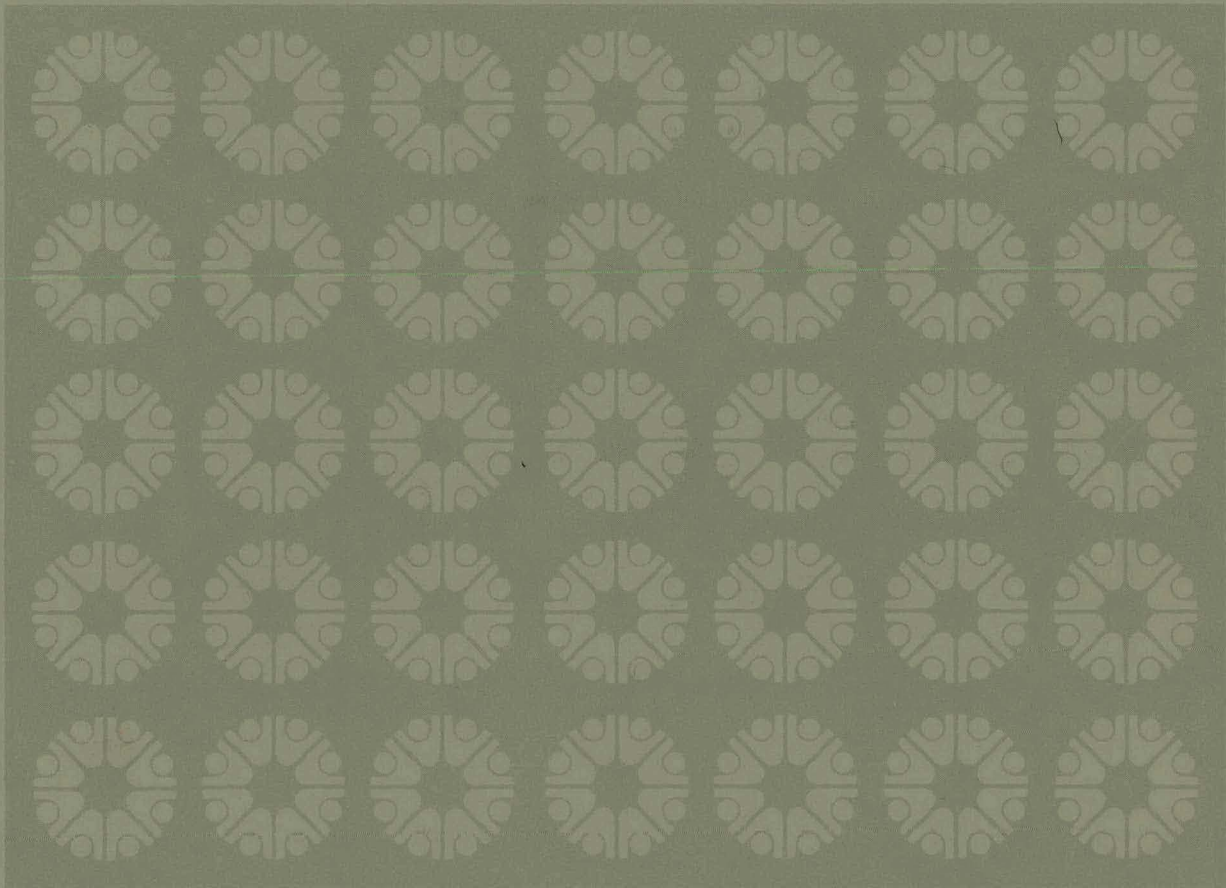2 cup

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Battelle

Pacific Northwest Laboratories

Battelle Boulevard / Richland, Washington 99352

RACER - A COMPUTER PROGRAM

FOR CALCULATING POTENTIAL EXTERNAL DOSE

FROM AIRBORNE FISSION PRODUCTS

FOLLOWING POSTULATED REACTOR ACCIDENTS

JUNE 1971

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D. L. Strenge
M. M. Hendrickson
E. C. Watson
Occupational and Environmental Safety Department

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In the spring of 1967 the USAEC Division of Reactor Development and
Technology (DRDT) invited several laboratories and companies to participate
in a survey designed to compare methods of calculating the potential
environmental radiological consequences of reactor accidents. The parti-
cipants were asked to calculate the time-integrated air concentration and
the thyroid and the whole body doses from an atmospheric release of radio-
nuclides. These doses were to be calculated at specified distances from
a specified fractional release of a reactor fission product inventory.
Other specified parameters were: the thermal power and operating time of
the reactor, the rate of release and the height of release. The same
atmospheric dispersion parameters were to be used by the participants.

A comparison of the calculation results showed reasonably good agreement for
the thyroid dose estimates but very poor agreement for estimates of the
external whole body dose.[1] As a result of this study, DRDT requested the
Pacific Northwest Laboratory, Battelle, to study existing whole body dose
calculational methods from atmospheric releases of reactor generated radio-
nuclides. As a result of this study, a computer program was developed for
calculating the external dose to people exposed to a passing cloud of radio-
nuclides released to the atmosphere following a postulated reactor accident.
This program, called RACER, is described in this report.

I. PROGRAM OBJECTIVE
    The objective of the program is to develop generally accepted methods
    which realistically estimate the whole body dose to man from airborne
    radionuclides which may be accidentally released to the atmosphere
    following a postulated nuclear reactor accident. Several tasks were
    established to accomplish this overall objective:

1. To develop mathematical methods for calculating the tissue dose
    from airborne radionuclides accidentally released to the atmosphere.

2. To program developed mathematical methods for solution by electronic
    data processing methods.

3. To assess the effect on dose estimates of uncertainties inherent in
    existing values of fission yields and decay energies of radionuclides
    anticipated to be released.

4. To identify those radionuclides which contribute to 90% of the
    whole body dose from an atmospheric release as a function of:
    irradiation history, time of release following shutdown, and
    fractionation of radionuclide inventory experienced during release.

Tasks 3 and 4 were abandoned because of funding problems.

Suggestions and comments were solicited from those scientists and
engineers knowledgeable in assessing the environmental consequences
of reactor accidents to help the investigators determine that the
recommended methods and the computerized codes are useful to the
nuclear industry. Comments and suggestions received are summarized
in Appendix H.

II. SUMMARY

Mathematical methods have been developed or adopted and corresponding
computer programs completed. These programs permit rapid and consis-
tent estimates of whole body dose resulting from exposure to a plume
of airborne radionuclides accidentally released from a nuclear reactor.
The method and computer program developed by this study incorporates
the desirable features of existing methods and programs into a single
efficient program. The desirability of the incorporated features was
determined after analysis of the results of a survey of investigators
experienced in the field of environmental radiation dose calculations.

III. MATHEMATICAL MODELS AND METHODS - TASK 1 & 2

It was decided early in this study to attempt mathematical description of the accidental release in order to make the dose calculations as realistic as practicable. An attempt has been made to make the program flexible enough so that new models, as they are developed, can be incorporated without extensive reprograming. Appendix A describes the mathematical models used in RACER.

Various stages which have been previously identified in the accidental release of reactor produced radionuclides, include:

- Source inventory - choice of radionuclides, dacay schemes (photon energies), quantity (initial activity).

- Aerosol behavior within containment spaces.

- Leakage rates from containment barriers (multiple containment effects).

- Atmospheric dispersion.

- Cloud size effect on external dose estimates.

- Build-up factors.

- Whole body dose from cloud.

Each of the preceding stages is incorporated into the model and program to be described; however, there are several possible starting places for the calculation depending on the information one has available. For example, if detailed information on the time and space distribution of each airborne radionuclide is available, the dose calculation would be primarily a shielding problem and the starting point would be a description of that time-space radionuclide distribution. Another starting point might be with the release rate to the atmosphere, of radionuclides at the containment vessel. Atmospheric dispersion would then be a

part of the problem. With such a starting point, the rate of release
of each isotope as a function of time would be necessary in order to
specify the source term. Earlier starting points would include descrip-
tion of the containment system action, aerosol behavior, and behavior
of fission products in the reactor core. The computer program, RACER,
has been designed so that several starting points are possible. It also
permits the user to incorporate different amounts of detail in account-
ing for the various release, transport and dispersion mechanisms between
the selected starting point and the dose calculation result. A brief
description of the program design is given in Appendix B and the program
flow diagram is presented in Appendix C.

Source Information

The simplest way of using the code is to specify an initial fission
product inventory in curies. This requires a minimum amount of input.
The computer code will then follow the chain decay of the radionuclides
specified and will determine the rate of release to the environs accord-
ing to the leak rate (or rates if multiple containers are involved)
specified. The specified inventory of radionuclides can be generated by
any code or taken from tables. To be used by this code, the inventory so
specified must be entered on punched cards in a simple prescribed format.
Input preparation is described in Appendix F. Parent nuclide decay and
buildup of daughter radionuclides which occur with time will be calcu-
lated utilizing internal library data on half lives and decay schemes.

An alternative to specifying the initial source inventory, is to use the
incorporated program of generating an inventory. This program is an
adaptation of the computer code RIBD.[2] This program provides a simple
method for specifying the source inventory as a function of irradiation
history. It accounts for only two fission sources and uses only a single
nuclear interaction cross section. Thus, its ability to provide accurate
inventory data is limited for reactors that have appreciable epithermal
neutron interactions, i.e. fission or capture. Its limitations become
of less concern for environmental radiation dose calculations when com-
pared with the uncertainties inherent in the other stages of the
calculation.

Source Inventory

It is required to supply an initial inventory, either by calculation or
by specification. Any number of radionuclides can be specified for the
inventory calculation. It may be possible to determine in advance which
nuclides are significant in the dose calculation for a given set of
conditions, however, in using this procedure the user must be careful
not to inadvertently overlook a nuclide that is relatively important at
some other internal of the dose calculation. For example, if one assumes
fractionation so that only the noble gases are released for the first
few hours after a reactor accident, ^{87}Kr and ^{88}Kr are the most important
radionuclides.   After a day or two, ^{133}Xe and ^{135}Xe become the most
significant and eventually after many days, ^{85}Kr becomes significant.
Flexibility is maintained in solving problems which assume different
fission product fractionation and decay times, by a library of informa-
tion in the computer code which contains data on 450 fission products
which may be of interest in reactor accident problems. This library is
open ended and the number of nuclides to be used in the calculation may
be reduced by removing the unwanted nuclides from the library. The data
library is described in Appendix D. With this variable library, it is
possible to conduct parametric studies, and determine which nuclides
are of importance for various types of problems.

Decay Scheme

There is no single source which contains all the suitable information for
the nuclides of interest in reactor accident dose calculations. Most of
the basic information on decay schemes is available from tables such
as those prepared by Lederer, Hollander, and Perlman[3] or Landolt and
Bornstein.[4] The use of these data in radiological dose calculations
involves intermediate calculations for items such as obtaining the effec-
tive gamma or beta energy per disintegration. Useful but incomplete
compilations of data in a form readily useable in radiation dose calcu-
lations have been prepared by Meek and Gilbert [5] and Meek and Rider.[6]

Aerosol Behavior

It is not desirable to settle on any one aerosol behavior model at this
time. So instead of including any aerosol behavior model a generalized
method of interfacing with aerosol behavior codes has been supplied.
This method permits use of either theoretically derived or experimentally
generated data. This allows the code to account for aerosol behavior
without being permanently tied to a specific model.

Terms describing aerosol behavior, filtration, and resuspension (for
noble gases produced from settled material) were considered necessary
for each barrier. At the present time, few models take credit for
aerosol behavior effects in containment; however, data now being gener-
ated indicate these effects can be significant. This is especially true
for the fast reactor where the postulated accident and subsequent
release involve large quantities of sodium. There is no general agree-
ment, however, on aerosol models for specific application to boiling
water reactors, pressurized water reactors, reactors with ice condensor
systems, or sodium cooled fast reactors.

The aerosol behavior section of the code was developed for analysis of
the FFTF site, specifically for analyzing FFTF accidents where hypothe-
tically there could be dense Na-Pu aerosols present. It became apparent
during this analysis that it was impractical to select any one aerosol
model because of the rapid progress being made in the study of the
behavior of Na-Pu aerosols. Therefore, the code programmed in RACER,
was designed to handle aerosol behavior effects in a generalized say.
The leak rates between barriers and to the atmosphere are also programmed
in a generalized way. Leak rates and aerosol removal rates are entered
as a function of time on data cards. The data is interpolated by the
code to determine the rates at the times needed.

Leakage Paths

One approach to modeling reactor fission product release calculations is
to assume a release from a single containment vessel. The single leakage
path approach is that used in TID 14844. It is assumed that the fission

products available for release are leaked at a constant rate to the atmosphere. This approach is not realistic for modern day reactor designs with their multiple containment barriers. The effect of multiple containment barriers in reducing the source term should not be ignored.

The model programmed in RACER considers the containment barriers to be a system of stirred vessels. Theoretically any number of containment barriers could be assumed but this code has been limited to three. The analytic solution to the containment problem beyond 3 barriers becomes extremely laborious to derive and takes considerable computer space to program. Thus the limit of three barriers avoids excessive demands on computer size. The arrangement of the containment system for a particular problem is determined by the leak rates supplied as input. Any release rates not specified are automatically set to zero and related calculations ignored. This specification procedure was selected for its simplicity.

Atmospheric Dispersion

The atmospheric dispersion of the released nuclides is described by a bivariate normal distribution model. This model was selected because of its widespread acceptance and not necessarily because of experimental evidence that it adequately describes plumes of released fission products.

The bivariate normal model requires as input the standard deviation of cloud concentration in the crosswind lateral and vertical directions. Several methods for estimating standard deviations are in common use and three of these have been programmed, i.e., Suttons parameters, the Hanford equations and Pasquill's curves. In addition, values of the standard deviations arbitrarily selected can be entered on punched cards as input.

Cloud depletion calculations due to ground deposition have not been included because of the large amount of computer space required and the long running time necessary for a comprehensive calculation. However,

a depletion factor as a function of distance may be included through a
punched card input.

Cloud Size Effect

The dose from gamma radiation to the whole body from airborne radio-
nuclides is a function of several variables, one of which is cloud
concentration. Initially, i.e., at close in distances, the cloud
plume size is small so that its concentration changes rapidly with
crosswind distance from its centerline. The cloud size increases at
large transport distances. A consequence of this growth is that
variation of concentration with distance from centerline is such that
for all practical purposes the cloud may be considered infinite in
size with a concentration equal to that at its centerline. At these
large distances, the cloud is described mathematically as a hemisphere
of infinite radius and dose calculations at the center of the hemisphere
are referred to as the "semi-infinite" cloud dose. Several methods have
been used to describe the cloud at close in distances where the cloud
cannot be considered an infinitely large hemisphere. A major reason
for the differences in the test problem results reported by Gammill and
VanderHoven[1] can be traced to methods of calculating the dose from
finite size clouds.

The model programmed as a result of this study utilizes a numerical
calculation over the finite size cloud. Because this method is time
consuming, hence more expensive, the "Semi-infinite" cloud calculation
has been incorporated also. Since it is less costly, the semi-infinite
cloud dose calculational method can be used at large distances and for
quick computations of upper limit dose estimates.

Dose Buildup Factors

A major contribution to the exposure dose from airborne radionuclides
emitting low energy gamma radiation may come from radiation reflected to
the point of interest after interacting with surrounding nuclei. The
ratio of the dose with such reflection to that without reflection is

called a dose buildup factor. These buildup factors, with air as the surrounding medium, are strongly dependent on energy of the emitted gamma rays.

The data selected for calculating dose buildup factors in this study are those calculated by Berger.[7] For analytic solution the model consists of fitting the buildup factors to a quadratic equation for each of twelve energy intervals. The optimum number and distribution of these energy intervals have not yet been determined.

Whole Body Tissue Dose From Cloud

The end result of the calculation is the tissue dose, in rads at a downwind location, delivered over a specified time, from gamma radiation emitted by a cloud of airborne radionuclides, the distribution of which is described by a bivariate normal equation. The location can be any point downwind of the release point on or off the centerline of the cloud at ground level. No restrictions have been placed on release height. The penetrating dose, i.e., dose at five centimeters, and the dose to organs of interest resulting from inhalation, have not been incorporated in the computer code.

One of the major problems encountered with the dose calculation has been in achieving sufficient accuracy in the integration over the cloud volume. Several integration techniques have been tried including (1) standard Gauss quadrature, (2) Gauss-Hermite quadrature and (3) a standard polynomial fit. The standard polynomial fit was selected because it performed best for several test cases. Another advantage of the polynomial fit is that the points for evaluation of the function are evenly spaced and the end points of the integration are variable. This allows the integrals to be broken into several smaller segments of varying length to give an efficient calculation, i.e. small intervals where the function changes rapidly and larger intervals where the function varies more smoothly.

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APPENDIX A

DESCRIPTION OF MATHEMATICAL MODELS

The mathematical description of a whole body tissue dose for exposure to an atmospheric release of radionuclides following a postulated reactor accident is complicated by: the number of fission products produced, the release path phenomena involved in leakage of these fission products from the fuel, through the containment system to the atmosphere, the unpredictable behavior of the atmosphere and the time dependence of the fission product concentrations. In this study, the problem has been reduced to three interrelated models: one for the fission product inventory calculation, one for the release path calculation and the third for the dose calculation. For convenience the expressions for dose will be discussed first.

Whole Body Tissue Dose

The basic equation for the external whole body tissue dose rate from a gamma emitting radionuclide present in an incremental volume of a cloud is:

$$d_\gamma = \frac{\chi B(\mu r) e^{-\mu r} K}{4\pi r^2} \, dxdydz$$

where:

d_γ · incremental dose rate to the whole body, rad/sec at a point and time after shutdown, from gamma radiation emitted from an incremental cloud volume, dx, dy, dz at a distance r meters from the point of interest.

χ · cloud concentration, Ci/m^3

$B(\mu r)$ · dose buildup factor for air

μ · total linear attenuation coefficient in air, m^{-1} for gamma energy E.

K · dose conversion factor, $(rad \cdot m^2)/(Ci \cdot sec)$ for a radionuclide with an effective energy $\hat{E}_\gamma$, Mev/disintegration.

The constant K is:

$$K = \frac{3.70 \times 10^{10} \frac{dis}{Ci \cdot sec} \quad 1.60 \times 10^{-6} \frac{erg}{Mev} \quad 10^{-4} \frac{m^2}{cm^2} \hat{E}_\gamma \left(\frac{\mu_a}{\rho}\right)}{100 \frac{ergs}{g \cdot rad}}$$

$$= 0.0592 \, \hat{E}_\gamma \left(\frac{\mu_a}{\rho}\right)$$

where:

$\left(\frac{\mu_a}{\rho}\right)$ · mass absorption coefficient, cm^2/g in tissue for gamma energy considered.

Dose rate is a function of γ-ray energy, therefore the gamma spectrum has been divided into energy groups. The incremental dose is then calculated separately for each energy group of each radionuclide, integrated over the cloud volume and then summed to obtain dose rate from the airborne radionuclides. The number and distribution of energy groups has not been optimized. Since the number of groups largely determines the computer running time for the cloud dose calculation, it would be desirable to use no more groups than is necessary to achieve the desired accuracy. Tests to determine the number to be used will have to be performed eventually.

To obtain the cloud concentration it is assumed that:

• Diffusion along the direction of cloud travel can be ignored.

• Lateral and vertical crosswind concentrations are normally distributed and that the standard deviation is a function of atmospheric stability and travel time.

• The dose receptor is at ground level downwind of the release point.

• Cloud depletion by fallout, washout and rainout can be described by a factor dependent on the distance of travel and assumed to be independent of travel time and displacement from the centerline.

- The quantity of radionuclides in the cloud at distances beyond 3
  standard deviations in the vertical and lateral directions is insignifi-
  cant and therefore makes little contribution to the dose rate.
  Radionuclides at distances greater than ± 400 meters in the direction
  of cloud travel are also ignored.

The cloud concentration is given by:

$$\bar{\chi} = \frac{Q' \, F_d}{\pi \, \sigma_y \, \sigma_z \, \bar{u}_h} \ \exp-[\frac{y^2}{2\sigma_y{}^2} + \frac{h^2}{2\sigma_z{}^2}]$$

where $\bar{u}_h$ = average wind speed at the height of release in direction of
travel, m/sec

σ_y = crosswind lateral standard deviation of cloud concentration, m

σ_z = crosswind vertical standard deviation of cloud concentration, m

y = lateral displacement from cloud centerline, m

z = vertical displacement from cloud centerline, m

h = height of release, m

F_d = radioactive decay factor

Q' = rate of release from source, curies/sec

The crosswind standard deviations, σ_y and σ_z, may be calculated by the
Hanford model, the Sutton model or taken from curves attributed to Pasquill.
Their values depend on atmospheric stability in all three methods.

The equations for the Hanford model are:

$$\sigma_y{}^2 = A[T - \alpha \, (1-e^{-T/\alpha})]$$
$$\sigma_z{}^2 = a(1-e^{-k^2T^2}) + bT$$

where:

$$A = [c + d \, (\sigma_\theta \bar{u})]$$

$$\alpha = \frac{A}{2(\alpha_\theta \bar{u})^2}$$

and a, b, c, d, k and $\alpha_\theta \bar{u}$ are parameters for describing the atmospheric condition and T is the time of travel to point of interest, sec.

The Sutton model defines the variances as:

$$\sigma_y = \frac{C_y \ x^{(2-n)/2}}{\sqrt{2}}$$

$$\sigma_z = \frac{C_z \ x^{(2-n)/2}}{\sqrt{2}}$$

where C_y, C_z and n are parameters describing the meteorological condition and x is downwind distance from the release point to the cloud position.

Values of the atmospheric parameters in use at Hanford are listed in Tables A-I and A-II.

The decay factor, F_d for each nuclide at a distance x from the source is calculated from the release rate at the source, Q', and the travel time. An eleaborate decay scheme is used to account for chain decay including transistions to and from isomeric states. The general equation for such a scheme is:

$$M_i^{'} = \frac{d \ M_i}{dt} = \sum_j \lambda_j K_j^i \ M_j - \lambda_i M_i$$

where: $M_i^{'}$ • rate of release for nuclide i after travel time T, atoms/sec.

