BNWL-462
UC-80

ALTHAEA
A ONE-DIMENSIONAL TWO-GROUP
DIFFUSION CODE WITH AN EFFECTIVE
FOUR-GROUP BURNUP

E. T. Merrill

MAY 1971

AEC RESEARCH &
DEVELOPMENT REPORT

BNWL-462

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UNITED STATES ATOMIC ENERGY COMMISSION UNDER CONTRACT AT(45-1)-1830

3 3679 00060 5941

BNWL-462
UC-80, Reactor Technology

ALTHAEA
A ONE-DIMENSIONAL TWO-GROUP DIFFUSION CODE
WITH AN EFFECTIVE FOUR-GROUP BURNUP

By
E. T. Merrill
Systems Engineering Department
Physics and Engineering Division

May 1971

BATTELLE MEMORIAL INSTITUTE
PACIFIC NORTHWEST LABORATORIES
RICHLAND, WASHINGTON 99352

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

ABSTRACT

The ALTHAEA code systems calculate nuclear reactor fuel burnups with reasonable accuracy and at computer speeds sufficient for preliminary economic analysis of fuel cycle costs. This first version of the code describes four neutron energy groups, and collapses these to two groups for one-dimensional space diffusion calculations. It provides for cross shielding of the epithermal resonances. It is written in FORTRAN II for the IBM 7090, or FORTRAN IV for the Univac 1108 computer.

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SUMMARY

The ALTHAEA code systems calculate nuclear reactor burnups with reasonable accuracy and at computer speeds sufficient for initial fuel cycle cost analysis. The code performs an approximate solution of the time-dependent diffusion equation in one dimension. It uses either slab or cylindrical geometry and leaves control transient problems to a subroutine package. Xenon and samarium may be considered to be in equilibrium.

The flux level is assumed to remain constant during each arbitrary burnup step, but cross sections are automatically adjusted as a function of blackness, the spectrum, and changes in isotopic concentrations.

Auxiliary output tapes can be used by other codes to calculate fuel cycle costs and to produce plots of physics parameters as functions of the fuel exposure.

The code may deal with up to 45 isotopes, which are arbitrarily divided into four arrays for coding convenience. A study involving only uranium and plutonium isotopes can be made at considerable speed.

Estimates of reactivity and burnup reaction rates are improved by making corrections for variations of epithermal flux from the assumed constant flux per unit lethargy. Corrections are also provided for resonance self-shielding and cross-shielding or interference. This representation provides more accurate treatment for transuranium isotopes, particularly Np^{237} , Pu^{239} , and Pu^{241} .

The code is calibrated in the zero dimensional mode by adjusting certain constants to fit data from critical experiments. Based on these results the one-dimensional mode can be calibrated by selecting the transverse buckling and the transport cross sections. The code may also be calibrated theoretically by matching its results against more rigorous codes such as the Hanford Monte Carlo Reactor Burnup Code and against data obtained from normalization of results from Hanford Revised GAM and THERMOS.

This report describes the basis for the ALTHAEA code using a 2-group diffusion theory and effective 4-group burnup. The code has recently been developed into a version which includes 4-group diffusion calculations and 4-group cross sections.

Calculations by ALTHAEA of concentrations of the higher transuranium isotopes agree well with actual concentrations produced in Yankee fuel rods.

A CHAIN arrangement of programs permits continuous calculations on one problem that would not otherwise be possible within the memory limits of the 7090 computer. Twelve parts of the CHAIN system are described.

ALTHAEA
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INTRODUCTION

In 1961 a reactor fuel burnup code called MELEAGER was written in Fortran II for the IBM-7090. Its primary output was fuel composition, reactivity, and power as a function of exposure for surveys of plutonium recycle in thermal reactors. A recognized need for calibration to a wide range of available experimental data has required updating of the code. A one-dimensional capability has been added and this revision has been called "ALTHAEA." It accommodates fewer isotopes than MELEAGER but the running time in the zero-dimensional mode has been reduced. The MELEAGER code required about 1 minute of 709 time for 20 isotopes exposed to 10,000 MWd/ton while ALTHAEA typically requires 10 seconds of 7090 time for 20 isotopes exposed to 20,000 MWd/ton. The current compilation of the code handles 20 fuel isotopes and 15 fission products, accounts for all major forms of nuclear destruction, and prints out intermediate compositions at any desired interval. In addition, it accounts for self-shielding and cross-shielding of all the isotope cross sections. When operating in the one-dimensional mode, the code is capable of handling up to 24 burnable regions.

This code is a "workhorse" code and its makeup has, to a significant degree, been altered to allow better fitting of burnup data, with the strong restraint that computation time should be less than 20 seconds of 7090 time for a 20,000 MWd/ton burnup. In effect, by using some adjustable coefficients, ALTHAEA approximates the accuracy of more basic codes, without any decrease in its computational speed.

Codes using more groups than ALTHAEA can also handle a wide variety of problems, but calibrating them to burnup data and simple flux spectrum data is fraught with the difficulty of selecting the right group or groups to adjust. The specific four-group structure used by ALTHAEA

was chosen mainly because one can visualize physically a fast, a slowing down, an epithermal-resonance, and a thermal flux breakdown, and one can reasonably estimate the adjustment required to fit burnup data. In addition, it was found that cross shielding of the low-lying transuranium isotope resonances, principally those of Pu^{239} , Pu^{241} , Np^{237} , and Am^{241} , was necessary to allow fitting of the data. The other major adjustable coefficients in the code involve resonance self-shielding, moderator effectiveness, calculation of effective neutron temperature, and thermal flux disadvantage factors.

CROSS SECTIONS AND FLUX APPROXIMATIONS

The Boltzmann transport equation is one of the fundamental relationships that describe a critical operating nuclear reactor. It is not normally solved by present day computers in complete detail for the entire reactor core but rather, various approximations are used. In the ALTHAEA burnup code it is assumed that the diffusion approximation will apply to a macroscopic description of the reactor core. The reactor can be described as consisting of up to 24 concentric cylindrical or adjacent slab macroscopic regions. Each of the regions requires homogenization to obtain the flux weighted cross sections and cell averaged parameters needed for the diffusion equation.

Cross Sections

The cross sections in the ALTHAEA code are derived from the infinitely dilute cross sections using a highly modified Westcott⁽¹⁾ system. (See also Appendix C). The dilute cross sections are altered by equations which have been shown to have the correct trends to account for the heterogeneities and resonance effects of the actual region. The coefficients for these equations are derived from slowing down, thermalization and transport calculations. In this manner it is possible to calibrate the thermal and epithermal reaction rates so that they are initially correct and so that as the burnup calculation progresses the effective cross sections and fluxes will be altered in the correct directions.

Flux Calculations

The time dependency of the diffusion equation is approximated by assuming that the transient or short-term effects can be ignored, and that the xenon and samarium can be accounted for by means of their equilibrium concentrations and cross sections, depending on the specific power in each homogenized region. Instantaneous solutions of the resulting diffusion equation, obtained by the forward substitution on backward elimination technique as developed by Keppler and Orr,⁽²⁾ are used to determine the two-group flux profile of the core, and this profile is normalized to provide the specified total core power. This solution of the fluxes is then utilized in each region to set the power level and to establish the spectral index by means of the relationship of the epithermal-to-thermal fluxes.

Regional flux and spectral index are assumed to remain constant for a finite period for the burnup calculation. The errors introduced by this assumption are larger for long periods, and smaller for shorter periods. At each calculation the error is reported for inspection by the user, whereupon the burnup time interval can be respecified. The burnup calculation is carried out sequentially for each region of the core for the specified period, after which the diffusion calculation is repeated with the new concentrations, and after which in turn, a burnup step is calculated, etc. During burnup the epithermal flux is corrected to three groups with self- and shadow-shielding to obtain the proper reaction rates in plutonium isotopes with large low-lying resonances and to properly account for the fast effect.

Burnup Calculations

The effective, or flux weighted, cross sections are not necessarily assumed to remain constant during the period but instead are assumed to vary according to the equations which had been initially calibrated to account both for thermal flux depression in the fuel and for the resonance self-shielding and resonance shadowing (cross-shielding) of competing isotopes. A means of obtaining the values for these equations (and the equations themselves) is discussed in a later section (see pages 9-17). Thus an

appropriate flux and cross section may be obtained to describe transmutation rates due to neutron events. Transmutation rates due to alpha, beta, electron capture or other neutron flux independent events may also be included in the data to be processed. The total rate of change of the concentration of any isotope may be obtained as the sum over all isotopes of the transmutation rates to isotope "i" minus the destruction rate of isotope "i." The burnup equation which is solved is of the form

$$\frac{dY_i}{dt} = \sum_{j=1-N11} Y_j \left[\lambda(j \rightarrow i) + \phi_w \hat{\sigma}(j \rightarrow i) \right] - 2Y_i \left[\lambda(i \rightarrow i) + \phi_w \hat{\sigma}_a(i \rightarrow i) \right] \quad (1)$$

where

$\frac{d}{dt} Y_i$ = the time rate of change of the concentration of the i^{th} isotope.

Y_j = the concentration of the j^{th} isotope in nuclei/barn centimeter where the value of j ranges over all the fuel isotopes.

$\lambda(j \rightarrow i)$ = the probability that isotope j will decay to form isotope i (per unit time).

ϕ_w = the Westcott flux.

$\hat{\sigma}(j \rightarrow i)$ = the Westcott form of the probability of forming the i^{th} isotope from the j^{th} isotope by a neutron event per unit of flux-time.

Y_i = concentration of i^{th} isotope.

$\lambda(i \rightarrow i)$ = the probability that isotope i will be destroyed by decay.

$\hat{\sigma}_a(i \rightarrow i)$ = the probability that isotope i will be destroyed by a neutron event.

Using the matrix form to simplify representation of transmutations, Equation (1) may be given as follows:

$$\frac{d}{dt} \begin{bmatrix} Y_1 \\ \vdots \\ Y_i \\ \vdots \\ Y_{List} \end{bmatrix} = \begin{bmatrix} V_{11} & \cdots & V_{1j} & \cdots & V_{1List} \\ \vdots & & \vdots & & \vdots \\ V_{i1} & \cdots & V_{ij} & \cdots & V_{iList} \\ \vdots & & \vdots & & \vdots \\ V_{List1} & \cdots & V_{Listj} & \cdots & V_{ListList} \end{bmatrix} \begin{bmatrix} Y_1 \\ \vdots \\ Y_j \\ \vdots \\ Y_{List} \end{bmatrix} - 2 \begin{bmatrix} V_{11} & \cdots & 0 \\ \vdots & & \vdots \\ 0 & \cdots & V_{ii} & \cdots & 0 \\ \vdots & & \vdots & & \vdots \\ 0 & \cdots & 0 & \cdots & V_{ListList} \end{bmatrix} \begin{bmatrix} Y_1 \\ \vdots \\ Y_i \\ \vdots \\ Y_{List} \end{bmatrix} \quad (2)$$

where

$$V_{ij} = \lambda_{(j \rightarrow i)} + \phi_w \hat{\sigma}_{(j \rightarrow i)}$$

$$List \equiv N+1$$

The cross sections and decay constants are stored in square arrays with the $\hat{\sigma}(j, i)$ representing the transmutation cross section of isotope j to form isotope i . The diagonal of this array is a convenient place to store the total absorption cross section of isotope i . The rate of production of an isotope is found by summing reactions across a row and subtracting the contribution due to the diagonal. The rate of change of concentration of the isotope then is obtained as the rate of production minus the rate of destruction. Thus the rate is the sum of the row minus twice the diagonal as indicated in Equation (1).

The effective Westcott cross sections in the equations depend on the thermal flux depression (a function of cell size and blackness) and on the epithermal resonance self-shielding and "cross-shielding" effects. The net result is that the effective Westcott cross sections for any isotope depends on the concentrations of all other isotopes. The total equation involves n simultaneous differential equations of the form of Equation (2) for a burnup calculation involving n isotopes. The "Gil version of Runge-Kutta"⁽³⁾ is used to perform a stepwise integration of these equations.

This method is described briefly on page 57. Error limits are required as input to control the length of the time steps (the Runge-Kutta calculation will in general take several time steps for each diffusion calculation).

ALTHAEA may deal with up to 40 isotopes, which are arbitrarily divided into four arrays for coding convenience.

The user of the ALTHAEA code may select up to 20 fuel isotopes (Array 1 isotopes) any one of which may decay to one or more of the others and any one of which may form one or more of the others by a neutron event. Thus, a rapid study can be made of only the usual two uranium and four plutonium isotopes; or, at a sacrifice of speed, the list of isotopes may be extended to 20 to include transplutonium isotopes in a more detailed study.

The fuel isotopes may have fission cross sections. Most other isotopes such as fission products and burnable poisons which do not have fission cross sections and which are not related to the fuel isotopes by radioactive decay events or neutron events are Array 2 isotopes. Any Array 2 isotope may decay to one or more other Array 2 isotopes and may form one or more other Array 2 isotopes by neutron events.

Fission products with high cross sections such that they may be considered saturated are Array 3 isotopes. There may be included up to ten Array 2 isotopes and up to five Array 3 isotopes. Any fissionable Array 1 isotope may be assigned a set of fractional fission yields to form one or more Array 2 or Array 3 isotopes.

Array 4 isotopes are a subset of Array 1 isotopes and will be described later under Special Options.

CROSS SECTIONS AND REACTION RATES

The cross section system used is based on dilute cross sections in the form proposed by Westcott.⁽¹⁾

$$\hat{\sigma} = \sigma_0 (g + rs)$$

Application of this idealized format for reactor cross sections requires considerable adjustment to the values of g , r , and s and also to the neutron temperature, which determines the division between the spectrum components g and s . A description of the Westcott system is given in Appendix C. The following paragraphs are devoted to the modifications which must be applied to the basic system to allow for deviations from the ideal due to the heterogeneous nature of a real system.

Thermal Cross Sections

In order to calculate the proper flux traverses and reaction rates, corrections must be applied to the dilute cross sections. The dilute cross section, defined as the cross section of a single isolated atom exposed to the idealized Maxwellian and $1/E$ slowing down neutron fluxes, is calculated as described in a later section on calibration. We will first consider the thermal flux and reaction rate. The desired flux is the cell average flux but the reaction rate is reduced by a disadvantage or flux depression factor. The reduction is calculated by applying the factor to g . This factor must be determined experimentally or by a P-3 or "S" type transport code.

If a burnup calculation is to be made, it is not sufficient to obtain only the initial thermal flux depression factor, but it is also necessary to obtain a "fit" of the thermal flux depression factor as a function of the fuel "blackness," since the blackness changes as the fuel concentration, and the composition is continually changing during the burnup cycle.

The following equation has been found to reproduce the thermal flux depression factor originally computed by transport theory as a function of the cell blackness.

$$\phi_c / \phi_f = \text{FCONST } \bar{\Sigma}_a + e^{-\text{FEXP } \bar{\Sigma}_a} \quad (4)$$

where

ϕ_c = the averaged cell flux

ϕ_f = the averaged fuel flux

$\bar{\Sigma}_a$ = the total thermal absorption cross section of the fuel

and

FCONST and FEXP are constants derived by the user from experimental data or computed by the JASON code⁽⁴⁾ in Battelle-Northwest Chain system.

This expression has been applied to a number of fuel elements, including those moderated with graphite or light or heavy water, and it correlates with transport theory calculations to 1 to 2% for an order of magnitude change in absorption cross section. The two constants have a

large leverage on the effective thermal group cross section. The effective value of g is given as

$$g_{i(\text{effective})} = \frac{g_i}{(\text{RTM})(\text{FCONST})(\text{SUM3}) + e^{-(\text{RTM})(\text{FEXP})(\text{SUM3})}} \quad (5)$$

where

$g_{i(\text{effective})}$ = the effective value of g for isotope i .
 g_i = the uncorrected value of g for isotope i .

$$\text{RTM} = \sqrt{\frac{293\pi}{4T}}$$

T = the modal temperature of the Maxwell distribution (see p. 18).

SUM3 = the uncorrected macroscopic absorption cross section.

The uncorrected absorption cross section is obtained by summing over all the fuel isotopes as follows:

$$\text{SUM3} = \sum_{\text{fuel}} Y_i \sigma_{oi} g_i \quad (6)$$

We have found that it is possible to calibrate the computations for the thermal group rather well and predict experimental burnup data by means of a careful choice of the parameters FCONST and FEXP in the otherwise homogeneous model.

Epithermal Cross Section

Next, attention is directed to the epithermal group where resonances of the fuel isotopes cause depressions in the flux, with spatial and energy dependence. This departure of flux from the assumed profile affects the absorptions of a single isotope and is frequently called self-shielding. A self-shielding formulation⁽⁵⁾ frequently applied to the epithermal absorbers for a single resonance is

$$I_{\text{effective}} = \frac{I_{\text{dilute}}}{\left(1 + \frac{YZ}{\text{SCA}}\right)^{1/2}} \quad (7)$$

where

- I = resonance integral
- Y = concentration of the isotope
- Z = effective height of the resonance
- SCA = FORTRAN acronym for resonance shielding index
calculated from the following expression as derived
by Bell:⁽⁶⁾

$$SCA = \Sigma_s^{\text{fuel}} + \left[\frac{4V}{S} + \frac{1}{[\Sigma_s^{\text{moderator}} (\text{Vol. mod. / Vol. fuel})]} \right]^{-1} \quad (8)$$

where $\frac{4V}{S}$ is the effective fuel element diameter.

Effective Resonance Integrals

A comparison of the effective resonance integrals for Pu^{240} computed from Equation (7) with the experimental values determined by Nichols⁽⁷⁾ is presented in Figure 1. This is nearly the theoretical situation since 98% or so of the resonance integral is due to the single resonance near 1 V. For other isotopes with several resonances providing major contributions to the resonance integral some sort of average resonance height is required for the value of Z. In practice, we have selected a value of Z for the resonance with the largest contribution to the resonance integral and then made arbitrary adjustments to obtain reaction rates that reproduce experimental data.

Equation (7) accounts for self-shielding, but it does not provide for the interference between isotopes competing for the neutrons at the resonance energy. The importance of such interferences is dramatized by the fact that, atom for atom, Pu^{239} has as much effect as Pu^{241} (or even more) on the resonance integral of Pu^{241} . Since there are more Pu^{239} atoms than Pu^{241} atoms, it would be much safer to neglect the self-shielding of the Pu^{241} than to neglect the interference of the Pu^{239} . A comparison of the cross sections of Pu^{239} and Pu^{241} is presented in Figure 2.

Figure 1. Comparison of Shielding Equation with Experimental Data for Pu-240

FIGURE 2. Cross Sections of Pu^{239} and Pu^{241}
Versus Energy Level

These difficulties are partially overcome in the ALTHAEA code by the use of the following expression including cross-shielding terms to calculate the epithermal cross section

$$I_{\text{eff}} = \frac{\text{CINTER}_{ii} I_{\text{dil}}}{\left(1 + \frac{Y_i Z_i}{\text{SCA}}\right)^{1/2}} + \frac{(1.0 - \text{CINTER}_{ii}) I_{\text{dil}}}{\left(1.0 + \frac{\sum \text{CINTER}_{ji} Y_j Z_j}{\text{SCA}}\right)^{1/2}} \quad (9)$$

where

CINTER_{ii} = the fraction of the resonance integral not subject to cross-shielding.

CINTER_{ji} = the coefficient of interference of the j^{th} isotope on the i^{th} isotope (note the value of CINTER_{ji} is always equal to unity for $j = i$).

The user should note that this cross-shielding is applied only to the isotopes in one particular region at a time, and does not correct the reaction rates in adjacent regions. This would be a severe shortcoming if one tried to apply this code to cell calculations rather than the intended macroscopic regions of a reactor.

Doppler Broadening

Variation of the effective height of any resonance as a function of absorber temperature is described by the Doppler broadening or Psi function. This function has been evaluated and tabulated by Roe⁽⁸⁾ and Rose.⁽⁹⁾ Machine evaluation involves solution of partially convergent series equations. These equations are too time consuming for use in a survey code, so the tabulated data were plotted and an analytical expression derived that is capable of rapid machine evaluation with existing subroutines. The function is

$$Z = \frac{Z_o}{1.0 + 1.155 D_c^2 T_f / (0.4 + D_c \sqrt{T_f})} \quad (10)$$

where

Z_o = cross section of nucleus at rest for neutrons of resonant energy.

T_f = absolute temperature of the fuel in degrees Kelvin.

D_c = a constant derived from resonance parameters.

The value of Z_o may be computed from the resonance parameters on the BNW Master Library tape by Stewart⁽¹⁰⁾ as follows

$$Z_o = \frac{(2.62 \times 10^6)(g) \Gamma_n \Gamma_a}{E_o \Gamma^2} \quad (11)$$

where g = a statistical weight

Γ = the total width ($\Gamma_n + \Gamma_a$) of the resonance

Γ_n = the neutron width of the resonance.

Γ_a = the capture plus fission width of the resonance.

E_o = the energy at the resonance.

This method produces Z values which are 12% high for Th^{232} and 21% low for U^{238} based on resonance escape probability measurements. An alternate method of calculation used in some cases is based on the known value of Z at a given temperature and on Equation (10).

In addition to Z_o the value of D_c may also be calculated from resonance parameters on the BNW tape as follows

$$D_c = \frac{2}{\Gamma} \sqrt{\frac{E_o K}{A}} \quad (12)$$

where

K = the Boltzmann constant

A = atomic weight of the nucleus

Whenever it becomes feasible to measure cross sections of the isotopes at a second temperature, it will then be possible to calculate new values of Z and from these to calculate empirical values for D_c by the use of Equation (10).

The error in the calculated values of the denominator of Equation (10) compared to the tabulated values of Roe and Rose is plotted on Figure 3. Since Z appears as a square root in the formulation for the effective resonance integral, these errors will be halved and are thus well within 1% for all isotopes at temperatures of interest.

Simulation of Three Epithermal Groups

To improve estimates of reactivity and burnup reaction rates, it is necessary to make corrections for variations of epithermal flux from the assumed constant flux per unit lethargy. Inspection of plots of flux per unit lethargy for typical cells, as computed by the RBU Monte Carlo code, led to the arbitrary selection of three epithermal flux groups--fast, slowing down, and epithermal. Figure 4 is a schematic representation of a typical flux calculation in a water-moderated cell. The dotted lines represent the flux groups used in ALTHAEA.

The fast flux (from 0.2 to 6 MeV) is usually more than double the average slowing down flux, while the flux just above thermal energy is usually reduced by an amount comparable to the resonance escape probability. The code therefore accepts the resonance integral data in three parts and forms a corrected sum for use in the burnup routine. The group limits are not built into the code, so the selection must be made by the user as he prepares the cross section data tape. Battelle-Northwest has selected limits at 3.681 kT, 0.68256 eV, 0.2 MeV, and 6 MeV. The lower limit is recommended by Westcott. Calculation of the neutron temperature is discussed in the next section. The value 0.68256 eV was selected for two reasons: first because it coincides with existing limits in GAM and THERMOS libraries, and second, because about half of the resonance integral for an isotope with $1/v$ cross section lies above this value at reactor operating temperatures. The limits for the fast group are based on inspection of Monte Carlo flux plots.

Correction of the lower (0.68 eV) resonance integral is based on the fraction of the neutrons absorbed at higher energy and on input parameters as follows

FIGURE 3. Percent Error in Simple ψ Function Versus Tabulated ψ Function Values of Roe and Rose

FIGURE 4. Schematic Flux Plot of a Reference Water-Moderated Lattice

$$SSUM = SUMR + SUMA' * P - SIGSUM * b \quad (13)$$

$$SUM = (SIGSUM + SNF + R * SSUM) / RAYK2 \quad (14)$$

$$P = 1.0 - SUMR * R / SUM \quad (15)$$

where

SUMR = the sum of the effective resonance integrals above the group limit

SUMA' = the sum of the effective resonance integrals in the lower group

P = the desired correction factor

SIGSUM = the effective thermal group macroscopic absorption cross section

b = the flux shape factor (1.176246)

SNF = the nonfuel macroscopic cross section

R = a spectral index defined on page 26

RAYK2 = an input parameter

Equations (13 to 15) and (35) (See also page 26) are iterated to four significant figure convergence. A value of zero for RAYK2 reduces this formulation to the assumed constant slowing down density condition. Calibration calculations with other codes indicate that a value of 0.6 is appropriate for a number of reactor designs.

Further correction of the cross sections of certain isotopes may be made to account for deviations from the assumed flux. The isotopes involved must be consecutive in the isotope list, and are designated when the cross section library tape is made up. The correction factor is computed for each isotope in turn, using the value of SUM computed in Equation (14) as follows

$$P_i = 1.0 - SUMP * R / SUM \quad (16)$$

where P_i is the correction factor for isotope i and SUMP is the sum of the effective resonance integrals of the isotopes preceding i that are designated as being responsible for flux deviation. The present practice at Battelle-Northwest is to correct U^{238} for Th^{232} , U^{234} for Th^{232} and U^{238} ; and Pu^{240} for Th^{232} , U^{238} , and U^{234} .

ALTHAEA has no direct method for calculation of reaction rates due to fast flux. To obtain the proper reaction rates, the resonance integrals for the fast energy group must be corrected by a factor obtained by experimental measurement or by a calculation in a cell code.

Correction of the fast group contribution is based on the input fast effect for one isotope (usually U^{238}). The correction factor required to produce the proper reaction rates for this isotope is applied to the resonance integrals for fission, absorption, and n-2n reactions for all isotopes. The output value of epsilon represents the gain in neutrons, considering fissions and absorptions in all isotopes due to the fast flux. The value of nu in this calculation is obtained from the second column of the nu array and represents an average typical of the fast flux. (See Appendix G for library description.)

Any of the fuel isotopes may have an n-2n cross section and descendant designated in the input data. The n-2n cross section is entered as a resonance integral, and is then converted to an equivalent effective cross section for use in the burnup routine by using the fast effect factor calculated as just described.

