AEC
RESEARCH
and
DEVELOPMENT
REPORT

BNWL-54

C-30-

**PYRE - A MULTIGROUP BURNUP CODE
FOR FAST REACTORS**

APRIL, 1965

BATTELLE-NORTHWEST
RICHLAND, WASHINGTON

PACIFIC NORTHWEST LABORATORY operated by BATTELLE MEMORIAL INSTITUTE

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

PACIFIC NORTHWEST LABORATORY

RICHLAND, WASHINGTON

Operated By

BATTELLE MEMORIAL INSTITUTE

For The

U. S. ATOMIC ENERGY COMMISSION Under Contract No. AT(45-1)-1830

PRINTED BY/FOR THE U. S. ATOMIC ENERGY COMMISSION

3 3679 00060 1114

BNWL-54

UC-34

PYRE - A MULTIGROUP BURNUP CODE
FOR FAST REACTORS

by

W. W. Little, Jr.

R. W. Hardie

April, 1965

Pacific Northwest Laboratory

Richland, Washington

Printed by/for the U.S. Atomic Energy Commission

SUMMARY

This report describes a zero dimensional burnup code capable of handling up to 26 energy groups and 40 isotopes (20 burnable isotopes). The running time is approximately 20 seconds for a typical fast reactor burnup calculation. Input information, a FORTRAN IV source deck listing, and a sample problem are given in the appendices.

I. INTRODUCTION

In survey reactor analysis, it is often desirable to have a rapid means of calculating reactor burnup. The code described in this report is designed to compute one region fuel burnup in fast spectrum reactors. Twenty-six energy groups can be used, with up to 10 downscattering transfers. The principal options available in the code are:

- 1) Burnup can be computed at constant power or constant flux;
- 2) Spectrum averaged cross sections are calculated using the spectrum in a critical reactor (adjusting B^2), or the spectrum at $k(t)$;
- 3) Neutronics format:
 - a) k_{eff} as a function of time for a given initial k_{eff} (adjusting B^2),
 - b) k_{eff} as a function of time for a specified buckling, (B^2),
 - c) B^2 as a function of time for a specified k_{eff} ;
- 4) Fission products can be burned out or simply accumulated;
- 5) Full or partial print. This option is demonstrated in Appendix II.

II. NEUTRON BALANCE

At each time step, the neutron balance is recomputed using a modification of the multigroup diffusion relationship

$$D_i \nabla^2 \phi_i - \Sigma_i^a \phi_i - \Sigma_i^r \phi_i + X_i \sum_{j=1}^G \left(\nu \Sigma^f \right)_j \phi_j + \sum_{j=1}^{i-1} \Sigma(j \rightarrow i) \phi_j = 0 \quad (1)$$

where:

G = number of energy groups,

X_i = fission spectrum contribution to group i ,

D_i = diffusion constant for group i ,

Σ_i^a = absorption cross section for group i ,

$(\nu \Sigma^f)_i$ = neutron source cross section for group i ,

$\Sigma(i \rightarrow j)$ = cross section for scattering from group i to group j ,

Σ_i^r = removal cross section for group i $\left(\Sigma_i^r = \sum_{j=i+1}^G \Sigma(i \rightarrow j) \right)$.

In order to make the problem tractable, the fundamental mode approximation,

$$D_i \nabla^2 \phi_i = - D_i B^2 \phi_i, \quad (2)$$

is employed to simulate neutron leakage. Placing Eq. (2) in (1), and rearranging terms, we find that

$$\phi_i = \frac{X_i \sum_{j=1}^G (\nu \Sigma^f)_j \phi_j + \sum_{j=1}^{i-1} \Sigma(j \rightarrow i) \phi_j}{D_i B^2 + \Sigma_i^a + \Sigma_i^r} \quad (3)$$

For convenience, we can normalize the fission source to unity, hence

$$\sum_{i=1}^G (\nu \Sigma^f)_i \phi_i = 1 . \quad (4)$$

Substituting Eq. (4) into (3), the final expression for flux becomes

$$\phi_i^{\eta+1} = \frac{X_i + \sum_{j=1}^{i-1} \Sigma(j \rightarrow i) \phi_j^{\eta}}{D_i B^2 + \Sigma_i^a + \Sigma_i^r} , \quad (5)$$

where η is the iteration index.

Note that for a given buckling, B^2 , Eq (5) can be solved explicitly for the multigroup flux.

The multiplication constant can now be obtained from the definition

$$k_{\text{eff}} = \frac{\sum_{i=1}^G (\nu \Sigma^f)_i \phi_i^{\eta+1}}{\sum_{i=1}^G (\nu \Sigma^f)_i \phi_i^{\eta}} , \quad (6)$$

or, from Eq. (4),

$$k_{\text{eff}} = \sum_{i=1}^G (\nu \Sigma^f)_i \phi_i^{\eta+1} . \quad (7)$$

For problems requiring the calculation of B^2 for a specified value of k_{eff} , the code iterates between Eqs. (5) and (7) to obtain the appropriate flux and buckling.

III. SPECTRUM AVERAGED CROSS SECTION

Before each time step, the spectrum averaged cross sections for capture, fission, and absorption are computed using the recipe

$$\bar{\sigma}^k = \frac{\sum_{i=1}^G \sigma_i^k \phi_i}{\sum_{i=1}^G \phi_i}, \quad (8)$$

where: σ_i^k = group i cross section for element k ,

$\bar{\sigma}^k$ = average cross section for element k . For this calculation, the code uses either the current flux distribution (corresponding to $k = k(t)$), or a critical flux distribution (corresponding to $k = 1$, adjusting B^2 for criticality).

IV. MARCH-OUT ALGORITHM

The basic burnup equation has the form

$$\begin{aligned} \frac{dN^i}{dt} = & -\lambda^i N^i - \bar{\sigma}_a^i \bar{\phi} N^i + \lambda^k N^k + S^i \\ & + \sum_{j=1}^4 \bar{\sigma}_c^j \bar{\phi} N^j + \sum_{m=1}^7 \bar{\sigma}_f^m \bar{\phi} N^m \end{aligned} \quad (9)$$

where: N^i = density of nuclide i ,

λ^i = decay constant of nuclide i ,

$\bar{\sigma}_a^i$ = spectrum averaged absorption cross section for nuclide i ,

$\bar{\phi}$ = total flux,

S^i = external source for nuclide i ,

$\lambda^k N^k$ = decay source from nuclide k .

The last two sum terms in Eq. (9) allow provision for four capture sources and seven fission sources. The latter, for example, would be used to compute the fission product buildup.

Eq. (9) is solved using the improved Euler algorithm. If we rewrite Eq. (9) as

$$\frac{d\vec{N}}{dt} = \vec{f}(\vec{N}, t), \quad (10)$$

the improved Euler algorithm can be written as

$$\vec{N}_{J+1} = \vec{N}_J + \frac{\delta T}{2} (\vec{f}_J + \vec{f}_{J+1}), \quad (11)$$

where J is the index on time, and δT is the time step.

Observe that Eq. (11) is implicit in the sense that $\vec{N}_{J+1}$ must be known in order to compute $\vec{f}_{J+1}$. One must therefore iterate on $\vec{N}$ at each time point. This procedure leads to the algorithm

$$\vec{N}_{J+1}^{v+1} = \vec{N}_J + \frac{\delta T}{2} \left[\vec{f}_J + \vec{f}_{J+1}^v \right], \quad (12)$$

where v is the iteration index.

The iteration is continued until the convergence criterion

$$\left| \frac{N_{J+1}^{v+1} - N_{J+1}^v}{N_{J+1}^v} \right| \leq \epsilon \quad (13)$$

is satisfied for all nuclides.

V. REMARKS ON CODE OPERATION

It should be clear from the mathematical model presented that relatively short time steps should be employed in order to minimize the discretization error. This error will grow roughly as the square of the time step. It has been found that typical fast reactor burnup calculations using 16 energy groups require about 20 seconds of IBM-7090 time.

APPENDIX I

INPUT INFORMATION

<u>Card</u>	<u>Column</u>	<u>Format</u>	<u>Remarks</u>
1	1-6	Fixed	Number of energy groups (≤ 26).
1	7-12	Fixed	Number of groups in downscattering matrix (≤ 10). For example, 5 for Hansen-Roach set, 6 for YOM set, 10 for Russian set.
1	13-18	Fixed	Number of burnable isotopes (≤ 20).
1	19-24	Fixed	Number of non-burnable isotopes (≤ 20).
1	25-30	Fixed	Print Option: = 0, full print; = 1, partial print.
2	1-6	Hollerith	Name of burnable isotope #1.
2	7-12	Floating	Atomic weight (a.m.u.).
2	13-18	Floating	Decay constant (days ⁻¹).
2	19-24	Floating	Energy released per fission (Mev/fission).
2	25-27	Fixed	= 0, no decay source; = n, where n is the isotope number of decay source.
2	28-30	Fixed	Number of capture sources (≤ 4).
2	31-33	Fixed	Capture source number (if any).
2	34-36	Fixed	Capture source number.
2	37-39	Fixed	Capture source number.
2	40-42	Fixed	Capture source number.
2	43-45	Fixed	Number of fission product sources (≤ 7).
2	46-48	Fixed	Fission product source number (if any).
2	49-51	Fixed	Fission product source number.
2	52-54	Fixed	Fission product source number.
2	55-57	Fixed	Fission product source number.
2	58-60	Fixed	Fission product source number.

<u>Card</u>	<u>Column</u>	<u>Format</u>	<u>Remarks</u>
2	61-63	Fixed	Fission product source number.
2	64-66	Fixed	Fission product source number.
3	Same as above for burnable isotope #2. Repeat for all burnable isotopes.		
4	1-6	Hollerith	Name of first non-burnable isotope.
4	7-12	Floating	Atomic weight (a.m.u.).
5	Same as above for next non-burnable element. Repeat for all non-burnable isotopes.		
6	1-12	Floating	Fission source in energy group 1.
6	13-24	Floating	Fission source in energy group 2.
...			
7	1-12	Floating	Lower energy bound for group 1.
7	13-24	Floating	Lower energy bound for group 2.
...			
8	1	Fixed	9
8	2-9	Floating	Cross section data for first group of first isotope. σ_f (barns)
8	10-18	Floating	σ_a
8	19-27	Floating	$\nu\sigma_f$
8	28-36	Floating	σ_{tr}
8	37-45	Floating	σ_{gg}
8	46-54	Floating	$\sigma_{g-1 \rightarrow g}$
8	55-63	Floating	$\sigma_{g-2 \rightarrow g}$
8	64-72	Floating	$\sigma_{g-3 \rightarrow g}$
9	1	Fixed	9
9	2-9	Floating	$\sigma_{g-4 \rightarrow g}$

<u>Card</u>	<u>Column</u>	<u>Format</u>	<u>Remarks</u>
9	10-18	Floating	σ_{g-5-g}
9	19-27	Floating	σ_{g-6-g}
9	28-36	Floating	σ_{g-7-g}
9	37-45	Floating	σ_{g-8-g}
9	46-54	Floating	σ_{g-9-g}
9	55-63	Floating	σ_{g-10-g}
10	Same as above for second group of isotope 1. Repeat cycle for all groups and isotopes.		
**11	2-72	Hollerith	Identification card
12	1-6	Fixed	<p>= 0, the code calculates the effective multiplication constant as a function of time for a given initial k_{eff};</p> <p>= 1, the code calculates the effective multiplication constant as a function of time for a given geometric buckling;</p> <p>=2, the code calculates the geometric buckling as a function of time for a specified k_{eff}.</p>
12	7-12	Fixed	<p>= 0, the calculation is performed for a constant power;</p> <p>= 1, the calculation is performed for a constant flux.</p>
12	13-18	Fixed	<p>= 0, spectrum averaged cross sections calculated at $k_{eff} = 1$;</p> <p>= 1, spectrum averaged cross sections calculated at appropriate k_{eff}.</p>
12	19-24	Fixed	<p>= 0, cost calculation;</p> <p>= 1, no cost calculation.</p>
12	25-30	Fixed	Number of time steps.
12	31-36	Fixed	<p>Number of intervals between prints (e.g.,</p> <p>= 1, print every time;</p> <p>= 2, print every other time).</p>

<u>Card</u>	<u>Column</u>	<u>Format</u>	<u>Remarks</u>
13	1-12	Floating	Length of time steps (days).
13	13-24	Floating	Volume of core (liters).
13	25-36	Floating	Power (Mwt) or flux ($n/(cm^2\text{-sec})$).
13	37-48	Floating	Buckling or effective multiplication constant.
13	49-60	Floating	Convergence criterion on march-out.
14	1-6	Fixed	Number of fertile isotopes (≤ 5). The data on this card is used to compute the breeding ratio only.
14	7-12	Fixed	Nuclide number of fertile isotope #1.
14	13-18	Fixed	Nuclide number of fertile isotope #2.
14	19-24	Fixed	Nuclide number of fertile isotope #3.
14	25-30	Fixed	Nuclide number of fertile isotope #4.
14	31-36	Fixed	Nuclide number of fertile isotope #5.
14	37-42	Fixed	Number of fissile isotopes (≤ 5).
14	43-48	Fixed	Nuclide number of fissile isotope #1.
14	49-54	Fixed	Nuclide number of fissile isotope #2.
14	55-60	Fixed	Nuclide number of fissile isotope #3.
14	61-66	Fixed	Nuclide number of fissile isotope #4.
14	67-72	Fixed	Nuclide number of fissile isotope #5.
15	1-12	Floating	Atom density for isotope #1 ($cm^{-3} \times 10^{24}$).
15	13-24	Floating	Atom density for isotope #2.
...			
16	1-12	Floating	Rate at which isotope #1 is being added to reactor (kg/day). List data for burnable isotopes only.
16	13-24	Floating	Rate at which isotope #2 is being added to reactor.
...			

**To run a series of cases, repeat from card 11.

APPENDIX II

SAMPLE CASES

Two sample cases are presented in this section to demonstrate the input and output organization to the reader. In these cases, the fuel burnup in a sodium cooled fast reactor fueled with PuO_2 is computed over a 100 day interval.

