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NBS TECHNICAL NOTE 576

Computer Code for the Calculation of Thermal Neutron Absorption in Spherical and Cylindrical Neutron Sources

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Computer Code for the Calculation of Thermal Neutron Absorption in Spherical and Cylindrical Neutron Sources

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Abstract

A computer code has been written in FORTRAN IV for the calculation of thermal neutron absorption in spherical and cylindrical neutron sources. The formalism of the calculation, the structure of the computer code, a listing of the code, and some sample results are presented. The comparison of the results of this calculation to experiment appears elsewhere (1).

Key words: Manganous sulfate bath calibration of neutron sources; neutron; neutron standards.

Computer Code for the Calculation of Thermal Neutron Absorption in Spherical and Cylindrical Neutron Sources

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1. INTRODUCTION

This calculation has been carried out in connection with a program to reduce the uncertainties in the corrections applied to the manganous sulfate bath calibration of neutron sources (2,3). The correction considered here is to account for the reduction of the manganese activity due to the loss of thermalized neutrons absorbed in the neutron source itself.

The source may be composed of up to three cladding and one, possibly fissionable, core material. The calculation is carried out in a single interaction approximation, i.e., the effects of elastic and inelastic scattering of thermal neutrons are neglected. This approximation is adequate because the neutrons are in thermal equilibrium and because the correction which is applied to the calibration is small (typically 1% or less). Two cases are available. The first is for a spherically symmetric source and the second is for a cylindrically symmetric source. Each consists of a core and up to three cladding layers. The thicknesses of the ends and side of a cladding cylinder may all be different. A measurement or knowledge of the thermal-neutron flux at the source location is required. The thermal-neutron flux is assumed to be isotropic, which enables one to carry out the computation as a sum of mono-directional fluxes from different directions. All integrations are performed with Weddle's formula (4).

2. DESCRIPTION OF THE CALCULATION

Part I. The probability of neutron loss for a given neutron direction and position.

If the source materials are labeled A, B, C, and D from the inside out, then the probability of a thermal neutron interacting in passing through the source is

$$P(t) = 1-e^{-\Sigma_a a(t)-\Sigma_b b(t)-\Sigma_c c(t)-\Sigma_d d(t)} \qquad (1)$$

where a(t) is the thickness of "A" material for this particular direction and location of passing through the source, b(t) is the total thickness of "B" material, etc., and "t" denotes any particular path through the source. The Σ's are the appropriate macroscopic cross sections, Σ_a being the sum of

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the absorption and fission cross sections for the "A" material, Σ_b being the absorption cross section for the "B" material, etc.

The probability of fission in the fissionable material is given by

$$P_f(t) = \frac{\Sigma_{af}}{\Sigma_a}\left(1-e^{-\Sigma_a a(t)}\right)\left(e^{-(\Sigma_d d'(t)+\Sigma_c c'(t)+\Sigma_b b'(t))}\right) \quad (2)$$

where $d'(t)$ is the thickness of the "D" layer passed in this direction and location going from the outside into the "A" material, $c'(t)$ the thickness of the "C" layer passed in going into the source, etc., and Σ_{af} is the macroscopic fission cross section of material "A".

Part II. Case 1. A spherically symmetric source.

In this case the probability of neutron disappearance for a neutron striking the source is given by

$$P = \frac{\int_0^R 2\pi y \, P(y) \, dy}{\pi R^2} \quad (3)$$

where R is the outer radius of the source and y is the perpendicular distance from the center of the source to the path through it. The probability of fission is similarly

$$P_f = \frac{\int_0^R 2\pi y \, P_f(y) \, dy}{\pi R^2} \quad (4)$$

The probability of loss of neutrons from the bath per neutron striking the source is therefore

$$P_L = P - kP_f \quad (5)$$

where k is the number of neutrons per fission interaction.

Part II. Case 2. A cylindrically symmetric source.

The notation for the dimensions of the source is given in Figure 1. Since an isotropic flux is assumed it is necessary only to consider monodirectional fluxes from an appropriate number of different directions. By symmetry it is most convenient to use cylindrical coordinates. For convenience in normalization and avoidance of solid angle considerations an imaginary sphere is placed about the source and all neutrons striking this sphere are included in the integrals and probabilities. This procedure does not affect the results when they are finally expressed in terms of neutrons lost as a percent of the source strength or in terms of a given thermal flux in the vicinity of the source.

Given that a neutron enters this imaginary sphere, let the probability of transmission without interaction be P_T, probability of interaction be P_I, probability of absorption be P_A and the probability of fission be P_f. The probability of neutron loss in the source P_L is

$$P_L = P_I - kP_f = 1 - P_T - kP_f \tag{6}$$

where k is the number of neutrons per fission. The program computes P_T and P_f by integration and P_I and P_A by $P_I = 1 - P_T$ and $P_A = P_I - P_f$.

Part III. The integration and normalization.

Consider the cylindrical source surrounded with a sphere centered about the core material with radius, R, just large enough to enclose the outer cylinder as shown in Figure 1(a). The thermal-neutron flux is taken to be isotropic. Neutron flux is the number of neutrons per second entering a sphere of 1 square centimeter cross section. The number of neutrons per second entering the sphere surrounding the cylindrical source is the cross sectional area of the surrounding sphere times the thermal flux.

Since the geometry is symmetric about the x-axis, the probability for various results for an isotropic flux will equal the probabilities evaluated assuming a flux uniformly distributed in z, w, and φ but all paths constrained to lie in planes parallel to the x-y plane. The neutron paths are perpendicular to all points along the w-axis within the sphere and φ is the angle between the y and w axes. The probability of transmission of a neutron through the imaginary sphere averaged over all directions is thus

$$P_T = \int_{-R}^{R} dz \int_{0}^{2\pi} d\varphi \int_{-r(z)}^{r(z)} dw \; P_T(z,w,\varphi)/2\pi(\pi R^2) \tag{7}$$

The probability of transmission, $P_T(z,w,\varphi)$ is

$$P_T(z,w,\varphi) = e^{-(\Sigma_a a(z,w,\varphi)+\Sigma_b b(z,w,\varphi)+\Sigma_c c(z,w,\varphi)+\Sigma_d d(z,w,\varphi))} \tag{8}$$

where the Σ's are the macroscopic cross sections. Parameters are as in the spherical case and the a, b, c, and d are the thicknesses of the A, B, C, and D materials as seen at the particular z, w, and φ. For a transparent source the macroscopic cross sections are all zero and equation (7) integrates to unity. The fission probability is similarly

$$P_F = \int_{-R}^{R} dz \int_{0}^{2\pi} d\varphi \int_{-r(z)}^{r(z)} dw \; P_f(z,w,\varphi)/2\pi(\pi R^2) \qquad (9)$$

where $P_f(z,w,\varphi) = (1-e^{-\beta})e^{-\gamma}(\sigma_{af}/\sigma_a)$.

In $P_f(z,w,\varphi)$;

$\beta = \Sigma_a \; a(z,w,\varphi)$,

$1-e^{-\beta}$ is the probability for interaction in the "A" material,

$\gamma = \Sigma_b b_i(z,w,\varphi)+\Sigma_c c_i(z,w,\varphi)+\Sigma_d d_i(z,w,\varphi)$,

where $b_i(z,w,\varphi)$ is the thickness of the "B" layer transversed passing into the "A" material and likewise for c_i and d_i; $e^{-\gamma}$ is the probability for passing through the cladding material without interaction; σ_{af} is the fission cross section for the "A" material, and (σ_{af}/σ_a) is the probability for fission, given that an interaction occurs in the "A" material.

3. DESCRIPTION OF THE CODE

3.1 Spherical Case Calculation

3.1.1 Organization of the Code

The program first reads the title card and input data, converts the source dimensions from inches to centimeters and then prints out an echo check of the title, input data and converted dimensions.

The approximate total number of subdivisions, NY, of the radius is specified in the input. Then each layer is subdivided in proportion to its thickness in the radial direction. The boundary between the central core and first encapsulation or the boundary between any of the encapsulations are always made an endpoint of subdivision so that any discontinuities occasioned by a change in derivative at a boundary do not occur within an interval of integration.

The integration of Equation (4) then begins with the central core and continues to the outside radius, layer by layer. All neutron paths at a given radius are perpendicular to a cross section through the center of the source and are equal in length. Subroutine Grand calculates the interaction probability in the A, B, C, and D materials and the transmission probability along the path specified. The probability of interaction in each material and the transmission probability with its error are summed and normalized in the main program. The results are then printed below the echo printout of the input data.

3.1.2 Approximations in the Numerical Integration

The numerical integration was performed using Weddle's Rule (4). If the core material has been divided into N subdivisions, Weddle's Rule replaces the integrand of the integral by N fifth-degree parabolas and numerically integrates over that region. The accuracy of the integration and estimate of the error depend upon the size of the integration intervals. Convergence of the integration procedure was checked for a plutonium-beryllium source with encapsulations of tantalum and stainless steel by varying the number of subdivisions, NY, of the radius between 10 and 1,000. In Table 1 the calculated transmissions and estimated errors are shown together with the actual error based upon accepting the finest subdivision as the correct answer. The estimated and actual error appear to agree for 20 subdivisions of NY, but this degree of accuracy is certainly not necessary. The difference in net neutron loss calculated for NY equal 10 and 1000 is only 0.005% out of 0.278%.

This program has been run in double precision, because it was created, for the most part, from the deck of cards to the cylindrical case, which had to be run in double precision. The number of additions involved in the transmission integral is approximately equal to 6 times the number of subdivisions, NY, of the radius specified in the input. Therefore no rounding error should be introduced by performing the program in single precision. The program takes about 3 seconds to compile and about 2 seconds to execute in the case of NY equal 100. It requires 1437 words of memory.

3.2 Cylindrical Case Calculation

3.2.1 Organization of the Code

Here too the title card and input data are read, the source dimensions are converted to centimeters, and an echo check of the input is printed.

According to the number, NZ, of subdivisions in the Z-direction, specified in the input, each layer in the Z-direction is subdivided in proportion to its thickness. The subdivision of the angle of integration from zero to $\pi/2$ is performed according to the number, NPHI, specified in the input. The subdivisions along the W-axis are set at the interface of any two materials in a plane parallel to the X-Y axis, distance Z above the origin. The D, C, and B materials in the negative W-direction, the core A material, and the B, C, and D materials in the positive W-direction are all divided into six subintervals. The limits of integration for each of these subintervals along the W-axis are passed to Subroutine IN. It checks the length of the interval and will further subdivide it by 6 or 36, if it is more than 6 times the length of a subdivision in the Z-direction. It sums and weights the contributions to the transmission, the error for the transmission, and the interaction in the core material for each path through the source perpendicular to the W-axis between the specified limits of integration and returns the result to the main program. These in turn are summed and weighted according to the Weddle Rule for the integration over angle, φ, and then over height, Z. The probabilities for transmission with its error, absorption, and fission are then printed.

Subroutine C2 calculates the X and Y coordinates of the points of incidence and the pathlength through any of the cylindrical source encapsulations or core material and returns the result to Subroutine Grand. Subroutine Grand calculates the segments of the path in each of the source materials in order to compute the transmission and fission probabilities along that path and returns the result to Subroutine PCE. Subroutine PCE subdivides by six the subinterval specified by Subroutine IN and performs the six point Weddle integration of transmission and fission probabilities with an estimate of the error for this segment of transmission. The seven values of the integrand are received from Subroutine Grand and after proper normalization the results are returned to Subroutine IN.

