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**INCITE -- A FORTRAN-IV PROGRAM
TO GENERATE THERMAL NEUTRON SPECTRA AND
MULTIGROUP CONSTANTS USING ARBITRARY SCATTERING KERNELS**

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U. S. ATOMIC ENERGY COMMISSION

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ABSTRACT

INCITE is a computer program to calculate energy-dependent thermal neutron spectra and appropriate average multigroup cross sections using arbitrary scattering kernels. The program employs a normalized Gauss-Seidel iteration technique to solve the energy-dependent integral form of the B-1 approximation to the Boltzmann transport equation. A maximum of 101 discrete energy points is utilized to represent the energy range. A library tape containing the proper absorption and scattering cross sections at these energy points is required. The current master library contains cross sections for 101 energy points in the energy range from 0.0 to 2.38 eV.

The INCITE code is written in FORTRAN-IV language and is presently operating on the IBM-7040 and the UNIVAC-1108 digital computers. The running time for INCITE depends on the number of isotopes in the problem as well as the size of the data records that must be processed for each isotope. However, the running time for a typical problem, such as the sample problem contained in the report, should rarely exceed seven minutes on the IBM-7040 or one minute on the UNIVAC-1108.

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I. INTRODUCTION

INCITE is a program employing an iterative solution to the integral formulation of the neutron thermalization problem. The code, which is written in FORTRAN-IV language for use on the IBM-7040 computer, calculates energy-dependent thermal spectra and performs appropriate cross section averaging to generate multigroup constants and cross sections for use in diffusion and transport codes. The most general types of neutron scattering kernels may be used with minimal approximation in obtaining the thermal spectra. An auxiliary code, LIBMOD [12], is presently under development to calculate free gas scattering kernels and make modifications to the library tape.

Various differential equations have been used in the past to solve the neutron thermalization problem for a homogeneous mixture of absorbers and moderators. These differential equations result from assumptions about the scattering kernels for the moderating atoms in the mixture. In almost all cases these assumptions approximate reality rather poorly. With the advent of thermal inelastic scattering kernels for water and other moderators, the solution of the neutron thermalization problem by means of gas differential equations is no longer tenable. The difficulty lies in the fact that it is not possible to treat the details of the differential inelastic scattering kernel within the framework of the gas equations in any form. In contrast, the integral form of the neutron thermalization equations does not place any restrictions on the scattering kernels employed, with the added advantage that mixtures of moderator kernels with greatly different characteristics may be used in the same problem. The disadvantages of the integral form of the equation are that large amounts of computer storage are required to handle the scattering matrices and that iterative solutions are necessary. However, with present day computers these are not serious drawbacks.

The utility of both the Wigner-Wilkins and Brown and St. John gas kernels is also enhanced when used in the integral form of the neutron thermalization equations. Approximations are necessary when using either of the above gas kernels in the differential form of the thermalization equations for every case except hydrogen. Because these gas kernels have been widely used in the past, and are advantageous when studying temperature effects for a large number of temperatures where it is costly to generate numerous inelastic kernels, they will be included as an option in the LIBMOD code. The cost of generating these gas kernels is negligible in comparison to the inelastic kernels presently in use.

Extensive experimental measurement and theoretical development of thermal inelastic scattering kernels have been underway at this laboratory for the past few years. The MR kernels [1] for water and D₂O are two of the products of these investigations. The INCITE code was based on considerations which would permit these and other thermal inelastic scattering kernels currently in use to be used without serious mathematical approximation

in reactor physics calculations for semihomogeneous systems. For these calculations to be practical, it is necessary to provide for gross leakage effects, realistic source spectra, and a complete library of thermal absorption cross sections with provision for treating low lying resonance absorption in mixtures of isotopes. The B-1 spherical harmonics equations for a bare homogeneous system provide a framework for computing spectra which satisfies these minimal requirements. The pertinent constants and cross sections necessary for diffusion and transport multigroup calculations can be obtained in a consistent manner from the B-1 spectra in a reasonable amount of computational time.

The B-1 equations have been cast in the form of a normalized Gauss Seidel iterative scheme where extrapolation and iteration in both directions have been included as options. The Gauss Seidel scheme with renormalization after every iteration has proved itself superior in the limited number of cases tested so far. The energy mesh presently being used has been set at 101 points to conform to the mesh used at General Atomics in the GATHER II code [2]. This was done to take advantage of the large number of inelastic scattering kernels already generated for this mesh. Although the number of points in this mesh is satisfactory for the H₂O scattering kernel, the choice of spacing is somewhat obscure. However, the advantages to be gained by using this mesh and the scattering kernels provided for it by General Atomics outweighed any considerations for a new mesh.

The library of cross sections presently available for INCITE is a revised version of a library obtained from General Atomics for the GATHER II code. Most of the heavy elements in the library do not contain scattering matrices. If scattering matrices are deemed necessary, they may be calculated from the gas kernels which will be available in the LIBMOD code. Some of the capture and scattering cross sections in the library have been revised to conform to recent measurements; however, most of the isotope cross sections in the library consist entirely of data obtained from General Atomics. The source of the kernel, for those moderator materials which possess an inelastic scattering matrix, is identified in the library listing.

At the present time no Doppler broadened resonance calculation from resonance parameters is included in INCITE. Smooth absorption cross sections are provided already Doppler broadened at room temperature. The present mesh is not deemed satisfactory to detect small differences in the Doppler temperature coefficient in the thermal region. The mesh will probably be changed in the future with provisions to accommodate an accurate multilevel and single-level resonance calculation.

A code presently under development, the LIBMOD code, will permit the revision of the INCITE cross section library. There will be provisions to modify or make additions to an existing cross section library or to create a complete new library tape. Options will be available for reading new cross section data from cards or, if the user desires, for calculating P_0 and P_1 elastic scattering kernels using the Wigner-Wilkins or Brown and St. John kernels for a monatomic gas.

II. DERIVATION OF EQUATIONS

The equations solved by INCITE are the energy-dependent B-1 spherical harmonics approximation to the transport equation originally developed by D. S. Selengut[3]. The physical model is a homogeneous, critical, one-dimensional, bare slab core. As such, the one-dimensional, steady state, energy-dependent, Boltzman transport equation applies[4].

$$\mu \frac{\partial}{\partial Z} \psi(Z, E, \mu) + \Sigma_t(E) \psi(Z, E, \mu) =$$

$$\frac{1}{4\pi^2} \int_{E'} \int_{\mu'} \int_{\phi'} \int_{\phi} \psi(Z, E', \mu') \Sigma_s(E' \rightarrow E, \mu_0) d\phi d\phi' d\mu' dE' + s(Z, E, \mu) \quad (1)$$

where $\psi(Z, E, \mu)$ is the neutron flux in the direction $\mu = \cos \theta$ relative to the one-dimensional Z axis of the slab at energy E and position Z, and ϕ and ϕ' are the azimuthal angles associated with μ and μ' , respectively.

$$\Sigma_s(E' \rightarrow E, \mu_0) = \sum_{k=1}^{NOI} N_k \sigma_s^k(E' \rightarrow E, \mu_0) \quad (2)$$

where $\sigma_s^k(E' \rightarrow E, \mu_0) = 2\pi \sigma_s^k(E' \rightarrow E, \vec{\Omega}' \rightarrow \vec{\Omega})$ is the laboratory system double differential scattering cross section for material k in the homogeneous mixture appropriately averaged over the thermal motion of the material k moderator atoms in a manner consistent with the specifications for the thermal scattering law. The cosine of the scattering angle, μ_0 , is $\vec{\Omega} \cdot \vec{\Omega}'$, where $\vec{\Omega}$ and $\vec{\Omega}'$ are taken relative to the fixed reactor reference frame. This form of the integral term in Equation (1) implies that the differential scattering law is azimuthally symmetric after appropriate averaging of the double differential scattering kernel for material k over orientations of molecules and thermal motion of the individual target atoms for molecular mixtures and gases. $s(Z, E, \mu)$, as yet unspecified, is the source of neutrons from outside the thermal energy range considered by Equation (1).

The spatial dependence of flux $\psi(Z, E, \mu)$ may be eliminated from Equation (1) by assuming that the solution is separable in (E, μ) and Z. A monoenergetic spatial solution for the fundamental mode of the critical homogeneous slab is $\cos Bx$ where B is the geometric buckling of the fundamental mode. If the scalar flux component of the function $\psi(Z, E, \mu)$ obeys diffusion theory boundary conditions for a homogeneous bare slab and if it is further assumed that the scalar flux is zero at the same extrapolated boundary for all energies E, $\psi(Z, E, \mu)$ may be represented as a separable function of (E, μ) and Z, namely,

$$\psi(Z, E, \mu) = \phi(E, \mu) e^{-iBZ} \quad (3)$$

where only the real part of e^{-iBZ} satisfies the spatial boundary conditions for the scalar flux. [See the derivation of the diffusion coefficient following Equation (46).] The complex form of the spatial dependence is necessary in order for the equations to remain homogeneous. Using Equation (3), Equation (1) becomes

$$\left[\Sigma_t(E) - iB\mu \right] \phi(E, \mu) = \frac{1}{4\pi^2} \int_{E'} \int_{\mu'} \int_{\phi'} \int_{\phi} \phi(E', \mu') \Sigma_s(E' \rightarrow E, \mu_0) d\phi d\phi' d\mu' dE' + S(E, \mu) \quad (4)$$

Equation (4) assumes that the source of neutrons scattered from the epithermal region above the thermal cutoff, E_0 , has the same spatial dependence as the thermal flux, namely, $s(Z, E, \mu) = S(E, \mu) e^{-iBZ}$. This assumption is also necessary in order for the equations to remain homogeneous.

The B-1 equations are obtained by expanding the scattering function into a Legendre series and truncating the expansion at $\ell = 1$. The flux is likewise expanded into a Legendre series, but truncation is not specified. The source term $S(E, \mu)$ is also truncated at $\ell = 1$.

$$\Sigma_s(E' \rightarrow E, \mu_0) = \frac{\Sigma_{s0}(E' \rightarrow E)}{2} + \frac{3}{2} \mu_0 \Sigma_{s1}(E' \rightarrow E) \quad (5)$$

$$S(E, \mu) = \frac{S_0(E)}{2} + i \frac{3}{2} \mu S_1(E) \quad (6)$$

$$\phi(E, \mu) = \sum_{\ell=0}^{\infty} \frac{2\ell+1}{2} \left[(i)^\ell F_\ell(E) \right] P_\ell(\mu) \quad (7)$$

If Equations (5), (6), and (7) are substituted on the r.h.s. of Equation (4), the result is

$$\begin{aligned}
\left[\Sigma_t(E) - iB\mu \right] \phi(E, \mu) &= \frac{1}{2} \int_{E'} \Sigma_{s0}(E' \rightarrow E) F_0(E') dE' \\
&+ \frac{1}{4\pi} \frac{3}{2} \int_{E'} \Sigma_{s1}(E' \rightarrow E) F_1(E') \int_0^{2\pi} \int_0^{2\pi} \int_{-1}^1 i \frac{3}{2} \mu_0 \mu' d\mu' d\phi' d\phi dE' \\
&+ \frac{S_0(E)}{2} + i \frac{3}{2} \mu S_1(E)
\end{aligned} \tag{8}$$

where the orthogonality of the Legendre polynomials has been used in the integral terms and the indicated integrations performed in the first term on the r.h.s. The second term may be integrated after use of the Legendre Polynomial addition theorem for μ_0 .

$$\mu_0 = \mu \mu' + \sin \theta \sin \theta' \cos(\phi - \phi') \tag{9}$$

Substituting into Equation (8) and performing the indicated integrations yields

$$\begin{aligned}
\left[\Sigma_t(E) - iB\mu \right] \phi(E, \mu) &= \frac{1}{2} \int_{E'} \Sigma_{s0}(E' \rightarrow E) F_0(E') dE' \\
&+ i \frac{3}{2} \mu \int_{E'} \Sigma_{s1}(E' \rightarrow E) F_1(E') dE' + \frac{S_0(E)}{2} + i \frac{3}{2} \mu S_1(E) .
\end{aligned} \tag{10}$$

The B-1 recursion equations are obtained by dividing Equation (10) through by $\Sigma_t(E) - iB\mu$ and expanding $\phi(E, \mu)$ in a Legendre series by means of the following relationship for the Legendre moments:

$$(i)^\ell F_\ell(E) = \int_{-1}^1 \phi(E, \mu) P_\ell(\mu) d\mu .$$

The result for the first two Legendre moments is

$$\begin{aligned}
F_0(E) &= \int_{-1}^1 \frac{1}{\Sigma_t(E) - iB\mu} \left\{ \frac{1}{2} \int_{E'} \Sigma_{s0}(E' \rightarrow E) F_0(E') dE' + \frac{S_0(E)}{2} \right. \\
&\quad \left. + i \frac{3}{2} \mu \left[\int_{E'} \Sigma_{s1}(E' \rightarrow E) F_1(E') dE' + S_1(E) \right] \right\} d\mu
\end{aligned} \tag{11}$$

and

$$i F_1(E) = \int_{-1}^1 \frac{\mu}{\Sigma_t(E) - 1B\mu} \left\{ \frac{1}{2} \int_{E'} \Sigma_{s0}(E' \rightarrow E) F_0(E') dE' + \frac{S_0(E)}{2} \right. \\ \left. + i \frac{3}{2} \mu \left[\int_{E'} \Sigma_{s1}(E' \rightarrow E) F_1(E') dE' + S_1(E) \right] \right\} d\mu \quad (12)$$

After rationalizing the μ integral, the integrations are elementary and yield[4]

$$F_0(E) = \frac{\beta}{\Sigma_t(E)} \left[\int_{E'} \Sigma_{s0}(E' \rightarrow E) F_0(E') dE' + S_0(E) \right] \\ - \frac{3(1 - \beta)}{\alpha \Sigma_t(E)} \left[\int_{E'} \Sigma_{s1}(E' \rightarrow E) F_1(E') dE' + S_1(E) \right] \quad (13)$$

$$F_1(E) = \frac{(1 - \beta)}{\alpha \Sigma_t(E)} \left[\int_{E'} \Sigma_{s0}(E' \rightarrow E) F_0(E') dE' + S_0(E) \right] \\ + \frac{3(1 - \beta)}{\alpha^2 \Sigma_t(E)} \left[\int_{E'} \Sigma_{s1}(E' \rightarrow E) F_1(E') dE' + S_1(E) \right] \quad (14)$$

where

$$\alpha = \frac{B}{\Sigma_t(E)}, \quad \beta = \frac{\tan^{-1} \alpha}{\alpha} \quad (15)$$

Equations (13) and (14) may be put into the form of the continuity equations for the energy-dependent neutron flux and current. First, multiply Equation (14) by α and add it to Equation (13) to obtain Equation (16) below. Second, eliminate the isotropic integrals from Equation (16) and Equation (13) to obtain Equation (17).

$$BF_1(E) + \Sigma_t(E)F_0(E) = \int_{E'} \Sigma_{s0}(E' \rightarrow E)F_0(E')dE' + S_0(E) \quad (16)$$

$$\gamma(E)\Sigma_t(E)F_1(E) - \frac{B}{3}F_0(E) = \int_{E'} \Sigma_{s1}(E' \rightarrow E)F_1(E')dE' + S_1(E) \quad (17)$$

where

$$\gamma(E) = \frac{1}{3} \frac{\beta \alpha^2}{(1 - \beta)} \quad (18)$$

$F_0(E)$ and $F_1(E)$ are interpreted as the energy-dependent flux and current, respectively, where $S_0(E)$ and $S_1(E)$ are interpreted as the energy-dependent isotropic and linearly anisotropic source of neutrons into the thermal energy range from the epithermal region. By examining Equations (16) and (17), it is seen that the B-1 approximation differs from the P-1 approximation by the factor $\gamma(E)$ in Equation (17). If $\gamma(E)$ is set equal to 1.0, Equations (16) and (17) become the continuity equations for the P-1 approximation. The factor $\gamma(E)$ tends to improve the B-1 estimation of the energy-dependent transport current, $F_1(E)$, for small homogeneous systems[3].

The GA energy mesh which is presently being used in INCITE has an upper energy cutoff, E_c , at 2.38 eV. Therefore, the source neutrons are those which result from scattering collisions above 2.38 eV. A reasonable assumption in this epithermal region is that the energy-dependent neutron flux and current have a $1/E'$ dependence and that the target atoms for all elements in the homogeneous mixture are at rest in the laboratory system of coordinates. With these two assumptions, the source terms $S_0^k(E)$ and $S_1^k(E)$ for isotope k become

$$S_0^k(E) = \int_{E_c}^{\frac{E}{\alpha_k}} N_k \sigma_{s0}^k \frac{dE'}{(E')^2 (1 - \alpha_k)} \quad (19)$$

and

$$S_1^k(E) = \int_{E_c}^{\frac{E}{\alpha_k}} N_k \mu_k \sigma_{s0}^k \frac{J(E') dE'}{(E') (1 - \alpha_k)} \quad (20)$$

where

$$\alpha_k = \left[\frac{A_k - 1}{A_k + 1} \right]^2$$

In integrating Equation (19), it is assumed that the scattering cross section of material k in the epithermal region is equal to the potential scattering cross section and therefore constant. The result is

$$S_0^k(E) = \frac{N_k \sigma_{s0}^k}{1 - \alpha_k} \left[\frac{1}{E_c} - \frac{\alpha_k}{E} \right] \quad (21)$$

In Equation (20), the assumption of isotropic scattering for targets at rest in the center of mass system yields for μ_k

$$\mu_k = \frac{A_k + 1}{2} \sqrt{\frac{E}{E'}} - \frac{A_k - 1}{2} \sqrt{\frac{E'}{E}} \quad (22)$$

The current $J(E')$ may be defined from the equation [see discussion following Equation (46)]

$$\begin{aligned} \vec{J}(E', Z) &= -D(E') \vec{k} \frac{d}{dZ} \left[F_0(E') \cos BZ \right] = \frac{D(E')B}{E'} \vec{k} \sin BZ \\ &= J(E') \vec{k} \sin BZ \end{aligned} \quad (23)$$

where $F_0(E')$ is assumed $1/E'$ as in Equation (19).