The Calculation of Neutron Temperature

It has been tacitly assumed up to this point in the derivation that the neutron temperature is known. Very often the neutron temperature either is not known or is known only for the initial conditions. It has been shown by Dawson and Steves⁽¹¹⁾ that the modal temperature of the Maxwellian (herein referred to as the neutron temperature) is related to the moderator temperature and the ratio of the thermal absorption cross section to the thermal neutron source term and this relationship may be correlated by a constant which is a function of the isotope as in the following expression:

$$T_n = T_m \left[1 + \frac{C_i Y_i \sigma_{a_{oi}} \sqrt{\frac{T_o}{T_m}}}{\xi \Sigma_s} \right] \quad (17)$$

where

- T_n = Maxwellian modal or neutron temperature, °K
- T_m = moderator temperature, °K
- T_o = temperature corresponding to the velocity of 2200 m/s
- C_i = an arbitrary dimensionless constant dependent on the isotope
- $\sigma_{a_{oi}}$ = 2200 m/s microscopic cross section for isotope i
- Y_i = concentration of the isotope
- ξ = average gain in lethargy per moderator collision
- Σ_s = macroscopic scattering cross section of the moderator

It was further demonstrated that composites of materials having different "C" values could be represented rather well by

$$T_n = T_m \left[1 + \frac{\sum_i C_i Y_i \sigma_{a_{oi}} \sqrt{\frac{T_o}{T_m}}}{\xi \Sigma_s} \right] \quad (18)$$

Supplying individual "C" values for each isotope would have necessitated adding yet another array to the physics constants or library tape (FORTRAN designation B-6), so an alternate means has been exploited to use constants which are available in the existing tape. Observation of the "C" values (Table I) indicates that those isotopes having smaller values are the isotopes with larger values of Westcott g, and the inverse is also true. The table also illustrates the degree of correlation of C and 1/g for the gas model kernel at a temperature of 20 °C and at 260 °C.

TABLE I. Correlation of C and Inverse G

Absorber Isotope	$\frac{0.61}{g(T)}$	"c"(20 °C) Spectrum	$\frac{0.61}{g(T)}$	"c"(260 °C) Spectrum
B-10(1/v)	0.61	0.61	0.61	0.61
U ²³⁵	0.6137	0.68	0.643	0.68
Pu ²³⁹	0.547	0.55	0.400	0.48
Pu ²⁴⁰	0.587	0.53	0.553	0.53

The use of the inverse $g(T)$ times the constant for 1/v material will in general give better estimates of the neutron temperature than the use of an overall constant, even though the correlations are poorer than might have been hoped. The equation used in the ALTHAEA code is

$$T_n = (T_m + T_o) \left\{ 1 + k_3 \frac{\sum [Y \sigma_{a_{oi}} / g(T)_i] \text{RTM}}{\xi \Sigma_s (\text{vol. mod. / vol. fuel})} \right\} - T_o \quad (19)$$

where

- T_n = Maxwellian modal temperature, °C
- T_m = moderator temperature, °C
- k_3 = a constant which is $\sqrt{4/\pi}$ times "C" for a 1/v material
(an input number identified as RAYK3 in Appendix D,
BNW is using 0.7)
- Y_i = concentration of the i^{th} isotope nuclei/barn cm in the
fuel
- $\sigma_{a_{oi}}$ = 2200 m/s absorption cross section of the i^{th} isotope
- $g(T)_i$ = the Westcott g as a function of the neutron temperature
for the i^{th} isotope

$$\text{RTM} = \sqrt{\frac{\pi T_o}{4(T_n + 273.1)}}$$

(where T_o is the temperature in °K corresponding to a velocity of 2200 m/s)

$\xi \Sigma_s (\text{vol. mod. / vol. fuel})$ = slowing down power per unit of fuel volume.

A better approach than basing the calculation on moderator temperature is to measure the neutron temperature and use this measured value in the code as the initial neutron temperature. Using Equation (19) ALTHAEA will then calculate an effective moderator temperature and change the neutron temperature as burnup proceeds.

THE ZERO-DIMENSIONAL BURNUP MODE

There are many modes of operation for the ALTHAEA code. Since the zero-dimensional mode is basic for any of the other modes, a detailed description of it will be presented first. The ALTHAEA code calculates the spectral index R for the zero-dimensional case in a manner consistent with use of a two-group one-dimensional diffusion equation to calculate the flux distribution and burnup of a reactor core.

The spectral index R as used in this document is the ratio of epithermal flux/unit lethargy to the "Westcott" flux:

$$R = \frac{\phi_1 / \Delta U}{\phi_w} \quad (20)$$

where

ϕ_1 = the epithermal flux

ΔU = the number of lethargy units in the epithermal group

ϕ_w = the Westcott flux, i. e., neutron density x 2200 meters/second.

The method is based upon the effective reaction rates in the thermal and epithermal groups as corrected for flux depression, resonance shielding, and interference and fast effect as described in the preceding section and an adjustment to the slowing down cross sections which will be described in the following paragraphs. Selections of a proper value for an input constant for this adjustment permits calibration of the computed value of R to an experimental value or to a value obtained from more detailed slowing down calculations.

The Westcott flux defined above is defined by Westcott to be composed of two components, the Maxwellian component ϕ_2 and an epithermal component related to the epithermal flux ϕ_1 as follows:

$$\phi_w = \phi_2 + b \frac{\phi_1}{\Delta U} \quad (21)$$

where

ϕ_2 = the Maxwellian component

b = the integral of the assumed flux shape from μkT to infinity as defined by Westcott and evaluated to be

$$\frac{4}{\sqrt{\pi\mu}}$$

Equation (21) may be substituted into Equation (20) and the numerator and denominator divided by ϕ_2 to express R as a function of the epithermal-to-thermal flux ratio as follows

$$R = \frac{\frac{\phi_1/\Delta U}{\phi_2}}{1 + b \frac{\phi_1/\Delta U}{\phi_2}} \quad (22)$$

In the following paragraphs the flux ratio will be derived using epithermal and thermal reaction rates.

The epithermal group of the two-group diffusion equation takes the form

$$D_1 \nabla^2 \phi_1 - \bar{\Sigma}_{r1} \phi_1 + S_1 = 0 \quad (23)$$

where

D_1 = the diffusion coefficient

∇^2 = differential operator

ϕ_1 = epithermal flux

$\bar{\Sigma}_{r1}$ = average group one removal cross section

S_1 = source term.

The Group 1 removal cross section consists of two components, the transfer from Group 1 to Group 2, and absorption of the neutrons. For an idealized case with no absorption, the removal cross section reduces to the expression:

$$\bar{\Sigma}_{r_1} = \frac{\xi \bar{\Sigma}_{s_1}}{\Delta U} \quad (24)$$

where

- ξ = the average logarithmic energy decrement
- $\bar{\Sigma}_{s_1}$ = the average Group 1 macroscopic scattering cross section
- ΔU = the number of lethargy units in Group 1 (the logarithm of the mean fission energy divided by the cutoff energy)

For the real case, the removal cross section increases because of the absorption of neutrons; however, the increase may not be computed by the direct addition of the macroscopic absorption cross section since they are averaged in different ways. We have, therefore, adopted the following form for removal cross sections:

$$\bar{\Sigma}_{r_1} = \frac{\xi \bar{\Sigma}_{s_1}}{\Delta U} + K_1 \bar{\Sigma}_{a_1} \quad (25)$$

where K_1 is an arbitrary input constant, and $\bar{\Sigma}_{a_1}$ is the macroscopic epithermal absorption cross section. The suitability of this expression is discussed in the next paragraph.

The transfer cross section from Group 1 to Group 2 which is the removal cross section minus the absorption cross section is therefore

$$\bar{\Sigma}_{s(1-2)} = \frac{\xi \bar{\Sigma}_{s_1}}{\Delta U} - (1.0 - K_1) \bar{\Sigma}_{a_1} \quad (26)$$

where $\bar{\Sigma}_{s(1-2)}$ is the transfer cross section from Group 1 to Group 2.

A value of 0.2 for K_1 for water and varying amounts of plutonium in a homogeneous system has been obtained from computer program GAM-1,⁽¹²⁾ of the Battelle-Northwest version known as HRG. It is believed that larger values of K would be appropriate for heterogeneous systems, and in fact, with a value of 0.65 the ALTHAEA computations confirmed experimental data reported for the Yankee reactor.⁽¹³⁾

Writing the equation for the thermal group of the two group system and using $\bar{\Sigma}_{s(1-2)}\phi_1$ as the source term for Group 2 we get

$$-D_2 \nabla^2 \phi_2 + \bar{\Sigma}_{a2} \phi_2 - \bar{\Sigma}_{s(1-2)} \phi_1 = 0 \quad (27)$$

Substituting the expression previously developed in (26) to consider Group 1 and assuming that the reactor is infinite which causes the first term for leakage to vanish,

$$\bar{\Sigma}_{a2} \phi_2 - \left[\frac{\xi \bar{\Sigma}_{s1}}{\Delta U} - (1 - K_1) \bar{\Sigma}_{a1} \right] \phi_1 = 0 \quad (28)$$

This expression can be solved for the ratio of the epithermal-to-thermal flux to give

$$\frac{\phi_1}{\phi_2} = \bar{\Sigma}_{a2} / \left[\left(\frac{\xi \bar{\Sigma}_{s1}}{\Delta U} \right) - (1 - K_1) \bar{\Sigma}_{a1} \right] \quad (29)$$

Equation (29) may be substituted into Equation (22) as follows

$$R = \frac{\frac{\bar{\Sigma}_{a2}}{\left[\frac{\xi \bar{\Sigma}_{s1}}{\Delta U} - (1 - K_1) \bar{\Sigma}_{a1} \right] \Delta U}}{1 + b \frac{\bar{\Sigma}_{a2}}{\left[\frac{\xi \bar{\Sigma}_{s1}}{\Delta U} - (1 - K_1) \bar{\Sigma}_{a1} \right] \Delta U}} = \frac{\bar{\Sigma}_{a2}}{\xi \bar{\Sigma}_{s1} - (1 - K_1) \bar{\Sigma}_{a1} \Delta U + b \bar{\Sigma}_{a2}} \quad (30)$$

$\bar{\Sigma}_{a_2}$ is the sum of the fuel plus nonfuel absorption cross sections. The effective fuel cross sections are summed over all isotopes as discussed earlier in Equations (5-6) and are given the name SIGSUM in the code. The nonfuel macroscopic absorption cross section per unit fuel volume is an input number called SNF.

$$\bar{\Sigma}_{a_2} = (\text{SIGSUM} + \text{SNF}) (\text{RTM}) \quad (31)$$

where

$$\text{RTM is } \sqrt{\frac{293 \pi}{4 T_n}}$$

$\xi \bar{\Sigma}_s$ is defined in Equation (24) and is expressed per unit fuel volume to be consistent with $\bar{\Sigma}_{a_2}$. It is represented by the name SDPV.

$$\xi \bar{\Sigma}_{s_1} = (\text{SDPV}) (\text{TNL}) \quad (32)$$

where

$$\text{TNL} = \frac{1}{1 + L^2 B^2}, \text{ the thermal nonleakage}$$

and

$$\begin{aligned} L &= \text{the thermal diffusion length} \\ B^2 &= \text{the geometric buckling} \end{aligned}$$

$\bar{\Sigma}_{a_1} \Delta U$ is the resonance integral for fuel plus nonfuel absorptions.

SUMA, which is the fuel contribution, is defined in terms of the quantities in Equation (13) as follows:

$$\text{SUMA} = \text{SUMR} + \text{SUMA}' * P \quad (33)$$

If the nonfuel absorption cross sections may be assumed to be $1/v$ the non-fuel contributions will be equal to b times SNF.

$$\bar{\Sigma}_{a_1} \Delta U = (\text{SUMA} + b \text{ SNF}) (\text{RTM}) \quad (34)$$

Equations (31), (32), and (34) may be substituted into Equation (30) to obtain the relation used in the program

$$\begin{aligned}
 R &= \frac{(\text{SIGSUM} + \text{SNF})(\text{RTM})}{(\text{SDPV})(\text{TNL}) - (1 - K_1)(\text{SUMA} + b \text{SNF})(\text{RTM}) + b(\text{SIGSUM} + \text{SNF})(\text{RTM})} \\
 &= \frac{\text{SIGSUM} + \text{SNF}}{(\text{SDPV})(\text{TNL})/\text{RTM} - \text{SUMA} + K_1(\text{SUMA} + b \text{SNF}) + b \text{SIGSUM}} \quad (35)
 \end{aligned}$$

Note that if K_1 is set equal to zero, Equation (35) will yield an expression which is identical to that derived by Triplett and Busselman.⁽¹⁴⁾

INPUT PREPARATION

Proper preparation of the input data for a given system is usually a many-step process due to the interdependence of both the nuclear constants and the calibratable constants. These constants must be assembled and given a preliminary test in a controlled system such as a critical experiment. Any necessary adjustments are then made in the values of the constants. These adjustments will now be described. Normally the calibration is based on the zero dimensional mode and then only a selection of transverse buckling and transport cross section is required for the one dimensional case.

Procedure

The first step in preparing input is to secure the necessary data to prepare the B6 library tape. Basic cross section data may be obtained from tabulations such as those of Westcott⁽¹⁾ or the BNW data by Liikala as revised by Stewart.⁽¹⁰⁾ The former presents values of the 2200 meter per second cross sections that are used directly, and tabulations of g and s as a function of temperature, which must be converted to polynomial coefficients of fourth or lower order. Stewart's data, which represent the cross section data as a function of energy over several energy ranges, must be converted to the Westcott parameters. E. Z. Block⁽¹⁵⁾ has written program GANDS to perform this function. GANDS reads the functional relationships from the library tape and flux weights cross sections to obtain a Maxwellian averaged cross section ($\bar{\sigma}$) and resonance integrals based on an assumed $1/E$ flux. The lower resonance integral (RI_3) is

calculated from μkT to the Group 3 cutoff energy. The energy limits are input values to the code, such as 3.681 for μ and 0.68256 for E_3 . Values of g and s are then computed as follows:

$$g = \sqrt{\frac{4T}{\pi T_0}} \frac{\bar{\sigma}}{\sigma_0} \quad (36)$$

$$S_3 = \sqrt{\frac{4T}{\pi T_0}} \frac{RI_3}{\sigma_0} - bg \quad (37)$$

where σ_0 is the 2200 meter/second cross section and T_0 is 293.16 °K. Since both $\bar{\sigma}$ and RI_3 are dependent on the neutron temperature T several values are computed and fitted versus T . The best set of polynomial coefficients of Order 4 or less is punched on cards ready for input to program PREP. (Appendix G) RI_1 and RI_2 are not dependent on temperature so one calculation for each is adequate and the values for S_1 and S_2 are computed by dividing each resonance integral by σ_0 . The term $\sqrt{4T/\pi T_0}$ would introduce a temperature dependence so it is not included in this calculation, but rather it is introduced in ALTHAEA when S_1 and S_2 are used.

During the reading of the functions describing the cross sections GANDS computes the maximum cross section of each resonance for use as Z_0 and the Doppler coefficient and tabulates them for user inspection. Suitable values may be selected for use in Arrays 16, 17, 21, and 22 of PREP input.

Data for the fission product model may be obtained from CRRP-913⁽¹⁶⁾ or for individual fission products from ANL-5800.⁽¹⁷⁾ Isotope decay constants may be obtained from BNWL-CC-325⁽¹⁰⁾ or a chart of the nuclides. Values for ν and resonance parameters for computation of Z may be obtained from Neutron Cross Sections and its supplements⁽¹⁸⁾ or BNWL-CC-325. Values of MeV per fission may be obtained from ANL-5800⁽¹⁷⁾ by summing fission fragment energies, prompt neutron and gamma energies, and an appropriate fraction of the fission product decay energies. These basic numbers may require some adjustment in the final stages of the calibration process to obtain detailed agreement of reactivity and isotopic content of the fuel.

As a second step in the input preparation process, reactor parameters are determined. Trial parameters are best obtained from a multigroup transport code analysis of a typical cell in the reactor. This code should yield values of epsilon; nonfuel macroscopic absorption cross section per unit fuel volume (SNF); coefficients in the thermal flux depression equation (FCONST and FEXP); moderator temperature weighted by thermal flux, volume, and macroscopic scattering cross section (TEM); the epithermal flux weighted slowing down power per unit fuel volume (SDPV); and the effective surface-to-volume ratio plus potential scattering cross section of the fuel (SCA). Program JASON(4) does these calculations.

The third step in the input preparation requires comparison of computed data from several trial runs with experimental data. Based on this data RAYK1 is adjusted to obtain the proper values of the epithermal index r such that experimental and computed results agree. Adjustment of RAYK2, and the fuel temperature may be required to match experimental values of resonance escape probability. Experimental measurements of thermal flux depression may be used to correct FCONST and FEXP; then SNF may be adjusted to obtain the correct thermal utilization. It may be adequate to adjust TEM or RAYK3 to obtain the proper relative reaction rates, but any major differences from burnup data are best corrected via the basic cross section data. Finally, epsilon may be adjusted to yield the correct reactivity or U^{238} reaction rate. Although all of these terms can be adjustable in practice, it has been found that when the JASON program is used only RAYK1 and RAYK2 actually require adjustment.

The following four-group structure is now being used in ALTHAEA. The fast group was arbitrarily set for 6 MeV down to 0.2 MeV. The slowing down group extends from 0.2 MeV to 0.683 eV, and the epithermal resonance group extends from 0.683 eV to μ KT. The thermal group is based on a Maxwellian flux. The slowing down group cross sections are adjusted to account for possible successive flux reductions due to the resonance absorption in the major fertile isotopes. We are presently applying this correction to Th^{232} , U^{238} , U^{234} , and Pu^{240} .

CALIBRATION

Calibration was accomplished in two ways—theoretically and experimentally. Theoretical calibration was performed by comparing more sophisticated and time consuming computer codes. These codes included RBU,⁽¹⁹⁾ ZODIAC G,⁽²⁰⁾ and HRG-THERMOS.^(21, 22) The experimental calibration used critical data obtained from BNL work⁽²³⁾ with U^{238} systems and from Battelle-Northwest work with plutonium systems. Burnup data were obtained by determining isotopic composition of UO_2 fuel elements for exposures up to 24,000 MWD/Tonne in the Yankee reactor.⁽²⁴⁾ Additional burnup data were obtained by chemical analysis of spent PRTR mixed oxide fuels.⁽²⁵⁾

The initial selection of the C_{ij} 's was made by graphically measuring the overlapping areas in the low-lying resonances. For example, Pu^{239} , U^{235} , and Am^{241} all have resonances near 0.3 eV, while Pu^{241} has a resonance at 0.25 eV and its reaction rate is also reduced by the total flux depression. To obtain the proper reaction rate of each isotope relative to the others, one must adjust the cross section of each isotope. The C_{ij} numbers have been refined empirically to fit burnup data.

If no other information is available, the calibratable coefficient Z can be estimated based upon the height in barns of the lowest lying resonance. This number usually is 1.5 to 2.0 times the cross section calculated by GANDS; for example, the factor is 1.95 to simulate the reaction rates of Pu^{240} and 1.82 for U^{238} .

RAYK2 is a calibratable coefficient, and if it is equal to zero, there is no correction for the Group 3 flux depletion; but if RAYK2 is equal to 1.0, the Group 3 flux is reduced by the fractional amount of the absorptions in Group 2. Because of the probability of a large energy decrement per scattering collision in water moderator reactors, intermediate values of RAYK2 are appropriate. Intermediate values will also apply in graphite reactors with large cells such that several scattering events occur before the neutron again passes through the fuel element.

Results of Calibration

Figure 5 shows a comparison between the results of the ALTHAEA calibration and the BNL data. Note the good agreement between the BNL data and the thermal utilization factor (f). The JASON code, with a P_3 transport calculation was used to derive the flux depression coefficients for ALTHAEA. The calculation of the resonance escape probability is done entirely within the ALTHAEA code. For these systems that are composed of slightly enriched uranium metal, the curve of p over the wide range of moderator-to-fuel ratios has the proper shape. Table II shows the comparison between experimental data reported in BNL-486 and theoretical data for the same lattices calculated by HRG-THERMOS and RBU. In all cases ALTHAEA agreed with theoretical calculations and only disagreed with experimental points at a moderator-to-fuel ratio of 3.0 where all other theoretical calculations also disagreed. All of the ALTHAEA results were calculated using a single set of the calibratable constants RAYK1, RAYK2, and Z's. All other inputs were directly calculated by the JASON- P_3 or were developed directly by the GANDS program from the BNW master library.

The ALTHAEA program was principally developed to generate burnup data for a wide range of reactor systems. Thus, an important phase of the calibration was checking the code's performance when compared with chemical analysis of spent fuel. Figure 6 shows excellent agreement between the ALTHAEA calculated isotopic concentrations and those experimentally determined from Yankee's core 1 and 2. Figure 7 shows the agreement for isotopes which were produced in much smaller concentrations in the Yankee experiments over the same burnup range as the data shown in Figure 6. Again, excellent agreement was obtained with the possible exception of Cm^{242} which, due to its short half life, is especially sensitive to reactor operating history and the time of chemical analysis.

Burnup data for $\text{PuO}_2\text{-UO}_2$ fuel irradiated in PRTR, which represents a different fuel and neutron spectrum than the Yankee reactor, again demonstrated excellent agreement between the ALTHAEA calculations and the experimental data points. (see Figure 8)

FIGURE 5. Comparison of ALTHAEA Calculations of p and f with Experimental Data

TABLE II. Experimental Values for f , p , and ϵ Compared with P_3 -Calibrated-ALTHAEA, HRG-THERMOS and RBU Calculations

Volume of Moderator	BNL-486*			P_3 -ALTHAEA			HRG-THERMOS**		RBU		
Volume of Fuel	f	p	ϵ	f	p	ϵ	f	p	f	p	ϵ
4.0	0.795	0.909 ± 0.008	1.042	0.797	0.905	1.048	0.771	0.911	0.792	0.871	1.037
3.0	0.845	0.868 ± 0.003	1.049	0.843	0.880	1.055	0.823	0.885			
2.0	0.893	0.841 ± 0.004	1.063	0.892	0.838	1.068	0.878	0.840			
1.5	0.920		1.075	0.916	0.802	1.079	0.906	0.799			
1.0	0.946	0.720 ± 0.027	1.099	0.942	0.743	1.095	0.934	0.735	0.945	0.743	1.099

* These are results of extensive experiments by BNL on water lattices at 20 °C.

** f from THERMOS is known to be slightly low.

Figure 6. Comparison of ALTHAEA Calculations With Experimental Yankee Burnup Data

Figure 7. Comparison of ALTHAEA Calculations With Experimental Yankee Burnup Data For Isotopes With Low Concentrations

Figure 8. Comparison of ALTHAEA Calculations With Experimental PRTR $\text{PuO}_2\text{-UO}_2$ Burnup Data

Theoretical burnup data and reactivity comparison was made by comparing ZODIAC G results with those obtained by the ALTHAEA code over a range of fuel exposures, H/U ratios, and rod sizes. The ZODIAC G code was chosen for this comparison because the cross sections are derived using the HRG-THERMOS system with burnup accomplished in the ALCHEMY program⁽²⁶⁾ which represents a very sophisticated method of determining reactivity and burnup concentrations. Another advantage of using this system was both ALTHAEA and ZODIAC G derived basic cross section data from the same source.

Table III shows the isotopic concentrations and reactivity calculated by ZODIAC G and the percentage variation between the ALTHAEA calculation and the ZODIAC G calculation. Reactivity was found to have a maximum variation of 0.5% while isotopic concentrations U^{235} and total plutonium were within 1%. The maximum variation occurred with Pu^{242} concentration, which was later found to be the result of an improperly derived cross section in ZODIAC G. Thus, the plutonium concentrations are closer than those shown. It is interesting to note that the ZODIAC G calculations took about 400 times more computer time than ALTHAEA to arrive at the results shown in Table III.

Table IV is another illustration of how well the ALTHAEA code can be calibrated. In this case, the fuel composition is approximately that found in a power reactor after an exposure of 8000 MWD/ton. For each isotope considered, the difference in cross section between the HRG-THERMOS codes and ALTHAEA is usually on the order of 1%.

Range of Applicability

So far, our calibrations have shown that for a given type of lattice, it is possible to calibrate the code readily so that a wide variety of burnups and critical experiments can be accommodated. With the coefficients available, there are a large number of combinations that will fit a single burnup case. However, as a given calibration is applied over a wider range, it is either necessary to accept a larger error or to provide a more general calibration by making a more adroit selection of the calibration coefficient adjustments. Further, it is necessary to know when to quit adjusting coefficients on ALTHAEA and when to secure the services of a more basic code. Enough experience has been accumulated to provide a "feel" for changing a coefficient

TABLE III. Comparison of Calculated Results with ZODIAC-G ALTHAEA

Cell		Exp.	K_{∞}	Net Change in Concentration, GM/MT					
				U^{235}	U^{238}	Pu^{239}	Pu^{240}	Pu^{241}	Pu^{242}
0.93 cm Rod H/U = 3.97	Zodiac	3,391	1.146	3,669	2,399	1,702	149	26.4	0.89
	% Var.		-0.17	+0.06	0.70	0.75	-0.67	1.15	5.3
	Zodiac	7,449	1.101	7,332	5,357	2,984	487	161.3	12.66
	% Var.		0.18	-0.15	0.38	0.37	-0.61	0.62	3.6
	Zodiac	10,824	1.073	9,946	7,918	3,698	801	327.9	39.20
	% Var.		-0.28	-0.19	0.17	-0.60	-0.41	-0.60	1.9
0.93 cm Rod H/U = 4.62	Zodiac	3,387	1.166	3,700	2,220	1,572	140	22.5	0.77
	% Var.		-0.26	+0.16	0.14	0.87	-0.72	0.91	4.9
	Zodiac	7,439	1.118	7,439	4,956	2,738	466	141.3	11.40
	% Var.		0.18	0	0.12	0.49	-0.46	-0.30	3.2
	Zodiac	10,136	1.088	9,615	6,854	3,262	712	258.2	29.93
	% Var.		0.28	-0.07	0.03	0.23	-0.29	-0.81	1.7
0.75 cm Rod H/U = 3.97	Zodiac	3,747	1.132	4,006	2,702	1,893	179	34.7	1.29
	% Var.		-0.26	+0.06	0.92	1.06	-0.83	1.20	5.5
	Zodiac	7,481	1.092	7,327	5,508	3,086	500	167.6	13.09
	% Var.		0	-0.20	0.36	0.56	-0.60	0.58	3.7
	Zodiac	11,210	1.055	10,116	8,417	3,893	852	358.1	44.19
	% Var.		0.28	-0.29	0.02	-0.36	-0.35	-0.67	1.8

TABLE IV. Comparison of HRG-THERMOS and P₃-Calibrated ALTHAEA Cross Section Values for Water Moderated Cells at 20°C

$\sigma_f U^{235}$		$\sigma_a U^{238}$		$\sigma_a Pu^{239}$		$\sigma_f Pu^{239}$		Moderator to Fuel Ratio
H-T	A/H-T	H-T	A/H-T	H-T	A/H-T	H-T	A/H-T	
469	1.00	8.13	1.01	1283	0.99	871	0.99	1.9
461	1.00	9.70	0.99	1314	1.00	885	1.00	1.5
$\sigma_a Pu^{240}$		$\sigma_a Pu^{241}$		$\sigma_f Pu^{241}$		$\sigma_a Pu^{242}$		
H-T	A/H-T	H-T	A/H-T	H-T	A/H-T	H-T	A/H-T	
861	0.99	1341	0.99	1028	0.99	263	0.99	1.9
958	1.00	1313	1.00	1010	1.00	323	1.00	1.5

Note: H-T = HRG-THERMOS Data

A = P₃-ALTHAEA

without throwing the others out of adjustment, and further refinements are being made. As more burnup data become available, it may be possible to use a formal statistical method to minimize all errors, rather than relying on intuition alone.