$M_{oi}^{'}$• initial rate of release for nuclide i, atom/sec

λ_i • decay rate constant for nuclide i (sec^{-1})

K_j • branching ratio nuclide j decaying to nuclide i.

$$F_d = \left(\frac{M}{M_o}\right)_i$$

The summation is over all nuclides decaying directly to nuclide i.

TABLE A-I

VALUES OF METEOROLOGICAL PARAMETERS
FOR THE HANFORD MODEL

Parameter	Moderately Stable Conditions	Very Stable Conditions
a m^2	97	34
b m^2/sec	0.33	0.025
c m^2/sec	13	13
d meters	230	230
k sec^{-2}	2.5×10^{-4}	8.8×10^{-4}

Commonly Used Values for $\sigma_\theta \bar{u}$

| Assumed Duration of Release, min | Average Wind Speed | | | |
	1 m/sec	2.5 m/sec	5 m/sec	10 m/sec
10	0.024	0.10	0.20	0.30
60	0.04	0.15	0.25	0.35
120	0.08	0.25	0.35	0.45
240	0.10	0.40	0.50	0.60
480	0.18	0.60	0.70	0.90

TABLE A-II

VALUES OF METEOROLOGICAL PARAMETERS FOR
SUTTON'S MODEL

Parameter	Release Level	Wind speed, m/sec	Atmospheric Condition	
			Unstable	Neutral
C_y	Ground	1	0.35	0.21
		5	0.30	0.15
		10	0.28	0.14
	Elevated	1	0.30	0.15
		5	0.26	0.12
		10	0.24	0.11
C_z	Ground	1	0.35	0.17
		5	0.30	0.14
		10	0.28	0.13
	Elevated	1	0.30	0.15
		5	0.26	0.12
		10	0.24	0.11
n	-	-	0.20	0.25

Containment Model

The release rate at the source, Q', is determined by a "multiple containment" model. The radioactive inventory at the onset of the release is calculated on the basis of irradiated time power level and time of initial release following shutdown.

The nuclides are then allowed to leak through the containment barriers with filtration, settling, plateout and rainout phenomena accounted for in each barrier as desired. The present model has three containment barriers with removal phenomena accounted for in each barrier. Each barrier also has a filter. A diagram of the system is given in Figure A-1 showing the barriers and leakage paths available. Several assumptions are necessary to obtain an analytic solution:

- There is instantaneous mixing in each vessel.

- Fallout rates and leak rates may be considered constant for short time intervals.

- Nuclides becoming airborne from material which has settled or plated out come only from the mass present at the beginning of the interval.

- Nuclides released from the filters come only from material present on the filters at the beginning of the interval.

These assumptions make the system of differential equations for nuclide concentrations in each barrier analytically solvable and of the form

$$M_i = \sum_j A_j e^{-B_j t}$$

where the summation is over all previous nuclides in the mass chain and all previous barriers (including amounts present as fallout and on filters). The A_j and B_j are functions of initial nuclide concentration Q_o, fractional leak rates, removal rates, filter efficiencies, nuclide half-lives and fractional decay yields. The release rate Q_i' at time t for nuclide i is readily calculated as:
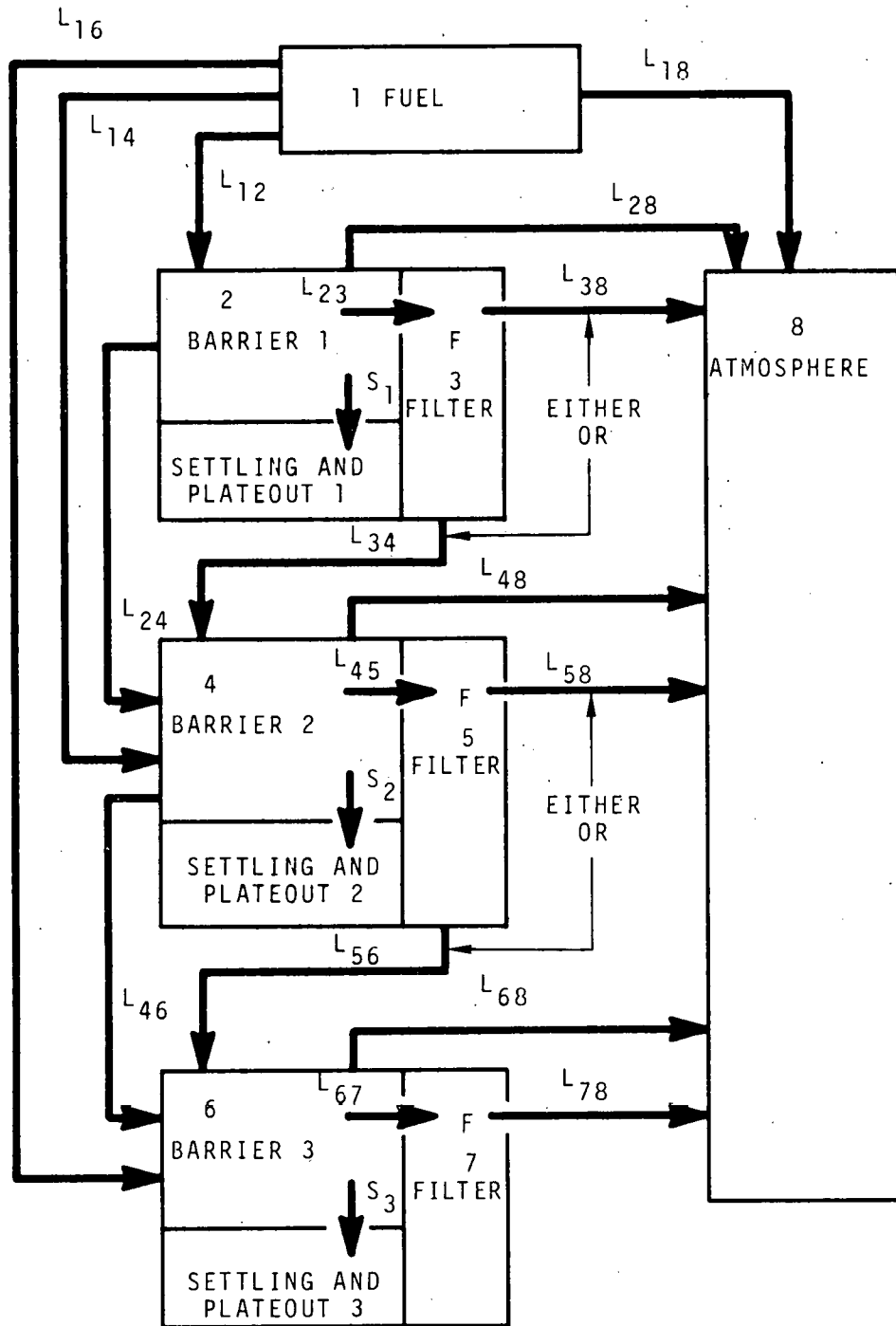
L_{ij} = LEAK RATE FROM COMPARTMENT i TO COMPARTMENT j , SEC^{-1}

S_i = REMOVAL RATE FOR BARRIER i , SEC^{-1}

F = FILTER FACTOR

FIGURE A-1. Multiple Containment Model

$$Q_i' = \sum M_{ij} L_j \lambda_i \; F_i, \; \text{curies/sec}$$

where M_{ij} = mass of nuclide i in barrier j at time t, curie seconds

L_j = leak rate from barrier J to the atmosphere at time t, sec^{-1}

λ_i = decay rate constant for nuclide i, sec^{-1}

F_i = filter factor for nuclide i

Nuclide Inventory Model

The containment model uses the initial nuclide inventory as a starting
point.   The accurate calculation of radionuclide inventories is complex
and depends on many variables. A more sophisticated calculation would use
a multigroup cross section model and a detailed description of core components,
fuel and geometry, however; the inventory of radiologically significant
radionuclides of most reactors can be described sufficiently for dose
estimate purposes by a simple one group cross section calculation. The
present model uses such a calculation with a multiple chain grid processor.
The processor calculates nuclide concentrations resulting from two fission
sources (for example, ^{235}U and ^{239}Pu) with normal down-chain decay by beta
emission and isomeric transfers and interchain coupling resulting from n-gamma
reactions. The general differential equation for a nuclide is

(Rate of change of nuclide i) = (fission rate source 1)(yield from source 1 fission)

+ (fission rate source 2)(yield from source 2 fission)

+ $\sum$ (neutron capture rate per atom of "parent")(atoms of "parents")

+ $\sum$ (decay rate of "parent")(atoms of "parent")

- (decay rate + neutron capture rate of nuclide)(atoms nuclide present)

Parameters to be specified include reactor operating power, operating time,
flux, energy per fission, physical half-lives, direct yields from source 1

fissions, direct yield from source 2 fissions, neutron capture cross sections, average γ decay energies per disintegration, average β decay energies per disintegration and branching ratios to account for various modes of transition to and from isomeric states.

Operation at several successive power levels may be calculated by using the final nuclide inventory at one level as the initial inventory for the next. Shutdown periods are accounted for by chain-decay calculations only (no fission or neutron capture).

APPENDIX B

COMPUTER CODE

The models described in Appendix A have been incorporated into a computer
code programmed for the Univac 1108 computer. The calculations are divided
into three main subroutines:

1. SOURCE - fission product inventory calculation or specification
2. BAFFLE - containment release calculation
3. EXDOSE - whole body tissue dose calculation

The subroutine SOURCE specifies the initial inventory in curies for each
nuclide; the subroutine BAFFLE calculates the rate of release of each
nuclide as a function of time; the subroutine EXDOSE calculates the
absorbed dose in tissue in rads. The code flow diagram is described in
Appendix D.

SOURCE

The subroutine SOURCE includes a modified version of the computer code
RIBD for the fission product inventory. RIBD performs the general calcula-
tion as described in Appendix A. Nuclide parameters are specified in a
library separate from the RIBD subroutine. This allows any data library
to be used (with the correct format) so that any other fuels with one or
two fissionable nuclides may be modeled such as ^{233}U - ^{239}Pu. The fission
decay yields of the library must correspond to the fissionable elements of
the fuel being studied.

A fission product inventory calculation does not have to be made; fission
product and activation product inventories (in curies) may be input on data
cards. The direct input option has been included to allow results of other
fission product inventory codes to be used.

The output from SOURCE may be put on tape (and saved for future use) or on
magnetic drums. The output is the activity of each radioactive fission
product and activation product in the nuclide library at the start of the
release calculation.

The fission product inventory code will calculate the fission product
inventory for any decay time desired. Any decay time may be used for the
initial inventory for the containment calculation. Inventories for as
many as ten decay times can be calculated during each run.

BAFFLE

The subroutine BAFFLE performs a multiple containment calculation as described
above. A few notes on the calculation follow.

Data for the multiple containment model are supplied to the computer by two
methods: as direct input and as stored library data. Data contained in
libraries include radionuclide physical half lives, branching ratios and
filter factors. (Presently two filter factors are stored for each nuclide:
one for release from fuel and the other for release through filters.) The
filter factors for a group of nuclides may be supplied as input. The nuclide
groupings are:

1. Noble gases: Ar, Kr, Xe
2. Halogens: Cl, Br, I
3. Volatile Solids; Se, Te, Cs
4. All Others

Filter factors for individual nuclides may be specified by updating the
filter library. Filter factors are specified as the fraction escaping the
filter. For example, a filter with an efficiency of 99.9% would have a
filter factor of 0.001.

The filter factor applied to the nuclide in the fuel was designed to have a
value of 0 to 1 to indicate whether or not the nuclide was released from the
fuel (1 for release). Three filters are included in the model (one for each
barrier) in addition to a pseudo filter representing release from the fuel.

Presently, the filters all use the same factors. A possible change would
be to have separate filter factors for each filter.

The libraries currently include data for 450 fission products plus 144
activiation products (including several transuranium elements).

Data to be supplied as input include length of release, time increments for
release, leak rates, settling rates and filter factors (optional). The leak
rates and settling rates are entered with corresponding time points to cover
the length of release.

The leak rates and settling rates are assumed to be constant over the time
increment specified for the calculation. The rates used are evaluated from
the input data for the time at the middle of the current time increment. A
linear interpolation of the input data is used.

The output from BAFFLE may be put on magnetic tape (for future use) or on
magnetic drum. The output is in terms of curie·sec/sec for each nuclide
at the specified time intervals.

EXDOSE

The subroutine EXDOSE calculates the whole body tissue dose from the radio-
nuclides released to the atmosphere (as calculated in BAFFLE). The dose is
calculated as the space and time integral of the cloud nuclide concentrations.

The actual integration is performed in the following order: y direction
(lateral), z direction (vertical), time and x direction. The first two
integrations are performed for each position in the x direction (as located
to give sufficient accuracy). The resulting double integral represents the
dose rate per unit length of downwind cloud per curie of a nuclide for
specified energy groups. The integration is performed for each energy group
at each x position. The release rate data is then used to calculate the

nuclide inventory at each position as a function of time. The inventories
are coupled with the dose rate per length factors to give the actual dose
rate per length as a function of time (summed over all energy groups). The
process is repeated for each 8 time points until the exposure time or
release time is reached. Then the dose per length values are integrated
over the x direction to give the total dose.

Cloud depletion factors may be applied to the dose rate per length values.
The depletion factors are input as a function of distance. The data input
is interpolated linearly to determine the depletion factor for each x
position.

The depletion factor represents the fraction of the cloud remaining at the
downwind position.

· APPENDIX C

CODE FLOW DIAGRAM

A general outline of the code operation is included here. Figure C-1
gives the relation of the main program and subroutines used by the code.
Several small subroutines have been omitted from Figure C-1, However,
their function is included in the code flow diagram, Figures C-2 thru C-17

The containment calculation is controlled through subroutine BAFFLE. The
subroutines used by BAFFLE (FIRST, SECOND, ...LAST) perform the multiple
containment calculation for each member of the nuclide chains. Subroutine
FIRST does the calculation for the first member of the chain, subroutine
SECOND does it for the second member, and so on until all chain members
have been taken care of. The subroutines calculate the necessary coefficients,
exponents and nuclide release rates. A series of equations similar to
FIRST is included in BAFFLE to handle activation products (simple decay).

The single containment calculation uses an effective decay constant
calculated as

$$L_{eff} = \lambda_r + (\text{LEAK RATE})(0 \text{ or } 1)$$

This equation is indicated in Figure C-5 For nuclides that are not allowed
to leak the leak rate is multiplied by zero. For nuclides that do leak the
leak rate is multiplied by 1. The factor zero or one is entered as the fuel
release factor which may only have the values 0 or 1.

Several parameters mentioned in the EXDOSE portion of the code are described
below.

Parameter	Description
TYPE	an integer variable for selection of the meteorological conditions to be modeled: 1 for Sutton equations, 2 for Hanford equations, 3 for direct input of σ_y and σ_z data, and 4-9 for Pasquill curves A-F.

Parameter	Description
THRU	a logical variable for the finite cloud calculation to indicate that the calculation may be stopped (THRU=true).

BURST(N,IE)

Photon energy data for nuclide N and energy group IE. BURST is the sum of $\lambda_N A_{KI} \dfrac{E_{KI}}{E_{IE}}$ for each photon of average energy E_{IE} where λ_N is the decay constant for nuclide N, sec^{-1}; A_{KI} is the abundance of the K-th photon from nuclide N; and E_{KI} is the energy of the K-th photon from nuclide N. E_{KI} is in group IE.

YLIST(IY,IE)

YLIST is the dose kernal for position (x, y, z) where IY is the index for the Y direction grid and IE is the energy group index.

ZLIST(IZ,IE)

ZLIST is the Y line integral for Z grid position IZ and energy group IE. ZLIST is calculated as

$$ZLIST(IZ,IE) = \int_Y YLIST(IY,IE)dy$$

DRDX(IX,IE)

Dose rate per curie per cm of downwind cloud at X position IX and energy group IE. DRDX is calculated as

$$DRDX(IX,IE) = \int_X ZLIST(IZ,IE)\, dZ$$

MASS(I)

MASS(I) is the curie seconds of nuclide I present at the position and time of interest.

Parameter	Description

DRDX(1,IE)
The semi-infinite calculation uses only one dimension of the dose rate array. DRDX(1,IE) is the dose rate at the exposure point per curie of a nuclide with a photon in energy group IE.

SUM(I,IX)
Total dose rate at time point I and X position IX. SUM is calculated as

$$SUM(I,IX) = \sum_N \left[\sum_{IE} DRDX(IX,IE) \times BURST(N,IE) \right] \times MASS(N)$$

DOSEDX(IX)
Time integrated dose for X position IX. DOSEDX is calculated as

$$DOSEDX(IX) = \int_0^{T_{exposure}} SUM(I,IX)\, dt$$

DOSE
For the semi-infinite cloud calculation the accumulated dose is calculated at each time increment as DOSE. DOSE is calculated as

$$DOSE = \int_0^T \sum_N \sum_{IE} DRDX(1,IE) \times MASS(N) \times BURST(N,IE)\, dt$$
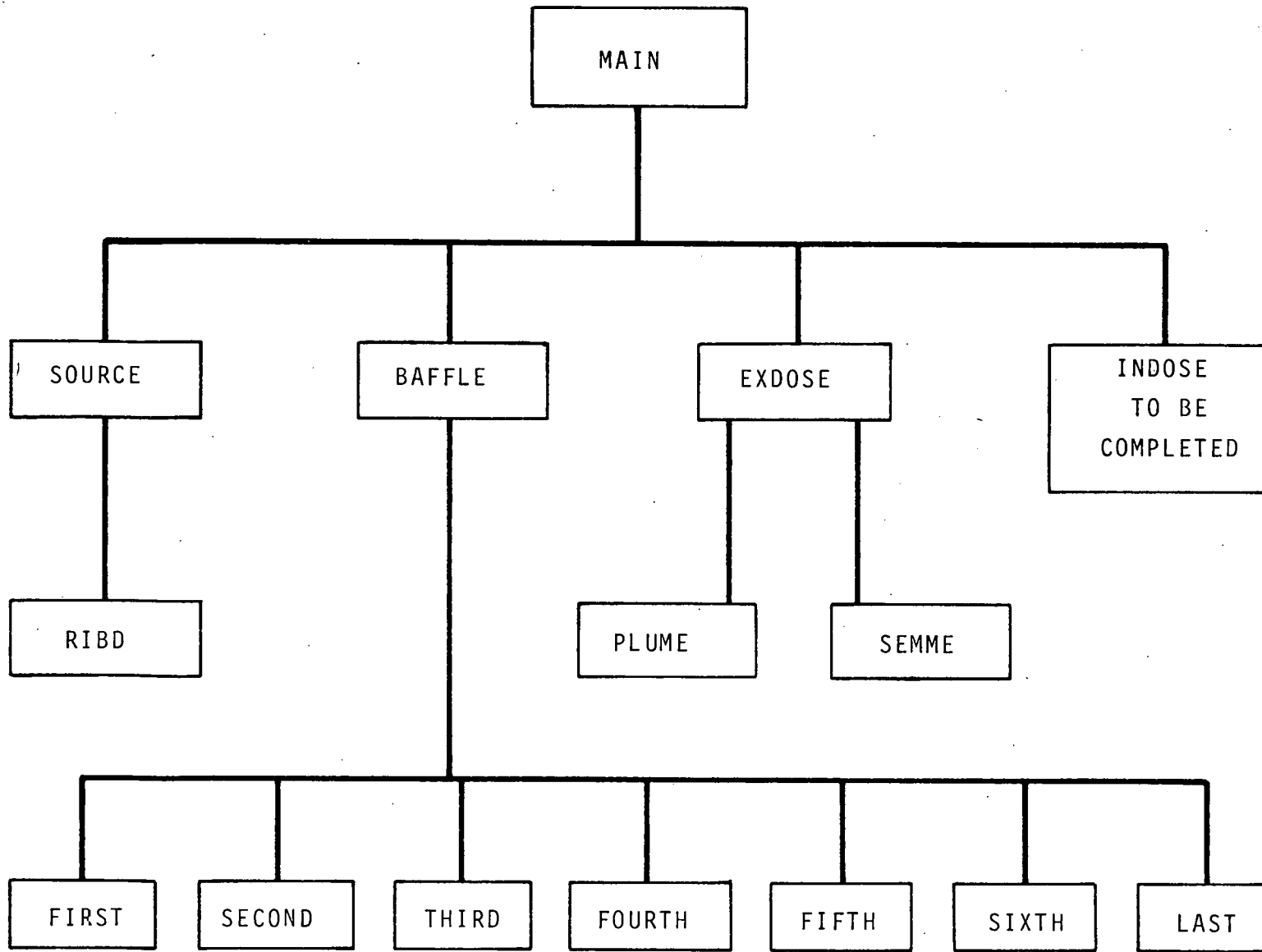
FIGURE C-1. Flow Diagram of Main Program and Subroutines

```
          ┌─────────────────────┐
          │    MAIN PROGRAM     │
          └─────────────────────┘
                    │
                    ▼
          ┌─────────────────────┐
          │ SET UP DRUM STORAGE │
          └─────────────────────┘
                    │            ┌─────────────────────┐
                    │◄───────────│   RETURN FROM        │
                    │            │   SUBROUTINES        │
                    ▼            └─────────────────────┘
                  ╱─────╲
                 ╱  END  ╲
                ╱ OF DATA ╲   YES        ╱─────╲
               ╲  FILE    ╱ ──────────► │ STOP  │
                ╲   ?    ╱               ╲─────╱
                 ╲─────╱
                    │ NO
                    ▼
        ╱──────────────────────────────╲
       ╱  READ AND PRINT TITLE CARD AND  ╲
      ╱   CONTROL CARD TO DETERMINE        ╲
      ╲   TYPE OF CALCULATION TO PERFORM   ╱
       ╲──────────────────────────────────╱
                    │
                    │        ┌──────────────────────────┐
                    ├───────►│  SOURCE; INVENTORY        │
                    │        └──────────────────────────┘
                    │
                    │        ┌──────────────────────────┐
                    ├───────►│  BAFFLE; CONTAINMENT      │
                    │        └──────────────────────────┘
                    │
                    ├───────►┌──────────────────────────┐
                    ┊        │  EXDOSE; CLOUD DOSE        │
                    ┊        └──────────────────────────┘
                    ┊
                    ┊        ┌──────────────────────────────────┐
                    └┈┈┈┈┈┈►│  INDOSE;  INHALATION DOSE,        │
                             │           INCOMPLETE               │
                             └──────────────────────────────────┘
```

