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**FORIG: A Computer Code
for Calculating Radionuclide
Generation and Depletion in
Fusion and Fission Reactors****User's Manual****James A. Blink****Manuscript date: March 1985****MASTER****LAWRENCE LIVERMORE NATIONAL LABORATORY**
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Preface

This manual was written for two reasons. First, the manual documents the changes made in the FORIG code since 1982. Second, and most important, the manual provides a stand-alone document that should be easily readable both by new users and by users with previous ORIGEN2 or FORIG experience. At the present time a FORIG user must consult the ORIGEN2 (fission-oriented) manual as well as the previous FORIG manual, which only concerns itself with differences between the two codes.

James A. Blink

FORIG User's Manual

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FORIG: A Computer Code for Calculating Radionuclide Generation and Depletion in Fusion and Fission Reactors

User's Manual

Abstract

In this manual we describe the use of the FORIG computer code to solve isotope-generation and depletion problems in fusion and fission reactors. FORIG runs on a Cray-1 computer and accepts more extensive activation cross sections than ORIGEN2 from which it was adapted. This report is an updated and a combined version of the previous ORIGEN2 and FORIG manuals.

1. Introduction

Neutrons arising from fusion reactions are absorbed in structures, coolants, and shields and consequently produce activation products. The subsequent decay of these radionuclides can be a troublesome heat source after shutdown (afterheat), and also may present a radiation hazard to occupational workers during reactor maintenance and to the public if the activated materials are dispersed in an accident. Finally, the long-lived activation products are hazardous waste that must be isolated from the environment for long periods of time.

The FORIG computer code was developed to calculate the time-dependent radionuclide populations in fusion reactors in terms of various figures of merit—*isotope mass, activity (curies), inhalation and ingestion biological hazard potentials (bhp's), afterheat (watts), gamma spectrum, surface dose rate, and waste disposal feasibility.* The FORIG code is an adaptation of the widely respected fission-reactor code ORIGEN2.¹

This user's manual is an updated and combined version of the information in the previous FORIG² and ORIGEN2 manuals.¹ Modifications for allowing fusion-reactor calculations are structured to permit continued use of the code for fission-reactor calculations.

Experienced ORIGEN2 users will wish to read Section 2, where differences in the input files of FORIG and ORIGEN2 are explained and code modifications are described, and then proceed to Section 3. *New users of the code should proceed directly to Section 3, where the source, controllee, input and output files are defined.* Sections 4 and 5 are detailed descriptions of the input and output formats, respectively.

The ORIGEN2 code, used to generate FORIG, was obtained from Oak Ridge National Laboratory (ORNL) in 1981. Users interested in fission-reactor calculations should contact Allen Croff at ORNL to obtain the ORIGEN2 improvements (or corrections) implemented since 1981.

2. A Comparison of FORIG and ORIGEN2

To permit fusion-reactor calculations, the major modification was to increase the maximum permitted number of removal paths (activation or decay) for each nuclide. The seven paths allowed in the ORIGEN2 code are inadequate. Compared to fission reactors, the high-energy flux in a fusion reactor drives more reactions, such as (n,np) , $(n,n\alpha)$, (n,d) , and (n,t) . The FORIG program allows 15 removal paths. Additionally, if the maximum removal paths are exceeded, the ORIGEN2 code proceeds after a warning error, whereas FORIG terminates.

In ORIGEN2, the decay and cross-section libraries are merged (concatenated) into a single file (TAPE9) before the code starts. In FORIG, the two libraries are kept separate. The decay library is in TAPE9 and the cross-section libraries are in TAPE8. To permit the separation of the decay and cross-section libraries, the LIB command was modified to

LIB NLIB(1) . . . NLIB(13)

The LIB command entry fields are defined as follows:

- NLIB(1) Printing of reaction-rate matrix if >0 . That is, for each nuclide, we obtain a printout of the production rate from each parent, in addition to the decay rate and the removal rate by activation.
- NLIB(2) Light-nuclide-decay library number.
- NLIB(3-4) Actinide and fission-product decay library numbers (zero for pure fusion problems).
- NLIB(5-7) The three cross-section library numbers. (Note: If a decay or cross-section library number is negative, replacement cards will also be read later in the "odd-reaction file" for that library.)
- NLIB(8) Input unit number for the cross-section library (use 8).
- NLIB(9) Input unit number for the decay library file (use 9).
- NLIB(10) Input unit number for the replacement decay and cross-section lines and for reading "odd-reaction" cross sections (use 3).
- NLIB(11) The number of odd reactions to be read. If NLIB(11) is negative, the data will be read on unit NLIB(10). If NLIB(11) is positive, the data will be read on unit NLIB(8). Use negative NLIB(11) to read the odd reaction on TAPE3.
- NLIB(12) Determines the set of actinides that have direct fission-product yields (use 0 for fusion problems).
- NLIB(13) Determines the set of variable actinide cross sections to be used (use 0 for fusion problems).

The change is in the addition of the NLIB(9) field.

A new first line was added to the instruction file (ORINPUT). If the line contains a negative integer, FORIG will not print neutron sources from spontaneous fission, from neutron-induced fission, or from (α,n) reactions. Such tables are not applicable to pure fusion problems. If the line is zero or a positive integer, FORIG will print the neutron-source information (like ORIGEN2).

To accommodate larger cross-section library entries, FORIG was modified to accept 120 character lines in the data files (ORIGEN2 only reads the first 80 characters). FORIG reads the cross-section libraries in the ORIGEN2 format, if the library number is less than 700. If the library number is greater than or equal to 700, the extended FORIG format is used.

The cross-section formats (used in the TAPE8 input file) are summarized in the following:

- For a library number less than 700, use standard ORIGEN2 library.

```
NLB TITLE (For the light-nuclide segment)
NLB ZZAAAI Sn $\gamma$  Sn2n Sn $\alpha$  Snp Sn $\gamma$ x Sn2nx -1.0
.
.
.
-1
```

```
NLB TITLE (For the actinide segment)
NLB ZZAAAI Sn $\gamma$  Sn2n Sn3n Snf Sn $\gamma$ x Sn2nx -1.0
.
.
.
-1
```

```
NLB TITLE (For the fission-product segment)
NLB ZZAAAI Sn $\gamma$  Sn2n Sn $\alpha$  Snp Sn $\gamma$ x Sn2nx 1.0
Y(1) Y(2) Y(3) Y(4) Y(5) Y(6) Y(7) Y(8)
.
.
.
-1
```

These entry fields are defined as follows:

NLB	Library number for each segment.
ZZAAAI	Nuclide identification (atomic number, atomic weight, and I = zero for the ground state or one for an isomer).
Sn γ	Cross section for the (n, γ) reaction; sn2nx is the cross section for the ($n, 2n$) reaction leaving an excited daughter, etc.
Y(1) through Y(8)	These represent the percent fission-product yield for fissioning of eight different actinides, as specified in Table 5.2 of the ORIGEN2 manual (or in Table 10 of this manual). The Y(1) through Y(8) line is omitted if the previous line ends in a negative number.

- For a light-nuclide library number greater than 699, use the ORLIB format.*

```
NLB TITLE (For a pure fusion problem)
Comment line
ZZAAAI Sn2n Sn2nx Sn3n Snp Snn $\alpha$  Snp Snd Snt Sn $\alpha$  Sn $\gamma$ x
.
.
.
-1
```

*ORLIB is a cross-section manipulation code³ available with FORIG. The code can be used to energy average the ACTL cross-section library,⁴ using either an input flux or a TARTNP-generated flux⁵ as a weighting function. One improvement has been made to ORLIB since Ref. 3 was published; a flag has been added to the ORLIB input file to allow production of libraries in one of four formats (0 or no entry = FORIG light-nuclide format, 1 = ORIGEN2 light-nuclide format, 2 = actinide format, 3 = fission-product format without fission product yield-curve entries). This input flag is an I1 format in Column 30 of line 2 of the ORLIB input.

This example includes no cross sections for actinides or fission products. If the problem includes either, they are entered in the original ORIGEN2 format following the light nuclide segment.

In both ORLIB and ORIGEN2 formats, the code will error exit if cross sections are entered for nuclides not in the appropriate decay-library segment.

The odd-reaction cross-section file (TAPE3) format was also modified to accommodate (optional) additional information available from the ORLIB cross-section code.

- For a light-nuclide library number less than 700, input the odd reactions in the ORIGEN2 format. The first and second entries on each line are the parent and daughter nuclide identifications. When this format is used, hydrogen and helium isotope production is not calculated for the odd reactions.

ZAAAI ZZAAAI SIGMA

.

- For a light-nuclide library number greater than 699, input the odd reactions in the ORLIB format.

NLIB TITLE

Comment Line

ZZAAAI ZZAAAI SIGMA TYPE

ZZAAAI ZZAAAI SIGMA TYPE

.

Here, TYPE is the reaction type number (Table 1). With this format, hydrogen and helium isotope production is calculated for both the odd and the standard reactions.

In either format, each odd reaction is added to the first library segment (light nuclide, actinide, or fission product) that includes the parent in the decay library. Also, if replacement decay or cross-section lines are to be inserted in any of the six library segments, these lines precede the odd-reaction formats shown. (The LIB and LPU commands force the code to read each line in its proper format, as discussed later.)

The decay library format was also altered to add an additional field at the end of each (two-line) nuclide entry. This field is the shallow-burial radioactivity concentration guide (SBRCG), i.e., the nuclide's allowable specific activity in curies per cubic meter if it is to be buried in a low-level disposal site as Class C waste as

Table 1. Reaction types available from the ORLIB code.

Type No.	Reaction	Type No.	Reaction
12	(n,2n)	29	(n,2np) α
13	(n,3n)	30	(n, γ na)
14	(n,4n)	31	(n, γ na)
15	(n,fiss)	37	(n,2 α) α
20	(n,np)	38	(n, 3 He α) α
21	(n,pn)	39	(n,tp) α
22	(n,nd)	40	(n,p)
23	(n,nd α) α	41	(n,d)
24	(n,nt)	42	(n,t)
25	(n, 3 He)	43	(n,t α) α
26	(n,n ω)	44	(n, 3 He)
27	(n,n2 α) α	45	(n, α)
28	(n,nt α) α	46	(n, γ)

specific activity in curies per cubic meter if it is to be buried in a low-level disposal site as Class C waste as specified⁶ in 10CFR61. Each SBRCG value assumes only that nuclide will be buried. If multiple nuclides are to be buried,

$$\sum_{\text{nuclides}} A_i (Ci/m^3) / SBRCG_i \leq 1 \quad (1)$$

or the waste must be isolated in a geologic (high-level waste) repository. The shallow burial index ($A_i / SBRCG_i$) is calculated by FORIG when Table 11 or 12 (for absolute or fractional values, respectively) is specified in the OPTL, OPTA, or OPTF commands.

To calculate the shallow-burial output, the code requires the volume of the activated material (waste). This value is input as a new command,

VOL vol ,

where vol is the material volume in cubic meters. The VOL command must precede the OUT command that outputs Table 11 or 12 (as previously specified by an OPTL, OPTA, or OPTF command).

One final command was added to the code. This command produces a crude calculation, suggested by D. Dorn of LLNL, of the gamma dose rate at the surface of a large (infinite) slab of a given thickness, material, and density and that contains the calculated activity density. The calculation is crude, because it does not consider down-scatter of photons in Compton scattering. Instead, an effective (energy-dependent) absorption cross section is used; this cross section is the sum of the (energy dependent) photoelectric and pair-production cross sections plus one-half the Compton cross section. The input format is

RADS rho thick mat'1 ,

where rho is the density in g/cm³, thick is the slab thickness in cm, and mat'1 is Fe, Ai, Mo, W, or Pb. The user must select the element that most closely approximates the actual material. If rho is entered as a negative number, the output is sent to the abbreviated (TAPE11) file, rather than to the full output file (OROUT). The RADS command must be preceded by both VOL and OUT commands. The RADS command was added to allow order of magnitude estimation of contact dose rates. More accurate estimates require a formal shielding calculation. Also, during operation, the dose rate from (n,γ) reactions (not calculated by FORIG) could be much higher than the dose rate from decay gamma rays.

3. Files and General Code Information

ORIGEN2 is a widely used computer code¹ for calculating the build-up, decay, and processing of radioactive materials in fission reactors. Here we document changes made in the ORIGEN2 code to allow its use on an LLNL Cray-1 computer. With only slight changes in the input file, any of the standard fission problems can run at LLNL as they do at Oak Ridge and elsewhere.

In addition, the code was modified to read activation cross-section libraries appropriate for fusion-reactor design problems. If the flux-weighted cross sections are constant in time, a single FORIG run can calculate activation and decay in a fusion-reactor component. To address fission-fusion hybrid-reactor designs, FORIG will require further modification to account for actinide burn-up and fission-product build-up. This may be done in a manner similar to the fission-reactor treatment in ORIGEN2, or it may be accomplished by cyclically using a neutron-flux code, a cross-section collapsing code, and FORIG.

3.1. Data Libraries

FORIG contains three separate and distinct nuclide lists for which nuclear data may be required:

- Activation products (sometimes called "light" nuclides) include low-Z impurities and structural materials.
- Actinides include all of the heavy isotopes ($Z > 90$) and all their decay daughters, including the final stable nuclides.
- Fission products include all nuclides having a significant fission product yield (either binary or ternary) and some nuclides resulting from neutron captures of the fission products.

Each of these three segments has three libraries that may be read:

- A decay data library that gives nuclide half-lives, decay modes, recoverable heat energy, natural abundances, and toxicities.
- A cross-section and fission-product-yield library that gives the cross sections for several reactions (as effective one-energy-group values in barns) and gives the fission product yields from ^{232}Th , ^{233}U , ^{235}U , ^{238}U , ^{239}Pu , ^{241}Pu , ^{245}Cm , and ^{252}Cf .
- A decay-photon yield library that gives the photons per disintegration in 18 energy groups. (Note that photons emitted in n, γ reactions are not tabulated by the FORIG or ORIGEN2 computer codes.)

In addition to these normal data-input library facilities in FORIG, two additional options are available to extend, update, or correct these libraries.

The first allows the user to input substitute decay-data lines and substitute cross-section and fission-product-yield data lines that override the corresponding data lines present in the main libraries. This option is an alternative to rebuilding the entire data library simply to change one or two items.

The second option allows the user to input any flux-dependent reaction rate between any two nuclides. While the user can duplicate the standard reaction types available in FORIG, the option is principally intended to allow inclusion of nonstandard reaction types.

The ORIGEN2 code cross-section library includes six standard reactions for nonfissionable nuclides: $(n, 2n)$, $(n, 2n)\text{exc}$, (n, p) , (n, α) , (n, γ) , and $(n, \gamma)\text{exc}$, where "exc" indicates the daughter is in an excited state.

Additional reactions may be added in a separate file, which is called "odd reactions" in the ORIGEN2 manual. For fission reactors, there are few additional reactions. However, fusion-reactor problems require an odd-reaction file many times larger than the standard file because of the higher energy neutrons. Further, ORIGEN2 ignores the production of helium and hydrogen isotopes from odd reactions.

The neutron-energy spectra differ between regions in a single fusion reactor design and differ between designs. FORIG requires flux-averaged (i.e., energy-weighted) cross sections. Hence, multiple cross-section libraries are required for fusion reactors and even for different regions of any specific design. The ORLIB code³ was written at LLNL to use the ACTL cross-section library⁴ and the TARTNP neutronics code⁵ to produce a one-energy-group cross-section library for any region of a fusion reactor.

The ORLIB output includes a cross-section file with 11 standard reactions:

$(n, 2n)$, $(n, 2n)exc$, $(n, 3n)$, (n, np) , $(n, n\alpha)$, (n, p) , (n, d) , (n, t) , (n, α) , (n, γ) , and $(n, \gamma)exc$.

ORLIB also produces an odd-reaction file. Table 1 lists the 26 reaction types processed by the code. FORIG will accept these files if the library number is set ≥ 700 . If the library number is less than 700, the standard ORIGEN2 cross-section format is required. FORIG also calculates helium and hydrogen isotope production from both standard and odd reactions.

FORIG, like ORIGEN2, divides problems into three parts: light nuclides (also called activation products), actinides, and fission products. In some problems, a nuclide may appear in two categories, and the code will treat the two separately. In pure fusion problems, there are no actinides or fission products, and no libraries are input for these categories. In this case, zeros are entered in the appropriate blocks in the LIB and PHO commands of the input file. (All FORIG commands are discussed later in this report).

3.2. FORIG Source and Control Files

The amount of memory required to solve a FORIG or ORIGEN2 problem is determined by the number of nuclides, cross sections, etc. that are involved. To prevent excessive use of memory, the source file was divided into two parts: first, a small main program (FORIGM) and, second, a large file containing all the subroutines required by the main program. The subroutines do not change from problem to problem, but the dimensions set up in FORIGM must be predetermined by the user. Figure 1 provides a listing of the main program as well as information required for setting the dimensions of the arrays. Once these are set, the program can be compiled.

3.2.1. FORIGM, the Main Program

FORIGM performs four major functions:

- (1) Provides a mechanism for variably dimensioning FORIG to accommodate different problem sizes.
- (2) Provides much of the framework necessary for putting some of the arrays to several different uses.
- (3) Calls for the subroutines that perform the desired operations.
- (4) Provides a mechanism for executing multiple FORIG problems with a single job.

The third function is handled automatically and will not be discussed. The fourth function is discussed in Section 4.2.1 (STP Command).

In Function 1, FORIG is variably dimensioned to allow the user to tailor the size of the executable module to the problem size or to the available computer space. The sizes of the FORIG executable modules range from about 175K (1K = 1024 bytes = 256 single precision words) to about 600K, the size depending mainly on the number of nuclides being considered.

FORIG keeps track of and prints the minimum required size of most of the variably dimensioned arrays (see Section 5.3 for output details). Table 2 is a summary of the recommended dimensions for several example problems. The magnitudes of the dimensions depend on the number of actinide nuclides having direct fission-product yields, which can range from zero to eight (see the LIB command). Dimensions are given in Table 2 for cases of 0, 4, 6, and 8 actinides having direct fission-product yields.

The variable NYTF in FORIGM (see Fig. 1) indicates whether thermal-reactor (NYTF = 0) or fast-reactor (NYTF = 1) neutron yields per neutron-induced fission are to be used (see also Section 4.1.1). The variables RMULV, DR, ER, FR, and LAM are related to a multiplier used by the MOV and ADD commands. LAM is the number of possible multipliers (presently four) in a given set of multipliers. These are specified by using DATA statements in FORIGM. Variables DR, ER, and FR are equivalenced to the appropriate portion of RMULV. The variable LAM is passed in subroutine parameter lists for variable dimensioning purposes.

AAAA	(13) No. output vectors	IIII	(2500) Maximum No. nonzero matrix elements for long-lived nuclides
BBBB	(900) Maximum No. nuclides	JJJJ	(10) No. storage vectors
CCCC	(15) Maximum No. nonzero cross-section and decay reactions per nuclide (cannot exceed 15 without redimensioning the subroutines)	KKKK	(300) No. nonzero natural abundance
DDDD	(2500) Total No. nonzero matrix elements	LLL	(3300) No. nonzero photon yields
EEEE	(4) No. nonzero fission-product yields	MMMM	(900) Maximum No. light nuclides
FFFF	(4) Maximum No. fission products	NNNN	(4) Maximum No. variable multipliers (in RMULV)
GGGG	(4) Maximum No. actinides + 1	PPPP	(3) No. actinides with both direct fission-product yields and a variable fission cross section (usually 3; can be 4 for plutonium-enriched thorium fuels)

```

1 c * generalized main program for the forig code on the 11n1 cray-1 * fln1
2 c forig is origen2 modified for fusion problems. fln1
3 c columns 74-77 contain information indicating the source of that line fln1
4 c orig is the original oak ridge origen2 fln1
5 c sla is a sandia albuquerque modification fln1
6 c 11n1 is a lawrence livermore modification fln1
7 c forig is an oak ridge modification for fusion problems fln1
8 c fln1 is a lawrence livermore modification for fusion problems fln1
9 c this main has been modified for a fusion library by allowing forig
10 c for up to 15 non-zero reactions per nuclide. variables coeff, fln1
11 c nprod, dum1, dum2, and lc are affected. forig
12 c updated mar 21, 85
13 program origen2 fln1
14 logical long orig
15 dimension xnew(AAAA,BBBB),coeff(CCCC,BBBB),nprod(CCCC,BBBB), fln1
16 $mmx(BBBB),kap(BBBB) forig
17 dimension sttfpb(JJJJ,10),istoti(JJJJ,03),is(JJJJ),rstoti(JJJJ) orig
18 dimension dr(NNNN),er(NNNN),fr(NNNN) orig
19 dimension yield(EEEE),nyield(FFFF),rmulv(NNNN,3) forig
20 dimension alphn(GGGG),nucan(GGGG),nucsfu(GGGG),ny(GGGG),yy(GGGG), forig
21 $ffsf(GGGG),ffa(GGGG) forig
22 common /junk/err,ldm(1),llite,iact,ifp,ltot,ilmax,imax,ifmax, orig
23 $itmax,izmax,axn,qxn,flux,power,index,tfpbav(4),ipmax orig
24 common /main03/nstp,anmul,anexp,nabmax, icnmax,iapmax,ifymax orig
25 c 1766 words are necessary in /nudscr/ beginning with s orig
26 c /nudscr/ is used for multiple purposes. orig
27 common /nudscr/dum1(CCCC,BBBB),dum2(CCCC,BBBB),s(2),cimn(BBBB), fln1
28 $csum(BBBB),nonp(BBBB),nq(BBBB),xp(BBBB),xpar(BBBB),xtemp(BBBB), forig
29 c max allowable size of reduced matrix increased from 1300 to 1111. forig
30 $d(BBBB),ap(IIII),locp(IIII),long(BBBB) forig
31 c increase the number of matrix elements per nuclide. forig
32 common /bit/nuc1(BBBB),q(BBBB),fg(0004),tocap(BBBB),genneu(GGGG), forig
33 $alphn(GGGG),sponf(GGGG),sfnu(GGGG),fiss(GGGG),nucab(BBBB), forig
34 $ampc(BBBB),wmpc(BBBB),xstore(JJJJ,BBBB),dis(BBBB),b(BBBB), forig
35 $abund(KKKK),non0(BBBB),kd(BBBB),loc(DDDD),ngf(BBBB),ngn(BBBB), forig
36 $ngr(LLLL),ggr(LLLL),a(DDDD),toca(DDDD),nfudfp(FFFF,PPPP) forig
37 dimension sh1(BBBB) fln1
38 c dr,er, and fr provide a convenient mechanism for initializing variable orig
39 c multiplier array rmulv. orig
40 equivalence (dr(1),rmulv(1,1)),(er(1),rmulv(1,2)), orig
41 $(fr(1),rmulv(1,3)) orig
42 c these equivalences have been realigned. forig
43 equivalence (dum1(1,1),coeff(1,1)),(dum2(1,1),nprod(1,1)), orig
44 $(xtemp(1),mmx(1)),(kap(1),d(1)),(xnew(1,1),dum1(1,1)) forig
45 equivalence (ap(1),alphn(1)),(alphn(GGGG),nucan(1)),(nucan(GGGG), forig
46 $nucsfu(1)),(nucsfu(GGGG),ny(1)),(ny(GGGG),yy(1)),(yy(GGGG), forig
47 $ffsf(1)),(ffsf(GGGG),yield(1)),(yield(EEEE),nyield(1)) forig

