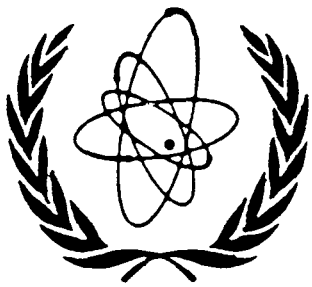
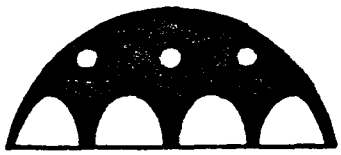
INDC(NDS)-167



International Atomic Energy Agency



Nuclear Reactor of the Technical University Budapest

ECN

Netherlands Energy Research Foundation ECN

TECHNICAL NOTE

ADDITIONAL DATA AND INFORMATION FOR THE REAL84 EXERCISE

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March 11, 1985

SUMMARY

In this technical note some additional data and information are given, which are needed by the participants in the REAL84 exercise. This information supplements the data on the magnetic tape R84.
This note contains also the questionnaire each participant should fill in with his output information and then submit to the evaluation team for the exercise.

INTRODUCTION

The aim and the merits of the REAL-84 exercise are described in [1]. This note gives some extra information and data to be used in combination with the magnetic tape.
The magnetic tape R84 which accompanies this note contains 45 files with data for the 7 input spectrum sets of the exercise , several utility and conversion programs and cross-section libraries which the participant may use.
A guide with some brief information on the tape R84 is given in table 1.
The first file of the tape R84 contains a list of the contents of the tape. This file has been reproduced partly as table 2 of this Technical Note.
Furthermore some extra information for the application of the libraries and programs of the R84 tape is given.
Also the questionnaire and the format of the magnetic tape with the output data from the participants is given.

THE UTILITY AND CONVERSION PROGRAMS

Several small utility programs are written on the tape for the convenience of the participants. In general, they convert the data from a format suitable to store them on a magnetic tape, either to a format which may be handy for the participant, or to a format required by conversion programs written also on the tape.
With some exceptions the programs are all written in ANSI STANDARD FORTRAN. Deviations from the STANDARD, if they occur, will be small in number.
The program written to calculate the group covariances (UNC33) was converted from FORTRAN77 especially for this exercise.
In the case that the participant prefers the FORTRAN77 version, he should request it from ECN.
The application of the programs and the input requirements are in most cases described with comment cards.
For the programs FITOCO and UNC33 (FORTRAN77) reports are included.

The program UNC33 (IV) on the tape was written in ANSI FORTRAN, which is different from FORTRAN-77 used in the version described in the ECN report on UNC33. Therefore minor but essential differences occur in the input deck. The user should take note of the comments in the program.

REACTION RATE COVARIANCES

For a number of input data sets the physical information for the reaction rate covariance matrices is contained only in the diagonal elements. In these cases the off-diagonal elements have been chosen rather arbitrarily.

CROSS-SECTION LIBRARIES

Two cross-section libraries are available:

- file 42 CS640 cross-section library in the extended SAND-II energy group structure with 640 groups.
  49 reaction cross-section sets are available.
- file 45 IRDF-85 cross-section library in the ENDF/B-V format processed in such a way that histogram cross-sections are available in the extended SAND-II energy group structure with 640 groups. The ENDF/B-V format is described in ENDF-102 [2].

The CS640 library contains all reaction cross-sections which are required in the REAL84 exercise (see table 3).
The reaction cross-sections comprise:
- activation reactions;
- gas production cross-sections (coded with GP and gas in their names;
- displacement cross-sections (coded with DISPL. The iron displacement cross-section is coded ST-D-AST, a short for STEEL-DISPL-ASTM)
- nickel double reaction cross-sections (coded with two times GP).

The IRDF-85 library contains also the reactions present in the CS640

library, except:
- the damage cross-sections for chromium and nickel;
- the nickel double reaction cross-sections.

A review of the reaction cross-sections present in the IRDF-85 is given in table 4. This table lists the file numbers (MF) and the reaction type codes (MT).
The key of these data can be found in ENDF-102 [2].
The gas production cross-sections in the IRDF-85 have MAT numbers starting with 5000.
The original point data in the ENDF/B-V format were treated with the programs LINEAR and GROUPIE and this resulted in fine group cross-section data in the extended SAND-II energy group structure for the activation and fission cross-section sets. The uncertainty data remained unaltered.

SOURCES OF CROSS-SECTION DATA SETS

Most of the cross-section data sets used in the exercise originate from the ENDF/B-V or IRDF-82.
The displacement cross-sections for chromium and nickel were made available by L.R. Greenwood (ANL). These cross-sections were calculated with the program DISCS in a 100 point energy grid using nuclear cross-sections from ENDF/B-V. Details on the calculation method will be published in the report "Specter: Neutron damage calculations for materials irradiations" by L.R. Greenwood and R.K. Smither.
The cross-section data sets for the nickel double reaction were also made available by L.R. Greenwood.

