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SECOND IAEA ADVISORY GROUP MEETING

ON

FISSION PRODUCT NUCLEAR DATA

Petten, Netherlands, 5-9 September 1977

Annexes and selected contributions

to the Review Papers

May 1978

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FOREWORD

This report is a supplement to the Proceedings of the Second IAEA Advisory Group Meeting on Fission Product Nuclear Data held at ECN Petten, Netherlands, which are published in 2 volumes as Technical Report IAEA-213.

The first part contains some Annexes to review papers, which were too bulky to be included in the Proceedings.

The second part contains selected contributions to review-papers, including - among others - all papers which were presented at the meeting in addition to the review papers.

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ANNEX TO REVIEW PAPER 4

(The importance of FPND in reactor design and operation)

C. Devillers
CEA Saclay, France

In this Annex the results of the sensitivity studies and of the error calculations for the FP decay heat are presented for the following five cases:

- ^{233}U thermal fission
- ^{235}U thermal fission
- ^{239}Pu thermal fission
- ^{235}U fast fission
- ^{239}Pu fast fission

In all cases, the irradiation time is 900 days, and the fission rate is 1 fission/sec. Neutron capture effects are not taken into account (see RP 4, § 4, Table VIII).

In the following there is to be found, for each of the 5 cases, one Table summarizing the errors as a function of cooling time, followed by Tables giving, for cooling times $10^0, 10^1, 10^2, \ldots 10^8$ sec, the errors due to those individual data for which the sensitivity is >0.01.

Note: The total error due to yields is calculated by:

$$\left(\frac{\Delta F}{F}\right)_y = \left\{ \sum_i \text{SIGN (i)} \left(\frac{\Delta F}{F}\right)_{yi}^2 \right\}^{1/2}$$

where $\left(\frac{\Delta F}{F}\right)_{yi}$ is the value given in column 5, and

SIGN (i) is the sign preceding this value.

LIST OF TABLES

^{233}U thermal

Table 1: "Uncertainties in FP afterheat vs cooling time"

Tables 2-10: "Analysis of afterheat uncertainties" for cooling times: $10^0, 10^1, 10^2, \ldots 10^8$ sec respectively

^{235}U thermal

Table 11: "Uncertainties in FP afterheat vs cooling time"

Tables 12-20: "Analysis of afterheat uncertainties" for cooling times: $10^0, 10^1, 10^2, \ldots 10^8$ sec respectively

^{239}Pu thermal

Table 21: "Uncertainties in FP afterheat vs cooling time"

Tables 22-30: "Analysis of afterheat uncertainties" for cooling times: $10^0, 10^1, 10^2, \ldots 10^8$ sec respectively

^{235}U fast

Table 31: "Uncertainties in FP afterheat vs cooling time"

Tables 32-40: "Analysis of afterheat uncertainties" for cooling times: $10^0, 10^1, 10^2, \ldots 10^8$ sec respectively

^{238}U fast

Table 41: "Uncertainties in FP afterheat vs cooling time"

Tables 42-50: "Analysis of afterheat uncertainties" for cooling times: $10^0, 10^1, 10^2, \ldots 10^8$ sec respectively

EXPLANATION OF HEADINGS

Tables 1,11,21,31

Title: Uncertainty in residual power due to fission products
Constant fission rate (1 fis./sec)
Irradiation time 7.776×10^7 sec

col. 1: cooling time (sec)
col. 2: afterheat (MeV/sec)
col. 3: error due to fission yields (%)
col. 4: error due to half-lives (%)
col. 5: error due to $E_{\beta+\gamma}$ (%)
col. 6: total error

Tables 2-10, 12-20, 22-30, 32-40, 42-50

Title: Residual power uncertainty analysis
Constant fission rate (1 fis./sec)
Irradiation time: 7.776×10^7 sec
Cooling time:
Residual power:
Uncertainty due to fission yields:
Uncertainty due to half-lives:
Uncertainty due to energies:
Total uncertainty

col. 1: nuclide

col. 2, col. 3, col. 4: afterheat sensitivity due to { fission yield / half-life / $\overline{E}_{\beta+\gamma}$ }

col. 5, col. 6, col. 7: afterheat uncertainty due to { fission yield / half-life / $\overline{E}_{\beta+\gamma}$ }

Table 1

ERREUR SUR LA PUISSANCE RESIDUELLE DUE AUX PRODUITS DE FISSION
***** U233 THERMIQUE *****
TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION 7.776E 07 S

TEMPS REF. (S)	PUISSANCE RESIDUELLE (MEV/S)	ERREUR DUE AU REND. (%)	ERREUR DUE A LA PER. (%)	ERREUR DUE AUX ENERGIES (%)	ERREUR TOTALE
1.0E 00	9.3849E 00	0.953	0.155	1.405	1.705
2.0E 00	9.1024E 00	0.919	0.134	1.395	1.676
5.0E 00	8.5655E 00	0.856	0.114	1.362	1.613
1.0E 01	8.0362E 00	0.796	0.136	1.311	1.540
2.0E 01	7.4262E 00	0.737	0.185	1.235	1.450
5.0E 01	6.5210E 00	0.665	0.253	1.115	1.323
1.0E 02	5.7848E 00	0.617	0.321	0.993	1.213
2.0E 02	5.0694E 00	0.590	0.382	0.837	1.093
5.0E 02	4.2133E 00	0.582	0.308	0.696	0.958
1.0E 03	3.6100E 00	0.585	0.146	0.645	0.883
2.0E 03	3.0190E 00	0.581	0.080	0.623	0.855
5.0E 03	2.3071E 00	0.538	0.105	0.665	0.862
1.0E 04	1.8490E 00	0.517	0.175	0.722	0.905
2.0E 04	1.4582E 00	0.517	0.311	0.776	0.983
5.0E 04	1.0557E 00	0.550	0.520	0.846	1.135
1.0E 05	8.4080E-01	0.608	0.478	0.940	1.217
2.0E 05	6.8683E-01	0.681	0.236	1.072	1.291
5.0E 05	5.2607E-01	0.802	0.112	1.267	1.504
1.0E 06	4.1357E-01	0.953	0.148	1.428	1.724
2.0E 06	3.0518E-01	1.175	0.198	1.628	2.018
5.0E 06	1.8490E-01	1.432	0.165	1.998	2.464
1.0E 07	1.1361E-01	1.373	0.126	2.445	2.807
2.0E 07	6.0153E-02	0.929	0.136	3.260	3.393
5.0E 07	2.2942E-02	0.578	0.554	3.754	3.838
1.0E 08	1.1686E-02	0.779	0.924	2.742	2.996

Table 2

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U233 THERMIQUE TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07	S
TEMPS DE REFROIDISSEMENT	1.000E 00	S
PUISSANCE RESIDUELLE	9.385E 00	(MEV/S)
ERREUR DUE AUX RENDEMENTS	0.953	%
ERREUR DUE AUX PERIODES	0.155	%
ERREUR DUE AUX ENERGIES	1.405	%
ERREUR TOTALE	1.705	%

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 86	0.007	-0.000	0.012	-0.11	0.00	0.11
BR 88	0.028	0.000	0.013	0.24	0.00	0.45
BR 89	0.014	0.001	0.005	0.26	0.00	0.16
KR 88	0.019	-0.001	0.012	-0.16	0.00	0.02
RB 88	0.001	0.000	0.022	-0.05	0.00	0.05
KR 89	0.034	0.000	0.019	0.07	0.00	0.05
RB 89	0.004	-0.001	0.021	-0.15	0.00	0.03
KR 90	0.032	0.000	0.013	0.13	0.00	0.03
RB 90F	0.005	0.000	0.023	-0.08	0.00	0.06
KR 91	0.021	0.000	0.005	0.10	0.00	0.18
RB 91	0.023	-0.000	0.024	0.27	0.00	0.09
RB 92	0.027	0.003	0.019	0.36	0.00	0.11
RB 93	0.019	0.001	0.011	0.34	0.00	0.03
SR 92	0.008	-0.000	0.010	-0.24	0.00	0.02
Y 92	0.000	0.000	0.011	-0.01	0.00	0.06
SR 93	0.019	0.000	0.018	-0.22	0.00	0.15
SR 94	0.026	-0.000	0.012	-0.14	0.00	0.28
Y 94	0.003	-0.000	0.020	-0.10	0.00	0.15
SR 95	0.029	0.001	0.014	0.34	0.01	0.32
Y 95	0.010	-0.000	0.013	-0.20	0.00	0.07
SR 96	0.013	0.001	0.004	0.25	0.00	0.15
Y 96	0.013	0.000	0.020	0.07	0.00	0.70
Y 97	0.013	0.003	0.006	0.10	0.02	0.05
ZR 98	0.011	0.001	0.006	-0.06	0.01	0.05
ZR 99	0.016	0.002	0.008	0.07	0.02	0.07
ZR100	0.014	0.000	0.004	0.16	0.00	0.14
NB100F	0.002	0.001	0.011	-0.09	0.02	0.32
TE132	0.011	0.000	0.002	-0.03	0.00	0.01
I132F	0.000	-0.000	0.014	-0.01	0.00	0.02
TE133M	0.013	0.000	0.008	-0.02	0.00	0.03
TE134	0.012	0.000	0.002	0.04	0.00	0.06
I134F	0.005	0.000	0.018	0.01	0.00	0.05
XE137	0.009	0.000	0.011	-0.02	0.00	0.08
XE138	0.021	-0.000	0.008	0.03	0.00	0.03
CS138F	0.002	0.001	0.017	-0.11	0.00	0.06
XE139	0.017	0.000	0.008	0.11	0.00	0.04
CS139	0.009	-0.000	0.012	-0.10	0.00	0.09
XE140	0.015	0.000	0.004	0.05	0.00	0.08
CS140	0.026	-0.000	0.019	0.14	0.00	0.08
CS141	0.015	0.000	0.007	0.23	0.00	0.16
LA140	0.000	-0.000	0.017	-0.00	0.00	0.03

Table 2 cont'd

I BA141	I	0.010	-0.000	0.011	I	-0.15	0.00	0.06 I
I BA142	I	0.022	-0.000	0.008	I	-0.08	0.00	0.07 I
I LA142	I	0.002	-0.000	0.021	I	-0.10	0.00	0.04 I
I BA143	I	0.018	0.000	0.008	I	0.13	0.00	0.30 I
I BA144	I	0.011	0.000	0.003	I	0.16	0.00	0.12 I

Table 3

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U233 THERMIQUE TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07 S
TEMPS DE REFROIDISSEMENT	1.000E 01 S
PUISSANCE RESIDUELLE	8.036E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS	0.796 %
ERREUR DUE AUX PERIODES	0.136 %
ERREUR DUE AUX ENERGIES	1.311 %
ERREUR TOTALE	1.540 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 86	0.007	0.001	0.013	-0.10	0.01	0.12
BR 87	0.010	0.001	0.009	0.05	0.00	0.24
BR 88	0.028	0.004	0.010	0.21	0.00	0.35
KR 87	0.004	0.000	0.010	-0.05	0.00	0.04
BR 89	0.012	0.002	0.001	0.19	0.00	0.04
KR 88	0.022	-0.001	0.014	-0.15	0.00	0.03
RB 88	0.001	0.000	0.026	-0.06	0.00	0.06
KR 89	0.039	0.001	0.022	0.15	0.00	0.06
RB 89	0.004	-0.001	0.025	-0.15	0.00	0.04
KR 90	0.035	0.003	0.012	0.14	0.00	0.03
RB 90M	0.007	0.001	0.011	0.04	0.01	0.02
RB 90F	0.006	0.001	0.026	-0.08	0.01	0.07
KR 91	0.020	0.003	0.003	0.09	0.03	0.10
RB 91	0.025	0.002	0.026	0.29	0.02	0.10
RB 92	0.019	0.008	0.005	0.17	0.01	0.03
RB 93	0.015	0.005	0.004	0.20	0.00	0.01
SR 92	0.009	-0.000	0.012	-0.14	0.00	0.03
Y 92	0.000	0.000	0.013	-0.01	0.00	0.07
SR 93	0.022	0.001	0.021	-0.15	0.00	0.17
Y 93F	0.000	0.000	0.011	-0.02	0.00	0.07

Table 3 cont'd

SR 94		0.029	0.001	0.013		0.10	0.00	0.31
Y 94		0.004	-0.000	0.024		-0.10	0.00	0.17
SR 95		0.030	0.004	0.013		0.35	0.04	0.29
Y 95		0.012	-0.000	0.016		-0.21	0.00	0.08
SR 96		0.011	0.002	0.001		0.17	0.01	0.03
Y 96		0.014	0.001	0.023		0.16	0.02	0.79
ZR 98		0.011	0.003	0.006		0.00	0.03	0.05
TE132		0.012	0.000	0.002		-0.03	0.00	0.01
I132F		0.000	-0.000	0.016		-0.01	0.00	0.03
TE133M		0.015	0.000	0.009		-0.01	0.00	0.03
TE134		0.014	0.000	0.003		0.05	0.00	0.07
I134F		0.006	0.000	0.021		0.01	0.00	0.05
I135		0.012	0.000	0.011		-0.05	0.00	0.02
XE137		0.010	0.001	0.013		-0.01	0.00	0.09
XE138		0.025	-0.000	0.009		0.10	0.00	0.04
CS138F		0.003	0.001	0.020		-0.10	0.00	0.06
XE139		0.018	0.002	0.008		0.12	0.00	0.04
CS139		0.010	-0.000	0.014		-0.09	0.00	0.10
XE140		0.015	0.002	0.003		0.05	0.02	0.06
CS140		0.029	0.001	0.021		0.15	0.01	0.09
CS141		0.016	0.002	0.007		0.23	0.00	0.14
LA140		0.000	-0.000	0.020		-0.00	0.00	0.04
BA141		0.012	-0.000	0.013		-0.16	0.00	0.07
BA142		0.026	-0.000	0.009		0.11	0.00	0.08
LA142		0.003	-0.000	0.025		-0.10	0.00	0.05
BA143		0.017	0.003	0.006		0.12	0.03	0.20
BA144		0.011	0.002	0.002		0.15	0.02	0.09

Table 4

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U233 THERMIQUE TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07 S
TEMPS DE REFROIDISSEMENT	1.000E 02 S
PUISSANCE RESIDUELLE	5.785E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS	0.617 %
ERREUR DUE AUX PERIODES	0.321 %
ERREUR DUE AUX ENERGIES	0.993 %
ERREUR TOTALE	1.213 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 88	0.025	0.002	0.000	0.05	0.00	0.01
KR 87	0.006	0.000	0.014	-0.04	0.00	0.06
BR 89	0.013	0.000	0.000	0.16	0.00	0.00
KR 88	0.030	-0.001	0.020	0.08	0.00	0.04
RB 88	0.001	0.000	0.035	-0.06	0.00	0.09
KR 89	0.047	0.009	0.021	0.20	0.01	0.06
RB 89	0.006	-0.001	0.034	-0.17	0.00	0.05
KR 90	0.028	0.009	0.002	0.09	0.00	0.01
RB 90M	0.008	0.004	0.013	0.10	0.06	0.02
RB 90F	0.005	0.009	0.027	-0.07	0.14	0.07
KR 91	0.014	0.001	0.000	0.04	0.01	0.00
RB 91	0.021	0.013	0.012	0.19	0.13	0.05
RB 92	0.019	0.000	0.000	0.03	0.00	0.00
SR 91	0.003	0.001	0.013	-0.15	0.01	0.05
RB 93	0.013	0.000	0.000	0.06	0.00	0.00
SR 92	0.013	0.000	0.017	0.02	0.00	0.04
Y 92	0.000	0.001	0.018	-0.01	0.00	0.10
SR 93	0.028	0.005	0.025	0.05	0.01	0.20
Y 93F	0.000	0.001	0.015	-0.03	0.01	0.10
SR 94	0.031	0.007	0.007	0.10	0.00	0.17
Y 94	0.005	0.000	0.032	-0.09	0.00	0.23
SR 95	0.025	0.006	0.002	0.20	0.06	0.04
Y 95	0.015	0.001	0.020	-0.09	0.00	0.10
Y 96	0.012	0.010	0.019	0.16	0.17	0.68
Y 97	0.013	0.000	0.000	-0.02	0.00	0.00
NB 97F	0.000	0.000	0.010	-0.01	0.00	0.04
SB131	0.011	0.000	0.007	-0.03	0.00	0.16
TE132	0.017	0.000	0.003	-0.03	0.00	0.01
I132F	0.001	-0.000	0.022	-0.01	0.00	0.04
TE133M	0.020	0.001	0.012	0.05	0.00	0.05
TE134	0.020	0.001	0.004	0.07	0.00	0.10
I134F	0.009	0.001	0.029	0.02	0.00	0.08
I135	0.016	0.001	0.015	0.04	0.00	0.03
XE137	0.010	0.004	0.014	0.04	0.00	0.09
XE138	0.033	0.001	0.012	0.18	0.00	0.05
CS138F	0.004	0.001	0.028	-0.10	0.00	0.09
XE139	0.017	0.005	0.002	0.08	0.01	0.01
CS139	0.013	0.002	0.018	0.01	0.01	0.13
XE140	0.011	0.001	0.000	0.02	0.01	0.00
CS140	0.027	0.011	0.011	0.11	0.11	0.05
CS141	0.014	0.002	0.001	0.11	0.00	0.01

Table 4 cont'd

I LA140	I	0.000	-0.000	0.028	I	-0.00	0.00	0.05 I
I CS142	I	0.010	0.000	0.000	I	0.06	0.00	0.00 I
I BA141	I	0.016	0.001	0.017	I	-0.04	0.00	0.09 I
I LA141	I	0.000	0.000	0.011	I	-0.03	0.00	0.08 I
I BA142	I	0.035	0.001	0.012	I	0.15	0.00	0.10 I
I LA142	I	0.004	-0.000	0.034	I	-0.13	0.00	0.07 I
I BA143	I	0.016	0.001	0.000	I	0.03	0.01	0.00 I
I LA143	I	0.006	0.001	0.012	I	-0.02	0.00	0.09 I
I PR146	I	0.000	0.000	0.011	I	-0.01	0.00	0.05 I

Table 5

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U233 THERMIQUE TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07 S
TEMPS DE REFROIDISSEMENT	1.000E 03 S
PUISSANCE RESIDUELLE	3.610E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS	0.585 %
ERREUR DUE AUX PERIODES	0.146 %
ERREUR DUE AUX ENERGIES	0.645 %
ERREUR TOTALE	0.883 %

I	I				I			
I NUCLEIDE	I	SENSIBILITE DE PUISSANCE RESIDUELLE A			I	ERREUR SUR PUISSANCE RESIDUELLE DUE A		
I	I	REND.	PERIODE	EBGM	I	REND (%)	PER. (%)	EBGM (%)
I BR 88	I	0.038	0.000	0.000	I	0.08	0.00	0.00 I
I KR 87	I	0.008	0.003	0.019	I	0.01	0.00	0.08 I
I KR 88	I	0.047	0.001	0.030	I	0.16	0.00	0.06 I
I RB 88	I	0.001	0.002	0.055	I	-0.09	0.00	0.14 I
I KR 89	I	0.031	0.012	0.001	I	0.08	0.01	0.00 I
I RB 89	I	0.005	0.016	0.031	I	-0.09	0.01	0.04 I
I KR 91	I	0.013	0.000	0.000	I	0.01	0.00	0.00 I
I RB 91	I	0.021	0.000	0.000	I	0.04	0.00	0.00 I
I RB 92	I	0.030	0.000	0.0	I	0.05	0.00	0.0 I
I SR 91	I	0.006	0.002	0.021	I	0.02	0.02	0.09 I
I Y 91F	I	0.000	0.001	0.010	I	-0.00	0.00	0.08 I
I RB 93	I	0.011	0.000	0.0	I	0.05	0.00	0.0 I
I SR 92	I	0.019	0.002	0.025	I	0.03	0.00	0.06 I
I Y 92	I	0.000	0.001	0.028	I	-0.01	0.00	0.16 I
I SR 93	I	0.023	0.016	0.009	I	0.02	0.03	0.07 I
I Y 93F	I	0.001	0.001	0.023	I	-0.02	0.01	0.15 I
I SR 94	I	0.023	0.002	0.000	I	0.04	0.00	0.00 I
I Y 94	I	0.005	0.017	0.029	I	-0.04	0.01	0.21 I
I SR 95	I	0.023	0.000	0.000	I	0.15	0.00	0.00 I
I Y 95	I	0.016	0.012	0.011	I	0.06	0.03	0.06 I
I ZR 95	I	0.001	0.003	0.014	I	-0.02	0.00	0.02 I
I NB 95F	I	0.000	0.000	0.013	I	-0.00	0.00	0.01 I

Table 5 cont'd

Y 97	0.021	0.000	0.0	-0.03	0.00	0.0
ZR 97	0.014	0.000	0.013	0.06	0.00	0.08
NB 97F	0.000	0.000	0.016	-0.01	0.00	0.06
SB131	0.013	0.004	0.007	0.02	0.01	0.16
TE132	0.028	0.001	0.004	0.05	0.00	0.02
I132F	0.001	0.000	0.036	-0.01	0.00	0.06
TE133M	0.029	0.004	0.016	0.11	0.00	0.06
I133F	0.003	-0.000	0.016	-0.02	0.00	0.07
TE134	0.029	0.003	0.005	0.11	0.01	0.12
I134F	0.011	0.005	0.042	0.02	0.01	0.11
I135	0.025	0.002	0.023	0.08	0.00	0.05
XE138	0.040	0.011	0.009	0.25	0.01	0.03
CS138F	0.004	0.007	0.039	-0.08	0.00	0.12
XE139	0.012	0.000	0.000	0.04	0.00	0.00
CS139	0.011	0.011	0.009	0.05	0.04	0.06
BA139	0.001	0.001	0.015	-0.01	0.00	0.10
XE140	0.012	0.000	0.000	0.00	0.00	0.00
CS140	0.030	0.000	0.000	0.02	0.00	0.00
CS141	0.015	0.000	0.000	0.09	0.00	0.00
BA140	0.014	-0.001	0.008	0.00	0.00	0.05
LA140	0.000	-0.000	0.045	-0.00	0.00	0.09
CS142	0.013	0.000	0.0	0.07	0.00	0.0
BA141	0.019	0.009	0.015	0.07	0.00	0.08
LA141	0.001	0.000	0.017	-0.03	0.00	0.13
BA142	0.044	0.009	0.007	0.16	0.01	0.05
LA142	0.005	0.003	0.052	-0.15	0.00	0.11
BA143	0.018	0.000	0.000	0.03	0.00	0.00
CE143	0.000	-0.000	0.011	-0.01	0.00	0.06
CE144	0.001	-0.011	0.001	0.00	0.00	0.01
PR144F	0.000	0.000	0.014	-0.00	0.00	0.10
CE146	0.010	0.003	0.002	-0.01	0.01	0.02
PR146	0.000	0.003	0.015	-0.02	0.01	0.07

Table 6

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U233 THERMIQUE TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07 S
TEMPS DE REFROIDISSEMENT	1.000E 04 S
PUISSANCE RESIDUELLE	1.849E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS	0.517 %
ERREUR DUE AUX PERIODES	0.175 %
ERREUR DUE AUX ENERGIES	0.722 %
ERREUR TOTALE	0.905 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 88	0.040	0.000	0.0	0.11	0.00	0.0
KR 87	0.004	0.014	0.009	0.01	0.01	0.04
KR 88	0.049	0.054	0.030	0.20	0.04	0.06
RB 88	0.000	0.007	0.058	-0.01	0.00	0.15
KR 89	0.014	0.000	0.000	-0.04	0.00	0.00
SR 89	0.000	0.001	0.019	-0.00	0.00	0.15
KR 91	0.023	0.000	0.0	0.01	0.00	0.0
RB 91	0.036	0.000	0.0	0.07	0.00	0.0
RB 92	0.040	0.000	0.0	0.07	0.00	0.0
SR 91	0.010	0.011	0.034	0.03	0.12	0.14
Y 91M	0.000	0.001	0.012	-0.00	0.00	0.18
Y 91F	0.000	0.001	0.020	-0.00	0.00	0.16
RB 93	0.012	0.000	0.0	0.05	0.00	0.0
SR 92	0.026	0.025	0.025	0.04	0.01	0.06
Y 92	0.000	0.009	0.048	-0.02	0.01	0.27
SR 93	0.026	0.000	0.000	-0.02	0.00	0.00
Y 93F	0.001	0.009	0.038	-0.01	0.09	0.25
SR 95	0.031	0.000	0.0	0.19	0.00	0.0
Y 95	0.022	0.000	0.000	0.10	0.00	0.00
ZR 95	0.002	0.005	0.027	0.01	0.00	0.05
NB 95F	0.000	0.001	0.025	-0.00	0.00	0.02
Y 97	0.038	0.000	0.0	-0.05	0.00	0.0
ZR 97	0.026	0.006	0.023	0.10	0.01	0.13
NB 97M	0.000	0.000	0.016	-0.01	0.00	0.00
NB 97F	0.000	0.001	0.029	-0.01	0.00	0.11
ZR 99	0.013	0.000	0.0	0.06	0.00	0.0
MO 99	0.000	0.001	0.014	0.00	0.00	0.08
TE131M	0.013	0.001	0.008	0.08	0.00	0.01
I131	0.000	-0.000	0.010	-0.00	0.00	0.04
SB132M	0.011	0.000	0.000	0.04	0.00	0.00
SB132F	0.011	0.000	0.000	0.04	0.00	0.00
TE132	0.053	0.002	0.008	0.11	0.00	0.04
I132F	0.001	0.002	0.067	-0.03	0.01	0.11
TE133M	0.025	0.012	0.004	0.07	0.01	0.02
I133F	0.005	0.002	0.029	-0.01	0.00	0.13
TE134	0.016	0.013	0.001	0.07	0.03	0.02
I134F	0.003	0.024	0.017	-0.01	0.05	0.05
TE135	0.012	0.000	0.0	0.11	0.00	0.0
I135	0.040	0.014	0.034	0.12	0.02	0.08
XE135F	0.001	0.002	0.016	-0.04	0.00	0.09

Table 6 cont'd

I CS138F	I	0.000	0.014	0.004	I -0.01	0.00	0.01	I
I BA139	I	0.001	0.010	0.008	I -0.00	0.00	0.06	I
I XE140	I	0.023	0.000	0.0	I 0.00	0.00	0.0	I
I CS140	I	0.059	0.000	0.0	I 0.04	0.00	0.0	I
I CS141	I	0.013	0.000	0.0	I 0.07	0.00	0.0	I
I BA140	I	0.028	-0.002	0.015	I 0.01	0.00	0.10	I
I LA140	I	0.000	-0.000	0.089	I -0.00	0.00	0.17	I
I BA141	I	0.016	0.003	0.000	I 0.06	0.00	0.00	I
I LA141	I	0.001	0.010	0.022	I -0.01	0.01	0.17	I
I BA142	I	0.026	0.004	0.000	I 0.08	0.00	0.00	I
I LA142	I	0.003	0.037	0.033	I -0.07	0.03	0.07	I
I BA143	I	0.021	0.000	0.0	I 0.02	0.00	0.0	I
I CE143	I	0.001	0.001	0.021	I -0.00	0.00	0.10	I
I BA144	I	0.014	0.000	0.0	I 0.01	0.00	0.0	I
I LA144	I	0.013	0.000	0.0	I 0.01	0.00	0.0	I
I CE144	I	0.003	-0.022	0.002	I 0.01	0.01	0.02	I
I PR144F	I	0.000	0.000	0.027	I -0.00	0.00	0.19	I

Table 7

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U233 THERMIQUE TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 05 S
PUISSANCE RESIDUELLE 8.408E-01 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.608 %
ERREUR DUE AUX PERIODES 0.478 %
ERREUR DUE AUX ENERGIES 0.940 %
ERREUR TOTALE 1.217 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
KR 89	0.029	0.000	0.0	0.07	0.00	0.0
SR 89	0.000	0.002	0.041	0.00	0.00	0.31
KR 91	0.021	0.000	0.0	0.01	0.00	0.0
RB 91	0.033	0.000	0.0	0.06	0.00	0.0
SR 91	0.009	0.034	0.011	0.03	0.36	0.04
Y 91F	0.000	0.003	0.043	-0.00	0.00	0.35
SR 93	0.011	0.000	0.0	0.02	0.00	0.0
Y 93F	0.000	0.030	0.014	-0.00	0.29	0.09
SR 95	0.067	0.000	0.0	0.43	0.00	0.0
Y 95	0.048	0.000	0.0	0.22	0.00	0.0
ZR 95	0.004	0.012	0.059	0.02	0.00	0.10
NB 95F	0.000	0.002	0.054	-0.00	0.00	0.04
Y 97	0.030	0.000	0.0	-0.04	0.00	0.0
ZR 97	0.021	0.058	0.017	0.08	0.07	0.10
NB 97M	0.000	0.000	0.012	-0.00	0.00	0.00
NB 97F	0.000	0.002	0.022	-0.00	0.00	0.08
ZR 99	0.022	0.000	0.0	0.10	0.00	0.0
MO 99	0.000	0.009	0.024	0.00	0.00	0.14
RU103	0.000	0.000	0.010	0.00	0.00	0.02
SB131	0.010	0.000	0.000	-0.03	0.00	0.00
TE131M	0.018	0.008	0.010	0.11	0.01	0.01
I131	0.000	0.001	0.020	-0.00	0.00	0.07
SB132M	0.019	0.000	0.0	0.08	0.00	0.0
SB132F	0.019	0.000	0.0	0.08	0.00	0.0
TE132	0.093	0.035	0.013	0.22	0.03	0.07
I132F	0.000	0.004	0.115	-0.00	0.03	0.18
TE133M	0.022	0.001	0.000	0.02	0.00	0.00
I133F	0.006	0.025	0.026	-0.00	0.01	0.11
XE133F	0.000	0.001	0.012	-0.00	0.00	0.08
I135	0.013	0.026	0.005	0.04	0.04	0.01
XE135F	0.001	0.011	0.009	-0.01	0.00	0.05
XE140	0.049	0.000	0.0	0.01	0.00	0.0
CS140	0.128	0.000	0.0	0.08	0.00	0.0
BA140	0.059	-0.000	0.032	0.02	0.00	0.21
LA140	0.000	0.003	0.192	-0.00	0.00	0.37
CE141	0.000	0.001	0.017	-0.00	0.00	0.34
BA143	0.032	0.000	0.0	0.04	0.00	0.0
LA143	0.013	0.000	0.000	0.01	0.00	0.00
CE143	0.001	0.016	0.026	-0.00	0.02	0.13
PR143	0.000	-0.000	0.020	-0.00	0.00	0.20
BA144	0.030	0.000	0.0	0.03	0.00	0.0
LA144	0.029	0.000	0.0	0.03	0.00	0.0
CE144	0.006	-0.048	0.005	0.01	0.02	0.05
PR144F	0.000	0.000	0.059	-0.00	0.00	0.41

Table 8

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U233 THERMIQUE TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07	S
TEMPS DE REFROIDISSEMENT	1.000E 06	S
PUISSANCE RESIDUELLE	4.136E-01	(MEV/S)
ERREUR DUE AUX RENDEMENTS	0.953	%
ERREUR DUE AUX PERIODES	0.148	%
ERREUR DUE AUX ENERGIES	1.428	%
ERREUR TOTALE	1.724	%

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 89	0.013	0.000	0.0	-0.08	0.00	0.0
KR 89	0.051	0.000	0.0	0.13	0.00	0.0
RB 89	0.010	0.000	0.0	0.11	0.00	0.0
SR 89	0.000	0.015	0.072	0.00	0.01	0.55
KR 91	0.027	0.000	0.0	0.01	0.00	0.0
RB 91	0.042	0.000	0.0	0.08	0.00	0.0
SR 91	0.011	0.001	0.000	0.04	0.01	0.00
Y 91F	0.000	0.016	0.077	-0.00	0.01	0.63
SR 95	0.128	0.000	0.0	0.81	0.00	0.0
Y 95	0.091	0.000	0.0	0.41	0.00	0.0
ZR 95	0.008	0.040	0.107	0.04	0.00	0.18
NB 95F	0.000	0.005	0.109	-0.00	0.00	0.08
MO 99	0.000	0.014	0.003	0.00	0.00	0.02
NB103	0.013	-0.000	0.0	0.06	0.00	0.0
RU103	0.000	0.005	0.017	0.00	0.00	0.03
I131	0.000	0.015	0.017	-0.00	0.00	0.06
TE132	0.024	0.074	0.003	0.06	0.06	0.01
I132F	0.0	0.001	0.023	0.0	0.01	0.04
XE140	0.059	0.000	0.0	0.01	0.00	0.0
CS140	0.154	0.000	0.0	0.10	0.00	0.0
CS141	0.011	0.000	0.0	0.06	0.00	0.0
BA140	0.072	0.138	0.035	0.02	0.12	0.23
LA140	0.000	0.033	0.237	-0.00	0.05	0.45
BA141	0.014	0.000	0.0	0.05	0.00	0.0
CE141	0.000	0.008	0.027	0.00	0.00	0.54
BA143	0.018	0.000	0.0	0.02	0.00	0.0
PR143	0.000	0.012	0.025	-0.00	0.01	0.25
BA144	0.060	0.000	0.0	0.06	0.00	0.0
LA144	0.058	0.000	0.0	0.06	0.00	0.0
CE144	0.013	-0.092	0.009	0.03	0.03	0.10
PR144F	0.0	0.000	0.116	0.0	0.00	0.81

Table 9

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U233 THERMIQUE TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07 S
TEMPS DE REFROIDISSEMENT	1.000E 07 S
PUISSANCE RESIDUELLE	1.136E-01 (MEV/S)
ERREUR DUE AUX RENDEMENTS	1.373 %
ERREUR DUE AUX PERIODES	0.126 %
ERREUR DUE AUX ENERGIES	2.445 %
ERREUR TOTALE	2.807 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 89	0.011	0.000	0.0	-0.07	0.00	0.0
KR 89	0.045	0.000	0.0	0.11	0.00	0.0
SR 89	0.000	0.104	0.059	0.00	0.05	0.45
KR 90	0.025	-0.000	0.0	0.04	0.00	0.0
SR 90	0.001	0.013	0.006	0.01	0.04	0.06
Y 90F	0.000	-0.000	0.032	-0.00	0.00	0.23
KR 91	0.028	0.000	0.0	0.01	0.00	0.0
RB 91	0.044	0.000	0.0	0.08	0.00	0.0
SR 91	0.012	0.001	0.0	0.04	0.01	0.0
Y 91F	0.000	0.125	0.077	-0.00	0.06	0.62
SR 95	0.188	0.000	0.0	1.19	0.00	0.0
Y 95	0.134	0.000	0.0	0.61	0.00	0.0
ZR 95	0.012	0.364	0.118	0.06	0.03	0.20
NB 95F	0.000	0.099	0.195	-0.00	0.01	0.14
RU103	0.000	0.022	0.009	0.00	0.01	0.01
MO106	0.018	-0.000	0.0	0.02	0.00	0.0
RU106	0.000	-0.022	0.000	0.00	0.01	0.00
RH106F	0.0	0.000	0.023	0.0	0.00	0.14
XE137	0.019	-0.000	0.0	0.02	0.00	0.0
BA137M	0.0	-0.000	0.020	0.0	0.00	0.00
BA140	0.002	0.039	0.000	0.00	0.03	0.00
CE141	0.000	0.027	0.010	0.00	0.01	0.19
BA144	0.165	0.000	0.0	0.18	0.00	0.0
LA144	0.160	0.000	0.0	0.17	0.00	0.0
CE144	0.035	-0.200	0.025	0.07	0.07	0.28
PR144F	0.0	0.000	0.324	0.0	0.00	2.27

Table 10

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U233 THERMIQUE TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07 S
TEMPS DE REFROIDISSEMENT	1.000E 08 S
PUISSANCE RESIDUELLE	1.169E-02 (MEV/S)
ERREUR DUE AUX RENDEMENTS	0.779 %
ERREUR DUE AUX PERIODES	0.924 %
ERREUR DUE AUX ENERGIES	2.742 %
ERREUR TOTALE	2.996 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
KR 85F	0.001	-0.008	0.012	0.01	0.00	0.11
BR 90	0.032	0.000	0.0	-0.19	0.00	0.0
KR 90	0.208	0.000	0.0	0.40	0.00	0.0
RB 90M	0.056	0.000	0.0	0.43	0.00	0.0
RB 90F	0.056	0.000	0.0	0.43	0.00	0.0
SR 90	0.009	-0.317	0.054	0.14	0.69	0.57
Y 90F	0.0	0.000	0.290	0.0	0.00	2.07
MO106	0.025	-0.000	0.0	0.03	0.00	0.0
RU106	0.001	0.051	0.000	0.00	0.03	0.00
RH106F	0.0	0.000	0.029	0.0	0.00	0.17
I137	0.063	0.000	0.0	-0.13	0.00	0.0
XE137	0.154	0.000	0.0	0.18	0.00	0.0
CS137	0.025	-0.217	0.052	0.12	0.07	0.55
BA137M	0.0	0.000	0.181	0.0	0.00	0.00
BA144	0.144	0.000	0.0	0.15	0.00	0.0
LA144	0.139	0.000	0.0	0.15	0.00	0.0
CE144	0.030	0.656	0.018	0.07	0.23	0.20
PR144F	0.0	0.000	0.224	0.0	0.00	1.57
CE147	0.013	0.000	0.0	0.04	0.00	0.0
PM147	0.000	0.003	0.019	0.00	0.00	0.24

Table 11

ERREUR SUR LA PUISSANCE RESIDUELLE DUE AUX PRODUITS DE FISSION
***** USTH 11/02/76 *****
TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION 7.776E 07 S

TEMPS REF. (S)	PUISSANCE RESIDUELLE (MEV/S)	ERREUR DUE AU REND. (%)	ERREUR DUE A LA PER. (%)	ERREUR DUE AUX ENERGIES (%)	ERREUR TOTALE
1.0E 00	1.1257E 01	0.605	0.309	1.678	1.811
2.0E 00	1.0753E 01	0.571	0.267	1.664	1.780
5.0E 00	9.8640E 00	0.535	0.203	1.619	1.717
1.0E 01	9.0467E 00	0.506	0.230	1.549	1.646
2.0E 01	8.1609E 00	0.465	0.298	1.453	1.555
5.0E 01	6.9469E 00	0.385	0.331	1.315	1.409
1.0E 02	6.0344E 00	0.315	0.363	1.180	1.274
2.0E 02	5.1975E 00	0.265	0.419	0.998	1.114
5.0E 02	4.2545E 00	0.256	0.321	0.819	0.916
1.0E 03	3.6157E 00	0.265	0.146	0.736	0.796
2.0E 03	2.9931E 00	0.272	0.085	0.669	0.727
5.0E 03	2.2511E 00	0.275	0.115	0.670	0.733
1.0E 04	1.7903E 00	0.285	0.177	0.720	0.794
2.0E 04	1.4196E 00	0.302	0.289	0.781	0.885
5.0E 04	1.0441E 00	0.313	0.478	0.853	1.026
1.0E 05	8.3396E-01	0.317	0.442	0.942	1.088
2.0E 05	6.8015E-01	0.332	0.220	1.070	1.141
5.0E 05	5.2291E-01	0.357	0.106	1.262	1.316
1.0E 06	4.1460E-01	0.370	0.143	1.424	1.478
2.0E 06	3.0986E-01	0.378	0.190	1.640	1.694
5.0E 06	1.9199E-01	0.388	0.155	2.087	2.129
1.0E 07	1.2057E-01	0.368	0.109	2.634	2.661
2.0E 07	6.5442E-02	0.339	0.103	3.500	3.518
5.0E 07	2.4786E-02	0.375	0.479	3.996	4.042
1.0E 08	1.1632E-02	0.448	0.840	2.724	2.886

Table 12

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
USTH 11/02/76 TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07	S
TEMPS DE REFROIDISSEMENT	1.000E 00	S
PUISSANCE RESIDUELLE	1.126E 01	(MEV/S)
ERREUR DUE AUX RENDEMENTS	0.605	%
ERREUR DUE AUX PERIODES	0.309	%
ERREUR DUE AUX ENERGIES	1.678	%
ERREUR TOTALE	1.811	%

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 88	0.021	0.000	0.011	0.05	0.00	0.38
BR 89	0.019	0.001	0.006	0.11	0.00	0.22
RB 88	0.000	0.000	0.012	-0.01	0.00	0.03
BR 90	0.013	0.002	0.004	0.21	0.02	0.14
KR 89	0.018	0.000	0.014	-0.06	0.00	0.04
RB 89	0.001	-0.001	0.014	-0.01	0.00	0.02
KR 90	0.023	0.000	0.011	-0.09	0.00	0.03
RB 90F	0.003	0.000	0.018	-0.02	0.00	0.05
KR 91	0.025	0.000	0.007	0.02	0.00	0.23
KR 92	0.013	0.001	0.002	0.04	0.00	0.02
RB 91	0.013	-0.000	0.020	-0.02	0.00	0.08
RB 92	0.022	0.002	0.018	0.04	0.00	0.10
RB 93	0.023	0.001	0.014	0.05	0.00	0.04
RB 94	0.010	0.001	0.004	0.02	0.00	0.11
SR 93	0.010	0.000	0.015	-0.03	0.00	0.12
SR 94	0.019	-0.000	0.010	0.07	0.00	0.25
Y 94	0.001	-0.000	0.016	-0.06	0.00	0.12
SR 95	0.032	0.001	0.019	0.29	0.01	0.41
Y 95	0.003	-0.000	0.012	-0.20	0.00	0.06
SR 96	0.019	0.001	0.007	0.22	0.01	0.23
Y 96	0.008	0.000	0.020	-0.16	0.00	0.71
Y 97	0.012	0.003	0.007	-0.08	0.02	0.06
ZR 99	0.015	0.002	0.009	-0.10	0.02	0.09
ZR100	0.022	0.000	0.006	0.13	0.00	0.22
NB100F	0.001	0.001	0.016	-0.07	0.02	0.45
ZR101	0.020	0.002	0.005	0.16	0.03	0.14
ZR102	0.010	0.001	0.001	0.13	0.04	0.03
NB102	0.011	0.002	0.010	-0.04	0.03	0.36
NB103	0.011	0.002	0.002	0.08	0.05	0.08
I132F	0.000	0.000	0.010	-0.00	0.00	0.02
SB133	0.011	0.000	0.006	0.06	0.00	0.11
TE133M	0.010	0.000	0.006	-0.04	0.00	0.02
TE134	0.022	0.001	0.004	-0.04	0.00	0.12
I134F	0.001	0.000	0.019	-0.00	0.00	0.05
TE135	0.018	0.000	0.011	0.06	0.01	0.40
TE136	0.010	0.000	0.004	0.26	0.00	0.15
I137	0.012	0.000	0.009	0.04	0.00	0.30
I138	0.013	0.000	0.006	0.07	0.00	0.20
XE138	0.022	-0.000	0.009	-0.01	0.00	0.04

Table 12 cont'd

I CS138F	I	0.001	0.001	0.019	I -0.01	0.00	0.06	I
I XE139	I	0.022	0.000	0.011	I -0.05	0.00	0.06	I
I CS139	I	0.003	-0.000	0.011	I -0.03	0.00	0.08	I
I XE140	I	0.032	0.001	0.010	I 0.11	0.01	0.19	I
I CS140	I	0.014	-0.000	0.020	I -0.02	0.00	0.08	I
I CS141	I	0.015	0.000	0.009	I 0.02	0.00	0.20	I
I LA140	I	0.000	-0.000	0.014	I -0.00	0.00	0.03	I
I CS142	I	0.014	0.002	0.005	I -0.01	0.01	0.12	I
I BA142	I	0.012	-0.000	0.007	I -0.07	0.00	0.05	I
I LA142	I	0.000	-0.000	0.016	I -0.01	0.00	0.03	I
I BA143	I	0.015	0.000	0.009	I 0.05	0.00	0.31	I
I BA144	I	0.017	0.000	0.006	I 0.05	0.00	0.20	I
I BA145	I	0.010	0.000	0.004	I 0.13	0.00	0.13	I
I LA146	I	0.013	0.000	0.006	I 0.12	0.02	0.21	I

Table 13

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U5TH 11/02/76 TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07 S
TEMPS DE REFROIDISSEMENT	1.000E 01 S
PUISSANCE RESIDUELLE	9.047E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS	0.506 %
ERREUR DUE AUX PERIODES	0.230 %
ERREUR DUE AUX ENERGIES	1.549 %
ERREUR TOTALE	1.646 %

I NUCLEIDE	I	SENSIBILITE DE PUISSANCE RESIDUELLE A			I	ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	I	REND.	PERIODE	EBGM	I	REND (%)	PER. (%)	EBGM (%)
I BR 88	I	0.023	0.004	0.009	I	0.05	0.00	0.31 I
I BR 89	I	0.017	0.003	0.002	I	0.09	0.01	0.06 I
I RB 88	I	0.000	0.000	0.015	I	-0.01	0.00	0.04 I
I BR 90	I	0.011	0.001	0.000	I	0.10	0.01	0.00 I
I KR 89	I	0.023	0.001	0.017	I	-0.02	0.00	0.04 I
I RB 89	I	0.001	-0.001	0.017	I	-0.01	0.00	0.03 I
I KR 90	I	0.027	0.003	0.011	I	0.07	0.00	0.03 I
I RB 90F	I	0.004	0.001	0.022	I	-0.02	0.01	0.06 I
I KR 91	I	0.026	0.004	0.004	I	0.03	0.04	0.14 I
I RB 91	I	0.015	0.001	0.024	I	0.02	0.01	0.09 I
I RB 92	I	0.016	0.008	0.006	I	0.02	0.01	0.03 I
I RB 93	I	0.019	0.007	0.006	I	0.03	0.00	0.01 I

Table 13 cont'd

Y 92	0.000	0.000	0.010	-0.00	0.00	0.06
SR 93	0.012	0.000	0.018	-0.01	0.00	0.14
SR 94	0.023	0.001	0.012	0.09	0.00	0.29
Y 94	0.001	-0.000	0.020	-0.07	0.00	0.15
SR 95	0.036	0.005	0.018	0.30	0.06	0.40
Y 95	0.004	-0.000	0.015	-0.22	0.00	0.07
SR 96	0.017	0.003	0.002	0.13	0.02	0.06
Y 96	0.010	0.001	0.025	-0.09	0.02	0.86
NB 98F	0.000	0.001	0.010	-0.00	0.02	0.08
ZR100	0.013	0.009	0.003	0.11	0.05	0.11
ZR101	0.015	0.002	0.000	0.07	0.04	0.01
TC102F	0.000	0.000	0.010	-0.00	0.00	0.28
NB103	0.010	0.000	0.000	0.03	0.01	0.00
I132F	0.000	0.000	0.012	-0.00	0.00	0.02
SB133	0.013	0.000	0.007	0.08	0.01	0.13
TE133M	0.013	0.000	0.008	-0.04	0.00	0.03
TE134	0.027	0.001	0.006	0.03	0.00	0.15
I134F	0.001	0.001	0.024	-0.00	0.00	0.06
TE135	0.018	0.004	0.010	0.06	0.04	0.34
TE136	0.011	0.002	0.004	0.29	0.02	0.13
I135	0.008	0.000	0.012	-0.03	0.00	0.03
I136F	0.005	0.000	0.011	0.06	0.00	0.04
I137	0.013	0.003	0.008	0.07	0.00	0.29
I138	0.012	0.003	0.003	0.05	0.00	0.10
XE137	0.006	0.001	0.012	-0.06	0.00	0.08
XE138	0.027	-0.000	0.012	0.03	0.00	0.05
CS138F	0.001	0.001	0.024	-0.01	0.00	0.08
XE139	0.025	0.002	0.012	0.03	0.00	0.06
CS139	0.004	-0.000	0.014	-0.03	0.00	0.10
XE140	0.035	0.004	0.007	0.12	0.04	0.14
CS140	0.016	0.001	0.023	-0.02	0.01	0.10
CS141	0.017	0.003	0.009	0.02	0.00	0.19
LA140	0.000	-0.000	0.018	-0.00	0.00	0.03
CS142	0.012	0.001	0.000	0.00	0.01	0.00
BA141	0.005	-0.000	0.010	-0.04	0.00	0.05
BA142	0.015	-0.000	0.008	-0.01	0.00	0.07
LA142	0.000	-0.000	0.020	-0.01	0.00	0.04
BA143	0.015	0.004	0.007	0.05	0.04	0.23
BA144	0.018	0.003	0.004	0.06	0.03	0.15
LA144	0.004	0.001	0.011	-0.05	0.00	0.26
LA146	0.013	0.002	0.005	0.11	0.16	0.17

Table 14

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
USTH 11/02/76 TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07	S
TEMPS DE REFROIDISSEMENT	1.000E 02	S
PUISSANCE RESIDUELLE	6.034E 00	(MEV/S)
ERREUR DUE AUX RENDEMENTS	0.315	%
ERREUR DUE AUX PERIODES	0.363	%
ERREUR DUE AUX ENERGIES	1.180	%
ERREUR TOTALE	1.274	%

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 88	0.022	0.002	0.000	0.02	0.00	0.01
BR 89	0.020	0.000	0.000	0.09	0.00	0.00
KR 88	0.011	-0.001	0.013	0.02	0.00	0.02
RB 88	0.000	0.000	0.023	-0.01	0.00	0.06
KR 89	0.029	0.008	0.018	0.05	0.00	0.05
RB 89	0.001	-0.001	0.026	-0.01	0.00	0.04
KR 90	0.023	0.009	0.002	-0.04	0.00	0.01
RB 90F	0.003	0.008	0.025	-0.02	0.13	0.07
KR 91	0.019	0.002	0.000	0.03	0.02	0.00
RB 91	0.013	0.013	0.012	0.04	0.13	0.05
RB 92	0.018	0.000	0.000	0.01	0.00	0.00
SR 91	0.001	0.001	0.012	0.01	0.01	0.05
RB 93	0.018	0.000	0.000	0.00	0.00	0.00
SR 92	0.006	0.000	0.015	0.02	0.00	0.03
Y 92	0.000	0.000	0.015	-0.00	0.00	0.09
RB 94	0.010	0.000	0.000	0.00	0.00	0.00
SR 93	0.016	0.004	0.023	0.03	0.01	0.18
Y 93F	0.000	0.000	0.013	-0.00	0.00	0.09
SR 94	0.026	0.007	0.007	0.08	0.00	0.18
Y 94	0.002	0.000	0.029	-0.07	0.00	0.21
SR 95	0.032	0.008	0.003	0.14	0.09	0.06
Y 95	0.006	0.001	0.021	-0.13	0.00	0.10
SR 96	0.014	0.000	0.000	0.07	0.00	0.00
Y 96	0.009	0.012	0.023	-0.01	0.20	0.80
Y 97	0.014	0.000	0.000	0.03	0.00	0.00
NB 97F	0.000	0.000	0.011	-0.00	0.00	0.04
ZR101	0.014	0.000	0.000	0.02	0.00	0.00
MO101	0.001	0.001	0.014	-0.00	0.00	0.05
TC102F	0.000	0.000	0.014	-0.00	0.00	0.38
SB132M	0.012	0.002	0.006	0.04	0.03	0.13
SB132F	0.011	0.003	0.007	0.03	0.05	0.16
I132F	0.000	0.000	0.019	-0.00	0.00	0.03
SB133	0.017	0.004	0.007	0.09	0.06	0.12
TE133M	0.019	0.001	0.012	-0.04	0.00	0.05
I133F	0.000	-0.000	0.011	-0.00	0.00	0.05
TE134	0.041	0.001	0.008	0.05	0.00	0.22
I134F	0.002	0.001	0.036	-0.01	0.00	0.09
TE135	0.013	0.002	0.000	0.03	0.02	0.01

Table 14 cont'd

I I135	I	0.013	0.001	0.018	I 0.01	0.00	0.04	I
I I138	I	0.013	0.000	0.000	I 0.05	0.00	0.00	I
I XE137	I	0.007	0.004	0.014	I -0.03	0.00	0.09	I
I XE138	I	0.039	0.001	0.016	I 0.06	0.00	0.06	I
I CS138F	I	0.002	0.001	0.035	I -0.01	0.00	0.11	I
I XE139	I	0.025	0.007	0.003	I 0.02	0.01	0.02	I
I CS139	I	0.006	0.001	0.019	I -0.02	0.00	0.14	I
I XE140	I	0.028	0.003	0.000	I 0.08	0.03	0.00	I
I CS140	I	0.016	0.012	0.014	I 0.01	0.12	0.06	I
I CS141	I	0.016	0.004	0.001	I 0.01	0.00	0.02	I
I LA140	I	0.000	-0.000	0.026	I -0.00	0.00	0.05	I
I CS142	I	0.017	0.000	0.000	I 0.01	0.00	0.00	I
I BA141	I	0.007	0.000	0.015	I 0.02	0.00	0.08	I
I BA142	I	0.022	0.001	0.011	I 0.02	0.00	0.09	I
I LA142	I	0.001	-0.000	0.029	I -0.01	0.00	0.06	I
I BA143	I	0.015	0.001	0.000	I 0.01	0.01	0.00	I
I LA143	I	0.002	0.000	0.012	I -0.01	0.00	0.09	I
I BA144	I	0.012	0.002	0.000	I 0.02	0.02	0.00	I
I LA146	I	0.012	0.001	0.000	I 0.02	0.05	0.00	I
I PR146	I	0.000	0.000	0.012	I -0.00	0.00	0.05	I

Table 15

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U5TH 11/02/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 03 S
PUISSANCE RESIDUELLE 3.616E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.265 %
ERREUR DUE AUX PERIODES 0.146 %
ERREUR DUE AUX ENERGIES 0.736 %
ERREUR TOTALE 0.796 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 88	0.035	0.000	0.000	0.04	0.00	0.00
KR 87	0.002	0.002	0.013	0.00	0.00	0.05
BR 89	0.015	0.000	0.0	0.08	0.00	0.0
KR 88	0.018	0.001	0.021	0.03	0.00	0.04
RB 88	0.000	0.001	0.037	-0.02	0.00	0.09
KR 89	0.020	0.010	0.001	-0.01	0.01	0.00
RB 89	0.001	0.012	0.024	-0.01	0.01	0.04
KR 91	0.019	0.000	0.000	0.03	0.00	0.00
KR 92	0.013	0.000	0.0	0.01	0.00	0.0
RB 91	0.014	0.000	0.000	0.05	0.00	0.00
RB 92	0.028	0.000	0.0	0.01	0.00	0.0
SR 91	0.001	0.001	0.019	0.06	0.01	0.08
RB 93	0.015	0.000	0.0	0.01	0.00	0.0
SR 92	0.009	0.002	0.023	0.03	0.00	0.05
Y 92	0.000	0.001	0.026	-0.00	0.00	0.15
SR 93	0.014	0.015	0.008	0.02	0.03	0.07
Y 93F	0.000	0.001	0.021	-0.00	0.01	0.14
SR 94	0.019	0.002	0.000	0.04	0.00	0.00
Y 94	0.002	0.015	0.028	-0.03	0.01	0.20
SR 95	0.031	0.001	0.000	0.06	0.01	0.00
Y 95	0.006	0.013	0.012	-0.04	0.04	0.06
ZR 95	0.000	0.002	0.014	-0.01	0.00	0.02
NB 95F	0.000	0.000	0.013	-0.00	0.00	0.01
SR 97	0.013	0.000	0.0	0.03	0.00	0.0
Y 97	0.024	0.000	0.0	0.05	0.00	0.0
ZR 97	0.008	0.000	0.014	0.03	0.00	0.08
NB 97F	0.000	0.000	0.018	-0.00	0.00	0.07
ZR101	0.013	0.000	0.0	0.02	0.00	0.0
MO101	0.001	0.010	0.011	-0.00	0.00	0.04
MO102	0.001	0.010	0.001	-0.00	0.02	0.06
SB131	0.013	0.005	0.009	0.03	0.01	0.20
SB132M	0.012	0.002	0.001	0.03	0.04	0.02
SB132F	0.011	0.001	0.000	0.02	0.01	0.00
TE132	0.010	0.000	0.004	0.02	0.00	0.02
I132F	0.000	0.000	0.031	-0.00	0.00	0.05
SB133	0.012	0.002	0.000	-0.05	0.03	0.00
TE133M	0.028	0.004	0.016	0.10	0.00	0.06
I133F	0.001	-0.000	0.018	-0.00	0.00	0.08

Table 15 cont'd

TE134		0.063	0.006	0.010		0.08	0.01	0.28
I134F		0.003	0.004	0.057		-0.01	0.01	0.15
TE135		0.021	0.000	0.000		0.05	0.00	0.00
I135		0.021	0.002	0.030		0.03	0.00	0.07
I138		0.016	0.000	0.0		0.08	0.00	0.0
XE138		0.049	0.016	0.012		0.09	0.01	0.05
CS138F		0.002	0.009	0.052		0.01	0.00	0.17
XE139		0.019	0.001	0.000		-0.02	0.00	0.00
CS139		0.005	0.012	0.010		-0.01	0.04	0.07
BA139		0.000	0.001	0.015		-0.01	0.00	0.11
XE140		0.031	0.000	0.000		0.04	0.00	0.00
CS140		0.019	0.000	0.000		0.01	0.00	0.00
CS141		0.018	0.000	0.000		0.01	0.00	0.00
LA140		0.000	-0.000	0.044		-0.00	0.00	0.08
CS142		0.023	0.000	0.0		0.01	0.00	0.0
BA141		0.008	0.009	0.013		0.03	0.00	0.07
LA141		0.000	0.000	0.015		-0.00	0.00	0.12
BA142		0.030	0.009	0.007		0.03	0.01	0.05
LA142		0.001	0.002	0.047		-0.01	0.00	0.10
BA143		0.017	0.000	0.000		0.01	0.00	0.00
CE143		0.000	-0.000	0.011		-0.00	0.00	0.06
BA144		0.013	0.000	0.000		0.01	0.00	0.00
CE144		0.000	-0.012	0.001		0.00	0.00	0.01
PR144F		0.000	0.000	0.016		-0.00	0.00	0.11
LA146		0.016	0.000	0.000		0.02	0.01	0.00
PR146		0.000	0.003	0.018		-0.00	0.01	0.08

Table 16

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U5TH 11/02/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 04 S
PUISSANCE RESIDUELLE 1.790E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.285 %
ERREUR DUE AUX PERIODES 0.177 %
ERREUR DUE AUX ENERGIES 0.720 %
ERREUR TOTALE 0.794 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 88	0.038	0.000	0.0	0.05	0.00	0.0
KR 88	0.020	0.039	0.022	0.04	0.03	0.04
RB 88	0.000	0.005	0.041	-0.00	0.00	0.10
SR 89	0.000	0.000	0.015	-0.00	0.00	0.12
KR 91	0.034	0.000	0.0	0.06	0.00	0.0
KR 92	0.018	0.000	0.0	0.01	0.00	0.0
RB 91	0.025	0.000	0.0	0.09	0.00	0.0
RB 92	0.040	0.000	0.0	0.02	0.00	0.0
SR 91	0.002	0.010	0.032	0.10	0.11	0.13
Y 91M	0.000	0.001	0.011	0.00	0.00	0.17
Y 91F	0.000	0.001	0.019	0.00	0.00	0.15
RB 93	0.018	0.000	0.0	0.01	0.00	0.0
SR 92	0.013	0.024	0.023	0.05	0.01	0.05
Y 92	0.000	0.008	0.045	-0.00	0.00	0.26
SR 93	0.016	0.000	0.000	0.03	0.00	0.00
Y 93F	0.000	0.009	0.036	-0.00	0.09	0.24
SR 95	0.043	0.000	0.0	0.04	0.00	0.0
ZR 95	0.000	0.004	0.029	0.00	0.00	0.05
NB 95F	0.000	0.001	0.026	-0.00	0.00	0.02
SR 97	0.024	0.000	0.0	0.05	0.00	0.0
Y 97	0.044	0.000	0.0	0.09	0.00	0.0
ZR 97	0.014	0.007	0.026	0.06	0.01	0.15
NB 97M	0.000	0.000	0.018	-0.00	0.00	0.00
NB 97F	0.000	0.001	0.033	-0.00	0.00	0.12
ZR 99	0.015	0.000	0.0	0.02	0.00	0.0
MO 99	0.000	0.001	0.018	0.00	0.00	0.11
SN132	0.012	0.000	0.0	0.03	0.00	0.0
SB132M	0.021	0.000	0.000	0.04	0.00	0.00
SB132F	0.021	0.000	0.000	0.04	0.00	0.00
TE132	0.020	0.002	0.007	0.05	0.00	0.04
I132F	0.000	0.001	0.062	-0.00	0.00	0.10
SB133	0.014	0.000	0.000	-0.05	0.00	0.00
TE133M	0.026	0.012	0.005	0.07	0.01	0.02
I133F	0.001	0.002	0.034	-0.00	0.00	0.15
TE134	0.035	0.031	0.002	0.05	0.06	0.04
I134F	0.001	0.035	0.031	-0.01	0.07	0.08
TE135	0.034	0.000	0.0	0.08	0.00	0.0
I135	0.034	0.018	0.046	0.04	0.03	0.10

Table 16 cont'd

XE135F		0.000	0.001	0.018		-0.00	0.00	0.10
CS138F		0.000	0.019	0.005		0.00	0.01	0.02
BA139		0.000	0.011	0.009		-0.00	0.00	0.06
XE140		0.062	0.000	0.0		0.11	0.00	0.0
CS140		0.039	0.000	0.0		0.03	0.00	0.0
CS141		0.015	0.000	0.0		0.01	0.00	0.0
BA140		0.007	-0.003	0.015		0.07	0.00	0.10
LA140		0.000	-0.000	0.089		-0.00	0.00	0.17
CS142		0.014	0.000	0.0		0.01	0.00	0.0
LA141		0.000	0.009	0.020		0.00	0.01	0.16
BA142		0.018	0.004	0.000		0.02	0.00	0.00
LA142		0.001	0.034	0.031		-0.01	0.03	0.06
BA143		0.021	0.000	0.0		0.01	0.00	0.0
CE143		0.000	0.001	0.022		-0.00	0.00	0.11
BA144		0.027	0.000	0.0		0.02	0.00	0.0
CE144		0.000	-0.025	0.003		0.00	0.01	0.03
PR144F		0.000	0.000	0.032		-0.00	0.00	0.23
PR145		0.000	0.004	0.010		-0.00	0.01	0.09

Table 17

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
USTH 11/02/76 TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 05 S
PUISSANCE RESIDUELLE 8.340E-01 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.317 %
ERREUR DUE AUX PERIODES 0.442 %
ERREUR DUE AUX ENERGIES 0.942 %
ERREUR TOTALE 1.088 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 89	0.012	0.000	0.0	0.04	0.00	0.0
KR 89	0.019	0.000	0.0	0.04	0.00	0.0
SR 89	0.000	0.001	0.032	0.00	0.00	0.25
KR 91	0.031	0.000	0.0	0.05	0.00	0.0
RB 91	0.022	0.000	0.0	0.08	0.00	0.0
SR 91	0.002	0.031	0.010	0.09	0.33	0.04
Y 91F	0.000	0.002	0.040	0.00	0.00	0.32
Y 93F	0.000	0.028	0.013	-0.00	0.27	0.09
RB 95	0.016	-0.000	0.0	-0.03	0.00	0.0
SR 95	0.091	0.000	0.0	0.09	0.00	0.0
Y 95	0.018	0.000	0.0	0.03	0.00	0.0
ZR 95	0.001	0.010	0.062	0.00	0.00	0.10
NB 95F	0.000	0.001	0.057	-0.00	0.00	0.04
SR 97	0.019	0.000	0.0	0.04	0.00	0.0
Y 97	0.034	0.000	0.0	0.07	0.00	0.0
ZR 97	0.011	0.065	0.019	0.05	0.08	0.11
NB 97M	0.000	0.000	0.013	-0.00	0.00	0.00
NB 97F	0.000	0.002	0.025	-0.00	0.00	0.09
Y 99	0.012	0.000	0.0	0.03	0.00	0.0
ZR 99	0.025	0.000	0.0	0.03	0.00	0.0
MO 99	0.000	0.012	0.029	0.00	0.00	0.17
NB103	0.015	0.000	0.0	0.04	0.00	0.0
RU103	0.000	0.001	0.019	0.00	0.00	0.03
I131	0.000	0.001	0.016	-0.00	0.00	0.06
SN132	0.020	0.000	0.0	0.04	0.00	0.0
SB132M	0.036	0.000	0.0	0.06	0.00	0.0
SB132F	0.036	0.000	0.0	0.06	0.00	0.0
TE132	0.035	0.031	0.013	0.08	0.03	0.06
I132F	0.000	0.003	0.107	-0.00	0.03	0.17
SB133	0.015	0.000	0.0	0.01	0.00	0.0
TE133M	0.022	0.001	0.000	0.02	0.00	0.00
I133F	0.001	0.029	0.030	-0.00	0.01	0.13
XE133F	0.000	0.000	0.013	-0.00	0.00	0.09
TE135	0.011	0.000	0.0	0.03	0.00	0.0
I135	0.011	0.033	0.007	0.01	0.05	0.01
XE135F	0.000	0.012	0.011	-0.00	0.00	0.06
XE140	0.129	0.000	0.0	0.24	0.00	0.0
CS140	0.081	0.000	0.0	0.07	0.00	0.0

Table 17 cont'd

BA140		0.015	-0.000	0.031		0.16	0.00	0.20
LA140		0.000	0.002	0.188		-0.00	0.00	0.36
CE141		0.000	0.001	0.015		-0.00	0.00	0.30
CS143		0.013	0.000	0.0		-0.01	0.00	0.0
BA143		0.032	0.000	0.0		0.02	0.00	0.0
CE143		0.000	0.017	0.027		0.00	0.03	0.13
PR143		0.000	-0.000	0.021		-0.00	0.00	0.20
BA144		0.057	0.000	0.0		0.03	0.00	0.0
LA144		0.017	0.000	0.0		0.02	0.00	0.0
CE144		0.001	-0.052	0.005		0.00	0.02	0.06
PR144F		0.000	0.000	0.069		-0.00	0.00	0.49

Table 18

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U5TH 11/02/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 06 S
PUISSANCE RESIDUELLE 4.146E-01 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.370 %
ERREUR DUE AUX PERIODES 0.143 %
ERREUR DUE AUX ENERGIES 1.424 %
ERREUR TOTALE 1.478 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 89	0.020	0.000	0.0	0.08	0.00	0.0
KR 89	0.033	0.000	0.0	0.07	0.00	0.0
SR 89	0.000	0.011	0.056	0.00	0.01	0.43
KR 91	0.038	0.000	0.0	0.07	0.00	0.0
RB 91	0.028	0.000	0.0	0.10	0.00	0.0
Y 91F	0.000	0.013	0.070	0.00	0.01	0.56
RB 95	0.031	-0.000	0.0	-0.05	0.00	0.0
SR 95	0.172	0.000	0.0	0.17	0.00	0.0
Y 95	0.035	0.000	0.0	0.05	0.00	0.0
ZR 95	0.001	0.035	0.110	0.00	0.00	0.18
NB 95F	0.000	0.004	0.113	-0.00	0.00	0.08
MO 99	0.000	0.017	0.004	0.00	0.00	0.02
NB103	0.024	0.000	0.0	0.06	0.00	0.0
MO103	0.011	0.000	0.0	0.03	0.00	0.0
RU103	0.000	0.008	0.032	0.00	0.00	0.05
MO106	0.011	0.0	0.0	0.02	0.0	0.0
RU106	0.000	-0.012	0.000	0.00	0.01	0.00
RH106F	0.0	0.000	0.012	0.0	0.00	0.07
I131	0.000	0.012	0.013	0.00	0.00	0.05
TE132	0.009	0.069	0.002	0.02	0.06	0.01
I132F	0.0	0.001	0.021	0.0	0.01	0.03
XE140	0.155	0.000	0.0	0.29	0.00	0.0
CS140	0.097	0.000	0.0	0.09	0.00	0.0
CS141	0.014	0.000	0.0	0.01	0.00	0.0
BA140	0.018	0.134	0.034	0.19	0.11	0.23
LA140	0.000	0.032	0.229	-0.00	0.05	0.44
CE141	0.000	0.007	0.024	0.00	0.00	0.48
BA143	0.017	0.000	0.0	0.01	0.00	0.0
PR143	0.000	0.013	0.025	-0.00	0.01	0.25
BA144	0.111	0.000	0.0	0.07	0.00	0.0
LA144	0.033	0.000	0.0	0.04	0.00	0.0
CE144	0.002	-0.100	0.011	0.01	0.04	0.12
PR144F	0.0	0.000	0.136	0.0	0.00	0.95

Table 19

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U5TH 11/02/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 07 S
PUISSANCE RESIDUELLE 1.206E-01 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.368 %
ERREUR DUE AUX PERIODES 0.109 %
ERREUR DUE AUX ENERGIES 2.634 %
ERREUR TOTALE 2.661 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 89	0.017	0.000	0.0	0.06	0.00	0.0
KR 89	0.028	0.000	0.0	0.06	0.00	0.0
SR 89	0.000	0.075	0.043	0.00	0.04	0.33
KR 90	0.021	-0.000	0.0	-0.04	0.00	0.0
Y 90F	0.000	-0.000	0.026	-0.00	0.00	0.19
KR 91	0.038	0.000	0.0	0.06	0.00	0.0
RB 91	0.027	0.000	0.0	0.09	0.00	0.0
Y 91F	0.000	0.104	0.066	0.00	0.05	0.53
RB 95	0.043	-0.000	0.0	-0.07	0.00	0.0
SR 95	0.239	0.000	0.0	0.24	0.00	0.0
Y 95	0.048	0.000	0.0	0.07	0.00	0.0
ZR 95	0.001	0.334	0.115	0.00	0.03	0.19
NB 95F	0.000	0.093	0.190	-0.00	0.01	0.14
NB103	0.014	0.000	0.0	0.04	0.00	0.0
RU103	0.000	0.039	0.016	0.00	0.02	0.03
MO106	0.031	0.0	0.0	0.05	0.0	0.0
RU106	0.000	-0.028	0.000	0.00	0.02	0.00
RH106F	0.0	0.000	0.033	0.0	0.00	0.20
I137	0.011	-0.000	0.0	-0.02	0.00	0.0
XE137	0.012	-0.000	0.0	0.03	0.00	0.0
BA137M	0.0	-0.000	0.017	0.0	0.00	0.00
BA140	0.000	0.036	0.000	0.00	0.03	0.00
CE141	0.000	0.022	0.008	0.00	0.01	0.16
CS144	0.020	0.000	0.0	-0.06	0.00	0.0
BA144	0.291	0.000	0.0	0.18	0.00	0.0
LA144	0.085	0.000	0.0	0.11	0.00	0.0
CE144	0.005	-0.192	0.028	0.02	0.07	0.31
PR144F	0.0	0.000	0.358	0.0	0.00	2.51

Table 20

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U5TH 11/02/76 TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07 S
TEMPS DE REFROIDISSEMENT	1.000E 08 S
PUISSANCE RESIDUELLE	1.163E-02 (MEV/S)
ERREUR DUE AUX RENDEMENTS	0.448 %
ERREUR DUE AUX PERIODES	0.840 %
ERREUR DUE AUX ENERGIES	2.724 %
ERREUR TOTALE	2.886 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 90	0.065	-0.000	0.0	-0.54	0.00	0.0
KR 90	0.187	0.000	0.0	0.50	0.00	0.0
RB 90M	0.027	0.000	0.0	0.42	0.00	0.0
RB 90F	0.039	0.000	0.0	0.08	0.00	0.0
SR 90	0.002	-0.283	0.047	0.03	0.79	0.50
Y 90F	0.0	0.000	0.253	0.0	0.00	1.80
MO106	0.044	0.0	0.0	0.07	0.0	0.0
RU106	0.000	0.077	0.000	0.00	0.04	0.00
RH106F	0.0	0.000	0.045	0.0	0.00	0.27
TE137	0.015	0.000	0.0	-0.12	0.00	0.0
I137	0.101	0.000	0.0	-0.18	0.00	0.0
XE137	0.108	0.000	0.0	0.24	0.00	0.0
CS137	0.005	-0.203	0.048	0.06	0.07	0.51
BA137M	0.0	0.000	0.166	0.0	0.00	0.00
CS144	0.018	0.000	0.0	-0.05	0.00	0.0
BA144	0.262	0.000	0.0	0.16	0.00	0.0
LA144	0.077	0.000	0.0	0.10	0.00	0.0
CE144	0.004	0.755	0.021	0.01	0.27	0.23
PR144F	0.0	0.000	0.265	0.0	0.00	1.85
LA147	0.011	-0.000	0.0	0.03	0.00	0.0
CE147	0.013	0.000	0.0	0.01	0.00	0.0
PM147	0.000	0.005	0.024	-0.00	0.00	0.32

Table 21

ERREUR SUR LA PUISSANCE RESIDUELLE DUE AUX PRODUITS DE FISSION

***** PUSTH 09/03/76 *****

TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION 7.776E 07 S

TEMPS REF. (S)	PUISSANCE RESIDUELLE (MEV/S)	ERREUR DUE AU REND. (%)	ERREUR DUE A LA PER. (%)	ERREUR DUE AUX ENERGIES (%)	ERREUR TOTALE
1.0E 00	9.3464E 00	0.999	0.260	1.949	2.205
2.0E 00	8.9930E 00	0.977	0.248	1.940	2.186
5.0E 00	8.3597E 00	0.946	0.221	1.914	2.146
1.0E 01	7.7644E 00	0.907	0.248	1.879	2.101
2.0E 01	7.0938E 00	0.862	0.307	1.824	2.041
5.0E 01	6.1222E 00	0.803	0.332	1.693	1.903
1.0E 02	5.3575E 00	0.760	0.343	1.546	1.756
2.0E 02	4.6407E 00	0.727	0.337	1.407	1.619
5.0E 02	3.8040E 00	0.709	0.240	1.219	1.431
1.0E 03	3.2066E 00	0.715	0.148	0.987	1.227
2.0E 03	2.6199E 00	0.726	0.106	0.730	1.035
5.0E 03	1.9634E 00	0.678	0.108	0.620	0.925
1.0E 04	1.5921E 00	0.643	0.133	0.660	0.931
2.0E 04	1.3095E 00	0.647	0.179	0.713	0.979
5.0E 04	1.0116E 00	0.644	0.266	0.783	1.048
1.0E 05	8.2620E-01	0.623	0.246	0.856	1.087
2.0E 05	6.7558E-01	0.607	0.149	0.960	1.145
5.0E 05	5.1098E-01	0.619	0.131	1.139	1.303
1.0E 06	3.9944E-01	0.673	0.159	1.313	1.484
2.0E 06	2.9874E-01	0.777	0.200	1.556	1.750
5.0E 06	1.9225E-01	0.952	0.187	2.058	2.276
1.0E 07	1.3003E-01	1.089	0.178	2.601	2.826
2.0E 07	8.0283E-02	1.267	0.192	3.244	3.488
5.0E 07	3.4409E-02	1.464	0.269	3.660	3.951
1.0E 08	1.3820E-02	1.237	0.497	2.787	3.089

Table 22

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9TH 09/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 00 S
PUISSANCE RESIDUELLE 9.346E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.999 %
ERREUR DUE AUX PERIODES 0.260 %
ERREUR DUE AUX ENERGIES 1.949 %
ERREUR TOTALE 2.205 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
KR 90	0.010	0.000	0.004	0.05	0.00	0.01
RB 92	0.012	0.001	0.008	0.14	0.00	0.05
RB 93	0.011	0.001	0.006	0.19	0.00	0.02
SR 93	0.011	0.000	0.010	-0.12	0.00	0.08
SR 94	0.017	-0.000	0.008	-0.05	0.00	0.19
Y 94	0.002	-0.000	0.013	-0.10	0.00	0.10
SR 95	0.025	0.001	0.013	0.28	0.01	0.29
Y 95	0.007	-0.000	0.011	-0.19	0.00	0.05
SR 96	0.013	0.001	0.005	0.23	0.00	0.16
Y 96	0.011	0.000	0.019	-0.10	0.00	0.66
Y 97	0.014	0.003	0.007	0.07	0.02	0.05
ZR 98	0.011	0.001	0.007	-0.09	0.01	0.06
NB 98F	0.000	0.000	0.011	-0.01	0.00	0.09
ZR 99	0.019	0.003	0.010	0.06	0.02	0.10
NB 99F	0.001	-0.000	0.010	-0.05	0.00	0.32
ZR100	0.025	0.000	0.007	0.28	0.00	0.25
NB100F	0.002	0.001	0.019	-0.12	0.03	0.54
ZR101	0.019	0.002	0.005	0.23	0.03	0.13
NB101	0.014	0.000	0.012	-0.13	0.00	0.21
MO101	0.001	-0.000	0.011	-0.09	0.00	0.04
NB102	0.020	0.003	0.015	0.19	0.05	0.52
TC102F	0.000	0.000	0.014	-0.00	0.00	0.38
NB103	0.023	0.003	0.005	0.20	0.10	0.17
MO103	0.017	-0.000	0.023	-0.13	0.00	0.79
MO104	0.024	-0.000	0.008	0.10	0.00	0.29
TC104	0.001	-0.000	0.016	-0.05	0.00	0.38
MO105	0.028	0.000	0.014	0.27	0.00	0.51
TC105	0.005	0.000	0.013	-0.14	0.00	0.33
MO106	0.012	0.000	0.003	0.16	0.00	0.10
TC106	0.008	0.000	0.012	0.05	0.00	0.35
I132F	0.000	-0.000	0.015	-0.01	0.00	0.02
TE134	0.017	0.000	0.003	0.03	0.00	0.09
I134F	0.007	0.001	0.022	-0.02	0.00	0.06
TE135	0.012	0.000	0.007	0.16	0.00	0.26
I135	0.012	0.000	0.012	-0.09	0.00	0.03
I136F	0.007	-0.000	0.010	0.07	0.00	0.04
I137	0.011	0.000	0.007	0.21	0.00	0.26
XE137	0.008	0.000	0.012	-0.11	0.00	0.08
XE138	0.023	-0.000	0.009	0.12	0.00	0.03
CS138F	0.001	0.001	0.018	0.04	0.00	0.06
XE139	0.018	0.000	0.009	0.13	0.00	0.04
CS139	0.006	-0.000	0.011	0.01	0.00	0.08
XE140	0.020	0.000	0.006	0.10	0.00	0.11

Table 22 cont'd

I CS140	I	0.022	-0.000	0.019	I	0.19	0.00	0.08 I
I CS141	I	0.016	0.000	0.008	I	0.17	0.00	0.18 I
I LA140	I	0.000	-0.000	0.015	I	-0.00	0.00	0.03 I
I CS142	I	0.011	0.001	0.004	I	0.13	0.01	0.08 I
I BA142	I	0.016	-0.000	0.007	I	-0.09	0.00	0.05 I
I LA142	I	0.001	-0.000	0.016	I	-0.03	0.00	0.03 I
I BA143	I	0.016	0.000	0.008	I	-0.03	0.00	0.28 I
I BA144	I	0.013	0.000	0.004	I	0.09	0.00	0.15 I
I LA146	I	0.013	0.000	0.006	I	0.12	0.02	0.21 I

Table 23

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9TH 09/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 01 S
PUISSANCE RESIDUELLE 7.764E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.907 %
ERREUR DUE AUX PERIODES 0.248 %
ERREUR DUE AUX ENERGIES 1.879 %
ERREUR TOTALE 2.101 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
KR 89	0.012	0.000	0.006	0.06	0.00	0.02
KR 90	0.012	0.001	0.004	0.05	0.00	0.01
RB 91	0.010	0.001	0.010	0.13	0.01	0.04
SR 93	0.013	0.000	0.012	-0.09	0.00	0.10
SR 94	0.020	0.001	0.009	0.11	0.00	0.21
Y 94	0.002	-0.000	0.016	-0.11	0.00	0.12
SR 95	0.027	0.004	0.012	0.29	0.04	0.27
Y 95	0.008	-0.000	0.013	-0.20	0.00	0.07
SR 96	0.011	0.002	0.001	0.16	0.01	0.04
Y 96	0.012	0.001	0.022	0.09	0.02	0.76
Y 97	0.010	0.000	0.000	0.06	0.00	0.00

Table 23 cont'd

I ZR 98	I	0.011	0.004	0.007	I 0.02	0.04	0.06	I
I NB 98F	I	0.000	0.001	0.012	I -0.00	0.03	0.09	I
I ZR 99	I	0.012	0.004	0.001	I 0.02	0.04	0.01	I
I ZR100	I	0.014	0.010	0.003	I 0.20	0.05	0.12	I
I ZR101	I	0.013	0.002	0.000	I 0.10	0.03	0.01	I
I NB101	I	0.013	0.005	0.006	I -0.02	0.02	0.11	I
I MO101	I	0.002	-0.000	0.014	I -0.07	0.00	0.04	I
I NB102	I	0.013	0.006	0.002	I 0.08	0.08	0.07	I
I TC102F	I	0.000	0.000	0.017	I -0.00	0.00	0.46	I
I NB103	I	0.020	0.000	0.000	I 0.09	0.01	0.00	I
I MO103	I	0.019	0.002	0.025	I 0.07	0.02	0.86	I
I MO104	I	0.028	0.001	0.009	I 0.11	0.01	0.32	I
I TC104	I	0.002	-0.000	0.019	I -0.06	0.00	0.46	I
I MO105	I	0.031	0.003	0.015	I 0.32	0.03	0.52	I
I TC105	I	0.006	0.000	0.015	I -0.15	0.00	0.40	I
I MO106	I	0.013	0.002	0.002	I 0.16	0.03	0.07	I
I TC106	I	0.009	0.002	0.013	I 0.05	0.02	0.38	I
I SB131	I	0.010	0.000	0.008	I -0.09	0.00	0.18	I
I TE132	I	0.011	0.000	0.002	I -0.15	0.00	0.01	I
I I132F	I	0.000	-0.000	0.018	I -0.01	0.00	0.03	I
I TE133M	I	0.011	0.000	0.007	I 0.11	0.00	0.03	I
I TE134	I	0.020	0.001	0.004	I 0.06	0.00	0.11	I
I I134F	I	0.008	0.001	0.027	I -0.02	0.00	0.07	I
I TE135	I	0.012	0.003	0.006	I 0.14	0.03	0.22	I
I I135	I	0.014	0.000	0.014	I -0.09	0.00	0.03	I
I I136F	I	0.008	0.000	0.012	I 0.09	0.00	0.05	I
I I137	I	0.012	0.002	0.007	I 0.21	0.00	0.24	I
I XE137	I	0.009	0.001	0.014	I -0.11	0.00	0.09	I
I XE138	I	0.027	0.000	0.011	I 0.15	0.00	0.04	I
I CS138F	I	0.002	0.001	0.022	I 0.05	0.00	0.07	I
I XE139	I	0.020	0.002	0.009	I 0.15	0.00	0.04	I
I CS139	I	0.007	-0.000	0.013	I 0.06	0.00	0.09	I
I XE140	I	0.021	0.003	0.004	I 0.09	0.03	0.08	I
I CS140	I	0.025	0.001	0.021	I 0.22	0.01	0.09	I
I CS141	I	0.018	0.002	0.008	I 0.18	0.00	0.17	I
I LA140	I	0.000	-0.000	0.018	I -0.00	0.00	0.03	I
I BA141	I	0.007	-0.000	0.011	I -0.12	0.00	0.06	I
I BA142	I	0.010	-0.000	0.008	I 0.01	0.00	0.06	I
I LA142	I	0.001	-0.000	0.019	I -0.03	0.00	0.04	I
I BA143	I	0.015	0.003	0.006	I 0.07	0.03	0.20	I
I BA144	I	0.013	0.002	0.003	I 0.09	0.02	0.11	I
I LA146	I	0.013	0.002	0.005	I 0.11	0.15	0.16	I

Table 24

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PUOTH 09/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 02 S
PUISSANCE RESIDUELLE 5.358E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.760 %
ERREUR DUE AUX PERIODES 0.343 %
ERREUR DUE AUX ENERGIES 1.546 %
ERREUR TOTALE 1.756 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
KR 89	0.014	0.003	0.006	0.09	0.00	0.02
RB 89	0.002	-0.000	0.010	-0.07	0.00	0.01
SR 93	0.016	0.003	0.015	0.04	0.01	0.12
SR 94	0.022	0.005	0.005	0.11	0.00	0.13
Y 94	0.003	0.000	0.023	-0.10	0.00	0.17
SR 95	0.023	0.006	0.002	0.14	0.06	0.04
Y 95	0.011	0.001	0.017	-0.11	0.00	0.09
Y 96	0.011	0.010	0.020	0.11	0.17	0.69
Y 97	0.015	0.000	0.000	0.08	0.00	0.00
NB 97F	0.000	-0.000	0.011	-0.01	0.00	0.04
ZR101	0.012	0.000	0.000	0.03	0.00	0.00
NB101	0.013	0.000	0.000	0.04	0.00	0.00
MO101	0.002	0.001	0.018	0.01	0.00	0.06
NB102	0.015	0.000	0.000	0.07	0.00	0.00
TC102F	0.000	0.000	0.023	-0.00	0.00	0.61
NB103	0.014	0.000	0.000	0.06	0.00	0.00
MO103	0.014	0.016	0.012	0.05	0.16	0.43
MO104	0.031	0.006	0.005	0.11	0.04	0.16
TC104	0.002	0.000	0.027	-0.05	0.00	0.65
MO105	0.029	0.009	0.005	0.26	0.09	0.16
TC105	0.008	0.002	0.020	-0.10	0.01	0.52
RU105	0.000	0.000	0.010	-0.01	0.00	0.04
RU106	0.002	-0.010	0.000	-0.05	0.01	0.00
RH106F	0.000	0.000	0.010	-0.00	0.00	0.06
SN130	0.011	0.002	0.003	0.24	0.01	0.12
SB131	0.015	0.001	0.011	0.07	0.00	0.25
SB132M	0.013	0.002	0.006	0.20	0.03	0.13
SB132F	0.011	0.003	0.005	0.12	0.05	0.13
TE132	0.016	0.001	0.003	-0.17	0.00	0.02
I132F	0.000	0.000	0.026	-0.02	0.00	0.04
SB133	0.010	0.002	0.004	0.18	0.03	0.08
TE133M	0.016	0.001	0.010	0.19	0.00	0.04
TE133F	0.012	0.000	0.011	-0.16	0.00	0.07
I133F	0.002	-0.000	0.012	-0.07	0.00	0.05
TE134	0.029	0.001	0.006	0.09	0.00	0.15
I134F	0.012	0.002	0.039	-0.03	0.00	0.10
I135	0.021	0.001	0.020	0.02	0.00	0.04
XE137	0.010	0.005	0.015	-0.02	0.00	0.10
XE138	0.038	0.001	0.014	0.21	0.00	0.06
CS138F	0.002	0.002	0.032	0.07	0.00	0.10
XE139	0.019	0.005	0.002	0.12	0.01	0.01
CS139	0.010	0.001	0.017	0.13	0.00	0.12

Table 24 cont'd

I XE140	I	0.017	0.002	0.000	I 0.05	0.02	0.00 I
I CS140	I	0.024	0.011	0.012	I 0.16	0.11	0.05 I
I CS141	I	0.016	0.003	0.001	I 0.10	0.00	0.02 I
I LA140	I	0.000	-0.000	0.026	I -0.00	0.00	0.05 I
I CS142	I	0.013	0.000	0.000	I 0.05	0.00	0.00 I
I BA141	I	0.010	0.001	0.015	I -0.05	0.00	0.08 I
I BA142	I	0.027	0.001	0.010	I 0.04	0.00	0.08 I
I LA142	I	0.001	-0.000	0.028	I -0.04	0.00	0.06 I
I BA143	I	0.015	0.001	0.000	I 0.01	0.01	0.00 I
I LA143	I	0.002	0.000	0.010	I -0.02	0.00	0.08 I
I LA146	I	0.011	0.001	0.000	I 0.02	0.05	0.00 I
I PR146	I	0.000	0.000	0.012	I -0.00	0.00	0.05 I

Table 25

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9TH 09/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 03 S
PUISSANCE RESIDUELLE 3.207E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.715 %
ERREUR DUE AUX PERIODES 0.148 %
ERREUR DUE AUX ENERGIES 0.987 %
ERREUR TOTALE 1.227 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
I BR 88	I 0.011	0.000	0.000	I -0.04	0.00	0.00 I
I KR 88	I 0.013	0.000	0.008	I 0.07	0.00	0.02 I
I RB 88	I 0.000	0.001	0.015	I -0.03	0.00	0.04 I
I KR 89	I 0.010	0.003	0.000	I 0.03	0.00	0.00 I
I RB 92	I 0.015	0.000	0.0	I 0.05	0.00	0.0 I
I SR 92	I 0.011	0.001	0.013	I 0.03	0.00	0.03 I
I Y 92	I 0.000	0.000	0.015	I -0.01	0.00	0.08 I
I SR 93	I 0.014	0.010	0.006	I 0.02	0.02	0.05 I
I Y 93F	I 0.000	0.001	0.015	I -0.01	0.01	0.10 I
I SR 94	I 0.016	0.001	0.000	I 0.05	0.00	0.00 I
I Y 94	I 0.003	0.012	0.022	I -0.04	0.01	0.16 I
I SR 95	I 0.023	0.000	0.000	I 0.08	0.00	0.00 I
I Y 95	I 0.012	0.011	0.010	I -0.01	0.03	0.05 I
I ZR 95	I 0.001	0.002	0.012	I -0.03	0.00	0.02 I
I NB 95F	I 0.000	0.000	0.011	I -0.00	0.00	0.01 I
I Y 97	I 0.025	0.000	0.0	I 0.14	0.00	0.0 I
I ZR 97	I 0.014	-0.000	0.015	I 0.10	0.00	0.09 I
I NB 97M	I 0.000	0.000	0.010	I -0.00	0.00	0.00 I
I NB 97F	I 0.000	0.000	0.019	I -0.01	0.00	0.07 I

Table 25 cont'd

MO 99	0.000	0.000	0.010	0.00	0.00	0.06
ZR101	0.012	0.000	0.0	0.03	0.00	0.0
NB101	0.012	0.000	0.0	0.04	0.00	0.0
MO101	0.002	0.013	0.014	0.01	0.00	0.05
TC101	0.000	0.002	0.010	-0.00	0.00	0.06
NB102	0.010	0.000	0.0	0.05	0.00	0.0
MO102	0.004	0.016	0.002	0.02	0.03	0.09
TC102F	0.0	0.000	0.015	0.0	0.00	0.39
RU103	0.000	0.000	0.011	0.00	0.00	0.02
MO104	0.024	0.002	0.000	0.07	0.01	0.00
TC104	0.002	0.015	0.025	0.01	0.03	0.61
MO105	0.021	0.001	0.000	0.17	0.01	0.00
TC105	0.007	0.013	0.008	0.04	0.07	0.22
RU105	0.000	0.000	0.017	-0.01	0.00	0.06
RU106	0.003	-0.017	0.000	0.02	0.01	0.00
RH106F	0.000	0.000	0.017	-0.00	0.00	0.10
SB131	0.019	0.007	0.012	0.17	0.02	0.26
SB132M	0.012	0.002	0.001	0.11	0.04	0.02
SB132F	0.011	0.001	0.000	0.03	0.01	0.00
TE132	0.026	0.001	0.005	-0.02	0.00	0.03
I132F	0.000	0.000	0.044	-0.02	0.00	0.07
TE133M	0.024	0.004	0.013	0.40	0.00	0.05
TE133F	0.013	0.006	0.009	-0.24	0.01	0.06
I133F	0.004	-0.000	0.021	-0.09	0.00	0.09
TE134	0.045	0.005	0.007	0.16	0.01	0.20
I134F	0.016	0.008	0.059	-0.06	0.02	0.16
TE135	0.013	0.000	0.000	0.06	0.00	0.00
I135	0.034	0.002	0.033	0.06	0.00	0.07
XE135F	0.001	0.000	0.012	-0.01	0.00	0.07
XE138	0.048	0.014	0.011	0.29	0.01	0.04
CS138F	0.003	0.009	0.047	0.10	0.00	0.15
XE139	0.014	0.001	0.000	0.08	0.00	0.00
CS139	0.009	0.011	0.009	0.12	0.04	0.06
BA139	0.001	0.001	0.015	-0.05	0.00	0.10
XE140	0.018	0.000	0.000	0.01	0.00	0.00
CS140	0.028	0.000	0.000	0.05	0.00	0.00
CS141	0.019	0.000	0.000	0.09	0.00	0.00
LA140	0.000	-0.000	0.044	-0.00	0.00	0.08
CS142	0.017	0.000	0.0	0.06	0.00	0.0
BA141	0.013	0.009	0.014	0.03	0.00	0.07
LA141	0.000	0.000	0.015	-0.01	0.00	0.12
BA142	0.036	0.008	0.006	0.04	0.01	0.05
LA142	0.001	0.002	0.044	-0.04	0.00	0.09
BA143	0.017	0.000	0.000	-0.01	0.00	0.00
CE144	0.000	-0.010	0.001	0.00	0.00	0.01
PR144F	0.000	0.000	0.013	-0.00	0.00	0.09
LA146	0.015	0.000	0.000	0.02	0.01	0.00
PR146	0.000	0.003	0.017	-0.00	0.01	0.07

Table 26

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9TH 09/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 04 S
PUISSANCE RESIDUELLE 1.592E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.643 %
ERREUR DUE AUX PERIODES 0.133 %
ERREUR DUE AUX ENERGIES 0.660 %
ERREUR TOTALE 0.931 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 88	0.012	0.000	0.0	-0.04	0.00	0.0
KR 88	0.014	0.016	0.009	0.08	0.01	0.02
RB 88	0.000	0.002	0.017	-0.00	0.00	0.04
RB 91	0.016	0.000	0.0	0.04	0.00	0.0
RB 92	0.021	0.000	0.0	0.07	0.00	0.0
SR 91	0.005	0.005	0.015	0.02	0.05	0.06
SR 92	0.015	0.013	0.013	0.04	0.00	0.03
Y 92	0.000	0.004	0.025	-0.01	0.00	0.14
SR 93	0.016	0.000	0.000	0.01	0.00	0.00
Y 93F	0.001	0.006	0.025	-0.00	0.06	0.16
SR 95	0.032	0.000	0.0	0.10	0.00	0.0
Y 95	0.017	0.000	0.000	0.05	0.00	0.00
ZR 95	0.001	0.004	0.025	0.01	0.00	0.04
NB 95F	0.000	0.001	0.023	-0.00	0.00	0.02
SR 97	0.014	0.000	0.0	0.16	0.00	0.0
Y 97	0.046	0.000	0.0	0.25	0.00	0.0
ZR 97	0.026	0.007	0.027	0.18	0.01	0.16
NB 97M	0.000	0.000	0.019	-0.00	0.00	0.00
NB 97F	0.000	0.002	0.035	-0.01	0.00	0.13
ZR 99	0.018	0.000	0.0	0.07	0.00	0.0
MO 99	0.000	0.001	0.020	0.00	0.00	0.12
NB103	0.013	0.000	0.0	0.05	0.00	0.0
MO103	0.013	0.000	0.0	0.05	0.00	0.0
RU103	0.000	0.001	0.023	0.00	0.00	0.04
MO105	0.021	0.000	0.0	0.17	0.00	0.0
RU105	0.000	0.010	0.023	0.00	0.00	0.08
MO106	0.015	0.0	0.0	0.06	0.0	0.0
TC106	0.014	0.000	0.0	0.06	0.00	0.0
RU106	0.006	-0.035	0.000	0.05	0.02	0.00
RH106F	0.0	0.000	0.034	0.0	0.00	0.20
I131	0.000	-0.000	0.012	-0.00	0.00	0.05
SB132M	0.021	0.000	0.000	0.11	0.00	0.00
SB132F	0.021	0.000	0.000	0.11	0.00	0.00
TE132	0.053	0.003	0.010	0.14	0.00	0.05
I132F	0.000	0.001	0.087	-0.04	0.01	0.14
TE133M	0.022	0.010	0.004	0.28	0.01	0.01
TE133F	0.016	0.000	0.000	-0.19	0.00	0.00
I133F	0.007	0.002	0.039	-0.06	0.00	0.17
TE134	0.025	0.022	0.001	0.11	0.04	0.03
I134F	0.005	0.037	0.027	-0.04	0.08	0.07
TE135	0.021	0.000	0.0	0.11	0.00	0.0
I135	0.055	0.019	0.051	0.09	0.03	0.11

Table 26 cont'd

XE135F		0.001	0.002	0.023	-0.02	0.00	0.13	
CS136F		0.000	0.017	0.005	0.01	0.01	0.01	
BA139		0.001	0.011	0.008	0.01	0.00	0.06	
XE140		0.036	0.000	0.0	0.02	0.00	0.0	
CS140		0.057	0.000	0.0	0.11	0.00	0.0	
CS141		0.016	0.000	0.0	0.07	0.00	0.0	
BA140		0.017	-0.002	0.015	0.04	0.00	0.10	
LA140		0.000	-0.000	0.089	-0.01	0.00	0.17	
BA141		0.011	0.003	0.000	0.03	0.00	0.00	
LA141		0.000	0.009	0.020	-0.00	0.01	0.16	
BA142		0.021	0.004	0.000	0.02	0.00	0.00	
LA142		0.001	0.033	0.029	-0.02	0.02	0.06	
BA143		0.021	0.000	0.0	0.02	0.00	0.0	
CE143		0.000	0.001	0.019	-0.00	0.00	0.09	
BA144		0.018	0.000	0.0	0.01	0.00	0.0	
CE144		0.001	-0.020	0.002	0.00	0.01	0.02	
PR144F		0.000	0.000	0.026	-0.00	0.00	0.18	

Table 27

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PUSTH 09/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 05 S
PUISSANCE RESIDUELLE 8.262E-01 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.623 %
ERREUR DUE AUX PERIODES 0.246 %
ERREUR DUE AUX ENERGIES 0.856 %
ERREUR TOTALE 1.087 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
SR 89	0.000	0.001	0.011	0.00	0.00	0.09
RB 91	0.013	0.000	0.0	0.03	0.00	0.0
SR 91	0.004	0.013	0.004	0.02	0.14	0.02
Y 91F	0.000	0.001	0.017	-0.00	0.00	0.14
Y 93F	0.000	0.017	0.008	0.00	0.17	0.05
SR 95	0.061	0.000	0.0	0.18	0.00	0.0
Y 95	0.033	0.000	0.0	0.10	0.00	0.0
ZR 95	0.003	0.008	0.048	0.01	0.00	0.08
NB 95F	0.000	0.001	0.044	-0.00	0.00	0.03
SR 97	0.010	0.000	0.0	0.11	0.00	0.0
Y 97	0.032	0.000	0.0	0.18	0.00	0.0
ZR 97	0.018	0.060	0.018	0.12	0.07	0.11
NB 97M	0.000	0.000	0.012	-0.00	0.00	0.00
NB 97F	0.000	0.002	0.023	-0.00	0.00	0.09
ZR 99	0.027	0.000	0.0	0.10	0.00	0.0
MO 99	0.000	0.012	0.030	0.00	0.00	0.18
NB103	0.025	0.000	0.0	0.10	0.00	0.0
MO103	0.024	0.000	0.0	0.09	0.00	0.0
RU103	0.000	0.002	0.044	0.00	0.00	0.07
MO106	0.030	0.0	0.0	0.12	0.0	0.0
TC106	0.028	0.000	0.0	0.11	0.00	0.0
RU106	0.012	-0.067	0.000	0.10	0.04	0.01
RH106F	0.0	0.000	0.066	0.0	0.00	0.39
SB131	0.013	0.000	0.000	-0.09	0.00	0.00
I131	0.000	0.001	0.022	-0.00	0.00	0.08
SB132M	0.033	0.000	0.0	0.18	0.00	0.0
SB132F	0.033	0.000	0.0	0.18	0.00	0.0
TE132	0.081	0.040	0.016	0.22	0.03	0.08
I132F	0.000	0.004	0.133	-0.00	0.03	0.21
TE133M	0.017	0.001	0.000	0.08	0.00	0.00
TE133F	0.016	0.000	0.000	-0.03	0.00	0.00
I133F	0.007	0.029	0.031	-0.02	0.01	0.13
XE133F	0.000	0.001	0.014	-0.00	0.00	0.09
I135	0.016	0.034	0.007	0.03	0.05	0.01
XE135F	0.000	0.014	0.012	-0.01	0.00	0.07
XE140	0.067	0.000	0.0	0.03	0.00	0.0
CS140	0.108	0.000	0.0	0.20	0.00	0.0
BA140	0.032	0.001	0.028	0.08	0.00	0.18
LA140	0.000	0.002	0.169	-0.01	0.00	0.32
CE141	0.000	0.001	0.014	-0.00	0.00	0.28
BA143	0.029	0.000	0.0	0.04	0.00	0.0
CE143	0.000	0.013	0.020	-0.00	0.02	0.10
PR143	0.000	0.000	0.016	-0.00	0.00	0.16
BA144	0.035	0.000	0.0	0.03	0.00	0.0
LA144	0.016	0.000	0.0	0.01	0.00	0.0
CE144	0.001	-0.039	0.004	0.00	0.01	0.04
PR144F	0.000	0.000	0.049	-0.00	0.00	0.35

Table 28

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9TH 09/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 06 S
PUISSANCE RESIDUELLE 3.994E-01 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.673 %
ERREUR DUE AUX PERIODES 0.159 %
ERREUR DUE AUX ENERGIES 1.313 %
ERREUR TOTALE 1.484 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
KR 89	0.015	0.000	0.0	0.05	0.00	0.0
SR 89	0.000	0.004	0.020	0.00	0.00	0.16
KR 91	0.010	0.000	0.0	0.01	0.00	0.0
RB 91	0.017	0.000	0.0	0.04	0.00	0.0
Y 91F	0.000	0.006	0.031	-0.00	0.00	0.25
SR 95	0.118	0.000	0.0	0.35	0.00	0.0
Y 95	0.063	0.000	0.0	0.19	0.00	0.0
ZR 95	0.005	0.029	0.088	0.03	0.00	0.15
NB 95F	0.000	0.004	0.090	-0.00	0.00	0.06
MO 99	0.000	0.018	0.004	0.00	0.00	0.02
NB103	0.042	0.000	0.0	0.17	0.00	0.0
MO103	0.040	0.000	0.0	0.16	0.00	0.0
RU103	0.000	0.021	0.074	0.00	0.01	0.11
MO106	0.060	0.0	0.0	0.24	0.0	0.0
TC106	0.056	0.000	0.0	0.23	0.00	0.0
RU106	0.024	-0.135	0.001	0.19	0.07	0.01
RH106F	0.0	0.000	0.134	0.0	0.00	0.79
SB131	0.010	0.000	0.0	-0.01	0.00	0.0
I131	0.000	0.017	0.018	-0.00	0.00	0.06
TE132	0.021	0.087	0.003	0.06	0.07	0.02
I132F	0.0	0.001	0.027	0.0	0.01	0.04
XE140	0.082	0.000	0.0	0.04	0.00	0.0
CS140	0.132	0.000	0.0	0.25	0.00	0.0
CS141	0.013	0.000	0.0	0.06	0.00	0.0
BA140	0.039	0.126	0.032	0.10	0.11	0.21
LA140	0.000	0.030	0.212	-0.00	0.05	0.40
CE141	0.000	0.007	0.023	0.00	0.00	0.46
BA143	0.016	0.000	0.0	0.02	0.00	0.0
PR143	0.000	0.010	0.020	-0.00	0.01	0.20
BA144	0.071	0.000	0.0	0.06	0.00	0.0
LA144	0.032	0.000	0.0	0.03	0.00	0.0
CE144	0.002	-0.076	0.008	0.00	0.03	0.09
PR144F	0.0	0.000	0.099	0.0	0.00	0.69

Table 29

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PUSTH 09/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 07 S
PUISSANCE RESIDUELLE 1.300E-01 (MEV/S)
ERREUR DUE AUX RENDEMENTS 1.089 %
ERREUR DUE AUX PERIODES 0.178 %
ERREUR DUE AUX ENERGIES 2.601 %
ERREUR TOTALE 2.826 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
KR 89	0.012	0.000	0.0	0.04	0.00	0.0
SR 89	0.090	0.025	0.014	0.00	0.01	0.11
RB 91	0.015	0.000	0.0	0.04	0.00	0.0
Y 91F	0.000	0.042	0.026	-0.00	0.02	0.21
SR 95	0.149	0.000	0.0	0.45	0.00	0.0
Y 95	0.080	0.000	0.0	0.24	0.00	0.0
ZR 95	0.006	0.241	0.082	0.04	0.02	0.14
NB 95F	0.000	0.068	0.135	-0.00	0.01	0.10
NB103	0.022	0.000	0.0	0.09	0.00	0.0
MO103	0.022	0.000	0.0	0.09	0.00	0.0
RU103	0.000	0.084	0.034	0.00	0.04	0.05
MO106	0.148	0.0	0.0	0.60	0.0	0.0
TC106	0.138	0.000	0.0	0.56	0.00	0.0
RU106	0.060	-0.295	0.002	0.48	0.16	0.03
RH106F	0.0	0.000	0.335	0.0	0.00	1.99
XE137	0.014	-0.000	0.0	0.04	0.00	0.0
BA137M	0.0	0.000	0.017	0.0	0.00	0.00
BA140	0.001	0.031	0.000	0.00	0.03	0.00
CE141	0.000	0.020	0.007	0.00	0.01	0.14
CS144	0.011	0.000	0.0	-0.07	0.00	0.0
BA144	0.166	0.000	0.0	0.13	0.00	0.0
LA144	0.075	0.000	0.0	0.06	0.00	0.0
CE144	0.004	-0.135	0.018	0.01	0.05	0.20
PR144F	0.0	0.000	0.233	0.0	0.00	1.63

Table 30

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9TH 09/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 08 S
PUISSANCE RESIDUELLE 1.382E-02 (MEV/S)
ERREUR DUE AUX RENDEMENTS 1.237 %
ERREUR DUE AUX PERIODES 0.497 %
ERREUR DUE AUX ENERGIES 2.787 %
ERREUR TOTALE 3.089 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
KR 90	0.065	0.000	0.0	0.05	0.00	0.0
RB 90M	0.018	0.000	0.0	0.12	0.00	0.0
RB 90F	0.018	0.000	0.0	0.12	0.00	0.0
SR 90	0.003	-0.094	0.014	0.02	0.26	0.15
Y 90F	0.0	0.000	0.077	0.0	0.00	0.55
MO106	0.186	0.0	0.0	0.76	0.0	0.0
TC106	0.173	0.000	0.0	0.70	0.00	0.0
RU106	0.075	0.716	0.003	0.60	0.39	0.04
RH106F	0.0	0.000	0.412	0.0	0.00	2.44
I137	0.071	0.0	0.0	-0.35	0.0	0.0
XE137	0.135	0.000	0.0	0.39	0.00	0.0
CS137	0.016	-0.201	0.043	0.18	0.07	0.46
BA137M	0.0	0.0	0.150	0.0	0.0	0.00
BA144	0.125	0.000	0.0	0.10	0.00	0.0
LA144	0.057	0.000	0.0	0.05	0.00	0.0
CE144	0.003	0.424	0.012	0.01	0.15	0.14
PR144F	0.0	0.000	0.156	0.0	0.00	1.10
CE147	0.011	0.000	0.0	0.02	0.00	0.0
PM147	0.000	0.005	0.019	-0.00	0.00	0.24

Table 31

ERREUR SUR LA PUISSANCE RESIDUELLE DUE AUX PRODUITS DE FISSION
***** USRAP 10/03/76 *****
TAUX DE FISSION CONSTANT (1 FTS./S.)

