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TRANSMUTATION OF HIGH-LEVEL
RADIOACTIVE WASTE WITH A
CONTROLLED THERMONUCLEAR REACTOR

W. C. Wolkenhauer, B. R. Leonard, Jr.
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WITH A CONTROLLED THERMONUCLEAR REACTOR

By*

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September 1973

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TABLE OF CONTENTS

	<u>Page No.</u>
I. INTRODUCTION.	1
II. OBJECTIVE AND TECHNICAL APPROACH.	4
III. SUMMARY AND CONCLUSIONS	5
IV. THE NEUTRON PHYSICS ANALYSIS MODEL.	11
A. Physical Description & Computational Procedure.	11
B. Selection of Blanket Moderating Material.	15
C. Selection of Cross-Section Data	21
1. Capture Cross-Sections.	21
2. Selection of n,2n and n,3n Cross-Section Data.	30
3. Selection of Cross Sections for Actinides	31
4. Other Fast Neutron Reaction Cross Sections of Fission Products.	31
D. Calculation of Neutron Source Strengths	45
1. Neutron Wall Loading.	45
2. The Thermal Neutron Flux.	48
3. The Fast Neutron Flux for Transmutation Calculations.	51
4. Flux Levels for the Deuterium-Deuterium (D-D) Thermonuclear Plasma.	52
V. RESULTS OF NEUTRONICS ANALYSIS.	56
A. Fission Product Transmutation	59
1. Transmutation of ^{90}Sr	59
2. Transmutation of ^{137}Cs	67
3. Transmutation of ^{85}Kr	86
4. Transmutation of ^{129}I	88
B. Actinide Transmutation.	91
C. Effect of Uncertainties of Neutron Cross Sections on Calculated Transmutation Rates.	103

TABLE OF CONTENTS (Continued)

	<u>Page No.</u>
VI. REFERENCES.	106
VII. APPENDIXES	
I. The Transmutation Chains Studies	I-1
II. The Transmutation Equations.	II-1
III. Summary of Isotopic Data From The Analysis	III-1
IV. Conversion Factors From NUC/Ton to A Hazard Index. . .	IV-1
V. Compilation of Hazard Index Values	V-1
VI. Analytical Model Development	VI-1
VII. Task Force Membership.	VII-1

LIST OF TABLES

<u>Table</u>		<u>Page No.</u>
1	Summary of Results of CTR Transmutation: Hazard Half-Life As A Function of Waste Constituent, Plasma Type, and Neutron Source Strength.	6
2	Comparison of the Characteristics of Beryllium and Graphite Blankets for a 10 MW/m ² D-T Neutron Source.	17
3	Comparison of Spectrum-Averaged Cross Sections With Values Calculated from Resonance Integrals . . .	24
4	Effective One-Group Capture Cross Sections of Fission Product Isotopes for Transmutation Calculations	26
5	Q Values and Threshold Energies for n,2n and n,3n Reactions for Fission Product Transmutation. . .	32
6	Data Sources for Cross Sections for Actinide Transmutation and n,2n and n,3n Q Values and Threshold Energies for New Files.	35
7	n,2n and n,3n Effective Cross Sections for the Neutron Flux Spectrum Used in the Transmutation Calculations.	36
8	Cross Sections of Actinides for Transmutation Calculations.	39
9	Comparison of Experimental Values of Cross Sections with Predicted Values for ⁸⁸ Sr ₃₈	42
10	The Results of ANISN Calculations for the D-D Neutron Source Placed in Two Different Energy Groups.	55
11	Summary of Calculations	57
12	Hazard Half-Life as a Function of Wall Loading for ⁹⁰ Sr in a D-T Flux.	61
13	Hazard Half-Life as a Function of Wall Loading for FP Sr in a D-T Flux	62
14	Hazard Half-Life as a Function of Wall Loading for ⁹⁰ Sr in a D-D Flux.	67

LIST OF TABLES (Continued)

<u>Table</u>		<u>Page No.</u>
15	Hazard Half-Life as a Function of Wall Loading for ^{137}Cs in a D-T Flux	72
16	Hazard Half-Life as a Function of Neutron Wall Loading for Cs in a D-T Flux	77
17	Hazard Half-Life as a Function of Wall Loading for ^{137}Cs in a D-D Flux.	83
18	Hazard Half-Life as a Function of Wall Loading for ^{85}Kr in a D-T Flux	88
19	Hazard Half-Life as a Function of Neutron Wall Loading for ^{129}I in a D-T Flux	89
20	Hazard Half-Life as a Function of Neutron Wall Loading for Actinide Transmutation in a D-T Flux . .	94
21	Hazard Half-Life as a Function of Neutron Wall Loading for Actinide Transmutation in a D-D Flux . .	95
A-III.1	Time Scale for Calculation	III-2
A-IV.1	Hazard Index Values.	IV-2

LIST OF FIGURES

<u>Figure</u>		<u>Page No.</u>
1	Analysis Model.	12
2	Be Versus Cs Spectrum	22
3	Transmutation of ^{90}Sr In A D-T Flux	60
4	Transmutation of Sr in a D-T Flux	63
5	Elemental Sr and ^{90}Sr Transmutation in a D-T Flux With a Wall Loading of 10 MW/m^2	64
6	Transmutation of ^{90}Sr In A D-D Flux	66
7	Transmutation of ^{90}Sr In A D-D and D-T Flux at 1.0 MW/m^2	68
8	Transmutation of ^{137}Cs in a D-T Flux at 10 MW/m^2 . .	70
9	Transmutation of ^{137}Cs In A D-T Flux at 1 MW/m^2 . .	71
10	Transmutation of ^{137}Cs In A D-T Flux As A Function Of Wall Loading	73
11	Transmutation of Cs In A D-T Flux At Wall Loading Of 1 MW/m^2	75
12	Transmutation of Cs In A D-T Flux For A Wall Loading of 10 MW/m^2	76
13	Transmutation of Cs In A D-T Flux As A Function of Wall Loading.	78
14	Elemental Cs and ^{137}Cs Transmutation In A D-T Flux With A Wall Loading of 10 MW/m^2	80
15	Transmutation of ^{137}Cs In A D-D Flux With A Neutron Wall Loading Of 1 MW/m^2	81
16	Transmutation of ^{137}Cs In A D-D Flux With A Neutron Wall Loading of 10 MW/m^2	82
17	Transmutation of ^{137}Cs In A D-D Flux As A Function of Neutron Wall Loading	84
18	Transmutation of ^{137}Cs In A D-T and D-D Flux at 1 MW/m^2	85

LIST OF FIGURES (Continued)

<u>Figure</u>		<u>Page No.</u>
19	Transmutation of ^{85}Kr In A D-T Flux As A Function Of Neutron Wall Loading	87
20	Transmutation of ^{129}I As A Function Of Neutron Wall Loading For A D-T Flux.	90
21	Transmutation of Actinides In A D-T Flux As A Function Of Neutron Wall Loading	93
22	Transmutation of Actinides In A D-D Flux As A Function of Neutron Wall Loading	96
23	Transmutation of Am and Cm In A D-T Flux With Neutron Wall Loading of 1 MW/m^2	98
24	Transmutation of ^{237}Np In A D-T Flux With A Neutron Wall Loading of 1.0 MW/m^2	99
25	Time Dependent Behavior of LWR Fuel Load In A D-T Flux At 1 MW/m^2	101
26	Time Dependent Behavior Of LWR Pu In A D-T Flux at 1 MW/m^2	102
A-I.1	Transmutation Chain For Sr^{90} and Sr in D-T Flux. . .	I-2
A-I.2	Transmutation Chain For Cs^{137} In D-T Flux.	I-4
A-I.3	Transmutation Chain For Cs In D-T Flux	I-5
A-I.4	Transmutation Chain For Kr^{85} In D-T Flux	I-6
A-I.5	Transmutation Chain For I^{129} In D-T Flux	I-7
A-I.6	Transmutation Chain For Sr^{90} In D-D Flux	I-8
A-I.7	Transmutation Chain For Cs^{137} In D-D Flux.	I-9
A-I.8	Transmutation Chain for Actinides.	I-10

I. INTRODUCTION

Due to rising concern over the increasing inventories of radioactive wastes from fission reactors, there has been a concerted effort under the sponsorship of the USAEC Division of Waste Management and Transportation to explore alternate methods of high-level radioactive waste disposal. At the same time the Controlled Thermonuclear Reactor Division of the USAEC has been exploring alternate uses of the Controlled Thermonuclear Reactor (CTR). As a result of this exploration, a task force made up of staff members of the USAEC and staff members of contractors to the USAEC was organized at a meeting in Washington, D. C. in May 1973. The task force membership is listed in Appendix VII. At this meeting it was decided that a survey study would be undertaken on use of the CTR for the transmutation of radioactive waste to expand the above mentioned disposal studies. Battelle-Northwest was given prime responsibility in this project, but various other laboratories have made significant contributions to the work as noted in this document.

The emphasis of this study is on surveying the neutron physics characteristics of radioactive waste transmutation in a CTR. Therefore, a parametric analysis was carried out to investigate the effects of; a) neutron source level, b) plasma type, and c) the characteristics of the waste, on the rate of transmutation of various constituents of the high-level waste. The major portion of the neutron physics effort was carried out by Battelle-Northwest members of the task force. Some specific

engineering constraints have been considered by the other members of the task force, and their evaluations are included in this report.

The first proposal for the transmutation of radioactive waste was made by Steinberg,⁽¹⁾ et al., in 1964. They proposed that special purpose high-flux burner reactors be used to reduce the stockpile of the problem fission products ^{85}Kr , ^{90}Sr and ^{137}Cs . In a later work Steinberg and Gregory⁽²⁾ considered the use of spallation accelerators for the transmutation of fission products. More recently Claiborne⁽³⁾ studied neutron-induced transmutation of high-level waste, emphasizing actinide recycle in a light water reactor.

In December 1970, Leonard⁽⁴⁾ described the possibility that a CTR could be used for the transmutation of fission products. In June 1971, Steiner⁽⁵⁾ pointed out the potential of a CTR for the transmutation of very long-lived wastes (i.e., ^{129}I , ^{99}Tc and some of the actinides). The first study of the use of a CTR for the transmutation of radioactive waste was made by Wolkenhauer⁽⁶⁾ in a study of the transmutation of fission products in a CTR. This study considered only the transmutation of isotopic ^{90}Sr and ^{137}Cs .

The high-level radioactive wastes which come from fission reactors can be divided into two categories with quite distinct characteristics. The first category, fission products, includes the particularly troublesome isotopes ^{85}Kr , ^{90}Sr , ^{137}Cs , and ^{129}I . The second category includes the heavy element wastes, such as the isotopes of americium and curium, which are derived from successive neutron capture and decay in uranium and plutonium. Members of this second category are usually referred to

as actinides, although they do not necessarily all belong to the actinide family, and this convention will be followed in this report. Typically, the most troublesome fission products have half-lives of the order of 30 years and relatively low neutron capture cross-section values. The isotope ^{129}I is an exception, since it has a very long half-life. However, it is included in the first category in this report. The actinide wastes typically have very long half-lives and relatively high neutron cross sections for transmutation, especially for the fission process. Therefore, the analysis of the transmutation of these two categories of wastes was carried out separately. Fission product transmutation in a CTR can be achieved by means of the n,γ and $n,2n$ reactions which transmute the fission products into stable isotopes. Actinide transmutation, on the other hand, takes place mainly by means of the fission process. Therefore, in the transmutation of the actinide wastes the character of the waste changes from long half-lived, highly-toxic actinide isotopes into fission products with much shorter half-lives.

II. OBJECTIVE AND TECHNICAL APPROACH

The objective of this study is to evaluate the potential of a CTR as a source supplying neutrons for the transmutation of high-level radioactive wastes into stable or less hazardous species. The technical approach employed is to select idealized analysis conditions in order to establish an apparent upper limit of the transmutation rates in a CTR for selected troublesome radionuclides. Should reasonable radionuclide transmutation rates not be achieved under these conditions, further study would not be warranted.

The achievement of this objective required completion of the following:

- Selection of the important parameters for transmutation of high-level radioactive wastes in the major CTR concepts, (i.e., neutron flux level, cross-section data, etc.).
- Establishment of the relationships between the parameters identified above.
- Establishment of the theoretical feasibility of the concept.

III. SUMMARY AND CONCLUSIONS

Calculations were performed to predict the rate at which selected radionuclides in high-level waste undergo transmutation in the blanket region of a CTR. Idealistic conditions were assumed to facilitate the calculations. These conditions are expected to produce optimistic results and thereby define an apparent upper limit for the transmutation rates. An upper limit helps to define theoretical feasibility and determine incentives for further study.

The waste constituents of the fission product category studied included; targets made of Sr and Cs, and the isotopes ^{90}Sr , ^{137}Cs , ^{85}Kr and ^{129}I . Isotopes between ^{232}U and ^{243}Cm were included in the actinide category.

Calculations were made for neutrons obtained from both a D-D plasma and a D-T plasma to determine the effect of plasma type on the transmutation rate. Two neutron flux levels were also studied to determine the effect of neutron source strength on the transmutation rate.

In this study a parameter called hazard half-life is used to describe the results of transmutation. It is defined as the time required to reduce the hazard from a particular isotope and its daughters to one-half of its original value. The reduction in hazard half-life due to neutron irradiation is a useful measure of the effectiveness of neutron transmutation of high-level waste.

Results of the analyses made in this study are summarized in Table 1 in terms of hazard half-life. The base case values are listed in column 3

TABLE 1

Summary of Results of CTR Transmutation: Hazard
Half-Life as a Function of Waste Constituent,
Plasma Type, and Neutron Source Strength

Target	Plasma Type	Hazard Half-Life (yrs)		
		Base Case (Decay Only)	Source Strength, P_w^*	
			$(P_w = 1.0 \text{ MW/m}^2)$	$(P_w = 10 \text{ MW/m}^2)$
^{90}Sr	D-T	32.0	7.6	2.4
Sr**	D-T	32.0	8.0	7.4
^{90}Sr	D-D	32.0	4.7	0.48
^{137}Cs	D-T	30.0	19.0	5.6
Cs**	D-T	14.0	12.0	4.0
^{137}Cs	D-D	30.0	16.0	4.4
^{85}Kr	D-T	11.0	8.6	6.1
^{129}I	D-T	1.6×10^7	0.35	0.027
Actinides	D-T	460***	0.60	0.056
Actinides	D-D	460***	0.23	0.037

* P_w is the neutron wall loading and is defined as the kinetic energy of source neutrons transported to the inner wall of a CTR blanket.

** Isotopic mixture of Sr or Cs from LWR fuel at 33,000 MWD/MTM and 10 years cooling.

*** Am^{241} only.

and represent natural decay (i.e., no induced neutron transmutation). The values listed in columns 4 and 5 represent two different neutron source levels described in terms of the neutron wall loading for the vacuum wall of a CTR. It is usually assumed that CTR's will have a neutron wall loading somewhere between these two values. Establishing theoretical feasibility requires setting a criterion as a basis for judging feasibility. For this study, if neutron transmutation results in a factor of two reduction in the hazard half-life relative to natural decay, the concept is deemed theoretically feasible. On the basis of the results presented in Table 1 and other findings of this study the following general and specific conclusions are drawn.

General Conclusions

- Within the limits of the data, techniques, and assumptions used in this analysis, CTR transmutation of high-level wastes appears theoretically feasible and effective. However, certain nuclides require larger neutron source strengths than others to meet the criterion for theoretical feasibility. Further study is indeed warranted.
- A Deuterium-Deuterium (D-D) fusion plasma produces a maximum thermal neutron flux which is 2.6 times that produced by a Deuterium-Tritium (D-T) plasma for equal neutron-energy first vacuum wall loadings. Since thermal neutron absorption is the dominant transmutation process, the D-D plasma calculations yield significantly larger values of transmutation rates.

Specific Conclusions Regarding Fission Product Transmutation

The conclusions arrived at by inspection of the results in Table 1 are:

- Transmutation of ^{90}Sr is theoretically feasible for the range of neutron source strengths studied.
- Transmutation of ^{137}Cs is theoretically feasible for source strengths in excess of 1.0 MW/m^2 .
- Transmutation of Sr which has cooled for 10 years is theoretically feasible. Transmutations of Cs which has cooled for 10 years is only feasible for larger source strengths. Moreover, isotopic separation of fission product Cs does not significantly improve transmutation rates. The improved transmutation rates of fission product Sr would only justify isotopic separation if the process were quite inexpensive. (Only Cs and Sr were studied from the standpoint of isotopic separation).
- Transmutation of ^{85}Kr is not theoretically feasible and therefore should be eliminated as a candidate for future CTR transmutation studies.
- Transmutation of ^{129}I is theoretically feasible.

Caution should be exercised in making definitive statements of theoretical feasibility based upon these findings. The uncertainties in the knowledge of the thermal neutron capture cross sections, particularly for ^{90}Sr and ^{137}Cs are so large that the theoretical feasibility of their transmutation in a CTR blanket cannot be established at reasonable levels

of confidence. New measurements of the thermal neutron cross sections are a prerequisite to accurately predicting the transmutation rates of these fission products. Perhaps technical feasibility might best be judged by conducting an irradiation of fission product bearing capsules in a high thermal-flux reactor and extrapolating the results to transmutation in CTR's.

Specific Conclusions Regarding Actinide Transmutation

The actinides represent a mixture of isotopes and these isotopes have a large range of decay half-lives. Therefore, to illustrate the effects of actinide transmutation the hazard half-life selected for the base case was taken as that for an isotope which has a natural decay half-life between 100 and 1000 years. Comparing the results in columns 4 and 5 in Table 1 with the base case value for the actinides shows substantial gains accruing via CTR transmutation. The hazard half-life is reduced from hundreds of years to fractions of years. Therefore, on this basis,

- It is concluded that actinide transmutation in CTR's appears to be an attractive actinide disposal scheme, particularly in view of the fact that the actinides constitute the long term hazard of high-level waste.

In addition, the transmutation of separate targets of ^{237}Np , a mixture of Am and Cm, and plutonium was studied. From these studies it was concluded that:

- Transmutation of pure targets of ^{237}Np is not feasible.
- Effective transmutation of Am-Cm targets is dependent on irradiation time. The hazard increase for irradiation up to 2 years and decreases for longer irradiation times.
- In a D-T CTR, the plutonium hazard index for Light Water Reactor (LWR) produced plutonium will ultimately decrease with exposure. This phenomenon offers a potential solution to the residual plutonium waste generated by the operation of a fission economy.

It should be pointed out that the results of these analysis probably represent optimistic predictions of actinide transmutation due to the assumptions made in performing the calculations. Nevertheless, the effect of using a more realistic representation of actual neutronic events is not expected to alter the results to the extent whereby actinide transmutation in CTR's is an unattractive disposal concept.

Likewise, the incremental radioactive buildup due to structural activation caused by the operation of CTR's for this purpose was not considered. If CTR's were operated for transmutation only, this structural activation would constitute a debit to the program. However, if transmutation were done in power generating CTR's this buildup would not constitute a debit.

Finally, the value of routine chemical partitioning of the targets at time intervals during the irradiation was not studied in this analysis. Future studies might investigate this strategy, as it could have a favorable impact on the results.

IV. THE NEUTRON PHYSICS ANALYSIS MODEL

The analysis model is comprised of an assumed geometrical configuration of a CTR and a combination of computer programs which are used to calculate the neutronic parameters of a CTR blanket. A physical description of the CTR, the calculational model used, the analysis procedure employed, and other analysis details are outlined below.

A. Physical Description & Calculational Procedure

1. Physical Description - The principal elements of a CTR consist of: 1) the plasma, 2) a void between the vacuum wall and the plasma, 3) the vacuum wall and structural materials, 4) the medium (i.e., the blanket) whereby neutrons are absorbed and tritium is produced.

For the transmutation analysis the following physical model was used.

- 1) A plasma region, 160 cm in radius.
- 2) A void out to 200 cm.
- 3) A 100 cm thick beryllium blanket at 80% theoretical density.

The vacuum wall and structural materials were omitted in the calculations. In the calculation, the transmutation of the waste occurred in the first centimeter of the blanket (i.e., adjacent to the void region). The basis for using a beryllium blanket is outlined in part B of this section.

2. Calculational Procedure - Elements of the calculational model used in the analysis are outlined in Figure 1. Each box in the figure identifies

FIGURE 1. ANALYSIS MODEL

a particular function, and where a computer program was used to perform that function, it is named and underlined.

The neutronics analysis is carried out in the following manner. For a given target nucleus, the probable transmutation reactions are determined and documented. For the targets considered in this study, the transmutation reactions are listed in Appendix I. Selection of the transmutation reactions determines the requirements for cross-section data. Generation of the necessary cross-section is described in part C of this section.

In addition to the reaction cross sections, representative wall loadings must be selected for the parametric study. The wall loadings determine the neutron source strengths to be used in the calculation. Details of the wall loading selection and the source strength calculation are given in part D of this section.

Another input required for the calculation is the atom density for each nuclide in the mixture of wastes to be transmuted. For this study the constituents of the waste mixture from PWR fuel discharged at an average fuel exposure of 33,000 MWD/MTM, and assuming 99.5% U and Pu separation in reprocessing, was taken from Reference 7, though it could have been generated using the ORIGEN code⁽⁸⁾ as shown in Figure 1.

The first step in the analysis is the calculation of the neutron flux spectrum and flux level per source neutron in a blanket surrounding a CTR device. The procedure employed was to compute transmutation rates

of radionuclides using the flux level and energy spectrum of neutrons calculated in the first centimeter of the blanket in the absence of radionuclides in this region. Thus, the effects on the flux level and spectrum of neutrons of the injection of finite amounts of radionuclides into this region has been ignored in this study. Doing so simplifies the analysis since a single ANISN calculation suffices for all transmutation target materials for all periods of irradiation. In addition, this procedure should yield upper limits of the predicted transmutation rates because self-shielding effects are not accounted for.

High energy $n,2n$ and $n,3n$ cross sections were generated by the code SOLWHO (see Appendix VI) which was written for this project and which uses the formalism of Pearlstein.⁽⁹⁾ Other reaction cross sections (i.e., n,γ , etc.) were obtained from ENDF/B⁽¹⁰⁾ and other sources as noted in the text. The $n,2n$ cross sections were averaged into one-group constants using the CLAPS code, (see Appendix VI) which was written for this project. The CLAPS code averages input cross sections by integration over the neutron spectrum calculated by the ANISN⁽¹¹⁾ code. The other reaction cross sections are averaged over this same spectrum as discussed in a later section.

The output from the CLAPS code, and the other spectrum-averaged reaction cross section (e.g., n,γ , n,f , etc.,) are used as input to the ALCHEMY⁽¹²⁾ code along with the calculated source strength and the waste composition. The ALCHEMY code, which is a generalized differential equation solution code, was programmed for the neutron reactions and

radioactive decay schemes determined to be of interest and, with the above input, was used to calculate the isotopic composition of the various waste mixtures as a function of time and neutron wall loading. The isotopic compositions thus generated were converted, by the TRUE code (see Appendix VI) into a hazard index, which is then used to draw conclusions regarding effectiveness of this concept. The hazard index used in this study is defined as the cubic meters of either air or water required to dilute a certain amount of noxious material to acceptable levels. A complete definition of this term is given in Appendix V. Descriptions of the codes which were written for this analysis are given in Appendix VI. The hazard values used by the TRUE code are based upon Radioactive Concentration Guide (RCG) values from 10:CFR:20,⁽¹³⁾ Appendix B, Table 2, Column 2, except as noted.

B. Selection of Blanket Moderating Material

One of the purposes of the transmutation study was to establish upper limits to the transmutation rates which could be obtained in hypothetical fusion reactor blankets. Previous calculations⁽¹⁴⁾ had demonstrated that a blanket of beryllium gave thermal and fast neutron fluxes which were enhanced over those in a graphite moderator. The thermal flux is here defined as that flux in thermal equilibrium with the surrounding media. The fast flux is here defined as the non-thermal flux. The flux enhancement is the result of neutron multiplication by the $^9\text{Be}(n,2n)$ reaction which has a threshold energy of only 1.85 MeV while the $^{12}\text{C}(n,2n)$ threshold energy is greater than 14 MeV.

Beryllium has not been widely used as a moderating material in nuclear fission reactors. There appear to be two dominant reasons why it has not been widely used. These are:

- The expense of the manufacture of beryllium into moderator material which is primarily due to its toxicity.
- The poor structural stability of beryllium under irradiation by neutrons. This is primarily due to the helium gas formed in the $n,2n$ reaction.

In view of these considerations, calculations were made of pure beryllium, pure graphite and combinations of the two materials in order to assess the credibility of the proposed usage of beryllium,

The first result of these calculations is that it is not necessary to use beryllium for the entire 100 cm. of blanket. If the first 20 cm cylindrical annulus of the blanket is beryllium followed by 20 cm or more of graphite, the maximum values of fast and thermal neutron fluxes are not significantly reduced (<10%) from those obtained in a 100 cm thickness of beryllium.

The results of ANISN calculations of pure beryllium and pure graphite blankets of 100 cm thickness are summarized on Table 2. In each of the calculations the moderating material was taken to be at 80 percent of its theoretical density to allow room for gas cooling channels. The maximum values of the thermal fluxes occur in the first centimeter of the blanket. The maximum thermal flux achievable in the beryllium is seen to be over 2.5

TABLE 2
Comparison Of The Characteristics Of
Beryllium And Graphite Blankets
For a 10 MW/m² D-T Neutron Source

<u>Characteristic</u>	<u>Be</u>	<u>C</u>
MAXIMUM THERMAL FLUX, ϕ_{Th} (n-cm ⁻² -sec ⁻¹)	2.925 x 10 ¹⁶	1.13 x 10 ¹⁶
MAXIMUM FAST FLUX, ϕ_f (n-cm ⁻² -sec ⁻¹)	8.2 x 10 ¹⁵	4.1 x 10 ¹⁵
GAS PRODUCTION AT INNER WALL (percent of initial blanket atoms)		
Helium	4.5% per yr.	2.2% per yr.
Tritium	0.20% per yr.	
RADIOACTIVITY (buildup per day per meter length)		
¹⁰ Be	5.4 curies	0.015 curies
¹⁴ C		0.83 curies
T	55,000 curies	
AFTERHEAT* (buildup per day per meter length)	2.5 Watts	0.3 x 10 ⁻³ Watts

*This value does not include the activation of impurities, which may well prove to be controlling.

times that of the graphite. The thermal flux decreases faster with distance into the beryllium blanket, however, such that about halfway into the blanket the thermal flux value is approximately equal to that in a graphite blanket.

The neutron multiplication of the beryllium blanket through the n,2n reaction is achieved at the expense of helium gas production via the reaction:

With the neutron spectra produced by the 14 MeV D-T source, additional helium and tritium gas are also produced by means of the reactions

and the secondary reaction

The calculated production rates of helium and tritium in the first cm of the beryllium blanket are shown in Table 2 for a 10 MW/m² D-T neutron wall loading. As a matter of secondary interest the total tritium production in the Be blanket is calculated to be 0.19 atoms per source neutron.

In the graphite blanket helium is produced by the reactions:

The calculated helium production rates in graphite are about half of the beryllium values. On the other hand, the gas production in the beryllium blanket is actually less than in the graphite per thermal neutron available for transmutation. In any event the gas production rates in either material are so high that neither material would have any structural stability. To be used in a 14 MeV D-T neutron flux of this magnitude either material would have to be encased in a structural material with a greater resistance to radiation damage.

Also shown on Table 2 are the activities of the radioactive daughters produced in the blanket. In beryllium the only long term activity is the 1.6×10^6 yr beta emitter ^{10}Be produced by neutron capture. The ^{10}Be activity is also produced in the graphite blanket by the n,alpha reaction on the 1.1% ^{13}C isotope. Since there are no experimental determinations of this cross section, the value shown in the table was obtained by assuming that the $^{13}\text{C}(n,\alpha)$ cross section was equal to the sum of the $^{12}\text{C}(n,\alpha)$ and $^{12}\text{C}(n,3\alpha)$ cross sections. Thus, the value shown may be erroneous by a factor of several. This is not of particular importance since the dominant activity in graphite is that of the 5730 yr beta emitter ^{14}C produced by neutron capture by ^{13}C . The ^{14}C production rate is actually time dependent since there will be appreciable buildup of ^{13}C during irradiation by neutron capture in ^{12}C . This will be compensated in part by the n,2n reaction on ^{13}C which has a threshold energy of 5.3 MeV. The analysis of the effects of these secondary reactions was not included in the present study.

The calculated values of activity and radioactive decay energy do not indicate any particular problems. Although the tritium activity is high, the tritium is useful for DT fuel and presumably the only potential problem is in recovery of the tritium.

The neutron multiplication in a beryllium blanket driven by a D-D neutron source is not significant (about 10%) and a graphite blanket would be as likely a moderating medium. However, a beryllium blanket was used in the analytical studies reported here in order that consistent comparisons to the D-T analysis can be made.

C. Selection of Cross-Section Data

The specific sources of the cross-section data used in the analysis and the methods used in compiling these data are reviewed below.

1. Capture Cross-Sections

The transmutation of fission products (FP) in the blanket of a CTR is based on the concept of placing FP isotopes or elemental mixtures of FP isotopes in a neutron moderating material. Calculations have previously been made⁽¹⁴⁾ for a 2m vacuum wall radius DT Tokamak in which the first 20 cm of thickness was a beryllium moderator containing ^{137}Cs followed by graphite and beryllium reflectors. In these studies ^{137}Cs (n, γ) and (n,2n) reaction rates were calculated for several different concentrations of ^{137}Cs varying from infinite dilution to complete displacement of the Be in this zone. The calculated fluxes per unit lethargy for the two extreme cases are shown on Figure 2. The calculated neutron energy spectra are seen to differ significantly from a 1/E distribution only for neutron energies above some 3 keV. In addition, except for the 14 MeV component, the neutron flux in this region was found to be very well thermalized. The results of these calculations were studied to see if the resonance integral for neutron capture at infinitely dilute concentration, I, could be used to represent the capture rate. This procedure would avoid the time-consuming and costly process of integrating over calculated neutron spectra. The capture cross sections of most of the isotopes are very poorly known and for the most part are based on nuclear model calculations. Thus the time and expense of performing the detailed calculations and processing the results for some forty isotopes is not warranted since the end result is having cross sections that are, generally, significantly uncertain.