The first two pages give the input data for the sample calculations. For brevity, only one cross section set is included. The code output is listed on the subsequent pages.

10	6	6	5	1													
PO24239.05	.0	211.	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
PO240240.05	.0	211.	0	1	1	0	0	0	0	0	0	0	0	0	0	0	0
PO241241.061.5e-4		211.	0	1	2	0	0	0	0	0	0	0	0	0	0	0	0
PO242242.06	.0	211.	0	1	3	0	0	0	0	0	0	0	0	0	0	0	0
FISPR 1.	.0	.0	0	1	6	0	0	0	4	1	2	3	4	0	0	0	0
FISPR 1.	.0	.0	0	1	5	0	0	0	4	1	2	3	4	0	0	0	0

OXYGEN 15.999

NA 22.990

IRON 55.847

NICKEL 58.71

CHROM 51.996

.132	.213	.232	.179	.116	.067
.034	.017	.01	.0	.0	.0
.0	.0	.0	.0	.0	.0

3.668	2.225	1.35	0.825	0.5	0.3
0.18	0.11	0.067	0.0407	0.025	0.015
0.0091	0.0055	0.0021	0.0005		

9	1.830	1.884	6.313	3.970	0.676	0.	0.	0.	00070405
9	0.	0.	0.						00070505
9	1.990	2.070	6.408	4.991	1.636	0.082	0.	0.	00070605
9	0.	0.	0.						00070705
9	1.960	2.057	6.076	5.140	2.037	0.146	0.238	0.	00070805
9	0.	0.	0.						00070905
9	1.810	1.914	5.466	5.070	2.606	0.241	0.274	0.358	00071005
9	0.	0.	0.						00071105
9	1.700	1.829	5.049	6.090	3.873	0.336	0.323	0.338	00071205
9	0.336	0.	0.						00071305
9	1.700	1.882	4.998	7.270	5.104	0.314	0.082	0.203	00071405
9	0.264	0.213	0.						00071505
9	1.700	1.990	4.964	8.660	6.458	0.225	0.050	0.051	00071605
9	0.112	0.150	0.183						00071705
9	1.700	2.109	4.947	10.203	7.901	0.207	0.033	0.024	00071805
9	0.040	0.054	0.113						00071905
9	1.800	2.334	5.238	11.502	9.049	0.178	0.005	0.012	00072005
9	0.	0.042	0.113						00072105
9	1.930	2.616	5.597	12.703	9.961	0.119	0.014	0.	00072205
9	0.007	0.	0.						00072305
9	2.170	3.062	6.293	13.303	10.604	0.126	0.	0.001	00072405
9	0.	0.004	0.						00072505
9	2.400	3.484	6.960	14.601	10.983	0.137	0.	0.	00072605
9	0.	0.	0.003						00072705
9	2.450	3.661	7.105	15.207	11.406	0.134	0.	0.	00072805
9	0.	0.	0.001						00072905
9	2.410	3.690	6.989	16.103	12.265	0.140	0.	0.	00073005
9	0.	0.	0.						00073105
9	3.280	5.171	9.512	18.008	12.734	0.148	0.	0.	00073205

0 8.000 10.020 17.400 13.501 8.481 0.000 0.0 0007400
 0 0.0 0.0 0.0 0.0 0.0 0.0 0.0 0007400

SAMPLE CASE 1 FULL PRINT Y O M CROSS SECTIONS

0 0 0 0 100 50
 1.0 206. 200. 1.05 .000001
 1 2 2 1 3
 2.13183E-031.604610E-04 0.0 0.0 0.0 0.0
 4.58459E-031.142250E-02 .0259 .002665 .00676
 0.0 0.0 0.0 0.0 0.0 0.0

SAMPLE CASE 2 PARTIAL PRINT Y O M CROSS SECTIONS

0 0 0 0 100 50
 1.0 206. 200. 1.05 .000001
 1 2 2 1 3
 2.13183E-031.604610E-04 0.0 0.0 0.0 0.0
 4.58459E-031.142250E-02 .0259 .002665 .00676
 0.0 0.0 0.0 0.0 0.0 0.0

***** ZERO DIMENSIONAL FUEL CYCLE CALCULATION *****

INPUT DATA LISTED BELOW

NUCLIDE NUMBER	NAME	ATM WEIGHT	LAMBDA(DAYS ⁻¹)	MEV/FISSION	SOURCE ELEMENT FOR DECAY	CAPTURE	FISSION
1	PU239	2.3905E 02	-0.	2.1100E 02	0 0 0		
2	PU240	2.4005E 02	-0.	2.1100E 02	0 1 0		
3	PU241	2.4106E 02	1.5000E-04	2.1100E 02	0 2 0		
4	PU242	2.4206E 02	-0.	2.1100E 02	0 3 0		
5	FISPR	1.0000E-00	-0.	-0.	0 6 1 2 3 4		
6	FISPR	1.0000E-00	-0.	-0.	0 5 1 2 3 4		

NON-BURNABLE NUCLIDES

7	OXYGN	1.5999E 01					
8	NA	2.2990E 01					
9	IRON	5.5847E 01					
10	NICKL	5.8710E 01					

GROUP	FISSION SOURCE	ENERGY LIMIT
-------	----------------	--------------

1	1,32000E-01	3,66800E 00
2	2,13000E-01	2,22500E 00
3	2,32000E-01	1,35000E 00
4	1,79000E-01	8,25000E-01
5	1,16000E-01	5,00000E-01
6	6,70000E-02	3,00000E-01
7	3,40000E-02	1,80000E-01
8	1,70000E-02	1,10000E-01
9	1,00000E-02	6,70000E-02
10	-0,	4,07000E-02
11	-0,	2,50000E-02
12	-0,	1,50000E-02
13	-0,	9,10000E-03
14	-0,	5,50000E-03
15	-0,	2,10000E-03
16	-0,	5,00000E-04

MICROSCOPIC CROSS SECTIONS FOR PU239

NUCLIDE NUMBER 1

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGKG	G=1XG	G=2XG	G=3XG	G=4XG	G=5XG	G=6XG
1	1.830	1.884	6.313	3.970	0.676	0.	0.	0.	0.	0.	0.
2	1.990	2.070	6.408	4.991	1.636	0.082	0.	0.	0.	0.	0.
3	1.960	2.057	6.076	5.140	2.037	0.146	0.238	0.	0.	0.	0.
4	1.810	1.914	5.466	5.070	2.606	0.241	0.274	0.358	0.	0.	0.
5	1.700	1.829	5.049	6.090	3.873	0.336	0.323	0.338	0.336	0.	0.
6	1.700	1.882	4.998	7.270	5.104	0.314	0.082	0.203	0.264	0.213	0.
7	1.700	1.990	4.964	8.660	6.458	0.225	0.050	0.051	0.112	0.150	0.183
8	1.700	2.109	4.947	10.203	7.901	0.207	0.033	0.024	0.040	0.054	0.113
9	1.800	2.334	5.238	11.502	9.049	0.178	0.005	0.012	0.	0.042	0.113
10	1.930	2.616	5.597	12.703	9.961	0.119	0.014	0.	0.007	0.	0.
11	2.170	3.062	6.293	13.803	10.604	0.126	0.	0.001	0.	0.004	0.
12	2.400	3.484	6.960	14.601	10.983	0.137	0.	0.	0.	0.	0.003
13	2.450	3.661	7.105	15.207	11.406	0.134	0.	0.	0.	0.	0.001
14	2.410	3.690	6.989	16.103	12.265	0.140	0.	0.	0.	0.	0.
15	3.280	5.171	9.512	18.008	12.784	0.148	0.	0.	0.	0.	0.
16	6.000	10.020	17.400	13.501	3.481	0.053	0.	0.	0.	0.	0.

MICROSCOPIC CROSS SECTIONS FOR PU240

NUCLIDE NUMBER 2

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G=1XG	G=2XG	G=3XG	G=4XG	G=5XG	G=6XG
1	1.540	1.558	5.929	3.900	0.849	0.	0.	0.	0.	0.	0.
2	1.630	1.669	5.721	4.630	1.570	0.087	0.	0.	0.	0.	0.
3	1.590	1.690	5.279	4.870	2.298	0.151	0.252	0.	0.	0.	0.
4	1.390	1.616	4.462	5.010	3.113	0.352	0.299	0.379	0.	0.	0.
5	0.760	1.003	2.394	5.591	4.310	0.183	0.393	0.368	0.356	0.	0.
6	0.191	0.413	0.594	6.881	6.067	0.278	0.072	0.095	0.288	0.225	0.
7	0.041	0.323	0.126	8.482	7.754	0.401	0.	0.024	0.013	0.163	0.194
8	0.	0.397	0.	10.101	9.441	0.405	0.	0.	0.002	0.029	0.122
9	0.	0.508	0.	11.502	10.854	0.204	0.	0.	0.	0.	0.
10	0.	0.648	0.	12.503	11.704	0.140	0.059	0.	0.	0.	0.
11	0.	0.788	0.	13.605	12.648	0.151	0.	0.	0.	0.	0.
12	0.	0.910	0.	14.306	13.230	0.169	0.	0.	0.	0.	0.
13	0.	1.054	0.	15.001	13.773	0.166	0.	0.	0.	0.	0.
14	0.	1.155	0.	15.103	13.774	0.174	0.	0.	0.	0.	0.
15	0.	1.435	0.	16.003	14.505	0.174	0.	0.	0.	0.	0.
16	0.	3.500	0.	14.006	10.506	0.063	0.	0.	0.	0.	0.

MICROSCOPIC CROSS SECTIONS FOR PU241

NUCLIDE NUMBER 3

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G=1XG	G=2XG	G=3XG	G=4XG	G=5XG	G=6XG
1	1.650	1.663	5.824	4.190	0.798	0.	0.	0.	0.	0.	0.
2	1.830	1.853	6.057	4.730	1.382	0.098	0.	0.	0.	0.	0.
3	1.890	1.929	6.010	4.830	1.715	0.159	0.292	0.	0.	0.	0.
4	1.830	1.907	5.673	4.730	2.232	0.255	0.322	0.440	0.	0.	0.
5	1.700	1.829	5.202	5.591	3.310	0.209	0.440	0.396	0.413	0.	0.
6	1.830	1.990	5.545	7.071	4.834	0.209	0.138	0.165	0.310	0.261	0.
7	2.050	2.247	6.171	9.170	6.729	0.222	0.118	0.093	0.134	0.176	0.225
8	2.270	2.517	6.810	10.763	8.014	0.186	0.025	0.052	0.040	0.058	0.132
9	2.750	3.072	8.223	12.401	9.110	0.202	0.008	0.	0.041	0.089	0.134
10	3.130	3.546	9.359	13.633	9.961	0.184	0.024	0.	0.	0.032	0.022
11	3.700	4.235	11.026	14.821	10.449	0.126	0.022	0.006	0.	0.	0.
12	4.430	5.099	13.201	16.033	10.800	0.137	0.	0.013	0.	0.	0.
13	5.290	6.123	15.764	17.637	11.374	0.134	0.	0.	0.	0.	0.
14	5.990	7.116	17.850	19.585	12.321	0.140	0.	0.	0.	0.	0.
15	6.910	8.305	20.592	21.110	12.752	0.148	0.	0.	0.	0.	0.
16	9.000	11.550	26.820	20.500	8.950	0.053	0.	0.	0.	0.	0.

MICROSCOPIC CROSS SECTIONS FOR PU242

NUCLIDE NUMBER 4

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G=1XG	G=2XG	G=3XG	G=4XG	G=5XG	G=6XG
1	1.540	1.558	5.929	3.900	0.849	0.	0.	0.	0.	0.	0.
2	1.370	1.409	4.809	4.630	1.573	0.087	0.	0.	0.	0.	0.
3	1.450	1.550	4.814	4.870	2.325	0.175	0.252	0.	0.	0.	0.
4	1.570	1.796	5.040	5.010	3.019	0.394	0.355	0.379	0.	0.	0.
5	0.660	0.903	2.079	5.591	4.380	0.132	0.444	0.437	0.356	0.	0.
6	0.179	0.401	0.555	6.881	6.074	0.308	0.047	0.108	0.342	0.225	0.
7	0.053	0.335	0.163	8.482	7.748	0.406	0.	0.015	0.014	0.194	0.194
8	0.023	0.420	0.070	10.101	9.433	0.399	0.	0.	0.001	0.035	0.145
9	0.	0.508	0.	11.502	10.854	0.196	0.	0.	0.	0.	0.
10	0.	0.648	0.	12.503	11.704	0.140	0.052	0.	0.	0.	0.
11	0.	0.788	0.	13.605	12.648	0.151	0.	0.	0.	0.	0.
12	0.	0.910	0.	14.306	13.230	0.169	0.	0.	0.	0.	0.
13	0.	1.054	0.	15.001	13.773	0.166	0.	0.	0.	0.	0.
14	0.	1.155	0.	15.103	13.774	0.174	0.	0.	0.	0.	0.
15	0.	1.435	0.	16.003	14.505	0.174	0.	0.	0.	0.	0.
16	0.	3.500	0.	14.006	10.506	0.063	0.	0.	0.	0.	0.

