DP-1283

AEC RESEARCH AND DEVELOPMENT REPORT

THE **HAMBUR** SYSTEM

D. R. FINCH

R. P. CHRISTMAN

DUPONT
REG. U. S. PAT. OFF.

Savannah River Laboratory

Aiken, South Carolina

Printed in the United States of America
Available from
National Technical Information Service
U. S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22151
Price: Printed Copy $3.00; Microfiche $0.95

663301

DP-1283

Mathematics and Computers
(TID-4500, UC-32)

THE **HAMBUR** SYSTEM

by

D. R. Finch
R. P. Christman

Approved by

P. L. Roggenkamp, Research Manager
Theoretical Physics Division

February 1972

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 HAMBUR System is a set of linked computer programs for the calculation of (1) infinite lattice parameters by multigroup transport theory, (2) composite reactor parameters by few-group diffusion theory using finite difference or source-sink methods, and (3) infinite lattice or reactor supercell depletion-production history by straight time march method. Procedures are given for using the system of linked reactor physics codes, and a detailed description of the methods used in generating the depletion-production history is given.

CONTENTS

CONTENTS (continued)

LIST OF TABLES

LIST OF FIGURES

1. INTRODUCTION

Computer programs for predicting the isotope inventory of a reactor core under specified operating conditions have been written for specialized applications. Ideally, it would be desirable to create a single computer program capable of treating any set of coupled isotopes with full capability to account for all time-dependent spectral and spatial effects on fluxes and cross sections, and to predict fission product nuclide poisoning as a function of time. Practically, the time required to perform such a calculation prevents this program from being written, and forces any real program to use more approximate methods suited to the specific application for which the program is designed.

Some approaches that are applicable over a wide range of problems have been taken in writing burnup programs:

- The computer programs CINDER[1] and GARGOYLE[2] are based on few (<10) energy groups and infinite media. They rely on other programs to provide the spatial dependence of the flux. These programs provide extremely fast calculations, and CINDER is well suited to the calculation of a large number of fission product nuclides.

- The computer program ASSAULT[3] predicts the average isotope inventory for a reactor system based on few-group diffusion theory in one or two space dimensions. This program relies on other programs for the spectrum-averaged cross sections and fission product nuclide inventory.

- The computer program LEOPARD[4] accounts for spectral and spatial dependences of cross sections and fluxes, but does so only for simple fuel rods in infinitely repeating arrays and for a restricted set of isotopes.

- A more general computer program LASER[5] accepts any general set of coupled isotopes, but restricts the cell geometry to four regions containing fuel, cladding, moderator, and a scattering ring.

The HAMBUR program was created in response to the need for a program that can predict spatial isotope inventories for complex annular assemblies with spectral effects and fission product poisoning accounted for. The nuclides that it can treat are dependent upon the existence of good differential cross section information; hence the isotopes are limited to those between ^{232}Th and ^{245}Cm.

The basic cell spectrum is calculated by a version of the HAMMER System[6] that has been extended to provide spatial detail in the calculation of resonance capture. HAMBUR may be used as a cell spectrum program in the same manner as the HAMMER System, and provides some advantages for hard spectrum reactors in its improved resonance capture treatment.

Parts of the HAMBUR system except for the BURNUP section are described in the references cited earlier. Only BURNUP will be discussed in detail in this report.

2. SUMMARY

The HAMBUR system will execute in sequence or iteratively the following programs:

2.1 CAPN

Program CAPN reads infinite lattice cell input from cards or Burnup History Tape and performs extensive error checking. A control path is established for the following programs.

2.2 THERMOS

Program THERMOS[7] performs 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.

2.3 HAMLET

Program HAMLET performs the same spatial flux calculation as THERMOS except the energy range is $0.625 \text{ ev} \leqslant E \leqslant 10$ Mev. Collision probabilities are computed under the assumption of isotropic angular distributions at region boundaries, and the energy spectrum during moderation is computed as in MUFT[8] or FORM.[9] For the calculation of resonance capture, extensively modified versions of ZUT and TUZ[10] are used as subroutines. These versions of ZUT and TUZ explicitly treat the spatial effects arising from separate resonance capture regions that may shield or otherwise interact with each other. In addition to the same output as THERMOS, a set of two-, three-, and four-group cross sections (10 Mev → 1.05 Mev → 9.12 kev → 0.625 ev) is produced, from which the asymptotic spectrum and buckling of an infinitely repeating lattice are determined.

2.4 FLOG

Program FLOG utilizes the four-group constants produced in THERMOS and HAMLET to perform a one-dimensional, many-region, diffusion theory reactor calculation. This is a modified version

of FOG.[11] FLOG performs the standard criticality searches of FOG, and automatically calculates a series of core size loadings for exponential assemblies.

2.5 HERESY

Program HERESY utilizes the two- or four-group constants produced in THERMOS and HAMLET to perform a two-dimensional, many-region, source-sink theory reactor calculation. This is a modified version of the HERESY III program.[12] A variety of criticality searches are available.

2.6 DIED

Program DIED is an edit program that combines the cross sections and fluxes obtained in THERMOS, HAMLET, and FLOG or HERESY to produce neutron balance sheets. For an infinitely repeating lattice cell edit (results of THERMOS and HAMLET only), the output is either (a) absorption versus fission by isotope and group, split into "smooth" and "resonance" event contribution, or (b) total absorptions versus total fissions by isotope or group. For a composite reactor edit (results from FLOG or HERESY), neutron balance sheets are produced for total absorptions versus 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 appropriate to the region average, the asymptotic core, the interface, or all of these.

2.7 BURNUP

Program BURNUP is a pointwise depletion and fission product program that calculates the burnup characteristics of infinitely repeating lattice cells, or repeating reactor supercells. Reaction rates, fluxes, and their spatial and energy distributions in an infinite lattice cell are provided by THERMOS and HAMLET. The reaction rates and fluxes are modified to correspond to a repeating reactor supercell by HERESY if required. Sixteen isotopes are considered in the calculation of fuel chain depletion and production. Two alternative fuel chains and a variety of peripheral isotope chains containing possible non-zero initial concentrations are provided. Reaction rates for fuel chain isotopes not computed explicitly in THERMOS and HAMLET are computed using cross sections obtained from fitted polynomial expansions as a function of spectral parameters. Fission product concentrations are calculated by a modified version of CINDER.[8] An edit of basic lattice cell parameters as a function of exposure

is given. Optionally, absolute fluxes, isotopic concentrations, isotopic cross sections, isotopic power, isotopic mass balance, fission product heat and rate of cooling, or all of these may be obtained.

The seven programs are executed in the order as given above. Figure 2.1 shows a diagram of the programs and the possible paths that may be taken through the programs. Jobs are processed in batches with each batch using a different set of the seven programs as specified. A batch contains any number of cases, each case describing an infinite lattice cell to be processed by THERMOS and/or HAMLET, or a set of lattice cells to be processed by FLOG or HERESY, or a set of lattice cells to be depleted as either an infinite lattice cell or a reactor supercell by BURNUP.

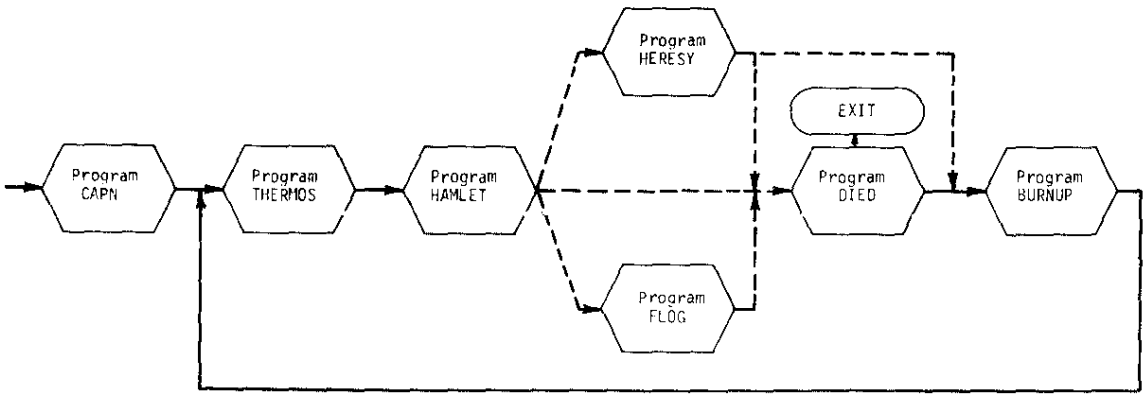


FIGURE 2.1 The HAMBUR System
(Broken lines indicate optional paths)

The results of THERMOS-HAMLET calculations may be saved on a Lattice Library Tape (LLT). Each lattice on LLT is assigned a unique permanent identification number specified by the user at its creation. FLOG and HERESY utilize results stored on the LLT to obtain group constants for the reactor calculations.

The results of a depletion history may be saved on a Burnup History Tape (BHT). The execution of a depletion calculation requires reiterating several times through THERMOS and HAMLET and possibly HERESY. The LLT is rewritten on each pass and is not preserved from pass to pass. The BHT contains the same information as the LLT as a function of exposure for each case. Case input may be read from a BHT in the CAPN program. An option allows fission products to be cooled for an arbitrary period after reactor shutdown; hence reactor staging operations may be simulated using data

stored in the BHT. The BHT is prepared as the last step in a depletion calculation, and is only produced for problems that successfully execute.

Information is passed from program to program on scratch data sets and on the LLT. All cases in a batch are executed by one program before going on to the next program in sequence.

3. BURNUP CALCULATION

3.1 GENERAL

The HAMBUR burnup system is designed to determine the depletion and production of isotopes as a function of time and specified operating conditions for an infinitely repeating lattice cell or in all lattice cells of a reactor supercell. The calculation is performed by iterating successively through the lattice cell calculation of THERMOS and HAMLET, optionally through the edit program DIED and the reactor supercell program HERESY, then through the depletion and production program BURNUP. The iteration is controlled in the BURNUP program. The frequency of the iteration is controlled by a three-level user option that allows the time period between spectrum recalculation to be adjusted to the accuracy required. Termination of the burnup calculation is also controlled by the BURNUP program as well as the optional final edits and writing of the Burnup History Tape (BHT). A final pass is made through the lattice cell calculation and reactor supercell calculation after termination to determine lattice and reactor constants.

During each iteration of the burnup calculation, two-group (10 Mev → 0.625 ev → 0.0) reaction rates by isotope and space point in the cell are generated in the THERMOS and HAMLET programs and passed to the BURNUP program on a scratch data set. These reaction rates are not preserved on a Lattice Library Tape (LLT); hence, each burnup problem must begin by executing the THERMOS and HAMLET programs for each lattice cell in the calculation. Several batches of problems may be executed and the reaction rates accumulated on the scratch data set, however, before entering the BURNUP program. Input for a lattice case may come from cards, a BHT, or a combination of these two. If a reactor supercell is calculated by the HERESY program, the results of this calculation are passed to the BURNUP program on a scratch data set.

During one pass through the BURNUP program, the following calculations and operations are performed for each lattice cell.

- Reaction rates by isotope and space point, fluxes by space point, and isotopic concentrations by space point are assembled.

- Reaction rates and fluxes are renormalized to conform to the reactor supercell calculation if required. Absolute fluxes and reaction rates are then derived from the specified power normalization.

- Cross sections by isotope and space point are derived from reaction rates, fluxes, and concentrations for each "fuel region" and "added isotope region" of the cell.

- A solution of the coupled depletion-production equations for each "fuel region" and/or "added isotope region" is performed to bring each lattice cell up to the next spectrum recalculation time. For reactor supercells the lattice cells are carried forward in parallel.

- For each "fuel region" a linearized chain fission product production calculation is performed using isotopic fission reaction rates from the "fuel chain" calculation just performed to determine fission product yields as a function of time. The total fission product poisoning is then expressed as equivalent xenon, samarium, and "lumped fission product" concentrations.

Each step in the above sequence will be described in detail below.

The terms "fuel region" and "added isotope region" will now be defined to understand the following description. A "fuel region" is any region in the lattice cell that contains any of the sixteen isotopes that are always carried through the calculation, and which normally control the spectrum of the lattice cell. Two different sets of isotopes may be specified on option and are shown in Figures 3.1 and 3.2.

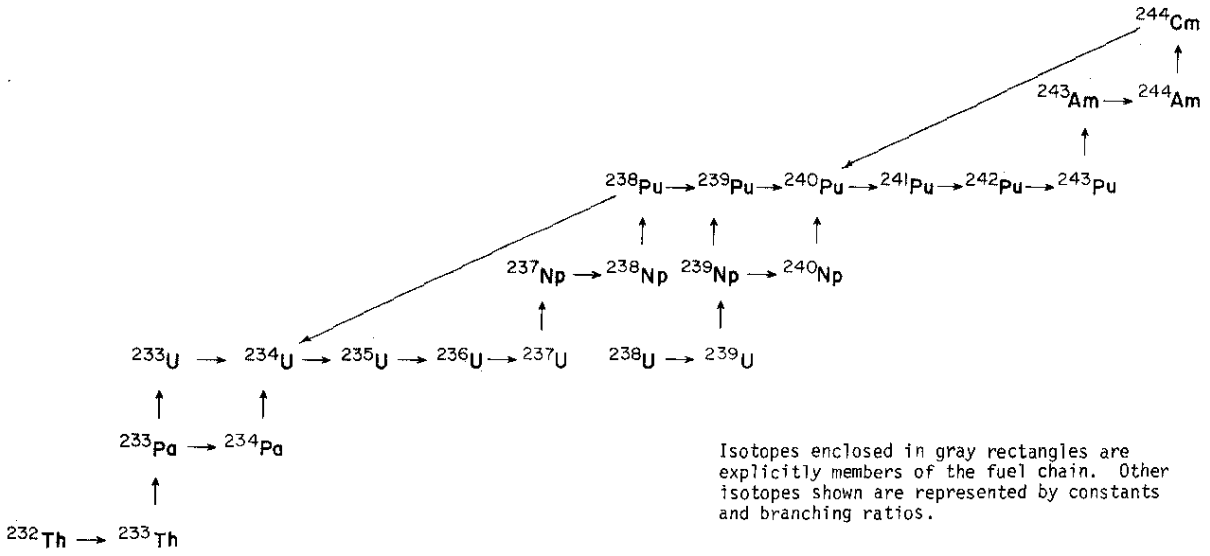


FIGURE 3.1 Fuel Chain 1 (Option KAM = 0)

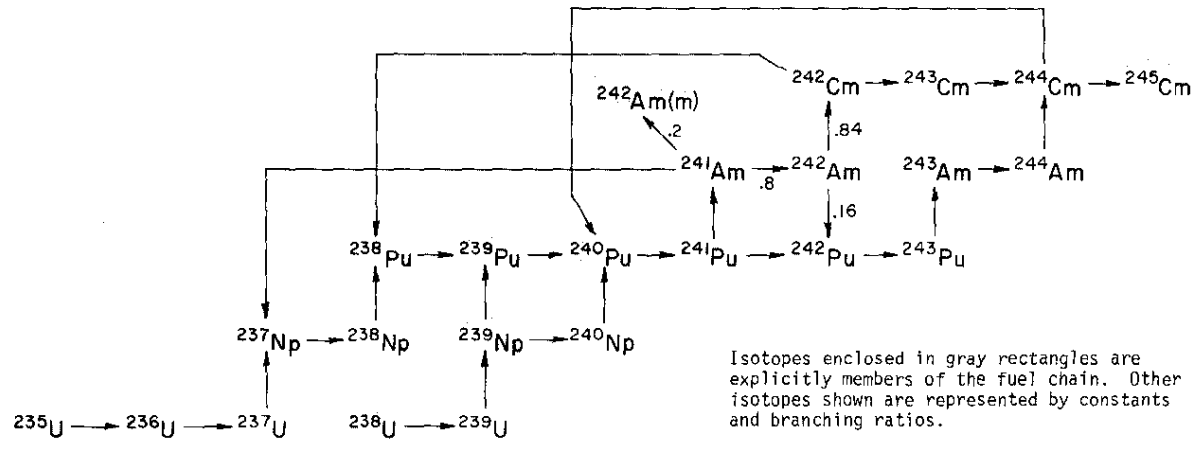
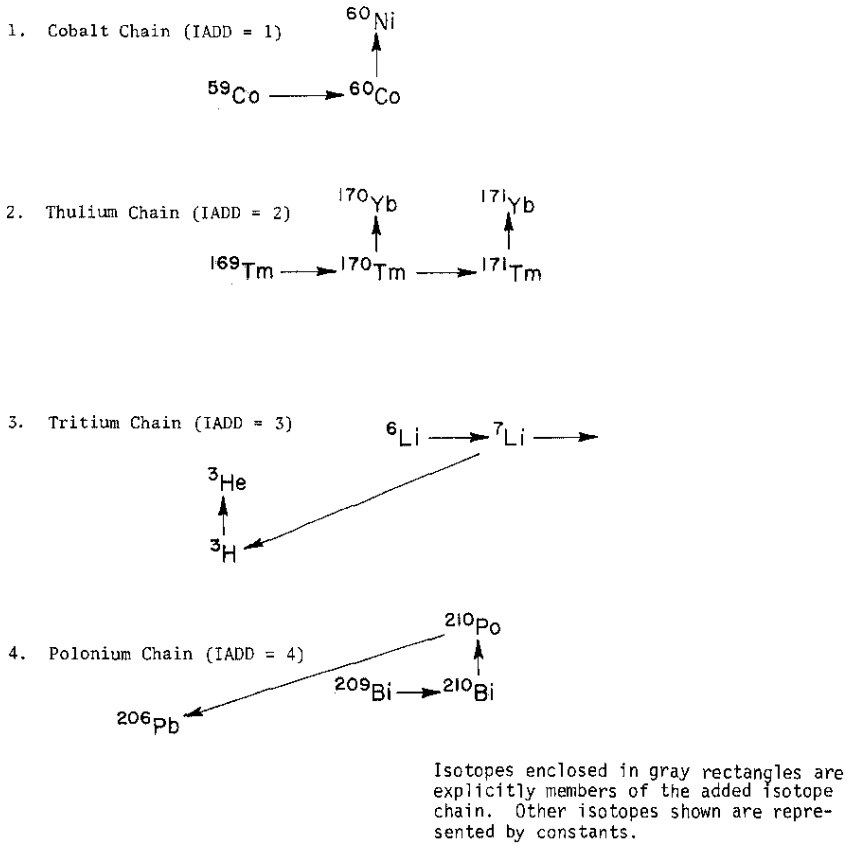


FIGURE 3.2 Fuel Chain 2 (Option KAM = 1)

An "added isotope region" is a region in the lattice cell that contains no "fuel chain isotopes," but does contain an isotope that may change concentration with time. The isotope may or may not have a significant effect on the cell spectrum, and may lead to the production of other isotopes that are of interest. An example of such an "added isotope" is a burnable poison which will have an influence on the cell spectrum if present in large concentrations. Several predefined "added isotope chains" are available and are diagrammed in Figure 3.3.



Isotopes enclosed in gray rectangles are explicitly members of the added isotope chain. Other isotopes shown are represented by constants.

FIGURE 3.3 Built-In "Added Isotope" Chains

3.2 MATERIALS AND ISOTOPE CONCENTRATIONS

The input for the lattice cell calls for several isotopes at specified concentrations to be lumped together and designated by a material number. Each region of the lattice cell contains one such material. During the burnup calculation, the isotopes will change concentration and the energy spectrum will change as a function of spatial position in the lattice cell. It is necessary, therefore, to initially specify each "fuel region" and each "added isotope region" as having a different material number to allow the spatial variations to be accounted for properly. There may be up to five "fuel regions" and three "added isotope regions" in a lattice cell.

A second consideration in making up the materials for the lattice cell is that some isotopes may not have any effect on the cell spectrum at the start of the problem but will build-in to a significant concentration as the burnup proceeds. Because of this, it is required that all materials contain all isotopes at the start of the burnup necessary to account for spectral effects throughout the calculation.

During the first pass through the BURNUP program, all "fuel chain" and "added isotope chain" isotope concentrations are set equal to a concentration of 1.0×10^{-15} atm/bn-cm. Next all isotope concentrations from the lattice cell for both "fuel chain" and "added isotope chain" isotopes are taken from the LIMP block of the LLT (Section 6.1). Finally any non-zero concentrations of "fuel chain" or "added isotope chain" isotopes may be entered in the BURNUP program input. This last feature allows isotopes to be carried in the burnup calculation but not the lattice cell calculation, or to start with non-zero concentrations of various isotopes in the "added isotope chains."

Between successive passes through the BURNUP program, these isotope concentrations are stored on a scratch data set. At the times at which spectrum recalculation is required, the LIMP block on the LLT is updated and used as input to the lattice cell calculation.

3.3 FLUXES AND REACTION RATES

During each lattice cell calculation, two-group fission and absorption (capture + fission) reaction rates per unit length by isotope and space point are written to a scratch data set by the THERMOS and HAMLET programs. The reaction rates are read by the BURNUP program for each lattice cell in the burnup calculation. If a reactor supercell calculation has been performed with HERESY, the two-group reaction rates must be renormalized to correspond

to the calculated supercell spectrum. The following assumptions are made for this renormalization.

- The spatial distribution of fluxes (and therefore reaction rates) is assumed constant; namely, that spatial distribution calculated in the lattice cell calculation.
- Cross section variation with energy within each of the two energy groups may be ignored.

Both of the above assumptions introduce small errors into the reaction rates; however, they are generally negligible.

The supercell calculation by the HERESY program produces the fast-to-thermal absorption ratios for each cell in the supercell. The THERMOS-HAMLET two-group reaction rates by isotope and space point for the cells are scaled by a constant C_k defined by

$$\frac{\sum_{i=1}^{NI}\sum_{j=1}^{NF}\Sigma^F_{ijk}\phi^F_{jk}V_{jk}}{\sum_{i=1}^{NI}\sum_{j=1}^{NF}\Sigma^T_{ijk}\phi^T_{jk}V_{jk}}\,C_k = \frac{I^F_k}{I^T_k} \qquad (3.1)$$

where

NI = number of isotopes in cell

NF = last point in cell within fuel assembly

Σ_{ijk} = macroscopic absorption cross section for i^{th} isotope at j^{th} space point in cell k

ϕ_{jk} = flux at j^{th} space point in k^{th} cell

V_{jk} = volume associated with j^{th} space point in k^{th} lattice cell

I_k = absorption in k^{th} assembly from the HERESY supercell calculation, normalized to the most absorbing assembly in the supercell absorbing one neutron

and superscripts F and T refer to fast and thermal energy groups, respectively.

The space-dependent fluxes are obtained from the LOUT block of the LLT by collapsing the four-group fluxes to two groups. The same renormalization factors C_k are used for fluxes as for reaction rates when a reactor supercell calculation is performed.

Reaction rates for isotopes that are members of the "fuel chain" or "added isotope chain" but do not explicitly appear in the lattice cell are computed from the fluxes as calculated above, regionwise atom densities specified in the input, and cross sections calculated as described in the next section.

Fluxes and reaction rates per unit length to this point have been to an arbitrary normalization. For the depletion calculation, the absolute fluxes must be derived from a known operating condition on the lattice cell or reactor supercell. The BURNUP program uses the total assembly power in the case of an infinitely repeating lattice cell or the total assembly power of the highest power assembly, total supercell power, or a specified assembly power in a reactor supercell to obtain the absolute flux normalization. For any assembly, it is assumed that the total power consists of four components: (1) fission energy release per second, (2) capture gamma ray energy per second arising from radiative capture in the assembly, (3) energy release per second from kinetic heating of moderator by neutrons (sometimes referred to as "inelastic scattering"), and (4) energy release per second from decay of fission products. These components are calculated by equations (3.2-3.6).