ALTHAEA With 4-Group Diffusion Theory

This report has described the basis for the ALTHAEA code as it is used with a 2-group diffusion code and effective 4-group burnup. The code has recently been developed into a version which includes 4-group diffusion calculations and 4-group cross sections.

Run Time

The time required to complete the computations for a given reactor may vary over wide limits depending on input values. Computation time in subroutines SIGMA and RK is essentially proportional to the square of the number of isotopes. For a typical arrangement with 14 isotopes in Group 1, SIGMA takes 60 milliseconds and RK 100. The number of passes through these routines is proportional to the reactor exposure and inversely proportional to the flux or power level. When economics data are being written on tape at least one SIGMA and RK calculation is made for each interval of PERIOD (in days). (See appendix for input definitions) If TSLIM (in seconds) is less than PERIOD, or if the estimated error in any isotopic concentration exceeds ERU causing a reduction in the time step, additional RK calculations will occur. Up to N15 calculations can be performed per SIGMA calculation. ERL determines the rate at which the time step doubles beginning at the input value of H until TSLIM is reached. The time step will usually double every iteration if ERL is 0.05 or greater, every other iteration at 0.03, and every fifth iteration at 0.01. Data written to Tape B8 requires about 65 milliseconds each PERIOD. The output to Tape KM requires from 70 to 600 milliseconds depending on the setting of NS3 and the number of isotopes, but there are N8 SIGMA calculations between output writings.

The time spent in a typical zero-dimensional case for a reactor operating for 1000 days at 20 MW/ton is presented in Table V.

TABLE V. Zero-Dimensional ALTHAEA Run Time

<u>Operation</u>	<u>Time (Seconds)</u>
Read case	0.41
Write back input data	0.27
Initialization	0.15
Time zero output to Tape 3	0.36
Time zero output to Tape 8	0.07
End point hunting	0.52
Final output to Tape 3	0.47
Final output to Tape 8	<u>0.06</u>
Subtotal	2.31
Intermediate output to Tape 3	1.05
Intermediate output to Tape 8	0.32
Subroutine SIGMA	0.80
Subroutine RK	<u>5.30</u>
Total	9.78

For a one-dimensional case with two burning regions and a reflector a typical breakdown of the time spent is presented in Table VI.

The sense lights serve as a guide to the user who is interested in a qualitative indication of the time spent in various parts of the program. Sense Light 1 is on while computing in subroutine SIGMA. Sense Light 2 is on during the Runge-Kutta calculation. Sense Light 3 is on during the the diffusion calculation. Sense Light 4 indicates that time is progressing in the forward direction. It is turned on when initialization is commenced and remains on until the "end point" hunting is commenced. If two or more end points are requested, it will come on again as "burn up" is progressing to the next "end point." If the user is at the computer console, he may follow the progress of the computations by depressing Sense Switch 6. As each new case is commenced, three numbers will be printed on line: the sequential case number, the economics ident number, and the recycle pass number.

TABLE VI. One-Dimensional ALTHAEA Run Time:
Two Burning Regions and One Reflector
Region

<u>Operation</u>	<u>Time (Seconds)</u>
Read case	1.30
Write back input data	0.90
Initialization	0.50
Time zero diffusion calculation	0.90
Time zero output to Tape 3	1.60
Time zero output to Tape 8	0.14
Final diffusion calculation	0.60
Final output to Tape 3	1.60
Final output to Tape 8	<u>0.12</u>
Subtotal	7.66
Intermediate output to Tape 3, Region	3.50
Intermediate output to Tape 3, Reactor	2.50
Intermediate output to Tape 8	0.64
Subroutine SIGMA	2.40
Subroutine RK	10.60
Diffusion calculations	<u>4.50</u>
Total	31.80

CODE MECHANICS

General Input Features

Input to the ALTHAEA code for both zero- and one-dimensional modes is prepared and punched on standard IBM cards which are then processed to magnetic tape on suitable off-line equipment. The magnetic tape is read by the 709/7090 computer. A detailed description of the input with examples, is given in Appendixes A, B, E, and F. The first nine cards have integers and constants for computational flow control. One card for each isotope follows, giving the isotope name, descendent number, concentration, and

n-2n descendant number. The physics cards are last and give the pertinent physics constants. Five of the first nine cards deal with special options. The control integers are selected such that zero or blank bypasses the option and the card is ignored. Thus, the code may be used with blank cards inserted for control of unwanted options.

Output Features

Output of ALTHAEA is written to the magnetic tape designated in input in a form ready for printing on suitable off-line equipment. (Samples of this output are presented in Appendix E and F.) The output may be quite limited or very detailed depending on control settings in the input. Detailed output will be written at the beginning and at the end of each case. The frequency and detail of output at intermediate times are controlled by input settings. A very large value for N8 will minimize or eliminate output at intermediate times, while a value of 1 will cause output each time new cross sections are calculated. The amounts of output written at each time will be determined by the setting of NS3. A negative value (-1 or -2) will actuate output of the effective microscopic cross sections in barns and the values of g and s used in their computation. Fission cross sections of the fissionable isotopes are given first, followed by the absorption cross sections of all isotopes. If NS3 is -2, the following two-group parameters based on unit fuel volume will be given next. (This option should not be used in the one-dimensional mode.)

MUKT	the cutoff energy in eV
RTM	$\sqrt{\pi T_o/4T}$
DU	number of lethargy units (factors of e) in the epithermal group
PHI 1	epithermal flux
PHI 2	thermal flux
SIGA 1	microscopic epithermal group absorption cross section
SIGA 2	microscopic thermal group absorption cross section
Y*SIGA 1	macroscopic epithermal group absorption cross section

Y*SIGA 2 macroscopic thermal group absorption cross section
 SIGF 1 microscopic epithermal group fission cross section
 SIGF 2 microscopic thermal group fission cross section
 Y*SIGF 1 macroscopic epithermal group fission cross section
 Y*SIGF 2 macroscopic thermal group fission cross section

If NS3 is zero or negative, a tabulation of data for each fuel (Group 1) isotope is given with each line containing the following data:
 (See Appendix A)

Isotope	name of isotope as provided in input
Descendant	name of isotope formed by neutron capture
Concentration	amount of isotope present in atoms per barn centimeter
Destroyed	summation from time zero of atoms destroyed by absorption and radioactive decay
ABS	macroscopic absorption cross section
ABSAV	flux weighted average from time zero
ALPHA	ratio of capture-to-fission cross section
HEAT	percent of total heat generated since time zero
Power	percent of total power at this time
CONV.	percent rate of formation per destruction of all fissile isotopes

The last entry line in the table reports the macroscopic absorption for the sum of all isotopes in Groups 2 and 3 (burnable poison isotopes and fission products). The concentrations of the isotopes in Groups 2, 3, and 4 are reported below the table.

If NS3 is positive, cross section data and the table are omitted but the concentrations of all isotopes are given. In any case, two additional lines of data report the following:

TIME	burnup time in seconds
PAV	average power in constant flux run or average flux in constant power run
PWR	power at this time
TNEUT	neutron temperature in degrees centigrade
PMXV	ratio of maximum-to-average power or flux

SIGSUM	sum over all isotopes of the thermal part of the cross section
ABSF	as above for fuel (Group 1) isotopes
XMLT	infinite multiplication constant
ZTA	ratio of neutrons produced excluding the fast group to the sum of all except fast absorptions in the fissile isotopes plus thermal absorptions in other fuel isotopes
EPS	gain in neutrons due to the fast group
CR	ratio of fissile isotope production to destruction, as percent
CAB	percent of neutrons absorbed in control material or leaked since time zero
HEAT	sum of fission energy released since time zero
PHI	flux
WRAT	change in WA per second
WA	macroscopic fission cross section times MeV/fission
PMNV	ratio of minimum-to-average power or flux
R	epithermal flux index
ABSR	macroscopic absorption cross section of the cell per unit fuel volume
AMLT	flux averaged infinite multiplication from time zero
J	ratio of all except fast absorptions in the fissile isotopes plus thermal absorptions in other fuel isotopes to all except fast absorptions in the cell minus epithermal absorptions in nonfissile fuel isotopes
Q	ratio of cell absorptions except epithermal in the non-fissile fuel isotopes to all absorptions in the cell (excluding the fast group)
FISBU	macroscopic absorption cross section of fissile isotopes
FRAB	percent of neutrons absorbed in fertile isotopes from time zero

Zero-Dimensional Output

When an "end-point" calculation based on reactivity, time, or exposure as directed by input is made, a special summary page of output is written. The heading states, "The following information is the beginning and end of this case." The concentrations and data at time zero are then written, followed by a statement regarding the degree of convergence to the end point. The final values of the nonfuel macroscopic absorption cross section and the slowing down power of the cell, both per unit fuel volume, are then given. The end-point data are presented on the following page of output. Samples of this output are presented in Appendix E.

One-Dimensional Output

With operation in the one-dimensional mode, additional output is written on tape (see sample in Appendix F) under the control of KOUT. If KOUT is negative, group constants per unit fuel volume will be written after the computations are completed for this region as follows:

SA1	macroscopic absorption resonance integral
SA2	macroscopic absorption cross section for thermal group
SF1	macroscopic fission resonance integral times MeV/ fission
SF2	macroscopic fission cross section for thermal group times MeV/fission
SFNU1	macroscopic fission resonance integral times nu
SFNU2	macroscopic fission cross section for thermal group times nu

Regardless of the setting of KOUT, a statement indicating the completion of computations for this region is written. When all regions are completed, a heading for the diffusion subroutine output is written, including a time reference and the value of power based on current cross sections and on fluxes from either input or the previous calculation. If the power is far from the desired value except at time zero, a smaller value of PERIOD in the input should be used.

In the event that any values of flux fail to meet the convergence criterion given in input after 100 iterations, a message to that effect, and the last ten values of reactivity will be written. If automatic refueling is done, a statement to that effect is written giving the reactivity before refueling. If the control rod adjustment loop fails to converge the reactivity, a message so stating is written. The reactivity with control rods withdrawn is then given. A message noting a flux reduction due to power peaking will be given next, if required by input. If all values of KOUT are positive, there will be no more output. If any value of KOUT is zero or negative, a table heading is written and the following items will be tabulated for those regions with zero or negative value of KOUT.

Radius	outer or right-hand boundary of the region
Power	specific power of the region, W/cc of fuel
Flux	average Westcott flux for region normalized to power given in input
Percent Change	change in flux since last calculation
R	average spectral index for this region
Percent Change	change in R since last calculation
SNFD	macroscopic cross section of control added
CFACT	fraction of "hot" control rod remaining in reactor
DFACT	fractional change to low-reactivity values of input constants (see NTRL = 3)

For those regions in which KOUT is - 2, the following values are tabulated

D1	epithermal group diffusion constant in centimeters
D2	thermal group diffusion constant in cm
XISIGS	transfer cross section from epithermal-to-thermal group
DU	number of lethargy units (factors of e) in epithermal group
SIG A1	epithermal macroscopic absorption cross section per unit cell
SIG A2	thermal macroscopic absorption cross section per unit cell

SIGR1	epithermal macroscopic removal cross section per unit cell
SIGR2	thermal macroscopic removal cross section per unit cell
SIGNU1	epithermal macroscopic fission cross section times nu
SIGNU2	thermal macroscopic fission cross section times nu

Following this table, the unnormalized two-group flux values are listed at each mesh point for any region in which KOUT is -2.

If KOUT is negative for any region, the next page of output will be a plot of the two-group fluxes for the entire reactor. The fluxes are normalized to 100 for the highest flux and plotted versus mesh point. The user must exercise care in interpreting this plot since it will be distorted if the mesh spacing is not the same in each region.

THE CHAIN SYSTEM

Several complete programs can be placed on tape in absolute binary form for successive use. This is called a CHAIN and permits continuous calculations in one job that would not otherwise be possible within the memory limits of the 7090 machine. Table VII lists the programs used in the ALTHAEA-ECONOMICS chain in the order in which they now appear on the tape.

TABLE VII. Economics Chain Programs

<u>Identification Number</u>	<u>Name</u>
2	ALTHAEA
1	SUPERCASE GENERATOR
6	SEED BLANKET EDITOR
4	PROTEUS
5	QUICK
3	PLOTTER
18	JASON CASE GENERATOR
8	JASON
7	CASE GENERATOR
10	OPTIMIZER
9	PREP
11	CURVES

Each link or program of the CHAIN calls a predetermined link or has an input integer on one of the control cards designating the link to call. To simplify use of the CHAIN system, a master binary chain tape is saved and a short call program may be used to call the desired program from the tape. In this way, a five-card binary deck can be used for access to any part of a system that otherwise requires about 3000 binary cards. This method is convenient for the user, saves time on the peripheral equipment and the machine, and reduces errors. A longer version of the call program permits updating the CHAIN master before the desired program is called. One or more newly compiled programs can be added to the master or substituted for older versions of the same program and/or the order of programs on the tape can be revised.

At the beginning of each FORTRAN job, zero is stored in each location of memory not actually occupied by program and the four sense lights are turned off. When CHAIN programs are read into memory, the sense lights are not altered and only the portion of memory containing the new program is altered. Thus, information can be passed from program to program via upper memory (common) or the sense lights. In developing this CHAIN system, the convention was adopted that Sense Light 1 would indicate that a prior program had prepared input tapes. Thus ALTHAEA, for example, will read input from Tape 2 if Sense Light 1 is off and from Tape 9 if it is on. It is automatically turned on by CASE GENERATOR code so that the cases stored on Tape B-9 are read by ALTHAEA.

SUPERCASE GENERATOR prepares cases for ALTHAEA in the zero-dimensional mode. Initial enrichment, reactivity, and final exposure for each case are saved on Tape 5 along with the data for the basic case. A control card is read from Tape 2 specifying the maximum exposure of interest, the maximum enrichment allowed, and the minimum and the maximum number of cases desired. Additional control cards are read to alter the base case and establish one or more trial enrichments for cases to be written on Tape 9 which ALTHAEA will read. A second order function relating enrichment and exposure is obtained and the enrichments for zero

and the maximum exposure computed. This enrichment range is divided into intervals equal in number to the minimum cases desired and additional cases prepared such that at least one case will be run in each interval provided the maximum number of cases is not exceeded.

Seed Blanket Editor reads the B8 data tape prepared by ALTHAEA and writes a new data tape on B7. The operation of this code is best described by an example: ALTHAEA might compute the history of a reactor loaded with one or more blanket regions and a seed region which is automatically refueled three times. The B7 tape would contain the histories of four reactors for economic comparison, one in which the seed and blanket life times were equal, one using two seeds to drive the blanket, a third using three seeds, as well as the computed case using four successive seeds.

PROTEUS reads the B8 data tape prepared by ALTHAEA and curve fits the data. It computes an average flux and using the curve fitted data prepares a data tape on B7 similar to B8 for the economics calculations.

QUICK reads the B7 or B8 data tape to obtain time, reactivity, concentration and exposure data, and computes fuel costs as a function of parameters read from Tape A2. Up to 18 sets of parameters may be used for each case each time the data tape is read. Up to 9 cases (enrichments) for a given reactor condition will be considered for cost minimization for each parameter set. A summary is written on Tape B9.

PLOTTER reads the B7 or B8 data tape and control information from Tape A2. Summary and "out-of-reactor" data may be obtained as requested by control integers as follows:

- Physics plot
- Fuel properties table
- Special products table at selected time intervals
- Special products table at reactor discharge
- Special product data after various decay periods
- Neutron balance data

The above may be presented after the reactor shutdown based on instantaneous reactivity (batch) only or based on batch, and repeated after reactor shutdown based on flux time averaged reactivity (graded). It may also be presented at every time interval if desired.

JASON CASE GENERATOR is used to prepare JASON cases and write them on B9 for JASON to read.

JASON is a one-dimensional P-3 transport code which will analyze a typical reactor cell and produce many of the homogenized cell constants used by ALTHAEA. It is particularly useful in the calculation of nonfuel absorptions (SNF or alternatively thermal utilization), slowing down power (SDPV), epithermal scattering index (SCA), the fast effect (epsilon), and the constants in the thermal flux depression correlation (FCONST and FEXP). JASON will prepare a series of control cards on Tape A5 for use by CASE GENERATOR or on Tape B10 for use by SUPERCASE Generator.

CASE GENERATOR prepares ALTHAEA cases on Tape B9. For survey studies, it is desirable to compute several cases with variations in one or more parameters such as enrichment, moderator-to-fuel ratio, fuel element size, temperature, etc. This could lead to handling hundreds of cards since each case requires 20 to 30 cards as input. Since successive cases may differ by only one number on one card in the case, a simple case generating code was written to prepare the cases. This code minimizes errors due to cards being out of order or omitted, saves user's time, and is more convenient since a study may be represented by a handful of cards rather than a drawer full.

The CASE GENERATOR reads a base case into memory and then reads control cards which direct the writing of cases on Tape B9. A blank card causes the writing of a case. Changes to the base case can be made if desired before it is written to tape by including the appropriate control cards before the first blank card. Changes to the base case are accomplished by storing a new value in the location in memory from which the specific parameter is transferred to tape. Each card of the base case has been given a number for the purpose of identifying the location in memory in the CASE GENERATOR code allocated to storing the data. For the user's convenience, it is suggested that these numbers be placed in Columns 78 through 80 of the base case although they are not used by the ALTHAEA code. The CASE GENERATOR code will place these identifying numbers on the storage tape (B9) even though they may not be on the base case.

2 5 1 4 1 3 2 · 3 4 5 6 6 E + 0 3

1-9	ALTHAEA flow control cards
11-99	Library change cards (Present only if changes are to be made in the basic library.)
101-135	IMY or isotope concentration cards
201-202	Physics cards
203-204	Diffusion data cards

Library changes are accomplished (one change to a card) by using cards having the same format and addresses as those used in preparing the B6 basic library tape; therefore they will not be described in detail here. Their use in the CASE GENERATOR code, however, has been made convenient by a special provision so that all of the changes which are to be made in a sequence of cases may be read in along with the base case. These changes can then be called from CASE GENERATOR memory by setting N10 (Card 4, Field 6) to the number of changes which are desired

in a specific case. N10 will automatically be reset to zero for the succeeding case unless it is again set by a change card to the number of change cards which are to be read into the case. N10 must be set in the base case itself to the number of change cards for the entire run and the entire batch of cards containing the changes is placed in the base case following the "9" card and preceding the IMY cards (isotope concentration cards). If no cross section changes are desired to the first case, N10 is set to 0 before the first blank card. If, for example, the first three cross section changes are to be made in the fourth case, N10 is set to 3 before the fourth case is written to tape. Suppose that the next two cross section changes (4th and 5th cross section change cards) are to be made in the sixth case, then N10 is set to 2 before the sixth case is written to tape. This process can be continued until all of the change cards (library) have been used.

Heading cards can be supplied with the base case following Card 202 (or 204 in a one-dimensional case), providing JMP (Card 202, Field 7) has been set to the number of cards. The CASE GENERATOR comment cards carry additional instructions on the use of the heading cards, which can be written out with the case control information.

During the generation of one-dimension cases, the value of NREG (Card 1, Field 8) is altered and must be reset when a new reactor representation is commenced.

OPTIMIZER reads data from the B9 summary of minimum fuel costs prepared by QUICK and control cards from Tape A2. The minimum fuel costs for each reactor parameter set are compared to obtain the optimum set of reactor parameters.

PREP code (see Appendix G) prepares the B6 library tape of physics data for use by ALTHAEA. Control and data cards are read from Tape A2. Provision is made for 22 arrays of data, 21 of which are now used as follows:

Number	Name	Content
1	SIGMA0	Fission cross section for Group 1 isotopes
2	SIGMA1	Group 1 absorption and capture cross sections
3	SIGMA2	Group 2 absorption and capture cross sections
4	SIGMA3	Group 3 absorption and capture cross sections
5	CINTER	Group 1 resonance shielding interference coefficients, lower integral
6	GF	Polynomial coefficients for fission g
7	SF	Polynomial coefficients for fission s, lower integral
8	GA	Polynomial coefficients for absorption g
9	SA	Polynomial coefficients for absorption s, lower integral
10	SLAM1	Group 1 decay constants
11	XLAM2	Group 2 decay constants
12	XLAM3	Group 3 decay constants
13		Not used
14	FYLD	Fission yield
15	XNU	Neutrons per fission
16	ZFO	Height of single NRIA resonance for fission
17	ZAO	Height of single NRIA resonance for absorption
18	XMEV	MeV per fission
19	SFRI	s for fission, upper resonance integrals
20	SARI	s for absorption, upper resonance integrals
21	DCF	Doppler coefficient for use with Array 16
22	DCA	Doppler coefficient for use with Array 17

Group 1 isotopes are the fuel isotopes, while Group 2 contains burnable poisons and fission products of low or moderate cross sections, and Group 3 contains saturable fission products. Arrays 1 and 18 are single-dimension arrays with one entry for each Group 1 isotope. Arrays 2, 3, 4, 10, 11, and 12 are two-dimensional arrays with an entry I, J indicating the production of Isotope J from Isotope I, and the diagonal (I, I) representing the total removal from Isotope I. Array 5 is also

two-dimensional with entry I, J indicating the shielding of the resonance of Isotope I due to the presence of Isotope J. The diagonal (entry I, I) is the fraction of the resonance integral not subject to cross shielding. Arrays 6, 7, 8, and 9 are two-dimensional arrays where an entry I, J is the I-1 order coefficient in a fourth order polynomial representation of g or s versus neutron temperature for Isotope J. Array 14 is a two-dimensional array in which an entry I, J represents the fractional fission yield from Group 1 Isotope I to the Jth isotope in Groups 2 and 3. If there are four isotopes in Group 2 and three in Group 3, J would range from 1 to 7 and J is 5 for the first isotope in Group 3. Array 15 is a two-dimensional array in which an entry I, J represents the neutrons per fission of Isotope I. For J equal to 1 the values are for thermal fission and the two low energy epithermal groups. For J equal to 2 the values are for the fast group. Arrays 16, 17, 21, and 22 are two-dimensional arrays in which an entry I, J is the value of Z_o or D_c used to calculate Z for Isotope I. For J equal to 1 the values are for the lower integral, and for J equal to 2 they are for the upper integral. Arrays 19 and 20 are two-dimensional arrays in which an entry I, J is the value of s for the resonance integral for Isotope I. For J equal to 1 the value applies to the upper epithermal range while for J equal to 2 it applies to the fast group.

To prepare a tape, the arrays are filled with data and then copied to tape as follows. First the arrays are zeroed, except the following:

<u>Array</u>	<u>Element</u>	<u>Entry</u>
5	Diagonal (I, I)	1.0
6	First Column (1, J)	1.0
8	First Column (1, J)	1.0
15	All	3.0
18	All	200.0

Second, a previously prepared tape may be read as directed by input control integers. Third, data cards of Format 3I4, E12.4 are read and the data entered in the arrays. The first I4 field indicates the array, the second, the index of a single dimensional array or the second and third together indicate the position in a two-dimensional array where the datum in the fourth field is to be stored.

CURVES reads selected data placed on tape by other programs, and prepares a tape for use by the Benson-Lehner PLOTTER for automatic graphing.

DESCRIPTION OF SUBROUTINES

The MELEAGER-ALTHAEA code is composed of nine Fortran subroutines plus various FAP coded service subroutines. As generally used it is "chain" linked to other cycle analysis codes to minimize user time and effort. Each subroutine will be discussed separately.

Main Program

The main or input program reads the input data from tape into the storage array section of memory. In the case of a zero-dimensional problem, as indicated by NREG = 0, only one case is read and subroutine "LOOP" is called. If a diffusion problem is being computed, one case for each region (Up to the maximum of 24) is read into the storage array. NREG is the number of regions in the problem. The dimensional constants which will be needed later for the diffusion calculation are calculated and subroutine LOOP is called. When all the computations are completed and loop has returned control to the main program, N6 is interrogated, and if additional problems are to be solved, the reading process is repeated. If there are no problems, the next chain linked program indicated by N1 is called.

Subroutine LOOP

This subroutine selects data from the storage arrays and transfers it to the working section of memory used by each calculation. The selected input data are printed out to provide a record to go with the calculated output. Subroutine SIGMA is called to calculate the physics data for the input conditions. If N18 is zero concentrations of saturable isotopes such as Xe^{135} are computed and SIGMA is again called to recalculate the physics data. If one of the special options, as directed by N17 is requested, subroutine SETZ is called to perform the required calculations.

If the calculations have proceeded satisfactorily to this point, subroutine ITR is called to carry out the desired burn-up calculations. If not,

a message commenting on the failure is printed and control is returned to the main program. Upon return of control from ITR if the problem is zero-dimensional and if NPASS indicates one or more recycle calculations are to be made, subroutine RECYC is called. NPASS is reduced by one and control is returned to a point near the beginning of LOOP for the burn-up calculation of recycled fuel.

If a diffusion (one-dimensional) problem is being computed, the necessary data to continue the burnup of the current region are moved from the working section of memory to the storage arrays and control is returned to the beginning of LOOP to process the next region. When all regions have been "burned" for the period of time given in the input data, subroutine ALTH is called to carry out the diffusion calculations. It should be noted that ITR is not called in a one-dimensional case until the switch KEY is set positive in subroutine ALTH. Thus ALTH may call for as many complete cross section calculations as the data require before burnup begins. If further burnup is to be calculated, ALTH resets the region index and control is returned to the beginning of LOOP to calculate the next burnup period. Upon completion, control returns to the main program to read the next problem.

Subroutine SIGMA

All of the physics data are computed in subroutine SIGMA. Concentrations of each isotope and microscopic cross sections are used to calculate the resonance shielding and thermal flux depression to obtain effective g and s factors for the calculation of effective cross sections. The effective neutron temperature and the spectral index r is calculated and the above calculation repeated if a change from the previous value in excess of one degree centigrade occurs.

Flux time integrals of neutron production, neutron absorption and reactivity, time integrals of flux and heat generated (exposure), and their instantaneous values are calculated. Maximum-to-average and minimum-to-average ratios of flux or power are calculated.

Subroutine SETZ

If N17 is nonzero, this subroutine is called to perform one of several initializing options. The value of N17 determines the computational branch entered as described under special options.