FIGURE C-2. Code Flow Diagram

SUBROUTINE SOURCE

```
                    ┌──────────┐         ┌─────────────────────────────┐
                    │  READ    │   YES   │ READ LIBRARY OF NUCLIDE DATA │
                    │ LIBRARY  │────────▶│ FOR INVENTORY CALCULATION    │
                    │    ?     │         │ AND NUCLIDE IDENTIFICATION   │
                    └──────────┘         └─────────────────────────────┘
                       │ NO
                       ▼
              ┌──────────────────────┐
              │ INITIALIZE PARAMETERS │
              └──────────────────────┘
                       │
                       ▼
                  ┌──────────┐   NO    ┌─────────────────────────┐
                  │  INPUT   │────────▶│ INVENTORY CALCULATION   │
                  │INVENTORY │         │ SUBROUTINE RIBD         │
                  │    ?     │         └─────────────────────────┘
                  └──────────┘                      │
                       │ YES                        ▼
                       │              YES      ┌──────────┐
                       │◀──────────────────────│  INPUT   │
                       │                       │ACTIVATION│
                       ▼                       │PRODUCTS  │
              ┌──────────────────┐             │    ?     │
              │ READ NUCLIDE     │             └──────────┘
              │ INVENTORY        │                  │ NO
              └──────────────────┘                  │
                       │                            │
                       ▼                            │
                  ┌──────────┐   NO    ┌──────┐     │
                  │   CAN    │────────▶│ STOP │     │
                  │NUCLIDES BE│        └──────┘     │
                  │IDENTIFIED │                     │
                  └──────────┘                      │
                       │ YES                        │
                       │◀───────────────────────────┘
                       ▼
              ┌──────────────────────┐
              │ WRITE INITIAL INVENTORY │
              │ TO DRUM OR TAPE         │
              └──────────────────────┘
                       │
                       ▼
              ┌──────────────────────┐
              │ DO NECESSARY DRUM OR  │
              │ TAPE HANDLING OPERATIONS │
              └──────────────────────┘
                       │
                       ▼
               ( RETURN TO MAIN )
```

FIGURE C-3.   Code Flow Diagram

SUBROUTINE BAFFLE

FIGURE C-4. Code Flow Diagram

FIGURE C-5. Code Flow Diagram

FIGURE C-6. Code Flow Diagram
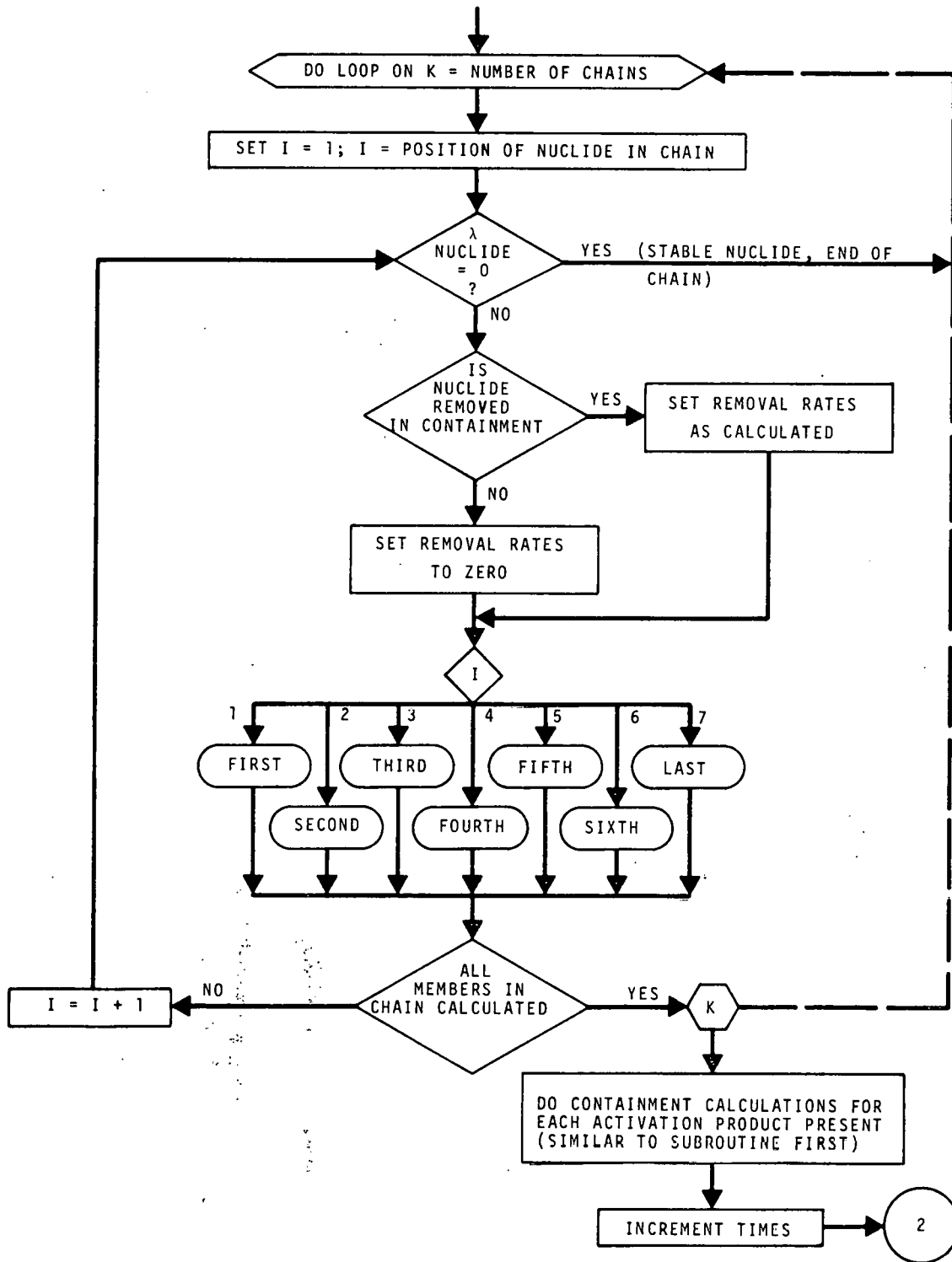
FIGURE C-7.   Code Flow Diagram

SUBROUTINE EXDOSE
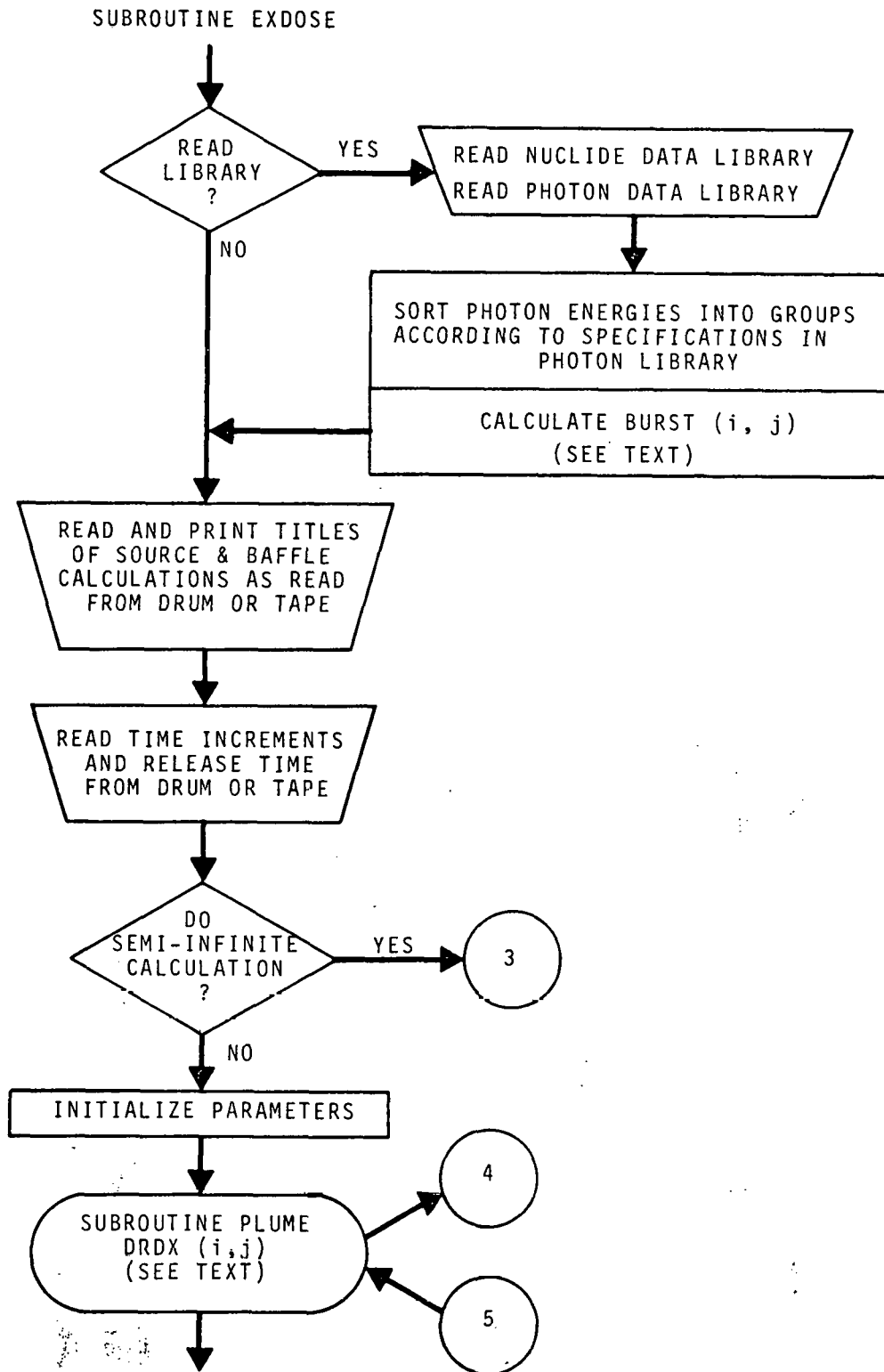
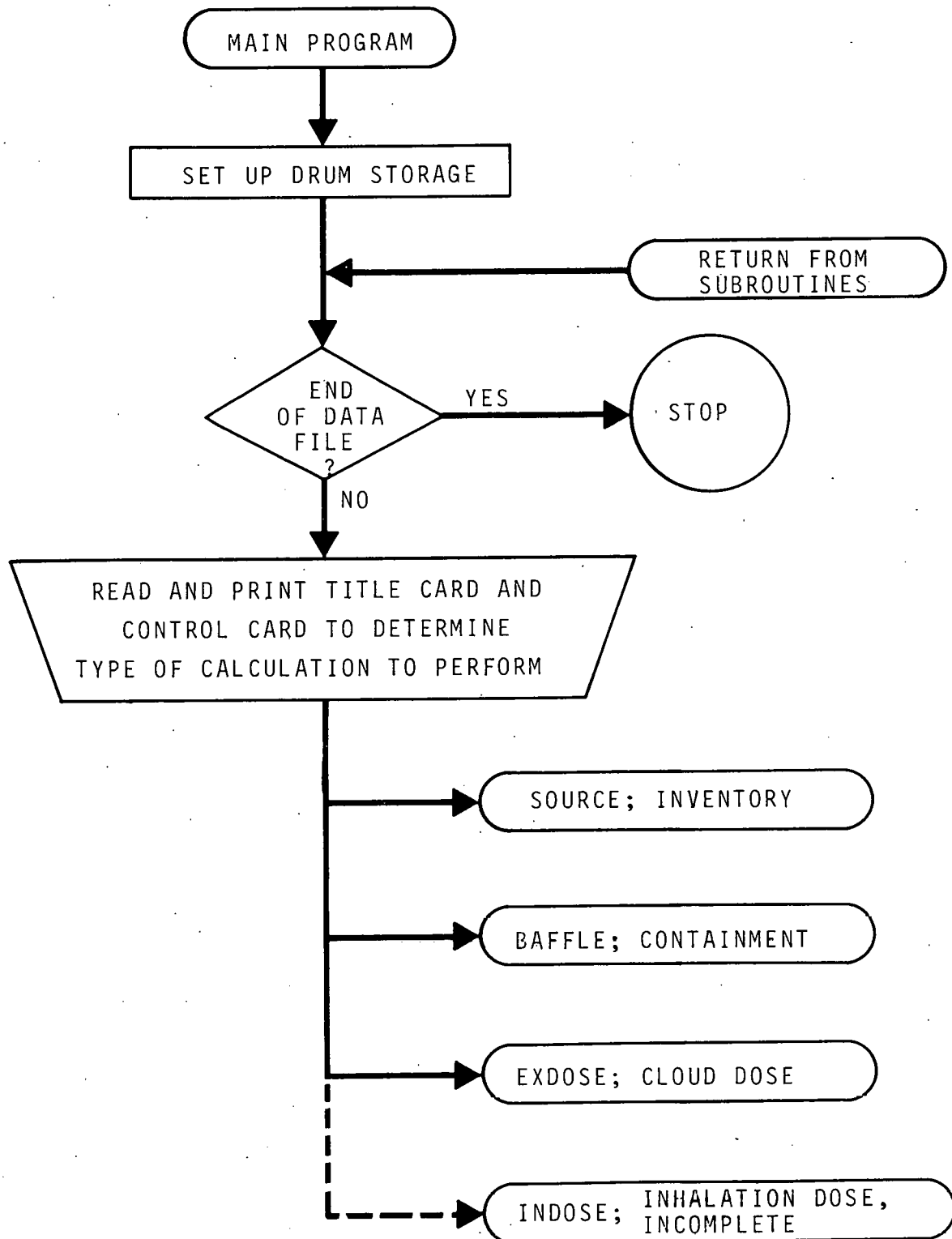
FIGURE C-8. Code Flow Diagram
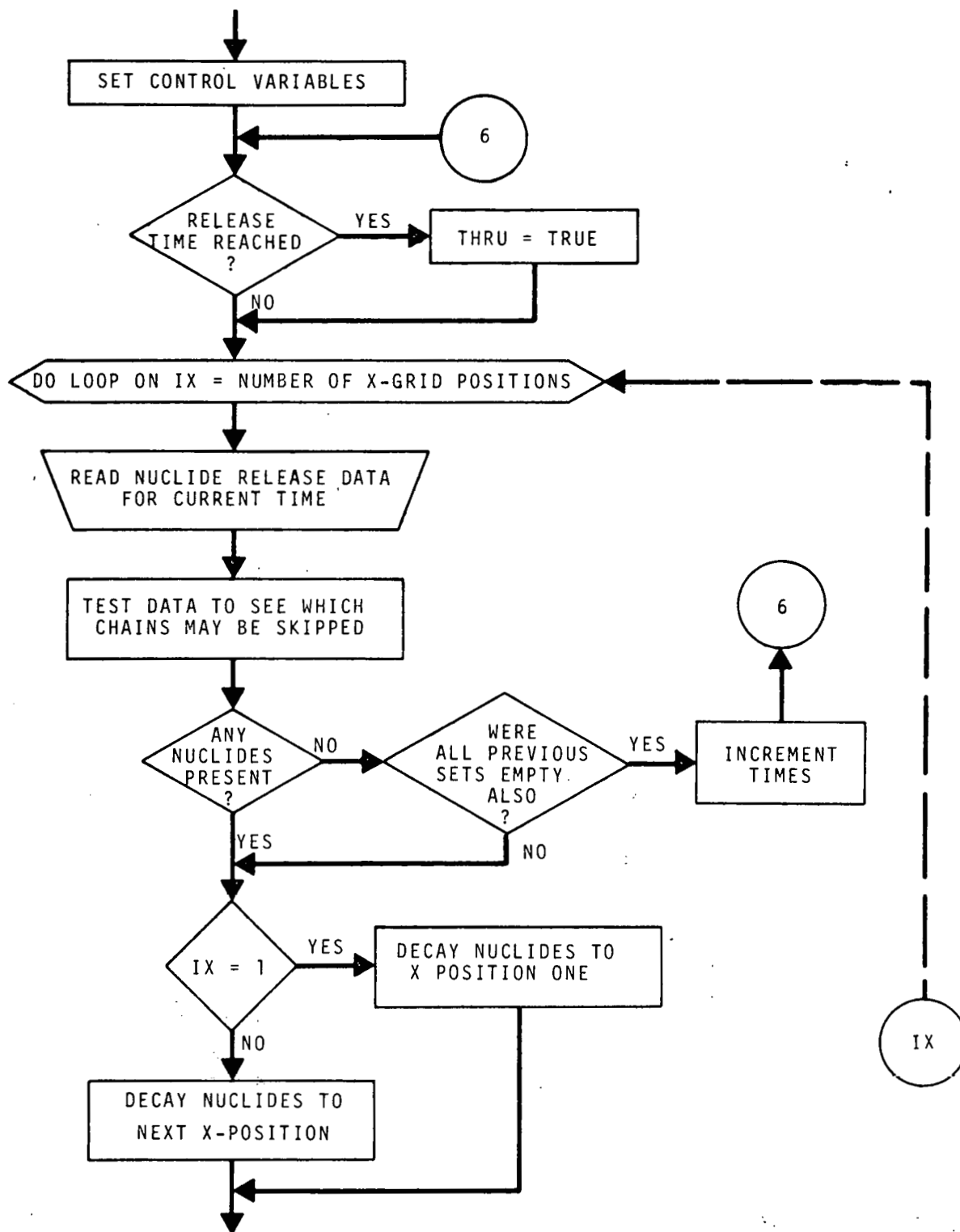
FIGURE C-9.   Code Flow Diagram
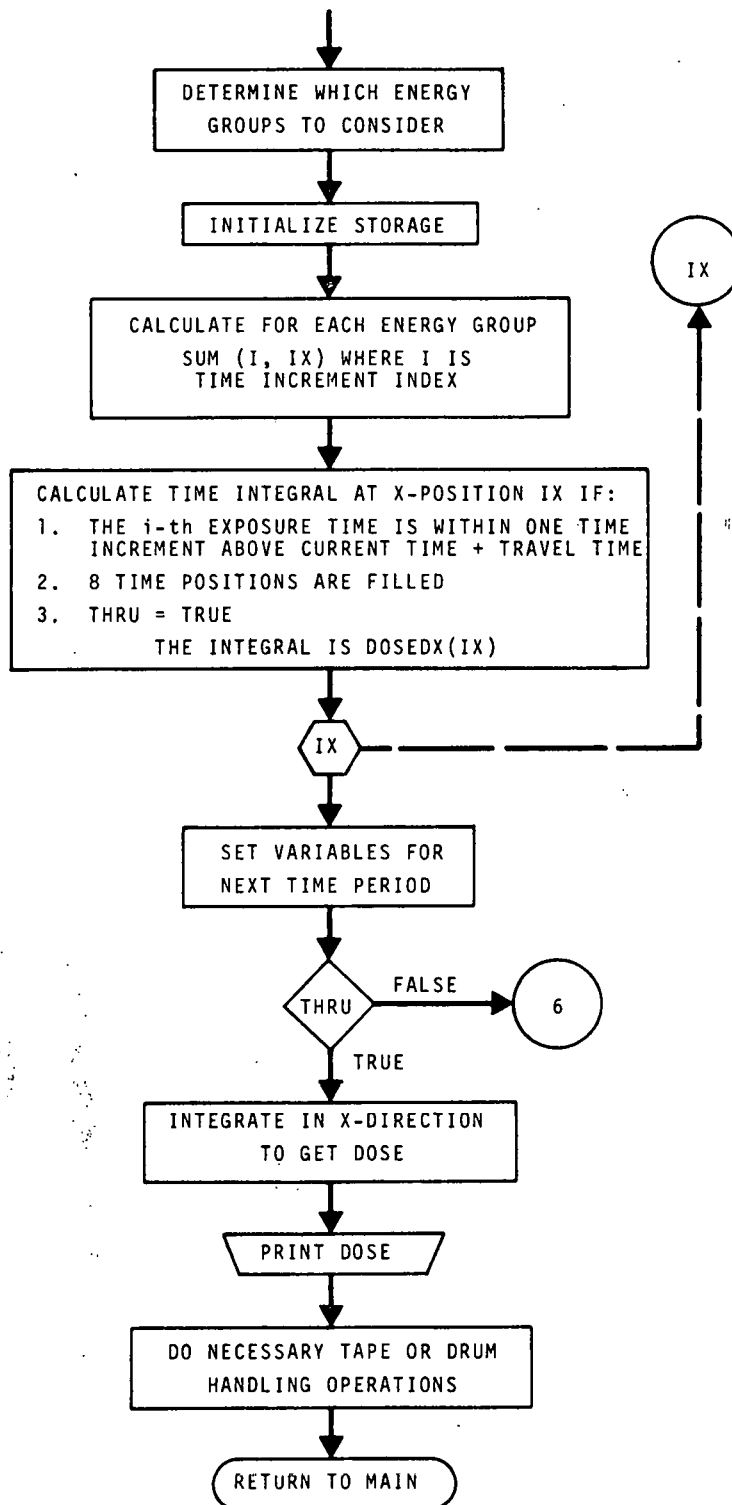
FIGURE C-10. Code Flow Diagram
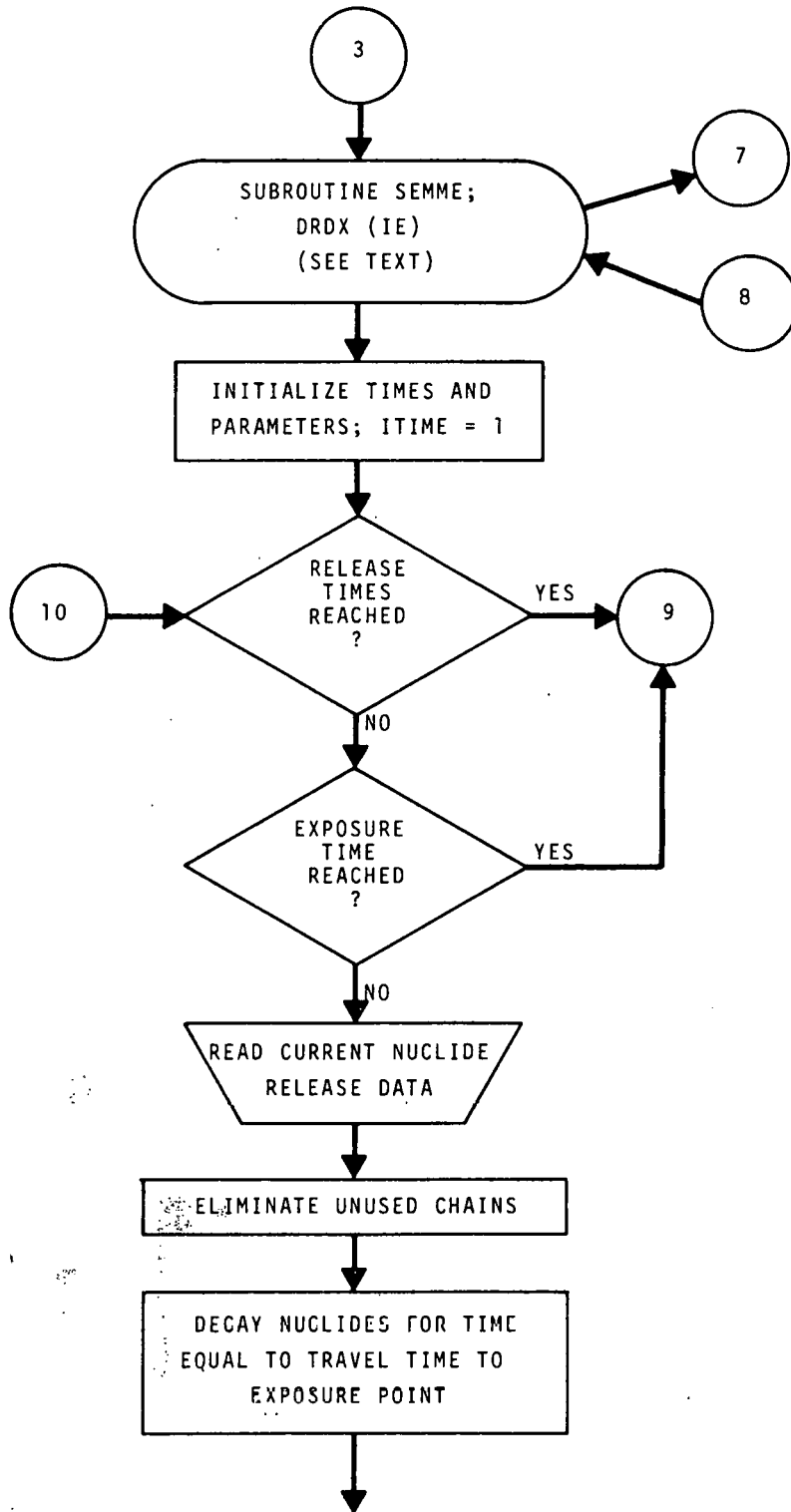
FIGURE C-11.   Code Flow Diagram.
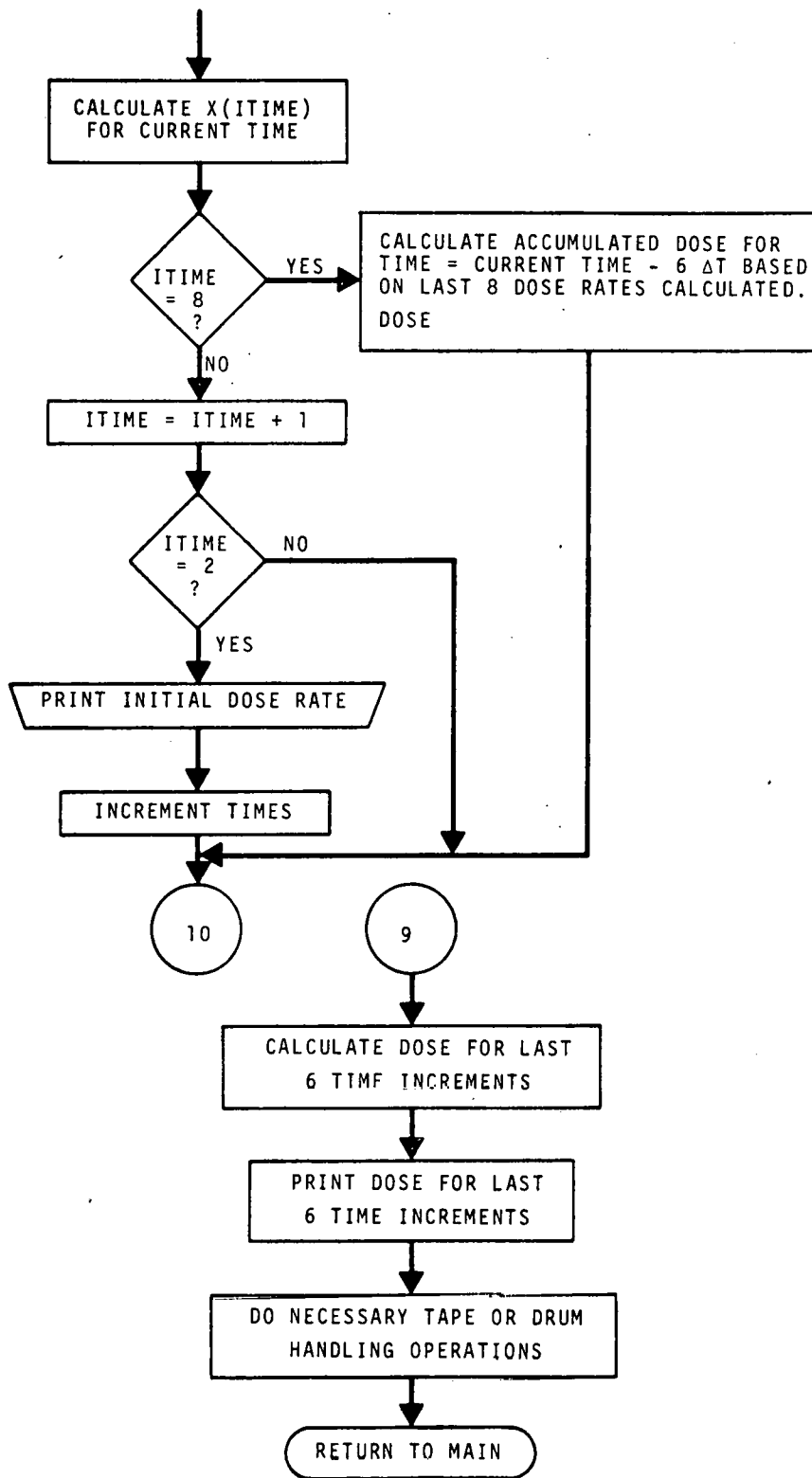
FIGURE C-12. Code Flow Diagram

