214
1-15-73
ANCR-1083

Sh. 2420
UC-32

DWARF
A FEW GROUP, ONE-DIMENSIONAL, DIFFUSION
THEORY PROGRAM FOR REACTOR DEPLETION
WITH ADJUSTABLE REACTIVITY FEEDBACK

G.E. Putnam and E.C. Anderson

Aerojet Nuclear Company

NATIONAL REACTOR TESTING STATION

Idaho Falls, Idaho — 83401

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DATE PUBLISHED—NOVEMBER 1972

PREPARED FOR THE

U. S. ATOMIC ENERGY COMMISSION

IDAHO OPERATIONS OFFICE UNDER CONTRACT AT(10-1)-1375

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

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

LEGAL NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

ANCR-1083

Mathematics and Computers
TID-4500

DWARF

A FEW GROUP, ONE-DIMENSIONAL, DIFFUSION THEORY PROGRAM
FOR REACTOR DEPLETION WITH ADJUSTABLE REACTIVITY FEEDBACK

Theory and Program By
E. C. Anderson and G. E. Putnam

Report Written By
G. E. Putnam

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

AEROJET NUCLEAR COMPANY

Date Published - November 1972

PREPARED FOR THE U. S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
UNDER CONTRACT NO. AT(10-1)-1375

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

89

ABSTRACT

The DWARF program solves the few group, neutron diffusion theory equations, together with the Bateman equations for the concentrations of nuclides at a time point, to allow a very general one-dimensional simulation of reactor burnup, neutron irradiation, or Xenon oscillation problems.

To solve for the fission source and flux distributions, the usual power method (fission source iteration) is used and either two-point or Chebyshev extrapolation can be used to accelerate convergence. Within each group, difference equations are solved simultaneously for the mesh point fluxes by Gaussian elimination of subdiagonal elements and subsequent backsolution.

The nuclide concentrations at each time point are obtained by the analytical solution of the Bateman equations with the assumption that the neutron fluxes are constant at their initial values throughout the time step.

A thermal feedback iterative loop is used to obtain consistent solutions for the distributions of the reactor power, neutron flux, and fuel and coolant properties. The neutron group constants for the macroscopic materials can be user specified functions of the coolant density and the average fuel temperature. The group constants can also be a mixture of nondepleting macroscopic and depleting microscopic material data in a region. The solutions for the thermal properties of the coolant cover the entire range from the subcooled to the superheated vapor regimes with constant pressure and mass flow rate assumed.

Automatic searches on buckling, poison, region dimensions, a region interface position, or total power can be done at each time point to adjust the eigenvalue. There can be up to four groups, 40 regions, 50 macroscopic materials (only 10 are functions of the feedback variables), 50 nuclides per region, and 250 mesh points. A flexible plot routine can plot up to three distributions (of fluxes, power, temperature, densities, enthalpies, concentrations, etc.) per frame versus the one-dimensional space variable.

Running times for a maximum sized problem on an IBM 360 vary from a fraction of a second to many seconds depending on the number of fission source and feedback loop iterations required per time point.

SUMMARY

The user is provided a brief description of the kind of problem the DWARF code will solve in an introduction that enumerates the major features of the program.

The mathematical problem is defined in detail in terms of the equations that must be simultaneously solved at each time point. The formulas are also given that are used to compute the neutron group constants, the neutron sources, the power distribution, and the thermal properties of the coolant and fuel. The iterative method of solution of the neutron diffusion equations and the thermal feedback equations is described in terms of the successive steps in the feedback iteration loop. The special iterative methods used to solve for the neutron flux, source, and power distributions on each pass through a feedback iteration are then described together with the options available to accelerate the convergence of these fission source iterations. The search options that can adjust various reactor parameters to maintain a desired reactor eigenvalue at each time point are described next. The equations used to compute the new nuclide concentrations for the next time point are also given together with detailed instructions to the user on how to define the nuclide production and depletion chains.

The program output edits are defined with instructions on how to call for the optional edits. The main edits consist of the processed input data, the iteration monitors, and all the processed problem solution data--including a diversity of plotted output data.

Much of the report consists of very detailed instructions for understanding and preparing the input data of a problem. The detailed tables are followed with a brief summary of the input data that experienced users will find sufficient for their purposes in setting up problems.

One section summarizes the flow of the program through all the major functions that the program overlays perform and it also presents special information relevant to the program language, storage requirements, and basic machine configuration requirements.

A detailed derivation of the difference equations and their solution for the neutron fluxes at each spatial mesh point is presented in Appendix B.

CONTENTS

ABSTRACT	ii
SUMMARY	iii
I. PROGRAM DESCRIPTION	1
II. PROBLEM DEFINITION AND SOLUTION METHODS	3
1. NEUTRON GROUP EQUATIONS AND BASIC DATA	3
1.1 The Multigroup Diffusion Equations	3
1.2 The Neutron Group Constants	6
1.3 The Feedback Equations	11
1.4 The Fixed Volume Source For a Group	17
1.5 The Fission Source For a Group	18
1.6 The Boundary Conditions For a Group	19
2. THE FLUX SOLUTION METHODS	22
2.1 The Fission Source Iterations	23
2.2 Fission Source Renormalization and Special Applications	25
2.3 Acceleration of the Iteration Convergence	27
3. AUTOMATIC SEARCHES FOR DESIRED REACTOR CONDITIONS	29
3.1 The Search Options	29
3.2 The Search Algorithms	31
3.3 Application of the Search Parameters	33
4. THE TIME DEPENDENT DEPLETION SOLUTIONS	36
III. PROGRAM OUTPUT	40
1. PROGRAM CHECKPOINTS	40
2. INPUT DATA CARD CONTENTS	40
3. PROCESSED INPUT DATA EDITS	40
4. ITERATION CONVERGENCE DATA EDITS	42
4.1 Fission Source Iteration Data	42
4.2 Feedback Loop Iteration Data	43
4.3 Search Iteration Data	43
5. PROBLEM SOLUTION EDITS	44

5.1	Nuclide Concentrations and Cross Section Contributions of Library Sets	44
5.2	Eigenvalue and Source Integral Summaries	44
5.3	Fission Source, Power, Group Fluxes and Total Flux	45
5.4	Peak-to-Average Ratios for Fluxes, Source, and Group	46
5.5	Neutron Balances and Averages by Region and Group	47
5.6	Net Currents and Leakage at Region Boundaries	48
5.7	Plotted Output	48
IV.	INPUT DATA PREPARATION	51
1.	GENERAL COMPOSITION AND ORDER OF THE INPUT DATA	51
2.	DETAILED DESCRIPTION OF THE INPUT DATA	52
2.1	The Time Range Title Card	52
2.2	The Time Range Integer Input Data	53
2.3	The Time Range Floating Point Input Data	64
2.4	The Library Set Title Card	74
2.5	The Library Set Integer Point Data	74
2.6	The Library Set Floating Point Data	74
3.	INPUT DATA SUMMARY	79
V.	PROGRAM CHARACTERISTICS, OPERATIONAL INFORMATION, AND MACHINE REQUIREMENTS	84
VI.	REFERENCES	88
APPENDIX A -- GLOSSARY		A-1
APPENDIX B -- DERIVATION AND SOLUTION OF THE DIFFERENCE EQUATIONS FOR A NEUTRON GROUP		B-1

FIGURES

1.	A Typical Nuclide Production and Depletion Scheme	37
2.	DWARF Program Flow	85

TABLES

I.	The External Boundary Condition Options	21
II.	Detailed Description of Time Range Integer Input Data	55
III.	Detailed Description of Time Range Floating Point Input Data	65
IV.	Detailed Description of Library Set Integer Input Data	76
V.	Detailed Description of Library Set Floating Point Data	77
VI.	The Time Range Title Card Summary	79
VII.	Summary of Time Range Integer Input Data	79
VIII.	Summary of Time Range Floating Point Input Data	81
IX.	The Library Set Title Card Summary	83
X.	Summary of Library Set Integer Input Data	83
XI.	Summary of the Library Set Floating Point Data	83

I. PROGRAM DESCRIPTION

The DWARF code is a few group (up to four), one-dimensional (1-D), neutron diffusion program with capability for both very general nuclide depletion and space dependent feedback. It is a combination of the CORA^[1] 1-D diffusion program and the SINBAD^[2] nuclide depletion program and allows the user wide flexibility in modeling spatially dependent reactivity feedback effects.

The code operates on IBM 360 Series machines with a requirement for about 380K bytes of core storage. It handles slab, cylindrical, or spherical geometries. The user-specified nuclide depletion transition schemes are solved analytically for each spatial mesh point at each time point. The chief approximation is that the neutron group fluxes remain constant at the initial values during a time interval. For reactivity feedback, the neutron group constants can be user specified functions of up to three spatially dependent feedback variables. The two variables in the current version of the code are the average fuel temperature and the coolant density. An iterative process can obtain the simultaneous solutions for the power, fluxes, feedback variables, etc., at each time point to an arbitrary degree of convergence.

The current version allows:

- 250 mesh points
- 40 regions
- 4 neutron groups
- 40 macroscopic (nonvariable) materials
- 10 macroscopic feedback-variable materials
- 40 depletion materials or microscopic library sets with independent transition schemes
- 50 nuclides per microscopic library set

(The neutron group constants in a region are a mixture of macroscopic material data and depleting microscopic material data, if any, in a region.) Basic input data and material constants can be changed freely at predetermined times throughout a time dependent problem. There are automatic searches on buckling, poison, region dimensions, total power, and a region interface position. Output options allow either pointwise values or simply region averages to be edited for concentrations and cross sections at a time point. The usual feedback data edited are the distributions of the density, enthalpy, temperature, and quality of the coolant, the fuel temperature, and the power density. Flux and source distributions; peak-to-average ratios for fluxes, sources and powers; neutron balance data by group and region, etc. can also be edited. A plot routine allows up to three spatial distributions per frame to be plotted, where the spatial distributions can include concentrations, fluxes, feedback variables, power, coolant and fuel properties, cross sections, etc.

Running times vary widely with problem sizes and convergence rates (of both the fission source iterations and the iterations through the feedback loop). On an IBM 360/75 with MVT, CPU time requirements have ranged from a fraction of a second to several seconds per time step.

The input data formats for the neutron group constants of the nonvariable materials are compatible with the punched output from the cross section codes PHROG^[3] and INCITE^[4]. The formats are therefore compatible with the input for the 1-D diffusion codes CORA^[1] and MONA^[5] except that DWARF allows scattering only to the next lowest energy group.

II. PROBLEM DEFINITION AND SOLUTION METHODS

1. NEUTRON GROUP EQUATIONS AND BASIC DATA

1.1 The Few Group Diffusion Equations

The range of the spatial variable $x \{x_1, x_{i_{\max}}\}$ in a DWARF problem is first divided into a number of "regions". The x range of a region is then divided into a number of equal mesh intervals that determine the basic spatial mesh points x_i of the problem. A difference equation for each mesh point is derived that is solved simultaneously with those of all the other mesh points to obtain a value of the neutron flux (for a given group) for each mesh point.

The time span of a problem also consists of one or more "time ranges". Each time range is subdivided into a number of equal "time steps" that comprise the basic time intervals of the problem. That is, solutions of the equations of a problem are obtained and output edits are made at the end of each time step (in addition to the initial time point of a time range).

DWARF is a reactor depletion program as opposed to a kinetics program in that the time independent forms of the neutron group equations are solved. That is, there are no time derivatives or delayed neutrons represented. Instead, quasi steady state solutions of the group equations are obtained that change in time solely in response to the changes in the concentrations of the nuclides represented in the problem. The solutions for the new spatially dependent concentrations obtained at each time point are analytic solutions that are rigorous within the limits of the assumption that the fluxes remain constant throughout the time step at their initial values.

The few group diffusion theory equations solved at each time point can be written in matrix form as

$$\bar{L} \cdot \bar{\phi} = \frac{\bar{X} \bar{F} \bar{\phi}}{\lambda} + \bar{S} e \quad (1)$$

for the usual real or direct flux problem.

$$\bar{L} \equiv \begin{pmatrix} (-\nabla \cdot D_1 \nabla + \Sigma_{r1}) & 0 & \dots & 0 & 0 \\ \Sigma_{s1} & (-\nabla \cdot D_2 \nabla + \Sigma_{r2}) & & & \\ 0 & \Sigma_{s2} & \ddots & & \\ \vdots & \vdots & \ddots & (-\nabla \cdot D_{j_{\max}-1} \nabla + \Sigma_{rj_{\max}-1}) & 0 \\ 0 & 0 & \dots & \Sigma_{sj_{\max}-1} & (-\nabla \cdot D_{j_{\max}} \nabla + \Sigma_{rj_{\max}}) \end{pmatrix}, \quad (2)$$

where $\Sigma_{rj} \equiv \Sigma_{aj} + \Sigma_{sj} + T_j \Sigma_p + D_j B_j^2.$ (3)

$$\bar{X} \equiv \begin{bmatrix} x_1 \\ x_2 \\ \vdots \\ x_{j_{\max}} \end{bmatrix} \cdot \bar{F} \equiv |\nu \Sigma_{f1} \nu \Sigma_{f2} \dots \nu \Sigma_{f_{j_{\max}}}|. \quad (4),(5)$$

$$\bar{\phi} \equiv \begin{bmatrix} \phi_1 \\ \phi_2 \\ \vdots \\ \phi_{j_{\max}} \end{bmatrix} \cdot \bar{S}_e \equiv \begin{bmatrix} S_{e1} \\ S_{e2} \\ \vdots \\ S_{ej_{\max}} \end{bmatrix} \quad (6),(7)$$

Equation (1) is solved subject to the boundary conditions

$$\bar{\alpha}_1 \bar{\phi}_1 - \bar{\beta}_1 \bar{J}_1 + \bar{\gamma}_1 = 0 \text{ and } \bar{\alpha}_2 \bar{\phi}_{i_{\max}} - \bar{\beta}_2 \bar{J}_{i_{\max}} + \bar{\gamma}_2 = 0, \quad (8),(9)$$

or, for the special periodic boundary condition case,

$$\bar{\phi}_1 = \bar{\phi}_{i_{\max}} \text{ and } x_1^{\rho} \bar{J}_1 = x_2^{\rho} \bar{J}_2, \quad (10),(11)$$

where

$$\bar{\alpha} \equiv \begin{bmatrix} \alpha_1 & 0 & \dots & 0 \\ 0 & \alpha_2 & \dots & 0 \\ \vdots & \vdots & \ddots & \vdots \\ \vdots & \vdots & \vdots & \vdots \\ 0 & 0 & \dots & \alpha_{j_{\max}} \end{bmatrix}, \quad \bar{\beta} \equiv \begin{bmatrix} \beta_1 & 0 & \dots & 0 \\ 0 & \beta_2 & \dots & 0 \\ \vdots & \vdots & \ddots & \vdots \\ \vdots & \vdots & \vdots & \vdots \\ 0 & 0 & \dots & \beta_{j_{\max}} \end{bmatrix}, \quad (12),(13)$$

$$\bar{\gamma} = \begin{bmatrix} \gamma_1 \\ \gamma_2 \\ \vdots \\ \gamma_{j_{\max}} \end{bmatrix}, \text{ and } \bar{J} = \begin{bmatrix} J_1 \\ J_2 \\ \vdots \\ J_{j_{\max}} \end{bmatrix}. \quad (14),(15)$$

The usual problem is a "renormalization" type of problem as signified by the integer input flag IREN=0. In this type of problem, a generation factor (or eigenvalue) λ is computed from

$$\lambda = \int_V (\bar{F}\bar{\phi}/F_0) dv. \quad (16)$$

Since λ divides the source in Equation (1), there is imposed the constraint that

$$\frac{1}{\lambda} \int_V \bar{F}\bar{\phi} dv = F_0. \quad (17)$$

The F_0 is a fixed quantity that is usually unity, but can be set to a given value with the floating point input word FISSOR. After the solution to Equation (1) is obtained for the time point (see Section 2.), the fluxes that are finally used to compute new concentrations in a depletion problem are multiplied (renormalized) by a constant C_p such that

$$\frac{C_p}{\lambda} \int_V \sum_j E \Sigma_{fj} \phi_j dv = P_{av}, \quad (18)$$

where P_{av} is the input quantity AVPOWR.

If the user sets the flag IREN=1 and there exists an extraneous source in the form of either a boundary source or a fixed volume source ($\bar{S}_e > 0$), Equation (1) with Equations (8) and (9) or (10) and (11) are solved with λ set to unity. In this case, the constraints of Equations (17) and (18) are not applied.

Sections 1.4 and 1.5 describe the fixed volume source $\bar{S}_e$ and the fission neutron source in greater detail, while Section 1.6 describes the detailed application of the boundary conditions.

Adjoint flux solutions can be obtained at the beginning or end of a time range by the artifice of running a dummy time range; that is, one of zero time width. In such cases the feedback routines should be by-passed (by setting IFEEED=0) so that no feedback vectors* (ρ_1, ρ_2, ρ_3) are re-computed as a function of the adjoint fluxes.

The adjoint fluxes are the solutions of

$$\bar{L}^T \cdot \bar{\phi}^* = \frac{\bar{F}^T \cdot \bar{\chi}^T}{\lambda} \bar{\phi}^* + \bar{S}_e, \quad (19)$$

where the T denotes a transposed matrix and the asterisk denotes the adjoint flux. The boundary conditions are the same as for the real flux problem.

*The term "feedback vector" is used to denote a spatial distribution ρ (in terms of values at the mesh points) that is computed in the feedback routines and that is used as one of the independent variables in computing most of the neutron group constants at a mesh point.

1.2 The Neutron Group Constants

The neutron group constants for a group j are Σ_{aj} , $\nu\Sigma_{fj}$, Σ_{fj} , E_j , Σ_{sj} , $\Sigma_{\mu j}$, B_j^2 , D_j , T_j , Σ_p , χ_j , and q_j^* . Of these, T_j , Σ_p , χ_j , and q_j are functions of neither any nuclide concentration nor any feedback vector.

In a DWARF problem, the user must assign to each region a macroscopic (macro) material number. The same macro material number can be assigned to any number of regions. The macro material number for region number IR (regions are numbered in the order of increasing x) is input as $MR(IR)$. Macro materials numbered less than or equal to 40 are not functions of the feedback vectors. Such materials are referred to as fixed macro (f. macro) materials. Those macro materials with numbers from 41 through 50 are functions of the feedback vectors and they are called variable macroscopic (v. macro) materials. The group constants in a region are, in general, (1) a function of macroscopic data read for the macro material assigned to the region, (2) a function of a set of microscopic (library set) data and the corresponding nuclide concentrations, or (3) both. If a region IR has time dependent group constants (because of time dependent nuclide concentrations in the region), then the user must assign a library set number (from 1 to 40) to the region in addition to the macro material number assigned. A library set number for region IR is entered as $MRMIC(IR)$.

For each library set number assigned in the problem, the user must supply (in a separate library set data deck): a list of the nuclides in the library set; the microscopic (micro) group constant data of each nuclide; the data describing all the transition schemes linking the productions and losses of the nuclides; and all other data needed for the transition scheme solutions such as decay constants and yields, capture transition cross sections, fission product yields, etc. Each nuclide in a library set is assigned an arbitrary number (from 1 to 50); and, since each library set is independent of any other, a nuclide, such as U-235 can have a different number in different library sets or two different nuclides might be assigned the same number in different library sets.

When the group constants are computed as a function of the feedback vectors or the nuclide concentrations, they are computed at every "source point". There is a source point, indexed by k , for every mesh point and an extra point for each region interface; hence, the source point index of mesh point i in region r is $k = i + r - 1$.

The detailed formulas for the group constants at a source point k are as follows.

*In any material that is not a function of a feedback vector (ρ_1, ρ_2, ρ_3), an internal boundary condition parameter q_j can be used to force a desired boundary condition at the surfaces of the regions in which the material occurs. In such regions, the constant q_j replaces all other group constants in the sense that no other data are relevant to the group flux solution in such cases.

First, for v. macro material m ($41 \leq m \leq 50$), the following arrays and the corresponding floating point input data are defined.

$$a_a(n, j, m') \equiv \text{ACOF}(N, J, M) \quad (\text{F-14, IM, J, N})^*; \quad (20)$$

$$a_v(n, j, m') \equiv \text{FNUCOF}(N, J, M) \quad (\text{F-15, IM, J, N}); \quad (21)$$

$$a_f(n, j, m') \equiv \text{FISCOF}(N, J, M) \quad (\text{F-16, IM, J, N}); \quad (22)$$

$$a_s(n, j, m') \equiv \text{SDCF}(N, J, M) \quad (\text{F-17, IM, J, N}); \quad (23)$$

$$a_\mu(n, j, m') \equiv \text{SCOF}(N, J, M) \quad (\text{F-18, IM, J, N}); \quad (24)$$

$$a_b(n, j, m') \equiv \text{BCOF}(N, J, M) \quad (\text{F-19, IM, J, N}); \quad (25)$$

$$a_d(n, j, m') \equiv \text{DCOF}(N, J, M) \quad (\text{F-13, IM, J, N}); \text{ and} \quad (26)$$

$$b(n', k', m') \equiv \text{COFMOD}(N', K, M) \quad (\text{F-20, IM, K, N}); \quad (27)$$

where $m' = m - 40$, $N = n \leq 4$; $J = j$; $M = m'$; $\text{IM} = m$ (hence $M = \text{IM} - 40$); $N' = n' \leq 5$, and $K = k'$. The dimensions of these arrays are presently $N = 4$, $J = 4$, $M = 10$ for ACOF through DCOF and $N' = 5$, $K = 7$, and $M = 10$ for COFMOD. As stated earlier, m is the number $\text{MR}(\text{IR})$ assigned to region $r = \text{IR}$.

Next, define a special function $h_{k, k', m}$:

$$h_{k, k', m} \equiv 1 \text{ for } m \leq 40; \quad (28)$$

$$h_{k, k', m} \equiv h[\rho_2(k), b(n', k', m')]^{**} \text{ for } m > 40. \quad (29)$$

The function of Equation (29) is represented in the program by the Fortran function $[\text{FMOD}(V, \text{COFMOD}, M, K)$; $V = \rho_2(k)$, $M = m' = m - 40$, $K = k'$].

Finally, define L_r as the set of nuclides specified for the library set assigned to region r . Then the group constants at source point k are:

$$\Sigma_{aj}(k) \equiv \left[\Sigma_{aj}^m(k) + \sum_{\ell \in L_r} N_\ell(k) \sigma_{aj}^\ell \right] h_{k, 2, m}, \quad (30)$$

where

*The expression, (F-14, IM, J, N), denotes time range floating point data corresponding to Index 1 = 14, Index 2 = IM, Index 3 = J, and Index 4 = N.

**In the present version of DWARF, the h function is a fourth order

polynomial; i.e., $h[\rho_2(k), b(n', k', m')] = \sum_{n'=1}^5 b(n', k', m') \rho_2^{n'-1}(k)$.

$$\Sigma_{aj}^m(k) \triangleq [\text{SIGA}(J, IM); J = j; IM = m] \text{ for } m \leq 40, \text{ and} \quad (31)$$

$$\Sigma_{aj}^m(k) \equiv g_a[\rho_1(i), a_a(n, j, m')]^*. \quad (32)$$

This g_a function is represented in the program by the Fortran function $[\text{ABSF}(V, \text{ACOF}, M, J); V = \rho_1(i); M = m'; J = j]$.

$$v\Sigma_{fj}^m(k) = \left[v\Sigma_{fj}^m(k) + \sum_{\ell \in L_r} N_\ell(k) v\sigma_{fj}^\ell \right] h_{k,3,m}, \quad (33)$$

where

$$v\Sigma_{fj}^m(k) \triangleq [\text{SIGNU}(J, IM); J = j; IM = m] \text{ for } m \leq 40, \text{ and} \quad (34)$$

$$v\Sigma_{fj}^m(k) \equiv g_v[\rho_1(i), a_v(n, j, m')] \text{ for } m > 40. \quad (35)$$

This g_v function is represented by the Fortran function $[\text{FNUF}(V, \text{FNUCOF}, M, J); V = \rho_1(i); M = m'; J = j]$.

$$E\Sigma_{fj}^m(k) \equiv \left[E_j^m \Sigma_{fj}^m(k) + \sum_{\ell \in L_r} N_\ell(k) E_j^\ell \sigma_{fj}^\ell \right] h_{k,4,m}, \quad (36)$$

where

$$E_j^m \triangleq [\text{ENERGY}(J, IM); J = j; IM = m], \quad (37)$$

$$\Sigma_{fj}^m(k) \equiv [\text{SIGFIS}(J, IM); J = j; IM = m] \text{ for } m \leq 40, \text{ and} \quad (38)$$

$$\Sigma_{fj}^m(k) \equiv g_f[\rho_1(i), a_f(n, j, m')] \text{ for } m > 40. \quad (39)$$

This g_f function is the Fortran function $[\text{FISF}(V, \text{FISCOF}, M, J); V = \rho_1(i); M = m'; J = j]$.

*In the present version of DWARF, all the g functions are third order polynomials; i.e., $g_x[\rho_1(i), a_x(n, j, m')] = \sum_{n=1}^4 a_x(n, j, m') \rho_1^{n-1}(i)$. Also, $h_{k,k',m}$ is applied as a correction factor on resonance energy group constants, only. That is, $h_{k,k',m}$ is assumed to be unity except in group $j_{\max}-1$, the one just above the thermal group.

$$\Sigma_{sj_{\max}}(k) \equiv 0. \quad (40)$$

For $j < j_{\max}$,

$$\Sigma_{sj}(k) \equiv \left[\Sigma_{sj}^m(k) + \sum_{\ell \in L_r} N_{\ell}(k) \sigma_{sj}^{\ell} \right] h_{k,5,m}; \quad (41)$$

where

$$\Sigma_{sj}^m(k) \triangleq [\text{SIGSD}(J, IM); J = j; IM = m] \text{ for } m \leq 40, \text{ and} \quad (42)$$

$$\Sigma_{sj}^m(k) \equiv g_s[\rho_1(i), a_s(n, j, m')] \text{ for } m > 40. \quad (43)$$

This g_s function is the Fortran function $[\text{SDSF}(V, \text{SDCF}, M, J); V = \rho_1(i); M = m'; J = j]$.

$$\Sigma_{\mu j}(k) \equiv \Sigma_{\mu j}^m(k) + \sum_{\ell \in L_r} N_{\ell}(k) \sigma_{\mu j}^{\ell} h_{k,6,m}, \quad (44)$$

where

$$\Sigma_{\mu j}^m(k) \triangleq [\text{SIGMU}(J, IM); J = j; IM = m] \text{ for } m \leq 40. \quad (45)$$

For $m > 40$, $\Sigma_{\mu j}^m(k)$ is computed from

$$\Sigma_{\mu j}^m(k) \equiv g_{\mu}[\rho_1(i), a_{\mu}(n, j, m')] h_{k,6,m} \text{ if } a_{\mu}(n, j, m') \neq 0 \text{ for} \quad (46)$$

some n , or from

$$\Sigma_{\mu j}^m(k) \equiv 1/\{3g_d[\rho_1(i), a_d(n, j, m')] h_{k,1,m}\} \text{ if } a_{\mu}(n, j, m) = 0 \text{ for} \quad (47)$$

all n , and

$$a_d(n, j, m') \neq 0 \text{ for some } n.$$

If both $a_{\mu}(n, j, m')$ and $a_d(n, j, m')$ are zero for all n , then

$$\Sigma_{\mu j}^m(k) \equiv 0. \quad (48)$$

The g_{μ} function is the Fortran function $[\text{SCSF}(V, \text{SCOF}, M, J); V = \rho_1(i); M = m'; J = j]$. The g_d function is described in the discussion of the diffusion coefficients, below.

The calculation of the diffusion coefficient is controlled by the IDCO flag and the IYUK(K) list. The IDCO flag specifies which of three formulas to use to compute the D's:

When IDCO = 0,

$$D_j(k) \equiv D_j^m(k) h_{k,1,m}, \quad (49)$$

where

$$D_j^m(k) \triangleq [DC(J,IM); J=j; IM=m] \text{ for } m \leq 40, \text{ and} \quad (50)$$

$$D_j^m(k) \equiv g_d[\rho_1(i), a_d(n,j,m')] \text{ for } m > 40. \quad (51)$$

The g_d function is the Fortran function [DCSF(V,DCOF,M,J); $V = \rho_1(i)$; $M = m'$; $J = j$].

When IDCO = 1,

$$D_j(k) \equiv 1/[3\Sigma_{\mu j}(k) + 3T_j^m \Sigma_p^m], \quad (52)$$

where

$$T_j^m \triangleq [T(J,IM); J=j; IM=m], \text{ and } \Sigma_p^m \triangleq [SIGPT(J,IM); J=j, IM=m]. \quad (53), (54)$$

When IDCO = 2, $D_j(k)$ is computed from

$$D_j(k) \equiv \Sigma_{rj}'(k)/\kappa_j^2, \quad (55)$$

where κ_j is the solution of the transcendental equation

$$\kappa_j/\Sigma_j(k) = \tanh \left\{ \kappa_j/[\Sigma_j(k) - \Sigma_{rj}'(k)] \right\}, \quad (56)$$

with

$$\Sigma_j(k) \equiv \Sigma_{\mu j}(k) + T_j^m \Sigma_p^m, \text{ and} \quad (57)$$

$$\Sigma_{rj}'(k) \equiv \Sigma_{aj}(k) + \Sigma_{sj}(k) + T_j^m \Sigma_p^m. \quad (58)$$

The formula specified by the IDC0 flag is applied in all materials of the problem unless there are one or more entries in the IYUK(K) list and IDC0 > 0. In the latter case, the specified formula is applied only in the materials listed in the IYUK(K) list and the formula corresponding to IDC0 = 0 is applied elsewhere.

$$B_j^{2r}(k) \equiv B_j^{2r}(k) h_{k,7,m}, \quad (59)$$

where

$$B_j^{2r}(k) \equiv g_b[\rho_1(i), a_b(n,j,m')] \text{ for } m > 40, \text{ and } \text{INBUCK} \leq 1; \quad (60)$$

otherwise,

$$B_j^{2r}(k) \equiv B_j^{2r}. \quad (61)$$

The g_b function is the Fortran function [BCSF(V,BCOF,M,J); V = $\rho_1(i)$; M = m' ; J = j]. The value of B_j^{2r} is determined by the INBUCK and IBSQ0 flags as shown in the table below.

I N B U C K							
	0	1	2	3	4		
$B_j^{2r} \triangleq$	BSQM(J,IM); J = j; IM = m	BSQR(IM); IM = m	BSQM(J,1); J = j	BSQM(1,1)	same as previous time range	0	IBSQ0
$B_j^{2r} \triangleq$	BSQM(J,IR); J = j; IR = r	BSQR(IR); IR = r	BSQM(J,1); J = j	BSQM(1,1)	same as previous time range	1	

1.3 The Feedback Equations

1.3.1 Calculation of q^u and H_j . As described in Section 1.2, space dependent feedback is effected through the dependency of the neutron group constants for the v. macro materials (numbered 41 through 50) on one or more of the feedback vectors ρ_1 , ρ_2 , and ρ_3^* .

The spatial distributions of the ρ 's are all computed in a single subroutine (FEEDER). For a given time point, the program first iteratively solves Equation (1) for the fluxes using the appropriate nuclide concentrations, but using the most recent values for the ρ distributions.

*In the present version of DWARF, only the first two feedback vectors, ρ_1 and ρ_2 are used.

(These most recent values are those obtained for the last time point or, at the initial time point of the problem, the values input by the user.) Then, if the input flag IFEED is 1, 2, or 3 the feedback routine is entered and new ρ 's are computed using the new power distribution $q^u(k)$, where u is the feedback iteration number (the number of times the feedback routine is entered on a time step) and k is the source point index $k=i+r-1$. The program then recomputes the fluxes in Equation (1) with the new ρ 's. At this point, the program checks to see if

$$| [q^u(k) - q^{u-1}(k)] / q^u(k) |_{\max} < \epsilon_4, \text{ or if } u = u_{\max}, \quad (62)$$

where $\epsilon_4 \triangleq \text{EPS4}$ (F-0,0,4,0) and $u_{\max} \triangleq \text{NFEEDL}$ (I-0,0,57,0). If either condition is satisfied, the program begins the calculations for the next time point unless a parameter search is called for (see Section III.). Hence, the fluxes obtained after feedback iteration are ϕ_j^{u+1} . That is, there is always one more flux calculation than the number of feedback vector calculations.

In the feedback routine, the formula for $q^u(k)$ is

$$q^u(k) = \frac{C_p}{\lambda} \sum_j E_{\Sigma f j}(k) \phi_{j i}, \quad (63)$$

where the $E_{\Sigma f j}(k)$ is computed from Equation (36) and where C_p is obtained (in the usual problem with $\text{IREN}=0$) from Equation (18). That is,

$$C_p = \lambda P_{av} / \int_{x_1}^{x_I} \sum_j E_{\Sigma f j} \phi_j x^\rho dx, \quad (64)$$

such that

$$\int_{x_1}^{x_I} C_p q^u(k) x^\rho dx = P_{av}. \quad (65)$$

Here, I denotes $i_{\max}$ and P_{av} is the floating point input word AVPOWR. As stated before, in inhomogeneous problems with $\text{IREN}=1$, $C_p = \lambda = 1$.

Since q , itself, is a function of the feedback vectors [by virtue of Equation (36) for $E_{\Sigma f j}(k)$] there is an iterative loop within the feedback subroutine that recomputes the $E_{\Sigma f j}(k)$'s, the q , and the ρ 's, in that order, until the following convergence criterion is satisfied:

$$\sum_k |q^u(k) - q^{u-1}(k)| \sum_k q(k) \leq \epsilon_5, \quad (66)$$

where the $q^{u-1}(k)$ is the old value and ϵ_5 is the floating point input word EPS5.

In the discussion to follow, the presently built-in algorithms and options for computing the feedback vectors are described. The user should keep in mind that other methods for computing the ρ 's can be easily substituted.