3.2.2 Approximations in the Numerical Integration

The comments about the use of the Weddle Rule in 3.1.2 also apply here. In this case we integrate over Z, φ, and W. The size of the integration steps for Z and φ are specified in the input and, because the interval along the W-axis varies with angle, φ , the thicknesses of the encapsulations, and the inside or core length, the length of the W integration step is tested against the length of the Z step and further subdivided, if necessary. The absolute values of the estimated error for each segment of integration along the W-axis are summed and added in quadrature with the estimated error in the φ integration. This error is then added in quadrature with the estimated error in the Z integration. It is also possible to test the relative error of any subinterval by testing against ET, which is specified in the input, and to further subdivide, if necessary. It has been found, however, that ET may safely be set as low as 1% or 0.5% without causing many subdivisions, but setting ET as low as o.1% or 0.05% will suddenly cause so many subdivisions that the program will be stopped automatically by exceeding the maximum time specified on the run card and all data are then lost. The transmission integral has also been found to be insensitive to the value of ET. Each time an interval in the W integration is subdivided once, twice or three times a counter, KB, KC, or KD is incremented and printed in the program output.

The convergence of the cylindrical program has been tested for two different plutonium-beryllium sources, one with a single nickel encapsulation and the other with encapsulations of tantalum and stainless steel. In Figure 2 are shown the calculated transmissions and estimated errors for the nickel encapsulated source for various specified subdivisions of the Z and φ integrals, NZ and NPHI, respectively. A solid line joins those points for which NPHI is held constant at 10 and a dashed line joins those points for which NZ is held constant at 10. Two points are also shown for both NPHI and NZ equal to 5 and 15. Taking finer steps in the Z integral, which forces finer steps in the W integral, increases the transmission for this source, whereas finer steps in the φ integral decreases it. Taking equal steps in Z and φ appear to result in a more rapid convergence, though this may be fortuitous. The difference in net neutron loss in the source for the two extreme points on the figure amounts to 0.01%, which is of no consequence in such a calculation. The convergence of the program for the tantalum and stainless steel encapsulated source was tested for NZ equal NPHI at values of 3, 6, and 10. The transmissions were 0.1839, 0.1842 and 0.1843, respectively. The negligible difference in transmission corresponded to run times of 188, 516, and 1172 seconds, respectively.

This program was affected by rounding errors when run in single precision by two different compilers. The results were only independent of choice of compiler, when run in double precision. The program takes about 5 seconds to compile and requires 2992 words of memory. Typical run times are listed in the above paragraph.

4.1 SPHERICAL CASE

4.1.1 LIST OF THE PROGRAM

```
      IMPLICIT DOUBLE PRECISION(A-H,O-Z)
      DIMENSION V(6),T(14),EH(6),         RZZ(5),KA(4)
      DATA V/2.D0,5.D0,1.D0,6.D0,1.D0,5.D0/,                EH/1.D0,-6.D0,
     115.D0,-20.D0,15.D0,-6.D0/,CC/1.D-8/
    1 FORMAT(13A6,A2/7G8.4,I4)
    3 FORMAT(12G6.3)
    4 FORMAT(1H1,13A6,A2/' SOURCE DIMENSIONS'25X'/ MACROSCOPIC CROSS SEC
     1    '17X'/ INTEGRATION PARAMETERS'/' RA    ='G14.6,' TB    ='G14.6,'
     1 / SIGAA='G14.6,' AK    ='G14.6,' / NY    ='I6/' RB    ='G14.6,' TC
     1 ='G14.6,' / SIGAF='G14.6,' F     ='G14.6,              /' RC    ='G1
     14.6,' TD    ='G14.6,' / SIGB ='G14.6,                 /' RD    ='G
     114.6,21X              ' / SIGC ='G14.6,21X' / NCY  ='I6/ 42X
     1                     ' / SIGD ='G14.6,21X' /'              /
     1   11X       ' IN INCHES,                        ' 2X    ' / COVER / WALL
     1 THICKNESS                                 '/13X'OUTSIDE DIAMETER='G12.6,
     1' /    B'4XG12.6,                          /43X'/    C'4XG12.6,
     1          /43X'/    D'4XG12.6)
    5 FORMAT(3X/    46H    FOR A SINGLE NEUTRON STRIKING THE SOURCE.../
     120X35HTHE TRANSMISSION PROBABILITY IS     22XG16.8/
     120X35HTHE INTERACTION PROBABILITY IS      22XG16.8/
     120X35HTHE ABSORPTION PROBABILITY IS       22XG16.8/
     120X44HTHE 'A' MATERIAL INTERACTION PROBABILITY IS 13XG16.8  /
     120X44HTHE 'B' MATERIAL INTERACTION PROBABILITY IS 13XG16.8  /
     120X44HTHE 'C' MATERIAL INTERACTION PROBABILITY IS 13XG16.8  /
     120X44HTHE 'D' MATERIAL INTERACTION PROBABILITY IS 13XG16.8  /
     120X57HTHE PROBABILITY FOR A FISSION ABSORPTION IS              G16
     1.8/16X'    NEUTRONS OUT PER NEUTRON STRIKE IS ' 22XG16.8
     1/20X'NEUTRONS LOST PER NEUTRON STRIKE    '22XG16.8,' .'
     1//38H    THE CROSS SECTION OF THE SOURCE IS G13.8,' CM**2.'
     1/3X'THE ERROR IN THE TRANSMISSION PROBABILITY IS ESTIMATED TO BE
     1'G12.3,'PERCENT.')
    7 FORMAT(3X/    44H    FOR A FLUX OF ONE NEUTRON PER CM**2-SEC.(G12.5,
     141H NEUTRONS PER SEC) STRIKING THE SOURCE.../
     120X26HTHE TRANSMISSION IS         G16.8,20H NEUTRONS PER SECOND/
     120X26HTHE INTERACTION IS          G16.8,20H NEUTRONS PER SECOND/
     120X26HTHE ABSORPTION IS           G16.8,20H NEUTRONS PER SECOND/
     120X26HTHE FISSION EMISSION IS     G16.8,20H NEUTRONS PER SECOND/
     116X26HAND THE RESULTANT LOSS IS 4XG16.8,' NEUTRONS PER SECOND.')
    8 FORMAT(3X/    35H    IF THE MEASURED THERMAL FLUX IS G12.4, ' PER CE
     1NT OF Q PER CM**2 THEN...'/
     120X26HTHE TRANSMISSION IS         G16.8,14H PER CENT OF Q/
     120X26HTHE INTERACTION IS          G16.8,14H PER CENT OF Q/
     120X26HTHE ABSORPTION IS           G16.8,14H PER CENT OF Q/
     120X26HTHE FISSION EMISSION IS     G16.8,14H PER CENT OF Q/
     116X26HAND THE RESULTANT LOSS IS 4XG16.8,' PER CENT OF Q.')
    9 FORMAT(1X'THE NUMBER OF GRAND CALLS WAS'I8,'  THE LARGEST SUBDIVIS
     1ION OF THE RADIUS WAS  'G16.8)
  100 READ(5,1)T,SIGAA,SIGAF,SIGB,SIGC,SIGD,AK,F,NY
      READ(5,3) DIA,TBI,TCI,TDI    'OUTSIDE DIAMETER+SHELL THICKNESSES
      IGRAND=0                     'COUNTER RESET FOR NUMBER OF SUB GRAND CALLS
      NCY=4                        'NCY EQUALS THE NUMBER OF LAYERS
      IF(TDI.LT.CC) NCY=3          '    3 LAYERS
      IF(TCI.LT.CC) NCY=2          '    2 LAYERS
      IF(TBI.LT.CC) NCY=1          '    1 LAYER
      TB=TBI*2.54D0                'THE INPUT DIMENSIONS WERE IN INCHES
      TC=TCI*2.54D0                'F=THE NEUTRON FLUX IN PERCENT OF Q
      TD=TDI*2.54D0                'T=TITLE,  AK=NEUTRONS PER FISSION
      RD=DIA*2.54D0/2.D0           'OUTSIDE RADIUS
      RC=RD-TD                     'FOR ONE LAYER A,B,C, AND D DIMENS. ARE EQUAL
      RB=RC-TC                     'FOR TWO LAYERS B,C, AND D DIMENSIONS ARE EQUAL
      RA=RB-TB                     'FOR THREE LAYERS C AND D DIMENSIONS ARE EQUAL
      DY=0.D0                      'LARGEST DIV. OF RADIUS, Y, IS MAX VALUE OF DA
      SIGAT=SIGAF+SIGAA            'SIGAT=TOTAL CROSS SECTION, 'A' MATERIAL
```

```
      WRITE(6,4)T,RA,TB,SIGAA,AK,NY,RB,TC,SIGAF,F,RC,TD,SIGB,RD,SIGC,
     1 NCY,SIGD,DIA,TBI,TCI,TDI
      RZZ(1)=0.D0
      RZZ(2)=RA
      RZZ(3)=RB
      RZZ(4)=RC
      RZZ(5)=RD
      EKI=0.D0
      STI=0.D0
      SYD=0.D0
      SYC=0.D0
      SYB=0.D0
      SYA=0.D0
      ANY=NY/RD                  'NY=APPROXIMATE NO. OF Y INTERVALS
      KA(1)=RA*ANY+.5D0          'THIS IS TO SUBDIVIDE THE Y-LAYERS IN
      KA(2)=(RB-RA)*ANY+.5D0     'PROPORTION TO THE THICKNESS OF EACH LAYER
      KA(3)=(RC-RB)*ANY+.5D0     'FROM ZERO TO RA, FROM RA TO RB, FROM RB
      KA(4)=(RD-RC)*ANY+.5D0     'TO RC, AND FROM RC TO RD. Y-INTEGRATION WILL
      DO 281 IAR=1,NCY           'ALWAYS OCCUR AT LAYER BOUNDARIES AND AVOID
      IF(KA(IAR).LT.1)KA(IAR)=1  'DISCONTINUITIES.
      DA=(RZZ(IAR+1)-RZZ(IAR))/(6.D0*KA(IAR))
      IF(DA.GT.DY)DY=DA          'LARGEST DIVISION OF RADIUS, Y
      NYP=6*KA(IAR)+1            'THE NUMBER OF PTS OF INTEGR. IN EACH LAYER
      IVY=2                      'IVY= Y INTEGRATION WEDDLE RULE COEF INDICATOR
      SPYT=0.D0                  'Y PARTIAL SUM RESET OF TRANSMISSION AND
      SPYD=0.D0                  'INTERACTION IN 'D' MATERIAL
      SPYC=0.D0                  'INTERACTION IN 'C' MATERIAL
      SPYB=0.D0                  'INTERACTION IN 'B' MATERIAL
      SPYA=0.D0                  'INTERACTION IN 'A' MATERIAL
      EK=0.D0                    'ERROR RESET FOR Y INTEGRAL IN DO LOOP 280
      TEK=0.D0                   'TEMPORARY EK
      DO 280  I=1,NYP            'BEGIN Y INTEGR.FR 0-RA-RB-RC-RD
      Y=DA*DBLE (I-1)+RZZ(IAR)
      RDY=0.D0
      RCY=0.D0
      RBY=0.D0
      RAY=0.D0
      GO TO (173,172,171,170),NCY
  170 IF(RD.GT.Y)RDY=DSQRT(RD**2-Y**2)
  171 IF(RC.GT.Y) RCY=DSQRT(RC**2-Y**2) 'HIT C LAYER, IF THERE IS ONE
  172 IF(RB.GT.Y) RBY=DSQRT(RB**2-Y**2) 'HIT B LAYER, IF THERE IS ONE
  173 IF(RA.GT.Y) RAY=DSQRT(RA**2-Y**2) 'HIT A LAYER
      CALL GRAND(Y,SPIA,SPIB,SPIC,SPID,SPT)
      IF(I.EQ.1.OR.I.EQ.NYP) GO TO 270
      GO TO(180,181,182,183),NCY
  183 SPYD=SPYD+V(IVY)*SPID
  182 SPYC=SPYC+V(IVY)*SPIC
  181 SPYB=SPYB+V(IVY)*SPIB
  180 SPYA=SPYA+V(IVY)*SPIA
      SPYT=SPYT+V(IVY)*SPT
      TEK=TEK+EH(IVY)*SPT 'SUM ABSOLUTE VALUE OF ERROR EACH 6 SUBINTERVALS
      IVY=IVY+1
      IF(IVY.GE.7) IVY=1
      IF(IVY.EQ.2)GO TO 269
      GO TO 280
  269 EK=EK+DABS(TEK)            'EK=ERROR IN Y INTEGRAL DUE TO WEDDLE RULE
      TEK=SPT
      GO TO 280
  270 GO TO(271,272,273,274),NCY'COEFF. OF FIRST AND LAST TERMS IS 1.
  274 SPYD=SPYD+SPID
  273 SPYC=SPYC+SPIC
  272 SPYB=SPYB+SPIB
  271 SPYA=SPYA+SPIA
      SPYT=SPYT+SPT
      EK=EK+SPT
  280 CONTINUE
      EKI=EKI+EK*DA/140.D0       ' Y NORMALIZATION,  WEDDLE RULE
      STI=.3D0*DA*SPYT+STI       'TRANSMISSION INTEGRAL
      GO TO(290,291,292,293),NCY
  293 SYD=SYD+0.3D0*DA*SPYD
  292 SYC=SYC+0.3D0*DA*SPYC
  291 SYB=SYB+0.3D0*DA*SPYB
```

```
290 SYA=SYA+0.3D0*DA*SPYA
281 CONTINUE                  'END OF Y INTEGRAL
    VNORM=2.D0/RD**2          'VNORM=2/RD**2      AND UNIT NORMALIZES
    STIS=STI*VNORM            'THE INTEGRALS
    SAFIS=SYA*VNORM           'INTERACTION IN 'A' MATERIAL
    SBFIS=SYB*VNORM           'INTERACTION IN 'B' MATERIAL
    SCFIS=SYC*VNORM           'INTERACTION IN 'C' MATERIAL
    SDFIS=SYD*VNORM           'INTERACTION IN 'D' MATERIAL
    IF(SIGAT.LT.1.D-4)GO TO 282
    SFIS=SAFIS*SIGAF/SIGAT 'FISSION PART OF 'A' INTERACTION
    GO TO 283
282 SFIS=0.D0
283 SNOS=SFIS*AK+STIS         'NEUTRONS OUT PER NEUTRON HITTING SPHERE
    ACROS=3.1416D0*RD**2      'SOURCE CROSS SECTION
    EM=EKI*1.D2/STI           'EM=ESTIMATE OF PER CENT ERROR IN STI
    FTSOU=STIS                'SINGLE STRIKE TRANSMISSION PROBABILITY
    FINSOU=1.D0-FTSOU         'SINGLE STRIKE INTERACTION PROBABILITY
    FMULSO=SFIS               'SINGLE STRIKE FISSION PROBABILITY
    FABSOU=FINSOU-FMULSO      'SINGLE STRIKE ABSORPTION PROBABILITY
    FOUTSO=FTSOU+FMULSO*AK 'NEUTRONS OUT PER NEUTRON STRIKING SOURCE
    FNOUT=1.D0-FOUTSO         'RESULTANT LOSS
    FPT=STIS*ACROS            'UNIT FLUX TRANSMISSION
    FPI=FINSOU*ACROS          'UNIT FLUX INTERACTION
    FPA=FABSOU*ACROS          'UNIT FLUX ABSORPTION
    FPF=SFIS*AK*ACROS         'UNIT FLUX FISSION
    FNLS=FPI-FPF              'UNIT FLUX, NET LOSS
    FQT=F*FPT                 'PERCENT Q OF TRANSMISSION
    FQI=F*FPI                 'PERCENT Q INTERACTION
    FQA=F*FPA                 'PERCENT Q ABSORPTION
    FQF=F*FPF                 'PERCENT Q FISSION GAIN
    FQL=F*FNLS                'PERCENT Q NET LOSS
    WRITE(6,9)IGRAND,DY
    WRITE(6,5) FTSOU,FINSOU,FABSOU,SAFIS,SBFIS,SCFIS,SDFIS,FMULSO,
   1FOUTSO,FNOUT,ACROS,EM
    WRITE(6,7)ACROS,FPT,FPI,FPA,FPF,FNLS
    WRITE(6,8)F,FQT,FQI,FQA,FQF,FQL
    GO TO 100

    SUBROUTINE GRAND(W,DUMA,DUMB,DUMC,DUMD,DMDA)
    IGRAND=IGRAND+1           'COUNT OF SUBROUTINE CALL
    DUMA=0.D0                 'CONTAINS MULTIPLICATION PART OF INTEGRAND
    DUMB=0.D0
    DUMC=0.D0
    DUMD=0.D0
    A=0.D0
    B=0.D0
    C=0.D0
    D=0.D0
    GO TO (370,360,350,340),NCY
340 D=(RDY-RCY)*2.D0            'THE AMOUNT OF A,B,C, AND D MATERIAL
350 C=(RCY-RBY)*2.D0            'TRAVERSED IS A,B,C, AND D, RESPECTIVELY.
360 B=(RBY-RAY)*2.D0
370 A=RAY*2.D0
    DMD=-(SIGAT*A+SIGB*B+SIGC*C+SIGD*D)
    DMDEX=DEXP(DMD)
    DMDA=W*DMDEX              'TRANSMISSION PART OF INTEGRAND
    DEXD=DEXP(-SIGD*D/2.D0)
    DEXC=DEXP(-SIGC*C/2.D0)
    DEXB=DEXP(-SIGB*B/2.D0)
    DEXA=DEXP(-SIGAT*A/2.D0)
    GO TO(400,401,402,403),NCY
403 DUMDA=W*(1-DMDEX)         'INTERACTION IN A,B,C,AND D MATERIAL
402 DUMCA=W*(DEXD-DMDEX/DEXD) 'INTERACTION IN A,B, AND C MATERIAL
401 DUMBA=W*(DEXD*DEXC-DMDEX/(DEXD*DEXC))'INTERACTION IN A AND B MATERIAL
400 DUMA=W*(DEXD*DEXC*DEXB-DMDEX/(DEXD*DEXC*DEXB)) 'INTERAC. IN A MAT.
    GO TO(410,411,412,413),NCY
413 DUMD=DUMDA-DUMCA          'INTERACTION IN D MATERIAL
412 DUMC=DUMCA-DUMBA          'INTERACTION IN C MATERIAL
411 DUMB=DUMBA-DUMA           'INTERACTION IN B MATERIAL
410 RETURN
    END
```