THE CALCULATION OF COARSE GROUP CROSS-SECTIONS

For the calculation of coarse group cross-section values in the users group structure both cross-section libraries (IRDF-85 an CS640) can be applied.
For the calculation of these coarse group cross-sections the participant has, in principle, four possibilities:

- To apply his own conversion procedure on the IRDF-85 or on the CS640 data, or probably on both libraries or on his own library;
- To apply the CS640 library in combination with the FITOCO program;
- To apply the output of the UNC33 program, which has as main purpose the calculation of the cross-section uncertainties, but in addition supplies also the coarse cross-section data themselves;
- To use the data from IRDF-85 in combination with the programs LINEAR an GROUPIE.

Fig. 1 gives a flow chart which illustrates these various possibilities.

Remark

In the application of the program FITOCO it should be realized that the lower energy boundary is always 10^{-10} MeV. If this is not required, an extra group can be introduced which has then to be removed from the FITOCO output before applying the coarse group data.

THE CROSS-SECTION UNCERTAINTY INFORMATION

The cross-section uncertainty information for most reactions can be found in files 32 and 33 of the IRDF-85 library. However, for the gas production and the displacement cross-section sets this information is not available in this library.
Where this uncertainty information is lacking, it should be generated by the participant. This procedure is an important part of this exercise in lack of better information, we propose to apply for these uncertainties the following data for the gas production and displacement crosssections.

material	standard deviation for all cross-section points (in per cent)
iron	10
nickel	12
chromium	18

The correlation of the uncertainties for the various energy points of a cross-section set can be described with a Gaussian function.
The correlation coefficient calculated with this function should depend on the difference of the natural logarithms of the energy points considered. the standard deviation of the Gaussian should be equal to $\sigma=0.9782$.

THE CALCULATION OF THE GROUP COVARIANCES

The participant can calculate the group cross-section covariances with his own procedure from the IRDF-85 or from another cross-section library using the file 32 and 33 information.
For this calculation also the conversion program UNC33 can be applied This program starts with data from the IRDF-85 files 32 and 33, but for simplification of the calculation the data from file 32 are already preprocessed in a fine group structure (640 groups SAND-II type). These preprocessed data are written on the R84 tape as files 35 ... 37. With utility programs these files are made suitable for the program UNC33.
For more details on the use of the program see fig. 1 and the program description.

REAL84 STEEL COMPOSITION

The participants are requested to calculate the output data for the following fictitious steel composition.

element	atom percentage
Fe	71
Cr	18
Ni	11

This composition does not refer to an actual steel type.
In pressure vessel steels more elements will be present, but the actual method to treat the data will be the same as the one to be used here.

OUTPUT DATA OF THE PARTICIPANT

The participants are requested to fill in the questionnaire and to return it to:

W.L. Zijp
Netherlands Energy Research Foundation ECN
P.O. Box 1
1755 ZG Petten
The Netherlands.

The questionnaire is added to this Note as appendix 1.
Please use this questionnaire to present your results. In case you need more copies of it, please reproduce this appendix 1.

It is appreciated if you write the output spectrum and covariance matrix on a magnetic tape. Please send that tape to:

D.E. Cullen
IAEA Nuclear Data Section
P.O. Box 100
A-1400 Vienna
Austria.

That magnetic tape (written with EBCDIC code) should contain the following information:

- spectrum identification;
- participant identification;
- the number of groups or points;
- for point data the interpolation scheme or otherwise upper energy;
- the definition of the fluence rate values (fluence rate per unit energy etc.);
- the lower energy boundaries of the groups (in increasing energy sequence);
- the fluence rate values in the same sequence;
- the description of the covariance matrix (i.e. relative or absolute and complete, upper triangle or lower triangle);
- the covariance matrix (same sequence as for energies).

The output can be given in a list directed way.
Each spectrum should be a file on the tape.

REFERENCES

[1] Zijp, W.L., Zsolnay, É.M., and Cullen, D.E.: "Information sheet for the REAL84 exercise", Report INDC(NDS)-166 (IAEA, NDS, Vienna, March 1st, 1985).

[2] Kinsey, R., and Magurno, B.A.: "ENDF-102. Data formats and procedures for the evaluated nuclear data file ENDF/B-V", BNL-NCS-50496 (ENDF-102) (Brookhaven National Laboratory, National Nuclear Data Center, New York, 1984).

Table 1. USER'S GUIDE TO THE REAL84 TAPE

THE TAPE HAS NO LABELS. (THE DATA FILES ARE NOT PRECEDED BY 'HEADER LABELS' AND THEY ARE NOT FOLLOWED BY 'EOF LABELS'.)

