obtained in a manner similar to those in the ^{137}Cs chain. The n, γ cross sections for the other isotopes were obtained by taking Cook's data and processing it directly into a GAM-II format with ETØG. All of the n,2n reaction cross sections in the ^{90}Sr burnup chain were obtained by the method described for the ^{137}Cs burnup chain. The reaction data for ^{90}Sr and ^{137}Cs is shown in Figures 8 and 9. Because of the direct processing of the other isotopes, there aren't any equivalent figures. The n,2n cross sections data for ^{91}Zr and ^{92}Zr is displayed in Figure 10. The 100 group reaction cross sections are given in Tables 13 thru 20.

By and large, the isotopes of interest to this study can be characterized as ones about which one can find relatively little nuclear data. There are exceptions, however. For instance, there is a wealth of data on ^{90}Zr . Therefore, the reader may find it curious that Cook's theoretical evaluation was used. In the final analysis, this data was used in order to be consistent with the other lesser known isotopes. It is felt that, particularly with a relative analysis such as this one, the overall consistency of the data is of primary importance.

TABLE 12

NUCLEAR DATA FOR ^{90}Sr TRANSMUTATION CHAIN

Isotope	Thermal n, γ Cross Section (0.025 eV)	Comment	Epithermal n, γ Cross Section	Comment	n,2n Measured Cross Section
^{90}Sr	Item 2	$\frac{1}{V}$ to 40 eV	Item 2	-	Item 2, at 15 MeV
^{89}Sr	Item 3	$\frac{1}{V}$ to cutoff	Item 3	Processed with ETOG	-
^{89}Y	Item 1, Vol. IIA, p. 39-0-1	$\frac{1}{V}$ to 1 KeV	Item 4	-	-
^{90}Y	Item 3	$\frac{1}{V}$ to cutoff	Item 3	Processed with ETOG	-
^{91}Y	Item 3	$\frac{1}{V}$ to cutoff	Item 3	Processed with ETOG	-
^{90}Zr	Item 3	$\frac{1}{V}$ to cutoff	Item 3	Processed with ETOG	-
^{91}Zr	Item 3	$\frac{1}{V}$ to cutoff	Item 3	Processed with ETOG	Item 2, at 15 MeV
^{92}Zr	Item 3	$\frac{1}{V}$ to cutoff	Item 3	Processed with ETOG	Item 2, at 15 MeV

Item 1 Neutron Cross Sections BNL-325, Supplement No. 2, May 1966

Item 2 B. R. Leonard Private Communication

Item 3 J. L. Cook, Fission Product Cross Sections, AAEC TM-549, June 1970

Item 4 Newsletter Bulletin 10, CCND-NW/10, December 1969

TABLE 13

⁹⁰Sr 100 GROUP REACTION CROSS SECTIONS

Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$
1	1.32-03	41	7.60-03	81	7.20-01
2	1.35-03	42	8.00-03	82	2.20-02
3	1.38-03	43	8.50-03	83	2.50-02
4	1.40-03	44	9.00-03	84	2.83-02
5	1.42-03	45	9.50-03	85	3.20-02
6	1.47-03	46	1.02-02	86	3.60-02
7	1.50-03	47	1.08-02	87	4.02-02
8	1.53-03	48	1.14-02	88	4.06-02
9	1.58-03	49	1.20-02	89	5.21-02
10	1.63-03	50	1.31-02	90	5.90-02
11	1.69-03	51	1.50-02	91	6.75-02
12	1.76-03	52	1.71-02	92	7.60-02
13	1.82-03	53	1.96-02	93	8.60-02
14	1.90-03	54	2.22-02	94	9.80-02
15	1.98-03	55	2.56-02	95	1.12-01
16	2.06-03	56	2.90-02	96	1.27-01
17	2.15-03	57	3.30-02	97	1.42-01
18	2.24-03	58	3.80-02	98	1.60-01
19	2.35-03	59	4.35-02	99	1.82-01
20	2.46-03	60	4.95-02	100	0.72-01
21	2.58-03	61	5.65-02		
22	2.70-03	62	6.50-02		
23	2.84-03	63	7.28-02		
24	3.00-03	64	8.40-02		
25	3.17-03	65	9.70-01		
26	3.31-03	66	1.09-01		
27	3.48-03	67	1.20-01		
28	3.68-03	68	1.42-01		
29	3.86-03	69	1.60-01		
30	4.10-03	70	1.80-01		
31	4.39-03	71	2.08-01	Group	$\sigma_{n,2n}$
32	4.60-03	72	2.33-01	1	1.50
33	4.87-03	73	2.66-01	2	1.38
34	5.10-03	74	3.00-01	3	1.18
35	5.45-03	75	3.40-01	4	0.85
36	5.75-03	76	3.80-01	5	0.35
37	6.05-03	77	4.20-01	6	0.10
38	6.40-03	78	5.00-01		
39	6.80-03	79	5.60-01		
40	7.20-03	80	6.40-01		

TABLE 14

⁸⁹Y 100 GROUP REACTION CROSS SECTIONS

Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$
1	0.0025	41	0.00940	81	0.0310
2	0.0009	42	0.00960	82	0.0352
3	0.00054	43	0.00995	83	0.0398
4	0.00034	44	0.0102	84	0.0451
5	0.000285	45	0.0106	85	0.0511
6	0.000281	46	0.0110	86	0.0580
7	0.000326	47	0.0114	87	0.0657
8	0.000380	48	0.0119	88	0.0744
9	0.000430	49	0.0123	89	0.0843
10	0.000572	50	0.0132	90	0.0956
11	0.000650	51	0.0145	91	0.108
12	0.000850	52	0.0161	92	0.123
13	0.001220	53	0.0184	93	0.139
14	0.00330	54	0.0216	94	0.158
15	0.00360	55	0.0250	95	0.179
16	0.00345	56	0.0300	96	0.202
17	0.00340	57	0.0353	97	0.229
18	0.00340	58	0.0430	98	0.260
19	0.00345	59	0.0520	99	0.294
20	0.00350	60	0.0630	100	1.28
21	0.00360	61	0.0750		
22	0.00390	62	0.0876		
23	0.00500	63	0.1000		
24	0.00540	64	0.119		
25	0.00564	65	0.140		
26	0.00590	66	0.153		
27	0.00630	67	0.175		
28	0.00800	68	0.196		
29	0.00960	69	0.00692		
30	0.00920	70	0.00784		
31	0.00895	71	0.00889		
32	0.00880	72	0.01007		
33	0.00865	73	0.0114		
34	0.00860	74	0.0129		
35	0.00860	75	0.0147		
36	0.00860	76	0.0166		
37	0.00875	77	0.0188		
38	0.00880	78	0.0213		
39	0.00900	79	0.0242		
40	0.00918	80	0.0274		

TABLE 15

⁸⁹Sr 100 GROUP REACTION CROSS SECTIONS

Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$
1	0.0000	41	4.2948-03	81	8.8314-03
2	0.0000	42	4.4149-03	82	1.0680-02
3	0.0000	43	4.5383-03	83	1.2768-02
4	0.0000	44	4.6652-03	84	1.5096-02
5	7.4697-07	45	4.7956-03	85	1.7684-02
6	1.2123-04	46	4.9297-03	86	2.0582-02
7	1.4489-04	47	5.0675-03	87	2.3788-02
8	1.6888-04	48	5.2091-03	88	2.7461-02
9	1.9916-04	49	5.3547-03	89	3.1362-02
10	2.3544-04	50	5.7045-03	90	3.6094-02
11	2.7866-04	51	6.7168-03	91	4.1002-02
12	4.3551-04	52	8.0681-03	92	4.7012-02
13	8.2556-04	53	9.8287-03	93	5.3264-02
14	1.2545-03	54	1.2076-02	94	6.0904-02
15	1.3193-03	55	1.5019-02	95	6.8854-02
16	1.3766-03	56	1.8777-02	96	7.8588-02
17	1.4152-03	57	2.3543-02	97	8.8783-02
18	1.4294-03	58	2.9548-02	98	1.0123-01
19	1.4438-03	59	3.7064-02	99	1.1424-01
20	1.4583-03	60	4.6253-02	100	
21	1.5195-03	61	5.7638-02		
22	1.6567-03	62	6.7919-02		
23	1.8068-03	63	6.5927-02		
24	1.9705-03	64	1.7610-04		
25	2.1490-03	65	2.7219-03		
26	2.3437-03	66	2.9059-01		
27	2.5561-03	67	4.5775-04		
28	2.7298-03	68	2.7112-04		
29	2.8476-03	69	2.7148-04		
30	2.9704-03	70	3.3921-04		
31	3.0986-03	71	4.6184-04		
32	3.2322-03	72	6.5926-04		
33	3.3717-03	73	9.4744-04		
34	3.5170-03	74	1.3469-03		
35	3.6400-03	75	1.8874-03		
36	3.7417-03	76	2.5792-03		
37	3.8463-03	77	3.4449-03		
38	3.9539-03	78	4.4988-03		
39	4.0644-03	79	5.7453-03		
40	4.1780-03	80	7.1811-03		