SUBROUTINE PLUME

(4)

INITIALIZE PARAMETERS

READ AND PRINT METEOROLOGY TYPE
AND POSITION DATA

READ AND PRINT
EXPOSURE TIMES

COUNT EXPOSURE TIMES AND
ARRANGE IN INCREASING ORDER

SET MAIN GRID POSITIONS
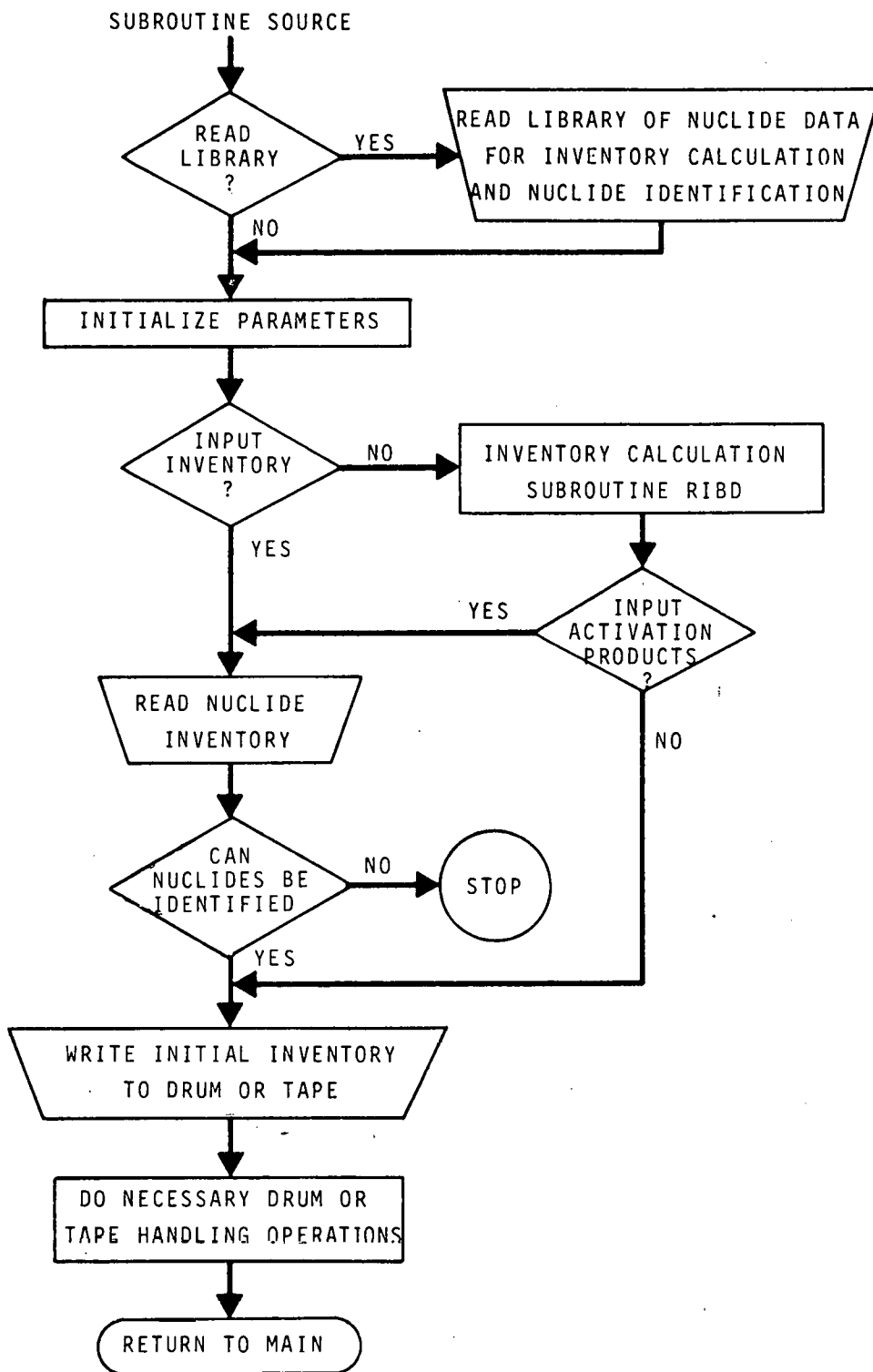IN X DIRECTION

TRANSFER ON TYPE

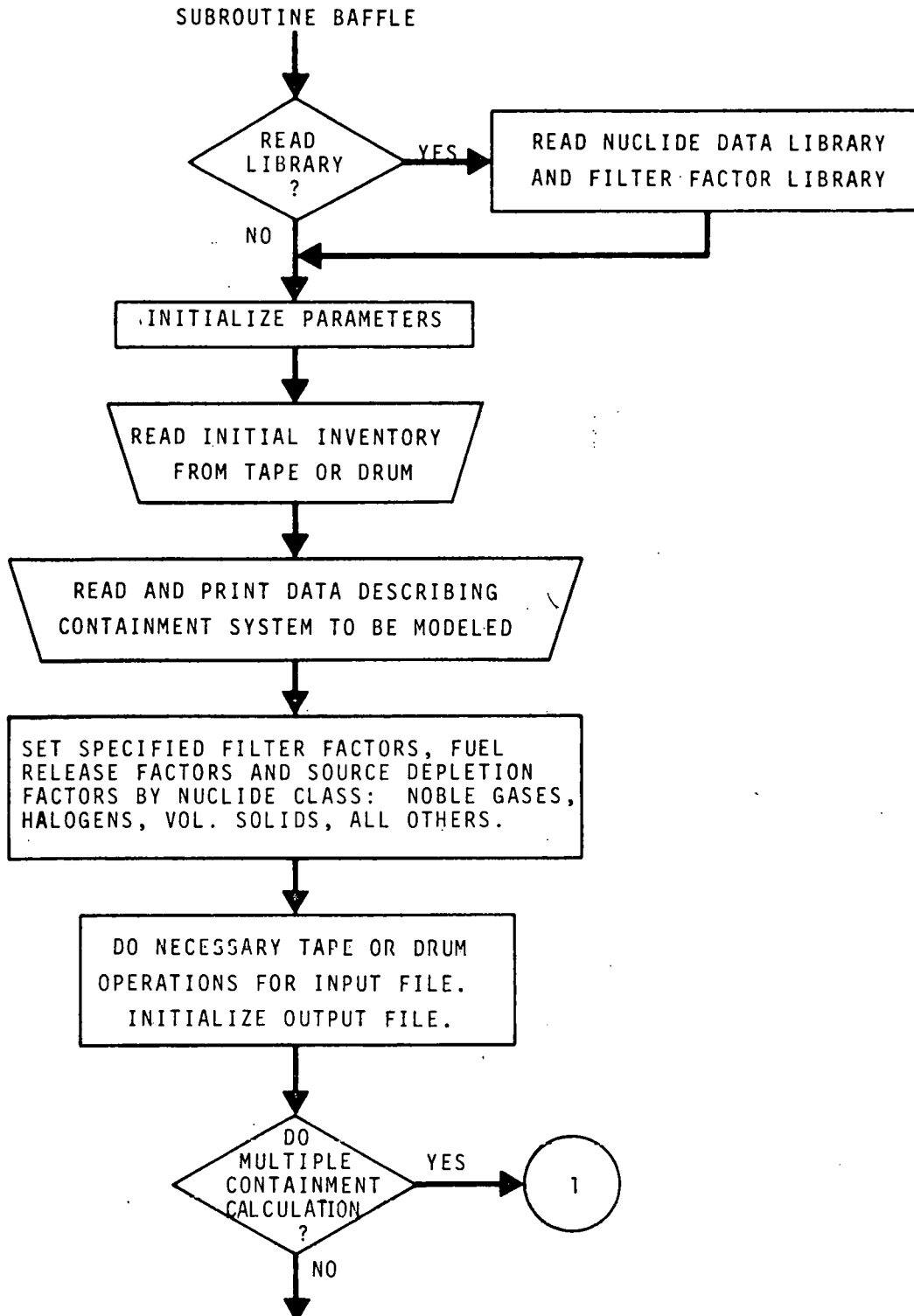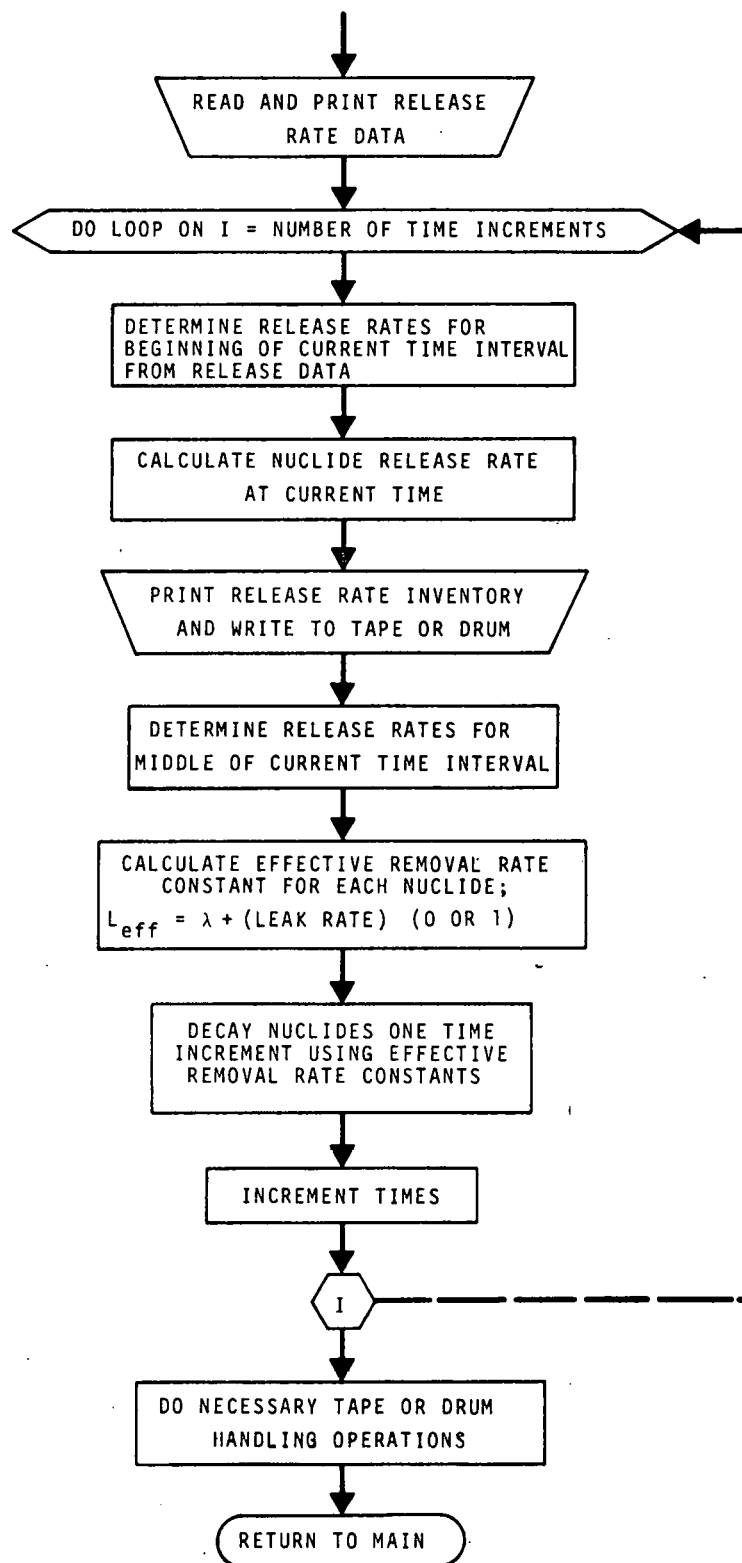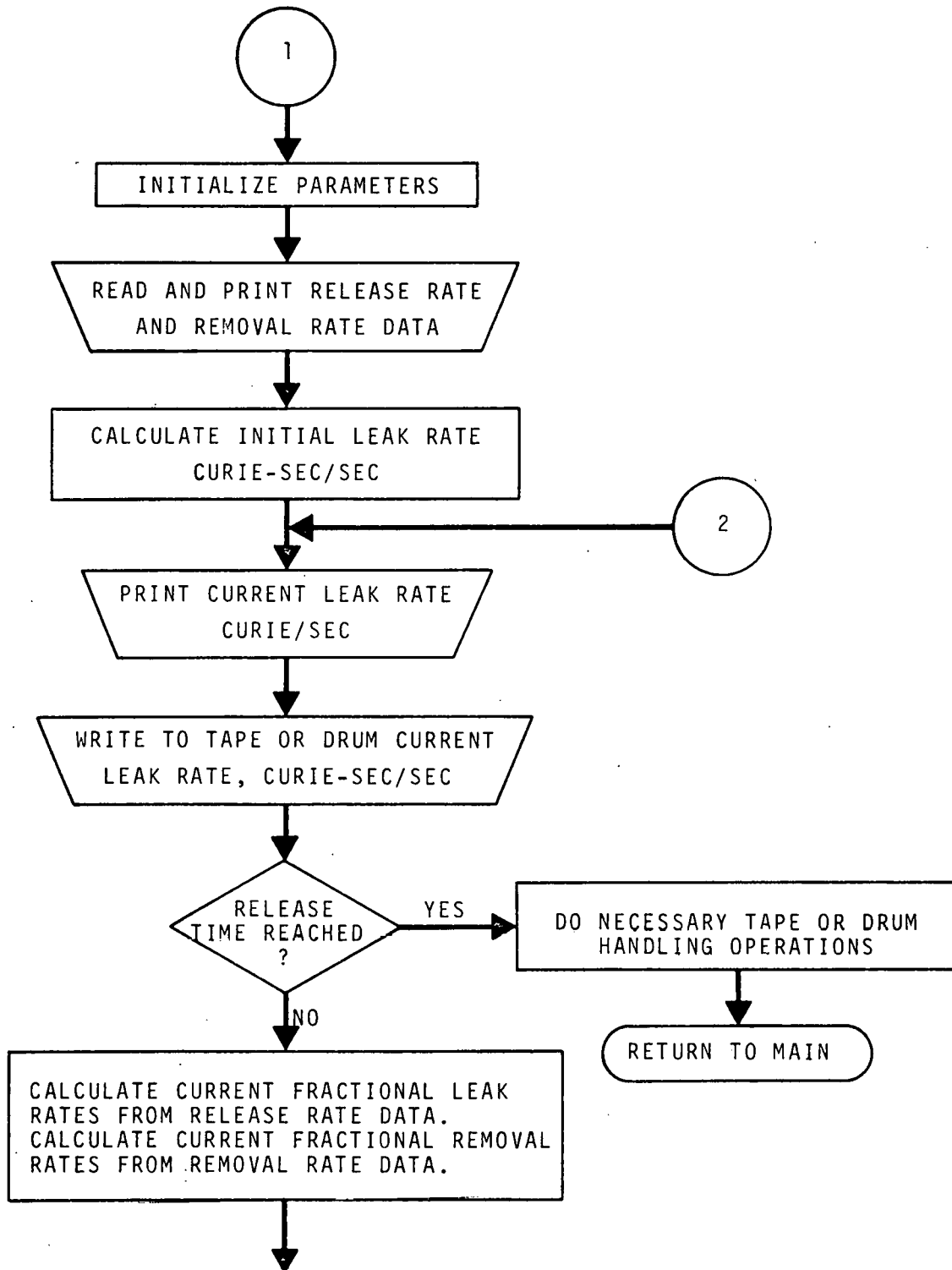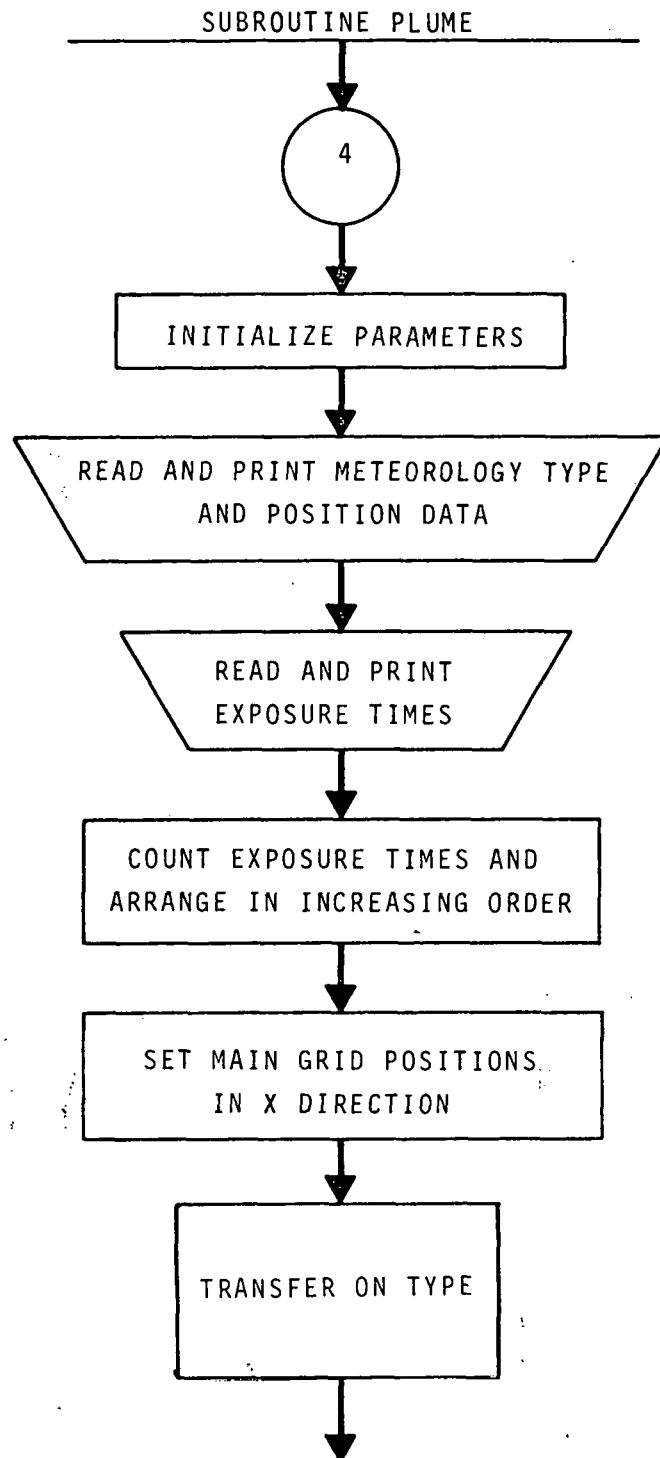FIGURE C-13. Code Flow Diagram
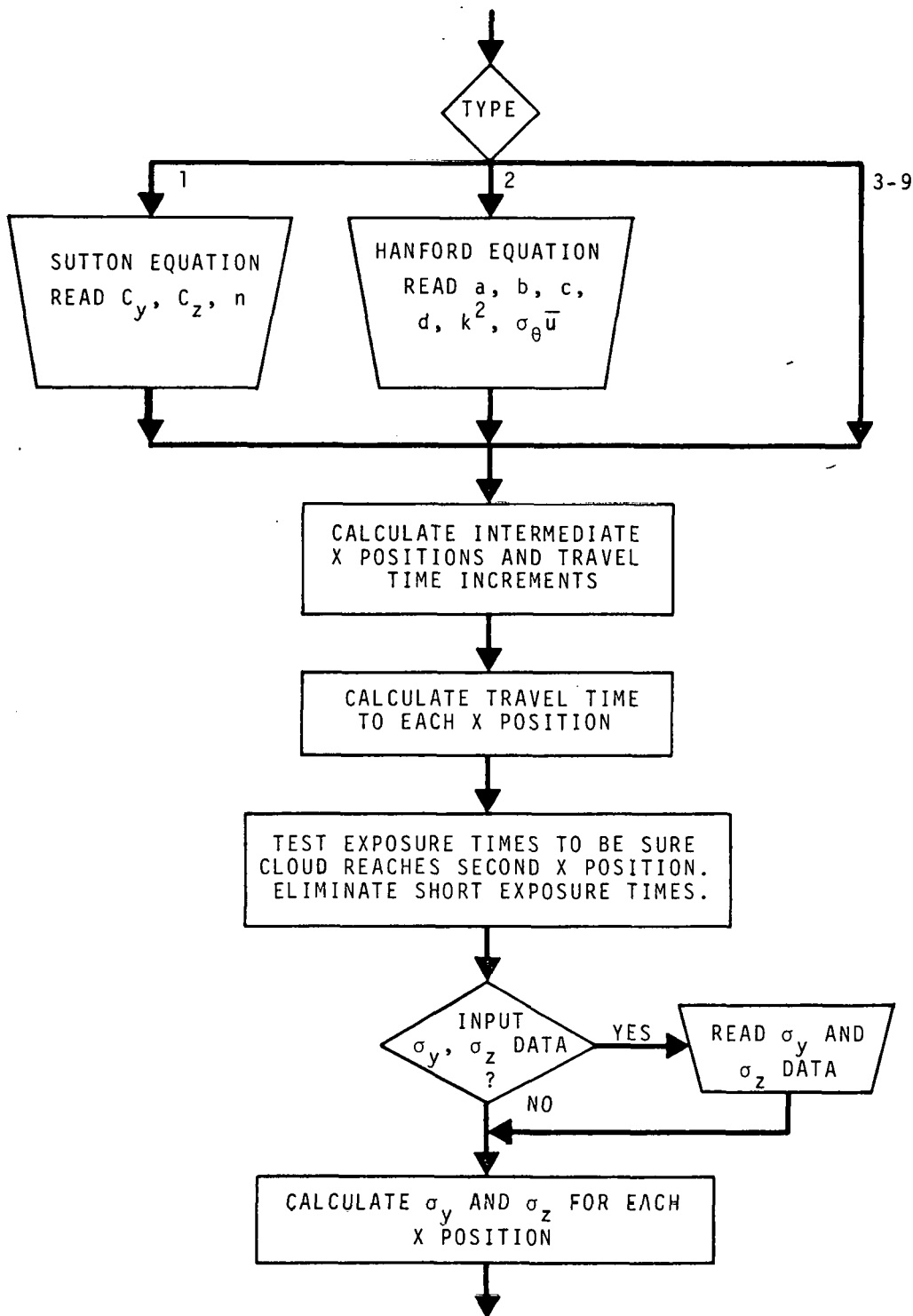
FIGURE C-14. Code Flow Diagram
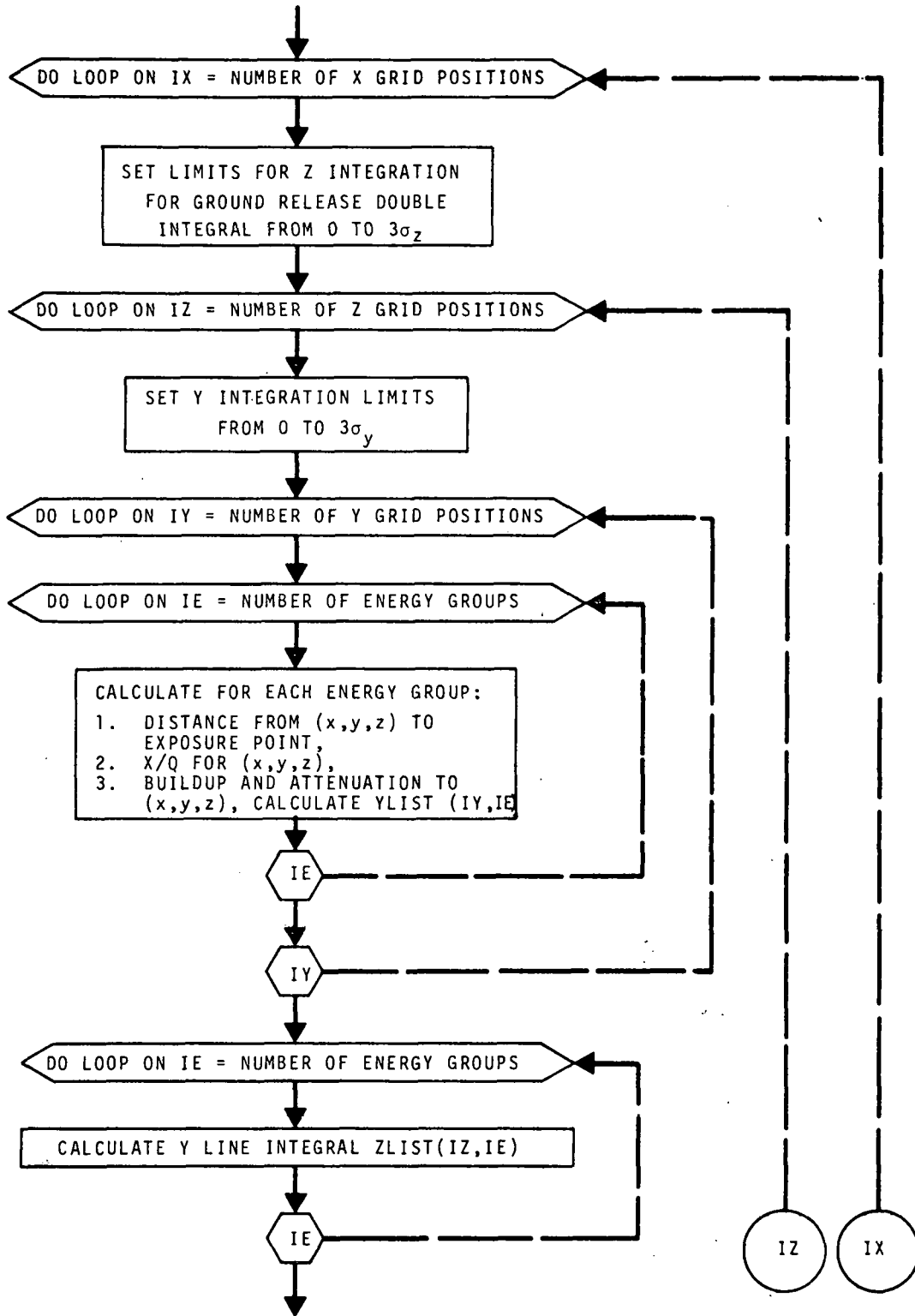
FIGURE C-15.   Code Flow Diagram
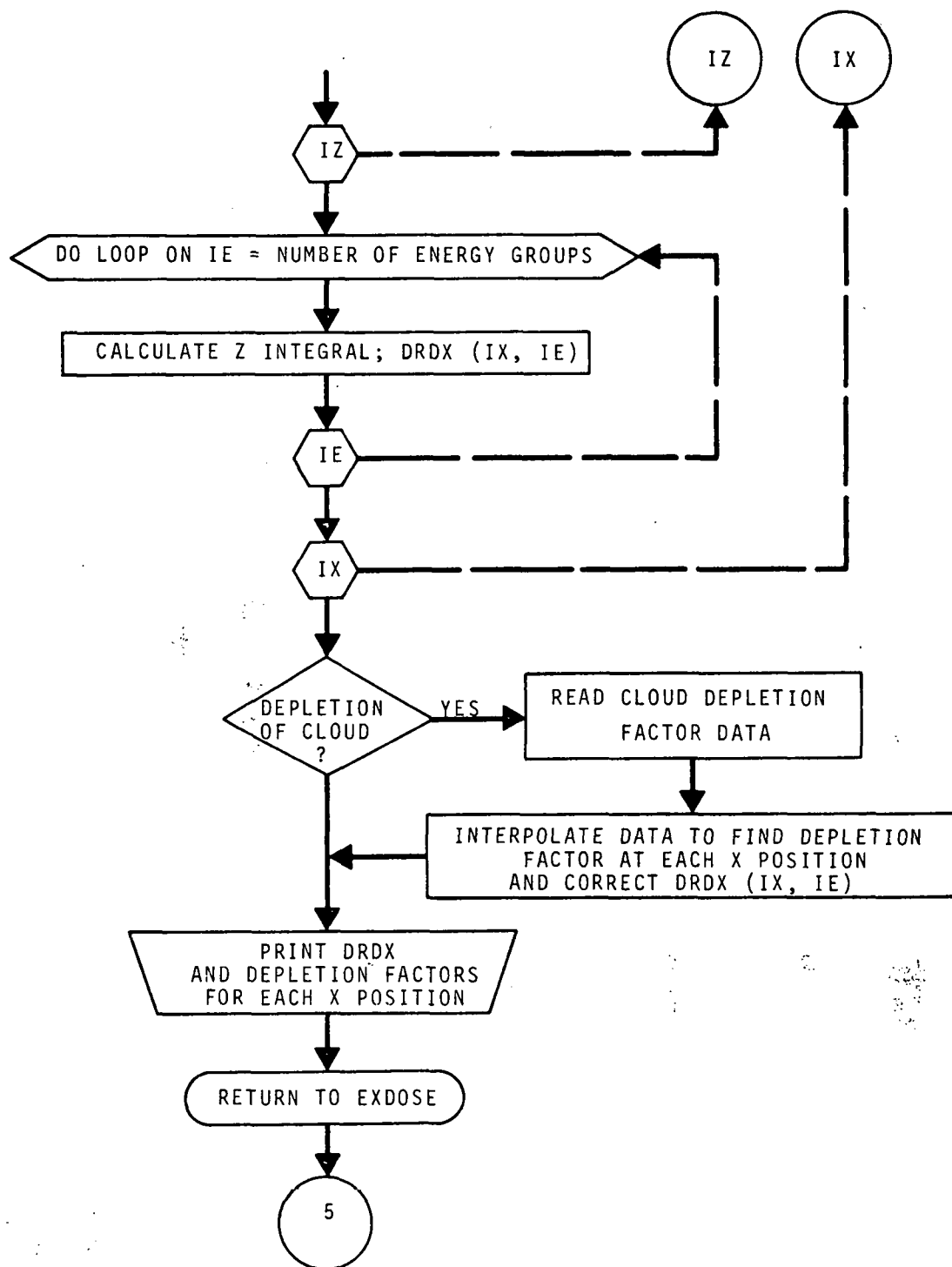
FIGURE C-16. Code Flow Diagram

SUBROUTINE SEMME

(7)

READ AND PRINT METEOROLOGY
TYPE, EXPOSURE POINT DATA
AND TIME DATA

TYPE

1

SUTTON EQUATION
READ C_y, C_z, n

2

HANFORD EQUATION
READ a, b, c, d
K^2, $\sigma_\theta u$

3

READ σ_y, σ_z

4-9

PASQUILL; CALCULATE
σ_y AND σ_z FROM
STORED DATA

CALCULATE X/Q AT
EXPOSURE POINT

CALCULATE DRDX(IE)
FOR
EACH ENERGY GROUP

PRINT DRDX(IE) AND
METEOROLOGICAL DATA

RETURN TO EXDOSE

(8)
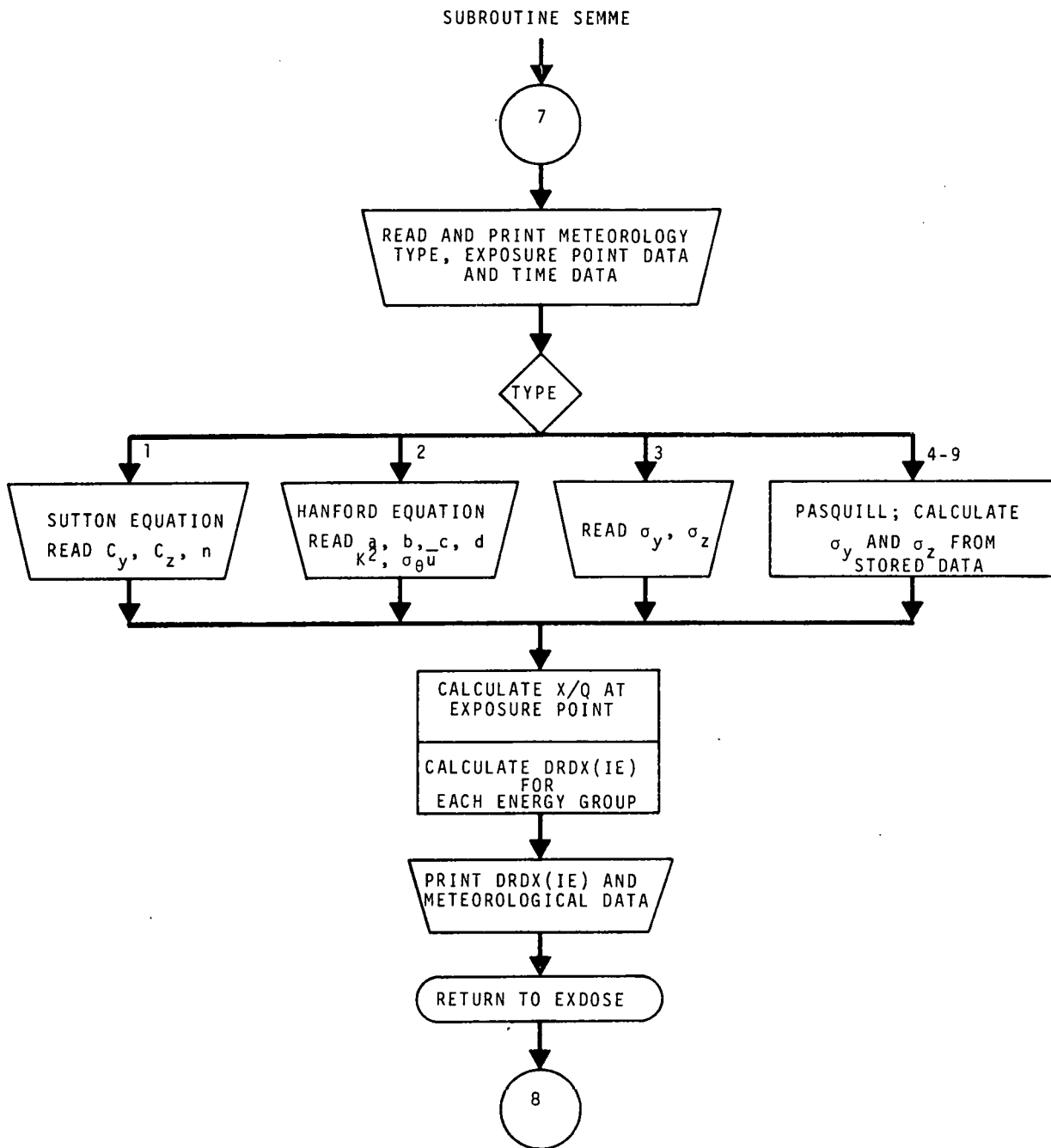
FIGURE C-17. Code Flow Diagram

Data Libraries

Three types of data libraries are used by the code.

1. Isotope Library (used by SOURCE, BAFFLE, and EXDOSE).

2. Photon Probability Library (used by EXDOSE only).

3. Filter Factor Library (used by BAFFLE only, optional).

The libraries have been constructed in a general manner so that data may be added or changed without difficulty.

Isotope Library

This library is used by SOURCE to calculate the fission product inventory. Activation product half lives are included at the end of the library. The order of isotopes in this library determines the order for which data is supplied in the other libraries. When the order of nuclides is changed in the isotope library, corresponding changes must be made in the other libraries to be used.

The first card of the isotope library contains the number of fission products in the library (I3). There is one card for each fission product, and they are in an ascending atomic number order within an ascending atomic mass sequence. The current isotope libraries contain 450 fission products.

The card after the last fission product card contains the number of activation products (I3). There is one card per activation product. The current isotope libraries contain 144 activation products.

