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MONA
A MULTIGROUP ONE-DIMENSIONAL NEUTRONICS
ANALYSIS CODE

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Idaho Falls, Idaho — 83401

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G. E. Putnam

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ACKNOWLEDGMENT

So many of the detailed algorithms and general methods of the MONA code were taken directly from the few group CORA code that the major contribution of E. C. Anderson in developing the latter must be gratefully acknowledged.

ABSTRACT*

The MØNA program solves the multigroup one-dimensional diffusion equations for either the direct neutron flux or the adjoint (importance) function in slab, cylindrical, or spherical geometry. Homogeneous problems can be solved for the fundamental function and eigenvalue, or extraneous sources and sinks can be specified as group dependent boundary sources and/or group and space dependent volume sources and the inhomogeneous problem solved. Very general group dependent boundary conditions can be specified or the periodic condition can be imposed in slab problems.

The usual power method (fission source iteration) is used and either two-point or Chebyshev extrapolation is optional for acceleration of convergence. All groups can be thermal (scattered into from lower groups) and optional inner iterations are allowed on them with several acceleration techniques available. Within each group, difference equations are solved simultaneously for the mesh point fluxes by Gaussian elimination of sub-diagonal elements and subsequent backsolution.

There can be automatic searches on buckling, poison, region dimensions, region boundary position, atomic concentrations, or total fission source, with the last used to obtain quick solutions to problems with extraneous sources in near critical reactors.

The diversity of output options allow edits of arbitrarily normalized group fluxes, fission source and power; flux spectra for regions and specified points, spatial distributions of median and average energies for the flux, fission rate, capture rate, and absorption rate; isotopic and material reaction rates by region; ratios of reaction rates; region dependent peak-to-average values for fluxes, source, and power; detailed neutron balance data; flux-weighted space and energy coalesced neutron group constants; and isotopic activation traverses.

The MØNA code operates on IBM 360 Series machines as an overlay program requiring about 300K bytes of core for program and input/output buffers. Variable dimensioning allows some trade-off among the mesh point, group, and region maxima, but the following upper limits also apply:

Quantity	Upper Limit
Number of Mesh Points	500
Number of Regions	50
Number of Groups	50
Number of Used (Mixed) Materials	20
Number of Library Materials	999

Running times vary from a few seconds for small problems to a few minutes with near maximum size, slowly convergent multithermal group problems.

* IDO-17255 and IN-1480 were previously assigned to this report prior to this publication.

SUMMARY

The user is provided a condensed review of what the MØNA one-dimensional, multigroup, neutron diffusion program can do in an introduction that enumerates the major features of the code.

After defining the mathematical problem posed by the set of multigroup neutron diffusion equations, the determination of each term in the neutron balance equation of each group by the program input data is detailed. A thorough description is then given of the method of solution of the problem by inner iteration on multiple thermal groups and by outer iteration on the fission neutron source. The options available for accelerating the convergence of both the outer and inner iterative processes are also elaborated.

The types of eigenvalue searches that can be applied are listed together with an outline of the mathematical algorithm used in a search. There is a complete description of how the particular group constants and parameters are modified in the course of each type of search.

The quantities output by the program 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.

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 an Appendix.

Finally, a sample problem and its solution are shown.

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MONA
A MULTIGROUP ONE-DIMENSIONAL NEUTRONICS ANALYSIS CODE

I. INTRODUCTION

Among the family of codes used in solving the larger part of the static reactor neutronics problems, the multigroup, one-dimensional (1-D) diffusion codes fill the need for efficient solution of those problems whose solutions are primarily a function of the neutron energy and only one space variable. These codes offer efficient solutions to problems where space-energy effects are not so separable as to allow cheaper few group (four or less) 1-D diffusion solutions, but where the detailed angular dependence of the flux is not so important as to require a much more expensive multigroup, 1-D transport solution by such codes as the S_n and P_n codes. At the same time, these 1-D codes run much faster than their respective 2-D counterparts so that there is a strong incentive to choose a physical model of a given reactor problem to allow a 1-D solution if at all possible.

The flexible 1-D, diffusion code, $CORA^{[1]}$, was developed at the National Reactor Testing Station (NRTS) to meet the need for a fast, up-to-date and general few group code. With that program as a basis, the MONA code was developed to meet the need for a similarly flexible multigroup, 1-D diffusion code.

The major characteristics and allowable options of the MONA code are:

1. Slab, cylindrical, or spherical geometry is allowed.
2. General linear boundary conditions can be specified, allowing separate specification of an equation of the form $\alpha\phi - \beta J + \gamma = 0$ for each group and boundary. A periodic condition that equates boundary fluxes and leakages is also allowed.
3. Real (direct) or adjoint (importance function) flux solutions are obtained.
4. Calculation of a generation factor is possible (with a renormalization type of fission source iteration) even in the presence of fixed boundary or volume sources.
5. General space and energy dependent fixed volume sources can be specified with separate spatial distributions in each group and, as implied above, group dependent boundary sources can be input.
6. Efficient solution (by the search method) is possible on extraneous source problems even when the eigenvalue is near unity.

7. An effective Chebyshev polynomial extrapolation technique for accelerating the outer (fission source) iterations is available, and in multithermal group problems inner iterations on the thermal groups can be performed with several acceleration techniques available.
8. Full up and down scattering is allowed.
9. The input data handling allows complete "change case" capability (subsequent problems require entry of only the changed data), strict ordering of the data is not required, and zero elements of a data block generally need not be entered.
10. There are automatic searches (to a user specified eigenvalue) on perpendicular buckling, material poison, region dimensions, nuclide concentrations, and the boundary position (continuously variable) between any two regions. If required, a search can be restricted to specified regions or materials.
11. There are a diversity of output options for processing and editing the solution data in forms useful for reactor analysis. Also, all the problem input and solution data can be stored for later processing by other codes for such things as perturbation theory purposes or graphical presentation. Auxiliary routines to process the solution data can be easily added as additional overlays.
12. The input formats for the neutron group constants are compatible in general with the CØRA code and the SCAMP^[2] 1-D, S_n code and the cross section codes PHROG^[3] and INCITE^[4] can be used to output the data in directly usable form.

II. PROBLEM DEFINITION AND SOLUTION METHODS

1. NEUTRON GROUP EQUATIONS AND BASIC DATA

1.1 The Multigroup Diffusion Equations

If the time independent form of the Boltzmann transport equation* is integrated first over all directions and then over the energy range of neutron group j , the basic neutron balance equation for the j^{th} group results:

---

*For example, see Equation (9.2) in Reference 5.

$$\nabla \cdot \bar{J}_j + \Sigma_{rj} \phi_j = S_j, \quad (1)$$

where Σ_{rj} , the total cross section for removal of neutrons from group j , is the sum

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

By convention, the energies of the neutron groups decrease as the group index j increases.

By assuming the diffusion theory approximation, the vector current $\bar{J}_j$ is expressed as

$$\bar{J}_j = -D_j \nabla \phi_j, \quad (3)$$

(where D_j is called the group j diffusion coefficient). Equation (1) can be written as the diffusion equation,

$$-\nabla \cdot D_j \nabla \phi_j + \Sigma_{rj} \phi_j = S_j \quad (4)$$

These neutron group equations are coupled through the group-to-group scattering and the fission source terms in S_j so that it is usual to express the multigroup diffusion theory problem in terms of the matrix equation

$$L \cdot \bar{\phi} = \frac{\bar{M} \cdot \bar{\phi}}{\lambda} + \bar{S}_e \quad (5)$$

In one dimension (with α , β , and γ values at the lower boundary of x denoted by the subscript 1 and those at the upper boundary by 2), the problem definition is complete if we specify, in addition to Equation (5), either the boundary condition equations,

$$\bar{\alpha}_1 \bar{\phi}_1 - \bar{\beta}_1 \bar{J}_1 + \bar{\gamma}_1 = 0 \text{ and } \bar{\alpha}_2 \bar{\phi}_{i_{\max}} - \bar{\beta}_2 \bar{J}_{i_{\max}} + \bar{\gamma}_2 = 0, \quad (6)$$

or the special periodic boundary condition equations,

$$\bar{\phi}_1 = \bar{\phi}_{i_{\max}} \text{ and } x_1^0 \bar{J}_1 = x_2^0 \bar{J}_{i_{\max}}, \quad (7)$$

where the subscripts 1 and $i_{\max}$ on $\bar{\phi}$ and $\bar{J}$ denote the values at mesh point $i=1$ and $i=i_{\max}$, respectively.

*See the Glossary, Appendix A for the definitions of all the neutron group constants.

MØNA attempts to find the solution to this problem either with λ set to unity (in most inhomogeneous problems) or with λ computed from the formulas

$$\lambda = \int_V \bar{F} \cdot \bar{\phi} / F_0 \text{ for a real flux problem and}$$

$$\lambda = \int_V \bar{X}^T \cdot \bar{\phi} / F_0 \text{ for an adjoint problem.} \quad (8)$$

When λ is the latter, the problem is called a "renormalization" type of problem.

In Equations (5) through (8), the following definitions apply.

$$\bar{L} \equiv \begin{vmatrix} (-\nabla \cdot D_1 \nabla + \Sigma_{r1}) & \Sigma_{2 \rightarrow 1} & \Sigma_{3 \rightarrow 1} & . & . & . & . & \Sigma_{j_{\max} \rightarrow 1} \\ \Sigma_{1 \rightarrow 2} & (-\nabla \cdot D_2 \nabla + \Sigma_{r2}) & \Sigma_{3 \rightarrow 2} & . & . & . & . & \Sigma_{j_{\max} \rightarrow 2} \\ \Sigma_{1 \rightarrow 3} & \Sigma_{2 \rightarrow 3} & (-\nabla \cdot D_3 \nabla + \Sigma_{r3}) & . & . & . & . & \Sigma_{j_{\max} \rightarrow 3} \\ . & . & . & . & . & . & . & . \\ . & . & . & . & . & . & . & . \\ \Sigma_{1 \rightarrow j_{\max}} & \Sigma_{2 \rightarrow j_{\max}} & \Sigma_{3 \rightarrow j_{\max}} & . & . & . & . & (-\nabla \cdot D_{j_{\max}} + \Sigma_{rj_{\max}}) \end{vmatrix} \quad (9)$$

$$\bar{\phi} \equiv \begin{vmatrix} \phi_1 \\ \phi_2 \\ \phi_3 \\ . \\ . \\ . \\ \phi_{j_{\max}} \end{vmatrix}, \quad \bar{J} \equiv \begin{vmatrix} J_1 \\ J_2 \\ J_3 \\ . \\ . \\ . \\ J_{j_{\max}} \end{vmatrix}, \quad \bar{S}_e \equiv \begin{vmatrix} S_{e1} \\ S_{e2} \\ S_{e3} \\ . \\ . \\ . \\ S_{ej_{\max}} \end{vmatrix}, \quad (10)$$

where $J_j = \bar{f} \cdot \bar{J}_j$.

Also, $\bar{M} \equiv \bar{X} \cdot \bar{F}$,

(11)

$$\text{where } \bar{X} \equiv \begin{vmatrix} x_1 \\ x_2 \\ x_3 \\ . \\ . \\ . \\ x_{j_{\max}} \end{vmatrix}; \quad \bar{F} = (v \Sigma_{f1} \ v \Sigma_{f2} \ v \Sigma_{f3} \ . \ . \ . \ v \Sigma_{fj_{\max}}). \quad (12)$$

$$\bar{\alpha} \equiv \begin{vmatrix} \alpha_1 & 0 & \dots & 0 \\ 0 & \alpha_2 & \dots & 0 \\ \cdot & \cdot & \cdot & \cdot \\ \cdot & \cdot & \cdot & \cdot \\ 0 & 0 & \dots & \alpha_{j_{\max}} \end{vmatrix} \quad \bar{\beta} \equiv \begin{vmatrix} \beta_1 & 0 & \dots & 0 \\ 0 & \beta_2 & \dots & 0 \\ \cdot & \cdot & \cdot & \cdot \\ \cdot & \cdot & \cdot & \cdot \\ 0 & 0 & \dots & \beta_{j_{\max}} \end{vmatrix}, \text{ and } \bar{\gamma} \equiv \begin{vmatrix} \gamma_1 \\ \gamma_2 \\ \cdot \\ \cdot \\ \gamma_{j_{\max}} \end{vmatrix} \quad (13)$$

For adjoint problems, the identical group constants and boundary conditions are used, but instead of Equation (5), the matrix equation solved for the adjoint flux (importance function) is

$$\bar{L}^T \cdot \bar{\phi} = \frac{\bar{M}^T}{\lambda} \cdot \bar{\phi} + \bar{S}_e, \quad (14)$$

where $\bar{L}^T$ and $\bar{M}^T$ are the transposes of $\bar{L}$ and $\bar{M}$; hence,

$$\bar{M}^T = (\bar{X} \cdot \bar{F})^T = \bar{F}^T \cdot \bar{X}^T. \quad (15)$$

1.2 The Neutron Group Constants

In a MØNA problem, the range $\{x_1, x_{i_{\max}}\}$ is divided into a number of regions. Within a region, the x-mesh interval size and the values of the neutron group constants (D_j , Σ_{rj} , Σ_{aj} , $\Sigma_{j \rightarrow j'}$, T_j , Σ_p , etc.) are fixed. Hence, a region is homogeneous (no spatial variation of physical properties) and consists of only one macroscopic (macro) material; however, the user may choose to assign several regions the same macro material. It is common to add regions solely to refine or coarsen the spatial mesh.

A macro material is characterized by its set of neutron group constants. These neutron group constants for a group j consist of the set D_j , Σ_{aj} , $\Sigma_{j \rightarrow j'}$ (for each j'), T_j , Σ_p , B_j^2 , $\nu \Sigma_{fj}$, χ_j , $\Sigma_{\mu j}$, Σ_{fj} , E_j , and q_j^* . (The constant $\Sigma_{\mu j}$ is used only in computing D_j ; and the fission cross section and energy per fission Σ_{fj} and E_j are used only to compute the power distribution at output edit time.) Of these group constants, T_j , Σ_p , B_j^2 , χ_j , and q_j are always input directly for each macro material m ; i.e., they are not computed from other input data (although Σ_p and B_j^2 are changed by a factor, of course, in poison and buckling

---

*In special materials, the internal logarithmic boundary condition parameter q_j is 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 solution of the equations.

searches). The group constant D_j is a special case and can also be input directly for macro material m , but it is usually computed from the mixed (averaged) data of one or more library materials - as is always required for the group constants Σ_{aj} , $\Sigma_{j \rightarrow j'}$, $\nu \Sigma_{fj}$, $\Sigma_{\mu j}$, Σ_{fj} , and E_j . With the exception of E_j , the latter group constants are computed from an equation of the form (for macro material m),

$$\Sigma^m = \sum_k \Sigma^{\ell_k^m}_k C^m_k, \quad (16)$$

where C^m_k is the k^{th} element in an input list of concentrations (weight factors) for material m . The $\Sigma^{\ell_k^m}_k$ values in Equation (16) are determined as follows. In addition to the list of concentrations C^m_k the user also inputs a corresponding list of integers ℓ_k^m ($k = 1, 2, 3$, etc.) for macro material m . If an element ℓ_k^m of the list is positive, then $\Sigma^{\ell_k^m}_k$ is a cross section of library material $\ell = \ell_k^m$ and the data are read from a stored library data set. If an element ℓ_k^m is negative, then the $\Sigma^{\ell_k^m}_k$ is a cross section of macro material $m' = \ell_k^m$ as mixed and used in the preceding MONA problem (hence, this option cannot be used on the first problem in a series).

The average energy E^m for a neutron group of macro material m is computed from

$$E^m = \sum_k E^{\ell_k^m}_k \Sigma^{\ell_k^m}_k C^m_k / \sum_k \Sigma^{\ell_k^m}_k C^m_k = \sum_k E^{\ell_k^m}_k \Sigma^{\ell_k^m}_k C^m_k / \Sigma^m_f. \quad (17)$$

The diffusion coefficient D_j for a macro material is computed from one of three different formulas at the user's option. D_j is computed from

$$D_j = \frac{1}{3\Sigma_{\mu j}}, \quad (18)$$

$$D_j = 1/[3(\Sigma_{\mu j} + T_j \Sigma_p)], \text{ or} \quad (19)$$

$$D_j = \Sigma_{rj}^1 / \kappa_j^2, \quad (20)$$

where κ_j is the solution of the transcendental equation

$$\kappa_j / \Sigma_j = \tanh [\kappa_j / (\Sigma_j - \Sigma_{rj}^1)], \quad (21)$$

with $\Sigma_j = \Sigma_{\mu j} + T_j \Sigma_p$ and $\Sigma_{rj}^1 = \Sigma_{rj} - D_j B_j^2$.

The user can require that Equation (19) or (20) be used in certain specified materials and Equation (18) be used elsewhere. The Σ_{uj} in these expressions is either simply $1/3D_j$, where D_j is directly input for macro material m or it is mixed according to Equation (16) using library material data and/or macro material data from the preceding problem.

As implied, a data set consisting of the group constant set Σ_{aj} , $\Sigma_{j \rightarrow j'}$, $\nu\Sigma_{fj}$, Σ_{uj} , Σ_{fj} , E_j , and D_j for each of one or more library materials must exist to allow MØNA to compute the necessary macro material group constants. This library data set consists of the group constants of from 1 to 999 library materials. It can be generated and updated during any MØNA problem. Each library material is indexed by a unique integer index ℓ , and, while the library materials must be stored on the data set in the order of ascending ℓ values, the ℓ values need not be consecutive.

1.3 The Fixed Volume Source for a Group

The elements of the fixed source vector $\bar{S}_e$ in Equation (5) are the group fixed volume sources S_{ej} . For a MØNA 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

$$S_{pj} = x_j \phi_j, \quad (22)$$

where x_j is any one of a number of 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

$$S_{ej} = S_{rj} + S_{pj}. \quad (23)$$

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.

---

*The term "source point" refers to an indexing scheme that allows for double values at region interfaces. The index is $k = i+r-1$ where i is the spatial mesh index and r is the region index; hence, the number of source points is $k_{\max} = i_{\max} + r_{\max} - 1$. This allows for the existence of different values of a source that exist just below and above a mesh point i that is an interface between two regions of dissimilar materials.

1.4 The Scattered Source for a Group

The neutron scattering-in source, stemming from the off-diagonal terms of $\bar{\Sigma}$ [see Equations (5) and (9)] consists for a given group j of the sum

$$S_{sj} = \sum_{j' \neq j} \Sigma_{j' \rightarrow j} \phi_{j'} \text{ for a real flux problem, and} \quad (24)$$

$$S_{sj} = \sum_{j' \neq j} \Sigma_{j \rightarrow j'} \phi_{j'} \text{ for an adjoint problem.} \quad (25)$$

On a given outer (fission source) iteration, the scattering-in source for group j (of a real flux problem) is actually computed as

$$S_{sj} = \sum_{j' < j} \Sigma_{j' \rightarrow j} \phi_{j'}^u + \sum_{j' > j} \Sigma_{j' \rightarrow j} \phi_{j'}^{u-1}, \quad (26)$$

where u is the "inner iteration" index. That is, to obtain the solution to the flux given the fission and other source distributions of a particular outer iteration, a Gauss-Seidel sweep through the group fluxes from group 1 to $j_{\max}$ is repeated as an inner iterative process.

The inner iterations are required because of the use of old (last iterate) fluxes to multiply the up-scatter cross sections in the last term of Equation (26). In practice, of course, the inner iteration sweeps are repeated only from the uppermost (in energy) thermal group to $j_{\max}$, where this highest thermal group is the highest in energy (lowest j) into which neutrons can be scattered from the lower groups. (Hence, it is the group of lowest j that has some $\Sigma_{j' \rightarrow j} \neq 0$ for $j' > j$.) Also, of course, no inner iteration is necessary when there is no up-scatter; i.e., only one thermal group.

For an adjoint problem, the sweep through the groups on both outer and inner iterations is from group $j_{\max}$ to 1, and inner iterations are repeated only for the thermal groups as defined above. In the inner iterative process, the source is computed from

$$S_{sj} = \sum_{j' > j} \Sigma_{j \rightarrow j'} \phi_{j'}^u + \sum_{j' < j} \Sigma_{j \rightarrow j'} \phi_{j'}^{u-1} \quad (27)$$

for an adjoint group j .

This inner iterative process is also accelerated by various means described in Section 2.4.

1.5 The Fission Source for a Group

The neutron source from fissions is represented by the terms $\bar{M} \cdot \bar{\phi} / \lambda$ and $\bar{M}^T \cdot \bar{\phi} / \lambda$ for real and adjoint problems, respectively. Since $\bar{M} = \bar{X} \cdot \bar{F}$ and $\bar{M}^T = \bar{F}^T \cdot \bar{X}^T$, the real and adjoint source terms are $\bar{X} \cdot \frac{\bar{F} \cdot \bar{\phi}}{\lambda}$ and $\bar{F}^T \cdot \frac{\bar{X}^T \cdot \bar{\phi}}{\lambda}$. Equation (12) can be used to enable the fission neutron source for a group j to be written

$$S_{fj} = x_j \frac{\bar{F} \cdot \bar{\phi}}{\lambda} = x_j \left[\sum_{j'} v \Sigma_{fj'} \phi_{j'} \right] / \lambda \text{ for a real flux problem, and} \quad (28)$$

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

and it should be recalled that either $\lambda=1$ or (from Equation (8) for re-normalization types of problems)

$$\lambda = \left[\int dv \sum_j v \Sigma_{fj} \phi_j \right] / F_0 \text{ for real flux problems, and} \quad (29)$$

$$\lambda = \left[\int dv \sum_j x_j \phi_j \right] / F_0 \text{ for adjoint problems.}$$

For later reference, Equations (29) obviously impose the constraint

$$\int \frac{1}{\lambda} \sum_j v \Sigma_{fj} \phi_j dv = F_0 \text{ for a real flux problem or} \quad (30)$$

$$\int \frac{1}{\lambda} \sum_j x_j \phi_j dv = F_0 \text{ for an adjoint problem.}$$

This constraint allows a definition of λ even in inhomogeneous (extraneous source and sink) types of problems, as will be discussed in Sections 2.1 and 2.2. 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 (5) is obtained by MØNA with the right hand side of Equation (5) known for each outer iteration. That is, MØNA really solves, iteratively, the problem

$$\bar{L} \cdot \bar{\phi}^t = \frac{(\bar{M} \cdot \bar{\phi}^{t-1})}{\lambda^{t-1}} + \bar{S}_e, \quad (31)$$

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

$$\lambda^{t-1} = [\int_V dv \sum_j v \Sigma_{fj} \phi_j^{t-1}] / F_0, \quad (32)$$

and the prime on the quantity $\bar{M} \cdot \phi^{t-1}$ denotes an extrapolated form.

1.6 The Boundary Conditions For A Group

As defined in Equations (6) and (7) and the Definitions (10), the boundary conditions for neutron group j are

$$\alpha_{1j} \phi_{j1} - \beta_{1j} J_{j1} + \gamma_{1j} = 0 \text{ at the inner (lower } x) \text{ boundary, and} \quad (33)$$

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

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

$$\phi_{j1} = \phi_{j\max}, \text{ and} \quad (34)$$

$$x_1^0 J_{j1} = x_{\max}^0 J_{j\max}.$$

(In these equations $J_{ji} \equiv \bar{i} \cdot \bar{J}_{ji}$, or, according to the diffusion theory approximation of Equation (3),

$$J_{ji} = \bar{i} \cdot \bar{J}_{ji} = -\bar{i} D_{ji} \nabla \phi_{ji} = -D_j \frac{d\phi_{ji}}{dx}.) \quad (35)$$

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[7].

Table 1. 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	$\frac{-1}{\phi_{j1\max}} \frac{d\phi_{j1\max}}{dx} = \frac{1}{a3D_jG_{2j}}$ *	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	$x_{1j}^0 J_{j1} = x_{1\max}^0 J_{j1\max}$ (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, $a = 0.710446$, $G_{1j} = [1.0 + 1.5207587 (3D_{j0}/2x_1)] / [1.0 + 0.8103127 (3D_{j0}/2x_1)]$ and $G_{2j} = 1.0 [1.0 + a (3D_{j0}/2x_{1\max})]$.

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

$$\frac{1}{\phi_{jib}} \frac{d\phi_{jib}}{dx} = \pm q_j^m \quad (36)$$

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 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} , $v\Sigma_{fj}$, 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

$$\frac{1}{\phi_{jib}} \frac{d\phi_{jib}}{dx} = -q_j^m \text{ (or, } -J_{jib} + D_{jib} q_j^m \phi_{jib} = 0), \quad (37)$$

*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

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

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 $-\phi_j + St_j = 0$, where St_j 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 SOLUTION METHODS

To solve the matrix Equations (5) or (14) subject to the boundary conditions (6), MØNA performs outer iterations with the fission neutron source for the second and higher iterations computed with fluxes from the previous iteration. When there are multiple thermal groups (scattering from lower energy groups to higher ones), inner iterations within each outer iteration are required to converge the flux solutions for a given outer iteration, as described in Section 1.4. By adopting these inner and outer iteration methods of solution, the problem is reduced to solving Equation (1) for each group in turn with the source term S_j a known quantity in terms of the contributions of S_{ej} , S_{sj} , and S_{fj} , the fixed volume, scattering-in, and fission sources, respectively. That is,

$$\nabla \cdot \bar{J}_j - \Sigma_{rj} \phi_j = S_{ej} + S_{sj} + S_{fj}. \quad (38)$$

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.

The sections to follow describe the outer and inner iteration processes exactly and explain the techniques at the disposal of the user for accelerating the convergence of the solutions.

2.1 Solution by Fission Source (Outer) Iteration

As implied in Section 1.5, the fission neutron source for group j in the t th outer iteration can be written $S_{fj} = x_j f^t$ or $S_{fj} = v \Sigma_{fj} f^t$ 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 the MØNA problem), 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 INSØRS values of $(\pm)1, 2, 3, 4$, and 5 , 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 (FISSØR) 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 (30).

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

$$s_f^t = \sum_j v \Sigma_{fj} \phi_j^t \text{ for a real flux problem and} \quad (39)$$

$$s_f^t = \sum_j x_j \phi_j^t \text{ for an adjoint problem.}$$

Next, a generation factor λ^t is computed from

$$\lambda^t = \int_V s_f^t dv / \int_V f^t dv. \quad (40)$$

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 all $\int_V f^t dv = F_0$, then an auxiliary distribution f^{*t+1} is computed from

$$f^{*t+1} = s_f^t / \lambda^t. \quad (41)$$

Otherwise, f^{*t+1} is simply set equal to s_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 (40) will be equal to $\int_V s_f^t dv / F_0$ in accordance with Equation (8), (29), and (32). Hence, the source f^{*t+1} will also now satisfy the fission source constraint of Equations (30); that is,

$$\int_V f^{*t+1} dv = \frac{1}{\lambda^t} \int_V s_f^t dv = \frac{F_0}{\int_V s_f^t dv} \int_V s_f^t dv = F_0. \quad (42)$$

When the problem is not a renormalization type and $f^{*t+1} = s_f^t$, then this is equivalent to the option corresponding to setting $\lambda=1$ in Equation (5) or (14) as described in Section 1.1.

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

$$f^{t+1} = f^t + \alpha(f^{*t+1} - f^t) + \beta(f^t - f^{t-1}), \quad (43)$$

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 (43) 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 NC0NVG equal to zero and iteration stops when

$$\left| \frac{\lambda^t - \lambda^{t-1}}{\lambda^t} \right| \leq \epsilon_1. \quad (44)$$

For NC0NVG = 1, pointwise convergence of the fission source is specified and the code computes

$$\bar{\lambda}^t = \left| \frac{f^{*t}}{f^{t-1}} \right|_{\text{maximum}} \quad \text{and} \quad \underline{\lambda}^t = \left| \frac{f^{*t}}{f^{t-1}} \right|_{\text{minimum}}. \quad (45)$$

The iteration is terminated when

$$\bar{\lambda}^t - \underline{\lambda}^{t-1} \leq \epsilon_1 \quad (46)$$

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

$$\left| \frac{1}{\lambda^t} - 1 \right| \text{ and } \left| \underline{\lambda}^t - 1 \right| \quad (47)$$

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 (41) 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

$$\int_V f^t dv = F_0 \quad (1 \leq t) \quad (48)$$

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 (48) 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.)

But one of the most notable options available to the user in a MONA solution is the option to call for fission source renormalization, i.e., the imposition of the constraint (48), even on inhomogeneous problems*. When the user calls for the renormalization type of solution (IREN=0) for an inhomogeneous problem, MONA performs the iterative solution for the neutron generation factor λ in the same manner as for the homogeneous case by means of the constraint (48) 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 (48) 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

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*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 (6) and (7), and/or by any nonzero values in the fixed volume sources, S_{ej} .

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 (48). (For example, see Equation (84), page 28, in Reference 6.)

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 s_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, s_f^t is the $t+1^{\text{th}}$ generation in both magnitude and distribution. And the ratio of the volume integrals

$$\lambda^t = \int_V s_f^t dv / \int_V f^t dv \quad (49)$$

is the total number of fission neutrons produced per fission neutron introduced in the f^t distribution. Hence, in the edits of MØNA 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 fission neutron introduced.) It is clear that if there is no extrapolation of the fission neutron sources in the course of a MØNA solution (in which case f^{t+1} is distributed exactly as s_f^t), 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 6.) Note: multiple thermal group solutions must also be converged.

2.3 Acceleration of the Outer Iteration Convergence

The Chebyshev extrapolation algorithm employed in the MØNA program is the one described in the very complete and comprehensive report by L. A. Hageman[8]. The only differences are that the initial estimate σ_0 of the dominance ratio σ^* is computed in MØNA from

*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.

$$\sigma_0 = \left| \bar{\Lambda}^{t'} - \underline{\Lambda}^{t'} \right| / \left| \bar{\Lambda}^{t'-1} - \underline{\Lambda}^{t'-1} \right| \quad (50)$$

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

$$\sigma_0 = \left| \lambda^{t'} - 1 \right| / \left| \lambda^{t'-1} - 1 \right| \quad (51)$$

when IREN=1 (and the problem is a non-renormalization type), provided $0 < \sigma_0 \leq 1$. 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 NPI0, in which case t' is set to NPI0+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 (43) with $\alpha_1 = 2/(2-\sigma_0)$ and $\beta_1 = 0$. Thereafter, the extrapolation parameters are computed from

$$\alpha_\ell = \frac{4}{\sigma_0} \left\{ \frac{\cosh [(\ell-1)\gamma]}{\cosh [\ell\gamma]} \right\} \quad (52)$$

and

$$\beta_\ell = \cosh [(\ell-2)\gamma] / \cosh [\ell\gamma], \quad (\ell \geq 2) \quad (53)$$

where

$$\gamma = \cosh^{-1}[(2-\sigma_0)/\sigma_0] = \ln \left[\left(\frac{2-\sigma_0}{\sigma_0} \right) + \sqrt{\left(\frac{2-\sigma_0}{\sigma_0} \right)^2 - 1} \right] \quad (54)$$

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

$$ER_\ell = (\bar{\Lambda}^\ell - \underline{\Lambda}^\ell) / (\bar{\Lambda}^2 - \underline{\Lambda}^2) \text{ when IREN} = 0 \quad (55)$$

or

$$ER_\ell = [(\bar{\Lambda}^\ell + \underline{\Lambda}^\ell)/2 - 1] / [(\bar{\Lambda}^2 + \underline{\Lambda}^2)/2 - 1] \text{ when IREN} = 1. \quad (56)$$

[The $\bar{\Lambda}^\ell$ and Λ^ℓ are the maximum and minimum values as defined by Equations (46) and (47) but 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

$$C_n(x) = 2x C_{n-1}(x) - C_{n-2}(x) \text{ for } n \geq 2, \quad (57)$$

$$C_1(x) = x \text{ and } C_0(x) = 1. \quad (58)$$

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

$$\sigma'_0 = \frac{\sigma_0}{2} \cosh \left\{ \frac{\ln(x + \sqrt{x^2-1})}{\ell-2} + 1 \right\} \quad (59)$$

where

$$x = ER_\ell C_{\ell-2}[(2-\sigma_0)/\sigma_0]. \quad (60)$$

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 (50) or (51) satisfies the condition

$$0 < \sigma_0 \leq 1. \quad (61)$$

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

$$f^{t+1} = f^{*t+1} + \theta(f^{*t+1} - f^t) \quad (62)$$

which is the same as Equation (43) with $\alpha = 1 + \theta$ and $\beta = 0$. The user should therefore choose θ less than or equal to one to avoid divergence.

2.4 Acceleration of the Multithermal Group (Inner Iteration) Convergence

In each iteration of the outer (fission source) iteration process, the neutron source magnitude and distribution from fissions and external sources, combined, are fixed. Therefore, if it were not for neutron up-scatter in the thermal groups, a single Gauss-Seidel sweep through the groups, as described in Section 1.4, would result in the final, correct flux solution corresponding to the assumed source magnitude and distribution of that outer iteration. As seen from Equations (26) and (27), if there were no scattering cross section $\Sigma_{j \rightarrow j'}$ such that $j' < j$, then the part of the source represented by neutrons scattered in to a group from the other groups does not depend on the old fluxes computed on the previous sweep through the groups. Hence, the final solution for the outer iteration is obtained in only one sweep--for either the real or the adjoint case.

When there is upscatter (some $\Sigma_{j \rightarrow j'} \neq 0$ for $j' < j$), the final solution for a given outer iteration can be obtained by the MØNA program only by repeated Gauss-Seidel sweeps through the groups; i.e., inner iterations. These inner iterations, indexed by u , can be continued until the flux solution converges or until $u = \text{MAXINS}$, an input integer.

The convergence of the inner iterative process is monitored in terms of a computed "source-to-loss" ratio R^u . This ratio R^u is defined as the entire neutron source to the thermal groups divided by the entire neutron loss in the thermal groups when the loss is computed in terms of the fluxes computed on the u^{th} inner iteration. This ratio defined in detail below, approaches unity as the inner iterations converge.

On a given outer iteration, the inner iterations are terminated when $|R^u - 1| \leq \epsilon_4$ or when $u = \text{MAXINS}$, where ϵ_4 is EPS4, an input value.

To help achieve a reasonable rate of convergence of the inner iterations, there are several options available in the MØNA program.

- a. Unmodified Gauss-Seidel iteration: With this option, the inner iterative process described above is unmodified; i.e., there is no action taken to accelerate the convergence rate. There are cases in which very near the maximum possible rate of convergence is obtained with this option. Although almost all real flux problems benefit considerably from some acceleration, this option should always be tried first for adjoint problems.
- b. Source-to-loss ratio multiplication: Inner iteration convergence is accelerated by multiplying the thermal group fluxes ϕ^u from the u^{th} inner iteration by the total source-to-loss ratio R^u before using the fluxes for iteration $u+1$. The basis of this method is described later in the section.
- c. Source-to-loss ratio multiplication conditional on convergence level: Understandably, the acceleration benefit from multiplying the thermal fluxes by R^u becomes small as R^u approaches unity.

Moreover, there is some unknown round-off error in R^u since it is the ratio of two sums of large numbers of quantities. To free the final flux and eigenvalue numbers from any round-off error from this source, MØNA allows the option to have this acceleration technique terminated when $|R^u - 1| \leq \epsilon_5$, where ϵ_5 is EPS5 an input value. It is usually good policy to use this option when source-to-loss ratio multiplication is used. An ϵ_5 value of about 0.01 is reasonable.

- d. Groupwise source-to-loss multiplication: Just as it is possible to find a total source-to-loss ratio R^u which, after multiplying the thermal fluxes, forces the total thermal group losses to balance the total source, it is possible to find a set of factors r_j^u , one for each thermal group, that forces a groupwise balance of sources and losses when each thermal ϕ_j^u is multiplied by r_j^u . While requiring extra computer time, this method has an advantage over multiplication of all groups by a single R^u factor when the degree of convergence varies greatly from group to group. The method for computing the r_j^u values is derived later in this section.
- e. Two-point, fixed factor extrapolation: From previous experience with a class of problem, one can often arrive at a reasonable estimate of an extrapolation factor τ , such that when the thermal fluxes to be used on iteration $u+1$ are computed from

$$\phi_j^u = \phi_j^{*u} + \tau(\phi_j^{*u} - \phi_j^{u-1}) \quad (63)$$

a nearly optimum rate of convergence is obtained. This extrapolation is applied at each spatial mesh point for each group, of course. This type of extrapolation is recommended where solid convergence is desired at all spatial points, including those unimportant in determining the thermal group sources and losses. Obviously, if a source-to-loss factor does not depend on the fluxes in certain relatively uncoupled or low-flux regions, it cannot be in general a relevant scaling factor for accelerating convergence in those regions. The factor τ is read in as THRMX to MØNA.

- f. Two-point extrapolation with automatic factor calculation: MØNA provides the option to have the two-point flux extrapolation factor τ computed automatically after an initial guess for τ is read in as THRMX. (A guess of about 0.4 for τ is reasonable.) A simple-minded algorithm updates the τ^u value after each inner iteration and generally results in good, but far from optimum, rates of convergence. (The τ^u values obtained by the algorithm are usually too low.) The algorithm monitors the convergence of the total source to loss ratio R^u and if all is going well [i.e., $R^u > 0$, $|R^u - 1| < |R^{u-1} - 1|$, and $(R^u - 1)/(R^{u-1} - 1) > 0$], a new value of τ is computed from

$$\tau^u = 0.2 + 0.8\tau^{u-1}. \quad (64)$$

If the tests fail the new τ is decreased by 10%; i.e.,

$$\tau^u = 0.9\tau^{u-1}. \quad (65)$$

These τ^u values are used to multiply the fluxes obtained on the u^{th} iteration as in Equation (63).

To derive the total and groupwise source-to-loss ratios R^u and r_j^u , one can start with the neutron balance equation for a thermal group written as follows (for a real flux solution).

$$\nabla \cdot \bar{J}_j + \Sigma_{rj} \phi_j - \sum_{j' \geq j}^{\text{th}} \Sigma_{j' \rightarrow j} \phi_{j'} = \sum_{j' < j}^{\text{th}} \Sigma_{j' \rightarrow j} \phi_{j'} + S_{fj} + S_{ej} \quad (66)$$

where j^{th} is the index of first (highest energy) thermal group and the scattering-in source has been split, with the part that does not change from one inner iteration to the next placed on the right hand side of Equation (66) with the other nonchanging sources. This nonchanging source represented on the right hand side of Equation (66) will henceforth be denoted by s_j^0 .

Next, the conventions and assumptions used in deriving the spatial difference equations for a group are used to derive an expression for the volume integral (divided by C_p) from x_1 to x_I (where $I \equiv i_{\text{max}}$) of Equation (66). After application of the boundary conditions, both external and internal, this expression can be written

$$\begin{aligned} & \left\{ [\beta_{1j}(a_{j1} + c_{j1}) - a_{j1} - x_1^0 \alpha_{1j}]^* \phi_{j1}^u + (c_{j2} + a_{j1} - \beta_{1j} a_{j1}) \phi_{j2}^u + \sum_{i=3}^{I-2} c_{ji} \phi_{ji}^u \right. \\ & + \sum_{i_b} D_j q_{ji_b} x_{i_b}^0 \phi_{ji_b} \\ & + (c_{jI-1} + a_{jI-1} - \beta_{2j} a_{jI-1}) \phi_{jI-1}^u + [\beta_{2j}(a_{jI-1} + c_{jI}) - a_{jI-1} + x_I^0 \alpha_{2j}]^* \phi_{jI}^u \Big\} r_j \\ & - \sum_{j' \geq j}^{\text{th}} \frac{1}{C_p} [\beta_{1j} \Sigma_{ij' \rightarrow j}^+ v_1^+ \phi_{j'1}^u + \sum_{i=2}^{I-1} (\Sigma_{ij' \rightarrow j}^- v_i^- + \Sigma_{ij' \rightarrow j}^+ v_i^+) \phi_{ji}^u + \beta_{2j} \Sigma_{Ij' \rightarrow j} v_I^- \phi_{j'I}^u] r_{j'} \\ & = \frac{1}{C_p} [\beta_{1j} s_{1j}^{0+} v_1^+ + \sum_{i=2}^{I-1} (s_{ij}^{0-} v_i^- + s_{ij}^{0+} v_i^+) + \beta_{2j} s_{Ij}^{0-} v_I^-] + x_1^0 \gamma_{1j} - x_I^0 \gamma_{2j}. \end{aligned} \quad (67)$$

In Equation (67), the r_j factors are required to make the left hand side equal to the right when an unconverged set of fluxes ϕ_{ji}^u is used in the equation. (Since the fluxes ϕ_{ji}^u were computed using some old ϕ_{ji}^{u-1} from the previous inner iteration in the scattering-in terms of the neutron balance equations, the balance equations are not satisfied when ϕ_{ji}^u is used everywhere.)

The terms within the braces multiplying r_j in Equation (67) can be viewed as the neutron loss in group j , the expression multiplying $r_{j'}$ is the scattering-in source to group j from group j' (where group j' is a thermal group), and the expression on the right hand side of the equation can be viewed as the nonchanging source to group j from all other processes (scattering from groups above thermal, fissions, and external sources, including boundary sources).