After computing q^u for each source point, the program computes a coolant enthalpy H_i for each mesh point under the assumed condition that the coolant flows from x_1 to x_I in the reactor, picking up all the energy being generated as it goes. The formula is

$$H_i = H_0 + (K_1/w_c) \int_{x_1}^{x_I} q^u dv = H_0 + (K_1/w_c) \int_{x_1}^{x_I} C_p q^u x^\rho dx \text{ (Btu/lb)*}, \quad (67)$$

where

H_0 = the inlet enthalpy of the coolant (Btu/lb) $\triangleq$ HINLET
(F-0,0,21,0)

K_1 = energy conversion factor (Btu/w-sec) $(9.48 \times 10^{-4}) \triangleq$ HCONV
(F-0,0,24,0)

w_c = mass flow rate of the coolant (lb/sec-cm² of core)** $\triangleq$ FLOW
(F-0,0,25,0)

The program then computes the distribution of the first feedback vector $\rho_1(i)$ according to whether IFEED is 1, 2, or 3 as follows.

1.3.2 Calculation of $\rho_1(i)$. When IFEED=1, $\rho_1(i)$ is considered the coolant density that is computed in a different manner according to whether the coolant is in the subcooled region, the transition region between nucleate boiling and bulk boiling, the bulk boiling region, or the superheated vapor region.

The subcooled region is identified by the fact that $H_i \leq H_{f\text{sat}}$ (Btu/lb) where $H_{f\text{sat}} \triangleq$ HFSAT, the enthalpy of the saturated fluid. In this region the quantities computed are:

*While the user is free to use any consistent set of units, the units used in the discussions are shown to help clarify the formulas and to serve as an example.

**The units shown for w_c are those appropriate for a slab geometry problem where the coolant flow is along the axis of the reactor, P_{av} is the power (watts) per cm² of core area perpendicular to the flow (axis), q^u is the power density (watts/cm²). The w_c is the product of the mass flow rate per unit area of flow channel and the flow channel area per unit area of core.

In cylindrical geometry, w_c is the mass flow rate per unit length of core in the radially outward direction, P_{av} is the power per unit length of core, and, again, q^u is the power density (watts/cm³).

In spherical geometry, w_c is the total mass flow rate (lb/sec) and P_{av} is the total power with q^u still the power density.

$$\rho_1(i) = P_\rho(H_i), \text{ the unsaturated fluid density (gm's/cm}^3\text{);} \quad (68)$$

$$T_\ell(i) = P_t(H_i), \text{ the unsaturated fluid temperature (}^\circ\text{F);} \quad (69)$$

$$Q(i) = 0., \text{ the quality;} \quad (70)$$

$$T_c(i) = T_\ell(i) + K_1 q^U (A_c h_c), \text{ the fuel cladding surface temperature (}^\circ\text{F).} \quad (71)$$

The functions $P_\rho(H_i)$ and $P_t(H_i)$ are fifth order polynomials of the enthalpy.

$$P_\rho(H_i) = \sum_{n=1}^6 a_\rho(n) H_i^{n-1}, \quad (72)$$

where $a_\rho(n) \triangleq \text{RHOLCS}(N) \text{ (F-0,6,1,N); } N = n.$

$$P_t(H_i) = \sum_{n=1}^6 a_t(n) H_i^{n-1}, \quad (73)$$

where $a_t(n) \triangleq \text{TEMPCS}(N) \text{ (F-0,6,2,N); } N = n.$

Again K_1 (Btu/w-sec) is the energy conversion factor. The area A_c (ft²/cm³) is the surface area of fuel clad per cm³ of core. $A_c \triangleq \text{AREA}$ (F-0,0,28,0). The h_c (Btu/ft²-sec-°F) is the film coefficient for heat transfer from the cladding surface to the bulk coolant. The cladding surface temperature $T_c(i)$ is computed in order to identify the start of nucleate boiling. At the first mesh point where $T_c(i) \geq T_{\text{sat}}$, the mesh point is stored as *inb*. The saturated fluid temperature T_{sat} is computed from

$$T_{\text{sat}} = P_t(H_{\text{fsat}}). \quad (74)$$

At the first mesh point where $H_i > H_{\text{fsat}}$, the mesh point is stored as *ibb*, the start of the bulk boiling region. In this region the quantities computed are:

$$Q(i) = (H_i - H_{\text{fsat}})/H_{\text{fg}}, \quad (75)$$

where H_{fg} (Btu/lb) is the enthalpy rise in going from the saturated fluid to vapor. $H_{\text{fg}} \triangleq \text{HFS}$ (F-0,0,23,0).

$$\rho_1(i) = \alpha_i \rho_g + (1 - \alpha_i) \rho_\ell, \quad (76)$$

where ρ_g (gms/cm³) $\triangleq$ GASRHO (F-0,0,26,0) is the saturated vapor density, ρ_ℓ (gms/cm³) is the saturated fluid density computed from

$$\rho_\ell = P_p(H_{fsat}), \quad (77)$$

and α_i is the vapor fraction computed from

$$\alpha_i = [0.833 + 0.167 Q(i)][Q(i)/\rho_g] / \{Q(i)/\rho_g + [1-Q(i)]/\rho_\ell\}. \quad (78)$$

The formula for the vapor fraction α is from the Armand correlation^[6]. The coolant temperature in the bulk boiling region is simply

$$T_\ell(i) = T_{sat}. \quad (79)$$

In the superheated vapor region, identified by the fact that $H_i \geq H_{fsat} + H_{fg}$, the quantities computed are

$$Q(i) = 1, \quad (80)$$

$$\rho_1(i) = \rho_v(i), \text{ and} \quad (81)$$

$$T_\ell(i) = \rho_g T_{sat} / \rho_v(i), \text{ where} \quad (82)$$

$$\rho_v(i) = \rho_g (H_{fsat} + H_{fg} - H_C) / (H_i - H_C) \text{ (gms/cm}^3\text{)}. \quad (83)$$

In Equation (83) for the super heated vapor density $\rho_v(i)$, the constant $H_C \triangleq HCON$ (F-0,0,32,0) is an arbitrary number used to help the formula fit the vapor density function at the given system pressure.

The $\rho_1(i)$ values are as shown in Equations (68) and (76) for the subcooled nucleate boiling region and the bulk boiling region, respectively, unless the user supplies values for $H_t \triangleq HFRAC$ (F-0,0,31,0) and $\alpha_t \triangleq CRITV$ (F-0,0,30,0). If these values are nonzero, the program uses them to identify a transition region extending from a mesh point $it1$ in the nucleate boiling region to a mesh point $it2$ in the bulk boiling region. The mesh point in the nucleate boiling region is determined as the first mesh point at which

$$H_i \geq H_t (H_{ibb} - H_{inb}), \quad (84)$$

where $H_{ibb} - H_{inb}$ is the enthalpy rise achieved in the nucleate boiling region, as follows from the definitions of the stored mesh points, ibb and inb .

The mesh point $it2$ in the bulk boiling region is the first mesh point where

$$\alpha_i \geq \alpha_t. \quad (85)$$

The program then recomputes the $\rho_1(i)$ values for all the mesh points between $it1$ and $it2$ by linear interpolation between the two densities $\rho_1(it1)$ and $\rho_1(it2)$ as follows

$$\rho_1(i) = \rho_1(it1) + [\rho_1(it2) - \rho_1(it1)](H_i - H_{it1}) / (H_{it2} - H_{it1}). \quad (86)$$

Obviously, H_t and α_t should be values less than unity to avoid extrapolation of densities outside the nucleate boiling and bulk boiling regions.

When $IFEED=2$, $\rho_1(i)$ is simply set to the enthalpy.

$$\rho_1(i) = H_i. \quad (87)$$

When $IFEED=3$, $\rho_1(i)$ is set to q^u as follows in a region r :

$$\rho_1(i) = q^u(k) \quad (88)$$

for all $k=i+r-1$ except at region interfaces. At region interfaces between region $r-1$ and region r ,

$$\rho_1(i) = [q^u(k-1) + q^u(k)]/2. \quad (89)$$

1.3.3 Calculation of $\rho_2(i)$. If $1 \leq IFEED \leq 3$ and $1 \leq JFEED \leq 3$, the program computes the second feedback vector $\rho_2(k)$ in addition to $\rho_1(i)$.

When $JFEED=1$, $\rho_2(k)$ is considered the effective fuel temperature for use in taking into account such effects as the Doppler broadening of resonances and the temperature expansion of the fuel elements. The formula used for a point in region r is

$$\rho_2(k) = T_g(i) + K_1 q^u(k) / (A_c h_f), \quad (90)$$

where $k=i+r-1$, and at a region interface, where q^u is double valued, the q^u value used is the upper value, or, the one of higher k . The K_1 and A_c values are as defined in Section 1.3.2 and h_f (Btu/ft²-sec-°F) $\triangleq$ HFUEL (F-0,0,37,0) is the effective fuel heat transfer coefficient relating the heat flux at the clad surface to the difference between the effective fuel temperature and the bulk coolant temperature. That is,

$$q'(\text{Btu/ft}^2\text{-sec}) = [T_{fue1}(k) - T_g(i)](^{\circ}\text{F}) h_f(\text{Btu/ft}^2\text{-sec-}^{\circ}\text{F}), \quad (91)$$

where $T_{fue1}(k)$ is the effective fuel temperature and q' is the heat flux at the clad surface

$$q' = q^U(k)(w/cm^3) K_1 (Btu/w-sec)/[A_C(ft^2/cm^3)]. \quad (92)$$

When JFEED=2, $\rho_2(k)$ is set to the enthalpy.

$$\rho_2(k) = H_i. \quad (93)$$

When JFEED=3, $\rho_2(k)$ is set to q^U .

$$\rho_2(k) = q^U(k) \text{ for } k = i+r-1. \quad (94)$$

1.3.4 Interpolation and Extrapolation of Feedback Vector Distributions.
After the necessary power and ρ distribution iterations have been performed to satisfy the convergence criterion of Equation (66), the values of the feedback vectors are extrapolated before leaving the feedback routine to return to the new calculation of the fluxes. The two-point formulas used are

$$\rho_1^U(i) = \rho_1(i) + \tau_1[\rho_1(i) - \rho_1^{U-1}(i)], \text{ and} \quad (95)$$

$$\rho_2^U(k) = \rho_2(k) + \tau_2[\rho_2(k) - \rho_2^{U-1}(k)], \quad (96)$$

where the extrapolation factors are $\tau_1 \triangleq \text{RHOEXT}(F-0,0,40,0)$ and $\tau_2 \triangleq \text{PHOEXT}(F-0,0,41,0)$ and $\rho_1(i)$ and $\rho_2(k)$ are the unextrapolated values computed with the formulas of Sections 1.3.2 and 1.3.3.

Values of τ between 0 and 1 can be used to accelerate the feedback iterations when convergence is slow. However, in some problems such as those for Xenon oscillation studies, the user may find it necessary to use negative values of τ (between -1 and 0) in order to prevent non-converging flip-flop of the power distribution from iteration to iteration.

1.4 The Fixed Volume Source for a Group

The elements of the fixed source vector $\bar{S}_e$ in Equation (1) are the group fixed volume sources S_{ej} . For a DWARF problem the S_{ej} for each group j is determined from the sum of two input quantities, S_{rj} and S_{pj} where S_{rj} is a source contribution constant in region r and S_{pj} is a spatially (pointwise) variable contribution. S_{pj} is input by source point. (There are also options to have S_{pj} computed automatically from

$$S_{pj} = x_j \phi_j, \quad (97)$$

where x_j is any one of a number of f. macro material group constants and ϕ_j is a flux either input directly or retained from the previous problem in the given computer run.

In any case, the fixed source S_{ej} for group j is always

$$S_{ej} = S_{rj} + S_{pj}. \quad (98)$$

Whenever a fixed volume source exists in a problem, the problem becomes an inhomogeneous one that may have no solution when the fission neutron multiplication of the source neutrons is greater than one. One should therefore exercise caution in fissioning problems when one makes the problem inhomogeneous by adding either fixed volume sources or sources that are specified by the boundary conditions.

1.5 The Fission Source for a Group

When Equation (1) or Equation (19) is expanded, the fission neutron source in a given group can be written

$$S_{fj} = x_j \frac{\bar{F} \cdot \bar{\phi}}{\lambda} = x_j \left[\sum_{j'} v \Sigma_{fj'} \phi_{j'} \right] / \lambda \text{ for a real flux problem, or} \quad (99)$$

$$S_{fj} = v \Sigma_{fj} \frac{\bar{X}^T \cdot \bar{\phi}}{\lambda} = v \Sigma_{fj} \left[\sum_{j'} x_{j'} \phi_{j'} \right] / \lambda \text{ for an adjoint problem.} \quad (100)$$

As stated earlier, λ is either unity or computed from Equation (16) (for renormalization types of problems) as:

$$\lambda = \left[\int_V dv \sum_j v \Sigma_{fj} \phi_j \right] / F_0 \text{ for real flux problems, or} \quad (101)$$

$$\lambda = \left[\int_V dv \sum_j x_j \phi_j \right] / F_0 \text{ for adjoint problems.} \quad (102)$$

Together with Equations (101) and (102), there is also imposed the constraint

$$\int_V \frac{1}{\lambda} \sum_j v \Sigma_{fj} \phi_j dv = F_0 \text{ for a real flux problem, or} \quad (103)$$

$$\int_V \frac{1}{\lambda} \sum_j x_j \phi_j dv = F_0 \text{ for an adjoint problem.} \quad (104)$$

This constraint allows a definition of λ even in inhomogeneous (extraneous source and sink) types of problems, as discussed in Sections 2.1 and 2.2 and Reference 7. Also discussed in these sections to follow is the fact that on a given outer iteration t , the neutron source from fissions is computed using fluxes from the $t-1$ iteration; hence, the solution to Equation (1) is obtained by DWARF with the right hand side of Equation (1) known for each source iteration. That is, DWARF really solves, iteratively, the problem

$$\overline{L} \cdot \overline{\phi}^t = \frac{(\overline{X} \cdot \overline{F} \cdot \overline{\phi}^{t-1})'}{\lambda^{t-1}} + \overline{S}_e, \quad (105)$$

where λ^{t-1} is, again, either unity or

$$\lambda^{t-1} = \left[\int_V dv \sum_j v \Sigma_{fj} \phi_j^{t-1} \right] / F_0 \text{ (for a real flux problem)}. \quad (106)$$

The prime on the quantity $\overline{X} \cdot \overline{F} \cdot \overline{\phi}^{t-1}$ denotes an extrapolated form. Exactly how the fission neutron source is computed and extrapolated from iteration to iteration is explained in Section 2.

1.6 The Boundary Conditions for a Group

As defined in Equations (8) through (15), the boundary conditions for neutron group j are

$$\alpha_{1j} \phi_{j1} - \beta_{1j} J_{j1} + \gamma_{1j} = 0 \text{ at the inner (lower } x) \text{ boundary, and} \quad (107)$$

$$\alpha_{2j} \phi_{ji_{\max}} - \beta_{2j} J_{ji_{\max}} + \gamma_{2j} = 0 \text{ at the outer (upper } x) \text{ boundary,} \quad (108)$$

except in the case of the special periodic boundary condition where the conditions are

$$\phi_{j1} = \phi_{ji_{\max}}, \text{ and} \quad (109)$$

$$x_1^0 J_{j1} = x_{i_{\max}}^0 J_{ji_{\max}}. \quad (110)$$

(In these equations J_{ji} is the net neutron current that, according to the diffusion theory, is

$$J_{ji} = -D_j \frac{d\phi_{ji}}{dx}.) \quad (111)$$

The values of the α 's, β 's, and γ 's can be directly input by the user. More often, the user will specify a type of boundary condition and the

program then automatically computes appropriate values for the boundary condition parameters α , β , and γ for either boundary in each group. (The point to remember is that the α 's, β 's, and γ 's are subject to automatic computation of new values that replace any that were input or computed on previous problems of a given run on the computer.)

Two other boundary condition data values for each group can be directly input by the user. These data are ω_{1j} for the inner boundary and ω_{2j} for the outer boundary. Each can be used in an automatic calculation of α , β , and γ at a boundary, as shown in Table I.

In Table I, all the external boundary condition options are shown and described. The user sets the input words NIBDRY and NOBDRY according to which of the eight options is desired at the inner boundary and at the outer boundary, respectively. The formulae for G_{1j} and G_{2j} , the correction factors on the extrapolation distance to zero flux for the inner and outer boundaries, respectively, were derived by E. R. Cohen^[8].

If the macro material in a region is not a v. macro material, special internal boundary conditions are permitted that allow the user to specify the value of the derivative of the logarithm of the flux at the surface of a region. (In a control rod or other highly absorbing material, there can be better diffusion theory representation of the loss of neutrons in the material when appropriate boundary conditions are applied at the material boundaries. Such boundary conditions are usually determined by higher order solutions of the transport problems from which one finds the correct albedos, flux extrapolation distances, current to flux ratios, etc.)

The form of the internal boundary condition applied in the program is

$$\frac{1}{\phi_{jib}} \frac{d\phi_{jib}}{dx} = \pm q_j^{m*} \quad (112)$$

where i_b denotes a mesh point at the surface of a region and the sign before q_j^m is minus at the lower i_b and plus at the upper i_b of the region. The superscript m denotes the index of the macro material the user specifies for the region; i.e., the user specifies the q_j^m values by neutron group

*This condition is equivalent to other familiar forms, as expressed by the multiple equation (for an upper boundary)

$$\frac{1}{\phi} \frac{d\phi}{dx} = q = \frac{1}{d_{ext}} = \frac{-J}{D\phi} = \frac{1}{2D} \frac{(1-\alpha)}{(1+\alpha)}$$

where D is the diffusion constant in the material adjacent to the boundary and d_{ext} is the so-called extrapolation distance (to a zero flux value). The albedo α is defined as J^+/J^- where J^+ and J^- are the partial currents in the positive and negative x directions, respectively (such that $J = J^+ - J^-$). And, as usual in diffusion theory, it has been assumed that

$$J^+ = \frac{\phi}{4} - \frac{D}{2} \frac{d\phi}{dx} \text{ and } J^- = \frac{\phi}{4} + \frac{D}{2} \frac{d\phi}{dx}.$$

Table I. The External Boundary Condition Options

Value of Boundary Option Indicator		Boundary Condition Applied	Source of Boundary Condition Data	Settings of Boundary Condition Parameters			Remarks
				α	β	γ	
INNER BOUNDARY	NIBDRY = 0	$\phi_{j1} = 0$ (all groups)	Computed	1.0	0.0	0.0	zero flux
	1	$J_{j1} = 0$ (all groups)	Computed	0.0	1.0	0.0	zero current, sometimes symmetry
	2	$\phi_{j1} = \omega_{1j}$	ω_{1j} is input	1.0	0.0	$-\omega_{1j}$	constant group flux
	3	$\frac{d\phi_{j1}}{dx} = \omega_{1j}$	ω_{1j} is input	0.0	1.0	$-\omega_{1j}D_j$	constant group flux derivative, $J_j = -\omega_{1j}D_j$
	4	$\frac{1}{\phi_{j1}} \frac{d\phi_{j1}}{dx} = \frac{1}{\omega_{1j}}$	ω_{1j} is input	1.0	$-\omega_{1j}/D_j$	0.0	logarithmic derivative condition, $J_j = -D_j\phi_j/\omega_{1j}$, flux extrapolates to zero at ω_{1j} from boundary
	5	$\frac{1}{\phi_{j1}} \frac{d\phi_{j1}}{dx} = \frac{1}{a3D_j}$ *	Computed	1.0	$-3a$	0.0	standard black boundary or vacuum
	6	$\frac{1}{\phi_{j1}} \frac{d\phi_{j1}}{dx} = \frac{1}{a3D_jG_{1j}}$ *	Computed	1.0	$-3aG_{1j}$ *	0.0	black boundary or vacuum, corrected for boundary curvature
	7	$\alpha_{1j}\phi_{j1} - \beta_{1j}J_{j1} + \gamma_{1j} = 0$	α_{1j} , β_{1j} , and γ_{1j} are input for all groups	as input	as input	as input	arbitrary condition
	8	$\phi_{j1} = \phi_{j1\max}$ (all groups)	Computed	1.0	0.0	0.0	periodic condition, boundary fluxes are equal

OUTER BOUNDARY	NØBDRY = 0	$\phi_{j1\max} = 0$ (all groups)	Computed	1.0	0.0	0.0	zero flux
	1	$J_{j1\max} = 0$ (all groups)	Computed	0.0	1.0	0.0	zero current, sometimes symmetry
	2	$\phi_{j1\max} = \omega_{2j}$	ω_{2j} is input	1.0	0.0	$-\omega_{2j}$	constant group flux
	3	$\frac{d\phi_{j1\max}}{dx} = \omega_{2j}$	ω_{2j} is input	0.0	1.0	$-\omega_{2j}D_j$	constant group flux derivative, $J_j = -\omega_{2j}D_j$
	4	$\frac{-1}{\phi_{j1\max}} \frac{d\phi_{j1\max}}{dx} = \frac{1}{\omega_{2j}}$	ω_{2j} is input	1.0	ω_{2j}/D_j	0.0	logarithmic derivative condition $J_j = D_j\phi_j/\omega_{2j}$, flux extrapolates to zero at ω_{2j} from boundary
	5	$\frac{-1}{\phi_{j1\max}} \frac{d\phi_{j1\max}}{dx} = \frac{1}{a3D_j}$ *	Computed	1.0	$3a$	0.0	standard black boundary or vacuum
	6	$\frac{-1}{\phi_{j1\max}} \frac{d\phi_{j1\max}}{dx} = \frac{1}{a3D_j G_{2j}}$ *	Computed	1.0	$3aG_{2j}$	0.0	black boundary or vacuum, corrected for boundary condition
	7	$\alpha_{2j}\phi_{j1\max} - \beta_{2j}J_{j1\max} + \gamma_{2j} = 0$	α_{2j} , β_{2j} , and γ_{2j} are input for all groups	as input	as input	as input	arbitrary condition
	8	$x_{1j}^p J_{j1} = x_{1\max}^p J_{j1\max}$ (all groups)	Computed	0.0	1.0	0.0	periodic condition, leakage into inner boundary equals leakage from outer boundary

*Values of D_j are those of the material adjacent to the boundary, $a = 0.710446$, $G_{1j} = [1.0 + 1.5207587 (3D_{jp}/2x_1)] / [1.0 + 0.8103127 (3D_{jp}/2x_1)]$ and $G_{2j} = 1.0 [1.0 + a (3D_{jp}/2x_{1\max})]$.

and material and not by region number, directly. In diffusion theory, the net current J is equal to $-D \frac{d\phi}{dx}$, so that if the user reads in a positive q_j^m value there is net leakage of j^{th} group neutrons into the material of the region at both boundaries. Negative q values cause homogeneous boundary sources at both boundaries.

Nonzero values of q_j^m are detected by the program and the corresponding group constants, D_j , Σ_{aj} , T_j , Σ_{uj} , $\nu\Sigma_{fj}$, Σ_{fj} , χ_j , and (for all j') $\Sigma_{j \rightarrow j'}$ and $\Sigma_{j' \rightarrow j}$, are automatically set to zero. This eliminates the possibility of a number of inconsistencies that might occur if fission neutron sources, scattering-in, scattering-out, etc., were allowed in the energy group.

The difference equations for the neutron energy group are affected at the boundaries of each region in which q_j^m is nonzero for the region material. For example, at the lower boundary of such a region where the internal boundary condition is

$$\frac{1}{\phi_{jib}} \frac{d\phi_{jib}}{dx} = -q_j^m \text{ (or, } -J_{jib} + D_{jib} q_j^m \phi_{jib} = 0), \quad (113)$$

the difference equation is derived in exactly the same way as the external boundary condition, Equation (B-14) (see Appendix B). The only difference, of course, is that i_b replaces $i_{\max}$ and the boundary condition equation corresponding to Equation (B-11) has been applied with the parameter values $\alpha = D_{jib} q_j^m$, $\beta = 1$, and $\gamma = 0$. At the upper boundary of the region an equation similar to Equation (B-12) is derived for the group with $\alpha = D_{jib} q_j^m$, $\beta = 1$, and $\gamma = 0$. The neutron group difference equations used by the program at all mesh points inside the region are changed to read $-\phi_i + S_{ti} = 0$, where S_{ti} is nonzero only if the user purposely reads in an external source (S_{rj} and/or S_{pj}) for the group and region. Hence, in a region and group in which q_j^m is nonzero, the flux values are ordinarily zero except at the boundaries.

2. THE FLUX SOLUTION METHODS

DWARF solves the matrix Equation (1) (on each feedback iteration of each time point) with fission neutron source iterations. The basic problem reduces to solving an equation for each group in turn, with a source term S_j known in terms of contributions from S_{ej} , S_{sj} , and S_{fj} , the fixed volume, scattering-in, and fission sources, respectively. That is,

$$-\nabla \cdot D_j \nabla \phi_j + \Sigma_{rj} \phi_j = S_{ej} + S_{sj} + S_{fj}, \text{ where} \quad (114)$$

$$S_{sj} = \Sigma_{sj-1} \phi_{j-1} \text{ for a real flux problem, or} \quad (115)$$

$$S_{sj} = \Sigma_{sj} \phi_{j+1} \text{ for an adjoint problem.} \quad (116)$$

With the boundary conditions given for group j , the spatial distribution of ϕ_j can be directly obtained in terms of the fluxes at the designated i mesh points on x from the easy solution of a set of difference equations. The derivation of the difference equations and their simultaneous solution for a group is detailed in Appendix B of Reference 5.

The sections to follow describe the source iteration process and explain the techniques at the disposal of the user for accelerating the convergence of the solution.

2.1 The Fission Source Iterations

The fission neutron source for group j in the t^{th} iteration can be written $S_{fj} = \chi_j f^t$ or $S_{fj} = \nu \Sigma_{fj} f^t$ for real and adjoint flux problems, respectively. (On a real flux problem, f^t is therefore simply the spatial distribution of the total neutrons produced per cm^3 per second.)

On the first iteration (the first solution for the group fluxes of a time range), f^1 can be:

1. set to unity at all x in each fissioning region,
2. set to the fission source distribution retained from the previous problem,
3. read in as input with one value per region,
4. read in as input with one value per source point, or
5. set to zero everywhere.

These five options correspond to INSORS values of $(\pm)0, 1, 2, 3$, and 4 , respectively. In all but the last case, there is the option to

"renormalize" f^1 such that $\int_V f^1 dv = F_0$, where F_0 can also be an input value (FISSOR) or simply unity. This renormalization of f^1 and all subsequent f^t values such that the volume integral equals a constant F_0 is controlled by the integer flag IREN and imposes the constraint of Equations (16), (103), and (104).

The f^t distributions for $t \geq 2$ are obtained as follows.

At the end of each fission source iteration, a new fission source distribution s_f^t is computed from the new fluxes ϕ_j^t obtained on the t^{th} iteration (solution). The new s_f^t is

$$s_f^t = \sum_j \nu \Sigma_{fj} \phi_j^t \text{ for a real flux problem, and} \quad (117)$$

$$s_f^t = \sum_j \chi_j \phi_j^t \text{ for an adjoint problem.} \quad (118)$$

Next, a generation factor λ^t is computed from

$$\lambda^t = \int_V \Delta_f^t dv / \int_V f^t dv. \quad (119)$$

This is always the generation factor edited and the definition applies whether or not the problem is the usual homogeneous eigenvalue problem. In the latter case, it is the fundamental eigenvalue at convergence.

If the problem is a renormalization type of problem (IREN=0) such that $\int_V f^t dv = F_0$ for all t , then an auxiliary distribution f^{*t+1} is computed from

$$f^{*t+1} = \Delta_f^t / \lambda^t. \quad (120)$$

Otherwise, f^{*t+1} is simply set equal to Δ_f^t (IREN=1). It is obvious that in renormalization types of problems, where $\int_V f^t dv = F_0$, the λ^t from Equation (119) will be equal to $\int_V \Delta_f^t dv / F_0$ in accordance with Equation (17), (101), (102), and (106). Hence, the source f^{*t+1} will also now satisfy the fission source constraint of Equations (103) and (104), that is,

$$\int_V f^{*t+1} dv = \frac{1}{\lambda^t} \int_V \Delta_f^t dv = \frac{F_0}{\int_V \Delta_f^t dv} \int_V \Delta_f^t dv = F_0. \quad (121)$$

When the problem is not a renormalization type and $f^{*t+1} = \Delta_f^t$, then this is equivalent to the option corresponding to setting $\lambda = 1$ in Equation (1) or (19) as described in Section 1.1.

Finally, the fission source distribution for the next outer iteration is calculated with the extrapolation formula

$$f^{t+1} = f^t + \alpha(f^{*t+1} - f^t) + \beta(f^t - f^{t-1}), \quad (122)$$

where α and β are two extrapolation constants defined in Section 2.3, and $\beta = 0$ for $t = 1$. Note that on renormalization types of problems, the extrapolation Equation (122) preserves the constraint in the sense that $\int_V f^{t+1} dv = F_0$.

The fission source iteration process is terminated when the number of iterations is equal to MAXOUT, as set by the user, or when a user specified convergence criterion is satisfied. If convergence is desired on the total fission source, i.e., the generation factor, itself, the user sets NCONVG equal to zero and iteration stops when

$$\left| \frac{\lambda^t - \lambda^{t-1}}{\lambda^t} \right| \leq \epsilon_1. \quad (123)$$

For NCONVG=1, pointwise convergence of the fission source is specified and the code computes

$$\bar{\Lambda}^t = \left| \frac{f^*t}{f^{t-1}} \right|_{\text{maximum}} \quad \text{and} \quad \underline{\Lambda}^t = \left| \frac{f^*t}{f^{t-1}} \right|_{\text{minimum}}. \quad (124)$$

The iteration is terminated when

$$\bar{\Lambda}^t - \underline{\Lambda}^{t-1} \leq \epsilon_1 \quad (125)$$

(for renormalization types of problems with IREN=0), or when the maximum of

$$\left| \bar{\Lambda}^t - 1 \right| \quad \text{and} \quad \left| \underline{\Lambda}^t - 1 \right| \quad (126)$$

is less than or equal to ϵ_1 (for non-renormalization types of problems with IREN=1).

The value of ϵ_1 is input as EPS1.

2.2 Fission Source Renormalization and Special Applications

As pointed out in the preceding section, when the renormalization is applied, the volume integrals of f^t , f^{*t+1} , and f^{t+1} are all constrained to equal F_0 . (Hence, the term "renormalization" is used with respect to the application of Equation (120) during the iterative process, for the successive fission source volume integrals are all "renormalized" to be equal to the same value.)

As stated before, this renormalization can be expressed as the constraint

$$\int_V f^t dv = F_0 \quad (1 \leq t) \quad (127)$$

where F_0 is the constant FISSOR that is either unity or some value input by the user.

With the usual homogeneous problem (in which the generation factor converges, with iteration, to the largest eigenvalue or k_{eff} of the reactor) the constraint (127) is necessary to preclude the chance of underflow or overflow of the solution. (The successive sources would otherwise each be the factor λ^t times the previous source.)

One of the options available to the user in a DWARF solution is the option to call for fission source renormalization, i.e., the imposition of the constraint (127), even on inhomogeneous problems*. When the user calls for the renormalization type of solution (IREN=0) for an inhomogeneous problem, DWARF performs the iterative solution for the neutron generation factor λ in the same manner as for the homogeneous case by means of the constraint (127) imposed on the volume integral of the fission neutron source.

There are two common uses for this option to supply fission source renormalization via constraint (127) in solving inhomogeneous problems.

First, there is the problem of an extraneous source in a reactor near critical (such as exists during a usual reactor start-up). The convergence of the fission source level to the proper steady state value is slow in this case when the inhomogeneous problem is solved by iteration with no fission source renormalization. The solution can be obtained quickly, however, if an ordinary search is called for with the constraint constant F_0 , itself, the search parameter and with the desired eigenvalue, λ_d , set to unity.

Second, the importance (or adjoint) weighted volume integral of a function can be evaluated in terms of the difference in the computed λ values when the difference is caused by adding the function as a small (compared to F_0) extraneous source to the reactor while imposing constraint (127). (For example, see Equation (84), page 28, in Reference 7.)

The direct analogy between unextrapolated fission neutron source iterations and successive neutron generations is also of use in certain instances. The analogy exists because a new fission source distribution Δ_f^t is composed of all the (first) fission neutron daughters produced as a result of the absorption and diffusion in the reactor of the neutron source distributed as f^t . Hence, if f^t is viewed as the distribution of the t^{th} generation of neutrons, Δ_f^t is the $t+1^{\text{th}}$ generation in both magnitude and distribution. And the ratio of the volume integrals

$$\lambda^t = \int_V \Delta_f^t dv / \int_V f^t dv \quad (128)$$

is the total number of fission neutrons produced per fission neutron introduced in the f^t distribution. Hence, in the edits of DWARF problems this λ^t is appropriately termed the "generation factor" for the t^{th} iteration. It is the reactor multiplication factor for the first generation of neutrons from the given fission source neutron distribution f^t . (There is an exception when there is an extraneous source of neutrons, but, even in that case λ^t remains the ratio of new fission neutrons per

*A problem becomes an inhomogeneous one when extraneous sources or sinks are present as signified by nonzero values of any γ_{1j} or γ_{2j} in the boundary conditions, Equations (8) and (9), and/or by any nonzero values in the fixed volume sources, S_{ej} .

fission neutron introduced.) It is clear that if there is no extrapolation of the fission neutron sources in the course of a DWARF solution (in which case f^{t+1} is distributed exactly as s_t^f), then the iterations can be viewed as the direct computation of the successive generations of neutrons resulting from the introduction of the initial source distribution f^1 . Because of this analogy, the generation factors for the successive λ^t values can be of various uses. (For example, see Sections 5.3 and 6.3 in Reference 7.)