The data format is:

Column	Format	Variable
1-3	I3	Atomic Mass Number
4-5	I2	Atomic Number
6-14	E9.3	Physical half life, days
15-23	E9.3	Fission yield from species one (^{235}U), %
24-32	E9.3	Fission yield from species two (^{239}Pu), %
33-41	E9.3	Absorption cross section, barns
42-46	F5.3	Fraction of (n,γ) captures that go to isomeric state
47-51	F5.3	If in isomeric state, fraction of decays by beta emission; if in ground state, fraction that decays to an isomer.
52-56	F5.3	Average beta energy released per disintegration, MeV.
57-61	F5.3	Average gamma energy released per disintegration, MeV.
69-71	A3	Isotope name (left adjusted)
72	I1	Isotope type: 1 - Isomeric State 2 - Ground State

Two isotope libraries are available: THERM for thermal reactor calculations and FAST for fast reactor calculations.

Photon Probability Library

This library contains data for use by EXDOSE in calculating the cloud gamma dose. The first card contains the number of energy groups (I3) for which data is given in the library (maximum of 16 groups). The second card contains the upper energy bound for each group (16F5.0) starting from the lowest energy group (number of entries is number of groups minus one). The next two cards contain the dose conversion factor for each energy group (8E10.0) in units of

$$\frac{(rads \ in \ tissue)(cm^2)}{curie \ sec}$$

The next cards contain the photon probabilities and energies for each
nuclide (fission products and activation products) contained in the isotope
library in the same order. More than one card may be used for each isotope
depending on the number of different gamma emissions involved. The first
card for each has the following format:

Column	Format	Variable
1-3	I3	Atomic mass number
4-5	I2	Atomic number
6-7	A2	Nuclide name
8	I1	Isotope state: 1 - isometric 2 - ground
9-10	I2	Number of photon energies to be read
11-15	F5.0	Abundance of first photon (if any)
16-20	F5.0	Energy of first photon, MeV
21-25	F5.0	Abundance of second photon (if any)
26-30	F5.0	Energy of second photon, MeV
.		
.		
.		
61-65	F5.0	Abundance of sixth photon (if any)
66-70	F5.0	Energy of sixth photon, MeV

If the nuclide has more than 6 photons, additional cards are needed. The
format is 14F5.0, with the abundances and energies submitted in pairs (for
up to 7 photons) as on the first card but starting in column 1. A maximum
of 20 photons may be supplied for each nuclide (3 cards).

