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CINDER-7: An Interim Report for Users

by

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ABSTRACT

CINDER, a general nuclide depletion and fission-product code, has been repro-
grammed to utilize improved data handling subroutines, improve roundoff control, and
incorporate new calculational features such as a gamma spectra subroutine. The new
coding uses a dynamic storage, variable dimensioning, and a free-form input format.
The format is not compatible with earlier versions of CINDER. This report describes
the format, its bases, and some of its extended features.

I. INTRODUCTION

CINDER[1,2] is a generalized code that computes
the temporal composition of coupled nuclides in a
time-dependent flux environment. The nuclides can
be coupled in any sequential mixture of radioactive
decay types and neutron or other particle reactions.
A neutron flux is normally used. The flux or fis-
sion power, nuclide parameters, couplings which form
nuclide chains, etc., can be specified by the user.
The code also computes the concentration of coupled
fission products generated by fission in fuel chains.
Various quantities related to the nuclide densities,
such as absorption build-up, fuel depletion, decay
heating, activity and decay spectra are also com-
puted.

Versions of the basic code or its derivatives
have been in use for more than 10 years. Since the
code's inception, the quantity of basic nuclide da-
ta has greatly increased; for example, the ENDF/B-IV
files contain neutron absorption and decay parameter
data for 824 nuclides classed as fission products.
Many quantities which depend on the time-dependent
concentration of nuclides can now be accurately com-
puted, and a nuclide code is needed that is capable
of utilizing the expanding data base as well as be-
ing suitable for the general user.

CINDER-7 is a reprogrammed and extended version
of the basic code and takes advantage of new coding
techniques which permit variable dimensioning, dy-
namic storage, and free-form input formats. It is
capable of treating a wider variety of coupled nu-
clide problems for a wider class of users; data in-
put is greatly simplified and less subject to error
in input preparation. Output options have been ex-
tended.

This report is essentially an interim manual
for users and assumes familiarity with earlier ver-
sions or with Refs. 1 and 2. Although the basic
concepts of CINDER remain unchanged, all input for-
mats have necessarily changed because of the new
coding bases outlined in the next section.

A large part of six of the seven subroutines
associated entirely with CINDER-7 was coded during
1970-72 while two of the authors were employed at
the Bettis Atomic Power Laboratory. During 1974-75,
the initial plans were extended and completed. In
addition, the original dependence of CINDER-7 on the
Bettis Environmental Package for CDC computers[3] was
eliminated (see App. A). The version of CINDER dis-
cussed in this report is now free standing and in
essentially standard FORTRAN. The only portion that
is nonstandard FORTRAN is already available on most
major computers and is expected to become a FORTRAN
standard in 197..

Additional code changes are in progress to facilitate use of the expanded decay and absorption data in ENDF/B-IV. A detailed report will be prepared for the final code and data when these are fully operational. However, CINDER-7 is the current validated version in use at LASL and is expected to remain in use for an indefinite period. It is more than sufficient for most users and is a considerable improvement over versions of CINDER in general circulation.

All capabilities of earlier versions are retained and extended. Some of the new features are described briefly in this report, and the general capabilities, such as a gamma spectra routine are described in Sec. III. Actual improvements in coding, such as an improved roundoff control, are not discussed in detail here.

Because the input format of CINDER-7 is incompatible with prior versions, the primary purpose of this report is to describe the new format for LASL users already acquainted with the basic code.

II. SUMMARY OF CODING BASES AND FEATURES

CINDER-7 exploits and extends the FORTRAN-IV capability of variable array dimensions. During the execution of a code, all variables are not usually of a maximum size at the same time. Several subroutines, originally developed at the Bettis Atomic Power Laboratory for other codes, have been modified and incorporated into CINDER. These permit a trade off of parts of a single, large array stored in blank common where they can be referenced by various FORTRAN subprograms. The dynamic storage is described in Ref. 3, and the original routines are available from the Argonne Code Center. Of the small number of these subroutines actually used, most have been modified to remove much of the machine dependence as well as interdependence on the large number of subroutines in the total Bettis FORTRAN Environmental Package. The names of the subroutines, identified in App. B, have not been changed, and Ref. 3 should be consulted for their basic functions. These modified subroutines account for less than 0.1% of the non-CDC versions of CINDER and only ~ 25% of the CDC version (App. A).

The recoded CINDER-7 subroutines fully exploit the variable array dimensions permitted by FORTRAN-IV. The recoding has reduced required input data

and, in addition to providing the dynamic storage, it has reduced the storage required by earlier versions of CINDER. Several quantities previously required as input are now computed directly from other input data. The flexible storage has permitted the addition of new options. All data are now input using a generalized format-free conversion routine which greatly simplifies input preparation. This, combined with other coding, also reduces input by eliminating the need to input classes of data which are unused in a particular case and some parameters which are zero or unity for any particular nuclide.

A. Summary of New Features

1. All data are input using a card index which identifies the nuclide, data type(s), and chain. The card order is *not* important and only those cards that are needed from case to case need be supplied. If there are duplicate card indices, then the data following the last duplicated index is used. Thus the user can readily replace or modify library data without, e.g., recourse to UPDATE.

2. Variable dimensions, dynamic storage, free input format, etc., are used as already described.

3. Because of the flexible storage, the number of input quantities is limited only by the size of the computer's memory. Flexible input includes the number of chains, nuclides per chain, time increments, and time-dependent quantities such as group fluxes, factors applied to fluxes and cross sections, and other quantities such as the number of decay energy groups, and nuclides in power that yield products on fission.