TABLE 16

 ^{90}Y 100 GROUP REACTION CROSS SECTIONS

Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$
1	0.0000	41	8.4690-03	81	2.8242-02
2	0.0000	42	8.6528-03	82	3.8654-02
3	0.0000	43	8.8407-03	83	5.1634-02
4	0.0000	44	9.0325-03	84	6.7347-02
5	5.3000-06	45	9.2234-03	85	8.5884-02
6	7.2873-04	46	9.4236-03	86	1.0726-01
7	9.1447-04	47	9.6331-03	87	1.3179-01
8	1.1242-03	48	9.8420-03	88	1.5993-01
9	1.3702-03	49	1.0056-02	89	1.8962-01
10	1.6676-03	50	1.0641-02	90	2.2556-01
11	2.0305-03	51	1.2721-02	91	2.6254-01
12	2.6703-03	52	1.5661-02	92	3.0719-01
13	3.6951-03	53	1.9420-02	93	2.0702-01
14	4.5825-03	54	2.4195-02	94	0.0000
15	4.7458-03	55	3.0311-02	95	0.0000
16	4.8993-03	56	3.8041-02	96	0.0000
17	5.7503-03	57	4.7660-02	97	0.0000
18	7.7639-03	58	5.9509-02	98	0.0000
19	1.0491-02	59	7.4133-02	99	0.0000
20	1.4177-02	60	9.1751-02	100	
21	1.6271-02	61	1.1337-01		
22	1.4836-02	62	1.3840-01		
23	1.3591-02	63	1.5099-01		
24	1.2409-02	64	3.0827-01		
25	1.1330-02	65	3.3612-01		
26	1.0346-02	66	4.0421-01		
27	9.4465-03	67	4.1320-01		
28	8.8746-03	68	7.3690-01		
29	8.6217-03	69	2.9575-03		
30	8.3763-03	70	3.5793-03		
31	8.1378-03	71	2.2995+00		
32	7.9061-03	72	1.4405-02		
33	7.6810-03	73	3.9270-03		
34	7.4620-03	74	3.2730-03		
35	7.4454-03	75	3.5680-03		
36	7.6070-03	76	4.7445-03		
37	7.7721-03	77	6.7207-03		
38	7.9497-03	78	9.7190-03		
39	8.1130-03	79	1.4088-02		
40	8.2891-03	80	2.0128-02		

TABLE 17

⁹¹Y 100 GROUP REACTION CROSS SECTIONS

Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$
1	0.0000	41	1.1134-02	81	1.3304-02
2	0.0000	42	1.1633-02	82	1.7039-02
3	0.0000	43	1.2101-02	83	2.1476-02
4	0.0000	44	1.2586-02	84	2.6535-02
5	1.2572-06	45	1.3092-02	85	3.2252-02
6	1.5078-04	46	1.3617-02	86	3.8670-02
7	1.6475-04	47	1.4154-02	87	4.5919-02
8	1.7903-04	48	1.4732-02	88	5.4130-02
9	2.1572-04	49	1.5325-02	89	6.2825-02
10	2.6435-04	50	1.6579-02	90	7.3306-02
11	3.2519-04	51	2.0015-02	91	8.4098-02
12	3.9357-04	52	2.4945-02	92	9.7205-02
13	4.7175-04	53	3.0932-02	93	1.1031-01
14	5.5864-04	54	3.8711-02	94	4.8272-02
15	6.4842-04	55	4.8342-02	95	0.0000
16	7.5233-04	56	6.0228-02	96	0.0000
17	1.0642-03	57	7.4734-02	97	0.0000
18	1.8446-03	58	9.2165-02	98	0.0000
19	3.2014-03	59	1.1301-01	99	0.0000
20	5.5568-03	60	1.3715-01	100	
21	7.6213-03	61	1.6602-01		
22	7.6789-03	62	2.0936-01		
23	7.7182-03	63	2.5204-01		
24	7.7577-03	64	2.7121-01		
25	7.7973-03	65	3.4204-01		
26	7.8372-03	66	4.6629-01		
27	7.8772-03	67	5.9092-01		
28	7.9469-03	68	7.5680-01		
29	8.0513-03	69	2.7491-01		
30	8.1572-03	70	8.9033-03		
31	8.2644-03	71	2.2304+00		
32	8.3731-03	72	4.6454-03		
33	8.4832-03	73	2.3251-03		
34	8.5949-03	74	2.1407-03		
35	8.6304-03	75	2.4619-03		
36	9.1850-03	76	3.1479-03		
37	9.5539-03	77	4.2202-03		
38	9.9375-03	78	5.7223-03		
39	1.0337-02	79	7.6932-03		
40	1.0752-02	80	1.0243-02		

TABLE 18

⁹⁰Zr 100 GROUP REACTION CROSS SECTIONS

Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$
1	0.0000	41	4.6980-03	81	2.3396-03
2	0.0000	42	4.8006-03	82	2.6733-03
3	0.0000	43	4.9049-03	83	3.0491-03
4	0.0000	44	5.0113-03	84	3.4727-03
5	3.0186-06	45	5.1200-03	85	3.9507-03
6	4.0435-04	46	5.2310-03	86	4.4906-03
7	5.0735-04	47	5.3445-03	87	5.1096-03
8	6.2372-04	48	5.4604-03	88	5.8063-03
9	7.6022-04	49	5.5791-03	89	6.5565-03
10	9.2522-04	50	5.8601-03	90	7.4749-03
11	1.1266-03	51	6.7451-03	91	8.4360-03
12	1.4815-03	52	7.2137-03	92	9.6131-03
13	2.0501-03	53	7.8444-03	93	1.0845-02
14	2.5424-03	54	1.0709-02	94	1.2356-02
15	2.6336-03	55	1.4064-02	95	1.3936-02
16	2.7182-03	56	1.6847-02	96	1.5874-02
17	3.1904-03	57	4.0959-03	97	1.7903-02
18	4.3076-03	58	3.9630-03	98	2.0349-02
19	5.8200-03	59	1.0537-03	99	2.2994-02
20	7.8655-03	60	1.2736-05	100	
21	9.0276-03	61	1.7727-05		
22	8.2589-03	62	2.6678-05		
23	7.5406-03	63	1.2968-02		
24	6.8847-03	64	5.7861-05		
25	6.2862-03	65	8.1983-05		
26	5.7399-03	66	1.1624-04		
27	5.2410-03	67	1.6076-04		
28	4.9237-03	68	2.1665-04		
29	4.7834-03	69	2.8465-04		
30	4.6472-03	70	3.6524-04		
31	4.5149-03	71	4.5833-04		
32	4.3863-03	72	5.6576-04		
33	4.2614-03	73	6.8662-04		
34	4.1402-03	74	8.2222-04		
35	4.1307-03	75	9.7344-04		
36	4.2294-03	76	1.1430-03		
37	4.3120-03	77	1.3316-03		
38	4.4055-03	78	1.5426-03		
39	4.5011-03	79	1.7745-03		
40	4.5986-03	80	2.0426-03		