2.3 Acceleration of the Iteration Convergence

The Chebyshev extrapolation algorithm employed in the DWARF program is the one described in the very complete and comprehensive report by L. A. Hageman^[9]. The only differences are that the initial estimate σ_0 of the dominance ratio σ^* is computed in DWARF from

$$\sigma_0 = \left| \bar{\lambda}^{t'} - \underline{\lambda}^{t'} \right| / \left| \bar{\lambda}^{t'-1} - \underline{\lambda}^{t'-1} \right| \quad (129)$$

when IREN=0 (and the problem is a renormalization type) and from

$$\sigma_0 = \left| \lambda^{t'} - 1 \right| / \left| \lambda^{t'-1} - 1 \right| \quad (130)$$

when IREN=1 (and the problem is a non-renormalization type), provided $0 < \sigma_0 < 1$. The iteration number t' , after which σ_0 is computed, is set to 5 by the program unless the user supplies a nonzero value for the input word NPIO, in which case t' is set to NPIO+1. The t' iterations are done with unextrapolated sources in order to permit the convergence process to "settle down" before the σ_0 estimate is obtained.

The first extrapolation of the source is performed according to Equation (122) with $\alpha_1 = 2/(2-\sigma_0)$ and $\beta_1 = 0$. Thereafter, the extrapolation parameters are computed from

$$\alpha_\ell = \frac{4}{\sigma_0} \left\{ \frac{\cosh [(\ell-1)\gamma]}{\cosh [\ell\gamma]} \right\} \quad (131)$$

and

$$\beta_\ell = \cosh [(\ell-2)\gamma] / \cosh [\ell\gamma], \quad (\ell \geq 2) \quad (132)$$

*The dominance ratio σ , as usually defined for homogeneous problems, is the absolute value of the ratio of the second largest eigenvalue to that of the largest. However, in inhomogeneous problems (with fission) the user may prefer to view σ as simply the ratio of the error reductions of successive iterations.

where

$$\gamma = \cosh^{-1}[(2-\sigma_0)/\sigma_0] = \ln \left[\left(\frac{2-\sigma_0}{\sigma_0} \right) + \sqrt{\left(\frac{2-\sigma_0}{\sigma_0} \right)^2 - 1} \right] \quad (133)$$

and ℓ is the subscript denoting both the order of the Chebyshev polynomials of the extrapolation procedure and the number of extrapolations done with the given σ_0 value.

It is recommended procedure to iterate with a series of low order Chebyshev polynomials rather than to let the order ℓ increase indefinitely. For this reason the maximum order can be set by the user (with the input word NCHMAX), or, if it is not set to one or higher, it is set to 15 by the program. It is usually good practice, also, to set the maximum order to at least four, because when $\ell \geq 4$ the program computes an updated estimate of σ to be used to try to increase the rate of convergence.

When $\ell \geq 4$ the program computes a value

$$ER_\ell = (\bar{\Lambda}^\ell - \underline{\Lambda}^\ell) / (\bar{\Lambda}^2 - \underline{\Lambda}^2) \text{ when } IREN=0 \quad (134)$$

or

$$ER_\ell = [(\bar{\Lambda}^\ell + \underline{\Lambda}^\ell)/2 - 1] / [(\bar{\Lambda}^2 + \underline{\Lambda}^2)/2 - 1] \text{ when } IREN=1. \quad (135)$$

[The $\bar{\Lambda}^\ell$ and $\underline{\Lambda}^\ell$ are the maximum and minimum values as defined by Equation (124), but they are superscripted by ℓ instead of the actual iteration number t .] Also computed is the Chebyshev polynomial $C_{\ell-2}$ of argument $(2-\sigma_0)/\sigma_0$ from the recursion relationship

$$C_n(x) = 2x C_{n-1}(x) - C_{n-2}(x) \text{ for } n \geq 2 \quad (136)$$

$$C_1(x) = x \text{ and } C_0(x) = 1. \quad (137)$$

Then, if $1 > ER_\ell > 1/C_{\ell-2}[(2-\sigma_0)/\sigma_0]$ a better estimate for σ is computed from

$$\sigma'_0 = \frac{\sigma_0}{2} \cosh \left\{ \frac{\ln(x + \sqrt{x^2 - 1})}{\ell - 2} + 1 \right\} \quad (138)$$

where

$$x = ER_\ell C_{\ell-2}[(2-\sigma_0)/\sigma_0]. \quad (139)$$

This better estimate for σ is never used in the same iteration in which it is found. Instead, if it is found when $\ell = \ell_0$, it is not used until $\ell = \ell_0 + \ell_1$ where ℓ_1 is two unless the user sets the input word NERCHK to one or more as a user specification of ℓ_1 . The purpose of this is to allow the Chebyshev polynomial to reach a reasonably high order with each σ_0 value used.

When $ER_\ell \geq 1$, signifying no error reduction, the whole extrapolation process is interrupted. The fission source for that iteration is not extrapolated, and the extrapolation is begun again with $\ell = 1$ on the next iteration for which σ_0 from Equation (129) or (130) satisfies the condition

$$0 < \sigma \leq 1. \quad (140)$$

If $ER_\ell \leq 1/C_{\ell-2}[(2-\sigma_0)/\sigma_0]$, then σ_0 is retained as the best estimate.

Although the three point Chebyshev polynomial convergence acceleration is highly recommended for all general cases in which extrapolation is desired, the user can specify his own two point extrapolation by setting the input word NEXT0 to one and providing a value for the extrapolation factor θ with the input word THETA. The two point formula used is

$$f_{t+1} = f_{*t+1} + \theta(f_{*t+1} - f_t) \quad (141)$$

which is the same as Equation (122) with $\alpha = 1 + \theta$ and $\beta = 0$. The user should therefore choose θ less than or equal to one to avoid divergence.

3. AUTOMATIC SEARCHES FOR DESIRED REACTOR CONDITIONS

3.1 The Search Options

At each time point, after the simultaneous solutions for the power distribution, the fluxes, and the feedback vectors have been obtained by means of the feedback iterations, any of a number of reactor properties can be automatically varied by means of the search algorithms of the search routines. By a number of "search iterations" for the time step, the specified reactor property can be adjusted until the generation factor λ^S attained on the s^{th} iteration satisfies the convergence criterion

$$\left| (\lambda^S - \lambda_D) / \lambda_D \right| \leq \epsilon_2, \quad (142)$$

where λ_D is the desired generation factor for each time point in the time range and ϵ_2 is the convergence level specified by the user as EPS2. The value of λ_D is input as EIGEND (F-0,0,14,0), except for the so-called "total source" search when it is unity.

Each search iteration consists of obtaining the generation factor finally resulting after the necessary feedback iterations, if any, given the particular values of the search parameters for the given search iteration. The resulting generation factor is then used with those of the previous search iterations for the time step to arrive at a new guess for the search parameter values. The new values are then used in the next search iteration.

For search iteration s , λ^s is usually the converged generation factor attained. However, if the input flag IGENOP is set to unity,

$$\lambda^s = \int_V dv \lambda_f^1 / \int_V dv \sum_j S_{ej} \quad (143)$$

where λ_f^1 is the fission source from the first source iteration [see Equations (117) and (118)] and S_{ej} is the fixed volume source of group j . The latter option allows the user to search on the value of the first generation factor of a particular input source distribution.

On each search iteration, a new value for a search parameter modifier, C_s , is computed. This value is then applied by either multiplying or dividing a search parameter to obtain a new value. Search parameters are determined by the user who sets the integer ISERCH as follows.

ISERCH Value	Search Parameter(s)
± 1	The perpendicular buckling squared B^2 [see Equation (3)] in one or more user specified regions, "Buckling Search"
± 2	The poison parameter Σ_p [see Equation (3)] in one or more user specified materials, "Poison Search"
± 3	The dimension or region width of one or more user specified regions, "Dimension Search"
± 4	The total fission neutron source F_0 (FISSOR) that is used in the renormalization of f^t [see Equation (127)] for special application in certain inhomogeneous problems, "Fission Source Search"
± 5	The region width of the lefthand (lower) region as the boundary between two contiguous regions is shifted, "Boundary Search"
± 6	The total power ($P_{av} = AVPOWER$) used in the normalization of the fluxes in computing $q^u(k)$ [see Equations (63) through (65)], "Power Search"

For the details concerning how the searches can be restricted to only specified materials or regions by the LRS(K) data and how to specify the initial guesses for the search parameter(s), see the notes on the application of the input data under ISERCH and LRS(K), Section IV-2.2, Table II.

A negative value of ISERCH indicates the user also desires the search parameters to satisfy the convergence criterion

$$\left| \frac{\lambda^s - \lambda^{s-1}}{\lambda^s} \right| \leq \epsilon_3 \quad (144)$$

where x^s is a search parameter value from search iteration s and ϵ_3 is the user specified convergence level, EPS3.

In setting search criteria, the user is warned to observe certain precautions:

- a. Allow enough fission source iterations (MAXOUT) per search iteration and set $\epsilon_1 = \text{EPS1}$ low enough [see Equation (125)] to assure an accurate value of λ^s for use in the search calculation of C_s . (Recommended is $\epsilon_1 \leq \epsilon_2/10$ with NCONVG set equal to one for pointwise source convergence.) Obviously, the feedback iteration convergence criterion $\epsilon_4 = \text{EPS4}$ should be correspondingly tight.
- b. Avoid setting ϵ_3 to too small a value. Often, λ^s changes so little with x^s that Equation (142) on λ^s is well satisfied before Equation (144) on x^s is satisfied. [Recommended is $\epsilon_2 \leq \epsilon_3/10$, but, if possible, the user should avoid specifying that Equation (144) be satisfied.]

3.2 The Search Algorithms

For all the searches except the boundary searches, the search parameter modifier C_s is found by means of an algorithm attributed originally to V. Bookston and C. Davis and used in the GNU II Code^[10]. The following description of the method was taken from Reference 8.

On the first search iteration, the value of C_1 is unity; that is, the initial guess for a search parameter is used, directly. For the second guess

$$C_2 = 1 + 10(\lambda_d - \lambda^1)/\lambda^1 \text{ if } \lambda^1 < \lambda_d \quad (145)$$

$$C_2 = 1/(1 + 10\omega) \text{ if } \lambda_d \leq \lambda^1 < 1.25\lambda_d \quad (146)$$

$$C_2 = 1/(1 - 2.62\omega + 50\omega^2) \text{ if } \lambda^1 \geq 1.25\lambda_d \quad (147)$$

where $\omega = (\lambda^1 - \lambda_d)/\lambda_d$. For the third guess,

$$\ln C_s = \ln C_{s-1} + \frac{(\ln C_{s-1} - \ln C_{s-2})}{(\lambda^{s-1} - \lambda^{s-2})} (\lambda_d - \lambda^{s-1}). \quad (148)$$

For the fourth and subsequent guesses, C_s is taken as the solution of

$$(C_s - a)(\lambda_d - b) = c \quad (149)$$

where a , b , and c are determined by solving the system:

$$\begin{aligned}
(C_{S-3}-a)(\lambda^{S-3}-b) &= c \\
(C_{S-2}-a)(\lambda^{S-2}-b) &= c \\
(C_{S-1}-a)(\lambda^{S-1}-b) &= c.
\end{aligned}
\tag{150}$$

The solution of System (150) involves the evaluation of a determinant. If the determinant is less than 10^{-20} , C_S is obtained from Equation (148). If C_S does not lie on the same branch of the hyperbola determined by Equation (149) as does C_{S-1} , then the linear extrapolation is performed

$$C_S = C_{S-1} + (C_{S-1} - C_{S-2})(\lambda^S - \lambda^{S-1}) / (\lambda^{S-1} - \lambda^{S-2}). \tag{151}$$

This hyperbolic extrapolation that is normally used on the fourth and higher search iterations works only if the generation factor λ^S increases as C_S increases. Therefore, on buckling, poison, total fission source, and total power searches the new parameter values are usually computed from $x^S = x^1/C_S$; and on a dimension search new parameters are usually computed from $x^S = x^1 C_S$. However, it is possible that the use of C_S in this manner can cause the λ^S to decrease as C_S is increased. To safeguard against this, the value $(\lambda^2 - \lambda^1)/(C_2 - C_1)$ is checked after the second search iteration; and, if it is not positive, then $x^S = x^1 C_S$ in buckling, poison, total fission source and power searches, and $x^S = x^1/C_S$ in dimension searches.

Negative values of C_S are not excluded in buckling and poison searches, but they cause a search failure (termination) in all other searches. All searches except boundary searches are terminated if a C_S value is obtained that does not satisfy the relationship,

$$10^{-10} C_1 < C_S < 10^{10} C_1. \tag{152}$$

Also (on all but boundary searches), whenever a value less than 0.001 is obtained for C_S , the search is restarted as if s were two and with new initial guess values. The new guess values are ten times the previous initial ones on buckling, poison, total fission source and total power searches and are simply the last values for a dimension search. Following any search failure the user is so informed and the problem is terminated and edited with the last search parameters as though the termination were normal. A boundary search is considered failed when a region width less than some minimum value is obtained twice. This minimum is the input word WMI that can be specified by the user.

In a boundary search, the width of the left-hand region is always found from $x_S = C_S x_1$. The C_S value is obtained by the linear extrapolation of Equation (151) for the third and subsequent search iterations, and C_2 is found from

$$C_2 = C_1 + .1\omega. \tag{153}$$

The value of ω in Equation (153) is

$$\omega = W_r/W_l \text{ if } W_r < W_l \quad (154)$$

and

$$\omega = -1 \text{ if } W_r \geq W_l \quad (155)$$

W_l and W_r are the initial widths of the left-hand and right-hand regions, respectively.

3.3 The Application of the Search Parameters

Buckling Searches. In a buckling search, a single value of the buckling squared B_s^2 , is applied in the formula for the total removal cross section in each group and in each region to which the search is to be supplied, hence, the formula for the removal cross section becomes

$$\Sigma_{rj} = \Sigma_{aj} + \sum_{j' \neq j} \Sigma_{j \rightarrow j'} + D_j B_s^2 + T_j \Sigma_p \quad (156)$$

in each search region that does not have a variable macro material. The regions to be searched are named in the LRS(K) list by the user. All regions are searched if LRS(1) = 0. The first guess is

- a. B^2 for group one in material one if all regions are searched,
- b. B^2 for group one in the material named by LRS(1) if LRS(1) > 0,
or
- c. $B^2 = 0.01$ if the value from a. or b., above is zero.

The bucklings B_s^2 are applied only in regions with f. macro materials ($m \leq 40$) unless INBUCK ≥ 2 (I-0,0,11,0). The buckling values found by a search will exist in the search regions in succeeding time ranges or problems.

Poison Searches. The value of the poison cross section parameter Σ_p in Equation (3) is modified by C_s for each material to which the search is to be applied. That is, the respective initial values of Σ_p in all the search materials are multiplied by a common factor on each search iteration. The user specifies which materials are search materials by listing them in the LRS(K) vector. If LRS(K) = 0, the search is applied to all materials. The final set of Σ_p values obtained in a poison search problem are retained in the respective materials for the next time range or problem.

In addition to modifying the removal cross section of group j by means of the term $T_j \Sigma_p$, in each search material, a poison search can also modify the diffusion coefficient D_j if the user calls for automatic calculation of D_j by Equations (52) and (55). See Section 1.2 and the discussion on the application of the input word IDC0.

Dimension Searches. The user can specify a number of regions to be dimension search regions by listing the regions, by number, in the LRS(K) vector. Regions are numbered sequentially from left (lower boundary) to right, starting with region one. If LRS(1) is zero, all regions are search regions. The region width RW(IR) (for region number IR) is modified by C_s in each search region on the s^{th} search iteration. That is, the dimension of each search region is increased or decreased by the same factor. The final region widths are retained in the respective regions for the next problem.

Total Fission Source Searches. The purpose of a total fission source search is to solve efficiently a certain class of problem. Inhomogeneous problems with fissions can be very slowly convergent when the k_{eff} of the reactor system approaches unity and when the conventional solution method is used (that iterates the fission neutron source without renormalization). See Section 2.2. Such problems can be solved very efficiently if an ordinary search is performed with $F_0 = \text{FISSOR}$, itself, the search parameter.

In an inhomogeneous problem with fissions, one can still solve for a generation factor λ^t from Equation (119) provided one applies the constraint (or renormalization) of Equation (127) to the fission source at the beginning of each fission source iteration. If one can find some value of F_0 for which the converged generation factor is unity, then the solution of that problem is the desired solution of the inhomogeneous problem. Hence, the user has the option in the DWARF program to call for the application of the search algorithm of Section 3.2 to the total fission source. The code then computes a value F_{0s} and applies the normalization.

$$\int_V f_s^t dv = F_{0s} \quad (157)$$

on each source iteration performed during the s^{th} search iteration.

Since renormalization is performed, the user should specify a zero value for the input word IREN.

If the value of FISSOR is zero or blank, or if it is unity from a previous problem, then it will be automatically set to twice the volume integral of the fixed volume source of the problem. That is,

$$F_{01} = 2 \int_V \sum_j S_{ej} dv = 2 \int_V \sum_j (S_{pj} + S_{rj}) dv. \quad (158)$$

This usually provides a reasonable guess for use on the first search iteration. [The total fixed volume source S_{ej} is determined by the floating point input data $SE0(K,J)$ for source point K and group J and $SEOR(J,IR)$ for group J and region IR. These data determine S_{pj} and S_{rj} , respectively.]

If there are no fixed volume sources in the problem and the inhomogeneity is due to the boundary conditions, then it is necessary to input a nonzero value of FISSOR for the first search guess, F_{01} . Otherwise, F_{0s} will remain zero and no solution can be obtained. Of course, any time a good guess is available for the total fission source, then that value should be input for FISSOR.

Boundary Searches. A specification of ± 5 for the value of ISERCH causes the position of the boundary between two contiguous regions to be varied with each search iteration until the search criteria are satisfied or until the search fails by finding (twice) that one of the region widths is less than a specified minimum value, $W_{min} = WMI (F-0,0,12,0)$. The value of WMI is set by the program to 0.001 unless the user supplies some value greater than zero. The program assumes the left-hand (lower) region is region one unless the user supplies a different region number by means of LRS(1). (Unlike a dimension search, a boundary search changes no other region boundary positions besides the one being searched.)

This search cannot properly be applied at the boundary of a region that is assigned a library set and nuclide concentrations.

On the first search iteration the boundary position is as determined by the user with the input data for the region widths, W_L and W_R for the left and right regions, respectively. The number of mesh intervals used in each region is also as initially specified by the user. Let n_L and n_R be the respective number of intervals in the left and right regions. On the second and subsequent search iterations, the number of intervals in each region is redistributed such that the number of intervals in the region of larger width is

$$k_s = \max \{W_{LS}, W_{RS}\} (n_L + n_R) / (W_L + W_R). \quad (159)$$

The mesh size in the larger region is then $\max \{W_{LS}, W_{RS}\} / k_s$ and in the smaller region it is $\min \{W_{LS}, W_{RS}\} / (n_L + n_R - k_s)$. Hence, the total number of intervals in the two regions is kept constant, but after the first search iteration, the intervals are redistributed such that the interval sizes of the two regions are as nearly equal as possible.

The exception is when $\min \{W_{LS}, W_{RS}\} \leq W_{min}$. In this case, the smaller region is allowed one mesh interval and the larger region is allotted $k_s = n_L + n_R - 1$. At the same time, the smaller region width is set equal to W_{min} and the larger region width is set equal to $W_L + W_R - W_{min}$. As stated before, the search is terminated if this case is encountered twice.

Power Searches. The power search option can be used in conjunction with the feedback options to adjust the generation factor of the reactor to a desired number at each time point. Since the feedback vectors can be functions of the power distribution q and the neutron group constants of the v . macro materials depend on the feedback vectors, the reactivity of the reactor depends on the value of the volume integral of the power P_{av} .

On each power search iteration, a new value of the power P_{AVS} is computed. This results in a new flux normalization factor C_p [see Equation (64)] that is subsequently used to compute new simultaneous solutions for λ , q , the feedback vectors, the neutron source, and the fluxes in a new set of feedback iterations.

The initial value of the search parameter P_{AV} is the input word AVPOWR (F-0,0,20,0), and the value of AVPOWR is therefore modified during a power search.

4. THE TIME DEPENDENT DEPLETION SOLUTIONS

The solution for the new nuclide concentrations of a time point is the analytic solution used in the SINBAD code. The solution method depends on four assumptions or restrictions that simplify the solution of the basic equation that is solved at each source point in each region that is assigned a library set number.

The equation solved for each nuclide is

$$\frac{dN_\ell}{dt} = \sum_{\ell'} P(\ell' \rightarrow \ell) N_{\ell'} - A_\ell N_\ell, \quad (160)$$

where the nuclide index ℓ of each nuclide of a library set is assigned by the user.

The $P(\ell' \rightarrow \ell)$ is a production coefficient (to be defined later) for the production of nuclide ℓ by nuclide ℓ' and the A_ℓ is the loss coefficient of nuclide ℓ ,

$$A_\ell = \lambda_\ell + \sum_j \sigma_{aj}^\ell \phi_j. \quad (161)$$

The four assumptions are:

1. $P(\ell' \rightarrow \ell)$ and A_ℓ are constant with respect to time during a time step. Hence, the neutron flux is assumed constant, since these coefficients are functions of it.
2. No nuclide can produce one of its predecessors in a transition scheme. In the transition scheme of Figure 1, the transition from nuclide 9 to nuclide 6 is therefore illegal.

Figure 1. A Typical Nuclide Production and Depletion Scheme

3. To facilitate handling of the production of fission product nuclides (those produced by the fission of other nuclides) the fission production is represented as production by an artificial nuclide and the fission rate is assumed constant during a time step.
4. No nuclide in a given transition scheme has the same loss coefficient A_ℓ . This restriction is satisfied by simply making each A_ℓ slightly different from any other. The user can set the input flag MODIFY (I-0,0,69,0) to cause the program either to warn the user of identical nuclide data or to automatically change one of the data by 0.1%.

The coefficient for the production of daughter nuclide ℓ by parent nuclide ℓ' is

$$P(\ell' \rightarrow \ell) = Y_d[m(i_\ell, \ell'), \ell'] \lambda_{\ell'} + \sum_j \sigma_c[j, |m(i_\ell, \ell')|, \ell'] \phi_j. \quad (162)$$

The ϕ_j is the flux computed by the program and the other quantities are set by the user as part of the library set data.

After assigning a unique index number ℓ (1 to 50) to each nuclide of a library set, each transition scheme of the set is defined by the following process.

First, the user assigns a branch number (1 to 4, in order) to each branch of a parent nuclide ℓ' that leads to the production of a daughter nuclide ℓ . The branch number leading to daughter ℓ is denoted by i_ℓ . Next the user assigns a "mode number" $m(i_\ell, \ell')$ to each branch of parent ℓ' . Radioactive decay transitions are assigned positive mode numbers (1 or 2), and neutron capture transitions are assigned negative mode numbers (-1 or -2). Then, for each decay (positive) mode number of the parent ℓ' , the user gives the yield fraction $Y_d[m(i_\ell, \ell'), \ell']$ that will multiply the decay constant $\lambda_{\ell'}$ in Equation (162). (If there is only one decay branch, the user can leave Y_d zero and the program will automatically set the Y_d to unity.) Finally, for each capture (negative) mode number of the parent ℓ' , the user gives the capture cross section $\sigma_c[j, |m(i_\ell, \ell')|, \ell']$ for the production of nuclide ℓ by neutron capture in group j by nuclide ℓ' . As indicated, the middle index of σ_c is the absolute value of the negative capture mode number.

The production of nuclides by fission is handled as production by an artificial nuclide f that has a unit concentration ($N_f = 1$), a zero loss coefficient ($A_f = 0$), and a production coefficient for producing nuclide ℓ that is given by

$$P(f \rightarrow \ell) = \sum_{\ell'} Y_f(n_{\ell'}, \ell) \bar{N}_{\ell'} \sum_j \sigma_{fj}^{\ell} \phi_j + Y_f(n_m, \ell) \sum_j \Sigma_{fj}^m \phi_j. \quad (163)$$

(In Figure 1, nuclide 6 is produced by fissions.) In Equation (163), Σ_{fj}^m is the macroscopic fission cross section as specified for macro material m according to Equation (38) or (39). The data for Σ_{fj}^m are read with the time range floating point data. The σ_{fj}^{ℓ} values for nuclide ℓ are microscopic fission cross sections that are read with the library set data. The fission yield $Y_f(n_{\ell'}, \ell)$ for producing fission product nuclide ℓ by fissions of nuclide ℓ' is determined as follows for each library set.

For each fissionable nuclide ℓ' the user assigns a fission yield set number $n_{\ell'}$, where $n_{\ell'} = 1, 2$, or 3 . The user then provides three sets of fission yields, such that each set specifies the yield per fission for each fission product nuclide ℓ . The program then chooses which of the three sets of yields to use with fissionable nuclide ℓ' according to the yieldset number $n_{\ell'}$ assigned to nuclide ℓ' . Similarly, the yield set to be used with macro material m in a region is specified by the yield set number n_m read for the material with the time range integer input data. The fission yield data used in this case are those of the library set assigned to the region.

The production coefficient $P(f \rightarrow \ell)$ is made to satisfy Assumption (1) that the production and loss coefficients be constant during a time step by using in Equation (163) the average concentration $\bar{N}_{\ell'}$ for nuclide ℓ' during the time step.

This average value for a time step is $\bar{N}_{\ell'} = \int_0^t N_{\ell'} dt / t$, where t is the time step length.

It is obvious that in order to compute the fission production $P(f \rightarrow \ell)$ for nuclide ℓ all the $\bar{N}_{\ell'}$ values must be evaluated first. To assure this, the user must observe the rule that all fissionable nuclides (or their parents) must be assigned numbers (ℓ') less than any fission product nuclide (ℓ). The SINBAD program processes the nuclides in the order in which they are numbered by the user; hence, if all fissionable nuclides (or their parents) have lower numbers than fission product nuclides, all the necessary $\bar{N}_{\ell'}$ values will be available by the time any fission product nuclide is processed.

In specifying the nuclide numbers for the transition scheme data of a library set, the user should try to observe the rules set forth in the SINBAD report, Section II-2.7, for reducing round-off errors in the concentration solutions. In brief, the rules are:

1. Eliminate all possible transitions where the $\sigma_c \phi$ or $Y_d \lambda$ values are so low that they can be neglected for the purposes of the problem. Aside from unduly complicating the scheme and requiring more computer time to solve, these low production coefficients are a major cause of round-off error.

2. Number the nuclides so that the shortest chains near the end of the transition scheme are processed first. Since the solution method considers lowest numbered nuclides first, the user should number the nuclides in a scheme so that the nuclide numbers decrease as one goes from the beginning to the end of the scheme.
3. Assign the branch numbers i_λ of a parent nuclide such that those with the largest production coefficients have the smallest i_λ values and are therefore processed first by the program.

The program allows the user the option to flag any of the nuclides of a library set as "nondepleting" nuclides. Such nuclides are flagged by assigning negative λ values for them in the list of nuclides for the library set. A nondepleting nuclide contributes to all the cross sections defined with Equations (30) through (44) and it contributes to fission production of other nuclides according to Equation (163), but it cannot otherwise produce other nuclides. That is, there is no production coefficient, Equation (162), considered for either producing it or producing other nuclides by it and there is no solution of Equation (160) performed for its concentration.

III. PROGRAM OUTPUT

1. PROGRAM CHECKPOINTS

At various points in the execution of a problem, the program prints a "checkpoint" line reading "TIME PT. N=X.XXXX", signifying that a particular point in the program flow has been reached and stating the machine (CPU) time that has elapsed, in seconds, since the last checkpoint was passed. These edits help the user predict required times for anticipated runs of similar problems and to know approximately what the program was doing when and if an error aborts the run.

(Each checkpoint and its number N is shown in Figure 2, page 85.)

2. INPUT DATA CARD CONTENTS

The contents of the DWARF time range title card are always edited as soon as the card is read. If the user has set the integer LIST=1 on the title card, the contents of each subsequent integer data card or floating point data card are also edited as soon as it is read. This editing continues for the title cards and data cards of any library set data decks that may follow the time range data deck.

3. PROCESSED INPUT DATA EDITS

After reading all the input data of the time range, the program processes the basic input data in preparation for problem solution.

The program then checks the input integer INED1. If INED1=0, editing proceeds for the following processed input data.

1. The complete list of single word integer data flags and indicators plus the two integer flags read on the time range title card.
2. All relevant (nonzero) integer data arrays including:
 - a. The list LRS(K) of macro materials or regions to which a search is to apply.
 - b. The list IYUK(K) of materials in which the formulas corresponding to IDC0=1 or 2 are to be used to compute the diffusion coefficients.
 - c. The list IREDT(K) of regions for which the concentrations and library set cross section contributions are to be edited as controlled by IOUT2.

3. The complete list of single word floating point data from EPS1 to VMAX3.
4. Any boundary condition parameters α_{1j} , β_{1j} , γ_{1j} , ω_{1j} , α_{2j} , β_{2j} , γ_{2j} , ω_{2j} that are input by the user.
5. The region dimension and mesh data consisting of the width RW(IR), number of intervals NI(IR), macro material number MR(IR), the region library set number MRMIC(IR), tl : mesh spacing, the upper bound mesh point, and the upper bound x value.
6. The transverse (perpendicular) buckling squared B_j^{2r} for each group and region as determined by the BSQM or BSQR arrays and the input flags INBUCK and IBSQ0 for all regions.
7. If v. macro materials are used, the polynomial coefficients $[a_p(n) \triangleq \text{RHOLCS}(N); N=n]$ and $[a_t(n) \triangleq \text{TEMPCS}(N); N=n]$ used to compute the coolant density and temperature as a function of enthalpy.
8. The macroscopic material contributions to the group constant data consisting of:
 - a. D_j , Σ_{aj} , $v\Sigma_{fj}$, Σ_{fj} , E_j , Σ_{sj} , Σ_{uj} , x_j , T_j , Σ_p , and the fission yield set number $[MY(IM); IM=m]$ for f. macro materials ($m \leq 40$).
 - b. The arrays defined by Equations (20) through (27) that determine the group constant contributions for v. macro materials ($m > 40$), followed by the group constants (listed in a. above) evaluated for a ρ_1 value of RHOGES and a ρ_2 value of PHOGES (for IRHO ≤ 2 , only).
9. If IRHO=4, the values read for $\rho_1(i)$, $1 \leq i \leq i_{\max}$; and if IPHO=4, the values read for $\rho_2(k)$, $1 \leq k \leq k_{\max}$, where $k_{\max}$ is the number of source points.
10. Any nonzero fixed volume source data for $S_{pj} \triangleq \text{SEO}(K,J)$ and $S_{rj} \triangleq \text{SEOR}(J,IR)$.
11. The fission source f^1 used on the first fission source iteration for the first time point in the time range if INSORS < 0.
12. Any values read with the floating point data of the time range for the concentration data RGCONC(L,IR) and PTCONC(K,L), provided there are any library sets used in the time range.
13. If there are any library sets used in the regions of the time range (i.e., some MRMIC(IR) > 0) the library set data for each nonzero MRMIC(IR) if LIBED=1 or for all library sets on the library unit if LIBED=2. {The library set data includes λ_ℓ , σ_{aj}^ℓ , $v\sigma_{fj}^\ell$, σ_{fj}^ℓ , the fission yield set number $n_\ell \triangleq \text{IY}(L)$, E_j^ℓ , σ_{sj}^ℓ , σ_{uj}^ℓ , the daughter ℓ' produced by branch i_ℓ of parent ℓ , the corresponding mode number

$m(i_{\ell'}, \ell)$, the decay yield $Yd[m(i_{\ell'}, \ell), \ell]$, the production cross sections $\sigma_c[j, m(i_{\ell'}, \ell), \ell]$, and the fission yields $Yf(n, \ell)$ for each nuclide ℓ of each yield set n .]

14. If $IOUT2 \geq 1$, the volume weighted average concentration for each region of each nuclide in the library set of the region; if $IOUT2 \geq 2$, the volume weighted average for the region of the cross section contributions of the library set nuclides; if $IOUT2 \geq 3$, the pointwise concentrations of the library set nuclides; if $IOUT2 = 4$, the pointwise cross section contributions of the library set nuclides. If $IREDT(1)$ is equal to a region number, the above data for $1 \leq IOUT2 \leq 3$ are edited only for the regions specified in the $IREDT(K)$ list. If $ISOSUM(1)$ is a nuclide number, then the cross section sum controlled by $IOUT2 = 4$ includes only the nuclides named in the $ISOSUM$ list. If $ISOED(1)$ is a nuclide number, then the pointwise nuclide concentration data corresponding to $IOUT2 = 3$ is edited only for the nuclides named in the $ISOED$ list.

4. ITERATION CONVERGENCE DATA EDITS

There are three major iterative loops that may be used to obtain the desired solution at a given time point:

1. The fission source iteration that computes new source and flux distributions for fixed nuclide concentrations and fixed values of the feedback vectors ρ_1 , ρ_2 , and ρ_3 until the convergence criterion of Equation (123) or (125) is met.
2. The feedback iteration that computes new values of the power distribution q and the corresponding distributions for ρ_1 and ρ_2 from the fluxes obtained by the source iteration until the convergence criterion of Equation (62) is met. [There is a minor iterative loop that is required to compute self-consistent values for q , ρ_1 , and ρ_2 within the accuracy on the residual defined by Equation (66).]
3. The search iteration that computes new reactor properties from the eigenvalues (corresponding to the self-consistently converged sources, fluxes, power distributions, and feedback vector distributions obtained by the source and feedback loops) until the convergence criteria of Equations (143) and, possibly, (144) are met.

The source iterative process can be extensively monitored, optionally; a minimum of data on feedback iterative process is edited; and data showing the status of each search iteration is always edited.

4.1 Fission Source Iteration Data

The fission source iterations are not performed and therefore not edited when there are no fissions; i.e., when $\nu \Sigma_{fj} = 0$ for all groups of all

materials. When there are fission neutrons produced, the outer iteration convergence data is printed (one line) following each outer iteration, provided the user specifies the integer ITDATA $\neq 0$.