START WITH PRINTING THE FILE 1 USING AN OFF-LINE PRINTER. THE CARRIAGE CONTROL CHARACTERS (BLANK, 0 AND 1) ARE USED. THE OUTPUT WILL BE THE TABLE OF CONTENTS OF THE REAL84 TAPE.

HAVING THIS TABLE OF CONTENTS, COMPILE THE UTILITY PROGRAMS GIVEN IN THE DATA FILES 2, 3, 4 AND 5. THESE PROGRAMS CAN BE TREATED BY MEANS OF ANY FORTRAN IV COMPILER. THE I/O FILES ARE DESCRIBED IN COMMENT LINES IN THE UTILITY PROGRAMS.

EXECUTE THESE UTILITY PROGRAMS WITH THE DATA SETS OF YOUR SELECTED SPECTRA TO GENERATE THE INPUT DATA FOR YOUR NEUTRON SPECTRUM ADJUSTMENT PROGRAM.

AS SOURCES FOR FINE GROUP CROSS SECTION DATA ARE HERE AVAILABLE
1) THE CROSS SECTION LIBRARY CS640,
2) THE CROSS SECTION LIBRARY IRDF-85.

IF YOU WANT TO USE THE CROSS SECTION LIBRARY CS640 GIVEN IN THE 640 GROUPS SAND-II STRUCTURE IN FILE 42, EXECUTE THE UTILITY PROGRAM OF FILE 40. IF THIS LIBRARY IS PLANNED TO SERVE AS BASIS FOR THE PROBLEM-DEPENDENT INFORMATION, YOU CAN USE THE 'FITOCD' UTILITY PROGRAM OF FILE 41 (SEE REPORT ECN-92).

IF YOU WANT TO USE THE IRDF-85 LIBRARY PRESENT IN FILE 45 INSTEAD OF THE GIVEN SAND-II TYPE CROSS SECTION LIBRARY CS640, THE PROGRAMS 'LINEAR' AND 'GROUPIE' ARE AVAILABLE IN THE FILES 43 AND 44. THE 'HOW TO USE' INFORMATION IS WRITTEN IN COMMENT LINES IN THESE PROGRAMS.

IF THE SELECTED SPECTRA CONTAIN ONE OR MORE OF THE REACTIONS NP237F, FE58G, CU63G, YOU CAN USE THE UTILITY PROGRAM OF FILE 34 TO GENERATE THE RELATIVE COVARIANCE MATRICES OF THE PERTINENT CROSS SECTIONS, BASED ON THE 'MF=32,MT=151' DATA OF THE IRDF-85, AS GIVEN IN THE FILE 45.

YOU CAN USE THE 'UNC33' PROGRAM TO GENERATE THE COMPLETE CROSS SECTION COVARIANCE LIBRARY (SEE REPORT ECN-85-30). THE PROGRAM IS MADE AVAILABLE IN ITS ORIGINAL FORTRAN77 VERSION AS FILE 39, AND ALSO IN FORTRAN IV VERSION AS FILE 38.

Table 2. TABLE OF CONTENTS PAGE 4

FILE NO.	NO. OF RECORDS	INFORMATION BLOCK ('FILE')
1	263	TABLE OF CONTENTS OF THIS TAPE
2	43	PROGRAM TO READ THE REACTION RATES (THIS PROGRAM READS THE DATA FILES FROM THIS TAPE AND MAKES THEM AVAILABLE FOR STORAGE IN YOUR SYSTEM.)
3	60	PROGRAM TO READ THE COVARIANCE MATRICES OF THE REACTION RATES. (THIS PROGRAM READS THE DATA FILES WRITTEN ON THIS TAPE IN COMPRSSED FORMAT AND MAKES THEM AVAILABLE FOR STORAGE IN YOUR SYSTEM.)
4	41	PROGRAM TO READ THE INPUT SPECTRUM (LIKE FOR FILE 2)
5	64	PROGRAM TO READ THE COVARIANCE MATRICES OF THE INPUT SPECTRA (LIKE FOR FILE 3)
6	4	REACTION RATES IN THE 'ANO' SPECTRUM (PRESSURE-VESSEL CAVITY OF THE ARKANSAS POWER AND LIGHT REACTOR, ARKANSAS NUCLEAR ONE-1) NUMBER OF REACTIONS = 6
7	5	COVARIANCE MATRIX OF THE REACTION RATES IN THE 'ANO' SPECTRUM
8	17	INPUT SPECTRUM FOR THE 'ANO' REACTOR NUMBER OF ENERGY GROUPS = 55
9	16	COVARIANCE MATRIX OF THE INPUT SPECTRUM FOR THE 'ANO' REACTOR ********** * NUMBER OF ENERGY GROUPS = 16 * **********
10	5	REACTION RATES IN THE 'PS1' SPECTRUM (OAK RIDGE RESEARCH REACTOR POOLSIDE FACILITY IN THE METALLURGICAL IRRADIATION EXPERIMENT, POSITION SIMULATED SURVEILLANCE CAPSULE) NUMBER OF REACTIONS = 10
11	8	COVARIANCE MATRIX OF THE REACTION RATES IN THE 'PS1' SPECTRUM ********** * DIAGONAL MATRIX ONLY * **********
12	13	INPUT SPECTRUM FOR THE 'PS1' CASE NUMBER OF ENERGY GROUPS = 37
13	56	COVARIANCE MATRIX OF THE INPUT SPECTRUM FOR THE 'PS1' CASE