```

Figure 1. FORIG main source file (FORIGM). The dimensions are set by using the comments above the listing. For a pure fusion problem with less than 900 nuclides, the chosen dimensions are listed parenthetically (continued on next page).

```

48      call dropfile('+forig')
49 c orinput is instructions, tape8 cross sections, tape9 decay library,
50 c tape10 photon library, and tape3 odd reactions.
51      call open(5,'orinput',2,nerr)
52      if(nerr.lt.0) call exit(1)
53      call open(8,'tape8',2,nerr)
54      if(nerr.lt.0) call exit(1)
55      call open(9,'tape9',2,nerr)
56      if(nerr.lt.0) call exit(1)
57      call open(10,'tape10',2,nerr)
58      if(nerr.lt.0) call exit(1)
59      call open(3,'tape3',2,nerr)
60      if(nerr.lt.0) call exit(1)
61 c
62      call open(4,'tape4',2,nerr)
63 c      if(nerr.lt.0) call exit(1)
64 c      tape4 statements should be un-commented if tape4 is to be used
65 c
66 c orout is main output, tape7 is exit material compositions,
67 c tape11 is alternate main output, tape12 is tape6 contents,
68 c tape13 is tape11 contents, tape15 is debug and error info,
69 c tape16 is variable actinide cross section info, and tape50
70 c is a working file.
71      call create(6,'orout',3,200000)
72      call create(7,'tape7',3,200000)
73      call create(11,'tape11',3,200000)
74      call create(12,'tape12',3,200000)
75      call create(13,'tape13',3,200000)
76      call create(15,'tape15',3,200000)
77      call create(16,'tape16',3,200000)
78      call create(50,'tape50',2,200000)
79      call create(17,'orplot',3,200000)
80 c initialize page counter
81      npage=ipage(0)
82      call nudoc(6)
83      ix=JJJJ
84      mx=AAAA
85      lc=CCCC
86      ilmax=MMMM
87      iamax=GGGG
88      ifmax=FFFF
89      itmax=BBBB
90      izmax=DDDD
91      iphmax=LLLL
92      iapmax=IIII
93      ifymax=EEEE
94      nabmax=KKKK
95      icnmax=PPPP
96      ifd=FFFF
97      lam=NNNN
98 c neutrons/neutron-induced fission: 0=thermal spectrum; 1=fast spectrum
99      nytf=1
100      nytf=0
101 c call subroutine to read card input from unit 5, print it on unit 6, &
102 c write it on unit 50. unit 50 is then rewound and origen2 reads the
103 c data from unit 50.
104      call listit(5,6,50)
105      rewind 50
106 c main1 handles the miscellaneous initialization data
107      1 call main1(nytf,sfnu,alphan,can,nucsfu,ny,yy,anmul,anexp)
108 c main2 reads the origen2 commands
109      2 call main2(nstp,shal)
110 c main3 executes the origen2 commands
111      3 call main3(
112      $ long,stoffb,istotf,ls,rstotf, ix, mx, lc, ifd,
113      $ nucab,nor0,kd,loc,ngf,ngn,ngr,nyield,nonp,nq,locc,mmax,kap,
114      $ loca,nfudfp, cinn,csum, s,
115      $ nucl,q,fg,tocap,genneu,alphan,spof,sfnu,fics,ampc,wmpc,xstore,
116      $ dsis,b,ggr,yield,a, xp,xpar,xtemp,d,ap,coeff,rprod, xnew,
117      $ salphn,nucan,nucsfu,ny,yy,ffsf,ffa,abund,rmulv,lam,shal )
118 c this "go to" provides the mechanism for executing multiple problems
119 c within a single job.
120      go to (1,2,3,4),nstp

```

Figure 1. (continued on next page).


```

121      4 continue                                orig
122      call timeused(icpu,ifio,isy)              flnl
123      ricpu=icpu/1000000.                        flnl
124      rifo=ifio/1000000.                        flnl
125      risys=isy/1000000.                        flnl
126      write(6,430)ricpu,rifo,risys             flnl
127  430 format(1hl,/" cpu time = ",f9.3," , io time = ",f9.3. flnl
128      $", system time = ",f9.3," seconds")      flnl
129      call lastmem(lwamem)                      flnl
130      rmem=lwamem/150000.                      flnl
131      write(6,435)rmem                          flnl
132  435 format(" memory used = ",f9.3,"X")         flnl
133      call timedate(time,date,mach)             flnl
134      write(6,436)time,date,mach               flnl
135  436 format(10x,a8,10x,a8,10x,a1," machine")  flnl
136      write(6,437)                             flnl
137  437 format(1hl)                              flnl
138      stop 100                                orig
139      end                                        orig

```

3.2.2. Compiling FORIG Source Files

The appropriate files may be obtained from RSIC or NMFECC or, at LLNL, in J. Blink's take directory.

XPORT / t v

.RD .080975:ORIGEN2 [FORIG FORIGM FORIGMG FORIGS BFORIGS]

XPORT is a file transport routine at LLNL.

FORIG is the controller set up for a pure fusion problem with less than 900 nuclides in the decay library. If the problem of interest does not require actinides or fission products and if the existing decay library is adequate, then only FORIG need be read and no compiling process is needed.

FORIGS is the source file containing all the FORIG subroutines. This file is only required if the subroutines must be changed. Once the changes are made, FORIGS is compiled into a library:

RCFT I=FORIGS / t v

The compiler RCFT produces the library file BFORIGS. If no changes are to be made in the subroutines, BFORIGS may be read directly from the take directory.

FORIGMG is the main program with generalized dimensions. After it is read and its dimensions adjusted, it may be renamed FORIGM and compiled:

RCFT I=FORIGM, X=FORIG, LIB=(BFORIGS,FORTLIB) / t v

The FORIGM file in the take directory is set up for a pure fusion problem with less than 900 nuclides. The controller is named FORIG.

3.3. FORIG Input Files

FORIG requires the following input files (which are now to be discussed individually):

TAPE9	Decay data library
TAPE8	Cross-section library
TAPE3	Odd-reaction cross-section library
TAPE10	Photon-decay data library
ORINPUT	Instruction list
TAPE4	Input composition file (optional)

Table 2. Dimensions for FORIG example cases. (Case 17 is for a fusion problem.)

Parameter	Case																
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
Segments considered ^a	APA,FP	APA,FP	APA,FP	A,FP	A,FP	A,FP	APA	A	AP	APA,FP	A	AP	FP	A,FP or A,AP	APA,FP	APA,FP	AP
Type of calculation ^b	Any	Any	Any	Any	Any	Any	Any	Any	Any	Decay	Decay	Decay	Decay	Any	Any	Any	Any
Number of actinides with direct fission product yields	4	6	8	4	6	8	0	0	0	0	0	0	0	4	6	4	0
Alphabetic array dimensions ^c																	
AAAA	13	13	13	13	13	13	13	13	13	13	13	13	13	13	13	13	13
BBBB ^d	1732	1676	1676	1000	1000	1000	820	132	700	1676	132	700	880	1000	1676	1676	900
CCCC	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	15
DDDD ^d	6400	7900	9600	5000	6600	8200	1800	400	1500	1700	280	600	1000	5000	8000	9996	2500
EEEE ^d	3300	5000	6600	3300	5000	6600	4	4	4	4	4	4	4	3300	5000	6600	4
FFFF ^d	880	880	880	880	880	880	4	4	4	880	4	4	880	880	880	880	4
GGGG ^d	132	132	132	132	132	132	132	132	4	132	132	4	4	132	132	132	4
IIII ^{d,e}	3500	4200	5000	2500	3100	3800	1700	500	1300	600	500	250	250	2500	4500	4500	2500
JJJJ	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10
KKKK ^d	502	450	450	160	160	160	300	12	300	450	12	300	160	300	450	450	300
LLLL ^d	7900	7900	7900	4700	4700	4700	4700	1500	3300	7900	1500	3300	3300	4700	7900	7900	3300
MMMM ^d	732	700	700	4	4	4	700	4	700	700	4	700	4	700	700	700	900
NNNN ^d	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4
PPPP ^f	3	3	3	3	3	3	3	3	3	3	3	3	3	3	4	6	3
Approximate amount of core required for execution (bytes) ^g	548K	560K	576K	396K	412K	428K	332K	132K	298K	494K	182K	286K	324K	396K	560K	576K	

^aAP = activation products; A = actinides and daughters; FP = fission products.

^bAny = either irradiation (i.e., IRP or IRF commands) or decay (i.e., DEC command) can be used. Decay = no irradiation, decay only.

^cSee Fig. 1 for details on the description and use of these dimensions.

^dArray dimensions should be evenly divisible by 4 to ensure word boundary alignment.

^eLarger dimensions may be required for small irradiation or decay time step. In the limit of zero time, IIII = DDDD.

^fDepends on reactor being considered; see Fig. 1, item PPPP.

^gCan vary, depending on the number of input/output units and buffer sizes.

3.3.1. TAPE9: Decay-Data Library

The first card of each of the three allowable segments (activation product, actinide, and fission product) of the decay-data library is a title card. It contains the number of the decay-library segment and the alphanumeric title of the segment. The lines following the title card are the decay data for the nuclides in a particular library segment. The decay data for each nuclide are specified on two sequential cards. The format of the decay library is given below.

The decay library serves other vitally important functions in the FORIG code, in addition to supplying decay data. The nuclide identifiers supplied by the decay libraries define the total list of all nuclides that will be considered in subsequent FORIG calculations. Thus, if a nuclide is to be used in a calculation (even if only the cross-section information is required), it *must* be present in the decay library. The decay library also defines the nuclide membership of each of the three library segments (activation product, fission product, and actinide) considered by the FORIG code. In some cases, a given nuclide can appear in two or even three library segments. Finally, the decay library defines the order in which the nuclides will be printed within each library segment during the normal output.

As a result of these considerations, the decay library *must* be input before the photon libraries (PHO command) or before the initial compositions (INP command). When the LIB command is invoked, the decay library is automatically read before the cross-section library.

The decay library is input in the following format. Only the final field (SBRCG) differs from that in ORIGEN2.

```
NLB TITLE (For the light-nuclide segment)
NLB ZZAAl IU THALF Fβ× Fβ+EC Fβ+ECx Fα Fit
NLB Fsf Fnβ QREC ABUND ARCG WRCG SBRCG
```

```
·
·
·
```

-1

```
NLIB TITLE (For the actinide segment)
```

```
·
·
·
```

-1

```
NLIB TITLE (For the fission-product segment)
```

```
·
·
·
```

-1

The data entry fields are defined as follows:

ZZAAAI	Nuclide identifier (atomic number, atomic weight, and 0 or 1 for ground or isomeric state).
IU	Time unit (see Table 3) for the half life, THALF.
Fβ	Fraction of all decays by negatron (beta) emission with an excited daughter.
Fβ+EC	Fraction of all decays by positron emission or electron capture.
Fβ+ECx	Fraction of all decays by positron emission or electron capture and with an excited daughter.
Fα	Fraction of all decays by alpha emission.
Fit	Fraction of all decays by internal fission.
Fsf	Fraction of all decays by spontaneous fission.
Fnβ	Fraction of all decays by neutron-plus-beta decay.

QREC	Total recoverable energy (not including neutrinos) released per decay (in MeV).
ABUND	Natural isotopic abundance.
ARCG	Radioactivity concentration guide (RCG) for continuous inhalation in unrestricted areas as given in Ref. 7 (Table II, column 1), 10CFR20, using the lower of the soluble and insoluble values.
WRCG	RCG for continuous ingestion from column 2 of the same table in 10CFR20.
SBRCG	Allowable curies of nuclide per cubic meter of waste for shallow burial if only that nuclide is buried (from Ref. 6, 10CFR61).

The fraction of negatron beta decays with a ground-state daughter is calculated from the above fractions (1 minus the sum of the entries).

Since the decay library is used to produce the nuclide list for the calculation, both radioactive and stable nuclides of interest must be included.

Table 3. Time-unit designations.

1	= seconds
2	= minutes
3	= hours
4	= days
5	= years
6	= stable
7	= 10^3 years (kY)
8	= 10^6 years (MY)
9	= 10^9 years (GY)

3.3.2. TAPE8: Cross-Section Library

- For a library number less than 700, use the standard ORIGEN2 library.

NLB TITLE (For the light-nuclide segment)
 NLB ZZAAAI Sn γ Sn2n Sn α Snp Sn γ x Sn2nx - 1.0

.
 .
 .
 -1

NLB TITLE (For the actinide segment)
 NLB ZZAAAI Sn γ Sn2n Sn3n Snf Sn γ x Sn2nx - 1.0

.
 .
 .
 -1

NLB TITLE (For the fission product segment)
 NLB ZZAAAI Sn γ Sn2n Sn α Snp Sn γ x Sn2nx 1.0
 Y(1) Y(2) Y(3) Y(4) Y(5) Y(6) Y(7) Y(8)

.
 .
 .
 -1

These entry fields are defined as follows:

NLB Library number for each segment.
 ZZAAAI Nuclide identification.
 Sn γ Cross section for the (n, γ) reaction.
 Sn2nx Cross section for the (n,2n) reaction leaving an excited daughter, etc.
 Y(1) Fission product yic'ds (%) from up to 8 actinides (see LIB command for more details).
 through The Y(1) through Y(8) line is omitted if the previous line ends in a negative number.
 Y(8)

- For a light-nuclide library number greater than 699, use the ORLIB format.

NLB TITLE (For a pure fusion problem)
 Comment line
 ZZAAAI Sn2n Sn2nx Sn3n Snp Snn α Snp Snd Snt Sn α Sn γ Sn γ x

-1

This example includes no cross sections for actinides or fission products. If the problem includes either, they are entered in the original ORIGEN2 format following the light-nuclide segment.

In both formats, the code will error exit if cross sections are entered for nuclides not in the appropriate decay-library segment. The input of TAPE8 is controlled by the LIB command in ORINPUT.

3.3.3. TAPE3: Odd-Reaction Cross-Section Library

- For a light-nuclide library number less than 700, input the odd reactions in the ORIGEN2 format.

ZZAAI ZZAAAI SIGMA

The first and second entries on each line are the parent and daughter nuclide identifications. When this format is used, hydrogen and helium isotope production is not calculated for the odd reactions.

- For a light-nuclide library number greater than 699, input the odd reactions in the ORLIB format.*

NLIB TITLE
 ZZAAAI ZZAAAI SIGMA TYPE
 ZZAAAI ZZAAAI SIGMA TYPE

TYPE is the reaction-type number (Table 1). With this format, hydrogen and helium isotope production is calculated for both the odd and the standard reactions.

*ORLIB is a cross-section manipulation code³ available with FORIG. The code can be used to energy average the ACTL cross-section library,⁴ using either an input flux or a TARTNP-generated flux⁵ as a weighting function. One improvement has been made to ORLIB since Ref. 3 was published; a flag has been added to the ORLIB input file to allow production of libraries in one of four formats (0 or no entry = FORIG light-nuclide format, 1 = ORIGEN2 light-nuclide format, 2 = actinide format, 3 = fission-product format without fission product yield-curve entries). This input flag is an I1 format in Column 30 of line 2 of the ORLIB input.

In either ORIGEN2 or ORLIB formats, each odd reaction is added to the first library segment (light nuclide, actinide, or fission product) that includes the parent in the decay library. Also, if replacement decay or cross-section lines are to be inserted in any of the six library segments, these lines *precede* the odd-reaction formats shown above. (The LIB and LPU commands force the code to read each line in its proper format, as is discussed later). Input for substitute decay or cross-section data is discussed in more detail in Sections 4.2.2 and 4.3.1.

3.3.4. TAPE10: Photon-Decay Data Library

- Input this library in the ORIGEN2 format.

```

NLB TITLE (For the light-nuclide segment)
NLB ZAAAAI NGP(1) RPH(1) ... NGP(J) RPH(J)
NLB NGP(J+1) RPH(J+1) ... NGP(18) RPH(18) -1

```

```

.
.
.

```

-1

These commands are defined as follows:

NGP Group number.

RPH Photon intensity per disintegration.

Only non-zero entries need be made. If actinides and fission products are used in the problem, their segments should be added to the example above. The photon-energy group structure is shown in Table 4. The input of the photon library is controlled by the PHO command in ORINPUT.

Table 4. Photon-energy group structures for activation products, actinides, and fission products.

Group	Group energy (MeV)		Average
	Lower boundary	Upper boundary	
1	0.0	0.02	0.01
2	0.02	0.03	0.025
3	0.03	0.045	0.0375
4	0.045	0.07	0.0575
5	0.07	0.1	0.085
6	0.1	0.15	0.125
7	0.15	0.3	0.225
8	0.3	0.45	0.375
9	0.45	0.7	0.575
10	0.7	1.0	0.85
11	1.0	1.5	1.25
12	1.5	2.0	1.75
13	2.0	2.5	2.25
14	2.5	3.0	2.75
15	3.0	4.0	3.5
16	4.0	6.0	5.0
17	6.0	8.0	7.0
18	8.0	11.0	9.5

3.3.5. ORINPUT: Instruction File

The input begins with four lines, each usually consisting of "-1". These negative entries suppress printing of neutron sources, decline to replace default fractional-recovery efficiencies of individual elements, decline to replace default fractional-recovery efficiencies of element groups, and decline to replace group-membership assignments of elements.

The file input continues with the individual commands, which are listed below by category. Detailed explanations are provided in Section 4.

3.3.5.1. Comment and Termination Commands.

RDA	Prints comments in the command list. (Comments may also be added at the end of most command lines.)
TIT	Specifies the title to be printed on the output table pages.
BAS	Prints a comment line on the output table pages. (For example, the comment can indicate that the calculation is based on one metric ton of steel in the reactor.)
HED	Allows a time-column heading to be changed from the actual time to a 10-character alphanumeric heading.
STP	Allows a new case to begin with new commands, new libraries, and/or initialization data.
END	Terminates execution.

3.3.5.2. Library Commands.

LIB	Inputs the decay and cross-section libraries. (The format of this command has been modified from the original ORIGEN2 format.)
LPU	Identifies nuclides requiring replacement lines in decay or cross-section library segments. One LPU command must precede the LIB command for each negative library number in the LIB command.
PHO	Inputs the photon data libraries.
LIP	Determines whether the decay, cross-section, and photon libraries are printed. The LIP command must precede the LIB command.

3.3.5.3. Material Input Commands.

INP	Forces the code to read the input material compositions. The program is set up to read the compositions at the end of the input command file (ORINPUT). However, a previous FORIG run could produce an output file in the proper format for input to this problem. If this output file is to be used directly, it should be named "TAPE4" and be opened in the FORIGM program. The INP command also allows input of continuous feed rates and continuous removal rates. These data follow the initial composition input at the end of ORINPUT.
WAC	Allows material to be continuously removed from the reactor with element-dependent efficiency. This waste accumulation can then be decayed.
PRO	Allows a material to be processed at a given time into two segments with element-dependent efficiency.
VOL	Inputs the material volume (to allow calculation of the shallow-burial index and the surface dose rate).

3.3.5.4. Material Manipulation Commands. FORIG maintains the material composition as a vector at each time step. Because only those vectors required for output or subsequent calculation need be kept, it is sometimes convenient to manipulate the vectors.

MOV	Allows a material composition to be moved from one vector to another. The command also allows a constant multiplier to be applied to the composition. The heading moves with the vector.
ADD	Allows addition of one material vector to a second material vector. The command also allows the material being added to be multiplied by a constant prior to the addition. The heading for the first vector will be moved to the second vector with this command.
KEQ	Allows two materials to be blended until the resultant material has an infinite neutron multiplication factor that matches either a third material or an input value.
FAC	Calculates the sum, product, or ratio of the total fission product plus actinide mass of two vectors or of a vector and an input constant. The resulting constant factor can then be used as a multiplier in subsequent ADD or MOV commands.

3.3.5.5. Commands that Take a Time Step

DEC	Allows a composition vector to decay for a specified time.
IRF	Allows irradiation and decay of a composition vector in a constant neutron flux for a specified time.
IRP	Allows irradiation and decay of a composition vector at a constant power for a specific time (only applied to fission problems).
BUP	Calculates the average burnup, flux, and specific power for an irradiation period (only applies to fission problems).

3.3.5.6. Loop Commands.

DOL	Sets up a "Do loop" through a set of instructions.
CON	Ends (continues) a Do loop.
REC	Counts the number of times (cycles) a Do loop has been executed (for printing in the output tables).

3.3.5.7. Output Commands.

OUT	Specifies composition vectors to be output.
OPTL	Specifies printing of <i>light-nuclide</i> output data for each vector in the OUT command. (Options include fractional or actual composition, activity, thermal-decay power, shallow-burial index, inhalation and ingestion biological-hazard potentials, and chemical-ingestion hazard. Output tables for neutron-absorption rate, neutron induced-fission rate, α -radioactivity, α -induced neutron production, and spontaneous-fission neutron production are automatically produced for the actinide segment. The photon-emission rate table is also automatically produced. The OPTL command also specifies the units of the various tables and whether the output will be by nuclide, element, and/or in a condensed format. The OPTL command must precede the OUT command.
OPTA	Output options for actinides (same format as OPTL).
OPTF	Output option for fission products (same format as OPTL).

CUT	Defines cutoff limits for condensed output tables. The nuclide is omitted if, for all the time steps being output on the page, the output for a particular nuclide is less than the product of the cutoff fraction and the total output at that time step.
PCH	Creates an output file containing the material composition. This file is in a format readable by future FORIG calculations.
RADS	Outputs the approximate surface-contact dose rate vs time.

3.4. FORIG Output Files

The following output files are generated by FORIG:

OROUT	Large file containing most of output (called TAPE6 in the ORIGEN2 manual).
TAPE12	Table of Contents for OROUT.
TAPE11	Abbreviated output file.
TAPE13	Table of Contents for TAPE11.
TAPE7	Output composition file (optional) written by the PCH command.
TAPE15	Debug information and error messages.
TAPE16	Actinide cross-section information.
TAPE50	Working file set up and used by FORIG.
ORPLOT	Lists total lines for the OROUT tables.

The "total" lines from the OROUT tables are repeated in ORPLOT to facilitate production of graphs. The OUT command, besides sending output to OROUT, can also specify sending the output to a second file (TAPE11 with table of contents TAPE13). The TAPE11 file is useful if an early, condensed hard-copy file is desired before generation of the full (OROUT) output, which is usually output as microfiche.

