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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

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

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

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

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

$$P_L = P_I - kP_f = 1 - P_T - kP_f \quad (6)$$

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

$$P_T = \int_{-R}^R dz \int_0^{2\pi} d\varphi \int_{-r(z)}^{r(z)} dw P_T(z, w, \varphi) / 2\pi(\pi R^2) \quad (7)$$

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

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

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

$$P_F = \int_{-R}^R dz \int_0^{2\pi} d\varphi \int_{-r(z)}^{r(z)} dw P_f(z, w, \varphi) / 2\pi(\pi R^2) \quad (9)$$

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

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

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

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

$$\gamma = \Sigma_b b_1(z, w, \varphi) + \Sigma_c c_1(z, w, \varphi) + \Sigma_d d_1(z, w, \varphi),$$

where $b_1(z, w, \varphi)$ is the thickness of the "B" layer transversed passing into the "A" material and likewise for c_1 and d_1 ;

$e^{-\gamma}$ 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, NPFI, 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 va-

ries 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 0.1% 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 NPFI, respectively. A solid line joins those points for which NPFI 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 NPFI 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 NPFI 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. LISTING OF THE PROGRAMS

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.DO,5.DO,1.DO,6.DO,1.DO,5.DO/, EH/1.DO,-6.DO,
115.DO,-20.DO,15.DO,-6.DO/,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 ='16/' 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 ='16/ 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.')
```

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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.')
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9 FORMAT(1X'THE NUMBER OF GRAND CALLS WAS'18,' THE LARGEST SUBDIVIS
1ION OF THE RADIUS WAS 'G16.8)
```

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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.DO '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.DO 'LARGEST DIV. OF RADIUS, Y, IS MAX VALUE OF DA
SIGAT=SIGAF+SIGAA 'SIGAT=TOTAL CROSS SECTION, 'A' MATERIAL
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WRITE(6,4)T,RA,TB,SIGAA,AK,NY,RB,TC,SIGAF,F,RC,TD,SIGB,RD,SIGC, 64
1  NCY,SIGD,DIA,TBI,TCI,TDI 65
RZZ(1)=0.DO 66
RZZ(2)=RA 67
RZZ(3)=RB 68
RZZ(4)=RC 69
RZZ(5)=RD 70
EKI=0.DO 71
STI=0.DO 72
SYD=0.DO 73
SYC=0.DO 74
SYB=0.DO 75
SYA=0.DO 76
ANY=NY/RD 'NY=APPROXIMATE NO. OF Y INTERVALS 77
KA(1)=RA*ANY+.5DO 'THIS IS TO SUBDIVIDE THE Y-LAYERS IN 78
KA(2)=(RB-RA)*ANY+.5DO 'PROPORTION TO THE THICKNESS OF EACH LAYER 79
KA(3)=(RC-RB)*ANY+.5DO 'FROM ZERO TO RA, FROM RA TO RB, FROM RB 80
KA(4)=(RD-RC)*ANY+.5DO 'TO RC, AND FROM RC TO RD. Y-INTEGRATION WILL 81
DO 281 IAR=1,NCY 'ALWAYS OCCUR AT LAYER BOUNDARIES AND AVOID 82
IF(KA(IAR).LT.1)KA(IAR)=1 'DISCONTINUITIES. 83
DA=(RZZ(IAR+1)-RZZ(IAR))/(6.DO*KA(IAR)) 84
IF(DA.GT.DY)DY=DA 'LARGEST DIVISION OF RADIUS, Y 85
NYP=6*KA(IAR)+1 'THE NUMBER OF PTS OF INTEGR. IN EACH LAYER 86
IVY=2 'IVY= Y INTEGRATION WEDDLE RULE COEF INDICATOR 87
SPYT=0.DO 'Y PARTIAL SUM RESET OF TRANSMISSION AND 88
SPYD=0.DO 'INTERACTION IN 'D' MATERIAL 89
SPYC=0.DO 'INTERACTION IN 'C' MATERIAL 90
SPYB=0.DO 'INTERACTION IN 'B' MATERIAL 91
SPYA=0.DO 'INTERACTION IN 'A' MATERIAL 92
EK=0.DO 'ERROR RESET FOR Y INTEGRAL IN DO LOOP 280 93
TEK=0.DO 'TEMPORARY EK 94
DO 280 I=1,NYP 'BEGIN Y INTEGR.FR 0-RA-RB-RC-RD 95
Y=DA*DBLE (I-1)+RZZ(IAR) 96
RDY=0.DO 97
RCY=0.DO 98
RBY=0.DO 99
RAY=0.DO 100
GO TO (173,172,171,170),NCY 101
170 IF(RD.GT.Y)RDY=DSQRT(RD**2-Y**2) 102
171 IF(RC.GT.Y)RCY=DSQRT(RC**2-Y**2) 'HIT C LAYER, IF THERE IS ONE 103
172 IF(RB.GT.Y)RBY=DSQRT(RB**2-Y**2) 'HIT B LAYER, IF THERE IS ONE 104
173 IF(RA.GT.Y)RAY=DSQRT(RA**2-Y**2) 'HIT A LAYER 105
CALL GRAND(Y,SPIA,SPIB,SPIC,SPID,SPT) 106
IF(I.EQ.1.OR.I.EQ.NYP)GO TO 270 107
GO TO(180,181,182,183),NCY 108
183 SPYD=SPYD+V(IVY)*SPID 109
182 SPYC=SPYC+V(IVY)*SPIC 110
181 SPYB=SPYB+V(IVY)*SPIB 111
180 SPYA=SPYA+V(IVY)*SPIA 112
SPYT=SPYT+V(IVY)*SPT 113
TEK=TEK+EH(IVY)*SPT 'SUM ABSOLUTE VALUE OF ERROR EACH 6 SUBINTERVALS 114
IVY=IVY+1 115
IF(IVY.GE.7)IVY=1 116
IF(IVY.EQ.2)GO TO 269 117
GO TO 280 118
269 EK=EK+DABS(TEK) 'EK=ERROR IN Y INTEGRAL DUE TO WEDDLE RULE 119
TEK=SPT 120
GO TO 280 121
270 GO TO(271,272,273,274),NCY'COEFF. OF FIRST AND LAST TERMS IS 1. 122
274 SPYD=SPYD+SPID 123
273 SPYC=SPYC+SPIC 124
272 SPYB=SPYB+SPIB 125
271 SPYA=SPYA+SPIA 126
SPYT=SPYT+SPT 127
EK=EK+SPT 128
280 CONTINUE 129
EKI=EKI+EK*DA/140.DO 'Y NORMALIZATION, WEDDLE RULE 130
STI=.3DO*DA*SPYT+STI 'TRANSMISSION INTEGRAL 131
GO TO(290,291,292,293),NCY 132
293 SYD=SYD+.3DO*DA*SPYD 133
292 SYC=SYC+.3DO*DA*SPYC 134
291 SYB=SYB+.3DO*DA*SPYB 135