4.1.2 LIST OF THE INPUT DATA DECK

```
     TITLE CARD.        FORMAT(13A6,A2)
EQUIVALENT SPHERICAL SOURCE FOR CYLINDRICAL SOURCE M-621                CARD 1
MACROSCOPIC CROSS SECTIONS AND INTEGRATION PARAMETERS.  FORMAT(7G8.4,I4)
12345678901234567890123456789012345678901234567890123456789012345678901234567890
1.865   4.305   1.16    .2811   .0      2.8     .122    100             CARD 2
SIGAA   SIGAF   SIGB    SIGC    SIGD    AK      F       NY
I       I       I       I       I       I       I       1/6 NUMBER OF INTEGRA-
I       I       I       I       I       I       I       TION STEPS
I       I       I       I       I       I       FLUX IN PERCENT OF SOURCE
I       I       I       I       I       I       STRENGTH PER CM**2
I       I       I       I       I       NUMBER OF NEUTRONS PER FISSION
I       I       I       I       ABSORPTION CROSS SECTION PER CM FOR D MATERIAL
I       I       I       ABSORPTION CROSS SECTION PER CM FOR C MATERIAL
I       I       ABSORPTION CROSS SECTION PER CM FOR B MATERIAL
I       FISSION CROSS SECTION PER CM FOR A MATERIAL
ABSORPTION CROSS SECTION PER CM FOR A MATERIAL

      SOURCE DIMENSIONS IN INCHES.   FORMAT(12G6.3)
12345678901234567890123456789012345678901234567890123456789012345678901234567890
1.913 .11   .071                                                        CARD 3
DIA   TBI   TCI   TDI
I     I     I     D SHELL THICKNESS IN INCHES
I     I     C SHELL THICKNESS IN INCHES
I     B SHELL THICKNESS IN INCHES
OUTSIDE DIAMETER OF SOURCE IN INCHES
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4.1.3 SIMPLIFIED SAMPLE OUTPUT

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  EQUIVALENT SPHERICAL SOURCE FOR THE CYLINDRICAL PU-BE M-621 SOURCE
  SOURCE DIMENSION INPUT IN INCHES, OD=1.913, B SHELL=0.110, C SHELL=0.071.
  MACROSCOPIC CROSS SECTIONS PER CM
  SIGAA= 1.865   , SIGAF=  4.305   ,  SIGB=  1.16    ,  SIGC=  0.2811
  NEUTRONS PER FISSION = 2.80     NY=100.
  FOR A SINGLE NEUTRON STRIKING THE SOURCE...
  THE TRANSMISSION PROBABILITY IS                               .13567724+0
  THE INTERACTION PROBABILITY IS                                .86432276+0
  THE ABSORPTION PROBABILITY IS                                 .59782176+0
  THE 'A' MATERIAL INTERACTION PROBABILITY IS                   .38195382+0
  THE 'B' MATERIAL INTERACTION PROBABILITY IS                   .36978466+0
  THE 'C' MATERIAL INTERACTION PROBABILITY IS                   .11258428+0
  THE PROBABILITY FOR A FISSION ABSORPTION IS                   .26650101+0
  NEUTRONS OUT PER NEUTRON STRIKE IS                            .88188005+0
  NEUTRONS LOST PER NEUTRON STRIKE IS                           .11811995+0
  THE CROSS SECTION OF THE SOURCE IN CM**2 IS                   .18543353+2
  THE PERCENT ERROR IN THE TRANSMISSION PROBABILITY IS          .0512
  IF THE MEASURED THERMAL FLUX IN PERCENT OF Q PER CM**2 IS .122
  THE TRANSMISSION IN PERCENT OF Q IS                           .30694114+0
  THE INTERACTION IN PERCENT OF Q IS                            .19553480+1
  THE ABSORPTION IN PERCENT OF Q IS                             .13524456+1
  THE FISSION IN PERCENT OF Q IS                                .16881265+1
  AND THE RESULTANT LOSS IN PERCENT OF Q IS                     .26722146+0

  EQUIVALENT SPHERICAL SOURCE FOR THE CYLINDRICAL PU-BE A SOURCE
  SOURCE DIMENSION INPUT IN INCHES, OD=1.180, B SHELL=0.044.
  MACROSCOPIC CROSS SECTIONS PER CM
  SIGAA= 1.214   , SIGAF=  2.802   ,  SIGB=  0.4106
  NEUTRONS PER FISSION = 2.80     NY=100.
  FOR A SINGLE NEUTRON STRIKING THE SOURCE...
  THE TRANSMISSION PROBABILITY IS                               .26256084+0
  THE INTERACTION PROBABILITY IS                                .73743916+0
  THE ABSORPTION PROBABILITY IS                                 .41973395+0
  THE 'A' MATERIAL INTERACTION PROBABILITY IS                   .45535478+0
  THE 'B' MATERIAL INTERACTION PROBABILITY IS                   .28208437+0
  THE PROBABILITY FOR A FISSION ABSORPTION IS                   .31770520+0
  NEUTRONS OUT PER NEUTRON STRIKE IS                            .11521354+1
  NEUTRONS LOST PER NEUTRON STRIKE IS                          -.15213542+0
  THE CROSS SECTION OF THE SOURCE IN CM**2 IS                   .70554114+1
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THE PERCENT ERROR IN THE TRANSMISSION PROBABILITY IS      .0185
IF THE MEASURED THERMAL FLUX IN PERCENT OF Q PER CM**2 IS .122
THE TRANSMISSION IN PERCENT OF Q IS                       .22600192+0
THE INTERACTION IN PERCENT OF Q IS                        .63475827+0
THE ABSORPTION IN PERCENT OF Q IS                         .36129028+0
THE FISSION IN PERCENT OF Q IS                            .76571038+0
AND THE RESULTANT LOSS IN PERCENT OF Q IS                -.13095211+0


EQUIVALENT SPHERICAL SOURCE FOR THE CYLINDRICAL AM-BE SOURCE
SOURCE DIMENSION INPUT IN INCHES, OD=1.551, B SHELL=0.080, C SHELL=0.042.
MACROSCOPIC CROSS SECTIONS PER CM
SIGAA= .1704   , SIGAF=  0.00082 ,  SIGB=  1.16    ,  SIGC=  0.281
NEUTRONS PER FISSION = 2.89      NY=100.
FOR A SINGLE NEUTRON STRIKING THE SOURCE...
THE TRANSMISSION PROBABILITY IS                           .34932104+0
THE INTERACTION PROBABILITY IS                            .65067896+0
THE ABSORPTION PROBABILITY IS                             .64994216+0
THE 'A' MATERIAL INTERACTION PROBABILITY IS               .15384751+0
THE 'B' MATERIAL INTERACTION PROBABILITY IS               .41839021+0
THE 'C' MATERIAL INTERACTION PROBABILITY IS               .78441236-1
THE PROBABILITY FOR A FISSION ABSORPTION IS               .73680038-3
NEUTRONS OUT PER NEUTRON STRIKE IS                        .35145039+0
NEUTRONS LOST PER NEUTRON STRIKE IS                       .64854961+0
THE CROSS SECTION OF THE SOURCE IN CM**2 IS               .12189389+2
THE PERCENT ERROR IN THE TRANSMISSION PROBABILITY IS      .0240
IF THE MEASURED THERMAL FLUX IN PERCENT OF Q PER CM**2 IS .121
THE TRANSMISSION IN PERCENT OF Q IS                       .51521920+0
THE INTERACTION IN PERCENT OF Q IS                        .95969683+0
THE ABSORPTION IN PERCENT OF Q IS                         .95861012+0
THE FISSION IN PERCENT OF Q IS                            .31406170-2
AND THE RESULTANT LOSS IN PERCENT OF Q IS                 .95655622+0
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4.2 CYLINDRICAL CASE

4.2.1 LIST OF THE PROGRAM