TEMPS D'IRRADIATION: 7.776E 07 S

TEMPS REF. (S)	PUISSANCE RESIDUELLE (MEV/S)	ERREUR DUE AU REID. (%)	ERREUR DUE A LA PER. (%)	ERREUR DUE AUX ENERGIES (%)	ERREUR TOTALE
1.0E 00	1.0640E 01	0.852	0.279	1.645	1.873
2.0E 00	1.0202E 01	0.816	0.242	1.632	1.840
5.0E 00	9.4199E 00	0.740	0.192	1.588	1.771
1.0E 01	8.6898E 00	0.705	0.214	1.528	1.696
2.0E 01	7.8865E 00	0.647	0.274	1.447	1.608
5.0E 01	6.7650E 00	0.576	0.312	1.325	1.478
1.0E 02	5.9064E 00	0.529	0.347	1.197	1.354
2.0E 02	5.1075E 00	0.501	0.399	1.026	1.210
5.0E 02	4.1883E 00	0.516	0.306	0.850	1.041
1.0E 03	3.5529E 00	0.560	0.143	0.748	0.946
2.0E 03	2.9323E 00	0.610	0.087	0.662	0.904
5.0E 03	2.2047E 00	0.602	0.113	0.659	0.899
1.0E 04	1.7608E 00	0.529	0.172	0.708	0.900
2.0E 04	1.4035E 00	0.493	0.280	0.765	0.952
5.0E 04	1.0375E 00	0.508	0.459	0.832	1.078
1.0E 05	8.3107E-01	0.523	0.424	0.918	1.138
2.0E 05	6.7807E-01	0.541	0.213	1.041	1.192
5.0E 05	5.1897E-01	0.570	0.112	1.231	1.361
1.0E 06	4.0955E-01	0.585	0.143	1.394	1.518
2.0E 06	3.0568E-01	0.596	0.186	1.607	1.724
5.0E 06	1.9001E-01	0.663	0.151	2.036	2.147
1.0E 07	1.1977E-01	0.753	0.107	2.562	2.673
2.0E 07	6.5316E-02	0.907	0.102	3.394	3.514
5.0E 07	2.4849E-02	1.060	0.452	3.862	4.030
1.0E 08	1.1554E-02	0.963	0.799	2.632	2.914

Table 32

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
USRAP 10/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07 S
TEMPS DE REFROIDISSEMENT	1.000E 00 S
PUISSANCE RESIDUELLE	1.064E 01 (MEV/S)
ERREUR DUE AUX RENDEMENTS	0.852 %
ERREUR DUE AUX PERIODES	0.270 %
ERREUR DUE AUX ENERGIES	1.645 %
ERREUR TOTALE	1.873 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 88	0.017	0.000	0.009	0.20	0.00	0.30
BR 89	0.012	0.001	0.004	0.19	0.00	0.14
RB 88	0.000	0.000	0.013	-0.02	0.00	0.03
KR 89	0.022	0.000	0.013	-0.07	0.00	0.03
RB 89	0.001	-0.001	0.014	-0.08	0.00	0.02
KR 90	0.028	0.000	0.011	0.18	0.00	0.03
RB 90F	0.003	0.000	0.017	-0.13	0.00	0.05
KR 91	0.021	0.000	0.006	0.10	0.00	0.18
SR 91	0.015	-0.000	0.010	0.15	0.00	0.07
RB 92	0.022	0.002	0.017	0.22	0.00	0.10
RB 93	0.021	0.001	0.012	0.27	0.00	0.03
RB 94	0.011	0.001	0.003	0.18	0.00	0.11
SR 93	0.012	0.000	0.015	-0.17	0.00	0.12
SR 94	0.020	-0.000	0.010	-0.10	0.00	0.25
Y 94	0.001	-0.000	0.016	-0.07	0.00	0.12
SR 95	0.033	0.001	0.018	0.20	0.01	0.40
Y 95	0.005	-0.000	0.012	-0.15	0.00	0.06
SR 96	0.018	0.001	0.006	0.21	0.01	0.22
Y 96	0.009	0.000	0.021	-0.13	0.00	0.73
Y 97	0.013	0.003	0.007	-0.02	0.02	0.06
NB 98F	0.000	0.000	0.010	-0.01	0.00	0.08
ZR 99	0.015	0.002	0.009	-0.07	0.02	0.08
ZR100	0.022	0.000	0.006	0.15	0.00	0.22
NB100F	0.001	0.001	0.016	-0.08	0.02	0.46
ZR101	0.020	0.002	0.005	0.19	0.03	0.14
NB101	0.009	0.000	0.010	-0.14	0.00	0.18
NB102	0.013	0.002	0.011	0.13	0.03	0.40
NB103	0.011	0.002	0.002	0.08	0.05	0.08
I132F	0.000	0.000	0.012	-0.00	0.00	0.02
TE134	0.019	0.000	0.004	0.05	0.00	0.11
I134F	0.002	0.000	0.020	-0.08	0.00	0.05
TE135	0.018	0.000	0.011	0.26	0.01	0.39
I136F	0.006	-0.000	0.010	0.09	0.00	0.04
I137	0.011	0.000	0.008	0.15	0.00	0.26
I138	0.011	0.000	0.005	0.22	0.00	0.17
XE138	0.022	-0.000	0.009	-0.12	0.00	0.04
CS138F	0.001	0.001	0.010	-0.06	0.00	0.06
XE139	0.022	0.000	0.010	-0.01	0.00	0.05
CS139	0.005	-0.000	0.011	-0.07	0.00	0.08
XE140	0.027	0.000	0.008	0.12	0.00	0.15
CS140	0.018	-0.000	0.020	0.11	0.00	0.08
CS141	0.017	0.000	0.009	0.17	0.00	0.20

Table 32 cont'd

I LA140	I	0.000	-0.000	0.014	I	-0.00	0.00	0.03 I
I CS142	I	0.014	0.002	0.005	I	0.18	0.01	0.11 I
I BA142	I	0.012	-0.000	0.006	I	-0.14	0.00	0.05 I
I LA142	I	0.000	-0.000	0.015	I	-0.02	0.00	0.03 I
I BA143	I	0.016	0.000	0.009	I	0.02	0.00	0.31 I
I BA144	I	0.015	0.000	0.005	I	0.10	0.00	0.18 I
I LA146	I	0.012	0.000	0.006	I	0.15	0.02	0.19 I

Table 33

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
USRAP 10/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 01 S
PUISSANCE RESIDUELLE 8.690E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.705 %
ERREUR DUE AUX PERIODES 0.214 %
ERREUR DUE AUX ENERGIES 1.528 %
ERREUR TOTALE 1.696 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
I BR 88	0.017	0.003	0.007	0.20	0.00	0.24
I BR 89	0.011	0.002	0.001	0.14	0.00	0.04
I KR 88	0.012	-0.001	0.009	-0.16	0.00	0.02
I RB 88	0.000	0.000	0.016	-0.02	0.00	0.04
I KR 89	0.026	0.001	0.016	0.07	0.00	0.04
I RB 89	0.001	-0.001	0.017	-0.08	0.00	0.02
I KR 90	0.031	0.003	0.011	0.23	0.00	0.03
I BR 90F	0.003	0.000	0.021	-0.14	0.01	0.06
I KR 91	0.021	0.003	0.003	0.09	0.03	0.10
I RB 91	0.017	0.002	0.021	0.17	0.02	0.09
I RB 92	0.016	0.008	0.005	0.11	0.01	0.03
I RB 93	0.017	0.006	0.005	0.17	0.00	0.01
I Y 92	0.000	0.000	0.010	-0.01	0.00	0.06
I SR 93	0.014	0.000	0.018	-0.12	0.00	0.14
I SR 94	0.023	0.001	0.012	0.09	0.00	0.28
I Y 94	0.001	-0.000	0.020	-0.07	0.00	0.15
I SR 95	0.036	0.005	0.018	0.22	0.05	0.39
I Y 95	0.006	-0.000	0.015	-0.16	0.00	0.08
I SR 96	0.016	0.003	0.002	0.14	0.02	0.05
I Y 96	0.011	0.001	0.025	-0.07	0.02	0.86
I Y 97	0.010	0.000	0.000	0.02	0.00	0.00
I NB 98F	0.000	0.001	0.011	-0.00	0.02	0.08

Table 33 cont'd

ZR100	0.013	0.003	0.003	0.12	0.05	0.11
ZR101	0.014	0.002	0.000	0.09	0.04	0.01
MO101	0.001	-0.000	0.011	-0.03	0.00	0.04
TC102F	0.000	0.000	0.012	-0.00	0.00	0.31
MO103	0.006	0.001	0.010	0.01	0.01	0.37
I132F	0.000	0.000	0.014	-0.01	0.00	0.02
SB133	0.011	0.000	0.006	0.21	0.01	0.11
TE134	0.023	0.001	0.005	0.11	0.00	0.13
I134F	0.003	0.001	0.025	-0.09	0.00	0.06
TE135	0.018	0.004	0.009	0.25	0.04	0.33
I135	0.008	0.000	0.012	-0.16	0.00	0.03
I136F	0.007	0.000	0.012	0.11	0.00	0.05
I137	0.012	0.002	0.007	0.15	0.00	0.25
XE137	0.007	0.001	0.012	-0.10	0.00	0.08
XE138	0.027	-0.000	0.011	-0.05	0.00	0.04
CS138F	0.001	0.001	0.023	-0.06	0.00	0.07
XE139	0.025	0.002	0.011	0.06	0.00	0.05
CS139	0.006	-0.000	0.014	-0.07	0.00	0.10
XE140	0.029	0.003	0.006	0.12	0.03	0.12
CS140	0.021	0.001	0.023	0.13	0.01	0.10
CS141	0.018	0.003	0.009	0.18	0.00	0.19
LA140	0.000	-0.000	0.018	-0.00	0.00	0.03
CS142	0.012	0.001	0.000	0.07	0.01	0.00
BA141	0.006	-0.000	0.011	-0.13	0.00	0.06
BA142	0.015	-0.000	0.008	0.04	0.00	0.06
LA142	0.001	-0.000	0.019	-0.02	0.00	0.04
BA143	0.016	0.004	0.007	0.07	0.04	0.23
BA144	0.016	0.002	0.004	0.11	0.03	0.14
LA144	0.005	0.001	0.011	-0.09	0.00	0.26
LA146	0.012	0.002	0.004	0.13	0.14	0.15

Table 34

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
USRAP 10/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07	S
TEMPS DE REFROIDISSEMENT	1.000E 02	S
PUISSANCE RESIDUELLE	5.906E 00	(MEV/S)
ERREUR DUE AUX RENDEMENTS	0.529	%
ERREUR DUE AUX PERIODES	0.347	%
ERREUR DUE AUX ENERGIES	1.197	%
ERREUR TOTALE	1.354	%

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A (%)		
	REND.	PERIODE	EBGM	REND	PER.	EBGM
BR 88	0.017	0.001	0.000	0.05	0.00	0.01
BR 89	0.012	0.000	0.000	0.13	0.00	0.00
KR 88	0.018	-0.001	0.013	0.06	0.00	0.02
RB 88	0.000	0.000	0.023	-0.02	0.00	0.06
KR 89	0.033	0.007	0.016	0.12	0.00	0.04
RB 89	0.002	-0.001	0.024	-0.09	0.00	0.04
KR 90	0.027	0.009	0.002	0.14	0.00	0.01
RB 90F	0.003	0.007	0.023	-0.12	0.12	0.06
KR 91	0.016	0.001	0.000	0.05	0.01	0.00
RB 91	0.015	0.012	0.011	0.13	0.12	0.04
RB 92	0.017	0.000	0.000	0.06	0.00	0.00
SR 91	0.002	0.001	0.011	-0.09	0.01	0.05
RB 93	0.016	0.000	0.000	0.07	0.00	0.00
SR 92	0.008	-0.000	0.014	0.02	0.00	0.03
Y 92	0.000	0.000	0.015	-0.01	0.00	0.09
SR 93	0.019	0.004	0.023	0.03	0.01	0.18
Y 93F	0.000	0.000	0.013	-0.01	0.00	0.08
SR 94	0.026	0.007	0.007	0.10	0.00	0.17
Y 94	0.002	0.000	0.029	-0.07	0.00	0.21
SR 95	0.031	0.008	0.003	0.10	0.09	0.06
Y 95	0.008	0.001	0.021	-0.09	0.00	0.10
SR 96	0.013	0.000	0.000	0.09	0.00	0.00
Y 96	0.010	0.012	0.023	0.03	0.19	0.79
Y 97	0.015	0.000	0.000	0.03	0.00	0.00
NB 97F	0.000	0.000	0.011	-0.02	0.00	0.04
ZR101	0.013	0.000	0.000	0.03	0.00	0.00
MO101	0.001	0.001	0.015	0.00	0.00	0.05
NB102	0.010	0.000	0.000	0.03	0.00	0.00
TC102F	0.000	0.000	0.016	-0.00	0.00	0.42
MO104	0.011	0.002	0.002	0.03	0.02	0.06
SB131	0.011	0.001	0.010	-0.02	0.00	0.21
SB132M	0.012	0.002	0.005	0.18	0.03	0.12
SB132F	0.010	0.003	0.006	0.09	0.05	0.14
TE132	0.010	0.000	0.003	-0.17	0.00	0.01
I132F	0.000	0.000	0.021	-0.01	0.00	0.03
SB133	0.014	0.003	0.005	0.22	0.05	0.10
TE133M	0.014	0.000	0.008	0.12	0.00	0.03
TE133F	0.010	0.000	0.011	-0.17	0.00	0.08
I133F	0.000	-0.000	0.010	-0.04	0.00	0.05
TE134	0.034	0.001	0.007	0.19	0.00	0.19
I134F	0.004	0.001	0.036	-0.13	0.00	0.10
TE135	0.013	0.002	0.000	0.07	0.02	0.01

Table 34 cont'd

I 1135	I	0.012	0.001	0.017	I -0.04	0.00	0.04 I
I I138	I	0.011	0.000	0.000	I 0.08	0.00	0.00 I
I XE137	I	0.008	0.004	0.014	I -0.05	0.00	0.09 I
I XE138	I	0.039	0.001	0.015	I 0.16	0.00	0.06 I
I CS138F	I	0.002	0.001	0.034	I -0.08	0.00	0.11 I
I XE139	I	0.024	0.006	0.003	I 0.04	0.01	0.02 I
I CS139	I	0.008	0.002	0.019	I -0.05	0.01	0.13 I
I XE140	I	0.023	0.003	0.000	I 0.06	0.03	0.00 I
I CS140	I	0.021	0.012	0.013	I 0.12	0.12	0.05 I
I CS141	I	0.017	0.003	0.001	I 0.09	0.00	0.02 I
I LA140	I	0.000	-0.000	0.026	I -0.00	0.00	0.05 I
I CS142	I	0.016	0.000	0.000	I 0.08	0.00	0.00 I
I BA141	I	0.008	0.000	0.015	I -0.06	0.00	0.08 I
I BA142	I	0.021	0.001	0.010	I 0.08	0.00	0.08 I
I LA142	I	0.001	-0.000	0.028	I -0.03	0.00	0.06 I
I BA143	I	0.015	0.001	0.000	I 0.02	0.01	0.00 I
I LA143	I	0.002	0.000	0.012	I -0.01	0.00	0.09 I
I BA144	I	0.011	0.002	0.000	I 0.05	0.02	0.00 I
I LA146	I	0.011	0.001	0.000	I 0.03	0.05	0.00 I
I PR146	I	0.000	0.000	0.012	I -0.00	0.00	0.05 I

Table 35

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
USRAP 10/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07 S
TEMPS DE REFROIDISSEMENT	1.000E 03 S
PUISSANCE RESIDUELLE	3.553E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS	0.560 %
ERREUR DUE AUX PERIODES	0.143 %
ERREUR DUE AUX ENERGIES	0.748 %
ERREUR TOTALE	0.946 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 88	0.026	0.000	0.000	0.05	0.00	0.00
KR 87	0.003	0.002	0.012	0.02	0.00	0.05
KR 88	0.028	0.001	0.020	0.12	0.00	0.04
RB 88	0.000	0.001	0.037	-0.03	0.00	0.09
KR 89	0.023	0.009	0.001	0.05	0.01	0.00
RB 89	0.002	0.012	0.023	-0.06	0.01	0.03
KR 91	0.016	0.000	0.000	0.02	0.00	0.00
RB 91	0.015	0.000	0.000	0.08	0.00	0.00
RB 92	0.028	0.000	0.0	0.09	0.00	0.0
SR 91	0.003	0.001	0.018	0.03	0.01	0.07
RB 93	0.013	0.000	0.0	0.06	0.00	0.0
SR 92	0.012	0.002	0.022	0.04	0.00	0.05

Table 35 cont'd

Y 92	0.000	0.001	0.025	-0.01	0.00	0.14
SR 93	0.016	0.014	0.008	0.02	0.03	0.07
Y 93F	0.000	0.001	0.021	-0.01	0.01	0.14
SR 94	0.020	0.002	0.000	0.05	0.00	0.00
Y 94	0.002	0.015	0.027	-0.03	0.01	0.20
SR 95	0.030	0.001	0.000	0.06	0.01	0.00
Y 95	0.009	0.013	0.012	-0.03	0.04	0.06
ZR 95	0.000	0.002	0.015	-0.01	0.00	0.02
NB 95F	0.000	0.000	0.013	-0.00	0.00	0.01
SR 97	0.011	0.000	0.0	0.03	0.00	0.0
Y 97	0.024	0.000	0.0	0.06	0.00	0.0
ZR 97	0.009	-0.000	0.014	0.04	0.00	0.08
NB 97F	0.000	0.000	0.018	-0.03	0.00	0.07
ZR101	0.013	0.000	0.0	0.03	0.00	0.0
MO101	0.001	0.011	0.012	0.00	0.00	0.04
MO102	0.002	0.011	0.001	0.01	0.02	0.07
TC102F	0.0	0.000	0.010	0.0	0.00	0.27
SB131	0.014	0.006	0.010	0.09	0.01	0.22
SB132M	0.011	0.002	0.001	0.09	0.04	0.01
SB132F	0.010	0.001	0.000	-0.03	0.01	0.00
TE132	0.017	0.001	0.004	-0.05	0.00	0.02
I132F	0.000	0.000	0.035	-0.01	0.00	0.06
SB133	0.010	0.002	0.000	-0.13	0.03	0.00
TE133M	0.021	0.003	0.012	0.34	0.00	0.05
TE133F	0.011	0.006	0.009	-0.20	0.01	0.06
I133F	0.001	-0.000	0.017	-0.05	0.00	0.08
TE134	0.053	0.006	0.009	0.35	0.01	0.24
I134F	0.006	0.005	0.057	-0.22	0.01	0.15
TE135	0.020	0.000	0.000	0.07	0.00	0.00
I135	0.020	0.002	0.028	0.05	0.00	0.06
I138	0.013	0.000	0.0	0.11	0.00	0.0
XE138	0.048	0.015	0.012	0.23	0.01	0.05
CS138F	0.002	0.008	0.049	-0.09	0.00	0.16
XE139	0.018	0.001	0.000	-0.02	0.00	0.00
CS139	0.007	0.012	0.009	-0.03	0.04	0.07
BA139	0.000	0.001	0.015	-0.01	0.00	0.11
XE140	0.024	0.000	0.000	0.02	0.00	0.00
CS140	0.024	0.000	0.000	0.07	0.00	0.00
CS141	0.019	0.000	0.000	0.07	0.00	0.00
LA140	0.000	-0.000	0.043	-0.00	0.00	0.08
CS142	0.022	0.000	0.0	0.11	0.00	0.0
BA141	0.010	0.009	0.014	0.01	0.00	0.07
LA141	0.000	0.000	0.016	-0.01	0.00	0.12
BA142	0.028	0.008	0.006	0.10	0.01	0.05
LA142	0.001	0.002	0.044	-0.03	0.00	0.09
BA143	0.018	0.000	0.000	0.02	0.00	0.00
CE143	0.000	-0.000	0.011	-0.00	0.00	0.05
BA144	0.012	-0.000	0.000	0.02	0.00	0.00
CE144	0.000	-0.012	0.001	0.00	0.00	0.01
PR144F	0.000	0.000	0.016	-0.00	0.00	0.11
LA146	0.014	0.000	0.000	0.03	0.01	0.00
PR146	0.000	0.003	0.017	-0.00	0.01	0.08

Table 36

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
USRAP 10/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07	S
TEMPS DE REFROIDISSEMENT	1.000E 04	S
PUISSANCE RESIDUELLE	1.761E 00	(MEV/S)
ERREUR DUE AUX RENDEMENTS	0.529	%
ERREUR DUE AUX PERIODES	0.172	%
ERREUR DUE AUX ENERGIES	0.708	%
ERREUR TOTALE	0.900	%

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 88	0.029	0.000	0.0	0.05	0.00	0.0
KR 88	0.031	0.038	0.021	0.14	0.03	0.04
RB 88	0.000	0.005	0.041	-0.00	0.00	0.10
KR 89	0.010	0.000	0.000	-0.04	0.00	0.00
SR 89	0.000	0.000	0.014	-0.00	0.00	0.11
KR 91	0.028	0.000	0.0	0.03	0.00	0.0
KR 92	0.013	0.000	0.0	0.02	0.00	0.0
RB 91	0.027	0.000	0.0	0.14	0.00	0.0
RB 92	0.039	0.000	0.0	0.13	0.00	0.0
SR 91	0.006	0.010	0.031	0.06	0.11	0.12
Y 91M	0.000	0.001	0.011	-0.00	0.00	0.16
Y 91F	0.000	0.001	0.018	-0.00	0.00	0.14
RB 93	0.016	0.000	0.0	0.07	0.00	0.0
SR 92	0.017	0.023	0.023	0.06	0.01	0.05
Y 92	0.000	0.008	0.044	-0.02	0.00	0.25
SR 93	0.019	0.000	0.000	0.02	0.00	0.00
Y 93F	0.000	0.008	0.035	-0.00	0.08	0.23
SR 95	0.042	0.000	0.0	0.09	0.00	0.0
Y 95	0.012	0.000	0.000	0.03	0.00	0.00
ZR 95	0.001	0.004	0.029	0.00	0.00	0.05
NB 95F	0.000	0.001	0.027	-0.00	0.00	0.02
SR 97	0.020	0.000	0.0	0.05	0.00	0.0
Y 97	0.045	0.000	0.0	0.12	0.00	0.0
ZR 97	0.017	0.007	0.026	0.08	0.01	0.15
NB 97M	0.000	0.000	0.018	-0.00	0.00	0.00
NB 97F	0.000	0.001	0.034	-0.03	0.00	0.13
ZR 99	0.015	0.000	0.0	0.03	0.00	0.0
MO 99	0.000	0.001	0.017	0.00	0.00	0.10
SB132M	0.020	0.000	0.000	0.06	0.00	0.00
SB132F	0.020	0.000	0.000	0.06	0.00	0.00
TE132	0.034	0.002	0.008	0.07	0.00	0.04
I132F	0.000	0.001	0.070	-0.02	0.01	0.11
SB133	0.011	0.000	0.000	-0.12	0.00	0.00
TE133M	0.019	0.009	0.003	0.25	0.01	0.01
TE133F	0.014	0.000	0.000	-0.16	0.00	0.00
I133F	0.001	0.002	0.033	-0.03	0.00	0.14
TE134	0.030	0.027	0.001	0.26	0.05	0.04
I134F	0.002	0.034	0.029	-0.13	0.07	0.08
TE135	0.033	0.000	0.0	0.12	0.00	0.0
I135	0.032	0.017	0.043	0.08	0.03	0.10
XE135F	0.000	0.001	0.017	-0.01	0.00	0.10
CS138F	0.000	0.018	0.005	-0.01	0.01	0.02

Table 36 cont'd

I BA139	I	0.000	0.011	0.008	I	-0.00	0.00	0.06 I
I XE140	I	0.049	0.000	0.0	I	0.04	0.00	0.0 I
I CS140	I	0.049	0.000	0.0	I	0.15	0.00	0.0 I
I CS141	I	0.016	0.000	0.0	I	0.05	0.00	0.0 I
I BA140	I	0.008	-0.002	0.015	I	0.05	0.00	0.10 I
I LA140	I	0.000	-0.000	0.087	I	-0.01	0.00	0.17 I
I CS142	I	0.013	0.000	0.0	I	0.07	0.00	0.0 I
I LA141	I	0.000	0.009	0.021	I	-0.00	0.01	0.16 I
I BA142	I	0.017	0.004	0.000	I	0.06	0.00	0.00 I
I LA142	I	0.001	0.032	0.029	I	-0.01	0.02	0.06 I
I BA143	I	0.022	0.000	0.0	I	0.03	0.00	0.0 I
I CE143	I	0.000	0.001	0.021	I	-0.00	0.00	0.10 I
I BA144	I	0.023	-0.000	0.0	I	0.05	0.00	0.0 I
I CE144	I	0.001	-0.024	0.002	I	0.00	0.01	0.03 I
I PR144F	I	0.000	0.000	0.032	I	-0.00	0.00	0.22 I
I PR145	I	0.000	0.004	0.010	I	-0.00	0.01	0.08 I

Table 37

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
USRAP 10/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07 S
TEMPS DE REFROIDISSEMENT	1.000E 05 S
PUISSANCE RESIDUELLE	6.311E-01 (MEV/S)
ERREUR DUE AUX RENDEMENTS	0.523 %
ERREUR DUE AUX PERIODES	0.424 %
ERREUR DUE AUX ENERGIES	0.918 %
ERREUR TOTALE	1.138 %

I	I				I			I
I	I				I			I
I NUCLEIDE	I	SENSIBILITE DE PUISSANCE			I	ERREUR SUR PUISSANCE		I
I	I	RESIDUELLE A			I	RESIDUELLE DUE A		I
I	I	REND.	PERIODE	EBGM	I	REND	PER.	EBGM I
I	I				I	(%)	(%)	(%) I
I KR 89	I	0.022	0.000	0.0	I	0.07	0.00	0.0 I
I SR 89	I	0.000	0.001	0.030	I	0.00	0.00	0.23 I
I KR 91	I	0.025	0.000	0.0	I	0.03	0.00	0.0 I
I RB 91	I	0.024	0.000	0.0	I	0.12	0.00	0.0 I
I SR 91	I	0.005	0.030	0.010	I	0.05	0.32	0.04 I
I Y 91F	I	0.000	0.002	0.038	I	-0.00	0.00	0.30 I
I Y 93F	I	0.000	0.026	0.012	I	0.00	0.26	0.08 I
I RB 95	I	0.012	0.000	0.0	I	-0.14	0.00	0.0 I
I SR 95	I	0.088	0.000	0.0	I	0.19	0.00	0.0 I
I Y 95	I	0.025	0.000	0.0	I	0.05	0.00	0.0 I
I ZR 95	I	0.001	0.010	0.061	I	0.00	0.00	0.10 I
I NB 95F	I	0.000	0.001	0.057	I	-0.00	0.00	0.04 I
I SR 97	I	0.016	0.000	0.0	I	0.04	0.00	0.0 I

Table 37 cont'd

Y 97	0.034	0.000	0.0	0.09	0.00	0.0
ZR 97	0.013	0.064	0.019	0.06	0.07	0.11
NB 97M	0.000	0.000	0.013	-0.00	0.00	0.00
NB 97F	0.000	0.002	0.025	-0.00	0.00	0.09
ZR 99	0.024	0.000	0.0	0.05	0.00	0.0
MO 99	0.000	0.011	0.028	0.00	0.00	0.16
NB103	0.014	0.000	0.0	0.04	0.00	0.0
RU103	0.000	0.001	0.020	0.00	0.00	0.03
SB131	0.011	0.000	0.000	-0.11	0.00	0.00
I131	0.000	0.001	0.019	-0.00	0.00	0.07
SN132	0.014	0.000	0.0	0.08	0.00	0.0
SB132M	0.034	0.000	0.0	0.11	0.00	0.0
SB132F	0.034	0.000	0.0	0.10	0.00	0.0
TE132	0.058	0.036	0.014	0.12	0.03	0.07
I132F	0.000	0.004	0.119	-0.00	0.03	0.19
SB133	0.012	0.000	0.0	-0.01	0.00	0.0
TE133M	0.016	0.001	0.000	0.08	0.00	0.00
TE133F	0.015	0.000	0.000	-0.03	0.00	0.00
I133F	0.002	0.027	0.029	-0.01	0.01	0.13
XE133F	0.000	0.001	0.013	-0.00	0.00	0.09
TE135	0.011	0.000	0.0	0.04	0.00	0.0
I135	0.010	0.031	0.006	0.03	0.05	0.01
XE135F	0.000	0.012	0.010	-0.00	0.00	0.06
XE140	0.102	0.000	0.0	0.08	0.00	0.0
CS140	0.102	0.000	0.0	0.32	0.00	0.0
BA140	0.017	-0.000	0.030	0.10	0.00	0.20
LA140	0.000	0.002	0.181	-0.02	0.00	0.35
CE141	0.000	0.001	0.015	-0.00	0.00	0.31
BA143	0.033	0.000	0.0	0.05	0.00	0.0
CE143	0.000	0.016	0.025	0.00	0.02	0.12
PR143	0.000	-0.000	0.020	-0.00	0.00	0.19
BA144	0.049	-0.000	0.0	0.10	0.00	0.0
LA144	0.021	0.000	0.0	0.04	0.00	0.0
CE144	0.001	-0.051	0.005	0.01	0.02	0.06
PR144F	0.000	0.000	0.067	-0.00	0.00	0.47
ND147	0.000	0.000	0.010	0.00	0.00	0.06

Table 38

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
USRAP 10/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07	S
TEMPS DE REFROIDISSEMENT	1.000E 06	S
PUISSANCE RESIDUELLE	4.096E-01	(MEV/S)
ERREUR DUE AUX RENDEMENTS	0.585	%
ERREUR DUE AUX PERIODES	0.143	%
ERREUR DUE AUX ENERGIES	1.394	%
ERREUR TOTALE	1.518	%

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 89	0.012	0.000	0.0	0.03	0.00	0.0
KR 89	0.038	0.000	0.0	0.12	0.00	0.0
SR 89	0.000	0.011	0.053	0.00	0.01	0.40
KR 91	0.031	0.000	0.0	0.04	0.00	0.0
RB 91	0.031	0.000	0.0	0.15	0.00	0.0
Y 91F	0.000	0.013	0.067	0.00	0.01	0.54
RB 95	0.022	0.000	0.0	-0.27	0.00	0.0
SR 95	0.167	0.000	0.0	0.35	0.00	0.0
Y 95	0.048	0.000	0.0	0.10	0.00	0.0
ZR 95	0.002	0.035	0.111	0.01	0.00	0.18
NB 95F	0.000	0.004	0.113	-0.00	0.00	0.08
MO 99	0.000	0.016	0.004	0.00	0.00	0.02
NB103	0.023	0.000	0.0	0.06	0.00	0.0
MO103	0.015	0.000	0.0	0.04	0.00	0.0
RU103	0.000	0.009	0.034	0.00	0.00	0.05
MO106	0.016	-0.000	0.0	0.10	0.00	0.0
RU106	0.000	-0.017	0.000	0.00	0.01	0.00
RH106F	0.0	0.000	0.018	0.0	0.00	0.10
I131	0.000	0.014	0.015	-0.00	0.00	0.05
TE132	0.015	0.076	0.003	0.03	0.06	0.01
I132F	0.0	0.001	0.024	0.0	0.01	0.04
XE140	0.123	0.000	0.0	0.10	0.00	0.0
CS140	0.123	0.000	0.0	0.38	0.00	0.0
CS141	0.014	0.000	0.0	0.05	0.00	0.0
BA140	0.020	0.130	0.033	0.12	0.11	0.22
LA140	0.000	0.031	0.223	-0.01	0.05	0.42
CE141	0.000	0.007	0.025	-0.00	0.00	0.50
BA143	0.018	0.000	0.0	0.03	0.00	0.0
PR143	0.000	0.012	0.024	-0.00	0.01	0.24
BA144	0.097	-0.000	0.0	0.20	0.00	0.0
LA144	0.041	0.000	0.0	0.08	0.00	0.0
CE144	0.003	-0.098	0.010	0.01	0.03	0.12
PR144F	0.0	0.000	0.133	0.0	0.00	0.93
ND147	0.000	0.008	0.010	0.00	0.01	0.06

Table 39

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
USRAP 10/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07	S
TEMPS DE REFROIDISSEMENT	1.000E 07	S
PUISSANCE RESIDUELLE	1.198E-01	(MEV/S)
ERREUR DUE AUX RENDEMENTS	0.753	%
ERREUR DUE AUX PERIODES	0.107	%
ERREUR DUE AUX ENERGIES	2.562	%
ERREUR TOTALE	2.673	%

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 89	0.010	0.000	0.0	0.02	0.00	0.0
KR 89	0.031	0.000	0.0	0.10	0.00	0.0
SR 89	0.000	0.071	0.040	0.00	0.03	0.31
KR 90	0.023	-0.000	0.0	0.05	0.00	0.0
Y 90F	0.000	-0.000	0.025	-0.00	0.00	0.18
KR 91	0.031	0.000	0.0	0.04	0.00	0.0
RB 91	0.030	0.000	0.0	0.15	0.00	0.0
Y 91F	0.000	0.099	0.063	0.00	0.05	0.51
RB 95	0.031	0.000	0.0	-0.37	0.00	0.0
SR 95	0.231	0.000	0.0	0.49	0.00	0.0
Y 95	0.067	0.000	0.0	0.14	0.00	0.0
ZR 95	0.003	0.334	0.115	0.01	0.03	0.19
NB 95F	0.000	0.093	0.189	-0.00	0.01	0.14
NB103	0.013	0.000	0.0	0.04	0.00	0.0
RU103	0.000	0.041	0.017	0.00	0.02	0.03
MO106	0.043	-0.000	0.0	0.28	0.00	0.0
RU106	0.000	-0.042	0.000	0.00	0.02	0.00
RH106F	0.0	0.000	0.049	0.0	0.00	0.29
XE137	0.014	-0.000	0.0	0.04	0.00	0.0
BA137M	0.0	0.000	0.017	0.0	0.00	0.00
BA140	0.001	0.035	0.000	0.00	0.03	0.00
CE141	0.000	0.023	0.008	0.00	0.01	0.17
CS144	0.024	0.000	0.0	-0.13	0.00	0.0
BA144	0.252	-0.000	0.0	0.51	0.00	0.0
LA144	0.106	0.000	0.0	0.22	0.00	0.0
CE144	0.007	-0.187	0.027	0.03	0.07	0.30
PR144F	0.0	0.000	0.347	0.0	0.00	2.43

Table 40

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
U5RAP 10/03/76 TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION	7.776E 07	S
TEMPS DE REFROIDISSEMENT	1.000E 08	S
PUISSANCE RESIDUELLE	1.155E-02	(MEV/S)
ERREUR DUE AUX RENDEMENTS	0.963	%
ERREUR DUE AUX PERIODES	0.799	%
ERREUR DUE AUX ENERGIES	2.632	%
ERREUR TOTALE	2.914	%

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
BR 90	0.026	-0.000	0.0	-0.42	0.00	0.0
KR 90	0.212	0.000	0.0	0.71	0.00	0.0
RB 90M	0.031	0.000	0.0	0.21	0.00	0.0
RB 90F	0.034	0.000	0.0	0.23	0.00	0.0
SR 90	0.003	-0.268	0.044	0.02	0.75	0.47
Y 90F	0.0	0.000	0.239	0.0	0.00	1.70
MO106	0.062	-0.000	0.0	0.41	0.00	0.0
TC106	0.012	0.000	0.0	0.16	0.00	0.0
RU106	0.000	0.115	0.000	0.01	0.06	0.01
RH106F	0.0	0.000	0.067	0.0	0.00	0.39
TE137	0.011	0.000	0.0	-0.09	0.00	0.0
I137	0.087	0.000	0.0	-0.32	0.00	0.0
XE137	0.125	0.000	0.0	0.41	0.00	0.0
CS137	0.006	-0.203	0.048	0.08	0.07	0.51
BA137M	0.0	0.0	0.165	0.0	0.0	0.00
CS144	0.022	0.000	0.0	-0.12	0.00	0.0
BA144	0.226	-0.000	0.0	0.46	0.00	0.0
LA144	0.095	0.000	0.0	0.19	0.00	0.0
CE144	0.006	0.730	0.020	0.02	0.26	0.22
PR144F	0.0	0.000	0.257	0.0	0.00	1.80
CE147	0.016	0.000	0.0	0.05	0.00	0.0
PM147	0.000	0.005	0.025	0.00	0.00	0.33

Table 41

ERREUR SUR LA PUISSANCE RESIDUELLE DUE AUX PRODUITS DE FISSION
***** PUSRAP 11/00/76 *****
TAUX DE FISSION CONSTANT (1 FIS./S.)

TEMPS D'IRRADIATION 7.776E 07 S

TEMPS REF. (S)	PUISSANCE RESIDUELLE (MEV/S)	ERREUR DUE AU RENO. (%)	ERREUR DUE A LA PER. (%)	ERREUR DUE AUX ENERGIES (%)	ERREUR TOTALE
1.0E 00	8.5108E 00	1.216	0.162	1.891	2.254
2.0E 00	8.2487E 00	1.197	0.171	1.881	2.236
5.0E 00	7.7594E 00	1.159	0.184	1.858	2.198
1.0E 01	7.2838E 00	1.111	0.216	1.830	2.152
2.0E 01	6.7288E 00	1.050	0.270	1.778	2.082
5.0E 01	5.8852E 00	0.962	0.298	1.655	1.937
1.0E 02	5.1978E 00	0.907	0.309	1.532	1.807
2.0E 02	4.5410E 00	0.883	0.294	1.426	1.703
5.0E 02	3.7484E 00	0.903	0.213	1.261	1.565
1.0E 03	3.1620E 00	0.944	0.143	1.024	1.400
2.0E 03	2.5756E 00	0.984	0.107	0.748	1.240
5.0E 03	1.9178E 00	0.941	0.108	0.620	1.132
1.0E 04	1.5505E 00	0.866	0.132	0.659	1.097
2.0E 04	1.2755E 00	0.859	0.175	0.712	1.130
5.0E 04	9.8862E-01	0.898	0.258	0.785	1.221
1.0E 05	8.1023E-01	0.919	0.239	0.863	1.283
2.0E 05	6.6390E-01	0.923	0.148	0.970	1.347
5.0E 05	5.0196E-01	0.960	0.135	1.157	1.509
1.0E 06	3.9214E-01	1.082	0.161	1.343	1.732
2.0E 06	2.9394E-01	1.320	0.198	1.598	2.082
5.0E 06	1.9050E-01	1.786	0.192	2.108	2.770
1.0E 07	1.2974E-01	2.252	0.190	2.653	3.485
2.0E 07	8.0773E-02	2.803	0.207	3.298	4.333
5.0E 07	3.4953E-02	3.263	0.273	3.729	4.963
1.0E 08	1.3974E-02	2.729	0.511	2.884	4.003

Table 42

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9RAP 11/00/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 00 S
PUISSANCE RESIDUELLE 8.511E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS 1.216 %
ERREUR DUE AUX PERIODES 0.162 %
ERREUR DUE AUX ENERGIES 1.891 %
ERREUR TOTALE 2.254 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A (%)		
	REND.	PERIODE	EBGM	REND	PER.	EBGM
KR 89	0.011	0.000	0.005	0.14	0.00	0.01
RB 92	0.010	0.001	0.006	0.17	0.00	0.04
SR 93	0.012	0.000	0.010	-0.17	0.00	0.08
SR 94	0.017	-0.000	0.007	0.08	0.00	0.18
Y 94	0.003	-0.000	0.014	-0.07	0.00	0.10
SR 95	0.020	0.000	0.010	0.30	0.00	0.22
Y 95	0.010	-0.000	0.011	-0.17	0.00	0.05
Y 96	0.012	0.000	0.017	0.12	0.00	0.58
Y 97	0.012	0.003	0.005	0.13	0.02	0.04
ZR 98	0.014	0.001	0.007	-0.03	0.01	0.06
NB 98F	0.000	0.000	0.011	-0.02	0.00	0.09
ZR 99	0.020	0.003	0.009	0.12	0.02	0.09
ZR100	0.019	0.000	0.005	0.21	0.00	0.19
NB100F	0.003	0.002	0.016	-0.09	0.03	0.46
ZR101	0.011	0.001	0.003	0.17	0.02	0.08
NB101	0.014	0.000	0.009	0.20	0.00	0.17
MO101	0.008	-0.000	0.014	-0.17	0.00	0.04
NB102	0.022	0.003	0.014	0.25	0.05	0.48
TC102F	0.000	0.000	0.017	-0.01	0.00	0.45
NB103	0.015	0.002	0.003	0.16	0.07	0.11
MO103	0.023	-0.000	0.023	-0.06	0.00	0.81
MO104	0.025	-0.000	0.009	0.12	0.00	0.31
TC104	0.003	-0.000	0.018	-0.07	0.00	0.45
MO105	0.025	0.000	0.012	0.39	0.00	0.43
TC105	0.008	-0.000	0.013	-0.18	0.00	0.34
MO106	0.012	0.000	0.003	0.16	0.00	0.10
TC106	0.013	0.001	0.014	0.16	0.01	0.43
TC107	0.011	0.000	0.008	0.21	0.00	0.28
SB131	0.010	0.000	0.007	0.02	0.00	0.16
TE132	0.012	0.000	0.002	-0.10	0.00	0.01
I132F	0.000	-0.000	0.017	-0.02	0.00	0.03
TE133M	0.011	0.000	0.006	0.10	0.00	0.03
TE134	0.018	0.001	0.004	0.18	0.00	0.10
I134F	0.005	0.001	0.024	-0.07	0.00	0.06
TE135	0.014	0.000	0.008	0.45	0.00	0.29
I135	0.011	0.000	0.012	-0.26	0.00	0.03
XE137	0.010	0.000	0.012	-0.07	0.00	0.08
XE138	0.027	-0.000	0.010	0.12	0.00	0.04
CS138F	0.003	0.001	0.022	-0.13	0.00	0.07
XE139	0.022	0.000	0.010	0.27	0.00	0.05
CS139	0.007	-0.000	0.013	-0.13	0.00	0.09
XE140	0.018	0.000	0.005	0.34	0.00	0.10

Table 42 cont'd

I CS140	I	0.024	-0.000	0.019	I	0.22	0.00	0.08 I
I CS141	I	0.016	0.000	0.008	I	0.23	0.00	0.18 I
I LA140	I	0.000	-0.000	0.016	I	-0.00	0.00	0.03 I
I BA141	I	0.008	-0.000	0.010	I	-0.18	0.00	0.05 I
I BA142	I	0.017	-0.000	0.007	I	-0.10	0.00	0.05 I
I LA142	I	0.001	-0.000	0.017	I	-0.05	0.00	0.03 I
I BA143	I	0.015	0.000	0.008	I	0.11	0.00	0.26 I
I BA144	I	0.011	0.000	0.004	I	0.12	0.00	0.12 I
I LA146	I	0.010	0.000	0.005	I	0.11	0.01	0.17 I

Table 43

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9RAP 11/00/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 01 S
PUISSANCE RESIDUELLE 7.284E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS 1.111 %
ERREUR DUE AUX PERIODES 0.216 %
ERREUR DUE AUX ENERGIES 1.830 %
ERREUR TOTALE 2.152 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
I KR 89	0.012	0.000	0.006	0.16	0.00	0.02
I RB 91	0.011	0.001	0.009	0.16	0.01	0.03
I SR 93	0.014	0.000	0.012	-0.10	0.00	0.09
I SR 94	0.019	0.001	0.008	0.10	0.00	0.19
I Y 94	0.004	-0.000	0.016	-0.07	0.00	0.12
I SR 95	0.021	0.003	0.009	0.30	0.03	0.21
I Y 95	0.011	-0.000	0.013	-0.18	0.00	0.06
I Y 96	0.014	0.001	0.019	0.16	0.01	0.65
I ZR 98	0.014	0.004	0.007	0.05	0.04	0.06
I NB 98F	0.000	0.001	0.011	-0.01	0.02	0.09
I ZR 99	0.012	0.004	0.001	-0.05	0.03	0.01
I ZR100	0.011	0.007	0.003	0.15	0.04	0.09
I NB101	0.012	0.004	0.005	0.14	0.02	0.08
I MO101	0.009	-0.000	0.016	-0.12	0.00	0.05
I NB102	0.014	0.005	0.002	0.10	0.07	0.06
I TC102F	0.000	0.000	0.020	-0.00	0.00	0.53
I NB103	0.013	0.000	0.000	0.08	0.01	0.00
I MO103	0.025	0.002	0.024	0.10	0.02	0.85
I MO104	0.029	0.001	0.009	0.14	0.01	0.33

Table 43 cont'd

I TC104	I	0.004	-0.000	0.021	I -0.08	0.00	0.52	I
I MO105	I	0.027	0.002	0.012	I 0.42	0.02	0.43	I
I TC105	I	0.009	0.000	0.015	I -0.19	0.00	0.39	I
I MO106	I	0.012	0.001	0.002	I 0.16	0.03	0.06	I
I TC106	I	0.013	0.002	0.015	I 0.16	0.02	0.44	I
I TC107	I	0.011	0.002	0.007	I 0.21	0.00	0.25	I
I SB131	I	0.012	0.000	0.009	I 0.03	0.00	0.19	I
I TE132	I	0.014	0.000	0.002	I -0.12	0.00	0.01	I
I I132F	I	0.000	-0.000	0.020	I -0.02	0.00	0.03	I
I TE133M	I	0.012	0.000	0.008	I 0.12	0.00	0.03	I
I TE134	I	0.021	0.001	0.004	I 0.21	0.00	0.11	I
I I134F	I	0.006	0.001	0.028	I -0.08	0.00	0.07	I
I TE135	I	0.013	0.003	0.007	I 0.39	0.03	0.24	I
I I135	I	0.013	0.000	0.014	I -0.26	0.00	0.03	I
I XE137	I	0.011	0.001	0.014	I -0.07	0.00	0.09	I
I XE138	I	0.031	0.000	0.011	I 0.21	0.00	0.04	I
I CS138F	I	0.004	0.001	0.026	I -0.14	0.00	0.08	I
I XE139	I	0.024	0.002	0.010	I 0.29	0.00	0.05	I
I CS139	I	0.008	-0.000	0.015	I -0.14	0.00	0.11	I
I XE140	I	0.018	0.002	0.004	I 0.33	0.02	0.07	I
I CS140	I	0.026	0.001	0.021	I 0.25	0.01	0.09	I
I CS141	I	0.017	0.002	0.008	I 0.24	0.00	0.17	I
I LA140	I	0.000	-0.000	0.018	I -0.00	0.00	0.03	I
I BA141	I	0.009	-0.000	0.012	I -0.17	0.00	0.06	I
I BA142	I	0.019	-0.000	0.008	I 0.08	0.00	0.06	I
I LA142	I	0.001	-0.000	0.019	I -0.05	0.00	0.04	I
I BA143	I	0.014	0.003	0.005	I 0.11	0.03	0.18	I
I BA144	I	0.011	0.002	0.003	I 0.12	0.02	0.09	I

Table 44

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9RAP 11/00/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 02 S
PUISSANCE RESIDUELLE 5.198E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.907 %
ERREUR DUE AUX PERIODES 0.309 %
ERREUR DUE AUX ENERGIES 1.532 %
ERREUR TOTALE 1.807 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
KR 88	0.012	-0.000	0.005	0.06	0.00	0.01
KR 89	0.015	0.002	0.006	0.19	0.00	0.02
RB 89	0.003	-0.000	0.011	-0.13	0.00	0.02
SR 93	0.018	0.002	0.014	0.09	0.01	0.11
SR 94	0.021	0.004	0.004	0.09	0.00	0.11
Y 94	0.005	0.000	0.022	-0.06	0.00	0.16
SR 95	0.017	0.004	0.001	0.16	0.04	0.03
Y 95	0.015	0.001	0.016	-0.07	0.00	0.08
Y 96	0.012	0.009	0.016	0.15	0.14	0.57
Y 97	0.012	0.000	0.000	0.03	0.00	0.00
ZR 97	0.013	-0.000	0.009	0.05	0.00	0.05
NB 97F	0.000	-0.000	0.011	-0.01	0.00	0.04
NB101	0.012	0.000	0.000	0.06	0.00	0.00
MO101	0.012	0.001	0.020	0.05	0.00	0.07
NB102	0.015	0.000	0.000	0.07	0.00	0.00
MO102	0.012	0.002	0.004	0.04	0.00	0.16
TC102F	0.000	0.000	0.025	-0.00	0.00	0.67
MO103	0.018	0.015	0.012	0.07	0.15	0.41
MO104	0.031	0.006	0.005	0.13	0.04	0.16
TC104	0.005	0.000	0.029	-0.07	0.00	0.71
MO105	0.024	0.007	0.004	0.32	0.07	0.13
TC105	0.012	0.002	0.019	-0.08	0.01	0.50
RU105	0.000	0.000	0.010	-0.04	0.00	0.04
RU106	0.002	-0.012	0.000	-0.05	0.01	0.00
RH106F	0.000	0.000	0.011	-0.00	0.00	0.07
SB131	0.017	0.001	0.012	0.10	0.00	0.25
SB132M	0.011	0.002	0.005	0.19	0.03	0.11
TE132	0.020	0.001	0.003	-0.13	0.00	0.02
I132F	0.001	-0.000	0.028	-0.03	0.00	0.05
TE133M	0.017	0.001	0.010	0.21	0.00	0.04
TE133F	0.012	0.000	0.010	-0.17	0.00	0.07
I133F	0.001	-0.000	0.012	-0.10	0.00	0.05
TE134	0.030	0.001	0.006	0.30	0.00	0.15
I134F	0.008	0.002	0.038	-0.11	0.00	0.10
I135	0.018	0.001	0.019	0.06	0.00	0.04
XE137	0.012	0.005	0.015	0.01	0.00	0.10
XE138	0.042	0.001	0.015	0.35	0.00	0.06
CS138F	0.005	0.002	0.036	-0.18	0.00	0.11
XE139	0.022	0.006	0.003	0.22	0.01	0.01
CS139	0.011	0.002	0.019	0.01	0.01	0.14
XE140	0.014	0.002	0.000	0.17	0.02	0.00
CS140	0.025	0.011	0.011	0.21	0.11	0.05

Table 44 cont'd

I CS141	I	0.015	0.003	0.001	I	0.12	0.00	0.02 I
I LA140	I	0.000	-0.000	0.026	I	-0.00	0.00	0.05 I
I CS142	I	0.011	0.000	0.000	I	0.07	0.00	0.00 I
I BA141	I	0.013	0.001	0.016	I	-0.04	0.00	0.08 I
I LA141	I	0.000	-0.000	0.010	I	-0.01	0.00	0.08 I
I BA142	I	0.026	0.001	0.010	I	0.12	0.00	0.08 I
I LA142	I	0.002	0.000	0.027	I	-0.07	0.00	0.06 I
I BA143	I	0.013	0.001	0.000	I	0.03	0.01	0.00 I
I PR146	I	0.000	0.000	0.012	I	-0.01	0.00	0.05 I

Table 45

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9RAP 11/00/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 03 S
PUISSANCE RESIDUELLE 3.162E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.944 %
ERREUR DUE AUX PERIODES 0.143 %
ERREUR DUE AUX ENERGIES 1.024 %
ERREUR TOTALE 1.400 %

I NUCLEIDE	I SENSIBILITE DE PUISSANCE RESIDUELLE A			I ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	I REND.	PERIODE	EBGM	I REND (%)	PER. (%)	EBGM (%)
I KR 88	I 0.018	0.000	0.008	I 0.12	0.00	0.01 I
I RB 88	I 0.001	0.001	0.015	I -0.07	0.00	0.04 I
I KR 89	I 0.010	0.003	0.000	I 0.09	0.00	0.00 I
I RB 92	I 0.011	0.000	0.0	I 0.06	0.00	0.0 I
I SR 92	I 0.016	0.001	0.012	I 0.06	0.00	0.03 I
I Y 92	I 0.000	0.000	0.014	I -0.02	0.00	0.08 I
I SR 93	I 0.015	0.009	0.005	I 0.06	0.02	0.04 I
I Y 93F	I 0.001	0.001	0.014	I -0.03	0.01	0.09 I
I SR 94	I 0.015	0.001	0.000	I 0.05	0.00	0.00 I
I Y 94	I 0.005	0.012	0.021	I 0.02	0.01	0.15 I
I SR 95	I 0.017	0.000	0.000	I 0.10	0.00	0.00 I
I Y 95	I 0.016	0.010	0.009	I 0.05	0.03	0.05 I
I ZR 95	I 0.002	0.002	0.012	I -0.07	0.00	0.02 I
I NB 95F	I 0.000	0.000	0.011	I -0.00	0.00	0.01 I
I Y 97	I 0.020	0.000	0.0	I 0.05	0.00	0.0 I
I ZR 97	I 0.022	-0.000	0.014	I 0.08	0.00	0.08 I
I NB 97F	I 0.000	0.000	0.018	I -0.01	0.00	0.07 I
I MO 99	I 0.000	0.000	0.010	I 0.00	0.00	0.06 I
I NB101	I 0.011	0.000	0.0	I 0.06	0.00	0.0 I
I MO101	I 0.011	0.015	0.016	I 0.05	0.00	0.05 I

Table 45 cont'd

TC101		0.000	0.002	0.011		-0.00	0.00	0.07
MO102		0.008	0.018	0.002		0.03	0.04	0.10
TC102F		0.0	0.000	0.016		0.0	0.00	0.42
RU103		0.000	0.000	0.011		0.00	0.00	0.02
MO104		0.023	0.001	0.000		0.08	0.01	0.00
TC104		0.004	0.016	0.026		-0.03	0.03	0.65
MO105		0.018	0.001	0.000		0.20	0.01	0.00
TC105		0.011	0.012	0.008		0.11	0.06	0.20
RU105		0.001	0.000	0.016		-0.03	0.00	0.06
TC106		0.010	0.000	0.000		0.09	0.00	0.00
RU106		0.003	-0.019	0.000		0.05	0.01	0.00
RH106F		0.000	0.000	0.019		-0.00	0.00	0.11
SB131		0.021	0.007	0.012		0.14	0.02	0.26
TE131F		0.002	0.002	0.011		-0.14	0.00	0.05
SB132M		0.010	0.002	0.001		0.11	0.03	0.01
TE132		0.032	0.001	0.005		0.08	0.00	0.03
I132F		0.001	0.000	0.046		-0.02	0.00	0.07
TE133M		0.025	0.004	0.014		0.44	0.00	0.05
TE133F		0.014	0.005	0.008		-0.22	0.01	0.06
I133F		0.002	-0.000	0.020		-0.13	0.00	0.09
TE134		0.045	0.005	0.007		0.50	0.01	0.19
I134M		0.012	0.001	0.000		-0.13	0.00	0.00
I134F		0.011	0.008	0.058		-0.20	0.02	0.15
TE135		0.014	0.000	0.000		0.08	0.00	0.00
I135		0.029	0.002	0.030		0.16	0.00	0.07
XE135F		0.001	0.000	0.011		-0.03	0.00	0.07
XE138		0.052	0.014	0.011		0.47	0.01	0.04
CS138F		0.006	0.010	0.051		-0.22	0.00	0.16
XE139		0.017	0.001	0.000		0.13	0.00	0.00
CS139		0.009	0.012	0.009		0.09	0.04	0.07
BA139		0.001	0.001	0.015		-0.03	0.00	0.11
XE140		0.015	0.000	0.000		0.03	0.00	0.00
CS140		0.028	0.000	0.000		0.06	0.00	0.00
CS141		0.017	0.000	0.000		0.11	0.00	0.00
LA140		0.000	-0.000	0.042		-0.00	0.00	0.08
CS142		0.014	0.000	0.0		0.08	0.00	0.0
BA141		0.015	0.009	0.014		0.08	0.00	0.07
LA141		0.000	0.000	0.016		-0.02	0.00	0.13
BA142		0.034	0.008	0.006		0.15	0.01	0.05
LA142		0.003	0.003	0.042		-0.08	0.00	0.09
BA143		0.015	0.000	0.000		0.03	0.00	0.00
PR144F		0.000	0.000	0.012		-0.00	0.00	0.08
LA146		0.011	0.000	0.000		0.03	0.00	0.00
PR146		0.000	0.003	0.016		-0.01	0.01	0.07

Table 46

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9RAP 11/00/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 04 S
PUISSANCE RESIDUELLE 1.551E 00 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.866 %
ERREUR DUE AUX PERIODES 0.132 %
ERREUR DUE AUX ENERGIES 0.659 %
ERREUR TOTALE 1.097 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
KR 88	0.020	0.015	0.008	0.15	0.01	0.02
RB 88	0.000	0.002	0.016	-0.01	0.00	0.04
RB 91	0.017	0.000	0.0	0.06	0.00	0.0
RB 92	0.016	0.000	0.0	0.09	0.00	0.0
SR 91	0.009	0.005	0.015	0.05	0.05	0.06
SR 92	0.022	0.013	0.013	0.09	0.00	0.03
Y 92	0.001	0.005	0.025	-0.04	0.00	0.14
SR 93	0.017	0.000	0.000	0.05	0.00	0.00
Y 93F	0.001	0.005	0.024	0.00	0.05	0.16
SR 95	0.024	0.000	0.0	0.08	0.00	0.0
Y 95	0.023	0.000	0.000	0.06	0.00	0.00
ZR 95	0.004	0.004	0.025	0.03	0.00	0.04
NB 95F	0.000	0.001	0.023	-0.00	0.00	0.02
Y 97	0.036	0.000	0.0	0.10	0.00	0.0
ZR 97	0.040	0.007	0.026	0.14	0.01	0.15
NB 97M	0.000	0.000	0.018	-0.02	0.00	0.00
NB 97F	0.000	0.001	0.033	-0.01	0.00	0.12
ZR 99	0.018	0.000	0.0	0.07	0.00	0.0
MO 99	0.000	0.001	0.020	0.00	0.00	0.12
MO103	0.016	0.000	0.0	0.04	0.00	0.0
RU103	0.000	0.001	0.023	0.00	0.00	0.03
MO105	0.017	0.000	0.0	0.19	0.00	0.0
TC105	0.011	0.001	0.000	0.12	0.00	0.00
RU105	0.001	0.010	0.022	0.02	0.00	0.08
MO106	0.014	0.0	0.0	0.09	0.0	0.0
TC106	0.020	0.000	0.0	0.19	0.00	0.0
RU106	0.005	-0.039	0.000	0.10	0.02	0.00
RH106F	0.0	0.000	0.038	0.0	0.00	0.23
SB131	0.010	0.004	0.000	-0.11	0.01	0.01
I131	0.000	-0.000	0.014	-0.00	0.00	0.05
SB132M	0.018	0.000	0.000	0.17	0.00	0.00
SB132F	0.018	0.000	0.000	0.17	0.00	0.00
TE132	0.066	0.003	0.011	0.21	0.00	0.05
I132F	0.001	0.002	0.092	-0.06	0.01	0.15
TE133M	0.023	0.011	0.004	0.31	0.01	0.02
TE133F	0.017	0.000	0.000	-0.18	0.00	0.00
I133F	0.004	0.002	0.037	-0.09	0.00	0.16
TE134	0.026	0.022	0.001	0.33	0.04	0.03
I134F	0.004	0.036	0.026	-0.13	0.07	0.07
TE135	0.023	0.000	0.0	0.12	0.00	0.0
I135	0.049	0.018	0.047	0.26	0.03	0.10
XE135F	0.002	0.002	0.022	-0.05	0.00	0.13

Table 46 cont'd

I CS138F	I	0.001	0.018	0.005	I -0.03	0.01	0.02	I
I BA139	I	0.000	0.012	0.009	I -0.00	0.00	0.06	I
I XE140	I	0.030	0.000	0.0	I 0.06	0.00	0.0	I
I CS140	I	0.057	0.000	0.0	I 0.12	0.00	0.0	I
I CS141	I	0.015	0.000	0.0	I 0.09	0.00	0.0	I
I BA140	I	0.019	-0.002	0.015	I 0.08	0.00	0.10	I
I LA140	I	0.000	0.000	0.086	I -0.01	0.00	0.16	I
I BA141	I	0.013	0.003	0.000	I 0.08	0.00	0.00	I
I LA141	I	0.000	0.010	0.022	I -0.00	0.01	0.17	I
I BA142	I	0.021	0.004	0.000	I 0.08	0.00	0.00	I
I LA142	I	0.002	0.032	0.028	I -0.04	0.02	0.06	I
I BA143	I	0.019	0.000	0.0	I 0.03	0.00	0.0	I
I CE143	I	0.000	0.001	0.017	I -0.00	0.00	0.09	I
I BA144	I	0.015	0.000	0.0	I 0.04	0.00	0.0	I
I LA144	I	0.011	0.000	0.0	I 0.03	0.00	0.0	I
I CE144	I	0.001	-0.019	0.002	I 0.01	0.01	0.02	I
I PR144F	I	0.000	0.000	0.024	I -0.00	0.00	0.17	I

Table 47

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9RAP 11/00/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 05 S
PUISSANCE RESIDUELLE 8.102E-01 (MEV/S)
ERREUR DUE AUX RENDEMENTS 0.919 %
ERREUR DUE AUX PERIODES 0.239 %
ERREUR DUE AUX ENERGIES 0.863 %
ERREUR TOTALE 1.283 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
SR 89	0.000	0.001	0.012	0.00	0.00	0.09
RB 91	0.013	0.000	0.0	0.04	0.00	0.0
SR 91	0.007	0.013	0.004	0.04	0.14	0.02
Y 91F	0.000	0.001	0.017	-0.00	0.00	0.14
Y 93F	0.000	0.016	0.008	0.00	0.16	0.05
SR 95	0.045	0.000	0.0	0.16	0.00	0.0
Y 95	0.044	0.000	0.0	0.11	0.00	0.0
ZR 95	0.007	0.008	0.047	0.05	0.00	0.08
NB 95F	0.000	0.001	0.043	-0.00	0.00	0.03
Y 97	0.025	0.000	0.0	0.07	0.00	0.0
ZR 97	0.028	0.057	0.017	0.10	0.07	0.10
NB 97M	0.000	0.000	0.012	-0.00	0.00	0.00
NB 97F	0.000	0.002	0.022	-0.00	0.00	0.08
ZR 99	0.026	0.000	0.0	0.10	0.00	0.0
MO 99	0.000	0.012	0.029	0.00	0.00	0.17
NB103	0.015	0.000	0.0	0.05	0.00	0.0
MO103	0.030	0.000	0.0	0.07	0.00	0.0
RU103	0.000	0.002	0.043	0.00	0.00	0.07
MO106	0.026	0.0	0.0	0.17	0.0	0.0
TC106	0.039	0.000	0.0	0.35	0.00	0.0
RU106	0.010	-0.074	0.000	0.19	0.04	0.01
RH106F	0.0	0.000	0.073	0.0	0.00	0.43
SB131	0.015	0.000	0.000	-0.15	0.00	0.00
TE131M	0.010	0.005	0.006	0.47	0.01	0.01
I131	0.000	0.002	0.024	-0.01	0.00	0.09
SB132M	0.028	0.000	0.0	0.28	0.00	0.0
SB132F	0.028	0.000	0.0	0.28	0.00	0.0
TE132	0.100	0.041	0.016	0.34	0.04	0.08
I132F	0.000	0.004	0.138	-0.01	0.04	0.22
TE133M	0.018	0.001	0.000	0.09	0.00	0.00
TE133F	0.017	0.000	0.000	0.03	0.00	0.00
I133F	0.004	0.028	0.030	-0.02	0.01	0.13
XE133F	0.000	0.001	0.013	-0.00	0.00	0.09
I135	0.014	0.031	0.006	0.08	0.05	0.01
XE135F	0.001	0.013	0.011	-0.02	0.00	0.06
XE140	0.056	0.000	0.0	0.11	0.00	0.0

Table 47 cont'd

I CS140	I	0.107	0.000	0.0	I	0.22	0.00	0.0 I
I BA140	I	0.036	0.001	0.027	I	0.15	0.00	0.18 I
I LA140	I	0.000	0.002	0.162	I	-0.02	0.00	0.31 I
I CE141	I	0.000	0.001	0.015	I	-0.00	0.00	0.30 I
I BA143	I	0.025	0.000	0.0	I	0.05	0.00	0.0 I
I CE143	I	0.000	0.012	0.019	I	-0.00	0.02	0.09 I
I PR143	I	0.000	0.000	0.015	I	-0.00	0.00	0.15 I
I BA144	I	0.028	0.000	0.0	I	0.08	0.00	0.0 I
I LA144	I	0.020	0.000	0.0	I	0.06	0.00	0.0 I
I CE144	I	0.002	-0.037	0.004	I	0.01	0.01	0.04 I
I PR144F	I	0.000	0.000	0.047	I	-0.00	0.00	0.33 I

Table 48

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9RAP 11/00/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 06 S
PUISSANCE RESIDUELLE 3.921E-01 (MEV/S)
ERREUR DUE AUX RENDEMENTS 1.082 %
ERREUR DUE AUX PERIODES 0.161 %
ERREUR DUE AUX ENERGIES 1.343 %
ERREUR TOTALE 1.732 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
KR 89	0.016	0.000	0.0	0.08	0.00	0.0
SR 89	0.000	0.005	0.022	0.00	0.00	0.17
RB 91	0.017	0.000	0.0	0.06	0.00	0.0
Y 91F	0.000	0.006	0.030	0.00	0.00	0.25
SR 95	0.087	0.000	0.0	0.30	0.00	0.0
Y 95	0.065	0.000	0.0	0.21	0.00	0.0
ZR 95	0.013	0.029	0.086	0.09	0.00	0.14
NB 95F	0.000	0.004	0.088	-0.00	0.00	0.06
MO 99	0.000	0.017	0.004	0.00	0.00	0.02
RB103	0.026	0.000	0.0	0.09	0.00	0.0
MO103	0.052	0.000	0.0	0.13	0.00	0.0
RU103	0.000	0.020	0.073	0.00	0.01	0.11
MO106	0.054	0.0	0.0	0.35	0.0	0.0
TC106	0.079	0.000	0.0	0.72	0.00	0.0
RU106	0.021	-0.149	0.001	0.37	0.08	0.01
RH106F	0.0	0.000	0.147	0.0	0.00	0.87
SB131	0.012	0.000	0.0	-0.03	0.00	0.0
I131	0.000	0.019	0.020	-0.00	0.00	0.07
TE132	0.026	0.090	0.003	0.09	0.08	0.02
I132F	0.0	0.001	0.028	0.0	0.01	0.05
XE140	0.068	0.000	0.0	0.14	0.00	0.0
CS140	0.132	0.000	0.0	0.27	0.00	0.0
CS141	0.012	0.000	0.0	0.07	0.00	0.0
BA140	0.044	0.121	0.030	0.18	0.10	0.20
LA140	0.000	0.029	0.203	-0.01	0.04	0.39
BA141	0.011	0.000	0.0	0.06	0.00	0.0
CE141	0.000	0.007	0.024	0.00	0.00	0.48
BA143	0.014	0.000	0.0	0.03	0.00	0.0
PR143	0.000	0.009	0.019	-0.00	0.01	0.18
BA144	0.056	0.000	0.0	0.17	0.00	0.0
LA144	0.040	0.000	0.0	0.12	0.00	0.0
CE144	0.005	-0.073	0.007	0.03	0.03	0.08
PR144F	0.0	0.000	0.094	0.0	0.00	0.66

Table 49

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9RAP 11/00/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 07 S
PUISSANCE RESIDUELLE 1.297E-01 (MEV/S)
ERREUR DUE AUX RENDEMENTS 2.252 %
ERREUR DUE AUX PERIODES 0.190 %
ERREUR DUE AUX ENERGIES 2.653 %
ERREUR TOTALE 3.485 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
KR 89	0.012	0.000	0.0	0.06	0.00	0.0
SR 89	0.000	0.027	0.015	0.00	0.01	0.11
RB 91	0.015	0.000	0.0	0.05	0.00	0.0
Y 91F	0.000	0.041	0.025	0.00	0.02	0.20
SR 95	0.109	0.000	0.0	0.38	0.00	0.0
Y 95	0.106	0.000	0.0	0.26	0.00	0.0
ZR 95	0.017	0.234	0.079	0.11	0.02	0.13
NB 95F	0.000	0.066	0.130	-0.00	0.01	0.09
NB103	0.014	0.000	0.0	0.05	0.00	0.0
MO103	0.027	0.000	0.0	0.07	0.00	0.0
RU103	0.000	0.081	0.033	0.00	0.04	0.05
MO106	0.130	0.0	0.0	0.85	0.0	0.0
TC106	0.192	0.000	0.0	1.75	0.00	0.0
RU106	0.050	-0.322	0.002	0.91	0.17	0.03
RH106F	0.0	0.000	0.362	0.0	0.00	2.14
XE137	0.016	-0.000	0.0	0.03	0.00	0.0
BA137M	0.0	-0.000	0.016	0.0	0.00	0.00
BA140	0.001	0.029	0.000	0.00	0.02	0.00
CE141	0.000	0.020	0.007	0.00	0.01	0.14
BA144	0.128	0.000	0.0	0.38	0.00	0.0
LA144	0.092	0.000	0.0	0.28	0.00	0.0
CE144	0.011	-0.128	0.017	0.06	0.05	0.19
PR144F	0.0	0.000	0.217	0.0	0.00	1.52

Table 50

ANALYSE DE L'ERREUR SUR LA PUISSANCE RESIDUELLE
PU9RAP 11/00/76 TAUX DE FISSION CONSTANT (1 FIS./S.)
TEMPS D'IRRADIATION 7.776E 07 S
TEMPS DE REFROIDISSEMENT 1.000E 08 S
PUISSANCE RESIDUELLE 1.397E-02 (MEV/S)
ERREUR DUE AUX RENDEMENTS 2.729 %
ERREUR DUE AUX PERIODES 0.511 %
ERREUR DUE AUX ENERGIES 2.884 %
ERREUR TOTALE 4.003 %

NUCLEIDE	SENSIBILITE DE PUISSANCE RESIDUELLE A			ERREUR SUR PUISSANCE RESIDUELLE DUE A		
	REND.	PERIODE	EBGM	REND (%)	PER. (%)	EBGM (%)
KR 90	0.046	0.000	0.0	0.23	0.00	0.0
RB 90M	0.025	0.000	0.0	0.13	0.00	0.0
RB 90F	0.025	0.000	0.0	0.13	0.00	0.0
SR 90	0.009	-0.092	0.014	0.05	0.26	0.15
Y 90F	0.0	0.000	0.075	0.0	0.00	0.54
MO106	0.161	0.0	0.0	1.05	0.0	0.0
TC106	0.237	0.000	0.0	2.16	0.00	0.0
RU106	0.062	0.763	0.003	1.13	0.41	0.04
RH106F	0.0	0.000	0.438	0.0	0.00	2.60
SB125	0.000	0.003	0.013	0.00	0.00	0.03
I137	0.051	0.000	0.0	0.20	0.00	0.0
XE137	0.149	0.000	0.0	0.31	0.00	0.0
CS137	0.011	-0.189	0.040	0.19	0.06	0.43
BA137M	0.0	0.000	0.140	0.0	0.00	0.00
BA144	0.095	0.000	0.0	0.28	0.00	0.0
LA144	0.069	0.000	0.0	0.21	0.00	0.0
CE144	0.008	0.388	0.011	0.05	0.14	0.13
PR144F	0.0	0.000	0.143	0.0	0.00	1.00
CE147	0.013	0.000	0.0	0.08	0.00	0.0
PM147	0.000	0.005	0.018	0.00	0.00	0.24

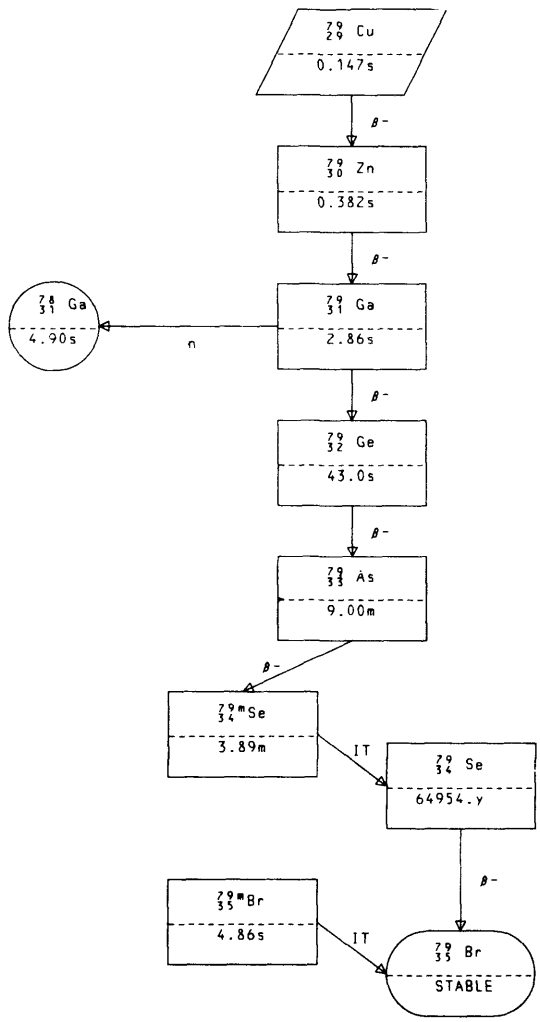
ANNEXES TO REVIEW PAPER 12

(Status of decay data of fission products)

J. Blachot
CEN Grenoble, France

Annex No. 1

Sample pages for A chain 79 from ENDF-243
"ENDF/B Fission Product Decay Data"
(BNL-NCS-5045)
by
P.F. Rose and T.W. Burrows



$^{79}_{29}$Cu

ENDF/B-IV FILE 1 COMMENTS
29-CU- 79 HEDL EVAL-APR74 R.E.SCHENTER
DIST-OCT74
REFERENCES
HALF LIFE-R SCHENTER,THEORY(9/73)

$^{79}_{29}$Cu

$T_{1/2}$ =.1474s
$\langle E_\beta \rangle$ PER DECAY =3912.
$\langle E_\gamma \rangle$ PER DECAY =2627.

FISSION YIELDS

^{235}U THERMAL	4.6025×10^{-8}
^{235}U FAST	8.8614×10^{-8}
^{238}U FAST	3.0297×10^{-7}

Q_β =10450.
BR_β =1.000

$^{79}_{30}$Zn

.3821s

79 - 29- 1

$^{79}_{30}$Zn

ENDF/B-IV FILE 1 COMMENTS
30-ZN- 79 HEDL EVAL-APR74 R.E.SCHENTER
DIST-OCT74
REFERENCES
HALF LIFE-R SCHENTER,THEORY(9/73)

$^{79}_{30}$Zn

$T_{1/2}$ =.3821s
$\langle E_\beta \rangle$ PER DECAY =3271.
$\langle E_\gamma \rangle$ PER DECAY =1990.

FISSION YIELDS

^{235}U THERMAL	1.4798×10^{-5}
^{235}U FAST	2.7555×10^{-5}
^{238}U FAST	2.7887×10^{-5}
^{239}PU THERMAL	1.1398×10^{-6}

Q_β =8660.
BR_β =1.000

$^{79}_{31}$Ga

2.860s

79 - 30- 1

$^{79}_{31}$Ga

ENDF/B-IV FILE 1 COMMENTS
31-GA- 79 HEDL EVAL-APR74 R.E.SCHENTER
DIST-OCT74
REFERENCES
DELAYED NEUTRON BRANCHING-T ENGLAND,THEORY(2/74)

$^{79}_{31}$Ga

$T_{1/2}$ =2.860s
$\langle E_\beta \rangle$ PER DECAY =2226.
$\langle E_\gamma \rangle$ PER DECAY =1276.

FISSION YIELDS

^{235}U THERMAL	1.3736×10^{-4}
^{235}U FAST	2.4827×10^{-4}
^{238}U FAST	8.4032×10^{-5}
^{239}PU THERMAL	5.2063×10^{-5}

Q_n =285.7
BR_n =.00140

Q_β =6060.
BR_β =.9986

$^{78}_{31}$Ga

4.900s

$^{79}_{32}$Ge

43.0±2.0s

79 - 31- 1

$^{79}_{32}$Ge

ENDF/B-IV FILE 1 COMMENTS
32-GE- 79 ANC EVAL-FEB74 C.W.REICH DECAY DATA
DIST-OCT74
FOR FILE DESCRIPTION SEE CW REICH,RG HELMER AND MH PUTMAN,
ANCR-1157,ENDF210,8/74.
REFERENCE
0- 1973 WAPSTRA-GOVE MASS TABLE

$^{79}_{32}$Ge

$T_{1/2}$ =43.0±2.0s
$\langle E_\beta \rangle$ PER DECAY =1893.
$\langle E_\gamma \rangle$ PER DECAY =251.4

FISSION YIELDS

^{235}U THERMAL	2.5084×10^{-4}
^{235}U FAST	3.6792×10^{-4}
^{238}U FAST	4.4076×10^{-5}
^{239}PU THERMAL	3.3455×10^{-4}

Q_β =4300.±200.
BR_β =1.000

$^{79}_{33}$As

9.000m

79 - 32- 1

PHOTON RADIATION TABLE

MEAN ENERGY	LINES	PHOTONS/100 DECAYS
230.4	1	20.00
467.8	1	6.000
542.5	1	12.00
633.4	1	4.000
745.5	1	2.000
755.0	1	6.000
781.5	1	3.400

<E_{PHOTON}> PER DECAY = 251.4

PARTICLE RADIATION TABLE

TYPE	E_{MAX}	MEAN ENERGY	INTENSITY/100 DECAYS
β^-	4070.0	1802.	20.00
β^-	4300.0	1915.	80.00

<E_{β}> PER DECAY = 1893.
<E_{ν}> PER DECAY = 2361.

PHOTON INTENSITY PLOT

CHARACTERISTIC RADIATION TABLE

TYPE	ENERGY	I/100 DECAYS
β^-	4300.	80.00
γ	230.4	20.00
β^-	4070.	20.00

$^{79}_{33}$As

END F/B-IV FILE 1 COMMENTS
33-AS- 79 HEDL EVAL-APR74 R.E.SCHENTER
DIST-OCT74

REFERENCES
OBETA -A TOBIAS(10/72) RD/B/M2453
EBETA-A TOBIAS(10/72) RD/B/M2453
EGAMMA-A TOBIAS(10/72) RD/B/M2453

$^{79}_{33}$As

$T_{1/2}$ =9.000m
<E_β> PER DECAY =861.0
<E_γ> PER DECAY =18.00

FISSION YIELDS
^{235}U THERMAL 1.0671x10^{-4}
^{235}U FAST 2.1013x10^{-5}
^{238}U FAST 7.9893x10^{-7}
^{239}Pu THERMAL 8.6968x10^{-5}

Q_β =1980.
BR_β =1.000

$^{79}_{34}$Se

3.890m

79 - 33- 1

$^{79m}_{34}$Se

END F/B-IV FILE 1 COMMENTS
34-SE- 79M HEDL EVAL-APR74 R.E.SCHENTER
DIST-OCT74

REFERENCES
OIT-R SCHENTER,THEORY(9/73)

$^{79m}_{34}$Se

$T_{1/2}$ =3.890m
<E_β> PER DECAY =.1000
<E_γ> PER DECAY =95.00

FISSION YIELDS
^{235}U THERMAL 8.1244x10^{-4}
^{235}U FAST 7.5172x10^{-4}
^{239}Pu THERMAL 1.6898x10^{-6}

Q_{IT}=250.0
BR_{IT}=1.000

$^{79}_{34}$Se

69554.y

$^{79}_{34}$Se

ENDF/B-IV FILE 1 COMMENTS
34-SE- 79 HEDL EVAL-APR74 R.E.SCHENTER
DIST-OCT74

REFERENCES
QBETA -A TOBIAS(10/72) RD/B/M2453
EBETA-A TOBIAS(10/72) RD/B/M2453
EGAMMA-A TOBIAS(10/72) RD/B/M2453

$^{79}_{34}$Se

$T_{1/2}$ =64954.y
$\langle E_\beta \rangle$ PER DECAY =42.00
$\langle E_\gamma \rangle$ PER DECAY =.1000

FISSION YIELDS
^{235}U THERMAL 8.1044×10^{-8}
^{235}U FAST 7.5112×10^{-8}
^{239}PU THERMAL 1.6898×10^{-6}

Q_β =154.0
BR_β =1.000

$^{79}_{35}$Br

STABLE OR LONG-LIVED

79 - 34- 1

$^{79m}_{35}$Br

ENDF/B-IV FILE 1 COMMENTS
35-BR- 79M HEDL EVAL-APR74 R.E.SCHENTER
DIST-OCT74

REFERENCES
QIT-C LEDERER ET AL TABLE OF ISOTOPES 6TH ED

$^{79m}_{35}$Br

$T_{1/2}$ =4.860s
$\langle E_\gamma \rangle$ PER DECAY =210.0

FISSION YIELDS
^{239}PU THERMAL 1.5698×10^{-9}

Q_{IT}=210.0
BR_{IT}=1.000

$^{79}_{35}$Br

STABLE OR LONG-LIVED

79m- 35- 1

$^{79}_{35}$Br

$^{79}_{35}$Br

STABLE OR LONG-LIVED

CROSS SECTIONS (BARNS)

σ TOTAL 2200M/S	$1.3468 \times 10^{+1}$
WESTCOTT G FACTOR	1.0229
σ CAPTURE 2200M/S	$1.1106 \times 10^{+1}$
WESTCOTT G FACTOR	1.0004
RESONANCE INTEGRAL TOTAL	$2.7710 \times 10^{+2}$
RESONANCE INTEGRAL CAPTURE	$1.3650 \times 10^{+2}$

FISSION YIELDS
^{239}PU THERMAL 1.7198×10^{-9}

79 - 35- 1

Annex n° 2

French File - C. FICHE, J. BLACHOT.
The description of this file can be found in the CEA Report (1), and some internal report (2). Some sample pages of the output are given in the following Tables,

Table 1 Listing for Ba 137 M and Pr 48

Table 2 Card images for Ba 137 M and Pr 148

Table 3 Sample of gamma rays ranged in order of increasing energy?