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FILE NO.	NO. OF RECORDS	INFORMATION BLOCK ('FILE')
14	4	REACTION RATES IN THE 'PS2' SPECTRUM (OAK RIDGE RESEARCH REACTOR POOLSIDE FACILITY IN THE METALLURGICAL IRRADIATION EXPERIMENT, T/4 POSITION OF THE SIMULATED PRESSURE VESSEL) NUMBER OF REACTIONS = 6
15	5	COVARIANCE MATRIX OF THE REACTION RATES IN THE 'PS2' SPECTRUM * DIAGONAL MATRIX ONLY * **********
16	13	INPUT SPECTRUM FOR THE 'PS2' CASE NUMBER OF ENERGY GROUPS = 37
17	56	COVARIANCE MATRIX OF THE INPUT SPECTRUM FOR THE 'PS2' CASE
18	5	REACTION RATES IN THE 'RTN' SPECTRUM (FUSION SIMULATION SPECTRUM MEASURED AT THE RTNS-II, A 14 MEV NEUTRON SOURCE AT LAWRENCE LIVERMORE LABORATORY. THE SPECTRUM IS A PRETTY FAIR SIMULATION OF A FUSION FIRST WALL SPECTRUM.) NUMBER OF REACTIONS = 12
19	9	COVARIANCE MATRIX OF THE REACTION RATES IN THE 'RTN' SPECTRUM
20	19	INPUT SPECTRUM FOR THE 'RTN' CASE NUMBER OF ENERGY GROUPS = 60
21	133	COVARIANCE MATRIX OF THE INPUT SPECTRUM FOR THE 'RTN' CASE
22	7	REACTION RATES IN THE 'TAN' SPECTRUM (ACCELERATOR SPECTRUM BE(D,N), WITH DEUTERON ENERGY OF 16 MEV) NUMBER OF REACTIONS = 17

23	16	COVARIANCE MATRIX OF THE REACTION RATES IN THE 'TAN' SPECTRUM
24	14	INPUT SPECTRUM FOR THE 'TAN' CASE NUMBER OF ENERGY GROUPS = 39
25	61	COVARIANCE MATRIX OF THE INPUT SPECTRUM FOR THE 'TAN' CASE
26	8	REACTION RATES IN THE 'U35' SPECTRUM (FISSION SPECTRUM OF U-235) NUMBER OF REACTIONS = 21

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FILE NO.	NO. OF RECORDS	INFORMATION BLOCK ('FILE')
27	21	COVARIANCE MATRIX OF THE REACTION RATES IN THE 'U35' SPECTRUM
28	10	INPUT SPECTRUM FOR THE 'U35' CASE NUMBER OF ENERGY GROUPS = 24
29	28	COVARIANCE MATRIX OF THE INPUT SPECTRUM FOR THE 'U35' CASE
30	8	REACTION RATES IN THE 'CFR' SPECTRUM (NEUTRON SPECTRUM IN THE CENTRE OF THE COUPLED FAST REACTIVITY MEASUREMENT FACILITY, CFRMF.) NUMBER OF REACTIONS = 23
31	23	COVARIANCE MATRIX OF THE REACTION RATES IN THE 'CFR' SPECTRUM
32	11	INPUT SPECTRUM FOR THE 'CFR' CASE NUMBER OF ENERGY GROUPS = 26
33	32	COVARIANCE MATRIX OF THE INPUT SPECTRUM FOR THE 'CFR' CASE
34	202	PROGRAM TO READ THE CROSS SECTION COVARIANCE MATRICES DERIVED FROM UNCERTAINTY FILE TYPE 32 IN ENDF/B-V FORMAT. (THIS PROGRAM READS THE DATA FILES FROM THIS TAPE AND MAKES THEM AVAILABLE FOR STORAGE IN YOUR SYSTEM.)
35	797	CROSS SECTION COVARIANCE MATRIX FOR THE NP237(N,F)F.P. REACTION DERIVED FROM THE FILE-32 IN THE ENDF/B-V FORMAT (MAT=6337, MF=32, MT=151) THE ENERGY BOUNDARIES OF THE GROUP STRUCTURE USED ARE CONSISTENT WITH THE 640 GROUPS STRUCTURE OF SAND-II TYPE.
36	7570	CROSS SECTION COVARIANCE MATRIX FOR THE FE58(N,G)FE59 REACTION DERIVED FROM THE FILE-32 IN THE ENDF/B-V FORMAT (MAT=6432, MF=32, MT=151) THE ENERGY BOUNDARIES OF THE GROUP STRUCTURE USED ARE CONSISTENT WITH THE 640 GROUPS STRUCTURE OF SAND-II TYPE.
37	1129	CROSS SECTION COVARIANCE MATRIX FOR THE CU63(N,G)CU64 REACTION DERIVED FROM THE FILE-32 IN THE ENDF/B-V FORMAT (MAT=6435, MF=32, MT=151) THE ENERGY BOUNDARIES OF THE GROUP STRUCTURE USED ARE CONSISTENT WITH THE 640 GROUPS STRUCTURE OF SAND-II TYPE.