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290	SYA=SYA+0.3D0*DA*SPYA		136
281	CONTINUE	'END OF Y INTEGRAL	137
	VNORM=2.D0/RD**2	'VNORM=2/RD**2 AND UNIT NORMALIZES	138
	STIS=STI*VNORM	'THE INTEGRALS	139
	SAFIS=SYA*VNORM	'INTERACTION IN 'A' MATERIAL	140
	SBFIS=SYB*VNORM	'INTERACTION IN 'B' MATERIAL	141
	SCFIS=SYC*VNORM	'INTERACTION IN 'C' MATERIAL	142
	SDFIS=SYD*VNORM	'INTERACTION IN 'D' MATERIAL	143
	IF(SIGAT.LT.1.D-4)GO TO 282		144
	SFIS=SAFIS*SIGAF/SIGAT	'FISSION PART OF 'A' INTERACTION	145
	GO TO 283		146
282	SFIS=0.D0		147
283	SNOS=SFIS*AK+STIS	'NEUTRONS OUT PER NEUTRON HITTING SPHERE	148
	ACROS=3.1416D0*RD**2	'SOURCE CROSS SECTION	149
	EM=EKI*1.D2/STI	'EM=ESTIMATE OF PER CENT ERROR IN STI	150
	FTSOU=STIS	'SINGLE STRIKE TRANSMISSION PROBABILITY	151
	FINSOU=1.D0-FTSOU	'SINGLE STRIKE INTERACTION PROBABILITY	152
	FMULSO=SFIS	'SINGLE STRIKE FISSION PROBABILITY	153
	FABSOU=FINSOU-FMULSO	'SINGLE STRIKE ABSORPTION PROBABILITY	154
	FOUTSO=FTSOU+FMULSO*AK	'NEUTRONS OUT PER NEUTRON STRIKING SOURCE	155
	FNOUT=1.D0-FOUTSO	'RESULTANT LOSS	156
	FPT=STIS*ACROS	'UNIT FLUX TRANSMISSION	157
	FPI=FINSOU*ACROS	'UNIT FLUX INTERACTION	158
	FPA=FABSOU*ACROS	'UNIT FLUX ABSORPTION	159
	FPF=SFIS*AK*ACROS	'UNIT FLUX FISSION	160
	FNLS=FPI-FPF	'UNIT FLUX, NET LOSS	161
	FQT=F*FPT	'PERCENT Q OF TRANSMISSION	162
	FQI=F*FPI	'PERCENT Q INTERACTION	163
	FQA=F*FPA	'PERCENT Q ABSORPTION	164
	FQF=F*FPF	'PERCENT Q FISSION GAIN	165
	FQL=F*FNLS	'PERCENT Q NET LOSS	166
	WRITE(6,9)IGRAND,DY		167
	WRITE(6,5)FTSOU,FINSOU,FABSOU,SAFIS,SBFIS,SCFIS,SDFIS,FMULSO,		168
	1FOUTSO,FNOUT,ACROS,EM		169
	WRITE(6,7)ACROS,FPT,FPI,FPA,FPF,FNLS		170
	WRITE(6,8)F,FQT,FQI,FQA,FQF,FQL		171
	GO TO 100		172
			173
	SUBROUTINE GRAND(W,DUMA,DUMB,DUMC,DUMD,DMDA)		174
	IGRAND=IGRAND+1	'COUNT OF SUBROUTINE CALL	175
	DUMA=0.D0	'CONTAINS MULTIPLICATION PART OF INTEGRAND	176
	DUMB=0.D0		177
	DUMC=0.D0		178
	DUMD=0.D0		179
	A=0.D0		180
	B=0.D0		181
	C=0.D0		182
	D=0.D0		183
	GO TO (370,360,350,340),NCY		184
340	D=(RDY-RCY)*2.D0	'THE AMOUNT OF A,B,C, AND D MATERIAL	185
350	C=(RCY-RBY)*2.D0	'TRAVERSED IS A,B,C, AND D, RESPECTIVELY.	186
360	B=(RBY-RAY)*2.D0		187
370	A=RAY*2.D0		188
	DMD=- (SIGAT*A+SIGB*B+SIGC*C+SIGD*D)		189
	DMDEX=DEXP(DMD)		190
	DMDA=W*DMDEX	'TRANSMISSION PART OF INTEGRAND	191
	DEXD=DEXP(-SIGD*D/2.D0)		192
	DEXC=DEXP(-SIGC*C/2.D0)		193
	DEXB=DEXP(-SIGB*B/2.D0)		194
	DEXA=DEXP(-SIGAT*A/2.D0)		195
	GO TO(400,401,402,403),NCY		196
403	DUMDA=W*(1-DMDEX)	'INTERACTION IN A,B,C,AND D MATERIAL	197
402	DUMCA=W*(DEXD-DMDEX/DEXD)	'INTERACTION IN A,B, AND C MATERIAL	198
401	DUMBA=W*(DEXD*DEXC-DMDEX/(DEXD*DEXC))	'INTERACTION IN A AND B MATERIAL	199
400	DUMA=W*(DEXD*DEXC*DEXB-DMDEX/(DEXD*DEXC*DEXB))	'INTERAC. IN A MAT.	200
	GO TO(410,411,412,413),NCY		201
413	DUMD=DUMDA-DUMCA	'INTERACTION IN D MATERIAL	202
412	DUMC=DUMCA-DUMBA	'INTERACTION IN C MATERIAL	203
411	DUMB=DUMBA-DUMA	'INTERACTION IN B MATERIAL	204
410	RETURN		205
	END		206