MICROSCOPIC CROSS SECTIONS FOR FISPR NUCLIDE NUMBER 5

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G=1XG	G=2XG	G=3XG	G=4XG	G=5XG	G=6XG
1	0.	0.020	0.	2.753	0.619	0.	0.	0.	0.	0.	0.
2	0.	0.040	0.	2.980	0.980	0.361	0.	0.	0.	0.	0.
3	0.	0.063	0.	3.406	1.807	0.477	0.557	0.	0.	0.	0.
4	0.	0.090	0.	4.142	3.043	0.502	0.553	0.521	0.	0.	0.
5	0.	0.092	0.	5.546	5.068	0.552	0.447	0.447	0.351	0.	0.
6	0.	0.095	0.	6.371	5.836	0.323	0.278	0.309	0.268	0.187	0.
7	0.	0.110	0.	7.652	7.212	0.362	0.026	0.110	0.163	0.129	0.137
8	0.	0.125	0.	8.227	7.813	0.270	0.042	0.033	0.016	0.072	0.086
9	0.	0.140	0.	7.995	7.598	0.289	0.038	0.036	0.004	0.053	0.043
10	0.	0.165	0.	7.626	7.225	0.257	0.	0.022	0.	0.	0.
11	0.	0.200	0.	7.411	6.975	0.236	0.	0.	0.	0.	0.
12	0.	0.250	0.	7.050	6.589	0.236	0.	0.	0.	0.	0.
13	0.	0.330	0.	6.970	6.431	0.211	0.	0.	0.	0.	0.
14	0.	0.450	0.	6.820	6.167	0.209	0.	0.	0.	0.	0.
15	0.	0.690	0.	6.830	6.072	0.203	0.	0.	0.	0.	0.
16	0.	1.540	0.	7.561	6.021	0.069	0.	0.	0.	0.	0.

MICROSCOPIC CROSS SECTIONS FOR FISPR

NUCLIDE NUMBER 6

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG	G-6XG
1	0.	0.020	0.	2,753	0,619	0.	0.	0.	0.	0.	0.
2	0.	0.040	0.	2,980	0,980	0,361	0.	0.	0.	0.	0.
3	0.	0.063	0.	3,406	1,807	0,477	0,557	0.	0.	0.	0.
4	0.	0.090	0.	4,142	3,043	0,502	0,553	0,521	0.	0.	0.
5	0.	0.092	0.	5,546	5,068	0,552	0,447	0,447	0,351	0.	0.
6	0.	0.095	0.	6,371	5,836	0,323	0,278	0,309	0,268	0,187	0.
7	0.	0.110	0.	7,652	7,212	0,362	0,026	0,110	0,163	0,129	0,137
8	0.	0.125	0.	8,227	7,813	0,270	0,042	0,033	0,016	0,072	0,086
9	0.	0.140	0.	7,995	7,598	0,289	0,038	0,036	0,004	0,053	0,043
10	0.	0.165	0.	7,626	7,225	0,257	0.	0,022	0.	0.	0.
11	0.	0.200	0.	7,411	6,975	0,236	0.	0.	0.	0.	0.
12	0.	0.250	0.	7,050	6,589	0,236	0.	0.	0.	0.	0.
13	0.	0.330	0.	6,970	6,431	0,211	0.	0.	0.	0.	0.
14	0.	0.450	0.	6,820	6,167	0,209	0.	0.	0.	0.	0.
15	0.	0.690	0.	6,830	6,072	0,203	0.	0.	0.	0.	0.
16	0.	1,540	0.	7,561	6,021	0,069	0.	0.	0.	0.	0.

MICROSCOPIC CROSS SECTIONS FOR OXYGN

NUCLIDE NUMBER 7

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG	G-6XG
1	0.	0.052	0.	1.140	0.733	0.	0.	0.	0.	0.	0.
2	0.	0.	0.	1.720	1.290	0.342	0.	0.	0.	0.	0.
3	0.	0.	0.	2.560	2.001	0.430	0.011	0.	0.	0.	0.
4	0.	0.	0.	3.510	2.717	0.559	0.	0.002	0.	0.	0.
5	0.	0.	0.	3.470	2.673	0.793	0.	0.	0.	0.	0.
6	0.	0.	0.	4.090	3.176	0.797	0.	0.	0.	0.	0.
7	0.	0.	0.	3.350	2.635	0.914	0.	0.	0.	0.	0.
8	0.	0.	0.	3.350	2.640	0.715	0.	0.	0.	0.	0.
9	0.	0.	0.	3.350	2.699	0.710	0.	0.	0.	0.	0.
10	0.	0.	0.	3.350	2.724	0.651	0.	0.	0.	0.	0.
11	0.	0.	0.	3.350	2.707	0.626	0.	0.	0.	0.	0.
12	0.	0.	0.	3.350	2.745	0.643	0.	0.	0.	0.	0.
13	0.	0.	0.	3.350	2.745	0.605	0.	0.	0.	0.	0.
14	0.	0.	0.	3.540	2.901	0.605	0.	0.	0.	0.	0.
15	0.	0.	0.	3.640	3.413	0.639	0.	0.	0.	0.	0.
16	0.	0.	0.	3.640	3.640	0.227	0.	0.	0.	0.	0.

MICROSCOPIC CROSS SECTIONS FOR NA NUCLIDE NUMBER 8

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGKG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG	G-6XG
1	0.	0.001	0.	1.300	0.688	0.	0.	0.	0.	0.	0.
2	0.	0.	0.	1.470	0.974	0.303	0.	0.	0.	0.	0.
3	0.	0.	0.	1.960	1.474	0.392	0.141	0.	0.	0.	0.
4	0.	0.	0.	2.890	2.097	0.486	0.029	0.084	0.	0.	0.
5	0.	0.	0.	3.930	3.100	0.658	0.	0.038	0.046	0.	0.
6	0.	0.	0.	3.220	2.720	0.721	0.135	0.	0.016	0.027	0.
7	0.	0.001	0.	3.490	2.971	0.500	0.056	0.	0.	0.014	0.011
8	0.	0.	0.	3.100	2.642	0.518	0.	0.030	0.	0.	0.007
9	0.	0.	0.	3.460	2.992	0.458	0.	0.	0.012	0.	0.
10	0.	0.003	0.	4.770	4.148	0.468	0.	0.	0.	0.011	0.
11	0.	0.	0.	3.880	3.362	0.619	0.	0.	0.	0.	0.
12	0.	0.	0.	4.100	3.585	0.518	0.	0.	0.	0.	0.
13	0.	0.	0.	5.010	4.381	0.515	0.	0.	0.	0.	0.
14	0.	0.	0.	7.231	6.324	0.629	0.	0.	0.	0.	0.
15	0.	0.377	0.	22.815	21.469	0.907	0.	0.	0.	0.	0.
16	0.	0.	0.	5.930	5.930	0.969	0.	0.	0.	0.	0.

MICROSCOPIC CROSS SECTIONS FOR IRON

NUCLIDE NUMBER 9

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG	G-6XG
1	0.	0.006	0.	1.990	0.680	0.	0.	0.	0.	0.	0.
2	0.	0.001	0.	2.190	1.199	0.362	0.	0.	0.	0.	0.
3	0.	0.002	0.	2.010	1.281	0.689	0.369	0.	0.	0.	0.
4	0.	0.003	0.	1.980	1.541	0.471	0.118	0.277	0.	0.	0.
5	0.	0.004	0.	2.300	2.145	0.181	0.256	0.086	0.162	0.	0.
6	0.	0.005	0.	2.970	2.775	0.151	0.076	0.	0.058	0.080	0.
7	0.	0.006	0.	2.420	2.266	0.190	0.	0.093	0.	0.007	0.054
8	0.	0.008	0.	2.890	2.706	0.148	0.	0.	0.056	0.	0.032
9	0.	0.010	0.	6.020	5.675	0.176	0.	0.	0.	0.030	0.
10	0.	0.010	0.	4.360	4.117	0.335	0.	0.	0.	0.	0.
11	0.	0.020	0.	12.513	11.806	0.233	0.	0.	0.	0.	0.
12	0.	0.005	0.	2.240	2.120	0.687	0.	0.	0.	0.	0.
13	0.	0.014	0.	5.660	5.353	0.116	0.	0.	0.	0.	0.
14	0.	0.013	0.	8.790	8.323	0.293	0.	0.	0.	0.	0.
15	0.	0.003	0.	5.761	5.655	0.454	0.	0.	0.	0.	0.
16	0.	0.020	0.	8.001	7.981	0.103	0.	0.	0.	0.	0.

MICROSCOPIC CROSS SECTIONS FOR NICKL

NUCLIDE NUMBER 10

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG	G-6XG
1	0.	0.002	0.	2.100	0.796	0.	0.	0.	0.	0.	0.
2	0.	0.003	0.	2.341	1.411	0.362	0.	0.	0.	0.	0.
3	0.	0.005	0.	2.671	2.300	0.349	0.364	0.	0.	0.	0.
4	0.	0.007	0.	2.971	2.787	0.143	0.251	0.277	0.	0.	0.
5	0.	0.007	0.	3.169	2.968	0.176	0.103	0.170	0.164	0.	0.
6	0.	0.007	0.	3.858	3.616	0.194	0.	0.054	0.090	0.081	0.
7	0.	0.009	0.	5.192	4.881	0.235	0.	0.	0.049	0.041	0.055
8	0.	0.010	0.	5.984	5.628	0.302	0.	0.	0.	0.008	0.026
9	0.	0.013	0.	7.825	7.398	0.346	0.	0.	0.	0.	0.009
10	0.	0.013	0.	7.474	7.080	0.414	0.	0.	0.	0.	0.
11	0.	0.014	0.	11.038	10.447	0.381	0.	0.	0.	0.	0.
12	0.	0.035	0.	31.746	30.148	0.577	0.	0.	0.	0.	0.
13	0.	0.016	0.	27.548	26.172	1.563	0.	0.	0.	0.	0.
14	0.	0.011	0.	14.881	14.137	1.360	0.	0.	0.	0.	0.
15	0.	0.178	0.	23.809	23.226	0.733	0.	0.	0.	0.	0.
16	0.	0.040	0.	15.015	14.975	0.405	0.	0.	0.	0.	0.

MICROSCOPIC CROSS SECTIONS FOR CROM

NUCLIDE NUMBER 11

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG	G-6XG
1	0.	0.001	0.	1.840	0.634	0.	0.	0.	0.	0.	0.
2	0.	0.001	0.	2.449	1.409	0.354	0.	0.	0.	0.	0.
3	0.	0.002	0.	2.341	1.530	0.407	0.340	0.	0.	0.	0.
4	0.	0.004	0.	2.310	2.145	0.103	0.280	0.249	0.	0.	0.
5	0.	0.007	0.	2.310	2.140	0.161	0.431	0.185	0.144	0.	0.
6	0.	0.008	0.	2.771	2.573	0.163	0.	0.148	0.097	0.070	0.
7	0.	0.008	0.	2.440	2.272	0.190	0.	0.	0.	0.043	0.048
8	0.	0.009	0.	5.447	5.083	0.160	0.	0.	0.	0.127	0.027
9	0.	0.010	0.	4.290	4.024	0.355	0.	0.	0.	0.	0.
10	0.	0.014	0.	3.890	3.653	0.256	0.	0.	0.	0.	0.
11	0.	0.013	0.	2.590	2.424	0.223	0.	0.	0.	0.	0.
12	0.	0.019	0.	2.979	2.795	0.153	0.	0.	0.	0.	0.
13	0.	0.020	0.	6.242	5.876	0.165	0.	0.	0.	0.	0.
14	0.	0.036	0.	17.182	16.193	0.346	0.	0.	0.	0.	0.
15	0.	0.057	0.	14.006	13.681	0.953	0.	0.	0.	0.	0.
16	0.	0.143	0.	4.301	4.158	0.268	0.	0.	0.	0.	0.

SAMPLE CASE 1 FULL PRINT Y O M CROSS SECTIONS

THIS IS A REACTIVITY VERSUS TIME CALCULATION FOR A CONSTANT GEOMETRIC BUCKLING

THE INITIAL MULTIPLICATION CONSTANT IS EQUAL TO 1.050000

THE BURNUP WILL BE COMPUTED USING A CONSTANT POWER OF 2.00000E 02 MWT.

AVERAGE CROSS SECTIONS WILL BE EVALUATED AT $K = 1$ (ADJUSTING B^2)

THE CONVERGENCE CRITERION ON THE MARCH-OUT ALGORITHM IS 1.00000E-06

TOTAL NUMBER OF TIME STEPS IS 100

EACH STEP IS 1.000E+00 DAYS.

TOTAL BURNUP TIME IS 1.000E 02 DAYS.

OUTPUT WILL BE GIVEN EVERY 50 STEPS.

THE BREEDING RATIO HAS 1 SOURCE TERMS AND 2 SINK TERMS:

NUCLIDE NUMBERS FOR SOURCE TERMS 2

NUCLIDE NUMBERS FOR SINK TERMS 1 3

SAMPLE CASE 1 FULL PRINT Y O M CROSS SECTIONS

THE INITIAL CORE LOADING IS GIVEN BELOW FOR A CORE VOLUME OF 206,000 LITERS

NUCLIDE	NAME	ATOM DENSITY(CM-3)	KG/LITER	KILOGRAMS	SOURCE(KG/DAY)
1	PU239	2.13183E-03	8.4611E-01	1.7430E 02	0.
2	PU240	1.60461E-04	6.3953E-02	1.3174E 01	0.
3	PU241	0.	0.	0.	0.
4	PU242	0.	0.	0.	0.
5	FISPR	0.	0.	0.	0.
6	FISPR	0.	0.	0.	0.
7	OXYGN	4.58459E-03	1.2178E-01	2.5087E 01	0.
8	NA	1.14225E-02	4.3600E-01	8.9816E 01	0.
9	IRON	2.59000E-02	2.4015E 00	4.9471E 02	0.
10	NICKL	2.66500E-03	2.5977E-01	5.3514E 01	0.
11	CROM	6.76000E-03	5.8358E-01	1.2022E 02	0.

SAMPLE CASE 1 FULL PRINT Y O M CROSS SECTIONS

TIME = 0. DAYS

K EFFECTIVE = 1.050000E 00

TOTAL POWER = 2.000000E 02 MW

NEUTRON FLUX = 7.195664E 15 N/CM2-SEC

BREEDING RATIO = 1.141254E 02

GEOMETRIC BUCKLING = 3.225511E 03 CM-2

*** FUEL INVENTORY ***

NUCLIDE	NAME	ATOM DENSITY(CM-3)	CURRENT LOAD(KG)	INITIAL LOAD(KG)	CHANGE(KG)
1	PU239	2.13183E-03	1.74299E 02	1.74299E 02	0.
2	PU240	1.60461E-04	1.31742E 01	1.31742E 01	0.
3	PU241	0.	0.	0.	0.
4	PU242	0.	0.	0.	0.
5	FISPR	0.	0.	0.	0.
6	FISPR	0.	0.	0.	0.