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FILE NO.	NO. OF RECORDS	INFORMATION BLOCK ('FILE')
38	984	PROGRAM FOR GENERATING THE CROSS SECTION COVARIANCE MATRICES STARTING FROM THE FILES 32+33 OF THE ENDF/B-V FORMAT. PROGRAM NAME: UNC33IV.
39	156	PROGRAM TO READ THE CROSS SECTION COVARIANCE LIBRARY GENERATED BY THE UNC33IV PROGRAM, WHICH IS GIVEN IN FILE 38 OF THIS TAPE.
40	88	PROGRAM TO READ THE CROSS SECTION LIBRARY IN THE ENERGY STRUCTURE OF THE 640 GR. SAND-II TYPE (LIKE FILE 2)
41	211	PROGRAM TO CONVERT FINE GROUP DATA TO COARSE GROUP DATA REFERENCE: H.CH. RIEFFE: FITOCO, A PROGRAM FOR THE CONVERSION OF FINE GROUP FLUX DENSITY AND CROSS SECTION DATA TO COARSE GROUP VALUES. REPORT ECN-92. NETHERLANDS ENERGY RESEARCH FOUNDATION, PETTEN, APRIL 1981.
42	4345	CROSS SECTION LIBRARY IN 640 GR. SAND-II TYPE (CS640) STRUCTURE, BASED ON THE IRDF-85 DATA NUMBER OF COVER MATERIALS = 3 NUMBER OF CROSS SECTION SETS = 49
43	2228	PROGRAM TO CONVERT ENDF/B FORMAT EVALUATED DATA TO A TABULAR FORM SUBJECT TO LINEAR INTERPOLATION IN ENERGY AND CROSS SECTION. PROGRAM NAME: LINEAR REFERENCE: D.E. CULLEN: PROGRAM LINEAR (VERSION 79-1) LINEARIZE DATA IN EVALUATED NUCLEAR DATA FILE/VERSION B (ENDF/B) FORMAT. REPORT UCRL 50400 VOL. 17, PART A, REV. 2 LAWRENCE LIVERMORE LABORATORY, OCTOBER 31, 1979.

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FILE NO.	NO. OF RECORDS	INFORMATION BLOCK ('FILE')
44	5497	PROGRAM TO CALCULATE AMONG OTHERS GROUP CROSS SECTION DATA FROM ENDF/B FORMAT DATA SUBJECT TO LINEAR INTERPOLATION PROGRAM NAME: GROUPIE REFERENCE: D.E. CULLEN: PROGRAM GROUPIE (VERSION 79-1) CALCULATION OF BONDARENKO SELFSHIELDED NEUTRON CROSS SECTIONS AND MULTIBAND PARAMETERS FROM DATA IN ENDF/B FORMAT REPORT UCRL 50400 VOL. 17, PART D, LAWRENCE LIVERMORE LABORATORY, JULY 4, 1980.
45	20435	CROSS SECTION LIBRARY IRDF-85 (INTERNATIONAL REACTOR DOSIMETRY FILE),IN HISTOGRAM FORMAT

Table 3. List of cross-section sets in CS640.