TABLE 19

^{91}Zr 100 GROUP REACTION CROSS SECTIONS

Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$
1	0.0000	41	1.5369-02	81	3.5912-02
2	0.0000	42	1.5987-02	82	4.1146-02
3	0.0000	43	1.6629-02	83	4.7128-02
4	0.0000	44	1.7297-02	84	5.3916-02
5	1.7043-06	45	1.7991-02	85	6.1595-02
6	2.0720-04	46	1.8713-02	86	7.0268-02
7	2.2641-04	47	1.9464-02	87	8.0060-02
8	2.4603-04	48	2.0246-02	88	9.1372-02
9	2.9645-04	49	2.1060-02	89	1.0338-01
10	3.6396-04	50	2.2958-02	90	1.1807-01
11	4.4687-04	51	2.7981-02	91	1.3342-01
12	5.4085-04	52	3.4949-02	92	1.5220-01
13	6.4829-04	53	4.2901-02	93	1.7184-01
14	7.6771-04	54	5.5976-02	94	1.9590-01
15	8.9107-04	55	7.1801-02	95	2.2108-01
16	1.0339-03	56	9.3673-02	96	8.9367-04
17	1.4625-03	57	1.1955-01	97	0.0000
18	2.5349-03	58	1.4999-01	98	0.0000
19	4.3996-03	59	1.8782-01	99	0.0000
20	7.6366-03	60	2.7089-01	100	
21	1.0474-02	61	3.2734-01		
22	1.0553-02	62	8.0194-04		
23	1.0607-02	63	6.5372-04		
24	1.0661-02	64	3.0230-01		
25	1.0716-02	65	8.0863-01		
26	1.0770-02	66	6.0937-01		
27	1.0825-02	67	1.0770+00		
28	1.0921-02	68	5.6033-03		
29	1.1065-02	69	3.9387-01		
30	1.1210-02	70	4.6714+00		
31	1.1357-02	71	1.2106-02	Group	$\sigma_{n,2n}$
32	1.1507-02	72	2.0146-02	1	1.39
33	1.1658-02	73	1.5683+01	2	1.31
34	1.1811-02	74	7.1914-01	3	1.20
35	1.2135-02	75	2.6252+00	4	1.03
36	1.2622-02	76	2.9991-02	5	0.64
37	1.3129-02	77	2.3557-02	6	0.25
38	1.3657-02	78	2.4823-02	7	0.052
39	1.4205-02	79	2.7650-02	8	0.0
40	1.4776-02	80	3.1402-02		

TABLE 20

⁹²Zr 100 GROUP REACTION CROSS SECTIONS

Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$	Group	$\sigma_{n,\gamma}$
1	0.0000	41	1.0005-02	81	5.0568-03
2	0.0000	42	1.0188-02	82	5.9551-03
3	0.0000	43	1.0375-02	83	6.9627-03
4	0.0000	44	1.0565-02	84	8.0912-03
5	1.6250-06	45	1.0758-02	85	9.3554-03
6	2.2043-04	46	1.0954-02	86	1.0772-02
7	2.9858-04	47	1.1155-02	87	1.2362-02
8	3.9715-04	48	1.1359-02	88	1.4189-02
9	5.2823-04	49	1.1567-02	89	1.6121-02
10	7.0264-04	50	1.2031-02	90	1.8476-02
11	9.3463-04	51	1.4734-02	91	2.0930-02
12	1.2078-03	52	1.7076-02	92	2.3928-02
13	1.5314-03	53	8.3462-03	93	2.7058-02
14	1.8500-03	54	2.0522-05	94	3.0885-02
15	2.0633-03	55	3.8747-05	95	3.4887-02
16	2.2976-03	56	4.3531-03	96	3.9785-02
17	2.4485-03	57	7.9557-03	97	4.4909-02
18	2.4827-03	58	1.0838-02	98	5.1184-02
19	2.5173-03	59	1.2925-04	99	5.7752-02
20	2.5524-03	60	2.2053-03	100	
21	2.6940-03	61	4.6026-02		
22	3.0097-03	62	1.0346-01		
23	3.3646-03	63	1.3107-01		
24	3.7614-03	64	2.9844-01		
25	4.2046-03	65	1.3233-02		
26	4.6999-03	66	1.4397-03		
27	5.2537-03	67	8.3639-04		
28	5.7519-03	68	6.7691-04		
29	6.1524-03	69	6.4035-04		
30	6.5805-03	70	6.6657-04		
31	7.0385-03	71	7.4127-04		
32	7.5284-03	72	8.6401-04		
33	8.0524-03	73	1.0396-03		
34	8.6126-03	74	1.2750-03		
35	8.9743-03	75	1.5770-03		
36	9.1384-03	76	1.9516-03		
37	9.3056-03	77	2.4035-03		
38	9.4758-03	78	2.9363-03		
39	9.6491-03	79	3.5534-03		
40	9.8256-03	80	4.2585-03		
				Group	$\sigma_{n,2n}$
				1	1.3
				2	1.12
				3	0.86
				4	0.50
				5	0.19
				6	0.0

^{90}Sr REACTION CROSS SECTIONS

FIGURE 8

^{89}Y REACTION CROSS SECTIONS

FIGURE 9

^{91}Zr AND ^{92}Zr N,2N CROSS SECTIONS

FIGURE 10

The 100-group cross-section sets developed in this manner have to be collapsed into effective one-energy group cross sections in order to perform the transmutation analysis. This collapsing was performed in the following manner.

A CTR blanket, labeled the standard blanket for identification purposes, has been the subject of intense neutronic analysis by several groups both in the United States and abroad. This blanket, while not necessarily optimum, was developed in 1971 at the ORNL international working session on the engineering problems of CTR power plants by an ad-hoc committee representing the CTR neutronics community. It has since been used for purposes of comparison of the different analytical techniques of various laboratories.

It was decided to use this standard blanket as the vehicle for transmutation of the fission products. Therefore, using the ANISN code, the hundred group cross-section sets for the isotopes of interest were collapsed over region 6 of this blanket. The blanket is shown in Figure 11.

The collapsing calculation was performed for a small isotopic content and therefore the isotopes did not have a material effect on the spectrum of the blanket. The effect of this assumption will be tested in later analysis. As we shall see, the decision to load the fission products in region 6 was not particularly astute and later analysis will be performed with fission products in different blanket locations. The effective one-energy group constants obtained in this manner were then used to solve the differential equations of transmutation for Sr^{90} and Cs^{137} . The values obtained from this analysis are reported in the next section.

STANDARD BLANKET CONFIGURATION

DISTANCES IN cm	0	150	200	200.5	203.5	204				264	294	300
ORIGIN	PLASMA	VAC.	Nb	Li 94		Nb		Li 94		C		Li 94
				6 Nb				6 Nb				6 Nb
							20 cm	20 cm	20 cm			
ZONE NUMBER	1	2	3	4	5			6		7	8	
REGION NUMBER	1	2	3	4	5	6	7	8		9	10	
MATERIAL	A	B	C	D	C			D		E	D	
NUMBER OF INTERVALS PER ZONE	1	1	3	6	3			30		15	3	
THICKNESS (cm)	150	50	0.5	3	0.5			60		30	6	

COMMENT: THE INTERVALS IN EACH ZONE ARE OF EQUAL STEP LENGTH. THERE ARE A TOTAL OF 62 INTERVALS IN THE PROBLEM.