Buildup and attenuation data are stored after the photon data. Constants
for the quadratic buildup equation $(B = 1 + A(\mu R) + \alpha(\mu R)^2)$ are given next
in format 8F10.0. The first card gives values for A for each energy group
(if there are more than 8 groups, more cards are needed). Next the values
of α are given, followed by values of α_1 (a dummy variable) and values of
μ last (total mass attentuation coefficient in air, cm^2/g).

The photon probability library is called ISOLIB.

Filter Factor Library

This library contains filter factor data for the isotope library fission products. The first card has the number of fission products to be read (I3). There is one card for each nuclide with data in the following format.

Column	Format	Variable
1-3	A3	Isotope name (left adjusted)
4-6	I3	Atomic mass number
7-16	F10.4	Fuel release factor
17-26	F10.4	Filter factor
28	L1	Logical variable to indicate isotopes that are resuspendable from filters or settled material
		T - resuspendable
		F - not resuspended

The filter factor library is called FILTER

Control Cards

When a CSC INFONET Univac 1108 is used, input cards for each run must be preceeded by DAT cards and an XQT card. When libraries are to be used, the input deck must contain a DAT card for each library. The DAT cards are prepared as follows:

Card	Library
∇ DAT Library Name, 1	Isotope library
∇ DAT Library Name, 2	Photon probability library
∇ DAT Library Name, 3	Filter factor library

The symbol ∇ is entered as a 7-8 punch. A DAT card must be supplied for each library to be used immediately before the XQT card. The XQT card for the current version of the code is: ∇ XQT RACER.

Input Preparation

The input to each main subroutine of RACER (SOURCE, BAFFLE, and EXDOSE) consists of a "packet" of cards describing the calculations to be performed. In each packet there is a title card, a "control word" card, and data cards. The title card contains 80 characters of alphameric data and is the first card of each packet. The title card must be present even though it has no effect on the calculations. The second card of each packet is the control word card. The first word of this card must be the name of one of the major subroutines: SOURCE, BAFFLE, or EXDOSE. The first word must start in column one. Other words (described below) can be anywhere on the card and in any order but must be separated by at least one blank space. None of the control words can have interior blanks. The remaining cards in each packet are data cards.

Only one main subroutine can be in the computer core at a time, due to size limitations on the computer. Therefore, it is necessary to store intermediate data on tape or drum. Figure F-1 illustrates the relation of tape drives and drums for the four main subroutines. This relation allows the subroutines to be called in any order. For example, a typical run may have the following calling sequence:

    SOURCE (for inventory)
    BAFFLE (for release)
    EXDOSE (gamma dose for 1st meteorological condition)
    EXDOSE (gamma dose for 2nd meteorological condition)
    BAFFLE (different release of same inventory)
    EXDOSE (gamma dose for 1st meteorological condition)
    EXDOSE (gamma dose for 2nd meteorological condition)
    SOURCE (new inventory)
    etc.

Details of the input and operation of each subroutine are described below. Appendix G contains two sample problems to illustrate input preparation.
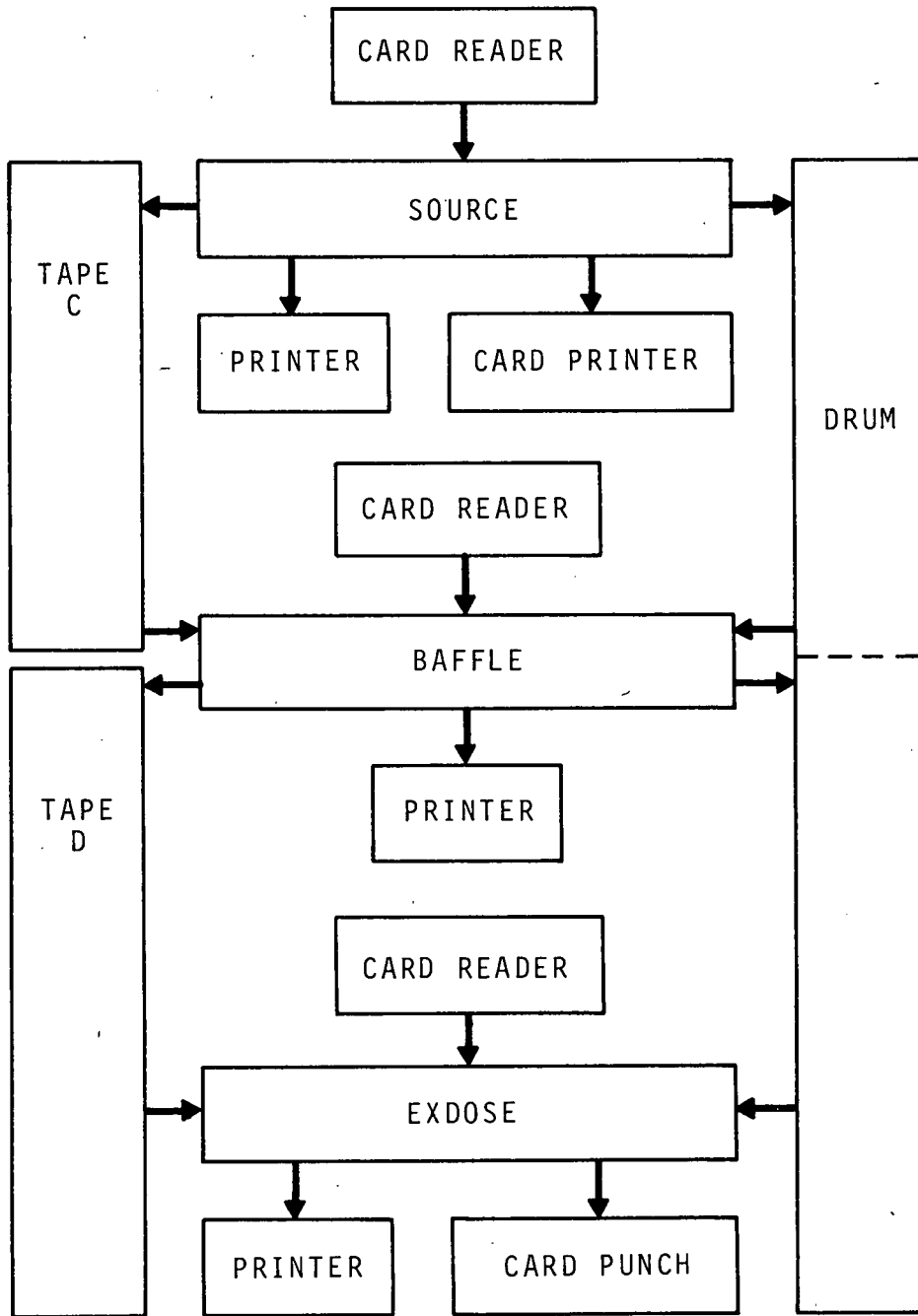
FIGURE F-1. Code Input-Output Diagram

SOURCE Input

The subroutine SOURCE specifies the radionuclide inventory at time zero.
The inventory may be input as curies of particular nuclides or a fission
product calculation (subroutine RIBD) may be made. The nuclides are
classified as either fission products or activation products; activation
products are those nuclides not listed as fission products in the program
libraries. Special provisions are included for specifying activation
products.

The control word card for the SOURCE packet may contain the following words:

Word	Purpose
SOURCE	Direct control to subroutine SOURCE. This must be the first word on the card.
BYPASS	Cause the RIBD fission product inventory calculation to be bypassed. If absent, a RIBD inventory calculation will be performed. When BYPASS is present, the curies of specific nuclides are to be input on data cards as specified below.
LIBRARY	Cause an isotope library to be read. LIBRARY must be present in the first SOURCE packet and in each SOURCE packet after another major subroutine has been run. A library DAT card must be supplied for the isotope library (See Appendix D).
ACTIVATION	Cause an activation product input when BYPASS is absent. ACTIVATION is ignored if BYPASS is present, in which case activation products may be specified with fission products.
TAPE	Cause SOURCE output to be written onto tape. If TAPE is absent, the output will be written to drum. If TAPE is used, a tape must be assigned to Unit C.
REWIND	Cause tape Unit C to be rewound.
BACKSPACE	Cause tape Unit C to be backspaced one file.

The data cards for the SOURCE packets depend on the control words BYPASS
and ACTIVATION. When a fission product inventory calculation is to be
made (BYPASS absent), the first data card contains reactor operating
parameters for the first period of operation. Data on this card are:

Parameter	Format	Description
P	F10.0	Reactor operating power in MW.
t	F10.0	Length of operating period in days.
T	F10.0	Exposure parameter, $T = t\sigma\phi$, where ϕ is neutron flux. T should not exceed 4.
σ	F10.0	The absorption cross section of the first fissionable species, barns.
Δ	F10.0	Fission ratio; initial ratio of second to first species fissions. High values of Δ (10,000) suppress first species fissions.
A	F10.0	A control number to signify whether the irradiation history is complete or whether additional operating data are to follow. A zero (or blank) signifies this to be a "one shot" buildup at the power level and for the time specified. A number greater than one marks this data card as the first in a series describing the irradiation history. Additional cards contain only P, t, and A in that order (3 F10.0). The last power level must be non-zero.
ENF	F10.0	Energy release per fission MeV/fission. If no value is specified, 200 MeV/fission will be used.

The card following the last irradiation history card contains 9 decay times in seconds (after shutdown) for which the fission product inventory is to be calculated. The format is 9F8.3 for the decay times. Column 73 is used to specify which decay time inventory is to be used as the initial fission product inventory for the BAFFLE calculation. The integer in column 73 (from 0 to 9) corresponds to the decay time to be used as entered on the card from left to right. A zero indicates using the shutdown inventory, a one, the first decay time on the card, etc. To have the nuclide inventory punched onto cards, place a T in column 75. A T in column 77 will suppress printing of γ and β power reports. The first 36 characters on the title card are printed at the top of each page of output for the fission product calculation.

When ACTIVATION is absent, the decay time card is the last card of the packet. If ACTIVATION is present (and BYPASS absent), the next card gives

the number of activation products to be read (I3). The following cards give the curies of the activation products with four nuclides per card. Twenty columns are used for each nuclide as follows:

Column	Use
1-3	Character name of the nuclide left adjusted. Isomers are indicated by an asterisk; i.e., KR*. Format is A3.
4-6	Atomic mass number of the nuclide right adjusted (I3).
7-20	Isotope inventory in curies (E14.0).
21-23	Character name of second nuclide.
etc.	

The nuclides specified must be taken from the list of activation products at the end of the program libraries associated with this code.

When BYPASS is present, no fission product inventory calculation is made. For this case, the first data card after the control word card gives the number of nuclides to be read (maximum of 450; fission products and/or activation products). The format is I3. The curies of each nuclide is entered according to the same format as specified for activation products above.

The output from SOURCE is the number of curies of each nuclide at time zero for the BAFFLE calculation. The output is to tape if TAPE is present on the control word card; otherwise, output is to drum. Output to tape may be saved for future runs.

BAFFLE Input

BAFFLE calculates the nuclide release rate for the nuclide inventory supplied on tape or drum. A diagram of the three barrier containment model is given in Figure A-1. A simplified version is also included in BAFFLE that allows for leakage from a single barrier with no filters.

The control words available with BAFFLE are:

Word	Purpose
BAFFLE	Direct control to subroutine BAFFLE (first word on the card).
BYPASS	Determine how the release to the atmosphere is to be calculated. If BYPASS is present, release will be from the single barrier system; if absent, release will be from a multiple containment system.
LIBRARY	Cause a filter factor library and a nuclide data library to be read. LIBRARY must be present in the first BAFFLE packet and after each major function is run.
TAPEIN	Indicates that input from SOURCE is on tape (Unit C). If TAPEIN is not present, input will be read from drum.
TAPEOUT	Indicates output from BAFFLE is to be put on tape (Unit D). If TAPEOUT is not present, output will be put on drum. A tape must be specified for Unit D.
REWINDIN	Causes tape Unit C to be rewound. The word is ignored if TAPEIN is absent.
REWINDOUT	Causes Unit D to be rewound. The word is ignored if TAPEOUT is absent.
BACKSPACEIN	Cause Unit C to be backspaced one file. The word is ignored if TAPEIN is absent.
BACKSPACEOUT	Causes Unit D to be backspaced one file. The word is ignored if TAPEOUT is absent.

The first data card after the control word card always has the same format.

Column	Format	Description
1-9	F9.0	ΔT, time increment for calculation of release. ΔT represents the time between release rate data points, seconds.
10	L1	Logical variable, FREAD, set to true (T) when the filter factor library is to be read.
11-15	I5	Number of release intervals to calculate, n. The total time of release is then $n\Delta T$.
16-20	I5	Number of leak rate data points to be read. The minimum is 1, and the maximum is 50.
21-25	I5	For the multiple containment calculation (BYPASS absent), the number indicates the number of leak rates to be specified.
26-30	I5	For the multiple containment calculation, this number indicates the number of removal rates to be specified. The maximum number is 3.

Column	Format	Description
31-35	I5	Control integer; >0 when filter factors are to be input.
36-40	I5	Control integer; >0 when fuel release factors are to be input.
41-45	I5	Control integer; >0 when source depletion factors are to be input.
46-50	I5	Control integer for report printing; >0 when release rates for each time interval are to be printed.

The first time point for leak rate data must be 0.0 seconds.

The filter factors represent the fraction of a nuclide penetrating the filters (on the barriers), the fuel release factors indicate whether or not the nuclide is released from fuel (= 1.0 for release and 0.0 for re-tention), and the source depletion factors reduce the initial inventory prior to the start of the calculation. The fuel release factors are used for both the single barrier release (BYPASS present) and the multiple containment model.

If the filter factor control integer is greater than zero, the next card contains filter factors for the four groups of nuclides in the order noble gases, halogens, volatile solids, and non-volatile solids (4 F10.0). A similar card is entered for each factor for which the control integer is greater than zero. For example, if a one appeared in columns 35 and 45, two cards would be supplied: the first would contain 4 filter factors, and the second would contain 4 source depletion factors. If all three control integers are zero (or blank), no factor cards are read. The input of the three types of factors is not dependent on the word BYPASS.

A filter library is included in the code for use in BAFFLE. The library contains fuel release factors, filter factors, and a logical variable to indicate resuspension for each nuclide. The library is read (if FREAD is true) prior to reading filter factor input data cards. Library factors

are used when input data are not given. All factors are set to 1.0 when the library is not read and input data are not given. Fuel depletion factors are all 1.0 (no depletion) unless entered differently on cards. To specify factors for individual nuclides (rather than groups of nuclides), the library may be updated prior to starting the run.

Specification of the leak rate data is dependent on the word BYPASS. If BYPASS is present, the next cards contain time leak rate data with format 8F10.0. Each pair of data entries is a time (sec) followed by the leak rate (sec^{-1}) at that time. Four data pairs are entered per card until the number entered is equal to the number in positions 16-20 on the first data card. This concludes the data for the single barrier calculation.

Specification of the leak rate data for the multiple containment model is more complicated. The first data card (after the "factor" cards, if present) gives the time points for which the leak rate data correspond. The format is 8F10.0; 8 times (seconds) per card in increasing order. Flow rates are specified next.

Consider again the multiple containment model as numbered in Figure A-1. The leak rates are indicated by two integers; the first integer represents the compartment the material comes from, and the second represents the compartment the material goes to. For example, the leak rate from barrier one to barrier two is indicated by 24 (bypassing the filter). For filtration, two equal leak rates must be specified. For example, consider leakage from barrier one (#2) through the filter (#3) to barrier two (#4). Leak rates 23 and 34 must be specified as equal. The leak rates are entered by placing the leak rate identification integer on one card (2I1), with the leak rates (sec^{-1}) on the following cards (8F10.0) in the order corresponding to the times submitted on the previous cards. These cards are repeated for each leak rate to be entered.

The removal rates are entered in a similar manner directly after the leak rates. The removal rates are identified by a single character (1, 2, or 3) corresponding to the barrier the removal occurs in. The identification

number is entered in format I1 with the removal rates on following cards
(8F10.0) corresponding to the times entered previously.

Any leak rates or removal rates not specified are considered to be zero.
Thus, the configuration of the particular containment system to be modeled
is determined by the leak rates and removal rates submitted.

The output from BAFFLE gives the nuclide release rate (curie sec/sec) at
each time point (i.e., at each ΔT time increment) for each nuclide.

EXDOSE Input

EXDOSE performs a gamma dose calculation using the release data generated
by BAFFLE. The calculation may be done using the semi-infinite cloud
assumption or a finite cloud integration may be performed. The code
calculates the whole body gamma dose as rads to tissue.

Control words for the EXDOSE subroutine are:

Word	Purpose
EXDOSE	Directs control to subroutine EXDOSE. This must be the first word on the card.
SEMI	Indicates that the semi-infinite cloud assumption is to be used in calculating the gamma dose. If SEMI is absent, a finite cloud integration will be performed.
LIBRARY	Causes an isotope library and a nuclide photon data library (ISOLIB) to be read. LIBRARY must be in the first EXDOSE packet and on any packet after another major function.
TAPE	Cause input from BAFFLE to be read from tape Unit D. If absent, input will be read from drum.
REWIND	Cause tape Unit D to be rewound.
BACKSPACE	Cause tape Unit D to be backspaced.

The first data card of the EXDOSE packets contains the following information:

Column	Format	Description
1-2	I2	Variable TYPE which indicates the type of meteorological model to be considered. TYPE has an integer value from 1 to 9 as follows:

TYPE	Meteorological Model
1	Sutton parameters
2	Hanford equations
3	σ_y & σ_z input
4	Pasquill curve type A
5	Pasquill curve type B
6	Pasquill curve type C
7	Pasquill curve type D
8	Pasquill curve type E
9	Pasquill curve type F

Column	Format	Description
3-5	I3	Integer variable NDT to indicate the relation between the BAFFLE time increment, ΔT, and the EXDOSE time increment. The relation is $\Delta T(EXDOSE) = NDT \times \Delta T(BAFFLE)$. If NDT is not specified, the two time increments will be equal (not used with SEMI).
6	L1	Logical variable, LP, set to true for printing of intermediate dose factors (not used with SEMI).
7	L1	Logical variable DK, true (T) when decay is not to be considered beyond release point, false otherwise.
8	L1	Logical variable METIN, true (T) when meteorological factors are to be input from cards as calculated from a previous run. When METIN is false (F), the dose rate per curie per cm factors will be calculated (not used with SEMI).
9	L1	Logical variable METOUT causes data to be punched for use in future runs. When METOUT is false (F), no data cards are punched (not used with SEMI). If METIN is true, METOUT is set false.
10	L1	Logical variable DEPLET which indicates if cloud depletion factors are to be submitted. When DEPLET is false (F), no depletion data is read (not used with SEMI).
11-20	F10.0	Average wind speed, $\bar{u}$, in meters/second.
21-30	F10.0	Distance from release point to exposure point, meters.
31-40	F10.0	Height of release, meters.

Column	Format	Description
41-50	F10.0	Lateral displacement of exposure point from plume centerline, meters.
51-60	F10.0	Length of release period measured from start of accident (time of exposure to each point of the cloud) hours. The release time must not be greater than $(I)(\Delta T)$ where ΔT is the time increment for the EXDOSE calculation and I is the largest integer less than (number of release intervals from BAFFLE)/NDT (not used with SEMI).

The following two parameters are used on this card only when SEMI is on the control word card.

Column	Format	Description
61-70	F10.0	Boundary time; length of exposure period measured from start of accident to passage of cloud, hours. The boundary time should be greater than release time plus travel time to exposure point.
71-80	F10.0	Time increment for semi-infinite calculation, seconds. If not specified, the time increment from the containment calculation (ΔT) will be used.

When a finite cloud calculation is desired (SEMI absent), the next card contains from one to five exposure times (hours) measured from the start of the accident.

The largest exposure time should be larger than (release time) + (travel time to exposure point) + $(400 + 2P)/\bar{u}$ where P is the perpendicular distance from the exposure point to the plume centerline. The expression represents the travel time from the exposure point to the downwind boundary of the cloud. The largest exposure time as calculated above is referred to as the boundary time for the finite cloud model.

The next set of data cards describe the meteorology and depend on the variable TYPE and the control word SEMI. Consider first the input for SEMI absent. The input is as follows:

TYPE   Input

1   One card containing Sutton equation parameters Cy, Cz, and n in
         format 3F10.0.

2   One card containing Hanford equation parameters in the order a,
         b, c, d, K^2, and $\sigma_\theta \bar{u}$ in format 6F10.0.

3   This mode indicates values of σ_y and σ_z are to be input directly.

4-9   Pasquill weather conditions--no additional cards needed.

When TYPE = 3 and SEMI is absent, values of σ_y and σ_z are calculated from
input data. The first card contains the number (I3) of positions for which
values of σ_y and σ_z to be read (maximum of 20). The next cards contain
the distances (meters) for which the σ_y and σ_z values correspond (8F10.0)
in increasing order. The next set of cards contains the σ_y values (meters)
corresponding to the distances (8F10.0), and the last set of cards contains
the values of σ_z (meters) corresponding to the distances. The program
performs a linear interpolation of the data to get values at the necessary
distances. When data is not supplied for large enough distances, the code
uses the values for the largest distance given.

When SEMI is present, the above data cards are the same, except the input
for TYPE = 3 is simpler. One card is necessary with values for σ_y and σ_z
(in that order) at the exposure point. The format is 2F10.0, and the units
are meters. When the variable DEPLET is true (T), depletion data are read
next (if DEPLET is false (F), there are not more data). The depletion
factors represent the fraction of material (of all nuclides) remaining
airborne at a given downwind position. The first data card for depletion
factors contains the number of factors to be entered (format I3). The
next cards contain pairs of distances (meters) and corresponding depletion
factors (format 8F10.8)--four pairs per card. The data pairs must be input
in order of increasing distances. The depletion data is interpolated in
the same manner as the σ_y and σ_z data described above.