NC =	3	
	CD	
	B	
	AL	
NL =	49	
1	AL27P	AL27(N,P)MG27
2	AL27A	AL27(N,A)NA24
3	MN552	MN55(N,2N)MN54
4	CO592	CO59(N,2N)CO58
5	CO59G	CO59(N,G)CO60
6	CO59A	CO59(N,A)MN56
7	NP237F	NP237(N,F)F.P.
8	AU197G	AU197(N,G)AU198
9	TH232F	TH232(N,F)F.P.
10	TH232G	TH232(N,G)TH233
11	U235F	U235(N,F)F.P.
12	U238F	U238(N,F)F.P.
13	U238G	U238(N,G)U239
14	PU239F	PU239(N,F)F.P.
15	LI6A	LI6(N,A)H3
16	B10A	B10(N,A)LI7
17	SC45G	SC45(N,G)SC46
18	TI46P	TI46(N,P)SC46
19	TI47P	TI47(N,P)SC47
20	TI48P	TI48(N,P)SC48
21	FE54P	FE54(N,P)MN54
22	FE56P	FE56(N,P)MN56
23	FE58G	FE58(N,G)FE59
24	NI582	NI58(N,2N)NI57
25	NI58P	NI58(N,P)CO58
26	NI60P	NI60(N,P)CO60
27	CU63G	CU63(N,G)CU64
28	CU63A	CU63(N,A)CO60
29	IN115N	IN115(N,N)IN115
30	IN115G	IN115(N,G)IN116
31	I1272	I127(N,2N)I126
32	S32P	S32(N,P)P32
33	CR-DISPL	CHROMIUM-DISPL.
34	CR-GP-H	CHROMIUM-GAS-H
35	CR-GP-D	CHROMIUM-GAS-D
36	CR-GP-T	CHROMIUM-GAS-T
37	CR-GPHE3	CHROMIUM-GAS-HE3
38	CR-GP-HE	CHROMIUM-GAS-HE
39	ST-D-AST	STEEL-DISPL-ASTM
40	FE-GP-H	IRON-GAS-H
41	FE-GP-HE	IRON-GAS-HE
42	NI-DISPL	NICKEL-DISPL.
43	NI-GP-H	NICKEL-GAS-H
44	NI-GP-D	NICKEL-GAS-D
45	NI-GP-HE	NICKEL-GAS-HE
46	NI59G-GP	NI59(N,G)NI60-GP
47	NI59P-GP	NI59(N,P)CO59-GP
48	NI59A-GP	NI59(N,A)FE56-GP
49	NI58G-GP	NI58(N,G)NI59-GP

Table 4. INDEX TO THE IRDF-85
TAPE NO = 9000

MAT	MATERIAL	MF/MT						
920	F-19	1/451	3/16	33/16				
1009	AM-241	1/451	2/151	3/18				
1120	NA-23	1/451	3/16	33/16				
1220	MG-24	1/451	3/103	33/103				
1520	P-31	1/451	3/103	33/103				
2920	CU-63	1/451	3/16	33/16				
3020	ZN-64	1/451	3/103	33/103				
4020	ZR-90	1/451	3/16	33/16				
4120	NB-93	1/451	3/51	33/51				
4520	RH-103	1/451	3/51	33/51				
5160	B-11	1/451	2/151	3/203	3/205	3/207		
5275	N-14	1/451	2/151	3/203	3/207			
5303	LI-6	1/451	2/151	3/203	3/204	3/205	3/207	
5304	BE-9	1/451	2/151	3/203	3/204	3/205	3/207	
5305	B-10	1/451	2/151	3/203	3/204	3/207		
5306	C-0	1/451	2/151	3/203	3/207			
5309	F-19	1/451	2/151	3/203	3/207			
5313	AL-27	1/451	2/151	3/203	3/207			
5314	SI-0	1/451	2/151	3/203	3/207			
5322	TI-0	1/451	2/151	3/203	3/207			
5323	V-0	1/451	2/151	3/203	3/207			
5324	CR-0	1/451	2/151	3/203	3/204	3/205	3/206	3/207
5325	MN-55	1/451	2/151	3/203	3/207			
5326	FE-0	1/451	2/151	3/203	3/207			
5327	CO-59	1/451	2/151	3/203	3/207			
5328	NI-0	1/451	2/151	3/203	3/204	3/207		
5329	CU-0	1/451	2/151	3/203	3/207			
5397	LI-7	1/451	2/151	3/203	3/204	3/205	3/207	33/203
5397	LI-7	33/204	33/205	33/207				
6311	NA-23	1/451	2/151	3/102	32/151	33/102		
6313	AL-27	1/451	2/151	3/103	3/107	33/103	33/107	
6325	MN-55	1/451	2/151	3/16	33/16			
6327	CO-59	1/451	2/151	3/16	3/102	3/107	33/16	33/102
6327	CO-59	33/107						
6337	NP-237	1/451	2/151	3/18	32/151	33/18		
6379	AU-197	1/451	2/151	3/102	33/102			
6390	TH-232	1/451	2/151	3/18	3/102	33/18	33/102	
6395	U-235	1/451	2/151	3/18	33/18			
6398	U-238	1/451	2/151	3/18	3/102	33/18	33/102	
6399	PU-239	1/451	2/151	3/18	33/18			
6424	LI-6	1/451	2/151	3/207	33/207			
6425	B-10	1/451	2/151	3/207	33/207			
6426	SC-45	1/451	2/151	3/102	33/102			
6427	TI-46	1/451	2/151	3/103	33/103			
6428	TI-47	1/451	2/151	3/28	3/103	33/28	33/103	
6429	TI-48	1/451	2/151	3/28	3/103	33/28	33/103	
6430	FE-54	1/451	2/151	3/103	33/103			
6431	FE-56	1/451	2/151	3/103	33/103			
6432	FE-58	1/451	2/151	3/102	32/151	33/102		
6433	NI-58	1/451	2/151	3/16	3/103	33/16	33/103	
6434	NI-60	1/451	2/151	3/103	33/103			
6435	CU-63	1/451	2/151	3/102	3/107	32/151	33/102	33/107
6436	CU-65	1/451	2/151	3/16	33/16			
6437	IN-115	1/451	2/151	3/51	3/102	9/102	33/51	33/102
6438	I-127	1/451	2/151	3/16	33/16			
6439	S-32	1/451	2/151	3/103	33/103			
8000	ST-ASTM	1/451	3/800					
8001	ST-EUR	1/451	3/801					