SAMPLE CASE 1. FULL PRINT Y O M CROSS SECTIONS.

★ ★ ★ N U C L E A R D A T A ★ ★ ★

NEUTRON SOURCE RATE(PER CC PER SEC) = 8.58859E 13
 NEUTRON FISSION RATE(PER CC PER SEC) = 2.87581E 13
 NEUTRON ABSORPTION RATE(PER CC PER SEC) = 3.58403E 13
 NEUTRON LEAKAGE RATE(PER CC PER SEC) = 4.59557E 13
 TOTAL NEUTRON LEAKAGE FROM CORE = 0.

GROUP	FLUX(PER CM2 PER SEC)	NU*SIGF*FLUX	SIGF*FLUX	INTG FLUX
1	1.48254E 14	2.136284E 12	6.150109E 11	2.060325E-02
2	3.36082E 14	4.899656E 12	1.513676E 12	6.730938E-02
3	6.22000E 14	8.583645E 12	2.757649E 12	1.537503E-01
4	8.22315E 14	1.017085E 13	3.356405E 12	2.680296E-01
5	1.10398E 15	1.230687E 13	4.135569E 12	4.214522E-01
6	1.12293E 15	1.207169E 13	4.104029E 12	5.775083E-01
7	9.87245E 14	1.046739E 13	3.584380E 12	7.147083E-01
8	7.21779E 14	7.611994E 12	2.615805E 12	8.150157E-01
9	4.78006E 14	5.337668E 12	1.834250E 12	8.814455E-01
10	3.37222E 14	4.023680E 12	1.387476E 12	9.283100E-01
11	1.62049E 14	2.173981E 12	7.496487E 11	9.508304E-01
12	1.63451E 14	2.425212E 12	8.362801E 11	9.735456E-01
13	8.00414E 13	1.212359E 12	4.180547E 11	9.846691E-01
14	4.28536E 13	6.385210E 11	2.201797E 11	9.906249E-01
15	4.02138E 13	8.154538E 11	2.811910E 11	9.962135E-01
16	2.72463E 13	1.010671E 12	3.485072E 11	1.000000E-00

SAMPLE CASE 1 FULL PRINT Y O M CROSS SECTIONS

THE MACROSCOPIC CROSS SECTIONS ARE GIVEN IN THE TABLE BELOW

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G=1XG	G=2XG	G=3XG	G=4XG	G=5XG	G=6XG
1	4.15E-03	4.68E-03	1.44E-02	9.87E-02	3.68E-02	0.	0.	0.	0.	0.	0.
2	4.50E-03	4.72E-03	1.46E-02	1.16E-01	6.51E-02	1.80E-02	0.	0.	0.	0.	0.
3	4.43E-03	4.73E-03	1.38E-02	1.21E-01	8.04E-02	2.83E-02	1.50E-02	0.	0.	0.	0.
4	4.08E-03	4.46E-03	1.24E-02	1.36E-01	1.04E-01	2.20E-02	6.58E-03	1.14E-02	0.	0.	0.
5	3.75E-03	4.23E-03	1.11E-02	1.58E-01	1.35E-01	1.81E-02	1.06E-02	5.14E-03	6.91E-03	0.	0.
6	3.65E-03	4.28E-03	1.08E-02	1.78E-01	1.56E-01	1.81E-02	3.70E-03	1.59E-03	3.19E-03	3.56E-03	0.
7	3.63E-03	4.54E-03	1.06E-02	1.68E-01	1.48E-01	1.73E-02	7.46E-04	2.52E-03	3.71E-04	1.09E-03	2.42E-03
8	3.62E-03	4.85E-03	1.05E-02	2.02E-01	1.80E-01	1.54E-02	7.04E-05	3.94E-04	1.54E-03	1.00E-03	1.42E-03
9	3.84E-03	5.42E-03	1.12E-02	2.87E-01	2.61E-01	1.68E-02	1.07E-05	2.56E-05	1.37E-04	8.67E-04	2.65E-04
10	4.11E-03	6.10E-03	1.19E-02	2.58E-01	2.33E-01	2.01E-02	3.93E-05	0.	1.49E-05	1.26E-04	0.
11	4.63E-03	7.30E-03	1.34E-02	4.62E-01	4.25E-01	1.88E-02	0.	2.13E-06	0.	8.53E-06	0.
12	5.12E-03	7.92E-03	1.48E-02	2.58E-01	2.33E-01	2.95E-02	0.	0.	0.	0.	6.40E-06
13	5.22E-03	8.51E-03	1.51E-02	3.70E-01	3.37E-01	1.73E-02	0.	0.	0.	0.	2.13E-06
14	5.14E-03	8.66E-03	1.49E-02	5.19E-01	4.77E-01	2.38E-02	0.	0.	0.	0.	0.
15	6.99E-03	1.65E-02	2.03E-02	6.26E-01	5.91E-01	3.38E-02	0.	0.	0.	0.	0.
16	1.28E-02	2.35E-02	3.71E-02	3.92E-01	3.68E-01	1.78E-02	0.	0.	0.	0.	0.

SAMPLE CASE 1 FULL PRINT Y O M CROSS SECTIONS

*** S P E C T R U M A V E R A G E D C R O S S S E C T I O N S ***

NUMBER	NAME	SIGMA F	SIGMA ABS	NU SIGMA F	SIGMA CAP
1	PU239	1.8302E 00	2.1367E 00	5.4567E 00	3.0652E-01
2	PU240	5.6867E-01	8.9265E-01	1.8748E 00	3.2398E-01
3	PU241	2.2115E 00	2.4187E 00	6.7496E 00	2.0721E-01
4	PU242	5.5116E-01	8.7514E-01	1.8138E 00	3.2398E-01
5	FISPR	0.	1.1622E-01	0.	1.1622E-01
6	FISPR	0.	1.1622E-01	0.	1.1622E-01
7	OXYGN	0.	1.1105E-03	0.	1.1105E-03
8	NA	0.	2.2672E-03	0.	2.2672E-03
9	IRON	0.	5.6629E-03	0.	5.6629E-03
10	NICKL	0.	9.6368E-03	0.	9.6368E-03
11	CROM	0.	8.2178E-03	0.	8.2178E-03

SAMPLE CASE 1: FULL PRINT Y O M CROSS SECTIONS:

T I M E = 50.00 D A Y S

K I E F F E C T I V E = 1.012021E 00

T O T A L P O W E R = 2.000000E 02 MWT

N E U T R O N F L U X = 7.654845E 15 N/CM2-SEC

B R E E D I N G R A T I O = 1.374664E-02

G E O M E T R I C B U C K L I N G = 3.225511E-03 CM-2

* * * F U E L I N V E N T O R Y * * *

NUCLIDE	NAME	ATOM DENSITY(CM-3)	CURRENT LOAD(KG)	INITIAL LOAD(KG)	CHANGE(KG)
1	PU239	1.99018E-03	1.62718E 02	1.74299E 02	-1.15816E 01
2	PU240	1.76316E-04	1.44760E 01	1.31742E 01	1.30175E 00
3	PU241	1.70429E-06	1.40515E-01	0.	1.40515E-01
4	PU242	5.71230E-09	4.72921E-04	0.	4.72921E-04
5	FISPR	1.24081E-04	4.24385E-02	0.	4.24385E-02
6	FISPR	1.24081E-04	4.24385E-02	0.	4.24385E-02

SAMPLE CASE 1 FULL PRINT Y O M CROSS SECTIONS

* * * N U C L E A R D A T A * * *

NEUTRON SOURCE RATE(PER CC PER SEC) = 8.58691E 13
 NEUTRON FISSION RATE(PER CC PER SEC) = 2.87581E 13
 NEUTRON ABSORPTION RATE(PER CC PER SEC) = 3.63977E 13
 NEUTRON LEAKAGE RATE(PER CC PER SEC) = 4.84514E 13
 TOTAL NEUTRON LEAKAGE FROM CORE = 4.19622E 25

GROUP	FLUX(PER CM2 PER SEC)	NU*SIGF*FLUX	SIGF*FLUX	INTG FLUX
1	1.53585E 14	2.091719E 12	6.014945E 11	2.006372E-02
2	3.48384E 14	4.797979E 12	1.480970E 12	6.557526E-02
3	6.46213E 14	8.422328E 12	2.703962E 12	1.499941E-01
4	8.57747E 14	1.001396E 13	3.302691E 12	2.620469E-01
5	1.16090E 15	1.216555E 13	4.086616E 12	4.137030E-01
6	1.18869E 15	1.195953E 13	4.065440E 12	5.689892E-01
7	1.05376E 15	1.044481E 13	3.576475E 12	7.066480E-01
8	7.75308E 14	7.642219E 12	2.626099E 12	8.079313E-01
9	5.17667E 14	5.403695E 12	1.856873E 12	8.755573E-01
10	3.69541E 14	4.122218E 12	1.421393E 12	9.238327E-01
11	1.79438E 14	2.250689E 12	7.760684E 11	9.472739E-01
12	1.83541E 14	2.546472E 12	8.780558E 11	9.712510E-01
13	9.10241E 13	1.289544E 12	4.446478E 11	9.831420E-01
14	4.91909E 13	6.857091E 11	2.364376E 11	9.895681E-01
15	4.68694E 13	8.889082E 11	3.065048E 11	9.956909E-01
16	3.29852E 13	1.143756E 12	3.943846E 11	1.000000E 00

SAMPLE CASE 1 FULL PRINT Y O M CROSS SECTIONS

THE MACROSCOPIC CROSS SECTIONS ARE GIVEN IN THE TABLE BELOW

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G=1XG	G=2XG	G=3XG	G=4XG	G=5XG	G=6XG
1	3.92E-03	4.45E-03	1.36E-02	9.89E-02	3.69E-02	0.	0.	0.	0.	0.	0.
2	4.25E-03	4.47E-03	1.38E-02	1.16E-01	6.52E-02	1.80E-02	0.	0.	0.	0.	0.
3	4.18E-03	4.49E-03	1.30E-02	1.21E-01	8.06E-02	2.84E-02	1.51E-02	0.	0.	0.	0.
4	3.85E-03	4.24E-03	1.17E-02	1.36E-01	1.05E-01	2.21E-02	6.68E-03	1.15E-02	0.	0.	0.
5	3.52E-03	4.01E-03	1.05E-02	1.59E-01	1.35E-01	1.82E-02	1.06E-02	5.21E-03	6.95E-03	0.	0.
6	3.42E-03	4.05E-03	1.01E-02	1.79E-01	1.57E-01	1.82E-02	3.76E-03	1.64E-03	3.22E-03	3.58E-03	0.
7	3.39E-03	4.29E-03	9.91E-03	1.69E-01	1.49E-01	1.73E-02	7.46E-04	2.54E-03	3.96E-04	1.10E-03	2.43E-03
8	3.39E-03	4.60E-03	9.86E-03	2.03E-01	1.81E-01	1.55E-02	7.61E-05	3.99E-04	1.53E-03	1.01E-03	1.43E-03
9	3.59E-03	5.14E-03	1.04E-02	2.88E-01	2.62E-01	1.68E-02	1.94E-05	3.28E-05	1.38E-04	8.74E-04	2.60E-04
10	3.85E-03	5.79E-03	1.12E-02	2.58E-01	2.34E-01	2.02E-02	3.83E-05	5.46E-06	1.39E-05	1.26E-04	3.75E-08
11	4.32E-03	6.93E-03	1.25E-02	4.62E-01	4.26E-01	1.88E-02	3.75E-08	2.00E-06	0.	7.96E-06	0.
12	4.78E-03	7.52E-03	1.39E-02	2.58E-01	2.34E-01	2.96E-02	0.	2.22E-08	0.	0.	5.97E-06
13	4.88E-03	8.10E-03	1.42E-02	3.69E-01	3.37E-01	1.73E-02	0.	0.	0.	0.	1.99E-06
14	4.81E-03	8.28E-03	1.39E-02	5.19E-01	4.77E-01	2.39E-02	0.	0.	0.	0.	0.
15	6.54E-03	1.60E-02	1.90E-02	6.25E-01	5.91E-01	3.38E-02	0.	0.	0.	0.	0.
16	1.20E-02	2.26E-02	3.47E-02	3.92E-01	3.69E-01	1.78E-02	0.	0.	0.	0.	0.