A special option is included for outputting the intermediate dose rate
per cm per curie factors onto cards. The cards may then be used as input
on future runs to avoid duplicating the calculation of the factors. This

option is controlled through the logical variables METIN and METOUT. When
data cards are to be prepared for future use, METOUT is set true (T) and
METIN false (F). When factors are to be input on subsequent runs, METIN is
true and METOUT is false. All of the necessary data cards are prepared so
only the title card and control word card need be added to the punched
output cards. The variables METIN and METOUT are given the proper value
on the punched cards (T and F, respectively). The cards should not be
altered or rearranged from the order in which they are punched, except that
the release time may be changed. A value is not punched for NDT; a blank
for NDT is interpreted as a one.

Sample Problems

To illustrate the use of the computer program, two sample problems are presented here. Both problems are based on a hypothetical reactor accident involving a boiling water reactor (BWR) with pressure suppression containment.

The first problem will include calculation of: 1) the fission product inventory, 2) release through the reactor containment system, and 3) the whole body gamma dose at several downwind positions. The second problem will use data produced by the first problem to calculate the whole body gamma dose from only the noble gases released from the fuel. The hypothetical accident for these example problems involves rupture of the reactor primary cooling-water system.

Figure G-1 provides a diagrammatic representation of a BWR system with pressure suppression containment. Release from the reactor pressure vessel is at a rate L_{12} into barrier one. The primary coolant water vapor, with the radionuclides, is expelled through barrier one to the pressure suppression pool by leakage path $L_{23} = L_{34}$. The effects of fog sprays and aerosol settling are included for barrier one as S_1. The pressure suppression pool is considered to act as filter F for the nuclides as they pass through.

The second barrier is taken as the space above the pool, with leakage only to the outer containment barrier L_{46} (barrier three). Leakage to the atmosphere is considered to occur through two paths: direct leakage to the atmosphere L_{68} and leakage through ventilation system filters L_{67} and L_{78}. Figure G-2 is a diagram of the containment system as it would be set up for the code showing the parts of the multiple containment model (Figure A-1) which must be specified.

Input for BAFFLE for the first problem is:
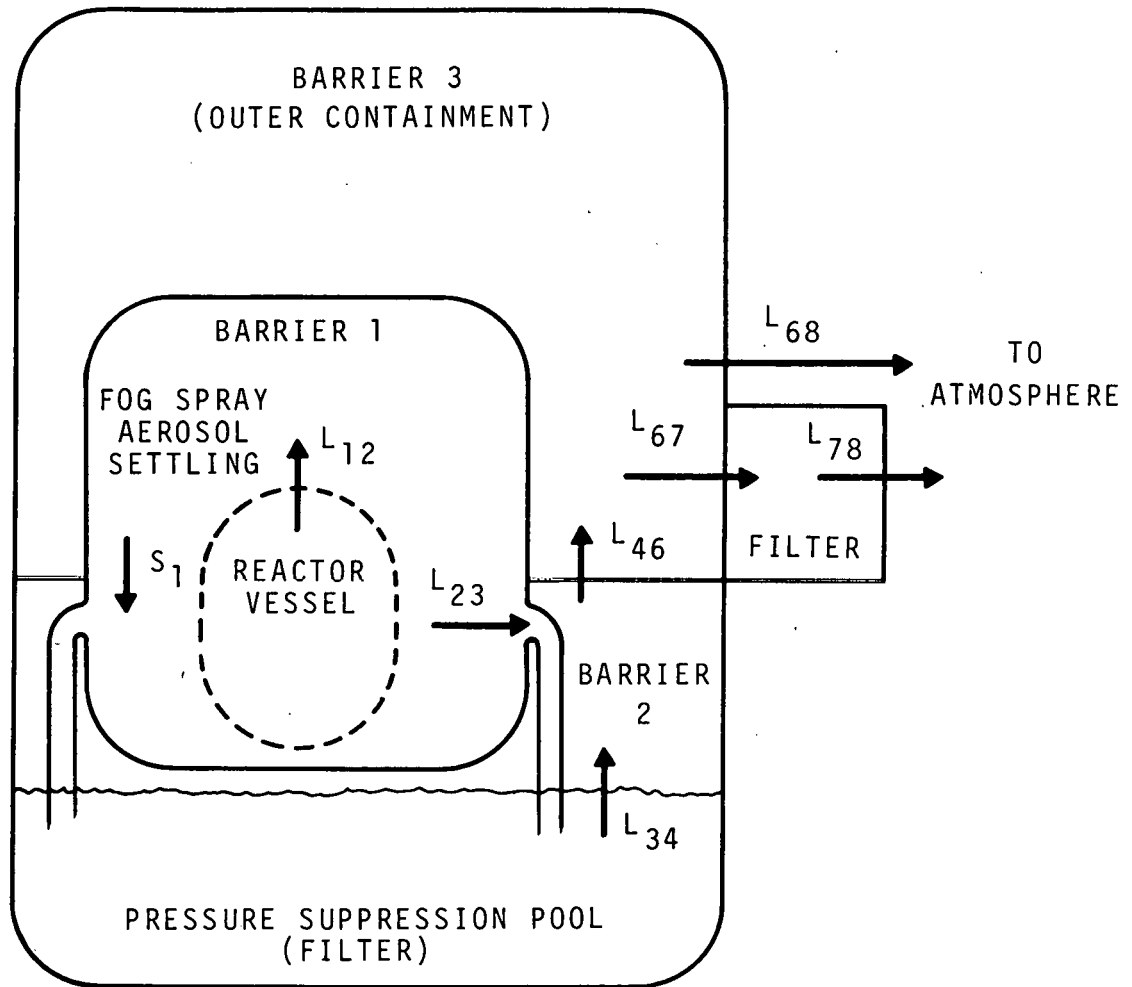
FIGURE G-1. Pressure Suppression Containment Diagram
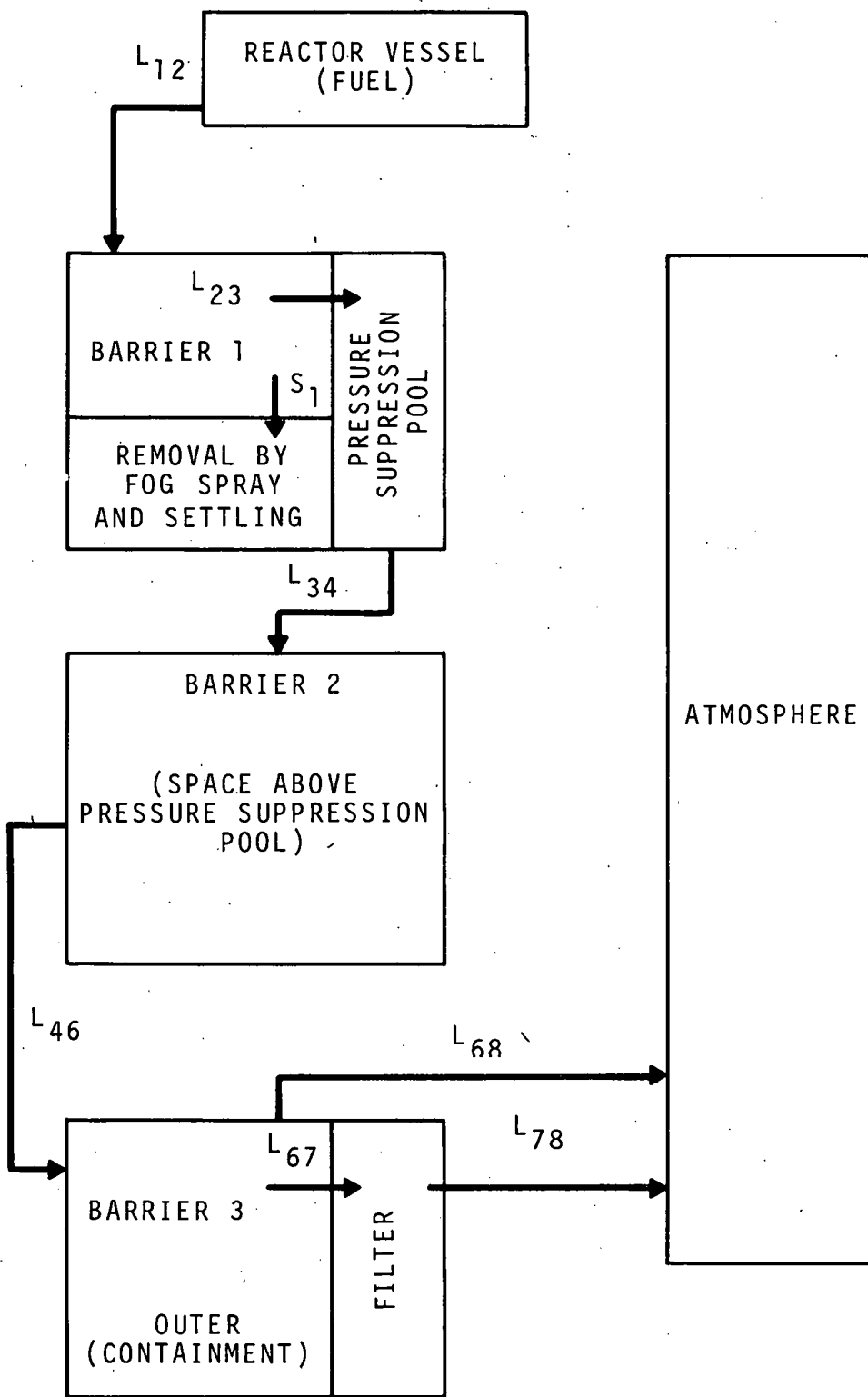
FIGURE G-2.   Pressure Suppression Containment Model

Parameter	Value
Time increments	60 sec
Print control variable	F
Number of increments	120 (for 2 hr release)
Time points	(0.0, ... 7200 sec)
Leak rates at time points selected, L_{12}, L_{23}, L_{34}, L_{46}, L_{67}, L_{78}, L_{68}	(appropriate values, sec^{-1})
Settling rate, S_1	(appropriate values, sec^{-1})
Fuel release factors	
Noble Gases	1.0
Halogens	.5
Volatile Solids	.5
All Remaining Fission Products	.01
Filter factors	
Noble Gases	1.0
Halogens	.05
Volatile Solids	.0005
All Remaining Fission Products	.0005

Note that the code does not apply the settling rate to noble gases. Input for the second problem containment calculation is the same, except the fuel release factors and filter factors are zero for all nuclides except the noble gases.

To identify the nuclides available for release, the operating history of the reactor must be specified. Assume the reactor has been operating for one year at 400 MWth at the time of the accident. Also assume that the following activation products will become airborne: 10^5 curies of ^{59}Fe and 10^3 curies of ^{51}Cr (these values are purely hypothetical).

The input to SOURCE is then:

Reactor Parameters

Power (megawatts)	400.
Time (days)	365.
Flux Parameter (dimensionless ratio)	0.82
Cross Section (barns)	650.
Initial Conversion Ratio	0.0689

Activation Products

^{59}Fe	10^5 curies
^{51}Cr	10^3 curies

The code uses both the calculated inventory for fission products and the supplied activity of activation products ^{59}Fe and ^{51}Cr.

Decay times for the RIBD calculation are:

Time No.	Time	Time (sec)
1	1 min.	60.
2	2 min.	120.
3	5 min.	300.
4	10 min.	600.
5	20 min.	1200.
6	1 hr.	3600.
7	2 hrs.	7200.
8	5 hrs.	18000.
9	10 hrs.	36000.

The shutdown inventory is to be used as the initial inventory for release. The inventory is also to be punched onto cards to be used in the second sample problem. Whole body gamma dose calculations for the sample problems will be made at 100, 300, 1,000, 3,000, and 10,000 meters. An effective release height of 60 meters is assumed for all releases. The calculation will be set to get the dose from the first two hours of released material.

To accomplish this the exposure time must be at least

$$2 \text{ hrs} + \frac{[\text{Distance to Exposure Point} + 400 + 2\cdot(\text{stack height})]}{(\text{wind speed})(3600 \text{ sec/hr})}$$

For a wind speed of 1 m/sec and a 60 meter stack height, the necessary exposure times are :

Distance, Meters	Exposure Time, Hours
100	2.18
300	2.23
1000	2.43
3000	2.98
10000	4.93

Pasquill Type F meteorological conditions are to be used. All exposure points are below the cloud centerline. Intermediate dose factor cards are to be prepared for use with the second sample problem. Input for both sample problems follows. The DAT cards necessary for the CSC INFONET system are also included. See Appendix D for a description of the DAT cards.

Note that for the SOURCE input to the second sample problem, the activation products are nuclides number 449 and 450. Since the code punches 450 nuclides (SOURCE calculation one), two of the fission products with zero curies were replaced by the two activation products. Changes were necessary for the last two nuclide cards only.

```
CONTROL CARDS FOR CSC INFONET UNIVAC 1108 COMPUTERS
-  DAT   THERM,1    (NOTE,  - PEPRESENTS A 7,8 PUNCH)
-  DAT   ISOLIB,2   •
-  XQT   RACER
*****    SAMPLE PROBLEM ONE DATA CARDS    *****
SOURCE CALCULATION FOR BWR SAMPLE PROBLEMS
SOURCE LIBRARY ACTIVATION
   400.         365.        .482      650.       .0689
    60.     120.     300.     600.    1200.   3600.   7200.  18000.  36000.   T T
    2
FE  59            1.E5CR  51          1.E3
RELEASE FROM BWR SAMPLE PROBLEM CONTAINMENT SYSTEM (PROBLEM ONE)
BAFFLE LIBRARY
    60.        120     8     7     1     1     1     0
   1.0        .05      .0005        .0005
   1.0        .5       .5           .01
   0.0        30.      90.          210.      330.      600.      3600.      7260.
  12
   3.0 E-2   1.2E-2    2.0E-3     5.6E-4    4.3E-4    3.5E-4    2.3E-4    2.0E-4
  23
   8.0E-5    2.0E-3    1.0E-3     2.3E-4    1.7E-4    1.3E-4    7.3E-5    6.2E-5
  34
   8.0E-5    2.0E-3    1.0E-3     2.3E-4    1.7E-4    1.3E-4    7.3E-5    6.2E-5
  46
   4.2E-5    5.6E-5    7.2E-5     7.6E-5    7.0E-5    5.7E-5    2.7E-5    2.1E-5
  67
   8.0E-5    8.0E-5    8.0E-5     8.0E-5    8.0E-5    8.0E-5    8.0E-5    8.0E-2
  78
   8.0E-5    8.0E-5    8.0E-5     8.0E-5    8.0E-5    8.0E-5    8.0E-5    8.0E-2
  68
   4.0E-6    4.0E-6    4.0E-6     4.0E-6    4.0E-6    4.0E-6    4.0E-6    4.0E-6
   1
   7.0E-4    5.6E-4    4.1E-4     3.3E-4    3.0E-4    2.6E-4    1.8E-4    1.5E-4
CLOUD GAMMA DOSE CALCULATION FOR BWR SAMPLE PROBLEM - 100 METERS
EXDOSE LIBRARY
   9  5FFFTF  1.0        100.      60.       0.0       2.0
   2.18
CLOUD GAMMA DOSE CALCULATION FOR BWR SAMPLE PROBLEM - 300 METERS
EXDOSE
   9  5FFFTF  1.0        300.      60.       0.0       2.0
   2.23
CLOUD GAMMA DOSE CALCULATION FOR BWR SAMPLE PROBLEM - 1000 METERS
EXDOSE
   9  5FFFTF  1.0       1000.      60.       0.0       2.0
   2.43
CLOUD GAMMA DOSE CALCULATION FOR BWR SAMPLE PROBLEM - 3000 METERS
EXDOSE
   9  5FFFTF  1.0       3000.      60.       0.0       2.0
   2.98
CLOUD GAMMA DOSE CALCULATION FOR BWR SAMPLE PROBLEM - 10,000 METERS
EXDOSE
   9  5FFFTF  1.0      10000.      60.       0.0       2.0
   4.93
*****    SAMPLE PROBLEM TWO DATA CARDS    *****
SOURCE FOR BWR SAMPLE PROBLEM TWO
SOURCE BYPASS LIBRARY
450
*****     THE NEXT 113 CARDS CONTAIN FISSION PRODUCTS AS CALCULATED FOR   *****
*****     PROBLEM ONE.  THE ACTIVATION PRODUCTS ARE ADDED IN THE LAST TWO *****
*****     POSITIONS, 449 AND 450.   *****
ZN   72      2.014+02GA   72      2.012+02GE   72      0.000    ZN   73      1.911+02
GA   73      1.950+02GE   73      0.000    GA   74      6.823+02GE   74      0.000
GA   75      7.798+02GE*  75      7.799+02GE   75      1.560+03AS   75      0.000
GA   76      1.950+03GE   76      0.000    AS   76      6.652+00SE   76      0.000
GE*  77      1.053+04GE   77      1.101+04AS   77      1.781+04SE*  77      5.714-02
SE   77      0.000    GE   78      3.510+04AS   78      3.900+04SE   78      0.000
AS   79      7.798+04SE*  79      7.800+04SE   79      9.922-01BR   79      0.000
AS   80      1.560+05SE   80      0.000    BR*  80      1.145-03BR   80      4.514-03
KR   80      0.000    AS   81      2.437+05SE*  81      1.640+04SE   81      2.602+05
BR   81      0.000    KR   81      0.000    SE   82      0.000    BR   82      7.392+02
KR   82      0.000    SE*  83      7.550+05SE   83      4.735+05BR   83      1.349+06
KR*  83      1.349+06KR   83      0.000    SE   84      1.617+05BR*  84      5.129+04
BR   84      2.566+05KR   84      0.000    AS   85      1.252+06SE   85      2.504+06
BR   85      3.511+06KR*  85      3.512+06KR   85      5.531+04RB   85      0.000
SE   86      2.510+06BR   86      5.021+06KR   86      0.000    RB*  86      2.491+02
RB   86      2.416+03SR   86      0.000    SE   87      6.575+06BR   87      6.575+06
KR   87      6.575+06RB   87      0.000    SR*  87      5.414-03SR   87      0.000
BR   88      7.221+06KR   88      9.238+06RB   88      9.239+06SR   88      0.000
BR   89      4.896+05KR   89      1.140+07RB   89      1.179+07SR   89      1.232+07
Y*   89      2.465+03Y    89      0.000    BR   90      6.465+06KR   90      1.293+07
RB   90      1.471+07SR   90      4.088+05Y*   90      8.523+00Y    90      4.156+05
ZR   90      0.000    KR   91      9.009+06RB*  91      4.994+06RB   91      9.971+06
SR   91      1.498+07Y*   91      8.840+06Y    91      1.555+07ZR   91      0.000
KR   92      7.444+06RB   92      1.510+07SR   92      1.637+07Y    92      1.637+07
ZR   92      0.000    KR   93      3.632+06RB   93      1.227+07SR   93      1.805+07
Y    93      1.833+07ZR   93      1.435+01NB*  93      3.234-01NB   93      0.000
KR   94      2.025+06RB   94      8.646+06SR   94      1.664+07Y    94      1.906+07
ZR   94      0.000    NB*  94      4.169-03NB   94      5.315-08MO   94      0.000
```