4. The actual storage required for input data and computations is calculated and printed on the output.

5. Nuclide cross sections can be changed individually in any neutron group at any time step by specification of factors on one card series. Other card series permit nonunity branching factors for decay or absorption per nuclide, summary calculations of the aggregate resonance integrals, and also an effective 2200 m/s cross section.

6. Extra summary listings include an option to selectively group nuclides, such as noble gases or major absorbers; the output print for these groupings also includes the aggregate resonance integral.

7. The irradiation history can be specified per time step in terms of:

(a) the group fluxes;

(b) group-4 flux and ratios of higher energy group fluxes to the fourth group, or

(c) the power and flux ratios.

The fluxes can also be modified as noted under 3, above.

8. Decay energies weighted by intensities can be input in up to 100 groups of arbitrary width, or energies and intensities can be stored by nuclide and the user can specify the desired group structure. Currently, stored ENDF/B-IV nuclide data are appropriate for photoneutron production from ^{9}Be and ^{2}H, and the decay information printed includes gammas per fission and per second for each nuclide and decay group.

9. The number and type of chains and the number of nuclides in each chain are now computed automatically.

10. Several new print options per time step and time-step summaries are available along with new punch options, as will be noted in Sec. III.

11. Decay heating (MeV/s and MeV/fission) is calculated per nuclide and decay group per time step; aggregate values, summed over all nuclides per decay group, are also printed.

12. The data in each card field is interpreted from context, as described in Sec. III.

As in earlier CINDER versions, CINDER-7 permits the user to adjust roundoff control parameters to match his computer word length and to specify other quantities such as the total energy per fission.

This version of CINDER has been tested only on CDC computers, but the subroutines have been further modified to work on all major computers without loss of any feature (see App. A).

III. INPUT FORMAT FOR CINDER-7

A. User Notes

1. The following pages specify 21 card, or card image, types each beginning with an index number which determines the type of data it carries. Cards are of the form

index, A, X_1, X_2, N_1, ...,

where X denotes a real number, N denotes integers and, when present, A denotes an alphanumeric. This notation is used for clarity; the code actually interprets the field or data type from context, as explained later in this section. A *field* consists of all entries between commas. (See Sec. III-C for a sample input listing.)

2. Any card can be continued by placing a plus (+) sign as the first nonblank character.

3. In addition to the specified card types, a title card can be entered. This begins with an equals (=) sign as the first nonblank character. The last such card found is used as the title.

4. Comment cards can be interspersed with data as desired. These begin with an asterisk (*) as the first nonblank character, and are printed in a card listing at the beginning of each output print but are not otherwise used.

5. The last card of a case has a period (.) or negative integer as the first nonblank character.

6. Except for the period card, the cards can be entered in random order. Cards following the period card are ignored.

7. If two or more cards have identical indices, the data on the last card found are used. This is also a redundancy check; data on duplicate card indices (including field continuations) are stored only for the last duplicate. Similarly, data on a card can be deleted by inserting a card having only the card index.

8. Most of the data cards described in this section are optional, but at least one of the following card types and the period card should appear in all cases: 1000001, 20TTTT, 1TCCCNN, 2TCCCNN.

9. Blank cards are ignored.

10. Any field can be repeated by having as its first nonblank character the slash (/), a repeat integer, and (/) preceding the field, i.e., /N/.10+10 repeats the .10+10 field N times.

11. The cards are scanned from left to right and each field is examined for the characters +, −, ., 0, 1, 2, 3, 4, 5, 6, 7, 8, 9, and blank. If any character is *not* one of these, the field is assumed to be nonnumeric, or Hollerith.

12. The card index, or first character, can begin in any column. Leading blanks are ignored. All blanks are ignored in numeric fields; leading and trailing blanks are ignored in Hollerith fields. However, Hollerith fields may be specified by enclosure in parentheses, ( ), in which case all blanks are retained. Any character except the dollar ($) sign may be included inside the parentheses. In numeric fields, a field containing only blanks is con-

verted to integer-zero. A field containing only a decimal point is converted as a real zero. In real fields the plus (+) sign is optional and discarded. The exponent is also optional if the decimal point is included; the decimal point is optional if the exponent is included and in such event the decimal is assumed to be to the left of the first nonblank character. For real fields, the following are all equivalent to 10^{-3}: 0.01-1, 01-1, .001. Leading zeros in an exponent or integer field are ignored and, if a decimal point is embedded in a real field, leading zeros are ignored. The E in real fields, e.g., 0.01E-1, should *not* be used.

13. A card is assumed to have 80 columns, but can be delimited by using the dollar ($) sign. This also eliminates the data on any continuation card(s).

14. The following limits are arbitrary and can readily be changed:

Decay energies can be input in up to 101 groups; output prints label the first group as β decay and the remainder as γ decay. The actual group structure is arbitrary but is fixed by the user's library. A second decay routine permits γ "line" data (energies and intensities) to be stored and the group structure is specified for each run by an input card which lists the desired decay energy bounds.

Up to nine yield sets can be specified, but these can be used for any fissionable nuclide any number of times, for example, the user can specify that ^{238}U yields be used for ^{238}U, ^{240}Pu, etc.

A sample input for a simple problem is provided in Sec. III-C.

Appendix A describes a slight variation in input required for a version which is essentially machine-independent. The following card image description applies to CDC computers.

B. Input Card Types

The card types (card images) described in this section need not be entered in order. For the convenience of the user, card descriptions are listed in the following order (any card described here as Card 2 through Card 21 need not be present if it has zero data values).

