

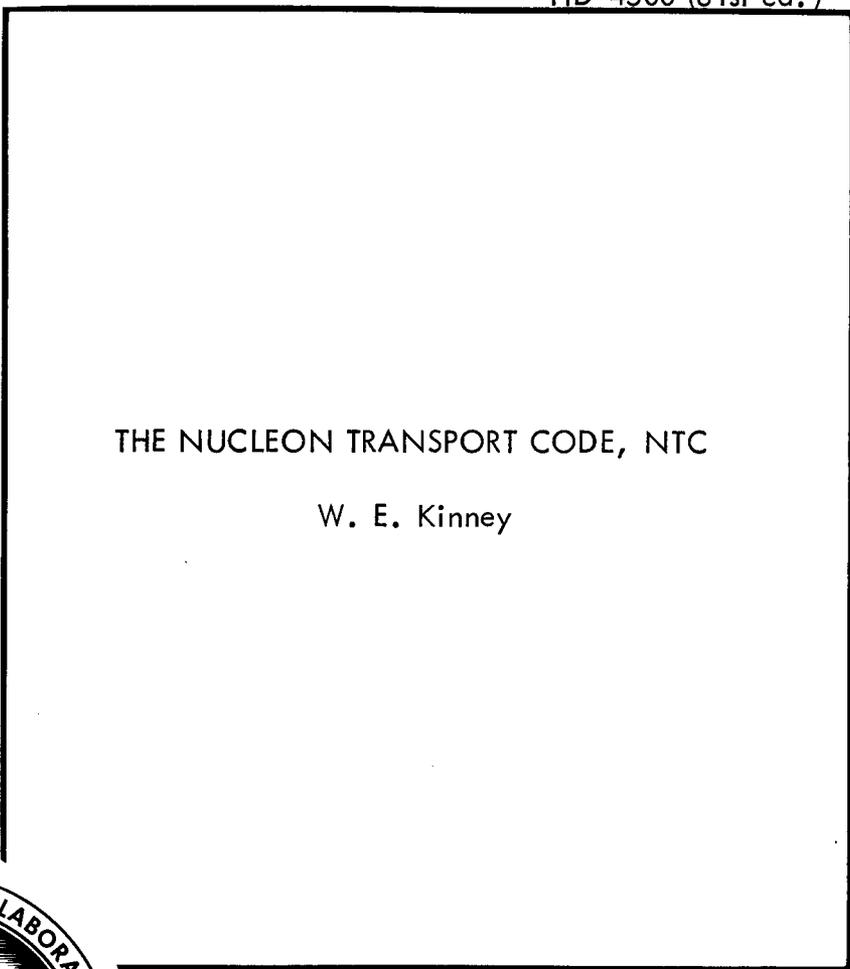
DOCUMENT COLLECTION

MARTIN MARIETTA ENERGY SYSTEMS LIBRARIES



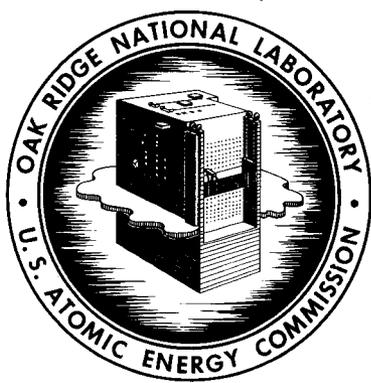
3 4456 0050152 5

ORNL-3610
UC-34 - Physics
TID-4500 (31st ed.)



THE NUCLEON TRANSPORT CODE, NTC

W. E. Kinney



OAK RIDGE NATIONAL LABORATORY

operated by

UNION CARBIDE CORPORATION

for the

U. S. ATOMIC ENERGY COMMISSION

Printed in USA. Price: \$2.25 Available from the
Office of Technical Services
U. S. Department of Commerce
Washington 25, D. C.

— LEGAL NOTICE —

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

- A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or
- B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

ORNL-3610

Contract No. W-7405-eng-26

Neutron Physics Division

THE NUCLEON TRANSPORT CODE, NTC *

W. E. Kinney

AUGUST 1964

*Work supported by the National Aeronautics and Space Administration under
NASA Order R-104.

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION



3 4456 0050152 5



Abstract

Detailed information concerning the interactions of high-energy (about 400 MeV) nucleons with matter is of particular interest in the design of radiation shields for space vehicles and high-energy particle accelerators.

The Monte Carlo method has been applied to the calculation of such interactions in the construction of the Nucleon Transport Code (NTC), a linked series of newly written codes for treatment of high-energy transport and previously written codes for the treatment of neutrons of energies below 50 MeV. NTC, written for the IBM-7090 computer, calculates the transport of nucleons having initial energies as high as 400 MeV through complex arbitrary configurations containing a maximum of four media, each of which may be composed of up to ten isotopes. Cascade processes and particle evaporation from excited nuclei are taken into account. The end result of NTC is one or more magnetic tapes containing the detailed records of all quantities pertinent to the transport of the nucleon. These tapes can be independently analyzed by the individual user of NTC to produce desired solutions to specific problems.

Appendices include typical analysis routines; a list of tape assignments; a users' manual for the low-energy portion of the code and for the geometry routine; and a full-scale demonstration problem, complete in all details, which can be used to verify satisfactory performance by all parts of the code group.

Acknowledgment

The group of computer programs reported on here represents the work of many people. O5R and the geometry routine were designed and programmed by R. R. Coveyou, J. G. Sullivan, and D. C. Irving. C. D. Zerby and especially R. R. Coveyou have given many helpful suggestions on the design of the high-energy transport code. Finally, H. W. Bertini and L. Dresner are to be commended for permitting their programs to suffer the indignity of being made into subroutines. The subroutines were prepared by V. E. Anderson.

Table of Contents

	<u>Page No.</u>
Abstract -----	iii
Acknowledgment -----	iv
1. Introduction -----	1
2. General Description -----	2
2.1. High-Energy Transport Code -----	2
2.2. The High-Energy Analysis Routine -----	8
2.3. The O5R Code -----	10
2.4. The O5R Analysis Routine -----	10
3. Input to Nucleon Transport Code (NTC) -----	10
4. High-Energy Collision Tape -----	12
5. Range-Energy Tables -----	15
6. Subroutine WEK -----	17
7. O5R Source Preparation -----	21
App. A. Typical High-Energy and O5R Analysis Routines -----	23
A1.1. Input to Main Routine -----	24
A1.2. Input to ANAL -----	24
A1.3. Input to ANAL2 -----	25
A1.4. Input to O5R Analysis Code -----	26
App. B. Tape Assignments -----	28
App. C. Routines in the High-Energy Nucleon Transport Code -----	29
App. D. O5R User's Manual -----	32
D.1. Introduction -----	32
D.2. Definitions -----	33
D.3. General Description of O5R Processes -----	34
D3.1. Construction of Neutron Histories -----	34
D3.2. Fissioning -----	36
D3.3. Thermal Neutrons -----	37
D3.4. Neutron Weights -----	38
D.4. Random Variable Selection in O5R -----	40
D.5. Cross Sections -----	43
D5.1. Cross-Section Representation -----	43
D5.2. Cross-Section Handling Code Package -----	43
D5.3. Code 1 and Code 2: Cross-Section Input Codes (Cards to Tape) -----	46
D5.4. Code 3: Cross-Section Edit Code -----	47
D5.5. Code 4: Cross-Section Deletion Code -----	47
D5.6. Code 5: Cross-Section Arithmetic Code -----	48
D5.7. Code 6: Reactor Data Preparation Code -----	49
D5.8. Code 7: Special Tape of Averaged Cross Sections -----	52

Table of Contents (cont.)

	<u>Page No.</u>
D.6. Geometry Routines -----	53
D6.1. Input for GEOM -----	55
D.7. O5R Input -----	58
D7.1. O5R Neutron Sources -----	58
D7.2. Reactor Data Input -----	60
D.8. Treatment of Nonelastic Scattering -----	65
D.9. Problem Analysis -----	68
D9.1. Collision Tape -----	68
D9.2. Parameter List -----	70

1. Introduction

The interaction of a high-energy proton or neutron with the material of a radiation shield initiates a complex avalanche of secondary particles which proceed through the shield, increasing in number and decreasing in total energy. In general, the direct interaction of the initial particle with the nucleons within the nuclei of the shield constituents produces several secondary nucleons having energies ranging from a few MeV to a large fraction of the incident particle energy. There is left a highly excited recoiling nucleus, which rids itself of most of its excess energy by evaporating nucleons and heavy particles of relatively low energy, of the order of a few MeV. Interaction of the high-energy secondaries continues and expands the cascade. It is essential to the design of the shield of a space vehicle to know the behavior of such avalanches as a function of energy and shield thickness.

A calculational method which is uniquely capable of encompassing all desired details of avalanche behavior is the Monte Carlo method. In effect it performs an idealized experiment upon a system whose properties are completely known, using counters of known resolution and perfect efficiency. Like all counting experiments, however, it is subject to counting uncertainties, and in application a balance must be struck between reduction of these uncertainties and computing costs.

The Monte Carlo method has been successfully applied to the treatment of neutron transport at fission energies and below in O5R, a code for the IBM 704 and 7090 computers.¹ Two circumstances, however, prevent a simple extension of O5R treating nucleons having energies greater than fission. First, at high energies nucleons may be absorbed, producing multiple secondary nucleons. In O5R a neutron never disappears by absorption at a collision, but rather continues with its statistical weight multiplied by the nonabsorption probability at the energy of the collision. Second, protons lose energy in flight due to ionization collisions, and

1. R. R. Coveyou, J. G. Sullivan, and H. P. Carter, "The O5R Code: A General Purpose Monte Carlo Reactor Code for the IBM-704 Computer," Codes for Reactor Computations, International Atomic Energy Agency, Vienna (1961), p. 267.

account must be taken of this effect. For these reasons, then, the approach to high-energy nucleon transport embodied in the present Nucleon Transport Code (NTC) has been to transport the nucleons from high energy down to some appropriate boundary energy by a separate code. Below this boundary neutrons are treated by O5R and protons are transported at the discretion of the user. A suitable boundary energy is 50 MeV, since most evaporation particles have energies < 50 MeV, the direct interaction model does not apply below this limit, and the proton range is small.

The purpose of the present paper is to facilitate use of the Nucleon Transport Code (NTC) by interested parties. A detailed description of the physical models and the mathematical methods employed is therefore reserved for a later publication. The series of codes comprising NTC will first be described in a general way, followed by a more detailed discussion of the high-energy transport and analysis codes. Running instructions and a description of input will also be given, and a sample problem will be solved.

2. General Description

The Nucleon Transport Code is actually a series of four basic codes, all written for the IBM-7090, which when combined calculate the transport of nucleons with energies up to 400 MeV through complex arbitrary configurations containing a maximum of four media, each of which may be composed of as many as 10 isotopes. Each basic code in turn employs a number of specific subroutines. The basic codes are the High-Energy Transport Code, the High-Energy Analysis Routine, the O5R, and the O5R Analysis Code. Discussed separately below, their order and general functions are outlined in the schematic flow chart of Fig. 2.1.

2.1. High-Energy Transport Code

The High-Energy Transport Code introduces the nucleons into the system and transports them down through the region between 400 and 50 MeV. The only tasks assigned to this code are the tracing of particle paths in phase space and the recording of information at significant points along the paths. When a nucleon has a collision, escapes from a medium, or slows down past 50 MeV, details of the event such as the position and velocity of

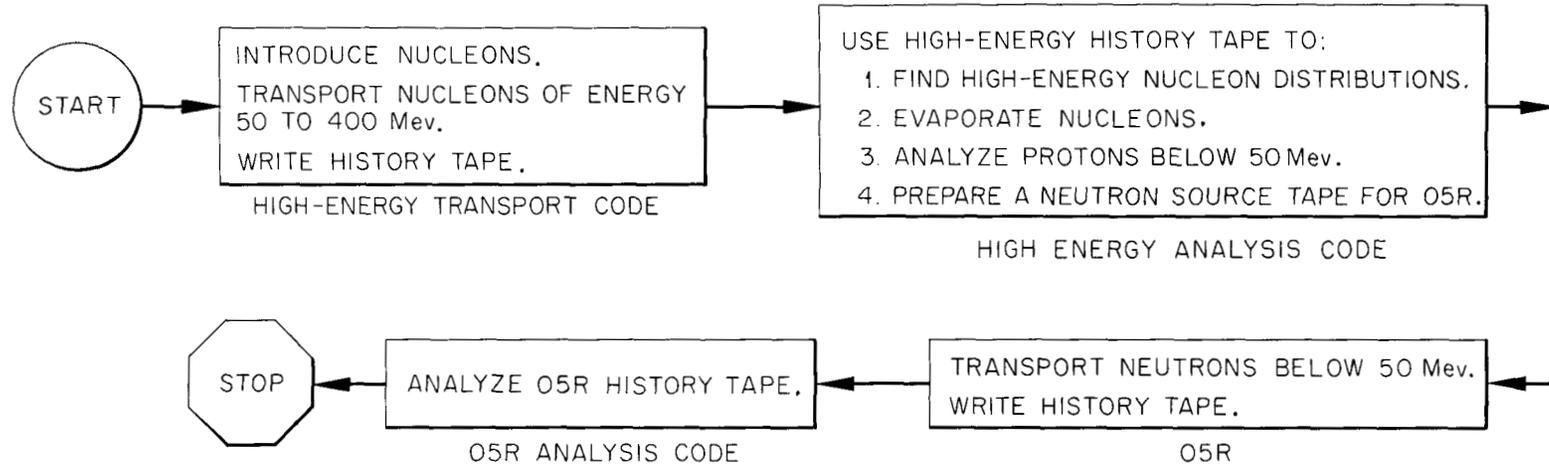


Fig. 2.1. Schematic Flow Chart of the Nucleon Transport Code

the nucleon at the time of the event are recorded on a history tape. This method was originally devised for O5R. It permits freedom in the manner of analyzing the particle trajectories and also allows saving the data for future reanalysis for additional information. A general flow chart of the code is shown in Fig. 2.2.

The computation starts with the specification of the upper- and lower-energy limits of the calculation; the atomic number, atomic weight, nuclear density, and effective ionization potential for each isotope of each medium; the number of source particles to be transported; a description of the geometry of the problem; and any source information required by the source subroutine. A range table containing values of the distance a proton will travel for each of 101 values of the energy equally spaced between the upper and lower limits is then computed, as well as an inverse range table containing values of the energy corresponding to 101 values of the range equally spaced between the maximum range and zero. Quantities in both tables are assumed to vary linearly between entries.

Flexibility in the geometry of the problem is achieved by using Irving's general-purpose geometry routine,² which subdivides the space of interest into a number of parallelepipeds, each of which may be further subdivided by several quadratic surfaces. Given the end points of a line, the subroutine specifies the coordinates of the intersection of the line with the nearest intervening surface, if any.

The repetitive portion of the High-Energy Transport Code starts by the source subroutine being called to select a source particle. This subroutine is arbitrarily written by the user to fit the requirements of the problem at hand. A flight distance, d , is next selected from the distribution $\Sigma e^{-\Sigma d}$, where Σ is the macroscopic total geometric cross section.³ If within the distance d the particle escapes the system, the position of the spatial boundary crossing is recorded on the history tape and interest in the particle ceases for the code. If the particle being treated is a

2. D. Irving, Neutron Phys. Div. Ann. Progr. Rept., Sept. 1, 1962, ORNL-3360, p. 230.

3. For hydrogen the largest proton cross section in the energy range, the n-p cross section at the lower energy limit, plays the role of a geometric cross section.

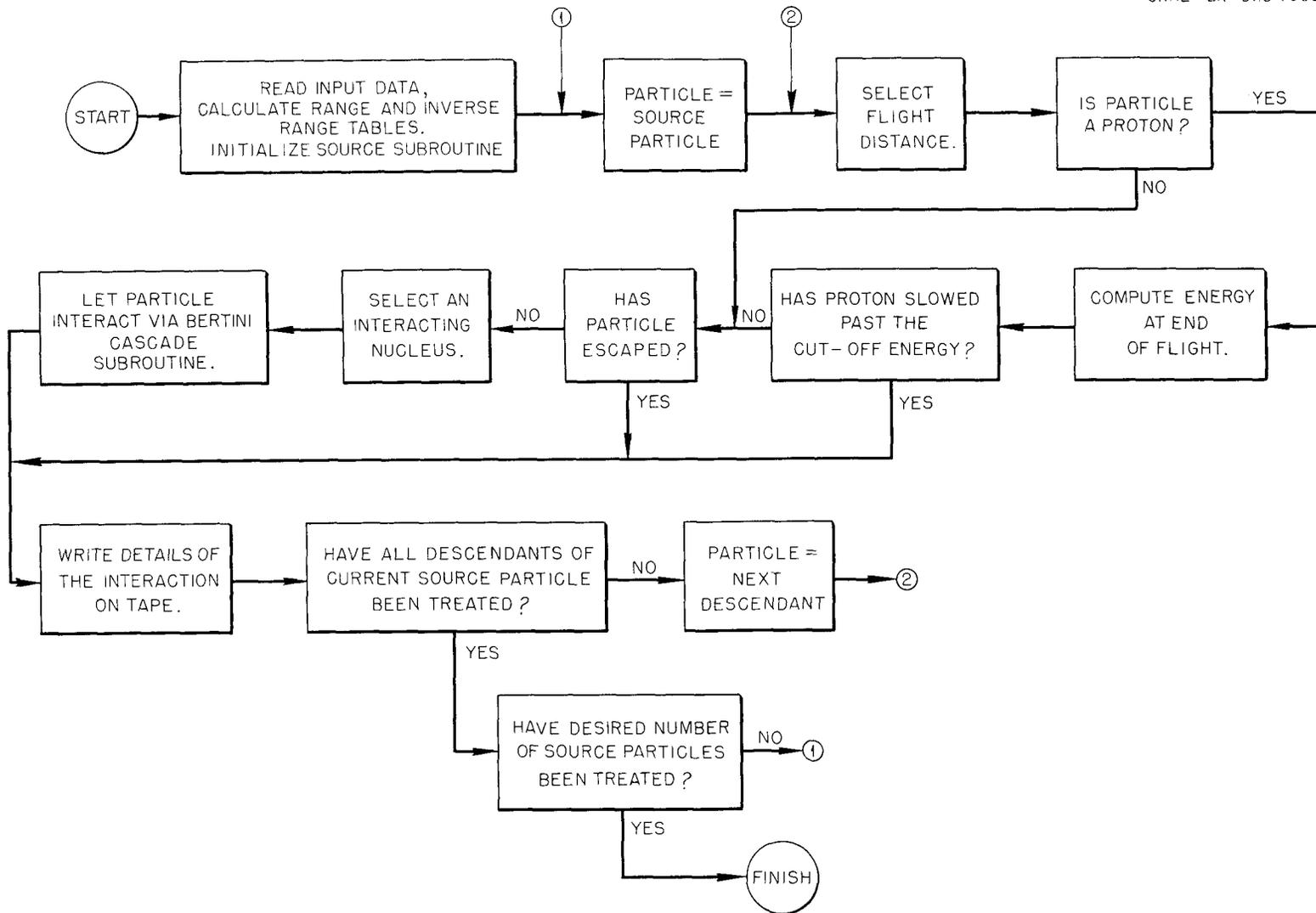


Fig. 2.2. Flow Chart of the High Energy Transport Code

proton, its energy at the end of the flight is computed by interpolation in the inverse range table. If the proton has crossed the lower energy limit, the position at which it crosses is recorded on the history tape and the particle is no longer considered by the code.

If the particle has neither slowed down below the minimum energy nor escaped, it may suffer a nuclear interaction. The j th nucleus is selected as the target nucleus by the requirement that

$$\sum_{i=1}^{j-1} (\Sigma_i/\Sigma) < R < \sum_{i=1}^j (\Sigma_i/\Sigma) ,$$

where R is a random number uniform on the unit interval, Σ_i is the i th isotope geometric cross section, and

$$\Sigma = \sum_i \Sigma_i .$$

If the target is not hydrogen, Bertini's cascade subroutine BERT⁴ is given the energy and type of incident particle and the atomic number and weight of the target. The subroutine calculates and returns the number, types, energies, and direction cosines of the cascade products. The energy variation of the interaction cross sections and the possibility of the passage of the particle through the nucleus without any interaction are both considered by this subroutine.

Collisions with hydrogen are treated by storing the n,p and p,p cross sections as functions of energy. A random number R is tested against $[\sigma_{n,p}(E)/\sigma_{\max}]$ or $[\sigma_{p,p}(E)/\sigma_{\max}]$ for incident neutrons or protons, respectively, where $\sigma_{x,p}$ is the microscopic cross section at energy E and σ_{\max} is the microscopic pseudogeometric cross section for hydrogen. If R is less than the ratio, a scattering takes place as an apparent cascade, with

4. H. W. Bertini, Monte Carlo Calculations on Intranuclear Cascades, ORNL-3383 (April 23, 1963).

two particles emitted. If R is greater than the ratio, the particle continues with no collision.

One version of the High-Energy Transport Code accounts for multiple Coulomb scattering (of the primary beam only) by means of Fermi's theory.⁵ A "Coulomb distance" for application of the deviation formulas is arbitrarily taken as 1/10 of the range at the current energy of the particle. However, the Coulomb distance must be greater than 1 cm.

When a nuclear collision occurs, the following details are recorded on the history tape:

1. the type of colliding particle (neutron or proton),
2. the position, energy, and velocity of the colliding particle at its birth,
3. the position of the collision and the energy of the colliding particle at collision,
4. the number, types, energies, and velocities of the product nucleons having energies above the lower-energy limit,
5. the number, types, energies, and velocities of the product nucleons having energies below the lower-energy limit,
6. the atomic number and weight and the excitation and recoil energies of the residual nucleus.

In addition to being put on the history tape, the data for the product nucleons having energies above the lower-energy limit are saved in the fast memory so that they may be transported in turn. The colliding particle, of course, disappears and the question is asked whether or not all the descendants of the original source particle having energies above the lower limit have been treated. If they have not, the next descendant becomes the particle of interest, a flight distance is selected, and the entire process is repeated. When all descendants have been treated, a

5. B. Rossi, High-Energy Particles, Prentice-Hall, Englewood, New Jersey (1952), p. 71.

test is made to determine whether or not the desired number of source particles has been considered. If not, another source particle is selected by calling the source routine and the steps following point 1 in the flow chart are repeated.