FIGURE 2. Be VERSUS Cs SPECTRUM

The epithermal capture cross sections of the Be-¹³⁷Cs systems were studied using the calculated activities of ¹³⁷Cs and of a unit 1/v cross section (i.e., a 1 barn thermal cross section with a 1/v energy variation at all energies). The calculated neutron fluxes and activities averaged over the 20 cm thickness were used in the comparison. The effective cross section, $\hat{\sigma}$, was defined to be

$$\hat{\sigma} = \frac{\text{Activity/Atom density}}{\text{Thermal flux}} \quad (7)$$

These cross sections were compared with the calculated values of

$$\hat{\sigma} = \sigma_{Th} + rI \quad (8)$$

where r was taken as the ratio of the epithermal flux per unit lethargy to the total thermal flux.

The results of this comparison are shown on Table 3 for the extreme cases of pure Be, pure Cs, and an intermediate concentration. Also shown are the numerical values of σ_{Th} , I , and r used in the comparison. The agreement for the unit 1/v absorber is seen to be essentially exact and the differences for ¹³⁷Cs are very small (much smaller than the uncertainties in the cross sections involved). In addition, it should be noted that the values of σ_{Th} and I for ¹³⁷Cs differ by only a factor of two. Since σ_{Th} and I are typically equal to about this extent for most of the isotopes under consideration, one can expect the use of calculated values of $\hat{\sigma} = \sigma_{Th} + rI$ to be adequate for the transmutation calculations.

TABLE 3

Comparison Of Spectrum-Averaged Cross Sections With
Values Calculated From Resonance Integrals

Vol % Cs Vol % Be	r	<u>1/v</u>		<u>¹³⁷Cs</u>		<u>Δ(%)</u> *
		<u>$\hat{\sigma}$</u>	<u>$\sigma_{Th} + rI$</u>	<u>$\hat{\sigma}$</u>	<u>$\sigma_{Th} + rI$</u>	
0/80	0.0193	1.0095	1.0095	0.1153	0.1145	0.7
25/55	0.02345	1.012	1.012	0.1167	0.1154	1.1
100/0	0.03867	1.019	1.019	0.1216	0.1189	2.3

Nuclear Data

	<u>1/v</u>	<u>¹³⁷Cs</u>
σ_{Th}	1.0	0.11
I	0.4944	0.231

$$* \Delta(\%) = \frac{\hat{\sigma} - (\sigma_{Th} + rI)}{\hat{\sigma}} \times 100$$

The values of thermal capture cross sections and resonance integrals used in this study were obtained primarily from the Chart of the Nuclides, March 1973 Revision,⁽¹⁵⁾ as shown on Table 4. The last column of the table gives the effective capture cross sections used in the transmutation calculations. These values are appropriate for the case of infinite dilution in a 1 cm thick 80 volume percent beryllium moderator. The thermal cross-section values given in the first column are, generally, the values at a neutron velocity of 2200 m/s, (σ_0), or the effective 2200 m/s cross section, ($g\sigma_0$), based on the Westcott flux convention.⁽¹⁶⁾ For the transmutation rate calculations using the ANISN calculated flux, the thermal Maxwellian-averaged cross section at room temperature have been used, (i.e., $\sqrt{\pi}/2 \times g\sigma_0$).

In the calculations of this study the CTR blanket was assumed to be at room temperature. For the fission product nuclei which were studied there is no significant differences between calculated values of reaction rates at room temperature or some elevated temperature. This is due to two reasons. First, because the FP nuclei were assumed to be in dilute concentration there is no resonance self shielding. If there is no resonance shielding then the resonance absorption is independent of temperature (Doppler-broadening). Secondly, the major FP nuclei have very small thermal absorption cross sections. For nuclei of this type this means that the thermal absorption cross sections have sensibly a $1/v$ energy dependence. Since all of the thermal neutron absorption cross sections in the system are $1/v$, the thermal neutron reaction rates are independent of temperature except for a small effect on neutron leakage from the blanket. Since the

TABLE 4

Effective One Group Capture Cross Sections Of
Fission Product Isotopes For Transmutation Calculations

Isotope	Thermal Capture (b) ⁽¹⁵⁾	σ_{Th} (b)	I (b)	$\sigma_{Th} + 0.013 I$ (b)
⁸⁸ Sr ₃₈	$\hat{\sigma} = 0.0057 \pm 0.0004$	0.00505	0.06 ± 0.03	0.0058
⁸⁹ Sr ₃₈	$\hat{\sigma} = 0.42 \pm 0.04$		UNK	0.37
⁹⁰ Sr ₃₈	$\hat{\sigma} = 0.8 \pm 0.5$		UNK	0.71
⁹¹ Sr ₃₈	UNK		UNK	1(c)
⁸⁸ Y ₃₉	UNK		UNK	2(c)
⁸⁹ Y ₃₉	$\sigma_0 = 1.28 \pm 0.01^{(17)}$	1.13	$0.09 \pm 0.03^{(17)}$	1.13
⁹⁰ Y ₃₉	$\hat{\sigma} < 6.5$		UNK	1(19)
⁹¹ Y ₃₉	$\hat{\sigma} = 1.4 \pm 0.3$		UNK	1.24
⁹² Y ₃₉	UNK		UNK	0.5 ⁽¹⁸⁾
⁹⁰ Zr ₄₀	$\hat{\sigma} = 0.03 \pm 0.02$		0.2 ± 0.1	0.03
⁹¹ Zr ₄₀	$\hat{\sigma} = 1.1 \pm 0.4$		5 ± 2	1.1
⁹² Zr ₄₀	$\hat{\sigma} = 0.26 \pm 0.08^{(17)}$		0.6 ± 0.2	0.24
⁹³ Zr ₄₀	$\hat{\sigma} = 1 \pm 3$		23 ± 10	1.3

TABLE 4 (Continued)

Isotope	Thermal Capture (b) ⁽¹⁵⁾	σ_{Th} (b)	I (b)	$\sigma_{Th} + 0.013I$ (b)
$^{133}_{Cs}_{55}$	$\sigma_0 = 29.5 \pm 2.0$	26.1	450 ± 50	32.0
$^{134}_{Cs}_{55}$	$\hat{\sigma} = 140 \pm 20$		UNK	125
$^{135}_{Cs}_{55}$	$\hat{\sigma} = 9 \pm 1$	8.0	62 ± 2	8.8
$^{136}_{Cs}_{55}$	UNK		UNK.	$100^{(17)}, 15^{(18)}$
$^{137}_{Cs}_{55}$	$\hat{\sigma} = 0.11 \pm 0.05$		UNK	0.11
$^{138}_{Cs}_{55}$	UNK		UNK	$1^{(18)}$
$^{134}_{Ba}_{56}$	$\hat{\sigma} = 2 \pm 2$		UNK	2
$^{135}_{Ba}_{56}$	$\hat{\sigma} = 6 \pm 1$	5.3	100 ± 50	6.6
$^{136}_{Ba}_{56}$	$\hat{\sigma} = 0.4 \pm 0.4$	0.35	20 ± 10	0.61
$^{137}_{Ba}_{56}$	$\hat{\sigma} = 5.1 \pm 0.4$	4.5	4 ± 2	4.6
$^{138}_{Ba}_{56}$	$\hat{\sigma} = 0.4 \pm 0.1$	0.35	0.3 ± 0.1	0.36
$^{139}_{Ba}_{56}$	$\hat{\sigma} = 6 \pm 2$		UNK	6
$^{139}_{La}_{57}$	$\sigma_0 = 9.2 \pm 3$	8.2	11.4 ± 0.7	8.35

TABLE 4 (Continued)

Isotope	Thermal Capture(b) ⁽¹⁵⁾	$\sigma_{Th}(b)$	I(b)	$\sigma_{Th} + 0.013I(b)$
⁸³ Kr ₃₆	$\sigma_o = 200 \pm 30$	177	150 $\pm$ 30	182
⁸⁴ Kr ₃₆	$\hat{\sigma} = 0.063 \pm 0.004^{**}$	0.056	4 $\pm$ 2 ^{**}	0.11 to ⁸⁵ Kr
	$= 0.07 \pm 0.01^{**}$	0.062	4 $\pm$ 2 ^{**}	0.11 to ⁸⁵ Br
⁸⁵ Kr ₃₆	$\hat{\sigma} = 1.66 \pm 0.20$	1.47	1.8 $\pm$ 1.0	1.5
⁸⁶ Kr ₃₆	$\hat{\sigma} = 0.06 \pm 0.03$	0.053	0.03 $\pm$ 0.03	0.054
¹²⁷ I ₅₃	$\sigma_o = 6.1 \pm 0.3$	5.4	150 $\pm$ 20	7.4
¹²⁸ I ₅₃	UNK	25 m Half-life	UNK	Not Required
¹²⁹ I ₅₃	$\sigma_o = 27 \pm 2$	24	36 $\pm$ 4	24.5
¹³⁰ I ₅₃	$\hat{\sigma} = 0.7 \pm 0.4$	0.62	8 $\pm$ 4	0.72

* UNK = Unknown (i.e., no experimental information).

** Values of Reference 15 adjusted for branching ratio of decay of isomeric state.

calculations were made for the inner edge of the blanket, the temperature dependent leakage effects are insignificant. For the actinides, many of the thermal neutron reaction cross sections are not $1/v$ and consequently the calculated reactions rates are temperature dependent. For ^{239}Pu in a blanket at $1000\text{ }^{\circ}\text{C}$, for example, the calculated reactions would be expected to be greater by a factor of more than three. For the actinides, however, the reaction rates calculated for the nuclides with non $1/v$ are all so high that changes of even this magnitude have little effect on the calculations. The actinide transmutation rates all depend primarily on even mass nuclides which do have thermal cross sections which are much smaller and have thermal cross-section energy variations which are sensibly $1/v$. Because of these reasons and because the calculated actinide transmutation rates are so high, the assumption of a significantly higher CTR blanket temperature is not expected to have a significant impact on the conclusions of this study.

It is interesting to note from Table 4 that experimental information on cross-section values is lacking for isotopes which are potentially sensitive to the transmutation calculations. For the FP isotope ^{90}Sr , there is no capture cross-section information available for the 9.48 hr ^{91}Sr neutron capture product. The capture cross section of ^{91}Sr is expected to be small since the nucleus contains 53 neutrons which is just over the magic number of 52 neutrons. For the ^{137}Cs Fp there is no capture cross-section information for either the 13 d ^{136}Cs n,2n product or the 32m ^{138}Cs capture product. Again, the ^{138}Cs capture cross section is expected to be

small ($N = 83$) and the half-life for decay is short. The ^{136}Cs half-life, however, is not negligible in a high neutron flux and the capture cross section may be very large ($N = 81$).

Receipt, while calculations were in progress, of two magnetic tapes containing fission product and actinide cross sections from M. R. Bhat and S. Pearlstein of the National Neutron Cross Section Center is acknowledged. This information is now available for use in subsequent studies.

2. Selection of $n,2n$ and $n,3n$ Cross-Section Data

The $n,2n$ and $n,3n$ cross sections as a function of neutron energy which were used in this study were calculated with the SOLWHO program which employs the formalism of Pearlstein.⁽⁹⁾ This formalism requires Q and threshold energy values as input. Definitions of these parameters are given in Reference 9. The Q values and threshold energies used by SOLWHO in these calculations are given in Table 5. Table 6 lists the data sources used in this evaluation. The values calculated with the SOLWHO program were compared for numerical accuracy with some of Pearlstein's published values. Effective cross sections were then calculated from these values using the formalism described previously. The effective cross-section values are given in Table 7 for the fission products. While actinide threshold data is given here for completeness, $n,2n$ and $n,3n$ cross sections were not used in the actinide analysis due to their relative insignificance when compared to the fission cross sections.

3. Selection of Cross Sections for Actinides

The reaction cross sections used in the actinide transmutation cases are summarized in Table 8. The procedure used here is the same as that outlined for the fission products.

4. Other Fast Neutron Reaction Cross Sections of Fission Products

The 14 MeV D-T source neutrons are, generally, at an energy above the threshold energy for a host of charged particle reaction products such as;

n,p

n,d

n,t

n, α

n,np

The effects of these reactions have not been included in the transmutation calculations. The cross sections are usually small and typically reach values of 1 to 100 mb at 14 MeV for FP nuclei considered here. There are only a few experimental values for these reaction cross sections which are relevant to the isotopes involved in these transmutation chains. In order to include these reactions in the transmutation calculations it would be necessary to make extensive use of values calculated from theoretical or semi-empirical models. The estimates obtained in this manner can be very unreliable as illustrated by comparing some experimental values with values predicted from Pearlstein's statistical model.⁽¹⁹⁾ Such a comparison can

TABLE 5

Q Values And Threshold Energies For n,2n and n,3n
Reactions For Fission Product Transmutation⁽²⁰⁾

<u>Isotope</u>	<u>n,2n</u>		<u>n,3n (If < 15MeV)</u>	
	<u>Q(MeV)</u>	<u>E_T(MeV)</u>	<u>Q(MeV)</u>	<u>E_T(MeV)</u>
⁸⁸ Sr ₃₈	-11.113	11.24		
⁸⁹ Sr ₃₈	-6.360	6.432		
⁹⁰ Sr ₃₈	-7.804	7.892	-14.164	14.323
⁹¹ Sr ₃₈	-5.827	5.892	-13.631	13.782
⁸⁸ Y ₃₉	-9.376	9.484		
⁸⁹ Y ₃₉	-11.468	11.598		
⁹⁰ Y ₃₉	6.860	6.937		
⁹¹ Y ₃₉	-7.946	8.034		
⁹² Y ₃₉	-6.557	6.629		
⁹⁰ Zr ₄₀	-11.983	12.117		
⁹¹ Zr ₄₀	-7.2026	7.283		
⁹² Zr ₄₀	-8.635	8.730		
⁹³ Zr ₄₀	-6.758	6.831		

TABLE 5 (Continued)

Isotope	n,2n		n,3n (If <15 MeV)	
	Q(MeV)	E _T (MeV)	Q(MeV)	E _T (MeV)
¹³³ Cs ₅₅	-8.979	9.047		
¹³⁴ Cs ₅₅	-6.891	6.943		
¹³⁵ Cs ₅₅	-8.825	8.891		
¹³⁶ Cs ₅₅	-6.769	6.819		
¹³⁷ Cs ₅₅	-8.2764	8.337		
¹³⁸ Cs ₅₅	-4.380	4.412 ^b	-12.656	12.749 ^a
¹³⁴ Ba ₅₆	-9.4644	9.535		
¹³⁵ Ba ₅₆	-6.9752	7.027		
¹³⁶ Ba ₅₆	-9.1071	9.175		
¹³⁷ Ba ₅₆	-6.9021	6.953		
¹³⁸ Ba ₅₆	-8.6115	8.674		
¹³⁹ Ba ₅₆	-4.7234	4.758 ^c	-13.335	13.432 ^b
¹³⁹ La ₅₇	-8.7778	8.842		

^a Not needed for transmutation calculations because of 32 min. half-life for radioactive decay.

^b Not needed for transmutation calculations because of 83 min. half-life for radioactive decay.

TABLE 5 (Continued)

<u>Isotope</u>	<u>n,2n</u>		<u>n,3n (If < 15 MeV)</u>	
	<u>Q(MeV)</u>	<u>E_T(MeV)</u>	<u>Q(MeV)</u>	<u>E_T(MeV)</u>
⁸³ Kr ₃₆	-7.467	7.558		
⁸⁴ Kr ₃₆	-10.518	10.644		
⁸⁵ Kr ₃₆	-7.111	7.195		
⁸⁶ Kr ₃₆	-9.860	9.976		
¹²⁷ I ₅₃	-9.139	9.212		
¹²⁸ I ₅₃	-6.8254	6.879 ^c		
¹²⁹ I ₅₃	-8.840	8.909		
¹³⁰ I ₅₃	-6.456	6.506		

^cNot needed for transmutation calculations because of 25 min. half-life for radioactive decay.

TABLE 6

Data Sources For Cross Sections For
Actinide Transmutation and n,2n and n,3n Q Values
And Threshold Energies For New Files

Isotope	Data File	n,2n		n,3n	
		Q(MeV)	E _T (MeV)	Q(MeV)	E _T (MeV)
²³⁷ Np	BNWML MAT 511				
²³⁸ Np	BNWML MAT 371				
²³⁸ Pu	BNWML MAT 507				
²³⁹ Pu	ENDF/B MAT 1159				
²⁴⁰ Pu	ENDF/B MAT 1105				
²⁴¹ Pu	ENDF/B MAT 1106				
²⁴² Pu	ENDF/B MAT 1161				
²⁴³ Pu ^a					
²⁴¹ Am	BNWML MAT 505				
²⁴² Am ^b					
²⁴³ Am	BNWML MAT 506				
²⁴⁴ Am ^c					
²⁴² Cm		-6.968	6.997	-13.038	13.092
²⁴³ Cm		-5.703	5.727	-12.671	12.724
²⁴⁴ Cm	ENDF/B MAT 1162				
²⁴⁵ Cm		-5.5199	5.543	-12.319	12.370
²⁴⁶ Cm		-6.4507	6.477	-11.9706	12.020

^a Reaction cross sections not needed for transmutation calculation because of 4.96 hr. half-life for radioactive decay.

^b The thermal fission cross section of ~2000b is only reaction cross section required for transmutation calculation because of 16 hr. half-life for radioactive decay.

^c The thermal fission cross section of ~2200b is the only reaction cross section required for transmutation calculation because of 10.1 hr. half-life for radioactive decay.

TABLE 7

n,2n and n,3n Effective Cross Sections For The
Neutron Flux Spectrum Used in the Transmutation Calculations

<u>Isotope</u>	<u>$\sigma_{n,2n}(\text{mb})$</u>	<u>$\sigma_{n,3n}(\text{mb})$</u>
$^{88}\text{Sr}_{38}$	77.75	
$^{89}\text{Sr}_{38}$	169.5	
$^{90}\text{Sr}_{38}$	151.0	
$^{91}\text{Sr}_{38}$	180.4	3.56
$^{88}\text{Y}_{39}$	106.8	
$^{89}\text{Y}_{39}$	61.45	
$^{90}\text{Y}_{39}$	155.2	
$^{91}\text{Y}_{39}$	142.0	
$^{92}\text{Y}_{39}$	167.0	
$^{90}\text{Zr}_{40}$	46.61	
$^{91}\text{Zr}_{40}$	146.1	
$^{92}\text{Zr}_{40}$	129.0	
$^{93}\text{Zr}_{40}$	162.2	

TABLE 7 (Continued)

Isotope	$\sigma_{n,2n}(\text{mb})$	$\sigma_{n,3n}(\text{mb})$
$^{133}\text{Cs}_{55}$	172.6	
$^{134}\text{Cs}_{55}$	212.6	
$^{135}\text{Cs}_{55}$	174.9	
$^{136}\text{Cs}_{55}$	220.4	
$^{137}\text{Cs}_{55}$	194.9	
$^{138}\text{Cs}_{55}$	220.6	48.5
$^{134}\text{Ba}_{56}$	159.7	
$^{135}\text{Ba}_{56}$	209.3	
$^{136}\text{Ba}_{56}$	167.0	
$^{137}\text{Ba}_{56}$	216.6	
$^{138}\text{Ba}_{56}$	187.0	
$^{139}\text{Ba}_{56}$	244.2	17.95
$^{139}\text{La}_{57}$	182.7	
$^{83}\text{Kr}_{36}$	146.6	
$^{84}\text{Kr}_{36}$	98.85	
$^{85}\text{Kr}_{36}$	161.3	
$^{86}\text{Kr}_{36}$	117.5	

TABLE 7 (Continued)

<u>Isotope</u>	$\sigma_{n,2n}(\text{mb})$	$\sigma_{n,3n}(\text{mb})$
$^{127}\text{I}_{53}$	171.7	
$^{128}\text{I}_{53}$	213.3	
$^{129}\text{I}_{53}$	178.9	
$^{130}\text{I}_{53}$	222.5	

TABLE 8

Cross Sections Of Actinides For Transmutation Calculations

Isotope	Capture			Fission		
	σ_0 (b)	I (b)	$\bar{\sigma}_c$ (b)	σ_0 (b)	I (b)	$\bar{\sigma}_f$ (b)
^{238}U (21)	78	237	72	81	280	75
^{237}Np (21)	169	664	159	0.017	0.067	0.016 (0.040)*
^{238}Np (21)	100	100	90	1600	1500	1440
^{239}Np (21)	60		60	1		1
^{236}Pu (21)	85	700	85	170	1400	170
^{238}Pu (21)	546	127	485	16.3	34.5	15
^{239}Pu (21)	271	200	275	742	304	697
^{240}Pu (21)	290	8484	374	0.058	9	0.17
^{241}Pu (21)	368	1162	344	1007	581	944
^{242}Pu (21)	20	1180	33		5.3	0.07
^{243}Pu (15)	100		90	200		180
^{241}Am (21) to ^{243}Am	94	230	86	3.15	7.5	2.9
to ^{242}Gm	486	1190	446			
$^{242\text{m}}\text{Am}$ (152 y) (15)	2000	2000	1800	6500	1570	5800
^{242}Gm (16 h) (15)	200 (18)	200 (18)	180	2000	2000 (18)	1800
^{243}Am (23)	90	1770	103		4.0 (22)	0.05
^{244}Am (15)	220 (18)	220 (18)	200	2200	2200 (18)	2000
^{242}Cm (21)	17.3	150	17.3	2.9	25	2.9
^{243}Cm (15)	300	400	270	700	1900	645

*Effective n,2n cross section to ^{236}Pu . To be used with $\bar{\sigma}_f$.

TABLE 8 (Continued)

<u>Isotope</u>	<u>Capture</u>			<u>Fission</u>		
	<u>$\sigma_0(b)$</u>	<u>I(b)</u>	<u>$\bar{\sigma}_Y(b)$</u>	<u>$\sigma_0(b)$</u>	<u>I(b)</u>	<u>$\bar{\sigma}_f(b)$</u>
^{244}Cm (23,24)	12.6	611	19.1	1.1	19(25)	1.34
^{245}Cm (23,24)	375	106	334	2160	768	1924
^{246}Cm (23,24)	1.2	121	2.64	0.2	10.6	0.34
^{247}Cm (23)	55	500	55	98	800	97
^{248}Cm (23,25)	1.35	240	4.3	0.37	14.2	0.18

be made for $^{88}\text{Sr}_{38}$ as shown on Table 9. The values shown illustrate two points. One is that where experimental data exist the accuracies are so poor that the resultant effects would vary over rather wide limits. The second point is that in the absence of experimental data the estimated value might differ by an order of magnitude from the true value. The effects of such large uncertainties in the transmutation chains would be difficult to assess.

As a general rule, the (n,p) and (n, γ) cross sections near 14 MeV neutron energy are larger than the other charged particle reactions which might be energetically possible. Consequently, an assessment of the uncertainties which might exist from the neglect of (n,p) and (n, α) processes in the transmutation calculations has been made.

The (n,p) reactions on the primary FP isotopes can generally be excluded from transmutation calculations since the product nuclei each undergo rapid beta decay to the original FP nucleus. Reactions for the FP isotopes considered here are:

TABLE 9

Comparison Of Experimental Values Of Cross Sections
With Predicted Values for $^{88}\text{Sr}_{38}$

E_n (MeV)	$\sigma(n,p)(\text{mb})$		$\sigma(n,\alpha)(\text{mb})$	
	Exp ⁽²⁶⁾	Theory (19)	Exp (26)+	Theory (19)
14.0	18 $\pm$ 3	8.8		2.0
14.5	17.7 $\pm$ 3.5		64 $\pm$ 20	
14.6	30 $\pm$ 12		87 $\pm$ 13	
14.8	11.2			
15.0		11		2.1

+ Cross section for production of 4.4h $^{85\text{m}}\text{Kr}$.

For ^{85}Kr , ^{90}Sr , and ^{137}Cs the half-lives of the product isotopes are sufficiently short such that the neutron reaction rates would be insignificant compared to the radioactive decay rates. The neutron reaction rates on the 33.4 d $^{129\text{m}}\text{Te}$ would be significant but the dominant neutron reactions on $^{129\text{m}}\text{Te}$ both lead to stable nuclei.

The (n,p) reactions on the products produced by neutron capture reactions on the primary FP isotopes can also be neglected because of the short half-lives of these isotopes or their products for radioactive decay. These are:

The possible effects of n,α reactions on the primary FP or transmutation product isotopes are much more complicated. The general effect is to produce isotopes which are already being produced at much faster rates by multiple neutron capture, $(n,2n)$, and radioactive decay. One possible exception which deserves study in future investigations has been noted. In the transmutation of ^{137}Cs or chemical FP Cs it is possible to form the isotope ^{135}Cs which has a radioactive decay half life of 2.3×10^6 years. The formation of significant quantities of ^{135}Cs is very undesirable. ^{135}Cs would be normally formed by

In addition, ^{135}Cs can result from

The thermal capture cross section of ^{134}Xe is 0.25b which is comparable to the ^{137}Cs capture cross section. Thus, if the $^{137}\text{Cs} (n,\alpha)$ rate is appreciable, the formation of ^{135}Cs via this chain might exceed the rate of production via successive $n,2n$ reactions from ^{137}Cs . No estimates of the $^{137}\text{Cs}(n,\alpha)$ cross section have been located as yet.

The fact that the charged particle reactions were not considered in the analysis due to the reasons given above does not mean, however, that future studies, done in more detail, should not include this effect.

D. Calculation of Neutron Source Strengths

In order to perform the required analysis, neutron wall loadings had to be selected and then converted into neutron source strengths. The following sections detail the procedure used in this study.

1. Neutron Wall Loading

(This portion of the analysis is based on comments received from G. L. Kulcinski, U. W.; G. A. Emmert, U.W.; C. Taylor, LLL; and D. Steiner, ORNL).

For a D-T Tokamak, the neutron current can be related to the plasma conditions by:

$$J = f \frac{n^2}{4} \langle \sigma v \rangle \frac{2\pi a^2 R}{4\pi^2 a R} \quad (24)$$

where:

J = neutron current

n = ion density

$\langle \sigma v \rangle$ = fusion reaction rate

a = minor radius of plasma

R = major radius of plasma

f = factor to account for spatial profile in density and temperature.

The following constraints are dictated by plasma physics considerations.

$$1) \quad \beta_p = \frac{2n kT}{\frac{B_p^2}{2\mu_0}} = \sqrt{A}, \quad A = \frac{R}{a}$$

$$2) \quad q = \frac{1}{A} \frac{B_t}{B_p} \geq 1.75$$

3) sufficient room for the core to energize the plasma current

$$r_o = a \left[\frac{R - \tau_b - a}{Rq} (\ln 8A - 1.75) \right]^{1/2}$$

$$r_o + H + \tau_b + a \leq R$$

where

β_p = ratio of magnetic pressure to the plasma pressure

k = Boltzmann constant

T = temperature

B_p = Poloidal field strength

μ_0 = permeability of free space

B_t = Toroidal field strength

τ_b = vacuum gap + blanket + shield thickness

H = Combined thickness of primary and toroidal magnet windings

Temporarily ignoring the last constraint and considering only the first two.

$$J = \frac{f \langle \sigma v \rangle}{4} \left[\frac{\beta_p B_p^2}{\mu_0 4kT} \right]^2 \frac{a}{2} \quad (25)$$

$$J = \frac{f B_T^4}{\mu_0^2 (128) q^4 A^3} \frac{\langle \sigma v \rangle}{(kT)^2} a \quad (26)$$

Hence, one wants small q , small A and $\frac{\sigma v}{T^2} = \max (T \approx 15 \text{ keV})$ and large B_T to maximize ϕ for a given a .

The minimum value for A is determined by the core constraint 3; generally the minimum acceptable A is in the range 2.5 - 3. The ability to reduce the blanket on the inside of the torus can perhaps reduce this, but not by much because of the need to shield the magnets.

The maximum J Tokamak differs from the most economic reactor essentially in B_T . Increasing B_T increases J drastically while it can hurt the economics of a power reactor. An approximate J scaled from the Wisconsin design (5000 MW) is

$$J = \phi_0 \left(\frac{B_m}{3.86} \right)^4 \left(\frac{a}{5} \right) \quad (27)$$

where

J_0 = 14 MeV current in "unstable" 5000 MW design

B_m = maximum allowed magnetic field strength in Tesla on the axis
of the plasma

a = minor radius of plasma in meters

R = 2.6 a

The device will necessarily produce a large amount of power. For example, one gets

$$P = P_0 \left(\frac{B_m}{3.86} \right)^4 \left(\frac{a}{5} \right)^3 \quad (28)$$

where $P_0 = 5000$ MW

In looking at mirror plasma devices, it would appear that in general they produce higher neutron wall loadings. Because operation at higher values of β is allowed, higher plasma reaction rates are possible. Recent D-T power reactor system design studies have used neutron wall loadings in the range from 1.6 to 3.0 MW/m². A recent study of high intensity beam driven minor devices using low temperature cold plasma targets has been analyzed and it was found that, in principle, these devices could produce neutron wall loadings in the range of 100 MW/m², or higher.

Thus, it is concluded that it would be extremely difficult to obtain a 10 MW/m² D-T neutron wall loading in a Tokamak device. It does appear possible, however to achieve a 10 MW/m² neutron wall loading in a mirror device. Thus, in the spirit of this analysis, which is to explore the possible range for CTR transmutation, neutron wall loadings of 1 MW/m² and 10 MW/m² were selected for use in the analysis.

2. The Thermal-Neutron Flux

The base calculational model is a Tokamak with 160 cm radius plasma and a beryllium blanket between the radii 200 cm and 300 cm. The flux spectra and intensities are obtained from the average values in the first cm of blanket thickness.

