664127

DP-1064

AEC RESEARCH AND DEVELOPMENT REPORT

The HAMMER System

HETEROGENEOUS ANALYSIS BY MULTIGROUP METHODS OF EXPONENTIALS AND REACTORS

J. E. SUICH & H. C. HONECK

DUPONT
REG. U. S. PAT. OFF.

Savannah River Laboratory

Aiken, South Carolina

LEGAL NOTICE

Printed in the United States of America
Available from
Clearinghouse for Federal Scientific and Technical Information
National Bureau of Standards, U. S. Department of Commerce
Springfield, Virginia 22151
Price: Printed Copy $3.00; Microfiche $0.65

DP-1064

Physics
(TID-4500)

The HAMMER System

HETEROGENEOUS ANALYSIS BY MULTIGROUP METHODS OF EXPONENTIALS AND REACTORS

by

John E. Suich
Savannah River Laboratory

and

Henry C. Honeck
United States Atomic Energy Commission

Approved by

G. Dessauer, Director
Physics Section

January 1967

E. I. DU PONT DE NEMOURS & COMPANY
SAVANNAH RIVER LABORATORY
AIKEN, S. C. 29801

CONTRACT AT(07-2)-1 WITH THE
UNITED STATES ATOMIC ENERGY COMMISSION

ABSTRACT

The HAMMER system comprises programs for calculations of infinite lattice parameters by multigroup transport theory, and of composite reactor parameters by few-group diffusion theory. Operating procedures for this system of linked reactor physics codes are presented for the HAMMER system and for the auxiliary cross section library codes.

PREFACE

The HAMMER system of linked reactor physics programs is subject to revision as theoretical knowledge and computing power increase. This initial document is intended to serve as an operating manual for the HAMMER system. Addenda will be issued to describe the physical equations and their solutions, modifications to the present programs, and additional programs. Therefore, the staples may be removed from this document and the pre-punched pages inserted in a looseleaf binder. Revisions to the code may then be easily inserted at the proper place.

CONTENTS

The HAMMER System

HETEROGENEOUS ANALYSIS BY MULTIGROUP METHODS OF EXPONENTIALS AND REACTORS

INTRODUCTION

Since the appearance of large-scale computing machinery in the 1950's there has evolved a considerable body of computer programs for, and experience in, nuclear reactor calculations. Moreover, a "standard" group of reactor codes has come into widespread use at U.S. laboratories for both reactor design problems and the analysis of lattice experiments. Recently, Honeck and Crandall[1] applied a selected set of these codes, namely THERMOS,[2] MUFT,[3] ZUT and TUZ,[4] and FOG[5] to the analysis of some one hundred critical experiments on clean, D_2O-uranium lattices. Good agreement between theory and experiment was obtained, the average calculated eigenvalue being 0.996. The same battery of codes has also been applied to a series of light water lattices with equally good results.[6] Furthermore, each of these codes has been proven individually through extensive use.

The experience gained in the above-mentioned series of calculations showed a definite need for the automation of these standard codes to eliminate the burdensome tasks of preparing redundant sets of input for the various lattice codes. At the same time, it was thought desirable to base all of the lattice codes on a consistent mathematical model, namely multigroup, multiregion integral transport theory for infinite lattice calculations, followed by homogenized-cell, Fourier transform leakage calculations. Finally, it was desired to produce a sufficiently general system to facilitate inter-laboratory comparisons of reactor calculations. The result of this project is the HAMMER system (an acronym for Heterogeneous Analysis by Multigroup Methods of Exponentials and Reactors), an IBM 704/7090 FORTRAN II code developed jointly at Brookhaven National Laboratory and at the Savannah River Laboratory. FORTRAN IV versions are being prepared for the IBM System 360 and CDC 6600.

Description of the HAMMER System

From a single set of data cards, the HAMMER system will sequentially perform any or all of the following operations:

(1) Program CAPN - interpret infinite lattice cell input data and perform extensive error checking; establish path of control among the following programs.

(2) Program THERMOS - perform a multigroup calculation of thermal flux distribution ($E < 0.625$ ev) from integral transport theory, followed by a multigroup Fourier transform leakage spectrum correction based on homogenized-cell parameters (multigroup flux-volume weighting). Cross section averages and diffusion parameters are provided in the output, as well as fluxes and reaction rates.

(3) Program HAMLET - same as (2) except the energy range is 0.625 ev $\leqslant E \leqslant 10$ Mev. This is a major block of new programming, since no suitable codes were available for epithermal integral transport. Collision probabilities are computed under the assumption of cosine currents crossing region boundaries, and the energy spectrum during moderation is computed as in MUFT. This approach automatically produces a heterogeneous calculation of ϵ and p; for the latter, ZUT/TUZ is employed as a subroutine. In addition to the output described in (2), a set of four-group cross sections (10 Mev - 1.05 Mev - 9.12 kev - 0.625 ev) is produced, from which the asymptotic spectrum and the buckling of the infinite lattice are determined.

(4) Program FLOG - utilizes the four-group constants produced in (2) and (3), to perform a one-dimensional, many-region reactor calculation. This program is a modification of FOG. In addition to the standard criticality searches, FLOG will perform least-squares fitting of reactor fluxes to obtain experimental values of the buckling, and will automatically calculate a series of core size loadings for exponential assemblies.

(5) Program DIED - The fluxes and cross sections obtained in (2) through (4) are combined to produce neutron balance sheets in this final edit program. For an infinite lattice cell edit (results of (2) and (3) only), the output is either (a) absorptions versus fissions by isotope and group, split into "smooth" and "resonance" event contributions, or (b) total absorptions versus total fissions by isotope and group. For a composite reactor edit (results from (4)), neutron balance sheets are produced for total absorptions vs. total fission neutron production by isotope and region. These latter tables can be produced for all regions, for all fissioning regions, or for selected regions. The flux weighting employed in the tables can be that appropriate to the region average, the asymptotic core, the interfaces, or all of these.

Work is processed in stacked batches; each batch of jobs may use a different set of the five programs. A batch contains any number of cases to be processed by the selected programs; each case describes either an infinite lattice type to be processed by THERMOS and/or HAMLET or a set of previously run lattice types to be assembled into a one-dimensional, multiregion reactor to be processed by FLOG. The results of THERMOS-HAMLET calculations are saved on a lattice library tape (LLT) charged to the individual user; each lattice type is identified, and retrieved from the user's tape by a permanent identification number assigned at the time of initial processing, by the user. Thus, on any given LLT volume, all lattices should have unique identification numbers. The individual programs process all cases within a batch before control is passed to the logically next program to be run.

Information is passed from program to program on scratch tapes in the LLT format, or on the LLT itself. Thus, in the event of a machine failure occurring in the middle of a batch, the already completed work can be bypassed (if the scratch tapes are saved) by exercising a restart option.

INPUT PREPARATION

General

All input cards have the same field definition (I1, 1X, I2, 1X, 2I2, I5, 2I2, I1, 1X, 5E 10) except for the batch control card. The data required are given on the input sheets.

Only the first case in a batch need be complete; subsequent cases must include the case lead card (type 2 or 6) and at least one additional card (types 3-5 and 7-11). Card images are retained in core; on subsequent cases, the card images with corresponding sequence numbers are overwritten. Hence only data to be changed need be input after the first case of a batch.

Sample Input Decks

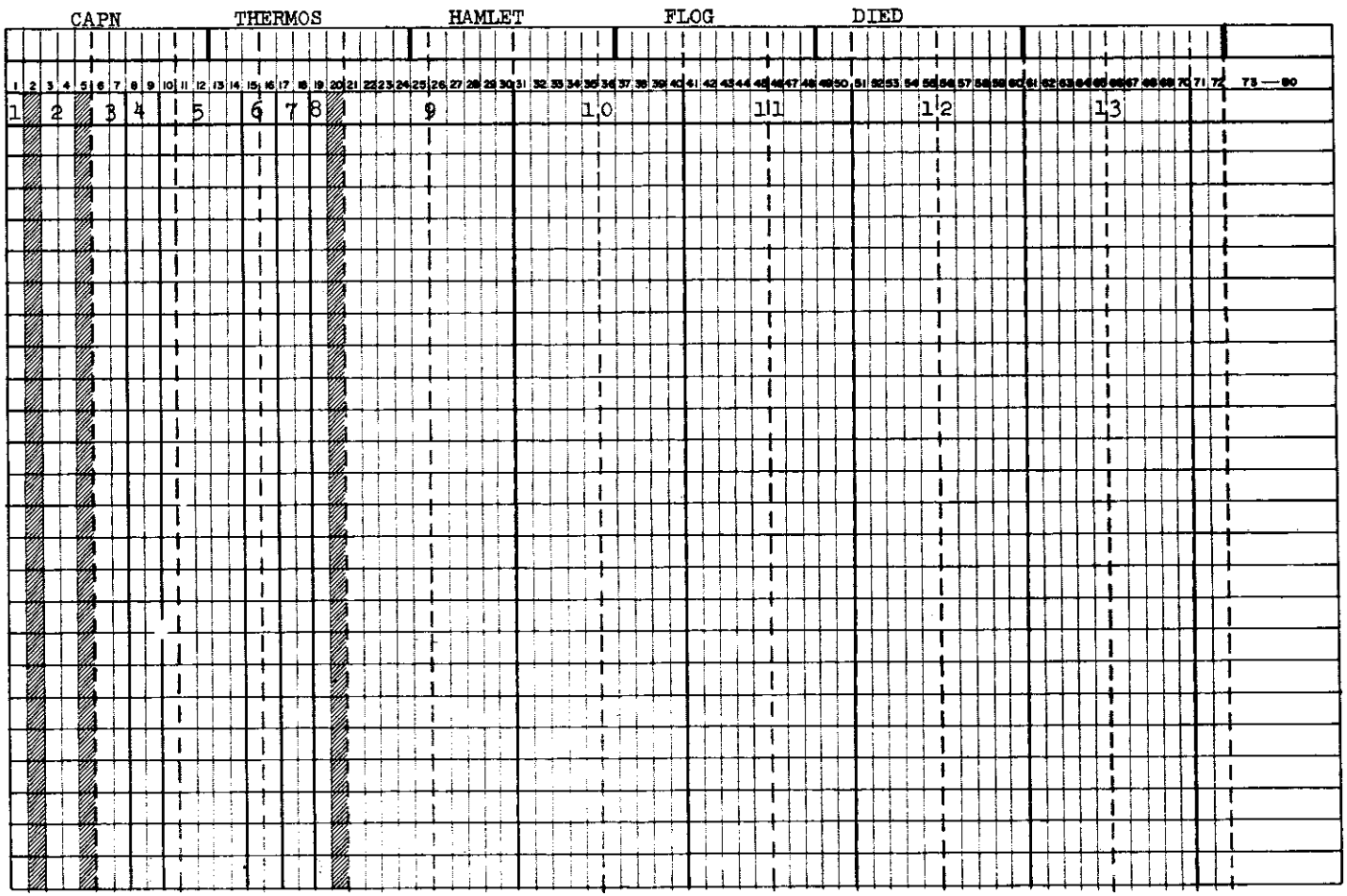
```
*I.D. (8277-15,,P,773A),COMPTON
*     XEQ
*     MOUNT EM0008 ON A4   EM0009 ON B5   EM0010 ON A5
*     PAUSE
*     CALL CHAIN 1,4   BINARY DECK
*     DATA
11          21          11 111111   11 11       1111311
   0     13066 1 11 TEST OF SINGLE ROD H2O CASES
   1     11006      1/16 METAL 1/1 M/F H2O
   2  3           2 10.       192235.   48000.
   3  1 1  100      .0625      18.9      3.        20.
1  4  2 2  300    1 .0883884  1.0                 20.
   1        1      TEST FLOG
   2  2 1   1      1.0
1  3  1 011006     18.937     29.32
1                                                 26111111111111
              1     LIST LLT      OPTION 6
```

OSR 24-F26

IBM 704 GENERAL PURPOSE CODING FORM

Programmer: _____ Page ____ of _____

Program: HAMMER SYSTEM



CAPN-THERMOS-HAMLET (Card Types 0-5)

CARD TYPE 0

Batch Control Card (One per batch)

Program	Column	Symbol	Description
CAPN	1	NCPRN	End job = 0; normal run = 1; process lattice input data only = 2
	2	NCPPT	Short print = 0; long print = 1
	3	NCPPN	Punch option (ignored)
THERMOS	13	NTHRN	No THERMOS run = 0; standard THERMOS run = 1; cosine current THERMOS run = 2
	14	NTHPT	Short print = 0; long print = 1
	15	NTHPN	Punch option (ignored)
HAMLET	25	NEPRN	No HAMLET run = 0; standard HAMLET run = 1
	26	NEPPT	Short print = 0; long print = 1
	27	NEPPN	Punch effective RI's = 1, no punch = 0
	28	NEPSG	Selengut-Goertzel approx = 0; no = 1 (recommended)
	29	NEPAG	Standard age approx = 0; self-consistent = 1 (recommended)
	30		
FLOG	37	NFGRN	No FLOG run = 0; standard FLOG run = 1
	38	NFGPT	Short print = 0; long print = 1
	39	NFGPN	Punch option (ignored)
	40	NFGPB	Point buckling calculation = 1; no = 0
	41	NFGJØ	J_o fit to thermal flux = 1; no = 0
	42	NFGB1	Boundary condition at origin

0 $d\phi^i/dr = 0$

1 $\phi^i = 0$

2 $\phi^i = w^i$

4 $\phi^i + w^i d\phi^i/dr = 0$

5 $\phi^i - 2.13\, D^i d\phi^i/dr = 0$

	43	NFGB2	Boundary condition at outer boundary 0 $\phi^i = 0$ 1 $d\phi^i/dr = 0$ 2 $\phi^i = w^i$ 4 $\phi^i + w^i_1 d\phi^i/dr = 0$ 5 $\phi^i - 2.13\ D^i d\phi^i/dr = 0$
DIED	49	NEDRN	No edit = 0 Neutron balance edit = 1 (this section) Lattice Library Tape edit = 2 (next section)
	50	NEDPT	Short print = 0; long print = 1
	51	NEDPN	Punch option (ignored)
	52	NEDNB	Detailed Neutron Balance sheet options 0 Ignore 1 Smooth vs. resonance reaction rates 2 Total reaction rates
	53	NEDFW	Flux weighting options for DNB 0 Reactor region average 1 Plus asymptotic for material of region 2 Plus interfaces of region 3 Plus midpoint of region
	54	NEDRB	Reactor regionwise balance sheet options 0 Ignore 1 All regions 2 All fissioning regions 3 Regions tagged on a data card following all FLOG input (40 I1; 1 = yes; 0 = no)
	55	NEDAX	Reactor regionwise average cross sections (as above for NEDRB)
	56	NEDNU	0 Balance absorptions vs. fissions 1 Balance absorptions vs. production
LLT EDIT	49	NEDRN	2 = Lattice Library Tape edit
	50	NEDPT	0 Print case ID, titles only 1 Plus integral reactivity parameters 2 Plus few-group data 3 Plus overlapping thermal group data 4 Case ID, title, region input data 5 Plus mesh point data 6 Complete listing of LLT data