If $D(E')$ and σ_{s0}^k are assumed constant in the epithermal region and Equations (22) and (23) are used in Equation (20), the result of integration is

$$\begin{aligned} S_1^k(E) &= - \frac{D B N_k \sigma_{s0}^k}{3 \sqrt{E} (1 - \alpha_k)} \left\{ (A_k + 1) E \left[\left(\frac{\alpha_k}{E} \right)^{\frac{3}{2}} - \left(\frac{1}{E_c} \right)^{\frac{3}{2}} \right] \right. \\ &\quad \left. - 3(A_k - 1) \left[\left(\frac{\alpha_k}{E} \right)^{\frac{1}{2}} - \left(\frac{1}{E_c} \right)^{\frac{1}{2}} \right] \right\} \end{aligned} \quad (24)$$

where

$$D = \frac{1}{3\Sigma_{tr}} = \frac{1}{3 \sum_{k=1}^{NOI} N_k \left(\sigma_a^k + \sigma_{s0}^k - \sigma_{s1}^k \right)} \quad (24A)$$

If a particular material possesses a scattering matrix, then σ_{s0}^k in Equations (21), (24), and (24A) is set equivalent to the free atom scattering cross section

contained on the library tape and $\sigma_{s1}^k = \bar{\mu}_k \sigma_{s0}^k$, where $\bar{\mu}_k = \frac{2}{3A_k}$. The absorption cross section is assumed to be the value of σ_a at 2.38 eV for that material. If the material does not have a scattering matrix then the microscopic cross sections in Equation (24A) are obtained from the library tape at 2.38 eV. All materials k are present in the diffusion coefficient since the net current flow above energy E_c receives a contribution from all materials in the mixture. The linearly anisotropic component of the collision density at energy E' for a single atom of material k is, therefore, proportional to $D \sigma_{s1}^k(E' \rightarrow E)$ when Equation (23) is used in Equation (20).

The total source contribution for the mixture is obtained by summing the sources for all materials k in the mixture which possess either an elastic or inelastic scattering matrix. Since no sources are computed or included on the library tape for nonmoderating materials, a scattering matrix must either be input to the library or calculated by means of the gas equations, included in the LIBMOD code, if the source terms from a particular isotope are desired.

III. NUMERICAL SOLUTION

In this section, the numerical equivalent of Equations (16) and (17) will be arranged for the Gauss Seidel iterative scheme. In INCITE, the thermal energy range from $E = 0$ to $E = E_c$ may be divided into a maximum of 101 energy points. $F_0(E)$ and $F_1(E)$, defined respectively as the scalar flux and current at energy E , take on values ϕ_i and J_i at each of the energy points E_i in the mesh. It is assumed that both $F_0(E)$ and $F_1(E)$ are linear functions between the discrete energy points E_i . In a similar way the singly differential scattering cross section is defined for transfer between the discrete energy points j and i . The numerical equivalent of Equations (16) and (17) for the i^{th} energy point is

$$B J_i + \Sigma_{t_i} \phi_i = \sum_{j=1}^{\text{NOG}} \Sigma_{s0_{ji}} \phi_j \Delta_j + S_{0_i} \quad (25)$$

and

$$\gamma_i \Sigma_{t_i} J_i - \frac{B}{3} \phi_i = \sum_{j=1}^{\text{NOG}} \Sigma_{s1_{ji}} J_j \Delta_j + S_{1_i} \quad (26)$$

where the integration is approximated using a trapezoidal scheme. In transforming to the numerical form, the following definitions were applied:

i and $j \Rightarrow$ energy points ($i=1$ is the lowest fine energy point)

NOI $\Rightarrow$ number of materials in the mixture (a molecular material such as H_2O possessing a combined scattering matrix is defined as a single material)

NOG $\Rightarrow$ number of fine energy points

$\phi_i = F_0(E_i) \Rightarrow$ flux at i^{th} energy point

$J_i = F_1(E_i) \Rightarrow$ current at i^{th} energy point

$\Delta_i \Rightarrow$ energy width associated with the i^{th} energy point

For the present energy structure, the Δ_i 's have been defined as

$$\Delta_i = \frac{E_{i+1} - E_{i-1}}{2}$$

with the exception that

$$\Delta_1 = 0.001 \text{ eV, ie, from } 0.0005 \text{ to } 0.0015 \text{ eV}$$

and

$$\Delta_{101} = 0.025 \text{ eV, ie, } \Delta_{101} = \frac{E_{101} - E_{100}}{2} .$$

The end point deltas are somewhat arbitrary and must be determined by the user if a new energy mesh is input.

B = buckling of fundamental mode defined in the conventional manner

$$\Sigma_{s0_{ji}} = \sum_{k=1}^{NOI} N_k \sigma_{s0_{ji}}^k \Rightarrow \text{differential scattering cross section from } j^{\text{th}} \text{ energy point to } i^{\text{th}} \text{ energy point}$$

$$\Sigma_{s1_{ji}} = \sum_{k=1}^{NOI} N_k \sigma_{s1_{ji}}^k \Rightarrow \text{linearly anisotropic component of the differential scattering cross section from } j^{\text{th}} \text{ energy point to } i^{\text{th}} \text{ energy point}$$

The $\sigma_{s0_{ji}}^k$ and $\sigma_{s1_{ji}}^k$ are obtained from the cross section library for moderating materials

$$S_{0_i} = \sum_{k=1}^{NOI} S_{0_i}^k \Rightarrow \text{scattering source term from above } E_c \text{ to the } i^{\text{th}} \text{ energy point}$$

$$S_{1_i} = \sum_{k=1}^{NOI} S_{1_i}^k \Rightarrow \text{linearly anisotropic component of the scattering source term from above } E_c \text{ to the } i^{\text{th}} \text{ energy point.}$$

Since the finite upper limit, E_c , is assumed in obtaining a solution to Equations (25) and (26), a provision has been made to account for scattering reactions into the energy range above this upper energy cutoff. In this case the total cross section for reactions at the i^{th} energy point is

$$\Sigma_{t_i} = \sum_{k=1}^{NOI} N_k \sigma_{a_i}^k + \sum_{k=1}^{NOI} N_k \left[\sum_{j=1}^{NOG} \sigma_{s0_{ij}}^k \Delta_j + \int_{E_c}^{\infty} \sigma_{s0}^k (E_i \rightarrow E'') dE'' \right] \quad (27)$$

where the integral term is defined for scattering to energies which lie above E_c . For each material, the cross section library contains a total scattering cross section at each energy point and may or may not possess a scattering matrix. Those library materials which possess scattering matrices contain

this outscatter contribution to above E_c as part of the total scattering cross sections. These contributions can be obtained on the basis of an extrapolation to the experimental free atom cross section or based on a scattering matrix with points lying above E_c . In the case where no scattering matrix is present in the library, a gas matrix may be calculated using the LIBMOD code. Then total P_0 and P_1 scattering cross sections will be calculated and placed in the library for that material. Options will be included in the LIBMOD code to permit the modification of total scattering cross sections at each mesh point to account for the scattering to above E_c .

To obtain the equations to be solved by the Gauss Seidel iterative scheme, first transpose the terms BJ_i and $\frac{B}{3} \phi_i$ to the r.h.s. of Equations (25) and (26), respectively. Then take the within-group scattering terms, ie, $\Sigma_{s0_{ii}} \phi_i \Delta_i$ and $\Sigma_{sl_{ii}} J_i \Delta_i$, from the scattering sums in equations (25) and (26) and transpose to the l.h.s. obtaining new coefficients for ϕ_i and J_i of $(\Sigma_{t_i} - \Sigma_{s0_{ii}} \Delta_i)$ and $(\gamma_i \Sigma_{t_i} - \Sigma_{sl_{ii}} \Delta_i)$, respectively. Then by dividing each equation through by the coefficient of ϕ_i and J_i , the flux and current for the i^{th} energy point can be isolated.

Now by assuming an initial flux and current guess, an improved solution can be obtained for each value of the flux (current) at each energy point in turn by using the fluxes (currents) at all other energy points and the current (flux) at that point. The code always calculates improved values of the flux or current in an orderly fashion, ie, either proceeding through the energy mesh in the direction of ascending or descending values of energy. Thus, on the $(m+1)^{st}$ iteration certain values of the flux or current have already been recalculated for the $(m+1)^{st}$ time, but other values are still those for the m^{th} iteration. In INCITE, when performing the $(m+1)^{st}$ iteration, the $(m+1)^{st}$ values of the fluxes or currents are used when possible, otherwise the m^{th} values are used. The flux and current equations for the $(m+1)^{st}$ iteration in the downscatter direction (ie, proceeding in the direction of decreasing energy) at the i^{th} energy point are[5]

$$\phi_i^{(m+1)} = \frac{S_{0_i} - B J_i^{(m)}}{\Sigma_{t_i} - \Sigma_{s0_{ii}} \Delta_i} + \sum_{j=1}^{i-1} a_{ji} \phi_j^{(m)} + \sum_{j=i+1}^{NOG} a_{ji} \phi_j^{(m+1)} \quad (28)$$

$$J_i^{(m+1)} = \frac{S_{1_i} + \frac{B}{3} \phi_i^{(m+1)}}{\gamma_i \Sigma_{t_i} - \Sigma_{sl_{ii}} \Delta_i} + \sum_{j=1}^{i-1} b_{ji} J_j^{(m)} + \sum_{j=i+1}^{NOG} b_{ji} J_j^{(m+1)} \quad (29)$$

$$\text{where } a_{ji} = \frac{\Sigma_{s0_{ji}} \Delta_j}{\Sigma_{t_i} - \Sigma_{s0_{ii}} \Delta_i} \text{ and } b_{ji} = \frac{\Sigma_{sl_{ji}} \Delta_j}{\gamma_i \Sigma_{t_i} - \Sigma_{sl_{ii}} \Delta_i}.$$

In iteratively solving the above equations, use can be made of the various techniques for aiding convergence. The INCITE code utilizes both linear extrapolation and renormalization. The procedure is to first calculate the extrapolated flux at each energy point and then solve for the extrapolated currents. After the fluxes and currents have been calculated at all energy points, a renormalization factor is determined and applied to all fluxes and currents.

If ω is the extrapolation factor, then the extrapolated values of the fluxes and currents in group i for the $(m+1)^{\text{st}}$ iteration are determined using the equations

$$\phi_i^{(m+1)} = (1-\omega) \phi_i^{(m)} + \omega \left[\frac{S_{0i} - B J_i^{(m)}}{\Sigma_{t_i} - \Sigma_{s0_{ii}} \Delta_i} + \sum_{j=1}^{i-1} a_{ji} \phi_j^{(m)} + \sum_{j=i+1}^{\text{NOG}} a_{ji} \phi_j^{(m+1)} \right] \quad (30)$$

and

$$J_i^{(m+1)} = (1-\omega) J_i^{(m)} + \omega \left[\frac{S_{1i} + \frac{B}{3} \phi_i^{(m+1)}}{\gamma_i \Sigma_{t_i} - \Sigma_{s1_{ii}} \Delta_i} + \sum_{j=1}^{i-1} b_{ji} J_j^{(m)} + \sum_{j=i+1}^{\text{NOG}} b_{ji} J_j^{(m+1)} \right] \quad (31)$$

The renormalization factor was derived from consideration of the neutron balance equation for the entire energy range.

$$\sum_{i=1}^{\text{NOG}} B J_i^{(m)} \alpha^{(m)} \Delta_i + \sum_{i=1}^{\text{NOG}} \Sigma_{t_i} \phi_i^{(m)} \alpha^{(m)} \Delta_i = \sum_{i=1}^{\text{NOG}} \sum_{j=1}^{\text{NOG}} \Sigma_{s0_{ji}} \phi_j^{(m)} \alpha^{(m)} \Delta_j \Delta_i + \sum_{i=1}^{\text{NOG}} S_{0i} \Delta_i \quad (32A)$$

where $\alpha^{(m)}$ is the renormalization factor for the m^{th} iteration. Now, making the substitution from Equation (27),

$$\Sigma_{t_i} = \Sigma_{a_i} + \sum_{j=1}^{\text{NOG}} \Sigma_{s0_{ij}} \Delta_j + \int_{E_c}^{\infty} \Sigma_{s0} (E_i \rightarrow E'') dE''$$

and canceling the corresponding terms in the scattering sums, Equation (32A) becomes

$$\sum_{i=1}^{\text{NOG}} \left[B J_i^{(m)} + \Sigma_{a_i} \phi_i^{(m)} + \phi_i^{(m)} \int_{E_c}^{\infty} \Sigma_{s0} (E_i \rightarrow E'') dE'' \right] \alpha^{(m)} \Delta_i = \sum_{i=1}^{\text{NOG}} S_{0i} \Delta_i \quad (32B)$$

Thus, the renormalization factor for the m^{th} iteration is

$$\alpha^{(m)} = \frac{\sum_{i=1}^{\text{NOG}} S_{0i} \Delta_i}{\sum_{i=1}^{\text{NOG}} \left[B J_i^{(m)} + \Sigma_{a_i} \phi_i^{(m)} + \phi_i^{(m)} \int_{E_c}^{\infty} \Sigma_{s0}(E_i \rightarrow E'') dE'' \right] \Delta_i} \quad (33)$$

where all terms in the denominator represent actual neutron loss. Experience has shown that the renormalization factor, which is calculated after every iteration, is very effective in speeding convergence. However, the extrapolation factor, which is input by the user, is not too helpful and can even impair convergence if used incorrectly. It has been found that a conservative value of ω between 1.0 and 1.1 should be used unless previous experience with a similar problem justifies the use of values greater than this. If this value is not specified by the user, the code sets ω equal to 1.0, ie, no extrapolation.

An option has been implemented in the INCITE code, on an experimental basis, to allow the iterative calculation to proceed in either direction, ie, to solve for the fluxes and currents at the highest energies first and proceed to the lowest energies or to start by calculating the fluxes and currents at the lowest energies and proceed to the highest energies. Also, any number of iterative sweeps can be made through the energy mesh in the downscatter direction for each sweep in the upscatter direction or vice-versa.

Intuitively, most calculations involving materials at or near room temperature should converge rapidly by doing the calculation exclusively in the downscatter direction since the determination of the correct downscatter terms probably will be more important than the upscatter terms. However, in some cases involving very high moderator temperatures it may be helpful to run a calculation in the upscatter direction after a certain number of iterations in the downscatter direction. The proper use of this option should become apparent after some experimentation by the user.

The user has been given two options for determining convergence of a problem. The first and most reliable option is to converge on pointwise values of the fluxes and currents. In this case the calculation will be converged when the following equations are satisfied at all energy points.

$$\frac{|\phi_i^{(m+1)} - \phi_i^{(m)}|}{\phi_i^{(m+1)}} < \epsilon \quad \text{and} \quad \frac{|J_i^{(m+1)} - J_i^{(m)}|}{J_i^{(m+1)}} < \epsilon$$

If the user desires to converge on the renormalization factor, then the problem will be considered converged when

$$|\alpha^{(m+1)} - \alpha^{(m)}| < \epsilon.$$

If the convergence criterion is not satisfied after the maximum number of iterations has been completed, a statement of this fact will be printed, the fluxes and currents obtained on the last and the next to the last iterations will be listed, and the code will proceed to the averaging routine where average cross sections will be calculated using the last iteration values of the fluxes and currents. In most cases run to date using a reasonable convergence criteria, ie, $\epsilon = 0.00001$, the problem has converged in less than 20 iterations.

IV. CALCULATION OF MULTIGROUP AVERAGE CROSS SECTIONS

In this section the equations for calculating the average cross sections have been set down in as complete a manner as practical. This should facilitate the proper use of the average cross sections generated by INCITE.

The diagram below illustrates the schemes for numbering the fine energy points and the broad energy groups.

In this example, E_I and E_{I+n} represent the lower and upper cutoff for the broad group G lying between the energy points I and I+n. In all cases the integration is completed over the energy range limited by the upper and lower energy points of each broad group using partial energy widths where necessary.

The flux and current at energy point I will be denoted by ϕ_I and J_I , respectively.

The integrated fluxes and currents for broad group G are calculated by the equations

$$\phi_G = \phi_I \left(\frac{E_{I+1} - E_I}{2} \right) + \sum_{i=I+1}^{I+n-1} \phi_i \Delta_i + \phi_{I+n} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right) \quad (34)$$

$$J_G = J_I \left(\frac{E_{I+1} - E_I}{2} \right) + \sum_{i=I+1}^{I+n-1} J_i \Delta_i + J_{I+n} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right) \quad (35)$$

where $\Delta_I = \frac{E_{I+1} - E_{I-1}}{2}$. In all cases involving the integration of fluxes and currents or the averaging of cross sections, when $I=1$ the quantity $\frac{E_{I+1} - E_I}{2}$ is set equal to Δ_1 , and when $I+n=NOG$ the quantity $\frac{E_{I+n} - E_{I+n-1}}{2}$ is set equal to Δ_{NOG} where Δ_1 and Δ_{NOG} represent the first and last energy widths associated with the first and last energy groups, respectively.

1. AVERAGE MACROSCOPIC CROSS SECTIONS

The following equations describe the average macroscopic cross sections for broad group G. σ (barns/atom) represents the microscopic cross section, and $\Sigma(\text{cm}^{-1}) = \sum_{k=1}^{\text{NOI}} N_k \sigma_k$ represents the macroscopic cross section where N_k is the number of atoms $\times 10^{-24}$ of type k per cubic centimeter.