3.3.1 Fission Power

This component is calculated from the unnormalized fission reaction rates per unit length and constant values of energy release per fission for each "fuel chain" isotope separately. The values of the energy release per fission are given in Table A-I in Appendix A. The following equation expresses this component:

$$F = \sum_{i=1}^{NI} \sum_{j=1}^{NX} \left(R_{ij}^{F} + R_{ij}^{T}\right)\chi_i H \Big/ 6.25 \times 10^{18} \qquad (3.2)$$

where

F = total fission power in assembly, MW

NI = number of isotopes in cell

NX = number of space points in cell

$R_{ij}^{F,T}$ = fast and thermal fission reaction rates per cm for isotope i at space point j

X_i = energy release per fission for isotope i, Mev

H = length of assembly, cm

3.3.2 Power Due to Radiative Capture Gamma Rays

This component is calculated assuming all gamma rays produced from radiative capture are absorbed in the assembly. Values of energy release per radiative capture are given in Table A-I in Appendix A. The following equation expresses this component:

$$RC = \sum_{i=1}^{NI} \sum_{j=1}^{NX} \left(C_{ij}^{F} + C_{ij}^{T} \right) \xi_i H \Big/ 6.25 \times 10^{18} \qquad (3.3)$$

where

RC = radiative capture power, MW

NI = number of isotopes in lattice cell

NX = number of space points in cell

$C_{ij}^{F,T}$ = fast and thermal neutron capture (absorption-fission) reaction rates per cm of length for isotope i and space point j

ξ_i = energy release per radiative capture, Mev

H = length of assembly, cm

3.3.3 Power Due to Kinetic Heating

This component of power is a composite term that includes heat arising from elastic scattering in moderators and inelastic scattering in fuel materials. It is calculated assuming an average energy release per second of 1.93 Mev per fission neutron produced in the lattice cell. This component is related to the fission reaction rate by

$$IS = \sum_{i=1}^{16} \sum_{j=1}^{NX} \left(R_{ij}^{F} + R_{ij}^{T} \right) \nu_i H(1.93) \Big/ 6.25 \times 10^{18} \qquad (3.4)$$

where

IS = power due to kinetic heating, MW

NX = number of space points in cell

$R_{ij}^{F,T}$ = fast and thermal fission reaction rates per cm of length for isotope i at space point j

ν_i = average number of neutrons produced per fission for isotope i

H = length of assembly, cm

Values of 1.93 ν_i are given for each of the sixteen "fuel chain" isotopes in Table A-I of Appendix A.

3.3.4 Fission Product Decay Power

Fission product decay power is computed from the concentrations of 142 fission products that are carried for each lattice cell. The power is derived from

$$DK = \sum_{i=1}^{142} \sum_{j=1}^{NX} \lambda_i N_{ij} \xi_i H \Big/ 6.25 \times 10^{18} \qquad (3.5)$$

where

DK = decay heat, MW

NX = number of space points in cell

λ_i = decay constant of i^{th} fission product

N_{ij} = concentration of i^{th} fission product at space point j, atm/cc

ξ_i = energy release per decay, Mev

H = length of assembly, cm

Values of energy release per decay are given in Table A-III of Appendix A.

The four components of the total power are computed from equations (3.2-3.5) using arbitrarily normalized reaction rates

per unit of length. The renormalization factor K_k necessary to obtain the absolute fluxes and reaction rates is then determined from

$$P_k = K_k(F_k + RC_k + IS_k + DF_k) \tag{3.6}$$

where

P_k = specified power for k^{th} lattice cell

K_k = flux renormalization factor for k^{th} lattice cell

F_k, RC_k, IS_k, DF_k = four arbitrarily normalized components of power for k^{th} lattice cell

3.4 CROSS SECTIONS

In the process of performing a burnup calculation, cross sections by isotope and space point are well known only at the times that lattice cell calculations are performed. To improve the cross sections between the known time points, the cross sections by isotope and space point are fitted to a quadratic form as a function of the remaining fraction of the initially most abundant fissile "fuel isotope" in the lattice cell. The fitting is constantly updated to use only the three most recent lattice cell calculations for the fitting. The first and second passes through the BURNUP program are somewhat different in that only one or two lattice cell calculations have been performed; hence, a constant and linear fit are used in these passes, respectively.

Cross sections for isotopes in the "fuel chain" and "added isotope chain" isotopes are obtained by two methods. For isotopes that are present in the lattice cell calculation, two group cross sections by isotope and space point are obtained from the reaction rates using the known fluxes and concentrations. Generally only about half of the "fuel chain" isotopes are present in the lattice cell calculation, and in many cases the coupling between these isotopes must be provided by isotopes not in the lattice cell. To obtain the necessary cross section for the "fuel chain" isotopes that do not appear in the lattice cell, a set of empirically derived polynomial expressions based on the methods used in the study of higher actinide element production[13] are built into the BURNUP program. The constants for these polynomials were derived from a series of calculations using the HAMMER System[6] for a wide range of isotopic concentrations.

The method is based on the fitting of the isotopic absorption cross section and absorption resonance integrals as a function of simply derived spectrum constants and shielding factors. The expressions involved are:

$$\sigma_{th}/\sigma_{2200} = A_0 + A_1(\phi_{epi}/\phi_{th}) + A_2(\phi_{epi}/\phi_{th})^2$$
$$\sigma_{fast}/I_{eff} = B_0 + B_1(\phi_{epi}/\phi_{th}) + B_2(\phi_{epi}/\phi_{th})^2 \quad (3.7)$$

where

σ_{th},σ_{2200} = thermal spectrum-averaged and 2200 m/sec cross section, respectively

σ_{fast} = epithermal spectrum-averaged cross section including resonance capture

I_{eff} = effective resonance integral

ϕ_{epi}/ϕ_{th} = epithermal to thermal flux ratio

A_0,A_1,A_2 = fitted constants for each isotope (Appendix A)

B_0,B_1,B_2 = .00267, .0554, -.00173 for uranium fuel
= .009, .044, -.00098 for plutonium fuel

The effective resonance integral is related to the infinitely dilute integral by

$$\log_{10}(I_\infty/I_{eff}) = C_0 + C_1 \log_e (\log_{10} \sigma_p) + C_2 \log_{10} \sigma_p \quad (3.8)$$

where

I_∞,I_{eff} = infinitely dilute and effective resonance integrals, respectively

σ_p = potential scattering cross section per absorber atom
= $(\Sigma_s + S_{eff}/4V)/N$

C_0,C_1,C_2 = fitted constants for each isotope (Appendix A)

The cross sections for an isotope at a given space point may be obtained from these expressions in the following manner. The potential scattering cross section (σ_p) is evaluated in the resonance integral treatment in the HAMLET program and passed to the

BURNUP program with the epithermal reaction rates. The epithermal to thermal flux ratio (ϕ_{epi}/ϕ_{th}) is evaluated at each space point from the known fluxes. Absorption cross sections are then derived from the expression:

$$\sigma^a_{th} = (\sigma_{th}/\sigma_{2200})\sigma^a_{2200}$$
$$\sigma^a_{fast} = (\sigma_{fast}/I_{eff})/(I_\infty/I_{eff})\ I^a_\infty \qquad (3.9)$$

where the ratios involved are calculated from equations (3.7 and 3.8), and the 2200 m/sec absorption cross section and infinitely dilute resonance integrals are stored constants. The values of the constants used in these calculations are shown in Table A-II of Appendix A.

The epithermal fission cross sections are derived from the absorption cross section using the ratio of infinitely dilute fission to absorption resonance integrals. Thermal fission cross sections are derived for most isotopes from the same fitted constants as for absorption. There are, however, two isotopes ^{239}Pu and ^{241}Pu that use a separate set of fitted coefficients in the expansion given in equation (3.7). These coefficients are shown in Table A-II.

If the epithermal-to-thermal flux ratio exceeds a value of 7.0, the expansions given in equation (3.7) no longer give reasonable cross section values. For these cases, these equations are replaced by

$$\sigma_{fast}/I_{eff} = 0.0602(\phi_{epi}/\phi_{th})$$
$$\sigma_{th}/\sigma_{2200} = \chi \qquad (3.10)$$

where χ is a spectral index. χ is calculated as the thermal spectrum-averaged value of a 1/v pure absorber with a 2200 m/sec cross section of 1 barn. The calculation of this spectral index requires the presence of this 1/v absorber in all "fuel" and "added isotope" regions.

3.5 FUEL CHAIN BURNUP

After isotopic concentrations, fluxes, reaction rates, and cross sections for a lattice cell are assembled, then the coupled first order depletion and production equations for the sixteen "fuel chain" isotopes are solved. The equations represent the

depletion or production of an isotope by the processes of neutron capture, fission, alpha decay, and beta decay. These equations are of the following form:

$$\frac{dN_{ij}}{dt} = -\left(R_{ij} + C_{ij} + \alpha_{ij} + \beta_{ij}\right) + \sum_{k\neq j}\left(\alpha_{i,k\rightarrow j} + \beta_{i,k\rightarrow j} + C_{i,k\rightarrow j}\right) \tag{3.11}$$

where

N_{ij} = concentration (atm/cm^3) of isotope j at space point i

R_{ij} = effective one group fission reaction rate per cm for isotope j at space point i

C_{ij} = effective one group capture reaction rate per cm for isotope j at space point i

α_{ij} = effective one group alpha decay reaction rate per cm for isotope j at space point i

β_{ij} = effective one group beta decay reaction rate per cm for isotope j at space point i

$C_{i,k\rightarrow j}$ = effective one group capture reaction rate per cm for isotope k to produce isotope j at space point i

$\alpha_{i,k\rightarrow j}$ = effective one group alpha decay reaction rate per cm for isotope k to produce isotope j at space point i

$\beta_{i,k\rightarrow j}$ = effective one group beta decay reaction rate per cm for isotope k to produce isotope j at space point i

Equation (3.11) is solved in one energy group for each isotope at each space point by defining effective thermal reaction rates and cross sections. These are defined by the following equations:

$$\sigma_{eff} = \sigma_{th} + \frac{\phi_{epi}}{\phi_{th}}\sigma_{fast}$$

$$R_{eff} = R_{th} + \frac{\phi_{epi}}{\phi_{th}}R_{fast} \tag{3.12}$$

Equation (3.12) provides accurate one group reaction rates when the epithermal-to-thermal flux ratio remains relatively constant. This flux ratio is assumed to be constant for each pass through

the BURNUP program; hence, it is the factor that determines how frequently lattice cell calculations must be performed. Lattice cells in which the spectrum changes very slowly can be depleted for much larger time segments than those in which the spectrum changes rapidly.

The basic time step used in each iteration of the Runge-Kutta-Gill solution of equation (3.11) is 20 hr except for the first iteration which is 5 hr. Additional time subdivisions are included during the iterative solution for specified edits at the proper times or exposures. For reactor supercell problems, all lattice cells are carried through a time step before proceeding to the next iteration. Care should be exercised in supercell calculations that the edit intervals do not create many very short time steps as this has a very strong effect on the total computer time for a problem.

Following each time interval in solving equation (3.11), the fluxes and reaction rates are renormalized to conform to the input power specification. Two effects are considered in this renormalization: cross section changes and concentration changes. Section 3.4 describes the fitting of cross sections as a function of time. The effects of the concentration changes on the fission reaction rate of the cell are determined, and the fluxes and reaction rates are scaled by a factor necessary to preserve the total power in the cell at its specified value.

Renormalization of power for a reactor supercell is more difficult than for an infinite uniform lattice. The supercell may consist of three types of assemblies: fuel, target, and control. Fuel and target assemblies differ in that the latter have cell multiplications less than one. Within the reactor supercell, each assembly has a different total power at each instant of time; hence, cumulative exposure will be different for each assembly over a fixed depletion interval. Further, the power in a target assembly may change with time due to production of fissile materials.

The reactor supercell is depleted by determining the initial power distribution by renormalizing the reaction rates for each assembly to conform to the supercell reaction rates, and then by carrying all assemblies forward in time by equal time intervals. Target assemblies increase or decrease power during these intervals from production or depletion of fissile isotopes. These power changes, however, are coupled to flux changes in all fuel assemblies. The method used to couple target and fuel assembly fluxes during a single time step will now be described.

A reactor supercell consists of "driven" assemblies (assemblies with cell multiplication less than 1.0) and "driver" assemblies (assemblies with cell multiplications greater than 1.0).

Flux levels in the "driven" assemblies are less dependent on fissions that occur in themselves than on the fluxes that exist in the "driver" assemblies. In order to establish a straight time march in the burnup of the "driven" assemblies, a coupling is established between "driven" and "driver" assemblies based on the reactor supercell calculation.

From the neutron balance for each assembly, the strength of each assembly as a "leakage source" of neutrons is

$$L = \text{Total leakage}$$

$$= \text{Neutrons produced} - \text{Neutrons absorbed} - \text{Axial leakage} \tag{3.13}$$

To distinguish between "driver" and "driven" elements, an index may be defined for each assembly by

$$C = \frac{\text{Total fission source}}{\text{Total fission source} + \text{Total leakage}} \tag{3.14}$$

where leakage into a lattice cell is taken as positive. The index from equation (3.14) classifies assemblies as having $C < 1.0$ for "driven" assemblies and $C \geq 1.0$ for "driver" assemblies. Initially the absolute fluxes for each lattice cell are determined from the power distribution in the reactor supercell. As the solution iterations of equation (3.11) proceed, a multiplier is calculated to scale the fluxes in the "driven" assemblies as follows:

$$\eta_j = \frac{\sum_{i=1}^{ND} \left(K_i^j / K_i^{j-1}\right) L_i N_i}{\sum_{i=1}^{ND} L_i N_i} \tag{3.15}$$

where

η_j = flux scale factor for j^{th} iteration

K_i^j = absolute flux normalization factor from equation (3.6) for i^{th} driver assembly and j solution iteration

L_i = total leakage for assembly i from equation (3.13)

N_i = number of assemblies of type i in supercell

ND = number of "driver" assemblies

The renormalization of fluxes and reaction rates for "driven" assemblies accounts for cross section changes in the same manner as for "driver" assemblies; however, flux changes are determined by the multiplier in equation (3.15) rather than from preserving the total number of fissions per unit time as constant. The power in these "driven" assemblies therefore will increase or decrease based on the coupling coefficient; whereas the "driver" assemblies are assumed to have constant power.

Equation (3.11) is solved using the Runge-Kutta-Gill iterative method with the fluxes and reaction rates renormalized as above following each iteration. This time iteration continues until a predetermined termination time is reached for a particular pass through the BURNUP program. The iterative solution of equation (3.11) is terminated when the concentration of the initially most abundant fissile "fuel chain" isotope is reduced to a specified fraction of its concentration at the start of a pass through the BURNUP program. Input options permit this change to be 3% (design calculation), 7% (intermediate calculation), or 10% (survey calculations) of the initial concentrations.

As equation (3.11) is being iterated, the fluxes and fission reaction rates by isotope and space point are saved at five time points equally spaced between start and finish of the solution iterations. These are stored for later use by the "added isotope" burnup and fission product calculations.

Termination of the entire burnup problem is determined by user option of one of the following conditions occurring:

- A lattice cell attains a specified exposure (megawatt days).
- A "fuel chain" isotope depletes to a specified fraction of its initial concentration.
- A fertile "fuel chain" isotope attains a concentration equal to a specified fraction of all "fuel chain" isotopes.

In a reactor supercell, the specifications are made for each cell separately and the first occurrence will terminate the problem.

3.6 ADDED ISOTOPE BURNUP

The purpose of the "added isotope" calculation is to allow non-"fuel chain" isotopes to be depleted and produced. Generally these "added isotopes" are either burnable poisons that have an effect on the cell spectrum, or they represent peripheral isotopes

that have no spectral effects but are of interest. A special option is available that will allow a chemical shim to be crudely approximated.

An "added isotope chain" can have no more than three isotopes with only the first isotope in the chain being in the lattice cell. Several standard chains are built into the BURNUP program as shown in Figure 3.3, and special chains may be defined as required.

The concentrations of the "added isotopes" as a function of time are obtained by performing a solution of the coupled differential equations in the same manner as for the "fuel chain" isotopes, except that only capture and decay reactions are considered. Absolute fluxes are obtained from fitting the five sets of spatial fluxes saved in the "fuel chain" calculation. Reaction rates are obtained from the lattice cell for the first isotope in the chain normally, and all others are determined from stored or supplied cross sections, fluxes, and concentrations. Spectral effects on cross sections are accounted for using the spectrum-averaged cross section of a pure 1/v absorber with $\sigma_{2200} = 1.0$ barn as a spectral index. The effective thermal cross section is then given by:

$$\sigma_{eff} = \sigma_{2200}\,\chi + \frac{\phi_{epi}}{\phi_{th}}\,\sigma_{fast} \qquad (3.16)$$

where χ is the thermal spectral index.

No self-shielding of the epithermal cross section is assumed because if the isotope is present in such concentrations, it should logically be included in the lattice cell. The computation of the spectral index requires that the 1/v pure absorber appear in each "added isotope" region at a low number density (1.0×10^{-15} atm/bn-cm).

The solution of the coupled depletion and production equations is performed by a Runge-Kutta-Gill method with a basic time step of 20 hours. Fluxes are corrected from the fitted values at each iteration, and reaction rates are corrected for flux and concentration changes. Cross sections are assumed to remain constant during this interval of time.

A special "added isotope" chain may be defined for a lattice cell using the type 19 and 20 input cards (Section 4). Each isotope in the chain is assigned a unique identification number, and its coupling to the next isotope, 2200 m/sec thermal absorption cross section, resonance integral (possibly shielded), and decay constant are required on type 20 cards. If the isotope is to have

an initial non-zero concentration it must be included on type 19 cards. If the identification number of the first isotope in the chain matches that of an isotope in some region of the lattice cell, the concentrations, reaction rates, and cross sections will be derived from the cell reaction rates.

A special option is available in the added isotope calculation to simulate a chemical control material distributed in a moderator or coolant region. For this case the concentration of the control isotope is taken from the lattice cell, but reaction rates are computed from input cross sections and decay constants. By adjustment of these constants, the behavior of the chemical control material may be simulated in the cell, reactor, and depletion calculations. The cross section and decay constants are not generally meaningful as physical quantities, but are chosen by trial and error to match the required behavior of the chemical control material.

3.7 FISSION PRODUCT CALCULATION

The final calculation for each lattice cell in a pass through the BURNUP program is a calculation of the fission product concentrations. This is performed by a modified version of the CINDER program.[1] This method of calculation represents the fission products by a series of decoupled linearized chains that may be solved analytically for the isotopic concentrations as a function of time.

The calculation contained in the BURNUP program contains 142 fission product nuclides in 32 linearized chains. This number of fission products and chains is much smaller than the usual number considered in a CINDER calculation; however, the chains and nuclides in the BURNUP program version are capable of accounting for more than 95% of the total fission product poisoning predicted by the normal CINDER program.

For the purposes of the lattice cell calculation, the fission product poisoning is represented by four components:

- ^{135}Xe concentration
- ^{149}Sm concentration
- Fully saturated lumped fission product concentration
- Unit 1/v pure absorber concentration

The purpose of the 1/v pure absorber is to approximate the cross section differences arising from differences between the spectrum over which the saturated lumped fission product cross sections are averaged and the actual lattice cell spectrum. This representation requires that each fuel region in the lattice cell contain the four isotopes above in at least trace concentration (1.0×10^{-15} atm/bn-cm).

The fission product calculation uses the fluxes and fission reaction rates from the "fuel chain" burnup calculation, which were saved at five time points in the calculation. Direct fission product yields as a function of time are determined by fitting the fission reaction rates to a quadratic form as a function of time, choosing a time step for solving the linearized chains, then determining the yields as a function of time using the isotopic yield tables given in Table A-IV. Fluxes are obtained as a function of time by a similar fitting procedure.

Effective cross sections for each fission product are obtained from the expression given in equation (3.16) using internal values of σ_{2200} and σ_{fast} (Table A-III). The thermal spectral index is obtained from the 1/v pure absorber. Cross sections for ^{135}Xe and ^{149}Sm are obtained from reaction rates from the lattice cell. Cross section and decay constants for the fission products are shown in Table A-III.

The basic time step used in the fission product calculation is 25 hours unless modified by special conditions determined within the program.

Once the concentrations of the fission products are known at a time when a spectrum renormalization must be performed, the cross sections of the many fission products must be expressed in an equivalent number density of lumped fission products. Because ^{135}Xe and ^{149}Sm are explicitly represented in the lattice cell, their concentrations are carried into the lattice cell. The remainder of the fission products are expressed in terms of the saturated lumped fission products and 1/v pure absorber. To do this the thermal and epithermal cross sections of the lumped fission products and the thermal cross section of the 1/v pure absorber are first derived from their reaction rates passed from the last lattice cell calculation. The concentration of saturated lumped fission products is then chosen to represent the total epithermal poisoning in the form

$$N_{LFP} = \frac{\sum_{i=1}^{All\ FP} N_i \sigma_i^F}{\sigma_{LFP}^F} \qquad (3.17)$$

where

N_{LFP} = concentration of lumped fission products, atm/bn-cm

σ^F_{LFP} = epithermal capture cross section for lumped fission products obtained from lattice cell reaction rates

N_i = concentration of i^{th} fission product, atm/bn-cm

σ^F_i = epithermal cross section of i^{th} fission product

The concentration of the lumped fission product given in equation (3.17) will also predict a thermal poisoning that will depend upon the lattice cell spectrum; however, the total thermal poisoning predicted by the fission product calculation may be different. The 1/v pure absorber concentration (positive or negative) is chosen to make up this difference and is given by

$$N_{1/v} = \frac{\sum_{i=1}^{All\ FP} N_i \sigma^T_i - N_{LFP}\sigma^T_{LFP}}{\sigma^T_{1/v}} \qquad (3.18)$$

where

$N_{1/v}$ = concentration of 1/v pure absorber, atm/bn-cm

N_i = concentration of i^{th} fission product, atm/bn-cm

σ^T_i = thermal capture cross section of i^{th} fission product

N_{LFP} = concentration of lumped fission product, atm/bn-cm

σ^T_{LFP} = thermal capture cross section for saturated lumped fission products obtained from lattice cell reaction rates

$\sigma^T_{1/v}$ = thermal capture cross section for 1/v pure absorber derived from lattice cell reaction rates

In addition to the fission product concentrations, the heat energy per unit time from the decay of fission products is determined for use in the flux normalization from total power.

3.8 FISSION PRODUCT COOLING

Following the termination of the burnup problem it is often of interest to follow the decay heat and activity of a lattice cell. This is provided by performing a 75-week fission product calculation under zero flux conditions. An optional edit will provide the decay heat, activity, and concentration of seven standard and one optional fission products at one-week intervals for up to 75 weeks. The standard isotopes in this calculation are ^{95}Zr, ^{95}Nb, ^{103}Ru, ^{106}Ru, ^{134}Cs, ^{137}Cs, and ^{144}Ce.

3.9 RESTRICTIONS TO LATTICE CELL IMPOSED BY BURNUP CALCULATION

In the preceding sections, several specific restrictions have been imposed. These restrictions are collected here for convenience.

- All lattice cell cases must originate from input by cards or from a Burnup History Tape at the start of a burnup job.
- Each "fuel region" and "added isotope region" in a lattice cell must contain a separate material mixture number.
- All isotopes that will have an appreciable effect on the lattice cell calculation at any time during the burnup problem must appear in the lattice cell at the start. All isotopes in the lattice cell must be at a non-zero concentration (1.0×10^{-15} atm/bn-cm).
- A "fuel region" in a lattice cell must contain the isotopes ^{135}Xe, ^{149}Sm, saturated lumped fission products, and 1/v pure absorber at a non-zero concentration.
- An "added isotope region" must contain a 1/v pure absorber at a non-zero concentration.
- A limit of 15 lattice cases may be processed through the BURNUP program at one time.
- Only one reactor supercell calculation may be performed during each iteration of a burnup job.

4. INPUT

4.1 GENERAL

Most input cards have the same field definition (I1,1X,I2, 1X,2I2,I5,2I2,I1,1X,5E10.0). The data required and changes to the above format are given on the input sheets below.

