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SOURCE, A NEUTRON DISTRIBUTION ROUTINE FOR THE O5R MONTE CARLO CODE

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ABSTRACT

SOURCE is a FORTRAN 63 program for determining the energy, the speed, and the initial spatial distribution of neutrons arising from fission. The energy distribution within the batch of neutrons, is obtained by sampling a Maxwellian fission spectrum which depends upon the energy of the neutrons producing fission. The program locates the neutrons at a point or distributes them spatially, either uniformly or according to a cosine function. Two isotopes producing fission neutrons having a slightly different energy distribution can be treated in this program by an approximation (using an average temperature) giving the correct first moment of the resulting overall energy distribution.

* Oak Ridge Gaseous Diffusion Plant

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I - INTRODUCTION

SOURCE is a FORTRAN 63 program for determining the energy, the speed, and the initial spatial distribution of neutrons arising from fission; these neutrons are then followed and described by the O5R Monte Carlo transport code.[1] The energy distribution within the batch of neutrons, of preselected size, is obtained by sampling a Maxwellian fission spectrum which depends upon the energy of the neutrons producing fission.

The program locates the neutrons at a point or distributes them spatially, either uniformly or according to a cosine function. At present no provision for anisotropic emission is included.

Two isotopes producing fission neutrons having a slightly different energy distribution can be treated in this program by an approximation (using an average temperature) giving the correct first moment of the resulting overall energy distribution.

II - THEORY

The spectrum of neutrons from fission in a single isotope is assumed to be Maxwellian:

$$f(E) = \frac{2}{\sqrt{\pi}\ T^{3/2}} E^{1/2}\ e^{-E/T} \tag{1}$$

where the nuclear temperature, T, is a function of the energy of the neutron inducing fission and E is the energy of the emitted neutron. The average energy of the emitted neutrons is related to the nuclear temperature by the relation

$$\langle E \rangle = (3/2)T \tag{2}$$

The average energy is also related to the average number of neutrons per fission, ν, by the relation[2]

$$\langle E \rangle = A + B\sqrt{\nu + 1} \tag{3}$$

where A and B are constants. Thus

$$T = 2/3\left(A + B\sqrt{\nu + 1}\right) \tag{4}$$

1. Coveyou, R. R., et al., O5R, A General-Purpose Monte Carlo Neutron-Transport Code, ORNL-3622 (1965).
2. J. Terrell, Phys. Rev. 127, 3, p. 899 (1962).
*. Oak Ridge Gaseous Diffusion Plant.

The value of the constants for ^{235}U and ^{238}U and some typical nuclear temperatures are given in Table I. An average nuclear temperature of a mixture of two isotopes is computed by the relation

$$T = \sum_i x_i(E) T_i(E) \tag{5}$$

where the fraction of neutrons emitted by the i^{th} isotope, x_i, is given by

$$x_i = \frac{\nu_i \; \Sigma_f^i}{\sum(\nu_i \; \Sigma_f^i)} \; . \tag{6}$$

If the average value of the m^{th} moment of the energy of the neutrons emitted in fission is compared to that calculated from the exact spectrum, which is a sum of two spectra of the form of Eq.(1), the following relation is obtained

$$\frac{\langle E^m \rangle \text{ Approx}}{\langle E^m \rangle \text{ Exact}} = \frac{(\sum x_i T_i)^m}{\sum x_i T_i^m} \; . \tag{7}$$

For two isotopes this can be written as

$$\frac{\langle E^m \rangle \text{ Approx}}{\langle E^m \rangle \text{ Exact}} = \frac{[x_1 + (1-x_1)\; T_2/T_1]^m}{[x_1 + (1-x_1)\; (T_2/T_1)^m]} \; . \tag{8}$$