```
      IMPLICIT DOUBLE PRECISION(A-H,O-Z)
      DIMENSION V(6),T(14),EH(6),AW(8),RZZ(5),KA(4)
      ALIM(RZABCD,DLRDP)=RZABCD/CX+(DLRDP-RZABCD*S/CX)*S
      DATA V/2.D0,5.D0,1.D0,6.D0,1.D0,5.D0/           ,EH/1.D0,-6.D0,
     115.D0,-20.D0,15.D0,-6.D0/,CC/1.D-8/
    1 FORMAT(13A6,A2/7G8.4,2I3,G8.4)
    3 FORMAT(12G6.3)
    4 FORMAT(1H1,13A6,A2/' SOURCE DIMENSIONS'25X'/ MACROSCOPIC CROSS SEC
     1    '17X'/ INTEGRATION PARAMETERS'/' RA    ='G14.6,' TRB   ='G14.6,'
     1 / SIGAA='G14.6,' AK    ='G14.6,' / NZ    ='I6/' RB    ='G14.6,' TRC
     1 ='G14.6,' / SIGAF='G14.6,' F     ='G14.6,' / NPHI ='I6/' RC    ='G1
     14.6,' TRD   ='G14.6,' / SIGB ='G14.6,21X' / ET    ='G14.6/' RD    ='G
     114.6,' TLB   ='G14.6,' / SIGC ='G14.6,21X' / NCY  ='I6/' DLA   ='G14
     1.6,' TLC   ='G14.6,' / SIGD ='G14.6,21X' /'            /21X' TLD   =
     1'G14.6,' /'/' IN INCHES, OUTSIDE LENGTH  ='G12.6,' / COVER / WALL
     1 THICKNESS / LEFT END    / RIGHT END'/12X'OUTSIDE DIAMETER='G12.6,
     1' /   B'4XG12.6,4X'/'G12.6, ' /'G12.6/42X'/   C'4XG12.6,4X'/'G12.6
     1,' /'G12.6/42X'/   D'4XG12.6,4X'/'G12.6,' /'G12.6)
    5 FORMAT(3X/   46H   FOR A SINGLE NEUTRON STRIKING THE SOURCE.../
     120X35HTHE TRANSMISSION PROBABILITY IS     22XG16.8/
     120X35HTHE INTERACTION PROBABILITY IS      22XG16.8/
     120X35HTHE ABSORPTION PROBABILITY IS       22XG16.8/
     120X55HTHE 'A' MATERIAL ABSORPTION INTERACTION PROBABILITY IS 2XG16
     1.8/20X52HTHE 'A' MATERIAL FISSION INTERACTION PROBABILITY IS 5XG16
     1.8/20X44HTHE 'A' MATERIAL INTERACTION PROBABILITY IS 13XG16.8 /
     120X49HTHE CLADDING MATERIAL INTERACTION PROBABILITY IS  8XG16.8
     1  /16X'    NEUTRONS OUT PER NEUTRON STRIKE IS ' 22XG16.8
     1/20X'NEUTRONS LOST PER NEUTRON STRIKE   '22XG16.8,' .'
     1//46H   THE AVERAGE CROSS SECTION OF THE SOURCE IS G13.8,' CM**2.'
     1)
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    6 FORMAT(3X/   62H   FOR A SINGLE THERMAL NEUTRON STRIKING A SPHERE
     1OF RADIUS R=G13.8,' CM, WHICH JUST SURROUNDS THE SOURCE...'/
     120X40HTHE PROBABILITY OF MISSING THE SOURCE IS17XG16.8/
     120X'THE TRANSMISSION (INCLUDING MISS) PROBABILITY IS'9XG16.8/
     120X35HTHE INTERACTION PROBABILITY IS      22XG16.8/
     120X35HTHE ABSORPTION PROBABILITY IS       22XG16.8/
     120X44HTHE 'A' MATERIAL INTERACTION PROBABILITY IS 13XG16.8  /
     120X57HTHE PROBABILITY FOR A FISSION ABSORPTION IS            G16
     1.8/16X'    NEUTRONS OUT PER NEUTRON INTO SPHERE'21XG16.8,2H ./3X'T
     1HE ERROR IN THE TRANSMISSION PROBABILITY IS ESTIMATED TO BE 'G12.3
     1,' PER CENT.')
    7 FORMAT(3X/   44H   FOR A FLUX OF ONE NEUTRON PER CM**2-SEC.(G12.5,
     151H NEUTRONS PER SEC) STRIKING THE THE ABOVE SPHERE.../
     120X26HTHE TRANSMISSION IS        G16.8,20H NEUTRONS PER SECOND/
     120X26HTHE INTERACTION IS         G16.8,20H NEUTRONS PER SECOND/
     120X26HTHE ABSORPTION IS          G16.8,20H NEUTRONS PER SECOND/
     120X26HTHE FISSION EMISSION IS    G16.8,20H NEUTRONS PER SECOND/
     116X26HAND THE RESULTANT LOSS IS 4XG16.8,' NEUTRONS PER SECOND.')
    8 FORMAT(3X/   35H   IF THE MEASURED THERMAL FLUX IS G12.4, ' PER CE
     1NT OF Q PER CM**2 THEN...'/
     120X26HTHE TRANSMISSION IS        G16.8,14H PER CENT OF Q/
     120X26HTHE INTERACTION IS         G16.8,14H PER CENT OF Q/
     120X26HTHE ABSORPTION IS          G16.8,14H PER CENT OF Q/
     120X26HTHE FISSION EMISSION IS    G16.8,14H PER CENT OF Q/
     116X26HAND THE RESULTANT LOSS IS 4XG16.8,' PER CENT OF Q.')
    9 FORMAT(1X'THE NUMBER OF GRAND CALLS WAS'I8,', 2ND 3RD, AND 4TH ORD
     1ER COUNTS,KB, KC, AND KD WERE'I8,','I8,', AND'I8,/1X'THE LARGEST R
     1ELATIVE ERRORS, ERL1, ERL2, AND ERL3, ENCOUNTERED AFTER THE 1ST, 2
     1ND, AND 3RD SUBDIVISIONS, THE LARGEST   '/1X'SUBDIVISION,DZ, ALONG
     1 THE Z-AXIS, AND THE LARGEST 3RD ORDER SUBDIVISION, DWD, OF THE W-
     1AXIS, WERE'/ 5(4X,G16.8))
  100 KB=0                      'SECOND ORDER COUNT, STATEMENT 60+2, SUB IN
      KC=0                      'THIRD ORDER COUNT, STATEMENT 65+2 SUB IN
      KD=0                      'FOURTH ORDER COUNT, STATEMENT 121 SUB IN
      READ(5,1)T,SIGAA,SIGAF,SIGB,SIGC,SIGD,AK,F,NPHI,NZ,ET
      IF(ET.LT..1D-2)ET=.1D-1'ET=SPECIFIED RELATIVE ERROR IN SUBROUTINE IN
      READ(5,3) AA,AB,TRBI,TLBI,TSB,TRCI,TLCI,TSC,TRDI,TLDI,TSD
      IGRAND=0                  'COUNTER RESET FOR NUMBER OF SUB GRAND CALLS
      NCY=4                     'AA=OUTSIDE LENGTH IN INCHES
      IF(TSD.LT.CC.AND.TRDI.LT.CC.AND.TLDI.LT.CC) NCY=3 '  3 LAYERS
      IF(TSC.LT.CC.AND.TRCI.LT.CC.AND.TLCI.LT.CC) NCY=2 '   2 LAYERS
      IF(TSB.LT.CC.AND.TRBI.LT.CC.AND.TLBI.LT.CC) NCY=1 '  1  LAYER
      TRB=2.54D0*TRBI           'THE INPUT DIMENSIONS WERE IN INCHES
      TLB=2.54D0*TLBI           'F=THE NEUTRON FLUX IN PERCENT OF Q
      TRC=2.54D0*TRCI           'AK=NEUTRONS PER FISSION
      TLC=2.54D0*TLCI           'NCY=THE NUMBER OF DIFFERENT MATERIALS
      TRD=2.54D0*TRDI           'T=TITLE
      TLD=2.54D0*TLDI           'TRB= THICKNESS, RIGHT END, B CYLINDER
      RD=AB*2.54D0/2.D0         'OUTSIDE RADIUS
      RC=RD-2.54D0*TSD          'FOR ONE LAYER A,B,C, AND D DIMENS. ARE EQUAL
      RB=RC-2.54D0*TSC          'FOR TWO LAYERS B,C, AND D DIMENSIONS ARE EQUAL
      RA=RB-2.54D0*TSB          'FOR THREE LAYERS C AND D DIMENSIONS ARE EQUAL
      DLA=2.54D0*AA-TRB-TRC-TRD-TLB-TLC-TLD
      BLA=DLA/2.D0              'ONE-HALF OF INSIDE LENGTH
      DRB=BLA+TRB               'AB=OUTSIDE DIAMETER OF SOURCE IN INCHES
      DRC=DRB+TRC               'SIGAA=MACROSCOPIC ABSORBTION CROSS SECTION OF
      DRD=DRC+TRD               'MATERIAL A IN 'PER CM.'.
      DLB=BLA+TLB               'SIGAF= FISSION IN 'A' IN PER CM.
      DLC=DLB+TLC               'SIGB=ABSORPTION IN 'B'
      DLD=DLC+TLD               'TLD=THICKNESS, LEFT END, CYLINDER D
      RTR=DSQRT(DRD**2+RD**2)'RA=RADIUS OF CYLINDER A
      RTL=DSQRT(DLD**2+RD**2)
      R=RTL
      IF(RTR.GT.RTL) R=RTR      'R=SURROUNDING SPHERE RADIUS
      ELM=R*0.001D0             'SEE ST 210+4. SKIPS W INTEGR.IF LIM.TOO SMALL
      ERL1=0.D0                 'LARGEST ERROR IN THE 1ST, 2ND, AND 3RD
      ERL2=0.D0                 'ORDER WEDDLE DIVISIONS IN SUBROUTINE IN.
      ERL3=0.D0
      DWD=0.D0                  'LARGEST DIVISION OF W-AXIS, SUBROUTINE IN
      DZ=0.D0                   'LARGEST DIV. OF Z-AXIS IS MAX VALUE OF DA
      SIGAT=SIGAF+SIGAA         'SIGAT=TOTAL CROSS SECTION, 'A' MATERIAL
      RK=DLD                    'NZ=NO. OF DIVS. OF Z FOR WEDDLE INTEGRATION
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      IF(DRD.GT.DLD)RK=DRD    'RK=LARGER OF DRD AND DLD
      U=DATAN2(RK,RD)         'ARCTAN(RK/RD)
      AG=U*(R**2)
      ARE=RD*RK
      AD=AG-ARE               'AD= CROSS SECTIONAL AREA OF SPHERE ABOVE
      NPHIP=6*NPHI+1          'CYLINDER AND NOT IN THE Z INTEGRAL RANGE.
      VKB=6.D0*NPHI           'VKB= THE NUMBER OF PHI INTERVALS PER 90 DEGR
      DB=1.5707963D0/VKB      'DB= PHI INTERVAL IN RADIANS
      WRITE(6,4)T,RA,TRB,SIGAA,AK,NZ,RB,TRC,SIGAF,F,NPHI ,RC,TRD,SIGB,ET
     1    ,RD,TLB,SIGC,NCY,DLA,TLC,SIGD,TLD,AA,AB,TSB,TLBI,TRBI,TSC,TLCI,
     1TRCI,TSD,TLDI,TRDI
      RZZ(1)=0.D0
      RZZ(2)=RA
      RZZ(3)=RB
      RZZ(4)=RC
      RZZ(5)=RD
      EJI=0.D0
      EKI=0.D0
      STI=0.D0
      SFI=0.D0
      SMI=0.D0
      ANZ=NZ/RD               'NZ=APPROXIMATE NO. OF Z INTERVALS
      KA(1)=RA*ANZ+.5D0       'THIS IS TO SUBDIVIDE THE Z-LAYERS IN
      KA(2)=(RB-RA)*ANZ+.5D0  'PROPORTION TO THE THICKNESS OF EACH LAYER
      KA(3)=(RC-RB)*ANZ+.5D0  'FROM ZERO TO RA, FROM RA TO RB, FROM RB
      KA(4)=(RD-RC)*ANZ+.5D0  'TO RC, AND FROM RC TO RD. Z-INTEGRATION WILL
      DO 281 IAR=1,NCY        'ALWAYS OCCUR AT LAYER BOUNDARIES AND AVOID
      IF(KA(IAR).LT.1)KA(IAR)=1 'DISCONTINUITIES.
      IF((RZZ(IAR+1)-RZZ(IAR)).LT.ELM)GO TO 281'IN CASE OF ZERO SIDEWALL
      DA=(RZZ(IAR+1)-RZZ(IAR))/(6.D0*KA(IAR))
      IF(DA.GT.DZ)DZ=DA       'LARGEST DIVISION OF Z-AXIS
      NZP=6*KA(IAR)+1         'THE NUMBER OF Z STEPS
      IVZ=2                   'IVZ= Z INTEGRATION WEDDLE RULE COEF INDICATOR
      SPZT=0.D0               'Z PARTIAL SUM RESET OF TRANSMISSION,
      SPZM=0.D0               'MULTIPLICATION, AND
      SPZMIS=0.D0             'MISSES
      EJ=0.D0                 'ERROR RESET FOR Z INTEGRAND IN DO LOOP 280
      EK=0.D0                 'ERROR RESET FOR Z INTEGRAL IN DO LOOP 280
      TEK=0.D0                'TEMPORARY EK
      DO 280  I=1,NZP         'BEGIN.Z INTEGR.FROM 0-RA-RB-RC-RD
      IVP=2  'ANGLE INTEGRATION WEDDLE RULE  COEFFICIENT INDICATOR.
      SPPT=0.D0               'ANGLE PARTIAL SUM RESET OF TRANSMISSION,
      SPPM=0.D0               'MULTIPLICATION, AND
      SPPMIS=0.D0             'MISSES
      EF=0.D0                 'ERROR RESET FOR PHI INTEGRAND IN DO LOOP 250
      EG=0.D0                 'ERROR RESET FOR PHI INTEGRAL IN DO LOOP 250
      TEG=0.D0                'TEMPORARY EG
      Z=DA*DBLE (I-1)+RZZ(IAR)
      RZ=DSQRT(R**2-Z**2)
      RZD=0.D0
      RZC=0.D0
      RZB=0.D0
      RZA=0.D0
      GO TO (173,172,171,170),NCY
  170 IF(RD.GT.Z)RZD=DSQRT(RD**2-Z**2)
  171 IF(RC.GT.Z) RZC=DSQRT(RC**2-Z**2) 'HIT C LAYER, IF THERE IS ONE
  172 IF(RB.GT.Z) RZB=DSQRT(RB**2-Z**2) 'HIT B LAYER, IF THERE IS ONE
  173 IF(RA.GT.Z) RZA=DSQRT(RA**2-Z**2) 'HIT A LAYER
      DO 250 J=1,NPHIP        'BEGINNING OF PHI INTEGRATION FOR PHI FROM
      PHI=DB*DBLE(J-1)        '0 TO 90 DEGREES ONLY. PHI IS THE ANGLE
      S=DSIN(PHI)             'BETWEEN THE Y-AXIS AND W-AXIS
      CX=DCOS(PHI)
      SPWT=0.D0               'W PARTIAL SUM RESET OF TRANSMISSION,
      SPWM=0.D0               'AND MULTIPLICATION
      IFLAG=3                 'PHI NOT EQUAL TO 0 OR 90 DEGREES
      IF(J.EQ.1) GO TO 180
      IF(J.EQ.NPHIP) GO TO 185
      GO TO 200
  180 IFLAG=1                 'PHI=0 DEGREES
      GO TO(184,183,182,181),NCY
  181 AW(1)=-RZD               'W(1) IS THE POSITION ALONG THE W-AXIS THAT A
      AW(8)=RZD                'NEUTRON, INCIDENT PERPENDICULARLY FROM THE
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182 AW(2)=-RZC              'RIGHT OR LEFT SIDE OF THE W-AXIS WILL JUST
    AW(7)=RZC               'STRIKE THE LEFT OUTER CORNER OF THE D
183 AW(3)=-RZB              'MATERIAL. AW(2), AW(3) , AND AW(4) WILL JUST
    AW(6)=RZB               'STRIKE THE C,B, AND A LAYERS, RESPECTIVELY.
184 AW(4)=-RZA              'AW(5), AW(6), AW(7), AND AW(8) WILL JUST
    AW(5)=RZA               'STRIKE THE  UPPER RIGHT OUTER CORNER OF THE
    GO TO 210               'A, B, C, AND D MATERIALS, RESPECTIVELY.  THIS
185 IFLAG=2'PHI=90 DEGREES. 'AVOIDS DISCONTINUITIES IN THE DERIVATIVES FOR
    GO TO(189,188,187,186),NCY'THE WEDDLE RULE.
186 AW(1)=-DLD
    AW(8)=DRD
187 AW(2)=-DLC
    AW(7)=DRC
188 AW(3)=-DLB
    AW(6)=DRB
189 AW(4)=-BLA
    AW(5)=BLA
    GO TO 210
200 GO TO(204,203,202,201),NCY
201 AW(1)=-ALIM(RZD,DLD)
    AW(8)=ALIM(RZD,DRD)
202 AW(2)=-ALIM(RZC,DLC)
    AW(7)=ALIM(RZC,DRC)
203 AW(3)=-ALIM(RZB,DLB)
    AW(6)=ALIM(RZB,DRB)
204 AW(4)=-ALIM(RZA,BLA)
    AW(5)=-AW(4)
210 LL=5-NCY
    LU=3+NCY
    DELW=AW(LU+1)-AW(LL)    'RQ=2* THE PORTION OF THE SPHERE CUT BEYOND
    RQ=2.D0*(2.D0*RZ-DELW)  'CYLINDER BECAUSE OF INCIDENCE FROM TWO
    IF(DELW.LT.ELM)GO TO 231                   'DIRECTIONS
    EZ=0.D0
    DO 230 I2=LL,LU
    CALL IN(AW(I2),AW(I2+1),E,SPT,SPM)
    EZ=EZ+E
    SPWT=SPWT+SPT
    SPWM=SPWM+SPM
230 CONTINUE                'END OF W INTEGRAL DO LOOP, WHICH BECOMES
231 GO TO(240,240,235),IFLAG'INTEGRAND FOR PHI INTGR. WITH ABS. ERROR EZ
235 SPPT=SPPT+V(IVP)*SPWT
    SPPM=SPPM+V(IVP)*SPWM
    SPPMIS=SPPMIS+V(IVP)*RQ 'MISS CYLINDER CALCULATION
    EF=EF+(V(IVP)* EZ)**2   'ERROR**2  IN PHI IN. DUE TO INTEGRND. UNCRT.
    TEG=TEG+EH(IVP)*SPWT'SUM ABSOLUTE VALUE OF ERROR EACH 6 SUBDIVISIONS
    IVP=IVP+1               'EH(IVP)=WEDDLE RULE ERROR CONSTANTS
    IF(IVP.GE.7) IVP=1
    IF(IVP.EQ.2)GO TO 239
    GO TO 250
239 EG=EG+DABS(TEG)         'EG=ERROR IN PHI INTEGRAL DUE TO WEDDLE RULE
    TEG=SPWT
    GO TO 250
240 SPPT=SPPT+SPWT          'COEFF. FOR FIRST AND LAST TERMS IS ONE
    SPPM=SPPM+SPWM
    SPPMIS=SPPMIS+RQ
    EG=EG+SPWT
    EF=EF+EZ**2
250 CONTINUE                'END OF PHI INTEGRAL DO LOOP, WHICH BECOMES
    SPPT=SPPT*0.3D0*DB      'INTEGRAND FOR THE Z INTEGRAL
    SPPM=SPPM*0.3D0*DB      'THIS IS THE WEDDLE RULE NORMALIZATION
    SPPMIS=SPPMIS*0.3D0*DB  'FOR THE PHI PART
    EF=.3D0*DB*DSQRT(EF)
    EG=EG*DB/140.D0
    EI=DSQRT(EG**2+EF**2)   'EI=TOTAL ABS. ERROR ESTIM. FOR PHI INTEGRAL
    IF(I.EQ.1.OR.I.EQ.NZP) GO TO 270
    SPZT=SPZT+V(IVZ)*SPPT
    SPZM=SPZM+V(IVZ)*SPPM
    SPZMIS=SPZMIS+SPPMIS*V(IVZ)
    EJ=EJ+(V(IVZ)*EI)**2    'ERROR**2 DUE TO INTEGRAND UNCERT. IN Z INTEG.
    TEK=TEK+EH(IVZ)*SPPT'SUM ABSOLUTE VALUE OF ERROR EACH 6 SUBDIVISIONS
    IVZ=IVZ+1
    IF(IVZ.GE.7) IVZ=1
```