The sum over the set of space points i_b represents the losses due to the application of the special internal boundary conditions at the surfaces of certain regions as described in Section 1.6.

In the case of periodic boundary conditions, Equation (67) applies with $\beta_{1j} = \beta_{2j} = 1$ and $\alpha_{1j} = \alpha_{2j} = \gamma_{1j} = \gamma_{2j} = 0$ for all thermal groups.

A term marked by an asterisk in the expression multiplying r_j is handled in a special manner when it contains a zero β . When β is zero in one of these terms, this means that the corresponding external boundary condition is of the form $\phi = \gamma/\alpha$, i.e., the end point flux is fixed and the corresponding term is a constant. Hence, it is not desired to have the term multiplying r_j , and it is therefore classified as a "source" term and moved to the right hand side of the equation. Keeping this special exception in mind, Equation (67) can be written in a simplified form:

$$\bar{L}_j \bar{\phi}_j r_j - \sum_{j' \geq j}^{\text{th}} \bar{T}_{j' \rightarrow j} \phi_{j'} r_{j'} = S_j^0, \quad (68)$$

where $\bar{\phi}_j$ is the column vector

$$\bar{\phi}_j \equiv \begin{vmatrix} \phi_{j1} \\ \phi_{j2} \\ \vdots \\ \phi_{ji} \\ \vdots \\ \phi_{jI} \end{vmatrix}$$

and the elements of the row vectors $\bar{L}_j$ and $\bar{T}_{j' \rightarrow j}$ are obvious from Equation (67). Equation (68) can now be written for each thermal group to form a matrix equation

$$\bar{B}\bar{r} = \bar{S}^0, \quad (69)$$

where

$$\bar{r} \equiv \begin{bmatrix} r_{jth} \\ r_{jth+1} \\ r_{jth+2} \\ \vdots \\ r_{jmax} \end{bmatrix}, \quad \bar{S}^0 \equiv \begin{bmatrix} S_{jth}^0 \\ S_{jth+1}^0 \\ S_{jth+2}^0 \\ \vdots \\ S_{jmax}^0 \end{bmatrix}$$

and the square matrix $\bar{B}$ has diagonal elements $b_{ii} = \bar{L}_{jth+i-1} \bar{\phi}_{jth+i-1}$ and off-diagonal elements $b_{ij} = \bar{T}_{jth+j-1 \rightarrow jth+i-1} \bar{\phi}_{jth+j-1}$ (for row i and column j).

When option d., above, is called for to accelerate the inner iterations, MØNA solves the matrix Equation (69) by a direct method. The subdiagonal elements are eliminated (by Gaussian elimination) and the resulting superdiagonal matrix allows backsolution for the r_j values, starting with r_{jmax} .

When options b. or c. are called for, it is assumed that all r_j are equal with the value R . Equation (68) is summed for all the thermal groups with the result

$$R^u = \left[\sum_{j \geq jth} S_j^0 \right] / \left[\sum_{j \geq jth} \bar{L}_j \bar{\phi}_j^u - \sum_{j \geq jth} \sum_{j' \geq jth} \bar{T}_{j' \rightarrow j} \bar{\phi}_{j'}^u \right]. \quad (70)$$

While these derivations correspond to real flux solutions, the equations for the r_j or R for an adjoint solution are identical except that the transfer coefficients $\bar{T}_{j' \rightarrow j}$ become $\bar{T}_{j \rightarrow j'}$.

3. AUTOMATIC SEARCHES FOR DESIRED REACTOR CONDITIONS

3.1 The Search Options

The MØNA program provides options for automatically varying a number of reactor properties in order to attain a desired condition or state. These searches proceed by solving a series of problems ("search iterations"), each solved as outlined in Section 2, until the generation factor λ^S attained on the s^{th} search iteration satisfies the convergence criterion

$$\left| \frac{\lambda^S - \lambda_d}{\lambda_d} \right| \leq \epsilon_2, \quad (71)$$

where λ_d is the desired generation factor and ϵ_2 is the convergence level specified by the user as EPS2. The value of λ_d is (a) the input word EIGEND (NSERCH=0) or (b) the last generation factor attained on the problem immediately preceding the search problem (NSERCH=1), where the value of NSERCH is set by the user. Also, for search iteration s , λ^s is either the converged generation factor attained as usual (IGENOP=0) or the ratio

$$\lambda^s = \int_V dv \, s_f^1 / \int_V dv \sum_j S_{ej} \quad (\text{IGENOP} = 1) \quad (72)$$

where s_f^1 is the fission source from the first source iteration [see Equations (39)] 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 (2)] in one or more user specified regions, "Buckling Search"
± 2	The poison parameter Σ_p [see Equation (2)] 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 (48)] 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 concentrations of all constituent materials named in the NUCSCH(K) list, "Concentration Search" (The concentrations of other constituent materials included in the ISUMS(K) list can also be affected.)

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

$$\left| \frac{x^S - x^{S-1}}{x^S} \right| \leq \epsilon_3 \quad (73)$$

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 outer (fission source) iterations (MAXOUT) per search iteration and set $\epsilon_1 = \text{EPS1}$ low enough [see Equation (46)] 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 NCQNVG set equal to 1 for pointwise source convergence.) Obviously, the inner 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 (71) on λ^S is well satisfied before Equation (73) on x^S is satisfied. [Recommended is $\epsilon_2 \leq \epsilon_3/10$, but, if possible, the user should avoid specifying that Equation (73) be satisfied.]

3.2 The Search Algorithms

For all the searches except the boundary search, 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^[9]. The following description of the method was taken from Reference 7.

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

$$C_2 = 1 + 10(\lambda_D - \lambda^1)/\lambda^1 \text{ if } \lambda^1 < \lambda_D \quad (74)$$

$$C_2 = 1/(1 + 10\omega) \text{ if } \lambda_D \leq \lambda^1 < 1.25\lambda_D \quad (75)$$

$$C_2 = 1/(1 - 2.62\omega + 50\omega^2) \text{ if } \lambda^1 \geq 1.25\lambda_D \quad (76)$$

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

$$\ln C_S = \ln C_{S-1} + \frac{(\ln C_{S-1} - \ln C_{S-2})}{(\lambda^{S-1} - \lambda^{S-2})} (\lambda_d - \lambda^{S-1}). \quad (77)$$

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

$$(C_S - a)(\lambda_d - b) = c \quad (78)$$

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

$$(C_{S-3} - a)(\lambda^{S-3} - b) = c$$

$$(C_{S-2} - a)(\lambda^{S-2} - b) = c \quad (79)$$

$$(C_{S-1} - a)(\lambda^{S-1} - b) = c.$$

The solution of System (79) involves the evaluation of a determinant. If the determinant is less than 10^{-20} , C_S is obtained from Equation (77). If C_S does not lie on the same branch of the hyperbola determined by Equation (78) as does C_{S-1} , then the linear extrapolation is performed

$$C_S = C_{S-1} + (C_{S-1} - C_{S-2})(\lambda_d - \lambda^{S-1}) / (\lambda^{S-1} - \lambda^{S-2}). \quad (80)$$

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, and total fission source searches the new parameter values are usually computed from $x^S = x^1/C_S$; and on a dimension search or a concentration 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, and total fission source searches and $x^S = x^1/C_S$ in dimension and concentration 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,

$$10^{-10} C_1 < C_S < 10^{10} C_1. \quad (81)$$

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 2 and with new initial guess values. The new guess values are ten times the previous initial ones on buckling, poison, and total fission source searches and

are simply the last values for a dimension search. On a concentration search, the new guess is one-tenth the previous guess. 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 (80) for the third and subsequent search iterations, and C_2 is found from

$$C_2 = C_1 + .1\omega. \quad (82)$$

The value of ω in Equation (82) is

$$\omega = W_r/W_l \text{ if } W_r < W_l$$

and

$$\omega = -1 \text{ if } W_r \geq W_l \quad (83)$$

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^2 , is applied in the formula for the total removal cross section in S each group and in each macro material to which the search is to be supplied, hence, the formula for the removal cross section becomes

$$\Sigma_{rj} = \Sigma_{aj} + \sum_{j' \neq j} \Sigma_{j \rightarrow j'} + D_j B_S^2 + T_j \Sigma_p \quad (84)$$

in each search material. The materials to be searched are named in the LRS(K) list by the user. All materials 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 buckling values found by a search will exist in the search materials on a succeeding problem.

Poison Searches. The value of the poison cross section parameter Σ_p in Equation (2) 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(1) = 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 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 (19) and (20). See Section 1.2 and the notes on the application of the input word IDCØ.

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 $F_0 = \text{FISSØR}$, itself, the search parameter.

In an inhomogeneous problem with fissions one can still solve for a generation factor λ^t from Equation (40) provided one applies the constraint (or renormalization) of Equation (48) 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 MØNA 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.

$$\int_V f_s^t dv = F_{0s} \quad (85)$$

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 FISSØR 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,

$$F_{01} = 2 \int_V \sum_j S_{ej} dv = 2 \int_V \sum_j (S_{pj} + S_{rj}) dv. \quad (86)$$

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 $SEØ(K,J)$ 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 FISSØR for the first search guess, F_{01} . Otherwise, F_{01} 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 FISSØR.

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.)

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

$$k_S = \max \{W_{LS}, W_{RS}\} (n_L + n_R) / (W_L + W_R). \quad (87)$$

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 $\min \{W_{\ell s}, W_{rs}\} \leq W_{\min}$. In this case, the smaller region is allowed one mesh interval and the larger region is allotted $k_s = n_{\ell} + n_r - 1$. At the same time, the smaller region width is set equal to $W_{\min}$ and the larger region width is set equal to $W_{\ell} + W_r - W_{\min}$. As stated before, the search is terminated if this case is encountered twice.

Concentration Searches. If the user sets ISERCH equal to ± 6 , the program applies the search algorithm to the concentrations of a set of specified constituent materials.

As explained in Section 1.2, most of the neutron group constants for any macro material m are mixed according to Equations (16) and (17) wherein the group constants of a number of constituent materials are weighted by the respective concentrations or weighting factors of each constituent. That is, a cross section, for example, of macro material m is computed from a formula of the type

$$\Sigma^m = \sum_{\ell \in M^m} \Sigma_{\ell}^m \rho_{\ell}^m, \quad (88)$$

where M^m is the set of constituent materials specified by the list $\ell_k^m \equiv \text{MIX}(K, IM)$ for material $m \equiv IM$, and $\rho_{\ell}^m \equiv C_k^m$ for $\ell \equiv \ell_k^m$. In a concentration search, the search parameters are the concentrations of the set of nuclides N specified by the user as the input list NUCSCH(K). If a constituent material ℓ belongs to this set (i.e., $\ell \in N$), then the concentration of that constituent is modified by the search parameter modifier C_s in each and every macro material in which the constituent material ℓ occurs--provided the user does not specify an LRS(K) list. When $\text{LRS}(1) \neq 0$, the code modifies the concentrations of the constituent materials $\ell \in N$ only in the macro materials specifically named in the LRS(K) list. That is, C_s is used to modify (multiply or divide) each ρ_{ℓ}^m for $\ell \in N$ for all m if $\text{LRS}(1) = 0$ or for $m \in \text{LRS}(K)$ if $\text{LRS}(1) \neq 0$.

The user can also specify a set of constituent materials I , where the set I is specified by the list ISUMS(K) and K denotes the K th word in the list. If $\text{ISUMS}(1) \neq 0$, the code maintains, throughout the search, the constraint

$$\sum_{\ell \in (M^m \cap I)} \rho_{s\ell}^m = \sum_{\ell \in (M^m \cap I)} \rho_{1\ell}^m, \quad (89)$$

in each macro material, where s is the search iteration number. That is, in each macro material m , the sum of the concentrations of all those constituents of the macro material that belong to the set I [specified by ISUMS(K)] is held constant at the original value. This is done (in each macro material m) by multiplying the concentrations of all constituents named in the ISUMS(K) list, but not in the NUCSCH(K) list by a single factor after the concentrations of the constituents named in NUCSCH(K) list have been modified in a given search iteration. This factor for macro material m is

$$\left[\sum_{\ell \in (M^m \cap I)} \rho_{S-1, \ell}^m - \sum_{\ell \in (M^m \cap I \cap N)} \rho_{S \ell}^m \right] / \sum_{\ell \in [M^m \cap I] - N} \rho_{S-1, \ell}^m; \text{ hence,} \quad (90)$$

for $\ell \in [M^m \cap I] - N$,

$$\rho_{S \ell}^m = \rho_{S-1, \ell}^m \left[\sum_{\ell' \in (M^m \cap I)} \rho_{S-1, \ell'}^m - \sum_{\ell' \in (M^m \cap I \cap N)} \rho_{S \ell'}^m \right] / \sum_{\ell' \in [M^m \cap I] - N} \rho_{S-1, \ell'}^m. \quad (91)$$

The final values of all the concentrations, as modified during a concentration search, can be edited by setting the integer input flag INED2 equal to 1.

III. PROGRAM OUTPUT

1. PROGRAM CHECKPOINTS

The first two lines printed by the MØNA program are "checkpoint" edits. A checkpoint edit is a one-line statement telling the user that a particular point in the program flow has been reached and what machine (CPU) time 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.

The first two checkpoint edits state, "Entering Main Link" and "Begin Problem".

2. INPUT DATA CARD CONTENTS

The contents of the MØNA problem title card are always edited as soon as the card is read. If the user has set the integer ILIST = 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 floating point data cards of any library material data decks that may follow the regular MØNA problem data.

3. PROCESSED INPUT DATA EDITS

After reading all the input data of a problem, including any library material decks following a problem, the program processes the basic input

data to the extent that a change case problem could be run if the program immediately returned to start reading the input data of another problem*. Then, if the integer flag $NØSØL = 3$ (read on the title card) the program does return to read the next problem without editing any data.

If $NØSØL < 3$, the program checks the input integer LIBED, and if $LIBED \neq 0$, an edit of the complete library material data is made, headed by the user specified library data set number. For each library material, the material number, the Hollerith contents of the library material title card and the group constants are edited. The group constants are D_j , Σ_{aj} , $\nu\Sigma_{fj}$, Σ_{fj} , E_j , $\Sigma_{\mu j}$, and $\Sigma_{j \rightarrow j'}$. (It is possible to input in-group scattering cross sections, $\Sigma_{j \rightarrow j'}$, and the edits include these data. However, these data are ignored in a MØNA solution.)

Following the library edit, the program completes the processing of the basic input data, in preparation for problem solution. If an input data flaw is found, the program returns to read the next problem data**. Otherwise, the program again checks $NØSØL$. If $NØSØL = 2$, the program returns to read the next problem; otherwise the program 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 three integer flags read on the problem title card.
2. All relevant (nonzero) integer data arrays including:
  - a. The list of mesh points ISPECT(K) at which flux spectrum edits are to be made.
  - b. The list of macro materials or regions to which a search is to apply.
  - c. The list of constituent materials NUCSCH(K) to which a concentration search is to apply.
  - d. The list of constituent materials ISUMS(K) to which the concentration search constraint, Equation (89), is to apply.
  - e. For each reaction rate ratio to be edited, the type and the list of constituent materials for both the numerator and the denominator.
3. The complete list of single word floating point data from EPS1 to FIXSRS.

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*Change cases require all macro materials to have been mixed and all automatic calculation of fixed volume sources, bucklings, fission spectra, etc. to be completed.

**No attempt is made to check for all possible input errors. If weird data is input, it is assumed that the user has a good reason and, perhaps, a new application in mind for the very flexible MØNA program.

4. The floating point data lists EBAR(J) (average group energy), DELE(J) (group energy width), and VRECIP(J) (average inverse group velocity).
5. The boundary condition parameters α_{1j} , β_{1j} , γ_{1j} , ω_{1j} , α_{2j} , β_{2j} , γ_{2j} , ω_{2j} .
6. The region dimension and mesh data consisting of the width RW(IR), upper bound, mesh spacing DELR(IR), macro material number MR(IR), number of intervals NI(IR), lower and upper mesh point indices, and the lower and upper source point indices.
7. The group constants for each macro material together with the number of downscatter groups, number of upscatter groups, number of thermal groups, the number of the first thermal group and the lists of constituent materials and respective concentrations. (The group constants are Σ_p , D_j , Σ_{aj} , B_j^2 , $\nu\Sigma_{fj}$, χ_j , Σ_{fj} , E_j , Σ_{rj} , Σ_{uj} , T_j , q_j , and $\Sigma_{j \rightarrow j'}$. Also included is a "normalization" cross section that is equal to $\Sigma_{rj} - \sum_{j'} \Sigma_{j \rightarrow j'}$ in real flux problems and $\Sigma_{rj} - \sum_{j'=j^{th}}^{j^{max}} \Sigma_{j' \rightarrow j}$ in adjoint problems.)
8. The fission source f^1 used on the first fission source (outer) iteration (provided the integer input word INSØRS < 0).
9. If there exists a fixed volume source for the problem:
  - a. The fixed volume source S_{ej} for each source point and each group ($S_{ej} = S_{rj} + S_{pj}$).
  - b. The basic fixed volume source data from which S_{ej} is computed; i.e., $S_{rj} \equiv SEØR(J,IR)$ for each group j and each region and $S_{pj} \equiv SEØ(K,J)$ for each source point and each group.

The integer input word INED1 can be set to suppress parts of the above input data edits as follows:

If INED1 $\geq$ 1, edit 7 of the macro material constants is suppressed except for the materials specifically listed in the integer input list IXLST(K).

If INED1 $\geq$ 2, edit 7 and edit 9a. are suppressed.

If INED1 $\geq$ 3, edit 7, edit 9a. and edit 9b. are suppressed.

If INED1 = 4, all the processed input data edits are suppressed.

4. ITERATION CONVERGENCE DATA EDITS

Following the processed input data edits, if any, the program checks the input integer NØSØL. If NØSØL = 1, the program returns to read the next problem, otherwise it proceeds with the solution of the problem. While

the solution is taking place, data showing the course of the convergence of the iterative processes can be edited.

4.1 Fission Source Iteration Data

The outer iterations are not performed and therefore not edited when there are no fissions; i.e., when $\nu \Sigma f_j = 0$ for all groups of all macro 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 $(\bar{\lambda}^t - \underline{\lambda}^t)/(\bar{\lambda}^{t-1} - \underline{\lambda}^{t-1})$ if IREN = 0 (renormalization type problem) and is $\max |(f^* \bar{\lambda}^t - f^* \bar{\lambda}^{t-1})/f^* \bar{\lambda}^{t-1}| / \max |(f^* \bar{\lambda}^{t-1} - f^* \bar{\lambda}^{t-2})/f^* \bar{\lambda}^{t-2}|$ 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 (59)

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 Multithermal Group Iteration Data

When there are more than one thermal group, data showing the course of the inner iteration convergence process can be edited if the user sets

the input integer INPRNT = 1. After each inner iteration, a line is printed showing the iteration number, u ; the total source-to-loss ratio, R^u ; the two point extrapolation factor being used, τ^u ; the convergence level $|R^u - 1|$; and the time, in seconds, of machine time (CPU) required for the iteration.

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 - $x^s = F_{0s}$.
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. Concentration search - x^s is the current multiplier of the concentrations of the nuclides specified by the NUCSCH(K) list. The initial multiplier is unity; i.e., $x^1 = 1$.

5. PROBLEM SOLUTION EDITS

5.1 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 and $\text{NRMFIX} > 0$, the program prints the normalized volume integral of the pointwise fixed volume source,

$\int_V dv N_f \sum_j S'_{pj}$, where the prime denotes the unnormalized value, together with the normalization factor used, N_f . 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 (46) and (47)]; and the desired convergence level, ϵ_1 . If pointwise convergence is called for ($\text{NCØNVG} = 1$), 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 $\text{ISERCH} < 0$, 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.

Whether or not there are fissions in the problem, a summary of the status of the convergence of the last inner iterations is edited if there are multiple thermal groups. Printed are the number of inner iterations done, the source-to-loss ratio R^u , the convergence level $|R^u - 1|$, and ϵ_4 , the desired convergence level.

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

If the integer input flag $\text{IOUT1} \geq 1$, 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 $\text{ILINES} = n$, 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 unextrapolated source Cf^{*t+1} from Equation (41), where t is the last source iteration performed. The normalizing constant C is usually 1.0, such that the volume integral of the source is F_0 (FISSØR), but if the integer flag NØRMAL 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 (41). Hence, if the source iterations are converged such that $f^{*t+1} = ft$, then the edited fluxes

are those that would result from an input source of Cf^{*t+1} . This means that if a source distribution Δf were computed using the edited fluxes from either $\sum_j v \Sigma_{fj} \phi_j$ or $\sum_j x_j \phi_j$ (for real and adjoint problems, respectively), then $\Delta f = \lambda^{t+1} Cf^{*t+1}$ where λ^{t+1} is the generation factor on a renormalization type problem ($IREN = 0$) and unity, otherwise.

The power distribution is always computed from

$$p = \frac{1}{\lambda^t} \sum_j E_j \Sigma_{fj} \phi_j \quad (92)$$

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 $\Sigma_{fj} \neq 0$.

If the integer input flag $ITOTPT > 0$, both the total flux and the total flux normalized at $ITOTPT$ are computed and edited. A line is printed for each mesh point, showing the mesh point index i , the position x_i , the total flux $\sum_j \phi_{ji}$, and the "normalized" flux $\sum_j \phi_{ji} / \sum_j \phi_{ji'}$, where $i' = ITOTPT$.

5.3 Flux Spectrum Data

The flux spectrum at specified mesh points will be edited if the desired mesh point numbers are given in the integer input list $ISPECT(K)$.

The real flux spectrum value for group j is computed by $\phi'_{ji} = \phi_{ji} / [\Delta E_j \sum_j \phi_{ji}]$, where ΔE_j is the energy (or lethargy) width of group j as specified by the $DELE(J)$ input list. The adjoint flux spectrum value listed is simply $\phi'_{ji} = \phi_{ji} / \sum_j \phi_{ji}$.

For each group j the program prints j , $\bar{E}_j$, ΔE_j , and ϕ'_{ji} , where $\bar{E}_j$ is the average energy of group j as specified by the $EBAR(J)$ input data list.

5.4 Median and Average Energies of Neutron Reaction Rates

If the integer input word $IOUT3$ is set to 1 or 3, the median energy and the average energy of various quantities are computed and edited.

The median and average energies of a function are defined as follows:

$$\text{Let } F_j \equiv \int_{\Delta E_j} f(E) dE, \text{ and} \quad (93)$$

$$\bar{E}_j F_j \equiv \int_{\Delta E_j} E f(E) dE; \quad (94)$$

then the median energy is the energy E_m at which

$$\int_0^{E_m} f(E) dE = \frac{1}{2} \int_0^{\infty} f(E) dE = \frac{1}{2} \sum_{j=1}^{j_{\max}} F_j, \quad (95)$$

and the average energy is

$$\bar{E} \equiv \sum_{j=1}^{j_{\max}} \bar{E}_j F_j / \sum_{j=1}^{j_{\max}} F_j. \quad (96)$$

The program finds the median energy, given a set of group values F_j , by assuming that the point function $f(E)$ in Equation (93) is flat in the group in which the median energy occurs and then linearly interpolating. The ΔE_j values in Equation (93) and (94) are the $DELE(J)$ values input and it is assumed that $j_{\max}$ is the group of lowest energy, such that the lower energy of group $j_{\max}$ is zero and the upper energy is $\Delta E_{j_{\max}}$, etc.

In finding average energies, the program assumes $\bar{E}_j \equiv EBAR(J)$, the list input for the group average energies.

The quantities for which median and average energies are edited are:

1. The neutron density, where $F_j = \phi_j (\frac{1}{V})_j$ and $(\frac{1}{V})_j \equiv VRECIP(J)$, the list input for the average reciprocal velocities of the group.
2. The neutron flux, where $F_j = \phi_j$.
3. The fission rate, where $F_j = \Sigma_{fj} \phi_j$.
4. The neutron absorption rate, where $F_j = \Sigma_{aj} \phi_j$.
5. The neutron capture rate, where $F_j = (\Sigma_{aj} - \Sigma_{fj}) \phi_j$.

The program first prints the median and average energies of each of these quantities at each mesh point. Then the median and average energies are printed for the volume integrals of the quantities for each region.

5.5 Macroscopic Material Mixture Data By Region

Of special usage in problems wherein the concentrations of the constituents of a macro material have been altered in the process of a concentration search, the new mixture data can be edited at this point if the user sets the integer input flag $INED2 = 1$. The list of constituent numbers and the respective values of the current concentrations are edited for each region.

5.6 Reaction Rates for Each Constituent of the Macroscopic Material in Each Region

If the user sets the integer input flag IOUT3 = 2 or 3, the program computes and edits a table of total (volume integrated) reaction rates for each region. For each constituent ℓ of the macro material m in a region, the following quantities are listed.

1. The constituent number ℓ is either a library material number or the number of a macro material used on the preceding problem.
2. The capture to fission ratio, computed from $\int_{V_r} dv \sum_j \phi_j (\Sigma_{aj}^\ell - \Sigma_{fj}^\ell) / \sum_j \phi_j \Sigma_{fj}^\ell$.
3. The neutron production rate, $\int_{V_r} dv \sum_j \phi_j \nu \Sigma_{fj}^\ell \rho_\ell^m$
4. The neutron fission rate, $\int_{V_r} dv \sum_j \phi_j \Sigma_{fj}^\ell \rho_\ell^m$
5. The neutron absorption rate, $\int_{V_r} dv \sum_j \phi_j \Sigma_{aj}^\ell \rho_\ell^m$
6. The neutron capture rate, $\int_{V_r} dv \sum_j \phi_j (\Sigma_{aj}^\ell - \Sigma_{fj}^\ell) \rho_\ell^m$
7. The neutron production fraction, $\int_{V_r} dv \sum_j \phi_j \nu \Sigma_{fj}^\ell \rho_\ell^m / \int_{V_r} dv \sum_j \phi_j \nu \Sigma_{fj}^m$
8. The fission fraction, $\int_{V_r} dv \sum_j \phi_j \Sigma_{fj}^\ell \rho_\ell^m / \int_{V_r} dv \sum_j \phi_j \Sigma_{fj}^m$
9. The neutron absorption fraction, $\int_{V_r} dv \sum_j \phi_j (\Sigma_{aj}^\ell - \Sigma_{fj}^\ell) \rho_\ell^m / \int_{V_r} dv \sum_j \phi_j (\Sigma_{aj}^m - \Sigma_{fj}^m)$
10. The neutron capture fraction, $\int_{V_r} dv \sum_j \phi_j (\Sigma_{aj}^\ell - \Sigma_{fj}^\ell) \rho_\ell^m / \int_{V_r} dv \sum_j \phi_j (\Sigma_{aj}^m - \Sigma_{fj}^m)$

In these formulas, V_r denotes integration over the volume of the r th region, m is the macro material of the region, and ρ_ℓ^m is the concentration of constituent ℓ in material m . The ρ_ℓ^m are defined by $\rho_\ell^m \equiv C_K^m \equiv \text{CON}(K, \text{IM})$ for $\ell = \ell_K^m$ where $\ell_K^m \equiv \text{MIX}(K, \text{IM})$ (for $k \equiv K$ and $m = \text{IM}$). If there has been a concentration search, the ρ_ℓ^m values are the modified values. All cross sections of the form Σ_j^m are defined by

$$\Sigma_j^m \equiv \sum_{\ell \in M^m} \Sigma_{j\ell}^\ell \rho_\ell^m, \quad (97)$$

where M^m is the set of constituent materials of macro material m , as specified by the list $\ell_K^m \equiv \text{MIX}(K, IM)$. Hence, the reaction rate fractions defined in 7. through 10., above, are the ratios of the reaction rate of the constituent to the total reaction rate of the macro material in the region.

For a given region, the quantities 1. through 10. are printed for each constituent. Then the totals (the corresponding quantities for macro material m) are printed.

5.7 Spatial Distributions of Reaction Rates and Ratios for Library Material Mixtures

The user can call for edits of the spatial distributions of reaction rates and ratios of reaction rates for sets of library materials selected from the constituents of the macro materials of a problem. To do this, the array $\text{MRATE}(K, N, L)$ is used as follows.

For each combination of the indices L and N , the user lists a set of library materials $\ell_K^{N, L}$, where $\ell_K^{N, L} \equiv \text{MRATE}(K, N, L)$ and K is the list index ($K = 1, 2, 3$, etc.).

The index L (set by the user) is the ratio number, where $1 \leq L \leq 5$ such that five reaction rate ratios can be defined by the user.

The index N (set by the user) is the reaction type number and it also indicates whether the list $\ell_K^{N, L}$ is to be used to compute the reaction rate for the numerator of the ratio ($N = +1$ through $+6$) or the reaction rate for the denominator of the ratio ($N = -1$ through -6). For each $|N|$ the reaction rate types are:

- $|N| = 1$ - absorption rate (Σ_a^ℓ data are used).
- $|N| = 2$ - fission rate (Σ_f^ℓ data are used).
- $|N| = 3$ - capture rate ($\Sigma_a^\ell - \Sigma_f^\ell$ data are used).
- $|N| = 4$ - neutron production rate ($\nu \Sigma_f^\ell$ data are used).
- $|N| = 5$ - power production rate ($E^\ell \Sigma_f^\ell$ data are used).
- $|N| = 6$ - "transport" collision rate (Σ_μ^ℓ data are used).

Since the numerator and the denominator of each reaction rate ratio is also printed, the user can call for editing of the spatial distributions of the sum of the capture rates of one set of nuclides, the sum of the fission rates of another set of nuclides, and the ratio of the two rates, for example.

Each reaction rate is computed from a formula of the type

$$R = \sum_{\ell'} \sum_j \phi_{ji} \Sigma_j^{\ell'} \rho_{\ell'} \quad (98)$$

where ℓ' corresponds to the library materials $\ell_K^{N,L}$ for a given combination of N and L, as described above, i is the spatial mesh point number, and $\rho_{\ell'}$ is set as follows.

If the library material ℓ' belongs to the set ℓ_K^m (specified by MIX(K,IM) for $\ell = K$ and $m = IM$) for the macro material m being used at mesh point i, then $\rho_{\ell'} = C_K^m$ as specified by the CON(K,IM) data. However, if $C_K^m = 0$, $\rho_{\ell'}$ is set to unity and used in Equation (98) for that mesh point.

If the library material ℓ' is not of member of the set ℓ_K^m for the macro material m of a given region, then $\rho_{\ell'}$ is zero for each mesh point i in the region.

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

If IQUT1 $\geq$ 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 $\nu\Sigma_{fj} \neq 0$). This ratio is computed for each group from

$$\left| \frac{\phi_{j\max}}{\phi_{j\text{av}}} \right| = \phi_j \left| \begin{array}{c} \text{max of} \\ \text{all fuel} \\ \text{regions} \end{array} \right| \left/ \left[\sum_{\text{fuel regions}} \int_V \phi_j dv \right] / \left[\sum_{\text{fuel regions}} \int_V dv \right] \right| \quad (99)$$

The point where the maximum fuel flux occurs and the average fuel flux used in the denominator of Equation (84) 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 (92).

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 (92).

5.9 Neutron Balances and Averages by Region and Group

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

1. The fission neutron source $x_j \int_{V_r} C f^t dv$ for real flux problems (100)
or $v \Sigma_{fj} \int_{V_r} C f^t dv$ for adjoint problems

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

2. The scattering-in source $\int_{V_r} \sum_{j' \neq j} \Sigma_{j' \rightarrow j} \phi_{j'} dv$ for real flux (101)
problems or $\int_{V_r} \sum_{j' \neq j} \Sigma_{j \rightarrow j'} \phi_{j'} dv$ for adjoint problems

3. The fixed volume source $\int_{V_r} S_{ej} dv = \int_{V_r} (S_{pj} + S_{rj}) dv$ (102)

4. The absorption loss $\int_{V_r} (\Sigma_{aj} + T_j \Sigma_p) \phi_j dv$ (103)

5. The scattering-out loss $\int_{V_r} \sum_{j' \neq j} \Sigma_{j \rightarrow j'} \phi_j dv$ (104)

6. The perpendicular leakage $\int_{V_r} D_j B_j^2 \phi_j dv$ (105)

7. The parallel leakage loss $\int_{V_r} \nabla \cdot \vec{J}_j dv = - \int_{V_r} \nabla \cdot D_j \nabla \phi_j dv$ (106)

8. The volume integrated flux $\int_{V_r} \phi_j dv \equiv \int_{V_r} C_\rho x^\rho \phi_j dx$, where V_r (107)
is the region volume

9. The average flux
$$\int_{V_r} \phi_j dv / \int_{V_r} dv \quad (108)$$
10. The flux spectrum
$$\int_{V_r} dv (\phi_j / \Delta E_j) / \int_{V_r} dv \sum_j \phi_j \quad (109)$$
11. The neutrons produced
$$\frac{1}{\lambda t} \int_{V_r} v \Sigma_f \phi_j dv, \text{ for real flux problems} \quad (110)$$

or
$$\frac{1}{\lambda t} \int_{V_r} \chi_j \phi_j dv \text{ for adjoint problems}$$
12. The parallel buckling
$$- \int_{V_r} \nabla \cdot D_j \nabla \phi_j dv / \int_{V_r} D_j \phi_j dv. \quad (111)$$

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 (92).

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 (92)], the flux times volume, and the out-scatter and the in-scatter.

6. OUTPUT FOR USE IN OTHER COMPUTER PROGRAMS

6.1 Punched Card Output

The integer input word IPUNCH can be set to cause the MØNA program to output a number of blocks of data on punched cards that can be used by a number of other computer programs. The options are as follows.

When IPUNCH = 1, a card deck is punched with the output fluxes of the problem. The deck is headed by a card with the Hollerith words "ordinary fluxes" or "adjoint fluxes" to identify the type of fluxes. Group number 1 fluxes are punched first, six words to a card and 12 columns per word (Fortran format 1P6E12.5). Each neutron group begins on a new card.

When IPUNCH = 2, the neutron group fluxes and the neutron fission source are punched in the format acceptable as input to codes such as CØRA, MØNA, and others. Heading the deck is a card with the Hollerith words "ordinary fluxes" or "adjoint fluxes", as appropriate.

When IPUNCH = 3, data are punched for use in a computer program that calculates the volume integrals of the various adjoint weighted reaction rates needed in perturbation theory analyses. Two data decks are needed; one for the real (ordinary, direct) flux problem and one for the adjoint (importance function) problem. The decks will be in the correct order if the user runs first the MONA real flux problem and then the adjoint problem, both with IPUNCH = 3, of course.

6.2 Output of Entire Problem Data

All of the data of a problem--the problem input data (in a compact form) and the problem solution data (group fluxes and neutron fission source) are written on Fortran logical unit 9 if the user sets the integer input word ITAPE > 0. The problem data on unit 9 are preceded by the word ITAPE, itself, to identify the data when it is to be read and used. Hence, the user should choose a different value for ITAPE for each problem of a series stored on a given computer run. The problem data of an adjoint problem are distinguished by preceding the data on unit 9 by -ITAPE.

The data stored on unit 9 can be used directly by the perturbation theory computer code referred to in 6.1, above. Other codes, such as one that plots the solution data in a variety of ways, can easily be written to process the problem data at a later time.

7. SPACE AND ENERGY COALESCED GROUP CONSTANTS

Neutron group constants for use in calculations with fewer groups and/or regions are calculated and edited if the user sets the integer input flag KØALES = 1 and defines the broad energy and region break points by means of the IEBK(K) and IRBK(K) integer data lists, respectively. The first broad group, J = 1, is assumed to consist of MØNA groups j = 1 through j = IEBK(1); broad group J = 2 extends from j = IEBK(1) + 1 through IEBK(2), etc. The broad regions are determined in an identical manner from the IRBK(K) data.

When KØALES = 1, there must be entries in both IEBK(K) and IRBK(K).

When there is coalescing the first data printed following the printed contents of the problem title card are the energy and region break points, IEBK(K) and IRBK(K). Then the region widths

$W_R = \sum_{r \in R} W_r$ and region volumes $V_R = \sum_{r \in R} V_r \equiv \sum_{r \in R} \int_{V_r} dv \equiv \sum_{r \in R} \int_{x_r} C_p x^p dx$ are listed where R is the index of the broad regions, W_r is the width of MØNA region r and x_r denotes integration over the range of x for region r.

Next, the average fluxes and the flux-volume integrals are printed for each broad group and broad region. For J and R, the average flux is

$$\phi_J^R \equiv \sum_{r \in R} \sum_{j \in J} \phi_j^r v_r / \sum_{r \in R} v_r, \quad (112)$$

where the flux-volume integral $\phi_j^r v_r$ is computed from

$$\phi_j^r v_r = \int_{v_r} dv \phi_j. \quad (113)$$

Coalesced boundary condition constants are next edited.

$$\alpha_{1j} \equiv \sum_{j \in J} |\alpha_{1j} \phi_{j1}| / \sum_{j \in J} \phi_{j1}. \quad (114)$$

$$\alpha_{2j} \equiv \sum_{j \in J} |\alpha_{2j} \phi_{jI}| / \sum_{j \in J} \phi_{jI}, \text{ where } I = i_{\max}. \quad (115)$$

$$\beta_{1J} \equiv \sum_{j \in J} (|\alpha_{1j} \phi_{j1}| + |\gamma_{1j}|) / \sum_{j \in J} [\alpha_{1j} \phi_{j1} + \gamma_{1j}] / \beta_{1j}. \quad (116)$$

$$\beta_{2J} \equiv \sum_{j \in J} (|\alpha_{2j} \phi_{jI}| + |\gamma_{2j}|) / \sum_{j \in J} [\alpha_{2j} \phi_{jI} + \gamma_{2j}] / \beta_{2j}. \quad (117)$$

$$\gamma_{1J} \equiv \sum_{j \in J} |\gamma_{1j}|. \quad (118)$$

$$\gamma_{2J} \equiv \sum_{j \in J} |\gamma_{2j}|. \quad (119)$$

The coalesced neutron group constants for all the broad groups are edited next for each broad region in turn. The broad group constants for region R are:

Diffusion coefficient,

$$D_J^R \equiv \sum_{r \in R} \sum_{j \in J} D_j \int_{v_r} dv |\Delta \phi_j| / \sum_{r \in R} \sum_{j \in J} \int_{v_r} dv |\Delta \phi_j| \quad (120)$$

where

$$\int_{v_r} dv |\Delta \phi| = \sum_{i=i_{r-1}}^{i_r-1} \left| \frac{\phi_{i+1} - \phi_i}{x_{i+1} - x_i} \right| \int_{x_i}^{x_{i+1}} c_p s^p dx \quad (121)$$

and i_r is the mesh index of the upper boundary of region r.

Absorption cross section,

$$\Sigma_{aJ}^R \equiv \sum_{r \in R} \sum_{j \in J} \Sigma_{aj} \phi_j^r v_r / \phi_J^R v_R, \quad (122)$$

where ϕ_J^R and ϕ_j^r are defined in Formulas (112) and (113).

Transverse (perpendicular buckling),

$$B_J^{2R} \equiv \sum_{r \in R} \sum_{j \in J} D_j B_j^2 \phi_j^r v_r / (D_J^R \phi_J^R v_R), \quad (123)$$

where B_j^2 are the perpendicular bucklings input as the BSQ array.

Neutron production cross section,

(124)

$$v \Sigma_{fJ}^R \equiv \left(\sum_{r \in R} \sum_{j \in J} v \Sigma_{fj} \phi_j^r v_r \right) \left(\sum_J \sum_{j \in J} \sum_{r \in R} x_j \int_{v_r} f^t dv \right) / \phi_J^R v_R \sum_{r \in R} \int_{v_r} f^t dv.$$

Fission spectrum fraction (chi),

$$\chi_J^R \equiv \sum_{r \in R} \sum_{j \in J} x_j \int_{v_r} f^t dv / \sum_J \sum_{j \in J} \sum_{r \in R} x_j \int_{v_r} f^t dv, \quad (125)$$

where f^t is the fission neutron source used on the last outer iteration t , as defined in Equations (39) through (43).

Fission cross section,

$$\Sigma_{fJ}^R \equiv \sum_{r \in R} \sum_{j \in J} \Sigma_{fj} \phi_j^r v_r / \phi_J^R v_R. \quad (126)$$

Energy per fission,

$$E_J^R \equiv \sum_{r \in R} \sum_{j \in J} E_j \Sigma_{fj} \phi_j^r v_r / \Sigma_{fJ}^R \phi_J^R v_R. \quad (127)$$

"Transport" cross section,

$$\Sigma_{\mu J}^R \equiv 1 / (3D_J^R - T_J^R \Sigma_p^R). \quad (128)$$

Poison cross section multiplier,

$$T_J^R \equiv \sum_{r \in R} \sum_{j \in J} T_j \Sigma_p \phi_j^r v_r / \Sigma_p^R \phi_J^R v_R. \quad (129)$$

Poison cross section,

$$\Sigma_p^R \equiv \sum_{r \in R} \sum_{j \in J_{\max}} T_j \Sigma_p \phi_j^r v_r, \quad (130)$$

where $J_{\max}$ is the last (lowest energy) broad group.