FIGURE 11

THE TRANSMUTATION CALCULATION

The transmutation chains displayed in Figure 1 can be described by sets of coupled linear differential equations. Once described in equation form, they can then be solved for isotopic densities as a function of fluence. The differential equation set for ^{137}Cs transmutation is:

$$\frac{dN_1}{dt} = -N_1 \left[(\sigma_1^1 + \sigma_1^2) \phi + \lambda_1 \right] + N_2 \sigma_2^1 \phi$$

$$\frac{dN_2}{dt} = N_1 \sigma_1^2 \phi - N_2 (\sigma_2^1 \phi + \lambda_2)$$

$$\frac{dN_3}{dt} = N_1 \sigma_1^1 \phi - N_3 \lambda_3$$

$$\frac{dN_4}{dt} = N_2 \lambda_2 + N_5 \sigma_5^1 \phi - N_4 (\sigma_4^1 + \sigma_4^2) \phi$$

$$\frac{dN_5}{dt} = N_4 \sigma_4^2 \phi - N_5 \sigma_5^1 \phi$$

$$\frac{dN_6}{dt} = N_1 \lambda_1 + N_4 \sigma_4^1 \phi + N_7 \sigma_7^2 \phi - N_6 \sigma_6^1 \phi$$

$$\frac{dN_7}{dt} = N_3 \lambda_3 + N_6 \sigma_6^1 \phi - N_7 (\sigma_7^1 + \sigma_7^2) \phi$$

$$\frac{dN_8}{dt} = N_7 \sigma_7^1 \phi - N_8 \lambda_8$$

$$\frac{dN_9}{dt} = N_8 \lambda_8$$

where:

$$N_1 = {}^{137}\text{Cs concentration}$$

$$N_2 = {}^{136}\text{Cs concentration}$$

$$N_3 = {}^{138}\text{Cs concentration}$$

$$N_4 = {}^{136}\text{Ba concentration}$$

$$N_5 = {}^{135}\text{Ba concentration}$$

$$N_6 = {}^{137}\text{Ba concentration}$$

$$N_7 = {}^{138}\text{Ba concentration}$$

$$N_8 = {}^{139}\text{Ba concentration}$$

$$N_9 = {}^{139}\text{La concentration}$$

and:

$$\sigma_i^1 = n, \gamma \text{ reaction cross section for isotope } i$$

$$\sigma_i^2 = n, 2n \text{ reaction cross section for isotope } i$$

$$\lambda_i = \text{decay constant for isotope } i$$

$$\phi = \text{average neutron flux during irradiation}$$

The differential equation set for ${}^{90}\text{Sr}$ transmutation is:

$$\frac{dN_1}{dt} = -N_1 \left[(\sigma_1^1 + \sigma_1^2) \phi + \lambda_1 \right] + N_2 \sigma_2^1 \phi$$

$$\frac{dN_2}{dt} = N_1 \sigma_1^2 \phi - N_2 (\sigma_2^1 \phi + \lambda_2)$$

$$\frac{dN_3}{dt} = N_1 \sigma_1^1 \phi - N_3 \lambda_3$$

$$\frac{dN_4}{dt} = N_2 \lambda_2 - N_4 \sigma_4^1 \phi$$

$$\frac{dN_5}{dt} = N_1\lambda_1 + N_4\sigma_4^1\phi - N_5(\lambda_5 + \sigma_5^1\phi)$$

$$\frac{dN_6}{dt} = N_3\lambda_3 + N_5\sigma_5^1\phi - N_6(\lambda_6 + \sigma_6^1\phi)$$

$$\frac{dN_7}{dt} = N_6\sigma_6^1\phi - N_7\lambda_7$$

$$\frac{dN_8}{dt} = N_5\lambda_5 + N_9\sigma_9^2\phi - N_8\sigma_8^1\phi$$

$$\frac{dN_9}{dt} = N_6\lambda_6 + N_8\sigma_8^1\phi + N_{10}\sigma_{10}^2\phi - N_9(\sigma_9^1 + \sigma_9^2)\phi$$

$$\frac{dN_{10}}{dt} = N_7\lambda_7 + N_9\sigma_9^1\phi - N_{10}\sigma_{10}^2\phi$$

where:

N_1 = ^{90}Sr concentration

N_2 = ^{89}Sr concentration

N_3 = ^{91}Sr concentration

N_4 = ^{89}Y concentration

N_5 = ^{90}Y concentration

N_6 = ^{91}Y concentration

N_7 = ^{92}Y concentration

N_8 = ^{90}Zr concentration

N_9 = ^{91}Zr concentration

N_{10} = ^{92}Zr concentration

and the other symbols are defined in the same manner as those used in the ^{137}Cs transmutation chain.

The preceding equations, when written in matrix notation, can be solved by the ALKEMY program. The ALKEMY program has the capability of being used as a generalized coupled linear differential equation set problem solver. The preceding equations, as written in matrix notation, are

for ^{137}Cs :

$$\begin{bmatrix} \dot{N}_1 \\ \dot{N}_2 \\ \dot{N}_3 \\ \dot{N}_4 \\ \dot{N}_5 \\ \dot{N}_6 \\ \dot{N}_7 \\ \dot{N}_8 \\ \dot{N}_9 \end{bmatrix} = \begin{bmatrix} -[(\sigma_1^1 + \sigma_1^2)\phi + \lambda_1] & \sigma_2^1\phi & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ \sigma_2^2\phi & (-\lambda_2 - \sigma_2^1\phi) & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ \sigma_1^1\phi & 0 & -\lambda_3 & 0 & 0 & 0 & 0 & 0 & 0 \\ 0 & \lambda_2 & 0 & -(\sigma_4^1 + \sigma_2^2)\phi & \frac{1}{5} & 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & \sigma_4^2\phi & -\sigma_5^1\phi & 0 & 0 & 0 & 0 \\ \lambda_1 & 0 & 0 & \sigma_4^1\phi & 0 & -\sigma_6^1\phi & \sigma_7^2\phi & 0 & 0 \\ 0 & 0 & \lambda_3 & 0 & 0 & \sigma_6^1\phi & -(\sigma_7^1 + \sigma_7^2)\phi & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 & 0 & \sigma_7^1\phi & -\lambda_8 & 0 \\ 0 & 0 & 0 & 0 & 0 & 0 & 0 & \lambda_8 & 0 \end{bmatrix} \begin{bmatrix} N_1 \\ N_2 \\ N_3 \\ N_4 \\ N_5 \\ N_6 \\ N_7 \\ N_8 \\ N_9 \end{bmatrix}$$

and ^{90}Sr :

$$\begin{bmatrix} \dot{N}_1 \\ \dot{N}_2 \\ \dot{N}_3 \\ \dot{N}_4 \\ \dot{N}_5 \\ \dot{N}_6 \\ \dot{N}_7 \\ \dot{N}_8 \\ \dot{N}_9 \\ \dot{N}_{10} \end{bmatrix} = \begin{bmatrix} -[(\sigma_1^1 + \sigma_2^1)\phi + \lambda_1] & \sigma_2^1\phi & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ \sigma_2^1\phi & -(\sigma_2^1\phi + \lambda_2) & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ \sigma_1^1\phi & 0 & -\lambda_3 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ 0 & \lambda_2 & 0 & -\sigma_4^1\phi & 0 & 0 & 0 & 0 & 0 & 0 \\ \lambda_1 & 0 & 0 & \sigma_4^1\phi & -(\sigma_5^1\phi + \lambda_5) & 0 & 0 & 0 & 0 & 0 \\ 0 & 0 & \lambda_3 & 0 & \sigma_5^1\phi & -(\sigma_6^1\phi + \lambda_6) & 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 & \sigma_6^1\phi & -\lambda_7 & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 & \lambda_5 & 0 & 0 & -\sigma_8^1\phi & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 & \lambda_6 & 0 & \sigma_8^1\phi & -(\sigma_9^1 + \sigma_9^2)\phi & \sigma_{10}^2\phi \\ 0 & 0 & 0 & 0 & 0 & 0 & \lambda_7 & 0 & \sigma_9^1\phi & -\sigma_{10}^2\phi \end{bmatrix} \begin{bmatrix} N_1 \\ N_2 \\ N_3 \\ N_4 \\ N_5 \\ N_6 \\ N_7 \\ N_8 \\ N_9 \\ N_{10} \end{bmatrix}$$