```
      IF(IVZ.EQ.2)GO TO 269
      GO TO 280
  269 EK=EK+DABS(TEK)         'EK=ERROR IN Z INTEGRAL DUE TO WEDDLE RULE
      TEK=SPPT
      GO TO 280
  270 SPZT=SPZT+SPPT          'COEFF. OF FIRST AND LAST TERMS IS 1.
      SPZM=SPZM+SPPM
      SPZMIS=SPZMIS+SPPMIS
      EJ=EJ+EI**2
      EK=EK+SPPT
  280 CONTINUE
      EJI=EJI+.3D0*DA*DSQRT(EJ)' Z NORMALIZATION, WEDDLE RULE
      EKI=EKI+EK*DA/140.D0
      STI=.3D0*DA*SPZT+STI   'TRANSMISSION INTEGRAL
      SFI=.3D0*DA*SPZM+SFI   'FISSION INTEGRAL
      SMI=.3D0*DA*SPZMIS+SMI 'MISSES INTEGRAL
  281 CONTINUE               'END OF Z INTEGRAL
      AR  =3.1416D0*(R**2)   'SPHERE CROSS SECTION
      VNORM=2.D0/(3.1416D0*AR) 'UNIT NORMALIZES THE INTEGRALS
      EL=DSQRT(EJI**2+EKI**2)
      STIS=STI*VNORM
      SAFIS=SFI*VNORM         'INTERACTION IN 'A' MATERIAL
      IF(SIGAT.LT.1.D-4)GO TO 282
      SFIS=SAFIS*SIGAF/SIGAT 'FISSION PART OF 'A' INTERACTION
      GO TO 283
  282 SFIS=0.D0
  283 SMIS=SMI*VNORM
      SLM=2.D0*AD/AR          'UNIT NORMALIZED, LUMP MISS PART
      SNOS=SFIS*AK+SLM+SMIS+STIS  'NEUTRONS OUT PER NEUTRON HITTING SPHERE
      FMIS=SLM+SMIS           'SINGLE NEUTRON MISS PROBABILITY
      ACROS=AR*(1.D0-FMIS)    'AVERAGE SOURCE CROSS SECTION
      FTRANS=SLM+STIS+SMIS    'SINGLE NEUTRON TRANSMISSION PROBABILITY
      EM=EL*1.D2*VNORM/FTRANS'EM=ESTIMATE OF PER CENT ERROR IN FTRANS
      FINT=1.D0-FTRANS        'SINGLE NEUTRON INTERACTION PROBABILITY
      FMUL=SFIS               'SINGLE NEUTRON FISSION PROBABILITY
      FABS=FINT-FMUL          'SINGLE NEUTRON ABSORPTION PROBABILITY
      FNO=FMUL*AK+FTRANS      'NEUTRONS OUT PER NEUTRON STRIKING SPHERE
      FTSOU=(FTRANS-FMIS)*AR/ACROS 'SINGLE STRIKE TRANSMISSION PROBABILITY
      FINSOU=1.D0-FTSOU       'SINGLE STRIKE INTERACTION PROBABILITY
      FSAFIS=SAFIS/(1.D0-FMIS)'SINGLE STRIKE 'A' INTERACTION PROBABILITY
      FMULSO=FMUL/(1.D0-FMIS)'SINGLE STRIKE FISSION PROBABILITY
      AAIP=FSAFIS-FMULSO      'SINGLE STRIKE 'A' ABSORPTION INTERACT. PROBAB.
      BCDIP=FINSOU-FSAFIS     'SINGLE STRIKE CLADDING  INTERACTION PROBABIL.
      FABSOU=FINSOU-FMULSO    'SINGLE STRIKE ABSORPTION PROBABILITY
      FOUTSO=FTSOU+FMULSO*AK 'NEUTRONS OUT PER NEUTRON STRIKING SOURCE
      FNOUT=1.D0-FOUTSO       'RESULTANT LOSS
      FPT=FTRANS*AR           'UNIT FLUX TRANSMISSION
      FPI=FINT*AR             'UNIT FLUX INTERACTION
      FPA=FABS*AR             'UNIT FLUX ABSORPTION
      FPF=FMUL*AR*AK          'UNIT FLUX FISSION
      FNLS=FPI-FPF            'UNIT FLUX, NET LOSS
      FQT=F*FPT               'PERCENT Q OF TRANSMISSION
      FQI=F*FPI               'PERCENT Q INTERACTION
      FQA=F*FPA               'PERCENT Q ABSORPTION
      FQF=F*FPF               'PERCENT Q FISSION GAIN
      FQL=F*FNLS              'PERCENT Q NET LOSS
      WRITE(6,9)IGRAND,KB,KC,KD,ERL1,ERL2,ERL3,DZ,DWD
      WRITE(6,5) FTSOU,FINSOU,FABSOU,AAIP,FMULSO,FSAFIS,BCDIP,FOUTSO,
     1FNOUT,ACROS
      WRITE(6,6)R,FMIS,FTRANS,FINT,FABS,SAFIS,FMUL,FNO,EM
      WRITE(6,7)AR,FPT,FPI,FPA,FPF,FNLS
      WRITE(6,8)F,FQT,FQI,FQA,FQF,FQL
      GO TO 100