The total neutron current from the plasma, J , is related to the neutron power wall loading- P_w , by:

$$\begin{aligned} P_w &= 1.609 \times 10^{-19} \frac{\text{W sec}}{\text{ev}} \times 1.406 \times 10^7 \frac{\text{eV}}{\text{n}} \cdot J \frac{\text{n}}{\text{cm}^2 \text{sec}} \times \frac{10^4 \text{cm}^2}{\text{m}^2} \times 10^{-6} \frac{\text{MW}}{\text{W}} \\ &= 2.262 \times 10^{-14} J \frac{\text{MW}}{\text{m}^2} \end{aligned} \quad (29)$$

For the 200 cm radius, 1 cm high cylinder, the current is related to the total neutron source rate, N , in the plasma by:

$$J = \frac{N}{2\pi \cdot 200} = 7.958 \times 10^{-4} N \quad (30)$$

$$N = 5.555 \times 10^{16} P_w \quad (31)$$

The reaction rate in a blanket zone of volume V_z for a material of density ρ (at-bn⁻¹-cm⁻¹) is related to the calculated total activity in the zone per source neutron, A , by

$$\begin{aligned} \frac{A}{\sigma \phi} &= \frac{AN \cdot 10^{-24}}{V_z \rho} \\ &= 5.555 \times 10^{-8} \frac{A P_w}{V_z \rho} \end{aligned} \quad (32)$$

For the reference 1 cm thick Be zone, the capture cross sections are obtained from the calculated ⁹Be (n, γ) activities.

$$A = 5.5099 \times 10^{-2} \text{ sec}^{-1}$$

$$V_z = 1.260 \times 10^3 \text{ cm}^3$$

$$\rho = 9.8 \times 10^{-2} \text{ at-bn}^{-1}\text{-cm}^{-1}$$

$$\overline{\sigma\phi} = \frac{5.555 \times 10^{-8} \times 5.5099 \times 10^{-2}}{1.260 \times 10^3 \times 9.8 \times 10^{-2}} P_w$$

$$= 2.479 \times 10^{-11} P_w \text{ sec}^{-1} \text{ MW}^{-1} \text{ m}^2$$

$$= 2.479 \times 10^{-10} \text{ sec}^{-1} \text{ for } P_w = 10 \text{ MW/m}^2$$

In order to scale capture cross sections of other isotopes, one must calculate an effective flux to use with the $(\sigma_{Th} + r I)$ formalism. For Be (n, γ)

$$\sigma_{Th} = 0.842 \times 10^{-26} \text{ cm}^2$$

$$I = 0.421 \times 10^{-26} \text{ cm}^2$$

$$r = \frac{\phi_{epi}}{\phi_{Th}} = \frac{8.502 \times 10^5}{6.615 \times 10^7} = 0.013$$

where ϕ_{Th} is the thermal flux in the zone and ϕ_{epi} is the low energy limit of the slowing down flux per unit lethargy. Then

$$\sigma_{Th} + rI = 0.8475$$

and

$$\bar{\phi} = \frac{2.479 \times 10^{-10}}{0.8475 \times 10^{-26}} = 2.925 \times 10^{16} \text{ n-cm}^{-2}\text{-sec}^{-1} \text{ for } P_w = 10 \text{ MW/m}^2 .$$

3. The Fast-Neutron Flux for Transmutation Calculations

In Figure 2, the neutron energy spectrum in the beryllium blanket region which was used in the transmutation calculations is seen to be significantly different from a $1/E$ shape at energies above approximately 3 keV. Thus, it was anticipated that the resonance integral formalism would not be an adequate representation for the flux or for the cross sections of threshold reactions. An effective fast flux, ϕ_f , was obtained from the calculated $\text{Be}(n,2n)$ events in the reference spectrum. Effective cross sections were calculated from

$$\bar{\sigma} = \frac{\int \phi(u) \sigma(u) du}{\int \phi(u) du} \quad (34)$$

where both integrations were carried out over the 99 lethargy groups of the GAM II structure⁽²⁷⁾ of the calculated flux spectrum above the thermal group. For the $\text{Be}(n,2n)$ reaction $\bar{\sigma}$ was found to be 84.66 mb. The Be activity predicted by the numerator of Eq. 34 was identical to the epithermal activity calculated by ANISN. The normalization of the fast flux is proportional to the neutron power wall loading (P_w). The following expression, similar to that used to obtain the thermal flux, was used to calculate the fast flux.

$$\begin{aligned} \langle \bar{\sigma} \phi_f \rangle &= 5.555 \times 10^{-8} \frac{A P_w}{V_z \rho} \\ &= \frac{5.555 \times 10^{-8} \times 0.154}{1.26 \times 10^3 \times 9.8 \times 10^{-2}} P_w \\ &= 0.6928 \times 10^{-10} P_w \text{ sec}^{-1} \text{ MW}^{-1} \text{ m}^2. \\ &= 6.928 \times 10^{-10} \text{ sec}^{-1} \text{ for } P_w = 10 \text{ MW/m}^2. \end{aligned} \quad (35)$$

Using the Be(n,2n) cross section —

$$\begin{aligned}\phi_f &= \frac{6.928 \times 10^{-10}}{8.466 \times 10^{-26}} \\ &= 8.183 \times 10^{15} \text{ n-cm}^{-2}\text{-sec}^{-1}\end{aligned}$$

for $P_w = 10 \text{ MW/m}^2$.

It should be pointed out that flux levels of the order of $10^{16} \text{ n-cm}^{-2}\text{-sec}^{-1}$ will undoubtedly cause severe engineering problems. However, this and other engineering considerations will need to be evaluated in future work.

4. Flux Levels For The Deuterium-Deuterium (D-D) Thermonuclear Plasma

The D-D thermonuclear plasma operates at ion temperatures of the order of 100 keV and produces neutrons with a wide spread of energies relative to the mean. The finite width of the neutron energy spectrum from the 10 keV DT plasma is in general, not important. This is due in part to the fact that there is usually little structure in the neutron cross sections of most elements around 14 MeV relative to the width of the neutron spectrum from the D-T source. The D-D source, however, yields a neutron spectrum covering some 0.5 to 1 MeV depending on the assumed ion temperature (if Maxwellian) or the characteristics of the driving component (e.g., if an accelerator driven ion source is employed). The details of the D-D neutron spectrum would be important in determining the neutronic characteristics of the blanket for most materials since the energy width of the spectrum would ordinarily cover appreciable fine structure in the microscopic cross sections (e.g., inelastic scattering channel). For the case of the pure

beryllium blanket there are two threshold reactions that present structure near the DD source energy. One of these is the ${}^9\text{Be} (n, \alpha) {}^6\text{He} \rightarrow {}^6\text{Li}$ reaction with a threshold energy of 0.67 MeV. The other, of greater potential importance, is the ${}^9\text{Be} (n, 2n)$ reaction with a threshold of 1.85 MeV.

In the transmutation calculations an extended D-D neutron spectrum was not used to calculate reaction rates. The GAM-II group structure used in ANISN would allow, at best, a rather coarse representation of the neutron source spectrum. Consequently the effect of an extended source was investigated by comparing calculations in which the neutron source was placed in each of two GAM energy groups which differed significantly in energy. The results of these calculations are shown on Table 10. The D-T plasma results are also shown for comparison. The beryllium activities shown are the reactions per source neutron in the first 1 cm zone of the blanket. It is noted that there is an order of magnitude difference in the $\text{Be}(n, 2n)$ rates for the two D-D neutron source energies. However, the beryllium capture rate in the two cases differs by less than 10% and the neutron spectrum coefficient, r , is not much different and also is little different from the D-T source result.

In order to make the transmutation calculations, essentially mean values were used which were obtained from the two D-D neutron energy source cases. These values appear to be within some 10% of what would be obtained from some model D-D plasma of about 100 keV ion temperature. The assumptions are:

1. The mean D-D source neutron energy is 2.75 MeV. This assumption leads to

$$\sigma\phi = 2.84 \times 10^{-7} \frac{AP_w}{V_z \rho} \quad (36)$$

where the symbols are the same as previously given for the D-T source.

2. The Be(n, γ) activity, A, is 0.022 per source neutron. Thus,

$$\langle \sigma\phi \rangle \cong 5.1 \times 10^{-10} \text{ sec}^{-1} \text{ for } P_w = 10 \text{ MW/m}^2$$

and

$$\phi_{th} \cong 6 \times 10^{16} \text{ n-cm}^{-2}\text{-sec}^{-1}.$$

3. Because of the relatively small changes in the spectral index, r, the capture cross sections used in the D-T source spectrum were unchanged for the D-D source transmutation calculations of fission products.

TABLE 10

The Results of ANISN Calculations for the D-D Neutron Source
Placed in Two Different Energy Groups

Energy Group in the Source in GAM-II Energy Group Boundaries (MeV)	D D Source		D T Source
	19	17	1
	2.23-2.47	2.73-3.01	13.5-14.9
Be Activity ^a			
n,γ	0.021	0.0229	0.0551
n,α	0.015	0.0227	0.0109
n,2n	0.0024	0.0328	0.154
$\phi_{th} (n\text{-cm}^{-2}\text{-s}^{-1})^a$	25.15	27.38	66.15
$\phi(u)$	0.357	0.403	0.85
r	0.0142	0.0147	0.0129
$\bar{E}_n$ (MeV)	2.35	2.87	14.2

a) Per source neutron

V. RESULTS OF NEUTRONICS ANALYSIS

An extensive series of calculations has been performed on the transmutation of fission products and actinide wastes in a Controlled Thermonuclear Reactor. These calculations, performed for a number of different target loadings, have been performed as a function of neutron wall loading (i.e., neutron source strength), plasma type (D-T, D-D, etc.), and time. Individual calculations are identified by calculational case number in Table 11. The results of the calculations are discussed below. Separate discussions are given for fission product and actinide transmutation due to the different phenomena involved. In the following discussion results are presented in terms of the radiological hazard associated with one ton of LWR fuel exposed to 33,000 MWD/MT, reprocessed, and cooled for 10 years. The radiological hazard is evaluated by use of the term hazard index, which is defined as the cubic meters of water required to dilute the present amount of waste to levels defined in the Code of Federal Regulations (10 CFR, Part 20).

The calculated hazard index is plotted as a function of time in the following figures. Also on these graphs can be found the hazard index associated with a ton of aged natural uranium. Aged uranium is uranium which has been mined and then stored (not exposed in a reactor). The hazard index associated with mined uranium constitutes an arbitrary measure of

TABLE 11
SUMMARY OF CALCULATIONS

<u>Case Number</u>	<u>Target Loading</u>	<u>Plasma</u>	<u>Comments</u>
1	^{90}Sr	D-T	Calculated for wall loading of $1 \frac{\text{MW}}{\text{m}^2}$, and $10 \frac{\text{MW}}{\text{m}^2}$
2	^{137}Cs	D-T	"
3	Sr^*	D-T	"
4	Cs^*	D-T	"
5	^{85}Kr	D-T	"
6	^{129}I	D-T	"
7	^{90}Sr	D-D	"
8	^{137}Cs	D-D	"
9	Actinides (AM+CM+NP) + 0.5% Pu	D-T	"
10	Actinides + 0.5% Pu	D-D	"
11	^{237}Np	D-T	Calculated for wall loading of $1 \frac{\text{MW}}{\text{m}^2}$
12	AM + CM	D-T	"
13	Pu	D-T	"
14	Actinides + Pu	D-T	"

*This is the Sr and Cs isotopic composition as derived from exposure of UO_2 fuel in a LWR to 33,000 $\frac{\text{MWD}}{\text{MTM}}$.

the effort required to reduce the hazard associated with exposure of the uranium in a reactor to that of unirradiated natural uranium.

In discussing the results of these calculations, extensive use will be made of the concept of a hazard half-life, which is here defined as the time required for the hazard index from a particular isotope and its daughters to decay to one half its original value. Notice that because the hazard index of the whole chain (i.e., parent plus daughters) is included in the definition, that the hazard half-life can vary from the decay half-life. One measure of the effectiveness of a transmutation concept is the change in hazard half-life associated with various levels of neutron irradiation. The concept of a hazard half-life is useful in explaining the results of the calculations and their meaning. In order to understand its applicability, however, one must understand how it is used in this document.

As a convention in this document, the hazard half-life is calculated from the slope of a straight line joining the end points of the curves on the semi-log plots of hazard index vs time. For data which is linear, this is analogous to the calculation of radioactive half-life from a decay curve. For nonlinear data, however, this assignment of hazard half-life has no more meaning than a corresponding assignment of decay half-life to a mixture of radioactive species. In particular, it may not be used to predict the time for reduction of the hazard to background levels.

Consequently, the reader should inspect the data as well as the corresponding hazard half life value to determine the relative usefulness of this assignment.

A. Fission Product Transmutation

Fission product transmutation in a CTR was studied for Sr, Cs, Kr, and I. These waste constituents were selected as prime transmutation possibilities because of their hazard potential. Sr, Cs and Kr contain most of the hazard (>90%) associated with fission products with Sr generally considered to be the most hazardous. ^{90}Sr and ^{137}Cs have natural half lives of the order of thirty years and ^{85}Kr of the order of ten years. Iodine, or more specifically, ^{129}I was selected as a possible target for other considerations. While its immediate hazard index is quite low relative to the other fission products considered, it is unique in that it has a decay half life of 1.6×10^7 years. Therefore, in contrast to Sr, Cs and Kr, which will essentially disappear by decay after a few hundred years, ^{129}I constitutes a very long-term hazard.

1. Transmutation of ^{90}Sr

^{90}Sr transmutation was evaluated as a function of neutron wall loading, plasma type, and target composition. The calculational cases pertinent to this analysis are cases 1, 3, and 7 listed in Table 11. The Sr transmutation chains analyzed are shown in Appendix I, Figure A-I.1, and Figure A-I.6. Cases 1 and 3 use the chain shown on Figure A-I.1 and case 7 uses the chain shown on Figure A-I.6.

Considering first the transmutation of isotopic ^{90}Sr , the results of the analysis for a D-T plasma are shown in Figure 3 as a function of neutron wall loading and time. This figure displays the change in isotopic hazard index per ton of fuel for the ^{90}Sr as a function of time and wall loading.

FIGURE 3. TRANSMUTATION OF ^{90}Sr IN A D-T FLUX

From Figure 3, the natural decay case, the hazard index is shown to be not quite linear. This is due to temporary daughter buildup early in time. This phenomenon is also present in the 10 MW/m^2 neutron wall loading case but is not evident in the 1 MW/m^2 case. (See Appendix V for detailed data breakdown).

The change in hazard half-life derived from these curves is listed in Table 12 as a function of wall loading.

TABLE 12

Hazard Half-Life As A Function Of
Wall Loading For ^{90}Sr In A D-T Flux

<u>Wall Loading (MW/m^2)</u>	<u>Hazard Half Life (yrs)</u>	<u>Time to Background*(yrs)</u>
Decay Only	31.9	439
1	7.55	104
10	2.35	17.8

The data in Table 12 shows, that in the absence of transmutation, the hazard half-life is not markedly different from the natural decay half-life of ^{90}Sr and that some 439 years are required to reduce the ^{90}Sr back to its pre-irradiation hazard index with no transmutation. Transmutation at a rate equivalent to 1 MW/m^2 reduces isotopic ^{90}Sr to background in 104 years and, at a wall loading of 10 MW/m^2 , the reduction to background is accomplished in 18 years.

* Time required for the hazard index to reach a value equivalent to natural uranium.

Sr does not, however, appear in the fission product mass as ^{90}Sr only. Rather, the Sr is made up of ^{88}Sr and ^{90}Sr . ^{88}Sr is stable and does not contribute to the Sr hazard index. It does, however, compete with ^{90}Sr for neutrons in the transmutation process, and does create hazardous daughters. Isotopic separation of ^{90}Sr from ^{88}Sr could be extremely expensive compared to chemical separation of Sr from the fission product mass. In addition, the separation process itself could generate significant waste. Thus, it was decided that an analysis of the Sr present as fission products (i.e., ^{88}Sr and ^{90}Sr) should be performed. The results of this analysis, as a function of wall loading, are displayed in Figure 4 for a D-T flux. The hazard half-lives as a function of wall loading are given in Table 13.

TABLE 13

Hazard Half-Life As A Function Of Wall Loading
For FP Sr In A D-T Flux

<u>Wall Loading (MW/m^2)</u>	<u>Hazard Half Life (yrs)</u>	<u>Time to Background (yrs)</u>
Decay Only	31.9	439
1	8.03	111
10	7.36	101

For purposes of comparison, the hazard for FP Sr vs ^{90}Sr are both shown in Figure 5 for a wall loading of $10 \text{ MW}/\text{m}^2$ in a D-T flux.

A data on Figure 4 indicates that the hazard half-life for $10 \text{ MW}/\text{m}^2$ may be much longer than that shown in Table 13. In this case, the hazard half-life was calculated using the beginning and end points of the $10 \text{ MW}/\text{m}^2$

FIGURE 4. TRANSMUTATION OF Sr IN A D-T FLUX

FIGURE 5. ELEMENTAL Sr AND ^{90}Sr TRANSMUTATION IN A D-T FLUX WITH A WALL LOADING OF 10 MW/m^2

curve. However, as can be seen from Figure 4, the slope of the 10 MW/m^2 curve approaches that of the no irradiation curve beyond 10 years. With extrapolation, it appears that the 10 MW/m^2 curve would eventually cross and be above the 1 MW/m^2 curve due to radioactive daughter buildup.

The data in Table 13 and on Figures 4 and 5 shows that isotopic separation is attractive at higher wall loadings (i.e., 10 MW/m^2), but that it is not attractive at 1 MW/m^2 . The higher wall loading results in buildup of hazardous daughters of ^{88}Sr which contribute greatly to the total hazard index. Thus, if the transmutation concept uses a wall loading of 1 MW/m^2 , isotopic separation is not required. However, for a wall loading of 10 MW/m^2 , needs separation of ^{90}Sr is required to make a significant change in the hazard half-life over that found for a wall loading of 1 MW/m^2 .

Transmutation in a CTR can potentially be carried out by means of a D-T or a D-D plasma source. For an equivalent neutron wall loading, the D-D flux will be higher. However, a D-T source potentially allows for transmutation events (i.e., $n, 2n$, etc.) that would not take place when a D-D plasma is used.

Figure 6 displays the results of an analysis of the transmutation of ^{90}Sr with a D-D plasma for wall loadings of 1.0, and 10.0 MW/m^2 . Table 14 gives the hazard half life as a function of wall loading as well as the time to background for the various wall loadings.

FIGURE 6. TRANSMUTATION OF ^{90}Sr IN A D-D FLUX

TABLE 14
Hazard Half-Life As A Function Of Wall Loading
For ^{90}Sr In a D-D Flux

<u>Wall Loading (MW/m^2)</u>	<u>Hazard Half Life (yrs)</u>	<u>Time to Background (yrs)</u>
Decay Only	31.9	439
1	4.68	64.5
10	0.479	6.6

The results shown in Table 14 when compared to those in Table 12 indicates the advantage of using a D-D plasma as a transmutation source for ^{90}Sr . The hazard half-life is almost halved going from a D-T to a D-D plasma at $1 \text{ MW}/\text{m}^2$ and is reduced by two-thirds for $10 \text{ MW}/\text{m}^2$. Figure 7 displays the change in hazard with time for both the D-T and D-D sources at $1 \text{ MW}/\text{m}^2$.

2. Transmutation of ^{137}Cs

Transmutation of ^{137}Cs was evaluated as a function of neutron wall loading, plasma type and target composition. The calculational cases pertinent to this analysis are cases 2, 4, and 8 of Table 11. The transmutation chains analyzed for Cs transmutation are shown in Appendix I, Figures A-I.2, A-I.3 and A-I.7.

FIGURE 7. TRANSMUTATION OF ^{90}Sr IN A D-D AND D-T FLUX AT 1.0 MW/m^2

The transmutation analysis for ^{137}Cs is somewhat more complex than that associated with the analysis of ^{90}Sr . This is because daughter products of ^{137}Cs are generated which have a high hazard index but which have a short half life. Thus, the immediate hazard index associated with ^{137}C rises while the true long term hazard index drops. This phenomenon is illustrated in Figure 8 which displays the transmutation of ^{137}C via the chain of Figure A-I.2 for a D-T plasma with a neutron wall loading of 10 MW/m^2 .

The upper curve on this figure shows that the hazard index as a function of time is everywhere higher than the initial hazard index (about 4×10^8) until about 55 years exposure. However, inspection of the individual components of this hazard index shows that a large percentage of the hazard index is contributed by ^{138}Cs and ^{139}Ba , both of which have half-lives much less than one day. If the target and its daughters are removed from the neutron flux and the ^{138}Cs and ^{139}Ba decay the long term hazard index composed mainly of ^{136}Cs and ^{137}Cs is the lower curve on this figure. Thus, transmutation of ^{137}Cs , while it obviously decreases the disposal hazard, actually increases the short term hazard index of the material while in the operating facility.

Figure 9 displays the transmutation of ^{137}Cs in a D-T flux with a neutron wall loading of 1 MW/m^2 . The same phenomenon is observed. The upper curve on this figure is the "as-produced" hazard index. The lower curve is the hazard index after decay of ^{138}Cs and ^{139}Ba .

FIGURE 8. TRANSMUTATION OF ^{137}Cs IN A D-T FLUX AT 10 MW/m^2

FIGURE 9. TRANSMUTATION OF ^{137}Cs IN A D-T FLUX AT 1 MW/m^2

Figure 10 is a display of the waste disposal hazard index from transmutation of ^{137}Cs in a D-T flux. The figure is made up of the lower, or decay, curves from Figures 8 and 9. Therefore, it shows the transmutation rate of ^{137}Cs as a function of neutron wall loading for no neutron transmutation and for wall loadings of 1 MW/m^2 and 10 MW/m^2 after decay of ^{138}Cs and ^{139}Ba . As shown, the hazard index is always decreasing. Table 15 gives the hazard half-life as a function of neutron wall loading for transmutation of ^{137}Cs in a D-T flux.

TABLE 15

Hazard Half-Life As A Function Of Wall Loading
For ^{137}Cs in a D-T Flux

<u>Wall Loading (MW/m^2)</u>	<u>Hazard Half Life (yrs)</u>	<u>Time to Background (yrs)</u>
Decay Only	30.0	147
1	18.6	91.0
10	5.61	27.5

The hazard half-life for ^{137}Cs with no transmutation is equal to its natural half-life, namely 30 years. This is because ^{137}Cs decays directly to a stable daughter. Also, the time for decay to background for ^{137}Cs (147 years) is markedly less than that of ^{90}Sr (439 years) indicating that ^{137}Cs is less of a waste problem than ^{90}Sr .

FIGURE 10. TRANSMUTATION OF ^{137}Cs IN A D-T FLUX AS A FUNCTION OF WALL LOADING

Transmutation of ^{137}Cs reduces the time to background to 91 years for a wall loading of 1 MW/m^2 , and for a wall loading of 10 MW/m^2 the time is reduced to 27.5 years. These numbers are comparable to the ^{90}Sr times to background, indicating that the lower transmutation cross section of ^{137}Cs is compensated for somewhat by its lower initial hazard index.

Cs is found in reprocessed LWR fuel in the form of ^{133}Cs , ^{134}Cs , ^{135}Cs , and ^{137}Cs . Figure 11 displays the hazard index of this FP mixture in a D-T flux at 1 MW/m^2 . As before, the upper curve is the initial hazard index and the lower curve on this figure is the decayed hazard index without ^{138}Cs and ^{139}Ba . The upper curve displays an early maximum before dipping to a pseudo-minimum, then rising for a long period of time before ultimately falling off. This structure can be traced to the buildup and decay of various daughter products. In contrast, the lower curve falls relatively uniformly.

Figure 12 is a graph of the transmutation of FP Cs in a D-T flux of 10 MW/m^2 . As before, the initial and decay curves for the chain are given. In this case, both curves seem reasonably uniform after an initial perturbation.

FIGURE 11. TRANSMUTATION OF CS IN A D-T FLUX AT WALL LOADING OF 1 MW/m²

FIGURE 12. TRANSMUTATION OF Cs IN A D-T FLUX FOR A WALL LOADING OF 10 MW/m^2

Figure 13 displays the transmutation hazard of Cs from reprocessed LWR fuel after the decay of short lived daughters as a function of neutron wall loading and time. From this figure, it would appear that a wall loading in the range of 10 MW/m^2 is needed in order to make a significant impact. Table 16 gives the hazard half-life as a function of neutron wall loading for Cs from reprocessed LWR fuel.

TABLE 16

Hazard Half-Life As A Function of Neutron Wall Loading
For Cs in a D-T Flux

<u>Wall Loading (MW/m^2)</u>	<u>Hazard Half Life (yrs)</u>	<u>Time to Background (yrs)</u>
Decay Only	14.45	150
1	12.29	127
10	3.71	38.5

A significantly shorter hazard half-life is obtained for Cs (14.45 years) natural decay associated with ^{137}Cs (30 years). This is because Cs from reprocessed LWR fuel contains some shorter half-lived Cs isotopes which result in a greater initial hazard index and a more rapidly decaying isotopic mixture. However, the time-to-background for no transmutation does not significantly change in spite of the higher initial hazard index of the mixture.

Comparing the transmutation rate for a wall loading of 1 MW/m^2 versus the decay only situation, there is obviously very little incentive

FIGURE 13. TRANSMUTATION OF Cs IN A D-T FLUX AS A FUNCTION OF WALL LOADING

to consider transmutation of Cs from reprocessed LWR fuel. However, the rate of transmutation associated with a neutron wall loading of 10 MW/m^2 appears to be more attractive.

Figure 14 shows the decayed transmutation hazard for elemental (i.e., reprocessed) Cs and ^{137}Cs for a neutron wall loading of 10 MW/m^2 in a D-T flux. The data presented in Tables 15 and 16, and on Figure 14 leads to concluding that there is limited incentive for isotopic separation of ^{137}Cs prior to transmutation. Irrespective of neutron wall loadings, the change in the time to background caused by isotopic separation is at best 28%.

As with other possible targets, transmutation of ^{137}Cs can potentially be carried out by means of a D-T or a D-D plasma. A calculation (Case 8 of Table 11) was made to examine the relative merits of this type of plasma as a transmutation source for ^{137}Cs .

Figure 15 displays the total decayed transmutation hazard index as a function of time for ^{137}Cs in a D-D flux with a neutron wall loading of 1 MW/m^2 . Notice that the discrepancy between the total and decayed curves is accentuated in this case relative to the D-T case. This is because all transmutation is taking place by n, γ reactions which in turn accelerates buildup of ^{138}Cs .

Figure 16 displays the transmutation hazard as a function of time for ^{137}Cs in a D-D flux with a neutron wall loading of 10 MW/m^2 . Both curves (i.e., total and decay) display some structure at about 40 years exposure time which is not seen in other analyses of Cs transmutation. This is a phenomenon due to the uniquely high flux of what associated with the D-D plasma and the large wall loading.

FIGURE 14. ELEMENTAL Cs AND ¹³⁷Cs TRANSMUTATION IN A D-T FLUX WITH A WALL LOADING OF 10 MW/m²

FIGURE 15. TRANSMUTATION OF ^{137}Cs IN A D-D FLUX WITH A NEUTRON WALL LOADING OF 1 MW/m^2

FIGURE 16. TRANSMUTATION OF ^{137}Cs IN A D-D FLUX WITH A NEUTRON WALL LOADING OF 10 MW/m^2

The decay curves from Figures 15 and 16, along with the natural decay curve for ^{137}Cs are combined in Figure 17 to show the transmutation of ^{137}Cs in a D-D flux as a function of neutron wall loading.

Table 17 gives the hazard half life as a function of neutron wall loading for transmutation of ^{137}Cs in a D-D flux.

TABLE 17

Hazard Half-Life As A Function Of Wall Loading
For ^{137}Cs in a D-D Flux

<u>Wall Loading (MW/m²)</u>	<u>Hazard Half Life (yrs)</u>	<u>Time to Background (yrs)</u>
Decay Only	30.0	147
1	15.8	77.4
10	4.38	21.5

Figure 18 gives the transmutation hazard for ^{137}Cs as a function of time both for a D-D flux and a D-T flux at a neutron wall loading of 1 MW/m². The data on Figure 18 and in Tables 15 and 17 is used to determine the impact of using a D-D flux as a transmutation source for ^{137}Cs . At a neutron wall loading of 1 MW/m², a D-D flux decreases the time-to-background by 15% and at a neutron wall loading of 10 MW/m², the time to background is decreased by 22% with a D-D flux. Thus, the impact of using the D-D plasma relative to the D-T plasma for transmutation of ^{137}Cs is much smaller than the relative impact of using it for transmutation of ^{90}Sr .

FIGURE 17. TRANSMUTATION OF ^{137}Cs IN A D-D FLUX AS A FUNCTION OF NEUTRON WALL LOADING

FIGURE 18. TRANSMUTATION OF ^{137}Cs IN A D-T AND D-D FLUX AT 1 MW/m^2

3. Transmutation of ^{85}Kr

^{85}Kr transmutation was evaluated as a function of neutron wall loading for a D-T plasma neutron source only. The calculational case which applies here is Case 5 of Table 11. The applicable transmutation chain is shown in Appendix I, Figure A-I.4.

Figure 19 shows the reduction of the hazard index as a function of neutron wall loading in a D-T plasma for ^{85}Kr . The initial hazard index for ^{85}Kr is higher than that for ^{137}Cs but lower than ^{90}Sr . However, ^{85}Kr is evaluated as an inhalation hazard whereas ^{137}Cs and ^{90}Sr are ingestion hazards.

It appears from Figure 19 that the curve for a neutron wall loading of 1 MW/m^2 might intersect the natural decay curve. If this were to happen, transmutation at 1 MW/m^2 would have to be stopped before this point in order to be effective. Unfortunately, the present analysis does not extend far enough to explore this possibility. In any case, the values of hazard half life for the 1 MW/m^2 curve have limited usefulness because of this curve structure. The curve for a 10 MW/m^2 neutron wall loading does not exhibit this behavior.

Table 18 gives the hazard-half life for the transmutation of ^{85}Kr in a D-T plasma as a function of neutron wall loading. In developing Table 18, the time to background is not calculated. The ^{85}Kr hazard index value corresponds to an ingestion hazard. Therefore, in this case, the aged uranium ingestion hazard index is not a proper background value and because the inhalation value is much different, the results cannot be

FIGURE 19. TRANSMUTATION OF ⁸⁵Kr IN A D-T FLUX AS A FUNCTION OF NEUTRON WALL LOADING

compared to other isotopes. The data in Table 8, show that transmutation at 1 MW/m^2 lowers the hazard half-life by about 20%, and transmutation at 10 MW/m^2 affects it by about 44% relative to the base case (decay only). Thus, transmutation is not deemed to be a theoretically feasible concept.

TABLE 18

Hazard Half-Life As A Function Of Wall Loading
For ^{85}Kr in a D-T Flux

<u>Wall Loading (MW/m^2)</u>	<u>Hazard Half-Life (yrs)</u>
Decay Only	10.8
1	8.63
10	6.07

As for ^{137}Cs , the hazard half-life for ^{85}Kr , with no transmutation, is identical to its natural half-life because of direct decay to a stable daughter.

4. Transmutation of ^{129}I

^{129}I transmutation was evaluated as a function of neutron wall loading using a D-T plasma neutron source only. The calculational case is number 6 in Table 11. The transmutation chain which was evaluated is shown on Figure A-I.5 of Appendix I.

Figure 20 shows the change in hazard as a function of neutron wall loading in a D-T flux for ^{129}I . Since the natural half-life for ^{129}I is very long the hazard index curve for no-transmutation is essentially a horizontal line. Significant reduction is shown for both a neutron wall loading of 1 MW/m^2 and 10 MW/m^2 . This is because of a large n, γ cross section of ^{129}I . Hazardous daughter buildup has a limited effect in this chain therefore these curves are straight lines. Table 19 gives the hazard half-life for ^{129}I as a function of neutron wall loading in a D-T flux.