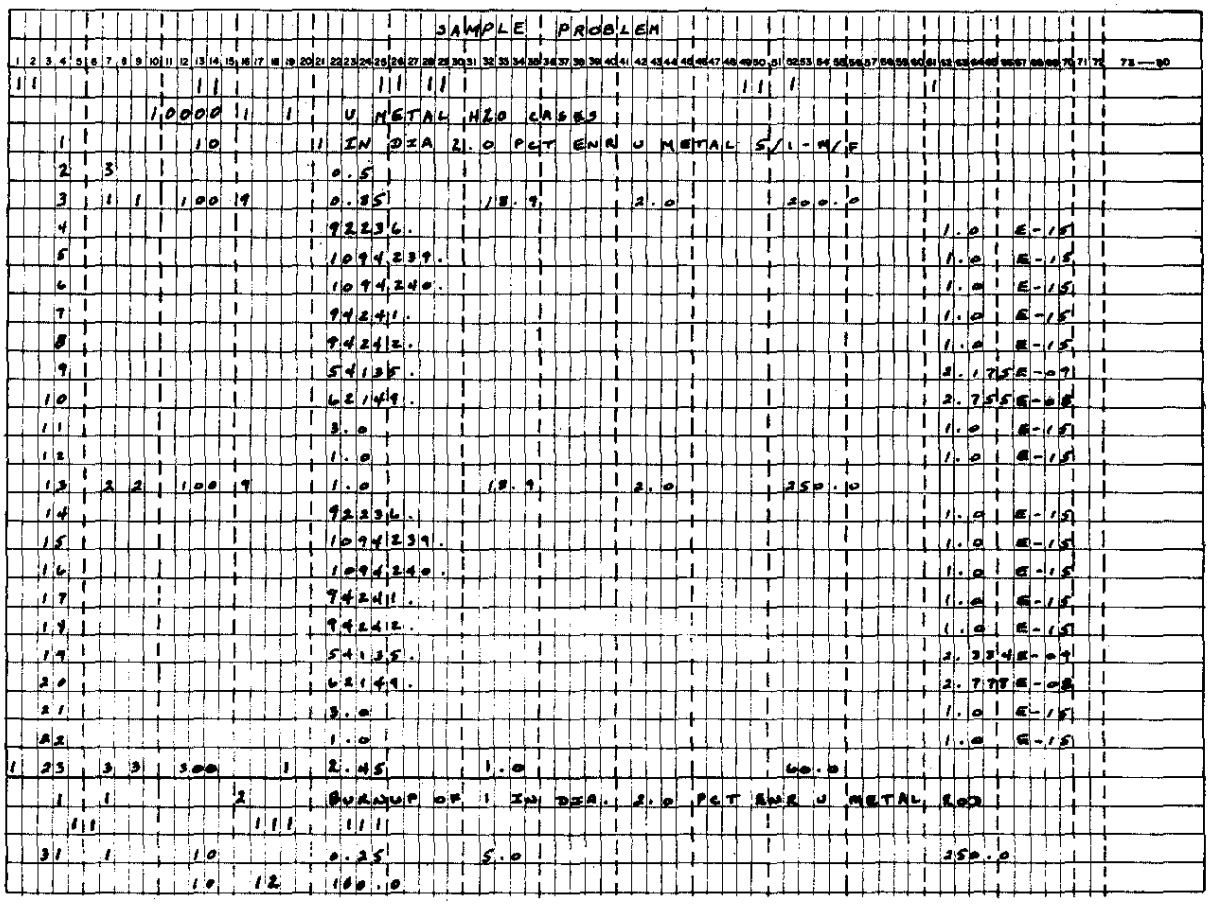
Only the first case in a batch of THERMOS-HAMLET problems need be complete; subsequent cases must include the Case Lead Card (type 2 as specified below) and at least one additional card (types 3-5). Card images are contained in core; on subsequent cases the card images with corresponding sequence numbers are overwritten; hence, only data to be changed need be input after the first card of a batch.

Input to the FLOG section is similar to the THERMOS-HAMLET batches. Only the first problem need be complete; subsequent cases must include the Reactor Lead Card (type 6) and at least one additional card (card types 7-11).

Input to the HERESY section is similar. Only the first problem need be complete; subsequent cases must include the Reactor Lead Card (type 12) and at least one additional type 13 and type 14 card.

Data to the BURNUP section of the code are input only once and are retained internally. Card types 15 to 20 are used.

The system executes the various programs in a fixed order of (1) CAPN, (2) THERMOS, (3) HAMLET, (4) FLOG or HERESY, (5) DIED, and (6) BURNUP as specified on the Batch Control Card. Input data blocks for each section are also ordered in this sequence for a given batch.


SAMPLE PROBLEM
10000
U METAL H2O CASES
IN DZA 21.0 PCT ENR U METAL S/L-H/F
0.5
0.35
18.9
2.0
200.0
92236.
10942390.
10942400.
942411.
942412.
54135.
621049.
3.0
1.0
1.0
18.9
2.0
250.0
92236.
10942390.
10942400.
942411.
942412.
54135.
621049.
3.0
1.0
2.45
1.0
60.0
DUPLICATE OF 1 IN DZA 2.0 PCT ENR U METAL BOX
0.25
5.0
350.0
10
12
100.0

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

CARD TYPE 0

Batch Control Card (One per Batch)

Program	Columns	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
THERMOS	13	NTHRN	No THERMOS run = 0; standard THERMOS run = 1; cosine current THERMOS run = 2
	14	NTHPT	Short print = 0; long print = 1
HAMLET	25	NEPRN	No HAMLET run = 0; standard HAMLET run = 1
	26	NEPPT	Short print = 0; long print = 1; long print with regionwise RI = 2
	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)
REACTOR	37	NFGRN	No reactor run = 0; standard FLOG run = 1, standard HERESY run = 2
	38	NFGPT	Short print = 0; long print = 1
	40	NFGPB	Point buckling calculation = 1; no = 0 (FLOG)
	42	NFGB1	Boundary condition at origin (FLOG) 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 (FLOG) 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 (LLT) edit = 2 (next section)
	50	NEDPT	Short print = 0; long print = 1
	52	NEDNB	Detailed Neutron Balance (DNB) 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 (If NFGRN = 2, only 0 or 1 is available) 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 (40I1; 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 = LLT edit
	50	NEDPT	0 Print case ident, titles only 1 Plus integral reactivity parameters 2 Plus few-group data 3 Plus overlapping thermal group data 4 Case ident, title, region input data 5 Plus mesh point data 6 Complete listing of LLT data
BURNUP	61	NBURN	No burnup run = 0; standard burnup run = 1
	62	NBURI	No RI print = 0; standard RI print = 1; Regionwise RI print = 2
	63	NBUPN	= 0, do not punch cell input cards at each spectrum renormalization; = 1, punch cell input cards at each spectrum normalization.

CARD TYPE 1

Batch Lead Card

Field	Columns	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 reactor cases in this batch (No. FLOG or HERESY problems)
8	19	I1	NEWLIB	= 1 if starting a new lattice library, = 0 if added to an old one
9	21-30		BTTL	Hollerith(12A4,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	Columns	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 ident; 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 (12A4,A2) lattice title
12	51-60			
13	61-70			

CARD TYPE 3

Lattice Description Card (One per Lattice)

Field	Columns	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 ≤ 20
5	10-14	I5	MX	Number of mixtures ≤ 10
6	15-16	I2	NP1B1	P1 = 0; B1 = 1 option for Fourier transform leakage calculations
7	17-18	I2	N4FT	= 0 calculate 4-group flux ratios at calculated B_m^2 = 1 calculate 4-group flux ratios at input B^2
8	19	I1	ISØXE	Number of foil materials to be edited (must *not* duplicate cell materials) ≤ 4
9	21-30	E10.0	BSQD	Material or leakage buckling (m^{-2}) for Fourier transform leakage calculations
10	31-40	E10.0	WSTBA(I), I=1, ISØXE	Foil identification numbers
11	41-50	E10.0		
12	51-60	E10.0		
13	61-70	E10.0		

CARD TYPE 3A*

Lattice Description Card for Burnup Recycle Input Option

Field	Columns	Format	Symbol	Description
1	1	I1	LCT	= 1 if last card, 0 otherwise
2	3-4	I2	NC	Card sequence number (within a case = 2)
3	6-7	I2	NGEØM	= 4 always
4	8-9			
5	10-14	I5	NCSID	Uniform lattice case ident or supercell rod type on Burnup History Tape used for input
6	15-16			
7	17-18	I2	N4FT	= 0 calculate 4-group flux ratios at calculated B_m^2 = 1 calculate 4-group flux ratios at input leakage buckling
8	19			
9	21-30	E10.0	BSQD	Material buckling or leakage buckling (m^{-2}) for Fourier transform leakage calculation
10	31-40	E10.0	XMWD**	Burnup (MWD) at which data are required
11	41-50	E10.0	XTIM**	Time (hr) at which data are required
12	51-60	E10.0	XFRAC**	Fraction of principal fissile material remaining at time data are required
13	61-70	E10.0	TIMC	Time (hr) after shutdown of previous cycle to start of new cycle

* This card replaces the normal type 3 input card when the burnup recycle input option is used. This card may be followed by type 3B cards only.

** The first non-zero value in fields 10, 11, and 12 will be used to determine data from the Burnup History Tape.

CARD TYPE 3B

Recycle Concentration Change Card

Columns	Format	Symbol	Description
1	I1	LCT	= 1 if last card, 0 otherwise
3-4	I2	NCT	Card sequence number
0-7			
8-9			
10-14	I5	IREGN	Region number in cell in which isotope concentration is changed
15-16			
17-18			
19			
21-30	E10.0	WSTBA	First ident of changed isotope
31-40	E10.0	WSTBB	Second ident of changed isotope
41-50			
51-60			
61-70	E10.0	CØNC	New isotope concentration (atm/bn-cm)

CARD TYPE 4

Lattice Region Description Card (One per Region)

Field	Columns	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
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	Columns	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 ident (see list following card types 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)
12	51-60	E10.0	A	Atomic weight this isotope
13	61-70	E10.0	CØNCTA	Concentration (atm/bn-cm); if this specification is given, D on card 4 is exclusive of this isotope

FLOG (Card Types 6-11)

CARD TYPE 6

Reactor Lead Card (One per Reactor)

Field	Columns	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 ident 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 (12A4,A2) reactor title
12	51-60			
13	61-70			

CARD TYPE 7

Reactor Description Card (One per Reactor)

Field	Columns	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, ≤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	Columns	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	Columns	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	Columns	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
5	10-14	I5	MAT	Material used (lattice ident) 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	-	

* See Table A-VI of Appendix A.

CARD TYPE 11

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

Field	Columns	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 ident
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

HERESY (Card Types 12-14)

CARD TYPE 12

Heterogeneous Reactor Lead Card

Field	Columns	Format	Mnemonic	Description
1	1			
2	3-4			
3	6-7			
4	8-9			
5	10-14	I5	NRID	Reactor case ident
6	15-16			
7	17-18			
8	19			
9	21-70	12A4,A2	TITLE(I)	Reactor case title
10				
11				
12				
13				

CARD TYPE 13

Heterogeneous Reactor Fixed Point Data Card

Field	Columns	Format	Mnemonic	Description
1	1	I1	LAST	= 0 except for last card of fixed point data
2	2-4	-		
3	6-7	I2	NUM	Number of data on this card (if zero assumes 5)
4	8-9	-		
5	10-14	I5	LØC*	Relative location of first datum on this card
6	15-16	-		
7	17-18	-		
8	19	-		
9	21-30	I10	JJ(1)	Datum 1
10	31-40	I10	JJ(2)	Datum 2
11	41-50	I10	JJ(3)	Datum 3
12	51-60	I10	JJ(4)	Datum 4
13	61-70	I10	JJ(5)	Datum 5

* See list following card type descriptions.

CARD TYPE 14

Heterogeneous Reactor Floating Point Data Card

Field	Columns	Format	Mnemonic	Description
1	1	I1	LAST	= 0 except for last card of floating point data
2	3-4	-		
3	6-7	I2	NUM	Number of data on this card (if zero assumes 5)
4	8-9	-		
5	10-14	I5	LØC*	Relative location of first datum on this card
6	15-16	-		
7	17-18	-		
8	19	-		
9	21-30	E10.0	BB(1)	Datum 1
10	31-40	E10.0	BB(2)	Datum 2
11	41-50	E10.0	BB(3)	Datum 3
12	51-60	E10.0	BB(4)	Datum 4
13	61-70	E10.0	BB(5)	Datum 5

* See list following card type descriptions.

BURNUP (Card Types 15-20)

CARD TYPE 15

Burnup Lead Card

Field	Columns	Format	Mnemonic	Description
1	1	I1	-	
2	3-4	I2	NCASES	Number of lattice cases for which data are to be read (≤15)
3	6-7	I2	NØLIST	Number of isotopes edit lists on type 16 cards to be read (≤3)
4	8-9	I2	NCD19	= 0, do not read type 19 cards = 1, read type 19 cards
5	10-14	-	-	
6	15-16	I2	ICALC	= 0 survey calculation (10% steps) = 1 intermediate calculation (7% steps) = 2 design calculation (3% steps)
7	17-18	I2	KAM	= 0 fuel chain 1* = 1 fuel chain 2*
8	19	I1	-	
9	21-70	(12A4,A2)	(BUTTL(I), I=1,13)	50 character burnup title

* See Figures 3.1 and 3.2 in Section 3.

CARD TYPE 16

Isotope Edit List

(Include NØLIST cards of this type)

Column	Isotope	Column	Isotope*
1	T	19	^{238}U
2	^{6}Li	20	^{237}Np
3	^{59}Co	21	^{238}Pu
4	^{60}Co	22	^{239}Np
5	^{135}Xe	23	^{239}Pu
6	^{149}Sm	24	^{240}Pu
7	^{169}Tm	25	^{241}Pu
8	^{170}Tm	26	^{242}Pu
9	^{171}Tm	27	^{243}Am
10	^{209}Bi	28	^{244}Cm
11	^{210}Bi	29	Gross Fission Products
12	^{210}Po	30	Special Added Isotope 1
13	^{232}Th	31	Special Added Isotope 2
14	^{233}Pa	32	Special Added Isotope 3
15	^{233}U	33	^{241}Am
16	^{234}U	34	^{242m}Am
17	^{235}U	35	^{242}Cm
18	^{236}U	36	^{245}Cm

* For each isotope indicate in the appropriate column if it is to be edited. A blank or zero indicates the isotope is not to be edited; any other number indicates it will be edited.

CARD TYPE 17

Burnup Parameter and Edit Option Card

(Include one card for each lattice case)

Field	Columns	Format	Mnemonic	Description
1	1-4	I4	NEDØPT*	Edit option for this case
2	6-7	I2	NLIST	Isotope list for this case
3	8-9	I2	NPWØPT	= 0, maximum assembly power given = -1, total supercell power given = n, rod type number for which power is given
4	10-14	I5	NCSID	Case ident for which these options apply
5	15-16	I2		
6	17-18	I2	IADD	Added isotope chain identification 0 - none 1 - cobalt 2 - thulium 3 - tritium 4 - polonium 5 - special chain
7	19	I1		
8	21-30	E10.0	ASSPWR	Assembly power (MW). For supercell meaning is specified by NPWØPT
9	31-40	E10.0	DELTT	Time step between edits (MWD)
10	41-50	E10.0	PCTCH**	Percent change of an isotope between edits
11	51-60	E10.0	WSØCH	Isotope ident for percent change edit
12	61-70	E10.0	XLNGTH	Length of assembly (cm)

* See Note c
** See Note d

CARD TYPE 18

Burnup Termination Option Card

(Include one card for each lattice case)

Field	Columns	Format	Mnemonic	Description
1	1-4	I4	IDFP*	Array number of any Special Fission Product to Edit following termination
2	6-7	I2		
3	8-9	I2		
4	10-14	I5	NCSID	Lattice ident to which these options apply
5	15-16	I2		
6	17-18	I2	NFPCL	Number of months for fission product cooling following termination
7	19	I1		
8	21-30	E10.0	TMWD	Total burnup at termination (MWD)
9	31-40	E10.0	FRCBRN**	Percent burnout of fissionable isotope at termination
10	41-50	E10.0	WSØBN	Fissionable isotope ident for termination
11	51-60	E10.0	FRCACC**	Percent accumulation of product at termination
12	61-70	E10.0	WSØACC	Product isotope ident for termination

* See Table A-III in Appendix A

** See Note d

CARD TYPE 19

Non-zero Added Isotope Card

(One card for each added isotope in each lattice case.
No more than three isotopes may be added per case)

Field	Columns	Format	Mnemonic	Description
1	1	I1	LAST	= 0 except for last type 19 card = 1
2	3-4	I2	NRGN(1)	1st region number to which isotope is added
3	6-7	I2	NRGN(2)	2nd region number to which isotope is added
4	8-9	I2	NRGN(3)	3rd region number to which isotope is added
5	10-14	I5	NCSID	Lattice case ident to which this isotope is added
6	15-16	I2	NRGN(4)	4th region number to which isotope is added
7	17-18	I2		
8	19			
9	21-30	E10.0	WISAD	Isotope ident to be added
10	31-40	E10.0	CØNCAD(1)	Concentration of isotope in 1st region (atm/bn-cm)
11	41-50	E10.0	CØNCAD(2)	Concentration of isotope in 2nd region (atm/bn-cm)
12	51-60	E10.0	CØNCAD(3)	Concentration of isotope in 3rd region (atm/bn-cm)
13	61-70	E10.0	CØNCAD(4)	Concentration of isotope in 4th region (atm/bn-cm)

CARD TYPE 20

Special Added Isotope Specification Card*

(Include only if IADD = 5 for a Case)

Field	Columns	Format	Mnemonic	Description
1	1	I1	LAST	= 0 except for last type 20 card = 1
2	3-4	-	-	
3	6-7	-	-	
4	8-9	-	-	
5	10-14	I5	NCSID	Case ident in which special isotopes are added
6	15-16	I2	ISØNØ	Isotope number in chain (1, 2, or 3)
7	17-18	I2	ICØUP	Coupling to next isotope in chain 0 - no coupling 1 - neutron capture 2 - decay (alpha or beta) 3 - pseudoshim material
8	19	-	-	
9	21-30	E10.0	SPISID	Isotope ident on type 19 card
10	31-40	E10.0	DECAY	Decay constant for this isotope (sec^{-1})
11	41-50	E10.0	XSECTN	2200 m/sec cross section for this isotope (barns)
12	51-60	E10.0	RINT	Resonance integral for this isotope (barns)
13	61-70	-	-	

* Non-zero concentrations for these Special Added Isotopes must be input on type 19 cards.

NOTES FOR CARD TYPES

Note a

Predefined Materials

Materials Available	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 (atm/bn-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.

Note c

BURNUP Edit Options

The edit option for a burnup case is obtained by adding together control numbers for each edit requested. This sum is the edit option to be entered on the type 17 card for a case. If plots of the edited information are desired, the option is entered as a negative number; a positive edit option omits all plots.

Edit	Control Number
Four-Group Absolute Fluxes	16
Cross Sections	8
Concentrations	4
Material Balance	2
Isotopic Power	1

Note d

Percent Edit and Termination Options

The edit and termination options of the BURNUP program that involve percent change, burnup, or production of an isotope are specified according to very restrictive rules. These rules are applicable to "fuel chain" isotopes only with atomic number (Z) greater than 80.

Fissile Isotopes

If the isotope being keyed on is fissile and decreasing in concentration, the edit frequency or termination concentration is determined from a concentration that equals the specified percent of the initial concentration. If the isotope is fissile and increasing in concentration, the edit frequency or termination concentration is determined from a concentration that equals the specified percent of all isotopes in the cell with atomic number greater than 80.

Nonfissile Isotopes

The edit frequency or termination concentration is determined from a concentration that equals the specified percent of all isotopes with atomic number greater than 80.

^{240}Pu

For the isotope ^{240}Pu, the specification is taken to mean the percent of ^{240}Pu relative to all plutonium isotopes (i.e., ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu).

Contents of HAMMER/HAMBUR Thermal Library (January 1972)

Isotope and Description	Ident	T/M	2200 m/sec Cross Section, barns σ_a	σ_f	σ_s
1/v unit absorber	1.	0.	1.0	0.0	0.0
Unit scatterer	2.	0.	1.0	0.0	0.0
Long-lived fission product pairs	3.	0.	70.0	0.0	0.0
Dummy zero cross section	9.	0.	0.0	0.0	0.0
304 Stainless	304.	0.	2.87	0.0	10.6
H Nelkin					
T = 293°K	1001.	1029.	0.328	0.0	20.4
T = 333°K	1001.	1033.	0.328	0.0	20.4
T = 363°K	1001.	1036.	0.328	0.0	20.4
T = 400°K	1001.	1040.	0.328	0.0	20.4
T = 506°K	1001.	1050.	0.328	0.0	20.4
T = 576°K	1001.	1057.	0.328	0.0	20.4
T = 600°K	1001.	1060.	0.328	0.0	20.4
H in C_6H_6					
T = 293°K	1001.	5029.	0.328	0.0	20.4
T = 373°K	1001.	5037.	0.328	0.0	20.4
T = 473°K	1001.	5047.	0.328	0.0	20.4
T = 536°K	1001.	5054.	0.328	0.0	20.4
T = 573°K	1001.	5057.	0.328	0.0	20.4
D Nelkin					
T = 293°K	1002.	1029.	0.00046	0.0	3.37
T = 333°K	1002.	1033.	0.00046	0.0	3.37
T = 363°K	1002.	1036.	0.00046	0.0	3.37
T = 400°K	1002.	1040.	0.00046	0.0	3.37
T = 500°K	1002.	1050.	0.00046	0.0	3.37
T = 600°K	1002.	1060.	0.00046	0.0	3.37
^{6}Li 1/v absorber	3006.	0.	945.0	0.0	1.52
^{7}Li zero absorber	3007.	0.	0.0	0.0	1.15
^{9}Be					
T = 293°K	4009.	29.	0.009	0.0	5.9
B natural	5000.	0.	755.	0.0	0.0
^{10}B 1/v absorber	5010.	0.	3837.	0.0	0.0
^{12}C gas					
T = 293°K	6012.	29.	0.0034	0.0	4.8
T = 453°K	6012.	45.	0.0034	0.0	4.8
T = 473°K	6012.	47.	0.0034	0.0	4.8
T = 553°K	6012.	55.	0.0034	0.0	4.8
T = 623°K	6012.	62.	0.0034	0.0	4.8
Graphite (Parks)					
T = 293°K	6012.	1029.	0.0034	0.0	4.8

Isotope and Description	Ident	T/M	2200 m/sec Cross Section, barns σ_a	σ_f	σ_s
Oxygen gas					
T = 293°K	8000.	0.	0.0	0.0	3.76
T = 293°K	8000.	29.	0.0	0.0	3.76
T = 333°K	8000.	33.	0.0	0.0	3.76
T = 363°K	8000.	36.	0.0	0.0	3.76
T = 400°K	8000.	40.	0.0	0.0	3.76
T = 500°K	8000.	50.	0.0	0.0	3.76
T = 576°K	8000.	57.	0.0	0.0	3.76
T = 600°K	8000.	60.	0.0	0.0	3.76
Mg	12000.	0.	0.073	0.0	3.7
Al	13000.	0.	0.23	0.0	1.4
Cr	24000.	0.	3.1	0.0	3.75
Fe	26000.	0.	2.53	0.0	11.5
Co	27000.	0.	37.2	0.0	7.1
Ni	28000.	0.	4.8	0.0	17.5
Zr	40000.	0.	0.18	0.0	6.0
Cd	48000.	0.	2550.	0.0	0.0
^{115}In	49115.	0.	195.	0.0	0.0
^{135}Xe	54135.	0.	3.08×10^6	0.0	4.3
^{149}Sm	62149.	0.	4.04×10^4	0.0	30.5
^{151}Eu	63151.	0.	8800.	0.0	5.0
^{152}Eu	63152.	0.	5000.	0.0	6.0
^{153}Eu	63153.	0.	390.	0.0	5.0
Gd	64000.	0.	46000.	0.0	5.0
^{164}Dy foil	66164.	0.	2704.	0.0	0.0
Tm	69000.	0.	127.	0.0	7.0
^{176}Lu foil	71176.	0.	2020.	0.0	0.0
^{197}Au foil	79197.	0.	98.8	0.0	0.0
Bi	83000.	0.	.034	0.0	9.0
^{232}Th	90232.	0.	7.56	0.0	12.5
^{232}Th	1090232.	0.	7.56	0.0	12.5
^{231}Pa	91231.	0.	200.0	0.0	11.0
^{233}Pa	91233.	0.	47.0	0.0	10.0
^{233}U	92233.	0.	573.	524.0	8.3
^{233}U fission foil	5092233.	0.	524.	0.0	0.0
^{234}U	92234.	0.	95.0	0.0	17.0
^{235}U	92235.	0.	682.	582.2	10.0
^{235}U fission foil	5092235.	0.	582.2	0.0	0.0
^{236}U	92236.	0.	5.119	0.0	10.4