Sample FORIG and ORIGEN2 files are listed in Table 5.

3.5. General Code Information

3.5.1. FORIG Input Data in Free Format

With few exceptions, input data to FORIG can be specified in free format. The restrictions on free-format input are as follows:

- (1) All data must appear in the correct order.
- (2) All data must be of the correct type (e.g., integer or real) and may be in I, F, E, or D format.
- (3) Each datum must be separated from the next by at least one space.
- (4) Zero data values must appear explicitly (i.e., a blank is not equivalent to a zero).
- (5) In general, data may be continued onto multiple records if desired.
- (6) Certain data must appear as the first datum on a new record. These instances are described later.
- (7) The maximum record length for ORINPUT is 80 bytes.
- (8) If an end of file is read, control is returned to the calling subroutine.

Thus, in general, the data being read must be in the correct order, must begin on a new card when required, and must be separated by blanks. Other than this, the data may appear at any place on an input record. In the special case of numbers in E or D format (e.g., 3.8E 01), the space after the E is acceptable and is not considered as the end of the number.

3.5.2. Composition "Vectors" that Store and Output Nuclide Data

Before attempting to read the operational details of FORIG, the user must understand the concept of a FORIG vector. A FORIG vector is a one-dimensional array that specifies the amount of each nuclide being

Table 5. Sample FORIG and ORIGEN2 files. (The "take directory" of J. Blink at LLNL contains a number of sample FORIG, ORLIB, and ORIGEN2 files.)

File description	Name of file
Directory table of contents	CONTENTS (in Directory .080975:ORIGEN2)
ORLIB files (see Ref. 3)	ORLIB, ORLIBS, EORLIBIN, AC820312, and ADEX8203 ^a
FORIG source and controllee files	FORIGS, BFORIGS, FORIGMG, FJRIGM, and FORIG
Example FORIG input and data files	EORINPUT, ETAPE3, ETAPE8, ETAPE9, and ETAPE10
Table of contents for ORIL-supplied file list for ORIGEN2	OR01C
Sample ORIGEN2 main programs for various fission reactors	OR03C-OR18C (ORO3C was modified to run on FORIG)
ORIGEN2 decay library	OR21C
ORIGEN2 cross-section libraries for various fission reactors	OR22C-OR44C
ORIGEN2 photon library for UO ₂ L ₁ bremsstrahlung	OR45C
ORIGEN2 photon library for H ₂ O bremsstrahlung	OR46C
ORIGEN2 photon library for vacuum (no bremsstrahlung)	OR47C
Sample ORIGEN2 output files for a single fission-reactor problem (TAPES 6, 7, 11, 12, 13, 14, and 16)	OR48C-OR54C
Example FORIG input and output files for a fusion problem	2ORINT, 2OROTT

^aAC files are ACTL libraries created on the given year-month-day; ADEX files are ACTL library indices created on the given year-month; and EORLIBIN is an example ORLIB command file.

considered in a FORIG case. It is printed as a single column of numbers in FORIG output. If the problem being run includes actinide, activation-product, and fission-product nuclides, a vector might specify the amounts of the nuclides in each of these segments at a given time.

As shown in the diagram (see Fig. 2) of the conceptual vector organization of FORIG, two basic types of vectors are accessible to the user: output vectors and storage vectors.

The twelve output vectors contained in FORIG are written when FORIG output is produced. Each vector is designated by positive integers corresponding to the relative location of the vector, the left-most vector on the output page being vector 1 and the right-most being vector 12. The information in the output vectors is retained under all conditions except one. This exception occurs (1) when a new set of FORIG commands is read during a single run using the `stp` command ("Execute previous commands and branch") and (2) when the new set of commands includes a `lib` command ("Read new decay and cross-section data libraries"). Then, the array containing the output vectors is used as scratch space to read the new libraries and the nuclide mass data are lost.

A variable number (*LX*) of storage vectors in FORIG depend on the variable dimensions employed (see variable `JJJJ` in Fig. 1). These vectors store intermediate results and cannot be output. The vectors are

Figure 2. Organization of FORIG vectors.

designated by using negative integers from -1 to -LX. The information in the storage vectors is retained under all circumstances, including those where the output vectors are overwritten.

3.5.3. Card Input Echo

Included in FORIG is a subroutine LISTIT whose function is to provide a command echo. The commands are read from ORINPUT, printed on OROUT, and written to TAPE50, which is a temporary file. Commands with a dollar sign (\$) in the first column are printed (in OROUT) but not written (to TAPE50), thus allowing for the inclusion of comments in the input stream without interfering with the operation of the code. TAPE50 is then rewound, and the rest of FORIG reads the commands from TAPE50.

3.5.4. Nuclide Identifier

The FORIG nuclide identifier is a six-digit integer that uniquely defines a particular nuclide:

$$\text{NUCLID} = 100 * Z + 10 * A + IS$$

where NUCLID is a six-digit nuclide identifier, Z is the atomic number of the nuclide, A is the atomic mass of the nuclide, IS is the isomeric-state indicator (0 = ground state, 1 = excited state, and 2 or greater is not permitted).

Thus, the nuclide identifier for ^{137}Cs ($Z = 55$, $A = 137$) would be 551370. The trailing zero (or one) is always required. A leading zero, such as for tritium ($\text{NUCLID} = 010030 = 10030$), is not required.

The six-digit identifier for an element is given by

$$\text{ELEMID} = 10000 * Z$$

where ELEMID is the element identifier and Z is defined as above. Thus, the ELEMID for cesium would be 550000.

3.5.5. Machine Compatibility Considerations

FORIG is designed to be as machine-compatible as possible by using only the FORTRAN computer language, using only standard FORTRAN functions (e.g., SQRT, etc.), usually using H-format specifications for literal data in FORMAT and Data statements, and minimizing the number of partial-word (i.e., one-byte and two-byte word) arrays.

FORIG runs on the CRAY systems at LLNL and at NMFEC. If use of CDC or IBM machines is desired, compatible versions of the ORIGEN2 code are available from ORNL. A user could then compare FORIG to the proper ORIGEN2 version and build a code in a reasonable time.

4. Instruction File for FORIG Computer Code (ORINPUT)

The command file ORINPUT controls the execution of the FORIG computer code and is divided into three sections:

- Miscellaneous initialization data. (For fusion problems, the default data are usually adequate.)
- The sequence of commands for selecting cross-section libraries, for determining the flux-time history, and for selecting the output options.
- The composition of the input material.

4.1. Miscellaneous Data in the Source Code (FORIGS)

Miscellaneous data related only by their invariance from case to case are included here. These data are all initialized in the Data subroutine contained in the source code (FORIGS) and include the following:

- Fission neutron yields per neutron-induced fission.
- (α, n) neutron-production rates.
- Neutron yield per spontaneous fission.
- Fractional reprocessing recoveries for individual elements.
- Fractional reprocessing recoveries for element groups.
- Assignment of individual elements to fractional reprocessing recovery groups.
- Elemental chemical toxicities

4.1.1. Suppression of Neutron-Source Printing

If printing of the spontaneous-fission neutron yield, (α, n) neutron-production rate, and neutron-induced-fission neutron yield is desired, the first line of ORINPUT should be a non-negative integer beginning in column 3. If printing is not desired, the usual case for fusion problems, the first line should be a negative integer beginning in column 3.

4.1.1.1. Fission Neutron Yield per Neutron-Induced Fission. The Data subroutine supplies spectrum-weighted, single-group, fission neutron yields per neutron-induced fission for a thermal reactor (PWR-U) and for a fast reactor (advanced-oxide LMFBR). These data are used in calculating the infinite neutron multiplication factor for a mixture of nuclides. These data cannot be altered except by changing the values in the Data subroutine and by recompiling the source file (FORIGS).

4.1.1.2. (α, n) Neutron-Production Rate. The Data subroutine supplies measured (α, n) neutron-production rates (in units of neutrons per gram per second) for nuclides in oxide fuels. These values override values calculated with an empirical equation in FORIG. The (α, n) neutron-production rates for those nuclides not listed explicitly are calculated from an empirical equation. The parameters in the equation and the explicit values can only be altered by changing the values in the Data subroutine and recompiling it in the source code (FORIGS).

4.1.1.3. Fission-Neutron Yield per Spontaneous Fission. The Data subroutine supplies measured values for neutron yields per spontaneous fission, which override values calculated with an empirical equation in FORIG. These neutron yields, denoted as "SF yields", are used to calculate the decay neutron activity of nuclide mixtures. The SF neutron yields for those nuclides not given explicitly are calculated from an empirical equation. These initialization data can only be altered by changing the values in the Data subroutine and recompiling the source code (FORIGS).

4.1.2. Miscellaneous Input Data

If neutron-source printing is not desired and if chemical reprocessing will not be required in the problem, the miscellaneous data input section consists of four lines, each with a negative integer in Columns 3 and 4.

4.1.2.1. Fractional-Reprocessing Recoveries for Individual Elements. The Data subroutine supplies fractional-reprocessing recoveries (FR's) for each individual element. The FR's are used to separate a specified elemental composition into two separate streams. The individual element FR's initially present in FORIG are given in Table 6. A single FR set specifies an FR for each of 99 elements. There are ten sets of individual FR's in FORIG.

The individual FR sets also serve another purpose under certain circumstances. If one or more WAC (nuclide accumulation) commands are used, then at least one individual-element or element-group FR set (see the following section) must contain continuous-removal rates (s^{-1}) for the elements. The continuous-removal rates specified in the FR set are those appropriate for a reactor with continuous fuel reprocessing (e.g., a Molten-Salt Breeder Reactor or MSBR). The specified continuous-removal rates are used by the WAC command to generate equivalent continuous-feed rates of waste during waste decay.

In either of the above cases, the initial data can be altered by the following input lines:

```

NE(1)      NS(1)      FR(1)
.           .           .
.           .           .
.           .           .

NE(M)      NS(M)      FR(M)
.           .           .
.           .           .
.           .           .

NE(MMAX)   NS(MMAX)   FR(MMAX)
-1

```

Table 6. FORIG defaults for individual-element fractional recoveries.

Element	Fractional recoveries									
	Set 1	Set 2	Set 3	Set 4	Set 5	Set 6	Set 7	Set 8	Set 9	Set 10
H	0.0	0.0	0.0005	0.0	1.0	1.0	0.0	0.0	0.9	1.0
He	0.0	0.0	0.0005	0.0	1.0	1.0	0.0	0.0	0.0	1.0
Li-B	0.0	1.0	0.0005	0.0	1.0	1.0	1.0	0.0	0.0	1.0
C, N	0.0	0.0	0.0005	0.0	1.0	1.0	0.0	0.0	0.0	1.0
O	0.0	1.0	0.0005	0.0	1.0	1.0	1.0	0.0	0.0	1.0
F	0.0	0.001	0.0005	0.0	1.0	1.0	1.0	0.0	0.0	1.0
Ne	0.0	0.0	0.0005	0.0	1.0	1.0	0.0	0.0	0.0	1.0
Na-S	0.0	1.0	0.0005	0.0	1.0	1.0	1.0	0.0	0.0	1.0
Cl	0.0	0.001	0.0005	0.0	1.0	1.0	1.0	0.0	0.0	1.0
Ar	0.0	0.0	0.0005	0.0	1.0	1.0	0.0	0.0	0.0	1.0
K-Se	0.0	1.0	0.0005	0.0	1.0	1.0	1.0	0.0	0.0	1.0
Br	0.0	0.001	0.0005	0.0	1.0	1.0	1.0	0.0	0.0	1.0
Kr	0.0	0.0	0.0005	0.0	1.0	1.0	0.0	0.0	0.0	1.0
Rb-Te	0.0	1.0	0.0005	0.0	1.0	1.0	1.0	0.0	0.0	1.0
I	0.0	0.001	0.0005	0.0	1.0	1.0	1.0	0.0	0.0	1.0
Xe	0.0	0.0	0.0005	0.0	1.0	1.0	0.0	0.0	0.0	1.0
Cs-Ac	0.0	1.0	0.0005	0.0	1.0	1.0	1.0	0.0	0.0	1.0
Rn	0.0	0.0	0.0005	0.0	1.0	1.0	0.0	0.0	0.0	1.0
Fr-Ac	0.0	1.0	0.0005	0.0	1.0	1.0	1.0	0.0	0.0	1.0
Th	0.0	1.0	0.0005	0.0	1.0	1.0	1.0	0.0	0.0	1.0
Pa	0.0	1.0	0.0005	0.0	1.0	1.0	1.0	0.0	0.0	1.0
U	0.995	1.0	0.0005	0.999	1.0	0.2	0.6	1.0	0.0	1.0
Np	0.0	1.0	0.0005	0.0	1.05	0.05	0.0	0.0	0.0	1.0
Pu	0.995	1.0	0.0005	0.9999	1.0	0.02	0.0	0.0	0.0	1.0
Am-Es	0.0	1.0	0.0005	0.0	1.001	0.001	0.0	0.0	0.0	1.0

These data fields are defined as follows:

- NE(M)** Element atomic number (one or two digits, 1 to 99) for the fractional recovery on the *M*th line of this input.
- NS(M)** Set number (1 to 10) for the individual fractional recovery on the *M*th line.
- FR(M)** Fractional recovery value replacing the initial value for element **NE(M)** in set **NS(M)**; or, continuous removal rate for the **WAC** command.
- KMAX** Number of individual-element fractional-recovery values being overridden.

The negative entry terminates reading of the altered data. If no changes are desired, a single negative line instructs the code to continue to the next section.

4.1.2.2. Fractional-Reprocessing Recoveries for Element Groups. The Data subroutine supplies **FR** values for groups of elements. These group **FR** values can be employed in essentially the same manner as those for individual elements (discussed in the preceding section). That is, the group values can be used to separate a single specified elemental composition into two streams or to designate continuous-removal rates for the **WAC** command. The **FR** values for the groups initially present in **FORIG** are given in Table 7. **FORIG** can monitor up to 20 groups of elements. Each of the ten sets of group **FR**'s in **FORIG** specifies the **FR**'s for all the element groups.

The **FR**'s for the initial groups of elements can be altered by the following input lines:

```

NG(1)      NS(1)      FR(1)
.           .           .
.           .           .
.           .           .

NG(L)      NS(L)      FR(L)
.           .           .
.           .           .
.           .           .

NG(LMAX)   NS(LMAX)   FR(LMAX)
-1

```

Table 7. **FORIG** defaults for element-group fractional recoveries.

Group	Fractional recoveries									
	Set 1	Set 2	Set 3	Set 4	Set 5	Set 6	Set 7	Set 8	Set 9	Set 10
1	0.0	1.0	0.0005	0.0	1.0	1.0	1.0	0.0	0.0	0.0
2	0.0	1.0	0.0005	0.0	1.0	1.0	0.0	0.0	0.0	0.0
3	0.0	1.0	0.0005	0.0	1.0	1.0	0.0	0.0	0.0	0.0
4	0.995	1.0	0.0005	0.999	1.0	0.2	0.6	1.0	0.0	0.0
5	0.0	1.0	0.0005	0.0	0.05	0.05	0.0	0.0	0.0	0.0
6	0.995	1.0	0.0005	0.9999	1.0	0.02	0.0	0.0	0.0	0.0
7	0.0	1.0	0.0005	0.0	0.001	0.001	0.0	0.0	0.0	0.0
8	0.0	1.0	0.0005	0.0	0.001	0.001	0.0	0.0	0.0	0.0
9	0.0	1.0	0.0005	0.0	0.001	0.001	0.0	0.0	0.0	0.0
10	0.0	1.0	0.0005	0.0	0.001	0.001	0.0	0.0	0.0	0.0
11	0.0	1.0	0.0005	0.0	0.001	0.001	0.0	0.0	0.0	0.0
12	0.0	0.001	0.0005	0.0	1.0	1.0	0.0	0.0	0.0	0.0
13	0.0	0.0	0.0005	0.0	1.0	1.0	0.0	0.0	0.0	0.0
14	0.0	0.0	0.0005	0.0	1.0	1.0	0.0	0.0	0.9	0.0
15-29	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

These data fields are defined as follows:

- NG(I)** Element group number (one or two digits, 1 to 20) for the fractional recovery on line I of this input section.
- NS(I)** Set number (1 to 10) for the fractional recoveries of the element-group on line I.
- FR(I)** Fractional recovery value replacing the initial value for group NG(I) in set NS(I); or, the continuous removal rate for the WAC command.
- IMAX** Number of group fractional-recovery values being overridden.

The negative entry terminates reading of the altered data. If no changes are desired, a single negative line instructs the code to continue to the next section.

4.1.2.3. Assignment of Elements to Fractional-Recovery Groups. The Data subroutine also assigns each of the 99 elements to one of the 20-element groups discussed in the preceding section. Any number of elements may be assigned to a given group, but an individual element can be a member of only one group. The initial membership of the FORIG element group is given in Table 8.

The assignment of elements to FR groups can be altered by using the following input lines:

```

NE(I)      NG(I)
.           .
.           .
.           .

NE(I)      NG(I)
.           .
.           .
.           .

NE(IMAX)   NG(IMAX)
-1

```

These data fields are defined as follows:

- NE(I)** Element atomic number (one or two digits, 1 to 99) on line I of this input section.
- NG(I)** Element group number (one or two digits, 1 to 20) where element NE(I) is to be assigned.
- IMAX** Number of element assignments being overridden.

Table 8. Composition of FORIG default-element groups.

Group	Elements in group	Group	Elements in group
1	All elements except those in groups 2-14	9	Bk
2	Th	10	Cf
3	Pa	11	Es
4	U	12	F, Cl, Br, I
5	Np	13	He, C, N, Ne, Ar, Kr, Xe, Rn
6	Pu	14	H
7	Am	15-20	None
8	Cm		

The negative entry terminates reading of the altered data. If no changes are desired, a single negative line instructs the code to continue to the next input section.

4.1.2.4. Elemental Chemical Toxicities. The Data subroutine supplies maximum permissible concentrations (MPCs in water) for each of the chemical elements. The MPC is used to calculate the volume of water required to dilute a given amount of an element to a concentration corresponding to its MPC. The sum of the volumes of water required for the elements in a mixture is assumed to yield the total volume of dilution water required and thus is a measure of the chemical toxicity of the elemental mixture. This measure is sometimes called the chemical biological hazard potential (bhp). The MPC data can only be altered by changing the values in the Data subroutine and recompiling the source code (FORIGS).

4.2. FORIG Commands

The FORIG commands (instructions) allow the user to precisely define the order of execution of the FORIG program functions. That is, the commands define a series of operations to be performed sequentially, the sequence being varied according to the option of the user. In this respect, the procedure is analogous to writing a FORTRAN program. The FORIG program flowsheet may use "DO loop" commands, which execute sets of instructions within the loop range a prescribed number of times. Coupled with other options, this allows the user to investigate the fuel-recycle rates (e.g., plutonium or tritium) and nuclear-fuel-cycle waste-production rates as a function of time.

The general format for the FORIG commands is

`COM PARM( 1 ) PARM( 2 ) PARM( I ) ,`

where COM represents a keyword that defines the instruction type and PARM(I) represents parameters that supply data necessary for execution of operational commands. Note that the command keyword always introduces the command. The FORIG commands and their functions are listed in Section 3 and are described in detail in the following.

Before you attempt to use FORIG, please note that certain restrictions are imposed on the order in which the commands must occur. The primary restriction is that the LIB command, which reads the decay and cross-section libraries, must precede most other commands because it defines the list of nuclides being considered. Other restrictions are noted in the individual command descriptions.

The number of times a FORIG command can occur in a single input stream is limited, the maximum limit depending on the specific command. This limit is specified in the section that describes each individual command. The limits can be changed by varying the dimensions of the appropriate array(s) within the FORIG source file (FORIGS) and then recompiling. The limit on the total number of FORIG commands that may be used is 300, a number which can also be changed by varying array dimensions within the source file (FORIGS) and then recompiling.

4.2.1. Comment and Termination Commands

RDA *Read Comments Regarding Case Being Input*

This command prints alphanumeric comments in the listing of the operational commands being input. The data sequence is

RDA COMMENT

COMMENT = Alphanumeric message.

The only limit on the number of RDA commands is that the maximum total number of commands cannot exceed 300. These RDA comments are printed in the listing created when FORIG is interpreting the commands.

TIT***Case Title***

This command supplies a title to be printed on most pages of the FORIG output. The data sequence is

TIT A(9) . . . A(80)

A(I) = Alphanumeric characters in columns 9 to 80 only.

The code allows up to 20 TIT commands. The TIT command is in effect until it is changed by a subsequent TIT command.

BAS***Case Basis***

This command supplies the case basis (e.g. 1 kg, 1 ton, all the material, etc.) printed on most pages of the FORIG output. It is purely a comment and has no influence on the code execution. The data sequence is

BAS A(9) . . . A(80)

A(I) = Alphanumeric characters in columns 9 to 80 only.

Up to ten BAS commands are allowed. The BAS message is printed until it is changed by a subsequent BAS command.

HED***Composition Vector Headings***

This command allows alphanumeric vector headings to be specified. The default is time (e.g., 1 yr after startup or shutdown). Example headings include "shutdown" and "startup." The data sequence is

HED NHED A(1) . . . A(10)

NHED = Number of vector to be given a heading.

A(I) = 10-character alphanumeric heading, placed anywhere to the right of NHED.

Up to 50 HED commands are allowed. The heading remains in effect until it is changed by a subsequent HED command or until the vector is overwritten (e.g., by an IRF or DEC command). The heading is moved with the vector when the MOV and ADD commands are used.

If a HED command will label either a vector of input concentrations [vector NINP(1)] or the vectors resulting from a PRO command [vectors NPRO(2) and NPRO(3)], the HED command must follow the INP or PRO command.

If A(1) is an apostrophe or asterisk (*), the ten characters immediately following A(1) are taken as the vector heading. This permits the inclusion of leading blanks.

STP***Execute Previous Commands and Branch***

This command allows multiple problems to be executed sequentially in the same run. The set of commands preceding the STP command is executed; then more commands are read. There is no limit to the number of STP commands. The data sequence is

STP NSTP

NSTP = Branching control character:

- 1 = Read new miscellaneous initialization data and a new set of commands and execute them.
- 2 = Read a new set of commands and execute them.
- 3 = Execute the preceding set of commands again. Additional input data (libraries and initial nuclide concentrations) is required.
- 4 = Terminate execution (same as **END**).

END *Terminate Execution*

This command terminates execution of **FORIG**. Only one **END** command is allowed. The data sequence is

END

4.2.2. Library Commands

LIB *Read Decay and Cross-Section Libraries*

This command reads decay and cross-section libraries, as well as substitute decay and cross-section entries and nonstandard-reaction (odd-reaction) cross sections.