CARD TYPE 1

Batch Lead Card

Field	Column	Format	Symbol	Description
1	1	I1	LCT	= 0 except last card of case = 1
2	3-4	I2	NC	Card sequence number (within a case) = 0
3	6-7	I2	NSTH	= 0 unless a "restart" job; then = number of THERMOS cases yet to be done
4	8-9	I2	NSEP	= 0 unless a "restart" job; then = number of HAMLET cases yet to be done
5	10-14	I5	IDBCH	Batch identification number for output page heading
6	15-16	I2	NCASE	Number of THERMOS/HAMLET cases in this batch (number of lattices)
7	17-18	I2	NFLØG	Number of FLOG cases in this batch (number of reactors)
8	19	I1	NEWLIB	= 1 if starting a new lattice library, = 0 if added to an old one
9	21-30		BTTL	Hollerith (8A6,A2) batch title for output page heading
10	31-40			
11	41-50			
12	51-60			
13	61-70			

CARD TYPE 2

Case Lead Card (One Per Lattice)

Field	Column	Format	Symbol	Description
1	1	I1	LCT	
2	3-4	I2	NC	Card sequence number (within a case) = 1
3	6-7	I2	-	
4	8-9	I2	-	
5	10-14	I5	IDENT	Lattice identification number; permanent ID; therefore must be unique (and non-zero) for a given lattice library file
6	15-16	I2	-	
7	17-18	I2	-	
8	19	I1	-	
9	21-30			
10	31-40			
11	41-50		HØL	Hollerith (8A6,A2) lattice title
12	51-60			
13	61-70			

CARD TYPE 3

Lattice Description Card (One Per Lattice)

Field	Column	Format	Symbol	Description
1	1	I1	LCT	= 0 except last card of case = 1
2	3-4	I2	NC	Card sequence number (within a case) = 2
3	6-7	I2	NGEØM	Slab = 1; Hex = 2; Square = 3
4	8-9	I2	NRX	Number of regions (ignored at present) $\leqslant$20
5	10-14	I5	MX	Number of mixtures (ignored at present) $\leqslant$10
6	15-16	I2	NPIBI	P1 = 0; B1 = 1 option for Fourier transform leakage calculations
7	17-18	I2	-	
8	19	I1	ISØXE	Number of foil materials to be edited (must not duplicate cell materials) $\leqslant$4
9	21-30	E10.0	BSQD	Material buckling (m^{-2}) for Fourier transform leakage calculations
10	31-40	E10.0	WSTBA(I), I=1, ISØXE	Foil ID numbers
11	41-50	E10.0		
12	51-60	E10.0		
13	61-70	E10.0		

CARD TYPE 4

Lattice Region Description Card (One Per Region)

Field	Column	Format	Symbol	Description
1	1	I1	LCT	= 0 except last card of case = 1
2	3-4	I2	NC	Card sequence number (within a case)
3	6-7	I2	NR	Region sequence number counting from center of cell = 1
4	8-9	I2	MXAS	Mixture number associated with this region; fields indicated by asterisk must be filled in on first appearance of new MXAS
5	10-14	I5	MR*	Predefined material in this region; table of possible values are given in Note a, page 26
6	15-16	I2	ISØXA*	Number of isotopes to be added to the predefined material; follow this card with ISOXA cards of type 5*
7	17-18	I2	NPT	Number of spatial mesh points assigned to this region; if left blank code will compute a value (only if THERMOS is to be run)
8	19	I1	ILHI*	= 1 if thermal neutron upscattering is important (moderator region); = 0 otherwise (fuel, clad, etc.)
9	21-30	E10.0	THT	Region OD or slab thickness (in.); for outermost region, a negative value denotes lattice pitch (in.); for slab cases, innermost and outermost region thicknesses are distances to reflecting boundaries
10	31-40	E10.0	D*	Mixture density (g/cc)
11	41-50	E10.0	P*	Weight percent of secondary component in the material, exclusive of added isotopes
12	51-60	E10.0	DEGC*	Region temperature (°C)
13	61-70	E10.0	FISD	Fission neutrons born/cc in this region; required only for HAMLET only runs

CARD TYPE 5

Added Isotope Card (As Required)

Field	Column	Format	Symbol	Description
1	1	I1	LCT	= 0 except last card of case = 1
2	3-4	I2	NC	Card sequence number (within a case)
3	6-7	I2	-	
4	8-9	I2	-	
5	10-14	I5	-	
6	15-16	I2	-	
7	17-18	I2	-	
8	19	I1	-	
9	21-30	E10.0	WSTBA	Isotope ID (see page 24 for possible values)
10	31-40	E10.0	WSTBB	Temperature/Model index; as above
11	41-50	E10.0	F	Weight fraction of this isotope in total mixture; use either fields 11 and 12, or 13 (see Note b, page 27)
12	51-60	E10.0	A	Atomic weight this isotope
13	61-70	E10.0	CØNCTA	Concentration (atoms/cc*10^{24}); if this specification is given, D on card 4 is exclusive of this isotope

CARD TYPE 6

Reactor Lead Card (One Per Reactor)

Field	Column	Format	Symbol	Description
1	1	I1	LCT	= 0 except last card of case = 1
2	3-4	I2	NC	Card sequence number (within a case) = 1
3	6-7	I2	-	
4	8-9	I2	-	
5	10-14	I5	IDENT	Reactor case ID for output page heading (non-zero, and unique within a batch)
6	15-16	I2	-	
7	17-18	I2	-	
8	19	I1	-	
9	21-30			
10	31-40			
11	41-50		HØL	Hollerith (8A6,A2) reactor title
12	51-60			
13	61-70			

CARD TYPE 7

Reactor Description Card (One Per Reactor)

Field	Column	Format	Symbol	Description
1	1	I1	LCT	= 0 except last card of case = 1
2	3-4	I2	NC	Card sequence number (within a case) = 2
3	6-7	I2	NGEØM (= N1)	Slab = 1; cylinder = 2; sphere = 3
4	8-9	I2	NRX	Number of regions, $\leqslant 40$; regions are numbered sequentially from the origin
				Search Options
5	10-14	I5	N20	Variable loading = 0; buckling search = 1; poison search = 2; dimension search = 3; poison boundary search = 4; fuel loading search = 5
6	15-16	I2	NVLR	Number of core radii to be considered in variable loading sequence (use if N20 = 0)
7	17-18	I2	NVLT	Reflector thickness constant = 0, reactor thickness constant = 1 (use if N20 = 0)
8	19	I1	-	
9	21-30	E10.0	EIGEN2	Value of the eigenvalue (k_{eff}) to be searched for
10	31-40	E10.0	RLN1	(N20 = 0) minimum value of $1/\kappa_z$ (cm); if fields 10, 11, and 12 are blank, use NVLR type 8 CDS;
			PG11	(N20 = 4, 5) initial guess at boundary, mesh point to left of body
11	41-50	E10.0	RLX	(N20 = 0) maximum value of $1/\kappa_z$ (cm);
			PG2	(N20 = 4, 5) lower bound for boundary, distance from origin
12	51-60	E10.0	RSV	(N20 = 0) reflector savings (cm);
			PG3	(N20 = 4, 5) second guess for boundary, distance from origin
13	61-70	E10.0	PG4	(N20 = 4, 5) upper bound for boundary, distance from origin

CARD TYPE 8

Core Size Card (As Required When N20 = 0)

Field	Column	Format	Symbol	Description
1	1	I1	LCT	= 0 except last card of case = 1
2	3-4	I2	NC	Card sequence number (within a case)
3	6-7	I2	-	
4	8-9	I2	-	
5	10-14	I5	-	
6	15-16	I2	-	
7	17-18	I2	-	
8	19	I1	-	
9	21-30	E10.0	FGCS(I), I=1, NVLR	Core radii (in.)
10	31-40	E10.0		
11	41-50	E10.0		
12	51-60	E10.0		
13	61-70	E10.0		

CARD TYPE 9

Control Poison Card (N20=2,4,5)

Field	Column	Format	Symbol	Description
1	1	I1	LCT	= 0 except last card of case = 1
2	3-4	I2	NC	Card sequence number (within a case)
3	6-7	I2	-	
4	8-9	I2	-	
5	10-14	I5	-	
6	15-16	I2	-	
7	17-18	I2	-	
8	19	I1	-	
9	21-30	E10.0	SGPT	Σ_p(thermal); this value is a scale factor on SIGPT(NR), card type 10
10	31-40	E10.0	T(1)	Σ_p (Group 1)/Σ_p (thermal)
11	41-50	E10.0	T(2)	Σ_p (Group 2)/Σ_p (thermal)
12	51-60	E10.0	T(3)	Σ_p (Group 3)/Σ_p (thermal)
13	61-70	E10.0	T(4)	Σ_p (Group 4)/Σ_p (thermal)

CARD TYPE 10

Reactor Region Description (One Per Region)

Field	Column	Format	Symbol	Description
1	1	I1	LCT	= 0 except last card of case = 1
2	3-4	I2	NC	Card sequence number (within a case)
3	6-7	I2	NR	Region number
4	8-9	I2	ITYPE	Few Group Data { from lattice library = 0; from type 11 cards = 1; "Built-in" reflector = 2 (see page 28)
5	10-14	I5	MAT	Material used (lattice ID) in this region
6	15-16	I2	-	
7	17-18	I2	NPT	Number of mesh points assigned to this region (optional)
8	19	I1	-	
9	21-30	E10.0	THT	Region thickness (in.)
10	31-40	E10.0	BSQD	Region transverse buckling (m^{-2}); if blank for region 2, the value for region 1 is used for all regions
11	41-50	E10.0	SIGAD	Added thermal absorption, Σ_a
12	51-60	E10.0	SIGPT	Control region tag; thermal poison multiplier of SGPT, card type 9
13	61-70	E10.0	-	

CARD TYPE 11

Few-Group Data Card (Sets of 4 as Required)

Field	Column	Format	Symbol	Description
1	1	I1	LCT	= 0 except last card of case = 1
2	3-4	I2	NC	Card sequence number (within a case)
3	6-7	I2	-	
4	8-9	I2	-	
5	10-14	I5	MAT	Lattice ID
6	15-16	I2	I	Group number
7	17-18	I2	-	
8	19	I1	-	
9	21-30	E10.0	FØGX(I,1)	D
10	31-40	E10.0	FØGX(I,2)	Σ_a
11	41-50	E10.0	FØGX(I,3)	$\nu\Sigma_f$
12	51-60	E10.0	FØGX(I,4)	Σ_{rem}
13	61-70	E10.0	FØGX(I,5)	Σ_f

THERMOS LIBRARY TAPE, *HAMMER* CODE

TAPE LABELED E-V-9 6/21/66

TAPE TO BE UPDATED LABELED E-V-9 6/21/66

ISOTOPES PREVIOUSLY ON TAPE

IDENT	T/M	KERNELS	DESCRIPTION
92235.	0.	0	URANIUM 235
92236.	-0.	0	U236 1/V(KT)= 6 B
92238.	0.	0	URANIUM 238
5010.	-0.	0	B10 1/V(KT)=3837 B
1.	0.	0	1/V UNIT ABSORBER
94239.	0.	0	PLUTONIUM 239
5000.	0.	0	BORON
48000.	0.	0	CADMIUM
192235.	0.	0	URANIUM 235 FISS
194239.	0.	0	PLUTONIUM 239 FISS
66164.	0.	0	DYSPROSIUM 164
71176.	0.	0	LUTETIUM 176
13000.	0.	1	ALUMINUM
2.	0.	0	UNIT SCATTERER
1001.	1029.	2	H NEL T=293K
1002.	1029.	2	D NEL T=293K
8000.	29.	1	OXYGEN GAS T=293K
8000.	-0.	1	OXYGEN GAS T=293K
1001.	1033.	2	H NEL T=333K
1002.	1033.	2	D NEL T=333K
8000.	33.	1	OXYGEN GAS T=333K
1001.	1036.	2	H NEL T=363K
1002.	1036.	2	D NEL T=363K
8000.	36.	1	OXYGEN GAS T=363K
40000.	0.	1	ZIRCONIUM
304.	0.	1	304 STAINLESS
26000.	0.	1	IRON
92233.	0.	0	URANIUM 233
94240.	0.	0	PLUTONIUM 240
94241.	0.	0	PLUTONIUM 241
90232.	0.	0	THORIUM 232
192233.	0.	0	URANIUM 233 FISS
79197.	0.	0	GOLD 197
63151.	0.	0	EUROPIUM 151
49115.	0.	0	INDIUM 115
3006.	-0.	1	LI6 1/V KT=945
3007.	-0.	1	LI7 ZERO ABS.
62149.	-0.	1	SM149 BNL325/0.12
54135.	-0.	1	XE135 BNL325-3+06
3.	-0.	0	LONG LIVED FP PRS

1001.	2029.	1	H BSJ T=293K
1002.	2029.	1	D BSJ T=293K
1001.	29.	1	H GAS T=293K
1002.	29.	1	D GAS T=293K
1001.	2033.	1	H BSJ T=333K
1002.	2033.	1	D BSJ T=333K
1001.	33.	1	H GAS T=333K
1002.	33.	1	D GAS T=333K
1001.	2036.	1	H BSJ T=363K
1002.	2036.	1	D BSJ T=363K
1001.	36.	1	H GAS T=363K
1002.	36.	1	D GAS T=363K
6012.	29.	1	C 12 GAS T=293K
6012.	47.	1	C 12 GAS T=473K
1001.	3062.	2	H SANTOWAX BSJ
1001.	3055.	2	H SANTOWAX BSJ
1001.	3045.	2	H SANTOWAX BSJ
1001.	3029.	2	H SANTOWAX BSJ
6012.	45.	1	C 12 GAS
6012.	55.	1	C 12 GAS
6012.	62.	1	C 12 GAS
101001.	1029.	2	H NEL T=293K
101002.	1029.	2	D NEL T=293K
1001.	1040.	2	H NEL T=400K
1002.	1040.	2	D NEL T=400K
8000.	40.	1	OXYGEN GAS T=400K
1001.	1050.	2	H NEL T=500K
1002.	1050.	2	D NEL T=500K
8000.	50.	1	OXYGEN GAS T=500K
1001.	1060.	2	H NEL T=600K
1002.	1060.	2	D NEL T=600K
8000.	60.	1	OXYGEN GAS T=600K
6012.	1029.	1	GRAPHITE PARKS 293
4009.	29.	1	BERYLIUM T=293 M=9
93237.	0.	0	NP237 RAZ
1090232.	0.	0	THORIUM 232
1092238.	0.	0	URANIUM 238
92234.	-0.	0	U 234 1/V SIG=95.
91231.	-0.	0	PA231 BNL325
91233.	-0.	0	PA233 BNL325
94238.	-0.	0	PU238 AI-65-190V10
94242.	-0.	0	PU242 JAS
95243.	-0.	0	AM243 JAS
96244.	-0.	0	CM244 AI-65-190
1001.	5029.	2	H NEL T=293K C6H6
6012.	47.	1	C 12 GAS T=473K
1001.	5047.	2	H NEL T=473K C6H6