Absorption

$$\Sigma_{a_G} = \frac{\Sigma_{a_I} \left(\frac{E_{I+1} - E_I}{2} \right) \phi_I + \sum_{i=I+1}^{I+n-1} \Sigma_{a_i} \phi_i \Delta_i + \Sigma_{a_{I+n}} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right) \phi_{I+n}}{\phi_G} \quad (36)$$

Fission

$$\bar{\Sigma}_{f_G} = \frac{\Sigma_{f_I} \left(\frac{E_{I+1} - E_I}{2} \right) \phi_I + \sum_{i=I+1}^{I+n-1} \Sigma_{f_i} \phi_i \Delta_i + \Sigma_{f_{I+n}} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right) \phi_{I+n}}{\phi_G} \quad (37)$$

Nu * fission (number of neutrons per fission times the fission cross section)

$$\overline{\nu \Sigma}_{f_G} = \frac{\nu \Sigma_{f_I} \left(\frac{E_{I+1} - E_I}{2} \right) \phi_I + \sum_{i=I+1}^{I+n-1} \nu \Sigma_{f_i} \phi_i \Delta_i + \nu \Sigma_{f_{I+n}} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right) \phi_{I+n}}{\phi_G} \quad (38)$$

Average Number of neutrons produced per fission

$$\nu_G = \frac{\overline{\nu \Sigma}_{f_G}}{\bar{\Sigma}_{f_G}}$$

Capture

$$\bar{\Sigma}_{c_G} = \bar{\Sigma}_{a_G} - \bar{\Sigma}_{f_G}$$

The average cross section for isotropic transfer from broad group G which lies between the energy point I and I+n to broad group G' which lies between the energy points J and J+m

$$\bar{\Sigma}_{s0_{G \rightarrow G'}} = \bar{\Sigma}_{s0_{G \rightarrow J}} + \bar{\Sigma}_{s0_{G \rightarrow \left(\begin{smallmatrix} J+1 \text{ to } J+m-1 \\ \text{Inclusive} \end{smallmatrix} \right)}} + \bar{\Sigma}_{s0_{G \rightarrow J+m}} \quad (39)$$

where

$$\begin{aligned} \bar{\Sigma}_{s0_{G \rightarrow J}} &= \frac{\Sigma_{s0_{I \rightarrow J}} \phi_I \left(\frac{E_{I+1} - E_I}{2} \right) \left(\frac{E_{J+1} - E_J}{2} \right)}{\phi_G} \\ &+ \frac{\Sigma_{s0_{I+n \rightarrow J}} \phi_{I+n} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right) \left(\frac{E_{J+1} - E_J}{2} \right)}{\phi_G} \\ &+ \frac{\sum_{i=I+1}^{I+n-1} \Sigma_{s0_{i \rightarrow J}} \phi_i \Delta_i \left(\frac{E_{J+1} - E_J}{2} \right)}{\phi_G} \\ \bar{\Sigma}_{s0_{G \rightarrow \left(\begin{smallmatrix} J+1 \text{ to } J+m-1 \\ \text{Inclusive} \end{smallmatrix} \right)}} &= \frac{\sum_{j=J+1}^{J+m-1} \Sigma_{s0_{I \rightarrow j}} \phi_I \Delta_j \left(\frac{E_{I+1} - E_I}{2} \right)}{\phi_G} \\ &+ \frac{\sum_{i=I+1}^{I+n-1} \sum_{j=J+1}^{J+m-1} \Sigma_{s0_{i \rightarrow j}} \phi_i \Delta_j \Delta_i + \sum_{j=J+1}^{J+m-1} \Sigma_{s0_{I+n \rightarrow j}} \phi_{I+n} \Delta_j \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right)}{\phi_G} \end{aligned}$$

and

$$\begin{aligned}
\bar{\Sigma}_{s0_{G \rightarrow J+m}} = & \frac{\Sigma_{s0_{I \rightarrow J+m}} \phi_I \left(\frac{E_{I+1} - E_I}{2} \right) \left(\frac{E_{J+m} - E_{J+m-1}}{2} \right)}{\phi_G} \\
& + \frac{\sum_{i=I+1}^{I+n-1} \Sigma_{s0_{i \rightarrow J+m}} \phi_i \Delta_i \left(\frac{E_{J+m} - E_{J+m-1}}{2} \right)}{\phi_G} \\
& + \frac{\Sigma_{s0_{I+n \rightarrow J+m}} \phi_{I+n} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right) \left(\frac{E_{J+m} - E_{J+m-1}}{2} \right)}{\phi_G}
\end{aligned}$$

The three terms in Equation (39) above represent, respectively, the scattering from all fine energy points I, weighted by their proper energy widths within G, to (a) energy point J, weighted by its proper partial energy width within G'; (b) energy points J+1 through J+m-1, weighted by their proper energy widths within G'; and (c) energy point J+m, weighted by its proper partial energy width within G'.

Equation (39) is also used to compute the average linearly anisotropic transfer cross section between broad groups after the following substitutions:

$$\begin{aligned}
\bar{\Sigma}_{s0_{G \rightarrow G'}} & \rightarrow \bar{\Sigma}_{s1_{G \rightarrow G'}} \\
\phi_I & \rightarrow J_I \\
\phi_G & \rightarrow J_G \\
\Sigma_{s0_{I \rightarrow J}} & \rightarrow \Sigma_{s1_{I \rightarrow J}}
\end{aligned} \tag{40}$$

Average scattering cross section from group G to above E_c

$$\bar{\Sigma}_{s0_{G \rightarrow x}} = \frac{\Sigma_{s0_{I \rightarrow x}} \phi_I \left(\frac{E_{I+1} - E_I}{2} \right) + \sum_{i=I+1}^{I+n-1} \Sigma_{s0_{i \rightarrow x}} \phi_i \Delta_i + \Sigma_{s0_{I+n \rightarrow x}} \phi_{I+n} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right)}{\phi_G} \tag{41}$$

where the same substitutions apply for the linearly anisotropic outscattering averages as are given in Equation (40).

Here the outscattering cross section $\Sigma_{s0_{I \rightarrow x}}$ is obtained from the step previous to Equation (32B) where the I^{th} group outscatter cross section is defined as

$$\Sigma_{s0_{I \rightarrow x}} = \int_{E_c}^{\infty} \Sigma_{s0}(E_I \rightarrow E'') dE'' = \Sigma_{s0_I} - \sum_{J=I}^{\text{NOG}} \Sigma_{s0_{I \rightarrow J}} \Delta_J \tag{42}$$

where

$$\Sigma_{s0_I} = \sum_{k=1}^{NOI} N_k \sigma_{s0_I}^k \quad \text{and} \quad \Sigma_{s0_{I \rightarrow J}} = \sum_{k=1}^{NOI} N_k \sigma_{s0_{I \rightarrow J}}^k.$$

The quantity $\sigma_{s0_I}^k$ is the total scattering cross section and $\sigma_{s0_{I \rightarrow J}}^k$ is the differential scattering cross section for each material k obtained from the cross section library.

Total Scattering Cross Section for Broad Group G

$$\bar{\Sigma}_{T_{s0_G}} = \sum_{G'=1}^{NOBG} \bar{\Sigma}_{s0_{G \rightarrow G'}} + \bar{\Sigma}_{s0_{G \rightarrow x}} \quad (43A)$$

$$\bar{\Sigma}_{T_{sl_G}} = \sum_{G'=1}^{NOBG} \bar{\Sigma}_{sl_{G \rightarrow G'}} + \bar{\Sigma}_{sl_{G \rightarrow x}} \quad (43B)$$

Total Cross Section for Broad Group G

$$\bar{\Sigma}_{T_G} = \bar{\Sigma}_{T_{s0_G}} + \bar{\Sigma}_{a_G}$$

Removal Cross Section From Broad Group G

$$\bar{\Sigma}_{R_G} = \sum_{\substack{G'=1 \\ G' \neq G}}^{NOBG} \bar{\Sigma}_{s0_{G \rightarrow G'}} + \bar{\Sigma}_{a_G} \quad (44)$$

Integrated Sources for Group G

$$S_{0_G} = S_{0_I} \left(\frac{E_{I+1} - E_I}{2} \right) + \sum_{i=I+1}^{I+n-1} S_{0_i} \Delta_i + S_{0_{I+n}} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right) \quad (45A)$$

$$S_{1_G} = S_{1_I} \left(\frac{E_{I+1} - E_I}{2} \right) + \sum_{i=I+1}^{I+n-1} S_{1_i} \Delta_i + S_{1_{I+n}} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right) \quad (45B)$$

Diffusion Coefficient for Broad Group G

$$\bar{D}_G = \frac{J_I \left(\frac{E_{I+1} - E_I}{2} \right) + \sum_{i=I+1}^{I+n-1} J_i \Delta_i + J_{I+n} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right)}{B \phi_I \left(\frac{E_{I+1} - E_I}{2} \right) + \sum_{i=I+1}^{I+n-1} B \phi_i \Delta_i + B \phi_{I+n} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right)} \quad (46)$$

Transport Cross Section for Group G

$$\bar{\Sigma}_{TRG} = \frac{1}{3 \bar{D}_G}$$

2. DERIVATION OF DIFFUSION COEFFICIENT AND TRANSPORT CROSS SECTION

Although it is not evident from a cursory examination of the B-1 equations, the mathematical structure of the system of equations resulting in Equations (16) and (17) is predetermined by the assumptions embodied in Equation (3). These assumptions, mathematically valid for a bare critical system satisfying simple diffusion boundary conditions, are characteristic of the broad generalities leading to the so-called First Fundamental Theorem of Reactor Theory [6]. The usual approach to the B-N energy-dependent equations involves taking the Fourier transform of the steady state transport equation for a slab with the implication that diffusion boundary conditions are somehow applicable to any order N.

The conceptual difficulty with this approach is that the resultant equations obscure any direct relationship between the spatial and energy dependence of the variables unless the resultant equations are first subjected to Fourier inversion. Such a relationship is required in order to derive a rigorous definition of the energy-dependent diffusion coefficient by means of Fick's Law. The procedure of deriving the B-1 equations by use of Equation (3) and diffusion theory boundary conditions for the scalar flux enables one to accomplish this end in a straightforward way.

Equation (3) establishes the desired relationship between the variables subject to the boundary conditions at the extrapolated boundary of the slab. Only the real part of e^{-iBZ} satisfies the boundary conditions for the spatially dependent scalar flux. The scalar flux is obtained from Equation (3) in the conventional manner.

$$\psi(Z, E, \mu) = \phi(E, \mu) e^{-iBZ}$$

$$\psi_0(Z, E) = \int_{-1}^1 \psi(Z, E, \mu) d\mu = e^{-iBZ} \int_{-1}^1 \phi(E, \mu) d\mu \quad (47)$$

From Equation (7) we have for the first two terms of the angular flux in the B-1 approximation.

$$\phi(E, \mu) = \frac{1}{2} F_0(E) + i \frac{3}{2} \mu F_1(E)$$

$$\begin{aligned} \therefore \psi_0(Z, E) &= e^{-iBZ} \left[\int_{-1}^1 \frac{1}{2} F_0(E) d\mu + i \frac{3}{2} \int_{-1}^1 \mu F_1(E) d\mu \right] \\ &= e^{-iBZ} F_0(E) \end{aligned} \quad (48)$$

Since only the real part of $\psi_0(Z, E)$ satisfies the boundary conditions for the scalar flux at the extrapolated boundary of the slab,

$$\text{Re} \left[\psi_0(Z, E) \right] = \cos BZ F_0(E). \quad (49)$$

From Fick's Law we obtain the relationship between the scalar flux and current,

$$\begin{aligned} \vec{J}(Z, E) &= -D(E) \vec{\nabla} \text{Re} \left[\psi_0(Z, E) \right] = -\vec{k} D(E) \frac{d}{dZ} \text{Re} \left[\psi_0(Z, E) \right] \\ &= D(E) B F_0(E) \left[\vec{k} \sin BZ \right] \end{aligned} \quad (50)$$

where $\vec{k}$ is the unit vector in the Z direction.

Therefore, $\vec{J}(Z, E)$ is also separable in space and energy, a result consistent with Equation (3).

$$\therefore \vec{J}(Z, E) = J(E) \left[\vec{k} \sin BZ \right]$$

where

$$J(E) = D(E) B F_0(E). \quad (51)$$

It has been pointed out that Equation (51) does not give a clearcut relationship between $J(E)$ and $F_1(E)$ in Equations (16) and (17) since Fick's Law is perfectly general. An alternate method of obtaining Equation (51) will be given which helps clear up any ambiguity which may exist. Using the same assumptions by which Equations (16) and (17) are derived we have

$$\psi(Z, E, \mu) = \phi(E, \mu) e^{-iBZ}$$

where

$$\phi(E, \mu) = \frac{1}{2} F_0(E) + i \frac{3}{2} \mu F_1(E)$$

$$\psi_0(Z, E) = \int_{-1}^1 \psi(Z, E, \mu) d\mu = F_0(E) e^{-iBZ}$$

and

$$\vec{J}^*(Z, E) = \vec{k} \int_{-1}^1 \mu \psi(Z, E, \mu) d\mu = i \vec{k} F_1(E) e^{-iBZ}.$$

Introducing Fick's Law without reference to the scalar flux boundary conditions we have

$$\vec{J}^*(Z, E) = -D(E) \vec{\nabla} \psi_0(Z, E) = i \vec{k} D(E) B F_0(E) e^{-iBZ}.$$

Now equate $\vec{J}^*(Z, E)$ from the two preceding equations to obtain a result identical to Equation (51).

$$\vec{J}^*(Z, E) = i \vec{k} F_1(E) e^{-iBZ} = i \vec{k} D(E) B F_0(E) e^{-iBZ}$$

$$\therefore J(E) = F_1(E) = D(E) B F_0(E)$$

The same relationship can be implied by assigning physical insight to Equation (17), the continuity equation for the energy-dependent neutron current [see equations (56) and (57)].

From Equation (51), $D(E)$ is defined as

$$D(E) = \frac{F_1(E)}{B F_0(E)} \quad (52)$$

or at each mesh point,

$$D_I = \frac{J_I}{B\phi_I} \quad (53)$$

The broad group flux and currents are defined as the integral values of $F_1(E)$ and $F_0(E)$ over the appropriate energy range,

$$\bar{D}_G = \frac{\int_G F_1(E) dE}{B \int_G F_0(E) dE}$$

from which Equation (46) follows. Equations (46), (52), and (53) give an unambiguous definition of the diffusion coefficient within the framework of the B-1 equations for a bare slab. From Equation (51), the definition of the B-1 transport cross section at energy E is

$$\Sigma_{tr}(E) = \frac{1}{3 D(E)} = \frac{B}{3} \frac{F_0(E)}{F_1(E)} \quad (54)$$

and for a broad group G is

$$\bar{\Sigma}_{trG} = \frac{1}{3 \bar{D}_G} = \frac{B}{3} \frac{\phi_G}{J_G} \quad (55)$$

In order to use Equations (46) and (55), it is necessary that the current from the solution run be nonzero. Thus, when a zero buckling is input to the code, neither the diffusion coefficient nor the transport cross section is calculated. This situation corresponds to an infinite medium problem for which there is no net neutron current flow. If a diffusion coefficient is desired for a near infinite medium situation, then a small but finite buckling input to the code will produce the desired results.

If a cell calculation is requested, ie, if another material is to be averaged over a spectrum calculated in a previous problem, then the transport cross section must be defined in terms of the cross sections of the material to be averaged, since the flux and current from the spectral problem are not characteristic of this material. In this case, the desired relationship is obtained from Equation (54) using Equation (17). Isolating $F_0(E)$ in Equation (17) we obtain

$$\frac{B}{3} F_0(E) = F_1(E) \left[\gamma(E) \Sigma_t(E) - \frac{1}{F_1(E)} \int_{E'} \Sigma_{s1}(E' \rightarrow E) F_1(E') dE' - \frac{S_1(E)}{F_1(E)} \right] \quad (56)$$

and then

$$\Sigma_{tr}(E) = \frac{B F_0(E)}{3 F_1(E)} = \gamma(E) \Sigma_t(E) - \frac{1}{F_1(E)} \int_{E'} \Sigma_{s1}(E' \rightarrow E) F_1(E') dE' - \frac{S_1(E)}{F_1(E)} \quad (57)$$

Equation (57) is an identity consistent with Fick's Law expressing the fact that the transport cross section for any "finite system" based on the transport equation cannot be expressed in terms of elementary cross sections alone, but requires the linearly anisotropic neutron currents also, which result from the finite boundary conditions.

The form of the r.h.s. of Equation (57) is used for obtaining the transport cross section for the material in the cell calculation. In this case, the current spectra from the spectral problem is used for averaging $\Sigma_t(E)$ as well as to obtain the integral term in Equation (57). $\gamma(E)$ is also obtained from the spectral problem in order to be consistent with the use of those currents. For a broad group G, the cell transport cross section is

$$\bar{\Sigma}_{trG}^* = \frac{\gamma_1 \Sigma_{tI}^* J_I \left(\frac{E_{I+1} - E_I}{2} \right) + \sum_{i=I+1}^{I+n-1} \gamma_i \Sigma_{ti}^* J_i \Delta_i + \gamma_{I+n} \Sigma_{tI+n}^* J_{I+n} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right)}{J_G} - \frac{\sum_{G'=1}^{NOBG} \Sigma_{s1G' \rightarrow G}^* J_{G'}}{J_G} - \frac{S_1^*}{J_G} \quad (58)$$

and

$$\bar{D}_{G_{cell}} = \frac{1}{3 \bar{\Sigma}_{trG_{cell}}^*} \quad (59)$$

where all quantities with the superscript * refer to the cell material including the source term S_1^* , which is recalculated for the cell material using Equation (24). In this way, if the original material from the spectral problem is used in Equations (58) and (59), an identity is established between Equations (59) and (46), which is equivalent to the identity expressed in Equation (57) for a single energy point E. S_1^* is required since in the limit for a cell material with zero cross section, $\bar{\Sigma}_{tr}^*$, must also approach zero. The source term from the spectral problem would not be consistent with this limit.

3. AVERAGE MICROSCOPIC CROSS SECTIONS

The equations used for calculating the average, broad group, microscopic cross sections are identical to equations (36) through (44) with σ replacing Σ .

The microscopic source terms used in Equations (45) for the fine groups I are obtained from Equations (21) and (24) with N_k in the numerator set to 1.0. Because N_k is given in units of 10^{24} atoms/cc, the microscopic source terms calculated by Equations (45) can be used validly only in cases where the source is summed for a macroscopic value as in Equation (58). The microscopic source as defined above has no physical meaning other than in units of 10^{24} atoms per cc.

A unique definition of a microscopic transport cross section presents a conceptual difficulty in the same way that a microscopic diffusion coefficient would. The macroscopic values are extensive quantities resulting from some form of Fick's Law, and there is no corresponding physically meaningful relationship for microscopic values. Rather, it has become traditional to define a microscopic transport cross section only in the metaphysical sense that "the whole is the sum of its assumed parts"; then from this, the macroscopic diffusion coefficient follows:

$$\bar{D}_G = \frac{1}{3 \sum_k N_k \bar{\sigma}_{tr,G}^k} = \frac{1}{3 \bar{\Sigma}_{tr,G}} \quad (60)$$

Since this is the case, the only thing required to define the microscopic transport cross section $\bar{\sigma}_{tr,G}^k$ in Equation (60) is a consistency relationship which will yield the proper macroscopic transport cross section for a particular mixture of materials as calculated by Equation (55). Any number of such consistency relationships could be defined and applied as long as the macroscopic transport cross section in the denominator of Equation (60), computed from microscopic values, gives the same result as Equation (55).