Subroutine ITR

This is the main iterative loop of ALTHAEA which produces the output from the program and performs all the necessary tests regarding termination of the calculation. Output is written on Tape KM upon entry to ITR and each N8 iterations thereafter. If the economics switch, NSTEP, is nonzero, output is written on Tape 8 at time 0.0 and at each NSTEP multiple of PERIOD thereafter. Subroutine RK is called to carry out the burnup and then subroutine SIGMA is called to recompute the physics data.

If end-point calculations are desired (NS5 and NS6) linear extrapolation or interpolation is used to estimate the time step required to reach the desired end-point. If additional end-points are desired (NS4) the next value is entered as BMULT and calculation continues. When all of the desired end-points have been calculated or when TMAX is reached, control returns to subroutine LOOP.

Subroutine RK

This subroutine calculates the concentrations of each isotope as burnup proceeds using the Gill modification of the Runge-Kutta method. In simple terms, this method solves each time step in four stages. First, the current concentrations are used to obtain a rate of change of each isotopic concentration and a prediction of the concentrations at the midpoint of the desired time step is made. Second, the predicted concentrations are used to obtain corrected midpoint concentrations. Third, the corrected midpoint concentrations are used to obtain predicted endpoint concentrations. Fourth, the predicted concentrations are used to obtain corrected final concentrations using essentially a Simpson's rule summation. An error function based on this last correction is computed for each isotope with nonzero concentration at the beginning of this time step for comparison to the input values of ERU and ERL. The rate of change of concentration of

an isotope is equal to the rate of production minus the rate of destruction and may be very small when these components are nearly equal (i.e., the isotope is "saturated"). The error function is therefore

$$E_i = \frac{Z_i}{(H)(D_i)} \quad (38)$$

where

E_i = the error function for Isotope i

Z_i = the correction computed in step four

H = the time step

D_i = the larger of the rate of change of the isotopic concentration or the rate of destruction.

The largest error is found and compared to ERU. If it is larger, the time step is halved and the calculation repeated. If it is smaller than ERL, the time step for the next calculation will be doubled or set equal to TSLIM from input, whichever is smaller.

If the code is using the constant power option ($N5 = 1$), the flux will vary as the fissionable isotopic concentration changes. The fractional change in flux is also compared to ERU and the time step is halved if the change is larger than ERU. When the economics switch, NSTEP, is a positive integer, the time step is also limited such that a calculation is made no further than the next integral multiple of PERIOD before control is returned to the calling routine. N15 steps will be made if this limit is not reached or imposed.

Subroutine RECYC

This subroutine is called to alter the isotopic concentrations based on the decay constants and thus calculates changes that occur while a fuel is outside the reactor being reprocessed or in the reactor during a period of zero power. The concentrations of all isotopes are altered using the decay constants and the value of D DAYS for the "out-of-pile" decay period. Up to eight isotopes may be recycled to the next burn-up calculation as designated by integers N34 through N41. N31 designates

the number of isotopes to be recycled using a separations plus fabrication loss of PERLOS and N32 designates the number, if any, to be recycled using the separations plus fabrication loss of PERLS. If N64 is zero, all other isotopes are reset to their time zero concentrations. If N64 is one, the isotopes not listed on Card 7 are recycled without loss. If N64 is two, the Group 4 concentrations times PERLS are subtracted from the Group 1 concentrations of the isotopes listed on Card 9 before the process loss PERLOS is applied to the isotopes listed on Card 7. The concentrations of the isotopes not listed on Card 7 are reset to their time zero values.

If N27 is zero or blank, the initialization in subroutine SETZ will be the same as the base case. If N27 is nonzero, a new control card to replace Card 5 will be read from Tape 2 (A2).

For the period of zero power calculation, the concentrations of all isotopes are altered using the decay constants and value of TIMEDR, the number of days the reactor is shut down. The shutdown schedule is provided on Card 8.

Subroutine ALTH

This subroutine prepares additional constants (not already computed in the main program) for the diffusion calculation and calls subroutine CALC to make the "control rods removed" calculation. If fuel shuffling is required, the pertinent data are moved in the storage array from the area representing one region to another as directed by N50 and N55 of the input. If control action is required, the corrections are made as directed by ACNTRL and NTRL. Subroutine CALC is called as required to make the necessary diffusion calculations. Diffusion output is prepared and written to Tape KM. The average flux and spectral index values for each region are computed and saved in the storage array for subsequent selection and use by LOOP and other subroutines.

If the reactivity is below the desired value (-BMULT) on the first calculation, switches are set to terminate the calculation. If the reactivity is below the desired value after the first calculation, switches are set for a one second terminal burnup and control returned to LOOP. At the second,

and each subsequent calculation when reactivity is above -BMULT, a linear extrapolation of reactivity is made and, if the required time increment is less than PERIOD, switches are set for a terminal burnup of the required amount and control returned to LOOP. When control returns to ALTH after a terminal burn-up fuel is shuffled if required and calculations proceed as if this was the first entry into ALTH or switches are set to read the next problem before control is returned to LOOP. In the event this was the last problem (N6 of the last region equaled one) control will go to subroutine EDIT to rewrite the B8 economics data tape before calling the chain link indicated by N1.

Subroutine CALC

Final constant preparation and the diffusion calculation is done in this subroutine. To maximize speed of calculation, macroscopic adsorption, scattering and fission cross sections which are generated as averages in each region are stored in arrays indexed by mesh point. The central loop of the diffusion calculation thus requires little or no indexing calculation. Thermal and epithermal fluxes by mesh point, as well as reactivity, are thus calculated. After convergence, or after 100 iterations in the event of failure to converge, region average fluxes and power are computed.

Subroutine EDIT

Economics data on the B8 tape are written as computed so the time zero data for Region 1 to NREG are first followed by data at Time = PERIOD and Time = 2 PERIOD, etc. This must be rearranged so that all of the data for Region 1 are first followed by the data for Region 2, etc. Subroutine EDIT accomplishes this by rewriting the B8 tape as follows:

As much as possible of B8 is read into memory and the balance is stored on a scratch tape on A5. B8 is rewound and the contents of memory is written on B8. The data for as many regions as possible are read from A5 to memory, the data for 1 region are copied directly to B8 and the balance is stored on a scratch tape on B7. The data from memory are written on B8. The data for as many regions as possible are read from B7 into

memory and the balance saved on A5, etc., until all the data are copied to B8.

This subroutine is written in FAP to permit variable array sizes and to buffer both reading and writing.

Subroutine RDYTP

This subroutine is used before a tape is to be written on or read the first time. It will rewind the tape if it is ready but if not a message is printed online telling the operator to make it ready. A second entry, UNLDTP, is used to unload a tape. This subroutine is coded in FAP and must be revised if this program is to be used on a 709 computer.

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APPENDIX A

ZERO-DIMENSIONAL FUEL CYCLE BURNUP

ZERO-DIMENSIONAL FUEL CYCLE BURNUP

Once the appropriate set of cell averaged constants to represent the characteristics of a reactor or portion of a reactor is chosen, the burnup calculation can be carried out. If the reactor is large and the fuel is managed in such a way that the chosen constants properly represent average conditions, a single zero-dimensional burnup calculation will be adequate. In other cases, it may be necessary to represent the reactor by several regions, with the diffusion subroutines calculating the flux for each. Since each region of a multiregion representation of a reactor is similar to the single region case in the zero-dimensional representation of a reactor, the input for a series of zero-dimensional cases will be described.

The user must first select an isotope arrangement and prepare a physics library tape. Using the input data sample (Appendix E) as a guide, the values of the required input constants to be selected will now be discussed.

INPUT CARDS

The first seven cards are of the same format, containing nine integers followed by four floating point numbers.

Card 1

The first three integers on Card 1 define the isotope array sizes, which must be consistent with the library data. The sample problem uses a typical arrangement of 14 fuel isotopes, four Array 2 isotopes consisting of one burnable poison and three pseudofission products to simulate the low and moderate cross section fission products, and three Array 3 high cross section or saturable fission products. The fifth integer is 3, designating the normal output tape to receive the output. The seventh integer is 1, designating the first case of a study. The fourth, sixth, and eighth through thirteenth numbers are zero, or blank, since this is a zero-dimensional example.

Card 2

The first integer of the second card is set equal to minus 1 to obtain the maximum output. To reduce the output such that cross section data are given at the beginning and end only, zero or blank would be used, while plus 1 would reduce the output to concentrations only at intermediate printouts.

Numbers 2, 3, 4, 10, 11, 12, and 13 on the second card deal with the termination of the calculation. BMULT, the tenth number, is the reactivity limit or negative of the exposure limit as determined by NS5, the third integer. If NS5 is 1, BMULT is the desired instantaneous or batch terminal reactivity, while 2 designates averaged flux time or graded reactivity. A value of 3 indicates BMULT is the negative of the desired terminal exposure, while 4 causes BMULT to be ignored and the calculation to continue to TMAX, the thirteenth number. If NS6, the fourth integer, is equal to 1, the first computation past the value of BMULT is accepted as the final calculation, while a value of 2 causes further calculations in an effort to obtain a final calculation closer to BMULT. The desired closeness of approach is EAE, the twelfth number. If, however, the time step in any iteration of the search procedure becomes less than HLO, the eleventh number, that calculation is accepted as the final calculation. NS4, the second integer, on Card 2 controls the number of end point calculations made. If it is 1, the end point calculation at BMULT is the end of this case; if it is 2, calculations will be resumed until TMAX is exceeded; and if it is 3, NS5 is set equal to 2 after the BMULT end point is computed and nonzero values of BMULT1, BMULT2, etc., from Card 3 are used for up to four additional end points. The fifth integer, N6, is set to 1 for the last case and to 2 for others. The sixth integer is the number of isotopes not subject to flux depression and is therefore normally zero. N8, the seventh integer, limits the frequency of output writing and is the number of physics calculations per output. N15, the eighth integer, is the maximum number of Runge-Kutta time steps permitted between physics calculations. The ninth integer, N9, is zero or blank for normal output, and nonzero if extra output of use only to the programmer is desired.

If the number of milliseconds between external signals (see 7090 reference manual) is entered as N9, the computation time for each case is given in the output. Negative values for N9 also provide information from the SIGMA routine and individual errors from the Runge-Kutta routine.

Card 3

The third card is left blank unless multiple end points described above or special options described on Page B-8 are used.

Card 4

The first integer on Card 4, N1, is the number of the next chain link on Tape 4 (A4) to be called. The second integer is not used.

The third integer, N3, if 1, indicates that SNF (Card 202) is the macroscopic nonfuel absorption cross section per unit fuel volume; or if 2, that SNF is the initial thermal utilization. N4, the fourth integer, controls the treatment of SPEC (Card 201). If N4 is 1 SPEC is SDPV, the slowing down power of the cell per unit fuel volume; if N4 is 2, SPEC is the initial spectral index R and SDPV is calculated; if N4 is 3, SPEC is R, which is held constant throughout the calculation. If N5, the fifth integer, is equal to 1, PWI (Card 202) is the fixed power; if it is 2, PWI is the fixed flux. The sixth integer N10, is the number of changes to the physics data on the library tape. Each change requires an input card of the same format as the PREP code following Card 9. The seventh, eighth, and ninth integers are used in the diffusion calculation and should be left blank. TSLIM, the tenth number on Card 4 is the upper limit for the Runge-Kutta time step in seconds.

ERL and ERU, the eleventh and twelfth numbers, are the lower and upper fractional error limits for use in Runge-Kutta. If the estimated error in burnup for each isotope is less than ERL in any given time step, the next time step will be doubled. If the estimated error for any isotope is greater than ERU, this time step is cut in half and the calculation repeated. H, the thirteenth number, is the initial Runge-Kutta time step in seconds.

Card 5

This card, described on page B-3, is concerned with special initialization options, and may be left blank.

Card 6

The integers and the thirteenth number on this card are concerned with special options and may be left blank. The tenth and eleventh numbers, RAYK1 and RAYK2, are used in the spectral index and cross section calculations. RAYK3, the twelfth number, is used in the calculation of neutron temperature.

Cards 7, 8, and 9

These three may be left blank since they are concerned with special options. For their use see Appendix B.

Library Change Cards and IMY Cards

The number of cards in this group varies. Library change cards, if any, as directed by N10, follow Card 9. In CASE GENERATOR, these cards are assigned numbers beginning with 11. Isotope title, descendent, and concentration cards are next, and are assigned numbers beginning with 101. There are $N11 + N12 + N13$ of these cards. Each card represents an isotope and may carry a six-letter title which is used for description in the output; however, all references to concentration, cross section, etc., are entirely due to card order in the deck. Radioactive decay, fission product production, and secondary product formation such as n-2n reactions are determined by the positions of the constants on the library tape.

The second field is the sequence number of the descendent produced by neutron capture or the isotope's own number, if there is no descendent in the list. The third field is the concentration of the isotope in atoms per barn-centimeter. The fourth field is the sequence number of the n-2n descendent. Blank or zero indicates no n-2n reaction.

Physics Cards

Two cards of physics data assigned Numbers 201 and 202 follow the isotope cards. The first number on Card 201, FCONST is the coefficient of the linear term in the flux disadvantage factor equation, while the second

number FEXP, is the coefficient of the exponential term. The third number, TEMF, is the fuel temperature in degrees centigrade used to calculate Z, the height of the NR1A resonance in Equation (10) (page 12). SPEC, the fourth number, is the slowing down power of the cell per unit fuel volume, or the spectral index, depending on the setting of N4.

The fifth number, TNL, is the thermal nonleakage probability for use in the calculation of the spectral index, but it is not used to correct the infinite multiplication constant of the cell. TEM, the sixth number, if positive, is the moderator temperature; if negative, is the neutron temperature at initial conditions from which a moderator temperature is computed. IE, the seventh number, is the sequence number of the isotope used to calculate the fast flux.

The first number on Card 202, SNF, is the initial thermal utilization, or the macroscopic absorption cross section for everything in the cell not represented by one of the isotopes in the isotope list (see N3 on Card 4). This cross section must be obtained by flux weighting the cross sections of clad, structure, and moderator. This weighted result is expressed in square centimeters per cubic centimeter of fuel. The second number, SCA, is the effective epithermal macroscopic potential scattering cross section and is used in the calculation of the resonance shielding of all isotopes. PWI, the third number, is the value of the flux or power input as directed by N5. DENS, the fourth number, is the density of the fuel isotopes in the fuel zone for use in the exposure calculation. E, the fifth number, is the ratio of total fissions to nonfast fissions if positive, or epsilon if negative. The value of E is based on the isotope designated by IE. The sixth number, XK, is the conversion factor required to obtain exposure in the desired units. If negative, an internally stored value of $1.4533\text{E} + 11$ is used. This produces exposure in megawatt days per U. S. ton if the fissionable isotope concentrations are in atoms per barn-centimeter, the fission cross section in barns, the flux in neutrons per barn second, the time in seconds, the fission energy in MeV per fission, and the density in grams per cubic centimeter. If XK is zero or left blank, $1.60206\text{E} + 11$ is used to produce MWd/tonne. JMP, the seventh number, is the number of cards following to be read and rewritten

as comments or heading cards for the case. Column one of the comment cards has the usual printer carriage control function (one for a new page, zero for double space, and blank for single space).

Subsequent cases follow the first, beginning with Card 2. Card 1 data are saved for use in all the cases.

APPENDIX B
SPECIAL OPTIONS

SPECIAL OPTIONS

The great utility of the ALTHAEA code lies in the ability to couple directly via Tape 8 to other codes, especially QUICK, the economics code, and PLOTTER, a code to summarize output. The special option of the preparation of this data is primarily controlled by Cards 8 and 9. Other options to be discussed in this section are as follows:

- An initialization option to simplify specification of starting conditions is controlled by Card 5.
- A recycle option to study successive use of bred fuel is controlled by Cards 6 and 7.
- The zone spectrum option is a special case of both the initialization and recycle options.
- The burnable poison option provides for special thermal self-shielding of lumped high cross section poisons and is controlled by Card 3.
- The one-dimensional option is controlled by Card 1 and permits combination of up to 24 zero-dimensional cases as a single problem.

DATA TAPE OPTION

NSTEP, the first integer of Card 8, controls the use of this option. Zero or blank will bypass the option. A negative value of NSTEP causes output to be written on Tape 8 at time zero and each time a new physics calculation is made. A positive value will cause output to be written at time 0.0 and at multiples of PERIOD thereafter.

PERIOD, the sixteenth number on Card 8 is a fraction of the accounting interval in days such that NSTEP times PERIOD is the accounting interval. TIMEDR and DEL, the sixteenth and seventeenth numbers on Card 9 are used to modify PERIOD. DEL is the delay time in days from the beginning of the first accounting period until the fuel is charged into the reactor. TIMEDR is the reactor down time in days for refueling. The burnup for the first partial accounting period will thus be PERIOD minus DEL and some subsequent burnups will be PERIOD minus TIMEDR;

therefore TIMEDR and DEL must not exceed PERIOD. IC1 through IC12, the fourth through fifteenth numbers on Card 8, are the sequence numbers of the partial accounting periods, following which the reactor is down for refueling. For example, assume that PERIOD is 182.5, IC1 is 2, IC2 is 4, and that TIMEDR and DEL are 30. Under these conditions the Tape 8 data for the first 2.5 years are as follows:

<u>Time in Days</u>	<u>Days of Burnup</u>
0.0	0.0
152.5	152.5
335.0	335.0
517.5	487.5
700.0	670.0
882.5	822.5

IDENT, a four digit number and one letter (Columns 3 through 7 on Card 8) is written to Tape 8 with the data as identification. NFS, the third number on Card 8, is the number of the fueling scheme used (see the QUICK code description, HW-71812) and is placed on Tape 8 for QUICK.

Proper accounting for use charges requires separate burnup of certain input isotopes in the presence of newly created isotopes, for example, plutonium in a reactor containing U^{238} . Card 9 controls this option. The first four integers (1D1 through 1D4) provide for the definition of up to four sub-groups that will be treated separately, for example, four plutonium isotopes, three americium isotopes and one neptunium isotope. The next ten integers (1F1 through 1F10) are the sequence numbers of the isotopes from the fuel, grouped as indicated by 1D1 through 1D4, whose initial concentrations and continuously computed cross sections are to be used. These isotopes are sometimes referred to as Array 4 isotopes.

Under some circumstances it may be desirable to obtain economics data for recycled fuel only so that integer fifteen on Card 9, N57, may be used as follows. NSTEP is initially zero, so no economics data are saved until the last recycle run is computed, then N57 is inserted in NSTEP.

INITIALIZATION OPTION

Card 5 controls the special initialization options in the subroutine SETZ. N17, the first integer, determines the option used, or if zero, bypasses the subroutine. There are nine options:

- Option 1 normalizes all the isotopic concentrations such that their sum equals TOT the thirteenth number on Card 5. This is predominantly used with the recycle option (p. B-4) and therefore considers the isotopes in two groups. Those isotopes listed on Card 7 (N34 through N41) are in one group and their concentrations are held constant, while all other isotopes are in the second group. The concentrations of the isotopes in the second group are adjusted up or down by the same factor to obtain the proper sum.
- Option 2 provides for adjustment of concentration to obtain a desired initial reactivity given as CMULT, the tenth number on Card 5. For this option, the isotopes are considered in three groups. The concentrations of the recycle isotopes, listed on Card 7, are held constant. The concentrations of the second group, as defined by N20 through N25 (the fourth through ninth integers on Card 5) are adjusted by a factor to obtain the desired reactivity. The concentrations of the remaining isotopes are adjusted by a factor so that the sum of all concentrations equals TOT. Adjustments are made in an iterative sequence involving subroutines SETZ and SIGMA, since cross sections are dependent upon concentrations. The iteration continues until the reactivity is within plus or minus CMATCH of CMULT. CMATCH is the eleventh number on Card 5. If some of the same isotopes are in both Groups 1 and 2, convergence will be slow and the relative concentrations in Group 2 will be altered.
- Option 3 treats the isotopes in three groups to obtain the fractional fissile isotope content given as PERFIS, the twelfth number

on Card 5. The groups are the same as Option 2. The recycle group (Card 7) is held constant while Groups 2 and 3 are adjusted to obtain the desired PERFIS and TOT. The allowable error of PERFIS is given as CMATCH.

- Option 4 is similar to Option 2 except that Card 7 is not considered.
- Option 5 is similar to Option 3 except that Card 7 is not considered.
- Option 6 is provided for use with the burnable poison mode (described on page B-8). First the isotopes listed in N20 through N25 are used to obtain the desired fractional fissile content as in Option 5. Then the concentrations of the isotopes listed as the second, fourth, sixth, and eighth integers on Card 3 are adjusted to obtain the reactivity given as CMULT.
- Options 7 and 8 are associated with the zone spectrum mode described on page B-6.
- Option 9 adjusts the reactivity to CMULT by adjusting the concentrations of the isotopes listed in N20 through N25 by adding amounts in the ratio found in the original input to any amount that may be present due to recycle.

N18, the second integer on Card 5, controls the calculation of the concentration of the high cross section or saturable fission products at time zero. If N18 is zero, the equilibrium value of the high cross section fission products is computed for the time zero fission rate. If N18 is nonzero no computation is made for these isotopes at time zero. This integer is effective regardless of the setting of N17. N19, the third number on Card 5, is the maximum number of iterations allowed. N20 through N25 are the sequence numbers of the isotopes to be used in the adjustment calculation. If fewer than six isotopes are to be named the remaining integers may be zero or blank.

RECYCLE OPTION

The special option for automatic recycle of isotopes is controlled by Cards 6 and 7. If NPASS, the first integer on Card 7, is zero this option is bypassed. The number of recycle calculations desired is entered in NPASS

to activate the option. The first step in the recycle process is to adjust the concentrations of all isotopes present at reactor discharges for radioactive decay. The time period used is DDAYS, the eleventh number on Card 7. Second, the concentrations of selected isotopes, which are listed as the second through ninth integers on Card 7 are adjusted for process loss, and the concentrations of all other isotopes are set as directed by N64, the fourth integer on Card 6. This integer may have four values.

- If N64 is zero the recycle procedure is controlled as follows:
The fifth integer on Card 6, N31, is the number of isotopes (listed on Card 7 beginning with N34) to be recycled subject to the fractional loss given as PERLOS, the tenth number on Card 7. The sixth integer on Card 6, N32, is the number of isotopes (following those mentioned above) to be recycled subject to the fractional loss given as PERLS, the thirteenth number on Card 7. The concentrations of the remaining isotopes are restored to their time zero values.
- If N64 is 1 the procedure is the same as for zero except that the isotopes not listed on Card 7 are recycled without loss.
- When N64 is 2 the residual amount of the initial concentration of each isotope listed on Card 9 times DEXP is subtracted from the total concentration of that isotope and then recycle proceeds as if N64 were zero.
- When N64 is three, N31 pairs of isotopes listed on Card 7 are recycled according to the following equations:

$$Y(N38) = Y(N34)*(1. - \text{PERLOS}) + Y(N38)*(1. - \text{PERLS})$$

$$Y(N39) = Y(N35)*(1. - \text{PERLOS}) + Y(N39)*(1. - \text{PERLS})$$

$$Y(N40) = Y(N36)*(1. - \text{PERLOS}) + Y(N40)*(1. - \text{PERLS})$$

$$Y(N41) = Y(N37)*(1. - \text{PERLOS}) + Y(N41)*(1. - \text{PERLS})$$

Concentrations of isotopes not listed, N38 through N41, are reset to their time zero values.

In some cases it is desirable to initialize each recycle calculation to a different value of PERFIS or CMULT. If N27, the first integer on Card 6, is nonzero a new Card 5 will be read from Tape 2 for each recycle calculation.

ZONE SPECTRUM OPTION

Another mode of operation referred to as the "Zone Spectrum" mode has been included in the consolidated version of the code. This mode treats a fuel management scheme in which the fuel is irradiated in varying spectra during successive portions of its burnup, presumably by shifting the fuel to a region of different moderator ratio or to another reactor. The code is run fictitiously in the recycle mode, but with the special provisions that no fuel modifications are made out of the reactor; and the exposure, power-time, and associated quantities are accumulated rather than reset to zero between recyclings.

An initial reactivity is set by adjusting the amount of moderator (really, slowing down power per unit volume of fuel). Burn-up proceeds until the reactivity reaches a low limit, and then the moderator is readjusted to yield the original reactivity. The process is repeated, and when reinitialization can no longer be attained even with a large moderator ratio, then the remnant fuel is given a final exposure with this large amount or an amount that produces the maximum reactivity. Dual limit operation is also possible in which the reactivity is constrained within specified limits in order to treat those fuels which exhibit a rising reactivity at the beginning of exposure.

SNF, representing the nonfuel absorptions, is adjusted at each shift to account for the change in the amount of moderator, or even the change in the relative flux in the moderator. The equation governing this is

$$SNF = A + B \left[1 + DFLUX \left(\frac{\Delta(SDPV)}{SDPV} \right) \right]$$

where

A = nonfuel, nonmoderator macroscopic cross section
(flux- and volume-weighted)

B = similar quantity for the moderator, corresponding to the reference value, SDPV.

DFLUX = 1 + K, where K relates the change in average flux in the moderator to change in SDPV.

$$K = \frac{\Delta(\bar{\phi}_{\text{mod}}) / \bar{\phi}_{\text{mod}}}{\Delta(\text{SDPV}) / \text{SDPV}}$$

Values for the SNF adjustment equation:

A is entered as CLOSE on Card 1.

B is entered as RAD on Card 1.

DFLUX is entered as DDAYS on Card 7.

Reference SDPV is entered as PERLOS on Card 7.

Limiting or final exposure value of SDPV is entered as RMATCH on Card 6. SDPV entered as SPEC on the first physics parameter card is only an estimate to get the initialization started, and for best operation should be substantially smaller than the value resulting from the initialization. When N33 = 1, high limit is entered as PERLS on Card 7, and DEXP on Card 7 is an increment by which the reinitialization values differ from the limits.

The following controls are necessary for Zone Spectrum Mode. N17 = 7 or 8 is the signal for the Zoned Spectrum Mode. If N17 = 8, PERFIS is used, as in N17 = 5, to set the fuel composition; then the reactivity initialization and burnup proceeds with N17 = 7.

N4 = 1

NS4 = 1

NS5 = 1 or 2

NREG and NNN1 (N51) must each be zero, thus disallowing simultaneous operation with either the diffusion or burnable poison modes.

N33 = 0 for single limit operation

= 1 for dual limit operation. (Allowed only for
N17 = 7 or 8)

NPASS = the number of acceptable SDPV adjustments

The logic for the handling and modification of SNF was derived with the assumption $N3 = 1$.