Table 4 List of the most abundant γ-rays (up to five) for each fission product

Table 5 Sample of the "Reference" file

Table 6 References ordered by year

Table 7 References ordered by isotopes

(1) J. BLACHOT et al. note CEA-N-1822 (Octobre 1975)

(2) C. FICHE - SEN-PA 77/16 Cadarache, France;

T A B L E 1

PRODUITS DE FISSION

Exemple de liste pour BA137M et PR148F

```
                    56 BA137M
                     ******

     PERIODE T1/2=   2.552 +/-  0.002 M          76MAR

          ENERGIE +/- ERREUR  KEV        BRANCHEMENT +/- ERREUR  % A
  QIT     661.647      0.009  76MAR         100.000      0.0

  ** GAMMAS      NG = 1        76MAR        ABSOLU

     ENERGIE +/- KEV     INTENSITE +/-          ALFA T    +/-     POLAR  % M
     661.640   0.010      89.930   0.400 *       0.112    0.0      M4    0.0

  COMMENTAIRES:EDITION DU  1/ 4/77

                    59 PR148F
                     ******

     PERIODE T1/2= 149.000 +/-  3.000 S          76MON

          ENERGIE +/- ERREUR  KEV        BRANCHEMENT +/- ERREUR  % A
  QBF-   3960.000    150.000  76MON         100.000      0.0

  ** BETAS-      NB-= 1        67NDS        ABSOLU

     ENERGIE +/- KEV     INTENSITE +/-
    4200.000   0.0       100.000   0.0

  ** GAMMAS      NG = 37       75LOH        RELATIF

     ENERGIE +/- KEV     INTENSITE +/-          ALFA T    +/-     POLAR  % M
     114.500   0.300       0.300   0.0           0.0      0.0            0.0
     214.200   0.300       1.500   0.0           0.0      0.0            0.0
     301.750   0.100     100.000   0.0   *       0.0      0.0            0.0
     343.600   0.300       1.800   0.0           0.0      0.0            0.0
     404.200   0.300       0.800   0.0           0.0      0.0            0.0
     450.600   0.100       3.900   0.0           0.0      0.0            0.0
     457.400   0.300       0.400   0.0           0.0      0.0            0.0
     467.200   0.300       1.000   0.0           0.0      0.0            0.0
     555.400   0.300       1.600   0.0           0.0      0.0            0.0
     615.000   0.300       3.900   0.0           0.0      0.0            0.0
     624.000   0.300       0.400   0.0           0.0      0.0            0.0

     ............
```

T A B L E 2

PRODUITS DE FISSION

Exemple de cartes pour BA 137M et PR 148F.

```
 56        BA137M     2.552          0.002  M          76MAR
EIT          661.647      0.00976MAR          100.000      0.0
GAMA          1                  A                                  76MAR
    661.64      0.01   89.93      0.400      0.112      0.0          1 4
CMN           1

 59        PR148F     149.       3.         S          76MON
QBF-       3960.      150.       76MON      100.
BTA-          1                  A                                  67NDS
 4200.0000              100.0000
GAMA         37                  R                                  75LOH
114.5      0.3        0.3
214.2      0.3        1.5
301.75     0.1        -100.
343.6      0.3         1.8
404.2      0.3        0.8
450.6      0.1        3.9
457.4      0.3        0.4
467.2      0.3        1.0
555.4      0.3        1.6
615.       0.3        3.9
624.       0.3        0.4
636.3      0.3        1.7
659.7      0.3        3.0
721.8      0.3        5.8
825.       0.3        3.0
869.3      0.3        6.9
902.9      0.3        1.9
947.       0.3        2.7
1023.5     0.3        -0.9
1153.2     0.3        1.4
1157.2     0.3        2.1
1171.1     0.5        -1.5
1210.1     0.5        2.6
1248.6     0.5        -6.
1344.                 2.0
1358.      0.5        9.9
1381.8     0.5        4.1
1420.0                1.5
1521.5     0.5        2.4
1772.0                1.
1907.5                0.5
 1931.5                0.5
2130.0                1.5

.......
```

T A B L E 3

PRODUITS DE FISSION

Example of the γ-ray table, ordered by energy

(The letter R means that the intensity is relative)

ENERGIE		INTENSITE	ELEMENT	PERIODE		ENERGIE		INTENSITE	ELEMENT	PERIODE	
1578.00		0.352	BR 84F	31.80	M	1617.91		0.428	ND151	12.40	M
1578.20	R	3.200	BR 87	55.70	S	1618.50		1.709	Y 95	10.60	M
1578.31		0.203	ND151	12.40	M	1620.22		0.015	KR 90	32.32	S
1579.00		0.007	TE131F	25.00	M	1620.60		1.098	SE 83F	22.60	M
1579.23		0.432	XE139	39.70	S	1620.90		0.441	CS139	8.980	M
1579.80		0.002	RH106F	29.90	S	1621.10		0.020	KR 92	1.920	S
1580.10	R	3.700	SB133	2.430	M	1621.10		0.000	RH106F	29.90	S
1581.90		1.896	SB130M	40.00	M	1621.50		0.200	RB 93	5.890	S
1582.30		0.346	KR 91	8.570	S	1622.30		0.053	BA141	18.00	M
1582.87		0.100	KR 89	3.180	M	1623.80		0.813	LA144	42.40	S
1583.00		0.070	MO101	14.62	M	1624.30		1.926	RB 91	58.20	S
1583.90		0.197	GA 76	27.60	S	1624.50		0.450	CD119M	110.0	S
1585.20		0.300	SB128F	10.00	M	1625.20	R	1.100	XE142	1.220	S
1585.21		0.176	XE139	39.70	S	1625.50		7.534	PM154F	2.800	M
1585.70	R	0.900	XE140	13.60	S	1625.76		0.300	RH104M	4.340	M
1588.20		0.290	TE133F	12.43	M	1627.30		0.116	CS140	63.70	S
1588.60		0.002	RH106F	29.90	S	1627.40		1.700	KR 93	1.200	S
1589.60		0.240	MO101	14.62	M	1627.80		1.284	RB 91	58.20	S
1590.30		0.156	RB 90F	2.550	M	1627.98		0.030	ND151	12.40	M
1590.70		2.857	Y 98M	2.000	S	1628.40		0.000	RH106F	29.90	S
1590.70		5.224	Y 98F	.6500	S	1629.00		0.020	MO101	14.62	M
1591.56		0.114	XE139	39.70	S	1629.24		0.258	I134F	52.60	M
1591.80		0.040	CS139	8.980	M	1631.40		1.174	LA144	42.40	S
1592.70		0.001	RH106F	29.90	S	1631.60		0.365	ZN 71F	2.400	M
1593.00		3.738	RH110M	28.50	S	1631.78		0.188	RB 90F	2.550	M

TABLE 4

PRODUITS DE FISSION

Sample of output for the 5 most abundant γ-rays

NUCLIDE	HALF-LIFE			E1	I1	E2	I2	E3	I3	E4	I4	E5	I5
53 I134F	52.60	M		847.02	95.400	884.09	65.254	1072.55	15.264	595.36	11.353	621.79	10.589
55CS134M	2.910	H		127.42	13.700	11.20	0.930	138.68	0.002				
55CS134F	2.061	A		604.64	97.540	795.84	85.130	569.35	15.000	802.07	8.800	563.20	8.400
50SN135	0.2000	S											
51SB135	1.700	S											
52TE135	18.00	S	R	603.50	100.000	870.30	23.000	266.80	15.000				
53 I135	6.700	H		1260.50	30.280	1131.60	22.952	526.50	11.718	1678.30	10.325	1458.10	9.447
54XE135M	15.65	M		526.62	81.200								
54XE135F	9.150	H		249.74	90.600	608.07	2.580	407.90	0.320	158.20	0.230	358.30	0.200
55CS135	0.3000E 07A												
56BA135M	28.70	H		268.24	15.580								
50SN136	0.2000	S											
51SB136	0.8200	S											
52TE136	20.90	S	R	2077.90	113.000	332.60	100.000	578.60	98.000	2569.40	82.000	3235.10	80.000
53 I136M	48.00	S	R	1313.00	100.000	381.30	98.000	197.30	85.000	370.10	19.000		
53 I136F	83.00	S		1313.00	69.300	1321.10	30.492	2289.80	12.474	2634.20	8.316	2415.00	7.623
55CS136	13.10	J		818.50	99.700	1048.07	79.720	340.57	46.753	1235.34	19.780	176.56	13.586
56BA136M	0.3200	S	R	163.70	100.000	818.00	100.000	1050.00	100.000				
51SB137	0.1000E 00S												
52TE137	3.500	S	R	243.00	100.000								
53 I137	24.60	S	R	1218.60	100.000	600.50	51.000	1302.80	41.000	1216.90	36.700	1426.80	25.100
54XE137	3.830	M		455.45	31.800	849.00	0.725	1783.40	0.487	2850.00	0.242	1273.20	0.238
55CS137	30.17	A											
56BA137M	2.552	M		661.64	89.930								
51SB138	0.1000E 00S												
52TE138	1.400	S											
53 I138	6.440	S		588.87	97.660	875.29	18.165	483.67	7.032	1809.50	5.664	1277.70	4.883
54XE138	14.17	M		258.31	29.000	434.49	19.400	1768.26	18.600	2015.82	13.700	2004.75	6.100
55CS138M	2.900	M		1435.86	25.000	462.80	24.625	191.90	20.000	112.50	2.000	324.30	1.550
55CS138F	32.20	M		1435.86	76.300	462.79	30.749	1009.78	29.833	2218.00	15.184	546.94	10.758
52TE139	0.7000	S											
53 I139	2.610	S	R	527.00	100.000	571.10	60.000	536.80	50.000	847.90	50.000	258.40	30.000
54XE139	39.70	S		218.59	56.830	296.53	22.107	174.92	21.027	289.78	9.343	393.50	6.820
55CS139	8.980	M		1283.30	8.020	627.30	1.845	1420.80	0.930	2100.00	0.642	2111.00	0.642
56BA139	82.70	M		165.85	18.800	1420.50	0.302	1254.70	0.047	1090.80	0.019	1310.60	0.018

TABLE 5

REFERENCE FILE

910*76HAN	*ZR 95	*T			HANSEN H.H. GROSSE G. ET AL Z. PHYS. A278(1976)317
	NB 95				
911*76HOI	*CE149	*T		G	HOLLAND G.E. NUCL. DAT. SH. 19(1976)337
	ND149				
	PM149				
	PR149				
912*76SCH	*TE136	*T		G	SCHUSSLER COMMUNICATION PRIVEE 1976
913*76TRA	*TC109	*			TRAUTMANN N. KAFFRELL ET AL PHYS REV C13N2(1976)872
	TC110				
914*76KOC2	*CO 58	*T		G	KOCHER D.C. AUBLE R.L. NUCL. DAT. SH. 19(1976)445
915*76BER	*AS 76	*G		G	BERTRAND F.E. AUBLE R.L. NUCL. DAT. SH. 19(1976)507
	GA 76				
	ZN 76				
916*76YOS	*SE 85	*T		G	YOSHIYUKI KISO ET AL AN REP RES REACT KYOTO 9(1976)117
	MO102				
	MO103				
	MO104				
	MO105				
	BA143				
	BA144				
917*77SIS	* Y 98	*T		G	SISTEMICH K. SADLER G. ET AL 1977 A PARAITRE
918*76SIN	*Y 96	*			SINGH B. TAYLOR H.W. J PHYS G NUCL PHYS 2 N6(1976)397
919*77PER	*MO 99	*			PEROLAT J.P. TABLE DES RADIONUCLEIDES SACLAY (1977)
920*77TUU	*DY167	*			TUURNALA T. ET AL. Z. PHYSIK A 280(1977)309
921*77WOI	*RB 98	*			WOLLNIK H. ET AL. A PARAITRE
922*76MON1	*CS143	*T	G		MONNAND E. SCHUSSLER F. ILL REPORT (1976)
	BA140				
923*69RAG	*RB 88	*			RAGAINI R.C. KNIGHT J.D. NUCL.PHYS. A125(1969)97
924*76MAP	*IN113M	*G			MARTIN M.J. NUCLEAR DECAY DATA ORNL 5114(1976)
	SN119M				
	I129				
925*76TAI	*RB 90	*			TALBERT J. COMMUNICATION PRIVEE (1976)
926*76END	*RH106F	*			ENSDF COMMUNICATION DU DR. EWBANK OCTOBRE 1976

TABLE 6

REFERENCE FILE

Sorted by year (example)

```
75LOH     LOHENGRIN COLLABORATION ILL GRENOBLE 1975
75MAC     MACIAS E.S PHELPS ET AL PHYS REV C VOL14N2(1976)639

76ALE     ALEKLETT K. LUND E. ET AL RESEARCH REPORT LF71 CARGESE 7
76AMI     AMIEL S. ENGLER G. CARGESE (1976)
76AUB     AUBLE R.L. TODD ET AL NUCL. DAT. SH. 19(1976)1
76BER     BERTRAND F.E. AUBLE R.L. NUCL. DAT. SH. 19(1976)507
76BLA     BLACHOT J. DOUSSON S. ET AL J. DE PHYS. 37(1976)275
76BUN1    BUNTING R.L. KRAUSHAAR J.J. NUCL. DAT SH. 18(1976)87
76BUR     BURROWS T.W. NUCL. DAT. SH. 18(1976)553
76ELL     ELLIS Y.A. NUCL. DAT. SH. 19(1976)103
76ELL1    ELLIS Y.A. NUCL. DAT. SH. 19(1976) 143
76ELL2    ELLIS Y.A. NUCL. DAT. SH. 19(1976) 181
76END     ENSDF COMMUNICATION DU DR. EWBANK OCTOBRE 1976
76FOG1    FOGELBERG B. KAWASE J. NUCL. PHYS. A267(1976)317
76HAN     HANSEN H.H. GROSSE G. ET AL Z. PHYS. A278(1976)317
76HAR1    HARMATZ B. NUCL. DAT. SH. 19(1976)33
76HOL     HOLLAND G.E. NUCL. DAT. SH. 19(1976)337
76JAC     JACKSON S.V. MEYER PHYS. REV C VOL13N1(1976)339
76JAN     JANECKE  GARVEY KELSON AT.NUCL.DAT.TAB.17(1976)1
76KAF     KAFFRELL N. FRANZ GARGESE 1976 CERN-76-13
76KOC2    KOCHER D.C. AUBLE R.L. NUCL. DAT. SH. 19(1976)445
76LEW     LEWIS L. NUMMELLEY ET AL PHYS. REV. 13 N5(1976)2017
76LOH     LOHENGRIN COLLABORATION ILL GRENOBLE 1976
76MAN     MANN L.G. WALTERS ET AL PHYS. REV. VOL14N3(1976)1141
76MAR     MARTIN M.J. NUCLEAR DECAY DATA ORNL 5114(1976)
76MEY     MEYER R.A. MOMYER F.F. PHYS. REV.C VOL 14 N 3(1976)1152
76MON     MONNAND E. J. BLACHOT CARGESE  CERN-76-13
76MON1    MONNAND E. SCHUSSLER F. ILL REPORT (1976)
76OTE     OTERO D. PROTO A.N. ET AL PHYS. REV. 13 N5 (1976)1996
76PAN     PANCHOLI S.C. MARTIN M.J. NUCL. DAT. SH. 18(1976)166
76PFE     PFEIFFER B. COMMUNICATION PRIVEE 1976
76PIN     PINSTON J.A. ROUSSILLE R. ET AL COMMUNICATION PRIVEE 197
76RUD     RUDSTAM G. LUND PHYS. REV. CVOL13N1(1976)321
76SAM     SAMUELSON L.E. KELLY W.H. ET AL NUCL. DAT SH. 18(1976)12
76SCH     SCHUSSLER COMMUNICATION PRIVEE 1976
76SCH1    SCHUSSLER F. BLACHOT J. & AL.-Z. PHY.-A PARAITRE 1977
76SIN     SINGH B. TAYLOR H.W. J PHYS G NUCL PHYS 2 N6(1976)397
76SIS     SISTEMICH K. SADLER G. ET AL CARGESE (1976)
76TAL     TALBERT J. COMMUNICATION PRIVEE (1976)
76TRA     TRAUTMANN N. KAFFRELL ET AL PHYS REV C13N2(1976)872
76VAI     VAISALA S. GRAEFFE PHYS. REV. VOL13N1(1976)372
76WES     WESTERN W.R. HILL ET AL PHYS. REV. CVOL13N1 (1976)275
76WOH     WOHN F.K. GLASCOCK ET AL PHYS. REV CVOL13N6(1976)2492
76YOS     YOSHIYUKI KISO ET AL AN REP RES REACT KYOTO 9(1976)117
76ZOL     ZOLLER H. WILLIAM ET AL PHYS. REV 13 N5 (1976)2024

77PER     PEROLAT J.P. TABLE DES RADIONUCLEIDES SACLAY (1977)
77PFE     PFEIFFER B. ROCQUET J.P. ET AL JOUR DE PHYS 38(1977)600
77SIS     SISTEMICH K. SADLER G. ET AL 1977 A PARAITRE
77TUU     TUURNALA T. ET AL.  Z. PHYSIK A 280(1977)309
77WAP     WAPSTRA A.H. ET AL ATOMIC NUCLEAR DATA TABLE A PARAITRE
77WOL     WOLLNIK H. ET AL.  A PARAITRE
```