Note a

Materials Available 9/1/66	MR	First Component	Second Component
No material; all added isotopes	0		
Fuel	100	^{238}U metal	^{235}U metal
	101	$^{238}UO_2$	$^{235}UO_2$
	102	^{232}Th metal	^{233}U metal
	103	$^{232}ThO_2$	$^{233}UO_2$
	104	Al	92% ^{239}Pu + 8% ^{240}Pu
	105	Al	84% ^{239}Pu + 16% ^{240}Pu
	106		
	107		
	108		
	109		
Cladding	200	Al	None
	201	304 stainless	None
	202	Zircaloy-2	None
	203	Al	^{6}Li
	204		
	205		
Coolant-Moderator	300	H_2O	B_2O_3
	301	D_2O	H_2O
	302	Graphite	None
	303	Organic ($C_{18}H_{14}$)	None
	304		
	305		

Note b

Breakdown of Mixtures by Isotope

Let F = wt fraction of second component in material

F_i = wt fraction of added isotope i

D = density (g/cc^3) of mixture

A_j = atomic weight of isotope j

N_j = number density (atoms/barn-cm) of isotope j

Then for the primary isotope (subscript p)

$$N_p = D * (1 - \sum_i F_i) * 0.6025/A_p * (1 - F)$$

for the secondary isotope (subscript s)

$$N_s = D * (1 - \sum_i F_i) * 0.6025/A_s * F$$

for the added isotopes (subscript i)

$$N_i = D * F_i * 0.6025/A_i$$

i.e., the mixture density is reduced by $(1 - \sum_i F_i)$ to obtain the material density.

If F_i is set to zero, and N_i is input instead, then the material density should be given in place of the mixture density.

Built-In Reflector Constants (FLØG)

Material		Description				
		Group 1	Group 2	Group 3	Group 4	
1	D	2.108	1.062	0.596	0.1557	H_2O
	Σ_a	0.000531	0.000013	0.000953	0.018922	
	Σ_r	0.1071	0.1494	0.1518	0	
2	D	2.226	1.214	1.225	0.8172	D_2O
	Σ_a	0.000524	0	0.000003	0.000092	
	Σ_r	0.08408	0.03819	0.02027	0	
3	D	2.410	1.049	0.934	0.897	Carbon
	Σ_a	0.000003	0	0	0.000238	
	Σ_r	0.02752	0.01125	0.00657	0	

HAMMER Error Stops

Program CAPN reads all cards of type 0-5. These cards describe the path of control between the linked codes, and the structure of infinite lattices to be treated by THERMOS and/or HAMLET. These data are processed by CAPN, and extensive checking is done. Input errors are listed for each infinite lattice case, and that case is deleted from the batch. The following table defines the input errors which produce error messages.

Card types 6-11 are read by an input routine in FLOG. These data are checked only for sequence errors and invalid lattice/material identification numbers. These error messages are also described in the following table.

HAMMER Error Stops

Code	Number	Output Message	Remarks
CAPN	100	CARD 1 IS ABSENT	Batch lead card missing
	101	SEQ NO. IS BAD	First lattice case cards out of order
	102	REGION IS UNDEF.	Lattice region description card missing
	103	MESH IS BAD	Sum of mesh points $\neq$ NX (slabs)
	104	MESH IS BAD	Sum of mesh points $\neq$ NX (cylinders)
	105	UPSCAT IS OVER 3	More than 3 mixtures with field 8, card type 4 = 1
	106	MIXES IS OVER 10	>10 mixtures are defined
	107	POINTS IS OVER 20	>20 space points are defined
	108	REG S IS OVER 20	>20 regions are defined
	109	ISO S IS OVER 18	>18 isotopes are called for
	110	A MIX IS VOID	Some mixture contains no isotope of non-zero concentration
	111	ISO CD IS ABSENT	Number of type 5 cards following a type 4 is less than field 6

Code	Number	Output Message	Remarks
	112	MAT,L IS UNDEF	Some material number (field 5, card type 4) is not defined
	113	NTHRN IS BIG-3	A request for THERMOS version >3
	114	NEPRN IS BIG-3	A request for HAMLET version >3
	115	NFGRN IS BIG-3	A request for FLOG version >3
	116	NEDRN IS BIG-3	A request for DIED version >3
	117	MESH IS BAD	Request for code to assign mesh points cannot be met
	118	THLIB IS BAD	Some isotope called for is not on the THERMOS cross-section library
	119	EP-LIB IS BAD	Same as 118, except HAMLET library
THERMOS	200	- None -	Same as 118
HAMLET	300	"	Same as 119
	400	"	No reactor lead card (type 6)
	401	"	First case cards out of order
	402	"	Region missing
FLOG	403	"	Lattice not on library
	405	"	Material ID <0
	406	"	Material ID not in built-in lib
	404	"	= FOG - N100 error stop

Data Set References

All data set references in the HAMMER system are symbolic, and are established for the entire system of codes by subroutine TAPE in program CAPN. The assignment and usage of these data sets (for IBM 7090 tapes) are given in the following tables.

Data Set Assignments

Symbol	Data Set	Description	Disposition	Read Only ?
-	1	Monitor system		Yes
INT2,INT1,NSC1	8, 3	Scratch	Scratch	No
LIB3	2	Lattice library*	Save*	No-Use caution
NTIN	5	Input	-	
NTØUT	6	Output	Print	
NTPUN	7	Punch	-	
NCHAIN	4	HAMMER program	Save	Yes
LIB1	9	THERMOS library	Save	Yes
LIB2	10	HAMLET library	Save	Yes

* The code stacks lattice output (binary !!) on the lattice library for subsequent use by FLOG and DIED. "Standard" lattices likely to be used again as a reactor region should be stacked on a lattice library and filed away to save repeating THERMOS/HAMLET cases. One shot calculations should be stacked on a scratch file to keep the permanent library search time short.

Batch Control versus Data Set Utilization

Code	Card Type 0 Column	Batch Type 0[a]	1	2	3	4	5	6	7	8	9
CAPN	1	2	1	1	1	1	1	1	1	1	1
THERMØS	13	1[b]	1,2	0	1,2	1,2	0	0	1,2	1,2	0
HAMLET	25	1[c]	0	1	1	1	0	0	1	1	0
FLØG	37	0	0	0	0	0	0	1	1	1	1
DIED	49	0	0	0	0	1,2	1,2	0	0	1	1

Symbol	Data Set	Usage (Y=Yes) (N=No)	0	1	2	3	4	5	6	7	8	9
INT2,INT1	8	Scratch	Y[c]	Y	Y	Y	Y	N	N	Y	Y	Y
NSC1	3	Scratch	Y[b]	Y	N[d]	Y	Y	N	N	Y	Y	Y
LIB3	2	Lattice	N	N	N	Y	Y	Y	Y[e]	Y	Y	Y[e]
NCHAIN	4	HAMMER	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y
LIB1	9	THERMOS	Y[b,f]	Y	N	Y	Y	N	N	Y	Y	N
LII32	10	HAMLET	Y[c]	N	Y	Y	Y	N	N	Y	Y	N

a This batch type preprocesses THERMOS/HAMLET data; FLOG data should not be included

b If THERMOS will be run with these data

c If HAMLET will be run with these data

d Will be used if resonance integrals are calculated by the integral equation treatment

e Will not be used if no reactor region takes data from lattice library (card type 10, field 4 $\neq$ 0)

f Will be used if any region is not assigned mesh points (card type 4, field 7 = 0)

OUTPUT INFORMATION

General

There are two modes of HAMMER output controlled by the second option for each program on the batch control card (card type 0; see input). In the short print mode, only summary information is selected from the edited quantities for printing; in the long print mode, all edited quantities are printed. In the following sections, quantities are described for the long print mode. An asterisk indicates those items which appear also in the short print mode. The primary quantity which the code computes is the macroscopic space energy distribution of neutrons either in an asymptotic core (lattice calculation) or in the composite reactor (reactor calculation). For the lattice calculation, the macroscopic distribution is assumed to be the product of the scalar neutron density per unit volume and energy as a function of position in the cell, times the homogenized-cell leakage spectrum correction for the asymptotic core. For the reactor calculation, the macroscopic flux is that which results from solution of the four-group diffusion equation in the finite reactor geometry. The edited quantities are primarily averages of these spectra over cross sections. The output from each program is described in the following separate sections.

CAPN Output

* The first block of the output from CAPN indicates the options which the user has selected for each of the five codes.

```
*HAMMER*  BNL-SRL LATTICE ANALYSIS CODE

OPTIONS
  CAPN      1 0-0-0-0-0-0-0-0-0-0-0-0
  THERMOS   2 0-0-0-0-0-0-0-0-0-0-0-0
  HAMLET    1 0-0 1 1 1 1 1 1-0-0-0
  FLOG      1 0-0 1 1-0-0-0-0-0-0-0
  DIED      1 0 1 1 3 1 1-0-0-0-0-0

BATCH13066        1 LATTICES        1 CORES      TEST OF SINGLE RCC H2O CASES
```

The second section is simply a printout of the input cards before any interpretation has been performed. Above each datum, CAPN indicates how it will interpret that number, so that the user may be sure he has entered his input data in the proper field on each card.

CASE 11006 1/16 METAL 1/1 M/F H20

INPUT CARDS

CARD	GEOM	REGS	MIX	P1-B1	BCT	FOILS	BUCKLING	FOIL 1	FOIL 2	FOIL 3	FOIL 4
2	3	-0	-0	-0	-0	2	1.00000E 01	1.92235E 05	4.80000E 04	-0.	-0.

	REG	MIX	MAT	ADD I	MESH	H-L	REG OD	DENSITY	WGT PCNT	TEMPERATURE	FISS/CC
3	1	1	100	-0	-0	-0	6.25000E-02	1.89000E 01	3.00000E 00	2.00000E 01	-0.

	REG	MIX	MAT	ADD I	MESH	H-L	REG OD	DENSITY	WGT PCNT	TEMPERATURE	FISS/CC
4	2	2	300	-0	-0	1	8.83884E-02	1.00000E 00	-0.	2.00000E 01	-0.

* CAPN reads the THERMOS and HAMLET library tapes if they will be required for this case and prints their labels.

LIBRARY TAPE LABELED E-V-9 6/02/66

EPITHERMAL LIBRARY TAPE LABELED E-V-10 6-6-66

CAPN prints the isotope table. This is a list of the isotopes whose cross sections will be required for this case versus their concentrations in the mixtures which have been defined for this case. In some cases, CAPN will renumber the mixtures as compared to their input numbering. Since this renumbering is a purely internal affair, it need not concern the user.

ISOTOPE TABLE

IDA	T/M	MIX 1	MIX 2
92235.	0.	0.0014537	0.
92238.	0.	0.0464102	0.
1001.1029.	0.		0.0669444
8000.	29.	0.	0.0334722
192235.	0.	FOIL	
48000.	0.	FOIL	

CAPN prints the region tables. For each region the number of space points which have been assigned to that region, its thickness in centimeters, its areal volume in cm^2, the temperatures in ^oC, the mixture number corresponding to the isotope table, the heavy/light moderator approximation to be used, and the fission distribution for that region are listed across the page.

REGION TABLES

REG	PTS	THICK	VOLUME	DEG C	MIX	H-L	FISS/CC
1	1	0.07937	1.9793E-02	20.00	1	-0	-0.
2	1	0.03288	1.9793E-02	20.00	2	1	-0.

THERMOS Output

* Following the THERMOS library tape label and geometry options, information relating to the iteration process for the space energy neutron distribution is printed. The quantity IT is the number of iterations required to converge the solution. This number should be less than 150, the maximum value permitted in the code, if the flux has converged. The quantity NORM is the renormalization factor applied at the last iteration. It should equal unity. The quantity CRIT is the pointwise convergence criterion which the code applied, and the quantity RES is the largest residual which remained when iteration ceased. Thus, RES should be less than CRIT if convergence was obtained.

```
1/16 METAL 1/1 M/F H20                                  CASE11006     THERMOS

LIBRARY TAPE LABELED   E-V-9  6/02/66

CYLINDRICAL GEOMETRY, COSINE CURRENTS

IT=  11   NORM=  1.0000   CRIT=  3.7737E-03   RES=  2.7471E-03
```

* The quantities ETA, F, and ETA*F all relate to fuel regions, where "fuel" is defined as a region in which thermal neutron fission occurs. All isotopes occurring in fuel regions are considered to be fuel isotopes. XA and NU*XF are the cell-averaged values of Σ_a and $\nu\Sigma_f$. The moderator absorption fraction is the number of neutrons absorbed in the outermost region of the cell relative to one neutron absorbed in the entire cell.

```
ETA= 1.81776   F= 0.97802   ETA*F= 1.77781   XA= 0.260580E-00
NU*XF= 0.463261E-00   MOD.ABS.FRACT= 0.02198
```

* The number of neutrons absorbed and the number of neutrons produced in fission per region are given in the following table.

ABSORPTION AND PRODUCTION BY ISOTOPE AND REGION

ISOTOPE	T/M	TYPE	REG 1	REG 2	REG
92235.	0.	ABS	8.587E-01	0.	
		PROD	1.778E 00	0.	
92238.	0.	ABS	1.194E-01	0.	
		PROD	0.	0.	
1001.	1029.	ABS	0.	2.198E-02	
		PROD	0.	0.	
800C.	29.	ABS	0.	0.	
		PROD	0.	0.	

* The next table gives: for each point, the radius associated with that point, the average neutron density, the average velocity, and the average flux in the volume associated with that point; and for each foil isotope specified, the activation (normalized to unity at point 1) and the average cross section. A volume average over all points in a given region is computed for these quantities. At the bottom of this table, the absolute activation at the origin of the cell for each of the foil isotopes is given so that the user may convert relative activations to absolute activations.