One of the peculiarities of the ALKEMY code is that it requires that the coefficient matrix be transposed. The transposed matrices for the ^{137}Cs and ^{90}Sr transmutation chains are

for ^{137}Cs :

$$\begin{bmatrix} \dot{N}_1 \\ \dot{N}_2 \\ \dot{N}_3 \\ \dot{N}_4 \\ \dot{N}_5 \\ \dot{N}_6 \\ \dot{N}_7 \\ \dot{N}_8 \\ \dot{N}_9 \end{bmatrix} = \begin{bmatrix} -[(\sigma_1^1 + \sigma_1^2)\phi + \lambda_1] & \sigma_1^2\phi & \sigma_1^1\phi & 0 & 0 & 0 & 0 & 0 & 0 \\ \sigma_2^1\phi & -(\sigma_2^1\phi + \lambda_2) & 0 & \lambda_2 & 0 & 0 & 0 & 0 & 0 \\ 0 & 0 & -\lambda_3 & 0 & 0 & 0 & \lambda_3 & 0 & 0 \\ 0 & 0 & 0 & -(\sigma_4^1 + \sigma_4^2)\phi & \sigma_4^2\phi & \sigma_4^1\phi & 0 & 0 & 0 \\ 0 & 0 & 0 & \sigma_5^1\phi & -\sigma_5^1\phi & 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 & -\sigma_6^1\phi & \sigma_6^1\phi & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 & \sigma_7^2\phi & -(\sigma_7^1 + \sigma_7^2)\phi & \sigma_7^1\phi & 0 \\ 0 & 0 & 0 & 0 & 0 & 0 & 0 & -\lambda_8 & 0 \\ 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \end{bmatrix} \begin{bmatrix} N_1 \\ N_2 \\ N_3 \\ N_4 \\ N_5 \\ N_6 \\ N_7 \\ N_8 \\ N_9 \end{bmatrix}$$

and ^{90}Sr :

$$\begin{bmatrix} \dot{N}_1 \\ \dot{N}_2 \\ \dot{N}_3 \\ \dot{N}_4 \\ \dot{N}_5 \\ \dot{N}_6 \\ \dot{N}_7 \\ \dot{N}_8 \\ \dot{N}_9 \\ \dot{N}_{10} \end{bmatrix} = \begin{bmatrix} -[(\sigma_1^1 + \sigma_1^2)\phi + \lambda_1] & \sigma_1^2\phi & \sigma_1^1\phi & 0 & \lambda_1 & 0 & 0 & 0 & 0 & 0 \\ \sigma_2^1\phi & -(\sigma_2^1\phi + \lambda_2) & 0 & \lambda_2 & 0 & 0 & 0 & 0 & 0 & 0 \\ 0 & 0 & -\lambda_3 & 0 & 0 & \lambda_3 & 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & -\sigma_4^1\phi & \sigma_4^1\phi & 0 & 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 & -(\lambda_5 + \sigma_5^1\phi) & \sigma_5^1\phi & 0 & \lambda_5 & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 & -(\sigma_6^1\phi + \lambda_6) & \sigma_6^1\phi & 0 & \lambda_6 & 0 \\ 0 & 0 & 0 & 0 & 0 & 0 & -\lambda_7 & 0 & 0 & \lambda_7 \\ 0 & 0 & 0 & 0 & 0 & 0 & 0 & -\sigma_8^1\phi & \sigma_8^1\phi & 0 \\ 0 & 0 & 0 & 0 & 0 & 0 & 0 & \sigma_9^2\phi & -(\sigma_9^1 + \sigma_9^2)\phi & \sigma_9^1\phi \\ 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & \sigma_{10}^2\phi & -\sigma_{10}^2\phi \end{bmatrix} \begin{bmatrix} N_1 \\ N_2 \\ N_3 \\ N_4 \\ N_5 \\ N_6 \\ N_7 \\ N_8 \\ N_9 \\ N_{10} \end{bmatrix}$$

The basic input required by the ALKEMY code is the neutron flux, the time in time steps, and the above transposed coefficient matrices entered in matrix format. The values of the coefficient used in the analysis are summarized in Table 21

TABLE 21
COEFFICIENT VALUES USED IN THE TRANSMUTATION ANALYSIS

	^{137}Cs			^{90}Sr		
	σ_i^1 (barns)	σ_i^2 (barns)	λ_i (sec^{-1})	σ_i^1 (barns)	σ_i^2 (barns)	λ_i (sec^{-1})
N_1	0.0444	0.147	0.8261×10^{-9}	0.0188	0.148	0.76318×10^{-9}
N_2	0.1168	-	0.617×10^{-6}	0.010616	-	0.15855×10^{-6}
N_3	-	-	0.3587×10^{-3}	-	-	0.19850×10^{-4}
N_4	0.06711	0.06352	-	0.010714	-	-
N_5	0.4565	-	-	0.032527	-	0.29991×10^{-5}
N_6	0.8652	-	-	0.041618	-	0.13598×10^{-6}
N_7	0.005912	0.1174	-	-	-	0.54544×10^{-4}
N_8	-	-	0.1359×10^{-3}	0.0051359	-	-
N_9	-	-	-	0.073411	0.14192	-
N_{10}	-	-	-	-	0.12315	-

In this analysis, one is primarily concerned with the parent/daughter balance. That is, one is trying to calculate the relative decrease in radioactivity as a function of reaction rate. Absolute destruction rates will be more accurately calculated in further analyses.

The calculation was performed using a nominal one-energy group neutron flux of 5×10^{15} n/cm² sec and time steps out to three years. The flux level is somewhat higher than that usually associated with CTR power plants. This value was selected based upon the hope that by the time transmutation is applied in a CTR, that the technology will have advanced far enough to allow for the implied vacuum wall loading. If this high a value proves to be unrealistic, longer irradiation times will be required. In fact, one should keep in mind that what is being calculated is a reaction rate (i.e., $\sigma_i^j \phi t$). It will be shown later that a more astute choice of target location can greatly increase σ_i^j and therefore the reaction rate term. This would, in turn, either allow one to operate with smaller flux values or shorter irradiation times.

THE NEUTRON BALANCE MODEL

One of the potential problems associated with neutron transmutation of fission products is the propensity of daughter products to compete with their parents for neutrons. If this effect becomes profound, one may have to routinely separate out the stable daughter products from the target material. On the other hand, because of the possibility of achieving significant (n,2n) reactions in the high energy neutron flux of a CTR, the scheme can be a net producer of neutrons.

In order to have an analysis tool to monitor the above phenomena, a neutron balance model for each transmutation chain was developed. Quite simply, this is a method of calculating the ratio of neutron gain through (n,2n) reactions to neutron loss through (n,γ) etc., reactions.

For the ¹³⁷Cs transmutation chain, the neutron gain and neutron loss terms are summarized below. The gain terms are multiplied by two because (n,2n) reactions are involved.

<u>Gain</u>	<u>Loss</u>
$2N_1\sigma_1^2\phi$	$N_1(\sigma_1^1 + \sigma_1^2)\phi$
$2N_4\sigma_4^2\phi$	$N_2\sigma_2^1\phi$
$2N_7\sigma_7^2\phi$	$N_4(\sigma_4^1 + \sigma_4^2)\phi$
	$N_5\sigma_5^1\phi$
	$N_6\sigma_6^1\phi$
	$N_7(\sigma_7^1 + \sigma_7^2)\phi$

The (n,2n) terms also show up in the loss column to account for the neutron loss from this reaction. Now, if one defines:

$$\text{Neutron Balance} = \frac{\text{Gain}}{\text{Loss}}$$

then:

$$\text{Neutron Balance} = \frac{2(N_1\sigma_1^2\phi + N_4\sigma_4^2\phi + N_7\sigma_7^2\phi)}{N_1(\sigma_1^1 + \sigma_1^2)\phi + N_2\sigma_2^1\phi + N_4(\sigma_4^1 + \sigma_4^2)\phi + N_5\sigma_5^1\phi + N_6\sigma_6^1\phi + N_7(\sigma_7^1 + \sigma_7^2)\phi}$$

By cancelling the flux term and doing some rearranging;

$$\text{Neutron Balance} = \frac{2(N_1\sigma_1^2 + N_4\sigma_4^2 + N_7\sigma_7^2)}{(N_1\sigma_1^2 + N_4\sigma_4^2 + N_7\sigma_7^2) + (N_1\sigma_1^1 + N_2\sigma_2^1 + N_4\sigma_4^1 + N_5\sigma_5^1 + N_6\sigma_6^1 + N_7\sigma_7^1)}$$

and then:

$$\text{Neutron Balance} = \frac{2}{1 + \frac{N_1\sigma_1^1 + N_2\sigma_2^1 + N_4\sigma_4^1 + N_5\sigma_5^1 + N_6\sigma_6^1 + N_7\sigma_7^1}{N_1\sigma_1^2 + N_4\sigma_4^2 + N_7\sigma_7^2}}$$

The above equation can be used to calculate the neutron balance as a function of exposure. An interesting calculation which can be done at this point is to calculate the neutron balance at time zero.