Isotope and Description	Ident	T/M	2200 m/sec Cross Section, barns σ_a	σ_f	σ_s
^{236}U	1092236.	0.	5.119	0.0	10.4
^{238}U	92238.	0.	2.71	0.0	8.3
^{238}U	1092238.	0.	2.71	0.0	8.3
^{237}Np	93237.	0.	170.	0.019	10.0
^{238}Np	93238.	0.	2270.	2070.0	6.0
^{238}Pu	94238.	0.	562.8	16.29	4.4
^{239}Pu	94239.	0.	1015.0	740.9	9.0
^{239}Pu	1094239.	0.	1015.0	740.9	9.0
^{239}Pu fission foil	5094239.	0.	740.9	0.0	0.0
^{240}Pu	94240.	0.	279.7	0.0485	2.0
^{240}Pu	1094240.	0.	279.7	0.0485	2.0
^{241}Pu	94241.	0.	1357.4	962.0	12.6
^{241}Pu	1094241.	0.	1357.4	962.0	12.6
^{242}Pu	94242.	0.	18.7	0.0	7.68
^{242}Pu	1094242.	0.	18.7	0.0	7.68
^{241}Am	95241.	0.	585.8	3.27	10.0
^{242}Am	95242.	0.	2500.0	2500.0	10.0
$^{242}Am(m)$	195242.	0.	8000.0	6400.0	10.0
^{243}Am	95243.	0.	90.0	0.0	10.0
^{242}Cm	96242.	0.	20.0	3.0	10.0
^{244}Cm	96244.	0.	14.5	1.2	7.5
^{245}Cm	96245.	0.	2550.0	2205.0	12.0
^{246}Cm	96246.	0.	1.33	0.0	9.8

Contents of HAMMER/HAMBUR Epithermal Library (January 1972)

Isotope	Ident	Resonance Treatment	Capture RI_{∞},[a] barns	Fission RI_{∞},[b] barns	Smooth Absorption Integral,[c] barns
1/v unit absorber	1.		0.0	0.0	0.401
Unit scatterer	2.		0.0	0.0	0.0
Long-lived fission product pairs	3.		0.0	0.0	213.4
Dummy zero cross section	9.		0.0	0.0	0.0
304 SS	304.		0.0	0.0	1.06
H	1001.		0.0	0.0	0.132
D	1002.		0.0	0.0	0.0638
^{6}Li	3006.		0.0	0.0	377.6
^{7}Li	3007.		0.0	0.0	0.0131
^{9}Be	4009.		0.0	0.0	0.644
B	5000.		0.0	0.0	303.4
^{10}B	5010.		0.0	0.0	1532.0
C	6012.		0.0	0.0	0.0034
O	8000.		0.0	0.0	0.1297
Mg	12000.		0.0	0.0	0.0467
Al	13000.	Equivalence	0.0689	0.0	0.205
Cr	24000.		0.0	0.0	0.927
Fe	26000.		0.0	0.0	1.06
Co	27000.	Equivalence	49.54	0.0	19.7
Ni	28000.		0.0	0.0	1.95
Zr	40000.	Equivalence	0.688	0.0	0.481
Cd	48000.		0.0	0.0	32.4
^{115}In	49115.	Equivalence	3288.0	0.0	78.8
^{135}Xe	54135.		0.0	0.0	3684.0
^{149}Sm	62149.		0.0	0.0	3021.0
^{151}Eu	63151.		0.0	0.0	2480.0
^{152}Eu	63152.		0.0	0.0	1200.0
^{153}Eu	63153.		0.0	0.0	1059.0
Gd	64000.	Equivalence	232.4	0.0	191.3
^{164}Dy Dummy 1/v	66164		0.0	0.0	0.4
Tm	69000.	Equivalence	1454.9	0.0	319.0
^{176}Lu Dummy 1/v	71176.		0.0	0.0	0.4
^{197}Au	79197.	Equivalence	1528.2	0.0	42.79
Bi	83000.		0.0	0.0	0.056
^{232}Th	90232.	Nordheim	86.02	0.0	2.32
^{232}Th	1090232.	Equivalence	86.02	0.0	2.32
^{231}Pa	91231.	Equivalence	356.7	0.0	50.25
^{233}Pa	91233.	Equivalence	537.6	0.0	268.18
^{233}U	92233.	Equivalence	101.2	378.3	521.5

Isotope	Ident	Resonance Treatment	Capture RI_{∞},[a] barns	Fission RI_{∞},[b] barns	Smooth Absorption Integral,[c] barns
^{233}U foil	5092233.	Equivalence	378.3	0.0	459.4
^{234}U	92234.	Equivalence	601.1	0.0	93.6
^{235}U	92235.	Equivalence	89.41	132.3	158.5
^{235}U foil	5092235.	Equivalence	132.3	0.0	117.7
^{236}U	92236.	Nordheim	331.34	0.0	17.36
^{236}U	1092236.	Equivalence	331.34	0.0	17.36
^{238}U	92238.	Nordheim	268.2	0.0	3.38
^{238}U	1092238.	Equivalence	268.2	0.0	3.74
^{237}Np	93237.	Equivalence	452.47	0.0063	31.2
^{238}Np	93238.	Equivalence	0.0	0.0	879.21
^{238}Pu	94238.	Equivalence	133.28	10.2	20.65
^{239}Pu	94239.	Equivalence	126.74	146.4	184.1
^{239}Pu	1094239.	Nordheim	126.76	146.5	184.1
^{239}Pu foil	5094239.	Equivalence	146.4	0.0	123.4
^{240}Pu	94240	Equivalence	7951.0	1.32	51.6
^{240}Pu	1094240.	Nordheim	7952.5	1.32	51.6
^{241}Pu	94241.	Equivalence	118.3	424.3	185.4
^{241}Pu	1094241.	Nordheim	118.3	424.3	185.4
^{242}Pu	94242.	Equivalence	1094.3	0.0	28.56
^{242}Pu	1094242.	Nordheim	1094.9	0.0	28.56
^{241}Am	95241.	Equivalence	836.4	4.77	272.4
^{242}Am	95242.		0.0	0.0	499.6
$^{242}Am(m)$	195242.		0.0	0.0	2005.0
^{243}Am	95243.	Equivalence	1642.1	0.0	102.53
^{242}Cm	96242.		0.0	0.0	8.02
^{244}Cm	96244.	Equivalence	629.29	16.36	12.0
^{245}Cm	96245.	Equivalence	41.81	281.8	541.7
^{246}Cm	96246.	Equivalence	87.79	0.0	30.63

a. Sum of capture integrals from resolved and unresolved resonances.
b. Fission integral from resolved and unresolved resonances.
c. Integral of smooth capture and fission.

4.2 INPUT TO HERESY PROGRAM

Data are broken into three types and input in the following order:

- Reactor Lead Card (type 12) (one card only)
- Fixed Point Data Cards (type 13)
- Floating Point Data Cards (type 14)

For each problem there must be at least one card of each type in the input deck. The three data types are described separately below.

4.2.1 Reactor Lead Card

This card consists of one type 12 card.

4.2.2 Fixed Point Data Cards (Card Type 13)

Fixed point data are entered on type 13 cards that allow any number of individual datum up to five to be read from a single card.

All fixed point data are stored in a large array in core. Each datum is assigned a relative location in this array, and a list of these locations appears below. The relative location of datum 1 on the card is punched into field 5. Data 2-5 are stored in sequential relative locations in the array.

If it is not desirable to input a full card of data (five data), it is only necessary to place the number of data actually appearing on the card in field 3 (a blank or a zero will cause five data to be read). For instance, if it is necessary to change one property of one rod type in a list of fifty rod types, the relative location of the datum may be determined from the list below and placed in field 5, number 1 placed in field 3, and the new datum placed in the datum 1 field as defined.

All fixed point data are entered to the extreme right of the defined field, blank columns being read as zeros.

The last card of fixed point data must have a 1 punched in field 1, to indicate it is the last card.

4.2.3 Floating Point Data Cards (Card Type 14)

Floating point data are input on type 14 cards that enable any number of individual data up to five to be read from a single card. A floating point datum may be entered anywhere within the defined field on the card so long as the decimal point is assigned the correct position in the number. If no decimal point is entered, it is automatically assigned as following the character appearing in the last column of the datum field. Blank columns are read as zeros.

All floating point data are stored in an array in core, each datum being assigned a relative location in this array. A list of these relative locations appears below.

The relative location of datum 1 on the card must be punched into field 5, and provision for reading less than five data is identical to that for fixed point data.

The last card of floating point data must have a 1 punched in field 1 to indicate it is the last.

HERESY DATA LOCATIONS

Relative locations for fixed point data are contained in the following list.

Geometry Data

1.	JJ(1) = KØØR	*Geometry Generator Control*
	= 0	Use the geometry of the previous problem.
	= 1	Generate a new lattice geometry.
	= 2	Read lattice geometry from cards.
2.	JJ(3) = INF	*Problem Geometry Control*
	= 0	Finite lattice geometry.
	= 1	Infinite lattice geometry.
3.	JJ(4) = LAT	*Lattice Geometry Control*
	= 0	Hexagonal lattice.
	= 1	Square lattice.
	= 2	Pattern of rods on circles about axis of reactor.
4.	JJ(5)••• = NICPRT(I)	Number of initial equivalent coordinates input for each rod type (enter for finite lattices only). (If zero are set equal to 1) ($1 \leqslant I \leqslant NTYPE$).
5.	JJ(55)••• = NRPRT(I)	Number of rods in lattice of each rod type (enter only if KØØR = 2). ($1 \leqslant I \leqslant NTYPE$).
6.	JJ(105) = KPLØT	*Geometry Plot Option*
	= 0	Do not plot geometry.
	= 1	Plot geometry.
7.	JJ(106) = KPRINT	*Geometry List Option*
	= 0	Do not list lattice coordinates.
	= 1	List lattice coordinates.

8.	JJ(107) = KPUNCH	*Geometry Punch Option*
	= 0	Do not punch lattice coordinate deck.
	= 1	Punch lattice coordinate deck.

Rod Specification Data

9.	JJ(111) = NTYPE	Number of rod types in lattice ($\leq$50).
10.	JJ(112) = NKIND	Number of rod kinds in lattice ($\leq$20).
11.	JJ(115)••• = NCASE(I)	Case identification number from which rod parameters are to be derived for each rod kind ($1 \leq I \leq$ NKIND) (enter zero if conversion is not to be performed for a rod kind I).
12.	JJ(135)••• = NLB(I)	Relative number of rods of each rod kind in lattice ($1 \leq I \leq$ NKIND).
13.	JJ(175)••• = KIND(I)	Rod kind associated with each rod type ($1 \leq I \leq$ NTYPE).
14.	JJ(225) = LCRIT	*Criticality Search Control*
	= 0	No criticality search.
	= 1	Search for criticality over thermal η parameter.
	= 2	Search for criticality over thermal f parameter.
	= 3	Search for criticality over all η parameters.
	= 4	Search for criticality over axial buckling.
15.	JJ(226) = KSERCH	Rod kind to be searched over in criticality search.

Miscellaneous Control Data

16.	JJ(231) = N2	Maximum number of inner iterations on eigenvalues and eigenvectors (set equal to 200 if left blank).
17.	JJ(232) = N3	Maximum number of iterations for outer eigenvalue iterations (set equal to 25 if left blank).

18. JJ(233) = N4 — Maximum number of iterations for criticality searches (set equal to 25 if left blank).

Relative locations for floating point data are contained in the following list.

Data for Geometry Generators

(Items 1-3 refer to finite lattices only.)

1. BB(1)••• = XØ(I,1)
BB(51)••• = XØ(I,2)
BB(101)••• = XØ(I,3) — Initial values of X coordinates for rods in finite geometry generators. ($1 \leqslant I \leqslant$ NTYPE). See Notes a and b.

2. BB(151)••• = YØ(I,1)
BB(201)••• = YØ(I,2)
BB(251)••• = YØ(I,3) — Initial values of Y coordinates for rods in finite geometry generators. ($1 \leqslant I \leqslant$ NTYPE). See Notes a and b.

3. BB(301)••• = THETA(I,1)
BB(351)••• = THETA(I,2)
BB(401)••• = THETA(I,3) — Angular separation between rods on same radius about axes of reactor (enter only if LAT = 2) ($1 \leqslant I \leqslant$ NTYPE). See Note b.

(Items 4 and 5 refer to finite reflected lattices only.)

4. BB(452) = RØUT — Reflector outer radius (cm).

5. BB(453) = RIN — Radius within which image rod effects may be ignored (cm).

(Items 6-10 refer to infinite lattices only.)

6. BB(454) = RADII — Practical outer radius of infinite lattice (cm).

7. BB(451) = ANG — Acute angle between X and Y axes (deg).

8. BB(455)••• = XPITCH(I) — X pitch for I^{th} overlapping lattice. ($1 \leqslant I \leqslant$ NTYPE). See Note b.

9.	BB(505)••• = YPITCH(I)	Y pitch for I^{th} overlapping lattice. ($1 \leqslant I \leqslant$ NTYPE). See Note b.
10.	BB(555)••• = XI(I)	X coordinate of rod nearest origin in I^{th} overlapping lattice. ($1 \leqslant I \leqslant$ NTYPE). See Note b.
11.	BB(605)••• = YI(I)	Y coordinate of rod nearest origin in I^{th} overlapping lattice. ($1 \leqslant I \leqslant$ NTYPE). See Note b.
12.	BB(655) = XCØR	Arbitrary scale factor on all X coordinates. See Note b.
13.	BB(656) = YCØR	Arbitrary scale factor on all Y coordinates. See Note b.
14.	BB(657) = XSCALE	Plotting interval along X axis in geometry plotter (if zero is set equal to XCØR).
15.	BB(658) = YSCALE	Plotting interval along Y axis in geometry plotter (if zero is set equal to YCØR).

(Items 15 and 16 refer to the KØØR = 2 option only.)

16.	BB(1151)••• = X(I)	X coordinates for all rods (cm). $\sum_{I=1}^{NTYPE}$ NRPRT(I) values See Note c.
17.	BB(6151)••• = Y(I)	Y coordinates for all rods (cm). $\sum_{I=1}^{NTYPE}$ NRPRT(I) values See Note c.

Data for Kernel Generators

18.	BB(661) = BSQ	Axial geometrical buckling ($1/cm^2$).
19.	BB(725)••• = AR(I,J)	Resonance absorption parameter for each rod kind and each resonance level ($1 \leqslant I \leqslant$ NKIND: $1 \leqslant J \leqslant 3$).

Nuclear Data for Individual Rods

20.	BB(905)••• = THERU(I)	Thermal utilization for each rod kind. ($1 \leqslant I \leqslant$ NKIND).
21.	BB(925)••• = ETAIN(I,J)	Eta value for each rod kind and each resonance level. ($1 \leqslant I \leqslant$ NKIND; $1 \leqslant J \leqslant 4$). (ETAIN (I, J_{max}) values are thermal values).
22.	BB(1125)••• = VCELL(I)	Cell volume for each rod kind. ($1 \leqslant I \leqslant$ NKIND).

Data for Criticality Searches

23.	BB(1145) = EIGEN	Eigenvalue to be searched for (if zero is set equal to 1.0).

Convergence Criteria

24.	BB(1146) = CR1	Convergence limit on matrix inversion (if zero is set equal to .000001)
25.	BB(1147) = CR2	Convergence limit on inner eigenvalue and eigenvector iteration (if zero is set equal to .001).
26.	BB(1148) = CR3	Convergence limit on outer eigenvalue iterations (if zero is set equal to .001)
27.	BB(1149) = CR4	Convergence limit on criticality searches (if zero is set equal to .001).

NOTES FOR HERESY DATA LOCATIONS

NOTE a

Entering the initial coordinates XØ(I,J), YØ(I,J), and THETA(I,J) for the finite lattice generator must be done in a special order. The matrices for these quantities are 50 by 3 (50 rows and 3 columns). Data are entered column-wise, which implies that for each value of J the index I runs through its full range. Hence, using XØ as an example, the values would be entered as follows:

XØ(1,1)•••, XØ(I,1)•••, XØ(50,1), XØ(1,2)•••, XØ(I,2)•••,

XØ(50,2), XØ(1,3)•••, XØ(I,3)•••, XØ(50,3).

Relative locations associated with these coordinates are:

BB(1) = XØ(1,1)	BB(151) = YØ(1,1)	BB(301) = THETA(1,1)
BB(51) = XØ(1,2)	BB(201) = YØ(1,2)	BB(351) = THETA(1,2)
BB(101) = XØ(1,3)	BB(251) = YØ(1,3)	BB(401) = THETA(1,3)

Note b

The scale factors XCØR and YCØR are used as arbitrary non-zero, non-negative multiplication factors to modify coordinate values input. These factors do not multiply all values, but only those indicated below:

XCØR multiplies the values of XØ(I,J), XPITCH(I), XI(I).

YCØR multiplies the values of YØ(I,J), YPITCH(I), YI(I).

If XCØR and/or YCØR is left blank or is entered negative, it is automatically set equal to 1.0.

Note c

Coordinates are input in sequence starting with rod type 1 and running over all rods of that type. These are followed by rod types 2, 3, 4, •••, where, for each rod type, the coordinates of all rods of that type are successively entered.

Note d

Entering the doubly indexed rod parameters must be done in the correct order, else a programmed EXIT will occur. Matrices for these parameters are dimensioned at 20 rows and 4 columns. Data are always entered column-wise which implies that the first index runs through its full range before the second index is incremented. Using the ETAIN array as an example, the parameters are entered as follows:

ETAIN(1,1)•••, ETAIN(I,1)•••, ETAIN(20,1), ETAIN(1,2)•••,

ETAIN(I,2)•••, ETAIN(I,4)•••, ETAIN(20,4).

Relative locations for these parameters are shown below:

ETAIN(1,1) = BB(925)	AR(1,1) = BB(725)
ETAIN(1,2) = BB(945)	AR(1,2) = BB(745)
ETAIN(1,3) = BB(965)	AR(1,3) = BB(765)
ETAIN(1,4) = BB(985)	AR(1,4) = BB(785)

4.3 HAMBUR 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 that 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.

Card types 12-14 are read by an input routine in HERESY. A gross check of the data is made before starting a problem; however, it is not possible to catch many of the errors that are possible. Constant monitoring during the program execution detects errors that occur. Valid error codes are listed in the following table.

Card types 15-20 are read by an input routine in BURNUP. Very little checking is done of these data. Most error conditions are diagnosed during program execution, and always lead to termination of the problem. Valid error codes are listed in the following table.

HAMBUR Error Stops

Program	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 ≠ NX (slabs)
	104	MESH IS BAD	Sum of mesh points ≠ NX (cylinders)
	105	UPSCAT IS OVER 3	More than 3 mixtures with field 8, card type 4 = 1
	106	POINTS 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
	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	Some is 118, except HAMLET library
	133	- None -	CINDER error 3 in Recycle Option
	134	"	CINDER error 4 in Recycle Option
	135	"	CINDER error 5 in Recycle Option

Program	Number	Output Message	Remarks
THERMOS	200	- None -	Same as 118
HAMLET	300	"	Same as 119
FLOG	400	"	No reactor lead card (type 6)
	401	"	First case cards out of order
	402	"	Region missing
	403	"	Lattice not on library
	405	"	Material ident <0
	406	"	Material ident not in built-in lib
	404	"	= FOG - N100 error stop

Program	Number	Reason for Error Condition	Action to be Taken by Code
HERESY	100	Data conversion is requested for a number of resonance levels not equal to 3.	Attempt next problem.
	110	A case is not in the lattice library.	Attempt next problem.
	210	The number of rods in the lattice exceeds 5000.	Exit machine.
	220	A machine branching error has occurred. Please re-submit problem.	Exit machine.
	230	The initial coordinates to the hexagonal lattice generator are in the wrong sector.	Exit machine.
	240	The intial coordinates to the square lattice generator are in the wrong sector.	Exit machine.
	310	A negative radial distance was found in tabulating the slowing down kernel. Check your geometry.	Exit machine.
	320	An error condition was found in the Bessel's function subroutine in the epithermal slowing down kernel tabulation. Check your geometry and/or kernel parameters.	Exit machine.
	330	A negative radial distance was encountered in tabulating the epithermal slowing down kernel. Check your geometry.	Exit machine.
	340	An error condition was found in the Bessel's function subroutine in tabulating the thermal diffusion kernel. Check your geometry and/or kernel parameters.	Exit machine.
	350	A negative radial distance was encountered in tabulating the thermal diffusion kernel. Check your geometry.	Exit machine.

Program	Number	Reason for Error Condition	Action to be Taken by Code
HERESY	360	The mesh spacing is too large. Check your geometry.	Exit machine.
	370	The mesh spacing is too small. Check your geometry.	Exit machine.
	410	The $(\underline{\underline{\Delta}}+\underline{\underline{D}})$ matrix failed to invert to the accuracy specified by CR1.	Attempt next problem.
	420	The $(\underline{\underline{\Delta}}+\underline{\underline{D}})$ matrix is singular. The problem is not well posed.	Attempt next problem.
	430	The $(\underline{\underline{I}}-\underline{\underline{R}})$ matrix failed to invert to the accuracy specified by CR1.	Attempt next problem.
	440	The $(\underline{\underline{I}}-\underline{\underline{R}})$ matrix is singular. The problem is not well posed.	Attempt next problem.
	450	The inner eigenvalue iterations have exceeded N2 iterations.	Attempt next problem.
	460	The outer eigenvalue iterations have exceeded N3 iterations.	Attempt next problem.
	510	The criticality search iterations have exceeded N4 iterations.	Attempt next problem.
	520	Criticality cannot be achieved with a positive axial buckling.	Attempt next problem.

Program	Number	Reason for Error
BURNUP	501	A burnup case is not on the Lattice Library Tape.
	502	A burnup case is not on the Reaction Rate data set.
	503	An internal error has occurred in data handling within the burnup sequence for the mixed lattice.
	504	A type 19 card has been read for a case with no type 17 or 18 card.
	505	A type 18 card has been read for a case with no type 17 card.
	506	An internal error has occurred in data handling in cycling the code sections.
	507	An internal error has occurred in separating cases in the mixed lattice.
	508	The number of burnup cases for the mixed lattice now exceeds 15.
	509	The number of rod kinds in the HERESY program supercell now exceeds 20.
	533	CINDER error 3 (see Reference 1)
	534	CINDER error 4 (see Reference 1)
	535	CINDER error 5 (see Reference 1)

5. OUTPUT

5.1 GENERAL

The output from a burnup problem can contain the results of lattice cell calculations, a reactor supercell calculation, neutron balance tables, and a variety of optional edits on the burnup history of each lattice. The output is produced in the following order.