One such equation which will yield the desired results is Equation (58) where Σ^* becomes the appropriate σ^k in each term of Equation (58). The microscopic transport cross section for a broad group is then given by

$$\bar{\sigma}_{tr,G}^k = \frac{\gamma_I \sigma_{T,I}^k J_I \left(\frac{E_{I+1} - E_I}{2} \right) + \sum_{i=I+1}^{I+n-1} \gamma_i \sigma_{T,i}^k J_i \Delta_i + \gamma_{I+n} \sigma_{T,I+n}^k J_{I+n} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right)}{J_G} - \frac{\sum_{G'=1}^{NOBG} \sigma_{s1,G' \rightarrow G} J_{G'}}{J_G} - \frac{S_{1C}^k}{J_G} \quad (61)$$

where the S_{1G}^k 's are the microscopic source terms for material k and broad group G defined above. Multiplying by N_k and summing over k will yield the same macroscopic transport cross section as that calculated by Equation (55). Equation (61) is only applied to those materials which possess a scattering matrix on the library tape, and it is only these materials for which a source

is calculated. For nonmoderating materials which do not possess a scattering matrix, the microscopic transport cross section is calculated from

$$\sigma_{tr_G}^k = \frac{\gamma_I \sigma_{T_I}^k J_I \left(\frac{E_{I+1} - E_I}{2} \right) + \sum_{i=I+1}^{I+n-1} \gamma_i \sigma_{T_i}^k J_i \Delta_i + \gamma_{I+n} \sigma_{T_{I+n}}^k J_{I+n} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right)}{J_G}$$

$$- \frac{\sigma_{sl_I}^k J_I \left(\frac{E_{I+1} - E_I}{2} \right) + \sum_{i=I+1}^{I+n-1} \sigma_{sl_i}^k J_i \Delta_i + \sigma_{sl_{I+n}}^k J_{I+n} \left(\frac{E_{I+n} - E_{I+n-1}}{2} \right)}{J_G} \quad (62)$$

where $\sigma_{T_I}^k = \sigma_{a_I}^k + \sigma_{s0_I}^k$. The parameters $\sigma_{a_I}^k$, $\sigma_{s0_I}^k$, and $\sigma_{sl_I}^k$ are the total cross sections obtained directly from the library for material k.

A measure of the "arbitrariness" of the definition of the microscopic transport cross section is that it is entirely possible that the microscopic transport cross section as given by Equation (61) can be negative for one or several materials in the mixture and yet when multiplied by N_k and summed over all materials in the mixture will yield the proper macroscopic transport cross section as given by Equation (55). This situation occurs when the mixture contains large amounts of a strong absorber and the resultant flux and current spectra are very hard. For moderating materials with a small absorption cross section, the second term in Equation (61) can then be the leading term due to the high current weights given the transfer cross sections $\sigma_{sl_{G' \rightarrow G}}^k$

when G lies in the energy range of depressed flux and current at low energy. The resultant microscopic transport cross sections, even though negative, are consistent in the sense of Equation (60) since "consistency" is all that is required of them in the definition.

The same pathologic situation can occur for cell materials being averaged over the flux and current from another spectral problem. If the spectral problem contained a very strong absorber producing a very hard spectra, and the cell material does not contain much absorption, then the macroscopic transport cross section from Equation (58) can be negative in low-energy groups. In this case, the negative result could be interpreted as indicating that the homogenization for the cell material was not very judiciously chosen.

4. EFFECTIVE DIFFUSION CONSTANTS (BLACKNESS THEORY)

Considerable use is made of diffusion theory in reactor design calculations; however, inherent in the diffusion theory approximation is its inadequacy in regions of high absorption and low scattering, ie, such as control rods. Since many reactor design calculations involve such regions, "Blackness Theory" has been developed to calculate effective diffusion theory constants.

These constants are defined such that when they are utilized in the solution of the diffusion equation across a thin, highly absorbing region, the result is a transport solution which matches, at the boundaries of the region, the solution of the diffusion equation in the surrounding regions. This solution assumes the surrounding regions can be adequately handled by diffusion theory.

This method allows the explicit representation of a highly absorbing region as a fictitious diffusion medium in the diffusion theory calculation. The equations incorporate the effect of the mesh spacing used in describing this region in a finite difference solution to the diffusion equation.

A complete development of this theory can be found in References 7, 8, 9, 10, and 11.

V. INCITE INPUT PREPARATION

The following input formats will be encountered:

- (1) Alphanumeric identification card
- (2) Integer data cards
- (3) Floating point data cards.

The type of input described in this section facilitates the running of a series of consecutive problems which have similar input. Successive problems are run by following the preceding problem with an identification card and at least one card containing integer data. It is only necessary to specify those data which are to be changed; however, caution must be used since all the data input to the preceding problem will be retained and used in the following problem unless it is either replaced by new input or set to zero.

One alphanumeric title card is required at the beginning of all problems (only one is permitted) and must be followed immediately by at least one integer data card. The last card containing integer data may or may not be followed by a floating point data card. In all cases, the integer data cards for any particular problem may be arranged in any order, but they must follow the title card and precede all the floating point data cards. The last card containing integer data must be denoted by a 1 or 2 in column 1. The floating point data cards for any particular problem may also be in any order; however, they must follow the last integer data card of the problem, and the last card containing floating point data must be denoted by a 1 in column 1.

1. ALPHANUMERIC IDENTIFICATION CARD

The information on this card is used for problem identification and is printed at the beginning of the problem output. With the exception of the first identification card of a series of problems, any valid numeric, alphabetic, or special characters may be entered in columns 1 through 72. Columns 1 through 6 on the identification card of the first problem of a set of problems must contain the library tape identification number.

By placing a 1 in column 80 of the identification card, all previous input data can be restored to zero. This option is very convenient if the user desires to perform a calculation which is independent of all previous calculations in a series of problems.

2. INTEGER DATA

The information on these cards is read-in using a (I1, 2X, I5, 2X, I2, 15I4) format where columns 1 through 12 contain information identifying the data on the card and columns 13 through 72 contain up to 15 words of integer data in 4-column fields. All integer data words must be positioned as far right as possible in their proper fields.

The following paragraph describes the information that must be specified in columns 1 through 12 of every card containing integer data:

Column 1		If this column is left blank or set to 0, then more cards containing integer data are to follow. If a 1 is placed in this column it signifies that this is the last card containing integer data and at least one card containing floating point data will follow. If a 2 is placed in this column it indicates that this is the last card containing integer data and there are no cards containing floating point data.
Columns 4-5	Index 1	These indices identify the type of integer data to be found on this card.
Columns 6-8	Index 2	
Columns 11-12		These columns specify the number of words of data on the card. If left blank, the program assumes 15 words of data.
Columns 13-16		These 4 column fields contain the integer data in consecutive order. Consecutive words of integer data on the card will be identified by consecutive values of the Index 2. Up to 15 words of data are permitted per card.
17-20		
21-24		
25-28		
29-32		
33-36		
37-40		
41-44		
45-48		
49-52		
53-56		
57-60		
61-64		
65-68		
69-72		

The mnemonic abbreviation assigned to each integer data word, the identifying values of the indices, and an explanation of the word are found on the following pages.

Index 1 Columns 4-5	Index 2 Columns 6-8	Mnemonic Name	Meaning
0	1	NTYPE	Type of calculation to determine fluxes and currents 1 -- B-1 approximation 2 -- P-1 approximation 3 -- Maxwellian fluxes 4 -- Cell: fluxes, currents, and gammas from the preceding problem will be used for averaging cross sections 5 -- Cell: read in fluxes, currents, and gammas to be used for averaging cross sections 6 -- Blackness: fluxes from the preceding problem will be used to obtain effective diffusion theory cross sections 7 -- Blackness: read in fluxes to use in obtaining effective diffusion -- theory cross sections
	2	NGESOP	Flux guess option 1 -- Maxwellian flux: current guess is the product of $1/(3(1-\mu)\Sigma_s)$ *Buckling*Flux Guess 2 -- Converged flux and current from preceding problem 3 -- Constant flux of unity: current guess is the product of $1/(3(1-\mu)\Sigma_s)$ *Buckling*1.0 4 -- Input flux and current

<u>Index 1</u> <u>Columns 4-5</u>	<u>Index 2</u> <u>Columns 6-8</u>	<u>Mnemonic</u> <u>Name</u>	<u>Meaning</u>
0	3	ISOPT	Isotropic source option 0 -- 1/E slowing down source from library 1 -- Input on cards 2 -- All set to 0.0
	4	IANOF	Anisotropic source option 0 -- 1/E slowing down source from library 1 -- Input on cards 2 -- All set to 0.0
	5	NOI	Number of materials in the spectrum, cell, or blackness calculation ($1 \leq \text{NOI} \leq 20$)
	6	NOBG	Number of broad groups ($1 \leq \text{NOBG} \leq 50$)
	7	IEND2	Highest energy point to be used as the upper limit of the one broad group calculation (if not specified or zero, code sets IEND2=54)

<u>Index 1</u> <u>Columns 4-5</u>	<u>Index 2</u> <u>Columns 6-8</u>	<u>Mnemonic</u> <u>Name</u>	<u>Meaning</u>
0	8	MAXIT	Maximum number of iterations (if not specified or zero, code sets MAXIT=30)
	9	IUPSCT	Iteration options 0 -- MAXIT iterations in the downscatter direction N -- N iterations in the downscatter direction for every iteration in the upscatter direction -N -- N iterations in the upscatter direction for every iteration in the downscatter direction
	10	KONVOP	Convergence option 0 -- Pointwise convergence on fluxes and currents 1 -- Convergence on renormalization factor
	11	MICR	Microscopic cross section option 0 -- No average microscopic data desired 1 -- Calculate average microscopic cross sections
	12	IEDIT	Editing option 0 -- Standard information 1 -- Fluxes and currents printed after each iteration

Index 1 <u>Columns 4-5</u>	Index 2 <u>Columns 6-8</u>	Mnemonic <u>Name</u>

1	I (broad group)	IEN(I

Meaning

Broad group energy structure

($1 \leq I \leq 50$)

The value I of Index 2 indicates that the first data word on the card specifies the energy point corresponding to the upper energy cutoff of broad group I. (In all cases, the upper limit of broad group number 1 must correspond to the number of the highest energy point on the library tape being used, ie, for group 1 the value 101 must be entered for a library with 101 fine groups.)

3. FLOATING POINT DATA

The information on these cards is read in using a (I1, 2I2, I3, 2X, I2, 5E12.8) format where columns 1 through 12 contain information identifying the data on the card and columns 13 through 72 contain up to five words of floating point data in 12-column fields.

Much the same type of information must be specified in columns 1 through 12 of the floating point data cards as was necessary for the cards containing integer data.

Column 1

If this column is left blank or set to 0, then more cards containing floating point data are to follow.

If a 1 is placed in this column, then it signifies that this is the last card containing floating point data. A title card for another problem may follow.

Columns 2-3 Index 1

Columns 4-5 Index 2

These indices identify the type of floating point data to be found on this card.

Columns 6-8 Index 3

Columns 11-12

These columns specify the number of words of data on the card. If left blank, the program assumes five words of data.

Columns 13-24
25-36
37-48
49-60
61-72

These 12 column fields contain the floating point data. Consecutive words of data on the card will be identified by consecutive values of the Index 3. Up to five words of data are permitted per card.

A few examples of acceptable ways of writing the floating point number +3.1415927 under a FORTRAN E12.8 format are as follows:

CARD COLUMNS

1	2	3	4	5	6	7	8	9	10	11	12
3	.	1	4	1	5	9	2	7			
	3	.	1	4	1	5	9	2	7		
+	3	.	1	4	1	5	9	2	7		
3	.	1	4	1	5	9	2	7		+	0
.	3	1	4	1	5	9	2	7		+	1
	.	3	1	4	1	5	9	2	7	+	1
3	.	1	4	1	5	9	2	7	E	+	0
	3	.	1	4	1	5	9	2	7	E	0
	3	1	4	1	5	9	2	7	E	+	1
3	1	4	1	5	9	2	7	E	+	0	1

The mnemonic abbreviation assigned to each word of floating point data, the identifying values of the indices, and an explanation of the word are found on the following pages.

<u>Index 1</u> <u>Columns 2-3</u>	<u>Index 2</u> <u>Columns 4-5</u>	<u>Index 3</u> <u>Columns 6-8</u>	<u>Mnemonic</u> <u>Name</u>	<u>Meaning</u>
0	0	1	TEMP	Temperature of Maxwellian flux to be used for averaging or as a flux guess (if not specified or zero, the code sets TEMP=293.0)
		2	CONV	Convergence criteria for fluxes and currents or for renormalization factor (if not specified or zero, the code sets CONV=0.00001)
		3	OMEGA	Extrapolation factor (if not specified or zero, the code sets OMEGA=1.0)
		4	BUCK	Buckling (B^2 , values must be positive)
		5		Presently not used
		6	TH	Thickness of highly absorbing region, in centimeters, for which a "Blackness theory" calculation of effective diffusion constants is desired
		7	H	Width of mesh, in centimeters, used to describe the interior of a highly absorbing region in a diffusion theory calculation

Index 1 Columns 2-3	Index 2 Columns 4-5	Index 3 Columns 6-8	Mnemonic Name

0

0

8

C

Meaning

A quantity occurring in the blackness theory equations which can be approximated by $\sqrt{3} \cdot \sqrt{\frac{\Sigma_{tr}}{\Sigma_t}}$, where Σ_{tr} and Σ_t represent the transport and total cross sections, respectively, for the regions immediately surrounding the blackness region. The ratio $\frac{\Sigma_{tr}}{\Sigma_t}$ is usually assumed to be unity and C is set to $\sqrt{3}$.

9

AK1

First dirty blackness coefficient
(usually set to 1.0)

10

AK2

Second dirty blackness coefficient
(usually set to 1.0)

1

I
(material)

ISC(I)

Material identification number
($1 \leq I \leq 20$)

The value I of Index 3 indicates that the first data field on the card contains the identification number for the Ith material.

The material identification numbers must always be ordered so that ascending values of I correspond to ascending values of the material identification number.

<u>Index 1</u> <u>Columns 2-3</u>	<u>Index 2</u> <u>Columns 4-5</u>	<u>Index 3</u> <u>Columns 6-8</u>	<u>Mnemonic</u> <u>Name</u>	<u>Meaning</u>
0	2	I (material)	DENS(I)	Atom density $\times 10^{-24}$ of the I^{th} material ($1 \leq I \leq 20$) The value I of Index 3 indicates that the first data field on the card contains the atom density for the I^{th} material
	3	I (material)	SSFF(I)	Self-shielding factor flag ($1 \leq I \leq 20$) 0.0 -- no self-shielding 1.0 -- self-shielding factors input by broad group 2.0 -- self-shielding factors input by fine group The value I of Index 3 indicates that the first data field on the card contains the self-shielding factor flag for the I^{th} material
1	I (material)	J (fine energy point or broad group)	SSF(I,J)	Self-shielding factors ($1 \leq I \leq 20$) , ($1 \leq J \leq 101$) These data are necessary only for those materials that require self-shielding factors The value I of the Index 2 indicates that the data words on the card specify the self-shielding factors for the I^{th} material. The value J of the Index 3 indicates that the first

Index 1 Columns 2-3	Index 2 Columns 4-5	Index 3 Columns 6-8	Mnemonic Name

Meaning

data field on the card contains the self-shielding factor
for the fine energy point J or the broad group J

2	1	$\left(\begin{array}{c} I \\ \text{fine energy} \\ \text{points} \end{array} \right)$	PHI(I)

Input fluxes

$(1 \leq I \leq 101)$

These data are required only if the fluxes are to be read in
as a flux guess or for averaging purposes

The value I of Index 3 indicates that the first data field
on the card contains the input flux for fine energy
point I

2	2	$\left(\begin{array}{c} I \\ \text{fine energy} \\ \text{points} \end{array} \right)$	CURR(I)

Input currents

$(1 \leq I \leq 101)$

These data are required only if the currents are to be read in
as a current guess or for averaging purposes

The value I of Index 3 indicates that the first data field
on the card contains the input current for fine energy
point I

<u>Index 1</u> <u>Columns 2-3</u>	<u>Index 2</u> <u>Columns 4-5</u>	<u>Index 3</u> <u>Columns 6-8</u>	<u>Mnemonic</u> <u>Name</u>
2	3	$\begin{pmatrix} I \\ \text{fine energy} \\ \text{points} \end{pmatrix}$	GAMMA(I)

Meaning

Input gammas

$(1 \leq I \leq 101)$

These data are required only if values of the flux and current are read in for averaging purposes. Gamma refers to the quantity appearing in the definition of the cell average transport cross section.

If not read in, gamma will be set to 1.0 at all fine energy points

The value I of Index 3 indicates that the first data field on the card contains the input gamma for fine energy point I

2	4	$\begin{pmatrix} I \\ \text{fine energy} \\ \text{points} \end{pmatrix}$	S(I)

Isotropic sources

$(1 \leq I \leq 101)$

These data will be used unchanged in the spectrum calculation and are required only if isotropic sources are to be read in as input

The value I of Index 3 indicates that the first data field on the card contains the input source for the fine energy point I.

<u>Index 1</u> <u>Columns 2-3</u>	<u>Index 2</u> <u>Columns 4-5</u>	<u>Index 3</u> <u>Columns 6-8</u>	<u>Mnemonic</u> <u>Name</u>	<u>Meaning</u>
2	5	(I fine energy points)	SAN(I)	Anisotropic sources (1 ≤ I ≤ 101)

These data will be used unchanged in the spectrum calculation
and are required only if anisotropic sources are to be
read in as input

The value I of Index 3 indicates that the first data field
on the card contains the input anisotropic source for
fine energy point I

4. CONDENSED DESCRIPTION OF INCITE INPUT

This section contains complete listing of the input data in a condensed form. This should facilitate the use of this code by those already familiar with the input. Those individuals inexperienced in the use of the code will find a detailed description of the input in Sections V-1, -2, and -3.