TABLE 7

REFERENCE FILE

Sorted by nuclide mass (example)

```
                    ***** 93 *****
KR 93     76RUD T     RUDSTAM G. LUND PHYS. REV. CVOL13N1(1976)321
RB 93     76RUD T     RUDSTAM G. LUND PHYS. REV. CVOL13N1(1976)321
SP 93     76ALF Q     ALEKLETT K. LUND E. ET AL RESEARCH REPORT LF71 CARGESE 7

                    ***** 94 *****
RB 94     76AMI T     AMIEL S. ENGLER G. CARGESE (1976)
RB 93     76RUD T     RUDSTAM G. LUND PHYS. REV. CVOL13N1(1976)321
SB 94     76AMI T     AMIEL S. ENGLER G. CARGESE (1976)

                    ***** 95 *****
NB 95     76HAN T     HANSEN H.H. GROSSE G. ET AL Z. PHYS. A278(1976)317
RB 95     76AMI T     AMIEL S. ENGLER G. CARGESE (1976)
RB 93     76RUD T     RUDSTAM G. LUND PHYS. REV. CVOL13N1(1976)321
SR 95     76MON TG    MONNAND E. J. BLACHOT CARGESE  CERN-76-13
ZP 95     76HAN T     HANSEN H.H. GROSSE G. ET AL Z. PHYS. A278(1976)317
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Annex No. 3

COMPARISON OF ENDF/B IV AND THE FRENCH FILE

C. Fiche, J. Blachot (CEA, France)

Dr. England (Los Alamos Scientific Laboratory) send to us a routine written on a magnetic tape that was interpreted and contained informations identical with that in Table VII in ENDF-223.

The table V (ENDF-223) lists the few changes from the original ENDF/B IV, and also some notes added by Dr. England.

A table similar to ENDF/B IV was obtained by computer program from the French File. The two tables were compared, and results are given in tables 1 to 3.

All the differences are given in percentage relative to the greater value.

+ means data in FF is greater than data in ENDF
- means data in FF is smaller than data in ENDF

T A B L E 1
242 Nuclides with experimental data in both files.

T A B L E 2
301 Nuclides with partial or without experimental data.

T A B L E 3 A)
Nuclides only the French File.

T A B L E 3 B)
Nuclides only on the ENDF/B IV file.

TABLE V (ENDF-223)

NUCLIDES HAVING TYPOGRAPHICAL OR SUSPECTED ERRORS IN ENDF/B-IV FISSION-PRODUCT FILES

Nuclide	Corrected in Table VII	Comments
^{97}Y	yes	$\bar{E}_\gamma = 9.35 \times 10^5$ eV.
^{104m}Rh	yes	Normalization factor (F) = 0 (F should be 1.8535×10^{-2}), and typographical errors in spectra.
^{126}Sn	yes	$\tau_{1/2} = 3.15569 \times 10^{12}$ s.
^{129m}Te	yes	Internal conversion energy of 0.6682×10^5 eV added to $\bar{E}_\gamma$. (F=0.10051)
^{130m}Sb	yes	$\bar{E}_\gamma$ too small ($E_\gamma \cong 3.04 \times 10^6$ eV). Change normalization to 1.17717.
^{131}Sb	yes	$\bar{E}_\gamma = 1.7025 \times 10^6$ eV.
^{133}Sb	yes	$\bar{E}_\gamma$ too large ($E_\gamma \cong 2.5 \times 10^6$ eV). Change normalization to 3.87351×10^{-1}.
^{136}Cs	no	$\bar{E}_\gamma$ includes some γ energy from ^{136}Ba.
^{138m}Cs	yes	$\bar{E}_\gamma$ too small ($\bar{E}_\gamma \cong 2.6 \times 10^6$ eV). Change normalization to 1.23827.
^{140}La	yes	$\bar{E}_\gamma$ too small ($\bar{E}_\gamma \cong 2.3$ v 10^6 eV). Change normalization to 1.03275.
^{142}La	yes	$\bar{E}_\gamma$ too large ($E_\gamma \cong 2.4 \times 10^6$ eV). Change normalization to 0.045854
^{152m}Pm	yes	$\bar{E}_\beta$ too small ($\bar{E}_\beta \cong 0.9 \times 10^6$ eV) and beta intensities do not sum to 1.0. Change normalization to 2.14551.
^{166}Er	yes	σ(0.0253 eV) too large ($\sigma \cong 20$ b).

Note 1: The table lists changes based on first BNL tape. The later issue corrects ^{126}Sn, ^{97}Y and ^{104}Rh.

Note 2: ^{98}Zr should decay 100% to ^{98}Nb, not to ^{98m}Nb.

Note 3: ^{113}Cd and ^{99}Tc have no MT 457 and therefore are stable by implication in ENDF/B-IV. The correct halflives are listed in the attached Table VII (but not in ENDF-223, LA-6116-MS).

Note 4: Since the publication of ENDF-223, LA-6116-MS, users have suggested:

a) Q value for ^{97m}Sc should be 9.6E+4

b) ^{110m}Ag and ^{110}Ag halflives should be interchanged

c) either the branchings for ^{130}Sn to ^{130}Sb isomers should be interchanged, or the halflives of the ^{130}Sb isomers (and associated data) should be reversed

TABLE 1

Differences (in %) between the data in the French File and in ENDF/B IV for the 242 nuclides with experimental data in both files

(EBMT average beta energy per decay
EGT total gamma energy per decay
NDK number of decay modes
RTYP type of decay
Q Q-value
B branching fraction for decay mode)

NUCLEIDE	T1/2	EBMT	EGT	NDK	RTYP	Q	B
30-ZN- 72	0.0	-2.9	10.3	1	1.0	0.2	-0.0
31-GA- 73	0.6	-3.2	2.1	2	1.0	100.0	100.0
					1.1	0.0	-0.0
30-ZN- 74	0.0	14.5	-55.8	1	1.0	6.0	-0.0
31-GA- 74	0.6	0.0	0.0	1	1.0	0.0	-0.0
31-GA- 75	9.5	3.9	23.2	2	1.0	0.0	-0.0
					1.1	-0.0	-0.0
32-GE- 75F	-0.0	-4.0	-0.4	1	1.0	-1.3	-0.0
31-GA- 76	1.8	-10.3	6.6	1	1.0	4.0	-0.0
33-AS- 76	0.1	-4.2	4.6	1	1.0	-0.4	-0.0
32-GE- 77M	-0.6	-1.7	-6.1	2	1.0	-1.6	-0.0
					3.0	0.4	-0.0
32-GE- 77F	0.0	-2.1	-2.5	1	1.0	-2.1	-0.0
33-AS- 77	0.0	-6.9	-92.9	2	1.0	0.1	-0.0
					1.1	18.2	11.8
32-GE- 78	0.0	-3.0	0.4	1	1.0	0.0	-0.0
33-AS- 78	-0.2	-6.8	20.2	1	1.0	0.9	-0.0
32-GE- 79	-2.3	-0.4	0.0	1	1.0	0.0	-0.0
33-AS- 79	-8.9	-2.2	6.1	2	1.0	100.0	100.0
					1.1	5.9	-1.3
33-AS- 80	0.0	-3.3	-0.0	1	1.0	-5.0	-0.0
34-SE- 81F	0.0	0.1	-7.7	1	1.0	0.2	-0.0
33-AS- 82M	-2.3	0.0	23.1	1	1.0	0.0	-0.0
33-AS- 82F	9.5	-1.5	27.7	1	1.0	0.0	-0.0
35-BR- 82M	0.5	-93.3	-14.5	2	1.0	0.1	-91.7
					3.0	0.0	2.2
35-BR- 82F	0.1	-4.6	-1.4	1	1.0	0.1	-0.0
34-SE- 83M	0.6	-5.2	4.8	2	1.0	-2.2	-0.0
					3.0	100.0	100.0
34-SE- 83F	0.4	4.2	-8.8	1	1.0	1.0	-0.0
35-BR- 83	-0.4	-1.5	90.9	2	1.0	100.0	100.0
					1.1	4.4	-0.0
34-SE- 84	-6.1	-0.8	-0.0	1	1.0	0.4	-0.0
35-BR- 84M	0.0	-21.0	0.6	1	1.0	-5.4	-0.0
35-BR- 84F	-0.0	2.7	-3.8	1	1.0	0.1	-0.0
34-SE- 85	-19.5	21.3	-46.2	1	1.0	-0.8	-0.0
36-KR- 85M	0.0	-1.0	21.0	2	1.0	0.1	-1.0
					3.0	0.1	3.6
36-KR- 85F	-0.3	-10.2	-0.9	1	1.0	-0.0	-0.0
35-BR- 86	-1.8	23.5	-17.6	1	1.0	0.0	-0.0
37-RB- 86F	0.8	-2.3	0.0	1	1.0	0.1	-0.0
36-KR- 87	0.4	-0.1	0.1	1	1.0	-0.1	-0.0
35-BR- 88	2.5	-19.5	38.5	2	1.0	-8.0	-0.0
					5.0	11.3	-0.0
36-KR- 88	0.0	32.8	-11.6	1	1.0	-0.6	-0.0
37-RB- 88	0.4	-0.4	1.4	1	1.0	0.2	-0.0
36-KR- 89	0.6	-0.5	-4.2	1	1.0	0.0	-0.0
37-RB- 89	-0.3	0.0	-0.8	1	1.0	0.0	-0.0
36-KR- 90	0.1	9.6	-70.0	2	1.0	-0.5	-0.0
					1.1	-0.5	0.0
37-RB- 90M	-2.3	11.6	-7.3	2	1.0	0.0	2.7
					3.0	0.5	-52.0
37-RB- 90F	-5.6	11.8	-100.0	1	1.0	0.0	-0.0
39- Y- 90M	2.8	-9.9	0.8	3	1.0	100.0	100.0
					-1.1	-100.0	-100.0
					3.0	-0.4	-0.0
39- Y- 90F	0.2	0.0	92.0	1	1.0	-0.0	-0.0
37-RB- 91	0.0	36.6	-56.3	1	1.0	0.4	-0.0

TABLE 1 cont'd

NUCLEIDE	T1/2	EBMT	EGT	NDK	RTYP	Q	B	NUCLEIDE	T1/2	EBMT	EGT	NDK	RTYP	Q	B
38-SR- 91	0.0	0.0	32.4	2	1.0	0.0	-38.7	44-RU-105	-0.9	0.5	-1.6	2	1.0	-0.0	7.2
					1.1	0.1	22.2						1.1	-2.5	-22.5
39- Y- 91F	-0.2	-2.6	-0.1	1	1.0	0.0	-0.0	45-RH-105F	-0.4	0.0	0.4	1	1.0	-0.1	-0.0
36-KR- 92	4.2	-0.2	-0.1	2	1.0	0.0	-0.0	45-RH-106M	0.0	0.0	7.4	2	1.0	1.3	-0.0
					5.0	100.0	-20.0						3.0	100.0	100.0
37-RB- 92	-1.1	1.5	2.1	2	1.0	1.6	0.0	45-RH-106F	0.0	1.7	1.3	1	1.0	0.0	-0.0
					5.0	100.0	-0.0	44-RU-107	7.7	-0.0	54.2	1	1.0	0.0	-0.0
38-SR- 92	0.0	0.9	0.0	1	1.0	0.0	-0.0	45-RH-107	0.0	0.0	1.3	1	1.0	0.0	-0.0
39- Y- 92	0.0	-0.1	-0.2	1	1.0	0.0	-0.0	44-RU-108	0.0	-0.6	12.6	1	1.0	0.0	-0.0
36-KR- 93	-5.5	2.8	-45.7	2	1.0	-5.5	-0.0	45-RH-108M	0.0	0.4	-15.3	1	1.0	1.6	-0.0
					5.0	-20.6	-0.0	45-RH-108F	-1.2	2.7	-51.0	1	1.0	0.0	-0.0
37-RB- 93	1.5	9.6	4.9	2	1.0	17.5	0.5	47-AG-108F	0.0	18.9	-94.6	2	1.0	0.2	-0.2
					5.0	30.8	-28.4						2.0	0.5	6.1
38-SR- 93	-2.7	-22.9	32.0	1	1.0	1.0	-0.0	45-RH-109	-11.1	18.6	-35.8	2	1.0	0.0	50.0
39- Y- 93F	-1.0	-0.0	-0.7	1	1.0	0.0	-0.0						1.1	2.6	-100.0
37-RB- 94	-0.7	27.3	-91.2	2	1.0	5.4	1.3	46-PD-109F	0.0	0.0	-100.0	2	1.0	0.0	84.0
					5.0	34.2	-10.8						1.1	-0.0	-0.1
38-SR- 94	0.7	3.7	-2.6	1	1.0	2.0	-0.0	45-RH-110M	89.5	-45.8	97.5	1	1.0	-1.8	-0.0
39- Y- 94	-1.6	5.9	-22.6	1	1.0	0.0	-0.0	47-AG-110M	100.0	-1.7	-1.1	2	1.0	0.8	-0.0
41-NB- 94M	0.0	32.5	8.5	2	1.0	-0.6	58.3						3.0	0.4	-0.0
					3.0	0.6	-0.3	47-AG-110F	-100.0	2.1	-7.4	2	1.0	0.1	-0.0
41-NB- 94F	1.2	-23.6	0.9	1	1.0	-0.2	-0.0						2.0	1.7	-0.0
38-SR- 95	7.1	12.4	-24.1	1	1.0	4.8	-0.0	46-PD-111M	0.0	-14.2	-26.8	3	1.0	0.0	73.3
39- Y- 95	0.9	-17.5	61.6	1	1.0	0.0	-0.0						1.1	-0.4	-84.3
40-ZR- 95	-2.3	-0.3	0.1	2	1.0	-0.1	0.6						3.0	-0.1	4.6
					1.1	-0.2	-50.0	46-PD-111F	0.0	-0.1	99.3	2	1.0	0.0	0.0
41-NB- 95F	0.1	0.4	-0.0	1	1.0	-0.0	-0.0						1.1	0.0	-0.0
39- Y- 96F	-92.8	-69.3	63.7	1	1.0	1.7	-0.0	47-AG-111F	0.1	-0.1	-2.0	1	1.0	0.0	-0.0
41-NB- 96	0.0	-7.3	-1.7	1	1.0	-0.1	-0.0	46-PD-112	0.1	-0.3	-75.7	1	1.0	1.0	-0.0
38-SR- 97	53.5	12.3	0.4	2	1.0	4.1	0.1	47-AG-112	-0.3	-0.0	2.3	1	1.0	0.0	-0.0
					-5.0	-100.0	-100.0	46-PD-113	9.6	28.1	-94.6	2	1.0	-2.8	-0.0
39- Y- 97F	70.0	-7.5	47.2	2	1.0	14.5	-0.0						1.1	2.9	0.0
					5.0	100.0	-0.0	47-AG-113F	1.3	25.4	-88.1	2	1.0	0.5	-0.0
40-ZR- 97	1.2	-0.3	79.5	2	1.0	-0.5	0.0						1.1	0.9	-0.0
					1.1	-0.8	-0.0	48-CD-113M	-4.4	15.0	-99.8	2	1.0	2.3	-0.0
41-NB- 97F	-2.2	0.3	-1.5	1	1.0	-0.0	-0.0						3.0	-2.3	-0.0
39- Y- 98F	53.8	2.4	-50.9	2	1.0	-10.4	0.5	46-PD-114	2.0	-6.8	-96.6	1	1.0	-31.0	-0.0
					-5.0	-100.0	-100.0	47-AG-114F	-2.2	1.1	24.9	1	1.0	0.2	-0.0
41-NB- 98M	0.6	0.0	1.5	1	1.0	-0.3	-0.0	49-IN-114F	0.0	-23.4	-99.5	2	1.0	-0.8	-1.4
41-NB- 98F	2.1	6.4	-43.2	1	1.0	6.2	-0.0						2.0	0.4	40.5
42-MO- 99	0.0	1.2	-18.1	2	1.0	-0.0	-12.4	47-AG-115F	-4.8	13.1	-3.7	2	1.0	-2.5	-0.0
					1.1	1.3	1.9						1.1	-2.0	0.0
41-NB-100F	-37.5	22.0	-50.6	1	1.0	-1.1	-0.0	48-CD-115M	0.4	23.6	-91.3	3	1.0	-1.3	-3.0
43-TC-100	-1.2	-3.8	1.2	1	1.0	-5.0	-0.0						1.1	-1.7	99.7
42-MO-101	0.1	-3.2	-0.8	1	1.0	-0.1	-0.0						3.0	100.0	100.0
43-TC-101	0.0	-2.4	1.5	1	1.0	0.1	-0.0	48-CD-115F	-0.9	0.7	-25.2	2	1.0	100.0	100.0
43-TC-102M	4.4	-0.0	-6.1	2	1.0	-1.9	-5.0						1.1	-0.3	-0.0
					3.0	100.0	100.0	49-IN-115M	-4.4	19.9	-1.7	2	1.0	0.5	-0.0
43-TC-102F	-5.7	0.0	0.2	1	1.0	0.0	-0.0						3.0	0.4	-0.0
43-TC-103	7.7	11.0	-23.1	1	1.0	0.0	-0.0	47-AG-116M	-1.1	-1.6	-30.9	2	1.0	-2.4	2.0
44-RU-103	-0.6	-4.4	7.2	2	1.0	100.0	100.0						-3.0	-100.0	-100.0
					1.1	0.2	-0.3	47-AG-116F	-0.0	-7.9	23.3	1	1.0	-3.2	-0.0
43-TC-104	1.1	0.0	25.7	1	1.0	21.3	-0.0	49-IN-116F	-0.7	29.2	-97.5	1	1.0	-1.5	-0.0
45-RH-104M	-0.2	-70.2	6.7	2	1.0	-0.1	-35.0	48-CD-117M	2.9	-66.0	62.4	2	1.0	-4.8	38.9
					3.0	-0.0	0.1						1.1	0.0	-81.1
45-RH-104F	0.7	0.3	0.8	2	1.0	-0.1	-0.4	49-IN-117F	-9.1	-39.5	50.4	1	1.0	-1.1	-0.0
					2.0	100.0	75.8	47-AG-118F	94.6	14.0	-3.1	1	1.0	-0.3	-0.0
43-TC-105	-3.8	9.4	-49.4	1	1.0	-6.2	-0.0	49-IN-118M	98.1	-64.6	91.9	2	1.0	0.0	-0.0
													3.0	100.0	100.0

TABLE 1 cont'd

NUCLEIDE	T1/2	EBMT	EGT	NDK	RTYP	Q	B
49-IN-118F	-98.1	64.6	-91.5	1	1.0	0.0	-0.0
48-CD-119F	-71.6	2.3	-1.4	2	1.0	100.0	100.0
					1.1	-1.9	-7.3
49-IN-119M	0.0	28.3	-95.4	2	1.0	-4.0	-2.1
					3.0	19.7	28.6
47-AG-120F	-10.0	-12.5	-65.0	1	1.0	-28.1	-0.0
49-IN-120M	93.5	-61.7	94.1	1	1.0	-3.6	-0.0
49-IN-120F	-93.7	55.8	-91.6	1	1.0	1.9	-0.0
49-IN-121F	-17.5	-1.7	-8.4	1	1.0	-6.7	-0.0
50-SN-121M	-0.3	-40.6	-77.4	1	1.0	-38.5	-0.0
49-IN-122M	85.0	-9.5	0.0	1	1.0	-7.1	-0.0
49-IN-122F	-86.4	13.1	-38.6	1	1.0	-3.7	-0.0
51-SB-122F	-0.2	0.6	-4.6	2	1.0	0.4	0.6
					2.0	-1.2	-20.0
50-SN-123M	0.2	-1.2	-65.5	2	1.0	-14.7	-0.0
					3.0	100.0	100.0
50-SN-123F	0.2	23.4	-98.3	1	1.0	-1.4	-0.0
48-CD-124	-94.2	50.2	-88.8	1	1.0	21.5	-0.0
51-SB-124F	-0.0	-54.8	60.3	1	1.0	-0.2	-0.0
50-SN-125M	0.0	1.0	2.2	2	1.0	-0.5	-0.0
					3.0	100.0	100.0
50-SN-125F	-0.1	-0.0	-0.6	1	1.0	-0.6	-0.0
51-SB-125	1.1	-2.3	6.0	2	1.0	0.0	1.8
					1.1	0.0	-6.1
50-SN-126	-0.3	1.3	54.8	2	1.0	100.0	100.0
					1.1	86.5	-14.0
51-SB-126M	0.0	-33.5	35.9	2	1.0	-7.5	-0.0
					3.0	-92.9	-0.0
51-SB-126F	-0.0	-10.2	6.5	1	1.0	-0.1	-0.0
51-SB-127	1.3	-2.7	3.2	2	1.0	0.0	-0.0
					1.1	-0.7	0.0
52-TE-127M	0.0	13.6	-4.6	2	1.0	0.3	-0.0
					3.0	-0.5	-0.0
52-TE-127F	0.0	-0.3	-25.4	1	1.0	0.0	-0.0
50-SN-128	0.2	-11.5	9.3	2	1.0	100.0	100.0
					-1.1	-100.0	-100.0
51-SB-129	-0.5	0.0	-5.9	2	1.0	0.0	-0.0
					1.1	-2.0	0.0
52-TE-129M	2.1	-0.1	-68.0	2	1.0	0.0	-0.0
					3.0	0.0	-0.0
52-TE-129F	-0.6	-1.0	-10.2	1	1.0	0.1	-0.0
53- I-129	-1.5	-34.9	-1.3	1	1.0	0.9	-0.0
50-SN-130F	0.0	-29.5	30.5	2	1.0	-2.9	90.0
					1.1	9.3	-100.0
51-SB-130M	7.5	-56.1	3.9	1	1.0	-15.8	-0.0
51-SB-130F	-4.5	-0.0	13.5	1	1.0	-1.6	-0.0
53- I-130F	-0.3	-0.2	-6.1	1	1.0	-0.2	-0.0
51-SB-131	0.0	-4.0	-4.4	2	1.0	-8.8	4.8
					1.1	0.0	-69.1
52-TE-131M	0.0	-16.3	-1.5	2	1.0	-7.4	-5.1
					3.0	-0.1	18.9
52-TE-131F	0.0	3.2	5.6	1	1.0	0.0	-0.0
53- I-131	-0.0	-0.0	1.2	2	1.0	-0.0	-0.0
					1.1	3.7	-0.0
51-SB-132M	0.0	0.0	15.4	1	1.0	-7.9	-0.0
51-SB-132F	25.0	0.0	21.1	1	1.0	-7.9	-0.0
52-TE-132	-0.3	2.5	-0.5	1	1.0	-2.4	-0.0
53- I-132F	0.7	-5.0	2.2	1	1.0	0.0	-0.0

NUCLEIDE	T1/2	EBMT	EGT	NDK	RTYP	Q	B
52-TE-133M	0.0	0.0	-6.6	2	1.0	-11.3	-0.0
					3.0	0.0	-0.0
52-TE-133F	-0.6	0.0	-3.2	1	1.0	0.3	-0.0
53- I-133F	0.0	-0.4	1.7	2	1.0	0.0	11.4
					1.1	0.0	-79.3
54-XE-133F	-0.8	0.0	0.1	1	1.0	-0.0	-0.0
52-TE-134	-0.5	42.8	10.1	1	1.0	17.6	-0.0
53- I-134F	0.0	-7.6	1.4	1	1.0	0.0	-0.0
55-CS-134F	-0.3	-0.4	-0.7	2	1.0	0.0	-0.0
					2.0	100.0	100.0
53- I-135	1.7	0.0	14.7	2	1.0	-0.1	-0.0
					1.1	-0.2	0.0
54-XE-135F	-0.2	-0.1	0.2	1	1.0	0.1	-0.0
53- I-136F	0.0	0.0	12.3	1	1.0	10.0	-0.0
55-CS-136	0.8	-12.6	2.0	2	1.0	-1.1	-0.0
					1.1	1.3	0.0
54-XE-137	-0.3	-1.6	2.5	1	1.0	-0.1	-0.0
53- I-138	-0.6	34.9	-62.0	2	1.0	9.9	-0.0
					5.0	8.3	-0.0
54-XE-138	-0.2	-6.8	-1.1	1	1.0	-3.2	-0.0
55-CS-138M	0.0	-74.6	-77.5	2	1.0	0.2	-75.0
					3.0	100.0	100.0
55-CS-138F	0.0	-1.6	1.4	1	1.0	0.2	-0.0
54-XE-139	-1.7	-0.0	17.8	1	1.0	0.0	-0.0
55-CS-139	-3.4	0.0	5.1	1	1.0	0.0	-0.0
56-BA-139	-0.7	2.4	-14.5	1	1.0	2.3	-0.0
55-CS-140	-0.2	-4.3	-4.0	1	1.0	-4.0	-0.0
56-BA-140	-0.0	2.0	-8.1	1	1.0	0.0	-0.0
57-LA-140	-0.1	-0.4	4.7	1	1.0	100.0	100.0
54-XE-141	0.0	32.7	-66.4	2	1.0	2.5	0.1
					5.0	100.0	-20.4
55-CS-141	-0.4	31.8	-84.4	2	1.0	-1.4	0.0
					5.0	-58.0	-27.4
57-LA-141	1.5	-0.5	22.7	1	1.0	0.0	-0.0
58-CE-141	-0.2	-6.8	29.6	1	1.0	-0.0	-0.0
56-BA-142	0.0	16.3	32.9	1	1.0	0.0	-0.0
57-LA-142	0.3	-4.6	6.4	1	1.0	0.0	-0.0
59-PR-142F	-0.2	5.4	0.2	2	1.0	0.2	-0.0
					2.0	100.0	100.0
57-LA-143	6.7	41.8	-97.3	1	1.0	2.4	-0.0
58-CE-143	-0.9	-5.0	-3.1	1	1.0	0.0	-0.0
55-CS-144	3.8	25.7	-45.5	2	1.0	0.0	-0.0
					5.0	12.3	-0.0
58-CE-144	0.1	-0.6	-11.1	2	1.0	0.0	-0.0
					1.1	0.0	-0.0
59-PR-144M	0.0	-1.6	13.5	2	1.0	-0.0	0.0
					3.0	0.1	-0.0
59-PR-144F	0.1	0.1	-6.5	1	1.0	-0.0	-0.0
56-BA-145	-30.2	26.6	-88.7	1	1.0	-16.2	-0.0
57-LA-145	-17.2	26.1	-58.4	1	1.0	-10.8	-0.0
58-CE-145	-9.7	6.6	11.8	1	1.0	0.4	-0.0
59-PR-145	0.0	-0.1	43.0	1	1.0	0.0	-0.0
58-CE-146	0.0	-0.3	3.5	1	1.0	0.0	-0.0
59-PR-146	0.0	1.2	-0.5	1	1.0	0.0	-0.0
59-PR-147	4.0	-0.0	-49.7	1	1.0	0.0	-0.0
60-ND-147	0.8	0.6	28.8	1	1.0	0.0	-0.0
61-PM-147	-0.3	-0.1	-2.6	1	1.0	0.0	-0.0
61-PM-148M	0.0	0.5	-0.3	2	1.0	-5.3	1.2
					3.0	-0.0	-18.3

TABLE 1 cont'd

NUCLEIDE	T1/2	EBMT	EGT	NDK	RTYP	Q	B
61-PM-148F	-0.0	1.5	-8.7	1	1.0	-0.0	-0.0
59-PR-149	-2.2	-2.3	31.4	1	1.0	0.0	-0.0
60-ND-149	0.0	0.8	21.8	1	1.0	0.0	-0.0
61-PM-149	-0.1	-0.0	-34.9	1	1.0	-0.0	-0.0
61-PM-150	0.7	7.5	5.7	1	1.0	2.0	-0.0
60-ND-151	0.0	0.0	25.0	1	1.0	-1.1	-0.0
61-PM-151	-1.4	-6.4	5.1	1	1.0	0.0	-0.0
62-SM-151	-6.7	1.3	-12.0	1	1.0	0.1	-0.0
61-PM-152M	0.0	37.4	-0.1	1	1.0	0.0	-0.0
61-PM-152F	0.0	-5.4	9.4	1	1.0	-2.8	-0.0
63-EU-152F	-4.5	-79.5	49.7	2	1.0	0.1	-0.4
					2.0	1.4	0.1
61-PM-153	-1.9	-0.2	-15.7	1	1.0	0.0	-0.0
62-SM-153	0.0	0.0	-17.2	1	1.0	-0.0	-0.0
61-PM-154F	0.0	26.8	69.5	1	1.0	0.0	-0.0
63-EU-154	-9.1	-5.5	0.5	1	1.0	-0.2	-0.0
62-SM-155	0.9	35.9	-80.5	1	1.0	-1.6	-0.0
63-EU-155	-3.4	-18.1	2.0	1	1.0	-1.4	-0.0
62-SM-156	0.0	20.8	-52.0	1	1.0	3.4	-0.0
62-SM-157	0.0	35.8	-41.1	1	1.0	5.4	-0.0
63-EU-157	-0.3	26.2	-21.6	1	1.0	6.6	-0.0
63-EU-159	3.2	39.0	-66.9	1	1.0	2.3	-0.0
64-GD-159	-0.2	38.7	-84.3	1	1.0	3.2	-0.0
63-EU-160	3.4	37.9	22.1	1	1.0	18.4	-0.0
64-GD-161	0.0	26.7	-46.9	1	1.0	-2.5	-0.0
65-TB-161	-0.1	22.2	-67.2	1	1.0	1.8	-0.0
64-GD-162	-18.0	39.5	9.4	2	1.0	28.6	2.0
					-1.1	-100.0	-100.0
65-TB-162F	3.6	-23.6	6.4	1	1.0	-13.9	-0.0
65-TB-163	0.0	-13.0	13.7	1	1.0	1.2	-0.0
65-TB-164	0.0	-14.0	31.3	1	1.0	1.8	-0.0
66-DY-165M	0.2	-6.7	-57.9	2	1.0	-0.6	-10.4
					3.0	0.1	0.3
66-DY-165F	-1.1	41.9	-93.5	1	1.0	-0.7	-0.0
66-DY-166	0.1	0.9	5.4	1	1.0	0.0	-0.0
67-HO-166M	-0.3	-96.1	55.3	2	1.0	-10.8	-0.0
					3.0	100.0	100.0
67-HO-166F	-0.0	44.7	-91.6	1	1.0	0.9	-0.0

COMPARAISON TERMINEE POUR 242 NUCLEIDES COMPLETS

TABLE 2

Comparison between the data in the French File and ENDF/B-IV (%) for the 301 nuclides with partial or without experimental data

(Explanations see Table 1)

NUCLEIDE	T1/2	EBMT	EGT	NDK	RTYP	Q	B
CU- 72	-91.7			1	1.0	-0.8	-0.0
GA- 72	0.0			1	1.0	0.0	-0.0
CU- 73	-87.3			1	1.0	-0.0	-0.0
ZN- 73	0.0			1	1.0	3.2	-0.0
GE- 73M	0.0			1	3.0	-0.4	-0.0
ZN- 75	11.8			1	1.0	-3.9	-0.0
GE- 75M	-1.2			1	3.0	0.1	-0.0
ZN- 76	5.3			1	1.0	-0.5	-0.0
ZN- 77	0.0			1	1.0	-4.4	-0.0
GA- 77	-2.3			2	1.0	11.0	0.0
					1.1	-1.3	-0.0
SE- 77M	0.0			1	3.0	-35.2	-0.0
ZN- 78	-34.1			1	1.0	5.7	-0.0
GA- 78	3.7			1	1.0	1.7	-0.0
ZN- 79	-8.4			1	1.0	0.0	-0.0
GA- 79	4.7			2	1.0	10.4	0.1
					-5.0	-100.0	-100.0
SE- 79M	0.5			1	3.0	-61.8	-0.0
SE- 79F	-0.3			1	1.0	-7.8	-0.0
BR- 79M	0.4			1	3.0	-1.3	-0.0
GA- 80	0.0			2	1.0	-8.9	0.9
					-5.0	-100.0	-100.0
GE- 80	2.0			1	1.0	29.3	-0.0
BR- 80M	0.0			1	3.0	-1.2	-0.0
BR- 80F	1.1			2	1.0	0.0	0.4
					2.0	0.1	-4.7
GA- 81	42.7			1	1.0	-2.8	-0.0
GE- 81	0.0			1	1.0	-1.4	-0.0
AS- 81	10.6			1	1.0	0.0	-0.0
SE- 81M	-0.0			2	1.0	100.0	100.0
					3.0	0.1	-0.6
KR- 81M	-2.3			2	1.0	100.0	100.0
					3.0	0.0	-0.0
KR- 81F	-0.3			1	2.0	-5.3	-0.0
GA- 82	74.4			1	1.0	0.0	-0.0
GE- 82	0.0			1	1.0	0.0	-0.0
GA- 83	52.4			1	1.0	0.0	-0.0
GE- 83	0.0			2	1.0	2.2	0.2
					-5.0	-100.0	-100.0
AS- 83	-1.5			2	1.0	11.5	-35.0
					1.1	12.6	16.4
KR- 83M	0.0			1	3.0	-0.6	-0.0
GA- 84	50.6			1	1.0	0.0	-0.0
GE- 84	0.0			2	1.0	-10.3	9.6
					-5.0	-100.0	-100.0
AS- 84	-6.9			2	1.0	-0.9	-0.0
					5.0	49.6	-0.0
GE- 85	41.5			1	1.0	2.4	-0.0
AS- 85	1.0			2	1.0	0.0	-3.8
					5.0	0.1	13.0
BR- 85	1.6			2	1.0	100.0	100.0
					-1.1	-100.0	-100.0
GE- 86	0.4			1	1.0	1.7	-0.0
AS- 86	-11.1			2	1.0	-10.3	-0.0
					5.0	17.3	-0.0
SE- 86	-3.6			2	1.0	3.4	50.0
					-1.1	-100.0	-100.0
RB- 86M	-0.1			1	3.0	0.0	-0.0

TABLE 2 cont'd

NUCLEIDE	T1/2	EBMT	EGT	NDK	RTYP	Q	B
-AS- 87	0.0			2	1.0	0.0	31.0
					5.0	0.1	-100.0
-SE- 87	3.4			2	1.0	0.0	-0.0
					5.0	28.3	-0.0
-BR- 87	-0.2			2	1.0	5.4	-0.0
					5.0	100.0	-0.0
-RB- 87	0.1			1	1.0	-2.3	-0.0
-SR- 87M	0.7			2	2.0	6.1	-0.0
					3.0	0.2	-0.0
-AS- 88	31.6			1	1.0	-18.1	-0.0
-SE- 88	1.3			2	1.0	0.0	-0.0
					5.0	69.9	0.0
-SE- 89	0.0			2	1.0	0.0	-0.0
					5.0	-1.9	-0.0
-BR- 89	1.1			2	1.0	-7.4	-0.0
					5.0	11.4	0.0
-SR- 89	-2.4			1	1.0	-0.0	-0.0
- Y- 89M	2.2			1	3.0	-0.1	-0.0
-SE- 90	-63.9			1	1.0	0.0	-0.0
-BR- 90	18.4			2	1.0	4.1	-12.0
					5.0	-17.4	46.9
-SR- 90	-0.3			1	1.0	0.0	-0.0
-ZR- 90M	0.0			1	3.0	0.0	-0.0
-SE- 91	31.7			2	1.0	0.0	-21.0
					5.0	100.0	100.0
-BR- 91	-9.8			2	1.0	0.0	-3.1
					5.0	6.6	29.3
-KR- 91	-1.5			1	1.0	1.3	-0.0
- Y- 91M	0.0			1	3.0	-0.0	-0.0
-SE- 92	-59.6			1	1.0	0.0	-0.0
-BR- 92	17.8			2	1.0	0.0	26.0
					5.0	0.3	-100.0
- Y- 93M	-99.6			1	3.0	67.0	-0.0
-ZR- 93	37.7			2	1.0	22.2	-0.0
					1.1	34.0	-0.0
-NB- 93M	26.6			1	3.0	-0.0	-0.0
-KR- 94	-4.8			2	1.0	0.0	-3.9
					5.0	1.5	45.7
-KR- 95	-80.0			1	1.0	2.1	-0.0
-RB- 95	0.0			2	1.0	8.4	-0.0
					5.0	39.2	-0.0
-NB- 95M	-0.0			1	3.0	-0.4	-0.0
-RB- 96	26.1			2	1.0	7.5	-0.0
					5.0	0.0	-0.0
-SR- 96	-77.5			1	1.0	19.4	-0.0
-RB- 97	1.2			2	1.0	0.3	1.2
					5.0	-23.6	-4.5
-NB- 97M	10.0			2	1.0	100.0	100.0
					3.0	0.0	-0.0
-PB- 98	-6.4			2	1.0	0.0	26.0
					-5.0	-100.0	-100.0
-SR- 98	-23.5			2	1.0	7.6	0.5
					-5.0	-100.0	-100.0
-ZR- 98	-1.0			1	1.0	44.2	-0.0
-RB- 99	0.0			2	1.0	0.0	37.0
					-5.0	-100.0	-100.0
-SR- 99	6.7			2	1.0	0.0	-2.2
					5.0	100.0	100.0

NUCLEIDE	T1/2	EBMT	EGT	NDK	RTYP	Q	B
- Y- 99	46.7			2	1.0	-1.8	2.6
					5.0	-100.0	-68.4
-ZR- 99	0.0			1	1.0	-1.2	-0.0
-NB- 99M	3.8			1	1.0	-15.5	-0.0
-NB- 99F	6.7			1	1.0	-1.8	-0.0
-TC- 99M	0.0			1	3.0	-0.0	-0.0
- Y-100	5.5			1	1.0	0.0	-0.0
-ZR-100	0.0			2	1.0	22.0	50.0
					-1.1	-100.0	-100.0
-NB-100M	22.3			1	1.0	-1.7	-0.0
- Y-101	-18.0			1	1.0	1.9	-0.0
-ZR-101	-39.4			1	1.0	-6.2	-0.0
-NB-101	1.4			1	1.0	-0.7	-0.0
- Y-102	45.5			1	1.0	1.5	-0.0
-ZR-102	-97.2			1	1.0	-21.1	-0.0
-NB-102	33.3			1	1.0	-3.6	-0.0
-MO-102	3.5			1	1.0	13.5	-0.0
-ZR-103	-43.5			1	1.0	2.6	-0.0
-NB-103	-90.4			1	1.0	3.8	-0.0
-MO-103	11.8			1	1.0	-1.7	-0.0
-RH-103M	0.2			1	3.0	-0.0	-0.0
-ZR-104	-97.4			1	1.0	-28.4	-0.0
-NB-104F	-20.0			1	1.0	3.3	-0.0
-MO-104	-37.5			1	1.0	22.1	-0.0
-NB-105	35.7			1	1.0	5.3	-0.0
-MO-105	-33.3			1	1.0	-0.6	-0.0
-RH-105M	17.8			1	3.0	-0.0	-0.0
-NB-106	46.5			1	1.0	2.4	-0.0
-MO-106	-8.9			1	1.0	6.9	-0.0
-TC-106	-2.7			1	1.0	-8.3	-0.0
-RU-106	-0.0			1	1.0	-0.0	-0.0
-MO-107	-45.2			1	1.0	0.0	-0.0
-TC-107	-26.9			1	1.0	-14.6	-0.0
-PD-107M	3.2			1	3.0	1.9	-0.0
-PD-107F	-0.3			1	1.0	-5.4	-0.0
-MO-108	0.0			1	1.0	8.6	-0.0
-TC-108	-3.8			1	1.0	1.8	-0.0
-AG-108M	-2.6			2	2.0	0.4	-0.9
					3.0	-0.2	9.4
-TC-109	-97.2			1	1.0	0.0	-0.0
-RU-109F	-1.4			2	1.0	-1.1	50.9
					-1.1	-100.0	-100.0
-PD-109M	0.0			1	3.0	0.5	-0.0
-AG-109M	0.0			1	3.0	0.4	-0.0
-CD-109	-0.2			2	2.0	100.0	100.0
					2.1	21.1	-0.0
-MO-110	-89.4			1	1.0	0.0	-0.0
-TC-110	-1.2			1	1.0	-12.9	-0.0
-RU-110	-21.2			1	1.0	-3.7	-0.0
-RH-110F	-88.6			1	1.0	0.0	-0.0
-RU-111	-80.5			1	1.0	0.9	-0.0
-RH-111	-82.5			2	1.0	-13.8	0.4
					1.1	-12.7	-100.0
-AG-111M	12.2			2	1.0	100.0	100.0
					3.0	-8.0	-0.3
-CD-111M	1.8			1	3.0	-0.0	-0.0
-TC-112	-66.2			1	1.0	1.2	-0.0
-RU-112	80.6			1	1.0	-2.7	-0.0

TABLE 2 cont'd

NUCLEIDE	T1/2	EBMT	EGT	NDK	RTYP	Q	B
-RH-112	-68.1			1	1.0	2.2	-0.0
-RU-113	7.8			1	1.0	0.6	-0.0
-RH-113	0.0			1	1.0	1.5	-0.0
-AG-113M	-0.5			3	1.0	-8.4	2.2
					1.1	-9.8	-68.9
					3.0	100.0	100.0
-IN-113M	0.0			1	3.0	-0.3	-0.0
-RH-114	-76.5			1	1.0	2.7	-0.0
-IN-114M	-0.0			2	2.0	0.4	-0.0
					3.0	-0.0	-0.0
-PD-115	-21.1			2	1.0	1.9	-0.0
					1.1	7.4	-0.0
-AG-115M	0.0			2	1.0	-9.4	27.0
					-1.1	-100.0	-100.0
-IN-115F	-100.0			1	1.0	0.6	-0.0
-PD-116	-2.9			2	1.0	-7.4	50.0
					1.1	1.5	-100.0
-IN-116M	-99.9			2	-1.0	-100.0	-100.0
					3.0	100.0	100.0
-IN-116N	99.9			2	1.0	100.0	100.0
					-3.1	-100.0	-100.0
-PD-117	0.0			2	1.0	0.0	50.0
					1.1	4.4	-100.0
-AG-117M	0.7			2	1.0	-8.9	50.0
					-1.1	-100.0	-100.0
-AG-117F	-0.7			2	1.0	-3.7	-0.0
					1.1	-0.6	-0.0
-CD-117F	0.0			2	1.0	0.2	7.9
					1.1	0.6	-0.6
-IN-117M	-0.5			2	1.0	-0.8	-0.0
					3.0	0.4	-0.0
-SN-117M	0.0			1	3.0	-0.8	-0.0
-PD-118	0.0			2	1.0	0.0	50.0
					1.1	3.3	-100.0
-AG-118M	-30.0			2	1.0	-3.6	8.5
					3.0	-48.9	-10.9
-CD-118	-0.0			1	1.0	-7.5	-0.0
-IN-118N	0.0			3	1.0	100.0	100.0
					3.0	100.0	100.0
					-3.1	-100.0	-100.0
-PD-119	-35.7			1	1.0	-3.2	-0.0
-AG-119	-65.0			2	1.0	-1.5	50.0
					-1.1	-100.0	-100.0
-CD-119M	-42.7			2	-1.0	-100.0	-100.0
					1.1	0.0	50.0
-IN-119F	11.8			2	1.0	-6.4	5.0
					1.1	-6.3	-100.0
-SN-119M	0.0			1	3.0	0.6	-0.0
-PD-120	-91.8			1	1.0	-1.5	-0.0
-CD-120	0.0			2	1.0	-3.4	50.0
					-1.1	-100.0	-100.0
-AG-121	-73.3			1	1.0	-12.7	-0.0
-CD-121F	-6.2			2	1.0	-5.3	18.0
					-1.1	-100.0	-100.0
-IN-121M	12.2			2	1.0	-4.6	-1.2
					3.0	100.0	100.0
-SN-121F	1.0			1	1.0	1.6	-0.0
-AG-122	93.3			1	1.0	-8.4	-0.0

NUCLEIDE	T1/2	EBMT	EGT	NDK	RTYP	Q	B
- I-133M	0.0			1	3.0	84.0	-0.0
-XE-133M	-1.8			1	3.0	-0.0	-0.0
-SN-134	18.8			1	1.0	0.0	-0.0
-SB-134M	-92.1			2	1.0	-1.1	-0.0
					5.0	100.0	-0.0
-SB-134F	91.7			2	1.0	0.0	-0.1
					5.0	100.0	100.0
- I-134M	5.3			2	1.0	100.0	100.0
					3.0	0.2	-2.0
-CS-134M	0.3			2	1.0	100.0	100.0
					3.0	0.8	-1.0
-SN-135	-31.3			1	1.0	0.0	-0.0
-SB-135	0.0			2	1.0	0.0	-0.0
					5.0	51.4	0.0
-TE-135	0.0			1	1.0	4.5	-0.0
-XE-135M	2.2			1	3.0	-0.0	-0.0
-CS-135	23.1			1	1.0	0.5	-0.0
-BA-135M	0.0			1	3.0	0.1	-0.0
-SN-136	-51.6			1	1.0	0.0	-0.0
-SB-136	71.8			1	1.0	0.0	-0.0
-TE-136	-0.5			2	1.0	12.0	0.5
					-5.0	-100.0	-100.0
- I-136M	0.0			1	1.0	10.0	-0.0
-BA-136M	3.8			1	3.0	-0.4	-0.0
-SB-137	-64.7			1	1.0	-3.0	-0.0
-TE-137	0.0			2	1.0	0.0	-0.0
					5.0	85.6	0.0
- I-137	0.0			2	1.0	-4.7	-0.0
					5.0	68.5	-0.0
-CS-137	-0.1			2	1.0	-0.0	-1.9
					1.1	0.2	0.1
-BA-137M	0.1			1	3.0	0.0	-0.0
-SB-138	-23.3			1	1.0	0.0	-0.0
-TE-138	-2.5			2	1.0	0.0	-6.3
					5.0	100.0	100.0
-TE-139	39.5			1	1.0	0.0	-0.0
- I-139	8.0			2	1.0	-1.0	-0.0
					5.0	9.0	0.0
-TE-140	-33.5			1	1.0	0.0	-0.0
- I-140	-31.4			2	1.0	0.0	-14.7
					5.0	-1.5	23.8
-XE-140	0.0			1	1.0	13.5	-0.0
-TE-141	57.1			1	1.0	-0.2	-0.0
- I-141	16.7			2	1.0	0.0	12.0
					5.0	0.6	-100.0
-BA-141	-1.6			1	1.0	0.0	-0.0
-XE-142	0.0			2	1.0	11.4	-0.0
					5.0	32.5	-11.8
-CS-142	0.0			2	1.0	-2.3	-0.1
					5.0	0.4	26.3
-CE-142	100.0			1	4.0	-9.0	-0.0
-PR-142M	0.0			1	3.0	-98.5	-0.0
-XE-143	68.8			2	1.0	0.0	1.1
					5.0	-0.5	-100.0
-CS-143	4.5			2	1.0	-1.4	-0.0
					5.0	65.9	-0.0
-BA-143	-11.8			1	1.0	-1.4	-0.0
-PR-143	0.1			1	1.0	0.1	-0.0

TABLE 2 cont'd

NUCLEIDE	T1/2	EBMT	EGT	NDK	RTYP	Q	B
-XE-144	-65.0			1	1.0	0.0	-0.0
-BA-144	-2.7			1	1.0	7.2	-0.0
-LA-144	5.7			1	1.0	-5.4	-0.0
-ND-144	-0.3			1	4.0	-3.4	-0.0
-XE-145	0.0			1	1.0	-11.8	-0.0
-CS-145	0.5			2	1.0	0.5	4.4
					-5.0	-100.0	-100.0
-CS-146	-0.5			2	1.0	0.0	3.9
					-5.0	-100.0	-100.0
-BA-146	-9.1			1	1.0	0.0	-0.0
-LA-146F	17.0			1	1.0	-7.8	-0.0
-XE-147	-62.1			1	1.0	-0.4	-0.0
-CS-147	-60.9			1	1.0	0.4	-0.0
-BA-147	-67.7			1	1.0	1.4	-0.0
-LA-147	-60.0			1	1.0	5.1	-0.0
-CE-147	-21.4			1	1.0	-7.2	-0.0
-SM-147	-100.0			1	4.0	-3.5	-0.0
-BA-148	-92.0			1	1.0	1.3	-0.0
-LA-148	23.5			1	1.0	0.0	-0.0
-CE-148	10.4			1	1.0	11.7	-0.0
-PR-148F	19.5			1	1.0	-18.5	-0.0
-LA-149	-91.3			1	1.0	0.7	-0.0
-CE-149	66.7			1	1.0	-0.8	-0.0
-SM-149	-96.0			1	4.0	-2.9	-0.0
-LA-150	-38.3			1	1.0	0.8	-0.0
-CE-150	83.1			1	1.0	0.0	-0.0
-PR-150	-50.8			1	1.0	-1.8	-0.0
-CE-151	-66.0			1	1.0	-1.5	-0.0
-PR-151	0.0			1	1.0	-6.4	-0.0
-PR-152	-95.2			1	1.0	0.5	-0.0
-ND-152	-0.9			1	1.0	17.9	-0.0
-EU-152M	-99.7			3	-1.0	-100.0	-100.0
					-2.0	-100.0	-100.0
					3.0	100.0	100.0
-EU-152N	82.8			3	1.0	100.0	100.0
					2.0	100.0	100.0
					-3.0	-100.0	-100.0
-GD-152	-0.3			1	4.0	-4.2	-0.0
-PR-153	-96.1			1	1.0	2.7	-0.0
-ND-153	-79.3			1	1.0	2.4	-0.0
-GD-153	0.5			1	2.0	1.6	-0.0
-ND-154	-100.0			1	1.0	0.0	-0.0
-PM-154M	-98.3			2	1.0	-13.9	10.0
					-3.0	-100.0	-100.0
-ND-155	-95.4			1	1.0	1.8	-0.0
-PM-155	-89.1			1	1.0	-1.0	-0.0
-PM-156	-87.0			1	1.0	6.4	-0.0
-EU-156	-0.0			1	1.0	0.0	-0.0
-PM-157	-98.5			1	1.0	0.0	-0.0
-SM-158	-95.0			1	1.0	-3.5	-0.0
-EU-158	0.0			1	1.0	-2.0	-0.0
-SM-159	-1.4			1	1.0	5.8	-0.0
-TB-160	0.0			1	1.0	1.6	-0.0
-EU-161	-40.6			1	1.0	6.3	-0.0
-EU-162	-98.1			1	1.0	5.9	-0.0
-TB-162M	-1.3			1	1.0	-19.6	-0.0
-GD-163	-99.4			1	1.0	-6.8	-0.0
-GD-164	-99.9			1	1.0	-6.0	-0.0

TABLE 3a

Half lives of nuclides that are only in the French File

NUCLEIDE	T1/2	NUCLEIDE	T 1/2
27-CO- 72	1.2266E-01	42-MO-111	3.9174E-01
28-NI- 72	2.4193E 00	43-TC-111	1.3358E 00
27-CO- 73	1.1551E-01	41-NB-112	8.5105E-02
28-NI- 73	3.9355E-01	42-MO-112	6.8924E-01
27-CO- 74	1.0752E-01	42-MO-113	1.9715E-01
28-NI- 74	6.4833E-01	43-TC-113	4.5835E-01
29-CU- 74	5.7315E-01	42-MO-114	3.2152E-01
27-CO- 75	8.0160E-02	43-TC-114	1.7340E-01
28-NI- 75	1.7963E-01	44-RU-114	5.0528E 00
29-CU- 75	7.6657E-01	42-MO-115	1.1596E-01
28-NI- 76	2.6838E-01	43-TC-115	2.2249E-01
29-CU- 76	2.2110E-01	44-RU-115	7.2941E-01
28-NI- 77	1.0281E-01	45-RH-115	6.0225E 00
29-CU- 77	2.9458E-01	43-TC-116	1.0620E-01
28-NI- 78	1.3765E-01	44-RU-116	1.4049E 00
29-CU- 78	1.2063E-01	45-RH-116	8.3326E-01
29-CU- 79	1.4744E-01	43-TC-117	1.3523E-01
29-CU- 80	9.1104E-02	44-RU-117	3.0891E-01
30-ZN- 80	7.1135E-01	45-RH-117	1.0761E 00
29-CU- 81	7.4469E-02	43-TC-118	7.7217E-02
30-ZN- 81	1.2937E-01	44-RU-118	6.1633E-01
30-ZN- 82	1.3526E-01	45-RH-118	2.9529E-01
30-ZN- 83	8.3858E-02	44-RU-119	1.7711E-01
31-GA- 85	9.1971E-02	45-RH-119	4.4774E-01
34-SE- 85M	1.9000E 01	44-RU-120	2.9316E-01
35-BR- 86M	4.5000E 00	45-RH-120	1.6241E-01
32-GE- 87	1.2551E-01	45-RH-121	2.2103E-01
32-GE- 88	1.4271E-01	46-PD-121	6.2214E-01
33-AS- 89	1.2942E-01	45-RH-122	1.0533E-01
33-AS- 90	9.0087E-02	46-PD-122	1.2701E 00
34-SE- 93	1.0677E-01	45-RH-123	1.3346E-01
35-BR- 93	2.0121E-01	46-PD-123	3.1002E-01
35-BR- 94	1.1050E-01	46-PD-124	5.6012E-01
35-BR- 95	1.1662E-01	47-AG-124	2.6853E-01
35-BR- 96	8.3793E-02	46-PD-125	1.8309E-01
36-KR- 96	4.4038E-01	46-PD-126	2.8703E-01
36-KR- 97	1.4852E-01	47-AG-126	1.5546E-01
36-KR- 98	2.2430E-01	47-AG-127	2.0518E-01
37-RB-100	1.0056E-01	48-CD-127	6.5903E-01
38-SR-100	1.0456E 00	47-AG-128	1.0245E-01
37-RB-101	1.1330E-01	48-CD-129	3.3773E-01
38-SR-101	2.5190E-01	48-CD-130	5.2403E-01
38-SR-102	4.1475E-01	48-CD-131	1.1926E-01
38-SR-103	1.3859E-01	48-CD-132	1.4479E-01
39- Y-103	3.6600E-01	49-IN-133	1.1392E-01
38-SR-104	1.9250E-01	49-IN-134	7.7543E-02
39- Y-104	1.4422E-01	54-XE-134M	2.9000E-01
39- Y-105	1.7362E-01	55-CS-135M	3.1800E 03
40-ZR-105	5.5859E-01	57-LA-138	3.3113E 18
39- Y-106	9.2921E-02	51-SB-139	1.7192E-01
40-ZR-106	9.8013E-01	52-TE-142	4.9127E-01
39- Y-107	1.0463E-01	53- I-142	1.9604E-01
40-ZR-107	2.4847E-01	53- I-143	3.2815E-01
41-NB-107	6.6943E-01	53- I-144	1.3270E-01
40-ZR-108	4.0762E-01	53- I-145	1.8670E-01
41-NB-108	2.2202E-01	54-XE-146	9.3718E-01
40-ZR-109	1.3368E-01	55-CS-148	2.0163E-01
41-NB-109	2.9607E-01	62-SM-148	2.5229E 23
42-MO-109	1.0328E 00	55-CS-149	2.7822E-01
45-RH-109M	5.0000E 01	56-BA-149	9.1747E-01
41-NB-110	1.2584E-01	55-CS-150	1.2437E-01
41-NB-111	1.5608E-01	56-BA-150	1.7975E 00

TABLE 3b

Half lives of nuclides that are only in ENDF/B-IV

NUCLEIDE	T1/2	NUCLEIDE	T 1/2
29-CU- 71	1.0000E 00	56-BA-151	4.3684E-01
30-ZN- 71M	1.4112E 04	57-LA-151	9.5359E-01
30-ZN- 71F	1.4400E 02	56-BA-152	7.5484E-01
39- Y- 96M	6.0000E 00	57-LA-152	3.0940E-01
39- Y- 97M	1.2100E 00	58-CE-152	1.4034E 01
39- Y- 98M	2.0000E 00	61-PM-152N	1.0800E 03
43-TC- 98	1.3209E 14	57-LA-153	4.3713E-01
43-TC- 99F	6.6988E 12	58-CE-153	1.7251E 00
41-NB-104M	4.8000E 00	57-LA-154	1.7533E-01
47-AG-107M	4.4300E 01	58-CE-154	3.5909E 00
44-RU-109M	1.2900E 01	59-PR-154	1.3072E 00
47-AG-114M	1.2000E 02	57-LA-155	2.2155E-01
47-AG-120M	3.2000E-01	58-CE-155	7.1253E-01
48-CD-121M	4.5000E 00	59-PR-155	1.8907E 00
49-IN-129M	1.2000E 00	58-CE-156	1.1624E 00
50-SN-130M	1.0200E 02	59-PR-156	5.1044E-01
49-IN-131M	2.7000E-01	60-ND-156	5.8494E 01
50-SN-131M	3.9000E 01	58-CE-157	3.6169E-01
53- I-132M	5.0160E 03	59-PR-157	6.7790E-01
57-LA-146M	6.2000E 00	60-ND-157	4.1488E 00
59-PR-148M	1.3400E 02	59-PR-158	2.6290E-01
65-TB-166	5.2000E 01	60-ND-158	7.8886E 00
66-DY-167	3.7200E 02	61-PM-158	3.8012E 00
67-HO-167	1.1160E 04	59-PR-159	3.1408E-01
66-DY-168	5.7600E 03	60-ND-159	1.4078E 00
67-HO-168	1.7940E 02	61-PM-159	4.2296E 00
66-DY-169	7.1000E 00	60-ND-160	2.1207E 00
67-HO-169	2.6400E 02	61-PM-160	9.9631E-01
68-ER-169	8.0352E 05	62-SM-160	3.4913E 02
66-DY-170	9.6000E 01	60-ND-161	5.5577E-01
67-HO-170M	1.6800E 02	61-PM-161	1.1882E 00
67-HO-170F	4.3000E 01	62-SM-161	1.2875E 01
69-TM-170	1.1232E 07	61-PM-162	3.9986E-01
RESTENT 33 NUCLEIDES SUR FICHIER 1		62-SM-162	1.9588E 01
		62-SM-163	2.5631E 00
		63-EU-163	1.4843E 01
		62-SM-164	4.2471E 00
		63-EU-164	2.1701E 00
		62-SM-165	9.2740E-01
		63-EU-165	2.5483E 00
		64-GD-165	1.0022E 02
		66-DY-165N	3.2000E 01

Annex n° 4

Half lives of fission products from the ENSDF file, July 1977

W.B. Ewbank, ORNL, USA

Table 1 Sorted by nuclides
Table 2 Sorted by half lives (only half lives >1 ms are listed here)

Table 1

Fission product half lives sorted by nuclide

Nucleus	$T_{1/2}$	E(level)	Jπ
^{72}Ga	<0.19 ns	208.42 5	1+
^{72}Ga	0.59 ns 3	161.44 7	1+
^{72}Ga	39.86 ms 13	119.58 8	
^{72}Ga	39.2 ns 7	16.44 5	2
^{72}Ga	14.1 h 2	0.0	3-
^{72}Ge	2.8 ps 7	1463.93 4	2+
^{72}Ge	3.3 ps 8	834.01 2	2+
^{72}Ge	0.43 μs 2	691.2 2	0+
^{72}Zn	46.5 h	0.0	0+
^{73}Ga	4.91 h 5	0.0	(3/2)-
^{73}Ge	0.53 s 5	66.70 2	1/2-
^{73}Ge	2.95 μs 5	13.26 2	5/2+
^{73}Ge	1.60 ns 14	68.75 1	(7/2)+
^{73}Zn	23.5 s 10	0.0	
^{74}Ga	< 0.5 μs	56.5 1	(3,4)-
^{74}Ga	8.1 m 1	0.0	(4)-
^{74}Ga	9.5 s 10	59.7 2	1+
^{74}Ge	11.8 ps 11	595.88 4	2+
^{74}Ge	5.4 ps 8	1204.31 4	2+
^{74}Zn	95 s 1	0.0	0+
^{75}As	1.3 fs 4	7646 5	1/2
^{75}As	1.7 ns 1	400.646 9	5/2+
^{75}As	17.53 ms 8	303.916 12	9/2+
^{75}As	0.28 ns 1	279.528 8	5/2-
^{75}As	11.8 ps 7	264.651 8	3/2-
^{75}As	0.87 ns 3	198.596 7	1/2-
^{75}Ga	2.17 m 10	0	(3/2-)
^{75}Ge	82.78 m 4	0.0	1/2-
^{75}Ge	48.3 s 6	139.3 2	(7/2)+
^{75}Zn	9 s 2	0	
^{76}As	26.32 h 7	0.0	2-
^{76}As	1.93 μs 6	46.043 9	(1,2)+
^{76}Ga	27.1 s 2	0.0	(3-)
^{76}Ge	18.2 ps 4	562.92 3	2+
^{76}Ge	9.1 ps 10	1108.45 4	2+
^{76}Zn	5.7 s 3	0.0?	0+
^{77}As	38.8 h 3	0.0	3/2-
^{77}As	75 ps 15	631.8 4	(5/2)
^{77}As	116 μs 4	475.0 3	(9/2)+
^{77}As	0.48 ns	264.5 5	(5/2-)
^{77}As	7.4 ns 3	194.9 5	(1/2-)
^{77}Ga	17.1 s 15	0.0	
^{77}Ge	11.30 h 1	0.0	(7/2+)
^{77}Ge	54 s 1	159.7 1	(1/2)-
^{77}Se	23 ps 2	439.6 5	5/2-
^{77}Se	9.3 ns 4	249.7 5	5/2-
^{77}Se	27 ps 7	239.0 5	3/2-
^{77}Se	17.5 s 1	161.9 5	7/2+
^{77}Zn	1.4 s 3	0.0?	
^{78}As	90.7 m 2	0.0	(2-)
^{78}Ga	4.8 s 13	0.0	
^{78}Ge	87 m 1	0.0	0+
^{78}Se	2.8 ps 3	1308.5 1	(2)+

Table 1 cont'd

Nucleus	$T_{1/2}$	E(level)	J^π
^{78}Se	8.8 ps 5	613.63 13	2+
^{79}As	9.01 m 15	0.0	3/2-
^{79}Br	0.20 ps 3	831.95 7	(3/2)-
^{79}Br	2.0 ps 2	605.97 6	(3/2)-
^{79}Br	1.3 ps 3	523.00 11	(5/2)-
^{79}Br	13 ps 3	397.44 6	(3/2)-
^{79}Br	8.3 ps 17	306.46 8	1/2-
^{79}Br	34 ps 23	217.03 6	5/2-
^{79}Br	4.88 s 5	207.2 4	9/2+
^{79}Br	1.2 ps 2	761.2 4	7/2-
^{79}Ga	3.00 s 9		
^{79}Ge	19.1 s 3		
^{79}Ge	82 s 2	0.0	(1/2)-
^{79}Se	≤ 65000. y	0.0	7/2+
^{79}Se	3.91 m 5	95.9 3	1/2-
^{80}As	16.5 s 3	0	1(+)
^{80}Br	17.4 m 2	0	1+
^{80}Br	7.37 ns 21	37.052	2-
^{80}Br	4.42 h 1	85.95 40	5-
^{80}Ga	1.7 s 2	0	
^{80}Ge	24.5 s 10	0	0+
^{80}Se	7.7 ps 2	666.41	2+
^{80}Se	1.01 ps 26	1449.50	2+
^{81}As	33 s	0.0	
^{81}Br	36 μs	536.1	9/2+
^{81}Ge	10.1 s	0	
^{81}Se	18.5 m	0.0	(1/2)-
^{81}Se	57.25 m	102.89	(7/2)+
^{82}As	21 s 2		(1+)
^{82}As	13.0 s 6		
^{82}Br	35.30 h 2	0.0	5-
^{82}Br	6.13 m 8	46 2	2-
^{82}Ge	4.6 s 4	0.0	0+
^{82}Se	1.4×10^{20} y	0.0	0+
^{82}Se	11 ps 1	654.8 2	2+
^{83}As	14.1 s 11	0.0	
^{83}Br	4.1 ns 4	1091.83 7	
^{83}Br	2.39 h 2	0.0	(3/2)-
^{83}Ge	1.9 s 4	0.0	
^{83}Kr	147 ns 4	9.40 1	7/2+
^{83}Kr	1.83 h 2	41.56 2	1/2-
^{83}Se	22.5 m 2	0.0	(9/2)+, 2.44
^{83}Se	70.4 s 3	220	(1/2)-, 0.13
^{84}As	5.8 s 5	0.0	
^{84}Br	6.0 m 2	320	
^{84}Br	31.80 m 8	0.0	2-
^{84}Kr	3.2 ps 5	881.5 1	2+
^{84}Se	3.3 m 2	0.0	0+
^{85}As	2.028 s 12	0.0	
^{85}Br	3.00 m 5	0.0	1/2-,3/2-
^{85}Kr	10.73 y 6	0.0	9/2+
^{85}Kr	4.36 h 9	305.0 2	1/2-
^{85}Rb	0.958 μs 12	513.98 3	(9/2)+
^{85}Rb	0.74 ns 5	151.28 10	3/2-
^{85}Se	39 s 4	0.0	
^{86}Br	54 s 2	0.0	1,2-
^{86}Rb	18.66 d 2	0.0	2-
^{86}Rb	1.017 m 3	555.9 3	(6)-
^{86}Se	16.7 s 3	0.0	
^{87}As	0.6 s 3	0.0	
^{87}Br	55.7 s	0.0	
^{87}Kr	76.4 m 10	0.0	(5/2+)
^{87}Rb	0.10 ns 5	403.0 5	5/2-
^{87}Rb	4.7×10^{10} y 1	0.0	3/2-
^{87}Se	5.8 s 3	0.0	
^{88}Br	16.3 s 3	0.0	(1-)
^{88}Kr	2.84 h 3	0.0	0+
^{88}Rb	17.8 m 1	0.0	2-
^{88}Se	1.53 s 6	0	0+
^{88}Sr	0.35 ps 21	2734.060 42	3-
^{88}Sr	0.119 ps 16	1836.035 37	2+
^{89}Br	4.53 s 10	0.0	
^{89}Kr	3.07 m 9	0.0	(5/2+)
^{89}Rb	15.2 m 1	0.0	(3/2-)
^{89}Se	0.41 s 4	0.0	
^{89}Sr	50.55 d 9	0.0	5/2+
^{89}Sr	0.33 ns 7	2079.0 5	(11/2)-
^{90}Br	1.71 s 14	0.0	
^{90}Kr	32.32 s 9	0.0	0+
^{90}Rb	153 s 3	0.0	(1-)
^{90}Rb	258 s 5	106.92 4	(4-)

Table 1 cont'd

' fission product half-lives ' continued

Nucleus	$T_{1/2}$	E(level)	Jπ
^{90}Sr	28.6 y 3	0.0	0+
^{90}Y	64.1 h 1	0.0	2-
^{90}Y	250 ps 7	202.51 3	3-
^{90}Y	3.19 h 1	682.04 5	7+
^{90}Zr	61.3 ns 25	1760.7 6	0+
^{90}Zr	125 ns 6	3589.44 19	8+
^{90}Zr	809.2 ns 20	2319.10 15	5-
^{90}Zr	84 fs 7	2186.50 15	2+
^{90}Zr	11.9 fs 4	3843 2	2+
^{91}Br	0.63 s 7	0.0	
^{91}Kr	8.6 s	0.0	
^{91}Rb	58 s	0	
^{91}Sr	9.5 h	0	5/2+
^{91}Y	49.71 m 4	555.57 5	9/2+
^{91}Y	58.51 d 6	0.0	1/2-
^{91}Zr	0.8 ps +6-3	1204.9 8	1/2+
^{91}Zr	1.5 ns 5	2170.1 9	(11/2)-
^{91}Zr	29.0 ns 8	2300	(15/2-)
^{92}Br	0.25 s 10	0.0	
^{92}Kr	1.840 s 8	0.0	
^{92}Rb	4.48 s 3	0.0	
^{92}Sr	2.71 h 1	0.0	0+
^{92}Y	3.54 h 1	0.0	2-
^{92}Zr	5.0 ps 4	934.46 7	2+
^{92}Zr	166 ps 28	1383.0 1	0+
^{93}Kr	1.289 s 12	0.0	
^{93}Nb	4.5 ps	949.9 2	13/2+
^{93}Nb	6.2 ps 3	808.7 2	5/2+
^{93}Nb	0.53 ps 11	744.0 2	7/2+
^{93}Nb	0.25 ps +26-11	686 1	(3/2)-
^{93}Nb	< 0.22 ps	1297.4 2	(9/2)+
^{93}Nb	0.31 ps 7	979.1 2	11/2+
^{93}Nb	13.6 y 3	30.4 3	1/2-
^{93}Rb	5.8 s 1	0.0	
^{93}Sr	7.6 m 2	0.0	(5/2+)
^{93}Y	10.1 h 2	0.0	1/2-
^{93}Y	≤ 0.13 ns	590.0	(3/2)-
^{93}Zr	1.45 ns 5	266.87 5	3/2+
^{93}Zr	1.53×10^{6} y	0.0	5/2+
^{94}Kr	0.20 s 1	0.0	0+
^{94}Nb	6.26 m 1	40.95 4	3+
^{94}Nb	36.4 ns 8	140.35 4	
^{94}Nb	2.03×10^{4} y 16	0.0	6+
^{94}Rb	2.69 s 4	0.0	
^{94}Sr	78 s 2	0.0	0+
^{94}Y	19.1 m 4	0.0	2-
^{94}Zr	7.7 ps	918.24 23	2+
^{94}Zr	320 ps	1299.99 29	0+
^{95}Kr	≈ 1 s	0.07	
^{95}Mo	2.05 ps 100	765.83 4	7/2+
^{95}Mo	0.68 ps 18	1074.1 2	(7/2)+
^{95}Mo	1.98 ps 50	947.8 2	9/2+
^{95}Mo	0.77 ns 1	203.94 8	3/2+
^{95}Nb	35.15 d 3	0.0	(9/2)+
^{95}Nb	86.6 h 8	234.70 14	(1/2)-
^{95}Rb	0.36 s 2	0.0	
^{95}Sr	26 s 1	0.0	
^{95}Y	10.7 m 2	0.0	(1/2-)
^{95}Zr	63.98 d 6	0.0	(5/2)+
^{96}Mo	0.41 ps 12	1869.5	4+
^{96}Mo	3.49 ps 16	778.26	2+
^{96}Mo	1.2 ps 3	1628.22	4+
^{96}Mo	60 ps 8	1147.9	(0+)
^{96}Mo	1.4 ps 4	1625.93	2+
^{96}Mo	0.7 ps 1	1497.82	2+
^{96}Nb	23.35 h 5	0.0	(6+)
^{96}Rb	0.207 s 3	0.0	
^{96}Sr	4.0 s 2	0.0	
^{96}Y	2.3 m 1	0.0	
^{97}Kr	< 0.1 s	0.0	

Table 1 cont'd

' fission product half-lives ' continued

Nucleus	$T_{1/2}$	E(level)	Jπ
^{97}Mo	1.2 ps 2	1116.7	9/2+
^{97}Mo	2.8 ps 3	888.2 4	1/2+
^{97}Mo	5.9 ps 6	480.9 1	(3/2+)
^{97}Mo	30 ps 2	679.6 1	(1/2+)
^{97}Mo	2.0 ps 5	657.92 10	7/2+
^{97}Mo	1.4 ps 3	1268.63 10	7/2+
^{97}Mo	1.4 ps 2	1024.53 30	7/2+
^{97}Nb	72.1 m 7	0.0	(9/2)+
^{97}Nb	60 s 8	743.36 10	(1/2)-
^{97}Rb	0.176 s 6	0.0	
^{97}Sr	≤ 0.2 s	0.0	
^{97}Y	1.11 s 14	0.0	
^{97}Zr	17.0 h 2	0.0	1/2+
^{97}Zr	17.0 h 2	0.0	1/2+
^{98}Mo	21.8 ns	734.9	0+
^{98}Nb	51.5 m 10		(4,5)
^{98}Nb	2.8 s 2		(1+)
^{98}Rb	0.136 s 8	0	
^{98}Sr	0.845 s 43	0	0+
^{98}Y	0.83 μs 10	300?	
^{98}Y	≈ 0.3 s	0	
^{98}Zr	31 s 3	0	0+
^{99}Mo	16.8 μs 16	98 1	(3/2,5/2)+
^{99}Mo	66.02 h 1	0	1/2+
^{99}Nb	14.3 s 6	0	(9/2)+
^{99}Nb	2.6 m 2	369 5	(1/2)-
^{99}Rb	0.076 s 5	0	
^{99}Ru	20.6 ns 2	89.36 4	3/2+
^{99}Sr	unknown	0.0?	
^{99}Tc	≤ 0.1 ns	920.5 1	(1/2,3/2)+
^{99}Tc	3.61 ns 7	181.07 5	(5/2)+
^{99}Tc	6.02 h 2	142.63 3	(1/2)-
^{99}Tc	196 ps 10	140.508 4	(7/2)+
^{99}Tc	2.13×10^5 y 5	0	9/2+
^{99}Y	0.8 s 7	0.0	
^{99}Zr	0.36 μs 5	≈ 120?	
^{99}Zr	2.4 s 1	0.0	
^{100}Mo	12.2 ps 6	535.6 3	2+
^{100}Mo	1.7 ns 2	694.4 6	0+
^{100}Mo	3.7 ps 3	1136.1 4	4+
^{100}Mo	5.4 ps 4	1063.7 3	2+
^{100}Nb	1.5 s 3		
^{100}Nb	3.1 s 3		
^{100}Ru	10.8 ps 8	539.59 5	2+
^{100}Ru	1.16 ps 16	1362.1 1	2+
^{100}Ru	254 ps 25	1226.5 3	4+
^{100}Tc	15.8 s 1	0.0	1+
^{100}Zr	7.1 s 4	0.0	0+
^{101}Mo	14.62 m 5	0.0	1/2+
^{101}Nb	7.0 s 3	0.0	
^{101}Ru	0.20 ns 4	325.2 2	1/2+
^{101}Ru	590 ps 30	127.22 5	3/2+
^{101}Ru	22.5 μs 5	526.8 6	(11/2-)
^{101}Tc	14.2 m 1	0.0	(9/2+)
^{101}Tc	760 μs 50	207.53 4	(1/2-)
^{101}Zr	3.3 s 6	0.0	
^{102}Mo	11.1 m 3	0	0+
^{102}Nb	2.9 s 4	0.0	
^{102}Ru	17.7 ps 16	475.07 4	2+
^{102}Ru	28 ps 5	1106.37 6	4+
^{102}Ru	3.8 ps 6	1103.15 5	2+
^{102}Tc	5.28 s 15	0	(1+)
^{102}Tc	4.35 m 7	< 500	
^{102}Y	0.9 s 3	0?	
^{102}Zr	0.8 s 3	0	0+
^{102}Zr	2.2 ns 2	151.8 2	(2+)
^{103}Mo	60 s 2	0	
^{103}Rh	1.11 ns 3	93.035 7	(9/2)+
^{103}Rh	56.12 m 1	39.750 7	7/2+
^{103}Rh	5.1 ps 4	294.98 1	3/2-

Table 1 cont'd

' fission product half-lives ' continued

Nucleus	$T_{1/2}$	E(level)	Jπ
^{103}Rh	71 ps 5	357.46 4	5/2-
^{103}Rh	0.9 ps 6	1277.2 5	3/2-
^{103}Rh	5.5 ps 5	920.0 4	(9/2)-
^{103}Rh	2.9 ps 2	880.4 3	5/2-
^{103}Ru	1.56 ms 5	237 5	(11/2)-
^{103}Ru	39.35 d 5	0.0	5/2+
^{103}Tc	50 s 2	0.0	
^{104}Mo	1.3 m 3	0.0	0+
^{104}Rh	< 0.6 ns	97.114 3	2+,1+
^{104}Rh	4.34 m 5	128.956 10	5+,4+
^{104}Rh	2.6 ms 2	51.422 2	2-,1-
^{104}Rh	42.3 s 4	0.0	1+
^{104}Ru	7.6 ps 8	988.1 3	(0+)
^{104}Ru	6.6 ps 12	893.0 1	2+
^{104}Ru	6.4 ps 12	888.5 1	4+
^{104}Ru	53.2 ps 24	357.99 3	2+
^{104}Tc	18.2 m 5	0.0	
^{105}Mo	42. s 3	0.0	
^{105}Nb	1.8 s?	0.0?	
^{105}Pd	94 ps 10	644.50 6	(7/2)-
^{105}Pd	36 μs 1	489.11 7	(11/2)-
^{105}Pd	33 ps 3	280.51 4	3/2+
^{105}Pd	3.8 ps 2	442.23 6	(5/2)+
^{105}Pd	1.8 ps 3	781.3 5	(5/2)+
^{105}Pd	1.01 ns 5	344.52 2	(1/2)+
^{105}Pd	25 ps 6	319.18 6	(5/2)+
^{105}Rh	≤ 0.2 ns	808.1 1	(3/2)+
^{105}Rh	< 0.3 ns	149.2 1	(9/2)+
^{105}Rh	≤ 0.2 ns	724.5 1	(5/2)+
^{105}Rh	≤ 0.4 ns	469.4 1	(5/2+)
^{105}Rh	45 s	129.7 1	(1/2)-
^{105}Rh	35.36 h 5	0.0	(7/2)+
^{105}Ru	4.44 h 5	0.0	(3/2+)
^{105}Ru	340 ns	20.57 11	(5/2)+
^{105}Tc	7.8 m 8	0.0	
^{106}Mo	9.5 s 5	0.0	0+
^{106}Pd	1.4 ps 2	1229.20 5	4+
^{106}Pd	≤ 3.6 ns	2757.3 5	(5)+
^{106}Pd	0.5 ps 2	2084.27 4	3-
^{106}Pd	11.3 ps 6	511.85 3	2+
^{106}Pd	3.0 ps 5	1128.02 4	2+
^{106}Pd	≤ 2.0 ns	2952.01 5	(5)+
^{106}Pd	6.8 ps 15	1133.6 2	0+
^{106}Rh	29.9 s 1	0.0	1+
^{106}Rh	132 m 1	≈ 138	(4,5,6)+
^{106}Ru	368.2 d 12	0.0	0+
^{106}Tc	36 s 1	0.0	
^{107}Ag	2.85 ns 10	125.7 4	(9/2)+
^{107}Ag	5.2 ps 9	324.6 1	3/2-
^{107}Ag	44.3 s 2	93.08 5	7/2+
^{107}Ag	40 ps 3	422.6 3	5/2-
^{107}Ag	1.30 ps 17	949.0 3	5/2-
^{107}Ag	0.27 ps 8	786.5 7	3/2-
^{107}Pd	0.85 μs 10	115.7 2	1/2+
^{107}Pd	21.3 s 3	214 1	(11/2)-
^{107}Pd	6.5×10^{6} y 3	0	(5/2)+
^{107}Rh	21.7 m 4	0.0	(5/2,7/2)+
^{107}Ru	4.2 m 3	0.0	
^{107}Tc	29 s 3	0.0	
^{108}Ag	127 y	109.58 8	(6)+
^{108}Ag	2.37 m 1	0.0	1+
^{108}Mo	1.5 s 5	0.0?	0.0
^{108}Pd	2.3 ps 3	1048.32 10	(4)+
^{108}Pd	23.8 ps 17	433.95 4	2+
^{108}Pd	5.3 ps 10	931.2 3	2+
^{108}Pd	4.6 ps 10	1052.80 5	(0)+
^{108}Rh	16.8 s 5	0.0	
^{108}Rh	5.9 m 2	0.0	
^{108}Ru	4.5 m 2	0.0	0+
^{108}Tc	8.3 s	0.0	

Table 1 cont'd

' fission product half-lives ' continued

Nucleus	$T_{1/2}$	E(level)	Jπ
^{109}Ag	39.6 s 2	88.032 2	7/2+
^{109}Ag	6.9 ns 7	311.4 1	3/2-
^{109}Ag	9.35 ns 20	132.8 2	(9/2)+
^{109}Ag	35 ps 4	415.3 3	5/2-
^{109}Ag	0.31 ps 9	1324.2 7	(3/2-)
^{109}Ag	0.5 ps 2	701.9 2	3/2-
^{109}Ag	1.3 ps 4	862.7 4	5/2-
^{109}Pd	4.69 m 1	188.9 1	11/2-
^{109}Pd	13.46 h 2	0.0	5/2+
^{109}Rh	90 s ?	0.0	
^{109}Ru	34.5 s 29	0.0	
^{110}Ag	24.57 s 12	0.0	1+
^{110}Ag	250.4 d 10	117.69 10	6+
^{110}Ag	37.5 ns 20	118.65 10	
^{110}Pd	3.8 ps 6	920.5 6	4+
^{110}Pd	8.0 ps 14	946.3 6	(0+)
^{110}Pd	14.0 ps 18	813.8 5	2+
^{110}Pd	43 ps 3	373.8 3	2+
^{110}Rh	3.0 s 2	0.0	
^{110}Rh	28.5 s 15	0.0	
^{110}Ru	0.23 ns 5	240.67	(2+)
^{110}Ru	15.9 s 5	0.0	0+
^{110}Tc	0.82 s 4	0.0?	
^{111}Ag	74 s 3	59.85	(7/2+)
^{111}Ag	7.47 d 2	0.0	1/2-
^{111}Cd	84.5 ns	245.35	5/2+
^{111}Cd	10.2 ps	620.0	5/2+
^{111}Cd	120 ps	416.64	7/2+
^{111}Cd	48.7 m 2	396.0	11/2-
^{111}Cd	≈ 40 ps	341.9	3/2+
^{111}Pd	22 m 1	0.0	(5/2+)
^{111}Pd	5.5 h 1	172.0 1	(11/2-)
^{112}Ag	3.12 h 1	0.0	2(-)
^{112}Cd	6.0 ps 3	617.4 3	2+
^{112}Cd	2.9 ps 5	1468.2 6	2+
^{112}Cd	0.9 ps 1	1414.2 6	4+
^{112}Cd	1.7 ps 3	1311.6 5	2+
^{112}Pd	20.12 h 6	0.0	0+
^{112}Pd	< 0.1 ns	348.8 5	2+
^{112}Rh	4.65 s 14	0.0?	
^{112}Ru	0.69 s 5#	0.0?	0+
^{112}Ru	0.20 ns	236.8? 5	(2+)
^{113}Ag	5.37 h 5	0.0	1/2(-)
^{113}Ag	1.20 m 15		(7/2+)
^{113}Cd	13.6 y 2	263.7 3	11/2-
^{113}Cd	> 3×10^{15} y	0.0	1/2+
^{113}Cd	32 ps 3	298.6 2	3/2+
^{113}Cd	9 ps 1	583.7 5	5/2+
^{113}Pd	1.4 m 1	0.0	
^{114}Ag	4.52 s 7	0	1+
^{114}Cd	10.2 ps 1	558.29 3	2+
^{114}Cd	≈ 5 ps	1283.30 3	4+
^{114}Pd	2.4 m 1	0	0+
^{115}Ag	55 s 2	0.0	(7/2+)
^{115}Ag	20.0 m 5	0.0	(1/2-)
^{115}Cd	44.6 d 3	173.4 3	11/2-
^{115}Cd	53.46 h 10	0.0	1/2+
^{115}In	57 ps 5	933.6 1	7/2+
^{115}In	0.45 ps 7	1290.5 1	(13/2)+
^{115}In	14 ps 3	941.2 5	5/2+
^{115}In	0.8 ps 1	1077.8 4	5/2+
^{115}In	5.1×10^{14} y	0.0	9/2+
^{115}In	1.0 ns 1	863.95 9	(1/2)+
^{115}In	5.7 ns 2	828.39 8	3/2+
^{115}In	4.3 h 1	336.23 5	1/2-
^{115}In	0.07 ps 2	1132.5 1	(11/2)+
^{115}Pd	37.4 s 4	0.0	
^{115}Sn	159 μs	714.6	(11/2)-
^{115}Sn	3.26 μs	613.9	(7/2)+
^{115}Sn	stable	0.0	1/2+

Table 1 cont'd

' fission product half-lives ' continued

Nucleus	$T_{1/2}$	E(level)	Jπ
^{116}Ag	10.4 s 8	81	
^{116}Ag	2.68 m 1	0.0	
^{116}Cd	2.1 ps	1213.6	2+
^{116}Cd	1.7 ps	1220.2	4+
^{116}Cd	12.2 ps	513.9	2+
^{116}Cd	1.2 ps	1381.3	0+
^{116}In	2.18 s	289.658	(8-)
^{116}In	14.10 s	0.0	1+
^{116}In	54.15 m	127.268	5+
^{116}Pd	< 0.1 ns	340.6?	(2+)
^{116}Pd	13.6 s 12	0.0	0+
^{116}Rh	0.60 ns	49.8	
^{116}Rh	unknown	0?	
^{116}Sn	0.75 μs	≈ 3300	
^{116}Sn	350 ns	2365.90	5-
^{116}Sn	0.362 ps	1293.54	2+
^{117}Sn	14.0 d 3	314.58 4	11/2-
^{117}Sn	0.279 ns 9	158.562 15	3/2+
^{118}Ag	2.8 s 3	127.7	
^{118}Ag	3.7 s 2	0.0	
^{118}Cd	50.3 m 2	0.0	0+
^{118}In	8.5 s 3	≈ 200	(8-)
^{118}In	4.45 m 5	≈ 60	(5+)
^{118}In	5.0 s 3	0.0	1+
^{118}Pd	3.1 s 3	0.0?	0+
^{118}Sn	0.465 ps 11	1229.64	2+
^{118}Sn	230 ns 10	2574.83	7-
^{118}Sn	21.7 ns 2	2321.15	5-
^{118}Sn	2.93 μs	3111.5	(10+)
^{119}Sn	17.75 ns 12	23.875 10	3/2+
^{119}Sn	245 d 20	89.54 1	11/2-
^{120}Ag	1.17 s 5	0.0	
^{120}Ag	0.32 s 4	203.0	
^{120}Cd	50.80 s 21	0.0	0+
^{120}In	44.4 s 10		(5)+
^{120}In	3.08 s 8		1+
^{120}Sn	0.632 ps 12	1171.5 2	2+
^{120}Sn	≤ 0.15 ns	2194.6 3	4+
^{120}Sn	5.53 ns 6	2284.6 4	5-
^{120}Sn	11.8 μs 5	2482.0 5	7-
^{121}Ag	≤ 3 s	0.0	
^{121}Cd	12.8 s 4	0.0	
^{121}In	3.1 m 3	325 50	(1/2)-
^{121}In	30 s 3	0.0	9/2+
^{121}Sb	2.1 ps 3	507.54 5	3/2+
^{121}Sb	8.5 ps 10	573.08 5	1/2+
^{121}Sb	3.5 ns 2	37.15	(7/2)+
^{121}Sn	27.06 h 4	0.0	3/2+
^{121}Sn	≈ 50 y	8 6	(11/2)-
^{122}Ag	1.5 s 5	0.0?	
^{122}Cd	5.5 s 1	0.0?	
^{122}In	10.0 s 5	0.0	(high)
^{122}In	1.5 s 3	0.0?	(low)
^{122}Sb	2.70 d 1	0	2-
^{122}Sb	530 μs 30	137.50 5	(5)+
^{122}Sb	1.8 μs 2	61.42 2	(3)+
^{122}Sb	4.2 m 2	163 1	
^{122}Sn	0.75 ps	1140.2 10	2+
^{123}In	47.8 s 5	320 10	(1/2)-
^{123}In	5.97 s 5	0.0	(9/2)+
^{123}Sb	7.6 ps	541.8	(3/2)+
^{123}Sb	0.61 ns	160.2	5/2+
^{123}Sn	129.2 d	0	(11/2)-
^{123}Sn	40.08 m	24	(3/2)+
^{124}In	3.21 s 6	0.0	
^{124}Sb	20.2 m 2	35	
^{124}Sb	93 s 5	10.73	(5+)
^{124}Sb	60.20 d 2	0.0	3-
^{124}Sn	0.95 ps 3	1131.4 9	2+
^{125}In	12? s		

Table 1 cont'd

' fission product half-lives ' continued

Nucleus	$T_{1/2}$	E(level)	Jπ
^{125}In	2.3? s		
^{125}Sn	9.64 d	0	11/2-
^{125}Te	x160 ps	525.1	(7/2-)
^{125}Te	12.8 ps 22	463.4	5/2+
^{125}Te	0.71 ns 2	321.02	(9/2)-
^{125}Te	58 d	144.73	11/2-
^{125}Te	1.49 ns	35.46	3/2+
^{125}Te	x600 ps	642.1	(7/2+)
^{125}Te	40 ps 20	636.1	7/2+
^{125}Te	1.06 ps 20	671.5	5/2+
^{126}In	1.53 s 1	0.1?	
^{126}Sb	78.0 ns 5	128.0 3	(2+)
^{126}Sb	553 ns 5	104.7 3	(3+)
^{126}Sb	≈ 11 s	40.4 3	(3-)
^{126}Sb	19.0 m 3	17.7	(5+)
^{126}Sb	5.1 ns 3	83.1 3	(3-)
^{126}Sb	12.4 d 1	0.0	(8-)
^{126}Sn	≈ 10×10^5 y	0.0	0+
^{126}Te	4.21 ps 21	666.2	2+
^{126}Te	13.5 ns 10	2975.2	(10+)
^{127}I	3.2 ps 3	417.9	5/2+
^{127}I	15 ps 3	374.96	1/2+
^{127}I	0.39 ns 1	202.84	3/2+
^{127}I	1.93 ns 6	57.60	7/2+
^{127}In	3.64? s		
^{127}In	2.0? s		
^{127}Sb	11 μs 1	1920.2	
^{127}Sb	3.83 d 8	0	(7/2)+
^{127}Sn	4.4 m 1	300	(3/2+)
^{127}Sn	2.10 h 8	0	(11/2-)
^{127}Te	109 d 2	88.26	11/2-
^{127}Te	9.35 h 7	0	3/2+
^{127}Te	0.30 ns 8	340.7	
^{128}I	24.99 m 2	0.0	1+
^{128}In	3.7 s 5		
^{128}Sb	9.01 h 3	0.0	
^{128}Sb	10.4 m 2	X+0.0	
^{128}Sn	60.0 m 4	0.0	0+
^{128}Te	3.2 ps 3	743.2	2+
^{129}I	1.57×10^7 y	0	7/2+
^{129}I	16.8 ns	27.77	5/2+
^{129}Sb	4.32 h 3	0.0	7/2+
^{129}Sn	7.5 m	0	
^{129}Te	33.6 d	105.50	11/2-
^{129}Te	69.6 m	0	3/2+
^{129}Xe	50 ps	411.3	(1/2,3/2)+
^{129}Xe	1.01 ns	39.58	3/2+
^{129}Xe	8.0 d	236.14	11/2-
^{130}I	9.0 m 1	48.2	2+
^{130}I	12.36 h	0.0	5+
^{130}In	0.53 s 5	0.0?	
^{130}Sb	40 m 1	0.0	(8-)
^{130}Sb	6.3 m 2	≈ 30	(4,5)+
^{130}Sb	3.6 ns 3	≈ 100	(3,4,5)+
^{130}Sn	3.72 m 11	0.0	0+
^{130}Sn	1.7 m 1	≥ 1800	(7-)
^{130}Te	2.27 ps 7	839.4	2+
^{130}Te	2.51×10^{21} y 27	0.0	0+
^{130}Te	115 ns 8	2146.0	(7)-
^{130}Te	9.8 ns 5	1815	(6)+
^{130}Xe	6.4 ps 6	536.09	2+
^{131}I	8.04 d 1	0.0	7/2+
^{131}I	5.9 ns 2	1797.07 8	
^{131}I	0.76 ns 5	149.716 5	5/2+
^{131}In	0.27 s 2	0.0?	(9/2+)
^{131}Sb	23 m 2	0.0	(7/2+)
^{131}Sn	61 s 3	0.0	(3/2+)
^{131}Te	30 h 2	182.25 2	11/2-
^{131}Te	25.0 m 1	0.0	3/2+
^{131}Xe	11.9 d 1	163.930 13	11/2-

Table 1 cont'd

' fission product half-lives ' continued

Nucleus	$T_{1/2}$	E(level)	Jπ
^{131}Xe	2.15 ns 7	341.140 16	9/2-
^{131}Xe	51 ps 5	364.480 10	5/2+
^{131}Xe	< 0.5 ns	404.808 10	3/2(+)
^{131}Xe	4.2 ps 5	636.973 10	7/2+
^{131}Xe	< 0.5 ns	666.921 16	7/2-
^{131}Xe	0.53 ps 9	722.893 10	5/2+
^{131}Xe	14 ns 3	1805.7	(19/2+)
^{131}Xe	477 ps 10	80.183 10	1/2+
^{132}I	0.93 ns 4	49.72 1	3+
^{132}I	1.42 ns 5	277.88 7	1+
^{132}I	83.6 m 17	≈ 120	(8-)
^{132}I	2.30 h 3	0	4+
^{132}In	0.12 s 2	0?	
^{132}Sb	14.8 ns 10	≈ 85	(3+)
^{132}Sb	4.2 m 1	0.0	(8-)
^{132}Sb	2.8 m 1	≈ 0	(4+)
^{132}Sn	40 s 1	0.00	0+
^{132}Te	78.2 h 8	0	0+
^{132}Te	9 μs 2	1924.7 2	(7)
^{132}Te	145 ns 8	1774.1 2	(6)+
^{132}Xe	8.4 ms 8	2752.6 10	(10+)
^{132}Xe	90 ns 10	2215.1 6	(7-)
^{132}Xe	5.06 ps 40	667.67 6	2+
^{133}Cs	27 ps 3	383.851	3/2+
^{133}Cs	6.30 ns 2	80.997	5/2+
^{133}I	9 s	1634	
^{133}I	20.8 h 1	0.0	7/2+
^{133}Sb	2.7 m 1	0.0	
^{133}Sn	1.7 s ?	0?	
^{133}Te	55.4 m 4	334.2	(11/2-)
^{133}Te	12.45 m 28	0.0	(3/2+)
^{133}Xe	2.191 d 30	233.5	11/2-
^{133}Xe	5.29 d 1	0.0	3/2+
^{134}Cs	≈ 53 ns	176.399	
^{134}Cs	2.062 y 5	0.0	4(+)
^{134}Cs	2.90 h 1	138.738	8(-)
^{134}Cs	45.7 ns 12	11.239	5(+)
^{134}I	< 150 ps	210.47	
^{134}I	52.6 m	0.0	
^{134}I	< 100 ps	180.89	
^{134}I	3.7 m 1	316.3	
^{134}I	1.62 ns 10	79.45	
^{134}Sb	0.85 s 10	300	
^{134}Sb	11.0 s 3	0	
^{134}Te	41.8 m 8	0.0	0+
^{134}Te	164 ns 3	1691.3	(6+)
^{134}Xe	0.290 s 17	1965.4	
^{135}Ba	28.7 h 2	268.24	11/2-
^{135}Cs	53 m 2	1627	(19/2-)
^{135}Cs	0.28 ns 8	249.79	5/2+
^{135}Cs	2.3×10^{6} y	0.0	7/2+
^{135}I	6.61 h	0	7/2+
^{135}Sb	≈ 2 s	0.0	
^{135}Te	18 s 1	0.0	
^{135}Xe	15.65 m	526.56	11/2-
^{135}Xe	9.083 h	0.0	3/2+
^{136}Ba	3.3 ns	2207.10	6+
^{136}Ba	x 1 ns	2140.15	5-
^{136}Ba	0.306 s	2030.46	7-
^{136}Cs	12.98 d	0.0	5+
^{136}I	83 s	1	(2-)
^{136}I	46 s	250	(5-)
^{136}Te	20.9 s	1?	0+
^{136}Xe	3.0 μs	1892.0	(6)+
^{137}Ba	2.552 m 1	661.640 19	11/2-
^{137}Cs	30.17 y 3	0.0	7/2+
^{137}I	24.7 s 2	0.0	(7/2+)
^{137}Te	3.5 s 5	0.0?	
^{137}Xe	3.83 m 1	0.0	(7/2)-
^{138}Ba	0.199 ns 12	1435.91 6	2+

Table 1 cont'd

' fission product half-lives ' continued

Nucleus	$T_{1/2}$	E(level)	Jπ
^{138}Ba	≤ 70 ps	3709.0 7	
^{138}Ba	2.17 ns 8	1898.71 7	4+
^{138}Ba	≤ 70 ps	3622.8 6	
^{138}Ba	≤ 70 ps	3184.0 5	
^{138}Ba	≤ 70 ps	3632.5 3	
^{138}Ba	≤ 70 ps	2930.2 6	
^{138}Ba	0.8 μs 1	2090.66 9	(6)+
^{138}Cs	32.2 m 1	0.0	3(-)
^{138}Cs	2.90 m 10	79.9	(6-)
^{138}I	6.40 s 6	0.0	(3-)
^{138}Te	1.4 s 4	0.0	0+
^{138}Xe	14.13 m 5	0.0	0+
^{139}Ba	84.9 m 11	0	(7/2)-
^{139}Cs	9.40 m 12	0	(7/2,9/2)
^{139}I	2.4 s 2	0	
^{139}La	1.50 ns 2	165.853 7	5/2+
^{139}Xe	39.5 s 2	0	(5/2-11/2)
^{140}Ba	12.789 d 6	0	0+
^{140}Ce	0.27 ns	1903.30	0+
^{140}Ce	3.45 ns	2083.5	8+
^{140}Ce	7 μs ?	2108.10	6+
^{140}Ce	55 ps	2412.28	3+
^{140}Ce	0.07 ps ?	2464.32	3-
^{140}Ce	0.1 ns	2521.72	2+
^{140}Ce	0.08 ps	1596.45	2+
^{140}Cs	63.7 s 3	0	
^{140}I	0.86 s	0	
^{140}La	40.22 h 2	0.0	3-
^{140}La	1e0.4 ns	29.9653	2-
^{140}La	1e7 ps	467.505	1-
^{140}La	1e10 ps	162.656	2-
^{140}La	0.5 ns	43.811	1-
^{140}Xe	13.60 s	0	0+
^{141}Ba	18.27 m 7	0.0	
^{141}Ce	32.50 d 7	0.0	7/2-
^{141}Cs	24.9 s 2	0.0	
^{141}I	0.43 s 8	0.0	
^{141}La	3.93 h 6	0.0	(7/2+)
^{141}La	1.25 ns 8	190.22 8	
^{141}Pr	4.7 ns 3	1118 1	11/2-
^{141}Pr	1.85 ns 3	145.440 10	7/2+
^{141}Xe	1.73 s 1	0.0	
^{142}Ba	10.7 m 1	0.0	0+
^{142}Ce	5.7 ps 1	641.2 1	2+
^{142}Ce	> 5×10^{16} y	0.0	0+
^{142}Cs	1.7 s 1	0.0	
^{142}La	92.7 m 7	0.0	2-
^{142}Pr	19.13 h 4	0.0	2-
^{142}Pr	14.6 m 5	3.683 4	5-
^{142}Xe	1.22 s 2	0.0	0+
^{143}Ba	20 s 1	0.0	
^{143}Ce	33.0 h 2	0.0	3/2-
^{143}Cs	1.68 s 7	0.0	
^{143}La	14.0 m 1	0.0	
^{143}Nd	2.5 ps	741.98	3/2-
^{143}Pr	13.56 d 2	0.0	7/2+
^{143}Xe	0.96 s 2	0.0	
^{143}Xe	0.30 s 3	0.0	
^{144}Ba	≤ 0.7 ns	199.3 2	(2+)
^{144}Ba	11.9 s 3	0.0	0+
^{144}Ce	284.31 d 28	0.0	0+
^{144}Cs	1.02 s 4	0.0	
^{144}La	40.7 s 5	0.0	
^{144}Nd	≈ 11 fs	2185.68 6	1-
^{144}Nd	3.43 ps 14	696.49 2	2+
^{144}Nd	2.1×10^{15} y 4	0.0	0+
^{144}Pr	7 ps 4	133.530 22	1-
^{144}Pr	17.28 m 3	0.0	0-
^{144}Pr	7.2 m 2	59.03 3	3-
^{144}Pr	0.66 ns 5	99.96 3	2-

Table 1 cont'd

' fission product half-lives ' continued

Nucleus	$T_{1/2}$	E(level)	Jπ
^{144}Pr	136 ps 19	80.12 3	1-
^{145}Ba	6.2 s 5	0.00	
^{145}Ce	3.0 m 1	0.00	
^{145}Cs	563 ms 27	0.00	
^{145}La	≈ 30 s	0.00?	
^{145}Nd	0.7 ns 1	72.2 2	5/2-
^{145}Nd	33 ns 3	67.0 2	3/2-
^{145}Pr	5.98 h 2	0.00	(7/2+,5/2+)
^{145}Xe	0.9 s 3	0.00	
^{146}Ba	1.91 s 16	0.0?	0+
^{146}Ba	≈ 0.85 ns	180.8 2	(2)+
^{146}Ce	14.2 m 5	0.0	0+
^{146}Ce	≈ 0.29 ns	258.32? 17	(2)+
^{146}Cs	189 ms 11	0	
^{146}La	8.8 s 4	0.0?	
^{146}Nd	4.0 ps 10	1043.4 6	4+
^{146}Nd	22.6 ps 13	453.77 17	2+
^{146}Pr	24.07 m 13	0.0	
^{147}Pm	2.6234 y 2	0.0	7/2+
^{147}Sm	1.07×10^{11} y 1	0.0	7/2-
^{148}Ba	0.47 s 20	0.0	0+
^{148}Ce	1.06 ns 8	158.45 8	2+
^{148}Ce	48 s 1	0.0	0+
^{148}La	1.29 s 8	0.0	
^{148}Nd	0.10 ns 2	301.7 1	2+
^{148}Nd	5.4 ps 20	752.0 3	4+
^{148}Pm	5.37 d 1	0.0	1-
^{148}Pm	41.3 d 1	137.2 2	6-
^{148}Pr	2.30 m 3	0.0	(3)
^{148}Sm	0.6 ps 1	1454.3 3	2+
^{148}Sm	0.10 ps 3	1465.1 4	1-
^{148}Sm	2.3 ps 6	1180.2 2	4+
^{148}Sm	8×10^{15} y 2	0.0	0+
^{148}Sm	7.3 ps 4	550.3 1	2+
^{149}Ce	≥ 5 s	0.0	
^{149}Nd	1.73 h 1	0.0 2	5/2-
^{149}Pm	53.08 h 5	0.0 1	7/2+
^{149}Pm	2.58 ns 4	114.32 10	5/2+
^{149}Pm	3.24 ns 14	188.64 10	5/2+,7/2+
^{149}Pm	80 ps 15	211.31 10	5/2+
^{149}Pm	35 μs 3	240.22 10	(11/2)-
^{149}Pm	2.64 ns 7	270.18 10	7/2-
^{149}Pm	≤ 50 ps	537.87 10	5/2-
^{149}Pm	≤ 0.18 ps	654.84 10	7/2-
^{149}Pm	0.6 ns 1	387.57 10	(1/2)+
^{149}Pr	2.5 m 2	0.0	(5/2+,7/2+)
^{149}Sm	$> 1\times10^{16}$ y	0.0	7/2-
^{149}Sm	7.13 ns 11	22.494 10	5/2-
^{149}Sm	≤ 0.2 ns	277.0 30	(5/2-)
^{149}Sm	≤ 0.2 ns	350.0 30	(3/2-)
^{149}Sm	0.262 ns 9	285.9 30	(9/2-)
^{150}Ce	4.0 s 6	0	0+
^{150}Ce	3.6 ns 1	97.8 6	2+
^{150}Nd	$> 5\times10^{17}$ y	0	0+
^{150}Nd	1.50 ns 2	130.12 6	2+
^{150}Nd	61 ps 5	381.5 6	4+
^{150}Nd	4.7 ps 4	1130.7 10	(8+)
^{150}Nd	6.5 ps 13	851.4 5	2+
^{150}Nd	12.0 ps 9	721.2 9	(6+)
^{150}Nd	7.4 ps 22	676.7 6	0+
^{150}Nd	1.42 ps 15	1062.4 5	(2+)
^{150}Pm	2.68 h 2	0	(1-)
^{150}Pr	6.2 s 2	0	(0-,1-,2-)
^{150}Sm	49.0 ps 6	333.95 1	2+
^{150}Sm	0.90 ps 6	1071.40 1	3-
^{150}Sm	1.27 ps 9	1193.81 1	2+
^{150}Sm	19.9 ps 20	740.42 4	0+
^{150}Sm	6.6 ps 3	773.35 1	4+
^{150}Sm	0.73 ps 21	1046.14 1	2+
^{151}Ce	1.02 s 6	0.0	

Table 1 cont'd

' fission product half-lives ' continued

Nucleus	$T_{1/2}$	E(level)	Jπ
^{151}Eu	9.5 ns 5	21.54 2	7/2+
^{151}Eu	0.24 ns 3	196.46 9	(3/2)+
^{151}Eu	0.36 ns 2	243.22 6	7/2-
^{151}Eu	62.7 μs 30	196.21 7	11/2-
^{151}Eu	<0.1 ns	349.76 6	9/2-
^{151}Nd	12.44 m 7	0.0	(3/2+)
^{151}Pm	28.40 h 4	0.0	5/2+
^{151}Pm	0.91 ns 3	255.6 2	3/2+
^{151}Pm	89 ps 15	116.7 2	5/2-
^{151}Pm	48 ps 10	1297.3 4	5/2+
^{151}Pr	4.0 s 7	0.0	
^{151}Pr	20 ns 8	96.8+X	
^{151}Sm	90 y 6	0.0	5/2-
^{151}Sm	0.4 ns 1	65.83 2	7/2-
^{151}Sm	35 ns 1	4.821 3	3/2-
^{151}Sm	77 ns 1	91.53 4	9/2+
^{151}Sm	< 0.5 ns	69.70 2	5/2-
^{151}Sm	0.48 ns 6	104.82 3	3/2-
^{151}Sm	0.38 ns 4	167.75 2	5/2+
^{151}Sm	< 0.4 ns	168.40 3	5/2-
^{151}Sm	≤ 0.1 ns	822.46 6	3/2+
^{151}Sm	1.4 μs 1	261.1 1	11/2-
^{151}Sm	≤ 0.1 ns	445.65 5	5/2+
^{151}Sm	≤ 0.09 ns	344.89 3	3/2+
^{151}Sm	≤ 0.1 ns	740.90 6	3/2
^{153}Eu	0.80 ns	83.367	7/2+
^{153}Eu	3.90 ns	103.179	3/2+
^{153}Eu	0.14 ns	172.851	5/2+
^{153}Eu	0.20 ns	97.430	5/2-
^{153}Eu	0.22 ns	193.064	9/2+
^{153}Pm	5.4 m 2	0.0	(5/2)-
^{153}Sm	46.7 h 1	0.0	3/2+
^{153}Sm	< 0.1 ns	35.843	3/2-
^{153}Sm	10.6 ms 3	98.4	(11/2-)
^{153}Sm	0.52 ns 16	90.874	5/2-
^{153}Sm	17.1 ns 66	182.90	5/2-
^{155}Eu	0.104 ns 10	104.32	5/2-
^{155}Eu	1.35 ns 5	245.73	3/2+
^{155}Eu	4.96 y 1	0.0	5/2+
^{155}Gd	1.16 ns 2	105.308	3/2+
^{155}Gd	0.19 ns 1	60.010	5/2-
^{155}Gd	6.5 ns 2	86.543	5/2+
^{155}Gd	0.10 ns 3	146.07	7/2-
^{155}Gd	31.97 ms 27	121.5	(11/2)-
^{155}Gd	≤ 8. ns	107.56	9/2+
^{155}Sm	22.1 m 2	0	3/2-
^{156}Eu	≤ 0.2 ns	291.06 17	1+
^{156}Eu	12.0 ns 3	87.25 19	1-
^{156}Eu	15.19 d 6	0.0	0+
^{156}Gd	2.17 ns 3	88.9652 15	2+
^{156}Gd	114 ps 16	238.182 4	4+
^{156}Gd	15.8 ps 12	584.72 4	6+
^{156}Gd	1.86 ps 11	1414.7 4	(10+)
^{156}Gd	1.9 ps 6	1154.105 23	2+
^{156}Gd	1.0 ps 4	1049.38 16	0+
^{156}Gd	4.4 ps 3	965.06 15	(8+)
^{156}Sm	≤ 160 ns	250.1 6	
^{156}Sm	9.4 h 2	0.0	0+
^{157}Eu	15.15 h 4	0	(5/2+)
^{157}Gd	< 0.1 ns	436.6	5/2-
^{157}Gd	17 μs 1	425	11/2-
^{157}Gd	< 0.1 ns	477.2	(3/2+)
^{157}Gd	0.46 μs 4	64	5/2+
^{157}Sm	83 s 2	0	
^{158}Eu	45.9 m	0	(1-)
^{158}Gd	0.162 ns 13	261.44	4+
^{158}Gd	2.54 ns 2	79.51	2+
^{159}Eu	18.7 m 4	0	(5/2+)
^{159}Gd	18.56 h 8	0.0	3/2-
^{159}Gd	26.2 ns 8	67.8	5/2+

Table 1 cont'd

' fission product half-lives ' continued

Nucleus	$t_{1/2}$	E(level)	Jπ
^{159}Tb	35.6 ps 43	137.5	7/2+
^{159}Tb	0.76 ps 10	580.8	1/2+
^{159}Tb	0.158 ns 7	363.3	5/2-
^{160}Eu	50 s 10	0.0	(0-)
^{160}Gd	2.66 ns 4	75.26	2+
^{160}Tb	72.3 d 2	0.0	3-
^{161}Dy	28.5 ns 6	25.65	5/2-
^{161}Dy	0.78 ns 6	43.84	7/2+
^{161}Dy	0.15 ns 2	131.77	5/2-
^{161}Dy	3.21 ns 3	74.58	3/2-
^{161}Gd	3.7 m 1	0	5/2-
^{161}Tb	6.91 d 2	0	3/2+
^{161}Tb	< 0.1 ns	480.0	(5/2)-
^{161}Tb	0.88 ns 2	417.0	7/2-
^{162}Dy	2.188 ns 35	80.660 2	2+
^{162}Dy	0.99 ps 10	1901 1	12+
^{162}Dy	1.8 ps 2	1375.1 1	10+
^{162}Dy	4.6 ps 5	921.3 1	8+
^{162}Dy	132 ps 8	265.665 4	4+
^{162}Dy	1.93 ns 11	1485.74 5	5-
^{162}Dy	0.21 ns 4	1148.29 5	(2)-
^{162}Dy	2.3 ps 1	888.22 5	2+
^{162}Gd	9 m 1	0	0+
^{162}Tb	7.7 m 2	0.0	(1)-
^{163}Dy	1.51 ns 5	73.4405 15	7/2-
^{163}Tb	19.5 m 3	0.0	3/2+
^{164}Dy	2.39 ns 4	73.392 5	2+
^{164}Dy	4.9 ps 4	761.78 8	2+
^{164}Dy	≈ 0.8 ps	1745.5 10	(12+)
^{164}Dy	≈ 2 ps	1261.3 2	10+
^{164}Dy	≈ 6 ps	843.67 7	8+
^{164}Tb	3.0 m 1	0.0	(5+)
^{165}Dy	10.7 ns 35	180.927 4	5/2-
^{165}Dy	< 35 ps	83.397	9/2+
^{165}Dy	2.334 h 6	0.0	7/2+
^{165}Dy	1.84 ns 99	158.591 4	3/2-
^{165}Dy	1.258 m 6	108.160	1/2-
^{165}Dy	1.07 ns 36	184.225 5	5/2-
^{165}Dy	< 35 ps	261.776 6	7/2-
^{167}Dy	6.8 m 4	0.0	(1/2-)

Table 2

Fission product half lives >1ms sorted by half life

* fission product half-lives * continued

$T_{1/2}$	Nucleus	E(level)	Jπ
1.56 ms 5	^{103}Ru	237 5	(11/2)-
8.4 ms 8	^{132}Xe	2752.6 10	(10+)
10.6 ms 3	^{153}Sm	98.4	(11/2-)
17.53 ms 8	^{75}As	303.916 12	9/2+
31.97 ms 27	^{155}Gd	121.5	(11/2)-
39.86 ms 13	^{72}Ga	119.58 8	
0.076 s 5	^{99}Rb	0	
< 0.1 s	^{97}Kr	0.0	
0.12 s 2	^{132}In	0?	
0.136 s 8	^{98}Rb	0	
0.176 s 6	^{97}Rb	0.0	
189 ms 11	^{146}Cs	0	
0.20 s 1	^{94}Kr	0.0	0+
≤ 0.2 s	^{97}Sr	0.0	
0.207 s 3	^{96}Rb	0.0	
0.25 s 10	^{92}Br	0.0	
0.27 s 2	^{131}In	0.0?	(9/2+)
0.290 s 17	^{134}Xe	1965.4	
≈ 0.3 s	^{98}Y	0	
0.30 s 3	^{143}Xe	0.0	
0.306 s	^{136}Ba	2030.46	7-
0.32 s 4	^{120}Ag	203.0	
0.36 s 2	^{95}Rb	0.0	
0.41 s 4	^{89}Se	0.0	
0.43 s 8	^{141}I	0.0	
0.47 s 20	^{140}Ba	0.0	0+
0.53 s 5	^{73}Ge	66.70 2	1/2-
0.53 s 5	^{138}In	0.0?	
563 ms 27	^{145}Cs	0.00	
0.6 s 3	^{87}As	0.0	
0.63 s 7	^{91}Br	0.0	
0.69 s 54	^{112}Rh	0.0?	0+
0.8 s 7	^{99}Y	0.0	
0.8 s 3	^{102}Zr	0	0+
809.2 ms 20	^{90}Zr	2319.10 15	5-
0.82 s 4	^{110}Tc	0.0?	
0.845 s 43	^{98}Sr	0	0+
0.85 s 10	^{134}Sb	300	
0.86 s	^{140}I	0	
0.9 s 3	^{102}Y	0?	
0.9 s 3	^{145}Xe	0.00	
0.96 s 2	^{143}Xe	0.0	
≈ 1 s	^{95}Kr	0.0?	
1.02 s 4	^{144}Cs	0.0	
1.02 s 6	^{151}Ce	0.0	
1.11 s 14	^{97}Y	0.0	
1.17 s 5	^{120}Ag	0.0	
1.22 s 2	^{142}Xe	0.0	0+
1.289 s 12	^{93}Kr	0.0	
1.29 s 8	^{148}La	0.0	
1.4 s 3	^{77}Zn	0.0?	
1.4 s 4	^{136}Te	0.0	0+
1.5 s 3	^{100}Nb		
1.5 s 5	^{108}Mo	0.0?	0.0
1.5 s 5	^{122}Ag	0.0?	
1.5 s 3	^{122}In	0.0?	(low)
1.53 s 6	^{88}Se	0	0+
1.53 s 1	^{124}In	0.1?	
1.68 s 7	^{143}Cs	0.0	
1.7 s 2	^{80}Ga	0	
1.7 s 7	^{133}Sn	0?	
1.7 s 1	^{142}Cs	0.0	
1.71 s 14	^{90}Br	0.0	
1.73 s 1	^{141}Xe	0.0	
1.8 s?	^{105}Nb	0.0?	
1.840 s 8	^{92}Kr	0.0	
1.9 s 4	^{83}Ge	0.0	
1.91 s 16	^{146}Ba	0.0?	0+
2.0? s	^{127}In		
≈ 2 s	^{135}Sb	0.0	

Table 2 cont'd

' fission product half-lives ' continued

$T_{1/2}$	Nucleus	E(level)	Jπ
2.028 s 12	^{85}As	0.0	
2.18 s	^{116}In	289.658	(8-)
2.37 s	^{125}In		
2.4 s 1	^{99}Zr	0.0	
2.8 s 2	^{139}I	0	
2.69 s 4	^{99}Rb	0.0	
2.8 s 2	^{98}Nb		(1+)
2.8 s 3	^{118}Ag	127.7	
2.9 s 4	^{102}Nb	0.0	
3.00 s 9	^{79}Ga		
3.0 s 2	^{110}Rh	0.0	
≤ 3 s	^{121}Ag	0.0	
3.08 s 8	^{120}In		1+
3.1 s 3	^{100}Nb		
3.1 s 3	^{118}Pd	0.0?	0+
3.21 s 6	^{124}In	0.0	
3.3 s 6	^{101}Zr	0.0	
3.5 s 5	^{137}Te	0.0?	
3.64? s	^{127}In		
3.7 s 2	^{119}Ag	0.0	
3.7 s 5	^{128}In		
4.0 s 2	^{96}Sr	0.0	
4.0 s 6	^{150}Ce	0	0+
4.0 s 7	^{151}Pr	0.0	
4.48 s 3	^{97}Rb	0.0	
4.52 s 7	^{114}Ag	0	1+
4.53 s 10	^{89}Br	0.0	
4.6 s 4	^{82}Ge	0.0	0+
4.65 s 14	^{112}Rh	0.0?	
4.8 s 13	^{78}Ga	0.0	
4.88 s 5	^{79}Br	207.2 4	9/2+
5.0 s 3	^{116}In	0.0	1+
≥ 5 s	^{149}Co	0.0	
5.28 s 15	^{102}Tc	0	(1+)
5.5 s 1	^{122}Cd	0.0?	
5.7 s 3	^{74}Zn	0.0?	0+
5.8 s 5	^{84}As	0.0	
5.8 s 3	^{87}Se	0.0	
5.8 s 1	^{93}Rb	0.0	
5.97 s 5	^{123}In	0.0	(9/2)+
6.2 s 5	^{145}Ba	0.00	
6.2 s 2	^{150}Pr	0	(0-,1-,2-)
6.40 s 6	^{138}I	0.0	(3-)
7.0 s 3	^{101}Nb	0.0	
7.1 s 4	^{100}Zr	0.0	0+
8.3 s	^{100}Tc	0.0	
8.5 s 3	^{118}In	≈ 200	(8-)
8.6 s	^{91}Kr	0.0	
8.8 s 4	^{146}La	0.0?	
9 s 2	^{75}Zn	0	
9 s	^{133}I	1634	
9.5 s 10	^{74}Ga	59.7 2	1+
9.5 s 5	^{106}Mo	0.0	0+
10.0 s 5	^{122}In	0.0	(high)
10.1 s	^{81}Ge	0	
10.4 s 8	^{116}Ag	81	
11.0 s 3	^{134}Sb	0	
≈ 11 s	^{126}Sb	80.4 3	(3-)
11.9 s 3	^{144}Ba	0.0	0+
127 s	^{125}In		
12.8 s 4	^{121}Cd	0.0	
13.0 s 6	^{82}As		
13.6 s 12	^{116}Pd	0.0	0+
13.60 s	^{140}Te	0	0+
14.1 s 11	^{83}As	0.0	
14.10 s	^{116}In	0.0	1+
14.3 s 6	^{99}Nb	0	(9/2)+
15.8 s 1	^{100}Tc	0.0	1+
15.9 s 5	^{110}Ru	0.0	0+
16.3 s 3	^{88}Br	0.0	(1-)

Table 2 cont'd

Fission product half-lives ' continued

$T_{1/2}$	Nucleus	E(level)	Jπ
16.5 s 3	^{80}As	0	1(+)
16.7 s 3	^{86}Se	0.0	
16.8 s 5	^{108}Rh	0.0	
17.1 s 15	^{77}Ga	0.0	
17.5 s 1	^{77}Se	161.9 5	7/2+
18 s 1	^{135}Te	0.0	
19.1 s 3	^{79}Ge		
20 s 1	^{143}Ba	0.0	
20.9 s	^{136}Te	17	0+
21 s 2	^{82}As		(1+)
21.3 s 3	^{107}Pd	214 1	(11/2)-
23.5 s 10	^{73}Zn	0.0	
24.5 s 10	^{80}Ge	0	0+
24.57 s 12	^{110}Ag	0.0	1+
24.7 s 2	^{137}I	0.0	(7/2+)
24.9 s 2	^{141}Cs	0.0	
26 s 1	^{93}Sr	0.0	
27.1 s 2	^{76}Ga	0.0	(3-)
28.5 s 15	^{110}Rh	0.0	
29 s 3	^{107}Tc	0.0	
29.9 s 1	^{106}Rh	0.0	1+
30 s 3	^{121}In	0.0	9/2
≈ 30 s	^{145}La	0.00?	
31 s 3	^{98}Zr	0	0+
32.32 s 9	^{90}Kr	0.0	0+
33 s	^{81}As	0.0	
34.5 s 29	^{109}Ru	0.0	
36 s 1	^{106}Tc	0.0	
37.4 s 4	^{115}Pd	0.0	
39 s 4	^{85}Se	0.0	
39.5 s 2	^{139}Xe	0	(5/2-11/2)
39.6 s 2	^{109}Ag	88.032 2	7/2+
40 s 1	^{132}Sn	0.00	0+
40.7 s 5	^{144}La	0.0	
42 s 2	^{79}Ge	0.0	(1/2)-
42. s 3	^{105}Ho	0.0	
42.3 s 4	^{104}Rh	0.0	1+
44.3 s 2	^{107}Ag	93.08 5	7/2+
44.4 s 10	^{120}In		(5)+
45 s	^{105}Rh	129.7 1	(1/2)-
46 s	^{136}I	250	(5-)
47.8 s 5	^{123}In	320 10	(1/2)-
48 s 1	^{148}Ce	0.0	0+
48.3 s 6	^{75}Ge	139.3 2	(7/2)+
50 s 2	^{103}Tc	0.0	
50 s 10	^{160}Eu	0.0	(0-)
50.80 s 21	^{120}Cd	0.0	0+
54 s 1	^{77}Ge	159.7 1	(1/2)-
54 s 2	^{88}Br	0.0	1,2-
55 s 2	^{115}Ag	0.0	(7/2+)
55.7 s	^{87}Br	0.0	
58 s	^{91}Rb	0	
60 s 8	^{97}Nb	743.36 10	(1/2)-
60 s 2	^{103}Ho	0	
1.017 m 3	^{86}Rb	555.9 3	(6)-
61 s 3	^{131}Sn	0.0	(3/2+)
63.7 s 3	^{140}Cs	0	
70.4 s 3	^{83}Se	220	(1/2)-, 0.13
1.20 m 15	^{113}Ag		(7/2+)
74 s 3	^{111}Ag	59.85	(7/2+)
78 s 2	^{94}Sr	0.0	0+
1.3 m 3	^{104}Ho	0.0	0+
83 s	^{136}I	1	(2-)
83 s 2	^{157}Sm	0	
1.4 m 1	^{113}Pd	0.0	
90 s ?	^{109}Rh	0.0	
93 s 5	^{129}Sb	10.73	(5+)
95 s 1	^{74}Zn	0.0	0+
1.7 m 1	^{130}Sn	≥ 1800	(7-)
2.17 m 10	^{75}Ga	0	(3/2-)

Table 2 cont'd

' fission product half-lives ' continued

$T_{1/2}$	Nucleus	E(level)	Jπ
2.3 m 1	^{96}Y	0.0	
2.30 m 3	^{148}Pr	0.0	(3)
2.37 m 1	^{108}Ag	0.0	1+
2.4 m 1	^{114}Pd	0	0+
2.5 m 2	^{149}Pr	0.0	(5/2+,7/2+)
153 s 3	^{90}Rb	0.0	(1-)
2.552 m 1	^{137}Ba	661.640 19	11/2-
2.6 m 2	^{99}Nb	369 5	(1/2)-
2.68 m 1	^{116}Ag	0.0	
2.7 m 1	^{133}Sb	0.0	
2.8 m 1	^{132}Sb	≈ 0	(4+)
2.90 m 10	^{138}Cs	79.9	(6-)
3.00 m 5	^{85}Br	0.0	1/2-,3/2-
3.0 m 1	^{145}Ce	0.00	
3.0 m 1	^{164}Tb	0.0	(5+)
3.07 m 9	^{89}Kr	0.0	(5/2+)
3.1 m 3	^{121}In	325 50	(1/2)-
3.3 m 2	^{84}Se	0.0	0+
3.7 m 1	^{134}I	316.3	
3.7 m 1	^{161}Gd	0	5/2-
3.72 m 11	^{130}Sn	0.0	0+
3.83 m 1	^{137}Xe	0.0	(7/2)-
3.91 m 5	^{79}Se	95.9 3	1/2-
4.2 m 3	^{107}Ru	0.0	
4.2 m 2	^{122}Sb	163 1	
4.2 m 1	^{132}Sb	0.0	(8-)
258 s 5	^{90}Rb	106.92 4	(4-)
4.34 m 5	^{104}Rh	128.956 10	5+,4+
4.35 m 7	^{102}Tc	< 500	
4.4 m 1	^{127}Sn	300	(3/2+)
4.45 m 5	^{118}In	≈ 60	(5+)
4.5 m 2	^{105}Ru	0.0	0+
4.69 m 1	^{109}Pd	188.9 1	11/2-
5.4 m 2	^{133}Pm	0.0	(5/2)-
5.9 m 2	^{106}Rh	0.0	
6.0 m 2	^{84}Br	320 130	
6.13 m 8	^{82}Br	46 2	2-
6.26 m 1	^{99}Nb	40.95 4	3+
6.3 m 2	^{130}Sb	≈ 30	(4,5)+
7.2 m 2	^{144}Pr	59.03 3	3-
7.5 m	^{129}Sn	0	
7.6 m 2	^{93}Sr	0.0	(5/2+)
7.7 m 2	^{162}Tb	0.0	(1)-
7.8 m 8	^{105}Tc	0.0	
8.1 m 1	^{74}Ga	0.0	(4)-
9.01 m 15	^{79}As	0.0	3/2-
9.0 m 1	^{130}I	48.2	2+
9 m 1	^{162}Gd	0	0+
9.40 m 12	^{139}Cs	0	(7/2,9/2)
10.4 m 2	^{128}Sb	X+0.0	
10.7 m 2	^{95}Y	0.0	(1/2-)
10.7 m 1	^{142}Ba	0.0	0+
11.1 m 3	^{102}Mo	0	0+
12.44 m 7	^{151}Nd	0.0	(3/2+)
12.45 m 28	^{133}Te	0.0	(3/2+)
14.0 m 1	^{143}La	0.0	
14.13 m 5	^{138}Xe	0.0	0+
14.2 m 1	^{101}Tc	0.0	(9/2+)
14.2 m 5	^{144}Ce	0.0	0+
14.6 m 5	^{142}Pr	3.683 4	5-
14.62 m 5	^{101}Mo	0.0	1/2+
15.2 m 1	^{89}Rb	0.0	(3/2-)
15.65 m	^{135}Xe	526.56	11/2-
17.28 m 3	^{144}Pr	0.0	0-
17.4 m 2	^{80}Br	0	1+
17.8 m 1	^{88}Rb	0.0	2-
18.2 m 5	^{104}Tc	0.0	
18.27 m 7	^{141}Ba	0.0	
18.5 m	^{81}Se	0.0	(1/2)-
18.7 m 4	^{159}Eu	0	(5/2+)

Table 2 cont'd

' fission product half-lives ' continued

$T_{1/2}$	Nucleus	E(level)	Jπ
19.0 m 3	^{126}Sb	17.7	(5+)
19.1 m 4	^{94}Y	0.0	2-
19.5 m 3	^{163}Tb	0.0	3/2+
20.0 m 5	^{115}Ag	0.0	(1/2-)
20.2 m 2	^{126}Sb	35	
21.7 m 4	^{107}Rh	0.0	(5/2,7/2)+
22 m 1	^{111}Pd	0.0	(5/2+)
22.1 m 2	^{153}Sm	0	3/2-