- THERMOS and HAMLET program output for each lattice cell at start of burnup history.
- Reactor supercell output from HERESY program if used, and DIED program edits for all lattice cells as requested.
- Burnup program input summary.
- Output from THERMOS, HAMLET, HERESY, and DIED programs for each spectrum calculation for each lattice cell during the burnup history.
- Summary edit and optional edits of burnup history for each lattice cell.
- THERMOS, HAMLET, HERESY, and DIED program output for each lattice cell at end of burnup history.

The output information from the programs THERMOS, HAMLET, FLOG, and DIED is identical to that provided by the HAMMER System. The user is referred to Reference 6 for a complete description of the output from these programs. Output from the HERESY program is described in detail in Reference 12.

5.2 BURNUP PROGRAM OUTPUT

The options for editing the burnup history of a lattice cell are designed to provide as much or as little information as is required by the user. If no optional edits are requested, the only edit will be a summary edit giving the lattice cell material buckling; k_∞; k_{eff}; fuel averaged unit 1/v cross section; cell-averaged, four-group, absolute fluxes; and epithermal-to-thermal flux ratio as a function of time, exposure, and fraction of initial most abundant "fuel chain" isotope. If a reactor supercell calculation has been performed, k_∞ and k_{eff} values are of no meaning since they are taken from the lattice cell calculation. An example of this summary edit is shown in Figure 5.1.

TOTAL TIME (HR)	TOTAL BURNUP (MWD)	FRAC U 235 IN ASSEM	MATERIAL BUCKLING (BUCKS)	K (INF)	K (EFF)	FUEL AVG. UNIT XSECTION	CELL AVG. ABSOLUTE FLUXES GRP 1	GRP 2	GRP 3	GRP 4	FAST/SLOW FLUX RATIO
0.0	0.0	1.0000	0.004705	1.1350	1.1334	6.41439E-01	1.63347E 13	1.86614E 13	1.57738E 13	3.68145E 13	1.37907E 00
480.0	5.00	0.9877	0.003717	1.1060	1.1044	6.42014E-01	1.64239E 13	1.87633E 13	1.58599E 13	3.70155E 13	1.37907E 00
960.0	10.00	0.9755	0.002730	1.0770	1.0755	6.42593E-01	1.64936E 13	1.88429E 13	1.59272E 13	3.71725E 13	1.37907E 00
1140.0	11.87	0.9710	0.002359	1.0661	1.0646	6.42811E-01	1.68471E 13	1.94126E 13	1.61301E 13	3.72614E 13	1.40601E 00
1440.0	15.00	0.9636	0.002172	1.0607	1.0593	6.43176E-01	1.68254E 13	1.93875E 13	1.61092E 13	3.72132E 13	1.40601E 00
1920.0	20.00	0.9518	0.002249	1.0630	1.0615	6.43765E-01	1.67704E 13	1.93242E 13	1.60567E 13	3.70918E 13	1.40601E 00
2340.0	24.37	0.9416	0.002308	1.0647	1.0632	6.44283E-01	1.69542E 13	1.95520E 13	1.62476E 13	3.74542E 13	1.40849E 00
2400.0	25.00	0.9402	0.002316	1.0649	1.0635	6.44358E-01	1.69594E 13	1.95580E 13	1.62526E 13	3.74658E 13	1.40849E 00
2880.0	30.00	0.9286	0.002373	1.0666	1.0651	6.44955E-01	1.70502E 13	1.96627E 13	1.63397E 13	3.76663E 13	1.40849E 00
3360.0	35.00	0.9170	0.002419	1.0680	1.0665	6.45557E-01	1.71849E 13	1.98180E 13	1.64687E 13	3.79638E 13	1.40849E 00
3560.0	37.08	0.9122	0.002435	1.0684	1.0670	6.45809E-01	1.70511E 13	1.96780E 13	1.63713E 13	3.77059E 13	1.40828E 00
3840.0	40.00	0.9056	0.002455	1.0690	1.0675	6.46164E-01	1.70842E 13	1.97162E 13	1.64031E 13	3.77791E 13	1.40828E 00
4320.0	45.00	0.8943	0.002481	1.0698	1.0683	6.46775E-01	1.71378E 13	1.97781E 13	1.64545E 13	3.78976E 13	1.40828E 00
4800.0	50.00	0.8832	0.002496	1.0703	1.0688	6.47391E-01	1.68339E 13	1.93514E 13	1.64675E 13	3.81007E 13	1.38194E 00
5280.0	55.00	0.8720	0.002501	1.0704	1.0689	6.48012E-01	1.70634E 13	1.96152E 13	1.66920E 13	3.86202E 13	1.38194E 00
5760.0	60.00	0.8608	0.002496	1.0703	1.0688	6.48637E-01	1.73684E 13	1.99659E 13	1.69904E 13	3.93105E 13	1.38194E 00
6060.0	63.12	0.8538	0.002487	1.0701	1.0686	6.49030E-01	1.69358E 13	1.94871E 13	1.66018E 13	3.84818E 13	1.37792E 00
6240.0	65.00	0.8497	0.002480	1.0699	1.0684	6.49267E-01	1.69567E 13	1.95112E 13	1.66223E 13	3.85293E 13	1.37792E 00
6720.0	70.00	0.8389	0.002454	1.0691	1.0677	6.49901E-01	1.69643E 13	1.95199E 13	1.66297E 13	3.85465E 13	1.37792E 00
7200.0	75.00	0.8282	0.002418	1.0681	1.0666	6.50540E-01	1.69313E 13	1.94819E 13	1.65973E 13	3.84715E 13	1.37792E 00
7360.0	76.67	0.8246	0.002403	1.0677	1.0662	6.50754E-01	1.70434E 13	1.96290E 13	1.67401E 13	3.88393E 13	1.37522E 00
7680.0	80.00	0.8175	0.002371	1.0668	1.0653	6.51184E-01	1.70568E 13	1.96906E 13	1.67925E 13	3.89610E 13	1.37522E 00
8160.0	85.00	0.8069	0.002314	1.0652	1.0637	6.51832E-01	1.71815E 13	1.97881E 13	1.68757E 13	3.91540E 13	1.37522E 00
8640.0	90.00	0.7964	0.002247	1.0632	1.0618	6.52485E-01	1.72670E 13	1.98865E 13	1.69596E 13	3.93488E 13	1.37522E 00
8680.0	90.41	0.7955	0.002241	1.0631	1.0616	6.52540E-01	1.71542E 13	1.97758E 13	1.68862E 13	3.92331E 13	1.37170E 00
9120.0	95.00	0.7859	0.002169	1.0610	1.0595	6.53143E-01	1.72396E 13	1.98742E 13	1.69702E 13	3.94284E 13	1.37170E 00
9600.0	100.00	0.7755	0.002081	1.0585	1.0570	6.53805E-01	1.73344E 13	1.99835E 13	1.70636E 13	3.96452E 13	1.37170E 00

FIGURE 5.1 Summary Edit for Uniform Lattice Case 10

If the optional edit of four-group absolute fluxes is requested, the absolute fluxes that were used in the calculation at each space point in a lattice cell are printed as a function of time, exposure, fraction of initial most abundant fuel isotope, and few group. The times of edit are determined by the specified edit frequency and the spectrum renormalization times. An example of this edit is shown in Figure 5.2.

TOTAL TIME (HRS)	TOTAL BURNUP (MWD)	FRAC U 235 IN ASSEM	GRP	PT 1	PT 2	PT 3	PT 4
0.0	0.0	1.0000	1	2.58787E 13	2.53488E 13	2.39222E 13	2.21785E 13
			2	2.55595E 13	2.50638E 13	2.38060E 13	2.24225E 13
			3	1.52973E 13	1.53427E 13	1.54537E 13	1.55707E 13
			4	1.25424E 13	1.33323E 13	1.56914E 13	1.93710E 13
480.0	5.00	0.9877	1	2.60200E 13	2.54872E 13	2.40528E 13	2.22996E 13
			2	2.56991E 13	2.52007E 13	2.39360E 13	2.25449E 13
			3	1.53809E 13	1.54265E 13	1.55381E 13	1.56557E 13
			4	1.26108E 13	1.34051E 13	1.57771E 13	1.94768E 13
960.0	10.00	0.9755	1	2.61304E 13	2.55953E 13	2.41548E 13	2.23942E 13
			2	2.58081E 13	2.53076E 13	2.40376E 13	2.26406E 13
			3	1.54461E 13	1.54919E 13	1.56040E 13	1.57221E 13
			4	1.26644E 13	1.34619E 13	1.58440E 13	1.95594E 13
1140.0	11.87	0.9710	1	2.66546E 13	2.61155E 13	2.46644E 13	2.28868E 13
			2	2.65729E 13	2.60606E 13	2.47604E 13	2.33288E 13
			3	1.56395E 13	1.56863E 13	1.58005E 13	1.59209E 13
			4	1.24114E 13	1.32192E 13	1.56337E 13	1.94104E 13
1440.0	15.00	0.9636	1	2.66202E 13	2.60818E 13	2.46325E 13	2.28573E 13
			2	2.65386E 13	2.60269E 13	2.47284E 13	2.32986E 13
			3	1.56193E 13	1.56660E 13	1.57801E 13	1.59003E 13
			4	1.23953E 13	1.32021E 13	1.56135E 13	1.93853E 13
1920.0	20.00	0.9518	1	2.65333E 13	2.59966E 13	2.45522E 13	2.27827E 13
			2	2.64520E 13	2.59420E 13	2.46477E 13	2.32226E 13
			3	1.55684E 13	1.56149E 13	1.57286E 13	1.58484E 13
			4	1.23549E 13	1.31590E 13	1.55625E 13	1.93220E 13

FIGURE 5.2 Absolute Four-Group Fluxes for Uniform Lattice Case 10

The optional edit of cross sections is performed for each isotope specified on the type 16 card associated with a lattice case. The cross sections at times intermediate between cell spectrum calculations are obtained by fitting the cross sections at the times cell spectrum calculations are performed to a quadratic polynomial as a function of exposure. This fitting is performed for epithermal and thermal fission and absorption cross sections as a function of space point in the cell. The pointwise cross sections are reduced to regionwise and assembly averaged cross sections by flux-volume weighting. One-group thermal effective cross sections are obtained using epithermal-to-thermal flux ratio at each space point. The edit provides the epithermal, thermal, and effective one-group regionwise and assembly averaged fission and absorption cross sections for an isotope as a function of time and exposure. An example of this edit is shown in Figure 5.3.

TOTAL TIME (HRS)	TOTAL BURNUP (MWD)	GRP	CROSS SECTION (BARNS) REGN 1 FISS	REGN 1 ABS	REGN 2 FISS	REGN 2 ABS	ASSEM AVG FISS	ASSEM AVG ABS
0.0	0.0	F	8.0	12.8	6.4	13.7	8.1	13.1
		S	600.0	910.8	650.2	975.6	619.1	935.5
		EFF	633.0	964.0	676.8	1018.7	649.7	984.9
480.0	5.00	F	7.9	12.8	8.4	13.7	8.1	13.1
		S	598.4	907.9	649.5	974.3	618.2	933.5
		EFF	631.5	961.2	676.1	1017.4	648.7	982.9
960.0	10.00	F	7.9	12.8	8.4	13.7	8.1	13.0
		S	596.9	905.1	648.9	973.0	617.2	931.7
		EFF	630.0	958.5	675.5	1016.2	647.8	981.1
1140.0	11.87	F	7.9	12.8	8.4	13.6	8.1	13.0
		S	596.3	904.1	648.6	972.5	616.9	931.0
		EFF	629.5	957.5	675.3	1015.7	647.4	980.4
1440.0	15.00	F	7.9	12.7	8.4	13.6	8.1	13.0
		S	595.4	902.5	648.2	971.7	616.3	929.8
		EFF	628.6	955.9	674.9	1014.9	646.9	979.2
1920.0	20.00	F	7.9	12.7	8.4	13.6	8.0	13.0
		S	594.1	900.0	647.6	970.5	615.4	928.1
		EFF	627.2	953.4	674.2	1013.7	646.0	977.5
2340.0	24.37	F	7.9	12.7	8.4	13.6	8.0	13.0
		S	592.9	897.8	647.0	969.4	614.6	926.6
		EFF	626.1	951.3	673.7	1012.7	645.2	975.9
2400.0	25.00	F	7.9	12.7	8.4	13.6	8.0	13.0
		S	592.7	897.6	647.0	969.3	614.5	926.4
		EFF	625.9	951.0	673.6	1012.5	645.1	975.7
2880.0	30.00	F	7.9	12.7	8.4	13.6	8.0	13.0
		S	591.5	895.3	646.4	968.1	613.7	924.7
		EFF	624.7	948.7	673.0	1011.3	644.3	974.0

FIGURE 5.3 Cross Section Edit for ^{239}Pu for Uniform Lattice Case 10

The optional concentration edit provides, for each region, the concentration (in atm/bn-cm) of each isotope specified on the type 16 card associated with a lattice case as a function of time and exposure. Fission product and added isotope concentrations are obtained by fitting the concentration at each spectrum calculation to a quadratic polynomial as a function of exposure. The times for editing are selected from the edit frequency specified and the spectrum renormalization times. An example is shown in Figure 5.4.

TOTAL TIME (HRS)	TOTAL BURNUP (MWD)	U 235	U 236	U 238	PU 239	PU 240
0.0	0.0	9.69127E-04	1.00000E-15	4.68886E-02	1.00000E-15	1.00000E-15
480.0	5.00	9.54186E-04	2.32754E-06	4.68818E-02	4.89685E-06	2.67707E-08
960.0	10.00	9.39412E-04	4.62654E-06	4.68750E-02	1.06317E-05	1.17663E-07
1140.0	11.87	9.33915E-04	5.48124E-06	4.68724E-02	1.27389E-05	1.67985E-07
1440.0	15.00	9.24883E-04	6.88605E-06	4.68682E-02	1.61288E-05	2.69400E-07
1920.0	20.00	9.10642E-04	9.09923E-06	4.68616E-02	2.13374E-05	4.76147E-07
2340.0	24.37	8.98391E-04	1.10009E-05	4.68558E-02	2.57166E-05	6.99028E-07
2400.0	25.00	8.96635E-04	1.12732E-05	4.68549E-02	2.63335E-05	7.34089E-07
2880.0	30.00	8.82674E-04	1.34378E-05	4.68481E-02	3.12842E-05	1.04229E-06
3360.0	35.00	8.68826E-04	1.55821E-05	4.68413E-02	3.61439E-05	1.39880E-06
3560.0	37.08	8.63082E-04	1.64705E-05	4.68384E-02	3.81454E-05	1.56129E-06
3840.0	40.00	8.55167E-04	1.76537E-05	4.68344E-02	4.08598E-05	1.80003E-06
4320.0	45.00	8.41729E-04	1.97688E-05	4.68276E-02	4.53600E-05	2.24212E-06
4800.0	50.00	8.28450E-04	2.18167E-05	4.68208E-02	4.97192E-05	2.72352E-06
5280.0	55.00	8.15240E-04	2.38517E-05	4.68137E-02	5.41105E-05	3.24673E-06
5760.0	60.00	8.02024E-04	2.58849E-05	4.68064E-02	5.85187E-05	3.81458E-06
6060.0	63.12	7.93740E-04	2.71578E-05	4.68018E-02	6.12846E-05	4.19312E-06
6240.0	65.00	7.88930E-04	2.78563E-05	4.67991E-02	6.28729E-05	4.42225E-06
6720.0	70.00	7.76231E-04	2.98445E-05	4.67921E-02	6.68408E-05	5.05486E-06
7200.0	75.00	7.63739E-04	3.17584E-05	4.67852E-02	7.06184E-05	5.71510E-06
7360.0	76.67	7.59626E-04	3.23880E-05	4.67828E-02	7.18336E-05	5.94056E-06
7680.0	80.00	7.51344E-04	3.36543E-05	4.67781E-02	7.42934E-05	6.40630E-06
8160.0	85.00	7.39034E-04	3.55345E-05	4.67709E-02	7.79374E-05	7.12805E-06
8640.0	90.00	7.26858E-04	3.73920E-05	4.67637E-02	8.14728E-05	7.87605E-06
8680.0	90.41	7.25850E-04	3.75457E-05	4.67631E-02	8.17625E-05	7.93951E-06
9120.0	95.00	7.14836E-04	3.92228E-05	4.67565E-02	8.48971E-05	8.65929E-06
9600.0	100.00	7.02946E-04	4.10310E-05	4.67493E-02	8.82148E-05	9.47289E-06

FIGURE 5.4 Concentrations (atm/bn-cm) in Region 2 for Uniform Lattice Case 10

The optional mass balance edit provides a detailed picture of the production and depletion of each isotope specified on the type 16 card for a lattice case itemized by the four reactions capture, fission, alpha decay, and beta decay. Edit times are selected from the specified edit frequency and the spectrum renormalization times. The total mass of each isotope is expressed in grams of material per axial length in feet as a convenient engineering unit. At each edit time a table is given for the production of the isotope by neutron capture, alpha decay, and beta decay since the last edit time. A similar table is given for the depletion of the isotope by neutron capture and fission, alpha decay, and beta decay. Finally, the current mass per unit of length is given for the time specified. An example of this edit is shown in Figure 5.5.

REGIONS 1 & 2 AND ASSEMBLY TOTAL

TOTAL TIME (HRS)	TOTAL BURNUP (MWD)	REGION 1 PRODUCTION REACTION	GMS/FT	REGION 1 DEPLETION REACTION	GMS/FT	REGION 2 PRODUCTION REACTION	GMS/FT	REGION 2 DEPLETION REACTION	GMS/FT	ASSEMBLY TOTAL PRODUCTION REACTION	GMS/FT	ASSEMBLY TOTAL DEPLETION REACTION	GMS/FT
0.0	0.0	BALANCE IN REGION	0.000 GM/FT			BALANCE IN REGION	0.000 GM/FT			BALANCE IN REGION	0.000 GM/FT		
480.0	5.00	CAPT	0.000	CAPT	0.000	CAPT	0.000	CAPT	0.000	CAPT	0.000	CAPT	0.001
		ALPHA	0.0	FISS	0.001	ALPHA	0.0	FISS	0.001	ALPHA	0.0	FISS	0.002
		BETA	0.128	ALPHA	0.0	BETA	0.004	ALPHA	0.0	BETA	0.213	ALPHA	0.0
				BETA	0.0			BETA	0.0			BETA	0.0
		TOTAL	0.128	TOTAL	0.001	TOTAL	0.004	TOTAL	0.001	TOTAL	0.213	TOTAL	0.003
		BALANCE IN REGION		0.127 GM/FT		BALANCE IN REGION		0.083 GM/FT		BALANCE IN REGION		0.210 GM/FT	
960.0	10.00	CAPT	0.000	CAPT	0.002	CAPT	0.000	CAPT	0.002	CAPT	0.000	CAPT	0.003
		ALPHA	0.0	FISS	0.003	ALPHA	0.0	FISS	0.003	ALPHA	0.0	FISS	0.006
		BETA	0.155	ALPHA	0.0	BETA	0.102	ALPHA	0.0	BETA	0.257	ALPHA	0.0
				BETA	0.0			BETA	0.0			BETA	0.0
		TOTAL	0.155	TOTAL	0.005	TOTAL	0.102	TOTAL	0.004	TOTAL	0.257	TOTAL	0.010
		BALANCE IN REGION		0.276 GM/FT		BALANCE IN REGION		0.181 GM/FT		BALANCE IN REGION		0.457 GM/FT	
1140.0	11.87	CAPT	0.000	CAPT	0.001	CAPT	0.000	CAPT	0.001	CAPT	0.000	CAPT	0.002
		ALPHA	0.0	FISS	0.002	ALPHA	0.0	FISS	0.002	ALPHA	0.0	FISS	0.004
		BETA	0.058	ALPHA	0.0	BETA	0.038	ALPHA	0.0	BETA	0.097	ALPHA	0.0
				BETA	0.0			BETA	0.0			BETA	0.0
		TOTAL	0.058	TOTAL	0.003	TOTAL	0.038	TOTAL	0.003	TOTAL	0.097	TOTAL	0.006
		BALANCE IN REGION		0.331 GM/FT		BALANCE IN REGION		0.217 GM/FT		BALANCE IN REGION		0.548 GM/FT	
1440.0	15.00	CAPT	0.000	CAPT	0.002	CAPT	0.000	CAPT	0.002	CAPT	0.000	CAPT	0.004
		ALPHA	0.0	FISS	0.004	ALPHA	0.0	FISS	0.003	ALPHA	0.0	FISS	0.008
		BETA	0.124	ALPHA	0.0	BETA	0.063	ALPHA	0.0	BETA	0.187	ALPHA	0.0
				BETA	0.0			BETA	0.0			BETA	0.0
		TOTAL	0.124	TOTAL	0.006	TOTAL	0.063	TOTAL	0.005	TOTAL	0.187	TOTAL	0.011
		BALANCE IN REGION		0.449 GM/FT		BALANCE IN REGION		0.274 GM/FT		BALANCE IN REGION		0.723 GM/FT	
1920.0	20.00	CAPT	0.000	CAPT	0.005	CAPT	0.000	CAPT	0.004	CAPT	0.000	CAPT	0.008
		ALPHA	0.0	FISS	0.009	ALPHA	0.0	FISS	0.007	ALPHA	0.0	FISS	0.017
		BETA	0.227	ALPHA	0.0	BETA	0.099	ALPHA	0.0	BETA	0.326	ALPHA	0.0
				BETA	0.0			BETA	0.0			BETA	0.0
		TOTAL	0.227	TOTAL	0.014	TOTAL	0.099	TOTAL	0.011	TOTAL	0.326	TOTAL	0.025
		BALANCE IN REGION		0.661 GM/FT		BALANCE IN REGION		0.363 GM/FT		BALANCE IN REGION		1.024 GM/FT	
2340.0	24.37	CAPT	0.000	CAPT	0.006	CAPT	0.000	CAPT	0.004	CAPT	0.000	CAPT	0.010
		ALPHA	0.0	FISS	0.011	ALPHA	0.0	FISS	0.008	ALPHA	0.0	FISS	0.019
		BETA	0.214	ALPHA	0.0	BETA	0.086	ALPHA	0.0	BETA	0.300	ALPHA	0.0
				BETA	0.0			BETA	0.0			BETA	0.0
		TOTAL	0.214	TOTAL	0.017	TOTAL	0.086	TOTAL	0.012	TOTAL	0.300	TOTAL	0.029
		BALANCE IN REGION		0.858 GM/FT		BALANCE IN REGION		0.437 GM/FT		BALANCE IN REGION		1.296 GM/FT	

FIGURE 5.5 Material Balance for ^{239}Pu for Uniform Lattice Case 10

The optional isotopic power edit breaks down the power in each "fuel region" of the lattice cell into the four components of fission power, fission product decay power, radiative capture power, and kinetic heating (called inelastic scattering). The fission power is further broken down into isotopic fission power for each "fuel chain" isotope specified on the type 16 card for the lattice case. These powers are given as a function of time and exposure for each "fuel region" and for the full assembly. An example is shown in Figure 5.6.