The data consists of:

1. The iteration number, t
2. The "generation factor", λ^t
3. The "convergence level", which is $(\lambda^t - \lambda^{t-1})/\lambda^t$ if NCONVG=0 and is $\bar{\lambda}^t - \underline{\lambda}^t$ if NCONVG = 1
4. The "maximum lambda", $\bar{\lambda}^t$, and the source point at which it occurred
5. The "minimum lambda", $\underline{\lambda}^t$, and the source point at which it occurred
6. The "Cheby Order", the Chebyshev polynomial order ℓ of the current outer iteration
7. The "dominance ratio" estimate σ_0 being used in the Chebyshev extrapolation
8. The "error ratio" or rate of error reduction, which is $(\bar{\lambda}^t - \lambda^t)/(\bar{\lambda}^{t-1} - \underline{\lambda}^{t-1})$ if IREN=0 (renormalization type problem) and is $\max |(f^*t - ft-1)/ft-1| / \max |(f^*t-1 - ft-2)/ft-2|$ when IREN=1 ("max" denotes the maximum of the function for the set of all source points)
9. The "updated dominance" or better estimate for σ_0 ; i.e., the σ_0' value from Equation (138)

When NCONVG=0, the data for $\bar{\lambda}^t$ and $\underline{\lambda}^t$ of 4. and 5. are not printed.

All the Chebyshev extrapolation data edits 6. through 9. are suppressed, of course, if the process is not used; i.e., if NEXT0 < 2.

4.2 Feedback Loop Iteration Data

If $1 \leq \text{IFEED} \leq 3$ and $\text{IOUT3} > 0$ the code prints a line (at the top of the edit of the feedback iteration solutions) that tells the number of minor iterations required for the residual of Equation (66) to become less than ϵ_5 (on the first feedback iteration) and the number of feedback iterations required for the power distribution q^u to satisfy the convergence criterion on ϵ_4 in Equation (62).

4.3 Search Iteration Data

When a search is performed, data showing the status of the search convergence is edited after each search iteration. In a two line statement are printed the search iteration number, s ; the number of outer iterations required to converge the fission source iterations for the current

search iteration; the desired eigenvalue or generation factor, λ_d ; the value of the eigenvalue obtained from the current source iteration, λ_s ; the search convergence level, $|(\lambda^s - \lambda_d)/\lambda_d|$; the desired convergence level, ϵ_2 ; and the current value of the search parameter x^s . (For the special option, IGENOP=1, the search generation factor λ^s is the ratio of the total number of fission neutrons produced in the first outer iteration of the search iteration and the volume integral of the fixed volume source.) If ISERCH is negative, indicating that the search parameters must satisfy the convergence criterion $|(x^s - x^{s-1})/x^s| \leq \epsilon_3$, then the current convergence level $|(x^s - x^{s-1})/x^s|$ is also printed together with the desired convergence level ϵ_3 .

For each different kind of search, the search parameter x^s printed is:

1. Buckling search - $x^s = B^2$.
2. Poison search - $x^s = \Sigma_p^m$, where m is the material in region 1 if LRS(1) = 0; otherwise, m is the material specified by LRS(1).
3. Dimension search - x^s is the region width of region 1 if LRS(1) = 0; otherwise, x^s is the region specified by LRS(1).
4. Total fission source search - $x^s = F_{os}$.
5. Boundary search - x^s is the region width of the lefthand and (lower) of the two regions between which the boundary is moved.
6. Total power search - x^s is the total power P_{av} . See Equations (18) and (65).

5. PROBLEM SOLUTION EDITS

5.1 Nuclide Concentrations and Cross Section Contributions of Library Sets

If there are any library sets used in the time range, the data described as item 14 in the processed input data edits of Section III-3 above, can be edited at each time point. As described in that section these edits are called for with the IOUT2 flag, but the amount of pointwise data edited can be reduced by use of the ISOSUM and ISOED input data lists. The edits can also be restricted to regions specified in the IREDT input data list except for the pointwise cross section edits.

5.2 Eigenvalue and Source Integral Summaries

After printing the contents of the problem title card at the head of the page, the program edits a summary of the major problem results.

Provided there are fissions in the problem (i.e., some $v\Sigma f_j \neq 0$), the following data are edited.

If there is a fixed volume source, the program prints the volume integral $\int_V dv \sum_j S_{pj}$. The program then prints the special fixed volume source generation factor, $\int_V dv s_f^1 / \int_V dv \sum_j S_{ej}$; followed by the values of $\int_V dv s_f^1$ and $\int_V dv \sum_j S_{ej}$.

Next, the final generation factor or eigenvalue λ^t is printed with the number of outer iterations performed, t ; the convergence level [see Equations (123) and (125)]; and the desired convergence level, ϵ_1 . If pointwise convergence is called for ($NCONVG=1$), the values of λ^t and $\underline{\lambda}^t$ are printed with the source points at which they occurred.

If a search was performed, the final results are printed, including λ_d , λ^s , s , $|(\lambda^s - \lambda_d)/\lambda_d|$, ϵ_2 , and, if $ISERCH < 0$, the value of the search parameter, x^s , is printed unless the search was a poison, dimension, or boundary search. In the latter cases all the new values of parameters are listed.

5.3 Fission Source, Power, Group Fluxes, and Total Flux

If the integer input flag $IOUT1 \geq 1$, the fission source density, the power density (if any), and the group neutron flux distributions are edited for each mesh point, with both left- and right-hand values given at region interfaces. If $ILINES = n$, where $n \geq 2$, the values are edited for the lower mesh point of each region and for every n^{th} mesh point of the region thereafter.

The normalized fission neutron source distribution edited is the normalizing factor C times the unextrapolated source f^{*t+1} from Equation (120), where t is the last source iteration performed. The constant C is usually 1.0, such that the volume integral of the source is F_0 (FISSOR), but if the integer flag $NORMAL$ is greater than zero, C is set to the appropriate normalizing value and edited as the "renormalization factor".

The group fluxes edited are always C times the fluxes from which f^{*t+1} was computed by means of Equation (120). Hence, if the source iterations are converged such that $f^{*t+1} = f^t$, then the edited fluxes are those that would result from an input source of Cf^{*t+1} . This means that if a source distribution s_f were computed using the edited fluxes from either

$$\sum_j v\Sigma f_j \phi_j \quad \text{or} \quad \sum_j x_j \phi_j \quad (\text{for real and adjoint problems, respectively}),$$

then $s_f = \lambda^{t+1} C f^{*t+1}$ where λ^{t+1} is the generation factor on a renormalization type problem ($IREN=0$) and unity, otherwise.

The power distribution edited in this part of the output edits is computed from

$$p = \frac{1}{\lambda^t} \sum_j E_j \Sigma_{fj} \phi_j \quad (164)$$

where ϕ_j are the edited fluxes and λ^t is the last generation factor computed on renormalization type problems and unity, otherwise. The power distribution is edited if some $\Sigma_{fj} \neq 0$. The volume integral of this power distribution is normalized to equal the value of the input word POWER.

It should be noted that this power p is not, in general, normalized in the same manner as the power q that is actually used in the time dependent calculations. The volume integral of q is always $P_{av} = AVPOWER$.

5.4 Peak-to-Average Ratios for Fluxes, Source, and Power

If $IOUT1 \geq 2$, the ratio of the maximum value to the average value of the flux is edited for each group in each region. The mesh point at which each maximum occurs is also printed.

The peak-to-average flux ratio of all the reactor fuel regions is edited next for each group if any fissionable region exists (i.e., some $\Sigma_{fj} \neq 0$). This ratio is computed for each group from

$$\left| \frac{\phi_{jmax}}{\phi_{jav}} \right| = \phi_j \left| \begin{array}{l} \text{max of} \\ \text{all fuel} \\ \text{regions} \end{array} \right| \left/ \left[\sum_{\text{fuel regions}} \int_v \phi_j dv \right] / \left[\sum_{\text{fuel regions}} \int_v dv \right] \right. \quad (165)$$

The point where the maximum fuel flux occurs and the average fuel flux used in the denominator of Equation (165) is also edited for each group.

The peak-to-average ratio, the point of maximum, and the average value of the flux in the entire reactor are next given for each group.

Next, the total reactor volume and the fuel volume are printed; and this is followed by the average fission source value (the average of Cf^{*t+1}) and the total volume integral of the fission source if there are any fissionable regions. If any $\Sigma_{fj} \neq 0$ in any region, the average fission power density and the total are printed next. These values are the volume weighted average and the volume integral of the distribution defined by Equation (164).

For each fuel (fissionable) region, an edit is next made of the volume weighted average value of the fission neutron source (Cf^{*t+1}), the ratio of the maximum source to the average source in the region, the ratio of the maximum source in the region to the average source for all fuel regions, and the source point at which the maximum source occurs in the

region. The ratio of the maximum source value to the average over all fuel regions and the source point of the maximum are given next.

For each fuel region, the preceding edits are then repeated for the power distribution of Equation (164).

5.5 Neutron Balances and Averages by Region and Group

If $IOUT \geq 3$, an edit occurs that identifies, for each region, the following quantities for each group:

1. The volume integrated flux $\int_{V_r} \phi_j dv \equiv \int_{V_r} C_p x^p \phi_j dx$, where V_r (166)
is the region volume

2. The average flux $\int_{V_r} \phi_j dv / \int_{V_r} dv$ (167)

3. The neutrons produced $\frac{1}{\lambda t} \int_{V_r} v \Sigma_f \phi_j$, for real flux problems (168)
or $\frac{1}{\lambda t} \int_{V_r} x_j \phi_j dv$ for adjoint problems

4. The fission neutron source $x_j \int_{V_r} C f^t dv$ for real flux problems (169)
or $v \Sigma_f \int_{V_r} C f^t dv$ for adjoint problems

where C is the normalizing constant and f^t is the fission source used on the last iteration

5. The fixed volume source $\int_{V_r} S_{ej} dv = \int_{V_r} (S_{pj} + S_{rj}) dv$ (170)

6. The scattering-in source $\int_{V_r} \sum_{j' \neq j} \Sigma_{j' \rightarrow j} \phi_{j'} dv$ for real flux (171)
problems or $\int_{V_r} \sum_{j' \neq j} \Sigma_{j \rightarrow j'} \phi_{j'} dv$ for
adjoint problems

7. The absorption loss $\int_{V_r} (\Sigma_{aj} + T_j \Sigma_p) \phi_j dv$ (172)

8. The perpendicular leakage $\int_{V_r} D_j B_j^2 \phi_j dv$ (173)

9. The scattering-out loss $\int_{V_r} \sum_{j' \neq j} \Sigma_{j \rightarrow j'} \phi_j dv$ (174)

10. The parallel leakage loss $\int_{V_r} \nabla \cdot \bar{J}_j dv = - \int_{V_r} \nabla \cdot D_j \nabla \phi_j dv$ (175)

11. The parallel buckling $-\int_{V_r} \nabla \cdot D_j \nabla \phi_j dv / \int_{V_r} D_j \phi_j dv$ (176)

In all these equations, ϕ_j is the normalized flux (as edited). Some of these groupwise values are then summed over all neutron groups and edited as the "Summed Region Balance Data", together with each region volume and the region power, which is the region volume integral of the power distribution of Equation (164).

Totals (summed over all groups and regions) are finally printed out for the parallel (net) leakage, the absorption, the perpendicular leakage, the fission source (above), the fixed volume source, the power [Equation (164)], the flux times volume, and the out-scatter and the in-scatter.

5.6 Net Currents and Leakage at Region Boundaries

If IOU1 = 4, four quantities are printed for each neutron group in each region. The two net currents J_{lower} and J_{upper} at the lower and upper bounds, respectively, of the region are printed first. Then, the total leakages corresponding to these currents, that is, $L_{lower} = -J_{lower} C_p x_{lower}^p$ and $L_{upper} = J_{upper} C_p x_{upper}^p$, are printed.

The program solves for the lower and upper bound currents by solving balance equations consistent with the difference equations to Appendix B at the region boundaries. To solve these equations, the currents at the interval midpoints (adjacent to the boundary mesh point) are obtained in terms of flux differences to be consistent with the derivation of the difference equations. As a result, region parallel net leakage computed from $L_{lower} + L_{upper}$ can be subject to considerable round-off error, and they are therefore less accurate than the net leakage computed for 5.5 above. The latter leakages are computed from a regionwise balance for each group as simply the sum of all sources minus all other losses.

5.7 Plotted Output

If the user sets the integer input flag IPLOT (I-0,0,41,0) equal to 1, 2, or 3, then plotted output is obtained at each time point. Up to three spatial (versus x) distributions can be plotted on the frame (graph)

produced for the time point. The values of the integer input flags IVEK1, IVEK2, and IVEK3 (I=0,0,42-44,0) determine what is plotted, with each flag controlling one plotted distribution (plot vector) as follows.

IVEK1 Value	Distribution Plotted as Plot Vector Number 1(VEK1)
0	None
1	$Cf \cdot t^{+1}$, fission neutron source computed for output edits
2	ft , fission neutron source used on last iteration
3	p , power computed for output edits
4	ρ_1 , feedback vector 1
5	ρ_2 , feedback vector 2
6	ρ_3 , feedback vector 3
7	ϕ_1 , neutron flux of first (highest) energy group
8	$\phi_{j_{max}}$, neutron flux of thermal or lowest energy group
9	$\sum_{\ell \in L_r} N_{\ell}(k) \sigma_{aj_{max}}^{\ell}$, contribution of the library sets to the thermal absorption cross section
10	$\sum_{\ell \in L_r} N_{\ell}(k) \nu \sigma_{fj_{max}}^{\ell}$, contribution of the library sets to the neutron production cross section
11	q , the power that is used in the feedback calculations and that determines the normalization of the fluxes in the time dependent concentration calculations
12	Q , the coolant quality
13	H , the coolant enthalpy
14	T_{ℓ} , the coolant temperature

The value of the flag IVEK2 independently controls what is plotted if anything, as plot vector 2 (VEK2) exactly as described for IVEK1. If IVEK3 is a number ℓ greater than zero, the concentration of nuclide ℓ is plotted as plot vector 3 (VEK3) in each region that nuclide ℓ occurs in a library set.

The scale in the x direction is automatically chosen. The full x range of a problem is always included in the frame. The scale of the ordinate is controlled by the value of IPLIT and the maximum value of plot vector 1, 2, or 3 (VEK1, VEK2, or VEK3) depending on whether IPLIT is 1, 2, or 3, respectively. The scale is chosen such that the maximum value of the designated distribution is close to the top of the frame. The other distributions plotted on the same frame are plotted multiplied by a factor $10^{\pm n}$ (where n is an integer) such that the maxima plotted are as large as possible while still fitting in the frame with the chosen ordinate scale.

The user, by reading in values for VMAX1, VMAX2, and VMAX3 (F-0,0,45-47, 0), can cause the program to consider these values to be the maxima (of plot vectors 1, 2, and 3, respectively) that are used in determining the ordinate scaling.

IV. INPUT DATA PREPARATION

1. GENERAL COMPOSITION AND ORDER OF THE INPUT DATA

A DWARF problem consists of one or more "time ranges". The purpose of dividing a problem into a series of time ranges is to allow the user to change any of the basic input data, as desired, to represent a given change in reactor conditions; hence, a separate data deck is read for each time range of a problem. A time range consists of one or more "time steps". Since the new fluxes, sources, nuclide concentrations, and feedback effects are recalculated and edited (as the edit options may specify) after each time step, the user can also change the data for a time range simply to change the kind and quantity of data edited.

If microscopic (micro) data are required in one or more of the spatial regions of a problem, in a given time range, then there must exist a data set on Fortran logical unit 8 that includes the "library set" of data for each different set of micro data required. Library sets required for solution during a time range will be placed on unit 8 and be available to the given time range if the input data deck of each different library set is read (in the order of increasing library set number) immediately following the data deck of the time range. A data deck read for a given library set can add the data to unit 8 as a new library set, replace all the data of an already existing set, or simply update some of the data of an existing set at the option of the user.

Nuclide concentrations required for use with the micro data of a time range are included in the time range data deck. For the first time range of a problem, all such concentrations are determined by the input data. In a given subsequent time range, the initial concentrations will be the ones existing at the end of the preceding time range, except as amended (totally or simply in part) by the input data of the given time range. In fact, the saved concentrations are the only data that link the solution in one time range to another. The solutions are otherwise totally independent.

The time range data decks, followed by any required library set data decks, can be stacked indefinitely to form the input data deck of a problem. "Change case" capability is built in such that only those data the user desires to change need be input in the data decks of either a time range or a library set. The data decks of complete problems can also be stacked indefinitely with the same change case capability. However, only the initial (not the final) concentrations of the preceding problem are available.

Each time range data deck must always include

1. one time range title card,
2. one or more integer input data cards,
3. one or more floating point data cards.

Following the time range data there must be a data deck for each library set to be either added or updated on unit 8 as required for the time range. The data decks of any library sets needed in succeeding time ranges can also be read. The library set data decks must be stacked in the order of increasing library set number, but the numbers (1 through 999) need not be successive.

The data deck of each library set must always include

1. one library set title card,
2. one or more integer input data cards, and
3. one or more floating point input data cards.

2. DETAILED DESCRIPTION OF THE INPUT DATA

2.1 The Time Range Title Card

The card format (Fortran) is A5,10A6,1X,2I2.

The time range title card is punched as follows:

Columns 1-5	NUPROB	The Hollerith word "START" punched in these columns means that this is the first time range of a problem or that this is not a time dependent problem. Punching any other characters in these columns means that this time range is not the first one of the problem. Any nonzero character in column one causes the printed edits for the time range to begin on a new page.
Columns 6-65	TITLE	Any identifying Hollerith characters can be punched here. This title is printed at various stages of the printed output editing to help identify the data.
Column 66		Always blank
Column 68	IZERO	Zero or blank means no action is taken. 1 punched means all previous problem data are set to zero before starting the present problem. This is operable only on the first time range of a problem.
Column 70	LIST	Zero or blank means no action is taken. 1 punched means that the contents of each input data card will be printed as soon as it is read.

2.2 The Time Range Integer Input Data

The Fortran format for an integer input data card is:

The first 12 columns of an integer input data card are devoted to the identification of the data on the cards as follows:

Column	1	IEND	Zero or blank means other integer input data cards follow. 1 punched means this is the last integer input data card.
Columns	2-3	INDEX 1	The values of these four indices identify the integer input data punched on the card; hence, the card order is irrelevant.
Columns	4-5	INDEX 2	
Columns	6-8	INDEX 3	
Columns	9-10	INDEX 4	
Columns	11-12	IWC	This word tells the number of integer input words punched on the card (1 to 15). If IWC is zero or blank, a value of 15 is assumed.
Columns	13-72		One to 15 (format I4) integer input words can be punched in these columns.

There must be at least one integer input data card per time range.

The value of IWC specifies the number of successive words of input data given on the card, excluding the data control information in card columns 1 to 12.

The integer input data cards can be in any order so long as the last card has a 1 punched in card column 1, but the integer input data deck must follow the title card and precede the floating point data deck of the time range.

Each integer input data word must be right adjusted; that is, the number must end in the extreme right column of the four column field.

The single word integer input data (starting with JMAX and ending with LIBMTL) can be punched in the order of increasing INDEX 3 values across a card (provided the INDEX 3 value in card columns 6-8 is that of the first data word on the card).

The indexing scheme described, above, in which the numbers punched for each of the four indices uniquely identify the input data on the card,

is used also with time range floating point data, the library set integer data, and the library set floating point data. Hence, as an aid in identifying and locating related data elsewhere in the text, a reference to a word or array will often be followed by a parenthetical expression, as in:

```
MR(IR) (I-0,1,IR,0);  
DC(J,IM) (F-1,IM,J,1);  
ISONUM(K) (LI-0,2,K,0); and  
SIGPRO(J,1,K) (LF-4,J,K,2)..
```

The I, F, LI, and LF designate time range integer input data, time range floating point data, library set integer input data, and library set floating point data, respectively. The four quantities separated by commas are the respective values of Index 1, 2, 3, and 4 that identify the data. This referencing system enables the user to immediately locate the array and its description in this section, for the data are described first in order of increasing Index 3 (with Indices 1 and 2 zero), then increasing Index 2, and finally increasing Index 1.

Table II. Detailed Description of Time Range Integer Input Data

Index 1 Card Cols. 2-3	Index 2 Card Cols. 4-5	Index 3 Card Cols. 6-8	Index 4 Card Cols. 9-10	Mnemonic Name of Data	Notes on Application of Data
Format I2	Format I2	Format I3	Format I2		
0 0 0	0 0 0	1 2 3	0 0 0	JMAX IRMAX NGEO	Number of energy groups; $1 \leq JMAX \leq 4$. Number of regions; $1 \leq IRMAX \leq 40$. Type of geometry indicator 0 or blank indicates slab geometry. 1 indicates cylindrical geometry. 2 indicates spherical geometry.
0	0	4	0	NIBDRY	Inner boundary condition flag Let j be the energy group number, ϕ_j be the neutron flux, J_j be the neutron current, D_j be the diffusion constant in the region adjacent to the boundary, G_{1j} be a geometric correction factor and x be the one-dimensional space variable. Also, define $\omega_{1j} \equiv OMEGA1(J)$, $\alpha_{1j} \equiv ALPHA(J)$, $\beta_{1j} \equiv BETA1(J)$, and $\gamma_{1j} \equiv GAMMA1(J)$, where $ALPHA(J)$, $BETA1(J)$, $GAMMA1(J)$, and $OMEGA1(J)$ are floating point input (F-0,5,-,J) for group $j \equiv J$. 0 or blank means $\phi_j = 0$ (zero flux at boundary for all groups). 1 means $J_j = 0$ (symmetry or zero current for all groups). 2 means $\phi_j = \omega_{1j}$ (groupwise constant flux). 3 means $\frac{d\phi_j}{dx} = \omega_{1j}$ (groupwise constant derivative of the flux). 4 means $\frac{1}{\phi_j} \frac{d\phi_j}{dx} = 1/\omega_{1j}$ (groupwise extrapolated flux is zero at a distance ω_{1j} from the boundary). 5 means $\frac{1}{\phi_j} \frac{d\phi_j}{dx} = \frac{1}{.71(3D_j)}$ (groupwise extrapolated flux is zero at a distance $.71(3D_j)$ from the boundary). 6 means $\frac{1}{\phi_j} \frac{d\phi_j}{dx} = \frac{1}{.71(3D_j G_{1j})}$ (groupwise extrapolated flux is zero at a distance $.71(3D_j G_{1j})$ from the boundary). At the inner boundary, $G_{1j} = \frac{1+1.5207587(3D_j \rho)/(2x_1)}{1+.8103127(3D_j \rho)/(2x_1)}$, where x_1 is the value of x at the boundary and $\rho = 0$ for a slab, 1 for a cylinder, and 2 for a sphere. See Reference 8. 7 means $\alpha_{1j}\phi_j - \beta_{1j}J_j + \gamma_{1j} = 0$ (groupwise application of general first order, linear boundary condition). 8 means $\phi_{1j} = \phi_{2j}$ and $J_{1j} = J_{2j}$, where the subscripts 1 and 2 denote the inner and outer boundaries, respectively (periodic boundary condition).
0	0	5	0	NOBDRY	Outer boundary condition flag Let j be the energy group number, ϕ_j be the neutron flux, J_j be the neutron current, D_j be the diffusion constant in the region adjacent to the boundary, G_{2j} be a geometric correction factor and x be the one-dimensional space variable. Also, define $\omega_{2j} \equiv OMEGA2(J)$, $\alpha_{2j} \equiv ALPHA2(J)$, $\beta_{2j} \equiv BETA2(J)$, and $\gamma_{2j} \equiv GAMMA2(J)$, where $ALPHA2(J)$, $BETA2(J)$, $GAMMA2(J)$, and $OMEGA2(J)$ are floating point input (F-0,5,-,J) for group $j \equiv J$. 0 or blank means $\phi_j = 0$ (zero flux at boundary for all groups). 1 means $J_j = 0$ (symmetry or zero current for all groups). 2 means $\phi_j = \omega_{2j}$ (groupwise constant flux). 3 means $\frac{d\phi_j}{dx} = \omega_{2j}$ (groupwise constant derivative of the flux). 4 means $\frac{-1}{\phi_j} \frac{d\phi_j}{dx} = 1/\omega_{2j}$ (groupwise extrapolated flux is zero at a distance ω_{2j} from the boundary). 5 means $\frac{-1}{\phi_j} \frac{d\phi_j}{dx} = \frac{1}{.71(3D_j)}$ (groupwise extrapolated flux is zero at a distance $.71(3D_j)$ from the boundary). 6 means $\frac{-1}{\phi_j} \frac{d\phi_j}{dx} = \frac{1}{.71(3D_j G_{2j})}$ (groupwise extrapolated flux is zero at a distance $.71(3D_j G_{2j})$ from the boundary). At the outer boundary, $G_{2j} = 1/[1+.710446(3D_j \rho)/(2x_1)]$, where x_1 is the value of x at the boundary and $\rho = 0$ for a slab, 1 for a cylinder, and 2 for a sphere. See Reference 8. 7 means $\alpha_{2j}\phi_j - \beta_{2j}J_j + \gamma_{2j} = 0$ (groupwise application of general first order, linear boundary condition). 8 means $\phi_{1j} = \phi_{2j}$ and $J_{1j} = J_{2j}$, where the subscripts 1 and 2 denote the inner and outer boundaries, respectively (periodic boundary condition). Note: $J_j = -D_j \frac{d\phi_j}{dx}$. Also, the constant ".71" above is actually .710446.

Table .II (Continued)

Index 1 Card Cols. 2-3 Format I2	Index 2 Card Cols. 4-5 Format I2	Index 3 Card Cols. 6-8 Format I3	Index 4 Card Cols. 9-10 Format I2	Mnemonic Name of Data	Notes on Application of Data
0	0	6	0	IREN	Fission neutron source "renormalization" flag 0 or blank indicates a normal "eigenvalue" or generation factor problem and it causes the total volume integral of the fission neutron source to be set to the value FISSOR (F-0,0,9,0) at the beginning of each fission source iteration. The problem may be either homogeneous or inhomogeneous, as noted below. Also, the fluxes used to compute new concentrations will be normalized such that the power AVPOWR is produced (F-0,0,20,0). 1 indicates an inhomogeneous problem (one with fixed volume sources or inhomogeneous boundary conditions) and any fission neutron source is not renormalized at the beginning of a fission source iteration. Also, the fluxes used to compute new concentrations will not be normalized. Note: If IREN=1 and fissions occur, the eigenvalue of the problem must be less than 1.0 for a solution to exist. Certain inhomogeneous problems can be run more quickly by setting IREN=0 and doing a search on the value of FISSOR by setting ISERCH=4 (I-0,0,61,0).
0	0	7	0	IADJ	Specifies type of flux solution 0 or blank indicates only the real or normal flux solution is to be performed. 1 indicates only the adjoint flux (or importance function) solution is to be performed.
0	0	8	0	IGENOP	Specifies formula for search generation factor 0 or blank specifies that the generation factor to which a search applies is the usual ratio of total fission neutrons produced per fission source neutron introduced (on the last source iteration). 1 specifies that the formula for the generation factor which is searched on is the ratio of fission neutrons produced on the first outer (source) iteration per fixed volume source neutron introduced.
0 0 0	0 0 0	9 10 11	0 0 0	INBUCK	Ignored Ignored Specifies type of input buckling 0 or blank specifies that the bucklings can vary by region or material and by energy group and that they are input directly as BSQM(J,K) (F-6,K,J,0) or they are computed for a v. macro material. 1 specifies that the bucklings can vary only by region or material and that they are input as BSQR(K) (F-0,2,K,0) or they are computed. 2 specifies that the bucklings can vary only by energy group and that they are input as BSQM(J,1), the values for region or material 1. (This means there must be a set of data read in for BSQM(J,1) even though there need not be a reference to material number 1 elsewhere in the problem.) 3 specifies that the bucklings vary neither by group nor by region or material and that they are input as BSQM(1,1), the value for region or material 1, group 1. (This value must be input regardless of whether or not BSQM(1,1) is used elsewhere in the problem.) 4 specifies that the buckling data is that saved from the previous time range. Note: The flag IBSQO, immediately below, determines whether the index K in BSQM(J,K) or BSQR(K) is interpreted as a macro material number (IBSQO=0) or a region number (IBSQO=1). When INBUCK ≤ 1, the bucklings for a variable macro material are determined by the variable buckling data. See BCOF(K,J,IM) (F-19,IM,J,K) and COFMO(K,7,IM) (F-20,IM,7,K).
0	0	12	0	IBSQO	Specifies whether the buckling data read is indexed by fixed macro material number or by region number. See the note under INBUCK, above. (A zero or blank means input by material; a 1 means input by region.)
0 0	0 0	13 14	0 0	IDCO	Ignored Specifies type of diffusion coefficient formula. Let $D_j(k)$ be the diffusion coefficient at source point k. 0 or blank specifies that $D_j(k)$ is computed from macro material data, only, as described in Section II-1.2. 1 specifies that $D_j(k) = 1/[3(\Sigma_{uj}(k) + T_{jp}^{m,m})]$, where $T_{jp}^{m,m} \equiv T(J,IM) \text{SIGPT}(IM)$ is the poison cross section of macro material m and $\Sigma_{uj}(k)$ is computed as shown in Section II-1.2. 2 specifies that $D_j(k)$ is the so-called asymptotic value computed from $D_j(k) = \Sigma_{rj}(k)/\kappa_j^2$, where κ_j is the solution of the transcendental equation $[\kappa_j/\Sigma_j(k)] = \tanh\{\kappa_j/[\Sigma_j(k) - \Sigma_{rj}(k)]\}$, and $\Sigma_{rj}(k) = \Sigma_{aj}(k) + \Sigma_{sj}(k) + T_{jp}^{m,m}$, $\Sigma_j(k) = \Sigma_{uj}(k) + T_{jp}^{m,m}$. See Section II-1.2 for the formulas of $\Sigma_{aj}(k)$, $\Sigma_{sj}(k)$ and $\Sigma_{uj}(k)$. Note: If any macro materials are listed in the IYUK(K) list (I-0,4,K,0) then the formula of IDCO=0 is applied to all materials not included in the list.

Table II (Continued)

Index 1 Card Cols. 2-3	Index 2 Card Cols. 4-5	Index 3 Card Cols. 6-8	Index 4 Card Cols. 9-10	Mnemonic Name of Data	Notes on Application of Data
Format I2	Format I2	Format I3	Format I2		
0	0	15	0	ISORS	Specifies method of automatic calculation of pointwise fixed volume source Pointwise fixed volume sources can be input directly as SEO(K,J) (F-10,J,K,0) for source point K and group J. The SEO(K,J) fixed volume sources can also be computed automatically as a function of fluxes obtained as a solution of the preceding problem. Define $\phi(I,J)$ as the flux at mesh point I for group J, and a source point K in the IRth region is defined by $K=I+IR-1$. If IM is the macroscopic material in the IRth region, then 0 or blank means, as in the normal case, the directly input values of SEO(K,J) remain. 1 means $SEO(K,J) = 0$. 2 means $SEO(K,J) = SIGNU(J,IM)\phi(I,J)$. 3 means $SEO(K,J) = CHIM(J,IM)\phi(I,J)$. 4 means $SEO(K,J) = SIGFIS(J,IM)\phi(I,J)$. 5 means $SEO(K,J) = ENERGY(J,IM)\phi(I,J)$. 6 means $SEO(K,J) = SIGMU(J,IM)\phi(I,J)$. 7 means $SEO(K,J) = SIGA(J,IM)\phi(I,J)$. 8 means $SEO(K,J) = T(J,IM)\phi(I,J)$. SIGNU(J,IM) is neutrons per fission times the fission cross section. CHIM(J,IM) is the fraction of fission neutrons released in group J for material IM. SIGFIS(J,IM) is the fission cross section. ENERGY(J,IM) is the energy per fission. SIGMU(J,IM) is the so-called "transport" cross section. SIGA(J,IM) is the absorption cross section. T(J,IM) is the fraction of the poison cross section SIGPT(IM) added to group J.
0	0	16	0	ISTORE	Flag controlling read in of stored concentrations for successive problems 0 or blank means any concentrations input will be read from cards, only. 1 means that the initial concentrations used in the first time range of the preceding problem will be read from auxiliary storage (unit 4) and used in the first time range of the present problem. The concentration data read can be replaced in part or full by card input, if desired.
0	0	17	0	ICHIM	Ignored
0	0	18	0		Ignored
0	0	19	0		Material independent fission spectra indicator 0 or blank indicates that the material dependent fission spectrum data $x_j = CHIIM(J,IM)$ are to be used, as input. $n > 0$ causes the fission spectrum data for all fissionable materials (those with some $v\sigma_{fj} \neq 0$) to be set equal to the fission spectrum data input for macro material $m = n$.
0	0	20	0	INED1	Ignored
0	0	21	0		Input data edit control flag 0 or blank causes all relevant, nonzero input data to be edited. 1 causes complete suppression of all the input data edits, with the exception of the library set edits which are controlled by LIBED (I-0,0,31,0).
0	0	22	0	INED2	Reserved
0	0	23	0	INED3	Reserved
0	0	24	0	IOUT1	Ignored
0	0	25	0		First output data edit control flag (for each time point) 0 or blank means only a summary of the problem solution will be edited, such as the eigenvalue, various convergence data, the results of searches, etc. 1 causes the pointwise fluxes, fission neutron source, and power (if any) to be listed, in addition. 2 causes additional editing of peak to average ratios for the fluxes by group and region as well as the fission neutron source and the power density by fuel (fissioning) region. 3 causes additional editing of detailed neutron balance data by region and neutron group, parallel bucklings, and other integral data for the entire reactor. 4 causes additional editing of the neutron net currents and neutron leakages at the region boundaries.
0	0	26	0	IOUT2	Second output data edit control flag (for each time point) 0 or blank suppresses all editing of library set contributions. 1 causes edits of the volume weighted average concentrations for each nuclide in each region. 2 causes additional editing of the regionwise averages of the contributions of the library set nuclides to the total cross sections. 3 causes additional editing of the pointwise concentrations of the nuclides. 4 causes additional editing of the pointwise contributions of the library set nuclides to the total cross sections. See also IREDT(K) (I-0,10,IR,0).