Scattering transfer cross section,

$$\Sigma_{J \rightarrow J'}^R \equiv \sum_{r \in R} \sum_{j' \in J'} \sum_{j \in J} \Sigma_{j \rightarrow j'} \phi_j^r v_r / \phi_J^R v_R. \quad (131)$$

Although it is edited, the within-group scatter $\Sigma_{J \rightarrow J}^R$ from Equation (131) is not a useful value because there are no within-group terms (with $\Sigma_{j \rightarrow j} \phi_j$) for the MØNA group included.

8. ANCILLARY ROUTINES TO PROCESS PROBLEM DATA

It is easy to add routines to the MØNA program to perform special operations on the problem input and solution data. The integer input data flags IPERT and IPLØT are reserved for controlling entry to these ancillary programs as they are added to MØNA to serve the special purposes of a user. These ancillary programs remain more or less independent of the rest of the MØNA program so that the user needs only a minimum knowledge of MØNA in order to write his own ancillary program.

Each such program is assigned either a unique IPERT or IPLØT value, such that the user can call for entry to his program (after all regular MØNA processing, solution and output, has been done) by setting IPERT or IPLØT to the value assigned for his program.

The user must also set up his own input data deck, if one is required. All ancillary program input data decks follow the data deck of the regular MØNA problem and the data decks of any library materials read. The input data deck of an ancillary program assigned an IPERT number is read before any deck of an ancillary program is assigned an IPLØT number.

✓

8.1 Activation Traverses for Library Materials

The IPERT = 1 value is assigned to an ancillary program of more or less general usefulness. It will be described here primarily to serve as a good example of ancillary program usage.

The data deck for this ancillary program consists only of one or more cards required to read an integer array IACT(K). This array is simply a selected list of library materials (existing on the MØNA library material data set) for which the activation traverse data are desired. The library materials listed must be in ascending order. The card format is the same as the MØNA integer data card format, with the word IEND in column 1 signaling the last card of the deck and with the word IWC in card columns (ccs) 11 and 12 telling the number of input data words to be read on the card. A value punched Index 3 in ccs 6-8 signifies that the input data on the card are the values of IACT(K) for K = INDEX3 through INDEX3 + IWC - 1. The dimension of IACT(K) is 50 and the last value of IACT(K) read must be zero to signify the end of the list.

After reading the data deck, the program prints the words "Activation Traverses"; followed by the contents of the MØNA problem title card. The program then searches the library data set unit for each material of the IACT(K) list. If a material cannot be found, the problem terminates. When a material is found, the data are read and the Hollerith contents of the library material data card and the material number are printed.

The first traverses printed include the total activation and the activation contributions of from one to five broad energy groups. The coarse (broad) energy groups are specified by the IEBK(K) data of the regular MØNA problem as discussed in Section 7., above.

The activation for coarse group J of library material λ is

$$A_J^\lambda \equiv \sum_{j \in J} \Sigma_{aj}^\lambda \phi_j, \quad (132)$$

and the total activation is

$$A^\lambda \equiv \sum_J A_J^\lambda. \quad (133)$$

The activation traverses for library material λ consist simply of all the values of A^λ and A_J^λ listed for each mesh point. If Σ_{aj}^λ is a microscopic cross section, the traverses can be considered the spatial distribution of the absorption rate per atom of library material λ .

Following the table of values for the total and coarse group activations for a material, a table of average activation cross sections for the material is printed. This table consists of the values, for each mesh point, of the total average,

$$\bar{\Sigma}_a^\ell \equiv \sum_j \Sigma_{aj}^\ell \phi_j / \sum_j \phi_j \quad (134)$$

and the coarse group averages,

$$\bar{\Sigma}_{aJ}^\ell \equiv \sum_{j \in J} \Sigma_{aj}^\ell \phi_j / \sum_{j \in J} \phi_j. \quad (135)$$

By applying this edit option to a dummy material of unit absorption cross section, the spatial distributions can be obtained for the total flux, $\sum_j \phi_j$, and for from one to five broad group fluxes, $\sum_{j \in J} \phi_j$.

IV. INPUT DATA PREPARATION

1. GENERAL COMPOSITION AND ORDER OF THE INPUT DATA

The data deck for a MØNA problem must always include

1. one problem title card,
2. one or more fixed point data cards, and
3. one or more floating point data cards.

If there is no library material data set on Fortran logical unit 8 that has been saved from a previous run and which has all the needed library materials, or if new library material data is to be read in, then one or more library material decks follow. A library material deck must always consist of

1. one library material title card, and
2. one or more floating point data cards.

The material numbers on these library material decks must be in increasing order, but the numbers need not be successive.

The input data decks of one or more ancillary problems can then follow. The data deck for an ancillary problem assigned an IPERT number is read before that of one assigned an IPLØT number. (Each ancillary problem is assigned a unique IPERT or IPLØT number to determine the calling sequence. The ancillary program assigned IPERT = 1 and instructions on its input data preparation are described in Section III-8.1.)

Data decks as described above can be stacked indefinitely in order to process problems successively in a given computer run. Data used (either input or automatically generated) in a MØNA problem are available

to the succeeding problem, with the exception of the material mixture data MIX(K,IM) and CON(K,IM). Hence, a minimum of input data is needed for succeeding problems of similar nature.

2. DETAILED DESCRIPTION OF THE INPUT DATA

2.1 The Problem Title Card

The card format (Fortran) is 11A6,3I2.

The problem title card is punched as follows:

Columns	1-66		Any identifying Hollerith characters can be punched here.
Column	68	NEWJOB	Zero or blank means no action is taken. 1 punched means that all previous problem data are set to zero before starting the present problem.
Column	70	ILIST	Zero or blank means no action is taken. 1 punched means that the contents of each card will be printed as soon as it is read.
Column	72	NOSOL	Zero or blank means a problem is to run normally. 1 punched means read the problem data deck and any library material data decks and do any input and library data set edits, but do not perform the solution of the problem. 2 punched means also do not do any input edits. 3 punched means also do not do any library data set edits. (Change case problems* can be run following a problem, regardless of the NOSOL value. See Sections III-2. through III-4.)

2.2 The Integer Input Data

The Fortran format for a integer input data card is:

*A change case is a succeeding problem that utilizes some of the retained data from the preceding problem.

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 problem or change case.

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 problem title card and precede the floating point data deck.

Each integer input data word must be right adjusted; that is, the number must end in the extreme right column of the 4 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 problem floating point data and library material 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 MIX(K,IM) (I-1,IM,K,0), RW(IR) (F-0,1,IR,0), and SIGA(J,L) (LF-1,L,J,2). The I, F, and LF designate integer input data, problem floating point input data, and library material floating point input 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 Integer 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	JMAX	Number of energy groups; $1 \leq JMAX \leq 50$ and $IMAX(JMAX+6) + 6(IRMAX) \leq 8705$, where IMAX is the number of mesh points on the one-dimensional axis of the problem and IRMAX is the number of regions. For example if IMAX is 500, one could have 17 groups and 34 regions.
0	0	2	0	IRMAX	Number of regions; $1 \leq IRMAX \leq 50$.
0	0	3	0	NGEØ	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 $\omega_{1j} \equiv \text{ØMEGA1}(J)$, $\alpha_{1j} \equiv \text{ALPHA1}(J)$, $\beta_{1j} \equiv \text{BETA1}(J)$, and $\gamma_{1j} \equiv \text{GAMMA1}(J)$, where $\text{ALPHA1}(J)$, $\text{BETA1}(J)$, $\text{GAMMA1}(J)$, and $\text{ØMEGA1}(J)$ are floating point input (F-0,5,-,J) for group $j \equiv J$. D_j is the value of $DC(J,IM)$ for group $j \equiv J$ of macroscopic material number IM in the region adjacent to the boundary. 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 7. 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	NØBDRY	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 $\omega_{2j} \equiv \text{ØMEGA2}(J)$, $\alpha_{2j} \equiv \text{ALPHA2}(J)$, $\beta_{2j} \equiv \text{BETA2}(J)$, and $\gamma_{2j} \equiv \text{GAMMA2}(J)$, where $\text{ALPHA2}(J)$, $\text{BETA2}(J)$, $\text{GAMMA2}(J)$, and $\text{ØMEGA2}(J)$ are floating point input (F-0,5,-,J) for group $j \equiv J$. D_j is the value of $DC(J,IM)$ for group $j \equiv J$ of macroscopic material number IM in the region adjacent to the boundary. 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 7. 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 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	6	0	IREN	Fission neutron source "renormalization" specification 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. 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 any outer iterations. 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	Flux solution type indicator 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. 2 indicates both the real and adjoint flux solutions are to be performed on the same problem.
0	0	8	0	IGENOP	Formula for search generation factor specification 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 1st 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. See Equation (72).
0	0	9	0	KOALE	Group and region coalescing indicator 0 or blank indicates no coalescing is to be done. 1 indicates that following the output edits, the average neutron group constants are to be obtained for the coalesced energy groups and/or regions defined by the group and region break points given as the IEBC(J) (I-0,9,K,0) and IRBK(IR) (I-0,10,K,0) lists.
0	0	10	0	IPERT	First ancillary program flag 0 or blank indicates that no ancillary programs assigned IPERT numbers are to be processed. 1 indicates that the ancillary program assigned the IPERT=1 value is to be used. See Section III-8.1. Following the solution and output edits, when all problem data are available, ancillary programs can be entered. These programs can be coded as the user desires. n > 0 indicates that the ancillary program assigned IPERT = n is to be processed.
0	0	11	0	INBUCK	Type of input buckling specification for group J and macroscopic material IM 0 or blank specifies that the bucklings can vary by both material and energy group and that they are input directly as BSQ(J,IM) (F-6,IM,J,0). 1 specifies that the bucklings can vary only by material and that they are input as BSQR(IM) (F-0,2,IM,0). 2 specifies that the bucklings can vary only by energy group and that they are input as BSQ(J,1), the values for material 1. (This means there must be a set of bucklings read in for macroscopic material 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 material and that they are input as BSQ(1,1), the value for material 1, group 1. (This value must be input regardless of whether or not material 1 is used elsewhere in the problem.)

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 0	0 0 0	12 13 14	0 0 0	IDCØ	Ignored Ignored Type of diffusion coefficient formula specification for group $j \equiv J$ and material $m \equiv IM$. Let D_j^m be the diffusion coefficient and Σ_{uj}^m be the corresponding macroscopic cross section. 0 or blank specifies that $D_j^m = 1/(3\Sigma_{uj}^m)$. 1 specifies that $D_j^m = 1/(3\Sigma_{uj}^m + T_j^m \Sigma_p^m)$, where $T_j^m \Sigma_p^m \equiv T(J, IM) \text{SIGPT}(IM)$ is the poison cross section. 2 specifies that D_j^m is the so-called asymptotic value computed from $D_j^m = \Sigma_{rj}^m / (\kappa_j^m)^2$, where $\Sigma_{rj}^m = \Sigma_{aj}^m + T_j^m \Sigma_p^m + \sum_{j' \neq j} \Sigma_{s,j \rightarrow j'}^m$, and κ_j^m is the solution of the transcendental equation $\frac{\kappa_j^m}{\Sigma_j^m} = \tanh \frac{\kappa_j^m}{\Sigma_j^m - \Sigma_{rj}^m}$. Also, $\Sigma_{aj}^m \equiv \text{SIGA}(J, IM)$, $\Sigma_{s,j \rightarrow j'}^m$ is the macroscopic scattering cross section for transfer from group j to group j' , and $\Sigma_j^m = \Sigma_{aj}^m + T_j^m \Sigma_p^m$.
0	0	15	0	ISØRS	Method of automatic calculation of pointwise fixed volume source specification Pointwise fixed volume sources can be input directly as $\text{SEØ}(K, J)$ (F-10, J, K, 0) for source point K and group J . The $\text{SEØ}(K, J)$ 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 IR^{th} region is defined by $K = I + IR - 1$. If IM is the macroscopic material in the IR^{th} region, then 0 or blank means, as in the normal case, the directly input values of $\text{SEØ}(K, J)$ remain. 1 means $\text{SEØ}(K, J) = 0$. 2 means $\text{SEØ}(K, J) = \text{SIGNU}(J, IM) \phi(I, J)$. 3 means $\text{SEØ}(K, J) = \text{CHIM}(J, IM) \phi(I, J)$. 4 means $\text{SEØ}(K, J) = \text{SIGFIS}(J, IM) \phi(I, J)$. 5 means $\text{SEØ}(K, J) = \text{ENERGY}(J, IM) \phi(I, J)$. 6 means $\text{SEØ}(K, J) = \text{SIGMU}(J, IM) \phi(I, J)$. 7 means $\text{SEØ}(K, J) = \text{SIGA}(J, IM) \phi(I, J)$. 8 means $\text{SEØ}(K, J) = T(J, IM) \phi(I, J)$. $\text{SIGNU}(J, IM)$ is neutrons per fission times the fission cross section. $\text{CHIM}(J, IM)$ is the fraction of fission neutrons released in group J for material IM . $\text{SIGFIS}(J, IM)$ is the fission cross section. $\text{ENERGY}(J, IM)$ is the energy per fission. $\text{SIGMU}(J, IM)$ is the so-called "transport" cross section. $\text{SIGA}(J, IM)$ is the absorption cross section. $T(J, IM)$ is the fraction of the poison cross section $\text{SIGPT}(IM)$ added to group J .
0	0	16	0	NRMFIX	Flag for normalization of the pointwise fixed volume source 0 specifies no normalization of the $S_{pj} \equiv \text{SEØ}(K, J)$ data. 1 designates that before problem solution, the pointwise part of the fixed volume source $S_{pj} \equiv \text{SEØ}(K, J)$ is to be multiplied by a normalization factor N_f such that $\int_V dv N_f \sum_j S_{pj} = \text{FIXSRS}$ (F-0, 0, 20, 0), where S_{pj} is either input or automatically computed as above when $\text{ISØRS} > 0$.
0 0	0 0	17 18	0 0	IPLØT	Ignored Second ancillary program flag 0 or blank indicates no ancillary programs assigned IPLØT numbers are to be processed. $n > 0$ indicates that the ancillary program assigned the IPLØT number n is to be processed following processing of any IPERT (I-0, 0, 10, 0) ancillary program.

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	19	0	ICHIM	Material independent fission spectra indicator 0 or blank indicates that the material dependent fission spectrum data $x_m^m = \text{CHIM}(J, IM)$ are to be used, as input. $n > 0$ causes the fission spectrum data for all fissionable materials (those with some $v_{fj} \neq 0$) to be set equal to the fission spectrum data input for macro material $m = n$.
0	0	20	0		Ignored
0	0	21	0	INED1	Input data edit control flag 0 or blank causes all relevant, nonzero input data to be edited. 1 causes suppression of the group constant data for the macro materials except for those named in the IXLST(K) list (I-0,15,K,0). 2 causes the additional suppression of the edits of the fixed volume sources used. (This is the sum of the regionwise $\text{SEOR}(J, IR)$ (F-0,IR,J,0) and the pointwise $\text{SEOR}(K, J)$ (F-10,J,K,0) data, provided they are nonzero.) 3 causes additional suppression of the edits of the basic fixed volume source data $\text{SEOR}(K, J)$. 4 causes additional suppression of all other input data edits, with the exception of library data set edits which are controlled by LIBED (I-0,0,31,0).
0	0	22	0	INED2	Flag controlling editing of macro material mixture data 0 or blank suppresses the edit. 1 causes the macro material mixture data consisting of the material numbers of the constituents and the corresponding concentrations to be edited for each region at problem output time (following solution). This is useful following a concentration search.
0	0	23	0	INED3	Reserved
0	0	24	0		Ignored
0	0	25	0	IOUT1	First output data edit control flag 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. See Section III-5. 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.
0	0	26	0	IOUT2	Second output data edit control flag (Reserved)
0	0	27	0	IOUT3	Third output data edit control flag 0 or blank suppresses the edits. 1 causes editing of spatially dependent median and average energies for a number of neutron reaction rates. See Section III-5.4. 2 causes editing of regionwise totals for the important neutron reaction rates of each constituent of the macro materials of the regions. See Section III-5.6. 3 causes editing of both of the above.
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 n th 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 which show the convergence of the fission source iteration.

Table II (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	30	0	INPRNT	Multiple thermal flux inner iteration data edit control 0 or blank causes suppression of inner iteration data. See Section III-4.2. 1 causes an edit (after each inner iteration on the multiple thermal flux groups) which lists the iteration number, the source to loss ratio for the thermal fluxes on the given iteration, and the degree of convergence of the ratio to unity.														
0	0	31	0	LIBED	Library material data set edit indicator 0 or blank means no edit is done. 1 means the data for each library material in the data set will be edited (following the updating of the data, if any).														
0	0	32	0	IPUNCH	Punched card output control 0 or blank means no data will be punched. 1 causes the fluxes (real or adjoint) to be punched six to the card, twelve columns to the word until the fluxes for a group are exhausted. A new card is then used to begin punching the next group. 2 causes the output fluxes and the fission neutron source acceptable to the MØNA, CØRA, and other programs to be punched. 3 causes problem data and fluxes to be punched as needed by a program that performs calculations for perturbation theory analyses.														
0	0	33	0	ITAPE	Note: Whenever options 1 or 2 are used, the first card of the deck is punched with the words "ØRDINARY FLUXES" or "ADJØINT FLUXES" to help identify the data of the deck. Indicator for saving problem data on Fortran logical unit 9 0 or blank means no problem data are saved. n means the data of the problem (input data, processed input data, and the complete set of solution data, including fluxes, fission sources, etc.) are written in binary form on Fortran logical unit 9. The data written for each problem is preceded on the tape by a single word record to identify the problem. This word is ITAPE for ordinary problems and -ITAPE for adjoint problems.														
0	0	34	0	NØRMAL	Ignored														
0	0	35	0		Ignored														
0	0	36	0		Output edit "normalization" specification 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 FISSØR (F-0,0,9,0). 1 specifies that all output data are normalized such that the total fission power rate equals PØWER (F-0,0,11,0). ENERGY(J,IM) and SIGFIS(J,IM) 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 = NØGRP at mesh point I = MESHNØ. 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 NØGRP or MESHNØ are zero or blank														
					<table><tr><th colspan="2"></th><th colspan="2">MESHNØ</th></tr><tr><th colspan="2"></th><th>0</th><th>Nonzero</th></tr><tr><td rowspan="2">NØGRP</td><td>0</td><td>The maximum flux</td><td>The maximum flux at MESHNØ</td></tr><tr><td>Non-Zero</td><td>The maximum flux in NØGRP</td><td>The flux value at MESHNØ in NØGRP</td></tr></table>			MESHNØ				0	Nonzero	NØGRP	0	The maximum flux	The maximum flux at MESHNØ	Non-Zero	The maximum flux in NØGRP
		MESHNØ																	
		0	Nonzero																
NØGRP	0	The maximum flux	The maximum flux at MESHNØ																
	Non-Zero	The maximum flux in NØGRP	The flux value at MESHNØ in NØGRP																
0	0	37	0	MESHNØ	3 specifies that all output are normalized such that the fission neutron source density is unity at the source point K = MESHNØ, unless MESHNØ 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.)														
0	0	38	0	NØGRP	4 specifies that all output data are simply multiplied by FLMULT (F-0,0,10,0). Mesh point or source point specification for output data "normalization" control when NØRMAL equals to 2 or 3, above. Group number specification for output data "normalization" control when NØRMAL equals to 2, above.														

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	39	0	ITØTPT	Total flux edit control flag 0 or blank suppresses the edit. n > 0 causes an output edit of the spatial distribution of the total neutron flux, $\sum_j \phi_j$, and the distribution of the total flux normalized to unity at space point n = ITØTPT.
0 0	0 0	40 41	0 0	INFLUX	Ignored Type and source of neutron flux guess specification 0 or blank indicates that the fluxes used for a first guess in a multiple thermal group problem or for automatically computing pointwise fixed volume sources [see ISØRS (I-0,0,15,0)] will be: a. the fluxes read from data cards (F-0,12,I,J) if any are read in. b. zero, if no fluxes are read in and this is a first problem; or c. the fluxes obtained in the preceding problem, if no fluxes are read in.
0	0	42	0	INCØNV	1 causes all the fluxes to be set to zero. Multiple thermal group inner iteration convergence acceleration control 0 or blank: No convergence acceleration is used. 1: Convergence is accelerated by multiplying all thermal group flux values by the thermal group source-to-loss ratio R^u that is computed each inner iteration. 2: Convergence is accelerated as for INCØNV = 1, above, but only until the source-to-loss ratio is within $\pm$ EPS5 (F-0,0,5,0) of unity; thereafter no acceleration multiplication is done. 3: Convergence is accelerated by multiplying the fluxes of each group by a groupwise source-to-loss ratio r_j^s until the total source-to-loss ratio R^u is within EPS5 of unity. 4: Convergence is accelerated by a two point extrapolation with the extrapolation factor τ set equal to THRMX (F-0,0,7,0). See Equation (63). 5: Convergence is accelerated by a two point extrapolation with the extrapolation factor τ computed automatically by Equations (64) and (65). The first guess for τ is THRMX (F-0,0,7,0).
0	0	43	0	MAXINS	Specification of maximum number of inner iterations on multiple thermal group problems 0 or blank causes MAXINS to be automatically set to 3. 1 indicates only one solution for the thermal groups will be performed per outer iteration (fission neutron source iteration). If the problem is a one outer iteration type problem, such as a fixed volume source problem with no fission, MAXINS should be large enough to allow convergence of the thermal groups. n indicates that the thermal group inner iterations performed for each outer iteration will be terminated on the n^{th} inner iteration--provided the thermal groups are not converged before reaching the n^{th} iteration. See EPS4 (F-0,0,4,0).
0	0	46	0	INSØRS	Type of fission neutron source guess specification 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 INSØRS = 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. A negative value of INSØRS causes the source guess to be edited, provided any problem input data edits are called for.

Table II (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	47	0	IGESS	Specification of 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 outer iteration performed. This allows a series of one outer iteration problems to be run with each using the same fission neutron source guess. See MAXOUT (1-0,0,49,0).
0	0	48	0	NCNVG	Convergence on total or pointwise fission neutron source specification 0 or blank specifies that the outer (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.
0	0	49	0	MAXOUT	The outer iterations are terminated when the change from the preceding iteration is less than $\pm EPS1$ (F-0,0,1,0). Also see MAXOUT, below. Maximum number of fission neutron source (outer) iterations specification 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 (outer) iterations will be terminated on the n^{th} outer iteration, unless the iterations are terminated earlier by meeting the convergence criterion imposed by EPS1 (F-0,0,1,0).
0	0	50	0	NEXT0	Fission neutron source extrapolation technique specification 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 $f_k^t = f_k^{t-1} + \theta(f_k^{t-1} - f_k^{t-2})$, where the θ 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 $f_k^t = f_k^{t-1} + \alpha_2(f_k^{t-1} - f_k^{t-2}) + \beta_2(f_k^{t-1} - f_k^{t-2})$. See Section II-2.3.
0	0	51	0	NPI0	Specification of the number of preliminary unextrapolated fission source iterations performed before beginning Chebyshev acceleration 0 or blank causes NPI0 to be automatically set to 4. n causes $n+1$ fission neutron source distribution 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 NPI0=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	Specification of 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 λ from 1 through no more than $n+1$ before beginning again with $\lambda=1$. See NEXT0, above.
0	0	53	0	NERCHK	Specification of 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	54-60	0		Note: A new Chebyshev extrapolation series starting with $\lambda=1$ is also begun (with the most recent estimate of the dominance ratio σ_0) if λ exceeds NCHMAX+1. See NCHMAX, above. Ignored

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	61	0	ISERCH	Type of search parameter specification 0 or blank specifies that no search is to be made. ±1 specifies that a "buckling search" is to be made. For use in all the groups, a single value of the perpendicular buckling squared B^2 is to be found which, when used in the search materials, causes the search convergence criteria to be met. The search parameter is the value of B^2, and the first value of B^2 tried is 1) the B^2 for the first group of the material in region 1 if all materials are to be searched, 2) the B^2 for the first group of the first material appearing in the LRS(K) (I-0,3,K,0) list if all materials are not searched, or 3) .01 if the value of B^2 picked up as in 1) or 2) is found to be zero or left blank. ±2 specifies that a "poison search" is to be done; that is, the poison parameter SIGPT(IM) (F-0,4,IM,0) is to be changed by the same multiplier in each search material until the search criteria are met. The search parameter value printed out is the value of SIGPT(IM) for the material of the first region if all materials are searched; and it is the value of SIGPT(IM) for the first material in the LRS(K) list, otherwise. ±3 specifies that a "dimension search" is to be done; that is, the region width RW(IR) (F-0,1,IR,0) is to be changed by the same multiplier in each search region until the search criteria are met. The search parameter value printed out is the width of either the first region or the first region in the LRS(K) list if LRS(1) ≠ 0. ±4 specifies that a "fission source search" is to be done. The value of the total number of neutrons produced per fission, FISSOR (F-0,0,9,0), to which the fission neutron source is renormalized at the beginning of each outer iteration, is to be changed until the generation factor or eigenvalue is close enough to unity to satisfy convergence criteria. (This relates to inhomogeneous problems, only, such as those with fixed volume sources or boundary sources.) The search parameter is FISSOR. ±5 specifies that a "boundary search" is to be performed. The x position of the boundary between two specified adjacent regions is to be varied until the search convergence criteria are met. Following the first search trial, the number of mesh intervals in each of the two regions is adjusted to make the interval widths in each region as close to equal as possible. The search is terminated if a region width less than the input word WMI (F-0,0,12,0) is required. The search parameter printed out is the width of the leftmost region. ±6 specifies that a "concentration search" is to be performed. The input concentrations, CON(K,IM) (F-11,IM,K,0), of the constituents named in the NUCSCH(K) list (I-0,13,K,0) are varied until the search criteria are met. If an ISUMS(K) list (I-0,14,K,0) is input, the sum of the concentrations of all constituents named in the list is held constant during the search. The search parameter printed is the multiplier of the concentrations of the NUCSCH(K) constituents. Note: The search convergence criteria to be satisfied are: first, the parameter values (the buckling, the poisoning, etc.) attained by the search must result in a generation factor or eigenvalue that is equal to the desired value EIGEND (F-0,0,14,0) to within the fractional error specified by EPS2 (F-0,0,2,0), and second, any parameter value itself must change by less than the fraction specified by EPS3 (F-0,0,3,0) from one search trial to the next; but this latter criterion need be met only if the ISERCH value is negative. Also, see fixed point input LRS(K) (I-0,3,K,0) for information on the specification of the search materials or search regions when ISERCH=1, 2, 3, 5, or 6. Following a search problem, the newly found values of the search parameter(s) are retained and can be used in a successive problem. The exception is that when INBUCK is equal to 1, the bucklings used in a problem are reset to values specified by the BSQR(IM) data, and the bucklings computed in a buckling search on the previous problems are destroyed. Following a concentration search problem the new macro material cross sections are retained, but the mixture data MIX(K,IM) and CON(K,IM) are reset to zero, as usual.

Table II (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	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 trials 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	0 0	64,65 66	0 0	LIBOLD	Ignored Identification number of library material data set saved from a previous computer run For each computer run, a library material data set must exist, either as saved from a previous run or written or updated on a current run with library material card input decks. The first record written on a library data set is the word LIBND, and this word serves as user's data set identification number. 0 or blank indicates that no library data set from a previous computer run exists, and LIBOLD is automatically set equal to LIBNEW below for succeeding problems. n > 0 indicates that at the beginning of the run, LIBOLD is to be compared to LIBND of the existing library data set 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 a library data set either newly written or updated on a current problem 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.
0	0	68	0	LIBMTL	Nonzero causes the new library material data set identification number (LIBND) to be set equal to LIBNEW regardless of the value of LIBOLD. The number of library material data decks which are to be read on the current problem A library material deck consists of a library material title card and one or more floating point data cards containing neutron group constants. 0 or blank indicates no library material data decks are to be read on the current problem. n indicates n library material decks are to be read immediately following the floating point data of the current problem.
					Note: Whenever LIBMTL is nonzero and new library material data are read from cards, this causes any existing library data set to be completely rewritten as the same data set with the new data incorporated. Therefore, care should be taken to avoid changing library data sets unintentionally.

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	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 IR th region; the next word pertains to the region number IR+1, etc. Each macroscopic material referred to by number in this data list MR(IR) on a current problem must be either 1) the same material as that used on an immediately preceding problem or 2) a material formed by mixing a set of constituent materials (either library materials from the tape or macroscopic materials used on the preceding problem) as specified by the mixture specification data MIX(K,IM) (I-1,IM,K,0) and CON(K,IM) (I-1,IM,K,0). A macroscopic material is considered "used" on a problem only if it appears in the MR(IR) list of the problem.
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 IR th 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 $IMAX = 1 + \sum_{IR=1}^{NI(IR)} NI(IR)$ where IMAX is the number of regions (less than or equal to 50). See the limitation on IMAX in the discussion for JMAX (I-0,0,1,0).
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 1st and 2nd 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 K th 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 macroscopic materials to which the search applies (all materials 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 concentration search, LRS(K) is the list of macro materials in which the concentrations are to be changed for each of the constituents named in the NUCSCH(K) list (I-0,13,K,U).

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	9	K (list word number)	0	IEBK(K)	Energy group break point specification for the groupwise coalescing. Each word in the IEBK(K) list is an energy group number. The value of Index 3 (K) indicates that the first data word on the card is the Kth word in the IEBK(K) list; and the next word on the card is the K+1 word in the list, etc. If the solution of a problem is to be followed by a calculation to obtain average neutron group constants for a new set of energy groups and regions which is comprised of coalesced energy groups and regions of the original problem, then there must be a nonzero list of group break points IEBK(K). K0ALES (1-0,0,9,0) must also be 1, and the IEBK(K) list, below, must be nonzero. The IEBK(K) list is interpreted as follows: the first new energy group is comprised of the coalesced original energy group(s) 1 through IEBK(1); the second energy group is comprised of the coalesced original energy group(s) IEBK(1)+1 through IEBK(2); etc; etc. The IEBK(K) list is terminated if a zero or blank word in the list is encountered.
0	10	K (list word number)	0	IRBK(K)	Region break point specification for the regionwise coalescing. Each word in the IRBK(K) list is a region number. The value of Index 3 (K) indicates that the first data word on the card is the Kth word in the IRBK(K) list; the next word on the card is the K+1 word in the list, etc. If the solution of a problem is to be followed by a calculation to obtain average neutron group constants for a new set of regions and energy groups which is comprised of coalesced regions and energy groups of the original problem, then there must be a nonzero list of region breakpoints IRBK(K). K0ALES (1-0,0,9,0) must also be 1, and the IEBK(K) list, must be nonzero. The IRBK(K) is interpreted as follows: the first new region is comprised of the coalesced original region(s) numbered 1 through IRBK(1); the second new region is comprised of the coalesced original region(s) numbered IRBK(1)+1 through IRBK(2); etc; etc. The IRBK(K) list is terminated if a zero or blank word in the list is encountered.
0	12	K (list word number)	0	ISPECT(K)	List of desired mesh points for flux spectrum edits. If ISPECT(1) = 0 the flux spectrum edits are suppressed. If ISPECT(1) > 0 the flux spectrum is computed and edited for every mesh point specified in the list. The list must be compact, for it is considered terminated when a zero value of ISPECT(K) is encountered. The K value in Index 3 indicates the first input word on the card is ISPECT(K), the next is ISPECT(K+1), etc. Note: $K \leq 25$.
0	13	K (list word number)	0	NUCSCH(K)	The spectrum calculation uses DELE(J) data (F-0,15,J,0). See Section III-5.3. List of constituent materials to which a concentration search is applied. If ISERCH = ±6 (1-0,0,61,0) a concentration search adjusts the concentration of each constituent material named in the NUCSCH(K) list by a common factor until the search is satisfied. The constituents named can be either library materials or macro materials saved from the previous problem. However, the search is applied in a given macro material of a problem only to those constituents that are named in the mixture list MIX(K,IM) (1-1,IM,K,0) as well as the NUCSCH(K) list. That is, the search skips any macro material for which MIX(1,IM) = 0. The search is also restricted to the macro materials listed in LRS(K) (1-0,3,K,0), if LRS(1) > 0. See Section II-3.3. The K of Index 3 indicates the first input word on the card is NUCSCH(K), the next is NUCSCH(K+1), etc. See, also, ISUMS(K), following. Note: $K \leq 20$.

Table II (Continued)

Index 1 Card Cols. 2-3 Format I 2	Index 2 Card Cols. 4-5 Format I 2	Index 3 Card Cols. 6-8 Format I 3	Index 4 Card Cols. 9-10 Format I 2	Mnemonic Name of Data	Notes on Application of Data
0	14	K (list word number)	0	ISUMS(K)	List of constituent materials to which the constraint on the "sum of the concentrations" applies during a concentration search. When a concentration search (ISERCH = ±6) is applied to a macro material, the concentrations of all those constituents of the material [MIX(K,IM)] that are also named in the NUCSCH(K) list, above, are multiplied by a common factor (the search parameter modifier). To account for the fact that varying the concentrations (volume fractions) of one set of nuclides can simultaneously require adjustment of the volume fractions of another set, the user can list both sets in the ISUMS(K) list of constituents. Then, in a given material, as the concentrations of the NUCSCH(K) constituents are changed, the concentrations of those constituents of the ISUMS(K) list that are not named in the NUCSCH(K) list are multiplied by a common factor to adjust the sum of the concentrations of all the constituents in the ISUMS(K) list to the original value. In applying the constraint in a given macro material, members of the ISUMS(K) list that are not members of the MIX(K,IM) list of the material are ignored. See Equations (89) through (91). The K of Index 3 indicates the first input word on the card is ISUMS(K), the next is ISUMS(K+1), etc. Note: $K \leq 20$.
0	15	K (list word number)	0	IXLST(K)	List of macro materials edited when INED1 ≠ 0. When the input edit flag INED1 ≥ 1 (I-0,0,21,0) the editing of the group constants for all the macro materials, as mixed, is suppressed. This suppression can be overridden for a specified list of macro materials by naming them in the IXLST(K) list. The K of Index 3 indicates that the first input word on the card is IXLST(K), the next is IXLST(K+1), etc. Note: $K \leq 20$.
1	IM (macro. mtl. number)	K (list word number)	0	MIX(K,IM)	The list of constituent materials to be mixed to form macroscopic material IM. If the MIX(K,IM) list is zero (that is, specifically, MIX(1,IM) is zero or blank), then no mixing is done and the data for macroscopic material IM as used on the immediately preceding problem is carried over to the current problem. If MIX(1,IM) is zero and the current problem is a new one or if the macroscopic material was not in the MR(IR) list (I-0,1,IR,0) of the preceding problem, then an error results. For the macroscopic material number IM specified as Index 2, the words in the MIX(K,IM) list specify, by number, which library materials from the library data set and/or macroscopic materials (as used in an immediately preceding problem) are to be mixed to form the material. There can be up to 50 materials listed here as mixture constituents, that is, $1 \leq K \leq 50$. Library materials from the library data set are referred to as positive numbers in the list. (Library materials are numbered 1 through 999.) Macroscopic materials used on the preceding problem can be used as mixture constituents by referring to them in the list as negative numbers. Macroscopic materials are numbered 1 through 20. The value of Index 3 (K) indicates that the first input word in the card is the K th word in the mixture list MIX(K,IM) for macroscopic material IM; the next word on the card is the K+1 word in the list, etc. The list is terminated if a zero or blank word in the list is encountered. A macro material is mixed using the CON(K,IM) (F-11,K,IM,0) data in the weighting formulas of Section II-1.2. Note: Both the MIX(K,IM) and CON(K,IM) lists are set to zero at the beginning of each problem so that the data are not available from problem to problem.

Table II (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
2	L (reaction ratio number)	K (list word number)	N (reaction rate type)	MRATE(K,N,L)	Array specifying the library materials and reaction rate types used to compute spatial distributions of reaction rate ratios. The value of L ($1 \leq L \leq 5$) in Index 2 indicates which of five reaction rate ratios the data on the card refers to. The sign of N indicates that the library materials listed on the card are to be used in forming the numerator (+ sign values) or the denominator (- sign values) of reaction rate ratio L. The value of N ($1 \leq N \leq 6$) indicates the type of reaction rate to be computed for the numerator (+N) or denominator (-N), as the case may be. The integers listed in the input data of the card specify the library materials (from the library data set, only) that are to be used in forming the given numerator or denominator of reaction ratio L. The K value in Index 3 indicates that the first word on the card is MRATE(K,N,L), the next word is MRATE(K+1,N,L), etc. Note: $K \leq 15$. The reaction rate types allowed are: Absorption (Σ_a^L used)----- N = 1. Fission (Σ_f^L used)----- N = 2. Capture ($\Sigma_a^L - \Sigma_f^L$ used)----- N = 3. Neutron Production ($\nu \Sigma_f^L$ used)---- N = 4. Power Production ($E^L \Sigma_f^L$ used)----- N = 5. "Transport" Collision (Σ_u^L used)-- N = 6. In forming a reaction rate, the flux-cross section product of a given library material is weighted by its concentration (either as input as CON(K,IM) data (F-11,IM,K,0) or as determined by a concentration search) unless the concentration is zero. Then a unit weighting is used. If a given library material is not a constituent of the macro material at a given space point, its weighting is zero; i.e., it is not allowed to contribute to the reaction rate at that point. See Section III-5.7. By means of these data, the user can cause to be edited, at output time, such things as the spatial distribution of the breeding ratio; i.e., ratio of the capture rate of all the fertile nuclides to the absorption rate of all the fissile ones. The rates corresponding to each numerator and denominator are also edited.

2.3 The Problem 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 problem or change case.

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

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 problem.

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

Table III. Detailed Description of 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. $\text{NC0NVG} = 0$ (I-0,0,48,0): $ (\lambda^t - \lambda^{t-1})/\lambda^t \leq \epsilon_1$ 2a. $\text{NC0NVG} = 1$ and $\text{IREN} = 0$: $\max_i \frac{f_i^{*t}}{f_i^{t-1}} - \min_i \frac{f_i^{*t}}{f_i^{t-1}} \leq \epsilon_1$ (I-0,0,6,0) 2b. $\text{NC0NVG} = 1$ and $\text{IREN} = 1$: $\max_i \left \frac{f_i^{*t}}{f_i^{t-1}} - 1 \right \leq \epsilon_1$
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 $\lambda_d^S \equiv$ 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 IGEN0P (I-0,0,8,0) as follows. $\text{IGEN0P} = 0$: $ (\lambda^S - \lambda_d)/\lambda_d \leq \text{EPS2}$. $\text{IGEN0P} = 1$: $ (\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 of the source-to-loss ratio for multiple thermal neutron energy groups EPS4 = 0 or blank causes it to be set automatically to .0001. When multiple thermal neutron groups exist, inner iterations are performed on the thermal groups, alone, within each outer iteration until either the convergence level specified by EPS4 is satisfied or until the number of inner iterations (per outer) reaches MAXINS (I-0,0,43,0). After each inner iteration u , the source-to-loss ratio R^u is computed. The source is the sum of all boundary sources, fixed volume sources, and scattering-in sources (from above the thermal groups); and the loss is the sum of all thermal group leakage and absorption. The convergence criterion is $ R^u - 1 \leq \text{EPS4}$.
0	0	5	0	EPS5	Convergence level at which acceleration of multiple thermal group flux convergence by source-to-loss multiplication can be terminated As a source-to-loss ratio approaches unity, multiplication of the fluxes by it has a vanishing acceleration effect. See Section II-2.4. When the convergence acceleration flag INC0NV (I-0,0,42,0) is 2 or 3, acceleration by multiplying by total source-to-loss ratio R^u or the groupwise source-to-loss ratio r_j^u , respectively, is terminated when $ R^u - 1 \leq \text{EPS5}$.