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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

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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

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IMPLICIT DOUBLE PRECISION(A-H,O-Z) 1
DIMENSION V(6),T(14),EH(6),AW(8),RZZ(5),KA(4) 2
ALIM(RZABCD,DLRDP)=RZABCD/CX+(DLRDP-RZABCD*S/CX)*S 3
DATA V/2.D0,5.D0,1.D0,6.D0,1.D0,5.D0/ ,EH/1.D0,-6.D0, 4
115.D0,-20.D0,15.D0,-6.D0/,CC/1.D-8/ 5
1 FORMAT(13A6,A2/7G8.4,2I3,G8.4) 6
3 FORMAT(12G6.3) 7
4 FORMAT(1H1,13A6,A2/' SOURCE DIMENSIONS'25X/' MACROSCOPIC CROSS SEC 8
1 '17X/' INTEGRATION PARAMETERS/' RA ='G14.6,' TRB ='G14.6,' 9
1 / SIGAA='G14.6,' AK ='G14.6,' / NZ ='I6/' RB ='G14.6,' TRC 10
1 ='G14.6,' / SIGAF='G14.6,' F ='G14.6,' / NPHI ='I6/' RC ='G1 11
14.6,' TRD ='G14.6,' / SIGB ='G14.6,21X' / ET ='G14.6/' RD ='G 12
114.6,' TLB ='G14.6,' / SIGC ='G14.6,21X' / NCY ='I6/' DLA ='G14 13
1.6,' TLC ='G14.6,' / SIGD ='G14.6,21X' / /21X' TLD = 14
1'G14.6,' /'/' IN INCHES, OUTSIDE LENGTH ='G12.6,' / COVER / WALL 15
1 THICKNESS / LEFT END / RIGHT END/12X'OUTSIDE DIAMETER='G12.6, 16
1' / B'4XG12.6,4X'/'G12.6,' /'G12.6/42X'/' C'4XG12.6,4X'/'G12.6 17
1,' /'G12.6/42X'/' D'4XG12.6,4X'/'G12.6,' /'G12.6) 18
5 FORMAT(3X/ 46H FOR A SINGLE NEUTRON STRIKING THE SOURCE.../ 19
12OX35HTHE TRANSMISSION PROBABILITY IS 22XG16.8/ 20
12OX35HTHE INTERACTION PROBABILITY IS 22XG16.8/ 21
12OX35HTHE ABSORPTION PROBABILITY IS 22XG16.8/ 22
12OX55HTHE 'A' MATERIAL ABSORPTION INTERACTION PROBABILITY IS 2XG16 23
1.8/20X52HTHE 'A' MATERIAL FISSION INTERACTION PROBABILITY IS 5XG16 24
1.8/20X44HTHE 'A' MATERIAL INTERACTION PROBABILITY IS 13XG16.8 / 25
12OX49HTHE CLADDING MATERIAL INTERACTION PROBABILITY IS 8XG16.8 26
1 /16X' NEUTRONS OUT PER NEUTRON STRIKE IS ' 22XG16.8 27
1/20X'NEUTRONS LOST PER NEUTRON STRIKE '22XG16.8,' . 28
1/46H THE AVERAGE CROSS SECTION OF THE SOURCE IS G13.8,' CM**2.' 29
1) 30