NEUTRON DENSITY, VBAR, AND FOIL ACTIVATION

N	R	DENSITY	VBAR	FLUX	FLUXVOL	ACT-	192235.-XSEC	ACT-	48000.-XSEC	ACT-
1	0.	4.7943E 01	1.9843	9.5133E 01	1.8830E 00	1.0000	2.673E 02	1.0000	2.880E 03	
AVERAGE		4.7943E 01	1.9843	9.513E 01	1.883E 00	1.0000		1.0000		
2	0.096	5.0575E 01	1.9525	9.8750E 01	1.9546E 00	1.0580	2.724E 02	1.0417	2.890E 03	
AVERAGE		5.0575E 01	1.9525	9.875E 01	1.955E 00	1.0580		1.0417		
				ABSOLUTE ACTIVATION AT R=0		2.543E 04		2.740E 05		

* In the next table, average cross sections are given for all isotopes appearing in the cell. In the section of the table labeled MACROSCOPIC, the output quantity is the macroscopic cross section times the flux integrated over the entire cell and divided by the flux integrated over the entire cell. In the section of the table entitled MICROSCOPIC, the value of the cross section is the flux volume integral over the entire cell of the macroscopic cross section divided by the flux volume integral over the entire cell of the isotopic concentration. These are the appropriate cross sections to multiply by the average flux volume in the cell to obtain the actual reaction rate in the cell.

AVERAGE CROSS SECTIONS

	MACROSCOPIC (1/CM)			MICROSCOPIC (BARNS)		
ISOTOPE	ABSORPTION	FISSION	NU*FISSION	ABSORPTION	FISSION	NU*FISSION
92235.	2.23751E-01	1.90642E-01	4.63261E-01	3.13691E 02	2.67273E 02	6.49475E 02
92238.	3.11005E-02	0.	0.	1.36573E 00	0.	0.
1001.	5.72779E-03	0.	0.	1.67987E-01	0.	0.
8000.	0.	0.	0.	0.	0.	0.

Microscopic cross sections are given by region for all isotopes in the cell. These are the cross sections averaged over the flux appearing in that region. The last two entries in this table are the flux and flux-volume by region, normalized so that the average flux in the fuel is unity.

MICROSCOPIC CROSS SECTIONS

ISOTOPE	SCALAR	REGION 1	2
92235.	A	0.E-00 3136	3196
	F	0.E-00 2672	2724
	NUF	0.E-00 6494	6619
92238.	A	1.E-03 1365	1387
1001.	A	1.E-03 165	167
	FLUX	1.E-03 999	1038
	FLUX*VOL	0.E-04 1979	2054

On the next page, the thermal flux is given as a function of space point and energy group. This flux, vN(v), is the zero buckling cell flux in dimensionless units, normalized to yield one absorption in the cell.

THERMAL FLUX, V*N(V)

GROUP	PT 1	PT 2	PT
1	2.8870E-02	5.7087E-02	
2	2.4341E-01	3.4857E-01	
3	8.1649E-01	1.0434E 00	
4	1.8902E 00	2.2782E 00	
5	3.5579E 00	4.1370E 00	
6	5.8501E 00	6.6387E 00	
7	8.7321E 00	9.7358E 00	
8	1.2109E 01	1.3321E 01	
9	1.5840E 01	1.7242E 01	
10	1.9752E 01	2.1317E 01	
11	2.3686E 01	2.5375E 01	
.			
.			
.			
20	2.9959E 01	3.0957E 01	
21	2.7517E 01	2.8365E 01	
22	2.5079E 01	2.5796E 01	
23	2.3098E 01	2.3719E 01	
24	1.9868E 01	2.0392E 01	
25	1.9212E 01	1.9758E 01	
26	1.7143E 01	1.7618E 01	
27	1.6776E 01	1.7099E 01	
28	1.6210E 01	1.6437E 01	
29	1.5179E 01	1.5344E 01	
30	1.3975E 01	1.4086E 01	

* At the bottom of this page, parameters derived from the Fourier transform leakage calculation are given. L2 is the diffusion area in cm^2, XA is the homogenized-cell absorption cross section, D is the diffusion constant, B^2 is the input value of the buckling, and IT is the number of iterations required to converge the leakage distribution. IT should be less than 200, the maximum permitted by the code.

P1 APPROX. L2= 1.17029E 00 XA= 2.60554E-01 D= 3.04955E-01 BSQD=10.00000E-04 IT= 16

HAMLET Output

* The label of the HAMLET library tape is printed at the head of the first page.

CASE 11006 1/16 METAL 1/1 M/F H2O

EPITHERMAL LIBRARY TAPE LABELED E-V-10 6-6-66

The first table of output relates to the computation of resonance integrals. For each resonance encountered in the calculations, the column labeled RI gives the effective value of the resonance integral used by the code. The column INFINITE RI gives the infinite resonance integral for this resonance. The column SHIELDING is the ratio RI/INF. RI. The geometrical cross section is the value of $S_{eff}/4V$ employed by the code in the resonance integral calculation. The column MESH gives the number of discrete lethargy intervals (fine groups) which the code used for the resonance integral calculation. A value of 0 indicates that this resonance was treated by equivalence principles. The next three columns, ABS for absorber, MOD1 for moderator 1, and MOD2 for moderator 2, indicate the number of mesh points within a collison interval for each of these isotopes. The column labeled CAPT. PROB. gives the capture probability for each resonance. All of the resonances appearing within a single broad group are summed to give the total effective resonance integral, the total infinite resonance integral, and the average shielding within that group.

RESONANCE INTEGRALS

GROUP	ISOTOPE	RI	INFINITE RI	SHIELDING	GEOM.CROSS SECTION	MESH	ABS	MOD1	MOD2	CAPT. PROB.
19	92238.	1.039689E-02	1.327940E-02	7.829336E-01	9.092317E-01	0	0	0	0	5.7825E-04
		1.039689E-02	1.327940E-02	7.829336E-01						
45	92238.	4.740976E 00	1.272038E 02	3.727071E-02	1.338866E 00	501	9	9	0	1.5691E-01
		4.740976E 00	1.272038E 02	3.727071E-02						
52	92235.	2.262114E 01	2.370487E 01	9.542826E-01	1.361520E 00	0	0	0	0	2.3044E-02
		2.262114E 01	2.370487E 01	9.542826E-01						
TOTALS										
	92235.	1.725618E 02	2.217273E 02	7.782614E-01						
	92238.	1.409128E 01	2.682237E 02	5.253554E-02						

* Between the last group values and the final grand totals over all groups, two lines of print appear, which give information relating to the convergence of the broad group flux calculation. The maximum iterations should be less than 100, the maximum built into the code, and the renormalization should not depart far from unity.

```
MAXIMUM ITERATIONS ON THE  FLUX SHAPE IN ANY MACRO GROUP =     2 GROUP 1
MAXIMUM RENORMALIZATION IN ANY GROUP =  0.99952 GROUP 6
```

The next page gives the space energy distribution computed at zero buckling. This is the distribution which is used to homogenize the cell on a flux volume basis in order to perform the Fourier transform leakage calculation.

ZERO BUCKLING FLUXES

GROUP	PT 1	PT 2	PT
1	5.3615E 00	5.3193E 00	
2	1.4928E 01	1.4803E 01	
3	3.3075E 01	3.2777E 01	
.			
.			
.			
52	1.6568E 01	1.6586E 01	
53	1.4883E 01	1.5016E 01	
54	1.4407E 01	1.4517E 01	

* The next page, OVERALL FEW-GROUP DATA, gives homogenized cell cross sections, capture escape probabilities, the age, and the diffusion coefficient for few-group schemes.

OVERALL FEW GROUP DATA

OUTPUT GROUP	MUFT GROUPS	SIGMA-A	SIGMA-R	SIGMA-F	NU SIGMA-F	P	AGE	DIFF.COEF.
1 OF 1	1 TO 54	0.024015	0.013877	0.007530	0.019091	0.366222	27.566	1.044496
1 OF 2	1 TO 25	0.008507	0.044623	0.004220	0.011184	0.839879	22.349	1.187604
2 OF 2	26 TO 54	0.063242	0.048978	0.015904	0.039091	0.436450	28.430	0.682492
1 OF 3	1 TO 10	0.010998	0.095130	0.009173	0.024645	0.896372	11.731	1.652862
2 OF 3	11 TO 25	0.006983	0.071932	0.001188	0.002946	0.911513	23.169	0.902860
3 OF 3	26 TO 54	0.063242	0.048978	0.015904	0.039091	0.436450	29.251	0.682492

The intermediate integrals used to produce these averages are listed below.

INTERMEDIATE INTEGRALS

CUTPUT GROUP	MUFT GROUPS	ABSORPTION	FLUX	FISSION	NU-FISSION	SOURCE	CURRENT
1 OF 1	1 TO 54	0.616776	25.683380	0.193398	0.490323	1.000000	0.848318
1 OF 2	1 TO 25	0.156591	18.406754	0.077673	0.205870	0.999810	0.691272
2 OF 2	26 TO 54	0.460185	7.276625	0.115726	0.284453	0.000190	0.157046
1 OF 3	1 TO 10	0.076855	6.988269	0.064105	0.172226	0.753200	0.365263
2 OF 3	11 TO 25	0.079735	11.418486	0.013568	0.033644	0.246610	0.326009
3 OF 3	26 TO 54	0.460185	7.276625	0.115726	0.284453	0.000190	0.157046

This output is based on the homogenized-cell leakage corrected flux at the input buckling. P is not the normally defined resonance escape probability, but is the ratio of neutrons slowing down out of the broad group divided by the neutrons slowing down plus those captured in smooth and resonance events by all isotopes.

The next page gives for each group: the flux and current per unit lethargy; η, the isotropic component of the slowing down density due to the hydrogen; ρ, the P_1 component of slowing down due to hydrogen; and q, the heavy element slowing down density.

CASE 11006 1/16 METAL 1/1 M/F H20

54 GROUP EDIT

GROUP	FLUX	CURRENT	ETA	RHO	Q
1	0.207157	0.016104	0.001573	0.000116	0.000150
2	0.576117	0.043590	0.006611	0.000465	0.000851
3	1.280372	0.090057	0.019766	0.001293	0.002764
.					
.					
.					
47	0.557125	0.012089	0.388206	0.005760	0.004972
48	0.524629	0.011602	0.381948	0.005617	0.004842
49	0.545542	0.011700	0.380317	0.005535	0.005135
50	0.537924	0.011503	0.378056	0.005453	0.004954
51	0.537547	0.011368	0.376657	0.005386	0.005006
52	0.480777	0.010604	0.367242	0.005236	0.004308
53	0.476903	0.009821	0.358755	0.005001	0.004491
54	0.464559	0.009428	0.352132	0.004822	0.004266

* The next page repeats one-group values of p, $\nu(p_f-1)$ the probability of fission, i.e. $\frac{\nu\Sigma_f}{\Sigma_a + \Sigma_r}$, and τ. If THERMOS has been run in this batch (so that a full, four-group set of cross sections are available), non-zero values will be provided for the material buckling, k_∞, k_{eff} at the input buckling, and the four-group fluxes which exist in the asymptotic core. These latter are energy integrated fluxes. If convergence to the material buckling cannot be obtained, a message to this effect will appear on the bottom of the preceding page.

CASE 11006 1/16 METAL 1/1 M/F H20

P = 3.662220E-01 NU(1-PF) = 5.038394E-01 TAU = 2.756566E 01
MATERIAL BUCKLING = 5.439608E-03 K-INFINITY = 1.154142E 00 K-EFFECTIVE AT INPUT BUCKLING = 1.123189E 00
ASYMPTOTIC FOUR GROUP FLUXES, 6.542813E 00, 1.036696E 01, 6.434009E 00, 1.201683E 00

For each isotope in the cell, two pages of output appear. The first page gives, for each of the 54 groups, the number of fissions occurring, the number of fission neutrons produced, the number of smooth absorptions, the number of resonance absorptions, and the number of resonance fissions.

ELEMENT NUMBER 92235. INTERMEDIATE INTEGRALS

GROUP	FISSION	NU-FISSION	SM. ABS.	RES. ABS.	RES. FISS.
1	0.00007381	0.00026367	0.00007491	0.	0.
2	0.00015558	0.00051771	0.00016037	0.	0.
3	0.00025965	0.00081363	0.00027198	0.	0.
.					
.					
.					
30	0.00382216	0.00939488	0.00554611	0.	0.
31	0.00524593	0.01289450	0.00765366	0.	0.
32	0.00651253	0.01600780	0.00954739	0.	0.
33	0.00344443	0.00846642	0.00506330	0.	0.
34	0.00317877	0.00781341	0.00468067	0.	0.
35	0.00351968	0.00865140	0.00519135	0.	0.
36	0.00360178	0.00885318	0.00531985	0.	0.
37	0.00431355	0.01060271	0.00638025	0.	0.
38	0.00750013	0.01843533	0.01110009	0.	0.
39	0.00789192	0.01939835	0.	0.01144931	0.00789192
40	0.00698971	0.01718070	0.	0.00953206	0.00698971
41	0.00776251	0.01908024	0.	0.01146073	0.00776251
42	0.00423707	0.01041473	0.	0.00591619	0.00423707
43	0.00351869	0.00864894	0.	0.01037818	0.00351869
44	0.00854985	0.02101554	0.	0.01363763	0.00854985
45	0.00080991	0.00199077	0.	0.00197197	0.00080991
46	0.00354960	0.00872492	0.	0.00615644	0.00354960
47	0.00022511	0.00055332	0.	0.00163207	0.00022511
48	0.00362150	0.00890164	0.	0.00546888	0.00362150
49	0.00017309	0.00042547	0.	0.00026830	0.00017309
50	0.00035092	0.00086257	0.	0.00125747	0.00035092
51	0.	0.	0.	0.	0.
52	0.00662712	0.01628947	0.	0.00879489	0.00662712
53	0.00567978	0.01396090	0.00650334	0.	0.
54	0.00469206	0.01153307	0.00491023	0.	0.

The second page gives macroscopic cross sections for each of these processes. These are flux-volume weighted macroscopic cross sections, and are the appropriate ones to multiply the cell average integrated flux to obtain the reaction rate in the cell.

ELEMENT NUMBER 92235. FEW GROUP DATA

OUTPUT GROUP	MUFT GROUPS	SMOOTH SIGMA-A	RESONANCE SIGMA-A	TOTAL SIGMA-F	RESONANCE SIGMA-F	NU SIGMA-F
1 OF 1	1 TO 54	0.00426606	0.00342339	0.00525399	0.00211447	0.01299264
1 OF 2	1 TO 25	0.00127847	0.	0.00104389	0.	0.00267521
2 OF 2	26 TO 54	0.01182339	0.01208310	0.01590373	0.00746322	0.03909137
1 OF 3	1 TO 10	0.00091100	0.	0.00082300	0.	0.00226804
2 OF 3	11 TO 25	0.00150336	0.	0.00117908	0.	0.00292439
3 OF 3	26 TO 54	0.01182339	0.01208310	0.01590373	0.00746322	0.03909137

* Following the cell isotope edits, a listing of the few-group cross sections versus space point within the cell is provided for foil materials.