SAMPLE CASE 1 FULL PRINT Y O M CROSS SECTIONS

*** S P E C T R U M A V E R A G E D C R O S S S E C T I O N S ***

NUMBER	NAME	SIGMA F	SIGMA ABS	NU SIGMA F	SIGMA CAP
1	PU239	1.8367E 00	2.1563E 00	5.4701E 00	3.1964E-01
2	PU240	5.4866E-01	8.8325E-01	1.8078E 00	3.3459E-01
3	PU241	2.2434E 00	2.4591E 00	6.8396E 00	2.1575E-01
4	PU242	5.3199E-01	8.6658E-01	1.7497E 00	3.3459E-01
5	FISPR	0.	1.1950E-01	0.	1.1950E-01
6	FISPR	0.	1.1950E-01	0.	1.1950E-01
7	OXYGN	0.	1.0526E-03	0.	1.0526E-03
8	NA	0.	2.5739E-03	0.	2.5739E-03
9	IRON	0.	5.7762E-03	0.	5.7762E-03
10	NICKL	0.	9.9409E-03	0.	9.9409E-03
11	CROM	0.	8.5088E-03	0.	8.5088E-03

SAMPLE CASE 1 FULL PRINT Y O M CROSS SECTIONS

T I M E = 100.00 D A Y S

K EFFECTIVE = 9.720033E+01

TOTAL POWER = 2.000000E+02 MWT

NEUTRON FLUX = 8.178242E+15 N/CM²-SEC

BREEDING RATIO = 1.653718E+02

GEOMETRIC BUCKLING = 3.225511E-03 CM⁻²

*** FUEL INVENTORY ***

NUCLIDE	NAME	ATOM DENSITY(CM ⁻³)	CURRENT LOAD(KG)	INITIAL LOAD(KG)	CHANGE(KG)
1	PU239	1.84819E+03	1.51109E+02	1.74299E+02	-2.31904E+01
2	PU240	1.92190E+04	1.57793E+01	1.31742E+01	2.60504E+00
3	PU241	3.60336E+06	2.97089E+01	0.	2.97089E+01
4	PU242	2.52695E+08	2.09206E+03	0.	2.09206E+03
5	FISPR	2.48253E+04	8.49079E+02	0.	8.49079E+02
6	FISPR	2.48253E+04	8.49079E+02	0.	8.49079E+02

SAMPLE CASE 1 FULL PRINT Y O M CROSS SECTIONS

★ ★ ★ N U C L E A R D A T A ★ ★ ★

NEUTRON SOURCE RATE(PER CC PER SEC) = 8.58552E 13
 NEUTRON FISSION RATE(PER CC PER SEC) = 2.87581E 13
 NEUTRON ABSORPTION RATE(PER CC PER SEC) = 3.70383E 13
 NEUTRON LEAKAGE RATE(PER CC PER SEC) = 5.12898E 13
 TOTAL NEUTRON LEAKAGE FROM CORE = 8.62898E 25

GROUP	FLUX(PER CM2 PER SEC)	NU*SIGF*FLUX	SIGF*FLUX	INTG FLUX
1	1.59671E 14	2.048302E 12	5.882515E 11	1.952387E-02
2	3.62416E 14	4.698610E 12	1.448868E 12	6.383856E-02
3	6.73747E 14	8.264154E 12	2.651116E 12	1.462214E-01
4	8.97869E 14	9.858908E 12	3.249394E 12	2.560090E-01
5	1.22513E 15	1.201904E 13	4.035742E 12	4.058126E-01
6	1.26284E 15	1.183459E 13	4.022436E 12	5.602271E-01
7	1.12889E 15	1.040938E 13	3.564131E 12	6.982631E-01
8	8.35996E 14	7.664028E 12	2.633475E 12	8.004850E-01
9	5.62837E 14	5.465406E 12	1.877993E 12	8.693063E-01
10	4.06597E 14	4.219685E 12	1.454921E 12	9.190232E-01
11	1.99514E 14	2.328404E 12	8.028244E 11	9.434188E-01
12	2.06987E 14	2.672407E 12	9.214285E 11	9.687283E-01
13	1.03971E 14	1.371192E 12	4.727702E 11	9.814415E-01
14	5.67151E 13	7.362381E 11	2.538415E 11	9.883763E-01
15	5.48820E 13	9.688985E 11	3.340652E 11	9.950871E-01
16	4.01792E 13	1.295987E 12	4.468560E 11	1.000000E-00

SAMPLE CASE 1 FULL PRINT Y O M CROSS SECTIONS

THE MACROSCOPIC CROSS SECTIONS ARE GIVEN IN THE TABLE BELOW

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG	G-6XG
1	3.68E-03	4.21E-03	1.28E-02	9.91E-02	3.70E-02	0.	0.	0.	0.	0.	0.
2	4.00E-03	4.21E-03	1.30E-02	1.16E-01	6.52E-02	1.81E-02	0.	0.	0.	0.	0.
3	3.93E-03	4.24E-03	1.23E-02	1.21E-01	8.08E-02	2.85E-02	1.53E-02	0.	0.	0.	0.
4	3.62E-03	4.02E-03	1.10E-02	1.36E-01	1.05E-01	2.22E-02	6.79E-03	1.16E-02	0.	0.	0.
5	3.29E-03	3.79E-03	9.81E-03	1.60E-01	1.36E-01	1.83E-02	1.07E-02	5.28E-03	7.00E-03	0.	0.
6	3.19E-03	3.81E-03	9.37E-03	1.79E-01	1.58E-01	1.82E-02	3.81E-03	1.69E-03	3.26E-03	3.60E-03	0.
7	3.16E-03	4.05E-03	9.22E-03	1.70E-01	1.50E-01	1.74E-02	7.45E-04	2.56E-03	4.21E-04	1.11E-03	2.44E-03
8	3.15E-03	4.34E-03	9.17E-03	2.03E-01	1.82E-01	1.55E-02	8.19E-05	4.04E-04	1.53E-03	1.02E-03	1.44E-03
9	3.34E-03	4.85E-03	9.71E-03	2.88E-01	2.63E-01	1.69E-02	2.81E-05	4.01E-05	1.39E-04	8.81E-04	2.55E-04
10	3.58E-03	5.48E-03	1.04E-02	2.59E-01	2.34E-01	2.02E-02	3.73E-05	1.09E-05	1.29E-05	1.26E-04	7.93E-08
11	4.02E-03	6.57E-03	1.17E-02	4.63E-01	4.26E-01	1.89E-02	7.93E-08	1.87E-06	0.	7.39E-06	0.
12	4.45E-03	7.11E-03	1.29E-02	2.58E-01	2.34E-01	2.96E-02	0.	4.68E-08	0.	0.	5.54E-06
13	4.55E-03	7.70E-03	1.32E-02	3.69E-01	3.38E-01	1.73E-02	0.	0.	0.	0.	1.85E-06
14	4.48E-03	7.90E-03	1.30E-02	5.18E-01	4.77E-01	2.39E-02	0.	0.	0.	0.	0.
15	6.09E-03	1.54E-02	1.77E-02	6.24E-01	5.91E-01	3.39E-02	0.	0.	0.	0.	0.
16	1.11E-02	2.16E-02	3.23E-02	3.92E-01	3.71E-01	1.78E-02	0.	0.	0.	0.	0.

SAMPLE CASE 1 FULL PRINT Y O M CROSS SECTIONS

* * * S P E C T R U M A V E R A G E D C R O S S S E C T I O N S * * *

NUMBER	NAME	SIGMA F	SIGMA ABS	NU SIGMA F	SIGMA CAP
1	PU239	1.8445E 00	2.1790E 00	5.4872E 00	3.3441E+01
2	PU240	5.2776E+01	8.7428E+01	1.7379E 00	3.4652E+01
3	PU241	2.2794E 00	2.5048E 00	6.9418E 00	2.2536E+01
4	PU242	5.1196E+01	8.5848E+01	1.6828E 00	3.4652E+01
5	FISPR	0.	1.2327E+01	0.	1.2327E+01
6	FISPR	0.	1.2327E+01	0.	1.2327E+01
7	OXYGN	0.	9.9393E-04	0.	9.9393E-04
8	NA	0.	2.9328E-03	0.	2.9328E-03
9	IRON	0.	5.8985E-03	0.	5.8985E-03
10	NICKL	0.	1.0286E-02	0.	1.0286E-02
11	CROM	0.	8.8415E-03	0.	8.8415E-03

***** ESTIMATE OF FUEL COST *****

SAMPLE CASE 1 FULL PRINT Y O M CROSS SECTIONS

DIRT CHEAP

***** ZERO DIMENSIONAL FUEL CYCLE CALCULATION *****

INPUT DATA LISTED BELOW

NUCLIDE NUMBER	NAME	ATM WEIGHT	LAMBDA(DAYS-1)	MEV/FISSION	SOURCE ELEMENT FOR DECAY	CAPTURE	FISSION
1	PU239	2.3905E 02	-0.	2.1100E 02	0 0 0		
2	PU240	2.4005E 02	-0.	2.1100E 02	0 1 0		
3	PU241	2.4106E 02	1.5000E-04	2.1100E 02	0 2 0		
4	PU242	2.4206E 02	-0.	2.1100E 02	0 3 0		
5	FISPR	1.0000E-00	-0.	-0.	0 6 1 2 3 4		
6	FISPR	1.0000E-00	-0.	-0.	0 5 1 2 3 4		

NON-BURNABLE NUCLIDES

7	OXYGN	1.5999E 01					
8	NA	2.2990E 01					
9	IRON	5.5847E 01					
10	NICKL	5.8710E 01					
11	CRON	5.1996E 01					

SAMPLE CASE 2 PARTIAL PRINT Y O M CROSS SECTIONS

THIS IS A REACTIVITY VERSUS TIME CALCULATION FOR A CONSTANT GEOMETRIC BUCKLING

THE INITIAL MULTIPLICATION CONSTANT IS EQUAL TO 1.050000

THE BURNUP WILL BE COMPUTED USING A CONSTANT POWER OF 2.00000E 02 MWt.

AVERAGE CROSS SECTIONS WILL BE EVALUATED AT $K = 1$ (ADJUSTING B2)

THE CONVERGENCE CRITERION ON THE MARCH-OUT ALGORITHM IS 1.00000E-06

TOTAL NUMBER OF TIME STEPS IS 100

EACH STEP IS 1.000E+00 DAYS

TOTAL BURNUP TIME IS 1.000E 02 DAYS

OUTPUT WILL BE GIVEN EVERY 50 STEPS

THE BREEDING RATIO HAS 1 SOURCE TERMS AND 2 SINK TERMS

NUCLIDE NUMBERS FOR SOURCE TERMS 2

NUCLIDE NUMBERS FOR SINK TERMS 1 3

SAMPLE CASE 2 PARTIAL PRINT Y O M CROSS SECTIONS

THE INITIAL CORE LOADING IS GIVEN BELOW FOR A CORE VOLUME OF 206,000 LITERS

NUCLIDE	NAME	ATOM DENSITY(CM-3)	KG/LITER	KILOGRAMS	SOURCE(KG/DAY)
1	PU239	2.13183E-03	8.4611E-01	1.7430E 02	0.
2	PU240	1.60461E-04	6.3953E-02	1.3174E 01	0.
3	PU241	0.	0.	0.	0.
4	PU242	0.	0.	0.	0.
5	FISPR	0.	0.	0.	0.
6	FISPR	0.	0.	0.	0.
7	OXYGN	4.58459E-03	1.2178E-01	2.5087E 01	0.
8	NA	1.14225E-02	4.3600E-01	8.9816E 01	0.
9	IRON	2.59000E-02	2.4015E 00	4.9471E 02	0.
10	NICKL	2.66500E-03	2.5977E-01	5.3514E 01	0.
11	CROM	6.76000E-03	5.8358E-01	1.2022E 02	0.

SAMPLE CASE 2 PARTIAL PRINT Y O M CROSS SECTIONS

T I M E = 0. D A Y S

K EFFECTIVE = 1.050000E 00

TOTAL POWER = 2.000000E 02 MW

NEUTRON FLUX = 7.195664E 15 N/CM2-SEC

BREEDING RATIO = 1.141254E-02

GEOMETRIC BUCKLING = 3.225511E-03 CM-2

*** FUEL INVENTORY ***

NUCLIDE	NAME	ATOM DENSITY(CM-3)	CURRENT LOAD(KG)	INITIAL LOAD(KG)	CHANGE(KG)
1	PU239	2.13183E-03	1.74299E 02	1.74299E 02	0.
2	PU240	1.60461E-04	1.31742E 01	1.31742E 01	0.
3	PU241	0.	0.	0.	0.
4	PU242	0.	0.	0.	0.
5	FISPR	0.	0.	0.	0.
6	FISPR	0.	0.	0.	0.

SAMPLE CASE 2 PARTIAL PRINT Y O M CROSS SECTIONS

*** N U C L E A R D A T A ***

NEUTRON SOURCE RATE(PER CC PER SEC) = 8.58859E 13
 NEUTRON FISSION RATE(PER CC PER SEC) = 2.87581E 13
 NEUTRON ABSORPTION RATE(PER CC PER SEC) = 3.58403E 13
 NEUTRON LEAKAGE RATE(PER CC PER SEC) = 4.59557E 13
 TOTAL NEUTRON LEAKAGE FROM CORE = 0.

GROUP	FLUX(PER CM2 PER SEC)	NU*SIGF*FLUX	SIGF*FLUX	INTG FLUX
1	1.48254E 14	2.136284E 12	6.150109E 11	2.060325E-02
2	3.36082E 14	4.899656E 12	1.513676E 12	6.730938E-02
3	6.22000E 14	8.583645E 12	2.757649E 12	1.537503E-01
4	8.22315E 14	1.017085E 13	3.356405E 12	2.680296E-01
5	1.10398E 15	1.230687E 13	4.135569E 12	4.214522E-01
6	1.12293E 15	1.207169E 13	4.104029E 12	5.775083E-01
7	9.87245E 14	1.046739E 13	3.584380E 12	7.147083E-01
8	7.21779E 14	7.611994E 12	2.615805E 12	8.150157E-01
9	4.78006E 14	5.337668E 12	1.834250E 12	8.814455E-01
10	3.37222E 14	4.023680E 12	1.387476E 12	9.283100E-01
11	1.62049E 14	2.173981E 12	7.496487E 11	9.508304E-01
12	1.63451E 14	2.425212E 12	8.362801E 11	9.735456E-01
13	8.00414E 13	1.212359E 12	4.180547E 11	9.846691E-01
14	4.28556E 13	6.385210E 11	2.201797E 11	9.906249E-01
15	4.02138E 13	8.154538E 11	2.811910E 11	9.962135E-01
16	2.72463E 13	1.010671E 12	3.485072E 11	1.000000E-00

SAMPLE CASE 2 PARTIAL PRINT Y O M CROSS SECTIONS

T I M E = 20.00 D A Y S

K E F F E C T I V E = 1.012021E 00
 T O T A L P O W E R = 2.000000E 02 MW
 N E U T R O N F L U X = 7.654845E 15 N/CM2-SEC
 B R E E D I N G R A T I O = 1.374664E 02
 G E O M E T R I C B U C K L I N G = 3.225511E-03 CM-2