BURNABLE POISON OPTION

A positive entry in the first control integer of Card 3 is the signal for the mode treating self-shielded, burnable poisons. The whole card then becomes interpreted in a special way and NS4 must equal one.

Definition of the parameters on Card 3 may be found in the code data input instructions (Appendix D). A description of the theory and application of this code can be found in the thesis, "Combinations of Self-Shielded Burnable Poisons for Nuclear Reactor Control," by G. J. Busselman, HW-SA-2578.

ONE-DIMENSION OPTION

A one-dimensional problem may have up to twenty-four regions, each region being represented by a case similar to the zero-dimensional case as described in Appendix A. Unlike the zero-dimensional mode, a new Card 1 is used for each problem. That is to say, the input for Region 1 of a problem begins with Card 1, and for the remaining regions of that problem begins with Card 2. First, we will consider the added data required for any problem.

The eighth integer on Card 1, NREG, is the signal indicating the diffusion mode if nonzero and is the number of regions to consider as a single problem. The sixth integer, N50, is the number of the region from which the diffusion subroutine will obtain its time scale (time does not advance in a nonburning region). IG, the ninth integer, has three functions. The first two digits of this number specify the left and right albedo's respectively (either zero or one). The third digit is a 1 or a 2 designating slab or cylindrical geometry. The power level per unit of transverse dimension for flux normalization is given as the tenth number on Card 1. In cgs units this would be megawatts per centimeter of height in a cylinder or megawatts per square centimeter for a slab. RAD, the thirteenth number on Card 1, is the value of the left hand mesh point in the diffusion calculation and is usually zero or blank.

It is necessary that the diffusion subroutine ALTH control the termination of the calculation and not subroutine ITR so NS5 on Card 2 must be 1, 2, or preferably 4, and BMULT the negative of the desired end point reactivity.

The eighth integer on Card 4, NDC, is used to keep a running tally on the number of diffusion calculations and should normally be zero or left blank. KEY, the ninth integer, controls the progress of burnup of the region. Minus one indicates no burnup such as a reflector or other nonfuel region, while zero or blank indicates burnup is to be calculated.

The frequency of diffusion calculations is controlled on Card 8, using the same switch and controls as the economics output so that economics data are written to Tape 8 for each burning region during a diffusion calculation. The first integer, NSTEP, is set equal to a positive integer to indicate time control is desired, and the sixteenth number, PERIOD, is set equal to the desired number of days between calculations. The percent change in flux and spectral index is part of the diffusion output so that the user can judge easily if he must reduce PERIOD to obtain the desired precision in the burnup calculation.

In the diffusion mode, two additional data cards are required. They are referred to as Cards 203 and 204. They follow Card 202 and precede comment cards, if any. The first two numbers on Card 203 are the macroscopic transport cross sections of the epithermal and thermal groups respectively. The third is the macroscopic scattering cross section for the epithermal group times ξ , the average logarithmic energy decrement per scattering event. The fourth number SAWD, is the weighting factor for the macroscopic absorption cross section in the calculation of the diffusion constant.

$$D = \frac{0.333333}{SS + (SAWD)(SA)}$$

where

D = the diffusion coefficient

SS = the transport cross section

SA = the absorption cross section

The fifth number, VR, is the ratio of fuel-to-cell volumes, and permits conversion of data from unit fuel basis to unit cell basis. For nonfuel regions VR should be 1.0 and SNF should be the macroscopic absorption cross section. The sixth number, WIDTH, is the mesh point interval used in the diffusion calculation for the region. NP, the seventh number, is the number of mesh intervals in this region. Thus the radius (or thickness) of the region is NP times WIDTH.

The first number on Card 204, CONV, is the fractional agreement required between two successive calculations of the flux for each mesh point in this region to be used as a convergence criterion. Thus, regions of interest may have a more stringent requirement than others if desired. If this number is zero, or blank, a value of 0.01 will be used. BUCK, the second number, is the transverse buckling which permits introduction of a leakage term for the dimension perpendicular to the direction of integration. The third number, PF, is the fraction of the reactor fuel inventory this region represents; it permits fuel cost weighting in the economics code. If zero or blank, the values of NP, VR, DENS, and WIDTH will be used to calculate PF. The fourth number PWMX, is the maximum power permitted in this region in watts/cc of fuel. A value of zero indicates no limit. The fifth number, AZERO, is the time limit for burnup in seconds. The seventh number, KOUT, controls the amount of detail in the diffusion output. Zero, or blank is the normal condition, while +1 will suppress output for regions of little interest. Minus 1 or minus 2 in any region will cause a plot by mesh point of two-group fluxes normalized to 100. Minus 2 for a region will cause extra output for the region.

Values for the flux distribution and spectral index for the steady state condition will usually differ from values for the transient condition when the reactivity is not equal to 1.0. Therefore, a means of control has been provided as a special option if the fourth integer on Card 1, NTRL, is nonzero:

- If NTRL is 1, poison is added simulating control rods or soluble poison.

- If NTRL is 2, APWR type control is used, simulating removal of fuel from the reactor and replacement with poison of about the same macroscopic absorption cross section.
- If NTRL is 3, a third type of control is used in an effort to simulate the "variable leakage seed" described by A. Radkowsky,⁽²⁷⁾ and to simulate the zone spectrum or spectral shift reactors. This mode controls reactivity by interpolating between high reactivity values and low reactivity values of the transport cross sections and the slowing down cross section (Σ_{fs}). The high reactivity values are given as the first three numbers on Card 203 (SS1, SS2, and SSXI), while the low reactivity values are obtained from the eleventh through thirteenth numbers on Card 7. Variation in geometry to increase leakage to an adjacent region is simulated by decreasing the transport cross sections (increase in the diffusion coefficients).

The sixth number on Card 204, ACNTRL, is used in conjunction with NTRL to determine regional control application. Zero or blank indicates no control in this region. The relative magnitude of ACNTRL in the various regions determines the relative strength of the control used.

In the case of fuel removal type control, a limit must be imposed since the region may be defined such that all of the fuel in the region might not be in control rods, and even if it were, some small fraction of the flux will still reach the withdrawn fuel. Most importantly, from the coding standpoint, the fuel concentrations cannot be set to zero because their values would be indeterminate when reinsertion is attempted. RMATCH, the thirteen number on Card 6, is provided for this limitation. If RMATCH is zero or blank 0.01 will be used. Up to ten adjustments of control will be made to bring the reactivity to $1.0 \pm \text{CLOSE}$ (the twelfth number on Card 1).

A special option is provided to simulate partial refueling of a reactor. This option is signaled by inserting the negative of the number of refuelings desired for RMULT, the eleventh number on Card 1. Each time the reactivity falls to the value specified by BMULT, partial refueling

will occur beginning with the region designated by N50, the sixth number on Card 1. That region is refueled from the region designated by N55, the seventh number on Card 4. The source region is refueled next. Refueling proceeds stepwise with the source of the fuel and the next region to be refueled indicated by the absolute value of N55. The outermost regions are nonburning regions (KEY = -1) which serve only as storage arrays for fresh fuel data. In each of these source regions the number of diffusion mesh intervals (NP) are set equal to zero so that they have no influence on the diffusion calculation. Source regions are not "refueled" and the value of N55 indicates only the number of the next region to be refueled. When the absolute value of N55 is found that is equal to NREG, the fueling sequence is completed and the problem continues. Any order of fuel movement through the various regions may be arranged, and if desired, not all of the regions need be included in the sequence. A negative value of N55 indicates concentrations, and associated data are moved from region to region. A positive value indicates that the physics characteristics of the fuel, as presented on Cards 201, 202, and the first five numbers on Card 203, are also moved.

Proper handling of the economics data will require a final or "graded end point" data set on the B8 tape for "discharged" regions at the time of refueling. Intermediate or "batch end point" data sets may be desirable for the other regions of the reactor so that refueling times can be recognized. The ninth integer on Card 3 is used for this purpose. A value of zero indicates that no data are desired on the B8 tape for this region upon refueling. A positive value indicates that final values are required, while a negative value indicates that intermediate values are required.

Selection of Options

Because of the very large number of possible combinations of control parameters for which settings are made, one must consider the influence that any one choice has on all the other parameters. After the selection has been made, it is advisable to review them to insure that they are consistent. Most incompatible selections are obvious, such as first

requiring adjustment of the concentration of an isotope during initialization by listing it on Card 5, and then requiring that its concentration be held constant by listing it on Card 7. Some of the less obvious conflicts are listed in Table B-I.

TABLE B-I. Conflicting Options

	<u>Zone Spectrum Mode</u>	<u>Burnable Poison Option</u>	<u>One- Dimensional Mode</u>
Zone Spectrum Mode		X	X
Burnable Poison Option	X		X
Initialization Options			X
Multiple End Points	X	X	X
Automatic Recycle	X		X

APPENDIX C
WESTCOTT DERIVATIONS

WESTCOTT DERIVATIONS

The cross-section and flux notation used in the ALTHAEA code is based on that of Westcott.⁽¹⁾ However, because the Westcott system was devised for well-moderated neutron interactions with extremely dilute fuel, the version of this system has been extensively modified to extend its applicability to a harder spectrum. The modifications primarily include: corrections to the reaction cross sections in order to account for the resonance shielding,⁽³⁾ prescriptions relating the effective neutron temperature to the temperature of the moderator,⁽¹¹⁾ and provisions for introducing a fast fission factor and other geometric effects. These items are discussed in the main body of the report. For the convenience of the reader the basic Westcott formulation is reviewed on the following pages.

BASIS OF WESTCOTT'S FORMULATION

The Westcott system is based on the usual formula for the reaction rate

$$R = \int_0^{\infty} N\sigma(v)n(v) dv \quad , \quad (C-1)$$

where

- R = reaction rate, events/cm³-sec
- N = number density of a given isotope, number/cm³
- v = speed of the neutron with respect to the given isotope, cm/sec
- σ = cross section for the reaction, cm² barns
- n = neutron density, number/cm³

It is understood the value of σ in Equation (C-1) corresponds to some particular reaction (that is, fission, capture, etc.).

Westcott's reaction rate formula is essentially an integrated form of Equation (C-1):

$$R = N \hat{\sigma} n_o v_o \quad , \quad (C-2)$$

where

$\hat{\sigma}$ = the effective cross section for the reaction, cm^2

n_o = the total neutron density, number/ cm^3

v_o = a representative neutron speed, cm/sec .

Because N , n_o , and v_o can be specified in a given situation, the problem of calculating the reaction rate is changed from that of integrating Equation (C-1) to that of calculating an effective cross section for the reaction.

An expression for the effective cross section can be obtained by equating the right-hand sides of Equations (C-1) and (C-2):

$$N \hat{\sigma} n_o v_o = N \int_0^{\infty} \sigma(v) n(v) v dv \quad .$$

Solving for this $\hat{\sigma}$ and noting that by definition

$$n_o = \int_0^{\infty} n(v) dv$$

gives

$$\hat{\sigma} = \frac{\int_0^{\infty} \sigma(v) n(v) v dv}{v_o \int_0^{\infty} n(v) dv} \quad . \quad (C-3)$$

A cross section that varies as $1/v^*$ can be expressed in the form

$$\sigma(v) = \frac{\sigma_o v_o}{v} \quad (C-4)$$

Substituting this relation into Equation (C-3) gives

$$\hat{\sigma} = \sigma_o \frac{\int_0^{\infty} n(v) dv}{\int_0^{\infty} n(v) dv} \quad ,$$

so that $\hat{\sigma} = \sigma_o$ for a $1/v$ absorber.

*A material having such a cross section is said to be a "1/v absorber."

Two things concerning the above result should be emphasized. The effective cross section for a $1/v$ absorber is completely independent of the neutron speed distribution, and there is no restriction on the choice of the representative neutron speed v_0 . Thus, $\hat{\sigma}$ will equal the cross section evaluated at whatever neutron speed is specified in the defining equation [that is, Equation (C-2)], regardless of the distribution of neutron speeds.

In most cases a $1/v$ approximation is not adequate--mainly because of resonance peaks, as shown in Figure C-1. However, much of the simplicity of the approach outlined for $1/v$ absorbers can be retained by introducing a "non- $1/v$ " correction factor, so that the effective cross section is given by the product $g\sigma_0$. The g factor is obtained from experimentally determined cross-section values as will be shown. Note that the neutron speed distribution must be specified in order to evaluate g .

MAXWELLIAN DISTRIBUTION OF NEUTRON SPEEDS

The speeds of neutrons completely in equilibrium with the atoms of a moderator are accurately represented by Maxwell's gas equation.

$$n(v)dv = n_0 A v^2 e^{-B^2 v^2} dv, \quad (C-5)$$

where

$n(v)dv$ = number of neutrons having a speed between v and $v + dv$, number/cm³

n_0 = total neutron density, number/cm³

v = neutron speed, cm/sec

and

$$A = 4 \pi \left(\frac{m}{2\pi kT} \right)^{3/2}$$

$$B = \left(\frac{m}{2kT} \right)^{1/2},$$

where

$$\begin{aligned} m &= \text{neutron mass} \\ &= 1.6747 \times 10^{-24} \text{ grams} \end{aligned}$$

FIGURE C-1. Fission Cross Section for Pu^{239} as a Function of the Speed of the Incident Neutron

$$\begin{aligned}
 k &= \text{Boltzmann's constant} \\
 &= 1.38044 \times 10^{-16} \text{ erg/}^\circ\text{K or} \\
 &\quad 8.6164 \times 10^{-5} \text{ eV/}^\circ\text{K}
 \end{aligned}$$

$$T = \text{effective neutron temperature, } ^\circ\text{K.}$$

The effect of the parameter T is shown by Figure C-2 wherein Equation (C-5) is graphed for three different values of the temperature. It is evident from the figure that the most probable neutron velocity (hereafter denoted as v_p) is a function of the temperature, and can be obtained by equating the first derivative with respect to velocity of Equation (C-5) to zero. This gives

$$v_p = \sqrt{\frac{2kT}{m}} \quad (\text{C-6})$$

E_p , the energy that corresponds to this velocity, is $1/2 mv_p^2$ so that from Equation (C-6) one obtains $E_p = kT$.

For well thermalized neutrons (namely, neutrons in thermal equilibrium with the moderator), the temperature of the neutrons is essentially that of the moderator. A convenient reference point, therefore, is room temperature 293°K . Let this temperature and the corresponding most probable velocity be denoted as T_0 and v_0 , respectively. From Equation (C-6) the most probable velocity for room temperature neutrons is found to be 2200 meters per second, while the corresponding energy is 0.0253 eV. Note that the most probable velocity for a Maxwellian distribution at temperature T is related to the most probable velocity for a room temperature distribution by the expression

$$v_p(T) = v_0 \sqrt{\frac{T}{T_0}}$$

The average velocity associated with the Maxwellian distribution can be found by substituting Equation (C-5) into the defining relation

$$\bar{v} = \frac{\int_0^\infty vn(v)dv}{\int_0^\infty n(v)dv} \quad (\text{C-7})$$

FIGURE C-2. Maxwellian Distribution of Neutron Speeds at Three Different Temperatures

to obtain

$$\bar{v} = A \int_0^{\infty} v^3 e^{-B^2 v^2} dv ,$$

with the result that*

$$\bar{v} = \sqrt{\frac{8kT}{\pi m}} . \quad (C-8)$$

Combining Equation (C-8) with Equation (C-6) for $T = T_o$ [that is, $v_p(T_o) = v_o$] gives the following relationship between the average velocity at a temperature T and the most probable velocity at room temperature:

$$\frac{\bar{v}}{v_o} = \sqrt{\frac{4T}{\pi T_o}} . \quad (C-9)$$

It is convenient at this point to introduce the average cross section $\bar{\sigma}$ and to relate it to the effective cross section $\hat{\sigma}$. The defining relation for the average cross section is

$$\bar{\sigma} = \frac{\int_0^{\infty} \sigma(v) n(v) v dv}{\int_0^{\infty} n(v) v dv} .$$

Comparing this formula with Equation (C-3), which defines $\hat{\sigma}$, and with Equation (C-7) shows that for an effective cross section at room temperature

$$\hat{\sigma} = \bar{\sigma} \frac{\bar{v}}{v_o} . \quad (C-10)$$

*The integral is most conveniently evaluated by changing the variable to $X = B^2 v^2$ for then

$$\begin{aligned} \bar{v} &= \frac{A}{2B^4} \int_0^{\infty} x e^{-x} dx = \frac{A}{2B^4} (x + 1) e^{-x} \Big|_0^{\infty} \\ &= \frac{A}{2B^4} \end{aligned}$$

The "non-1/v" factor g can be defined as the ratio

$$g = \frac{\bar{\sigma}}{\sigma_0}$$

Substituting the value of the effective cross section from Equation (C-10) gives

$$g = \left(\frac{\bar{\sigma}}{\sigma_0} \right) \left(\frac{\bar{v}}{v_0} \right)$$

With Equation (C-9), this can be written

$$g(T) = \frac{\bar{\sigma}}{\sigma_0} \sqrt{\frac{4T}{\pi T_0}} \quad (C-11)$$

Equation (C-11) is actually used to compile the g factors as a function of temperature for various fuel isotopes.⁽¹⁵⁾

THE WESTCOTT FLUX SPECTRUM

In the following discussion it will be convenient to characterize the neutron speed distribution $n(v)$ as a flux ϕ , where

$$\phi = n(v)v ,$$

and to consider the neutron energy $E = 1/2 mv^2$ as the independent variable.* The functional dependence of the neutron flux upon the energy is called a flux spectrum.

The thermal reactor flux spectrum assumed by Westcott is a combination of Maxwellian distribution for the low or thermal neutron energies, a $1/E$ distribution for the high or epithermal energies, and a joining or cutoff function. The joining function is a particularly sensitive formulation because some important near-thermal resonances lie near the cutoff point (see Figure C-1). Much of the disagreement between various investigators in this field is in their choice of a joining function.

*This energy is considered to be with respect to the target nucleus and is not an absolute value.

The equation for the flux spectrum suggested by Westcott has the form

$$\phi(E) = \frac{E}{(kT)^2} e^{-E/kT} + \frac{\beta\Delta}{E} \quad , \quad \text{C-12}$$

where

- ϕ = neutron flux, number/cm²-sec
- E = neutron energy, eV
- T = neutron temperature, °K
- k = Boltzmann constant
- β = epithermal-to-thermal flux ratio
- Δ = joining function between the Maxwellian and the 1/E tail.

For thermal energies, Equation (C-12) will be essentially a Maxwellian distribution--a statistical distribution derived from the kinetic theory for a nonabsorbing medium. For epithermal energies, Equation (C-12) will be essentially a 1/E variation which is in accord with theoretical calculations for slowing down a nonabsorbing medium. (Note that the 1/E approximation is most accurate for slowing down in hydrogen and holds fairly well in all media for energies sufficiently below those of fission neutrons.) The joining function originally suggested by Westcott was designed to be zero for energies less than some "cutoff energy," μkT , and 1 for energies greater than this.* Westcott later modified the step function approach in order to present data based on four different formulations of Δ . These are shown in Figure C-3. The $\Delta 4$ formulation is generally used, however.

Using the neutron spectrum represented by Equation (C-12) gives the following formula for the effective cross section.

$$\hat{\sigma} = \sigma_0 (g + rs) \quad , \quad \text{(C-13)}$$

where

- σ_0 = "2200 m/sec cross section," cm²
- g = non-1/v factor for thermal energies

*Note that the ratio of the cutoff energy to the most probable energy in the Maxwellian is simply μ which in the $\Delta 4$ formulation is 3.681.

FIGURE C-3. Westcott Cutoff, or Joining, Functions

s = non-1/v factor for epithermal energies
 r = spectral index.

Both g and s are functions of the neutron temperature, except when the absorber obeys the 1/v law. In this case, g = 1, and s = 0. The spectral index r is a parameter that characterizes the fraction of epithermal neutrons in the spectrum and is related to β , the epithermal-to-thermal flux ratio, by the expression

$$r = \frac{\beta}{1 + b\beta} \quad , \quad (C-14)$$

where

$$b = \frac{4}{\sqrt{\pi\mu}} \quad . \quad (C-15)$$

Finally, the computational formula used to obtain the s factor is

$$s = \frac{\Sigma}{\sigma_o} \sqrt{\frac{4T}{\pi T_o}} - bg \quad , \quad (C-16)$$

where Σ is the infinite dilution resonance integral in cm^2 , and g is obtained from Equation (C-11).

APPENDIX E
ZERO-DIMENSIONAL SAMPLE PROBLEM

14	4	3	1	4	4	1.03	10000.	.0001	3.00E+08	DEM	2
-1	1	2	2	1						DEM <td>3</td>	3
0	1	1	1			2.000E+06	.02	.05	86400.	DEM <td>4</td>	4
						.65	.7	.7		DEM <td>5</td>	5
										DEM <td>6</td>	6
										DEM <td>7</td>	7
										DEM <td>8</td>	8
										DEM <td>9</td>	9
U233	M=	7	Y=							DEM	101
U235	M=	10	Y=	7.22000E-04						DEM	102
PU239	M=	8	Y=							DEM	103
PU241	M=	11	Y=							DEM	104
TH232	M=	9	Y=							DEM	105
U238	M=	3	Y=	2.11460E-02	N-ZN= 12					DEM	106
U234	M=	2	Y=	7.00000E-06						DEM	107
PU240	M=	4	Y=							DEM	108
PA233	M=	7	Y=							DEM	109
U236	M=	12	Y=	3.00000E-06						DEM	110
PU242	M=	14	Y=							DEM	111
NP237	M=	12	Y=							DEM	112
AM241	M=	13	Y=							DEM	113
AM243	M=	14	Y=							DEM	114
POISON	M=	15	Y=							DEM	115
FPA	M=	16	Y=							DEM	116
FPB	M=	17	Y=							DEM	117
FPC	M=	18	Y=							DEM	118
XE135	M=	19	Y=							DEM	119
SM149	M=	20	Y=							DEM	120
SM151	M=	21	Y=							DEM	121
.62414		.31207		735.		1.8284	.972	301.		6DEM	201
.1		1.0613		30.		8.6385	-1.0443			1DEM	202

FWR SIMULATION (DIABLO CANYON)

INPUT DATA

I= U233 M= 7 Y= -0.00000	I= U235 M= 10 Y= 7.22000-04	I= PU239 M= 6 Y= -0.00000	I= PU241 M= 11 Y= -0.00000
I= T232 M= 9 Y= -0.00000	I= U238 M= 3 Y= 2.11400-02	I= U234 M= 2 Y= 7.00000-00	I= PU240 M= 4 Y= -0.00000
I= PA233 M= 7 Y= -0.00000	I= U236 M= 12 Y= 3.00000-06	I= PU242 M= 14 Y= -0.00000	I= NP237 M= 12 Y= -0.00000
I= AM241 M= 13 Y= -0.00000	I= AM243 M= 14 Y= -0.00000	I= XE135 M= 15 Y= -0.00000	I= FPA M= 16 Y= -0.00000
I= FPU M= 17 Y= -0.00000	I= FPC M= 18 Y= -0.00000	I= XE135 M= 15 Y= -0.00000	I= SM149 M= 20 Y= -0.00000
I= SM151 M= 21 Y= -0.00000	I=		
N11 N12 N13 NTRL N4 N50 N16 NREG IG POWER RMULT CLOSE RAD			
14 4 3 -0 3 -0 1 -0 -0 -0.0000 -0.00 0 -0.0000 -0.0000			
N53 N54 N55 N56 N6 N7 N8 N15 N9 RMULT HLO EAE TMAX			
-1 1 2 2 1 -0 4 4 -0 1.0300+00 1.00 0+04 1.0000-04 3.0000+08			
N51 N52 N53 N54 N6 N7 N8 N15 N9 RMULT1 RMULT BMULT3 RMULT4			
-0 -0 -0 -0 -0 -0 -0 -0 -0 -0.0000 -0.00 0 -0.0000 -0.0000			
N1 N2 N3 N4 N5 N10 N55 NDC KEY TSLIM ERL ERU H			
0 -0 1 1 1 -0 -0 -0 -0 2.0000+06 2.0000-02 5.0000-02 8.6400+04			
N17 N18 N19 N20 N21 N22 N23 N24 N25 CMULT CMATC PERFIS TOT			
-0 -0 -0 -0 -0 -0 -0 -0 -0 -0.0000 -0.00 0 -0.0000 -0.0000			
N27 N28 N31 N32 N33 RAYK1 RAYK2 RAYK3 RMATCH			
-0 -0 -0 -0 -0 -0 -0 -0 6.5000-01 7.00 0-01 7.0000-01 -0.0000			
NPASS N34 N35 N36 N37 N38 N39 N40 N41 PERLCS DDAYS DEXP PERLS			
-0 -0 -0 -0 -0 -0 -0 -0 -0.0000 -0.00 0 -0.0000 -0.0000			
NSTEP IDENT NFS IC1 IC2 IC3 IC4 IC5 IC6 IC7 IC8 IC9 IC10 IC11 IC12 PERIOD W/F FUEL VOL			
-0 -0 -0 -0 -0 -0 -0 -0 -0 -0 -0 -0 -0 -0 -0.0000-0.0000			
ID1 ID2 ID3 ID4 IF1 IF2 IF3 IF4 IF5 IF6 IF7 IF8 IF9 IF10 IN57 TIMEDR DEL			
-0 -0 -0 -0 -0 -0 -0 -0 -0 -0 -0 -0 -0 -0 -0			
FCONST= 6.2414-01 FEXP= 3.1207-01 TEF= 7.3500+02 SPEC= 1.8244+00 TKE= 9.720 -01 TEM= 3.0104+02 IE= 6			
SNF= 1.0000-01 SCA= 1.0613+00 PIF= 3.3000+01 DENS= 8.6385+00 Z= -1.044 +00 XK= 1.8021+11 J/P= 1			

ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF
U233	6.422+02	.8970	1.7223	U235	4.764+02	.8406	-1.1032	PU239	1.309+03	1.5081	1.3715
PU241	1.272+03	1.1572	.3187	T-232	3.373-01	.0000	1.8144	U238	1.332+00	.0000	7.1719
U234	5.997+00	.9062	533.2167	PL240	7.015+00	1.1777	700.2001	P-233	0.000	.9040	-1.1195
U236	2.499+00	.0000	13.454	PL242	6.449+00	.8140	1683.1845	NP237	5.535+00	.9195	1563.2696
AM241	1.309+01	1.6550	13.606	AM243	4.413+00	.0000	23.7551				
ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA
U233	7.273+02	.9021	1.9377	U235	5.830+02	.8447	-.0627	PU239	2.055+03	1.7002	1.8225
PU241	1.778+03	1.2345	.3347	T-232	3.250+01	.9042	18.7776	U238	1.382+01	.9146	16.4972
U234	2.565+02	.9056	9.7657	PL240	2.556+03	1.0380	43.3711	PA233	2.978+02	.8384	29.2654
U236	1.290+02	.7692	111.605	PL242	3.849+02	.8154	75.3029	NP237	3.597+02	.9836	6.1612
AM241	1.493+03	1.4847	5.7447	AM243	5.544+02	.5279	11.5236	DISON	0.000	.9040	-1.1195
FPA	7.252+02	.9040	.0135	FPA	1.525+03	.9040	23.5698	FPC	4.532+01	.9040	.0135
XE135	2.220+06	1.1580	-1.306	SM149	6.725+04	2.0401	-2.1999	SM151	1.433+04	.9045	.2728
ISOTOPE	DESCENDANT	CONCENTRATION	DESTROYED	AMS	ABSAV	ALPHA	HEAT	POWER	CORV.		
U233	U234	-0.00000	0.00000	.00000	.00000	.13249	.00 0	.0000	.0000		
U235	U236	7.22000-04	0.00000	.42089	.42089	.22376	92.48 3	92.4863	.4200		
PU239	PU240	-0.00000	0.00000	.00000	.00000	.57033	.00 0	.0000	47.6819		
PU241	PU242	-0.00000	0.00000	.00000	.00000	.39857	.00-0	.0000	.0000		
T-232	PA233	-0.00000	0.00000	.00000	.00000	95.43949	.00 0	.0000			
U233	PU239	2.11460-02	0.00000	.22886	.22886	7.12416	7.5006	7.5006			
U234	U235	7.00000-06	0.00000	.00181	.00181	42.10825	.0111	.0111			
PU240	PU241	-0.00000	0.00000	.00000	.00000	363.36560	.0000	.0000			
PA233	U234	-0.00000	0.00000	.00000	.00000	.00000	.0000	.0000			
U235	NP237	3.00000-06	0.00000	.00039	.00039	50.61059	.00 0	.0020			
PU242	AM243	-0.00000	0.00000	.00000	.00000	56.66473	.00 0	.0000			
NP237		-0.00000	0.00000	.00000	.00000	.00000	.00 0	.0000			
AM241		-0.00000	0.00000	.00000	.00000	.00000	.0000	.0000			
AM243		-0.00000	0.00000	.00000	.00000	.00000	.0000	.0000			
FISSION PRODUCTS				.00305	.02305						
POISO= -0.000	FPA= -0.000	FPC= -0.000	FPC= -0.000	X135= 7.144-09	SM149= 7.536-03						
SM151= 1.479-07											
TIME/HEAT	PAV/PHI	PWR/HEAT	TRUIT/NA	PXV/PXVW	SIGSUM/R	ABSF/ABSP	XMLT/AM	TZTA/J	EPS/G	CR/FISHU	CAR/FRAH
0.00000	2.14283-11	3.00000+01	386.844	1.00000	.49625	.46765	1.17966	1.80207	1.05446	4.1019	.0000
0.00000	2.14283-11	0.00000	75.491	1.00000	.19575	.77636	1.17966	.78780	.79438	.4209	26.0771

ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF
U233	6.473+02	.8932	1.705	U235	4.787+02	.8948	1.705	PU239	1.141+03	1.3443	.9656
PU241	1.134+03	1.1156	.2294	TH232	3.479+01	.0000	1.779	U238	1.375+00	.0000	7.0340
U234	6.192+00	.9025	523.2634	PU240	7.014+03	1.1731	.071.9975	PA233	0.000	.9003	-.1361
U236	2.580+00	.0000	13.125	PU242	6.861+00	.8116	1051.9250	NP237	5.712+00	.9151	1533.2512
AM241	1.232+01	1.3880	13.9275	AM243	4.525+00	.0000	23.2983				
ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA
U233	7.339+02	.9983	1.9193	U235	5.875+02	.8483	.0815	PU239	1.708+03	1.4912	1.3416
PU241	1.040+03	1.1514	.2284	TH232	3.367+01	.9005	18.6058	U238	1.119+01	.9138	16.3801
U234	2.081+02	.9019	9.8143	PU240	2.032+03	1.0314	3.7033	PA233	3.097+02	.8353	29.1066
U236	1.143+02	.7670	93.5282	PU242	4.016+02	.8129	74.6498	NP237	3.657+02	.9763	6.0722
AM241	1.158+03	1.2774	5.3393	AM243	5.701+02	.5264	13.4453	POISON	0.000	.9003	-.1363
FPA	7.214+02	.9003	.0076	FPR	1.644+03	.9303	22.4315	FPC	4.509+01	.9003	.0076
XE135	2.185+06	1.1555	-1.3033	SM149	6.609+04	2.0316	-2.1936	SM151	1.429+04	.9009	.2655
ISOTOPE	DESCENDANT	CONCENTRATION	DESTROYED	ABS	ABSAV	ALPHA	HEAT	POWER	CONV.		
U233	U234	4.70541-14	3.83487-15	.00000	.00000	.13385	.00 0	.0000	.0003		
U235	U236	5.89370-04	1.33711-04	.34628	.33252	.22734	83.07 8	75.2922	.3797		
PU239	PU240	5.23436-05	1.79914-05	.02252	.05147	.54913	9.17 6	16.5664	46.8543		
PU241	PU242	1.35133-06	2.63625-07	.00222	.00072	.38543	.14 1	.4452	2.3219		
TH232	PA233	-0.00000	0.00000	.00000	.00000	95.77913	.00 0	.0000			
U236	PU239	2.10650-02	6.11509-05	.23566	.23318	7.13571	7.58 0	7.6536			
U234	U235	6.38705-06	6.17599-07	.00171	.00177	42.30360	.0108	.0104			
PU240	PU241	5.05880-06	1.53337-06	.01028	.00430	266.07447	.0037	.0093			
PA233	U234	2.77290-14	5.34093-14	.00000	.00000	.00000	.00 0	.0000			
U236	NP237	2.68139-05	6.36075-07	.00307	.00182	43.31191	.00 0	.0182			
PU242	AM243	6.08635-08	2.45877-09	.00002	.00001	59.29732	.00 0	.0001			
NP237		1.07365-06	6.03942-07	.00039	.00017	.00000	.00 0	.0016			
AM241		9.53202-09	1.24544-09	.00001	.00000	.00000	.00 0	.0000			
AM243		1.76263-09	9.30555-11	.00000	.00000	.00000	.00 0	.0000			
FISSION PRODUCTS				.04813	.03638						
POISON = -0.000 FPA = -1.229-06 FPR = 1.276-05 FPC = 9.220-05 X135 = 7.237-09 SM149 = 8.307-08 SM151 = 1.783-07											
TIME/HEAT	PAV/PHI	PWR/WRAT	TNEUT/KA	PMXV/PMNV	SIGSUM/KA	ABSF/ABSR	XSLT/AM T	ZTA/J	EPS/G	CR/FISBU	CAP/FRAH
1.64300+07	2.12107-11	3.00000+01	380.095	1.01026	.51161	.47522	1.12208	1.77907	1.05903	49.5566	15.1529
5.72222+03	2.12652-11	-7.04875-08	76.370	.99575	.19549	.64171	1.15153	.76329	.78779	.4411	25.6575
ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF
U233	6.403+02	.8962	1.670	U235	4.858+02	.8529	1.0745	PU239	1.070+03	1.2926	.7566
PU241	1.152+03	1.0932	.1625	TH232	3.261+01	.0000	1.0575	U238	1.283+00	.0000	6.5517
U234	5.822+00	.9657	488.5417	PU240	6.370+03	1.1776	.004.8093	PA233	0.000	.9035	-.1628
U236	2.419+00	.0000	12.2911	PU242	6.262+00	.8157	1542.5280	NP237	5.358+00	.9174	1428.3744
AM241	1.172+01	1.3176	12.3285	AM243	4.270+00	.0000	21.7012				
ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA
U233	7.329+02	.9013	1.8828	U235	5.977+02	.8561	.1174	PU239	1.646+03	1.4173	1.0945
PU241	1.590+03	1.1244	.1762	TH232	3.358+01	.9037	18.4663	U238	1.109+01	.9138	16.0719
U234	2.730+02	.9051	9.9354	PU240	1.356+03	1.0316	19.2843	PA233	3.094+02	.8366	28.6649
U236	9.668+01	.7726	77.9603	PU242	3.956+02	.8171	73.1676	NP237	3.622+02	.9750	5.9363
AM241	1.339+03	1.2236	5.1533	AM243	5.755+02	.5343	13.3098	POISON	0.000	.9035	-.1628
FPA	7.216+02	.9035	.0073	FPR	1.642+03	.9035	23.2242	FPC	4.510+01	.9035	.0073
XE135	2.195+06	1.1629	-1.3111	SM149	6.619+04	2.0387	-2.1994	SM151	1.429+04	.9040	.2483
ISOTOPE	DESCENDANT	CONCENTRATION	DESTROYED	ABS	ABSAV	ALPHA	HEAT	POWER	CONV.		
U233	U234	6.61507-13	1.53976-13	.00000	.00000	.13395	.00 0	.0000	.0000		
U235	U236	3.88630-04	3.35468-04	.23229	.31768	.23042	70.94 1	55.3476	.3358		
PU239	PU240	1.00951-04	1.14630-04	.16616	.10823	.53802	19.97 0	32.9211	49.1910		
PU241	PU242	1.07371-05	7.15181-06	.01707	.00641	.38006	1.33 1	3.8048	7.5288		
TH232	PA233	-0.00000	0.00000	.00000	.00000	101.94962	.00 0	.0000			
U236	PU239	2.68987-02	2.47969-04	.23172	.23412	7.60026	7.68 7	7.8212			
U234	U235	5.27625-06	1.71179-06	.00142	.00164	45.37533	.01 0	.0089			
PU240	PU241	2.22138-05	1.64096-05	.03149	.01596	211.44600	.01 0	.0427			
PA233	U234	1.44559-13	8.42552-13	.00000	.00000	.00000	.00 0	.0000			
U236	NP237	6.09944-05	3.94466-07	.00590	.00373	38.97367	.00 0	.0426			
PU242	AM243	1.66467-06	1.92182-07	.00066	.00018	62.18274	.00 0	.0030			
NP237		4.64736-06	7.62511-07	.00168	.00074	.00000	.00 0	.0072			
AM241		2.21346-07	8.65549-08	.00029	.00000	.00000	.00 0	.0007			
AM243		1.63742-07	2.18181-08	.00009	.00002	.00000	.00 0	.0002			
FISSION PRODUCTS				.07280	.05396						
POISON = -0.000 FPA = -3.343-06 FPR = 2.443-05 FPC = 2.721-04 X135 = 6.709-09 SM149 = 8.142-08 SM151 = 1.922-07											
TIME/HEAT	PAV/PHI	PWR/WRAT	TNEUT/KA	PMXV/PMNV	SIGSUM/KA	ABSF/ABSR	XSLT/AM T	ZTA/J	EPS/G	CR/FISBU	CAP/FRAH
4.64400+07	2.18469-11	3.00000+01	370.140	1.06935	.49221	.44737	1.32403	1.74023	1.05289	57.0556	9.5742
1.08333+04	2.33620-11	-2.34452-07	68.242	.96675	.19678	.28290	1.09574	.72598	.76959	.4163	26.4517

THE FOLLOWING INFORMATION IS THE BEGINNING AND END OF THIS CASE

U233 = -0.000 U235 = 7.224-04 PU239 = -0.000 PU241 = -0.000 T 232 = -0.000 U238 = 2.115-02
 U234 = 7.000-06 PU240 = -0.000 PA233 = -0.000 U236 = 3.000-10 P 242 = -0.000 NP 37 = -0.000
 AM241 = -0.000 AM243 = -0.000 POISON = -0.000 FPA = -0.000 F B = -0.000 FPC = -0.000
 XE135 = 7.144-09 SM149 = 7.514-03 SM151 = 1.479-07

TIME/HEAT PAV/PHI PWR/WRAT TFEUT/HA DMXV/PKXV SIGSUM/P ABSF/ABSR XNLT/AM T ZTA/J EPS/HA CR/FISRU CAR/FRAN
 .00000 2.14283-11 3.00000+01 306.864 1.00000 .49625 .46785 1.17900 1.00207 1.05440 4.01019 .0000
 .00000 2.14283-11 0.00000 75.491 1.00000 .18575 .76738 1.17900 .76780 .79438 .4209 26.0771

THIS DATA SET WITHIN 3.1282+04SEC OF THE SELECTED VALUE OF BNULT WHICH IS 1.0300+00

CALCULATED SNF= 1.0000-01 CALCULATED SDPV= 1.8284+00

ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF
U233	6.367+02	.9052	1.6305	U235	4.961+02	.8634	-.0397	PU239	1.044+03	1.2784	.6847
PU241	1.144+03	1.0928	.1512	T-232	2.771-01	.0000	1.4807	U238	1.095+00	.0000	5.8529
U234	4.976+00	.9149	438.2623	PU240	5.362+00	1.1901	534.2315	PA233	0.000	.9128	-.1828
U236	2.055+00	.0000	10.3801	PL242	5.337+00	.8251	381.0291	NP237	4.556+00	.9258	1276.4303
AM241	1.081+01	1.2595	11.5054	AM243	3.628+00	.0000	19.3870				
ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA
U233	7.234+02	.9103	1.4427	U235	6.107+02	.8665	-.1705	PU239	1.600+03	1.3977	1.0122
PU241	1.577+03	1.1227	.1455	T-232	3.203+01	.9129	18.2507	U238	1.049+01	.9229	15.6833
U234	2.618+02	.9143	9.8274	PU240	1.035+03	1.0389	14.1273	PA233	2.945+02	.8475	28.6304
U236	8.249+01	.7528	69.2757	PL242	3.556+02	.8287	68.6451	NP237	3.497+02	.9804	5.8165
AM241	1.264+03	1.2129	5.0602	AM243	5.466+02	.5437	13.0963	POISON	0.000	.9128	-.1828
FPA	7.265+02	.9128	-.0245	FPB	1.567+03	.9128	23.6310	FPC	4.541+01	.9128	-.0245
XE135	2.255+06	1.1780	-1.3273	SM149	6.774+04	2.0596	-2.2234	SM151	1.435+04	.9133	.2302

ISOTOPE	DESCENDANT	CONCENTRATION	DESTROYED	ABS	ABSAV	ALPHA	HEAT	POWER	CONV.
U233	U234	2.49894-12	1.41984-12	.00000	.00000	.13262	.00 0	.0000	.0000
U235	U236	1.98942-04	5.24137-04	.12150	.24248	.23118	58.61 0	35.1428	.2860
PU239	PU240	1.13521-04	3.13439-04	.10123	.14663	.53221	29.45 0	46.1951	54.9893
PU241	PU242	2.45470-05	4.08766-05	.03271	.01789	.37885	4.0422	10.4902	13.7886
TH232	PA233	-0.00000	0.00000	.00000	.00000	114.59181	.0000	.0000	
U233	PU239	2.26469-02	4.92876-04	.21667	.22904	8.36019	7.7903	7.9742	
U234	U235	3.91563-06	3.07024-05	.01102	.00142	51.50055	.0090	.0068	
PU240	PU241	4.72542-05	6.32252-05	.04691	.02903	192.33501	.04 9	.0889	
PA233	U234	3.42593-13	4.02002-12	.00000	.00000	.70000	.00 0	.0000	
U236	NP237	8.71191-05	1.13635-05	.07735	.00531	39.14084	.00 0	.0643	
PU242	AM243	8.44913-06	2.17205-06	.00302	.00103	65.62302	.00 0	.0159	
NP237		1.34932-05	3.82591-06	.0367	.00173	.00000	.0000	.0168	
AM241		7.88814-07	8.27955-17	.0100	.00033	.00000	.00 0	.0030	
AM243		1.64639-06	4.93400-07	.0090	.00023	.00000	.00 0	.0021	
FISSION PRODUCTS				.02206	.06672				

POISON = -0.000 FPA = -5.212-03 FEB = 2.728-05 FFB = 5.091-04 X 135 = 5.622-09 SM149 = 6.254-03
 SM151 = 1.815-07

TIME/HEAT PAV/PHI PWR/WRAT TFEUT/HA DMXV/PKXV SIGSUM/P ABSF/ABSR XNLT/AM T ZTA/J EPS/HA CR/FISRU CAR/FRAN
 9.20154+07 2.36715-11 3.00000+01 350.035 1.00000 .44377 .52242 .91009 1.68084 1.05122 64.0638 3.0045
 3.19491+04 2.63762-11 -2.61070-07 57.007 .49223 .18715 .81716 1.00000 .68867 .74851 .3529 27.6129

END OF CASE NUMBER 1 RECYCLE PASS NUMBER 0

TOTAL COMPUTATION TIME 3.443 SECONDS OR .05739 MINUTES.

HEAT LINK = 0

APPENDIX F
ONE-DIMENSIONAL SAMPLE PROBLEM

Page F-1, below, is a listing of the input cards for a three region, one dimensional case. Pages F-2 through F-13 are the output for that case.

															Link Number for Program CALI	
14	4	3	1	3	1	1	3102	8.8			.001		DEM	1	↑ Basic Deck for ALTHAEA CASE GENERATOR Program ↓	
1	1	4	2	2		4	4	-1.01	10000.		.0001	3.00E+08	DEM	2		
0		1	3	2				2.000E+06	.02		.05	86400.	DEM	3		
								.65	.7		.7		DEM	4		
													DEM	5		
													DEM	6		
													DEM	7		
													DEM	8		
													DEM	9		
													DEM	101		
1	1D							90.					DEM	102		
U233	M=	7	Y=										DEM	103		
U235	M=	10	Y=	4.76000E-04									DEM	104		
PU239	M=	8	Y=										DEM	105		
PU241	M=	11	Y=										DEM	106		
TH232	M=	9	Y=										DEM	107		
U238	M=	3	Y=	2.13920E-02	N=2N=	12							DEM	108		
U234	M=	2	Y=	7.00000E-06									DEM	109		
PU240	M=	4	Y=										DEM	110		
PA233	M=	7	Y=										DEM	111		
U236	M=	12	Y=	2.00000E-06									DEM	112		
PU242	M=	14	Y=										DEM	113		
NP237	M=	12	Y=										DEM	114		
AM241	M=	13	Y=										DEM	115		
AM243	M=	14	Y=										DEM	116		
POISON	M=	15	Y=										DEM	117		
FPA	M=	16	Y=										DEM	118		
FPA	M=	17	Y=										DEM	119		
FPC	M=	18	Y=										DEM	120		
XE135	M=	19	Y=										DEM	121		
SM149	M=	20	Y=										DEM	122		
SM151	M=	21	Y=										DEM	123		
.62414		.31207		735.		0.18		.972		301.		6DEM	201			
.052		1.7613		2.000E-11		8.6385		-1.7443				1DEM	202			
.48		.87		.57		.2		.3117		15.2		9DEM	203			
.001		.0006						3.0000E+08		1.0		-2DEM	204			
1																
PWR SIMULATION (DIABLO CANYON)																
102	3	.000598		106	3	.02127		107	3	6.	E-06110	3	2.5	E-06		
202	7			203	6	3.416		204	6	1.3						
0																
102	3			106	3			107	3		117	3				
203	1	.58		203	2	1.09		203	3	.96	203	5	1.			
4	9			-1202	1	.015		203	6	3.38	204	6	1.7			
2	5			1												
0																

Region 1
Change
Cards for
Region 2
Change
Cards
for
Region
3

PWR SIMULATION (DIABLO CANYON)

INPUT DATA

I= U233 M= 7 Y= -0.00000	I= U235 M= 10 Y= 4.70000-04	I= PU239 M= 8 Y= -0.00000	I= PU241 M= 11 Y= -0.00000														
I= TH232 M= 9 Y= -0.00000	I= U238 M= 3 Y= 2.13920-02	I= U234 M= 2 Y= 7.00000-06	I= PU240 M= 4 Y= -0.00000														
I= PA233 M= 7 Y= -0.00000	I= U236 M= 12 Y= 2.00000-06	I= PU242 M= 14 Y= -0.00000	I= NP237 M= 12 Y= -0.00000														
I= AM241 M= 13 Y= -0.00000	I= AM243 M= 14 Y= -0.00000	I= PFI30N M= 15 Y= -0.00000	I= FPA M= 16 Y= -0.00000														
I= FPB M= 17 Y= -0.00000	I= FPC M= 16 Y= -0.00000	I= XE135 M= 14 Y= -0.00000	I= SM149 M= 20 Y= -0.00000														
I= SM151 M= 21 Y= -0.00000	I=																
N11 14	N12 4	N13 3	NTRL 1	KM 3	N50 1	N16 1	NREG 3	IG 102	POWER 8.8000+00	RMULT -0.0000	CLOSE 1.0000-03	RAD -0.0000					
NS3 1	NS4 1	NS5 4	NS6 2	N6 2	N7 -0	N8 4	N15 4	N9 -0	UMULT -1.0100+00	HLO 1.0000+04	EAE 1.0000-04	TMAX 3.0000+08					
NS1 -0	NS2 -0	NS3 -0	NS4 -0	-0	-0	-0	-0	-0	BMULT1 -0.0000	BMULT2 -0.0000	BMULT3 -0.0000	BMULT4 -0.0000					
N1 0	N2 -0	N3 1	N4 3	N5 2	N10 -0	N55 -0	NCC -0	KEY -0	ISLIM 2.0000+06	ERL 2.0000-02	ERU 5.0000-02	H 8.6400+04					
N17 -0	N18 -0	N19 -0	N20 -0	N21 -0	N22 -0	N23 -0	N24 -0	N25 -0	CMULT -0.0000	CMATCH -0.0000	PERFIS -0.0000	TOT -0.0000					
N27 -0	N28 -0	-0	-0	N31 -0	N32 -0	N33 -0	-0	-0	RAYK1 6.5000-01	RAYK2 7.0000-01	RAYK3 7.0000-01	RMATCH 1.0000-02					
NPASS -0	N34 -0	N35 -0	N36 -0	N37 -0	N38 -0	N39 -0	N40 -0	N41 -0	PERLOS -0.0000	DDAYS -0.0000	DEXP -0.0000	PERLS -0.0000					
NSTEP 1	IDENT 10	NFS -0	IC1 -0	IC2 -0	IC3 -0	IC4 -0	IC5 -0	IC6 -0	IC7 -0	IC8 -0	IC9 -0	IC10 -0	IC11 -0	IC12 -0	PERIOD 90.0	M/F -0.0000	HEIGHT -0.0000
ID1 -0	ID1 -0	ID3 -0	ID4 -0	IF1 -0	IF2 -0	IF3 -0	IF4 -0	IF5 -0	IF6 -0	IF7 -0	IF8 -0	IF9 -0	IF10 -0	IN57 -0	TIMEDR -0	DEL -0	

FCNST= 6.2414-01 FEXP= 3.1207-01 TEMF= 7.3500+02 SPEC= 1.8000-01 TALE= 9.7200-01 TEM= 3.0100+02 IE= 6
 SNF= 5.2000-02 SCA= 1.0613+00 PAI= 2.0000-11 DENSE= 8.6325+00 E= -1.0443+00 XK= 1.6021+11 JMP= 1
 SS1= 4.8000-01 SS2= 8.0000-01 SSX1= 5.7000-01 SA0= 2.0000-01 VR= 3.1170-01 WIDTH= 1.5200+01 NP= 9
 CONVE= 1.0000-03 BUCK= 6.0000-05 PF= 6.6663-01 PMX= -0.0000 AZER= 3.0000+08 ACNTRL= 1.3000+00 KOUT= -2
 SAI= 1.3202+00 SA2= 3.5589-01 SF1= 7.2359+01 SF2= 4.8427+01 SFNU1= 9.1046-01 SFNU2= 5.7970-01

BURN-UP OF REGION 1 IDENTIFIED AS 10 COMPLETED.

INPUT DATA

I= U233 M= 7 Y= -0.00000	I= U235 M= 10 Y= 5.98000-04	I= PU239 M= 8 Y= -0.00000	I= PU241 M= 11 Y= -0.00000														
I= TH232 M= 9 Y= -0.00000	I= U238 M= 3 Y= 2.12700-02	I= U234 M= 2 Y= 5.00000-06	I= PU240 M= 4 Y= -0.00000														
I= PA233 M= 7 Y= -0.00000	I= U236 M= 12 Y= 2.50000-06	I= PU242 M= 14 Y= -0.00000	I= NP237 M= 12 Y= -0.00000														
I= AM241 M= 13 Y= -0.00000	I= AM243 M= 14 Y= -0.00000	I= POISON M= 15 Y= -0.00000	I= FPA M= 16 Y= -0.00000														
I= FPB M= 17 Y= -0.00000	I= FPC M= 18 Y= -0.00000	I= XE135 M= 19 Y= -0.00000	I= SM149 M= 20 Y= -0.00000														
I= SM151 M= 21 Y= -0.00000	I=																
N11 14	N12 4	N13 3	NTRL 1	KM 3	N50 1	N16 1	NREG 3	IG 102	POWER 8.8000+00	RMULT -0.0000	CLOSE 1.0000-03	RAD -0.0000					
NS3 1	NS4 1	NS5 4	NS6 2	N6 2	N7 -0	N8 4	N15 4	N9 -0	BMULT -1.0100+00	HLO 1.0000+04	EAE 1.0000-04	TMAX 3.0000+08					
NS1 -0	NS2 -0	NS3 -0	NS4 -0	-0	-0	-0	-0	-0	BMULT1 -0.0000	BMULT2 -0.0000	BMULT3 -0.0000	BMULT4 -0.0000					
N1 0	N2 -0	N3 1	N4 3	N5 2	N10 -0	N55 -0	NCC -0	KEY -0	ISLIM 2.0000+06	ERL 2.0000-02	ERU 5.0000-02	H 8.6400+04					
N17 -0	N18 -0	N19 -0	N20 -0	N21 -0	N22 -0	N23 -0	N24 -0	N25 -0	CMULT -0.0000	CMATCH -0.0000	PERFIS -0.0000	TCT -0.0000					
N27 -0	N28 -0	-0	-0	N31 -0	N32 -0	N33 -0	-0	-0	RAYK1 6.5000-01	RAYK2 7.0000-01	RAYK3 7.0000-01	RMATCH 1.0000-02					
NPASS -0	N34 -0	N35 -0	N36 -0	N37 -0	N38 -0	N39 -0	N40 -0	N41 -0	PERLOS -0.0000	DDAYS -0.0000	DEXP -0.0000	PERLS -0.0000					
NSTEP 1	IDENT 20	NFS -0	IC1 -0	IC2 -0	IC3 -0	IC4 -0	IC5 -0	IC6 -0	IC7 -0	IC8 -0	IC9 -0	IC10 -0	IC11 -0	IC12 -0	PERIOD 90.0	M/F -0.0000	HEIGHT -0.0000
ID1 -0	ID1 -0	ID3 -0	ID4 -0	IF1 -0	IF2 -0	IF3 -0	IF4 -0	IF5 -0	IF6 -0	IF7 -0	IF8 -0	IF9 -0	IF10 -0	IN57 -0	TIMEDR -0	DEL -0	

FCNST= 6.2414-01 FEXP= 3.1207-01 TEMF= 7.3500+02 SPEC= 1.8000-01 TALE= 9.7200-01 TEM= 3.1000+02 IE= 6
 SNF= 5.2000-02 SCA= 1.0613+00 PAI= 2.0000-11 DENSE= 8.6325+00 E= -1.0443+00 XK= 1.6021+11 JMP= -1
 SS1= 4.8000-01 SS2= 8.0000-01 SSX1= 5.7000-01 SA0= 2.0000-01 VR= 3.1170-01 WIDTH= 3.4100+00 NP= 9
 CONVE= 1.0000-03 BUCK= 6.0000-05 PF= 3.3332-01 PMX= -0.0000 AZER= 3.0000+08 ACNTRL= 1.3000+00 KOUT= -2
 SAI= 1.4389+00 SA2= 4.2628-01 SF1= 4.9654+01 SF2= 5.9991+01 SFNU1= 1.1291+00 SFNU2= 7.1812-01

BURN-UP OF REGION 2 IDENTIFIED AS 20 COMPLETED.