The libraries supplied by ORNL with **ORIGEN2** are listed in Table 9. For fusion problems, the user must supply the cross-section libraries (e.g., by running **TARTNP**⁵ to obtain the flux and then running **ORLIB**³ to energy-average the **ACTL** cross-section library,⁴ using the flux as a weighting function). The data sequence is

LIB **NLIB(1)** . . . **NLIB(13)**

- NLIB(1)** = Control character. The reaction rate matrix is printed if **NLIB(1)** is greater than zero. (That is, the production rate from each parent, the decay rate, and the removal rate by activation are printed for each nuclide.) Printing the matrix allows determination of the pathways that dominate the activation; without the matrix, only the overall activation results are available.
- NLIB(2)** = The light-nuclide decay library number.
- NLIB(3)** = The actinide decay library number (zero for pure fusion problems).
- NLIB(4)** = The fission-product decay library number (zero for pure fusion problems).
- NLIB(5-7)** = The three cross-section library numbers. (Note: If a decay or cross-section library number is negative, replacement cards will also be read later in the odd-reaction file for that library.)
- NLIB(8)** = The input unit number for the cross-section library file (use 8 for **TAPE8**).
- NLIB(9)** = The input unit number for the decay library file (use 9 for **TAPE9**).
- NLIB(10)** = The input unit number for the replacement decay and cross-section lines and for reading odd reactions (use 3 for **TAPE3**).
- NLIB(11)** = The number of odd reactions to be read. If **NLIB(11)** is negative, the data will be read on unit **NLIB(10)**. If **NLIB(11)** is positive, the data will be read on unit **NLIB(8)**. (Use negative **NLIB(11)** to read the odd reactions, i.e., use **TAPE3**.)

Table 9. Identification numbers of FORIG data libraries.

File ^a	Type of library	Activation product [NLIB(2 or 5)]	Actinide [NLIB(3 or 6)]	Fission product [NLIB(4 or 7)]	Variable actinide NLIB(13)
Decay		1	2	3	—
Photon	UO ₂ Bremsstrahlung	101	102	103	—
	H ₂ O Bremsstrahlung	104	105	106	—
	NO Bremsstrahlung	107	108	109	—
<u>Cross-section libraries</u>					
22	PWR: ²³⁵ U-enriched UO ₂ ; 33,000 MWd/metric ton	204	205	206	1
23	PWR: ²³⁵ U-enriched UO ₂ in a self generated Pu recycle reactor	207	208	209	2
24	PWR: Pu-enriched UO ₂ in a self-generated Pu recycle reactor	210	211	212	3
30	BWR: ²³⁵ U-enriched UO ₂	251	252	253	4
31	BWR: ²³⁵ U-enriched fuel in a self-generated Pu recycle reactor	254	255	256	5
32	BWR: Pu-enriched fuel in a self-generated Pu recycle reactor	257	258	259	6
25	PWR: ThO ₂ -enriched with denatured ²³³ U	213	214	215	7
28	PWR: Pu-enriched ThO ₂	216	217	218	8
29	PWR: ²³⁵ U-enriched UO ₂ ; 50,000 MWd/metric ton	219	220	221	9
27	PWR: ThO ₂ -enriched with makeup denatured ²³⁵ U	222	223	224	10
26	PWR: ThO ₂ enriched with recycled, denatured ²³³ U	225	226	227	11
LMFBR:	Early oxide, LWR-Pu/U/U/U				
34	Core	301	302	303	18
35	Axial blanket	304	305	316	19
36	Radial blanket	307	308	309	20
LMFBR:	Advanced oxide, LWR-Pu/U/U/U				
37	Core	311	312	313	12
38	Axial blanket	314	315	316	13
39	Radial blanket	317	318	319	14
LMFBR:	Advanced oxide, recycle-Pu/U/U/U				
40	Core	321	322	323	15
41	Axial blanket	324	325	326	16
42	Radial blanket	327	328	329	17
33	Thermat: 0.0253-eV cross sections	201	202	203	0
43	CANDU: Natural uranium feed	401	402	403	—
44	Slightly enriched feed	404	405	406	—

^aThe file number refers to the file number on the RSIC-supplied ORIGEN2 tape, and in J. Blink's LLNL OCTOPUS take directory, .R:ORIGEN2 (e.g., file 22 is stored as OR22C where the "OR" stands for ORIGIN and the "C" stands for CRAY).

NLIB(12) = The set of actinides (Table 10) having direct fission-product yields (use zero for fusion problems).

NLIB(13) = The set of variable actinide cross sections (Table 9) to be used (use zero for fusion problems).

Up to five LIB commands are allowed. The libraries remain in effect until another LIB command is read.

If substitute decay or cross-section lines are to be read, the LPU command(s) must precede the LIB command in which the lines are to be read. The formats for the decay and cross-section libraries were discussed in Section 3. The LIB command must precede the material input (INP) commands.

LPU

Data Library Replacement Cards

This command reads nuclide identifiers for replacement decay and cross-section data lines that will later be read by the LIB command. The data sequence is

LPU NLPU(1) . . . NLPU(MAX) -1

NLPU(I) = Nuclide identifiers for replacement data lines in the order they occur in the original data library.

MAX = Number (up to 100) of nuclide identifiers to be read for a given LPU command.

Up to nine LPU commands are allowed. The LPU data remain valid until a second LIB command is read. If less than 100 nuclide identifiers are specified, a "-1" must appear after the last identifier. As many lines may be used as are required.

The LPU command(s) must precede the LIB command in which the replacement data cards will be read. The first LPU command is associated with the first *negative* library number in the NLIB(2)-to-NLIB(7) set of library numbers in "the LIB command". The second LPU command is associated with the second *negative* library number, etc.

PHO

Read Photon Libraries

This command reads the photon production rate per disintegration in 18 energy groups. The data sequence is

PHO NPHO(1) . . . NPHO(4)

NPHO(1) = Identification number of the activation-product photon library to be read (see Table 9).

NPHO(2) = Identification number of the actinide-nuclide photon library to be read (see Table 9).

Table 10. Actinide sets with direct fission-product yields.

NLIB(12)	Actinides								
	²³² Th	²³³ U	²³⁴ U	²³⁵ U	²³⁸ U	²³⁹ Pu	²⁴¹ Pu	²⁴⁵ Cm	²⁵² Cf
1				X	X	X	X		
2	X	X		X		X			
3	X	X		X	X	X	X		
4	X	X	X		X	X	X	X	X

NPHO(3) = Identification number of the fission-product nuclide photon library to be read (see Table 9).

NPHO(4) = Number of input unit on which the photon libraries are to be read (use 10 for TAPE10).

If **NPHO(1)**, **NPHO(2)**, or **NPHO(3)** is less than zero, no photon library is read for that segment. Up to five **PHO** commands are allowed. The libraries remain in effect until a subsequent **PHO** command is read. The photon library format was described in Section 3.

LIP

Library Print Control

This command controls the printing of the input data libraries. Up to five **LIP** commands are allowed. The **LIP** command must precede the **LIB** command. The data sequence is

LIP NLIP(1) NLIP(2) NLIP(3)

NLIP(1) = Control character for decay library print.

NLIP(2) = Control character for cross-section library print.

NLIP(3) = Control character for photon library print.

If an **NLIP** entry is zero, the library will not be printed; if an entry is positive, that library will be printed.

4.2.3. Material Input Commands

INP

Read Input Composition, and Continuous Removal and Feed Rates

The **INP** command provides read locations and units for input nuclide composition, continuous nuclide-feed rate, or continuous element-removal rate. The actual format for the data is discussed later in Section 4.3.2.

The initial composition can be input in the final (third section) of the **ORINPUT** file. Alternatively, it can be input by a separate (**TAPE4**) file if the main program **FORIGM** is altered to open **TAPE4** and then is recompiled. Either gram or gram-atom mass units can be used. The data sequence is

INP NINP(1) . . . NINP(6)

NINP(1) = Number of the vector in which the initial compositions are to be stored.

NINP(2) = Initial nuclide composition:

= 0, not read.

= 1, in grams (read at end of **ORINPUT**).

= 2, in g-atoms (read at end of **ORINPUT**).

= -1, in grams (read in **TAPE4**).

= -2, in g-atoms (read in **TAPE4**).

NINP(3) = Continuous nuclide-feed rate:

= 0, not read.

= 1, in grams/(time); see **NINP(5)** for time units.

= 2, in g-atoms/(time); see **NINP(5)** for time units.

NINP(4) = Element-removal rate:

< 0, not read.

= 0, not read but propagates previously read values.

> 0, reads **NINP(4)** data pairs (see Section of 4.3.2.3 for details); see **NINP(6)** for time units.

NINP (5) = Time units of continuous nuclide-feed rate data (see Table 3).

NINP (6) = Time units of continuous element-removal rate data (see Table 3).

Up to 15 **INP** commands are allowed. It is recommended that the initial composition data be entered as either 1.0 g or 1.0 g-atom. Then, the actual mass can be specified with the **MOV** command, and the composition entries can be reused in subsequent runs having different masses but unchanged initial compositions. The **BAS** command provides a repeated reminder (in the output) of the amount of mass in the calculation (of course the comment must agree with the input mass).

WAC

Waste Nuclide Accumulation

This command multiplies a concentration vector by a fractional recovery vector and stores the result, which subsequently can be used as a continuous feed rate to a waste system. The data sequence is

WAC NWAC (1) NWAC (2)

NWAC (1) = Number of the fractional recovery vector to be used to multiply concentration vector **NWAC (2)**. Fractional recovery **NWAC (1)** should contain the removal rate (sec^{-1}) of each element from the system (equivalent to the feed rate to the waste system being analyzed). A positive value selects an individual element fractional-recovery set (Section 4.1.2.1); a negative value selects a group fractional-recovery set (Sections 4.1.2.2 and 4.1.2.3).

NWAC (2) = Number of concentration vector that is to be multiplied by fractional recovery vector **NWAC (1)**.

Up to two **WAC** commands are allowed. The **WAC** command remains in effect until a subsequent **WAC** or **LIB** command is input. Once reactor equilibrium is attained, the **WAC** command calculates the continuous accumulation of waste from a reactor with continuous reprocessing (e.g., an MSBR). The steady-state fuel composition in vector **NWAC (2)** is multiplied by the appropriate continuous-removal rates stored in fractional recovery vector **NWAC (1)**. Then, the waste can be decayed.

A typical sequence of waste accumulation commands follows:

WAC 1 3 Calculate continuous feed rate to the waste system from vector 3 and recovery set 1.

MOV 2 2 0 0 0 Initialize vector 2 to all zeros.

DEC 3 0 2 3 5 2 Add three years of waste accumulation and decay and place result in vector 3.

WAC End of **WAC** sequence.

The result in vector 3 is based on continuous feed for three years coupled with continuous decay of the accumulating material.

PRO

Reprocess Fuel

This command reprocesses fuel (or other material) into two product compositions. The data sequence is

PRO NPRO (1) . . . NPRO (4)

NPRO(1) = Number of vector where composition of the material to be reprocessed is stored.

NPRO(2) = Number of vector where amount of recovered material will be stored. The recovered amount of an isotope of element **NE** is given by

$$[\text{Mass of isotope NE}] [f(\text{NPRO}(4))]$$

The fraction $f[\text{NPRO}(4)]$ is the fractional recovery of element **NE** specified by variable **NPRO(4)** (see below).

NPRO(3) = Number of vector where amount of unrecovered material will be stored. The amount of an isotope of element **NE** that is not recovered is given by

$$[\text{Mass of isotope NE}] [1.0 - f(\text{NPRO}(4))]$$

NPRO(4) = Number of set for fractional recoveries that are to be used in this reprocessing operation. If **NPRO(4)** is greater than zero, individual fractional recoveries are to be used. If **NPRO(4)** is less than zero, group fractional recoveries are to be used. (See Section 4.1 for a description of fractional recovery data.)

Up to 20 **PRO** commands are allowed. The reprocessing action is an instantaneous activity. If continuous separation is desired, it can be approximated by using either the **WAC** or the **INP** commands.

VOL *Material Volume*

This command inputs the material volume to allow calculation of the shallow-burial index and the surface (gamma) dose rate. It must precede the **OUT** and **RADS** commands. The data sequence is

VOL RVOL

RVOL = Material volume in m^3 .

Up to 20 **VOL** commands are allowed. The **VOL** command remains in effect until changed by a new **VOL** command.

4.2.4. Material Manipulation Commands

FORIG maintains the material composition as a vector at each specified time step. Since only those vectors required for output or subsequent calculation need be stored, it is sometimes convenient to manipulate the vectors.

MOV *Move Nuclide Composition from Vector to Vector*

Moves (i.e., copies) the nuclide concentration data in one composition vector to another vector, nuclide by nuclide. The data sequence is

MOV NMOV(1) NMOV(2) NMOV(3) RMOV(1)

NMOV(1) = Number of the vector where the concentration to be moved is presently stored.

NMOV(2) = Number of the vector where the concentration in vector **NMOV(1)** will be moved; it may be the same as **NMOV(1)**.

NMOV(3) = Source of additional multiplier:

- > 0, number of the variable multiplier vector that contains the additional factors for multiplying vector **NMOV(1)** before moving it to vector **NMOV(2)**. The variable multipliers are in array **RMULV** and are initialized with a Data statement in **FORIGM**. The particular element of **RMULV** used is

$$RMULV[NREC, NMOV(3)]$$

where **NREC** is the recycle number (from the **REC** command). The total multiplier **RMULT** is given by

$$RMULT = RMULV[NREC, NMOV(3)] * RMOV(1)$$

NREC must be defined to use the variable multiplier option.

- = 0, no additional multiplier is used; that is,

$$RMULT = RMOV(1)$$

- < 0, an additional multiplier is used, which was previously calculated by an **FAC** command and designated as **FACTOR[NFAC(1)]** at that time. To use this factor, set **NMOV(3) = -NFAC(1)**; the total multiplier is then given by

$$RMULT = FACTOR[-NMOV(3)] * RMOV(1)$$

RMOV(1) = Factor by which vector **NMOV(1)** is to be multiplied before being stored in vector **NMOV(2)**, subject to the conditions specified by **NMOV(3)** above.

Up to 99 **MOV** commands are allowed. An example **MOV** command is

MOV -1 1 0 3.56E02

which moves the composition in vector "-1" (which could be 1 gram if the **INP** command filled vector "-1") to vector "1", after multiplying it by 356 (i.e., vector 1 contains 356 grams). In another example, vector **NMOV(2)** can be zeroed by moving a vector to **NMOV(2)** with **RMOV(1) = 0.0**. The information in vector **NMOV(1)** is not destroyed by the **MOV** command. Vector **NMOV(2)** will have the same heading as vector **NMOV(1)** after the **MOV** command is executed.

ADD

Add Two Composition Vectors

This command adds the nuclide concentration data in one vector to that in another vector, nuclide by nuclide. The data sequence is

ADD NADD(1) NADD(2) NADD(3) RADD(1)

NADD(1) = Number of the vector where the concentration to be added is presently stored.

NADD(2) = Number of the vector where the concentration in vector **NADD(1)** will be added.

NADD(3) = Source of additional multiplier:

- > 0, number of the variable multiplier vector that contains the additional factors by which vector **NADD(1)** is to be multiplied before being added to vector

NADD(2). The variable multipliers are in array RMULV and are initialized with a Data statement in FORIGM. The particular element of RMULV used is

$$RMULV[NREC, NADD(3)]$$

where NREC is the recycle number (from the REC command). The total multiplier, RMULT, is given by

$$RMULT = RMULV[NREC, NADD(3)] * RADD(1)$$

NREC must be defined to use this option.

= 0, no additional multiplier used; that is,

$$RMULT = RADD(1)$$

< 0, this additional multiplier to be used was previously calculated by a FAC command and designated as FACTOR[NFAC(1)]. To use this factor, set NADD(3) = -NFAC(1); the total multiplier is then given by

$$RMULT = FACTOR[-NADD(3)] * RADD(1)$$

RADD(1) = Factor by which vector NADD(1) is to be multiplied before being added to vector NADD(2), subject to the conditions specified by NADD(3) above.

Up to 30 ADD commands are allowed. Vector NADD(1) may be subtracted from vector NADD(2) by setting RADD(1) = -1.0. **(CAUTION: Negative nuclide concentrations can result in fatal errors and ungraceful termination!)** The information in vector NADD(1) is not altered by the ADD command. After the ADD command has been executed, vector NADD(2) will have the same headings as vector NADD(1).

KEQ *Match Infinite Multiplication Factors*

Blend materials in two vectors so that the resulting infinite (neutron) multiplication factor (IMF) matches that of another vector or an input value. The data sequence is

$$KEQ \ NKEQ(1) \ . \ . \ . \ NKEQ(5) \ RKEQ(1)$$

NKEQ(1) = Vector whose IMF will be matched by vector NKEQ(4).

NKEQ(2) = Vector whose material will be wholly included in the final blended material in vector NKEQ(4).

NKEQ(3) = Vector whose material is to be apportioned to obtain the proper IMF for vector NKEQ(4).

NKEQ(4) = Vector containing all material in vector NKEQ(2) plus part of the material in NKEQ(3) and having the same IMF as either vector NKEQ(1) or RKEQ(1); that is,

$$NKEQ(4) = NKEQ(2) + f * NKEQ(3)$$

where f is the factor by which NKEQ(3) must be multiplied to obtain the correct IMF for NKEQ(4).

NKEQ(5) = vector containing the portion of NKEQ(3) not blended into NKEQ(4); that is,

$$NKEQ(5) = (1 - f) * NKEQ(3)$$

If $(1 - f)$ is less than zero, then $NKEQ(5)$ is set to zero.

RKEQ(1) = Desired IMF data source:

- > 0, the desired final IMF for vector $NKEQ(4)$.
- = 0, the IMF is equal to $RMULV(NREC, 1)$. The $RMULV$ values are specified in a Data statement in **FORIGM**; the $NREC$ parameter is from the **RRC** command.
- < 0, the IMF of vector $NKEQ(4)$ is matched to that of vector $NKEQ(1)$.

Up to 3 **KEQ** commands are allowed. The equation used to calculate f , the parameter for multiplying vector $NKEQ(3)$ before being combined with material in vector $NKEQ(2)$ and being placed in vector $NKEQ(4)$, is given by

$$f = [(k_2 - k_1) * D_2] / [(k_1 - k_3) * D_3]$$

k_1 = IMF to be matched from vector $NKEQ(1)$ or $RKEQ(1)$.

k_2 = IMF of material in vector $NKEQ(2)$.

k_3 = IMF of material in vector $NKEQ(3)$.

D_2 = Neutron absorption rate (neutrons/s) of material in vector $NKEQ(2)$.

D_3 = Neutron absorption rate (neutrons/s) of material in vector $NKEQ(3)$.

TAPE15 file (Debug Information and Error Messages) contains some of the intermediate information calculated by the **KEQ** command.

FAC

Calculate a Multiplication Factor Based on Total Vector Masses

This command calculates a multiplication factor, **FACTOR[NFAC(1)]**, based on the total actinide plus fission product masses in one or two vectors, for use in **MOV** or **ADD** commands. The data sequence is

FAC NFAC(1) . . . NFAC(4) RFAC(1)

NFAC(1) = Number of factor calculated by this command (must be greater than zero and less than or equal to the maximum number of **FAC** commands).

NFAC(2) = Vector number.

NFAC(3) = Vector number.

NFAC(4) = Method for calculating **FACTOR[NFAC(1)]**:

- = 1, **FACTOR[NFAC(1)]** = $T[NFAC(2)] + T[NFAC(3)]$
- = 2, **FACTOR[NFAC(1)]** = $T[NFAC(2)] - T[NFAC(3)]$
- = 3, **FACTOR[NFAC(1)]** = $T[NFAC(2)] * T[NFAC(3)]$
- = 4, **FACTOR[NFAC(1)]** = $T[NFAC(2)] / T[NFAC(3)]$
- = 5, **FACTOR[NFAC(1)]** = $T[NFAC(2)]$
- = 6, **FACTOR[NFAC(1)]** = $T[NFAC(3)]$
- = 7, **FACTOR[NFAC(1)]** = $1.0 / T[NFAC(2)]$
- = 8, **FACTOR[NFAC(1)]** = $1.0 / T[NFAC(3)]$

where the $T[NFAC(I)]$ is the total (kg) fission-product-plus-actinide mass for the indicated vectors.

RFAC(1) = Constant value (kg) used in place of $T[NFAC(I)]$:

- > 0, substitute **RFAC(1)** for $T[NFAC(2)]$ when calculating **FACTOR[NFAC(1)]**.
- = 0, use the $T[NFAC(I)]$ as defined.
- < 0, substitute $[-RFAC(I)]$ for $T[NFAC(3)]$ when calculating **FACTOR[NFAC(1)]**.

Up to 20 FAC commands are allowed. The result, NFAC(1), remains in effect until another FAC command with the same value of NFAC(1) is executed. Some intermediate results from this command are printed in the TAPE15 output file (Debug Information and Error Messages).

4.2.5. Commands that Take a Time Step

DEC *Decay*

This command allows a composition vector to decay for a specified time. The data sequence is

DEC DEC(1) NDEC(1) . . . NDEC(4)

- DEC(1) = Time at which this decay interval ends.
- NDEC(1) = Number of the vector where the material composition at the beginning of this decay interval is stored.
- NDEC(2) = Number of the vector where the material composition at the end of this decay interval is stored.
- NDEC(3) = Time units of DEC(1) (see Table 3).
- NDEC(4) = Specification of time at which this decay interval begins:
 - = 0, starting time is the end of the previous IRF, IRP, or DEC interval. All reactivity and burnup information is retained, and the MIRR array is not altered (used for continuing irradiation/decay on the same output page).
 - = 1, starting time is set to zero. All reactivity and burnup information is retained, and MIRR is set to zero (used for beginning a new irradiation on the same output page).
 - = 2, starting time is set to zero. All reactivity and burnup information and MIRR are set to zero (used to begin a new irradiation/decay on a new output page).
 - = 3, same as NDEC(4) = 0, except the first seven lines of the reactivity and burnup information are set to zero (used for continuing irradiation to a new output page).
 - = 4, same as NDEC(4) = 1, except the first seven lines of the reactivity and burnup information are set to zero (used to begin a new output page for the decay following irradiation while at the same time retaining the average irradiation parameters).

The total number of IRF + IRP + DEC commands must not exceed 150. The "reactivity and burnup information" referred to in NDEC(4) consists of seven lines of data characteristic of an individual vector (e.g., time, infinite multiplication factor, neutron flux) and three lines containing irradiation parameters (e.g., burnup) averaged over the range of the BUP commands. For fusion problems, the "reactivity and burnup information" is largely irrelevant, and NDEC(4) is 2 when the clock is reset to zero; when the clock is not reset (but just continues), NDEC(4) should be zero.