TABLE 19

Hazard Half-Life As A Function Of Neutron Wall
Loading For ^{129}I In A D-T Flux

<u>Wall Loading (MW/m^2)</u>	<u>Hazard Half-Life (yrs)</u>
Decay Only	1.6×10^6 yrs
1	0.352
10	0.027

Calculation of the time to background is pointless as ^{129}I is initially below this figure.

The data presented on Figure 20 and in Table 19 shows that transmutation of ^{129}I proceeds at a fast rate because of no daughter buildup. Thus even at a neutron wall loading of 1 MW/m^2 it appears to be an attractive disposal scheme.

Transmutation of ^{129}I in a CTR rapidly reduces the hazard index of this isotope. However, the incentive for transmuting this isotope is marginal since it has such a small initial hazard index value. This isotope would be used as a transmutation target only if no other effective means of disposal were available.

FIGURE 20. TRANSMUTATION OF ^{129}I AS A FUNCTION OF NEUTRON WALL LOADING FOR A D-T FLUX.

B. Actinide Transmutation

The problem of disposing actinide wastes have somewhat different characteristics than that of disposal of the fission products. Typically, the hazard is lower but of much longer duration. Transmutation of actinides occurs through the fission process rather than the process of conversion to stable isotopes by (n,γ) , $(n,2n)$ or similar reactions. Thus, the objective of actinide transmutation is to convert the waste from an actinide composition to a fission product composition. The fission products require much shorter storage time to decay to background radiation levels than do the actinides. Therefore, in reviewing the results of these analyses, it should be noted that the hazard index given, is the hazard associated with the remaining (non-transmuted) actinides. The hazard associated with fission product buildup is not accounted for because it is assumed that an actinide fission replaces a uranium/plutonium fission. Hence, the total fission product waste is not perturbed. In other words, it is assumed that efficient use will be made of the energy generated by this process.

In the actinide analysis reported here, no provision was made for long-term daughter buildup due to α decay. Actinides, when stored for a long period of time (i.e., $10^4 \leftrightarrow 10^6$ yr), exhibit a decrease in hazard index followed by an increase at typically about 10^3 years before ultimately decreasing. Actinides, when transmuted in fission reactors have shown a similar behavior.⁽³⁾ The specific daughters which cause the hazard index buildup are of the radon family. In fission reactor actinide transmutation, this phenomenon did not result in a problem because the effect of transmutation was to shift the curve downward while retaining its shape (Figure 3, page 53 of Reference 3). Since the flux of the CTR is higher than that of a fission reactor, this

should accentuate the effect. However, the long term aging of transmuted wastes was not explored in this analysis, but, as was pointed out with regard to the n,p etc. reactions, should probably be explored in future work.

Actinide transmutation has been evaluated here as a function of neutron wall loading and plasma neutron source. In addition, the affects of irradiating separate targets of ^{237}Np , Am and Cm, and Pu have been explored. Finally, the last case examined was having to transmute all of the plutonium generated in the fission economy, rather than just the 0.5% is not recovered in reprocessing which ends up in the high-level waste stream. This latter case is analogous to a hybrid (fission-fusion) waste burner.

The numerical cases used in this analysis are Case 9 through 14 of Table 11 and the transmutation chain used in this analysis is shown in Appendix I, Figure A-I.8.

The change in hazard index obtained for the transmutation of actinides in a D-T neutron flux as a function of neutron wall loading is shown in Figure 21. The base case is a straight line demonstrating the long-term hazard of actinides. The change in hazard with time is shown to be a strong function of the wall loading. The curves shown in Figure 21 and in all following actinide related figures are decay curves. That is, the curves for wall loadings of 1 and 10 MW/m^2 the hazard is plotted for those isotopes with half-lives longer than 1 year. It is assumed that the process will be managed such that short-lived isotopes will be allowed to decay. As the concepts of this approach are presented in the section where the Cs analysis was discussed, the details are given here, although exactly the

FIGURE 21. TRANSMUTATION OF ACTINIDES IN A D-T FLUX AS A FUNCTION OF NEUTRON WALL LOADING

same technique was used. The selection of 1 year as a half-life cutoff is made arbitrarily.

Figure 21 illustrates the magnitude of the actinide waste problem in that, for no neutron transmutation, the hazard index curve is flat for long time periods at a relatively high hazard index level (i.e., almost as high as ^{137}Cs , which is one of the more bothersome fission products). At a wall loading of 1 MW/m^2 , an initial increase in hazard index is observed but the curve reaches a maximum at about 2 months and then decreases at an appreciable rate. At a neutron wall loading of 10 MW/m^2 the curve is very steep and the hazard reaches that of natural uranium in less than 3 months.

Table 20 gives the hazard half-life for actinide transmutation in a D-T flux as a function of neutron wall loading.

TABLE 20

Hazard Half Life As A Function Of Neutron Wall
Loading for Actinide Transmutation in a D-T Flux

<u>Wall Loading</u> <u>(MW/m²)</u>	<u>Hazard Half-Life</u> <u>(Year)</u>	<u>Time to Background</u> <u>(Year)</u>
Decay Only	458*	462*
1	0.603	2.73
10	0.056	0.254

* ^{241}Am only.

Calculation of a hazard half-life for the actinide chain is very misleading due to the detailed structure of the curve. The hazard is truly long-lived but is made up of an intricate complex of short and long-lived isotopes such that linear representation is difficult. Thus, for illustrative purposes only, the half-life of one of the contributors, ^{241}Am is used and the time for this isotope to approach background is given.

The data in Table 20, shows that CTR transmutation of actinides is very effective. At 1 MW/m^2 , the hazard half-life is less than 1 year, and even more dramatically, at 10 MW/m^2 , the time to background is 3 months.

Figure 22 gives the change in hazard index for transmutation of actinides in a D-D plasma as a function of neutron wall loading. The curves have similar shapes to those given for a D-T plasma (Figure 21) except that the slopes are greater. The 1 MW/m^2 curve has the same early maximum seen on Figure 21 so that the increase in thermal neutron flux associated with a D-D plasma is not sufficient to depress this feature.

Table 21 gives the hazard half-life for transmutation of actinides in a D-D plasma as a function of neutron wall loading.

TABLE 21

Hazard Half Life as a Function of Neutron Wall
Loading for Actinide Transmutation in a D-D flux

Wall Loading (MW/M^2)	Hazard Half Life (yr)	Time to Background (yr)
	458*	462*
1	0.23	1.05
10	0.0367	0.166

* ^{241}Am only.

FIGURE 22. TRANSMUTATION OF ACTINIDES IN A D-D FLUX AS A FUNCTION OF NEUTRON WALL LOADING

The results shown in Figure 22 and Table 21 are truly impressive. The analysis indicates that a D-D CTR can eliminate the actinide hazard in times of the order of less than 1 year for achievable neutron wall loadings, whereas storage of actinides would require hundreds to thousands of years to achieve the same reduction in hazard index.

Figures 23 and 24 illustrate that over the time span examined not only is the hazard index from actinide transmutation continuously decreasing after an early modest maximum over the range examined, but also, the hazard index of the components of the mixture appear to ultimately decrease. Thus, much complex shuffling of waste targets in and out of CTR machines may be avoidable.

The results of the transmutation calculations for a target of only LWR produced Am and Cm isotopes in a D-T fusion CTR with a neutron energy wall loading of 1 MW/m^2 are shown on Figure 23. Although a short term irradiation increases the hazard appreciably, after 2 years irradiation the hazard has fallen below its initial value and is decreasing with a half life of about 4 months. The use of such Am plus Cm targets in a neutron flux of this magnitude or larger would appear to be a candidate for further transmutation studies.

The calculated change in hazard for the transmutation of a target of only ^{237}Np is shown in Figure 24. After the calculations were made an error was discovered in an assignment of the transmutation chain which was used as shown in Figure A-I.8 of Appendix I.

The ^{236}Np which is formed from the $n,2n$ reaction on ^{237}Np was assumed to decay only to ^{236}Pu . Actually, ^{236}Np decays with essentially equal probability to ^{236}U . Successive neutron capture and decay will return to ^{237}Np . The isotopic contents shown for Case 11 in Appendix III indicate that the fraction of ^{236}Np which is formed is so small that the neglect of ^{236}U production results in a

FIGURE 23. TRANSMUTATION OF Am AND Cm IN A D-T FLUX WITH NEUTRON WALL LOADING OF 1 MW/m²

FIGURE 24. TRANSMUTATION OF ²³⁷Np IN A D-T FLUX WITH A NEUTRON WALL LOADING OF 1.0 MW/m²

negligible error in the ^{237}Np content at any time. The production of ^{236}Pu in calculation was also overestimated by a factor of two. This results in no significant error in the calculated hazard index as shown on Figure 24.

The results of the transmutation calculations for a pure ^{237}Np target show that the hazard has been increased from a near negligible level to a significant one by the production of progeny with a wide range of half life values. Clearly ^{237}Np , by itself, is not a candidate for neutron-induced transmutation.

Figures 25 and 26 illustrate different characteristics of actinide transmutation in a CTR. The initial composition of the actinides studied to this point consists of the actinides and 0.5% of the total Pu generated in the LWR. This is due to the fact that a normal fuel reprocessing plant leaves a 0.5% of the Pu in the waste. Figure 25 depicts the time dependent behavior of the actinides and all of the Pu generated by the LWR. Figure 26 is a comparison curve for Pu only.

The interesting point is that both curves, under the conditions of a D-T plasma at a neutron wall loading of 1 MW/m^2 , show ultimately decreasing hazard index values after an early false minimum and a maximum. Comparing Figures 25 and 26, it appears that the actinides tend to fill in somewhat the early minimum and add a bit to the maximum. The fact that both of these curves decrease is somewhat surprising, at least in this time scale, and is, perhaps, quite important.

It has been postulated that the plutonium hazard index would be continuously rising as plutonium is exposed to a neutron flux. This postulate is based upon the buildup of higher plutonium isotopes. This may be true if a continuous source of uranium is imposed. Whether or not it is true for a fission reactor fueled in once a year batch quantities has not been shown. These data show that, for a D-T CTR at a neutron wall loading of 1 MW/m^2 , the hazard

FIGURE 25. TIME DEPENDENT BEHAVIOR OF LWR FUEL LOAD IN A D-T FLUX AT 1 MW/m²

FIGURE 26. TIME DEPENDENT BEHAVIOR OF LWR Pu IN A D-T FLUX AT 1 MW/m²

index decreases after an initial increase of less than an order of magnitude and that within two years, the hazard index is lower than the original value and continuously decreasing.

Thus, contrary to the postulate that the generation and use of plutonium results in a continuously growing waste stockpile, this analysis shows that a D-T CTR can remove the potential Pu hazard at reasonable wall loadings and at reasonable times.

C. The Effect of Uncertainties of Neutron Cross Section on Calculated Transmutation Rates

In all of the calculations of this study the dominant transmutation process is neutron capture or fission for thermal neutrons. For the actinide transmutations these thermal cross sections are generally well known, i.e., a few percent uncertainty, although there are exceptions where the uncertainties may be 25 percent. Clearly the calculated compositions of some of the curium isotopes after a year or so may be uncertain by a factor of two. However, the calculated concentrations of higher actinide nuclei are so small that even an uncertainty of this magnitude is not expected to have a significant impact on the conclusions of this study.

For the fission product nuclei, however, the uncertainties in the thermal neutron capture cross sections dominates the conclusions which can be derived from this study. These uncertainties are primarily in the thermal neutron capture cross section values for ^{90}Sr and ^{137}Cs . A literature search has been made for information on these cross sections using the research citations of CINDA-72 and its supplement. (28,29) The result of this search has been that apparently only a single value with large uncertainty has been obtained for each nuclide.

For ^{90}Sr there is a single activation measurement by Zeisel.⁽³⁰⁾ This experiment yielded a value of 0.8 ± 0.5 b. The result was published in an Australian physics journal and has not been obtained as yet for critical review. The quoted error, however, is so large that the calculated transmutation rates of ^{90}Sr are greatly uncertain. For example, there is approximately a 20 percent probability that the thermal capture cross section is really as small as the assumed ^{137}Cs value of 0.11 b. Thus, the possibility of transmutation of ^{90}Sr may really be no more attractive than that indicated in these studies for ^{137}Cs . On the other hand, there is an almost equal probability that the calculated transmutation rates for ^{90}Sr are too small by a factor of two. In this case the feasibility of transmutation of ^{90}Sr would be greatly enhanced.

The single measurement of the thermal neutron capture cross section of ^{137}Cs was reported by Stupegia.⁽³¹⁾ In this experiment, a value of 0.110 b was reported for the activation of the ^{138g}Cs ground state. The uncertainty on this cross section value is erroneously cited as ± 0.033 b in BNL-325.⁽³²⁾ This error value was, however, only the statistical error assignment of Stupegia. The author assigned a significantly larger total uncertainty of ± 0.050 b to the experimental value. Thus, there is almost a 30 percent probability that the calculated transmutation rates of ^{137}Cs are a factor of two too large. If this were the case the feasibility of the transmutation of ^{137}Cs would be greatly reduced. Based on the assigned error, there is only a few percent probability that the calculated ^{137}Cs transmutation rates could be a factor of two larger. However, it is now known⁽³³⁾ that, in addition to the ^{138g}Cs ground state which beta decays with a half life of 32.2 min., there is also an isomeric state with a half life of 2.9 min., which decays both to the

^{138g}Cs (75%) ground state and by direct beta decay to stable ^{138}Ba (25%).

The isomeric state was not observed in the work of Stupegia. Consequently, his value of the thermal capture cross section is biased low to the extent that the isomer undergoes direct beta decay. The fraction of thermal neutron captures which produce the isomeric state is apparently not known at this time. Thus, the calculated transmutation rates of this study for the fission products must be viewed with significant reservation.

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APPENDIX I

THE TRANSMUTATION CHAINS STUDIED

The transmutation chains used in the analysis are displayed in this Appendix. By the nature of these chains, they have to be arbitrarily terminated at some point. However, every attempt was made to include major daughter products. The $n,2n$ and $n,3n$ reactions were, wherever theoretically possible, allowed for fission products. The $n,2n$ and $n,3n$ reactions were ignored for the actinide case with one exception. Also, fission was allowed for almost all heavy elements in the actinide chain.

FIGURE A-I.1.1 TRANSMUTATION CHAIN FOR Sr^{90} AND Sr IN D-T FLUX

FIGURE A-I.2 TRANSMUTATION CHAIN FOR Cs^{137} IN D-T FLUX

FIGURE A-I.3 TRANSMUTATION CHAIN FOR Cs IN D-T FLUX

FIGURE A-I.4 TRANSMUTATION CHAIN FOR Kr^{85} IN D-T FLUX

FIGURE A-I.5 TRANSMUTATION CHAIN FOR I^{129} IN D-T FLUX

FIGURE A-I.6 TRANSMUTATION CHAIN FOR Sr^{90} IN D-D FLUX

FIGURE A-I.7 TRANSMUTATION CHAIN FOR Cs^{137} IN D-D FLUX

FIGURE A-I.8 TRANSMUTATION CHAIN FOR ACTINIDES

APPENDIX II

THE TRANSMUTATION EQUATIONS

The transmutation equations used in this study which describe the transmutation chains shown in Appendix I are given in this appendix. The symbolism used in these equations is defined below.

N_i : Number density of isotope i (gms/Ton of Fuel Charged)

σ_i^1 : n, γ cross section for isotope i (barns)

σ_i^2 : $n, 2n$ cross section for isotope i (barns)

σ_i^3 : $n, 3n$ cross section for isotope i (barns)

σ_i^f : n , fission cross section for isotope i (barns)

λ_i : radioactive decay constant for decay from the ground state of isotope i (sec^{-1})

λ_i^2 : radioactive decay constant for decay of isomer to the ground state of isotope i (sec^{-1})

F_i^j : Branching between ground state and metastable state

for isotope i (i.e., F_i^1 = ground state, F_i^2 = metastable state)

TRANSMUTATION EQUATIONS FOR Sr-90 and Sr in D-T FLUX

$N_1 = {}^{85}\text{Sr}$	$N_5 = {}^{88}\text{Y}$	$N_9 = {}^{92}\text{Y}$	$N_{13} = {}^{93}\text{Zr}$
$N_2 = {}^{89}\text{Sr}$	$N_6 = {}^{89}\text{Y}$	$N_{10} = \text{Y}^{90}\text{Zr}$	$\sigma^1 = n, \gamma$
$N_3 = {}^{90}\text{Sr}$	$N_7 = {}^{90}\text{Y}$	$N_{11} = {}^{91}\text{Zr}$	$\sigma^2 = n, 2n$
$N_4 = {}^{91}\text{Sr}$	$N_8 = {}^{91}\text{Y}$	$N_{12} = {}^{92}\text{Zr}$	$\sigma^3 = n, 2n$

$$\frac{dN_1}{dt} = N_2 \sigma_2^2 + N_5 \lambda_5 - N_1 \sigma_1^1 \phi - N_1 \sigma_1^2 \phi$$

$$\frac{dN_2}{dt} = N_1 \phi_1^1 + N_3 \sigma_3^2 \phi - N_2 \lambda_2 - N_2 \sigma_2^1 \phi - N_2 \sigma_2^2 \phi + N_4 \sigma_4^3 \phi$$

$$\frac{dN_3}{dt} = N_2 \sigma_2^1 \phi + N_4 \sigma_4^2 \phi - N_3 \lambda_3 - N_3 \sigma_3^2 \phi - N_3 \sigma_3^1 \phi$$

$$\frac{dN_4}{dt} = N_3 \sigma_3^1 \phi - N_4 \lambda_4 - N_4 \sigma_4^2 \phi - N_4 \sigma_4^1 \phi - N_4 \sigma_4^3 \phi$$

$$\frac{dN_5}{dt} = N_6 \sigma_6^2 \phi - N_5 \lambda_5 - N_5 \sigma_5^1 \phi - N_5 \sigma_5^2 \phi$$

$$\frac{dN_6}{dt} = N_5 \sigma_5^1 \phi + N_2 \lambda_2 + N_7 \sigma_7^2 \phi - N_6 \sigma_6^2 \phi - N_6 \sigma_6^1 \phi - N_6 \sigma_6^3 \phi$$

$$\frac{dN_7}{dt} = N_6 \sigma_6^1 \phi + N_8 \sigma_8^2 \phi + N_3 \lambda_3 - N_7 \sigma_7^2 \phi - N_7 \sigma_7^1 \phi - N_7 \sigma_7^3 \phi$$

$$\frac{dN_8}{dt} = N_7 \sigma_7^1 \phi + N_9 \sigma_9^2 \phi + N_4 \lambda_4 - N_8 \lambda_8 - N_8 \sigma_8^1 \phi - N_8 \sigma_8^2 \phi$$

$$\frac{dN_9}{dt} = N_8 \sigma_8^1 \phi - N_9 \sigma_9^2 \phi - N_9 \lambda_9 - N_9 \sigma_9^1 \phi$$

$$\frac{dN_{10}}{dt} = N_7 \lambda_7 + N_{11} \sigma_{11}^2 \phi - N_{10} \sigma_{10}^1 \phi - N_{10} \sigma_{10}^2 \phi$$

Transmutation Equations for Sr-90 and Sr in D-T Flux (Continued)

$$\frac{dN_{11}}{dt} = N_8 \lambda_8 + N_{10} \sigma_{10}^1 \phi + N_{12} \phi_{12}^2 \phi - N_{11} \sigma_{11}^2 \phi - N_{11} \sigma_{11}^1 \phi$$

$$\frac{dN_{12}}{dt} = N_9 \lambda_9 + N_{11} \sigma_{11}^1 \phi + N_{13} \sigma_{13}^2 \phi - N_{12} \sigma_{12}^2 \phi - N_{12} \sigma_{12}^1 \phi$$

$$\frac{dN_{13}}{dt} = N_{12} \sigma_{12}^1 \phi - N_{13} \sigma_{13}^2 \phi - N_{13} \sigma_{13}^1 \phi$$

TRANSMUTATION EQUATIONS FOR Cs-137 in D-T FLUX

$$N_1 = {}^{136}\text{Cs}$$

$$N_2 = {}^{137}\text{Cs}$$

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

$$N_4 = {}^{135}\text{Ba}$$

$$N_5 = {}^{136}\text{Ba}$$

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

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

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

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

$$\frac{dN_1}{dt} = N_2 \sigma_2^2 \phi - N_1 \lambda_1 - N_1 \sigma_1^1 \phi - N_1 \sigma_1^2 \phi + N_3 \sigma_3^3 \phi$$

$$\frac{dN_2}{dt} = N_1 \sigma_1^1 \phi + N_3 \sigma_3^2 \phi - N_2 \sigma_2^2 \phi - N_2 \sigma_2^1 \phi - N_2 \lambda_2$$

$$\frac{dN_3}{dt} = N_2 \sigma_2^1 \phi - N_3 \lambda_3 - N_3 \sigma_3^2 \phi - N_3 \sigma_3^1 \phi - N_3 \sigma_3^3 \phi$$

$$\frac{dN_4}{dt} = N_5 \sigma_5^2 \phi - N_4 \sigma_4^1 \phi - N_4 \phi_4^2 \phi$$

$$\frac{dN_5}{dt} = N_1 \lambda_1 + N_4 \sigma_4^1 \phi + N_6 \sigma_6^2 \phi - N_5 \sigma_5^2 \phi - N_5 \sigma_5^1 \phi$$

$$\frac{dN_6}{dt} = N_2 \lambda_2 + N_5 \sigma_5^1 \phi + N_7 \sigma_7^2 \phi - N_6 \sigma_6^2 \phi - N_6 \sigma_6^1 \phi + N_8 \sigma_8^3 \phi$$

$$\frac{dN_7}{dt} = N_3 \lambda_3 + N_6 \sigma_6^1 \phi + N_8 \sigma_8^2 \phi - N_7 \sigma_7^2 \phi - N_7 \sigma_7^1 \phi$$

$$\frac{dN_8}{dt} = N_7 \sigma_7^1 \phi - N_8 \sigma_8^2 \phi - N_8 \lambda_8 - N_8 \sigma_8^1 \phi - N_8 \sigma_8^3 \phi$$

$$\frac{dN_9}{dt} = N_8 \lambda_8 - N_9 \left(\sigma_9^1 + \sigma_9^2 \right) \phi$$

TRANSMUTATION EQUATIONS FOR Cs in D-T FLUX

$$\begin{array}{lll}
N_1 = {}^{133}\text{Cs} & N_5 = {}^{137}\text{Cs} & N_9 = {}^{136}\text{Ba} & N_{13} = {}^{139}\text{La} \\
N_2 = {}^{134}\text{Cs} & N_6 = {}^{128}\text{Cs} & N_{10} = {}^{137}\text{Ba} \\
N_3 = {}^{135}\text{Cs} & N_7 = {}^{134}\text{Ba} & N_{11} = {}^{138}\text{Ba} \\
N_4 = {}^{136}\text{Cs} & N_8 = {}^{135}\text{Ba} & N_{12} = {}^{139}\text{Ba}
\end{array}$$

$$\frac{dN_1}{dt} = N_2 \sigma_2^2 \phi - N_1 \sigma_1^1 \phi - N_1 \sigma_1^2 \phi$$

$$\frac{dN_2}{dt} = N_1 \sigma_1^1 \phi + N_3 \sigma_3^2 \phi - N_2 \lambda_2 - N_2 \sigma_2^1 \phi - N_2 \sigma_2^2 \phi$$

$$\frac{dN_3}{dt} = N_2 \sigma_2^1 \phi + N_4 \sigma_4^2 \phi - N_3 \sigma_3^1 \phi - N_3 \sigma_3^2 \phi - N_3 \lambda_3$$

$$\frac{dN_4}{dt} = N_3 \sigma_3^1 \phi + N_5 \sigma_5^1 \phi - N_4 \lambda_4 - N_4 \sigma_4^1 \phi - N_4 \sigma_4^2 \phi + N_6 \sigma_6^3 \phi$$

$$\frac{dN_5}{dt} = N_4 \sigma_4^1 \phi + N_6 \sigma_6^2 \phi - N_5 \lambda_5 - N_5 \sigma_5^1 \phi - N_5 \sigma_5^2 \phi$$

$$\frac{dN_6}{dt} = N_5 \sigma_5^1 \phi - N_6 \lambda_6 - N_6 \sigma_6^2 \phi - N_6 \sigma_6^1 \phi - N_6 \sigma_6^3 \phi$$

$$\frac{dN_7}{dt} = N_2 \lambda_2 + N_8 \sigma_8^2 \phi - N_7 \sigma_7^1 \phi - N_7 \sigma_7^2 \phi$$

$$\frac{dN_8}{dt} = N_7 \sigma_7^1 \phi + N_9 \sigma_9^2 \phi - N_8 \sigma_8^1 \phi - N_8 \sigma_8^2 \phi + N_3 \lambda_3$$

$$\frac{dN_9}{dt} = N_4 \lambda_4 + N_8 \sigma_8^1 \phi + N_{10} \sigma_{10}^2 \phi - N_9 \sigma_9^1 \phi - N_9 \sigma_9^2 \phi + N_{11} \sigma_{11}^3 \phi$$

$$\frac{dN_{10}}{dt} = N_5 \lambda_5 + N_9 \sigma_9^1 \phi + N_{11} \sigma_{11}^2 \phi - N_{10} \sigma_{10}^1 \phi - N_{10} \sigma_{10}^2 \phi + N_{12} \sigma_{12}^3 \phi$$

$$\frac{dN_{11}}{dt} = N_6 \lambda_6 + N_{10} \sigma_{10}^1 \phi + N_{12} \sigma_{12}^2 \phi - N_{11} \sigma_{11}^1 \phi - N_{11} \sigma_{11}^2 \phi - N_{11} \sigma_{11}^3 \phi$$

$$\frac{dN_{12}}{dt} = N_{11} \sigma_{11}^1 \phi - N_{12} \sigma_{12}^2 \phi - N_{12} \lambda_{12} - N_{12} \sigma_{12}^1 \phi - N_{12} \sigma_{12}^3 \phi$$

$$\frac{dN_{13}}{dt} = N_{12} \lambda_{12} - N_{13} \left(\sigma_{13}^1 + \sigma_{13}^2 \right) \phi$$