Nuclide	Value	Nuclide	Value	Nuclide	Value	Nuclide	Value
KR 95	1.037+07	RB 95	2.074+07	SR 95	2.074+07	Y 95	2.074+07
ZR 95	2.046+07	NB* 95	4.089+05	NB 95	2.005+07	MO 95	0.000
Y 96	9.922+06	ZR 96	0.000	NB 96	2.062+04	MO 96	0.000
KR 97	7.828+06	RB 97	1.566+07	SR 97	1.566+07	Y 97	1.566+07
ZR 97	2.014+07	NB* 97	1.813+07	NB 97	2.014+07	MO 97	0.000
ZR 98	5.324+05	NB* 98	2.673+07	NB 98	1.043+07	MO 98	0.000
ZR 99	1.029+07	NB 99	2.058+07	MO 99	2.059+07	TC* 99	1.894+07
TC 99	6.660+01	RU 99	0.000	NB 100	2.240+07	MO 100	0.000
TC 100	3.292+05	RU 100	0.000	NB 101	9.077+06	MO 101	1.816+07
TC 101	1.816+07	RU 101	0.000	MO 102	1.672+07	TC 102	1.672+07
RU 102	0.000	TC 103	7.061+06	RU 103	1.349+07	RH 103	1.343+07
RH 103	0.000	MO 104	1.196+07	TC 104	1.196+07	RU 104	0.000
RH* 104	8.027+04	RH 104	1.130+06	PD 104	0.000	MO 105	4.874+06
TC 105	7.312+06	RU 105	7.314+06	RH* 105	1.968+06	RH 105	6.682+06
PD 105	0.000	RU 106	2.606+06	RH* 106	4.411+05	RH 106	2.782+06
PD 106	0.000	TC 107	5.441+06	RU 107	5.520+06	RH 107	5.519+06
PD 107	3.595-01	AG 107	0.000	RU 108	3.148+06	RH 108	3.148+06
PD 108	0.000	AG* 108	1.483-06	AG 108	3.723-03	CD 108	0.000
RH 109	2.049+06	PD* 109	2.050+06	PD 109	2.065+06	AG* 109	2.065+06
AG 109	0.000	PD 110	0.000	AG* 110	8.173+02	AG 110	8.293+04
CD 110	0.000	PD* 111	1.619+04	PD 111	3.596+05	AG* 111	3.596+05
AG 111	3.588+05	CD* 111	2.438+00	CD 111	0.000	PD 112	1.921+05
AG 112	1.920+05	CD 112	0.000	PD 113	3.569+04	AG* 113	3.569+04
AG 113	3.212+04	CD* 113	5.915-02	CD 113	0.000	IN 113	0.000
PD 114	3.374+04	AG 114	3.374+04	CD 114	0.000	IN* 114	6.000-05
IN 114	1.117-04	PD 115	2.491+04	AG* 115	6.975+03	AG 115	1.934+04
CD* 115	1.813+03	CD 115	2.468+04	IN* 115	2.468+04	IN 115	0.000
SN 115	0.000	PD 116	1.950+04	AG 116	1.950+04	CD 116	0.000
IN* 116	3.027+03	IN 116	8.539+02	SN 116	0.000	AG 117	3.712+04
CD* 117	1.856+04	IN 117	1.859+04	IN 117	1.859+04	IN 117	2.228+04
SN* 117	4.012-02	SN 117	0.000	CD 118	2.092+04	IN 118	2.092+04
SN 118	0.000	CD* 119	2.519+04	CD 119	2.519+02	IN* 119	2.520+02
IN 119	2.495+04	SN* 119	1.761+03	SN 119	0.000	IN 120	1.046+04
SN 120	0.000	CD 121	4.050+03	IN* 121	4.050+03	IN 121	1.258+04
SN* 121	1.535+02	SN 121	8.656+04	SB 121	0.000	IN 122	1.338+04
SN 122	0.000	SB* 122	3.242+00	SB 122	3.200+02	TE 122	0.000
IN 123	1.882+03	SN* 123	1.665+03	SB 123	2.918+04	SB 123	0.000
TE* 123	5.092-02	TE 123	0.000	SN 124	0.000	SB* 124	1.275+00
SB 124	5.837+01	TE 124	0.000	SN* 125	2.287+04	SN 125	1.225+05
SB 125	2.636+04	TE* 125	5.151+03	TE 125	0.000	SN 126	2.942+00
SB* 126	2.941+00	SB 126	1.440+03	TE 126	0.000	SN 127	1.024+06
SB 127	1.055+06	TE* 127	1.821+05	TE 127	1.023+06	I 127	0.000
SN 128	1.291+06	SB* 128	1.363+05	SB 128	1.551+06	TE 128	0.000
I 128	4.511+03	XE 128	0.000	SB 129	3.374+06	TE* 129	1.214+06
TE 129	3.374+06	I 129	1.378-01	XE 129	0.000	SN 130	6.749+06
SB 130	6.749+06	TE 130	0.000	I 130	7.428+04	XE 130	0.000
SB 131	9.681+06	TE* 131	1.715+06	TE 131	9.683+06	I 131	1.140+07
XE* 131	9.334+04	XE 131	0.000	SB 132	1.390+07	TE 132	1.609+07
I 132	1.609+07	XE 132	0.000	SB 133	1.168+07	TE* 133	1.198+07
TE 133	8.390+06	I 133	2.009+07	XE* 133	4.596+05	XE 133	2.249+07
CS 133	0.000	SB 134	1.084+07	TE 134	2.575+07	I 134	2.550+07
XE 134	0.000	CS* 134	4.138+04	CS 134	6.844+04	BA 134	0.000
SB 135	9.950+06	I 135	9.950+06	I 135	2.178+07	XE* 135	5.974+06
XE 135	5.432+06	CS 135	1.838+00	BA* 135	2.683+00	BA 135	0.000
I 136	9.036+06	XE 136	0.000	CS 136	1.973+05	BA 136	0.000
I 137	1.682+07	XE 137	2.154+07	CS 137	4.864+05	BA* 137	4.475+05
BA 137	0.000	I 138	1.190+07	XE 138	1.914+07	CS 138	2.030+07
BA 138	0.000	I 139	5.931+06	XE 139	1.557+07	CS 139	1.967+07
BA 139	2.007+07	LA 139	0.000	XE 140	1.192+07	CS 140	1.939+07
BA 140	2.001+07	LA 140	2.043+07	CE 140	0.000	XE 141	5.077+06
CS 141	1.600+07	BA 141	1.964+07	LA 141	1.983+07	CE 141	1.982+07
PR 141	0.000	XE 142	1.124+06	CS 142	1.116+07	BA 142	1.806+07
LA 142	1.864+07	CE 142	0.000	PR 142	1.536+05	ND 142	0.000
XE 143	9.819+06	CS 143	1.964+07	BA 143	1.964+07	LA 143	1.964+07
CE 143	1.966+07	PR 143	1.963+07	ND 143	0.000	XE 144	7.933+06
CS 144	1.587+07	BA 144	1.587+07	LA 144	1.587+07	CE 144	1.081+07
PR 144	1.108+07	ND 144	0.000	CE 145	1.266+07	PR 145	1.266+07
ND 145	0.000	CE 146	9.943+06	PR 146	1.014+07	ND 146	0.000
CE 147	2.051+06	PR 147	4.102+06	ND 147	8.236+06	PM 147	1.698+06
SM 147	0.000	CE 148	5.799+06	PR 148	5.799+06	ND 148	0.000
PM* 148	5.609+05	PM 148	1.136+05	SM 148	0.000	ND 149	4.432+06
PM 149	4.856+06	SM 149	0.000	ND 150	0.000	SM 150	0.000
ND 151	2.079+06	SM 151	2.116+06	PM 151	1.644+07	EU 151	0.000
PM 152	7.341+05	SM 152	0.000	EU* 152	5.052+02	EU 152	1.308+01
GD 152	0.000	PM 153	4.098+05	SM 153	1.274+06	EU 153	0.000
PM 154	2.943+05	SM 154	0.000	EU 154	2.721+03	GD 154	0.000
SM 155	3.899+05	EU 155	1.627+04	GD 155	0.000	SM 156	1.819+05
EU 156	5.658+05	GD 156	0.000	EU 157	1.197+05	GD 157	0.000
EU 158	6.089+04	GD 158	0.000	EU 159	1.603+04	GD 159	3.232+04
TB 159	0.000	EU 160	3.786+03	GD 160	0.000	TB 160	4.281+02
DY 160	0.000	GD 161	2.116+04	TB 161	2.131+04	DY 161	0.000
GD 162	8.442+02	TB 162	8.438+02	DY 162	0.000	TB 163	5.695+02
DY 163	0.000	DY 164	0.000	DY* 165	3.942+02	DY 165	4.710+02
HO 165	0.000	DY 166	9.666+01	HO 166	1.077+02	ER* 167	5.460-03

```
FE   59           1.E5CR   51           1.E3
RELEASE OF NOBLE GASES FROM BWR SAMPLE PROBLEM CONTAINMENT SYSTEM (PROBLEM TWO)
BAFFLE LIBRARY
   60.       120      8      7      1      1      1      0      0
     1.0      0.0          0.0          0.0
     1.0      0.0          0.0          0.0
     0.0     30.         90.         210.       330.      600.     3600.     7260.
```

```
       3.0 E-2    1.2E-2    2.0E-3    5.6E-4    4.3E-4    3.5E-4    2.3E-4    2.0E-4
23
       8.0E-5    2.0E-3    1.0E-3    2.3E-4    1.7E-4    1.3E-4    7.3E-5    6.2E-5
34
       8.0E-5    2.0E-3    1.0E-3    2.3E-4    1.7E-4    1.3E-4    7.3E-5    6.2E-5
46
       4.2E-5    5.6E-5    7.2E-5    7.6E-5    7.0E-5    5.7E-5    2.7E-5    2.1E-5
67
       8.0E-5    8.0E-5    8.0E-5    8.0E-5    8.0E-5    8.0E-5    8.0E-5    8.0E-2
78
       8.0E-5    8.0E-5    8.0E-5    8.0E-5    8.0E-5    8.0E-5    8.0E-5    8.0E-2
68
       4.0E-6    4.0E-6    4.0E-6    4.0E-6    4.0E-6    4.0E-6    4.0E-6    4.0E-6
1
       7.0E-4    5.6E-4    4.1E-4    3.3E-4    3.0E-4    2.6E-4    1.8E-4    1.5E-4
CLOUD GAMMA DOSE CALCULATION FOR BWR SAMPLE PROBLEM - 100 METERS
EXDOSE LIBRARY
*****   THE FOLLOWING CARDS WERE PREPARED BY THE COMPUTER FOR THE FIRST  *****
*****   PROBLEM.   *****
CLOUD GAMMA DOSE CALCULATION FOR BWR SAMPLE PROBLEM - 100 METERS
EXDOSE LIBRARY
 9  1FFTFF      1.00     100.00     60.00       .00      2.00
      2.180     -.000     -.000     -.000     -.000                  \
36
  .1000+02   .4086+03   .8071+03   .1206+04   .1604+04   .2003+04   .2401+04   .2800+04
  .3829+04   .4857+04   .5886+04   .6914+04   .7943+04   .8971+04   .1000+05   .1103+05
  .1206+05   .1309+05   .1411+05   .1514+05   .1617+05   .1720+05   .1931+05   .2143+05
  .2354+05   .2566+05   .2777+05   .2989+05   .3200+05   .3629+05   .4057+05   .4486+05
  .4914+05   .5343+05   .5771+05   .6200+05
  .5585-14   .3068-11   .5194-11   .9686-11   .1791-10   .3165-10   .5123-10   .7130-10
  .9743-10   .1285-09   .1567-09   .1911-09
  .7836-14   .3488-11   .5733-11   .1056-10   .1931-10   .3392-10   .5497-10   .7652-10
  .1045-09   .1376-09   .1677-09   .2044-09
  .1096-13   .3964-11   .6327-11   .1150-10   .2082-10   .3638-10   .5901-10   .8219-10
  .1122-09   .1476-09   .1796-09   .2188-09
  .1530-13   .4502-11   .6981-11   .1253-10   .2245-10   .3902-10   .6340-10   .8835-10
  .1206-09   .1584-09   .1926-09   .2344-09
  .2126-13   .5110-11   .7699-11   .1365-10   .2420-10   .4188-10   .6815-10   .9503-10
  .1297-09   .1701-09   .2067-09   .2514-09
  .2945-13   .5795-11   .8487-11   .1487-10   .2609-10   .4495-10   .7329-10   .1023-09
  .1395-09   .1829-09   .2219-09   .2698-09
  .4060-13   .6565-11   .9350-11   .1618-10   .2812-10   .4827-10   .7887-10   .1102-09
  .1502-09   .1966-09   .2385-09   .2898-09
  .5571-13   .7429-11   .1029-10   .1759-10   .3030-10   .5163-10   .8489-10   .1187-09
  .1618-09   .2116-09   .2564-09   .3114-09
  .1226-12   .1014-10   .1312-10   .2176-10   .3664-10   .6228-10   .1027-09   .1440-09
  .1962-09   .2559-09   .3096-09   .3755-09
  .2563-12   .1364-10   .1655-10   .2668-10   .4403-10   .7458-10   .1239-09   .1742-09
  .2374-09   .3091-09   .3733-09   .4524-09
  .4993-12   .1790-10   .2050-10   .3223-10   .5230-10   .8854-10   .1483-09   .2092-09
  .2850-09   .3704-09   .4467-09   .5411-09
  .8838-12   .2268-10   .2474-10   .3806-10   .6093-10   .1033-09   .1744-09   .2466-09
  .3361-09   .4362-09   .5255-09   .6363-09
  .1378-11   .2733-10   .2871-10   .4344-10   .6889-10   .1171-09   .1989-09   .2819-09
  .3843-09   .4983-09   .5999-09   .7262-09
  .1833-11   .3084-10   .3163-10   .4735-10   .7466-10   .1272-09   .2170-09   .3081-09
  .4200-09   .5442-09   .6550-09   .7927-09
  .2025-11   .3216-10   .3270-10   .4879-10   .7679-10   .1309-09   .2237-09   .3178-09
  .4332-09   .5613-09   .6754-09   .8174-09
  .1841-11   .3084-10   .3161-10   .4733-10   .7464-10   .1272-09   .2169-09   .3080-09
  .4199-09   .5441-09   .6548-09   .7924-09
  .1388-11   .2732-10   .2868-10   .4340-10   .6883-10   .1170-09   .1987-09   .2817-09
  .3840-09   .4979-09   .5995-09   .7256-09
  .8910-12   .2265-10   .2470-10   .3799-10   .6084-10   .1032-09   .1741-09   .2463-09
  .5356-09   .4356-09   .5248-09   .6354-09
  .5029-12   .1786-10   .2045-10   .3215-10   .5218-10   .8835-10   .1480-09   .2087-09
  .2844-09   .3696-09   .4458-09   .5400-09
  .2577-12   .1359-10   .1649-10   .2660-10   .4390-10   .7437-10   .1236-09   .1738-09
  .2368-09   .3082-09   .3723-09   .4512-09
  .1229-12   .1010-10   .1307-10   .2168-10   .3650-10   .6206-10   .1023-09   .1435-09
  .1955-09   .2551-09   .3085-09   .3743-09
  .5568-13   .7386-11   .1024-10   .1751-10   .3017-10   .5163-10   .8456-10   .1182-09
  .1612-09   .2108-09   .2554-09   .3102-09
  .9843-14   .3786-11   .6104-11   .1115-10   .2025-10   .3545-10   .5749-10   .8005-10
  .1093-09   .1458-09   .1751-09   .2134-09
  .1599-14   .1911-11   .3608-11   .7051-11   .1358-10   .2461-10   .3979-10   .5534-10
  .7579-10   .1005-09   .1230-09   .1505-09
  .2479-15   .9617-12   .2132-11   .4460-11   .9148-11   .1730-10   .2809-10   .3914-10
  .5382-10   .7194-10   .8866-10   .1090-09
  .3747-16   .4848-12   .1263-11   .2827-11   .6187-11   .1230-10   .2018-10   .2826-10
  .3907-10   .5272-10   .6544-10   .8097-10
  .5582-17   .2453-12   .7507-12   .1796-11   .4201-11   .8823-11   .1472-10   .2077-10
  .2891-10   .3941-10   .4930-10   .6142-10
  .8241-18   .1246-12   .4477-12   .1144-11   .2861-11   .6374-11   .1087-10   .1549-10
  .2174-10   .2996-10   .3780-10   .4745-10
  .1210-18   .6353-13   .2678-12   .7300-12   .1954-11   .4631-11   .8105-11   .1170-10
  .1657-10   .2310-10   .2941-10   .3722-10
  .2457-20   .1640-13   .9518-13   .2951-12   .9073-12   .2455-11   .4581-11   .6832-11
  .9887-11   .1414-10   .1835-10   .2365-10
  .4963-22   .4287-14   .3409-13   .1199-12   .4237-12   .1317-11   .2650-11   .4113-11
  .6108-11   .8973-11   .1189-10   .1563-10
  .1004-23   .1132-14   .1229-13   .4888-13   .1987-12   .7122-12   .1559-11   .2532-11
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.3874-11   .5856-11   .7930-11   .1066-10
.2034-25   .3016-15   .4450-14   .1999-13   .9341-13   .3873-12   .9285-12   .1586-11
.2507-11   .3905-11   .5409-11   .7438-11
.4134-27   .8090-16   .1618-14   .8193-14   .4403-13   .2116-12   .5580-12   .1006-11
.1648-11   .2649-11   .3757-11   .5290-11
.6448-29   .2182-16   .5899-15   .3364-14   .2078-13   .1159-12   .3378-12   .6451-12
.1098-11   .1822-11   .2647-11   .3820-11
.0000      .5914-17   .2156-15   .1383-14   .9824-14   .6365-13   .2055-12   .4169-12
.7386-12   .1268-11   .1887-11   .2794-11
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***** DATA CARDS FOR THE REMAINING DISTANCES ARE SIMILAR TO THE CARDS *****
***** FOR 100 METERS AND WILL BE OMITTED. *****

APPENDIX H

SOLICITED COMMENTS
of
OTHER INVESTIGATORS

A brief summary of the calculational method and questionaire, designed
to elicit comments and suggestions regarding dose calculations from a
cloud of airborne radionuclides, were distributed to forty investigators.
The individual questions and the summarized responses received from
thirteen of these, are:

Question 1 "Are the proposed methods of specifying inventory
            adequate for present and future problems as you
            now see them?"

There were three "No" answers to Question 1. Two of these responses were
concerned about the need for calculating inventories for different types
of reactors and different types of fuel both of which can be handled by
separate specialized codes and used as input source inventories submitted
on punched cards. The third "No" response argued for calculating the
inventory in units of Mev/sec to avoid ambiguity, however, the total energy
of gamma radiation emitted per unit time is not a measure of effective
gamma energy per unit time, i.e. one gamma ray per disintegration of 1 Mev
does not have the same effect as 4 gamma rays per disintegration each with
an energy of 0.25 Mev.

Question 2 "Does the proposed containment model allow
            adequate description of containment system
            response for reactor designs and accidents
            of interest to you?"

There were 6 "No" responses. Several individuals think the accounting of
containment system response should be more detailed. Specifically, a more
detailed accounting was suggested for the "release from fuel" and for the

effect of recirculating filters. The concentration remaining airborne
within each containment volume was felt to be a desirable output. One
respondent felt there was a need to include the effect of sprays and
reentrainment from sump pools of airborne radionuclides.

At this time more detail cannot be included and still have a usable-practical
code. But as the code is debugged and running time is reduced it may be
feasible to provide a more complete and comprehensive method for describing
containment system response.

Question 3   "Is the proposed method of describing leakage rates
             adequate and acceptable?"

There was only one "No" response. All other respondents felt the proposed
method of describing leakage rates adequate.

Question 4   "Have you made any estimates or parametric studies
             of the impact on environmental consequences of
             accounting for the effects of multiple barrier
             containment systems? (i.e. reduction of release
             because of radioactive decay during travel of
             an aerosol through successive containers.)"

Eight investigators answered "Yes". Three of these had conducted studies
but their work was unpublished and considered proprietary. Three others
had documented their studies of the impact on environmental consequences
of accounting for the effects of multiple barrier systems. These documents
are:

"Consolidated Edison Unit No. 4 PSAR" and Public Service - Newbold Island
PSAR both reported by John Szeligowski

"Site and Safeguard Requirements for Radio iodine Control" MDAC PAPER
WD 1106 August 1969 by C. A. Willis

"Estimation of Radiation Doses Following A Reactor Accident" ORNL 4086
February 1968 by F. T. Binford, J. Barish, and F. B. Kam.

Question 5 "Do you feel that the model should account for
            removal mechanisms in containment, i.e. settling,
            plate out, fog spray, rain out?"

All respondents felt that removal mechanisms in containment should be
accounted for. Some were emphatic in stating that the effect cannot
be ignored.

Question 6 "Is the proposed generalized method of accounting
            for removal mechanisms adequate?"

There were two "No" responses. The effect of recirculation was the concern
of these investigators as well as some of those who answered "Yes".
Recirculating cleanup can be accounted for in the code by assigning a
single removal rate factor which is applicable to each container. To
input it as a separate factor would probably mean interfacing sophisticated
aerosol behavior codes with the proposed code. This may be easily done if
the more sophisticated code is designed to provide the input this code
needs, i.e., the removal rate at specified times.

Question 7 "Have you made any estimates or parametric studies
            of the impact on environmental consequences of
            accounting for removal mechanisms within containment
            barriers?"

Eleven investigators have made estimates of the impact on environmental
consequences of accounting for removal mechanisms within containment
barriers. Several document references supplied by respondents are:

"Estimation of Radiation Doses Following A Reactor Accident" ORNL 4086
February, 1968 by F. T. Binford, J. Barish, and F. B. Kam.

"Accident Confinement System of the Savannah River Plant Reactor" DP 1071
August, 1966 by W. S. Durant, R. C. Milham, Dr. Muhlbair, and others

"The SRL Meteorological Program and Offsite Dose Calculations" DP 1163
R. E. Cooper and B. V. Rusche, September 1968

Question 8   "Three dispersion models are included in the
             code. (See description of code) Do these 3
             models provide sufficient flexibility for your
             problems?"

There were seven "No" responses and it was correctly pointed out that the
code incorporates only one model with several different means of determining
its parameters. Most comments were related to the need for handling volume
source, building wake, fumigation condition, variable wind and non uniform
distribution effects.

Provision has been made for direct input of selected values of the parameters
σ_y and σ_z for desired distances and the code interpolates. Building wake
effects will be incorporated in the code. Incorporating the remaining effects
is under study.

Question 9   "Please comment on the importance (or lack of
             importance) of cloud depletion effects on cloud
             dose and need for inclusion of some way of
             accounting for the effect."

Opinion was about evenly divided between those that felt cloud depletion
effects were important and those that felt they were not.

Question 10   "Should a one group or multigroup calculation
              be performed? Why? How many groups?"

Only one investigator felt that one energy group was sufficient for the
calculation. Most investigators felt 5-10 groups would be desirable for
their calculations.

Since "interaction cross sections and buildup factors are functions of gamma
ray energy", it seems desirable to use multigroup calculations for some
problems. The code now uses a 12 group calculation.

Question 11   "Taylor's buildup formula is being used. Is this
              a suitable choice?"

Taylor's buildup formula appears to be OK for higher energies but is
inadequate for low energy buildup. As a result of the responses to the
questionaire Berger's values are now being used (see Appendix A).

Question 12   "Proposed limits for the space integration are:
              y direction $\pm$ 3 σ_y
              z direction $\pm$ 3 σ_z
              x direction $\pm$ 420 meters from exposure point except
                        if exposure point is less than 420 meters
                        from the release point in which case the
                        x integration extends from the release point
                        to 420 meters downwind of the exposure point.

              Are these limits acceptable? (Please comment on this
              choice of limits.)"

Only two investigators felt these limits were inadequate. Actual
integration limits are still to be studied.

Question 13   "Close to a reactor containment building the external
              whole body dose from contained fission products may
              overshadow the cloud dose. Do you make calculations
              at distances this close? By separate code? Point
              or extended source? Include shielding effects of
              containment and/or air? Buildup?"

Two investigators reported that they do not make this calculation, two
calculate it with their cloud dose code in use and the remaining nine
calculate it using a separate code or by hand.

Question 14   "The ICRP suggested method (page 22 of ICRP
               Publication 2) of calculating maximum permissible
               concentration (mpc's) in air includes β radiation
               exposure as part of the external whole body exposure
               if the β radiation has an end point energy greater
               than 0.1 Mev."

        14a.   "Do you do this in your cloud dose calculations?"
        14b.   "Do you include exposure from Bremsstrahlung as
               part of the external whole body exposure for β
               emitters such as ^{85}Kr?"
        14c.   "Do you calculate the β dose as part of cloud
               dose calculations? (how do you calculate or
               suggest calculating the β dose? Semi-infinite
               cloud? Dose versus depth? Skin dose? etc.)"

The consensus is that the dose from beta particles can generally be ignored
but in any case where it is required the infinite cloud assumption can be
used to provide a ready answer. Only one investigator reported accounting
for Bremsstrahlung.

Question 15   Deleted prior to distribution.

Question 16   "We propose to follow ICRP Committee II recommendations
               for making internal dose calculations. Do you have
               other methods that you use? (Please describe them and
               any observed differences in results between the methods.)"

All reported using ICRP Committee II recommendations. A few have revised
biological parameters of ICRP Publication 2 according to ICRP publication
10 and only one investigator uses the new ICRP lung model.

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