Alphanumeric Identification Card

Columns 1-6	Library tape identification number (necessary only on first problem of a series of problems)
Columns 7-72	Numeric, alphabetic or special characters (if not the first title card of a series of problems then columns 1-72 may be used)
Column 80	A nonzero value causes all previous input data to be set to zero

Integer Data

<u>Index 1</u> <u>Columns 2-3</u>	<u>Index 2</u> <u>Columns 4-5</u>	<u>Mnemonic</u> <u>Name</u>	<u>Meaning</u>
0	1	NTYPE	Type of calculation: 1-B1, 2-P1, 3-Maxwellian, 4-cell, 5-cell (input spectra), 6-blackness, 7-blackness (input spectra)
	2	NGESOP	Flux guess option: 1-Maxwellian, 2-from preceding problem, 3-constant, 4-input
	3	ISOPT	Isotropic source option: 0-1/E, 1-input, 2-set to 0.0
	4	IANOP	Anisotropic source option: 0-1/E, 1-input, 2-set to 0.0
	5	NOI	Number of materials
	6	NOBG	Number of broad groups
	7	IEND2	Upper limit of one broad group
	8	MAXIT	Maximum number of iterations
	9	IUPSCT	Iteration options: 0-all in downscatter direction, N-N down for each up, -N-N up for each down
	10	KONVOP	Convergence option: 0-pointwise, 1-renormalization factor
	11	MICR	Microscopic cross section option: 0-No, 1-yes
	12	IEDIT	Editing option: 0-standard, 1-extra information
1	I	IEN(I)	Upper energy point for broad group I

Floating Point Data

<u>Index 1</u> <u>Columns 2-3</u>	<u>Index 2</u> <u>Columns 4-5</u>	<u>Index 3</u> <u>Columns 6-8</u>	<u>Mnemonic</u> <u>Name</u>	<u>Meaning</u>
0	0	1	TEMP	Temperature of Maxwellian flux
		2	CONV	Convergence criteria
		3	OMEGA	Extrapolation factor
		4	BUCK	Buckling
		5		Presently not used
		6	TH	Thickness in blackness calculation
		7	H	Mesh width in blackness calculation
		8	C	Blackness theory quantity
		9	AK1	First dirty blackness coefficient
		10	AK2	Second dirty blackness coefficient
	1	I	ISO(I)	Identification number of I th material
	2	I	DENS(I)	Atom density of I th material
	3	I	SSFF(I)	Self-shielding flag for I th material
1	I	J	SSF(I,J)	Self-shielding factor for I th material in J th fine or broad group
2	1	I	PHI(I)	Input flux at energy point I
2	2	I	CURR(I)	Input current at energy point I
2	3	I	GAMMA(I)	Input gamma at energy point I
2	4	I	S(I)	Input isotropic sources at energy point I
2	5	I	SAN(I)	Input anisotropic sources an energy point I

1. INPUT

[illegible]

BF - 0001

INCITE SAMPLE PROBLEM

LIBRARY TAPE NO, 112166 WAS REQUESTED AND TAPE NO, 112166 WAS USED

INPUT DATA

PROBLEM TYPE ***** 1
 NO. OF BROAD GROUPS ** 6
 NO. OF ISOTOPES ***** 8
 ISO, SOURCE OPTION *** 0
 ANISO, SOURCE OPTION * 0
 FLUX GUESS OPTION **** 1
 BUCKLING (B**2) ***** 0,42000E-02
 CONVERGENCE OPTION *** 0
 MAX, NO. OF ITERATIONS 20
 CALC, OPTION (IUPSCT) 0
 IEDIT ***** 0
 TEMPERATURE ***** 0,29300E 03
 CONVERGENCE CRITERION 0,10000E-04
 OMEGA ***** 0,11000E 01

ISOTOPE ID. NO. DENSITY SELF SHIELDING OPTION

1,00293 0,1674100E-01 1,0

BROAD GROUP
 SELF SHIELDING FACTORS

GROUP	1	2	3	4	5	6
	1,0008700 E 00	1,0208700 E 00	1,0287700 E 00	1,0609900 E 00	1,1219100 E 00	1,2598200 E 00

8,00303 0,1897000E-01 1,0

BROAD GROUP
 SELF SHIELDING FACTORS

GROUP	1	2	3	4	5	6
	9,8935000 E-01	9,7427000 E-01	9,6454000 E-01	9,2476000 E-01	8,5191000 E-01	6,8945000 E-01

24,00000 0,1397000E-02 1,0

BROAD GROUP
 SELF SHIELDING FACTORS

GROUP	1	2	3	4	5	6
	9,9897000 E-01	9,9984000 E-01	1,0001900 E 00	1,0008300 E 00	9,9080000 E-01	9,5691000 E-01

2. OUTPUT

		25.00000		0.1408000E-03		1.0
				BROAD GROUP		
				SELF SHIELDING FACTORS		
GROUP	1	2	3	4	5	6
	9,9897000 E-01	9,9984000 E-01	1,0001900 E 00	1,0008300 E 10	9,9080000 E-01	9,5691000 E-01
		26.00000		0.5003000E-02		1.0
				BROAD GROUP		
				SELF SHIELDING FACTORS		
GROUP	1	2	3	4	5	6
	9,9897000 E-01	9,9984000 E-01	1,0001900 E 00	1,0008300 E 10	9,9080000 E-01	9,5691000 E-01
		28.00000		0.7042000E-03		1.0
				BROAD GROUP		
				SELF SHIELDING FACTORS		
GROUP	1	2	3	4	5	6
	9,9897000 E-01	9,9984000 E-01	1,0001900 E 00	1,0008300 E 10	9,9080000 E-01	9,5691000 E-01
		92.23500		0.4607000E-03		1.0
				BROAD GROUP		
				SELF SHIELDING FACTORS		
GROUP	1	2	3	4	5	6
	9,8935000 E-01	9,7427000 E-01	9,6454000 E-01	9,2476000 E-01	8,5191000 E-01	6,8945000 E-01
		92.23800		0.9024000E-02		1.0
				BROAD GROUP		
				SELF SHIELDING FACTORS		
GROUP	1	2	3	4	5	6
	9,8935000 E-01	9,7427000 E-01	9,6454000 E-01	9,2476000 E-01	8,5191000 E-01	6,8945000 E-01

MAXWELLIAN FLUX GUESS

3,8229953 E-01	7,3490852 E-01	1,3578829 E 00	1,6314429 E 00	2,1100802 E 00	2,3178772 E 00
2,6766988 E 00	3,2937329 E 00	3,6026721 E 00	3,6944751 E 00	3,6367316 E 00	3,2632116 E 00
2,7450515 E 00	2,2168025 E 00	1,9700930 E 00	1,7404800 E 00	1,5297632 E 00	1,3386167 E 00
1,1667634 E 00	1,0134539 E 00	8,7757193 E-01	7,5780373 E-01	4,1183998 E-01	2,1760345 E-01
1,1262849 E-01	5,7384056 E-02	2,8876146 E-02	1,4385417 E-02	1,0120987 E-02	7,1072488 E-03
4,9822496 E-03	3,4870175 E-03	2,4369117 E-03	1,7007062 E-03	1,1853989 E-03	8,2524531 E-04
5,7387685 E-04	3,9865965 E-04	2,7666960 E-04	1,9183227 E-04	1,3289420 E-04	9,1988969 E-05
4,3975151 E-05	1,2462580 E-05	9,9690580 E-06	6,8686041 E-06	3,2553901 E-06	2,2394618 E-06
1,5398516 E-06	1,2766511 E-06	1,0583213 E-06	7,2705556 E-07	4,9927143 E-07	1,4957646 E-07
7,5806368 E-08	2,9444151 E-08	1,6679234 E-08	1,1414861 E-08	4,4176103 E-09	1,7069030 E-09
4,8540089 E-10	2,5372882 E-10	3,7524007 E-11	5,5247702 E-12	8,1025096 E-13	2,9818948 E-13
1,7400897 E-13	8,0577428 E-14	3,7294566 E-14	1,7253476 E-14	7,9783913 E-15	5,4245643 E-15
3,6878172 E-15	2,5068548 E-15	9,5464378 E-16	3,6332466 E-16	2,4683506 E-16	1,6767949 E-16
1,1389774 E-16	7,7359430 E-17	3,5677953 E-17	1,9962842 E-17	1,6449246 E-17	7,5815149 E-18
1,0919828 E-18	1,9700755 E-19	2,2538762 E-20	3,2307018 E-21	9,7562466 E-23	9,4402375 E-24
1,9185107 E-25	3,8837023 E-27	1,7107203 E-28	7,5202784 E-30	1,5756341 E-30	3,1599785 E-32
6,3215884 E-34	1,2617752 E-35	0,	0,	0,	

CONVERGED FLUXES AND CURRENTS (9 ITERATIONS)

POINT	ENERGY	FLUX	CURRENT	E*FLUX	E*CURRENT	GAMMA
1	0.0010	0.18078974E 01	0.55905258E-02	0.18078973E-02	0.55905258E-05	0.10000238E 01
2	0.0020	0.35041761E 01	0.13275879E-01	0.70083519E-02	0.26551758E-04	0.10000342E 01
3	0.0040	0.67138254E 01	0.31063025E-01	0.26855301E-01	0.12425210E-03	0.10000493E 01
4	0.0050	0.81934904E 01	0.40056538E-01	0.40967451E-01	0.20028269E-03	0.10000551E 01
5	0.0070	0.10936265E 02	0.58926613E-01	0.76553856E-01	0.41248629E-03	0.10000644E 01
6	0.0080	0.12180252E 02	0.68239141E-01	0.97442013E-01	0.54591312E-03	0.10000674E 01
7	0.0100	0.14439221E 02	0.88641498E-01	0.14439221E 00	0.88641497E-03	0.10000708E 01
8	0.0150	0.19056156E 02	0.13897923E 00	0.28584234E 00	0.20846885E-02	0.10000854E 01
9	0.0200	0.22096045E 02	0.18064519E 00	0.44192088E 00	0.36129038E-02	0.10001040E 01
10	0.0253	0.23919875E 02	0.21699682E 00	0.60517283E 00	0.54900194E-02	0.10001224E 01
11	0.0300	0.26285473E 02	0.27289261E 00	0.78856418E 00	0.81867781E-02	0.10001396E 01
12	0.0400	0.26621225E 02	0.32918664E 00	0.10646490E 01	0.13167465E-01	0.10001610E 01
13	0.0500	0.25333311E 02	0.35496996E 00	0.12666655E 01	0.17748297E-01	0.10002009E 01
14	0.0600	0.23212237E 02	0.35807188E 00	0.13927342E 01	0.21484312E-01	0.10002636E 01
15	0.0650	0.21897220E 02	0.34837804E 00	0.14233193E 01	0.22644572E-01	0.10003036E 01
16	0.0700	0.20616776E 02	0.33945961E 00	0.14431743E 01	0.23762172E-01	0.10003232E 01
17	0.0750	0.19302255E 02	0.32685965E 00	0.14476691E 01	0.24514473E-01	0.10003364E 01
18	0.0800	0.18040025E 02	0.31321418E 00	0.14432019E 01	0.25057134E-01	0.10003488E 01
19	0.0850	0.16846123E 02	0.29964217E 00	0.14319204E 01	0.25469584E-01	0.10003583E 01
20	0.0900	0.19727122E 02	0.28596596E 00	0.14154410E 01	0.25736935E-01	0.10003638E 01
21	0.0950	0.14649044E 02	0.27102713E 00	0.13916591E 01	0.25747577E-01	0.10003546E 01
22	0.1000	0.13550846E 02	0.25487851E 00	0.13550846E 01	0.25487850E-01	0.10002884E 01
23	0.1200	0.10918501E 02	0.22959399E 00	0.13102202E 01	0.27551278E-01	0.10002851E 01
24	0.1400	0.86645448E 01	0.19626418E 00	0.12130363E 01	0.27476985E-01	0.10003217E 01
25	0.1600	0.70871706E 01	0.17136758E 00	0.11339473E 01	0.27418813E-01	0.10003575E 01
26	0.1800	0.60091231E 01	0.15415234E 00	0.10816422E 01	0.27747421E-01	0.10003922E 01
27	0.2000	0.51713191E 01	0.13881772E 00	0.10342638E 01	0.27763543E-01	0.10004214E 01
28	0.2200	0.45959351E 01	0.12899972E 00	0.10111057E 01	0.28379938E-01	0.10005011E 01
29	0.2300	0.42948386E 01	0.12195309E 00	0.98781285E 00	0.28049211E-01	0.10005567E 01
30	0.2400	0.40735695E 01	0.11797007E 00	0.97765666E 00	0.28312816E-01	0.10005735E 01
31	0.2500	0.38677532E 01	0.11358233E 00	0.96693829E 00	0.28395583E-01	0.10005818E 01
32	0.2600	0.36917805E 01	0.10983892E 00	0.95986290E 00	0.28558120E-01	0.10005914E 01
33	0.2700	0.35284914E 01	0.10615358E 00	0.95269266E 00	0.28661466E-01	0.10005970E 01
34	0.2800	0.33947921E 01	0.10339615E 00	0.95053058E 00	0.28950921E-01	0.10006034E 01
35	0.2900	0.32679471E 01	0.10064530E 00	0.94770464E 00	0.29187137E-01	0.10006114E 01
36	0.3000	0.31461412E 01	0.97344322E-01	0.94384233E 00	0.29203296E-01	0.10006172E 01
37	0.3100	0.30488005E 01	0.95350982E-01	0.94512813E 00	0.29558804E-01	0.10006282E 01
38	0.3200	0.29622977E 01	0.93695855E-01	0.94793525E 00	0.29982673E-01	0.10006399E 01
39	0.3300	0.28948997E 01	0.92663685E-01	0.95530369E 00	0.30579016E-01	0.10006518E 01
40	0.3400	0.28273917E 01	0.91493590E-01	0.96131318E 00	0.31107821E-01	0.10006632E 01
41	0.3500	0.27652110E 01	0.90330109E-01	0.96782383E 00	0.31615537E-01	0.10006703E 01
42	0.3600	0.26952612E 01	0.88932602E-01	0.97029403E 00	0.32015736E-01	0.10006803E 01
43	0.3800	0.25589672E 01	0.85474138E-01	0.97240750E 00	0.32480172E-01	0.10005474E 01
44	0.4140	0.24615556E 01	0.85292970E-01	0.10190840E 01	0.35311290E-01	0.10006414E 01
45	0.4200	0.24438054E 01	0.84913039E-01	0.10263982E 01	0.35663476E-01	0.10007437E 01
46	0.4300	0.23717170E 01	0.82706426E-01	0.10198383E 01	0.35563763E-01	0.10007231E 01
47	0.4500	0.23232844E 01	0.82607475E-01	0.10454780E 01	0.37173363E-01	0.10007648E 01
48	0.4600	0.22593986E 01	0.80534853E-01	0.10393049E 01	0.37046032E-01	0.10008140E 01
49	0.4700	0.22268623E 01	0.79903379E-01	0.10466253E 01	0.37554587E-01	0.10008364E 01
50	0.4750	0.22008186E 01	0.79089944E-01	0.10453888E 01	0.37567723E-01	0.10008430E 01
51	0.4800	0.21674032E 01	0.77896403E-01	0.10403535E 01	0.37390273E-01	0.10008354E 01
52	0.4900	0.21313348E 01	0.76968352E-01	0.10443940E 01	0.37714492E-01	0.10008241E 01
53	0.5000	0.20776987E 01	0.75102171E-01	0.10388494E 01	0.37551086E-01	0.10007314E 01
54	0.5320	0.19983007E 01	0.73642945E-01	0.10630959E 01	0.39178047E-01	0.10007380E 01

55	0.9500	0.19233250E 01	0.71073730E-01	0.10578287E 01	0.39090551E-01	0.10007797E 01
56	0.9750	0.18533700E 01	0.69196970E-01	0.10656877E 01	0.39788257E-01	0.10008145E 01
57	0.9900	0.18141091E 01	0.68191182E-01	0.10703243E 01	0.40232797E-01	0.10008862E 01
58	0.6000	0.17666839E 01	0.66305602E-01	0.10600103E 01	0.39783361E-01	0.10008424E 01
59	0.6250	0.17026820E 01	0.64386684E-01	0.10641762E 01	0.40241678E-01	0.10008038E 01
60	0.6500	0.16415884E 01	0.62478211E-01	0.10670325E 01	0.40610837E-01	0.10007815E 01
61	0.6830	0.15639155E 01	0.59966471E-01	0.10681542E 01	0.40957100E-01	0.10008277E 01
62	0.7000	0.14945276E 01	0.56970000E-01	0.10461693E 01	0.39879000E-01	0.10007426E 01
63	0.7500	0.14149997E 01	0.54584970E-01	0.10612498E 01	0.40938728E-01	0.10006747E 01
64	0.8000	0.13148242E 01	0.50945894E-01	0.10518593E 01	0.40756714E-01	0.10006950E 01
65	0.8500	0.12402441E 01	0.48525539E-01	0.10542075E 01	0.41246708E-01	0.10007927E 01
66	0.8760	0.12116672E 01	0.47765319E-01	0.10614204E 01	0.41842419E-01	0.10009249E 01
67	0.8900	0.11735239E 01	0.46121837E-01	0.10444363E 01	0.41048434E-01	0.10009416E 01
68	0.9100	0.11429195E 01	0.44938208E-01	0.10400568E 01	0.40893769E-01	0.10009215E 01
69	0.9300	0.11199328E 01	0.44159225E-01	0.10415375E 01	0.41068079E-01	0.10009303E 01
70	0.9500	0.10851814E 01	0.42718348E-01	0.10309223E 01	0.40582430E-01	0.10009180E 01
71	0.9700	0.10784665E 01	0.42743827E-01	0.10461125E 01	0.41461511E-01	0.10009586E 01
72	0.9800	0.10537611E 01	0.41610030E-01	0.10326859E 01	0.40777829E-01	0.10009623E 01
73	0.9900	0.10402713E 01	0.41071730E-01	0.10298686E 01	0.40661012E-01	0.10009577E 01
74	1.0000	0.10264198E 01	0.40495302E-01	0.10264198E 01	0.40495302E-01	0.10009224E 01
75	1.0250	0.99800194E 00	0.39380860E-01	0.10229520E 01	0.40365380E-01	0.10008924E 01
76	1.0500	0.97438705E 00	0.38520871E-01	0.10231064E 01	0.40446915E-01	0.10009193E 01
77	1.0600	0.95338910E 00	0.37550928E-01	0.10105924E 01	0.39803984E-01	0.10009386E 01
78	1.0700	0.93697985E 00	0.36820002E-01	0.10025684E 01	0.39397402E-01	0.10009348E 01
79	1.0800	0.93114658E 00	0.36654378E-01	0.10056383E 01	0.39586728E-01	0.10009302E 01
80	1.0900	0.91586848E 00	0.36004530E-01	0.99829663E 00	0.39244937E-01	0.10009104E 01
81	1.1100	0.89145647E 00	0.34958959E-01	0.98951668E 00	0.38804444E-01	0.10008978E 01
82	1.1250	0.88734776E 00	0.34939313E-01	0.99826623E 00	0.39306727E-01	0.10009009E 01
83	1.1300	0.87879570E 00	0.34593435E-01	0.99303913E 00	0.39045381E-01	0.10008906E 01
84	1.1500	0.86330076E 00	0.33938725E-01	0.99279587E 00	0.39029533E-01	0.10008141E 01
85	1.2000	0.85468093E 00	0.34089898E-01	0.10256171E 01	0.40907877E-01	0.10007940E 01
86	1.2500	0.84063247E 00	0.33923839E-01	0.10507906E 01	0.42404798E-01	0.10008246E 01
87	1.3000	0.81694221E 00	0.33183824E-01	0.10620249E 01	0.43138971E-01	0.10008431E 01
88	1.3500	0.77807938E 00	0.31574334E-01	0.10504072E 01	0.42625351E-01	0.10007680E 01
89	1.4400	0.74707047E 00	0.30672750E-01	0.10757815E 01	0.44168760E-01	0.10007848E 01
90	1.5000	0.70237260E 00	0.28681558E-01	0.10535589E 01	0.43022337E-01	0.10007745E 01
91	1.6000	0.66309028E 00	0.27282443E-01	0.10609444E 01	0.43651908E-01	0.10007488E 01
92	1.7000	0.62788824E 00	0.26118377E-01	0.10674100E 01	0.44401241E-01	0.10007994E 01
93	1.7800	0.60406934E 00	0.25428916E-01	0.10752363E 01	0.45263470E-01	0.10008561E 01
94	1.8600	0.56328104E 00	0.23808142E-01	0.10477027E 01	0.44283144E-01	0.10009117E 01
95	1.9000	0.54702289E 00	0.23030312E-01	0.10393435E 01	0.43757593E-01	0.10008830E 01
96	2.0000	0.50833768E 00	0.21312494E-01	0.10166754E 01	0.42624989E-01	0.10007884E 01
97	2.1000	0.48421238E 00	0.20558989E-01	0.10168460E 01	0.43173876E-01	0.10008106E 01
98	2.2000	0.46012069E 00	0.19796866E-01	0.10122655E 01	0.43553105E-01	0.10008438E 01
99	2.2900	0.43957490E 00	0.19102170E-01	0.10066265E 01	0.43743968E-01	0.10009329E 01
100	2.3300	0.42723015E 00	0.18578014E-01	0.99544623E 00	0.43286773E-01	0.10009839E 01
101	2.3800	0.42221944E 00	0.18486788E-01	0.10048727E 01	0.43998555E-01	0.10010601E 01