2.2. The High-Energy Analysis Routine

The High-Energy Analysis Routine uses the history tape, usually called the collision tape, from the High-Energy Transport Code to do four things: (1) to estimate the high-energy nucleon distributions, (2) to complete the intranuclear cascades by evaporating nucleons from the highly excited residual nuclei, (3) to estimate the distribution of protons below the boundary energy, and (4) to prepare an O5R source tape for neutrons having energies below the boundary energy of 50 MeV. A flow chart of the routine is shown in Fig. 2.3.

The analysis is initiated when a record from the High-Energy Transport Code collision tape is read into the memory and an arbitrary "Analysis I" subroutine is called to analyze the particle history and determine its contribution to whatever distributions are being estimated. It is important to emphasize that this subroutine must be arbitrarily written by the user to extract the information required for a particular problem. If the record is that of a particle which escaped from the system, the code obviously has no further interest in the particle. If the record is that of a proton that has slowed down past the lower-energy limit of the high-energy code, a second completely arbitrary analysis subroutine, "Analysis II", is called. If the record is that of a particle that has suffered an interaction, the code determines whether there are any product nucleons below the lower-energy limit and, if there are, treats protons with the Analysis II subroutine and writes the neutron data on the O5R source tape. Finally, particles are evaporated from the excited residual nucleus by calling a subroutine version of the evaporation code constructed by Dresner,⁶ with the resulting protons being channeled to Analysis II and

6. L. Dresner, EVAP - A Fortran Program for Calculating the Evaporation of Various Particles from Excited Compound Nuclei, ORNL CF 61-12-30 (December 19, 1961).

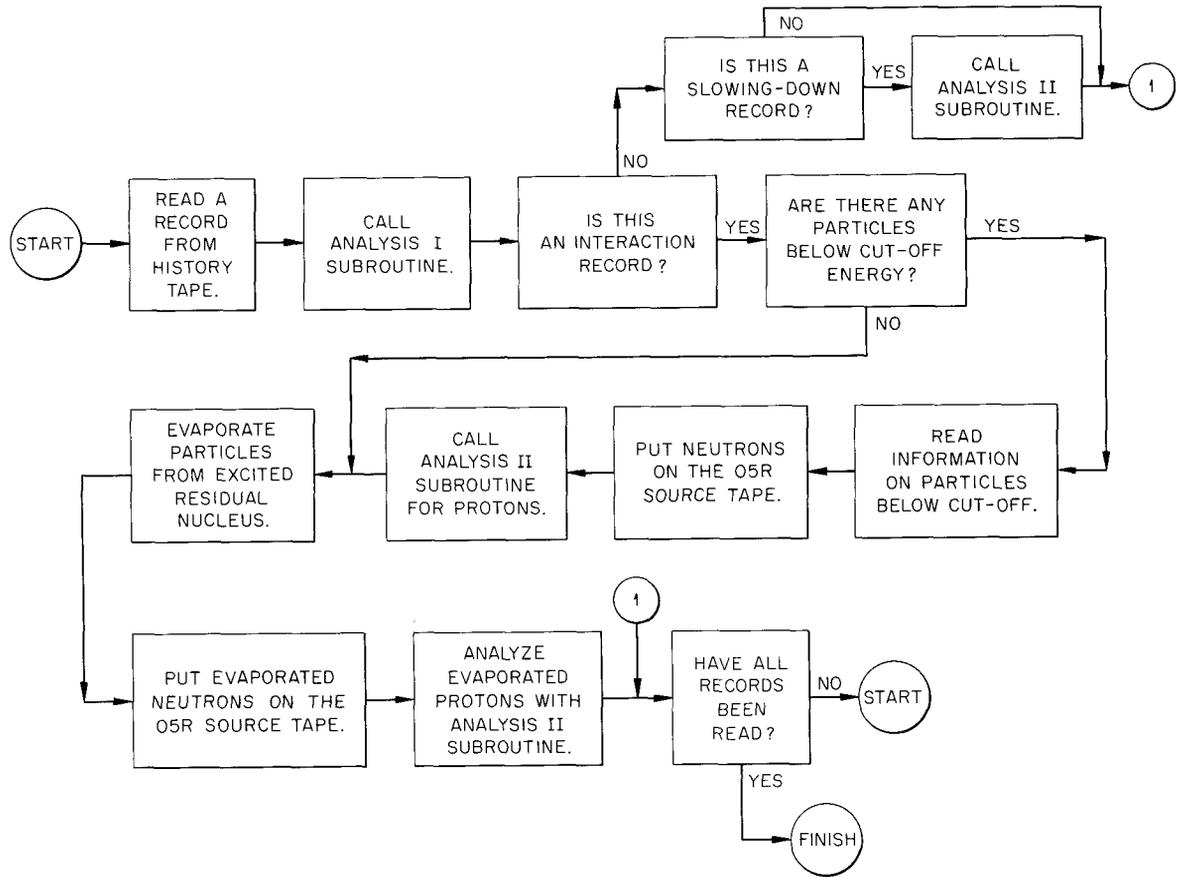


Fig. 2.3. Flow Chart of the High Energy Analysis Code

the neutrons put on the O5R source tape. Provision is also made in the code for recording the spatial distribution of evaporated heavy-particle energy and final residual nucleus energy.

Some details of a typical High-Energy Analysis Routine are given in Appendix A.

2.3. The O5R Code

The O5R code has been described elsewhere.¹ Using the source tape produced by the High-Energy Analysis Routine, O5R traces the neutron histories from 50 MeV to thermal energies, producing a history tape similar to the one produced by the High-Energy Transport Code. Some 34 parameters may be put on this history tape.

2.4. The O5R Analysis Routine

As is the case with the high-energy histories, the analysis routines for the O5R histories are arbitrarily constructed by the user for the particular problem involved. Since the histories can be saved, later re-analysis of the data for additional information is practical.

3. Input to Nucleon Transport Code (NTC)

The following is the input to NTC Chain (1,3) and Chain (2,3) to produce a collision tape designated as logical tape 6.

Chain (1,3) reads in:

Card 1. Format (80H)

A card containing Hollerith information for identification of printed output.

Card 2. Format (ϕ 12)

a. NRAN The starting random number which is stored in 90 of ~~C~~MMON, the position of the current random number.

Card 3. Format (E10.3, E10.3, I10, I10, I10)
 a b c d e

- a. EMAX The maximum energy in the range-energy tables, in MeV; $EMAX \leq 400$.
- b. EMIN The minimum energy in the range-energy tables, in MeV; $EMIN \sim 50$. Particles are not followed below EMIN in the High-Energy Transport Code.
- c. MXMAT The maximum number of media in the system ($MXMAT \leq 4$).
- d. MAXCAS The number of source particles in each batch.
- e. MAXBCH The number of batches to be run. The user may or may not wish to make use of this number.

The media must be specified in order to compute cross sections and range-energy tables. For each medium cards 4a and 4b must be supplied.

Card 4a. Format (E10.3, I10)
 a b

- a. DENH(M) The hydrogen nuclear density $\times 10^{-24}$ for the Mth medium.
- b. NEL(M) The number of elements in medium M with mass > 1 ; $NEL(M) \leq 10$.

Card 4b. Format (E10.3, E10.3, E10.3)
 a b c

- a. Z(I,M) The charge number of the Ith isotope in the Mth medium.
- b. A(I,M) The mass number of the Ith isotope in the Mth medium.
- c. DEN(I,M) The nuclear density $\times 10^{-24}$ of the Ith isotope in the Mth medium.

The index I runs from 1 to NEL(M); the index M from 1 to MXMAT.

Chain (2,3) reads in:

1. The input for the geometry routine, GEOM. This input is described in the ϕ 5R manual in Appendix D.
2. The input for the source routine, SORS, if any.

SORS is a subroutine written by the user which gives the starting parameters for the source nucleons. It is called by the statement:

```
CALL SORS(M,TIP,X,Y,Z,E,SSQ,U,V,W,WT)
```

where

M = 0 to initialize the source routine (one time only)
= 1 if source data is to be provided,

TIP = 0. if particle is a neutron
= 1. if particle is a proton,

X,Y,Z = starting coordinates, in cm,

E = particle energy, in MeV,

SSQ = (speed)² of the particle, in cm²/sec²,

U,V,W = velocity components of the particle, in cm/sec,

WT = weight of the source particle (usually 1.).

CAUTION: Do not start source particles on GEOM zone or block boundaries
(see GEOM specifications in Appendix D).

Chain (2,3) transports MAXCAS source particles and their products, writing the history tape described below on logical tape 6. It then calls Chain (3,3) which is the start of the arbitrary analysis routines.

4. High-Energy Collision Tape

The high-energy collision tape may be analyzed for the desired nucleon distributions between the upper- and lower-energy limits. The cascade process is not complete, however, and subroutine WEK, described in Sec. 5, must be used to evaporate additional particles from the excited residual nuclei. Neutrons with energies below the lower-energy limit may be counted

and reported on magnetic tape to enable the preparation of the O5R input.

The high-energy collision tape format is as follows:

1. NØCAS The number of the source particle which initiated the current cascade. Source particles are numbered from 1 through MAXCAS.
2. NCØL An integer indicating the kind of event:
 - 1 = nuclear interaction,
 - 2 = escape from the system,
 - 3 = slowing down of a proton below EMIN,
 - 4 = escape from a region,
 - 5 = nuclear interaction with no products resulting,
 - 7 = Coulomb interaction (primaries only).
3. NAME The number of the particle considered in this event. NAME starts with 1 for each source particle and increases by 1 for each additional product particle in the cascade.
4. TIP The type of particle:
 - 0. = neutron,
 - 1. = proton.
5. E The energy of the particle at birth, in MeV.
6. SSQ The (speed)² of the particle at birth, in cm²/sec².
7. U } The velocity of the particle in the x, y, and z
8. V } directions, respectively, in cm/sec.
9. W }
10. X } The position coordinates of the particle at birth,
11. Y } in cm.
12. Z }
13. WT The weight of the particle.

14. THEFT The mean free (geometrical) flight time at energy E , equal to $1/(\Sigma s)$, where Σ is the macroscopic geometrical cross section and s is the speed.
15. XC }
16. YC } The position coordinates of the event, in cm.
17. ZC }
18. EC The energy of the particle at the start of the event.
19. SSQC The (speed)² of the particle at the start of the event.
20. NMED The number of the medium in which the event occurred.
21. NAB ϕ V The number of cascade products having energies above EMIN.
22. NL ϕ The number of cascade products having energies below EMIN.
23. ZPR The charge number, Z , of the residual nucleus.
24. APR The mass number, A , of the residual nucleus.
25. EX The excitation energy of the residual nucleus, in MeV.
26. EREC The recoil energy of the residual nucleus, in MeV.

Quantities 21 through 26 are nonzero only if $NC\phi L = 1$ or 5.

If a nuclear interaction takes place and $NAB\phi V$ is greater than zero, a record of the products having energies greater than EMIN is given in the following form:

NAME(I) The name of the product.

TIP(I) The type of particle:

0. = neutron,

1. = proton.

E(I) The energy of the particle, in MeV.

SSQ(I) The (speed)² of the particle, in cm^2/sec^2 .

U(I) }
V(I) } The velocity components in the x, y, and z directions,
W(I) } in cm/sec.

WT(I) The statistical weight of the particle.

The index I runs from 1 to NAB ϕ V.

If particles having energies below EMIN are produced(NL ϕ is greater than zero), they are reported on one record as follows:

TIP(I) The type of particle:

0. = neutron,

1. = proton.

E(I) The energy of the particle, in MeV.

ALPHA(I) } The direction cosines of the particle flight, referred to
BETA(I) } the coordinate system x'', y'', and z''. In this system the
GAMMA(I) } z'' axis is coincident with the velocity vector of the parent
particle. The relation between the x'', y'', z'', and
x, y, z axes is shown in Fig. 4.1.

The index I runs from 1 to NL ϕ .

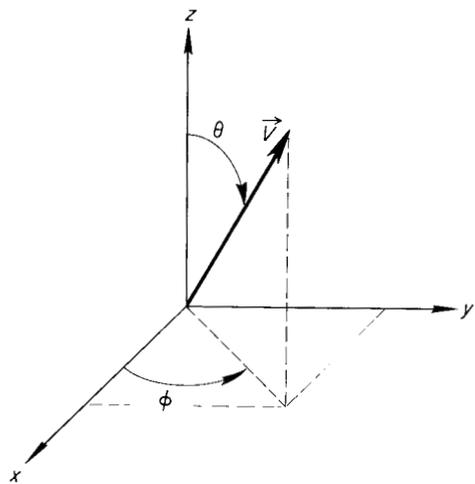
5. Range-Energy Tables

The range-energy tables were briefly mentioned in connection with the High-Energy Transport Code (Sec. 2.1). These tables are retained in COMMON for use in analyzing the high-energy proton distributions. A set of two tables is prepared for each medium, a range table and an inverse-range or energy table. Each range table entry is the distance travelled by a proton of specified energy before its energy is reduced to EMIN by ionization losses. The ranges are stored in descending order for 101 equally spaced values of energy, beginning with EMAX and ending with EMIN. To illustrate the use of the range table, assume a proton of energy E lying between EMAX and EMIN. The distance the proton can travel before its energy reaches EMIN can be computed by the following FORTRAN statements:

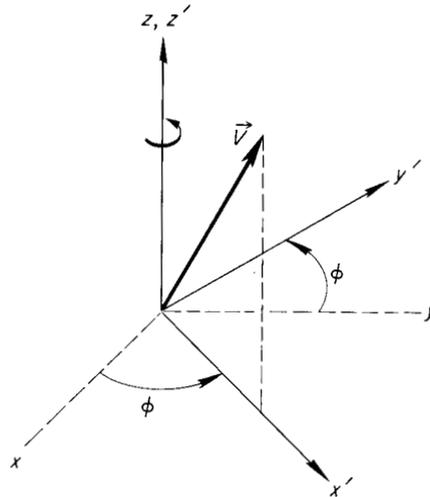
$$I = (EMAX - E) / DELER + 1.$$

$$FI = I$$

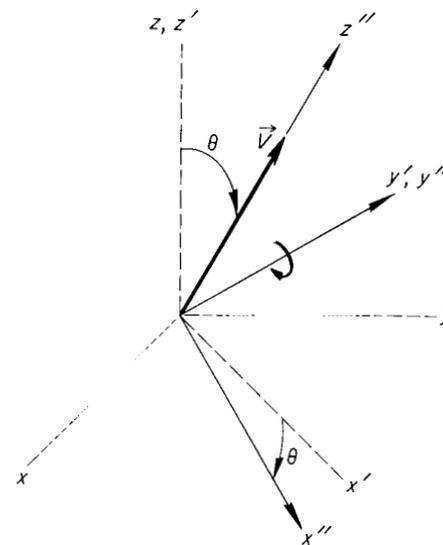
$$DIST = RNGE(I + 1, M) + (RNGE(I, M) - RNGE(I + 1, M)) * \\ (E - EMAX + FI * DELER) / DELER$$



CONSIDER THE VELOCITY OF THE
 INCIDENT NUCLEON \vec{V} WITH POLAR
 ANGLE θ AND AZIMUTHAL ANGLE ϕ .



ROTATE ABOUT z AXIS AN ANGLE
 ϕ . \vec{V} NOW LIES IN $x'-z'$ PLANE.



ROTATE ABOUT y' AXIS AN ANGLE
 θ . \vec{V} IS NOW COINCIDENT WITH z'' AXIS.

Fig. 4.1. Transformation from x, y, z to x'', y'', z'' Coordinate System.

where

DELER = the width of the uniform energy interval in the range table
(DELER = (EMAX - EMIN/100),

RNGE(I,M) = the Ith entry in the range table for medium M.

The inverse-range or energy table entries are the energy a proton must have in order to travel a given distance before ionization losses reduce its energy to EMIN. The inverse ranges are stored in descending order at 101 equally spaced values of the range beginning with the range at EMAX and ending with zero. As an illustration of the use of the inverse-range table, consider a proton whose range to EMIN is R. Its energy can be computed by means of the following FORTRAN statements:

$$I = (\text{RNGMX}(M) - R) / \text{DELRNG}(M) + 1.$$

$$FI = I$$

$$\text{ENERGY} = \text{ER}(I + 1, M) + (\text{ER}(I, M) - \text{ER}(I + 1, M) * (R - \text{RNGMX}(M) + FI * \text{DELRNG}(M)) / \text{DELRNG}(M)$$

where

RNGMX(M) = the range at EMAX in medium M,

DELRNG(M) = the width of the range interval in the inverse-range tables for medium M (= RNGMX(M)/100),

ER(I,M) = the Ith entry in the inverse-range table for medium M.

In addition to the range-energy tables, a number of other quantities which may be of interest are stored in ~~COMMON~~. A complete listing and description of the ~~COMMON~~ storage is given in Table 1.

6. Subroutine WEK

Subroutine WEK, which calculates the evaporation of particles from the excited residual nuclei, is the evaporation program of Dresner⁶ put into subroutine form so that it may be used in an arbitrary analysis code to complete the cascades. When used in the NTC code, the WEK data are stored on logical tape 1 after the 19 records required for the cascade data; therefore, a 19-record skip on the tape is necessary to enter the

Table 1. List and Description of Quantities
Stored in COMMON

Position in COMMON	Quantity	Dimension	Description
1-85	DUMMY	85	Used by GEOM
86	EMAX	1	See input specifications
87	EMIN	1	See input specifications
88	MXMAT	1	See input specifications
89	MAXCAS	1	See input specifications
90	NRAN	1	See input specifications
91	MAXBCH	1	See input specifications.
92	NBCH	1	Batch counter. Set initially to 00. Not used by Chain (1,3) or Chain (2,3).
93	DELER	1	Width of the uniform energy interval in the range tables. It is the same for all media.
94-97	DENH(M)	4	See input specifications.
98-101	NEL(M)	4	See input specifications
102-141	Z(I,M)	(10,4)	See input specifications.
142-181	A(I,M)	(10,4)	See input specifications.
182-221	EION(I,M)	(10,4)	The average ionization potential for the <u>I</u> th element in the <u>M</u> th medium. It is calculated by $\begin{aligned} EION(eV) &= 9.0 \times Z, & Z \geq 30; \\ &= Z[12.8 - 3.8(Z - 5)/25], & 5 \leq Z < 30; \\ &= 59.0, & Z = 4; \\ &= 34.5, & Z = 3; \\ &= 44., & Z = 2; \\ &= 17.5, & Z = 1. \end{aligned}$
222-261	DEN(I,M)	(10,4)	See input specifications
262-265	SIGT(M)	4	Macroscopic geometric total cross sec- tion for medium M.
266-305	SIG(I,M)	(10,4)	Ratio of macroscopic geometric total cross section for <u>I</u> th element in medium M to SIGT(M).
306-309	SIGH	4	See input specifications.

Table 1 (cont.)

Position in COMMON	Quantity	Dimension	Description
310	HSIGMX	1	Hydrogen n,p cross section at EMIN
311-314	RNGMX(M)	4	Range at EMAX for medium M
315-318	DELRNG(M)	4	Range interval in inverse-range tables for medium M.
319-722	RNGE(I,M)	(101,4)	Range tables: I = entry; M = medium
723-1126	ER(I,M)	(101,4)	Inverse-range tables: I = entry; M = medium

WEK data. This skip cannot be a FORTRAN skip since the cascade data is neither written nor read in FORTRAN. A subroutine called SKIP has been written in FAP to perform the skip. The WEK data is read into COMMON where it is used by some additional subroutines called by WEK. The COMMON must be assigned by means of appropriate DIMENSION and COMMON statements in WEK, DOST, WAPTB, and ENERGY to preserve other needed COMMON. In the High Energy Analysis Routine discussed in Appendix A, the WEK data starts in 1947 of COMMON.

WEK is called by the statement:

```
CALL WEK(MI,NAPR,NZPR,EX,M4,NH,NPART,EPART,SUM,U)
```

where

MI = 1 if the subroutine is being entered for the first time
= 2 for all subsequent entries.

If MI = 1, input tables are read from logical tape 1. The other parameters in the call statement are defined as:

NAPR = The mass number of the residual nucleus (an integer).

NZPR = The charge number of the residual nucleus (an integer).

EX = The excitation energy, in MeV.

M4 Not used at present.

NH = The number of histories, set equal to 1 for this application. (WEK may be used independently of the NTC code to estimate the probability of (n,n'), (n,p), (n,2n), and similar reactions by setting NH equal to the number of trials desired for the estimation.)

NPART = a dimensioned array indicating number of particles produced in NH trials:

NPART(1) = number of neutrons,

NPART(2) = number of protons,

NPART(3) = number of deuterons,

NPART(4) = number of tritons,

NPART(5) = number of He³ nuclei,

NPART(6) = number of alpha particles.

EPART = A two-dimensional array, EPART(100,2), indicating the energy of the Ith evaporated nucleon, in MeV.

EPART(I,1) = neutron (index I goes from 1 to NPART(1)),

EPART(I,2) = proton (index goes from 1 to NPART(2)).

SUM = The sum of the kinetic energies of all of the charged particles.

U = The energy remaining in the residual nucleus after all of the particles have been evaporated. For an individual history, U may be negative because of the statistical nature of the calculation. If a residual nucleus can evaporate a particle, it will. The particle, however, may come off with a kinetic energy greater than the excitation energy less the binding energy (since no test is made on the kinetic energy), thus yielding a negative residual energy. The residual energy averaged over many histories, however, will be positive.

7. Ø5R Source Preparation

Neutrons having energies below the boundary energy EMIN result from both direct interactions of incident nucleons within the nucleus and evaporation from a highly excited compound nucleus. The required parameters of these neutrons are put on tape to form a source for Ø5R treatment with one FORTRAN record per neutron. Each record is assumed to have the form:

1. NEUTINØ The number of the neutron. A negative value of NEUTINØ is a convenient way to signal the end of the source tape.
2. E The neutron energy, in eV.
3. SSQ The (speed)² of the neutron, in cm²/sec².
4. U }
5. V } The velocity components of the neutron, in cm/sec.
6. W }

- 7. X
 - 8. Y
 - 9. Z
10. WT The statistical weight of the neutron.

The position coordinates of the neutron, in cm.

A complete user's manual for $\phi 5R$ is given in Appendix D. The user has considerable liberty in choosing the format of the collision tape which is produced by $\phi 5R$, and complete freedom in constructing the necessary analysis routines.