EACH 1/451 SECTION CONTAINS THE COMMENT:
********** PROGRAM GROUPIE (84-2) **********
UNSHIELDED GROUP AVERAGES USING 640 GROUPS (SAND-II) EXTEND)

..

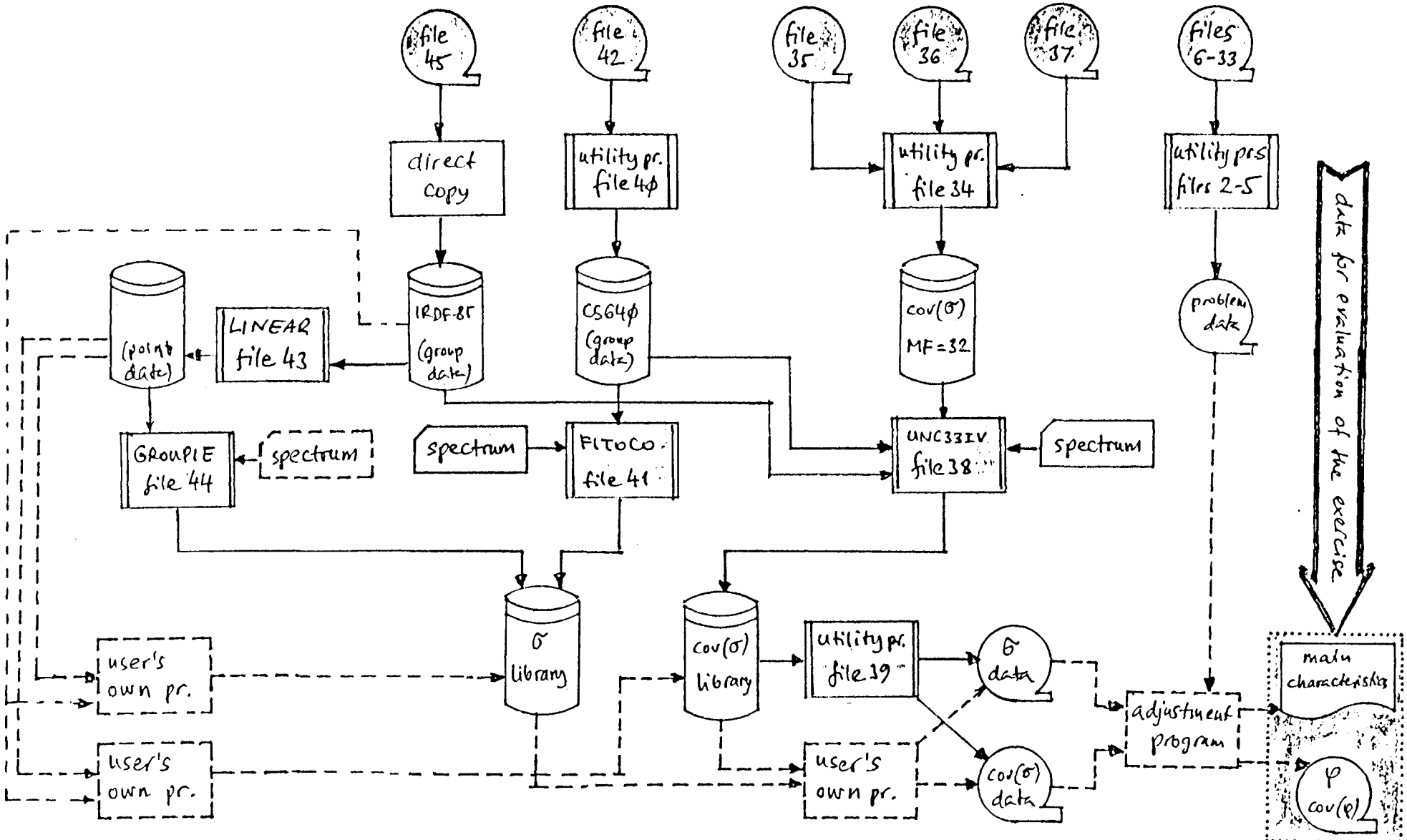

Fig. 1. Flow chart for data treatement of the REAL84 tape

Appendix.