IRF *Flux Irradiation*

This command allows irradiation for a single interval with the neutron flux specified. The data sequence is

IRF RIRF(1) RIRF(2) NIRF(1) . . . NIRF(4)

RIRF(1) = Time at which this irradiation interval ends.

RIRF(2) = The neutron flux (neutrons/cm²/s) during this irradiation interval:
> 0, the flux is the specified value.
< 0, the flux is given by:

$$NEWFLUX = OLDFLUX * [-RIRF(2)]$$

NEWFLUX = Flux (neutrons/cm²/s) to be used during this interval.

OLDFLUX = Flux (neutrons/cm²/s) for the same time period but from the previous irradiation sequence.

For this option to be used, the time steps for the current irradiation and decay sequence must correspond *exactly* to those in the previous sequence. The scale factors [-RIRF(2)] from previous irradiations *do* propagate.

NIRF(1) = Number of the vector where the material composition at the beginning of this irradiation interval is stored.

NIRF(2) = Number of the vector where the material composition at the end of this irradiation interval will be stored.

NIRF(3) = Time units of RIRF(1) (see Table 3).

NIRF(4) = Specification of time at which this irradiation interval begins: Values of 0, 1, 2, 3, or 4 are permitted. The definitions are identical to those of the DEC command.

The total number of IRF + IRP + DEC commands must not exceed 150. Intermediate parameters related to the flux/specific power calculations are printed in TAPE15 (Debug Information and Error Messages).

IRP

Specific Power Irradiation

This command allows activation and decay of a composition vector at a constant power for a specified time interval. It only applies to standard fission-reactor problems. The data sequence is

IRP RIRP(1) RIRP(2) NIRP(1) . . . NIRP(4)

RIRP(1) = Time at which this irradiation interval ends.

RIRP(2) = Power level during this irradiation interval.

> 0, thermal energy (MW_t) per unit of fuel input.

< 0, specific power, MW_t per metric ton of heavy metal.

(NOTE: Both actinides and fission products must be present for the specific power option to work correctly.)

NIRP(1) = Number of the vector where the material composition at the beginning of this irradiation interval is stored.

NIRP(2) = Number of the vector where the material composition at the end of this irradiation interval is to be stored.

NIRP(3) = Time units of RIRP(1) (see Table 3).

NIRP(4) = Specification of the time at which this irradiation interval begins: Values of 0, 1, 2, 3, or 4 are permitted. The definitions are identical to those of the DEC command. The total number of IRF + IRP + DEC commands must not exceed 150. Intermediate parameters related to the flux/specific power calculations are printed in TAPE15 (Debug Information and Error Messages).

BUP *Burnup Calculation*

This command defines the basis and calculates the average burnup, flux, and specific power for an irradiation in a fission reactor. The data sequence is

**BUP
IRRADIATION
BUP**

IRRADIATION = the operational commands (generally several **IRPs** or **IRFs**) that describe the fuel irradiation to be used as the basis for the burnup calculation.

Up to twenty (10 pairs) of **BUP** commands are allowed. The resulting values remain valid until superseded by other **BUP** commands.

A **BUP** command must appear both *before* and *after* the irradiation operational commands on which the burnup calculation is to be based. Other commands may be present between the **BUP** statements. The **BUP** sequence is required to properly use the variable cross sections available for fission problems.

4.2.6. Loop Commands

DOL *Do Loop*

This command initiates a "Do loop" which executes the commands within its range a prescribed number of times. The data sequence is

DOL NDOL (1) NDOL (2)

NDOL (1) = Number of the **CON** command that defines the range of this Do loop. Each **DOL** must have a unique **CON** associated with it.

NDOL (2) = The total number of times the instructions within the loop are to be executed.

Up to two **DOL** commands are allowed.

CON *Continuation (End of Do Loop)*

This command defines the range of a **DOL** command. Up to two **CON** commands are allowed. There must be one, and only one, **CON** command for each **DOL** command. The data sequence is

CON NCON

NCON = Number of this **CON** command; must be equal to **NDOL (1)** for the **DOL** command that is associated with this **CON** command.

REC *Loop Counter*

This command counts the number of times a loop (**DOL** command) is executed. The data sequence is

REC

Only one **REC** command is allowed. It must precede the first do loop to which it applies. This counter is output as the "Recycle #" in the **FORIG** output.

4.2.7. Output Commands

OUT *Print Calculated Results*

This command calls for the calculated results in some or all of the output vectors to be printed. Up to 20 OUT commands are allowed. The data sequence is

OUT NOUT(1) . . . NOUT(4)

NOUT(1) = Number of vectors to be printed (beginning with first vector. The absolute value of NOUT(1) cannot exceed 12.

> 0, output in file OROUT.

< 0, output in file TAPE11.

NOUT(2) = Frequency of print if instruction is in a Do loop. Printing occurs the first time through loop and every NOUT(2)th recycle thereafter. A REC command is required if this entry is not 1.

NOUT(3) = Print number of present recycle.

> 0, yes.

≤ 0, no.

NOUT(4) = Type of summary table to be printed. (Vectors are tested in accordance with the CUT command.)

> 0, only vector NOUT(4) tested to see if a nuclide is included in the summary table.

= 0, all vectors tested for inclusion in summary table.

< 0, all vectors tested for inclusion in summary table except vector "-NOUT(4)".

OPTL *Specify Activation-Product Output Options*

This command specifies the output tables (1 through 24) and the table types (nuclide, element, or summary) to be printed for the activation products. The data sequence is

OPTL NOPTL(1) . . . NOPTL(24)

NOPTL(I) = Control character indicating the output-table types to be printed for the activation products (see Table 11); I = table number (see Table 12 for output-table description).

Table 11. Types of output tables to be printed.

NOPTL(I) NOPTA(I) NOPTF(I)	Table type—printed? (yes or no)		
	Nuclide	Element	Summary
1	Yes	Yes	Yes
2	Yes	Yes	No
3	Yes	No	Yes
4	No	Yes	Yes
5	Yes	No	No
6	No	Yes	No
7	No	No	Yes
8	No	No	No

Table 12. Description of FORIG output tables.

Table No.	Description of Table	Units
1	Isotopic composition of each element	atom fraction
2	Isotopic composition of each element	weight fraction
3	Composition	gram-atoms
4	Composition	atom fraction
5	Composition	grams
6	Composition	weight fraction
7	Radioactivity (total)	Ci
8	Radioactivity (total)	fractional
9	Thermal power	watts
10	Thermal power	fractional
11	Shallow burial index	dimensionless ^a
12	Shallow burial index	fractional
13	Radioactive Inhalation hazard	m ³ air
14	Radioactive Inhalation hazard	fractional
15	Radioactive ingestion hazard	m ³ water
16	Radioactive ingestion hazard	fractional
17	Chemical ingestion hazard	m ³ water
18	Chemical ingestion hazard	fractional
19	Neutron absorption rate	neutrons/sec
20	Neutron absorption rate	fractional
21	Neutron-induced fission rate	fissions/sec
22	Neutron-induced fission rate	fractional
23	Radioactivity (alpha)	Ci
24	Radioactivity (alpha)	fractional
25	(alpha,n) neutron production	neutrons/sec
26	Spontaneous fission neutron production	neutrons/sec
27	Photon emission rate	photons/sec
28	Set test parameter ERR	—

^aTotal values less than 1.0 indicate the material is suitable for disposal by shallow land burial. Total values exceeding 1.0 indicate the need for either dilution or geologic (high-level) waste disposal.

Up to 20 OPTL commands are allowed. The command remains in effect until it is superseded by a new OPTL command.

The nuclide tables include results for each nuclide, the element tables include results summed by element, and the summary tables (by nuclide and element) include only the most significant results (as determined by the CUT command). The NOPTL(1) must all be on a single line.

If NOPTL(1) is less than one, only a summary table (grams) is printed for all nuclides (including actinides and fission products) until new commands (after an STP command) are read. Only the first 24 tables listed in Table 12 are controlled by the OPTL command. The OPTL command must precede the first OUT command to which it applies.

OPTA *Specify Actinide Output Options*

This command specifies the output tables (1 through 24) and the table types (nuclide, element, or summary) to be printed for the actinide nuclides. The data sequence is

OPTA NOPTA(1) . . . NOPTA(24)

NOPTA(I) = control character indicating the output-table types to be printed for the actinide nuclides (see Table 11); I = table number (see Table 12).

Up to 20 OPTA commands are allowed. The command remains in effect until it is changed by a new OPTA command. The NOPTA(I) must all be on a single line.

If NOPTA(1) is less than 1, only a summary (grams) table is printed for all nuclides (including activation and fission products) until new commands (after a STP command) are read. Only the first 24 tables listed in Table 12 are controlled by the OPTA command. The OPTA command must precede the first OUT command to which it applies.

OPTF *Specify Fission Product Output Options*

This command specifies the output tables (1 through 24) and the table types (nuclide, element, or summary) to be printed for fission product nuclides. The data sequence is

OPTF NOPTF(1) . . . NOPTF(24)

NOPTF(1) = The output-table types to be printed for the fission product nuclides (see Table 11); I = table number (see Table 12).

Up to 20 OPTF commands are allowed. The command remains in effect until it is changed by a new OPTF command. The NOPTF(I) must all appear on a single line.

If NOPTF(1) is less than 1, only a summary (grams) table is printed for all nuclides (including activation products and actinides) until new commands (after a STP command) are read. Only the first 24 tables in Table 12 are controlled by the OPTF command. The OPTA command must precede the first OUT command to which it applies.

CUT *Cutoff Fractions for Summary Tables*

This command overrides default cutoff fractions for summary output tables. The data sequence is

CUT NCUT(1) RCUT(1) . . . NCUT(NT) RCUT(NT) -1

NCUT(I) = Output table number to which cutoff fraction RCUT(I) is to apply (see Table 12 for table numbers and descriptions).

RCUT(I) = New cutoff fraction for table number NCUT(I).

NT = Total number of default cutoff values that are being overridden with this CUT command.

Up to three CUT commands are allowed. The CUT command remains in effect until changed by a subsequent CUT command. The CUT command must precede the first OUT command to which it applies.

If an output value for a particular nuclide is less than the cutoff fraction multiplied by the total (all nuclides) value for each vector being tested, then that particular nuclide is not printed.

Table 28 can be used to override the default value for ERR, presently set at 1.0E-25. ERR is used in logical IF statements instead of 0.0.

An integer (-1) must follow RCUT(NT) unless all 28 cutoff fractions are specified. If only the defaults are to be used, a "CUT -1" line is required. The default cutoff fractions for the first 26 tables (see Table 12) are 0.001; for the 27th table, the cutoff is 0.01.

The "NCUT(I) RCUT(I)" pairs may continue onto subsequent cards; no command keyword is used on the additional cards.

The application of the cutoff value to photon tables only applies to the "principle contributor" tables; the photon summation tables include no cutoff tests.

PCH "Punch" an Output Vector

This command writes a designated output vector to a FORIG-readable file, TAPE7. The data sequence is

PCH NPCH(1) NPCH(2) NPCH(3)

NPCH(1) = Control character for light nuclide and structural material (activation product) output.

NPCH(2) = Control character for actinide nuclide output.

NPCH(3) = Control character for fission product nuclide output.

With NPCH(1):

= 0, no output.

> 0, number of output vector to be output.

< 0, number of storage vector to be output.

Up to 54 PCH commands are allowed. The *punched* output is in g-atom units. The last record (line) written by each PCH command is

B BURNUP FLUX SPECIFIC POWER

The burnup, flux, and specific power are average values produced by the BUP command and *must* be present for a file read on TAPE4 (in a subsequent problem). These parameters are not necessary for input material compositions read in ORINPUT (in a subsequent problem).

RADS Surface Gamma-Dose-Rate Output

This command calculates the surface dose rate from gamma radiation for 12 output vectors. The dose rate only includes decay gamma rays; the dose rate from (n, γ) reactions is not calculated. The data sequence is

RADS RRAD(1) RRAD(2) RRAD(3)

RRAD(1) = The material density (g/cm³) is the absolute value of RRAD(1). If RRAD(1) is negative, the output is to TAPE11, otherwise the output is to OROUT.

RRAD(2) = The material thickness (cm).

RRAD(3) = The material type (FE, AL, MO, W, or PB).

The RADS command must follow the OUT command that covers the same time intervals (i.e., output vectors). Also, the RADS command must be preceded by a VOL command. Up to 20 RADS commands are allowed. The user must select the material type that most closely approximates the actual material.

The command uses RRAD(1), (2), and (3) to perform a crude photon-transport calculation from an infinite-slab gamma source. (The source is generated by the OUT command). Since Compton scattering is approximated as pure absorption by using only one-half the total cross section, the calculation is only by an order-of-magnitude. Yet it does allow an easy estimate of the maintainability of a fusion reactor; a formal calculation would require a shielding code.

4.3. Input at the End of ORINPUT

4.3.1. Substitute Decay or Cross-Section Library Data

Substitute decay, cross-section, and fission-product yield data can be read by invoking the LPU command. This procedure is an alternative to having to rebuild an entire data library in order to change a few parameters. It may also be used for parametric studies of output sensitivity to nuclear data changes.

If NLIB(10) from the LIB command is 3, the substitute data is read at the beginning of TAPE3. If NLIB(10) is 5, the data is read at the end of the ORINPUT file (after the last command, END, but before the initial material compositions).

An LPU command must be supplied for each library requiring substitute lines. (The LPU commands must precede the LIB command.) The LPU command lists the nuclides to be replaced in the order in which they appear in the library. Libraries requiring substitute lines are indicated by negative library numbers in the LIB command [NLIB(2) through (7)]. The LPU commands are in the same sequence as the LIB data entries:

- (1) Substitute activation-product decay data.
- (2) Substitute actinide-decay data.
- (3) Substitute fission-product-decay data.
- (4) Substitute activation product cross-section data.
- (5) Substitute actinide cross-section data.
- (6) Substitute fission-product cross-section and yield data.

The format for each substitute line is identical to the format of the appropriate library. A replacement fission-product-yield card can never appear alone; it must always follow a cross-section card for the same nuclide.

4.3.2. Specification of Materials and Rates

In this section, we describe the options available to the user relative to the specification of the initial material compositions, the continuous nuclide-feed rates, and the continuous element-removal (reprocessing) rates. The most-often-used option by far is the specification of the initial composition of some material. The initial composition can be specified on either a nuclide-by-nuclide basis or as amounts of naturally occurring elements. The amount of a naturally occurring element is converted to a nuclide-by-nuclide basis internally, using the natural isotopic abundances input in the decay library. The amounts of individual nuclides or naturally occurring elements may be specified in grams-atoms or grams, depending on the control characters of the INP command.

The continuous-nuclide-feed option allows the user to specify the continuous feed rate of individual nuclide or naturally occurring elements. Both the mass units (grams or gram-atoms) and the time units are specified by using the INP command. This option is useful in stimulating the continuous feed of nuclides to a fluid-fuel reactor (e.g., a MSBR) or to a radioactive waste tank.

The continuous-element-removal option allows the user to specify the continuous removal rates (fraction/time unit) of elements during irradiation. The time units are specified by the INP command. This option is most useful when simulating the continuous reprocessing that would be expected to occur during the operation of a fluid-fuel reactor. If this option will be used to calculate continuous element removal in the absence of irradiation, a small (negligible) neutron flux must be specified.

4.3.2.1. Specification of Initial Material Composition. This input data specifies initial amounts of individual nuclides or naturally occurring elements. The data sequence is

```
NEXT NUCLID(1) RCOMP(1) . . . NUCLID(JMAX) RCOMP(JMAX)
```

- NEXT = Segment for which the information is intended and the type of information (i.e., nuclides or elements):
- = 1, individual activation-product nuclides.
 - = 2, individual actinide nuclides.
 - = 3, individual fission-product nuclides.

- = 4, naturally occurring activation-product elements.
- = 5, naturally occurring actinide elements.
- = 6, naturally occurring fission-product elements.

NUCLID(J) = Six-digit identifier (ZZAAAI) for nuclide (or element) J.

RCOMP(J) = Amount of nuclide (or element) **NUCLID(J)** initially present. The units of **RCOMP(J)** are specified with the **INP** command.

JMAX = The maximum number of **NUCLID(J)**, **RCOMP(J)** pairs specified on each line. **JMAX** must not exceed four. If **JMAX** is less than four, a zero must be added to the line.

A line with a single zero entry instructs the code to proceed to the next section of input (if any).

If a given nuclide is specified more than once for a single value of **NEXT**, all **RCOMP(J)** values for that nuclide, on lines having this same **NEXT** value, are added together.

Initial composition lines with different **NEXT** values may occur in any order as long as the **NUCLID(J)** and **RCOMP(J)** values on any given line correspond to the **NEXT** value on that line.

4.3.2.2. Specification of Continuous Nuclide Feed Rates. This input data specifies constant-feed rates of individual nuclides or naturally occurring elements. The data sequence is

NEXT NUCLID(1) RRATE(1) . . . NUCLID(JMAX) RRATE(JMAX)

NEXT = Segment for which the information is intended and the type of information:

- = 1, individual activation-product nuclides.
- = 2, individual actinide nuclides.
- = 3, individual fission-product nuclides.
- = 4, naturally occurring activation-product elements.
- = 5, naturally occurring actinide elements.
- = 6, naturally occurring fission-product elements.

NUCLID(J) = The six-digit nuclide identifier (ZZAAAI) for nuclide (or element) J.

RRATE(J) = The feed rate of nuclide (or element) **NUCLID(J)**. The units of **RRATE(J)** are specified with the **INP** command.

JMAX = Maximum number of **NUCLID(J)**, **RRATE(J)** pairs specified on each card; **JMAX** must not exceed four. If **JMAX** is less than four, a zero must be added to the line. A line with a single zero entry instructs the code to proceed to the next input section (if any).

If the feed rate of a given nuclide is specified more than once for a single value of **NEXT**, the total feed rate for **NUCLID(J)** is computed by adding together all of the nuclide's **RRATE(J)** values that are on lines having that particular **NEXT** value.

Continuous-feed-rate lines with different **NEXT** values may occur in any order as long as the **NUCLID(J)** and **RRATE(J)** values and the **NEXT** value on the same line all correspond.

4.3.2.3. Specification of Continuous Element-Removal Rates. In this data section, the user specifies continuous-element-removal rates during irradiation. The data sequences are as follows:

Group 1 (1-card set)

RREM(1) NPROS(1) . . . RREM(M) NPROS(M) . . .
RREM(MMAX) NPROS(MMAX)

Group 2 [MMAX card sets (M = 1 to MMAX)]

NZ(M,1) . . . NZ(M,N) . . . NZ[M,NPROS(M)]

RREM(M) = The removal rate of elements **NZ(M,1)** through **NZ(M,NPROS(M))**. The units of **RREM(M)** are specified with the **INP** command.

NPROS(M) = The number of elements in card set **M** of Group 2; that is, the number of elements with a continuous removal rate equal to **RREM(M)**.

MMAX = The number of continuous reprocessing rates to be read. Also, the number of card sets in Group 2. **MMAX** is specified as **NINP(4)** with the **INP** command.

NZ(M,N) = The two-digit atomic number of an element (e.g., He = 02) with removal rate **RREM(M)**.

Continuous element removal will occur only during irradiation. If continuous removal is desired in a situation where no neutron flux is present, use the **IRF** command with a negligibly small flux.

5. Organization of FORIG Output Files

FORIG produces a voluminous but well-organized output. In this section we describe each output file in detail and use a sample fusion problem, activation of a $\text{Li}_7\text{Pb}_{83}$ coolant, to demonstrate the output files. Users desiring a sample fission-reactor problem should consult the user's manual for the ORIGEN2 Computer Code.¹

5.1. Sample Fusion-Problem Input File (ORINPUT)

For the sample fusion problem, we used the input command file (ORINPUT), shown in Fig. 3. The example also required the data libraries (TAPE9, TAPE8, TAPE3, and TAPE10) and the controller (FORIG).

The individual commands are discussed in detail in Section 4.2. In this section, we show, by example, how the commands can be combined to perform the desired calculation.

```

-1
-1
-1
-1
bas    all material in zone 1, 1982 wppm'bi in pb 20rotl
cul -1
lip 1 1 1
lib 1 1 0 0 720 0 0 8 9 3 -67 0 0
pho 107 0 0 10
optl 1 1 8 8 1 1 1 1 1 1 1 1 1 1 4 4 8 8 8 8 8 8
opto 24*8
optf 24*8
inp -1 1 -1 -1 1 1
tit    benchmark 1d problem for po build up during opn
mov -1 1 0 6.3150e+09
hed 1 fresh
vol 683.00
irf 1. 4.2575e14 1 2 3 2 2 is after 1 hour
irf 3. 4.2575e14 2 3 3 0
irf 10. 4.2575e14 3 4 3 0
irf 1. 4.2575e14 4 3 4 0 3 is after 1 day
irf 3. 4.2575e14 3 5 4 0
irf 10. 4.2575e14 5 4 4 0 4 is after 10 days
irf 30. 4.2575e14 4 5 4 0 5 is after 1 month
irf 90. 4.2575e14 5 6 4 0 6 is after 3 months
irf 180. 4.2575e14 6 7 4 0 7 is after 6 months
irf 1. 4.2575e14 7 8 5 0 8 is after 1 year
irf 3. 4.2575e14 8 9 5 0 9 is after 3 years
irf 5. 4.2575e14 9 10 5 0 10 is after 5 years
irf 10 4.2575e14 10 12 5 0
irf 21 4.2575e14 12 11 5 0 11 is after 21 years
irf 30 4.2575e14 11 12 5 0 12 is after 30 years
out 12 1 -1 0
rads 9.246 20. pb
optl 8 8 8 8 8 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
out -12 1 -1 0
mov 12 1 0 1.0
hed 1 discharge
tit    benchmark 1d problem for po build up after opn
dec 1. 1 2 1 2
dec 3. 2 3 1 0
dec 10. 3 2 1 0
dec 30. 2 3 1 0
dec 100. 3 2 1 0
dec 300. 2 3 1 0
dec 900. 3 2 1 0
dec 1800. 2 3 1 0

```

Figure 3. Example of ORINPUT file (continued on next page).

```

dec 1. 3 2 3 0 2 is after 1 hour
dec 3. 2 4 3 0
dec 10. 4 5 3 0
dec 1. 5 3 4 0
dec 3. 3 5 4 0
dec 10. 5 4 4 0
dec 30. 4 5 4 0
dec 90. 5 7 4 0
dec 180. 7 8 4 0
dec 1. 8 3 5 0 3 is after 1 year
dec 3. 3 5 5 0
dec 5. 5 4 5 0 4 is after 5 years
dec 10. 4 5 5 0 5 is after 10 years
dec 15. 5 6 5 0 6 is after 15 years
dec 20. 6 7 5 0 7 is after 20 years
dec 25. 7 8 5 0 8 is after 25 years
dec 30. 8 9 5 0 9 is after 30 years
dec 35. 9 10 5 0 10 is after 35 years
dec 40. 10 11 5 0 11 is after 40 years
dec 45. 11 12 5 0 12 is after 45 years
out -12 1 -1 0
optl 1 1 8 8 1 1 1 1 1 1 1 1 1 1 4 4 8 8 8 8 8 8
out 12 1 -1 0
rads 9.246 20. pb
mov 12 2 0 1.0
dec 50. 2 1 5 0 1 is after 50 years
dec 55. 1 2 6 0 2 is after 55 years
dec 60. 2 3 5 0 3 is after 60 years
dec 65. 3 4 5 0 4 is after 65 years
dec 70. 4 5 5 0 5 is after 70 years
dec 75. 5 6 5 0 6 is after 75 years
dec 80. 6 7 5 0 7 is after 80 years
dec 85. 7 8 5 0 8 is after 85 years
dec 90. 8 9 5 0 9 is after 90 years
dec 95. 9 10 5 0 10 is after 95 years
dec 100. 10 11 5 0 11 is after 100 years
dec 300. 11 12 5 0 12 is after 300 years
out 12 1 -1 0
rads 9.246 20 pb
optl 8 8 8 8 5 3 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
out -12 1 -1 0
end
4 260000 .085038 0 0.0
1 30060 .004867 30070 .000630 0 0.0
4 820000 .907662 830000 .001803 0 0.0
0

```

The first four lines of ORINPUT (see Fig. 3) operate to decline printing of the neutron sources and to accept the default chemical reprocessing data tables (which are not required for this particular calculation). The commands after the first four lines are described in the following:

- BAS** Reminder of the mass included in the calculation. Because it will be printed on almost every page, the BAS command can also include a user-selected calculation identifier (2OROUT in this case).
- CUT** Accepts the default cutoff levels for the summary tables. Nuclides with levels below the cutoff in *all* columns of the table will not be printed.
- LIP** Forces printing of the decay, cross-section, and photon libraries (22 pages of output in this example). For parametric studies that use the same libraries, the printing can be suppressed by zeroing the arguments.