This ratio is given for some values of m, T_2/T_1 and x in Table II. The approximation gives the correct average energy. For the case of equal number of fissions in each isotope and a temperature ratio of 1.1 the moments less than the fourth are smaller than the exact values by no more than 1%. For the case of 95% of the fissions in one isotope and $T_2/T_1 = 1.1$ moments up to the fifth are less than 0.5% too small. The approximation, therefore, introduces little error in the fission neutron spectrum in uranium containing 93.2% ^{235}U. The dependence of the fission neutron yield of ^{235}U on the incident neutron energy is taken as a quadratic function up to 8 Mev according to the data of Moat, Mather and Fieldhouse.[3]

$$\nu(^{235}U) = 2.43 + 0.088E + 0.0088E^2 \tag{9}$$

where E is in Mev. The yield is taken to be linear between 8 and 15 Mev having, at the highest energy, a value of 4.41. For ^{238}U the energy dependence of the fission neutron yield is linear between 1 and 15 Mev with values of 2.48 and 4.58, respectively, corresponding

3. D. S. Mather, P. Fieldhous, and A. Moat, Phys. Rev. 133, B, 1403 (1964).

Table I. Constants and Typical Nuclear Temperatures for ^{235}U and ^{238}U Expressed in Mev.

	^{235}U[a]	^{238}U[b]
A	0.74	0.80
B	0.653	0.698
T[c]	1.3	-
T[d]	1.5	1.6

a. J. Terrell, Phys. Rev. 127, 3, 899 (1962).

b. The ^{238}U constants are chosen to match the ratio of T_{28}/T_{25} at 14.3 Mev measured by Yu A. Vasil'ev et al., JETP, 11, 3, 483 (1960) using their measured values of the number of neutrons produced per fission.

c. Incident neutrons at thermal energy.

d. Incident neutrons at 14.3 Mev.

Table II. Ratio of the Approximate Moments of Energy to the Exact Values for Two Fissionable Isotopes.

Order of Moments m	$\langle E^m \rangle$ Approx/$\langle E^m \rangle$ Exact x = 0.5 $T_2/T_1 = 1.1$	 x = 0.95 $T_2/T_1 = 1.1$
1	1.0	1.0
2	0.9977	0.9995
3	0.9932	0.9985
4	0.9866	0.9970
5	0.9778	0.9949

a. x is the fraction of fissions occurring in the isotope at temperature T_1; T_2 is the temperature of the other isotope.

to $d\nu/dE$ of 0.15 neutrons per Mev.

III - GENERAL DESCRIPTION OF THE PROGRAM

SOURCE is a subroutine which calculates the energy, the speed, and the initial spatial distribution of the neutrons arising from fission in two fissionable isotopes; these neutrons are then followed and described by the Monte Carlo calculation. In the following discussion these neutrons are called "source neutrons." SOURCE transmits the information concerning the source neutrons to O5R through a list of parameters used as arguments in the calling statement. These parameters are as follows:

SPDSQ: The square of the speed of the neutron emitted in fission. The factor for converting neutron energy in eV to (neutron speed)^2 in cm^2/sec^2 is 1.913220092 x 10^{12} cm^2/(sec^2 eV).

U,V,W: The directional cosines of the source neutron.

X,Y,Z: The coordinates of the source neutron.

WATE: The weighting factor initially assigned to a source neutron.

N: An integer associated with a particular neutron.

NMED: The medium in which X, Y, and Z lie.

NREG: The region in which X, Y, and Z lie.

NMEM: The number of initial source neutrons.

Certain of the above parameters are not associated with either the energy or the spatial distribution of the source spectrum and are, therefore, taken from information furnished in the O5R input data (Cards G and H). The neutron starting weight (WATE), the number of initial source neutrons (NMEM), and the name of each source neutron (N) are examples of information not calculated by SOURCE. At present the directional cosines (U, V, and W), are those associated with isotropic emission although the routine can be modified to include anisotropic emission.

At the beginning of a batch, SOURCE is entered once for each neutron until all of the source neutrons (NMEM) have been processed. The distribution of these neutrons in space is calculated by sampling from a delta, a uniform, or a cosine function. A uniform function indicates that all initial coordinates, along each of three axes, within the whole system are equally probable. The cosine function samples the initial coordinates (X, Y, Z) of the source neutrons from cosine functions along each coordinate axis. In the uniform and cosine distributions each position is checked to insure that no source position will be picked which lies in either an internal or an external void. This check is not made for a point source distribution. The coordinates (X, Y, Z) of each source neutron are supplied to O5R by SOURCE. It should be noted that the information calculated in SOURCE will take precedence over O5R input data, that is, a value of the starting coordinates, XSTRT, YSTRT, and ZSTRT appearing on card G of the O5R input data section will establish a starting point for the source neutrons only if a "1" indicating a point source is placed in the control card input to SOURCE.