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IF (DRD.GT.DLD)RK=DRD      'RK=LARGER OF DRD AND DLD      103
U=DATAN2(RK,RD)            'ARCTAN(RK/RD)                104
AG=U*(R**2)                '                                105
ARE=RD*RK                  '                                106
AD=AG-ARE                  'AD= CROSS SECTIONAL AREA OF SPHERE ABOVE  107
NPHIP=6*NPHI+1            'CYLINDER AND NOT IN THE Z INTEGRAL RANGE.  108
VKB=6.D0*NPHI             'VKB= THE NUMBER OF PHI INTERVALS PER 90 DEGR  109
DB=1.5707963D0/VKB       'DB= PHI INTERVAL IN RADIANS                110
WRITE(6,4)T,RA,TRB,SIGAA,AK,NZ,RB,TRC,SIGAF,F,NPHI ,RC,TRD,SIGB,ET  111
1 ,RD,TLB,SIGC,NCY,DLA,TLC,SIGD,TLD,AA,AB,TSB,TLBI,TRBI,TSC,TLCI,  112
]TRCI,TSD,TLDI,TRDI      '                                113
RZZ(1)=0.D0               '                                114
RZZ(2)=RA                 '                                115
RZZ(3)=RB                 '                                116
RZZ(4)=RC                 '                                117
RZZ(5)=RD                 '                                118
EJI=0.D0                  '                                119
EKI=0.D0                  '                                120
STI=0.D0                  '                                121
SFI=0.D0                  '                                122
SMI=0.D0                  '                                123
ANZ=NZ/RD                 'NZ=APPROXIMATE NO. OF Z INTERVALS          124
KA(1)=RA*ANZ+.5D0        'THIS IS TO SUBDIVIDE THE Z-LAYERS IN      125
KA(2)=(RB-RA)*ANZ+.5D0   'PROPORTION TO THE THICKNESS OF EACH LAYER  126
KA(3)=(RC-RB)*ANZ+.5D0   'FROM ZERO TO RA, FROM RA TO RB, FROM RB   127
KA(4)=(RD-RC)*ANZ+.5D0   'TO RC, AND FROM RC TO RD. Z-INTEGRATION WILL  128
DO 281 IAR=1,NCY          'ALWAYS OCCUR AT LAYER BOUNDARIES AND AVOID  129
IF(KA(IAR).LT.1)KA(IAR)=1 'DISCONTINUITIES.                          130
IF((RZZ(IAR+1)-RZZ(IAR)).LT.ELM)GO TO 281 'IN CASE OF ZERO SIDEWALL                    131
DA=(RZZ(IAR+1)-RZZ(IAR))/(6.D0*KA(IAR))  132
IF(DA.GT.DZ)DZ=DA        'LARGEST DIVISION OF Z-AXIS                 133
NZP=6*KA(IAR)+1         'THE NUMBER OF Z STEPS                      134
IVZ=2                    'IVZ= Z INTEGRATION WEDDLE RULE COEF INDICATOR  135
SPZT=0.D0                'Z PARTIAL SUM RESET OF TRANSMISSION,        136
SPZM=0.D0                'MULTIPLICATION, AND                        137
SPZMIS=0.D0              'MISSES                                     138
EJ=0.D0                  'ERROR RESET FOR Z INTEGRAND IN DO LOOP 280  139
EK=0.D0                  'ERROR RESET FOR Z INTEGRAL IN DO LOOP 280  140
TEK=0.D0                 'TEMPORARY EK                               141
DO 280 I=1,NZP           'BEGIN.Z INTEGR.FROM 0-RA-RB-RC-RD          142
IVP=2 'ANGLE INTEGRATION WEDDLE RULE COEFFICIENT INDICATOR.  143
SPPT=0.D0                'ANGLE PARTIAL SUM RESET OF TRANSMISSION,    144
SPPM=0.D0                'MULTIPLICATION, AND                        145
SPPMIS=0.D0              'MISSES                                     146
EF=0.D0                  'ERROR RESET FOR PHI INTEGRAND IN DO LOOP 250  147
EG=0.D0                  'ERROR RESET FOR PHI INTEGRAL IN DO LOOP 250  148
TEG=0.D0                 'TEMPORARY EG                               149
Z=DA*DBLE (I-1)+RZZ(IAR)  150
RZ=DSQRT(R**2-Z**2)      151
RZD=0.D0                 152
RZC=0.D0                 153
RZB=0.D0                 154
RZA=0.D0                 155
GO TO (173,172,171,170),NCY  156
170 IF(RD.GT.Z)RZD=DSQRT(RD**2-Z**2)  157
171 IF(RC.GT.Z) RZC=DSQRT(RC**2-Z**2) 'HIT C LAYER, IF THERE IS ONE  158
172 IF(RB.GT.Z) RZB=DSQRT(RB**2-Z**2) 'HIT B LAYER, IF THERE IS ONE  159
173 IF(RA.GT.Z) RZA=DSQRT(RA**2-Z**2) 'HIT A LAYER  160
DO 250 J=1,NPHIP        'BEGINNING OF PHI INTEGRATION FOR PHI FROM  161
PHI=DB*DBLE(J-1)        '0 TO 90 DEGREES ONLY. PHI IS THE ANGLE  162
S=DSIN(PHI)             'BETWEEN THE Y-AXIS AND W-AXIS  163
CX=DCOS(PHI)           164
SPWT=0.D0               'W PARTIAL SUM RESET OF TRANSMISSION,        165
SPWM=0.D0               'AND MULTIPLICATION  166
IFLAG=3                 'PHI NOT EQUAL TO 0 OR 90 DEGREES  167
IF(J.EQ.1) GO TO 180    168
IF(J.EQ.NPHIP) GO TO 185  169
GO TO 200               170
180 IFLAG=1             'PHI=0 DEGREES  171
GO TO(184,183,182,181),NCY  172
181 AW(1)=-RZD          'W(1) IS THE POSITION ALONG THE W-AXIS THAT A  173
AW(8)=RZD              'NEUTRON, INCIDENT PERPENDICULARLY FROM THE  174