22.5 m 2	^{83}Se	0.0	(9/2)+, 2.48
23 m 2	^{131}Sb	0.0	(7/2+)
24.07 m 13	^{140}Pr	0.0	
24.99 m 2	^{128}I	0.0	1+
25.0 m 1	^{131}Te	0.0	3/2+
31.80 m 8	^{84}Br	0.0	2-
32.2 m 1	^{136}Cs	0.0	3(-)
40 m 1	^{130}Sb	0.0	(8-)
40.08 m	^{123}Sn	24	(3/2)+
41.8 m 8	^{134}Te	0.0	0+
45.9 m	^{150}Eu	0	(1-)
48.7 m 2	^{111}Cd	396.0	11/2-
49.71 m 4	^{91}Y	555.57 5	9/2+
50.3 m 2	^{115}Cd	0.0	0+
51.5 m 10	^{98}Nb		(4,5)
52.6 m	^{134}I	0.0	
53 m 2	^{135}Cs	1627	(19/2-)
54.15 m	^{116}In	127.268	5+
55.4 m 4	^{133}Te	334.2	(11/2-)
56.12 m 1	^{103}Rh	39.750 7	7/2+
57.25 m	^{81}Se	102.89	(7/2)+
60.0 m 4	^{128}Sn	0.0	0+
69.6 m	^{129}Te	0	3/2+
72.1 m 7	^{97}Nb	0.0	(9/2)+
76.4 m 10	^{87}Kr	0.0	(5/2+)
82.78 m 4	^{75}Ge	0.0	1/2-
83.6 m 17	^{132}I	≈ 120	(8-)
84.9 m 11	^{139}Ba	0	(7/2)-
87 m 1	^{78}Ge	0.0	0+
90.7 m 2	^{76}As	0.0	(2-)
92.7 m 7	^{142}La	0.0	2-
1.73 h 1	^{149}Nd	0.0 2	5/2-
1.83 h 2	^{83}Kr	41.56 2	1/2-
2.10 h 4	^{127}Sn	0	(11/2-)
132 m 1	^{106}Rh	≈ 138	(4,5,6)+
2.30 h 3	^{132}I	0	4+
2.39 h 2	^{83}Br	0.0	(3/2)-
2.68 h 2	^{150}Pm	0	(1-)
2.71 h 1	^{92}Sr	0.0	0+
2.84 h 3	^{88}Kr	0.0	0+
2.90 h 1	^{134}Cs	138.738	8(-)
3.12 h 1	^{112}Ag	0.0	2(-)
3.19 h 1	^{90}Y	682.04 5	7+
3.54 h 1	^{92}Y	0.0	2-
3.93 h 6	^{141}La	0.0	(7/2+)
4.3 h 1	^{115}In	336.23 5	1/2-
4.32 h 3	^{129}Sb	0.0	7/2+
4.36 h 9	^{85}Kr	305.0 2	1/2-
4.42 h 1	^{80}Br	85.95 80	5-
4.44 h 5	^{105}Ru	0.0	(3/2+)
4.91 h 5	^{73}Ga	0.0	(3/2)-
5.37 h 5	^{113}Ag	0.0	1/2(-)
5.5 h 1	^{111}Pd	172.0 1	(11/2-)
5.98 h 2	^{145}Pr	0.00	(7/2+,5/2+)
6.02 h 2	^{99}Tc	142.63 3	(1/2)-
6.61 h	^{135}I	0	7/2+
9.01 h 3	^{128}Sb	0.0	
9.083 h	^{135}Xe	0.0	3/2+
9.35 h 7	^{127}Te	0	3/2+
9.4 h 2	^{156}Sm	0.0	0+
9.5 h	^{91}Sr	0	5/2+
10.1 h 2	^{93}Y	0.0	1/2-

Table 2 cont'd

' fission product half-lives ' continued

$T_{1/2}$	Nucleus	E(level)	Jπ
11.30 h 1	^{77}Ge	0.0	(7/2+)
12.36 h	^{130}I	0.0	5+
13.46 h 2	^{109}Pd	0.0	5/2+
14.1 h 2	^{72}Ga	0.0	3-
15.15 h 4	^{157}Eu	0	(5/2+)
17.0 h 2	^{97}Zr	0.0	1/2+
17.0 h 2	^{97}Zr	0.0	1/2+
18.56 h 8	^{159}Gd	0.0	3/2-
19.13 h 4	^{142}Pr	0.0	2-
20.12 h 6	^{112}Pd	0.0	0+
20.8 h 1	^{133}I	0.0	7/2+
23.35 h 5	^{96}Nb	0.0	(6+)
26.32 h 7	^{76}As	0.0	2-
27.06 h 4	^{121}Sn	0.0	3/2+
28.40 h 4	^{151}Pm	0.0	5/2+
28.7 h 2	^{135}Ba	268.24	11/2-
30 h 2	^{131}Te	182.25 2	11/2-
33.0 h 2	^{143}Ce	0.0	3/2-
35.30 h 2	^{82}Br	0.0	5-
35.36 h 5	^{105}Rh	0.0	(7/2)+
38.8 h 3	^{77}As	0.0	3/2-
40.22 h 2	^{140}La	0.0	3-
46.5 h	^{72}Zn	0.0	0+
46.7 h 1	^{153}Sm	0.0	3/2+
2.191 d 30	^{133}Xe	233.5	11/2-
53.08 h 5	^{149}Pm	0.0 1	7/2+
53.46 h 10	^{115}Cd	0.0	1/2+
64.1 h 1	^{90}Y	0.0	2-
2.70 d 1	^{122}Sb	0	2-
66.02 h 1	^{99}Mo	0	1/2+
78.2 h 8	^{132}Te	0	0+
86.6 h 8	^{95}Nb	234.70 14	(1/2)-
3.83 d 4	^{127}Sb	0	(7/2)+
5.29 d 1	^{133}Xe	0.0	3/2+
5.37 d 1	^{148}Pm	0.0	1-
6.91 d 2	^{161}Tb	0	3/2+
7.47 d 2	^{111}Ag	0.0	1/2-
8.0 d	^{129}Xe	236.14	11/2-
8.04 d 1	^{131}I	0.0	7/2+
9.64 d	^{125}Sn	0	11/2-
11.9 d 1	^{131}Xe	163.930 13	11/2-
12.4 d 1	^{126}Sb	0.0	(8-)
12.789 d 6	^{140}Ba	0	0+
12.98 d	^{136}Cs	0.0	5+
13.56 d 2	^{143}Pr	0.0	7/2+
14.0 d 3	^{117}Sn	314.58 4	11/2-
15.19 d 6	^{156}Eu	0.0	0+
18.66 d 2	^{86}Rb	0.0	2-
32.50 d 7	^{141}Ce	0.0	7/2-
33.6 d	^{129}Te	105.50	11/2-
35.15 d 3	^{95}Nb	0.0	(9/2)+
39.35 d 5	^{103}Ru	0.0	5/2+
41.3 d 1	^{146}Pm	137.2 2	6-
44.6 d 3	^{115}Cd	173.4 3	11/2-
50.55 d 9	^{89}Sr	0.0	5/2+
58 d	^{125}Te	144.73	11/2-
58.51 d 6	^{91}Y	0.0	1/2-
60.20 d 2	^{124}Sb	0.0	3-
63.98 d 6	^{95}Zr	0.0	(5/2)+
72.3 d 2	^{160}Tb	0.0	3-
109 d 2	^{127}Te	88.26	11/2-
129.2 d	^{123}Sn	0	(11/2)-
245 d 20	^{119}Sn	89.54 1	11/2-
250.4 d 10	^{110}Ag	117.69 10	6+
284.31 d 24	^{144}Ce	0.0	0+
368.2 d 12	^{106}Ru	0.0	0+
2.062 y 5	^{134}Cs	0.0	4(+)
2.6234 y 2	^{147}Pm	0.0	7/2+
4.96 y 1	^{155}Eu	0.0	5/2+
10.73 y 6	^{85}Kr	0.0	9/2+

Table 2 cont'd

' fission product half-lives ' continued

$T_{1/2}$	Nucleus	E(level)	Jπ
13.6 y 3	^{93m}Nb	30.4 3	1/2-
13.6 y 2	^{113}Cd	263.7 3	11/2-
28.6 y 3	^{90}Sr	0.0	0+
30.17 y 3	^{137}Cs	0.0	7/2+
≈ 50 y	^{121}Sn	8 6	(11/2)-
90 y 6	^{151}Sm	0.0	5/2-
127 y	^{108}Ag	109.58 8	(6)+
2.03×10^{4} y 16	^{94}Nb	0.0	6+
≤ 65000. y	^{79}Se	0.0	7/2+
2.13×10^{5} y 5	^{99}Tc	0	9/2+
≈ 10×10^{5} y	^{126}Sn	0.0	0+
1.53×10^{6} y 10	^{93}Zr	0.0	5/2+
2.3×10^{6} y	^{135}Cs	0.0	7/2+
6.5×10^{6} y 3	^{107}Pd	0	(5/2)+
1.57×10^{7} y	^{129}I	0	7/2+
4.7×10^{10} y 1	^{87}Rb	0.0	3/2-
1.07×10^{11} y 1	^{147}Sm	0.0	7/2-
5.1×10^{14} y 4	^{115}In	0.0	9/2+
2.1×10^{15} y 4	^{144}Nd	0.0	0+
> 3×10^{15} y	^{113}Cd	0.0	1/2+
8×10^{15} y 2	^{148}Sm	0.0	0+
> 1×10^{16} y	^{149}Sm	0.0	7/2-
> 5×10^{16} y	^{142}Ce	0.0	0+
> 5×10^{17} y	^{150}Nd	0	0+
1.4×10^{20} y	^{82}Se	0.0	0+
2.51×10^{21} y 27	^{130}Te	0.0	0+
stable	^{115}Sn	0.0	1/2+

Annex NO 5

Q calculated with the different mass prediction formulae, compared to the Wapstra + Bos[1] evaluated data

ELEMENT	Z	N	A	AYE/S[2]	GR/HI/TA[3]	SEE/HO[4]	LIR/ZEL[5]	BE/LOM[6]	JA/GA/KE[7]	CO/KE[8]	JA/EY[9]	WAP/BOS[10]
CU	29	38	67	590.0	910.0	700.0	1120.0	-600.0	600.0	620.0	530.0	575.0
ZN	30	37	67	-1530.0	-1360.0	-1100.0	-1370.0	-1800.0	-1070.0	-1030.0	-1030.0	-1001.0
CU	29	39	68	4040.0	4280.0	5100.0	4990.0	0.0	4610.0	4640.0	4490.0	4616.0
ZN	30	38	68	-3280.0	-2820.0	-3500.0	-3160.0	0.0	-3120.0	-3010.0	-3060.0	-2921.0
GA	31	37	68	80.0	-220.0	400.0	-20.0	0.0	-120.0	-150.0	40.0	-114.0
CU	29	40	69	2210.0	2780.0	2400.0	2940.0	1100.0	2620.0	2490.0	2370.0	2477.0
ZN	30	39	69	180.0	560.0	800.0	580.0	100.0	900.0	970.0	900.0	905.0
GA	31	38	69	-1620.0	-1680.0	-1900.0	-1820.0	-3000.0	-2160.0	-2250.0	-2000.0	-2226.0
GE	32	37	69	-3630.0	-3910.0	-3800.0	-4240.0	-4900.0	-4170.0	-4040.0	-3950.0	-3966.0
CU	29	41	70	5590.0	6150.0	6800.0	6760.0	0.0	6510.0	6370.0	6210.0	6330.0
ZN	30	40	70	-1610.0	-910.0	-1600.0	-1330.0	0.0	-1090.0	-1030.0	-1220.0	-655.0
GA	31	39	70	1820.0	1670.0	2200.0	2050.0	0.0	1850.0	1750.0	1960.0	1656.0
GE	32	38	70	-5280.0	-5320.0	-6000.0	-6030.0	0.0	-6210.0	-6200.0	-5990.0	-6222.0
AS	33	37	70	-2250.0	-2760.0	-2200.0	-2760.0	0.0	-2650.0	-2380.0	-2430.0	-2749.0
CU	29	42	71	3780.0	4430.0	4200.0	4620.0	3200.0	4350.0	4260.0	4010.0	0.0
ZN	30	41	71	1760.0	2470.0	2500.0	2350.0	2000.0	2800.0	2810.0	2620.0	2820.0
GA	31	40	71	-20.0	200.0	-100.0	80.0	-1100.0	-140.0	-330.0	-160.0	-235.0
GE	32	39	71	-1790.0	-1980.0	-2100.0	-2250.0	-3000.0	-2200.0	-2120.0	-2040.0	-2009.0
AS	33	38	71	-3890.0	-4170.0	-4300.0	-4610.0	-5000.0	-4690.0	-4500.0	-4460.0	0.0
CU	29	43	72	7080.0	7440.0	8200.0	8310.0	0.0	8160.0	8040.0	7830.0	0.0
ZN	30	42	72	-10.0	780.0	100.0	410.0	0.0	630.0	610.0	420.0	458.0
GA	31	41	72	3230.0	3550.0	3900.0	3890.0	0.0	3750.0	3740.0	3670.0	3690.0
GE	32	40	72	-3520.0	-3410.0	-4300.0	-4150.0	0.0	-4180.0	-4360.0	-4160.0	-4351.0
AS	33	39	72	-420.0	-860.0	-300.0	-640.0	0.0	-680.0	-380.0	-510.0	-340.0
CU	29	44	73	5280.0	5730.0	5700.0	6210.0	5400.0	5950.0	5800.0	5650.0	0.0
ZN	30	43	73	3280.0	3800.0	3800.0	4000.0	3200.0	4450.0	4460.0	4240.0	4700.0
GA	31	42	73	1360.0	1870.0	1800.0	1810.0	1000.0	1590.0	1550.0	1480.0	1564.0
GE	32	41	73	-220.0	-50.0	-300.0	-410.0	-1000.0	-300.0	-340.0	-330.0	-337.0
AS	33	40	73	-2100.0	-2290.0	-2600.0	-2670.0	-3200.0	-2660.0	-2720.0	-2640.0	-2740.0
SE	34	39	73	-4170.0	-4430.0	-4400.0	-4960.0	-6500.0	-4720.0	-4630.0	-4480.0	-4637.0
CU	29	45	74	8500.0	8690.0	9700.0	9720.0	0.0	9630.0	9460.0	9430.0	0.0
ZN	30	44	74	1520.0	2130.0	1500.0	2150.0	0.0	2240.0	2160.0	2060.0	2100.0
GA	31	43	74	4600.0	4870.0	5300.0	5490.0	0.0	5400.0	5360.0	5300.0	5502.0
GE	32	42	74	-2120.0	-1650.0	-2400.0	-2350.0	0.0	-2460.0	-2510.0	-2530.0	-2562.0
AS	33	41	74	1240.0	1020.0	1400.0	1250.0	0.0	1220.0	1340.0	1190.0	1351.0
SE	34	40	74	-5810.0	-5640.0	-6700.0	-6960.0	0.0	-6700.0	-6970.0	-6600.0	-6917.0
CU	29	46	75	6680.0	6990.0	7400.0	7810.0	7100.0	7520.0	7390.0	7260.0	0.0
ZN	30	45	75	4770.0	5100.0	5300.0	5600.0	4700.0	5910.0	5770.0	5840.0	0.0
GA	31	44	75	2820.0	3190.0	3300.0	3430.0	3100.0	3200.0	3170.0	3120.0	3296.0
GE	32	43	75	970.0	1390.0	1200.0	1290.0	500.0	1350.0	1220.0	1300.0	1176.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/HO [4]	LIR/ZEL [5]	BE/LOM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BOS [10]
AS	33	42	75	-550.0	-550.0	-800.0	-880.0	-1200.0	-940.0	-820.0	-1010.0	-865.0
SE	34	41	75	-2440.0	-2560.0	-2800.0	-3080.0	-3400.0	-2820.0	-2940.0	-2780.0	-3010.0
CU	29	47	76	9860.0	9910.0	10900.0	11110.0	0.0	10930.0	10820.0	10660.0	0.0
ZN	30	46	76	2980.0	3430.0	3100.0	3930.0	0.0	3810.0	3640.0	3680.0	0.0
GA	31	45	76	6030.0	6140.0	7000.0	6920.0	0.0	6860.0	6730.0	6900.0	0.0
GE	32	44	76	-850.0	-280.0	-1000.0	-600.0	0.0	-850.0	-940.0	-880.0	-924.0
AS	33	43	76	2520.0	2500.0	3100.0	2890.0	0.0	2870.0	2930.0	2810.0	2969.0
SE	34	42	76	-4140.0	-4080.0	-5000.0	-5100.0	0.0	-4970.0	-4980.0	-4980.0	-4957.0
BR	35	41	76	-1130.0	-1500.0	-1300.0	-1300.0	0.0	-1190.0	-1070.0	-1230.0	-1201.0
CU	29	48	77	8050.0	8220.0	9000.0	9530.0	8700.0	9080.0	8850.0	8590.0	0.0
ZN	30	47	77	6150.0	6350.0	6500.0	7230.0	6200.0	7210.0	7090.0	7060.0	0.0
GA	31	46	77	4270.0	4480.0	4800.0	5030.0	4800.0	4760.0	4620.0	4740.0	0.0
GE	32	45	77	2360.0	2600.0	2800.0	2880.0	1900.0	2820.0	2700.0	2900.0	2707.0
AS	33	44	77	540.0	890.0	800.0	770.0	1000.0	670.0	700.0	640.0	684.0
SE	34	43	77	-960.0	-1010.0	-1300.0	-1340.0	-2000.0	-1170.0	-1320.0	-1160.0	-1365.0
BR	35	42	77	-2800.0	-3000.0	-3300.0	-3500.0	-3700.0	-3340.0	-3060.0	-3440.0	-3000.0
CU	29	49	78	11170.0	11090.0	12200.0	12570.0	0.0	12420.0	12030.0	11880.0	0.0
ZN	30	48	78	4370.0	4710.0	4600.0	5810.0	0.0	5360.0	5120.0	5010.0	0.0
GA	31	47	78	7420.0	7370.0	8300.0	8270.0	0.0	8170.0	8000.0	8150.0	0.0
GE	32	46	78	610.0	970.0	400.0	1160.0	0.0	710.0	640.0	730.0	980.0
AS	33	45	78	3670.0	3700.0	4400.0	4330.0	0.0	4340.0	4320.0	4420.0	4290.0
SE	34	44	78	-2910.0	-2550.0	-3200.0	-3330.0	0.0	-3370.0	-3540.0	-3330.0	-3574.0
BR	35	43	78	400.0	40.0	400.0	390.0	0.0	470.0	610.0	370.0	694.0
KR	36	42	78	-6440.0	-6480.0	-7000.0	-7810.0	0.0	-7370.0	-7160.0	-7320.0	0.0
CU	29	50	79	9350.0	9430.0	10700.0	11550.0	10500.0	10700.0	10200.0	10070.0	0.0
ZN	30	49	79	7490.0	7570.0	7900.0	8960.0	-4000.0	8710.0	8290.0	8310.0	0.0
GA	31	48	79	5650.0	5730.0	6200.0	6670.0	18100.0	6320.0	6020.0	6070.0	0.0
GE	32	47	79	3780.0	3870.0	4100.0	4460.0	3400.0	4120.0	4070.0	4150.0	4300.0
AS	33	46	79	1920.0	2030.0	2200.0	2350.0	2700.0	2230.0	2220.0	2250.0	2199.0
SE	34	45	79	70.0	280.0	0.0	270.0	-600.0	300.0	130.0	450.0	156.0
BR	35	44	79	-1450.0	-1470.0	-1200.0	-1810.0	-1500.0	-1740.0	-1680.0	-1800.0	-1631.0
KR	36	43	79	-3200.0	-3420.0	-3700.0	-3910.0	-4500.0	-3560.0	-3460.0	-3510.0	-3584.0
CU	29	51	80	14800.0	14950.0	15200.0	17550.0	0.0	15500.0	15120.0	14800.0	0.0
ZN	30	50	80	5720.0	5940.0	6100.0	7660.0	0.0	6980.0	6430.0	6490.0	0.0
GA	31	49	80	8740.0	8570.0	9500.0	9640.0	0.0	9670.0	9190.0	9370.0	0.0
GE	32	48	80	2030.0	2270.0	2100.0	2960.0	0.0	2270.0	2030.0	2080.0	0.0
AS	33	47	80	5070.0	4900.0	5700.0	5640.0	0.0	5630.0	5690.0	5670.0	5997.0
SE	34	46	80	-1660.0	-1400.0	-2100.0	-1580.0	0.0	-1800.0	-1930.0	-1720.0	-1871.0
BR	35	45	80	1440.0	1370.0	2000.0	1850.0	0.0	1930.0	1950.0	1990.0	2011.0
KR	36	44	80	-4960.0	-4910.0	-5300.0	-6000.0	0.0	-5770.0	-5710.0	-5690.0	-5707.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/HO [4]	LIR/ZEL [5]	BE/LOM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BUS [1]
CU	29	52	81	12970.0	13330.0	14000.0	15210.0	14000.0	14590.0	14200.0	14010.0	0.0
ZN	30	51	81	11150.0	11470.0	10900.0	13060.0	10700.0	11770.0	11330.0	11230.0	0.0
GA	31	50	81	6970.0	6940.0	7800.0	8540.0	8200.0	7940.0	7360.0	7560.0	0.0
GE	32	49	81	5140.0	5120.0	5200.0	6100.0	4800.0	5620.0	5290.0	5380.0	0.0
AS	33	48	81	3320.0	3290.0	3900.0	3920.0	4200.0	3790.0	3770.0	3600.0	3797.0
SE	34	47	81	1480.0	1480.0	1400.0	1820.0	1100.0	1600.0	1530.0	1700.0	1584.0
BR	35	46	81	-340.0	-340.0	-200.0	-240.0	200.0	-180.0	-290.0	-190.0	-311.0
KR	36	45	81	-2050.0	-2030.0	-2000.0	-2270.0	-3000.0	-2090.0	-2170.0	-1900.0	-2262.0
RB	37	44	81	-3630.0	-3850.0	-3700.0	-4330.0	-4000.0	-4270.0	-3880.0	-4090.0	-3986.0
CU	29	53	82	15960.0	16130.0	16700.0	18830.0	0.0	16890.0	16680.0	16360.0	0.0
ZN	30	52	82	9350.0	9880.0	9700.0	10810.0	0.0	10880.0	10380.0	10430.0	0.0
GA	31	51	82	12360.0	12430.0	12500.0	14420.0	0.0	12730.0	12270.0	12310.0	0.0
GE	32	50	82	3400.0	3520.0	3600.0	4730.0	0.0	3890.0	3360.0	3570.0	0.0
AS	33	49	82	6400.0	6130.0	6900.0	6930.0	0.0	7130.0	6980.0	6910.0	0.0
SE	34	48	82	-220.0	-100.0	-400.0	170.0	0.0	-250.0	-300.0	-380.0	-84.0
BR	35	47	82	2770.0	2510.0	3300.0	3140.0	0.0	3240.0	3130.0	3230.0	3093.0
KR	36	46	82	-3850.0	-3710.0	-4300.0	-4250.0	0.0	-4210.0	-4300.0	-4070.0	-4376.0
RB	37	45	82	-710.0	-1000.0	-200.0	-590.0	0.0	-600.0	-270.0	-310.0	-219.0
CU	29	54	83	14130.0	14510.0	15100.0	16340.0	15000.0	15860.0	15440.0	15320.0	0.0
ZN	30	53	83	12350.0	12690.0	12400.0	14440.0	12000.0	13180.0	12860.0	12790.0	0.0
GA	31	52	83	10570.0	10850.0	11400.0	12700.0	12100.0	11830.0	11330.0	11510.0	0.0
GE	32	51	83	8780.0	9010.0	8200.0	9830.0	8200.0	8580.0	8250.0	8320.0	0.0
AS	33	50	83	4660.0	4530.0	5400.0	5620.0	5900.0	5410.0	5060.0	5100.0	0.0
SE	34	49	83	2860.0	2740.0	2600.0	3380.0	2500.0	3100.0	3120.0	2930.0	3579.0
BR	35	48	83	1070.0	930.0	1500.0	1300.0	1700.0	1380.0	1100.0	1170.0	963.0
KR	36	47	83	-730.0	-850.0	-1000.0	-740.0	-1600.0	-790.0	-930.0	-660.0	-996.0
RB	37	46	83	-2530.0	-2640.0	-2300.0	-2750.0	-2100.0	-2710.0	-2370.0	-2460.0	-2250.0
CU	29	55	84	17070.0	17280.0	18000.0	19900.0	0.0	18300.0	18130.0	17790.0	0.0
ZN	30	54	84	10550.0	11100.0	11000.0	12230.0	0.0	12160.0	11610.0	11760.0	0.0
GA	31	53	84	13540.0	13640.0	14100.0	16170.0	0.0	14130.0	13810.0	13870.0	0.0
GE	32	52	84	7020.0	7460.0	7100.0	8290.0	0.0	7780.0	7280.0	7530.0	0.0
AS	33	51	84	10000.0	9990.0	10000.0	10980.0	0.0	10200.0	9760.0	9840.0	0.0
SE	34	50	84	1150.0	1160.0	1200.0	1850.0	0.0	1370.0	1310.0	1130.0	1617.0
BR	35	49	84	4130.0	3750.0	4500.0	4380.0	0.0	4730.0	4540.0	4470.0	4672.0
KR	36	48	84	-2410.0	-2400.0	-2900.0	-2540.0	0.0	-2640.0	-2800.0	-2720.0	-2680.0
RB	37	47	84	570.0	190.0	900.0	700.0	0.0	690.0	960.0	930.0	889.0
SR	38	46	84	-5980.0	-5950.0	-6300.0	-6820.0	0.0	-6680.0	-6660.0	-6360.0	-6949.0
CU	29	56	85	15240.0	15660.0	16400.0	17380.0	16300.0	17170.0	16700.0	16630.0	0.0
ZN	30	55	85	13500.0	13870.0	14000.0	15760.0	13500.0	14590.0	14300.0	14230.0	0.0
GA	31	54	85	11750.0	12070.0	12700.0	14200.0	13200.0	13110.0	12570.0	12650.0	0.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/HO [4]	LIR/ZEL [5]	BE/LOM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BUS [7]
GE	32	53	85	9990.0	10250.0	9800.0	11630.0	9600.0	10080.0	9760.0	9880.0	0.0
AS	33	52	85	8250.0	8440.0	8800.0	9830.0	9900.0	9300.0	8910.0	9060.0	0.0
SE	34	51	85	6480.0	6620.0	5700.0	6510.0	6000.0	6160.0	5970.0	5860.0	0.0
BR	35	50	85	2420.0	2180.0	3100.0	2850.0	3500.0	3010.0	2800.0	2670.0	2802.0
KR	36	49	85	650.0	420.0	300.0	780.0	-200.0	700.0	500.0	580.0	687.0
RB	37	48	85	-1100.0	-1350.0	-900.0	-1250.0	-500.0	-1150.0	-1070.0	-1130.0	-1064.0
SR	38	47	85	-2830.0	-3120.0	-3400.0	-3220.0	-3900.0	-3280.0	-3220.0	-2970.0	-3260.0
CU	29	57	86	18130.0	18380.0	19600.0	20890.0	0.0	20040.0	19660.0	19480.0	0.0
ZN	30	56	86	11690.0	12290.0	12400.0	13440.0	0.0	13460.0	12860.0	13090.0	0.0
GA	31	55	86	14630.0	14810.0	15600.0	17570.0	0.0	15540.0	15280.0	15320.0	0.0
GE	32	54	86	8240.0	8700.0	8400.0	9960.0	0.0	9060.0	8560.0	8860.0	0.0
AS	33	53	86	11200.0	11220.0	11700.0	12960.0	0.0	11600.0	11420.0	11420.0	0.0
SE	34	52	86	4750.0	5100.0	4500.0	5510.0	0.0	5260.0	4980.0	5100.0	0.0
BR	35	51	86	7720.0	7610.0	7500.0	7710.0	0.0	7800.0	7580.0	7410.0	7304.0
KR	36	50	86	-1030.0	-1120.0	-1100.0	-880.0	0.0	-1030.0	-1050.0	-1210.0	-526.0
RB	37	49	86	1930.0	1440.0	2100.0	1900.0	0.0	2200.0	1930.0	2170.0	1775.0
SR	38	48	86	-4530.0	-4630.0	-5100.0	-5130.0	0.0	-5130.0	-5220.0	-5040.0	-5273.0
CU	29	58	87	15310.0	16780.0	17600.0	18360.0	17900.0	18560.0	18120.0	18000.0	0.0
ZN	30	57	87	14600.0	15010.0	15500.0	16900.0	14900.0	16330.0	15820.0	15930.0	0.0
GA	31	56	87	12880.0	13240.0	14000.0	15430.0	14500.0	14420.0	13860.0	14180.0	0.0
GE	32	55	87	11170.0	11450.0	11400.0	13220.0	11100.0	11490.0	11250.0	11340.0	0.0
AS	33	54	87	9440.0	9680.0	10200.0	11500.0	11100.0	10570.0	10200.0	10400.0	0.0
SE	34	53	87	7720.0	7880.0	7400.0	8470.0	7500.0	7570.0	7400.0	7460.0	0.0
BR	35	52	87	5970.0	6090.0	6300.0	6910.0	7900.0	6890.0	6600.0	6640.0	0.0
KR	36	51	87	4260.0	4300.0	3300.0	3510.0	3000.0	3770.0	3710.0	3530.0	3890.0
RB	37	50	87	250.0	-90.0	700.0	270.0	1000.0	470.0	280.0	380.0	273.0
SR	38	49	87	-1470.0	-1840.0	-2000.0	-1750.0	-2100.0	-1780.0	-2000.0	-1740.0	-1861.0
Y	39	48	87	-3220.0	-3570.0	-3200.0	-3710.0	-3300.0	-3960.0	-3900.0	-3780.0	-3578.0
ZN	30	58	88	12790.0	13440.0	13600.0	14490.0	0.0	14860.0	14280.0	14460.0	0.0
GA	31	57	88	15770.0	15940.0	17100.0	18750.0	0.0	17270.0	16840.0	17030.0	0.0
GE	32	56	88	9400.0	9910.0	9800.0	11340.0	0.0	10370.0	9830.0	10200.0	0.0
AS	33	55	88	12360.0	12400.0	13200.0	14570.0	0.0	13010.0	12900.0	12890.0	0.0
SE	34	54	88	5980.0	6370.0	6000.0	7260.0	0.0	6540.0	6180.0	6430.0	0.0
BR	35	53	88	8930.0	8860.0	9200.0	9640.0	0.0	9200.0	9070.0	9010.0	0.0
KR	36	52	88	2560.0	2810.0	2200.0	2790.0	0.0	2860.0	2640.0	2750.0	2908.0
RB	37	51	88	5490.0	5290.0	5100.0	4880.0	0.0	5270.0	5250.0	5130.0	5306.0
SR	38	50	88	-3120.0	-3340.0	-3500.0	-3450.0	0.0	-3510.0	-3740.0	-3540.0	-3619.0
Y	39	49	88	-200.0	-800.0	-100.0	-550.0	0.0	-610.0	-790.0	-470.0	-677.0
GA	31	58	89	13970.0	14320.0	15200.0	16450.0	15600.0	15800.0	15320.0	15560.0	0.0
GE	32	57	89	12260.0	12560.0	12900.0	14580.0	13100.0	13240.0	12810.0	13060.0	0.0

ELEMENT	Z	N	A	MYERS[2]	GR/HI/TA[3]	SEE/HO[4]	LIR/ZEL[5]	BE/LOM[6]	JA/GA/KE[7]	CO/KE[8]	JA/EY[9]	WAP/BUS[1]
AS	33	56	89	10590.0	10860.0	11500.0	12870.0	12200.0	11880.0	11480.0	11750.0	0.0
SE	34	55	89	8910.0	9110.0	8900.0	10190.0	8900.0	8980.0	8860.0	8930.0	0.0
BR	35	54	89	7200.0	7340.0	7800.0	8580.0	9000.0	8170.0	7900.0	7980.0	0.0
KR	36	53	89	5510.0	5570.0	5000.0	5370.0	4900.0	5170.0	5010.0	5120.0	4928.0
RB	37	52	89	3800.0	3810.0	4200.0	4170.0	5300.0	4350.0	4310.0	4350.0	4486.0
SR	38	51	89	2100.0	2040.0	800.0	980.0	1300.0	1290.0	1310.0	1220.0	1492.0
Y	39	50	89	-1850.0	-2300.0	-1500.0	-2190.0	-1400.0	-2340.0	-2760.0	-2280.0	-2831.0
ZR	40	49	89	-3550.0	-4210.0	-4200.0	-4180.0	-4700.0	-4430.0	-4280.0	-4490.0	-4251.0
GE	32	58	90	10470.0	10960.0	11000.0	12480.0	0.0	11760.0	11280.0	11590.0	0.0
AS	33	57	90	13330.0	13410.0	14700.0	15940.0	0.0	14750.0	14450.0	14610.0	0.0
SE	34	56	90	7070.0	7560.0	7300.0	8740.0	0.0	7850.0	7430.0	7790.0	0.0
BR	35	55	90	10100.0	10060.0	10700.0	11310.0	0.0	10610.0	10600.0	10480.0	0.0
KR	36	54	90	3810.0	4090.0	3700.0	4460.0	0.0	4140.0	3890.0	4100.0	4390.0
RB	37	53	90	6730.0	6550.0	6800.0	6590.0	0.0	6660.0	6570.0	6720.0	6317.0
SR	38	52	90	420.0	580.0	0.0	260.0	0.0	390.0	410.0	440.0	546.0
Y	39	51	90	3350.0	3050.0	2900.0	2500.0	0.0	2450.0	2270.0	2470.0	2276.0
ZR	40	50	90	-5170.0	-5690.0	-5800.0	-5890.0	0.0	-6160.0	-6250.0	-6290.0	-6111.0
AS	33	58	91	11560.0	11820.0	12500.0	14020.0	13300.0	13280.0	12920.0	13140.0	0.0
SE	34	57	91	9810.0	10080.0	10600.0	11690.0	10900.0	10710.0	10370.0	10660.0	0.0
BR	35	56	91	8160.0	8480.0	9100.0	10000.0	10200.0	9490.0	9200.0	9340.0	0.0
KR	36	55	91	6670.0	6810.0	6500.0	7080.0	6400.0	6570.0	6520.0	6600.0	6200.0
RB	37	54	91	5040.0	5070.0	5500.0	5720.0	6700.0	5640.0	5480.0	5700.0	5704.0
SR	38	53	91	3350.0	3330.0	2700.0	2610.0	2900.0	2690.0	2700.0	2810.0	2684.0
Y	39	52	91	1680.0	1590.0	1800.0	1710.0	2800.0	1550.0	1450.0	1700.0	1543.0
ZR	40	51	91	10.0	-350.0	-1200.0	-1180.0	-900.0	-1370.0	-1290.0	-1540.0	-1257.0
SE	34	58	92	8070.0	8510.0	8500.0	9990.0	0.0	9240.0	8830.0	9200.0	0.0
BR	35	57	92	10910.0	10960.0	12100.0	12800.0	0.0	12340.0	12150.0	12210.0	0.0
KR	36	56	92	4700.0	5250.0	5100.0	5920.0	0.0	5460.0	5190.0	5460.0	5970.0
RB	37	55	92	7850.0	7770.0	8400.0	8210.0	0.0	8060.0	7970.0	8200.0	7701.0
SR	38	54	92	1680.0	1880.0	1200.0	1790.0	0.0	1670.0	1700.0	1800.0	1930.0
Y	39	53	92	4600.0	4310.0	4600.0	3980.0	0.0	3850.0	3710.0	4060.0	3634.0
ZR	40	52	92	-1640.0	-1770.0	-2200.0	-2030.0	0.0	-2270.0	-2110.0	-2310.0	-2008.0
NB	41	51	92	1250.0	680.0	600.0	380.0	0.0	420.0	420.0	190.0	360.0
BR	35	58	93	9170.0	9400.0	10000.0	11230.0	11100.0	10870.0	10630.0	10750.0	0.0
KR	36	57	93	7450.0	7710.0	8100.0	8650.0	8300.0	8320.0	8090.0	8340.0	0.0
RB	37	56	93	5850.0	6210.0	6900.0	7100.0	8100.0	6950.0	6760.0	7060.0	0.0
SR	38	55	93	4510.0	4590.0	4300.0	4230.0	4400.0	4090.0	4110.0	4300.0	4186.0
Y	39	54	93	2930.0	2860.0	3100.0	3130.0	4100.0	2830.0	2730.0	3050.0	2890.0
ZR	40	53	93	1260.0	960.0	500.0	260.0	500.0	30.0	140.0	60.0	93.0
NB	41	52	93	-360.0	-740.0	-500.0	-530.0	800.0	-480.0	-370.0	-580.0	-407.0

ELEMENT	Z	N	A	MYERS[2]	GR/HI/TA[3]	SEE/HO[4]	LIR/ZEL[5]	BE/LOM[6]	JA/GA/KE[7]	CO/KE[8]	JA/EY[9]	WAP/BUS[1]
BR	35	59	94	12040.0	12030.0	12600.0	14120.0	0.0	13680.0	13610.0	13480.0	0.0
KR	36	58	94	5740.0	6160.0	6100.0	7240.0	0.0	6840.0	6580.0	6880.0	0.0
RB	37	57	94	8570.0	8650.0	9900.0	9710.0	0.0	9810.0	9670.0	9940.0	0.0
SR	38	56	94	2490.0	3070.0	2700.0	3200.0	0.0	2980.0	3000.0	3170.0	3421.0
Y	39	55	94	5750.0	5550.0	6100.0	5490.0	0.0	5260.0	5080.0	5550.0	4883.0
ZR	40	54	94	-370.0	-470.0	-800.0	-610.0	0.0	-1000.0	-970.0	-960.0	-897.0
NB	41	53	94	2530.0	1970.0	2200.0	1790.0	0.0	1830.0	2040.0	1790.0	2045.0
BR	35	60	95	10370.0	10570.0	11300.0	12320.0	12600.0	11880.0	11930.0	11720.0	0.0
KR	36	59	95	8600.0	8790.0	8900.0	10070.0	10600.0	9650.0	9520.0	9610.0	0.0
RB	37	58	95	6870.0	7080.0	7600.0	8360.0	8500.0	8330.0	8240.0	8490.0	0.0
SR	38	57	95	5200.0	5490.0	5800.0	5770.0	6300.0	5840.0	5780.0	6040.0	0.0
Y	39	56	95	3740.0	4090.0	4600.0	4440.0	5400.0	4140.0	4080.0	4420.0	4430.0
ZR	40	55	95	2490.0	2220.0	2100.0	1790.0	2200.0	1440.0	1330.0	1550.0	1122.0
NB	41	54	95	870.0	550.0	1000.0	820.0	1900.0	790.0	890.0	780.0	926.0
MO	42	53	95	-740.0	-1130.0	-1600.0	-1740.0	-1100.0	-1780.0	-1660.0	-1850.0	-1700.0
KR	36	60	96	6970.0	7350.0	7400.0	8420.0	0.0	7840.0	7830.0	7850.0	0.0
RB	37	59	96	9700.0	9680.0	10500.0	11100.0	0.0	11150.0	11220.0	11220.0	0.0
SR	38	58	96	3500.0	3920.0	3900.0	4500.0	0.0	4360.0	4330.0	4590.0	0.0
Y	39	57	96	6370.0	6520.0	7400.0	6950.0	0.0	7000.0	6970.0	7300.0	0.0
ZR	40	56	96	570.0	790.0	700.0	740.0	0.0	320.0	150.0	420.0	166.0
NB	41	55	96	3730.0	3220.0	3800.0	3240.0	0.0	3230.0	3260.0	3280.0	3188.0
MO	42	54	96	-2370.0	-2520.0	-2900.0	-2780.0	0.0	-2820.0	-2880.0	-2860.0	-2974.0
KR	36	61	97	9820.0	10000.0	10500.0	11360.0	11200.0	10920.0	10910.0	10990.0	0.0
RB	37	60	97	8070.0	8230.0	8800.0	9520.0	10200.0	9340.0	9540.0	9460.0	0.0
SR	38	59	97	6330.0	6520.0	6800.0	7220.0	7900.0	7170.0	7240.0	7330.0	0.0
Y	39	58	97	4650.0	4890.0	5400.0	5670.0	6400.0	5530.0	5680.0	5860.0	0.0
ZR	40	57	97	3110.0	3320.0	3700.0	3280.0	4000.0	3180.0	2930.0	3290.0	2662.0
NB	41	56	97	1930.0	1800.0	2400.0	2090.0	3200.0	2110.0	1930.0	2150.0	1927.0
MO	42	55	97	490.0	150.0	0.0	-240.0	400.0	-380.0	-290.0	-350.0	-320.0
KR	36	62	98	8200.0	8580.0	8700.0	9540.0	0.0	9020.0	8990.0	9070.0	0.0
RB	37	61	98	10840.0	10860.0	11900.0	12370.0	0.0	12420.0	12620.0	12610.0	0.0
SR	38	60	98	4730.0	5090.0	5100.0	5730.0	0.0	5370.0	5490.0	5570.0	0.0
Y	39	59	98	7450.0	7440.0	8300.0	8330.0	0.0	8330.0	8340.0	8600.0	0.0
ZR	40	58	98	1390.0	1730.0	1800.0	2000.0	0.0	1710.0	1860.0	1840.0	2222.0
NB	41	57	98	4380.0	4330.0	5400.0	4660.0	0.0	4970.0	4730.0	5030.0	4605.0
MO	42	56	98	-1150.0	-1250.0	-1500.0	-1450.0	0.0	-1510.0	-1650.0	-1480.0	-1582.0
TC	43	55	98	1730.0	1160.0	1600.0	1330.0	0.0	1390.0	1760.0	1390.0	1793.0
RB	37	62	99	9280.0	9450.0	10100.0	10620.0	12100.0	10510.0	10700.0	10690.0	0.0
SR	38	61	99	7540.0	7700.0	8000.0	8560.0	8900.0	8440.0	8540.0	8720.0	0.0
Y	39	60	99	5840.0	6000.0	6500.0	6840.0	7900.0	6540.0	6500.0	6830.0	0.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/HO [4]	LIR/ZEL [5]	BE/LOM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BUS [10]
ZR	40	59	99	4160.0	4290.0	4800.0	4700.0	5300.0	4510.0	4720.0	4590.0	4495.0
NB	41	58	99	2610.0	2710.0	3400.0	3300.0	4300.0	3490.0	3530.0	3590.0	3625.0
MO	42	57	99	1370.0	1340.0	1700.0	1210.0	2300.0	1360.0	1260.0	1390.0	1356.0
TC	43	56	99	120.0	-240.0	100.0	10.0	800.0	270.0	310.0	270.0	294.0
RU	44	55	99	-1120.0	-1350.0	-1200.0	-1300.0	0.0	-1300.0	-1200.0	-1250.0	-1253.0
RH	45	54	99	-2400.0	-2600.0	-2500.0	-2600.0	0.0	-2600.0	-2500.0	-2550.0	-2553.0
RB	37	63	100	12090.0	12050.0	12900.0	13550.0	0.0	13630.0	13790.0	13830.0	0.0
SR	38	62	100	5950.0	6320.0	6600.0	6890.0	0.0	6540.0	6610.0	6800.0	0.0
Y	39	61	100	8650.0	8590.0	9100.0	9630.0	0.0	9610.0	9620.0	9990.0	0.0
ZR	40	60	100	2550.0	2880.0	3000.0	3210.0	0.0	2710.0	3020.0	2830.0	0.0
NB	41	59	100	5330.0	5220.0	6200.0	6000.0	0.0	6300.0	6480.0	6330.0	5999.0
MO	42	58	100	-520.0	-250.0	-100.0	-180.0	0.0	-110.0	-230.0	-50.0	-170.0
TC	43	57	100	2310.0	2370.0	3300.0	2740.0	0.0	3120.0	3190.0	3140.0	3203.0
RU	44	56	100	-3060.0	-3240.0	-3400.0	-3480.0	0.0	-3490.0	-3610.0	-3520.0	-3630.0
SR	38	63	101	8760.0	8920.0	9300.0	9810.0	8300.0	9660.0	9680.0	9940.0	0.0
Y	39	62	101	7050.0	7200.0	7600.0	7960.0	9700.0	7700.0	7720.0	8070.0	0.0
ZR	40	61	101	5360.0	5480.0	5900.0	6030.0	6900.0	5790.0	6100.0	5990.0	0.0
NB	41	60	101	3710.0	3800.0	4200.0	4450.0	5200.0	4500.0	4770.0	4570.0	0.0
MO	42	59	101	2150.0	2220.0	2900.0	2590.0	3700.0	2690.0	2740.0	2700.0	2811.0
TC	43	58	101	920.0	840.0	1300.0	1240.0	1800.0	1650.0	1620.0	1700.0	1625.0
RU	44	57	101	-250.0	-620.0	-200.0	-600.0	300.0	-630.0	-600.0	-640.0	-542.0
SR	38	64	102	7170.0	7540.0	7900.0	8020.0	0.0	7810.0	7900.0	8100.0	0.0
Y	39	63	102	9840.0	9790.0	10300.0	10830.0	0.0	10830.0	10820.0	11210.0	0.0
ZR	40	62	102	3790.0	4110.0	4200.0	4380.0	0.0	3880.0	4190.0	4070.0	0.0
NB	41	61	102	6490.0	6380.0	7200.0	7260.0	0.0	7580.0	7860.0	7730.0	0.0
MO	42	60	102	520.0	780.0	1000.0	1000.0	0.0	890.0	1050.0	950.0	0.0
TC	43	59	102	3440.0	3240.0	4200.0	4070.0	0.0	4460.0	4610.0	4440.0	0.0
RU	44	58	102	-1960.0	-2030.0	-2100.0	-2160.0	0.0	-2110.0	-2290.0	-2080.0	-2323.0
SR	38	65	103	9950.0	10120.0	10500.0	10960.0	11200.0	10780.0	10730.0	11000.0	0.0
Y	39	64	103	8250.0	8410.0	8800.0	9060.0	9800.0	8980.0	9060.0	9370.0	0.0
ZR	40	63	103	6580.0	6700.0	7000.0	7260.0	7300.0	7000.0	7270.0	7220.0	0.0
NB	41	62	103	4910.0	5010.0	5400.0	5560.0	7200.0	5670.0	5960.0	5820.0	0.0
MO	42	61	103	3280.0	3350.0	4000.0	3880.0	4900.0	3970.0	4140.0	4100.0	0.0
TC	43	60	103	1760.0	1770.0	2200.0	2380.0	3100.0	2660.0	2900.0	2690.0	2351.0
RU	44	59	103	550.0	400.0	1000.0	780.0	1600.0	700.0	720.0	670.0	763.0
RH	45	58	103	-620.0	-990.0	-800.0	-690.0	-500.0	-500.0	-550.0	-500.0	-546.0
Y	39	65	104	11000.0	10970.0	11500.0	11960.0	0.0	11950.0	11900.0	12280.0	0.0
ZR	40	64	104	5020.0	5350.0	5400.0	5500.0	0.0	5150.0	5500.0	5380.0	0.0
NB	41	63	104	7680.0	7580.0	8200.0	8440.0	0.0	8800.0	9030.0	8970.0	0.0
MO	42	62	104	1710.0	1990.0	2100.0	2140.0	0.0	2060.0	2250.0	2180.0	0.0
TC	43	61	104	4470.0	4290.0	5300.0	5290.0	0.0	5730.0	6020.0	5850.0	0.0
RU	44	60	104	-1240.0	-1110.0	-1000.0	-940.0	0.0	-1100.0	-1030.0	-1090.0	-1147.0

ELEMENT	Z	N	A	MYERS[2]	GR/HI/TA[3]	SEE/HO[4]	LIR/ZEL[5]	BE/LOM[6]	JA/GA/KE[7]	CO/KE[8]	JA/EY[9]	WAP/BOS[1]
RH	45	59	104	2050.0	1530.0	2500.0	2310.0	0.0	2310.0	2440.0	2260.0	2447.0
Y	39	66	105	9430.0	9590.0	10100.0	10130.0	10900.0	10250.0	10100.0	10610.0	0.0
ZR	40	65	105	7760.0	7910.0	8200.0	8420.0	9200.0	8120.0	8330.0	8290.0	0.0
NB	41	64	105	6120.0	6230.0	6500.0	6620.0	7400.0	6940.0	7260.0	7120.0	0.0
MO	42	63	105	4470.0	4550.0	5100.0	5070.0	5300.0	5190.0	5340.0	5340.0	0.0
TC	43	62	105	2890.0	2930.0	3200.0	3470.0	4800.0	3820.0	4160.0	3940.0	3396.0
RU	44	61	105	1410.0	1380.0	2200.0	2050.0	3100.0	1980.0	2090.0	2070.0	1918.0
RH	45	60	105	300.0	30.0	200.0	510.0	900.0	510.0	670.0	510.0	567.0
Y	39	67	106	12140.0	12120.0	12600.0	13000.0	0.0	13200.0	13060.0	13580.0	0.0
ZR	40	66	106	6200.0	6550.0	6800.0	6610.0	0.0	6420.0	6520.0	6610.0	0.0
NB	41	65	106	8850.0	8770.0	9200.0	9530.0	0.0	9910.0	10080.0	10050.0	0.0
MO	42	64	106	2930.0	3220.0	3500.0	3230.0	0.0	3340.0	3580.0	3500.0	0.0
TC	43	63	106	5620.0	5470.0	6100.0	6410.0	0.0	6950.0	7230.0	7090.0	0.0
RU	44	62	106	-180.0	0.0	100.0	190.0	0.0	60.0	230.0	160.0	40.0
RH	45	61	106	2820.0	2440.0	3400.0	3550.0	0.0	3590.0	3730.0	3670.0	3540.0
PO	46	60	106	-2560.0	-2730.0	-2900.0	-2880.0	0.0	-3200.0	-2970.0	-3220.0	-2982.0
ZR	40	67	107	8920.0	9080.0	9300.0	9500.0	10100.0	9380.0	9470.0	9590.0	0.0
NB	41	66	107	7300.0	7420.0	7600.0	7660.0	8400.0	8210.0	8270.0	8370.0	0.0
MO	42	65	107	5660.0	5770.0	6500.0	6180.0	7300.0	6300.0	6410.0	6420.0	0.0
TC	43	64	107	4070.0	4130.0	4300.0	4490.0	4800.0	5100.0	5450.0	5250.0	0.0
RU	44	63	107	2530.0	2530.0	3200.0	3210.0	3800.0	3190.0	3370.0	3320.0	3150.0
RH	45	62	107	1140.0	1020.0	1100.0	1620.0	2800.0	1680.0	1800.0	1750.0	1511.0
PO	46	61	107	90.0	-270.0	400.0	240.0	1100.0	-120.0	90.0	-50.0	33.0
ZR	40	68	108	7380.0	7730.0	7900.0	7710.0	0.0	7440.0	7490.0	7610.0	0.0
NB	41	67	108	9980.0	9930.0	10300.0	10530.0	0.0	11160.0	11220.0	11340.0	0.0
MO	42	66	108	4140.0	4440.0	4700.0	4290.0	0.0	4600.0	4600.0	4750.0	0.0
TC	43	65	108	6770.0	6650.0	7200.0	7450.0	0.0	8070.0	8260.0	8180.0	0.0
RU	44	64	108	990.0	1200.0	1300.0	1240.0	0.0	1340.0	1620.0	1480.0	1200.0
RH	45	63	108	3790.0	3520.0	4300.0	4670.0	0.0	4810.0	4930.0	4910.0	4504.0
PO	46	62	108	-1580.0	-1710.0	-1700.0	-1680.0	0.0	-2040.0	-1860.0	-1960.0	-1923.0
ZR	40	69	109	10050.0	10220.0	10200.0	10510.0	11000.0	10470.0	10690.0	10500.0	0.0
NB	41	68	109	8440.0	8580.0	9000.0	8700.0	9600.0	9230.0	9230.0	9370.0	0.0
MO	42	67	109	6830.0	6950.0	7600.0	7190.0	8200.0	7550.0	7550.0	7730.0	0.0
TC	43	66	109	5250.0	5320.0	5200.0	5480.0	6200.0	6370.0	6440.0	6500.0	0.0
RU	44	65	109	3690.0	3730.0	4300.0	4250.0	5300.0	4310.0	4410.0	4400.0	0.0
RH	45	64	109	2220.0	2170.0	2100.0	2660.0	3200.0	2950.0	3200.0	3080.0	0.0
PO	46	63	109	1030.0	730.0	1600.0	1430.0	1600.0	1100.0	1220.0	1200.0	1116.0
AG	47	62	109	-220.0	-560.0	-600.0	-290.0	900.0	-370.0	-200.0	-370.0	-192.0
NB	41	69	110	11100.0	11060.0	11500.0	11480.0	0.0	12260.0	12420.0	12260.0	0.0
MO	42	68	110	5300.0	5620.0	6000.0	5330.0	0.0	5620.0	5570.0	5760.0	0.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/HO [4]	LIR/ZEL [5]	BE/LUM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BOS [1]
TC	43	67	110	7930.0	7820.0	8200.0	8390.0	0.0	9320.0	9390.0	9480.0	0.0
RU	44	66	110	2170.0	2410.0	2300.0	2250.0	0.0	2610.0	2570.0	2740.0	0.0
RH	45	65	110	4900.0	4670.0	5100.0	5670.0	0.0	5920.0	6140.0	6000.0	5405.0
PD	46	64	110	-530.0	-620.0	-600.0	-560.0	0.0	-760.0	-620.0	-640.0	-879.0
AG	47	63	110	2440.0	1890.0	2600.0	2830.0	0.0	2770.0	2890.0	2790.0	2893.0
NB	41	70	111	9550.0	9710.0	10400.0	9730.0	10600.0	10300.0	10430.0	10400.0	0.0
MO	42	69	111	7960.0	8090.0	8600.0	8150.0	9100.0	8660.0	8780.0	8650.0	0.0
TC	43	68	111	6400.0	6500.0	6600.0	6450.0	7300.0	7380.0	7390.0	7510.0	0.0
RU	44	67	111	4850.0	4910.0	5400.0	5210.0	6500.0	5560.0	5520.0	5720.0	0.0
RH	45	66	111	3330.0	3350.0	3000.0	3620.0	4300.0	4220.0	4290.0	4330.0	0.0
PD	46	65	111	2050.0	1860.0	2500.0	2500.0	3400.0	2220.0	2300.0	2290.0	2196.0
AG	47	64	111	870.0	530.0	300.0	860.0	1300.0	910.0	1070.0	960.0	1028.0
CD	48	63	111	-560.0	-910.0	100.0	-740.0	700.0	-880.0	-820.0	-900.0	-849.0
MO	42	70	112	6440.0	6760.0	7400.0	6370.0	0.0	6690.0	6790.0	6790.0	0.0
TC	43	69	112	9040.0	8960.0	9200.0	9270.0	0.0	10420.0	10580.0	10410.0	0.0
RU	44	68	112	3350.0	3600.0	3800.0	3230.0	0.0	3630.0	3520.0	3740.0	0.0
RH	45	67	112	6040.0	5830.0	5800.0	6580.0	0.0	7170.0	7240.0	7320.0	0.0
PD	46	66	112	520.0	560.0	600.0	470.0	0.0	510.0	460.0	620.0	294.0
AG	47	65	112	3500.0	2970.0	3300.0	3920.0	0.0	3880.0	3940.0	3890.0	3958.0
CD	48	64	112	-2110.0	-2230.0	-2100.0	-2560.0	0.0	-2730.0	-2580.0	-2730.0	-2579.0
MO	42	71	113	9060.0	9190.0	10000.0	9060.0	10300.0	9700.0	9660.0	9730.0	0.0
TC	43	70	113	7510.0	7630.0	7900.0	7410.0	8500.0	8460.0	8600.0	8560.0	0.0
RU	44	69	113	6010.0	6070.0	6400.0	6090.0	7200.0	6660.0	6710.0	6640.0	0.0
RH	45	68	113	4520.0	4520.0	4200.0	4550.0	8700.0	5240.0	5230.0	5350.0	0.0
PD	46	67	113	3210.0	3030.0	3400.0	3460.0	1300.0	3470.0	3420.0	3610.0	0.0
AG	47	66	113	1930.0	1660.0	1300.0	1930.0	2300.0	2170.0	2080.0	2220.0	2010.0
CD	48	65	113	520.0	220.0	1100.0	510.0	2100.0	240.0	300.0	200.0	322.0
MO	42	72	114	7490.0	7840.0	8800.0	7430.0	0.0	7730.0	7740.0	7730.0	0.0
TC	43	71	114	10140.0	10050.0	10500.0	10100.0	0.0	11480.0	11460.0	11500.0	0.0
RU	44	70	114	4490.0	4750.0	5200.0	4200.0	0.0	4700.0	4720.0	4790.0	0.0
RH	45	69	114	7190.0	6980.0	6700.0	7410.0	0.0	8270.0	8410.0	8250.0	0.0
PD	46	68	114	1720.0	1740.0	1500.0	1450.0	0.0	1530.0	1440.0	1640.0	0.0
AG	47	67	114	4530.0	4110.0	4400.0	4900.0	0.0	5130.0	4930.0	5210.0	4859.0
CD	48	66	114	-1050.0	-1100.0	-1000.0	-1330.0	0.0	-1460.0	-1490.0	-1470.0	-1444.0
IN	49	65	114	1690.0	1160.0	1900.0	1820.0	0.0	1960.0	1920.0	2010.0	1985.0
TC	43	72	115	8560.0	8700.0	9300.0	8390.0	9500.0	9500.0	9530.0	9500.0	0.0
RU	44	71	115	7130.0	7180.0	7700.0	6940.0	13400.0	7710.0	7590.0	7740.0	0.0
RH	45	70	115	5690.0	5670.0	5600.0	5440.0	1600.0	6320.0	6410.0	6400.0	0.0
PD	46	69	115	4360.0	4210.0	4200.0	4350.0	5400.0	4560.0	4630.0	4540.0	0.0
AG	47	68	115	2970.0	2790.0	2300.0	2920.0	3300.0	3200.0	2890.0	3240.0	3162.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/HO [4]	LIR/ZEL [5]	BE/LOM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BOS [1]
CO	48	67	115	1550.0	1320.0	2100.0	1660.0	3400.0	1490.0	1410.0	1530.0	1449.0
IN	49	66	115	150.0	-150.0	-400.0	90.0	-300.0	260.0	290.0	340.0	494.0
TC	43	73	116	11100.0	11040.0	11800.0	10920.0	0.0	12460.0	12490.0	12620.0	0.0
RU	44	72	116	5600.0	5860.0	6400.0	5170.0	0.0	5740.0	5660.0	5740.0	0.0
RH	45	71	116	8310.0	8090.0	8100.0	8190.0	0.0	9320.0	9280.0	9350.0	0.0
PD	46	70	116	2800.0	2930.0	2900.0	2390.0	0.0	2610.0	2670.0	2700.0	0.0
AG	47	69	116	5530.0	5180.0	5200.0	5810.0	0.0	6230.0	5990.0	6140.0	0.0
CD	48	68	116	10.0	0.0	-100.0	-190.0	0.0	-450.0	-560.0	-450.0	-464.0
IN	49	67	116	2720.0	2260.0	2900.0	3090.0	0.0	3220.0	3190.0	3340.0	3272.0
TC	43	74	117	9500.0	9660.0	10300.0	9390.0	10700.0	10470.0	10560.0	10650.0	0.0
RU	44	73	117	8110.0	8210.0	8800.0	7770.0	9100.0	8710.0	8620.0	8860.0	0.0
RH	45	72	117	6730.0	6760.0	6800.0	6340.0	7600.0	7350.0	7350.0	7350.0	0.0
PD	46	71	117	5350.0	5300.0	5600.0	5180.0	6300.0	5620.0	5560.0	5650.0	0.0
AG	47	70	117	3960.0	3860.0	3500.0	3860.0	4700.0	4270.0	4060.0	4300.0	4176.0
CD	48	69	117	2560.0	2390.0	3000.0	2710.0	4100.0	2590.0	2550.0	2460.0	2529.0
IN	49	68	117	1190.0	940.0	700.0	1360.0	1000.0	1280.0	1230.0	1370.0	1455.0
SN	50	67	117	-2080.0	-2060.0	-1400.0	-1910.0	-2100.0	-1660.0	-1750.0	-1620.0	-1745.0
RU	44	74	118	6530.0	6860.0	7400.0	6160.0	0.0	6710.0	6670.0	6890.0	0.0
RH	45	73	118	9240.0	9100.0	9400.0	8960.0	0.0	10320.0	10320.0	10480.0	0.0
PD	46	72	118	3780.0	3980.0	4100.0	3310.0	0.0	3650.0	3630.0	3650.0	0.0
AG	47	71	118	6490.0	6210.0	6300.0	6650.0	0.0	7280.0	6960.0	7250.0	0.0
CD	48	70	118	1030.0	1080.0	1000.0	890.0	0.0	630.0	580.0	620.0	744.0
IN	49	69	118	3720.0	3320.0	3900.0	4260.0	0.0	4310.0	4300.0	4270.0	4204.0
SN	50	68	118	-3590.0	-3350.0	-3400.0	-3710.0	0.0	-3590.0	-3710.0	-3580.0	-3695.0
RU	44	75	119	9030.0	9190.0	9700.0	8620.0	10000.0	9530.0	9540.0	9830.0	0.0
RH	45	74	119	7660.0	7750.0	8000.0	7260.0	8800.0	8330.0	8380.0	8510.0	0.0
PD	46	73	119	6300.0	6320.0	6800.0	5980.0	7200.0	6610.0	6610.0	6780.0	0.0
AG	47	72	119	4920.0	4880.0	4600.0	4760.0	5800.0	5320.0	5000.0	5250.0	0.0
CD	48	71	119	3560.0	3450.0	4100.0	3690.0	4900.0	3640.0	3540.0	3570.0	3498.0
IN	49	70	119	2180.0	2000.0	1700.0	2530.0	2300.0	2350.0	2290.0	2440.0	2339.0
SN	50	69	119	-1020.0	-960.0	-200.0	-800.0	-1100.0	-550.0	-570.0	-680.0	-583.0
RU	44	76	120	7450.0	7840.0	8300.0	7210.0	0.0	7630.0	7750.0	7920.0	0.0
RH	45	75	120	10140.0	10070.0	10500.0	9750.0	0.0	11150.0	11250.0	11450.0	0.0
PD	46	74	120	4740.0	4990.0	5200.0	4240.0	0.0	4620.0	4680.0	4810.0	0.0
AG	47	73	120	7420.0	7210.0	7500.0	7440.0	0.0	8270.0	8030.0	8360.0	0.0
CD	48	72	120	2020.0	2140.0	2300.0	1900.0	0.0	1670.0	1480.0	1580.0	0.0
IN	49	71	120	4690.0	4350.0	4700.0	5340.0	0.0	5370.0	5240.0	5390.0	5320.0
SN	50	70	120	-2540.0	-2250.0	-2300.0	-2600.0	0.0	-2520.0	-2610.0	-2520.0	-2680.0
SB	51	69	120	150.0	-30.0	400.0	760.0	0.0	700.0	920.0	550.0	984.0
RH	45	76	121	8570.0	8720.0	9000.0	8230.0	9600.0	9250.0	9470.0	9540.0	0.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/MO [4]	LIR/ZEL [5]	BE/LOM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BUS [10]
PO	46	75	121	7210.0	7310.0	7700.0	6790.0	8200.0	7440.0	7540.0	7740.0	0.0
AG	47	74	121	5870.0	5890.0	5900.0	5650.0	6900.0	6290.0	6120.0	6430.0	0.0
CD	48	73	121	4510.0	4460.0	5200.0	4590.0	5900.0	4630.0	4400.0	4700.0	0.0
IN	49	72	121	3170.0	3050.0	2800.0	3620.0	3400.0	3390.0	3260.0	3400.0	3366.0
SN	50	71	121	-10.0	100.0	800.0	230.0	-300.0	500.0	320.0	440.0	386.0
SB	51	70	121	-1330.0	-1310.0	-1700.0	-1100.0	-2600.0	-1270.0	-1070.0	-1280.0	0.0
RH	45	77	122	11010.0	11010.0	11500.0	10590.0	0.0	11920.0	12160.0	12500.0	0.0
PO	46	76	122	5670.0	5980.0	6300.0	5190.0	0.0	5530.0	5760.0	5840.0	0.0
AG	47	75	122	8330.0	8190.0	8500.0	8220.0	0.0	9110.0	9010.0	9360.0	0.0
CD	48	74	122	2970.0	3160.0	3300.0	2860.0	0.0	2650.0	2470.0	2740.0	0.0
IN	49	73	122	5650.0	5360.0	5900.0	6330.0	0.0	6350.0	6180.0	6530.0	6504.0
SN	50	72	122	-1520.0	-1180.0	-1300.0	-1570.0	0.0	-1470.0	-1650.0	-1550.0	-1622.0
SB	51	71	122	1140.0	1030.0	1500.0	1780.0	0.0	1740.0	1850.0	1680.0	1980.0
TE	52	70	122	-4220.0	-3960.0	-4200.0	-4250.0	0.0	-4140.0	-4140.0	-4340.0	-4142.0
PO	46	77	123	8110.0	8270.0	8600.0	7640.0	10100.0	8210.0	8470.0	8810.0	0.0
AG	47	76	123	6740.0	6860.0	7200.0	6540.0	7900.0	7210.0	7240.0	7450.0	0.0
CD	48	75	123	5440.0	5470.0	5900.0	5450.0	6800.0	5460.0	5340.0	5690.0	0.0
IN	49	74	123	4110.0	4060.0	4000.0	4610.0	4300.0	4370.0	4240.0	4560.0	4379.0
SN	50	73	123	980.0	1140.0	1800.0	1200.0	700.0	1490.0	1290.0	1590.0	1397.0
SB	51	72	123	-370.0	-260.0	-500.0	-80.0	-1400.0	-230.0	-130.0	-320.0	-52.0
TE	52	71	123	-1700.0	-1660.0	-1200.0	-1380.0	-2300.0	-1130.0	-1240.0	-1370.0	-1204.0
PO	46	78	124	6560.0	6940.0	7300.0	6200.0	0.0	6420.0	6730.0	7000.0	0.0
AG	47	77	124	9220.0	9150.0	9700.0	9000.0	0.0	9880.0	9960.0	10420.0	0.0
CD	48	76	124	3910.0	4160.0	4400.0	3780.0	0.0	3560.0	3610.0	3780.0	0.0
IN	49	75	124	6570.0	6350.0	6900.0	7220.0	0.0	7190.0	7060.0	7510.0	0.0
SN	50	74	124	-540.0	-150.0	-400.0	-580.0	0.0	-500.0	-650.0	-380.0	-628.0
SB	51	73	124	2130.0	2060.0	2400.0	2730.0	0.0	2740.0	2800.0	2820.0	2905.0
TE	52	72	124	-3200.0	-2930.0	-3100.0	-3200.0	0.0	-3110.0	-3170.0	-3360.0	-3157.0
PO	46	79	125	8990.0	9200.0	9600.0	8570.0	9900.0	8970.0	9250.0	9620.0	0.0
AG	47	78	125	7660.0	7830.0	8200.0	7450.0	8900.0	8090.0	8240.0	8620.0	0.0
CD	48	77	125	6350.0	6430.0	7100.0	6270.0	7600.0	6240.0	6330.0	6740.0	0.0
IN	49	76	125	5040.0	5050.0	5100.0	5500.0	5500.0	5280.0	5240.0	5610.0	5419.0
SN	50	75	125	1930.0	2160.0	2500.0	2140.0	1600.0	2320.0	2240.0	2570.0	2356.0
SB	51	74	125	610.0	770.0	500.0	880.0	-400.0	750.0	790.0	860.0	766.0
TE	52	73	125	-710.0	-610.0	-400.0	-400.0	-1400.0	-150.0	-200.0	-230.0	-177.0
I	53	72	125	-2030.0	-1940.0	-1800.0	-1700.0	-2700.0	-1810.0	-1770.0	-2270.0	-1734.0
AG	47	79	126	10070.0	10080.0	10800.0	9850.0	0.0	10650.0	10750.0	11240.0	0.0
CD	48	78	126	4830.0	5130.0	5200.0	4660.0	0.0	4440.0	4610.0	4950.0	0.0
IN	49	77	126	7460.0	7310.0	8000.0	8010.0	0.0	7960.0	7960.0	8570.0	0.0
SN	50	76	126	420.0	870.0	600.0	390.0	0.0	420.0	440.0	660.0	478.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/HO [4]	LIR/ZEL [5]	BE/LOM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BOS
SB	51	75	126	3080.0	3070.0	3300.0	3620.0	0.0	3560.0	3630.0	3810.0	3566.0
TE	52	74	126	-2210.0	-1880.0	-2100.0	-2220.0	0.0	-2130.0	-2170.0	-2180.0	-2156.0
I	53	73	126	440.0	300.0	700.0	1070.0	0.0	1150.0	1230.0	860.0	1251.0
AG	47	80	127	8540.0	8750.0	9100.0	8400.0	9900.0	8920.0	9000.0	9710.0	0.0
CD	48	79	127	7230.0	7380.0	7800.0	7090.0	8500.0	7000.0	7130.0	7570.0	0.0
IN	49	78	127	5930.0	6010.0	6200.0	6290.0	6400.0	6170.0	6220.0	6770.0	0.0
SN	50	77	127	2870.0	3150.0	3300.0	3050.0	2400.0	3090.0	3150.0	3630.0	0.0
SB	51	76	127	1570.0	1780.0	1500.0	1800.0	800.0	1660.0	1830.0	1910.0	1581.0
TE	52	75	127	250.0	410.0	700.0	520.0	-500.0	690.0	670.0	770.0	694.0
I	53	74	127	-1050.0	-960.0	-900.0	-760.0	-1500.0	-840.0	-700.0	-1100.0	-664.0
AG	47	81	128	10930.0	10980.0	11900.0	10750.0	0.0	11320.0	11360.0	12190.0	0.0
CD	48	80	128	5700.0	6080.0	6100.0	5540.0	0.0	5280.0	5360.0	6040.0	0.0
IN	49	79	128	8330.0	8250.0	8900.0	8710.0	0.0	8720.0	8730.0	9410.0	0.0
SN	50	78	128	1360.0	1860.0	1500.0	1330.0	0.0	1300.0	1410.0	1830.0	1350.0
SB	51	77	128	4000.0	4040.0	4200.0	4500.0	0.0	4340.0	4550.0	4880.0	4251.0
TE	52	76	128	-1240.0	-850.0	-1100.0	-1280.0	0.0	-1210.0	-1160.0	-1140.0	-1258.0
I	53	75	128	1400.0	1320.0	1600.0	1940.0	0.0	1970.0	2130.0	1860.0	2128.0
XE	54	74	128	-3840.0	-3580.0	-3800.0	-3820.0	0.0	-3860.0	-3960.0	-4350.0	-3907.0
CD	48	81	129	8090.0	8310.0	8700.0	7910.0	9400.0	7680.0	7730.0	8530.0	0.0
IN	49	80	129	6800.0	6950.0	7200.0	6980.0	7500.0	7000.0	6960.0	7880.0	0.0
SN	50	79	129	3790.0	4120.0	4200.0	3960.0	3300.0	3860.0	3930.0	4460.0	3998.0
SB	51	78	129	2490.0	2760.0	2500.0	2710.0	1800.0	2550.0	2830.0	3080.0	2376.0
TE	52	77	129	1190.0	1410.0	1300.0	1400.0	500.0	1460.0	1610.0	1840.0	1504.0
I	53	76	129	-90.0	50.0	100.0	140.0	400.0	70.0	280.0	-50.0	190.0
XE	54	75	129	-1390.0	-1300.0	-1300.0	-1120.0	-3100.0	-1040.0	-1140.0	-1390.0	-1136.0
CD	48	82	130	6560.0	7000.0	7000.0	6420.0	0.0	6390.0	6440.0	7210.0	0.0
IN	49	81	130	9190.0	9170.0	9700.0	9320.0	0.0	9410.0	9310.0	10360.0	0.0
SN	50	80	130	2270.0	2840.0	2500.0	2270.0	0.0	2130.0	2170.0	2940.0	2000.0
SB	51	79	130	4900.0	4990.0	5200.0	5370.0	0.0	5100.0	5340.0	5710.0	4967.0
TE	52	78	130	-300.0	150.0	-400.0	-360.0	0.0	-320.0	-110.0	40.0	-450.0
I	53	77	130	2320.0	2300.0	2500.0	2780.0	0.0	2740.0	3010.0	2930.0	2984.0
XE	54	76	130	-2850.0	-2550.0	-2900.0	-2890.0	0.0	-2940.0	-2970.0	-3290.0	-3019.0
CD	48	83	131	10990.0	10870.0	11300.0	0.0	12700.0	10450.0	10330.0	11330.0	0.0
IN	49	82	131	7660.0	7870.0	8100.0	7580.0	8500.0	8110.0	8010.0	9040.0	0.0
SN	50	81	131	4660.0	5060.0	4900.0	4880.0	4000.0	4540.0	4530.0	5430.0	0.0
SB	51	80	131	3390.0	3720.0	3600.0	3630.0	2800.0	3380.0	3580.0	4190.0	0.0
TE	52	79	131	2120.0	2380.0	2200.0	2290.0	1400.0	2230.0	2390.0	2670.0	2256.0
I	53	78	131	830.0	1040.0	900.0	1000.0	600.0	960.0	1310.0	1140.0	971.0
XE	54	77	131	-430.0	-300.0	-400.0	-250.0	-1200.0	-270.0	-290.0	-320.0	-355.0
CS	55	76	131	-1710.0	-1640.0	-1500.0	-1500.0	-2000.0	-1490.0	-1350.0	-1910.0	-1341.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/HO [4]	LIR/ZEL [5]	BE/LOM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BUS [10]
IN	49	83	132	12070.0	11720.0	12500.0	0.0	0.0	12170.0	11900.0	13170.0	0.0
SN	50	82	132	3160.0	3780.0	3400.0	3370.0	0.0	3250.0	3240.0	4110.0	3220.0
SB	51	81	132	5770.0	5930.0	6000.0	6280.0	0.0	5780.0	5940.0	6680.0	5601.0
TE	52	80	132	620.0	1120.0	500.0	560.0	0.0	500.0	620.0	1150.0	505.0
I	53	79	132	3240.0	3270.0	3500.0	3620.0	0.0	3510.0	3870.0	3770.0	3580.0
XE	54	78	132	-1910.0	-1550.0	-2100.0	-2000.0	0.0	-2050.0	-2030.0	-2110.0	-2103.0
CS	55	77	132	690.0	600.0	1000.0	1110.0	0.0	1190.0	1310.0	1060.0	1269.0
IN	49	84	133	10540.0	10390.0	11300.0	0.0	12700.0	10840.0	10700.0	11830.0	0.0
SN	50	83	133	7560.0	7630.0	7600.0	8630.0	7100.0	7310.0	7130.0	8240.0	0.0
SB	51	82	133	4270.0	4660.0	4500.0	4500.0	4500.0	4490.0	4640.0	5360.0	3950.0
TE	52	81	133	3010.0	3330.0	3100.0	3210.0	2000.0	2910.0	2980.0	3640.0	2971.0
I	53	80	133	1750.0	2010.0	1700.0	1880.0	1600.0	1790.0	2100.0	2250.0	1760.0
XE	54	79	133	480.0	680.0	400.0	600.0	-300.0	490.0	530.0	530.0	427.0
CS	55	78	133	-770.0	-650.0	-600.0	-650.0	-1000.0	-590.0	-470.0	-730.0	-520.0
BA	56	77	133	-2040.0	-1970.0	-2200.0	-1880.0	-2800.0	-1900.0	-1950.0	-2380.0	0.0
SN	50	84	134	6060.0	6330.0	6500.0	8120.0	0.0	5980.0	5920.0	6900.0	0.0
SB	51	83	134	8660.0	8490.0	8800.0	9200.0	0.0	8550.0	8540.0	9500.0	0.0
TE	52	82	134	1520.0	2070.0	1400.0	1520.0	0.0	1610.0	1670.0	2320.0	0.0
I	53	81	134	4110.0	4210.0	4400.0	4470.0	0.0	4200.0	4430.0	4750.0	4155.0
XE	54	80	134	-980.0	-560.0	-1300.0	-1110.0	0.0	-1230.0	-1230.0	-1000.0	-1217.0
CS	55	79	134	1610.0	1570.0	1700.0	1920.0	0.0	1950.0	2080.0	1910.0	2058.0
BA	56	78	134	-3500.0	-3210.0	-3600.0	-3610.0	0.0	-3680.0	-3690.0	-4170.0	-3700.0
SN	50	85	135	8380.0	8480.0	8500.0	10130.0	8500.0	8090.0	8070.0	8960.0	0.0
SB	51	84	135	7160.0	7190.0	7600.0	8620.0	8400.0	7220.0	7330.0	8160.0	0.0
TE	52	83	135	5890.0	5910.0	5700.0	6230.0	5800.0	5680.0	5570.0	6450.0	6195.0
I	53	82	135	2640.0	2950.0	2700.0	2720.0	2500.0	2900.0	3130.0	3430.0	2710.0
XE	54	81	135	1380.0	1640.0	1400.0	1470.0	500.0	1180.0	1160.0	1510.0	1159.0
CS	55	80	135	140.0	320.0	0.0	190.0	100.0	230.0	280.0	380.0	205.0
BA	56	79	135	-1110.0	-990.0	-1400.0	-1040.0	-1800.0	-1130.0	-1170.0	-1530.0	-1200.0
SN	50	86	136	6880.0	7180.0	7400.0	8910.0	0.0	6910.0	6870.0	7740.0	0.0
SB	51	85	136	9460.0	9330.0	9800.0	10710.0	0.0	9330.0	9470.0	10220.0	0.0
TE	52	84	136	4410.0	4630.0	4500.0	5460.0	0.0	4350.0	4360.0	5130.0	0.0
I	53	83	136	6990.0	6770.0	7000.0	7150.0	0.0	6950.0	7000.0	7560.0	7005.0
XE	54	82	136	-80.0	390.0	-400.0	-180.0	0.0	-110.0	-140.0	190.0	-69.0
CS	55	81	136	2500.0	2520.0	2700.0	2740.0	0.0	2640.0	2630.0	2860.0	2548.0
BA	56	80	136	-2570.0	-2220.0	-3000.0	-2740.0	0.0	-2850.0	-2900.0	-3050.0	-2874.0
LA	57	79	136	20.0	-100.0	100.0	260.0	0.0	360.0	390.0	-40.0	460.0
SB	51	86	137	7960.0	8040.0	8600.0	9640.0	9100.0	8150.0	8280.0	9010.0	0.0
TE	52	85	137	6730.0	6760.0	6700.0	7370.0	6800.0	6460.0	6500.0	7180.0	0.0
I	53	84	137	5500.0	5490.0	5800.0	6340.0	7000.0	5630.0	5780.0	6230.0	5495.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/HO [4]	LIR/ZEL [5]	BE/LOM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BOS [10]
XE	54	83	137	4260.0	4210.0	3800.0	4260.0	4100.0	3940.0	3780.0	4330.0	4343.0
CS	55	82	137	1040.0	1270.0	1200.0	1080.0	1000.0	1350.0	1290.0	1570.0	1173.0
BA	56	81	137	-200.0	-20.0	-600.0	-210.0	-1100.0	-440.0	-510.0	-550.0	0.0
LA	57	80	137	-1450.0	-1330.0	-1500.0	-1450.0	-1700.0	-1360.0	-1370.0	-1560.0	0.0
SB	51	87	138	10250.0	10150.0	10800.0	12300.0	0.0	10370.0	10510.0	11160.0	0.0
TE	52	86	138	5240.0	5480.0	5600.0	6210.0	0.0	5280.0	5300.0	5970.0	0.0
I	53	85	138	7810.0	7620.0	7900.0	8320.0	0.0	7740.0	7920.0	8300.0	0.0
XE	54	84	138	2790.0	2940.0	2800.0	3260.0	0.0	2620.0	2510.0	2990.0	2740.0
CS	55	83	138	5360.0	5080.0	5200.0	5370.0	0.0	5400.0	5230.0	5710.0	5161.0
BA	56	82	138	-1660.0	-1260.0	-2100.0	-1800.0	0.0	-1740.0	-1810.0	-1860.0	-1749.0
LA	57	81	138	910.0	860.0	1100.0	1040.0	0.0	1050.0	1020.0	940.0	1040.0
SB	51	88	139	8750.0	8860.0	9400.0	10670.0	10100.0	8980.0	9040.0	9840.0	0.0
TE	52	87	139	7530.0	7600.0	7800.0	8750.0	7500.0	7490.0	7530.0	8130.0	0.0
I	53	86	139	6330.0	6330.0	6700.0	7260.0	8900.0	6570.0	6730.0	7080.0	0.0
XE	54	85	139	5100.0	5080.0	5000.0	5100.0	4100.0	4720.0	4580.0	5060.0	4880.0
CS	55	84	139	3890.0	3810.0	4100.0	4400.0	5600.0	4080.0	4020.0	4380.0	4293.0
BA	56	83	139	2660.0	2540.0	2200.0	2480.0	2700.0	2310.0	2160.0	2280.0	2305.0
LA	57	82	139	-550.0	-370.0	-600.0	-560.0	-700.0	-240.0	-280.0	-370.0	-260.0
CE	58	81	139	-1750.0	-1660.0	-2300.0	-1850.0	-2800.0	-2080.0	-2100.0	-2060.0	-2110.0
TE	52	88	140	6050.0	6320.0	6400.0	7080.0	0.0	6100.0	6050.0	6820.0	0.0
I	53	87	140	8610.0	8450.0	8900.0	9850.0	0.0	8780.0	8950.0	9240.0	0.0
XE	54	86	140	3630.0	3810.0	3800.0	3950.0	0.0	3550.0	3420.0	3850.0	4060.0
CS	55	85	140	6190.0	5930.0	6200.0	6310.0	0.0	6180.0	6100.0	6440.0	6053.0
BA	56	84	140	1200.0	1290.0	1100.0	1370.0	0.0	1000.0	930.0	950.0	1035.0
LA	57	83	140	3760.0	3410.0	3600.0	3660.0	0.0	3810.0	3670.0	3770.0	3760.0
CE	58	82	140	-3200.0	-2870.0	-3700.0	-3450.0	0.0	-3380.0	-3390.0	-3370.0	-3388.0
TE	52	89	141	8320.0	8420.0	8800.0	9900.0	8500.0	8400.0	8510.0	9080.0	0.0
I	53	88	141	7130.0	7170.0	7600.0	8280.0	8800.0	7380.0	7470.0	7930.0	0.0
XE	54	87	141	5910.0	5920.0	6000.0	6430.0	5800.0	5770.0	5680.0	6010.0	6000.0
CS	55	86	141	4720.0	4670.0	5000.0	5240.0	7500.0	5000.0	4960.0	5240.0	4980.0
BA	56	85	141	3510.0	3410.0	3100.0	3140.0	2500.0	3100.0	2990.0	3010.0	3025.0
LA	57	84	141	2300.0	2160.0	2600.0	2680.0	4200.0	2490.0	2420.0	2440.0	2430.0
CE	58	83	141	1090.0	910.0	300.0	720.0	1000.0	680.0	540.0	770.0	581.0
TE	52	90	142	6840.0	7150.0	7300.0	7940.0	0.0	6400.0	6470.0	7010.0	0.0
I	53	89	142	9390.0	9250.0	9900.0	11110.0	0.0	9690.0	9930.0	10200.0	0.0
XE	54	88	142	4440.0	4660.0	4700.0	4830.0	0.0	4370.0	4220.0	4700.0	4900.0
CS	55	87	142	7000.0	6770.0	7300.0	7720.0	0.0	7220.0	7160.0	7390.0	6870.0
BA	56	86	142	2050.0	2160.0	1900.0	2010.0	0.0	1920.0	1830.0	1810.0	2204.0
LA	57	85	142	4600.0	4280.0	4600.0	4510.0	0.0	4600.0	4520.0	4510.0	4517.0
CE	58	84	142	-360.0	-330.0	-600.0	-360.0	0.0	-650.0	-710.0	-550.0	-743.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/HO [4]	LIR/ZEL [5]	BE/LOM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BUS [1]
PR	59	83	142	2190.0	1770.0	1900.0	1940.0	0.0	2200.0	2140.0	2210.0	2160.0
I	53	90	143	7910.0	7980.0	8300.0	9250.0	9500.0	7690.0	7900.0	8120.0	0.0
XE	54	89	143	6720.0	6740.0	6800.0	7560.0	7000.0	6670.0	6680.0	6970.0	0.0
CS	55	88	143	5520.0	5510.0	5900.0	6180.0	7200.0	5830.0	5710.0	6090.0	0.0
BA	56	87	143	4330.0	4270.0	4200.0	4390.0	4300.0	4140.0	4020.0	3970.0	0.0
LA	57	86	143	3130.0	3030.0	3500.0	3450.0	5100.0	3410.0	3330.0	3300.0	3305.0
CE	58	85	143	1950.0	1780.0	1300.0	1370.0	1700.0	1460.0	1430.0	1520.0	1451.0
PR	59	84	143	750.0	540.0	1000.0	1040.0	3000.0	880.0	910.0	890.0	933.0
I	53	91	144	10150.0	10050.0	10200.0	12220.0	0.0	10170.0	10460.0	10560.0	0.0
XE	54	90	144	5250.0	5490.0	5600.0	5720.0	0.0	4670.0	4660.0	4900.0	0.0
CS	55	89	144	7780.0	7580.0	7700.0	8880.0	0.0	8130.0	8180.0	8360.0	0.0
BA	56	88	144	2870.0	3020.0	3000.0	2810.0	0.0	2740.0	2540.0	2660.0	0.0
LA	57	87	144	5420.0	5120.0	5400.0	5820.0	0.0	5640.0	5550.0	5470.0	0.0
CE	58	86	144	490.0	550.0	300.0	260.0	0.0	280.0	260.0	310.0	315.0
PR	59	85	144	3040.0	2650.0	2900.0	2850.0	0.0	2990.0	3010.0	2960.0	2997.0
ND	60	84	144	-1880.0	-1930.0	-2200.0	-2020.0	0.0	-2240.0	-2300.0	-2210.0	-2330.0
XE	54	91	145	7490.0	7550.0	7600.0	8600.0	0.0	7150.0	7230.0	7350.0	0.0
CS	55	90	145	6320.0	6330.0	6400.0	7090.0	0.0	6130.0	6170.0	6300.0	0.0
BA	56	89	145	5130.0	5100.0	4900.0	5430.0	0.0	5050.0	4970.0	4930.0	0.0
LA	57	88	145	3960.0	3880.0	4000.0	4270.0	5600.0	4230.0	4070.0	4160.0	0.0
CE	58	87	145	2770.0	2640.0	2500.0	2550.0	2700.0	2500.0	2490.0	2480.0	2485.0
PR	59	86	145	1590.0	1410.0	1700.0	1790.0	3900.0	1810.0	1820.0	1760.0	1805.0
ND	60	85	145	400.0	180.0	-200.0	-290.0	100.0	-120.0	-180.0	-140.0	-160.0
XE	54	92	146	6050.0	6300.0	6900.0	6710.0	0.0	5540.0	5710.0	5770.0	0.0
CS	55	91	146	8520.0	8400.0	8400.0	9910.0	0.0	8600.0	8770.0	8750.0	0.0
BA	56	90	146	3650.0	3860.0	3800.0	3670.0	0.0	3050.0	2930.0	2870.0	0.0
LA	57	89	146	6200.0	5930.0	5800.0	6830.0	0.0	6540.0	6500.0	6440.0	0.0
CE	58	88	146	1330.0	1410.0	1100.0	980.0	0.0	1110.0	1020.0	1170.0	1080.0
PR	59	87	146	3850.0	3500.0	3600.0	4060.0	0.0	4020.0	4080.0	3920.0	4084.0
ND	60	86	146	-1020.0	-1040.0	-1300.0	-1380.0	0.0	-1300.0	-1420.0	-1340.0	-1460.0
XE	54	93	147	8320.0	8410.0	8700.0	9630.0	0.0	7870.0	7880.0	8110.0	0.0
CS	55	92	147	7060.0	7190.0	7500.0	8080.0	0.0	7000.0	7270.0	7170.0	0.0
BA	56	91	147	5840.0	5940.0	5800.0	6440.0	0.0	5520.0	5480.0	5330.0	0.0
LA	57	90	147	4610.0	4670.0	4800.0	5070.0	0.0	4540.0	4500.0	4370.0	0.0
CE	58	89	147	3450.0	3420.0	3100.0	3480.0	0.0	3410.0	3410.0	3440.0	0.0
PR	59	88	147	2400.0	2200.0	2100.0	2500.0	4400.0	2630.0	2610.0	2620.0	2705.0
ND	60	87	147	1240.0	1050.0	700.0	830.0	1200.0	920.0	880.0	830.0	896.0
PM	61	86	147	80.0	-180.0	100.0	180.0	2100.0	150.0	230.0	180.0	225.0
CS	55	93	148	9330.0	9280.0	9500.0	10930.0	0.0	9320.0	9460.0	9510.0	0.0
BA	56	92	148	4400.0	4730.0	5000.0	4680.0	0.0	3920.0	3970.0	3750.0	0.0

ELEMENT	Z	N	A	MYERS[2]	GR/HI/TA[3]	SEE/HO[4]	LIR/ZEL[5]	BE/LOM[6]	JA/GA/KE[7]	CO/KE[8]	JA/EY[9]	WAP/BOS[10]
LA	57	91	148	6820.0	6740.0	6500.0	7750.0	0.0	7010.0	7080.0	6840.0	0.0
CE	58	90	148	1920.0	2190.0	1900.0	1760.0	0.0	1410.0	1390.0	1380.0	0.0
PR	59	89	148	4400.0	4210.0	4100.0	4920.0	0.0	4930.0	5010.0	4890.0	0.0
ND	60	88	148	-300.0	-310.0	-600.0	-730.0	0.0	-480.0	-560.0	-470.0	-535.0
PM	61	87	148	2320.0	1850.0	2000.0	2390.0	0.0	2380.0	2460.0	2350.0	2464.0
CS	55	94	149	7930.0	8080.0	8700.0	9140.0	0.0	8100.0	7980.0	8300.0	0.0
BA	56	93	149	6660.0	6820.0	6900.0	7480.0	0.0	6250.0	6130.0	6100.0	0.0
LA	57	92	149	5400.0	5540.0	5800.0	6020.0	0.0	5400.0	5570.0	5250.0	0.0
CE	58	91	149	4130.0	4260.0	3900.0	4400.0	0.0	3880.0	3960.0	3850.0	0.0
PR	59	90	149	2910.0	2980.0	2800.0	3190.0	0.0	2940.0	3020.0	2830.0	2996.0
ND	60	89	149	1710.0	1720.0	1300.0	1650.0	0.0	1820.0	1790.0	1800.0	1678.0
PM	61	88	149	650.0	490.0	700.0	820.0	2800.0	980.0	1050.0	1050.0	1072.0
CS	55	95	150	10170.0	10160.0	10600.0	11920.0	0.0	10290.0	10320.0	10550.0	0.0
BA	56	94	150	5260.0	5630.0	6200.0	5800.0	0.0	5020.0	4640.0	4880.0	0.0
LA	57	93	150	7660.0	7620.0	7800.0	8730.0	0.0	7740.0	7740.0	7600.0	0.0
CE	58	92	150	2730.0	3070.0	3100.0	2740.0	0.0	2270.0	2440.0	2270.0	0.0
PR	59	91	150	5130.0	5050.0	4700.0	5730.0	0.0	5400.0	5570.0	5300.0	0.0
ND	60	90	150	250.0	520.0	200.0	-40.0	0.0	-170.0	-210.0	-260.0	-134.0
PM	61	89	150	2700.0	2510.0	2300.0	3130.0	0.0	3280.0	3440.0	3320.0	3501.0
SM	62	88	150	-2020.0	-1980.0	-2100.0	-2370.0	0.0	-2040.0	-2240.0	-1970.0	-2294.0
BA	56	95	151	7510.0	7700.0	8100.0	8550.0	0.0	7200.0	6980.0	7140.0	0.0
LA	57	94	151	6260.0	6430.0	6900.0	7080.0	0.0	6510.0	6270.0	6390.0	0.0
CE	58	93	151	4990.0	5150.0	5200.0	5430.0	0.0	4610.0	4610.0	4610.0	0.0
PR	59	92	151	3730.0	3870.0	3900.0	4060.0	0.0	3790.0	4100.0	3730.0	0.0
ND	60	91	151	2490.0	2590.0	2000.0	2480.0	0.0	2300.0	2430.0	2200.0	2467.0
PM	61	90	151	1250.0	1310.0	1200.0	1400.0	0.0	1290.0	1330.0	1270.0	1166.0
SM	62	89	151	60.0	50.0	-400.0	-60.0	0.0	260.0	170.0	310.0	76.0
BA	56	96	152	6130.0	6510.0	7100.0	6960.0	0.0	5930.0	5510.0	5800.0	0.0
LA	57	95	152	8500.0	8500.0	8800.0	9710.0	0.0	8700.0	8600.0	8650.0	0.0
CE	58	94	152	3610.0	3970.0	4500.0	3900.0	0.0	3380.0	3120.0	3410.0	0.0
PR	59	93	152	5980.0	5940.0	5900.0	6630.0	0.0	6130.0	6300.0	6070.0	0.0
ND	60	92	152	1100.0	1420.0	1300.0	890.0	0.0	680.0	940.0	630.0	1142.0
PM	61	91	152	3470.0	3380.0	2900.0	3830.0	0.0	3760.0	3820.0	3730.0	3473.0
SM	62	90	152	-1370.0	-1140.0	-1400.0	-1750.0	0.0	-1730.0	-1840.0	-1750.0	-1877.0
BA	56	97	153	8350.0	8580.0	9000.0	9570.0	0.0	8090.0	7870.0	7950.0	0.0
LA	57	96	153	7120.0	7310.0	7900.0	8170.0	0.0	7420.0	7150.0	7310.0	0.0
CE	58	95	153	5850.0	6040.0	6200.0	6520.0	0.0	5560.0	5460.0	5670.0	0.0
PR	59	94	153	4610.0	4760.0	5200.0	5090.0	0.0	4900.0	4810.0	4860.0	0.0
ND	60	93	153	3340.0	3490.0	3300.0	3470.0	0.0	3020.0	3170.0	2980.0	0.0
PM	61	92	153	2100.0	2210.0	2200.0	2190.0	0.0	2150.0	2330.0	2160.0	1799.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/HO [4]	LIR/ZEL [5]	BE/LUM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BUS [1]
SM	62	91	153	850.0	930.0	400.0	680.0	0.0	740.0	790.0	720.0	807.0
EU	63	90	153	-360.0	-320.0	-600.0	-290.0	0.0	-390.0	-280.0	-420.0	-243.0
LA	57	97	154	9340.0	9360.0	9800.0	10670.0	0.0	9580.0	9500.0	9470.0	0.0
CE	58	96	154	4480.0	4870.0	5400.0	5100.0	0.0	4290.0	4000.0	4330.0	0.0
PR	59	95	154	6840.0	6820.0	6800.0	7600.0	0.0	7090.0	7120.0	7120.0	0.0
ND	60	94	154	1980.0	2320.0	2700.0	2030.0	0.0	1790.0	1640.0	1780.0	0.0
PM	61	93	154	4340.0	4280.0	4000.0	4660.0	0.0	4480.0	4660.0	4510.0	3996.0
SM	62	92	154	-520.0	-220.0	-400.0	-880.0	0.0	-870.0	-730.0	-860.0	-724.0
EU	63	91	154	1860.0	1730.0	1400.0	2030.0	0.0	2080.0	2180.0	2050.0	1975.0
LA	57	98	155	7950.0	8170.0	8800.0	9240.0	0.0	8430.0	8050.0	8260.0	0.0
CE	58	97	155	6710.0	6910.0	7300.0	7580.0	0.0	6450.0	6360.0	6480.0	0.0
PR	59	96	155	5470.0	5650.0	6200.0	6190.0	0.0	5810.0	5660.0	5790.0	0.0
ND	60	95	155	4210.0	4390.0	4300.0	4550.0	0.0	3980.0	3930.0	4040.0	0.0
PM	61	94	155	2980.0	3110.0	3200.0	3180.0	0.0	3250.0	3210.0	3310.0	0.0
SM	62	93	155	1720.0	1850.0	1600.0	1610.0	0.0	1470.0	1620.0	1490.0	1629.0
EU	63	92	155	490.0	570.0	500.0	410.0	0.0	460.0	540.0	480.0	246.0
GD	64	91	155	-730.0	-680.0	-1200.0	-1010.0	0.0	-870.0	-840.0	-930.0	-817.0
CE	58	98	156	5340.0	5750.0	6300.0	6270.0	0.0	5290.0	4910.0	5270.0	0.0
PR	59	97	156	7690.0	7690.0	8000.0	8570.0	0.0	7970.0	8000.0	7950.0	0.0
ND	60	96	156	2860.0	3220.0	3700.0	3250.0	0.0	2710.0	2470.0	2710.0	0.0
PM	61	95	156	5190.0	5170.0	5000.0	5580.0	0.0	5430.0	5450.0	5560.0	0.0
SM	62	94	156	380.0	690.0	700.0	230.0	0.0	240.0	290.0	300.0	715.0
EU	63	93	156	2730.0	2640.0	2400.0	2790.0	0.0	2800.0	2760.0	2830.0	2453.0
GD	64	92	156	-2090.0	-1830.0	-2100.0	-2560.0	0.0	-2480.0	-2420.0	-2500.0	-2438.0
CE	58	99	157	7540.0	7770.0	8100.0	8610.0	0.0	7370.0	7020.0	7360.0	0.0
PR	59	98	157	6320.0	6520.0	7100.0	7280.0	0.0	6820.0	6550.0	6740.0	0.0
ND	60	97	157	5080.0	5270.0	5500.0	5630.0	0.0	4870.0	4810.0	4870.0	0.0
PM	61	96	157	3830.0	4010.0	4300.0	4270.0	0.0	4160.0	3990.0	4230.0	0.0
SM	62	95	157	2620.0	2740.0	2600.0	2650.0	0.0	2420.0	2490.0	2560.0	2597.0
EU	63	94	157	1360.0	1490.0	1500.0	1350.0	0.0	1580.0	1460.0	1630.0	1360.0
GD	64	93	157	140.0	230.0	-100.0	-150.0	0.0	-150.0	-150.0	-150.0	-58.0
CE	58	100	158	6180.0	6600.0	6900.0	7430.0	0.0	6410.0	6040.0	6360.0	0.0
PR	59	99	158	8520.0	8540.0	9000.0	9500.0	0.0	8900.0	8630.0	8820.0	0.0
ND	60	98	158	3730.0	4110.0	4600.0	4470.0	0.0	3710.0	3360.0	3660.0	0.0
PM	61	97	158	6040.0	6040.0	6100.0	6520.0	0.0	6330.0	6280.0	6390.0	0.0
SM	62	96	158	1270.0	1610.0	1800.0	1450.0	0.0	1140.0	990.0	1230.0	0.0
EU	63	95	158	3590.0	3630.0	3300.0	3660.0	0.0	3760.0	3710.0	3890.0	3453.0
GD	64	94	158	-1200.0	-910.0	-900.0	-1510.0	0.0	-1380.0	-1470.0	-1350.0	-1217.0
TB	65	93	158	1130.0	1030.0	800.0	1040.0	0.0	1040.0	1020.0	1050.0	934.0
PR	59	100	159	7150.0	7370.0	7800.0	8350.0	0.0	7940.0	7650.0	7820.0	0.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/HO [4]	LIR/ZEL [5]	BE/LOM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BOS [1]
ND	60	99	159	5920.0	6130.0	6400.0	6680.0	0.0	5790.0	5450.0	5740.0	0.0
PM	61	98	159	4700.0	4890.0	5100.0	5350.0	0.0	5170.0	4840.0	5190.0	0.0
SM	62	97	159	3480.0	3630.0	3700.0	3730.0	0.0	3310.0	3070.0	3390.0	0.0
EU	63	96	159	2250.0	2390.0	2500.0	2400.0	0.0	2480.0	2490.0	2570.0	2634.0
GD	64	95	159	1010.0	1140.0	900.0	840.0	0.0	810.0	800.0	910.0	969.0
TB	65	94	159	-200.0	-110.0	0.0	-380.0	0.0	-190.0	-360.0	-150.0	-365.0
PR	59	101	160	9330.0	9370.0	9700.0	10450.0	0.0	9750.0	9600.0	9630.0	0.0
ND	60	100	160	4570.0	4970.0	5300.0	5650.0	0.0	4830.0	4460.0	4740.0	0.0
PM	61	99	160	6900.0	6900.0	7100.0	7460.0	0.0	7250.0	6970.0	7270.0	0.0
SM	62	98	160	2130.0	2500.0	2700.0	2650.0	0.0	2150.0	1710.0	2190.0	0.0
EU	63	97	160	4450.0	4410.0	4300.0	4580.0	0.0	4650.0	4620.0	4730.0	0.0
GD	64	96	160	-310.0	10.0	100.0	-330.0	0.0	-470.0	-510.0	-420.0	-108.0
TB	65	95	160	2000.0	1920.0	1800.0	1850.0	0.0	2000.0	1910.0	2110.0	1837.0
PR	59	102	161	7980.0	8200.0	8600.0	9410.0	0.0	8990.0	8600.0	8750.0	0.0
ND	60	101	161	6750.0	6980.0	7200.0	7730.0	0.0	6650.0	6410.0	6550.0	0.0
PM	61	100	161	5540.0	5740.0	6000.0	6430.0	0.0	6280.0	6000.0	6280.0	0.0
SM	62	99	161	4330.0	4510.0	4700.0	4770.0	0.0	4230.0	3830.0	4270.0	0.0
EU	63	98	161	3100.0	3280.0	3400.0	3470.0	0.0	3490.0	3170.0	3520.0	0.0
GD	64	97	161	1900.0	2030.0	1800.0	1880.0	0.0	1700.0	1690.0	1760.0	1963.0
TB	65	96	161	670.0	790.0	800.0	610.0	0.0	720.0	540.0	780.0	584.0
DY	66	95	161	-530.0	-440.0	-700.0	-880.0	0.0	-870.0	-920.0	-780.0	-852.0
ND	60	102	162	5400.0	5810.0	6200.0	6790.0	0.0	5870.0	5410.0	5670.0	0.0
PM	61	101	162	7720.0	7740.0	7800.0	8420.0	0.0	8110.0	7960.0	8090.0	0.0
SM	62	100	162	2990.0	3370.0	3600.0	3840.0	0.0	3270.0	2870.0	3270.0	0.0
EU	63	99	162	5290.0	5280.0	5400.0	5490.0	0.0	5570.0	5370.0	5610.0	0.0
GD	64	98	162	560.0	900.0	1000.0	860.0	0.0	540.0	250.0	550.0	1400.0
TB	65	97	162	2880.0	2820.0	2600.0	2710.0	0.0	2880.0	2680.0	2950.0	2421.0
DY	66	96	162	-1860.0	-1560.0	-1700.0	-2050.0	0.0	-2140.0	-2290.0	-2100.0	-2135.0
ND	60	103	163	7560.0	7790.0	8000.0	8790.0	0.0	7670.0	7240.0	7480.0	0.0
PM	61	102	163	6370.0	6590.0	6900.0	7500.0	0.0	7340.0	6990.0	7210.0	0.0
SM	62	101	163	5160.0	5360.0	5400.0	5830.0	0.0	5090.0	4810.0	5090.0	0.0
EU	63	100	163	3960.0	4140.0	4300.0	4530.0	0.0	4610.0	4410.0	4610.0	0.0
GD	64	99	163	2760.0	2910.0	2900.0	2910.0	0.0	2610.0	2330.0	2640.0	0.0
TB	65	98	163	1540.0	1690.0	1700.0	1640.0	0.0	1730.0	1330.0	1750.0	1702.0
DY	66	97	163	340.0	450.0	300.0	100.0	0.0	20.0	-30.0	60.0	-3.0
ND	60	104	164	6210.0	6640.0	7200.0	7900.0	0.0	6900.0	6460.0	6710.0	0.0
PM	61	103	164	12340.0	12770.0	13200.0	14410.0	0.0	13450.0	12670.0	13240.0	0.0
EU	63	101	164	6140.0	6130.0	6100.0	6440.0	0.0	6430.0	6360.0	6430.0	0.0
GD	64	100	164	1410.0	1780.0	1900.0	2030.0	0.0	1650.0	1300.0	1640.0	0.0
TB	65	99	164	3730.0	3690.0	3600.0	3600.0	0.0	3810.0	3410.0	3830.0	3855.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/HO [4]	LIR/ZEL [5]	BE/LOM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BOS [1]
DY	66	98	164	-970.0	-660.0	-700.0	-900.0	0.0	-1130.0	-1340.0	-1140.0	-1028.0
HO	67	97	164	1320.0	1240.0	1000.0	960.0	0.0	1080.0	1060.0	1070.0	1004.0
PM	61	104	165	13160.0	13590.0	14300.0	15470.0	0.0	14480.0	13720.0	14270.0	0.0
EU	63	102	165	4790.0	4990.0	5200.0	5600.0	0.0	5650.0	5390.0	5560.0	0.0
GD	64	101	165	3590.0	3770.0	3800.0	3950.0	0.0	3480.0	3220.0	3450.0	0.0
TB	65	100	165	2400.0	2560.0	2500.0	2680.0	0.0	2850.0	2370.0	2850.0	0.0
DY	66	99	165	1200.0	1340.0	1300.0	1100.0	0.0	940.0	820.0	950.0	1285.0
HO	67	98	165	10.0	120.0	-100.0	-130.0	0.0	-70.0	-350.0	-130.0	-375.0
PM	61	105	166	9310.0	9340.0	9600.0	10460.0	0.0	10180.0	9930.0	9930.0	0.0
SM	62	104	166	4650.0	5050.0	5600.0	6110.0	0.0	5340.0	4880.0	5260.0	0.0
EU	63	103	166	6930.0	6950.0	7000.0	7440.0	0.0	7450.0	7230.0	7370.0	0.0
GD	64	102	166	2270.0	2650.0	2900.0	3180.0	0.0	2700.0	2220.0	2580.0	0.0
TB	65	101	166	4560.0	4540.0	4400.0	4530.0	0.0	4670.0	4240.0	4660.0	0.0
DY	66	100	166	-110.0	220.0	300.0	230.0	0.0	-20.0	-150.0	-50.0	484.0
HO	67	99	166	2180.0	2120.0	1800.0	1810.0	0.0	2010.0	1820.0	1960.0	1856.0
SM	62	105	167	6780.0	7000.0	7300.0	7960.0	0.0	7160.0	6740.0	6940.0	0.0
EU	63	104	167	5600.0	5810.0	6200.0	6680.0	0.0	6680.0	6480.0	6600.0	0.0
GD	64	103	167	4420.0	4610.0	4600.0	5020.0	0.0	4500.0	4040.0	4400.0	0.0
TB	65	102	167	3240.0	3410.0	3500.0	3730.0	0.0	3890.0	3340.0	3790.0	0.0
DY	66	101	167	2050.0	2210.0	2200.0	2110.0	0.0	1810.0	1620.0	1770.0	0.0
HO	67	100	167	860.0	1000.0	800.0	880.0	0.0	1050.0	890.0	970.0	970.0
SM	62	106	168	5450.0	5850.0	6300.0	7170.0	0.0	6380.0	5950.0	6050.0	0.0
EU	63	105	168	7740.0	7760.0	8000.0	8480.0	0.0	8500.0	8350.0	8280.0	0.0
GD	64	104	168	3090.0	3480.0	4000.0	4300.0	0.0	3720.0	3270.0	3630.0	0.0
TB	65	103	168	5380.0	5370.0	5200.0	5520.0	0.0	5700.0	5200.0	5610.0	0.0
DY	66	102	168	730.0	1090.0	1300.0	1360.0	0.0	1030.0	590.0	900.0	0.0
HO	67	101	168	3020.0	2970.0	2600.0	2710.0	0.0	2860.0	2770.0	2790.0	2716.0
SM	62	107	169	7570.0	7780.0	7900.0	8990.0	0.0	8080.0	7680.0	7740.0	0.0
EU	63	106	169	6400.0	6610.0	7300.0	7740.0	0.0	7720.0	7570.0	7390.0	0.0
GD	64	105	169	5230.0	5430.0	5500.0	6080.0	0.0	5550.0	5130.0	5320.0	0.0
TB	65	104	169	4060.0	4240.0	4600.0	4810.0	0.0	4920.0	4480.0	4840.0	0.0
DY	66	103	169	2880.0	3050.0	3000.0	3160.0	0.0	2820.0	2370.0	2720.0	0.0
HO	67	102	169	1710.0	1860.0	1600.0	1910.0	0.0	2100.0	1840.0	1920.0	2048.0
EU	63	107	170	8510.0	8530.0	8800.0	9530.0	0.0	9410.0	9300.0	9090.0	0.0
GD	64	106	170	3900.0	4290.0	4900.0	5370.0	0.0	4770.0	4360.0	4430.0	0.0
TB	65	105	170	6190.0	6180.0	6200.0	6560.0	0.0	6740.0	6360.0	6520.0	0.0
DY	66	104	170	1560.0	1930.0	2400.0	2470.0	0.0	2050.0	1600.0	1960.0	0.0
HO	67	103	170	3850.0	3820.0	3300.0	3680.0	0.0	3900.0	3530.0	3740.0	4005.0
EU	63	108	171	7180.0	7370.0	8200.0	8730.0	0.0	8600.0	8440.0	8090.0	0.0
GD	64	107	171	6020.0	6220.0	6300.0	7130.0	0.0	6460.0	6080.0	6130.0	0.0

ELEMENT	Z	N	A	MYERS [2]	GR/HI/TA [3]	SEE/HO [4]	LIR/ZEL [5]	BE/LOM [6]	JA/GA/KE [7]	CO/KE [8]	JA/EY [9]	WAP/BOS [1]
TB	65	106	171	4860.0	5050.0	5600.0	5860.0	0.0	5960.0	5620.0	5640.0	0.0
DY	66	105	171	3690.0	3880.0	4000.0	4230.0	0.0	3870.0	3460.0	3640.0	0.0
EU	63	109	172	9270.0	9260.0	9800.0	10550.0	0.0	10620.0	10400.0	10080.0	0.0
GD	64	108	172	4700.0	5070.0	5800.0	6340.0	0.0	5640.0	5210.0	5130.0	0.0
TB	65	107	172	6960.0	6970.0	7000.0	7630.0	0.0	7660.0	7360.0	7330.0	0.0
DY	66	106	172	2380.0	2760.0	3300.0	3530.0	0.0	3090.0	2710.0	2760.0	0.0
GD	64	109	173	6790.0	6960.0	7300.0	8120.0	0.0	7670.0	7170.0	7120.0	0.0
TB	65	108	173	5540.0	5830.0	6600.0	6880.0	0.0	6830.0	6510.0	6340.0	0.0
DY	66	107	173	4500.0	4680.0	4900.0	5280.0	0.0	4790.0	4430.0	4460.0	0.0
GD	64	110	174	5460.0	5800.0	6600.0	7220.0	0.0	6830.0	6290.0	6310.0	0.0
TB	65	109	174	7740.0	7720.0	8200.0	8670.0	0.0	8860.0	8500.0	8340.0	0.0
DY	66	108	174	3190.0	3540.0	4300.0	4520.0	0.0	3970.0	3560.0	3460.0	0.0
GD	64	111	175	7540.0	7640.0	8000.0	9050.0	0.0	8780.0	8260.0	7880.0	0.0
TB	65	110	175	6410.0	6560.0	7600.0	7830.0	0.0	8020.0	7640.0	7540.0	0.0
DY	66	109	175	5270.0	5440.0	5800.0	6280.0	0.0	6000.0	5520.0	5450.0	0.0
TB	65	111	176	8470.0	8400.0	9100.0	9690.0	0.0	9970.0	9620.0	9090.0	0.0