A Maxwellian fission spectrum, with a specified nuclear temperature, is sampled to give the energies of the neutrons in the first batch. This method of sampling is provided in the O5R random number package. The energy spectrum for the second and succeeding batches of neutrons is also picked from the Maxwellian spectrum but now information from the previous batch is used to select the nuclear temperatures in the second and succeeding batches. In this manner SOURCE iterates on the fission energy spectrum to obtain a converged spectrum. The nuclear temperature as a function of energy within each region can be calculated by the cross section routine of O5R and put on data cards. These cards, which follow the source control card specifying the spatial distribution of the first batch of neutrons, are then read by SOURCE for storage as a table. A search of this table, which may contain up to 100 values of energy and their associated temperatures, is used to obtain the nuclear temperature. This search is easily performed since the value of SPDSQ upon entry into SOURCE is proportional to the energy of the neutron producing the fission.

In the application of SOURCE to one-velocity problems, the neutron velocity is set to a constant. The control card specifying the type of spatial distribution to be given the initial source batch must be present.

IV - INPUT SPECIFICATIONS

The cards on which the SOURCE input data are specified are arranged as follows:

Card 1: FORMAT (E10.5[a], I5[b])

a. TEMPER: Temperature of the Maxwellian used in the first batch.

b. NSORCE: A variable used to indicate the type of spatial spectrum desired for the first generation neutrons. A "1" punched in Column 5 indicates that a point source is desired and the values of XSTRT, YSTRT, and ZSTRT on card H os the O5R input data are used for these coordinates. A "2" punched in Column 5 indicates that a uniform distribution is to be sampled. A "3" punched in Column 5 indicates that a cosine distribution along each of the coordinate axes is to be sampled.

Card 2: FORMAT ($\overset{a}{\text{E10.5}}$, $\overset{b}{\text{E10.5}}$, $\overset{c}{\text{E10.5}}$, $\overset{d}{\text{E10.5}}$, $\overset{e}{\text{E10.5}}$, $\overset{f}{\text{E10.5}}$)

a. XZERO: The X coordinate corresponding to the center of the cosine or of the uniform distribution.

b. XLENTH: The X coordinate at which the source spatial distribution becomes zero. For the cosine distri-
ZLENTH corresponds to L in the relationship
$S(X) = \cos \frac{\pi}{2L} X.$

c. YZERO: Same as (a) for the Y axis.

d. YLENTH: Same as (b) for the Y axis.

e. ZZERO: Same as (a) for the Z axis.

f. ZLENTH: Same as (b) for the Z axis.

Card 3: FORMAT ($\overset{a}{\text{E10.5}}$, $\overset{b}{\text{E10.5}}$)

a. E: The energy in ev. Energies are given in decreasing order.

b. TBAR: The nuclear temperature corresponding to the energy specified in (a) for the Maxwellian distribution.

As many as one hundred values of E and of TBAR may be used with one pair of values per card.

Card 4: BLANK

A blank card is used to signify the end of the E and the TBAR input data. Cards 3 and 4 are deleted in the solution of one velocity problems.

The following outline summarizes the arrangement and data necessary for SOURCE.

A. Multi-Velocity O5R

1. O5R INPUT DATA

2. GEOM INPUT DATA

3. SOURCE INPUT DATA

a. CARD 1

b. CARD 2

c. CARD 3

.
.
.

