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BNWL-84

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C R U N C H - A GROUP COLLAPSING
CODE FOR FAST REACTOR ANALYSIS

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April, 1965

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PACIFIC NORTHWEST LABORATORY

RICHLAND, WASHINGTON

Operated By

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For The

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

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

BNWL-84

C R U N C H - A GROUP COLLAPSING CODE
FOR FAST REACTOR ANALYSIS

by

W. W. Little, Jr.

R. W. Hardie

April 29, 1965

Pacific Northwest Laboratory
Richland, Washington

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

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SUMMARY

This paper describes a group collapsing code for condensing fast reactor cross section data. The code uses the fundamental mode spectrum as a weighting function in computing few group cross sections. The input cross sections can have up to 26 energy groups with 10 downscattering elements. The collapsed set can contain any number of groups ($1 \leq n \leq \text{number in input set}$).

A FORTRAN IV source deck listing, input information, and a sample case are given in the appendices.

I. INTRODUCTION

Fast reactor cross sections are often compiled in sets consisting of many energy groups — the number of groups usually lying between 16 and 26. For a large class of design analysis problems, such as fuel cycle and multi-dimensional analyses, many group models are both unnecessary and costly. The CRUNCH code is designed to rapidly compute few group cross sections for such analyses.

II. OUTLINE OF CODE

Before presenting the details of the mathematical model, it is perhaps helpful to the reader to see an overview of the code logic. Such a picture is shown in Figure 2.1.

Although Figure 2.1 is almost self-explanatory, we shall follow through one complete cycle. First, the program reads and prints the multigroup cross section data. For each energy group, the input data must be arranged in the format:

$$\sigma_f \quad \sigma_a \quad v\sigma_f \quad \sigma_{tr} \quad \sigma_{gg} \quad \sigma_{g-1-g} \quad \sigma_{g-2-g} \quad \dots,$$

where g is the group index. See Appendix B for details.

After reading the data for the new case, the code computes and prints the fundamental mode energy spectrum. The neutronics model for this calculation is discussed in Section III.

Using the fundamental code flux distribution as a weighting function, the program now calculates, prints, and punches the collapsed group cross sections. A number of options are available for computing σ_{tr} . These options are presented in Section IV.

Figure 2.1. Schematic Diagram of CRUNCH Code.

Finally, for additional information and cross checking, the code recomputes and prints the fundamental mode spectrum using the collapsed cross section data. If certain options are used for the calculation of σ_{tr} , the initial neutron balance and the few group neutron balance will be identical.

III. NEUTRONICS MODEL

The flux spectrum is obtained from a modification of the well-known diffusion equation

$$\begin{aligned} \frac{1}{3\Sigma_i^{tr}} \nabla^2 \phi_i - \Sigma_i^a \phi_i - \Sigma_i^r \phi_i \\ + X_i \sum_{j=1}^G (\nu \Sigma^f)_j \phi_j + \sum_{j=1}^{i-1} \Sigma(j \rightarrow i) \phi_j = 0, \end{aligned} \quad (3.1)$$

where: ϕ_i = flux in group i,

G = total number of energy groups,

X_i = fission source in group i,

Σ_i^{tr} = transport cross section for group i,

Σ_i^r = removal cross section for group i

$$= \sum_{j=i+1}^G \Sigma(i \rightarrow j),$$

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

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

We now assume that the leakage can be simulated by the fundamental mode approximation

$$\nabla^2 \phi_i = -B^2 \phi_i . \quad (3.2)$$

Placing (3.2) into (3.1), we find that

$$\phi_i = \frac{X_i \sum_{j=1}^G (\nu \Sigma^f)_j \phi_j + \sum_{j=1}^{i-1} \Sigma(j \rightarrow i) \phi_j}{\frac{B^2}{3\Sigma_i^{tr}} + \Sigma_i^a + \Sigma_i^r} . \quad (3.3)$$

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

$$\sum_{i=1}^G (\nu \Sigma^f)_i \phi_i = 1 . \quad (3.4)$$

Substituting (3.4) into (3.3), the final expression for flux becomes

$$\phi_i^{\eta+1} = \frac{X_i + \sum_{j=1}^{i-1} \Sigma(j \rightarrow i) \phi_j^{\eta}}{\frac{B^2}{3\Sigma_i^{tr}} + \Sigma_i^a + \Sigma_i^r} , \quad (3.5)$$

where η is the iteration index.

Note that for a given buckling, B^2 , (3.5) can be solved explicitly for the multigroup flux. The buckling, in turn, can be determined from the effective multiplication constant. From the life cycle definition of k_{eff} ,

$$k_{eff} = \frac{\sum_{i=1}^G (\nu \Sigma^f)_i \phi_i^{\eta+1}}{\sum_{i=1}^G (\nu \Sigma^f)_i \phi_i^{\eta}} \quad (3.6)$$

or, from (3.4)

$$k_{\text{eff}} = \sum_{i=1}^G (\nu \Sigma^f)_i \phi_i^{\eta+1} . \quad (3.7)$$

By iteration, the buckling is computed so that the flux found from (3.5) agrees with the normalization given by (3.7).

IV. CROSS SECTION AVERAGING PROCEDURE

For convenience, let:

K = index for new collapsed energy group,

ψ_K = total flux in group K ,

$\bar{\sigma}_K$ = average microscopic cross section for the K^{th} collapsed group.

We now define ψ_K by the equation

$$\psi_K = \sum_{i=1}^K \phi_i , \quad (4.1)$$

where the symbol $\sum_{i=1}^K$ denotes a sum over the original microscopic energy groups comprising the K^{th} collapsed group.

For each isotope and collapsed energy group, the fission, capture, and neutron source ($\nu \sigma_f$) cross sections are obtained using the recipe

$$\bar{\sigma}_K = \frac{\sum_{i=1}^K \sigma_i \phi_i}{\psi_K} . \quad (4.2)$$

Observe that the rate of fission, capture, and neutron emission is unaltered by this averaging procedure.

Unfortunately, the neutron leakage rate will not be conserved if the collapsed transport cross sections are computed by the simple recipe given above. This difficulty arises, of course, from the inverse dependence of leakage on atom density.

For convenience, four options are available in the code for computing the average transport cross sections. These are:

- (1) Reciprocal weighting, defined to be

$$\bar{\sigma}_K^{\text{tr}} = \frac{\psi_K}{\sum_i \frac{\phi_i}{\sigma_i^{\text{tr}}}} ; \quad (4.3)$$

- (2) Normalized reciprocal weighting, defined by the equation

$$\bar{\sigma}_K^{\text{tr}} = \gamma_K \frac{\psi_K}{\sum_i \frac{\phi_i}{\sigma_i^{\text{tr}}}} , \quad (4.4)$$

where

$$\gamma_K = \frac{\psi_K}{\sum_N (\bar{\sigma}_K^{\text{tr}}) \sum_i \left(\frac{\phi_i}{\sum_N \bar{\sigma}_i^{\text{tr}}} \right)} . \quad (4.5)$$

The symbol $\sum_N$ denotes a sum over the isotopes in the mixture.