Table II (Continued)

Index 1 Card Cols. 2-3	Index 2 Card Cols. 4-5	Index 3 Card Cols. 6-8	Index 4 Card Cols. 9-10	Mnemonic Name of Data	Notes on Application of Data															
Format I2	Format I2	Format I3	Format I2																	
0	0	27	0	IOUT3	Third (feedback) output data edit control flag (for each time point) 0 or blank suppresses printing of the feedback data distributions. 1 causes editing of the computed distributions for $\rho_1(i)$, $\rho_2(k)$, $H(i)$, etc.															
0	0	28	0	ILINES	Pointwise flux edit frequency control 0 or blank means the source, power, and fluxes at every mesh point are listed. n means that only every nth mesh point is edited per region.															
0	0	29	0	ITDATA	Fission neutron source data edit control 0 or blank causes suppression of the fission source iteration data. 1 causes an edit, each outer iteration, of the generation factor (or eigenvalue) and other data that show the convergence of the fission source iteration.															
0	0	30	0	LIBED	Ignored															
0	0	31	0		Flag controlling library set edits (for each time range) 0 or blank means no edits are done. 1 causes editing of the data of each library set used in the time range after any writing or updating of the library set data on unit 8. 2 causes editing of the data of every library set on unit 8.															
0	0	32	0	NORMAL	Ignored															
0	0	33	0		Ignored															
0	0	34	0		Ignored															
0	0	35	0		Ignored															
0	0	36	0		Specifies the output edit "normalization" (does not affect any feedback or depletion normalization) 0 or blank specifies that all output data (fluxes, fission source, power, and integrals of these quantities are to be unnormalized; i.e., they simply correspond to the fission neutron source magnitude as determined by the floating point input data word FISSOR (F-0,0,9,0). 1 specifies that all output data are normalized such that the total fission power rate equals POWER (F-0,0,11,0). Energies per fission and fission cross sections must be nonzero in some material IM and group J of the problem. 2 specifies that all output data are multiplied by the reciprocal of the neutron flux $\phi(I,J)$ in group J=NOGRP at mesh point I = MESHNO. See (I-0,0,37,0) and (I-0,0,38,0), below. The following table shows which normalizing flux $\phi(I,J)$ is used if NOGRP or MESHNO is zero or blank.															
						<table><tr><td colspan="4">MESHNO</td></tr><tr><td colspan="2"></td><td>0</td><td>Nonzero</td></tr><tr><td rowspan="2">NOGRP</td><td>0</td><td>The maximum flux</td><td>The maximum flux at MESHNO</td></tr><tr><td>Non-Zero</td><td>The maximum flux in NOGRP</td><td>The flux value at MESHNO in NOGRP</td></tr></table>	MESHNO						0	Nonzero	NOGRP	0	The maximum flux	The maximum flux at MESHNO	Non-Zero	The maximum flux in NOGRP
MESHNO																				
		0	Nonzero																	
NOGRP	0	The maximum flux	The maximum flux at MESHNO																	
	Non-Zero	The maximum flux in NOGRP	The flux value at MESHNO in NOGRP																	
					3 specifies that all output are normalized such that the fission neutron source density is unity at the source point K=MESHNO, unless MESHNO is zero. In the latter case, the data are normalized such that the maximum fission neutron source density is unity. (Source point K=I+IR-1 at mesh point I in region IR.)															
					4 specifies that all output data are simply multiplied by FLMULT (F-0,0,10,0).															
0	0	37	0	MESHNO	Mesh point or source point specification for output data "normalization" control when NORMAL equals to 2 or 3, above.															
0	0	38	0	NOGRP	Group number specification for output data "normalization" control when NORMAL equals to 2, above.															
0	0	39	0	IGRID	Ignored															
0	0	40	0		Flag for background grid control on plotted output (see IPLOT, below) 0 or blank suppresses plotting of any background grid lines 1 causes plotting of a coarse background grid of vertical and horizontal lines at each tic mark on the ordinate and abscissa. 2 causes the addition of a fine background grid of both vertical and horizontal lines between those of the coarse grid.															
0	0	41	0	IPLOT	Flag for plotted output and ordinate scale control 0 suppresses all plotted output. n ($1 \leq n \leq 3$) means plotted output is required and the maximum value on the ordinate scale will be the value found for plot vector n. See VMAX1, VMAX2, and VMAX3 (F-0,0,45-47,0) and IVEC1, IVEC2, and IVEC3 immediately following.															

Table II (Continued)

Index 1 Card Cols. 2-3 Format I2	Index 2 Card Cols. 4-5 Format I2	Index 3 Card Cols. 6-8 Format I3	Index 4 Card Cols. 9-10 Format I2	Mnemonic Name of Data	Notes on Application of Data
0	0	42	0	IVEK1	Specifies the quantity plotted as plot vector number 1 0 or blank suppresses plotting of plot vector 1. n ($1 \leq n \leq 14$) means plot vector 1 will be plotted and the quantity plotted will be: 1-output fission neutron source, $Cf \cdot t+1$; 2-fission neutron source used on last iteration, ft ; 3-power, p ; 4-feedback vector number 1, p_1 ; 5-feedback vector number 2, p_2 ; 6-feedback vector number 3, p_3 ; 7-first group flux, ϕ_1 ; 8-thermal group flux, ϕ_{jmax} ; 9-thermal contribution to the absorption cross section, $\sum_{k \in L_r} N_k(k) \sigma_{a,jmax}^k$; 10-thermal contribution to the neutron production cross section, $\sum_{k \in L_r} N_k(k) \nu \sigma_{f,jmax}^k$; 11-the feedback power, q ; 12-the quality, Q ; 13-the enthalpy, H ; 14-fluid temperature, T_L .
0	0	43	0	IVEK2	Specifies the quantity plotted as plot vector number 2 0 or blank suppresses plotting of plot vector 2.
0	0	44	0	IVEK3	n ($1 \leq n \leq 14$): same as above for IVEK1. Specifies the concentration plotted as plot vector number 3 0 or blank suppresses plotting of plot vector 3 x causes the concentration $N_k(k)$ to be plotted in all regions in which a library set contains a nuclide x.
0	0	45	0	INSORS	Ignored
0	0	46	0		Specifies type of fission neutron source guess 0 or blank indicates a unit, flat source density is automatically used in all fission neutron producing regions (those for which some $SIGNU(J,IM)$, the neutrons per fission times the fission cross section, is nonzero). ±1: The source density saved from the preceding problem is used, provided no $F(K)$ ($F-0,10,K,0$) values are read in. See the next fixed point word, IGESS ($I-0,0,47,0$). ±2: The source density used in each region is flat and, for the IR^{th} region, is equal to $FR(IR)$ ($F-0,3,IR,0$). Note: $FR(IR)$ data is not saved from problem to problem whenever $INSORS=0$ ($I-0,0,46,0$). ±3: The source density used will be that read in from cards as problem floating point data, $F(K)$ ($F-0,10,K,0$), unless none is read in; then that saved from a previous problem is used. ±4: The fission neutron source guess is set to zero.
0	0	47	0	IGESS	A negative value of INSORS causes the source guess to be edited, provided any problem input data edits are called for. Specifies which fission neutron source is saved for next problem 0 or blank specifies that the fission neutron source distribution saved for use as a first guess on the next problem will be the one obtained as the solution on the last outer iteration. 1 specifies that the source distribution saved for use as a guess on the next problem will be the next to the last one obtained; or, identically, it will be the one used as the source on the last source iteration performed. This allows a series of one source iteration problems to be run with each using the same fission neutron source guess. See MAXOUT ($I-0,0,49,0$).
0	0	48	0	NCONVG	Specified convergence on total or pointwise fission neutron source 0 or blank specifies that the fission neutron source iterations will be terminated on the basis of the convergence of the total volume integral of the fission neutron source. 1 specifies that the outer iterations will be terminated on the basis of the convergence of the fission neutron source density at each source point. The outer iterations are terminated when the change from the preceding iteration is less than $ \pm EPS $ ($F-0,0,1,0$). Also see MAXOUT immediately following.
0	0	49	0	MAXOUT	Maximum number of fission neutron source iterations 0 or blank causes MAXOUT to be automatically set to 50. 1 indicates that only one solution will be performed for the neutron fluxes and the associated fission neutron source distribution. n indicates that the fission neutron source (iterations will be terminated on the n^{th} source iteration, unless the iterations are terminated earlier by meeting the convergence criterion imposed by $EPS1$ ($F-0,0,1,0$)).

Table II (Continued)

Index 1 Card Cols. 2-3 Format I2	Index 2 Card Cols. 4-5 Format I2	Index 3 Card Cols. 6-8 Format I3	Index 4 Card Cols. 9-10 Format I2	Mnemonic Name of Data	Notes on Application of Data
0	0	50	0	NEXT0	Specifies fission neutron source extrapolation technique 0 or blank indicates no extrapolation will be done. 1 indicates that the convergence of the fission neutron source distribution is to be accelerated by a two-point extrapolation technique. The extrapolated value at each source point k on the t th iteration is $f_k^t = f_k^{*t} + \theta(f_k^{*t} - f_k^{t-1})$, where the $*$ indicates the unextrapolated value of the source density f . [θ is the input word THETA (F-0,0,8,0)]. 2 indicates that a three-point Chebyshev polynomial extrapolation technique is to be used to accelerate the outer iterations. The extrapolation is $f_k^t = f_k^{t-1} + \alpha_2(f_k^{*t} - f_k^{t-1}) + \beta_2(f_k^{t-1} - f_k^{t-2})$. See Section II-2.3.
0	0	51	0	NPIO	Specifies the number of preliminary unextrapolated fission source iterations performed before beginning Chebyshev acceleration 0 or blank causes NPIO to be automatically set to 4. n causes $n+1$ fission neutron source distributions to be computed with no extrapolation before a dominance ratio is estimated and an extrapolated fission source is computed for use as the guess on the $n+2$ iteration, or flux solution. For example, if NPIO=4 an extrapolated source is computed after the fifth solution for the fluxes and source and this extrapolated source is used as the guess for the sixth flux and source solution.
0	0	52	0	NCHMAX	Specifies the maximum order of Chebyshev polynomial to be allowed during Chebyshev extrapolation of the fission neutron source 0 or blank causes NCHMAX to be automatically set to 15. n causes the Chebyshev extrapolation coefficients α_2 and β_2 to be computed for successive values of 2 from 1 through no more than $n+1$ before beginning again with $2=1$. See NEXT0, above.
0	0	53	0	NERCHK	Specifies number of extra Chebyshev extrapolations to be performed with the existing dominance ratio before a new estimate can be used 0 or blank causes NERCHK to be interpreted to be equal to 2. n causes $n+1$ more fission neutron sources to be extrapolated with the α_2 and β_2 Chebyshev polynomial coefficients corresponding to the existing dominance ratio σ_0 before beginning a new extrapolation series with a new estimate of σ_0 . See NEXT0, above.
0	0	54	0	IFEED	Note: A new Chebyshev extrapolation series starting with $2=1$ is also begun (with the most recent estimate of the dominance ratio σ_0) if 2 exceeds NCHMAX+1. See NCHMAX, above. Flag for feedback control and specification of how feedback vector number 1, ρ_1 , is computed 0 or blank means there is to be no feedback. That is, although distributions may be input for the feedback vectors (ρ_1 , ρ_2 , and ρ_3) and there may be variable macro materials in the problem that are dependent on the input distributions, no feedback vector distributions will be recomputed as a function of energy depositions by means of the feedback routines. n ($1 \leq n \leq 3$) means that there is feedback and ρ_1 will be recomputed in the feedback routines. The built-in algorithms compute ρ_1 according to n as follows: 1 - the density of the axially flowing coolant; 2 - the enthalpy of the axially flowing coolant, H ; 3 - the feedback power density, q .
0	0	55	0	JFEED	Flag for specification of how feedback vector number 2, ρ_2 , is computed (provided IFEED, above, is 1 to 3) 0 or blank means ρ_2 is not recomputed in the feedback routines. Any distribution input directly can be used in computing constants for variable macro materials, however. n ($1 \leq n \leq 3$) means that ρ_2 will be computed in the feedback routines. The built-in algorithms compute ρ_2 according to n as follows: 1 - the average temperature of the fuel; 2 - the enthalpy of the axially flowing coolant, H ; 3 - the feedback power density, q .
0	0	56	0	KFEED	Flag for specification of how feedback vector number 3, ρ_3 , is computed (provided IFEED, above, is 1 to 3) There are no built-in algorithms for computing ρ_3 or for using it if input.

Table II (Continued)

Index 1 Card Cols. 2-3 Format I2	Index 2 Card Cols. 4-5 Format I2	Index 3 Card Cols. 6-8 Format I3	Index 4 Card Cols. 9-10 Format I2	Mnemonic Name of Data	Notes on Application of Data
0	0	57	0	NFEEDL	Maximum number of feedback iterations 0 or blank causes NFEEDL to be set to 3. n means that, for a given time point, the feedback vectors ρ_1 , ρ_2 , and ρ_3 will be computed a maximum of n times. A "feedback iteration" begins with the calculation of new flux distributions using the new nuclide concentrations, if any, of the time point, but using the last computed ρ 's from the previous time point. The feedback vectors are then recomputed in the feedback routine (provided IFEEED > 0) as a function of the new fluxes. A new feedback iteration is then begun. The iterations are terminated after the new flux calculation when n new ρ 's have been computed or when the power distribution q is point-wise converged according to $ (q^u - q^{u-1})/q^{u-1} < \epsilon_4$.
0	0	58	0	IRHO	Flag controlling the initial distribution of feedback vector number 1 for the time range 0 or blank causes ρ_1 to be set to zero everywhere. 1 causes ρ_1 to be set to unity everywhere. 2 causes ρ_1 to be set to RHOGES (F-0,0,34,0) everywhere. 3 causes ρ_1 to remain as it was at the end of the previous time range or as it was input with RHO(I) (F-0,7,1,0) in the present time range.
0	0	59	0	IPHO	Flag controlling the initial distribution of feedback vector number 2 for the time range 0 or blank causes ρ_2 to be set to zero everywhere. 1 causes ρ_2 to be set to unity everywhere. 2 causes ρ_2 to be set to PHOGES (F-0,0,35,0) everywhere. 3 causes ρ_2 to remain as it was at the end of the previous time range or as it was input with PHO(K) (F-0,8,K,0) in the present time range.
0	0	60	0	ISHO	Flag controlling the initial distribution of feedback vector number 3 for the time range 0 or blank causes ρ_3 to be set to zero everywhere. 1 causes ρ_3 to be set to unity everywhere. 2 causes ρ_3 to be set to SHOGES (F-0,0,36,0) everywhere. 3 causes ρ_3 to remain as it was at the end of the previous time range or as it was input with SHO(K) (F-0,9,K,0) in the present time range.
0	0	61	0	ISERCH	Flag controlling type of reactor parameter search 0 or blank specifies that no parameter search is to be made. ±1 specifies that a "buckling search" is to be made for a single value of B^2 applied in all search regions that do not have v. macro materials. ±2 specifies that a "poison search" is to be done for a factor multiplying the respective SIGPT(IM) (F-0,4,IM,0) values of the search materials. ±3 specifies that a "dimension search" is to be made for the factor multiplying the respective RW(IR) (F-0,1,IR,0) values of the search regions. ±4 specifies that a "fission source search" is to be made for the total number of fission neutrons produced, FISSOR (F-0,0,9,0). ±5 specifies that a boundary search is to be made for the position of the boundary between the two search regions. ±6 specifies that a "power search" is to be made for the total power, $P_{av} \equiv AVPOWR$ (F-0,0,20,0) used to normalize the fluxes used in the feedback and nuclide concentration calculations. Also, see Section II-3 and LRS(K) (I-0,3,K,0).

Table II (Continued)

Index 1 Card Cols. 2-3 Format I2	Index 2 Card Cols. 4-5 Format I2	Index 3 Card Cols. 6-8 Format I3	Index 4 Card Cols. 9-10 Format I2	Mnemonic Name of Data	Notes on Application of Data
0	0	62	0	MAXSCH	Maximum number of search trials 0 or blank causes MAXSCH to be automatically set to 20. n causes the search to be terminated after n different values of the search parameter (such as the buckling value, poisoning values, etc.) have been tried in the course of the search. If a search is terminated by reaching the number of trial set by MAXSCH, a normal output edit is performed, regardless, for the last value of the search parameter, just as in the case of normal satisfaction of the search criterion.
0	0	63	0	NSERCH	Indicator of how the desired generation factor EIGEND for a search is to be set For a fission source search (ISERCH = ±4), the value of NSERCH is ignored because EIGEND, the desired search eigenvalue, is automatically unity. For all other kinds of searches, the value of EIGEND is set as follows. 0 or blank indicates that EIGEND, the desired search eigenvalue, is either input directly as floating point data (F-0,0,14,0) or, if not, it is the value of EIGEND retained from the preceding problem. 1 indicates that EIGEND is to be set equal to the eigenvalue computed on the preceding problem.
0	0	64	0	IENPF	Flag controlling how the energy per fission values are set 0 or blank causes all E_f^m to be set to EPFISS (F-0,0,15,0). 1 causes E_f^m for each group of macro material m to be set to ENERGY(1,IM) (F-1,IM,J,1 and 5), the first group value. 2 leaves E_f^m as read directly with the ENERGY(J,IM) data. This flag affects macro material values, only. It cannot be used to automatically set the $E_f^j \equiv EFIS(L,J)$ data of the library set data.
0 0	0 0	65 66	0 0	LIBOLD	Ignored Identification number of library set data saved from a previous computer run on unit 8 When library set data are to be used in a given computer run, a data set with the library set data must exist, either as saved from a previous run or written or updated on the current run with library set card input decks. The first record written on a library data set is the word LIBNO, and this word serves as user's data set identification number. 0 or blank indicates that no data set of library sets from a previous computer run exists, and LIBOLD is automatically set equal to LIBNEW below for succeeding time ranges or problems. n > 0 indicates that at the beginning of the run, LIBOLD is to be compared to LIBNO of the existing library set data to assure that the proper data set has been found. The run is terminated if the two numbers are not identical.
0	0	67	0	LIBNEW	Identification number to be written at the beginning of the data set with the library set data if it is either newly written or updated following the current time range data decks 0 or blank is illegal if LIBOLD is 0 or blank. 0 or blank causes LIBNEW to be set automatically to LIBOLD if LIBOLD is nonzero. Nonzero causes the new library material data set identification number (LIBNO) to be set equal to LIBNEW regardless of the value of LIBOLD.
0	0	68	0	LIBMTL	The number of library set data decks that are to be read following the current time range data decks 0 or blank indicates no library set data decks are to be read following the data decks of the current time range. n indicates n library set decks are to be read immediately following the data decks of the current time range. Note: Whenever LIBMTL is nonzero and new library set data are read from cards, this causes any existing data set of library sets to be completely rewritten as the same data set with the new data incorporated. Therefore, care should be taken to avoid changing data sets unintentionally.
0	0	69	0	MODIFY	Flag controlling the automatic checking for illegal equality of nuclide loss terms in each library set 0 or blank causes no checking. 1 causes the λ_k and σ_a^k for each nuclide in a library set to be checked against those of every other nuclide to insure that they are unequal. If equal values are found, the one of higher λ is modified by the factor 1.001. See Section II-4.
0	0	70	0	ISHTDN	Flag specifying a zero flux, shutdown reactor condition 0 or blank causes the flux solutions to be computed, as usual. 1 causes suppression of the neutron flux and power solutions to greatly speed the time dependent solutions during the time range when the reactor is in a shutdown state.

Table II (Continued)

Index 1 Card Cols. 2-3	Index 2 Card Cols. 4-5	Index 3 Card Cols. 6-8	Index 4 Card Cols. 9-10	Mnemonic Name of Data	Notes on Application of Data
Format I2	Format I2	Format I3	Format I2		
0	1	IR (region number)	0	MR(IR)	The number of the macroscopic material to be used in region number IR. The number IR punched as Index 3 indicates that the first word in the data field of the card is the macroscopic material number to be used in the IRth region; the next word pertains to the region number IR+1, etc. Values of MR(IR) less than or equal to 40 denote fixed macro materials, while values greater than 40 denote variable macro materials that are functions of the feedback vectors. Each region must be assigned a macro material number whether or not a library set of micro material data is assigned to the region by means of the MRMIC(IR) list.
0	2	IR (region number)	0	NI(IR)	The number of mesh intervals to be used in region number IR. The value of Index 3, IR, indicates that the first data word on the card is the number of intervals into which the width of the IRth region is to be partitioned. The next data word on the card pertains to the IR+1 region, etc. This partitioning defines the number of mesh points IMAX, since $IMAX = 1 + \sum_{IR=1}^{IRMAX} NI(IR)$, where IRMAX is the number of regions (less than or equal to 40).
0	3	K (list word number)	0	LRS(K)	List of specific materials or regions to which a search applies. If a search is to be performed [see ISERCH (I-0,0,61,0)]. This LRS(K) list specifies the materials or regions in which the search parameters (buckling, poisoning, etc.) are to be changed. If LRS(1) = 0 or blank the search will apply automatically to all materials or regions or to the first and second regions of a boundary search. If LRS(1) is nonzero, the search parameter is changed in only those materials or regions referred to by number in the LRS(K) list. Each word in the LRS(K) list is a macroscopic material number or a region number. The value of Index 3 (K) indicates that the first data word on the card is the Kth word in the LRS(K) list, the next word on the card is the K+1 word in the list, etc. The LRS(K) list is assumed to be terminated if a zero or blank word in the list is encountered. The various searches (ISERCH values) cause the LRS(K) list to be interpreted in the following ways. For ISERCH = ±1, a buckling search, LRS(K) is the list of regions to which the search applies (all regions if LRS(1) = 0). For ISERCH = ±2, a poison search, LRS(K) is the list of macroscopic materials to which the search applies (all materials if LRS(1) = 0). For ISERCH = ±3, a dimension search, LRS(K) is the list of regions to which the search applies (all regions if LRS(1) = 0). For ISERCH = ±4, a fission source search, LRS(K) is not applicable. For ISERCH = ±5, a boundary search, the value of LRS(1) is the leftmost of the two regions to which the search applies. (The leftmost region is assumed to be region 1 if LRS(1) = 0.) For ISERCH = ±6, a power search, LRS(K) is not applicable.
0	4	K (list word number)	0	IYUK(K)	List of specific macro materials in which the diffusion coefficients are to be computed automatically from the formulae of Section II-1.2. When IDCO > 0 and IYUK(1) is zero, the diffusion coefficients are computed from the formulae in all materials. See IDCO (I-0,0,14,0). Each word in the list is a macro material number as assigned by the MR(IR) data.
0	7	IR (region number)	0	MRMIC(IR)	The number of the library set to be used in region number IR. In each region that microscopic cross section data are to be used in computing the neutron group constants, a library set number must be assigned by means of the MRMIC(IR) list. The number IR punched as Index 1 indicates that the first integer data word on the card is the library set number for region IR; the next word, if any, is the set number for region IR+1, etc. These library set numbers are used during the execution of a problem to pick up the appropriate library set data from logical unit 8.
0	8	IM (macro material)	0	MY(IM)	The number of the fission yield set to be used for macro material IM. The fission yield set number n_m [MY(IM); IM=m] specifies which of the three sets to use for macro material m = IM in computing the production of fission product nuclides by fission in macro material m. See Equation (163). The first integer input word on the card is the n_m value for macro material m = IM, as indicated by the value of IM punched as Index 3; the next word is n_{m+1} , etc. The fission yield data used are those read in for the library set assigned to the region.
0	10	K (list word number)	0	IREDT(K)	List of regions for which the library set cross section and concentration edits under control of $1 \leq IOUT2 \leq 3$ are to be made. Each word in the list is a region number IR.

2.3 The Time Range Floating Point Input Data

The Fortran format for a floating point data card is:

The first 12 columns of a floating point data card are devoted to the identification of the data on the card.

Column	1	IEND	Zero or blank means other floating point data cards follow. 1 punched means this is the last floating point data card.
Columns	2-3	INDEX 1	The values of these four indices identify the floating point data punched on the card; hence, the card order is irrelevant.
Columns	4-5	INDEX 2	
Columns	6-8	INDEX 3	
Columns	9-10	INDEX 4	
Columns	11-12	IWC	This word tells the number of floating point words punched on the card (1 to 5). If IWC is zero or blank, a value of 5 is assumed.
Columns	13-72		One to 5 floating point words of format E12.8 can be punched in these columns.

There must be at least one floating point data card per time range.

The value of IWC specifies the number of successive words of floating point data given on the card.

The floating point data cards can be in any order so long as the last card has a 1 punched in card column 1. The floating point data deck must follow the integer input data deck of the time range.

The single word input data (starting with EPS1 and ending with VMAX3) can be punched in the order of increasing Index 3 values across a card (provided the Index 3 value in card columns 6-8 corresponds to the data word in the first floating point data field on the card).

Table III. Detailed Description of Time Range Floating Point Input Data

Index 1 Card Cols. 2-3 Format I2	Index 2 Card Cols. 4-5 Format I2	Index 3 Card Cols. 6-8 Format I3	Index 4 Card Cols. 9-10 Format I2	Mnemonic Name of Data	Notes on Application of Data
0	0	1	0	EPS1	Desired fission neutron source convergence level EPS1 is relevant only if fission neutrons are produced. If EPS1 = 0 or blank, it is automatically set to .0001. Define: $\epsilon_1 \equiv \text{EPS1}$, $\lambda^t \equiv$ fission neutrons produced per fission neutron introduced on the t th outer iteration, $f_i^{*t} \equiv$ fission neutron source density produced (before any extrapolation) at mesh point i on the t th outer iteration, and f_i^t as the value after extrapolation, if any. The fission source convergence criterion under the various possible circumstances is as follows. 1. $\text{NCONVG} = 0$ (I-0,0,48,0): $ (\lambda^t - \lambda^{t-1}) /\lambda^t \leq \epsilon_1$ 2a. $\text{NCONVG} = 1$ and $\text{IREN} = 0$: $\max \frac{f_i^{*t}}{f_i^{t-1}} - \min \frac{f_i^{*t}}{f_i^{t-1}} \leq \epsilon_1$ (I-0,0,6,0) 2b. $\text{NCONVG} = 1$ and $\text{IREN} = 1$: $\max \left \frac{f_i^{*t}}{f_i^{t-1}} - 1 \right \leq \epsilon_1$
0	0	2	0	EPS2	Desired generation factor convergence level for searches If EPS2 = 0 or blank, it is automatically set to .0001. Define: $\lambda^s \equiv$ the total fission neutrons produced per fission neutron introduced as obtained on the final outer iteration for the value of the search parameter (buckling, poisoning, etc.) used on the s th search trial, and $\lambda_V^s \equiv$ the total fission neutrons produced per fixed volume source neutron introduced, as obtained on the first outer iteration for the value of the search parameter used on the s th search trial; and let $\lambda_d \equiv \text{EIGEND}$ (F-0,0,14,0) be the desired search eigenvalue. The search generation factor convergence criterion is dependent on IGENOP (I-0,0,8,0) as follows. $\text{IGENOP} = 0$: $ (\lambda^s - \lambda_d)/\lambda_d \leq \text{EPS2}$. $\text{IGENOP} = 1$: $ (\lambda_V^s - \lambda_d)/\lambda_d \leq \text{EPS2}$. See ISERCH (I-0,0,61,0).
0	0	3	0	EPS3	Desired convergence level for the search parameter value EPS3 = 0 or blank causes it to be set automatically to .0001. This convergence criterion is ignored unless ISERCH (I-0,0,61,0) is a negative number. Let x^s be the value of the search parameter on the s th search trial. If ISERCH is negative, the convergence criterion that must be satisfied on the search parameter is $ (x^s - x^{s-1})/x^s \leq \text{EPS3}$.
0	0	4	0	EPS4	Desired convergence level for the pointwise power distribution EPS4 = 0 or blank causes it to be set automatically to .0001. The feedback loop iterations described in Section II-1.3.1 repeatedly compute the fluxes, power distribution q^u , and feedback vectors until the maximum value of $ q^u(k) - q^{u-1}(k) /q^u(k) < \epsilon_4$, k is the source point index, and u is feedback iteration number.
0	0	5	0	EPS5	Desired convergence level for the power residual (on inner feedback iterations) EPS5 = 0 or blank causes it to be set automatically to .0001. During the feedback iterations an inner iterative loop converging the power distribution and the feedback vectors is repeated until $\sum_k q^u(k) - q^{u-1}(k) / \sum_k q^u(k) \leq \epsilon_5$ where $\epsilon_5 = \text{EPS5}$ and $q^u(k)$ and $q^{u-1}(k)$ are new and old power values, respectively, at source point k .
0	0	6	0	RAD1	The position of mesh point number 1, the lower boundary The variable x is the one-dimensional space variable of the problem. The range of x to which the problem pertains extends from x_1 to x_1 , where x_1 is the lower bound and x_1 the upper bound. The value of RAD1 defines x_1 , i.e., $x_1 \equiv \text{RAD1}$. Because of the flexibility of the problem boundary conditions, RAD1 need not be zero for cylindrical or spherical problems. Note: IRMAX (I-0,0,2,0) is the number of regions; hence, $x_1 = x_1 + \sum_{\text{IR}=1}^{\text{IRMAX}} \text{RW}(\text{IR}) \text{ (F-0,1,IR,0)}, \text{ and } I = 1 + \sum_{\text{IR}=1}^{\text{IRMAX}} \text{NI}(\text{IR})$ (I-0,2,IR,0).
0 0	0 0	7 8	0 0	THETA	Ignored The value of the two-point extrapolation factor for acceleration of the fission neutron source convergence A two-point extrapolation formula is used to accelerate the convergence of the fission neutron source distribution if the extrapolation flag NEXT0 (I-0,0,50,0) is set to 1. Define: $\theta \equiv \text{THETA}$ and $f_k^t \equiv$ the extrapolated fission neutron source density at source point k following the t th outer iteration. The two-point extrapolation formula is $f_k^t = f_k^{*t} + \theta(f_k^{*t} - f_k^{t-1})$, where the $*$ indicates an unextrapolated value. (The source point k in the IRth region at mesh point i is given by $k = i + \text{IR} - 1$.)

Table III (Continued)

Index 1 Card Cols. 2-3 Format 12	Index 2 Card Cols. 4-5 Format 12	Index 3 Card Cols. 6-8 Format 13	Index 4 Card Cols. 9-10 Format 12	Mnemonic Name of Data	Notes on Application of Data
0	0	9	0	FISSOR	The value F_0 to which the total volume integral of the fission neutron source is to be set at the beginning of each source iteration. By setting $IREN=0$ (1-0,0,6,0), it is indicated that the volume integral of a fission neutron source is to be always "normalized" to equal F_0 , satisfying the constraint defined by Equations (117) and (118). This normalizing is done to the source guess before the first source iteration, and thereafter it is done to the newly computed fission neutron source at the end of each source iteration. (This normalization procedure is usually desired for all but some inhomogeneous or noneigenvalue types of problems). When $IREN=0$, and FISSOR is 0 or blank, the code sets $FISSOR = F_0 = 1$ except when F_0 is to be the search parameter in a fission source search problem. In the latter case when $ISERCH = \pm 4$ (1-0,0,60,0); a detected value of FISSOR of either 0 or 1 causes F_0 to be set to twice the volume integral of the fixed volume source, as a first guess for F_0 . Subsequently, $FISSOR = F_0$ is changed in the course of the search. See Section II-3.3. When $IREN=1$, the volume integral of the fission neutron source is set equal to FISSOR at the beginning of only the first source iteration. However, if FISSOR=0 then the source is not normalized.
0	0	10	0	FLMULT	Optional multiplier of all flux, source, and power output data. When the fixed point data word NORMAL (1-0,0,36,0) is set equal to 4, all neutron group fluxes, fission neutron sources, power distributions, and all other output quantities such as volume integrals of these quantities are multiplied by the value of FLMULT prior to any output edits.
0	0	11	0	POWER	The total volume integral of the fission power for use in "normalizing" the output fluxes, source, and fission power. When the fixed point data word NORMAL (1-0,0,36,0) is set equal to 1, an unnormalized power distribution p' is first computed from the unnormalized fluxes as $p' = \frac{1}{\lambda t} \sum_j E_j \bar{x}_{fj} \phi_j'$, where the primes denote the unnormalized values. Then the normalizing factor C is computed from $C = P / \int_V dv p'$, where $P \equiv \text{POWER}$. The output neutron source, power, and fluxes are then CF^{*t+1} , Cp' , and $C\phi_j'$, respectively.
0	0	12	0	WMT	The minimum region width allowed before terminating a boundary search problem. If WMT is 0 or blank, it is set automatically to .001. When a boundary search is specified (that is, $ISERCH = \pm 5$ and the x position of the boundary between two adjacent regions is to be varied until the search criteria are satisfied), the search is considered to have failed if the width of either of the regions must be less than the value of WMT in order to satisfy the search criteria. The first time the search attempts to use a region width less than WMT, the region width is set, instead, to the value of WMT and the search is continued. However, the second time the search attempts to use a region width smaller than WMT, the search is terminated and the problem solution (as edited) pertains to the last set of region widths used just before the second encounter of the illegal region width.
0 0	0 0	13 14	0 0	EIGEND	Ignored. The desired value of the generation factor or eigenvalue which is to be attained by a search problem. EIGEND is set automatically to unity if either 1) EIGEND is 0 or blank and $NSERCH=0$ (1-0,0,63,0), or 2) $ISERCH = \pm 4$ (1-0,0,61,0). EIGEND is set automatically to the eigenvalue λ of the preceding problem if $NSERCH=1$ and $ISERCH \neq \pm 4$. In all other cases, the value of EIGEND is the input value. A search computes and uses different values of the search parameter until the eigenvalue or generation factor corresponding to the search parameter is within a specified fraction EPS2 (F-0,0,2,0) of being equal to EIGEND.
0	0	15	0	EPFISS	Optional energy per fission value. The energy per fission value is $E_j^m \equiv \text{ENERGY}(J,IM)$ for group $j=J$ of macro material number $m \equiv IM$. This energy per fission value for every group of a macro material $m \equiv IM$ can be set equal to EPFISS by setting the word IENPF (1-0,0,64,0) equal to 0 or blank.
0	0	16	0	TSTART	Starting time T_0 for the present time range. The user can set the initial time of each time range independently. The program will solve Equation (160) for new nuclide concentrations at each time point $T_n = T_0 + n\Delta t$ (1 $\leq n$) until $ T_e - T_n \leq \Delta t/2$, where $T_e \equiv \text{TIMEND}$ and $\Delta t \equiv \text{DELTIM}$, below.
0	0	17	0	TIMEND	Ending time T_e for the present time range. See TSTART, above.
0	0	18	0	DELTIM	The time step size Δt to be used in the present time range. See TSTART, above.