      SUBROUTINE IN(FLL,FUL,E,ANS,QE)  'TO GIVE THE WEDDLE RULE INTEGRAL TO
      DW1=(FUL-FLL)/36.D0             'A SPECIFIED RELATIVE ERROR ET.
      DW2=DW1/6.D0            'IF THE SUBDIVISION ALONG THE W-AXIS IS 6
      IF(DW2.GT.DA)GO TO 65  'TIMES LARGER THAN IN THE Z-DIRECTION, THEN
      IF(DW1.GT.DA)GO TO 60  'IT IS AUTOMATICALLY DIVIDED DOWN BY 6 OR 36
      CALL PCE(FLL,FUL,ANS,AE,QE)
      IF(ANS.LT.1.D-4)GO TO 50
      EE=DABS(AE/ANS)
```

```
      IF(EE.GT.ERL1)ERL1=EE  'ERL1 IS LARGEST ERROR IN 1ST ORDER WEDDLE DIV.
      IF(EE.GT.ET)GO TO 60             'FLL=LOWER LIMIT
      E=AE                             'FUL=UPPER LIMIT
      RETURN                           'ANS=INTEGRAL
   50 E=0.D0
      RETURN
   60 ADIV=6.D0
      IA1=6
      KB=KB+1                          'RESET COUNTER IN STATEMENT 100
      GO TO 70
   65 ADIV=36.D0
      IA1=36
      KC=KC+1                          '3RD ORDER COUNT
      DW3=DW2/6.D0
      IF(DW3.GT.DWD)DWD=DW3  'DWD IS LARGEST 3RD ORDER SUBDIV. OF W-AXIS
   70 ED=(FUL-FLL)/ADIV                'PCE RETURNS ABSOLUTE ERROR.
      P1I=0.D0
      QE=0.D0
      E=0.D0
      FL2=FLL
      FU2=FLL+ED
      DO 120 IA=1,IA1
      CALL PCE(FL2,FU2,PI,PEI,QF)      '2ND ORDER OR 3RD ORDER
      P1I=P1I+PI
      QE=QE+QF
      E=E+PEI
      FL2=FU2                          'INCREMENT LIMITS
  120 FU2=FU2+ED
      ANS=P1I
      IF(ANS.LT.1.D-4)GO TO 122
      EE=DABS(E/ANS)
      IF(IA1.EQ.36)GO TO 121
      IF(EE.GT.ERL2)ERL2=EE  'ERL2 IS LARGEST ERROR IN 2ND ORDER WEDDLE DIV.
      IF(EE.GT.ET)GO TO 65   '4TH ORDER COUNT,KD,USED HERE TO COUNT NO OF
      RETURN                 'TIMES 3RD ORDER WEDDLE ERROR FAILS TO
  121 IF(EE.GT.ET)KD=KD+1    'SATISFY RELATIVE ERROR ET IN INPUT
      IF(EE.GT.ERL3)ERL3=EE  'ERL3 IS LARGEST ERROR IN 3RD ORDER WEDDLE DIV.
      RETURN
  122 E=0.D0
      RETURN

      SUBROUTINE PCE(FL,FU,AI,AE,QC)'GIVES THE WEDDLE CORRECT SUM BETWEEN FL
      DIMENSION VII(7),VIE(7)          'AND FU USING 6 SEGMENTS.
      DATA VII/1.D0,5.D0,1.D0,6.D0,1.D0,5.D0,1.D0/,VIE/1.D0,-6.D0,15.D0,
     1-20.D0,15.D0,-6.D0,1.D0/,CA/3.D-1/, CB/.71428571D-2/
      EC=(FU-FL)/6.D0        'AI IS THE FIRST INTEGRAL, TRANSMISSION PART
      PI=0.D0                'QC IS THE SECOND INTEGRAL, MULTIPLICATION PART
      QB=0.D0                'AE IS THE ABSOLUTE ERROR IN THE FIRST INTEGRAL
      PE=0.D0                'EC IS THE INCREMENT SIZE
      X=FL                   'X IS THE INTEGRAND EVALUATION POINT
      DO 100 ID=1,7
      CALL GRAND(X,QA,QG)    'QA IS THE MULTIPLICATION PART, INTEGRAND
      PI=PI+QG*VII(ID)       'QG IS THE TRANSMISSION PART, INTEGRAND
      QB=QB+VII(ID)*QA       'PI IS THE TRANSMISSION PART PARTIAL SUM
      PE=PE+QG*VIE(ID)       'QB IS THE MULTIPLICATION PARTIAL SUM
  100 X=X+EC
      AI=CA*EC*PI            'PE IS THE ERROR PARTIAL SUM (6TH DIFFERENCE)
      AE=DABS(CB*EC*PE)      'QC IS THE MULTIPLICATION PART ANSWER
      QC=CA*EC*QB
      RETURN