- (3) Linear weighting, defined by the equation

$$\bar{\sigma}_K^{\text{tr}} = \frac{\sum_i \sigma_i^{\text{tr}} \phi_i}{\psi_K} ; \quad (4.6)$$

- (4) Normalized linear weighting, defined as

$$\bar{\sigma}_K^{\text{tr}} = \gamma_K \frac{\sum_i \sigma_i^{\text{tr}} \phi_i}{\psi_K} , \quad (4.7)$$

where, as before, the normalization factor is obtained from Eq. (4.5).

Observe that since the neutron leakage rate is conserved in the normalized options, the total neutron balance using the initial and few group cross sections will be identical.

The collapsed downscattering cross sections are computed by matching the slowing down currents. If we define $\bar{\sigma}(K \rightarrow J)$ as the collapsed scattering probability from group K to group J, a current balance gives the algorithm

$$\bar{\sigma}(K \rightarrow J) \psi_K = \sum_{i=1}^K \sum_{j=1}^J \sigma(i \rightarrow j) \phi_i, \quad (4.8)$$

or

$$\bar{\sigma}(K \rightarrow J) = \frac{\sum_{i=1}^K \sum_{j=1}^J \sigma(i \rightarrow j) \phi_i}{\psi_K}. \quad (4.9)$$

The final cross section computed is the in-group scattering probability, $\bar{\sigma}^{gg}$. For each isotope and energy group, $\bar{\sigma}^{gg}$ is calculated using the expression

$$\bar{\sigma}_K^{gg} = \bar{\sigma}_K^{tr} - \bar{\sigma}_K^a - \sum_{j=1}^J \bar{\sigma}(K \rightarrow j). \quad (4.10)$$

It should be mentioned that $\bar{\sigma}^{gg}$, as defined and used in this paper, is actually superfluous information. It is carried along as a dummy in the CRUNCH code because it is required input in many neutron transport codes.

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APPENDIX A

FORTRAN IV SOURCE DECK LISTING

```

COMMON      NG,NXS,NXCM,NP,NAVX,NOP, NXCMP,NV
COMMON      CHI,S,DN,SM,PDI,FPHI,EL,HOL,SIGR,POWER, SG, NPN, CHIP
COMMON      BSO,EPS,AK,RS,RA,RL,B1,B2, CXX, NAX
DIMENSIONCHI(26),S(20,15,26),DN(20),SM(15,26),PHI(26),FPHI(26),
1  EL(26),HOL(12),SIGR(26) , POWER(26), NPN(26), SG(20, 15, 26) ,
2  CHIP(26), NV(26), CXX(26)
CALL INPUT
BSO=.0

1  CONTINUE

CALL START
CALL CALC
CALL PRINT

2  CALL AXCS

JJ = NXCMP + 5
DO 100 I = 1, NP
DO 90 K = 1, NXS
DO 90 J = 1, JJ
G = S(K, J, I)
S(K, J, I) = SG(K, J, I)

9  0  SG(K, J, I) = G

CXX(1) = CHI(1)
CHI(1) = CHIP(1)

1  00  CONTINUE

NMG = NG
NXX = NXCM
NG = NP
NXCM = NXCMP
DO 101 I = 1, NG
KK = NXCM + 5
DO 101 K = 1, KK
SM(K, I) = 0.0
DO 101 J = 1, NXS

1  01  SM(K, I) = SM(K, I) + DN(J)*S(J, K, I)

CALL CALC
CALL PRINT
NG = NMG

```

-11-

NUMBER

F O R T R A N S T A T E M E N T

* IDENTIFICATION

NXCM = NXX

DO 105 I = 1, NP

DO 102 K = 1, NXS

DO 102 J = 1, JJ

1 02 S(K, J, I) = SG(K, J, I)
1 05 CHI(I) = CXX(I)

GO TO 1

END

```

SUBROUTINE INPUT
COMMON      NG,NXS,NXCM,NP,NAVX,NOP, NXCMP,NV
COMMON  CHI,S,DN,SM,PHI,FPHI,EL,HOL,SIGR,POWER, SG, NPN, CHIP
COMMON  BSQ,EPS,AK,RS,RA,RL,B1,B2
DIMENSION CHI(26),S(20,15,26),DN(20),SM(15,26),PHI(26),FPHI(26),
1 EL(26),HOL(12),SIGR(26), POWER(26), NPN(26), SG(20, 15, 26),
2 CHIP(26), NV(26), CXX(26)
READ (2,2)      NG,NXS,NXCM

2      FORMAT (3I6)

      READ (2,4)      (CHI(I),I =1,NG)
      READ (2,4)      (EL(I),I=1,NG)

4      FORMAT (6E12.4)

      KK=NXCM+5
      DO 11 J=1,NXS
      DO 11 I=1,NG
      READ (2,6)      KD,(S(J,K,I),K=1,8)

6      FORMAT (11,E8.3,7E9.4)

      READ (2,6)      KD,(S(J,K,I),K=9,KK)

1 1      CONTINUE

      WRITE (3,13)      NG,NXS

1 3      FORMAT (39H1.....C R U N C H ...../////

1 32H THE NUMBER OF ENERGY GROUPS IS 12///
2 27H THE NUMBER OF NUCLIDES IS 12///
3 42H   GROUP      FISSION SOURCE      ENERGY LIMIT///)
      DO 18 I=1,NG
      WRITE (3,15)      I,CHI(I),EL(I)
      WRITE (3,16)

1 5      FORMAT (18,1P2E16.5)
1 6      FORMAT(1H /)
1 8      CONTINUE

2 1      CONTINUE

      KK=0
      DO 81 K=1,NXS

```


NUMBER

F O R T R A N S T A T E M E N T

* IDENTIFICATION

```

WRITE (3,25) K
2 5  FORMAT (40H1MICROSCOPIC CROSS SECTIONS FOR NUCLIDE 127777
      1 112H GROUP SIGF SIGA NUSIGF SIGTR SIGXG G-1XG
      2 G-2XG G-3XG G-4XG G-5XG G-6XG777)
      KKK=NXCM+5
      IF(KKK - 11) 26, 26, 100
2 6  DO 28 I = 1, NG
      WRITE (3,27) I, ( S(K,J,I),J=1,KKK)
2 7  FORMAT (13,11F10.3)
      WRITE (3,1)
1  FORMAT (1H /)
2 8  CONTINUE
      GO TO 81
1 00  CONTINUE
      DO 128 I = 1, NG
      WRITE (3,27) I, (S(K, J, I ), J = 1, 11)
      WRITE (3, 27) I, (S(K, J, I), J = 12, KKK)
      WRITE (3,1)
1 28  CONTINUE
8 1  CONTINUE
9 9  CONTINUE
      RETURN
      END