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	6	0	RAD1	The position of mesh point number 1, the inner 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_I , where x_1 is the lower bound and x_I the upper bound. The value of RAD1 defines x_I , i.e., $x_I \equiv \text{RAD1}$. 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, $x_I = x_1 + \sum_{IR=1}^{\text{IRMAX}} \text{RW}(\text{IR}) (\text{F}-0,1,\text{IR},0), \text{ and } I = 1 + \sum_{IR=1}^{\text{IRMAX}} \text{NI}(\text{IR}) (\text{I}-0,2,\text{IR},0).$
0	0	7	0	THRMX	The value of the two-point extrapolation factor τ for accelerating multiple thermal flux inner iteration convergence. When INCONV (I-0,0,42,0) is 4 or 5, the inner iterations performed to converge the multiple thermal fluxes are accelerated with the two-point formula of Equation (63). The extrapolation factor τ is equal to THRMX when $\text{INCONV} = 4$. When $\text{INCONV} = 5$, τ is automatically computed with Equations (64) and (65), but THRMX is the first guess τ^1 .
0	0	8	0	THETA	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 is the extrapolated fission neutron source density at source point k following the t^{th} outer iteration. The two-point extrapolation formula is $f_k^t = f_k^{t-1} + \theta(f_k^{t-1} - f_k^{t-2})$, where the * indicates an unextrapolated value. (The source point k in the IR^{th} region at mesh point i is given by $k = i + \text{IR} - 1$.)
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 outer iteration. By setting $\text{IREN} = 0$ (I-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 (39) and (42). This normalizing is done to the source guess before the first outer iteration, and thereafter it is done to the newly computed fission neutron source at the end of each outer iteration. (This normalization procedure is usually desired for all but some inhomogeneous or non-eigenvalue types of problems). When $\text{IREN} = 0$, and FISSOR is 0 or blank, the code sets $\text{FISSOR} \equiv F_0 = 1$ except when F_0 is to be the search parameter in a fission source search problem. In the latter case when $\text{ISERCH} = \pm 4$ (I-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_{0S} . Subsequently, $\text{FISSOR} = F_{0S}$ is changed in the course of the search. See Section II-3.3. When $\text{IREN} = 1$, the volume integral of the fission neutron source is set equal to FISSOR at the beginning of the first outer iteration, only.
0	0	10	0	FLMULT	Optional multiplier of all flux, source, and power output data. When the fixed point data word NORMAL (I-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	PPOWER	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 (I-0,0,36,0) is set equal to 1, an unnormalized power distribution p' is first computed from the unnormalized fluxes as $p' = \frac{1}{\lambda t} \sum_j E_j \Sigma_f \phi_j$, where the primes denote the unnormalized values. Then the normalizing factor C is computed from $C = P / \int_V dv p'$, where $P \equiv \text{PPOWER}$. The output neutron source, power, and fluxes are then $C f^{*t+1}$, $C p'$, and $C \phi_j'$, respectively. See Section III-5.2.

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	0	12	0	WMI	The minimum region width allowed before terminating a boundary search problem. If WMI is 0 or blank, it is set automatically to .001. When a boundary search is specified (that is, ISERCH = ±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 WMI in order to satisfy the search criteria. The first time the search attempts to use a region width less than WMI, the region width is set, instead, to the value of WMI and the search is continued. However, the second time the search attempts to use a region width smaller than WMI, 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. Ignored
0 0	0 0	13 14	0 0	EIGEND	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 (I-0,0,63,0), or 2) ISERCH = ±4 (I-0,0,61,0). EIGEND is set automatically to the eigenvalue λ of the preceding problem if NSERCH = 1 and ISERCH ≠ ±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 $E_j \equiv \text{ENERGY}(J,IM)$ for group $j \equiv J$ of library material number $\lambda \equiv LM$. This energy per fission value for every group of a library material $\lambda \equiv LM$ can be set equal to EPFISS at the time the library material deck is read or updated by setting the word IENPF equal to 0 or blank on the library material title card. See Section 2.4, The Library Material Title Card for IENPF, and see floating point library material data, for $\text{ENERGY}(J,IM)$ (LF-1,L,J,5).
0 0	0 0	16-19 20	0 0	FIXSRS	Ignored Volume integral for optional normalization of the pointwise part of the fixed volume source. When the flag NRMFIX = 1 (I-0,0,16,0), the pointwise part S_{pj} of the total fixed volume source, $S_{ej} = S_{pj} + S_{rj}$, is normalized such that $\int dv \sum_j S_{pj} = \int dv N_f \sum_j S'_{pj} = \text{FIXSRS}$ where N_f is the appropriate normalizing factor needed to multiply the unnormalized fixed source S'_{pj} . The S_{pj} data are the $SE\theta(K,J)$ data (F-10,J,K,0), either as input or automatically computed when ISQRS > 0 (I-0,0,15,0).
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 IRth region, the next word pertains to the IR+1 region, etc.
0	2	IM (macro. mtl. number)	0	BSQR(IM)	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 (I-0,0,61,0). Optional specification of buckling values for macroscopic material IM 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 IM given as Index 3; the next data word pertains to material IM+1, etc. These buckling squared data, as read in, remain available from problem to problem regardless of buckling searches, changes in the other buckling data BSQ(J,IM) (F-6,IM,J,0), etc. If INBUCK = 1 (I-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 BSQ(J,IM) data are used instead.)

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	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 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. mtl. 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 $\gamma_{jm}^{pm} = T(J,IM) \text{ SIGPT}(IM)$. However, unlike the ordinary absorption cross section, this poisoning absorption can cause automatic modification of the diffusion coefficient during poison searches as discussed under IDC0 (I-0,0,14,0), when IDC0 equals 1 or 2. See Section II-1.2.
0	5	1	J (group number)	0MEGA1(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 $j = J+1$, etc. If the inner boundary condition indicator NIBDRY (I-0,0,4,0) is 2, 3, or 4, the value of $\omega_{1j} \equiv 0\text{MEGA1}(J)$ 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)	0MEGA2(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 $j = J+1$, etc. If the outer boundary condition indicator N0BDRY (I-0,0,5,0) is 2, 3, or 4, the value of $\omega_{2j} \equiv 0\text{MEGA2}(J)$ is used in defining the boundary condition for the $j \equiv J$ th energy group. See the discussion under N0BDRY.
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, $J_j(x_1) = 0$.

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	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 NØBDRY = 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 NØBDRY = 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 NØBDRY = 8 (the periodic boundary condition), the outer boundary condition is defined by the equation $\alpha_{2j}\phi_{jI} - \beta_{2j}J_{jI} + \gamma_{2j} = 0$, where ϕ_{jI} and J_{jI} 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, $J_{jI}(x_I) = -1.8$.
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 K th 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, $K = I + IR - 1$. If F(K) is input on cards, the value of INSØRS (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 FISSØR (F-0,0,9,0). See the discussion for FISSØR 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 $\phi_{jj} \equiv \text{PHI}(I,J)$ in energy group $j \equiv J$ at mesh point $i \equiv I$, the next word is the flux at mesh point $I+1$, etc. Flux data read on these cards can be used to provide a flux guess from which to start a multiple thermal group problem. In this case, INFLUX (I-0,0,41,0) should be 0 or blank and the corresponding pointwise fission source should also be input (see F(K) above). These input fluxes can also be used to compute automatically a fixed volume source by various means. See fixed point data word ISØRS (I-0,0,15,0). The flux values obtained in a previous 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 flux deck is followed by the punched fission neutron source deck, in this case.

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	14	J (group number)	0	EBAR(J)	The average energy of neutron energy group J The first input word on the card is the average energy $\bar{E}_j \equiv \text{EBAR}(J)$ ($j \equiv J$) of energy group J, the next word is the value for group J+1, etc. These average energies are used to compute the spatial distributions of the average energies of the neutron density, neutron flux, fission rate, absorption rate, and capture rate for output editing as discussed in Section III-5.4.
0	15	J (group number)	0	DELE(J)	The energy width of neutron energy group J The first input word on the card is the energy width $\Delta E_j \equiv \text{DELE}(J)$ ($j \equiv J$) of energy group J, the next word is the value for group J+1, etc. These energy widths are used to compute the flux spectra and the spatial distributions of median energies for the neutron density, flux, fission rate, absorption rate, and capture rate. The output editing of these data is discussed in Sections III-5.3. and 5.4.
0	16	J	0	VRECIP(J)	The average of the reciprocal of the neutron velocity of group J The first word on the card is the average reciprocal velocity $\left(\frac{1}{v}\right)_j \equiv \text{VRECIP}(J)$ ($j \equiv J$) for group J, the next word is the value for group J+1, etc.
1	IM (macro. mtl. number)	J (energy group number)	1	DC(J,IM)	These data are used in computing the average and median energies of the of the neutron density for output editing as discussed in Section 5.4. Optional specification of the diffusion coefficient for group J of macroscopic material IM The first data word on the card is the value of the diffusion coefficient for group J, the next word is the value for group J+1, etc. As noted in the discussion in Section II-1.2. and the discussion in Section IV-2.5. for the library input data DC(J,IM) (LF-1,L,J,0), the value of D_j^m , the diffusion coefficient, as used in a problem for group $j \equiv J$ of macroscopic material $m \equiv \text{IM}$ is usually computed by first calculating a Σ_{uj}^m value as a mixture of a given set of materials. The D_j^m is then computed from this Σ_{uj}^m value according to whatever formula is specified by the value of IDC0 (I-0,0,14,0). However, if the value of Σ_{uj}^m , as mixed, is found to be equal to zero, then it is assumed that the Σ_{uj}^m is to be computed instead from the macroscopic DC(J,IM) value that is input here. The Σ_{uj}^m is computed as $\Sigma_{uj}^m = 1/[3\text{DC}(J,\text{IM})]$, in this case. Thereafter, the value of D_j^m is calculated as a function of this Σ_{uj}^m and the value of T_{jp}^m (the poison cross section) just as before. Hence, when $\Sigma_{uj}^m = 0$ and IDC0 = 0, the D_j^m values computed are exactly equal to the DC(J,IM) values input here. It should be emphasized, however, that the D_j^m values finally used in a problem are not, in general, equal to the input DC(J,IM) values when IDC0 = 1 or 2. In the latter cases these DC(J,IM) values are used only to compute the Σ_{uj}^m from which D_j^m is then computed. Note: D_j^m is the only neutron group constant that can be determined either as a function of mixed constituent material data or as direct input for the macroscopic material.

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
5	IM (macro. mtl. 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 number $m \equiv IM$. 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 number $m \equiv IM$. Hence, if the total fission neutron source density is f , then the product $x_j^m f$ is the fission source to group j , Sf_j , as defined in Equation (28). 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.
6	IM (macro. mtl. number)	J (energy group number)	0	BSQ(J,IM)	The buckling squared values for energy group J of macroscopic material number IM . 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. The buckling squared values B^2 are set by the $BSQ(J,IM)$ 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.
7	IM (macro. mtl. number)	J (energy group number)	0	T(J,IM)	The multiplier T_j^m of the macroscopic poison cross section $\Sigma_p^m \equiv SIGPT(IM)$ for energy group number $j \equiv J$ of macroscopic material number $m \equiv IM$. The first data word on the card is the value of the fraction T_j^m of Σ_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. 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.
8	IM (macro. mtl. number)	J (energy group number)	0	Q(J,IM)	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 macroscopic material $m \equiv IM$ appears. 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. 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 $\frac{1}{\phi_{jib}} \frac{d\phi_{jib}}{dx} = -q_j^m \text{ at the left (lower) boundary of the region and}$ $\frac{1}{\phi_{jib}} \frac{d\phi_{jib}}{dx} = +q_j^m \text{ at the right (upper) boundary of the region.}$ 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. 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 <u>periodic boundary condition</u> is specified. See Section II-1.6, also. Note: To avoid ambiguity concerning any intended meaning, a nonzero value of q_j^m for group j and material m causes all the neutron group constants (D_j , χ_j , $v\Sigma_{fj}$, Σ_{fj} , Σ_{aj} , Σ_{ujtj} , Σ_{p1} , $\Sigma_{sj'+j}$, and $\Sigma_{sj+j'}$) associated with the j th neutron group of that material to be set to zero.

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
9	IR (region number)	J (energy group number)	0	SEØR(J,IR)	The value of the flat fixed volume neutron source density S_{rj} in energy group J for the IRth region. 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. 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. The fixed volume source density, S_{ej}, applied in a problem is always the sum of the SEØR(J,IR) values and the pointwise variable fixed sources, S_{pj}, specified by SEØ(K,J), below.
10	J (energy group number)	K (source point number)	0	SEØ(K,J)	The pointwise (space dependent) fixed volume neutron source density S_{pj} for energy group number J at source point K. 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. These input data facilitate specification of a pointwise variable fixed volume neutron source where it is necessary. The fixed volume source density S_{ej} applied in a problem is always the sum of the SEØ(K,J) values and the regionwise constant fixed sources specified by SEØR(J,IR), above. There is an option to compute, automatically, the SEØ(K,J) values as a function of existing fluxes and various neutron group constants. See ISØRS (I-0,0,15,0). When this option is used (i.e., ISØRS $\neq 0$), 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.
11	IM (macro. mtl. number)	K (list word number)	0	CØN(K,IM)	Note: It is common for both fission produced neutron sources and fixed volume sources to be double valued at region or material interfaces; that is, at a mesh point on a region boundary, there can be one source intensity on the immediate left side of the point and another intensity (or zero) on the immediate right side. Allowance for these "double values" results in there being an additional point for a source at each mesh point on a region interface. (There are, of course, IRMAX - 1 region interfaces, where IRMAX is the number of regions in the problem.) Therefore, the source points are indexed by a number K that is related to the mesh point Index I by the equation $K = I + IR - 1$ when the source is in region IR. The list of concentrations or weighting factors C_k^m of the constituent materials to be mixed to form macroscopic material number $m \equiv IM$. The value of Index 3 (K) indicates that the first data word on the card is the Kth word in the concentration (or weighting factor) list for macroscopic material IM, the next data word is the $K+1$ word in the list for the material, etc. The value of Kth word in the CØN(K,IM) list is used to weight the data (neutron group constants) of the constituent material named (by its number) as the Kth word in the MIX(K,IM) (I-1,IM,K,0) list in order to form macroscopic material IM. See Section II-1.2. There can be from 1 to 50 values given in the CØN(K,IM) list for material IM. Both the CØN(K,IM) and MIX(K,IM) are set to zero at the beginning of each problem before any input data are read. Also, the CØN(K,IM) data are changed during the solution of a concentration search problem. See Section II-3.3. The final values obtained by the search can be edited by setting INED2 = 1 (I-0,0,22,0).

2.4 The Library Material Title Card

The card format is I1,2I2,I3,2I2,10A6.

The first 12 columns of a library material card are devoted to material identification data and information controlling how the words on the library material data cards are to be handled and interpreted.

Column 1		This is always blank.
Columns 2-3	IAD	Zero or blank means the input data read for this library material are new or completely replaces the data on the library data set for any material of the same library material number L 1 punched means the material input data read in simply change, or update, the data for the material that were stored previously on the library data set and only those data to be changed need be read in.
Columns 4-5	IDIFF	Zero or blank means the values of $\Sigma_{\mu j}^L$ (used to compute diffusion coefficients) for group $j \equiv J$ and library material $L \equiv L$ will be read in, as normal. 1 punched means $\Sigma_{\mu j}^L$ will be computed as $1/3D_j^L \equiv 1/3[DC(J,L)]$ where $DC(J,L)$ values are the library material diffusion coefficients read in directly.
Columns 6-8	L	This is the library material number that identifies the material data. This number L must be greater than that of any preceding library material deck.
Columns 9-10	IENPF	Zero or blank means that the energy per fission E_f^L used in all groups for the material will be set equal to EPFISS (F-0,0,15,0), a value read in with the problem floating point data. 1 punched means that the energy per fission used in all groups will be set equal to $ENERGY(1,L)$, the energy per fission value read in with this library material data as the value for the first group. 2 punched means that the energy per fission used in each group of this library material will be read in directly with the data for the material.
Columns 11-12		These are always blank.
Columns 13-72		Any Hollerith characters can be punched here to further identify the material.

2.5 The Library Material 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 problem floating point data cards in Section 2.3, above. Following the library material title card, there must be at least one floating point data card per library material 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.

The macro materials used in a problem are those that appear in the MR(IR) (I=0,1,IR,0) list of the problem fixed point data. The IRth number given in the MR(IR) list is the number m of the macro material that is to be used in the IRth region of the problem. The macro materials are numbered 1 through 20.

Some macro material data are read with the problem floating point data. These data include the buckling data B_j^{2m} ; the poison cross section data T_j^m and Σ_p^m ; the fission spectrum fractions x_j^m ; and the logarithmic boundary condition parameters q_j^m .

The remaining macro material data for the neutron group constants are formed by "mixing" the data of a set of constituent materials. These mixed macro data include the diffusion coefficient data Σ_{uj}^m and D_j^{m*} ; the absorption cross sections Σ_{aj}^m ; the neutrons per fission times the fission cross sections $\nu\Sigma_{fj}^m$; the fission cross sections Σ_{fj}^m ; the energy per fission data E_j^m ; and the scattering transfer cross sections $\Sigma_{sj\rightarrow j'}^m$. See the mixing formulas of Section II-1.2.

Constituent materials can be either library materials or macro materials. However, on the first problem of a run on the computer, all the constituent materials must, of course, be library materials.

On any problem after the first problem of a given computer run, a constituent material can be the data of a macro material m' used (that is, named in the MR(IR) list) in the previous problem.

It is permissible for a macro material m to be a mixture containing the macro constituent material m', even for m=m'.

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*Whenever the mixed value of Σ_{uj}^m is zero, an alternate value is computed from the formula, $\Sigma_{uj}^m = 1/[3DC(J,IM)]$, where the DC(J,IM) (F-1,IM,J,0) value is as input with the problem floating point data for group $j \equiv J$ of macro material $m \equiv IM$. (See Section 2.3.) The D_j^m values used in the problem, however, remain the usual functions of Σ_{uj}^m and $T_j^m \Sigma_p^m$. Hence, the D_j^m value used in the problem can be equal to DC(J,IM) only when IDC0 = 0. When IDC0 equals 1 or 2, D_j^m is a function of other neutron group constants in addition to the DC(J,IM) values, as input.

Library materials are those stored on the library data set. There can be up to 999 library materials, and the library material numbering is completely independent from the macro material numbering. The data for library material ℓ consist of the diffusion coefficient data Σ_{Dj}^{ℓ} and D_j^{ℓ} ; the absorption cross sections Σ_{aj}^{ℓ} ; the neutrons per fission times the fission cross sections $\nu\Sigma_{fj}^{\ell}$; the fission cross sections Σ_{fj}^{ℓ} ; the energy per fission data, E_j^{ℓ} ; and the scattering transfer cross sections $\Sigma_{sj \rightarrow j'}^{\ell}$. The ℓ superscript denotes a library material, as opposed to a macro material which is denoted by the superscript m .

The data for a library material must first be placed on the library data set before they can be available for mixing in a given problem. The data of a given library material read in immediately following the problem floating point data of a given problem are placed on the data set and are available for mixing in the given problem and on all subsequent problems.

Once the data of a library material are on the library data set, they can be replaced by a completely different set of data for a library material of the same number or they can be "updated" by reading in a deck for the material that changes only part of the material data. See data word IAD in Section 2.4, The Library Material Title Card.

Library materials on the library data set must be written in the order of increasing library material number, but the numbers need not be successive integers. For example, the materials on a tape might be 6, 7, 10, 93, 95, and 980. The library material number for a given library material deck is defined by the value of L on the library material title card. (The value of Index 2 (L) on the library material floating point data cards is ignored in a MØNA problem.)

The library material data decks read with a given problem must be stacked so that the library material numbers are in increasing order (the data deck for the lowest material number is read first, etc.).

The total number of library material decks read following the data of a given problem must agree with the value of LIBMTL (I-0,0,68,0) of that problem.

Examples of Mixing

Example 1. In a given problem, the mixture specification, as defined by the MIX(K,IM) and CØN(K,IM) lists for the macro material $m \equiv IM = 3$, might be as follows.

K (list word number)	1	2	3	4	5
MIX(K,3)	5	2	4	902	30
CØN(K,3)	.05	$3.1 \cdot 10^{23}$	$.0045 \cdot 10^{22}$	10.3	.387

These data cause macro material number 3 to be formed by mixing library materials 5, 2, 4, 902, and 30. The mixture is formed by weighting each cross section or product (Σ_{uj}^{ℓ} , Σ_{aj}^{ℓ} , $\nu\Sigma_{fj}^{\ell}$, Σ_{fj}^{ℓ} , $\Sigma_{sj \rightarrow j}^{\ell}$, or $E_j^{\ell}\Sigma_{fj}^{\ell}$) of library material ℓ in the MIX(K,IM) list by the corresponding value of CØN(K,IM). In this example, the macroscopic absorption cross section for group 2 is computed by the formula,

$$\Sigma_{a2}^{m=3} = (.05)\Sigma_{a2}^{\ell=5} + (3.1 \cdot 10^{23})\Sigma_{a2}^{\ell=2} + (.0045 \cdot 10^{22})\Sigma_{a2}^{\ell=4} + (10.3)\Sigma_{a2}^{\ell=902} + (.387)\Sigma_{a2}^{\ell=30}$$

Note: D_j^m is always computed from the value of Σ_{uj}^m according to the formula specified by the IDCØ (I-0,0,14,0) value. E_j^m , the average energy per fission, is computed from the mixed value of the product ($E_j^m\Sigma_{fj}^m$) and the mixed value of Σ_{fj}^m according to the formula, $E_j^m = (E_j^m\Sigma_{fj}^m) / \Sigma_{fj}^m$. Also, the energy per fission values E_j^{ℓ} of a library material can be set automatically rather than read directly. See IENPF in Section 2.4, The Library Material Title Card, and EPFISS (F-0,0,15,0).

Example 2: In a problem which follows another one in a given computer run, the mixture specification lists might be as follows for macro material 19.

K	1	2	3	4	5
MIX(K,19)	6	-19	1	19	-5
CØN(K,19)	$2.5 \cdot 10^{23}$	.30	.0087	$1.7 \cdot 10^{20}$	.04

The data for macro material 19 on this problem are obtained by weighting the cross sections of library materials 6, 1, and 19 by the factors $2.5 \cdot 10^{23}$, .0087, and $1.7 \cdot 10^{20}$, respectively, and then weighting the cross sections of the macro materials 19 and 5 (as they existed on the preceding problem) by the factors .30 and .04, respectively. The cross section $\Sigma_{a2}^{m=19}$ in this case would be given by

$$\Sigma_{a2}^{m=19} = (2.5 \cdot 10^{23})\Sigma_{a2}^{\ell=6} + (.30)\Sigma_{a2}^{m'=19} + (.0087)\Sigma_{a2}^{\ell=1} + (1.7 \cdot 10^{20})\Sigma_{a2}^{\ell=19} + (.04)\Sigma_{a2}^{m'=5}$$

At the beginning of each problem of a computer run, every word in the MIX(K,IM) list and the CØN(K,IM) lists are automatically set to zero. Upon detecting a zero value for MIX(1,IM), the code assumes that the data for macro material IM are the same as on the preceding problem. This also means that if a macro material is not exactly the same as it was on the previous problem, entire new lists [MIX(K,IM) and CØN(K,IM)] must be read in for the material.

Table IV. Detailed Description of the Library Material Floating Point 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		
1	*L	J (energy group number)	0	DC(J,L), SIGA(J,L), SIGNU(J,L), SIGFIS(J,L), ENERGY(J,L)	The values of the diffusion coefficient D_0^j, absorption cross section $\Sigma_{a,j}$, neutrons per fission times the fission cross section $\nu\Sigma_{f,j}$, the fission cross section $\Sigma_{f,j}$, and the energy per fission E_j^x for energy group $j \equiv J$ of library material number $L \equiv L$. The first data word on the card is DC(J,L), the next is SIGA(J,L), etc., in the order given in the list at left under the Mnemonic Name of Data. The word DC(J,L) on a card is not used if, on the library material title card, the value of the word IDIFF (card columns 4-5) is 0 or blank. In this case, the $\Sigma_{u,j}^x \equiv \text{SIGMU}(J,L)$ (LF-4,L,J,0) values for the library material $L \equiv L$ are assumed to be those read in directly as outlined below. On the other hand, if IDIFF=1 on the library material title card, all the $\Sigma_{u,j}^x \equiv \text{SIGMU}(J,L)$ values for the material are computed from the DC(J,L) input values by the formula $\Sigma_{u,j}^x = 1/3DC(J,L)$, and these values replace any SIGMU(J,L) (LF-4,L,J,0) values read in. $\Sigma_{u,j}^x$ is used in the mixing calculations to obtain the Σ_m^m value for macroscopic material m, and the D_0^m values for the macroscopic material m are normally computed from the mixed cross section Σ_m^m [as determined by the value of the problem fixed point word IDCP (1-0,0,14,0)]. However, if a $\Sigma_{u,j}^x$ value calculated by this mixing process is zero, it is assumed that the value DC(J,IM) (F-1,IM,J,1) for group $j \equiv J$ of macroscopic material $m \equiv IM$ (which is optionally input as problem floating point data) is to be used to compute the $\Sigma_{u,j}^m$ value by the formula $\Sigma_{u,j}^m = 1/3D_0^m$. See Section 2.3 for a discussion of the optionally input diffusion coefficient data DC(J,IM) (F-1,IM,J,1). See Section 2.4, The Library Material Title Card; data word IENPF, for optional ways of setting $E_j^x \equiv \text{ENERGY}(J,L)$. Optional specification of diffusion coefficient data D_0^j for group $j \equiv J$ of library material $L \equiv L$. An Index 4 value of 1 indicates that the first data word on the card is the value of DC(J,L) for group J, the next word is the value for group J+1, etc. The DC(J,L) values can be specified by the cards of either this format or the format corresponding to (LF-1,L,J,0) above. Cards of both types can be used in the same deck for a given library material. If two values of DC(J,L) are read (for the same energy group J) the value on the last card read is used. See the discussion above for general information on how the DC(J,L) data are used.
1	*L	J (energy group number)	1	DC(J,L)	Optional specification of the absorption cross section $\Sigma_{a,j}$ for group $j \equiv J$ of library material $L \equiv L$. With this format (with an Index 4 value of 2) the first data word on the card is the value of $\Sigma_{a,j}$ for group number $j \equiv J$, the next word is the value for group J+1, etc. The SIGA(J,L) 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 library material. The last card(s) read determine the values used if the same data are read more than once.
1	*L	J (energy group number)	2	SIGA(J,L)	Optional specification of neutrons per fission times the fission cross section $\nu\Sigma_{f,j}$ for group $j \equiv J$ of library material $L \equiv L$. The first data word on the card is the value of $\nu\Sigma_{f,j}$ for group number $j \equiv J$, the next word is the value for group J+1, etc. The SIGNU(J,L) 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 library material. The last card(s) read determine the values used if the same data are read more than once.
1	*L	J (energy group number)	3	SIGNU(J,L)	

*For a MØNA problem, Index 2 is ignored. The library material number is the L specified on the library material title card.

Table IV (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		
1	*L	J (energy group number)	4	SIGFIS(J,L)	Optional specification of fission cross section Σ_{fj}^L for group $j \equiv J$ of library material $L \equiv L$. The first data word on the card is the value of Σ_{fj}^L for group number $j \equiv J$, the next word is the value for group $J+1$, etc. The SIGFIS(J,L) 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 library material. The last card(s) read determine the values used if the same data are read more than once.
1	*L	J (energy group number)	5	ENERGY(J,L)	Optional specification of the energy per fission E_j^L for group $j \equiv J$ of library material $L \equiv L$. The first data word on the card is the value of E_j^L for group number $j \equiv J$, the next word is the value for group $J+1$, etc. The ENERGY(J,L) 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 library material. The last card(s) read determine the values used if the same data are read more than once. See Section 2.4, The Library Material Title Card, data word IENPF, for optional ways of setting $E_j^L \equiv \text{ENERGY}(J,L)$. See also, Section 2.3, The Problem Floating Point Data, data word EPFISS (F-0,0,15,0).
2	*L	J (energy group number)	N	TRMAT(J,J+N)	Optional specification of the scattering cross sections $\Sigma_{sj-j'}^L \equiv \text{TRMAT}(J,J+N)$ for the scattering transfer of neutrons from energy group $j \equiv J$ to group $j' \equiv J+N$ for library material $L \equiv L$. For specifying the scattering cross sections from group J to a group $J+1$, for example, the optional format corresponding to Index 1=2 may be much more convenient than the format of Index 1=3, below. (This format is particularly convenient when there is scattering only from a given group to the next group.) The first data word on the card is the value of the cross section $\Sigma_{sj-j'}^L$ for scattering neutrons from energy group $j \equiv J$ to group $j' \equiv J+N$, the next word on the card is the cross section for scattering from group $j \equiv J+1$ to group $j' \equiv J+N+1$, etc. If it happens that $j \equiv j'$ for some word on a card, the word is ignored.
3	*L	J (energy group number)	K	TRMAT(J,K)	Specification of the scattering transfer cross sections $\Sigma_{sj-j'}^L \equiv \text{TRMAT}(J,K)$ for scattering neutrons from energy group $j \equiv J$ to group $j' \equiv K$ for library material $L \equiv L$. The first data word on a card is the value of the cross section $\Sigma_{sj-j'}^L$ for scattering neutrons from energy group $j \equiv J$ (Index 3) to group $j' \equiv K$ (Index 4), the next word is the value for scattering from group $j \equiv J$ to group $j' \equiv K+1$, the next is for scattering from group $j \equiv J$ to group $j' \equiv K+2$, etc. If it happens that $j \equiv j'$ for some word on a card, the word is ignored. (Hence, the input formats remain compatible with those used in various transport codes that require the "within-group" scattering cross sections $\Sigma_{sj-j'}^L$.)
4	*L	J (energy group number)	0	SIGMU(J,L)	Specification of the cross section $\Sigma_{\mu j}^L$ for energy group $j \equiv J$ and library material number $L \equiv L$. The first data word on the card is the value of $\Sigma_{\mu j}^L \equiv \text{SIGMU}(J,L)$ for energy group $j \equiv J$, the next data word is the value for group $J+1$, etc. Instead of being input directly on these cards with Index 1=4, these values can be automatically computed from the $\text{DC}(J,L)$ values that can be input as described above. In this case, $\Sigma_{\mu j}^L = 1/3\text{DC}(J,L)$.

*For a MØNA problem, Index 2 is ignored. The library material number is the L specified on the library material title card.

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 2.5 for the detailed descriptions of the input data.

Table V. The Problem Title Card Summary

Information or Mnemonic Name of Data	Card Columns	Description or Function
Any Hollerith Characters NEWJOB ILIST NOSOL	1-66 67-68 69-70 71-72	Problem title information A nonzero value indicates a completely new problem A nonzero value causes edits of contents of all input data cards Nonzero values suppress problem solutions and input data edits

See Section 2.1 for more detail.

Table VI. Summary of Problem 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	NGEØ	Geometry indicator: 0-slab, 1-cylinder, 2-sphere	(3)
0, 0, 4,0	NIBDRY	Inner boundary condition indicator: $0-\phi=0$, $1-J=0$, $2-\phi=\omega_1$, $3-\frac{d\phi}{dx}=\omega_1$, $4-\frac{1}{\phi}\frac{d\phi}{dx}=1/\omega_1$, $5-\frac{1}{\phi}\frac{d\phi}{dx}=\frac{1}{.71\lambda_{tr}}=1/((.71)(3D)$, $6-\frac{1}{\phi}\frac{d\phi}{dx}=1/((.71)(3D)(G_1)$, $7-\alpha\phi-\beta J+Y=0$, 8-periodic	(3)
0, 0, 5,0	NØBDRY	Outer boundary indicator: $0-\phi=0$, $1-J=0$, $2-\phi=\omega_2$, $3-\frac{d\phi}{dx}=\omega_2$, $4-\frac{1}{\phi}\frac{d\phi}{dx}=1/\omega_2$, $5-\frac{1}{\phi}\frac{d\phi}{dx}=\frac{1}{.71\lambda_{tr}}=1/((.71)(3D)$, $6-\frac{1}{\phi}\frac{d\phi}{dx}=1/((.71)(3D)(G_2)$, $7-\alpha\phi-\beta J+Y=0$, 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, 2-both	(3)
0, 0, 8,0	IGENØP	Controls how generation factor is computed for searches: 0-usual eigenvalue, 1-fission neutrons per fixed source neutron on first iteration	(3)
0, 0, 9,0	KØALES	Group and region coalescing indicator: 0-no coalescing, 1-coalesce	(3)
0, 0,10,0	IPERT	First ancillary program flag: 0-no IPERT ancillary run, n-run nth IPERT ancillary	(3)
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)
0,0,12-13,0		Ignored	
0, 0,14,0	IDCØ	Specifies type of diffusion coefficient formula used: 0-D = $1/3\Sigma_t$, 1-D = $1/3(\Sigma_t+\Sigma_p)$, 2-D = Σ_r/κ^2	(3)
0, 0,15,0	ISØRS	Specifies type of pointwise fixed volume source automatically computed: 0-none computed, 1-Sp = 0, 2-Sp = $v\Sigma_f\phi$, 3-Sp = $x\phi$, 4-Sp = $\Sigma_f\phi$, 5-Sp = $E\phi$, 6-Sp = $\Sigma_p\phi$, 7-Sp = $\Sigma_a\phi$, 8-Sp = $T\phi$	(3)
0, 0,16,0	NRMFIX	Pointwise fixed volume source normalization flag: 0-no normalization, $1-\int dv N_f \sum_j S_{pj} = \text{FIXSRS}$	(3)
0, 0,17,0		Ignored	
0, 0,18,0	IPLØT	Second ancillary program flag: 0-no IPLØT ancillary run, n-run nth IPLØT ancillary	(3)
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 macro material edits except for those in IXLST(K), 2-also no summed fixed source (S_{ej}) edits, 3-also no input fixed source data (S_{rj} or S_{pj}) edits, 4-no edits	(3)
0, 0,22,0	INED2	Final (after searches) macro material mixture data edit flag: 0-no edit, 1-edit	(3)
0, 0,23,0	INED3	Reserved	
0, 0,24,0		Ignored	
0, 0,25,0	IØUT1	First output data edit control flag: 0-problem summary, 1-plus fluxes and sources, 2-plus peak to averages, 3-plus neutron balances	(3)
0, 0,26,0	IØUT2	Reserved	
0, 0,27,0	IØUT3	Third output data edit control flag: 0-no edits, 1-edit median and average energies, 2-edit constituent material reaction rates by region, 3-edit both	(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 nth line per region	(3)
0, 0,29,0	ITDATA	Controls edits of fission source iteration data: 0-no edit, 1-edit	(3)
0, 0,30,0	INPRNT	Controls edits of multiple thermal flux inner iteration data: 0-no edit, 1-edit	(3)
0, 0,31,0	LIBED	Library material data set edit indicator: 0-no edit, 1-edit	(3)
0, 0,32,0	IPUNCH	Controls type and format of output data punched	(3)
0, 0,33,0	ITAPE	Indicator for saving complete program data on unit 9: 0-do not save n $\geq$ 1-save	(3)
0,0,34-35,0		Ignored	

Table VI (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, 0,36,0	NORMAL	Specifies type of "normalization" to be used during output edits: 0-no normalization, 1-fission rate=POWER, 2- ϕ =1, 3-fission source=1, 4-multiply by FLMULT	(3)
0, 0,37,0	MESHND	Specifies mesh or source point used as basis for normalization when NORMAL = 2 or 3	(3)
0, 0,38,0	N0GRP	Specifies neutron group used as basis for normalization when NORMAL = 2	(3)
0, 0,39,0	ITBTPT	Total flux edit control flag: 0-no edit, n-edit total flux distribution, normalized at mesh point n	(3)
0, 0,40,0		Ignored	
0, 0,41,0	INFLUX	Specifies type and source of neutron flux guess for multiple thermal group problems: 0- ϕ as is, 1- ϕ =0	(3)
0, 0,42,0	INC0NV	Controls multiple thermal group inner iteration acceleration: 0-none, 1-use R^U , 2-use R^U until $ R^U-1 \leq \epsilon_5$, 3-use r_j^U until $ R^U-1 \leq \epsilon_5$, 4-use input τ , 5-use computed τ	(3)
0, 0,43,0	MAXINS	Specifies maximum number of thermal group iterations	(3)
0,0,44-45,0		Ignored	
0, 0,46,0	INS0RS	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, 0,47,0	IGESS	Specifies fission neutron source saved for next problem: 0-last source computed, 1-source used on last iteration	(3)
0, 0,48,0	NC0NVG	Specifies convergence based on total or pointwise fission source: 0-total source, 1-pointwise source	(3)
0, 0,49,0	MAX0UT	Specifies maximum number of fission source iterations	(3)
0, 0,50,0	NEXT0	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	NPI0	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)
0,0,54-60,0		Ignored	
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 -concentrations	(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)
0,0,64-65,0		Ignored	
0, 0,66,0	LIB0LD	Data set identification number written on the saved and mounted library material data set	(3)
0, 0,67,0	LIBNEW	Data set identification number to be written on the updated library data set	(3)
0, 0,68,0	LIBMTL	Number of library materials to be read from cards	(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) $\neq$ 0.	(3-K)
0, 9, K,0	IEBK(K)	These energy group break points define the broad coalesced groups.	(3-K)
0,10, K,0	IRBK(K)	These region break points define the broad coalesced regions.	(3-K)
0,12, K,0	ISPECT(K)	The list of mesh points at which flux spectra are edited	(3-K)
0,13, K,0	NUCSCH(K)	The concentrations of the constituent materials named in this list are changed directly in a concentration search.	(3-K)
0,14, K,0	ISUMS(K)	The sum in each macro material of the concentrations of the constituent materials named in this list is held constant during a concentration search.	(3-K)
0,15, K,0	IXLST(K)	The list of macro materials edited when INED1 > 0	(3-K)
1,IM, K,0	MIX(K,IM)	The list of constituent materials mixed to form macro material IM (+ numbers are library materials, - numbers are previously used macros)	(3-K)
2, L, K,N	MRATE(K,N,L)	The list of library materials used to compute the numerator (+N) or denominator (-N) of reaction rate ratio L [Rate computed: N =0-none, 1-absorption (Σ_a), 2-fission (Σ_f), 3-capture ($\Sigma_a-\Sigma_f$), 4-production ($\nu\Sigma_f$), 5-power ($E\Sigma_f$), 6-transport collision (Σ_u)]	(3-K)

Table VII. Summary of Problem 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 multiple thermal group neutron balance	(3)
0, 0, 5,0	EPS5	Convergence level of thermal inner iterations at which acceleration by source-to-loss multiplication can be cut off	(3)
0, 0, 6,0	RAD1	Position of the first mesh point, the inner boundary x_1	(3)
0, 0, 7,0	THRMX	Extrapolation factor τ for acceleration of inner (thermal group) iterations	(3)
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	PPOWER	Total volume integral of fission power	(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_d	(3)
0, 0,15,0	EPFISS	Optional energy per fission E_f to use as library data set is written	(3)
0,0,16-19,0		Ignored	
0, 0,20,0	FIXSRS	The total volume integral of pointwise fixed volume source when NRMFIX=1	(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
0,10, K,0	F(K)	Pointwise fission neutron source guess for source point K	(3-K)
0,12, I,J	PHI(I,J)	Pointwise flux input for group J and mesh point I	(3-I)
0,14, J,0	EBAR(J)	Average energy $\bar{E}_j$ of group $j \equiv J$	(3-J)
0,15, J,0	DELE(J)	Energy width ΔE_j of group $j \equiv J$	(3-J)
0,15, J,0	VRECIP(J)	Average reciprocal velocity of group $j \equiv J$	(3-J)
1,IM, J,1	DC(J,IM)	Optional diffusion coefficient for group J of macroscopic material IM	(3-J)
5,IM, J,0	CHIM(J,IM)	Fraction of total fission neutrons released in group J of macroscopic material IM	(3-J)
6,IM, J,0	BSQ(J,IM)	Buckling squared value for group J of macroscopic material IM	(3-J)
7,IM, J,0	T(J,IM)	Fraction T_j of macroscopic poison cross section SIGPT(IM) used in group $j \equiv J$	(3-J)
8,IM, J,0	Q(J,IM)	Logarithmic boundary condition parameter applied in group J of macroscopic material IM	(3-J)
9,IR, J,0	SEOR(J,IR)	Flat fixed volume source S_{rj} in group $j \equiv J$ and region IR	(3-J)
10, J, K,0	SEOR(K,J)	Pointwise fixed volume source S_{pj} for group $j \equiv J$ and source point K	(3-K)
11,IM, K,0	CON(K,IM)	The values listed here respectively multiply the cross sections of the constituent materials listed as MIX(K,IM) to mix macroscopic material IM.	(3-K)

Table VIII. The Library Material Title Card Summary

Information or Mnemonic Name of Data	Card Columns	Description or Function
IAD	2-3	A value of 1 means update rather than replace or add this material to the library tape.
IDIFF	4-5	A nonzero value means the SIGMU(J,L) values are computed from the input D(J,L) values instead of read directly.
L	6-8	Library Material identifying number $L \equiv L$
IENPF	9-10	Controls how the energy per fission values are input: 0- $E_j^f =$ EPFISS, 1- $E_j^f - E_j^f$, 2- E_j^f as input
Any Hollerith Characters	13-72	Library material title information

Table IX. Summary of the Library Material Floating Point Input Data

Indices 1,2,3 and 4 In Card Cols. 2-10	Mnemonic Name of Data	Description or Function of Data
1, L, J, 0	DC(J,L), SIGA(J,L), SIGNU(J,L), SIGFIS(J,L), ENERGY(J,L)	Diffusion coefficient, absorption cross section, neutrons per fission times the fission cross section, fission cross section, and energy per fission for group J of library material L; values are listed across a card in the order given; DC(J,L) is not used unless IDIFF=1 in the title card
1, L, J, 1*	DC(J,L)	Diffusion coefficient for group J of library material L (when IDIFF=1 on the title card)
1, L, J, 2*	SIGA(J,L)	Absorption cross section for group J of library material L
1, L, J, 3*	SIGNU(J,L)	Neutrons per fission times fission cross section for group J of library material L
1, L, J, 4*	SIGFIS(J,L)	Fission cross section for group J of library material L
1, L, J, 5*	ENERGY(J,L)	Energy per fission for group J of library material L
2, L, J, N*	TRMAT(J,J+N)	Optional scattering transfer cross section $\Sigma_{s,j \rightarrow j}^L$ (for neutron scatter) from group J to group J+N for library material $L \equiv L$
3, L, J, K**	TRMAT(J,K)	General scattering transfer cross sections $\Sigma_{s,j \rightarrow j}^L$ from group J to group K for library material $L \equiv L$
4, L, J, 0*	SIGMU(J,L)	Cross section $\Sigma_{t,j}^L$ for group $j \equiv J$ of library material $L \equiv L$ (used in group $j \equiv J$ for mixing the macroscopic cross section $\Sigma_{t,j}^m$ from which the macroscopic diffusion coefficient D_j^m for macroscopic material m is computed) These input data are used only if IDIFF=0 or blank on the title card.