c. CARD 3

d. CARD 4

B. ONE VELOCITY O5R

1. O5R INPUT DATA

2. GEOM INPUT DATA

3. SOURCE INPUT DATA

a. CARD 1

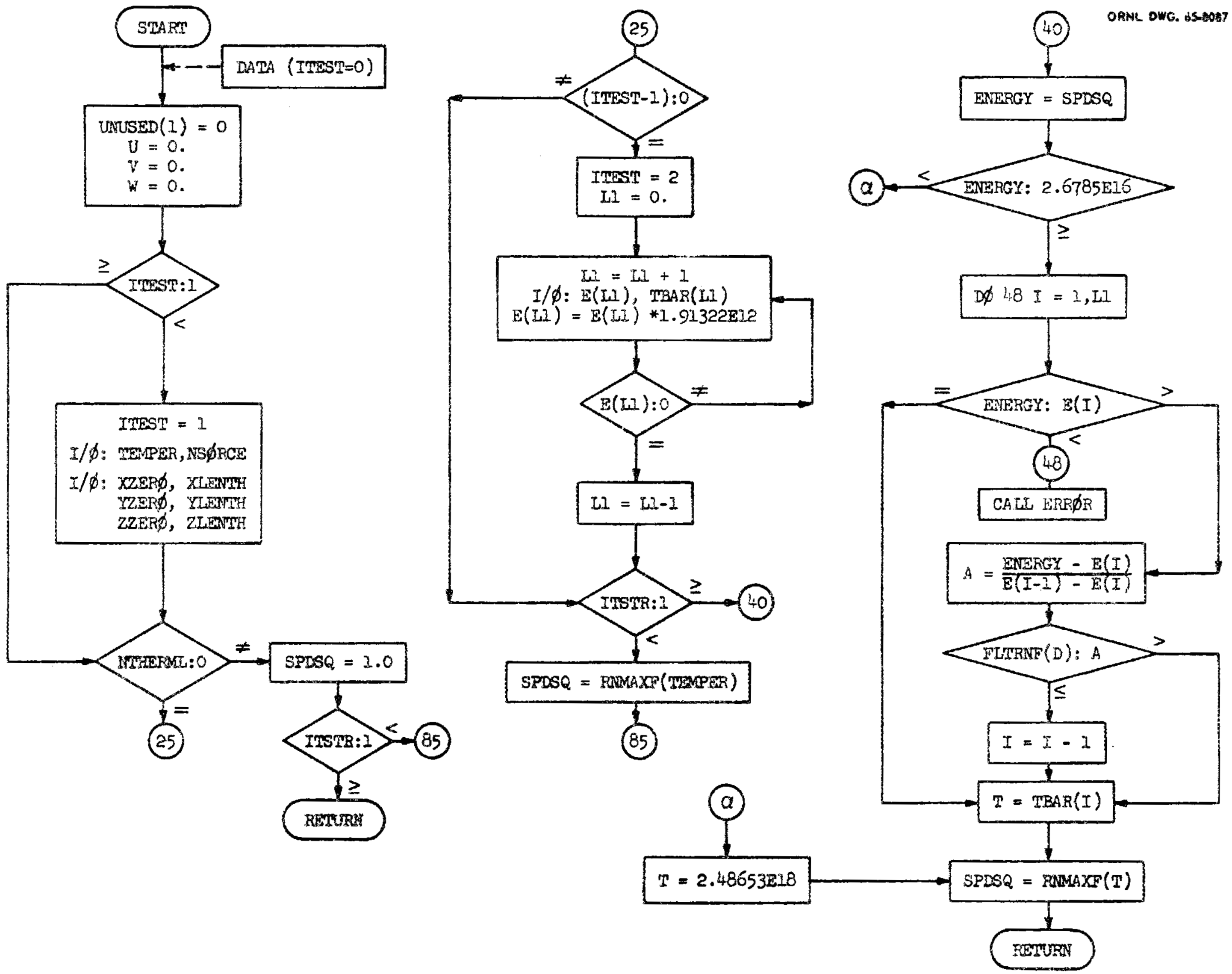
b. CARD 2

V - FORTRAN STATEMENTS AND FLOW CHART FOR SOURCE

```
      SUBROUTINE SOURCE(SPDSQ,U,V,W,X,Y,Z,DUM1,DUM2,DUM3,DUM4,DUM5)
      TYPE INTEGER UNUSED
      DIMENSION AS(2), E(100,2),  TBAR(100,2)
      COMMON/SINGLES/BLZON,EBOT,ECUT,EGROUP,EINC,EMONO,ESOUR,
     1ETAPE,ETA,ETATH,ETAUSD,ETOP,FONE,FTOTL,FWATE,ITERS,ITSTR,
     2LELEM,LF,MARK,MAXGP,MEDIA,MGPREG,MFISTP,MXREG,N,NCOLPR,
     3NWPCOL,NCONT1,NCONT2,NCONTP,NEWNM,NFISH,NFONE,NFPT,NGEOM,
     4NGROUP,NGWT,NHISMX,NHISTR,NINC,NFINC,NPINC,NITS,NKILL,
     5NLAST,NGLAST,NSIGL,NPLAST,NLEFT,NMEM,NMOST,NOEL,NPCOF,
     6NPTAPE,NQUIT,NROOM,NSOUR,NSPLT,NSTAPE,NSTRT,NTHERM,NTHRML,
     7NTYPE,OLDWT,PSIE,SPOLD,THETM,TNUC,UINP,UOLD,VINP,VOLD,
     8WATEF,WINP,WOLD,WTAVR,WTHIR,WTLOR,WTRED,WTSTRT,XOLD,XSTRT,
     9YOLD,YSTRT,ZOLD,ZSTRT,UNUSED(10)
      COMMON/JOMIN2/DUM9(5),XPBD,XMBD,YPBD,YMBD,ZPBD,ZMBD
      COMMON/GEOM/N9,DAM(6),NMEDG,NZ,EU,E3,B3
      DATA(ITEST#0)
      UNUSED(1)#0
      U#0.0
      V#0.0
      W#0.0
      IF(ITEST-1)1,15,15
    1 ITEST#1
      READ 5,TEMPER,NSORCE
    5 FORMAT(E10.5,I5)
      PRINT 6,TEMPER,NSORCE
    6 FORMAT(1HK,7NUCLEAR TEMPERATURE #7,E11.4,10X,7NSORCE #7,I3)
      IF(NSORCE-1)15,8,8
    8 READ 10,XZERO,XLENTH,YZERO,YLENTH,ZZERO,ZLENTH