CONSTANTS FOR ONE BROAD GROUP
FROM 0.0 EV TO 0.5320 EV

DIFFUSION COEFF,	= 0.30969903E 00
SIGMA TRANSPORT	= 0.10763138E 01
SIGMA ABSORPTION	= 0.18005052E 00
SIGMA CAPTURE	= 0.55671817E-01
SIGMA FISSION	= 0.12437870E 00
NU SIGMA FISSION	= 0.30224025E 00
AVERAGE NJ	= 0.24300000E 01
SIGMA TOTAL	= 0.20640559E 01
TOTAL P0 SCATTER	= 0.18840054E 01
TOTAL P1 SCATTER	= 0.61785483E 00
AVERAGE 1/V SEC/KM	= 0.27345552E 00
INTEGRATED FLUX	= 0.38107158E 01
INTEGRATED CURRENT	= 0.76484077E-01
ISOTROPIC SOURCE	= 0.16524585E 00
ANISOTROPIC SOURCE	= 0.15053001E-02

BROAD GROUP ENERGY LIMITS			BROAD GROUP FLUXES AND CURRENTS	
GROUP	UPPER ENERGY	LOWER ENERGY	INTEGRATED FLUXES	INTEGRATED CURRENTS
1	2,3800	1,4400	0,52315E 00	0,21976E-01
2	1,4400	0,5320	0,10425E 01	0,40630E-01
3	0,5320	0,4140	0,26083E 00	0,93243E-02
4	0,4140	0,1000	0,15505E 01	0,40724E-01
5	0,1000	0,0253	0,16123E 01	0,23603E-01
6	0,0253	0,0000	0,38710E 00	0,28334E-02

AVERAGED BROAD GROUP MACROSCOPIC CROSS SECTIONS

GROUP	DIFFUSION COEFF,	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA REMOVAL	SIGMA TOTAL	AVERAGE 1/V (SEC/KM)
1	0,64818396E 00	0,17951141E-01	0,16529961E-01	0,57941231E 00	0,11630913E 01	0,53379355E-01
2	0,60137116E 00	0,42127809E-01	0,64843890E-01	0,46132119E 00	0,11735861E 01	0,78215164E-01
3	0,55160715E 00	0,59167258E-01	0,96241985E-01	0,71336631E 00	0,12101225E 01	0,10559151E 00
4	0,40528418E 00	0,11785102E 00	0,19482607E 00	0,47409398E 00	0,16447140E 01	0,16823569E 00
5	0,22388457E 00	0,21024221E 00	0,35670426E 00	0,51833682E 00	0,22328258E 01	0,31522832E 00
6	0,11294464E 00	0,38488120E 00	0,64442535E 00	0,10709848E 01	0,36160921E 01	0,63401660E 00

GROUP	SIGMA TRANSPORT	SIGMA CAPTURE	SIGMA FISSION	AVERAGE NU	TOTAL P0 SCATTER	TOTAL P1 SCATTER
1	0,51425728E 00	0,11148688E-01	0,68024534E-02	0,24299999E 01	0,11451402E 01	0,57065122E 00
2	0,55428884E 00	0,15443081E-01	0,26684728E-01	0,24300000E 01	0,11314583E 01	0,50820642E 00
3	0,60429479E 00	0,19561503E-01	0,39605755E-01	0,24300000E 01	0,11509553E 01	0,45901402E 00
4	0,82246814E 00	0,37675682E-01	0,80175339E-01	0,24300000E 01	0,15268630E 01	0,63343010E 00
5	0,14756799E 01	0,63450340E-01	0,14679188E 00	0,24300000E 01	0,20225836E 01	0,64958714E 00
6	0,29512984E 01	0,11968558E 00	0,26519562E 00	0,24300000E 01	0,32312109E 01	0,65236449E 00

SOURCES FROM ABOVE 2,380 EV

GROUP	ISOTROPIC	ANISOTROPIC
1	0,30089771E 00	0,74297523E-02
2	0,28230149E 00	0,51988804E-02
3	0,36686760E-01	0,47199652E-03
4	0,97624084E-01	0,91072209E-03
5	0,23224584E-01	0,10703340E-03
6	0,77104376E-02	0,15548168E-04

UP SCATTERING TO ABOVE 2,380 EV

P0 SCATTERING	P1 SCATTERING
-0,20666357E-07	-0,26263202E-07
0,44487197E-08	-0,13989125E-07
-0,16301762E-07	-0,13078919E-07
-0,43775614E-07	-0,26813893E-07
0,16012338E-07	-0,33279893E-07
-0,75290599E-07	-0,53976928E-07

MACROSCOPIC P0 SCATTERING MATRIX

FROM GROUP

		1	2	3	4	5	6
TO GROUP	1	5.8367898 E+01	3.4999972 E+03	0.	0.	0.	0.
	2	3.7352242 E+01	7.1226487 E+01	2.2277729 E+02	2.6409225 E+05	2.0849292 E+09	7.2546282 E+10
	3	4.4166249 E+02	1.1434331 E+01	4.9675621 E+01	4.9514833 E+03	2.2804385 E+07	7.6029078 E+08
	4	1.1568372 E+01	2.4971686 E+01	5.4310558 E+01	1.1706201 E+00	1.0331409 E+01	2.8404033 E+02
	5	2.3410207 E+02	4.4657617 E+02	7.7479320 E+02	3.0985550 E+01	1.7144890 E+00	6.5769955 E+01
	6	4.6786407 E+03	6.9756210 E+03	1.1336470 E+02	4.1409630 E+02	2.0478030 E+01	2.5451073 E+00

MACROSCOPIC P1 SCATTERING MATRIX

FROM GROUP

		1	2	3	4	5	6
TO GROUP	1	3.0470303 E+01	3.3933558 E+03	0.	0.	0.	0.
	2	2.1767001 E+01	3.7034259 E+01	1.6647020 E+02	1.6079056 E+05	-1.0030685 E+10	-6.1595657 E+11
	3	1.8324460 E+02	5.8349790 E+02	2.3988591 E+01	4.2633998 E+03	-8.8033061 E+09	-8.7002439 E+09
	4	2.8739063 E+02	7.6153421 E+02	2.0839156 E+01	5.9908538 E+01	3.2680806 E+02	-1.4681083 E+03
	5	1.1587496 E+03	2.4279672 E+04	-4.7490058 E+03	3.2749797 E+02	6.1743503 E+01	4.0471302 E+02
	6	5.5957789 E+05	-2.7549989 E+04	-1.1614435 E+03	-2.6845063 E+03	-5.2864826 E+04	6.1336137 E+01

AVERAGE MICROSCOPIC CROSS SECTIONS

(UNSELF-SHIELDED)

MCMURRY WATER MODEL H2O MOLECULE 293K 8/5/66 (1/V ABS .664)

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,77976649E-01	0,	0,	0,77976649E-01	0,
2	0,11425664E 00	0,	0,	0,11425664E 00	0,
3	0,15424825E 00	0,	0,	0,15424825E 00	0,
4	0,24575897E 00	0,	0,	0,24575897E 00	0,
5	0,46048604E 00	0,	0,	0,46048604E 00	0,
6	0,92617247E 00	0,	0,	0,92617247E 00	0,

GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,34700892E 02	0,53127192E 02	0,53049176E 02	0,33323577E 02	0,32277312E 02
2	0,23085051E 02	0,51980200E 02	0,51865944E 02	0,29404194E 02	0,24119782E 02
3	0,20451429E 02	0,52845014E 02	0,52690765E 02	0,26340550E 02	0,36237689E 02
4	0,27526639E 02	0,72496105E 02	0,72250347E 02	0,35213890E 02	0,19603426E 02
5	0,56010367E 02	0,95746086E 02	0,95285600E 02	0,34245593E 02	0,16112461E 02
6	0,11455221E 03	0,14486340E 03	0,14393723E 03	0,30741069E 02	0,32756733E 02

SOURCES FROM ABOVE 2.380 EV

UP SCATTERING TO ABOVE 2.380 EV

GROUP	ISOTROPIC	ANISOTROPIC	P0 SCATTERING	P1 SCATTERING
1	0,17457169E 02	0,15176450E-01	=0,	0,
2	0,16862881E 02	0,10462653E-01	=0,	0,
3	0,21914319E 01	0,94988445E-03	=0,	0,
4	0,58314369E 01	0,18328117E-02	=0,	0,
5	0,13872877E 01	0,21540277E-03	=0,	0,
6	0,46057211E 00	0,31290406E-04	=0,	0,

MICROSCOPIC P0 SCATTERING MATRIX

FROM GROUP

		1	2	3	4	5	6
TO GROUP							
1		2.0849841 E 01	1.8816591 E-01	0.	0.	0.	0.
2		2.1078801 E 01	2.7860419 E 01	1.1312415 E 00	1.3592765 E-03	1.1084729 E-07	3.4397373 E-08
3		2.6133212 E 00	6.2462134 E 00	1.6607325 E 01	2.4391882 E-01	1.2124702 E-05	3.6048719 E-06
4		6.8450759 E 00	1.4550546 E 01	2.9800650 E 01	5.2892680 E 01	5.2599342 E 00	1.3464416 E 00
5		1.3852627 E 00	2.6125166 E 00	4.4940000 E 00	1.6806069 E 01	7.9633625 E 01	3.0484116 E 01
6		2.7687678 E-01	4.0808422 E-01	6.5754957 E-01	2.3063213 E 00	1.0392030 E 01	1.1210667 E 02

MICROSCOPIC P1 SCATTERING MATRIX

FROM GROUP

		1	2	3	4	5	6
TO GROUP							
1		1.7314476 E 01	1.8737549 E-01	0.	0.	0.	0.
2		1.3152482 E 01	2.1222572 E 01	9.3238124 E-01	9.8013689 E-04	-5.3574356 E-09	-2.9205200 E-09
3		1.0841954 E 00	3.5091771 E 00	1.3309467 E 01	2.3309344 E-01	-4.6739861 E-07	-4.1251669 E-07
4		1.7005084 E 00	4.4869471 E 00	1.2441601 E 01	3.3273757 E 01	1.7584823 E 00	-6.9355149 E-02
5		6.8601125 E-02	1.4236754 E-02	-2.7553461 E-01	1.8564317 E 00	3.2519360 E 01	1.9882236 E 00
6		3.3150120 E-03	-1.6113724 E-02	-6.7364722 E-02	-1.5037098 E-01	-3.2248311 E-02	2.8822201 E 01

AVERAGE MICROSCOPIC CROSS SECTIONS
(SELF-SHIELDED)

MCMURRY WATER MODEL H2O MOLECULE 293K 8/5/66 (1/V ABS .664)

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,78713546E-01	0,	0,	0,78713546E-01	0,
2	0,11666111E 00	0,	0,	0,11666111E 00	0,
3	0,15883570E 00	0,	0,	0,15883570E 00	0,
4	0,26252605E 00	0,	0,	0,26252605E 00	0,
5	0,51981645E 00	0,	0,	0,51981645E 00	0,
6	0,11668106E 01	0,	0,	0,11668106E 01	0,

GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,35035292E 02	0,53426419E 02	0,53547705E 02	0,33636275E 02	0,32588687E 02
2	0,23655316E 02	0,53172468E 02	0,52955807E 02	0,30021613E 02	0,24629637E 02
3	0,21217809E 02	0,54417750E 02	0,54258915E 02	0,27124724E 02	0,37326688E 02
4	0,29684707E 02	0,77373607E 02	0,77111082E 02	0,37513249E 02	0,21049803E 02
5	0,63224738E 02	0,10400512E 03	0,10748531E 03	0,38514549E 02	0,18351003E 02
6	0,14338068E 03	0,18250180E 03	0,18133499E 03	0,38728211E 02	0,41267582E 02

SOURCES FROM ABOVE 2 380 EV

UP SCATTERING TO ABOVE 2,380 EV

GROUP	ISOTROPIC	ANISOTROPIC	P0 SCATTERING	P1 SCATTERING
1	0,17457169E 02	0,15176450E-01	-0,13795269E-06	0,26780716E-06
2	0,16862881E 02	0,11462653E-01	-0,27826144E-06	0,77774358E-07
3	0,21914319E 01	0,94988445E-03	-0,37829712E-06	0,78695427E-07
4	0,58314369E 01	0,18328117E-02	0,25444886E-06	-0,35581137E-07
5	0,13872877E 01	0,2540277E-03	0,28932645E-05	-0,11576579E-05
6	0,46057211E 00	0,32290406E-04	-0,54018319E-06	-0,15286988E-05

MICROSCOPIC P0 SCATTERING MATRIX

		FROM GROUP					
		1	2	3	4	5	6
TO GROUP	1	2.1037732 E 01	1.9209293 E-01	0.	0.	0.	0.
	2	2.1283716 E 01	2.8442832 E 01	1.1638134 E 00	1.4421805 E-03	1.2454029 E-07	4.3334497 E-08
	3	2.6382086 E 00	6.3791391 E 00	1.7091063 E 01	2.5879562 E-01	1.3621878 E-05	4.5414896 E-06
	4	6.9102029 E 00	1.4857507 E 01	3.0698754 E 01	5.6323805 E 01	5.9085595 E 00	1.6962740 E 00
	5	1.3983755 E 00	2.6675601 E 00	4.6281177 E 00	1.8053854 E 01	8.9654121 E 01	3.8404496 E 01
	6	2.7947200 E-01	4.1667884 E-01	6.7716803 E-01	2.4731873 E 00	1.1922611 E 01	1.4123422 E 02

MICROSCOPIC P1 SCATTERING MATRIX

		FROM GROUP					
		1	2	3	4	5	6
TO GROUP	1	1.7470794 E 01	1.9128601 E-01	0.	0.	0.	0.
	2	1.3281650 E 01	2.1666270 E 01	9.5922037 E-01	1.0399154 E-03	-6.0215545 E-09	-3.6793295 E-09
	3	1.0945857 E 00	3.5842646 E 00	1.3696946 E 01	2.4730981 E-01	-5.2585310 E-07	-5.1969676 E-07
	4	1.7166872 E 00	4.5817466 E 00	1.2821610 E 01	3.5403350 E 01	1.9727421 E 00	-8.7374994 E-02
	5	6.9216269 E-02	1.4503120 E-02	-2.8367516 E-01	2.0217201 E 00	3.6532371 E 01	2.5048039 E 00
	6	3.3425596 E-03	-1.6456598 E-02	-6.9377184 E-02	-1.6017053 E-01	9.4378291 E-03	3.6310785 E 01

AVERAGE MICROSCOPIC CROSS SECTIONS

(UNSELF-SHIELDED)

OXYGEN 303K GAS ABS 1/V (.178MB)

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,20903379E-04	0,	0,	0,20903379E-04	0,
2	0,30629092E-04	0,	0,	0,30629092E-04	0,
3	0,41349682E-04	0,	0,	0,41349682E-04	0,
4	0,65881168E-04	0,	0,	0,65881168E-04	0,
5	0,12344354E-03	0,	0,	0,12344354E-03	0,
6	0,24828117E-03	0,	0,	0,24828117E-03	0,

GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,34076744E 01	0,39789834E 01	0,39789625E 01	0,34150908E 00	0,91937494E 00
2	0,36723795E 01	0,38924039E 01	0,38923733E 01	0,24224884E 00	0,47787974E 00
3	0,36942064E 01	0,38485643E 01	0,38485229E 01	0,20696462E 00	0,17508991E 01
4	0,37212493E 01	0,39089795E 01	0,39089136E 01	0,24632039E 00	0,48606483E 00
5	0,37686042E 01	0,39455682E 01	0,39454447E 01	0,23119172E 00	0,60497361E 00
6	0,41307193E 01	0,41547846E 01	0,41545363E 01	0,22898793E 00	0,11300537E 01

SOURCES FROM ABOVE 2.380 EV

UP SCATTERING TO ABOVE 2,380 EV

GROUP	ISOTROPIC	ANISOTROPIC	P0 SCATTERING	P1 SCATTERING
1	0,45583867E 00	-0,30422470E-03	0,14960842E-07	0,12858151E-07
2	0,	-0,	0,12488718E-07	-0,45388293E-08
3	0,	-0,	-0,52747680E-08	-0,61978275E-08
4	0,	-0,	0,11881436E-07	0,28444219E-08
5	0,	-0,	0,28927528E-07	-0,24393663E-08
6	0,	0,	0,26452499E-07	0,32271340E-08