ORDER	CARD INDEX[*]
1	100001
2	20TTTT
3	1TCCCNN
4	2TCCCNN
5	3TCCCNN
6	4CCCNN
7	5TCCCNN
8	7TCCCNN
9	100008
10	100009
11	8CCCFS
12	9TCCCNN
13	901
14	1JJSS
15	100002
16	2TTTT
17	1
18	2
19	3
20	4
21	5

1. **Control Card. Card 10001**

100001,X_1,X_2,N_1,X_3,X_4,X_5,N_2

X_1 = Group-4 spectrum or shielding factor. X_1 multiplies all group-4 cross sections except those given individual factors on card series 5TCCCNN. If this factor is not the same for all time steps, specify 0.0 and input time-dependent values on time card series 20TTTT.

X_2 = Joules/fission (K, Eq. 3, Ref. 1) (for reference, 200 MeV = 0.32042 x 10^{-10}).

N_1 = 1 -- print: Contribution to (nuclide X) from (nuclide Y) and preceding nuclides in chain (Z) set to 0 (see Sec. 2V, Ref. 1).

= 0: this flag is not printed.

---

[*] T denotes a depletion or fuel chain (T = 1) or a fission product chain (T = 2).

TTTT denotes a time-step number.

CCC denotes a chain sequence number beginning with 1 for T = 1, and T = 2 type chains.

NN denotes the nuclide sequence in each chain.

F, S, JJ, and SS are defined on the card type descriptions.

Unless otherwise specified, TCCCNN and the nuclide ID are used throughout the code as a cross reference to the input data library.

X_3 = Coupling criterion. Check against near zero divisors from β and $\Delta\beta$. If zero, blank, or negative, 10^{-12} is used (see criterion 14, Ref. 1). Here $\beta \equiv \lambda + \sum_{g} \sigma\phi$.[*]

X_4 = Roundoff criterion. If zero, blank, or negative, 10^{-12} is used (see criterion 13, Ref. 1).[*]

X_5 = If greater than 0.0, then print option 3 on the time card series, 20TTTT, will print only data for those nuclides having decay energies (MeV/s) exceeding X_5 percent of the total decay energy. If X_5 = 0.0, only the (nonredundant) nuclides are printed. If a small positive value is given, only radioactive nuclide data are printed, etc.

N_2 = 0,ϕ_4: Specified on series 20TTTT (flux groups 1 to 3 are ratios to group 4).

    1: Power is specified instead of ϕ_4.

    2: ϕ's are specified instead of ratios.

2. Time Card Series. Cards 20TTTT

20TTTT,X_1,X_2,X_3,X_4,N_1,N_2,X_5,X_6

TTTT = The time-step number . If the same values are to be used for a series of time steps, TTTT is the number of the last time step in the series (e.g., a single card, 20100, would use the X values for all time steps from the last specified time card through time step 100).

X_1 = Time-step length in hours.

X_2 = ϕ_1/ϕ_4 ⎫
X_3 = ϕ_2/ϕ_4 ⎬ Fast-to-group-4 flux ratios (or ϕ's if N_2 = 2 on card 100001).
X_4 = ϕ_3/ϕ_4 ⎭

N_1 = 0: Omit the fission product print this time step (TS).

    1: Full print this TS.

    2: Delete all print this TS.

    3: Print selected decay energies (listing of nuclides in sum having a total MeV/s exceeding the percent, X_5, on control card).

    4: Print all depletion chain information and all fission product information except the macroscopic absorption data.

    5: Print information for all depletion chain nuclides and only those fission product nuclides in sum (N1 = 1 on card 1TCCCNN).

---
[*] The coupling criteria are flagged, but the code now automatically adjusts parameters to permit computations to continue. Here, β denotes the total destruction rate per nuclide, including any shielding and flux adjustment factors.

N_2 = 0: (Option to be added later.)

X_5 = Either ϕ_4 (n/cm^2-s) or power density (W/cm^3), depending on option specified on control card 100001.

X_6 = Spectrum or shielding factor for each time step. X_6 multiplies all group-4 cross sections except those given individual factors on card series 5TCCCNN. If X_1 on the control card (100001) is not equal to 0.0, X_6 is ignored and therefore may be omitted.

3. Number Density and Nuclide Library, Card Series 1. Cards 1TCCCNN

1TCCCNN,A_1,X_1,X_2,N_1,N_2

T   = 1: Denotes a depletion or fuel chain.

T   = 2: Denotes a fission product chain.

CCC = The nuclide chain sequence.

NN = The nuclide sequence in the chain.

A_1 = Nuclide identification -- alphabetic and/or numeric information such as U235, U238, etc.

X_1 = Initial number density (atoms/barn-cm).

X_2 = λ: Radioactive decay constant (s^{-1}).

N_1 = 1: Include this nuclide in total absorption, barns/fission, disintegration rate summations, etc.

    0: Omit this nuclide in summations.

N_2 = Type of coupling to next nuclide.

    1: Last nuclide in chain (no coupling).

    2: Radioactive decay ($\gamma = \lambda$).

    3: Total absorption cross section ($\gamma = A^a$).[†]

    4: Capture cross section ($\gamma = A^a - A^f$).