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182 AW(2)=-RZC          'RIGHT OR LEFT SIDE OF THE W-AXIS WILL JUST 175
    AW(7)=RZC          'STRIKE THE LEFT OUTER CORNER OF THE D 176
183 AW(3)=-RZB          'MATERIAL. AW(2), AW(3) , AND AW(4) WILL JUST 177
    AW(6)=RZB          'STRIKE THE C,B, AND A LAYERS, RESPECTIVELY. 178
184 AW(4)=-RZA          'AW(5), AW(6), AW(7), AND AW(8) WILL JUST 179
    AW(5)=RZA          'STRIKE THE UPPER RIGHT OUTER CORNER OF THE 180
    GO TO 210          'A, B, C, AND D MATERIALS, RESPECTIVELY. THIS181
185 IFLAG=2*PHI=90 DEGREES. 'AVOIDS DISCONTINUITIES IN THE DERIVATIVES FOR182
    GO TO(189,188,187,186),NCY'THE WEDDLE RULE. 183
186 AW(1)=-DLA          184
    AW(8)=DRD          185
187 AW(2)=-DLC          186
    AW(7)=DRC          187
188 AW(3)=-DLB          188
    AW(6)=DRB          189
189 AW(4)=-BLA          190
    AW(5)=BLA          191
    GO TO 210          192
200 GO TO(204,203,202,201),NCY 193
201 AW(1)=-ALIM(RZD,DLA) 194
    AW(8)=ALIM(RZD,DRD) 195
202 AW(2)=-ALIM(RZC,DLC) 196
    AW(7)=ALIM(RZC,DRC) 197
203 AW(3)=-ALIM(RZB,DLB) 198
    AW(6)=ALIM(RZB,DRB) 199
204 AW(4)=-ALIM(RZA,BLA) 200
    AW(5)=-AW(4) 201
210 LL=5-NCY 202
    LU=3+NCY 203
    DELW=AW(LU+1)-AW(LL) 'RQ=2* THE PORTION OF THE SPHERE CUT BEYOND 204
    RQ=2.00*(2.00*RZ-DELW) 'CYLINDER BECAUSE OF INCIDENCE FROM TWO 205
    IF(DELW.LT.ELM)GO TO 231 'DIRECTIONS 206
    EZ=0.00 207
    DO 230 I2=LL,LU 208
    CALL IN(AW(I2),AW(I2+1),E,SPT,SPM) 209
    EZ=EZ+E 210
    SPWT=SPWT+SPT 211
    SPWM=SPWM+SPM 212
230 CONTINUE 'END OF W INTEGRAL DO LOOP, WHICH BECOMES 213
231 GO TO(240,240,235),IFLAG'INTEGRAND FOR PHI INTGR. WITH ABS. ERROR EZ 214
235 SPPT=SPPT+V(IVP)*SPWT 215
    SPPM=SPPM+V(IVP)*SPWM 216
    SPPMIS=SPPMIS+V(IVP)*RQ 'MISS CYLINDER CALCULATION 217
    EF=EF+V(IVP)*EZ**2 'ERROR**2 IN PHI IN. DUE TO INTEGRND. UNCRT. 218
    TEG=TEG+EH(IVP)*SPWT'SUM ABSOLUTE VALUE OF ERROR EACH 6 SUBDIVISIONS 219
    IVP=IVP+1 'EH(IVP)=WEDDLE RULE ERROR CONSTANTS 220
    IF(IVP.GE.7) IVP=1 221
    IF(IVP.EQ.2)GO TO 239 222
    GO TO 250 223
239 EG=EG+DABS(TEG) 'EG=ERROR IN PHI INTEGRAL DUE TO WEDDLE RULE 224
    TEG=SPWT 225
    GO TO 250 226
240 SPPT=SPPT+SPWT 'COEFF. FOR FIRST AND LAST TERMS IS ONE 227
    SPPM=SPPM+SPWM 228
    SPPMIS=SPPMIS+RQ 229
    EG=EG+SPWT 230
    EF=EF+EZ**2 231
250 CONTINUE 'END OF PHI INTEGRAL DO LOOP, WHICH BECOMES 232
    SPPT=SPPT*0.300*DB 'INTEGRAND FOR THE Z INTEGRAL 233
    SPPM=SPPM*0.300*DB 'THIS IS THE WEDDLE RULE NORMALIZATION 234
    SPPMIS=SPPMIS*0.300*DB 'FOR THE PHI PART 235
    EF=.300*DB*DSQRT(EF) 236
    EG=EG*DB/140.00 237
    EI=DSQRT(EG**2+EF**2) 'EI=TOTAL ABS. ERROR ESTIM. FOR PHI INTEGRAL 238
    IF(I.EQ.1.OR.I.EQ.NZP) GO TO 270 239
    SPZT=SPZT+V(IVZ)*SPPT 240
    SPZM=SPZM+V(IVZ)*SPPM 241
    SPZMIS=SPZMIS+SPPMIS*V(IVZ) 242
    EJ=EJ+V(IVZ)*EI**2 'ERROR**2 DUE TO INTEGRAND UNCERT. IN Z INTEG. 243
    TEK=TEK+EH(IVZ)*SPPT'SUM ABSOLUTE VALUE OF ERROR EACH 6 SUBDIVISIONS 244
    IVZ=IVZ+1 245
    IF(IVZ.GE.7) IVZ=1 246

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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
EJ=EJ+.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(EJ**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)

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IF(EE.GT.ERL1)ERL1=EE 'ERL1 IS LARGEST ERROR IN 1ST ORDER WEDDLE DIV.319
IF(EE.GT.ET)GO TO 60 'FLL=LOWER LIMIT 320
E=AE 'FUL=UPPER LIMIT 321
RETURN 'ANS=INTEGRAL 322
50 E=0.D0 323
RETURN 324
60 ADIV=6.D0 325
IA1=6 326
KB=KB+1 'RESET COUNTER IN STATEMENT 100 327
GO TO 70 328
65 ADIV=36.D0 329
IA1=36 330
KC=KC+1 '3RD ORDER COUNT 331
DW3=DW2/6.D0 332
IF(DW3.GT.DWD)DWD=DW3 'DWD IS LARGEST 3RD ORDER SUBDIV. OF W-AXIS 333
70 ED=(FUL-FLL)/ADIV 'PCE RETURNS ABSOLUTE ERROR. 334
PII=0.D0 335
QE=0.D0 336
E=0.D0 337
FL2=FLL 338
FU2=FLL+ED 339
DO 120 IA=1,IA1 340
CALL PCE(FL2,FU2,PI,PEI,QF) '2ND ORDER OR 3RD ORDER 341
PII=PII+PI 342
QE=QE+QF 343
E=E+PEI 344
FL2=FU2 'INCREMENT LIMITS 345
120 FU2=FU2+ED 346
ANS=PII 347
IF(ANS.LT.1.D-4)GO TO 122 348
EE=DABS(E/ANS) 349
IF(IA1.EQ.36)GO TO 121 350
IF(EE.GT.ERL2)ERL2=EE 'ERL2 IS LARGEST ERROR IN 2ND ORDER WEDDLE DIV.351
IF(EE.GT.ET)GO TO 65 '4TH ORDER COUNT,KD,USED HERE TO COUNT NO OF 352
RETURN 'TIMES 3RD ORDER WEDDLE ERROR FAILS TO 353
121 IF(EE.GT.ET)KD=KD+1 'SATISFY RELATIVE ERROR ET IN INPUT 354
IF(EE.GT.ERL3)ERL3=EE 'ERL3 IS LARGEST ERROR IN 3RD ORDER WEDDLE DIV.355
RETURN 356
122 E=0.D0 357
RETURN 358
359
SUBROUTINE PCE(FL,FU,AI,AE,QC)'GIVES THE WEDDLE CORRECT SUM BETWEEN FL360
DIMENSION VII(7),VIE(7) 'AND FU USING 6 SEGMENTS. 361
DATA VII/1.D0,5.D0,1.D0,6.D0,1.D0,5.D0,1.D0/,VIE/1.D0,-6.D0,15.D0, 362
1-20.D0,15.D0,-6.D0,1.D0/,CA/3.D-1/,CB/.71428571D-2/ 363
EC=(FU-FL)/6.D0 'AI IS THE FIRST INTEGRAL, TRANSMISSION PART 364
PI=0.D0 'QC IS THE SECOND INTEGRAL, MULTIPLICATION PART365
QB=0.D0 'AE IS THE ABSOLUTE ERROR IN THE FIRST INTEGRAL366
PE=0.D0 'EC IS THE INCREMENT SIZE 367
X=FL 'X IS THE INTEGRAND EVALUATION POINT 368
DO 100 ID=1,7 369
CALL GRAND(X,QA,QG) 'QA IS THE MULTIPLICATION PART, INTEGRAND 370
PI=PI+QG*VII(ID) 'QG IS THE TRANSMISSION PART, INTEGRAND 371
QB=QB+VII(ID)*QA 'PI IS THE TRANSMISSION PART PARTIAL SUM 372
PE=PE+QG*VIE(ID) 'QB IS THE MULTIPLICATION PARTIAL SUM 373
100 X=X+EC 374
AI=CA*EC*PI 'PE IS THE ERROR PARTIAL SUM (6TH DIFFERENCE) 375
AE=DABS(CB*EC*PE) 'QC IS THE MULTIPLICATION PART ANSWER 376
QC=CA*EC*QB 377
RETURN 378
379
SUBROUTINE GRAND(W,DUMC,DUMD) 380
C FINAL EVALUATION OF THE INTEGRAND, WHICH IS PUT INTO PCE 381
C CONTRIBUTIONS ARE ADDED FOR NEUTRONS INCIDENT ON SOURCE FROM OPPOSITE 382
C DIRECTIONS, SO THAT THE PHI INTEGRAL IS CARRIED OUT FROM 0 TO PI/2 383
C INSTEAD OF FROM 0 TO PI. DUMD=TRANSMISSION PART OF INTEGRAND. 384
IGRAND=IGRAND+1 385
DUMC=0.D0 'CONTAINS MULTIPLICATION PART OF INTEGRAND 386
A=0.D0 387
B=0.D0 388
C=0.D0 389
D=0.D0 390

```