Table III (Continued)

Index 1 Card Cols. 2-3 Format I2	Index 2 Card Cols. 4-5 Format I2	Index 3 Card Cols. 6-8 Format I3	Index 4 Card Cols. 9-10 Format I2	Mnemonic Name of Data	Notes on Application of Data
0	0	19	0	RNDOFF	Optional round-off error bound ϵ_r . Round-off error in the solution of the Bateman equations for the new nuclide concentrations at a time point is detected by comparing the difference between various quantities to ϵ_r . See the SINBAD report, pages 23 and 72. If RNDOFF is zero or left blank, it is set to 10^{-13} in the IBM 360, double precision version of DWARF.
0	0	20	0	AVPOWR	The total power P_{av} used to normalize the fluxes for the feedback vector and nuclide concentration calculations. A factor C_p is found that multiplies the fluxes such that the volume integral (one-dimensional form) of q^u is equal to P_{av} . See Equations (18), (63), (64), and (65).
0	0	21	0	HINLET	The inlet enthalpy of the coolant (usually Btu/lb). The enthalpy of the coolant H_i at a mesh point i is computed using Equation (67) and the input value of H_0 for mesh point 1. The first feedback vector $\rho_1(i)$ is a function of H_i .
0	0	22	0	HFSAT	The enthalpy H_{fsat} of the saturated coolant fluid. The saturated fluid enthalpy is used in testing for the subcooled state and in Equations (74) to (83).
0	0	23	0	HFG	The enthalpy rise H_{fg} of the coolant in going from saturated fluid to vapor. The coolant quality is computed using H_{fg} in Equation (75).
0	0	24	0	HCONV	The energy conversion factor K_1 . The factor K_1 is used in Equation (67) in converting the volume integral (one-dimensional form) of q^u , the power density, to enthalpy unit. (The factor is usually 9.48×10^{-4} Btu/w-sec.)
0	0	25	0	FLOW	The mass flow rate of the coolant w_c . The mass flow rate w_c is used in Equation (67) for the coolant enthalpy. See the footnote on units in Section II-1.3.1.
0	0	26	0	GASRHO	The saturated vapor density ρ_g of the coolant. The density ρ_g is used in Equations (76) to (83) in computing the coolant density $\rho_1(i)$ (feedback vector 1) for the two-phase and the superheated vapor states.
0 0	0 0	27 28	0 0	PRESSR AREA	The system coolant pressure (not used in the present DWARF version). The surface area of the fuel cladding material per unit volume of core A_c . The area A_c is usually given in units of ft^2/cm^3 and it is used in Equation (71) to compute the surface temperature of the cladding in the subcooled coolant region.
0	0	29	0	HFILM	The heat transfer film coefficient h_c . The film coefficient h_c (usually in $\text{Btu}/\text{ft}^2\text{-sec-}^\circ\text{F}$) is used in Equation (71) with A_c to compute the temperature of the fuel cladding surface.
0	0	30	0	CRITV	The "critical" vapor fraction. The critical vapor fraction α_t is used (if the user supplies a non-zero value) together with a critical enthalpy fraction H_t to identify a special transition region from the subcooled nucleate boiling region to the bulk boiling region in the coolant. See Equations (84) to (86).
0	0	31	0	HFRACT	The "critical" enthalpy fraction. See CRITV, above.
0	0	32	0	HCON	The density formula fitting parameter H_c . The arbitrary parameter H_c (enthalpy units) is used in the formula for the superheated coolant vapor density to help the function of enthalpy fit the true curve. See Equation (83).
0	0	33	0		Ignored.

Table III (Continued)

Index 1 Card Cols. 2-3 Format 12	Index 2 Card Cols. 4-5 Format 12	Index 3 Card Cols. 6-8 Format 13	Index 4 Card Cols. 9-10 Format 12	Mnemonic Name of Data	Notes on Application of Data
0	0	34	0	RHOGES	Optional input value for feedback vector 1, ρ_1 . If the integer input flag IRHO (1-0,0,58,0) is set to 2, the program sets $\rho_1(i)$ equal to RHOGES at each mesh point at the start of the time range.
0	0	35	0	PHOGES	Optional input value for feedback vector 2, ρ_2 . If the integer input flag IPHO (1-0,0,59,0) is set to 2, the program sets $\rho_2(k)$ equal to PHOGES at each mesh point at the start of the time range.
0	0	36	0	SHOGES	Optional input value for feedback vector 3, ρ_3 . If the integer input flag ISHO (1-0,0,60,0) is set to 2, the program sets $\rho_3(k)$ equal to SHOGES at each mesh point at the start of the time range.
0	0	37	0	HFUEL	The effective heat transfer coefficient (fuel to coolant), h_f . The effective heat transfer coefficient h_f is used in Equation (90) to compute an effective (average) fuel temperature, given the coolant temperature and the power $q''(k)$ at a point.
0	0	38	0	RHOEXT	Ignored
0	0	39	0		Ignored
0	0	40	0		Extrapolation factor for feedback vector 1, ρ_1 . The point values of ρ_1 computed on each feedback iteration can be interpolated or extrapolated using RHOEXT as the factor in Equation (95) to retard or accelerate, respectively, the change in successive values.
0	0	41	0	PHOEXT	Extrapolation factor for feedback vector 2, ρ_2 . The point values of ρ_2 computed on each feedback iteration can be interpolated or extrapolated using PHOEXT as the factor in Equation (96) to retard or accelerate, respectively, the change in successive values.
0	0	42	0	SHOEXT	Not used in the present DWARF version
0	0	43	0	VMAX1	Ignored
0	0	44	0		Ignored
0	0	45	0		Optional maximum value for plot vector 1. By reading in a nonzero value, the user can cause VMAX1 to be used as the maximum value of plot vector 1 instead of the actual value in determining the ordinate scaling of plotted output. See IPLIT (1-0,0,41,0).
0	0	46	0	VMAX2	Optional maximum value for plot vector 2. A nonzero value read for VMAX2 replaces the actual maximum of plot vector 2 in plotted output. See VMAX1, above.
0	0	47	0	VMAX3	Optional maximum value for plot vector 3. A nonzero value read for VMAX3 replaces the actual maximum of plot vector 3 in plotted output. See VMAX1, above.
0	1	IR (region number)	0	RW(IR)	The region width of region number IR. The number IR punched as Index 3 indicates that the first data word on the card is the width (the range of the space variable x) of the IR th region, the next word pertains to the IR+1 region, etc. Note: On a boundary search problem, the region widths of the two searched regions are altered and the final RW(IR) values obtained by the search are retained for use in a successive problem. See ISERCH = ± 5 (1-0,0,61,0).
0	2	K (macro. material, or region number)	0	BSQR(K)	Optional specification of buckling values for macroscopic material IM = K or region IR = K when the bucklings of a problem do not vary by energy group. The first data word on the card is a group independent buckling squared value for the macroscopic material number or region number K given as Index 3; the next data word pertains to material K+1, etc. The integer input word IBSQO (1-0,0,12,0) determines whether K is a material or region number. These buckling squared data, as read in, remain available from problem to problem regardless of buckling searches, changes in the other buckling data BSQM(J,IM) (F-6,IM,J,0), etc. If INBUCK = 1 (1-0,0,11,0), the buckling squared values of material IM are set to BSQR(IM) in all energy groups. (For all other values of INBUCK, the BSQM(J,IM) data are used instead.)

Table III (Continued)

Index 1 Card Cols. 2-3	Index 2 Card Cols. 4-5	Index 3 Card Cols. 6-8	Index 4 Card Cols. 9-10	Mnemonic Name of Data	Notes on Application of Data
Format I2	Format I2	Format I3	Format I2		
0	3	IR (region number)	0	FR(IR)	Specification of the values for a region dependent (but flat in each region) neutron fission source guess. The first data word on the card is the value of the fission neutron source density guess to be used in the region number IR given as Index 3, the next word is the value for the region IR+1, etc. A value FR(IR) is used as the source density f^1 (neutrons per unit volume and unit time) applied throughout region IR in the first outer iteration of a problem. These values are not used as the fission source guess unless INSORS = ± 2 (I-0,0,46,0). Also, if INSORS = 0, the FR(IR) values are automatically set to 0 in nonfissioning regions and 1.0 in fissioning regions.
0	4	IM (macro. material number)	0	SIGPT(IM)	Specification of the poison cross section for macroscopic material number IM. The first data word on the card is the value of the "poison" cross section to be used in the macroscopic material number IM specified by the value of Index 3, the next data word pertains to material IM+1, etc. A given value of SIGPT(IM) is used in conjunction with the T(J,IM) (F-7,IM,J,0) data to compute a macroscopic "poison" absorption cross section for each group of macroscopic material IM. This "poison" cross section is treated simply as an addition to the macroscopic absorption cross section of the material; and for group $j \equiv J$ and material $m \equiv IM$, this addition is computed as $T_{p,m}^m = T(J,IM) \text{ SIGPT}(IM)$. However, unlike the ordinary absorption cross section, this poisoning absorption can cause automatic modification of the diffusion coefficient during poison searches as discussed under IDCO (I-0,0,14,0), when IDCO equals 1 or 2. See Section II-1.2.
0	5	1	J (group number)	OMEGA1(J)	Specification of the optional boundary condition parameter ω_{1j} for the inner boundary. The first data word is ω_{1j} for group $j \equiv J$ given as Index 4, the second data word is the value for group $j \equiv J+1$, etc. If the inner boundary condition indicator NIBDRY (I-0,0,4,0) is 2, 3, or 4, the value of $\omega_{1j} \equiv \text{OMEGA1}(J)$ is used in defining the boundary condition for the $j \equiv J$ energy group. See the discussion under NIBDRY.
0	5	2	J (group number)	OMEGA2(J)	Specification of the optional boundary condition parameter ω_{2j} for the outer boundary. The first data word on the card is ω_{2j} for group $j \equiv J$ given as Index 4, the second word is for group $j \equiv J+1$, etc. If the outer boundary condition indicator NOBDRY (I-0,0,5,0) is 2, 3, or 4, the value of $\omega_{2j} \equiv \text{OMEGA2}(J)$ is used in defining the boundary condition for the $j \equiv J$ th energy group. See the discussion under NOBDRY.
0	5	3	J (group number)	ALPHA1(J)	Optional specification of the inner boundary condition parameter, α_{1j} , the neutron flux multiplier for group $j \equiv J$.
0	5	4	J (group number)	BETA1(J)	Optional specification of the inner boundary condition parameter, β_{1j} , the neutron current multiplier for group $j \equiv J$.
0	5	5	J (group number)	GAMMA1(J)	Optional specification of the inner boundary condition constant, γ_{1j} for group $j \equiv J$. The first data word on one of these cards gives the value for group $j \equiv J$, the second word pertains to group $J+1$, etc. Except when NIBDRY = 7, the values of α_{1j} , β_{1j} , and γ_{1j} are computed automatically; and they replace any values input on cards or used on a previous problem. When NIBDRY = 7 (I-0,0,4,0), the values of α_{1j} , β_{1j} , and γ_{1j} are read from cards. (In this case, if no values are read in, the values from the preceding problem are used.) Except for NIBDRY = 8 (the periodic boundary condition), the inner boundary condition is defined by the equation $\alpha_{1j}\phi_{j1} - \beta_{1j}J_{j1} + \gamma_{1j} = 0$, where ϕ_{j1} and J_{j1} are the flux and the net neutron current in group j at the inner boundary. For example, a "zero current" boundary condition for group j is specified if $\alpha_{1j} = 0$, $\beta_{1j} = 1$, and $\gamma_{1j} = 0$. This makes the boundary condition equation read, $J_j(x_1) = 0$.
0	5	6	J (group number)	ALPHA2(J)	Optional specification of the outer boundary condition parameter, α_{2j} , the neutron flux multiplier for group $j \equiv J$.
0	5	7	J (group number)	BETA2(J)	Optional specification of the outer boundary condition parameter, β_{2j} , the neutron current multiplier for group $j \equiv J$.
0	5	8	J (group number)	GAMMA2(J)	Optional specification of the outer boundary condition constant γ_{2j} for group $j \equiv J$. The first data word on one of these cards gives the value for group $j \equiv J$, the second word pertains to group $J+1$, etc. Except when NOBDRY = 7 (I-0,0,5,0), the values of α_{2j} , β_{2j} , and γ_{2j} are computed automatically; and they replace any values input on cards or used on a previous problem. When NOBDRY = 7, the values of α_{2j} , β_{2j} , and γ_{2j} are read from cards. (In this case, if no values are read in, the values from the preceding problem are used.) Except for NOBDRY = 8 (the periodic boundary condition) the outer boundary condition is defined by the equation $\alpha_{2j}\phi_{j1} - \beta_{2j}J_{j1} + \gamma_{2j} = 0$, where ϕ_{j1} and J_{j1} are the flux and the net neutron current in group j at the outer boundary. For example, a boundary condition specifying an inwardly directed net neutron current equal to, say, 1.8 neutrons per unit area and unit time in group j at the outer boundary is attained if $\alpha_{2j} = 0$, $\beta_{2j} = 1$, and $\gamma_{2j} = -1.8$. This makes the boundary condition equation read, $J_j(x_1) = -1.8$.

Table III (Continued)

Index 1 Card Cols. 2-3 Format I2	Index 2 Card Cols. 4-5 Format I2	Index 3 Card Cols. 6-8 Format I3	Index 4 Card Cols. 9-10 Format I2	Mnemonic Name of Data	Notes on Application of Data
0	6	1	N (index number)	RHOLCS(N)	Polynomial coefficients for computing the coolant density (for the unsaturated fluid) The first data word on the card is the coefficient $a_p(n) = \text{RHOLCS}(N)$ ($N=n$), the next word is $a_p(n+1)$, etc. The density of the coolant liquid is computed with the enthalpy polynomial $P_p(H_i)$ of Equation (72) using the $a_p(n)$ coefficients. The density is then used for obtaining the first feedback vector ρ_1 . ($1 \leq N \leq 6$)
0	6	2	N (index number)	TEMPCS(N)	Polynomial coefficients for computing the coolant temperature (of the unsaturated fluid) The first data word on the card is the coefficient $a_t(n) = \text{TEMPCS}(N)$ ($N=n$), the next word is $a_t(n+1)$, etc. The temperature of the coolant liquid is computed with the enthalpy polynomial $P_t(H_i)$ of Equation (73) using the $a_t(n)$ coefficients. ($1 \leq N \leq 6$)
0	6	3	N (index number)	SLIPCS(N)	Optional coefficients, $1 \leq N \leq 6$ (not used in the present DWARF version)
0	7	I (mesh point number)	0	RHO(I)	Optional specification of the pointwise distribution for $\rho_1(i)$, feedback vector 1 at beginning of time range If $\text{IRHO} = 3$ (I-0,0,58,0), any data read here are used to set $\rho_1(i) = \text{RHO}(I)$ ($I=i$) at the start of the time range.
0	8	K (source point number)	0	RHO(K)	The first data word on the card is $\text{RHO}(I)$, the next is $\text{RHO}(I+1)$, etc. Optional specification of the pointwise distribution for $\rho_2(k)$, feedback vector 2 If $\text{IRHO} = 3$ (I-0,0,59,0); any data read here are used to set $\rho_2(k) = \text{PHO}(K)$ ($K=k$) at the start of the time range.
0	9	K (source point number)	0	SHO(K)	The first data word on the card is $\text{PHO}(K)$, the next is $\text{PHO}(K+1)$, etc. Optional specification of the pointwise distribution for $\rho_3(k)$, feedback vector 3 (not used in the present DWARF version)
0	10	K (source point number)	0	F(K)	Optional specification of the guess for the pointwise fission neutron source distribution to be used on the first outer iteration The first data word on the card is the value of the fission neutron source density (neutrons per unit volume and time) at the Kth source point, the next data word pertains to the source point number $K+1$, etc. A "source point" K is related to a mesh point I in region number IR by the equality, $K = I + \text{IR} - 1$. If F(K) is input on cards, the value of INSORS (I-0,0,46,0) must be ± 3 (± 1 also works); and the cards must be read with the problem in which the F(K) values for the source guess are to be used. (This is because the F(K) values are changed on each outer iteration.) Note: This F(K) fission neutron source distribution is usually normalized to a volume integral total equal to FISSOR (F-0,0,9,0). See the discussion for FISSOR and IREN (I-0,0,6,0). The fission neutron source density values obtained in a given problem can be punched out in this card format for input here by specifying a value of IPUNCH (I-0,0,32,0) equal to 2. The fission source data deck follows the punched flux deck in this case.
0	12	I (mesh point number)	J (energy group number)	PHI(I,J)	Optional specification of the pointwise flux guess for each group to be used on the first outer iteration The first data word on the card is the value of the neutron scalar flux $\phi_{ij} = \text{PHI}(I,J)$ in energy group $j = J$ at mesh point $i = I$, the next word is the flux at mesh point $I+1$, etc. These input fluxes can be used to compute automatically a fixed volume source by various means. See fixed point data word ISORS (I-0,0,15,0).

Table III (Continued)

Index 1 Card Cols. 2-3 Format I2	Index 2 Card Cols. 4-5 Format I2	Index 3 Card Cols. 6-8 Format I3	Index 4 Card Cols. 9-10 Format I2	Mnemonic Name of Data	Notes on Application of Data
1	IM (f. macro material number)	J (energy/ group number)	0	DC(J,IM), SIGA(J,IM), SIGNU(J,IM), SIGFIS(J,IM), ENERGY(J,IM)	The values of the diffusion coefficient D_j , absorption cross section Σ_{aj} , neutrons per fission times the fission cross section $\nu\Sigma_{fj}$, the fission cross section Σ_{fj} , and the energy per fission E_j for energy group $j \equiv J$ of macro material number $m \equiv IM \leq 40$. The first data word on the card is DC(J,IM), the next is SIGA(J,IM) etc., in the order given in the list at left under the Mnemonic Name of Data.
1	IM (f. macro material number)	J (energy/ group number)	1	DC(J,IM)	Optional specification of diffusion coefficient data D_j for group $j \equiv J$ of f. macro material $m \equiv IM \leq 40$. An Index 4 value of 1 indicates that the first data word on the card is the value of DC(J,IM) for group J, the next word is the value for group J+1, etc. The DC(J,IM) values can be specified by the cards of either this format or the format corresponding to (F-1,IM,J,0) above. Cards of both types can be used in the same deck for a given f. macro material. If two values of DC(J,IM) are read (for the same energy group J) the value on the last card read is used. See Section II-1.2 for specifying information on how the DC(J,IM) data are used.
1	IM (f. macro material number)	J (energy/ group number)	2	SIGA(J,IM)	Optional specification of the absorption cross section Σ_{aj} for group $j \equiv J$ of f. macro material $m \equiv IM \leq 40$. With this format (with an Index 4 value of 2) the first data word on the card is the value of Σ_{aj} for group number $j \equiv J$, the next word is the value for group J+1, etc. The SIGA(J,IM) can be specified by cards of either the format with Index 4=0, above, or by this format with Index 4=2. Cards of both types can be used in the same deck of a given f. macro material. The last card(s) read determine the values used if the same data are read more than once.
1	IM (f. macro material number)	J (energy/ group number)	3	SIGNU(J,IM)	Optional specification of neutrons per fission times the fission cross section $\nu\Sigma_{fj}$ for group $j \equiv J$ of f. macro material $m \equiv IM \leq 40$. The first data word on the card is the value of $\nu\Sigma_{fj}$ for group number $j \equiv J$, the next word is the value for group J+1, etc. The SIGNU(J,IM) can be specified by cards of either the format with Index 4=0, above, or by this format with Index 4=3. Cards of both types can be used in the same deck of a given material. The last card(s) read determine the values used if the same data are read more than once.
1	IM (f. macro material number)	J (energy/ group number)	4	SIGFIS(J,IM)	Optional specification of fission cross section Σ_{fj} for group $j \equiv J$ of f. macro material $m \equiv IM \leq 40$. The first data word on the card is the value of Σ_{fj} for group number $j \equiv J$, the next word is the value for group J+1, etc. The SIGFIS(J,IM) can be specified by cards of either the format with Index 4=0, above, or by this format with Index 4=4. Cards of both types can be used in the same deck of a given material. The last card(s) read determine the values used if the same data are read more than once.
1	IM (f. macro material number)	J (energy/ group number)	5	ENERGY(J,IM)	Optional specification of the energy per fission E_j for group $j \equiv J$ of f. macro material $m \equiv IM \leq 40$. The first data word on the card is the value of E_j for group number $j \equiv J$, the next word is the value for group J+1, etc. The ENERGY(J,IM) can be specified by cards of either the format with Index 4=0, above, or by this format with Index 4=5. Cards of both types can be used in the same deck of a given material. The last card(s) read determine the values used if the same data are read more than once.
2	IM (f. macro material number)	J (energy/ group number)	0	SIGSD(J,IM)	Set the discussion for the integer input data word IENPF (I-0,0,64,0) for optional ways of setting $E_j \equiv ENERGY(J,IM)$. See also, data word EPFISS (F-0,0,15,0). Specification of the scattering cross sections $\Sigma_{sj} \equiv SIGSD(J,IM)$ for the scattering transfer of neutrons from energy group $j \equiv J$ to group $j+1$ for f. macro material $m \equiv IM \leq 40$. The first data word on the card is the value of the cross section Σ_{sj} for scattering neutrons from energy group $j \equiv J$ to group $j+1$, the next word on the card is the cross section for scattering from group $j+1$ to group $j+2$, etc.
4	IM (f. macro material number)	J (energy/ group number)	0	SIGMU(J,IM)	Specification of the cross section $\Sigma_{\mu j}$ for energy group $j \equiv J$ and f. macro material number $m \equiv IM \leq 40$. The first data word on the card is the value of $\Sigma_{\mu j} \equiv SIGMU(J,IM)$ for energy group $j \equiv J$, the next data word is the value for group $j+1$, etc. See Section II-1.2 for specific information on how these data are used.
5	IM (f. macro material number)	J (energy/ group number)	0	CHIM(J,IM)	The fraction x_j^m of the fission produced source neutrons that are released in energy group $j \equiv J$ of macroscopic material $m \equiv IM$. The $x_j^m \equiv CHIM(J,IM)$ value is the fraction of the total fission neutrons produced that are released in the $j \equiv J$ th energy group of macroscopic material $m \equiv IM$. Hence, if the total fission neutron source density is f , then the product $x_j^m f$ is the fission source to group j , S_{fj} , as defined in Equation (99). The first data word on the card is the value of x_j^m for the $j \equiv J$ th energy group, the next word is the value for group J+1, etc.

Table III (Continued)

Index 1 Card Cols. 2-3	Index 2 Card Cols. 4-5	Index 3 Card Cols. 6-8	Index 4 Card Cols. 9-10	Mnemonic Name of Data	Notes on Application of Data
Format I2	Format I2	Format I3	Format I2		
6	IM (macro. material number)	J (energy group number)	0	BSQM(J,IM)	<p>The buckling squared values for energy group J of macroscopic material number IM</p> <p>The first data word on the card is the value of the buckling squared (B^2) for group number J, the next word is the value for the J+1 group, etc.</p> <p>For fixed macro materials, the buckling squared values B^2 are set by the BSQM(J,IM) data on these cards if INBUCK = 0 (or blank), 2, or 3 (1-0,0,11,0). For these values of INBUCK, the B^2 values are 1) group and material dependent, 2) only group dependent, or 3) neither group nor material dependent, respectively. Otherwise, (for INBUCK = 1), the B^2 values are set according to the BSQR(IM) (F-0,2,IM,0) data when only material dependence is desired. For variable macro materials, the B^2 values are functions of the BCOF and COFMOD arrays when INBUCK ≤ 1, otherwise the B^2 data are set as for f. macro materials.</p> <p>The multiplier T_j^m of the macroscopic poison cross section $\lambda_p^m = \text{SIGPT}(IM)$ for energy group number $j \equiv J$ of macroscopic material number $m \equiv IM$</p> <p>The first data word on the card is the value of the fraction T_j^m of λ_p^m to be applied in group $j \equiv J$ of the macroscopic material $m \equiv IM$, the next word is the fraction for the group J+1, etc.</p> <p>See the floating point data SIGPT(IM) (F-0,4,IM,0) and Section II-1.2 for discussions of how the T(J,IM) and SIGPT(IM) data are applied.</p> <p>Specification of the logarithmic boundary condition parameter q_j^m to be applied in energy group $j \equiv J$ at both boundaries of any region in which fixed macroscopic material $m \equiv IM$ appears</p> <p>The first data word on a card is the value q_j^m, the logarithmic boundary condition parameter, for the $j \equiv J$th energy group, the next word is the value for group J+1, etc.</p> <p>If $q_j^m \equiv Q(J,IM) \neq 0$, then in any region in which the macroscopic material IM appears, a nonzero value of q_j^m results in the application of the boundary conditions</p> $\frac{1}{\phi_{j1b}} \frac{d\phi_{j1b}}{dx} = -q_j^m \text{ at the left (lower) boundary of the region and}$ $\frac{1}{\phi_{j1b}} \frac{d\phi_{j1b}}{dx} = +q_j^m \text{ at the right (upper) boundary of the region.}$ <p>This parameter is most usually applied in representing control rods or other highly absorbing regions. As can be noted, a positive value of q_j^m causes neutrons to leak into the material m in group j; a negative q_j^m causes homogeneous boundary sources (leakage out of the material) into the adjacent regions. The net neutron currents at the region surfaces are $J(x) = D_j q_j^m \phi_j(x)$ at the left boundary and $J(x) = -D_j q_j^m \phi_j(x)$ at the right boundary, where the D_j values are those for the respective adjacent regions.</p> <p>It is permissible to specify nonzero $q_j^m \equiv Q(J,IM)$ values in adjacent regions, but if q_j^m is nonzero in the first or the last regions, certain limitations must be observed. Specifically, it is not permissible to specify a nonzero Q(J,IM) value for materials appearing in either region number 1 or the last region when a periodic boundary condition is specified. See Section II-1.6, also. Also, it is nonpermissible to have a nonzero Q(J,IM) in a region with a v. macro material (with IM > 40).</p> <p>Note: To avoid ambiguity concerning any intended meaning, a nonzero value of q_j^m for group j and f. macro material m causes all the neutron group constants (D_j, χ_j, $\nu \Sigma f_j$, Σ_{aj}, Σ_{ujtj}, Σ_{pj}, $\Sigma_{sj}^{1 \rightarrow j}$, and $\Sigma_{sj}^{j \rightarrow 1}$) associated with the Jth group of that material to be set to zero.</p>
7	IM (macro. material number)	J (energy group number)	0	T(J,IM)	
8	IM (f. macro material number)	J (energy group number)	0	Q(J,IM)	
9	IR (region number)	J (energy group number)	0	SEOR(J,IR)	<p>The value of the flat fixed volume neutron source density S_{rj} in energy group J for the IRth region</p> <p>The first data word on the card is the value of the flat fixed volume neutron source density (neutrons per unit volume and time), S_{rj}, in the IRth region in energy group $j \equiv J$, the next word is the value for group J+1, etc.</p> <p>These input data facilitate specification of a fixed volume source in any region in which the fixed volume source density is not a varying function of the space variable.</p> <p>The fixed volume source density, S_{ej}, applied in a problem is always the sum of the SEOR(J,IR) values and the pointwise variable fixed sources, S_{pj}, specified by SEO(K,J), below.</p>
10	J (energy group number)	K (source point number)	0	SEO(K,J)	<p>The pointwise (space dependent) fixed volume neutron source density S_{pj} for energy group number J at source point K</p> <p>The first data word on the card is the value of the fixed neutron source density S_{pj} at the source point number K for group number j, the next word is the source density value at source point number K+1, etc.</p> <p>These input data facilitate specification of a pointwise variable fixed volume neutron source where it is necessary.</p> <p>The fixed volume source density S_{ej} applied in a problem is always the sum of the SEO(K,J) values and the regionwise constant fixed sources specified by SEOR(J,IR), above.</p> <p>There is an option to compute, automatically, the SEO(K,J) values as a function of existing fluxes and various neutron group constants. See ISORS (I-0,0,15,0). When this option is used (i.e., ISORS $\neq 0$), all previously input values including any values input on the current problem, are replaced by the computed values in all groups and at all space points.</p>

Table III (Continued)

Index 1 Card Cols. 2-3	Index 2 Card Cols. 4-5	Index 3 Card Cols. 6-8	Index 4 Card Cols. 9-10	Mnemonic Name of Data	Notes on Application of Data
Format I2	Format I2	Format I3	Format I2		
11	IR (region number)	L (nuclide number)	0	RGCONC(L,IR)	Regionwise constant concentration for nuclide L in region IR. Provided there is a library set assigned to region IR (by means of the MRMIC(IR) data) and there is a nuclide numbered L in the library set, the initial concentration of nuclide L (for the time range) will be set to the value read for RGCONC(L,IR) at every source point in region IR. The first data word on the card is the concentration of nuclide L, the next is the concentration of nuclide L+1, etc.
12	L	K	0	PTCONC(K,L)	Pointwise concentrations for nuclide L. The initial concentrations of the nuclides used in a time range can be input pointwise with the PTCONC(K,L) input array. The first data word on the card is the concentration of nuclide number L at source point K=I+IR-1 (where I is the mesh point number and IR is the region number). The next word is the concentration at point K+1, etc.
13	IM (v. macro material number)	J (energy group number)	N (index number)	DCOF(N,J,M)	Function coefficients for computing the diffusion coefficients D_j for v. macro material IM > 40. The first data word on the card is the coefficient of index N = n [DCOF(N,J,M) = ad(n,J,M)] for computing D in group J = j of v. macro material m = IM = M+40. The next word is the coefficient of index N+1, etc. The maximum value of N is 4. See the discussion for IDCO (I-0,0,14,0) and for SCOF(N,J,M), below. The coefficients are used with $\rho_1(i)$ in the Fortran function DCSF(V,DCOF,M,J) of Equation (47) and (51).
14	IM (v. macro material number)	J (energy group number)	N (index number)	ACOF(N,J,M)	Function coefficients for computing the absorption cross section Σ_{aj} for v. macro material IM > 40. These coefficients are used with $\rho_1(i)$ in the Fortran function ABSF(V,ACOF,M,J) to compute Σ_{aj} for group j = J of v. macro material m = IM = M+40. The first word on the card is the coefficient for index N, the next is for index N+1, etc. See Equation (32) and the footnote on page 8.
15	IM (v. macro material number)	J (energy group number)	N (index number)	FNVCOF(N,J,M)	Function coefficients for computing the neutron production cross section $\nu\Sigma_{fj}$ for v. macro material IM > 40. These data are entered and read in the same manner as for DCOF and ACOF, above.
16	IM (v. macro material number)	J (energy group number)	N (index number)	FISCOF(N,J,M)	Function coefficients for computing the fission cross section Σ_{fj} for v. macro material IM > 40. These data are entered and read in the same manner as for DCOF and ACOF, above.
17	IM (v. macro material number)	J (energy group number)	N (index number)	SDCF(N,J,M)	Function coefficients for computing the neutron slowing down cross section (from group j to j+1) Σ_{sj} for v. macro material IM > 40. These data are entered and read in the same manner as for DCOF and ACOF, above.
18	IM (v. macro material number)	J (energy group number)	N (index number)	SCOF(N,J,M)	Function coefficients for computing the "transport" cross section $\Sigma_{\mu j}$ for v. macro material IM > 40. These data are entered and used in the same manner as for DCOF and ACOF, above. A $\Sigma_{\mu j}$ value for group j of a material is computed with the SCOF coefficients unless none are read (i.e., SCOF(N,J,M) = 0 for all N), then $\Sigma_{\mu j}$ is computed from the DCOF coefficients above. The diffusion coefficient D_j is computed using $\Sigma_{\mu j}$ when IDCO (I-0,0,14,0) is 1 or 2.
19	IM (v. macro material number)	J (energy group number)	N (index number)	BCOF(N,J,M)	Function coefficients for computing the buckling squared values B_j^2 for v. macro material IM > 40. These data are entered and read in the same manner as for DCOF and ACOF, above. If the integer input flag INBUCK (I-0,0,11,0) is 2 or 3, the B_j^2 values are set with the BSQM(J,IM) or BSQR(K) data, even for a v. macro material.
20	IM (v. macro material number)	K (Type of Group Constant Indicator)	N (index number)	COFMOD(N,K,M)	Function coefficients for computing the special functions $h_{k,k',m}$ for v. macro material IM > 40. The first data word on the card is the coefficient COFMOD(N,K,M) = b(n',k',m') for index N = n'; type K = k', and v. macro material IM = M+40 = m'+40; the next word is for index N+1, etc. These coefficients are used with feedback vector $\rho_2(k)$ in the Fortran function FMOD(V,COFMOD,M,K) of Equation (29) to compute the special function $h_{k,k',m}$ at source point k for v. macro material m > 40. The k' = K indicates which group constant is multiplied by the $h_{k,k',m}$ value as follows: k' = 1-D _j '; k' = 2- Σ_{aj} '; k' = 3- $\nu\Sigma_{fj}$ '; k' = 4- Σ_{fj} '; k' = 5- Σ_{sj} '; k' = 6- $\Sigma_{\mu j}$ '; k' = 7- B_j^2 '; where j' = j _{max} -1, the group above the thermal group. (The $h_{k,k',m}$ multiplication is applied only in group j _{max} -1.) See Equations (30) through (59).