*Index 3 (J) increases as words are added to a card.

**Index 4 (K) increases as words are added to a card.

V. PROGRAM CHARACTERISTICS, OPERATIONAL INFORMATION, AND MACHINE REQUIREMENTS

The MØNA program language is entirely FORTRAN IV and it operates on IBM 360 series machines with no installation dependent routines except for a subroutine called TIMJ(X). This is a small clock (elapsed CPU time) routine that can be easily replaced or dummied.

The major restrictions on the problem dimensions are as follows.

Dimension	Maximum	
Mesh points, I	500	} Also, $I(J+6)+6IR \leq 8705$ must be satisfied.
Energy groups, J	50	
Regions, IR	50	
Macro (mixed) materials, M	20	
Library (stored) materials, L	999	
Constituent materials per macro material	50	
Total number of transfer cross sections ($\sum_{j \rightarrow j'} N$), N	12250	

The last restriction on the number of transfer cross sections allowed applies to the sum

$$N = \sum_m \left[N_d^m J - N_d^m (N_d^m + 1)/2 + N_u^m N_{th}^m - N_u^m (N_u^m + 1)/2 \right],$$

where $m \equiv$ the set of macro materials of the problem,

$J \equiv$ number of groups,

$N_d^m \equiv$ maximum number of groups any group can transfer to by downscatter for macro material m ,

$N_u^m \equiv$ maximum number of groups any group can transfer to by upscatter for macro material m , and

$N_{th}^m \equiv$ number of thermal groups for material m (number of groups that upscatter plus one).

Round off errors in the program solutions are determined primarily by the following considerations.

Almost all basic data (neutron group constants, fixed sources, concentrations, etc.) are stored as single precision, 4 byte, words.

All arithmetic is done in double precision, and fission sources are stored in double precision.

Although computed in double precision, the fluxes are stored in single precision. This is probably the major source of round off error in the program. However, no cases have yet been observed where meaningful pointwise convergence of the source to 10^{-6} or less could not be attained. Hence, eigenvalues can be attained by within a fractional error of about 10^{-7} or less.

The MØNA program load module requires 260K bytes of storage.

At execution time, the significant requirements are:

1. 270K-300K bytes of core storage (depending on I/O buffer sizes specified)
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 (intermediate storage - 120K bytes required)
5. Fortran logical unit 3 (intermediate storage - 90K bytes required)
6. Fortran logical unit 4 (intermediate storage - 12K bytes required per library material)
7. Fortran logical unit 8 (library material storage - 12K bytes required per library material)
8. Fortran logical unit 9 (problem data storage when ITAPE > 0 - allow 160K bytes per problem stored)
9. Unit for punched card output.

The order in which MØNA performs the various major input, solution, and output functions is shown on Figure 1, where the steps carried out in each of the 8 overlays are outlined.

The reader is referred to the sample problem described in Appendix C for an example of CPU time required for a problem solution on an IBM 360/75 operating with MVT with a load module compiled using the Fortran H compiler, optimization level 2.

Figure 1. MØNA Overlay Flow And Functions

VI. REFERENCES

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APPENDIX A

GLOSSARY

The major indices, subscripts or superscripts are:

- a - symbol for neutron absorption
- b - symbol for region boundary
- e - symbol for external source
- f - symbol for neutron fission
- i - mesh index of the space variable x
- j - neutron energy group index
- jth - index denoting a thermal neutron group
- k - source point index ($k=i+r-1$), or the index of data words input as a list or a set of vector elements.
- ℓ - Chebyshev extrapolation polynomial order, or constituent material index
- m - macroscopic material index
- p - symbol for neutron poison (absorption), or symbol for pointwise
- r - region index, or symbol for total neutron removal
- s - search iteration index
- T - matrix transpose symbol
- t - fission neutron source iteration index
- u - inner (scattering source) iteration index
- μ - symbol for transport corrected value

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

The most commonly referred to symbols are:

- 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
- $\overline{\Lambda}^t$ - $(f^*t/f^{t-1})_{\text{maximum}}$
- $\underline{\Lambda}^t$ - $(f^*t/f^{t-1})_{\text{minimum}}$
- $\nu \Sigma_{fj}$ - 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), or a constituent concentration or weighting factor used in mixing macro material group constants
- σ_0, σ_0' - dominance ratio, estimate of dominance ratio
- Σ_j - total macroscopic cross section of group j
- Σ_{aj} - macroscopic absorption cross section of group j
- Σ_{fj} - the fission cross section
- $\Sigma_{j \rightarrow j'}$ - macroscopic cross section for scatter transfer from group j to group j'
- Σ_p - "poison" cross section
- Σ_{rj} - total macroscopic cross section for the effective absorption (removal) of neutrons, $\Sigma_{rj} = \Sigma_{aj} + T_j \Sigma_p + D_j B_j^2 + \sum_{j' \neq j} \Sigma_{j \rightarrow j'}$
- Σ_{uj} - a cross section that can be used to compute the diffusion coefficient
- τ - flux extrapolation factor for inner iterations
- ϕ - scalar neutron flux: track lengths per unit volume and per unit time
- ϕ^* - unextrapolated flux during inner iterations
- x_j - the fraction of fission neutrons appearing in energy group j
- ω - an optional boundary condition parameter
- a_i - $D_i [x_{i+1} + x_i] / 2]^p / (x_{i+1} - x_i)$

- B_j^2 - perpendicular buckling (squared)
- C - normalization constant multiplying output sources and fluxes
- C_ρ - geometry constant (1, 2π , and 4π for $\rho = 0, 1$, and 2)
- C_s - search parameter modifier
- D_j - diffusion coefficient
- ΔE_j - energy width of group j
- E_j - the energy per fission for fissions caused by absorption of neutrons in group j
- $\bar{E}_j$ - average energy of 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_1, G_2 - lower and upper boundary condition correction factors
- $\bar{i}$ - unit vector in the x direction
- J - $\bar{J} \cdot \bar{i}$, the x component of $\bar{J}$
- J'_i - $\bar{J} \cdot \bar{i}$ at $(x_i + x_{i+1})/2$
- $\bar{J}$ - neutron current or vector flux, neutrons per unit area and per unit time
- N_f - normalization factor for multiplying pointwise fixed volume source
- q_j - a constant determining internal logarithmic boundary conditions at the upper and lower boundaries of a region for group j
- r - groupwise source-to-loss ratio for a thermal neutron group
- R - total source to total loss ratio for all thermal groups

$s_i - C_p \left(\frac{x_i + x_{i+1}}{2} \right)$, 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

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 $(x_i + x_{i+1})/2$

x - the space variable.

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.

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

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

and

$$J_{i_{\max}-1}^+ s_{i_{\max}-1} - J_{i_{\max}} C_p x_{i_{\max}}^p - (\Sigma_{r,i_{\max}}^- v_{i_{\max}}^-) \phi_{i_{\max}} \quad (B-4)$$

$$+ S_{i_{\max}}^- v_{i_{\max}}^- = 0 \quad (i = i_{\max}).$$

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,

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

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

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_p and multiplied by β_1 while Equation (B-10) is multiplied by x_1^p ; 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

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

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

---

*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 $J_1 x_1^p = J_{i_{\max}} x_{i_{\max}}^p$ are handled in a special manner as described later.

and

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

The new coefficients here are

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

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

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

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

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

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

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

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

$$\delta_1 = \beta_1 S_{t1} + x_1^0 \gamma_1; \text{ then (for } i = 2, 3, \dots, i_{\max-1}), \quad (B-24)$$

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

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

and

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

Finally,

$$Z_{i_{\max}} = 1/[-x_{i_{\max}}^0 \alpha_2 - \beta_2 c_{i_{\max}} - \beta_2 a_{i_{\max}-1} (a_{i_{\max}-1}' + 1)] \quad (B-28)$$

and

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

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

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

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

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

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

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

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

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

The new difference equations that result are

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

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

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).

APPENDIX C

SAMPLE PROBLEM

The complete output of a sample MØNA problem is shown in Figure C-1. With ILIST=1 in card columns (ccs) 64-70, the contents of each input card are also listed; hence, the figure also shows the complete input data deck followed by the data of four library material data decks. In the figure these input data cards occur between the statements "BEGIN PROBLEM" and "BEGIN INITIALIZATION".

As can be seen from the input card edit and the processed input data edit that follows it, the major characteristics of the sample problem are as follows.

There are JMAX = 22 neutron groups, IRMAX = 3 regions, and the geometry is cylindrical (NGEØ = 1). The boundary conditions for all groups are specified as symmetry, or zero current, at the center (NIBDRY = 1) and a corrected extrapolated flux condition at the outer boundary (NØBDRY = 6).

The problem is the usual renormalization or eigenvalue type (IREN = 0). Group and region coalesced constants are to be computed (KØALES = 1). Following the problem solution the ancillary program assigned IPERT = 1 will be entered. The group buckling (B^2) values are group dependent, only (INBUCK = 2); hence, the B^2 values used in all materials will be those read for macro material 1.

On the second integer input data card, ICHIM = 1 will cause the fission spectrum (χ values) in each fissionable material to be that of macro material 1. An INED1 value of 1 suppresses the editing of macro material data during the input data edit of all macro materials except material 3 (as specified by the IXLST data on the 15th integer input card). The INED2 = 2 value calls for an edit of the final constituents and concentrations of the macro materials at output edit time. All the output edits controlled by the flags IØUT1 and IØUT3 are called for. During the source and flux edits, values for only every other mesh point will be printed (ILINES = 2). The ITDATA value, 1, calls for editing of the outer iteration convergence data.

Setting ITØTPT = 1 on the third integer input card will cause an edit of two total flux distributions where one of them will be normalized to unity at mesh point 1.

On the next card, NCØNVG = 1 calls for convergence of the fission source to within an error of ϵ_1 at each mesh point unless the number of outer iterations reaches MAXØUT = 30, and NEXTØ = 2 will apply Chebyshev acceleration to the iterative process.

With ISERCH = 6 on the fifth integer card, a concentration search will be done with no more than MAXSCH = 10 search iterations to be performed.

On the sixth integer card $LIBOLD = 0$, as it must be when no saved library data set exists, $LIBNEW$ is set to 1 to identify the new data set, and $LIBMTL = 4$ tells the program that the decks of four library materials follow the problem input data.

The next two cards assign macro materials 1, 3, and 2 to regions 1, 2, and 3, respectively, and specify 16, 20, and 4 mesh intervals to regions 1, 2, and 3.

Integer card 9, with the LRS array, restricts the concentration search to macro materials 1 and 3. (This card is actually superfluous because the library material concentrations affected by the search, 1 and 3, occur only in these materials, anyway, as specified by the 14th integer input card.)

The 10th integer card sets the first four values of the IEBK list to 5, 11, 17, and 22 to signify that, for the coalesced constants to be computed, the first broad group will consist of MØNA groups 1 through 5, the second will consist of MØNA groups 6 through 11, etc. The following card sets the region break points with the IRBK list so that constants for MØNA regions 1 and 2 will be coalesced for the first broad region and MØNA region 3 constants will be used for the second broad region.

The first four words of the ISPECT list are set to 1, 11, 16, and 21 on the 12th integer card; hence, flux spectra will be edited for those mesh points as well as for each region.

The next two integer cards specify that the concentration search is to vary directly the concentration of library material 1 ($NUCSCH(1) = 1$), and that, in every macro material searched, the original sum of the concentrations of library materials 1 and 3 is to be preserved throughout the search [$ISUMS(1) = 1$ and $ISUMS(2) = 3$].

The 15th integer card sets the first word of the IXLST list to 3, and, since $INED1$ has been set to 1, the input data edits will list only the group constants for macro material 3.

The next three integer cards specify the mixture constituents for macro materials 1, 2, and 3, in that order, by means of the $MIX(K,IM)$ data. Macro material 1 will be mixed from data for library materials 1, 3, 4, and 5, while library materials 4 and 5 will be used for macro material 2 and library materials 1, 3, and 5 will be used for macro material 3.

The last four integer data cards specify the $MRATE(K,N,L)$ array for reaction rate ratio edits. On the first of these cards, $L = 1$ and $N = 3$ so that the numerator of reaction rate ratio number 1 will be the capture ($\Sigma_a - \Sigma_f$) rate for the library materials listed on the card (simply material 1, in this case). The next card specifies that the denominator of ratio 1 is to be the fission (Σ_f) rate ($N = -2$) of library material 1. In the same manner, the last two cards specify that for reaction ratio 4, the numerator will be the capture rate of library material 3 and the denominator will be the absorption (Σ_a) rate of library material 1. (Hence, reaction ratio 4 is a breeding ratio in this case.)

The problem floating point data cards specify the following values.

The fission source convergence level ϵ_1 is 10^{-6} .

The search convergence level ϵ_2 is 10^{-5} , that is, ten times ϵ_1 , as recommended.

The search eigenvalue, EIGEND, to be attained is 1.01.

The optional energy per fission value EPFISS is set to 3.14×10^{-11} . This value will be used to set the group-independent values for each library material that has the IENPF flag set to zero on the title card at the time the library material data is stored on the library data set.

Region widths are set to 24.0, 50.0, 8.0 cm in regions 1, 2, and 3, respectively.

Fission spectrum values $[x_j]$ as set by the CHIM(J,IM) array] are read only for macro material 1, but will be used also for the fissionable macro material 3 because ICHIM has been set to 1 in the integer input data.

Also, buckling (B^2) values are read only for material 1, but will be used in all macro materials because INBUCK = 2.

The original concentrations of the constituents of each macro material are set with the CONC(K,IM) array. For example, the respective concentrations of library materials 1, 3, 4, and 5 that are used in macro material 1 are 0.0075, 0.0065, 0.005, and 0.2.

Finally, values for $\bar{E}_j$, ΔE_j , and $\left(\frac{1}{v}\right)_j$, are read (for use in the output edits) by means of the EBAR, DELE, and VRECIP data lists.

The problem data decks are followed by four library material data decks. These library data will be stored on a library data set for permanent use, if desired, and for use in subsequent MØNA problems in the same computer run.

On all four library material title cards, IENPF = 0 and IDIFF = 0. Hence, the energy per fission values stored on the library data set will be EPFISS (from the MØNA problem data) in all cases; and no diffusion coefficient data are needed because all $\Sigma_{\mu j}^0$ values are to be those input directly (instead of computed from $1/3D_j^0$).

The data for library materials 1, 3, and 4 are microscopic cross section values for U-235, U-238, and Na, respectively. The appropriate concentrations used in weighting these data to form macros are therefore atomic densities (atoms per cm^3). Library material 5 is already a macro material (stainless steel); hence, a volume fraction should be used for its "concentration" in forming macros.

The processed input data edits are largely self-explanatory. It might be noted, however, that various data not read in have been set to their "default" values, automatically. The automatically computed boundary condition parameters α , β , and γ at either boundary are also output.

The iteration convergence data edits show that on the first search iteration 20 outer (source) iterations were required for convergence to an eigenvalue of 1.0910984. It is noted that these iterations took 7.022 seconds, or about 0.35 seconds of CPU time per iteration for this 22 group, 41 mesh point problem.

Three more search iterations were required to reach the desired eigenvalue of 1.01 to within a fractional error of 10^{-5} . The final search parameter value of 0.84281251 is the multiplier of the original concentration of library material 1 required to satisfy the search.

The problem output data edits are self-explanatory; and following the edit of the coalesced group and region data the program enters the ancillary program corresponding to IPERT=1. Since ILIST=1, the contents of the single input data card read in this ancillary program are printed immediately following the coalesced data edits. The activation traverses for library material 4 (as called for on the data card) are then computed and edited as shown.

The entire run required a total of 25.65 seconds of CPU time on the IBM 360/75 operating with MVT with a load module compiled using the Fortran H compiler, optimization level 2.

Figure C.1 MØNA Sample Problem Output Edit

```

ENTERING MAIN LINK          0.0
BEGIN PROBLEM              0.017

22 GROUP MØNA SAMPLE PROBLEM
0 0 0 1 011 22 3 1 1 6 0 0 0 1 1 2 0 0 1 1 0
0 0 0 19 011 1 0 1 1 0 0 0 3 0 3 2 1 0 0 0 0
0 0 0 39 0 1 1 0 0 0 0 0 0 0 0 0 0 0 0 0 0
0 0 0 48 0 3 1 30 2 0 0 0 0 0 0 0 0 0 0 0
0 0 0 61 0 2 6 10 0 0 0 0 0 0 0 0 0 0 0 0
0 0 0 66 0 3 0 1 4 0 0 0 0 0 0 0 0 0 0 0
0 0 1 1 0 3 1 3 2 0 0 0 0 0 0 0 0 0 0 0
0 0 2 1 0 3 16 20 4 0 0 0 0 0 0 0 0 0 0
0 0 3 1 0 2 1 3 0 0 0 0 0 0 0 0 0 0 0
0 0 9 1 0 4 5 11 17 22 0 0 0 0 0 0 0 0
0 0 10 1 0 2 2 3 0 0 0 0 0 0 0 0 0 0
0 0 12 1 0 4 1 11 16 21 0 0 0 0 0 0 0 0
0 0 13 1 0 1 1 0 0 0 0 0 0 0 0 0 0 0
0 0 14 1 0 2 1 3 0 0 0 0 0 0 0 0 0 0
0 0 15 1 0 1 3 0 0 0 0 0 0 0 0 0 0 0
0 1 1 1 0 4 1 3 4 5 0 0 0 0 0 0 0 0
0 1 2 1 0 2 4 5 0 0 0 0 0 0 0 0 0 0
0 1 3 1 0 3 1 3 5 0 0 0 0 0 0 0 0 0
0 2 1 1 3 1 1 0 0 0 0 0 0 0 0 0 0
0 2 1 1 4 1 1 0 0 0 0 0 0 0 0 0 0
0 2 4 1 3 1 3 0 0 0 0 0 0 0 0 0 0
1 2 4 1 1 1 0 0 0 0 0 0 0 0 0 0
0 0 0 1 0 0 1.000000-06 1.000000-05 0.0 0.0 0.0
0 0 0 14 0 2 1.010000 00 3.140000-11 0.0 0.0
0 0 1 1 0 3 2.400000 01 5.000000 01 8.000000 00 0.0
0 0 1 1 0 5 1.303500-01 2.108900-01 2.322300-01 1.794100-01 1.148700-01
0 0 1 6 0 5 6.484100-02 3.409700-02 1.712500-02 8.395900-03 4.069700-03
0 0 1 11 0 5 1.953200-03 9.332830-04 4.395200-04 2.667700-04 6.814700-05
0 0 1 16 0 5 1.845900-05 2.237900-05 6.118900-06 7.159600-06 4.406200-06
0 0 1 21 0 2 1.564100-06 3.927000-07 0.0 0.0 0.0
0 0 1 1 5 5 1.990000-03 1.980000-03 1.960000-03 1.810000-03 1.650000-03
0 0 1 6 0 5 1.500000-03 1.410000-03 1.310000-03 1.230000-03 1.140000-03
0 0 1 11 0 5 1.070000-03 9.940000-04 9.000000-04 8.190000-04 6.840000-04
0 0 1 16 0 5 5.640000-04 1.530000-04 1.100000-05 1.000000-05 1.000000-05
0 0 1 21 0 2 1.000000-05 1.000000-05 0.0 0.0 0.0
0 1 1 1 0 4 7.500000-03 6.500000-03 5.000000-03 2.000000-01 0.0
0 1 2 1 0 2 1.500000-03 9.500000-01 0.0 0.0 0.0
0 1 3 1 0 3 6.200000-05 2.400000-02 2.200000-01 0.0 0.0
0 0 14 1 0 5 6.840000 03 2.960000 03 1.790000 03 1.090000 03 6.600000 02
0 0 14 6 0 5 3.620000 02 2.430000 02 1.470000 02 8.920000 01 5.420000 01
0 0 14 11 0 5 3.290000 01 1.990000 01 1.210000 01 6.720000 00 3.460000 00
0 0 14 16 0 5 2.330000 00 1.640000 00 1.100000 00 7.720000-01 3.620000-01
0 0 14 21 0 2 1.880000-01 6.800000-02 0.0 0.0 0.0
0 0 15 1 0 0 6.320000 03 1.450000 03 8.800000-02 5.290000 02 3.230000-02
0 0 15 6 0 0 1.960000 02 1.190000 02 7.200000 01 4.360000 01 2.650000 01
0 0 15 11 0 0 1.610000 01 9.800000 00 5.880000 00 4.810000 00 1.700000 00
0 0 15 16 0 0 5.700000-01 8.100000-01 2.689000-01 3.782000-01 3.075000-01
0 0 15 21 0 2 1.741000-01 1.013000-01 0.0 0.0 0.0
0 0 16 1 0 0 2.780000-08 4.200000-08 5.410000-08 6.900000-08 8.930000-08
0 0 16 6 0 0 1.200000-07 1.470000-07 1.870000-07 2.440000-07 3.090000-07
0 0 16 11 0 0 4.080000-07 5.070000-07 6.540000-07 8.850000-07 1.230000-06
0 0 16 16 0 0 1.520000-06 1.790000-06 2.170000-06 3.700000-06 4.130000-06
1 0 16 21 0 2 5.260000-06 8.770000-06 0.0 0.0 0.0
0 0 0 1 0 0 LIB. MFL. 1: MICRO U-235
0 1 1 1 0 0 0.0 1.279590 00 3.985040 00 1.260350 00 0.0
0 1 1 2 0 0 0.0 1.335810 00 3.586950 00 1.297410 00 0.0
0 1 1 3 0 0 0.0 1.383150 00 3.436700 00 1.314920 00 0.0
0 1 1 4 0 0 0.0 1.362410 00 3.181840 00 1.256610 00 0.0
0 1 1 5 0 0 0.0 1.313910 00 2.893490 00 1.163640 00 0.0
0 1 1 6 0 0 0.0 1.444800 00 3.042400 00 1.235650 00 0.0
0 1 1 7 0 0 0.0 1.633000 00 3.319540 00 1.355890 00 0.0
0 1 1 8 0 0 0.0 1.904740 00 3.699540 00 1.516430 00 0.0
0 1 1 9 0 0 0.0 2.221740 00 4.152940 00 1.705810 00 0.0
0 1 1 10 0 0.0 2.616290 00 4.756880 00 1.956310 00 0.0
0 1 1 11 0 0.0 3.064740 00 5.498600 00 2.262960 00 0.0
0 1 1 12 0 0.0 3.612970 00 6.424940 00 2.645380 00 0.0
0 1 1 13 0 0.0 4.146760 00 7.535650 00 3.103600 00 0.0
0 1 1 14 0 0.0 5.273500 00 9.461280 00 3.897420 00 0.0
0 1 1 15 0 0.0 6.803330 00 1.200260 01 4.944810 00 0.0
0 1 1 16 0 0.0 8.275200 00 1.440400 01 5.934410 00 0.0
0 1 1 17 0 0.0 9.490060 00 1.635730 01 6.739330 00 0.0
0 1 1 18 0 0.0 1.145090 01 1.947120 01 8.022440 00 0.0
0 1 1 19 0 0.0 1.302100 01 2.195890 01 9.047490 00 0.0
0 1 1 20 0 0.0 1.619160 01 2.694350 01 1.101380 01 0.0
0 1 1 21 0 0.0 2.171940 01 3.558420 01 1.466170 01 0.0
0 1 1 22 0 0.0 2.822630 01 4.618860 01 1.903110 01 0.0
0 3 1 1 2 0 2.233880-01 4.306790-01 4.936720-01 3.704450-01 2.118120-01
0 3 1 1 7 0 1.021390-01 4.432870-02 1.803500-02 7.053580-03 2.693190-03
0 3 1 1 12 2 1.013390-03 3.649760-04 0.0 0.0 0.0
0 3 1 2 3 0 3.269280-01 4.895840-01 4.627410-01 3.047150-01 1.601610-01
0 3 1 2 8 0 7.325490-02 3.076960-02 1.226960-02 4.740190-03 1.796410-03
0 3 1 2 12 2 6.728890-04 1.555400-04 0.0 0.0 0.0
0 3 1 3 4 0 4.441670-01 4.449190-01 3.184120-01 1.761700-01 8.314900-02
0 3 1 3 9 0 3.560110-02 1.436270-02 5.588320-03 2.126970-03 7.987910-04
0 3 1 3 14 1 2.975630-04 0.0 0.0 0.0 0.0
0 3 1 4 5 0 4.956470-01 3.384450-01 1.912730-01 9.145810-02 3.947040-02
0 3 1 4 10 0 1.600070-02 6.243900-03 2.380730-03 8.950570-04 3.336420-04
0 3 1 5 6 0 5.092360-01 2.552690-01 1.223330-01 5.253490-02 2.133730-02
0 3 1 5 11 4 8.335990-03 3.180640-03 1.196300-03 5.489090-04 0.0
0 3 1 6 7 3 5.200190-01 1.383210-01 5.277240-02 1.732550-02 4.331200-03
0 3 1 6 12 3 1.404990-03 5.039860-04 1.706500-04 0.0 0.0
0 3 1 7 8 0 4.218730-01 8.649510-02 3.886480-02 1.846310-02 5.910070-03
0 3 1 7 13 2 2.451360-03 1.173020-03 0.0 0.0 0.0
0 3 1 8 9 0 3.792250-01 7.757600-02 2.937750-02 8.538040-03 4.238190-03
0 3 1 8 14 4 2.398330-03 6.258470-04 1.458930-04 8.893990-05 0.0
0 3 1 9 10 0 2.644050-01 8.379990-02 3.910070-02 6.937050-03 3.743930-03
0 3 1 9 15 3 5.995150-04 8.723880-04 1.382610-04 0.0 0.0
0 3 1 10 11 0 1.820350-01 2.011520-02 2.403210-02 1.523770-02 3.676830-03
0 3 1 10 16 4 9.723880-04 8.675410-04 1.491020-04 9.114170-05 0.0
0 3 1 11 12 2 1.479600-01 5.200830-03 0.0 0.0 0.0
0 3 1 12 13 0 1.146390-01 1.081920-02 1.468100-03 4.688760-04 4.884540-04
0 3 1 12 18 1 8.902890-05 0.0 0.0 0.0 0.0
0 3 1 13 14 1 1.344730-01 0.0 0.0 0.0 0.0
0 3 1 14 15 1 1.106850-01 0.0 0.0 0.0 0.0
0 3 1 15 16 1 7.739280-02 0.0 0.0 0.0 0.0
0 2 1 16 1 0 5.286470-01 1.332000-01 5.591130-01 1.622860-01 9.805640-02
0 2 1 21 1 1 4.098970-02 0.0 0.0 0.0 0.0
0 4 1 1 0 0 4.363890 00 4.925720 00 4.783170 00 4.865830 00 5.482580 00
0 4 1 6 0 0 6.680590 00 9.214200 00 7.983310 00 1.220400 00 1.241040 00
0 4 1 11 0 0 1.339340-01 1.426240-01 1.470720-01 1.861200 01 2.107410-01
0 4 1 16 0 0 2.390050 01 2.338470 01 2.783630 01 2.764590 01 2.919020 01
1 4 1 21 0 2 3.160460 01 3.888650 01 0.0 0.0 0.0

```

C 0 0 3 0 0 LIB. MTL. 3, MICRO U-238

0 1 3	1 0 0 0 0	6.384660-01	1.974490	00	6.194300-01	0.0
0 1 3	2 0 0 0 0	5.968490-01	1.586030	00	5.731800-01	0.0
0 1 3	3 0 0 0 0	4.723020-01	1.077940	00	4.119600-01	0.0
0 1 3	4 0 0 0 0	1.552920-01	6.405980-02	2.543000-02	0.0	0.0
0 1 3	5 0 0 0 0	1.301060-01	3.071940-03	1.229800-03	0.0	0.0
0 1 3	6 0 0 0 0	1.158980-01	0.0	0.0	0.0	0.0
0 1 3	7 0 0 0 0	1.392420-01	0.0	0.0	0.0	0.0
0 1 3	8 0 0 0 0	1.913040-01	0.0	0.0	0.0	0.0
0 1 3	9 0 0 0 0	2.594510-01	0.0	0.0	0.0	0.0
0 1 3	10 0 0 0 0	3.354770-01	0.0	0.0	0.0	0.0
0 1 3	11 0 0 0 0	4.050190-01	0.0	0.0	0.0	0.0
0 1 3	12 0 0 0 0	5.023080-01	0.0	0.0	0.0	0.0
0 1 3	13 0 0 0 0	6.358130-01	0.0	0.0	0.0	0.0
0 1 3	14 0 0 0 0	7.860230-01	0.0	0.0	0.0	0.0
0 1 3	15 0 0 0 0	7.088650-01	0.0	0.0	0.0	0.0
0 1 3	16 0 0 0 0	8.505780-01	0.0	0.0	0.0	0.0
0 1 3	17 0 0 0 0	7.710140-01	0.0	0.0	0.0	0.0
0 1 3	18 0 0 0 0	1.134490	00	0.0	0.0	0.0
0 1 3	19 0 0 0 0	1.312790	00	0.0	0.0	0.0
0 1 3	20 0 0 0 0	1.246670	00	0.0	0.0	0.0
0 1 3	21 0 0 0 0	1.678550	00	0.0	0.0	0.0
0 1 3	22 0 0 0 0	1.784180	00	0.0	0.0	0.0
0 3 3	1 2 0	2.823300-01	5.599350-01	6.385440-01	4.784350-01	2.734630-01
0 3 3	1 7 0	1.318670-01	5.723450-02	2.328720-02	9.108210-03	3.477800-03
0 3 3	112 2	1.308650-03	4.668650-04	0.0	0.0	0.0
0 3 3	2 3 0	4.280290-01	6.714650-01	6.357700-01	4.191600-01	2.204870-01
0 3 3	2 8 0	1.008970-01	4.239330-02	1.690780-02	6.532860-03	2.475970-03
0 3 3	213 2	9.274720-04	2.143930-04	0.0	0.0	0.0
0 3 3	3 4 0	6.712730-01	6.563000-01	4.874740-01	2.400840-01	1.130900-01
0 3 3	3 9 0	4.836080-02	1.949560-02	7.581900-03	2.884930-03	1.083260-03
0 3 3	314 1	4.034900-04	0.0	0.0	0.0	0.0
0 3 3	4 5 0	5.119400-01	2.709960-01	2.267240-01	7.104970-02	2.316700-02
0 3 3	410 4	8.116110-03	3.141870-03	1.181010-03	3.948020-04	0.0
0 3 3	5 6 0	5.058000-01	4.180960-03	3.395750-02	2.214390-02	1.012870-02
0 3 3	511 4	3.903690-03	1.479470-03	5.488570-04	1.477640-04	0.0
0 3 3	6 7 2	5.122730-01	1.888040-02	0.0	0.0	0.0
0 3 3	7 8 0	4.690260-01	2.035040-02	9.108210-03	3.477800-03	1.308650-03
0 3 3	713 2	4.668650-04	2.143930-04	0.0	0.0	0.0
0 3 3	8 9 0	5.393660-01	1.104240-02	2.115090-03	1.833080-03	7.277890-04
0 3 3	814 1	3.487940-04	0.0	0.0	0.0	0.0
0 3 3	910 3	4.205410-01	1.188100-01	1.562510-02	0.0	0.0
0 3 3	1011 0	1.936640-01	4.869930-02	2.991230-02	1.471330-02	2.832030-03
0 3 3	1016 4	6.739030-04	6.718150-04	1.570260-04	1.550610-04	0.0
0 3 3	1112 1	1.652610-01	0.0	0.0	0.0	0.0
0 3 3	12 1 0	1.652590-01	1.820980-01	1.115670-01	9.428360-02	6.975780-01
0 3 3	17 1 0	1.644770-01	8.037450-01	2.347300-01	1.030330-01	3.074360-01
0 4 3	1 0 0	4.085040	00	4.520750	00	4.562780
0 4 3	6 0 0	7.015220	00	8.414750	00	9.901570
0 4 3	11 0 0	1.309450	01	1.417680	01	1.491170
0 4 3	16 0 0	1.531190	01	1.247840	01	1.159370
0 4 3	21 0 0	1.149550	01	9.922430	00	0.0
0 0 0	4 0 0	LIB. MTL. 4, MICRO NA				
0 1 4	1 2 0	8.781560-03	0.0	7.105140-05	1.923410-04	3.075600-04
0 1 4	6 2 0	4.808560-04	6.776740-04	9.038600-04	1.555140-03	2.999340-03
0 1 4	11 2 0	2.491750-03	1.592920-03	1.247840-03	1.260520-03	4.680910-02
0 1 4	16 2 0	6.295120-02	1.622520-02	7.645380-03	6.509570-03	6.297270-03
0 3 4	1 2 0	2.985140-01	2.101250-01	1.834180-01	1.152090-01	5.886530-02
0 3 4	1 7 0	2.646890-02	1.100390-02	4.360640-03	1.678330-03	6.345920-04
0 3 4	112 2	2.352810-04	4.824850-05	0.0	0.0	0.0
0 3 4	2 3 0	6.685260-01	7.864070-02	9.356140-02	4.929330-02	1.949870-02
0 3 4	2 8 0	7.550770-03	2.808960-03	9.267190-04	3.184910-04	1.194440-04
0 3 4	213 1	2.762760-05	0.0	0.0	0.0	0.0
0 3 4	3 4 0	1.5828180-01	0.0	0.0	0.0	0.0
0 3 4	4 5 2	8.331250-01	1.073850-01	0.0	0.0	0.0
0 3 4	5 6 0	7.210480-01	1.089310-01	4.327040-02	1.206750-02	1.416750-03
0 3 4	6 7 0	6.481790-01	0.0	0.0	1.719640-03	1.140440-03
0 3 4	612 2	4.343170-04	1.669270-04	0.0	0.0	0.0
0 3 4	7 8 1	5.967810-01	0.0	0.0	0.0	0.0
0 3 4	8 9 1	6.492110-01	0.0	0.0	0.0	0.0
0 3 4	910 1	6.630620-01	0.0	0.0	0.0	0.0
0 4 4	1 0 0	5.878200-01	8.085940-01	3.690250-01	6.559630-01	9.779840-01
0 4 4	15 1 0	9.776990-01	2.320690-01	2.073980-01	8.132050-01	2.677450-01
0 4 4	20 1 0	1.598990-01	8.419320-02	0.0	0.0	0.0
0 4 4	1 0 0	1.297900-00	1.936010-00	1.947780-00	2.818360-00	4.530350-00
0 4 4	6 0 0	3.706260	00	3.931340	00	3.904950
0 4 4	11 0 0	3.855360	00	4.076080	00	4.940610
0 4 4	16 0 0	9.741050	00	4.315160	00	3.054330
0 4 4	21 0 2	3.021760	00	3.026640	00	0.0
0 0 0	5 0 0	LIB. MTL. 5, MACRO STAINLESS STEEL				
0 1 5	1 2 0	4.042980-03	1.523000-03	3.075660-04	2.500640-04	3.942260-04
0 1 5	6 2 0	4.354780-04	5.118400-04	6.362910-04	9.112440-04	1.277400-03
0 1 5	11 2 0	2.182990-03	1.571020-03	2.265760-03	4.710290-03	2.315050-03
0 1 5	16 2 0	1.928250-03	3.279940-03	3.268380-02	3.575720-03	3.281490-03
0 3 5	1 2 0	3.027280-02	2.718370-02	2.493550-02	1.563660-02	8.022670-03
0 3 5	1 7 0	3.487320-03	1.364030-03	5.450620-04	2.074590-04	7.710990-05
0 3 5	112 2	2.854140-05	5.706160-06	0.0	0.0	0.0
0 3 5	2 3 0	5.620840-02	1.503980-02	5.660810-03	3.844680-03	2.231090-03
0 3 5	2 8 0	1.072610-03	4.647080-04	1.772330-04	5.876090-05	1.510210-05
0 3 5	213 1	3.043130-07	0.0	0.0	0.0	0.0
0 3 5	3 4 0	3.862270-02	2.499030-02	2.562460-03	1.254110-03	6.472430-04
0 3 5	3 9 0	3.002520-04	1.257310-04	4.588390-05	1.348310-05	3.988330-06
0 3 5	314 1	1.059300-06	0.0	0.0	0.0	0.0
0 3 5	4 5 0	1.493460-02	1.066030-02	5.375340-03	2.633710-03	1.070190-03
0 3 5	410 0	4.210310-04	1.595030-04	6.166640-05	2.280020-05	5.103390-06
0 3 5	5 6 0	1.895880-02	9.617500-05	6.719950-05	3.059080-05	1.568070-05
0 3 5	511 4	6.133880-06	1.458980-06	2.280020-05	5.103390-06	0.0
0 3 5	6 7 1	1.258220-02	0.0	0.0	0.0	0.0
0 3 5	7 8 1	2.340820-02	0.0	0.0	0.0	0.0
0 3 5	8 1 0	1.530660-02	1.936670-02	2.378310-02	2.661370-02	2.863560-02
0 3 5	13 1 0	2.979880-02	2.220760-02	2.513950-02	7.948500-02	1.534440-02
0 3 5	18 1 4	7.245960-02	2.700980-02	1.718250-02	8.103470-03	0.0
0 4 5	1 0 0	4.767880-01	1.815980-01	1.839180-01	3.711240-01	1.968520-01
0 4 5	0 0 0	2.386920-01	2.394270-01	2.979310-01	3.799800-01	9.742800-01
0 4 5	11 0 0	5.787010-01	3.501270-01	6.494300-01	1.178170	00
0 4 5	16 0 0	5.796010-01	6.193720-01	7.170050-01	7.597120-01	8.277090-01
0 4 5	21 0 2	8.766890-01	8.831070-01	0.0	0.0	0.0