   10 FORMAT(6E10.5)
      PRINT12,XZERO,YZERO,ZZERO,XLENTH,YLENTH,ZLENTH
   12 FORMAT(1H0,7HXZERO #,E11.4,5X,7HYZERO #,
     1E11.4,5X,7HZZERO #,E11.4/1H0,8HXLENTH #,
     2E11.4,5X,8HYLENTH #,E11.4,5X,8HZLENTH #,E11.4/1H1)
   15 IF(NTHRML)18,25,18
   18 SPDSQ#1.0
      IF(ITSTR-1)85,20,20
   20 RETURN
   25 IF(ITEST-1)36,28,36
   28 LI#0
      ITEST#2
   30 LI#LI+1
      READ 32,E(LI),TBAR(LI)
   32 FORMAT(2E10.5)
      E(LI)#E(LI)*1.913220092E12
      IF(E(LI))35,35,30
```

```
 35 LI#LI-1
 36 IF(ITSTR-1)38,40,40
 38 SPDSQ#RNMAXF(TEMPER)
    GO TO 85
 40 ENERGY#SPDSQ
    IF(ENERGY.LE.2.6785E16)42,43
 42 T#2.48635E18
    GO TO 53
 43 DO48I#1,LI
    IF(ENERGY-E(I))48,52,50
 48 CONTINUE
    CALL ERROR
 50 IF(FLTRNF(D)-(ENERGY-E(I))/(E(I-1)-E(I)))51,51,52
 51 I#I-1
 52 T#TBAR(I)
 53 SPDSQ#RNMAXF(T)
 55 RETURN
 85 GO TO(70,200,300),NSORCE
 70 RETURN
200 X#FLTRNF(D)*XLENTH
    X#SIGNF(X,SFLRANF(D)) + XZERO
    Y#FLTRNF(D)*YLENTH
    Y#SIGNF(Y,SFLRANF(D)) + YZERO
    Z#FLTRNF(D)*ZLENTH
    Z#SIGNF(Z,SFLRANF(D)) + ZZERO
    CALL LOOKZ(X,Y,Z)
    IF(NMEDG.EQ.1000.OR.NMEDG.EQ.0)200,70
300 AI#2.0/3.14159265
    X#ASINF(FLTRNF(DUM))*AI
    X#SIGNF(X*XLENTH,SFLRANF(D))+XZERO
    Y#ASINF(FLTRNF(DUM))*AI
    Y#SIGNF(Y*YLENTH,SFLRANF(D))+YZERO
    Z#ASINF(FLTRNF(DUM))*AI
    Z#SIGNF(Z*ZLENTH,SFLRANF(D))+ZZERO
    CALL LOOKZ(X,Y,Z)
    IF(NMEDG.EQ.1000.OR.NMEDG.EQ.0)300,70
    END
```