FOIL 192235.

PT	GROUP 1	GROUP 2	GROUP 3	GROUP 4
1	1.125789E 00	1.617495E 00	2.063837E 01	2.672729E 02
2	1.125854E 00	1.620408E 00	2.065057E 01	2.724170E 02

The leakage corrected neutron spectrum by point and group is the product of the normalized space energy distribution within the cell at zero buckling times the leakage correction spectrum.

LEAKAGE CORRECTED SPECTRUM

GROUP	PT 1	PT 2	PT
1	5.2537E 00	5.2123E 00	
2	1.4614E 01	1.4492E 01	
3	3.2490E 01	3.2197E 01	
.			
.			
.			
34	2.1659E 01	2.1745E 01	
35	2.2453E 01	2.2532E 01	
36	2.0896E 01	2.0992E 01	
37	2.2067E 01	2.2167E 01	
38	2.1143E 01	2.1333E 01	
39	1.4980E 01	1.4983E 01	
40	1.8912E 01	1.8915E 01	
41	1.2295E 01	1.2300E 01	
42	1.7076E 01	1.7077E 01	
43	1.6697E 01	1.6710E 01	
44	1.5634E 01	1.5638E 01	
45	5.4027E 00	5.4069E 00	
46	1.2825E 01	1.2824E 01	
47	1.4064E 01	1.4083E 01	
48	1.3251E 01	1.3255E 01	
49	1.3771E 01	1.3791E 01	
50	1.3584E 01	1.3593E 01	
51	1.3568E 01	1.3590E 01	
52	1.2138E 01	1.2152E 01	
53	1.1994E 01	1.2101E 01	
54	1.1690E 01	1.1780E 01	

Finally, the few-group fluxes within the cell are output. Again, these are energy integrated fluxes normalized to the asymptotic core fluxes.

FEW GROUP FLUXES

GROUP	PT 1	PT 2
1	1.6623E 02	1.6432E 02
2	2.6264E 02	2.6112E 02
3	1.6236E 02	1.6270E 02
4	2.9790E 01	3.0922E 01

FLOG Output

* As in CAPN, the raw input data, labeled to indicate how FLOG will interpret these data, are printed out.

```
CASE     1      TEST FLOG                                                    FLOG

INPUT CARDS
CARD   GEOM   REGS   S-OPT   NVL   S-TYP        EIGENVALUE
  2      2      1       1    -0     -0        1.00000E 00 -0.          -0.          -0.          -0.

       REG   LIB    MAT          MESH          THICKNESS   BUCKLING     ADDED XA     CONTROL XA     POIS ID
  3      1     0 11006           -0          1.89370E 01  2.93200E 01 -0.          -0.          -0.
```

The next page is the interpreted data: N1 is the geometry option; N20, the search option; EIGEN2, the value of k_{eff} to be searched on; and EPS1, the pointwise convergence criterion for the flux calculation.

```
INTERPRETED DATA
N1= 2   N20= 1   EIGEN2= 1.00000   ESP1=  1.0000E-05
```

* A reactor region table is output, giving for each region: its thickness in centimeters, the number of points which the user has assigned to that region, the axial buckling of that region in cm^{-2}, the thermal group control poison distribution, and the four-group cross section data to be employed in that region.

```
REG WIDTH(CM) POINTS BUCKLING   CONT POIS GROUP     D          XA          NU*XF       XR          XF
 1   48.1000     -0   2.932E-03 -0.          1    1.652862   0.010998   0.024645   0.095130   0.009173
                                             2    0.902860   0.006983   0.002946   0.071932   0.001188
                                             3    0.682492   0.063242   0.039091   0.048978   0.015904
                                             4    0.304955   0.260580   0.463261              0.190642
```

The standard input options which the code will employ are printed out along with radial increments in centimeters.

```
INPUT OPTIONS

GEOMETRY N1=2     BOUNDARY CONDITION AT ORIGIN N2=0     OUTER BOUNDARY CONDITION N3=0
CONVERGENCE N4=1     EXTRAPOLATION FACTOR N5=0     SOURCE N6=0
BUCKLING BY REGION N7=1     BUCKLING BY GROUP N8=0     CHI OPTION N9=1
OUTPUT FORMAT N10=0     FLUX PRINT N11=0     GROUP DATA PRINTOUT N12=0
FLUX MULTIPLICATION N13=0     FLUX NORMALIZATION N14=0     N15=  0     N16=0
GROUP PRINT N17=0     CHECKS N18=1     BUCKLING ITERATION N19=0
CRITICALITY SEARCH N20= 1     ADJOINT FLUX N22=0     ADJOINT FLUX OUTPUT N23=0
```

```
DIMENSIONS

NO. OF REGIONS M= 1      NO. OF POINTS N=150      NO. OF GROUPS NOG= 4
NO. OF GROUPS FROM WHICH THERE IS SCATTERING NDS= 1

 1 NO. OF INTERVALS IN REGIONS

LM=   149

POINTWISE CONVERGENCE EPS1=  0.09999999E-04

EXTRAPOLATION FACTOR

THETA=  0.80000000E 00

R(1)=  0.

 1 DELTA R

DELR=  0.32281864E-00

CRITICALITY SEARCH - CONVERGENCE CRITERION EPS2=  0.09999999E-04

CONTROL REGIONS
              - SEARCH IN ALL REGIONS
```

Few-group cross sections are given by region and group.

```
DATA FOR GROUP  4

T=  0.

                          REGION  1
CHI                       0.
SIGMA POISON              -0.
BUCKLING                  0.29319999E-02
SIGMA ABSORPTION          0.26057974E-00
TOTAL REMOVAL             0.26147386E-00
DIFFUSION COEFFICIENT     0.30495502E-00
NU SIGMA FISSION          0.46326058E-00
TRANSFER COEFFICIENTS     0.48994840E-01
```

At each iteration in the eigenvalue search, the current value of the eigenvalue, the previous value of the eigenvalue, and the current value for the search parameter are listed. No more than 10 such iterations are permitted.

```
THE EIGENVALUE TO BE SEARCHED FOR IS .09999999E 01

ITERATION COUNT   EIGENVALUE      PREVIOUS VALUE OF EIGENVALUE      BUCKLING
      1        1.0002037E 00        1.0002034E 00              2.9319999E-03
      2        1.0001733E 00        1.0001735E 00              2.9331942E-03
      3        1.0000000E 00        9.9999996E-01              2.9400226E-03
```

* The results of the J_o fit to the thermal flux are given in the form of data pairs. Fitting is done in the central region of the core. All points from the origin out to successive values of R are considered. For each value of R_{max} the corresponding value of the reflector savings, λ, is printed out. A value of -1.000 for the reflector savings indicates that the code was not able to obtain a fit. Fitting is continued for successively higher values of R_{max} until the code can no longer obtain a fit to λ.

```
JO FIT TO THERMAL FLUX, (MAX R, LAMBDA)  CORE RADIUS =  48.1CC
(  0.646,-1.000)  (  0.968,-1.000)  (  1.291, 0.000)  (  1.614, 0.000)  (  1.937, 0.000)  (  2.260, 0.000)
(  2.583, 0.000)  (  2.905, 0.000)  (  3.228, 0.000)  (  3.551, 0.000)  (  3.874, 0.000)  (  4.197, 0.000)
(  4.519, 0.000)  (  4.842, 0.000)  (  5.165, 0.000)  (  5.488, 0.000)  (  5.811, 0.000)  (  6.134, 0.000)
(  6.456, 0.000)  (  6.779, 0.000)  (  7.102, 0.000)  (  7.425, 0.000)  (  7.748, 0.000)  (  8.070, 0.000)
(  8.393, 0.000)  (  8.716, 0.000)  (  9.039, 0.000)  (  9.362, 0.000)  (  9.685, 0.000)  ( 10.007, 0.000)
( 10.330, 0.000)  ( 10.653, 0.000)  ( 10.976, 0.000)  ( 11.299, 0.000)  ( 11.621, 0.000)  ( 11.944, 0.000)
( 12.267, 0.000)  ( 12.590, 0.000)  ( 12.913, 0.000)  ( 13.236, 0.000)  ( 13.558, 0.000)  ( 13.881, 0.000)
```

The next output section, FLOG EDIT, gives few-group fluxes and the power density (assumed proportional to the fission density) by space point.

FLOG EDIT, FLUX AND POWER BY SPACE POINT

POINT	REGION	RADIUS	FLUX 1	FLUX 2	FLUX 3	FLUX 4	POWER
1	1	0.	2.0850E-03	3.3036E-03	2.0503E-03	3.8294E-04	1.2866E-04
2	1	0.3228	2.0849E-03	3.3034E-03	2.0502E-03	3.8291E-04	1.2866E-04
3	1	0.6456	2.0844E-03	3.3028E-03	2.0498E-03	3.8284E-04	1.2863E-04
.							
.							
.							
148	1	47.4543	3.5171E-05	5.5727E-05	3.4586E-05	6.4596E-06	2.1704E-06
149	1	47.7771	1.7528E-05	2.7773E-05	1.7237E-05	3.2193E-06	1.0817E-06
150	1	48.1000	0.	0.	0.	0.	0.

* For each region, the average flux, power, power density, and peak-to-average power is output, followed by reactor averages.

FLOG EDIT, REGION AVERAGE FLUX AND POWER

REGION	FLUX 1	FLUX 2	FLUX 3	FLUX 4	REG. POWER	SPEC. POWER	PEAK/AVE POWER
1	9.0017E-04	1.4263E-03	8.8520E-04	1.6533E-04	4.0375E-01	5.5549E-05	2.31622
REACTOR	9.0017E-04	1.4263E-03	8.8520E-04	1.6533E-04	4.0375E-01	5.5549E-05	2.31622

* A neutron balance sheet shows the number of neutrons absorbed, the number of neutrons which leak out of the reactor in the axial and radial directions, the number of neutrons absorbed in control poison, and the production of neutrons. The column EIGEN ABS gives the number of neutrons lost to the system when k_{eff} is not equal to unity.

FLOG EDIT, REGION ABSORPTION AND PRODUCTION

REGION	ABS	Z LEAKAGE	R LEAKAGE	NET LEAKAGE	CONTROL ABS	TOT. ABS +LEAKAGE	PROD	EIGEN ABS
1	864.3803	73.3004	62.3195	135.6199	-0.	1000.0002	1000.0002	0.
REACTOR	864.3803	73.3004	62.3195	135.6199	0.	1000.0002	1000.0002	0.

* The results of the point buckling calculation are given for each central core considered. The neutron balance at the origin of the reactor is used to determine the radial buckling at that point and the reflector savings at that point.

	POINT RADIAL BUCKLINGS X 10000				AND REFLECTOR SAVINGS			
CORE	RADIUS	GROUP 1	GROUP 2	GROUP 3	GROUP 4	GROUP 1	GROUP 2	GROUP 3	GROUP 4
1	48.100	24.996	24.996	24.996	24.996	0.0006	0.0007	0.0005	0.0007

DIED Output

Version 1

* DIED Version 1 is normally used following a lattice and/or reactor calculation. There are three sections of output. Neutron balance sheets based on 1000 neutrons are printed out. If Column 56, card type 0, of the input is equal to 0, the balance is cast in the form of number of absorption events versus number of fission events. If Column 56 of card type 0 equals 1, the balance is cast in the form of number of neutrons absorbed versus number of neutrons produced in fission. No simple relationship exists between these two forms since the number of neutrons produced per fission is a function of energy and isotope. The first of these tables is appropriate for comparison with experimental measurements and the second is appropriate for reactivity considerations. These tables may be obtained in one of two forms controlled by Column 52 of card type 0. If that column equals 1, events are separated into smooth and resonance events. If that column equals 2, the smooth and resonance events are summed to give only total events. In either case, events are separated by group and by isotope; losses to control poison, leakage, and eigenvalue absorptions are also shown.

LATTICE 11006 1/16 METAL 1/1 M/F H2O

REGION 1 AVERAGE FLUX WEIGHTING

NEUTRON BALANCE SHEET BASED ON 1000 NEUTRONS

ISO-TOPE	SMOOTH ABSORPTION GROUP 1	GROUP 2	GROUP 3	GROUP 4	TOTAL	RESONANCE ABSORPTION GROUP 1	GROUP 2	GROUP 3	GROUP 4	TOTAL	TOTAL ABSORPT.
92235.	5.9605	15.5853	76.0718	268.8783	366.4959	0.	0.	77.7428	0.	77.7428	444.2387
92238.	64.7455	43.1466	11.8669	37.3730	157.1319	0.	13.5966	239.0219	0.	252.6185	409.7505
8000.	1.2418	0.	0.	0.	1.2418	0.	0.	0.	0.	0.	1.2418
1001.	0.0085	0.0641	2.1936	6.8830	9.1492	0.	0.	0.	0.	0.	9.1492
OTHER				0.	0.						0.
	71.9564	58.7960	90.1323	313.1342	534.0189	0.	13.5966	316.7647	0.	330.3613	864.3802
CNTRL	-0.	-0.	-0.	-0.	0.						0.
ZLEAK	31.7945	27.5184	12.9101	1.0774	73.3004						73.3004
RLEAK	27.0315	23.3960	10.9760	0.9160	62.3194						62.3194
LEAK	58.8259	50.9144	23.8861	1.9934	135.6198						135.6198
EIGEN	0.	0.	0.	0.	0.						0.
TOTAL	130.7823	109.7104	114.0184	315.1276	669.6386	0.	13.5966	316.7647	0.	330.3613	1000.0000

ISO-TOPE	SMOOTH FISSION GROUP 1	GROUP 2	GROUP 3	GROUP 4	TOTAL	RESONANCE FISSION GROUP 1	GROUP 2	GROUP 3	GROUP 4	TOTAL	TOTAL FISSION
92235.	5.3847	12.2235	54.3064	229.0915	301.0061	0.	0.	48.0184	0.	48.0184	349.0245
92238.	54.6336	0.0951	0.	0.	54.7288	0.	0.	0.	0.	0.	54.7288
8000.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
1001.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	60.0184	12.3187	54.3064	229.0915	355.7349	0.	0.	48.0184	0.	48.0184	403.7533

AVERAGE NEUTRONS PER FISSION 2.4768

TOTAL NEUTRONS 1000.0000

If DIED is being used following a FLOG run, the user has his choice of several flux weightings to be used in computing the balance sheets. If Column 53 of card type 0 is 0, the average flux (in the reactor region in which this lattice has been placed) will be used for the neutron balance. If Column 53 equals 1, the balance tables will also be computed based on the asymptotic flux for that lattice. If Column 53 equals 2, the fluxes existing at the interfaces of that region will also be used; and if Column 53 equals 3, the flux existing at the midpoint of this reactor region will also be used. These latter flux weightings should be useful in determining how nearly a reactor region approaches its asymptotic behavior. If a reactor calculation has not been done, only the asymptotic flux weighting is available.