BEGIN INITIALIZATION

2.812

INPUT DATA
CONCENTRATION SEARCH

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PROBLEM SPECIFICATION DATA*****
NEWJOB 1 NEW PROBLEM INDICATOR
ILIST 1 INPUT CARD EDIT INDICATOR
MUSOL 0 SOLUTION OR EDIT SKIP INDICATOR
JMAX 22 NUMBER OF NEUTRON GROUPS
IRMAX 3 NUMBER OF REGIONS
NGEO 1 GEOMETRY INDICATOR (SLAB, CYLINDER, OR SPHERE)
NIBDRY 1 INNER (LEFT) BOUNDARY CONDITION TYPE
NUBDRY 6 OUTER (RIGHT) BOUNDARY CONDITION TYPE
IKEN 0 FISSION SOURCE RENORMALIZATION (OR NOT)
IADJ 0 TYPE OF FLUXES COMPUTED (REGULAR, ADJOINT, OR BOTH)
IGENOP 0 TYPE OF GENERATION FACTOR USED IN SEARCHES
KDALES 1 GROUP OR REGION COALESCE INDICATOR
IPERT 1 AUXILIARY DATA INDICATOR
INBUCK 2 BUCKLING DATA FORM SPECIFICATION
IDCO 0 DIFFUSION COEFFICIENT COMPUTATION METHOD
ISORS 0 TYPE OF AUTOMATIC FIXED VOLUME SOURCE CALCULATION USING FLUXES FROM PREVIOUS PROBLEM
NRMFIX 0 POINTWISE FIXED SOURCE NORMALIZATION
IPLOT 0 AUXILIARY PLOTTING PROBLEM INDICATOR
ICHIM 1 MATERIAL NO. FOR SINGLE FISSION SPECTRUM
INED1 1 INPUT EDIT SPECIFICATION
INED2 1 REGIONWISE MIXTURE DATA EDIT
IOUT1 3 OUTPUT EDIT SPECIFICATION
IOUT3 3 AVERAGE ENERGY EDIT AND/OR REGIONWISE EDIT OF LIB. MTL. REACTION RATES
ILINES 2 MESH POINT SKIP CONTROL FOR FLUX EDITS
ITDATA 1 FISSION SOURCE ITERATION DATA EDIT
INPRNT 0 THERMAL FLUX ITERATION DATA EDIT
LIBED 0 LIBRARY TAPE DATA EDIT
IPUNCH 0 PUNCHED CARD FLUX OUTPUT
ITAPE 0 SAVE PROBLEM DATA ON TAPE INDICATOR
ITOTR 1 TOTAL FLUX EDIT AND NORMALIZING POINT SPECIFICATION
NORMK 0 TYPE OF OUTPUT NORMALIZATION
MESHNO 0 MESH OR SOURCE POINT USED IN RENORMALIZING
NGGRP 0 GROUP NUMBER USED IN RENORMALIZING
INFLUX 0 THERMAL FLUX GUESS SPECIFICATION
INCONV 0 TYPE OF CONVERGENCE FOR THERMAL FLUX GROUPS
MAXINS 3 MAXIMUM NUMBER OF THERMAL FLUX GROUP ITERATIONS (PER OUTER ITERATION)
INSORS 0 FISSION SOURCE GUESS SPECIFICATION
IGESS 0 FISSION SOURCE SAVED FOR NEXT PROBLEM (LAST OR NEXT TO LAST)
NCONVG 1 TYPE OF CONVERGENCE FOR FISSION SOURCE
MAXOUT 30 MAXIMUM NUMBER OF OUTER (FISSION SOURCE) ITERATIONS
NEXO 2 TYPE OF FISSION SOURCE EXTRAPOLATION
NPID 0 NO. OF OUTER ITERATIONS BEFORE TSCHEBYSHEFF EXTRAPOLATION
NCHMAX 15 MAXIMUM ORDER OF TSCHEBYSHEFF POLYNOMIAL
NERCHK 0 NO. OF EXTRA EXTRAPOLATIONS AFTER FINDING A BETTER SIGMA
ISERCH 6 TYPE OF SEARCH AND CONVERGENCE SPECIFICATION
MAXSCH 10 MAXIMUM NUMBER OF SEARCH ITERATIONS
NSERCH 0 SEARCH EIGENVALUE SOURCE INDICATOR
LIBOLD 1 OLD LIBRARY TAPE NUMBER
LIBNEW 1 NEW LIBRARY TAPE NUMBER
LIBNTL 4 NUMBER OF LIBRARY MATERIALS ON DATA CARDS
MESH POINTS AT WHICH FLUX SPECTRA WILL BE EDITED
1, 11, 16, 21,
MATERIALS OR REGIONS TO WHICH SEARCH APPLIES
1, 3,
SPECIFIED LIST OF CONCENTRATION SEARCH LIBRARY MATERIALS
1, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0
0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0
SPECIFIED LIST OF LIB. MTLs. FOR CONSTRAINT ON CONCENTRATION SEARCH
1, 3, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0
0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0
EDIT REACTION RATE RATIO DISTRIBUTION NO. 1, NUMERATOR REACTION TYPE IS 3, DENOMINATOR REACTION TYPE IS 2
NUMERATOR MATERIALS ARE 1, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0
DENOMINATOR MTLs. ARE 1, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0
EDIT REACTION RATE RATIO DISTRIBUTION NO. 4, NUMERATOR REACTION TYPE IS 3, DENOMINATOR REACTION TYPE IS 1
NUMERATOR MATERIALS ARE 3, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0
DENOMINATOR MTLs. ARE 1, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0, 0
EPS1 1.00000000-06 DESIRED FISSION SOURCE CONVERGENCE LEVEL
EPS2 1.00000000-05 DESIRED CONVERGENCE LEVEL FOR SEARCH GENERATION FACTOR
EPS3 9.99999900-05 DESIRED CONVERGENCE LEVEL FOR SEARCH PARAMETER
EPS4 9.99999900-05 DESIRED CONVERGENCE LEVEL FOR THERMAL FLUX INNER ITERATIONS
EPS5 9.99999930-04 CUTOFF LEVEL OF THERMAL FLUX RENORMALIZATION
KADI 0.0 COORDINATE VALUE AT FIRST MESH POINT
THRMX 0.0 EXTRAPOLATION FACTOR FOR THERMAL GROUPS
THETA 0.0 FIXED EXTRAPOLATION FACTOR FOR SOURCE
FISSOR 1.00000000 00 VOLUME INTEGRAL OF FISSION NEUTRON SOURCE
EIGEND 1.01000000 00 DESIRED SEARCH GENERATION FACTOR
EPFISS 3.14000000-11 OPTIONAL ENERGY PER FISSION
FIXSRS 0.0 POINTWISE FIXED SOURCE TOTAL, IF NORMALIZED
ENERGY AVERAGE NEUTRON GROUP RECIPROCAL OF
GROUP NO. NEUTRON ENERGY ENERGY WIDTH GROUP VELOCITY
1 6.84000000 03 6.32000000 03 2.78000000-08
2 2.96000000 03 1.45000000 03 4.20000000-08
3 1.79000000 03 8.80000000 02 5.41000000-08
4 1.09000000 03 5.29000000 02 6.90000000-08
5 6.60000000 02 3.23000000 02 8.93000000-08
6 3.62000000 02 1.96000000 02 1.20000000-07
7 2.43000000 02 1.19000000 02 1.47000000-07
8 1.47000000 02 7.20000000 01 1.87000000-07
9 8.92000000 01 4.36000000 01 2.44000000-07
10 5.42000000 01 2.65000000 01 3.09000000-07
11 3.29000000 01 1.61000000 01 4.08000000-07
12 1.99000000 01 9.80000000 00 5.07000000-07
13 1.21000000 01 5.88000000 00 6.54000000-07
14 6.72000000 01 4.81000000 00 8.85000000-07
15 3.46000000 00 1.70000000 00 1.23000000-06
16 2.33000000 00 5.70000000 01 1.52000000-06
17 1.64000000 00 8.10000000 01 1.79000000-06
18 1.10000000 00 2.68900000 01 2.17000000-06
19 7.72000000 01 3.78200000 01 3.70000000-06
20 3.62000000 01 3.07500000 01 4.13000000-06
21 1.88000000 01 1.74100000 01 5.26000000-06
22 6.80000000 02 1.01300000 01 8.77000000-06

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BOUNDARY GROUP	CONDITION	CONSTANTS FOR INNER (LEFT) BOUNDARY	OMEGA1
ALPHA1	BETA1	GAMMA1	
1	0.0	1.00000000 00	0.0
2	0.0	1.00000000 00	0.0
3	0.0	1.00000000 00	0.0
4	0.0	1.00000000 00	0.0
5	0.0	1.00000000 00	0.0
6	0.0	1.00000000 00	0.0
7	0.0	1.00000000 00	0.0
8	0.0	1.00000000 00	0.0
9	0.0	1.00000000 00	0.0
10	0.0	1.00000000 00	0.0
11	0.0	1.00000000 00	0.0
12	0.0	1.00000000 00	0.0
13	0.0	1.00000000 00	0.0
14	0.0	1.00000000 00	0.0
15	0.0	1.00000000 00	0.0
16	0.0	1.00000000 00	0.0
17	0.0	1.00000000 00	0.0
18	0.0	1.00000000 00	0.0
19	0.0	1.00000000 00	0.0
20	0.0	1.00000000 00	0.0
21	0.0	1.00000000 00	0.0
22	0.0	1.00000000 00	0.0

BOUNDARY GROUP	CONDITION	CONSTANTS FOR OUTER (RIGHT) BOUNDARY	OMEGA2
ALPHA2	BETA2	GAMMA2	
1	1.00000000 00	2.07562160 00	0.0
2	1.00000000 00	2.07997320 00	0.0
3	1.00000000 00	2.08060000 00	0.0
4	1.00000000 00	2.07738360 00	0.0
5	1.00000000 00	2.08473840 00	0.0
6	1.00000000 00	2.09232190 00	0.0
7	1.00000000 00	2.09249190 00	0.0
8	1.00000000 00	2.09984950 00	0.0
9	1.00000000 00	2.10646590 00	0.0
10	1.00000000 00	2.10617270 00	0.0
11	1.00000000 00	2.11484630 00	0.0
12	1.00000000 00	2.10442430 00	0.0
13	1.00000000 00	2.11665150 00	0.0
14	1.00000000 00	2.12321010 00	0.0
15	1.00000000 00	2.12060560 00	0.0
16	1.00000000 00	2.11512670 00	0.0
17	1.00000000 00	2.11592860 00	0.0
18	1.00000000 00	2.11795760 00	0.0
19	1.00000000 00	2.11869950 00	0.0
20	1.00000000 00	2.11972610 00	0.0
21	1.00000000 00	2.12036800 00	0.0
22	1.00000000 00	2.12044700 00	0.0

REGION DIMENSION AND MESH DATA

REGION NUMBER	REGION WIDTH	UPPER VALUE	MESH SPACING	MATERIAL NUMBER	NUMBER OF INTERVALS	MESH POINTS	SOURCE POINTS
1	2.40000000 01	2.40000000 01	1.50000000 00	1	16	1 TO 17	1 TO 17
2	5.00000000 01	7.40000000 01	2.50000000 00	3	20	17 TO 37	18 TO 38
3	8.00000000 00	8.20000000 01	2.00000000 00	2	4	37 TO 41	39 TO 43

MACROSCOPIC MATERIAL CONSTANTS (IN ORDER OF USAGE IN REGIONS) THE TOTAL NUMBER OF MATERIALS USED IS 3

MATERIAL NUMBER: 3

POISON CROSS SECTION= 0.0 , DOWNSCATTER GROUPS= 12, UPSCATTER GROUPS= 0
NO. OF THERMAL GROUPS IS 17 FIRST THERMAL GROUP IS 22

MATERIAL COMPOSITION BY LIBRARY MATERIAL AND CONCENTRATION
(NEGATIVE NOS. ARE MACROSCOPIC MATERIALS FROM THE PRECEEDING PROBLEM)

LIBRARY MATERIAL 1 3 5
CONCENTRATION 6.1999992E-05 2.39999996E-02 2.1999997E-01

NEUTRON GROUP	DIFFUSION	ABSORPTION	BUCKLING	NU TIMES	FISSION	FISSION	ENERGY
GROUP NUMBER	COEFFICIENT	CROSS SECTION	SQUARED	SIGMA FISSON	SPECTRUM, CHI	CROSS SECTION	PER FISSON
1	2.4648705E-00	1.6291969E-02	1.9899998E-03	4.7634806E-02	1.3034999E-01	1.4944457E-02	3.1399980E-11
2	2.2408190E-00	1.4742251E-02	1.9799999E-03	3.8287081E-02	2.1089000E-01	1.3836756E-02	3.1399980E-11
3	2.2183008E-00	1.1488665E-02	1.9599998E-03	2.6083630E-02	2.3222995E-01	9.9685602E-03	3.1399980E-11
4	2.2586918E-00	3.8664902E-03	1.8099998E-03	1.7347087E-03	1.7940998E-01	6.8822965E-04	3.1399980E-11
5	1.8894973E-00	3.2907347E-03	1.6500000E-03	2.5197165E-04	1.1486995E-01	1.0166080E-04	3.1399980E-11
6	1.5063076E-00	2.9666850E-03	1.4999998E-03	1.8862874E-04	6.4840972E-02	7.6610231E-05	3.1399980E-11
7	1.3064861E-00	3.5556580E-03	1.4099998E-03	2.0581139E-04	3.4096997E-02	8.4065119E-05	3.1399980E-11
8	1.0972490E-00	4.8493706E-03	1.3100000E-03	2.2937139E-04	1.7124999E-02	9.4018629E-05	3.1399980E-11
9	9.3354535E-01	6.5672025E-03	1.2299998E-03	2.5748205E-04	8.3958991E-03	1.0576018E-04	3.1399980E-11
10	8.7266123E-01	8.5042827E-03	1.1399998E-03	2.9492634E-04	4.0696971E-03	1.2129116E-04	3.1399980E-11
11	7.5344890E-01	1.0390725E-02	1.0700000E-03	3.4091296E-04	1.9532000E-03	1.4030347E-04	3.1399980E-11
12	7.9715192E-01	1.2625016E-02	9.9399989E-04	3.9834599E-04	9.3328347E-04	1.6401347E-04	3.1399980E-11
13	6.6445118E-01	1.6015075E-02	8.9999987E-04	4.6721008E-04	4.3951999E-04	1.9242313E-04	3.1399980E-11
14	5.3638804E-01	2.0227768E-02	8.1899995E-04	5.8659911E-04	2.6676990E-04	2.4163999E-04	3.1399980E-11
15	5.8244824E-01	1.7943874E-02	6.8399985E-04	7.4416096E-04	6.8146997E-05	3.0657812E-04	3.1399980E-11
16	6.7139393E-01	2.1351144E-02	5.6399987E-04	8.9304778E-04	1.8458988E-05	3.6793319E-04	3.1399980E-11
17	7.6243949E-01	1.9814301E-02	1.5299999E-04	1.0141523E-03	2.2378998E-05	4.1783834E-04	3.1399980E-11
18	7.6152939E-01	3.5084121E-02	1.0999999E-05	1.2072136E-03	6.1188994E-06	4.9739098E-04	3.1399980E-11
19	7.5864720E-01	3.3100910E-02	9.9999997E-06	1.3614511E-03	7.1595996E-06	5.6094420E-04	3.1399980E-11
20	7.2201693E-01	3.1645875E-02	9.9999997E-06	1.6704963E-03	4.4061999E-06	6.8285266E-04	3.1399980E-11
21	7.0813054E-01	4.2297475E-02	9.9999997E-06	2.2062198E-03	1.5640999E-06	9.0902392E-04	3.1399980E-11
22	7.6657832E-01	4.5641378E-02	9.9999997E-06	2.8636926E-03	3.9269997E-07	1.1799259E-03	3.1399980E-11

GROUP NUMBER	TOTAL REMOVAL CROSS SECTION	TRANSPORT CROSS SECTION	SIGMA POISON MULTIPLIER	LOGARITHMIC BOUNDARY CONST.	NORMALIZATION CROSS SECTION
1	1.04931870-01	1.3523358E-01	0.0	0.0	2.11970580-02
2	9.90311330-02	1.4875507E-01	0.0	0.0	1.91790720-02
3	8.49687380-02	1.5026516E-01	0.0	0.0	1.58365330-02
4	4.26047770-02	1.4757800E-01	0.0	0.0	7.95472060-03
5	2.46260300-02	1.7830098E-01	0.0	0.0	6.37540520-03
6	2.07874640-02	2.2129166E-01	0.0	0.0	5.22614600-03
7	2.26781070-02	2.5513715E-01	0.0	0.0	5.39780320-03
8	2.30157440-02	3.0378991E-01	0.0	0.0	6.28676660-03
9	2.53203290-02	3.5706174E-01	0.0	0.0	7.71546290-03
10	2.17421970-02	3.8197333E-01	0.0	0.0	9.49911630-03
11	2.10276830-02	4.4240999E-01	0.0	0.0	1.11969150-02
12	2.32113630-02	4.1815531E-01	0.0	0.0	1.34173850-02
13	2.75475010-02	5.0166714E-01	0.0	0.0	1.66130810-02
14	2.82372090-02	6.2144065E-01	0.0	0.0	2.06670690-02
15	2.61405600-02	5.7229686E-01	0.0	0.0	1.83422680-02
16	5.11911490-02	4.9647951E-01	0.0	0.0	2.17298100-02
17	2.72624230-02	4.3719316E-01	0.0	0.0	1.97309550-02
18	7.03573290-02	4.3771565E-01	0.0	0.0	3.50924980-02
19	4.746942240-02	4.3937856E-01	0.0	0.0	3.31084970-02
20	3.79121120-02	4.6166968E-01	0.0	0.0	3.16530960-02
21	5.14683160-02	4.7072297E-01	0.0	0.0	4.23045560-02
22	4.56490430-02	4.3483269E-01	0.0	0.0	4.56490430-02

SCATTERING MATRIX (ROW INDEX IS GROUP SCATTERED FROM)

1	2	3	4	5	6	7
1	0.0	1.3449777E-02	1.9445546E-02	2.0841464E-02	1.4945451E-02	8.3412267E-03
2	0.0	0.0	2.2658806E-02	1.9454263E-02	1.6532540E-02	1.0924555E-02
3	0.0	0.0	0.0	2.4635077E-02	2.1276642E-02	1.2282852E-02
4	0.0	0.0	0.0	0.0	1.5602894E-02	8.8701472E-03
5	0.0	0.0	0.0	0.0	0.0	1.6341701E-02
6	0.0	0.0	0.0	0.0	0.0	1.5094869E-02
7	0.0	0.0	0.0	0.0	0.0	0.0
8	0.0	0.0	0.0	0.0	0.0	0.0
9	0.0	0.0	0.0	0.0	0.0	0.0
10	0.0	0.0	0.0	0.0	0.0	0.0
11	0.0	0.0	0.0	0.0	0.0	0.0
12	0.0	0.0	0.0	0.0	0.0	0.0
13	0.0	0.0	0.0	0.0	0.0	0.0
14	0.0	0.0	0.0	0.0	0.0	0.0
15	0.0	0.0	0.0	0.0	0.0	0.0
16	0.0	0.0	0.0	0.0	0.0	0.0
17	0.0	0.0	0.0	0.0	0.0	0.0
18	0.0	0.0	0.0	0.0	0.0	0.0
19	0.0	0.0	0.0	0.0	0.0	0.0
20	0.0	0.0	0.0	0.0	0.0	0.0
21	0.0	0.0	0.0	0.0	0.0	0.0
22	0.0	0.0	0.0	0.0	0.0	0.0
8	9	10	11	12	13	14
1	1.6775734E-03	6.7992415E-04	2.6467512E-04	1.0059832E-04	3.7749502E-05	1.2482736E-05
2	2.6620424E-03	1.1215822E-03	4.4553890E-04	1.7000987E-04	6.2787425E-05	2.2335895E-05
3	2.8617077E-03	1.2289213E-03	4.9644546E-04	1.9240633E-04	7.2336450E-05	2.6925176E-05
4	2.2902787E-03	7.9389662E-04	2.8840522E-04	1.1088263E-04	4.2058426E-05	1.4946777E-05
5	8.2774810E-04	5.4144044E-04	2.4786126E-04	9.5554802E-05	3.6025434E-05	1.8262770E-05
6	4.6170526E-04	3.2718881E-04	1.0741805E-04	2.6853417E-07	8.7109356E-08	3.1247122E-08
7	1.6432565E-02	6.9377210E-04	2.2100657E-04	8.4611893E-05	3.1774005E-05	1.1356738E-05
8	0.0	1.6335741E-02	2.6982720E-04	5.2583549E-05	4.4523258E-05	1.7729690E-05
9	0.0	0.0	1.4370043E-02	2.8566350E-03	3.7742639E-04	4.3009697E-07
10	0.0	0.0	0.0	9.8914951E-03	1.1700301E-03	7.1938499E-04
11	0.0	0.0	0.0	0.0	9.8304451E-03	3.2245134E-07
12	0.0	0.0	0.0	0.0	0.0	9.7931512E-03
13	0.0	0.0	0.0	0.0	0.0	6.7079009E-07
14	0.0	0.0	0.0	0.0	0.0	1.0934420E-02
15	0.0	0.0	0.0	0.0	0.0	0.0
16	0.0	0.0	0.0	0.0	0.0	0.0
17	0.0	0.0	0.0	0.0	0.0	0.0
18	0.0	0.0	0.0	0.0	0.0	0.0
19	0.0	0.0	0.0	0.0	0.0	0.0
20	0.0	0.0	0.0	0.0	0.0	0.0
21	0.0	0.0	0.0	0.0	0.0	0.0
22	0.0	0.0	0.0	0.0	0.0	0.0
15	16	17	18	19	20	21
1	0.0	0.0	0.0	0.0	0.0	0.0
2	0.0	0.0	0.0	0.0	0.0	0.0
3	0.0	0.0	0.0	0.0	0.0	0.0
4	0.0	0.0	0.0	0.0	0.0	0.0
5	0.0	0.0	0.0	0.0	0.0	0.0
6	0.0	0.0	0.0	0.0	0.0	0.0
7	0.0	0.0	0.0	0.0	0.0	0.0
8	3.8802501E-08	9.0453618E-09	5.5142699E-09	0.0	0.0	0.0
9	3.7169919E-08	5.4088034E-08	8.5721794E-09	0.0	0.0	0.0
10	6.8196663E-05	1.6227743E-05	1.6177335E-05	3.7778673E-06	3.7271138E-06	0.0
11	0.0	0.0	0.0	0.0	0.0	0.0
12	9.1022173E-08	2.9070303E-08	3.0284127E-08	5.5197873E-09	0.0	0.0
13	0.0	0.0	0.0	0.0	0.0	0.0
14	7.5701401E-03	0.0	0.0	0.0	0.0	0.0
15	0.0	7.7982917E-03	0.0	0.0	0.0	0.0
16	0.0	0.0	2.9461339E-02	0.0	0.0	0.0
17	0.0	0.0	0.0	7.3314682E-03	0.0	0.0
18	0.0	0.0	0.0	0.0	3.5264831E-02	0.0
19	0.0	0.0	0.0	0.0	0.0	1.1585727E-02
20	0.0	0.0	0.0	0.0	0.0	6.2590167E-03
21	0.0	0.0	0.0	0.0	0.0	0.0
22	0.0	0.0	0.0	0.0	0.0	0.0
22						
1	0.0					
2	0.0					
3	0.0					
4	0.0					
5	0.0					
6	0.0					
7	0.0					
8	0.0					
9	0.0					
10	0.0					
11	0.0					
12	0.0					
13	0.0					
14	0.0					
15	0.0					
16	0.0					
17	0.0					
18	0.0					
19	0.0					
20	0.0					
21	9.1637596E-03					
22	0.0					

BEGIN SOLUTION

0.649

FISSION SOURCE ITERATION DATA (DESIRED CONVERGENCE LEVEL IS 1.0000000E-06, MAXIMUM NO. OF ITERATIONS IS 30)

ITER. NO.	GENERATION FACTOR	CONVERGENCE LEVEL	MAXIMUM LAMBDA	PT. NO.	MINIMUM LAMBDA	PT. CHEBY NO. ORDER	DOMINANCE RATIO	ERROR RATIO	UPDATED DOMINANCE
1	4.52049140-01	5.12405690 00	5.44953520 00	17	0.32547830 00	38	0 0.0	5.124060 00	0.0
2	7.37999930-01	1.66733860 00	1.95076120 00	1	0.28342270 00	38	0 0.0	3.253940-01	0.0
3	9.66850260-01	1.01346210 00	1.26499970 00	1	0.25153160 00	38	0 0.0	6.078320-01	0.0
4	1.05649680 00	7.94310840-01	1.07833590 00	1	0.28402500 00	37	0 0.0	7.837600-01	0.0
5	1.08200350 00	5.98661680-01	1.02423760 00	1	0.42557590 00	37	0 0.0	7.536870-01	0.0
6	1.09272370 00	1.76820100 00	2.76616240 00	37	0.99796140 00	6	1 7.536870-01	2.953590 00	0.0
7	1.09270030 00	6.03406560-01	1.59842330 00	37	0.99501670 00	1	2 7.536870-01	3.412550-01	0.0
8	1.09058980 00	1.30498620-01	1.00076580 00	1	0.87026720 00	37	3 7.536870-01	2.162700-01	0.0
9	1.09103340 00	3.64537970-03	1.00437550 00	37	0.99873010 00	29	4 7.536870-01	4.326010-02	0.0
10	1.09115970 00	1.58814020-02	1.01574620 00	37	0.99986480 00	1	5 7.536870-01	2.813170 00	0.0
11	1.09109260 00	4.50360340-03	1.00006790 00	20	0.99556430 00	37	6 7.536870-01	2.835770-01	0.0
12	1.09109410 00	4.92486390-04	1.00001940 00	2	0.99952690 00	36	7 7.536870-01	1.093540-01	0.0
13	1.09109960 00	6.02406410-04	1.00005950 00	37	0.99999340 00	20	8 7.536870-01	1.223190 00	0.0
14	1.09109840 00	1.10287990-04	1.00000360 00	21	0.99989340 00	37	9 7.536870-01	1.830800-01	0.0
15	1.09109820 00	3.68900410-05	1.00000040 00	18	0.99998360 00	37	10 7.536870-01	3.344880-01	0.0
16	1.09109840 00	2.24704900-05	1.00002190 00	37	0.99999940 00	21	11 7.536870-01	6.091210-01	0.0
17	1.09109830 00	2.34355480-06	1.00000030 00	18	0.99999800 00	37	12 7.536870-01	1.042950-01	0.0
18	1.09109830 00	2.29404060-06	1.00000020 00	20	0.99999790 00	36	13 7.536870-01	9.790270-01	0.0
19	1.09109830 00	1.02984680-06	1.00000090 00	36	0.99999990 00	20	14 7.536870-01	4.488510-01	0.0
20	1.09109840 00	1.39720430-07	1.00000010 00	38	0.99999990 00	31	15 7.536870-01	1.356710-01	0.0

SEARCH ITERATION NO. 1 CONVERGED ON FISSION SOURCE ITERATION 20, (DESIRED LAMBDA IS 1.01000000 00).
OBTAINED LAMBDA IS 1.09109840 00, SEARCH CONVERGENCE LEVEL IS 8.02953970-02 (DESIRED CONVERGENCE LEVEL IS 1.0000-05).
PARAMETER VALUE IS 1.00000000 00
BEGIN INITIALIZATION 7.022
BEGIN SOLUTION 0.017

FISSION SOURCE ITERATION DATA (DESIRED CONVERGENCE LEVEL IS 1.0000000E-06, MAXIMUM NO. OF ITERATIONS IS 30)

ITER. NO.	GENERATION FACTOR	CONVERGENCE LEVEL	MAXIMUM LAMBDA	PT. NO.	MINIMUM LAMBDA	PT. CHEBY NO. ORDER	DOMINANCE RATIO	ERROR RATIO	UPDATED DOMINANCE
1	8.33712380-01	3.56469130-01	1.30600110 00	18	0.94953200 00	17	0 0.0	3.564690-01	0.0
2	8.27999420-01	9.68026340-02	1.08979660 00	27	0.99299400 00	1	0 0.0	2.715600-01	0.0
3	8.26234690-01	4.20967530-02	1.03919740 00	31	0.99710060 00	1	0 0.0	4.348720-01	0.0
4	8.25666700-01	1.85192730-02	1.01744520 00	34	0.99892590 00	1	0 0.0	4.399220-01	0.0
5	8.25480430-01	7.82850270-03	1.00745370 00	36	0.99962520 00	1	0 0.0	4.227220-01	0.0
6	8.25402580-01	1.85141000-03	1.00179100 00	37	0.99993960 00	1	1 4.227220-01	2.364960-01	0.0
7	8.25387630-01	1.29059160-04	1.00005980 00	38	0.99993070 00	28	2 4.227220-01	6.970860-02	0.0
8	8.25388460-01	5.52212640-05	1.00000210 00	2	0.99994670 00	36	3 4.227220-01	4.278760-01	0.0
9	8.25388870-01	1.25260820-05	1.00000020 00	5	0.99998770 00	37	4 4.227220-01	2.268340-01	0.0
10	8.25388920-01	6.51234160-07	1.00000030 00	31	0.99999960 00	37	5 4.227220-01	5.199030-02	4.301240-01

SEARCH ITERATION NO. 2 CONVERGED ON FISSION SOURCE ITERATION 10, (DESIRED LAMBDA IS 1.01000000 00).
OBTAINED LAMBDA IS 8.25388920-01, SEARCH CONVERGENCE LEVEL IS 1.82783250-01 (DESIRED CONVERGENCE LEVEL IS 1.0000-05).
PARAMETER VALUE IS 5.54645330-01
BEGIN INITIALIZATION 3.977
BEGIN SOLUTION 0.017

FISSION SOURCE ITERATION DATA (DESIRED CONVERGENCE LEVEL IS 1.0000000E-06, MAXIMUM NO. OF ITERATIONS IS 30)

ITER. NO.	GENERATION FACTOR	CONVERGENCE LEVEL	MAXIMUM LAMBDA	PT. NO.	MINIMUM LAMBDA	PT. CHEBY NO. ORDER	DOMINANCE RATIO	ERROR RATIO	UPDATED DOMINANCE
1	9.97645380-01	2.11138410-01	1.04011690 00	17	0.82897850 00	18	0 0.0	2.111380-01	0.0
2	1.00358290 00	6.86832850-02	1.00519700 00	1	0.93651370 00	27	0 0.0	3.253000-01	0.0
3	1.00519830 00	2.86838530-02	1.00207740 00	1	0.97399350 00	31	0 0.0	4.717620-01	0.0
4	1.00565940 00	1.12373170-02	1.00074830 00	1	0.98951100 00	34	0 0.0	3.917650-01	0.0
5	1.00579470 00	4.08285830-03	1.00025430 00	1	0.99617140 00	36	0 0.0	3.633300-01	0.0
6	1.00584400 00	7.76717530-04	1.00004590 00	1	0.99926910 00	37	1 3.633300-01	1.902390-01	0.0
7	1.00585280 00	3.23338620-05	1.00001660 00	29	0.99998430 00	37	2 3.633300-01	4.162890-02	0.0
8	1.00585280 00	1.31521270-05	1.00001270 00	36	0.99999960 00	7	3 3.633300-01	4.067600-01	0.0
9	1.00585270 00	2.02432360-06	1.00000010 00	37	0.99999990 00	9	4 3.633300-01	1.539160-01	0.0
10	1.00585270 00	2.93745530-07	1.00000010 00	24	0.99999990 00	35	5 3.633300-01	1.451080-01	0.0

SEARCH ITERATION NO. 3 CONVERGED ON FISSION SOURCE ITERATION 10, (DESIRED LAMBDA IS 1.01000000 00).
OBTAINED LAMBDA IS 1.00585270 00, SEARCH CONVERGENCE LEVEL IS 4.10625200-03 (DESIRED CONVERGENCE LEVEL IS 1.0000-05).
PARAMETER VALUE IS 8.35352440-01
BEGIN INITIALIZATION 4.227
BEGIN SOLUTION 0.017

FISSION SOURCE ITERATION DATA (DESIRED CONVERGENCE LEVEL IS 1.0000000E-06, MAXIMUM NO. OF ITERATIONS IS 30)

ITER. NO.	GENERATION FACTOR	CONVERGENCE LEVEL	MAXIMUM LAMBDA	PT. NO.	MINIMUM LAMBDA	PT. CHEBY NO. ORDER	DOMINANCE RATIO	ERROR RATIO	UPDATED DOMINANCE
1	1.00984930 00	4.68369160-03	1.00074990 00	17	0.99606620 00	18	0 0.0	4.683690-03	0.0
2	1.00995440 00	1.27267290-03	1.00008910 00	1	0.99881640 00	27	0 0.0	2.717240-01	0.0
3	1.00998250 00	4.96089190-04	1.00003570 00	2	0.99953970 00	31	0 0.0	3.897980-01	0.0
4	1.00999050 00	1.89674620-04	1.00001300 00	1	0.99982330 00	34	0 0.0	3.823430-01	0.0
5	1.00999280 00	6.87326690-05	1.00000450 00	1	0.99993580 00	36	0 0.0	3.623710-01	0.0
6	1.00999370 00	1.32836230-05	1.00000080 00	3	0.99998750 00	35	1 3.623710-01	1.932650-01	0.0
7	1.00999390 00	8.03654080-07	1.00000040 00	27	0.99999960 00	37	2 3.623710-01	6.049960-02	0.0

SEARCH ITERATION NO. 4 CONVERGED ON FISSION SOURCE ITERATION 7, (DESIRED LAMBDA IS 1.01000000 00).
OBTAINED LAMBDA IS 1.00999390 00, SEARCH CONVERGENCE LEVEL IS 6.07321030-06 (DESIRED CONVERGENCE LEVEL IS 1.0000-05).
PARAMETER VALUE IS 8.42812510-01
BEGIN OUTPUT EDIT 2.047

22 GROUP MONA SAMPLE PROBLEM

OUTPUT EDIT
*****

FINAL SOURCE ITERATION DATA

GENERATION FACTOR ***** 1.00999390 00
SOURCE ITERATIONS ***** 7
SOURCE CONVERGENCE LEVEL ***** 8.03654080-07
DESIRED CONVERGENCE LEVEL ***** 1.00000000-06
MAXIMUM LAMBDA ***** 1.00000040 00
SOURCE POINT NUMBER ***** 27
MINIMUM LAMBDA ***** 9.99999620-01
SOURCE POINT NUMBER ***** 37

FINAL SEARCH ITERATION DATA

DESIRED GENERATION FACTOR *** 1.01000000 00
GENERATION FACTOR ***** 1.00999390 00
SEARCH ITERATIONS ***** 4
SEARCH CONVERGENCE LEVEL ***** 6.07321030-06
DESIRED CONVERGENCE LEVEL *** 1.00000000-05
SEARCH PARAMETER ***** 8.42812510-01

FISSION SOURCE AND FLUX OUTPUT

MESH POINT	MESH POINT POSITION	FISSION SOURCE DENSITY	POWER DENSITY	NORMALIZED FLUXES			
				GROUP NO. 1	2	3	4
*****[REGION 1, MATERIAL 1]*****							
1	0.0	6.4109441D-04	7.9192566E-15	9.1324747E-04	1.7334609E-03	2.6499359E-03	3.6239494E-03
3	3.0000000D 00	6.3634896D-04	7.8611061E-15	9.0523739E-04	1.7181160E-03	2.6258002E-03	3.5919661E-03
5	6.0000000D 00	6.2223778D-04	7.6882538E-15	8.8118576E-04	1.6720383E-03	2.5534439E-03	3.4965118E-03
7	9.0000000D 00	5.9913858D-04	7.4055515E-15	8.4098848E-04	1.5950236E-03	2.4329575E-03	3.3390871E-03
9	1.2000000D 01	5.6767375D-04	7.0210188E-15	7.8432867E-04	1.4864686E-03	2.2642773E-03	3.1221886E-03
11	1.5000000D 01	5.2870252D-04	6.3458184E-15	7.1043638E-04	1.3449327E-03	2.0469507E-03	2.8493367E-03
13	1.8000000D 01	4.8331058D-04	5.9941947E-15	6.1769388E-04	1.1674243E-03	1.7791437E-03	2.5251762E-03
15	2.1000000D 01	4.3280247D-04	5.3835483E-15	5.0301943E-04	9.4829244E-04	1.4579650E-03	2.1557326E-03
17	2.4000000D 01	3.7870149D-04	4.7346262E-15	3.6090915E-04	6.7752833E-04	1.0775877E-03	1.7489505E-03
*****[REGION 2, MATERIAL 3]*****							
17	2.4000000D 01	7.5880293D-05	8.6058399E-16	3.6090915E-04	6.7752833E-04	1.0775877E-03	1.7489505E-03
19	2.9000000D 01	3.2295187D-05	3.6926126E-16	1.3945298E-04	2.6463764E-04	4.6633091E-04	1.0065564E-03
21	3.4000000D 01	1.4325278D-05	1.6527147E-16	5.5752214E-05	1.0664608E-04	2.0316517E-04	5.5641797E-04
23	3.9000000D 01	6.6542773D-06	7.7536046E-17	2.3103377E-05	4.4419969E-05	8.9501627E-05	3.0074012E-04
25	4.4000000D 01	3.2527443D-06	3.8303584E-17	9.9706995E-06	1.9213490E-05	4.0100087E-05	1.6053506E-04
27	4.9000000D 01	1.6768483D-06	1.9955183E-17	4.5096967E-06	8.6854091E-06	1.8399645E-05	8.5188862E-05
29	5.4000000D 01	9.0935036D-07	1.0926986E-17	2.1494770E-06	4.1284266E-06	8.7097587E-06	4.5152265E-05
31	5.9000000D 01	5.1475118D-07	6.2365630E-18	1.0808471E-06	2.0674088E-06	4.2781830E-06	2.3986155E-05
33	6.4000000D 01	3.0037380D-07	3.6636020E-18	5.6813190E-07	1.0832655E-06	2.1822389E-06	1.2797113E-05
35	6.9000000D 01	1.7746344D-07	2.1771940E-18	3.0100968E-07	5.7719063E-07	1.1437514E-06	6.8550708E-06
37	7.4000000D 01	1.0303035D-07	1.2745670E-18	1.3761934E-07	2.8060492E-07	5.9474803E-07	3.6724541E-06
*****[REGION 3, MATERIAL 2]*****							
37	7.4000000D 01	0.0	0.0	1.3761934E-07	2.8060492E-07	5.9474803E-07	3.6724541E-06
39	7.8000000D 01	0.0	0.0	5.2754638E-08	1.2246528E-07	3.0073568E-07	2.0094021E-06
41	8.2000000D 01	0.0	0.0	2.0741414E-08	4.9924481E-08	1.3078665E-07	9.3359137E-07

FLUX OUTPUT (CONTINUED)

MESH POINT	MESH POINT POSITION	FLUXES				
		GROUP NO. 5	6	7	8	9
*****[REGION 1, MATERIAL 1]*****						
1	0.0	4.5102313E-03	4.7844648E-03	3.6154268E-03	2.5553217E-03	1.4530083E-03
3	3.0000000D 00	4.4734180E-03	4.7479752E-03	3.5915443E-03	2.5402007E-03	1.4454520E-03
5	6.0000000D 00	4.3639466E-03	4.6468899E-03	3.5208999E-03	2.4955883E-03	1.4232711E-03
7	9.0000000D 00	4.1848011E-03	4.4797696E-03	3.4065577E-03	2.4237705E-03	1.3879621E-03
9	1.2000000D 01	3.9410517E-03	4.2550042E-03	3.2536844E-03	2.3285996E-03	1.3420768E-03
11	1.5000000D 01	3.6400980E-03	3.9821379E-03	3.0866613E-03	2.2155680E-03	1.2893227E-03
13	1.8000000D 01	3.2920246E-03	3.6740052E-03	2.8642530E-03	2.0919221E-03	1.2347249E-03
15	2.1000000D 01	2.9102757E-03	3.3472523E-03	2.6498605E-03	1.9686285E-03	1.1848751E-03
17	2.4000000D 01	2.5127055E-03	3.0231506E-03	2.4418777E-03	1.8516099E-03	1.1483124E-03
*****[REGION 2, MATERIAL 3]*****						
17	2.4000000D 01	2.5127055E-03	3.0231506E-03	2.4418777E-03	1.8516099E-03	1.1483124E-03
19	2.9000000D 01	1.7028858E-03	2.2541746E-03	1.9129175E-03	1.5275516E-03	1.0103595E-03
21	3.4000000D 01	1.0868683E-03	1.5692972E-03	1.3980512E-03	1.1694161E-03	8.1029045E-04
23	3.9000000D 01	6.7053013E-04	1.0488543E-03	9.7905356E-04	8.5427752E-04	6.1476626E-04
25	4.4000000D 01	4.0508644E-04	6.8240357E-04	6.6581555E-04	6.0388213E-04	4.4896617E-04
27	4.9000000D 01	2.4141758E-04	4.3561612E-04	4.4302549E-04	4.1634147E-04	3.1858380E-04
29	5.4000000D 01	1.4256799E-04	2.7474388E-04	2.8963434E-04	2.8121914E-04	2.2081213E-04
31	5.9000000D 01	8.3637948E-05	1.7055646E-04	1.8627745E-04	1.8644493E-04	1.4982620E-04
33	6.4000000D 01	4.8765345E-05	1.0498868E-04	1.1743244E-04	1.2118365E-04	9.9387165E-05
35	6.9000000D 01	2.8165217E-05	6.3809697E-05	7.1404938E-05	7.6780241E-05	6.3921441E-05
37	7.4000000D 01	1.5911195E-05	3.7968595E-05	3.9510865E-05	4.6918547E-05	3.8776692E-05
*****[REGION 3, MATERIAL 2]*****						
37	7.4000000D 01	1.5911195E-05	3.7968595E-05	3.9510865E-05	4.6918547E-05	3.8776692E-05
39	7.8000000D 01	9.0240246E-06	2.2406355E-05	2.2373963E-05	2.8497627E-05	2.2468881E-05
41	8.2000000D 01	4.0005298E-06	2.2764185E-06	9.0852172E-06	1.0639964E-05	7.0899132E-06

FLUX OUTPUT (CONTINUED)