LIB	Determines the decay and cross-section library numbers. If these library numbers are not present in TAPE9 and TAPE8, respectively, the code will error exit. The LIB command also states the number of odd reactions to be read from TAPE3. The first argument forces the printing of the activation/decay matrix (17 pages of output in this example). This matrix is useful in determining which parents contribute to the production of a particular nuclide.
PHO	Determines the photon library number to be read from TAPE10.
OPTL , OPTA , OPTF	Determine which tables (grams, curies, etc.) will be printed and in OPTA , which formats (by nuclide, by element, or by summary using the OPTF cutoff levels).
INP	Instructs the code to read the initial composition (from the end of the ORINPUT file) into vector "-1" but not to read continuous feed or removal rates. The composition is entered in weight-fraction units; hence, at this point, 1.0 gram of material is being considered.
TIT	Determines the title to be printed on most pages of the output.
MOV	Multiplies the input composition (vector "-1") by 6.3×10^9 and then stores it in vector "1". This use of the INP and MOV commands allows a change in the mass to be calculated without editing the (lengthy) composition listing.
HED	Changes the column heading of vector "1" to "fresh" (rather than a numerical time heading).
VOL	Establishes the material volume in cubic meters.
IRF	Allows passage of time with a constant neutron flux. The content of vector "1" is irradiated and stored in vector "2"; vector "2" is irradiated and stored in vector "3"; and "3" is irradiated and stored in vector "4". Vector "4" is irradiated, and the result overwrites vector "3". This procedure allows a reasonable increase in the time step (<3 is a good rule of thumb) while saving only the desired output vectors. Once the IRF commands are completed, 12 vectors (the maximum that can be output on one page) are ready to be output.
OUT , RADS	Sends the 12 output vectors to file OROUT in the formats determined by the most recent OPTL , OPTA , and OPTF commands. Then, the RADS command sends approximate dose rates to OROUT . Finally, the OPTL command is altered to a minimum amount of output, and the vectors are sent to file TAPE11. File OROUT is designed to be sent to microfiche; it contains 350 pages in this example. File TAPE11 is designed to be printed as it is much shorter.
MOV	This MOV command copies the final composition (vector "12") into vector "1". The HED and TIT commands change the vector "1" heading and the title, respectively.

In the above sequence, the first set of **DEC** commands allows the passage of more time but without a neutron flux. Once 12 output vectors accumulate, they are output to **OROUT** and TAPE11 using the **OUT**, **OPTL**, **OUT**, and **RADS** commands. The decay process is continued with the third **MOV** command, another set of **DEC** commands, and a final set of output commands. Finally, the **END** command instructs the code to stop execution. The **END** command is followed by the composition data requested by the earlier **INP** command.

5.2. TAPE12: The Table of Contents for File **OROUT**

The complete TAPE12 for the fusion-problem example is given in Fig. 4. In this section, we discuss the organization of TAPE12 (and hence **OROUT**). The individual output segments are discussed in the next section.

The output begins with a standard set of information: a few code description pages, a listing of the commands (as read), the chemical reprocessing data, and the chemical toxicities of the elements. Then, the instructions are repeated as they are "compiled" (but not yet executed). Next, the data libraries are listed.

1	page	table of contents on unit = 12 for output unit = 6
1		forig overview and documentation
4		input echo; read on 5 list on 6 copy to 50
6		individual element fractional recoveries
8		group elemental fractional recoveries
9		elemental assignment to frac recovery group
10		elemental chemical toxicities
11		forig instructions for this case
13		nuclide data libraries
13		decay data library----- decay library: activation product
27		cross section library-----uses ac840216,bench2 1d,1cm fe at 5
33		add reactions
35		non-zero matrix elements--
52		daughters not in the decay library
59		photon library----- photon lib: activation prod
70		output tables--title=benchmark 1d problem for po build up during opn
70		reactivity and burnup data
		*activation products*****activation products*****activation pro
71		isotopic composition of each element, atom fraction n
75		isotopic composition of each element, weight fraction n
79		concentrations, grams nuclide table:
83		concentrations, grams element table:
84		concentrations, grams summary table:
86		concentrations, fractional grams nuclide table:
90		concentrations, fractional grams element table:
91		concentrations, fractional grams summary table:
93		radioactivity, curies nuclide table:
96		radioactivity, curies element table:
97		radioactivity, curies summary table:
99		radioactivity, fractional curies nuclide table:
102		radioactivity, fractional curies element table:
103		radioactivity, fractional curies summary table:
105		thermal power, watts nuclide table:
108		thermal power, watts element table:
109		thermal power, watts summary table:
111		thermal power, fractional watts nuclide table:
114		thermal power, fractional watts element table:
115		thermal power, fractional watts summary table:
117		shallow burial index, per cent nuclide table:
120		shallow burial index, per cent element table:
121		shallow burial index, per cent summary table:
123		shallow burial index, fractional per cent nuclide table
126		shallow burial index, fractional per cent element table
127		shallow burial index, fractional per cent summary table
129		radioactive inhalation hazard, m**3 air of rpg nuclide t
132		radioactive inhalation hazard, m**3 air of rpg element t

Figure 4. Example of TAPE12 (OROUT Table of Contents) (continued on next page).

②

1	page	table of contents on unit = 12 for output unit = 6
133	radioactive inhalation hazard, m**3 air at rcg	summary t
135	radioactive inhalation hazard, fractional m**3 air at rcg	n
138	radioactive inhalation hazard, fractional m**3 air at rcg	e
139	radioactive inhalation hazard, fractional m**3 air at rcg	s
141	radioactive ingestion hazard, m**3 water at rcg	nucii
144	radioactive ingestion hazard, m**3 water at rcg	eleme
145	radioactive ingestion hazard, m**3 water at rcg	summa
147	radioactive ingestion hazard, fractional m**3 water at rcg	
150	radioactive ingestion hazard, fractional m**3 water at rcg	
151	radioactive ingestion hazard, fractional m**3 water at rcg	
153	chemical ingestion hazard, m**3 water	element table:
154	chemical ingestion hazard, m**3 water	summary table:
156	chemical ingestion hazard, fractional m**3 water	element t
157	chemical ingestion hazard, fractional m**3 water	summary t
159	activation product photon table	
164	activation product dose rate table	
170	output tables--little=benchmark 1d problem for po build up after opn	
170	reactivity and burnup data	
	*activation products*****activation products*****activation pro	
171	isotopic composition of each element, atom fraction	n
175	isotopic composition of each element, weight fraction	n
179	concentrations, grams	nuclide table:
183	concentrations, grams	element table:
184	concentrations, grams	summary table:
186	concentrations, fractional grams	nuclide table:
190	concentrations, fractional grams	element table:
191	concentrations, fractional grams	summary table
193	radioactivity, curies	nuclide table:
196	radioactivity, curies	element table:
197	radioactivity, curies	summary table:
199	radioactivity, fractional curies	nuclide table:
202	radioactivity, fractional curies	element table:
203	radioactivity, fractional curies	summary table:
205	thermal power, watts	nuclide table:
208	thermal power, watts	element table:
209	thermal power, watts	summary table:
211	thermal power, fractional watts	nuclide table:
214	thermal power, fractional watts	element table:
215	thermal power, fractional watts	summary table:
217	shallow burial index, per cent	nuclide table:
220	shallow burial index, per cent	element table:
221	shallow burial index, per cent	summary table:
223	shallow burial index, fractional per cent	nuclide table
226	shallow burial index, fractional per cent	element table
227	shallow burial index, fractional per cent	summary table
229	radioactive inhalation hazard, m**3 air at rcg	nuclide t
232	radioactive inhalation hazard, m**3 air ol rcg	element t
233	radioactive inhalation hazard, m**3 air at rcg	summary l

②

Figure 4. (continued on next page).

1	page	table of contents on unit = 12 for output unit = 6
235		radioactive inhalation hazard, fractional m**3 air at rcg n
238		radioactive inhalation hazard, fractional m**3 air at rcg e
239		radioactive inhalation hazard, fractional m**3 air at rcg s
241		radioactive ingestion hazard, m**3 water at rcg nucli
244		radioactive ingestion hazard, m**3 water at rcg eleme
245		radioactive ingestion hazard, m**3 water at rcg summa
247		radioactive ingestion hazard, fractional m**3 water at rcg
250		radioactive ingestion hazard, fractional m**3 water at rcg
251		radioactive ingestion hazard, fractional m**3 water at rcg
253		chemical ingestion hazard, m**3 water element table:
254		chemical ingestion hazard, m**3 water summary table:
256		chemical ingestion hazard, fractional m**3 water element t
257		chemical ingestion hazard, fractional m**3 water summary t
259		activation product photon table
265		activation product dose rate table
272	output tables--title=benchmark 1d problem for po build up after opn	
272		reactivity and burnup data
		activation products*****activation products*****activation pro
273		isotopic composition of each element, atom fraction n
276		isotopic composition of each element, weight fraction n
279		concentrations, grams nuclide table:
282		concentrations, grams element table:
283		concentrations, grams summary table:
285		concentrations, fractional grams nuclide table:
288		concentrations, fractional grams element table:
289		concentrations, fractional grams summary table:
291		radioactivity, curies nuclide table:
292		radioactivity, curies element table:
293		radioactivity, curies summary table:
295		radioactivity, fractional curies nuclide table:
296		radioactivity, fractional curies element table:
297		radioactivity, fractional curies summary table:
299		thermal power, watts nuclide table:
300		thermal power, watts element table:
301		thermal power, watts summary table:
303		thermal power, fractional watts nuclide table:
304		thermal power, fractional watts element table:
305		thermal power, fractional watts summary table:
307		shallow burial index, per cent nuclide table:
308		shallow burial index, per cent element table:
309		shallow burial index, per cent summary table:
311		shallow burial index, fractional per cent nuclide table
312		shallow burial index, fractional per cent element table
313		shallow burial index, fractional per cent summary table
315		radioactive inhalation hazard, m**3 air at rcg nuclide t
316		radioactive inhalation hazard, m**3 air at rcg element t
317		radioactive inhalation hazard, m**3 air at rcg summary t
319		radioactive inhalation hazard, fractional m**3 air at rcg n

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Figure 4. (continued on next page).

1	page	table of contents on unit = 12 for output unit = 6
320		radioactive inhalation hazard, fractional m**3 air at rcg e
321		radioactive inhalation hazard, fractional m**3 air at rcg s
323		radioactive ingestion hazard, m**3 water at rcg nucli
324		radioactive ingestion hazard, m**3 water at rcg eleme
325		radioactive ingestion hazard, m**3 water at rcg summa
327		radioactive ingestion hazard, fractional m**3 water at rcg
328		radioactive ingestion hazard, fractional m**3 water at rcg
329		radioactive ingestion hazard, fractional m**3 water at rcg
331		chemical ingestion hazard, m**3 water element table:
332		chemical ingestion hazard, m**3 water summary table:
334		chemical ingestion hazard, fractional m**3 water element t
335		chemical ingestion hazard, fractional m**3 water summary t
337		activation product photon table
341		activation product dose rate table

Finally, the main body of the output begins. This information is organized in a hierarchical fashion. For each OUT command, there is a title, reactivity and burnup data, the activation product tables, the actinide tables, and finally the fission product tables (if requested). These are followed by the neutron production tables for actinides and the photon production tables for activation products, actinides, and fission products.

Within a segment (e.g., activation products), the requested tables (1 through 24) are printed in the sequence shown in Table 12. For each table type, the data are printed by nuclide, by element, summarized by nuclide, and summarized by element (if requested).

The hierarchical organization is illustrated in Fig. 5.

5.3. The OROUT Output File

The organization of the OROUT file was discussed in the previous section. In the following we show the detailed format of some of the individual output-data tables.

The first page from the decay library echo is given in Fig. 6. This output table is much more readable than the original (TAPE9) data file. Similarly, Fig. 7 is the first page from the standard cross-section file (TAPE8) echo, and Fig. 8 is the first page of the odd cross-section file (TAPE3) echo. The units for all cross sections are in barns.

Figure 9 is one page from the activation-decay matrix. The entries are for the nuclide sequence number (from TAPE9), the nuclide identification (ZZAAAI), its disintegration constant (per second), its total cross section in barns, and the number of different possible daughters from the nuclide (including those from activation and decay, and including protons, deuterons, tritons, and alphas generated by activation and decay). Then, all the possible parents for the nuclide are listed along with the reaction cross section (if the nuclide is produced by activation) or the parent-decay constant multiplied by the branching ratio (if the nuclide is produced by decay).

When a decay or activation reaction produces a nuclide that is not in the decay library (TAPE9), that mass is effectively removed from the calculation since no further decay or activation will be calculated because of lack of data. Figure 10 is a page from the listing of these "limbo" reaction products. This output data should be reviewed, because it indicates potential improvement points for the decay and cross-section libraries. To aid in this review, we screen the data for two conditions. First, for a "naturally occurring" parent, the input composition should be checked, because if that parent element is input, significant quantities of the limbo daughter could be produced. Second, if the parent "is not produced," the daughter cannot be produced unless the parent is input in the initial composition.

Figure 11 is a page from the photon library (TAPE10) echo. Each nuclide entry occupies two lines to show the photon production per disintegration in the 18 photon-energy groups. This data file contains only decay-gamma information. Gamma production from (n, γ) reactions is not calculated by FORIG. However, we can obtain that information from neutron-photon transport codes such as TARTNP.⁵

Command code documentation input echo

Miscellaneous input

- Fission neutron yield per neutron-induced fission
- (alpha,n) neutron production rate
- Fission neutron yield per spontaneous fission
- Individual-element fractional reprocessing recoveries
- Element-group fractional reprocessing recoveries
- Assignment of elements to fractional recovery groups
- Elemental chemical toxicities

Listing of ORIGEN2 commands

Data libraries

Decay

- Activation product segment
- Actinide segment
- Fission product segment
- Cross section [same segments as for Decay data library]
- Photon [same segments as for Decay data library]

First Output Group (from a single OUT command)

Reactivity and burnup data

Activation product segment

Table type ^a

- Nuclide aggregation^b
- Element aggregation
- Summary isotope aggregation
- Summary element aggregation

Actinide segment [same table types and aggregations as for activation products]

Fission product segment [same table types and aggregations as for activation products]

Neutron production rates for the actinide segment

(alpha,n)

Spontaneous fission

Photon production rates

Activation product segment

- Summation tables
- Principal contributor tables

Actinide segment [same aggregations as for activation products]

Fission product segment [same aggregations as for activation products]

Final Output Group

^aTable types are defined in Table 12. The table types printed are controlled by the entries in the OPTL, OPTA, and OPTF commands.

^bThe aggregations printed are controlled by the magnitude of the entries in the OPTL, OPTA, and OPTF commands.

Figure 5. Hierarchical organization of FORIG output file, OROUT.

1

decay library: activation products updated jan 8 85 output unit = 6

nuclide	decay	con f	bela ex f	posit f	pos ex f	int tr f	alpha f	sp fis f	n+beta	q value	nat abund	air
h 1	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	9.998e+01	1.
h 2	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.500e-02	1.
h 4	6.931e+02	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	3.
he 3	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.300e-04	1.
he 4	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.000e+02	1.
he 6	8.577e-01	0.	0.	0.	0.	0.	0.	0.	0.	1.568e+00	0.	3.
li 6	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	7.500e+00	1.
li 7	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	9.250e+01	1.
li 8	8.232e-01	0.	0.	0.	0.	0.	0.	0.	0.	6.290e+00	0.	3.
be 9	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.000e+02	1.
be 10	1.373e-14	0.	0.	0.	0.	0.	0.	0.	0.	2.025e-01	0.	1.
be 11	5.097e-02	0.	0.	0.	0.	0.	0.	0.	0.	1.151e+01	0.	3.
b 10	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	2.000e+01	1.
b 11	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	8.000e+01	1.
b 12	3.415e+01	0.	0.	0.	0.	0.	0.	0.	0.	1.337e+01	0.	3.
b 13	4.000e+01	0.	0.	0.	0.	0.	0.	0.	0.	1.305e+01	0.	3.
c 12	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	9.889e+01	1.
c 13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.110e+00	1.
c 14	3.834e-12	0.	0.	0.	0.	0.	0.	0.	0.	4.947e-02	0.	1.
c 15	2.830e-01	0.	0.	0.	0.	0.	0.	0.	0.	2.871e+00	0.	3.
n 13	1.159e-03	0.	1.00e+00	0.	0.	0.	0.	0.	0.	1.511e+00	0.	3.
n 14	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	9.963e+01	1.
n 15	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	3.660e-01	1.
n 16	9.735e-02	0.	0.	0.	0.	0.	0.	0.	0.	7.311e+00	0.	3.
o 16	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	9.976e+01	1.
o 17	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	3.800e-02	1.
o 18	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	2.040e-01	1.
o 19	2.390e-02	0.	0.	0.	0.	0.	0.	0.	0.	4.819e+00	0.	3.
f 19	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.000e+02	1.
f 20	6.080e-02	0.	0.	0.	0.	0.	0.	0.	0.	7.030e+00	0.	3.
ne 20	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	9.051e+01	1.
ne 21	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	2.700e-01	1.
ne 22	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	9.220e+00	1.
ne 23	1.861e-02	0.	0.	0.	0.	0.	0.	0.	0.	2.068e+00	0.	3.
na 22	8.442e-09	0.	1.00e+00	0.	0.	0.	0.	0.	0.	2.387e+00	0.	3.
na 23	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.000e+02	1.
na 24	1.284e-05	0.	0.	0.	0.	0.	0.	0.	0.	4.675e+00	0.	5.
na 24m	3.483e+01	0.	0.	0.	1.00e+00	0.	0.	0.	0.	4.720e-01	0.	3.
na 25	1.163e-02	0.	0.	0.	0.	0.	0.	0.	0.	1.936e+00	0.	3.
mg 24	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	7.899e+01	1.
mg 25	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.000e+01	1.
mg 26	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.101e+01	1.
mg 27	1.221e-03	0.	0.	0.	0.	0.	0.	0.	0.	1.593e+00	0.	3.
mg 28	9.208e-06	0.	0.	0.	0.	0.	0.	0.	0.	1.533e+00	0.	1.
al 26	3.051e-14	0.	1.00e+00	0.	0.	0.	0.	0.	0.	3.153e+00	0.	1.
al 26m	1.090e-01	0.	1.00e+00	0.	0.	0.	0.	0.	0.	2.627e+00	0.	3.
al 27	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.000e+02	1.
al 28	5.157e-03	0.	0.	0.	0.	0.	0.	0.	0.	3.026e+00	0.	3.
al 29	1.772e-03	0.	0.	0.	0.	0.	0.	0.	0.	2.351e+00	0.	3.
al 30	1.881e-01	0.	0.	0.	0.	0.	0.	0.	0.	5.723e+00	0.	3.
si 28	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	9.223e+01	1.
si 29	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	4.670e+00	1.

Figure 6. Decay-data library (TAPE9) echo.