Appendix A. Typical High-Energy and Ø5R Analysis Routines

In the preceding discussion it has been emphasized that the primary functions of both the High-Energy Transport Code and the Ø5R code are to produce particle histories which can be arbitrarily analyzed by the user. This flexibility may be more clearly demonstrated by an examination of some portions of an analysis routine specifically written for infinite-slab geometry. This routine assumes that source nucleons are run in batches, and it calls Chain (5,3) to process the data batch by batch, thus furnishing a means for estimating statistical errors. The main routine merely allows for the tabulation of the energy deposited locally by evaporated particles with $A > 1$ and the energy of the residual nucleus after evaporation. It then calls the ANAL subroutine to analyze the nucleon distributions above EMIN and the ANAL 2 subroutine to analyze the proton distributions below EMIN. Chain (3,3) computes range-energy tables from 1 MeV to EMIN.

ANAL estimates the angular distributions of uniformly wide energy groups of nucleons escaping from the rear faces of as many as four infinite slabs which are integral subdivisions of the infinite slab for which the high-energy transport calculation was made. The uniform energy group width is arbitrarily specified. The number of nucleons in each group is reported as a function of arbitrarily wide (not necessarily equal) intervals of the cosine of the angle between the normal to the slab and its rear face, in units of $\text{number} \cdot \text{MeV}^{-1} \cdot (\text{unit cosine})^{-1} \cdot (\text{incident source nucleon})^{-1}$. The statistical error attached to each estimate is also computed. In addition, a summation over all angles is made to give an estimate of the energy spectrum of nucleons emerging from the rear face of each subslab. ANAL also permits study of the spread of the beam as it proceeds through a medium by estimating the variation with radius r of the fraction of particles emerging from the rear face of the slab within a circle of radius r centered at the intersection of the velocity vector of the primary beam with the slab face. Finally, an estimate of the number of neutrons with energies greater than EMIN emerging from the rear faces of the subslabs per incident source particle and an estimate of the total number of neutrons with energies greater than EMIN escaping from both front and rear faces of the subslabs per incident source particle are reported.

ANAL 2 estimates the angular distributions of uniformly wide energy groups of protons having energies below EMIN within the subslabs, in the hope that distributions of such protons emerging from the slabs can be sufficiently well determined from these data. The proton angular flux is reported as a function of arbitrary intervals of the cosine of the angle between the normal to the slab and its face, in units of $\text{cm}\cdot\text{cm}^{-3}\cdot\text{MeV}^{-1}\cdot(\text{unit cosine})^{-1}\cdot(\text{incident source particle})^{-1}$. Integration over all angles gives the energy-dependent flux.

Input to the main routine, called for first, and input to ANAL and ANAL 2 are described in detail below.

Al.1. Input to Main Routine

 a b c d
Format (E10.3,I10,I10,I10)

- a. XCAP Thickness of the infinite slab, in cm.
- b. JMAX The total number of subslabs in which the energy deposited by evaporated particles with $A > 1$ and the residual energy of the nuclei after evaporation are to be estimated.
- c. NREC The number of input cards from the GEOM input to the end of the input, inclusive.
- d. NRECL The number of input cards from this card to the end of the input, inclusive.

NREC and NRECL are used to hunt back on the input tape in running successive batches.

Al.2. Input to ANAL

 a b c d
Card 1. Format (E10.3,I10,I10,I10)

- a. T The slab thickness, in cm.
- b. NX The number of subslabs into which the infinite slab is to be divided; $NX \leq 4$. All quantities are computed at the rear for each subslab. For a 10-cm-thick slab and $NX = 4$, for

example, the leakage spectra, angular distributions, etc., of nucleons emerging from 2.5-, 5.0-, 7.5-, and 10-cm-thick slabs will be computed.

- c. NR The number of radial boundaries for the computation of spreading; $NR \leq 16$. See below.
- d. NA The number of angular groups in the computation of angular distribution; $NA \leq 10$.

Card 2. Format (7E10.3)

FMU(I) The cosines of the lower limit of the angular groups with cosines in increasing order.

I=1,NA It is assumed that the upper limit of group 1 is 90° ($\cos^{-1} = 0$).

Card 3. Format (7E10.3)

R(I) The upper boundaries of the Ith radial groups in increasing order.

I=1,NR The spreading is computed at each subslab and is reported as the fraction of the total number of particles emerging from the slabs lying within R.

Card 4. Format (E10.3, E10.3, I10)

- a. EHI The upper energy for the energy spectrum calculation, MeV.
- b. EL \emptyset The lower energy for the energy spectrum calculation, MeV.
- c. NE The number of (equal) energy intervals between EHI and EL \emptyset .

A1.3. Input to ANAL2

Card 1. Format (I10, I10, I10, E10.3)

- a. NX The number of subslabs. See ANAL input.
- b. NE The number of energy groups below EMIN for the calculation of proton spectra; $NE \leq 5$.

- c. NA The number of angular groups in the computation of angular distribution; $NA \leq 10$.
- d. T The slab thickness, in cm.

Card 2. Format (7E10.3)

E(I) The lower-energy limit, in MeV, of the Ith group in descending order with EMIN as the upper limit for group 1.

Card 3. Format (7E10.3)

FMU(I) The cosine of the lower limit of the angular group.

I=1,NA Same as in Card 2 of ANAL input.

The neutrons which appear with energies below EMIN are put on the ϕ 5R source tape and are transported by ϕ 5R until they either leak from the system or are killed by Russian Roulette. The ϕ 5R Analysis Routine estimates the angular distributions of arbitrarily wide energy groups of neutrons having energies below EMIN which emerge from the rear faces of the subslabs mentioned in the description of ANAL. The number of neutrons, in units of $\text{number} \cdot \text{Mev}^{-1} \cdot (\text{unit cosine})^{-1} \cdot (\text{high-energy source particle})^{-1}$, is reported in the same format as the nucleon analysis of ANAL. The angular groups are summed over all angles to give an energy spectrum and, again as in ANAL, an estimate is made of neutrons emerging from the rear faces of the subslabs as well as of neutrons emerging from both front and rear faces of the subslabs, both normalized to unit incident high-energy source particle.

The ϕ 5R collision parameters are NC ϕ LL, NAME, X1, U ϕ , X ϕ , WT ϕ , λ , and SO, and the first collision tape is logical tape 1.

A1.4. Input to ϕ 5R Analysis Code

Card 1. Format (I10,I10,E10.3,E10.3,F10,F10,E10.3)

- | | | | | | | | |
|--|---|---|---|---|---|---|---|
| | a | b | c | d | e | f | g |
|--|---|---|---|---|---|---|---|
- a. NX The number of subslabs into which the slab is subdivided (same as NX in ANAL input).
 - b. NA The number of angular groups (same as NA in ANAL input).

- c. TLIM1 The upper-energy limit, in MeV, of the first major energy group for energy spectra calculation.
- d. TLIM2 The upper-energy limit, in MeV, of the second major energy group for energy spectra calculation.
- e. DL1 The energy group width, in MeV, of the minor energy groups in the first major energy group.
- f. DL2 The energy group width, in MeV, of the minor energy groups in the second major energy group.
- g. T ϕ T The total number of source nucleons, i.e., MAXCAS*MAXBCH, which gave rise to the neutrons in ϕ 5R. This value is used to normalize results to unit source strength.

The energy spectra are calculated for groups of width DL1 from 0 to TLIM1 and for groups of width DL2 from TLIM1 to TLIM2. The total number of energy groups must be equal to or less than 22.

Card 2. Format (7E10.3)

FMU(I) The same quantity as FMU(I) in ANAL; I = 1,NA.

Card 3. Format (E10.3)

T The thickness of the slab, in cm.

Note: This analysis routine assumes the following quantities to be written for each collision: NCOLL, NAME, X1, UO, XO, WTO, LAMBDA, and SO. It is therefore required that item C on card I of the O5R input (see Appendix D) have the value 11000010000100100100000000000110.

Appendix B. Tape Assignments

The tape assignments for running the High-Energy Transport Code of NTC are as follows:

<u>Log. Tape No.</u>	<u>Absolute Assignment</u>	<u>Function</u>
1	A5	Contains BERT and WEK data.
2	Channel B	Scratch tape used by BERT subroutine.
6		High-energy history tape.
9		Standard output tape.
10		Standard input tape.
-	B3	Chain tape.

Only the above are used for running Chain (1,3) and Chain (2,3). If Chain (3,3), Chain (4,3), and Chain (5,3) of Appendix A are also used, the additional tapes listed below are required:

<u>Log. Tape No.</u>	<u>Function</u>
7	Scratch tape to store batch results.
8	Ø5R neutron source tape.

Appendix C. Routines in the High-Energy
Nucleon Transport Code

<u>Routine</u>	<u>Language</u>	<u>Function</u>
		<u>Chain (1,3)</u>
MAIN	FØRTRAN	Reads some input, computes and stores cross sections, sets up range tables.
GTHSIG	FØRTRAN	Given the energy and type of particle, finds the proton scattering cross section.
NPSIG	FAP	Sets the cross sections for n,p scattering at the energies which bracket E in GTHSIG. Cross sections exist for every 10 MeV from zero.
PPSIG	FAP	Sets the p,p cross sections at the energies which bracket E in GTHSIG.
READ	FAP	Reads BERT data tape on channel A5 and picks off geometric cross sections for A = 1 to 239.
RANGE	FØRTRAN	Computes and stores range and inverse range tables.
INC	FØRTRAN	Computes dx/dE at energy E for use in "RANGE."
ZFØI	FØRTRAN	Computes effective ionization potential.
		<u>Chain (2,3)</u>
MAIN	FØRTRAN	Reads geometry and source input data, transports nucleons, and writes history tape on logical tape 6.
JØSET	FAP	Establishes the storage for the GEØM routine and transfers to the appropriate entry in the GEØM routine to read in the geometry specifications.
SØRS	FØRTRAN	Reads in source input data if first argument is 0 or provides data for a source particle if the first argument is 1.

GØMSØR	FØRTRAN	Calls LØØKZ to set necessary quantities in GEØM.
ECØL	FØRTRAN	Given the range, computes the energy of a proton.
SPREAD	FØRTRAN	Computes Coulomb small-angle scattering.
GAURN	FØRTRAN	Picks from a Gaussian distribution.
GØMPRP	FØRTRAN	Sets up and calls GEØM for a particle flight to see if particle has escaped from region.
CASCAD	FØRTRAN	Selects a target nucleus and calls the necessary routines to perform the nuclear interactions or, in the case of hydrogen, the elastic scattering.
GØHSIG	FØRTRAN	} Same as in Chain (1,3)
NØSIG	FAP	
PPSIG	FAP	
PSCAT	FØRTRAN	Computes n,p or p,p elastic scattering.
BERT	FAP	Performs the intranuclear cascade.
RECØLL	FØRTRAN	Computes recoil energy of residual nucleus.
SPDSQ	FØRTRAN	Given an energy, computes a speed squared.
RANDØM PACKAGE	FAP	Selects random variables from various distributions.
GEØM PACKAGE	FAP	See Appendix D.
BERT	FAP	Performs intranuclear cascade calculation.
<u>Chain (3,3)</u>		
MAIN	FØRTRAN	Computes range-energy relationships for various media in the energy range from 1 to 50 MeV.
RANGE	FØRTRAN	} Same as Chain (1,3)
INC	FØRTRAN	

Chain (4,3)

MAIN	FØRTRAN	Calls ANAL and ANAL2 to put batch results on logical tape 7. Puts Ø5R source neutrons on logical tape 8. Decides if all batches are run. Assumes slab geometry.
SKIP	FAP	Hunts 19 records on A5 past BERT data to WEK data.
WEK	FØRTRAN	Evaporates particles from excited nuclei.
ANAL	FØRTRAN	Analyzes nucleon distributions above cutoff energy.
ANAL2	FØRTRAN	Analyzes proton distributions below cutoff energy.
RAMDØM PACKAGE	FAP	Same as Chain (2,3).

Chain (5,3)

MAIN	FØRTRAN	Reads in, from logical tape 7, results of Chain (4,3) analyzes batch by batch, computes standard deviation and reports results.
ANAL	FØRTRAN	Analyzes the batch results from ANAL in Chain (4,3).
ANAL2	FØRTRAN	Analyzes the batch results from ANAL2 in Chain (4,3).
DØST	FØRTRAN	Returns k_p , C_p , or k_α for given Z (see ref. 6, p 4).
ENERGY	FØRTRAN	Returns either experimental or, if there is none, calculated binding energy for given A and Z.
WAPTB	FØRTRAN	Fills in the gaps in the table of experimental binding energies with calculated values.
ECØL	FØRTRAN	Same as in Chain (2,3).

Appendix D. O5R User's Manual

R. R. Coveyou and J. G. Sullivan

D.1. Introduction

The O5R code, a Monte Carlo neutron transport code for the IBM 7090 computer, has three basic purposes. It may be used for the generation of comparison standards for the evaluation of other methods of computation; it may be used to do research on the capabilities of the Monte Carlo method; or it may be used to do specific reactor physics calculations as comprehensive as the study of complete reactor systems. Principally because of the first purpose, great care has been taken to assure an adequate representation of neutron cross sections. Cross sections may be given at as many as 10^4 energy values.

Speed in the construction of neutron histories is achieved in spite of this volume of cross-section data by constructing many neutron histories in parallel. Neutron interactions are followed in space and energy from birth of the neutron in the system until its disappearance.

The details of every collision experienced by a neutron during its lifetime are recorded on one or several history or collision tapes. These tapes are then available as input to analysis routines specially tailored by the user to calculate the answers to a given problem. The separation of the generation of the histories from their analysis not only allows flexibility in the types of problems which may be treated but also permits re-analysis of histories for additional information.

Part of the O5R system is a geometry routine which can handle all systems whose material boundaries can be described as quadric or plane surfaces, the single restriction being that curved boundaries cannot intersect. As many as six different material media are allowed, with each medium allowed up to seven scatterers. Any scatterer may scatter anisotropically in the center-of-mass system.

The O5R source specification permits variation in space, energy, and initial direction of the neutron. In addition, in fissioning systems the neutrons produced by one fission generation may be used as the source for the succeeding generation.

D.2. Definitions

The definitions listed below will be found helpful in understanding the discussions of the various features of O5R which follow. Other terms are defined as they are encountered in the text.

- Batch: Some number (1-1000) of neutron histories handled in parallel.
- Run: One or more batches.
- FPI: Fortran-type integer.
- History: The details of the interactions of a neutron from its appearance in the system until its interactions are no longer of interest.
- Collision Tape: A tape or tapes containing all the histories from one run.
- AC: After the collision.
- BC: Before the collision.
- Medium: The element or compound within which the flight of the neutron is to take place. Media are numbered and identified by NMED, FPI's beginning with 1.
- Region: A subdivision of the volume of interest, primarily used in the application of weight standards associated with the position of the neutron. Only one region is permitted in the O5R portion of the Nucleon Transport Code.
- Supergroup: A division of the neutron energy range, having boundaries a factor of two apart. The energy range encompassed by O5R, from 77 MeV to 10^{-4} eV, is divided into 40 supergroups. Supergroups are numbered consecutively, with the highest supergroup on the reactor data tape being numbered 1.
- Element: The particular scatterer with which a neutron collision takes place. ELEM for each medium begins with 1 and follows the medium element specifications used in preparing the reactor data tape.

- Name: A FTI (NAME_n) identifying the n th neutron, beginning at 1 for each batch.
- s_n^2 : The energy of the n th neutron; the energy is measured in units such that it is numerically equal to the square of the speed in cm/sec.
- u_n, v_n, w_n : The velocity vector in the laboratory system of coordinates associated with the n th neutron; $u_n^2 + v_n^2 + w_n^2 = s_n^2$.
- x_n, y_n, z_n : The position of the n th neutron.
- $\theta(s^2, m)$: The mean free flight time for a neutron of energy s^2 in medium m . $\theta = 1/s\Sigma_{T_m}$, where Σ_{T_m} is the macroscopic total cross section in medium m .
- η : The number of mean free paths to be travelled to a collision. It is a random variable drawn from an exponential distribution.
- $\psi(s^2, m)$: The nonabsorption probability for a neutron of energy s^2 in medium m .
- $P_l(s^2, m, l)$: The probability that a neutron with energy s^2 will have a scattering collision with the l th nuclide in medium m .
- $P_f(s^2, m)$: The probability that a fission neutron of unit weight will be produced by a neutron of energy s^2 and unit weight having a collision in medium m .
- $f_1(s^2, l)$: The average value of the cosine of the scattering angle in the center-of-mass system at energy s^2 for nuclide l .

D.3. General Description of O5R Processes

D.3.1. Construction of Neutron Histories

The procedure followed by O5R in constructing the neutron histories is given below in outline form. The details of the more important operations are discussed in later sections.

1. Each neutron is given a name, a FTI which distinguishes it from every other neutron.
2. An initial speed, s_0 , and direction of the source flight is assigned.
3. The coordinates of the birth position of the neutron, x_0 , y_0 , and z_0 , are assigned.
4. An initial statistical weight, w_{T0} , is assigned to the neutron. See discussion below.
5. A geometry routine, GEOM, determines in which medium the source neutron lies.

The initial conditions having been established, the slowing-down process then proceeds as follows.

6. If weight standards are being applied, the weight of the neutron is tested for "splitting" or "killing."
7. A possible position for the next collision, based upon a consideration of the mean free path in the medium in which the neutron is travelling, is calculated from the relationships

$$x_1 = x_0 + \eta \cdot \theta \cdot u_0 ,$$

$$y_1 = y_0 + \eta \cdot \theta \cdot v_0 ,$$

$$z_1 = z_0 + \eta \cdot \theta \cdot w_0 ,$$

where the subscripts zero denote the origin of the flight path and the subscripts one the possible collision point; η is the number of mean free paths, θ is the mean free flight time in the medium, and u , v , and w are the velocity components of the neutron flight.

If the flight path is entirely in the original medium, then the calculated x_1 , y_1 , and z_1 give the point at which the next collision will occur. If, however, the flight path enters another medium, the position of the nearest boundary crossing along the flight path from x_0 , y_0 , and z_0 is calculated and the number of mean free paths, η , is reduced by the number of paths,

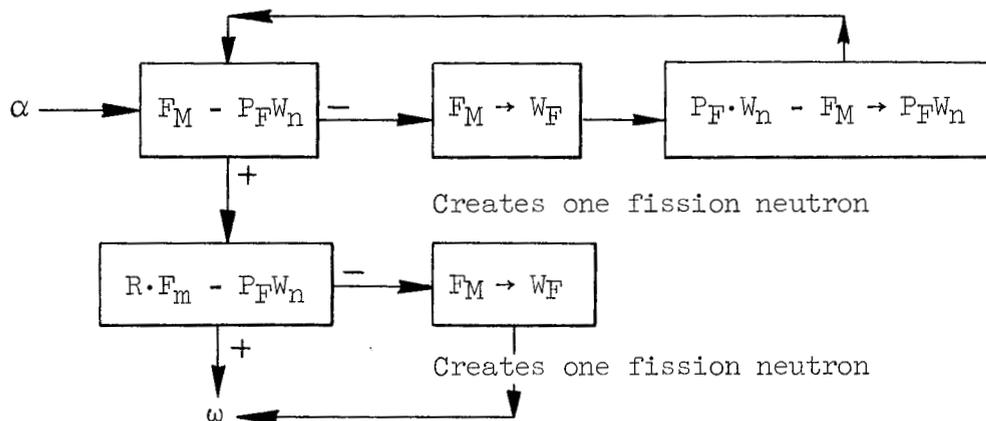
η' , needed to be travelled in order to arrive at the boundary. The coordinates of the boundary-flight path intersection then describe the starting point and the process is repeated, using the reduced η , to find another tentative collision point. Unless the neutron escapes from the system the repetitions continue until a collision point is accepted. A geometry routine of considerable generality calculates the boundary crossing points and performs the necessary bookkeeping.

8. When a collision point is found, a new neutron statistical weight, wT_1 , is computed by multiplying wT_0 by the survival probability, ψ , in the medium of the collision point. This procedure treats every collision as a scattering collision, allowing for absorption by means of the weight change.
9. The nuclide from which the neutron is to be scattered, L , is chosen by considering the scattering cross sections of all of the nuclides of the medium in which the collision occurs.
10. From the mass of nuclide L , the neutron velocity before collision (in the laboratory system) and the average value of the cosine of the scattering angle in the center-of-mass system, f_1 , a new velocity for the neutron in the laboratory system (u_1, v_1, w_1) is selected.
11. The resulting energy of the neutrons, s_1^2 , ($= u^2 + v^2 + w^2$) is compared with an input cutoff energy. If the neutron energy is greater than the cutoff energy, the slowing-down process is continued by repeating the procedures from 6 above. If the neutron energy is below the cutoff, the slowing-down process is ended and the neutron enters the thermal group.

D.3.2. Fissioning

Provision is made for allowing fast and/or thermal fissioning in any medium. At each collision in a medium in which fissioning may occur the code decides whether or not one fission neutron should be introduced at (x,y,z) , the collision position, in the next batch. The weight to be

given the fission neutron in a medium is specified as an input parameter. The process to determine whether or not a fissioning should occur is as follows:



where F_M is the weight to be assigned the fission neutron, P_F is the mean number of fission neutrons produced per collision, R is a random number uniform on the interval $(0,1)$, W_n is the weight of this neutron having the collision, and W_F is the weight of the fission neutron.

The parameters associated with the position and weight of the fission neutrons are stored on a tape and at the completion of a batch the next generation is read from the tape and becomes the source for the new batch. It is also possible simply to count up the number of fissions per batch without using these produced fissions as the source for the next generation. For instance, if the source distribution over the reactor is already known, then the average number of fissions produced by several batches using the known source would give the multiplication constant, k , for this particular source.

D.3.3. Thermal Neutrons

At present a simple one-velocity model is used for treatment of thermal neutrons. For each medium two parameters which characterize the diffusion process must be supplied: the mean free path, θ , for thermal neutrons in that medium, and the survival probability, ψ , for a collision in the same medium. In general, $\theta = 1/\Sigma_T$ and $\psi = \Sigma_S/\Sigma_T$, where Σ_S and Σ_T are the macroscopic scattering and total cross sections for thermal

neutrons in the medium. If, however, the available information includes a value, calculated or experimental, for D , the diffusion constant, and Σ_a , the macroscopic absorption cross section, one can, following diffusion theory, calculate θ and ψ as $\theta = 3D$ and $\psi = 1 - 3 \Sigma_a D$. Note, however, that this prescription is suitable only if $3 \Sigma_a D$ is small.

Finally, if thermal fission is to occur in a medium a third parameter, $\nu \Sigma_f / \Sigma_t$, must be supplied, where ν is the mean number of neutrons per fission and Σ_f is the macroscopic thermal fission cross section.