TRANSMUTATION EQUATIONS FOR Kr-85 in D-T FLUX

$$N_1 = {}^{84}\text{Kr}$$

$$N_2 = {}^{85}\text{Kr}$$

$$N_3 = {}^{85}\text{Kr}$$

$$N_4 = {}^{85}\text{Rb}$$

$$\frac{dN_1}{dt} = N_2 \sigma_2^2 \phi - N_1 \sigma_1^1 \phi - N_1 \sigma_1^2 \phi$$

$$\frac{dN_2}{dt} = N_1 \sigma_1^1 \phi + N_3 \sigma_3^2 \phi - N_2 \lambda_2 - N_2 \sigma_2^1 \phi - N_2 \sigma_2^2 \phi$$

$$\frac{dN_3}{dt} = N_2 \sigma_2^1 \phi - N_3 \sigma_3^2 \phi - N_3 \sigma_3^1 \phi$$

$$\frac{dN_4}{dt} = N_2 \lambda_2$$

TRANSMUTATION EQUATIONS FOR I-129 in D-T FLUX

$$N_1 = {}^{128}\text{I}$$

$$N_4 = {}^{128}\text{Xe}$$

$$N_2 = {}^{129}\text{I}$$

$$N_5 = {}^{130}\text{Xe}$$

$$N_3 = {}^{130}\text{I}$$

$$\frac{dN_1}{dt} = N_2 \sigma_2^2 \phi - N_1 \sigma_1^1 \phi - N_1 \lambda_1 - N_1 \sigma_1^2 \phi$$

$$\frac{dN_2}{dt} = N_1 \sigma_1^1 \phi + N_3 \sigma_3^2 \phi - N_2 \sigma_2^1 \phi - N_2 \sigma_2^2 \phi$$

$$\frac{dN_3}{dt} = N_2 \sigma_2^1 \phi - N_3 \sigma_3^2 \phi - N_3 \lambda_3 - N_3 \sigma_3^1 \phi$$

$$\frac{dN_4}{dt} = N_1 \lambda_1$$

$$\frac{dN_5}{dt} = N_3 \lambda_3$$

TRANSMUTATION EQUATIONS FOR Sr-90 in D-D FLUX

$$N_1 = {}^{90}\text{Sr}$$

$$N_5 = {}^{92}\text{Y}$$

$$N_2 = {}^{91}\text{Sr}$$

$$N_6 = {}^{90}\text{Zr}$$

$$N_3 = {}^{90}\text{Y}$$

$$N_7 = {}^{91}\text{Zr}$$

$$N_4 = {}^{91}\text{Y}$$

$$N_8 = {}^{92}\text{Zr}$$

$$\frac{dN_1}{dt} = N_1 \lambda_1 - N_1 \sigma_1^1 \phi$$

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

$$\frac{dN_3}{dt} = N_1 \lambda_1 - N_3 \lambda_3 - N_3 \sigma_3^1 \phi$$

$$\frac{dN_4}{dt} = N_2 \lambda_2 + N_3 \sigma_3^1 \phi - N_4 \lambda_4 - N_4 \sigma_4^1 \phi$$

$$\frac{dN_5}{dt} = N_4 \sigma_4^1 \phi - N_5 \lambda_5 - N_5 \sigma_5^1 \phi$$

$$\frac{dN_6}{dt} = N_3 \lambda_3 - N_6 \sigma_6^1 \phi$$

$$\frac{dN_7}{dt} = N_6 \sigma_6^1 \phi + N_4 \lambda_4 - N_7 \sigma_7^1 \phi$$

$$\frac{dN_8}{dt} = N_5 \lambda_5 + N_7 \sigma_7^1 \phi - N_8 \sigma_8^1 \phi$$

EQUATIONS FOR ACTINIDE CHAIN

$N_1 = {}^{232}\text{U}$	$N_7 = {}^{239}\text{Pu}$	$N_{13} = {}^{242}\text{Am}$	$N_{19} = {}^{244}\text{Cm}$
$N_2 = {}^{237}\text{Np}$	$N_8 = {}^{240}\text{Pu}$	$N_{14} = {}^{242\text{m}}\text{Am}$	$N_{20} = {}^{245}\text{Cm}$
$N_3 = {}^{238}\text{Np}$	$N_9 = {}^{241}\text{Pu}$	$N_{15} = {}^{243}\text{Am}$	$N_{21} = {}^{246}\text{Cm}$
$N_4 = {}^{239}\text{Np}$	$N_{10} = {}^{242}\text{Pu}$	$N_{16} = {}^{244}\text{Am}$	$N_{22} = {}^{247}\text{Cm}$
$N_5 = {}^{236}\text{Pu}$	$N_{11} = {}^{243}\text{Pu}$	$N_{17} = {}^{242}\text{Cm}$	$N_{23} = {}^{248}\text{Cm}$
$N_6 = {}^{238}\text{Pu}$	$N_{12} = {}^{241}\text{Am}$	$N_{18} = {}^{243}\text{Cm}$	$N_{24} = \text{F.P.}$

$$\frac{dN_1}{dt} = N_5\lambda_5 - N_1(\sigma_1^1 + \sigma_1^f)\phi$$

$$\frac{dN_2}{dt} = N_{12}\lambda_{12} - N_2(\sigma_2^1 + \sigma_2^2 + \sigma_2^f)\phi$$

$$\frac{dN_3}{dt} = N_2\sigma_2^1\phi - N_3[(\sigma_3^1 + \sigma_3^f)\phi + \lambda_3]$$

$$\frac{dN_4}{dt} = N_3\sigma_3^1\phi + N_{15}\lambda_{15} - N_4[(\sigma_4^1 + \sigma_4^f)\phi + \lambda_4]$$

$$\frac{dN_5}{dt} = N_2\sigma_2^2\phi - N_5[(\sigma_5^1 + \sigma_5^f)\phi + \lambda_5]$$

$$\frac{dN_6}{dt} = N_3\lambda_3 + N_{17}\lambda_{17} - N_6[(\sigma_6^1 + \sigma_6^f)\phi + \lambda_6]$$

$$\frac{dN_7}{dt} = N_{18}\lambda_{18} + N_4\lambda_4 - N_7[(\sigma_7^1 + \sigma_7^f)\phi + \lambda_7]$$

$$\frac{dN_8}{dt} = N_7\sigma_7^1\phi + N_{19}\lambda_{19} - N_8[(\sigma_8^1 + \sigma_8^f)\phi + \lambda_8]$$

$$\frac{dN_9}{dt} = N_8\sigma_8^1\phi + N_{20}\lambda_{20} - N_9[(\sigma_9^1 + \sigma_9^f)\phi + \lambda_9]$$

$$\frac{dN_{10}}{dt} = N_9\sigma_9^1\phi + N_{13}\lambda_{13} - N_{10}[(\sigma_{10}^1 + \sigma_{10}^f)\phi + \lambda_{10}]$$

$$\frac{dN_{11}}{dt} = N_{10}\sigma_{10}^1\phi - N_{11}[(\sigma_{11}^1 + \sigma_{11}^f)\phi + \lambda_{11}]$$

$$\frac{dN_{12}}{dt} = N_9\lambda_9 - N_{12}[(\sigma_{12}^1 + \sigma_{12}^f)\phi + \lambda_{12}]$$

$$\frac{dN_{13}}{dt} = N_{14}\lambda_{14} + N_{12}^F\sigma_{12}^1\phi - N_{13}[(\sigma_{13}^1 + \sigma_{13}^f)\phi + \lambda_{13} + \lambda_{13}^2]$$

$$\frac{dN_{14}}{dt} = N_{12}\sigma_{12}^1 F_{12}^2 \phi - N_{14}[(\sigma_{14}^1 + \sigma_{14}^f)\phi + \lambda_{14}]$$

$$\frac{dN_{15}}{dt} = N_{13}\sigma_{13}^1 \phi + N_{14}\sigma_{14}^1 \phi + N_{11}\lambda_{11} - N_{15}(\sigma_{15}^1 + \sigma_{15}^f)\phi$$

$$\frac{dN_{16}}{dt} = N_{15}\sigma_{15}^1 \phi - N_{16}[(\sigma_{16}^1 + \sigma_{16}^f)\phi + \lambda_{16}]$$

$$\frac{dN_{17}}{dt} = N_{13}\lambda_{13} - N_{17}[(\sigma_{17}^1 + \sigma_{17}^f)\phi + \lambda_{17}]$$

$$\frac{dN_{18}}{dt} = N_{17}\sigma_{17}^1 \phi - N_{18}[(\sigma_{18}^1 + \sigma_{18}^f)\phi + \lambda_{18}]$$

$$\frac{dN_{19}}{dt} = N_{16}\lambda_{16} + N_{18}\sigma_{18}^1 \phi - N_{19}[(\sigma_{19}^1 + \sigma_{19}^f)\phi + \lambda_{19}]$$

$$\frac{dN_{20}}{dt} = N_{19}\sigma_{19}^1 \phi - N_{20}[(\sigma_{20}^1 + \sigma_{20}^f)\phi + \lambda_{20}]$$

$$\frac{dN_{21}}{dt} = N_{20}\sigma_{20}^1 \phi - N_{21}[(\sigma_{21}^1 + \sigma_{21}^f)\phi + \lambda_{21}]$$

$$\frac{dN_{22}}{dt} = N_{21}\sigma_{21}^1 \phi - N_{22}[(\sigma_{22}^1 + \sigma_{22}^f)\phi]$$

$$\frac{dN_{23}}{dt} = N_{22}\sigma_{22}^1 \phi - N_{23}(\sigma_{23}^1 + \sigma_{23}^f)\phi$$

$$\frac{dN_{24}}{dt} = \sum_{i=1}^{23} N_i \sigma_i^f \phi$$

APPENDIX III

SUMMARY OF ISOTOPIC DATA FROM THE ANALYSIS

III-1

The analysis model calculates the composition of the waste being transmuted as a function of time. This data is compiled in this appendix for the transmutation chains calculated in this analysis. The data is given in the units of nuclei per ton (Nuc/T) of fuel. This unit means that this is the number of nuclei at any given time for a given isotope which has, as its original source, a metric ton of slightly enriched uranium fuel exposed in a LWR for given conditions.

The time scale used in these calculations is given in Table A-III.1. The time scale for Cases 1-8 is given in Column 1 and the time scale for Cases 9-14 is given in Column 2. The difference in time scales is due to the markedly different transmutation rates of fission products on the one hand, and actinides on the other.

In some cases, the calculation was not done for all the time steps. This is due to the fact that the ALCHEMY code as run on the 1108 computer has a lower limit on the size of the exponential it can compute. This effect results in shortened table lengths for some transmutation chains.

TABLE A-III.1

TIME SCALE FOR CALCULATION

Column 1 Fission Product Transmutation Time Scale	Column 2 Actinide Transmutation Time Scale
0	0
1/2 yr	1/2 mo
1 yr	1 mo
2 yr	1 1/2 mo
3 yr	2 mo
5 yr	3 mo
8 yr	6 mo
10 yr	8 mo
15 yr	10 mo
20 yr	1 yr
30 yr	15 mo
40 yr	18 mo
50 yr	21 mo
60 yr	2 yr

III-3

CASE 1. Sr-90 TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA

<u>Sr-88 Nuc/T</u>	<u>Sr-89 Nuc/T</u>	<u>Sr-90 Nuc/T</u>	<u>Sr-91 Nuc/T</u>	<u>Y-88 Nuc/T</u>
0.00	0.00	3.63 + 24	0.00	0.00
4.17 + 18	2.50 + 21	3.47 + 24	3.63 + 20	9.73 + 17
1.25 + 19	2.59 + 21	3.31 + 24	3.46 + 20	3.84 + 18
4.04 + 19	2.38 + 21	3.02 + 24	3.15 + 20	1.06 + 19
8.23 + 19	2.16 + 21	2.75 + 24	2.87 + 20	1.64 + 19
1.99 + 20	1.79 + 21	2.28 + 24	2.38 + 20	2.48 + 19
4.17 + 20	1.36 + 21	1.72 + 24	1.80 + 20	3.09 + 19
5.75 + 20	1.13 + 21	1.43 + 24	1.50 + 20	3.22 + 19
9.59 + 20	7.06 + 20	8.95 + 23	9.37 + 19	3.00 + 19
1.28 + 21	4.43 + 20	5.62 + 23	5.88 + 19	2.46 + 19
1.70 + 21	1.75 + 20	2.21 + 23	2.32 + 19	1.39 + 19
1.90 + 21	6.91 + 19	8.74 + 22	9.15 + 18	6.92 + 18
1.97 + 21	2.70 + 19	3.40 + 22	3.55 + 18	3.19 + 18
1.97 + 21	1.12 + 19	1.39 + 22	1.46 + 18	1.46 + 18
<u>Y-89 Nuc/T</u>	<u>Y-90 Nuc/T</u>	<u>Y-91 Nuc/T</u>	<u>Y-92 Nuc/T</u>	<u>Zr-90 Nuc/T</u>
0.00	0.00	0.00	0.00	0.00
4.29 + 21	8.89 + 20	4.65 + 22	3.09 + 18	4.19 + 22
1.04 + 22	8.56 + 20	4.96 + 22	3.30 + 18	8.32 + 22
2.11 + 22	7.92 + 20	4.58 + 22	3.05 + 18	1.61 + 23
2.97 + 22	7.33 + 20	4.17 + 22	2.77 + 18	2.34 + 23
4.19 + 22	6.26 + 20	3.46 + 22	2.30 + 18	3.66 + 23
5.04 + 22	4.95 + 20	2.62 + 22	1.74 + 18	5.26 + 23
5.19 + 22	4.22 + 20	2.17 + 22	1.44 + 18	6.15 + 23
4.77 + 22	2.81 + 20	1.36 + 22	9.04 + 17	7.81 + 23
3.89 + 22	1.86 + 20	8.54 + 21	5.68 + 17	8.87 + 23
2.17 + 22	8.04 + 19	3.36 + 21	2.24 + 17	9.89 + 23
1.08 + 22	3.42 + 19	1.33 + 21	8.83 + 16	1.01 + 24
4.97 + 21	1.41 + 19	5.16 + 20	3.43 + 16	1.00 + 24
2.27 + 21	6.06 + 18	2.12 + 20	1.41 + 16	9.76 + 23
<u>Zr-91 Nuc/T</u>	<u>Zr-92 Nuc/T</u>	<u>Zr-93 Nuc/T</u>		
0.00	0.00	0.00		
6.61 + 22	3.08 + 21	1.13 + 19		
1.65 + 23	1.16 + 22	8.55 + 19		
3.43 + 23	4.25 + 22	6.16 + 20		
4.88 + 23	8.83 + 22	1.89 + 21		
6.92 + 23	2.12 + 23	7.38 + 21		
8.40 + 23	4.33 + 23	2.32 + 22		
8.71 + 23	5.87 + 23	3.82 + 22		
8.15 + 23	9.35 + 23	8.54 + 22		
6.83 + 23	1.19 + 24	1.35 + 23		
4.19 + 23	1.42 + 24	2.10 + 23		
2.46 + 23	1.40 + 24	2.40 + 23		
1.50 + 23	1.26 + 24	2.37 + 23		
1.02 + 23	1.10 + 24	2.16 + 23		

III-4

CASE 1. Sr-90 TRANSMUTATION FOR WALL LOADING OF 10 MW/m²; D-T PLASMA

<u>Sr-88 Nuc/T</u>	<u>Sr-89 Nuc/T</u>	<u>Sr-90 Nuc/T</u>	<u>Sr-91 Nuc/T</u>	<u>Y-91 Nuc/T</u>
0.00	0.00	3.63 + 24	0.00	0.00
3.50 + 20	1.92 + 22	2.54 + 24	2.66 + 21	3.17 + 23
8.50 + 20	1.47 + 22	1.78 + 24	1.86 + 21	2.43 + 23
1.73 + 21	7.27 + 21	8.70 + 23	9.10 + 20	1.20 + 23
2.29 + 21	3.56 + 21	4.26 + 23	4.46 + 20	5.88 + 22
2.72 + 21	8.46 + 20	1.01 + 23	1.06 + 20	1.39 + 22
2.70 + 21	1.03 + 20	1.20 + 22	1.26 + 19	1.67 + 21
2.58 + 21	2.67 + 19	2.89 + 21	3.02 + 18	3.99 + 20
2.28 + 21	3.64 + 18	1.76 + 20	1.84 + 17	2.25 + 19
2.01 + 21	2.16 + 18	2.02 + 19	2.11 + 16	2.49 + 18
<u>Y-88 Nuc/T</u>	<u>Y-89 Nuc/T</u>	<u>Y-90 Nuc/T</u>	<u>Y-92 Nuc/T</u>	<u>Zr-80 Nuc/T</u>
0.00	0.00	0.00	0.00	0.00
6.20 + 19	3.04 + 22	1.09 + 21	2.11 + 20	4.97 + 22
1.58 + 20	5.08 + 22	1.10 + 21	1.62 + 20	1.12 + 23
1.91 + 20	4.86 + 22	7.99 + 20	7.98 + 19	2.25 + 23
1.35 + 20	3.21 + 22	4.83 + 20	3.91 + 19	2.97 + 23
4.41 + 19	1.01 + 22	1.42 + 20	9.27 + 18	3.50 + 23
6.24 + 18	1.40 + 21	1.91 + 19	1.11 + 18	3.39 + 23
1.60 + 18	1.58 + 20	4.82 + 18	2.65 + 17	3.19 + 23
9.87 + 16	2.43 + 19	3.19 + 17	1.50 + 16	2.66 + 23
4.05 + 16	1.07 + 19	1.23 + 17	1.64 + 15	2.21 + 23
<u>Zr-91 Nuc/T</u>	<u>Zr-92 Nuc/T</u>	<u>Zr-93 Nuc/T</u>		
0.00	0.00	0.00		
4.34 + 23	2.27 + 23	7.84 + 21		
7.30 + 23	6.42 + 23	4.14 + 22		
7.16 + 23	1.36 + 24	1.51 + 23		
5.01 + 23	1.69 + 24	2.45 + 23		
2.02 + 23	1.59 + 24	2.93 + 23		
6.61 + 22	1.01 + 24	2.07 + 23		
4.02 + 22	6.99 + 23	1.46 + 23		
1.80 + 22	2.76 + 23	5.78 + 22		
1.05 + 22	1.19 + 23	2.44 + 22		

CASE 2. Cs-137 TRANSMUTATION FOR WALL LOADING OF 10 MW/m^2 ; D-T PLASMA

<u>Cs-133 Nuc/T</u>	<u>Cs-134 Nuc/T</u>	<u>Cs-135 Nuc/T</u>	<u>Cs-136 Nuc/T</u>	<u>Cs-137 Nuc/T</u>
0.00	0.00	0.00	0.00	5.40 + 24
1.50 + 13	6.21 + 15	1.59 + 19	2.28 + 21	5.06 + 24
1.11 + 13	5.94 + 15	1.52 + 19	2.13 + 21	4.73 + 24
9.70 + 12	5.20 + 15	1.33 + 19	1.87 + 21	4.14 + 24
8.49 + 12	4.55 + 15	1.16 + 19	1.63 + 21	3.63 + 24
6.49 + 12	3.48 + 15	8.91 + 18	1.25 + 21	2.77 + 24
4.36 + 12	2.34 + 15	5.98 + 18	8.40 + 20	1.86 + 24
3.34 + 12	1.79 + 15	4.59 + 18	6.44 + 20	1.43 + 24
1.72 + 12	9.20 + 14	2.35 + 18	3.30 + 20	7.32 + 23
8.83 + 11	4.74 + 14	1.21 + 18	1.70 + 20	3.77 + 23
2.39 + 11	1.28 + 14	3.27 + 17	4.60 + 19	1.02 + 23
6.16 + 10	3.30 + 13	8.45 + 16	1.19 + 19	2.63 + 22
2.18 + 10	1.17 + 13	3.00 + 16	4.22 + 18	9.37 + 21
1.01 + 10	5.45 + 12	1.40 + 16	1.98 + 18	4.39 + 21
<u>Cs-138 Nuc/T</u>	<u>Ba-134 Nuc/T</u>	<u>Ba-135 Nuc/T</u>	<u>Ba-136 Nuc/T</u>	<u>Ba-137 Nuc/T</u>
0.00	0.00	0.00	0.00	0.00
4.53 + 19	8.50 + 17	9.92 + 19	2.00 + 22	2.80 + 22
4.24 + 19	3.24 + 18	2.11 + 20	3.44 + 22	3.40 + 22
3.71 + 19	7.92 + 18	3.44 + 20	5.06 + 22	3.80 + 22
3.25 + 19	1.05 + 19	3.94 + 20	5.63 + 22	3.87 + 22
2.49 + 19	1.12 + 19	3.82 + 20	5.34 + 22	3.54 + 22
1.67 + 19	8.76 + 18	2.88 + 20	3.99 + 22	2.71 + 22
1.28 + 19	6.95 + 18	2.27 + 20	3.14 + 22	2.18 + 22
6.57 + 18	3.67 + 18	1.19 + 20	1.65 + 22	1.18 + 22
3.38 + 18	1.90 + 18	6.17 + 19	8.52 + 21	6.22 + 21
9.14 + 17	5.11 + 17	1.66 + 19	2.29 + 21	1.69 + 21
2.36 + 17	1.33 + 17	4.32 + 18	5.96 + 20	4.39 + 20
8.40 + 16	4.31 + 16	1.42 + 18	1.97 + 20	1.44 + 20
3.94 + 16	1.80 + 16	6.04 + 17	8.44 + 19	6.15 + 19
<u>Ba-138 Nuc/T</u>	<u>Ba-139 Nuc/T</u>	<u>La-139 Nuc/T</u>		
0.00	0.00	0.00		
2.77 + 23	2.09 + 19	8.89 + 21		
5.15 + 23	3.89 + 19	1.95 + 22		
8.53 + 23	6.45 + 19	3.50 + 22		
1.05 + 24	7.90 + 19	4.39 + 22		
1.15 + 24	8.72 + 19	4.94 + 22		
1.00 + 24	7.57 + 19	4.33 + 22		
8.40 + 23	6.35 + 19	3.64 + 22		
4.80 + 23	3.63 + 19	2.09 + 22		
2.57 + 23	1.94 + 19	1.12 + 22		
7.01 + 22	5.30 + 18	3.05 + 21		
1.83 + 22	1.38 + 18	7.98 + 20		
5.77 + 21	4.36 + 17	2.50 + 20		
2.30 + 21	1.74 + 17	9.96 + 19		

CASE 2. Cs-137 TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA

<u>Cs-133 Nuc/T</u>	<u>Cs-134 Nuc/T</u>	<u>Cs-135 Nuc/T</u>	<u>Cs-136 Nuc/T</u>	<u>Cs-137 Nuc/T</u>
0.00	0.00	0.00	0.00	5.41 + 24
5.27 + 11	6.62 + 14	2.07 + 18	9.32 + 20	5.31 + 24
1.63 + 12	1.26 + 15	3.53 + 18	9.15 + 20	5.21 + 24
3.17 + 12	1.89 + 15	5.06 + 18	8.82 + 20	5.03 + 24
3.82 + 12	2.13 + 15	5.61 + 18	8.51 + 20	4.84 + 24
4.03 + 12	2.16 + 15	5.68 + 18	7.90 + 20	4.50 + 24
3.72 + 12	1.98 + 15	5.19 + 18	7.08 + 20	4.03 + 24
3.47 + 12	1.84 + 15	4.83 + 18	6.58 + 20	3.75 + 24
2.89 + 12	1.54 + 15	4.02 + 18	5.48 + 20	3.12 + 24
2.40 + 12	1.28 + 15	3.35 + 18	4.57 + 20	2.60 + 24
1.67 + 12	8.88 + 14	2.33 + 18	3.17 + 20	1.80 + 24
1.16 + 12	6.16 + 14	1.61 + 18	2.20 + 20	1.25 + 24
8.02 + 11	4.27 + 14	1.12 + 18	1.52 + 20	8.68 + 23
5.60 + 11	2.98 + 14	7.80 + 17	1.06 + 20	6.06 + 23
<u>Cs-138 Nuc/T</u>	<u>Ba-134 Nuc/T</u>	<u>Ba-135 Nuc/T</u>	<u>Ba-136 Nuc/T</u>	<u>Ba-137 Nuc/T</u>
0.00	0.00	0.00	0.00	0.00
4.76 + 18	6.65 + 15	7.79 + 18	8.46 + 21	5.58 + 22
4.67 + 18	5.28 + 16	3.00 + 19	1.73 + 22	1.00 + 23
4.51 + 18	3.63 + 17	1.02 + 20	3.40 + 22	1.63 + 23
4.34 + 18	1.03 + 18	1.92 + 20	4.94 + 22	2.01 + 23
4.04 + 18	3.38 + 18	3.86 + 20	7.66 + 22	2.37 + 23
3.62 + 18	8.44 + 18	6.40 + 20	1.08 + 23	2.45 + 23
3.36 + 18	1.22 + 19	7.76 + 20	1.24 + 23	2.39 + 23
2.80 + 18	2.09 + 19	1.01 + 21	1.50 + 23	2.16 + 23
2.33 + 18	2.69 + 19	1.11 + 21	1.60 + 23	1.92 + 23
1.62 + 18	3.14 + 19	1.11 + 21	1.54 + 23	1.49 + 23
1.12 + 18	2.96 + 19	9.66 + 20	1.32 + 23	1.14 + 23
7.78 + 17	2.50 + 19	7.82 + 20	1.06 + 23	8.59 + 22
5.43 + 17	2.01 + 19	6.14 + 20	8.28 + 22	6.49 + 22
<u>Ba-138 Nuc/T</u>	<u>Ba-139 Nuc/T</u>	<u>La-139 Nuc/T</u>		
0.00	0.00	0.00		
3.30 + 22	2.49 + 17	2.29 + 20		
7.53 + 22	5.70 + 17	9.09 + 20		
1.79 + 23	1.35 + 18	3.47 + 21		
2.98 + 23	2.25 + 18	7.26 + 21		
5.51 + 23	4.17 + 18	1.70 + 22		
9.05 + 23	6.84 + 18	3.23 + 22		
1.11 + 24	8.41 + 18	4.18 + 22		
1.52 + 24	1.15 + 19	6.10 + 22		
1.78 + 24	1.34 + 19	7.36 + 22		
1.99 + 24	1.50 + 19	8.47 + 22		
1.94 + 24	1.47 + 19	8.37 + 22		
1.76 + 24	1.33 + 19	7.33 + 22		
1.53 + 24	1.16 + 19	6.68 + 22		

CASE 3. Sr TRANSMUTATION FOR WALL LOADING OF 10 MW/m^2 ; D-T PLASMA

<u>Sr-88 Nuc/T</u>	<u>Sr-89 Nuc/T</u>	<u>Sr-90 Nuc/T</u>	<u>Sr-91 Nuc/T</u>	<u>Y-88 Nuc/T</u>
2.40 + 24	0.00	3.63 + 24	0.00	0.00
2.37 + 24	2.14 + 22	2.54 + 24	2.66 + 21	6.83 + 19
2.34 + 24	1.70 + 22	1.78 + 24	1.86 + 21	1.76 + 20
2.28 + 24	9.55 + 21	8.71 + 23	9.11 + 20	2.24 + 20
2.22 + 24	5.78 + 21	4.27 + 23	4.47 + 20	1.72 + 20
2.11 + 24	2.96 + 21	1.02 + 23	1.07 + 20	8.25 + 19
1.96 + 24	2.06 + 21	1.30 + 22	1.36 + 19	4.22 + 19
1.86 + 24	1.89 + 21	3.81 + 21	3.98 + 18	3.58 + 19
1.64 + 24	1.65 + 21	9.85 + 20	1.03 + 18	3.02 + 19
1.45 + 24	1.45 + 21	7.34 + 20	7.66 + 17	2.66 + 19
1.12 + 24	1.13 + 21	5.35 + 20	5.59 + 17	2.07 + 19
7.72 + 23	8.74 + 20	4.48 + 20	4.68 + 17	1.60 + 19
6.74 + 23	6.76 + 20	3.61 + 20	3.77 + 17	1.24 + 19
5.26 + 23	5.27 + 20	2.40 + 20	2.51 + 17	9.67 + 18
<u>Y-89 Nuc/T</u>	<u>Y-90 Nuc/T</u>	<u>Y-91 Nuc/T</u>	<u>Y-92 Nuc/T</u>	<u>Zr-90 Nuc/T</u>
0.00	0.00	0.00	0.00	0.00
3.36 + 22	1.13 + 21	3.17 + 23	2.11 + 20	5.04 + 22
5.72 + 22	1.17 + 21	2.43 + 23	1.62 + 20	1.15 + 23
5.79 + 22	9.01 + 20	1.20 + 23	7.99 + 19	2.36 + 23
4.23 + 22	5.95 + 20	5.89 + 22	3.92 + 19	3.18 + 23
2.03 + 22	2.53 + 20	1.40 + 22	9.36 + 18	3.90 + 23
1.10 + 22	1.24 + 20	1.80 + 21	1.20 + 18	4.03 + 23
9.44 + 21	1.04 + 20	5.26 + 20	3.50 + 17	3.97 + 23
8.03 + 21	8.79 + 19	1.34 + 20	8.95 + 16	3.71 + 23
7.06 + 21	7.73 + 19	1.01 + 20	6.73 + 16	3.43 + 23
5.48 + 21	6.00 + 19	7.44 + 19	4.95 + 16	2.87 + 23
4.26 + 21	4.66 + 19	6.17 + 19	4.10 + 16	2.37 + 23
3.29 + 21	3.61 + 19	4.95 + 19	3.29 + 16	1.93 + 23
2.57 + 21	2.81 + 19	3.36 + 19	2.23 + 16	1.56 + 23
<u>Zr-91 Nuc/T</u>	<u>Zr-92 Nuc/T</u>	<u>Zr-93 Nuc/T</u>		
0.00	0.00	0.00		
4.34 + 23	2.27 + 23	7.84 + 21		
7.30 + 23	6.42 + 23	4.15 + 22		
7.17 + 23	1.36 + 24	1.51 + 23		
5.02 + 23	1.69 + 24	2.45 + 23		
2.03 + 23	1.59 + 24	2.93 + 23		
6.83 + 22	1.01 + 24	2.08 + 23		
4.28 + 22	7.07 + 23	1.47 + 23		
2.15 + 22	2.88 + 23	5.98 + 22		
1.45 + 22	1.34 + 23	2.70 + 22		
9.75 + 21	5.25 + 22	9.85 + 21		
7.77 + 21	3.61 + 22	6.58 + 21		
6.29 + 21	2.86 + 22	5.18 + 21		
5.08 + 21	2.30 + 22	4.18 + 21		

CASE 3. Sr TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA

<u>Sr-88 Nuc/T</u>	<u>Sr-89 Nuc/T</u>	<u>Sr-90 Nuc/T</u>	<u>Sr-91 Nuc/T</u>	<u>Y-88 Nuc/T</u>
2.40 + 24	0.00	3.63 + 24	0.00	0.00
2.39 + 24	2.74 + 21	3.47 + 24	3.63 + 20	1.06 + 18
2.39 + 24	2.84 + 21	3.31 + 24	3.46 + 20	4.20 + 18
2.38 + 24	2.63 + 21	3.02 + 24	3.15 + 20	1.16 + 19
2.38 + 24	2.42 + 21	2.75 + 24	2.87 + 20	1.81 + 19
2.37 + 24	2.04 + 21	2.28 + 24	2.38 + 20	2.76 + 19
2.35 + 24	1.61 + 21	1.72 + 24	1.80 + 20	3.49 + 19
2.34 + 24	1.37 + 21	1.43 + 24	1.50 + 20	3.68 + 19
2.31 + 24	9.51 + 20	8.95 + 23	9.37 + 19	3.57 + 19
2.28 + 24	6.85 + 20	5.62 + 23	5.88 + 19	3.10 + 19
2.22 + 24	4.11 + 20	2.22 + 23	2.32 + 19	2.06 + 19
2.17 + 24	2.99 + 20	8.75 + 22	9.16 + 18	1.37 + 19
2.11 + 24	2.51 + 20	3.40 + 22	3.56 + 18	9.91 + 18
2.06 + 24	2.30 + 20	1.40 + 22	1.47 + 18	8.04 + 18
<u>Y-89 Nuc/T</u>	<u>Y-90 Nuc/T</u>	<u>Y-91 Nuc/T</u>	<u>Y-92 Nuc/T</u>	<u>Zr-90 Nuc/T</u>
0.00	0.00	0.00	0.00	0.00
4.68 + 21	8.89 + 20	4.65 + 22	3.09 + 18	4.19 + 22
1.13 + 22	8.57 + 20	4.96 + 22	3.30 + 18	8.32 + 22
2.32 + 22	7.94 + 20	4.58 + 22	3.05 + 18	1.62 + 23
3.28 + 22	7.37 + 20	4.17 + 22	2.77 + 18	2.35 + 23
4.66 + 22	6.32 + 20	3.46 + 22	2.30 + 18	3.67 + 23
5.71 + 22	5.02 + 20	2.62 + 22	1.74 + 18	5.29 + 23
5.96 + 22	4.30 + 20	2.17 + 22	1.44 + 18	6.19 + 23
5.70 + 22	2.91 + 20	1.36 + 22	9.04 + 17	7.90 + 23
4.91 + 22	1.98 + 20	8.54 + 21	5.68 + 17	9.01 + 23
3.26 + 22	9.25 + 19	3.37 + 21	2.24 + 17	1.01 + 24
2.17 + 22	4.63 + 19	1.33 + 21	8.84 + 16	1.05 + 24
1.58 + 22	2.61 + 19	5.17 + 20	3.44 + 16	1.05 + 24
1.28 + 22	1.77 + 19	2.13 + 20	1.42 + 16	1.03 + 24
<u>Zr-91 Nuc/T</u>	<u>Zr-92 Nuc/T</u>	<u>Zr-93 Nuc/T</u>		
0.00	0.00	0.00		
6.61 + 22	3.08 + 21	1.13 + 19		
1.65 + 23	1.16 + 22	8.56 + 19		
3.43 + 23	4.25 + 22	6.16 + 20		
4.88 + 23	8.83 + 22	1.89 + 21		
6.92 + 23	2.12 + 23	7.38 + 21		
8.40 + 23	4.33 + 23	2.32 + 22		
8.71 + 23	5.87 + 23	3.82 + 22		
8.15 + 23	9.35 + 23	8.54 + 22		
6.83 + 23	1.19 + 24	1.35 + 23		
4.19 + 23	1.42 + 24	2.10 + 23		
2.47 + 23	1.40 + 24	2.40 + 23		
1.52 + 23	1.26 + 24	2.37 + 23		
1.04 + 23	1.10 + 24	2.17 + 23		