At time zero:

$$N_2 = N_4 = N_5 = N_6 + N_7 = 0.0$$

so that:

$$\text{Neutron Balance} = \frac{2}{1 + \frac{\sigma_1^2}{\sigma_1^1}}$$

From Table 21, $\sigma_1^1 = 0.0444$ and $\sigma_1^2 = 0.147$, so that:

$$\text{Neutron Balance} = \frac{2}{1 + \frac{0.0444}{0.147}} = 1.536$$

Therefore, at time zero, the transmutation chain for ^{137}Cs is a net producer of neutrons.

In a similar manner, the gain and loss terms for the ^{90}Sr transmutation chain are:

Gain	Loss
$2N_1\sigma_1^2\phi$	$N_1(\sigma_1^1 + \sigma_1^2)\phi$
$2N_9\sigma_9^2\phi$	$N_2\sigma_2^1\phi$
$2N_{10}\sigma_{10}^2\phi$	$N_4\sigma_4^1\phi$
	$N_5\sigma_5^1\phi$
	$N_6\sigma_6^1\phi$
	$N_8\sigma_8^1\phi$
	$N_9(\sigma_9^1 + \sigma_9^2)\phi$
	$N_{10}\sigma_{10}^2\phi$

As before:

$$\text{Neutron Balance} = \frac{\text{Gain}}{\text{Loss}}$$

and:

$$\text{Neutron Balance} = \frac{2(N_1\sigma_1^2\phi + N_9\sigma_9^2\phi + N_{10}\sigma_{10}^2\phi)}{N_1(\sigma_1^1 + \sigma_1^2)\phi + N_2\sigma_2^1\phi + N_4\sigma_4^1\phi + N_6\sigma_6^1\phi + N_8\sigma_8^1\phi + N_9(\sigma_9^1 + \sigma_9^2)\phi + N_{10}\sigma_{10}^2\phi}$$

By rearranging terms and cancelling the flux terms;

$$\text{Neutron Balance} = \frac{2}{N_1\sigma_1^2 + N_9\sigma_9^2 + N_{10}\sigma_{10}^2 + N_1\sigma_1^1 + N_2\sigma_2^1 + N_4\sigma_4^1 + N_5\sigma_5^1 + N_6\sigma_6^1 + N_8\sigma_8^1 + N_9\sigma_9^1}$$

and then:

$$\text{Neutron Balance} = \frac{2}{1 + \frac{N_1\sigma_1^1 + N_2\sigma_2^1 + N_4\sigma_4^1 + N_5\sigma_5^1 + N_6\sigma_6^1 + N_8\sigma_8^1 + N_9\sigma_9^1}{N_1\sigma_1^2 + N_9\sigma_9^2 + N_{10}\sigma_{10}^2}}$$

The above equation describes the neutron balance for the ^{90}Sr transmutation chain as a function of exposure. Looking at the time zero conditions for this chain

$$N_2 = N_4 = N_5 = N_6 = N_8 = N_9 = N_{10} = 0$$

and:

$$\text{Neutron Balance} = \frac{2}{1 + \frac{\sigma_1^1}{\sigma_1^2}}$$

as before.

From Table , $\sigma_1^1 = 0.0188$ and $\sigma_1^2 = 0.148$ for the ^{90}Sr chain, so that,

$$\text{Neutron Balance} = \frac{2}{1 + \frac{0.0188}{0.148}} = 1.775$$

Therefore, the ^{90}Sr chain is an even higher net producer of neutrons than the ^{137}Cs chain at time zero. The results of the ALKEMY analysis, which provide values of N_i as a function of time, will allow calculation of the neutron balance as a function of time for both transmutation chains.

THE DAUGHTER PRODUCTS

The daughter product concentrations as a function of time which result from the ALKEMY calculations are shown in Figures 12 and 13. As can be seen from studying these figures, the large majority of the daughter products which are created are stable and after about an equivalent reaction rate of one year at 5×10^{15} flux level, are well out of the transient accumulation stage. Therefore, this analysis demonstrates that, for these particular isotopes in a neutron flux of this spectrum, the reduction in radioactivity of the fission products is paramount. That is, in large measure, transmutation of Sr^{90} and Cs^{137} , result in stable daughter products in a CTR flux. Therefore, this concept passes its first test for viability. As pointed out earlier, isotopes which on the periodic chart are bordered on the left and above by stable isotopes have an excellent chance of success with regard to neutron transmutation concepts. Sr^{90} and Cs^{137} meet these criterion and, as is shown here, indeed transmute to stable daughter products. The probable rate at which this transmutation will occur is the subject of a later discussion.

BUILDUP OF ^{90}Sr DAUGHTER PRODUCTS WITH TIME

FIGURE 12

BUILDUP OF ^{137}Cs DAUGHTER PRODUCTS WITH TIME

FIGURE 13

THE NEUTRON BALANCE AND CONSEQUENT ECONOMIC IMPLICATIONS

The neutron balance for the ^{137}Cs and ^{90}Sr transmutation chains as a function of time are displayed in Table 22. These results are based upon the ALKEMY transmutation calculation results reported earlier in this report. The results are also displayed in Figure 14.

TABLE 22

NEUTRON BALANCE FOR ^{137}Cs AND ^{90}Sr

<u>Time</u>	<u>^{137}Cs Neutron Balance</u>	<u>^{90}Sr Neutron Balance</u>
0.0	1.54	1.78
3 mo	1.53	1.77 ⁺
6 mo	1.52 ⁺	1.77 ⁺
9 mo	1.52 ⁻	1.77 ⁺
12 mo	1.51	1.77 ⁻
25 mo	1.49	1.76 ⁺
36 mo	1.46	1.76 ⁻

These results show that the ^{90}Sr and ^{137}Cs transmutation chains are net exporters of neutrons. In addition, these results are excellent indicators of the effect of the presence of daughter products in the loading. Notice that the neutron balance tends to decrease with time. This is due primarily to parasitic neutron absorption by the daughter products. Overall, it appears that for extended exposure times (up to three years at high flux levels) the neutron balance for both chains is only slightly depressed by this parasitic absorption. It would appear, therefore, that from a neutronics standpoint, early separation of the daughter products from the mixture is not of vital importance to the success of this concept.

$^{90}\text{Sr} + ^{137}\text{Cs}$
NEUTRON BALANCE

FIGURE 14

The fact that these transmutation chains appear to be net exporters of fission products is also of major importance to the probable economics of the process.

If a transmutation process is a net absorber of neutrons, the process could prove to be quite expensive relative to other means of waste disposal.

An initial evaluation of the minimum cost of such a process can be made by examining the worth in dollars of the neutrons consumed in the process. It is felt that the neutron worth can be the largest single cost factor in such a process. The best present and most probable future use for neutrons is in the generation of plutonium for fission power plants. The present and probable future worth of plutonium has been estimated and the results of these calculations are shown in Figure 15.

The present value of plutonium is of the order of \$6/gm. The peak of the curve corresponds to the large-scale introduction of commercial Liquid Metal Fast Breeder Reactor (LMFBR) power plants. At this point, the value of plutonium is estimated to be \$15/gm. Because the LMFBR's are predicted to produce more plutonium than they require, the excess plutonium will probably be used in thermal neutron power plants. Large-scale production of plutonium from LMFBR's will tend to depress the plutonium value in the early 2000's and by the year 2020 plutonium should asymptotically approach a value of about \$2/gm. It is in this time frame that one could expect that a CTR fission product transmutation scheme might be employed.

One gram of plutonium would contain a certain number (N^{Pu}) atoms of plutonium. This number can be calculated by the following equation.

$$N = \text{number of atoms/gm} = \frac{\text{Avogadro's Number}}{A}$$

where

Avogadro's Number = The number of atoms in a gram atomic weight of any substance (6.023×10^{23})

A = Atomic weight of the element.