*** F U E L I N V E N T O R Y ***

NUCLIDE	NAME	ATOM DENSITY(CM-3)	CURRENT LOAD(KG)	INITIAL LOAD(KG)	CHANGE(KG)
1	PU239	1.99018E-03	1.62718E 02	1.74299E 02	-1.15816E 01
2	PU240	1.76316E-04	1.44760E 01	1.31742E 01	1.30175E 00
3	PU241	1.70429E-06	1.40515E-01	0.	1.40515E-01
4	PU242	5.71230E-09	4.72921E-04	0.	4.72921E-04
5	FISPR	1.24081E-04	4.24385E-02	0.	4.24385E-02
6	FISPR	1.24081E-04	4.24385E-02	0.	4.24385E-02

SAMPLE CASE 2 PARTIAL PRINT / O M CROSS SECTIONS

*** N U C L E A R D A T A ***

NEUTRON SOURCE RATE(PER CC PER SEC) = 8.28691E 13
 NEUTRON FISSION RATE(PER CC PER SEC) = 2.87581E 13
 NEUTRON ABSORPTION RATE(PER CC PER SEC) = 3.63977E 13
 NEUTRON LEAKAGE RATE(PER CC PER SEC) = 4.84514E 13
 TOTAL NEUTRON LEAKAGE FROM CORE = 4.19922E 25

GROUP	FLUX(PER CM2 PER SEC)	NU*SIGF*FLUX	SIGF*FLUX	INTG FLUX
1	1.53585E 14	2.091719E 12	6.014945E 11	2.006372E-02
2	3.48384E 14	4.797979E 12	1.480970E 12	6.557526E-02
3	6.46213E 14	8.422328E 12	2.703962E 12	1.499941E-01
4	8.57747E 14	1.001396E 13	3.302691E 12	2.620469E-01
5	1.16090E 15	1.216555E 13	4.086616E 12	4.137030E-01
6	1.18869E 15	1.195953E 13	4.065440E 12	5.689892E-01
7	1.05376E 15	1.044481E 13	3.576475E 12	7.066480E-01
8	7.75308E 14	7.642219E 12	2.626099E 12	8.079313E-01
9	5.17667E 14	5.403695E 12	1.856873E 12	8.755573E-01
10	3.69541E 14	4.122218E 12	1.421393E 12	9.238327E-01
11	1.79438E 14	2.250689E 12	7.760684E 11	9.472739E-01
12	1.83541E 14	2.546472E 12	8.780558E 11	9.712510E-01
13	9.10241E 13	1.289544E 12	4.446478E 11	9.831420E-01
14	4.91909E 13	6.857091E 11	2.364376E 11	9.895681E-01
15	4.68694E 13	8.889082E 11	3.065048E 11	9.956909E-01
16	3.29852E 13	1.143756E 12	3.943846E 11	1.000000E 00

SAMPLE CASE 2 PARTIAL PRINT Y U M CROSS SECTIONS

TIME = 100.00 DAYS

K EFFECTIVE = 9.720033E-01

TOTAL POWER = 2.000000E 02 MW

NEUTRON FLUX = 8.178242E 15 N/CM2-SEC

BREEDING RATIO = 1.653718E-02

GEOMETRIC BUCKLING = 3.225511E-03 CM-2

*** FUEL INVENTORY ***

NUCLIDE	NAME	ATOM DENSITY(CM-3)	CURRENT LOAD(KG)	INITIAL LOAD(KG)	CHANGE(KG)
1	PU239	1.84819E-03	1.51109E 02	1.74299E 02	-2.31904E 01
2	PU240	1.92190E-04	1.57793E 01	1.31742E 01	2.60504E 00
3	PU241	3.60336E-06	2.97089E-01	0.	2.97089E-01
4	PU242	2.52595E-08	2.09206E-03	0.	2.09206E-03
5	PIPR	2.48253E-04	8.49079E-02	0.	8.49079E-02
6	PIPR	2.48253E-04	8.49079E-02	0.	8.49079E-02

SAMPLE CASE 2 PARTIAL PRINT Y O M CROSS SECTIONS

*** N U C L E A R D A T A ***

NEUTRON SOURCE RATE(PER CC PER SEC) = 8.58552E 13
 NEUTRON FISSION RATE(PER CC PER SEC) = 2.87581E 13
 NEUTRON ABSORPTION RATE(PER CC PER SEC) = 3.70383E 13
 NEUTRON LEAKAGE RATE(PER CC PER SEC) = 5.12898E 13
 TOTAL NEUTRON LEAKAGE FROM CORE = 8.62898E 25

GROUP	FLUX(PER CM2 PER SEC)	NU*SIGF*FLUX	SIGF*FLUX	INTG FLUX
1	1.59671E 14	2.048302E 12	5.882515E 11	1.952387E-02
2	3.62416E 14	4.698610E 12	1.448868E 12	6.383856E-02
3	6.73747E 14	8.264154E 12	2.651116E 12	1.462214E-01
4	8.97869E 14	9.658908E 12	3.249394E 12	2.560090E-01
5	1.22513E 15	1.201904E 13	4.035742E 12	4.058126E-01
6	1.26284E 15	1.183459E 13	4.022436E 12	5.602271E-01
7	1.12889E 15	1.040938E 13	3.564131E 12	6.982631E-01
8	8.35996E 14	7.664028E 12	2.633475E 12	8.004850E-01
9	5.62837E 14	5.465406E 12	1.877993E 12	8.693063E-01
10	4.06597E 14	4.219685E 12	1.454921E 12	9.190232E-01
11	1.99514E 14	2.328404E 12	8.028244E 11	9.434188E-01
12	2.06987E 14	2.672407E 12	9.214285E 11	9.687283E-01
13	1.03971E 14	1.371192E 12	4.727702E 11	9.814415E-01
14	5.67151E 13	7.362381E 11	2.538415E 11	9.883763E-01
15	5.48820E 13	9.688985E 11	3.340652E 11	9.950871E-01
16	4.01792E 13	1.295987E 12	4.468560E 11	1.000000E-00

***** ESTIMATE OF FUEL COST *****

SAMPLE CASE 2 PARTIAL PRINT Y O M CROSS SECTIONS

DIRT CHEAP

APPENDIX III

LISTING OF FORTRAN IV SOURCE DECK

SIGFTC MAIN - LIST

```

  DIMENSION CHI(26), EL(26), ATW(40), SORCE(40), ALAM(20), FE(20),
1 HOL(12), HOLN(40), AN(20,2), DNM(20), DNMI(40), SM(15,26), DN(40)
2 , SIGR(26), S(40,15,26), ACS(40,4), PHI(26), FPHI(26)
  DIMENSION LD(20), LC(20), LF(20), LCN(20,4), LFN(20,7),
1 LBLN(5), LBSN(5)
  COMMON NUMT, NPRT, NBE, NNE, NXS, NG, NP, NXCM, NSP, NFP, NOP,
1 NCOST, NCAL, NBS, NBL
  COMMON H, HS, VOL, POW, FLX, EPSL, BSQ, AK, T, BR, TOTL, RS,
1 RA, RL, RF
  COMMON LD, LC, LF, LCN, LFN, LBSN, LBLN
  COMMON CHI, EL, ATW, SORCE, ALAM, FE, HOL, HOLN, DN, AN, DNM, DNMI, SM,
1 SIGR, S, ACS, PHI, FPHI
C   OD FUEL CYCLE
  CALL INPUT1
  CALL PRINT1
1  CONTINUE
  BSO = .001
  CALL INPUT2
  CALL START
  NPRTG = 10000
  T = .0
  IF (NUMT) 11, 11, 212
212 DO 8 N = 1, NUMT
  NPRTG = NPRTG + 1
  CALL AVXS
  CALL NORM
  IF (NOP) 100, 100, 101
100 NOP = 1
101 IF (NPRTG - NPRT) 6, 2, 2
2  CALL PRINT
  NPRTG = 0
6  T = T + H
  CALL MARCH
8  CONTINUE
11 CONTINUE
  CALL AVXS
  CALL NORM
  CALL PRINT
  IF (NCOST) 9, 9, 1
9  CALL COST
  GO TO 1
  END
```

```

$IBFIC START LIST
SUBROUTINE START
  DIMENSION CHI(26), EL(26), ATW(40), SORCE(40), ALAM(20), FE(20),
1 HOL(12), HOLN(40), AN(20,2), DNM(20), DNMI(40), SM(15,26), DN(40)
2 , SIGR(26), S(40,15,26), ACS(40,4), PHI(26), FPHI(26)
  DIMENSION LD(20), LC(20), LF(20), LCN(20,4), LFN(20,7),
1 LBLN(5), LBSN(5)
  COMMON NUMT, NPRT, NBE, NNE, NXS, NG, NP, NXCM, NSP, NFP, NOP,
1 NCOST, NCAL, NBS, NBL
  COMMON H, HS, VOL, POW, FLX, EPSL, BSQ, AK, T, BR, TOTL, RS,
1 RA, RL, RF
  COMMON LD,LC,LF,LCN,LFN,LBSN,LBLN
  COMMON CHI,EL,ATW,SORCE,ALAM,FE,HOL,HOLN,DN,AN,DNM,DNMI,SM,
1 SIGR,S,ACS,PHI,FPHI
  READ (2,80) (DN(J),J=1, NXS)
  READ (2,80) (SORCE(I),I=1,NBE)
80  FORMAT (6E12.4)
  WRITE (3,81)
81  FORMAT (1H1)
  WRITE (3,82) (HOL(I),I=1,12)
82  FORMAT (12A6)
  WRITE (3,84) VOL
84  FORMAT (1H0///62H THE INITIAL CORE LOADING IS GIVEN BELOW FOR A CO
1RE VOLUME OF F9.3,7H LITERS////
281H NUCLIDE NAME ATOM DENSITY(CM-3) KG/LITER KILOGRAMS
3 SOURCE(KG/DAY)///)
  KK = NBE + 1
  DO 90 J = KK,NXS
90  SORCE(J) = .0
  DO 120 J = 1,NXS
  DNMI(J) = VOL*DN(J)*ATW(J)/.6023
  G = DNMI(J)/VOL
  WRITE (3,101) J,HOLN(J), DN(J), G, DNMI(J),SORCE(J)
101  FORMAT (I7,5H ,A6,1PE18.5,1P3E14.4)
  WRITE (3,103)
103  FORMAT (1H /)
120  CONTINUE
  G = .602/VOL
  GG = 24.*3600.
  DO 150 J = 1,NBE
  DNM(J) = DNMI(J)
  SORCE(J) = SORCE(J)*G/(GG*ATW(J))
150  CONTINUE
  RETURN
  END

```

SIOTFC INP1 LIST

```
      SUBROUTINE INPUT1
      DIMENSION CHI(26), EL(26), ATW(40), SORCE(40), ALAM(20), FE(20),
1  HOL(12), HOLN(40), AN(20,2), DNM(20), DNMI(40), SM(15,26), DN(40)
2  , SIGR(26), S(40,15,26), ACS(40,4), PHI(26), FPHI(26)
      DIMENSION LD(20), LC(20), LF(20), LCN(20,4), LFN(20,7),
1  LBLN(5), LPSN(5)
      COMMON NUNT, NPRT, NBE, NNE, NXS, NG, NP, NXCM, NSP, NFP, NOP,
1  NCOST, NCAL, NBS, NBL
      COMMON H, HS, VOL, POW, FLX, EPSL, BSQ, AK, T, BR, TOTL, RS,
1  RA, RL, RE
      COMMON LD, LC, LF, LCN, LFN, LPSN, LBLN
      COMMON CHI, EL, ATW, SORCE, ALAM, FE, HOL, HOLN, DN, AN, DNM, DNMI, SM,
1  SIGR, S, ACS, PHI, FPHI
      READ (2,2) NG, NXCM, NBE, NNE, NP
2  FORMAT (5I6)
      NXS = NBE + NNE
      DO 4 J = 1, NBE
      READ (2,3) HOLN(J), ATW(J), ALAM(J), FE(J), LD(J), LC(J),
1  (LCN(J,K), K=1,4), LF(J), (LFN(J,K), K=1,7)
3  FORMAT (A6,3E6.1,14I3)
4  CONTINUE
      KK = NBE + 1
      DO 6 J = KK, NXS
      READ (2,5) HOLN(J), ATW(J)
5  FORMAT (A6,5E6.1)
6  CONTINUE
      READ (2,9) (CHI(I), I=1,NG)
      READ (2,9) (EL(I), I=1,NG)
9  FORMAT (6E12.4)
      KK = NXCM + 5
      DO 11 J = 1, NXS
      DO 11 I = 1, NG
      READ (2,10) KD, (S(J,K,I), K=1,8)
      READ (2,10) KD, (S(J,K,I), K=9, KK)
10  FORMAT (I1, E8.3, 7E9.4)
11  CONTINUE
      RETURN
      END
```

```

SIRFTC INP2      LIST
      SUBROUTINE INPUT2
      DIMENSION CHI(26), EL(26), ATW(40), SORCE(40), ALAM(20), FE(20),
1  HOL(12), HOLN(40), AN(20,2), DNM(20), DNMI(40), SM(15,26), DN(40)
2  , SIGR(26), S(40,15,26), ACS(40,4), PHI(26), FPHI(26)
      DIMENSION LD(20), LC(20), LF(20), LCN(20,4), LFN(20,7),
1  LBLN(5), LBSN(5)
      COMMON NUMT, NPRT, NBE, NNE, NXS, NG, NP, NXCM, NSP, NFP, NOP,
1  NCOST, NCAL, NBS, NBL
      COMMON H, HS, VOL, POW, FLX, EPSL, BSQ, AK, T, BR, TOTL, RS,
1  RA, RL, RF
      COMMON LD,LC,LF,LCN,LFN,LBSN,LBLN
      COMMON CHI,EL,ATW,SORCE,ALAM,FE,HOL,HOLN,DN,AN,DNM,DNMI,SM,
1  SIGR,S,ACS,PHI,FPHI
      TOTL = .0
      READ (2,2) (HOL(I),I=1,12)
2  FORMAT (12A6)
      WRITE (3,1) (HOL(I),I=1,12)
1  FORMAT (10H1      12A6)
      READ (2,4) NOP, NFP, NSP, NCOST, NUMT, NPRT
4  FORMAT (6I6)
      READ (2,6) H, VOL, G, GG, EPSL
6  FORMAT (5E12.4)
      HS = H * 24.* 3600.
      IF(NOP -1) 10,20,30
10  AK = GG
      WRITE (3,11) AK
11  FORMAT (1H0////87H      THIS IS A REACTIVITY VERSUS TIME CALC
1  ULATION FOR A CONSTANT GEOMETRIC BUCKLING///
249H THE INITIAL MULTIPLICATION CONSTANT IS EQUAL TO F8.6///)
      GO TO 35
20  BSQ = GG
      WRITE (3,21) BSQ
21  FORMAT (1H0////87H      THIS IS A REACTIVITY VERSUS TIME CALCUL
1  ATION FOR A CONSTANT GEOMETRIC BUCKLING///
226H THE BUCKLING IS EQUAL TO 1PE16.4, 6H CM-2.///)
      GO TO 35
30  AK = GG
      WRITE (3,31) AK
31  FORMAT (1H0////78H      THIS IS A BUCKLING VERSUS TIME CALCULA
1  TION FOR A CONSTANT REACTIVITY///
225H K EFFECTIVE IS EQUAL TO F8.6///)
35  CONTINUE
      IF (NFP) 37,37,45
37  POW = G
      WRITE (3,39) POW