2.4 The Library Set Title Card

The card format is I1,2I2,I3,2I2,15A4.

Column	1		Leave blank.
Columns	2-3	IAD	Zero or blank means the input data read for this library set are either new or are to completely replace the data of a library set with the same number on unit 8 where the library set data are stored. 1 punched means that the library set data read simply change or update the data already on unit 8 for library set number LIB.
Columns	4-5		Leave blank.
Columns	6-8	LIB	This is the library set number that identifies the library set data on unit 8. This number LIB must be greater than that of any preceding library set deck in the same stack. There must be a library set on unit 8 for every $MRMIC(IR) = LIB$ assigned to the regions in a time range.
Columns	9-10		Leave blank.
Columns	11-72		Any Hollerith characters can be punched here to further identify the library set.

2.5 The Library Set Integer Input Data

The format is I1,2I2,I3,2I2,15A4, and the data control information in card columns 1-12 consisting of IEND, Index 1, Index 2, Index 3, Index 4, and IWC have exactly the same function as described for the time range integer input data cards in Section 2.2, above. Following the library set title card, there must be at least one integer input data card per library set data deck.

The last integer input data card of each library set deck must have IEND=1 punched in card column 1 to signal the end of the integer input deck for that library set.

Each integer input word or array is described in detail in Table IV.

2.6 The Library Set Floating Point Data

The format is I1,2I2,I3,2I2,5E12.8, and the data control information in card columns 1-12 consisting of IEND, Index 1, Index 2, Index 3, Index 4, and IWC have exactly the same function as described for the time range

floating point data cards in Section 2.3, above. Following the library set integer input data deck, there must be at least one floating point data card per library set data deck.

The last floating point data card of each library material deck must have IEND=1 punched in card column 1 to signal the end of the deck for that material.

Each floating point data word or array is described in detail in Table V.

Table IV. Detailed Description of Library Set Integer Input Data

Index 1 Card Cols. 2-3 Format 12	Index 2 Card Cols. 4-5 Format 12	Index 3 Card Cols. 6-8 Format 13	Index 4 Card Cols. 9-10 Format 12	Mnemonic Name of Data	Notes on Application of Data
0	2	K (list word number)	0	ISONUM(K)	Specification of the list of nuclide λ numbers for the library set. The nuclide λ numbers assigned by the user to the nuclides of the library set are input by means of the ISONUM(K) list. The first data word on the card is the nuclide number given as the K th word of the list, the next data word is the K+1 th word in the ISONUM(K) list, etc. The library set can contain data for up to 50 nuclides, but only the nuclides appearing in the ISONUM(K) list are considered by the program.
0	6 (nuclide number)	L	0	IY(L)	Fission yield set number n_L for nuclide λ . The user specifies which of the three possible sets of fission yields to use for a fissionable nuclide λ by setting IY(L) = n_L (L = λ) equal to 1, 2, or 3. The first data word on the card is the yield set number for nuclide L, the next word is the set number for nuclide L+1, etc. See FYIELD (LF-7,N,L,0) for the fission yield set data of the library set.
0	7	K (list word number)	0	ISOSUM(K)	List of nuclides to be included in pointwise cross section edits. If the user sets the input flag IOUT2 = 4 (I-0,0,26,0), there is an edit every time point of the pointwise contributions of the library set nuclides to the total macroscopic cross sections. These edited contributions can be restricted to a smaller number of nuclides by listing the desired nuclides of the set in the ISOSUM list. The first data word on the card is the nuclide number λ given as the K th word in the ISOSUM list, the next word is the nuclide number appearing K+1 th in the list, etc.
0	8	K (list word number)	0	ISOED(K)	List of nuclides to be included in the pointwise concentration edits. If the user sets the input flag IOUT2 $\geq$ 3 (I-0,0,26,0), there is an edit every time point of the pointwise concentrations of each nuclide in the library set. The edited concentrations can be limited to a selected number of nuclides by listing them in the ISOED list. The first word on the card is the nuclide number λ appearing K th in the list, the next word is the nuclide number appearing K+1 th in the list, etc.
1	0	0	0	L,IK(I,L), MODE(I,L)	Specification of daughters and modes of production of each parent nuclide in the library set. There can be four branches (each to a daughter nuclide) from each parent nuclide. The IK(I,L) input data determine the flow of a production and depletion scheme. IK(I,L) must be set to the nuclide λ' number of the daughter produced by the I th branch of parent nuclide L $\equiv \lambda$. The MODE(I,L) input data identify which cross section(s) or which decay yield is to be used in producing the daughter. Positive MODE(I,L) values from 1 to 2 denote decay production; and a value of, say, 2 causes the program to use DYIELD(2,L) as the fraction of the decay of parent L to the daughter. Negative values of MODE(I,L) of -1 and -2 denote neutron capture production; and a value of, say, -2 causes the program to use the cross section SIGPRO(J,2,L) in calculating the production of the daughter by capture of group J neutrons in parent nuclide L. The first word (in columns 13-16) on each of these cards must be the λ number (L) of a parent nuclide. The next word is IK(1), the nuclide λ' number of the daughter produced by the first branch. The next word is the mode value MODE(1,L) for the first branch. If there is more than one branch from the parent, the next two words are, respectively, IK(2,L) and MODE(2,L). If there are more branches, the pairs of IK(I,L) and MODE(I,L) values are listed in turn across the card in like manner. The daughters and modes for other parent nuclides can be specified on the same card if a single blank word is left between the last specified MODE(I,L) value and the L value of the next parent. [However, the IK(I,L) and MODE(I,L) values of a parent L cannot be continued on a second card; i.e., all the data for a given parent must fit on the card on which it is begun.] The following example shows how these data are punched on a card for a parent nuclide $\lambda = 15$ that produces daughters $\lambda' = 10$ and $\lambda' = 6$ by decay and nuclides $\lambda' = 27$ and $\lambda' = 7$ by neutron capture.
<div> <div>Cols 13-16 17-20 21-24 25-28 29-32 33-36 37-40 41-44 45-48 49-52 </div> <div>Word 15 10 1 27 -1 6 2 7 -2 blank </div> </div>					
2	0	0	0	L,IZN(L), IAN(L)	Another parent could be handled on the same card if it had two or less branches. Specification of nuclide element and mass numbers for the library set. The element number $n_Z(\lambda)$ (such as 92 for uranium) and the mass number $n_A(\lambda)$ (such as 235 for U-235) for nuclide λ can be optionally specified by means of these data. The nuclides for which these data are specified will be identified throughout most of the output edits by these numbers together with the element symbol (such as "U" for uranium). The first word (in columns 13-16) on each card must be the λ number (L) of a nuclide, the next word is the element number $n_Z(\lambda) \equiv$ IZN(L) for the nuclide, and the next word is the mass number $n_A(\lambda) \equiv$ IAN(L). Other nuclides can be similarly identified on the same card by following the IAN(L) word for one nuclide by the L, IZN(L), and IAN(L) values of another. No blanks or zero words can occur between the triplets; hence, up to five nuclides can be identified on a card.

Table V. Detailed Description of Library Set Floating Point Data

Index 1 Card Cols. 2-3	Index 2 Card Cols. 4-5	Index 3 Card Cols. 6-8	Index 4 Card Cols. 9-10	Mnemonic Name of Data	Notes on Application of Data
Format I2	Format I2	Format I3	Format I2		
2	0	L (nuclide number)	0	DECAY(L)	The radioactive decay constants λ_L of the nuclides The decay constant (or reciprocal lifetime) λ_L of each radioactive nuclide is given in the DECAY(L) input data list. The constant is in reciprocal time units and is given by $\lambda_L \pm 693/T_{1/2L}$, where $T_{1/2L}$ is the half-life of nuclide L. The value of Index 3 (L) indicates that the first input word on the card is the decay constant for nuclide $L \equiv L$, the next word is for nuclide $L+1$, etc.
2	M (decay mode number)	L (parent nuclide number)	0	DYIELD(M,L)	The radioactive decay yield fractions $Y_d(m,L)$ of the radioactive parent nuclides No decay yield fractions [DYIELD(M,L)] need be read in for parent nuclides that produce only one daughter by radioactive disintegration. The program appropriately assumes a value of unity for the single decay fraction in such cases. For each parent nuclide L that produces two or more daughters by radioactive decay, a positive mode number [$m(i_L',L) \equiv \text{MODE}(I,L)$] of 1 or 2 must be assigned to each decay production branch. The fraction of the disintegrations that produce the daughter L' of that branch i_L' is then supplied in terms of the assigned mode number as $Y_d(m,L)$ for parent nuclide L. There can be yield fraction data for only one parent nuclide on a card. The value of Index 2 (M) indicates that the first word on the card is the yield fraction for decay mode $m(i_L',L) = M$, the next word on the card is the yield for decay mode $M+1$, if any.
4	J (neutron group number)	L (nuclide number)	0	ABSM(L,J), SIGPRO(J,1,L), SIGPRO(J,2,L), SCATM(L,J) & SLODNM(L,J)	Microscopic cross sections of nuclide $L \equiv L$ for neutron energy group $j \equiv J$ The following definitions apply (for group $j \equiv J$ and nuclide $L \equiv L$): ABSM(L,J) $\equiv \sigma_{aj}$, microscopic absorption cross section. SIGPRO(J,1,L) $\equiv \sigma_{c1}$, microscopic capture production cross section for the first capture mode ($m = \text{MODE}(I,L) = 1$, where I denotes the Ith branch from parent $L \equiv L$). SIGPRO(J,2,L), microscopic capture production cross section for the second capture mode ($m = 2$). SCATM(L,J) $\equiv \sigma_{tj}$, microscopic "transport" cross section. SLODNM(L,J) $\equiv \sigma_{sj}$, microscopic slowing down cross section for scattering neutrons from group J to J+1. With this card format, Index 2 (J) indicates that the data on the card are for group J and Index 3 (L) indicates the data are for nuclide $L \equiv L$. The first word on the card is ABSM(L,J), and it is followed by SIGPRO(J,1,L), SIGPRO(J,2,L), and SCATM(L,J), in turn. The last (fifth) word on the card is SLODNM(L,J). Since most users prefer to input the microscopic cross sections in barn (10^{-24}cm^2) units, it is assumed that the values punched on these cards are in barn units. All the data are multiplied automatically by 10^{-24} .
4	J (neutron group number)	L (nuclide number)	N (data type indicator)	ABSM(L,J), or SIGPRO(J,1,L), or SIGPRO(J,2,L), or SCATM(L,J), or SLODNM(L,J)	Alternate specification of microscopic cross sections (or the optional thermal cross section multiplier for nuclide $L \equiv L$ for neutron energy group J See the discussion immediately preceding for the detailed description of the data. All cross section values read are multiplied by 10^{-24} in the program to convert automatically from barn units to cm^2 units. It is sometimes more convenient to input these data in the order permitted by this alternate type of card. As in the previous card type, Index 2 (J), indicates the neutron group number of the data. A nonzero value of Index 4 (N), however, indicates the data on the card are as follows: N=1 indicates that the first word on the card is ABSM(L,J) for nuclide L, the next word on the card is ABSM(L+1,J) (for nuclide L+1), etc. for up to five nuclides. N=2 indicates that the first word on the card is SIGPRO(J,1,L) for nuclide L, the next is SIGPRO(J,1,L+1) (for nuclide L+1), etc. N=3 indicates that the first word on the card is SIGPRO(J,2,L) for nuclide L, the next word is SIGPRO(J,2,L+1), etc. N=4 indicates that the first word on the card is SCATM(L,J), the next is SCATM(L+1,J), etc., for up to five nuclides. N=5 indicates that the first word on the card is SLODNM(L,J) for nuclide L, the next word is SLODNM(L+1,J), etc.

Table V (Continued)

Index 1 Card Cols. 2-3 Format 12	Index 2 Card Cols. 4-5 Format 12	Index 3 Card Cols. 6-8 Format 13	Index 4 Card Cols. 9-10 Format 12	Mnemonic Name of Data	Notes on Application of Data
5	J (neutron group number)	L (nuclide number)	0	GNUF(L,J), FISM(L,J), EFIS(L,J)	Microscopic, fission-related cross sections and energy per fission of nuclide $\lambda \equiv L$ for neutron energy group $j \equiv J$. The following definitions apply: $GNUF(L,J) \equiv v\sigma_f^j$, the product of neutrons per fission times the microscopic fission cross section. $FISM(L,J) \equiv \sigma_f^j$, the microscopic fission cross section. $EFIS(L,J) \equiv \sigma_f^j$, the energy per fission of nuclide $\lambda \equiv L$ when the fissions result from absorption of a group $j \equiv J$ neutron. The first word on the card is GNUF(L,J), the next is FISM(L,J), and the third word is EFIS(L,J). The last two words on the card are blank with this format. (See immediately below for an alternate format for reading these data.) It is assumed that the units of GNUF(L,J) and FISM(L,J) are barns; hence, the program automatically multiplies these data by 10^{-24} as soon as they are read. EFIS(L,J) is not multiplied. Alternate specification of the fission-related microscopic cross sections and energy per fission of nuclide $\lambda \equiv L$ for neutron energy group $j \equiv J$. See the immediately preceding discussion for the detailed descriptions of these data. The GNUF(L,J) and FISM(L,J) data are multiplied by 10^{-24} as soon as they are read to convert from barn units to cm^2 . EFIS(L,J) values are not multiplied. As with the previous format, Index 2 (J) is the group number of the data on the card. However, when Index 4 (N) is 1, 2, or 3, the data on the card is interpreted as follows: N=1 indicates that the first word on the card is GNUF(L,J) for nuclide L, the next word is GNUF(L+1, J) (the value for nuclide L+1), etc.; such that the GNUF(L,J) values can be read for five nuclides on one card. N=2 indicates that the first word on the card is FISM(L,J) for nuclide L, the next word is FISM(L+1,J), etc. N=3 indicates that the first word on the card is EFIS(L,J) for nuclide L, the next word is EFIS(L+1,J), etc.
5	J (neutron group number)	L (nuclide number)	N (data type indicator)	GNUF(L,J) or FISM(L,J) or EFIS(L,J)	
7	N (yield set number)	L (nuclide number)	0	FYIELD(N,L)	The fission yield fraction, $Y_f(N,\lambda)$ for the direct fission production of fission product nuclide $\lambda \equiv L$ as specified for fission yield set $n \equiv N$. Instead of specifying a different set of fission product yield fractions for each fissionable nuclide, the user specifies only up to three different sets of yield fractions and then specifies which of these three sets to use for each of the fissionable nuclides or macro material. (The former is done by means of the IY(L) input list; and the user specifies which of the three sets to use for the nondepleting macro material fissions by means of the MY(IM) input list.) Index 2 (N) is 1, 2, or 3 to indicate which of the three sets the data on the card specifies. Index 3 (L) indicates that the first word on the card is the direct fission yield fraction, $FYIELD(N,L) \equiv Y_f(n,\lambda)$, for fission product nuclide $\lambda \equiv L$, the next word on the card is the yield for nuclide L+1, etc., for up to five consecutively numbered nuclides.

3. INPUT DATA SUMMARY

A compact summary of the input data is provided to serve as a convenient memory aid for the experienced user. See Section 2.1 through 2.6 for the detailed descriptions of the input data.

Table VI. The Time Range Title Card Summary

Information or Mnemonic Name of Data	Cards Columns	Description or Function
NUPROB	1-5	"START" punched indicates this is the first time range of a problem
Any Hollerith Characters	6-65	Problem title information
IZERO	67-68	A nonzero value indicates a completely new problem
ILIST	69-70	A nonzero value causes edits of contents of all input data cards

See Section 2.1 for more detail.

Table VII. Summary of Time Range Integer Input Data

Indices 1,2,3 and 4 in Card Cols. 2-10	Mnemonic Name of Data	Description or Function of Data	Index (1,2,3 or 4) That Increases As Words Are Added To Card
0, 0, 1,0	JMAX	Number of energy groups	(3)
0, 0, 2,0	IRMAX	Number of regions	(3)
0, 0, 3,0	NGEO	Geometry indicator: 0-slab, 1-cylinder, 2-sphere	(3)
0, 0, 4,0	NIBDRY	Inner boundary condition indicator: $0-\phi=0$, $1-J=0$, $2-\phi=\omega_1$, $3-\frac{d\phi}{dx}=\omega_1$, $4-\frac{1}{\phi}\frac{d\phi}{dx}=1/\omega_1$, $5-\frac{1}{\phi}\frac{d\phi}{dx}=\frac{1}{.71\lambda_{tr}}=1/(.71)(3D)$, $6-\frac{1}{\phi}\frac{d\phi}{dx}=1/(.71)(3D)(G_1)$, $7-\alpha\phi-\beta J+\gamma=0$, 8-periodic	(3)
0, 0, 5,0	NOBDRY	Outer boundary indicator: $0-\phi=0$, $1-J=0$, $2-\phi=\omega_2$, $3-\frac{d\phi}{dx}=\omega_2$, $4-\frac{1}{\phi}\frac{d\phi}{dx}=1/\omega_2$, $5-\frac{1}{\phi}\frac{d\phi}{dx}=\frac{1}{.71\lambda_{tr}}=1/(.71)(3D)$, $6-\frac{1}{\phi}\frac{d\phi}{dx}=1/(.71)(3D)(G_2)$, $7-\alpha\phi-\beta J+\gamma=0$, 8-periodic	(3)
0, 0, 6,0	IREN	Fission neutron source renormalization indicator: 0-normalization, 1-no normalization	(3)
0, 0, 7,0	IADJ	Type of flux solution indicator: 0-real, 1-adjoint	(3)
0, 0, 8,0	IGENOP	Controls how generation factor is computed for searches: 0-usual eigenvalue, 1-fission neutrons per fixed source neutron on first iteration	(3)
0,0,9-10,0		Ignored	
0, 0,11,0	INBUCK	Specifies degree of detail and form of input bucklings: 0-material and group dependent, 1-material dependent, 2-group dependent, 3-constant	(3)
0, 0,12,0	IBSQO	Specifies whether buckling data are entered by region or by macro material: 0-by macro material, 1-by region	(3)
0, 0,13,0		Ignored	
0, 0,14,0	IDCO	Specifies type of diffusion coefficient formula used: 0-D is computed from macro material data, 1-D = $1/3(\Sigma_u + \Sigma_p)$, 2-D = Σ_r/κ^2	(3)
0, 0,15,0	ISORS	Specifies type of pointwise fixed volume source automatically computed: 0-none computed, 1-Sp = 0, 2-Sp = $v\Sigma_f\phi$, 3-Sp = $\chi\phi$, 4-Sp = $\Sigma_f\phi$, 5-Sp = $E\phi$, 6-Sp = $\Sigma_u\phi$, 7-Sp = $\Sigma_a\phi$, 8-Sp = $T\phi$	(3)
0, 0,16,0	ISTORE	Indicates source of nuclide concentration data for first time range of a problem: 0-card input, only, 1-initial concentrations used on time range of previous problem read from unit 4	(3)
0,0,17-18,0		Ignored	
0, 0,19,0	ICHIM	Nonmaterial dependent fission spectra indicator: 0-all x_j^m are read in, n-all $x_j^m = x_j^n$	(3)
0, 0,20,0		Ignored	
0, 0,21,0	INED1	Input data edit control flag: 0-all edits done, 1-no edits done of time range input data	(3)
0, 0,22,0	INED2	Reserved	
0, 0,23,0	INED3	Reserved	
0, 0,24,0		Ignored	
0, 0,25,0	IOUT1	First output data edit control flag: 0-problem summary, 1-plus fluxes and sources, 2-plus peak to averages, 3-plus neutron balances, 4-plus currents and leakages	(3)
0, 0,26,0	IOUT2	Second output data edit control flag: 0-no edits, 1-edit region averaged concentrations, 2-plus region averaged library set cross sections, 3-plus pointwise concentrations, 4-plus pointwise library set cross sections	(3)
0, 0,27,0	IOUT3	Third output data edit control flag: 0-no edits, 1-edit feedback vector distributions and related data each time point	(3)
0, 0,28,0	ILINES	Controls the number of lines of neutron fluxes printed per region: 0 or 1- every line, n $\geq$ 2-every n th line per region	
0, 0,29,0	ITDATA	Controls edits of fission source iteration data: 0-no edit, 1-edit	(3)
0, 0,30,0		Ignored	
0, 0,31,0	LIBED	Library set data edit indicator: 0-no edit, 1-edit data of sets used in the time range, 2-edit data of all sets on unit 4	(3)
0,0,32-35,0		Ignored	
0, 0,36,0	NORMAL	Specifies type of "normalization" to be used during output edits: 0-no normalization, 1-fission rate = POWER, 2- $\phi=1$, 3-fission source = 1, 4-multiply by FLMULT	(3)

Table VII (Continued)

Indices 1,2,3 and 4 In Card Cols. 2-10	Mnemonic Name of Data	Description or Function of Data	Index (1,2,3 or 4) That Increases As Words Are Added To Card
0, 0,37,0	MESHNO	Specifies mesh or source point used as basis for normalization when NORMAL = 2 or 3	(3)
0, 0,38,0	NOGRP	Specifies neutron group used as basis for normalization when NORMAL = 2	(3)
0, 0,39,0	Ignored		
0, 0,40,0	IGRID	Background grid control flag for plotted output: 0-no grid, 1-coarse grid, 2-fine grid	(3)
0, 0,41,0	IPLOT	Plotted output control flag: 0-no plotted output, n-ordinate scale determined by maximum value for plot vector n	(3)
0, 0,42,0	IVEK1	Plot vector 1 control flag: 0-no plot, 1-Cf*tt ¹ , 2-ft, 3-p, 4-ρ ₁ , 5-ρ ₂ , 6-ρ ₃ , 7-φ ₁ , 8-φ _{jmax} , 9- $\sum_{k \in L_r} N_k(k) \sigma_{a,j}^k$, 10- $\sum_{k \in L_r} N_k(k) v \sigma_{f,j}^k$, 11-q, 12-Q, 13-H, 14-T _k	(3)
0, 0,43,0	IVEK2	Plot vector 2 control flag: 0-no plot, 1- through 14-same as for IVEK1, above	(3)
0, 0,44,0	IVEK3	Plot vector 3 control flag: 0-no plot, 2-plot concentration N _k (k) for nuclide number 2	(3)
0, 0,45,0	Ignored		
0, 0,46,0	INSORS	Specifies type of fission neutron source guess: 0-unity, ±1-as saved, ±2-flat and equal to FR(IR), ±3-input pointwise, ±4-set to zero	(3)
0, 0,37,0	IGESS	Specifies fission neutron source saved for next problem: 0-last source computed, 1-source used on last iteration	(3)
0, 0,48,0	NCONVG	Specifies convergence based on total or pointwise fission source: 0-total source, 1-pointwise source	(3)
0, 0,49,0	MAXOUT	Specifies maximum number of fission source iterations	(3)
0, 0,50,0	NEXTO	Specifies type of neutron fission source extrapolation: 0-no extrapolation, 1-two point type using THETA, 2-Chebyshev three point	(3)
0, 0,51,0	NPIO	Number of unextrapolated source iterations (on Chebyshev type)	(3)
0, 0,52,0	NCHMAX	Maximum order of Chebyshev polynomial allowed	(3)
0, 0,53,0	NERCHK	Iterations with old Chebyshev parameters before using new ones	(3)
0, 0,54,0	IFEED	First feedback control flag: 0-no feedback iterations performed, 1-iterate and compute ρ ₁ = coolant density, 2-iterate and compute ρ ₁ = H, 3-iterate and compute ρ ₁ = q	(3)
0, 0,55,0	JFEED	Second feedback controlling ρ ₂ calculation: 0-ρ ₂ not computed, 1-ρ ₂ = Tfuel, 2-ρ ₂ = H, 3-ρ ₂ = q (provided 1 ≤ IFEED ≤ 3)	(3)
0, 0,56,0	KFEED	Third feedback flag controlling ρ ₃ calculation (presently not used)	(3)
0, 0,57,0	NFEEDL	Feedback iteration maximum: 0-set to 3, n-compute feedback vectors no more than n times	(3)
0, 0,58,0	IRHO	Specifies initial distribution of ρ ₁ for the time range: 0-ρ ₁ = 0, 1-ρ ₁ = 1, 2-ρ ₁ = RHOGES, 3-ρ ₁ = RHO(I) or as saved	(3)
0, 0,59,0	IPHO	Specifies initial distribution of ρ ₂ for the time range: 0-ρ ₂ = 0, 1-ρ ₂ = 1, 2-ρ ₂ = PHOGES, 3-ρ ₂ = PHO(K) or as saved	(3)
0, 0,60,0	ISHO	Specifies initial distribution of ρ ₃ for the time range: 0-ρ ₃ = 0, 1-ρ ₃ = 1, 2-ρ ₃ = SHOGES, 3-ρ ₃ = SHO(K) or as saved	(3)
0, 0,61,0	ISERCH	Specifies type of search to employed: 0-none, ±1-buckling, ±2-poison, ±3-dimension, ±4-total fission neutron source, ±5-boundary between two regions, ±6-total power	(3)
0, 0,62,0	MAXSCH	Maximum number of search trials	(3)
0, 0,63,0	NSERCH	Specifies source of the desired search eigenvalue EIGEND: 0-EIGEND input, 1-EIGEND equals last problem eigenvalue	(3)
0, 0,64,0	IENPF	Flag controlling input of macro material energies per fission: 0-E _j ^m = EPFISS, 1-E _j ^m = ENERGY(1,IM), 2-E _j ^m = ENERGY(J,IM)	(3)
0, 0,65,0	Ignored		
0, 0,66,0	LIBOLD	Data set identification number written on the saved library set unit	(3)
0, 0,67,0	LIBNEW	Data set identification number to be written on the updated library set unit	(3)
0, 0,68,0	LIBMTL	Number of library set decks to be read following the deck of this time range	(3)
0, 0,69,0	MODIFY	Flag controlling modification of nuclide loss coefficients: 0-no checking done, 1-check and modify λ's and σ _a 's such that no loss coefficients are equal in a library set	(3)
0, 0,70,0	ISHTON	Reactor "shutdown" flag: 0-compute fluxes as usual, 1-φ = 0 for all concentration calculations of the time range	(3)
0, 1,IR,0	MR(IR)	Macroscopic material number to be used in region IR	(3-IR)
0, 2,IR,0	NI(IR)	Number of mesh intervals to be used in region IR	(3-IR)
0, 3,IR,0	LRS(K)	A search is restricted to the materials or regions named in this list if LRS(1) ≠ 0	(3-K)
0, 4, K,0	IYUK(K)	List of macro materials for which diffusion coefficients are to be automatically computed if IDCO = 1 or 2	(3-K)
0, 7,IR,0	MRMIC(IR)	Library set number to be used in region IR	(3-IR)
0, 8,IM,0	MY(IM)	Fission product yield set number n _m to be used with macro material m = IM	(3-IM)
0,10, K,0	IREDT(K)	List of regions for which the nuclide cross section and concentration edits of 1 ≤ IOUT2 ≤ 3 are to be made	(3-K)

Table VIII. Summary of Time Range Floating Point Input Data

Indices 1,2,3 and 4 In Card Cols. 2-10	Mnemonic Name of Data	Description or Function of Data	Index (1,2,3 or 4) That Increases As Words Are Added To Card
0, 0, 1,0	EPS1	Desired convergence level for either total or pointwise fission source	(3)
0, 0, 2,0	EPS2	Desired convergence level for the generation factor on searches	(3)
0, 0, 3,0	EPS3	Desired convergence level for the search parameter on searches	(3)
0, 0, 4,0	EPS4	Desired convergence level for pointwise power on feedback iterations	(3)
0, 0, 5,0	EPS5	Convergence level of power residual on inner feedback iterations	(3)
0, 0, 6,0	RADI	Position of the first mesh point, the inner boundary x_1	(3)
0, 0, 7,0		Ignored	
0, 0, 8,0	THETA	Fixed extrapolation factor for neutron fission source acceleration	(3)
0, 0, 9,0	FISSOR	Total volume integral of fission neutron source F_0	(3)
0, 0,10,0	FLMULT	Fixed multiplier for output source and fluxes	(3)
0, 0,11,0	POWER	Total volume integral of fission power for output edits	(3)
0, 0,12,0	WMI	Minimum allowable region width on boundary searches	(3)
0, 0,13,0		Ignored	
0, 0,14,0	EIGEND	Desired generation factor to be attained on a search λ_{dm}	(3)
0, 0,15,0	EPFISS	Optional energy per fission for setting macro material E_j^m values	(3)
0, 0,16,0	TSTART	Time range starting time T_0	(3)
0, 0,17,0	TIMEND	Time range ending time T_e	(3)
0, 0,18,0	DELTIM	Time range time step size Δt	(3)
0, 0,19,0	RNDOFF	Round off error bound in concentration calculations	(3)
0, 0,20,0	AVPOWR	Total volume integral of fission power for feedback and concentration calculations	(3)
0, 0,21,0	HINLET	Coolant inlet enthalpy H_0	(3)
0, 0,22,0	HFSAT	Enthalpy of the saturated coolant fluid H_{fsat}	(3)
0, 0,23,0	HFG	Fluid to vapor enthalpy rise of coolant fluid h_{fg}	(3)
0, 0,24,0	HCONV	Energy conversion factor K_1 (usually 9.48×10^{-4} Btu/w-sec)	(3)
0, 0,25,0	FLOW	The mass flow rate of the coolant w_c	(3)
0, 0,26,0	GASRHO	The density of the saturated coolant vapor ρ_g	(3)
0, 0,27,0	PRESSR	The pressure of the coolant system (not presently used)	(3)
0, 0,28,0	AREA	The surface area A_c of fuel clad per unit volume of core	(3)
0, 0,29,0	HFILM	The film coefficient h_c for the clad to bulk coolant heat transfer	(3)
0, 0,30,0	CRITV	The "critical" vapor fraction α_t for defining a transition region from the nucleate to bulk boiling regions	(3)
0, 0,31,0	HFRAC	The "critical" enthalpy fraction for defining a transition region from the nucleate to bulk boiling regions	(3)
0, 0,32,0	HCON	The density formula fitting parameter H_c for the superheated vapor region	(3)
0, 0,33,0		Ignored	
0, 0,34,0	RHOGES	Optional input value for initial value of ρ_1 for the time range	(3)
0, 0,35,0	PHOGES	Optional input value for initial value of ρ_2 for the time range	(3)
0, 0,36,0	SHOGES	Optional input value for initial value of ρ_3 for the time range	(3)
0, 0,37,0	HFUEL	Effective heat transfer coefficient h_f for heat transfer from fuel to coolant	(3)
0,0,38-39,0		Ignored	
0, 0,40,0	RHOEXT	Extrapolation factor for ρ_1 distribution	(3)
0, 0,41,0	PHOEXT	Extrapolation factor for ρ_2 distribution	(3)
0, 0,42,0	SHOEXT	Extrapolation factor for ρ_3 distribution (not presently used)	(3)
0,0,43-44,0		Ignored	
0, 0,45,0	VMAX1	Optional maximum value for plot vector 1	(3)
0, 0,46,0	VMAX2	Optional maximum value for plot vector 2	(3)
0, 0,47,0	VMAX3	Optional maximum value for plot vector 3	(3)
0, 1,IR,0	RW(IR)	Region width for region number IR	(3-IR)
0, 2,IM,0	BSQR(IM)	Buckling squared value (group independent) for macroscopic material IM	(3-IM)
0, 3,IR,0	FR(IR)	Optional flat, region dependent, fission neutron source guess for region IR	(3-IR)
0, 4,IM,0	SIGPT(IM)	Poison cross section for macroscopic material IM	(3-IM)
0, 5, 1,J	OMEGA1(J)	Inner boundary condition parameter for group J	4-J
0, 5, 2,J	OMEGA2(J)	Outer boundary condition parameter for group J	4-J
0, 5, 3,J	ALPHA1(J)	Inner boundary condition neutron flux multiplier for group J	4-J
0, 5, 4,J	BETA1(J)	Inner boundary condition neutron current multiplier for group J	4-J
0, 5, 5,J	GAMMA1(J)	Inner boundary condition arbitrary constant for group J	4-J
0, 5, 6,J	ALPHA2(J)	Outer boundary condition neutron flux multiplier for group J	4-J
0, 5, 7,J	BETA2(J)	Outer boundary condition neutron current multiplier for group J	4-J
0, 5, 8,J	GAMMA2(J)	Outer boundary condition arbitrary constant for group J	4-J
0, 6, 1,N	RHOLCS(N)	Polynomial coefficients for computing unsaturated coolant fluid density	(4-N)
0, 6, 2,N	TEMPCS(N)	Polynomial coefficients for computing unsaturated coolant fluid temperature	(4-N)
0, 6, 3,N	SLIPCS(N)	Optional coefficients (not presently used)	
0, 7, 1,0	RHO(I)	Optional pointwise specification of ρ_1 at start of time range	(3-I)
0, 8, K,0	PHO(K)	Optional pointwise specification of ρ_2 at start of time range	(3-K)
0, 9, K,0	SHO(K)	Optional pointwise specification of ρ_3 at start of time range	(3-K)
0,10, K,0	F(K)	Pointwise fission neutron source guess for source point K	(3-K)
0,12, 1,J	PHI(1,J)	Pointwise flux input for group J and mesh point I	(3-I)

Table VIII (Continued)