QUESTIONNAIRE ON REAL84 SOLUTIONS

The participants are requested to fill in as many of the questions as possible.

GENERAL QUESTIONS

Name of participant:
Name of laboratory :
Detailed address :

1. Name of adjustment code:

2. Literature reference for adjustment code:

3. What is the advantage of your code from a physics point of view:

4. Which source of cross section data did you use in the adjustment?
   - 0 the International Reactor Dosimetry File IRDF-85
   - 0 the 640 group Cross Section file CS640
   - 0 other source (please specify):

5. For the calculation of coarse group cross section data and the corresponding cross section covariance matrices one needs a weighting spectrum. Which method was applied to derive the required shape of the weighting spectrum from the coarse group spectrum data?
   - 0 log-log interpolation of ϕ_E point values for the energy points calculated from the group energy boundaries E_L and E_U with the formula

$$\bar{E} = (E_U - E_L) \; / \; (\ell n \; E_U - \ell n \; E_L)$$

O spline method with conservation of group neutron fluences

O other method (please specify):

6. Which calculation method was applied for the determination of cross section covariances?

O the program UNC33 present on the R84 tape

O other method (please specify):

7. Which method was applied for the conversion of point or fine group data to coarse group data?

O the program GROUPIE present on the R84 tape

O the program FITOCO present on the R84 tape

O other method (please specify):

8. Which method was applied in the treatment of covariance matrices for the conversion from one group structure to another?
Please specify, or give the relevant literature reference:

9. If you calculated a fitting parameter, how is this defined?
Please specify, or give the relevant literature reference:

10. In which way is your normalization factor defined?
With the notation of the final report on REAL80, [1].

O $f_1 = \Sigma \alpha_i^c . \alpha_i^m \; / \; \Sigma (\alpha_i^c)^2$

O $f_2 = (\Sigma \alpha_i^m \; / \; \alpha_i^c \;) / \; n$

O $f_3 = [\Sigma (\alpha_i^c \; / \; \alpha_i^m)] \; / \; [\Sigma (\alpha_i^c \; / \; \alpha_i^m)^2]$

- $f_4 = \Sigma\alpha_i^m \; / \; \Sigma\alpha_i^c$

- f_0, based on least squares fit.

QUESTIONS FOR SPECTRUM ...

11. Number of groups used in adjustment:

12. Number of reactions used in adjustment:

13. Value of fitting parameter (if available):

14. Value of your normalization factor of input spectrum:

15.

no	reaction name	data for input spectrum				data for output spectrum		
		A_c	$s_\sigma(A_c)$	$s_\phi(A_c)$	$s_p(A_c)^*$	A_c	$s_\phi(A_c)$	$s_p(A_c)^*$
1								
2								
3								
.								
.								
.								
.								
.								
.								
.								
.								
.								
.								
23								

Specification (if relevant) of quantity $s_p(A_c)$, listed by participant

16.

Spectrum characteristics		
quantity	value	standard deviation*
ϕ (>0,1 MeV) ϕ (>1,0 MeV) ϕ_{int} ϕ_{Ni} ϕ_{tot} ϕ_{th} ϕ_{Co}		

17.

Integral damage characteristics		
quantity	value	standard deviation*
R_{dpa}(Fe) R_{dpa}(St) R_{He}(St) R_{H}(St) N_{He}(St) for 1 year		

* Are correlation matrices for these quantities at the left side and for quantities at the right side available? If so, please transmit them.

18. Did you observe inconsistencies or clearly incorrect values in the input data set?
Please describe:

19. In the case of inconsistent or incorrect input values, what was your remedy?
Please describe:

20. Which changes or additions in the input data are required to obtain improved characterization of the damage characteristics?

21. And what gain do you expect and did you experience from changes (or additions)?

22. Have you comments on the group structure of the input spectrum of this data set?
Please describe:

23. Which uncertainty values in the input data set will (in your opinion) determine mainly the uncertainty in the integral damage characteristics?

And, if relevant, the energy range?
Could you elaborate on this question?

ADDENDUM

Additional Data and Information for the REAL84 Exercise

INDC(NDS)-167

The information in INDC(NDS)-167 concerning the contents of the International Reactor Dosimetry File (IRDF-85) have been superseded by the recent release of IRDF-85 and its documentation "The International Reactor Dosimetry File" (IRDF-85, IAEA-NDS-11, Rev. 1, April 1985). In addition to the data described in INDC(NDS)-167 the new IRDF-85 library contains neutron cross section for 58-Ni and 59-Ni and damage cross section for Nickel and Chromium.

The length of the current IRDF85 file is 21600 records (as opposed to the 20435 records stated at the bottom of p. 15 of INDC(NDS)-167).