CASE 4. Cs TRANSMUTATION FOR WALL LOADING OF 10 MW/m^2 ; D-T PLASMA

<u>Cs-133 Nuc/T</u>	<u>Cs-134 Nuc/T</u>	<u>Cs-135 Nuc/T</u>	<u>Cs-136 Nuc/T</u>	<u>Cs-137 Nuc/T</u>
4.57 + 24	7.41 + 23	1.45 + 24	0.00	5.40 + 24
1.12 + 19	6.80 + 19	1.57 + 23	1.69 + 22	1.02 + 25
9.19 + 18	2.86 + 18	2.79 + 21	4.59 + 21	9.71 + 24
9.19 + 18	1.70 + 18	6.09 + 19	3.83 + 21	8.50 + 24
6.88 + 18	1.61 + 18	4.85 + 19	3.36 + 21	7.44 + 24
2.00 + 19	5.81 + 18	1.25 + 20	2.57 + 21	5.69 + 24
9.19 + 18	1.70 + 18	4.51 + 19	1.73 + 21	3.82 + 24
<u>Cs-138 Nuc/T</u>	<u>Ba-134 Nuc/T</u>	<u>Ba-135 Nuc/T</u>	<u>Ba-136 Nuc/T</u>	<u>Ba-137 Nuc/T</u>
0.00	0.00	0.00	0.00	0.00
9.18 + 19	6.52 + 21	8.51 + 21	9.78 + 23	1.50 + 23
8.70 + 19	2.68 + 21	7.14 + 21	8.08 + 23	1.75 + 23
7.62 + 19	5.31 + 20	4.21 + 21	5.32 + 23	1.47 + 23
6.67 + 19	1.59 + 20	2.77 + 21	3.63 + 23	1.22 + 23
5.10 + 19	5.13 + 19	1.43 + 21	1.92 + 23	8.95 + 22
3.43 + 19	2.34 + 19	7.19 + 20	9.83 + 22	6.04 + 22
<u>Ba-138 Nuc/T</u>	<u>Ba-139 Nuc/T</u>	<u>La-139 Nuc/T</u>		
0.00	0.00	0.00		
5.69 + 23	4.30 + 19	1.70 + 22		
1.26 + 24	9.52 + 19	4.67 + 22		
2.19 + 24	1.66 + 20	8.99 + 22		
2.64 + 24	1.99 + 20	1.11 + 23		
2.75 + 24	2.08 + 20	1.18 + 23		
2.24 + 24	1.69 + 20	9.71 + 22		

CASE 4. Cs TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA

<u>Cs-133 Nuc/T</u>	<u>Cs-134 Nuc/T</u>	<u>Cs-135 Nuc/T</u>	<u>Cs-136 Nuc/T</u>	<u>Cs-137 Nuc/T</u>
4.57 + 24	7.41 + 23	1.45 + 24	0.00	5.41 + 24
1.05 + 24	3.46 + 23	4.00 + 24	1.13 + 23	5.67 + 24
2.39 + 23	8.06 + 22	3.48 + 24	1.01 + 23	6.07 + 24
1.25 + 22	4.82 + 21	1.71 + 24	5.10 + 22	6.52 + 24
7.08 + 20	5.47 + 20	7.71 + 23	2.36 + 22	6.59 + 24
5.48 + 19	7.82 + 19	1.51 + 23	5.51 + 21	6.32 + 24
		1.31 + 22	1.38 + 21	5.71 + 24
		2.72 + 21	1.01 + 21	5.31 + 24
		3.10 + 20	7.85 + 20	4.42 + 24
		6.67 + 19	6.48 + 20	3.68 + 24
		6.11 + 18	4.49 + 20	2.55 + 24
		6.35 + 19	3.13 + 20	1.77 + 24
			2.12 + 20	1.22 + 24
			1.50 + 20	8.55 + 23
<u>Cs-138 Nuc/T</u>	<u>Ba-134 Nuc/T</u>	<u>Ba-135 Nuc/T</u>	<u>Ba-136 Nuc/T</u>	<u>Ba-137 Nuc/T</u>
0.00	0.00	0.00	0.00	0.00
5.09 + 18	1.04 + 23	5.67 + 21	7.76 + 23	6.56 + 22
5.44 + 18	1.23 + 23	1.59 + 22	1.82 + 24	1.48 + 23
5.84 + 18	1.09 + 23	3.32 + 22	3.13 + 24	3.35 + 23
5.91 + 18	9.13 + 22	4.27 + 22	3.64 + 24	5.01 + 23
5.67 + 18	6.29 + 22	4.67 + 22	3.72 + 24	7.01 + 23
5.12 + 18	3.64 + 22	3.93 + 22	3.33 + 24	7.70 + 23
4.76 + 18	2.53 + 22	3.34 + 22	3.05 + 24	7.50 + 23
3.96 + 18	1.03 + 22	2.27 + 22	2.41 + 24	6.47 + 23
3.30 + 18	4.37 + 21	1.63 + 22	1.90 + 24	5.42 + 23
2.29 + 18	9.47 + 20	9.41 + 21	1.19 + 24	3.79 + 23
1.59 + 18	3.33 + 20	5.81 + 21	7.51 + 23	2.67 + 23
1.10 + 18	1.45 + 20	3.62 + 21	4.71 + 23	1.88 + 23
7.67 + 17	9.75 + 19	2.34 + 21	3.06 + 23	1.35 + 23
<u>Ba-138 Nuc/T</u>	<u>Ba-139 Nuc/T</u>	<u>La-139 Nuc/T</u>		
0.00	0.00	0.00		
3.43 + 22	2.59 + 17	2.34 + 20		
8.53 + 22	6.45 + 17	9.83 + 20		
2.46 + 23	1.86 + 18	4.36 + 21		
4.77 + 23	3.61 + 18	1.06 + 22		
1.08 + 24	8.14 + 18	3.07 + 22		
2.03 + 24	1.54 + 19	6.97 + 22		
2.61 + 24	1.97 + 19	9.57 + 22		
3.74 + 22	2.83 + 19	1.49 + 23		
4.41 + 24	3.34 + 19	1.82 + 23		
4.86 + 24	3.68 + 19	2.08 + 23		
4.61 + 24	3.49 + 19	2.00 + 23		
4.06 + 24	3.07 + 19	1.77 + 23		
3.43 + 24	2.59 + 19	1.50 + 23		

CASE 5. Kr-85 TRANSMUTATION FOR WALL LOADING OF 10 MW/m^2 ; D-T PLASMA

<u>Kr-84 Nuc/T</u>	<u>Kr-85 Nuc/T</u>	<u>Kr-86 Nuc/T</u>	<u>Rb-85 Nuc/T</u>
0.00	2.01 + 23	0.00	0.00
2.77 + 21	9.61 + 22	9.58 + 22	4.56 + 21
3.81 + 21	4.72 + 22	1.38 + 23	6.76 + 21
3.98 + 21	1.36 + 22	1.62 + 23	8.46 + 21
3.50 + 21	6.02 + 21	1.62 + 23	9.03 + 21
2.50 + 21	3.61 + 21	1.49 + 23	9.59 + 21
1.55 + 21	2.95 + 21	1.29 + 23	1.02 + 22
1.17 + 21	2.64 + 21	1.17 + 23	1.06 + 22
6.52 + 20	2.02 + 21	9.14 + 22	1.13 + 22
4.25 + 20	1.56 + 21	7.14 + 22	1.19 + 22
2.27 + 20	9.45 + 20	4.33 + 22	1.27 + 22
1.34 + 20	5.73 + 20	2.63 + 22	1.32 + 22
8.06 + 19	3.45 + 20	1.58 + 22	1.35 + 22
4.93 + 19	2.11 + 20	9.69 + 21	1.36 + 22

Kr-85 TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA

<u>Kr-84 Nuc/T</u>	<u>Kr-85 Nuc/T</u>	<u>Kr-86 Nuc/T</u>	<u>Rb-85 Nuc/T</u>
0.00	2.01 + 23	0.00	0.00
3.94 + 20	1.81 + 23	1.31 + 22	6.13 + 21
7.44 + 20	1.63 + 23	2.49 + 22	1.17 + 22
1.33 + 21	1.33 + 23	4.50 + 22	2.11 + 22
1.80 + 21	1.08 + 23	6.12 + 22	2.89 + 22
2.44 + 21	7.18 + 22	8.47 + 22	4.04 + 22
2.93 + 21	3.95 + 22	1.05 + 23	5.07 + 22
3.06 + 21	2.68 + 22	1.12 + 23	5.49 + 22
3.07 + 21	1.08 + 22	1.19 + 23	6.05 + 22
2.89 + 21	5.17 + 21	1.20 + 23	6.30 + 22
2.42 + 21	2.37 + 21	1.15 + 23	6.51 + 22
2.00 + 21	1.90 + 21	1.09 + 23	6.64 + 22
1.66 + 21	1.73 + 21	1.03 + 23	6.76 + 22
1.38 + 21	1.62 + 21	9.72 + 22	6.87 + 22

III-12

CASE 6. I-129 TRANSMUTATION FOR WALL LOADING OF 10 MW/m^2 ; D-T PLASMA

<u>I-128 Nuc/T</u>	<u>I-129 Nuc/T</u>	<u>I-130 Nuc/T</u>	<u>Xe-128 Nuc/T</u>	<u>Xe-130 Nuc/T</u>
0.00	1.07 + 24	0.00	0.00	0.00
5.55 + 13	1.75 + 19	8.07 + 17	2.18 + 21	1.07 + 24
1.39 + 13	4.40 + 18	1.95 + 17	2.18 + 21	1.07 + 24
1.39 + 13	4.40 + 18	1.95 + 17	2.18 + 21	1.07 + 24
2.96 + 12	9.37 + 17	4.15 + 16	2.18 + 21	1.07 + 24
1.06 + 13	3.37 + 18	1.51 + 17	2.18 + 21	1.07 + 24
1.39 + 13	4.40 + 18	1.95 + 17	2.18 + 21	1.07 + 24

I-129 TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA

<u>I-128 Nuc/T</u>	<u>I-129 Nuc/T</u>	<u>I-130 Nuc/T</u>	<u>Xe-128 Nuc/T</u>	<u>Xe-130 Nuc/T</u>
0.00	1.07 + 24	0.00	0.00	0.00
1.09 + 17	3.45 + 23	1.55 + 21	1.48 + 21	7.21 + 23
3.53 + 16	1.11 + 23	4.99 + 20	1.95 + 21	9.55 + 23
3.67 + 15	1.16 + 22	5.20 + 19	2.16 + 21	1.06 + 24
3.89 + 14	1.23 + 21	5.50 + 18	2.18 + 21	1.07 + 24
1.76 + 13	5.54 + 19	2.47 + 17	2.18 + 21	1.07 + 24

CASE 7. Sr-90 TRANSMUTATION FOR WALL LOADING OF 10 MW/m^2 ; D-D PLASMA

<u>Sr-90 Nuc/T</u>	<u>Sr-91 Nuc/T</u>	<u>Y-90 Nuc/T</u>	<u>Y-91 Nuc/T</u>	<u>Y-92 Nuc/T</u>
3.63 + 24	0.00	0.00	0.00	0.00
1.84 + 24	3.94 + 21	4.55 + 20	4.34 + 23	5.92 + 20
9.27 + 23	1.99 + 21	2.30 + 20	2.35 + 23	3.21 + 20
2.37 + 23	5.07 + 20	5.87 + 19	6.03 + 22	8.23 + 19
6.04 + 22	1.29 + 20	1.50 + 19	1.54 + 22	2.10 + 19
3.88 + 21	8.33 + 18	9.64 + 17	9.88 + 20	1.35 + 18
1.85 + 19	3.99 + 16	4.76 + 15	7.20 + 18	9.84 + 15
1.08 + 19	2.32 + 16	2.67 + 15	2.41 + 18	3.30 + 15
<u>Zr-90 Nuc/T</u>	<u>Zr-91 Nuc/T</u>	<u>Zr-92 Nuc/T</u>		
0.00	0.00	0.00		
2.95 + 22	5.64 + 23	6.97 + 23		
4.41 + 22	6.16 + 23	1.48 + 24		
5.31 + 22	2.63 + 23	1.88 + 24		
5.31 + 22	8.13 + 22	1.50 + 24		
4.82 + 22	7.06 + 21	6.85 + 23		
4.08 + 22	1.22 + 21	1.85 + 23		
3.64 + 22	1.03 + 21	7.76 + 22		

Sr-90 TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-D PLASMA

<u>Sr-90 Nuc/T</u>	<u>Sr-91 Nuc/T</u>	<u>Y-90 Nuc/T</u>	<u>Y-91 Nuc/T</u>	<u>Y-92 Nuc/T</u>
3.63 + 24	0.00	0.00	0.00	0.00
3.36 + 24	7.21 + 20	8.52 + 20	9.16 + 22	1.25 + 19
3.10 + 24	6.66 + 20	7.88 + 20	9.43 + 22	1.29 + 19
2.65 + 24	5.68 + 20	6.72 + 20	8.15 + 22	1.11 + 19
2.26 + 24	4.85 + 20	5.74 + 20	6.96 + 22	9.50 + 18
1.64 + 24	3.53 + 20	4.17 + 20	5.06 + 22	6.90 + 18
1.02 + 24	2.20 + 20	2.60 + 20	3.16 + 22	4.31 + 18
7.47 + 23	1.60 + 20	1.90 + 20	2.30 + 22	3.14 + 18
3.38 + 23	7.25 + 19	8.57 + 19	1.04 + 22	1.42 + 18
1.54 + 23	3.30 + 19	4.00 + 19	4.73 + 21	6.45 + 17
3.17 + 22	6.81 + 18	8.05 + 18	9.77 + 20	1.33 + 17
6.62 + 21	1.42 + 18	1.68 + 18	2.04 + 20	2.78 + 16
1.46 + 21	3.13 + 17	3.71 + 17	4.48 + 19	6.11 + 15
<u>Zr-90 Nuc/T</u>	<u>Zr-91 Nuc/T</u>	<u>Zr-92 Nuc/T</u>		
0.00	0.00	0.00		
4.09 + 22	1.30 + 23	1.25 + 22		
7.95 + 22	3.10 + 23	4.60 + 22		
1.48 + 23	5.92 + 23	1.58 + 23		
2.05 + 23	7.73 + 23	3.09 + 23		
2.96 + 23	9.27 + 23	6.57 + 23		
3.84 + 23	8.78 + 23	1.13 + 24		
4.22 + 23	7.70 + 23	1.37 + 24		
4.71 + 23	4.77 + 23	1.68 + 24		
4.85 + 23	2.67 + 23	1.69 + 24		
4.76 + 23	7.78 + 22	1.32 + 24		
4.54 + 23	2.76 + 22	9.15 + 23		
4.29 + 23	1.53 + 22	6.10 + 23		
		4.13 + 23		

CASE 8. Cs-137 TRANSMUTATION FOR WALL LOADING OF 10 MW/m^2 ; D-D PLASMA

<u>Cs-137 Nuc/T</u>	<u>Cs-138 Nuc/T</u>	<u>Ba-137 Nuc/T</u>	<u>Ba-138 Nuc/T</u>	<u>Ba-139 Nuc/T</u>
5.41 + 24	0.00	0.00	0.00	0.00
4.82 + 24	8.86 + 19	1.29 + 22	4.88 + 23	7.56 + 19
4.29 + 24	7.89 + 19	1.17 + 22	7.92 + 23	1.23 + 20
3.41 + 24	6.27 + 19	9.29 + 21	1.04 + 24	1.60 + 20
2.70 + 24	4.97 + 19	7.38 + 21	1.03 + 24	1.59 + 20
1.70 + 24	3.12 + 19	4.63 + 21	7.82 + 23	1.21 + 20
8.52 + 23	1.57 + 19	2.32 + 21	4.27 + 23	6.61 + 19
5.38 + 23	9.90 + 18	1.47 + 21	2.74 + 23	4.24 + 19
1.69 + 23	3.11 + 18	4.61 + 20	8.70 + 22	1.35 + 19
5.50 + 22	1.01 + 18	1.50 + 20	2.80 + 22	4.33 + 18
5.69 + 21	1.05 + 17	1.55 + 19	2.86 + 21	4.43 + 17
2.28 + 21	4.19 + 16	6.08 + 18	8.65 + 20	1.34 + 17
4.40 + 20	8.09 + 15	1.17 + 18	1.58 + 20	2.45 + 16
4.09 + 20	7.53 + 15	1.09 + 18	1.40 + 20	2.16 + 16

La-139 Nuc/T

0.00
 1.87 + 22
 3.27 + 22
 4.43 + 22
 4.45 + 22
 3.40 + 22
 1.86 + 22
 1.19 + 22
 3.79 + 21
 1.22 + 21
 1.25 + 20
 3.74 + 19
 6.82 + 18
 6.01 + 18

CASE 8. Cs-137 TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-D PLASMA

<u>Cs-137 Nuc/T</u>	<u>Cs-138 Nuc/T</u>	<u>Ba-137 Nuc/T</u>	<u>Ba-138 Nuc/T</u>	<u>Ba-139 Nuc/T</u>
5.41 + 24	0.00	0.00	0.00	0.00
5.29 + 24.	9.73 + 18	5.00 + 22	6.62 + 22	1.03 + 18
5.18 + 24	9.52 + 18	8.13 + 22	1.46 + 23	2.27 + 18
4.95 + 24	9.11 + 18	1.12 + 23	3.22 + 23	4.99 + 18
4.74 + 24	8.72 + 18	1.21 + 23	4.97 + 23	7.71 + 18
4.34 + 24	7.98 + 18	1.19 + 23	8.10 + 23	1.26 + 19
3.81 + 24	7.00 + 18	1.06 + 23	1.15 + 24	1.79 + 19
3.49 + 24	6.42 + 18	9.75 + 22	1.31 + 24	2.04 + 19
2.80 + 24	5.15 + 18	7.82 + 22	1.53 + 24	2.37 + 19
2.25 + 24	4.14 + 18	6.29 + 22	1.57 + 24	2.43 + 19
1.45 + 24	2.67 + 18	4.06 + 22	1.37 + 24	2.13 + 19
9.40 + 23	1.73 + 18	2.63 + 22	1.07 + 24	1.67 + 19
6.03 + 23	1.11 + 18	1.64 + 22	7.81 + 23	1.21 + 19
3.90 + 23	7.17 + 17	1.09 + 22	5.52 + 23	8.57 + 18

LA-139 Nuc/T

0.00
 8.44 + 20
 2.98 + 21
 9.35 + 21
 1.67 + 22
 3.08 + 22
 4.69 + 22
 5.46 + 22
 6.52 + 22
 6.76 + 22
 5.99 + 22
 4.71 + 22
 3.44 + 22
 2.44 + 22

CASE 9. ACTINIDE TRANSMUTATION FOR WALL LOADING OF 10 MW/m^2 D-T PLASMA

<u>U-232 Nuc/T</u>	<u>Np-237 Nuc/T</u>	<u>Np-238 Nuc/T</u>	<u>Np-239 Nuc/T</u>	<u>Pu-236 Nuc/T</u>
0.00	1.22 + 24	0.00	0.00	0.00
2.88 + 15	2.73 + 21	2.89 + 20	7.05 + 20	3.10 + 17
1.93 + 13	6.10 + 18	6.45 + 17	2.36 + 18	7.09 + 14
8.85 + 10	1.36 + 16	1.44 + 15	6.31 + 15	1.58 + 12
3.59 + 08	3.05 + 13	3.23 + 12	6.76 + 13	3.54 + 09
3.92 + 04	4.55 + 11	4.22 + 10	4.41 + 12	1.91 + 07
3.41 + 04	4.55 + 11	4.22 + 10	2.74 + 10	1.90 + 07
1.10 + 04	1.37 + 11	1.48 + 10	7.69 + 09	6.07 + 06
<u>Pu-238 Nuc/T</u>	<u>Pu-239 Nuc/T</u>	<u>Pu-240 Nuc/T</u>	<u>Pu-241 Nuc/T</u>	<u>Pu-242 Nuc/T</u>
2.11 + 21	6.63 + 22	2.71 + 22	1.26 + 22	4.38 + 21
1.53 + 20	9.89 + 19	1.22 + 20	3.94 + 19	8.51 + 21
1.83 + 19	3.39 + 17	4.44 + 18	1.32 + 18	2.43 + 21
7.79 + 18	1.96 + 15	1.93 + 18	5.69 + 17	6.91 + 20
3.36 + 18	4.66 + 14	9.13 + 17	2.69 + 17	1.97 + 20
6.26 + 17	8.57 + 13	2.01 + 17	5.92 + 16	1.65 + 19
4.05 + 15	5.50 + 11	2.00 + 15	5.90 + 14	2.30 + 16
1.41 + 14	2.31 + 10	9.05 + 13	2.67 + 13	7.57 + 14
<u>Pu-243 Nuc/T</u>	<u>Am-241 Nuc/T</u>	<u>Am-242 Nuc/T</u>	<u>Am-242m Nuc/T</u>	<u>Am-243 Nuc/T</u>
0.00	1.12 + 23	0.00	2.32 + 21	2.27 + 23
1.80 + 20	5.71 + 15	1.13 + 15	6.58 + 13	7.59 + 21
5.16 + 19	1.54 + 14	2.91 + 13	1.75 + 12	1.05 + 21
1.46 + 19	6.38 + 13	1.20 + 13	7.24 + 11	2.77 + 20
4.17 + 18	3.02 + 13	5.68 + 12	3.43 + 11	7.83 + 19
3.49 + 17	6.63 + 12	1.25 + 12	7.53 + 10	6.50 + 18
4.83 + 14	5.74 + 10	1.11 + 10	7.21 + 08	8.16 + 15
1.59 + 13	1.34 + 10	2.47 + 09	1.59 + 08	2.56 + 14
<u>Am-244 Nuc/T</u>	<u>Cm-242 Nuc/T</u>	<u>Cm-243 Nuc/T</u>	<u>Cm-244 Nuc/T</u>	<u>Cm-245 Nuc/T</u>
0.00	1.33 + 22	2.70 + 20	7.73 + 22	4.65 + 21
2.81 + 20	1.18 + 22	2.30 + 20	6.96 + 22	5.93 + 20
3.84 + 19	5.12 + 21	9.93 + 19	3.44 + 22	2.94 + 20
1.01 + 19	2.21 + 21	4.29 + 19	1.65 + 22	1.41 + 20
2.85 + 18	9.54 + 20	1.85 + 19	7.79 + 21	6.65 + 19
2.37 + 17	1.78 + 20	3.45 + 18	1.71 + 21	1.46 + 19
2.96 + 14	1.15 + 18	2.23 + 16	1.71 + 19	1.46 + 17
9.31 + 12	4.00 + 16	7.74 + 14	7.71 + 17	6.58 + 15
<u>Cm-246 Nuc/T</u>	<u>Cm-247 Nuc/T</u>	<u>Cm-248 Nuc/T</u>	<u>F. P. Nuc/T</u>	
5.29 + 20	0.00	0.00	0.00	
9.80 + 21	1.48 + 20	1.47 + 20	1.52 + 24	
1.39 + 22	2.33 + 20	5.09 + 20	1.58 + 24	
1.49 + 22	2.58 + 20	9.12 + 20	1.60 + 24	
1.45 + 22	2.54 + 20	1.27 + 21	1.61 + 24	
1.23 + 22	2.17 + 20	1.74 + 21	1.62 + 24	
6.33 + 21	1.20 + 20	1.81 + 21	1.62 + 24	
4.00 + 21	7.09 + 19	1.45 + 21	1.63 + 24	

CASE 9. ACTINIDE TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA

<u>U-232 Nuc/T</u>	<u>Np-237 Nuc/T</u>	<u>Np-238 Nuc/T</u>	<u>Np-239 Nuc/T</u>	<u>Pu-236 Nuc/T</u>
0.00	1.22 + 24	0.00	0.00	0.00
1.30 + 17	6.65 + 23	3.95 + 22	3.24 + 21	2.38 + 19
2.56 + 17	3.61 + 23	2.15 + 22	1.81 + 21	2.18 + 19
2.86 + 17	1.96 + 23	1.17 + 22	9.81 + 20	1.51 + 19
2.55 + 17	1.07 + 23	6.33 + 21	5.33 + 20	9.43 + 18
1.46 + 17	3.14 + 22	1.87 + 21	1.57 + 20	3.20 + 18
1.11 + 16	8.06 + 20	4.79 + 19	4.04 + 18	9.05 + 16
1.51 + 15	7.01 + 19	4.16 + 18	3.53 + 17	7.94 + 15
1.89 + 14	6.10 + 18	3.62 + 17	3.18 + 16	6.92 + 14
2.24 + 13	5.30 + 17	3.15 + 16	3.42 + 15	6.02 + 13
8.67 + 11	1.36 + 16	8.08 + 14	4.45 + 14	1.54 + 12
3.21 + 10	3.50 + 14	2.08 + 13	1.94 + 14	3.97 + 10
1.16 + 09	1.09 + 13	6.40 + 11	1.00 + 14	1.10 + 09
<u>Pu-238 Nuc/T</u>	<u>Pu-239 Nuc/T</u>	<u>Pu-240 Nuc/T</u>	<u>Pu-241 Nuc/T</u>	<u>Pu-242 Nuc/T</u>
2.11 + 21	6.63 + 22	2.71 + 22	1.26 + 22	4.38 + 21
1.06 + 23	5.52 + 21	1.47 + 22	5.28 + 21	2.20 + 22
7.71 + 22	2.61 + 21	5.53 + 21	1.98 + 21	2.46 + 22
4.53 + 22	1.41 + 21	2.47 + 21	8.47 + 20	2.34 + 22
2.55 + 22	7.65 + 20	1.28 + 21	4.26 + 20	2.14 + 22
8.24 + 21	2.26 + 20	4.90 + 20	1.55 + 20	1.72 + 22
6.50 + 20	5.86 + 18	1.55 + 20	4.57 + 19	8.42 + 21
2.80 + 20	5.42 + 17	1.13 + 20	3.34 + 19	5.23 + 21
1.51 + 20	6.57 + 16	8.50 + 19	2.50 + 19	3.25 + 21
8.48 + 19	1.63 + 16	6.34 + 19	1.87 + 19	2.05 + 21
3.61 + 19	5.48 + 15	4.05 + 19	1.19 + 19	1.04 + 21
1.54 + 19	2.34 + 15	2.57 + 19	7.57 + 18	5.37 + 20
6.56 + 18	1.02 + 15	1.62 + 19	4.78 + 18	2.83 + 20
		1.02 + 19	3.00 + 18	1.53 + 20
<u>Pu-243 Nuc/T</u>	<u>Am-241 Nuc/T</u>	<u>Am-242 Nuc/T</u>	<u>Am-242m Nuc/T</u>	<u>Am-243 Nuc/T</u>
0.00	1.12 + 23	0.00	2.32 + 21	2.27 + 23
5.34 + 19	1.43 + 22	1.15 + 21	1.74 + 20	1.60 + 23
6.02 + 19	1.84 + 21	1.48 + 20	2.24 + 19	1.10 + 23
5.74 + 19	2.37 + 20	1.90 + 19	2.89 + 18	7.69 + 22
5.25 + 19	3.09 + 19	2.48 + 18	3.76 + 17	5.41 + 22
4.22 + 19	7.15 + 17	5.62 + 16	8.55 + 15	2.78 + 22
2.06 + 19	5.19 + 16	3.82 + 15	5.90 + 14	5.66 + 21
1.28 + 19	3.74 + 16	2.75 + 15	4.25 + 14	2.74 + 21
8.00 + 18	2.80 + 16	2.06 + 15	3.18 + 14	1.54 + 21
5.03 + 18	2.09 + 16	1.54 + 15	2.37 + 14	9.31 + 20
2.55 + 18	1.34 + 16	9.83 + 14	1.52 + 14	4.60 + 20
1.32 + 18	8.50 + 15	6.24 + 14	9.64 + 13	2.34 + 20
6.95 + 17	5.36 + 15	3.94 + 14	6.08 + 13	1.22 + 20
3.75 + 17	3.37 + 15	2.48 + 14	3.83 + 13	6.50 + 19

CASE 9. ACTINIDE TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA (Con't)

<u>Am-244 Nuc/T</u>	<u>Cm-242 Nuc/T</u>	<u>Cm-243 Nuc/T</u>	<u>Cm-244 Nuc/T</u>	<u>Cm-245 Nuc/T</u>
0.00	1.33 + 22	2.70 + 20	7.73 + 22	4.65 + 21
1.89 + 21	5.33 + 22	8.95 + 20	1.25 + 23	1.02 + 21
1.31 + 21	5.22 + 22	1.00 + 21	1.55 + 23	1.29 + 21
9.12 + 20	4.60 + 22	9.03 + 20	1.70 + 23	1.43 + 21
6.41 + 20	4.00 + 22	7.88 + 20	1.77 + 23	1.49 + 21
3.29 + 20	3.01 + 22	5.93 + 20	1.74 + 23	1.47 + 21
6.68 + 19	1.28 + 22	2.53 + 20	1.27 + 23	1.08 + 21
3.23 + 19	7.27 + 21	1.43 + 20	9.67 + 22	8.25 + 20
1.82 + 19	4.12 + 21	8.11 + 19	7.27 + 22	6.20 + 20
1.10 + 19	2.33 + 21	4.59 + 19	5.42 + 22	4.62 + 20
5.42 + 18	9.94 + 20	1.96 + 19	3.46 + 22	2.95 + 20
2.76 + 18	4.23 + 20	8.34 + 18	2.19 + 22	1.87 + 20
1.44 + 18	1.80 + 20	3.56 + 18	1.38 + 22	1.18 + 20
7.65 + 17	7.69 + 19	1.52 + 18	8.68 + 21	7.41 + 19
<u>Cm-246 Nuc/T</u>	<u>Cm-247 Nuc/T</u>	<u>Cm-248 Nuc/T</u>	<u>F. P. Nuc/T</u>	
5.29 + 20	0.00	0.00	0.00	
2.16 + 21	1.22 + 19	1.17 + 18	3.96 + 23	
3.62 + 21	2.93 + 19	5.42 + 18	6.14 + 23	
5.32 + 21	5.12 + 19	1.37 + 19	7.39 + 23	
7.13 + 21	7.70 + 19	2.69 + 19	8.14 + 23	
1.08 + 22	1.35 + 20	6.98 + 19	8.96 + 23	
1.96 + 22	3.00 + 20	3.32 + 20	9.93 + 23	
2.35 + 22	3.80 + 20	5.90 + 20	1.03 + 24	
2.60 + 22	4.34 + 20	8.84 + 20	1.06 + 24	
2.76 + 22	4.68 + 20	1.19 + 21	1.07 + 24	
2.85 + 22	4.92 + 20	1.66 + 21	1.10 + 24	
2.84 + 22	4.94 + 20	2.09 + 21	1.11 + 24	
2.76 + 22	4.84 + 20	2.47 + 21	1.12 + 24	
2.65 + 22	4.65 + 20	2.80 + 21	1.12 + 24	