```

\$ IREF C START LIST

```

SUBROUTINE START
COMMON      NG,NXS,NXCM,NP,NAVX,NOP, BXCMP,NV
COMMON      CHI,S,DN,SM,PHI,FPHI,EL,HOL,SIGR,POWER, SG, NPN, CHIP
COMMON      BSQ,EPS,AK,RS,RA,RL,B1,B2, CXX, NXA
DIMENSIONCHI(26),S(20,15,26),DN(20),SM(15,26),PHI(26),FPHI(26),
1 EL(26),HOL(12),SIGR(26), POWER(26), NPN(26), SG(20, 15, 26),
2 CHIP(26), NV(26), CXX(26)
READ (2,3)      (HOL(I),I=1,12)
WRITE (3,1)

1      FORMAT (1H1)

      WRITE (3,3)      (HOL(I),I=1,12)

3      FORMAT (12A6)

      READ (2,5)  NOP, NP, NAVX,G

5      FORMAT(3I6,1E12.4)

      READ (2,6)  (NPN(K), K = 1, NP)

6      FORMAT (12I6)

      EPS = .000001
      IF(NOP) 7,7,21

7      IF(G) 11,11,9

9      BSQ = G

1 1      CONTINUE

      WRITE (3,13)      BSQ,EPS

1 3      FORMAT ( 1H0//39H.....THIS IS A REACTIVITY CALCULATION///

1      17H THE BUCKLING IS 1PE16.6///
2      30H THE CONVERGENCE CRITERION IS 1PE16.6////////)
      GO TO 49

2 1      AK=G

      WRITE (3,23)      AK,EPS

2 3      FORMAT (1H0//38H.....THIS IS A BUCKLING CALCULATION///

```

NUMBER	FORTRAN STATEMENT	IDENTIFICATION
1	25H K EFFECTIVE IS EQUAL TO F8.6///	
2	30H THE CONVERGENCE CRITERION IS 1PE16.6////////)	
4 9	CONTINUE	
	READ (2,51) (DN(J),J=1,NXS)	
5 1	FORMAT (6E12.4)	
	WRITE (3,53)	
5 3	FORMAT (39H0THE NUCLIDE DENSITIES ARE LISTED BELOW///	
1	19H ELEMENT ATOMS/CC///)	
	WRITE (3,55) (J,DN(J),J=1,NXS)	
5 5	FORMAT(I5,1PE16.6)	
	DO 61 I = 1,NG	
	KK=NXCM +5	
	DO 61 K = 1,KK	
	SM(K,I) =.0	
	DO 61 J = 1,NXS	
6 1	SM(K,I) = SM(K,I) + DN(J)*S(J,K,I)	
7 1	CONTINUE	
	WRITE (3, 403)	
4 03	FORMAT (1H1)	
	WRITE(3,404) (HOL(I), I = 1,12)	
4 04	FORMAT (12A6)	
	WRITE (3,73)	
7 3	FORMAT (60H0THE MACROSCOPIC CROSS SECTIONS ARE GIVEN IN THE TABLE	
1	BELOW///	
1	112H GROUP SIGF SIGA NUSIGF SIGTR SIGXG G-1XG	
2	G-2XG G-3XG G-4XG G-5XG G-6XG///)	
	IF(NXCM - 6) 178, 178, 101	
1 01	DO 191 I = 1, NG	

NUMBER

F O R T R A N S T A T E M E N T

* IDENTIFICATION

WRITE (3,74) I, (SM(K,I), K=1,11)
WRITE (3,74) I, (SM(K,I), K = 12, KK)
WRITE (3, 75)

1 91

CONTINUE

GO TO 99

1 78

DO 78 I=1,NG

WRITE (3,74) I, (SM(K,I),K=1,KK)

7 4

FORMAT (I4, 1P11E10.2)

WRITE (3,75)

7 5

FORMAT (1H /)

7 8

CONTINUE

9 9

CONTINUE

RETURN
END

\$ IBF: C CALC LIST

```

SUBROUTINE CALC
COMMON      NG,NXS,NXCM,NP,NAVX,NOP, NXCMP,NV
COMMON      CHI,S,DN,SM,PHI,FPHI,EL,HOL,SIGR,POWER, SG, NPN, CHIP
COMMON      BSQ,EPS,AK,RS,RA,RL,B1,B2, CXX, NXX
DIMENSION CHI(26),S(20,15,26),DN(20),SM(15,26),PHI(26),FPHI(26),
1 EL(26),HOL(12),SIGR(26), POWER(26), NPN(26), SG(20, 15, 26),
2 CHIP(26), NV(26), CXX(26)
IF(NOP) 99,99,3

3      BSQ =.0001

      ITER = 0

9  9      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

1  00      SIGR(I) = SIGR(I) + SM(K,NN)
1  01      CONTINUE
1  06      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

1  11      SUM = SUM + PHI(KKK)*SM(KK, I )
1  12      CONTINUE
1  99      PHI(I) = (100.*CHI(1)+ SUM)/(.33333*BSQ/SM(4,1)+SM(2,1)+ SIGR(I))

      CONTINUE
      RS=0.
      RL=0.
      RA=0.
      DO 412 I =1,NG
      FPHI(I) = PHI(I)*SM(3,I)
      RS =RS + FPHI(I)
      RL =RL + BSQ*PHI(I)/(3.*SM(4,1))

```

NUMBER	FORTRAN STATEMENT	* IDENTIFICATION
4 12	RA = RA + PHI(1)*SM(2,1)	
	IF(NOP) 511,511,441	
4 41	TAK = RS/(RL+RA)	
	IF(ABS (AK-TAK)-EPS) 511,511,601	
6 01	IF(ITER - 200) 605,605, 511	
6 05	G=0.	
	DO 611 I =1,NG	
6 11	G = G + PHI(1)/(3.*SM(4,1))	
	BSQ =(RS/AK - RA)/G	
	GO TO 106	
5 11	AK = RS/(RL+RA)	
5 12	CONTINUE	
	RETURN	
	END	

5 1857 C PRINT LIST

SUBROUTINE PRIN

COMMON NG,HXS,NAQT,NP,NAQX,NOP,NAQMP,NV

COMMON CH1,SDN,SM,FPH1,FPHI,EL,HOL,STGR,POWER,SG,NPN,CHIP

COMMON BSQ,EP,AK,RS,RA,RL,B1,B2,CAX,XXX

DIMENSIONCH1(26),S(20,15,26),DN(26),SM(15,26),PHI(26),FPHI(26),

1 EL(26),HOL(12),STGR(26),POWER(26),NPHI(26),SG(20,15,26),

2 CHIP(26),NV(26),CAX(26)

WRITE (3,1)

1 FORMAT (1H1)

WRITE(3,2) (HOL(I), I = 1, 12)

2 FORMAT (12A6)

WRITE (3,3) AK,BSQ,RS,RA,RL

3 FORMAT (39H0.....FINAL RESULTS BELOW////

1 15H K EFFECTIVE = 1PE16.6//

2 16H BUCKLING(B2) = 1PE16.6//

3 15H SOURCE RATE = 1PE16.6//

4 19H ABSORPTION RATE = 1PE16.6//

5 16H LEAKAGE RATE = 1PE16.6////

6 58H GROUP FLUX NO*SIGF*FLUX SIGF*FLUX INTG FLUX//)

PP = .0

SUM = 0.