      SUBROUTINE GRAND(W,DUMC,DUMD)
C  FINAL EVALUATION OF THE INTEGRAND, WHICH IS PUT INTO PCE
C CONTRIBUTIONS ARE ADDED FOR NEUTRONS INCIDENT ON SOURCE FROM OPPOSITE
C DIRECTIONS, SO THAT THE PHI INTEGRAL IS CARRIED OUT FROM 0 TO PI/2
C  INSTEAD OF FROM 0 TO PI.  DUMD=TRANSMISSION PART OF INTEGRAND.
      IGRAND=IGRAND+1
      DUMC=0.D0              'CONTAINS MULTIPLICATION PART OF INTEGRAND
      A=0.D0
      B=0.D0
      C=0.D0
      D=0.D0
```

```
      GO TO(90,60,30,10),NCY   'NCY=NUMBER OF LAYERS
   10 CALL C2(IB,DRD,DLD,RZD,W,XID,YID,D)
      GO TO (20,30),IB
   20 DUMD=2.D0                'TRANSMISSION. IT MISSES IN BOTH DIRECTIONS
      RETURN
   30 CALL C2(IB,DRC,DLC,RZC,W,XIC,YIC,C)
      IF(IB.EQ.2) GO TO 60
      IF(NCY.EQ.3) GO TO 20
      GO TO 130
   60 CALL C2(IB,DRB,DLB,RZB,W,XIB,YIB,B)
      IF(IB.EQ.2) GO TO 90
      IF(NCY.EQ.2) GO TO 20
      GO TO 130
   90 CALL C2(IB,BLA,BLA,RZA,W,XIA,YIA,A)
      IF(IB.EQ.2) GO TO 130
      IF(NCY.EQ.1) GO TO 20
  130 GO TO (170,160,150,140),NCY
  140 D=D-C                    'THE AMOUNT OF A,B,C, AND D MATERIAL
  150 C=C-B                    'TRAVERSED IS A,B,C, AND D, RESPECTIVELY.
  160 B=B-A
  170 DUM=-(SIGAT*A+SIGB*B+SIGC*C+SIGD*D)
      DUMD=2.D0*DEXP(DUM)      'TRANSMISSION
      IF(IB.EQ.1) RETURN       'DID NOT HIT A MATERIAL. NO FISSION
      DIDI=0.D0
      DIDO=0.D0
      DICI=0.D0
      DICO=0.D0
      DBPBA=-SIGAT*A
      DBPBE=(1.D0-DEXP(DBPBA))
      GO TO (220,210,200,190),NCY
  190 DIDI=DSQRT((XID-XIC)**2+(YID-YIC)**2) 'ENTRY THICK. FOR FISSION CALC.
      DIDO=D-DIDI                           'SAME, FROM OTHER SIDE
  200 DICI=DSQRT((XIC-XIB)**2+(YIC-YIB)**2)
      DICO=C-DICI
  210 DIBI=DSQRT((XIB-XIA)**2+(YIB-YIA)**2)
      DIBO=B-DIBI
      DUMBI=-(SIGD*DIDI+SIGC*DICI+SIGB*DIBI)
      DUMBO=-(SIGD*DIDO+SIGC*DICO+SIGB*DIBO)
      DUMCI=DEXP(DUMBI)                             'FISSION EFFECT
      DUMCO=DEXP(DUMBO)                             'FROM OPPOSITE SIDE
      DUMC=(DUMCI+DUMCO)*DBPBE
      RETURN
  220 DUMC=2.D0*DBPBE
      RETURN

      SUBROUTINE C2(IB,DR,DL,C2R,C2W,XI,YI,DTHRU)
      YPF(X)=C2W*CX-S*(X-C2W*S)/CX ' Y-COORDINATE, GIVEN X, PHI, AND C2W(=W)
      XPF(Y)=C2W*S+CX*(C2W*CX-Y)/S ' X-COORDINATE, GIVEN Y, PHI, AND C2W(=W)
      IF(IFLAG.EQ.1)GO TO 101
      IF(IFLAG.EQ.2)GO TO 102
      TR=YPF(DR)
      TL=YPF(-DL)
      DTHRU=0.D0
      IF(C2R.GT.TR) GO TO 30
   20 IB=1                     'NEUTRON MISSES CYLINDER.
      RETURN
   30 IF(-C2R.GE.TL) GO TO 20
      IB=2                     'NEUTRON HITS CYLINDER.
      IF(-C2R.GT.TR) GO TO 60
      XI=DR                    'NEUTRON HITS END
      YI=TR
      GO TO 70
   60 XI=XPF(-C2R)             'NEUTRON HITS BOTTOM.
      YI=-C2R
   70 IF(C2R.GT.TL) GO TO 90
      XO=XPF(C2R)              'NEUTRON EXITS FROM THE TOP.
      YO=C2R
      GO TO 100
   90 XO=-DL                   'NEUTRON EXITS FROM THE END
      YO=TL
  100 DTHRU=DSQRT((XI-XO)**2+(YI-YO)**2)
      RETURN
```

```
101 DTHRU=DR+DL
    XI=DR
    YI=C2W
    IB=2
    RETURN
102 DTHRU=2.D0*C2R
    XI=C2W
    YI=-C2R
    IB=2
    RETURN
    END
```

4.2.2 LIST OF THE INPUT DATA DECK

```
     TITLE CARD.        FORMAT(13A6,A2)
SOURCE M-621  PU-BE  80 GRAM, TA AND STAINLESS STEEL ENCAPSULATED         CARD 1
MACROSCOPIC CROSS SECTIONS AND INTEGRATION PARAMETERS.  FORMAT(7G8.4,2I3,G8.4)
12345678901234567890123456789012345678901234567890123456789012345678901234567890
1.865   4.305   1.16    .2811   .0      2.8     .122     10 10.01         CARD 2
SIGAA   SIGAF   SIGB    SIGC    SIGD    AK      F       N- NZ ET
I       I       I       I       I       I       I       PHII  I
I       I       I       I       I       I       I       I  I  MINIMUM FRACTION-
I       I       I       I       I       I       I       I  I  AL ERROR ALLOWED
I       I       I       I       I       I       I       I  I  IN SUBROUTINE PCE
I       I       I       I       I       I       I       I  1/6 NUMBER OF Z INTE-
I       I       I       I       I       I       I       I  GRATION STEPS
I       I       I       I       I       I       I       1/6 NUMBER OF PHI INTE-
I       I       I       I       I       I       I       GRATION STEPS
I       I       I       I       I       I       FLUX IN PERCENT OF SOURCE
I       I       I       I       I       I       STRENGTH PER CM**2
I       I       I       I       I       NUMBER OF NEUTRONS PER FISSION
I       I       I       I       ABSORPTION CROSS SECTION PER CM FOR D MATERIAL
I       I       I       ABSORPTION CROSS SECTION PER CM FOR C MATERIAL
I       I       ABSORPTION CROSS SECTION PER CM FOR B MATERIAL
I       FISSION CROSS SECTION PER CM FOR A MATERIAL
ABSORPTION CROSS SECTION PER CM FOR A MATERIAL

     SOURCE DIMENSIONS IN INCHES.   FORMAT(12G6.3)
12345678901234567890123456789012345678901234567890123456789012345678901234567890
2.72  1.310 .100  .250  .070  .250  .100  .03   .0    .0    .0            CARD 3
AA    AB    TRBI  TLBI  TSB   TRCI  TLCI  TSC   TRDI  TLDI  TSD
I     I     I     I     I     I     I     I     I     I     D SIDE WALL IN IN.
I     I     I     I     I     I     I     I     I     D LEFT END IN INCHES
I     I     I     I     I     I     I     I     D RIGHT END IN INCHES
I     I     I     I     I     I     I     C SIDE WALL IN INCHES
I     I     I     I     I     I     C LEFT END IN INCHES
I     I     I     I     I     C RIGHT END IN INCHES
I     I     I     I     B SIDE WALL IN INCHES
I     I     I     B LEFT END IN INCHES
I     I     B RIGHT END IN INCHES
I     OUTSIDE DIAMETER IN INCHES
OUTSIDE LENGTH IN INCHES
```

4.2.3 SIMPLIFIED SAMPLE OUTPUT

```
CYLINDRICAL PU-BE M-621 SOURCE. ENCAPSULATED IN TANTALUM AND ST. STEEL.
SOURCE DIMENSION INPUT IN INCHES, L = 2.72, OD = 1.31,  B WALL = 0.07,
B ENDS = 0.25 AND 0.1,  C WALL = 0.03,  C ENDS = 0.1 AND 0.25.
MACROSCOPIC CROSS SECTIONS PER CM
SIGAA= 1.865   , SIGAF=  4.305    ,  SIGB=  1.16    ,  SIGC=  0.2811
NEUTRONS PER FISSION = 2.80.    NZ=10.    NPHI=10.
FOR A SINGLE NEUTRON STRIKING THE SOURCE...
THE TRANSMISSION PROBABILITY IS                              .18433302+0
THE INTERACTION PROBABILITY IS                               .81566698+0
THE ABSORPTION PROBABILITY IS                                .54509177+0
THE 'A' MATERIAL ABSORPTION INTERACTION PROBABILITY IS       .11721783+0
THE 'A' MATERIAL FISSION INTERACTION PROBABILITY IS          .27057522+0
THE 'A' MATERIAL INTERACTION PROBABILITY IS                  .38779305+0
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THE CLADDING MATERIAL INTERACTION PROBABILITY IS           .42787393+0
NEUTRONS OUT PER NEUTRON STRIKE IS                         .94194362+0
NEUTRONS LOST PER NEUTRON STRIKE IS                        .58056380+0
THE AVERAGE CROSS SECTION OF THE SOURCE IN CM**2 IS        .20170300+2
THE PERCENT ERROR IN THE TRANSMISSION PROBABILITY IS       .0162
IF THE MEASURED THERMAL FLUX IN PERCENT OF Q PER CM**2 IS .122
THE INTERACTION IN PERCENT OF Q IS                         .20071742+1
THE ABSORPTION IN PERCENT OF Q IS                          .13413491+1
THE FISSION IN PERCENT OF Q IS                             .18643104+1
AND THE RESULTANT LOSS IN PERCENT OF Q IS                  .14286378+0