4. Microscopic Absorption Cross Section Library, Card Series 2. Cards 2TCCCNN

2TCCCNN,X_1,X_2,X_3,X_4

X_1 ⎫
X_2 ⎬ Total nuclide microscopic absorption cross
X_3 ⎪ sections (barns) in each energy group.[‡]
X_4 ⎭

[†] $A^a \equiv \sum_j g_j \sigma_{a_j} \phi_j f_j$ where j = neutron group, g_j = time-dependent *shielding* or spectrum factor, f_j = time-dependent flux factor, f and g are 1.0 unless specified otherwise on cards 2TTTT or 5TCCCNN. The g_j factors are also applied to the fission cross sections to compute, e.g., A^f.

Card series 7TCCCNN is used to insert all nonunity branching factors which are applied to the γ defined by the expressions under N_2.

[‡] CINDER-7 presently permits one to four neutron groups. This is being changed to permit an arbitrary number of groups.

5. **Microscopic Fission Cross Section Library, Card Series 3. Cards 3TCCCNN**

$3TCCCNN, X_1, X_2, X_3, X_4$

$$\left.\begin{array}{l} X_1 \\ X_2 \\ X_3 \\ X_4 \end{array}\right\} = \text{Total nuclide microscopic fission cross sections (barns) in each energy group.*}$$

6. **Initial Fission and Number Density Card Series.† (Optional) Cards 4CCCNN**

$4CCCNN, X_1, X_2$

X_1 = Initial fission density (fission/barn-cm).

X_2 = Initial number density to be used in "fuel-fraction" calculation (atoms/barn-cm). This need not be the same value as X_1 in card series 1TCCCNN.

Supply this card only if card 100002 is present, and then only for the fissile nuclides included in the power calculation. The T is omitted in the card number because this series is used only for depletion chain nuclides.

If there are no fissionable nuclides included in power, supply some value for X_1 to prevent a zero in the barns/fission calculations.

7. **Shielding Factor Card Series. (Optional) Cards 5TCCCNN**

$5TCCCNN, X_1, X_2, X_3, \ldots, X_{NT}$

T = 1: Depletion chain; group 1 shielding factor.

   2: Fission product chain; group 1 shielding factor.

   3: Depletion chain; group 2 shielding factor.

   4: Fission product chain; group 2 shielding factor.

   5: Depletion chain; group 3 shielding factor.

   6: Fission product chain; group 3 shielding factor.

   7: Depletion chain; group 4 shielding factor.

   8: Fission product chain; group 4 shielding factor.

CCC = The nuclide chain.

---

*All nuclides, including fission products, can have a cross section specified here. Card 100008 specifies which nuclides are in power and contribute to fission product yields; in effect, card 10008 determines whether the cross sections specified here are actual fission cross sections. Otherwise, this card is used to provide an exact representation of (n,γ) activations where a single branching fraction is adequate (see series 7TCCCNN).

†This special purpose card should not be confused with card 1TCCCNN.

NN = The nuclide in the chain.

X_{NT} = Shielding factor multiplying cross sections of this nuclide for time step or expansion NT.

If more than one card is needed for a nuclide, replace the card number by + on subsequent cards. If N consecutive time-step factors are the same, $/N/X_{NT}$ may be used.

The group-4 factor overrides any specified value of X_6 on time series cards, 20TTTT, and X_1 on the control card, 100001.

8. **Branching Factor Card Series. (Optional) Cards 7TCCCNN**

$7TCCCNN, X$

X = α: The fraction of total specified coupling (decay or absorption) to a daughter nuclide (if card is not supplied, X = 1.0).

In decay heat calculations where even the shortest lived nuclides are important and where multiple branches are the rule rather than the exception, a convenient new feature is the ability to specify this branching factor, α. α multiplies the coupling factor $(\lambda$ or $\sum_g \sigma\phi)$, producing the direct daughter nuclide in a linear chain (depletion or fission product). It does *not* directly affect any other computation, such as λN. Note that in the case of neutron capture, unless all group cross sections divide in the same proportion, a single α factor cannot be used. However, the practice of using fictitious fission cross sections on card series 3TCCCNN and type 4 coupling is precise and convenient.

Example: 7200203, .942+0

The 0.942 factor is applied to λ, A^a, or $(A^a - A^f)$ depending on the type of coupling specified for the third nuclide of fission product chain two.

The next three card descriptions (100008, 100009, 8CCCFS) are supplied if a nuclide is to be included in the power calculation.

9. **Depletion Chain Nuclides in Power. Card 10008**

$100008, N_1, N_2, N_3, \ldots, N_j$

N_j = The nuclide to be included in the power calculation (Eq. 3, Ref. 1). N_j is a five-digit fixed point number corresponding to CCCNN on the depletion chain Number Density and Nuclide Library Card Series, 1TCCCNN.

NOTE: Yield fractions, even if zero, must be supplied for all fissile nuclides included in the power calculation, as noted for series 100009 and 8CCCFS.

Example: 100008,00101,00102,00203,00204.

This example specifies that nuclides in power are the first nuclide in the first chain, the second nuclide in the first chain, the third nuclide in the second chain, etc.

10. **Yield Deck Assignments. Card 100009**

$100009, N_1, N_2, N_3, \ldots, N_j$

N_j = The F on card series 8CCCFS. Tells which yield deck identified by F corresponds to which nuclide (N_j on card 100008).

Example: 100009,1,3,3,2 (refer to example on card 100008).

The yields in deck 1 are to be used for nuclide 1, depletion chain 1. The yields in deck 3 are to be used for nuclide 2, depletion chain 1. The yields in deck 3 are to be used for nuclide 3, depletion chain 2. The yields in deck 2 are to be used for nuclide 4, depletion chain 2.