DO 18 I=1,NG

1 8 SUM = SUM + PHI(I)

WWWL=0.0

DO 11 I =1,NG

WWWL=WWWL+PHI(I)/SUM

POWER(I) = PHI(I)*SM(1,I)

PP = PP + POWER(I)

WRITE (3,5) 1,PHI(I),FPHI(I),POWER(I),WWWL

5 FORMAT (13, 1P4E14.5)

1 1 CONTINUE

WRITE (3,17)

1 7 FORMAT (1H0)

WRITE (3,19) SUM

NUMBER

F O R T R A N S T A T E M E N T

* IDENTIFICATION

```
1 9      FORMAT (19H THE TOTAL FLUX IS IPE16.477)
        WRITE (3,190) PP
1 90     FORMAT (21H THE FISSION RATE IS IPE16.477)
        RETURN
        END
```


\$ IBFI C AXCS LIST

SUBROUTINE AXCS

COMMON NG,NXS,NXCM,NP,NAVX,NOP,NXCMP,NV

COMMON CHI,S,DN,SM,PHI,FPHI,LL,HOL,SIGR,POWER,SG,NPN,CHIP

COMMON BSQ,EPS,AK,RS,RA,RL,B1,B2,CXX,NXX

DIMENSIONCHI(26),S(20,15,26),DN(20),SM(15,26),PHI(26),FPHI(26),

1 EL(26),HOL(12),SIGR(26),POWER(26),NPN(26),SG(20,15,26),

2 CHIP(26),NV(26),CXX(26)

C MISCELLANEOUS DATA

NXCMP = NXCM

IF (NXCM-NP+1) 130,130,110

1 10 NXCMP = NP - 1

1 30 CONTINUE

I = 0

DO 160 K = 1,NP

FPHI(K)=.0

CHIP(K) = .0

KK = NPN(K)

DO 160 L = 1,KK

I = I + 1

CHIP(K) = CHIP(K) + CHI(I)

1 60 FPHI(K) = FPHI(K) + PHI(I)

WRITE(3,1)

1 FORMAT(1H1)

WRITE(3,2) (HOL(I),I=1,12)

2 FORMAT (12A6)

WRITE (3, 699)

6 99 FORMAT (1H //)

WRITE(3,6) NP

6 FORMAT (50H DATA FOR CROSS SECTION AVERAGING LISTED BELOW///

2 42H THE NUMBER OF ENLARGED ENERGY GROUPS IS 13///)

IF (NAVX-2) 3,12,23

3 WRITE(3,4)

NUMBER	FORTRAN STATEMENT	* IDENTIFICATION
4	FORMAT (57HUSIGMA TRANSPORT IS CALCULATED USING RECIPROCAL WEIG	
1	HTING///)	
	GO TO 38	
1 2	WRITE (3,14)	
1 4	FORMAT (68HUSIGMA TRANSPORT IS CALCULATED USING NORMALIZED RECIP	
1	ROCAL WEIGHTING///)	
	GO TO 38	
2 3	IF (NAVX-3) 26,26,30	
2 6	WRITE (3,28)	
2 8	FORMAT (53HUSIGMA TRANSPORT IS CALCULATED USING LINEAR WEIGHTIN	
1	G///)	
	GO TO 38	
3 0	WRITE(3,31)	
3 1	FORMAT (64HUSIGMA TRANSPORT IS CALCULATED USING NORMALIZED LINE	
1	AR WEIGHTING///)	
3 8	CONTINUE	
	WRITE (3, 91)	
9 1	FORMAT (78HNEW GROUP OLD GROUPS PER NEW GROUP FISSION SOURC	
1	E FOR NEW GROUP FLUX///)	
	DO 94 K = 1, NP	
	WRITE (3, 92) K, NPN(K), CHIP(K), FPHI(K)	
9 2	FORMAT(I3,20H 13,20H E16.5,5	
1	H E16.5//)	
9 4	CONTINUE	
C	CALCULATION OF LINEAR AVERAGES	
	KK=0	
	II= 0	
	DO 260 K = 1, NP	
	II = II+KK	

NUMBER	F O R T R A N S T A T E M E N T	* IDENTIFICATION
	KK = NPN(K) DO 260 M = 1,NXS DO 260 MM =1,4 SG(M,MM,K) =.0 DO 260 L = 1, KK I = L + II	
2 60	SG(M,MM,K) = SG(M,MM,K) + S(M,MM,I)*PHI(I)/FPHI(K)	
C	CALCULATION OF SIGTR	
	IF (NAVX - 3) 300,400 ,380	
3 00	KK=0	
	II =0 DO 360 K=1,NP II = II + KK KK = NPN(K) DO 360 M=1,NXS SG(M,4,K) =.0 DO 360 L= 1,KK I = L + II	
3 60	SG(M,4,K) = SG(M,4,K) + PHI(I)/S(M,4,I)	
	DO 370 K=1,NP DO 370 M=1,NXS	
3 70	SG(M,4,K) = FPHI(K)/SG(M,4,K)	
	IF(NAVX -2) 400, 380,380	
3 80	I=0	
	WRITE (3, 991)	
9 91	FORMAT(58H1GROUP NORMALIZATION FACTOR FOR TRANSPORT CROSS SECTIO	
1	NS///) DO 390 K = 1, NP G = .0 DO 381 N=1,NXS	
3 81	G=G + DN(N)*SG(N,4,K)	

NUMBER	FORTRAN STATEMENT	* IDENTIFICATION
	GGG = FPHI(K)/G	
	KK = NPN(K)	
	GG=0.	
	DO 386 I=1, KK	
	I=I+1	
	G = .0	
	DO 384 N=1, NXS	
3 84	G = G + DN(N)*S (N,4,I)	
3 86	GG = GG + PHI(I)/G	
	DO 389 M =1, NXS	
3 89	SG(M,4,K) = GGG*SG(M,4,K)/GG	
	GXG = GGG/GG	
	WRITE (3, 992) K, GXG	
9 92	FORMAT(I6,16H ,1PE16.5)	
	WRITE(3,993)	
9 93	FORMAT(1H /)	
3 90	CONTINUE	
4 00	CONTINUE	
	CALL AXCS2	
	RETURN	
	END	

9 IBF C AXCS2 LIST

```

SUBROUTINE AXCS2
COMMON NG,NXS,NXCM,NP,NAVX,NOP,NXCMP,NV
COMMON CHI,S,DN,SM,PHI,FPHI,EL,HOL,SIGR,POWER,SG,NPN,CHIP
COMMON BSO,EPS,AK,RS,RA,RL,B1,B2,CXX,NXX
1 DIMENSIONCHI(26),S(20,15,26),DN(20),SM(15,26),PHI(26),FPHI(26),
2 EL(26),HOL(12),SIGR(26),POWER(26),NPN(26),SG(20,15,26),
CHIP(26),NV(26),CXX(26)

```

C CALCULATION OF SCATTERING MATRIX

```

LL = NXCMP + 5
DO 630 N=1,NXS
DO 630 K=1,NP
JJ = K + 5
IF (JJ - LL) 621, 621, 630

```

```

6 21 DO 619 L = JJ, LL
6 19 SG(N, L, K) = 0.
6 30 CONTINUE

```

```

IF(NXCMP) 700,700,640

```

```

6 40 CONTINUE

```

```

NV(1) = 1
DO 650 I=2,NP

```

```

6 50 NV(I) = NV(I-1) + NPN(I-1)

```

```

IL = NP-1
DO 690 N=1,NXS
DO 690 I=1,IL
JL = I+1
DO 690 J=JL,NP
MN = 5 + J - I
IF (J-I-NXCMP) 655,655,690

```

```

6 55 SG(N,MN,J) = .0

```

```

I4 = NPN(I)
DO 680 III=1,I4
II = NV(I) + III - 1
J4 = NPN(J)
DO 680 JJJ = 1,J4
JJ = NV(J) + JJJ - 1
NN = 5 + JJ - II