MICROSCOPIC P0 SCATTERING MATRIX

FROM GROUP

		1	2	3	4	5	6
TO GROUP	1	3.0596085 E 00	1.5375556 E-02	0.	0.	0.	0.
	2	9.1935407 E-01	3.4145242 E 00	1.5272214 E-01	1.2915233 E-04	0.	0.
	3	0.	4.0895190 E-01	2.0976652 E 00	3.5284529 E-02	0.	0.
	4	0.	5.3521699 E-02	1.5981357 E 00	3.4229146 E 00	2.7220748 E-01	5.1298860 E-04
	5	0.	0.	0.	4.5022168 E-01	3.3405945 E 00	1.1292924 E 00
	6	0.	0.	0.	3.6365919 E-04	3.3264273 E-01	3.0247309 E 00

MICROSCOPIC P1 SCATTERING MATRIX

FROM GROUP

		1	2	3	4	5	6
TO GROUP	1	5.9077425 E-01	1.0336428 E-02	0.	0.	0.	0.
	2	-2.4926518 E-01	3.5117199 E-01	3.2172990 E-02	-7.5824630 E-05	0.	0.
	3	0.	-8.9475494 E-02	5.1669284 E-01	7.0220815 E-03	0.	0.
	4	0.	-2.9784067 E-02	-3.4190120 E-01	3.0191975 E-01	-2.1344670 E-02	-4.1009144 E-04
	5	0.	0.	0.	-6.2358949 E-02	2.9444252 E-01	-1.1175430 E-01
	6	0.	0.	0.	-1.8666519 E-04	-4.1906118 E-02	3.4115233 E-01

AVERAGE MICROSCOPIC CROSS SECTIONS
(SELF-SHIELDED)

OXYGEN 303K GAS ABS 1/V (.178MB)

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,20665515E-04	0,	0,	0,20665515E-04	0,
2	0,29834490E-04	0,	0,	0,29834490E-04	0,
3	0,39833863E-04	0,	0,	0,39833863E-04	0,
4	0,60354217E-04	0,	0,	0,60354217E-04	0,
5	0,10415460E-03	0,	0,	0,10415460E-03	0,
6	0,17117745E-03	0,	0,	0,17117745E-03	0,

GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,33697622E 01	0,39340644E 01	0,39340437E 01	0,33768894E 00	0,90733173E 00
2	0,35794161E 01	0,37916055E 01	0,37915767E 01	0,23598372E 00	0,46506134E 00
3	0,35644651E 01	0,37077838E 01	0,37077459E 01	0,19941308E 00	0,16854417E 01
4	0,34278582E 01	0,35894119E 01	0,35893515E 01	0,22648777E 00	0,43459785E 00
5	0,32026013E 01	0,33384810E 01	0,33383769E 01	0,19607931E 00	0,50525793E 00
6	0,29092620E 01	0,28645153E 01	0,28643451E 01	0,15787572E 00	0,77911544E 00

SOURCES FROM ABOVE 2,380 EV

UP SCATTERING TO ABOVE 2,380 EV

GROUP	ISOTROPIC	ANISOTROPIC	P0 SCATTERING	P1 SCATTERING
1	0,45583867E 00	-0,80422470E-03	0,66282233E-08	0,12976401E-07
2	0,	-0,	0,48540530E-07	-0,43262232E-08
3	0,	-0,	-0,25274316E-07	-0,58061165E-08
4	0,	-0,	0,48973650E-07	0,25740231E-08
5	0,	-0,	0,11866068E-07	-0,40830426E-08
6	0,	C,	-0,26335699E-07	0,31943602E-08

MICROSCOPIC P0 SCATTERING MATRIX

FROM GROUP

		1	2	3	4	5	6
TO GROUP	1	3.0267326 E 00	1.4979943 E-02	0.	0.	0.	0.
	2	9.0731110 E-01	3.3265452 E 00	1.4730249 E-01	1.1943491 E-04	0.	0.
	3	0.	3.9800395 E-01	2.0223441 E 00	3.2629721 E-02	0.	0.
	4	0.	5.2047621 E-02	1.5380994 E 00	3.1548141 E 00	2.3188676 E-01	3.5367998 E-04
	5	0.	0.	0.	4.0147197 E-01	2.8332231 E 00	7.7859065 E-01
	6	0.	0.	0.	3.1638952 E-04	2.7326707 E-01	2.0854008 E 00

MICROSCOPIC P1 SCATTERING MATRIX

FROM GROUP

		1	2	3	4	5	6
TO GROUP	1	5.8429355 E-01	1.0070472 E-02	0.	0.	0.	0.
	2	-2.4660462 E-01	3.4209586 E-01	3.1033811 E-02	-7.0119582 E-05	0.	0.
	3	0.	-8.7210142 E-02	4.9811112 E-01	6.4937402 E-03	0.	0.
	4	0.	-2.8972456 E-02	-3.2973184 E-01	2.7799313 E-01	-1.8179785 E-02	-2.8273754 E-04
	5	0.	0.	0.	-5.7766015 E-02	2.5045558 E-01	-7.7049003 E-02
	6	0.	0.	0.	-1.6295221 E-04	-3.6196474 E-02	2.3520745 E-01

AVERAGE MICROSCOPIC CROSS SECTIONS

(UNSELF-SHIELDED)

CHROMIUM

1/V (3.1 B), CONST SCAT (3.0 B)

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,36404762E 00	0,	0,	0,36404752E 00	0,
2	0,53342801E 00	0,	0,	0,53342801E 00	0,
3	0,72013490E 00	0,	0,	0,72013490E 00	0,
4	0,11473687E 01	0,	0,	0,11473687E 01	0,
5	0,21498596E 01	0,	0,	0,21498596E 01	0,
6	0,43239981E 01	0,	0,	0,43239981E 01	0,

GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,33255934E 01	0,33540476E 01	0,30000000E 01	0,38454175E-01	0,36404762E 00
2	0,34949739E 01	0,35534281E 01	0,30000000E 01	0,38454174E-01	0,53342801E 00
3	0,36816807E 01	0,37201349E 01	0,30000000E 01	0,38454175E-01	0,72013490E 00
4	0,41089145E 01	0,41473686E 01	0,30000000E 01	0,38454174E-01	0,11473687E 01
5	0,51114053E 01	0,51498595E 01	0,30000000E 01	0,38454174E-01	0,21498596E 01
6	0,72855438E 01	0,73239981E 01	0,30000000E 01	0,38454174E-01	0,43239981E 01

AVERAGE MICROSCOPIC CROSS SECTIONS
(SELF-SHIELDED)

CHROMIUM

1/V (3.1 B), CONST SCAT (3.0 B)

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,36368796E 00	0,	0,	0,36368796E 00	0,
2	0,53334674E 00	0,	0,	0,53334674E 00	0,
3	0,72028560E 00	0,	0,	0,72028560E 00	0,
4	0,11469541E 01	0,	0,	0,11469541E 01	0,
5	0,21264180E 01	0,	0,	0,21264180E 01	0,
6	0,41376769E 01	0,	0,	0,41376769E 01	0,

GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,33222938E 01	0,33607097E 01	0,29970218E 01	0,38415967E 01	0,36368796E 00
2	0,34944366E 01	0,35328848E 01	0,29995381E 01	0,38448241E 01	0,53334674E 00
3	0,36824477E 01	0,37209099E 01	0,30006243E 01	0,38462156E 01	0,72028560E 00
4	0,41083522E 01	0,41468142E 01	0,29998601E 01	0,38461952E 01	0,11469541E 01
5	0,50572009E 01	0,50952733E 01	0,29688554E 01	0,38072240E 01	0,21264180E 01
6	0,69716096E 01	0,70084068E 01	0,28707299E 01	0,36797183E 01	0,41376769E 01

AVERAGE MICROSCOPIC CROSS SECTIONS

(UNSELF-SHIELDED)

MANGANESE (NATURAL) 1/V I 13.2 B), CONST SCAT (2.3 B)

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,15501382E 01	0,	0,	0,15501382E 01	0,
2	0,22713709E 01	0,	0,	0,22713709E 01	0,
3	0,30663809E 01	0,	0,	0,30663809E 01	0,
4	0,48855699E 01	0,	0,	0,48855699E 01	0,
5	0,91542406E 01	0,	0,	0,91542406E 01	0,
6	0,18411863E 02	0,	0,	0,18411863E 02	0,

GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,38222239E 01	0,38501382E 01	0,23000000E 01	0,27914342E 01	0,15501382E 01
2	0,45434565E 01	0,45713708E 01	0,22999999E 01	0,27914345E 01	0,22713709E 01
3	0,53384664E 01	0,53663807E 01	0,22999999E 01	0,27914345E 01	0,30663809E 01
4	0,71576554E 01	0,71855698E 01	0,22999999E 01	0,27914347E 01	0,48855699E 01
5	0,11426326E 02	0,11454241E 02	0,23000000E 01	0,27914378E 01	0,91542406E 01
6	0,20683948E 02	0,20711863E 02	0,23000000E 01	0,27914498E 01	0,18411863E 02

AVERAGE MICROSCOPIC CROSS SECTIONS
(SELF-SHIELDED)

MANGANESE (NATURAL)   1/V (13.2   E), CONST SCAT (2.3   B)

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,15486068E 01	0,	0,	0,15486068E 01	0,
2	0,22710249E 01	0,	0,	0,22710249E 01	0,
3	0,30670226E 01	0,	0,	0,30670226E 01	0,
4	0,48838046E 01	0,	0,	0,48838046E 01	0,
5	0,90544250E 01	0,	0,	0,90544250E 01	0,
6	0,17618495E 02	0,	0,	0,17618495E 02	0,

GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,38184368E 01	0,38463235E 01	0,22977167E 01	0,27886606E-01	0,15486068E 01
2	0,45427606E 01	0,45706707E 01	0,22996459E 01	0,27910038E-01	0,22710249E 01
3	0,53395809E 01	0,53675012E 01	0,23004786E 01	0,27920138E-01	0,30670226E 01
4	0,71557772E 01	0,71836972E 01	0,22998927E 01	0,27919993E-01	0,48838046E 01
5	0,11302910E 02	0,11330547E 02	0,22761224E 01	0,27637126E-01	0,90544250E 01
6	0,19792676E 02	0,19819388E 02	0,22008930E 01	0,26711662E-01	0,17618495E 02

AVERAGE MICROSCOPIC CROSS SECTIONS
(UNSELF-SHIELDED)

IRON (NATURAL)   1/4 (2.62 B), CONST SCAT (11.0 B)

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,30767895E 00	0,	0,	0,30767895E 00	0,
2	0,45083270E 00	0,	0,	0,45083270E 00	0,
3	0,60863014E 00	0,	0,	0,60863014E 00	0,
4	0,96971159E 00	0,	0,	0,96971159E 00	0,
5	0,18169780E 01	0,	0,	0,18169780E 01	0,
6	0,36544757E 01	0,	0,	0,36544757E 01	0,

GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,11176375E 02	0,11307679E 02	0,11000000E 02	0,13130414E 00	0,30767895E 00
2	0,11319529E 02	0,11450833E 02	0,11000000E 02	0,13130414E 00	0,45083270E 00
3	0,11477326E 02	0,11608630E 02	0,11000000E 02	0,13130414E 00	0,60863014E 00
4	0,11838407E 02	0,11969711E 02	0,11000000E 02	0,13130414E 00	0,96971159E 00
5	0,12685674E 02	0,12816973E 02	0,11000000E 02	0,13130414E 00	0,18169780E 01
6	0,14523171E 02	0,14654475E 02	0,11000000E 02	0,13130414E 00	0,36544757E 01

AVERAGE MICROSCOPIC CROSS SECTIONS
(SELF-SHIELDED)

IRON (NATURAL)   1/V (2.62   B), CONST SCAT (11.0   B)

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,30737498E 00	0,	0,	0,30737498E 00	0,
2	0,45076402E 00	0,	0,	0,45076402E 00	0,
3	0,60875750E 00	0,	0,	0,60875750E 00	0,
4	0,96936119E 00	0,	0,	0,96936119E 00	0,
5	0,17971661E 01	0,	0,	0,17971661E 01	0,
6	0,34970044E 01	0,	0,	0,34970044E 01	0,

GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,11165281E 02	0,11296455E 02	0,10989080E 02	0,13117358E 00	0,30737498E 00
2	0,11317786E 02	0,11449070E 02	0,10998306E 02	0,13128388E 00	0,45076402E 00
3	0,11479715E 02	0,11611046E 02	0,11002289E 02	0,13133140E 00	0,60875750E 00
4	0,11837517E 02	0,11968848E 02	0,10999487E 02	0,13133070E 00	0,96936119E 00
5	0,12552969E 02	0,12682969E 02	0,10885803E 02	0,13000000E 00	0,17971661E 01
6	0,13397368E 02	0,14023014E 02	0,10526010E 02	0,12964625E 00	0,34970044E 01

AVERAGE MICROSCOPIC CROSS SECTIONS
(UNSELF-SHIELDED)

NICKEL (NATURAL) 1/V (4,6 R), CONST SCAT (17,5 B)

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,54019968E 00	0,	0,	0,54019968E 00	0,
2	0,79153834E 00	0,	0,	0,79153834E 00	0,
3	0,10685873E 01	0,	0,	0,10685873E 01	0,
4	0,17025470E 01	0,	0,	0,17025470E 01	0,
5	0,31901141E 01	0,	0,	0,31901141E 01	0,
6	0,64162551E 01	0,	0,	0,64162551E 01	0,

GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,17841415E 02	0,18040199E 02	0,17500000E 02	0,19878482E 00	0,54019968E 00
2	0,18092753E 02	0,18291538E 02	0,17500000E 02	0,19878483E 00	0,79153834E 00
3	0,18369802E 02	0,18568587E 02	0,17500000E 02	0,19878483E 00	0,10685873E 01
4	0,19003762E 02	0,19202547E 02	0,17500000E 02	0,19878483E 00	0,17025470E 01
5	0,20491329E 02	0,20690114E 02	0,17500000E 02	0,19878483E 00	0,31901141E 01
6	0,23717470E 02	0,23916255E 02	0,17500000E 02	0,19878482E 00	0,64162551E 01

AVERAGE MICROSCOPIC CROSS SECTIONS
(SELF-SHIELDED)

NICKEL (NATURAL)   1/V (4,6   E), CONST SCAT (17,5   B)

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,53966599E 00	0,	0,	0,53966599E 00	0,
2	0,79141773E 00	0,	0,	0,79141773E 00	0,
3	0,10688109E 01	0,	0,	0,10688109E 01	0,
4	0,17019319E 01	0,	0,	0,17019319E 01	0,
5	0,31553299E 01	0,	0,	0,31553299E 01	0,
6	0,61397786E 01	0,	0,	0,61397786E 01	0,

GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,17823705E 02	0,18022293E 02	0,17482627E 02	0,19859732E 00	0,53966599E 00
2	0,18089969E 02	0,18288723E 02	0,17497306E 02	0,19875416E 00	0,79141773E 00
3	0,18373626E 02	0,18572452E 02	0,17503642E 02	0,19882608E 00	0,10688109E 01
4	0,19002290E 02	0,19201116E 02	0,17499184E 02	0,19882502E 00	0,17019319E 01
5	0,20276842E 02	0,20473693E 02	0,17318323E 02	0,19681045E 00	0,31553299E 01
6	0,22695484E 02	0,22885703E 02	0,16745925E 02	0,19021918E 00	0,61397786E 01

AVERAGE MICROSCOPIC CROSS SECTIONS
(UNSELF-SHIELDED)

URANIUM 235 NASA REPORT DEC 1964

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,21385526E 02	0,36295215E 02	0,14936302E 02	0,64492227E 01	0,24300000E 01
2	0,70165142E 02	0,14449998E 03	0,59465014E 02	0,10700128E 02	0,24300000E 01
3	0,10225110E 03	0,21689173E 03	0,89255858E 02	0,12995237E 02	0,24300000E 01
4	0,22879357E 03	0,46148119E 03	0,18990996E 03	0,38883612E 02	0,24300000E 01
5	0,44217655E 03	0,91824536E 03	0,37787875E 03	0,64297795E 02	0,24300000E 01
6	0,98195499E 03	0,20288579E 04	0,83492094E 03	0,14703406E 03	0,24300000E 01
GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,35693439E 02	0,35734133E 02	0,14348608E 02	0,40494313E-01	0,21385526E 02
2	0,85318989E 02	0,85362080E 02	0,15196938E 02	0,43090529E-01	0,70165142E 02
3	0,11820178E 03	0,11824715E 03	0,15996060E 02	0,45373861E-01	0,10225110E 03
4	0,24544254E 03	0,24548976E 03	0,16696196E 02	0,47219347E-01	0,22879357E 03
5	0,45951478E 03	0,45956405E 03	0,17387502E 02	0,49264361E-01	0,44217655E 03
6	0,99957642E 03	0,99962652E 03	0,17671531E 02	0,50099327E-01	0,98195499E 03

AVERAGE MICROSCOPIC CROSS SECTIONS
(SELF-SHIELDED)

URANIUM 235 NASA REPORT DEC 1964

GROUP	SIGMA ABSORPTION	NL SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,21144060E 02	0,35880099E 02	0,14765473E 02	0,63785873E 01	0,24300000E 01
2	0,68345340E 02	0,14075079E 03	0,57922136E 02	0,10423203E 02	0,24300000E 01
3	0,98482800E 02	0,20890381E 03	0,85968646E 02	0,12514153E 02	0,24300000E 01
4	0,20971361E 03	0,42289140E 03	0,17402938E 03	0,35684225E 02	0,24300000E 01
5	0,37283177E 03	0,77426582E 03	0,31862791E 03	0,54213861E 02	0,24300000E 01
6	0,67700885E 03	0,13987961E 04	0,57563624E 03	0,10137261E 03	0,24300000E 01
GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,35290092E 02	0,35330327E 02	0,14186266E 02	0,40234496E-01	0,21144060E 02
2	0,83105630E 02	0,83148605E 02	0,14803265E 02	0,41974686E-01	0,68345340E 02
3	0,11384971E 03	0,11389343E 03	0,15410628E 02	0,43714825E-01	0,98482800E 02
4	0,22500088E 03	0,22504432E 03	0,15330716E 02	0,43444592E-01	0,20971361E 03
5	0,38750293E 03	0,38754473E 03	0,14712957E 02	0,41793640E-01	0,37283177E 03
6	0,68915793E 03	0,68919248E 03	0,12183637E 02	0,34540981E-01	0,67700885E 03