INPUT DATA

12 U233 M= 7 Y= -0.00000	12 U234 M= 8 Y= -0.00000	12 U235 M= 9 Y= -0.00000	12 U236 M= 10 Y= -0.00000	12 U237 M= 11 Y= -0.00000	12 U238 M= 12 Y= -0.00000	12 U239 M= 13 Y= -0.00000	12 U240 M= 14 Y= -0.00000	12 U241 M= 15 Y= -0.00000	12 U242 M= 16 Y= -0.00000	12 U243 M= 17 Y= -0.00000	12 U244 M= 18 Y= -0.00000	12 U245 M= 19 Y= -0.00000	12 U246 M= 20 Y= -0.00000	12 U247 M= 21 Y= -0.00000	12 U248 M= 22 Y= -0.00000	12 U249 M= 23 Y= -0.00000	12 U250 M= 24 Y= -0.00000	12 U251 M= 25 Y= -0.00000	12 U252 M= 26 Y= -0.00000	12 U253 M= 27 Y= -0.00000	12 U254 M= 28 Y= -0.00000	12 U255 M= 29 Y= -0.00000	12 U256 M= 30 Y= -0.00000	12 U257 M= 31 Y= -0.00000	12 U258 M= 32 Y= -0.00000	12 U259 M= 33 Y= -0.00000	12 U260 M= 34 Y= -0.00000	12 U261 M= 35 Y= -0.00000	12 U262 M= 36 Y= -0.00000	12 U263 M= 37 Y= -0.00000	12 U264 M= 38 Y= -0.00000	12 U265 M= 39 Y= -0.00000	12 U266 M= 40 Y= -0.00000	12 U267 M= 41 Y= -0.00000	12 U268 M= 42 Y= -0.00000	12 U269 M= 43 Y= -0.00000	12 U270 M= 44 Y= -0.00000	12 U271 M= 45 Y= -0.00000	12 U272 M= 46 Y= -0.00000	12 U273 M= 47 Y= -0.00000	12 U274 M= 48 Y= -0.00000	12 U275 M= 49 Y= -0.00000	12 U276 M= 50 Y= -0.00000	12 U277 M= 51 Y= -0.00000	12 U278 M= 52 Y= -0.00000	12 U279 M= 53 Y= -0.00000	12 U280 M= 54 Y= -0.00000	12 U281 M= 55 Y= -0.00000	12 U282 M= 56 Y= -0.00000	12 U283 M= 57 Y= -0.00000	12 U284 M= 58 Y= -0.00000	12 U285 M= 59 Y= -0.00000	12 U286 M= 60 Y= -0.00000	12 U287 M= 61 Y= -0.00000	12 U288 M= 62 Y= -0.00000	12 U289 M= 63 Y= -0.00000	12 U290 M= 64 Y= -0.00000	12 U291 M= 65 Y= -0.00000	12 U292 M= 66 Y= -0.00000	12 U293 M= 67 Y= -0.00000	12 U294 M= 68 Y= -0.00000	12 U295 M= 69 Y= -0.00000	12 U296 M= 70 Y= -0.00000	12 U297 M= 71 Y= -0.00000	12 U298 M= 72 Y= -0.00000	12 U299 M= 73 Y= -0.00000	12 U300 M= 74 Y= -0.00000	12 U301 M= 75 Y= -0.00000	12 U302 M= 76 Y= -0.00000	12 U303 M= 77 Y= -0.00000	12 U304 M= 78 Y= -0.00000	12 U305 M= 79 Y= -0.00000	12 U306 M= 80 Y= -0.00000	12 U307 M= 81 Y= -0.00000	12 U308 M= 82 Y= -0.00000	12 U309 M= 83 Y= -0.00000	12 U310 M= 84 Y= -0.00000	12 U311 M= 85 Y= -0.00000	12 U312 M= 86 Y= -0.00000	12 U313 M= 87 Y= -0.00000	12 U314 M= 88 Y= -0.00000	12 U315 M= 89 Y= -0.00000	12 U316 M= 90 Y= -0.00000	12 U317 M= 91 Y= -0.00000	12 U318 M= 92 Y= -0.00000	12 U319 M= 93 Y= -0.00000	12 U320 M= 94 Y= -0.00000	12 U321 M= 95 Y= -0.00000	12 U322 M= 96 Y= -0.00000	12 U323 M= 97 Y= -0.00000	12 U324 M= 98 Y= -0.00000	12 U325 M= 99 Y= -0.00000	12 U326 M= 100 Y= -0.00000	12 U327 M= 101 Y= -0.00000	12 U328 M= 102 Y= -0.00000	12 U329 M= 103 Y= -0.00000	12 U330 M= 104 Y= -0.00000	12 U331 M= 105 Y= -0.00000	12 U332 M= 106 Y= -0.00000	12 U333 M= 107 Y= -0.00000	12 U334 M= 108 Y= -0.00000	12 U335 M= 109 Y= -0.00000	12 U336 M= 110 Y= -0.00000	12 U337 M= 111 Y= -0.00000	12 U338 M= 112 Y= -0.00000	12 U339 M= 113 Y= -0.00000	12 U340 M= 114 Y= -0.00000	12 U341 M= 115 Y= -0.00000	12 U342 M= 116 Y= -0.00000	12 U343 M= 117 Y= -0.00000	12 U344 M= 118 Y= -0.00000	12 U345 M= 119 Y= -0.00000	12 U346 M= 120 Y= -0.00000	12 U347 M= 121 Y= -0.00000	12 U348 M= 122 Y= -0.00000	12 U349 M= 123 Y= -0.00000	12 U350 M= 124 Y= -0.00000	12 U351 M= 125 Y= -0.00000	12 U352 M= 126 Y= -0.00000	12 U353 M= 127 Y= -0.00000	12 U354 M= 128 Y= -0.00000	12 U355 M= 129 Y= -0.00000	12 U356 M= 130 Y= -0.00000	12 U357 M= 131 Y= -0.00000	12 U358 M= 132 Y= -0.00000	12 U359 M= 133 Y= -0.00000	12 U360 M= 134 Y= -0.00000	12 U361 M= 135 Y= -0.00000	12 U362 M= 136 Y= -0.00000	12 U363 M= 137 Y= -0.00000	12 U364 M= 138 Y= -0.00000	12 U365 M= 139 Y= -0.00000	12 U366 M= 140 Y= -0.00000	12 U367 M= 141 Y= -0.00000	12 U368 M= 142 Y= -0.00000	12 U369 M= 143 Y= -0.00000	12 U370 M= 144 Y= -0.00000	12 U371 M= 145 Y= -0.00000	12 U372 M= 146 Y= -0.00000	12 U373 M= 147 Y= -0.00000	12 U374 M= 148 Y= -0.00000	12 U375 M= 149 Y= -0.00000	12 U376 M= 150 Y= -0.00000	12 U377 M= 151 Y= -0.00000	12 U378 M= 152 Y= -0.00000	12 U379 M= 153 Y= -0.00000	12 U380 M= 154 Y= -0.00000	12 U381 M= 155 Y= -0.00000	12 U382 M= 156 Y= -0.00000	12 U383 M= 157 Y= -0.00000	12 U384 M= 158 Y= -0.00000	12 U385 M= 159 Y= -0.00000	12 U386 M= 160 Y= -0.00000	12 U387 M= 161 Y= -0.000
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BURN-UP OF REGION 3 IDENTIFIED AS 30 COMPLETED.

DIFFUSION OUTPUT AT 0.0000 SECONDS. POWER BASED ON OLD FLUX IS 4.8682+00

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FLUX 1, REGION 1, POINT 1, FAILED TO CONVERGE IN 100 ITERATIONS, LAST TEN VALUES OF K ARE AS FOLLOWS.
1.00016496 1.00016367 1.00016245 1.00016127 1.00016001 1.00015879 1.00015761 1.00015645 1.00015530 1.00015412
FINAL REACTIVITY FOR EACH CONTROL ITERATION WAS
1.00916271 1.00015412

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REACTIVITY WITH CONTROL RODS WITHDRAWN IS 1.17189

REGIO.	RADIUS	POWEP	FLUX	PERCENT CHANGE	R	PERCENT CHANGE	SNFD	CFACT	DFACT
1	136.800	288.130	3.54075-11	43.515	.1538	-17.002	.0707	1.0000	.0000
2	167.544	384.157	3.78841-11	47.207	.1732	-3.931	.0920	1.0000	.0000
3	197.964	.000	5.27266-12	100.000	.0445	-304.673	.0376	1.0000	.0000

REGIO.	D1	D2	XISIGS	DU	SIG1	SIG2	SIG1	SIG2	SIGFNU1	SIGFNU2
1	.89601	.407647	.029578	16.066349	.015197	.965815	.046477	.088530	.010480	.167205
2	.689135	.405977	.029012	16.663664	.016543	.75727	.047544	.103551	.012982	.132626
3	.574262	.304075	.058977	16.666337	.000000	.000000	.061289	.131116	.000000	.000000

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 1

FAST	3.841-04	3.841-04	4.035-04	4.390-04	4.908-04	5.595-04	6.471-04	7.542-04	8.833-04	1.048-03
TRFMA	1.285-04	1.285-04	1.350-04	1.469-04	1.642-04	1.873-04	2.164-04	2.522-04	2.951-04	3.315-04

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 2

FAST	1.048-03	1.093-03	1.088-03	1.051-03	9.945-04	6.927-04	7.780-04	6.432-04	4.866-04	2.872-04
TERMA	3.315-04	3.064-04	2.984-04	2.870-04	2.687-04	2.457-04	2.129-04	1.786-04	1.493-04	1.066-04

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 3

FAST	2.852-04	9.705-05	3.368-05	1.162-05	4.917-06	1.364-06	4.775-07	1.642-07	5.493-08	1.393-08
TRKKA	1.600-04	1.404-04	8.180-05	3.924-05	1.742-05	7.377-06	3.522-06	1.198-06	4.343-07	6.623-08

NORMALIZED DIFFUSION FLUXES BY MESH POINT

SA1= 1.3479+00 SA2= 3.5740-01 SF1= 7.7283+01 SF2= 4.9419+01 SFNU1= 9.7729-01 SFNU2= 5.7960-01

BURN-UP OF REGION 1 IDENTIFIED AS 10 COMPLETED.

SA1= 1.4473+00 SA2= 4.2679-01 SF1= 9.1121+01 SF2= 5.9956+01 SFNU1= 1.1490+00 SFNU2= 7.1773-01

BURN-UP OF REGION 2 IDENTIFIED AS 20 COMPLETED.

DIFFUSION OUTPUT AT 0.0000 SECONDS. POWER BASED ON OLD FLUX IS 8.8907+00

FLUX IN REGION 1, POINT 1, FAILED TO CONVERGE IN 100 ITERATIONS. LAST TEN VALUES OF K ARE AS FOLLOWS.

1.00047804 1.00047807 1.00047818 1.00047819 1.00047825 1.00047830 1.00047833 1.00047837 1.00047836 1.00047840

FINAL REACTIVITY FOR EACH CONTROL ITERATION WAS

1.00732346 1.00047840

REACTIVITY WITH CONTROL RODS WITHDRAWN IS 1.17029

REGION	RADIUS	POWER	FLUX	PERCENT CHANGE	R	PERCENT CHANGE	SNFD	CFACT	DFACT
1	136.800	318.468	3.85292-11	8.102	.1562	1.536	.0717	1.0000	.0000
2	167.544	323.479	3.17682-11	-19.251	.1749	.997	.0940	1.0000	.0000
3	197.964	.000	4.22953-12	-24.663	.0452	1.497	.0385	1.0000	.0000

REGION	C1	C2	XISIGS	DU	SIGAI	SIGAI2	SIGRI	SIGRI2	SIGFNU1	SIGFNU2
1	.689466	.407518	.029372	16.084441	.015641	.066709	.046743	.089828	.011341	.108183
2	.689088	.405868	.026943	16.068-12	.016677	.073394	.047640	.106452	.013239	.132894
3	.574254	.334046	.052964	16.066337	.000000	.000000	.061315	.031652	.000000	.000000

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 1

FAST	5.754-04	5.754-04	5.977-04	6.153-04	6.509-04	6.951-04	7.469-04	8.036-04	8.635-04	9.332-04
THERMAL	1.883-04	1.863-04	1.930-04	2.013-04	2.129-04	2.275-04	2.443-04	2.628-04	2.821-04	2.899-04

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 2

FAST	9.332-04	9.532-04	9.343-04	8.605-04	8.256-04	7.421-04	6.419-04	5.275-04	3.972-04	2.304-04
THERMAL	2.898-04	2.639-04	2.532-04	2.404-04	2.227-04	2.002-04	1.736-04	1.447-04	1.204-04	1.340-04

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 3

FAST	2.304-04	7.943-05	2.739-05	9.449-05	3.260-06	1.121-06	3.880-07	1.334-07	4.462-08	1.132-08
THERMAL	1.340-04	1.175-04	6.541-05	3.125-05	1.561-05	5.821-06	2.377-06	9.385-07	3.391-07	5.170-08

NORMALIZED DIFFUSION FLUXES BY MESH POINT

ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF
U233	6.224+02	.9233	1.6400	U235	4.929+02	.8646	-.0905	PU239	1.276+03	1.5134	1.3150
PU241	1.278+03	1.2076	.2867	TH232	2.284+01	.0000	1.4620	U238	9.029+01	.0000	5.7791
U234	4.113+00	.9331	432.7573	PU240	4.904+00	1.2132	564.3883	PA233	0.000	.9308	-.1668
U236	1.094+00	.0000	10.9417	PU242	4.425+00	.8404	1371.2429	NP237	3.760+00	.9452	1260.4822
AM241	1.102+01	1.6450	12.0109	AM243	2.991+00	.0000	19.1427				
ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA
U233	7.019+02	.9286	1.3526	U235	6.006+02	.8688	.0939	PU239	1.992+03	1.7002	1.7664
PU241	1.784+03	1.2531	.3072	TH232	2.813+01	.9310	16.3711	U238	9.149+00	.9414	15.5021
U234	2.261+02	.9324	9.2525	PU240	2.123+03	1.0628	41.5353	PA233	2.549+02	.8639	28.8433
U236	1.084+02	.7960	110.5150	PU242	3.227+02	.8418	74.0341	NP237	3.277+02	1.0047	5.9785
AM241	1.380+03	1.4831	5.6013	AM243	4.794+02	.5504	13.2898	POISON	0.000	.9308	-.1668
FPA	7.412+02	.9308	-.0278	FPC	1.368+03	.9308	22.2182	FPC	4.632+01	.9308	-.0278
XE135	2.395+06	1.1985	-1.3512	SM149	7.202+04	2.1023	-2.2717	SM151	1.445+04	.9314	.2291

ISOTOPE	DESCENDANT	CONCENTRATION	DESTROYED	ABS	ABSAV	ALPHA	HEAT	POWER	CO ₂ V.
U233	U234	-0.00000	0.00000	.00000	.00000	.12773	.0000	.0000	.0000
U235	U236	4.76000-04	0.00000	.28549	.28589	.21801	92.4506	92.4506	.5434
PU239	PU240	-0.00000	0.00000	.00000	.00000	.56124	.0000	.0000	61.7007
PU241	PU242	-0.00000	0.00000	.00000	.00000	.39663	.0000	.0000	.0000
TH232	PA233	-0.00000	0.00000	.00000	.00000	122.14077	.0000	.0000	.0000
U234	PU239	2.13920-02	0.00000	.19571	.19571	9.13200	7.5369	7.5369	.0112
U234	U235	7.00000-06	0.00000	.00158	.00158	53.96359	.0112	.0112	.0000
PU240	PU241	-0.00000	0.00000	.00000	.00000	431.87363	.0000	.0000	.0000
PA233	U234	-0.00000	0.00000	.00000	.00000	.00000	.0000	.0000	.0000
U236	NP237	2.00000-06	0.00000	.00022	.00022	62.93056	.0013	.0013	.0000
PU242	AM243	-0.00000	0.00000	.00000	.00000	71.61735	.0000	.0000	.0000
NP237	-	-0.00000	0.00000	.00000	.00000	.00000	.0000	.0000	.0000
AM241	-	-0.00000	0.00000	.00000	.00000	.00000	.0000	.0000	.0000
AM243	-	-0.00000	0.00000	.00000	.00000	.00000	.0000	.0000	.0000
FISSION PRODUCTS				.01774	.01774	.00000	.0000	.0000	.0000

POISON = -0.000 FPA = -0.000 FPC = -0.000 FPC = -0.000 XE135 = 5.344-9 SM149 = 4.851-08
SM151 = 9.945-08

TIME/HEAT PAV/PHI PWR/WRAT THERM/WM PMXV/PMNV SIGSUM/K A.SF/ABSR X-ALT/AMLT Z-A/J EPS/Q CR/FISBU CAR/FRAB
0.00000 3.68082+01 3.68082+01 376.054 1.00000 .35767 .32654 1.12738 1.71795 1.05094 62.2441 .0000
0.00000 3.85292-11 0.00000 51.513 1.00000 .15624 .54406 1.12738 .82492 .76071 .2859 32.1159
-SA1 = 1.5369+00 SA2 = 3.8162-01 SF1 = 9.7977+01 SF2 = 4.9482+1 SF3 = 1.1364+00 SF4 = 6.0702-01

BURN-UP OF REGION 1 IDENTIFIED AS 10 COMPLETED.

ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF	ISOTOPE	SIGMAF	GF	SF
U233	6.350+02	.9101	1.6763	U235	4.837+02	.8524	1.1014	PU239	1.294+03	1.5148	1.3064
PU241	1.275+03	1.1989	.7883	TU232	2.825+01	.0000	1.6430	U238	1.117+00	.0000	6.3841
U234	5.057+00	.9195	476.5305	PU240	5.974+00	1.1452	637.4712	PA233	0.000	.9173	-.1547
U236	2.095+00	.0000	11.9765	PU242	5.946+00	.8266	150.9050	NP237	4.644+00	.9324	13.91991
AM241	1.211+01	1.6560	12.6519	AM243	3.690+00	.0000	21.1460				

ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA	ISOTOPE	SIGMAA	GA	SA
U233	7.180+02	.9153	1.8903	U235	5.917+02	.8500	.0750	PU239	2.024+03	1.7053	1.7501
PU241	1.761+03	1.2456	.3043	TU232	3.085+01	.9175	1.5846	U238	1.012+01	.9279	15.0036
U234	2.431+02	.9189	9.4874	PU240	2.367+03	1.0509	42.1453	PA233	2.819+02	.8510	29.0867
U236	1.214+02	.7828	111.1608	PL242	3.615+02	.8283	74.7435	NP237	3.400+02	.9948	6.0422
AM241	1.447+03	1.4884	5.6228	AM243	5.266+02	.5383	13.4214	POISON	0.000	.9173	-.1547
FPA	7.321+02	.9173	-.0121	FPA	1.504+03	.9173	23.4138	FPC	4.571+01	.9173	-.0121
XE135	2.292+06	1.1775	-1.3290	SM149	6.924+04	2.0711	-2.2368	SM151	1.441+04	.9178	.2463

ISOTOPE	DESCENDANT	CONCENTRATION	DESTROYED	APS	ABSAV	ALPHA	HEAT	POWER	CONV.
U233	U234	-0.00000	0.00000	.00000	.00000	.13000	.0000	.0000	.0000
U235	U236	5.98000-04	0.00000	.35356	.35356	.22233	92.4697	92.4697	.0074
PU239	PU240	-0.00000	0.00000	.00000	.00000	.56670	.0000	.0000	54.1618
PU241	PU242	-0.00000	0.00000	.00000	.00000	.39701	.0000	.0000	.0000
TU232	PA233	-0.00000	0.00000	.00000	.00000	108.16391	.0000	.0000	.0000
U233	PU239	2.12700-02	0.00000	.21525	.21525	18.26151	7.5191	7.5191	
U234	U235	6.30000-06	0.00000	.00147	.00147	47.7289	.0096	.0096	
PU240	PU241	-0.00000	0.00000	.00000	.00000	395.31516	.0000	.0000	
PA233	U234	-0.00000	0.00000	.00000	.00000	.00000	.0000	.0000	
U236	NP237	2.50000-06	0.00000	.00030	.00030	56.92889	.0016	.0016	
PU242	AM243	-0.00000	0.00000	.00000	.00000	85.45101	.0000	.0000	
NP237	-	-0.00000	0.00000	.00000	.00000	.00000	.0000	.0000	
AM241	-	-0.00000	0.00000	.00000	.00000	.00000	.0000	.0000	
AM243	-	-0.00000	0.00000	.00000	.00000	.00000	.0000	.0000	
FISSION PRODUCTS				.02112	.02112				

POISON = 0.000 FPA = -0.030 FPB = -0.000 FPC = -0.000 XE135 = 6.559-09 SM149 = 6.220-08
SM151 = 1.233-07

TIME/HEAT	PAV/PHI	PWR/WRT	TLEUT/WA	PMXV/PMNV	SIGSUM/H	ASF/ABSR	XVLT/AMLT	ZTA/J	EPS/Q	CR/FISBU	CAP/FRAH
0.00000	3.74115+01	3.74115+01	380.194	1.00000	.42819	.43245	1.19435	1.76775	1.03267	54.5692	.0000
0.00000	3.117882-11	0.00000	63.570	1.00000	.17494	.64487	1.19435	.84303	.76793	.3536	29.9186
SA1 =	1.6190+03	SA2 =	4.4652-01	SF1 =	1.0011+02	SF2 =	6.0247+1	SF1U1 =	1.2847+00	SFNU2 =	7.3479-01

BURN-UP OF REGION 2 IDENTIFIED AS 20 COMPLETED.

DIFFUSION OUTPUT AT 7.7760+06 SECONDS. POWER BASED ON OLD FLUX IS 9.1712+00

FLUX IN REGION 1, POINT 1, FAILED TO CONVERGE IN 100 ITERATIONS. LAST TEN VALUES OF K ARE AS FOLLOWS.
1.00030510 1.00030544 1.00030570 1.00030603 1.00030628 1.00030653 1.00030680 1.00030705 1.00030731 1.00030755

FINAL REACTIVITY FOR EACH CONTROL ITERATION WAS
1.00623904 1.00630755

REACTIVITY WITH CONTROL RODS WITHDRAWN IS 1.13961

REGION	RADIUS	POWER	FLUX	PERCENT CHANGE	R	PERCENT CHANGE	SNFD	CFACT	DFACT
1	136.800	322.062	3.69646-11	-4.233	.1647	5.140	.0637	1.0000	.0000
2	167.544	316.291	3.01283-11	-5.443	.1807	3.178	.0835	1.0000	.0000
3	197.964	.000	4.42159-12	4.344	.0432	-4.620	.0344	1.0000	.0000

REGION	D1	D2	XISIGS	DU	SIG1	SIG2	SIGR1	SIGR2	SIGFNU1	SIGFNU2
1	.688843	.407157	.028580	16.096858	.017929	.071696	.048138	.093446	.013256	.113983
2	.688535	.405672	.028236	16.080777	.018754	.063173	.048877	.108432	.014881	.136871
3	.574289	.304180	.059526	16.066337	.000000	.000000	.061199	.029224	.000000	.000000

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 1

FAST	5.826-04	5.826-04	5.760-04	6.196-04	6.523-04	6.926-04	7.384-04	7.873-04	8.369-04	8.915-04
THERMAL	1.783-04	1.783-04	1.724-04	1.496-04	1.996-04	2.119-04	2.259-04	2.408-04	2.558-04	2.607-04

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 2

FAST	8.315-04	9.056-04	8.459-04	8.438-04	7.825-04	7.042-04	6.110-04	5.044-04	3.824-04	2.247-04
THERMAL	2.607-04	2.395-04	2.299-04	2.182-04	2.023-04	1.821-04	1.583-04	1.328-04	1.120-04	1.297-04

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 3

FAST	2.246-04	7.753-05	2.776-05	9.240-04	3.191-06	1.102-06	3.805-07	1.310-07	4.383-08	1.112-08
THERMAL	1.299-04	1.192-04	6.312-05	3.330-05	1.504-05	8.461-06	2.700-06	1.080-06	3.993-07	6.091-08

NORMALIZED DIFFUSION FLUXES BY MESH POINT

U233 = 8.044-15 U235 = 3.979-04 PU239 = 4.051-05 PU241 = 6.967-07 TH232 = -0.000 U238 = 2.133-02
 U234 = 6.540-06 PU240 = 3.545-06 PA233 = 1.059-14 U236 = 1.580-05 PU242 = 2.699-08 NP237 = 5.485-07
 AM241 = 2.290-09 AM243 = 5.377-10 POISON = -0.000 FPA = -7.581-07 FPB = 8.252-06 FPC = 5.574-05
 XE135 = 5.584-09 SM149 = 5.494-08 SM151 = 1.269-07

TIME/HEAT PAV/PHI PWR/WRAT TNEUT/WA PMXV/PMNV SIGSUM/R ABSF/ABSR XMLT/AMLT ZTA/J EPS/G CR/FISBU CAB/FPA
 7.77000+06 3.80079+01 3.71030+01 364.543 1.01772 .38216 .35519 1.10057 1.70897 1.05107 60.6403 12.3346
 3.42072+03 3.69646-11 0.00000 54.123 .96843 .16471 .61150 1.12453 .80791 .76109 .3133 30.9631
 SA1 = 1.6422+00 SA2 = 3.8971-01 SF1 = 8.9496+01 SF2 = 4.3732+01 SFU1 = 1.1681+00 SFNU2 = 6.0761-01

BURN-UP OF REGION 1 IDENTIFIED AS 10 COMPLETED.