D.3.4. Neutron Weights

Each neutron, i , of a batch is assigned an initial weight, WT_i (the value normally assigned is 1.), and the relative statistical importance of

this neutron in a batch of neutrons is equal to $WT_i / \sum_{n=1}^N WT_n$. After

each collision the weight of the neutron is multiplied by the probability of its not being absorbed on that collision, ψ . Thus the probability of the i th neutron not having been absorbed after j collisions is $WT_i^j = WT_i^0 \times \psi_1 \times \psi_2 \times \dots \times \psi_j$, where the superscripts on the WT_i represent the collision number.

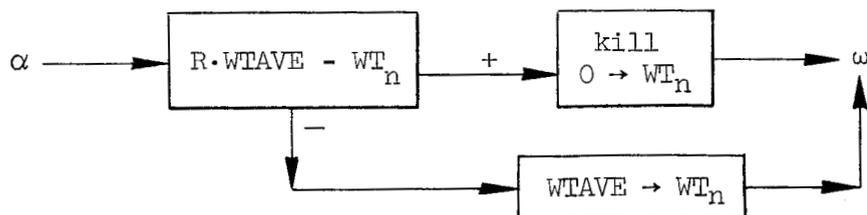
One advantage arising from the use of this method of weighting the neutrons is that if there is an appreciable amount of absorption in the reactor, the variance of the results computed by analysis routines may be reduced from that obtained by allowing absorption to take place. The weighting system also makes it rather simple to concentrate neutrons in a particular volume of the reactor where greater accuracy might be required.

Consider the following example: In a reactor core there are initially 200 neutrons of original weight 1. After the neutrons have had several collisions there might then be 100 neutrons with weight .9, and 100 with weight .01. Let us assume that the spatial position of each neutron does not matter and we are interested in the number of neutrons escaping from the core. It is clear then that if we have to spend an equal amount of computing time on each neutron (regardless of its weight) before it escapes, then the computing time spent on all 100 of the neutrons of

weight .01 yields statistical information which is approximately equal in importance to the information gained from the computing time spent on only one of the high weight neutrons. In other words, about half the computing time would be wasted gaining little additional information. To prevent this sort of imbalance from occurring the use of "weight standards," is encouraged so that the amount of computing time spent on each neutron will produce information approximately equal in statistical value. When called for by certain input parameters, the weight of the neutron after each collision is compared with two numbers, WT_{HIT} and WT_{LOW} , called weight standards. These numbers may be a function of the position (for this purpose, the reactor is divided into geometric regions) and energy group of the neutron. The region boundaries and medium boundaries need not be the same.

If $WT_{HIT} < WT_n$ then the neutron is "split in two," that is, the original neutron has its weight cut in half and another neutron (with a different "NAME" a weight one-half the original, but with all other neutron details the same) is added to the neutron list. These two neutrons then generate independent histories from the position where the original neutron was split (their first flight from the position, however, will be in the same direction).

If $WT_{LOW} > WT_n$ then, as illustrated in the flow chart,



where R is a random number uniformly distributed between 0 and 1, the neutron's weight is either set to zero (the neutron is "killed") or is replaced by an input weight, WT_{AVE} , which is a function of region and energy group. This process, known as "Russian Roulette" is simply a device to make the code run faster and more efficiently by not compelling it to follow neutrons of low statistical weight through the reactor.

NOTE: If few neutrons are expected to escape from a thermal reactor, then the only way in which histories can be terminated is by the neutron weight becoming less than WTL/W , the lowest weight allowed. In running almost all reactors a lower weight standard must be specified for reasons of economy. Determining exactly what this weight ought to be is a matter for inspired guesswork, but probably $1/50$ of the original weight of a source neutron is much too low and, in most problems, $1/20$ of the original weight would be quite adequate.

D.4. Random Variable Selection in O5R

A general-purpose Monte Carlo reactor code such as O5R requires efficient and precise techniques for the selection of random variables from the probability distributions involved. Among the more important techniques required are the following:

1. A fast and completely reliable method for the selection of pseudo-random numbers, i.e., numbers on the unit interval so chosen as to approximate choices of a random variable uniformly distributed on the unit interval. The method chosen for the generation of pseudo-random numbers is the "congruential multiplication" method.

Let P be the integral capacity of the machine; $P - 1$ is the largest integer expressible as a fixed point number by the machine. Let λ be an appropriately chosen integer, known as the generator of the random number sequence. Let X_n be the n th random number in the sequence. Then

$$PX_n \equiv \lambda^n \pmod{P} .$$

For the IBM 7090 machine, $P = 2^{35}$, while λ was chosen to be 5^{15} . It can be shown that this choice of λ leads to a period of 2^{33} for the sequence, that is, $X_1, \dots, X_{2^{33}}$ are distinct, while $X_1 = X_{2^{33}+1}$.

This sequence has been subjected to a very wide variety of statistical tests and no deviation from satisfactory random behavior has been observed. Further, it can be shown analytically that

the correlation between successive members of the sequence is acceptably small.

2. A method for choosing a random free path, the distance traveled by a neutron between two collisions, measured in units of a mean free path. If n is such a random variable, then

$$\text{Prob} (X \leq u \leq X + dX) = e^{-X} dX .$$

It is readily seen that the negative of the logarithm of a random number is so distributed. There is, however, another and very clever technique for generation of such a random variable, due to John von Neumann, which may best be expressed by the flow chart shown in Fig. D.1.

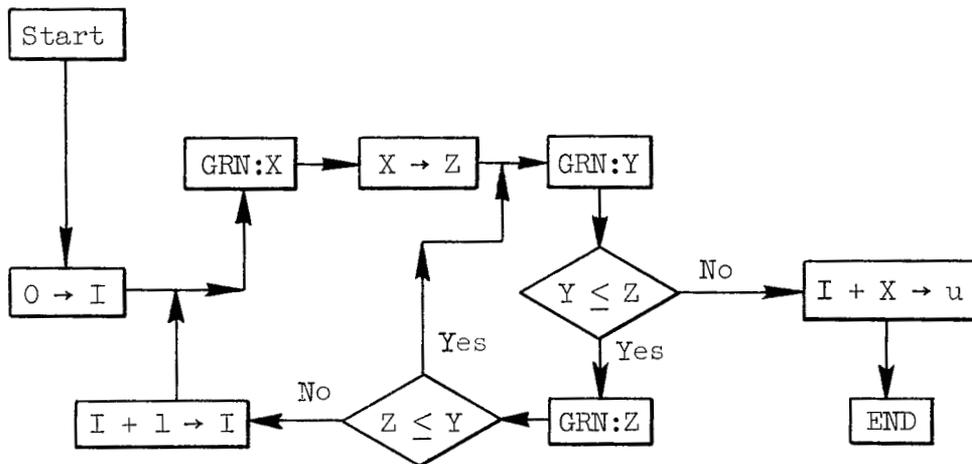


Fig. D.1. Selection of Random Number of Mean Free Paths:

$$\text{Prob} (X \leq u \leq X + dX) = e^{-X} dX; \quad 0 < X < \infty$$

The random-variable subroutines which are available for use in or with O5R are listed in Table D.1.

Table D.1. O5R Random-Variable Subroutines*

Routine	Function
FLTRN	Places in the accumulator a floating-point pseudo-random number uniformly distributed on the interval [0,1].
RNSGN	Gives a random sign to the contents of the accumulator.
FXDRN	Places in the MQ register a fixed-point random number uniformly distributed on the interval [0,1].
EXPRN	Places in the accumulator a floating-point number chosen from the distribution function $P(x) dx = e^{-x} dx$.
AZIRN	Stores in cell 1,4 the sine of an angle α and stores in cell 2,4 the cosine of an angle α where α is uniformly distributed on $(0,2\pi)$. Both numbers are in floating point.
PØLRN	Stores in cell 1,4 the sine of an angle β and stores in cell 2,4 the cosine of an angle β , where $\cos\beta$ is chosen uniformly on the interval $(-1,1)$. Both numbers are in floating point.
FISRN	Places in the accumulator a floating point $(\text{speed})^2$ chosen from the Nerenson-Rosen fission spectrum. The result is $< 3.02 \times 10^{20}$.
MAXRN	Places in the accumulator a floating-point random number from the distribution $P(x) dx = \sqrt{2/\pi} x^2 e^{-x^2/2} dx$ ($0 < x < 17$).
GAURN	Places in the accumulator a floating-point random number from the distribution $P(x) dx = (1/\sqrt{2\pi}) e^{-x^2/2} dx$ ($-15 < x < 15$).

*All subroutines return control to cell 1,4 except AZIRN and PØLRN which return control to cell 3,4.

D.5. Cross Sections

D.5.1. Cross-Section Representation

The O5R code is designed to encompass a neutron energy range extending from 77 MeV to 10^{-4} eV. This range is divided into 40 so-called "super-groups," whose energy boundaries are a factor of two apart, as shown in Table D.2. Each supergroup in turn is divided into 2^n subgroups of equal energy width, where n is equal to or less than 8, at the option of the user. The data are assumed constant within each subgroup. This method of representation reproduces fairly well the general tendency of cross sections to vary rather more regularly with lethargy than with energy and takes into account the fact that within a supergroup it is easier to work with equal energy spacings. With equal spacings the energy of a neutron is, within a linear transformation, the machine memory address of a wanted cross section. Some tests have been performed which indicate that for cross sections which do not fluctuate too abruptly 128 subgroups per supergroup are adequate.

The huge volume of cross-section data implied by the above scheme cannot, of course, be contained in the fast storage of the computer and at first glance it would seem that a continuous demand for cross sections from the magnetic tape storage would inordinately slow the computation. The way around this difficulty, however, is clear. It consists of using the cross sections available in the fast storage with maximum efficiency by constructing many (200-1000) neutron histories in parallel. Each neutron is processed as far as possible by using the supergroup data then in fast storage, then stored until all other neutrons have undergone similar treatment. When all neutrons have been processed through a supergroup, cross sections for a succeeding supergroup are read from tape and the processing repeated.

D.5.2. Cross-Section Handling Code Package

The cross-section package consists of a group of seven numbered codes, performing the functions listed below.

Table D.2. Upper Energy and Speed-Squared Limits of O5R Supergroups

Energy (eV)	Speed ² (cm ² /sec ²)	Energy (eV)	Speed ² (cm ² /sec ²)
x 10 ⁶		x 10 ⁰	
77.13	2 ⁶⁷	588.5	2 ⁵⁰
38.57	2 ⁶⁶	294.2	2 ⁴⁹
19.28	2 ⁶⁵	147.1	2 ⁴⁸
9.642	2 ⁶⁴	73.56	2 ⁴⁷
4.821	2 ⁶³	36.78	2 ⁴⁶
2.410	2 ⁶²	18.39	2 ⁴⁵
1.205	2 ⁶¹	9.195	2 ⁴⁴
		4.598	2 ⁴³
x 10 ³		2.299	2 ⁴²
602.6	2 ⁶⁰	1.149	2 ⁴¹
301.3	2 ⁵⁹		
150.7	2 ⁵⁸	x 10 ⁻³	
75.33	2 ⁵⁷	574.7	2 ⁴⁰
37.66	2 ⁵⁶	287.3	2 ³⁹
18.83	2 ⁵⁵	143.7	2 ³⁸
9.416	2 ⁵⁴	71.84	2 ³⁷
4.708	2 ⁵³	35.92	2 ³⁶
2.354	2 ⁵²	17.96	2 ³⁵
1.177	2 ⁵¹	8.980	2 ³⁴
		4.490	2 ³³
		2.245	2 ³²
		1.122	2 ³¹
		0.561	2 ³⁰
		0.281	2 ²⁹
		0.1403	2 ²⁸

- Code 1: Accepts point cross section data together with an element and type-of-cross-section indicator and generates a cross-section tape.
- Code 2: Adds cross sections to the cross-section tape.
- Code 3: Edits (prepares a printout) of the cross-section tape.
- Code 4: Deletes specified cross sections from the cross-section tape.
- Code 5: Performs arithmetic on any two specified cross sections from the cross-section tape in order to produce a new cross section.
- Code 6: Prepares a tape containing data needed by the O5R code.
- Code 7: Averages selected microscopic cross sections over the group structure used by the O5R code.

The cross-section tape contains two records (logical FORTRAN type) per cross section. The first record identifies the element, the type of cross section, the manner in which interpolation between successive data points is to be made, and specifies the number of point cross sections given in the second record. At the present time three choices of interpolation methods are available: The logarithm of the cross section is assumed linear in lethargy; the cross section is assumed linear in lethargy; or the cross section is assumed linear in energy.

The second record contains a lethargy point (a lethargy zero of 100 MeV is used to convert energy to lethargy) followed by the cross section or its logarithm at the given lethargy, for as many points as the first record has specified. A record with a negative element identifier marks the end of the cross-section tape.

A particular code is called for by an input card having the letters "CODE" punched in columns 1 through 4, no punch in column 5, and the code number punched in column 6. Following this, in columns 7-10, is punched a logical tape number designating variously a scratch tape required for codes 4 and 5, the reactor data tape from code 6, or the tape of averaged cross sections produced by code 7. If this logical tape number is not provided, logical tape No. 12 will be used. The format is (1A4,1X,11,14).

The calling card is followed by the particular input data required by the code. Another code-calling card may follow. In the paragraphs which follow, references to "CODE 1" through "CODE 7" denote a card punched in the above fashion.

D5.3. Code 1 and Code 2: Cross-Section Input Codes (Cards to Tape)

The purpose of these codes is to write point cross sections on logical tape 11 in a convenient form for preparing the reactor data needed by the O5R code. The only difference between Code 1 and Code 2 is that Code 1 initializes the cross-section tape by writing the end-of-tape marker and rewinding the tape. It then proceeds as if Code 2, which simply adds cross sections to the tape, had been called.

The input contains an "identification card" listing the element, type of cross section, and type of interpolation to be used between successive cross-section points. This card is followed by "cross-section cards" which give the energy in eV and the cross section in barns at that energy. The cross-section cards must be arranged in order of descending energy, with the highest energy appearing first.

After each set of cross sections is written on logical tape 11 an "end-of-tape" marker is written, so that if a set of data is subsequently punched incorrectly the cross-section tape will still be in usable form and will still contain all cross-section entries prior to the one in error. The tape will not be correct, however, if an error occurs during the process of writing cross sections. A copy should be made before attempting to update a tape.

Each set of cross sections successfully loaded on logical tape 11 will have its identification card printed out. If an erroneous input occurs the data in error will lie in the set immediately following the last identification printed.

The input to Codes 1 and 2 is as follows:

Card A: Format (I5,I5,I5,I5,F10.5,5A6)

- a. > 0: Element identifier
Blank: End of all cross sections being loaded.
- b. Type of cross section
- | | | |
|---|-------------------------|--------------------|
| { | 1. Total | 5. ν x fission |
| | 2. Elastic scattering | 6. ν |
| | 3. Inelastic scattering | 7. f_1 |
| | 4. Fission | |

The parameter ν is the average number of neutrons per fission; f_1 is the average cosine of the angle of scattering in the center-of-mass system.

- c. Method to be used in interpolating between points:
- c = {
- 0: Logarithm of cross section assumed linear between points;
 - 8: Cross sections assumed linear in lethargy;
 - 8: Cross sections assumed linear in energy.
- d. Used by code for counting cross-section points.
- e. A decimal number (suggested use: Mass).
- f. Thirty characters of Hollerith input.

Card B: Format (E15.5,E15.5)

- a. The energy, in eV.
- b. The cross section, in barns, f_1 values, or ν values. A blank card B denotes the end of the list of cross sections for a particular element.

Cards are loaded in the following sequence: CODE 1 or CODE 2, A,B,B,...B,B(blank); A,B,B,...B(blank); A,...,B(blank), A(blank).

D5.4. Code 3: Cross-Section Edit Code

Code 3 "edits," i.e., prints out, the contents of the cross-section tape produced by Codes 1 or 2. It is called by "CODE 3" and no other input is required.

D5.5. Code 4: Cross-Section Deletion Code

Code 4 is used to delete specified cross sections from the cross-section tape. "CODE 4" calls the deletion code. It is followed by as

many cards as needed -- one card for each cross section to be deleted. Using Format (2I5), these cards give the element identifier and the cross-section identifier. A blank card signals the end of the list. The code prints out both the identification portion of each cross section deleted and a list of the cross sections finally remaining on the tape. After the deletions are accomplished the remaining cross sections are written on logical tape 11. Tape I, a scratch tape, is a copy of tape 11.

D5.6. Code 5: Cross-Section Arithmetic Code

Code 5 adds, subtracts, divides, or multiplies any two cross sections from the cross-section tape and produces a new cross section. A cross section is produced for every energy point listed in either of the two original cross sections. The new data are written on output tape logical number L, and may be punched on cards, added to the cross-section tape, or both.

The input to Code 5 is as follows:

Card A: Format (12A6)

72 columns of Hollerith characters.

Card B: Format (I5,I5,E10.5,I5,I5,E10.5,I5,I5,I5)

- a. ID1: Element 1 identifier.
- b. ISIG1: Element 1 cross-section identifier.
- c. DENS1: A number by which all element 1 cross sections are multiplied.
- d. ID2: Element 2 identifier.
- e. ISIG2: Element 2 cross-section identifier.
- f. DENS2: A number by which all element 2 cross sections are multiplied.

- g. NARITH $\left\{ \begin{array}{l} \leq 0: \text{ Add cross sections} \\ = 1: \text{ Subtract element 1 cross section from} \\ \quad \text{element 2 cross section} \\ = 2: \text{ Multiply cross sections} \\ = 3: \text{ Divide element 1 cross section by} \\ \quad \text{element 2 cross section.} \end{array} \right.$

- h. NNEG $\left\{ \begin{array}{l} = 0: \text{ Negative results produced by subtraction are set equal to zero} \\ \neq 0: \text{ Negative results produced by subtraction are accepted} \end{array} \right.$
- i. NOUT $\left\{ \begin{array}{l} = 1: \text{ Produce a BCD (binary coded decimal) tape of resultant cross sections} \\ = 2: \text{ Perform both NOUT = 1 and NOUT = 3} \\ = 3: \text{ Add resultant cross section to cross-section tape} \\ = 4: \text{ Do neither NOUT = 1 nor NOUT = 3.} \end{array} \right.$
- a-d e f

Card C: Format (4I5,F10.5,5A6)

Card C is the element identification card for the cross section resulting from the operations of Code 5. This card is identical with card A of Codes 1 and 2. Cards are loaded in the sequence: CODE 5, A, B, and C.

The manner of interpolating between points of the cross sections resulting from the above arithmetic is left to the user. Some caution should be exercised, especially over intervals where the cross sections change value rapidly. Logical tape L is written in such a manner that cards punched from it will be in the proper format for later loading by Codes 1 or 2.

D5.7. Code 6: Reactor Data Preparation Code

Code 6 prepares a reactor data tape needed for the O5R code. The preparation of these data is based upon the supergroup-subgroup structure previously defined and upon the media which make up the system. The group structure data for a medium are averages of cross sections over each of N subgroups within a supergroup. Given N, the number of subgroups, the averaged cross sections between the supergroup limits 2^{k-1} and 2^k [the energy in units of (cm/sec)²] are

$$\bar{\sigma}(i) = \frac{\int_{u_{i-1}}^{u_i} \sigma(u) du}{\int_{u_{i-1}}^{u_i} du} \quad 1 \leq i \leq N$$

where u_i is the lethargy associated with the lower energy limit of the i th group and is equal to $\ln(1.9132 \times 10^{20} / [2^k(2N - i)/2N])$

The number of subgroups, N , may be different for each medium.

For each supergroup the code computes and writes on tape a record of mean-free-flight times, nonabsorption probabilities, and probabilities for scattering for each nuclide. If anisotropic scattering is specified the average cosine of scattering angle (in the center-of-mass system) is computed and written. If fissioning may occur in a medium the fission probability is computed and written. Data are stored on the tape in order of descending energy and an "end-of-file" is written at the end of the information. In computing the data, the mean-free-flight time for the subgroup between i and $i - 1$ in a supergroup spanning 2^{k-1} to 2^k is equal to $1/s(i) \Sigma_{\text{T}}(i)$, where s is given by

$$s^2(i) = \frac{4N - 2i + 1}{4N} 2^k, \quad 1 \leq i \leq N,$$

and

$$\Sigma_{\text{T}}(i) = \sum_{j=1}^J N_j \bar{\sigma}_{\text{T}_j}(i),$$

where J is the number of nuclides composing the medium and N_j is the atomic density of nuclide j .

The nonabsorption probability for a medium is computed from $\Sigma_{\text{S}}(i)/\Sigma_{\text{T}}(i)$, where

$$\Sigma_{\text{S}}(i) = \sum_{j=1}^J N_j \bar{\sigma}_{\text{S}_j}(i)$$

and J is the number of scattering nuclides in the medium.

The probability for scattering with a particular nuclide, j , is computed from

$$N_j \bar{\sigma}_{s_j}(i) / \Sigma_s(i) \quad .$$

The fission probability is $\nu \Sigma_f(i) / \Sigma_T(i)$, where

$$\nu \Sigma_f(i) = \sum_{j=1}^J N_j \overline{\nu \sigma_{f_j}}(i) \quad .$$

The input to Code 6 is as follows:

Card A: Format (12A6)

72 columns of Hollerith identification.

Card B: Format (I5,E10.5,E10.5)

- a. NMED: Number of media.
- b. ETOP: Highest energy of cross sections needed, in eV.
- c. ELW: Lowest energy of cross sections needed, in eV.

Card C_M: Format (I5,I5) (One Card for Each Medium)

- a. N~~Ø~~ELEM: Number of elements in medium M.
- b. NPT: Number of points per supergroup for medium M
(NPT = 2^{j_M}, 0 ≤ j_M ≤ 8).

Card D_{M,L}: Format (I5,I5,I5,I5,I5,I5,E10.5) (One Card for Each Element of Each Medium)

- a. Element "L" identifier.
- b. Cross-section identifiers for element "L." If less than the 5 allowed are needed, blanks or zeros may be used to satisfy the format.
- c. 10⁻²⁴ x atoms/cm³ of element "L." Note: The order in which elements are specified here determines the order of specifying masses on card H, as well as the order in which the values of LFl are given on card F of the 05R input.

Card E: Format (10I5)

A list of anisotropic scatterers by element identifier. The order in which anisotropic scatterers are specified here determines the integer "N" used for LFL of card F of the O5R input.

The order of loading the above cards is: CODE 6, A,B,C,D,...,D,C,D,..., D,C,D,...,D,E.