MESH POINT	MESH POINT POSITION	FLUXES				
		GROUP NO. 11	12	13	14	15
*****[REGION 1, MATERIAL 1]*****						
1	0.0	3.6930782E-04	1.7158414E-04	5.8742604E-05	2.2709210E-05	5.6038634E-06
3	3.0000000D 00	3.6817277E-04	1.7125506E-04	5.8646881E-05	2.2659762E-05	5.5929377E-06
5	6.0000000D 00	3.6496506E-04	1.7036058E-04	5.8397723E-05	2.2528577E-05	5.5654064E-06
7	9.0000000D 00	3.6030193E-04	1.6918572E-04	5.8114776E-05	2.2371198E-05	5.5385926E-06
9	1.2000000D 01	3.5527465E-04	1.6821752E-04	5.8013189E-05	2.2293898E-05	5.5468236E-06
11	1.5000000D 01	3.5155844E-04	1.6816158E-04	5.8432837E-05	2.2480977E-05	5.6523650E-06
13	1.8000000D 01	3.5159476E-04	1.6995837E-04	5.9891667E-05	2.3250323E-05	5.9676022E-06
15	2.1000000D 01	3.5888981E-04	1.7478800E-04	6.3182364E-05	2.5165413E-05	6.6992570E-06
17	2.4000000D 01	3.7848786E-04	1.8403864E-04	6.9549409E-05	2.9263654E-05	8.2361712E-06
*****[REGION 2, MATERIAL 3]*****						
17	2.4000000D 01	3.7848786E-04	1.8403864E-04	6.9549409E-05	2.9263654E-05	8.2361712E-06
19	2.9000000D 01	3.9290451E-04	1.9377531E-04	7.8402707E-05	3.5187724E-05	1.0308279E-05
21	3.4000000D 01	3.5061711E-04	1.7813501E-04	7.4099076E-05	3.4026787E-05	1.0324761E-05
23	3.9000000D 01	2.8860453E-04	1.5107216E-04	6.393331E-05	2.9738687E-05	9.2641476E-06
25	4.4000000D 01	2.2545193E-04	1.2128420E-04	5.2133008E-05	2.4435154E-05	7.7713139E-06
27	4.9000000D 01	1.6952855E-04	9.3471928E-05	4.0715502E-05	1.9223880E-05	6.2262034E-06
29	5.4000000D 01	1.2364586E-04	6.9708505E-05	3.0756681E-05	1.4638895E-05	4.8343609E-06
31	5.9000000D 01	8.778259E-05	5.0514704E-05	2.2615612E-05	1.0899515E-05	3.7028585E-06
33	6.4000000D 01	6.0638704E-05	3.5600649E-05	1.6322214E-05	8.0958407E-06	2.9040366E-06
35	6.9000000D 01	4.0575076E-05	2.4337176E-05	1.1849580E-05	6.3652851E-06	2.5382142E-06
37	7.4000000D 01	2.6109410E-05	1.6032718E-05	9.4298075E-06	6.1486171E-06	2.8070317E-06
*****[REGION 3, MATERIAL 2]*****						
37	7.4000000D 01	2.6109410E-05	1.6032718E-05	9.4298075E-06	6.1486171E-06	2.8070317E-06
39	7.8000000D 01	1.4533990E-05	1.0222429E-05	6.5582053E-06	4.9645561E-06	2.4686951E-06
41	8.2000000D 01	3.5099265E-06	3.4573150E-06	1.5822243E-06	7.7503915E-07	4.6684926E-07

FLUX OUTPUT (CONTINUED)

MESH POINT	MESH POINT POSITION	FLUXES GROUP NO.17	18	19	20	21	22
[REGION 1, MATERIAL 1]							
1	0.0	7.1049931E-07	8.6095667E-08	8.0277573E-08	2.9850831E-08	7.4227664E-09	1.4663917E-09
3	3.00000000 00	7.0916650E-07	8.5802128E-08	7.9889674E-08	2.9657755E-08	7.3695112E-09	1.4556241E-09
5	6.00000000 00	7.0604801E-07	8.5010981E-08	7.8796973E-08	2.9095322E-08	7.2119661E-09	1.4236528E-09
7	9.00000000 00	7.0409556E-07	8.4027363E-08	7.7248274E-08	2.8222576E-08	6.9575066E-09	1.3715435E-09
9	1.20000000 01	7.0933344E-07	8.3496559E-08	7.5797573E-08	2.7172728E-08	6.6212458E-09	1.3013994E-09
11	1.50000000 01	7.3315783E-07	8.4708233E-08	7.5629828E-08	2.6238919E-08	6.2334742E-09	1.2176182E-09
13	1.80000000 01	7.9731564E-07	9.0300830E-08	7.9352844E-08	2.6095670E-08	5.8604890E-09	1.1321326E-09
15	2.10000000 01	9.4493248E-07	1.0591430E-07	9.2926598E-08	2.8375165E-08	5.6641731E-09	1.0870513E-09
17	2.40000000 01	1.2659593E-06	1.4412655E-07	1.3040176E-07	3.7188986E-08	6.0775776E-09	1.2553181E-09
[REGION 2, MATERIAL 3]							
17	2.40000000 01	1.2659593E-06	1.4412655E-07	1.3040176E-07	3.7188986E-08	6.0775776E-09	1.2553181E-09
19	2.90000000 01	1.7961174E-06	2.0579432E-07	1.9190531E-07	5.4208634E-08	7.1349007E-09	1.5416244E-09
21	3.40000000 01	1.8887358E-06	2.1646719E-07	2.0232420E-07	5.8371818E-08	7.2344228E-09	1.5163930E-09
23	3.90000000 01	1.7553266E-06	2.0105580E-07	1.8771311E-07	5.5393908E-08	6.7689250E-09	1.3957488E-09
25	4.40000000 01	1.5194819E-06	1.7412918E-07	1.6256632E-07	4.9097352E-08	6.0279604E-09	1.2496304E-09
27	4.90000000 01	1.2583878E-06	1.4457368E-07	1.3540159E-07	4.2033619E-08	5.2588121E-09	1.1266119E-09
29	5.40000000 01	1.0208469E-06	1.1798573E-07	1.1151747E-07	3.6034685E-08	4.7042761E-09	1.0897192E-09
31	5.90000000 01	8.4088850E-07	9.8313762E-08	9.4776738E-08	3.2759583E-08	4.7034447E-09	1.2506680E-09
33	6.40000000 01	7.5079924E-07	8.9252921E-08	8.9189882E-08	3.4424705E-08	5.9710707E-09	1.8359082E-09
35	6.90000000 01	7.9435745E-07	9.5400139E-08	1.0098614E-07	4.5114987E-08	1.0433372E-08	3.2958281E-09
37	7.40000000 01	1.0366657E-06	1.2260074E-07	1.4293846E-07	7.3572096E-08	2.3905184E-08	6.3455907E-09
[REGION 3, MATERIAL 2]							
37	7.40000000 01	1.0366657E-06	1.2260074E-07	1.4293846E-07	7.3572096E-08	2.3905184E-08	6.3455907E-09
39	7.80000000 01	1.0075519E-06	1.2013231E-07	1.5308018E-07	8.6781938E-08	3.5601389E-08	8.5057437E-09
41	8.20000000 01	2.6593841E-07	2.9211648E-08	3.5568316E-08	1.9162126E-08	7.7432780E-09	1.8280868E-09

MESH POINT	MESH POINT POSITION	TOTAL FLUX	NORMALIZED TOTAL FLUX
[REGION 1, MATERIAL 1]			
1	0.0	7.0737100E-02	7.7777772E-01
3	3.00000000 00	2.7035058E-02	9.9257059E-01
5	6.00000000 00	2.6433624E-02	9.7048945E-01
7	9.00000000 00	2.5450177E-02	9.3438298E-01
9	1.20000000 01	2.4113055E-02	8.8529161E-01
11	1.50000000 01	2.2461932E-02	8.2467193E-01
13	1.80000000 01	2.0547926E-02	7.5440074E-01
15	2.10000000 01	1.8434092E-02	6.7679301E-01
17	2.40000000 01	1.6196478E-02	5.9464079E-01
[REGION 2, MATERIAL 3]			
17	2.40000000 01	1.6196478E-02	5.9464079E-01
19	2.90000000 01	1.1645425E-02	4.2755250E-01
21	3.40000000 01	8.1550293E-03	2.9940540E-01
23	3.90000000 01	5.6066774E-03	2.0584469E-01
25	4.40000000 01	3.8005654E-03	1.3953473E-01
27	4.90000000 01	2.5860701E-03	9.3476936E-02
29	5.40000000 01	1.6874189E-03	6.1952240E-02
31	5.90000000 01	1.1060054E-03	4.0606106E-02
33	6.40000000 01	7.1513467E-04	2.6255599E-02
35	6.90000000 01	4.5324932E-04	1.6640687E-02
37	7.40000000 01	2.7797557E-04	1.0205652E-02
[REGION 3, MATERIAL 2]			
37	7.40000000 01	2.7797557E-04	1.0205652E-02
39	7.80000000 01	1.6586711E-04	6.0896787E-03
41	8.20000000 01	5.7285797E-05	2.1032024E-03

NORMALIZED FLUX SPECTRA FOR SPECIFIED MESH POINTS
MESH POINTS ARE 1 11 16 21

GROUP NUMBER	AVERAGE GROUP ENERGY	GROUP ENERGY WIDTH	FLUX SPECTRA POINT 1	11	16	21
1	6.84000000 03	6.32000000 03	5.3052463E-06	5.0045028E-06	3.9806437E-06	1.0817312E-06
2	2.96000000 03	1.45000000 03	4.3891472E-05	4.1293839E-05	3.2649710E-05	9.0188541E-06
3	1.79000000 03	8.80000000 02	1.1055716E-04	1.0355149E-04	8.3669700E-05	2.8310067E-05
4	1.09000000 03	5.29000000 02	2.5151297E-04	2.3979545E-04	2.1346194E-04	1.2897927E-04
5	6.60000000 02	3.23000000 02	5.1266118E-04	5.0172210E-04	4.8464630E-04	4.1261851E-04
6	3.62000000 02	1.96000000 02	8.9621334E-04	9.0450933E-04	9.3743508E-04	9.8180119E-04
7	2.43000000 02	1.19000000 02	1.1154413E-03	1.1484083E-03	1.2339044E-03	1.4406235E-03
8	1.47000000 02	7.20000000 01	1.3030085E-03	1.3699522E-03	1.5288808E-03	1.9916408E-03
9	8.92000000 01	4.36000000 01	1.2235332E-03	1.3165215E-03	1.5413361E-03	2.2789179E-03
10	5.42000000 01	2.65000000 01	1.0645755E-03	1.1848470E-03	1.4769072E-03	2.5352442E-03
11	3.29000000 01	1.61000000 01	8.4216404E-04	9.7213034E-04	1.3148263E-03	2.6704329E-03
12	1.99000000 01	9.80000000 00	6.4281374E-04	7.6392991E-04	1.0528814E-03	2.2289362E-03
13	1.21000000 01	5.88000000 00	3.6678370E-04	4.4241757E-04	6.4666802E-04	1.5451852E-03
14	6.72000000 00	4.81000000 00	1.7333693E-04	2.0807645E-04	3.2216730E-04	8.6746155E-04
15	3.46000000 00	1.70000000 00	1.2102435E-04	1.4802467E-04	2.4892855E-04	7.4474141E-04
16	2.33000000 00	5.70000000 01	7.0798735E-05	8.6402360E-05	1.3504984E-04	3.5823067E-04
17	1.64000000 00	8.10000000 01	3.2204218E-05	4.0296305E-05	7.6605735E-05	2.8593047E-04
18	1.10000000 00	2.68900000 01	1.1755055E-05	1.4024509E-05	2.5959729E-05	9.8713310E-05
19	7.72000000 01	3.78200000 01	7.7930390E-06	8.9027553E-06	1.6366525E-05	6.5599525E-05
20	3.62000000 01	3.07500000 01	3.5640633E-06	3.7988630E-06	5.9183294E-06	2.3277287E-05
21	1.88000000 01	1.74100000 01	1.5653122E-06	1.5939850E-06	1.9052577E-06	5.0954150E-06
22	6.80000000 02	1.01300000 01	5.3146505E-07	5.3512412E-07	6.3893725E-07	1.8355940E-06

NORMALIZED REGION-AVERAGED FLUX SPECTRA

GROUP NUMBER	AVERAGE GROUP ENERGY	GROUP ENERGY WIDTH	FLUX SPECTRA REGION 1	2	3
1	6.84000000 03	6.32000000 03	4.7821795E-06	1.1859092E-06	5.9112793E-08
2	2.96000000 03	1.45000000 03	3.79426013E-05	9.8116643E-06	5.7126988E-07
3	1.79000000 03	8.80000000 02	9.9417273E-05	2.8795548E-05	2.2084241E-06
4	1.09000000 03	5.29000000 02	2.3470464E-04	1.1697707E-04	2.4063076E-05
5	6.60000000 02	3.23000000 02	4.9894955E-04	3.7702522E-04	1.7433747E-04
6	3.62000000 02	1.96000000 02	9.1188052E-04	9.2978799E-04	7.0090103E-04
7	2.43000000 02	1.19000000 02	1.1657383E-03	1.4106038E-03	1.1654310E-03
8	1.47000000 02	7.20000000 01	1.4012700E-03	2.0254534E-03	2.3870820E-03
9	8.92000000 01	4.36000000 01	1.3610851E-03	2.3967742E-03	3.0970417E-03
10	5.42000000 01	2.65000000 01	1.2428248E-03	2.7956513E-03	4.1459464E-03
11	3.29000000 01	1.61000000 01	1.0419125E-03	3.0637465E-03	5.4493248E-03
12	1.99000000 01	9.80000000 00	8.2096830E-04	2.6839659E-04	6.1590672E-03
13	1.21000000 01	5.88000000 00	4.8428634E-04	1.9223420E-03	6.3012838E-03
14	6.72000000 00	4.81000000 00	2.3262655E-04	1.1095903E-03	5.5035688E-03
15	3.46000000 00	1.70000000 00	1.7031611E-04	1.0176993E-03	7.6213628E-03
16	2.33000000 00	5.70000000 01	9.6764721E-05	5.1288027E-04	5.4646768E-03
17	1.64000000 00	8.10000000 01	4.8569505E-05	4.5315037E-04	6.4872503E-03
18	1.10000000 00	2.68900000 01	1.6802514E-05	1.5839841E-04	2.3152763E-03
19	7.72000000 01	3.78200000 01	1.0686717E-05	1.0910969E-04	2.0533646E-03
20	3.62000000 01	3.07500000 01	4.3176251E-06	4.6007393E-05	1.4000630E-03
21	1.88000000 01	1.74100000 01	1.6710701E-06	1.3679635E-05	9.7302347E-04
22	6.80000000 02	1.01300000 01	5.6197212E-07	6.1092678E-06	4.0558889E-04

SPATIAL DISTRIBUTIONS OF MEDIAN AND AVERAGE ENERGIES

MESH POINT	MESH POINT POSITION	MEDIAN NEUTRON ENERGY	AVERAGE NEUTRON ENERGY	MEDIAN FLUX ENERGY	AVERAGE FLUX ENERGY	MEDIAN FISSION ENERGY	AVERAGE FISSION ENERGY	MEDIAN ABSORB ENERGY	AVERAGE ABSORB ENERGY	MEDIAN CAPTURE ENERGY	AVERAGE CAPTURE ENERGY
*** (REGION 1, MATERIAL 1) ***											
1	0.0	2.53E 02	4.87E 02	4.90E 02	9.63E 02	4.94E 02	1.10E 03	4.32E 02	9.71E 02	2.49E 02	5.21E 02
2	1.500000 00	2.53E 02	4.86E 02	4.90E 02	9.62E 02	4.94E 02	1.10E 03	4.31E 02	9.71E 02	2.49E 02	5.21E 02
3	3.000000 00	2.52E 02	4.86E 02	4.90E 02	9.62E 02	4.93E 02	1.10E 03	4.31E 02	9.70E 02	2.49E 02	5.21E 02
4	4.500000 00	2.52E 02	4.85E 02	4.89E 02	9.60E 02	4.92E 02	1.09E 03	4.29E 02	9.68E 02	2.48E 02	5.19E 02
5	6.000000 00	2.51E 02	4.83E 02	4.87E 02	9.58E 02	4.90E 02	1.09E 03	4.28E 02	9.65E 02	2.47E 02	5.18E 02
6	7.500000 00	2.49E 02	4.81E 02	4.86E 02	9.55E 02	4.88E 02	1.09E 03	4.25E 02	9.62E 02	2.45E 02	5.15E 02
7	9.000000 00	2.47E 02	4.78E 02	4.83E 02	9.52E 02	4.85E 02	1.08E 03	4.22E 02	9.57E 02	2.44E 02	5.12E 02
8	1.050000 01	2.45E 02	4.75E 02	4.81E 02	9.47E 02	4.81E 02	1.08E 03	4.18E 02	9.51E 02	2.41E 02	5.09E 02
9	1.200000 01	2.42E 02	4.70E 02	4.77E 02	9.40E 02	4.76E 02	1.07E 03	4.13E 02	9.43E 02	2.38E 02	5.04E 02
10	1.350000 01	2.39E 02	4.65E 02	4.72E 02	9.32E 02	4.69E 02	1.06E 03	4.07E 02	9.32E 02	2.35E 02	4.98E 02
11	1.500000 01	2.35E 02	4.58E 02	4.67E 02	9.21E 02	4.61E 02	1.04E 03	3.99E 02	9.19E 02	2.30E 02	4.90E 02
12	1.650000 01	2.30E 02	4.49E 02	4.59E 02	9.08E 02	4.52E 02	1.03E 03	3.89E 02	9.03E 02	2.25E 02	4.80E 02
13	1.800000 01	2.24E 02	4.39E 02	4.50E 02	8.90E 02	4.39E 02	1.00E 03	3.77E 02	8.81E 02	2.18E 02	4.69E 02
14	1.950000 01	2.16E 02	4.25E 02	4.39E 02	8.67E 02	4.24E 02	9.74E 02	3.62E 02	8.54E 02	2.10E 02	4.54E 02
15	2.100000 01	2.07E 02	4.09E 02	4.25E 02	8.37E 02	4.04E 02	9.35E 02	3.42E 02	8.18E 02	2.00E 02	4.36E 02
16	2.250000 01	1.95E 02	3.89E 02	4.08E 02	7.98E 02	3.80E 02	8.85E 02	3.19E 02	7.72E 02	1.88E 02	4.13E 02
17	2.400000 01	1.82E 02	3.64E 02	3.86E 02	7.46E 02	3.49E 02	8.18E 02	2.93E 02	7.13E 02	1.77E 02	3.86E 02
*** (REGION 2, MATERIAL 3) ***											
17	2.400000 01	1.82E 02	3.64E 02	3.86E 02	7.46E 02	2.37E 03	3.09E 03	4.46E 02	1.23E 03	2.08E 02	4.27E 02
18	2.650000 01	1.64E 02	3.14E 02	3.41E 02	6.36E 02	2.27E 03	2.99E 03	3.05E 02	9.96E 02	1.76E 02	3.68E 02
19	2.900000 01	1.48E 02	2.78E 02	3.03E 02	5.53E 02	2.19E 03	2.89E 03	2.37E 02	8.11E 02	1.59E 02	3.24E 02
20	3.150000 01	1.34E 02	2.50E 02	2.79E 02	4.89E 02	2.14E 03	2.79E 03	1.85E 02	6.68E 02	1.43E 02	2.90E 02
21	3.400000 01	1.22E 02	2.28E 02	2.59E 02	4.39E 02	2.08E 03	2.68E 03	1.61E 02	5.57E 02	1.30E 02	2.63E 02
22	3.650000 01	1.11E 02	2.10E 02	2.42E 02	3.99E 02	2.01E 03	2.56E 03	1.41E 02	4.71E 02	1.18E 02	2.41E 02
23	3.900000 01	1.04E 02	1.95E 02	2.26E 02	3.67E 02	1.95E 03	2.44E 03	1.25E 02	4.04E 02	1.08E 02	2.23E 02
24	4.150000 01	9.80E 01	1.83E 02	2.11E 02	3.40E 02	1.87E 03	2.32E 03	1.11E 02	3.51E 02	1.02E 02	2.07E 02
25	4.400000 01	9.25E 01	1.72E 02	1.98E 02	3.17E 02	1.79E 03	2.19E 03	1.04E 02	3.10E 02	9.63E 01	1.94E 02
26	4.650000 01	8.74E 01	1.63E 02	1.85E 02	2.98E 02	1.69E 03	2.07E 03	9.71E 01	2.77E 02	9.11E 01	1.82E 02
27	4.900000 01	8.26E 01	1.55E 02	1.77E 02	2.82E 02	1.59E 03	1.95E 03	9.13E 01	2.50E 02	8.63E 01	1.73E 02
28	5.150000 01	7.82E 01	1.48E 02	1.70E 02	2.68E 02	1.48E 03	1.83E 03	8.61E 01	2.29E 02	8.19E 01	1.64E 02
29	5.400000 01	7.40E 01	1.42E 02	1.63E 02	2.56E 02	1.35E 03	1.72E 03	8.14E 01	2.11E 02	7.78E 01	1.56E 02
30	5.650000 01	7.00E 01	1.36E 02	1.57E 02	2.45E 02	1.14E 03	1.62E 03	7.71E 01	1.96E 02	7.40E 01	1.49E 02
31	5.900000 01	6.66E 01	1.31E 02	1.52E 02	2.36E 02	9.28E 02	1.52E 03	7.31E 01	1.84E 02	7.03E 01	1.43E 02
32	6.150000 01	6.44E 01	1.26E 02	1.46E 02	2.28E 02	6.94E 02	1.44E 03	6.92E 01	1.74E 02	6.70E 01	1.38E 02
33	6.400000 01	6.21E 01	1.22E 02	1.42E 02	2.21E 02	6.79E 02	1.37E 03	6.63E 01	1.65E 02	6.50E 01	1.33E 02
34	6.650000 01	5.96E 01	1.17E 02	1.37E 02	2.14E 02	3.91E 02	1.30E 03	6.41E 01	1.57E 02	6.29E 01	1.28E 02
35	6.900000 01	5.64E 01	1.13E 02	1.32E 02	2.08E 02	3.09E 02	1.23E 03	6.15E 01	1.49E 02	6.05E 01	1.23E 02
36	7.150000 01	5.18E 01	1.07E 02	1.27E 02	2.02E 02	2.63E 02	1.15E 03	5.81E 01	1.41E 02	5.71E 01	1.18E 02
37	7.400000 01	4.38E 01	9.87E 01	1.20E 02	1.95E 02	2.06E 02	1.04E 03	5.26E 01	1.30E 02	5.16E 01	1.10E 02
*** (REGION 3, MATERIAL 2) ***											
37	7.400000 01	4.38E 01	9.87E 01	1.20E 02	1.95E 02	0.0	0.0	4.16E 01	1.14E 02	4.16E 01	1.14E 02
38	7.600000 01	3.83E 01	9.19E 01	1.16E 02	1.88E 02	0.0	0.0	3.87E 01	1.04E 02	3.87E 01	1.04E 02
39	7.800000 01	3.59E 01	8.90E 01	1.16E 02	1.86E 02	0.0	0.0	3.74E 01	9.99E 01	3.74E 01	9.99E 01
40	8.000000 01	3.69E 01	9.19E 01	1.22E 02	1.91E 02	0.0	0.0	3.83E 01	1.03E 02	3.83E 01	1.03E 02
41	8.200000 01	5.48E 01	1.14E 02	1.48E 02	2.18E 02	0.0	0.0	5.44E 01	1.30E 02	5.44E 01	1.30E 02

REGION VALUES OF MEDIAN AND AVERAGE ENERGIES

REGION NUMBER	MATERIAL NUMBER	MEDIAN NEUTRON ENERGY	AVERAGE NEUTRON ENERGY	MEDIAN FLUX ENERGY	AVERAGE FLUX ENERGY	MEDIAN FISSION ENERGY	AVERAGE FISSION ENERGY	MEDIAN ABSORB ENERGY	AVERAGE ABSORB ENERGY	MEDIAN CAPTURE ENERGY	AVERAGE CAPTURE ENERGY
1	1	2.27E 02	4.43E 02	4.54E 02	8.95E 02	4.44E 02	1.01E 03	3.82E 02	8.88E 02	2.21E 02	4.73E 02
2	3	1.06E 02	2.12E 02	2.40E 02	4.30E 02	2.12E 03	2.74E 03	1.42E 02	5.48E 02	1.11E 02	2.45E 02
3	2	3.88E 01	9.34E 01	1.19E 02	1.91E 02	0.0	0.0	3.92E 01	1.06E 02	3.92E 01	1.06E 02

MACROSCOPIC MATERIAL MIXTURE DATA BY REGION

REGION NO. 1, MACROSCOPIC MATERIAL NO. 1

LIBRARY MATERIAL 1 3 4 5
CONCENTRATION 6.32109E-03 7.67891E-03 5.00000E-03 2.00000E-01

REGION NO. 2, MACROSCOPIC MATERIAL NO. 3

LIBRARY MATERIAL 1 3 5
CONCENTRATION 5.22544E-05 2.40097E-02 2.20000E-01

REGION NO. 3, MACROSCOPIC MATERIAL NO. 2

LIBRARY MATERIAL 4 5
CONCENTRATION 1.50000E-03 9.50000E-01

REACTION RATES OF THE ISOTOPES AND/OR MATERIALS COMPRISING THE MACROSCOPIC MATERIAL IN EACH REGION

REGION NUMBER 1 (MATERIAL NO. 1)

LIB. MTL. CAPTURES PER NEUTRON FISSION ABSORPTION CAPTURE NEUT. PROD. FISSION ABSORPTION CAPTURE	NUMBER	FISSION	PRODUCTION	PRODUCTION	RATE	RATE	FRACTION	FRACTION	FRACTION	FRACTION
1	1.7429E-01	8.4370E-01	3.3584E-01	3.9437E-01	5.8533E-02	9.1816E-01	9.2557E-01	8.3965E-01	5.4785E-01	
3	1.5813E 00	7.5205E-02	2.7006E-02	6.9711E-02	4.2705E-02	8.1842E-02	7.4429E-02	1.4842E-01	3.9970E-01	
4	0.0	0.0	0.0	1.7115E-04	1.7115E-04	0.0	0.0	3.6440E-04	1.6019E-03	
5	0.0	0.0	0.0	5.4328E-03	5.4328E-03	0.0	0.0	1.1567E-02	5.0849E-02	
(TOTAL)	2.94450-01	9.18910-01	3.62850-01	4.69690-01	1.06840-01	1.00000 00	1.00000 00	1.00000 00	1.00000 00	

REGION NUMBER 2 (MATERIAL NO. 3)

LIB. MTL. CAPTURES PER NEUTRON FISSION ABSORPTION CAPTURE NEUT. PROD. FISSION ABSORPTION CAPTURE	NUMBER	FISSION	PRODUCTION	PRODUCTION	RATE	RATE	FRACTION	FRACTION	FRACTION	FRACTION
1	2.3555E-01	9.6650E-03	3.9251E-03	4.8496E-03	9.2455E-04	1.0611E-01	1.1761E-01	1.7657E-02	3.8317E-03	
3	7.8764E 00	8.1422E-02	2.9449E-02	2.6140E-01	2.3195E-01	8.9389E-01	8.8239E-01	9.5171E-01	9.6130E-01	
5	0.0	0.0	0.0	8.4140E-03	8.4140E-03	0.0	0.0	3.0634E-02	3.4871E-02	
(TOTAL)	7.22990 00	9.10870-02	3.33740-02	2.74660-01	2.41290-01	1.00000 00	1.00000 00	1.00000 00	1.00000 00	

REGION NUMBER 3 (MATERIAL NO. 2)

LIB. MTL. CAPTURES PER NEUTRON FISSION ABSORPTION CAPTURE NEUT. PROD. FISSION ABSORPTION CAPTURE	NUMBER	FISSION	PRODUCTION	PRODUCTION	RATE	RATE	FRACTION	FRACTION	FRACTION	FRACTION
4	0.0	0.0	0.0	2.1049E-06	2.1049E-06	0.0	0.0	3.0958E-03	3.0958E-03	
5	0.0	0.0	0.0	6.7781E-04	6.7781E-04	0.0	0.0	9.9690E-01	9.9690E-01	
(TOTAL)	0.0	0.0	0.0	6.79910 04	6.79910 04	1.00000 00	1.00000 00	1.00000 00	1.00000 00	

SPATIAL DISTRIBUTIONS OF REACTION RATIOS

MATERIALS IN NUMERATOR OF RATIO 1 ARE	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
THE DENOMINATOR MATERIALS ARE	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
MATERIALS IN NUMERATOR OF RATIO 2 ARE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
THE DENOMINATOR MATERIALS ARE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
MATERIALS IN NUMERATOR OF RATIO 3 ARE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
THE DENOMINATOR MATERIALS ARE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
MATERIALS IN NUMERATOR OF RATIO 4 ARE	3	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
THE DENOMINATOR MATERIALS ARE	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
MATERIALS IN NUMERATOR OF RATIO 5 ARE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
THE DENOMINATOR MATERIALS ARE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

	RATIO 1	RATIO 2	RATIO 3	RATIO 4	RATIO 5
NUMERATOR REACTION TYPE	3	0	0	3	0
DENOMINATOR REACTION TYPE	2	0	0	1	0

DISTRIBUTIONS FOR REACTION RATIO NUMBER 1

MESH POINT	MESH-POINT POSITION	NUMERATOR REACTION RATE	DENOMINATOR REACTION RATE	REACTION RATE RATIO
*** (REGION 1, MATERIAL 1) ***				
1	0.0	3.8741709E-05	2.3361962E-04	1.6583240E-01
2	1.50000000 00	3.8680970E-05	2.3319840E-04	1.6587150E-01
3	3.00000000 00	3.8499391E-05	2.3193771E-04	1.6599017E-01
4	4.50000000 00	3.8199156E-05	2.2984737E-04	1.6619354E-01
5	6.00000000 00	3.7783888E-05	2.2694365E-04	1.6649014E-01
6	7.50000000 00	3.7258695E-05	2.2324943E-04	1.6689265E-01
7	9.00000000 00	3.6630343E-05	2.1879433E-04	1.6741920E-01
8	1.05000000 01	3.5906982E-05	2.1361378E-04	1.6809297E-01
9	1.20000000 01	3.5098681E-05	2.0775103E-04	1.6894585E-01
10	1.35000000 01	3.4217213E-05	2.0125568E-04	1.7001861E-01
11	1.50000000 01	3.3276374E-05	1.9418432E-04	1.7136484E-01
12	1.65000000 01	3.2292024E-05	1.8660093E-04	1.7305392E-01
13	1.80000000 01	3.1282470E-05	1.7857709E-04	1.7517626E-01
14	1.95000000 01	3.0268828E-05	1.7019286E-04	1.7785013E-01
15	2.10000000 01	2.9275208E-05	1.6153710E-04	1.8122900E-01
16	2.25000000 01	2.8329421E-05	1.5270880E-04	1.8551266E-01
17	2.40000000 01	2.7463699E-05	1.4381835E-04	1.9096100E-01
*** (REGION 2, MATERIAL 3) ***				
17	2.40000000 01	2.2703369E-07	1.1888933E-06	1.9096220E-01
18	2.65000000 01	2.0905816E-07	1.0300437E-06	2.0296043E-01
19	2.90000000 01	1.8826927E-07	8.8418784E-07	2.1292901E-01
20	3.15000000 01	1.6678962E-07	7.5333412E-07	2.2140187E-01
21	3.40000000 01	1.4589961E-07	6.3787002E-07	2.2872937E-01
22	3.65000000 01	1.2633461E-07	5.3724153E-07	2.3515421E-01
23	3.90000000 01	1.0847623E-07	4.5038945E-07	2.4084979E-01
24	4.15000000 01	9.2478615E-08	3.7601438E-07	2.4594432E-01
25	4.40000000 01	7.8351945E-08	3.1273879E-07	2.5053477E-01
26	4.65000000 01	6.6017947E-08	2.5920451E-07	2.5469440E-01
27	4.90000000 01	5.5349187E-08	2.1412899E-07	2.5848520E-01
28	5.15000000 01	4.6191108E-08	1.7633437E-07	2.6195100E-01
29	5.40000000 01	3.8381497E-08	1.4476137E-07	2.6513630E-01
30	5.65000000 01	3.1759416E-08	1.1847158E-07	2.6807624E-01
31	5.90000000 01	2.6172369E-08	9.6644726E-08	2.7081007E-01
32	6.15000000 01	2.1480311E-08	7.8571418E-08	2.7338576E-01
33	6.40000000 01	1.7558058E-08	6.3645814E-08	2.7587134E-01
34	6.65000000 01	1.4296742E-08	5.1356555E-08	2.7838200E-01
35	6.90000000 01	1.1604847E-08	4.1280394E-08	2.8112255E-01
36	7.15000000 01	9.4095043E-09	3.3078344E-08	2.8464114E-01
37	7.40000000 01	7.6584499E-09	2.6493847E-08	2.8906518E-01
*** (REGION 3, MATERIAL 2) ***				
37	7.40000000 01	0.0	0.0	0.0
38	7.60000000 01	0.0	0.0	0.0
39	7.80000000 01	0.0	0.0	0.0
40	8.00000000 01	0.0	0.0	0.0
41	8.20000000 01	0.0	0.0	0.0

DISTRIBUTIONS FOR REACTION RATIO NUMBER 4

MESH POINT	MESH-POINT POSITION	NUMERATOR REACTION RATE	DENOMINATOR REACTION RATE	REACTION RATE RATIO
[REGION 1, MATERIAL 1]**				
1	0.0	2.8786890E-05	2.7236016E-04	1.0569417E-01
2	1.50000000 00	2.8739028E-05	2.7187821E-04	1.0570550E-01
3	3.00000000 00	2.8595896E-05	2.7043629E-04	1.0573983E-01
4	4.50000000 00	2.8358991E-05	2.6804511E-04	1.0579932E-01
5	6.00000000 00	2.8030758E-05	2.6472611E-04	1.058586E-01
6	7.50000000 00	2.7614733E-05	2.6050699E-04	1.0600376E-01
7	9.00000000 00	2.7115486E-05	2.5542360E-04	1.0615885E-01
8	1.05000000 01	2.6538692E-05	2.4952018E-04	1.0635889E-01
9	1.20000000 01	2.5891059E-05	2.4284986E-04	1.0661340E-01
10	1.35000000 01	2.5180721E-05	2.3547305E-04	1.0693669E-01
11	1.50000000 01	2.4416935E-05	2.2746078E-04	1.0734564E-01
12	1.65000000 01	2.3610497E-05	2.1889299E-04	1.0786319E-01
13	1.80000000 01	2.2773747E-05	2.0985969E-04	1.0851890E-01
14	1.95000000 01	2.1920889E-05	2.0046177E-04	1.0935193E-01
15	2.10000000 01	2.1068176E-05	1.9081238E-04	1.1041301E-01
16	2.25000000 01	2.0234307E-05	1.8103835E-04	1.1176807E-01
17	2.40000000 01	1.9440791E-05	1.7128212E-04	1.1350155E-01
[REGION 2, MATERIAL 3]**				
17	2.40000000 01	6.0705911E-05	1.4159286E-06	4.2930054E-01
18	2.65000000 01	5.4955555E-05	1.2391029E-06	4.4351074E-01
19	2.90000000 01	4.8762493E-05	1.0724525E-06	4.5468201E-01
20	3.15000000 01	4.2673695E-05	9.2012408E-07	4.6378189E-01
21	3.40000000 01	3.6948259E-05	7.8377002E-07	4.7141708E-01
22	3.65000000 01	3.1717413E-05	6.6357643E-07	4.7797668E-01
23	3.90000000 01	2.7033588E-05	5.5886619E-07	4.8372192E-01
24	4.15000000 01	2.2701467E-05	4.6049340E-07	4.8863224E-01
25	4.40000000 01	1.9297775E-05	3.9109085E-07	4.9343445E-01
26	4.65000000 01	1.6183738E-05	3.2522291E-07	4.9761978E-01
27	4.90000000 01	1.3513097E-05	2.6947822E-07	5.0145401E-01
28	5.15000000 01	1.1237168E-05	2.2252544E-07	5.0498337E-01
29	5.40000000 01	9.3081517E-06	1.8314279E-07	5.0824554E-01
30	5.65000000 01	7.6808483E-06	1.5023102E-07	5.1126907E-01
31	5.90000000 01	6.3137559E-06	1.2281703E-07	5.1407822E-01
32	6.15000000 01	5.1696361E-06	1.0005175E-07	5.1669617E-01
33	6.40000000 01	4.2157581E-06	8.1203837E-08	5.1915741E-01
34	6.65000000 01	3.4239529E-06	6.5653069E-08	5.2152206E-01
35	6.90000000 01	2.7707056E-06	5.2885277E-08	5.2390854E-01
36	7.15000000 01	2.2372869E-06	4.2487869E-08	5.2657074E-01
37	7.40000000 01	1.8101591E-06	3.4152322E-08	5.3002518E-01
[REGION 3, MATERIAL 2]**				
37	7.40000000 01	0.0	0.0	0.0
38	7.60000000 01	0.0	0.0	0.0
39	7.80000000 01	0.0	0.0	0.0
40	8.00000000 01	0.0	0.0	0.0
41	8.20000000 01	0.0	0.0	0.0

REGION PEAK TO REGION AVERAGE FLUX RATIOS

REGION	GROUP NO. 1 RATIO	POINT	GROUP NO. 2 RATIO	POINT	GROUP NO. 3 RATIO	POINT	GROUP NO. 4 RATIO	POINT	GROUP NO. 5 RATIO	POINT
1	1.4130220E 00	1	1.4179688E 00	1	1.4164276E 00	1	1.3649235E 00	1	1.3087091E 00	1
2	1.5084668E 01	17	1.4918440E 01	17	1.3321438E 01	17	8.8537588E 00	17	6.4636145E 00	17
3	2.2381353E 00	37	2.0582170E 00	37	1.8594007E 00	37	1.7528896E 00	37	1.7167807E 00	37
PEAK TO AVERAGE FLUX RATIOS IN FUEL										
	1.0216254E 01	1	1.0243336E 01	1	9.8448009E 00	1	7.9466524E 00	1	6.3492107E 00	1
PEAK TO AVERAGE FLUX RATIOS FOR ENTIRE REACTOR										
	1.2542609E 01	1	1.2575520E 01	1	1.2085189E 01	1	9.7475147E 00	1	7.7731056E 00	1
AVERAGE FLUXES IN REACTOR										
	7.2811599E-05		1.3784408E-04		2.1927136E-04		3.7178188E-04		5.8023538E-04	
AVERAGE FLUXES IN FUEL										
	8.9391615E-05		1.6922817E-04		2.6917108E-04		4.5603467E-04		7.1036094E-04	

REGION PEAK TO REGION AVERAGE FLUX RATIOS

REGION	GROUP NO. 6 RATIO	POINT	GROUP NO. 7 RATIO	POINT	GROUP NO. 8 RATIO	POINT	GROUP NO. 9 RATIO	POINT	GROUP NO. 10 RATIO	POINT
1	1.2318225E 00	1	1.2187310E 00	1	1.1843901E 00	1	1.1449852E 00	1	1.0910282E 00	1
2	5.1966848E 00	17	4.5569954E 00	17	3.9773989E 00	17	3.4423342E 00	17	2.8806705E 00	17
3	1.6792564E 00	37	1.7309647E 00	37	1.6586313E 00	37	1.7447882E 00	37	1.7498436E 00	37
PEAK TO AVERAGE FLUX RATIOS IN FUEL										
	5.1859846E 00	1	4.5676813E 00	1	3.9709463E 00	1	3.3636007E 00	1	2.6895332E 00	1
PEAK TO AVERAGE FLUX RATIOS FOR ENTIRE REACTOR										
	6.7323188E 00	1	5.9720920E 00	1	4.8275766E 00	1	4.0823116E 00	1	3.2555256E 00	1
AVERAGE FLUXES IN REACTOR										
	7.553893E-04		6.4885011E-04		5.2931765E-04		3.5592774E-04		2.3602991E-04	
AVERAGE FLUXES IN FUEL										
	9.2257583E-04		7.9152337E-04		6.4350432E-04		4.3198001E-04		2.8570066E-04	

REGION PEAK TO REGION AVERAGE FLUX RATIOS

REGION	GROUP NO. 11 RATIO	POINT	GROUP NO. 12 RATIO	POINT	GROUP NO. 13 RATIO	POINT	GROUP NO. 14 RATIO	POINT	GROUP NO. 15 RATIO	POINT
1	1.0551119E 00	17	1.0696945E 00	17	1.1421347E 00	17	1.2230034E 00	17	1.3302221E 00	17
2	2.5159235E 00	18	2.3078194E 00	19	2.1728497E 00	19	2.0653334E 00	19	1.9019747E 00	20
3	1.8081455E 00	37	1.6138802E 00	37	1.5463276E 00	37	1.4112148E 00	37	1.3704500E 00	38
PEAK TO AVERAGE FLUX RATIOS IN FUEL										
	2.2177601E 00	18	2.0784693E 00	19	2.0262918E 00	19	1.9810581E 00	19	1.8780546E 00	20
PEAK TO AVERAGE FLUX RATIOS FOR ENTIRE REACTOR										
	2.6739340E 00	18	2.4916506E 00	19	2.4018202E 00	19	2.3037596E 00	19	2.1217146E 00	20
AVERAGE FLUXES IN REACTOR										
	1.4815584E-04		7.7769830E-05		3.2643031E-05		1.5274039E-05		4.9508471E-06	
AVERAGE FLUXES IN FUEL										
	1.7863020E-04		9.3229799E-05		3.8692699E-05		1.7762082E-05		5.5931732E-06	