CYLINDRICAL SOURCE PU-BE A. ENCAPSULATED IN NICKEL.
SOURCE DIMENSION INPUT IN INCHES, L = 1.031, OD = 1.031, B WALL = 0.128,
B ENDS = 0.112 AND 0.129.
MACROSCOPIC CROSS SECTIONS PER CM
SIGAA= 1.214   , SIGAF=  2.802    ,  SIGB=  0.4106.
NEUTRONS PER FISSION = 2.80.    NZ=15.   NPHI=15.
FOR A SINGLE NEUTRON STRIKING THE SOURCE...
THE TRANSMISSION PROBABILITY IS                            .31458511+0
THE INTERACTION PROBABILITY IS                             .68541489+0
THE ABSORPTION PROBABILITY IS                              .38173937+0
THE 'A' MATERIAL ABSORPTION INTERACTION PROBABILITY IS     .13157105+0
THE 'A' MATERIAL FISSION INTERACTION PROBABILITY IS        .30367551+0
THE 'A' MATERIAL INTERACTION PROBABILITY IS                .43524656+0
THE CLADDING MATERIAL INTERACTION PROBABILITY IS           .25016833+0
NEUTRONS OUT PER NEUTRON STRIKE IS                         .11648765+1
NEUTRONS LOST PER NEUTRON STRIKE IS                       -.16487655+0
THE AVERAGE CROSS SECTION OF THE SOURCE IN CM**2 IS        .77943026+1
THE PERCENT ERROR IN THE TRANSMISSION PROBABILITY IS       .0270
IF THE MEASURED THERMAL FLUX IN PERCENT OF Q PER CM**2 IS .122
THE INTERACTION IN PERCENT OF Q IS                         .65176438+0
THE ABSORPTION IN PERCENT OF Q IS                          .36299785+0
THE FISSION IN PERCENT OF Q IS                             .80854631+0
AND THE RESULTANT LOSS IN PERCENT OF Q IS                 -.15678192+0
CYLINDRICAL AM-BE SOURCE. ENCAPSULATED IN TANTALUM AND STAINLESS STEEL.
SOURCE DIMENSION INPUT IN INCHES, L = 1.355, OD = 1.355, B WALL = 0.07
B ENDS = 0.07,  C WALL = 0.03,  C ENDS = 0.05.
MACROSCOPIC CROSS SECTIONS PER CM
SIGAA= .1704   , SIGAF=  0.00082 ,  SIGB=  1.16    ,   SIGC=  0.281
NEUTRONS PER FISSION = 2.89.    NZ=10.   NPHI=10.
FOR A SINGLE NEUTRON STRIKING THE SOURCE...
THE TRANSMISSION PROBABILITY IS                            .40678453+0
THE INTERACTION PROBABILITY IS                             .59321547+0
THE ABSORPTION PROBABILITY IS                              .59253429+0
THE 'A' MATERIAL ABSORPTION INTERACTION PROBABILITY IS     .14155335+0
THE 'A' MATERIAL FISSION INTERACTION PROBABILITY IS        .68118396-3
THE 'A' MATERIAL INTERACTION PROBABILITY IS                .14223453+0
THE CLADDING MATERIAL INTERACTION PROBABILITY IS           .45098094+0
NEUTRONS OUT PER NEUTRON STRIKE IS                         .40875315+0
NEUTRONS LOST PER NEUTRON STRIKE IS                        .59124685+0
THE AVERAGE CROSS SECTION OF THE SOURCE IN CM**2 IS        .13463194+2
THE PERCENT ERROR IN THE TRANSMISSION PROBABILITY IS       .0356
IF THE MEASURED THERMAL FLUX IN PERCENT OF Q PER CM**2 IS .121
THE INTERACTION IN PERCENT OF Q IS                         .96637561+0
THE ABSORPTION IN PERCENT OF Q IS                          .96526593+0
THE FISSION IN PERCENT OF Q IS                             .32069762-2
AND THE RESULTANT LOSS IN PERCENT OF Q IS                  .96316863+0
```

Table 1.

Neutron Transmission Convergence Test for the Spherical Source Program

NY	Transmission	Estimated Error	Actual Error*
10	0.14653888	0.819%	1.205%
20	0.14542886	0.438%	0.438%
50	0.14495702	0.113%	0.113%
100	0.14485061	0.041%	0.039%
500	0.14479752	0.008%	0.002%
1000	0.14479397	0.004%	-

* Actual Error listed here assumes that transmission for NY = 1000 is correct.

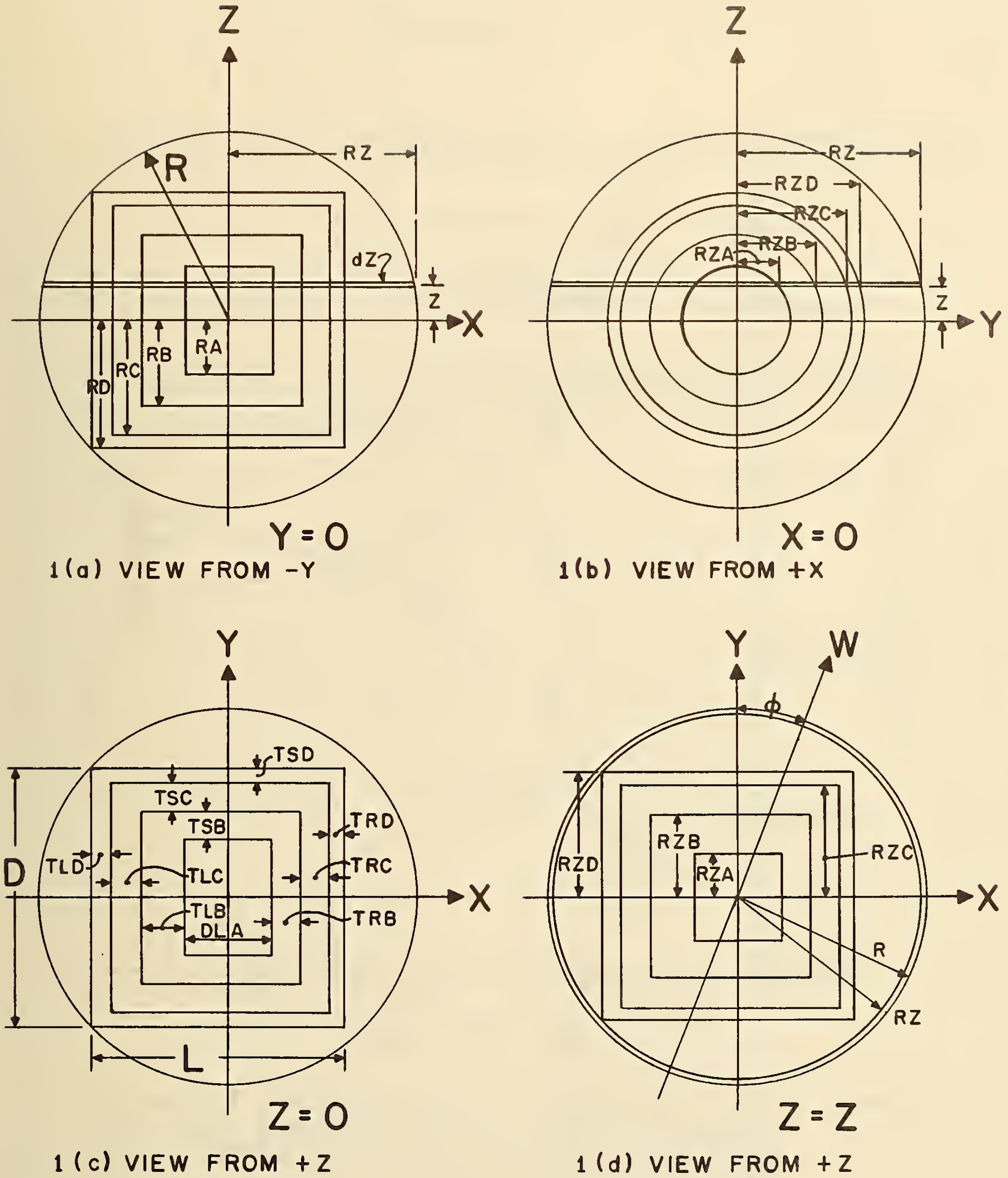

Figure 1. Diagram for the cylindrical source program. The cylindrical source is surrounded by an imaginary sphere with center on the X axis in the middle of the core material. Figure 1(d) shows the intersection of a plane parallel to the X-Y axis with the source at Z.

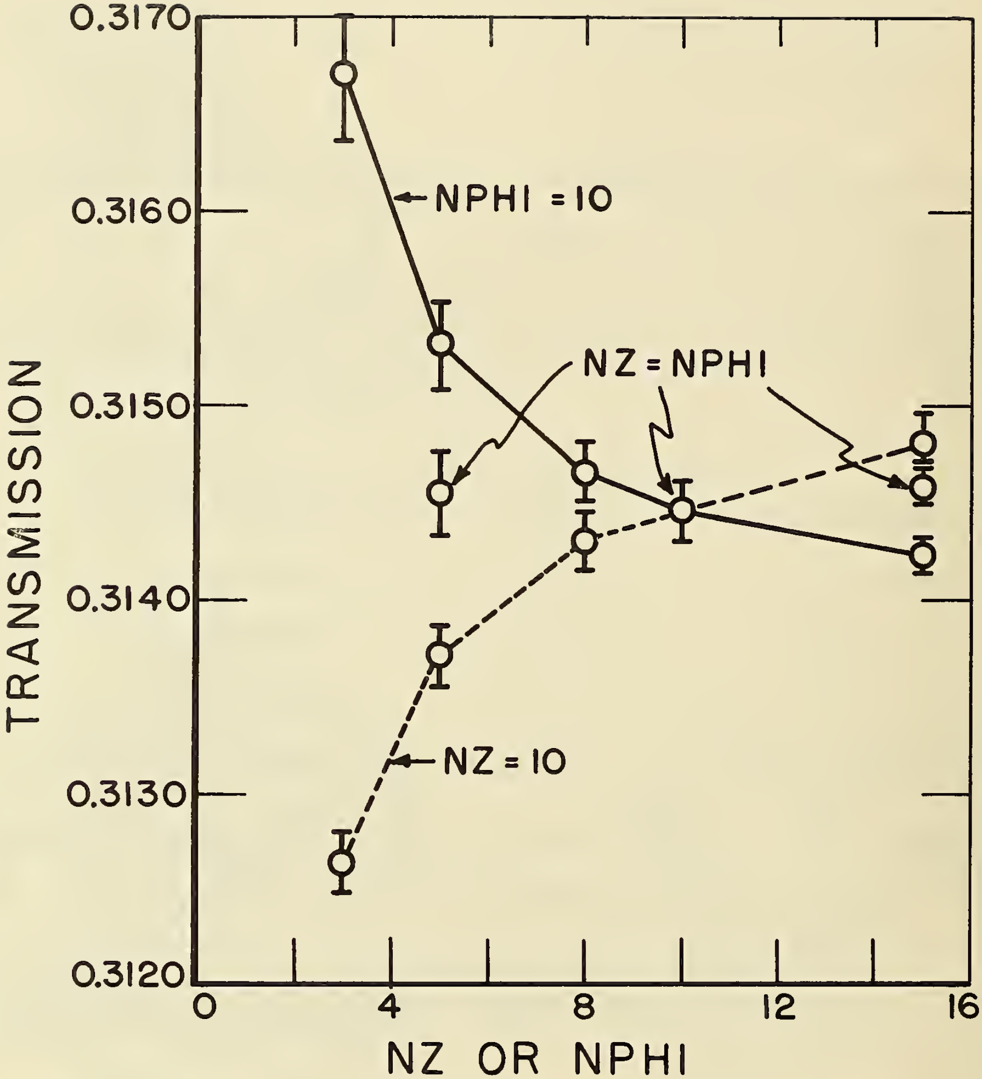

Figure 2. The calculated Transmission and Estimated Error for a nickel encapsulated Pu-Be cylindrical source for various specified subdivisions of the Z and φ integrals, NZ and NPHI. The solid line joins points for which NPHI equals 10 and the dashed line joins points for which NZ equals 10. Two points are shown for both NPHI and NZ equal to 5 and 15.

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16. ABSTRACT (A 200-word or less factual summary of most significant information. If document includes a significant bibliography or literature survey, mention it here.)

A computer code has been written in FORTRAN IV for the calculation of thermal neutron absorption in spherical and cylindrical neutron sources. The formalism of the calculation, the structure of the computer code, a listing of the code, and some sample results are presented. The comparison of the results of this calculation to experiment appears in Metrologia 7, No. 1, 34-38 (1971).

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