The code is limited to not more than 6 media, not more than 8 scatterers per medium, not more than a total of 10 anisotropic scatters, a maximum energy (ETOP) of < 77 MeV, and a minimum energy (EMIN) of > 0.141×10^{-3} eV. The data are written on logical tape "L" in binary format.

D5.8. Code 7: Special Tape of Averaged Cross Sections

Code 7 produces a magnetic tape of microscopic cross sections averaged as in Code 6, the reactor data tape.

If σ_L ($L = 1, \text{NSIGS}$), where NSIGS is the number of cross sections requested) is the order of loading, then the FORTRAN statement used for writing this tape is WRITE TAPE 12, ($\sigma(L,I), I=\text{NPT}, L=1, \text{NSIGS}$). The first supergroup will bracket the energy ETOP and the last supergroup the energy ELW in the same manner as explained for code 5. In each record $I = 1$ corresponds to the highest energy and $I = \text{NPT}$ corresponds to the lowest energy of each supergroup.

The input to Code 7 is as follows:

Card A: Format (12A6)

72 columns of Hollerith identification.

Card B: Format (I5, I5, E10.5, E10.5)

- a. NSIGS: Number of cross sections requested.
- b. NPT: Number of points per supergroup ($\text{NPT} = 2^j, 0 \leq j \leq 8$).
- c. ETOP: Highest energy of cross sections needed, in eV.
- d. ELW: Lowest energy of cross sections needed, in eV.

Card C: Format (I^a5,I^b5)

- a. Element identifier.
- b. Cross-section identifier.

The order of loading the above cards is: C~~Ø~~DE 7,A,B,C,C,...,C. The code is limited to not more than 12 NSIGS, an E~~TØ~~P of < 77 MeV, and an E~~LØ~~W of > 0.141 x 10⁻³ eV. The cross-section output is in the same order as the order of the C-card input.

D.6. Geometry Routines *

The geometry routine currently being used with O5R is GE~~Ø~~M. It takes the starting and ending points of a neutron flight and determines (a) whether or not the flight lies entirely within a medium and (b) the point at which the flight leaves the original medium, if it does so. The geometries accepted by GE~~Ø~~M are of the following type:

The entire system is enclosed in a parallelepiped whose faces are parallel to the coordinate planes. The enclosing parallelepiped is cut into zones by planes parallel to the coordinate planes. Each plane used as a zone boundary cuts completely across the system, dividing it into two groups of zones. Each zone is similarly cut into blocks. Each plane used as a block boundary cuts completely across a zone, but need not be used as a block boundary in an adjoining zone. Each block may contain one or two media. If a block contains two media, the boundary between the two may be any quadric surface, thus a medium boundary may be either a block boundary or a quadric surface.

The purpose of the block system is twofold. Wherever possible block boundaries serve as medium boundaries; elsewhere all other medium boundaries are separated so that no block contains more than two media separated by a quadric surface. Through the use of the zone system the number of blocks required is reduced, since one zone may require a number of small

*This portion of this report was prepared by D. C. Irving.

blocks to separate the quadric surfaces while an adjoining zone may require only one block.

Upon entry GEØM assumes the following:

<u>Absolute Octal Address</u>	<u>Quantity</u>	<u>Meaning</u>
77453	ZØLD: } YØLD: } ZØLD: }	The coordinates of the starting point of the neutron flight.
77452		
77451		
77456	ZØNE: } YØNE: } ZØNE: }	The coordinates of the tentative end of flight.
77455		
77454		
77457	ETA:	The number of mean free paths for this flight.
77461	NMED:	The number of the medium of the start- ing point.
77450	UØLD: } VØLD: } WØLD: }	The X, Y, and Z direction cosines of the neutron velocity.
77447		
77446		
77445	SPØLD:	The speed squared of the neutron.
77336	BLZØN or LØCATE:	A packed word containing the zone and block location of the neutron at the start of flight.

If the neutron flight lies entirely within the medium of the starting point, GEØM leaves everything else unchanged and supplies the following:

<u>Absolute Octal Address</u>	<u>Quantity</u>	<u>Meaning</u>
77460	MARK:	Contains 1 in decrement.
77336	LØCATE:	A packed word containing the block and zone location of the terminal point of the neutron flight.

If the flight leaves the original medium, GEØM supplies:

<u>Absolute Octal Address</u>	<u>Quantity</u>	<u>Meaning</u>
77456	XØNE: YØNE: ZØNE: }	The coordinates of the point at which the neutron flight leaves the original medium.
77455		
77454		
77461	NMED:	The new medium of the neutron.
77336	LOCATE:	The block and zone location of the point at which the neutron leaves the original medium.
77460	MARK:	MARK contains -1 if the neutron escaped the system; -2 if the neutron entered an internal void; 0 otherwise.

Space outside the system of interest, yet within the enclosing parallelepiped, is assumed to be exterior void.

An interior void is a region within which a neutron has an infinitely long mean free path. A neutron encountering an interior void is carried across the void until it enters another medium; its path through the void is of length zero in mean free paths.

D6.1. Input for GEØM

Input will be on cards in columns 1-72. All blanks will be ignored. Numbers have a variable length field, and integers are unsigned strings of digits. Floating point numbers have the form $+X.YE+Z$, where X, Y, and Z are integers. Signs will be assumed to be positive if omitted and the fields X, Y, and EZ will be assumed zero if omitted. There are three sections to the input.

- a. Zone boundaries: The X, Y, and Z values used as zone boundaries, including the outer limits of the system. A typical example might be as follows:

```
X ZØNE BØUNDARIES (0, .1, .5, 8E-1, 1)
Y ZØNE BØUNDARIES (0, .5, 1)
Z ZØNE BØUNDARIES (0, .3, .65, 1.0)
```

The order of the listing is irrelevant.

- b. The quadratic functions defining the curved surfaces:

The quadratic surfaces used as interfaces between media may be defined by $f(X,Y,Z) = 0$, where f is of second order or less in X , Y , and Z . One medium then lies on the positive side of the surface, in the region where $f(X,Y,Z) > 0$, and the other medium lies on the negative side. The format is:

CURVED BOUNDARIES (n/l,f₁(x,y,z)/2,f₂(x,y,z)/.../n,f_n(x,y,z))

where n is the number of curved surfaces. To identify the surfaces they are numbered consecutively from 1 to n ; however, they need not be listed in consecutive order in the input.

Each $f(x,y,z)$ is written as one would normally write a function except that XSQ, YSQ, and ZSQ are used for X^2 , Y^2 , and Z^2 . Parentheses may not be used, therefore expressions such as $(x + 1)^2$ must be expanded as XSQ + 2.X + 1. Coefficients are in floating point, and if omitted are taken to be 1., i.e., "-XSQ" is equivalent to "-1.XSQ." Note that $f(x,y,z)$ is a function and not an equation; an equal sign is not permitted.

- c. For each zone in the system the following specifications must be made:

1. "ZONE (l,m,n)" identifies the description to follow as that for the zone which is the l th in the x direction, the m th in the y direction, and the n th in the z direction.
2. The x , y , and z values used as block boundaries in that zone, including the boundaries of the zone. A typical example might be as follows.

X BOUNDARIES (.1,.5) Z BOUNDARIES (.65,.7,.8,1.)
Y BOUNDARIES (.5,.75,1.).

The order of the three lists is irrelevant.

3. The list of media in each block. If a block contains only one medium, this specification is merely the medium number.

If there are two media in a block, however, the specification is "n/i/m," where n is the number of the medium on the positive side of the curved surface, i is the number of the curved surface in the block, and m is the medium number on the negative side of the curved surface. Interior voids are designated as medium No. 0 and exterior voids as medium No. 1000.

The specification for each block is separated from the next by a comma. The blocks must be listed in the order (1,1,1), (2,1,1), (3,1,1), ..., (n,1,1), (1,2,1), (2,2,1), ..., (n,m,1), (1,1,2), ..., (n,m,l), where (i,j,k) denotes the block which is the ith in the x direction jth in the y direction and kth in the z direction.

An example of a block specification is

~~BL~~CKS 3,3/1/2, 3/1/2,2

which would describe 4 blocks. The first contains only medium 3; the second contains medium 3 on the positive side of quadric surface 1 and medium 2 on the negative side of quadric surface 1; the third contains medium 3 on the positive side of surface 1 and medium 2 on its negative side; and the fourth block contains only medium 2.

The end of the input is signaled by an END card having "END" punched in columns 8-10.

An example of a complete input for ~~GEOM~~ follows:

~~XZ~~ONE ~~B~~OUNDARIES (0,1.)
~~YZ~~ONE ~~B~~OUNDARIES (0,1.)
~~ZZ~~ONE ~~B~~OUNDARIES (0,.5,1.)
CURVED ~~B~~OUNDARIES (1/1,XSQ+YSQ+ZSQ-X-Y-Z+.5)
~~Z~~ONE (1,1,1) XB(0,.5,1.) YB(0,.5,1.) ZB(0,.5)
~~BL~~CKS 1,1,1,1
~~Z~~ONE (1,1,2) XB(0,1.) YB(0,1.) ZB(.5,1.)
~~BL~~CKS 2/1/1
END

The procedure for reading input into GEOM is as follows:

- a. Place the address of the last location to be used for input storage in XR1 (Index Register 1).
- b. Then TSX \$JMIN,4. JMIN will store input in successively lower locations and return to 1,4 with the last address not used for storage in XR1. If there are errors in the input JMIN will write appropriate error flags and transfer to the Monitor Exit Routine. Input is assumed to be on logical tape 10 and output (for error flags) on logical tape 9.

The present limitations on GEOM are:

ZONES: 1023

BLOCKS: 1023 per zone (not more than 31 blocks in either the X, Y, or Z directions in a single zone).

MEDIA: 511, not counting voids.

There is no limit upon the number of quadric surfaces allowed.

A rough estimate of the amount of storage used for input is given by: 11 x number of curved surfaces + 14 x number of zones + 4 x total number of blocks.

D.7. O5R Input

The input required to run the O5R code consists of the appropriate neutron source specifications, a description of the geometry of the system being studied, the reactor data, i.e., a complete specification of the system, and a suitable subroutine for treating inelastic scattering, if it is being considered. Each section of the input is discussed below.

D7.1. O5R Neutron Sources

The information required by O5R for each source neutron are the position of the neutron, its energy, its statistical weight, and the direction of its first flight. For the ensuing discussion the following definitions may be helpful. SSQ is the neutron energy, in cm^2/sec^2 . X, Y, and Z are the X-axis, Y-axis, and Z-axis coordinates, in cm. U, V,

and W are the X-, Y-, and Z-direction cosines, respectively, and $U^2 + V^2 + W^2 = 1$. WT is the statistical weight of the neutron. NEUT is an integer associated with a particular neutron, and NEUTS is the number of neutrons to be given source data.

Three choices are available for E. It may be the same for all source neutrons, it may be chosen from an empirical fission spectrum energy distribution by O5R, or it may be computed by the subroutine SOURCE, written by the user.

Two choices are available for X, Y, and Z. They may be the same for all neutrons or they may be computed by SOURCE.

Three choices are available for U, V, and W. They may be the same for all neutrons, they may be chosen isotropically by O5R, or they may be computed by SOURCE.

Two choices are available for WT. It may be the same for all neutrons or it may be computed by SOURCE.

Two methods of supplying the source data are used. Any quantity which is to be the same for all neutrons may be given to O5R on an input card. This card is Card D of the O5R reactor data input described below, and the permissible alternatives available by using this card are covered in that description. If any source parameter may vary from neutron to neutron or from batch to batch, however, it must be computed or supplied by Subroutine SOURCE. SOURCE data always take precedent over input card data. The call statement is

```
CALL SOURCE (SSQ,U,V,W,X,Y,Z,WT,NEUT,NEUTS,NMED)
```

This subroutine must be included if any of the values needed to specify the source cannot be supplied by the source input card. Arguments must appear in the order given in the FORTRAN statement above, even though some of them may not be used. The subroutine is entered once for each neutron until the total number of source neutrons, NEUTS, has been processed. NEUT is initially 1 and is incremented by 1 on each subsequent entry until NEUT = NEUTS, at which point the data for the final neutron will be processed. As an example, suppose that half of the

source neutrons are to have an energy of 1. MeV and the other half an energy of 2. MeV, with all other parameters to be supplied by source card. A subroutine to provide for this might be as follows:

```
SUBROUTINE SOURCE (SSQ,U,V,W,X,Y,Z,WT,NEUT,NEUTS,NMED)
  IF (NEUTS-NEUT-NEUT)2,1,1
1  SSQ = 2.*1.913220092E18
  RETURN
2  SSQ = 1.913220092E18
  RETURN
  END
```

As is evident in the subroutine above, the conversion factor from energy in eV units to energy in speed squared units, the units used in O5R, is $1.913220092 \times 10^{12}$.

If it is intended to compute U, V, and W by SOURCE, at least one of U, V, or W must be nonzero on the input card. The sum of $U^2 + V^2 + W^2$ must equal 1.

CAUTION: Do not start a neutron on a zone or block boundary. This will lead to serious difficulties with the GEOM package.

D7.2. Reactor Data Input

Much of the versatility of O5R is obtained through the flexibility of its data input specifications coupled, of course, with equally flexible source and geometry systems. Input to O5R is straightforward, a single general caution being that some of the quantities specified in input are also found in some of the cross-section, geometry, and source routines, and care should be taken to insure internal agreement.

The input cards for the O5R reactor data input are individually considered below.

Card A: Format (12A6)

A card containing 72 columns of Hollerith information for identification of the printed output.

Card B: a b c d e f g h
Format (I5,I5,I5,I5,I5,I5,E10.5,I5)

- a. NSTRT: The number of neutrons with which to start off each batch. $0 < \text{NSTRT} \leq \text{NMOST}$.
- b. NMOST: The maximum number of neutrons permitted to appear during one batch. $\text{NSTRT} \leq \text{NMOST} < 1000$. (The number of neutrons which can appear from splitting by applying weight standards can be controlled.)
- c. NITS: The number of batches in the run.
- d. NHOW: Directs whether the original source is to be repeated for each batch (NHOW > 0) or whether succeeding source batches are to be made up out of fission neutrons from the preceding batch (NHOW = 0).
- e. QUIT: Number of runs (> 0).
- f. FISTP: The logical scratch tape number available for storing intermediate fission source data. (If no fissioning is permitted, FISTP can be left blank or 0.)
- g. ECUT: The energy, in eV, below which neutrons are considered to be in the thermal group.
- h. THERM: Determines the manner in which the thermal group is to be treated:
 - THERM = 0: No thermal group.
 - THERM = 1: One-velocity thermal group. (The one-velocity parameters will be given on card G of this input.)
 - THERM = 2: Thermal parameters will be computed by a subroutine SNAFU (written by the user, but must be called SNAFU.)

Card C: a b c d e
Format (I5,I5,I5,I5,E10.5)

- a. MEDIA: The number of media, exclusive of voids, in the reactor. This number must agree with the number appearing on the reactor data tape. $0 < \text{MEDIA} < 6$.

- b. $N\phi$ NE: The number of anisotropic scatterers appearing on the reactor data tape.
- c. CONT1: The logical tape number of the reactor data tape.
- d. CONT2: The logical tape number of a copy of the reactor data tape. (For problems which involve a wide range between the highest and lowest energy on the reactor data tape and several media, a considerable fraction of machine time is expended in rewinding the data tape. To get around this difficulty a second tape, identical with the first, may be specified so the O5R may shuttle between them, rewinding one tape while using data from the other.)
- e. ETAPE: The maximum energy, in eV, of the reactor data tape. This value must be the same as that given to the reactor data code.

Card D: Format (E10.4, E10.4, E10.4, E10.4, E10.4)

Card D is the source input card. See the preceding discussion of "O5R Neutron Sources" for further explanation.

- a. ESOURCE: The source energy, in eV.
- b. ULNP: }
c. VINP: } The direction cosines of the source neutrons. If
d. WINP: } these are to be supplied by subroutine SOURCE, at
least one of them must be nonzero on this card.
- e. WTSTRT: The starting weight of the source neutrons.

Card E: Format (E10.4, E10.4, E10.4, I5, I5)

- a. XSTRT: }
b. YSTRT: } The starting coordinates of the source neutrons,
c. ZSTRT: } unless otherwise given by subroutine
SOURCE.
- d. NMED: The medium number of XSTRT, YSTRT, ZSTRT.
- e. NREG: The region number of XSTRT, YSTRT, ZSTRT.

Card(s) F: a b c d e
Format (I5,I5,I5,E10.5,5X,8I2)

- a. SCATS: The number of scatterers in the medium.
- b. PØINTS: The number of subgroups per supergroup on the reactor data tape for this medium. The value of PØINTS must agree with the specification of the reactor data tape.
- c. FISSIØN: Indicates whether or not there is fast fissioning in this medium. If FISSIØN = 0, no fast fissioning is permitted; if FISSIØN = 1, fast fissioning is allowed.
- d. FWLØW: The weight to be assigned to neutrons produced by fission (usually 1.).
- e. LFl: Used to inform O5R whether or not scattering is anisotropic for each scatterer in the medium and, if some of the nuclides of the medium do scatter anisotropically, which position on the reactor data tape is to be used. A value of LFl must be given for each scatterer in the medium in the order in which the scatterers are specified in cards $D_{M,L}$ of Code 6 of the cross-section package. If a scatterer is an isotropic scatterer, then no f_1 's appear for it and LFl = 0. If a nuclide is an anisotropic scatterer, then LFl is the position the scatterer occupies on card E of Code 6. For example, if a medium contains H, C, and O, with O and C listed as anisotropic scatterers on card E of Code 6 in that order, and if H, C, and O are specified as the elements in the medium on cards $D_{M,L}$ of Code 6 in that order, then LFl would be ① 0 ① 2 ① 1. (The symbol ① is used to indicate a space.) If the elements on card $D_{M,L}$ were specified in the order C, H, and O, then LFl would be ① 2 ① 0 ① 1.

There will be an individual card F for each medium, and they are loaded in the same order as the media appear on the reactor data tape.

Card(s) G: Format (E10.5, E10.5, E10.5) (One-velocity parameters)

- a. ~~SL~~OTH: The thermal neutron mean free path for this medium.
- b. ~~SL~~OPS: The thermal-neutron nonabsorption probability for this medium.
- c. ~~SL~~WF: The average number of fission neutrons produced by a collision at thermal energies in this medium, given by $\nu\Sigma_f/\Sigma_t$.

If one-velocity parameters are called for on card B, by setting THERM = 1, then a G card immediately follows each F card for each medium.

Card H: Format (7E10.5)

The H cards are loaded by medium - one card to a medium. The atomic masses, in amu, are specified for each scatterer of a medium in the same order as the scatterers appear on the reactor data tape, which is the order in which they were specified on card D_{M,L} of Code 6. An inelastic scatterer is signaled by a mass of zero which, when encountered by O5R, triggers the calling of subroutine KINNY which treats inelastic scattering. For example, if the medium consisted of water, with hydrogen, scattering elastically, and oxygen scattering both elastically and inelastically, card H would show three masses: 1.00000, 16.00000, and .00000.

Card I: Format (I5, I5, 5X, 36I1)

- a. HISTR: The logical tape number to be assigned to the first collision tape to be used.
- b. HISMX: The highest logical tape number a collision tape may be assigned. $HISTR \leq HISMX$. When the first collision tape, HISTR, has been filled the logical tape number is increased by one, the logical number increasing by one for each succeeding tape until HISMX is reached. After HISMX is filled the logical tape number is set

back to one and the process continues. (It is expected that enough time for machine operating personnel to remove a filled tape and replace it with a pool tape will be available.) None of these collision tapes may be on the same machine channel as the reactor data tape or the fission tape.

- c. NBIND: Selects from the list of parameters available at each collision, as shown in the discussion of analysis routines below. For each of the parameters in the bank list, either a 0 or a 1 must be punched. A 1 selects the parameter, a 0 does not select it.

Card J: Format (1^a012)

- a. RANDM: The octal representation of the initial random number to be used in the generation of random numbers. If left blank the code will insert 343277244615. Starting random numbers must end in 1 or 5 in order that the period of the random number sequence be the maximum for the IBM 7090. The maximum period is 2^{33} .

Card K: Format (I^a5,I^b5,I^c5,I^d5)

- a. NSPLT: An index controlling splitting. If NSPLT = 1, splitting is allowed; if NSPLT = 0, splitting is prohibited.
- b. NKILL: An index controlling Russian Roulette. If NKILL = 1, Russian Roulette is allowed; if NKILL = 0, Russian Roulette is prohibited.
- c. REGMX: The maximum number of regions in the reactor. REGMX must equal 1 in the present version, for reasons which are given below.
- d. GPMAX: The number of fast groups plus the number of thermal groups. The latter quantity is either zero or one. If GPMAX is left blank, the code supplies the value 40.

Card(s) L: a b c d e f g h i
Format (I5,I5,I5,I5,I5,I5,E10.5,E10.5,E10.5) These cards
are included only if NSPLT or NKILL (Card K) are nonzero.

- a. NGP1: These parameters select particular groups and
b. NDG: regions in which the weight standards are to be
c. NGP2: applied. For group NGP1 in steps of NDG through
d. NRG1(=1)* group NGP2 and for region NRG1 in steps of NDRG
e. NDRG(=1)* through region NRG2 the weight standards assigned
f. NRG2(=1)* by g, h, and i, below, are to be applied. Setting
 NGP1 = 0 causes the weight standards g, h, and i
 to be applied to all regions. A last card with
 NGP1 = -1 must be supplied to signal that all
 weight standards have been loaded.
- g. WITHH: Neutrons having weights above this value in the regions
and groups indicated will be split.
- h. WTLØW: Neutrons having weights below this value in the regions
and groups indicated will undergo Russian Roulette.
- i. WTAVE: The weight which replaces the current weight of a
neutron surviving Russian Roulette.

The format of this card may be easier to understand if the following FORTRAN statements are considered.

```
DØ 1, I = NGP1, NGP2, NDG  
DØ 1, J = NGR1, NGR2, NDRG  
WITHH(I,J) = WITHH  
WTLØW(I,J) = WTLØW  
WTAVER(I,J) = WTAVE.
```

*The geometry routine included here does not provide for spatial regions in which different weight standards for Russian Roulette and splitting apply, thus the number of regions must be 1 throughout the calculation. Different weight standards may be used in different energy groups, however.