* The next type of output available from DIED Version 1 is neutron balance sheets in abbreviated form, displaying the total number of neutrons produced and the total number of neutrons absorbed by each isotope in each region of the reactor.

CASE 1 TEST FLNG
REACTOR REGION NEUTRON BALANCES

ISO-TOPE	REGION 1 LATTICE 11006 ABS	PROD
92235.	0.44424	0.85336
92238.	0.40975	0.14664
CTHER	0.01039	0.
SUB	0.86438	1.00000
CNTRL	0.	
ZLEAK	0.07330	
RLEAK	0.06232	
LEAK	0.13562	
EIGEN	0.	
TOTAL	1.00000	1.00000

Column 54 of card type 0 controls those regions for which results will be printed. If it is 0, no results will be obtained; if it is 1, all regions in the reactor will be considered; and if it is 3, a card will be read to indicate whether or not each region is to be considered. This card should follow all lattice and reactor data for this batch.

* Few-group cross sections averaged over regions within the reactor are available. The regionwise options are the same as for the regionwise balance sheets. Column 55 of card type 0, however, controls the cross section options. A four-group set by isotope and group is provided, as well as a two-group set in which groups 1, 2, and 3 of the four-group set are collapsed into a single group, given as group 1, and group 4 is simply reprinted as group 2. A one-group set is also available in which all four groups are collapsed to one group. The averaging is done on a flux-volume basis over all the regions specified by the region option.

FEW GROUP CROSS SECTIONS AVERAGED OVER REGIONS 1,

FOUR GROUP SET

ISO-TOPE	GROUP 1 ABSORPTION	GROUP 1 NUFISSION	GROUP 2 ABSORPTION	GROUP 2 NUFISSION	GROUP 3 ABSORPTION	GROUP 3 NUFISSION	GROUP 4 ABSORPTION	GROUP 4 NUFISSION
92235.	9.11004E-04	2.26804E-03	1.50336E-03	2.92440E-03	2.39065E-02	3.90914E-02	2.23751E-01	4.63261E-01
92238.	9.89567E-03	2.23770E-02	5.47346E-03	2.20276E-05	3.89942E-02	0.	3.11005E-02	0.
CTHER	1.91104E-04	0.	6.17958E-06	0.	3.40943E-04	0.	5.72779E-03	0.
CONTROL	0.	0.	0.	0.	0.	0.	0.	0.
D	1.65286E 00		9.02860E-01		6.82492E-01		3.04955E-01	
REM	9.51300E-02		7.19324E-02		4.89784E-02			

TWO GROUP SET

ISO-TOPE	GROUP 1 ABSORPTION	GROUP 1 NUFISSION	GROUP 2 ABSORPTION	GROUP 2 NUFISSION
92235.	7.51208E-03	1.27088E-02	2.23751E-01	4.63261E-01
92238.	1.59519E-02	6.28163E-03	3.11005E-02	0.
CTHER	1.50278E-04	0.	5.72779E-03	0.
CONTROL	0.	0.	0.	0.
D	1.05233E 00		3.04955E-01	
REM	1.34994E-02			

ONE GROUP SET

ABSORPTION	NUFISSION
1.01647E-01	2.07748E-01
2.83065E-02	5.97410E-03
2.56210E-03	0.
0.	0.
1.01574E 00	

Version 2

DIED Version 2 provides the user with the capability of dumping a lattice library tape. Lattice after lattice is read in from the tape, and the data which exist on the tape is labeled and printed.

* If the short print option has been selected, only a table of contents will be given, listing the identification number and the Hollerith identification of each lattice on the tape.

```
*HAMMER* EDIT2

LISTING OF LATTICE LIBRARY AS OF 00000000

EDIT OPTION  6

LATTICE NUMBER  11006, RUN ON 00000000 AT  0.    , BY *HAMMER*
**********
TITLE...1/16 METAL 1/1 M/F H2O                                    ...

LIBRARY CONTAINS    1 LATTICES
```

Long print options 1-6 appear as follows:

Print Options 1 2 3 4 5 6

```
THERMOS LIBRARY E-V-9  6/21/66     ,  HAMLET LIBRARY E-V-10 08-01-66

MISCELLANEOUS PARAMETERS
INPUT BUCKLING=  10.00000 (M-2)                                  ETA=    1.81776
 THERMAL UTIL.=    0.97802          FRACT. THERMAL ABS. IN MOD.=    0.02198
            P=    0.36622                          NU*(1-PF)=    0.50384
          AGE=   27.56566              MATERIAL BUCKLING(M-2)=   54.39608
    K INFINITY=    1.15414          EIGENVALUE AT INPUT BUCKLING=   1.12319
FEW GROUP ASYMPTOTIC FLUXES
PHI1= 6.5428E 00  PHI2= 1.0367E 01  PHI3= 6.4340E 00  PHI4= 1.2017E 00

FEW GROUP CROSS SECTIONS BY ISOTOPE, ARRAY BALT                  CASE  11006
ISOTOPE GRP  XA(TOT)      XF(TOT)      NU*XF(TOT)   XA(RES)      XF(RES)
------- --- ----------   ----------   ----------   ----------   ----------
  92235. 1  9.1100E-04   8.2300E-04   2.2680E-03   0.           0.
         2  1.5034E-03   1.1791E-03   2.9244E-03   0.           0.
         3  2.3906E-02   1.5904E-02   3.9091E-02   1.2083E-02   7.4632E-03
         4  2.2375E-01   1.9064E-01   4.6326E-01   0.           0.
  92238. 1  9.8957E-03   8.3502E-03   2.2377E-02   0.           0.
         2  5.4735E-03   9.1782E-06   2.2028E-05   1.3115E-03   0.
         3  3.8994E-02   0.           0.           3.7150E-02   0.
         4  3.1101E-02   0.           0.           0.           0.
   1001. 1  1.3001E-06   0.           0.           0.           0.
         2  6.1796E-06   0.           0.           0.           0.
         3  3.4094E-04   0.           0.           0.           0.
         4  5.7278E-03   0.           0.           0.           0.
   8000. 1  1.8980E-04   0.           0.           0.           0.
         2  0.           0.           0.           0.           0.
         3  0.           0.           0.           0.           0.
         4  0.           0.           0.           0.           0.

AVE FOIL CROSS SECTION BY GROUP AND MESH POINT                   CASE  11006
  FOIL   REGION   GROUP 1     GROUP 2     GROUP 3     GROUP 4
-------- ------ ----------  ----------  ----------  ----------
 192235.   1     1.1113E 00  1.6175E 00  2.0637E 01  2.6727E 02
           2     1.1114E 00  1.6204E 00  2.0650E 01  2.7242E 02
  48000.   1     2.4883E-02  2.0980E-01  2.6253E 00  2.8800E 03
           2     2.4857E-02  2.1044E-01  2.6313E 00  2.8900E 03
```

FEW GROUP DATA FOR FLOG CALCULATIONS, ARRAY FOGX CASE 11006

GROUP	D	XA	NU*XF	XR	XF	CHI
1	1.6529E 00	1.0998E-02	2.4645E-02	9.5130E-02	9.1732E-03	7.5320E-01
2	9.0286E-01	6.9830E-03	2.9464E-03	7.1932E-02	1.1883E-03	2.4661E-01
3	6.8249E-01	6.3242E-02	3.9091E-02	4.8978E-02	1.5904E-02	1.8998E-04
4	3.0496E-01	2.6058E-01	4.6326E-01	0.	1.9064E-01	0.

AVERAGE FLUX BY GROUP AND SPACE MESH POINT, ARRAY FLX, FISD CASE 11006

MESH POINT	GROUP 1	GROUP 2	GROUP 3	GROUP 4	TH. FISS./CC
1	1.6623E 02	2.6264E 02	1.6236E 02	2.9790E 01	
69012	0.0115E-39	1.6432E 02	2.6112E 02	1.6270E 02	
68590	0.				

OVERLAPPING THERMAL GROUP DATA, ARRAYS OTGC AND OTGA CASE 11006
AVE S(V)= 2.1145E-01 AVE V*S(V)= 7.0432E-01

GROUP	V	V*N(V)*DV	XA(V)	D(V)	NU*XF(V)	V*X-(V*P)
1	1.0000E-01	4.4339E-05	4.0970E 00	2.0891E-02	1.9306E 00	-4.2686E 00
2	2.0000E-01	3.0530E-04	2.4796E 00	3.9144E-02	4.4732E 00	-1.3752E 00
3	3.0000E-01	9.5911E-04	1.7555E 00	5.4514E-02	3.1747E 00	-5.6412E-01
4	4.0000E-01	2.1495E-03	1.3535E 00	6.7707E-02	2.4497E 00	-1.8738E-01
5	5.0000E-01	3.9678E-03	1.0982E 00	7.9632E-02	1.9879E 00	3.2278E-02
6	6.0000E-01	6.4398E-03	9.2157E-01	9.1108E-02	1.6676E 00	1.7657E-01
7	7.0000E-01	9.5230E-03	7.9179E-01	1.0279E-01	1.4319E 00	2.7759E-01
8	8.0000E-01	1.3114E-02	6.9217E-01	1.1518E-01	1.2507E 00	3.5064E-01
9	9.0000E-01	1.7060E-02	6.1313E-01	1.2867E-01	1.1067E 00	4.0419E-01
10	1.0000E 00	2.1179E-02	5.4871E-01	1.4355E-01	9.8912E-01	4.4333E-01
11	1.1000E 00	2.5300E-02	4.9296E-01	1.6010E-01	8.9118E-01	4.6940E-01
12	1.2000E 00	2.9162E-02	4.4736E-01	1.7790E-01	8.0792E-01	4.9014E-01
13	1.3000E 00	3.2541E-02	4.0813E-01	1.9627E-01	7.3685E-01	5.0918E-01
14	1.4000E 00	3.5118E-02	3.7442E-01	2.1357E-01	6.7559E-01	5.3649E-01
15	1.5000E 00	3.6580E-02	3.4480E-01	2.2791E-01	6.2160E-01	5.8036E-01
16	1.6050E 00	4.0465E-02	3.1769E-01	2.3898E-01	5.7119E-01	6.4427E-01
17	1.7200E 00	4.3094E-02	2.9203E-01	2.4886E-01	5.2353E-01	7.1554E-01
18	1.8450E 00	4.4907E-02	2.6843E-01	2.6146E-01	4.7863E-01	7.6858E-01
19	1.9800E 00	4.6145E-02	2.4676E-01	2.7805E-01	4.3668E-01	7.9423E-01
20	2.1225E 00	4.5551E-02	2.2637E-01	2.9855E-01	3.9774E-01	8.1149E-01
21	2.2775E 00	4.7552E-02	2.0894E-01	3.1459E-01	3.6335E-01	8.5506E-01
22	2.4550E 00	4.9852E-02	1.9348E-01	3.3295E-01	3.3343E-01	9.0765E-01
23	2.6600E 00	5.3122E-02	1.8126E-01	3.5432E-01	3.1191E-01	9.5485E-01
24	2.8975E 00	5.2954E-02	1.7656E-01	3.6241E-01	3.0563E-01	1.0435E 00
25	3.1725E 00	5.9299E-02	1.8880E-01	3.8612E-01	3.2407E-01	1.1815E 00
26	3.4900E 00	6.0968E-02	1.8314E-01	3.8995E-01	3.2225E-01	1.2974E 00
27	3.8550E 00	6.8158E-02	1.2744E-01	4.0487E-01	2.2798E-01	1.2227E 00
28	4.2725E 00	7.4955E-02	9.2747E-02	4.0859E-01	1.6072E-01	1.1950E 00
29	4.7475E 00	7.9538E-02	7.1939E-02	3.9726E-01	1.2113E-01	1.2263E 00

LATTICE INPUT DATA BLOCK LIMP CASE 11006
MESH POINTS= 2 MIXTURES= 2 REGIONS= 2 ISOTOPES= 4 FOILS= 2
GEOMETRY= 3 P1(0) OR B1(1)=-0 BOUND.COND.= 0 MAT. BUCKLING=10.0000E-04

REGION DATA CASE 11006

REGION	THICKNESS	PTS	MIX	H-L	BREAK	TEMP	VOLUME	
1	0.07937	1	1	-0	1	20.00	1.9793E-02	0.0115E-39
3757	0.00000	2	1	268416		0.02		

ISOTOPE CONCENTRATION DATA CASE 11006

ISOTOPE	T/M	MIXTURE 1	MIXTURE 2	MIXTURE 3	MIXTURE 4	MIXTURE 5
92235.	0.	1.4537E-03	0.	0.	0.	0.
922380.	0.	4.6410E-02	0.	0.	0.	0.
10010.	10290.	0.	6.6944E-02	0.	0.	0.
80000.	290.	0.	3.3472E-02	0.	0.	0.