11. **Yield Card Series. Cards 8CCCFS**

$8CCCFS, Y_1, Y_2, Y_3, \ldots, Y_{NN}$

CCC = The fission product chain number

F = The depletion nuclide in power (each F actually designates a deck). F = 1 to 9.

S = Sequence number (if more than one card is needed for a chain).

Y_{NN} = The yield fraction (*not* percent) for nuclide NN in fission product chain CCC.

Example: 800111, .1-3, .2-3, 8-3
     800211, .4-3, .5-3
      .
      .
      .
     805311, .6-3, .7-3
     800121, .8-3, /2/ .9-3
      .
      .
      .
     805321, .1-2, .2-2
     800131, /3/ 0.0
     800231, .3-2, .4-2
     800331, /3/ .5-2, .6-2
      .
      .
      .
     802731, /10/ .7-2, .8-2
      .
      .
      .
     805331, /2/ .9-2

.1-3 is the yield for nuclide 1, fission product chain 1

.6-2 is the yield for nuclide 4, fission product chain 3

.7-2 is the yield for nuclide 10, fission product chain 27

.2-2 is the yield for nuclide 2, fission product chain 53.

NOTE: The first card of each input yield deck must be present even if all the values on it are equal to zero.

12. **Energy Card Series. (Optional) Cards 9TCCCNN**

$9TCCCNN, X_1, X_2, X_3, \ldots, X_N$

$X_1 = E^\beta$, beta energy per decay in MeV.

X_N = The energy per decay in N energy groups. These can be arbitrary in the sense of energy range per group, type of decay, and energy units per decay. But corresponding to expected usage, the program output labels the first group as a β and the remaining groups as γ-decay energies. The output print includes a summation over the groups 2-N per nuclide in addition to aggregate summations over all N groups.

13. **γ-Spectra Energy Band Card.* (Optional) Card 901**

$901, X_1, X_2, X_3, \ldots, X_N$

X_N = N energy values defining N – 1 decay groups (group energy bounds) of arbitrary width:

$X_{N-1} < X_g \leq X_N$, where X_g denotes all γ energies appearing in group N.

---

*The listed nuclides have γ energies, intensities, and β^- end-point energies in data statements. The user must still specify nuclide chains and all other chain data. The presence of the 901 card forces the code to look up the spectral data when one or more of the above ID's *as listed* occurs in a chain. This option should not be used when the 9TCCCNN card(s) is present.

These 32 nuclides are the only fission products having γ energies above the ^9Be and $^2H(\gamma,N)$ threshold and $\tau_{1/2} \geq$ 15 min (above 1.67 MeV). A chain set in which they appear and ENDF/B-IV data are available.

Energy bounds are assumed to be in MeV.

Data for other nuclides can be stored in the SPECT routine.

At present, the output print is formatted for up to 101 groups, but this can be increased if desired.

If only the following list of nuclides is of interest, this card may be used in place of the Energy Card Series, 9TCCCNN.

BR0840	RH1060	LA1400	TC1040	LA1430
KR0870	AG1120	LA1410	SN1250	PR1460
KR0880	TE1311	LA1420	SB1290	
RB0880	I1320	PR1440	SB1310	
Y0900	I1340	RB0890	TE1331	
Y0920	I1350	Y0940	TE1330	
Y0930	CS1380	Y0950	BA1410	
ZR0970		NB0980		

14. Extra Summary Listings Card. (Optional) Cards 1JJSS

$1JJSS, N_1, N_2, \ldots, N_i$

JJ = A number designating a group of nuclides (N_1, $N_2, \ldots, N_i$) to be summed and listed as a separate summary output. JJ can go from 01-99.

SS = Sequence number.

N_i = A nuclide to be included in the sum of nuclides in group JJ. N_i is a six-digit, fixed point number corresponding to TCCCNN on the Number Density and Nuclide Library Card Series, 1TCCCNN. The barns/fission for groups 1, 2, 3, and 4 and the number density are summed over all nuclides, N_i, input per group JJ and listed by time step as a separate summary. Resonance integrals are also included in this summary.

Example: 10101, 204204, 204502, 204602 .

Nuclide 4 in chain 42

Nuclide 2 in chain 45

Nuclide 2 in chain 46

These nuclides can be, for example, all ^{135}Xe.

15. Continuation Card. (Optional) Card 100002

$100002, X_1, X_2, X_3, X_4$

Initial values of the following quantities can be supplied:

X_1 = Elapsed time.

X_2 = Thermal flux time (n/cm^2).

X_3 = Integrated power (W-s).

X_4 = Integrated neutron absorption (n/cm^3).

Any of the above quantities may be zero. This card does *not* refer to a restart of a previous problem. It simply permits specification of initial values of the quantities noted, which are otherwise set to zero initially.

16. Time-Dependent Flux Factor Card. (Optional) Cards 2TTTT

$2TTTT, X_1, X_2, X_3, X_4$

$TTTT$ = The time-step number. If the same values are to be used for a series of time steps, $TTTT$

is the number of the last time step in the series.

X_1 = Factor on ϕ_1.

X_2 = Factor on ϕ_2.

X_3 = Factor on ϕ_3.

X_4 = Factor on ϕ_4.

The number of these cards must be less than or equal to the number of time cards.

(These factors provide an additional degree of freedom in modifying neutron absorption in addition to the spectrum and shielding factors which are applied to the cross sections on series 20TTTT and 5TCCCNN.)