```

NUMBER

FORTRAN STATEMENT

* IDENTIFICATION

IF (JJ - 1) - NXCM) 670,670,680

6 70 SG(N,MN,J) = SG(N,MN,J) + S(N,NN,JJ) * PHI(11) / FPHI(1)
 6 80 CONTINUE
 6 90 CONTINUE

7 00 CONTINUE

0 CALCULATION OF SIGG

DO 760 N = 1, NXS
 DO 760 K = 1, NP
 G = .0
 DO 740 L = 1, NXCMP
 M = L + 5
 KK = K + L
 IF(KK - NP) 740, 740, 760

7 40 G = G + SG(N, M, KK)
 7 60 SG(N, 5, K) = SG(N, 4, K) - SG(N, 2, K) - G

DO 55 J=1, NXS
 WRITE (3,20)

2 0 FORMAT(1H1)
 WRITE (3,22) (HOL(I),I=1,12)

2 2 FORMAT(12A6)

4 0 WRITE (3,41) J
 4 1 FORMAT (85H0THE COLLAPSED GROUP MICROSCOPIC CROSS SECTIONS AR

1 E LISTED BELOW FOR NUCLIDE NUMBER 13777
 2 114H GROUP SIGF SIGA NUSIGF SIGTR SIGXG G-1X
 3 G G-2XG G-3XG G-4XG G-5XG G-6XG 777)
 KK = NXCMP + 5
 IF (NXCMP - 6) 49, 49, 45

4 5 CONTINUE

DO 48 I=1,NP
 WRITE (3,47) I, (SG(J,K,I),K=1,11)
 WRITE (3,47) I, (SG(J,K,I),K=12,KK)

4 7 FORMAT (I3,11F10.3)

NUMBER	F O R T R A N S T A T E M E N T	* IDENTIFICATION
	WRITE (3, 56)	
5 6	FORMAT(1H /)	
4 8	CONTINUE	
	GO TO 855	
4 9	CONTINUE	
	DO 53 I=1,NP	
	WRITE (3, 47) I, (SG(J, K, I), K = 1, KK)	
	WRITE (3, 56)	
5 3	CONTINUE	
8 55	CONTINUE	
	KD = 9	
	DO 54 I= 1,NP	
	PUNCH 554, KD, (SG(J,K,I), K=1,8)	
	IF(KK-9) 54,544,544	
5 44	PUNCH 554, KD, (SG(J,K,I), K= 9, KK)	
5 54	FORMAT (11, F8.3, 7F9.3)	
5 4	CONTINUE	
	PUNCH 555	
5 55	FORMAT (1H)	
5 5	CONTINUE	
	RETURN	
	END	

APPENDIX B

CODE INPUT INFORMATION

<u>Card</u>	<u>Column</u>	<u>Format</u>	<u>Remarks</u>
1	1-6	Fixed	Number of original energy groups (≤ 26).
1	7-12	Fixed	Number of isotopes (≤ 20).
1	13-18	Fixed	Number of groups in original downscattering matrix (≤ 10). (e.g., 5 for Hansen-Roach set, 6 for YOM set).
2	1-12	Floating	Fission source in group 1.
2	13-24	Floating	Fission source in group 2.
...			
3	1-12	Floating	Lower energy bound for group 1.
3	13-24	Floating	Lower energy bound for group 2.
...			
4	1	Fixed	One digit number or blank.
4	2-9	Floating	Cross section data for first group of first isotope. σ_f (barns)
4	10-18	Floating	σ_a
4	19-27	Floating	σ_f
4	28-36	Floating	σ_{tr}
4	37-45	Floating	σ_{gg}
4	46-54	Floating	σ_{g-1-g}
4	55-63	Floating	σ_{g-2-g}
4	64-72	Floating	σ_{g-3-g}
5	1	Fixed	One digit number or blank.
5	2-9	Floating	σ_{g-4-g}
5	10-18	Floating	σ_{g-5-g}
5	19-27	Floating	σ_{g-6-g}
...			

<u>Card</u>	<u>Column</u>	<u>Format</u>	<u>Remarks</u>
6			Same as above for second group of isotope 1. Repeat for all energy groups of isotope 1, then repeat cycle for all isotopes.
** 7	2-72	Hollerith	Identification card
8	1-6	Fixed	= 1; buckling calculation. The code computes the buckling corresponding to the value of k_{eff} given on the fourth word of this card. = 0; reactivity calculation. The code computes the reactivity corresponding to the buckling given on the fourth word of this card.
8	7-12	Fixed	Number of crunched groups.
8	13-18	Fixed	= 1; $\bar{\sigma}_{tr}$ calculated using reciprocal weighting. = 2; $\bar{\sigma}_{tr}$ calculated using normalized reciprocal weighting. = 3; $\bar{\sigma}_{tr}$ calculated using linear weighting. = 4; $\bar{\sigma}_{tr}$ calculated using normalized linear weighting.
8	19-30	Floating	Effective multiplication constant if this is a buckling calculation. Buckling (B^2) if this is a reactivity calculation.
9	1-6	Fixed	Number of original groups in first crunched group.
9	7-12	Fixed	Number of original groups in second crunched group.
...			
10	1-12	Floating	Atom density for isotope 1 ($\text{cm}^{-3} \times 10^{-24}$).
10	13-24	Floating	Atom density for isotope 2.
...			

** To run a series of cases, repeat from card 7 (ID card).

-30-

APPENDIX C

SAMPLE CASES

1	6	6	5			
.	204	.344	.168	.18	.09	.014
.	0	.0	.0	.0	.0	.0
3	000.	1400.	900.	400.	100.	17.
3	.	.55	.1	.03	.01	.003

.	001	.0004	.0001	.000025
---	-----	-------	-------	---------

9	1 .	900	1.930	6.612	4.250	1.050	0.	0.	0.	00009 609
9	0 .		0.							00009 709
9	1 .	950	2.000	6.026	4.500	1.420	0.200	0.	0.	00009 809
9	0 .		0.							00009 909
9	1 .	830	1.900	5.472	4.800	2.090	0.180	0.270	0.	00010 009
9	0 .		0.							00010 109
9	1 .	700	1.810	4.981	5.700	3.550	0.450	0.500	0.450	00010 209
9	0 .		0.							00010 309
9	1 .	670	1.840	4.810	8.400	6.510	0.290	0.300	0.350	00010 409
9	0 .	310	0.							00010 509
9	1 .	920	2.300	5.491	13.200	10.850	0.050	0.050	0.060	00010 609
9	0 .	050	0.040							00010 709
9	2 .	500	3.500	7.150	13.500	9.960	0.050	0.	0.	00010 809
9	0 .		0.							00010 909
9	4 .	200	6.700	12.012	16.700	9.960	0.040	0.	0.	00011 009
9	0 .		0.							00011 109
9	14 .	000	21.500	40.000	31.500	9.960	0.040	0.	0.	00011 209
9	0 .		0.							00011 309
9	6 .	000	10.000	17.200	20.000	9.940	0.040	0.	0.	00011 409
9	0 .		0.							00011 509
9	17 .	000	25.000	48.600	35.000	9.930	0.060	0.	0.	00011 609
9	0 .		0.							00011 709
9	4 .	000	7.000	11.400	17.000	9.940	0.070	0.	0.	00011 809
9	0 .		0.							00011 909
9	20 .	000	23.000	57.200	33.000	9.930	0.060	0.	0.	00012 009
9	0 .		0.							00012 109
9	80 .	000	123.000	229.000	133.000	9.920	0.070	0.	0.	00012 209
9	0 .		0.							00012 309
9	760 .	000	1210.000	2174.000	1220.000	9.940	0.080	0.	0.	00012 409
9	0 .		0.							00012 509
9	705 .	000	980.000	2016.300	990.000	10.000	0.060	0.	0.	00012 609
9	0 .		0.							00012 709