```

```

      GO TO 50
45     FLX = G
      WRITE (3,46) FLX
46     FORMAT (54H THE BURNUP WILL BE COMPUTED USING A CONSTANT FLUX OF 1
1PE16.5,11H N/CM2-SEC.////)
50     CONTINUE
      IF(NSP) 52,52,60
52     CONTINUE
      WRITE (3,53)
53     FORMAT (65H AVERAGE CROSS SECTIONS WILL BE EVALUATED AT K = 1 (ADJ
1USTING B2)////)
      GO TO 65
60     CONTINUE
      WRITE (3,61)
61     FORMAT (82H AVERAGE CROSS SECTIONS WILL BE EVALUATED AT K(T). NO
1ADJUSTMENT FOR CRITICALITY.////)
65     CONTINUE
      WRITE (3,67) EPSL
67     FORMAT (57H THE CONVERGENCE CRITERION ON THE MARCH-OUT ALGORITHM I
1S 1PE16.5////)
      G = NUMT
      G = G*H
      WRITE (3,70) NUMT, H, G, NPRT
70     FORMAT (31H TOTAL NUMBER OF TIME STEPS IS 15///
114H EACH STEP IS 1PE16.3,5H DAYS///
222H TOTAL BURNUP TIME IS 1PE16.3,5H DAYS///
328H OUTPUT WILL BE GIVEN EVERY 14,6H STEPS///)
      READ (2,140) LBS, (LBSN(I),I=1,5),LBL,(LBLN(I),I = 1,5)
140    FORMAT (12I6)
      IF(LBS) 170,170,141
141    CONTINUE
      WRITE (3,150) LBS, LBL,(LBSN(I),I=1,LBS)
150    FORMAT (24H THE BREEDING RATIO HAS 12,18H SOURCE TERMS AND 12,11H
1SINK TERMS///
239H      NUCLIDE NUMBERS FOR SOURCE TERMS 516)
      WRITE (3,160) (LBLN(I),I = 1,LBL)
160    FORMAT (39H0      NUCLIDE NUMBERS FOR SINK TERMS 516)
170    CONTINUE
      RETURN
      END

```

```

SUBFTC PRT1 LIST
SUBROUTINE PRINT1
  DIMENSION CHI(26), EL(26), ATW(40), SORCE(40), ALAM(20), FE(20),
1 HOL(12), HOLN(40), AN(20,2), DNM(20), DNMI(40), SM(15,26), DN(40)
2 , SIGR(26), S(40,15,26), ACS(40,4), PHI(26), FPHI(26)
  DIMENSION LD(20), LC(20), LF(20), LCN(20,4), LFN(20,7),
1 LBLN(5), LBSN(5)
  COMMON NUMT, NPRT, NBE, NNE, NXS, NG, NP, NXCM, NSP, NFP, NOP,
1 NCOST, NCAL, NBS, NBL
  COMMON H, HS, VOL, POW, FLX, EPSL, BSQ, AK, T, BR, TOTL, RS,
1 RA, RL, RF
  COMMON LD, LC, LF, LCN, LFN, LBSN, LBLN
  COMMON CHI, EL, ATW, SORCE, ALAM, FE, HOL, HOLN, DN, AN, DNM, DNMI, SM,
1 SIGR, S, ACS, PHI, FPHI
  WRITE (3,2)
2 FORMAT(108H1* * * * * ZERO DIMENSIONAL FUE
1 L CYCLE CALCULATION * * * * *//)
2 24H INPUT DATA LISTED BELOW//
3117H NUCLIDE NUMBER NAME ATM WEIGHT LAMBDA(DAYS-1) MEV
4/FISSION SOURCE ELEMENT FOR DECAY CAPTURE FISSION//)
  GG = 24. * 3600.
  DO 6J = 1, NBE
  KK = LC(J)
  KKK = LF(J)
  J = J
  WRITE (3,4) J, HOLN(J), ATW(J), ALAM(J), FE(J), LD(J),
1 (LCN(J,K), K=1, KKK), (LFN(J,K), K=1, KKK)
4 FORMAT (18,7H ,A6,1P3E16.4,17,11I3)
  WRITE (3,5)
5 FORMAT (1H0)
  ALAM(J) = ALAM(J)/GG
6 CONTINUE
  WRITE (3,8)
8 FORMAT (32H0 NON-BURNABLE NUCLIDES//)
  KK = NBE + 1
  DO 14 J = KK, NXS
  WRITE (3,10) J, HOLN(J), ATW(J)
10 FORMAT (18,7H ,A6,1PE16.4)
  WRITE (3,5)
14 CONTINUE
  IF(NP) 16,16,99
16 CONTINUE
  WRITE (3,20)
20 FORMAT (42H1 GROUP FISSION SOURCE ENERGY LIMIT//)
  DO 28 I = 1, NG
  WRITE (3,21) I, CHI(I), EL(I)
  WRITE (3,22)

```

```

22  FORMAT (1H /)
28  CONTINUE
    DO 38 K = 1,NXS
    WRITE (3,35) HOLN(K),K
35  FORMAT (32H1MICROSCOPIC CROSS SECTIONS FOR A6,21H          NUCLIDE NUM
1BER 12////
2111H GROUP  SIGF      SIGA      NUSIGF      SIGTR      SIGXG      G-1XG
3      G-2XG      G-3XG      G-4XG      G-5XG      G-6XG////)
    KKK = NXCM + 5
    IF (NXCM - 6) 41, 41, 43
43  CONTINUE
    DO 44 I = 1,NG
    WRITE (3,36) I, (S(K,J,I), J = 1,11)
    WRITE (3,37)
44  CONTINUE
    WRITE (3,40)
40  FORMAT (44H1GROUP  G-7XG      G-8XG      G-9XG      G-10XG////)
    DO 49 I = 1,NG
    WRITE (3,42) I, (S(K,J,I), J = 12, KKK)
42  FORMAT (13,4F10.3)
    WRITE (3,37)
49  CONTINUE
    GO TO 99
41  CONTINUE
    DO 38 I = 1, NG
    WRITE (3,36) I, (S(K,J,I),J=1,KKK)
36  FORMAT (13,11F10.3)
    WRITE (3,37)
37  FORMAT (1H /)
38  CONTINUE
99  CONTINUE
    RETURN
    END

```

SIRFTC AVXS LIST

SUBROUTINE AVXS

```

DIMENSION CHI(26), EL(26), ATW(40), SORCE(40), ALAM(20), FE(20),
1 HOL(12), HOLN(40), AN(20,2), DNM(20), DNMI(40), SM(15,26), DN(40)
2 , SIGR(26), S(40,15,26), ACS(40,4), PHI(26), FPHI(26)
DIMENSION LD(20), LC(20), LF(20), LCN(20,4), LFN(20,7);

```

```

1 LDLN(5), LBSN(5)

```

```

COMMON NUMT, NPRT, NBE, NNE, NXS, NG, NP, NXCM, NSP, NFP, NOP,

```

```

1 NCOST, NCAL, NBS, NBL

```

```

COMMON H, HS, VOL, POW, FLX, EPSL, BSQ, AK, T, BR, TOTL, RS,

```

```

1 RA, RL, RF

```

```

COMMON LD, LC, LF, LCN, LFN, LBSN, LBLN

```

```

COMMON CHI, EL, ATW, SORCE, ALAM, FE, HOL, HOLN, DN, AN, DNM, DNMI, SM,

```

```

1 SIGR, S, ACS, PHI, FPHI

```

C THIS ROUTINE CALCULATES SPECTRUM AVERAGED CROSS SECTIONS

```

KK = NXCM + 5

```

```

DO 999 I = 1, NG

```

```

DO 999 K = 1, KK

```

```

SM(K,I) = .0

```

```

DO 999 J = 1, NXS

```

```

999 SM(K,I) = SM(K,I) + DN(J)*S(J,K,I)

```

```

IF(NSP) 100,100,200

```

```

100 TEMK = AK

```

```

TEMB = BSQ

```

```

AK = 1.0

```

```

NCAL = 1

```

```

CALL CALC

```

```

AK = TEMK

```

```

BSQ = TEMB

```

```

GO TO 1

```

```

200 CONTINUE

```

```

IF(NOP-1) 220,210,220

```

```

210 NCAL = 0

```

```

CALL CALC

```

```

GO TO 1

```

```

220 NCAL = 1

```

```

CALL CALC

```

```

1 CONTINUE

```

```

SUM = .0

```

```

DO 4 I = 1, NG

```

```

4 SUM = SUM + PHI(I)

```

```

DO 99 J = 1, NXS

```

```

DO 11 K = 1, 4

```

```

11 ACS(J,K) = .0

```

```

DO 21 I = 1, NG

```

```

DO 20 K = 1, 3

```

```

20 ACS(J,K) = ACS(J,K) + PHI(I)*S(J,K,I)

```



```
DO 99 K= 1, 4  
ACS(J,K) = ACS(J,K)/SUM  
99 CONTINUE  
RETURN  
END
```

SIBETC WAR LIST

SUBROUTINE MARCH

DIMENSION CHI(26), EL(26), ATW(40), SORCE(40), ALAM(20), FE(20),
1 HOL(12), HOLN(40), AN(20,2), DNM(20), DNMI(40), SM(15,26), DN(40)
2 , SIGR(26), S(40,15,26), ACS(40,4), PHI(26), FPHI(26)

DIMENSION LD(20), LC(20), LF(20), LCN(20,4), LFN(20,7),
1 LBLN(5), LBSN(5)

COMMON NUMT, NPRT, NBE, NNE, NXS, NG, NP, NXCM, NSP, NFP, NOP,
1 NCOST, NCAL, NBS, NBL

COMMON H, HS, VOL, POW, FLX, EPSL, BSQ, AK, T, BR, TOTL, RS,
1 RA, RL, RF

COMMON LD, LC, LF, LCN, LFN, LBSN, LBLN

COMMON CHI, EL, ATW, SORCE, ALAM, FE, HOL, HOLN, DN, AN, DNM, DNMI, SM,

1 SIGR, S, ACS, PHI, FPHI

C THIS ROUTINE COMPUTES THE TIME DEPENDENT ISOTOPE CONCENTRATION

NCOV = 0

FLX = FLX * 10.**(-24)

C ESTIMATE OF N(T+1) USING EULERS METHOD

DO 20 J = 1, NBE

G = -(ALAM(J) + ACS(J,2)*FLX)* DN(J) + SORCE(J)

IF (LD(J)) 6,6,4

4 KK = LD(J)

G = G + ALAM(KK) * DN(KK)

6 IF (LC(J)) 12,12,8

8 KK = LC(J)

DO 10 K = 1, KK

JJ = LCN(J,K)

10 G = G + DN(JJ)*ACS(JJ,4)*FLX

12 IF (LF(J)) 18,18,14

14 KK = LF(J)

DO 16 K = 1, KK

JJ = LFN(J,K)

16 G = G + DN(JJ)*ACS(JJ,1)*FLX

18 AN(J,1) = DN(J) + HS*G

20 CONTINUE

30 CONTINUE

C ESTIMATE OF N(T+1) USING MODIFIED EULER EQUATION

DO 40 J = 1, NBE

GG = (AN(J,1) + DN(J))*.5

G = -(ALAM(J) + ACS(J,2)*FLX)*GG + SORCE(J)

IF (LD(J)) 32,32,32

32 KK = LD(J)

G = G + ALAM(KK)*(DN(KK)+AN(KK,1)) *.5

33 IF (LC(J)) 36,36,34

34 KK = LC(J)

DO 35 K = 1, KK

```

36 IF (LF(J)) 39,39,37
37 KK = LF(J)
DO 38 K = 1, KK
JJ = LFN(J, K)
38 G = G + ACS(JJ, 1) * FLX * (AN(JJ, 1) + DN(JJ)) / 2.
39 AN(J, 2) = DN(J) + HS * G
40 CONTINUE
C CHECK ON ACCURACY
DO 50 J = 1, NBE
IF (ABS((AN(J, 2) - AN(J, 1)) / AN(J, 1)) - EPSL) 50, 50, 60
50 CONTINUE
GO TO 100
60 NCOV = NCOV + 1
IF (NCOV - 100) 80, 80, 65
65 WRITE (3, 67)
67 FORMAT (28H1NO CONVERGENCE ON MARCH-OUT///
150H0 NUCLIDE DN(T) AN(T+1, 1) AN(T+1, 2)///)
DO 69 J = 1, NBE
WRITE (3, 68) J, DN(J), AN(J, 1), AN(J, 2)
68 FORMAT (17, 1P3E16.5)
69 CONTINUE
CALL EXIT
RETURN
80 CONTINUE
DO 85 J = 1, NBE
85 AN(J, 1) = AN(J, 2)
GO TO 30
100 CONTINUE
TOTL = TOTL + RL * HS * VOL * 1000.
FLX = FLX * 10. ** (24)
DO 106 J = 1, NBE
DN(J) = AN(J, 2)
106 DNM(J) = VOL * DN(J) * ATW(J) / .6023
RETURN
END