17. Punched Output for Plots. (Optional) Card 1

$1, N_1, N_2, M_1, M_2, \ldots, M_j$

N_1 = The number assigned to the curve on the graph (N_1 = 1 to 9).

N_2 = 1: Elapsed time is the abscissa.

2: Accumulated fission density is the abscissa.

M_j = 1: Punch fuel fraction for the nuclide specified on Card 5.

2: Punch resonance integral (RI).

3: Punch 2200 m/s σ.

4: Punch σ effective.

The M's are position-dependent in that the value of j designates the graph number.

Example: 1,3,1,4,1,2.

This set of curves is the third set on the graphs to be plotted.

Elapsed time will be punched on card series 101.

σ effective will be punched on card series 10301.

Fuel fraction will be punched on card series 20301.

Resonance integral will be punched on card series 30301.

18. Punched Output of Summary Data. (Optional) Card 2

$2, 1$

This card enables the user to obtain three subsets of punched output. Each set of punched data is preceded by a header card labeling the punched columns. The entire set is sequentially numbered except for the header cards. Each subset, useful for different purposes, contains some punches redundant with the other sets.

PUNCH SET 1

Format	Columns	Data
6X	1-6	Blank
I3	7-9	Time-step number
2X	10-11	Blank
F14.8	12-25	TS increment (h)
2X	26-27	Blank
F14.6	18-41	Accumulated time (h)
2X	42-43	Blank
E12.6	44-55	Fission density (fissions/barn-cm)
2X	56-57	Blank
E12.6	58-69	Accumulated fission density (fission/barn-cm)
3X	70-72	Blank
IS	73-80	Sequential card number

PUNCH SET 2

6X	1-6	Blank
I	7-8	Time-step number
F14.8	9-22	TS increment (h)
F14.6	23-36	Accumulated time (h)
E12.5	37-48	Total β energy rate (MeV/s)
E12.5	49-60	Total γ energy rate (MeV/s)
E12.5	61-72	Total $\beta + \gamma$ energy rate (MeV/s)
I8	73-80	Sequential card numbers (continued from SET 1)

PUNCH SET 3

Same as SET 2, except energy rates are in MeV/fission.

All punched data appear in the CINDER output summary.

19. Resonance Integral Group Lethargies.* (Optional) Card 3

$3, X_1, X_2, X_3, X_4$

$\left. \begin{array}{l} X_1 \\ X_2 \\ X_3 \\ X_4 \end{array} \right\}$ = The values for ℓ_i (lethargies) in the resonance integral calculation (card type 5)

If this card is not input, the following values are used:

$\ell_1 = 2.5$
$\ell_2 = 5.0$
$\ell_3 = 9.088399$
$\ell_4 = 0.0$.

---
*RI is printed on the output summary, not after each detailed time-step print.

20. Parameters for Calculation of Aggregate Resonance Integral and 2200 m/s Cross Section. (Optional) Card 4

$4, X_1, X_2, X_3, X_4$

$\left. \begin{array}{l} X_1 \\ X_2 \\ X_3 \\ X_4 \end{array} \right\}$ = The values of g_i used in the resonance integral and 2200 m/s cross section calculations (see card type 5).

If this card is not input, all values of g are 1.0.

21. Extra Summary Print and Spectra Conversion Card. (Optional) Card 5

$5, A, X$

A = The alphanumeric ID of the fuel fraction desired to be stored over time and printed in the summary.

X = The factor used for the conversion of $\hat{\sigma}_{eff}$ (defined below) to other spectra. X is normally zero.

$$\text{Resonance integral} = \sum_{i=1}^{4} \ell_i \cdot \frac{\hat{\sigma}_i}{g_i} \text{ per energy group } i$$

where ℓ_i = input on card 3

g_i = input on card 4

$\hat{\sigma}_f$ = total/barns/fission.

$$2200 \text{ m/s Sigma} = \frac{\hat{\sigma}_4}{g_4}$$

$$\text{Sigma eff} = \frac{\hat{\sigma}_4 + \sum_{i=1} \frac{\phi_i}{\phi_4} \cdot \frac{\hat{\sigma}_i}{g_i}}{g_4} \left(1.0 - X \cdot \frac{\hat{\sigma}_3}{\phi_4} \right).$$

When X = 0, σ_{eff} is an effective 2200 m/s barns/fission, i.e., total reaction rate = $g_4 \hat{\sigma}_{eff} \phi_4 = \sum_{i=1}^{4} g_i \hat{\sigma}_i \phi_i$.

The caret [^] denotes barns per fission.

C. Sample Input for CINDER-7

The following input sample specifies:

2 depletion chains with 5 nuclides in the first chain and 6 in the second.

2 fission product chains with 2 and 1 nuclides, respectively.

4 time steps with zero power at TS 2 and 4 and 2.86×10^5 W at TS 1 and 3. Time increments are 51.84, 164.16, 326.4, and 33.6 h.

Ratios of the 3 fast flux groups to the group-4 flux are 5.06, 7.95, and 5.98 during the two TS at power.

9 nuclides are in power (out of the 11 in the depletion chains). All nuclides except ^{232}Th and ^{233}Pa in the first depletion chain contribute to aggregate summations (of barns/fission, macroscopic absorption, decay heating, etc.).

The 5TCCCNN series cards specify factors which alter the cross sections at various time steps.