```

GO TO(90,60,30,10),NCY 'NCY=NUMBER OF LAYERS 391
10 CALL C2(IB,DRD,DLB,RZD,W,XID,YID,D) 392
GO TO (20,30),IB 393
20 DUMD=2.D0 'TRANSMISSION. IT MISSES IN BOTH DIRECTIONS 394
RETURN 395
30 CALL C2(IB,DRC,DLB,RZC,W,XIC,YIC,C) 396
IF(IB.EQ.2) GO TO 60 397
IF(NCY.EQ.3) GO TO 20 398
GO TO 130 399
60 CALL C2(IB,DRB,DLB,RZB,W,XIB,YIB,B) 400
IF(IB.EQ.2) GO TO 90 401
IF(NCY.EQ.2) GO TO 20 402
GO TO 130 403
90 CALL C2(IB,BLA,BLA,RZA,W,XIA,YIA,A) 404
IF(IB.EQ.2) GO TO 130 405
IF(NCY.EQ.1) GO TO 20 406
130 GO TO (170,160,150,140),NCY 407
140 D=D-C 'THE AMOUNT OF A,B,C, AND D MATERIAL 408
150 C=C-B 'TRAVERSED IS A,B,C, AND D, RESPECTIVELY. 409
160 B=B-A 410
170 DUM=- (SIGAT*A+SIGB*B+SIGC*C+SIGD*D) 411
DUMD=2.D0*DEXP(DUM) 'TRANSMISSION 412
IF(IB.EQ.1) RETURN 'DID NOT HIT A MATERIAL. NO FISSION 413
DIDI=0.D0 414
DIDO=0.D0 415
DICI=0.D0 416
DICO=0.D0 417
DBPBA=-SIGAT*A 418
DBPBE=(1.D0-DEXP(DBPBA)) 419
GO TO (220,210,200,190),NCY 420
190 DIDI=DSQRT((XID-XIC)**2+(YID-YIC)**2) 'ENTRY THICK. FOR FISSION CALC. 421
DIDO=D-DIDI 'SAME, FROM OTHER SIDE 422
200 DICI=DSQRT((XIC-XIB)**2+(YIC-YIB)**2) 423
DICO=C-DICI 424
210 DIBI=DSQRT((XIB-XIA)**2+(YIB-YIA)**2) 425
DIBO=B-DIBI 426
DUMBI=- (SIGD*DIDI+SIGC*DICI+SIGB*DIBI) 427
DUMBO=- (SIGD*DIDO+SIGC*DICO+SIGB*DIBO) 428
DUMCI=DEXP(DUMBI) 'FISSION EFFECT 429
DUMCO=DEXP(DUMBO) 'FROM OPPOSITE SIDE 430
DUMC=(DUMCI+DUMCO)*DBPBE 431
RETURN 432
220 DUMC=2.D0*DBPBE 433
RETURN 434
435
SUBROUTINE C2(IB,DR,DL,C2R,C2W,XI,YI,DTHRU) 436
YPF(X)=C2W*CX-S*(X-C2W*S)/CX ' Y-COORDINATE, GIVEN X, PHI, AND C2W(=W) 437
XPF(Y)=C2W*S+CX*(C2W*CX-Y)/S ' X-COORDINATE, GIVEN Y, PHI, AND C2W(=W) 438
IF(IFLAG.EQ.1)GO TO 101 439
IF(IFLAG.EQ.2)GO TO 102 440
TR=YPF(DR) 441
TL=YPF(-DL) 442
DTHRU=0.D0 443
IF(C2R.GT.TR) GO TO 30 444
20 IB=1 'NEUTRON MISSES CYLINDER. 445
RETURN 446
30 IF(-C2R.GE.TL) GO TO 20 447
IB=2 'NEUTRON HITS CYLINDER. 448
IF(-C2R.GT.TR) GO TO 60 449
XI=DR 'NEUTRON HITS END 450
YI=TR 451
GO TO 70 452
60 XI=XPF(-C2R) 'NEUTRON HITS BOTTOM. 453
YI=-C2R 454
70 IF(C2R.GT.TL) GO TO 90 455
XO=XPF(C2R) 'NEUTRON EXITS FROM THE TOP. 456
YO=C2R 457
GO TO 100 458
90 XO=-DL 'NEUTRON EXITS FROM THE END 459
YO=TL 460
100 DTHRU=DSQRT((XI-XO)**2+(YI-YO)**2) 461
RETURN 462

```