U233 = 8.314-15 U235 = 5.170-04 PU239 = 3.781-05 PU241 = 5.132-07 TH232 = -0.000 U238 = 2.122-02
 U234 = 5.647-06 PU240 = 2.783-06 PA233 = 1.073-14 U236 = 1.702-05 PU242 = 1.613-08 NP237 = 5.532-07
 AM241 = 1.684-09 AM243 = 2.945-10 POISON = -0.000 FPA = -7.457-07 FPB = 8.320-06 FPC = 5.582-05
 XE135 = 6.717-09 SM149 = 6.787-08 SM151 = 1.482-07

TIME/HEAT PAV/PHI PWR/WRAT TNEUT/WA PMXV/PMNV SIGSUM/R ABSF/ABSR XMLT/AMLT ZTA/J EPS/G CR/FISBU CAB/FPA
 7.77000+06 3.81372+01 3.65053+01 373.934 1.00976 .44675 .41555 1.16256 1.75488 1.05267 53.7630 14.0406
 3.43234+03 3.01283-11 0.00000 65.334 .95721 .18068 .69507 1.18514 .82380 .76924 .3759 29.0836
 SA1 = 1.7203+00 SA2 = 4.5067-01 SF1 = 1.0155+02 SF2 = 5.9001+01 SFU1 = 1.3153+00 SFNU2 = 7.2895-01

BURN-UP OF REGION 2 IDENTIFIED AS 20 COMPLETED.

DIFFUSION OUTPUT AT 1.5552E7 SECONDS. POWER BASED ON OLD FLUX IS 8.7296E00

FLUX IN REGION 1: POINT 1. FAILED TO CONVERGE IN 100 ITERATIONS. LAST TEN VALUES OF K ARE AS FOLLOWS:
 .99993665 .99998795 .99999719 .99999843 .99998577 .99998500 .99999430 .99998300 .99998298 .99998230

FINAL REACTIVITY FOR EACH CONTROL ITERATION WAS
 1.00400074 .99998230

REACTIVITY WITH CONTROL RODS WITHDRAWN IS 1.10354

REGION	RADIUS	POWER	FLUX	PERCENT CHANGE	R	PERCENT CHANGE	SNFD	CFACT	DFACT
1	136.800	296.717	3.42855-11	-7.814	.1644	-.187	.0476	1.0000	.0000
2	167.544	366.982	3.53942-11	14.878	.1779	-1.540	.0624	1.0000	.0000
3	197.964	.000	6.36358-12	30.517	.0376	-14.821	.0257	1.0000	.0000

REGION	D1	D2	XISIG5	D0	SIG1	SIG2	SIGR1	SIGR2	SIGFNU1	SIGFNU2
1	.088549	.407298	.028213	16.099527	.019179	.073270	.048802	.092037	.013642	.114246
2	.688271	.405961	.027902	16.085066	.019965	.084127	.049470	.105505	.015264	.136074
3	.574363	.304404	.053157	16.060337	.000000	.000000	.060956	.024121	.000000	.000000

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 1

FAST	4.250-04	4.250-04	4.416-04	4.720-04	5.161-04	5.743-04	6.467-04	7.336-04	8.356-04	9.604-04
THERMA	1.304-04	1.304-04	1.355-04	1.449-04	1.584-04	1.762-04	1.984-04	2.251-04	2.561-04	2.825-04

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 2

FAST	9.614-04	9.913-04	9.339-04	9.496-04	8.917-04	8.125-04	7.143-04	5.987-04	4.629-04	2.773-04
THERMA	2.825-04	2.655-04	2.590-04	2.492-04	2.339-04	2.132-04	1.879-04	1.603-04	1.393-04	1.694-04

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 3

FAST	2.773-04	9.588-05	3.316-05	1.147-05	3.969-06	1.374-06	4.752-07	1.639-07	5.495-08	1.394-08
THERMA	1.694-04	1.656-04	9.958-05	5.116-05	2.431-05	1.102-05	4.632-06	2.045-06	7.813-07	1.193-07

NORMALIZED DIFFUSION FLUXES BY MESH POINT

RADIUS	0	10	20	30	40	50	60	70	80	90	100
0.000	*	*	*	*	*	*	*	*	*	*	*
15.200	*	T	*	*	F	*	*	*	*	*	*
30.400	*	*	T	*	F	F	*	*	*	*	*
45.600	*	*	*	T	*	F	*	*	*	*	*
60.800	*	*	*	*	T	F	F	*	*	*	*
76.000	*	*	*	T	*	*	F	*	*	*	*
91.200	*	*	T	T	*	*	F	F	*	*	*
106.400	*	*	*	T	*	*	*	F	*	*	*
121.600	*	*	*	*	T	*	*	*	F	*	*
136.800	*	*	*	*	*	T	*	*	*	F	*
140.216	*	*	*	*	*	*	T	*	*	*	F
143.632	*	*	*	*	*	*	*	T	*	*	F
147.048	*	*	*	*	*	*	*	*	T	*	F
150.464	*	*	*	*	*	*	*	*	*	T	*
153.880	*	*	*	*	*	*	*	*	*	*	T
157.296	*	*	*	*	*	*	*	*	*	*	*
160.712	*	*	*	*	*	*	*	*	*	*	*
164.128	*	*	*	*	*	*	*	*	*	*	*
167.544	*	*	*	*	*	*	*	*	*	*	*
170.924	*	*	*	*	*	*	*	*	*	*	*
174.304	*	F	T	*	*	*	*	*	*	*	*
177.684	*	F	T	*	*	*	*	*	*	*	*
181.064	*	F	T	*	*	*	*	*	*	*	*
184.444	*	F	T	*	*	*	*	*	*	*	*
187.824	*	T	*	*	*	*	*	*	*	*	*
191.204	*	T	*	*	*	*	*	*	*	*	*
194.584	*	T	*	*	*	*	*	*	*	*	*
197.964	*	T	*	*	*	*	*	*	*	*	*
RADIUS	0	10	20	30	40	50	60	70	80	90	100

U233 = 5.222-14 U235 = 3.349-04 PU239 = 0.524-05 PU241 = 3.137-06 TH232 = -0.000 U238 = 2.128-02
 U234 = 6.114-06 PU240 = 9.801-05 FA233 = 3.445-14 U236 = 2.000-05 PU242 = 2.574-07 NP237 = 1.370-05
 AM241 = 2.190-08 AM243 = 1.100-04 POISCN = -0.000 FPA = -1.508-06 FPB = 1.404-05 FPC = 1.104-01
 XE135 = 5.552-09 SM149 = 5.819-08 SX151 = 1.412-07

TIME/HEAT PAV/PHI PWR/WRAF TENDT/KA PMXV/PMNV SIGSUM/R ABSF/ABSR XMLT/AMLT ZTA/J EPS/G CR/FISBU CAP/FRA7
 1.55520+07 3.75398+01 3.43638+01 359.485 1.83041 .38974 .35875 1.07213 1.09711 1.05108 61.6291 11.2843
 6.75717+03 3.42855-11 0.00000 54.044 .91540 .16440 .63054 1.10488 .79259 .75992 .3194 30.9851
 SA1= 1.7103+00 SA2= 3.8483-01 SF1= 8.9881+01 SF2= 4.7343+01 SFNU1= 1.1422+00 SFNU2= 5.9730-01

BURN-UP OF REGION 1 IDENTIFIED AS 10 COMPLETED.

U233 = 5.330-14 U235 = 4.501-04 PU239 = 0.207-05 PU241 = 2.475-06 TH232 = -0.000 U238 = 2.117-02
 U234 = 5.324-06 PU240 = 7.839-05 FA233 = 3.426-14 U236 = 2.072-05 PU242 = 1.579-07 NP237 = 1.354-05
 AM241 = 1.688-08 AM243 = 6.021-09 POISCN = -0.000 FPA = -1.406-06 FPB = 1.418-05 FPC = 1.093-04
 XE135 = 6.618-09 SM149 = 7.059-08 SX151 = 1.612-07

TIME/HEAT PAV/PHI PWR/WRAF TENDT/KA PMXV/PMNV SIGSUM/R ABSF/ABSR XMLT/AMLT ZTA/J EPS/G CR/FISBU CAP/FRA7
 1.55520+07 3.72480+01 4.25553+01 367.788 1.14249 .45054 .41570 1.13533 1.74309 1.05258 54.4161 12.9283
 6.70463+03 3.53942-11 0.00000 64.831 .97219 .17794 .71514 1.16580 .80838 .76967 .3794 29.0790
 SA1= 1.8018+00 SA2= 4.4756-01 SF1= 1.0185+02 SF2= 5.0662+01 SFNU1= 1.3307+00 SFNU2= 7.0863-01

BURN-UP OF REGION 2 IDENTIFIED AS 20 COMPLETED.

DIFFUSION OUTPUT AT 2.3328+07 SECONDS. POWER BASED ON OLD FLUX IS 8.6002+00

FINAL REACTIVITY FOR EACH CONTROL ITERATION WAS
 1.00004433

REACTIVITY WITH CONTROL RODS WITHDRAWN IS -1.06908

REGION	RADIUS	POWER	FLUX	PERCENT CHANGE	R	PERCENT CHANGE	SNFD	CFACT	DFACT
1	136.800	331.195	3.55660-11	3.600	.1614	-1.840	.0319	1.0000	.0000
2	167.544	358.026	3.56136-11	.616	.1731	-2.820	.0418	1.0000	.0000
3	197.964	.000	7.79161-12	18.328	.0313	-18.366	.0173	1.0000	.0000

REGION	D1	D2	XISIGS	DU	SIGAI	SIGAI2	SIGR1	SIGR2	SIGFNU1	SIGFNU2
1	.688371	.407567	.027983	16.104509	.020018	.073489	.049198	.089328	.013837	.112599
2	.688061	.406361	.027626	16.092594	.020982	.063875	.049934	.101471	.015497	.132800
3	.574435	.304742	.059285	16.068337	.000000	.000000	.060719	.019127	.000000	.000000

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 1

	FAST	1.616-04	5.157-04	5.223-04	5.358-04	5.589-04	5.949-04	6.469-04	7.170-04	8.053-04	9.164-04
Thermal	1.616-04	1.616-04	1.637-04	1.630-04	1.752-04	1.865-04	2.028-04	2.248-04	2.523-04	2.760-04	

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 2

	FAST	9.164-04	9.444-04	9.376-04	9.066-04	8.545-04	7.831-04	6.943-04	5.893-04	4.637-04	2.834-04
Thermal	2.700-04	2.603-04	2.542-04	2.450-04	2.308-04	2.116-04	1.882-04	1.630-04	1.459-04	1.850-04	

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 3

	FAST	2.834-04	9.819-05	3.403-05	1.179-05	4.089-06	1.414-06	4.914-07	1.698-07	5.702-08	1.446-08
Thermal	1.859-04	1.940-04	1.235-04	6.731-05	3.398-05	1.640-05	7.663-06	3.448-06	1.383-06	2.117-07	

NORMALIZED DIFFUSION FLUXES BY MESH POINT

U233 = 1.456-13 U235 = -2.853-04 PU239 = 8.021-05 PU241 = 6.261-06 TH232 = -0.000 U238 = 2.122-02
 U234 = 5.742-06 PU240 = 1.627-05 PA233 = 6.364-14 U236 = 3.496-05 PU242 = 7.777-07 NP237 = 2.256-06
 AM241 = 6.782-08 AV243 = 5.060-08 PO150N = -0.000 FPA = -2.148-06 FPB = 1.786-05 FPC = 1.604-04
 XE135 = 5.359-09 SM149 = 5.859-08 SM151 = 1.474-07

TIME/HEAT PAV/PHI PWR/WRAT TLEUT/WA PMXV/PMNV SIGSUM/R ABSF/ABSR XMLT/AMLT ZTA/J EPS/Q CR/FISBU CAB/FRA3
 2.33280+07 3.63667+01 3.49337+01 385.455 1.06365 .38985 .35649 1.04858 1.68386 1.05102 63.1109 10.0041
 9.81900+03 3.55660-11 0.00000 52.963 .92604 .16143 .64482 1.08959 .78136 .75896 .3171 31.0217
 SA1= 1.7584+00 SA2= 3.8642-01 SF1= 8.9309+01 SF2= 4.5566+01 SF1U1= 1.1825+00 SFNU2= 5.8094-01

BURN-UP OF REGION 1 IDENTIFIED AS 10 COMPLETED.

U233 = 1.463-13 U235 = 3.822-04 PU239 = 8.075-05 PU241 = 5.792-06 TH232 = -0.000 U238 = 2.111-02
 U234 = 4.970-06 PU240 = 1.463-05 PA233 = 6.751-14 U236 = 4.024-05 PU242 = 6.212-07 NP237 = 2.466-06
 AM241 = 5.649-08 AV243 = 3.862-08 PO150N = -0.000 FPA = -2.248-06 FPB = 1.908-05 FPC = 1.707-04
 XE135 = 6.608-09 SM149 = 7.042-08 SM151 = 1.683-07

TIME/HEAT PAV/PHI PWR/WRAT TLEUT/WA PMXV/PMNV SIGSUM/R ABSF/ABSR XMLT/AMLT ZTA/J EPS/Q CR/FISBU CAB/FRA3
 2.33280+07 3.88086+01 4.15658+01 362.115 1.09654 .44727 .40876 1.10502 1.72867 1.05240 55.9900 11.3266
 1.04783+04 3.56136-11 0.00000 62.933 .93310 .17306 .72138 1.14717 .79287 .76904 .3732 29.0859
 SA1= 1.8528+00 SA2= 4.3908-01 SF1= 1.0067+02 SF2= 5.3928+01 SF1U1= 1.3248+00 SFNU2= 6.8142-01

BURN-UP OF REGION 2 IDENTIFIED AS 20 COMPLETED.

DIFFUSION OUTPUT AT 3.1104+07 SECONDS. POWER RATIO ON CL₁ FLUX IS 8.5157+00

FLUX IN REGION 1, POINT 1, FAILED TO CONVERGE IN 100 ITERATIONS. LAST TEN VALUES OF K ARE AS FOLLOWS.

.99949716 .99949768 .99949826 .99949881 .99949936 .99949990 .99950047 .99950097 .99950153 .99950203

FINAL REACTIVITY FOR EACH CONTROL ITERATION WAS
.99950203

REACTIVITY WITH CONTROL RODS WITHDRAWN IS 1.03835

REGION	RADIUS	POWER	FLUX	PERCENT CHANGE	R	PERCENT CHANGE	SNFD	CFACT	DFACT
1	136.800	295.598	3.60431-11	1.324	.1576	-2.448	.0180	1.0000	.0001
2	167.544	369.221	3.83289-11	7.084	.1666	-3.904	.0236	1.0000	.0000
3	197.964	.000	1.04672-11	25.562	.0262	-21.019	.0098	1.0000	.0000

REGION	D1	D2	XISIG5	DU	SIGM1	SIGM2	SIGR1	SIGR2	SIGFNU1	SIGFNU2
1	.688252	.407871	.027825	16.109625	.020627	.073025	.049460	.086287	.013871	.109785
2	.687943	.406823	.027466	16.099927	.021642	.062575	.050188	.096813	.015475	.128150
3	.674500	.304989	.059399	16.066337	.000000	.000000	.060507	.014689	.000000	.000000

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 1

FAST	4.860-04	4.860-04	4.952-04	5.127-04	5.399-04	5.783-04	6.318-04	7.016-04	7.914-04	9.107-04
THERMAL	1.568-04	1.568-04	1.598-04	1.654-04	1.742-04	1.866-04	2.039-04	2.264-04	2.552-04	2.833-04

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 2

FAST	9.107-04	9.417-04	9.395-04	9.138-04	8.674-04	8.022-04	7.196-04	6.203-04	4.980-04	3.109-04
THERMAL	2.833-04	2.701-04	2.653-04	2.573-04	2.441-04	2.259-04	2.034-04	1.796-04	1.661-04	2.211-04

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 3

FAST	3.169-04	1.079-04	3.746-05	1.301-05	4.517-06	1.569-06	5.447-07	1.885-07	6.341-08	1.609-08
THERMAL	2.211-04	2.459-04	1.664-04	9.664-05	5.217-05	2.700-05	1.355-05	6.519-06	2.756-06	4.213-07

NORMALIZED DIFFUSION FLUXES BY MESH POINT

RADIUS	0	10	20	30	40	50	60	70	80	90	100
15.200	*		T			F					*
30.400	*		T			F					*
45.600	*		T			F					*
60.800	*		T			F					*
76.000	*		T			F					*
91.200	*		T			F					*
106.400	*		T			F					*
121.600	*		T			F					*
136.800	*		T			F					*
140.216	*		T			F					*
143.632	*		T			F					*
147.048	*		T			F					*
150.464	*		T			F					*
153.880	*		T			F					*
157.296	*		T			F					*
160.712	*		T			F					*
164.128	*		T			F					*
167.544	*		T			F					*
170.960	*		T			F					*
174.376	*	F		T							*
177.684	* F		T								*
181.064	* F	T									*
184.444	* F	T									*
187.824	* F	T									*
191.204	* F	T									*
194.584	* F										*
197.964	* F										*
RADIUS	0	10	20	30	40	50	60	70	80	90	100

U233 = 2.841-13 U235 = 2.414-04 PU239 = 9.017-05 PU241 = 9.549-06 TH232 = -0.000 U238 = 2.117-02
 U234 = 5.382-06 PU240 = 2.300-05 PA233 = 9.546-14 U236 = 4.203-05 PU242 = 1.643-06 NP237 = 3.274-06
 AM241 = 1.354-07 AM243 = 1.431-07 PO150 = -0.000 FPA = -2.727-06 FPU = 2.066-05 FPC = 2.105-04
 KE135 = 5.180-09 SM149 = 5.773-08 SM151 = 1.503-07

TIME/HEAT PAV/PHI PWR/WRAT TNEUT/WA PMXV/PMNV SIGSUM/R ALSF/ABSR XMLT/AMLT ZTA/J EPS/G CR/FISBU CAB/FRAA
 3.11040+07 3.58740+01 3.43027+01 352.050 1.07026 .38619 .35000 1.02578 1.67015 1.05093 65.0353 8.7162
 1.29140+06 3.60431-11 0.00000 51.318 .03676 .15757 .04400 1.07545 .77090 .75769 .3101 31.0950

U233 = 4.518-13 U235 = 2.060-04 PU239 = 9.597-05 PU241 = 1.244-05 TH232 = -0.000 U238 = 2.112-02
 U234 = 5.070-06 PU240 = 2.897-05 PA233 = 1.260-13 U236 = 4.744-05 PU242 = 2.723-06 NP237 = 4.128-06
 AM241 = 2.125-07 AM243 = 2.922-07 PO150 = -0.000 FPA = -3.183-06 FPU = 2.249-05 FPC = 2.554-04
 KE135 = 4.973-09 SM149 = 5.636-08 SM151 = 1.506-07

TIME/HEAT PAV/PHI PWR/WRAT TNEUT/WA PMXV/PMNV SIGSUM/R ALSF/ABSR XMLT/AMLT ZTA/J EPS/G CR/FISBU CAB/FRAA
 3.82764+07 3.54702+01 3.31356+01 350.484 1.09054 .38118 .34411 .99874 1.65605 1.05070 68.0374 7.5835
 1.57130+04 3.60431-11 0.00000 49.572 .03418 .15757 .04445 1.06335 .76168 .75233 .3021 31.1900
 SA1 = 1.7890+00 SA2 = 3.8118-01 SF1 = 8.8225+01 SF2 = 4.3785+01 SF3 = 1.1743+00 SFNU2 = 5.3294-01

BURN-UP OF REGION 1 IDENTIFIED AS 10 COMPLETED.

U233 = 3.025-13 U235 = 3.241-04 PU239 = 9.250-05 PU241 = 9.346-06 TH232 = -0.000 U238 = 2.105-02
 U234 = 4.642-06 PU240 = 2.158-05 PA233 = 1.066-13 U236 = 4.977-05 PU242 = 1.446-06 NP237 = 3.654-06
 AM241 = 1.228-07 AM243 = 1.240-07 PO150 = -0.000 FPA = -2.927-06 FPU = 2.242-05 FPC = 2.209-04
 KE135 = 6.273-09 SM149 = 6.854-08 SM151 = 1.702-07

TIME/HEAT PAV/PHI PWR/WRAT TNEUT/WA PMXV/PMNV SIGSUM/R ALSF/ABSR XMLT/AMLT ZTA/J EPS/G CR/FISBU CAB/FRAA
 3.11340+07 3.92865+01 4.29048+01 357.222 1.09210 .43865 .33804 1.07919 1.71302 1.05219 57.7949 9.8076
 1.41431+04 3.83289-11 0.00000 60.359 .02175 .16650 .71507 1.13063 .78095 .76861 .3613 29.1504

U233 = 4.915-13 U235 = 2.747-04 PU239 = 9.937-05 PU241 = 1.267-05 TH232 = -0.000 U238 = 2.099-02
 U234 = 4.344-06 PU240 = 2.819-05 PA233 = 1.452-13 U236 = 5.751-05 PU242 = 2.588-06 NP237 = 4.827-06
 AM241 = 1.997-07 AM243 = 2.783-07 PO150 = -0.000 FPA = -3.485-06 FPU = 2.465-05 FPC = 2.856-04
 KE135 = 5.979-09 SM149 = 6.576-08 SM151 = 1.690-07

TIME/HEAT PAV/PHI PWR/WRAT TNEUT/WA PMXV/PMNV SIGSUM/R ALSF/ABSR XMLT/AMLT ZTA/J EPS/G CR/FISBU CAB/FRAA
 3.82764+07 3.97758+01 4.08906+01 355.212 1.07866 .42833 .36596 1.04377 1.69629 1.05181 61.3473 8.4311
 1.76213+04 3.83289-11 0.00000 57.525 .01041 .16650 .71032 1.11600 .76915 .76110 .3475 29.2469
 SA1 = 1.8846+00 SA2 = 4.2833-01 SF1 = 9.8829+01 SF2 = 5.1070+01 SF3 = 1.3083+00 SFNU2 = 6.5105-01

BURN-UP OF REGION 2 IDENTIFIED AS 20 COMPLETED.

DIFFUSION OUTPUT AT 3.8276+07 SECONDS. POWER BASED ON OLD FLUX IS 3.4818+00

REACTIVITY WITH CONTROL RODS WITHDRAWN IS 1.01129

REGION	RADIUS	POWER	FLUX	PERCENT CHANGE	R	PERCENT CHANGE	SNFD	CFACT	DFACT
1	136.800	296.237	3.74765-11	3.825	.1513	-4.128	-.0000	1.0000	.0000
2	167.544	367.942	4.02073-11	4.672	.1562	-6.644	-.0000	1.0000	.0000
3	197.964	.0000	1.69074-11	38.091	.0186	-41.030	-.0000	1.0000	.0000

REGION	D1	D2	XISIS	DU	SIGA1	SIGA2	SIGR1	SIGR2	SIGFNU1	SIGFNU2
1	.688209	.408292	.027762	16.114337	.021027	.072193	.049549	.082066	.013802	.106617
2	.687911	.407435	.027412	16.106485	.022077	.080818	.050247	.090654	.015326	.122841
3	.574584	.305313	.059547	16.660337	.000000	.000000	.000231	.008891	.000000	.000000

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 1

FAST	6.275-04	6.275-04	6.275-04	6.276-04	6.280-04	6.296-04	6.356-04	6.551-04	7.082-04	8.292-04
THERMAL	2.123-04	2.123-04	2.123-04	2.123-04	2.125-04	2.130-04	2.151-04	2.219-04	2.399-04	2.720-04

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 2

FAST	8.292-04	8.669-04	8.775-04	8.684-04	8.414-04	7.970-04	7.359-04	6.568-04	5.495-04	3.569-04
THERMAL	2.720-04	2.647-04	2.642-04	2.607-04	2.524-04	2.393-04	2.221-04	2.043-04	2.014-04	2.892-04

UNNORMALIZED DIFFUSION FLUXES BY MESH POINT FOR REGION 3

FAST	3.569-04	1.242-04	4.320-05	1.503-05	5.233-06	1.822-06	6.340-07	2.199-07	7.410-08	1.880-08
THERMAL	2.892-04	3.575-04	2.684-04	1.740-04	1.056-04	6.172-05	3.496-05	1.879-05	8.629-06	1.320-06

NORMALIZED DIFFUSION FLUXES BY MESH POINT

RADIUS	0	10	20	30	40	50	60	70	80	90	100
.000	*	*	*	*	*	*	*	*	*	*	*
15.200	*	*	*	T	*	*	*	F	*	*	*
30.400	*	*	*	T	*	*	*	F	*	*	*
45.600	*	*	*	T	*	*	*	F	*	*	*
60.800	*	*	*	T	*	*	*	F	*	*	*
76.000	*	*	*	T	*	*	*	F	*	*	*
91.200	*	*	*	T	*	*	*	F	*	*	*
106.400	*	*	*	T	*	*	*	F	*	*	*
121.600	*	*	*	T	*	*	*	F	*	*	*
136.800	*	*	*	T	*	*	*	F	*	*	*
140.216	*	*	*	T	*	*	*	F	*	*	*
143.632	*	*	*	T	*	*	*	F	*	*	*
147.048	*	*	*	T	*	*	*	F	*	*	*
150.464	*	*	*	T	*	*	*	F	*	*	*
153.880	*	*	*	T	*	*	*	F	*	*	*
157.296	*	*	*	T	*	*	*	F	*	*	*
160.712	*	*	*	T	*	*	*	F	*	*	*
164.128	*	*	*	T	*	*	*	F	*	*	*
167.544	*	*	*	T	*	*	*	F	*	*	*
170.960	*	*	*	T	*	*	*	F	*	*	*
174.376	*	*	*	T	*	*	*	F	*	*	*
177.792	*	*	*	T	*	*	*	F	*	*	*
181.208	*	*	*	T	*	*	*	F	*	*	*
184.624	*	*	*	T	*	*	*	F	*	*	*
188.040	*	*	*	T	*	*	*	F	*	*	*
191.456	*	*	*	T	*	*	*	F	*	*	*
194.872	*	*	*	T	*	*	*	F	*	*	*
197.964	*	*	*	T	*	*	*	F	*	*	*

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