D.8. Treatment of Nonelastic Scattering

The O5R code handles nonelastic scattering by calling subroutine

"KINNY(SSQ,NMED,NELEM,X,Y,Z,NEUTS,SLIST,WT)"

whenever a scatterer of zero mass is encountered by the subroutine which selects the nuclide struck by the neutron. (See card H of O5R input, above.) The arguments are:

SSQ: The speed squared of the incident neutron, in cm^2/sec^2 .

NMED: The number of the medium within which the event occurred.

NELEM: The number of the element with which the neutron collided - counted from 1 in the order punched on card H of the O5R input.

X,Y,Z: The spatial coordinates of the event.

NEUTS: The number of neutrons resulting from the collision. NEUTS \leq 10 and provides for (n,2n), (n,3n), ..., etc. events.

SLIST: A list of the energies (speed squared) of each of the product neutrons: SLIST(I), I = 1, NEUTS. It is assumed that the product neutrons are emitted isotropically in the laboratory system of reference. It must be dimensioned as SLIST(10).

WT: The weight of the neutron going into the nonelastic event. This weight is assigned to each of the product neutrons.

KINNY must be self-contained and read in its own input as required.

D.9. Problem Analysis

In order to insure maximum flexibility in the range of problems O5R can handle, the production of the neutron histories pertinent to a particular problem is completely divorced from their analysis. Collision data generated by O5R are transmitted to the problem analysis code by means of a "collision" tape or tapes. The input data for O5R permits the programmer to choose the collision parameters required for the problem analysis routine. At each collision the selected parameters are written on the collision tape in a particular manner explained below. A program can then be written to accept data from the collision tape for analysis. The analysis code may be written completely in FORTRAN if desired, since the parameters available and the collision tape itself are compatible with the FORTRAN system.

Among the advantages of this separation of codes are the reduction of core storage problems, the fact that debugging of analysis codes does not require a rerunning of O5R for each debugging run, and the possibility of multiple use of the collision tape, that is, a collision tape prepared for the solution of one problem may be reanalyzed for the solution of another.

It appears to be a reasonable rule of thumb that any datum which might conceivably be desirable for the problem analysis should be called for on the collision tape. It is much less expensive to call for too much data than it is to rerun O5R to get a quantity which suddenly turns out to be necessary. Some quantities put on the collision tape are redundant; for instance, S^2 (the speed squared) is determined if U , V , and W , the direction cosines, are given ($S^2 = U^2 + V^2 + W^2$), but it is nevertheless almost always simpler and in the long run cheaper to include such numbers as, in this case, S^2 .

D9.1. Collision Tape

Each record written on the collision tape is composed of fewer than 129 words, so that the tape is compatible with FORTRAN. The first word of each record contains the integer 1. This number indicates the

number of physical records contained in the logical record and is required by the FORTRAN tape routines.

The second word of each record contains the run record number, and is increased by 1 as each successive record is written on tape during a run.

The third word of each record contains a FORTRAN-type integer, NTYPE, which indicates the type of data filling out the rest of the record. NTYPE = 0 indicates a normal collision record. If NTYPE = 0, the remainder of the record contains the requested collision parameters for as many collisions as there is room. For instance, if 10 parameters per collision are called for, then one tape record would contain the results of 12 neutron collisions, since this results in 120 words of data plus the 3 required words at the beginning of the record, or 123 words, which does not exceed the permissible maximum of 128 words. When the last record of a batch is not completely filled with collision parameters, the remainder of the record is set to zero. As each batch is completed a record is written with NTYPE = 1 and the rest of the record filled in with zeros. If, in writing the collisions an end of tape is reached, the code writes a record with NTYPE = 2 and continues on to another tape. After all batches of a run have been processed a record is written with NTYPE = 3.

It is intended that all data concerned with significant events in the history of a neutron shall be available for the subsequent analysis of the system. Thus not only is the data concerning the real collisions of neutrons with nuclides recorded, but other available information ("pseudo collisions") is also recorded. Pseudo collisions are recorded when changes in the neutron weight occur through the application of weight standards, when medium boundaries are crossed, as soon as the source data for a neutron are available, and when the neutron escapes from the system of interest. Each of these events is identified by a marker index which tells whether or not the collision is real and, if it is a pseudo collision, what has taken place. This index is one of the parameters which may be selected to be written with each record, and is described as item 1 below.

D9.2. Parameter List

Any selection from the following list of parameters may be made by appropriately punching item c of card I of the O5R input. The parameters selected will be written at each collision in the order listed. All numbers are floating point except when indicated otherwise.

1. $NCOLL$: A FORTRAN-type integer indicating the type of collision.
 $NCOLL = 1$: Source data appears.
2: A real collision of a neutron with a nuclide.
3: Neutron killed by Russian Roulette.
4: Neutron escaped from the reactor.
5: Splitting occurs through the application of weight standards. The data given are those for the original neutron with its new weight.
6: Same as 5 except that the data for the duplicate neutron are given.
7: Neutron crosses a medium boundary.
8: Neutron survives Russian Roulette and weight is increased.
9: Neutron participates in a nonelastic interaction.
2. $NAME_n$: A FORTRAN-type integer identifying the neutron having the collision.
3. S_1^2 : The speed squared of the neutron after collision.
4. U_1 :
5. V_1 :
6. W_1 : } The neutron velocity in the X, Y, and Z directions, respectively, after collision.
7. X_1 :
8. Y_1 :
9. Z_1 : } The coordinates of the collision location.
10. WT_1 : The neutron weight after collision.
11. S_0^2 : The speed squared of the neutron before collision.

- 12. U_0 : } The neutron velocity in the X, Y, and Z directions,
- 13. V_0 : } respectively, before collision.
- 14. W_0 : }

- 15. X_0 : } The coordinates of the directly previous event.
- 16. Y_0 : }
- 17. Z_0 : }

- 18. WT_0 : The neutron weight before collision.

- 19. $\text{THETA}(S_0^2, m)$: The mean-free-flight time to the collision point.

- 20. $\text{PSI}(S_0^2, m)$: The nonabsorption probability at the collision.

- 21. ETA : The number of mean free paths used to arrive at the collision point.

- 22. $\text{GR}\phi\text{UP}$: A FORTRAN-type integer identifying the energy group within which S_0^2 lies.

- 23. ELEM : A FORTRAN-type integer identifying the nuclide collided with.

- 24. NREG : A FORTRAN-type integer identifying the region within which the collision occurred.

- 25. NMED : A FORTRAN-type integer identifying the medium within which the collision occurred.

- 26. NAMEX_n : A FORTRAN-type integer giving the original NAME of the neutron from which the current neutron was produced by splitting.

- 27. MDRGL_n : A packed word containing the medium and region where the collision occurred, the nuclide collided with, and the energy group associated with S_1^2 . MDRGL_n is of the form GGGMMMLLLRRR, where GGG = group; MMM = medium; LLL = element position in medium MMM; and RRR = region.

- 28. BLZNT : A packed word giving the block and zone location of X_1 , Y_1 , and Z_1 .

- 29. BLZON : A packed word giving the block and zone location of X_0 , Y_0 , and Z_0 .

- 30. LAMBDA: The mean free path of the neutron in the medium before collision.
- 31. S_0 : The speed of the neutron before collision.
- 32. S_1 : The speed of the neutron after collision.

In order to make clear exactly which of the above parameters have significance for various types of real or pseudo collisions, the following comments are appended:

- a. $NC\emptyset LL = 1$, a source collision. The code, before allowing any neutron to have a flight from the source position, records the source data. Significant items are 1-10, 24, 25, 28, 29.
- b. $NC\emptyset LL = 2$, a real collision. All items are significant.
- c. $NC\emptyset LL = 3$, neutron killed by Russian Roulette. Item 10 will be 0, items 1-18, 22, 24-26, and 28 will have significance. Items 3-10 will be the same as items 11-18.
- d. $NC\emptyset LL = 4$, neutron escaped from the reactor. All items except 23 will have significance. Items 3-6 and 10 will be the same, respectively, as items 11-14 and 18. Items 7-9 will locate the point at which the escape was located.
- e. $NC\emptyset LL = 5$, neutron split in two. Items 1-18, 22, and 24-29 will have significance. Item 2, NAME, will be the same as NAME before splitting. Items 3-9 will be the same as items 11-17. Item 8 will be twice item 10.
- f. $NC\emptyset LL = 6$, neutron split in two. A new NAME will have been assigned to the new neutron and item 10 will contain the NAME of the original neutron. The significant items will be the same as for $NC\emptyset LL = 5$. $NC\emptyset LL = 6$ data will immediately follow $NC\emptyset LL = 5$ data on the tape.
- g. $NC\emptyset LL = 7$, neutron crosses a medium boundary. All items except 23 will have significance. Items 3-6 and 10 will be the same as items 11-14 and 18. Item 25 will refer to the medium being entered and items 7-9 will locate the crossing point.

- h. $NC\phi_{LL} = 8$, neutron survives Russian Roulette and weight is increased. Item 9 will be the new weight, item 18 the old weight. Items 1-18, 22, and 24-30 will have significance. Items 3-9 will be the same as items 11-17.
- i. $NC\phi_{LL} = 9$. If no product neutrons are born from a nonelastic collision, then $NC\phi_{LL} = 9$ and $WT_1 = 0$. If only one neutron results, then $NC\phi_{LL} = 2$ and $WT_1 = WT_0$. If more than one neutron results, the first has $NC\phi_{LL} = 2$ and $WT_1 = WT_0$, while the additional neutrons appear with $NC\phi_{LL} = 9$ and $WT_1 = WT_0$.

Appendix E. Demonstration Problem

As an example of the input preparation for the Nucleon Transport Code and to show the results obtained from the various codes, the details of a typical problem are presented below. It is hoped that reference to this material will clarify any vague points which may exist in the descriptions of input. In addition, the proper functioning of NTC in the hands of the user can be verified by running this problem and duplicating the results presented.

The problem considers the distributions in angle and energy of the neutrons and protons resulting from 4,000 160-MeV protons normally incident on an infinite slab of aluminum 10 cm thick. The source protons were run in batches of 2,000 each, and the desired distributions were obtained from the analysis routines described in Appendix A. The 10-cm-thick slab was divided into four equal subslabs for the analysis but, in order to save space, only the results for the 10-cm slab are presented.

The problem input is shown in Fig. E1. The first five lines comprise the input to NTC Chain (1,3), discussed on pages 10 and 11 of this report. The next six lines are the GEOM input read into Chain (2,3) as noted on page 12. The next line is the source input. The lines which follow are the inputs to the analysis routines, which were discussed on pages 24 through 26.

The output from the high-energy transport analysis, for the 10-cm-thick slab only, is shown in Figs. E2 through E11, and is generally self-explanatory. As the batches run, there are printed the total number of source particles run, up to and including the current batch, and the total number of neutrons which have been put on logical 8, the O5R source tape. The GEOM input follows each of these printouts, since the input tape is repositioned so that a new batch may be run. Figure E2 is a sample of two such printouts.

The time required to run and analyze the two batches was 19.2 min.

Before O5R could be run, it was necessary to prepare a master cross-section tape for aluminum and, from it, a reactor data tape. Code 1 (see page 46) was run to prepare the master cross-section tape, and Code 3 (see page 47) was used to print the results, which are shown in Figs. E10 through E14.

2 BATCHES OF 2000 160 MEV PROTONS INTO 10 CM. OF ALUMINUM.
262746553275

```
160.      50.      1      2000      2
  0        1
13. 26.9815 .0603116
X ZONE BOUNDARIES (0,10.)
Y ZONE BOUNDARIES (-10.E+20,10.E+20)
Z ZONE BOUNDARIES (-10.E+20,10.E+20)
ZONE(1,1,1)XB(0,10.)YB(-10.E+20,10.E+20)ZB(-10.E+20,10.E+20)
BLOCKS1
  END
160.      1
  10.      10      19      12
  10.      4      16      9
.1737     .342     .5      .6428     .766     .866     .9397
.9848     1.
.05       .1      .15     .2      .3      .4      .5
.6        .7      .8      .9      1.      1.5     2.
  3.      5.
160.      50.      11
  4        5      9      10.
  40.      30.      20.      10.      0.
.1737     .342     .5      .6428     .766     .866     .9397
.9848     1.
```

Fig. E1. Problem Input: Nucleon Transport Code

```

2 BATCHES OF 2000 160 MEV PROTONS INTO 10 CM. OF ALUMINUM.
INITIAL RANDOM NUMBER # 262746553275
EMAX #      160.00  EMIN #      50.00  MAXMAT # 1  MAXCAS # 2000  MAXBCH # 2
THIS IS MEDIUM NUMBER 1
DENS(M) # 0.      NEL1 # 1
Z # 13.0  A # 27.0  DEN # 0.603116E-01
X ZONE BOUNDARIES (0,10.)
Y ZONE BOUNDARIES (-10.E+20,10.E+20)
Z ZONE BOUNDARIES (-10.E+20,10.E+20)
ZCNE(1,1,1)XB(0,10.)YB(-10.E+20,10.E+20)ZB(-10.E+20,10.E+20)
BLOCKSI
      END

BATCH 1      SOURCE PARTICLES# 2000

05R NEUTS#      350
X ZONE BOUNDARIES (0,10.)
Y ZONE BOUNDARIES (-10.E+20,10.E+20)
Z ZONE BOUNDARIES (-10.E+20,10.E+20)
ZCNE(1,1,1)XB(0,10.)YB(-10.E+20,10.E+20)ZB(-10.E+20,10.E+20)
BLOCKSI
      END

BATCH 2      SOURCE PARTICLES# 4000

05R NEUTS#      701

```

Fig. E2. Output from High Energy Transport Analysis

T# 1.000E 01 NX# 4 NR# 16 NA# 9
COSINE GROUP LIMITS
1.737E-01 3.420E-01 5.000E-01 6.428E-01 7.660E-01 8.660E-01 9.397E-01 9.848E-01 1.000E 00
RADIAL GROUP LIMITS
5.000E-02 1.000E-01 1.500E-01 2.000E-01 3.000E-01 4.000E-01 5.000E-01 6.000E-01 7.000E-01 8.000E-01
9.000E-01 1.000E 00 1.500E 00 2.000E 00 3.000E 00 5.000E 00
EUP# 1.600E 02 ELO# 5.000E 01 DELTA E# 1.000E 01

Fig. E3. Output from High Energy Transport Analysis

	CHARGED		RESIDUAL	
1	5.653E-01	1.746E-02	1.311E-01	3.483E-02
2	5.286E-01	3.694E-02	1.240E-01	2.013E-02
3	5.504E-01	1.016E-01	1.111E-01	1.191E-02
4	6.736E-01	9.742E-02	1.571E-01	1.064E-02
5	6.926E-01	9.898E-04	1.568E-01	1.769E-02
6	6.649E-01	2.253E-01	1.727E-01	7.135E-03
7	4.274E-01	7.225E-02	1.301E-01	8.276E-03
8	9.174E-02	2.472E-02	3.567E-02	4.230E-03
9	9.739E-03	9.739E-03	5.412E-03	5.412E-03
10	7.971E-04	7.971E-04	1.855E-03	3.613E-04

Fig. E4. Output from High Energy Transport Analysis: Charged Particle Energy and Standard Deviation and Residual Nucleus Energy and Standard Deviation, by Spatial Groups.

SUB-SLAB THICKNESS# 1.000E 01

ENERGY-ANGULAR DISTRIBUTION, (NO./MEV/COSINE)

UPPER E# 1.600E 02 LOWER E# 1.500E 02

ANG. GP.	PROTON	S.D.	NEUTRON	S.D.
1	0.	0.	0.	0.
2	0.	0.	0.	0.
3	0.	0.	0.	0.
4	0.	0.	0.	0.
5	0.	0.	0.	0.
6	0.	0.	0.	0.
7	0.	0.	0.	0.
8	0.	0.	0.	0.
9	0.	0.	0.	0.

UPPER E# 1.500E 02 LOWER E# 1.400E 02

ANG. GP.	PROTON	S.D.	NEUTRON	S.D.
1	0.	0.	0.	0.
2	0.	0.	0.	0.
3	0.	0.	0.	0.
4	0.	0.	0.	0.
5	0.	0.	0.	0.
6	0.	0.	0.	0.
7	0.	0.	0.	0.
8	0.	0.	0.	0.
9	0.	0.	1.645E-03	1.645E-03

UPPER E# 1.400E 02 LOWER E# 1.300E 02

ANG. GP.	PROTON	S.D.	NEUTRON	S.D.
1	0.	0.	0.	0.
2	0.	0.	0.	0.
3	0.	0.	0.	0.
4	0.	0.	0.	0.
5	0.	0.	0.	0.
6	0.	0.	0.	0.
7	0.	0.	3.392E-04	3.392E-04
8	0.	0.	1.663E-03	5.543E-04
9	0.	0.	4.934E-03	1.645E-03

UPPER E# 1.300E 02 LOWER E# 1.200E 02

ANG. GP.	PROTON	S.D.	NEUTRON	S.D.
1	0.	0.	0.	0.
2	0.	0.	0.	0.
3	0.	0.	0.	0.
4	0.	0.	0.	0.
5	0.	0.	0.	0.
6	0.	0.	0.	0.
7	0.	0.	3.392E-04	3.392E-04
8	0.	0.	1.109E-03	0.
9	0.	0.	3.289E-03	3.289E-03

UPPER E# 1.200E 02 LOWER E# 1.100E 02

ANG. GP.	PROTON	S.D.	NEUTRON	S.D.
1	0.	0.	0.	0.

Fig. E5. Output from High Energy Transport Analysis

2	0.	0.	0.	0.
3	0.	0.	0.	0.
4	0.	0.	0.	0.
5	0.	0.	2.029E-04	2.029E-04
6	0.	0.	2.500E-04	2.500E-04
7	0.	0.	2.035E-03	0.
8	0.	0.	2.217E-03	2.217E-03
9	0.	0.	3.289E-03	3.289E-03
UPPER E# 1.100E 02 LOWER E# 1.000E 02				
ANG. GP.	PROTON	S.D.	NEUTRON	S.D.
1	0.	0.	0.	0.
2	0.	0.	0.	0.
3	0.	0.	0.	0.
4	0.	0.	0.	0.
5	0.	0.	2.029E-04	2.029E-04
6	0.	0.	0.	0.
7	0.	0.	6.784E-04	0.
8	0.	0.	1.109E-03	0.
9	0.	0.	0.	0.
UPPER E# 1.000E 02 LOWER E# 9.000E 01				
ANG. GP.	PROTON	S.D.	NEUTRON	S.D.
1	0.	0.	0.	0.
2	0.	0.	0.	0.
3	0.	0.	0.	0.
4	0.	0.	1.751E-04	1.751E-04
5	0.	0.	6.088E-04	2.029E-04
6	0.	0.	2.500E-04	2.500E-04
7	0.	0.	1.357E-03	6.784E-04
8	0.	0.	0.	0.
9	0.	0.	1.645E-03	1.645E-03
UPPER E# 9.000E 01 LOWER E# 8.000E 01				
ANG. GP.	PROTON	S.D.	NEUTRON	S.D.
1	0.	0.	0.	0.
2	0.	0.	0.	0.
3	0.	0.	0.	0.
4	0.	0.	0.	0.
5	0.	0.	2.029E-04	2.029E-04
6	0.	0.	2.000E-03	5.000E-04
7	0.	0.	1.018E-03	3.392E-04
8	0.	0.	1.109E-03	1.109E-03
9	0.	0.	0.	0.
UPPER E# 8.000E 01 LOWER E# 7.000E 01				
ANG. GP.	PROTON	S.D.	NEUTRON	S.D.
1	0.	0.	1.439E-04	1.439E-04
2	0.	0.	0.	0.
3	0.	0.	1.582E-04	1.582E-04
4	0.	0.	5.252E-04	5.252E-04
5	0.	0.	6.088E-04	2.029E-04
6	0.	0.	1.250E-03	7.500E-04
7	0.	0.	1.357E-03	6.784E-04
8	0.	0.	2.772E-03	5.543E-04
9	0.	0.	1.645E-03	1.645E-03

Fig. E6. Output from High Energy Transport Analysis

UPPER E# 7.000E 01		LOWER E# 6.000E 01			
ANG. GP.	PROTON	S.D.	NEUTRON	S.D.	
1	0.	0.	0.	0.	0.
2	0.	0.	0.	0.	0.
3	0.	0.	0.	0.	0.
4	0.	0.	1.582E-04	1.582E-04	1.582E-04
5	0.	0.	1.751E-04	1.751E-04	1.751E-04
6	0.	0.	8.117E-04	4.058E-04	4.058E-04
7	0.	0.	2.500E-03	10.000E-04	10.000E-04
8	0.	0.	1.696E-03	1.018E-03	1.018E-03
9	0.	0.	2.217E-03	0.	0.
			4.934E-03	4.934E-03	4.934E-03

UPPER E# 6.000E 01		LOWER E# 5.000E 01			
ANG. GP.	PROTON	S.D.	NEUTRON	S.D.	
1	0.	0.	0.	0.	0.
2	0.	0.	2.971E-04	0.	0.
3	0.	0.	7.911E-04	0.	0.
4	0.	0.	8.754E-04	1.582E-04	1.582E-04
5	0.	0.	6.088E-04	1.751E-04	1.751E-04
6	0.	0.	2.000E-03	2.029E-04	2.029E-04
7	0.	0.	2.035E-03	0.	0.
8	0.	0.	1.109E-03	1.357E-03	1.357E-03
9	0.	0.	0.	1.109E-03	1.109E-03
			0.	0.	0.

ENERGY SPECTRUM

ENERGY GP.	PROTON	S.D.	NEUTRON	S.D.	
1	0.	0.	0.	0.	0.
2	0.	0.	2.500E-05	2.500E-05	2.500E-05
3	0.	0.	1.750E-04	2.500E-05	2.500E-05
4	0.	0.	1.250E-04	2.500E-05	2.500E-05
5	0.	0.	3.500E-04	0.	0.
6	0.	0.	1.250E-04	2.500E-05	2.500E-05
7	0.	0.	2.500E-04	5.000E-05	5.000E-05
8	0.	0.	3.500E-04	0.	0.
9	0.	0.	5.750E-04	7.500E-05	7.500E-05
10	0.	0.	7.000E-04	1.000E-04	1.000E-04
11	0.	0.	7.750E-04	2.500E-05	2.500E-05

RADIAL SPREAD

RADIAL GP.	NUMBER	WITHIN	S.D.	
1	0.	0.	0.	0.
2	0.	0.	0.	0.
3	0.	0.	0.	0.
4	0.	0.	0.	0.
5	0.	0.	0.	0.
6	0.	0.	0.	0.
7	0.	0.	0.	0.
8	7.143E-03	7.143E-03	7.143E-03	7.143E-03
9	2.143E-02	2.143E-02	2.143E-02	2.143E-02
10	5.021E-02	5.021E-02	3.550E-02	3.550E-02
11	7.185E-02	7.185E-02	4.244E-02	4.244E-02
12	7.899E-02	7.899E-02	4.958E-02	4.958E-02
13	1.370E-01	1.370E-01	4.874E-02	4.874E-02
14	2.235E-01	2.235E-01	7.647E-02	7.647E-02
15	3.826E-01	3.826E-01	1.032E-01	1.032E-01
16	6.515E-01	6.515E-01	4.853E-02	4.853E-02
17	1.000E 00	1.000E 00	0.	0.