For plutonium, the value of N^{Pu} is:

$$N^{\text{Pu}} = \frac{6.023 \times 10^{23}}{239} = 2.51 \times 10^{21} \text{ atoms/gm}$$

This implies that at least 2.51×10^{21} neutrons were required to produce this amount of plutonium. Therefore, the neutron value in \$/neutron, $(N.V.)^{\text{Pu}}$, in 1995 when applied to plutonium production is:

$$(N.V.)^{\text{Pu}} = \frac{\$2.00}{2.52 \times 10^{21}} = 7.94 \times 10^{-22} \text{ \$/neutron.}$$

Actually, more neutrons than this would be required for production of this amount of plutonium due to inevitable leakage and parasitic losses. This would tend to somewhat depress the magnitude of the $(N.V.)^{\text{Pu}}$.

The current best alternative solution to the fission product problem, storage in deep salt caverns, has been estimated to cost of the order of 45 mills per megawatt hour electric (MWhr_e). If so, the cost of storage for fission products per megawatt hour thermal (MWhr_{th}) is then 15 mills, presuming an average of 33% efficiency in energy conversion.

One fission produces on the average two fission products and 3.2×10^{-11} W-sec or 1.152×10^{-21} MWhr_{th} . Only 5% of the fission products need be considered. Therefore, one can develop the following argument.

$$1 \text{ Fission} = >2 \times (0.05) \text{ Fission Products} = >1.152 \times 10^{-21} \text{ MWhr}_{th}$$

$$\frac{0.1 \text{ Fission Products}}{1.152 \times 10^{-21}} = >15 \text{ mills}$$

$$8.681 \times 10^{19} \text{ Fission Products} = >15 \text{ mills}$$

so that, if the minimum number of neutrons required to transmute this amount of fission products is 8.681×10^{19} neutrons, this amount of neutrons

PRESENT AND PROBABLE FUTURE WORTH OF PLUTONIUM

FIGURE 15

has a value of 15 mills when compared to long-term storage. The transmutation process has been estimated to be relatively inefficient (~50%) because of the low absorption cross sections which would double the number of required neutrons to 1.736×10^{20} . Then, the Neutron Value when used for transmutation of fission products, $(NV)^{F.P.}$, can be estimated as:

$$(NV)^{F.P.} = \frac{1.5 \times 10^{-2} \$}{1.736 \times 10^{20}} = 8.6 \times 10^{-23} \text{ \$/neutron}$$

From these calculations, one finds that in 2020 a neutron would have ten times as much value when used for plutonium production as compared to being used for fission product transmutation. From this kind of preliminary argument, one might conclude that fission product transmutation will be of the order of ten times as expensive as the best current alternate solution to the problem. If the other isotopes of cesium are considered, the process becomes twenty times as expensive.

The conclusion that one draws from this discussion is that, if a transmutation process is a net neutron absorber, as would be likely with neutrons of energies below 8 MeV (the typical $n,2n$ threshold), the process is relatively expensive. On the other hand, if transmutation is a net neutron producer, as seems likely in a CTR flux, then the neutrons produced can generate significant revenue. This opens up the possibility that fission product transmutation in a CTR could prove to be a revenue producing process and could perhaps be justified on economic grounds quite separate from its waste management function.

FUSION-FUSSION TRANSMUTATION BALANCE

One question which was discussed earlier was the question of the minimum required flux level for adequate transmutation of fission products. This was established in a preliminary manner by reaction rate considerations. Another question of a scoping nature is to try to determine at what rate one might burn fission products in a CTR relative to their production in fission reactors. That is, one can try to find what fraction of the electric economy would have to be fusion power, considering the rest to be fission, in order that the particularly noxious fission products be transmuted at the rate that they are created.

For a D-T CTR, one neutron is produced for every 17.6 MeV of thermal power. If both systems use state-of-the-art heat transfer technology, it is reasonable to assume that the efficiency of conversion for each system is approximately the same. For the purposes of this calculation, ^{90}Sr and ^{137}Cs , which are shown elsewhere to be the most noxious fission products for at least one situation, can be taken to be approximately 5% of the total fission product population. Therefore:

$$\begin{aligned} & \frac{200 \text{ MeV/Fission}}{2 \text{ Fission Products}} \times 0.05 \frac{^{90}\text{Sr or } ^{137}\text{Cs}}{\text{Fission Product Inventory}} \\ &= 2000 \frac{\text{MeV}}{^{90}\text{Sr or } ^{137}\text{Cs atom}} \end{aligned}$$

Fission Product transmutation can be expected to utilize a certain fraction of the total available neutrons in a CTR. If a D-T CTR has a breeding ratio of 1.25, then some 0.25 neutrons are available for transmutation per 17.6 MeV of thermal energy. The amount of CTR energy required, for a breeding ratio of 1.25, in order to transmute fission products at a certain efficiency is summarized in Table 23.

TABLE 23
CTR ENERGY FOR TRANSMUTATION

<u>Neutron Transmutation Efficiency</u>	<u>Equivalent CTR Thermal Energy (MeV)</u>
0.1	880
0.2	440
0.4	220
0.5	176
0.7	127

The fraction of power which must come from CTR devices in order to transmute at a break even level is:

$$F_p = \frac{X_1}{X_1 + X_2}$$

where

X_1 = Equivalent CTR thermal energy required to transmute 1 noxious fission product atom

X_2 = Amount of Fission thermal energy required to produce 1 noxious fission product.

The break even fraction, as a function of neutron transmutation efficiency, is summarized in Table 24.

TABLE 24
REQUIRED CTR POWER FRACTION

<u>Neutron Transmutation Efficiency Fraction</u>	<u>Required CTR Power Fraction</u>
0.1	0.31
0.2	0.18
0.4	0.10
0.5	0.08
0.7	0.06

Therefore, if one could load noxious fission products into the blanket such that the available neutrons were used with 50% efficiency, then 8% of the total power would have to be CTR power in order to transmute on a break even basis.

The preceding calculation was done on the basis of isotopically pure ^{137}Cs and ^{90}Sr . If the other isotopes in the mixture will tend to increase the CTR power fraction required.

LOADING PATTERN FOR FISSION PRODUCTS

Another question which must be explored is to determine, at least in a preliminary manner, just what fraction of the available neutrons can be reasonably absorbed in a fission product region of a CTR blanket.

A simple model, which can be used for comparative analysis, is to consider a plasma source which is an infinitely long cylinder and which is surrounded by a blanket of fission products. This geometric model can be described by the equation:

$$\epsilon = 1 - \frac{r}{r_0} \times \frac{1}{2} \times E_1 \left[\Sigma_a (r_0 - r) \right]$$

where

ϵ = efficiency of absorption

r_0 = plasma + blanket radius

r = plasma radius

Σ_a = macroscopic absorption cross section

$E_1(x)$ = an exponential function

This simple model assumes that the fission products are loaded into the inner region of the blanket at full density. While this strategy is perhaps debatable, some reasons will be given in the summary for following such an approach.

In order to achieve 50% utilization of the available neutrons from a D-T plasma with a breeding ratio of 1.25, one must absorb $1/5 \times .5 = 0.1$ or 10% of the total source neutrons. The approximate density for ^{137}Cs and ^{90}Sr is 1.9 gm/cc and 2.5 gm/cc, respectively. For purposes of illustration, the appropriate number density range is then:

$$N_0^{\text{Cs}} = 8.3 \times 10^{21}$$

$$N_0^{\text{Sr}} = 1.14 \times 10^{22}$$

assuming a microscopic cross section of the order of 1 barn and taking the lower number in the above range, one can calculate a macroscopic absorption cross section of:

$$\Sigma_a = N_o \sigma_a = 8.3 \times 10^{21} \times 1.0 \times 10^{-24}$$

$$\Sigma_a = 8.3 \times 10^{-3}$$

If one assumes a source radius of 100 cm, the following table can be developed based on the above absorption cross section and this calculational model.

TABLE 25
FISSION PRODUCT ATTENUATION FROM CYLINDRICAL SOURCE

<u>r, cm</u>	<u>r₀, cm</u>	<u>r - r₀, cm</u>	<u>ε</u>
100	110	10	0.045
100	115	15	0.237
100	120	20	0.368
100	125	25	0.464

If this data is plotted, it indicates that a fission product loading approximately 12 cm in thickness gives the required amount of attenuation.