SA M PLE CASE FOR C R U M C H USING HANSEN-ROACH CROSS SECTIONS

1 4 1 1.0

4 3 5 4

2.24 9 28E-031.693000E-044.837150E-031.142250E-022.526300E-028.876190E-03

IDENT CUS485 XD
START TIME GL 06 35 00 CL 10 03 30 DT 04 28 65 C SHIFT JOB NUMBER CUS485 1 LOG 010317 CBN016113
\$EXECUTE IBJOB
IBJOB VERSION 4
\$IBJOB NOSOURCE
BEGIN EXECUTION

THE NUMBER OF ENERGY GROUPS IS 16

THE NUMBER OF NUCLIDES IS 6

GROUP	FISSION SOURCE	ENERGY LIMIT
1	2.04000E-01	3.00000E 03
2	3.44000E-01	1.40000E 03
3	1.68000E-01	9.00000E 02
4	1.80000E-01	4.00000E 02
5	9.00000E-02	1.00000E 02
6	1.40000E-02	1.70000E 01
7	-0.	3.00000E 00
8	-0.	5.50000E-01
9	-0.	1.00000E-01
10	-0.	3.00000E-02
11	-0.	1.00000E-02
12	-0.	3.00000E-03
13	-0.	1.00000E-03
14	-0.	4.00000E-04
15	-0.	1.00000E-04
16	-0.	2.50000E-05

MICROSCOPIC CROSS SECTIONS FOR NUCLIDE 1

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG
1	1,900	1,930	6,612	4,250	1,050	0,	0,	0,	0,	0,
2	1,950	2,000	6,026	4,500	1,420	0,200	0,	0,	0,	0,
3	1,830	1,900	5,472	4,800	2,090	0,180	0,270	0,	0,	0,
4	1,700	1,810	4,981	5,700	3,550	0,450	0,500	0,450	0,	0,
5	1,670	1,840	4,810	8,400	6,510	0,290	0,300	0,350	0,310	0,
6	1,920	2,300	5,491	13,200	10,850	0,050	0,050	0,060	0,050	0,040
7	2,500	3,500	7,150	13,500	9,960	0,050	0,	0,	0,	0,
8	4,200	6,700	12,012	16,700	9,960	0,040	0,	0,	0,	0,
9	14,000	21,500	40,000	31,500	9,960	0,040	0,	0,	0,	0,
10	6,000	10,000	17,200	20,000	9,940	0,040	0,	0,	0,	0,
11	17,000	25,000	48,600	35,000	9,930	0,060	0,	0,	0,	0,
12	4,000	7,000	11,400	17,000	9,940	0,070	0,	0,	0,	0,
13	20,000	23,000	57,200	33,000	9,930	0,060	0,	0,	0,	0,
14	80,000	123,000	229,000	133,000	9,920	0,070	0,	0,	0,	0,
15	760,000	1210,000	2174,000	1220,000	9,940	0,080	0,	0,	0,	0,
16	705,000	980,000	2016,300	990,000	10,000	0,060	0,	0,	0,	0,

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG
1	1.600	1.650	5.952	4.250	1.030	0.	0.	0.	0.	0.
2	1.540	1.620	5.267	4.500	1.620	0.240	0.	0.	0.	0.
3	1.390	1.500	4.587	4.650	2.240	0.220	0.330	0.	0.	0.
4	0.770	0.920	2.487	5.450	4.070	0.500	0.580	0.550	0.	0.
5	0.050	0.280	0.159	8.200	7.870	0.400	0.350	0.400	0.400	0.
6	0.	0.450	0.	11.450	10.950	0.050	0.060	0.060	0.060	0.050
7	0.	0.700	0.	11.700	10.950	0.050	0.	0.	0.	0.
8	0.	2.000	0.	13.000	10.950	0.050	0.	0.	0.	0.
9	0.	11.000	0.	22.000	10.950	0.050	0.	0.	0.	0.
10	0.	5.000	0.	16.000	10.920	0.050	0.	0.	0.	0.
11	0.	12.000	0.	24.000	11.910	0.080	0.	0.	0.	0.
12	0.	24.000	0.	39.000	14.900	0.090	0.	0.	0.	0.
13	0.	300.000	0.	808.000	504.140	0.100	0.	0.	0.	0.
14	0.	120.000	0.	158.000	37.650	3.860	0.	0.	0.	0.
15	0.	140.000	0.	151.000	10.930	0.350	0.	0.	0.	0.
16	0.030	252.030	0.090	263.030	11.000	0.070	0.	0.	0.	0.

MICROSCOPIC CROSS SECTIONS FOR NUCLIDE 3

-36-

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG
1	0.	0.040	0.	1.330	0.866	0.	0.	0.	0.	0.
2	0.	0.	0.	1.180	0.989	0.424	0.	0.	0.	0.
3	0.	0.	0.	3.230	2.328	0.191	0.	0.	0.	0.
4	0.	0.	0.	3.630	3.074	0.902	0.	0.	0.	0.
5	0.	0.	0.	3.710	3.373	0.556	0.	0.	0.	0.
6	0.	0.	0.	3.260	3.029	0.337	0.	0.	0.	0.
7	0.	0.	0.	3.550	3.295	0.231	0.	0.	0.	0.
8	0.	0.	0.	3.640	3.370	0.255	0.	0.	0.	0.
9	0.	0.	0.	3.640	3.370	0.270	0.	0.	0.	0.
10	0.	0.	0.	3.640	3.260	0.270	0.	0.	0.	0.
11	0.	0.	0.	3.640	3.226	0.380	0.	0.	0.	0.
12	0.	0.	0.	3.640	3.260	0.414	0.	0.	0.	0.
13	0.	0.	0.	3.640	3.226	0.380	0.	0.	0.	0.
14	0.	0.	0.	3.640	3.142	0.414	0.	0.	0.	0.
15	0.	0.	0.	3.640	3.309	0.498	0.	0.	0.	0.
16	0.	0.	0.	3.641	3.641	0.331	0.	0.	0.	0.