TOTAL NEUTS#	3.450E-02	S.D.#	5.000E-04
TRANSMITTED NEUTS#	3.450E-02	S.D.#	5.000E-04

Fig. E7. Output from High Energy Transport Analysis

THICKNESS#	1.000E 01		
LOWER ENERGY#	4.000E 01	UPPER ENERGY#	5.000E 01
ANG. GP.	FLUX	S.D.	
1	0.	0.	
2	0.	0.	
3	9.573E-06	9.573E-06	
4	0.	0.	
5	0.	0.	
6	1.847E-05	1.847E-05	
7	0.	0.	
8	0.	0.	
9	1.345E-01	3.668E-04	
TOTAL FLUX#	2.048E-03	S.D.#	5.234E-06
LOWER ENERGY#	3.000E 01	UPPER ENERGY#	4.000E 01
ANG. GP.	FLUX	S.D.	
1	0.	0.	
2	0.	0.	
3	7.850E-06	7.850E-06	
4	0.	0.	
5	0.	0.	
6	2.356E-05	2.356E-05	
7	0.	0.	
8	9.484E-05	7.348E-05	
9	6.203E-01	1.923E-03	
TOTAL FLUX#	9.437E-03	S.D.#	2.483E-05
LOWER ENERGY#	2.000E 01	UPPER ENERGY#	3.000E 01
ANG. GP.	FLUX	S.D.	
1	0.	0.	
2	0.	0.	
3	6.005E-06	6.005E-06	
4	0.	0.	
5	0.	0.	
6	2.231E-05	1.373E-05	
7	3.965E-05	2.312E-05	
8	3.255E-04	3.162E-05	
9	4.751E-01	1.537E-03	
TOTAL FLUX#	7.243E-03	S.D.#	2.266E-05

Fig. E8. Output from High Energy Transport Analysis

ANG. GP.	FLUX	S.D.
1	0.	0.
2	0.	0.
3	3.992E-06	3.992E-06
4	0.	0.
5	0.	0.
6	2.431E-05	3.572E-07
7	7.239E-05	9.940E-08
8	3.883E-04	5.908E-05
9	3.160E-01	1.053E-03
TOTAL FLUX#	4.828E-03	S.D.# 1.932E-05

ANG. GP.	FLUX	S.D.
1	0.	0.
2	9.979E-08	9.979E-08
3	1.617E-06	1.617E-06
4	0.	0.
5	0.	0.
6	9.852E-06	1.448E-07
7	2.980E-05	4.245E-07
8	1.782E-04	2.399E-05
9	1.280E-01	4.264E-04
TOTAL FLUX#	1.958E-03	S.D.# 7.882E-06

Fig. E9. Output from High Energy Transport Analysis

ELEMENT# 13
 SIGMA# 1
 INTERPOLATION 8
 POINTS# 188
 MASS# 26.98000

TOTAL CROSS SECTION

ENERGY (EV)	CROSS SECTION	ENERGY (EV)	CROSS SECTION	ENERGY (EV)	CROSS SECTION	ENERGY (EV)	CROSS SECTION
0.10000E 09	0.17500E 01	0.25600E 07	0.31000E 01	0.34700E 06	0.33000E 01	0.46800E 05	0.31000E 01
0.18000E 08	0.17500E 01	0.24400E 07	0.30000E 01	0.33000E 06	0.36000E 01	0.44500E 05	0.35100E 01
0.17100E 08	0.17100E 01	0.23200E 07	0.29500E 01	0.31400E 06	0.39500E 01	0.42400E 05	0.44500E 01
0.16300E 08	0.16900E 01	0.22100E 07	0.31600E 01	0.29900E 06	0.42200E 01	0.40300E 05	0.58500E 01
0.15500E 08	0.16800E 01	0.21000E 07	0.32500E 01	0.28400E 06	0.42900E 01	0.38300E 05	0.92000E 01
0.14750E 08	0.16700E 01	0.20000E 07	0.32500E 01	0.27000E 06	0.26300E 01	0.36500E 05	0.22000E 02
0.14000E 08	0.16600E 01	0.19000E 07	0.26000E 01	0.25700E 06	0.27000E 01	0.34700E 05	0.25000E 02
0.13300E 08	0.16600E 01	0.18100E 07	0.28000E 01	0.24400E 06	0.29600E 01	0.33000E 05	0.10600E 02
0.12700E 08	0.16600E 01	0.17200E 07	0.33000E 01	0.23300E 06	0.32800E 01	0.31400E 05	0.32000E 01
0.12100E 08	0.16600E 01	0.16300E 07	0.30500E 01	0.22100E 06	0.59000E 01	0.29900E 05	0.85000E 00
0.11500E 08	0.16700E 01	0.15500E 07	0.30500E 01	0.21000E 06	0.62800E 01	0.28400E 05	0.56100E 00
0.10900E 08	0.16700E 01	0.14800E 07	0.30800E 01	0.20000E 06	0.40000E 01	0.27000E 05	0.51100E 00
0.10400E 08	0.16700E 01	0.14100E 07	0.31000E 01	0.19000E 06	0.36100E 01	0.25700E 05	0.50500E 00
0.98900E 07	0.16800E 01	0.13400E 07	0.31000E 01	0.18100E 06	0.42200E 01	0.24400E 05	0.57600E 00
0.94100E 07	0.16800E 01	0.12700E 07	0.34000E 01	0.17200E 06	0.55500E 01	0.23300E 05	0.62200E 00
0.89500E 07	0.16900E 01	0.12100E 07	0.41500E 01	0.16300E 06	0.98000E 01	0.22100E 05	0.66700E 00
0.85100E 07	0.17300E 01	0.11500E 07	0.39000E 01	0.15500E 06	0.10100E 02	0.21000E 05	0.72300E 00
0.81000E 07	0.18400E 01	0.10960E 07	0.28000E 01	0.14800E 06	0.99000E 01	0.20000E 05	0.76000E 00
0.77000E 07	0.19500E 01	0.10420E 07	0.26000E 01	0.14100E 06	0.51000E 01	0.19000E 05	0.81300E 00
0.73300E 07	0.20200E 01	0.99100E 06	0.27000E 01	0.13400E 06	0.14000E 01	0.18100E 05	0.84800E 00
0.69700E 07	0.20600E 01	0.94300E 06	0.28500E 01	0.12700E 06	0.16400E 01	0.17200E 05	0.89000E 00
0.66300E 07	0.21100E 01	0.89700E 06	0.32000E 01	0.12100E 06	0.11200E 02	0.16300E 05	0.93100E 00
0.63000E 07	0.21500E 01	0.85300E 06	0.40500E 01	0.11500E 06	0.22100E 01	0.15500E 05	0.97000E 00
0.60000E 07	0.21500E 01	0.81200E 06	0.60000E 01	0.10960E 06	0.30000E 01	0.14800E 05	0.10100E 01
0.57000E 07	0.21500E 01	0.77200E 06	0.35700E 01	0.10420E 06	0.39000E 01	0.14100E 05	0.10400E 01
0.54300E 07	0.21500E 01	0.73400E 06	0.34600E 01	0.99100E 05	0.65000E 01	0.13400E 05	0.10700E 01
0.51600E 07	0.22000E 01	0.69900E 06	0.34900E 01	0.94300E 05	0.11000E 02	0.12700E 05	0.11200E 01
0.49100E 07	0.22400E 01	0.66600E 06	0.34900E 01	0.89700E 05	0.17000E 02	0.12100E 05	0.11500E 01
0.46700E 07	0.22500E 01	0.63200E 06	0.34200E 01	0.85300E 05	0.11500E 02	0.11500E 05	0.11800E 01
0.44400E 07	0.23000E 01	0.60100E 06	0.37000E 01	0.81200E 05	0.46000E 01	0.10960E 05	0.12100E 01
0.42300E 07	0.23700E 01	0.57200E 06	0.40000E 01	0.77200E 05	0.23500E 01	0.10430E 05	0.12700E 01
0.40200E 07	0.25500E 01	0.54400E 06	0.37700E 01	0.73400E 05	0.16200E 01	0.99100E 04	0.13000E 01
0.38200E 07	0.27000E 01	0.51800E 06	0.35600E 01	0.69900E 05	0.14800E 01	0.94300E 04	0.94300E 01
0.36400E 07	0.27000E 01	0.49200E 06	0.35000E 01	0.66600E 05	0.15100E 01	0.89700E 04	0.13000E 01
0.34600E 07	0.27000E 01	0.46800E 06	0.36500E 01	0.63200E 05	0.15700E 01	0.85300E 04	0.13100E 01
0.32900E 07	0.27000E 01	0.44500E 06	0.50000E 01	0.60100E 05	0.16500E 01	0.81200E 04	0.13200E 01
0.31300E 07	0.27000E 01	0.42400E 06	0.59500E 01	0.57200E 05	0.18000E 01	0.77200E 04	0.13400E 01
0.29700E 07	0.27400E 01	0.40300E 06	0.36700E 01	0.54400E 05	0.20400E 01	0.73400E 04	0.13600E 01
0.28300E 07	0.29000E 01	0.38300E 06	0.34000E 01	0.51800E 05	0.23000E 01	0.69900E 04	0.14400E 01
0.26900E 07	0.30600E 01	0.36500E 06	0.34000E 01	0.49200E 05	0.25700E 01	0.66600E 04	0.15300E 01

Fig. E10. Total Cross Sections for Aluminum; $100 \text{ MeV} \geq E_n \geq 6.66 \text{ keV}$

ELEMENT# 13
 SIGMA# 2
 INTERPOLATION 8
 POINTS# 188
 MASS# 26.98000

SCAT. CROSS SECTION

ENERGY (EV)	CROSS SECTION	ENERGY (EV)	CROSS SECTION	ENERGY (EV)	CROSS SECTION	ENERGY (EV)	CROSS SECTION
0.10000E 09	0.16930E 01	0.25600E 07	0.31040E 01	0.34700E 06	0.33000E 01	0.46800E 05	0.31000E 01
0.18000E 08	0.16930E 01	0.24400E 07	0.29990E 01	0.33000E 06	0.36000E 01	0.44500E 05	0.35100E 01
0.17100E 08	0.16450E 01	0.23200E 07	0.29510E 01	0.31400E 06	0.39500E 01	0.42400E 05	0.44500E 01
0.16300E 08	0.16190E 01	0.22100E 07	0.31560E 01	0.29900E 06	0.42200E 01	0.40300E 05	0.58500E 01
0.15500E 08	0.16050E 01	0.21000E 07	0.32500E 01	0.28400E 06	0.42900E 01	0.38300E 05	0.92000E 01
0.14750E 08	0.15970E 01	0.20000E 07	0.32500E 01	0.27000E 06	0.26300E 01	0.36500E 05	0.22000E 02
0.14000E 08	0.15850E 01	0.19000E 07	0.26030E 01	0.25700E 06	0.27000E 01	0.34700E 05	0.25000E 02
0.13300E 08	0.15770E 01	0.18100E 07	0.27980E 01	0.24400E 06	0.29600E 01	0.33000E 05	0.10600E 02
0.12700E 08	0.15760E 01	0.17200E 07	0.32950E 01	0.23300E 06	0.32800E 01	0.31400E 05	0.32000E 01
0.12100E 08	0.15700E 01	0.16300E 07	0.30530E 01	0.22100E 06	0.59000E 01	0.29900E 05	0.85000E 00
0.11500E 08	0.15750E 01	0.15500E 07	0.30500E 01	0.21000E 06	0.62800E 01	0.28400E 05	0.56100E 00
0.10900E 08	0.15710E 01	0.14800E 07	0.30770E 01	0.20000E 06	0.40000E 01	0.27000E 05	0.51100E 00
0.10400E 08	0.15690E 01	0.14100E 07	0.30980E 01	0.19000E 06	0.36100E 01	0.25700E 05	0.50500E 00
0.98900E 07	0.15650E 01	0.13400E 07	0.30960E 01	0.18100E 06	0.42200E 01	0.24400E 05	0.57600E 00
0.94100E 07	0.15630E 01	0.12700E 07	0.34010E 01	0.17200E 06	0.55500E 01	0.23300E 05	0.62200E 00
0.89500E 07	0.15730E 01	0.12100E 07	0.41500E 01	0.16300E 06	0.78000E 01	0.22100E 05	0.66700E 00
0.85100E 07	0.16420E 01	0.11500E 07	0.39000E 01	0.15500E 06	0.10100E 02	0.21000E 05	0.72300E 00
0.81000E 07	0.17520E 01	0.10960E 07	0.27960E 01	0.14800E 06	0.99000E 01	0.20000E 05	0.76000E 00
0.77000E 07	0.18720E 01	0.10420E 07	0.25980E 01	0.14100E 06	0.51000E 01	0.19000E 05	0.81300E 00
0.73300E 07	0.19500E 01	0.99100E 06	0.27000E 01	0.13400E 06	0.14000E 01	0.18100E 05	0.84800E 00
0.69700E 07	0.19980E 01	0.94300E 06	0.28500E 01	0.12700E 06	0.16400E 01	0.17200E 05	0.89000E 00
0.66300E 07	0.20530E 01	0.89700E 06	0.32000E 01	0.12100E 06	0.11200E 02	0.16300E 05	0.93100E 00
0.63000E 07	0.20970E 01	0.85300E 06	0.40500E 01	0.11500E 06	0.22100E 01	0.15500E 05	0.97000E 00
0.60000E 07	0.21070E 01	0.81200E 06	0.60000E 01	0.10960E 06	0.30000E 01	0.14800E 05	0.10100E 01
0.57000E 07	0.21020E 01	0.77200E 06	0.35700E 01	0.10420E 06	0.39000E 01	0.14100E 05	0.10400E 01
0.54300E 07	0.21200E 01	0.73400E 06	0.34600E 01	0.99100E 05	0.65000E 01	0.13400E 05	0.10700E 01
0.51600E 07	0.21730E 01	0.69900E 06	0.34900E 01	0.94300E 05	0.11000E 02	0.12700E 05	0.11200E 01
0.49100E 07	0.22240E 01	0.66600E 06	0.34900E 01	0.89700E 05	0.17000E 02	0.12100E 05	0.11500E 01
0.46700E 07	0.21860E 01	0.63200E 06	0.34200E 01	0.85300E 05	0.11500E 02	0.11500E 05	0.11800E 01
0.44400E 07	0.22870E 01	0.60100E 06	0.37000E 01	0.81200E 05	0.46000E 01	0.10960E 05	0.12100E 01
0.42300E 07	0.23540E 01	0.57200E 06	0.40000E 01	0.77200E 05	0.23500E 01	0.10430E 05	0.12700E 01
0.40200E 07	0.25440E 01	0.54400E 06	0.37700E 01	0.73400E 05	0.16200E 01	0.99100E 04	0.13000E 01
0.38200E 07	0.26940E 01	0.51800E 06	0.35600E 01	0.69900E 05	0.14800E 01	0.94300E 04	0.13000E 01
0.36400E 07	0.26960E 01	0.49200E 06	0.35000E 01	0.66600E 05	0.15100E 01	0.89700E 04	0.13000E 01
0.34600E 07	0.27000E 01	0.46800E 06	0.36500E 01	0.63200E 05	0.15700E 01	0.85300E 04	0.13100E 01
0.32900E 07	0.26980E 01	0.44500E 06	0.50000E 01	0.60100E 05	0.16500E 01	0.81200E 04	0.13200E 01
0.31300E 07	0.26980E 01	0.42400E 06	0.59500E 01	0.57200E 05	0.18000E 01	0.77200E 04	0.13400E 01
0.29700E 07	0.27400E 01	0.40300E 06	0.36700E 01	0.54400E 05	0.20400E 01	0.73400E 04	0.13600E 01
0.28300E 07	0.29010E 01	0.38300E 06	0.34000E 01	0.51800E 05	0.23000E 01	0.69900E 04	0.14400E 01
0.26900E 07	0.30590E 01	0.36500E 06	0.34000E 01	0.49200E 05	0.25700E 01	0.66600E 04	0.15300E 01

Fig. E12. Scattering Cross Sections for Aluminum; $100 \text{ MeV} \geq E_p \geq 6.66 \text{ keV}$

ELEMENT# 13
 SIGMA# 7
 INTERPOLATION 8
 POINTS# 139
 MASS# 26.98000

F 1

ENERGY (EV)	CROSS SECTION						
0.10000E 09	0.70900E 00	0.25600E 07	0.36200E-00	0.34700E 06	0.98000E-01	0.46800E 05	1.00000E-02
0.18000E 08	0.70900E 00	0.24400E 07	0.35300E-00	0.33000E 06	0.82000E-01	0.44500E 05	0.90000E-02
0.17100E 08	0.70300E 00	0.23200E 07	0.34400E-00	0.31400E 06	0.70000E-01	0.42400E 05	0.90000E-02
0.16300E 08	0.70000E 00	0.22100E 07	0.33800E-00	0.29900E 06	0.68000E-01	0.40300E 05	0.80000E-02
0.15500E 08	0.69500E 00	0.21000E 07	0.33000E-00	0.28400E 06	0.68000E-01	0.38300E 05	0.80000E-02
0.14750E 08	0.68600E 00	0.20000E 07	0.32100E-00	0.27000E 06	0.69000E-01	0.36500E 05	0.70000E-02
0.14000E 08	0.68000E 00	0.19000E 07	0.31500E-00	0.25700E 06	0.73000E-01	0.34700E 05	0.70000E-02
0.13300E 08	0.67300E 00	0.18100E 07	0.31200E-00	0.24400E 06	0.77000E-01	0.33000E 05	0.70000E-02
0.12700E 08	0.66400E 00	0.17200E 07	0.30900E-00	0.23300E 06	0.79000E-01	0.31400E 05	0.60000E-02
0.12100E 08	0.65700E 00	0.16300E 07	0.30600E-00	0.22100E 06	0.79000E-01	0.29900E 05	0.60000E-02
0.11500E 08	0.64800E 00	0.15500E 07	0.30500E-00	0.21000E 06	0.78000E-01	0.28400E 05	0.60000E-02
0.10900E 08	0.63900E 00	0.14800E 07	0.30300E-00	0.20000E 06	0.73000E-01	0.27000E 05	0.60000E-02
0.10400E 08	0.63300E 00	0.14100E 07	0.28500E-00	0.19000E 06	0.65000E-01	0.25700E 05	0.50000E-02
0.98900E 07	0.62200E 00	0.13400E 07	0.27300E-00	0.18100E 06	0.59000E-01	0.24400E 05	0.50000E-02
0.94100E 07	0.61500E 00	0.12700E 07	0.31900E-00	0.17200E 06	0.52000E-01	0.23300E 05	0.50000E-02
0.89500E 07	0.60400E 00	0.12100E 07	0.31800E-00	0.16300E 06	0.44000E-01	0.22100E 05	0.40000E-02
0.85100E 07	0.59800E 00	0.11500E 07	0.28000E-00	0.15500E 06	0.38000E-01	0.21000E 05	-0.
0.81000E 07	0.59000E 00	0.10900E 07	0.24000E-00	0.14800E 06	0.33000E-01	0.17200E 04	-0.
0.77000E 07	0.58000E 00	0.10420E 07	0.24100E-00	0.14100E 06	0.31000E-01	0.10000E-03	-0.
0.73300E 07	0.57000E 00	0.99100E 06	0.25600E-00	0.13400E 06	0.29000E-01	0.	0.
0.69700E 07	0.55900E 00	0.94300E 06	0.29000E-00	0.12700E 06	0.27000E-01	0.	0.
0.66300E 07	0.55000E 00	0.89700E 06	0.32500E-00	0.12100E 06	0.26000E-01	0.	0.
0.63000E 07	0.54000E 00	0.85300E 06	0.31300E-00	0.11500E 06	0.24000E-01	0.	0.
0.60000E 07	0.53000E 00	0.81200E 06	0.25300E-00	0.10960E 06	0.23000E-01	0.	0.
0.57000E 07	0.52200E 00	0.77200E 06	0.22000E-00	0.10420E 06	0.22000E-01	0.	0.
0.54300E 07	0.51100E 00	0.73400E 06	0.21300E-00	0.99100E 05	0.21000E-01	0.	0.
0.51600E 07	0.49900E-00	0.69900E 06	0.21000E-00	0.94300E 05	0.20000E-01	0.	0.
0.49100E 07	0.49000E-00	0.66600E 06	0.21000E-00	0.89700E 05	0.19000E-01	0.	0.
0.46700E 07	0.47800E-00	0.63200E 06	0.21000E-00	0.85300E 05	0.18000E-01	0.	0.
0.44400E 07	0.46500E-00	0.60100E 06	0.21000E-00	0.81200E 05	0.17000E-01	0.	0.
0.42300E 07	0.45800E-00	0.57200E 06	0.21000E-00	0.77200E 05	0.16000E-01	0.	0.
0.40200E 07	0.44900E-00	0.54400E 06	0.20600E-00	0.73400E 05	0.15000E-01	0.	0.
0.38200E 07	0.43800E-00	0.51800E 06	0.19300E-00	0.69900E 05	0.15000E-01	0.	0.
0.36400E 07	0.42900E-00	0.49200E 06	0.17500E-00	0.66600E 05	0.14000E-01	0.	0.
0.34600E 07	0.41900E-00	0.46800E 06	0.15000E-00	0.63200E 05	0.13000E-01	0.	0.
0.32900E 07	0.40900E-00	0.44500E 06	0.13300E-00	0.60100E 05	0.13000E-01	0.	0.
0.31300E 07	0.40100E-00	0.42400E 06	0.12500E-00	0.57200E 05	0.12000E-01	0.	0.
0.29700E 07	0.39000E-00	0.40300E 06	0.12000E-00	0.54400E 05	0.11000E-01	0.	0.
0.28300E 07	0.38200E-00	0.38300E 06	0.11600E-00	0.51800E 05	0.11000E-01	0.	0.
0.26900E 07	0.37100E-00	0.36500E 06	0.10700E-00	0.49200E 05	1.00000E-02	0.	0.