A simple calculation based upon a wall loading of 10 MW/M² and a loading a 12 cm of fission products at a density of 1.9 gm/cc shows that 24 KG of fission products would be loaded per MW of CTR power. (i.e.)

$$\pi \left[(112)^2 - (100)^2 \right] \times 10^{-3} \text{ KG/g} \times 1.9 \text{ gm/cc} \times 10^2 \text{ cm/}\mu \quad 1 \mu$$

$$2 \times \pi \times 100 \times 10^{-2} \mu/\text{cm} \times 10 \text{ MW/}\mu^2 \times 1 \mu = 24 \text{ KG/MW}$$

From Table 1, 0.8867 KG of ^{137}Cs , 0.5587 KG of ^{135}Cs , 0.004742 KG of ^{134}Cs , 0.9861 KG of ^{133}Cs , 0.373 KG of ^{90}Sr , and 0.3111 KG of ^{88}Sr are created for every 20 MW of thermal fission power after ten years cooling. This results in a production ratio of 0.1555 KG/MW of noxious fission products from fission power. Using a fusion-fission ratio of 0.16 as determined by the neutron balance, $0.1555/0.16 = 0.97 \text{ KG/gm}$. Therefore, more fission products need to be loaded into the blanket to achieve 50% utilization of neutrons than are available from an on-balance situation. This indicates that a few CTR's would be used to burn fission products rather than scattering the load out among many.

The above preliminary calculation indicates that fission products can perhaps be loaded into CTR blankets without space limitations being a problem. Further iterative calculation of a much more detailed nature are required to determine the proper loading scheme. It appears, however, that a loading of the order of 12 cm thickness in a CTR achieves about 50% available neutron utilization.

FUTURE ANALYSIS

The intent of the work done to date is to establish whether or not transmutation of noxious fission products by means of a CTR device is a viable concept for waste management. In order to accomplish this end, there are a number of analyses that must be completed in addition to the work described here.

First of all, the calculations were performed on the basis of isotopic Sr^{90} and Cs^{137} loading. Any practical scheme would probably involve elemental rather than isotopic loadings. It is, therefore, necessary to perform the additional calculations which would account for the transmutation and decay of the other isotopes of Cs and Sr.

Secondly, the present analysis was done on the basis that small fission product loadings were placed into the fusion standard blanket assembly in such a manner that the neutron flux profile in that assembly was not disturbed. This approach is useful for an initial analysis. However, in the next phase of the analysis, one must determine the effect on the neutron flux profile and thus on the performance of the blanket when significant fission product loading are introduced into the blanket such that the base neutron flux profile is perturbed.

In line with the above analysis, one must also try to determine the optimum loading pattern in the CTR blanket which results in the highest transmutation rate while at the same time not materially affecting the other functions of the blanket. This will involve a series of complex parametric studies involving iteration between target loadings and resultant flux profiles. In addition, design changes to the blanket could be contemplated which enhance its value as a transmutation device but which

would not materially affect its other missions.

Another analysis which needs to be performed is to look at transmutation of other isotopes when Sr^{90} and Cs^{137} in a CTR blanket. Certain other process fission product transmutation chains need to be looked at and, in addition, transmutation of trans-uranic isotopes needs to be analyzed. The latter isotopes appear, on initial inspection, to be peculiarly susceptible to transmutation processes and, indeed, may prove to be the most attractive for such a concept.

When the above information has been developed, it should be possible to be able to draft a statement on the viability of transmutation with a CTR as a useful means of Waste Management. A final piece of work, which is not necessarily required for one to make a viability statement but which, nevertheless, is of prime importance to any of these approaches is to continuously encourage the collection and evaluation of neutronic data on the isotopes of interest. There appears to be an alarming lack of available evaluated data on the isotopes of interest for many of the important reactions.

CONCLUSIONS AND SOME SPECULATIONS

The major conclusion which can be drawn about the work which has been done to date is that the process appears to be viable at this point. This conclusion, however, will not be firm until the previously noted unfinished portions of the analysis have been completed.

First of all, transmutation of the specific isotopes analyzed indicates that the radiological hazard is decreased. That is, the ultimate daughter products created are stable and for the most part are non-radioactive. This is absolutely necessary for the feasibility of any transmutation scheme. It is a function of the neutron flux spectrum and this conclusion does not necessarily apply to all neutron transmutation schemes.

Secondly, while a particularly attractive transmutation rate has not been demonstrated by the analysis to this point, it appears that if detailed parametric studies are performed on specific target loading techniques, that a transmutation rate several times that of the natural decay of these isotopes can be achieved.

It appears that there will be adequate physical space in a CTR blanket for fission product loadings and that the thermal burden of these fission products on the blanket will not be too great.

It has been shown that a few CTR power plants can, in principle, transmute all of the Sr^{90} and Cs^{137} created by the rest of the electrical economy, if the economy were all nuclear power.

The daughter products produced appear to be approaching well defined equilibriums after a year or so operation. They should, therefore, cause few perturbations on the characteristics of operating CTR power plants.

Finally, the question of the economics of CTR power plants as fission product burners was found to be most interesting. First of all, it was shown that a transmutation scheme which is a net absorber of neutron would be relatively expensive. Then, it was shown that a CTR transmutation scheme, because of its unique neutronic characteristics, is a net producer of neutrons for long irradiation periods. This finding has a number of implications. First of all, it indicates that the process could be very cheap and indeed, could be a net producer of income. Secondly, it shows that the process does not require early separation of the daughter products. Finally, because certain fission products may prove to be of considerable value to CTR power plants, this finding has some implications as to what strategy should be used in Waste Storage.

In addition to the above conclusions, there are some inferences which can be drawn from the above analysis which, while they border on being speculations, can perhaps be presented here as discussion points.

First of all, one should notice that transmutation is not an absolute method of removing all fission products from the environment. It probably can be used only on certain isotopes and therefore should be considered an one of an arsenal of tools to be used, in conjunction with other approaches, to solve the total waste management problem.

Secondly, because the process is by its nature exponential in character, it depends upon having a certain fission product inventory in order to be successful. As such, it will never completely eliminate all of the fission products which it is brought to bear on. That is, the transmutation rate drops as the inventory goes down. Therefore, a transmutation scheme determines the absolute upper limit of the inventory of a particular isotope in the fuel cycle rather than bringing about its total removal from the environment.

A final speculation has to do with the application of CTR power plants to the transmutation of transuranium wastes (Pu^{239} , etc.). As a rule of thumb, the longer the natural half life of a particular isotope and the larger the cross section of that isotope, the more attractive that isotope is for transmutation. On this basis, while Sr^{90} and Cs^{137} have been found to be perhaps adequate transmutation targets, at least as far as this analysis has been carried, the CTR should be able to do a much better job on the transmutation of the transuranium wastes because, in general, their half lives are orders of magnitude higher, and their neutron cross sections are also significantly higher. In addition, the CTR is likely to be able to outperform most other approaches if, in the final analysis, the CTR is found to have the high neutron flux and excess neutron capacity currently predicted. Another reason why CTR transmutation of transuranium isotopes appears particularly attractive in a CTR has to do with the specific neutronic characteristics of a CTR. Recalling that this analysis demonstrated the importance of high energy neutron threshold reactions (i.e., $n, 2n$, etc.) in CTR transmutation, and looking at the cross sections for these isotopes for the $n, 2n, n, 3n$, and fission cross sections as a function of energy, one can tentatively conclude that these reactions would be of major importance in the transmutation of transuranium isotopes in a CTR. The net effect could be to transmute these isotopes back up the chain (i.e., $\text{Pu}^{240} \rightarrow \text{Pu}^{239}$) or alternately, to fission them. In either case, the net effect could be a rapid decrease of the radiological burden on the environment. Of course, if the plutonium were originally bred in a CTR blanket, as has been proposed, it is likely to have a higher Pu^{239} content than present transuranium isotopic mixtures and therefore would be more attractive material both economically and environmentally.

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