Indices 1,2,3 and 4 In Card Cols. 2-10	Mnemonic Name of Data	Description or Function of Data	Index (1,2,3 or 4) That Increases As Words Are Added To Card
1,IM, J,0	DC(J,IM) SIGA(J,IM), SIGNU(J,IM), SIGFIS(J,IM), ENERGY(J,IM)	Diffusion coefficient, absorption cross section, neutrons per fission times the fission cross section, fission cross section, and energy per fission for group J of f. macro material IM; values are listed across a card in the order given (all five words are read, regardless of IWC)	(None)
1,IM, J,1	DC(J,IM)	Diffusion coefficient for group J of f. macro material IM	(J)
1,IM, J,2	SIGA(J,IM)	Absorption cross section for group J of f. macro material IM	(J)
1,IM, J,3	SIGNU(J,IM)	Neutrons per fission times fission cross section for group J of f. macro material IM	(J)
1,IM, J,4	SIGFIS(J,IM)	Fission cross section for group J of f. macro material IM	(J)
1,IM, J,5	ENERGY(J,IM)	Energy per fission for group J of f. macro material IM	(J)
2,IM, J,0	SIGSD(J,IM)	Optional scattering transfer cross section $\Sigma_{s,j}^m$ (for neutron scatter) from group J to group J+1 for f. macro material $m \equiv IM$	(J)
4,IM, J,0	SIGMU(J,IM)	Cross section $\Sigma_{t,j}^m$ for group $j \equiv J$ of f. macro material $m \equiv IM$	(J)
5,IM, J,0	CHIM(J,IM)	Fraction of total fission neutrons released in group J of macro material IM	(J)
6, K, J,0	BSQM(J,K)	Buckling squared value for group J of macro material $IM \equiv K$ or region $IR \equiv K$ [see fixed point data, INBUCK (0,0,11,0) and IBSQO (0,0,12,0)]	(J)
7,IM, J,0	T(J,IM)	Fraction of poison cross section SIGPT(IM) used in group J	(J)
8,IM, J,0	Q(J,IM)	Logarithmic boundary condition parameter applied in group J of f. macro material IM	(J)
9,IM, J,0	SEOR(J,IR)	Flat fixed volume source in group J and region IR	(J)
10, J, K,0	SEO(K,J)	Pointwise fixed volume source for group J and source point K	(K)
11,IR, L,0	RGONC(L,IR)	Regionwise constant initial concentration of nuclide L in region IR	(3-L)
12, L, K,0	PTCONC(K,L)	Concentration of nuclide L at source point $K = I+IR-1$	(3-K)
13,IM, J,N	DCOF(N,J,M)	Function coefficients for computing D_j^m for v. macro material $m = IM > 40$	(4-N)
14,IM, J,N	ACOF(N,J,M)	Function coefficients for computing $\Sigma_{a,j}^m$ for v. macro material $m = IM > 40$	(4-N)
15,IM, J,N	FNUCOF(N,J,M)	Function coefficients for computing $\Sigma_{f,j}^m$ for v. macro material $m = IM > 40$	(4-N)
16,IM, J,N	FISCOF(N,J,M)	Function coefficients for computing $\Sigma_{f,j}^m$ for v. macro material $m = IM > 40$	(4-N)
17,IM, J,N	SDCF(N,J,M)	Function coefficients for computing $\Sigma_{s,j}^m$ for v. macro material $m = IM > 40$	(4-N)
18,IM, J,N	SCOF(N,J,M)	Function coefficients for computing $\Sigma_{u,j}^m$ for v. macro material $m = IM > 40$	(4-N)
19,IM, J,N	BCOF(N,J,M)	Function coefficients for computing B_j^m for v. macro material $m = IM > 40$	(4-N)
20,IM, K,N	COFMOD(N,K,M)	Function coefficients for computing the special functions $h_{k,k',m}$ that modify the group constants of the epithermal group j_{max-1} for v. macro material $m = IM > 40$	(4-N)

Table IX. The Library Set Title Card Summary

Information or Mnemonic Name of Data	Card Columns	Description or Function
blank	1	None
IAD	2-3	A value of 1 means update rather than replace or add this library set to the library set data unit
blanks	4-5	None
LIB	6-8	Library set identification number LIB
blanks	9-10	None
Any Hollerith Characters	11-72	Library set title information

Table X. Summary of Library Set Integer Input Data

Indices 1,2,3 and 4 In Card Cols. 2-10	Mnemonic Name of Data	Description or Function of Data	Index (1,2,3 or 4) That Increases As Words Are Added To Card
0, 2, K, 0 0, 6, L, 0 0, 7, K, 0 0, 8, K, 0 1, 0, 0, 0	ISONUM(K) IY(L) ISOSUM(K) ISOED(K) L,IK(I,L) MODE(I,L)	List of nuclide λ numbers to be used in the library set in the time range Fission yield set number n_2 to use for fissionable nuclide $\lambda = L$ List of nuclides to be included in pointwise cross section edits List of nuclides to be included in pointwise concentration edits Parent nuclide number $\lambda = L$, daughter $\lambda' = IK(I,L)$ produced by branch $i_{\lambda'} = I$ and production mode number $m(i_{\lambda'}, \lambda) = MODE(I,L)$ of the branch Assign each branch from a parent nuclide L a distinct number I . Assign a mode number $[MODE(I,L)]$ to each branch using mode numbers +1 or +2 for decay branches and mode numbers -1 or -2 for neutron capture branches. The first data word on a card (columns 13-16) is always a parent nuclide L number; the next two words are the λ' number, $IK(I,L)$, of the daughter and the mode number of the first branch ($I=1$); the next two words are the daughter and mode numbers of the second branch, etc. The daughters and modes of another parent nuclide can be put on the same card provided there is room for their complete specification after leaving one blank word between the last mode number and the L number of the new parent. Specification of the element numbers and mass numbers of the nuclides For each nuclide L to be identified, give the triplet L [the nuclide λ number], $IZN(L)$ [the element number $nz(\lambda)$], and $IAN(L)$ [the mass number $nA(\lambda)$] of the nuclide beginning with the first data word (columns 13-16) on a card. The triplet for any other nuclide can be given in the next three words, etc., for a maximum of five nuclides per card.	(3-K) (3-L) (3-K) (3-K) (None)
2, 0, 0, 0	L,IZN(L), IAN(L)		(None)

Table XI. Summary of Library Set Floating Point Data

Indices 1,2,3 and 4 In Card Cols. 2-10	Mnemonic Name of Data	Description or Function of Data	Index (1,2,3 or 4) That Increases As Words Are Added To Card
2, 0, L, 0 2, M, L, 0	DECAY(L) DYIELD(M,L)	The decay constant λ_{λ} of nuclide $\lambda = L$ (in reciprocal time units) The $Y_d(m, \lambda)$ of radioactive disintegrations of parent nuclide $\lambda = L$ that result in production of daughter nuclide λ' by means of branch $i_{\lambda'}$ with an assigned mode number of $m(i_{\lambda'}, \lambda)$	(3-L) (2-M)
4, J, L, 0	ABSM(L,J) SIGPRO(J,1,L) SIGPRO(J,2,L) SCATM(L,J) SLODNM(L,J)	Microscopic cross sections σ_{aj}^{λ} , $\sigma_{c(j,1,\lambda)}$, $\sigma_{c(j,2,\lambda)}$, $\sigma_{u,j}^{\lambda}$, and σ_{sj}^{λ} for group $j = J$ of nuclide $\lambda = L$	(None)
4, J, L, 1 4, J, L, 2 4, J, L, 3 4, J, L, 4 4, J, L, 5	ABSM(L,J) SIGPRO(J,1,L) SIGPRO(J,2,L) SCATM(L,J) SLODNM(L,J)	Micro absorption cross section σ_{aj}^{λ} for group $j = J$ of nuclide $\lambda = L$ Micro capture cross section $\sigma_{c(j,1,\lambda)}$ for group $j = J$ of nuclide $\lambda = L$ Micro capture cross section $\sigma_{c(j,2,\lambda)}$ for group $j = J$ of nuclide $\lambda = L$ Micro transport cross section $\sigma_{u,j}^{\lambda}$ for group $j = J$ of nuclide $\lambda = L$ Micro slowing down cross section σ_{sj}^{λ} for scattering from group $j = J$ to group $j+1$ of nuclide $\lambda = L$	(3-L) (3-L) (3-L) (3-L) (3-L)
5, J, L, 0 5, J, L, 1	GNUF(L,J) FISM(L,J) EFIS(L,J) GNUF(L,J)	The micro neutron production cross section $\nu \sigma_{fj}^{\lambda}$, the micro fission cross section σ_{fj}^{λ} , and the energy per fission E_f^{λ} for group $j = J$ of nuclide $\lambda = L$ (punched in the order given as the first three data words on the card) The micro neutron production cross section $\nu \sigma_{fj}^{\lambda}$ for neutron group $j = J$ of nuclide $\lambda = L$	(None) (3-L)
5, J, L, 2 5, J, L, 3 7, N, L, 0	FISM(L,J) EFIS(L,J) FYIELD(N,L)	The micro fission cross section σ_{fj}^{λ} for group $j = J$ of nuclide $\lambda = L$ The energy per fission E_f^{λ} for group $j = J$ of nuclide $\lambda = L$ The fission yield fraction $Y_f(n, \lambda)$ for the direct fission production of fission product nuclide $\lambda = L$ for fission yield set number $n = N$	(3-L) (3-L) (3-L)

V. OPERATIONAL PROGRAM INFORMATION AND MACHINE REQUIREMENTS

The DWARF program language is Fortran IV H. It operates on IBM 360 series machines with standard OS. It is compiled by the Fortran H compiler at optimization level 2. Two routines are installation dependent: an elapsed CPU time routine (easily replaced or dummied) and a fairly elaborate plotting routine.

All arithmetic is done in double precision, but to save core storage most basic input data is stored in single precision.

At execution time, the significant machine requirements are:

1. 300K-380K bytes of core storage (depending on the user specified I/O buffer sizes).
2. Fortran logical unit 5 (card input).
3. Fortran logical unit 6 (printed output--allow about 8K bytes per page).
4. Fortran logical unit 1 for temporary storage of nuclide concentrations (bytes of storage required is about $8 \sum_r L_r(n_r+1)$, where the sum is over the region number r , n_r is the number of mesh intervals in the region and L_r is the number of nuclides listed for the library set assigned the region).
5. Fortran logical unit 3 for temporary storage of all library set data (allow about 12K bytes per library set).
6. Fortran logical unit 4 for temporary storage of library sets used in the time range (allow 12K bytes per library set).
7. Fortran logical unit 8 for storage of all library set data--possibly saved permanently by the user (allow about 12K bytes per library set).
8. Fortran logical unit 9 for temporary storage of nuclide concentrations (storage required is same as unit 1).

The order in which DWARF performs the various major input, solution, and output functions is shown on Figure 2, where the steps carried out in each of the five overlays are outlined. The flags determining the major flow decisions are also shown on Figure 2 to enable a user with some familiarity with the coding to understand the logical flow of the programming. The logical flow of the fourth overlay (DWARF4) is shown in more detail to help the user follow the iterative loop nesting of the solution methods.

Figure 2. DWARF Program Flow
(Only Major Branches Shown)

First Page

VI. REFERENCES

1. E. C. Anderson and G. E. Putnam, CORA - A Few Group Diffusion Theory Code for One-Dimensional Reactor Analysis, IN-1416, (August 1970).
2. E. C. Anderson and G. E. Putnam, SINBAD, A Space Independent Nuclide Buildup and Depletion Code, IDO-17217, (January 1969).
3. R. L. Curtis et al, PHROG - A Fortran IV Program to Generate Fast Neutron Spectra and Average Cross Sections, IN-1435, (April 1971).
4. R. L. Curtis and R. A. Grimesey, INCITE - A Fortran IV Program to Generate Thermal Neutron Spectra and Multigroup Constants Using Arbitrary Scattering Kernals, IN-1062, (November 1967).
5. G. E. Putnam, MONA - A Multigroup One-Dimensional Neutronics Analysis Code, ANCR-1051, (March 1972).
6. A. A. Armand (translated by V. Beck), The Resistance During the Movement of a Two-Phase System in Horizontal Pipes, AERE-Trans 828, (March 1969).
7. G. E. Putnam, Some Methods of Calculation of the Parameters in the Reactor Kinetics Equations, IDO-16983, (February 1965).
8. H. P. Flatt and D. C. Baller, AIM-5, A Multigroup, One-Dimensional Diffusion Equation Code, NAA-SR-4696, (March 1960), Appendix IV by E. R. Cohen.
9. L. A. Hageman, Numerical Methods and Techniques Used in the Two-Dimensional Neutron-Diffusion Program PDQ-5, WAPD-TM-364, (February 1963).
10. J. Bookston and C. Davis, An IBM-704 Program for Criticality Search on Reactor Parameters, GMR-107, (November 1957).
11. D. K. Fadeeva and V. N. Fadeeva (translated by Robert C. Williams), Computational Methods of Linear Algebra, pp 128-130, San Francisco & London, W. H. Freeman and Company, (1963).

APPENDIX A

GLOSSARY

The major indices, subscripts or superscripts are:

- a - denoting neutron absorption
- b - denoting the buckling squared; denoting a region boundary
- d - denoting neutron diffusion coefficient
- e - denoting external source
- f - denoting neutron fission
- i - mesh index of the space variable x
- j - neutron energy group index
- k - source point index ($k = i+r-1$); index of data words input as a list or a set of vector elements
- k' - index for type of group constant modified by the functions $h_{k,k',m}$
- ℓ - Chebyshev extrapolation polynomial order
- m - macroscopic material index
- p - denoting neutron poison (absorption), or denoting pointwise
- r - region index, or symbol for total neutron removal
- s - symbol for neutron slowing down, search iteration index
- T - matrix transpose symbol
- t - fission neutron source iteration index
- u - feedback iteration index
- μ - denoting transport corrected value
- v - denoting neutron production
- x - index indicating type of group constant

To denote a maximum, an index is appended with "max"; e.g., "j_{max}".

The most commonly referred to symbols are:

- α - vapor fraction in coolant, albedo, extrapolation constant
- β - extrapolation constant
- α, β, γ - constants that determine a boundary condition by means of an equation of the form, $\alpha\phi - \beta J + \gamma = 0$
- ϵ - convergence criterion constant
- θ - two-point extrapolation constant for fission source iteration
- λ - fission source divisor: either a generation factor or unity
- λ^t - generation factor: fission neutrons produced per fission neutron introduced on the t^{th} source iteration
- $\bar{\lambda}^t$ - $(f^*t/f^{t-1})_{\text{maximum}}$
- $\underline{\lambda}^t$ - $(f^*t/f^{t-1})_{\text{minimum}}$
- $\nu\Sigma_{fj}$ - neutron production cross section, the neutrons per fission (ν) times the fission cross section for neutrons absorbed in group j
- ρ - geometry index (0-slab, 1-cylinder, 2-sphere); the distribution of a feedback variable or feedback vector; density (of coolant liquid, saturated vapor, superheated vapor, etc.)
- σ_0, σ'_0 - dominance ratio, estimate of dominance ratio
- σ_x - microscopic cross section for group constant denoted by x
- Σ_j - total macroscopic cross section of group j
- Σ_{aj} - macroscopic absorption cross section of group j
- Σ_{fj} - the fission cross section
- Σ_{sj} - macroscopic cross section for scatter transfer from group j to group $j+1$
- Σ_p - "poison" cross section
- Σ_{rj} - total macroscopic cross section for the effective absorption (removal) of neutrons, $\Sigma_{rj} = \Sigma_{aj} + T_j\Sigma_p + D_jB_j^2 + \Sigma_{sj}$
- $\Sigma_{\mu j}$ - a cross section that can be used to compute the diffusion coefficient
- ϕ - scalar neutron flux: track lengths per unit volume and per unit time

- x_j - the fraction of fission neutrons appearing in energy group j
- ω - an optional boundary condition parameter
- $a_i - D_i[(x_{i+1} + x_i)/2]^p / (x_{i+1} - x_i)$
- $a_x(n, j, m')$ - function coefficients for computing the neutron group constant denoted by the subscript x for group j of variable macroscopic material $m = m' + 40$ ($1 \leq n \leq 4$)
- A_c - fuel clad surface area per unit volume of core
- A_ℓ - loss coefficient of nuclide ℓ
- $b(n', k', m')$ - function coefficients for computing special functions $h_{k, k', m}$ that modify group constants of group $j_{\max} - 1$ of variable macroscopic material $m = m' + 40$ ($1 \leq n \leq 5$)
- B_j^2 - perpendicular buckling (squared)
- C - normalization constant multiplying output sources and fluxes
- C_n - Chebyshev polynomial of order n
- C_p - normalization constant multiplying fluxes used in feedback calculations and nuclide concentration calculations
- C_ρ - geometry constant ($1, 2\pi$, and 4π for $\rho = 0, 1$, and 2)
- C_s - search parameter modifier
- d_{ext} - zero flux extrapolation distance at a boundary surface
- D_j - diffusion coefficient
- E_j - the energy per fission for fissions caused by absorption of neutrons in group j
- f^{t+1} - fission neutron source distribution from the t^{th} iteration after renormalization and extrapolation, if any, and ready for use as the source for the $t+1^{\text{th}}$ source iteration
- f^{*t+1} - fission neutron source distribution from the t^{th} iteration after renormalization, if any
- F_0 - renormalization constant (The volume integral of the fission neutron source is set equal to F_0 on renormalization problems.)

g_x - A function of the first feedback vector ρ_1 used in computing the group constant denoted by the subscript x for a variable macroscopic material
 G_1, G_2 - lower and upper boundary condition correction factors
 $h_{k,k',m}$ - the special modifying function (of the second feedback vector) that multiplies the group constant of type k' in group $j_{\max}-1$ at source point k for variable macroscopic material m
 h - heat transfer coefficient
 H - coolant enthalpy
 H_C - fitting parameter in coolant density function
 J - $\vec{J} \cdot \vec{i}$, the x component of $\vec{J}$, where i is the unit vector in the x direction
 J^+, J^- - partial (hemispherical) currents in the positive and negative x directions, respectively
 J_i^1 - $\vec{J} \cdot \vec{i}$ at $(x_i + x_{i+1})/2$
 $\vec{J}$ - neutron current or vector flux, neutrons per unit area and per unit time
 K_1 - energy conversion factor
 N_ℓ - the concentration of nuclide number ℓ
 p - power distribution used in output edits
 $P(\ell' - \ell)$ - production coefficient for nuclide ℓ by ℓ'
 P_{av} - total power used in normalizing fluxes for the feedback or nuclide concentration calculations
 P_p - polynomial function of enthalpy for coolant unsaturated liquid density
 P_t - polynomial function of enthalpy for coolant unsaturated liquid temperature
 q_j - a constant determining internal logarithmic boundary conditions at the upper and lower boundaries of a region for group j
 q^u - the fission power distribution used in the feedback calculations
 q^1 - heat flux
 Q - boiling liquid quality

$s_i - C_p \left(\frac{x_i + x_{i+1}}{2} \right)$, the surface perpendicular to the x-axis at $\left(\frac{x_i + x_{i+1}}{2} \right)$

s_f^t - fission neutron source distribution from the t^{th} source iteration before renormalization or extrapolation, if any

S_j - effective neutron source in group j, neutrons per unit volume and per unit time

S_{ej} - total fixed volume source, $S_{rj} + S_{pj}$

S_{fj} - fission neutron source

S_{pj} - fixed volume source, input by source point and group

S_{rj} - fixed volume neutron source, input by region and group

S_{sj} - neutron scattering-in source

T_j - "poison" cross section multiplier for group j

T - temperature

v - volume, or neutron speed

$v_i^- - \frac{C_p}{\rho+1} \left[x_i^{\rho+1} - \left(\frac{x_{i-1} + x_i}{2} \right)^{\rho+1} \right]$, the volume between $(x_{i-1} + x_i)/2$ and x_i

$v_i^+ - \frac{C_p}{\rho+1} \left[\left(\frac{x_i + x_{i+1}}{2} \right)^{\rho+1} - x_i^{\rho+1} \right]$, the volume between x_i and $(x_i + x_{i+1})/2$

w_c - mass flow rate of coolant

w - region width

x - the space variable; search parameter

Y_d - nuclide decay transition field (fractional)

Y_f - fission product yield per fission

APPENDIX B

DERIVATION AND SOLUTION OF THE DIFFERENCE EQUATIONS FOR A NEUTRON GROUP

The basic neutron balance equation for a neutron group is

$$-\nabla \cdot \vec{J} - \Sigma_r \phi + S = 0, \quad (B-1)$$

where the group index j has been dropped.

The spatial mesh of $i_{\max}$ distinct points partitions the x -axis with $x_i < x_{i+1}$. An equation is derived for each mesh point by integrating Equation (B-1) over the volume assigned to the point. For this integration, Green's theorem is used to express the volume integral $\int \nabla \cdot \vec{J} dv$ as the surface integral $\oint \vec{J} \cdot \vec{n} ds$, where the unit vector $\vec{n}$ is outward and normal to the surface element ds .

At mesh point 1 the volume integration extends from x_1 to $(x_1 + x_2)/2$, at mesh point i for $1 < i < i_{\max}$ it extends from $(x_{i-1} + x_i)/2$ to $(x_i + x_{i+1})/2$, and at $i_{\max}$ it extends from $(x_{i_{\max}-1} + x_{i_{\max}})/2$ to $x_{i_{\max}}$. The equations, after integration, can be written.

$$J_1 C_p x_1^\rho - J_1' s_1 - \Sigma_{r,1}^+ v_1^+ \phi_1 + S_1^+ v_1^+ = 0 \quad (i=1), \quad (B-2)$$

$$J_{i-1}' s_{i-1} - J_i' s_i - (\Sigma_{r,i}^- v_i^- + \Sigma_{r,i}^+ v_i^+) \phi_i + (S_i^- v_i^- + S_i^+ v_i^+) = 0 \quad (1 < i < i_{\max}), \quad (B-3)$$

and

$$J_{i_{\max}-1}' s_{i_{\max}-1} - J_{i_{\max}} C_p x_{i_{\max}}^\rho - (\Sigma_{r,i_{\max}}^- v_{i_{\max}}^-) \phi_{i_{\max}} + S_{i_{\max}}^- v_{i_{\max}}^- = 0 \quad (i = i_{\max}). \quad (B-4)$$

By definition,

$$\phi_1 \equiv \int_{v_1^+} \Sigma_r \phi dv / (\Sigma_{r,1}^+ v_1^+), \quad (B-5)$$

$$\phi_i \equiv \int_{v_i^- + v_i^+} \Sigma_r \phi dv / (\Sigma_{r,i}^- v_i^- + \Sigma_{r,i}^+ v_i^+), \quad (B-6)$$

and

$$\phi_{i_{\max}} \equiv \int_{v_{i_{\max}}} \Sigma_r \phi dv / (\Sigma_{r,i_{\max}}^- v_{i_{\max}}^-). \quad (B-7)$$

The Σ_r and S values superscripted by - and + represent the appropriate averages for the lower half-interval and the upper, respectively.

To this point, no approximations or assumptions have been made except that the physical problem is one-dimensional in nature. Equations (B-2), (B-3), and (B-4) are simply the integrals of Boltzmann's transport equation over all angle and over appropriate energy and volume intervals. The final difference equations (that will be solved for the average fluxes ϕ_i) result from two assumptions.

The first assumption is that

$$J_i' = -D_i \frac{\Delta \phi_i}{\Delta x_i} = -D_i (\phi_{i+1} - \phi_i) / (x_{i+1} - x_i), \quad (B-8)$$

where D_i is the diffusion coefficient associated with the interval midpoint $(x_{i+1} + x_i)/2$.

The second assumption is that the true physical boundary conditions can be represented by a boundary condition expression of the form

$$\alpha \phi - \beta J + \gamma = 0^* \quad (B-9)$$

where ϕ and J are the values at the boundary and α , β , and γ are arbitrary constants. Hence,

$$\alpha_1 \phi_1 - \beta_1 J_1 + \gamma_1 = 0 \quad \text{and} \quad (B-10)$$

$$\alpha_2 \phi_{i_{\max}} - \beta_2 J_{i_{\max}} + \gamma_1 = 0.^{**} \quad (B-11)$$

Approximation (B-8) is applied in each of the Equations (B-2), (B-3), and (B-4) to replace each J_i . Next, Equation (B-2) is divided by C_0 and multiplied by β_1 while Equation (B-10) is multiplied by x_1^0 ; and the two resulting equations are added to eliminate the J_1 terms. Equations (B-4) and (B-11) are similarly manipulated to eliminate the $J_{i_{\max}}$ terms. The final difference equations are obtained in the form

$$- \left[-x_1^0 \alpha_1 + \beta_1 (a_1 + c_1) \right] \phi_1 + \beta_1 a_1 \phi_2 + \beta_1 S_{t1} + x_1^0 \gamma_1 = 0, \quad (B-12)$$

$$a_{i-1} \phi_{i-1} - (a_{i-1} + c_i + a_i) \phi_i + a_i \phi_{i+1} + S_{ti} = 0, \quad (1 < i < i_{\max}), \quad (B-13)$$

*At control rod region boundaries, there is allowance for special internally applied boundary conditions also of this form; namely $J = \pm qD\phi$. How the application of these special conditions affects the difference equations is explained in Section III-1.6.

**Periodic boundary conditions of the form $\phi_1 = \phi_{i_{\max}}$ and $J_1 x_1^0 = J_{i_{\max}} x_{i_{\max}}^0$ are handled in a special manner as described later.

and

$$\begin{aligned} \beta_2 a_{i_{\max}-1} \phi_{i_{\max}-1} - \left[x_{i_{\max}}^0 \alpha_2 + \beta_2 (a_{i_{\max}-1} + c_{i_{\max}}) \right] \phi_{i_{\max}} \\ + \beta_2 S_{t i_{\max}} - x_{i_{\max}}^0 \gamma_2 = 0. \end{aligned} \quad (B-14)$$

The new coefficients here are:

$$a_i = D_i s_i / [C_p (x_{i+1} - x_i)] = \frac{D_i [(x_{i+1} + x_i)/2]^p}{(x_{i+1} - x_i)}, \quad (B-15)$$

$$c_1 = \Sigma_{r1}^+ v_1^+ / C_p, \quad (B-16)$$

$$c_i = (\Sigma_{ri}^- v_i^- + \Sigma_{ri}^+ v_i^+) / C_p, \quad (B-17)$$

$$c_{i_{\max}} = \Sigma_{ri_{\max}}^- v_{i_{\max}}^- / C_p, \quad (B-18)$$

$$S_{t1} = S_1^+ v_1^+ / C_p, \quad (B-19)$$

$$S_{ti} = (S_i^- v_i^- + S_i^+ v_i^+) / C_p, \text{ and} \quad (B-20)$$

$$S_{t i_{\max}} = S_{i_{\max}}^- v_{i_{\max}}^- / C_p. \quad (B-21)$$

The set of difference equations for each neutron group pose a simple matrix problem. The matrix equation is of the form $A\bar{\phi} + \bar{S} = 0$, where the elements of the vector $\bar{\phi}$ are the ϕ_i values and A is a tri-diagonal matrix except when periodic boundary conditions are applied. The solution of the tri-diagonal matrix problem is described first and the additional operations necessitated by periodic boundary conditions are outlined later.

The matrix problem is solved directly with Gauss's method^[10]. The lower diagonal elements are first eliminated by the elementary row operations corresponding to the usual Gaussian elimination that results in an upper triangular matrix with unit diagonal elements. These operations are carried out with the following recursive formulas.

$$Z_1 = 1/[x_1^0 \alpha_1 - \beta_1 (a_1 + c_1)], \quad (B-22)$$

$$a_1' = \beta_1 a_1 Z_1, \quad (B-23)$$

and

$$\delta_1 = \beta_1 S_{t1} + x_1^p \gamma_1; \text{ then (for } i = 2, 3, \dots, i_{\max}-1), \quad (\text{B-24})$$

$$Z_i = 1/[-c_i - a_i - a_{i-1}(a_{i-1}' + 1)], \quad (\text{B-25})$$

$$a_i' = a_i Z_i, \quad (\text{B-26})$$

and

$$\delta_i = S_{ti} - a_{i-1}' \delta_{i-1}. \quad (\text{B-27})$$

Finally,

$$Z_{i_{\max}} = 1/[-x_{i_{\max}}^p \alpha_2 - \beta_2 c_{i_{\max}} - \beta_2 a_{i_{\max}-1}(a_{i_{\max}-1}' + 1)] \quad (\text{B-28})$$

and

$$\delta_{i_{\max}} = S_{ti_{\max}} - a_{i_{\max}-1}' \beta_2 \delta_{i_{\max}-1}. \quad (\text{B-29})$$

The new set of equations, ready for the backsolution for the ϕ_i values, is

$$\phi_i + a_i' \phi_{i+1} + \delta_i Z_i = 0 \quad (1 \leq i < i_{\max}) \quad (\text{B-30})$$

$$\phi_{i_{\max}} + \delta_{i_{\max}} Z_{i_{\max}} = 0 \quad (i = i_{\max}). \quad (\text{B-31})$$

Equation (B-31) is solved directly for $\phi_{i_{\max}}$ and the other fluxes are obtained recursively from

$$\phi_i = -\delta_i Z_i - a_i' \phi_{i+1}. \quad (\text{B-32})$$

When periodic boundary conditions are to be imposed, the Equations (B-10) and (B-11) are replaced by

$$\phi_1 - \phi_{i_{\max}} = 0, \text{ and} \quad (\text{B-33})$$

$$x_1^p J_1 - x_{i_{\max}}^p J_{i_{\max}} = 0. \quad (\text{B-34})$$

The new difference equations that result are

$$\phi_1 - \phi_{i_{\max}} = 0, \quad (\text{B-35})$$

$$a_{i-1} \phi_{i-1} - (a_{i-1} + c_i + a_i) \phi_i + a_i \phi_{i+1} + S_{ti} = 0, \quad (1 < i < i_{\max}), \quad (\text{B-36})$$

and

$$-(a_1+c_1)\phi_1 + a_1\phi_2 + a_{i_{\max}-1}\phi_{i_{\max}-1} - (a_{i_{\max}-1}+c_{i_{\max}})\phi_{i_{\max}} + (S_{t1}+S_{ti_{\max}}) = 0 \quad (B-37)$$

where the last equation derives from substituting values of $x_1^p J_1$ and $x_{i_{\max}}^p J_{i_{\max}}$ from Equations (B-2) and (B-4) into Equation (B-34) and then applying approximation (B-8) to replace the J_1' and $J_{i_{\max}-1}'$ terms.

The solution of these equations proceeds exactly as before with regard to calculating the Z_i , a_i , and δ_i by means of the recursion formulas shown in Equations (B-22) through (B-29) with $\alpha_1 = 1$, $\beta_2 = 1$, and $\beta_1 = \gamma_1 = \alpha_2 = \gamma_2 = 0$, as automatically set by the program. In addition, a ω_i for $1 \leq i \leq i_{\max}-1$ is computed from

$$\omega_1 = -x_1^p \text{ and } \omega_i = -a_{i-1}' \omega_{i-1} \text{ (for } i \geq 2). \quad (B-38)$$

The difference equations can then be written in the form

$$\phi_i + a_i' \phi_{i+1} + \omega_i Z_i \phi_{i_{\max}} + \delta_i Z_i = 0 \quad (1 \leq i < i_{\max}) \quad (B-39)$$

and

$$\begin{aligned} &-(a_1+c_1)\phi_1 + a_1\phi_2 + a_{i_{\max}-1}\phi_{i_{\max}-1} \\ &- (a_{i_{\max}-1}+c_{i_{\max}})\phi_{i_{\max}} + (S_{t1}+S_{ti_{\max}}) = 0. \end{aligned} \quad (B-40)$$

(Note that $a_1' = \delta_1 = 0$.)

After subtracting $a_{i_{\max}-1}$ times the Equation (B-39) for $i = i_{\max}-1$ from Equation (B-40), the last equation can be written

$$-(a_1+c_1)\phi_1 + t_2\phi_2 + u_2\phi_{i_{\max}} + y_2 = 0 \quad (B-41)$$

where

$$t_2 = a_1, \quad (B-42)$$

$$y_2 = S_{t1} - a_{i_{\max}-1}' \delta_{i_{\max}-1} + S_{ti_{\max}}, \quad (B-43)$$

$$u_2 = [(1/Z_{i_{\max}}) - a_{i_{\max}-1}' \omega_{i_{\max}-1}]. \quad (B-44)$$

Equation (B-41) can be solved for $\phi_{i_{\max}} = \phi_1$ if the ϕ_2 term can be eliminated; hence, t_2 times the equation in (B-39) for $i=2$ is subtracted from Equation (B-41) to eliminate the ϕ_2 term. This elimination generates

a ϕ_3 term that, after being similarly eliminated, leaves a ϕ_4 term, etc. The elimination process continues until the $\phi_{i_{\max}-1}$ term is eliminated and the final equation can be solved for $\phi_{i_{\max}}$. The elimination process generates the terms

$$t_i = -a_{i-1}t_{i-1} \text{ (for } 3 \leq i \leq i_{\max}\text{)}, \quad (\text{B-45})$$

$$y_i = y_{i-1} - t_{i-1}\delta_{i-1}Z_{i-1} \text{ (for } 3 \leq i \leq i_{\max}\text{)}, \quad (\text{B-46})$$

$$u_i = u_{i-1} - t_{i-1}\omega_{i-1}Z_{i-1} \text{ (for } 3 \leq i < i_{\max}\text{)}, \text{ and} \quad (\text{B-47})$$

$$u_i = u_{i-1} - t_i - t_{i-1}\omega_{i-1}Z_{i-1} \text{ (for } i = i_{\max}\text{)}. \quad (\text{B-48})$$

The value of $\phi_{i_{\max}}$ can then be computed from

$$\phi_{i_{\max}} = -y_{i_{\max}}/(u_{i_{\max}} - a_1 - c_1) \quad (\text{B-49})$$

and all the other fluxes are obtained by backsolving through Equation (B-39).