```

101 DTHRU=DR+DL                                463
    XI=DR                                        464
    YI=C2W                                       465
    IB=2                                         466
    RETURN                                       467
102 DTHRU=2.DO*C2R                              468
    XI=C2W                                       469
    YI=-C2R                                      470
    IB=2                                         471
    RETURN                                       472
    END                                          473

```

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)
1234567890123456789012345678901234567890123456789012345678901234567890
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 I 1/6 NUMBER OF Z INTE-
I      I      I      I      I      I      I      I GRATION STEPS
I      I      I      I      I      I      I      I 1/6 NUMBER OF PHI INTE-
I      I      I      I      I      I      I      I GRATION STEPS
I      I      I      I      I      I      I      FLUX IN PERCENT OF SOURCE
I      I      I      I      I      I      I      STRENGTH PER CM**2
I      I      I      I      I      I      I      NUMBER OF NEUTRONS PER FISSION
I      I      I      I      I      I      I      ABSORPTION CROSS SECTION PER CM FOR D MATERIAL
I      I      I      I      I      I      I      ABSORPTION CROSS SECTION PER CM FOR C MATERIAL
I      I      I      I      I      I      I      ABSORPTION CROSS SECTION PER CM FOR B MATERIAL
I      I      I      I      I      I      I      FISSION CROSS SECTION PER CM FOR A MATERIAL
ABSORPTION CROSS SECTION PER CM FOR A MATERIAL

```

```

SOURCE DIMENSIONS IN INCHES.  FORMAT(12G6.3)
1234567890123456789012345678901234567890123456789012345678901234567890
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  I  D SIDE WALL IN IN.
I  I  I  I  I  I  I  I  I  I  I  D LEFT END IN INCHES
I  I  I  I  I  I  I  I  I  I  I  D RIGHT END IN INCHES
I  I  I  I  I  I  I  I  I  I  I  C SIDE WALL IN INCHES
I  I  I  I  I  I  I  I  I  I  I  C LEFT END IN INCHES
I  I  I  I  I  I  I  I  I  I  I  C RIGHT END IN INCHES
I  I  I  I  I  I  I  I  I  I  I  B SIDE WALL IN INCHES
I  I  I  I  I  I  I  I  I  I  I  B LEFT END IN INCHES
I  I  I  I  I  I  I  I  I  I  I  B RIGHT END IN INCHES
I  I  I  I  I  I  I  I  I  I  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

```

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.

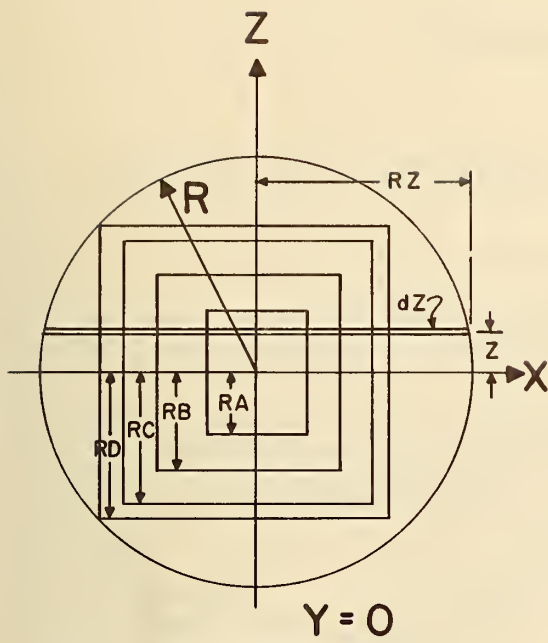
MACROSCOPIC CROSS SECTIONS PER CM
 SIGAA= .1704 , SIGAF= 0.00082 , SIGB= 1.16 , SIGC= 0.281
 NEUTRONS PER FISSION = 2.89. NZ=10. NPHI=10.
 FOR A SINGLE NEUTRON STRIKING THE SOURCE...
 THE TRANSMISSION PROBABILITY IS .40678453+0
 THE INTERACTION PROBABILITY IS .59321547+0
 THE ABSORPTION PROBABILITY IS .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.

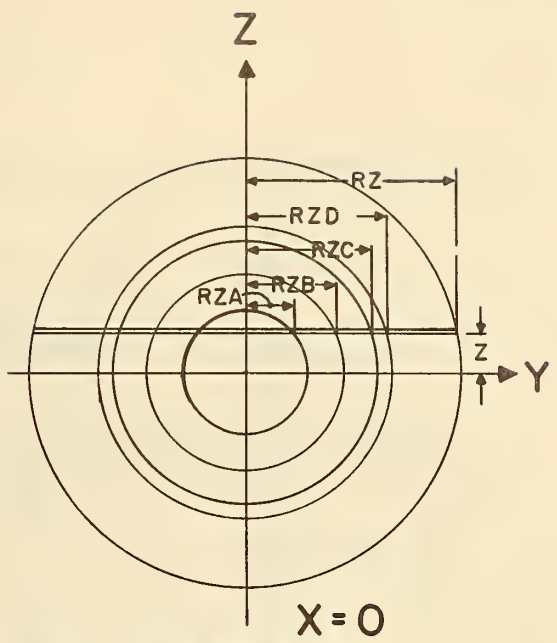
Neutron Transmission Convergence Test for the Spherical
Source Program

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%	-

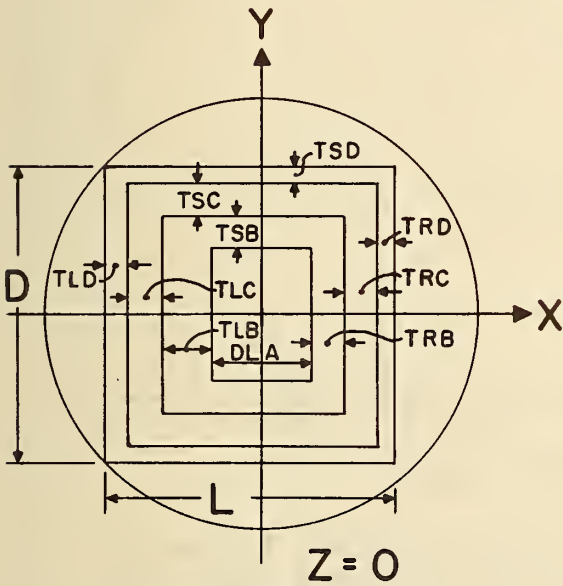
* Actual Error listed here assumes that transmission
for NY = 1000 is correct.