1

720

output unit = 6

uses ac840216, bench2 id, lcm fe at 5m, 2.4, 1.0m pbli w 90% iif 10v% fe

nuclide	(n,2n)	(n,2n)ex	(n,3n)	(n,np)	n,n-alpha	(n,p)	(n,d)	(n,t)	(n,alpha)	(n,ga)
h 2	0.	0.	0.	0.	0.	0.	0.	0.	0.	3.000e
he 3	0.	0.	0.	0.	0.	2.237e+03	1.999e-03	0.	0.	4.200e
li 6	0.	0.	0.	0.	0.	3.607e-04	0.	8.100e-01	0.	4.672e
li 7	1.278e-03	0.	0.	0.	0.	0.	0.	0.	0.	1.495e
be 9	2.070e-02	0.	0.	0.	0.	0.	0.	6.595e-04	3.688e-03	8.334e
b 10	0.	0.	0.	0.	0.	0.	0.	0.	1.679e+00	6.019e
b 11	4.591e-04	0.	0.	0.	0.	1.179e-04	0.	3.629e-04	8.905e-04	8.188e
c 12	0.	0.	0.	0.	0.	3.647e-05	0.	0.	7.195e-04	5.280e
c 13	4.188e-04	0.	0.	0.	1.794e-03	4.519e-06	0.	6.472e-06	2.201e-03	1.731e
c 14	3.022e-04	1.937e-07	0.	2.291e-04	0.	0.	0.	0.	9.206e-04	0.
n 14	1.449e-04	0.	5.997e-03	0.	1.552e-02	1.544e-03	8.228e-04	9.162e-03	4.793e	
o 16	0.	0.	7.447e-06	3.810e-03	1.016e-03	7.150e-04	0.	4.031e-03	0.	0.
o 17	0.	0.	0.	0.	0.	0.	0.	4.367e-02	0.	0.
f 19	5.570e-04	0.	2.777e-04	1.327e-03	5.728e-04	3.169e-04	1.965e-04	1.594e-03	1.669e	
na 23	3.752e-04	0.	1.752e-03	2.456e-04	1.387e-03	0.	0.	4.393e-03	1.383e	
mg 24	0.	0.	1.094e-04	0.	4.866e-03	0.	0.	2.443e-03	6.886e	
mg 25	1.571e-03	0.	0.	0.	1.443e-03	0.	0.	2.311e-03	2.285e	
mg 26	5.906e-03	0.	0.	0.	1.391e-03	0.	0.	2.342e-03	5.711e	
mg 27	1.319e-02	0.	0.	0.	0.	0.	0.	4.039e-03	3.808e	
al 26	4.374e-03	0.	1.019e-03	1.361e-05	4.299e-02	8.133e-04	4.484e-05	5.737e-03	3.041e	
al 27	5.627e-04	2.583e-06	0.	0.	2.389e-03	6.459e-04	6.702e-07	3.314e-03	2.374e	
si 28	0.	0.	5.845e-06	0.	8.366e-03	0.	0.	2.895e-03	2.295e	
si 29	3.292e-03	0.	0.	0.	7.876e-03	0.	0.	2.851e-03	1.724e	
si 30	3.043e-03	0.	0.	0.	6.057e-03	0.	0.	2.278e-03	1.734e	
si 31	4.766e-03	0.	0.	0.	0.	0.	0.	2.132e-03	1.173e	
p 31	2.181e-04	0.	2.507e-03	0.	4.448e-03	0.	0.	3.321e-03	3.713e	
s 32	0.	0.	1.749e-03	0.	1.111e-02	0.	1.481e-07	1.016e-02	1.646e	
s 33	5.085e-04	0.	1.974e-03	3.705e-04	1.190e-02	3.769e-04	0.	1.396e-02	9.915e	
s 34	3.741e-04	0.	1.439e-04	1.444e-04	2.542e-03	3.451e-04	0.	2.276e-03	8.315e	
s 35	1.057e-03	0.	0.	5.807e-05	0.	0.	0.	3.137e-03	6.709e	
s 36	1.444e-03	0.	0.	0.	0.	0.	0.	0.	5.089e	
s 37	2.289e-03	0.	0.	0.	0.	0.	0.	0.	4.254e	
cl 35	6.921e-05	6.593e-05	0.	1.741e-03	5.249e-04	1.864e-01	0.	4.727e-03	3.457e	
cl 36	1.011e-03	0.	2.209e-03	1.142e-04	4.453e-02	0.	0.	2.973e-03	7.426e	
cl 37	1.098e-03	0.	2.400e-04	3.482e-05	4.331e-04	0.	5.564e-07	1.297e-03	4.400e	
cl 38	2.146e-03	0.	1.294e-04	0.	1.003e-03	0.	0.	0.	2.561e	
or 36	0.	0.	0.	0.	8.596e-03	0.	0.	5.946e-03	8.093e	
or 37	4.056e-03	0.	0.	0.	1.502e-02	0.	0.	7.508e-03	5.826e	
or 38	2.630e-03	0.	0.	0.	6.733e-03	0.	0.	5.472e-03	4.462e	
or 39	1.035e-02	0.	0.	0.	7.042e-03	0.	0.	1.321e-02	3.642e	
or 40	1.306e-02	0.	0.	0.	6.353e-03	0.	0.	5.190e-03	2.028e	
or 41	1.796e-02	0.	0.	0.	0.	0.	0.	5.436e-03	1.755e	
or 42	1.898e-02	0.	0.	0.	0.	0.	0.	0.	1.810e	
k 39	3.172e-05	7.168e-06	4.713e-03	0.	1.284e-02	0.	2.074e-07	4.804e-03	4.007e	
k 40	7.703e-04	0.	0.	0.	3.480e-02	3.121e-04	0.	2.489e-03	6.135e	
k 41	8.063e-03	0.	0.	0.	2.322e-03	0.	3.345e-07	9.566e-04	4.502e	
k 42	1.356e-02	0.	0.	0.	1.344e-03	0.	0.	5.524e-04	4.636e	
k 43	1.519e-02	0.	0.	0.	0.	0.	0.	0.	1.946e	
k 44	1.967e-02	0.	0.	0.	0.	0.	0.	0.	3.681e	
ca 40	0.	0.	3.817e-03	5.210e-04	2.152e-02	0.	4.022e-06	9.455e-03	3.034e	
ca 41	6.049e-03	0.	1.824e-03	1.241e-03	7.652e-03	0.	2.568e-05	3.973e-03	5.107e	
ca 42	4.093e-03	0.	2.406e-04	3.071e-04	4.912e-03	0.	0.	3.211e-03	1.145e	

Figure 7. Standard cross-section library (TAPE8) echo.

odd reaction cross sections

reaction types (lrrtyp): (a=alpha, h=3He, (a) = residual is an alpha)

12 n,2n	13 n,3n	14 n,4n	15 n,1is	20 n,np
21 n,pn	22 n,nd	23 n,nda(a)	24 n,nt	25 n,nh
26 n,no	27 n,n2a(a)	28 n,nla(a)	29 n,2np(a)	30 n,gna
31 n,2np(a)	37 n,2a(a)	38 n,ha(a)	39 n,tp(a)	40 n,p
41 n,d	42 n,t	43 n,la(a)	44 n,h	45 n,a
46 n,g				

odd rxn; parent = 30060	daughter = 20040	cross section= 1.9390e-02 barns	type = 22
odd rxn; parent = 30060	daughter = 20040	cross section= 1.8800e-03 barns	type = 29
odd rxn; parent = 30070	daughter = 20040	cross section= 9.8060e-03 barns	type = 24
odd rxn; parent = 50100	daughter = 20040	cross section= 2.8620e-03 barns	type = 23
odd rxn; parent = 50100	daughter = 20040	cross section= 5.2310e-04 barns	type = 29
odd rxn; parent = 50100	daughter = 20040	cross section= 4.6070e-03 barns	type = 43
odd rxn; parent = 60120	daughter = 20040	cross section= 9.9260e-03 barns	type = 27
odd rxn; parent = 230490	daughter = 210461	cross section= 1.2710e-03 barns	type = 45
odd rxn; parent = 260530	daughter = 250521	cross section= 1.0550e-02 barns	type = 20
odd rxn; parent = 260540	daughter = 250521	cross section= 2.5340e-07 barns	type = 42
odd rxn; parent = 280580	daughter = 270581	cross section= 8.4030e-03 barns	type = 40
odd rxn; parent = 280590	daughter = 270581	cross section= 2.9390e-03 barns	type = 20
odd rxn; parent = 280600	daughter = 270601	cross section= 1.8950e-03 barns	type = 40
odd rxn; parent = 280600	daughter = 270581	cross section= 6.1610e-07 barns	type = 42
odd rxn; parent = 280610	daughter = 270601	cross section= 1.7000e-04 barns	type = 20
odd rxn; parent = 280610	daughter = 270601	cross section= 7.1080e-05 barns	type = 41
odd rxn; parent = 280620	daughter = 270621	cross section= 5.0410e-04 barns	type = 40
odd rxn; parent = 280620	daughter = 270601	cross section= 1.7880e-08 barns	type = 42
odd rxn; parent = 290620	daughter = 270581	cross section= 4.8350e-04 barns	type = 26
odd rxn; parent = 290630	daughter = 270601	cross section= 6.3540e-04 barns	type = 45
odd rxn; parent = 290640	daughter = 270601	cross section= 3.8390e-05 barns	type = 26
odd rxn; parent = 290650	daughter = 270621	cross section= 1.7900e-04 barns	type = 45
odd rxn; parent = 300680	daughter = 290681	cross section= 2.0370e-04 barns	type = 40
odd rxn; parent = 300700	daughter = 290701	cross section= 1.2290e-04 barns	type = 40
odd rxn; parent = 320720	daughter = 300691	cross section= 3.8100e-04 barns	type = 45
odd rxn; parent = 380860	daughter = 360831	cross section= 2.3650e-05 barns	type = 45
odd rxn; parent = 390890	daughter = 370861	cross section= 8.5530e-05 barns	type = 45
odd rxn; parent = 410930	daughter = 390891	cross section= 6.4720e-05 barns	type = 26
odd rxn; parent = 420920	daughter = 410911	cross section= 8.8310e-05 barns	type = 20
odd rxn; parent = 420920	daughter = 410921	cross section= 2.1170e-03 barns	type = 40
odd rxn; parent = 420920	daughter = 410911	cross section= 5.1100e-04 barns	type = 41
odd rxn; parent = 420930	daughter = 410921	cross section= 1.7770e-03 barns	type = 41
odd rxn; parent = 420931	daughter = 410921	cross section= 2.7770e-03 barns	type = 41
odd rxn; parent = 420940	daughter = 410921	cross section= 1.4850e-08 barns	type = 42
odd rxn; parent = 471070	daughter = 451041	cross section= 2.7220e-04 barns	type = 45
odd rxn; parent = 491130	daughter = 481131	cross section= 2.2520e-04 barns	type = 40
odd rxn; parent = 491130	daughter = 471101	cross section= 1.8780e-05 barns	type = 45
odd rxn; parent = 491150	daughter = 481151	cross section= 2.0680e-04 barns	type = 40
odd rxn; parent = 501120	daughter = 491121	cross section= 1.2890e-03 barns	type = 40

Figure 8. Odd-reaction cross-section library (TAPE3) echo.

non-zero matrix elements; i=# of daughter:										output unit = 6			
i	nuc1(i)	dis(i)	locap(i)	mmax	rate	parent	rate	parent	rate	parent	rate	parent	rat
95	210441	3.288e-06	1.179e-01	8	2.423e-03	210450							
96	210450	0.	3.639e-02	8	4.923e-08	200450	2.814e-02	210440	2.814e-02	210441	1.259e		
					1.263e-02	210461	1.011e-04	220460	6.871e-03	230480			
97	210460	9.574e-08	2.371e-02	7	3.713e-02	210461	1.658e-02	210450	1.391e-02	210470	7.671e		
					5.945e-05	220470	1.939e-03	230490					
98	210461	3.713e-02	2.408e-02	7	9.421e-03	210450	1.271e-03	230490					
99	210470	2.394e-06	3.661e-02	7	1.769e-06	200470	7.066e-03	210460	7.066e-03	210461	1.896e		
					6.195e-03	220470	1.188e-05	220480	1.501e-03	230500	9.276e		
100	210480	4.395e-06	2.833e-02	7	2.000e-02	210470	1.946e-03	220480	1.319e-05	220490	8.553e		
101	210490	2.009e-04	0.	1	1.313e-03	200490	6.751e-03	210480	9.522e-04	220490	1.953e		
102	210500	6.762e-03	0.	1	4.444e-04	220500							
103	220460	0.	1.452e-02	9	9.574e-08	210460	7.900e-03	220470	3.786e-03	240490			
104	220470	0.	2.195e-02	8	2.394e-06	210470	3.544e-04	230470	4.964e-03	220460	5.978e		
					4.063e-03	230480	1.156e-03	240500					
105	220480	0.	2.280e-02	8	4.395e-06	210480	5.020e-07	230480	6.427e-03	220470	1.565e		
					1.822e-01	230480	2.191e-03	230490	1.261e-03	240510			
106	220490	0.	2.286e-02	8	2.009e-04	210490	2.431e-08	230490	1.382e-02	220480	1.280e		
					4.960e-03	230490	8.776e-04	230500	3.641e-04	240520			
107	220500	0.	1.453e-02	6	6.762e-03	210500	3.844e-25	230500	5.460e-03	220490	2.181e		
					8.520e-03	230500	4.657e-04	230510	7.226e-04	240530			
108	220510	2.006e-03	2.538e-02	7	1.105e-03	220500	1.213e-03	230510	1.501e-04	230520	3.028e		
109	230470	3.544e-04	3.533e-02	2	4.219e-03	230480							
110	230480	5.020e-07	3.738e-01	8	3.533e-02	230470	6.821e-03	230490	7.737e-03	240490	6.789e		
111	230490	2.431e-08	3.484e-02	9	2.755e-04	240490	1.764e-01	230480	1.719e-02	230500	7.200e		
					3.669e-03	240500	4.916e-03	250520					
112	230500	5.491e-25	2.627e-01	5	1.766e-02	230490	1.483e-02	230510	8.170e-03	240500	1.500e		
					3.103e-08	240520	2.296e-03	250530					
113	230510	0.	2.667e-02	8	2.006e-03	220510	2.895e-07	240510	2.346e-01	230500	2.485e		
					8.893e-03	240510	1.722e-04	240520	2.593e-06	240530	1.209e		
114	230520	3.081e-03	7.164e-02	8	9.212e-03	230510	2.417e-03	240520	7.263e-05	240530	1.029e		
					8.678e-04	250550							
115	230530	7.175e-03	0.	1	4.387e-02	230520	1.215e-03	240530	2.602e-06	240540	8.016e		
116	230540	1.260e-02	0.	1	3.404e-04	240540	3.558e-04	250570					
117	240490	2.755e-04	1.397e-01	9	2.019e-04	240500							
118	240500	0.	3.136e-02	10	1.647e-25	230500	5.569e-02	240490	6.413e-03	240510	1.002e		
119	240510	2.895e-07	5.510e-02	8	2.501e-04	250510	1.816e-02	240500	5.936e-03	240520	3.060e		
					3.112e-03	260540							
120	240520	0.	1.103e-02	10	3.081e-03	230520	1.435e-06	250520	5.379e-04	250521	3.703e		
					1.525e-02	240530	1.889e-01	250520	1.700e-03	250530	3.470e		
121	240530	0.	4.194e-02	10	7.175e-03	230530	5.780e-15	250530	2.141e-03	240520	1.743e		
					3.626e-03	250530	7.499e-04	250540	1.226e-05	250550	1.870e		
122	240540	0.	1.949e-02	10	1.260e-02	230540	2.467e-08	250540	2.468e-02	240530	2.172e		
					2.878e-03	250540	5.994e-04	250550	1.518e-03	260570			
123	240550	3.254e-03	3.845e-02	7	1.410e-03	240540	2.297e-02	240560	1.084e-03	250550	9.127e		
					4.909e-04	260580							
124	240560	1.958e-03	2.365e-02	3	1.645e-02	240550	7.815e-04	250560	2.365e-05	250570	2.781e		
125	250510	2.501e-04	5.315e-02	3	2.540e-03	250520							
126	250520	1.435e-06	2.247e-01	8	9.581e-06	250521	1.755e-02	250510	1.728e-03	250530	1.055e		
					1.048e-06	260540							
127	250521	5.475e-04	0.	2	3.560e-02	250510	1.358e-03	250530	1.055e-02	260530	2.534e		
128	250530	5.780e-15	6.931e-02	9	1.358e-03	260530	4.566e-03	260531	2.529e-02	250520	1.825e		
					5.542e-01	260530	5.077e-03	260540					
129	250540	2.567e-08	3.995e-02	8	5.860e-02	250530	1.809e-02	250550	1.558e-02	260540	1.889e		
					5.104e-07	260560	1.985e-05	270580	1.896e-05	270581			
130	250550	0.	4.203e-02	10	3.254e-03	240550	8.448e-09	260550	1.686e-02	250540	2.720e		
					7.187e-03	260550	2.060e-03	260560	3.027e-07	260570	1.511e		

Figure 9. Page from the reaction-matrix table.

```

daughters produced that are not in the decay library
daughter = 481050; parent = 481060
481060 is not produced.
481060 is a naturally occurring nuclide.
daughter = 491120; parent = 491130
491130 is a naturally occurring nuclide.
daughter = 481131; parent = 491130
491130 is a naturally occurring nuclide.
daughter = 501110; parent = 501120
501120 is not produced.
501120 is a naturally occurring nuclide.
daughter = 491120; parent = 501120
501120 is not produced.
501120 is a naturally occurring nuclide.
daughter = 491110; parent = 501120
501120 is not produced.
501120 is a naturally occurring nuclide.
daughter = 491121; parent = 501120
501120 is not produced.
501120 is a naturally occurring nuclide.
daughter = 491220; parent = 501220
501220 is a naturally occurring nuclide.
daughter = 491240; parent = 501240
501240 is not produced.
501240 is a naturally occurring nuclide.
daughter = 511200; parent = 511210
511210 is a naturally occurring nuclide.
daughter = 591400; parent = 591410
591410 is a naturally occurring nuclide.
daughter = 601410; parent = 601420
601420 is a naturally occurring nuclide.
daughter = 621430; parent = 621440
621440 is not produced.
621440 is a naturally occurring nuclide.
daughter = 631501; parent = 631510
631510 is a naturally occurring nuclide.
daughter = 631500; parent = 631510
631510 is a naturally occurring nuclide.
daughter = 10030; parent = 631510
631510 is a naturally occurring nuclide.
daughter = 10030; parent = 631530
631530 is a naturally occurring nuclide.
daughter = 671621; parent = 671630
daughter = 671640; parent = 671630
daughter = 671641; parent = 671630
daughter = 671641; parent = 671650
671650 is a naturally occurring nuclide.
daughter = 671640; parent = 671650
671650 is a naturally occurring nuclide.
daughter = 671670; parent = 671660
daughter = 671670; parent = 671661
daughter = 691680; parent = 691690
691690 is a naturally occurring nuclide.
daughter = 711741; parent = 711750
711750 is a naturally occurring nuclide.

```

Figure 10. Page from the listing of "Daughters not in the decay library."

		output unit = 6									
		photon lib: activation products, no Brem. (nov 4, 83)									
nuclide		mean group energy, mev									
		0.0100	0.0250	0.0375	0.0575	0.0850	0.1250	0.2250	0.3750	0.5750	0.8500
		2.2500	2.7500	3.5000	5.0000	7.0000	11.0000				
be 8	0.	0.	0.	0.	0.	0.	0.	8.38e-09	0.	0.	3.34e
	4.77e-10	2.39e-10	1.76e-10	5.24e-11	3.40e-12	2.15e-13					
b 13	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
	0.	0.	9.40e-02	0.	0.	0.	0.				
n 13	9.42e-08	0.	0.	0.	0.	0.	0.	0.	0.	1.77e+00	0.
	0.	0.	0.	0.	0.	0.	0.				
n 16	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	4.66e
	0.	7.59e-03	0.	0.	6.55e-01	6.54e-04					
ne 23	0.	0.	0.	0.	0.	0.	0.	0.	3.87e-01	0.	0.
	9.32e-04	4.12e-04	0.	0.	0.	0.					
na 22	1.06e-04	0.	0.	0.	0.	0.	0.	0.	0.	1.60e+00	0.
	0.	0.	0.	0.	0.	0.					
na 24	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.17e
	0.	1.00e+00	6.85e-04	7.12e-06	0.	0.					
na 25	0.	0.	0.	0.	0.	0.	0.	0.	1.34e-01	1.28e-01	1.69e
	1.00e-03	5.81e-04	0.	0.	0.	0.					
mg 27	0.	0.	0.	0.	0.	0.	0.	6.07e-03	0.	0.	7.13e
	0.	0.	0.	0.	0.	0.					
mg 28	1.44e-03	0.	5.39e-01	0.	0.	0.	0.	0.	3.91e-01	1.12e-03	4.24e
	0.	0.	0.	0.	0.	0.					
al 26	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.64e+00	0.
	0.	2.75e-03	0.	0.	0.	0.					
al 28	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
	0.	0.	0.	0.	0.	0.					
al 29	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	1.78e
	8.99e-02	0.	0.	0.	0.	0.					
al 30	0.	0.	0.	0.	0.	0.	0.	0.	1.12e-03	0.	0.
	6.37e-01	6.83e-02	3.18e-01	2.01e-02	0.	0.					
si 31	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
	0.	0.	0.	0.	0.	0.					
cl 36	3.00e-04	0.	0.	0.	0.	0.	0.	0.	0.	2.65e-04	0.
	0.	0.	0.	0.	0.	0.					
cl 38	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
	4.24e-01	0.	2.72e-04	0.	0.	0.					
cl 38m	1.60e-05	0.	0.	0.	0.	0.	0.	0.	0.	1.17e+00	0.
	0.	0.	0.	0.	0.	0.					
ar 37	2.22e-02	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
	0.	0.	0.	0.	0.	0.					
ar 41	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
	0.	0.	0.	0.	0.	0.					
k 40	2.78e-03	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
	0.	0.	0.	0.	0.	0.					
k 42	0.	0.	0.	0.	0.	0.	0.	0.	2.66e-03	3.88e-05	5.45e
	2.28e-04	0.	0.	0.	0.	0.					
k 43	0.	0.	0.	0.	0.	0.	0.	4.24e-02	9.90e-01	9.78e-01	5.21e
	0.	0.	0.	0.	0.	0.					
ca 45	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
	0.	0.	0.	0.	0.	0.					
ca 47	0.	0.	6.97e-05	0.	0.	0.	0.	0.	0.	5.83e-02	6.72e
	0.	0.	0.	0.	0.	0.					
sc 44	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
	3.26e-03	0.	6.50e-06	0.	0.	0.					
sc 44m	0.	0.	0.	0.	0.	0.	0.	8.66e-01	0.	0.	0.
	0.	0.	0.	0.	0.	0.					

Figure 11. Photon-decay library (TAPE10) echo.

Figure 12 is the first reactivity and burnup-data table from the example (i.e., the data taken during the neutron flux). The table begins with the title (from the `TIT` command). Then, the power, burnup, and flux are listed. For fusion problems, these values are not computed, but instead a value of 1.0 is always printed. For fission problems, these values are computed by the `BUP` commands. Next, the `BAS` comments are printed. The first two rows of the time-dependent output are for the time in seconds (useful for plotting) and for the neutron flux ($n/cm^2 \cdot s$). The entries for "size of `MMAX(i)`" indicate the number of nuclides with "`i`" daughter nuclides (including decay and activation reactions as well as the `p`, `d`, `t`, and `a` particles produced by those reactions). In the example, one nuclide had 14 possible daughters. In `ORIGEN2`, a maximum of only seven daughters was permitted. (The higher energy spectrum in fusion reactors can drive more reaction types than the spectrum in fission reactors.) The final segment of the reactivity and burnup data indicates the size of the arrays. These sizes should not exceed the dimensions of the code (see Fig. 1). If the dimensions are exceeded (say by an overly large decay library), the arrays will overflow and overwrite other arrays. A comparison is shown in Table 13 between the numbers printed in the output and the dimensions given in Fig. 1.

Figure 13 illustrates a typical output table, that is, a summary table by element for radioactivity in curies. The "sumtot" line is the total of the table entries, and the "total" line is the overall total. The difference indicates activity "lost" by using the cutoff values. [Similarly, inspection of the "totals" line of the "grams" table as a function of time indicates the mass lost to the production of limbo daughters (i.e., daughters not in the decay library).] The "cumulative table totals" are a running total by segment. For the activation-product output, the other two segments are not yet accumulated, and the totals "ap & fp," "act & fp," and "ap & act & fp" should be ignored.