MICROSCOPIC CROSS SECTIONS FOR NUCLIDE 4

-37-

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG
1	0.	0.030	-0.	1.607	1.137	-0.	-0.	-0.	-0.	-0.
2	0.	-0.	-0.	2.260	1.704	0.440	-0.	-0.	-0.	-0.
3	0.	0.	-0.	2.990	1.799	0.556	-0.	-0.	-0.	-0.
4	0.	0.	-0.	3.119	2.471	1.191	-0.	-0.	-0.	-0.
5	0.	0.001	-0.	3.107	2.911	0.604	-0.	-0.	-0.	-0.
6	0.	0.001	-0.	3.884	3.695	0.195	0.036	-0.	-0.	-0.
7	0.	0.001	-0.	6.089	5.787	0.188	-0.	0.008	-0.	-0.
8	0.	0.001	-0.	4.545	4.310	0.301	-0.	-0.	-0.	-0.
9	0.	0.005	-0.	3.010	2.850	0.234	-0.	-0.	-0.	-0.
10	0.	0.011	-0.	3.010	2.783	0.155	-0.	-0.	-0.	-0.
11	0.	0.018	-0.	3.030	2.776	0.216	-0.	-0.	-0.	-0.
12	0.	0.032	-0.	3.069	2.818	0.236	-0.	-0.	-0.	-0.
13	0.	0.057	-0.	3.099	2.804	0.219	-0.	-0.	-0.	-0.
14	0.	0.100	-0.	3.210	2.816	0.238	-0.	-0.	-0.	-0.
15	0.	0.190	-0.	3.350	2.962	0.294	-0.	-0.	-0.	-0.
16	0.	0.447	-0.	3.747	3.300	0.198	-0.	-0.	-0.	-0.

MICROSCOPIC CROSS SECTIONS FOR NUCLIDE 5

-38-

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG
1	0.	0.003	0.	2.243	1.630	0.	0.	0.	0.	0.
2	0.	0.005	0.	2.200	1.575	0.610	0.	0.	0.	0.
3	0.	0.010	0.	1.960	1.470	0.400	0.	0.	0.	0.
4	0.	0.010	0.	2.290	2.157	0.230	0.200	0.	0.	0.
5	0.	0.010	0.	2.440	2.359	0.123	0.230	0.020	0.	0.
6	0.	0.010	0.	2.230	2.170	0.071	0.	0.020	0.	0.
7	0.	0.010	0.	5.650	5.520	0.050	0.	0.	0.	0.
8	0.	0.011	0.	7.114	6.952	0.120	0.	0.	0.	0.
9	0.	0.027	0.	10.968	10.708	0.151	0.	0.	0.	0.
10	0.	0.055	0.	11.364	10.977	0.233	0.	0.	0.	0.
11	0.	0.098	0.	11.364	10.901	0.332	0.	0.	0.	0.
12	0.	0.170	0.	11.370	10.871	0.365	0.	0.	0.	0.
13	0.	0.310	0.	11.370	10.702	0.329	0.	0.	0.	0.
14	0.	0.510	0.	11.370	10.431	0.358	0.	0.	0.	0.
15	0.	0.910	0.	11.570	10.390	0.429	0.	0.	0.	0.
16	0.	2.240	0.	13.040	10.800	0.270	0.	0.	0.	0.

MICROSCOPIC CROSS SECTIONS FOR NUCLIDE 6

-39-

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG
1	0.	0.010	-0.	2.290	1.370	-0.	-0.	-0.	-0.	-0.
2	0.	0.010	-0.	2.330	1.720	0.910	-0.	-0.	-0.	-0.
3	0.	0.010	-0.	2.530	2.320	0.300	-0.	-0.	-0.	-0.
4	0.	0.010	-0.	2.450	2.340	0.200	0.210	-0.	-0.	-0.
5	0.	0.010	-0.	3.380	3.290	0.100	-0.	0.090	-0.	-0.
6	0.	0.020	-0.	5.170	5.050	0.080	-0.	-0.	-0.	-0.
7	0.	0.040	-0.	14.540	14.210	0.100	-0.	-0.	-0.	-0.
8	0.	0.040	-0.	15.680	15.320	0.290	-0.	-0.	-0.	-0.
9	0.	0.050	-0.	16.680	16.300	0.320	-0.	-0.	-0.	-0.
10	0.	0.100	-0.	16.730	16.160	0.330	-0.	-0.	-0.	-0.
11	0.	0.180	-0.	17.300	16.590	0.470	-0.	-0.	-0.	-0.
12	0.	0.310	-0.	17.310	16.520	0.530	-0.	-0.	-0.	-0.
13	0.	0.560	-0.	17.510	16.420	0.480	-0.	-0.	-0.	-0.
14	0.	0.940	-0.	18.200	16.620	0.530	-0.	-0.	-0.	-0.
15	0.	1.600	-0.	19.000	16.970	0.640	-0.	-0.	-0.	-0.
16	0.	4.600	-0.	21.210	16.610	0.430	-0.	-0.	-0.	-0.

SAMPLE CASE FOR C R U N C H USING HANSEN-ROACH CROSS SECTIONS

.....THIS IS A BUCKLING CALCULATION

K EFFECTIVE IS EQUAL TO 1.000000

THE CONVERGENCE CRITERION IS 1.000000E-06

THE NUCLIDE DENSITIES ARE LISTED BELOW

ELEMENT ATOMS/CC

1	2.249280E+03
2	1.693000E+04
3	4.837150E+03
4	1.142250E+02
5	2.526300E+02
6	8.876190E+03

SAMPLE CASE FOR C R U N C H USING HANSEN-ROACH 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
1	4.54E-03	5.32E-03	1.59E-02	1.12E-01	7.31E-02	-0.	-0.	-0.	-0.	-0.
2	4.65E-03	4.99E-03	1.44E-02	1.19E-01	8.28E-02	3.11E-02	-0.	-0.	-0.	-0.
3	4.35E-03	4.87E-03	1.31E-02	1.33E-01	9.46E-02	2.05E-02	6.63E-04	-0.	-0.	-0.
4	3.95E-03	4.57E-03	1.16E-02	1.47E-01	1.27E-01	2.66E-02	8.14E-03	1.11E-03	-0.	-0.
5	3.76E-03	4.54E-03	1.08E-02	1.65E-01	1.54E-01	1.43E-02	6.54E-03	2.16E-03	7.65E-04	-0.
6	4.32E-03	5.69E-03	1.24E-02	1.94E-01	1.83E-01	6.48E-03	5.34E-04	6.50E-04	1.23E-04	9.84E-05
7	5.62E-03	8.61E-03	1.61E-02	3.91E-01	3.72E-01	5.54E-03	-0.	9.14E-05	-0.	-0.
8	9.45E-03	1.61E-02	2.70E-02	4.28E-01	4.01E-01	1.04E-02	-0.	-0.	-0.	-0.
9	3.15E-02	5.14E-02	9.00E-02	5.52E-01	4.88E-01	1.07E-02	-0.	-0.	-0.	-0.
10	1.35E-02	2.57E-02	3.87E-02	5.35E-01	4.93E-01	1.20E-02	-0.	-0.	-0.	-0.
11	3.82E-02	6.25E-02	1.09E-01	5.76E-01	4.94E-01	1.70E-02	-0.	-0.	-0.	-0.
12	9.00E-03	2.72E-02	2.56E-02	5.38E-01	4.94E-01	1.88E-02	-0.	-0.	-0.	-0.
13	4.50E-02	1.16E-01	1.29E-01	7.07E-01	5.71E-01	1.71E-02	-0.	-0.	-0.	-0.
14	1.80E-01	3.19E-01	5.15E-01	8.29E-01	4.87E-01	1.93E-02	-0.	-0.	-0.	-0.
15	1.71E 00	2.78E 00	4.89E 00	3.29E 00	4.87E-01	2.25E-02	-0.	-0.	-0.	-0.
16	1.59E 00	2.35E 00	4.54E 00	2.85E 00	5.00E-01	1.46E-02	-0.	-0.	-0.	-0.