Cards beginning with (*) are optional comment cards; these are discarded by the code before execution. Data storage for this sample equals 364_{10}; 628_{10} are additionally required for calculations.

```
= SAMPLE INPUT FOR CINDER-7
100001, 0.0, 1.6942-8, 1, 0. , 0.0, 1
* FISS. HEAT FLUX HISTORY
200001, 51.84, 5.06, 7.95, 5.98, 1, 0, 2.86+5, 0.9567
200002, 164.16, 0.0, 0.0, 0.0, 1, 0, 0.0, 1.0
200003, 326.40, 5.06, 7.95, 5.98, 1, 0, 2.86+5, 1.0
200004, 33.6, 0.0, 0.0, 0.0, 1, 0, 0.0, 1.0
* NUMBER DENSITY AND LIB. CARD, DEP. CHAIN 1
1100101,TH-232, 1.4465-2, 2.286-18, 0, 4
1100102,PA-233, 0.0, 4.225-7, 0, 4
1100103,U-234, 0.0, 1.281-13, 1,4
1100104,U-235, 0.0, 4.457-16, 1, 4
1100105,U-236, 0.0, 1.330-15, 1, 1
* NUMBER DENSITY AND LIB. CARD, DEP. CHAIN 2
1100201,TH-232, 1.4465-2, 2.286-18, 1, 4
1100202,PA-233, 0.0, 4.225-7, 1, 2
1100203,U-233, 0.0, 1.944-13, 1, 4
 1100204,U-234, 0.0, 1.281-13, 1, 4
1100205,U-235, 0.6323-3, 4.457-16, 1, 4
1100206,U-236, 0.0, 1.330-15, 1, 1
* NUMBER DENSITY AND LIB. CARD, FISS. PROD CHAIN 1
1200101,XE-136, 0.0, 0.0, 1, 3
1200102,CS-137, 0.0, 7.471-10, 1, 1
* NUMBER DENSITY AND LIB. CARD, FISS. PROD CHAIN 2
1200201,ZR-95, 0.0, 1.234-7, 1, 1
* MICROSCOPIC ABS. CX CARD, DEP. CHAIN 1
2100101, .1619, .2941, 1.7338, 7.3287
2100102, .7503, .1982, 83.2456, 41.8705
2100103, 1.5548, .7994, 70.6076, 92.0176
2100104, 1.3363, 2.2562, 28.4158, 619.4244
2100105, .9169, .5461, 42.3300, 6.0
* MICROSCOPIC ABS. CX CARD, DEP. CHAIN 2
2100201, .1619, .2941, 1.7338, 7.3287
2100202, .7503, .1982, 83.2456, 41.8705
2100203, 1.9377, 3.2252, 87.7112, 583.7101
2100204, 1.5548, .7994, 70.6076, 92.0176
2100205, 1.3363, 2.2562, 28.4158, 619.4244
2100206, .9169, .5461, 42.3300, 6.0
* MICROSCOPIC ABS. CX CARD FP CHAIN 1
2200101, 0.0, 0.0, 0.011, 0.15
2200102, 0.0, 0.0, 0.033, 0.11
* MICROSCOPIC ABS. CX CARD FP CHAIN 2
 2200201, 0.0, 0.0, 0.0, 0.0
* MICROSCOPIC FISSION CX CARD, DEP. CHAIN 1
3100101, 0.0890, 0.0, 0.0, 0.0
3100102, .7503, .0083, 0.0, 0.0
3100103, 1.4077, .2419, 0.0, 0.0
3100104, 1.2733, 1.7643, 19.2419, 525.6581
3100105, .7512, 0.8241, 0.0, 0.0
* MICROSCOPIC FISSION CX CARD, DEP. CHAIN 2
3100201, .0890, 0.0, 0.0, 0.0
3100202, .7503, .0083, 0.0, 0.0
3100203, 1.8790, 2.9231, 74.8007, 533.3536
3100204, 1.4077, 0.2419, 0.0, 0.0
3100205, 1.2733, 1.7643, 19.2419, 525.6581
3100206, .7512, .8241, 0.0, 0.0
```

10

```
* UEP CHAIN NUCLIDES IN POWER, CHAIN 1 AND 2
100008.00101.00102.00103.00104.00105.00203.00204.00205.00206
* YIELD CARD SERIES, FP CHAIN 1
100009, 1, 2, 2, 3, 3, 2, 2, 3, 3
* YIELD CARD SERIES, FP CHAIN 1
* ASSUMING NO CONTRIBUTIONS TO CS-137 FROM I-135, XE-135,
* DECK 1, TH-232
* DECK 2, PA-233=U-233=U-234
* DECK 3, U-235=U-236
A00111, 6.40-2, 6.70-2
A00121, 6.68-2, 6.64-2
A00131, 6.46-2, 6.15-2
* YIELD CARD SERIES, FP CHAIN 2
A00211, 5.09-2
800221, 6.15-2
800231, 6.20-2
* YIELDS FOR U-234 = U-233, AND U-236=U-235, AND PA-233=U-233
10101.100105.100206
10201.100104.100205
10301.100103.100204
* FLUX FACTORS
2000004.1.0.1.0,1.0.1.0
5700101./2/1.0,.9567,1.0
5700102./2/1.0,.9567,1.0
5700103./2/1.0,.9567,1.0
5700104./2/1.0,.9567,1.0
5700105./2/1.0,.9567,1.0
5700201./2/1.0,.9567,1.0
5700202./2/1.0,.9567,1.0
5700203./2/1.0,.9567,1.0
5700204./2/1.0,.9567,1.0
5700205./2/1.0,.9567,1.0
5700206./2/1.0,.9567,1.0
5A00101./2/1.0,.9567,1.0
5A00102./2/1.0,.9567,1.0
5A00201./2/1.0,.9567,1.0
```

APPENDIX A

STANDARD FORTRAN AND CDC VERSIONS OF CINDER-7

The subroutines abstracted from Ref. 3 were
very interdependent, many were unnecessary to CINDER,
some were in the obsolete ASCENT language, and some
interacted with the I/O and the dayfile.