AVERAGE MICROSCOPIC CROSS SECTIONS
(UNSELF-SHIELDED)

URANIUM 238 NASA REPORT JANUARY 1965

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,47612393E 00	0,	0,	0,47612393E 00	0,
2	0,54723237E 00	0,	0,	0,54723237E 00	0,
3	0,67869070E 00	0,	0,	0,67869070E 00	0,
4	0,10282081E 01	0,	0,	0,10282081E 01	0,
5	0,18801258E 01	0,	0,	0,18801258E 01	0,
6	0,37556647E 01	0,	0,	0,37556647E 01	0,

GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,11176068E 02	0,11206124E 02	0,10730000E 02	0,30056119E-01	0,47612393E 00
2	0,11247176E 02	0,11277232E 02	0,10730000E 02	0,30056119E-01	0,54723237E 00
3	0,11378634E 02	0,11408691E 02	0,10730000E 02	0,30056118E-01	0,67869070E 00
4	0,11728152E 02	0,11758208E 02	0,10730000E 02	0,30056117E-01	0,10282081E 01
5	0,12580069E 02	0,12610126E 02	0,10730000E 02	0,30056119E-01	0,18801258E 01
6	0,14455608E 02	0,14485665E 02	0,10730000E 02	0,30056124E-01	0,37556647E 01

AVERAGE MICROSCOPIC CROSS SECTIONS
(SELF-SHIELDED)

URANIUM 238 KASA REPORT JANUARY 1965

GROUP	SIGMA ABSORPTION	NU SIGMA FISSION	SIGMA FISSION	SIGMA CAPTURE	AVERAGE NU
1	0,47074375E 00	0,	0,	0,47074375E 00	0,
2	0,53304382E 00	0,	0,	0,53304382E 00	0,
3	0,65381905E 00	0,	0,	0,65381905E 00	0,
4	0,94212291E 00	0,	0,	0,94212291E 00	0,
5	0,15864052E 01	0,	0,	0,15864052E 01	0,
6	0,25893430E 01	0,	0,	0,25893430E 01	0,

GROUP	SIGMA TRANSPORT	SIGMA TOTAL	SIGMA P0 SCATTER	SIGMA P1 SCATTER	SIGMA REMOVAL
1	0,11049820E 02	0,11079537E 02	0,10608793E 02	0,29717042E-01	0,47074375E 00
2	0,10955882E 02	0,10985160E 02	0,10452116E 02	0,29278005E-01	0,53304382E 00
3	0,10962291E 02	0,10991248E 02	0,10337429E 02	0,28957517E-01	0,65381905E 00
4	0,10768822E 02	0,10796480E 02	0,98543570E 01	0,27657655E-01	0,94212291E 00
5	0,10641125E 02	0,10666625E 02	0,90802195E 01	0,25499611E-01	0,15864052E 01
6	0,99664190E 01	0,99871413E 01	0,73977984E 01	0,20722195E-01	0,25893430E 01

VII. DESCRIPTION OF THE PRESENT LIBRARY TAPE

The present library tape contains cross section data for several materials at 101 energy points. However, INCITE allows a completely arbitrary energy point structure with a maximum of 101 energy points. The LIBMOD code, which is presently under development, will permit the user to modify, make additions to, or create cross section libraries for materials. These tasks will be accomplished by reading cross section data from cards or by using built-in routines to calculate $1/v$ absorption or fission cross sections, free gas scattering matrices, or $1/E$ slowing down sources. The program will also possess options for forming mixtures of material data as well as condensing an existing library tape.

The following types of data appear on the library tape for each material k and fine group I . However, only reactions which are pertinent to a particular material are included in its library.

- First data group
- (a) Material identification number (ISO).
  - (b) Material description (WORD(I), $I = 1,12$)
  - (c) Absorption flag (NABS)
  - (d) Fission flag (NFIS)
  - (e) Scattering flag (NSCT)
  - (f) Kernel flag (NKER)
  - (g) Average number of neutrons per fission (ANU)

- Second data group
- (a) If NABS=1; absorption cross sections $\left(\sigma_{aI}^k \right)$
  - (b) If NFIS=1; fission cross sections $\left(\sigma_{fI}^k \right)$
  - (c) If NSCT=1; P_0 and P_1 total scattering cross sections $\left(\sigma_{s0I}^k \text{ and } \sigma_{s1I}^k \right)$
  - (d) If NKER=1; P_0 and P_1 sources into energies below E_c $\left(S_{0I}^k \text{ and } S_{1I}^k \right)$

The following data groups are present only if NKER=1, ie, if a scattering matrix is furnished for this material.

Third data group   Flags indicating the size of the scattering matrix

Fourth data group P_0 differential scattering matrix $\begin{pmatrix} \sigma^k \\ s0_{I \rightarrow J} \end{pmatrix}$

Fifth data group P_1 differential scattering matrix $\begin{pmatrix} \sigma^k \\ sl_{I \rightarrow J} \end{pmatrix}$

1. MATERIALS ON LIBRARY TAPE

<u>Material Identification No.</u>	<u>Material Description</u>	<u>Material Identification No.</u>	<u>Material Description</u>
1.00293	H ₂ O Molecule 300 K (McMurry-Russell Kernel)	16.0	Sulfur
1.00295	H ₂ O Molecule 295 K (Nelkin Kernel)	19.0	Potassium
1.00302	Hydrogen in H ₂ O 300 K (Nelkin Kernel)	20.0	Calcium
1.00303	Hydrogen in Polyethylene 300 K (Parks Kernel)	22.0	Titanium
1.00304	Hydrogen in ZrH 300 K (GAKER Kernel)	23.0	Vanadium
1.20301	Deuterium in D ₂ O 300 K (Gas Kernel)	24.0	Chromium
1.20302	Deuterium in D ₂ O 300 K (Nelkin Kernel)	25.0	Manganese (Natural)
3.0	Lithium (Natural)	26.0	Iron (Natural)
3.006	Lithium-6	27.0	Cobalt (Natural)
3.007	Lithium-7	28.0	Nickel (Natural)
4.00302	Beryllium (Metal) 300 K (SUMMIT Kernel)	29.0	Copper (Natural)
5.0	Boron (Natural)	30.0	Zinc
5.01	Boron-10	31.0	Gallium
6.00302	Carbon 300 K (Parks Kernel)	34.0	Selenium
6.00602	Carbon 600 K (Parks Kernel)	34.082	Selenium-82
6.00902	Carbon 900 K (Parks Kernel)	35.081	Bromine-81
6.01202	Carbon 1200 K (Parks Kernel)	36.083	Krypton-83
7.0	Nitrogen	36.084	Krypton-84
8.00303	Oxygen 303 K (Gas Kernel)	36.085	Krypton-85
8.00873	Oxygen 873 K (Gas Kernel)	36.086	Krypton-86
9.0	Flourine	37.0	Rubidium (Natural)
11.0	Sodium	37.085	Rubidium-85
12.0	Magnesium	37.087	Rubidium-87
13.0	Aluminum	38.088	Strontium-88
14.0	Silicon	38.09	Strontium-90
15.0	Phosphorus	39.089	Yttrium-89 (Natural)
		39.091	Yttrium-91
		40.0	Zirconium (Natural)
		40.091	Zirconium-91
		40.092	Zirconium-92
		40.093	Zirconium-93
		40.094	Zirconium-94

<u>Material Identification No.</u>	<u>Material Description</u>	<u>Material Identification No.</u>	<u>Material Description</u>
40.096	Zirconium-96	49.0	Indium (Natural)
41.093	Niobium-93 (Natural)	49.113	Indium-113
42.0	Molybdenum (Natural)	49.115	Indium-115
42.095	Molybdenum-95	50.0	Tin (Natural)
42.096	Molybdenum-96	52.126	Tellurium-126
42.097	Molybdenum-97	52.128	Tellurium-128
42.098	Molybdenum-98	52.13	Tellurium-130
42.1	Molybdenum-100	53.127	Iodine-127 (Natural)
43.099	Technetium-99	53.129	Iodine-129
44.1	Ruthenium-100	53.131	Iodine-131
44.101	Ruthenium-101	54.131	Xenon-131
44.102	Ruthenium-102	54.132	Xenon-132
44.104	Ruthenium-104	54.133	Xenon-133
45.103	Rhodium-103 (Natural)	54.134	Xenon-134
46.104	Palladium-104	54.135	Xenon-135
46.105	Palladium-105	54.136	Xenon-136
46.106	Palladium-106	55.133	Cesium-133 (Natural)
46.107	Palladium-107	55.134	Cesium-134
46.108	Palladium-108	55.135	Cesium-135
46.11	Palladium-110	55.137	Cesium-137
47.109	Silver-109	56.138	Barium-138
48.0	Cadmium (Natural)	57.139	Lanthanum-139
48.11	Cadmium-110	58.14	Cerium-140
48.111	Cadmium-111	58.142	Cerium-142
48.112	Cadmium-112	59.141	Praseodymium-141 (Natural)
48.113	Cadmium-113	60.143	Neodymium-143
48.114	Cadmium-114	60.144	Neodymium-144
48.116	Cadmium-116	60.145	Neodymium-145

<u>Material Identification No.</u>	<u>Material Description</u>	<u>Material Identification No.</u>	<u>Material Description</u>
60.146	Neodymium-146	66.161	Dysprosium-161
60.148	Neodymium-148	66.162	Dysprosium-162
60.15	Neodymium-150	66.163	Dysprosium-163
61.147	Promethium-147	66.164	Dysprosium-164
61.148	Promethium-148	68.167	Erbium-167
61.1482	Promethium-148	71.0	Lutetium (Natural)
62.0	Samarium (Natural)	71.175	Lutetium-175
62.147	Samarium-147	71.176	Lutetium-176
62.148	Samarium-148	72.0	Hafnium (Natural)
62.149	Samarium-149	72.174	Hafnium-174
62.15	Samarium-150	72.176	Hafnium-176
62.151	Samarium-151	72.177	Hafnium-177
62.152	Samarium-152	72.178	Hafnium-178
62.154	Samarium-154	72.179	Hafnium-179
63.0	Europium (Natural)	72.18	Hafnium-180
63.151	Europium-151	73.0	Tantalum (Natural)
63.153	Europium-153	73.18	Tantalum-180
63.154	Europium-154	73.181	Tantalum-181
63.155	Europium-155	74.0	Tungsten (Natural)
64.0	Gadolinium (Natural)	74.18	Tungsten-180 (Wolfram)
64.154	Gadolinium-154	74.182	Tungsten-182 (Wolfram)
64.155	Gadolinium-155	74.183	Tungsten-183 (Wolfram)
64.156	Gadolinium-156	74.184	Tungsten-184 (Wolfram)
64.157	Gadolinium-157	74.186	Tungsten-186 (Wolfram)
64.158	Gadolinium-158		
64.16	Gadolinium-160		
65.159	Terbium-159 (Natural)		
66.0	Dysprosium (Natural)		

<u>Material Identification No.</u>	<u>Material Description</u>	<u>Material Identification No.</u>	<u>Material Description</u>
75.0	Rhenium (Natural)	92.235	Uranium-235
75.185	Rhenium-185	92.236	Uranium-236
75.187	Rhenium-187	92.237	Uranium-237
77.0	Iridium (Natural)	92.238	Uranium-238
77.191	Iridium-191	93.237	Neptunium-237
77.193	Iridium-193	93.239	Neptunium-239
79.197	Gold-197 (Natural)	94.239	Plutonium-239
80.0	Mercury (Natural)	94.240	Plutonium-240
80.196	Mercury-196	94.241	Plutonium-241
80.198	Mercury-198	94.242	Plutonium-242
80.199	Mercury-199	95.241	Americium-241
80.2	Mercury-200	95.243	Americium-243
80.201	Mercury-201	102.0	Th-232 (Non-saturating Fission Products)
80.202	Mercury-202	123.0	U-233 (Non-saturating Fission Products)
82.0	Lead	125.0	U-235 (Non-saturating Fission Products)
90.232	Thorium-232	128.0	U-238 (Non-saturating Fission Products)
91.231	Protactinium-231	149.0	Pu-239 (Non-saturating Fission Products)
91.233	Protactinium-233	201.0	Hastelloy-X (Macroscopic)
92.232	Uranium-232	202.0	Stainless Steel (West)
92.233	Uranium-233 (12-20-62)	203.0	Stainless Steel (Type 347)
92.2332	Uranium-233 (NSE, Vol 16, p 2, 1963)		
92.2333	Uranium-233 (Oct, 1964)		
92.234	Uranium-234		

2. INITIAL ENERGY POINT STRUCTURE

<u>Energy Point</u>	<u>Energy (eV)</u>	<u>Energy Width</u>	<u>Energy Point</u>	<u>Energy (eV)</u>	<u>Energy Width</u>
1	0.001	0.0010	26	0.18	0.02
2	0.002	0.0015	27	0.20	0.02
3	0.004	0.0015	28	0.22	0.015
4	0.005	0.0015	29	0.23	0.01
5	0.007	0.0015	30	0.24	0.01
6	0.008	0.0015	31	0.25	0.01
7	0.010	0.0035	32	0.26	0.01
8	0.015	0.005	33	0.27	0.01
9	0.020	0.00515	34	0.28	0.01
10	0.0253	0.0050	35	0.29	0.01
11	0.030	0.00735	36	0.30	0.01
12	0.040	0.01	37	0.31	0.01
13	0.050	0.01	38	0.32	0.01
14	0.060	0.0075	39	0.33	0.01
15	0.065	0.005	40	0.34	0.01
16	0.070	0.005	41	0.35	0.01
17	0.075	0.005	42	0.36	0.015
18	0.080	0.005	43	0.38	0.027
19	0.085	0.005	44	0.414	0.02
20	0.090	0.005	45	0.42	0.008
21	0.095	0.005	46	0.43	0.015
22	0.100	0.0125	47	0.45	0.015
23	0.120	0.02	48	0.46	0.01
24	0.140	0.02	49	0.47	0.0075
25	0.160	0.02	50	0.475	0.005

<u>Energy Point</u>	<u>Energy (eV)</u>	<u>Energy Width</u>	<u>Energy Point</u>	<u>Energy (eV)</u>	<u>Energy Width</u>
51	0.48	0.0075	77	1.06	0.01
52	0.49	0.01	78	1.07	0.01
53	0.50	0.021	79	1.08	0.01
54	0.532	0.025	80	1.09	0.015
55	0.55	0.0215	81	1.11	0.0175
56	0.575	0.02	82	1.125	0.01
57	0.59	0.0125	83	1.13	0.0125
58	0.60	0.0175	84	1.15	0.035
59	0.625	0.025	85	1.20	0.05
60	0.65	0.029	86	1.25	0.05
61	0.683	0.025	87	1.30	0.05
62	0.70	0.0335	88	1.35	0.07
63	0.75	0.05	89	1.44	0.075
64	0.80	0.05	90	1.5	0.08
65	0.85	0.038	91	1.6	0.10
66	0.876	0.02	92	1.7	0.09
67	0.89	0.017	93	1.78	0.08
68	0.91	0.02	94	1.86	0.06
69	0.93	0.02	95	1.9	0.07
70	0.95	0.02	96	2.0	0.10
71	0.97	0.015	97	2.1	0.10
72	0.98	0.01	98	2.2	0.095
73	0.99	0.01	99	2.29	0.065
74	1.0	0.0175	100	2.33	0.045
75	1.025	0.025	101	2.38	0.025
76	1.05	0.0175			

VIII. OPERATING INSTRUCTIONS

The INCITE code has been written in FORTRAN-IV and consists of three computational links and a main link. It was originally written for the IBM-7040 and requires a 32k memory and utilizes the standard FORTRAN-IV monitor system with a chain feature.

The code uses six tapes which are allocated as follows:

<u>Description of Tape</u>	<u>FORTRAN Logical Tape Number</u>
Input Tape	5
Output Tape	6
Cross Section Library Tape	9
Scratch Tape	3
" "	4
" "	8

The library tape on FORTRAN logical 9 is a binary data tape and contains both numeric and alphanumeric data. A code, LIBMOD, is presently being written for the maintenance of this tape.

IX. INCITE -- SIMPLIFIED BLOCK DIAGRAM

X. REFERENCES

1. H. L. McMurry, G. J. Russell, R. M. Brugger, "Slow-Neutron Scattering by Water", Nucl. Sci. Eng., 25 (July 1966) pp 248-260.
2. G. D. Joanou, C. V. Smith, H. A. Vieweg, GATHER-II: An IBM-7090 FORTRAN-II Program for the Computation of Thermal-Neutron Spectra and Associated Multigroup Cross Section, GA-4132 (July 1963).
3. D. Selengut, Critical Mass Calculations for Bare Hydrogen Moderated Reactors by Means of Transport Theory, APEX-121 (September 1952).
4. R. A. Grimesey, F. E. Mullen, L. J. Gannon, MUFT Revision - A Fast Neutron Spectrum Code for the IBM-650, IDO-16735 (March 1962).
5. R. S. Varga, Matrix Iterative Analysis, Chapter 3, Basic Iterative Methods and Comparison Theorems, Englewood Cliffs: Prentice-Hall Inc., 1962.
6. A.M. Weinberg and E. P. Wigner, The Physical Theory of Neutron Chain Reactors, Chicago: The University of Chicago Press (1958) p 381.
7. C. W. Maynard, Blackness Theory and Coefficients for Slab Geometry, WAPD-TM-168 (May 1959).
8. A. F. Henry, A Theoretical Method for Determining the Worth of Control Rods, WAPD-218 (August 1959).
9. A. D. Voorhis et al, Experimental and Theoretical Study of Critical Slabs: Effect of Absorbing Membranes of Cadmium, Gold, and Boron, WAPD-170 (April 1957).
10. H. Bohl, Jr., and A. P. Hemphill, MUFT-5- A Fast Neutron Spectrum Program for the Philco-2000, WAPD-TM-218 (February 1961).
11. A. Radkowsky, ed., Naval Reactors Physics Handbook. Volume I. Selected Basic Techniques, TID-7050 (1964).
12. R. L. Curtis and R. A. Grimesey, LIBMOD - A FORTRAN IV Program To Modify and Update the INCITE Cross Section Library, IN-1123, (to be published).