ORNL DWG. 65-8087



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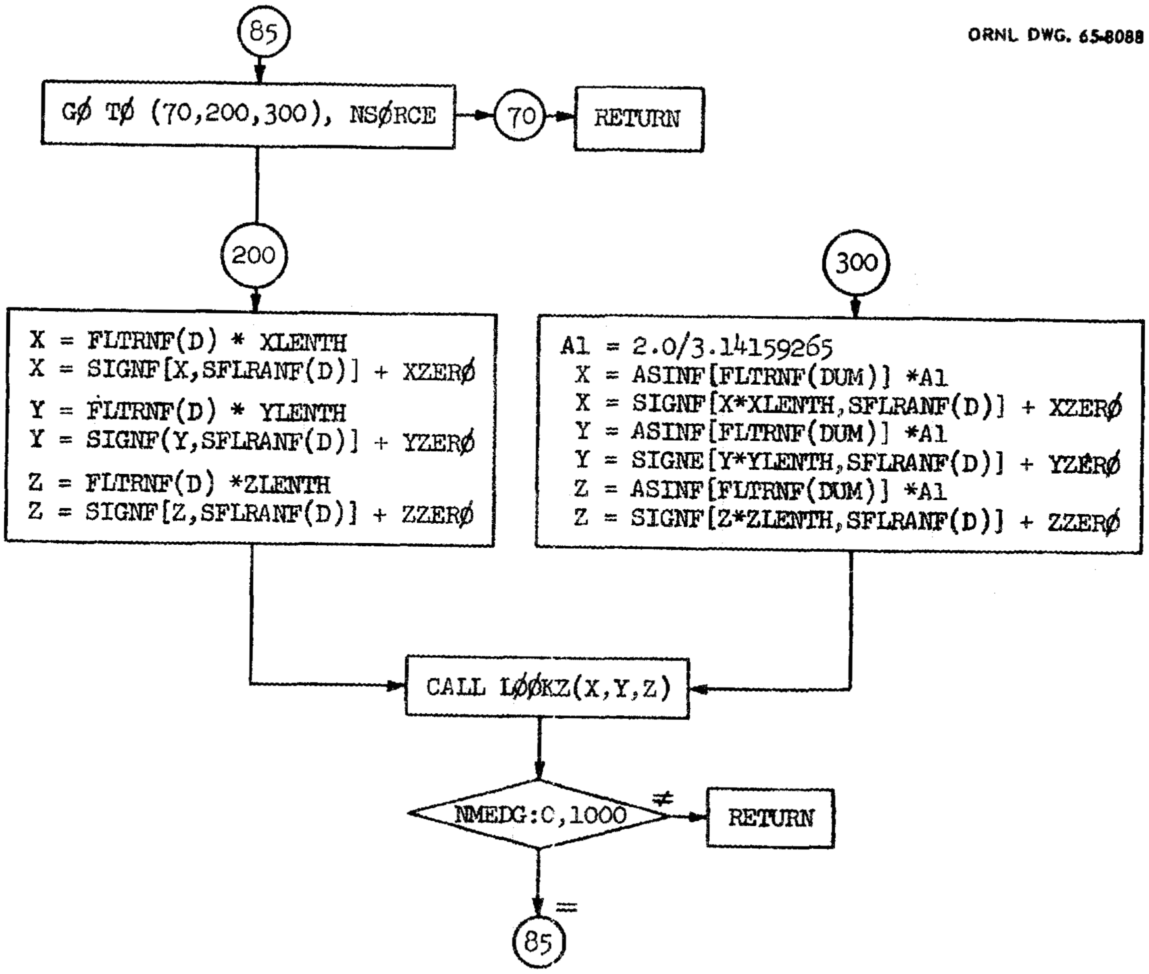

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