DY	66	110	176	3970.0	4290.0	5200.0	5430.0	0.0	5160.0	4650.0	4660.0	0.0
TB	65	112	177	7150.0	7190.0	8300.0	8730.0	0.0	8920.0	8570.0	8020.0	0.0
DY	66	111	177	6040.0	6130.0	6800.0	7240.0	0.0	7110.0	6620.0	6220.0	0.0
DY	66	112	178	4710.0	4940.0	6000.0	6260.0	0.0	6050.0	5560.0	5140.0	0.0
DY	66	113	179	6760.0	6660.0	7400.0	8170.0	0.0	8150.0	7770.0	7000.0	0.0
DY	66	114	180	5440.0	5310.0	6500.0	7060.0	0.0	6960.0	6370.0	5560.0	0.0

1) A.H. Wapstra, K. Bos: Atomic Data and Nucl. Data Tables 17 (1976) 474
2) W.D. Myers: see reference 1) p. 411
3) H.V. Groote, E.R. Hilf, K. Takahashi: see reference 1) p. 418
4) P.A. Seeger, W.M. Howard: see reference 1) p. 428
5) S. Liran, N. Zeldes: see reference 1) p. 431
6) M. Beiner, R.J. Lombard, D. Mas: see reference 1) p. 450
7) J. Jänecke: see reference 1) p. 455
8) E. Comay, I. Kelson: see reference 1) p. 463
9) J. Jänecke, B.P. Eynon: see reference 1) p. 467

ANNEXES TO REVIEW PAPER 15

(Integral determination of fission product inventory and decay power)

R.E. Schenter, T.R. England

APPENDIX A1

A1. ADDITIONAL EXPERIMENTAL RESULTS

This appendix provides tables and figures of experimental results for ^{235}U thermal fission. Recent measurements on ^{239}Pu and ^{233}U have been made at IRT, LASL and ORNL. The results are preliminary and are not reported here. However, comparisons of the LASL measurements with ENDF/B-IV calculations show agreement within the experimental uncertainties. The current estimate of uncertainty in the preliminary measurement of data are approximately 5%. The final uncertainty is expected to be smaller when the radio chemical determination of the number of fissions is complete, and of course, this determination could also alter the agreement with calculation.

LASL RESULTS

TABLE I

EXPERIMENTAL AND CALCULATED DECAY HEAT FOR THE PRODUCTS OF THERMAL FISSION OF ^{235}U.

	2×10^4 s Irradiation				Infinite Irradiation			
Cooling Time (s)	Experimental Decay Heat (MeV/fiss)	Experimental Uncertainty (1 σ in %)	Calculated Decay Heat using CINDER-10 and ENDF/B-IV (MeV/fiss)	Ratio Exp/Calc Decay Heat	Experimental Decay Heat Extended by CINDER-10 and ENDF/B-IV (MeV/fiss)	Experimental Uncertainty[a] (1 σ in %)	Calculated Decay Heat using CINDER-10 and ENDF/B-IV (MeV/fiss)	Ratio Exp/Calc Decay Heat
10	8.10	4.1	7.780	1.041	9.65	3.5	9.327	1.035
15	7.38	3.0	7.239	1.019	8.93	2.5	8.786	1.016
20	6.933	2.6	6.842	1.013	8.480	2.2	8.389	1.011
25	6.595	2.4	6.531	1.010	8.142	2.0		
30	6.335	2.3	6.276	1.009	7.882	1.9		
35	6.109	2.2	6.060	1.008	7.656	1.8		
40	5.920	2.1	5.873	1.008	7.467	1.7	7.420	1.006
45	5.758	2.1	5.709	1.009	7.305	1.7		
50	5.614	2.0	5.562	1.009	7.160	1.6		
55	5.481	2.0	5.429	1.010	7.027	1.6		
60	5.358	2.0	5.309	1.009	6.904	1.6	6.855	1.007
65	5.244	1.9	5.198	1.009	6.790	1.5		
70	5.141	1.9	5.097	1.009	6.687	1.5		
75	5.047	1.9	5.003	1.009	6.593	1.5		
80	4.958	1.8	4.915	1.009	6.504	1.5	6.461	1.007
85	4.881	1.8	4.834	1.010	6.427	1.4		
90	4.806	1.8	4.758	1.010	6.351	1.4		

TABLE I

(continued)

	2×10^4 s Irradiation				Infinite Irradiation			
Cooling Time (s)	Experimental Decay Heat (MeV/fiss)	Experimental Uncertainty (1 σ in %)	Calculated Decay Heat using CINDER-10 and ENDF/B-IV (MeV/fiss)	Ratio Exp/Calc Decay Heat	Experimental Decay Heat Extended by CINDER-10 and ENDF/B-Iv (MeV/fiss)	Experimental Uncertainty[a] (1 σ in %)	Decay Heat from CINDER-10 & ENDF/B-IV (MeV/fiss)	Ratio Exp/Calc Decay Heat
95	4.734	1.8	4.686	1.010	6.279	1.4		
100	4.667	1.8	4.619	1.010	6.212	1.4	6.164	1.008
110	4.544	1.8	4.496	1.011	6.089	1.4		
120	4.426	1.7	4.385	1.009	5.971	1.4		
130	4.339	1.7	4.286	1.012	5.884	1.4		
140	4.251	1.7	4.195	1.013	5.795	1.4		
150	4.170	1.7	4.112	1.014	5.714	1.4	5.655	1.010
160	4.092	1.7	4.035	1.014	5.636	1.4		
170	4.021	1.7	3.964	1.014	5.565	1.3		
180	3.960	1.7	3.899	1.016	5.503	1.3		
190	3.899	1.6	3.838	1.016	5.442	1.3		
200	3.841	1.6	3.780	1.016	5.384	1.3	5.323	1.011
250	3.608	1.6	3.541	1.019	5.150	1.3		
300	3.419	1.6	3.355	1.019	4.960	1.3		
350	3.265	1.6	3.205	1.019	4.805	1.3		
400	3.135	1.6	3.078	1.019	4.673	1.3	4.616	1.012
450	3.022	1.6	2.969	1.018	4.559	1.3		
500	2.920	1.6	2.873	1.016	4.456	1.3		
600	2.746	1.5	2.709	1.014	4.280	1.2	4.243	1.009
700	2.598	1.5	2.572	1.010	4.130	1.2		
800	2.474	1.5	2.455	1.008	4.004	1.2	3.984	1.005
900	2.363	1.5	2.351	1.005	3.890	1.2		
1000	2.264	1.5	2.258	1.003	3.789	1.2	3.783	1.002
1100	2.173	1.5	2.174	1.000	3.696	1.2		
1200	2.093	1.5	2.098	0.998	3.614	1.2		
1300	2.020	1.5	2.027	0.997	3.539	1.2		
1400	1.950	1.5	1.962	0.994	3.467	1.2		
1500	1.886	1.5	1.901	0.992	3.401	1.2	3.416	0.996
1600	1.827	1.5	1.845	0.990	3.340	1.2		
1700	1.773	1.5	1.791	0.990	3.284	1.2		
1800	1.721	1.5	1.741	0.989	3.229	1.2		
1900	1.671	1.5	1.694	0.986	3.177	1.2		
2000	1.627	1.5	1.650	0.986	3.131	1.2	3.154	0.993
2500	1.431	1.5	1.460	0.980	2.925	1.3		
3000	1.283	1.5	1.311	0.979	2.768	1.3		
3500	1.166	1.5	1.191	0.979	2.641	1.3		
4000	1.067	1.5	1.092	0.977	2.533	1.3	2.557	0.991
4500	0,9808	1.5	1.008	0.973	2.438	1.3		
5000	0.9111	1.5	0.9362	0.973	2.360	1.3		
6000	0.7998	2.3	0.8198	0.976	2.231	1.5	2.251	0.991
7000	0.7195	2.0	0.7287	0.987	2.135	1.5		
8000	0.6480	1.6	0.6553	0.989	2.049	1.5	2.056	0.997
9000	0.5886	1.6	0.5948	0.990	1.975	1.5		
10000	0.5401	1.7	0.5440	0.993	1.912	1.5	1.916	0.998
15000	0.3803	1.8	0.3778	1.007	1.691	1.6	1.689	1.001
20000	0.2918	2.0	0.2874	1.015	1.552	1.7	1.547	1.003
25000	0.2359	2.2	0.2311	1.021	1.453	1.7		
30000	0.1947	2.3	0.1923	1.012	1.375	1.7		
62183	0.0823	2.2	0.0832	0.989	1.105	1.9		
100000	0.0454	2.2	0.0455	0.998	0.971	1.9	0.969	1.002

[a] Calculated on the assumptions that the calculated infinite irradiation decay heat curve has 2% uncertainty for $t_{cool} \geq 2 \times 10^4$ s, and that this uncertainty is uncorrelated with the experimental uncertainty.

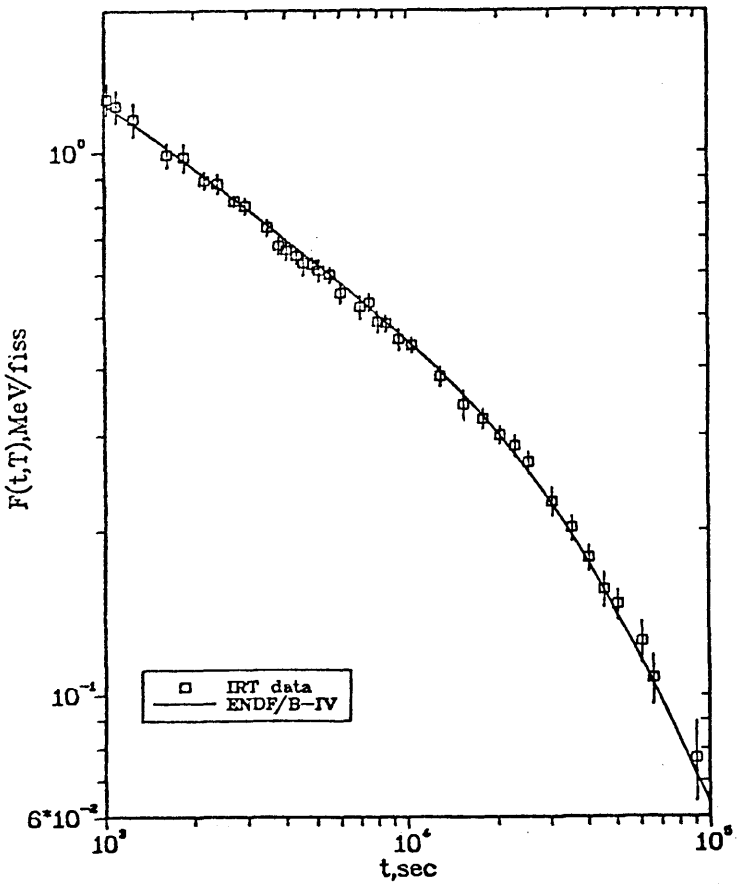
24 H Total Decay Heat

IRT RESULTS

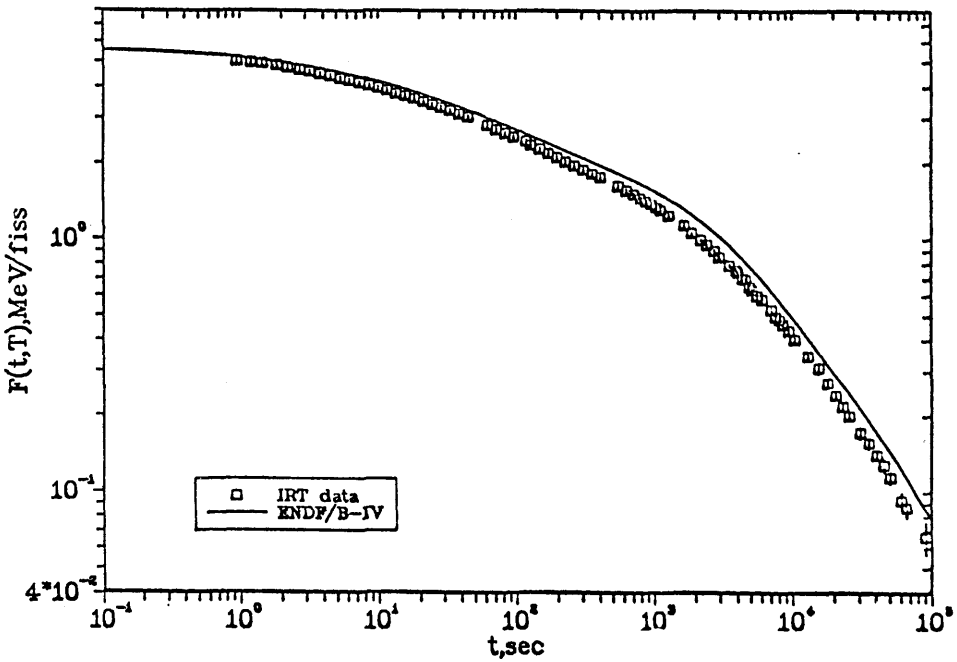
SAMPLE NO 12526 NORMALIZATION= 1.0000E+00 IREP= 0

		T	FH	STD DEV	MEV/FIS	STD DEV
200	1	.05	.0559826	.0004687	11.1965275	.0937423
201	2	1.05	.0553822	.0003142	11.0764465	.0628314
202	3	1.35	.0546496	.0003114	10.9299145	.0622764
203	4	1.70	.0534932	.0002830	10.6986485	.0565902
204	5	2.10	.0524594	.0002789	10.4918804	.0557733
205	6	2.55	.0517378	.0002610	10.3474083	.0521901
206	7	3.05	.0501680	.0002551	10.0335960	.0510136
207	8	3.60	.0490868	.0002404	9.8161602	.0480749
208	9	4.30	.0477978	.0002216	9.5595531	.0443178
209	10	5.10	.0467991	.0002100	9.3598278	.0435912
210	11	6.00	.0457293	.0002051	9.1458603	.0410209
211	12	7.05	.0443779	.0001969	8.8755760	.0393827
212	13	8.25	.0432393	.0001878	8.6478634	.0375530
213	14	9.65	.0422265	.0001805	8.4453011	.0360938
214	15	11.25	.0409207	.0001727	8.1841431	.0345377
215	16	13.10	.0397948	.0001653	7.9589605	.0330620
216	17	15.25	.0387226	.0001589	7.7445269	.0317725
217	18	17.80	.0375481	.0001524	7.5096138	.0304871
218	19	20.65	.0364743	.0001465	7.2948513	.0292910
219	20	24.00	.0353608	.0001405	7.0721550	.0280976
220	21	27.90	.0342993	.0001350	6.8598671	.0269933
221	22	32.40	.0333590	.0001304	6.6718073	.0260724
222	23	37.55	.0322848	.0001258	6.4568071	.0251542
223	24	43.60	.0312088	.0001207	6.2417536	.0241440
224	25	50.55	.0302885	.0001161	6.0416956	.0232266
225	26	58.60	.0292304	.0001120	5.8460875	.0224017
226	27	68.00	.0282362	.0001079	5.6460476	.0215768
227	28	78.95	.0272355	.0001038	5.4471073	.0207525
228	29	91.55	.0262650	.0000996	5.2530060	.0199283
229	30	106.15	.0252501	.0000960	5.0580164	.0191950
230	31	123.15	.0243819	.0000923	4.8763723	.0184556
231	32	142.75	.0234698	.0000890	4.6937971	.0178087
232	33	165.50	.0225777	.0000853	4.5155428	.0170698
233	34	191.00	.0217845	.0000826	4.3569012	.0165161
235	35	259.50	.0201573	.0000766	4.0314600	.0153175
236	36	300.40	.0194118	.0000738	3.8823590	.0147645
237	37	347.80	.0187021	.0000711	3.7404127	.0142120
238	38	402.65	.0180172	.0000680	3.6034369	.0137524
240	39	541.30	.0166425	.0000637	3.3285012	.0127420
241	40	619.65	.0159035	.0000614	3.1966991	.0122033
242	41	690.00	.0154206	.0000596	3.0841131	.0119170
243	42	776.35	.0149572	.0000578	2.9914408	.0115584
244	43	854.60	.0144784	.0000564	2.8956881	.0112764
245	44	932.90	.0140810	.0000549	2.8162088	.0109812
246	45	1011.20	.0136774	.0000854	2.7354898	.0010724
247	46	1089.50	.0133308	.0000522	2.6661696	.0104491
249	47	1248.58	.0126654	.0000504	2.5330763	.0100785
250	48	1783.00	.0108385	.0000910	2.1676946	.0103586
251	49	1986.10	.0104446	.0000467	2.0889173	.0093394
252	50	2436.60	.0096652	.0000891	1.9330468	.0178125
253	51	2632.70	.0091767	.0000663	1.8353364	.0132511
254	52	3081.60	.0085981	.0000556	1.7196131	.0111142
256	53	3728.80	.0077905	.0000310	1.5581808	.0061921
257	54	3923.70	.0077073	.0001050	1.5414593	.0209955
258	55	4390.50	.0070654	.0000677	1.4130876	.0135410
259	56	4599.80	.0070170	.0000617	1.4034061	.0123345
260	57	5040.20	.0066746	.0000906	1.3349292	.0181181
261	58	5587.80	.0063627	.0000278	1.2725446	.0055518
262	59	6530.60	.0058239	.0000269	1.1647885	.0053789
264	60	8397.90	.0050596	.0000209	1.0119116	.0041758
265	61	9323.60	.0046736	.0000246	.9347277	.0049182
266	62	10202.20	.0045015	.0000149	.9003047	.0029866
267	63	11232.40	.0042986	.0000106	.8597126	.0037223
268	64	14448.70	.0036212	.0000150	.7242475	.0029919
269	65	16920.80	.0032669	.0000141	.6533873	.0028125
270	66	19425.00	.0029588	.0000096	.5915929	.0019166
271	67	21959.10	.0027295	.0000114	.5458950	.0022741
272	68	24467.50	.0025149	.0000087	.5029870	.0017458
273	69	26947.00	.0023378	.0000096	.4675626	.0019219
274	70	31920.30	.0020719	.0000079	.4143791	.0015781
275	71	36942.40	.0018289	.0000079	.3641086	.0015885
276	72	41949.70	.0016221	.0000079	.3244125	.0015827
277	73	46932.70	.0014688	.0000055	.2937684	.0011020
278	74	51935.20	.0013414	.0000059	.2682836	.0011704
279	75	64986.60	.0010759	.0000063	.2151748	.0012545
280	76	74896.00	.0009123	.0000055	.1824598	.0010973
282	77	94923.59	.0007099	.0000055	.1419835	.0010916
283	78	100083.50	.0006724	.0000062	.1344726	.0012490
284	79	106588.00	.0006198	.0000054	.1239661	.0010875
285	80	129893.00	.0004881	.0000054	.0976169	.0010730
286	81	154877.59	.0004049	.0000061	.0809763	.0012238

FISSION-PRODUCT DECAY-HEAT COMPARISON
of ENDF/B-IV and IRT data (1976)
Zr 98 and Y 96 corrected
Beta Heat 24-hr. exposure



FISSION-PRODUCT DECAY-HEAT COMPARISON
of ENDF/B-IV and IRT data (1976)
Zr 98 and Y 96 corrected
Gamma Heat 24-hr. exposure



ORNL RESULTS

Gamma-ray Energy Release and Yields from Fission Created by Thermal-Neutron Fission of ^{235}U

Irradiation Time (sec)	Waiting Time[c] (sec)	Counting Time[d] (sec)	"Average" Time[e] (sec)	Experiment[a] Energy Release (MeV/fission)	Experiment[a] Yield (Gammas/fission)	Calculation[b] Energy Release (MeV/fission)	Ratio Experiment/ Calculation
1.0 ± 0.1	1.7 ± 0.1	1	2.7	0.188 ± 0.010	0.206 ± 0.012	0.150	1.25
	2.7	1	3.7	0.146 ± 0.007	0.164 ± 0.008	0.120	1.22
	3.7	1	4.7	0.119 ± 0.005	0.133 ± 0.006	0.0990	1.20
	4.7	2	6.2	0.188 ± 0.008	0.212 ± 0.009	0.160	1.18
	6.7	3	8.7	0.210 ± 0.008	0.237 ± 0.010	0.184	1.14
	9.7	5	12.7	0.248 ± 0.008	0.281 ± 0.011	0.227	1.09
	14.7	5	17.7	0.182 ± 0.006	0.209 ± 0.009	0.173	1.05
	19.7	5	22.7	0.143 ± 0.005	0.166 ± 0.007	0.139	1.03
	24.7	10	30.2	0.223 ± 0.007	0.253 ± 0.010	0.217	1.03
	34.7	10	40.2	0.170 ± 0.005	0.189 ± 0.008	0.166	1.02
	44.7	15	52.7	0.199 ± 0.006	0.216 ± 0.008	0.193	1.03
	59.7	15	67.7	0.154 ± 0.005	0.164 ± 0.006	0.150	1.03
	75	15	83	0.124 ± 0.004	0.128 ± 0.005	0.121	1.02
	90	20	100	0.131 ± 0.004	0.138 ± 0.005	0.130	1.00
10.0	10.7	6	18.7	0.216 ± 0.007	0.249 ± 0.010	0.202	1.07
	16.7	8	25.7	0.211 ± 0.007	0.241 ± 0.010	0.202	1.04
	24.7	10	34.7	0.198 ± 0.006	0.226 ± 0.009	0.192	1.03
	34.7	10	44.7	0.156 ± 0.005	0.171 ± 0.006	0.151	1.03
	44.7	10	54.7	0.127 ± 0.004	0.138 ± 0.005	0.124	1.02
	54.7	20	69.7	0.197 ± 0.006	0.211 ± 0.007	0.194	1.01
	74.7	20	89.7	0.149 ± 0.004	0.157 ± 0.005	0.147	1.01
	95	20	110	0.117 ± 0.004	0.122 ± 0.004	0.122	0.96
	115	40	140	0.175 ± 0.005	0.185 ± 0.006	0.176	1.00
	155	60	190	0.177 ± 0.005	0.191 ± 0.007	0.180	0.98
	215	80	260	0.158 ± 0.005	0.176 ± 0.007	0.161	0.98
	295	100	350	0.137 ± 0.004	0.161 ± 0.006	0.138	0.99
	395	200	500	0.186 ± 0.006	0.225 ± 0.010	0.186	1.00
	595	200	700	0.132 ± 0.005	0.166 ± 0.010	0.131	1.00

BAPL RESULTS

Total Decay Power Following Thermal Neutron Fission Comparing Gunst et al. ("Measured") with ENDF/B-IV (Calc)

	Watts from sample		
Cooling time (h)	Calculated*	"Measured"	ANS Std.
14.55	61.40	65.1 63.7†	63.8
14.97	60.53	64.4 62.9†	62.9
15.45	59.59	63.0 61.1†	61.8
15.90	58.73	61.9 60.5†	60.9
34.49	39.46	41.49 40.68†	39.99
35.49	38.88	41.15 40.35†	39.36
81.77	25.28	25.83 25.51†	25.30
83.34	25.03	24.72 24.41†	25.04
201.5	15.08	15.73 15.60†	15.12
202.2	15.04	15.36 15.23†	15.09
296.3	11.45	11.24 11.15†	11.51
297.6	11.41	11.20 11.11†	11.48
1067	3.357	3.23 3.21†	3.35
1068	3.354	3.35 3.32†	3.35
2099	1.581	1.57 1.56†	1.55
2436	1.354	1.16 1.15†	1.32
2770	1.180	1.15 1.15†	1.15
3108	1.039	1.10 1.10†	1.01
4142	0.730	0.69 0.68†	0.71
4551	0.641	0.64 0.63†	0.64

* Using 0.6718 gms of U235 fissioned in the sample

† Corrected values August 10, 1977

A2. ADDITIONAL SUMMATION CALCULATION RESULTS

TABLE I - TOTAL INTEGRAL AFTERHEAT FOLLOWING THERMAL NEUTRON FISSION OF U235

Comparisons { Total Decay Power-Calorimetry (Lott 1973)
Calculations using ORIGEN (ENDF/IV, 1975)
Calculations using PEPIN 1976 (French Library, CEA-N-1822)

Irradiation time (s)	100 seconds					1 000 seconds					5 000 seconds				
Cooling time (s)	Total Integral Afterheat (MeV/fission)			$\frac{\text{Calc - Meas}}{\text{Meas}} \times 100$		Total Integral Afterheat (MeV/fission)			$\frac{\text{Calc - Meas}}{\text{Meas}} \times 100$		Total Integral Afterheat			$\frac{\text{Calc - Meas}}{\text{Meas}} \times 100$	
	Measured	Calculated				Measured	Calculated				Measured	Calculated			
		ORIGEN	PEPIN	ORIGEN	PEPIN		ORIGEN	PEPIN	ORIGEN	PEPIN		ORIGEN	PEPIN	ORIGEN	PEPIN
70	0.982	1.121	1.125	14.1	14.5			2.941					4.268		
100	0.755	0.838	0.842	11.0	11.5			2.497					3.800		
150	0.539	0.577	0.580	7.1	7.6			2.029					3.303		
200	0.411	0.433	0.435	5.4	5.8	1.692	1.746	1.733	3.2	2.42			2.975		
300	0.275	0.285	0.286	3.6	4.0	1.385	1.389	1.370	0.3	-1.08	2.579	2.679	2.554	3.9	- 0.97
500	0.170	0.170	0.169		- 0.5	1.031	1.029	1.001	- 0.2	-2.9	2.158	2.211	2.083	2.4	- 3.5
700	0.124	0.124	0.122		- 1.6	0.835	0.835	0.804		-3.7	1.862	1.927	1.797	3.5	- 3.5
1000	0.090	0.0894	0.0863	- 0.7	- 4.1	0.655	0.660	0.626	0.7	-4.4	1.552	1.635	1.510	-5.3	- 2.7
1500	0.0622	0.0618	0.0583	- 0.6	- 6.3	0.473	0.489	0.455	3.4	-3.8	1.220	1.313	1.197	7.6	- 1.9
2000	0.0471	0.0467	0.0433	- 0.8	- 8.0	0.362	0.383	0.351	5.8	-3.03	1.004	1.093	0.989	8.9	- 1.5
3000	0.0293	0.0301	0.0273	2.7	- 7.0	0.237	0.258	0.233	8.9	-1.7	0.735	0.807	0.725	9.8	- 1.4
5000			0.0145			0.132	0.146	0.130	10.6	-1.5	0.463	0.512	0.461	10.6	- 0.4
7000			0.0094			0.0861	0.0968	0.0873	12.4	1.4	0.331	0.363	0.332	9.7	+ 0.3
10000			0.006C			0.0546	0.0608	0.0563	11.4	3.1	0.225	0.242	0.228	4.4	1.3
15000			0.0035			0.0322	0.0346	0.0335	7.5	4.0	0.140	0.146	0.144	4.3	2.8
20000			0.0024			0.0214	0.0230	0.0229	7.5	7.0	0.098	0.101	0.102	3.1	4.1
30000			0.0014					0.0135			0.0605	0.0614	0.0626	1.5	3.5
50000			0.00070					0.0069			0.0334	0.0325	0.0329	- 2.7	- 1.5
70000			0.00040					0.0043			0.0201	0.0207	0.0207	3.0	3.0

TABLE II - TOTAL INTEGRAL AFTERHEAT (MeV/fission)

U235 THERMAL FISSION - 2.10^4s. IRRADIATION TIME

comparisons { Total Decay Power Calorimetry (YARNELL and BENDT - Los Alamos)
Calculations using CINDER 10 (ENDF/IV)
Calculations using PEPIN 1976

Cooling Time (s)	Calorimeter (MeV/fiss)	Gamma Escape (MeV/fiss)	Total decay Heat (MeV/fiss)	Calc. Decay Heat (MeV/fiss.)		Decay Heat (Exp./Calc.)	
				CINDER 10	PEPIN 76	CINDER 10	PEPIN 76
10	8.078	0.240	8.318	7.780	7.647	1.069	1.088
15	7.336	0.221	7.557	7.239	7.131	1.044	1.060
20	6.835	0.207	7.042	6.842	6.757	1.029	1.042
30	6.212	0.189	6.401	6.276	6.224	1.020	1.028
50	5.498	0.164	5.662	5.562	5.543	1.018	1.021
80	4.854	0.141	4.995	4.915	4.916	1.016	1.016
100	4.560	0.130	4.690	4.619	4.624	1.015	1.014
150	4.071	0.112	4.183	4.112	4.118	1.017	1.016
200	3.755	0.101	3.856	3.780	3.784	1.020	1.019
300	3.338	0.088	3.426	3.355	3.351	1.021	1.022
500	2.852	0.073	2.925	2.873	2.858	1.018	1.023
800	2.421	0.061	2.482	2.455	2.433	1.011	1.020
1000	2.216	0.056	2.272	2.258	2.235	1.006	1.017
1500	1.849	0.046	1.895	1.901	1.879	0.997	1.009
2000	1.596	0.040	1.636	1.650	1.631	0.992	1.003
3000	1.259	0.031	1.290	1.311	1.300	0.984	0.992
5000	0.8974	0.0222	0.9196	0.9362	0.935	0.982	0.984
8000	0.6400	0.0149	0.6549	0.6553	0.661	0.999	0.991
10000	0.5344	0.0120	0.5464	0.5440	0.552	1.004	0.990
15000	0.3740	0.0077	0.3817	0.3778	0.386	1.010	0.988
20000	0.2847	0.0054	0.2901	0.2874	0.294	1.009	0.985
30000	0.1896	0.0032	0.1928	0.1923	0.196	1.003	0.984
100000	0.0445	0.0006	0.0451	0.0455	0.0454	0.991	0.993

TABLE III TOTAL INTEGRAL AFTERHEAT (MeV/fission)

U235 THERMAL FISSION - 1 YEAR IRRADIATION TIME

Cooling time (s.)	CINDER 10 (ENDF/IV)	PEPIN 1976	Calc. $\frac{\text{PEPIN 1976}}{\text{CINDER 10}}$
1.	$1.146\ 10^{1}$	$1.128\ 10^{1}$	0.984
2.	$1.094\ 10^{1}$	$1.077\ 10^{1}$	0.984
5.	$1.001\ 10^{1}$	9.869	0.986
$1.\ 10^{1}$	9.167	9.045	0.987
$2.\ 10^{1}$	8.228	8.152	0.991
$5.\ 10^{1}$	6.944	6.931	0.998
$1.\ 10^{2}$	5.998	6.007	1.002
$2.\ 10^{2}$	5.156	5.163	1.001
$5.\ 10^{2}$	4.242	4.230	0.997
$1.\ 10^{3}$	3.616	3.594	0.994
$2.\ 10^{3}$	2.987	2.969	0.994
$5.\ 10^{3}$	2.218	2.217	1.
$1.\ 10^{4}$	1.752	1.757	1.003
$2.\ 10^{4}$	1.386	1.389	1.002
$5.\ 10^{4}$	1.025	1.020	0.995
$1.\ 10^{5}$	$8.185\ 10^{-1}$	$8.130\ 10^{-1}$	0.993
$2.\ 10^{5}$	$6.656\ 10^{-1}$	$6.601\ 10^{-1}$	0.992
$5.\ 10^{5}$	$5.111\ 10^{-1}$	$5.057\ 10^{-1}$	0.989
$1.\ 10^{6}$	$3.991\ 10^{-1}$	$3.948\ 10^{-1}$	0.989
$2.\ 10^{6}$	$2.910\ 10^{-1}$	$2.884\ 10^{-1}$	0.991
$5.\ 10^{6}$	$1.703\ 10^{-1}$	$1.690\ 10^{-1}$	0.992
$1.\ 10^{7}$	$1.021\ 10^{-1}$	$1.009\ 10^{-1}$	0.988
$2.\ 10^{7}$	$4.904\ 10^{-2}$	$4.829\ 10^{-2}$	0.985
$5.\ 10^{7}$	$1.590\ 10^{-2}$	$1.610\ 10^{-2}$	1.013

TABLE IV : U 235 THERMAL FISSION - 1 YEAR IRRADIATION TIME

BETA AND GAMMA AFTERHEAT (MeV/fission)

Cooling Time (s)	BETA			GAMMA		
	CINDER 10	PEPIN 1976	PEPIN / CINDER10	CINDER 10	PEPIN 1976	PEPIN / CINDER 10
1.	5.805	5.929	1.021	5.661	5.349	0.945
2.	5.5	5.609	1.020	5.44	5.162	0.949
5.	4.96	5.044	1.017	5.05	4.825	0.955
$1.\ 10^{1}$	4.465	4.546	1.018	4.702	4.499	0.957
$2.\ 10^{1}$	3.931	4.022	1.023	4.297	4.13	0.961
$5.\ 10^{1}$	3.245	3.331	1.026	3.699	3.6	0.973
$1.\ 10^{2}$	2.763	2.825	1.022	3.255	3.183	0.978
$2.\ 10^{2}$	2.343	2.378	1.015	2.813	2.786	0.990
$5.\ 10^{2}$	1.884	1.905	1.011	2.358	2.324	0.986
$1.\ 10^{3}$	1.576	1.592	1.010	2.041	2.003	0.981
$2.\ 10^{3}$	1.285	1.295	1.007	1.702	1.674	0.983
$5.\ 10^{3}$	$9.675\ 10^{-1}$	$9.727\ 10^{-1}$	1.005	1.251	1.244	0.994
$1.\ 10^{4}$	$7.859\ 10^{-1}$	$7.875\ 10^{-1}$	1.002	$9.658\ 10^{-1}$	$9.701\ 10^{-1}$	1.044
$2.\ 10^{4}$	$6.279\ 10^{-1}$	$6.244\ 10^{-1}$	0.994	$7.584\ 10^{-1}$	$7.646\ 10^{-1}$	1.008
$5.\ 10^{4}$	$4.426\ 10^{-1}$	$4.350\ 10^{-1}$	0.983	$5.82\ 10^{-1}$	$5.847\ 10^{-1}$	1.005
$1.\ 10^{5}$	$3.386\ 10^{-1}$	$3.317\ 10^{-1}$	0.980	$4.799\ 10^{-1}$	$4.813\ 10^{-1}$	1.003
$2.\ 10^{5}$	$2.703\ 10^{-1}$	$2.645\ 10^{-1}$	0.979	$3.953\ 10^{-1}$	$3.956\ 10^{-1}$	1.0008
$5.\ 10^{5}$	$2.119\ 10^{-1}$	$2.07\ 10^{-1}$	0.977	$2.992\ 10^{-1}$	$2.987\ 10^{-1}$	0.998
$1.\ 10^{6}$	$1.733\ 10^{-1}$	$1.69\ 10^{-1}$	0.975	$2.258\ 10^{-1}$	$2.258\ 10^{-1}$	1.000
$2.\ 10^{6}$	$1.346\ 10^{-}$	$1.31\ 10^{-1}$	0.973	$1.564\ 10^{-1}$	$1.574\ 10^{-1}$	1.006
$5.\ 10^{6}$	$8.67\ 10^{-2}$	$8.412\ 10^{-2}$	0.970	$8.36\ 10^{-2}$	$8.492\ 10^{-2}$	1.016
$1.\ 10^{7}$	$5.684\ 10^{-2}$	$5.505\ 10^{-2}$	0.968	$4.53\ 10^{-2}$	$4.583\ 10^{-2}$	1.012
$2.\ 10^{7}$	$3.323\ 10^{-2}$	$3.223\ 10^{-2}$	0.970	$1.581\ 10^{-2}$	$1.606\ 10^{-2}$	1.016
$5.\ 10^{7}$	$1.356\ 10^{-2}$	$1.32\ 10^{-2}$	0.973	$2.34\ 10^{-3}$	$2.898\ 10^{-3}$	1.238

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April 1 - June 30 1976
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FISP RESULTS (1977)

FISSION PRODUCT DECAY HEAT FOLLOWING AN INSTANTANEOUS BURST OF THERMAL FISSIONS IN 235U (MEV/FISS/SEC)

DECAY TIME S	(1) E-BETA	(2) E-GAMMA	(3) E-CE+X	1+2+3
0	0.6540	0.5828	4.666E-5	1.237
1	0.3912	0.3305	6.868E-5	0.7218
1.5	0.3251	0.2697	7.908E-5	0.5949
2	0.2770	0.2264	8.864E-5	0.5035
3	0.2116	0.1690	1.048E-4	0.3807
5	0.1403	0.1089	1.260E-4	0.2493
7	0.1034	0.07935	1.360E-4	0.1829
10	0.07338	0.05665	1.385E-4	0.1302
15	0.04908	0.03932	1.284E-4	0.08853
20	0.03683	0.03076	1.141E-4	0.06770
30	0.02465	0.02197	8.896E-5	0.04671
50	0.01489	0.01423	5.841E-5	0.02918
70	0.01052	0.01345	4.401E-5	0.02101
100	7.088E-3	7.332E-3	3.411E-5	0.01445
150	4.396E-3	4.722E-3	2.692E-5	9.144E-3
200	3.098E-3	3.372E-3	2.487E-5	6.494E-3
300	1.883E-3	2.042E-3	1.771E-5	3.943E-3
500	1.008E-3	1.078E-3	9.993E-6	2.096E-3
700	6.737E-4	7.243E-4	7.116E-6	1.405E-3
1000	4.457E-4	4.909E-4	5.393E-6	9.420E-4
1.5E03	2.816E-4	3.248E-4	4.276E-6	6.107E-4
2.0E03	2.003E-4	2.412E-4	3.652E-6	4.452E-4
3.0E03	1.186E-4	1.536E-4	2.799E-6	2.750E-4
5.0E03	5.884E-5	8.237E-5	1.769E-6	1.430E-4
7.0E03	3.777E-5	5.471E-5	1.186E-6	9.367E-5
1.0E04	2.446E-5	3.422E-5	6.989E-7	5.938E-5
1.5E04	1.539E-5	1.896E-5	3.419E-7	3.469E-5
2.0E04	1.116E-5	1.210E-5	2.066E-7	2.347E-5
3.0E04	7.035E-6	6.589E-6	1.212E-7	1.375E-5
5.0E04	3.648E-6	3.365E-6	7.715E-8	7.090E-6
7.0E04	2.190E-6	2.198E-6	5.744E-8	4.445E-6
1.0E05	1.221E-6	1.384E-6	4.147E-8	2.646E-6
1.5E05	6.192E-7	7.999E-7	2.891E-8	1.448E-6
2.0E05	3.805E-7	5.426E-7	2.261E-8	9.457E-7
3.0E05	2.000E-7	3.313E-7	1.580E-8	5.471E-7
5.0E05	1.066E-7	2.002E-7	9.466E-9	3.163E-7
7.0E05	7.654E-8	1.469E-7	6.579E-9	2.303E-7
1.0E06	5.484E-8	1.225E-7	4.466E-9	1.618E-7
1.5E06	3.737E-8	6.501E-8	2.875E-9	1.053E-7
2.0E06	2.821E-8	4.605E-8	2.093E-9	7.635E-8
3.0E06	1.840E-8	2.707E-8	1.314E-9	4.678E-8
5.0E06	1.012E-8	1.265E-8	6.721E-10	2.344E-8
7.0E06	6.817E-9	7.813E-9	3.921E-10	1.502E-8
1.0E07	4.539E-9	4.984E-9	1.940E-10	9.717E-9
1.5E07	2.786E-9	2.800E-9	7.044E-11	5.656E-9
2.0E07	1.916E-9	1.588E-9	3.148E-11	3.535E-9
3.0E07	1.132E-9	5.137E-10	1.290E-11	1.659E-9
5.0E07	5.958E-10	8.570E-11	7.695E-12	6.892E-10
7.0E07	3.636E-10	4.102E-11	6.055E-12	4.107E-10
1.0E08	1.920E-10	2.942E-11	4.843E-12	2.263E-10
1.5E08	9.067E-11	2.354E-11	4.047E-12	1.103E-10

Fission Product Decay Heat Following ^{235}U Fast Fission

MeV Fission^{-1} 10^{5} s^{-1} Irradiation

Decay Time s	Fisp Calculation		
	E_{β}	E_{γ}	$E_{conv.el. + X rays}$
0	6.355	6.135	4.425×10^{-2}
1	5.839	5.687	4.422×10^{-2}
2	5.500	5.406	4.428×10^{-2}
5	4.897	4.928	4.445×10^{-2}
10	4.381	4.532	4.545×10^{-2}
20	3.857	4.117	4.950×10^{-2}
50	3.164	3.498	5.963×10^{-2}
100	2.650	2.991	8.049×10^{-2}
200	2.179	2.493	8.945×10^{-2}
500	1.634	1.907	4.013×10^{-2}
10^{3}	1.291	1.540	2.869×10^{-2}
2×10^{3}	9.861×10^{-1}	1.192	2.416×10^{-2}
5×10^{3}	6.610×10^{-1}	7.730×10^{-1}	1.647×10^{-2}
10^{4}	4.795×10^{-1}	5.165×10^{-1}	1.090×10^{-2}
2×10^{4}	3.249×10^{-1}	3.243×10^{-1}	7.362×10^{-3}
5×10^{4}	1.530×10^{-1}	1.649×10^{-1}	4.700×10^{-3}
10^{5}	6.944×10^{-2}	8.800×10^{-2}	3.097×10^{-3}
2×10^{5}	2.797×10^{-2}	4.293×10^{-2}	1.935×10^{-3}
5×10^{5}	9.877×10^{-3}	1.869×10^{-2}	8.777×10^{-4}
10^{6}	5.326×10^{-3}	9.810×10^{-3}	4.355×10^{-4}
2×10^{6}	2.799×10^{-3}	4.480×10^{-3}	2.114×10^{-4}
5×10^{6}	1.032×10^{-3}	1.266×10^{-3}	6.990×10^{-5}
10^{7}	4.702×10^{-4}	5.081×10^{-4}	2.047×10^{-5}
2×10^{7}	1.995×10^{-4}	1.619×10^{-4}	3.368×10^{-6}
5×10^{7}	6.294×10^{-5}	9.494×10^{-6}	8.169×10^{-7}
10^{8}	2.026×10^{-5}	3.309×10^{-6}	$5:071 \times 10^{-7}$
2×10^{8}	6.072×10^{-6}	2.223×10^{-6}	3.834×10^{-7}
5×10^{8}	3.798×10^{-6}	1.681×10^{-6}	2.975×10^{-7}
10^{9}	2.561×10^{-6}	1.164×10^{-6}	2.060×10^{-7}

Calculated Decay Energies after an Instantaneous Irradiation (n-thermal) on ^{235}U (MeV/Fission)

NAIG M.Iida & T. Yoshida

Cooling Time(sec)	Present Calculation*		Tasaka's Original	
	Gamma	Beta	Gamma	Beta
1	0.316	0.364	0.321	0.328
4	0.559	0.599	0.539	0.559
10	0.642	0.655	0.589	0.622
40	0.726	0.676	0.647	0.668
100	0.705	0.642	0.655	0.649
400	0.527	0.499	0.517	0.499
1000	0.462	0.455	0.462	0.456

*) E_β and E_γ are Estimated for data-unknown-nuclides ($\sim$300) with the aid of Gross Theory (T.Yoshida, NSE,63,376(1977)). Other necessary data taken from Tasaka's data library (K. Tasaka, JAERI-M 5997)

APPENDIX A3

A3. ADDITIONAL BETA AND GAMMA SPECTRAL COMPARISONS

This appendix includes an additional 32 spectral comparisons of the 102 made to date using ENDF/B-IV data. These are selected to show comparisons over the full range of experimental cooling times. At the shortest irradiation and cooling times the ENDF/B-IV spectral data is likely inadequate, and at the longest cooling times (compared to the associated irradiation times) the quality of the measurements deteriorates.

As noted in the main text, ENDF/B-IV includes data for 824 fission products of which 711 are unstable and contribute to the decay energy. Of the 711, 180 have either β and/or γ spectral data. The decay heat contribution of the 180 nuclides to the total energy is summarized in the main text for each experiment.

The ORNL experiments include both β and γ spectra at each irradiation and cooling time. Therefore, we have included both spectral types at corresponding times on each page.

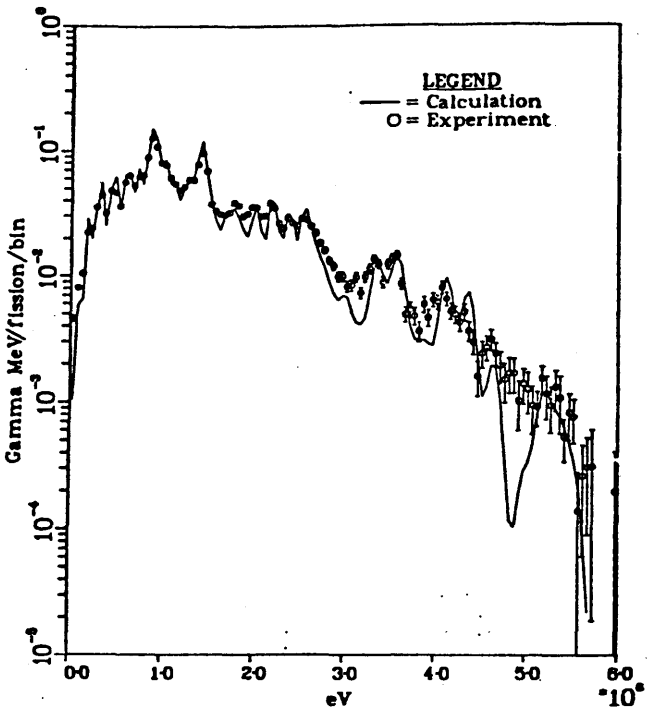

Fig. 1 Comparison of calculation with LASL 5.56 h irradiation, 70 s decay

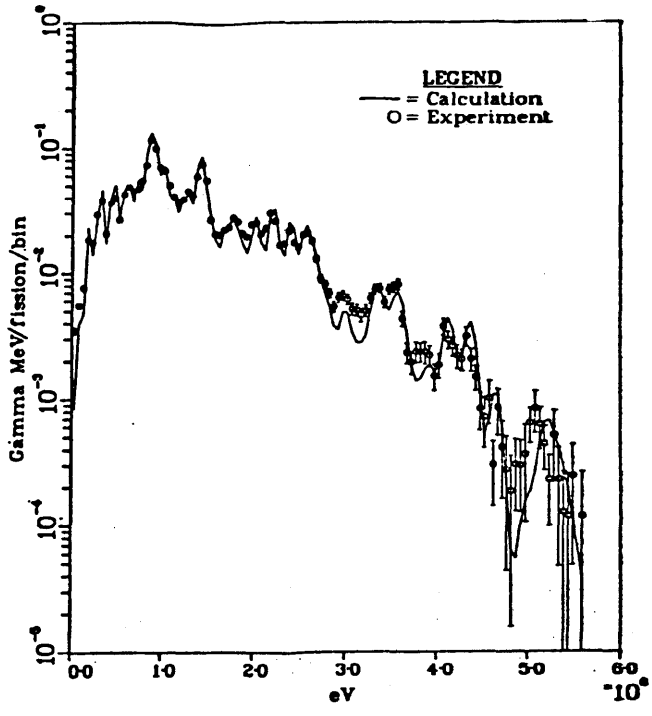

Fig. 2 Comparison of calculation with LASL 5.56 h irradiation, 199 s decay

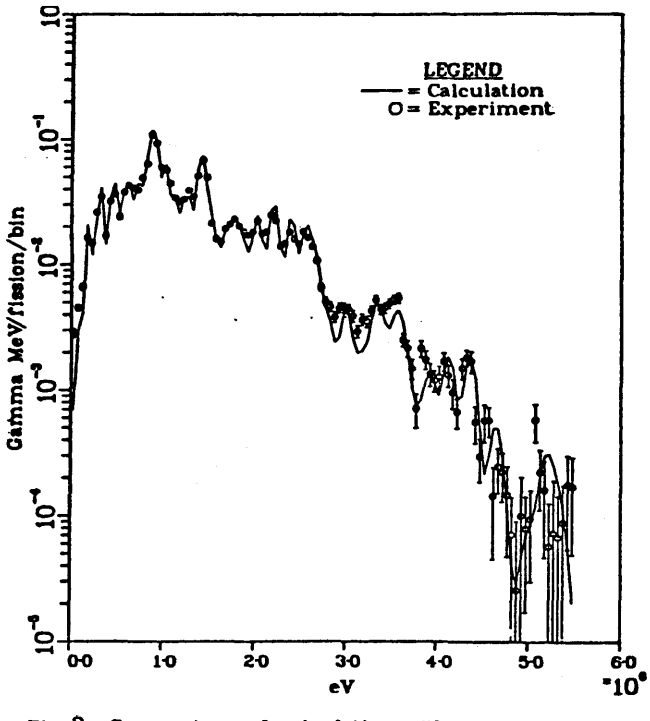

Fig. 3 Comparison of calculation with LASL 5.56 h irradiation, 388 s decay

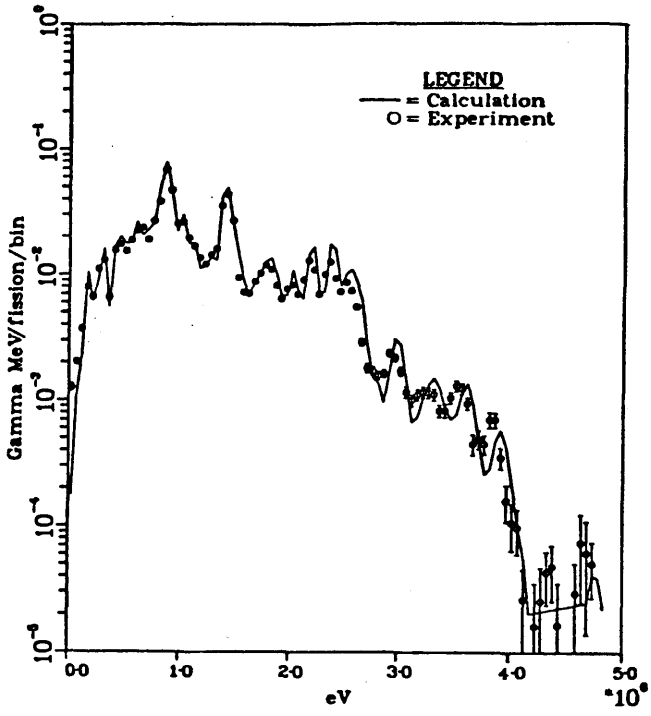

Fig. 4 Comparison of calculation with LASL 5.56 h irradiation, 2214 s decay

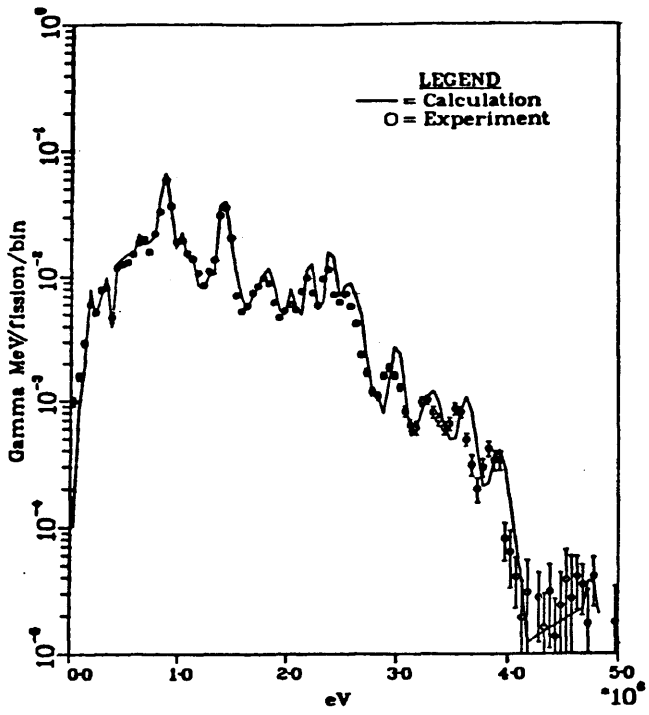

Fig. 5 Comparison of calculation with LASL 5.56 h irradiation, 3234 s decay

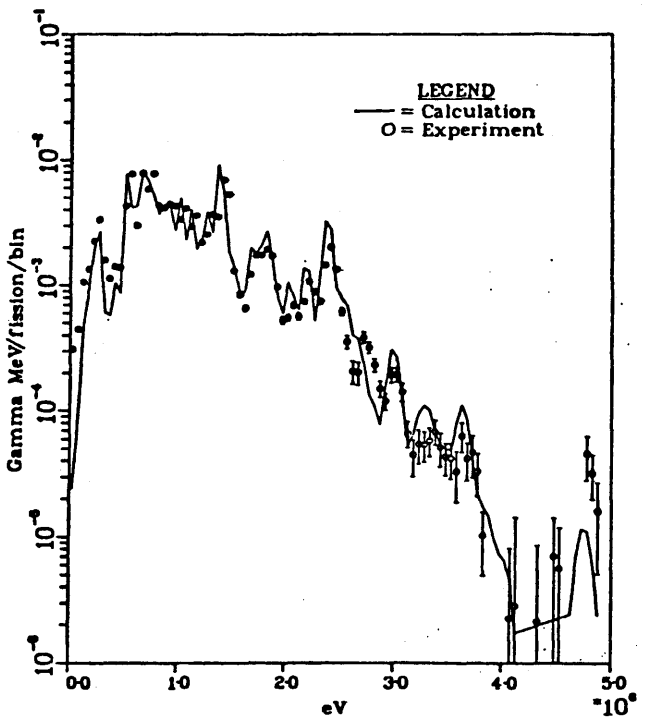

Fig. 6 Comparison of calculation with LASL 5.56 h irradiation, 20845 s decay

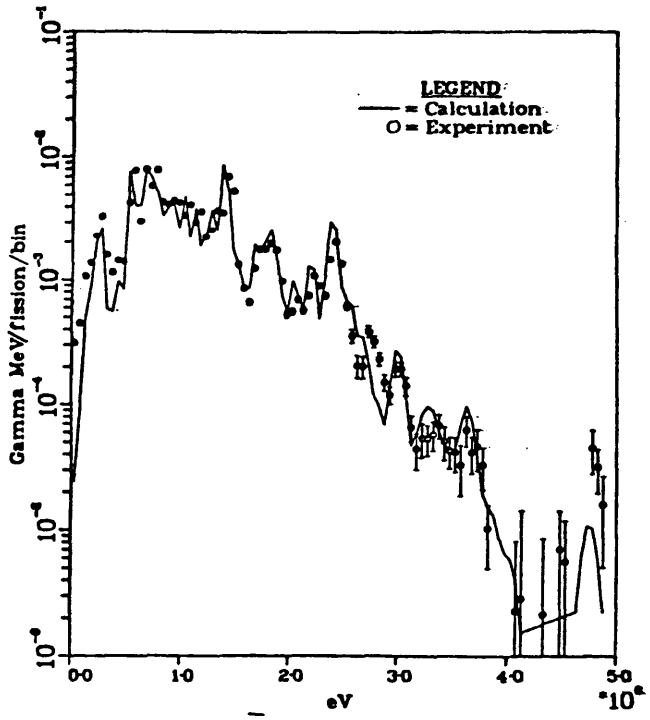

Fig. 7 Comparison of calculation with LASL 5.56 h irradiation, 21845 s decay

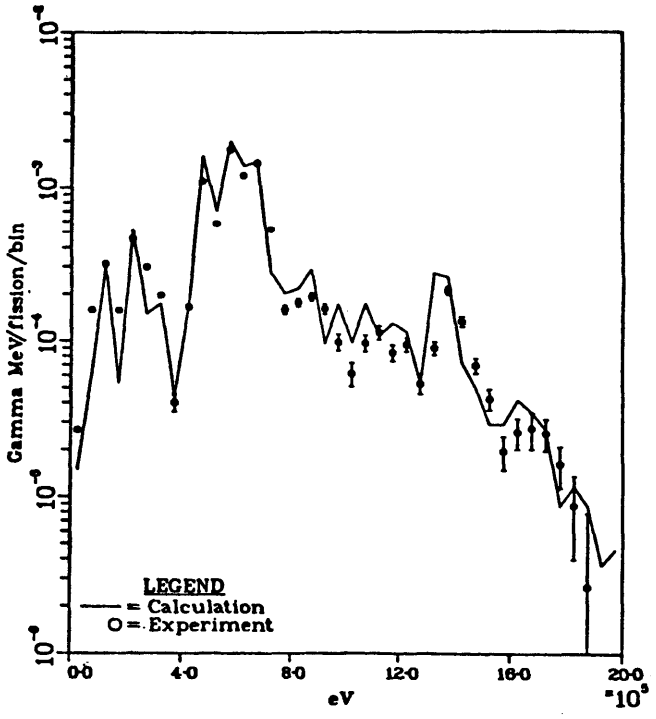

Fig. 8 Comparison of calculation with LASL 5 h irradiation, 151200 s decay

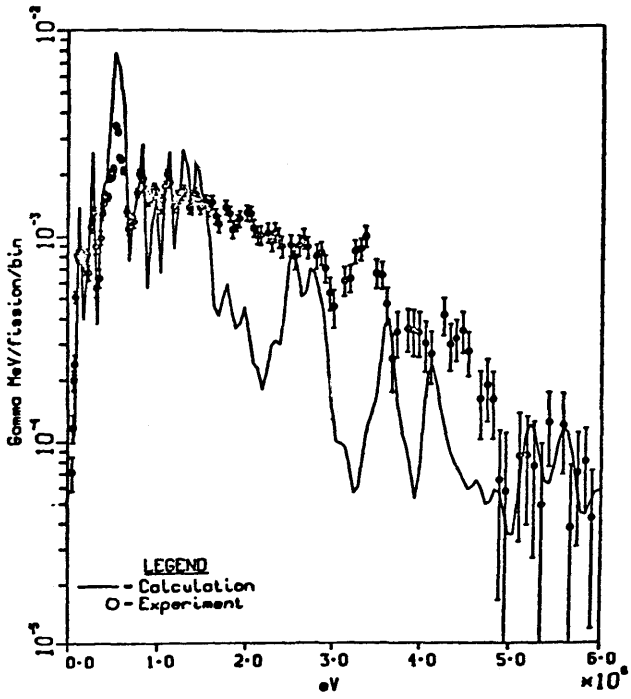

Fig. 9 Comparison of calculation with ORNL 1 s irradiation experiment, 5.7 s decay

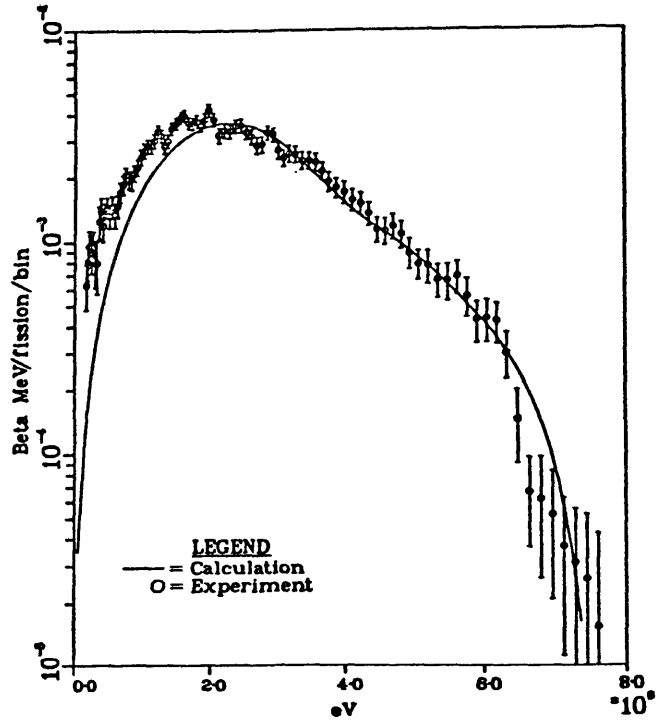

Fig.10 Irradiation time = 1 s, decay time = 5.7 s

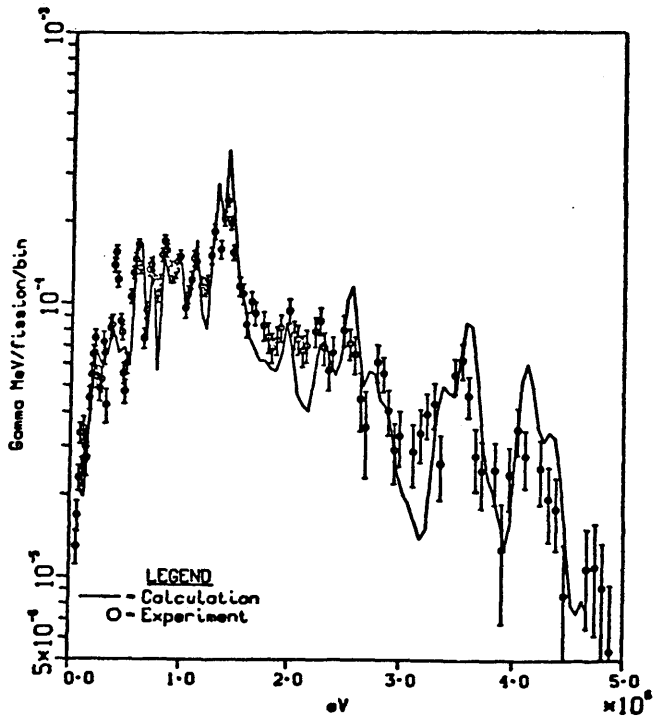

Fig.11 Comparison of calculation with ORNL 1 s irradiation experiment, 99.7 s decay

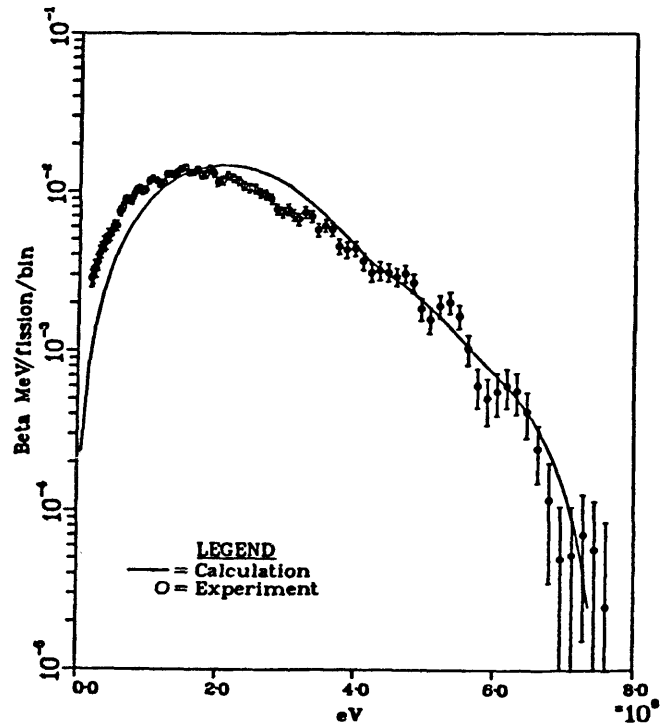

Fig.12 Irradiation time = 10 s, decay time = 13.7 s

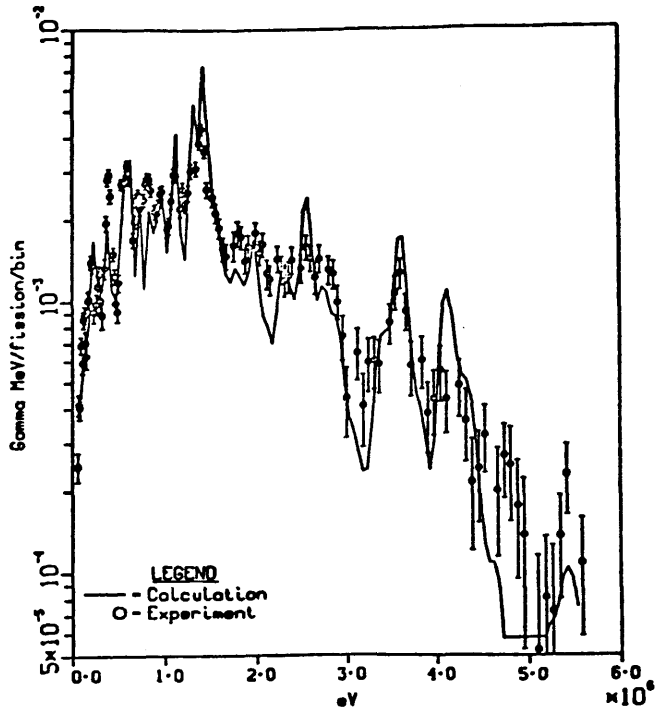

Fig.13 Comparison of calculation with ORNL 10 s irradiation experiment, 49.7 s decay

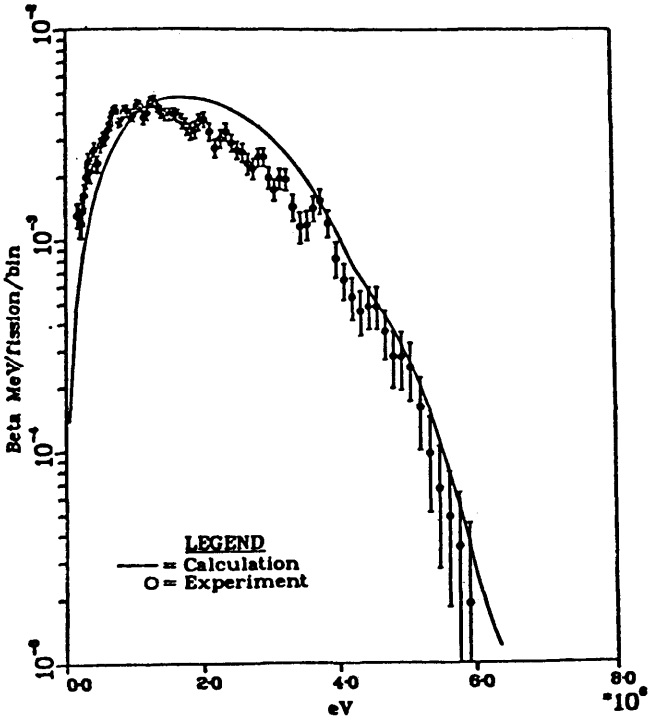

Fig.14 Irradiation time = 10 s, decay time = 49.7 s

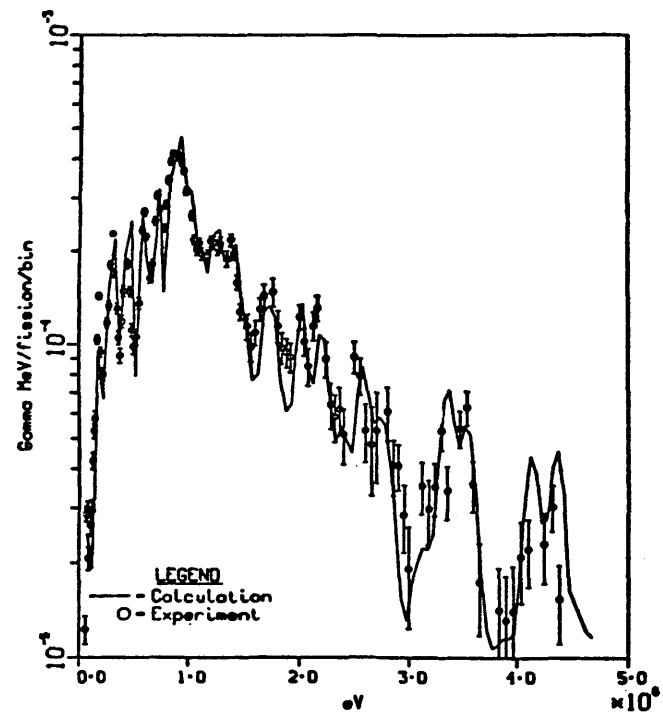

Fig.15 Comparison of calculation with ORNL 10 s irradiation experiment, 494.7 s decay

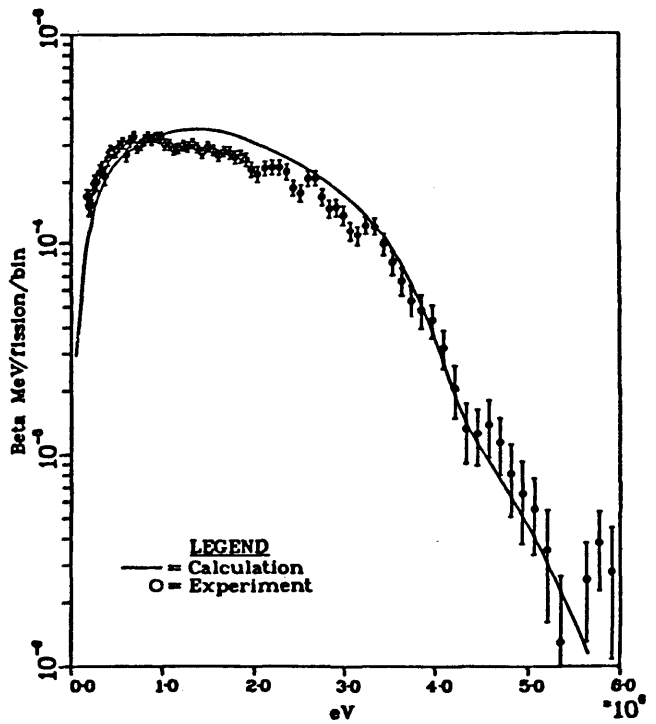

Fig.16 Irradiation time = 10 s, decay time = 494.7 s

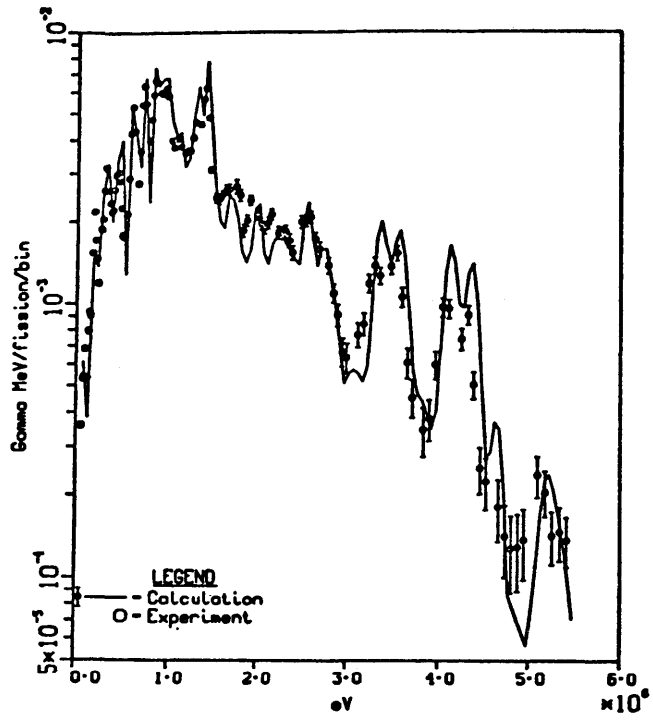

Fig. 17 Comparison of calculation with ORNL 100 s irradiation experiment, 210.0 s decay

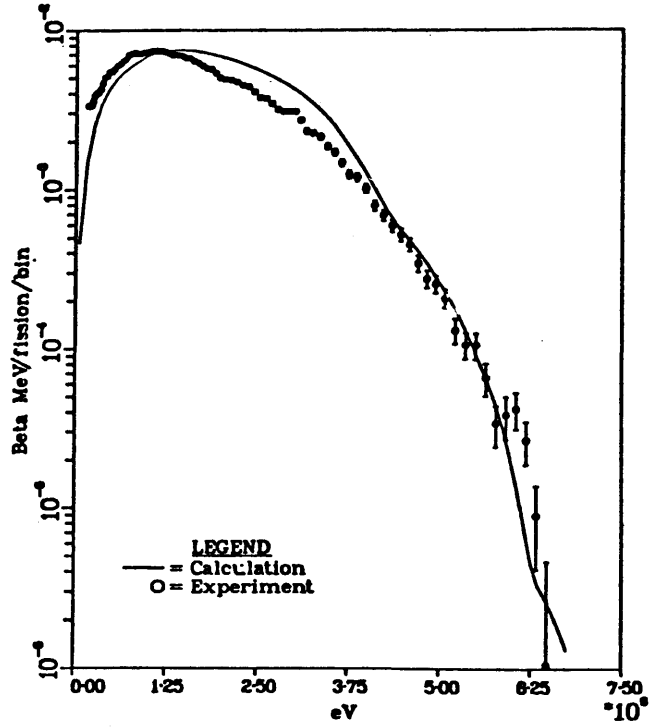

Fig. 18 Irradiation time = 100 s, decay time = 210 s

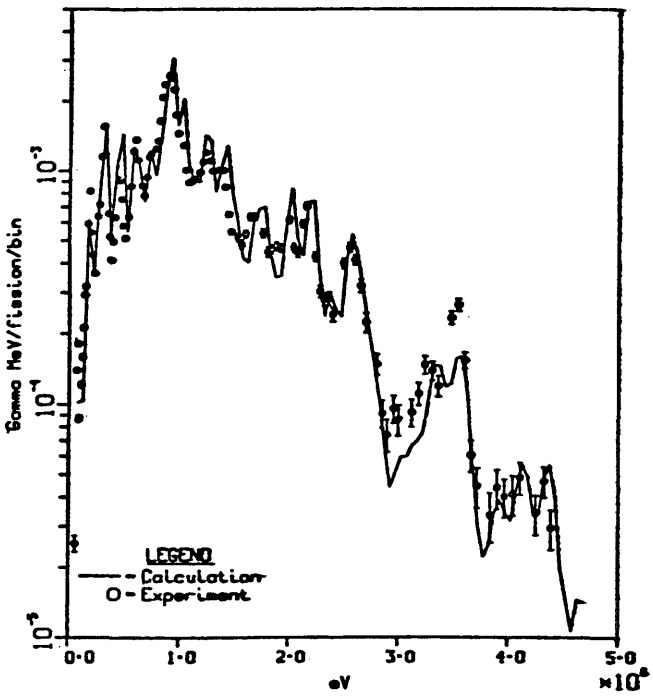

Fig. 19 Comparison of calculation with ORNL 100 s irradiation experiment, 950.0 s decay

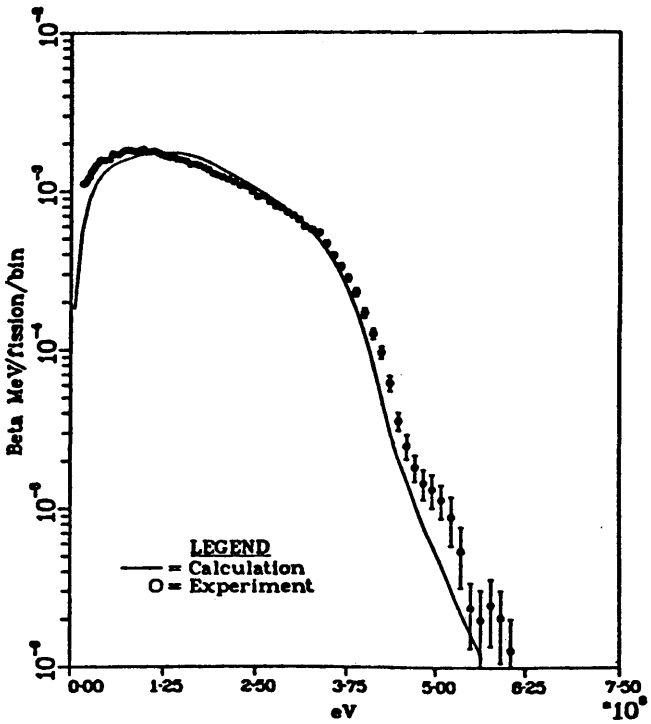

Fig. 20 Irradiation time = 100 s, decay time = 950 s

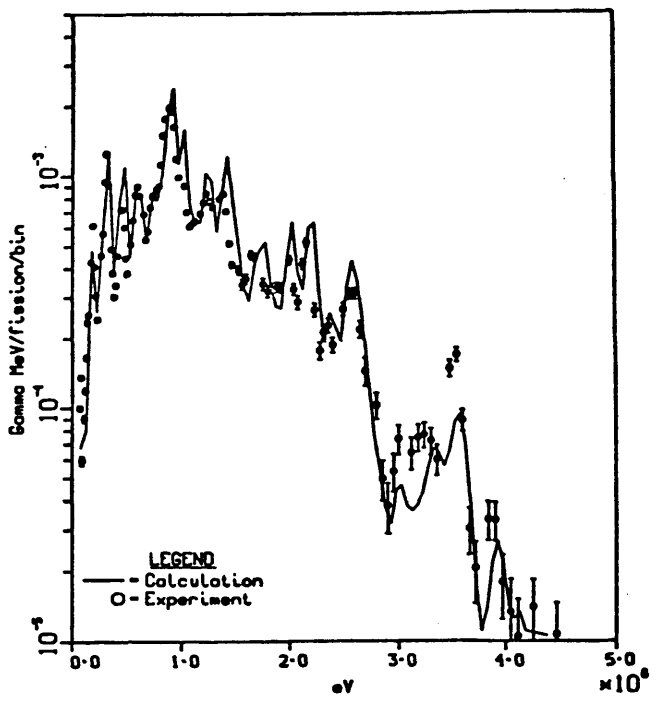

Fig. 21 Comparison of calculation with ORNL 100 s irradiation experiment, 1350.0 s decay

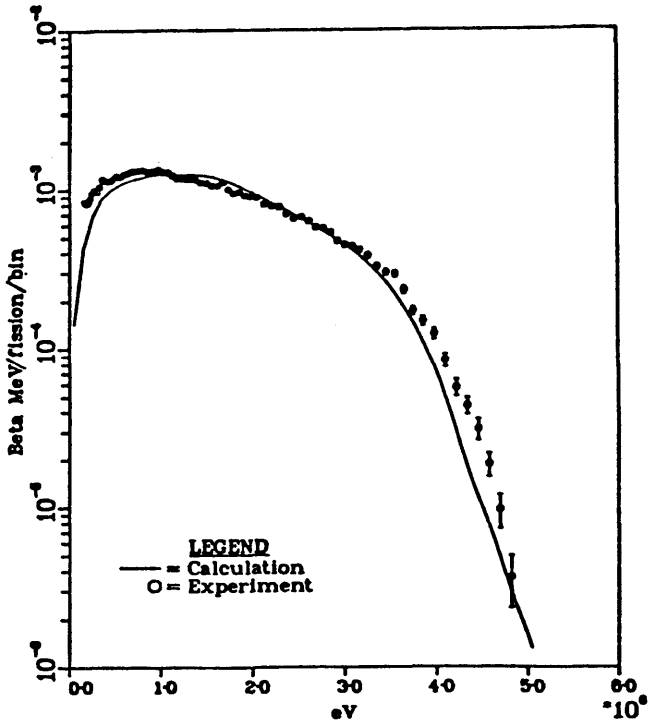

Fig. 22 Irradiation time = 100 s, decay time = 1350 s

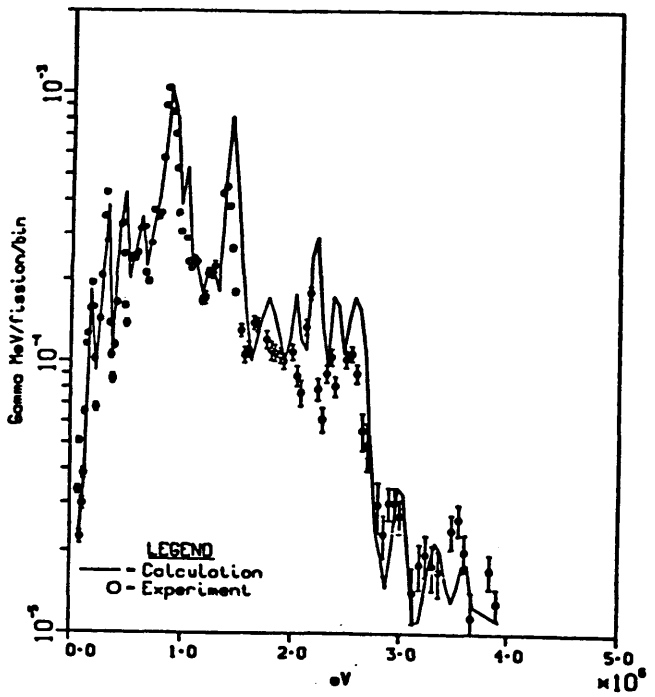

Fig. 23 Comparison of calculation with ORNL 100 s irradiation experiment, 3450.0 s decay

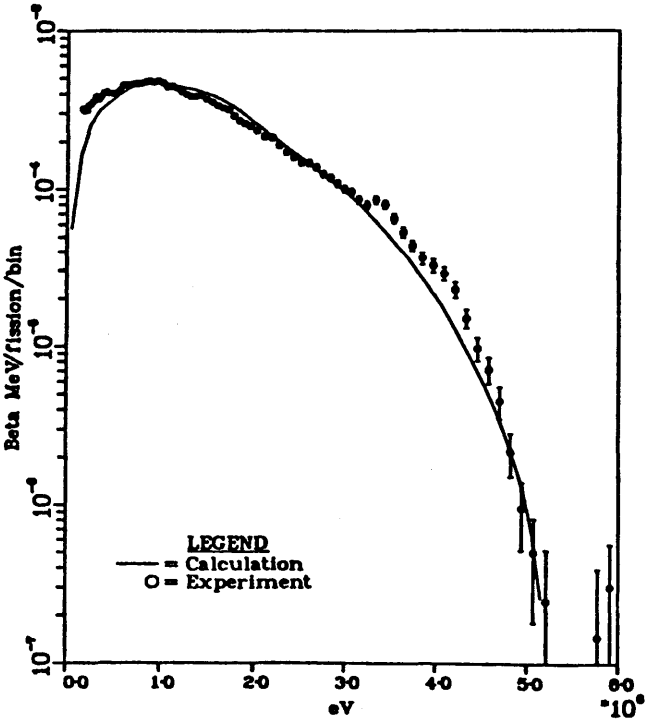

Fig. 24 Irradiation time = 100 s, decay time = 3450 s

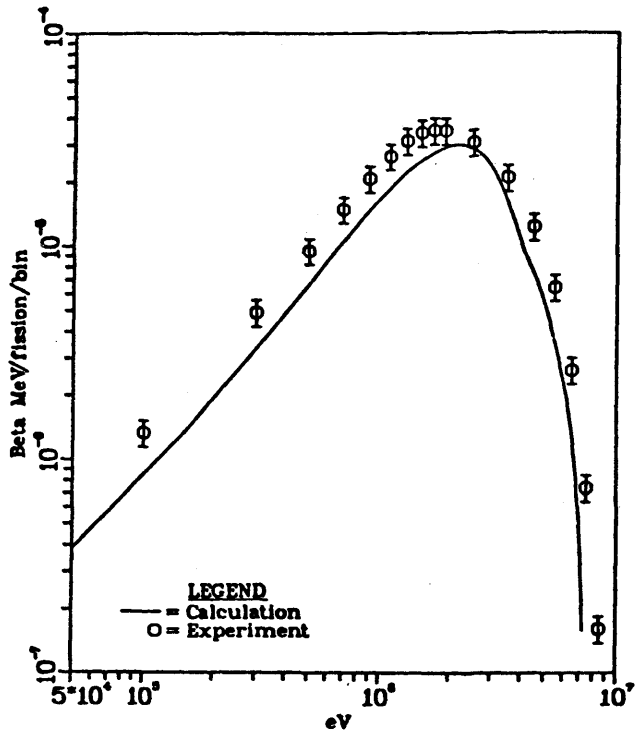

Fig.25 Comparison of calculation with UI
15 ms irradiation experiment, 13 s decay

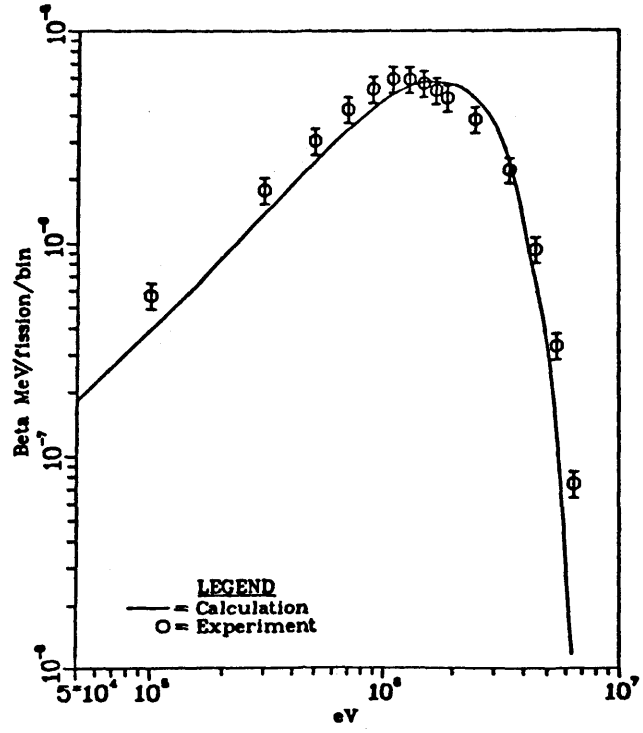

Fig.26 Comparison of calculation with UI
15 ms irradiation experiment, 66 s decay

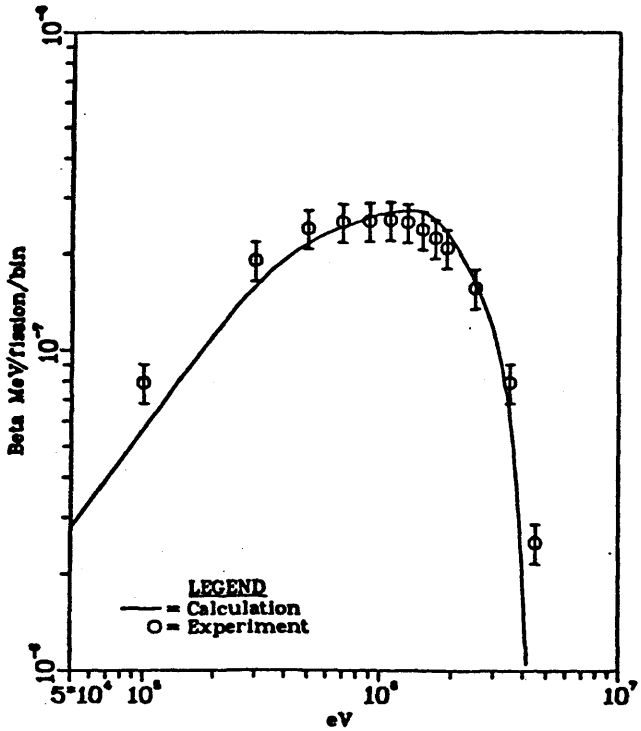

Fig 27 Comparison of calculation with UI
15 ms irradiation experiment, 960 s decay

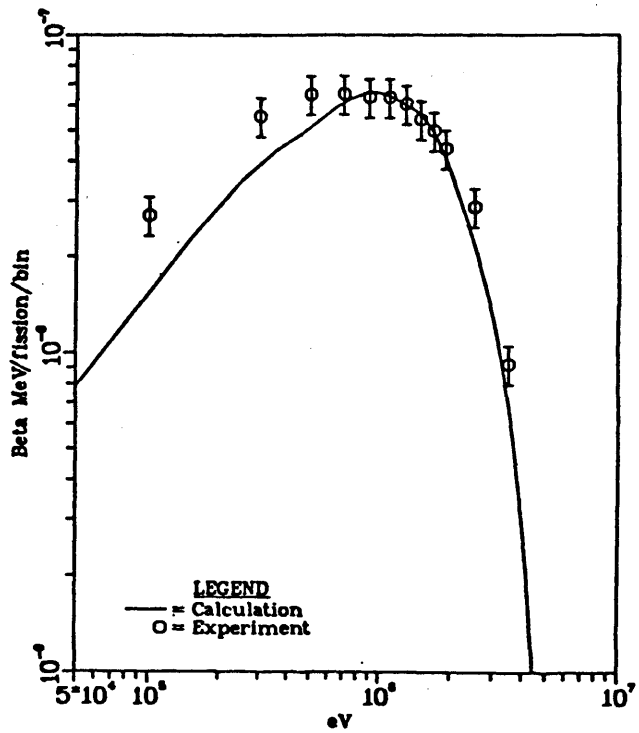

Fig. 28 Comparison of calculation with UI
15 ms irradiation experiment, 3750 s decay

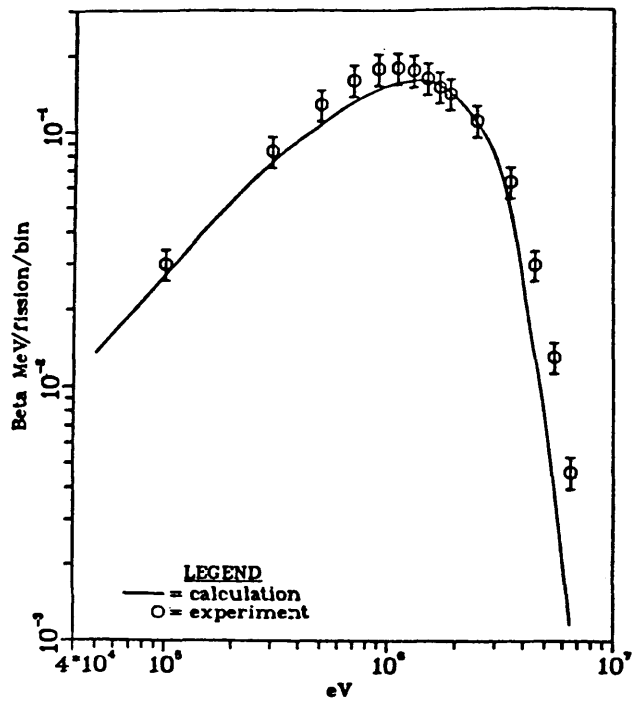

Fig. 29 Comparison of calculation with UI
8 h irradiation, 6 s decay

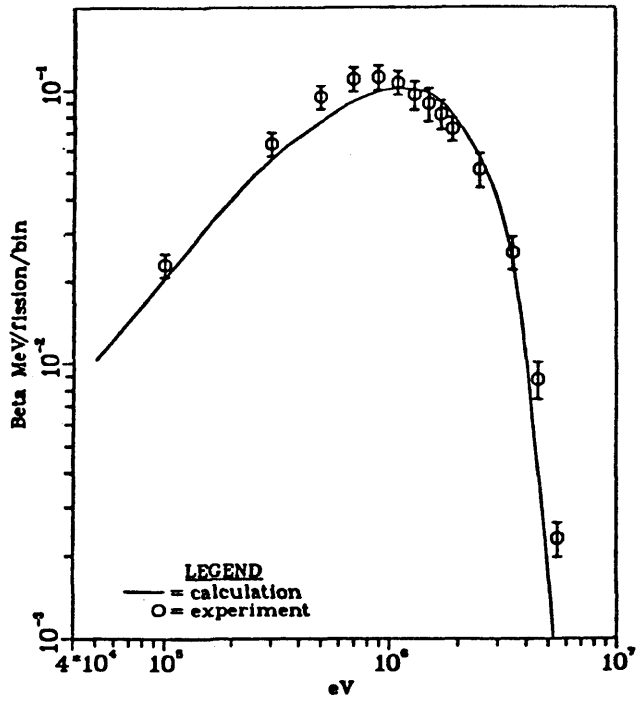

Fig. 30 Comparision of calculation with UI
8 h irradiation, 66 s decay

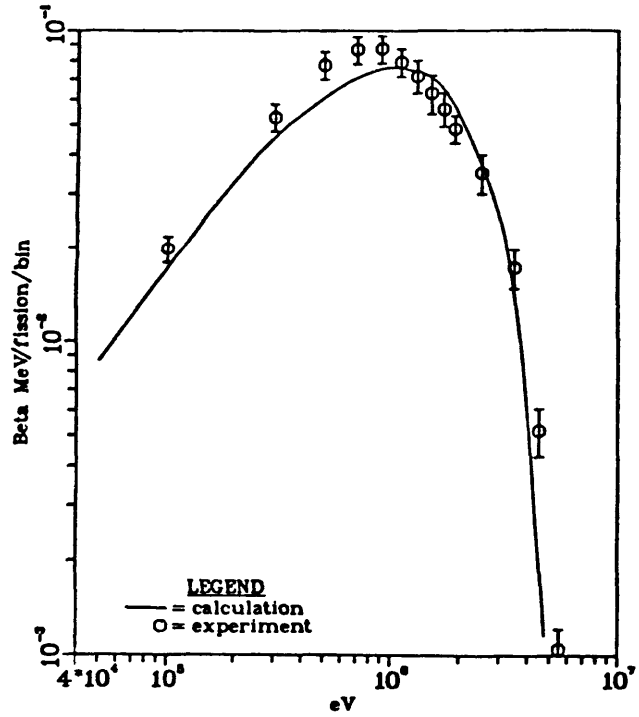

Fig. 31 Comparison of calculation with UI
8 h irradiation, 210 s decay

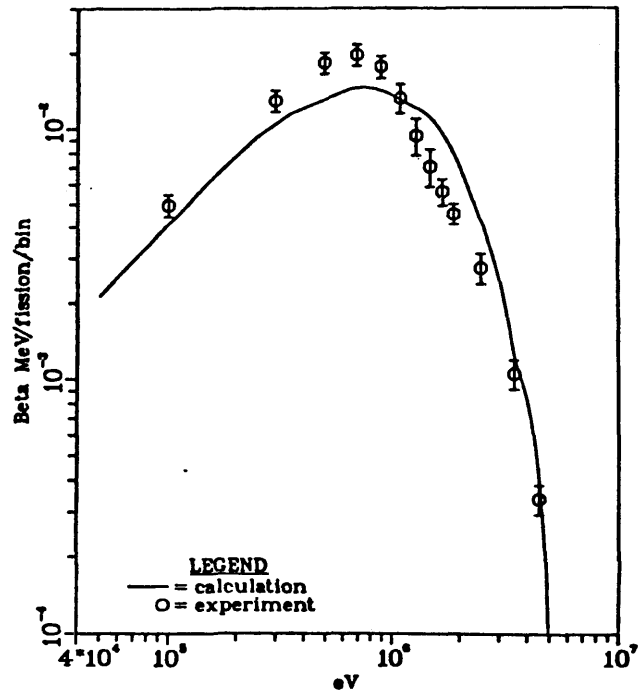

Fig. 32 Comparison of calculation with UI
8 h irradiation, 10950 s decay

Contribution to Review Paper 7

FOUR GROUP FISSION-PRODUCT CROSS SECTIONS FOR THERMAL REACTOR-APPLICATIONS

by

W. B. Wilson, T. R. England, and R. J. LaBauve

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Fission-product data are given for 824 nuclides in the ENDF/B-IV fission-product library. of these, 181 nuclides (MATs) are complete with evaluated neutron cross sections over the range 1 X 10^{-5} eV - 20 MeV. Each of these 181 nuclides have evaluated neutron cross sections for total (MT=1), elastic (MT=2), total inelastic (MT=4), and radiative capture (MT=102) reactions. Thirty-six of these have additional evaluated neutron cross sections for other absorption reactions such as (n,α), (n,p), etc. The ENDF/B-IV fission-product data library also contains radioactive decay constants, decay spectra, decay branching, and neutron absorption branching data. These parameters, including a few corrections, resonance integrals, and 2200 M/s cross sections, have been concisely listed in Ref. 1.

The evaluated cross sections of the ENDF/B-IV fission-product data library have been processed into the Power Reactor Studies (PRS) 154-Group Structure[2] using the NJOY cross-section processing code.[3] The 154-group cross sections were subsequently collapsed with the TOAFEW cross-section collapsing code[4] to the four-group structure described in Table I. The flux weighting function used in cross-section processing and subsequent collapsing is the PRS Flux Weighting Function, shown in Fig. 1. The PRS Flux Weighting Function, typical of light water reactors, is described in Table II by a set of 115 pairs of flux- and neutron-energy values between which log-log interpolation is used.

All cross sections listed in the ENDF/B-IV fission-product data library were processed at infinite dilution using σ_0 = 10^{10} b in the Bondarenko cross-section resonance self-shielding formalism.[5] Thirty-six nuclides (MATs) appearing on both the ENDF/B-IV fission-product and general-purpose data files were processed from the latter at a temperature of 900 K, using PENDF[5] files retained from previous processing. The remaining 145 MATS were processed from the fission-product files at 1000 K. Without exception, fission-product neutron absorption cross sections for reactions other than radiative capture were observed to be negligibly small and unnecessary for fission-product absorption buildup calculations.

The four-group radiative capture cross sections are listed in Table III. These cross sections have been incorporated with ENDF/B-IV fission-product decay constants, decay and absorption branching fractions, and fission yields in a library of fission-product data for the CINDER depletion and fission-product absorption code.[7]

TABLE I

FOUR CROUP STRUCTURE

Group	Energy Bound
	$1.000\ 000\ 000 \times 10^{7}$ eV
1	
	$8.208\ 499\ 862 \times 10^{5}$ eV
2	
	$5.530\ 843\ 701 \times 10^{3}$ eV
3	
	$6.250\ 600\ 000 \times 10^{-1}$ eV
4	
	$1.000\ 000\ 000 \times 10^{-5}$ eV

REFERENCES

1. T. R. England and R. E. Schenter, "ENDF/B-IV Fission-Product Files: Summary of Major Nuclide Data," Los Alamos Scientific Laboratory report LA-6116-MS (ENDF-233) (1975).

2. R. J. LaBauve and W. B. Wilson, "Proposal to Extend CSEWG Neutron and Photon Multigroup Structures for Wider Applications," Los Alamos Scientific Laboratory report LA-6240-P (1976).

3. R. E. MacFarlane and R. M. Boicourt, "NJOY - A Neutron and Photon Cross Section Processing System," Trans. Am. Nucl. Soc. 22, 720 (1975).

4. W. B. Wilson, Los Alamos Scientific Laboratory, unpublished codes for the CDC-7600. (A data library containing the 154-group processed cross sections, group structure, weighting flux, and collapsing code will soon be released by the authors.)

5. I. I. Bondarenko et al., Group Constants for Nuclear Reactor Calculations (Consultants Bureau Enterprises, Inc., N.Y., 1964).

6. R. J. LaBauve, C. R. Weisbin, R. E. Seamon, M. E. Battat, D. R. Harris, P. G. Young, and M. M. Klein, "PENDF: A Library of Nuclear Data for Monte Carlo Calculations Derived from Data in the ENDF/B Format," Los Alamos Scientific Laboratory report LA-5687 (1974).

7. T. R. England, W. B. Wilson, and M. G. Stamatelatos, "Fission-Product Data for Thermal Reactors, Part 1, A Data Set for EPRI-CINDER using ENDF/B-IV," LA-6745-MS, and "Fission Product Data for Thermal Reactors, Part 2, Users Manual for EPRI-CINDER Code and Data," LA-6745-MS (to be published by Electric Power Research Institute).

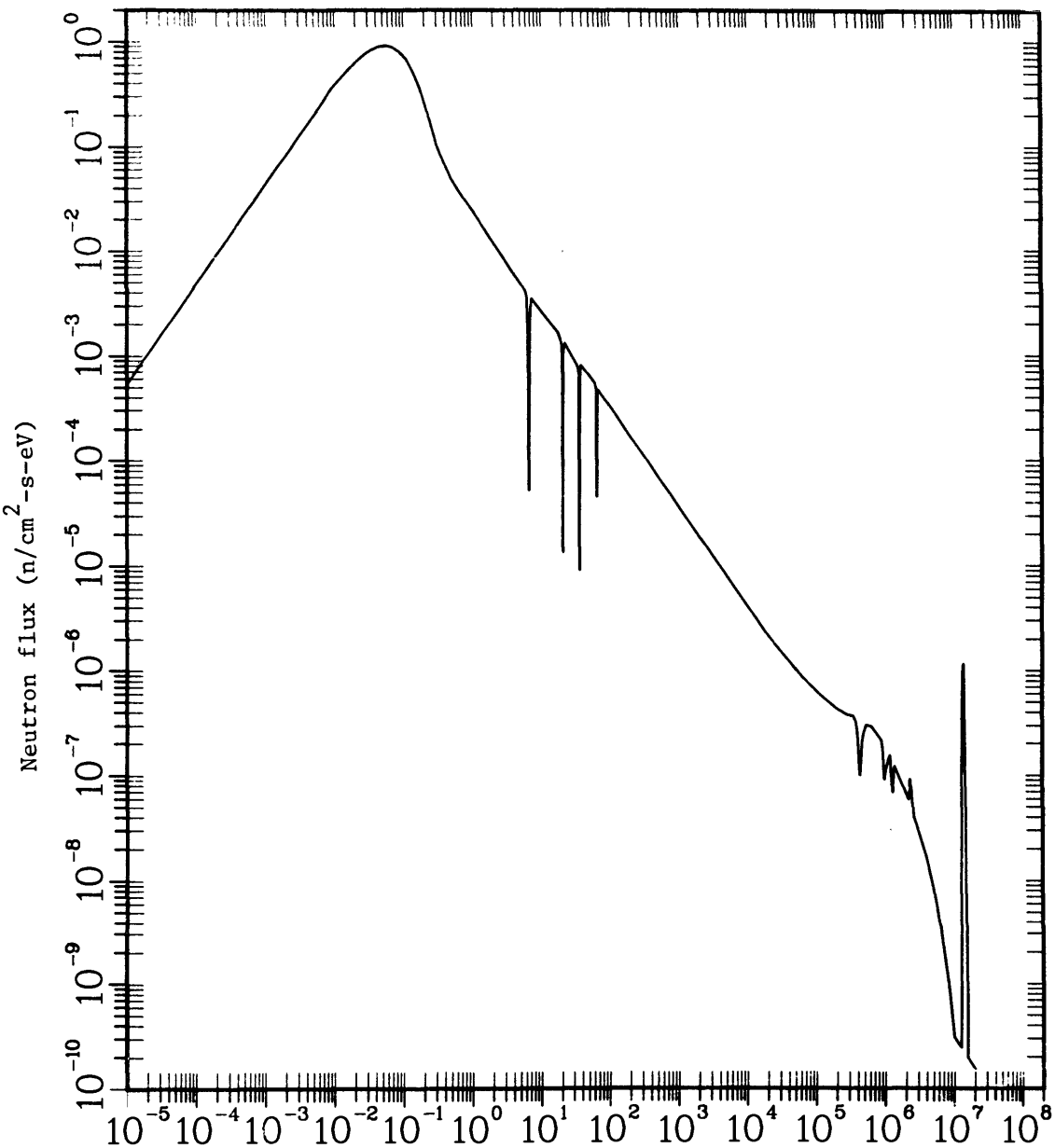

Fig. 1. Power Reactor Studies (PWR) neutron flux weighting function.

TABLE II

PRS FLUX WEIGHTING FUNCTION

POINT	ENERGY, (EV)	FLUX	POINT	ENERGY, (EV)	FLUX	POINT	ENERGY, (EV)	FLUX
1	1.0000E-05	5.2500E-04	40	3.5600E+01	7.4897E-04	78	3.0000E+06	3.1142E-08
2	9.0000E-03	3.5500E-01	41	3.5900E+01	6.7872E-04	79	4.0000E+06	1.7073E-08
3	1.6000E-02	5.5200E-01	42	3.6700E+01	9.1595E-06	80	5.0000E+06	9.0679E-09
4	2.4000E-02	7.1200E-01	43	3.7400E+01	6.5453E-04	81	6.0000E+06	4.7153E-09
5	2.9000E-02	7.8500E-01	44	3.8700E+01	8.2618E-04	82	8.0000E+06	1.2276E-09
6	3.3000E-02	8.2900E-01	45	6.1200E+01	5.5873E-04	83	1.0000E+07	3.0953E-10
7	4.3000E-02	8.9800E-01	46	6.4900E+01	4.8243E-04	84	1.2570E+07	2.4619E-10
8	5.0000E-02	9.1800E-01	47	6.6000E+01	4.5797E-05	85	1.2600E+07	3.4731E-10
9	5.4000E-02	9.2100E-01	48	6.7100E+01	4.7226E-04	86	1.2700E+07	1.0357E-09
10	5.9000E-02	9.1800E-01	49	6.8200E+01	4.8362E-04	87	1.2800E+07	2.8436E-09
11	7.0000E-02	8.9200E-01	50	1.0100E+03	3.7829E-05	88	1.2900E+07	7.1910E-09
12	9.0000E-02	7.9900E-01	51	2.0000E+04	2.2257E-06	89	1.3000E+07	1.6776E-08
13	1.1200E-01	6.8600E-01	52	3.0700E+04	1.5571E-06	90	1.3100E+07	3.6122E-08
14	1.4000E-01	5.2000E-01	53	6.0700E+04	9.1595E-07	91	1.3200E+07	7.1864E-08
15	1.7000E-01	3.8300E-01	54	1.2000E+05	5.7934E-07	92	1.3300E+07	1.3222E-07
16	2.1000E-01	2.5200E-01	55	2.0100E+05	4.3645E-07	93	1.3400E+07	2.2511E-07
17	3.0000E-01	1.0800E-01	56	2.8300E+05	3.8309E-07	94	1.3500E+07	3.5512E-07
18	4.0000E-01	6.8700E-02	57	3.5600E+05	3.6926E-07	95	1.3600E+07	5.1946E-07
19	4.9000E-01	5.1000E-02	58	3.7700E+05	3.4027E-07	96	1.3700E+07	7.0478E-07
20	5.7000E-01	4.3700E-02	59	3.9900E+05	2.7387E-07	97	1.3800E+07	8.8825E-07
21	6.0000E-01	4.1300E-02	60	4.4200E+05	1.0075E-07	98	1.3900E+07	1.0408E-06
22	1.0000E+00	2.4914E-02	61	4.7400E+05	2.1754E-07	99	1.4070E+07	1.1540E-06
23	1.3518E+00	1.8502E-02	62	5.0200E+05	2.6333E-07	100	1.4200E+07	1.0870E-06
24	4.0100E+00	6.3200E-03	63	5.4000E+05	3.0501E-07	101	1.4300E+07	9.5757E-07
25	5.5047E+00	4.6164E-03	64	6.5000E+05	2.9493E-07	102	1.4400E+07	7.8704E-07
26	5.8842E+00	4.1950E-03	65	7.7000E+05	2.5005E-07	103	1.4500E+07	6.0403E-07
27	6.1350E+00	3.7279E-03	66	9.0000E+05	2.1479E-07	104	1.4600E+07	4.3317E-07
28	6.4490E+00	1.6524E-03	67	9.4100E+05	1.7861E-07	105	1.4700E+07	2.9041E-07
29	6.6700E+00	5.3125E-05	68	1.0000E+06	9.1595E-08	106	1.4800E+07	1.8213E-07
30	6.8940E+00	1.7632E-03	69	1.0500E+06	1.1518E-07	107	1.4900E+07	1.0699E-07
31	7.0100E+00	2.9219E-03	70	1.1200E+06	1.3648E-07	108	1.5000E+07	5.8832E-08
32	7.3080E+00	3.6042E-03	71	1.1900E+06	1.5479E-07	109	1.5100E+07	3.0354E-08
33	1.7530E+01	1.7156E-03	72	1.2100E+06	1.5022E-07	110	1.5200E+07	1.4687E-08
34	1.9860E+01	1.3858E-03	73	1.3100E+06	6.8696E-08	111	1.5300E+07	6.6688E-09
35	2.0370E+01	1.0973E-03	74	1.4000E+06	1.2182E-07	112	1.5400E+07	2.8450E-09
36	2.0900E+01	1.3739E-05	75	2.2200E+06	5.9033E-08	113	1.5500E+07	1.1406E-09
37	2.1400E+01	1.0588E-03	76	2.3500E+06	9.1595E-08	114	1.5676E+07	1.9780E-10
38	2.2500E+01	1.3565E-03	77	2.6300E+06	3.9981E-08	115	2.0000E+07	1.5477E-10
39	3.4400E+01	8.1519E-04						

FLUX VALUES ARE GIVEN IN UNITS OF NEUTRONS/ CM**2 SEC. EV.

TABLE III

FOUR GROUP (N,GAMMA) CROSS SECTIONS

NUCLIDE	Z	A	S	MAT	TEMP DEG.KEL	GROUP 1 CROSS SECTION	GROUP 2 CROSS SECTION	GROUP 3 CROSS SECTION	GROUP 4 CROSS SECTION	EFFEC.THERM. CROSS SECTION
GE 72	32	72	0	48	1.0E+03	2.099716E-02	5.432289E-02	8.043696E-02	5.476937E-01	9.885840E-01
GE 73	32	73	0	49	1.0E+03	4.379093E-02	1.769910E-01	8.051782E+00	8.381628E+00	1.512879E+01
GE 74	32	74	0	51	1.0E+03	9.484258E-03	3.185640E-02	4.402824E-02	2.141804E-01	3.865945E-01
GE 76	32	76	0	54	1.0E+03	2.668907E-03	9.284226E-03	1.632492E-01	7.941654E-02	1.433464E-01
AS 75	33	75	0	68	1.0E+03	3.174023E-02	2.399267E-01	7.066404E+00	2.408837E+00	4.347936E+00
SE 76	34	76	0	85	1.0E+03	3.043571E-02	1.025647E-01	3.876426E+00	4.754810E+01	8.582406E+01
SE 77	34	77	0	86	1.0E+03	2.397912E-02	2.570206E-01	3.571884E+00	2.346095E+01	4.234688E+01
SE 78	34	78	0	88	1.0E+03	1.233979E-02	4.559968E-02	5.350889E-01	2.234162E-01	4.032650E-01
SE 80	34	80	0	91	1.0E+03	1.224080E-02	4.557911E-02	8.982662E-02	3.412236E-01	6.159067E-01
SE 82	34	82	0	94	1.0E+03	3.747510E-03	1.079513E-02	1.686126E-03	2.513117E-02	4.536163E-02
BR 79	35	79	0	108	1.0E+03	6.182080E-02	4.360678E-01	1.200472E+01	6.222752E+00	1.123203E+01
BR 81	35	81	0	112	1.0E+03	4.217740E-02	2.809681E-01	5.719193E+00	1.503798E+00	2.714348E+00
KR 80	36	80	0	131	1.0E+03	3.268930E-02	1.358841E-01	6.958419E+00	7.974773E+00	1.439442E+01
KR 82	36	82	0	134	1.0E+03	3.209819E-02	1.243721E-01	1.984202E+01	1.685921E+01	3.043078E+01
KR 83	36	83	0	135	1.0E+03	6.710060E-02	3.993954E-01	1.841473E+01	1.119001E+02	2.019790E+02
KR 84	36	84	0	137	1.0E+03	4.472712E-03	1.912670E-02	4.376474E-01	4.633101E-02	8.362721E-02
KR 85	36	85	0	138	1.0E+03	4.363354E-03	2.864559E-02	1.600506E-01	9.295652E-01	1.677860E+00
KR 86	36	86	0	140	1.0E+03	1.108820E-03	3.162755E-03	1.545394E-02	3.441791E-02	6.212414E-02
RB 85	37	85	0	153	1.0E+03	2.080914E-02	2.157803E-01	5.395468E-01	2.576161E-01	4.649955E-01
RB 86	37	86	0	154	1.0E+03	4.296598E-03	9.237312E-02	2.813636E+00	2.735252E+00	4.937115E+00
RB 87	37	87	0	156	1.0E+03	1.287127E-03	8.266649E-03	2.536605E-01	6.710316E-02	1.211208E-01
SR 86	38	86	0	172	1.0E+03	1.688432E-02	4.158945E-02	5.748211E-01	1.588873E+00	2.867907E+00
SR 87	38	87	0	173	1.0E+03	1.669480E-02	7.775602E-02	1.045327E+01	9.270864E+00	1.673386E+01
SR 88	38	88	0	175	1.0E+03	1.329117E-03	1.164227E-03	2.201289E-04	3.248619E-03	5.863738E-03
SR 89	38	89	0	176	1.0E+03	8.295338E-03	2.038847E-02	3.876896E-02	2.348515E-01	4.239056E-01
SR 90	38	90	0	177	1.0E+03	4.920037E-03	1.560162E-02	3.315065E-02	5.026414E-01	9.072648E-01
Y 89	39	89	0	192	1.0E+03	7.136131E-03	2.253723E-02	8.228018E-02	7.165644E-01	1.293395E+00
Y 90	39	90	0	194	1.0E+03	1.138581E-02	1.146555E-01	4.323381E-01	1.954493E+00	3.527849E+00
Y 91	39	91	0	196	1.0E+03	4.217073E-03	3.284697E-02	1.639328E-01	7.827059E-01	1.412780E+00
ZR 90	40	90	0	215	1.0E+03	1.508096E-02	2.520652E-02	1.987868E-02	5.586877E-02	1.008428E-01
ZR 91	40	91	0	217	1.0E+03	1.715130E-02	5.507396E-02	6.700606E-01	5.758814E-01	1.039463E+00
ZR 92	40	92	0	218	1.0E+03	1.042367E-02	3.130879E-02	8.308115E-02	1.452876E-01	2.622433E-01
ZR 93	40	93	0	219	1.0E+03	2.089521E-02	5.150402E-02	3.654891E+00	1.396955E+00	2.521495E+00
ZR 94	40	94	0	220	1.0E+03	5.447404E-03	2.595021E-02	2.019570E-02	3.128906E-02	5.647657E-02
ZR 95	40	95	0	221	1.0E+03	3.771503E-02	1.168058E-01	5.797035E-01	2.741127E-01	4.947719E-01
ZR 96	40	96	0	222	1.0E+03	2.648158E-02	2.165040E-02	6.357653E-01	8.862928E-03	1.599754E-02
NB 93	41	93	0	1189	9.0E+02	1.604464E-02	1.581963E-01	1.036643E+00	6.392347E-01	1.153815E+00
NB 94	41	94	0	238	1.0E+03	5.739899E-03	1.688129E-01	1.173435E+01	7.640611E+00	1.379126E+01
NB 95	41	95	0	240	1.0E+03	5.266054E-02	2.650329E-01	2.503124E+00	8.380113E-01	1.512606E+00
MO 94	42	94	0	264	1.0E+03	1.622583E-02	3.166593E-02	9.483320E-02	8.925915E-03	1.611123E-02
MO 95	42	95	0	265	1.0E+03	3.417280E-02	2.320601E-01	1.257062E+01	8.099988E+00	1.462043E+01
MO 96	42	96	0	266	1.0E+03	1.622694E-02	4.895069E-02	2.259064E+00	5.590317E-01	1.009049E+00

NUCLIDE	Z	A	S	MAT	TEMP DEG.KEL	GROUP 1 CROSS SECTION	GROUP 2 CROSS SECTION	GROUP 3 CROSS SECTION	GROUP 4 CROSS SECTION	EFFEC.THERM. CROSS SECTION
MO 97	42	97	0	267	1.0E+03	2.907559E-02	2.248807E-01	1.719731E+00	1.215824E+00	2.194556E+00
MO 98	42	98	0	268	1.0E+03	2.516355E-02	7.168378E-02	7.678803E-01	7.125803E-02	1.286203E-01
MO 99	42	99	0	269	1.0E+03	4.286153E-02	4.063070E-01	2.849176E+00	9.492203E-01	1.713337E+00
MO100	42	100	0	270	1.0E+03	9.074271E-03	5.494819E-02	4.423330E-01	1.111105E-01	2.005538E-01
TC 99	43	99	0	1137	9.0E+02	3.673952E-02	3.614800E-01	3.069278E+01	1.083246E+01	1.955253E+01
RU 99	44	99	0	308	1.0E+03	3.826665E-02	3.753455E-01	1.421606E+01	2.833373E+00	5.114222E+00
RU100	44	100	0	309	1.0E+03	3.858774E-02	1.136596E-01	1.132016E+00	3.239899E+00	5.847999E+00
RU101	44	101	0	310	1.0E+03	2.690347E-02	4.480972E-01	9.421788E+00	1.743797E+00	3.147543E+00
RU102	44	102	0	311	1.0E+03	6.954545E-02	1.577507E-01	3.499625E-01	7.279318E-01	1.313913E+00
RU103	44	103	0	312	1.0E+03	2.468305E-02	3.671989E-01	7.404899E+00	4.301005E+00	7.763290E+00
RU104	44	104	0	313	1.0E+03	3.070218E-02	1.059028E-01	7.196752E-01	2.440178E-01	4.404508E-01
RU105	44	105	0	314	1.0E+03	2.325016E-02	3.233188E-01	6.399468E-01	1.117743E-01	2.017520E-01
RU106	44	106	0	315	1.0E+03	8.344953E-03	5.233637E-02	2.475373E-01	8.153410E-02	1.471686E-01
RH103	45	103	0	1125	9.0E+02	4.648621E-02	5.571160E-01	9.047802E+01	9.701270E+01	1.751074E+02
RH105	45	105	0	334	1.0E+03	9.125643E-02	4.919683E-01	7.002262E+02	4.934551E+03	8.906837E+03
PD104	46	104	0	358	1.0E+03	8.530149E-02	2.169388E-01	2.016123E+00	2.166387E-01	3.910315E-01
PD105	46	105	0	359	1.0E+03	8.900761E-02	7.121473E-01	9.287985E+00	7.843382E+00	1.415726E+01
PD106	46	106	0	360	1.0E+03	4.789705E-02	1.283831E-01	7.720379E-01	1.332195E-01	2.404604E-01
PD107	46	107	0	361	1.0E+03	6.726203E-02	4.894277E-01	7.143654E+00	5.588543E+00	1.008729E+01
PD108	46	108	0	363	1.0E+03	4.149806E-02	1.255796E-01	2.259456E+01	6.845444E+00	1.235599E+01
PD110	46	110	0	366	1.0E+03	1.580939E-02	7.311286E-02	8.358860E-01	1.229825E-01	2.219828E-01
AG107	47	107	0	1138	9.0E+02	8.334117E-02	5.664923E-01	1.210986E+01	2.047736E+01	3.696153E+01
AG109	47	109	0	1139	9.0E+02	4.458553E-02	3.535820E-01	1.302363E+02	5.257063E+01	9.488969E+01
AG111	47	111	0	391	1.0E+03	2.611568E-02	3.334252E-01	1.231271E+01	1.677295E+00	3.027507E+00
CD108	48	108	0	415	1.0E+03	9.591343E-02	1.805074E-01	3.702015E-01	6.153806E-01	1.110758E+00
CD110	48	110	0	417	1.0E+03	1.158785E-01	2.033491E-01	4.663890E+00	6.207640E+00	1.120475E+01
CD111	48	111	0	418	1.0E+03	3.549300E-02	3.246907E-01	5.577368E+00	1.357310E+01	2.449936E+01
CD112	48	112	0	420	1.0E+03	9.319133E-02	1.924481E-01	1.147311E+00	1.230079E+00	2.220285E+00
CD113	48	113	0	1282	9.0E+02	4.035980E-02	2.936839E-01	2.050547E+01	2.727591E+04	4.923287E+04
CD114	48	114	0	423	1.0E+03	9.644373E-02	2.058760E-01	2.208012E+00	1.877895E-01	3.389589E-01
CD115M	48	115	1	425	1.0E+03	3.428619E-02	2.347852E-01	2.226469E+01	1.732369E+01	3.126916E+01
CD116	48	116	0	426	1.0E+03	4.177477E-02	9.279391E-02	2.235261E-01	4.293487E-02	7.749719E-02
IN113	49	113	0	445	1.0E+03	2.409397E-01	5.236885E-01	2.162369E+01	6.590786E+00	1.189633E+01
IN115	49	115	0	449	1.0E+03	1.228192E-01	3.755484E-01	2.854464E+02	1.289674E+02	2.327855E+02
SN115	50	115	0	482	1.0E+03	3.729064E-03	3.183642E-02	2.008304E+00	2.796581E+01	5.047812E+01
SN116	50	116	0	483	1.0E+03	3.175388E-02	4.364059E-02	1.356454E+00	6.732990E-02	1.215301E-01
SN117	50	117	0	484	1.0E+03	3.274490E-02	1.661797E-01	1.984943E+00	1.455753E+00	2.627627E+00
SN118	50	118	0	486	1.0E+03	7.690040E-02	8.667272E-02	6.911670E-01	4.610512E-02	8.321948E-02
SN119	50	119	0	487	1.0E+03	6.371977E-03	4.040596E-02	3.974212E-01	1.284014E+00	2.317637E+00
SN120	50	120	0	489	1.0E+03	1.800320E-02	2.933619E-02	1.337315E-01	7.881781E-02	1.422657E-01
SN122	50	122	0	492	1.0E+03	1.337318E-02	1.827124E-02	7.595520E-02	1.010517E-01	1.823978E-01
SN123	50	123	0	493	1.0E+03	3.947307E-02	8.523844E-02	2.774607E-01	1.845241E-02	3.330650E-02
SN124	50	124	0	495	1.0E+03	1.849913E-02	2.415400E-02	8.102987E-01	7.272250E-02	1.312637E-01
SN125	50	125	0	496	1.0E+03	9.919385E-03	4.686948E-02	1.984130E+00	3.074827E-01	5.550044E-01
SN126	50	126	0	498	1.0E+03	5.678309E-03	7.303296E-03	1.098527E-02	1.677718E-01	3.028272E-01
SB121	51	121	0	511	1.0E+03	1.106325E-01	4.009508E-01	1.677825E+01	3.560680E+00	6.427007E+00
SB123	51	123	0	514	1.0E+03	5.059958E-02	2.100477E-01	8.328224E+00	2.434903E+00	4.394987E+00
SB124	51	124	0	515	1.0E+03	2.543705E-02	5.863286E-01	2.475458E+00	3.630309E+00	6.552687E+00

NUCLIDE	Z	A	S	MAT	TEMP DEG.KEL	GROUP 1 CROSS SECTION	GROUP 2 CROSS SECTION	GROUP 3 CROSS SECTION	GROUP 4 CROSS SECTION	EFFEC.THERM. CROSS SECTION
SB125	51	125	0	518	1.0E+03	7.517359E-02	2.103150E-01	2.103414E+00	5.592590E-01	1.009459E+00
SB126	51	126	0	519	1.0E+03	1.848180E-02	3.033790E-01	5.136429E+00	3.247778E+00	5.862222E+00
TE122	52	122	0	538	1.0E+03	1.202321E-01	2.504961E-01	8.412152E+00	1.569423E+00	2.832800E+00
TE123	52	123	0	539	1.0E+03	6.157976E-02	3.666051E-01	4.865476E+02	2.468661E+02	4.455918E+02
TE124	52	124	0	541	1.0E+03	1.104859E-01	1.937552E-01	7.708467E-01	3.808186E+00	6.873755E+00
TE125	52	125	0	542	1.0E+03	2.613155E-02	2.557467E-01	2.647459E+00	8.658353E-01	1.562828E+00
TE126	52	126	0	544	1.0E+03	3.714886E-02	8.357152E-02	1.185596E+00	5.779812E-01	1.043253E+00
TE127M	52	127	1	546	1.0E+03	5.419348E-02	2.648949E-01	4.726406E+00	5.264135E+00	9.501734E+00
TE128	52	128	0	547	1.0E+03	4.529906E-02	8.662364E-02	2.271824E-01	1.202259E-01	2.170070E-01
TE129M	52	129	1	549	1.0E+03	9.472597E-03	8.205984E-02	6.885660E-01	6.142364E-01	1.108693E+00
TE130	52	130	0	550	1.0E+03	3.760774E-03	1.329507E-02	2.826283E-02	1.621755E-01	2.927259E-01
TE132	52	132	0	553	1.0E+03	5.710111E-04	4.443987E-04	7.327881E-05	1.117629E-03	2.017314E-03
I127	53	127	0	565	1.0E+03	5.610752E-02	3.939301E-01	1.539254E+01	3.478504E+00	6.278681E+00
I129	53	129	0	567	1.0E+03	4.800948E-02	2.804701E-01	3.709139E+00	1.510053E+01	2.725637E+01
I130	53	130	0	568	1.0E+03	1.837486E-02	2.696179E-01	1.985659E+01	1.004999E+01	1.814018E+01
I131	53	131	0	570	1.0E+03	9.472026E-03	7.367629E-02	9.685281E-01	3.911037E-01	7.059399E-01
I135	53	135	0	576	1.0E+03	4.684394E-04	5.736313E-04	7.324264E-04	1.118905E-02	2.019617E-02
XE128	54	128	0	1173	9.0E+02	5.312305E-02	2.997089E-01	2.526227E+01	9.969592E+00	1.799506E+01
XE129	54	129	0	589	1.0E+03	5.323277E-02	3.033197E-01	2.522329E+01	1.003302E+01	1.810954E+01
XE130	54	130	0	1174	9.0E+02	3.404525E-02	1.127239E-01	3.154451E-01	3.446107E+00	6.220203E+00
XE131	54	131	0	1175	9.0E+02	1.332154E-02	1.837029E-01	8.832394E+01	5.042924E+01	9.102450E+01
XE132	54	132	0	1176	9.0E+02	1.814070E-02	6.651676E-02	1.568604E-01	2.501073E-01	4.514423E-01
XE133	54	133	0	595	1.0E+03	5.950709E-03	5.286344E-02	3.435562E+01	1.062848E+02	1.918434E+02
XE134	54	134	0	1177	9.0E+02	1.054011E-02	3.432525E-02	3.981692E-02	1.434189E-01	2.588703E-01
XE135	54	135	0	1294	9.0E+02	5.564618E-04	3.218521E-03	3.634285E+02	1.670657E+06	3.015526E+06
XE136	54	136	0	1178	9.0E+02	1.072229E-03	2.288886E-03	1.069663E-02	8.874087E-02	1.601768E-01
CS133	55	133	0	1141	9.0E+02	5.103384E-02	3.777516E-01	3.378401E+01	1.663776E+01	3.003107E+01
CS134	55	134	0	614	1.0E+03	1.286883E-02	3.448577E-01	2.079022E+01	7.828318E+01	1.413007E+02
CS135	55	135	0	616	1.0E+03	3.180215E-03	4.148685E-02	4.381765E+00	4.866696E+00	8.784359E+00
CS136	55	136	0	618	1.0E+03	1.942282E-02	1.635070E-01	3.303652E+00	7.287809E-01	1.315445E+00
CS137	55	137	0	619	1.0E+03	1.502562E-03	5.543158E-03	5.867613E-02	6.142019E-02	1.108631E-01
BA134	56	134	0	634	1.0E+03	3.524396E-02	8.612855E-02	2.724576E+00	1.205309E+00	2.175576E+00
BA135	56	135	0	635	1.0E+03	2.867491E-02	2.214519E-01	1.119982E+01	3.250888E+00	5.867834E+00
BA136	56	136	0	637	1.0E+03	1.312605E-02	3.583927E-02	2.133971E-01	2.289892E-01	4.133242E-01
BA137	56	137	0	639	1.0E+03	7.225130E-03	3.743974E-02	4.969716E-01	2.849852E+00	5.143966E+00
BA138	56	138	0	641	1.0E+03	9.214468E-03	5.529264E-03	1.282347E-02	1.955539E-01	3.529736E-01
BA140	56	140	0	643	1.0E+03	1.689069E-03	3.743996E-03	1.567531E+00	8.912482E-01	1.608698E+00
LA139	57	139	0	657	1.0E+03	6.009337E-03	2.967405E-02	1.244460E+00	5.013105E+00	9.048625E+00
LA140	57	140	0	658	1.0E+03	2.948971E-02	1.295762E-01	7.885135E+00	1.508462E+00	2.722765E+00
CE140	58	140	0	674	1.0E+03	1.703531E-02	1.730715E-02	2.903654E-02	3.190750E-01	5.759285E-01
CE141	58	141	0	675	1.0E+03	4.510816E-02	1.054578E-01	2.342997E+00	1.619745E+01	2.923631E+01
CE142	58	142	0	676	1.0E+03	1.485239E-02	3.294434E-02	5.925282E-02	5.307171E-01	9.579413E-01
CE143	58	143	0	677	1.0E+03	9.577707E-03	7.577132E-02	5.095466E+00	3.349870E+00	6.046496E+00
CE144	58	144	0	678	1.0E+03	1.090923E-02	2.405806E-02	2.314846E-01	5.586255E-01	1.008316E+00
PR141	59	141	0	692	1.0E+03	2.548600E-02	9.959114E-02	2.105239E+00	6.439460E+00	1.162319E+01
PR142	59	142	0	693	1.0E+03	1.604911E-02	2.547188E-01	1.527271E+01	1.117627E+01	2.017310E+01
PR143	59	143	0	695	1.0E+03	4.842014E-02	2.161496E-01	1.913981E+01	4.977789E+01	8.984881E+01
ND142	60	142	0	713	1.0E+03	2.773231E-02	3.127668E-02	7.312854E-01	1.044265E+01	1.884892E+01

NUCLIDE	Z	A	S	MAT	TEMP DEG.KEL	GROUP 1 CROSS SECTION	GROUP 2 CROSS SECTION	GROUP 3 CROSS SECTION	GROUP 4 CROSS SECTION	EFFEC.THERM. CROSS SECTION
ND143	60	143	0	714	1.0E+03	5.923333E-02	1.938598E-01	1.209111E+01	1.773106E+02	3.200445E+02
ND144	60	144	0	715	1.0E+03	4.188978E-02	6.863372E-02	5.867006E-01	2.013982E+00	3.635226E+00
ND145	60	145	0	716	1.0E+03	6.803414E-02	1.983824E-01	2.256161E+01	2.355985E+01	4.252540E+01
ND146	60	146	0	717	1.0E+03	5.710881E-02	1.082823E-01	2.963151E-01	7.827704E-01	1.412896E+00
ND147	60	147	0	718	1.0E+03	5.106341E-02	3.928675E-01	6.013407E+01	2.740049E+01	4.945773E+01
ND148	60	148	0	719	1.0E+03	1.435365E-01	1.318842E-01	2.284639E+00	1.399507E+00	2.526102E+00
ND150	60	150	0	721	1.0E+03	5.230297E-02	1.742186E-01	1.865556E+00	6.713311E-01	1.211749E+00
PM147	61	147	0	733	1.0E+03	1.382808E-01	8.170927E-01	1.918335E+02	9.917021E+01	1.790017E+02
PM148	61	148	0	734	1.0E+03	4.921928E-01	2.574234E+00	2.332976E+03	2.594150E+03	4.682426E+03
PM148M	61	148	1	735	1.0E+03	4.921928E-01	2.574234E+00	3.182157E+02	1.909160E+04	3.446023E+04
PM149	61	149	0	736	1.0E+03	5.268164E-01	2.133433E+00	7.332632E+01	7.811904E+02	1.410044E+03
PM151	61	151	0	738	1.0E+03	1.012214E-03	2.037358E-02	1.742050E+02	3.910634E+02	7.058672E+02
SM147	62	147	0	753	1.0E+03	1.180276E-01	5.432755E-01	7.514796E+01	3.454724E+01	6.235757E+01
SM148	62	148	0	754	1.0E+03	2.023254E-01	2.596780E-01	3.134887E+00	1.509450E+00	2.724549E+00
SM149	62	149	0	1027	9.0E+02	5.637011E-02	9.344686E-01	2.770684E+02	4.841072E+04	8.738108E+04
SM150	62	150	0	756	1.0E+03	1.313093E-01	2.862238E-01	1.160902E+01	5.471646E+01	9.876290E+01
SM151	62	151	0	757	1.0E+03	2.315641E-01	1.356167E+00	2.894601E+02	4.669738E+03	8.428850E+03
SM152	62	152	0	758	1.0E+03	7.520442E-02	2.877061E-01	2.755117E+02	1.173567E+02	2.118281E+02
SM153	62	153	0	759	1.0E+03	2.260670E-04	1.191724E-02	2.418654E+02	1.846696E+02	3.333276E+02
SM154	62	154	0	760	1.0E+03	5.706727E-02	1.425114E-01	3.851160E+00	3.076205E+00	5.552533E+00
EU151	63	151	0	1290	9.0E+02	4.181577E-01	2.402754E+00	2.044278E+02	4.717469E+03	8.515005E+03
EU152	63	152	0	1292	9.0E+02	1.744222E-01	3.245252E+00	3.252586E+02	7.677840E+02	1.385846E+03
EU153	63	153	0	1291	9.0E+02	2.431446E-01	1.539698E+00	1.478631E+02	2.197333E+02	3.966174E+02
EU154	63	154	0	1293	9.0E+02	7.241066E-02	1.875663E+00	2.228416E+02	4.972313E+02	8.974997E+02
EU155	63	155	0	778	1.0E+03	9.062116E-01	1.543263E+00	1.540107E+02	2.258836E+03	4.077187E+03
EU156	63	156	0	779	1.0E+03	5.008924E-04	3.539738E-02	1.016701E+02	2.696344E+02	4.866885E+02
EU157	63	157	0	780	1.0E+03	3.221641E-03	2.526199E-02	8.471598E+01	1.059332E+02	1.912089E+02
GD154	64	154	0	791	1.0E+03	3.041554E-01	6.280531E-01	2.531751E+01	4.755081E+01	8.582895E+01
GD155	64	155	0	792	1.0E+03	1.251320E-01	1.467746E+00	1.313204E+02	1.920502E+04	3.466494E+04
GD156	64	156	0	793	1.0E+03	1.177736E-01	2.925910E-01	1.328447E+01	8.292483E-01	1.496788E+00
GD157	64	157	0	794	1.0E+03	1.068337E-01	7.103253E-01	8.871128E+01	8.261695E+04	1.491231E+05
GD158	64	158	0	795	1.0E+03	4.647452E-02	1.882511E-01	6.519249E+00	1.403779E+00	2.533814E+00
GD160	64	160	0	797	1.0E+03	8.083934E-02	1.503781E-01	9.311768E-01	4.300564E-01	7.762493E-01
TB159	65	159	0	803	1.0E+03	9.138282E-02	7.922879E-01	4.634842E+01	1.437843E+01	2.595299E+01
TB160	65	160	0	804	1.0E+03	1.434563E-03	7.845290E-02	8.018238E+01	2.928139E+02	5.285275E+02
DY160	66	160	0	811	1.0E+03	7.939346E-01	1.200462E+00	1.508596E+02	3.609387E+01	6.514924E+01
DY161	66	161	0	812	1.0E+03	5.945760E-02	1.227134E+00	1.166236E+02	3.112586E+02	5.618201E+02
DY162	66	162	0	813	1.0E+03	2.803023E-01	5.297747E-01	2.471440E+02	1.147570E+02	2.071357E+02
DY163	66	163	0	814	1.0E+03	5.480995E-02	5.190151E-01	1.336405E+02	7.776202E+01	1.403600E+02
DY164	66	164	0	1031	9.0E+02	9.006775E-02	1.905334E-01	2.300536E+01	1.300572E+03	2.347525E+03
HO165	67	165	0	820	1.0E+03	8.922674E-02	8.753142E-01	7.475623E+01	3.753369E+01	6.774810E+01
ER166	68	166	0	823	1.0E+03	1.431152E-01	2.983558E-01	1.481533E+01	1.954315E+01	3.527527E+01
ER167	68	167	0	824	1.0E+03	6.106636E-02	8.024943E-01	1.590069E+02	1.101003E+03	1.987304E+03

NOTE THAT THE EFFECTIVE THERMAL CROSS SECTION IS THE GROUP 4 CROSS SECTION DIVIDED BY SIGMA(1/V),
WHERE SIGMA(1/V) IS THE GROUP 4 VALUE OF A 1/V CROSS SECTION EQUAL TO UNITY AT 2200 M/S (.554018)

Contribution to Review Paper 7

EMPIRICAL CRITERIA AND THEORETICAL BACKGROUND IN THE METHODOLOGY OF DETERMINING SETS OF CONSISTENT PARAMETERS FOR NEUTRON CROSS SECTION CALCULATIONS FAILING EXPERIMENTAL INFORMATION

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Abstract

The problems reviewed are those connected with the determination of sets of consistent parameters for neutron cross section calculations with emphasis on the analysis of the systematic behaviour of relevant parameters.

INTRODUCTION

As a result of the close cooperation between CEA and CNEN a common methodology is being developed, while joining forces allow each side to go deeper into the various subjects.

Within this framework it has been suggested to outline what has been done at Bologna on the use of models, as the keystone to a methodology aimed at determining consistent sets of parameters when available experimental information is poor.

1. LEVEL DENSITY PARAMETRIZATION

Current level density descriptions are based on the Fermi gas hypothesis. According to this hypothesis the level density parameter "a" is given by the single particle state density at the Fermi level and therefore it is expected to be independent of model formulation and constant with excitation energy.

In practice "a" is determined from mean spacing ($\bar{D}_{OBS}$) of observed neutron resonances and it turns out to depend on the formula adopted, the conventions adopted for the determination of effective excitations and the assumed spin distribution.

Different values of "a" due to different assumptions imply differing behaviours of the predicted level density at increasing excitations.

In what follows the composite level density elaboration by Gilbert and Cameron |1| has been adopted.

1.1. SYSTEMATIC BEHAVIOUR OF "a"

An overall and a local systematic behaviour of "a" can be considered.

The overall systematic (O.S.) is based on the assumption that, in medium and heavy nuclei, the contributions to the density of single particle states at the Fermi level come essentially from neutron states, because of neutron excess.

In fig. 1 "a" values in MeV^{-1} are plotted vs neutron number N .

The dips around magic numbers are typical shell closure effects. The plotted curve fits experimental data to within 15%. A comparable uncertainty may be estimated for unknown "a" values determined by the use of O.S. -

Uncertainty in the determination of unknown mean level spacings $\bar{D}(B)$ at the neutron binding by use of "a" values from O.S., due to the level density exponential law may rise well beyond 50%, depending on the value of effective excitations. Now, greater accuracy than this is needed, which suggests adopting a more theoretical approach.

Single particle states calculated by means of the Nilsson model have been used |2| for the determination of "a" as a sum of neutron and proton contributions and as a function of deformations.

About 30% of the $\bar{D}_{OBS}$ predicted were in fair agreement with experiment.

The result was encouraging because the calculations could be improved on a great deal.

In fig. 2, calculated single particle level densities g in $\hbar\omega(\delta)$ units for typical deformations $\beta = .1, .2, .3, -.1$ are plotted as a function of neutron and proton numbers N and Z, respectively.

From this attempt some interesting information emerges:

a) typical shell closure effects are punctually reproduced,
b) important proton contributions can be observed,
c) important deformation effects are observed,
d) the O.S. trend finds a significant explanation,
e) deviations from the O.S. may be expected in the sense of a local behaviour of those nuclei where more pronounced P and/or deformation effects are present.

On this basis, and pending a more thorough application of the model, local systematics for "a" have been determined empirically for several families of isotopes. In fig. 3 an example is given for Zr, Mo, Sn, Nd, Sm families. Local systematics put in evidence certain residual odd-even effects in deformed nuclei such as Sm isotopes |3|, |4|, |5|. It should be verified whether these are real odd-even effects or they are not due to neglecting a spin dependent deformation energy correction.

In practice it can be estimated that local systematics improve the accuracy in the determination of "a" by 5% and is helpful in interpolating "a" values for families with many isotopes and data; using the overall systematic helps to estimate unknown "a" values, in families with few or unaccurate data. Knowledge of "a" particularly may affect the calculations of radiative widths and of all non dominant competitions in statistical decay of compound nucleus.

1.2. SPIN DISTRIBUTION

a. Determination of the spin cut off factor σ^2 at low energies

A model description of the spin distribution of levels is given only at high excitation, and is completely characterized by the spin cut off factor $\sigma^2(E)$ (which carries on the energy dependence). Nevertheless, application of the statistical description of the spin distribution to low lying levels, has been suggested by Schmittroth in ref.|6|. Successively I analysed the spin distribution of the experimental discrete levels of about 100 isotopes.

In 40% of the cases considered a good fit was achieved provided the factor σ^2 was appropriately chosen |6| for each case. In the remaining 60% of cases, either the discrete level schemes were not well known, or the extrapolation really failed to work (see fig. 4), that is, the procedure adopted at intermediate energies, of interpolating the spin cut off factor in between the σ^2 values obtained, from the fit of known levels and from the model, at times can be somewhat arbitrary.

The staircase plots of the cumulative number of experimental levels for given J, that is $N(E,J)$, and of the cumulative number of all levels $N(E)$ vs E can in general be fitted on a lin-log scale, by a family of parallel straight lines. This supports the assumption that σ^2 does not depend on energy, at low energies. In fig. 5 the case of Zr-92 is considered, because discrete levels are available up to 3 MeV.

Where level schemes are not available or unreliable, any choice for σ^2 at low energy is doubtful.

b. Determination of $\sigma^2(E)$ at high energies

At high energies a model prediction of $\sigma^2(E)$ is given by |1|,

$$\sigma^2(E) = \langle m^2 \rangle \, g \, T(E)$$

in terms of single particle state density g, nuclear temperature $T(E)$ and variance of angular momenta projection of single particle states around the Fermi level.

The quantities $\langle m^2 \rangle$, g, T vary from nucleus to nucleus. The quantities g and T can be related to "a" values, while $\langle m^2 \rangle$ should be determined for each case.

The Nilsson model also permits a determination of $\langle m^2 \rangle$. In fig. 6 the calculated values of $\langle m^2 \rangle$ are plotted as a function of mass number A for typical deformation $\beta = .1, .2, .3, -.1$.

One may observe that $\langle m^2 \rangle$ is an oscillating function of A which depends on N, Z, β and does not exhibit usual shell effects.

Cameron |1| proposed an average value $\langle m^2 \rangle = .146\ A^{2/3}$ which is visualized in fig. 6 by the dashed straight line.

The average value adopted at Bologna is $\langle m^2 \rangle = .24\ A^{2/3}$ |7| which is represented by the solid straight line. A good knowledge of $\sigma^2(E)$ is important because it affects the spin distribution of levels and consequently the determination of "a" and the level density description. In addition, more proper choices for $\sigma^2(E)$ are expected to yield a sizeable improvement in the accuracy obtainable from "a" systematics.

Non dominant process calculations are rather sensitive to σ^2 choices.

Some improvements in the application of the Nilsson model for consistently determining both "a" and $\langle m^2 \rangle$ are in progress at Bologna, towards a better determination of the effective excitations and of potential well parameters.

The Hartree-Fock theory must be mentioned here as a valuable, advanced alternative.

1.3. PARITY DISTRIBUTION

Suggestions for an empirical determination for the parity distribution law, of low lying levels, come from the analysis of the experimental parity distribution of known levels, where they are not all characterized by the same parity.

From the analysis of a large number of cases it has been seen that in general, because both total level density and the density of levels of a given parity, can be fitted by an exponential law at low energies, the probability of a given parity can also be fitted by a low exponential with energy.

Beyond the value of the median, $E_{1/2}$, of the distribution, the probability is assumed to be 1/2, for both parities.

In fig. 7 the case of In-116 is illustrated, where the fit of $N(E)$, $N(E,\pi=+)$, $N(E,\pi=-)$ is achieved with a value for the median $E_{1/2} = 4.433$ MeV.

Even though this procedure proved satisfactory in the cases tested, it should be used with care, because it has no theoretical support. Variations in parity distribution do not affect neutron inelastic cross sections, on the contrary, due to E_1 selection rules, they may strongly affect radiative width calculations, and consequently capture calculations where capture widths cannot be renormalized.

Where only levels of a given parity are known, or no levels at all are known, any of the commonly adopted procedures in the determination of the parity distribution of low-lying levels is arbitrary.

1.4. NUCLEAR TEMPERATURE AND MATCHING ENERGY U_x IN COMPOSITE LEVEL DENSITY DESCRIPTION

Nuclear temperature T, at low excitations can be assumed to be constant and can be determined from the fitting conditions imposed on the composite level density description.

Results of calculations performed in the F.P. range are shown in fig. 8. T appears as a smooth function of N, characterized by small fluctuations representative of weak shell closure effects.

The uncertainty in interpolating unknown T values can be estimated to within 15% when using the solid straight line, and reduces to 10% if the dashed curve is used.

Nuclear temperature is useful to the determination of evaporative spectra.

To the extent that the relation among T, "a" and the matching energy U_x

$$1/T \simeq (a/U_x)^{\frac{1}{2}}$$