MESH POINT DATA CASE 11006

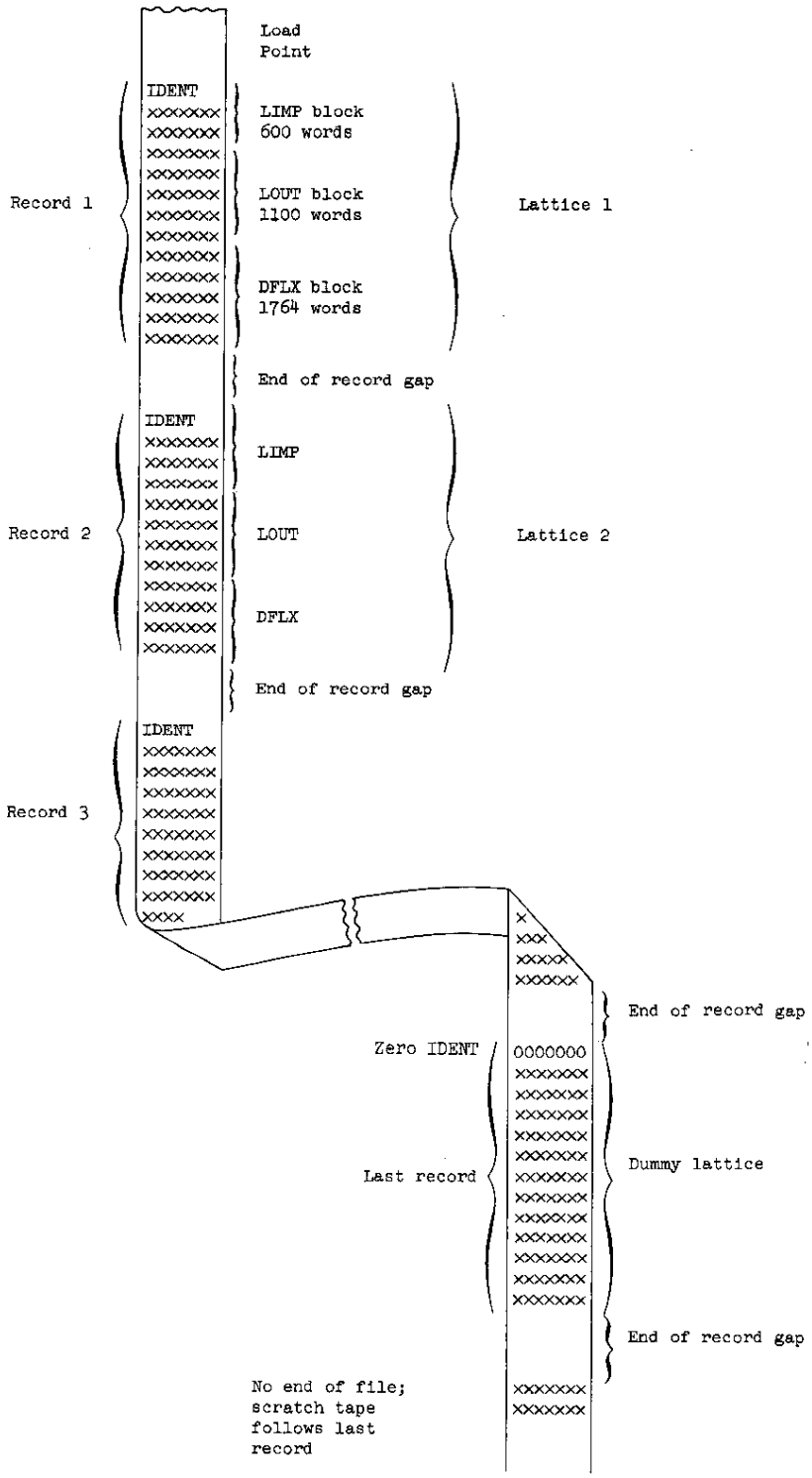
POINT	REGION	MIXTURE	R(INNER)	R(AVE)	R(OUTER)	VOLUME
1	1	1	0.	0.	7.9375E-02	1.9793E-02
2	2	2	7.9375E-02	9.5814E-02	1.1225E-01	1.9793E-02

LATTICE LIBRARY TAPE

Lattice Library Tape Format

The Lattice Library Tape (LLT) contains sufficient information about each lattice to enable the individual user to edit quantities of interest to him, to punch input cards for other codes, etc. It is expected that individuals will utilize these tapes as their primary source of data for secondary computations - not the paper output immediately following a HAMMER run. The information contained in this section is primarily reference material which will be required in the writing of special edit programs. The author may be consulted about detailed questions arising during the writing of such programs.

The following pages contain a listing of the detailed structure of blocks for the Lattice Input (LIMP), the Lattice Output (LOUT), and the Detailed Flux (DFLX). The LIMP block contains the detailed physical description of the lattice; the LOUT block few-group averaged quantities; and the DFLX block the data required to produce additional cross section edits (in conjunction with the THERMOS and HAMLET library tapes.



LATTICE LIBRARY TAPE FORMAT

Control Block - CNTRL

Relative Location	Equivalence to CNTRL(35)	Symbol	Description
1	35	IDBCH	Batch identification number
2	34	NCASE	Number of lattices to be run
3	33	NFLOG	Number of FLOG cases to be run
4	32	BTTL(9)	Batch title, 54 Hollerith characters
13	23	NTIN	System input tape number
14	22	NTOUT	System print output tape number
15	21	NTPUN	System punch output tape number
16	20	NCHAIN	HAMMER program tape number
17	19	LIB1	THERMOS library tape number
18	18	LIB2	HAMLET library tape number
19	17	LIB3	Lattice library tape number
20	16	INT1	Intermediate tape 1 number
21	15	INT2	Intermediate tape 2 number
22	14	NSC1	Scratch tape number
23	13	LNK(10)	Linkage director. LNK(N) gives exit link number for the N^{th} program Program 1 is CAPN Program 2 is THERMOS Program 3 is HAMLET Program 4 is FLOG Program 5 is DIED Program 6 Program 7 Program 8 Program 9 Program 10
33	3		
34	2		
35	1		

CNTRL Block (Continued)

Answer all questions with 0 (no) or 1 (yes)

Relative Location	Equivalence to CNTRL(200)	Symbol	Description
129	72	NCPRN	= 0, end job; = 1, run CAPN; ≥1, end job and unload tapes
130	71	NCPPT	CAPN long print?
131	70		
132	69		
133	68		
134	67		
135	66		
136	65		
137	64		
138	63		
139	62		
140	61		
141	60	NTHRN	= 0, do not run THERMOS = 1, run standard THERMOS = 2, run cosine current THERMOS
142	59	NTHPT	THERMOS long print?
143	58	NTHPN	THERMOS punch?
144	57		
145	56		
146	55		
147	54		
148	53		
149	52		
150	51		
151	50		
152	49		

CNTRL Block (Continued)

Relative Location	Equivalence to CNTRL(200)	Symbol	Description
153	48	NEPRN	= 0, do not run HAMLET = 1, run HAMLET, Version 1 = 2, run HAMLET, Version 2 = 3, run HAMLET, Version 3
154	47	NEPPT	HAMLET long print?
155	46	NEPPN	HAMLET punch?
156	45	NEPSG	= 0, use Selengut-Goertzel approximation
157	44	NEPAG	Self-consistent age approx.?
158	43		
159	42		
160	41		
161	40		
162	39		
163	38		
164	37		
165	36	NFGRN	= 0, do not run FLOG = 1, run FLOG, Version 1 = 2, run FLOG, Version 2 = 3, run FLOG, Version 3
166	35	NFGPT	FLOG long print?
167	34	NFGPN	FLOG punch?
168	33	NFGPB	Point buckling calculation?
169	32	NFGJO	J_o fit to thermal flux?
170	31	NFGB1	Boundary condition at origin
171	30	NFGB2	Boundary condition at outer boundary
172	29		
173	28		
174	27		
175	26		
176	25		

CNTRL Block (Continued)

Relative Location	Equivalence to CNTRL(200)	Symbol	Description
177	24	NEDRN	= 0, do not run DIED = 1, run DIED, Version 1 = 2, run DIED, Version 2 = 3, run DIED, Version 3
178	23	NEDPT	DIED long print
179	22	NEDPN	DIED punch?
180	21	NEDNB	Detailed neutron balance test = 0, ignore = 1, smooth and resonance breakdown by group = 2, total only by group
181	20	NEDFW	Averaging flux used = 0, ignore = 1, asymptotic = 2, asymptotic + interfaces = 3, asymptotic + interfaces + region midpoints
182	19	NEDRB	Regionwise neutron balance = 0, ignore = 1, all regions = 2, all fission regions = 3, read table from cards
183	18	NEDAX	Compute average cross sections = 0, ignore = 1, all regions = 2, all fission regions = 3, read table from cards
184	17	NEDNU	= 0, Balance absorption vs. fissions = 1, Balance absorption vs. productic = 2, = 2, = 3,
185	16		= 0, = 1, = 2, = 3,

Lattice Input Data Block - LIMP

Relative Location	Equivalence to LIMP(600)	Symbol	Description
1	600	IDENT	Case identification number
2	599	NX	Total number of mesh point $\leqslant 20$
3	598	MX	Total number of mixtures $\leqslant 10$
4	597	NRX	Total number of regions $\leqslant 20$
5	596	ISOX	Total number of isotopes $\leqslant 18$
6	595	ISOXE	Total number of foil isotopes, ISOX + ISOXE $\leqslant 18$
7	594	NGEOM	1 = slab, 2 = hexagonal array, 3 = square array
8	593	NP1B1	0 = P_1 approx, 1 = B_1 approx
9	592		
10	591		
11	590	BSQD	Approximate material buckling (cm^{-2})
12	589	HOL(9)	Case title, 54 Hollerith characters
21	580	NPT(20)	Number of mesh points in each region
41	560	THT(20)	Thickness (cm) of each region
61	540	MXAS(20)	Mixture assigned to each region
81	520	ILHI(20)	0 = treat mixture as a heavy scatterer 1 = treat mixture as a light scatterer
101	500	NRBP(20)	Number of the last mesh point in each region
121	480	DEGC(20)	Temperature (^{o}C) of each region
141	460	ISTBA(18)	Isotope identification numbers
159	442	ISTBB(18)	Temperature index for isotope
177	424	CONCTA(18,10)	CONCTA(I,J) is the concentration (atoms/barn-cm) of the I isotope in mixture J
357	244	LIMT(10)	Location in array P of the kernel associated with each mixture
367	234	RIN(20)	Inner radius (cm) of each subregion (mesh point)
387	214	RAD(20)	Central radius (cm) of each subregion (mesh point)
407	194	ROUT(20)	Outer radius (cm) of each subregion (mesh point)
427	174	VOL(20)	Volume (cc) of each subregion (mesh point)
447	154	MTBL(20)	Mixture associated with each mesh point
467	134	REGV(20)	Volume (cc) of each region
487	114	FISD(20)	Fission neutrons/cc in each region

Lattice Output Data Block - LOUT

Relative Location	Equivalence to LOUT(1100)	Symbol	Description
1	1100	ID	Lattice identification number
2	1099	HOLA(9)	Date (A6,A2), time, THERMOS library label (3A6), HAMLET library label (3A6)
11	1090	BALT(18,4,5)	BALT(I,J,K) is the K^{th} cross section (cm^{-1}) of isotope I in group J. K=1 denotes Σ_a (total) 2 denotes Σ_f (total) 3 denotes $\nu\Sigma_f$ (total) 4 denotes Σ_a (resonance) 5 denotes Σ_f (resonance)
371	730	FOIL(20,5,4)	FOIL(I,J,K) is the activity of foil J in lattice region I in group K
771	330	FOGX(4,6)	FOGX(I,J) is the total cross section of type J in group I J=1 denotes D 2 denotes Σ_a 3 denotes $\nu\Sigma_f$ 4 denotes Σ_R 5 denotes Σ_f 6 denotes χ
795	306	FLX(20,4)	FLX(I,J) is the average flux in group J at lattice mesh point I
875	226	DUMB(20)	Miscellaneous parameters DUMB(1) = η 2 = f 3 = fraction absorbed in moderator 4 = p 5 = $\nu(1-p_f)$ 6 = τ 7 = B_m^2 8 = k_∞ 9 = eigenvalue at B^2 10 = ψ_1 at B_m^2 11 = ψ_2 at B_m^2 12 = ψ_3 at B_m^2 13 = ψ_4 at B_m^2
895	206	ØTGC(5)	Overlapping thermal group arrays
900	201	ØTGA(20,6)	

Detailed Flux Block - DFLX

Relative Location	Equivalence to DFLX(1764)	Symbol	Description
1	1764	FT(20, 30)	THERMOS zero buckling flux (vn(r,v)dv*V) by space point and energy group
601	1164	FTB(30)	THERMOS homogenized-cell spectrum at input buckling
631	1134	FH(20, 54)	HAMLET zero buckling flux (Φ(r,u)) by space point and lethargy group
1711	54	FHB(54)	HAMLET homogenized-cell spectrum at input buckling

Library for Thermal Energies

The code LITHE (Library for Thermal Energies) is used to prepare a cross section library tape for THERMOS calculations in the HAMMER system. The program accepts absorption and fission cross sections as constant, 1/v, or tabulated. Scattering cross sections and kernels may be either input on binary decks or computed from the Brown-St. John form of the free gas kernel. In this model, the cross section is described by

$$\sigma_s(v_r) = \sum_{n=1}^{NTERM} A_n e^{-\kappa_n v_r^2}$$

where v_r is the relative velocity, and A_n, κ_n, and M_n (the mass) are input parameters.

The units used are:

cross sections - barns

velocity - 2200 m/sec

temperature - ^oK

Structure of the Binary Tape LIB1

The first record on tape LIB1 contains

IX, HØL, V, DV

where IX is the number of velocity groups,

HØL is an 18-character (3 words) tape identification,

V are the velocity points (30 words), and

DV are the velocity increments (30 words).

The second record contains

WA, WB, HØLB, NKERN, VNU, XA, XS, XF, S

where WA is the isotope identification number,

HØLB is an 18-character (3 words) isotope identification,

NKERN is the number of Legendre components of the scattering kernel included ($\leqslant 4$),

VNU is the value of ν (neutrons/fission),

XA are the absorption cross sections at each velocity (30 words),

XS are the scattering cross sections at each velocity (30 words),

XF are the fission cross sections at each velocity (30 words), and

S are the slowing down sources into each group (30 words).

The next few records contain the Legendre components of the scattering kernel in the form

$$P_{nij} = 4\pi(0.0253)v_i v_j \Delta v_j \int_{-1}^{1} d\mu p_n(\mu) \frac{d\sigma(E_i \leftarrow E_j, \mu)}{d\Omega d\epsilon}$$

where $p_n(\mu)$ is the Legendre polynomial.

If NKERN = 0, no kernels are used and no records written.

If NKERN = 1, the isotropic (P_0) component is used and one record written.

If NKERN = 2, the isotropic (P_0) components and P_1 component are used and two records written, etc.

The pattern is repeated from the second record for each isotope. The last record is similar to the second record but WA $\leqslant$ 0 and signals the end of the tape.