Figure 14 shows the (decay) photon-emission rates as functions of both time and photon energy. The upper table is in terms of photons per second for each photon-energy group. Its line of totals is immediately followed by the total photon-energy release rate (MeV/s). The lower table is the photon-energy release rate (MeV/s) divided by the reactor power (MW). For fusion problems, the reactor power is not computed by `FORIG`, and 1 MW is assumed in the table. Hence, in the lower table, the total line is the gamma-energy release rate (MeV/MW·s). The final line ("gam pow") is more easily understood; it is the actual decay gamma power in watts. In the example, the 30 year-shutdown decay-gamma power is 2.4 MW, or less than 0.1% of the reactor fusion power.

Figure 15 shows one page of the principal contributors to the various photon-energy groups. The units are in photons per second, and the sums for each group (not printed) are the entries in the top half of Fig. 14.

The approximate gamma dose rate (mR/hr) is printed in a similar format. First, the dose rates for all 18 photon-energy groups are printed in one table. Then, each energy group is reprinted to show the individual contributions from the most significant nuclides. Finally, the total dose rate is broken down to show the contributions of the most significant contributors. Also, in this final table, the "sum" of the significant contributors is shown as well as the "total" from all nuclides. The difference between these two totals can be reduced by changing the `CUT` command.

5.4. TAPE13: Table of Contents and TAPE11: Abbreviated Output

The organization of these files is identical to that of TAPE12 and of `OROUT`, except that output begins with the first requested "Reactivity and Burnup" table.

5.5. TAPE15: Debug Information

This file serves three principal functions. For the first function, which is useful in some debugging situations, a single line of information is printed just before each command is executed. The number of each command type is also printed. However, for counting the number of commands of a particular type, all `IRP`, `IRF`, and `DEC` commands are counted as `IRF` commands. There will always be a total of zero `IRP` and `DEC` instructions.

The second function of the output on TAPE15 is to provide selected internal information calculated by `FORIG`. This type of information is printed for the `IRP`, `IRF`, `KEQ`, and `FAC` commands. The significance of the printed information is discussed below.

benchmark 1d problem for po build up during opn
power= 1.00000e+00mw, burnup= 1.00000e+00mwd, flux= 1.00e+00n/cm**2-sec

reactivity and burnup data									
basis=all material in zone 1, 1982 wppm bi in pb 2orott									
	fresh	1.0hr	1.0d	10.0d	30.0d	90.0d	180.0d	1.0yr	3.0yr
time, sec	0.	3.60e+03	8.64e+04	8.64e+05	2.59e+06	7.78e+06	1.56e+07	3.16e+07	9.47e+07
neut. flux	0.	4.26e+14	4.26e+14	4.26e+14	4.26e+14	4.26e+14	4.26e+14	4.26e+14	4.26e+14
sp pow,mw	0.	0.	0.	0.	0.	0.	0.	0.	0.
burnup,mwd	0.	0.	0.	0.	0.	0.	0.	0.	0.
k infinity	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
neut prodn	0.	0.	0.	0.	0.	0.	0.	0.	0.
neut destn	0.	3.65e+06	3.65e+06	3.65e+06	3.65e+06	3.64e+06	3.64e+06	3.62e+06	3.57e+06
tot burnup	0.	0.	0.	0.	0.	0.	0.	0.	0.
avg n flux	0.	0.	0.	0.	0.	0.	0.	0.	0.
avg sp pow	0.	0.	0.	0.	0.	0.	0.	0.	0.

size of mmax(i): mmax= 1 # = 277 mmax= 2 # = 79 mmax= 3 # = 21 mmax= 4 # = 26 mmax= 5 #
mmax= 7 # = 32 mmax= 8 # = 54 mmax= 9 # = 23 mmax=10 # = 16 mmax=11 # = 9 mmax=12
mmax=13 # = 4 mmax=14 # = 1 mmax=15 # = 0

the number of non-zero terms in a=2036
the number of non-zero fission product yields= 2
ilite= 722 iact= 0 ifp= 0 itot= 722
the number of non-zero natural abundances= 284
the number of non-zero photon yields= 1633
the maximum number of terms in ap= 2002

Figure 12. Reactivity and burnup data.

Table 13. Comparison of code sizes with dimensions selected by the user.

Code size (from the table of reactivity and burnup data)	Dimension (from Fig. 1)
The number of non-zero terms in "a" (2036)	Total number of non-zero matrix elements: DDDD (2500)
The number of nonzero fission product yields (2)	Number of nonzero fission product yields = EEEE (4)
ilite (722)	Maximum number of light nuclides = MMMM (900)
iact (0)	Maximum number of actinides = GGGG - 1 (4 - 1 = 3)
ifp (0)	Maximum number of fission products = FFFF (4)
itot (722)	Maximum number of non-nuclides = BBBB (900)
The number of nonzero natural abundances (284)	Number of nonzero natural abundances = KKKK (300)
The number of nonzero photon yields (1633)	Number of nonzero photon yields = LLLL (3300)
The maximum number of terms in "ap" (2002)	Maximum number of nonzero matrix elements for long-lived nuclides = IIII (2500)

```

1
benchmark 1d problem for pa build up during opn
power= 1.00000e+00mw, burnup= 1.00000e+00mwd, flux= 1.00e+00n/cm**2-sec
output unit = 6
activ

summary table: radioactivity, curies
all material in zone 1, 1982 wppm bi in pb 2orott
fresh      1.0hr      1.0d      10.0d      30.0d      90.0d      180.0d      1.0yr      3.0yr

he      0.      1.281e+07 1.281e+07 1.281e+07 1.280e+07 1.277e+07 1.274e+07 1.267e+07 1.239e+07 1.
cr      0.      1.073e+05 3.922e+05 2.759e+06 6.450e+06 1.087e+07 1.200e+07 1.213e+07 1.214e+07 1.
mn      0.      4.621e+07 1.883e+08 1.897e+08 1.923e+08 1.993e+08 2.083e+08 2.219e+08 2.437e+08 2.
fe      0.      3.571e+05 8.171e+05 5.093e+06 1.446e+07 4.162e+07 7.998e+07 1.513e+08 3.544e+08 4.
tl      0.      3.735e+06 3.736e+06 3.738e+06 3.744e+06 3.767e+06 3.802e+06 3.879e+06 4.172e+06 4.
pb      9.892e-07 1.068e+08 5.995e+08 6.078e+08 6.078e+08 6.077e+08 6.076e+08 6.074e+08 6.065e+08 6.
sumtot  9.892e-07 1.700e+08 8.055e+08 8.220e+08 8.376e+08 8.761e+08 9.244e+08 1.009e+09 1.233e+09 1.

total   9.892e-07 1.701e+08 8.056e+08 8.226e+08 8.385e+08 8.774e+08 9.260e+08 1.011e+09 1.235e+09 1.

cumulative table totals

ap+fp   9.892e-07 1.701e+08 8.056e+08 8.226e+08 8.385e+08 8.774e+08 9.260e+08 1.011e+09 1.235e+09 1.
act+fp  0.      0.      0.      0.      0.      0.      0.      0.      0.      0.
ap+act+fp 9.892e-07 1.701e+08 8.056e+08 8.226e+08 8.385e+08 8.774e+08 9.260e+08 1.011e+09 1.235e+09 1.

```

Figure 13. Typical FORIG output page.

Basically the same information is printed for both the IRP and IRF commands. Most of the parameters printed are intermediate values used in SUBROUTINE FLUXO to calculate the flux when the power is given, or vice versa. (These values will not be described in detail, but the nomenclature in the TAPE15 output is the same as that in FLUXO. Thus the interested user can readily perform the flux/power calculation with a hand calculator if required.)

The following parameters printed in TAPE15 are of more general interest:

TSEC	Absolute time at the end of the current irradiation step (s).
DELT	Duration of the current irradiation step (s).
EPF1, EPF2, EPF3	Recoverable energy per fission associated with the zero-, first-, and second-time derivatives used in the flux/power calculation (MeV/fission).
EPFAVG	Average recoverable energy per fission for this time step (MeV/fission).
FLUX	Calculated or specified flux for the irradiation step (neutrons/s/cm ²).
POWER	Calculated or specified power for the irradiation step (MW per basis unit).

In some situations, this type of information can be useful either as input to auxiliary hand calculations or in finding errors.

The internal information printed for the KEQ command is related to the calculated neutron production and destruction rates, to the infinite multiplication factors, and to the fraction of the allocated material that is included in the final mixture. The parameters are defined as follows:

NPROA, NPROB, NPROC	Relative neutron production rates of vectors NKEQ(1), NKEQ(2), and NKEQ(3), respectively.
NDESA, NDESB, NDESC	Relative neutron destruction rates of vectors NKEQ(1), NKEQ(2), and NKEQ(3), respectively.

output unit = 6

photon spectrum for activation products

benchmark 1d problem for po build up during opn

power= 1.00 mw, burnup= 1. mwd, flux= 1.00e+00 n/cm²-sec

18 group photon release rates, photons/second

basis=all material in zone 1, 1982 wppm bi in pb 20r011

emear	fresh	1.0hr	1.0d	10.0d	30.0d	90.0d	180.0d	1.0yr	3.0yr
1.500e-02	0.	1.284e+16	1.469e+16	3.070e+16	5.931e+16	1.139e+17	1.645e+17	2.347e+17	3.452e+17
2.500e-02	0.	7.245e+09	3.086e+10	3.260e+10	3.567e+10	4.088e+10	4.364e+10	4.498e+10	4.727e+10
3.750e-02	0.	9.594e-08	1.280e-03	9.502e-01	1.570e+01	2.187e+02	8.845e+02	3.673e+03	3.333e+04
5.750e-02	0.	6.717e+03	2.289e+06	1.306e+08	1.000e+09	6.773e+09	1.948e+10	4.934e+10	1.732e+11
8.500e-02	0.	1.309e+12	3.106e+13	2.905e+14	7.566e+14	1.551e+15	1.958e+15	2.096e+15	2.116e+15
1.250e-01	0.	9.690e+15	9.690e+15	9.691e+15	9.693e+15	9.700e+15	9.710e+15	9.730e+15	9.810e+15
2.250e-01	3.066e-04	4.828e+17	3.716e+18	3.877e+18	3.881e+18	3.887e+18	3.890e+18	3.890e+18	3.884e+18
3.750e-01	0.	2.953e+16	1.825e+17	1.973e+17	2.089e+17	2.230e+17	2.273e+17	2.295e+17	2.357e+17
5.750e-01	0.	9.508e+15	3.764e+16	3.903e+16	3.903e+16	3.903e+16	3.904e+16	3.906e+16	3.911e+16
8.500e-01	1.222e-04	1.603e+18	6.778e+18	6.832e+18	6.925e+18	7.181e+18	7.505e+18	8.001e+18	8.791e+18
1.250e+00	6.952e-05	5.414e+15	1.232e+16	1.243e+16	1.249e+16	1.253e+16	1.259e+16	1.274e+16	1.349e+16
1.750e+00	3.483e-05	4.572e+17	1.932e+18	1.935e+18	1.935e+18	1.935e+18	1.935e+18	1.935e+18	1.937e+18
2.250e+00	1.745e-05	2.192e+17	9.279e+17	9.293e+17	9.293e+17	9.293e+17	9.293e+17	9.293e+17	9.299e+17
2.750e+00	8.745e-06	3.083e+16	1.300e+17	1.302e+17	1.302e+17	1.302e+17	1.302e+17	1.303e+17	1.303e+17
3.500e+00	6.440e-06	2.650e+15	1.115e+16	1.117e+16	1.117e+16	1.117e+16	1.117e+16	1.117e+16	1.118e+16
5.000e+00	1.917e-06	1.306e+12	1.306e+12	1.306e+12	1.306e+12	1.306e+12	1.307e+12	1.308e+12	1.313e+12
7.000e+00	1.244e-07	5.019e-02	2.614e+01	1.780e+03	8.512e+03	2.653e+04	4.530e+04	6.554e+04	8.101e+04
1.100e+01	7.867e-09	3.173e-03	1.653e+00	1.125e+02	5.382e+02	1.678e+03	2.865e+03	4.144e+03	5.123e+03
total	5.679e-04	2.862e+18	1.375e+19	1.400e+19	1.414e+19	1.447e+19	1.486e+19	1.543e+19	1.633e+19
mev/sec	4.171e-04	2.883e+18	1.257e+19	1.267e+19	1.275e+19	1.298e+19	1.326e+19	1.368e+19	1.436e+19

18 group specific energy release rates, mev/watt-sec

basis=all material in zone 1, 1982 wppm bi in pb 20r011

emear	fresh	1.0hr	1.0d	10.0d	30.0d	90.0d	180.0d	1.0yr	3.0yr
1.500e-02	0.	1.925e+08	2.203e+08	4.606e+08	8.897e+08	1.709e+09	2.467e+09	3.521e+09	5.178e+09
2.500e-02	0.	1.811e+02	7.715e+02	8.149e+02	8.917e+02	1.022e+03	1.091e+03	1.124e+03	1.182e+03
3.750e-02	0.	3.598e-15	4.800e-11	3.563e-08	5.888e-07	8.202e-06	3.317e-05	1.377e-04	1.250e-03
5.750e-02	0.	3.862e-04	1.316e-01	7.509e+00	5.752e+01	3.895e+02	1.120e+03	2.837e+03	9.960e+03
8.500e-02	0.	1.113e+05	2.640e+06	2.469e+07	6.431e+07	1.318e+08	1.665e+08	1.781e+08	1.799e+08
1.250e-01	0.	1.211e+09	1.211e+09	1.211e+09	1.212e+09	1.212e+09	1.214e+09	1.216e+09	1.226e+09
2.250e-01	6.899e-11	1.086e+11	8.360e+11	8.723e+11	8.732e+11	8.747e+11	8.753e+11	8.752e+11	8.738e+11
3.750e-01	0.	1.107e+10	6.844e+10	7.400e+10	7.833e+10	8.363e+10	8.524e+10	8.605e+10	8.840e+10
5.750e-01	0.	5.467e+09	2.105e+10	2.244e+10	2.244e+10	2.244e+10	2.245e+10	2.246e+10	2.249e+10
8.500e-01	1.039e-10	1.362e+12	5.762e+12	5.807e+12	5.887e+12	6.104e+12	6.380e+12	6.801e+12	7.472e+12
1.250e+00	8.690e-11	6.768e+09	1.539e+10	1.554e+10	1.561e+10	1.566e+10	1.574e+10	1.593e+10	1.686e+10
1.750e+00	6.096e-11	8.001e+11	3.381e+12	3.387e+12	3.387e+12	3.387e+12	3.387e+12	3.387e+12	3.389e+12
2.250e+00	3.927e-11	4.931e+11	2.088e+12	2.091e+12	2.091e+12	2.091e+12	2.091e+12	2.091e+12	2.092e+12
2.750e+00	2.405e-11	8.477e+10	3.576e+11	3.582e+11	3.582e+11	3.582e+11	3.582e+11	3.582e+11	3.584e+11
3.500e+00	2.254e-11	9.274e+09	3.904e+10	3.910e+10	3.910e+10	3.910e+10	3.910e+10	3.910e+10	3.913e+10
4.000e+00	9.586e-12	6.528e+06	6.529e+06	6.529e+06	6.530e+06	6.532e+06	6.535e+06	6.541e+06	6.564e+06
7.000e+00	8.708e-13	3.513e-07	1.830e-04	1.246e-02	9.588e-02	1.857e-01	3.171e-01	4.588e-01	5.671e-01
1.100e+01	8.653e-14	3.491e-08	1.818e-05	1.238e-03	5.921e-03	1.846e-02	3.151e-02	4.559e-02	5.635e-02
total	4.171e-10	2.883e+12	1.257e+13	1.267e+13	1.275e+13	1.298e+13	1.326e+13	1.368e+13	1.436e+13
gpm pow	6.687e-17	4.621e+05	2.015e+06	2.031e+06	2.044e+06	2.080e+06	2.125e+06	2.193e+06	2.302e+06

Figure 14. Photon-decay emission rates as a function of time and photon energy.

photon spectrum for activation products

		principal photon sources in group 1,photons/sec mean energy= 0.015mev							
nuclide	fresh	1.0hr	1.0d	10.0d	30.0d	90.0d	180.0d	1.0yr	3.0yr
cr 51	0.	5.196e+13	1.232e+15	1.104e+16	2.633e+16	4.462e+16	4.931e+16	4.985e+16	4.981e+16
mn 54	0	2.845e+13	6.820e+14	6.753e+15	1.982e+16	5.571e+16	1.013e+17	1.709e+17	2.813e+17
mn 57	0.	1.275e+16	1.275e+16	1.276e+16	1.276e+16	1.277e+16	1.278e+16	1.281e+16	1.291e+16
		principal photon sources in group 2,photons/sec mean energy= 0.025mev							
nuclide	fresh	1.0hr	1.0d	10.0d	30.0d	90.0d	180.0d	1.0yr	3.0yr
cr 56	0	7.245e+09	3.086e+10	3.260e+10	3.565e+10	4.076e+10	4.328e+10	4.408e+10	4.420e+10
co 58m	0	9.864e-01	9.290e+03	2.180e+06	1.780e+07	1.230e+08	3.548e+08	8.953e+08	3.074e+09
		principal photon sources in group 3,photons/sec mean energy= 0.038mev							
nuclide	fresh	1.0hr	1.0d	10.0d	30.0d	90.0d	180.0d	1.0yr	3.0yr
ca 47	0.	9.594e-08	1.280e-03	9.502e-01	1.570e+01	2.187e+02	8.845e+02	3.673e+03	3.333e+04
		principal photon sources in group 4,photons/sec mean energy= 0.058mev							
nuclide	fresh	1.0hr	1.0d	10.0d	30.0d	90.0d	180.0d	1.0yr	3.0yr
co 60m	0.	1.334e+03	1.231e+06	1.176e+08	9.603e+08	6.637e+09	1.914e+10	4.831e+10	1.659e+11
co 61	0.	2.504e-04	2.252e-01	2.778e+04	6.946e+05	1.523e+07	9.337e+07	5.100e+08	5.533e+09
au199	0.	1.094e-04	5.187e+00	4.548e+03	9.439e+04	1.636e+06	7.382e+06	3.318e+07	3.095e+08
pb210	0.	5.383e+03	1.058e+06	1.292e+07	3.933e+07	1.188e+08	2.386e+08	4.875e+08	1.500e+09
		principal photon sources in group 5,photons/sec mean energy= 0.085mev							
nuclide	fresh	1.0hr	1.0d	10.0d	30.0d	90.0d	180.0d	1.0yr	3.0yr
hg203	0.	1.302e+12	3.103e+13	2.905e+14	7.565e+14	1.551e+15	1.958e+15	2.095e+15	2.113e+15
bi208	0.	1.213e+08	2.912e+09	2.913e+10	8.741e+10	2.626e+11	5.261e+11	1.072e+12	3.264e+12
		principal photon sources in group 6,photons/sec mean energy= 0.125mev							
nuclide	fresh	1.0hr	1.0d	10.0d	30.0d	90.0d	180.0d	1.0yr	3.0yr
mn 57	0.	9.690e+15	9.690e+15	9.691e+15	9.693e+15	9.700e+15	9.710e+15	9.730e+15	9.810e+15

Figure 15. Principal nuclide contributors to the photon-decay emission rates.

IMFA, Infinite multiplication factors (= $NPRon/NDESn$) of vectors NKEQ(1), NKEQ(2), and
 IMFB, NKEQ(3), respectively.

FRC (IMFB-IMFA)/(IMFA-IMFC).

FRD FRC*NDESB/NDESC.

The neutron production and destruction rates are relative, because they have not been multiplied by the neutron flux.

The internal information printed for the **FAC** command is relatively simple compared with that for the irradiation and **KEQ** commands. The **FAC** output information in **TAPE15** consists of the value of **NFAC (1)** in the **FAC** instruction and the value of **FACTOR[NFAC (1)]**.

The third function of **TAPE15** is to provide a mechanism for printing internal **FORIG** error messages. Three general types of error messages are contained in **FORIG**.

- The first type of error message is related to size of specified problem. If the specification requires arrays that exceed the size of the arrays actually present, an error message will be output to indicate the dimension exceeded. (Although this was also the intention of the **ORIGEN2** author, some runs produced crashes, indicating that not all possible errors are monitored. Hence, the user should compare the dimensions with the actual array sizes as indicated in the output table of reactivity and burnup data.)
- The second type of message is similar except it is the individual command count that is checked. That is, if the times a particular command is actually used exceeds the allowable number given in Section 4.2, an error message will be printed. Neither of these two error types will stop program execution.
- The third type of error is printed when the command key word, defining the type of command, fails to match one of the 30 key words contained internally in **ORIGEN2**. In this case, a message will be printed and program execution will be terminated.

5.6. TAPE16: Variable Actinide Cross Sections

The output on **TAPE16** relates to the changing of the variable actinide cross sections included in **FORIG**. These cross sections are altered by linear interpolation based on the anticipated burnup during the next irradiation step. Thus, the first output on **TAPE16** contains parameters related to the anticipated burnup during the next irradiation step and also contains the weighting factors used in the cross-section interpolation. A small table is output that contains the following information for each nuclide with a variable cross section.

NUCLID	Six-digit nuclide identifier.
XSEC TYPE	Type of cross section; 1 = (n,gamma), 2 = (n,gamma) to an excited state of the daughter, 4 = (n,fission).
TOCAP (I)	I is the location of the cross section in array TOCAP that contains the total neutron absorption cross section (not meaningful for fission cross sections).
A (N)	N is the location of the reaction rate corresponding to the cross section being varied in the matrix of reaction rates (i.e., A).
FP YIELD INDIC ARR	Number of the array containing the locations of the fission-product yields that must be adjusted when fission cross sections are varied.
FISS (J)	J is the location of the fission cross section in array FISS , which contains all fission cross sections.
OLD XSEC	Value of the cross section during the previous irradiation step.
NEW XSEC	Value of the cross section during the upcoming irradiation step.

Each of these pieces of information, in one way or another, indicates whether the routines that vary the actinide cross sections are functioning properly. Under normal circumstances, this output is not useful and can be suppressed.

5.7. TAPE7: Compositions

This output is generated by the **PCH** commands. The format of the output written on **TAPE7** is the same as the format for **FORIG** input material compositions. The **PCH** command also outputs the average

burnup, flux, and associated power for each material. These values are required for compositions to be read by a subsequent FORIG problem on Unit 4 (TAPE4) but are ignored for compositions to be read from ORINPUT. If TAPE4 is used to input the compositions, the FORIGM (main) program must be altered to open TAPE4 and then must be recompiled.

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6. *Title 10, Code of Federal Regulations, Part 61* ("Licensing Requirement for Land Disposal of Radioactive Waste Disposal"), Federal Register (Dec. 27, 1982).
7. *Title 10, Code of Federal Regulations, Part 20* ("Standards for Protection Against Radiation"), Federal Register (1978).