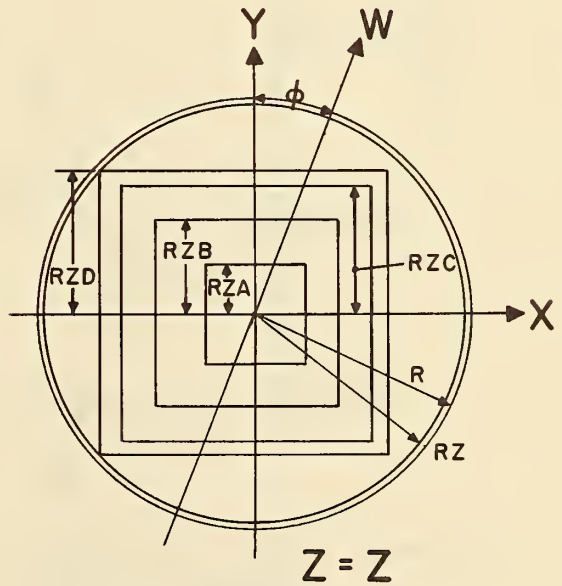
1(a) VIEW FROM -Y



1(b) VIEW FROM +X



1(c) VIEW FROM +Z



1(d) VIEW FROM +Z

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.

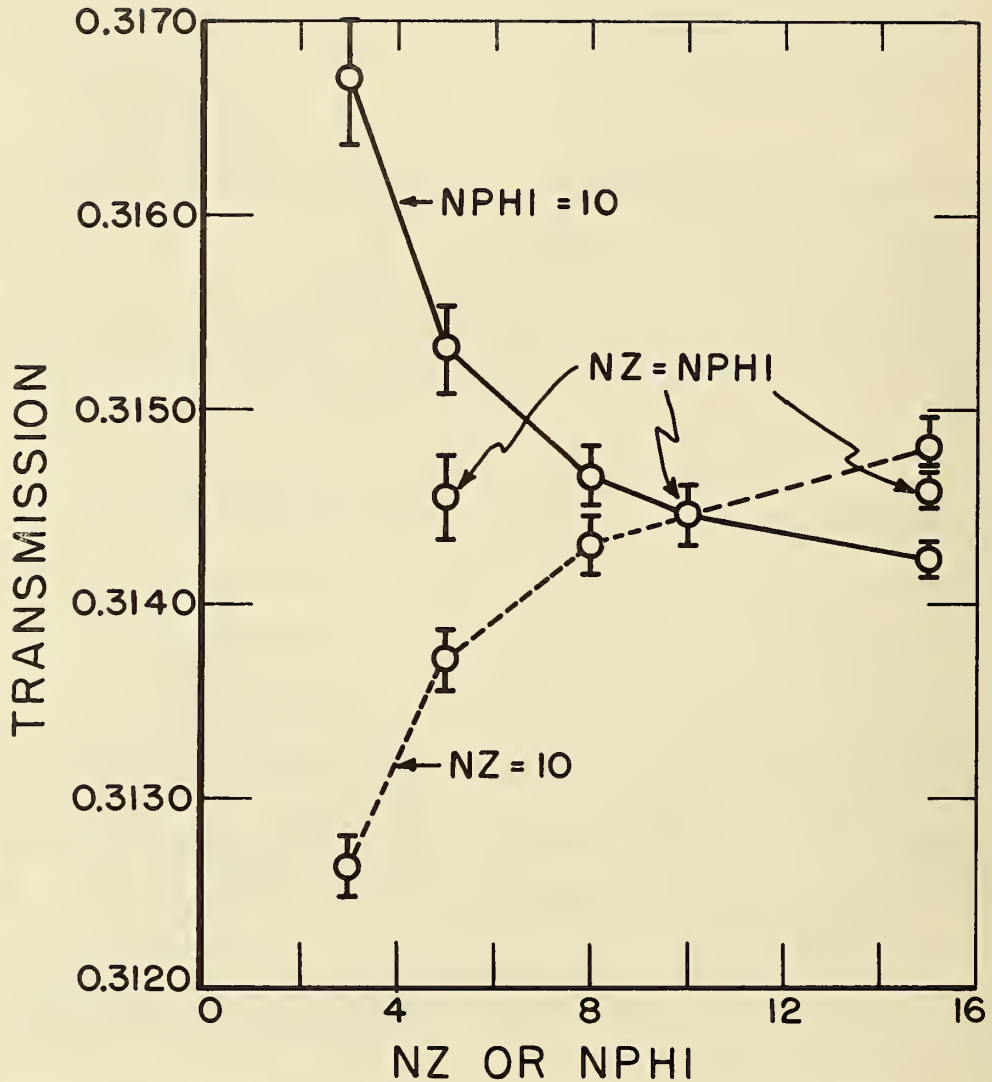


Figure 2. The calculated Transmission and Estimated Error for a nickel encapsulated Pu-Be cylindrical source for various specified subdivisions of the Z and φ integrals, NZ and NPHI. The solid line joins points for which NPHI equals 10 and the dashed line joins points for which NZ equals 10. Two points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

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7. AUTHOR(S) V. Spiegel, Jr. and W. M. Murphey		8. Performing Organization	
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