Data Set Assignments

Symbol	Data Set	Description	Disposition	Read Only ?
NTIN	5	Input (BCD)	-	
NTOUT	6	Output (BCD)	Print	
LIB1	9	Input (BIN) (old library)	Save	Yes
LIB2	10	Output (BIN) (new library)	Save	No
NTEMP	2	Scratch	Scratch	No

LITHE Input Data

Item	Column	Format	Symbol	Description
1	1-18	3A6	HØL	18-character description of tape
2	1-5	I5	ITAG	= 0 make up a new tape ≠ 0 update existing tape
	6-10	I5	IX	Number of velocity groups (<30)
	11-15	I5	NISD	Number of isotopes to be deleted from existing tape (<100)
3	-	7E10.5	V(I),I=1,IX	The velocity points in increasing order. Omit if ITAG ≠ 0
4	-	7E10.5	DV(I),I=1,IX	Velocity increments. Omit if ITAG ≠ 0
5	1-10 11-20	E10.0 E10.0	(WAP(L), WBP(L), L=1,NISD)	Identification number and Temperature/Model index for isotopes to be deleted. One card/isotope. Omit if NISD = 0
6	1-18	3A6	HØLB	18-character description of isotope
	21-30	F10.0	WA	Isotope identification number
	31-40	F10.0	WB	Temperature/Model index
	41-45	I5	LXA	= 0 $\sigma_a(v)$ = VXA (see item 7) = 1 $\sigma_a(v)$ = VXA/v (see item 7) = 2 $\sigma_a(v)$ tabulated
	46-50	I5	LXS	= 0 $\sigma_s(v)$ = VXS (see item 7) = 1 $\sigma_s(v)$ computed from gas model = 2 $\sigma_s(v)$ computed from input kernel
	51-55	I5	LXF	= 0 $\sigma_f(v)$ = VXF (see item 7) = 1 $\sigma_f(v)$ = VXF/v (see item 7) = 2 $\sigma_f(v)$ tabulated
	56-60	I5	NKERN	Number of kernels to be put on tape
	61-65	I5	NTERM	Number of terms in BSJ formula
7	1-10	E10.5	VXA	Value of σ_a
	11-20	E10.5	VXS	Value of σ_s
	21-30	E10.5	VXF	Value of σ_f
	31-40	E10.5	VNU	Value of ν
	41-50	E10.5	AM	Mass used in source calculation (amu)
	51-60	E10.5	HXS	Cross section used in source calculation (barns)
	61-70	E10.5	TP	Temperature used in source calculation (°K)
8	-	7E10.5	(XA(I),I=1,IX)	Tabulated values of $\sigma_a(v_i)$. Omit if LXA ≠ 2
9	-	7E10.5	(XF(I),I=1,IX)	Tabulated values of $\sigma_f(v_i)$. Omit if LXF ≠ 2
10	-	7E10.5	T,	Temperature (°K) used in the kernel calculation
			(ZAMT(N), ZAT(N), ZAKT(N), N=1,NTERM)	M_n used in the BSJ formula (amu) A_n used in the BSJ formula (barns) κ_n used in the BSJ formula Omit if LXS ≠ 1
11	-	-	$n, I_o, J_o, (P_n(I_o,J),$ $J=J_o, J_o+5$	The NKERN kernel decks in order P_o, P_1, ... Omit if LXS ≠ 2
12	-	-	-	Repeat items 6-11 for each isotope to be added to the library. Follow last isotope with a blank card to signal end of input data.

Isotope and Material Identification System

An isotope or material is identified by a seven-digit number,

euzzaaa

For a single isotope,

aaa is the mass number (e.g., 235 in $_{92}U^{235}$), and

zz is the atomic number (e.g., 92 in $_{92}U^{235}$).

For a natural mixture of isotopes,

aaa is 000, and

zz is the atomic number (e.g., 5 in $_{5}B$).

For any other mixture,

aaa is a mixture code number, and

zz is 00.

The first digit, e (for evaluation), is generally zero, but may assume any value less than 10 to distinguish different evaluations of the same material on a library tape.

The second digit, u (for usage), describes the cross section type entered in the fields normally used for absorption cross sections. This feature permits foil materials to be treated as ordinary materials. The assignments for u are

u	Cross section in σ_a field
0	σ_a
1	σ_f
2	
3	
4	
5	
6	
7	
8	
9	

A second number, called the Temperature/Model index (T/M), is required to further specify materials for thermalization calculations. The T/M index is a four-digit number,

The group ttt is the absolute temperature divided by ten and rounded.

For example, if T = 293°K, ttt = 029. If T = 478°K, ttt = 048.

The digit m (for model) denotes the model used to compute the scattering kernel.

m = 0 Free gas
1 Bound model (Nelkin, Parks)
2 Pseudo-free-gas (Brown-St. John)
3
4
5 Bound model (Ardente benzene)
6
7
8
9

HAMMER Epithermal Library Program

The code HELP (HAMMER Epithermal Library Program) is used to prepare, update, and copy a cross section library tape for HAMLET calculations in the HAMMER system. All cross sections, scattering matrices, and resonance parameters to be used are input to the program in a single fixed BCD format. The collection of these data for a single isotope (or element), along with a single Hollerith header card constitute an "isotopic data block." Users will normally maintain a card file of these BCD isotopic data blocks as well as the resultant binary library tape file.

The units used throughout are:

energy - ev
cross section - barns
scattering matrices - barns/unit lethargy
mass - amu

Structure of an Isotopic Data Block

Card Type 0: Header Card (F7.0, 1 X E8.5, 2 X 9 A6)

Item	Symbol	Description
1	WDI	Isotope ID (see LITHE description)
2	AMAS	Atomic mass
3-10	HØL(9)	Chemical symbol and documentation

Card Type 1: Smooth Data (I1, 2I3, 1 X 8 E8.6)

(one card per significant lethargy group)

Item	Symbol	Description
1	I	Card type, always 1
2	J	Group number (sequential within block)
3	K	Ignored
4	SMX(1,J)	P_0 component of smooth scattering, σ_{so}
5	SMX(2,J)	Smooth capture cross section, σ_c
6	SMX(3,J)	Inelastic cross section, σ_{in}
7	SMX(4,J)	Smooth fission cross section, σ_f
8	SMX(5,J)	Age number, λ
9	SMX(6,J)	P_1 component of smooth scattering, σ_{s1}
10	SMX(7,J)	Number of neutrons per fission, ν
11	SMX(8,J)	Microscopic slowing down power, $\xi\sigma_{so}$

These data are identical to those input to the MUFT library. An invaluable reference in preparing such data is WCAP-3709, Compilation of a MUFT Library.[7]

OSR 24-F26

Programmer: SUICH **Page** ____ **of** ____

Program: HELP DATA FORMAT - ISOTOPIC DATA BLOCKS

ID (F7.0) AT. WT. (E8.5) HOLLERITH (FIRST WORD IS CHEM SYMBOL + A)

1	2–4	5–7	9–16	17–24	25–32	33–40	41–48	49–56	57–64	65–72	73–80
I	J	K	DATA (N), N = 1,8 (8E8.6)								
I	GRP		σ_{So}	σ_c	σ_{in}	σ_f	λ	σ_{Si}	ν	$\xi\sigma_{So}$	ID + SEQ
2	FIN. GRP	INIT. GRP.	$\sigma_{in}(K,J)$	$\sigma_{in}(K+1,J)$	etc.						"
3	GRP	SEQ No.	E_o	Γ_n	Γ_R	Γ_f	g				"
			or R	n	α						"
4	GRP	SEQ No.	D	$\langle\Gamma_n^o\rangle$	Γ_R	Γ_f	g				"
9	0	0	END OF ISOTOPE								"

Card Type 2: Inelastic Scattering Matrix (I1, 2I3, 1 X 8 E8.6)

(use as required)

Item	Symbol	Description
1	I	Card type, always 2
2	J	Final group of these data
3	K	Initial group of first datum
4-11	XIN(L,J) L = K,K+7	Matrix elements from higher groups into group J, given sequentially beginning at group K. Ignore all fields after the diagonal element (J = L).

See WCAP-3709 for details.[7]

Card Type 3: Resonance Parameters (I1, 2I3, 1 X 8 E8.6)

(use as required)

Item	Symbol	Description
1	I	Card type, always 3
2	J	Group in which this resonance occurs
3	K	Sequential number of this resonance within this group

A) Data to be treated by the Nordheim method:

4	ER	Resonance energy, E_o
5	GMN	Neutron width, Γ_n
6	GMGM	Radiation width, Γ_γ
7	GMF	Fission width, Γ_f
8	SPIN	Spin factor, g

B) Data to be treated by the Equivalence Principle[7]:

4	R	Shielding parameter, $r = \sigma_o \Gamma_a/\Gamma$
5	M	Infinite integral, $m = \sigma_o \Gamma_a/E_o$
6	ALPHA	Fission to absorption ratio, $\alpha = \Gamma_f/\Gamma_a$

Card Type 4: Unresolved Resonance Data (I1, 2I3, 1 X 8 E8.6)

(use as required)

Item	Symbol	Description
1	I	Card type, always 4
2	J	Initial group in which resonances occur
3	K	Final group in which resonances occur
4	ER	Average level spacing, D
5	GMN	Average reduced neutron width, $\langle\Gamma_n^o\rangle$
6	GMGM	Average radiation width, Γ_γ
7	GMF	Average fission width, Γ_f
8	SPIN	Spin factor, g

Card Type 9: End of Block Flag (I1)

(separates data blocks)

1	I	Card type, always 9

STRUCTURE OF THE BINARY LIBRARY TAPE

Record 1: NG, NISØ, DU, EN, HØLB

where NG is the number of lethargy groups,

NISØ is the number of isotopes,

DU are the lethargy increments (NG words),

EN are the energy breakpoints (NG + 1 words), and

HØLB is the library tape label (12 words).

Record 2: S, the source values (NG words)

Records 3 to (NISØ + 2): IFLT, WD, NIN, NR, TB

where IFLT is the number of cross section data in group 1, this isotope,

WD is the isotopic identification number,

NIN is the number of inelastic scattering data in group 1, this isotope,

NR is the number of sets of resonance data in group 1, this isotope and

TB is the data for this isotope, group 1:

mass, smooth cross sections (8 words),
inelastic cross sections (NIN words),
resonance parameters (5*NR words), and
Hollerith description (9 words).

Record (NISØ + 3): As above, with a negative/zero WD, to flag end of group 1.

Records (NISØ + 4) to (2NISØ + 4): As above, with data for group 2, and the Hollerith data (last 9 words) omitted

The pattern is repeated for each group. An end of file is placed after the last (negative/zero WD) isotope of the last group.

HELP Input Data

Card	Format	Symbol	Description
1	12A6	HØLB	Library tape label
2	I5	NG	Number of groups. (If $\leq$ 0, an update or copy is assumed, skip to card 5)
3	7E10.5	DU	Lethargy increments (NG words) highest energy is assumed to be 10 Mev
4	7E10.5	S	Source values (NG words) Skip to card 6
5	(I10	NDEL	Number of isotopes to be deleted (= 0 for tape copy)
	6E10.0/(7E10.0))	WDD	Identification numbers of isotopes to be deleted (ignore for tape copy)
6	Isotopic data blocks to be added to the library (ignore for tape copy)		
7	Blank card (zero WD) to end run		

Error Detection

The HELP program checks input data cards for several common types of errors. The following errors cause the program execution to be terminated with a message indicating the error type:

Type	Error
1	Cross section type (data word 1) not 1, 2, 3, 4, or 9, or not in sequential order
2	Group index out of range
3	-
4	Number of resonances exceeds 300/isotope
5	Resolved and unresolved resonance groups overlap
6	Isotopic mass $\leqslant 0$

The following errors produce a message in the BCD (printed) output, but do not terminate execution:

Type	Error
1	Negative data
2	Null resonance parameters
3	Resonance energy out of range of group to which it is assigned

Data Set Assignments

Symbol	Data Set	Description	Disposition	Read Only ?
NTIN	5	Input (BCD)	-	-
NTOUT	6	Output (BCD)	Print	-
LIB	9	Output (BIN)	Save	No
LIBO	4	Input (BIN) (new library)	Save	Yes
NSC1,NSC2	3,8	Scratch (old library)	Scratch	No

Library Conversion

The code LIBCON (Library Conversion) is used to convert cross section data cards in the MUFT/FORM excess 50 notation to the HELP input format. The program processes data (from the input to the output/punch tapes) in blocks belonging to a particular isotope. This order is different from that of a MUFT library. Block after block is processed until a blank header card is encountered. An isotopic data block consists of any or all of the following MUFT/FORM data decks, in the following order, preceded by a header card:

(i) Header Card: Format (A5, 6I5, 5X, F7.2 4A6, 1X)

HOLA, ID, IDO, ISM, IIN, IRM, IAL, IRES, AMAS,

(HOLB(J), J = 1, 4)

where

HOLA = chemical symbol, e.g., ^{239}Pu

ID = ZA; e.g., 94239

IDO = old (MUFT) ID; e.g., 21

ISM = 1 if item (ii) follows

IIN = 1 if item (iii) follows

IRM = 1 if item (iv) follows

IAL = 1 if item (v) follows

AMASS = mass (in amu) this isotope, e.g., 239

HOLB = documentation, e.g., MUFT LIB 567-WCAP 3709[7]

(ii) Microscopic smooth cross section deck (54 cards)

(iii) Inelastic matrix deck (48 cards); must not be given unless (ii) above is also given; deck will be renormalized to σ_{in} given in (ii)

(iv) r and m factor decks (60 cards)

(v) Alpha factor decks (30 cards); must not be given unless (iv), above, is also given

ACKNOWLEDGMENT

The development of this program was a joint BNL-SRL project. The interest and encouragement of J. Chernick (BNL) and J. L. Crandall (SRL) are gratefully acknowledged. They were instrumental in bringing the authors together for the summer of 1964 at BNL, where the initial version of the program was designed and coded. Members of the Reactor Physics groups at BNL and SRL have been most helpful in the checking of the program. Especial thanks are due R. L. Hellens, H. R. Connell, Bal Raj Sehgal, and L. G. Joiner of BNL and W. E. Graves of SRL.

REFERENCES

1. H. C. Honeck and J. L Crandall. The Physics of Heavy Water Lattices, USAEC Report BNL-8253, Brookhaven National Laboratory and E. I. du Pont de Nemours and Co., Savannah River Laboratory (1964).

2. H. C. Honeck. THERMOS, A Thermalization Transport Theory Code for Reactor Lattice Calculations. USAEC Report BNL-5826, Brookhaven National Lab., Upton, N. Y. (1961).

3. H. Bohl, Jr., E. M. Gelbard, and G. H. Ryan. MUFT-4 - Fast Neutron Spectrum Code for the IBM-704. USAEC Report WAPD-TM-72, Westinghouse Electric Corp., Bettis Plant, Pittsburgh, Pa. (1957).

4. G. F. Kuncir. A Program for the Calculation of Resonance Integrals. USAEC Report GA-2525, General Dynamics Corp., General Atomic Div., San Diego, Calif. (1961).

5. H. P. Flatt. The FOG One-Dimensional Diffusion Equation Codes. USAEC Report NAA-SR-6104, North American Aviation, Inc., Atomics International, Canoga Park, Calif. (1961).

6. R. L. Hellens and H. C. Honeck. "A Summary and Preliminary Analysis of the BNL Slightly Enriched Uranium, Water Moderator Lattice Measurements," Light Water Lattices. Technical Report Series No. 12, IAEA, Vienna, Austria, pp 27-71 (1962).

7. R. A. Dannels and D. J. Bredin. Compilation of a MUFT Library (Library 567). USAEC Report WCAP-3709, Westinghouse Electric Corporation, Atomic Power Division, Pittsburgh, Pa. (1962).