CASE 10. ACTINIDE TRANSMUTATION FOR WALL LOADING OF 10 MW/m^2 ; D-D PLASMA

<u>U-232 Nuc/T</u>	<u>Np-237 Nuc/T</u>	<u>Np-238 Nuc/T</u>	<u>Np-239 Nuc/T</u>	<u>Pu-236 Nuc/T</u>
0.00	1.22 + 24	0.00 0	0.00	0.00
7.17 + 12	4.46 + 18	4.95 + 17	2.57 + 19	5.19 + 14
9.92 + 07	1.66 + 13	1.84 + 12	2.51 + 15	1.91 + 09
1.04 + 04	2.32 + 11	2.46 + 10	2.56 + 12	1.09 + 07
1.29 + 04	3.35 + 11	3.51 + 10	2.08 + 11	1.49 + 07
9.36 + 03	2.32 + 11	2.46 + 10	2.03 + 10	1.09 + 07
9.36 + 03	2.32 + 11	2.46 + 10	1.87 + 10	1.09 + 07
<u>Pu-238 Nuc/T</u>	<u>Pu-239 Nuc/T</u>	<u>Pu-240 Nuc/T</u>	<u>Pu-241 Nuc/T</u>	<u>Pu-242 Nuc/T</u>
2.11 + 21	6.63 + 22	2.71 + 22	1.26 + 22	4.38 + 21
6.92 + 18	1.71 + 18	3.28 + 18	1.01 + 18	2.12 + 21
1.30 + 18	3.45 + 14	3.15 + 17	9.30 + 16	1.58 + 20
2.49 + 17	3.40 + 13	6.69 + 16	1.97 + 16	1.20 + 19
4.76 + 16	6.49 + 12	1.41 + 16	4.14 + 15	9.60 + 17
1.74 + 15	2.38 + 11	6.11 + 14	1.80 + 14	9.37 + 15
8.74 + 10	1.34 + 09	4.80 + 10	1.50 + 10	3.84 + 11
<u>Pu-243 Nuc/T</u>	<u>Am-241 Nuc/T</u>	<u>Am-242 Nuc/T</u>	<u>Am-242m Nuc/T</u>	<u>Am-243 Nuc/T</u>
0.00	1.12 + 23	0.00	2.32 + 21	2.27 + 23
7.92 + 19	6.25 + 13	1.33 + 13	7.14 + 11	7.92 + 20
5.92 + 18	5.08 + 12	1.05 + 12	5.77 + 10	5.44 + 19
4.50 + 17	1.08 + 12	2.23 + 11	1.22 + 10	4.12 + 18
3.59 + 16	2.23 + 11	4.63 + 10	2.54 + 09	3.25 + 17
3.48 + 14	1.23 + 10	2.53 + 09	1.26 + 08	2.95 + 15
1.42 + 10	2.15 + 09	4.96 + 08	8.40 + 06	1.80 + 11
<u>Am-244 Nuc/T</u>	<u>Cm-242 Nuc/T</u>	<u>Cm-243 Nuc/T</u>	<u>Cm-244 Nuc/T</u>	<u>Cm-245 Nuc/T</u>
0.00	1.33 + 22	2.70 + 20	7.73 + 22	4.65 + 21
3.29 + 19	3.99 + 21	7.71 + 19	2.56 + 22	1.18 + 20
2.25 + 18	7.62 + 20	1.47 + 19	5.52 + 21	4.71 + 19
1.71 + 17	1.45 + 20	2.82 + 18	1.17 + 21	9.99 + 18
1.34 + 16	2.78 + 19	5.38 + 17	2.46 + 20	2.10 + 18
1.22 + 14	1.01 + 18	1.96 + 16	1.07 + 19	9.12 + 16
7.39 + 09	4.93 + 13	9.54 + 11	8.17 + 14	6.97 + 12
<u>Cm-246 Nuc/T</u>	<u>Cm-247 Nuc/T</u>	<u>Cm-248 Nuc/T</u>	<u>F. P. Nuc/T</u>	
5.29 + 20	0.00	0.00	0.00	
1.15 + 22	1.95 + 20	4.51 + 20	1.62 + 24	
1.17 + 22	2.05 + 20	1.07 + 21	1.65 + 24	
9.77 + 21	1.73 + 20	1.44 + 21	1.65 + 24	
7.85 + 21	1.39 + 20	1.58 + 21	1.66 + 24	
4.93 + 21	8.73 + 19	1.45 + 21	1.66 + 24	
1.21 + 21	1.14 + 19	5.73 + 20	1.66 + 24	

CASE 10. ACTINIDE TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-D PLASMA

<u>U-232 Nuc/T</u>	<u>Np-237 Nuc/T</u>	<u>Np-238 Nuc/T</u>	<u>Np-239 Nuc/T</u>	<u>Pu-236 Nuc/T</u>
0.00	1.22 ++24	0.00	0.00	0.00
1.27 + 17	3.50 + 23	2.77 + 22	5.13 + 21	2.16 + 19
1.22 + 17	1.00 + 23	7.93 + 21	1.51 + 21	9.03 + 18
6.81 + 16	2.86 + 22	2.27 + 21	4.33 + 20	2.96 + 18
3.07 + 16	8.18 + 21	6.48 + 20	1.24 + 20	8.97 + 17
4.70 + 15	6.68 + 20	5.29 + 19	1.01 + 19	7.61 + 16
7.95 + 12	3.65 + 17	2.89 + 16	6.13 + 15	4.20 + 13
9.13 + 10	2.43 + 15	1.93 + 14	2.59 + 14	2.80 + 11
9.82 + 08	1.67 + 13	1.32 + 12	8.43 + 13	1.89 + 09
1.02 + 07	5.15 + 10	4.60 + 09	3.27 + 13	1.06 + 07
1.15 + 05	2.83 + 11	2.11 + 10	8.51 + 12	1.21 + 07
2.63 + 05	6.79 + 11	4.93 + 10	2.42 + 12	3.03 + 07
3.02 + 05	7.90 + 11	5.73 + 10	7.56 + 11	3.41 + 07
<u>Pu-238 Nuc/T</u>	<u>Pu-239 Nuc/T</u>	<u>Pu-240 Nuc/T</u>	<u>Pu-241 Nuc/T</u>	<u>Pu-242 Nuc/T</u>
2.11 + 21	6.63 + 22	2.71 + 22	1.26 + 22	4.38 + 21
4.83 + 22	3.47 + 21	5.67 + 21	1.97 + 21	2.25 + 22
1.53 + 22	1.06 + 21	1.47 + 21	4.86 + 20	1.98 + 22
4.72 + 21	3.03 + 20	4.70 + 20	1.53 + 20	1.59 + 22
1.60 + 21	8.67 + 19	1.86 + 20	5.88 + 19	1.25 + 22
3.50 + 20	7.12 + 18	6.89 + 19	2.06 + 19	7.56 + 21
6.58 + 19	1.32 + 16	2.53 + 19	7.45 + 18	1.68 + 21
2.69 + 19	3.82 + 15	1.37 + 19	4.05 + 18	6.27 + 20
1.10 + 19	1.55 + 15	7.38 + 18	2.18 + 18	2.40 + 20
4.49 + 18	6.31 + 14	3.94 + 18	1.16 + 18	9.51 + 19
1.17 + 18	1.65 + 14	1.52 + 18	4.48 + 17	2.54 + 19
3.07 + 17	4.32 + 13	5.81 + 17	1.71 + 17	7.41 + 18
8.01 + 16	1.14 + 13	2.22 + 17	6.54 + 16	2.35 + 18
		8.44 + 16	2.49 + 16	7.98 + 17
<u>Pu-243 Nuc/T</u>	<u>Am-241 Nuc/T</u>	<u>Am-242 Nuc/T</u>	<u>Am-242m Nuc/T</u>	<u>Am-243 Nuc/T</u>
0.00	1.12 + 23	0.00	2.32 + 21	2.27 + 23
1.11 + 20	1.66 + 21	2.14 + 20	2.02 + 19	1.09 + 23
9.78 + 19	2.49 + 19	3.21 + 18	3.03 + 17	5.21 + 22
7.87 + 19	4.73 + 17	5.97 + 16	5.68 + 15	2.61 + 22
6.18 + 19	4.48 + 16	5.24 + 15	5.18 + 14	1.40 + 22
3.74 + 19	1.18 + 16	1.34 + 15	1.34 + 14	5.05 + 21
8.27 + 18	4.07 + 15	4.58 + 14	4.62 + 13	7.61 + 20
3.10 + 18	2.22 + 15	2.49 + 14	2.51 + 13	2.78 + 20
1.19 + 18	1.19 + 15	1.34 + 14	1.35 + 13	1.05 + 20
4.70 + 17	6.35 + 14	7.15 + 13	7.21 + 12	4.11 + 19
1.25 + 17	2.45 + 14	2.76 + 13	2.78 + 12	1.07 + 19
3.66 + 16	9.39 + 13	1.06 + 13	1.07 + 12	3.05 + 18
1.16 + 16	3.58 + 13	4.03 + 12	4.07 + 11	9.46 + 17
3.94 + 15	1.36 + 13	1.53 + 12	1.55 + 11	3.15 + 17

CASE 10. ACTINIDE TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-D PLASMA (Cont)

<u>Am-244 Nuc/T</u>	<u>Cm-242 Nuc/T</u>	<u>Cm-243 Nuc/T</u>	<u>Cm-244 Nuc/T</u>	<u>Cm-245 Nuc/T</u>
0.00	1.33 + 22	2.70 + 20	7.73 + 22	4.65 + 21
2.11 + 21	4.36 + 22	8.32 + 20	1.37 + 23	1.14 + 21
1.01 + 21	3.54 + 22	6.90 + 20	1.52 + 23	1.29 + 21
5.06 + 20	2.83 + 22	5.52 + 20	1.48 + 23	1.25 + 21
2.70 + 20	2.26 + 22	4.42 + 20	1.35 + 23	1.15 + 21
9.71 + 19	1.45 + 22	2.82 + 20	1.06 + 23	9.01 + 20
1.46 + 19	3.78 + 21	7.38 + 19	4.43 + 22	3.78 + 20
5.32 + 18	1.55 + 21	3.01 + 19	2.41 + 22	2.05 + 20
2.02 + 18	6.32 + 20	1.23 + 19	1.29 + 22	1.10 + 20
7.86 + 17	2.58 + 20	5.04 + 18	6.89 + 21	5.88 + 19
2.05 + 17	6.75 + 19	1.32 + 18	2.65 + 21	2.27 + 19
5.83 + 16	1.76 + 19	3.44 + 17	1.02 + 21	8.67 + 18
1.81 + 16	4.61 + 18	8.99 + 16	3.88 + 20	3.31 + 18
6.03 + 15	1.20 + 18	2.35 + 16	1.47 + 20	1.26 + 18
<u>Cm-246 Nuc/T</u>	<u>Cm-247 Nuc/T</u>	<u>Cm-248 Nuc/T</u>	<u>F. P. Nuc/T</u>	
5.29 + 20	0.00	0.00	0.00	
3.48 + 21	2.91 + 19	5.60 + 18	8.04 + 23	
6.62 + 21	7.34 + 19	2.68 + 19	1.06 + 24	
9.79 + 21	1.26 + 20	6.82 + 19	1.15 + 24	
1.27 + 22	1.78 + 20	1.31 + 20	1.20 + 24	
1.74 + 22	2.69 + 20	3.11 + 20	1.25 + 24	
2.39 + 22	4.07 + 20	1.08 + 21	1.32 + 24	
2.46 + 22	4.27 + 20	1.62 + 21	1.34 + 24	
2.39 + 22	4.19 + 20	2.09 + 21	1.35 + 24	
2.26 + 22	3.98 + 20	2.48 + 21	1.36 + 24	
2.02 + 22	3.57 + 20	2.89 + 21	1.37 + 24	
1.78 + 22	3.14 + 20	3.12 + 21	1.37 + 24	
1.55 + 22	2.75 + 20	3.21 + 21	1.37 + 24	
1.35 + 22	2.39 + 20	3.20 + 21	1.37 + 24	

CASE 11. Np-237 TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA

<u>U-232 Nuc/T</u>	<u>Np-237 Nuc/T</u>	<u>Np-238 Nuc/T</u>	<u>Np-239 Nuc/T</u>	<u>Pu-236 Nuc/T</u>
0.00	1.22 + 24	0.00	0.00	0.00
1.30 + 17	6.65 + 23	3.95 + 22	3.24 + 21	2.38 + 19
2.56 + 17	3.61 + 23	2.15 + 22	1.81 + 21	2.18 + 19
2.86 + 17	1.06 + 23	1.17 + 22	9.81 + 20	1.51 + 19
2.55 + 17	1.07 + 23	6.33 + 21	5.33 + 20	9.43 + 18
1.46 + 17	3.14 + 22	1.87 + 21	1.57 + 20	3.20 + 18
1.11 + 16	8.06 + 20	4.79 + 19	4.03 + 18	9.05 + 16
1.51 + 15	7.01 + 19	4.16 + 18	3.51 + 17	7.94 + 15
1.89 + 14	6.10 + 18	3.62 + 17	3.06 + 16	6.92 + 14
2.24 + 13	5.30 + 17	3.15 + 16	2.73 + 15	6.02 + 13
8.67 + 11	1.36 + 16	8.08 + 14	1.05 + 14	1.54 + 12
3.21 + 10	3.49 + 14	2.07 + 13	1.91 + 13	3.96 + 10
1.16 + 09	1.03 + 13	6.04 + 11	8.32 + 12	1.07 + 09
<u>Pu-238 Nuc/T</u>	<u>Pu-239 Nuc/T</u>	<u>Pu-240 Nuc/T</u>	<u>Pu-241 Nuc/T</u>	<u>Pu-242 Nuc/T</u>
0.00	0.00	0.00	0.00	0.00
1.04 + 23	3.93 + 21	1.60 + 21	3.43 + 20	1.32 + 20
7.53 + 22	2.57 + 21	2.15 + 21	6.29 + 20	7.93 + 20
4.36 + 22	1.41 + 21	1.52 + 21	4.82 + 20	1.40 + 21
2.41 + 22	7.65 + 20	9.11 + 20	2.97 + 20	1.71 + 21
7.14 + 21	2.26 + 20	2.85 + 20	9.40 + 19	1.73 + 21
1.83 + 20	5.79 + 18	8.21 + 18	2.68 + 18	9.24 + 20
1.60 + 19	5.04 + 17	1.61 + 18	4.92 + 17	5.61 + 20
1.40 + 18	4.39 + 16	9.96 + 17	2.92 + 17	3.39 + 20
1.25 + 17	3.90 + 15	8.36 + 17	2.45 + 17	2.05 + 20
5.14 + 15	1.43 + 14	6.25 + 17	1.83 + 17	9.69 + 19
9.74 + 14	2.42 + 13	4.39 + 17	1.29 + 17	4.61 + 19
4.00 + 14	1.04 + 13	2.96 + 17	8.72 + 16	2.21 + 19
		1.95 + 17	5.75 + 16	1.07 + 19
<u>Pu-243 Nuc/T</u>	<u>Am-241 Nuc/T</u>	<u>Am-242 Nuc/T</u>	<u>Am-242m Nuc/T</u>	<u>Am-243 Nuc/T</u>
0.00	0.00	0.00	0.00	0.00
3.04 + 17	1.55 + 17	1.00 + 16	1.59 + 15	3.04 + 18
1.91 + 18	5.70 + 17	4.09 + 16	6.35 + 15	4.80 + 19
3.41 + 18	5.86 + 17	4.33 + 16	6.67 + 15	1.48 + 20
4.18 + 18	4.10 + 17	3.06 + 16	4.72 + 15	2.62 + 20
4.24 + 18	1.41 + 17	1.06 + 16	1.63 + 15	4.22 + 20
2.27 + 18	4.01 + 15	3.01 + 14	4.62 + 13	3.74 + 20
1.38 + 18	6.26 + 14	4.65 + 13	7.16 + 12	2.47 + 20
8.31 + 17	3.27 + 14	2.40 + 13	3.71 + 12	1.54 + 20
5.03 + 17	2.70 + 14	1.98 + 13	3.06 + 12	9.41 + 19
2.38 + 17	2.03 + 14	1.49 + 13	2.31 + 12	4.45 + 19
1.13 + 17	1.44 + 14	1.05 + 13	1.63 + 12	2.11 + 19
5.42 + 16	9.74 + 13	7.15 + 12	1.10 + 12	1.01 + 19
2.62 + 16	6.43 + 13	4.72 + 12	7.30 + 11	4.85 + 18

CASE 11. Np-237 TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA (Cont)

<u>Am-244 Nuc/T</u>	<u>Cm-242 Nuc/T</u>	<u>Cm-243 Nuc/T</u>	<u>Cm-244 Nuc/T</u>	<u>Cm-245 Nuc/T</u>
0.00	0.00	0.00	0.00	0.00
3.12 + 16	2.86 + 16	2.13 + 14	1.26 + 17	6.02 + 14
5.37 + 17	3.62 + 17	4.63 + 15	5.55 + 18	3.64 + 16
1.70 + 18	8.66 + 17	1.38 + 16	3.16 + 19	2.31 + 17
3.04 + 18	1.21 + 18	2.13 + 16	8.70 + 19	6.71 + 17
4.93 + 18	1.35 + 18	2.56 + 16	2.67 + 20	2.16 + 18
4.40 + 18	7.00 + 17	1.38 + 16	7.79 + 20	6.54 + 18
2.91 + 18	4.01 + 17	7.90 + 15	8.76 + 20	7.41 + 18
1.81 + 18	2.28 + 17	4.50 + 15	8.35 + 20	7.08 + 18
1.11 + 18	1.30 + 17	2.56 + 15	7.28 + 20	6.19 + 18
5.24 + 17	5.64 + 16	1.11 + 15	5.42 + 20	4.61 + 18
2.48 + 17	2.47 + 16	4.86 + 14	3.78 + 20	3.22 + 18
1.19 + 17	1.10 + 16	2.16 + 14	2.54 + 20	2.17 + 18
5.71 + 16	4.99 + 15	9.79 + 13	1.67 + 20	1.43 + 18
<u>Cm-246 Nuc/T</u>	<u>Cm-247 Nuc/T</u>	<u>Cm-248 Nuc/T</u>	<u>F. P. Nuc/T</u>	
0.00	0.00	0.00	0.00	
9.62 + 13	1.00 + 11	2.04 + 09	2.75 + 23	
1.45 + 16	3.35 + 13	1.60 + 12	1.55 + 23	
1.61 + 17	5.90 + 14	4.72 + 13	5.54 + 23	
7.08 + 17	3.51 + 15	4.10 + 14	6.08 + 23	
4.16 + 18	3.04 + 16	6.14 + 15	6.54 + 23	
3.83 + 19	4.63 + 17	2.49 + 17	6.73 + 23	
7.22 + 19	1.00 + 18	8.25 + 17	6.74 + 23	
1.06 + 20	1.59 + 18	1.83 + 18	6.74 + 23	
1.34 + 20	2.12 + 18	3.23 + 18	6.75 + 23	
1.65 + 20	2.73 + 18	5.87 + 18	6.75 + 23	
1.83 + 20	3.11 + 18	8.84 + 18	6.75 + 23	
1.91 + 20	3.28 + 18	1.18 + 19	6.75 + 23	
1.91 + 20	3.33 + 18	1.47 + 19	6.75 + 23	

CASE 12. Am AND Cm TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA

<u>U-232 Nuc/T</u>	<u>Np-237 Nuc/T</u>	<u>Np-238 Nuc/T</u>	<u>Np-239 Nuc/T</u>	<u>Pu-236 Nuc/T</u>
0.00	0.00	0.00	0.00	0.00
1.84 + 11	2.02 + 18	1.14 + 17	1.44 + 17	4.68 + 13
5.62 + 11	1.36 + 18	7.99 + 16	1.01 + 17	6.27 + 13
7.61 + 11	7.71 + 17	4.57 + 16	6.80 + 16	5.05 + 13
7.52 + 11	4.23 + 17	2.51 + 16	4.59 + 18	3.38 + 13
4.81 + 11	1.25 + 17	7.43 + 15	2.13 + 16	1.22 + 13
4.03 + 10	3.21 + 15	1.91 + 14	2.71 + 15	3.59 + 11
5.62 + 09	2.82 + 14	1.67 + 13	9.64 + 14	3.17 + 10
7.12 + 08	2.66 + 13	1.57 + 12	4.63 + 14	2.86 + 09
8.65 + 07	3.82 + 12	2.22 + 11	2.69 + 14	3.20 + 08
4.36 + 06	1.13 + 12	6.42 + 10	1.37 + 14	5.70 + 07
7.89 + 05	6.82 + 11	3.85 + 10	7.47 + 13	3.22 + 07
4.23 + 05	4.27 + 11	2.41 + 10	4.19 + 13	2.01 + 07
2.63 + 05	2.67 + 11	1.51 + 10	2.39 + 13	1.26 + 07
<u>Pu-238 Nuc/T</u>	<u>Pu-239 Nuc/T</u>	<u>Pu-240 Nuc/T</u>	<u>Pu-241 Nuc/T</u>	<u>Pu-242 Nuc/T</u>
0.00 + 0.00	0.00	0.00	0.00	0.00
1.30 + 21	3.42 + 17	2.90 + 19	6.07 + 18	7.99 + 21
1.73 + 21	3.74 + 17	7.17 + 19	1.85 + 19	8.19 + 21
1.64 + 21	3.16 + 17	1.03 + 20	2.83 + 19	7.39 + 21
1.45 + 21	2.59 + 17	1.22 + 20	3.45 + 19	6.56 + 21
1.09 + 21	1.77 + 17	1.35 + 20	3.90 + 19	5.18 + 21
4.66 + 20	6.70 + 16	1.05 + 20	3.08 + 19	2.62 + 21
2.64 + 20	3.72 + 16	7.95 + 19	2.34 + 19	1.68 + 21
1.50 + 20	2.09 + 16	5.92 + 19	1.75 + 19	1.10 + 21
8.47 + 19	1.19 + 16	4.38 + 19	1.29 + 19	7.22 + 20
3.61 + 19	5.08 + 15	2.77 + 19	8.15 + 18	3.93 + 20
1.54 + 19	2.19 + 15	1.74 + 19	5.13 + 18	2.19 + 20
6.56 + 18	9.44 + 14	1.09 + 19	3.22 + 18	1.24 + 20
2.79 + 18	4.10 + 14	6.82 + 18	2.01 + 18	7.18 + 19
<u>Pu-243 Nuc/T</u>	<u>Am-241 Nuc/T</u>	<u>Am-242 Nuc/T</u>	<u>Am-242m Nuc/T</u>	<u>Am-243 Nuc/T</u>
0.00	1.12 + 23	0.00	2.32 + 21	2.27 + 23
1.95 + 19	1.43 + 22	1.15 + 21	1.74 + 20	1.58 + 23
2.01 + 19	1.84 + 21	1.47 + 20	2.24 + 19	1.08 + 23
1.81 + 19	2.36 + 20	1.89 + 19	2.87 + 18	7.37 + 22
1.61 + 19	3.03 + 19	2.43 + 18	3.68 + 17	5.04 + 22
1.27 + 19	5.39 + 17	4.29 + 16	6.52 + 15	2.38 + 22
6.41 + 18	3.43 + 16	2.52 + 15	3.89 + 14	3.16 + 21
4.13 + 18	2.62 + 16	1.93 + 15	2.97 + 14	1.15 + 21
2.69 + 18	1.96 + 16	1.44 + 15	2.22 + 14	5.60 + 20
1.77 + 18	1.45 + 16	1.06 + 15	1.64 + 14	3.27 + 20
9.63 + 17	9.16 + 15	6.72 + 14	1.03 + 14	1.67 + 20
5.36 + 17	5.76 + 15	4.22 + 14	6.53 + 13	9.15 + 19
3.04 + 17	3.61 + 15	2.65 + 14	4.10 + 13	5.13 + 19
1.76 + 17	2.26 + 15	1.66 + 14	2.56 + 13	2.93 + 19

CASE 12. Am and Cm TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA (Cont)

<u>Am-244 Nuc/T</u>	<u>Cm-242 Nuc/T</u>	<u>Cm-243 Nuc/T</u>	<u>Cm-244 Nuc/T</u>	<u>Cm-245 Nuc/T</u>
0.00	1.33 + 22	0.00	0.00	0.00
1.88 + 21	5.33 + 22	8.86 + 20	5.36 + 22	4.08 + 20
1.28 + 21	5.21 + 22	1.00 + 21	8.84 + 22	7.20 + 20
8.75 + 20	4.60 + 22	9.03 + 20	1.08 + 23	9.01 + 20
5.98 + 20	4.00 + 22	7.88 + 20	1.18 + 23	9.95 + 20
2.83 + 20	3.01 + 22	5.93 + 20	1.22 + 23	1.03 + 21
3.74 + 19	1.28 + 22	2.53 + 20	9.02 + 22	7.69 + 20
1.36 + 19	7.27 + 21	1.43 + 20	6.81 + 22	5.81 + 20
6.59 + 18	4.11 + 21	8.11 + 19	5.06 + 22	4.32 + 20
3.85 + 18	2.33 + 21	4.59 + 19	3.74 + 22	3.19 + 20
1.97 + 18	9.93 + 20	1.96 + 19	2.36 + 22	2.01 + 20
1.08 + 18	4.23 + 20	8.34 + 18	1.48 + 22	1.27 + 20
6.03 + 17	1.80 + 20	3.55 + 18	9.30 + 21	7.93 + 19
3.45 + 17	7.69 + 19	1.51 + 18	5.82 + 21	4.96 + 19

<u>Cm-246 Nuc/T</u>	<u>Cm-247 Nuc/T</u>	<u>Cm-248 Nuc/T</u>	<u>F. P. Nuc/T</u>
0.00	0.00	0.00	0.00
2.41 + 20	6.52 + 17	3.20 + 16	3.66 + 22
9.74 + 20	5.05 + 18	5.44 + 17	5.85 + 22
2.01 + 21	1.45 + 19	2.50 + 18	7.67 + 22
3.20 + 21	2.85 + 19	6.89 + 18	9.27 + 22
5.73 + 21	6.51 + 19	2.58 + 19	1.20 + 23
1.21 + 22	1.82 + 20	1.76 + 20	1.78 + 23
1.50 + 22	2.39 + 20	3.38 + 20	2.04 + 23
1.68 + 22	2.79 + 20	5.28 + 20	2.22 + 23
1.79 + 22	3.03 + 20	7.31 + 20	2.36 + 23
1.86 + 22	3.21 + 20	1.04 + 21	2.50 + 23
1.86 + 22	3.24 + 20	1.32 + 21	2.59 + 23
1.81 + 22	3.17 + 20	1.58 + 21	2.65 + 23
1.74 + 22	3.06 + 20	1.80 + 21	2.69 + 23

CASE 13. PLUTONIUM TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA

<u>U-232 Nuc/T</u>	<u>Np-237 Nuc/T</u>	<u>Np-238 Nuc/T</u>	<u>Np-239 Nuc/T</u>	<u>Pu-236 Nuc/T</u>
0.00	0.00	0.00	0.00	0.00
2.26 + 09	5.47 + 16	2.78 + 15	1.23 + 17	8.03 + 11
1.39 + 10	7.31 + 16	4.12 + 15	3.21 + 17	2.21 + 12
2.73 + 10	5.53 + 16	3.22 + 15	4.67 + 17	2.57 + 12
3.38 + 10	3.45 + 16	2.03 + 15	5.46 + 17	2.12 + 12
2.78 + 10	1.11 + 16	6.58 + 14	5.69 + 17	9.44 + 11
3.01 + 09	3.28 + 14	1.94 + 13	3.43 + 17	3.35 + 10
4.69 + 08	7.00 + 13	4.00 + 12	2.14 + 17	4.74 + 09
9.11 + 07	4.56 + 13	2.57 + 12	1.31 + 17	2.16 + 09
3.94 + 07	3.85 + 13	2.17 + 12	7.96 + 16	1.75 + 09
2.61 + 07	2.88 + 13	1.62 + 12	3.76 + 16	1.32 + 09
1.87 + 07	2.02 + 13	1.14 + 12	1.79 + 16	9.35 + 08
1.29 + 07	1.36 + 13	7.68 + 11	8.56 + 15	6.35 + 08
<u>Pu-238 Nuc/T</u>	<u>Pu-239 Nuc/T</u>	<u>Pu-240 Nuc/T</u>	<u>Pu-241 Nuc/T</u>	<u>Pu-242 Nuc/T</u>
4.23 + 23	1.33 + 25	5.42 + 24	2.52 + 24	8.76 + 23
6.20 + 22	3.18 + 23	2.59 + 24	9.83 + 23	2.77 + 24
9.13 + 21	7.61 + 21	6.48 + 23	2.62 + 23	3.12 + 24
1.39 + 21	1.83 + 20	1.55 + 23	6.32 + 22	2.92 + 24
2.54 + 20	5.01 + 18	3.72 + 22	1.51 + 22	2.61 + 24
5.12 + 19	6.96 + 17	2.79 + 21	1.05 + 21	2.04 + 24
2.02 + 19	4.27 + 17	1.37 + 21	3.94 + 20	9.53 + 23
1.15 + 19	2.68 + 17	1.40 + 21	4.07 + 20	5.75 + 23
6.54 + 18	1.64 + 17	1.27 + 21	3.70 + 20	3.48 + 23
3.74 + 18	9.93 + 16	1.07 + 21	3.14 + 20	2.11 + 23
1.63 + 18	4.70 + 16	7.77 + 20	2.28 + 20	9.98 + 22
7.25 + 17	2.23 + 16	5.35 + 20	1.57 + 20	4.76 + 22
3.28 + 17	1.07 + 16	3.56 + 20	1.05 + 20	2.29 + 22
<u>Pu-243 Nuc/T</u>	<u>Am-241 Nuc/T</u>	<u>Am-242 Nuc/T</u>	<u>Am-242m Nuc/T</u>	<u>Am-243 Nuc/T</u>
0.0	0.0	0.0	0.0	0.0
6.72 + 21	1.33 + 21	9.86 + 19	1.52 + 19	2.05 + 23
7.63 + 21	5.91 + 20	4.53 + 19	6.94 + 18	4.49 + 23
7.15 + 21	1.80 + 20	1.39 + 19	2.13 + 18	6.13 + 23
6.40 + 21	4.81 + 19	3.74 + 18	5.71 + 17	6.96 + 23
4.99 + 21	3.14 + 18	2.44 + 17	3.73 + 16	7.07 + 23
2.34 + 21	4.18 + 17	3.05 + 16	4.72 + 15	4.19 + 23
1.41 + 21	4.41 + 17	3.23 + 16	4.99 + 15	2.61 + 23
8.53 + 20	4.06 + 17	2.98 + 16	4.60 + 15	1.60 + 23
5.17 + 20	3.47 + 17	2.55 + 16	3.94 + 15	9.68 + 22
2.45 + 20	2.54 + 17	1.86 + 16	2.88 + 15	4.58 + 22
1.17 + 20	1.75 + 17	1.29 + 16	1.99 + 15	2.18 + 22
5.62 + 19	1.17 + 17	8.61 + 15	1.33 + 15	1.04 + 22
2.74 + 19	7.68 + 16	5.64 + 15	8.71 + 14	5.04 + 21

CASE 13 PLUTONIUM TRANSMUTATION FOR WALL LOADING OF 1 MW/M^2 ; D-T PLASMA (Cont.)