To create a free-standing version of CINDER of
reasonable size, we initially abstracted about 1800
of the 22 000 cards in the input package, and elim-
inated the obsolete ASCENT language. We later elim-

inated 9 subroutines unnecessary to CINDER by eliminating calls and modifying the CINDER and remaining input package routines as necessary. Some modifications were also needed to remove communication with the dayfile, etc. The 19 subroutines remaining are relatively small, totaling ~1300 cards (in CDC version). The most important functions of these routines (Ref. 3) have been retained. The input specifications of Sec. III refer to these modifications, and this version of the code is specifically designed for CDC computers. Machine dependence for this version is based primarily on the 60-bit word length, but statements in Ref. 3 regarding dependence still apply.

STANDARD FORTRAN VERSION

An even larger effort was made to produce all CINDER subroutines in standard FORTRAN which would be operational on most large computers without loss of the features described in the body of this report. Here the major problem was with a format-free input. However, a version which is essentially standard FORTRAN was finally coded and tested.

Twelve of the 19 remaining routines from the modified Bettis input package were eliminated and the remainder recoded. This part of the code now totals ~200 cards. The exposition in the body of this report also applies to the general FORTRAN version using the following input conventions:

Card 1 - Hollerith title card

Data cards in any order

Last card - any negative integer.

The remaining machine and compiler dependencies are summarized in Table A-I.

TABLE A-I

MACHINE AND COMPILER DEPENDENCIES OF CINDER-7

Necessary changes:

1. Program card.
2. Size of arrays in blank common and variables IASIZE and NCDSIZ defining their size.
3. Subroutine INTTAP. Set I to the integer value of your input unit.
4. Subroutine ØUTTAP. Set L to the integer value of your output unit.
5. Subroutine INP. Change the free-format read statement (list-directed read) INPUT (IN) NC (NCARD+1),B. For IBM 360 it would be: READ (IN,*) NC(NCARD+1),B. This is nonstandard. CDC and RCA compilers do not have free-form reads.

Nonstandard Features:

1. Implied do loop notation in data statements. Some compilers allow this and some allow an array name without subscript. All major FORTRAN compilers* allow one or the other.
2. Mixed-mode arithmetic (integer and real). All major FORTRAN compilers allow this.
3. Punch statements. All major FORTRAN compilers allow this.
4. "A" type formats depend on word and bit size of each machine. (NOTE: Precede A fields with NH, i.e., 4HU235.)

*Major FORTRAN compilers are those of CDC, GE, Honeywell, IBM, Univac, XDS, Burroughs, and DEC.

APPENDIX B

LIST OF SUBROUTINES IN CINDER-7

There are two versions of CINDER-7 subroutines. Subroutines listed in Table B-I apply to the standard FORTRAN version as described in App. A. It will work with major FORTRAN compilers (CDC, GE, Honeywell, IBM, Univac, XDS, Burroughs, and DEC). As noted in App. A, it requires only minor changes from machine to machine. The subroutines listed in Table B-II are used specifically for CDC compilers that do not permit the nonstandard FORTRAN statements required for the free-format read. This version is currently being used for informal communication with BAPL, but both versions are operational at LASL.

TABLE B-I

STANDARD FORTRAN VERSION OF CINDER-7

Subroutine	Octal Length
(CINDER Subroutines)	
CINDR	204
RD1	5210
WRTINP	1757
CALCUL	4720
TALEN	4563
SUMRY	2365
(Data Storage and Interpretation Subroutines)	
INTTAP	14
ØUTTAP	14
INP	723
LINC	74
MOVER	343
LINE	44
FINISH	66

TABLE B-II

CDC 6600 and 7600 VERSION OF CINDER-7

Subroutine	Octal Length
(CINDER Subroutines)	
CINDR	213
DR1	4646
WRTINP	1564
CALCUL	4630
SPECT	2416
TALEN	2306
SUMRY	2142
(Data Storage and Interpretation Subroutines)	
FINISH	66
ØUTTAP	14
INTTAP	14
LINES	13
INP	301
INP2	512
LINK	133
TITLE	43
CARDS	1774
INPFIX	132
INPFLT	357
INPSEP	76
INPCKN	33
INPWCH	107
PACK	10
UNPACK	10
INPRINT	60
INPHOL	224
MODER	274

DATA STORAGE

In either code, essentially all data storage is in a single blank-common array. The size is currently set at $30\ 000_{10}$. This is specified in the CINDER routine as IBSIZ. The code output print includes a statement giving the actual storage used and the additional storage required for running each case. If the net storage exceeds the specified size of IBSIZ, the user is informed by an output print and the case is not run.

REFERENCES

1. T. R. England, "CINDER -- A One-Point Depletion and Fission Product Program," Bettis Atomic Power Laboratory report WAPD-TM-334(Rev) (1964).

2. T. R. England, "An Investigation of Fission Product Behavior and Decay Heating in Nuclear Reactors," Thesis, University of Wisconsin, 1969 (Univ. Microfilms Order No. 70-12, 727).

3. C. J. Pfeifer, "CDC-6600 FORTRAN Programming -- Bettis Environmental Report," Bettis Atomic Power Laboratory report WAPD-TM-668 (1967).