TOTAL TIME (HRS)	TOTAL BURNUP (MWD)	ISOTOPE	REGION 1 POWER IN MW FROM FISSION	F.P. DECAY	RAD. CAP.	INEL SCAT	REGION 2 POWER IN MW FROM FISSION	F.P. DECAY	RAD. CAP.	INEL SCAT
0.0	0.0	ALL	0.157	0.000	0.002	0.004	0.082	0.000	0.001	0.002
		U 235	0.145				0.078			
		U 236	0.0				0.0			
		U 238	0.012				0.004			
		PU 239	0.000				0.000			
		PU 240	0.0				0.0			
		PU 241	0.000				0.000			
480.0	5.00	ALL	0.157	0.000	0.003	0.004	0.082	0.000	0.001	0.002
		U 235	0.144				0.077			
		U 236	0.0				0.0			
		U 238	0.012				0.004			
		PU 239	0.001				0.001			
		PU 240	0.0				0.0			
		PU 241	0.000				0.000			
960.0	10.00	ALL	0.157	0.001	0.003	0.004	0.082	0.000	0.001	0.002
		U 235	0.143				0.076			
		U 236	0.0				0.0			
		U 238	0.012				0.004			
		PU 239	0.002				0.002			
		PU 240	0.0				0.0			
		PU 241	0.000				0.000			
1140.0	11.87	ALL	0.157	0.001	0.003	0.004	0.082	0.000	0.001	0.002
		U 235	0.142				0.076			
		U 236	0.0				0.0			
		U 238	0.012				0.004			
		PU 239	0.002				0.002			
		PU 240	0.0				0.0			
		PU 241	0.000				0.000			
1440.0	15.00	ALL	0.157	0.001	0.003	0.004	0.082	0.000	0.001	0.002
		U 235	0.141				0.075			
		U 236	0.0				0.0			
		U 238	0.013				0.004			
		PU 239	0.003				0.002			
		PU 240	0.0				0.0			
		PU 241	0.000				0.000			
1920.0	20.00	ALL	0.157	0.001	0.003	0.004	0.081	0.000	0.001	0.002
		U 235	0.139				0.074			
		U 236	0.0				0.0			
		U 238	0.014				0.004			
		PU 239	0.004				0.003			
		PU 240	0.0				0.0			
		PU 241	0.000				0.000			
2340.0	24.37	ALL	0.157	0.001	0.003	0.004	0.082	0.000	0.001	0.002
		U 235	0.139				0.074			
		U 236	0.0				0.0			
		U 238	0.012				0.004			
		PU 239	0.006				0.004			
		PU 240	0.0				0.0			
		PU 241	0.000				0.000			

FIGURE 5.6 Isotope Power Edit for Uniform Lattice Case 10

The optional fission product cooling edit is obtained by specifying the number of months (<18) over which the edit is desired. The edit provides the total activity, total fission product decay power, and the concentration (in atm/bn-cm) of eight standard and one optional fission product as a function of time (in weeks) after shutdown. An example is shown in Figure 5.7.

TIME AFTER SHUTDOWN (WEEKS)	TOTAL ACTIVITY	TOTAL F.P. DECAY HEAT (MW)	FISSION PRODUCT CONCENTRATIONS							
			ZR 95	NB 95	RU 103	RU 106	CS 134	CS 137	CE 144	CS 137
1	3.4331E-12	4.3761E-03	3.2156E-06	1.8365E-06	1.1235E-06	1.5257E-06	1.5918E-07	3.2674E-06	8.4711E-06	3.2674E-06
3	2.2728E-12	4.2119E-03	2.7697E-06	1.7799E-06	8.7982E-07	1.4861E-06	1.5735E-07	3.2644E-06	8.1826E-06	3.2644E-06
5	1.7847E-12	4.1314E-03	2.3857E-06	1.6832E-06	6.8901E-07	1.4475E-06	1.5554E-07	3.2614E-06	7.9039E-06	3.2615E-06
7	1.4808E-12	2.9117E-04	2.0549E-06	1.5635E-06	5.3958E-07	1.4099E-06	1.5376E-07	3.2585E-06	7.6346E-06	3.2585E-06
9	1.2632E-12	2.4832E-04	1.7700E-06	1.4329E-06	4.2256E-07	1.3733E-06	1.5199E-07	3.2555E-06	7.3746E-06	3.2556E-06
11	1.0951E-12	2.1355E-04	1.5245E-06	1.2996E-06	3.3092E-07	1.3376E-06	1.5024E-07	3.2526E-06	7.1234E-06	3.2526E-06
13	9.5929E-13	1.8447E-04	1.3131E-06	1.1689E-06	2.5915E-07	1.3029E-06	1.4852E-07	3.2496E-06	6.8807E-06	3.2497E-06
15	8.4667E-13	1.5976E-04	1.1311E-06	1.0443E-06	2.0295E-07	1.2691E-06	1.4681E-07	3.2467E-06	6.6464E-06	3.2468E-06
17	7.5170E-13	1.3860E-04	9.7422E-07	9.2797E-07	1.5893E-07	1.2361E-06	1.4512E-07	3.2438E-06	6.4200E-06	3.2438E-06
19	6.7080E-13	1.2040E-04	8.3913E-07	8.2085E-07	1.2446E-07	1.2040E-06	1.4346E-07	3.2408E-06	6.2013E-06	3.2409E-06
21	6.0142E-13	1.0471E-04	7.2278E-07	7.2336E-07	9.7471E-08	1.1727E-06	1.4181E-07	3.2379E-06	5.9900E-06	3.2379E-06
23	5.4164E-13	9.1179E-05	6.2256E-07	6.3544E-07	7.6332E-08	1.1423E-06	1.4018E-07	3.2350E-06	5.7860E-06	3.2350E-06
25	4.8995E-13	7.9502E-05	5.3624E-07	5.5671E-07	5.9778E-08	1.1126E-06	1.3857E-07	3.2320E-06	5.5889E-06	3.2321E-06
27	4.4509E-13	6.9424E-05	4.6188E-07	4.8663E-07	4.6813E-08	1.0837E-06	1.3698E-07	3.2291E-06	5.3985E-06	3.2292E-06
29	4.0607E-13	6.0725E-05	3.9784E-07	4.2454E-07	3.6661E-08	1.0556E-06	1.3540E-07	3.2262E-06	5.2146E-06	3.2262E-06
31	3.7202E-13	5.3216E-05	3.4267E-07	3.6976E-07	2.8710E-08	1.0282E-06	1.3385E-07	3.2233E-06	5.0370E-06	3.2233E-06
33	3.4222E-13	4.6733E-05	2.9516E-07	3.2159E-07	2.2484E-08	1.0015E-06	1.3231E-07	3.2203E-06	4.8654E-06	3.2204E-06
35	3.1608E-13	4.1135E-05	2.5423E-07	2.7934E-07	1.7607E-08	9.7546E-07	1.3079E-07	3.2174E-06	4.6997E-06	3.2175E-06
37	2.9308E-13	3.6299E-05	2.1898E-07	2.4238E-07	1.3789E-08	9.5013E-07	1.2929E-07	3.2145E-06	4.5396E-06	3.2146E-06
39	2.7277E-13	3.2119E-05	1.8862E-07	2.1012E-07	1.0798E-08	9.2546E-07	1.2780E-07	3.2116E-06	4.3850E-06	3.2117E-06
41	2.5478E-13	2.8505E-05	1.6246E-07	1.8201E-07	8.4565E-09	9.0143E-07	1.2634E-07	3.2087E-06	4.2356E-06	3.2088E-06
43	2.3880E-13	2.5378E-05	1.3993E-07	1.5754E-07	6.6225E-09	8.7802E-07	1.2488E-07	3.2058E-06	4.0914E-06	3.2058E-06
45	2.2456E-13	2.2671E-05	1.2053E-07	1.3629E-07	5.1862E-09	8.5522E-07	1.2345E-07	3.2029E-06	3.9520E-06	3.2029E-06
47	2.1181E-13	2.0324E-05	1.0382E-07	1.1783E-07	4.0615E-09	8.3301E-07	1.2203E-07	3.2000E-06	3.8174E-06	3.2000E-06
49	2.0037E-13	1.8288E-05	8.9422E-08	1.0183E-07	3.1806E-09	8.1138E-07	1.2063E-07	3.1971E-06	3.6873E-06	3.1971E-06
51	1.9005E-13	1.6519E-05	7.7023E-08	8.7966E-08	2.4909E-09	7.9031E-07	1.1924E-07	3.1942E-06	3.5617E-06	3.1943E-06

FIGURE 5.7 Fission Product Cooling Edit for Uniform Lattice Case 10

6. TAPE LIBRARIES AND DATA SETS

6.1 LATTICE LIBRARY TAPE

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 HAMBUR run. The information contained in this section is primarily reference material, which will be required in the writing of special edit 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).

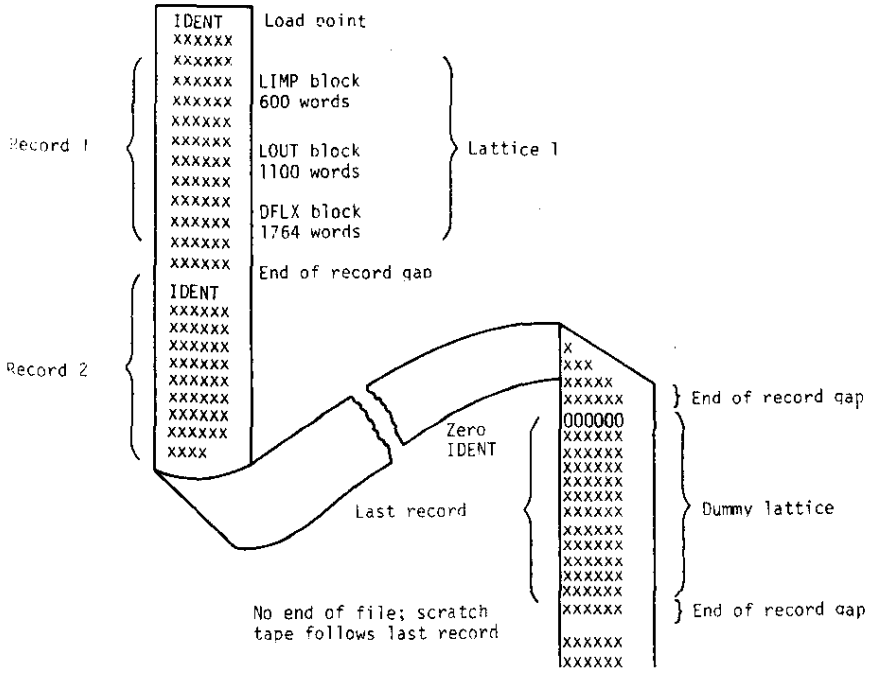


FIGURE 6.1 Lattice Library Tape Format

Lattice Input Data Block - LIMP

Relative Location	Symbol	Description
1	IDENT	Case ident number
2	NX	Total number of mesh point $\leqslant 20$
3	MX	Total number of mixtures $\leqslant 10$
4	NRX	Total number of regions $\leqslant 20$
5	ISØX	Total number of isotopes $\leqslant 18$
6	ISØXE	Total number of foil isotopes, ISØX + ISØXE $\leqslant$ 18
7	NGEØM	1 = slab, 2 = hexagonal array, 3 = square array
8	NP1B1	0 = P_1 approx, 1 = B_1 approx
9	N4FT	0 = flux ratios at B_m^2 1 = flux ratios at input leakage B^2
10		
11	BSQD	Input leakage buckling (cm^{-2})
12	HOL(9)	Case title, 36 Hollerith characters
21	NPT(20)	Number of mesh points in each region
41	THT(20)	Thickness (cm) of each region
61	MXAS(20)	Mixture assigned to each region
81	ILHI(20)	0 = treat mixture as a heavy scatterer 1 = treat mixture as a light scatterer
101	NRBP(20)	Number of the last mesh point in each region
121	DEGC(20)	Temperature (°C) of each region
141	WSTBA(18)	Isotope ident numbers

Relative Location	Symbol	Description
159	WSTBB(18)	Temperature index for isotope
177	CØNCTA(18,10)	CØNCTA(I,J) is the concentration (atm/bn-cm) of the I isotope in mixture J
357	LIMT(10)	Location in array P of the kernel associated with each mixture
367	RIN(20)	Inner radius (cm) of each sub-region (mesh point)
387	RAD(20)	Central radius (cm) of each sub-region (mesh point)
407	RØUT(20)	Outer radius (cm) of each sub-region (mesh point)
427	VØL(20)	Volume (cc) of each subregion (mesh point)
447	MTBL(20)	Mixture associated with each mesh point
467	REGV(20)	Volume (cc) of each region
487	FISD(20)	Fission neutrons/cc in each region

Lattice Output Data Block - LOUT

Relative Location	Symbol	Description
1	ID	Lattice ident number
2	HØLA(9)	Date (A6,A2), time, THERMOS library label (3A6), HAMLET library label (3A6)
11	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	FØIL(20,5,4)	FOIL(I,J,K) is the activity of foil J in lattice region I in group K
771	FØGX(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	FLX(20,4)	FLX(I,J) is the average flux in group J at lattice mesh point I
875	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^2_m 8 = K_∞ 9 = eigenvalue at B^2 10 = ψ_1 at B^2_m 11 = ψ_2 at B^2_m 12 = ψ_3 at B^2_m 13 = ψ_4 at B^2_m
895	ØTGC(5)	Overlapping thermal group arrays
900	ØTGA(20,6)	

Detailed Flux Block - DFLX

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

6.2 BURNUP HISTORY TAPE

The Burnup History Tape (BHT) is designed to store as much information as possible for further utilization in production codes, for further editing of results, and to provide a means of storing the results of lengthy calculations for common usage. The information includes the Lattice Library Tape blocks, absolute fluxes, cross sections, reaction rates, logical information concerning the cases, and selected spectral data. The tape may be further utilized to re-enter information into HAMBUR for recycling of assemblies.

The tape contains four types of records identified by numbers 1 to 4. Each record has its record type as the first datum in the record. All records are 6700 single precision words in length. The four record types contain the following information.

Record Type 1

Identification record - contains signals to indicate if tape contains a uniform lattice history or reactor supercell history. If the tape is for a supercell history, further type 1 records are given that contain the HERESY input blocks and calculated output.

Record Type 2

Flux, Cross Section, and Reaction Rate Record - contains data at each edit point of the burnup run on fluxes, cross sections, reaction rates, and other miscellaneous data. At each edit point and each spectrum recalculation point a type 2 record is output for each case run.

Record Type 3

Fission Product Record - contains quadratic fitted coefficients for each fission product carried by CINDER as of function of exposure in MWD, and fuel tube in assembly.

Record Type 4

Fission Product Cooling Record - contains total activity, total decay heat, and fission product concentrations for a 75-week period following shutdown.

Detailed descriptions of the data contained in each record are given below. Each record type contains its type number as the first word of the record except for the last record on the tape which contains a negative number in a dummy record.

RECORD TYPE 1

Burnup History Tape Ident Record

Relative Location	Mnemonic	Description
1	KREC	Always = 1
2	ICØDE	= 0 for uniform lattice tape = 1 for supercell history tape
3	ICTMAX	If ICØDE = 1, this gives the number of supercell reactor solution blocks to follow. If ICØDE = 0, no blocks follow and this is ignored
	(Items 4-6 present only if ICØDE = 1)	
4	NCASES	Number of unique mixed lattice cases for edit
5	NREDUN(I,J)	Rod Type Numbers (J) for each unique rod case I in edit ($I \leq 15$, $J \leq 20$)
305	NØREDN(I)	Number of equivalent rod types for edit Case I ($I \leq 15$)

RECORD TYPE 1a

Supercell Reactor Solution Record

Relative Location	Mnemonic	Description
1	KREC	Always = 1
2	XTIM	Maximum time (hr) for which this reactor solution is valid
3	XMWD	Maximum burnup (MWD) for which this reactor solution is valid
4	XPOW(I)	Power for each depletion Case I at start of interval
19	ABLK(18)	Reactor Case Title (alphanumeric)
37	IMP(255)	Reactor fixed point input
292	XIMP(1150)	Reactor floating point input
1442	XKEF	k_{eff} from reactor solution
1443	ABSRB(I,J)	Absorption in the I^{th} rod type and J^{th} energy group ($1 \leq I \leq 50$), ($1 \leq J \leq 4$)

RECORD TYPE 2

Case EDIT Record

Relative Location	Mnemonic	Description
1	KREC	Always = 2
2	NCSID	Case ident
3	PWRASS	Assembly input power, or maximum assembly power for supercell
4	PØWER	Actual power (MW) at this edit point
5	FISSP	Total fission power (MW) at this edit point
6	UNIS	Fuel avg unit cross section (barns)
7	TØTDEL	Total time (hr) from start of burnup
8	TØTMWD	Total burnup (MWD) since start of burnup
9	FRAC	Fraction of initially most abundant fissile element remaining at this edit point
10	LATBLK(3464)	Lattice Library Block at this point
3474	TØTRTS(I,J,K)	Total events of type I occurring in fuel isotope J in K^{th} fuel region since last edit I = 1 - absorption I = 2 - fission I = 3 - beta decay I = 4 - alpha decay
3858	TRABS(I,J,K)	Total events of type I occurring in added isotope J in K^{th} region since last edit I = 1 - absorption I = 2 - alpha or beta decay
3894	REAT(I,J,K)	Absolute reaction rate of I^{th} type for J^{th} fuel isotope at K^{th} space point I = 1 - absorption I = 2 - fission I = 3 - alpha decay I = 4 - beta decay

Relative Location	Mnemonic	Description
5174	CØNC(I,J)	Concentration (atm/bn-cm) of the I^{th} fuel isotope at J^{th} space point
5494	ACØNC(I,J)	Concentration (atm/bn-cm) of the I^{th} added isotope at J^{th} space point
5554	PHI(I,J)	Absolute flux (n/cm^2-sec) for I^{th} group at J^{th} space point
5634	FISSC(I,J)	Total cumulative fission in I^{th} fuel isotope at J^{th} space point (events)
5954	FSECT(I,J,K,L)	Microscopic cross section of K^{th} type for I^{th} fuel isotope in J^{th} group and L^{th} fuel region (barns) J = 1 - fast K = 1 - absorption J = 2 - thermal K = 2 - fission J = 3 - effective
6510	ASECT(I,J,K)	Microscopic absorption cross section for I^{th} added isotope in J^{th} group and K^{th} region (barns) J = 1 - epithermal J = 2 - thermal J = 3 - effective
6564	SMXT(I,J)	Samarium absorption cross section in I^{th} group and J^{th} fuel region (barns)
6582	XEXT(I,J)	Xenon absorption cross section in I^{th} group and J^{th} fuel region (barns)
6600	XUNIT(I)	Unit cross section in I^{th} fuel region (barns)
6600	BPF(I,J)	Barns/fission of gross fission product in I^{th} group and J^{th} fuel region I = 1 - epithermal I - 2 - thermal

Relative Location	Mnemonic	Description
6618	TØTAC(I)	Total activations in I^{th} fuel region (events)
6624	RDKHT(I)	Fission product decay heat in I^{th} fuel region (MW)
6630	RCHT(I)	Radiative capture heat in I^{th} fuel region (MW)
6636	XISHR(I)	Kinetic heat in I^{th} fuel region (MW)
6642	NFUEL(I)	Fuel point identification = 0 - not fuel point = 1 - fuel point = 2 - target point
6662	MREGN(I)	Region numbers of I^{th} fuel and/or target region
6668	MMREGN(I)	Region numbers of I^{th} fuel only regions
6674	KREGN(I)	Region number of I^{th} added isotope region
6680	XLNGTH	Effective length of assembly (cm)
6681	IADD	Added isotope option
6682	KAM	= 0 fuel chain 1 = 1 fuel chain 2

RECORD TYPE 3

Fission Product Record[a]

Relative Location	Mnemonic	Description
1	KREC	Always = 3
2	NCSID	Case ident
3	FPCØEF(I,J,K)	Quadratic expansion coefficients for K^{th} fission product in J^{th} fuel region ($1 \leqslant I \leqslant 3$)
2703	LREGN(I)	Region number of I^{th} fuel region

a. Concentration of fission products are expanded as follows:

$$CT(K,J) = FPCØEF(1,J,K) + XMWD*(FPCØEF(2,J,K) + XMWD*FPCØEF(3,J,K)$$

where

CT(K,J) = the concentration of the K^{th} fission product in the J^{th} region (atm/cm^3)

XMWD = exposure, MWD

RECORD TYPE 4

Fission Product Cooling Record

Relative Location	Mnemonic	Description
1	KREC	Always = 4
2	NCSID	Case ident
3	TØTAC(I)	Total activity at I^{th} week after shutdown (events/sec)
78	TDKHT(I)	Total decay heat at I^{th} week after shutdown (MW)
153	SPISØC(I,J)	Concentration of I^{th} special isotope at J^{th} ($\leq$75) week after (atm/bn-cm)
903	ØNDC(I)	Concentration of I^{th} fission product after 75 weeks ($1 \leq I \leq 150$) (atm/bn-cm)
1053	MISØ	Fission product ident number of I^{th} fission product in SPISØC(I,J)

6.3 AUXILIARY PROGRAMS

The only auxiliary programs required by HAMBUR are those necessary to make and maintain the THERMOS program and HAMLET program libraries. These are the same auxiliary programs used by the HAMMER system and are designated the LITHE program and HELP programs, respectively. A complete description of these programs may be found in Reference 6.

6.4 DATA SETS USED IN HAMBUR

Use	Data Set Reference No.	Mnemonic Reference
Thermal Library	09	LIB1
Epithermal Library	10	LIB2
Lattice Library Tape	04	LIB3
Burnup History Input Tape	17	LIB4
Burnup History Output Tape	26	LIB5
Scratch	02,03,11,12,14, 15,16,18,25	INT1 - INT10, NSC1
Scratch (HERESY)	20,21,22,23,30, 31,32,33,40,41, 42,43,51,52	

6.5 COMMON CONTROL BLOCK

A block of blank common storage (designated CNTRL) is used to store various flags that control the sequence of calculation through HAMBUR. The contents of this block are described below.

Control Block - CNTRL

Relative Location	Symbol	Description
1	IDBCH	Batch ident number
2	NCASE	Number of lattices to be run
3	NFLØG	Number of reactor cases to be run
4	BTTL(9)	Batch title, 54 Hollerith characters
13	NTIN	System input tape number
14	NTØUT	System print output tape number
15	NTPUN	System punch output tape number
16	NCHAIN	HAMMER program tape number
17	LIB1	THERMOS library tape number
18	LIB2	HAMLET library tape number
19	LIB3	Lattice library tape number
20	INT1	Intermediate tape 1 number
21	INT2	Intermediate tape 2 number
22	NSC1	Scratch tape number
23	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 or HERESY Program 5 is DIED Program 6 is BURNUP
33	NTHCS	Number of THERMOS cases
34	NEPCS	Number of HAMLET cases

Relative Location	Symbol	Description
36	INT3	Intermediate tape 3 number
37	INT4	Intermediate tape 4 number
38	INT5	Intermediate tape 5 number
39	INT6	Intermediate tape 6 number
40	INT7	Intermediate tape 7 number
41	INT8	Intermediate tape 8 number
42	LIB4	Burnup History Input tape number
43	ICØUNT	Iteration counter through BURNUP section
44	TMZRØ	Starting time for BURNUP calculation
45	INT9	Intermediate tape 9 number
47	INT10	Intermediate tape 10 number
48	LIB6	Burnup History Output tape number
129	NCPRN	= 0, end job; = 1, run CAPN
130	NCPPT	CAPN long print?*
141	NTHRN	= 0, do not run THERMOS = 1, run standard THERMOS = 2, run cosine current THERMOS
142	NTHPT	THERMOS long print?
153	NEPRN	= 0, do not run HAMLET = 1, run HAMLET.
154	NEPPT	= 0, HAMLET short print = 1, HAMLET long print = 2, HAMLET long print plus region-wise resonance integrals
155	NEPPN	HAMLET punch?
156	NEPSG	= 0, use Selengut-Goertzel approximation
157	NEPAG	Self-consistent age approx?
165	NFGRN	= 0, do not run reactor = 1, run FLOG = 2, run HERESY

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

Relative Location	Symbol	Description
166	NFGPT	FLOG or HERESY long print?
167	NFGPN	FLOG punch?
168	NFGPB	Point buckling calculation?
170	NFGB1	Boundary condition at origin
171	NFGB2	Boundary condition at outer boundary
177	NEDRN	= 0, do not run DIED = 1, run DIED, Version 1 = 2, run DIED, Version 2
178	NEDPT	DIED long print?
179	NEDPN	DIED PUNCH?
180	NEDNB	Detailed neutron balance test = 0, ignore = 1, smooth and resonance breakdown by group = 2, total only by group
181	NEDFW	Averaging flux used = 0, ignore = 1, asymptotic = 2, asymptotic + interfaces = 3, asymptotic + interfaces + region midpoints
182	NEDRB	Regionwise neutron balance = 0, ignore = 1, all regions = 2, all fission regions = 3, read table from cards
183	NEDAX	Compute average cross sections = 0, ignore = 1, all regions = 2, all fission regions = 3, read table from cards
184	NEDNU	= 0, balance absorption vs fissions = 1, balance absorption vs production

Relative Location	Symbol	Description
189	NBURN	= 0, do not run BURNUP = 1, run BURNUP
190	NBURI	= 0, do not print resonance integral in HAMLET = 1, standard print of resonance integral in HAMLET = 2, regionwise print of resonance integral in HAMLET
191	NBUPN	= 0, do not punch cell input at each spectrum renormalization = 1, punch cell input at each spectrum renormalization

7. REFERENCES

1. T. R. England. *CINDER - A One Point Depletion and Fission Product Program.* USAEC Report WAPD-TM-334 (Rev.), Westinghouse Electric Corp., Bettis Atomic Power Laboratory, West Mifflin, Pa. (1964).