Fig. E14. Average Cosines of the Scattering Angle in the Center-of-Mass System for Aluminum;
 $100 \text{ MeV} \geq E_n \geq 100 \text{ eV}$

The reactor data tape was prepared for running Code 6 (see pages 49-52) with the input shown in Fig. E15.

```
CODE 6
O5R REACTOR DATA TAPE FOR ALUMINUM.
  1      5.E+07      1.
  1      128
 13      1      2      .0603116
 13
```

Fig. E15.

O5R and its analysis routines were then run as a chain job with O5R as the first link and the systems EXIT subroutine replaced by an O5R EXIT subroutine which merely called CHAIN (2,3).

The input to O5R and its analysis routine is shown in Fig. E16. The first eight lines are the O5R input, the next six the input for GEOM, and the last four the input to the O5R Analysis Code, which was described on pages 26 and 27.

The results of running O5R and the analysis code are shown in Figs. E17 through E20. At the start of the run, O5R prints the input information, as shown in Fig. E17. At the beginning of each batch O5R prints, as shown in Fig. E18, the starting random number for the batch, the sums of the values of the X coordinates, the Y coordinates, the Z coordinates, and the sum of the values of the weights for the batch. NRNM is the number of storage cells available for neutrons produced by splitting during the run of the batch (= NMOST-NSTRT). NEWNM is the last name given to a neutron. NMEN is the number of cells actually required for the NEWNM neutrons. The latter two quantities are of little or no interest to the user. ERRS indicates the number of tape errors which occurred during the run of the batch. NFISH is the number of neutrons put on the fission tape. FWATE is the product of the weight, an input number, and NFISH, while

```

701 NEUTS FROM 4000 160 MEV P ON 10 CM AL.
701 750 1 1 1 0 1 0
1 1 3 0 5.E7
0. 1. 0. 0. 1.
0. 0. 0. 1 1
1 128 0 0. 1
27.
1 2 11000010000100100100000000000110

```

```

X ZONE BOUNDARIES(0.,10.)
Y ZONE BOUNDARIES(-1.E20,1.E20)
Z ZONE BOUNDARIES(-1.E20,1.E20)
ZONE(1,1,1)XB(0.,10.)YB(-1.E20,1.E20)ZB(-1.E20,1.E20)
BLOCKS1

```

```

END
4 9 10. 50. 1. 10. 4000.
.1737 .342 .5 .6428 .766 .866 .9397
.9848 1.
10.

```

Fig. E16. Input to O5R and O5R Analysis Routine

```

701 NEUTS FROM 4000 160 MEV P ON 10 CM AL.
NSTART# 701    NMOST# 750    NITS# 1    NHOW# 1    QUIT# 1    FISTAPE# 0    ECUT# 0.10000E 01    THERM# 0
MEDIA# 1    FCNES# 1    CONSTANTS TAPES# 3, 3    EMAX OF TAPE# 0.50000E 08
ESOURCE# 0.    U# 0.1000E 01    V# 0.    W# 0.    WEIGHT# 0.1000E 01
X# 0.    Y# 0.    Z# 0.    MEDIUM# 1    REGION# 1
MEDIUM    SCATS    PCINTS    FISSION    FLOW    ANISCTROPY
1    1    128    NO    0.    1
MEDIUM    MASSES
1    27.000000
COLLISION TAPES FROM 1 TO 2
WORDS PER COLLISION 8    COLLISIONS PER RECORO 15    PARAMETERS 1100001000010010010000000000001100000
COLLISION TAPE PARAMETERS SELECTED
NCCLL
NAME
X1
U0
X0
WTO
LAPDA
SQ
RANDOM# 343277244615
NSPLT# NO    NKILL# NO    RGMAX# -0    GPMAX# 40
X ZONE BCUNDARIES(0.,10.)
Y ZONE BCUNDARIES(-1.E20,1.E20)
Z ZONE BCUNDARIES(-1.E20,1.E20)
ZONE(1,1,1)XB(0.,10.)YB(-1.E20,1.E20)ZB(-1.E20,1.E20)
BLOCKSI
END

```

Fig. E17. 05R Output

RANDOM	XAVE	YAVE	ZAVE	WEIGHT	NROOM	NEWM	NMEM	ERRS	NFISH	FWATE	FTOTL
343277244615	0.25197E 04	0.10045E 02	0.22894E 02	0.70100E 03	49	701	701	0	0 0.	0.	
105673570231-0.	0.	-0.	0.		49	701	701	0	0 0.	0.	

Fig. E18. 05R Output

NX# 4 NA# 9 TLIM1# 0.100E 02 TLIM2# 0.500E 02 DLI# 1. DL2# 10. TOTWT# 0.400E 04 T# 0.100E 02
 ANGULAR INTERVALS
 1.737E-01 3.420E-01 5.000E-01 6.428E-01 7.660E-01 8.660E-01 9.397E-01 9.848E-01
 1.000E 00

XLOW# 7.5000E 07 XHIGH# 1.0000E 01

ENERGY GROUP	1	2	3	4	5	6	7	8	9	10	11
8.76E-04	2.11E-03	6.89E-03	6.47E-03	9.27E-03	1.69E-02	6.67E-03	1.97E-02	2.93E-02	7.18E-03		
9.14E-05	5.47E-03	4.17E-03	8.03E-03	9.31E-03	1.00E-02	2.71E-02	8.97E-03	7.05E-02	8.36E-03		
4.24E-03	5.85E-03	8.09E-03	5.28E-03	1.34E-02	7.26E-03	1.08E-02	1.04E-02	4.72E-02	8.11E-03		
4.18E-05	3.54E-03	4.98E-03	2.12E-03	5.89E-03	8.99E-03	1.35E-02	2.03E-02	1.13E-02	5.40E-03		
3.24E-03	5.18E-03	4.74E-03	4.05E-03	8.36E-03	1.05E-02	1.36E-02	5.41E-03	0.	6.09E-03		
7.12E-05	4.90E-03	8.02E-03	6.58E-03	7.16E-03	9.77E-03	1.11E-02	6.11E-03	1.02E-02	6.15E-03		
4.76E-05	4.80E-03	5.12E-03	3.13E-03	5.49E-03	1.95E-03	5.23E-03	7.80E-03	2.64E-02	4.08E-03		
2.92E-05	2.17E-03	2.40E-03	2.96E-03	1.04E-02	0.	3.82E-03	6.13E-03	1.12E-02	3.18E-03		
1.31E-03	3.86E-03	6.78E-03	4.39E-03	4.81E-03	7.67E-03	0.	4.96E-03	0.	4.16E-03		

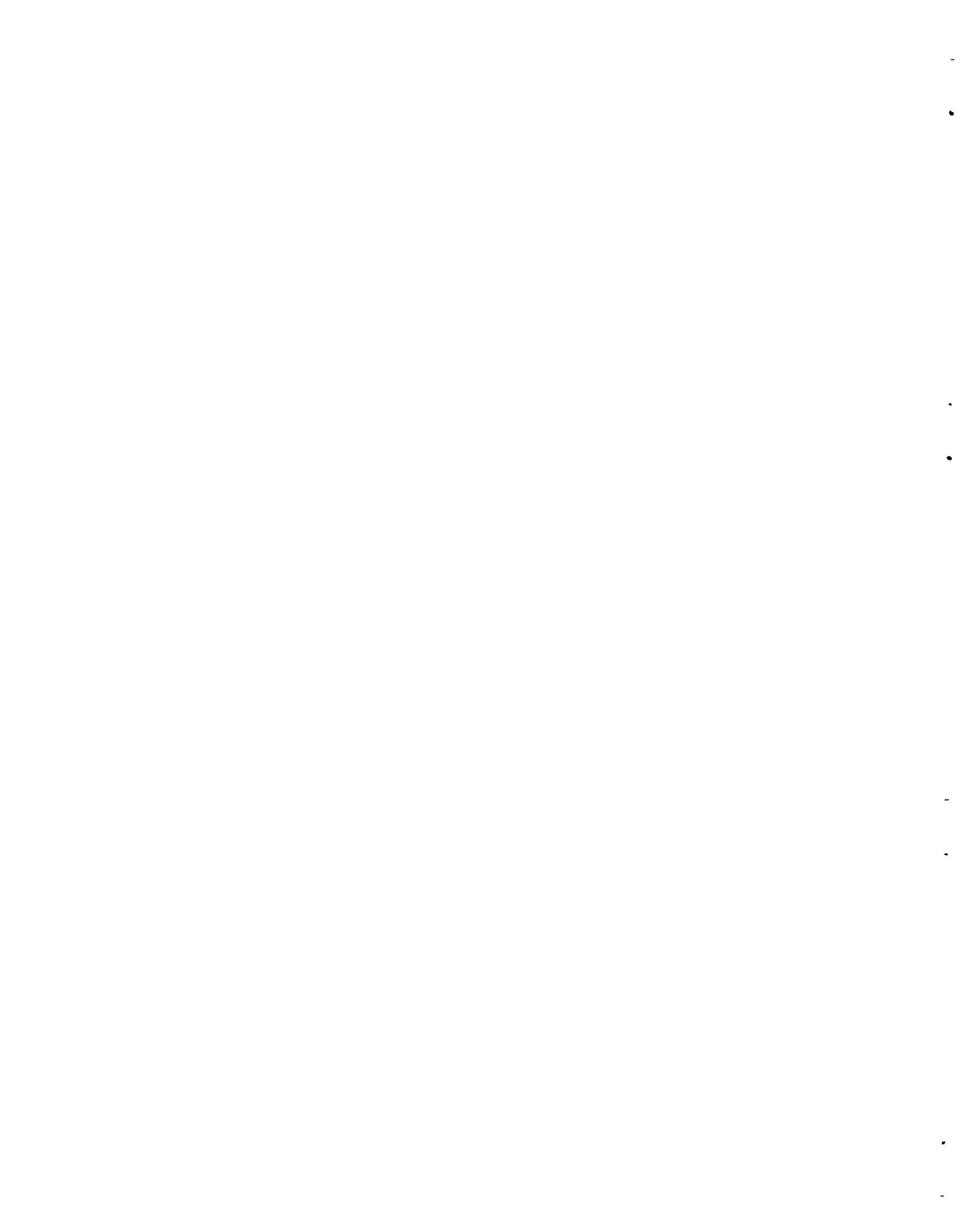
Fig. E19. 05R Output

ENERGY GROUP 10	7.61E-08	2.66E-03	1.94E-03	3.59E-03	3.11E-03	6.53E-03	1.18E-02	0.	0.	3.17E-03
ENERGY GROUP 11	0.	0.	0.	0.	0.	0.	0.	0.	0.	6.78E-04
ENERGY GROUP 12	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.48E-03
ENERGY GROUP 13	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.16E-03
ENERGY GROUP 14	0.	0.	0.	0.	0.	0.	0.	0.	0.	2.08E-03
TOTAL NEUTRONS										
RNEUT	1.4859E 02	3.3181E 02	4.6285E 02	4.3954E 02						
QNEUT	2.1243E 02	4.8633E 02	6.8405E 02	7.0327E 02						
QNEUT(J)/TOTWT	5.3107E-02	1.2158E-01	1.7101E-01	1.7582E-01						
RNEUT(J)/TCTWT	3.7149E-02	8.2952E-02	1.1571E-01	1.0988E-01						

Fig. E20. O5R Output. The R neutrons are those penetrating the slab, QNEUT is the total of neutrons penetrating and reflected from the slab.

FTOTL is the total probability for the batch of producing fissions. The next line gives the value of the random number at the end of the batch and the sums of the values of X, Y, and Z coordinates of those members of the batch which survive to energy ECUT, along with the total weight of the survivors. (In the example shown no neutrons survived.) Other entries on this line are not relevant.

The total O5R running time for the 701 neutrons was 4.2 min.



ORNL-3610
 UC-34 - Physics
 TID-4500 (31st ed.)

INTERNAL DISTRIBUTION

- | | |
|-------------------------------------|-----------------------------------|
| 1. Biology Library | 71. F. C. Maienschein |
| 2-4. Central Research Library | 72. J. A. Martin |
| 5. Reactor Division Library | 73. Betty F. Maskewitz |
| 6-7. ORNL - Y-12 Technical Library | 74. J. Neufeld |
| Document Reference Section | 75. R. Peelle |
| 8-57. Laboratory Records Department | 76. S. K. Penny |
| 58. Laboratory Records, ORNL R.C. | 77. R. T. Santoro |
| 59. R. G. Alsmiller, Jr. | 78. M. J. Skinner |
| 60. R. D. Birkhoff | 79. J. A. Swartout |
| 61. E. P. Blizzard | 80. J. T. Thomas |
| 62. H. P. Carter | 81. J. E. Turner |
| 63. J. K. Dickins | 82. H. A. Wright |
| 64. L. C. Emerson | 83. A. M. Weinberg |
| 65. R. M. Freestone, Jr. | 84. R. A. Charpie (consultant) |
| 66. W. A. Gibson | 85. P. F. Gast (consultant) |
| 67. F. F. Haywood | 86. M. L. Goldberger (consultant) |
| 68. W. H. Jordan | 87. R. F. Taschek (consultant) |
| 69. W. E. Kinney | 88. T. J. Thompson (consultant) |
| 70. C. E. Larson | |

EXTERNAL DISTRIBUTION

89. Air Force Weapons Laboratory, WLRB-1 (Attn: Lt. Duane Adams), Kirtland Air Force Base, New Mexico
90. Lt. Col. Ralph G. Allen, Jr., USAF Aerospace Medical Center, Brooks AFB, Texas
91. C. K. Bauer, Lockheed-Georgia Company, Marietta, Georgia
92. N. Barr, Chief, Radiological Physics Branch, Division of Biology and Medicine, U.S. Atomic Energy Commission, Washington, D.C.
93. E. R. Beever, North American Aviation, Inc., Downey, California
94. M. J. Berger, National Bureau of Standards, Washington, D.C.
95. S. Bresticker, Grumman Aircraft Engineering Corporation, Bethpage, Long Island, New York
96. Fred Casal, NASA Headquarters, Washington, D.C.
97. Mac C. Chapman, Northrop Space Laboratories, Hawthorne, California
98. T. H. Colvin, Bendix Systems Division, Ann Arbor, Michigan
99. Colonel J. Conner, National Aeronautics and Space Administration, Washington, D.C.
100. Capt. R. F. Cooper, ASRPE-20, Wright-Patterson AFB, Ohio
101. R. B. Curtis, University of Indiana, Bloomington, Indiana
102. C. A. Dempsey, 6570 AMRL, Wright-Patterson AFB, Ohio
103. T. W. De Vries, General Dynamics/Fort Worth, Fort Worth, Texas.
104. Ed Divita, 1504 Doxbury Road, Towson 4, Maryland
105. D. L. Dye, Boeing Aircraft Company, Seattle, Washington

106. D. W. Drawbaugh, Westinghouse Electric Company, Pittsburgh, Pennsylvania
107. J. E. Duberg, NASA, Langley Field, Virginia
108. F. Felberg, Jet Propulsion Laboratory, Pasadena, California
109. Trutz Foelsche, NASA, Langley Research Center, Hampton, Virginia
110. R. E. Fortney, Northrop Space Laboratories, Hawthorne, California
111. E. M. Finkelman, Grumman Aircraft, Bethpage, New York
112. Stan Freden, Aerospace Corporation, El Segundo, California
113. K. D. George, Reactor Requirements Office, Pictinny Arsenal, Dover, New Jersey
114. R. A. Glass, Lockheed Missiles and Space Co., Sunnyvale, California
115. R. V. Glowszowski, McDonnell Aircraft, St. Louis, Missouri
- 116-120. W. L. Gill, NASA, Manned Spacecraft Center, Houston, Texas
121. R. C. Good, Jr., General Electric Company, Philadelphia, Pennsylvania
122. Air Force Space Systems Division, SSTA (Attn: Lt. Col. Edward Harney), Air Force Unit Post Office, Los Angeles, California
123. R. L. Harvey, General Electric Company, San Jose, California
124. Wilmot N. Hess, Chief, NASA, Goddard Space Flight Center, Greenbelt, Maryland
125. W. C. Hulten, NASA - IRD, Langley Research Center, Hampton, Virginia.
126. R. I. Hildebrand, Lewis Research Center, Cleveland, Ohio
127. C. W. Hill, Lockheed-Georgia Company, Marietta, Georgia
128. George Joanou, General Atomic, San Diego, California
129. Jim Johnson, LTV Research Center, Dallas, Texas
130. C. F. Johnson, General Dynamics/Fort Worth, Fort Worth, Texas
131. I. M. Karp, Lewis Research Center, Cleveland, Ohio
132. F. L. Keller, Aerospace Corporation, El Segundo, California
133. J. W. Keller, NASA Headquarters, Washington, D.C.
134. J. F. Kenney, Boeing Scientific Research Laboratories, Seattle, Washington
135. M. R. Kinsler, North American Aviation, Inc., Downey, California
136. Sol Krasner, Office of Naval Research, Washington, D.C.
137. E. C. Kidd, Zone S-71, General Dynamics/Fort Worth, Fort Worth, Texas
138. David Langford, Pratt and Whitney Aircraft, East Hartford, Connecticut
139. W. H. Langham, Los Alamos Scientific Laboratory, Los Alamos, New Mexico
140. Borje Larsson, The Gustaf Werner Institute, University of Uppsala, Uppsala, Sweden
141. L. R. Lewis, Bendix Systems Division, Ann Arbor, Michigan
142. Martin Leimdorfer, Research Institute of National Defense, FOA4, Stockholm 80, Sweden
143. S. H. Levine, Northrop Space Laboratories, Hawthorne, California
144. Richard Madey, Republic Aviation Corporation, Farmingdale, L. I., New York

145. Brian Mar, Boeing Aircraft Company, Seattle, Washington
146. R. J. Macklin, Jr., Jet Propulsion Laboratory, Pasadena, California
147. R. V. Meghreblian, Jet Propulsion Laboratory, Pasadena, California
148. R. A. Miller, General Dynamics/Fort Worth, Fort Worth, Texas
149. Phil Mittleman, United Nuclear Corporation, White Plains, New York
150. Keith More, Bendix Systems Division, Ann Arbor, Michigan
151. Winnie M. Morgan, National Aeronautics and Space Administration, Washington, D.C.
152. J. L. Modisette, Manned Spacecraft Center, Houston, Texas
153. ASD(ASRSSV-e), Shielding Research Reports, ASR55-54 (T. J. McGuire), Wright-Patterson AFB, Ohio
154. L. W. McCleary, North American Aviation, Downey, California
155. J. C. Noyes, Boeing Scientific Research Laboratories, Seattle, Washington
156. J. P. Neissel, GE Company, Schenectady, New York
157. Wade Patterson, University of California, Radiation Laboratory, Berkeley, California
158. Maynard Pearson, Boeing Aircraft Company, Seattle, Washington
159. G. F. Pieper, NASA Headquarters, Greenbelt, Md.
160. Robert Pruett, Aerospace Corporation, El Segundo, California
161. Loren Pittman, 6570 AMRL (MRBBR), Wright-Patterson AFB, Ohio
162. Col. J. E. Pickering, USAF Aerospace Medical Center, Brooks AFB, Texas
163. Frederick Raymes, North American Aviation, Inc., Downey, California
164. Arthur Reetz, NASA Headquarters, Washington, D.C.
165. Evalyn Repplinger, Crew Systems Division, Houston, Texas
- 166-170. O. Reynolds, Director, National Aeronautics and Space Administration, Washington, D.C.
171. D. H. Robey, General Dynamics/Astronautics, San Diego, California
172. Sidney Russak, The Martin Company, Baltimore, Maryland
173. T. J. Rock, General Dynamics/Fort Worth, Fort Worth, Texas
174. Hermann Schaefer, U.S. Naval School of Aviation Medicine, Pensacola, Florida
175. F. E. Schwamb, Applied Physics Research, Republic Aviation Corporation, Farmingdale, L. I., New York
176. H. J. Schulte, Bellcomm, Inc., Washington, D.C.
177. W. M. Schofield, Advanced Research Corporation, Atlanta, Georgia
178. S. P. Shen, New York University, New York, New York
179. E. C. Smith, Advanced Research Corporation, Atlanta, Georgia
180. G. D. Smith, Ames Research Center, Moffett Field, California
181. Jerry Speakman, 6570 AMRL (MRBBR), Wright-Patterson AFB, Ohio
182. D. F. Spencer, Jet Propulsion Laboratory, Pasadena, California
183. R. T. Siegel, College of William and Mary, Williamsburg, Virginia
184. R. H. Steelle, NASA, Manned Spacecraft Center, Houston, Texas
185. H. E. Stern (M-RP-N), NASA, George C. Marshall Space Flight Center, Huntsville, Alabama
186. T. R. Strayhorn, General Dynamics/Fort Worth, Fort Worth, Texas

187. William Steigelmann, Kuljian Corporation, Philadelphia
Pennsylvania
188. Cornelius Tobias, University of California, Radiation Laboratory,
Berkeley, California
189. M. A. Van Dilla, Los Alamos Scientific Laboratory, Los Alamos,
New Mexico
190. L. F. Vosteen, NASA, Langley Research Center, Hampton, Virginia
191. Frank Voorhis, USAF (MC), OART/RBH, NASA, Washington, D.C.
192. F. Voris, National Aeronautics and Space Administration,
Washington, D.C.
193. G. P. Wachtell, Franklin Institute, Philadelphia, Pennsylvania
194. G. A. Whan, University of New Mexico, Albuquerque, New Mexico
195. G. T. Western, General Dynamics/Fort Worth, Fort Worth, Texas
196. Maurice Wilkinson, Boeing Aircraft Company, Seattle, Washington
197. Clayton Zerby, Union Carbide Research Institute, Tarrytown,
New York
198. K. Ziock, University of Virginia, Charlottesville, Virginia
199. V. B. Bhanot, Physics Department, Panjab University, Chandigarh-
3, India
200. Research and Development Division, AEC, ORO
201. A. M. Sarius, Centro di Calcolo, Vera Mazzimi 2, Bologna, Italy
202. Pierre LaFore, Commissariat a L'Energie Atomique, Centre D'Etudes
Nucleaires, de Fontenay-Aux-Roses (Seine), Boite Postale No. 6,
France
203. Gen. C. Roadman, Director, National Aeronautics and Space Admin-
istration, Washington, D.C.
- 204-843. Given distribution as shown in TID-4500 (31st ed.) under Physics
category (75 copies - OTS)