is valid, a correlation among the systematic trends of "a", T and U_x is expected. Results of the investigation are shown in fig. 10. Marked shell closure effects appear as well as the local behaviour of SN isotopes (represented by dotted circles).

The uncertainty over unknown U_x taken out of the systematics, can be estimated to within 25%.

It is important to have a systematic of U_x because knowledge of it is sufficient for a complete determination of the total level density, at low energies.

2. RADIATIVE WIDTHS

2.1. MODELS

In ref. |8| it was indicated that, provided certain refinements are introduced, the Brink-Axel (B.A.) model may give absolute values for average total radiative widths at the neutron binding $\overline{\Gamma_\gamma(B)}$ for medium and heavy nuclei, where only statistical processes are involved. In particular important spin and parity dependence was found especially for lighter nuclei.

Calculations over 75 F.P. have shown that 56% of the average radiative widths calculated at neutron binding fell within experimental error bars, while 18% differed by more than 30% from experimental ones.

Nevertheless, taking into account that important valency contributions have been demonstrated for certain isotopes, and considering the large uncertainties in low lying level schemes and in significant parameters, one concludes that the full degree of the model's validity can only be ascertained after the parameters involved have been more precisely determined.

Preliminary calculations with recent level schemes, and taking into account the considerations in Sect. 1, are in progress and seem to confirm a general improvement.

A new approach to the problem has been attempted |9|, based on the assumption that the nucleus irradiates like a black body formed by an assembly of oscillators.

By straightforward application of the concepts of statistical

thermodynamics a simple expression for $\overline{\Gamma_\gamma(B)}$ can be found in terms of mass number A and thermodynamic temperature T .

The model was checked for all isotopes from Ca to Am, for which experimental values were available.

In fig. 9 the ratios $\overline{\Gamma_\gamma^{EXP}}(Bn)/\overline{\Gamma_\gamma^{CALC}}(Bn)$ are plotted vs. A .

The model has not yet been refined in order to account for spin and parity dependence of radiative widths.

2.2. SYSTEMATICS

According to the B.A. model as outlined in ref. |8| it appears that, for the most part, the contributions to total $\overline{\Gamma_\gamma(B)}$ are due to transitions to levels lying below the matching energy U_x . This implies that the success of the model in particular depends on the knowledge of low lying excitation levels and (owing to E1 selection rules) on spin and parity distributions.

Because no systematic behaviour of spin and parity distributions of low lying levels can be envisaged, no systematic behaviour of $\overline{\Gamma_\gamma(B,J,\pi)}$ should be expected to be found, valid in all cases.

On the contrary, according to the black body model, a systematic behaviour for $\overline{\Gamma_\gamma(Bn)}A^{1/3}$ vs $[(B-\Delta)/a]^{3/2}$ is predicted; A, B, Δ, a being mass number, neutron binding energy, pairing energy, level density parameter, respectively.

Results of investigation for most isotope rich families of U, Gd, Se, Ge are shown in fig. 10. The solid line, which accounts for experimental data to within 50%, represents the overall trend predicted by the model. Local systematics are given by dashed lines.

2.3. GIANT RESONANCE PARAMETER SYSTEMATICS

The systematic trend of all giant resonance parameters (G.R.P.) both for spherical and deformed nuclei has been determined |10| on the

basis of the hydrodynamic model. This model is able consistently to describe data for all medium and heavy nuclei, provided the deformation parameter β is known.

Fig. 11 shows the systematic trend of peak energies E_o, half maximum widths Γ_o and peak cross section σ_o for magic nuclei. Maximum uncertainties for unknown values of E_o, Γ_o/E_o, σ_o can be taken as less than 3%, 20% and 15%, respectively.

In fig. 12 the trends are shown of the characteristics of the split giant resonances in deformed nuclei. Labels a and b refer to semimajor and semiminor axes, respectively.

Uncertainties in determining unknown values for E_a/E_o, E_b/E_o can be estimated to less than 10%, while for Γ_a/E_a, Γ_b/E_b and for $\sigma_a E_a$, $\sigma_b E_b$ to within 30%.

G.R.P. do not play an important role in neutron capture calculations when $\overline{\Gamma_\gamma(B)}$ is known and when the excitations involved are far from giant resonance peak energy.

On the contrary an important role is played in calculations of absolute values of Γ_γ with the B.A. model.

3. NEUTRON CAPTURE CROSS SECTION SYSTEMATICS

3.1. STATISTICAL PROCESSES

In cases of statistical processes where only s-wave neutron capture is involved and for values of the radiative transmission coefficients $T_\gamma(E)$ that are negligible with respect to neutron transmission coefficients, the statistical model formula, for neutron capture, reduces to a linear expression in $T_\gamma(E)$.

Experimental data for $\sigma_{n,\gamma}$ at 25 keV, renormalized and plotted against $T_\gamma(E)$ confirm the model predictions to within an uncertainty of 40% as can be seen in fig. 13.

This systematic helps to check, where necessary, the consistency

of adopted values of $\overline{\Gamma_\gamma}(B)$ and $\bar{D}_{OBS}$, when $\sigma_{n,\gamma}(25)$ is known, or gives supplementary information for the determination of unknown values of $\overline{\Gamma_\gamma}(B)$ or $\bar{D}_{OBS}$.

In addition, while a given experimental capture cross section at 25 keV can be reproduced by an infinite number of couples $(\overline{\Gamma_\gamma}(B), \bar{D}_{OBS})$ in a fixed ratio $T_\gamma(E)$, on the contrary owing to the relation between $\overline{\Gamma_\gamma}(B)$ and $\bar{D}_{OBS}$ one and only one couple $(\overline{\Gamma_\gamma}(B), \bar{D}_{OBS})$ should be found also to satisfy the systematic at 25 keV.

By the joint use of all these systematics, in principle, one has the possibility of iterating until a couple $(\overline{\Gamma_\gamma}, \bar{D}_{OBS})$ is found that reasonably accounts for all experimental information and systematics, and which fulfills consistency requirements.

3.2. DIRECT PROCESSES SYSTEMATICS

A satisfactory agreement between experimental data on fast neutron capture cross sections and calculations accounting for both compound nucleus and direct-semidirect reaction mechanisms has been illustrated in ref. |11|.

Because this formulation implies rather a large man- and computer-time waste, where more rough estimates are sufficient, one can use the Lane and Lynn |12| formula for direct process contribution with Brown enhancement factor |13| accounting for collective reaction mechanism.

This recipe (see also ref. |14|) depends on a free parameter K.

A systematic behaviour of K has been found by plotting the ratios

$$K = \sigma_{n,\gamma}^{EXP}(14.5) \; / \; \sigma_{n,\gamma}^{CALC}(14.5)$$

against mass number A. Results are shown in fig. 14.

Experimental data on capture cross sections at 14.5 MeV measured by the activation method were not considered because of possible contaminations due to capture of neutrons from inelastic scattering.

CONCLUSIONS

While the adopted formalism for neutron cross section calculations is going to be increasingly sophisticated and very much standardized in most laboratories, substantial disagreements still appear in the methodology of determining recommended sets of parameters, in the absence of experimental information.

For these reasons greater efforts are desirable in the study and development of models aimed to improve parameter consistency and accuracy.

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FIGURE CAPTIONS

Fig. 1 Level density parameter a (MeV^{-1}) vs neutron number N .

Fig. 2 Single particle level density g in $\hbar\omega(\delta)$ units vs neutron or proton numbers N and Z , respectively for typical deformation β = .1 , .2 , .3 , -.1 .

Fig. 3 Local systematic for the level density parameter a(MeV^{-1}) vs. neutron number N for Zr, Mo, Sn, Nd, Sm isotopes. Overall systematic is also shown.

Fig. 4 Comparison between the spin distributions of known discrete levels and the predictions according to the law $f(J) = (2J+1)\exp\left[-(s+\frac{1}{2})^2/2\sigma^2\right]$, when σ^2 is obtained according to maximum likelihood estimator | 6 |.

Fig. 5 Trends of the cumulative number of levels. N(E) , N(E,J=0) , N(E,J=2) vs E , according to all J , J=2 , J=0 in Zr-92.

Fig. 6 Variance of angular momentum projection of single particle state $\langle m^2 \rangle$ vs neutron or proton numbers N and Z , respectively for typical deformation β = .1 , .2 , .3 , -.1 . Straight solid line gives $\langle m^2 \rangle$ as suggested by | 7 | and dashed straight line gives $\langle m^2 \rangle$ as suggested by | 1 |.

Fig. 7 Cumulative number of levels N(E) , N(E,π=+) , N(E,π=-) vs E in In-116.

Fig. 8 Systematic trend of nuclear temperature T(MeV) vs neutron number N at low excitations, and systematic trend of matching energies U_x(MeV) vs N . Dotted circles represent tin isotopes.

Fig. 9 Ratios of experimental to calculated radiative width at neutron binding B , $\overline{\Gamma_\gamma}^{EXP} / \overline{\Gamma_\gamma}^{CALL}$ vs mass number A .

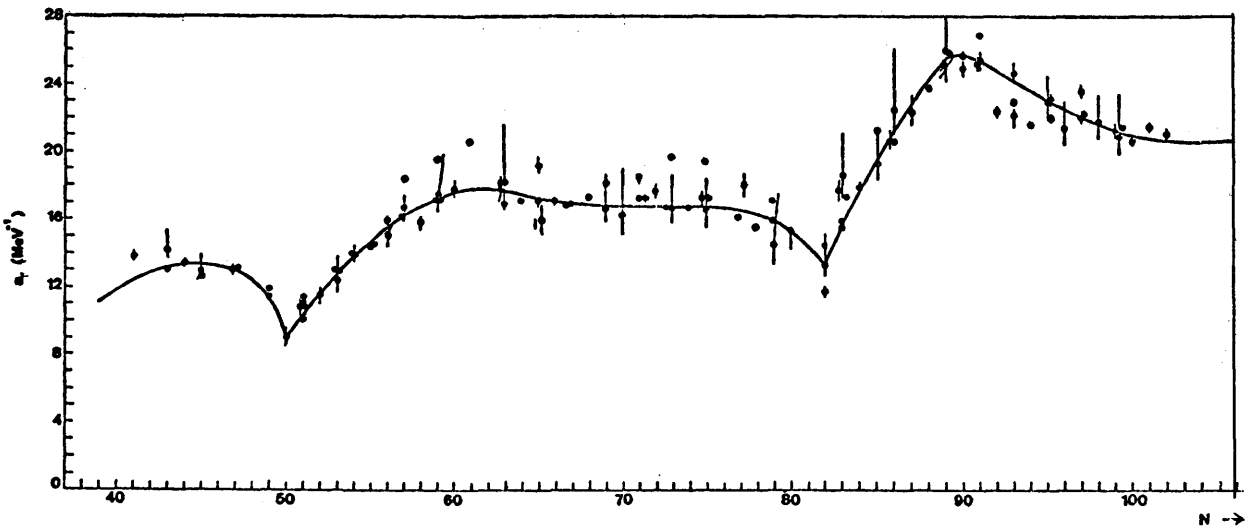
Fig. 10 Systematic trend of experimental radiative widths $\overline{\Gamma_\gamma}(B)A^{1/3}$ vs $\left[(B-\Delta)/a\right]^{3/2}$. The solid line represents the black model predictions. Local systematics are given by dashed lines.

Fig. 11 Trend of giant resonane parameters for spherical nuclei. E_o(MeV) , Γ_o/E_o σ(barn) vs mass number A .

Fig. 12 Trend of giant resonance parameters $E_{a \text{ or } b} / E_o$, $\frac{\Gamma_a}{E_o}$, $\frac{\Gamma_b}{E_o}$ vs deformation parameter β , and trends of $\sigma_a E_o$, $\sigma_b E_o$ vs mass number A for deformed nuclei.

Fig. 13 Systematic of 25 keV experimental neutron capture cross section vs $\overline{\Gamma_\gamma}^{EXP} / \left[\overline{D}^{EXP}(2I+1)\right]$, I being the target spin.

Fig. 14 Systematic trend of the normalization constant K in direct-semidirect neutron capture vs mass number A as derived from calculated and experimental neutron capture at 14.5 MeV.



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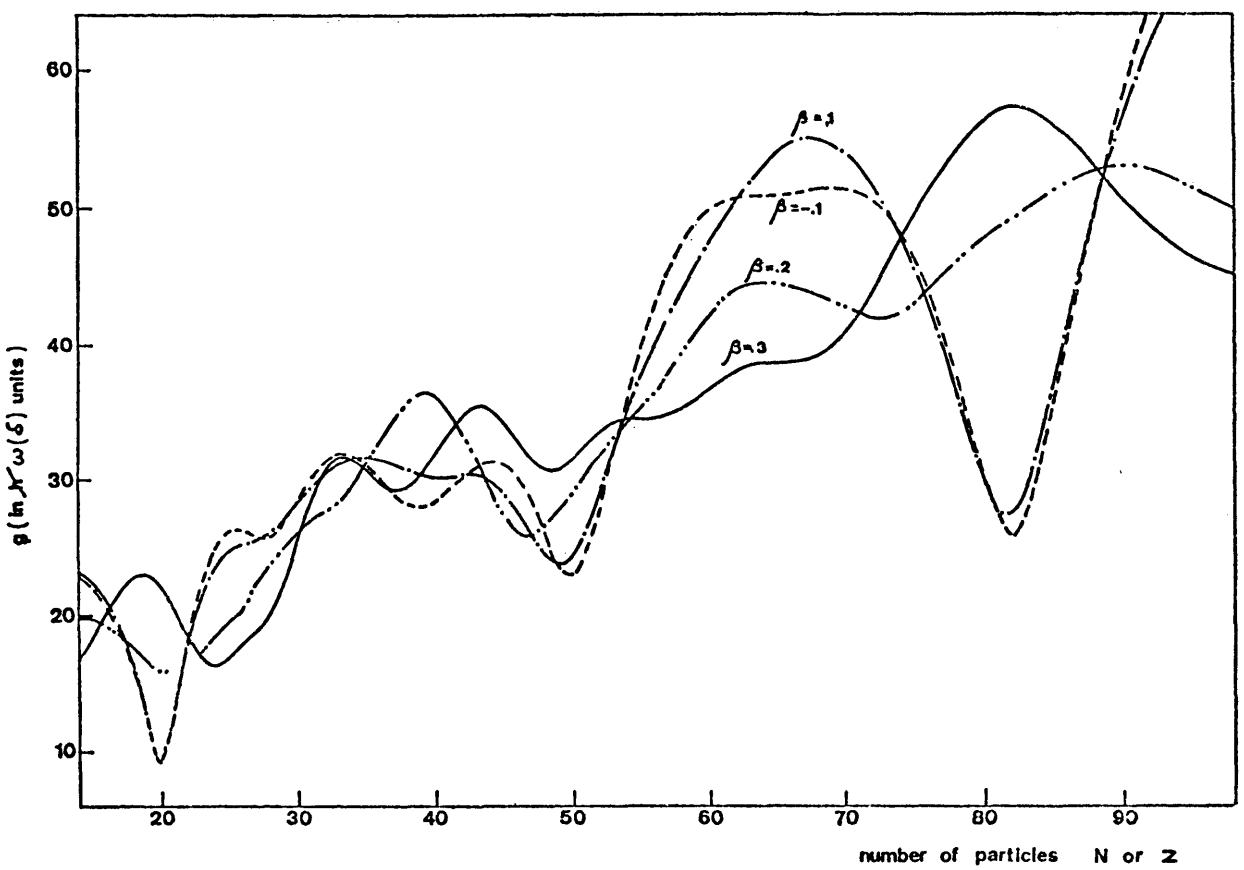

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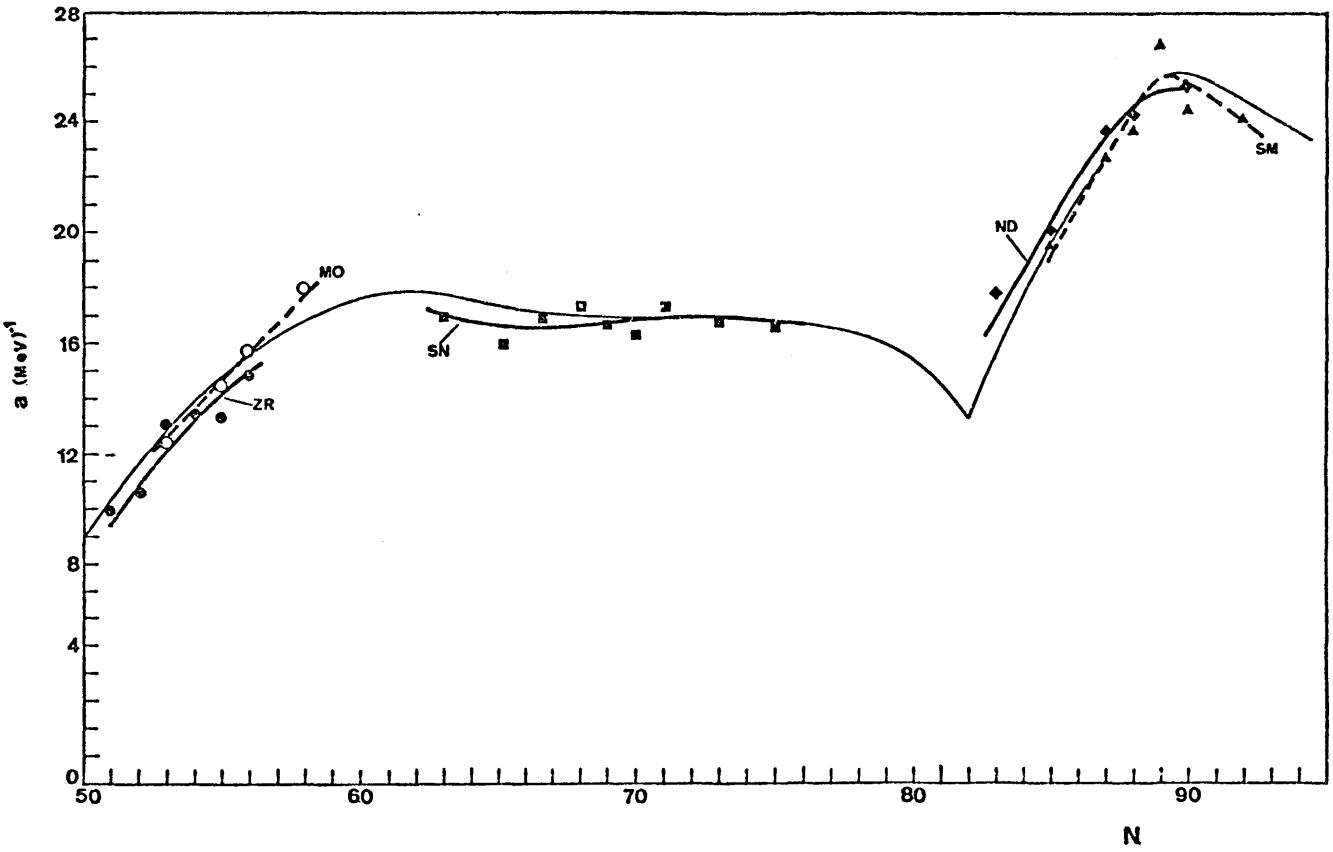

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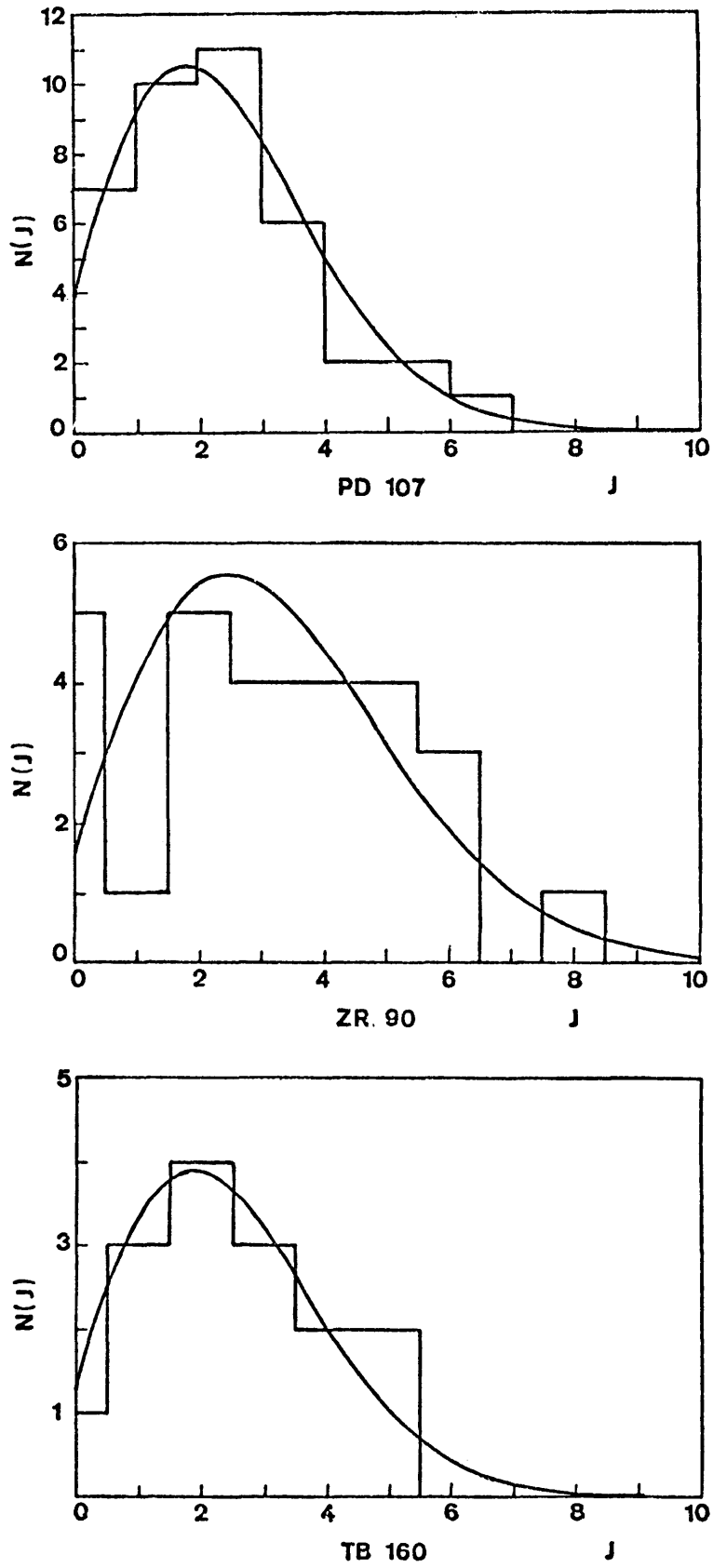

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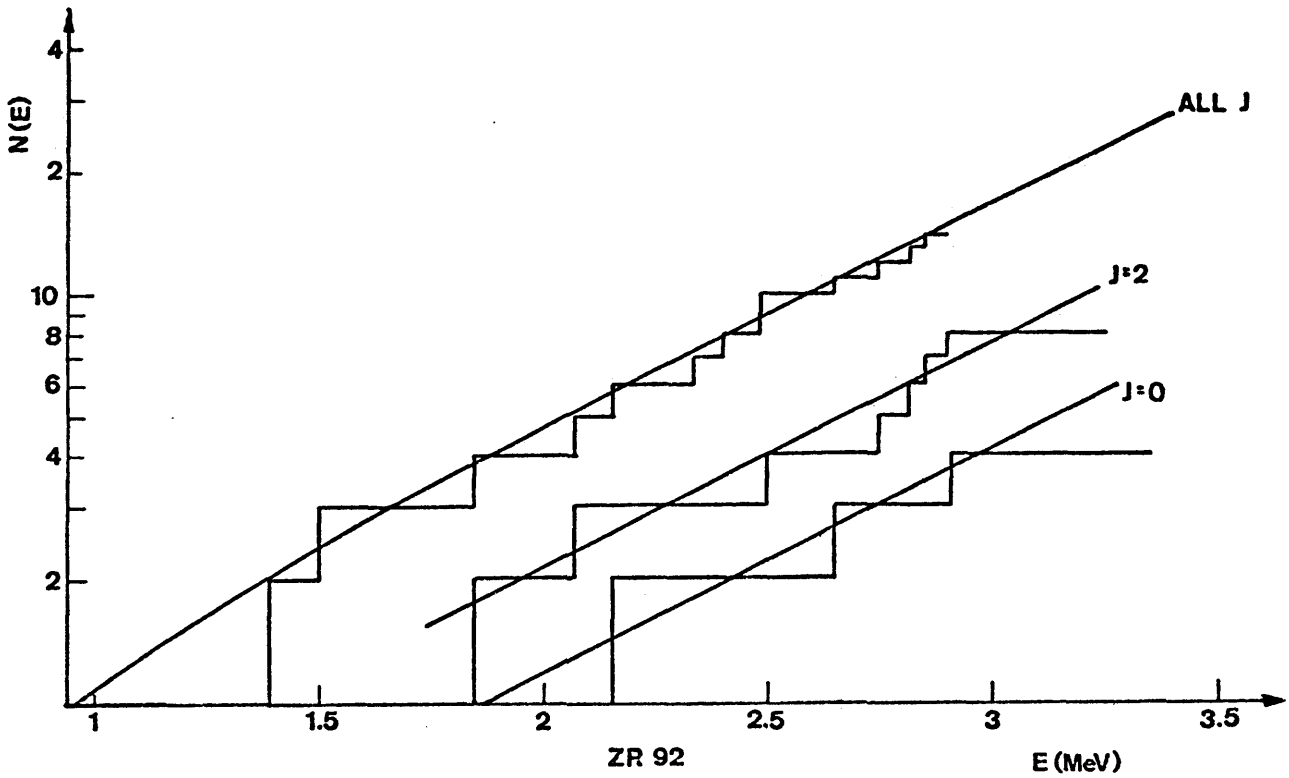

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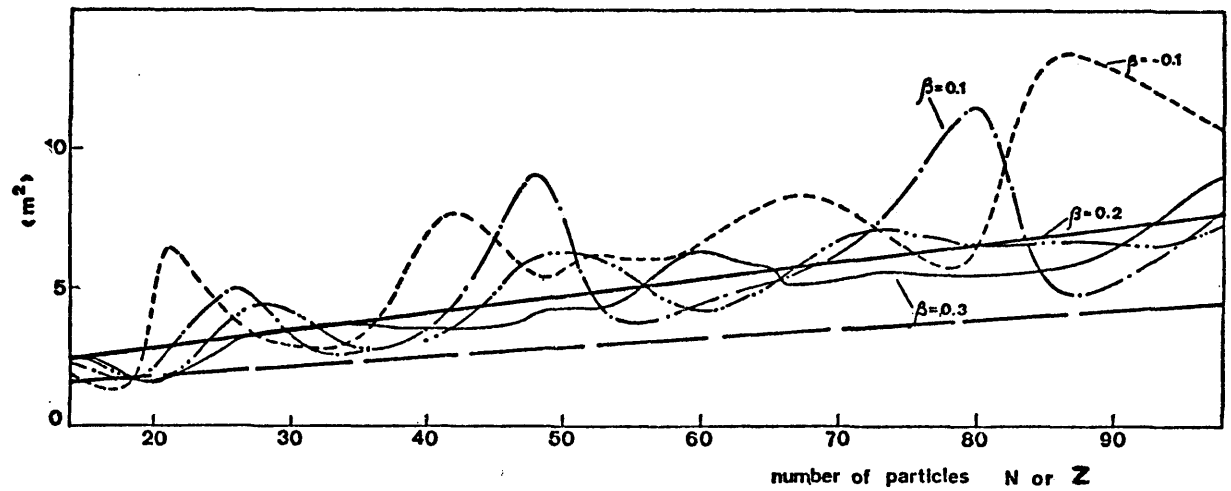

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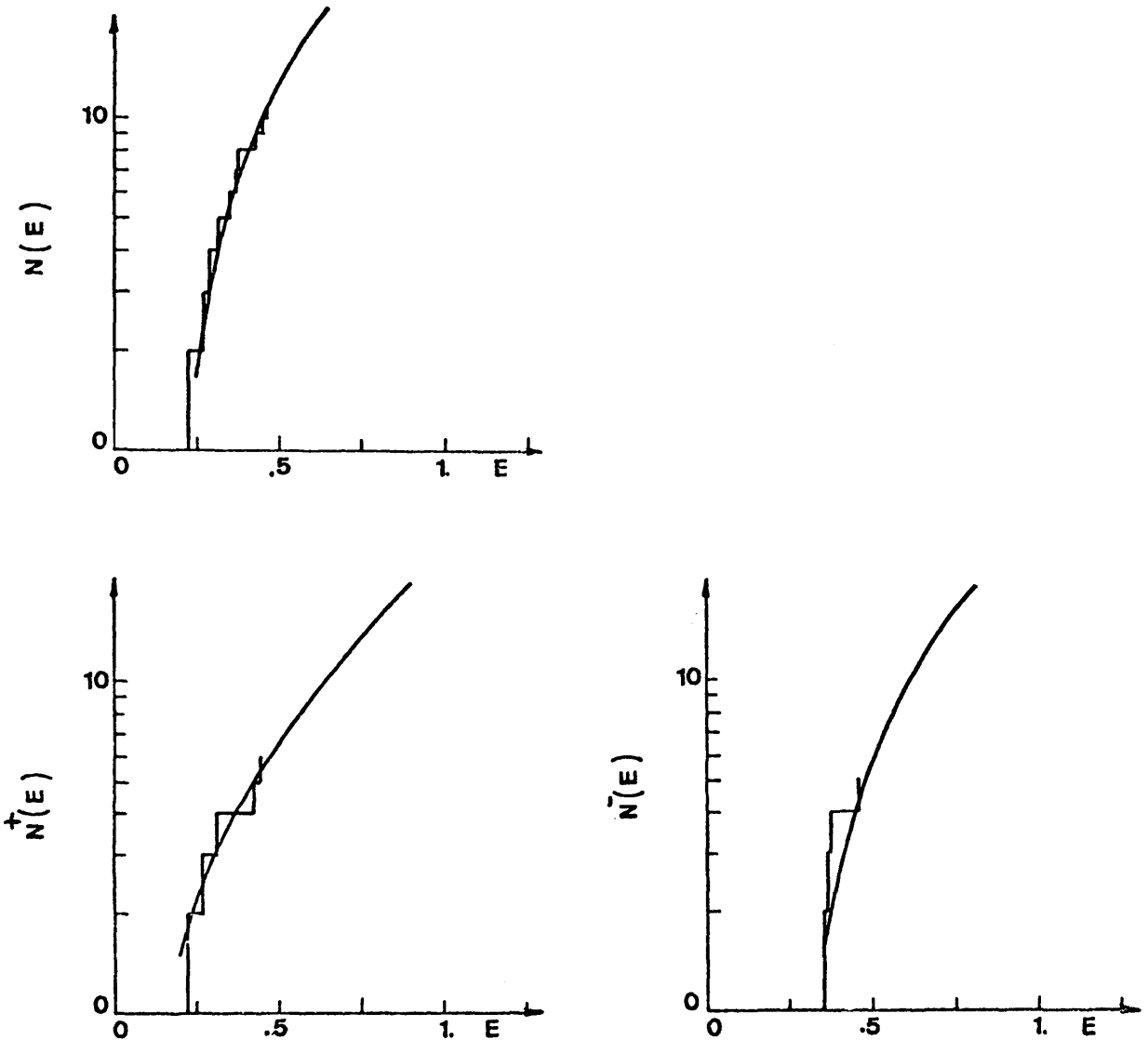

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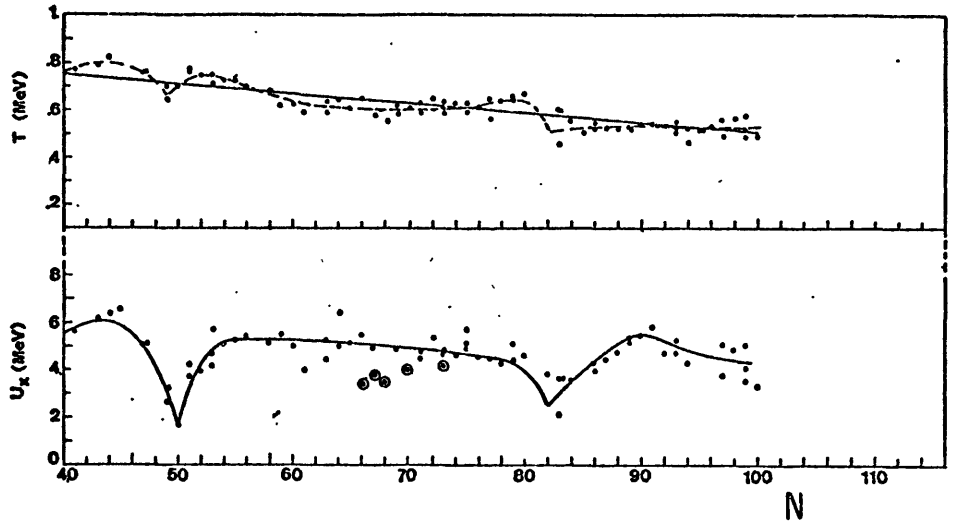

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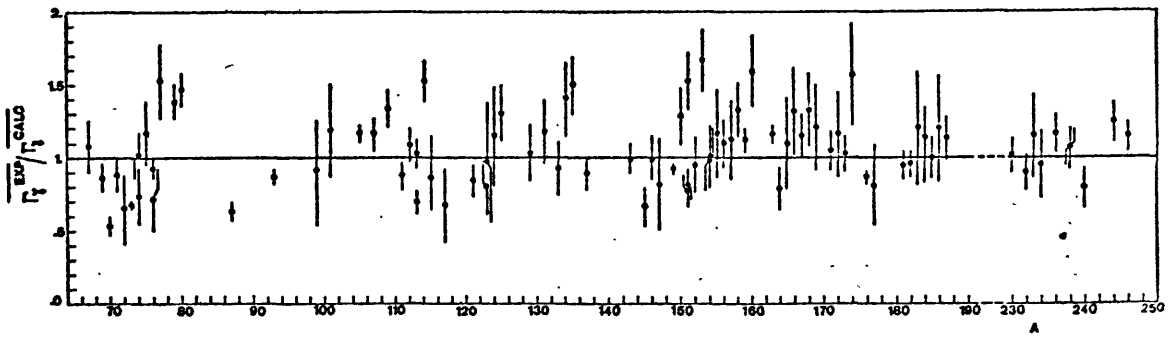

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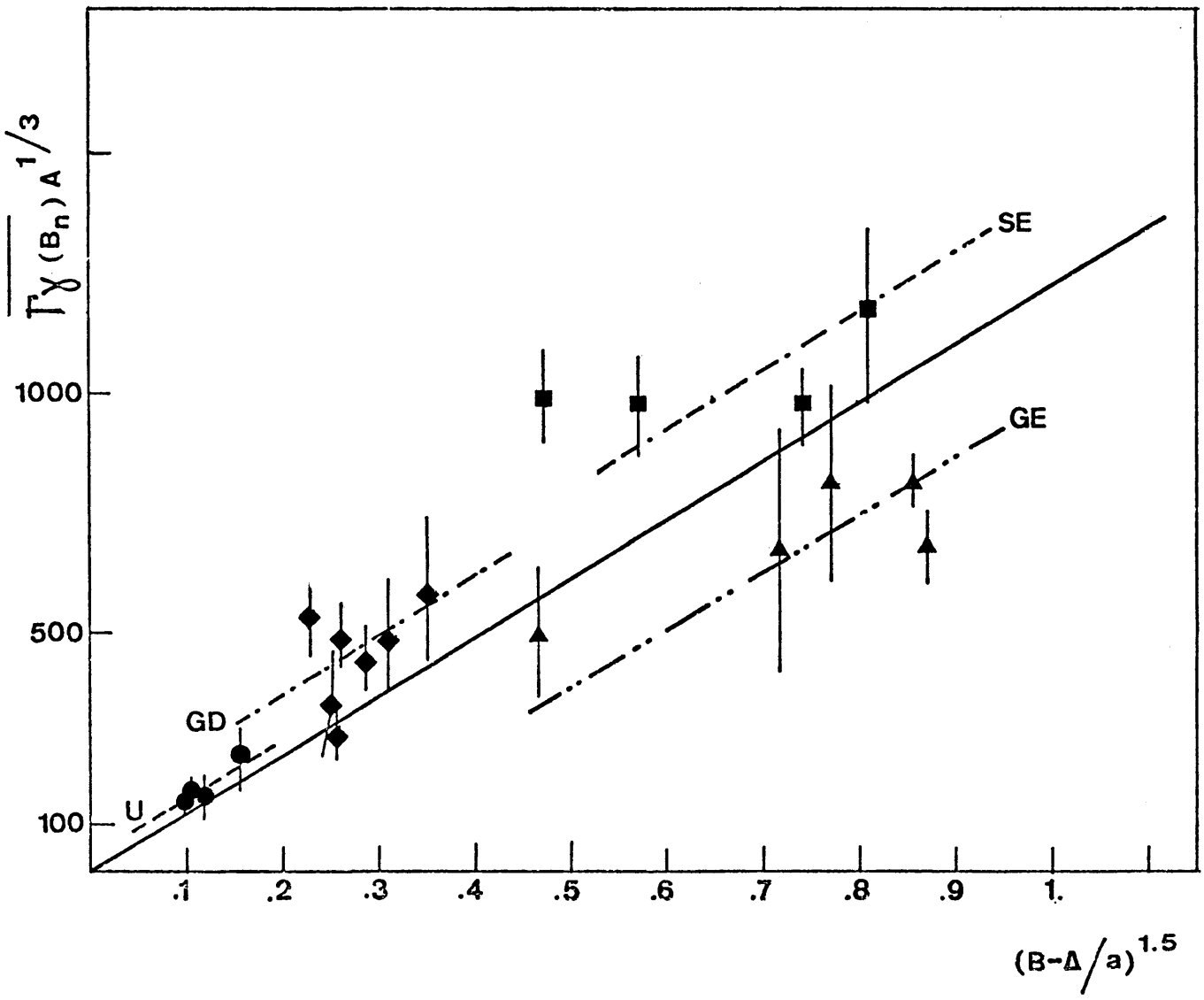

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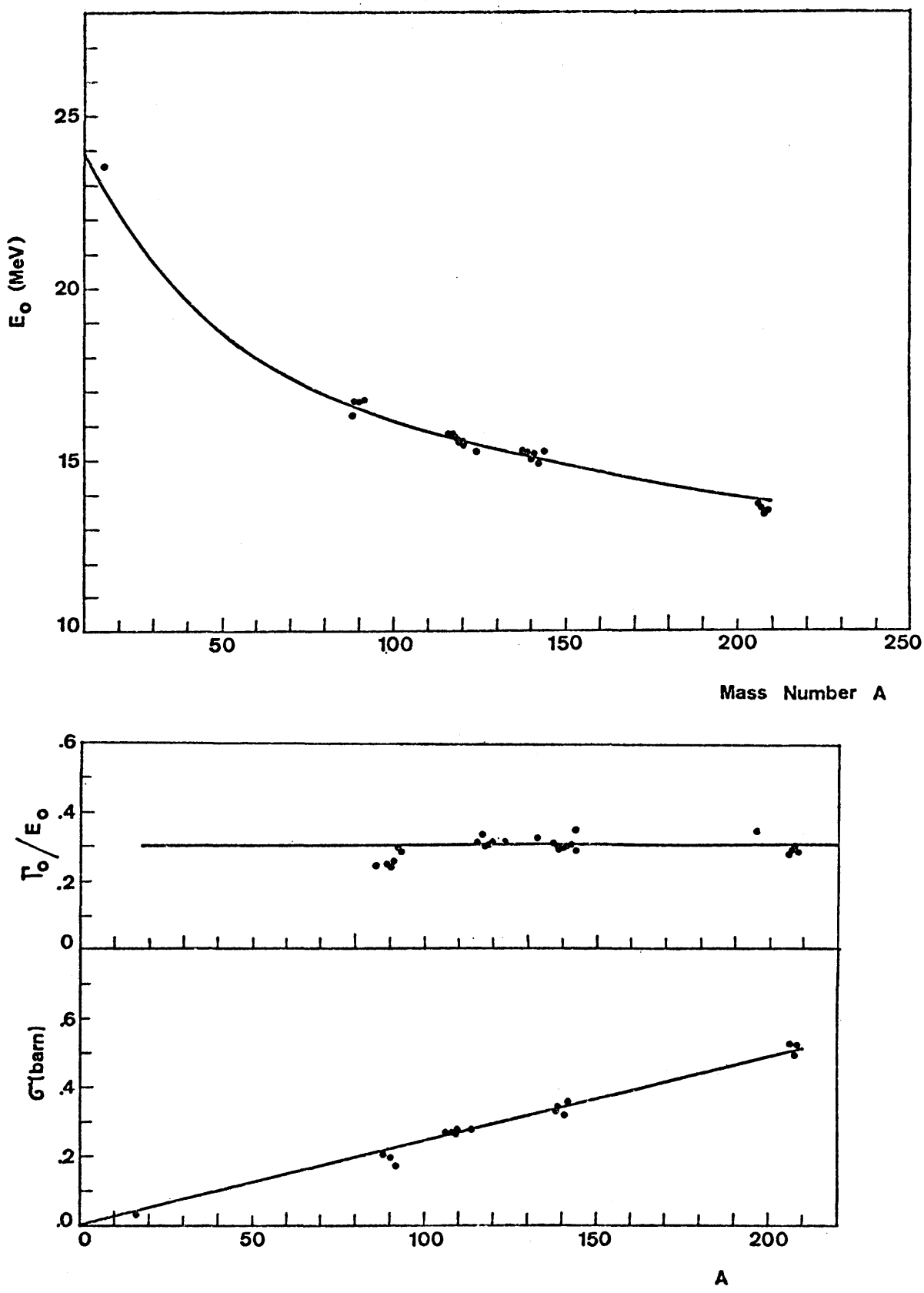

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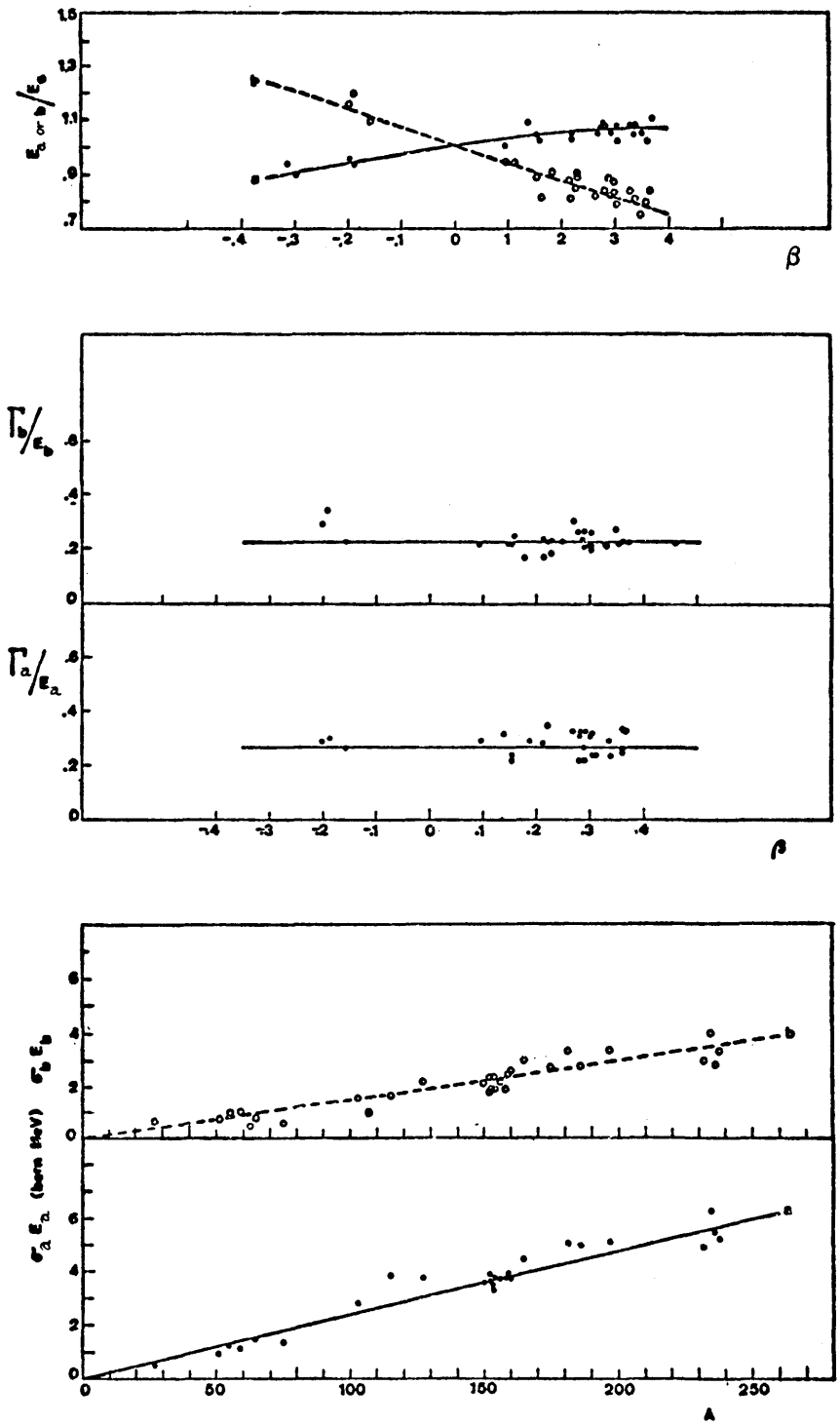

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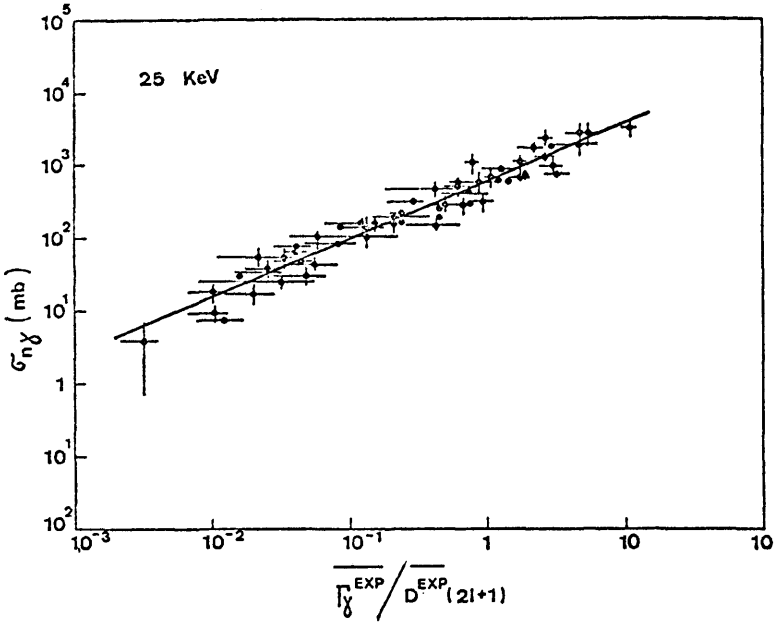

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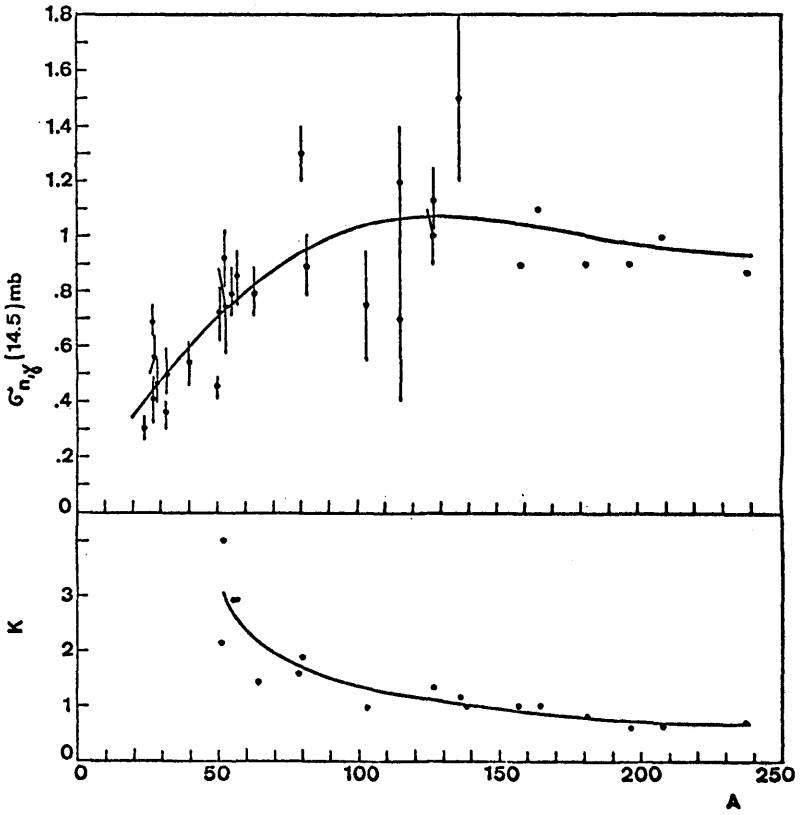

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Contribution to Review Paper 10

SYSTEMATICS OF THE ODD-EVEN EFFECT IN LOW ENERGY FISSION OF EVEN Z NUCLIDES AND SYSTEMATICS OF THE HALF-LIVES OF VARIOUS FISSION PRODUCTS

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Abstract

Independent yields of fission products of the elements Kr, Sn, Sb and Xe in the fast neutron fission of ^{232}Th and of Rb, Sr, Cs and Ba in the thermal neutron fission of ^{235}U were measured. The isotopic distributions show a significant odd-even effect. The dependence of the proton odd-even effect on major parameters of the fissioning system is discussed and the magnitude of the effect is found to decrease with increasing excitation energy and atomic number of the fissionable nuclide. Whereas the odd-even effect systematics fits very well in the central region of the isotopic distribution of yields in thermal neutron fission of ^{235}U, a significant enhancement of the yields in the heavy mass wing of the distribution was found. An interpretation of this effect is suggested. Half-lives of various fission products were measured and compared with theoretical predictions.

1. Introduction

The paucity of available data on fission product mass distributions limits the extent of our comprehension of several features of the fission process. Additional experimental data of improved precision is needed to check theoretical models which were recently formulated and studies of variegated fissioning systems may reveal the finest details of the fission process.

The distribution of yields of fission products has not yet been determined with adequate precision, except for the fission of ^{233}U and ^{235}U induced by thermal neutrons[1-4]. The analysis of 19 mass chains in the thermal neutron fission of ^{235}U has shown that a Gaussian with a constant

standard deviation of σ_Z=0.56±0.06 charge units does describe, on the average, the fractional independent yields in any isobaric chain[5]. Closer examination of even-Z fissioning nuclides, however, revealed systematic fluctuations about the Gaussian, so that the independent yields of fission products are higher than predicted by the calculations for those nuclides with an even number of protons and lower for those with an odd number of protons[1-6]. A quantitative approach to this odd-even effect showed[1,2] that the average difference from the normal distribution[5] is (22±7)% in the thermal neutron fission of both ^{233}U and ^{235}U.

The present study was aimed in two directions:

a) Establishing the dependence of the proton odd-even effect on the fissionable nuclide and on the excitation energy using published fission yields of uranium and plutonium isotopes, as well as our recently measured fission yields in the fast neutron (fission spectrum) fission of ^{232}Th.

b) Detailed examination of existing systematics of fission yields of isotopes far from the center of the nuclear charge distribution using on-line measurements of mass separated fission products of the alkali and alkaline earth elements.

2. Experimental

The experimental work was based on ionization of fission products and their isotopic separation accomplished by accelerating the ions and subsequent electromagnetic mass separation. The experiments were performed using the on-line isotope separator SOLIS coupled on-line with an integrated target-ion source. The fission events were induced by a thermal neutron flux (6×10^8 ncm^{-2}sec^{-1}) bombarding a target made of ^{235}U enriched to 93%. Two different experimental assemblies were used for the study of the various investigated elements:

(1) The study of the alkali and alkaline earth elements, Rb, Sr, Cs and Ba, was carried out with an uranium-graphite target, heated to 2000°C, integrated with a surface ionization source[7,8]. The delay half-times obtained were 0.27±0.03 sec for Rb and Cs isotopes and 1.4±0.3 and 1.0±0.4 sec for Sr and Ba isotopes, respectively. The alkali elements were ionized on a Ta surface and the alkaline earth elements on Re. The present version of the integrated target-ion source is very efficient, selective and reliable[7,8]. Beta activity measurements of the fission products were made at the collector

chamber of the separator by a 300μm silicon surface barrier detector. Delay and decay times were determined from growth and decay measurements of a mass aligned along the axis of the collector chamber. Isotopic fission yields were determined in beta activity scans over a mass region. A scan is done by a gradual variation of the magnetic field of the separator.

(2) The study of the rare gases Kr and Xe was carried out with an emanating target containing a mixture of U_3O_8 or uranyl stearate with barium stearate[9,10]. Gaseous fission products were transferred into the plasmatron ion source (modified Nielsen type). The mean transfer time was about 0.7 sec, calculated from the rate of activity accumulation. The overall efficiency of the system was 0.1%-1% according to the mode of collecting the activity. The beta activity in on-line measurements was detected by a telescope system consisting of NE-102 plastic scintillators. In off-line measurements, the beta activity was collected on an aluminum strip which afterwards was cut into mass segments. These were counted in gas flow proportional counters. Both techniques were used to measure fission yields.

In another set of experiments, radiochemical separations were used in an off-line method to measure independent yields in the fission of ^{232}Th induced by the fast neutron flux of the "rabbit" transfer system[11]. The elements studied were Kr, Sn, Sb and Xe. Gamma activities of fission products were measured with a Ge(Li) detector. In most cases, counting could be started within ~20 sec after the end of irradiation.

3. Results

3.1 Half-lives

The experimental data in activity decay measurements were analyzed by the program DECAY which fits, by the least squares method, a series of decaying exponentials to the experimental curve. The half-lives are given in Table 1. All measurements were made by beta multiscaling, except in the case of ^{96}Sr, which was treated by following the decay of the 122 keV gamma transition in the excited levels of ^{96}Y. Therefore the result for ^{96}Sr was very precise. The 0.6±0.1 sec component of mass 98 was attributed to both ^{98g}Sr and ^{98m}Y. Very good agreement with published values was obtained in most cases. The three nuclides ^{147}Ba, ^{148}Ba and ^{149}La were identified for the first time in this work and their half-lives determined. For some other nuclei, meaningful results were obtained instead of rough estimates. The shortest half-life measured was 0.1 sec.

3.2 Fission Yields

Independent fission yields of isotopes of Kr and Xe produced in ^{235}U (n_{th},f) were published[1,10] prior to the former Bologna FPND panel and therefore are not reported here again.

Independent yields of Rb, Sr, Cs and Ba[12], measured in this work, are given in Table 2. The distributions, compared with the normal distributions of Wahl, are shown in Fig. 1. The present distributions include values for ^{99}Rb, $^{147-148}Cs$ and $^{145-149}Ba$ determined for the first time. The $^{147-148}Cs$ and $^{147-148}Ba$ isotopes were also identified for the first time in the yield measurements. The precision of the results is very satisfactory as indicated, for example, by the value of ^{149}Ba. The error here is ±28% mainly due to the statistical error in the measurement of the activity and the error in the value of the delay half-time. The distributions of the independent yields in the light mass region are not completely determined, since only cumulative yields could be measured there. This is because the rare gases Kr and Xe are produced in fission in high yields in this region and are trapped in the target, thus leading to a significant production of the alkali isotopes by β decay.

The independent yields measured in the fast neutron fission of ^{232}Th are given in Table 3. The yields of ^{91}Rb, ^{131}Sb and ^{140}Cs were evaluated by subtracting the measured fractional yields from 1.0. The distributions of independent yields are compared with the normal distribution in Fig. 2.

4. Discussion

4.1 Half-lives

The measured half-lives were compared (see Fig. 3) with the theoretical prediction of Takahashi et al.[13] evaluated within the framework of the Gross Theory of beta decay, and the agreement was better than an order of magnitude. The agreement between calculated and experimental data was better when the modified Lorentz single particle strength function was assumed, rather than a Gaussian. There is a similarity between the systematic behavior of the experimental values and that of the predicted values. The tooth structure, due to pairing, in the predicted curve also appears in the experimental curve, but with a smaller amplitude. These findings permit extrapolation of the experimental curve and prediction of values which have not yet been measured. The experimental and theoretical curves of Cs, from mass 143 upwards, are nearly straight lines when plotted on semilog scales and therefore, a linear extrapolation with a slight tooth - modulation is

justified. By this procedure, the half-lives of $^{147-148}Cs$ and ^{149}Ba were predicted and used in the analysis of independent yields. A similar nearly straight line is also found in the case of La.

4.2 Independent yields of Rb, Sr, Cs and Ba in ^{235}U (n_{th},f)

The central region of the distributions of these elements (see Fig.1) shows an odd-even effect in complete agreement with the findings of the recent studies of this phenomenon. The heavy mass wing of the distributions is characterized by a significant enhancement of the yields relative to the normal values calculated according to the prescription of Wahl. This enhancement cannot be explained by the odd-even effect. The nuclear charge distributions obtained by Siegert et al.[4] show a similar feature on the wings when compared with the normal distributions. By using chain yields, isotopic independent yields may be obtained and these indicate the same effect i.e. an enhancement in the heavy mass wing.

A model was developed[12] to interpret this effect based on calculation of the total energy released in the thermal neutron fission of ^{235}U into a fragment pair of complementary elements (e.g. Rb and Cs). By applying statistical considerations, the isotopic distribution may be obtained and its shape is found to be represented by a Gaussian. The width of this Gaussian is corrected for the additional dispersion caused by prompt neutron emission. The corrected width parameter is in very good agreement with the width parameter of a Gaussian fitted to the measured distribution. The width parameter σ_A of the Rb distribution is 2.1 amu, compared with 1.59 amu obtained according to the UCD postulate (σ_Z=0.56 charge units).

4.3 Independent yields in ^{232}Th (n_F,f)

The independent yields determined in the fast neutron fission of ^{232}Th do not include more than 2 values in the central region of the isotopic distribution of the elements studied. These yields, together with the published data on Te and I, were analyzed and compared with the normal values. The average odd-even effect for even-Z products is (28±10)% and for odd-Z products (-32±12)%. These results were averaged again to (30±12)% and corrected for particle-hole excitation at the saddle, induced by the high energy part of the neutron spectrum. The corrected value, (45±15)%, is introduced into the systematics of the odd-even effect (see 4.4).

4.4 Systematics of the odd-even effect

The measurements of independent and cumulative fission yields of various products by on-line and off-line techniques were aimed at establishing the systematics of the distribution of fission products. The normal distribution suggested by Wahl was found inadequate when applied to experimental yields of various elements[5]. Consequently, the analysis of all published independent yields in thermal neutron fission of ^{235}U revealed a systematic enhancement of the yields of even-Z products relative to a normal Gaussian, and a corresponding attenuation of odd-Z products[1]. The magnitude of this effect averaged over all fission events in ^{235}U (n_{th},f) was found to be (22±7)%. Further investigation of fission yields in ^{233}U (n_{th},f) revealed an effect of the same magnitude[2]. Recent data[3,4] on the nuclear charge distribution of light fragments in ^{235}U (n_{th},f) gave an average value which is in excellent agreement with the odd-even systematics[1,2].

The odd-even effect was found to be dependent on two major parameters[14]: the excitation energy at the saddle, and the atomic number Z_F of the fissionable nuclide. The effect decreases when the excitation energy at the saddle exceeds the pairing gap and therefore, the effect in the fast neutron fission of ^{235}U is only 10%. This indicates that the odd-even effect is a result of a partial preservation of the original proton pairs. When the available excitation energy is sufficiently high, more proton pairs may be broken at the saddle.

The Z_F dependence of the effect is shown in Fig. 4 for the following nuclei: ^{232}Th (fast), ^{233}U (thermal), ^{235}U (thermal), ^{238}U (fast) and ^{239}Pu (thermal). In the cases of ^{232}Th and ^{238}U, the fission barrier is high and the excitation energy in fast neutron fission is below the pairing gap. Therefore, the "thermal" odd-even effect is preserved. While the odd-even effect in the 3 isotopes of uranium is roughly the same (within the error limits), there is a clear tendency towards a decrease with increasing Z_F. It is predicted that for $Z_F \geq 95$ the odd-even effect is practically zero. The decrease of the odd-even effect at high atomic numbers may be explained by the increasing density of the pairing levels at the descent from saddle to scission which facilitates more level crossings and subsequent proton pair-breaking.

The evaluation of the odd-even effect in the distribution of fission yields and its dependence on the excitation energy and atomic number of the fissioning nuclide enable us to obtain good agreement between calculated and measured fission yields and a reliable prediction of the yet unmeasured values.

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Table 1: Half-lives (in sec) of fission products measured with the SOLIS on-line isotope-separator. Value in parentheses is the ratio exp./calc., where calc. refers to data from ref. 13.

Mass	91	92	93	94	95	96	97m	97g	98m	98g
Rb	59.6±0.2 (5.18)	4.57±0.07 (1.34)	5.92±0.09 (1.44)	2.80±0.04 (1.81)	0.402±0.008 (0.25)	0.22±0.01 (0.30)		0.18±0.01 (0.26)		0.10±0.02 (0.26)
Sr				76.7±0.9 (1.15)	25.1 ±0.2 (3.77)	1.10±0.02 (0.10)		0.43±0.03 (0.18)		0.6 ±0.1 (0.20)
Y						6.3 ±0.2 (0.83)	1.3±0.1	3.3 ±0.2 (0.34)	0.6±0.1	2.1 ±0.3 (0.69)
Mass	141	142	143	144	145	146	147	148	149	
Cs	29.3±0.1 (3.26)	1.78±0.02 (0.41)	1.78±0.01 (0.44)	1.04±0.03 (0.44)	0.65±0.03 (0.31)	0.31±0.06 (0.24)				
Ba			15.2±0.2 (1.07)	12.0±0.4 (0.58)	2.7 ±0.1 (0.41)	2.2 ±0.2 (0.31)	0.72±0.07 (0.24)	0.47±0.20 (0.17)		
La				40±3.5 (2.07)	21±5 (0.93)	9.0 ±0.6 (0.95)	4.4 ±0.5 (0.48)	2.6 ±0.6 (0.58)	1.2±0.4 (0.30)	
Ce									4.7±0.8 (0.31)	

Table 2: Independent yields (IY) in ^{235}U (n_{th},f) measured with the SOLIS on-line isotope separator.

Mass	IY (%)		Mass	IY (%)	
	Rb	Sr		Cs	Ba
92	3.24±0.28		142	2.34±0.33	
93	3.12±0.11		143	1.45±0.17	4.42±1.15
94	2.26±0.34	5.77±1.6	144	0.43±0.06	3.90±0.32
95	0.89±0.10	4.45±0.14	145	0.100±0.028	2.60±0.55
96	0.35±0.06	2.56±0.72	146	0.045±0.009	0.79±0.17
97	0.10±0.02	1.63±0.46	147	0.0097±0.0024	0.37±0.09
98	0.036±0.009	0.49±0.14	148	0.0024±0.0008	0.19±0.04
99	0.0064±0.0018	0.18±0.06	149		0.060±0.017
100		0.057±0.025			

Table 3: Independent yields measured in the fast neutron fission of ^{232}Th (in %)

Kr		Xe		Sn		Sb	
90	5.543±0.848	139	5.296±0.558	131	1.228±0.340		
91	5.075±0.323	140	5.935±0.434	132	1.953±0.451	132	0.765±0.327

Figure Captions

Fig. 1: Independent yield distributions of Rb, Sr, Cs and Ba in ^{235}U (n_{th},f) measured in the present work compared with the normal distributions.

Fig. 2: Independent yield distributions in ^{232}Th (n_F,f) measured in the present work compared with the normal distributions.

Fig. 3: Half-lives measured in the present work compared with calculated values[13].

Fig. 4: Odd-even effect in different fissioning systems. Z_F is the atomic number of the fissionable nuclide.

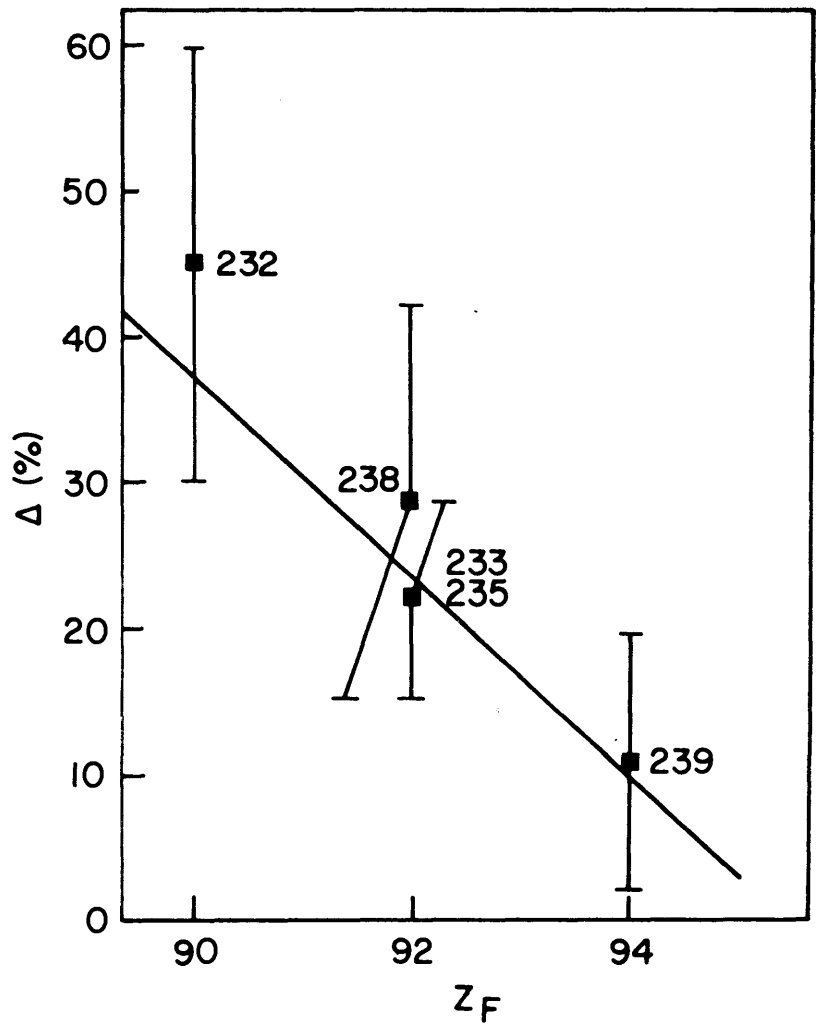

Fig. 4: Odd-even effect in different fissioning systems. Z_F is the atomic number of the fissionable nuclide.

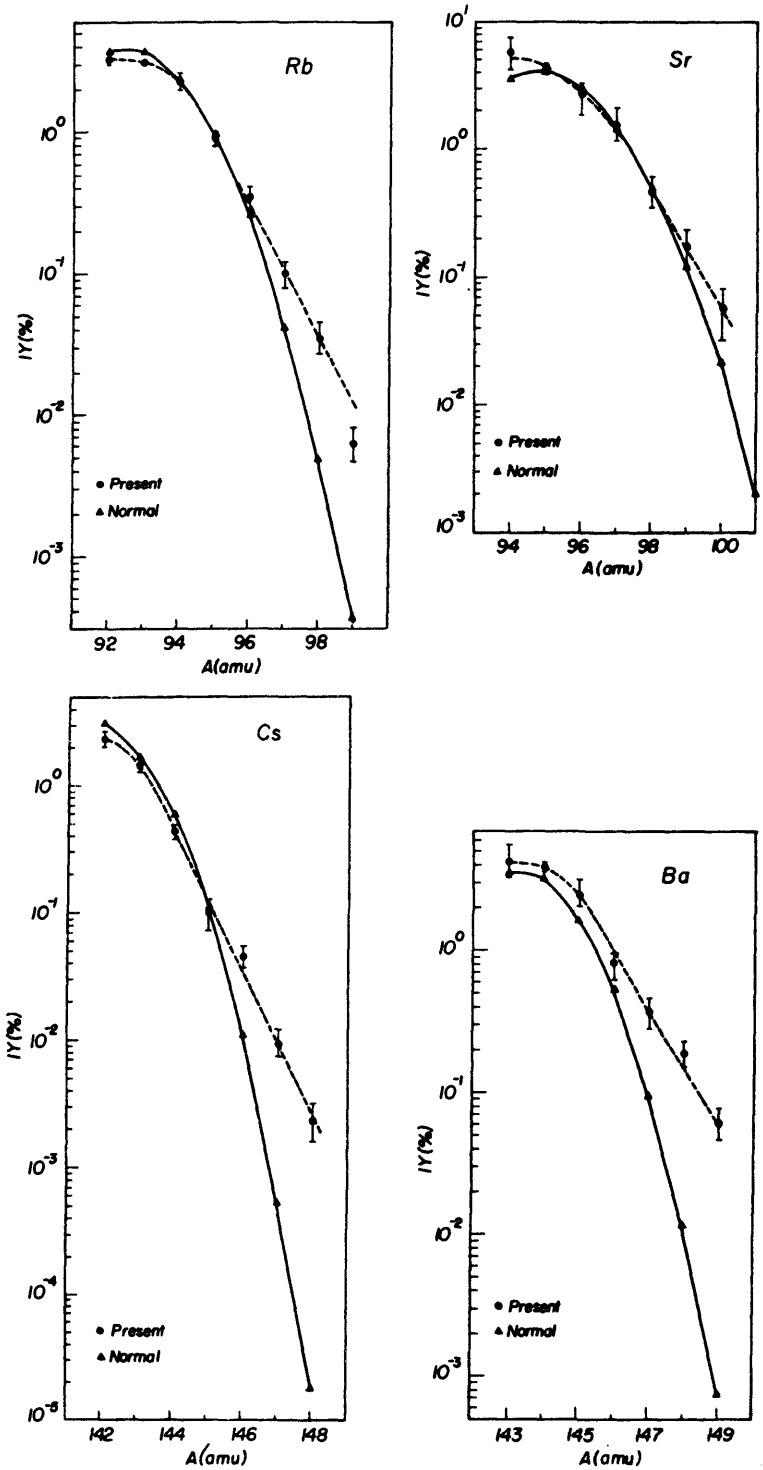

Fig. 1: Independent yield distributions of Rb, Sr, Cs and Ba in ^{235}U (n_{th},f) measured in the present work compared with the normal distributions.

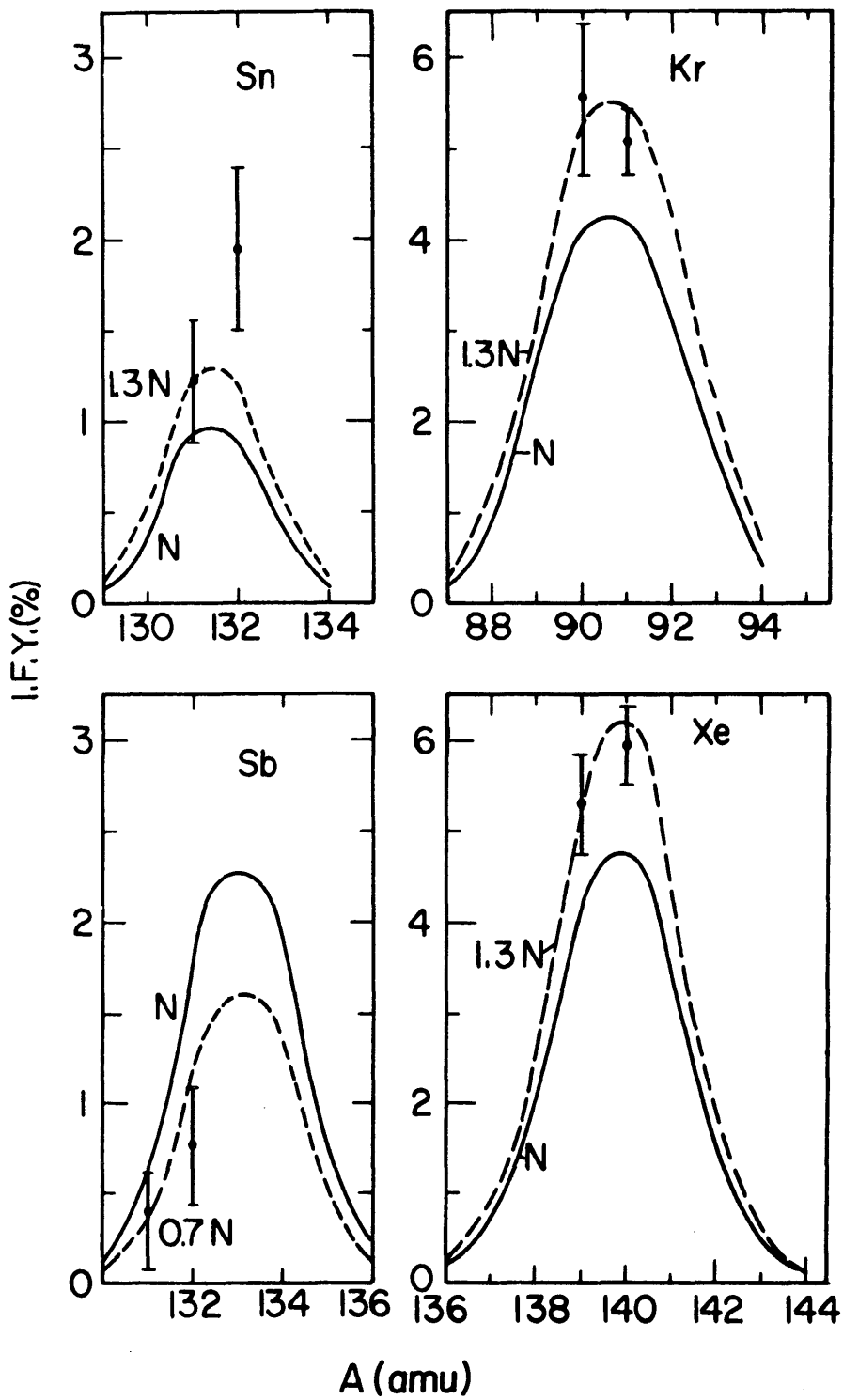

Fig. 2: Independent yield distributions in ^{232}Th (n_F,f) measured in the present work compared with the normal distributions.

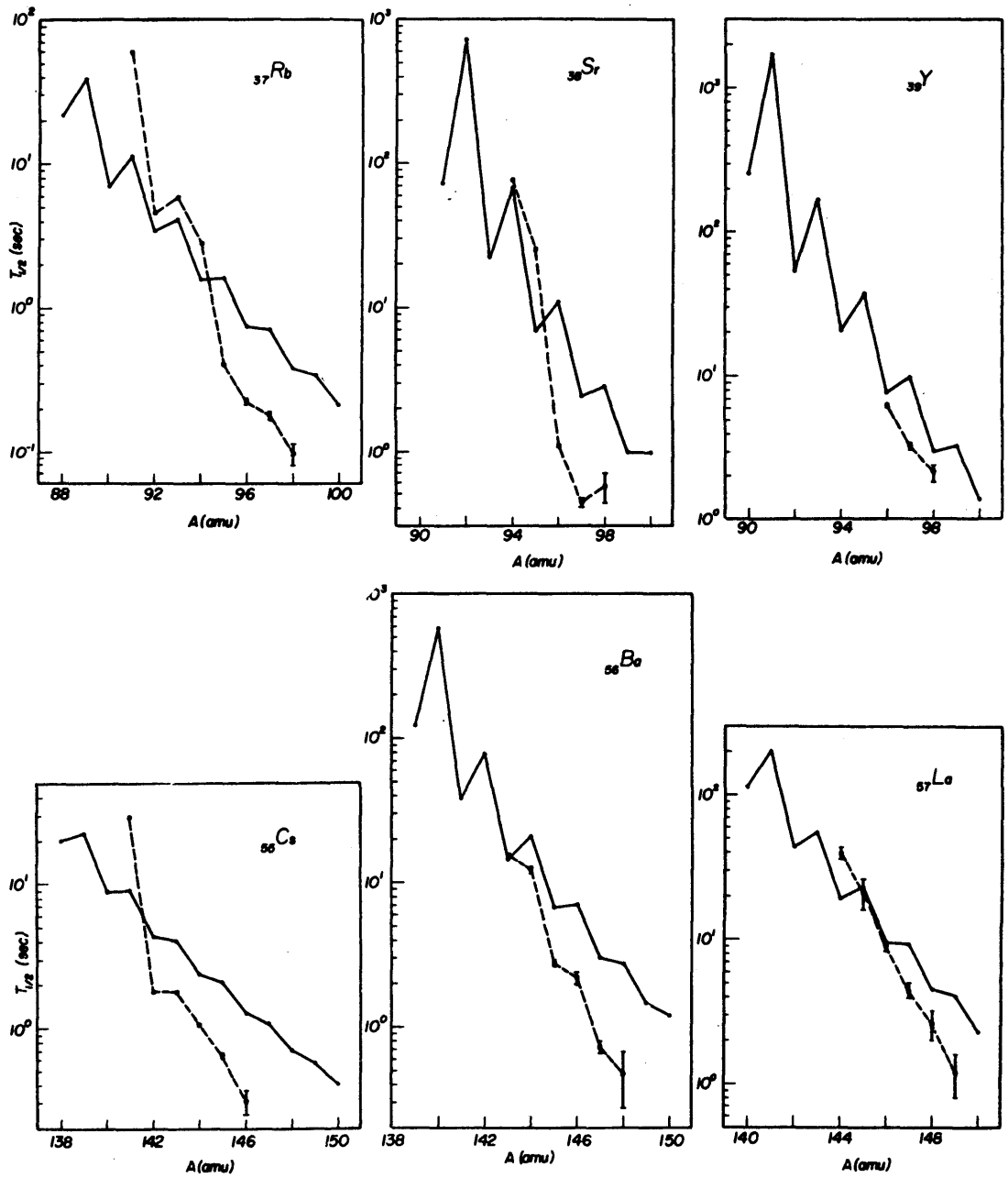

Fig. 3: Half-lives measured in the present work compared with calculated values[13].

Contribution to Review Paper 10

STATUS REPORT ON FAST REACTOR FISSION YIELDS IN THE TACO EXPERIMENT

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Abstract
Fast reactor fission yields for ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu have been measured for the heavy mass range. The analytical procedure, the evaluation method and the accuracies of the determinations are given. The results are supported by an observed systematic shift in the mass-yields.

In the frame of the TACO Experiment*, fissile nuclides have been irradiated in the fast reactor Rapsodie for the purpose of measuring their cumulative fission yields. The fission source material (^{232}Th, ^{233}U, ^{235}U, ^{236}U, ^{238}U, $^{239-242}Pu$, ^{237}Np, ^{241}Am, ^{243}Am) was separately encapsulated, inserted into standard rods of a fuel bundle and irradiated up to an approximate integrated flux of 10^{23} n/cm^2 during 1970/71.

ANALYTICAL PROCEDURE:

The mass distribution of the fission products was obtained by the measurement of selected fission products lying on the heavy hump of the yield curve and covering about 70% of the fission products.

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*The TACO (TAux de COmbustion) Experiment was conducted in cooperation with the CEA, France and included, as well as cumulative fission yield measurement, the determination of integral neutron cross-sections in heavy nuclides and fission products.

The fission gas released from the target but contained in the irradiation capsule was determined by isotope dilution analysis. Under vacuum the capsule was opened by means of a laser beam and the freed gas mixed with a spike of natural xenon by cryo-pumping.

The fission gas retained in the target was liberated during dissolution and again measured by the isotope dilution technique [1].

In the dissolved sample, ^{125}Sb, ^{137}Cs and ^{144}Ce were measured directly by γ-spectrometry. Using a mass-spectrometric isotope-dilution technique the isotopic concentrations of Cs, Nd, Sm Eu and Gd were determined [2]. The determination of Ba and Ce failed due to heavy contamination with the naturally occuring isotopes. A simple separation scheme was used to condition the sample for the purpose of the mass spectrometric analysis (see fig. 1).

EVALUATION PROCEDURE:

The fission yields were obtained by normalizing the fission product nuclide abundance in the sample to 200 percent. As only nuclides on the heavy hump of mass distribution were measured the normalisation condition is:

$$n \sum_{l=a}^{\leq 160} N_l = 100\%$$

(n = normalisation factor, N_ℓ = fission product abundance, a = the mean mass number obtained from $a = (N_f + 1 + \nu)/2$, N_f = mass number of fission source and ν number of emitted neutrons).
The missing nuclides were obtained by interpolation using thermal fission yields (if available) or estimated yields as illustrated in the case of ^{235}U [3].

For the masses 125, 131 to 137, 143 to 153 the averaged results (as obtained by analyses of two irradiated ^{235}U capsules) were completed for the masses not measured by using the recommended thermal yields [4] and fast fission yields [5]. The difference for the two interpolations amounts only to 0.7 % for the yields of the measured nuclides. Nevertheless we relied rather on the thermal yields because the difference in the two sets are mainly caused by Ba and Ce nuclides, which - in our experience - are extremely difficult to measure by mass spectrometry. However this technique was used to determine the fast yields (fig. 2).

RESULTS AND DISCUSSION:

The analyses of the nuclides ^{235}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu are complete. The results have been corrected for contributions to the fissions from grown-in daughter products for ^{239}Pu, ^{240}Pu and ^{241}Pu. The latter contained a significant amount of ^{241}Am, for which the fission yield determinations have not yet been finished. The same situation also holds for ^{243}Am and ^{236}U. However, major changes due to the evaluation procedure in the present data sets for ^{235}U, ^{241}Pu and ^{242}Pu are not expected. The influence of neutron capture of the fission products themselves to their mass distribution appears negligible. Measurements aimed at the determination of fission product cross-sections under the conditions of the experiment are underway.

The results obtained derive from the analysis of two independent irradiations of each nuclide (exept for ^{239}Pu where three capsules were irradiated). For each capsule the fission products were determined in at least three assays (except for the fission gases), from which the results are averaged (table 1).

The results obtained in the TACO experiment compare poorly with the recommended data [4]. For ^{235}U and ^{239}Pu the new data obtained by W.J. Maeck [5] match within the data accuracy. For ^{240}Pu, ^{241}Pu and ^{242}Pu only estimations of Sidebotham [6]

exist. Here an agreement is observed for the thermal yields. As the data for ^{235}U and ^{239}Pu did not show a great difference from the revised thermal ones [7], the agreement with "thermal" estimations is not surprising. In this case, however, the term "thermal" should be more exactly replaced by "thermal and fast reactor", which makes sense of the fissionability of nuclides such as ^{240}Pu and ^{242}Pu [8].

The Pu fission sources, having the same proton number, show a steady mass shift in the yield distributions, which is different from nuclides with another number of protons (^{235}U, ^{238}U, ^{237}Np, ^{241}Am). This is clearly illustrated in the case of nuclides lying on the heavy side of the mass yield curve (fig. 3). The same systematics cannot be observed for all measured fission products of the Pu-sources (fig. 4).

Lit.

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MASS	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{235}U
125	0.10±12 %	0.08± 62%	0.09± 11%	0.06± 16%	0.06±9.4%
(126)	0.20±2.9%	0.22± 32%	0.16±9.4%	0.20± 0%	0.10±0.5%
(127)	0.48±3.2%	0.43±9.3%	0.30±8.3%	0.34± 16%	0.20±0.3%
(128)	0.82±1.9%	0.82± 27%	0.57± 7%	0.53± 11%	0.49±0.1%
(129)	1.50±3.7%	1.46±8.9%	1.07± 9%	1.06± 10%	1.09± 0%
(130)	2.52±3.5%	2.49±2.4%	2.26±1.6%	1.91±2.4%	2.00±0.2%
131	4.23±2.9%	4.03±5.1%	3.74±1.9%	3.15± 4%	3.21±0.6%
132	5.73±2.7%	5.68±1.5%	5.07±1.8%	4.38± 2%	4.68±0.4%
133	7.21±1.2%	6.87±0.5%	6.68±0.7%	6.43±0.3%	6.85±0.7%
134	7.83±3.5%	7.99±2.3%	7.88±1.8%	7.25±0.3%	7.58±0.3%
135	7.60±1.4%	7.23±0.8%	7.22±1.3%	7.04±3.5%	6.72±0.2%
136	7.04±1.1%	6.68±2.7%	6.80±0.9%	6.67±2.8%	6.19±0.4%
137	6.43±1.0%	6.32±1.7%	6.31±0.9%	6.27±2.4%	6.27±0.3%
(138)	5.55±1.5%	6.06±0.5%	6.36±0.6%	6.22±2.2%	6.69±0.2%
(139)	5.47±2.1%	5.89±4.7%	6.10±0.2%	6.06±2.2%	6.44±1.2%
(140)	5.31±2.1%	5.51±3.1%	5.72±0.4%	5.76±2.8%	6.06±0.2%
(141)	5.01±2.8%	5.21±3.3%	4.68± 1%	5.23±0.4%	5.57±0.4%
(142)	4.69±3.7%	4.70±2.2%	4.61±1.4%	4.62±0.6%	5.62±0.3%
143	4.27±1.4%	4.20±0.7%	4.38±0.7%	4.46±0.5%	5.67±0.3%
144	3.58±0.6%	3.54±3.7%	4.00±2.2%	4.14±0.7%	5.17±1.1%
145	2.97±1.6%	2.93±0.5%	3.12±0.3%	3.31±0.6%	3.73±0.4%
146	2.44±1.3%	2.45±0.8%	2.65±0.4%	2.87±0.5%	2.91±0.4%
(147)	2.03±0.7%	1.99±0.5%	2.17±0.7%	2.35±0.4%	2.11±2.1%
148	1.64±1.3%	1.68±0.3%	1.84±0.5%	1.97± 0%	1.67±0.6%
149	1.25±0.5%	1.32±0.7%	1.43±0.7%	1.63±0.6%	1.02±1.0%
150	0.97±1.6%	1.01± 0%	1.15±0.9%	1.30± 0%	0.68±0.8%
151	0.79±3.1%	0.85±1.8%	0.90±3.3%	1.02± 2%	0.41±2.5%
152	0.64±0.9%	0.67±5.3%	0.71±1.4%	0.85±1.2%	0.24±5.7%
153	0.47±6.4%	0.46± 21%	0.53±3.8%	0.65±1.5%	0.16±8.2%
154	0.31± 14%	0.34± 26%	0.38±5.3%	0.47±2.1%	0.07± 0%

Table 1: Cumulative fission yields for $^{239-242}Pu$ and ^{235}U averages and percent deviation.

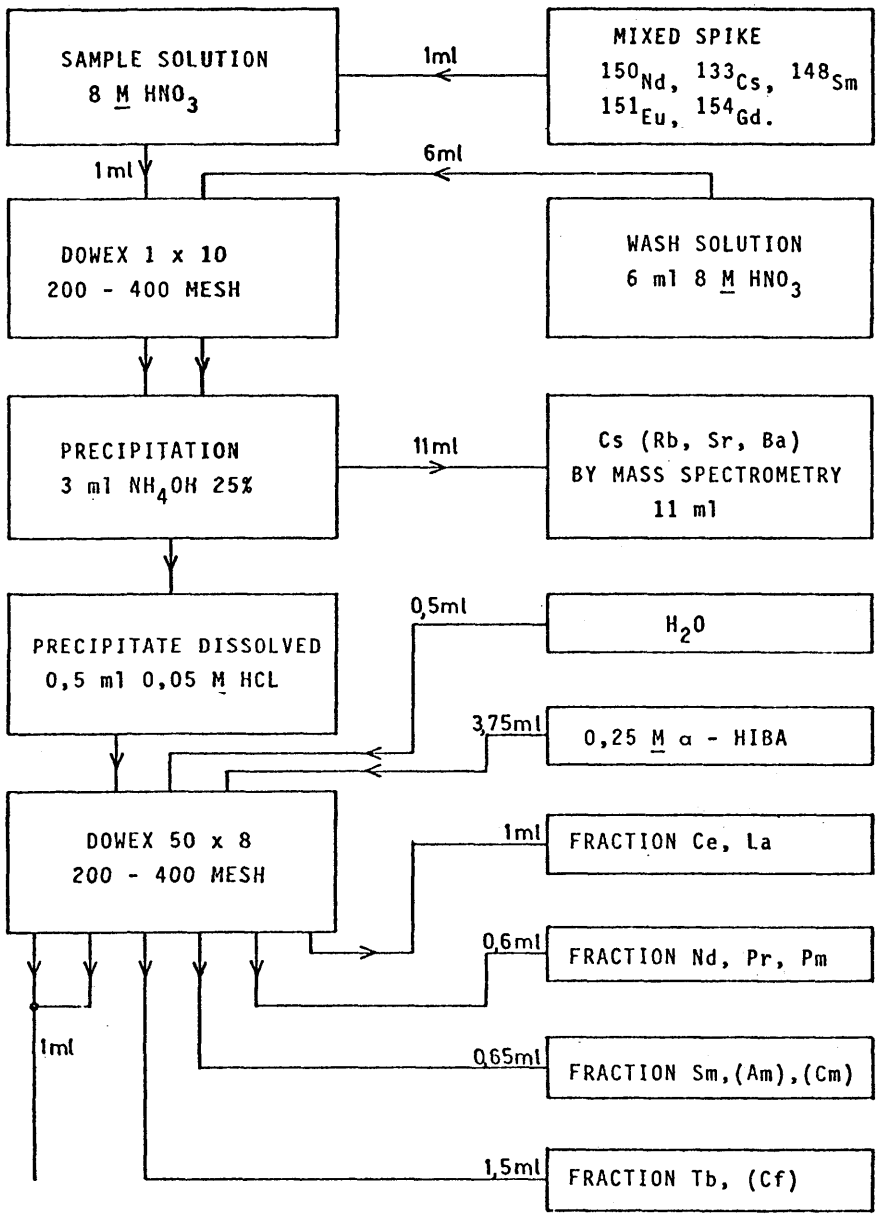

Fig. 1: Chemical conditioning of samples for isotope dilution mass spectrometry.

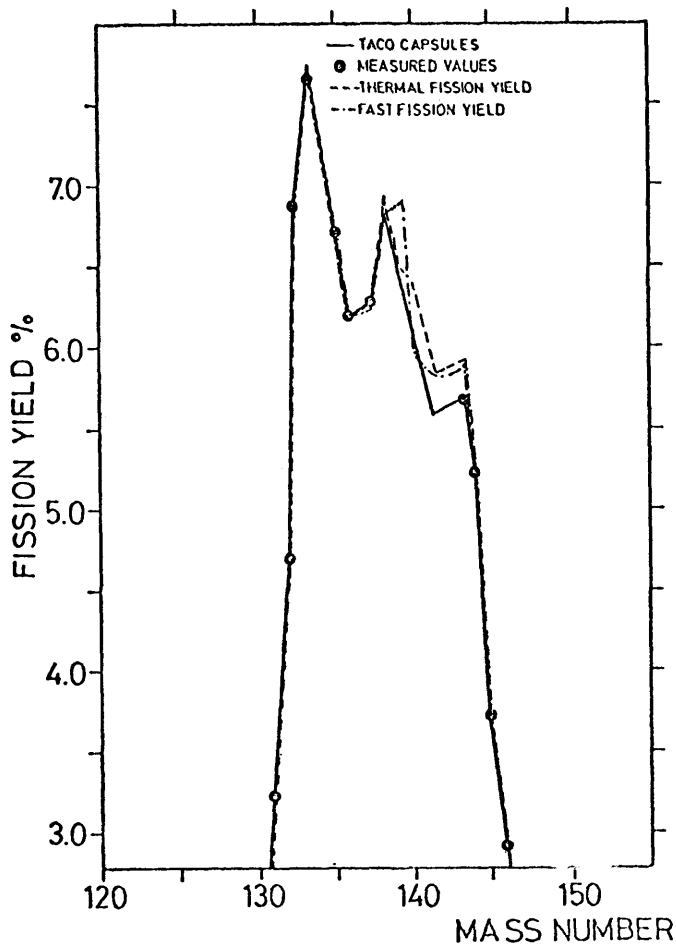

Fig. 2: Mass distribution in heavy nuclides in ^{235}U fast reactor fission and interpolation with recommended data.

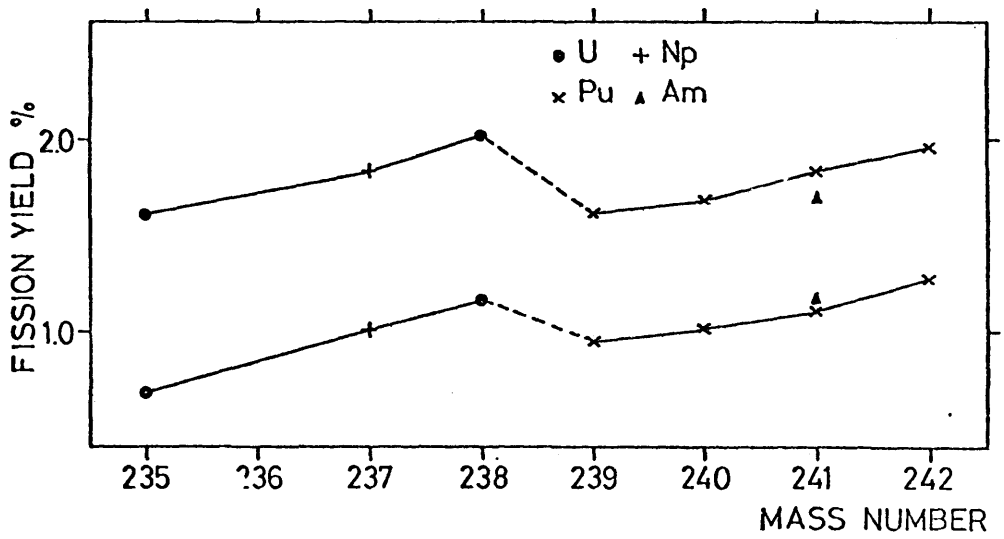

Fig. 3: Systematic mass shift of ^{148}Nd and ^{150}Nd yields for different fission sources.

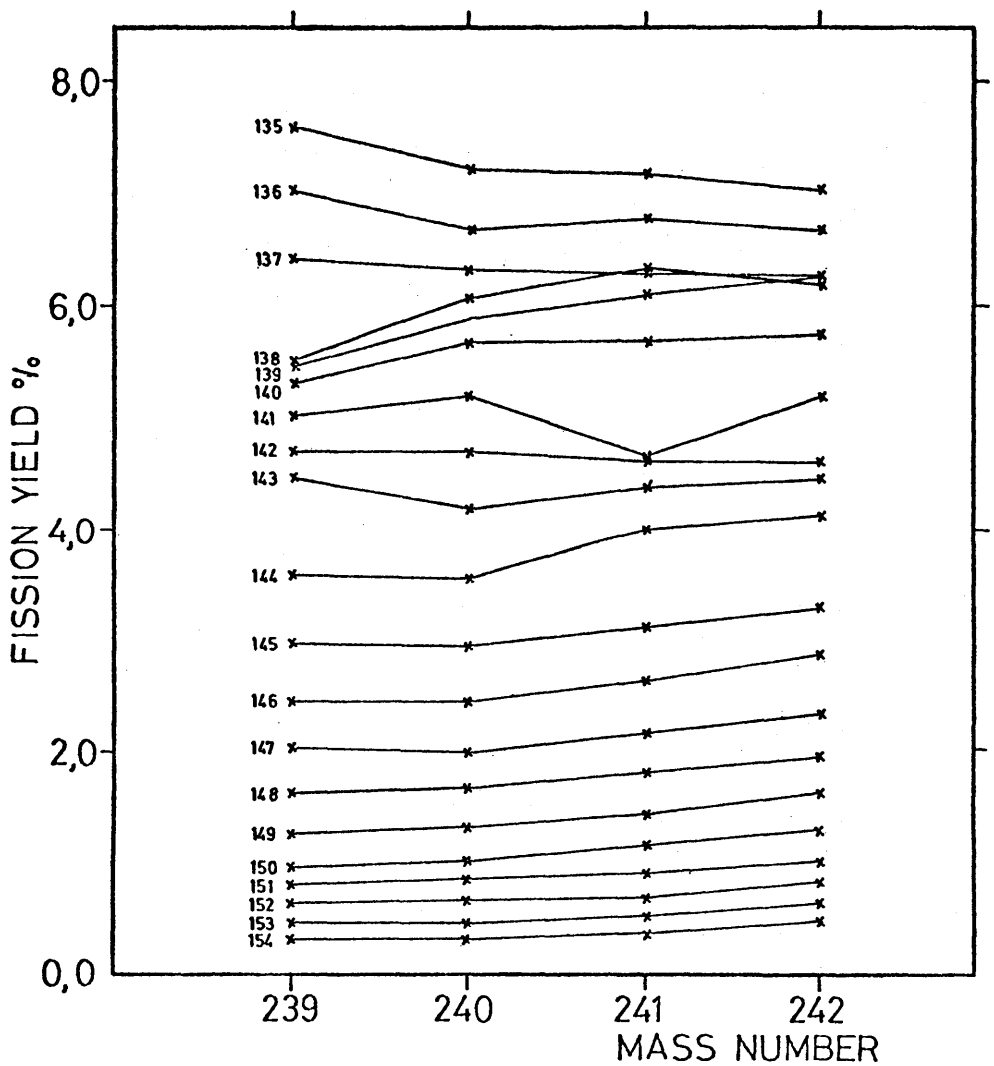

Fig. 4: Mass shift of cumulative yields for Pu-fission-sources in a fast reactor.

Contribution to Review Paper 10

FISSION PRODUCT YIELDS IN ^{235}U FISSION BY FAST AND THERMAL NEUTRONS

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Abstract: Relative yields of fission products in ^{235}U fission induced by thermal and ^{252}Cf spontaneous fission neutrons were determined by means of γ-ray spectrometry. Fast-to-thermal yield ratios are given for 20 fission products. In some cases, in particular on the wings of the mass yield curve, our results differ from those obtained with reactor fast neutron spectra.

This work has been started with the aim to contribute to the investigation of the energy dependence of so-called fast fission yields. Most of the data reported in the literature have been obtained from measurements in nuclear reactors. Since the fast neutron spectra differ for the various reactor types and for different irradiation positions within the same reactor, the evaluated fission yields in tabulations based on these measurements are valid for a rather poorly defined mean reactor fast neutron spectrum.

This does not matter, perhaps, for fission products in the maxima of the mass yield curve where fission yields are not expected to vary considerably. In the valley and on the wings of the curve, however, a clear energy dependence has been observed.

We have determined relative fission product yields in ^{235}U fission by neutrons emitted in ^{252}Cf spontaneous fission. The ^{252}Cf source and the ^{235}U samples (discs of 10 mm diameter and 0.2 mm thickness, enriched to 93 % ^{235}U) were mounted in a low-scattering open-air mast construction /1/. The ^{252}Cf neutron spectrum is known quite well, and it is not expected to be considerably distorted by the source mounting materials. A total of six irradiations lasting

from two hours to twelve days were performed. In addition three samples were irradiated in the thermal neutron spectrum of the PTB reactor FMRB. The fission rate was of the same order of magnitude as for the ^{252}Cf irradiations.

Gamma-ray spectra of the irradiated samples were repeatedly taken with a Ge(Li)-spectrometer (active volume 70 cm^3, resolution 2.2 keV at 1.33 MeV) at cooling times between half an hour and 100 days. Because the number of fissions was not determined absolutely, only relative cumulative yields could be derived from the number of counts in the peaks corresponding to the respective fission products. Any results showing a dependence of the yield on the irradiation time or on the cooling time (due, for example, to another nearby γ-ray) were eliminated.

In Table 1 we have given the "fast/thermal" yield ratios with respect to ^{95}Zr along with the chain yield ratios taken from the Meek and Rider tables /2/. For the fission products under study the chain yield ratios are not expected to differ from the cumulative yield ratios. The ratio for ^{95}Zr in column 2 has been set equal to that in column 3. Our results represent the mean of all values obtained from the different irradiations and, in some cases, from several prominent lines. When forming fission yield ratios, any errors due to γ-ray self-absorption in the sample, to the efficiency calibration, to γ-ray emission probabilities and to real summing corrections cancel. Dead time and pile-up counting losses were accounted for by the pulser method. The fast yields were corrected for contributions of ^{238}U fission. Uncertainties quoted in the Table are mainly of statistical nature and are valid at a confidence level of about 68 %.

Our results reflect the energy dependence of fission yields in particular on the wings of the mass yield curve (e.g. mass numbers 103, 105, 131, 132). A comparison with the Meek and Rider values in column 3 of the Table makes evident that ^{235}U fission yields in reactor neutron spectra clearly differ from the yields in the harder neutron spectrum of ^{252}Cf.

References

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(2) M.E. Meek, B.F. Rider: NEDO-12154-1, January 26, 1974.

Table I: Ratios of fission product yields Y from ^{235}U fission by fast neutrons (Cf: ^{252}Cf fission neutrons, r reactor spectrum neutrons) to the yields Y^{th} from thermal neutron induced fission (Y_c cumulative yield, Y_t total yield). The uncertainties quoted in column 2 are mainly statistical ones and relate to a confidence level of 68 %.

Fission product	$\left[Y_c^{Cf} / Y_c^{th}\right]_{rel}$ *) Debertin, 1977	Y_t^r / Y_t^{th} Meek and Rider, 1974
^{87}Kr	0,94 ± 0,04	0,950 ± 0,021
$^{88}Kr/Rb$	1,010 ± 0,015	0,992 ± 0,022
$^{91}Sr/Y^m$	0,922 ± 0,010	0,946 ± 0,021
$^{92}Sr/Y$	0,966 ± 0,010	0,961 ± 0,021
^{93}Y	0,950 ± 0,018	0,958 ± 0,021
$^{95}Zr/Nb$	0,990 ± (0,010) **)	0,990 ± 0,012
$^{97}Zr/Nb$	1,016 ± 0,007	1,004 ± 0,020
^{99}Mo	1,013 ± 0,016	0,930 ± 0,016
^{103}Ru	1,099 ± 0,007	1,051 ± 0,030
^{105}Rh	1,264 ± 0,021	1,17 ± 0,14
^{127}Sb	2,95 ± 0,09	2,42 ± 0,32
^{129}Sb	1,54 ± 0,10	1,58 ± 0,16
^{131}I	1,258 ± 0,019	1,143 ± 0,023
$^{132}Te/I$	1,136 ± 0,008	1,106 ± 0,019
^{133}I	1,000 ± 0,008	0,954 ± 0,015
^{134}Te	0,93 ± 0,04	0,992 ± 0,023
$^{135}I/Xe$	1,013 ± 0,008	0,946 ± 0,024
^{138}Cs	0,94 ± 0,03	0,944 ± 0,015
$^{140}Ba/La$	0,960 ± 0,009	0,954 ± 0,014
^{142}La	0,95 ± 0,04	0,922 ± 0,027
^{143}Ce	0,892 ± 0,008	0,946 ± 0,011
^{147}Nd	0,97 ± 0,05	1,045 ± 0,034
^{151}Pm	1,26 ± 0,07	1,036 ± 0,024

*) normalized to the ratio for ^{95}Zr from Meek and Rider

**) not included in the uncertainties of the other ratios.

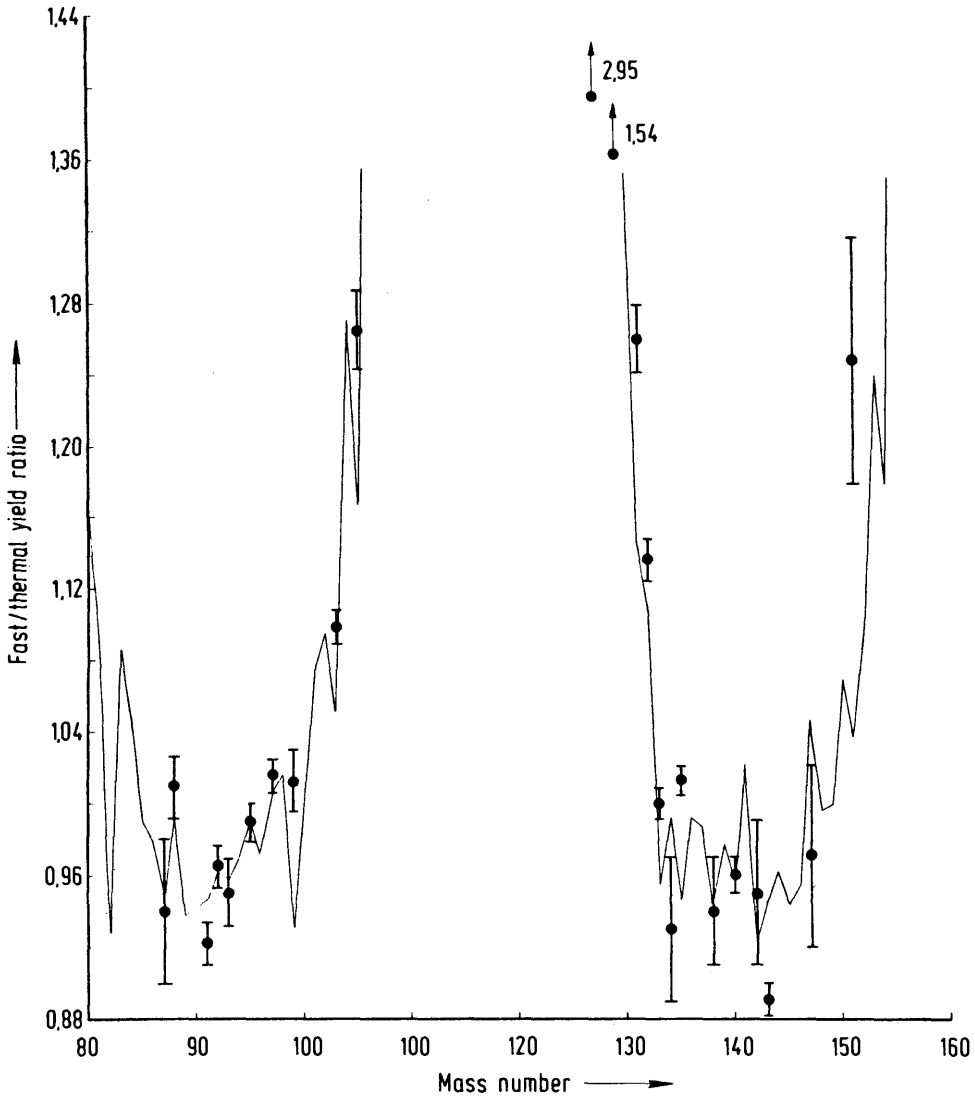

Fig. 1 Fast to thermal yield ratio for ^{235}U fission (see also Table I).

——— Y_t^r / Y_t^{th} from Meek and Rider (2)

Y_c^{cf} / Y_c^{th}, this work

Contribution to Review Paper 11

NUCLEAR-CHARGE DISTRIBUTION IN FISSION - INVESTIGATION OF SYSTEMATICS AND METHODS FOR ESTIMATION OF INDEPENDENT YIELDS*

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Abstract:

Three mathematical models for calculation of independent fission-product yields are described, and, with the parameters derived for each from the data for thermal-neutron-induced fission of ^{235}U, the three models represent the data about equally well. Two models are of conventional form with charge dispersion for each mass number, the maxima occuring at Z_p and the widths being related to σ_Z, but differ in the method of representing the proton-pairing effect. In one the effect is introduced as a constant multiplier (1.25 for even Z's and 1/1.25 for odd Z's) of independent yields derived from a Gaussian dispersion with constant values of σ_Z and $\Delta Z_p = |Z_p - Z(UCD)|$, UCD representing unchanged charge

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distribution. In the second model pairing is represented by oscillating ΔZ_p and σ_Z functions. For both models abrupt changes in values of the functions are observed near Z = 50, and the changes are attributed to the effect of the stable 50-proton shell on the fission process.

The third model describes the dispersion of fragment mass numbers for each atomic number, maxima occurring at A'_p and the widths being related to $\sigma_{A'}$, which is nearly constant. For this model an abrupt change in element yield ($\gtrsim$40 fold) is observed between Z = 49 and 50, an effect also attributed to the stability of the 50-proton shell and related to asymmetric mass division in low-energy fission processes.

A modification of the first model described has been used to represent data for thermal-neutron-induced fission of ^{233}U and ^{239}Pu and for spontaneous fission of ^{252}Cf. The $\overline{\sigma_Z}$ values for the various fission processes are nearly constant, and $\overline{\Delta Z_p}$ values vary only a little, being largest for ^{239}Pu. However, the proton-pairing effect, which is prominent in the fission of both ^{235}U and ^{233}U, is essentially absent in the fission of ^{239}Pu and ^{252}Cf.

The possibilities and limitations of using the models for estimation of unmeasured independent yields is discussed, and it is pointed out that uncertainties in the estimates can be quite large.

1. INTRODUCTION

The distribution of nuclear charge in fission is of both theoretical and practical interest. Independent (or primary) yields of fission products are related to the probabilities of various mass and charge divisions and to the dependence of prompt-neutron yields on mass and atomic numbers, excitation energy, etc. of a fissioning nucleus so independent yields and/or systematic methods of representing them are useful for comparisons with theoretical predictions. Independent yields are also useful for interpretation of delayed-neutron yields and for determination of delayed-neutron emission probabilities. In addition, independent yields are useful in reactor design, e.g., for calculation of heating effects from fission-product decay following a reactor shut down.

Most fission products decay rapidly and/or are rarely formed, so independent yields are generally difficult to measure and are available for only a small fraction of products from the many fission processes that occur, even for thermal-neutron-induced fission of ^{235}U, for which by far the most information is available. Therefore, it is desirable to deduce systematic trends from the independent yields that have been measured in order both to understand better the mechanism of nuclear fission and to estimate as well as possible the unmeasured yields.

The method of estimation of independent yields described in this article involves derivation of models, involving simple mathematical functions,

from the data available for thermal-neutron-induced fission of ^{235}U and testing how well each model represents tha data. Then a suitable model is chosen for use with other fission processes, and the mathematical functions are fitted to the available data for each process to determine the values of the parameters for the model. The magnitude of the parameters and their variation with mass and atomic numbers and excitation energy of a fissioning nucleus could give information about the mechanism of nuclear-fission processes and could also allow extrapolation of the parameters to fissioning systems for which few or no data exist.

This article reports the current status of a continuing investigation of nuclear-charge distribution in fission; previous articles [1,2] describe some of the procedures that have been developed and are still used. The fission processes discussed in this article include thermal-neutron-induced fission of ^{235}U, ^{233}U, and ^{239}Pu and spontaneous fission of ^{252}Cf, the symbols [3] used for these processes being U235T, U233T, PU239T, and CF252S, respectively. It is hoped that discussion of these systems will illustrate some uses of the methods described and, as the investigation continues, that the methods can be improved and extended to still other fissioning systems.

The conventional description of nuclear-charge distribution in fission involves for each mass number (A) a charge-dispersion curve with a maximum at charge Z_p, which is usually non-integer. For many mass numbers the curve is approximately Gaussian in shape [1,2,4] with a full-width at half maximum of $\sim$1.5 charge units, but the experimental data for U235T

are better represented if the independent yields of fission products with even atomic numbers (Z) are increased and those with odd Z's are decreased from the Gaussian values [2,4,5,6] . The magnitude of this even-odd-proton effect is ~25 %, on the average, and the magnitude of the corresponding even-odd-neutron effect is smaller by at least a factor of three [6] , probably, at least in part, because the primary effect is dispersed by prompt neutron emission. Values of Z_p for light fission products are larger, and those for heavy products are smaller than would be expected if protons and neutrons were distributed uniformly between the two fragments (i.e., unchanged charge distribution - symbol: UCD); for U235T the average deviation from UCD is ~0.5 charge units [2,4] .

The effects discussed above deduced [2,4,5,6] from radiochemical and mass-spectrometric data for U235T are generally supported by the data available for several other low-energy fission processes [4,6,7,8] and by X-ray measurements [9] for U233T, PU239T, and CF252S. The effects were also observed for light fission products from U235T with near average kinetic energies and ionic charges separated according to mass number, ionic charge, and kinetic energy by the LOHENGRIN fission-product separator at Grenoble [10,11] . Also, the complete set of independent yields for these selected fission products (a very small fraction of the total for each A and Z with independent yields $\gtrsim$~0.1 %) allowed additional observations to be made. (a) The slope of the $\overline{Z(A)}$ function oscillates in such a way that $\overline{Z(A)}$ values tend to remain close to even-Z values, and this effect is especially large at Z = 42, which is complementary to Z = 50. (b) Charge dispersion curves vary in width and deviate from the Gaussian shape, the

widths and deviations being strongly correlated with the even-odd-proton effect and to a lesser extent also with nuclear-shell and even-odd-neutron effects. (c) The magnitude of the even-odd-proton and -neutron effects vary with A.

2. COMPILATION AND SELECTION OF DATA

Radiochemical and mass-spectrometric data were obtained from several compilations [2,3,7,8,12] and from recent literature, unpublished reports and theses, and personal communications. Some data were not included because the method of measurement had not been adequately tested and/or results were in conflict with other data that appeared to be more reliable. Thus no results from hot-atom chemical experiments, from mass-separation of bromine and iodine fission products, or from the Munich fission-product separator were included. Data for U235T [10,11] from the fission-product separator, LOHENGRIN, for fission products with near average kinetic energies and ionic charge states were included, after comparison with radiochemical and mass-spectrometric data as discussed below.

A qualitative comparison [13] of radiochemical and LOHENGRIN data indicated general agreement, so it seemed probable that the complete sets of 'partial' yields (since measurements were made for only a very small fraction of each fission product) from LOHENGRIN experiments were reasonably representative of 'total' yields as measured by radiochemical and mass-spectrometric techniques. A detailed comparison of fractional

independent yields (FI) for 45 light fission products for which both 'total' and 'partial' yields were available showed for both types of LOHENGRIN measurements [10,11] , that ~65 % of the values agreed within the sum of the reported experimental uncertainties, that ~25 % of the values differed by 1 to 2 times the sum of the uncertainties, and that ~10 % of the values differed by more than 2 times the sum of uncertainties. About one-half of the differences exceeding the sum of experimental uncertainties were for small yields, the LOHENGRIN values generally being larger than the radiochemical values. It was concluded that LOHENGRIN 'partial' yields could be used to represent 'total' yields, but with increased uncertainties. The estimated uncertainty was the larger of ±0.04 or 15 % of a value; thus, for the average of two yields, one obtained by each method [10,11] , the estimated uncertainty was the larger of ±0.03 or 10 % of the average value. For large values a somewhat smaller uncertainty was derived from the smaller uncertainties for yields of isobars, since the sum of FI values for each A must be one. Further evidence that the 'partial' yields are resonable approximations of 'total' yields is given in section 6, where it is shown that essentially the same average values for parameters associated with a model are derived with or without the inclusion of LOHENGRIN data.

The yield value and uncertainty selected for each fission product was consistent with all data considered to be reliable; where two or more

data were not in agreement, an intermediate value was selected and a large uncertainty was assigned. It was also required that the sum of FI values for each A be one, within assigned uncertainties and usually within ± 0.01[1].

3. CALCULATIONS

Two basic models were considered: (a) the conventional one discussed in the introduction involving charge dispersion as a function of A, and (b) a model involving mass-number dispersion as a function of Z. It was hoped that comparison of the two models would reveal strengths and weaknesses of each and perhaps reveal or emphasize the more important factors to be considered in describing nuclear-charge distribution in fission.

It was desirable, especially, when few data were available, to establish the complementarity between light and heavy fission products in order that data for both could be used in the same mass-number region, a technique that was used earlier [1,2] . For this purpose Terrell's method [14] was used, as before, but with recent mass-yield data [15] to derive the total

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[1] A limited number of copies of the compilation of experimental data and selected yields in the form of computer output will be available fairly soon for those who have current need for the compilation. A planned future publication will include the compilation and estimated yields from model calculations.

number of neutrons, ν_T, emitted to form products with complementary mass numbers. An estimate of the average mass-number (A') of the primary fragment precursors of each final product was obtained by multiplying ν_T by a ratio, $R = (\nu_p/\nu_T)$, derived by Musgrove, Cook, and Trimble [4] from data obtained by physical measurements of the average number of neutrons (ν_f) emitted by primary fragments from U235T [16,17].

$$A' = A + R\nu_T \qquad (1)$$

Because of lack of better information, the same ratio R was also used for fission processes other than U235T. It should be noted, however, that although an incorrect R value would result in poor estimates of A', it does not destroy the complementarity derived since the relationship, $R(A_L) = 1\text{-}R(A_H)$, was used in which $A_L = A_F\text{-}A_H\text{-}\nu_T$, the subscript L, H, and F refering to light, heavy, and fissioning nuclei, respectively.

Dispersion in both Z and A were assumed to be Gaussian in shape, but modified by even-odd-proton, EOZ(A), and even-odd-neutron, EON(A) or EON(Z), factors, and equations essentially equivalent to the following were used for calculations of FI and IN (independent-yield) values[2].

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[2]The ratio IN(A,Z)/FI(A,Z) = Y(A), the mass or chain yield for A, Y(A) values being relatively well known [3,12]. Also, fractional-cumulative yields, $FC(A,Z) = \sum_{0}^{Z} FI(A,Z)$, and cumulative yields, $CU(A,Z) = \sum_{0}^{Z} IN(A,Z)$, are also sometimes measured and are readily interconverted: CU(A,Z)/FC(A,Z) = Y(A).

$$FI(A,Z) = \frac{[EOF(A)][NF(A)]}{[\pi c_{A'}(Z)]^{1/2}} \exp\left\{[Z-Z_p(A)]^2/c_Z(A)\right\} \qquad (2)$$

$$IN(A,Z) = \frac{[EOF(Z)][Y(Z)]}{[\pi c_{A'}(Z)]^{1/2}} \exp\left\{[A'-A'_p(Z)]^2/c_{A'}(Z)\right\} \qquad (3)$$

In Eq. 2 $EOF(A) = [EOZ(A)][EON(A)]$ if both Z and N are even or the reciprocal of the appropriate factor if Z and/or N is odd. The symbol NF(A) in Eq. 2 is a normalization factor required because application of even-odd factors destroys the inherent normalization property of a Gaussian distribution, e.g., $FI(A) = 1.00$ [4]. In Eq. 3 the even-odd factor $EOF(Z) = EON(Z)$ if the neutron number (N) is even or $1/EON(Z)$ if N is odd. The normalization factor cannot be separated from the element yield, Y(Z), a parameter that must be determined or estimated for each pair of complementary Z's, since Y(Z) values are not accurately known.

Calculations were made by the method of least squares using the general least-squares program, ORGLS, of Busing and Levy [18] after suitable modifications had been made. The Gaussian equations used in the program were in cumulative from [1] and, therefore, involved error rather than exponential functions and the width parameter[3], σ, which is related to c approximately through Sheppard's correction.

$$\sigma \simeq (c/2 - 1/12)^{1/2}$$

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[3]The 'σ' values reported with the later LOHENGRIN results [10,11] are the square root of the second moment of the charge distribution and are applicable to any distribution; if the distribution were Gaussian, however, this 'σ' = $(c/2)^{1/2}$, i.e., it would differ from the σ discussed in this article by not including Sheppard's correction.

When normalization was necessary, two calculation subroutines were used in each cycle of the least-squares iteration procedure, the first to calculate the normalization factor appropriate for the parameter values being used in that cycle, and the second to calculate FI and FC values for comparison with the data.

In the initial phase of the current investigation, when models were being chosen and parameters for them were being derived from U235T data, each A or Z was considered separately. Later, before extending the investigation to other fission processes, for which too few data exist to allow this approach, computer programs were modified to allow derivation directly in one calculation of all the parameters for a model (e.g., $\overline{\Delta Z_p} = \overline{|Z_p - Z(UCD)|}$, $\overline{\sigma_Z}$, $\overline{EOZ}$, and $\overline{EON}$) from data for all mass numbers, expect those near symmetric mass and charge division, where few data exist and require special treatment, as will be discussed in sections 7 and 8.

Very small values of FI or IN, although often measured reliably and accurately, were of limited use for deriving and testing the simple models to be discussed, partly because in using the conventional least-squares weighting factors ($1/(\text{error})^2$) the very small absolute errors gave undue weight to the small values. However, a more fundamental reason is that most very small yield values are for fission products near the region of beta stability and are formed by neutron emission from primary fission fragments with very much higher yields, so that even small differences in primary yields or in neutron-emission probabilities have large effects on the very small yields of the final products. Thus, small yields for fission

products near beta stability cannot be estimated accurately by use of the simple models to be discussed, and, conversely, they are of little use in deriving the models. Therefore, one of the following two procedures was followed: (a) Only FI or FC values ≥ 0.01 were used. (b) All values were used, but the error was taken to be the largest of (i) the experimental uncertainty, (ii) the estimated uncertainty assigned for use of 'partial' yields from LOHENGRIN as 'total' yields, (iii) the estimated uncertainty in the values calculated by use of a model, an uncertainty derived from the function,

$$ME = 0.1/(FI)^{1/3} \qquad (4)$$

which gave error values (error = +(ME)(FI) or = -(ME)(FI)/(1 + ME)) that approximated the author's expectations of how well a simple model might be able to represent data and, possibly, to allow estimation of yields[4].

4. DERIVATION OF MODELS FOR U235T

It seemed clear from a study of the LOHENGRIN results [10,11] that the oscillating natures of the Z_p or $\bar{Z}$ and 'σ' functions were due mainly to the even-odd-proton effect; however, it was not clear whether the data could be better represented by a model involving even-odd effects or by

[4]For example, the estimated model error (ME) is 11 % for FI = 0.8, 20 % for FI = 0.1, 45 % for FI = 0.01, a factor of 2 for FI = 10^{-4}, and a factor of 100 for FI = 10^{-9}.

one involving oscillating Z_p and σ functions, and it seemed possible that a more complex model involving both and/or other factors might be required. As a start in exploring this problem, two versions of the conventional model were derived from the selected data with FI(FC)> 0.01, one with constant average values, $\overline{\Delta Z_p} = \overline{\Delta Z_p(L)} = \overline{-\Delta Z_p(H)} = 0.50$, $\overline{\sigma_Z} = 0.54$, and $\overline{EOZ} = 1.25$ (called the EOZ model), and one with oscillating $\Delta Z_p(A')$ and $\sigma_Z(A')$ functions (called the OSC model). Equations for the oscillating functions are given below, and plots of the functions are shown as lines in Figs. 1 and 2.

$$\Delta Z_p(A') = 0.44 + 0.005 \left[A'(H) - (50)(236)/92\right] + 0.095 \sin \left[(Z_p(H) - 50)\pi\right] \quad (5)$$

For light fission products: $A'(H) = 236 - A'(L)$, and $Z_p(H) = 92 - Z_p(L)$.

$$\sigma_Z = 0.54 - 0.13 \cos \left[(Z_p - 50)\pi\right] \quad (6)$$

Since charge dispersion is small, there are relatively few (usually only 3 or 4) FI values ≥ 0.01 for each A, and it proved impractical to determine several parameters simultaneously. Therefore, the average charge, $\overline{Z(A)}$, was first determined for each A and assumed to be $\simeq Z_p(A)$. It was from these values that $\overline{\Delta Z_p} = 0.50$ and the parameters given in Eq. 5 were derived[5].

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[5] In later calculations, described in section 6, slopes, $\partial \Delta Z_p / \partial A'$, were found to be essentially zero rather than finite as given in Eq. 5.

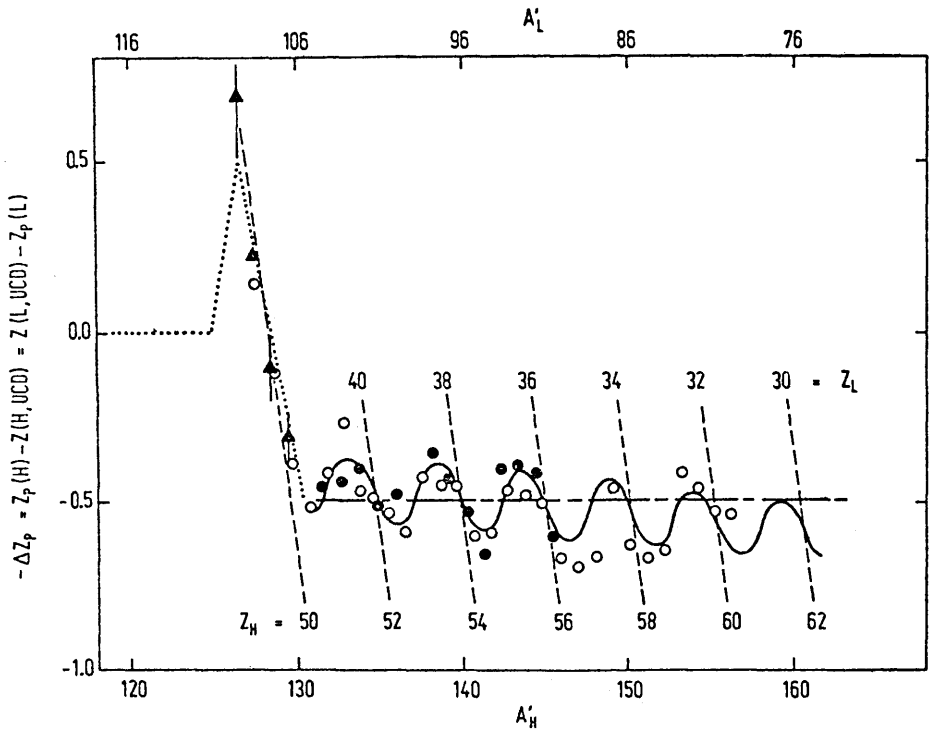

Fig. 1. Z_P functions. Symbols: O , $Z(L,UCD)-\overline{Z(L)}$; ● , $\overline{Z(H)}-Z(H,UCD)$; ▲ , Z_P calculated from FI for $^{126-9}Sb$ assuming $\sigma_Z = 0.33$ and EOZ = 1.00; ——— , OSC ΔZ_P function; ——– , EOZ ΔZ_P function; ·····, estimated ΔZ_P function; — —, lines of constant Z.

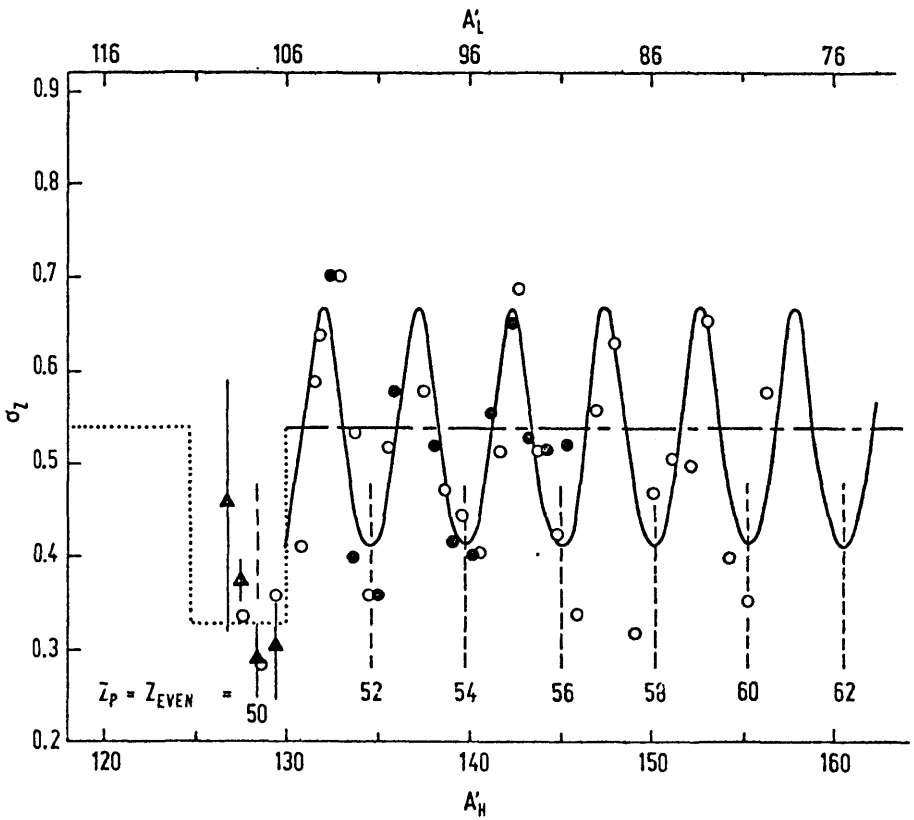

Fig. 2. σ_Z functions. Symbols: O , $\sigma_Z(L)$; ● , $\sigma_Z(H)$; ▲ , σ_Z calculated from FI for $^{126-9}Sb$ assuming estimated ΔZ_P function and EOZ = 1.00; ——— , OSC σ_Z function; ——– , EOZ σ_Z function; ····· estimated σ_Z function; — —, $Z_P = Z_{EVEN}$.

The abrupt changes in the ΔZ_p and σ_Z functions near Z = 50, 42 shown in Figs. 1 and 2, indicated that data for A = 105-129 required special treatment, which is discussed in sections 7 and 8, and these data were not used in derivation of the ΔZ_p and σ_Z functions discussed above and below, respectively.

Values of σ_Z were next calculated for each A using Z_p values calculated from Eq. 5 and EOF = 1.00. Values are plotted in Fig. 2 and were used in the derivation of $\overline{\sigma_Z}$ = 0.54 and the parameters given in Eq. 6.