<u>Am-244 Nuc/T</u>	<u>Cm-242 Nuc/T</u>	<u>Cm-243 Nuc/T</u>	<u>Cm-244 Nuc/T</u>	<u>Cm-245 Nuc/T</u>
2.32 + 21	9.68 + 20	1.21 + 19	2.41 + 22	1.56 + 20
5.19 + 21	1.73 + 21	2.97 + 19	1.15 + 23	8.61 + 20
7.15 + 21	1.84 + 21	3.46 + 19	2.58 + 23	2.03 + 21
8.14 + 21	1.69 + 21	3.28 + 19	4.26 + 23	3.44 + 21
8.30 + 21	1.30 + 21	2.55 + 19	7.54 + 23	6.23 + 21
4.93 + 21	5.56 + 20	1.09 + 19	1.26 + 24	1.06 + 22
3.08 + 21	3.16 + 20	6.22 + 18	1.25 + 24	1.06 + 22
1.88 + 21	1.80 + 20	3.55 + 18	1.11 + 24	9.46 + 21
1.14 + 21	1.03 + 20	2.03 + 18	9.35 + 23	7.95 + 21
5.39 + 20	4.51 + 19	8.87 + 17	6.72 + 23	5.72 + 21
2.56 + 20	2.00 + 19	3.94 + 17	4.60 + 23	3.92 + 21
1.23 + 20	9.10 + 18	1.79 + 17	3.06 + 23	2.61 + 21
5.94 + 19	4.25 + 18	8.32 + 16	1.99 + 23	1.70 + 21
<u>Cm-246 Nuc/T</u>	<u>Cm-247 Nuc/T</u>	<u>Cm-248 Nuc/T</u>	<u>F.P. Nuc/T</u>	
0.00	0.00	0.00	0.00	
5.36 + 19	1.03 + 17	3.72 + 15	1.52 + 25	
6.41 + 20	2.37 + 18	1.83 + 17	1.75 + 25	
2.44 + 21	1.29 + 19	1.57 + 18	1.80 + 25	
5.89 + 21	3.93 + 19	6.67 + 18	1.82 + 25	
1.81 + 22	1.63 + 20	4.50 + 19	1.85 + 25	
8.37 + 22	1.10 + 21	7.41 + 20	1.93 + 25	
1.34 + 23	1.96 + 21	1.94 + 21	1.97 + 25	
1.78 + 23	2.77 + 21	3.75 + 21	2.02 + 25	
2.14 + 23	3.46 + 21	6.06 + 21	2.05 + 25	
2.50 + 23	4.19 + 21	1.01 + 22	2.09 + 25	
2.69 + 23	4.60 + 21	1.44 + 22	2.11 + 25	
2.75 + 23	4.76 + 21	1.87 + 22	2.13 + 25	
2.73 + 23	4.76 + 21	2.26 + 22	2.14 + 25	

CASE 14. FUEL LOADING TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA

<u>U-232 Nuc/T</u>	<u>Np-237 Nuc/T</u>	<u>Np-238 Nuc/T</u>	<u>Np-239 Nuc/T</u>	<u>Pu-236 Nuc/T</u>
0.00	1.22 + 24	0.00	0.00	0.00
1.30 + 17	6.65 + 23	3.95 + 22	3.24 + 21	2.38 + 19
2.56 + 17	3.61 + 23	2.15 + 22	1.81 + 21	2.18 + 19
2.86 + 17	1.96 + 23	1.17 + 22	9.82 + 20	1.51 + 19
2.55 + 17	1.07 + 23	6.33 + 21	5.33 + 20	9.43 + 18
1.46 + 17	3.14 + 22	1.87 + 21	1.58 + 20	3.20 + 18
1.11 + 16	8.06 + 20	4.79 + 19	4.38 + 18	9.05 + 16
1.51 + 15	7.01 + 19	4.16 + 18	5.66 + 17	7.94 + 15
1.89 + 14	6.10 + 18	3.62 + 17	1.62 + 17	6.92 + 14
2.24 + 13	5.30 + 17	3.15 + 16	8.26 + 16	6.02 + 13
8.67 + 11	1.36 + 16	8.10 + 14	3.79 + 16	1.55 + 12
3.21 + 10	3.70 + 14	2.19 + 13	1.80 + 16	4.06 + 10
1.17 + 09	2.44 + 13	1.40 + 12	8.62 + 15	1.73 + 09
5.09 + 07	1.08 + 13	6.11 + 11	4.17 + 15	5.18 + 08
<u>Pu-238 Nuc/T</u>	<u>Pu-239 Nuc/T</u>	<u>Pu-240 Nuc/T</u>	<u>Pu-241 Nuc/T</u>	<u>Pu-242 Nuc/T</u>
4.23 + 23	1.33 + 25	5.42 + 24	2.52 + 24	8.76 + 23
1.68 + 23	3.21 + 23	2.59 + 24	9.83 + 23	2.77 + 24
8.62 + 22	1.02 + 22	6.50 + 23	2.63 + 23	3.13 + 24
4.66 + 22	1.59 + 21	1.57 + 23	6.38 + 22	2.93 + 24
2.58 + 22	7.70 + 20	3.83 + 22	1.55 + 22	2.62 + 24
8.29 + 21	2.26 + 20	3.27 + 21	1.20 + 21	2.04 + 24
6.70 + 20	6.28 + 18	1.51 + 21	4.38 + 20	9.56 + 23
2.92 + 20	8.08 + 17	1.50 + 21	4.38 + 20	5.77 + 23
1.57 + 20	2.29 + 17	1.34 + 21	3.93 + 20	3.49 + 23
8.86 + 19	1.15 + 17	1.13 + 21	3.31 + 20	2.12 + 23
3.77 + 19	5.22 + 16	8.14 + 20	2.39 + 20	1.00 + 23
1.61 + 19	2.45 + 16	5.58 + 20	1.64 + 20	4.79 + 22
6.89 + 18	1.16 + 16	3.71 + 20	1.09 + 20	2.31 + 22
2.95 + 18	5.58 + 15	2.42 + 20	7.12 + 19	1.13 + 22
<u>Pu-243 Nuc/T</u>	<u>Am-241 Nuc/T</u>	<u>Am-242 Nuc/T</u>	<u>Am-242m Nuc/T</u>	<u>Am-243 Nuc/T</u>
0.00	1.12 + 23	0.00	2.32 + 21	2.27 + 23
6.74 + 21	1.57 + 22	1.25 + 31	1.90 + 20	3.64 + 23
7.65 + 21	2.43 + 21	1.93 + 20	2.93 + 19	5.57 + 23
7.17 + 21	4.16 + 20	3.29 + 19	5.01 + 18	6.87 + 23
6.42 + 21	7.88 + 19	6.20 + 18	9.44 + 17	7.47 + 23
5.01 + 21	3.84 + 18	2.99 + 17	4.56 + 16	7.31 + 23
2.35 + 21	4.68 + 17	3.42 + 16	5.29 + 15	4.22 + 23
1.42 + 21	4.76 + 17	3.49 + 16	5.39 + 15	2.63 + 23
8.56 + 20	4.32 + 17	3.17 + 16	4.90 + 15	1.60 + 23
5.19 + 20	3.67 + 17	2.69 + 16	4.15 + 15	9.73 + 22
2.46 + 20	2.66 + 17	1.95 + 16	3.02 + 15	4.60 + 22
1.18 + 20	1.83 + 17	1.34 + 16	2.08 + 15	2.19 + 22
5.67 + 19	1.22 + 17	8.97 + 15	1.39 + 15	1.05 + 22
2.76 + 19	7.98 + 16	5.86 + 15	9.05 + 14	5.08 + 21

CASE 14. FUEL LOADING TRANSMUTATION FOR WALL LOADING OF 1 MW/m^2 ; D-T PLASMA (Cont)

<u>Am-244 Nuc/T</u>	<u>Cm-242 Nuc/T</u>	<u>Cm-243 Nuc/T</u>	<u>Cm-244 Nuc/T</u>	<u>Cm-245 Nuc/T</u>
0.00	1.3 + 22	2.70 + 20	7.73 + 22	4.65 + 21
4.20 + 21	5.42 + 22	9.07 + 20	1.49 + 23	1.17 + 21
6.47 + 21	5.39 + 22	1.03 + 21	2.69 + 23	2.14 + 21
8.03 + 21	4.78 + 22	9.37 + 20	4.28 + 23	3.45 + 21
8.75 + 21	4.17 + 22	8.20 + 20	6.01 + 23	4.91 + 21
8.59 + 21	3.14 + 22	6.18 + 20	9.24 + 23	7.67 + 21
4.97 + 21	1.34 + 22	2.64 + 20	1.38 + 24	1.16 + 22
3.09 + 21	7.58 + 21	1.49 + 20	1.34 + 24	1.14 + 22
1.89 + 21	4.30 + 21	8.46 + 19	1.18 + 24	1.00 + 22
1.14 + 21	2.43 + 21	4.79 + 19	9.84 + 23	8.38 + 21
5.42 + 20	1.04 + 21	2.05 + 19	7.03 + 23	5.99 + 21
2.58 + 20	4.43 + 20	8.73 + 18	4.80 + 23	4.09 + 21
1.24 + 20	1.90 + 20	3.73 + 18	3.18 + 23	2.71 + 21
5.98 + 19	8.12 + 19	1.60 + 18	2.07 + 23	1.77 + 21
<u>Cm-246 Nuc/T</u>	<u>Cm-247 Nuc/T</u>	<u>Cm-248 Nuc/T</u>	<u>F.P. Nuc/T</u>	
5.29 + 20	0.00	0.00	0.00	
2.21 + 21	1.23 + 19	1.18 + 18	1.55 + 25	
4.26 + 21	3.17 + 19	5.61 + 18	1.80 + 25	
7.75 + 21	6.40 + 19	1.52 + 19	1.87 + 25	
1.29 + 22	1.16 + 20	3.35 + 19	1.90 + 25	
2.88 + 22	2.97 + 20	1.15 + 20	1.93 + 25	
1.03 + 23	1.40 + 21	1.07 + 21	2.02 + 25	
1.57 + 23	2.33 + 21	2.52 + 21	2.07 + 25	
2.03 + 23	3.19 + 21	4.62 + 21	2.11 + 25	
2.40 + 23	3.91 + 21	7.22 + 21	2.15 + 25	
2.78 + 23	4.66 + 21	1.17 + 22	2.19 + 25	
2.96 + 23	5.07 + 21	1.65 + 22	2.21 + 25	
3.01 + 23	5.22 + 21	2.11 + 22	2.23 + 25	
2.98 + 23	5.20 + 21	2.53 + 22	2.25 + 25	

The output of the alchemy analysis consists of data for each isotope in a given chain in nuclei per ton of fresh fuel. It is necessary that this data be converted into a hazard index in order that the total hazard of the chain as a function of fluence be determined. The hazard index (H) used in this analysis is:

$$H = \frac{N_i \times \lambda_i \times 10^{-6}}{3.7 \times 10^4 \times \text{RCG}}$$

where H = hazard index

N_i = number of nuclei of isotope i per ton of fresh fuel

λ_i = decay constant for isotope i in sec

3.7×10^4 = conversion factor from disintegrations per sec to microcuries

RCG = radioactive concentration guide in microcuries per milliliter

10^{-6} = conversion factor from milliliters to cubic meters.

The RCG values used here are from 10 CFR 20, Appendix B, Table 2, Column 2, which is the measure of the allowable hazard to the general public from ingestion of these isotopes. An exception to this is Kr-85 which is evaluated on the basis of Table 1, Column 2. Thus, Kr-85 is evaluated based on the inhalation hazard.

Notice that the hazard index has the units of M^3 of water. This is the amount of water required to dilute the mixture to a point at which it can be released. The lower the value the less the hazard.

In the following table, the RCG and hazard index values used in this study are listed.

TABLE A-IV.1

HAZARD INDEX VALUES

Isotope	RCG $\left(\frac{\mu\text{C}}{\text{m}\ell}\right)$	Hazard Index
		$\left(\frac{\text{m}^3 \text{ of H}_2\text{O}}{\text{Nuc per ton of fuel}}\right)$
^{89}Sr	3×10^{-6}	0.14279×10^{-11}
^{90}Sr	3×10^{-7}	0.68757×10^{-13}
^{91}Sr	7×10^{-5}	0.76641×10^{-11}
^{88}Y	3×10^{-6}	0.67523×10^{-12}
^{90}Y	2×10^{-5}	0.40527×10^{-11}
^{91}Y	3×10^{-5}	0.12252×10^{-12}
^{92}Y	6×10^{-5}	0.24568×10^{-10}
^{133}Cs	-	
^{134}Cs	-	
^{135}Cs	8×10^{-4}	
^{136}Cs	9×10^{-5}	0.18531×10^{-13}
^{137}Cs	2×10^{-5}	0.99014×10^{-16}
^{138}Cs	3×10^{-6}	0.32324×10^{-8}
^{139}Ba	3×10^{-6}	0.12541×10^{-8}
^{84}Kr	9×10^{-6}	0.31441×10^{-13}
$^{85}\text{Kr}^*$	3×10^{-7}	0.18396×10^{-12}
^{128}I	-	0.41631×10^{-8}
^{129}I	-	0.12378×10^{-19}
^{130}I	-	0.14459×10^{-9}
^{238}Np	3×10^{-6}	0.34414×10^{-10}
^{239}Np	1×10^{-4}	0.92259×10^{-12}
^{236}Pu	3×10^{-6}	0.69489×10^{-13}
^{238}Pu	5×10^{-6}	0.13351×10^{-14}
^{239}Pu	5×10^{-6}	0.48773×10^{-17}
^{240}Pu	5×10^{-6}	0.17811×10^{-16}
^{241}Pu	2×10^{-4}	0.22851×10^{-16}
^{242}Pu	5×10^{-6}	0.31346×10^{-18}
^{243}Pu	3×10^{-4}	0.34829×10^{-11}

* ^{85}Kr is evaluated for inhalation hazard

TABLE A-IV.1 (Continued)

HAZARD INDEX VALUES

<u>Isotope</u>	RCG $\left(\frac{\mu\text{C}}{\text{ml}}\right)$	Hazard Index
		$\left(\frac{\text{m}^3 \text{ of H}_2\text{O}}{\text{Nuc per ton of fuel}}\right)$
^{241}Am	4×10^{-6}	0.32426×10^{-15}
$^{242\text{m}}\text{Am}$	4×10^{-6}	0.97703×10^{-15}
^{242}Am	4×10^{-4}	0.32516×10^{-11}
^{243}Am	4×10^{-6}	0.19412×10^{-16}
^{244}Am	5×10^{-3}	0.10405×10^{-12}
^{242}Cm	2×10^{-5}	0.66514×10^{-13}
^{243}Cm	7×10^{-6}	0.37130×10^{-14}
^{244}Cm	7×10^{-6}	0.46873×10^{-14}
^{245}Cm	4×10^{-6}	0.15932×10^{-6}
^{246}Cm	4×10^{-6}	0.27101×10^{-16}

Below are listed the cumulative hazard index values for the cases calculated in this analysis. They are in m^3 of H_2O per ton of fuel charged to the reactor. (Except for Case 5, ^{85}Kr , which is in m^3 of air.) These values were arrived at by multiplying each decaying isotope in the chain by its hazard index conversion value as given in Appendix IV and then summing the values for each chain at each time point. The time points used for each case are given in Appendix III. Thus, the values given here are the total hazard index values for each chain as a function of time and wall loading. Therefore, the values are the data from which the conclusions have been drawn for this analysis.

CASE 1 (m^3 of H_2O per Ton of Fuel)

Wall Loading Time Pt.	Decay Only	$P_w = 1 \frac{\text{MW}}{\text{m}^2}$	$P_w = 10 \frac{\text{MW}}{\text{m}^2}$
1	2.4986×10^{11}	2.4986×10^{11}	2.4986×10^{11}
2	2.6053×10^{11}	2.4855×10^{11}	2.7111×10^{11}
3	2.6594×10^{11}	2.3755×10^{11}	1.9577×10^{11}
4	2.6678×10^{11}	2.1648×10^{11}	9.7234×10^{10}
5	2.6287×10^{11}	1.9723×10^{11}	4.8008×10^{10}
6	2.5437×10^{11}	1.6350×10^{11}	1.1499×10^{10}
7	2.3705×10^{11}	1.2385×10^{11}	1.3878×10^9
8	2.2594×10^{11}	1.0282×10^{11}	3.3596×10^8
9	2.0362×10^{11}	6.4468×10^{10}	2.3194×10^7
10	1.8061×10^{11}	4.0548×10^{10}	5.5062×10^6
11	1.3970×10^{11}	1.5998×10^{10}	-
12	1.0984×10^{11}	6.3283×10^9	-
13	8.6039×10^{10}	2.4618×10^9	-
14	6.7913×10^{10}	1.0123×10^9	-

CASE 2* (m^3 of H_2O per Ton of Fuel)

1	5.3542×10^8	5.3542×10^8	5.3542×10^8
2	5.2927×10^8	-	-
3	5.2320×10^8	4.9321×10^8	4.4817×10^8
4	5.1126×10^8	-	-
5	4.9960×10^8	-	-
6	4.7689×10^8	-	-
7	4.4515×10^8	-	-
8	4.2507×10^8	-	-
9	3.7860×10^8	-	-
10	3.3746×10^8	2.4604×10^8	3.5717×10^7
11	2.6791×10^8	-	-
12	2.1269×10^8	1.1850×10^8	2.4914×10^6
13	1.6824×10^8	-	-
14	1.3405×10^8	5.7316×10^7	4.1596×10^5

*These values are after irradiation to the time point indicated and then subsequent decay for two years.

CASE 3 (m^3 of H_2O per Ton of Fuel)Wall
LoadingTime
Pt.

	Decay Only	$P_w = 1 \frac{\text{MW}}{\text{m}^2}$	$P_w = 10 \frac{\text{MW}}{\text{m}^2}$
1	2.4986×10^{11}	2.4986×10^{11}	2.4986×10^{11}
2	2.6053×10^{11}	2.5453×10^{11}	2.7427×10^{11}
3	2.6594×10^{11}	2.4393×10^{11}	1.9942×10^{11}
4	2.6678×10^{11}	2.2241×10^{11}	1.0100×10^{11}
5	2.6287×10^{11}	2.0267×10^{11}	5.1751×10^{10}
6	2.5437×10^{11}	1.6807×10^{11}	1.5093×10^{10}
7	2.3705×10^{11}	1.2741×10^{11}	4.7278×10^9
8	2.2594×10^{11}	1.0585×10^{11}	3.2759×10^9
9	2.0362×10^{11}	6.6517×10^{10}	2.7595×10^9
10	1.8061×10^{11}	4.1984×10^{10}	2.4259×10^9
11	1.3970×10^{11}	1.6183×10^{10}	1.8822×10^9
12	1.0984×10^{11}	6.8777×10^9	1.4629×10^9
13	8.6039×10^{10}	2.9034×10^9	1.1321×10^9
14	6.7913×10^{10}	1.4079×10^9	8.8090×10^8

CASE 4 (m^3 of H_2O Per Ton of Fuel)

1	2.3853×10^{10}	2.3853×10^{10}	2.3853×10^{10}
2	2.0302×10^{10}	-	-
3	1.7290×10^{10}	1.9222×10^9	9.1989×10^8
4	1.2568×10^{10}	-	-
5	9.1692×10^9	-	-
6	4.9361×10^9	-	-
7	2.1128×10^9	-	-
8	1.2868×10^9	-	-
9	5.4339×10^8	-	-
10	3.6931×10^8	3.4838×10^8	7.3401×10^7
11	2.6909×10^8	-	-
12	2.1273×10^8	1.6755×10^8	5.1361×10^6
13	1.6824×10^8	-	-
14	1.3405×10^8	8.0908×10^7	3.2451×10^5

CASE 5 (m^3 of Air Per Ton of Fuel)Wall
LoadingTime
Pt.

	Decay Only	$P_w = 1.0 \frac{\text{MW}}{\text{m}^2}$	$P_w = 10.0 \frac{\text{MW}}{\text{m}^2}$
1	3.6889×10^{10}	3.6889×10^{10}	3.6889×10^{10}
2	3.5723×10^{10}	3.3269×10^{10}	1.7672×10^{10}
3	3.4592×10^{10}	3.0009×10^{10}	8.6842×10^9
4	3.2437×10^{10}	2.4428×10^{10}	2.4940×10^9
5	3.0416×10^{10}	1.9899×10^{10}	8.5101×10^8
6	2.6717×10^{10}	1.3200×10^{10}	6.6385×10^8
7	2.2051×10^{10}	7.2668×10^9	5.4286×10^8
8	1.9389×10^{10}	4.9241×10^9	4.8546×10^8
9	1.4042×10^{10}	1.9862×10^9	3.7172×10^8
10	1.0191×10^{10}	9.5139×10^8	2.8779×10^8
11	5.356×10^9	4.3622×10^8	1.7393×10^8
12	2.8152×10^9	3.5026×10^8	1.0547×10^8
13	1.4646×10^9	3.1901×10^8	6.3495×10^7
14	7.7769×10^8	2.9713×10^8	3.8840×10^7

CASE 6 (m^3 of H_2O per Ton of Fuel)

1	1.3232×10^4	1.3232×10^4	1.3232×10^4
2	1.3232×10^4	4.2710×10^3	2.1674×10^{-1}
3	1.3232×10^4	1.3783×10^3	5.4407×10^{-2}
4	1.3232×10^4	1.4352×10^2	5.4407×10^{-2}
5	1.3232×10^4	1.5189×10^1	1.1595×10^{-2}
6	1.3232×10^4	6.8611×10^{-1}	4.1698×10^{-2}
7	1.3232×10^4	-	5.4407×10^{-2}
8	1.3232×10^4	-	-
9	1.3232×10^4	-	-
10	1.3232×10^4	-	-
11	1.3232×10^4	-	-
12	1.3232×10^4	-	-
13	1.3232×10^4	-	-
14	1.3232×10^4	-	-

CASE 7 (m^3 of H_2O per Ton of Fuel)Wall
LoadingTime
Pt.

Decay Only

$$P_w = 1.0 \frac{\text{MW}}{\text{m}^2}$$

$$P_w = 10.0 \frac{\text{MW}}{\text{m}^2}$$

1	2.4986×10^{11}	2.4986×10^{11}	2.4986×10^{11}
2	2.6053×10^{11}	2.5137×10^{11}	2.2594×10^{11}
3	2.6594×10^{11}	2.3345×10^{11}	1.1661×10^{11}
4	2.6678×10^{11}	1.9942×10^{11}	2.9798×10^{10}
5	2.6287×10^{11}	1.7024×10^{11}	7.6055×10^9
6	2.5437×10^{11}	1.2375×10^{11}	4.8899×10^8
7	2.3705×10^{11}	7.7175×10^{10}	2.7191×10^6
8	2.2594×10^{11}	5.6242×10^{10}	1.3115×10^6
9	2.0362×10^{11}	2.5434×10^{10}	1.3115×10^5
10	1.8061×10^{11}	1.1566×10^{10}	-
11	1.3970×10^{11}	2.3884×10^9	-
12	1.0984×10^{11}	4.9836×10^8	-
13	8.6039×10^{10}	1.0990×10^8	-
14	6.7913×10^{10}	3.4780×10^7	-

CASE 8 (m^3 of H_2O per Ton of Fuel)

1	5.3542×10^8	5.3542×10^8	5.3542×10^8
2	5.2927×10^8	5.2380×10^8	4.7702×10^8
3	5.2320×10^8	5.1214×10^8	4.2499×10^8
4	5.1126×10^8	4.9045×10^8	3.3734×10^8
5	4.9960×10^8	4.6940×10^8	2.6776×10^8
6	4.7689×10^8	4.2965×10^8	1.6804×10^8
7	4.4515×10^8	3.7699×10^8	8.4354×10^7
8	4.2507×10^8	3.4534×10^8	5.3273×10^7
9	3.7860×10^8	2.7702×10^8	1.6756×10^7
10	3.3746×10^8	2.2287×10^8	5.4415×10^6
11	2.6791×10^8	1.4364×10^8	5.6363×10^5
12	2.1269×10^8	9.3087×10^7	2.2546×10^5
13	1.6824×10^8	5.9722×10^7	4.3550×10^4
14	1.3405×10^8	3.8616×10^7	4.0497×10^4

CASE 9 (m^3 of H_2O per Ton of Fuel)Wall
LoadingTime
Pt.

	Decay Only	$P_w = 1.0 \frac{\text{MW}}{\text{m}^2}$	$P_w = 10.0 \frac{\text{MW}}{\text{m}^2}$
1	4.1265×10^8	4.1265×10^8	4.1265×10^8
2	-	7.4263×10^8	3.2761×10^8
3	-	8.3732×10^8	1.6219×10^8
4	-	8.6591×10^8	7.7809×10^7
5	-	8.6789×10^9	3.7004×10^7
6	-	8.2772×10^8	8.3730×10^6
7	-	5.9721×10^8	2.5149×10^5
8	-	4.5507×10^8	1.12148×10^5
9	-	3.4194×10^8	-
10	-	2.5508×10^8	-
11	-	1.6303×10^8	-
12	-	1.0356×10^8	-
13	-	6.5554×10^7	-
14	3.8547×10^8	4.1438×10^7	-

CASE 10 (m^3 of H_2O per Ton of Fuel)

1	4.1265×10^8	4.1265×10^8	4.1265×10^8
2	-	7.1538×10^8	1.2044×10^8
3	-	7.3956×10^8	2.6259×10^7
4	-	7.0092×10^8	5.7643×10^6
5	-	6.3733×10^8	1.3679×10^6
6	-	4.9775×10^8	1.8381×10^5
7	-	2.0887×10^8	3.2703×10^4
8	-	1.1368×10^8	3.2703×10^4
9	-	6.1297×10^7	-
10	-	3.2920×10^7	-
11	-	1.2996×10^7	-
12	-	5.2469×10^6	-
13	-	2.2376×10^6	-
14	3.8547×10^8	1.0574×10^6	-

NOTE: Cases 9-14 include the hazard from only those isotopes with half lives longer than 1 yr. Everything else is assumed to have decayed away.

Case 11 (m^3 of H_2O per
Ton of Fuel)Case 12 (m^3 of H_2O per
Ton of Fuel)Wall
LoadingTime
Pt.

$$P_w = 1.0 \text{ MW/m}^2$$

$$P_w = 1.0 \frac{\text{MW}}{\text{m}^2}$$

1	0.0	4.2902×10^7
2	1.4127×10^8	2.6425×10^8
3	1.0227×10^8	4.2292×10^8
4	5.9576×10^7	5.1490×10^8
5	3.3313×10^7	5.6117×10^8
6	1.1048×10^7	5.7447×10^8
7	3.9111×10^6	4.2467×10^8
8	4.1369×10^6	3.2037×10^8
9	3.9215×10^6	2.3814×10^8
10	3.4201×10^6	1.7598×10^9
11	2.5445×10^6	1.1123×10^8
12	1.7781×10^6	7.0073×10^7
13	1.1980×10^6	4.4087×10^7
14	7.8904×10^5	2.7759×10^7

Case 13 (m^3 of H_2O per
ton of Fuel)Case 14 (m^3 of H_2O per
Ton of Fuel)

1	1.3014×10^9	1.7075×10^9
2	4.7333×10^8	1.2136×10^9
3	6.3304×10^8	1.4672×10^9
4	1.2438×10^9	2.1035×10^9
5	2.0166×10^9	2.8744×10^9
6	3.5500×10^9	4.3600×10^9
7	5.9006×10^9	6.4684×10^9
8	5.8622×10^9	6.2880×10^9
9	5.2295×10^9	5.5452×10^9
10	4.3899×10^9	4.6231×10^9
11	3.1577×10^9	3.3049×10^9
12	2.1644×10^9	2.2572×10^9
13	1.4410×10^9	1.4994×10^9
14	9.4186×10^8	9.7860×10^8

APPENDIX VI

ANALYTICAL MODEL DEVELOPMENT

Several computer codes were used in this analysis including:

- ANISN
- ALCHEMY
- SOLWHO
- CLAPS
- TRUE

The first two codes are adequately described in the literature. The latter three codes, however, were written for this study and, as such, are not described in previous literature. Therefore, a brief description of each code follows.

TRUE is an auxiliary program that converts isotopic number densities calculated at a series of time intervals into hazard index values. The proper hazard index factor is multiplied with each number density and a hazard array is printed out in a format that is analogous to the number density array. In addition, the total hazard index from all isotopes is summed for each time step and printed out.

The $(n,2n)$ and $(n,3n)$ cross sections were computed using the SOLWHO code which employs the formalism of Pearlstein⁽⁹⁾ based on the statistical model of nuclear reactions. Specifically, Eqs. 1 through 6 of Reference (9) were programmed, and the average spin values were input as 2_{j+1} values from Table I of Ref. (9). Above the $(n,3n)$ threshold the calculated $(n,2n)$ cross sections were modified by subtraction of the $(n,2n)$ cross sections as per the discussion on p. 245 of Ref. (9). Cross section values were calculated for the midpoint energies of the 100-group GAM-II structure and these values were used as average values in the 100-group format.

Effective one-group cross sections for the (n,2n) and (n,3n) reactions were calculated using the program CLAPS. This algorithm used the 99-group epithermal flux from the ANISN calculation to calculate the flux weighted average of the multigroup cross sections produced by the program SOLWHO.

APPENDIX VII

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Listed below are the names, organizations and functions of the task force members who contributed to the development of this study.

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