REGION PEAK TO REGION AVERAGE FLUX RATIOS

REGION	GROUP NO. 16		GROUP NO. 17		GROUP NO. 18		GROUP NO. 19		GROUP NO. 20	
	RATIO	POINT	RATIO	POINT	RATIO	POINT	RATIO	POINT	RATIO	POINT
1	1.2076530E 00	17	1.5047855E 00	17	1.4917030E 00	17	1.5087643E 00	17	1.3098688E 00	17
2	1.8138768E 00	20	1.6119452E 00	21	1.5920508E 00	21	1.5359230E 00	21	1.6270760E 00	37
3	1.3303366E 00	38	1.3104095E 00	38	1.3154984E 00	38	1.3119583E 00	38	1.3036108E 00	38
PEAK TO AVERAGE FLUX RATIOS IN FUEL	1.7646961E 00	20	1.6612215E 00	21	1.6420355E 00	21	1.5935650E 00	21	1.6953144E 00	37
PEAK TO AVERAGE FLUX RATIOS FOR ENTIRE REACTOR	1.9315767E 00	20	1.7384443E 00	21	1.7128401E 00	21	1.5915823E 00	21	1.9047890E 00	38
AVERAGE FLUXES IN REACTOR	8.7625898E-07		1.0864514E-06		1.2637906E-07		1.2712138E-07		4.8494080E-08	
AVERAGE FLUXES IN FUEL	9.5912310E-07		1.1369557E-06		1.3182853E-07		1.2696324E-07		4.3397314E-08	

REGION PEAK TO REGION AVERAGE FLUX RATIOS

REGION	GROUP NO. 21		GROUP NO. 22	
	RATIO	POINT	RATIO	POINT
1	1.1930971E 00	1	1.2045622E 00	1
2	3.1443110E 00	37	3.2120333E 00	37
3	1.2768793E 00	39	1.2786074E 00	38
PEAK TO AVERAGE FLUX RATIOS IN FUEL	3.2055702E 00	37	3.3471556E 00	37
PEAK TO AVERAGE FLUX RATIOS FOR ENTIRE REACTOR	3.1650763E 00	39	3.0889997E 00	38
AVERAGE FLUXES IN REACTOR	1.1248190E-08		2.7990414E-09	
AVERAGE FLUXES IN FUEL	7.4573876E-09		1.8958155E-09	

REACTOR VOLUME= 2.1124069D 04
 FUEL VOLUME= 1.7203362D 04
 AVERAGE FISSION NEUTRON SOURCE DENSITY= 5.8128174D-05
 TOTAL FISSION NEUTRON SOURCE= 1.0000000D 00
 AVERAGE FISSION POWER DENSITY= 7.1603417D-16
 TOTAL FISSION POWER= 1.2318195D-11

PEAK TO AVERAGE FISSION NEUTRON SOURCE DENSITIES

FUEL REGION NUMBER	REGION AVERAGE VALUE	REGION MAX TO REGION AVG	REGION MAX TO FUEL AVG	POINT NUMBER OF MAX
1	5.0278264E-04	1.2750921E 00	1.1028978E 01	1
2	5.8585692E-06	1.2952018E 01	1.3053961E 00	17

FUEL MAXIMUM TO FUEL AVERAGE 1.1028978D 01 1

PEAK TO AVERAGE POWER DENSITIES

FUEL REGION NUMBER	REGION AVERAGE VALUE	REGION MAX TO REGION AVG	REGION MAX TO FUEL AVG	POINT NUMBER OF MAX
1	6.2339152E-15	1.2703495E 00	1.1059886E 01	1
2	6.7401522E-17	1.2768020E 01	1.2018747E 00	17

FUEL MAXIMUM TO FUEL AVERAGE 1.1059886D 01 1

DETAILED REGION NEUTRON BALANCE DATA

REGION NUMBER 1 (MATERIAL NO. 1)

GROUP NUMBER	FISSION SOURCE TO GROUP	NEUTRON SOURCE	SCATTERING IN SOURCE	FIXED VOLUME SOURCE	ABSORPTION LOSS	SCATTERING OUT LOSS	PERPENDICULAR LEAKAGE LOSS	PARALLEL LEAKAGE LOSS
1	1.18594290-01	0.0	0.0	0.0	1.6190540E-02	6.7644060E-02	7.8355968E-03	2.6924089E-02
2	1.91870730-01	1.3013564E-02	0.0	0.0	2.9491667E-02	1.1702615E-01	1.3053503E-02	4.5312975E-02
3	2.11286130-01	5.9905235E-02	0.0	0.0	4.2086389E-02	1.4738065E-01	1.9784603E-02	6.1939716E-02
4	1.63229770-01	1.0503900E-01	0.0	0.0	4.7349844E-02	1.3365549E-01	2.5393728E-02	6.1869703E-02
5	1.04510320-01	1.4398324E-01	0.0	0.0	5.8526509E-02	1.1786193E-01	2.4541669E-02	4.7563463E-02
6	5.89932440-02	1.6461670E-01	0.0	0.0	6.9762170E-02	1.0027254E-01	2.1297734E-02	3.2277506E-02
7	3.10219360-02	1.4383441E-01	0.0	0.0	6.1718218E-02	8.1434190E-02	1.3705511E-02	1.7998427E-02
8	1.55805690-02	1.0731262E-01	0.0	0.0	5.3255349E-02	5.3670026E-02	7.8533106E-03	8.1145018E-03
9	7.63870900-03	6.6966474E-02	0.0	0.0	3.7262693E-02	3.2094281E-02	3.5944243E-03	1.6537844E-03
10	3.70268800-03	3.6736775E-02	0.0	0.0	2.7408424E-02	1.4650811E-02	1.7786750E-03	6.9846539E-04
11	1.77704930-03	1.9877471E-02	0.0	0.0	1.4885422E-02	7.5316690E-03	7.2355289E-04	1.4861233E-03
12	8.49114650-04	1.0178402E-02	0.0	0.0	8.4112845E-03	2.9565776E-03	3.5641366E-04	6.9675897E-04
13	3.99881570-04	4.0357150E-03	0.0	0.0	3.4769611E-03	1.2658695E-03	9.1303824E-05	3.9853761E-04
14	2.42711070-04	1.8053222E-03	0.0	0.0	1.7457281E-03	4.7142548E-04	2.3158806E-05	1.9227904E-04
15	6.20011130-05	5.5579725E-04	0.0	0.0	5.5061886E-04	1.2466903E-04	4.3021391E-06	6.1818660E-05
16	1.67942510-05	1.5950530E-04	0.0	0.0	1.2707816E-04	5.1692972E-05	9.2607968E-07	7.3976626E-06
17	2.03607320-05	7.0419555E-05	0.0	0.0	1.0145792E-04	9.4551378E-06	1.9954479E-07	2.0332300E-05
18	5.56706210-06	1.2368220E-05	0.0	0.0	1.5320766E-05	4.9414721E-06	1.5131871E-09	2.3584683E-06
19	6.51390610-06	7.1931945E-06	0.0	0.0	1.4566297E-05	1.4965799E-06	1.2170582E-09	2.3569928E-06
20	4.00882360-06	1.4965799E-06	0.0	0.0	5.7853968E-06	2.8909153E-07	3.7718495E-10	5.6946192E-07
21	1.42304040-06	2.8909153E-07	0.0	0.0	1.6979729E-06	5.2478196E-08	7.8426612E-11	3.8397609E-08
22	3.57284050-07	5.2478196E-08	0.0	0.0	2.549561E-07	0.0	1.4292800E-11	1.5747659E-08

GROUP NUMBER	VOLUME INTEGRAL OF FLUX	AVERAGE FLUX	FLUX SPECTRUM	GROUP NEUTRON PRODUCTION	PARALLEL BUCKLING
1	1.1695509E-00	6.4630760E-04	4.78217970-06	4.6725601E-02	6.8378821E-03
2	2.2121763E-00	1.2224957E-03	3.94260270-05	7.6336741E-02	6.8732239E-03
3	3.3894296E-00	1.8708983E-03	9.94172800-05	1.0056144E-01	6.1361715E-03
4	4.8044767E-00	2.6550561E-03	2.34704650-04	9.8014772E-02	4.4099130E-03
5	6.2363138E-00	3.4463198E-03	4.98949640-04	1.3076986E-01	3.1978118E-03
6	6.9161234E-00	3.8219972E-03	9.11880520-04	1.3168979E-01	2.2733046E-03
7	5.3680534E-00	2.9665006E-03	1.16573850-03	1.1523871E-01	1.8516474E-03
8	3.9041195E-00	2.1574995E-03	1.40127020-03	9.0394914E-02	1.3535682E-03
9	2.2963629E-00	1.2690190E-03	1.36108520-03	5.9685577E-02	5.6591933E-04
10	1.2744551E-00	7.0429104E-04	1.24282490-03	3.7941981E-02	4.4766464E-04
11	6.4912128E-01	3.5871821E-04	1.04191250-03	2.2338353E-02	2.1976994E-03
12	3.1133020E-01	1.7204770E-04	8.20968420-04	1.2518823E-02	1.9431859E-03
13	1.1019158E-01	6.0894206E-05	4.84286510-04	5.1968731E-03	3.9284639E-03
14	4.3298524E-02	2.3927685E-05	2.32626560-04	2.5638714E-03	6.7998543E-03
15	1.1204008E-02	6.1913734E-06	1.70316120-04	8.4163132E-04	9.8285824E-03
16	2.1343254E-03	1.1794737E-06	9.67647250-05	1.9240522E-04	4.5053102E-03
17	1.5223604E-03	8.4128880E-07	4.85695110-05	1.5584835E-04	1.5589684E-02
18	1.7483720E-04	9.6618749E-08	1.68025160-05	2.1305867E-05	1.6926620E-02
19	1.5639923E-04	8.6429509E-08	1.06867180-05	2.1494037E-05	1.9366305E-02
20	5.1375828E-05	2.8391376E-08	4.31762570-06	8.6633536E-06	1.5097678E-02
21	1.1258025E-05	6.2214234E-09	1.67107020-06	2.5072204E-06	4.8959889E-03
22	2.2028908E-06	1.2173642E-09	5.61972150-07	6.3679704E-07	1.1017893E-02

REGION NUMBER 2 (MATERIAL NO. 3)

GROUP NUMBER	FISSION SOURCE TO GROUP	NEUTRON SOURCE	SCATTERING IN SOURCE	FIXED VOLUME SOURCE	ABSORPTION LOSS	SCATTERING OUT LOSS	PERPENDICULAR LEAKAGE LOSS	PARALLEL LEAKAGE LOSS
1	1.17556980-02	0.0	0.0	0.0	5.9981197E-03	3.0841976E-02	1.8066068E-03	2.6891004E-02
2	1.90192500-02	4.9538352E-03	0.0	0.0	1.0301527E-02	5.5830576E-02	3.1019412E-03	4.5260958E-02
3	2.09438070-02	2.3004211E-02	0.0	0.0	1.4294904E-02	8.6093843E-02	5.4141432E-03	6.1854869E-02
4	1.61802040-02	5.1957536E-02	0.0	0.0	1.1721663E-02	1.0536385E-01	1.2431953E-02	6.1379720E-02
5	1.03596200-02	9.1007590E-02	0.0	0.0	1.9623656E-02	1.0919404E-01	1.8459424E-02	4.5909908E-02
6	5.84772470-03	1.5077162E-01	0.0	0.0	2.6451878E-02	1.3933825E-01	2.0233821E-02	2.9404610E-02
7	3.07505960-03	1.6919857E-01	0.0	0.0	2.9209863E-02	1.4253622E-01	1.5195392E-02	1.4667843E-02
8	1.54429100-03	1.5763223E-01	0.0	0.0	3.4632489E-02	1.1988890E-01	1.0300804E-02	5.6455322E-03
9	7.57189550-04	1.2738774E-01	0.0	0.0	3.3625361E-02	9.0411365E-02	5.8966205E-03	2.1157980E-03
10	6.70282500-04	8.0844479E-02	0.0	0.0	3.0879438E-02	4.4573251E-02	3.6217475E-03	2.2370701E-03
11	1.76150590-04	5.3059889E-02	0.0	0.0	2.5123443E-02	2.3829263E-02	1.9541504E-03	3.3291775E-03
12	8.41687700-05	3.1098370E-02	0.0	0.0	1.6279098E-02	1.2659281E-02	1.0241671E-03	1.2199921E-03
13	3.96383930-05	1.0707858E-02	0.0	0.0	8.8766143E-03	6.0738102E-03	3.3216722E-04	4.6490901E-04
14	2.40588150-05	7.5160302E-03	0.0	0.0	5.2936450E-03	1.9854156E-03	1.1522169E-04	1.4580680E-04
15	6.14588070-06	2.2341539E-03	0.0	0.0	1.5206914E-03	6.6300412E-04	3.3873352E-05	2.2930879E-05
16	1.66473570-06	7.2239549E-04	0.0	0.0	3.0568708E-04	4.2323186E-04	5.4407628E-06	1.0299475E-05
17	2.01826430-06	4.8222276E-04	0.0	0.0	3.5585999E-04	1.3224398E-04	2.1045980E-06	5.9675331E-06
18	5.51836870-07	1.4600382E-04	0.0	0.0	7.3222618E-05	7.3816278E-05	1.7539502E-08	5.0078035E-07
19	6.45693090-07	8.7387321E-05	0.0	0.0	6.6890512E-05	2.3494911E-05	1.5389404E-08	2.3677976E-06
20	3.97375970-07	2.3494911E-05	0.0	0.0	2.1899075E-05	4.3513264E-06	5.0213558E-09	2.3631355E-06
21	1.41059350-07	4.3513264E-06	0.0	0.0	4.9273904E-06	1.0727763E-06	8.2909968E-10	1.5086098E-06
22	3.54158990-08	1.0727763E-06	0.0	0.0	1.3801855E-06	0.0	2.3327940E-10	2.7222654E-07

GROUP NUMBER	VOLUME INTEGRAL OF FLUX	AVERAGE FLUX	FLUX SPECTRUM	GROUP NEUTRON PRODUCTION	PARALLEL BUCKLING
1	3.6830544E-01	2.3925561E-05	1.18591000-06	1.7363410E-02	2.9620759E-02
2	6.9911724E-01	4.5415494E-05	9.81166430-06	2.6488796E-02	2.8890517E-02
3	1.2452240E-00	8.0891245E-05	2.87955570-05	3.2130226E-02	2.2392355E-02
4	3.0408554E-00	1.9753761E-04	1.16977080-04	5.1313192E-03	8.9364201E-03
5	5.9842854E-00	3.8874615E-04	3.77025250-04	1.3260453E-03	4.1036643E-03
6	8.9552851E-00	5.8174599E-04	9.29788170-04	1.4096117E-03	2.1798601E-03
7	8.2488108E-00	5.3585251E-04	1.41060380-03	1.4166839E-03	1.3610474E-03
8	7.1663208E-00	4.6553276E-04	2.02546350-03	1.3716638E-03	7.1796775E-04
9	5.1351480E-00	3.3358531E-04	2.39677440-03	1.1033476E-03	4.4134271E-05
10	3.6405602E-00	2.3649515E-04	2.79565130-03	8.9597120E-04	7.0415158E-04
11	2.4239187E-00	1.5746066E-04	3.06374660-03	6.8956288E-04	1.2753466E-03
12	1.2925358E-00	8.3964667E-05	2.68396700-03	4.2965054E-04	1.1840565E-03
13	5.5545300E-01	3.6082885E-05	1.92234200-03	2.1655671E-04	1.2596771E-03
14	2.6226908E-01	1.7037310E-05	1.10959040-03	1.2838098E-04	1.0363997E-03
15	8.5017383E-02	5.5228311E-06	1.01769930-03	5.2794276E-05	4.6303961E-04
16	1.4365818E-02	9.3322075E-07	5.12880500-04	1.0705755E-05	1.0676626E-03
17	1.8037107E-02	1.1717120E-06	4.53150570-04	1.5264479E-05	4.3382728E-04
18	2.0930567E-03	1.3596747E-07	1.58398420-04	2.1085179E-06	3.1406712E-04
19	2.0277959E-03	1.3172803E-07	1.09109710-04	2.3037665E-06	1.5385894E-03
20	6.9520390E-04	4.5161279E-08	4.80074060-05	9.6910298E-07	4.7061667E-03
21	1.1703416E-04	7.6026758E-09	1.36796360-05	2.1546344E-07	1.8195752E-02
22	3.0411509E-05	1.9755679E-09	6.10926830-06	7.2673572E-08	1.1669543E-02

REGION NUMBER 3 (MATERIAL NO. 2)

GROUP NUMBER	FISSION SOURCE TO GROUP	NEUTRON SOURCE	SCATTERING IN SOURCE	FIXED VOLUME SOURCE	ABSORPTION LOSS	SCATTERING OUT LOSS	PERPENDICULAR LEAKAGE LOSS	PARALLEL LEAKAGE LOSS
1	0.0	0.0	0.0	0.0	9.2911592E-07	2.5927642E-05	9.9090084E-07	-2.1847651E-05
2	0.0	7.0411488E-06	0.0	0.0	7.7337830E-07	4.3786582E-05	2.0110629E-06	-3.9529870E-05
3	0.0	3.5380304E-05	0.0	0.0	3.6655933E-07	8.2785700E-05	4.6122113E-06	-5.2384159E-05
4	0.0	6.0588005E-05	0.0	0.0	1.9537438E-06	2.8739707E-04	2.9712508E-05	-2.5847531E-04
5	0.0	1.6315250E-04	0.0	0.0	1.3625603E-05	7.1126013E-04	1.0312178E-04	-6.6485489E-04
6	0.0	7.8518083E-04	0.0	0.0	3.6738202E-05	1.1462756E-03	1.9079245E-04	-5.8862520E-04
7	0.0	1.2004692E-03	0.0	0.0	4.3607113E-05	2.0702551E-03	1.8025105E-04	-1.0936444E-03
8	0.0	2.0971247E-03	0.0	0.0	6.7191024E-05	1.7207318E-03	1.6763849E-04	-1.4156335E-04
9	0.0	1.7315191E-03	0.0	0.0	7.5634438E-05	1.6898052E-03	9.7371565E-05	-1.3129206E-04
10	0.0	1.6942264E-03	0.0	0.0	8.6354776E-05	1.6643584E-03	7.4305106E-05	-1.3079190E-04
11	0.0	1.6660690E-03	0.0	0.0	1.1762112E-04	1.5000531E-03	3.6346959E-05	-1.2047851E-05
12	0.0	1.5006734E-03	0.0	0.0	5.8223755E-05	1.0811309E-03	3.8098296E-05	-3.2322016E-04
13	0.0	1.0821242E-03	0.0	0.0	5.1508687E-05	7.0036808E-04	1.1488019E-05	-3.1875935E-04
14	0.0	7.0058531E-04	0.0	0.0	7.6472104E-05	3.8545020E-04	4.1206404E-06	-2.3454236E-04
15	0.0	3.8545020E-04	0.0	0.0	1.8974621E-05	2.1193508E-04	2.2768441E-06	-1.5231365E-04
16	0.0	2.1193508E-04	0.0	0.0	3.8718144E-06	1.5247719E-04	6.6854273E-07	-5.4917531E-05
17	0.0	1.5247719E-04	0.0	0.0	1.0648130E-05	5.0483475E-05	2.9070230E-07	-9.1054884E-05
18	0.0	5.0483475E-05	0.0	0.0	1.2402366E-05	2.8143535E-05	2.1481623E-09	-9.9354256E-06
19	0.0	2.8143535E-05	0.0	0.0	1.7071843E-06	1.3059799E-05	2.3000386E-09	-1.3374251E-05
20	0.0	1.3059799E-05	0.0	0.0	8.6868141E-07	4.5998040E-06	1.1709391E-09	-7.5901426E-06
21	0.0	4.5998040E-06	0.0	0.0	3.1549757E-07	8.5534708E-07	4.3514459E-10	-3.4285240E-06
22	0.0	8.5534708E-07	0.0	0.0	1.2308305E-07	0.0	1.0477379E-10	-7.3215921E-07

GROUP NUMBER	VOLUME INTEGRAL OF FLUX	AVERAGE FLUX	FLUX SPECTRUM	GROUP NEUTRON PRODUCTION	PARALLEL BUCKLING
1	-2.4107809E-04	-6.1488379E-08	-5.91127950E-08	0.0	-5.5925641E-02
2	5.3452561E-04	1.3633394E-07	5.71269910E-07	0.0	-3.8919277E-02
3	1.2540778E-03	3.1986002E-07	2.20842490E-06	0.0	-2.2261105E-02
4	8.2142167E-03	2.0950847E-06	2.40630860E-05	0.0	-1.5745558E-02
5	3.6337283E-02	9.2680411E-06	1.74337480E-04	0.0	-1.0638006E-02
6	8.8648617E-02	2.2610358E-05	7.00901080E-04	0.0	-4.6277381E-03
7	8.9493752E-02	2.2825916E-05	1.18543110E-03	0.0	-8.5549429E-03
8	1.1090702E-01	2.8287497E-05	2.38708220E-03	0.0	-1.1062373E-03
9	8.7135017E-02	2.2224296E-05	3.09704180E-03	0.0	-1.6584836E-03
10	7.0897162E-02	1.8082736E-05	4.14594760E-03	0.0	-2.0066281E-03
11	5.6614544E-02	1.4439878E-05	5.44932830E-03	0.0	-3.5467045E-04
12	3.8949359E-02	9.9342669E-06	6.15906910E-03	0.0	-8.4329396E-03
13	2.3909230E-02	6.0981920E-06	6.30128460E-03	0.0	-2.4972398E-02
14	1.7082393E-02	4.3569662E-06	5.50356970E-03	0.0	-4.6616573E-02
15	8.3606653E-03	2.1324377E-06	7.62136480E-03	0.0	-4.6784811E-02
16	2.0100125E-03	5.1266574E-07	5.46468000E-03	0.0	-4.6329819E-02
17	3.3908214E-03	8.6484931E-07	6.48725140E-03	0.0	-4.7923233E-02
18	4.0174718E-04	1.0246799E-07	2.31527650E-03	0.0	-5.0875884E-02
19	5.0112582E-04	1.2781510E-07	2.05336470E-03	0.0	-5.8147930E-02
20	2.7781259E-04	7.0857766E-08	1.40006320E-03	0.0	-6.4820945E-02
21	1.0931538E-04	2.7881541E-08	9.73023540E-04	0.0	-7.8790367E-02
22	2.6512746E-05	6.7622317E-09	4.05588930E-04	0.0	-6.9879949E-02

SUMMED REGION BALANCE DATA

REGION	VOLUME	NET LEAKAGE	ABSORPTION	PERP. LEAKAGE	FISSION SOURCE	FIXED SOURCE
1	1.80955740 03	3.0008709E-01	4.6968812E-01	1.4003861E-01	9.0981412E-01	0.0
2	1.59393804 04	-2.8440624E-01	-2.7466202E-01	-9.9929571E-02	9.0185583E-02	0.0
3	3.92070770 03	-1.6239653E-03	6.7991088E-04	9.4405306E-04	0.0	0.0
TOTALS	2.11240690 04	-1.4056884E-02	7.45030050E-01	-2.40912230E-01	9.99999700E-01	0.0

REGION	POWER	FLUX TIMES VOL.	OUT-SCATTER	IN-SCATTER	SCATTER ERROR
1	1.12806290E-11	3.8696228E 01	8.7810826E-01	8.7810802E-01	0.0
2	1.03796580E-12	4.9140472E-01	9.6994126E-01	9.6994102E-01	0.0
3	0.0	6.4529628E-01	1.3571139E-02	1.3571136E-02	0.0
TOTALS	1.23181950E-11	8.84820050E 01	1.86162070 00	1.86162020 00	

BEGIN AUXILIARY EDITS

3.727

22 GROUP MONA SAMPLE PROBLEM

COALESCED GROUP AND REGION DATA

ENERGY GROUP BREAK POINTS
5' 11' 17' 22'

REGION BREAK POINTS
2 3

COALESCED REGION DIMENSIONS AND FLUX DATA

	REGION 1	REGION 2
WIDTH	7.4000000 01	8.0000000 00
VOLUME	1.72033610 04	3.92070760 03
GROUP NO.	REGION 1	REGION 2
	AVERAGE FLUX	AVERAGE FLUX
1	1.6941866E-03	1.1880806E-05
2	3.2539146E-03	1.2847071E-04
3	1.5737300E-04	2.3899367E-05
4	3.1154207E-07	3.3573470E-07
GROUP NO.	REGION 1	REGION 2
	FLUX*VOLUME	FLUX*VOLUME
1	2.9145706E 01	4.6581168E-02
2	5.5978271E 01	5.0367614E-01
3	2.7073584E 00	9.3702435E-02
4	5.3595714E-03	1.3165139E-03

BOUNDARY CONDITION CONSTANTS

ALPHA1	BETA1	GAMMA1	ALPHA2	BETA2	GAMMA2
0.0	1.00000000 00	0.0	1.00000000 00	2.08320880 00	0.0
0.0	1.00000000 00	0.0	1.00000000 00	2.09980870 00	0.0
0.0	1.00000000 00	0.0	1.00000000 00	2.11128700 00	0.0
0.0	1.00000000 00	0.0	1.00000000 00	2.11805010 00	0.0

AVERAGED MACROSCOPIC CROSS SECTIONS AND GROUP CONSTANTS

AVERAGED REGION 1 (CONSISTS OF ORIGINAL REGIONS 1 THROUGH 2)

POISON CROSS SECTION= 0.0

GROUP NUMBER	DIFFUSION COEFFICIENT	ABSORPTION CROSS SECTION	BUCKLING SQUARED	NU TIMES SIGMA FISSION	FISSION SPECTRUM, CHI	FISSION CROSS SECTION	ENERGY PER FISSION
1	2.2768879E 00	8.7692104E-03	1.98644130-03	1.7921120E-02	8.67750010-01	6.7525357E-03	3.1399966E-11
2	1.2711105E 00	7.8872517E-03	1.49190340-03	8.3079152E-03	1.30481790-01	3.3960170E-03	3.1399966E-11
3	7.3592126E-01	1.7376542E-02	9.98430200-04	8.3276331E-03	1.74855860-03	3.4293467E-03	3.1399966E-11
4	7.5278437E-01	3.8457483E-02	1.04626880-05	1.1358965E-02	1.96415020-05	4.6742149E-03	3.1399966E-11

GROUP NUMBER	TRANSPORT CROSS SECTION	SIGMA POISON MULTIPLIER
1	1.4639860E-01	0.0
2	2.6223785E-01	0.0
3	4.5294702E-01	0.0
4	4.4280070E-01	0.0

SCATTERING MATRIX (ROW INDEX IS GROUP SCATTERED FROM)

	1	2	3	4
1	1.6910318E-02	1.6344596E-02	5.6519464E-05	0.0
2	0.0	1.4361918E-02	0.2603027E-04	5.7726822E-07
3	0.0	0.0	9.8526850E-03	5.2405390E-05
4	0.0	0.0	0.0	2.0433493E-02

AVERAGED REGION 2 (CONSISTS OF ORIGINAL REGIONS 3 THROUGH 3)

POISON CROSS SECTION= 0.0

GROUP NUMBER	DIFFUSION COEFFICIENT	ABSORPTION CROSS SECTION	BUCKLING SQUARED	NU TIMES SIGMA FISSION	FISSION SPECTRUM, CHI	FISSION CROSS SECTION	ENERGY PER FISSION
1	1.7791595E 00	3.7887390E-04	1.69469560-03	0.0	0.0	0.0	0.0
2	1.0969572E 00	8.4802462E-04	1.35142280-03	0.0	0.0	0.0	0.0
3	6.6797620E-01	2.3446467E-03	9.08965170-04	0.0	0.0	0.0	0.0
4	4.5121771E-01	1.1710331E-02	1.03681890-05	0.0	0.0	0.0	0.0

GROUP NUMBER	TRANSPORT CROSS SECTION	SIGMA POISON MULTIPLIER
1	1.8735433E-01	0.0
2	3.0387086E-01	0.0
3	4.9901986E-01	0.0
4	7.3874176E-01	0.0

SCATTERING MATRIX (ROW INDEX IS GROUP SCATTERED FROM)

	1	2	3	4
1	5.7139359E-03	1.8961381E-02	3.7580015E-05	0.0
2	0.0	1.6461000E-02	2.9782504E-03	0.0
3	0.0	0.0	2.7014866E-02	5.3876359E-04
4	0.0	0.0	0.0	3.5440940E-02
1 0 0 1 0 2 4	0 0 0 0 0 0 0 0 0 0 0 0 0 0			

ACTIVATION TRAVERSES

22 GROUP MONA SAMPLE PROBLEM

1 1 0

LIB. MTL. 4, MICRO NA	MESH POINT POSITION	TOTAL ACTIVATION	COARSE GROUP NO.	(MATERIAL NO. 4)				
				ACTIVATIONS				
				1	2	3	4	5
1	0.0	2.3556359E-05	1.0292218E-05	1.2544919E-05	7.1828049E-07	1.4431298E-09	0.0	0.0
2	1.50000000 00	2.3516404E-05	1.0269831E-05	1.2527203E-05	7.1792959E-07	1.4414825E-09	0.0	0.0
3	3.00000000 00	2.3396318E-05	1.0202688E-05	1.2474293E-05	7.1690687E-07	1.4366086E-09	0.0	0.0
4	4.50000000 00	2.3196345E-05	1.0090829E-05	1.2386972E-05	7.1531520E-07	1.4287469E-09	0.0	0.0
5	6.00000000 00	2.2915614E-05	9.9343079E-06	1.2266560E-05	7.1332948E-07	1.4183175E-09	0.0	0.0
6	7.50000000 00	2.2560678E-05	9.7331631E-06	1.2114913E-05	7.1120601E-07	1.4059689E-09	0.0	0.0
7	9.00000000 00	2.2132503E-05	9.4873749E-06	1.1934454E-05	7.0929445E-07	1.3926500E-09	0.0	0.0
8	1.05000000 01	2.1634420E-05	9.1968077E-06	1.1728181E-05	7.0805567E-07	1.3797194E-09	0.0	0.0
9	1.20000000 01	2.1070315E-05	8.8611423E-06	1.1499716E-05	7.0808608E-07	1.3691257E-09	0.0	0.0
10	1.35000000 01	2.0444641E-05	8.4797884E-06	1.1253343E-05	7.1015239E-07	1.3636752E-09	0.0	0.0
11	1.50000000 01	1.9762403E-05	8.0517621E-06	1.0994041E-05	7.1523687E-07	1.3674499E-09	0.0	0.0
12	1.65000000 01	1.9029088E-05	7.5755443E-06	1.0727568E-05	7.2460085E-07	1.3864616E-09	0.0	0.0
13	1.80000000 01	1.8250736E-05	7.0487159E-06	1.0460527E-05	7.3986814E-07	1.4296673E-09	0.0	0.0
14	1.95000000 01	1.7433820E-05	6.4687192E-06	1.0200464E-05	7.6313808E-07	1.5105668E-09	0.0	0.0
15	2.10000000 01	1.6585356E-05	5.8306050E-06	9.9559784E-06	7.9713504E-07	1.6497126E-09	0.0	0.0
16	2.25000000 01	1.5712838E-05	5.1286825E-06	9.7368729E-06	8.4540460E-07	1.8786930E-09	0.0	0.0
17	2.40000000 01	1.4824243E-05	4.3551108E-06	9.5543182E-06	9.1256811E-07	2.264581E-09	0.0	0.0
18	2.55000000 01	1.2939860E-05	2.9115236E-06	9.0053754E-06	1.0200820E-06	2.8788982E-09	0.0	0.0
19	2.70000000 01	1.1294818E-05	1.7750887E-06	8.7489132E-06	1.0675185E-06	3.2369620E-09	0.0	0.0
20	3.15000000 01	9.8317760E-06	1.3580066E-06	7.4010932E-06	1.0692793E-06	3.3963985E-09	0.0	0.0
21	3.40000000 01	8.5226866E-06	9.4532567E-07	6.5360809E-06	1.0378662E-06	3.4130734E-09	0.0	0.0
22	3.65000000 01	7.3929143E-06	6.650433E-07	5.7000625E-06	9.8361761E-07	3.3290131E-09	0.0	0.0
23	3.90000000 01	6.3113257E-06	4.7331571E-07	4.9200398E-06	9.1479347E-07	3.1763865E-09	0.0	0.0
24	4.15000000 01	5.3906660E-06	3.3973640E-07	4.2101346E-06	8.3781515E-07	2.9801326E-09	0.0	0.0
25	4.40000000 01	4.5821016E-06	2.4587320E-07	3.5759213E-06	7.5754809E-07	2.7597584E-09	0.0	0.0
26	4.65000000 01	3.8768748E-06	1.7925606E-07	3.0174360E-06	6.7758180E-07	2.5306508E-09	0.0	0.0
27	4.90000000 01	3.2656162E-06	1.3154511E-07	2.5312775E-06	6.0049859E-07	2.3051154E-09	0.0	0.0
28	5.15000000 01	2.7392853E-06	7.7092538E-08	2.1120431E-06	5.2805620E-07	2.0932707E-09	0.0	0.0
29	5.40000000 01	2.2887380E-06	7.2027433E-08	1.7533139E-06	4.6149256E-07	1.9038797E-09	0.0	0.0
30	5.65000000 01	1.9053337E-06	5.3667144E-08	1.4403121E-06	4.0160887E-07	1.7451753E-09	0.0	0.0
31	5.90000000 01	1.5810683E-06	4.0132694E-08	1.1903221E-06	3.4898784E-07	1.6257478E-09	0.0	0.0
32	6.15000000 01	1.3087529E-06	3.0094803E-08	9.7295742E-07	3.0414469E-07	1.5555588E-09	0.0	0.0
33	6.40000000 01	1.0821477E-06	2.2603810E-08	7.9030406E-07	2.6769317E-07	1.5471837E-09	0.0	0.0
34	6.65000000 01	8.9611018E-07	1.6974308E-08	6.3698985E-07	2.4052849E-07	1.6174448E-09	0.0	0.0
35	6.90000000 01	7.4674546E-07	1.2705602E-08	5.0820694E-07	2.2404316E-07	1.7897155E-09	0.0	0.0
36	7.15000000 01	6.3161337E-07	9.4257224E-09	3.9969433E-07	2.2039592E-07	2.0973892E-09	0.0	0.0
37	7.40000000 01	5.500595E-07	6.8507795E-09	3.0770684E-07	2.3285889E-07	2.5894196E-09	0.0	0.0
38	7.60000000 01	4.859725E-07	5.2466120E-09	2.4034608E-07	2.3750340E-07	3.0791580E-09	0.0	0.0
39	7.80000000 01	3.8326789E-07	3.6465542E-09	1.7672983E-07	2.0005632E-07	2.8351475E-09	0.0	0.0
40	8.00000000 01	2.5152377E-07	2.5276601E-09	1.1599445E-07	1.3104483E-07	1.9568138E-09	0.0	0.0
41	8.20000000 01	1.0377073E-07	1.6014050E-09	5.7267645E-08	4.4245152E-08	6.5656547E-10	0.0	0.0

MESH POINT	MESH POINT POSITION	TOTAL AVERAGE	COARSE GROUP NO.	AVERAGE				
				1	2	3	4	5
1	0.0	8.6487131E-04	7.6631317E-04	9.2610228E-04	2.7578494E-03	7.0357695E-03	0.0	0.0
2	1.50000000 00	8.6499541E-04	7.6630618E-04	9.2626130E-04	2.7578187E-03	7.0358701E-03	0.0	0.0
3	3.00000000 00	8.6536980E-04	7.6628174E-04	9.2674326E-04	2.7577493E-03	7.0361719E-03	0.0	0.0
4	4.50000000 00	8.6600357E-04	7.6622609E-04	9.2756283E-04	2.7577251E-03	7.0366897E-03	0.0	0.0
5	6.00000000 00	8.6691137E-04	7.6611480E-04	9.2874654E-04	2.7578946E-03	7.0374347E-03	0.0	0.0
6	7.50000000 00	8.6811441E-04	7.6590921E-04	9.3033211E-04	2.7584967E-03	7.0384406E-03	0.0	0.0
7	9.00000000 00	8.6964085E-04	7.6555181E-04	9.3237031E-04	2.7598911E-03	7.0397258E-03	0.0	0.0
8	1.05000000 01	8.7152594E-04	7.6495740E-04	9.3492749E-04	2.7626029E-03	7.0413128E-03	0.0	0.0
9	1.20000000 01	8.7381341E-04	7.6400256E-04	9.3808770E-04	2.7673808E-03	7.0432052E-03	0.0	0.0
10	1.35000000 01	8.7655638E-04	7.6250499E-04	9.4195572E-04	2.7752740E-03	7.0453696E-03	0.0	0.0
11	1.50000000 01	8.7981764E-04	7.6019357E-04	9.4665890E-04	2.7877179E-03	7.0476905E-03	0.0	0.0
12	1.65000000 01	8.8367122E-04	7.5668539E-04	9.5235230E-04	2.8066314E-03	7.0499256E-03	0.0	0.0
13	1.80000000 01	8.8820327E-04	7.5136637E-04	9.5921988E-04	2.8345045E-03	7.0516579E-03	0.0	0.0
14	1.95000000 01	8.9351274E-04	7.4331812E-04	9.6747698E-04	2.8744573E-03	7.0522763E-03	0.0	0.0
15	2.10000000 01	8.9971139E-04	7.3108403E-04	9.7736623E-04	2.9302279E-03	7.0510395E-03	0.0	0.0
16	2.25000000 01	9.0672216E-04	7.1228668E-04	9.8915305E-04	3.0060431E-03	7.0473403E-03	0.0	0.0
17	2.40000000 01	9.1527542E-04	6.8286736E-04	1.0031092E-03	3.1063163E-03	7.0410781E-03	0.0	0.0
18	2.65000000 01	9.3839527E-04	6.0926145E-04	1.0357606E-03	3.2310877E-03	7.0322342E-03	0.0	0.0
19	2.90000000 01	9.6989283E-04	5.5172178E-04	1.0652130E-03	3.3241736E-03	7.0279390E-03	0.0	0.0
20	3.15000000 01	1.0061867E-03	5.0642411E-04	1.0924831E-03	3.3975695E-03	7.0255101E-03	0.0	0.0
21	3.40000000 01	1.0450834E-03	4.7058053E-04	1.1181275E-03	3.4580058E-03	7.0240274E-03	0.0	0.0
22	3.65000000 01	1.0852066E-03	4.4212327E-04	1.1424718E-03	3.5096186E-03	7.0230439E-03	0.0	0.0
23	3.90000000 01	1.1256796E-03	4.1949633E-04	1.1657164E-03	3.5551102E-03	7.0223138E-03	0.0	0.0
24	4.15000000 01	1.1659411E-03	4.0151319E-04	1.1879879E-03	3.5963608E-03	7.0216767E-03	0.0	0.0
25	4.40000000 01	1.2056369E-03	3.8725429E-04	1.2093678E-03	3.6348028E-03	7.0210248E-03	0.0	0.0
26	4.65000000 01	1.2445578E-03	3.7602149E-04	1.2299067E-03	3.6717085E-03	7.0202723E-03	0.0	0.0
27	4.90000000 01	1.2826105E-03	3.6723749E-04	1.2496333E-03	3.7084681E-03	7.0193522E-03	0.0	0.0
28	5.15000000 01	1.3198173E-03	3.6046002E-04	1.2685568E-03	3.7469433E-03	7.0182011E-03	0.0	0.0
29	5.40000000 01	1.3563542E-03	3.5532610E-04	1.2866715E-03	3.7900044E-03	7.0167929E-03	0.0	0.0
30	5.65000000 01	1.3926406E-03	3.5155516E-04	1.3039513E-03	3.8424132E-03	7.0151575E-03	0.0	0.0
31	5.90000000 01	1.4295310E-03	3.4882646E-04	1.3203500E-03	3.9123073E-03	7.0136513E-03	0.0	0.0
32	6.15000000 01	1.4686745E-03	3.4695351E-04	1.3357957E-03	4.0137134E-03	7.0119761E-03	0.0	0.0
33	6.40000000 01	1.5132087E-03	3.4564456E-04	1.3501821E-03	4.1705482E-03	7.0111603E-03	0.0	0.0
34	6.65000000 01	1.5690729E-03	3.4453184E-04	1.3633564E-03	4.4224747E-03	7.0112981E-03	0.0	0.0
35	6.90000000 01	1.6475378E-03	3.4300308E-04	1.3750987E-03	4.8312917E-03	7.0121512E-03	0.0	0.0
36	7.15000000 01	1.7701529E-03	3.3987267E-04	1.3850837E-03	5.4810941E-03	7.0125759E-03	0.0	0.0
37	7.40000000 01	1.9786113E-03	3.3261650E-04	1.3928048E-03	6.4532943E-03	7.0105158E-03	0.0	0.0
38	7.60000000 01	2.1937760E-03	3.2337452E-04	1.3898176E-03	7.2975047E-03	7.0124269E-03	0.0	0.0
39	7.80000000 01	2.3106926E-03	3.1683315E-04	1.3792226E-03	7.7492073E-03	7.0159249E-03	0.0	0.0
40	8.00000000 01	2.2637967E-03	3.1277142E-04	1.3512389E-03	7.7524260E-03	7.0187226E-03	0.0	0.0
41	8.20000000 01	1.8114571E-03	3.1182589E-04	1.2626103E-03	6.6035874E-03	7.0210807E-03	0.0	0.0

BEGIN PROBLEM

0.965