```

SIEFIC CALC LIST

```

SUBROUTINE CALC
  DIMENSION CHI(26), EL(26), ATW(40), SORCE(40), ALAM(20), FE(20),
1 HOL(12), HOLN(40), AN(20,2), DNM(20), DNMI(40), SM(15,26), DN(40)
2 , SIGR(26), S(40,15,26), ACS(40,4), PHI(26), FPHI(26)
  DIMENSION LD(20), LC(20), LF(20), LCN(20,4), LFN(20,7),
1 LBLN(5), LBSN(5)
  COMMON KUMT, KPRT, NBE, NNE, NXS, NG, NP, NXCM, NSP, NFP, NOP,
1 NCOST, NCAL, NBS, NBL
  COMMON H, HS, VOL, POW, FLX, EPSL, BSC, AK, T, BR, TOTL, RS,
1 RA, RL, RF
  COMMON LD, LC, LF, LCN, LFN, LBSN, LBLN
  COMMON CHI, EL, ATW, SORCE, ALAM, FE, HOL, HOLN, DN, AN, DNM, DNMI, SM,
1 SIGR, S, ACS, PHI, FPHI
C THIS ROUTINE COMPUTES THE NEUTRON SPRECTRUM
  EPS = .000001
  IF (NCAL) 2,2,1
1 CONTINUE
  ITER = 0
2 CONTINUE
  KK = NXCM + 5
  DO 101 I = 1, NG
    SIGR(I) = .0
  DO 100 K = 6, KK
    NN = I + K - 5
    IF (NN - NG) 100,100,101
100 SIGR(I) = SIGR(I) + SM(K,NN)
101 CONTINUE
106 CONTINUE
  ITER = ITER + 1
  DO 199 I = 1, NG
    SUM = .0
    DO 111 N=1, NXCM
      KKK = I - N
      KK = N + 5
      IF(KKK) 112,112,111
111 SUM = SUM + PHI(KKK)*SM(KK,I)
112 CONTINUE
199 PHI(I) = (100.*CHI(I) + SUM)/(.33333*BSC/SM(4,I)+SM(2,I)+SIGR(I))
  RS = 0.
  RL = 0.
  RA = 0.
  DO 412 I = 1, NG
    FPHI(I) = PHI(I) * SM(3,I)
    RS = RS + FPHI(I)

```

```

441  TAK = RS/(RL + RA)
      IF (ABS(AK - TAK) - EPS) 512,512,601
601  IF (ITER - 200) 605,605,602
602  WRITE (3,603)
603  FORMAT (27HNO CONVERGENCE ON SPECTRUM)
      GO TO 511
605  G = .0
      DO 611 I = 1,NG
611  G = G + PHI(I)/(3.*SM(4,I))
      BSG = (RS/AK - RA)/G
      GO TO 106
511  AK = RS/(RL + RA)
512  CONTINUE
      RETURN
      END

```

```

SUBROUTINE PRINT
  DIMENSION CHI(26), EL(26), ATW(40), SORCE(40), ALAM(20), FE(20),
1 HOL(12), HOLN(40), AN(20,2), DNM(20), DNMI(40), SM(15,26), DN(40)
2 , SIGR(26), S(40,15,26), ACS(40,4), PHI(26), FPHI(26)
  DIMENSION LD(20), LC(20), LF(20), LCN(20,4), LFN(20,7),
1 LBLN(5), LBSN(5)
  COMMON NUMT, NPRT, NBE, NNE, NXS, NG, NP, NXCM, NSP, NFP, NOP,
1 NCOST, NCAL, NBS, NBL
  COMMON H, HS, VOL, POW, FLX, EPSL, BSQ, AK, T, BR, TOTL, RS,
1 RA, RL, RF
  COMMON LD,LC,LF,LCN,LFN,LBSN,LBLN
  COMMON CHI,EL,ATW,SORCE,ALAM,FE,HOL,HOLN,DN,AN,DNM,DNMI,SM,
1 SIGR,S,ACS,PHI,FPHI
C THIS ROUTINE PRINTS ALL APPROPRIATE INFORMATION AT ANY TIME DURING
C THE PROGRAM
  WRITE (3,2)
2 FORMAT (1H1)
  WRITE (3,4) (HOL(I),I=1,12)
4 FORMAT (12A6)
  WRITE (3,6) T, AK, POW, FLX, BR, BSQ
6 FORMAT (1H0////27H TIME = F8.2,9H DAY S///
115H K EFFECTIVE = 1PE16.6//
215H TOTAL POWER = 1PE16.6,4H MWT//
315H NEUTRON FLUX = 1PE16.6,10H N/CM2-SEC//
418H BREEDING RATIO = 1PE16.6//
522H GEOMETRIC BUCKLING = 1PE16.6,5H CM-2/////))
C WE WILL NOW PRINT FUEL INVENTORY
  WRITE (3,10)
10 FORMAT (47H * * * F U E L I N V E N T O R Y * * *///
192H NUCLIDE NAME ATOM DENSITY(CM-3) CURRENT LOAD(KG) I
2INITIAL LOAD(KG) CHANGE(KG)///)
  DO 20 J=1, NBE
  DNM (J) = VOL*DN(J)*ATW(J)/.6023
  G = DNM(J) - DNMI(J)
  WRITE (3,12) J, HOLN(J), DN(J), DNM(J), DNMI(J), G
  WRITE (3,14)
12 FORMAT (16,5H ,A6,1PE19.5,1PE20.5,1PE18.5,1PE16.5)
14 FORMAT (1H /)
20 CONTINUE
  WRITE (3,30)
30 FORMAT (1H1)
  WRITE (3,32) (HOL(I),I=1,12)
32 FORMAT (12A6)
  WRITE (3,34) RS,RF,RA,RL,TOTL
34 FORMAT (1H0////45H * * * N U C L E A R D A T A * * *///
140H NEUTRON SOURCE RATE(PER CC PER SEC) = 1PE16.5//
240H NEUTRON FISSION RATE(PER CC PER SEC) = 1PE16.5//

```

```

343H NEUTRON ABSORPTION RATE(PER CC PER SEC) = 1PE16.5//
443H NEUTRON LEAKAGE RATE(PER CC PER SEC) = 1PE16.5//
535H TOTAL NEUTRON LEAKAGE FROM CORE = 1PE16.5////
673H GROUP      FLUX(PER CM2 PER SEC)      NU*SIGF*FLUX      SIGF*FLUX
7 INTG FLUX//)
P = .0
DO 40 I = 1, NG
G = PHI(I) * SM(1,I)
P = P + PHI(I)/FLX
WRITE (3,36) I, PHI(I), FPHI(I),G,P
36  FORMAT (I4,1PE24.5,1P3016.6)
40  CONTINUE
IF (NP) 300,300,500
300  CONTINUE
WRITE (3,301)
301  FORMAT (1H1)
WRITE (3,302) (HCL(I),I=1,12)
302  FORMAT (12A6)
WRITE (3,304)
304  FORMAT (1H0///60H THE MACROSCOPIC CROSS SECTIONS ARE GIVEN IN THE
1TABLE BELOW//)
2112H GROUP  SIGF      SIGA      NUSIGF      SIGTR      SIGXG      G-1XG
3      G-2XG      G-3XG      G-4XG      G-5XG      G-6XG//)
KK = NXCM + 5
IF(NXCM - 6) 341, 341, 343
343  CONTINUE
DO 354 I = 1, NG
WRITE (3, 306) I, (SM(K,I), K = 1, 11)
WRITE (3, 308)
354  CONTINUE
390  FORMAT (44H1GROUP  G-7XG      G-8XG      G-9XG      G-10XG//)
WRITE (3, 390)
DO 349 I = 1, NG
352  FORMAT (I4, 1P4E10.2)
WRITE (3, 352) I, (SM(K,I), K = 12, KK)
WRITE (3, 308)
349  CONTINUE
GO TO 399
341  CONTINUE
DO 310 I = 1, NG
WRITE (3,306) I, (SM(K,I),K=1, KK)
306  FORMAT (I4,1P11E10.2)
WRITE (3,308)
308  FORMAT (1H /)
310  CONTINUE
399  CONTINUE
WRITE (3,320)
320  FORMAT (1H1)

```

```

      WRITE (3,330) (HOL(I),I=1,12)
330  FORMAT (12A6)
      WRITE (3,340)
340  FORMAT (1H0////83H      * * *      S P E C T R U M      A V E R A G E D
10 R O S S      S E C T I O N S      * * *////
27CH NUMBER      NAME      SIGMA F      SIGMA ABS      NU SIGMA F      SI
3GMA CAP//)
      DO 350 J=1,NXS
      WRITE (3,342) J, HOLN(J),(ACS(J,K), K=1,4)
      WRITE (3,344)
342  FORMAT (I5,4H      A6,1P4E14.4)
344  FORMAT (1H /)
350  CONTINUE
500  CONTINUE
      RETURN
      END

```


SUBFC NORM LIST

SUBROUTINE NORM

```

DIMENSION CHI(26), EL(26), ATW(40), SORCE(40), ALAM(20), FE(20),
1 HOL(12), HOLN(40), AN(20,2), DNM(20), DNMI(40), SM(15,26), DN(40)
2 , SIGR(26), S(40,15,26), ACS(40,4), PHI(26), FPHI(26)
DIMENSION LD(20), LC(20), LF(20), LCN(20,4), LFN(20,7),
1 LBLN(5), LBSN(5)
COMMON NUNT, NPRT, RDE, RNE, RXS, NG, NP, NXCH, MSP, NFP, NOP,
1 NCOST, NCAL, NBS, NBL
COMMON H, HS, VOL, POW, FLX, EPSL, BSG, AK, T, BR, TOTL, RS,
1 RA, RL, RF
COMMON LD,LC,LF,LCN,LFN,LBSN,LBLN
COMMON CHI,EL,ATW,SORCE,ALAM,FE,HOL,HOLN,DN,AN,DNM,DNMI,SM,
1 SIGR,S,ACS,PHI,FPHI
IF (NSP) 2,2,50
2 CONTINUE
IF(NOP - 1) 22,21,22
21 NCAL = 0
CALL CALC
GO TO 50
22 NCAL = 1
CALL CALC
50 CONTINUE
IF (NFP) 100,100,52
52 CONTINUE
G = .0
DO 54 I= 1,NG
54 G = G + PHI(I)
G = FLX/G
RS = G * RS
RA = G * RA
RL = G * RL
RF = 0.
POW = 0.
DO 60 I= 1,NG
PHI(I) = G*PHI(I)
FPHI(I) = G * FPHI(I)
DO 60 J=1,NBE
RF = RF + DN(J)*S(J,1,I)*PHI(I)
60 POW = POW + DN(J)*S(J,1,I)*PHI(I)*FE(J)
POW = POW * VOL * 1000.*1.6*10.**(-19)
GO TO 200
100 CONTINUE
G = 0.
DO 110 I=1,NG
DO 110 J=1,NBE
110 G = G + DN(J)*S(J,1,I)*PHI(I)*FE(J)
G = G * VOL * 1000.*1.6*10.**(-19)

```

```

G = POW/G
RS = G * RS
RA = G * RA
RL = G * RL
RF = .0
FLX = .0
DO 130 I=1,NG
PHI(I) = G * PHI(I)
FLX = FLX + PHI(I)
FPHI(I) = G * FPHI(I)
DO 130 J= 1,NBE
130 RF = RF + DN(J)*S(J,1,I)*PHI(I)
200 CONTINUE
G = .0
GG = .0
DO 210 K=1,LBS
KK = LBSN(K)
210 G = G + DN(KK)*ACS(KK,4)
DO 220 K=1,LBL
KK = LBLN(K)
220 GG = GG + DN(KK)*ACS(KK,2)
BR = G/GG
RETURN
END

```

318FTC COST LIST

SUBROUTINE COST

```

DIMENSION CHI(26), EL(26), ATW(40), SORCE(40), ALAM(20), FE(20),
1 HOL(12), HOLR(40), AN(20,2), DRN(20), DNMI(40), SM(15,26), DI(40)
2 , SIGR(26), S(40,15,26), ACS(40,4), PHI(26), FPHI(26)
DIMENSION LD(20), LC(20), LF(20), LCR(20,4), LFN(20,7),
1 LGLL(5), LBSR(5)
COMMON NUNT, KPRT, KBL, NRE, RXS, RG, RP, NXCH, NSP, NFP, KOP,
1 KOST, KCAL, NBS, NBL
COMMON H, HS, VOL, POW, FLX, EPSL, BSC, AK, T, BR, TOTL, RS,
1 RA, RL, RF
COMMON LD,LC,LF,LCH,LFN,LBSR,LGLL
COMMON CHI,EL,ATW,SORCE,ALAM,FE,HOL,HOLR,DI,AN,DNR,DNMI,SM,
1 SIGR,S,ACS,PHI,FPHI
WRITE (3,2)
2 FORMAT (62H1* * * * * E S T I M A T E O F F U E L C O S T
1 * * * * * //)
WRITE (3,4) (HOL(I), I = 1,12)
4 FORMAT (12A6)
WRITE (3,20)
20 FORMAT (11HODIRT CHEAP)
RETURN
END

```

INTERNAL DISTRIBUTION

Copy Number

1	G. J. Busselman
2	F. G. Dawson
3	D. E. Deonigi
4	B. H. Duane
5	E. A. Eschbach
6	J. C. Fox
7	R. W. Hardie
8	R. A. Harris
9	P. L. Hofmann
10	P. L. Hofmann
11	C. W. Lindenmeier
12	W. W. Little
13-27	L. L. Maas
28	R. E. Peterson
29-33	300 Files

EXTERNAL DISTRIBUTION

34-38	DTIE
39	US AEC RLOO R. K. Sharp

 