2. F. W. Todt. *GARGOYLE - An Infinite Medium Fuel Cycle Analysis Code, with Fuel and Poison Searches.* USAEC Report GA-6622, General Dynamics Corp., General Atomic Div., San Diego, Calif. (1966).

3. D. R. Vondy, T. B. Fowler, and M. L. Tobias. *Reactor Depletion Code ASSAULT (Two-Dimensional Multi-Neutron-Group, Diffusion).* USAEC Report ORNL-TM-1302, Union Carbide Corp., Oak Ridge National Laboratory, Oak Ridge, Tenn. (1966).

4. R. F. Barry. *LEOPARD - A Spectrum-Dependent Non-Spatial Depletion Code for the IBM-7094.* USAEC Report WCAP-3269-26, Westinghouse Electric Corp., Atomic Power Div., Pittsburgh, Pa. (1963).

5. C. G. Poncelet. *LASER - A Depletion Program for Lattice Calculations Based on MUFT and THERMOS.* USAEC Report WCAP-6073, Westinghouse Electric Corp., Atomic Power Div., Pittsburgh, Pa. (1966).

6. J. E. Suich and H. C. Honeck. *The HAMMER System.* USAEC Report DP-1064, E. I. du Pont de Nemours and Co., Savannah River Laboratory, Aiken, S. C. (1967).

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

8. 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 Atomic Power Laboratory, West Mifflin, Pa. (1957).

9. D. J. McGoff. *FORM - A Fourier Transform Fast Spectrum Code for the IBM-709.* USAEC Report NAA-SR-Memo-5766, North American Aviation, Inc., Atomics International Div., Canoga Park, Calif. (1960).

10. 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).

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

12. D. R. Finch. *The HERESY III Code.* USAEC Report DP-1284, E. I. du Pont de Nemours and Co., Savannah River Laboratory, Aiken, S. C. (to be issued).

13. J. A. Smith, Personal Communication, Savannah River Laboratory.

14. T. R. England. *Time-Dependent Fission-Product Thermal and Resonance Absorption Cross Sections.* USAEC Report WAPD-TM-333, Westinghouse Electric Corp., Bettis Atomic Power Laboratory, West Mifflin, Pa. (1962), and Addendum 1 (1965).

15. W. E. Holden and F. W. Walker. *Chart of the Nuclides.* 10th ed., General Electric Corp., Knolls Atomic Power Laboratory, Schenectady, N. Y. (1968).

APPENDIX A – NUCLEAR DATA

The programs in the HAMBUR System contain a large amount of nuclear data and constants stored internally. Tables A-I through A-VI summarize these data for user reference.

Table A-I gives the energy release per fission and per radiative capture, the energy release in kinetic heating, and the decay constants for α and β decays. Energy yield data for isotopes through plutonium are from Reference 4. Energy yield data for americium and heavier isotopes are based on best-guess estimates. Decay constants were obtained from Reference 15.

Table A-II contains fitted cross section coefficients, 2200 m/sec absorption and fission cross section, and infinitely dilute absorption resonance integrals used to obtain "fuel chain" and "added isotope" cross sections. Cross sections as a function of the constants and coefficients are:

$$\sigma^a_{th}/\sigma^a_{2200} = A_0 + A_1(\phi_{epi}/\phi_{th}) + A_2(\phi_{epi}/\phi_{th})^2 \qquad (A.1)$$

$$\sigma^f_{th} = \sigma^a_{th}(\sigma^f_{2200}/\sigma^a_{2200}) \qquad (A.2)$$

Equation (A.2) is used except for ^{239}Pu and ^{241}Pu where

$$\sigma^f_{th}/\sigma^f_{2200} = D_0 + D_1(\phi_{epi}/\phi_{th}) + D_2(\phi_{epi}/\phi_{th})^2 \qquad (A.3)$$

The following expression is used for all isotopes

$$\sigma_{fast}/I_{eff} = 0.00267 + 0.0554(\phi_{epi}/\phi_{th}) - 0.00173(\phi_{epi}/\phi_{th})^2 \qquad (A.4)$$

The relation between I_{eff} and I_∞ is given by

$$\log_{10}(I_\infty/I_{eff}) = C_0 + C_1 \log_e(\log_{10}\sigma_p) + C_2 \log_{10}\sigma_p \qquad (A.5)$$

Each of these expressions has been discussed in detail in Section 3.4.

Table A-III contains a description of the isotopes and chains used in the fission product poisoning calculation, and the thermal and epithermal cross sections, decay constants, Mev/decay, and coupling to the next isotope in the chain. Isotopes included in the lumped fission product are indicated as well as which entry or entries are included in the case of multiple production chains for a fission product. The chains and data are derived from Reference 14 by eliminating the very short lived and low yield isotopes and chains from the larger set given there. These chains in HAMBUR have been shown to represent over 95% of the total fission product poisoning that the more extensive chains of Reference 14 predict.

Table A-IV shows the direct fission yields of the basic six isotopes ^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, and ^{241}Pu normalized to a total yield of two fragments per fission. For isotopes of higher atomic numbers than Pu, the yields for ^{241}Pu are assumed. The data in Table A-IV are taken from Reference 14.

Table A-V shows the 2200 m/sec absorption cross section, resonance integrals, and decay constants used for all built-in "added isotopes." These data were taken from the HAMMER/HAMBUR libraries.

Table A-VI are built-in reflector contents for H_2O, D_2O, and graphite moderator reflectors. These are used in the FLOG program as an option.

TABLE A-I

Nuclear Constants for Fuel Chain Isotopes

Isotope	Mev/Fission	Mev/Capture	Total Kinetic Heat, Mev per Fission (1.93 ν)	Decay Constants, sec^{-1}	
				α	β
^{232}Th	186.0	4.92	5.03		
^{233}Pa		5.16			2.92×10^{-7}
^{233}U	187.0	5.38	4.83		
^{234}U		5.15			
^{235}U	189.0	6.43	4.68		
^{236}U		5.45			
^{238}U	186.7	4.062	5.03		
^{237}Np		4.0			
^{239}Np					3.41×10^{-6}
^{238}Pu		4.0		2.51×10^{-10}	
^{239}Pu	196.0	6.42	5.57		
^{240}Pu		5.71			
^{241}Pu	198.0	6.06	5.40		1.68×10^{-9}
^{242}Pu		5.50			
^{241}Am	200.0	4.0	5.0	4.65×10^{-11}	
^{242m}Am	200.0	4.0	5.0		
^{243}Am	200.0	4.0	5.0		
^{242}Cm	200.0	4.0	5.0	4.91×10^{-8}	
^{244}Cm	200.0	4.0	5.0	1.25×10^{-9}	
^{245}Cm	200.0	4.0	5.0		

TABLE A-II

Fitted Cross Section Coefficients

Isotope	2200 m/sec Cross Section, barns		Uranium Fuel			Plutonium Fuel		
	σ_a	σ_f	A_0	A_1	A_2	A_0	A_1	A_2
^{232}Th	7.4	0.0	.797	-.0859	.00673	.790	-.0387	.00231
^{233}Pa	47.0	0.0	.750	-.0722	.00551	.790	-.0387	.00231
^{233}U	573.0	524.0	.797	-.0859	.00673	.790	-.0387	.00231
^{234}U	95.0	0.0	.797	-.0859	.00673	.790	-.0387	.00231
^{235}U	678.0	577.0	.756	-.0867	.00667	.790	-.0387	.00231
^{236}U	6.0	0.0	.797	-.0859	.00673	.790	-.0387	.00231
^{238}U	2.73	0.0	.797	-.0859	.00673	.790	-.0387	.00231
^{237}Np	169.0	0.02	.784	.00293	.00097	.790	-.0387	.00231
^{239}Np	60.0	0.0	.797	-.0859	.00673	.790	-.0387	.00231
^{238}Pu	500.0	16.8	.797	-.0859	.00673	.790	-.0387	.00231
^{239}Pu	1015.0	741.0	.938	.0724	-.00513	.942	-.000772	-.000489
^{240}Pu	286.0	0.03	.797	-.0451	.0036	.807	-.0229	.00151
^{241}Pu	1375.0	950.0	.870	-.0391	.00273	.876	-.0295	.00151
^{242}Pu	30.0	0.0	.797	-.0859	.00673	.790	-.0387	.00231
^{241}Am	600.0	3.1	.797	-.0859	.00673	.790	-.0387	.00231
^{242m}Am	8000.0	6400.0	.797	-.0859	.00673	.790	-.0387	.00231
^{243}Am	85.0	0.0	.797	-.0859	.00673	.790	-.0387	.00231
^{242}Cm	20.0	3.0	.797	-.0859	.00673	.790	-.0387	.00231
^{244}Cm	20.0	1.67	.797	-.0859	.00673	.790	-.0387	.00231
^{245}Cm	2869.0	2438.5	.797	-.0859	.00673	.790	-.0387	.00231
			D_0	D_1	D_2	D_0	D_1	D_2
^{239}Pu		741.0	.892	-.037	-.00217	.896	-.00854	.000053
^{241}Pu		950.0	.855	-.032	.00273	.863	-.0280	-.00141

Table A-II (Contd)

Isotope	I_∞^a barns	$\frac{I_\infty^f}{I_\infty^a}$	C_0	C_1	C_2
^{232}Th	83.0	0.0	1.2795	0.11776	-0.33603
^{233}Pa	817.0	0.0	0.55887	0.11697	-0.18758
^{233}U	887.0	0.851	0.4186	-0.29474	-0.002622
^{234}U	661.0	0.0	1.4479	0.48395	-0.44545
^{235}U	402.0	0.667	0.33177	-0.12706	-0.038037
^{236}U	320.0	0.0	1.44	0.38569	-0.44056
^{238}U	278.0	0.0	1.6965	-0.14264	-0.32293
^{237}Np	847.0	0.0	0.33546	0.084125	-0.10807
^{239}Np	415.0	0.0	0.0	0.0	0.0
^{238}Pu	152.0	0.0	0.26061	0.019659	-0.070813
^{239}Pu	428.0	0.61	0.53063	0.08	-0.15329
^{240}Pu	8220.0	0.0	2.0925	0.52221	-0.55858
^{241}Pu	697.0	0.805	0.54679	-0.20829	-0.062126
^{242}Pu	1100.0	0.0	1.6996	0.5388	-0.51329
^{241}Am	1400.0	0.0	1.25	-0.0263	-0.284
^{242m}Am	2000.0	0.85	0.0	0.0	0.0
^{243}Am	1470.0	0.0	1.0248	0.30166	-0.32626
^{242}Cm	8.0	0.0	0.0	0.0	0.0
^{244}Cm	621.0	0.0	2.17	-0.165	-0.394
^{245}Cm	75.0	0.80	0.0	0.0	0.0

TABLE A-III

Fission Product Cross Sections and Constants

Chain No.	Isotope No.	Isotope Name	Cross Sections σ_a^{fast}	Cross Sections σ_a^{th}	Decay Constant β, sec^{-1}	Coupling	Mev/Decay	Included in Lumped Fission Products
1	1	^{81}Br	6.74	3.3	0.0	C	0.0	✓
	2	^{82}Br	0.0	0.0	5.363×10^{-6}	β	2.56	✓
	3	^{82}Kr	21.55	45.0	0.0	C	0.0	✓
	4	^{83}Kr	24.11	205.0	0.0		0.0	✓
2	5	^{90}Sr	0.198	1.0	7.84×10^{-10}	C	0.191	✓
	6	^{91}Sr	0.0	0.0	1.985×10^{-5}	β	1.64	✓
	7	^{91}Y	0.0466	1.07	1.383×10^{-7}	β	1.07	✓
	8	^{91}Zr	0.99	1.58	0.0		0.0	✓
3	9	^{93}Y	0.0	0.0	1.869×10^{-5}	β	1.07	✓
	10	^{93}Zr	3.08	1.1	0.0		0.0	✓
4	11	^{95}Zr	0.0	0.0	1.234×10^{-7}	β	0.85	✓
	12	^{95}Nb	0.0	0.0	2.292×10^{-7}	β	0.82	✓
	13	^{95}Mo	11.61	13.9	0.0		0.0	✓
5	14	^{97}Zr	0.0	0.0	1.133×10^{-5}	β	1.36	✓
	15	^{97}Mo	1.76	2.2	0.0	C	0.0	✓
	16	^{98}Mo	0.616	0.51	0.0	C	0.0	✓
	17	^{99}Mo	0.0	0.0	2.895×10^{-6}	β	0.576	✓
	18	$^{99}Tc(m)$	0.0	0.0	3.209×10^{-5}	β	0.126	✓
	19	^{99}Tc	15.4	22.0	0.0		0.0	✓
6	20	^{100}Mo	0.695	0.5	0.0	C	0.0	✓
	21	^{101}Ru	8.47	5.0	0.0	C	0.0	✓
	22	^{102}Ru	1.21	1.44	0.0	C	0.0	✓
	23	^{103}Ru	0.0	0.0	2.021×10^{-7}	β	0.548	✓
	24	^{103}Rh	113.3	149.0	0.0		0.0	✓
7	25	^{104}Ru	0.88	0.7	0.0	C	0.0	✓
	26	^{105}Ru	0.009	0.2	4.327×10^{-5}	β	1.03	✓
	27	^{105}Rh	653.1	15000.0	5.348×10^{-6}	β	0.275	✓
	28	^{105}Pd	8.36	11.0	0.0		0.0	✓
8	29	^{106}Ru	1.72	9.0	2.175×10^{-8}	β	0.19	✓
	30	^{106}Pd	1.32	6.0	1.6×10^{-10}	C	0.0	✓
	31	^{107}Pd	4.4	10.0	0.0	C	0.0	✓
	32	^{108}Pd	16.98	10.4	0.0	C	0.0	✓
	33	^{109}Pd	0.0	0.0	1.437×10^{-5}	β	0.474	✓
	34	^{109}Ag	158.14	87.0	0.0		0.0	✓

Table A-III (Contd)

Chain No.	Isotope No.	Isotope Name	Cross Sections σ_a^{fast}	Cross Sections σ_a^{th}	Decay Constant β, sec^{-1}	Coupling	Mev/Decay	Included in Lumped Fission Products
9	35	^{110}Pd	1.1	0.21	0.0	C	0.0	✓
	36	^{111}Ag	0.0	0.0	1.056×10^{-6}	β	0.39	✓
	37	^{111}Cd	5.72	2.0	0.0	C	0.0	✓
	38	^{112}Cd	1.43	0.03	1.6×10^{-10}	C	0.0	✓
	39	^{113}Cd	71.73	20000.0	0.0	C	0.0	✓
	40	^{114}Cd	1.65	1.24	0.0	C	0.0	✓
	41	^{115}Cd	0.0	0.0	3.633×10^{-10}	β	0.506	✓
	42	^{115}In	363.0	207.0	0.0		0.0	✓
10	43	^{127}Sb	0.0	0.0	2.116×10^{-6}	β	1.43	✓
	44	^{127}Te	0.0	0.0	2.07×10^{-5}	β	0.245	✓
	45	^{127}I	17.87	7.0	0.0		0.0	✓
11	46	^{128}Te	1.13	0.3	0.0	C	0.0	✓
	47	^{129}I	4.26	32.0	0.0		0.0	✓
12	48	$^{131}Te(m)$	0.0	0.0	6.418×10^{-6}	β	1.52	✓
	49	^{131}I	2.18	50.0	9.966×10^{-7}	β	0.585	✓
	50	^{131}Xe	87.98	120.0	0.0		0.0	✓
13	51	^{133}I	0.0	0.0	9.257×10^{-6}	β	0.922	✓
	52	^{133}Xe	8.27	190.0	5.1522×10^{-6}		0.18	✓
14	53	^{133}I	0.0	0.0	9.257×10^{-6}	β	0.922	✓
	54	^{133}Xe	8.27	190.0	5.1522×10^{-6}	β	0.18	✓
	55	^{133}Cs	46.21	29.0	0.0	C	0.0	✓
	56	^{134}Cs	5.83	134.0	9.55×10^{-7}		1.80	✓
15	57	^{137}Cs	0.033	0.11	7.471×10^{-10}	β	0.772	
	58	^{137}Ba	0.22	5.10	0.0	C	0.0	✓
	59	^{138}Ba	0.055	0.7	0.0	C	0.0	✓
	60	^{139}Ba	1.21	8.9	0.0		0.0	✓
16	61	^{140}Ba	0.522	12.0	6.768×10^{-7}		0.58	✓
17	62	^{141}Ce	0.0	0.0	2.431×10^{-7}	β	0.272	✓
	63	^{141}Pr	2.59	11.6	0.0		0.0	✓
18	64	^{142}Ce	0.143	1.0	0.0	C	0.0	✓
	65	^{143}Ce	0.261	6.0	5.835×10^{-6}	β	0.764	✓
	66	^{143}Pr	20.91	89.0	5.856×10^{-7}	β	0.31	✓
	67	^{143}Nd	14.3	324.0	0.0		0.0	✓
19	68	^{143}Ce	0.261	6.0	5.835×10^{-6}	C	0.764	
	69	^{144}Ce	0.290	0.96	2.865×10^{-8}		0.124	✓

Table A-III (Contd)

Chain No.	Isotope No.	Isotope Name	Cross Sections σ_a^{fast}	Cross Sections σ_a^{th}	Decay Constant β, sec^{-1}	Coupling	Mev/Decay	Included in Lumped Fission Products
20	70	^{145}Pr	0.0	0.0	3.231×10^{-5}	β	1.68	✓
	71	^{145}Nd	26.96	60.0	0.0	C	0.0	✓
	72	^{146}Nd	2.75	10.0	0.0	C	0.0	✓
	73	^{147}Nd	0.0	0.0	7.228×10^{-7}	β	0.406	✓
	74	^{147}Pm	354.3	235.0	8.289×10^{-9}	C	0.070	✓
	75	$^{148}Pm(m)$	3521.0	27000.0	1.976×10^{-7}	C	2.08	✓
	76	^{149}Pm	0.0	0.0	3.626×10^{-6}	β	0.382	✓
	77	^{149}Sm	374.1	40800.0	0.0	C	0.0	
	78	^{150}Sm	28.0	97.0	0.0	C	0.0	✓
	79	^{151}Sm	275.1	15000.0	2.746×10^{-10}	C	0.029	✓
	80	^{152}Sm	275.1	216.0	0.0		0.0	✓
21	81	^{145}Pr	0.0	0.0	3.231×10^{-5}	β	1.68	
	82	^{145}Nd	26.96	60.0	0.0	C	0.0	
	83	^{146}Nd	2.75	10.0	0.0	C	0.0	
	84	^{147}Nd	0.0	0.0	7.228×10^{-7}	β	0.406	
	85	^{147}Pm	354.3	235.0	8.289×10^{-9}	C	0.070	
	86	^{148}Pm	4841.3	1500.0	1.488×10^{-6}	C	1.49	✓
	87	^{149}Pm	0.0	0.0	3.626×10^{-6}	β	0.382	✓
	88	^{149}Sm	374.1	40800.0	0.0	C	0.0	
	89	^{150}Sm	28.0	97.0	0.0	C	0.0	✓
	90	^{151}Sm	275.1	15000.0	2.746×10^{-10}	C	0.029	✓
	91	^{152}Sm	275.1	216.0	0.0		0.0	✓
22	92	^{145}Nd	26.96	60.0	0.0	C	0.0	
	93	^{146}Nd	2.75	10.0	0.0	C	0.0	
	94	^{147}Nd	0.0	0.0	7.228×10^{-7}	β	0.122	
	95	^{147}Pm	354.3	235.0	8.289×10^{-9}	β	0.070	
	96	^{147}Sm	77.87	87.0	0.0		0.0	✓
23	97	^{148}Nd	5.28	3.4	0.0	C	0.0	✓
	98	^{149}Pm	0.0	0.0	3.626×10^{-6}	β	0.382	✓
	99	^{149}Sm	374.1	40800.0	0.0		0.0	✓
24	100	$^{148}Pm(m)$	5.28	3.4	0.0	C	0.0	
	101	^{149}Pm	0.0	0.0	3.626×10^{-6}	β	0.382	
	102	^{149}Sm	374.1	40800.0	0.0	C	0.0	
	103	^{150}Sm	28.0	97.0	0.0	C	0.0	✓
	104	^{151}Sm	275.1	15000.0	2.746×10^{-10}	C	0.029	✓
	105	^{152}Sm	275.1	216.0	0.0	C	0.0	✓
	106	^{153}Sm	0.0	0.0	4.097×10^{-6}	β	0.316	✓
	107	^{153}Eu	151.8	450.0	0.0	C	0.0	✓
	108	^{154}Eu	65.31	15000.0	1.373×10^{-9}	C	1.48	✓
	109	^{155}Eu	610.0	14000.0	5.491×10^{-9}		0.18	✓

Table A-III (Contd)

Chain No.	Isotope No.	Isotope Name	Cross Sections σ_a^{fast}	Cross Sections σ_a^{th}	Decay Constant β, sec^{-1}	Coupling	Mev/Decay	Included in Lumped Fission Products
25	110	^{150}Nd	1.54	3.0	0.0	C	0.0	✓
	111	^{151}Pm	0.0	0.0	6.78×10^{-6}	β	0.43	✓
	112	^{151}Sm	275.1	15000.0	2.746×10^{-10}	C	0.029	✓
	113	^{152}Sm	275.1	216.0	0.0	C	0.0	✓
	114	^{153}Sm	0.0	0.0	4.097×10^{-6}	β	0.31	✓
	115	^{153}Eu	151.8	450.0	0.0	C	0.0	✓
	116	^{154}Eu	65.31	15000.0	1.37×10^{-9}	C	1.48	✓
	117	^{155}Eu	610.0	14000.0	5.491×10^{-9}		0.17	
26	118	^{150}Nd	1.54	3.0	0.0	C	0.0	
	119	^{151}Pm	0.0	0.0	6.78×10^{-6}	β	0.43	
	120	^{151}Sm	275.1	15000.0	2.746×10^{-10}	C	0.029	
	121	^{152}Sm	275.1	216.0	0.0	C	0.0	
	122	^{153}Sm	0.0	0.0	4.097×10^{-6}	β	0.31	
	123	^{153}Eu	151.8	450.0	0.0	C	0.0	
	124	^{154}Eu	65.31	15000.0	1.37×10^{-9}	C	1.48	
	125	^{155}Eu	610.0	14000.0	5.491×10^{-9}		0.17	✓
27	126	^{154}Sm	2.75	5.5	0.0	C	0.0	✓
	127	^{155}Eu	610.0	14000.0	5.491×10^{-9}		0.17	✓
28	128	^{157}Eu	0.0	0.0	1.25×10^{-5}	β	0.716	
	129	^{157}Gd	81.42	242000.0	0.0	C	0.0	
	130	^{158}Gd	3.19	3.9	1.6×10^{-10}	C	0.0	
	131	^{159}Gd	0.0	0.0	1.07×10^{-5}	β	0.367	
	132	^{159}Tb	42.71	46.0	0.0		0.0	
29	133	^{135}I	0.0	0.0	2.874×10^{-5}	β	2.26	✓
	134	^{135}Xe	500.0	2720000.0	2.093×10^{-5}	β	0.57	✓
	135	^{135}Cs	6.82	8.7	0.0		0.0	
30	136	^{149}Pm	0.0	0.0	3.626×10^{-6}	β	0.38	✓
	137	^{149}Sm	374.1	40800.0	0.0		0.0	
31	138	Pseudo	0.65	4.75	0.0	0	0.0	✓
32	139	^{135}I	0.0	0.0	2.874×10^{-5}	β	2.26	
	140	^{135}Xe	500.0	2720000.0	2.093×10^{-5}	C	0.57	
	141	^{136}Xe	0.011	0.15	0.0	C	0.0	✓
	142	^{137}Xe	0.033	0.11	7.471×10^{-10}		0.74	✓