SAMPLE CASE FOR C R U N C H USING HANSEN-ROACH CROSS SECTIONS

.....FINAL RESULTS BELOW

KI EFFECTIVE = 9.999999E-01
 BUCKLING(B2) = 3.457151E-03
 SOURCE RATE = 1.000006E-02
 ABSORPTION RATE = 4.100491E-01
 LEAKAGE RATE = 5.899567E-01

GROUP	FLUX	NU*SIGF*FLUX	SIGF*FLUX	INTG FLUX
1	4.13863E-02	6.57210E-00	1.88080E-00	5.28606E-02
2	1.03613E-03	1.49677E-01	4.81469E-00	1.85200E-01
3	8.08747E-02	1.05822E-01	3.51927E-00	2.88497E-01
4	1.77050E-03	2.05816E-01	7.00081E-00	5.14634E-01
5	2.34415E-03	2.54246E-01	8.82518E-00	8.14041E-01
6	1.06210E-03	1.31178E-01	4.58682E-00	9.49698E-01
7	2.75468E-02	4.43017E-00	1.54901E-00	9.84882E-01
8	9.69637E-01	2.61980E-00	9.16014E-01	9.97267E-01
9	1.58918E-01	1.42980E-00	5.00430E-01	9.99297E-01
10	4.24309E-00	1.64155E-01	5.72633E-02	9.99839E-01
11	8.66173E-01	9.46857E-02	3.31205E-02	9.99949E-01
12	3.50700E-01	8.99257E-03	3.15529E-03	9.99994E-01
13	4.37161E-02	5.62446E-03	1.96660E-03	1.00000E-00
14	2.45548E-03	1.28478E-03	4.41845E-04	1.00000E-00
15	1.97555E-05	9.66033E-05	3.37712E-05	1.00000E-00
16	1.23140E-07	5.58470E-07	1.95269E-07	1.00000E-00

THE TOTAL FLUX IS 7.8293E-03

THE FISSION RATE IS 3.3689E-01

SAMPLE CASE FOR C R U N C H USING HANSEN-ROACH CROSS SECTIONS

DATA FOR CROSS SECTION AVERAGING LISTED BELOW

THE NUMBER OF ENLARGED ENERGY GROUPS IS 4

SIGMA TRANSPORT IS CALCULATED USING RECIPROCAL WEIGHTING

NEW GROUP	OLD GROUPS PER NEW GROUP	FISSION SOURCE FOR NEW GROUP	FLUX
1	4	0.89600E+00	0.40292E+04
2	3	0.10400E+00	0.36817E+04
3	5	0.	0.11832E+03
4	4	0.	0.46191E+01

SAMPLE CASE FOR C R U N C H USING HANSEN-ROACH CROSS SECTIONS

THE COLLAPSED GROUP MICROSCOPIC CROSS SECTIONS ARE LISTED BELOW FOR NUCLIDE NUMBER 1

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG
1	1,811	1,889	5,516	4,995	2,745	0.	0.	0.		
2	1,804	2,097	5,182	9,690	7,591	0,360	0.	0.		
3	5,674	8,941	16,223	18,013	9,072	0,003	0.	0.		
4	23,508	28,826	67,243	34,389	5,563	0,000	0.	0.		

SAMPLE CASE FOR C R U N C H USING HANSEN-ROACH CROSS SECTIONS

THE COLLAPSED GROUP MICROSCOPIC CROSS SECTIONS ARE LISTED BELOW FOR NUCLIDE NUMBER 2

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG
1	1.178	1.291	3.979	4.876	3.135	0.	0.	0.		
2	0.032	0.360	0.101	9.154	8.790	0.449	0.	0.		
3	0.	3.455	0.	13.934	10.478	0.004	0.	0.		
4	0.000	290.363	0.000	661.993	371.630	0.000	0.	0.		

SAMPLE CASE FOR C R U N C H USING HANSEN-ROACH CROSS SECTIONS

THE COLLAPSED GROUP MICROSCOPIC CROSS SECTIONS ARE LISTED BELOW FOR NUCLIDE NUMBER 3

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG
1	0.	0.004	0.	2.091	1.842	0.	0.	0.		
2	0.	0.	0.	3.556	3.537	0.244	0.	0.		
3	0.	0.	0.	3.640	3.639	0.019	0.	0.		
4	0.	0.	0.	3.640	3.640	0.001	0.	0.		

SAMPLE CASE FOR C R U N C H USING HANSEN-ROACH CROSS SECTIONS

THE COLLAPSED GROUP MICROSCOPIC CROSS SECTIONS ARE LISTED BELOW FOR NUCLIDE NUMBER 4

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G-1XG	G-2XG	G-3XG	G-4XG	G-5XG
1	0.	0.003	-0.	2.593	2.305	0.	0.	0.		
2	0.	0.001	-0.	3.431	3.407	0.285	0.	0.		
3	0.	0.002	-0.	4.163	4.160	0.023	-0.	0.		
4	0.	0.059	-0.	3.105	3.045	0.001	0.	0.		

SAMPLE CASE FOR C R U N O H USING HANSEN-ROACH CROSS SECTIONS

THE COLLAPSED GROUP MICROSCOPIC CROSS SECTIONS ARE LISTED BELOW FOR NUCLIDE NUMBER 5

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G=1XG	G=2XG	G=3XG	G=4XG	G=5XG
1	0.	0.008	0.	2.188	2.071	0.	0.	0.		
2	0.	0.010	0.	2.478	2.459	0.109	0.	0.		
3	0.	0.016	0.	7.604	7.587	0.009	0.	0.		
4	0.	0.321	0.	11.370	11.049	0.001	0.	0.		

SAMPLE CASE FOR C R U N G H USING HANSEN-ROACH CROSS SECTIONS

THE COLLAPSED GROUP MICROSCOPIC CROSS SECTIONS ARE LISTED BELOW FOR NUCLIDE NUMBER 6

GROUP	SIGF	SIGA	NUSIGF	SIGTR	SIGXG	G=1XG	G=2XG	G=3XG	G=4XG	G=5XG
1	0.	0.010	0.	2.416	2.339	0.	0.	0.		
2	0.	0.015	0.	4.011	3.974	0.067	0.	0.		
3	0.	0.045	0.	15.859	15.812	0.022	0.	0.		
4	0.	0.581	0.	17.546	16.965	0.001	0.	0.		

SAMPLE CASE FOR C R U N C H USING HANSEN-ROACH CROSS SECTIONS

.....FINAL RESULTS BELOW

K EFFECTIVE = 1.000000E 00
 BUCKLING(B2) = 3.383756E-03
 SOURCE RATE = 1.000006E 02
 ABSORPTION RATE = 4.101237E 01
 LEAKAGE RATE = 5.898821E 01

GROUP	FLUX	NU*SIGF*FLUX	SIGF*FLUX	INTG FLUX
1	4.02336E 03	5.26266E 01	1.71904E 01	5.13879E-01
2	3.68735E 03	4.30382E 01	1.49839E 01	9.84843E-01
3	1.18625E 02	4.32872E 00	1.51393E 00	9.99994E-01
4	4.63021E-02	7.00314E-03	2.44826E-03	1.00000E-00

THE TOTAL FLUX IS 7.8294E 03

THE FISSION RATE IS 3.3691E 01

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