TABLE A-IV

Fission Product Yields

Chain No.	Isotope No.	Isotope Name	^{232}Th	^{233}U	^{235}U	^{238}U	^{239}Pu	^{241}Pu and Higher
1	1	^{81}Br	0.0036	0.0046	0.0014	0.0003	0.0011	0.0
	2	^{82}Br	0.01	0.0069	0.0028	0.0005	0.0017	0.0007
	3	^{82}Kr	0.0	0.0	0.0	0.0	0.0	0.0
	4	^{83}Kr	0.0199	0.0114	0.00544	0.0009	0.0029	0.001
2	5	^{90}Sr	0.073	0.0618	0.0577	0.034	0.022	0.014
	6	^{91}Sr	0.07164	0.0654	0.0581	0.0438	0.0254	0.0169
	7	^{91}Y	0.00036	0.0003	0.0003	0.0002	0.0001	0.0001
	8	^{91}Zr	0.0	0.0	0.0	0.0	0.0	0.0
3	9	^{93}Y	0.0596	0.0664	0.0610	0.0435	0.0369	0.0237
	10	^{93}Zr	0.0034	0.0038	0.0035	0.0025	0.0021	0.0013
4	11	^{95}Zr	0.0509	0.0615	0.0620	0.0475	0.0495	0.0396
	12	^{95}Nb	0.0	0.0	0.0	0.0	0.0	0.0
	13	^{95}Mo	0.0006	0.0007	0.0007	0.0005	0.0005	0.0004
5	14	^{97}Zr	0.0439	0.0531	0.0590	0.0533	0.0543	0.0552
	15	^{97}Mo	0.0014	0.0017	0.0019	0.0017	0.0017	0.0018
	16	^{98}Mo	0.0360	0.0525	0.0578	0.0590	0.0590	0.0600
	17	^{99}Mo	0.027	0.050	0.061	0.063	0.061	0.062
	18	$^{99}Tc(m)$	0.0	0.0	0.0	0.0	0.0	0.0
	19	^{99}Tc	0.0	0.0	0.0	0.0	0.0	0.0
6	20	^{100}Mo	0.0104	0.0449	0.063	0.063	0.071	0.063
	21	^{101}Ru	0.0062	0.0289	0.050	0.063	0.059	0.064
	22	^{102}Ru	0.0036	0.0216	0.0410	0.063	0.059	0.063
	23	^{103}Ru	0.0016	0.016	0.030	0.062	0.056	0.062
	24	^{103}Rh	0.0	0.0	0.0	0.0	0.0	0.0
7	25	^{104}Ru	0.00084	0.0092	0.018	0.048	0.059	0.060
	26	^{105}Ru	0.0005	0.005	0.009	0.037	0.055	0.058
	27	^{105}Rh	0.0	0.0	0.0	0.0	0.0	0.0
	28	^{105}Pd	0.0	0.0	0.0	0.0	0.0	0.0
8	29	^{106}Ru	0.00041	0.0022	0.0038	0.029	0.045	0.055
	30	^{106}Pd	0.0	0.0	0.0	0.0	0.0	0.0
	31	^{107}Pd	0.00049	0.0012	0.0019	0.013	0.036	0.050
	32	^{108}Pd	0.00053	0.0007	0.0009	0.0064	0.026	0.040
	33	^{109}Pd	0.00055	0.0004	0.0003	0.003	0.0155	0.031
	34	^{109}Ag	0.0	0.0	0.0	0.0	0.0	0.0

Table A-IV (Contd)

Chain No.	Isotope No.	Isotope Name	^{232}Th	^{233}U	^{235}U	^{238}U	^{239}Pu	^{241}Pu and Higher
9	35	^{110}Pd	0.00061	0.0003	0.00021	0.0014	0.0055	0.022
	36	^{111}Ag	0.00064	0.0003	0.00019	0.0007	0.0022	0.013
	37	^{111}Cd	0.0	0.0	0.0	0.0	0.0	0.0
	38	^{112}Cd	0.0	0.0	0.0	0.0	0.0	0.0
	39	^{113}Cd	0.00068	0.0002	0 00012	0.0005	0.0007	0.004
	40	^{114}Cd	0.00068	0.0002	0.00011	0.00045	0.0004	0.002
	41	^{115}Cd	0.00058	0.00017	0.00009	0.00034	0.00029	0.00084
	42	^{115}In	0.00004	0.00001	0.000006	0.00002	0.000019	0.00005
10	43	^{127}Sb	0.00037	0.0025	0.0010	0.00053	0.00151	0.0016
	44	^{127}Te	0.00041	0.00277	0.00114	0.00059	0.00168	0.00182
	45	^{127}I	0.0	0.0	0.0	0.0	0.0	0.0
11	46	^{128}Te	0.00154	0.0092	0.0036	0.00228	0.0065	0.0061
	47	^{129}I	0.0022	0.0096	0.0045	0.00298	0.0079	0.0067
12	48	$^{131}Te(m)$	0.00255	0.00512	0.0044	0.00345	0.0057	0.0051
	49	^{131}I	0.01445	0.02898	0.0249	0.0196	0.0323	0.0289
	50	^{131}Xe	0.0	0.0	0.0	0.0	0.0	0.0
13	51	^{133}I	0.037	0.0588	0.0659	0.052	0.069	0.060
	52	^{133}Xe	0.0	0.0	0.0	0.0	0.0	0.0
14	53	^{133}I	0.037	0.0588	0.0659	0.052	0.069	0.060
	54	^{133}Xe	0.0	0.0	0.0	0.0	0.0	0.0
	55	^{133}Cs	0.0	0.0	0.0	0.0	0.0	0.0
	56	^{134}Cs	0.0	0.0	0.0	0.0	0.0	0.0
15	57	^{137}Cs	0.067	0.0664	0.0615	0.060	0.065	0.064
	58	^{137}Ba	0.0	0.0	0.0	0.0	0.0	0.0
	59	^{138}Ba	0.073	0.071	0.0574	0.060	0.063	0.063
	60	^{139}Ba	0.0751	0.0661	0.0655	0.060	0.060	0.062
16	61	^{140}Ba	0.0759	0.0662	0.0634	0.0572	0.0552	0.0601
17	62	^{141}Ce	0.084	0.0624	0.064	0.054	0.056	0.060
	63	^{141}Pr	0.0	0.0	0.0	0.0	0.0	0.0
18	64	^{142}Ce	0.083	0.0679	0.0601	0.051	0.050	0.058
	65	^{143}Ce	0.0796	0.0588	0.060	0.0478	0.0458	0.0557
	66	^{143}Pr	0.0004	0.0003	0.0003	0.0002	0.0002	0.0003
	67	^{143}Nd	0.0	0.0	0.0	0.0	0.0	0.0

Table A-IV (Contd)

Chain No.	Isotope No.	Isotope Name	^{232}Th	^{233}U	^{235}U	^{238}U	^{239}Pu	^{241}Pu and Higher
19	68	^{143}Ce	0.0796	0.0588	0.060	0.0478	0.0458	0.0557
	69	^{144}Ce	0.075	0.0451	0.0562	0.045	0.038	0.048
20	70	^{145}Pr	0.066	0.0338	0.0398	0.039	0.031	0.036
	71	^{145}Nd	0.0	0.0	0.0	0.0	0.0	0.0
	72	^{146}Nd	0.055	0.0258	0.0307	0.033	0.026	0.028
	73	^{147}Nd	0.0382	0.0193	0.0236	0.028	0.0205	0.022
	74	^{147}Pm	0.0	0.0	0.0	0.0	0.0	0.0
	75	$^{148}Pm(m)$	0.0	0.0	0.0	0.0	0.0	0.0
	76	^{149}Pm	0.0	0.0	0.0	0.0	0.0	0.0
	77	^{149}Sm	0.0	0.0	0.0	0.0	0.0	0.0
	78	^{150}Sm	0.0	0.0	0.0	0.0	0.0	0.0
	79	^{151}Sm	0.0	0.0	0.0	0.0	0.0	0.0
	80	^{152}Sm	0.0	0.0	0.0	0.0	0.0	0.0
21	81	^{145}Pr	0.066	0.0338	0.0398	0.039	0.031	0.036
	82	^{145}Nd	0.0	0.0	0.0	0.0	0.0	0.0
	83	^{146}Nd	0.055	0.0258	0.0307	0.033	0.026	0.028
	84	^{147}Nd	0.0382	0.0193	0.0236	0.028	0.0205	0.022
	85	^{147}Pm	0.0	0.0	0.0	0.0	0.0	0.0
	86	^{148}Pm	0.0	0.0	0.0	0.0	0.0	0.0
	87	^{149}Pm	0.0	0.0	0.0	0.0	0.0	0.0
	88	^{149}Sm	0.0	0.0	0.0	0.0	0.0	0.0
	89	^{150}Sm	0.0	0.0	0.0	0.0	0.0	0.0
	90	^{151}Sm	0.0	0.0	0.0	0.0	0.0	0.0
	91	^{152}Sm	0.0	0.0	0.0	0.0	0.0	0.0
22	92	^{145}Nd	0.066	0.0338	0.0398	0.039	0.031	0.036
	93	^{146}Nd	0.055	0.0258	0.0307	0.033	0.026	0.028
	94	^{147}Nd	0.0382	0.0193	0.0236	0.028	0.0205	0.022
	95	^{147}Pm	0.0	0.0	0.0	0.0	0.0	0.0
	96	^{147}Sm	0.0	0.0	0.0	0.0	0.0	0.0
23	97	^{148}Nd	0.018	0.0128	0.0171	0.024	0.017	0.016
	98	^{149}Pm	0.0	0.0	0.0	0.0	0.0	0.0
	99	^{149}Sm	0.0	0.0	0.0	0.0	0.0	0.0
24	100	$^{148}Pm(m)$	0.018	0.0128	0.0171	0.024	0.017	0.016
	101	^{149}Pm	0.00945	0.0077	0.0113	0.021	0.013	0.012
	102	^{149}Sm	0.0	0.0	0.0	0.0	0.0	0.0
	103	^{150}Sm	0.0	0.0	0.0	0.0	0.0	0.0
	104	^{151}Sm	0.0	0.0	0.0	0.0	0.0	0.0
	105	^{152}Sm	0.0	0.0	0.0	0.0	0.0	0.0
	106	^{153}Sm	0.0	0.0	0.0	0.0	0.0	0.0

Table A-IV (Contd)

Chain No.	Isotope No.	Isotope Name	^{232}Th	^{233}U	^{235}U	^{238}U	^{239}Pu	^{241}Pu and Higher
	107	^{153}Eu	0.0	0.0	0.0	0.0	0.0	0.0
	108	^{154}Eu	0.0	0.0	0.0	0.0	0.0	0.0
	109	^{155}Eu	0.0	0.0	0.0	0.0	0.0	0.0
25	110	^{150}Nd	0.004	0.0055	0.0067	0.013	0.010	0.008
	111	^{151}Pm	0.002	0.0035	0.0044	0.009	0.008	0.005
	112	^{151}Sm	0.0	0.0	0.0	0.0	0.0	0.0
	113	^{152}Sm	0.00085	0.0022	0.00281	0.006	0.006	0.003
	114	^{153}Sm	0.00034	0.0013	0.0015	0.0034	0.003	0.0018
	115	^{153}Eu	0.00004	0.0002	0.00019	0.0004	0.0004	0.0002
	116	^{154}Eu	0.0	0.0	0.0	0.0	0.0	0.0
	117	^{155}Eu	0.0	0.0	0.0	0.0	0.0	0.0
26	118	^{150}Nd	0.004	0.0055	0.007	0.013	0.01	0.008
	119	^{151}Pm	0.002	0.0035	0.0044	0.009	0.008	0.005
	120	^{151}Sm	0.0	0.0	0.0	0.0	0.0	0.0
	121	^{152}Sm	0.00085	0.0022	0.00281	0.006	0.006	0.003
	122	^{153}Sm	0.00034	0.0013	0.0015	0.0034	0.003	0.0018
	123	^{153}Eu	0.00004	0.0002	0.00019	0.0004	0.0004	0.0002
	124	^{154}Eu	0.0	0.0	0.0	0.0	0.0	0.0
	125	^{155}Eu	0.0	0.0	0.0	0.0	0.0	0.0
27	126	^{154}Sm	0.00018	0.00047	0.00077	0.00022	0.0025	0.00015
	127	^{155}Eu	0.00009	0.0003	0.0033	0.0013	0.0016	0.001
28	128	^{157}Eu	0.002	0.00005	0.000018	0.00035	0.0007	0.0005
	129	^{157}Gd	0.0	0.0	0.0	0.0	0.0	0.0
	130	^{158}Gd	0.000015	0.00003	0.00002	0.00017	0.00035	0.00040
	131	^{159}Gd	0.00001	0.00001	0.00001070	0.00008	0.00015	0.0002
	132	^{159}Tb	0.0	0.0	0.0	0.0	0.0	0.0
29	133	^{135}I	0.0539	0.0562	0.0617	0.0578	0.0693	0.0626
	134	^{135}Xe	0.0021	0.0022	0.0024	0.0022	0.0027	0.0024
	135	^{135}Cs	0.0	0.0	0.0	0.0	0.0	0.0
30	136	^{149}Pm	0.00745	0.0077	0.0113	0.021	0.013	0.012
	137	^{149}Sm	0.0	0.0	0.0	0.0	0.0	0.0
31	138	Pseudo	0.26	0.26	0.26	0.26	0.26	0.26
32	139	^{135}I	0.0539	0.0562	0.0617	0.0578	0.0693	0.0626
	140	^{135}Xe	0.0021	0.0022	0.0024	0.0022	0.0027	0.0024
	141	^{136}Xe	0.064	0.0668	0.0646	0.060	0.066	0.066
	142	^{137}Xe	0.067	0.0664	0.0615	0.060	0.065	0.064

TABLE A-V

Cross Sections and Decay Constants for "Added Isotopes"

Isotope Name	σ^a_{2200}	I^a_∞, barns	Decay Constant, sec^{-1}	Decay Mode
T	0.0	0.0	9.752×10^{-9}	β^-
^{6}Li	950.0	50.0		
^{59}Co	37.0	7.7		
^{60}Co	6.0	0.45	4.2×10^{-9}	β^-
^{169}Tm	127.0	13.0		
^{170}Tm	150.0	16.0	6.417×10^{-8}	β^-
^{171}Tm	150.0	16.0	1.51×10^{-9}	β^-
^{209}Bi	0.034	0.015		
^{210}Bi	0.0	0.0	1.604×10^{-6}	β^-
^{210}Po	0.0	0.0	7.10×10^{-8}	α

TABLE A-VI

Built-In Reflector Constants (FLØG)

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

APPENDIX B – COMPUTER REQUIREMENTS

The HAMBUR program was written in FORTRAN IV for the IBM System/360-65 computer. It requires 250,000 bytes (62,500 single precision words) of resident core excluding the operating system. The program is overlayed in core, and the overlay tree is shown in Figure B-1. Table B-I gives a breakdown of subroutines contained in each segment.

Extensive use is made of scratch disk data sets for passing data from one program to another and for temporary storage. Table B-II gives the data sets, uses, and approximate space requirements.

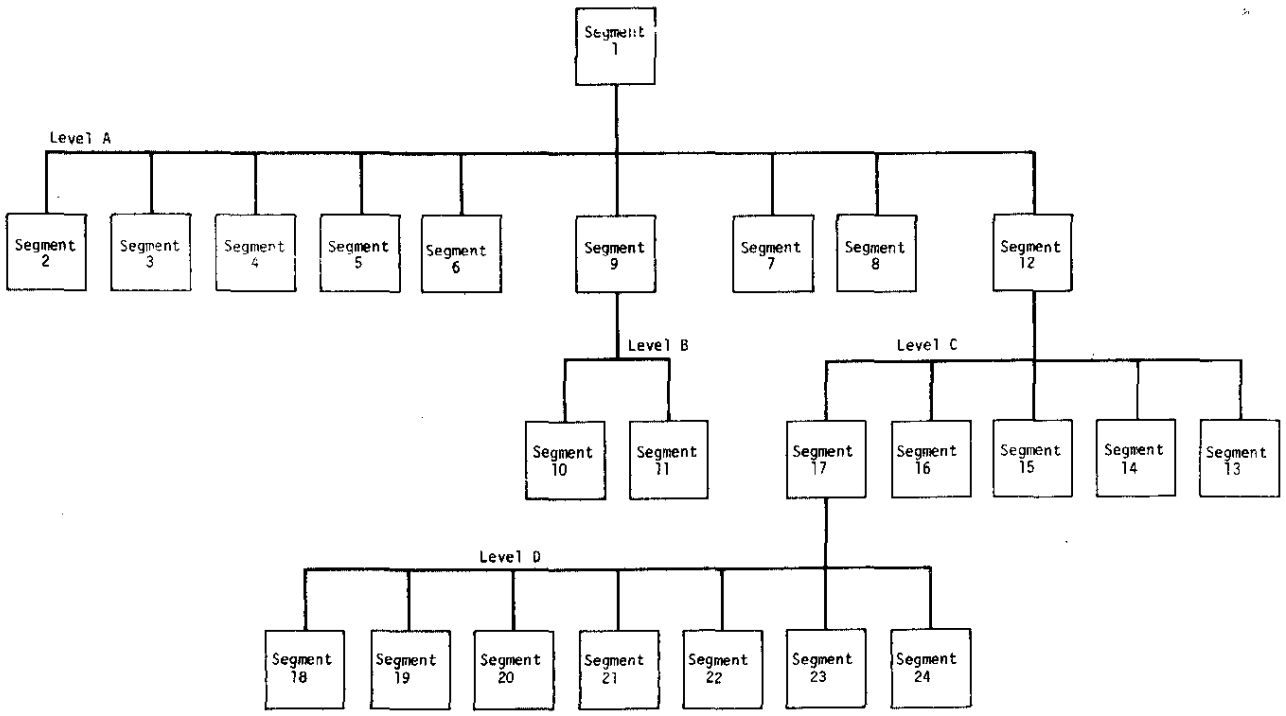


FIGURE B-1. HAMBUR Overlay

TABLE B-I

Subroutines in HAMBUR
(*indicates a possible system dependence)

Segment 1	Segment 2	Segment 3	Segment 4
1. MAIN	1. CAPN	1. THRMØS	1. THRMS2
2. CHAIN	2. LINK	2. EDIT	2. EDIT2
3. TIMEX*	3. TAPE	3. GEØR	3. GEØR2
4. TIMEZ*	4. INPA	4. GEØX	4. GEØX2
	5. INPB	5. THEFT	5. THEFT2
	6. MESH	6. ITER	6. ITER2
	7. MATC	7. SLABK	7. ESCAP1
	8. CØØLER	8. EI	8. EDEH2
	9. FPC	9. EDEH	9. SCALE2
	10. STØP	10. SCALE	10. STØP12
		11. STØP1	

Segment 5			Segment 6	
1. HAMLET	16. NIT	31. TK	1. FLØG	11. KERC1
2. INPUT	17. RESØL	32. TXI	2. INPC	12. EXTRP1
3. LIB	18. CRØSS	33. A	3. JØCØS	13. ANPR
4. RTBLIB	19. TUZ	34. B	4. FLED	14. ØMCAL
5. ZUTIN	20. BARJ	35. FLØUT	5. LLIB	15. FLUX
6. SØRE	21. SIMPS	36. EXTRAP	6. BILIB	16. CØNS
7. FLUX1	22. SAMPS	37. SLØDØN	7. BSQCD	17. CRIT1
8. GEØM	23. SØMPS	38. CLAPS	8. DATIN	18. CRIT2
9. ESCAPE	24. SEMPS	39. KEFC	9. DAPR	19. CHEV
10. RESC	25. GRID	40. EDIT12	10. LSQ	20. STØP3
11. PCALC	26. SCALC	41. ED12P		
12. PZCALC	27. GAMINC	42. EDIT3		
13. PTBL	28. FJ	43. ED3P		
14. DTBL	29. DANC	44. PUNL		
15. PSIFNC	30. TJ	45. STØP2		

TABLE B-I (continued)

Segment 7	Segment 8	Segment 9	
1. DIED	1. DIED2	1. HERESY	9. MXMPMX
2. BALANZ	2. LLED	2. ERRHLT	10. MXSBMX
3. REGØUT		3. ABMXVC	11. RVMPCV
		4. DGMPMX	12. RVMPDG
		5. MXADDG	13. RVMPMX
		6. MXADMX	14. SCMPMX
		7. MXMPCV	15. MAXVAL
		8. MXMPDG	

Segment 10		Segment 11	
1. FIRST	14. HSCALE	1. SECØND	13. SEARCH
2. HDATIN	15. HMESH	2. DELTAS	14. HEAD
3. RESA	16. ATYPE	3. TYPE	15. SUMRY
4. CONVTH	17. MATGEN	4. DATPRT	16. CØNV
5. DATCHK	18. MATRIX	5. MATDMP	17. ETASER
6. HGEØM	19. BTERP	6. FØRMQ	18. FSER
7. INGEØM	20. KERG1	7. FØRMDZ	19. RETA
8. HEX	21. KERG2	8. FØRMKA	20. BUCKLE
9. CIRCLE	22. KERGZ	9. FØRMN	21. HEDIT
10. SQUARE	23. TFUNC	10. FØRMA	22. DIEDTP
11. PRINT	24. EØNE	11. EIGEN1	23. MINV
12. PUNCH	25. BESK	12. CØNVER	
13. PLØT			

Segment 12	Segment 13	Segment 14	Segment 15	Segment 16
1. BURNUP	1. MIXLAT	1. VARINT	1. RUNGKT	1. CINDER
2. REGURG	2. BURIN	2. XPØWER	2. NUFLUX	2. CØNVRT
3. BERRØR		3. RR	3. KDELT	3. CYIELD
4. XLSQF		4. RDCAP	4. VYTSTP	4. TYIELD
5. REDIT		5. CHIMP	5. SAXE	5. FPCØNC
6. CVTPWR		6. XSCFIT	6. ADDISØ	
7. LSQFIT				
8. PHCASE				

Segment 17	Segment 18	Segment 19	Segment 20	Segment 21
1. EDHIST	1. MIXMAP	1. FXEDIT	1. CSEDIT	1. CTEDIT
2. PLØT1*			2. CSPRT	

Segment 22	Segment 23	Segment 24
1. MBEDIT	1. IPEDIT	1. TYPEII
2. MBPRT		

TABLE B-II

Data Sets used in HAMBUR

Reference No.	Use	Device Type	Disk Space[a]
2,3	Scratch	Disk	350,000
4	Lattice Library	Disk or Tape[b]	350,000
5	Input	System Input	
6	Output	System Printer	
9	Lithe Library	Disk or Tape[b]	As required
10	Help Library	Disk or Tape[b]	As required
11,12,16,18	Scratch	Disk	350,000
13,14	Scratch	Disk	3,500
15	Scratch	Disk	750,000
17[b]	Burnup History Input Tape	Disk or Tape[b]	1,500
19,25	Scratch	Disk	1,500
20	Scratch	Disk	10,000
26	Burnup History Output Tape	Disk or Tape[b]	350,000
21-23,30-33, 40-43,51,52	Scratch	Disk	2,500

a. Single precision words.
b. Indicates an optional usage.