For the less conventional model describing mass-number dispersion at constant Z (called the A'_p model), the dispersion is quite large, and 8 to 14 data with FI values $\geq$0.01 were available for 17 elements, 11 with $Z \leq 42$ and 6 for $Z \geq 50$. For each of these elements, values of A'_p, $\sigma_{A'}$, and Y(Z) were derived, assuming EON(Z) = 1.00.[6] Plots of $\sigma_{A'}$ and $\Delta A'_p$ are shown in Fig. 3, and a plot of Y(Z) is shown in Fig. 4. Results for complementary elements are reasonably consistent, and values of $\sigma_{A'}$ do not vary greatly from $\overline{\sigma_{A'}}$ = 1.5. Parameters for calculations were taken from the lines shown in Figs. 3 and 4. The Y(Z) values used are the averages for complementary Z's after normalization of calculated IN values to add to the recommended Y(A) [15] by use of a computer program, which calculates IN(A,Z), Y(A), etc. from $A'_p(Z)$, $\sigma_{A'}(Z)$, and Y(Z) input parameters.

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[6] Values of A_p, σ_A, and Y(Z) were also derived, without addition of ν_p to A, and were quite similar to A'_p, $\sigma_{A'}$, and Y(Z), except σ_A varied more than $\sigma_{A'}$. In one calculation EON(Z) was also derived; there was considerable variation, but $\overline{EON(Z)} \simeq 1.07$.

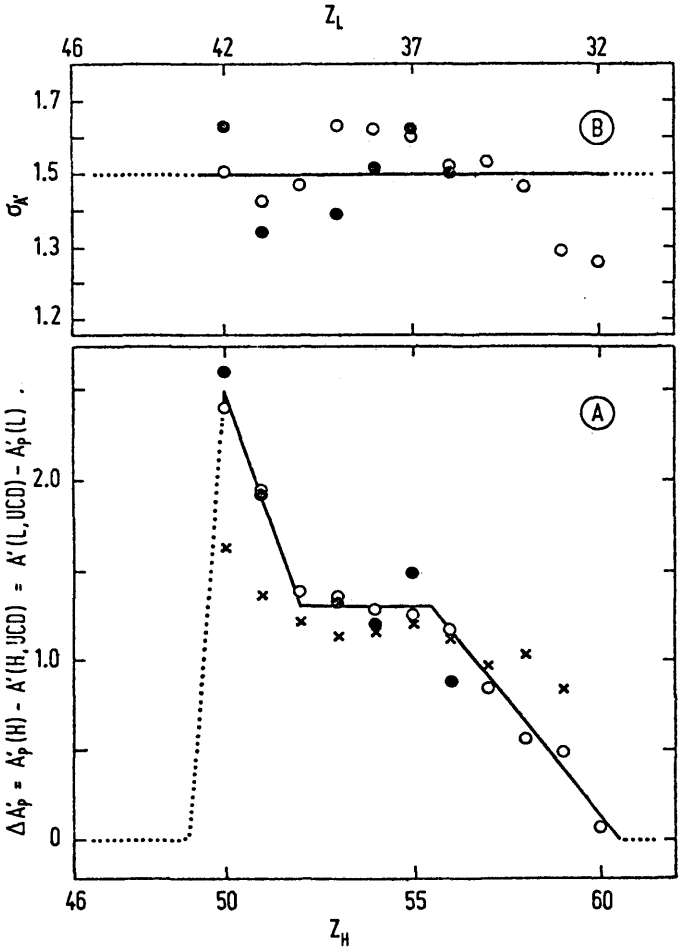

Fig. 3. Functions for A'_p model. Symbols: ○ , Z_L; ● , Z_H; X, from x-ray data [9] ; ——— , ····· , functions used in calculations.

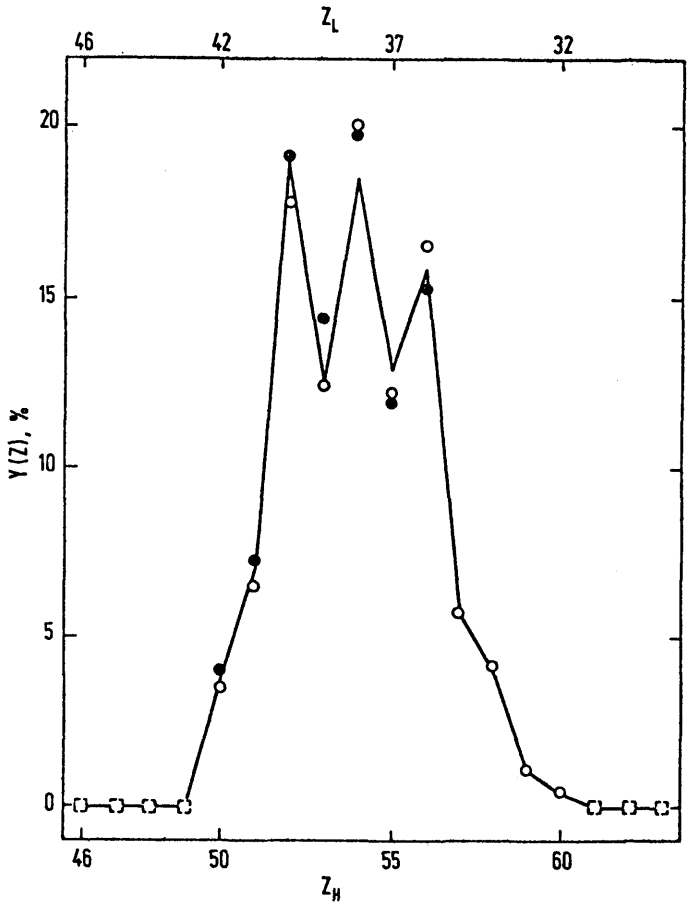

Fig. 4. Element yields, Y(Z), for A'_p model. Symbols: ○ , Z_L; ● , Z_H; ⊡ estimated Y(Z); ——— , normalized to Y(A), Y(Z) average for complementary Z's, used in calculations.

The even-odd-proton effect shows up prominently in Fig. 4, which also shows that the effect was largely lost in the X-ray measurements [9]. The derived $\Delta A'_p$ values shown in Fig. 3A are resonably consistent with those from the X-ray measurements, expect the differences are quite large for Z = 40, 42 and 51, 41.

5. TESTS OF MODELS FOR U235T

The parameters associated with each model were used with computer programs to calculate FI, FC, IN, Y(A), and Y(Z), for all fission products from U235T, and experimental and calculated values were compared in several ways. The three simple models were found to represent the data, from which they are derived, about equally well. For example, 60-65 % of the experimental and calculated values agreed within experimental error (or the estimated assigned error for LOHENGRIN data), and only 3-4 % of the values disagreed by more than a factor of two. The worst discrepency for all three models occurred for ^{77}Ga; much larger FC values were calculated (0.78-0.90) than had been measured (0.573 ± 0.014 [19]). Since the difference between calculated and experimental values was ~20 times the reported experimental uncertainty, the value was not used in later calculations.

Other comparisons between calculated and experimental values included calculation of 'goodness-of-fit' values (called FIT and defined and discussed in section 6), the values for the three models being about the

same ($\sim$1.5). Also, for the EOZ and the OSC models, the change in Z_p (for constant σ_Z) and the change in σ_Z (for constant Z_p) required to bring a calculated value into agreement with an experimental one were determined. For both models 88 % required a change in Z_p of <0.1, and only 4 % required a change of >0.2; 80 % required a change in σ_Z <0.05, and 12 % required a change of >0.1.

In addition, semi-logarithmic plots of experimental and calculated IN values vs. A and vs. Z were made for visual comparison; in most plots, involving all three models, agreement between calculated and experimental values was satisfactory. It was demonstrated, as expected, that the near Gaussian dispersion of yields with respect to both A and Z, observed in the high and nearly constant mass-yield regions, does not hold in regions where mass-yields change rapidly. This effect is illustrated for technetium yields in Fig. 5.

The simplest of the three models is the EOZ model, and, because of its simplicity, it is most easily applied to other fissioning systems for which much less data exist than for U235T. The EON factor was added, and the results of calculations with this model, modified to derive $\overline{\Delta Z_p}$, $\overline{\sigma_Z}$, $\overline{EOZ}$, $\overline{EON}$, and $\partial \Delta Z_p / \partial A'$ values from data for many different mass numbers simultaneously are discussed in section 6.

The less conventional A'_p model is perhaps the most fundamental of the three tested, and it is encouraging that it represents the U235T data as well as do the more conventional models. The even-odd-proton effect is included naturally in the Y(Z) values, as shown in Fig. 4., and, as will be discussed in section 7, the peak-to valley transition, which is awkward for the

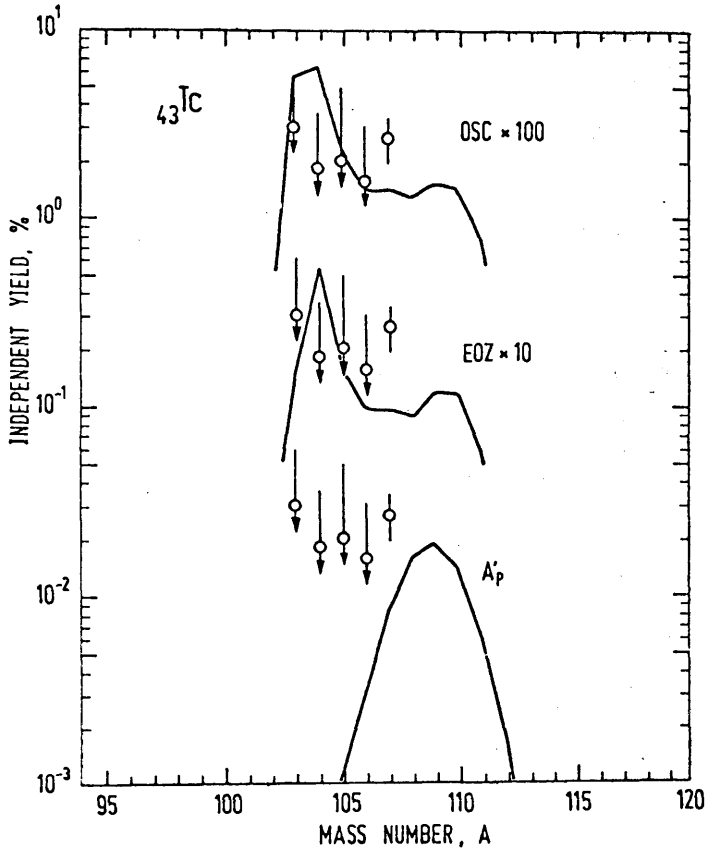

Fig. 5. Independent yields for $_{43}$Tc isotopes.

Symbols: O , measured (arrows indicate upper limits); ——— , calculated for indicated model.

conventional models, is accounted for quite naturally by a large relative change in yield ($\gtrsim\sim$40 fold) between $_{50}$Sn, $_{42}$Mo and $_{49}$In, $_{43}$Tc. Unfortunately, the model is not useful at present for fission processes other than U235T because Y(Z) values are not accurately known experimental quantities, and, therefore, many parameters would need to be determined from the limited data available.

6. USES OF MODIFIED EOZ MODEL AND APPLICATION TO OTHER FISSION PROCESSES

One use of the modified EOZ model, which determines $\overline{\Delta Z_p}$, $\overline{\sigma_Z}$, $\overline{EOZ}$, $\overline{EON}$, and $\partial\Delta Z_p/\partial A'$ from data for many mass numbers simultaneously, was to test the over-all consistency of two different data sets by comparing the derived parameters. For example, as shown in Table I, removal of most LOHENGRIN data from a U235T data set or use of data for only heavy fission products (no LOHENGRIN data) results in essentially the same average

TABLE I

AVERAGE VALUES FOR PARAMETERS FOR U235T

(EON = 1.00 , $\partial\Delta Z_p/\partial A' = 0.0$)

Calculation (No. of data)[a]	$\overline{\Delta Z_p}$	$\overline{\sigma_Z}$	$\overline{EOZ}$	FIT[b]
EOZ model (146)[c]	(0.50)	(0.54)	(1.25)	1.5
Modified EOZ model (146)[d]	0.51±0.01	0.53±0.01	1.28±0.02	1.4
Modified EOZ model (79)[e]	0.52±0.01	0.53±0.01	1.29±0.02	1.7
Modified EOZ model (48)[f]	0.50±0.01	0.53±0.01	1.33±0.03	1.6

[a]Data with FI(FC) $\geq$0.01 (A = 79-104, 131-144) were used.

[b]$FIT = \left[\left(\frac{1}{N-V}\right)\sum_N\left(\frac{\text{exp.-calc.}}{\text{error}}\right)^2\right]^{1/2}$

N = number of observations.

V = number of parameters varied.

exp. = experimental value.

calc. = calculated value.

error = uncertainty in experimental value.

(If the functions used to calculate values are the correct ones, and there are many data with statistical variations and errors, FIT should equal one.)

[c]Parameters derived from treatment of individual mass numbers by the EOZ model as discussed in section 4.

[d]Same data as used for the EOZ model treated by the modified EOZ model in one calculation.

[e]67 values derived only from LOHENGRIN data were not included.

[f]Only data for heavy fission products (A = 131-144) were used.

values as those obtained from all data or as those derived by treating mass numbers separately as described in section 4. Thus, at least on the average, the LOHENGRIN data are consistent with the radiochemical and mass spectrometric data, a conclusion consistent with that drawn in section 2 from direct comparison of 45 individual yields. Ofcourse, in the more detailed comparison, differences did occur, and a number of interesting results that were derived from the LOHENGRIN data [10,11] are not reflected in the average values.

The method was also applied to CF252S, for which relatively few radiochemical data exist, but a number of independent yields had been determined for even-even fission products from intensities of $2^+ \rightarrow 0^+$ ground-state gamma transitions [20]. As can be seen in Table II, the different data sets give average values that are consistent with the average values for the combined set. It may also be noted that the method of weighting and selecting data does affect the results to a small extent. The last entry shows that use of representative values of $\overline{\Delta Z_p}$ and $\overline{\sigma_Z}$ derived for other fission process, as will be discussed, still reveals no even-odd-proton effect, although most CF252S data are for even-even nuclei. If for same data, EOZ is fixed at 1.25, the FIT value increases to 2.0.

The results of calculations for U233T and PU239T, as well as results of new calculations for U235T after addition of new data [21], are summarized in Table III. The values given were obtained from a number of different calculations and represent approximately the median of the values obtained, and the uncertainties given cover the range of values obtained and are equal to or larger than the standard deviation obtained from a single calculation. In one set of calculations only FI and FC values $\geq$0.01 were used,and in another the estimated uncertainty in the model was

used in deriving the weighting factor as described in section 3. In each set of calculations 3, 4, or 5 parameters were determined; $\overline{\Delta Z_p}$, $\overline{\sigma_Z}$, and $\overline{EOZ}$ were always determined, and $\overline{EON}$ and/or $\partial \Delta Z_p / \partial A'$ were sometimes determined. The latter two parameters were always small, and inclusion or omission of them affected the values of the other parameters very

TABLE II

AVERAGE VALUES FOR PARAMETERS FOR CF252S

(EON = 1.00, $\partial \Delta Z_p / \partial A' = 0.0$)

Data used (No.)	$\overline{\Delta Z_p}$	$\overline{\sigma_Z}$	$\overline{EOZ}$	FIT
Rad. Chem.[a] (7)	0.59±0.07	0.56±0.04	0.97±0.10	2.5
Rad. Chem.[b] (11)	0.55±0.03	0.57±0.02	1.05±0.07	0.7
$\gamma(2+ \rightarrow 0+)$[a,c] (27)	0.48±0.03	0.60±0.02	0.93±0.05	1.3
All[a,c] (34)	0.55±0.03	0.58±0.02	1.01±0.04	1.6
All[b,c] (38)	0.53±0.03	0.59±0.02	0.97±0.04	1.2
All[a,c] (34)	(0.53)[d]	(0.54)[d]	0.93±0.06	1.5

[a]Only FI(FC) $\geq$0.01 were used.

[b]Weights included the estimated uncertainity in the model, as discussed near the end of section 3.

[c]Since experimental uncertainities were not given [20], estimated uncertainities of 15 % or more of a value were assigned to the gamma-ray data. The experimental IN values were divided by appropriate Y(A) values [3] to obtain FI values for analysis. The experimental values for ^{96}Sr, ^{98}Zr, ^{132}Te, and ^{150}Nd, all with reported small IN values, were found to be smaller by~5 estimated errors than the calculated values in preliminary calculations and, therefore, were not included in the data set.

[d]Assigned value.

TABLE III

SUMMARY OF AVERAGE VALUES FOR PARAMETERS FOR U235T, U233T, PU239T and CF252S

$(\partial \Delta Z_p / \partial A' = 0.0)$

Fission process	$\overline{\Delta Z_p}$	$\overline{\sigma_Z}$	$\overline{EOZ}$	$\overline{EON}$	FIT (No.of data) (a)	(b)
U235T	0.51±0.01	0.53±0.01	1.26±0.02	1.07±0.02	1.3 (156)	0.8 (191)
U233T	0.51±0.02	0.54±0.03	1.30±0.05	1.07±0.02	2.9 (45)	1.1 (58)
PU239T	0.57±0.03	0.56±0.03	1.07±0.04	1.00±0.03	2.1 (48)	1.3 (59)
CF252S	0.54±0.04	0.58±0.02	1.00±0.05	(1.00)[c]	1.6 (34)	1.2 (38)

[a] Only FI(FC) $\geq$0.01 were used.

[b] All data were used, and weights included the estimated uncertainty in the model as discussed near the end of section 3.

[c] Assumed value.

little. The $\overline{EON}$ values are given in Table III, and values of $\partial \Delta Z_p / \partial A'$ were all small and negative (-0.01 to 0.00). Inclusion of small negative slopes in ΔZ_p did not improve the representation of the data appreciably, so values are not given.

For the several fission processes discussed, the $\overline{\sigma_Z}$ is nearly constant; a value of 0.54 ± 0.02 could represent all quite well, except possibly for CF252S. The $\overline{\Delta Z_p}$ is somewhat larger for PU239T than for U235T or U233T, an observation that had been made previously, although based on less data [4,7,22] . The largest difference in parameter values is for the $\overline{EOZ}$ factor, which is close to one for PU239T and CF252S, again an observation

that had been made previously for PU239T, although based on less data [4,7,22]. The same conclusion was also suggested earlier because of differences in the amount of fine structure observed in the fragment mass-yield curves [23,24].

7. THE 50-PROTON EFFECT

In the mass number regions (A = 105-107 and A = 129-126), where mass yields decrease sharply, there is evidence that abrupt changes in the charge-distribution systematics occur for U235T. Figs. 1 and 2 show a sharp rise in ΔZ_p and a decrease in σ_Z near Z = 50, 42, and Fig. 4 shows an abrupt change in relative yield between Z = 49, 43 and 50, 42, an increase from $\lesssim\sim$0.1% to 3.7 %, a factor of $\gtrsim\sim$40.[7]

A more quantitative evaluation of the changes in ΔZ_p and in $\overline{\sigma_Z}$ that occur was made by treating the data for A = 105-107 (9 yields, 3 each for $_{41}$Nb, $_{42}$Mo, and $_{43}$Tc isotopes) and for A = 126-129 (4 $_{51}$Sb yields) by the method of least squares. If EOZ = 1.30 and EON = 1.06 were assumed, $\partial\Delta Z_p/\partial A' = -0.23 \pm 0.08$ and $\overline{\sigma_Z} = 0.40 \pm 0.03$; if EOZ = EON = 1.00 were assumed, $\partial\Delta Z_p/\partial A' = -0.28 \pm 0.06$ and $\overline{\sigma_Z} = 0.33 \pm 0.02$. These values may be compared to $\partial\Delta Z_p/\partial A' = -0.01$ to 0.00 and $\sigma_Z = 0.53 \pm 0.01$ for A < 105 and A > 129 derived in section 6. The large negative $\partial\Delta Z_p/\partial A'$ values approach that for $\partial Z(50)/\partial A' = -0.39$, so Z_p values remain close to 50, 42 over a range of several mass numbers, presumably due to the stability of the 50-proton shell. However, as noted in section 5, this treatment does lead to strange distributions of yields for $_{43}$Tc and $_{49}$In fission products, as is illustrated in Fig. 5.

[7] Although the yields of $_{43}$Tc and $_{49}$In have not been determined accurately, there is considerably evidence that the yields from U235T are small [9,10,25,29].

The above discussion suggests that a general requirement for high yields at low excitation energies is that one of the products have 50 or more protons, and this leads to asymmetric mass and charge division when $Z_F < 100$. Although maximum yields occur for elements with $Z > 50$ and appear to be associated with the stability of neutron shells [27], the abrupt change from low to high yields occurs at $Z = 50$ and is most probably associated with the effect of the stable 50-proton shell on the fission process.

8. ESTIMATION OF FISSION-PRODUCT YIELDS

Of the models derived and discussed in previous sections the modified EOZ model, which gave the results discussed in section 6, is probably the most useful one, at present, for estimation of unmeasured fission-product yields. The parameters summarized in Table III can be used for the fission processes listed and giving products with $A > 129$, or less than the complementary A.

It should be noted that if either $\overline{EOZ}$ or $\overline{EON}$ is not one, a normalization factor (NF(A), Eq. 1) should be calculated for each A by calculating FI for each Z and summing. However, the normalization correction is usually small ($\lesssim\sim$10 % of a value for U235T) and may often be omitted along with the $\overline{EON}$ factor in making estimates of FI values, since the uncertainties introduced are probably no larger than those inherent in the simple model.

The uncertainty in the estimated yields is at least as large as the 'model error' (ME = $0.1/(FI)^{1/3}$) discussed in section 3, since an estimated value should be less certain than values calculated for data from which the model was derived.

For other system fissioning at low excitation energies, $\overline{\sigma_Z} = 0.54$ and $\Delta Z_p = 0.53$ might be used along with whatever estimate of $\overline{EOZ}$ can be made, possibly based on the amount of fine structure observed in a fragment mass-yield curve [24] or on other criteria [28]. Of course, the uncertainities in the estimated yields would be still larger than those for the systems for which parameters were derived.

Yield estimates for products from near symmetric mass and charge division would be very uncertain, but if necessary, the dotted lines for ΔZ_p an σ_Z shown in Figs. 1 and 2 might be used. The behaviors of these functions near Z = 50 were discussed for U235T in section 7; the rest of the functions towards symmetry (A' = 118) were estimated. The assumption of UCD near as well as at symmetry for both Z_p (Fig. 1) and A'_p (Fig. 3A) was one way to achieve reasonably consistent yield estimates from calculations based on the two types of models.

A more natural transition from peak to valley yields is given by the A_P' model. As illustrated in Fig. 6, assumption of Gaussian dispersion of IN yields for isotopes of elements and use of $\sigma_{A'} = 1.5$ and A_P' values derived from Fig. 3A give reasonable approximations of experimental Y(A) values from the sums of calculated IN values.

The nearly constant position of the heavy mass-yield peak for many fission processes [24], especially the constant position of the light side of the peak [26], suggest that similar treatments of the transition from peak to valley yields could be applied to many fission processes other than U235T. The change in peak-to-valley yield ratios with changing excitation energy would be associated with changing yields of elements with $Z < 50$ and $Z > Z_F - 50$, relative to those with values of Z outside this range.

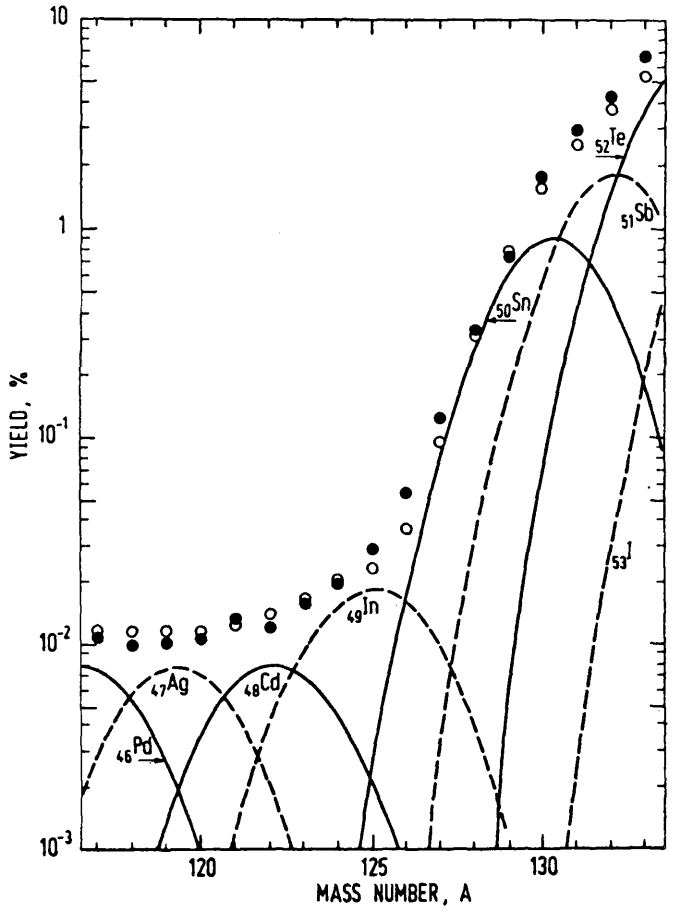

Fig. 6: Mass yields derived from summation of independent yields from A_p' model. Symbols: o , sum of independent yields; o , experimental mass yields /15/; ——— , ---- , independent yields of isotopes of elements calculated from parameters for A_p' model shown in Figs. 3 and 4.

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Contribution to Review Paper 12

THE USE OF DATA FROM β-STRENGTH FUNCTION EXPERIMENTS TO CALCULATE AVERAGE β- AND γ-DECAY ENERGIES

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Abstract

In this paper, we suggest that data from recent experiments designed to measure β-strength functions for a large number of short-lived fission products may also be used to provide average β- and γ-decay energies for these nuclides as well. The methods employed to do this are given and the results for the average β-decay energies, $\langle E_\beta \rangle$, per decay are presented. For ~10 cases, $\langle E_\beta \rangle$ values from decay-schemes studies are available. A comparison of our deduced values with these latter ones is presented and generally good agreement is found.

1. INTRODUCTION

The libraries of Fission-Product Nuclear Data (see, e.g., [1,2,3]) utilized by "summation-calculation" codes to calculate the fission-product decay-heat source term incorporate data on large numbers (typically several hundred) of individual nuclides. For each nuclide, the radioactive-decay information required include the half-life, decay modes and the average β- and γ-decay energy ($\langle E_\beta \rangle$ and $\langle E_\gamma \rangle$) emitted per decay. At the present time, a significant fraction of these data (particularly the average-energy values) results from other than direct experimental determination. The ENDF/B-IV Fission-Product File [3], for example, contains data for 711 radioactive nuclear species. Of these, ~380 represent cases for which the experimental data are either nonexistent or sufficiently sparse that the $\langle E_\beta \rangle$ and $\langle E_\gamma \rangle$ values cannot be obtained directly. These nuclides are

generally those with short half-lives, which, because of the attendant experimental difficulties, have remained relatively unstudied. Moreover, they make the bulk of their contribution to the decay heat source term at relatively short times (~10^1-10^3sec), which is the region of importance for the assessment of the effects of loss-of-coolant accidents (LOCA). Different fission-product data libraries employ different methods to estimate values for these unmeasured quantities.

It is clearly desirable to have these average-energy values determined from experimental measurements wherever possible. In this context, we wish to call attention to the existence of recent experiments [4,5] designed to measure β-strength functions for a large number (~70) of short-lived fission-product nuclides. The bulk of these nuclides presently have no measurement-based average-energy values. In this paper we show that information produced from these experiments may provide a means of deducing "experimental" $\langle E_\beta \rangle$ and $\langle E_\gamma \rangle$ values for these nuclides. It is also suggested that this information may make it possible, through extrapolation, to obtain improved estimates of $\langle E_\beta \rangle$ and $\langle E_\gamma \rangle$ values for a wider range of presently unstudied fission products.

2. GENERAL CONSIDERATIONS

The experimental techniques and the methods of data analysis employed to obtain the β-strength functions for the decay of fission-product nuclides has been described in detail elsewhere [4,5]. For purposes of this discussion, we will define only the pertinent quantities involved. A schematic decay scheme for a nuclide with relatively large Q value is shown in Figure 1. The energy available for the decay is Q_β. The β-decay branching intensity to an energy interval, dE, centered at an excitation energy, E, in the daughter nucleus is defined as b(E)dE. At excitation energies of the order of a few MeV, this energy interval generally contains a relatively large number of discrete energy levels. The quantity of interest for nuclear structure, in this case, is the β-strength function, $S_\beta(E)$, which

is defined by the relation [4,5,6]

$$S_\beta(E) = \frac{b(E)}{f(Z,Q-E)\cdot T_{1/2}} \ ,$$

where $T_{1/2}$ is the parent-nucleus β-decay half-life and f(Z,Q-E) is the integrated rate function (or Fermi function).

From the experiment, values are deduced for the quantity b(E)dE as a function of E. For experimental reasons, b(E)dE cannot be measured below a certain energy, E_0. For the experimental conditions of Ref. [5] (from which the data presented below were taken), E_0 ~305 keV and dE ~142 keV. It should be noted that these measurements give absolute values for b(E)dE, not just relative ones.

3. ANALYTICAL PROCEDURES AND RESULTS

Our interest here centers on the β-feeding intensities, $b(E_i)dE$, since they can be used to infer $\langle E_\beta \rangle$ (and hence $\langle E_\gamma \rangle$) values. These β-decay branching data are not explicitly given in [5]. However, they have been supplied to us by Dr. K. Aleklett.

3.1. Procedure for estimating the $\langle E_\beta \rangle$ values

By analogy with the more customary situation in which the energies and intensities of the discrete β-decay branches are known from decay-scheme studies (see, e.g., [3]), we can calculate an average β energy, $\langle E_\beta \rangle_\alpha$, corresponding to the fraction, α, of the β intensity to levels above E_0, using the relation

$$\langle E_\beta \rangle_\alpha = \sum_i E_{\beta i} \, I_{\beta i} \, f_i(E_{\beta i}) \ , \qquad (1)$$

where

$E_{\beta i} \equiv Q_\beta - E_i$ is the end-point energy of the $i^{\underline{th}}$ β-decay branch,

$I_{\beta i} \equiv b(E_i)dE$ is the intensity (expressed as a fraction per decay) of the $i^{\underline{th}}$ β-decay branch,

and $f_i(E_{\beta i})$ gives that fraction of the available energy $(Q_\beta - E_i)$ which is, on the average, carried by the β radiation.

Similarly, the average neutrino energy, $\langle E_\nu \rangle_\alpha$, corresponding to these transitions is given by the expression

$$\langle E_\nu \rangle_\alpha = \sum_i E_{\beta i} \, I_{\beta i} \, [1 - f_i(E_{\beta i})]. \qquad (2)$$

The $f_i(E_{\beta i})$ values were calculated using a computer program described in [7]. In the present work, we have assumed that all the β transitions have an allowed shape. In the absence of direct experimental evidence to the contrary, it is generally assumed (see, e.g., [3]) that allowed and first-forbidden nonunique β transitions have allowed shape factors (and, hence, f_i values); and this assumption is not believed to introduce any appreciable error into the results.

To obtain values for $\langle E_\beta \rangle$, account must be taken of that fraction, 1-α, of the β-decay intensity which proceeds to daughter-nucleus levels below E_0. To do this, we have considered three different situations, which cover the range of possible distributions of this remaining β intensity. These situations are as follows:

(1) the remaining fraction (1-α) goes directly to the ground state;

(2) this fraction goes to an assumed state halfway between the ground state and E_0;

(3) this fraction is split equally among four states (including the ground state) assumed equally spaced in the region below E_0.

The resulting average-energy values, $\langle E_\beta \rangle_{1-\alpha}$, can then be combined with the $\langle E_\beta \rangle_\alpha$ value to yield an overall $\langle E_\beta \rangle$ value consistent with each of

these three cases. The $\langle E_\beta \rangle$ values calculated according to situation (3) above lie between those calculated according to situations (1) and (2). Consequently, in the absence of other information regarding the distribution of this intensity, the $\langle E_\beta \rangle$ values given below have been calculated assuming the applicability of this third situation.

3.2. Results of the $\langle E_\beta \rangle$ calculations

The results of the $\langle E_\beta \rangle$ calculations for those fission-product nuclides studied in [5], together with other pertinent information, are summarized in Table I. In several cases, two entries are given for an individual nuclide. These nuclides are delayed-neutron precursors (for which Q_β is greater than the neutron separation energy, Sn, in the daughter nucleus, as illustrated in Figure 1); the two entries correspond to β-feeding distributions deduced [5] without consideration of the effect of delayed-neutron emission and with account taken of this effect using the listed delayed-neutron branching (P_n). For these latter cases ($P_n \neq 0$), all the neutron decay was assumed to populate only the ground state of the daughter nucleus (Z,A-1).

The $\langle E_\beta \rangle_\alpha$ values in column 5 of Table I represent the calculated contribution to the total average β energy due to feeding of levels above E_o. As discussed above, three different situations have been considered concerning how the remaining fraction (1-α) of the β intensity is assumed to be distributed over the levels below E_o in the daughter nucleus. To illustrate the sensitivity of the derived $\langle E_\beta \rangle$ value to this choice we consider as a representative case the nuclide ^{114}Ag, for which only ~31% of the β decay proceeds to levels above E_o. We find for this case the following values of $\langle E_\beta \rangle$ calculated for these three situations: (1) 2061 keV; (2) 2011 keV; and (3) 2023 keV. Even for this relatively extreme case, the range of $\langle E_\beta \rangle$ values is rather small. We have found, quite generally, that the uncertainty introduced into the $\langle E_\beta \rangle$ values from the problems associated with the unknown distribution of the (1-α) fraction of the β intensity is smaller than that due to other sources of error. As stated in Sect. 3.1. above, in cases where this distribution is unknown, the $\langle E_\beta \rangle$ values given in Table I are

those calculated according to situation (3). In cases where the J^{π} values are such that ground-state β transitions are unlikely, and where levels below E_o are expected to occur, the listed $\langle E_\beta \rangle$ value is that calculated according to the second situation. The $\langle E_\beta \rangle$ value based on the first situation was chosen for the decay of odd-odd nuclei where spin considerations indicated that ground-state β transitions could occur.

It is difficult at present to provide a realistic assessment of the uncertainties to be associated with the $\langle E_\beta \rangle$ values listed in column 6 of Table I. The uncertainties associated with these values have been estimated from a consideration of only two possible sources of error: (1) that associated with the distribution of the fraction (1-α) of the β intensity; and (2) possible errors in the value of α itself. This first source has been treated by direct calculation, as indicated above. No estimate of the magnitude of possible errors in the α values is given in [5]. In order to provide some estimate of the influence of errors in α on the derived $\langle E_\beta \rangle$ values, we have assigned, somewhat arbitrarily, an uncertainty of 10% to each of the listed α values. This seems to represent a reasonable estimate. Values of α greater than unity are, by definition, not possible. Of the α-values given in Table I, only 5 (excluding those of some delayed neutron precursors) are greater than unity and, of these, only two (for which α ~1.2) are significantly larger than 1.1. In calculating the overall uncertainty in $\langle E_\beta \rangle$, the error propagation took explicit account of the fact that $\langle E_\beta \rangle_\alpha$ and $\langle E_\beta \rangle_{1-\alpha}$ are constrained by the requirement that the total β-decay intensity is unity (or, for delayed-neutron emission, is $1-P_n$). No contribution from the uncertainty in the Q_β values is included in the $\langle E_\beta \rangle$ uncertainties in Table I.

3.3. Comparison of calculated and previously measured $\langle E_\beta \rangle$ values

Among the nuclides listed in Table I, there are several whose average-energy values have also been determined from the discrete β-decay data from decay-scheme studies. A comparison of the $\langle E_\beta \rangle$ values from these data with those deduced above from the β-feeding data is summarized in Table II. The experimental $\langle E_\beta \rangle$ values in Table II are taken from the

decay-scheme data as given in the Fission-Product File currently being prepared for Version V of ENDF/B [3,8]. Since the Q_β values given in [5] differ in some cases from those in ENDF/B, it has been necessary to change one or the other of them in order to effect a meaningful comparison. By way of further illustration, a comparison of the β-feeding data of Ref. [5] with the experimental β intensities is given for three cases in Figure 2. In Figure 2, the shaded area gives, as a histogram, the β-feeding data while the β intensities obtained from decay-scheme studies are shown by the heavy vertical lines.

In general, as inspection of Table II indicates, the agreement between the two sets of $\langle E_\beta \rangle$ values is quite good, particularly for those nuclides situated in the heavy-mass peak of the fission-yield distribution. The nuclide, ^{140}Cs, appears to represent an exception, but here there is considerable uncertainty in the intensities of the β branches to the ground and first excited states of the daughter nuclide, ^{140}Ba. (In ENDF/B-V the sum of these two β-transition intensities is ~30%, while some data exist which indicate that this sum may be $\gtrsim$40%. Use of this latter value would significantly reduce the difference in the two $\langle E_\beta \rangle$ values.) Consequently, ^{140}Cs may not represent a really good test case. The ^{76}Ga decay leads to doubly even ^{76}Ge. For this relatively light-mass daughter, the first excited-state energy is fairly high (~563 keV) and the level density up to excitation energies of ~5-6 MeV is low. Consequently, the assumptions underlying the data-analysis procedures of Ref. [5], namely that the γ decay of the levels excited following β decay can be treated statistically, may not be valid. Even in this case, however, the difference in the two $\langle E_\beta \rangle$ values is only about 240 keV (~14%).

The thrust of the discussion thus far has been only the evaluation of the $\langle E_\beta \rangle$ values, since our interest has centered on the validity of the idea that the β-strength function experiments can serve as a reliable source of average-energy data for otherwise unstudied short-lived fission products. The $\langle E_\beta \rangle$ values are the important ones from this point of view, since the $\langle E_\gamma \rangle$ values can then be inferred from the relation

$$\langle E_\gamma \rangle = Q_\beta - \langle E_\beta \rangle - \langle E_\nu \rangle ,$$

with the $\langle E_\nu \rangle$ values being determined from the same b(E)dE data as the $\langle E_\beta \rangle$ values (see Eqs. (1) and (2)).

4. CONCLUSIONS

In this report, we have called attention to the existence of experimental studies [4,5] of the β-strength functions of a number of short-lived fission-product nuclides and have suggested that data from these experiments may make it possible to provide experimentally based estimates of the $\langle E_\beta \rangle$ (and hence $\langle E_\gamma \rangle$) values for these nuclides, many of which have not at present been studied by any other means. For those cases where the $\langle E_\beta \rangle$ values calculated as described above can be compared with corresponding values from decay-scheme studies, the agreement between the two sets of values is reasonably good. This seems particularly to be the case for the heavier-mass fission products, where the level densities in the daughter nuclides are sufficiently high that the statistical assumptions made [5] in the data analysis to yield the β-feeding distributions are more likely to be justified.

It appears to us that the data of Ref. [5] can be used to provide realistic $\langle E_\beta \rangle$ and $\langle E_\gamma \rangle$ values for a number of short-lived fission products for which at present only theoretically estimated values are available. As such, they can serve to extend the range of nuclides on the various fission-product libraries for which "experimental" average-energy values are available. Of further interest is the possibility that these data, in some form (such as, e.g., the average beta feeding function [5]) may provide a basis for extrapolation to other unmeasured cases and, hence, to provide a means of generating a set of $\langle E_\beta \rangle$ and $\langle E_\gamma \rangle$ data that are more closely linked to experimental measurements than is presently the case. Additional effort is required in order to determine if, or to what extent, such an extrapolation procedure can be defined that will lead to actual improvement in these values.

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TABLE I

Summary of the average β-energy values calculated from the β-strength-function related measurements given in [5]. The quantities in parentheses represent the uncertainties in the least significant figure (or figures) of the associated value. For further discussion, see the text.

Nuclide	$T_{1/2}$(sec)	α	Q_β(keV)	$\langle E_\beta \rangle_\alpha$ (keV)	$\langle E_\beta \rangle$ (keV)	Comments
^{76}Zn	5.70	0.87	4760	1525	1787(16)	$\Delta J^\pi = 0^+ \rightarrow (3^-)$; several final levels $< E_0$
^{77}Zn	1.40	1.07	6000	1493	1493(149)	
^{76}Ga	27.6	0.78	6100	1242	1837(61)	$\Delta J^\pi = (3^-) \rightarrow 0^+$; "NDS" $I_{\beta_0} = 0$, no levels $< E_0$
^{77}Ga	13.2	0.47	4700	701	1784(18)	
^{78}Ga	5.09	0.82	8010	1785	2442(85)	
^{79}Ga	3.00	0.43	6800	856	2601(33)	"NDS" Z-assignment uncertain
^{80}Ga	1.70	0.90	9440	1709	2144(155)	
^{81}Ga	2.20	0.56	7443	1175	2659(50)	
^{79}Ge	19.1	0.30	4300	409	1689(10)	$\Delta J^\pi = (\text{high } ?) \rightarrow 3/2^-$;
^{81}Ge	13.4	0.44	5575	894	2273(13)	
^{82}As	16.3	1.22	7083	2311	2311(231)	"NDS" gives two isomers with $T_{1/2}$=21 s and 13 s. Strong I_{β_0} for 21 s activity.
^{83}As	6.50	0.77	5086	1145	1685(40)	"NDS" $T_{1/2}$=14.1 (1.1) s
^{85}Se	19.0	0.18	5921	351	2503(9)	"NDS" $T_{1/2}$= 39 s
^{85}Br	175.0	0.08	2800	51	1084(3)	
^{86}Br	55.7	0.87	8000	1748	2216(98)	$I_{\beta_0} = 0$, no final levels $< E_0$
^{87}Br	55.8	0.70	7060	963	1914(89)	P_n assumed zero

Table I (continued)

^{87}Br	55.8	0.71	7060	951	1870(92)	P_n=0.024, S_n=5510
^{88}Br	16.5	1.07	8980	2603	2603(260)	$\Delta J^{\pi}=(1^-)\rightarrow 0^+$; "NDS" I_{β_o} = 5.5% from systematics; P_n assumed zero
"	16.5	1.12	8980	2626	2626(263)	P_n= 0.05, S_n = 7150
^{89}Br	3.60	0.81	8040	2184	2877(54)	P_n assumed zero
"	3.60	0.90	8040	2261	2625(72)	P_n = 0.089, S_n = 5220
^{89}Kr	281.0	0.96	5000	1360	1447(51)	$\Delta J^{\pi}=(5/2^+)\rightarrow(3/2,5/2^-)$; "NDS" I_{β_o}=14%
^{90}Kr	32.2	0.68	4350	902	1498(26)	Daughter isomer fed on β- decay
^{91}Kr	8.70	0.76	6120	1090	1742(69)	
^{92}Kr	1.84	0.21	5970	379	2468(12)	"NDS" measured $I_{\beta_o} = 94\%^{+6}_{-20}$
^{90}Rb	186.8	1.20	6870	2569	2569(257)	
^{91}Rb	58.2	0.76	5680	967	1567(66)	
^{92}Rb	4.50	0.15	7580	334	3215(12)	
^{93}Rb	5.81	0.58	7230	1155	2519(51)	P_n assumed zero
"	5.81	0.59	7230	1164	2495(53)	P_n = 0.0163, S_n = 5140
^{94}Rb	2.72	0.94	9450	2881	3140(83)	P_n assumed zero
"	2.72	1.06	9450	2964	2964(296)	P_n = 0.111, S_n = 7170
^{114}Ag	4.35	0.31	5000	545	2023(8)	"NDS" $I_{\beta_o} \geq 75\%$
^{116}Ag	153.0	0.37	5330	256	1705(42)	
^{116}Ag	8.60	0.49	5410	480	1672(47)	
^{117}Ag	5.30	0.53	4180	784	1607(10)	

Table I (continued)

^{117}Ag	72.7	0.46	4180	348	1295(32)	
^{118}Ag	3.70	0.72	7230	1353	2250(67)	"NDS" assumes I_{β_o} = 15%
^{119}Ag	2.10	0.92	5365	1597	1782(38)	
^{120}Ag	1.17	0.41	8342	978	3178(39)	
^{120}In	3.08	0.25	5600	434	2249(12)	$\Delta J^{\pi}=1^{+}\rightarrow 0^{+}$; "NDS" I_{β_o} = 80.6%
^{121}In	23.1	1.08	3400	1045	1045(104)	$\Delta J^{\pi}=9/2^{+}\rightarrow 3/2^{-}$ & $(11/2^{-}$ @ 8keV)
^{122}In	9.20	0.68	6700	1511	2453(35)	
^{123}In	6.01	0.67	4500	1022	1640(16)	$\Delta J^{\pi}=(9/2^{+})\rightarrow 11/2^{-}$
^{123}In	48.0	0.60	4820	762	1582(33)	
^{124}In	3.17	0.86	7140	1837	2274(60)	
^{125}In	2.33	0.62	5420	1057	1935(26)	
^{125}In	12.2	0.21	5420	292	2116(14)	
^{126}In	1.53	1.00	8151	2598	2598(260)	
^{127}In	3.10	0.43	6372	805	2393(28)	
^{128}In	0.80	0.91	8919	2679	3038(67)	"NDS" $T_{1/2}$ = 3.7 s (?)
^{129}In	0.80	0.36	7310	698	2765(33)	
^{132}Sb	185.0	0.52	5600	599	1757(46)	$\Delta J^{\pi}=(4^{+})\rightarrow 0^{+}$; no final levels $<E_o$
^{134}Sb	10.7	0.56	8300	1404	3025(47)	
^{135}Te	19.2	0.75	6200	1642	2291(22)	
^{136}I	83.0	0.54	6590	1040	2360(36)	Recent data show I_{β_o} = 0, no final levels$<E_o$

Table I (continued)

^{137}I	24.6	0.32	6200	497	2324(26)	$\Delta J^{\pi}=(7/2^{+})\rightarrow 7/2^{-}$; P_n assumed zero
"	24.6	0.35	6200	506	2252(31)	P_n = 0.055, S_n = 3860
^{139}I	2.60	0.43	6500	845	2456(26)	P_n assumed zero
"	2.60	0.52	6500	849	2206(44)	P_n = 0.14, S_n = 3630
^{137}Xe	230.0	0.46	4348	741	1718(7)	$\Delta J^{\pi}=(7/2^{-})\rightarrow 7/2^{+}$; "ENDF/B" I_{β_o}=66(3)%, no levels $<E_o$
^{139}Xe	40.4	0.91	4570	1356	1529(27)	"ENDF/B" I_{β_o} = 24(6)%
^{140}Xe	13.7	0.74	4060	824	1259(29)	
^{141}Xe	1.72	0.42	6000	685	2186(28)	
^{139}Cs	562.0	0.18	4440	212	1729(9)	
^{140}Cs	64.0	0.56	5800	896	1991(35)	"ENDF/B" I_{β_o} = 10.7(3.0)%, no final levels $<E_o$
^{141}Cs	25.2	0.99	4980	996	1017(77)	
^{142}Cs	1.70	0.38	6890	651	2511(35)	P_n assumed zero
"	1.70	0.38	6890	652	2511(35)	P_n = 0.0021, S_n = 6200
^{143}Cs	1.65	0.70	5730	1073	1810(46)	P_n assumed zero
"	1.65	0.71	5730	1078	1790(47)	P_n = 0.0113, S_n = 4090
^{143}Ba	13.6	0.71	4000	756	1232(29)	
^{144}Ba	10.7	0.97	3200	987	1025(17)	
^{145}Ba	4.80	0.76	4950	1030	1531(39)	
^{144}La	40.3	0.87	5500	1036	1340(71)	
^{145}La	29.3	0.81	4130	700	1023(48)	

TABLE II

Comparison of our calculated average β-energy values with those obtained from decay scheme studies. The Q_β-value adjustments necessary to complete the comparison are also listed. The quantities in parentheses represent the uncertainties in the least significant figure (or figures) of the associated value. For further discussion, see the text.

Nuclide	$T_{1/2}$(sec)	α	ΔJ^π	Reference	Q(keV)	$\langle E_\beta \rangle$
^{76}Ga	27.6	0.78	$(3^-)\rightarrow 0^+$	THIS WORK	6100	1837(61)
				ENDF	6100	1601(10)
^{86}Br	55.7	0.87	$(2^-)\rightarrow 0^+$	THIS WORK	8000	2227(102)
				ENDF	8000(438)	2090
^{89}Kr	281.0	0.96	$(5/2^+)\rightarrow(3/2, 5/2^-)$	THIS WORK	4930	1414(48)
				ENDF	4930(60)	1346(16)
^{90}Kr	32.2	0.68	$0^+\rightarrow(1^-)$	THIS WORK	4390	1517(26)
				ENDF	4390(30)	1316(4)
^{91}Kr	8.70	0.76		THIS WORK	6120	1742(69)
				ENDF	6120(70)	1941(15)
^{91}Rb	58.2	0.76	$?\rightarrow(5/2^+)$	THIS WORK	5680	1567(66)
				ENDF	5680(40)	1462(26)
^{92}Rb	4.50	0.15		THIS WORK	7580	3215(12)
				ENDF	7700(250)	3446
^{137}Xe	230.0	0.46	$7/2^-\rightarrow 7/2^+$	THIS WORK	4348	1718(7)
				ENDF	4347(24)	1722
^{139}Xe	40.4	0.91		THIS WORK	4880	1671(28)
				ENDF	4880(60)	1715(9)
^{139}Cs	562.0	0.18	$(7/2, 9/2)\rightarrow(7/2)^-$	THIS WORK	4290	1660(9)
				ENDF	4290(70)	1696(16)
^{140}Cs	64.0	0.56	$?\rightarrow 0^+$	THIS WORK	6050	2106(35)
				ENDF	6050(250)	1696(17)

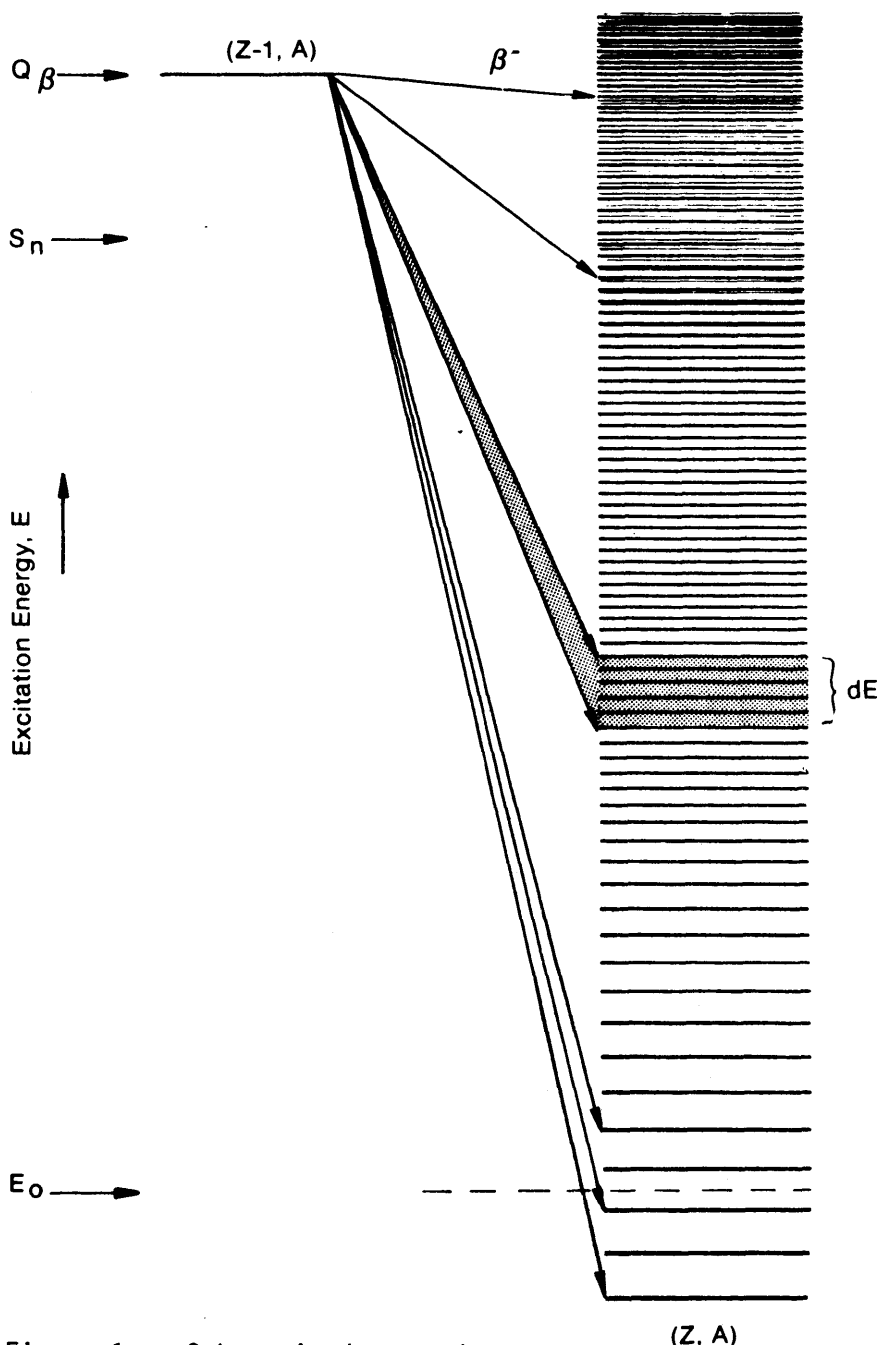

Figure 1. Schematic decay scheme illustrating the quantities discussed in the text.

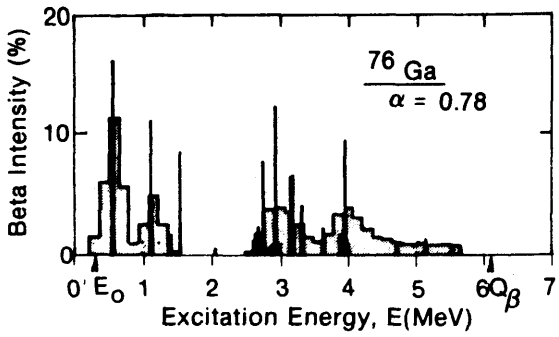

Fig. 2a) ^{76}Ga

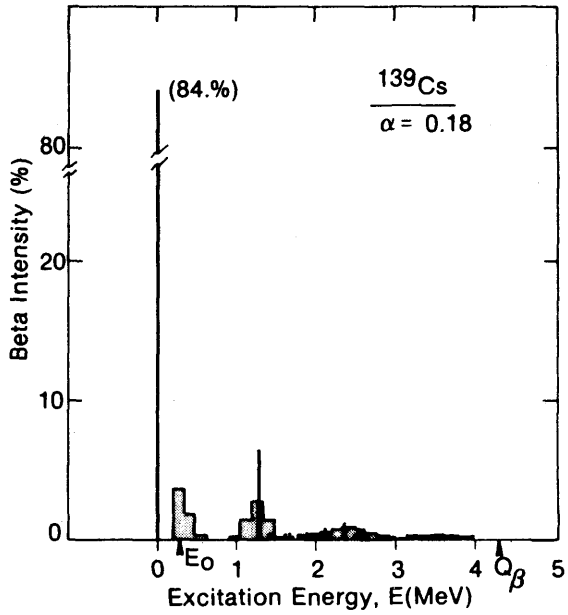

Fig. 2c) ^{139}Cs

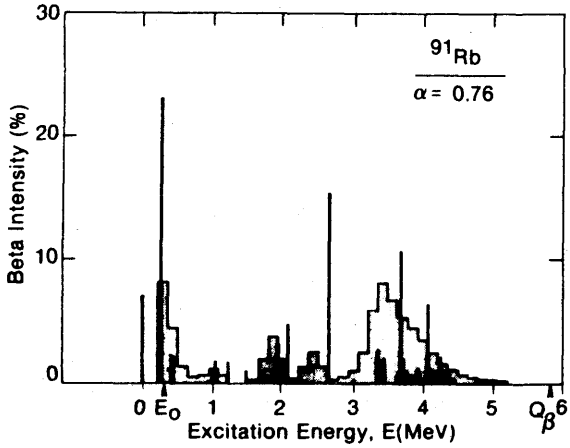

Fig. 2b) ^{91}Rb

Figure 2. Comparison of the β-intensity distributions deduced from the experimental β-strength function measurements [5] with those from decay-scheme studies for the decay of a) ^{76}Ga, b) ^{91}Rb and c) ^{139}Cs.

Contribution to Review Paper 12

GAMMA RAY EMISSION PROBABILITIES AND HALF LIVES OF SELECTED FISSION PRODUCTS

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Abstract: Gamma-ray emission probabilities per decay (also called absolute γ-ray intensities) were determined for the most abundant lines of ^{103}Ru, ^{132}Te, ^{134}Cs, ^{140}Ba and ^{140}La. Half-lives were measured for ^{103}Ru, ^{106}Ru, ^{131}I, ^{132}Te, ^{140}Ba and ^{140}La.

1. Gamma-ray emission probabilities

In contributions to the "First Panel on Fission Product Nuclear Data" in Bologna /1/ and to the INDC (NDS)-series on "Progress in Fission Product Nuclear Data" /2/, we described our program and method for the accurate determination of γ-ray emission probabilities I_γ per decay. The main emphasis of this program is on fission products. Results for ^{144}Ce, ^{144}Pr, ^{106}Rh and ^{95}Zr were already published /3/, those for ^{140}Ba and ^{140}La were submitted for publication /4/. New results are presented in this contribution for ^{103}Ru, ^{132}Te and ^{134}Cs. In Table I we have summarized the values for I_γ hitherto obtained by our group. For comparison purposes we have also listed values taken from American and French data files. In the following we give some comments on such data which are frequently used in various application fields, but which have been rather uncertain up to now.

^{103}Ru

For $I_\gamma(496)$ values between 86.4 % and 90 % have been reported. Helmer and Greenwood recommend in their evaluation /7/ $I_\gamma = (89 \pm 3)$ %. Only relative γ-ray emission probabilities I_r were measured in the the past. In order to arrive at values for I_γ, assumptions on the decay scheme and in particular on the abundance of the β⁻-decay

branch to the 56 min isomeric state of ^{103}Rh have to be made. Our method [+)] does not depend on this information.

^{106}Rh

S.T.Hsue et al./8/ report a value $I_\gamma(512) = (18.8 \pm 1.4)$ % which is derived from relative measurements of I_r and from an assumption on the abundance of the β^- decay branch to the ground state of ^{106}Rh. These authors suspect that in all other relevant papers a contribution of annihilation radiation to the 512 keV peak has been underestimated. They infer from a very questionable double peak evaluation an annihilation contribution of about 5 % which is subtracted from the total peak area. Annihilation radiation is present in the background and can originate from higher energy γ-rays of ^{106}Rh interacting in the surroundings of the detector. We can hardly imagine that measuring conditions are so badly chosen that the annihilation contribution is larger than 1 %. In our measurements this contribution was far less than 1 % which was assured by separate measurements of the background and of the 511 keV peak observed in spectra of other radionuclides emitting high energy γ-rays (^{88}Y, ^{24}Na).

^{132}Te

Helmer and Greenwood /7/ recommend $I_\gamma(228) = (88.5 \pm 6.0)$ % while the Oak Ridge Nuclear Data Group /5/ quotes (88.20 ± 0,20) % with a surprisingly low uncertainty.

$^{140}Ba/^{140}La$

The value $I_\gamma(537) = 20$ % given in the Nuclear Data Sheets /9/ is certainly wrong. In two recent publications, values of (25.7 ± 0.6) % /10/ and (23.57 ± 0.54) % /11/ are reported. The results for ^{140}La in references /10/ and /11/ differ by 12 % to 16 % for the strong lines at 329 keV, 487 keV and 816 keV though uncertainties of 2 % to 3 % are claimed by both authors.

+) Determination of the ^{103}Ru activity by $4\pi\beta$-γ coincidence measurements with application of the extrapolation method; accuracy ± 0.3 % at the 68 % confidence level.

^{144}Pr

Our result for $I_\gamma(696)$ already published in 1974 is by about 10 % lower than that recommended in references /5/ and /7/. Recent measurements at the LMRI /12/, however, support our value.

2. Half-lives

Half-lives of ^{103}Ru, ^{106}Ru, ^{131}I, ^{140}Ba and ^{140}La were determined by following the radioactive decay with an open air ionisation chamber and a high pressure argon filled chamber. In addition, Ge(Li) measurements were carried out for ^{131}I, ^{132}Te and ^{140}Ba. The results are given in Table II.

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Table I. Gamma-ray emission probabilities I_γ per decay.

Nuclide	E_γ in keV	I_γ in % this work	I_γ in % Ref. /5/	I_γ in % Ref. /6/
^{95}Zr	234.70	0.292 ± 0.015	-	0.25
	724.23	44.0 ± 0.5	44.4	44.4
	756.74	54.3 ± 0.5	54.6	54.6
^{103}Ru	497.08	90.9 ± 0.7	86.4 ± 2.4	89.35 ± 0.5
	557.04	0.841 ± 0.012	0.76 ± 0.03	0.800 ± 0.04
	610.33	5.65 ± 0.07	5.3 ± 0.3	5.720 ± 0.1
^{106}Rh	512	20.47 ± 0.23	20.6	20.6 ± 2.0
	616	0.735 ± 0.013	0.700	0.700
	622	9.95 ± 0.08	9.806	9.806
	873	0.421 ± 0.007	0.416	0.416
	1050	1.452 ± 0.013	1.463	1.463
	1128	0.376 ± 0.006	0.385	
	1562	0.148 ± 0.002	0.146	
^{132}Te	228.16	88.5 ± 1.3	88.20 ± 0.20	88.0
^{134}Cs	475.35	1.51 ± 0.03	1.46 ± 0.04	1.49 ± 0.08
	563.227	8.34 ± 0.12	8.38 ± 0.05	8.40 ± 0.4
	569.315	15.38 ± 0.22	15.43 ± 0.11	15.00 ± 0.7
	604.699	97.6 ± 0.1	97.6 ± 0.3	97.54 ± 0.1
	795.845	85.3 ± 0.9	85.4 ± 0.4	85.13 ± 0.8
	801.932	8.64 ± 0.12	8.73 ± 0.04	8.80 ± 0.6
	1038.57	0.998 ± 0.013	1.000 ± 0.010	1.02 ± 0.05
	1167.94	1.800 ± 0.020	1.80 ± 0.03	1.88 ± 0.07
	1365.15	3.02 ± 0.03	3.04 ± 0.04	3.20 ± 0.2
^{140}Ba	162.64	6.21 ± 0.08		6.15
	304.83	4.30 ± 0.05		4.10
	423.81	3.15 ± 0.04		3.20
	437.60	1.93 ± 0.04		2.00
	537.32	24.39 ± 0.22		24.0

Table I (cont.)

Nuclide	E_γ in keV	I_γ in % this work	Ref. /5/	Ref. /6/
^{140}La	241.97	0.473 ± 0.028		0.48 ± 0.05
	266.55	0.452 ± 0.025		0.52 ± 0.05
	328.77	20.74 ± 0.18		19.5 ± 0.8
	432.53	2.99 ± 0.04		2.86 ± 0.11
	487.03	45.94 ± 0.38		45.3 ± 1.9
	751.83	4.41 ± 0.04		4.17 ± 0.19
	815.85	23.64 ± 0.17		22.7 ± 0.7
	867.87	5.59 ± 0.05		5.3 ± 0.2
	919.63	2.68 ± 0.03		2.86 ± 0.14
	925.24	7.05 ± 0.08		7.10 ± 0.3
	951.4	0.539 ± 0.019		0.51 ± 0.04
	1596.49	95.40 ± 0.08		95.33 ± 0.16
	2348.1	0.846 ± 0.017		0.83 ± 0.04
	2521.7	3.43 ± 0.08		3.46 ± 0.14
^{144}Ce	133.53	11.09 ± 0.16	10.8 ± 0.7	11.10[a] ± 0.5
^{144}Pr	696.49	1.342 ± 0.014	1.48 ± 0.06	1.34[a] ± 0.04
	1489.15	0.279 ± 0.003	0.300 ± 0.016	0.28[a] ± 0.01
	2185.70	0.700 ± 0.010	0.77 ± 0.05	0.70[a] ± 0.08

[a] based on our results

Table II. Half-lives $T_{1/2}$

Nuclide	$T_{1/2}$		
	this work	Ref. /5/	Ref. /6/
^{103}Ru	(39.276 ± 0.009)d	(39.35 ± 0.05)d	39.35 d
^{106}Ru	(371.63 ± 0.17)d		369.0 d
^{131}I	(8.02 ± 0.01)d	(8.04 ± 0.01)d	8.04 d
^{132}Te	(76.9 ± 0.3)h	(78.2 ± 0.8)h	77.8 h
^{140}Ba	(12.746 ± 0.010)d		12.79 d
^{140}La	(40.272 ± 0.007)h		40.20 h

Contribution to Review Paper 13

DELAYED-NEUTRON SPECTROSCOPY WITH ^{3}He IONIZATION CHAMBERS

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ABSTRACT

The characteristics of ^{3}He ionization chambers have been examined. The detector response function is given in the energy range o.o19 - 2.77 MeV. A considerably improved energy resolution has been achieved by minimizing effects due to pulse summing, to detection of scattered neutrons, and microphonics, and is most evident in the low energy part of the spectra. The influences to the spectrum shape and the deduced average neutron energies for spectra from individual precursors and equilibrium and time-dependent spectra are discussed.

1. INTRODUCTION

Substantial improvement in fast neutron spectrometers has been achieved a few years ago by Shalev and Cuttler [1,2] who developed a high pressure ^{3}He ionization chamber. Though detectors of this design have become available commercially, delayed-neutron spectroscopy is not yet a straightforward technique and the properties of these detectors must be well understood. In this contribution to RP 13 some details on these characteristics are given with the emphasis on the spectroscopy of neutrons in the o.o1 - 2 MeV energy region.

2. ENERGY CALIBRATION AND RESPONSE FUNCTION

The spectrometers used in this study were cylindrical gridded ^{3}He ionization chambers with guard tubes to reduce fringing fields. Neutrons are detected through the reaction

$$^3\text{He} + n \longrightarrow p + {}^3\text{H} + 763.8 \text{ keV}.$$

Details on the description of the spectrometers and the electronic system used for high-resolution delayed-neutron spectroscopy can be found in Ref. [3].

The absolute energy calibration of the ^{3}He spectrometers and their response function to monoenergetic neutrons was obtained using neutrons from the $^{7}Li(p,n)^{7}Be$ reaction produced with Van de Graaff accelerators. An accuracy of about o.1% of the incident neutron energy was achieved.

The response function of the ^{3}He spectrometers is shown in Figure 1 for some examples. Scattered neutrons from the detector and the surrounding material and wall effects give a continuous pulse distribution between the fast and thermal neutron peak. In addition, recoil effects in the elastic scattering of neutrons with protons and ^{3}He nuclei appear beyond the thermal peak for neutron energies larger than approximately 1.o MeV. The total response function was found to be extremely sensitive to the experimental conditions giving rise to large variations in the peak-to-plateau ratios.

3. SPECTRAL EFFECTS

The major difficulties with ^{3}He spectrometers which have an impact on neutron spectroscopy are:

a) wall effects resulting from incomplete energy loss of one or both reaction products within the active counter volume,
b) recoil effects from elastic scattering of neutrons with ^{3}He and proton nuclei,
c) spread in pulse rise time due to location and orientation of the reaction product tracks relative to the central wire,
d) γ-ray sensitivity; γ-rays interact with the spectrometer walls and the counter gas giving rise to pulses smaller than the thermal neutron peak, but high γ-ray input rates cause spectrum distortion due to neutron-γ pile-up,
e) acoustic effects, which have a similar influence as γ-ray pile-up and may lead to spurious pulses in the neutron spectrum.

3.1 Microphonics

The thin, long anode wire in the electric field of the charge collecting voltage of the spectrometer is very sensitive to

vibrations. Effects from acoustic noise were studied with sounds of various frequencies from a sound generator. Figure 2 shows a series of pulser peaks obtained without and with the influence of sound waves of 11.4 kHz of increasing sinus-to-noise amplitude ratios. In addition to a considerable broadening of the peak already observed at low signal-to-noise ratio, pulses of random amplitudes appear at higher ratios lying beyond and below the original peak. In practice these effects have been observed during experiments, even though the noise level in that particular case was lower than the noise level usually occurring at reactor installations.

3.2 Neutron scattering

Distortion of neutron spectra due to room scattering and scattering from the experimental set-up and the counting sample itself is an important effect which cannot be suppressed but only minimized. To investigate this effect, the known delayed-neutron spectrum of ^{17}N [4] was measured under conditions particularly favourable to detection of scattered neutrons. From Figure 3 it is evident that the main intensity of the scattered neutrons is found in the thermal neutron peak region, exponentially decreasing with increasing energy. For our normal experimental conditions (Fig.3, spectrum (a)), the amount of scattered neutrons with $E_n \geq 40$ keV is less than 7%. However, strong spectral distortion is observed under conditions favouring neutron scattering (Fig.3, spectrum (b)): The intensity of the thermal neutron peak is increased by a factor of 50, and neutron scattering also extends in energy to $E_n \approx 1.1$ MeV.

4. EFFICIENCY MEASUREMENTS

Data on the efficiency of the ^3He spectrometers were obtained with monoenergetic neutrons from the $^7Li(p,n)^7Be$ reaction and were normalized to an absolutely calibrated long counter [3]. The data in the energy range 19 keV to 2.77 MeV are shown in Figure 4 along with the $^3He(n,p)^3H$ cross section. Including

uncertainties in corrections applied for the magnet calibration data of the Van de Graaff, degradation of proton energy in the LiF target used, and the absolute efficiency of the BF_3 long counter, our efficiency measurements should be accurate to about 5%.

5. ANALYSIS OF THE PULSE HEIGHT DISTRIBUTION

The method for spectral analysis developed by Greenberger and Shalev [5] uses a set of premeasured detector response functions for monoenergetic neutrons to fit the entire spectrum in a least squares approach. Since the response function showed to be extremely sensitive to the experimental conditions which was disregarded in Shalev's analysis we chose another method: In order to avoid introducing systematic or statistical errors into the spectrum analysis that may be caused by the different corrections of thermal neutron peak, γ-ray pile-up and response function, the experimental spectra were analyzed independently for peak area and centroid using two different codes [3]. With these results, the detector response function was adapted to the experimental conditions by varying the peak-to-plateau ratio. The total plateau distribution obtained in this way (usually containing between 5 and 2o% of the spectrum intensity) is then subtracted from the pulse height spectrum, after its correction for thermal neutrons and γ-ray pile-up determined independently. The residual spectrum is then corrected for efficiency.

6. APPLICATION TO DELAYED-NEUTRON MEASUREMENTS

Up to now, a total of 17 high-resolution delayed-neutron spectra of individual precursors has been measured by the Mainz-group (^{85}As, ^{87}Br, $^{93-97}Rb$, ^{135}Sb, ^{136}Te, $^{137-138}I$, $^{141-146}Cs$). In comparison with the spectra reported by Shalev and Rudstam [6] our data exhibit improved resolution by about a factor of two and we have been able to extend measurements over the entire

energy range from the thermal neutron peak to ($Q_\beta - B_n$). Our data also show considerably more intensity associated with the peak structure which can be traced to the near absence of scattered neutrons in our measurements and the relatively small corrections required to account for the detector response function.

As an example, Figure 5 shows the pulse height distribution from ^{87}Br decay obtained directly by accumulating data from 28o sources along with the final spectrum after correction for thermal neutrons and detector efficiency. The neutron intensity contained in the response function distribution was determined to some 4% of the total spectrum intensity. In comparing our spectrum with that of Shalev and Rudstam [7] fairly good agreement is found for line energies and relative intensities in the energy range above loo keV but our absolute intensities are typically a factor of two smaller [8]. Part of this discrepancy may lie in the rather large percentage (~35%) of the total intensity that lies below loo keV which was not resolved from thermal and background events toward lower energies in the spectrum of Shalev and Rudstam. However, it would appear that this cannot account for the complete difference; it may be due to the additional effect of scattered neutrons in the energy range loo - 4oo keV giving rise to a considerably larger 'continuum' underlying the peak structure in their spectrum.

The spectrum from ^{135}Sb decay shown in Figure 6 represents the hardest neutron spectrum known so far amongst fission product precursors. The three strongest lines above 1 MeV contain some 25% of the total spectrum intensity, and about 55% are contained in the range 1.o - 2.5 MeV. In comparison with the spectrum of Shalev and Rudstam [7], the neutron intensity below 1 MeV in our data is only about one half that in their spectrum as indicated by curve (a) in Figure 6. That this discrepancy is not only due to response function corrections is indicated by curve (b) representing the envelope of our ^{135}Sb spectrum prior to subtraction of the total plateau distribution. Again, the strong spectrum distortion in the low energy part must be due to background from scattered neutrons.

7. AVERAGE NEUTRON ENERGIES AND INTEGRAL NEUTRON SPECTRA

The discrepancies in the experimental data which appear in most spectra that have been measured by Shalev and Rudstam [6] and by the Mainz-group have a strong influence not only on the interpretation of delayed-neutron spectra, they also show an impact on the data needed for application, i.e. average neutron energies and the calculation of the effective neutron energy distribution in nuclear fuel from an 'analytical representation' of the spectra [9]. As can be seen from Figure 6, curve (c), which represents an older description of Shalev and Rudstam's experimental ^{135}Sb delayed-neutron spectrum by a set of l_n-waves [7], this analytical representation strongly overestimates the contribution of low l_n-waves because of the background problems discussed before. Thus, any integral delayed-neutron spectrum obtained from summation of individual spectra [10] or their analytical expression [9] cannot be used as an 'accurate representation of the non-moderated delayed-neutron spectrum in fission' as is stated in RP 13, p.44.

To illustrate the discrepancies, Table 1 gives the average neutron energies, $\bar{E}_n$, deduced from our spectra along with those presented in Table 4 of RP 13. It is seen that the neutron spectra obtained by the Mainz-group are in the average 95 keV harder than those from the OSIRIS-group; the most prominent differences are observed for ^{85}As, ^{94}Rb and ^{135}Sb. Again, these discrepancies arise from the combined effects of insuffiently known detector response functions and backround due to scattered neutrons. A comparison with the results of the SOLAR-group using a direct method [11] is difficult. However, as their detector arrangement was mainly calibrated with neutron spectra of Shalev and Rudstam [7], it is somewhat embarrassing that their results generally are in better agreement with our data.

The sensitivity of $\bar{E}_n$ to the energy range of the delayed-neutron spectrum measured is demonstrated in Table 2 for two representative nuclides. In the case of the rather soft ^{87}Br spectrum the effect due to neutrons from the energy range o - loo keV is dominant, whereas for the hard spectrum of ^{135}Sb

the sensitivity of the high energy region is clearly illustrated. It should be noted here that the experimental spectra published by Shalev and Rudstam [6] only cover the energy range 100 to 1600 keV. From our results we conclude that the uncertainties in the average neutron energies given in Table 1, including statistical and systematic errors, are in the order of 20 to 50 keV.

8. CONCLUSIONS

The discrepancies noted in delayed-neutron spectroscopy using ^{3}He ionization chambers would request:

(1) further studies on the properties of the ^{3}He spectrometers, especially the influence of experimental conditions on their response function and detector efficiency;
(2) measurement of the low energy part (0 - 100 keV) for the remaining precursors of interest;
(3) a considerable reduction of uncertainties in average neutron energies to about 20 keV;
(4) measurements of equilibrium and time-dependent spectra, as the summation methods using current spectra from individual precursors contain large uncertainties in both their low (0 - 300 keV) and high energy parts ($\gtrsim$ 1600 keV).

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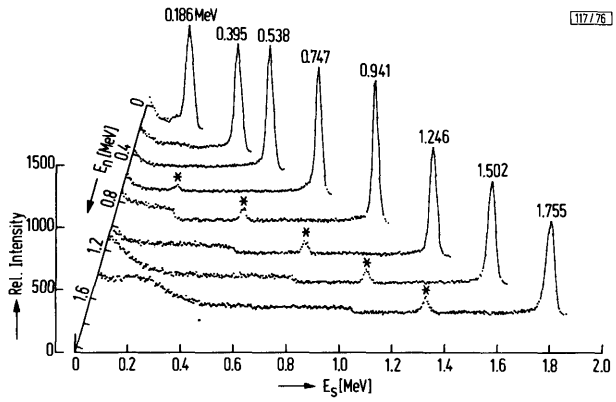


FIGURE 1: Response of the ^{3}He ionization chamber to monoenergetic neutrons produced by the ^{7}Li(p,n)^{7}Be reaction. E_n is the incident neutron energy deduced from the measured proton energy and the reaction kinematics; E_s is the pulse height distribution obtained from monoenergetic neutrons with E_n. The peaks marked with an asterisk are due to a secondary reaction ^{7}Li(p,n)^{7}Be*.

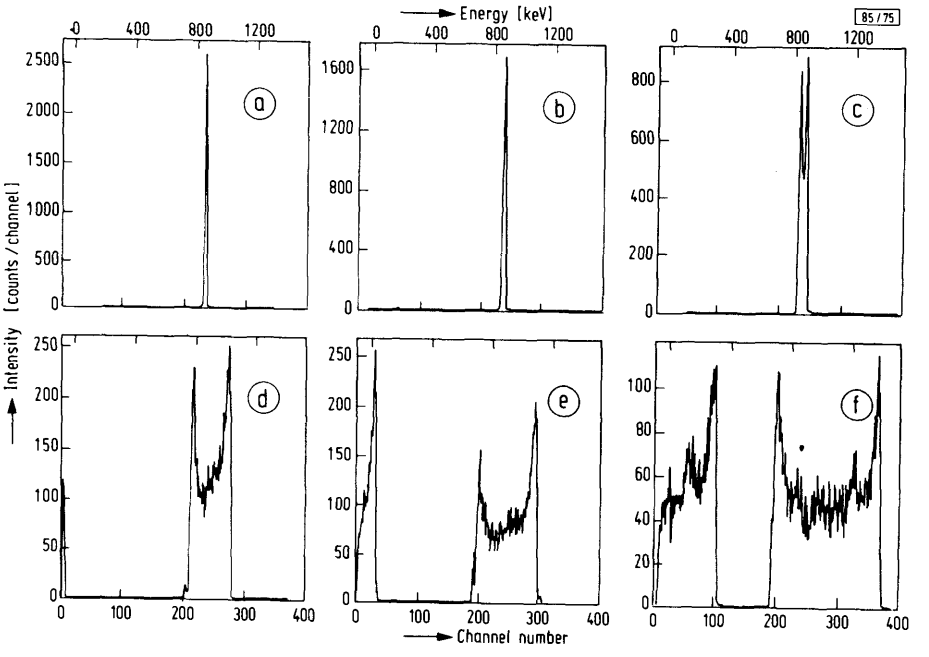


FIGURE 2: Distortion of a pulser peak by sound waves at various sinus-to-noise amplitude ratios, U_s/U_n; sound wave frequency 11.4 kHz; (a) U_s/U_n= o, (b) 1.3, (c) 3.8, (d) 1o, (e) 13, (f) 2o.

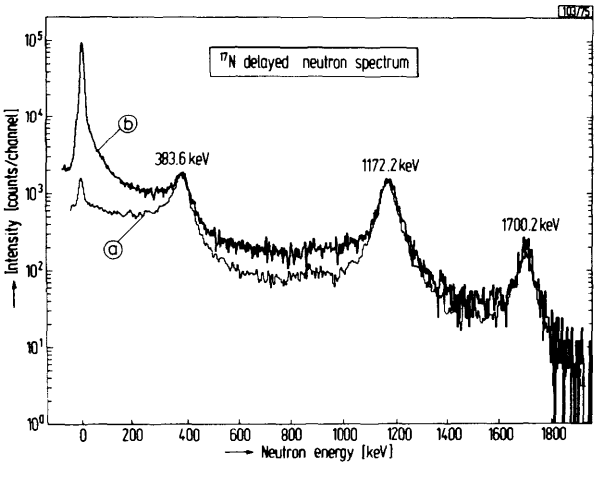


FIGURE 3: Delayed-neutron spectrum of ^{17}N; (a) measured under optimum experimental conditions, (b) measured under conditions favouring scattering.

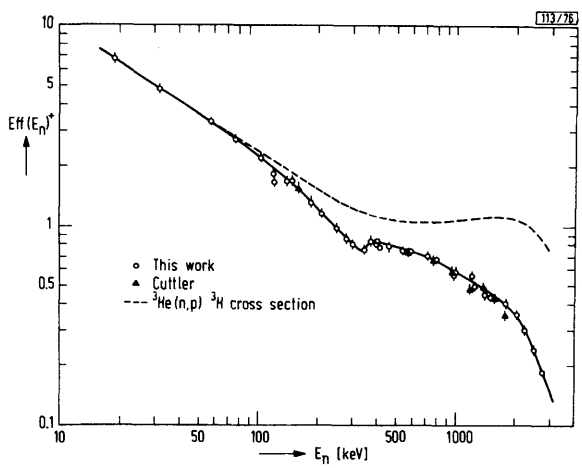
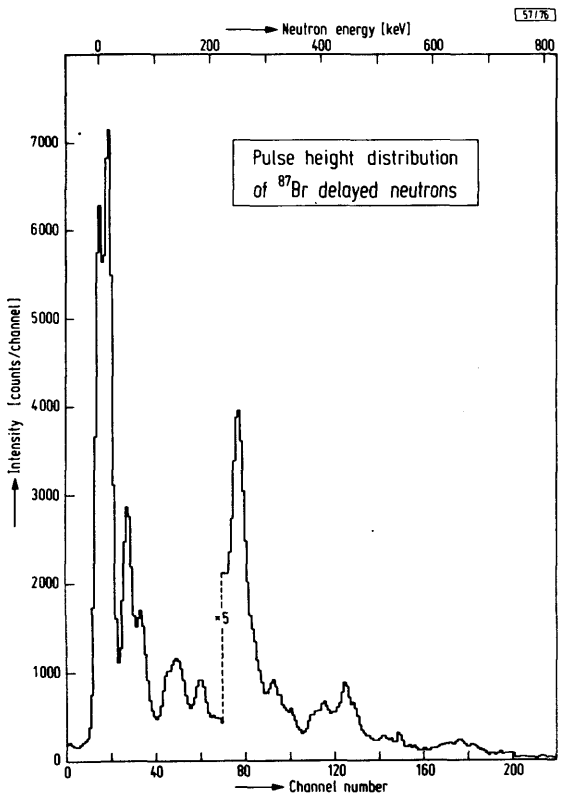


FIGURE 4: Relative efficiency of the ^{3}He ionization chamber, along with the $^{3}He(n,p)^{3}H$ cross section.



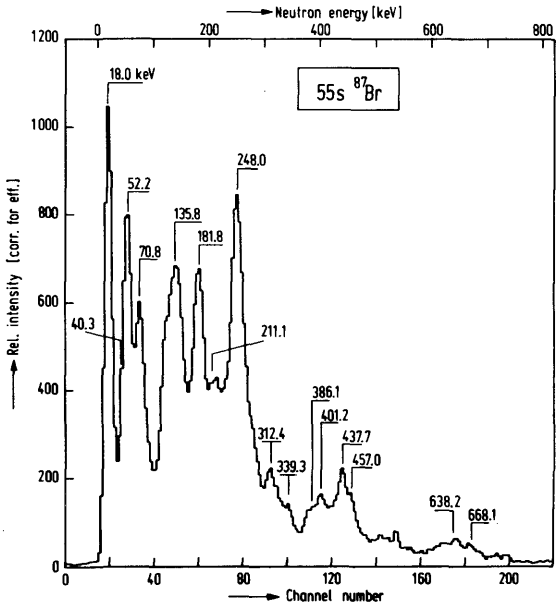


FIGURE 5: Delayed-neutron spectrum from ^{87}Br decay. Experimental pulse height distribution (left part); neutron spectrum after correction for detector efficiency and thermal neutrons (right part).

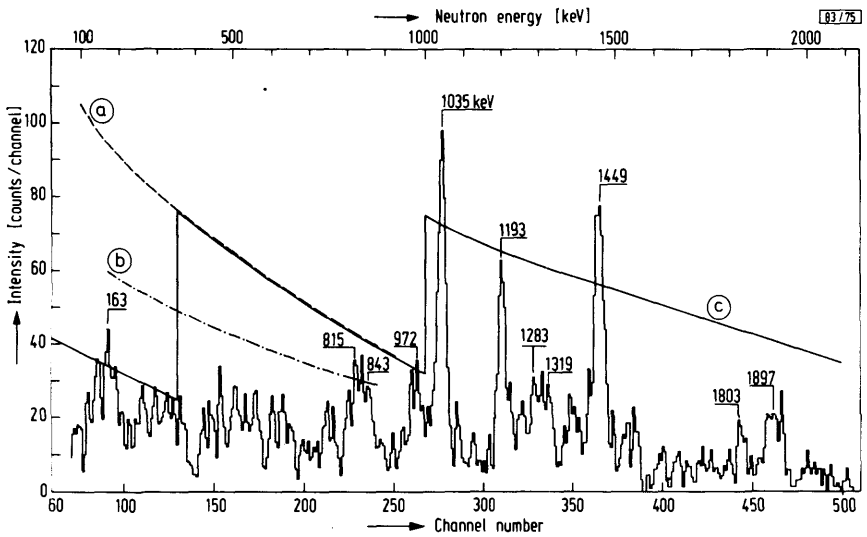


FIGURE 6: Delayed-neutron spectrum from ^{135}Sb decay after correction for thermal neutrons and total detector response function. Experimental spectrum envelope according to Shalev and Rudstam (a); envelope of the Mainz-spectrum prior to response function corrections (b); predicted spectrum envelope (c) according to Shalev and Rudstam [7].

TABLE 1: Average neutron energy $\overline{E}_n$ from individual delayed neutron precursors.

Precursor	Average neutron energy $\overline{E}_n$ [keV]				
	Mainz	OSIRIS		SOLAR	
		Exp. spectrum	Anal. repr.	Method A	Method B
^{85}As	752	610	570		
^{87}Br	217	170	220		150 ± 10
^{93}Rb	414	340	330	560 ± 10	630 ± 10
^{94}Rb	474	350	310	570 ± 10	610 ± 10
^{95}Rb	508	520	540	530 ± 10	540 ± 10
^{96}Rb	481			560 ± 10	540 ± 10
^{97}Rb	754			> 720	> 620
^{135}Sb	1033	610	600		
^{136}Te	305	220	210		
^{137}I	579	510	490		530 ± 50
^{138}I	467	390	370		
^{142}Cs	252	(200)	(200)	240 ± 60	130 ± 20
^{143}Cs	276	240	220	350 ± 10	320 ± 20
^{144}Cs	312	280	310	290 ± 20	330 ± 20
^{145}Cs	331			460 ± 30	540 ± 20

TABLE 2: Sensitivity of the average neutron energy $\bar{E}_n$ to the range of the energy spectrum.

^{87}Br		^{135}Sb	
Neutron energy range [keV]	$\bar{E}_n$ [keV]	Neutron energy range [keV]	$\bar{E}_n$ [keV]
100 - 750	266	100 - 1600	862
0 - 750	206	0 - 1600	850
100 - 1300	270	100 - 3060	1036
0 - 1300	217	0 - 3060	1033

Contribution to Review Paper 13

THE SYSTEMATICS OF RELATIVE DELAYED NEUTRON YIELDS

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ABSTRACT

The author considers the dependence of relative delayed neutron yields on charge and mass of the fissionable nucleus with a view to predicting these data in the case of nuclei for which there are no experimental data available. The author indicates the nuclei for which it is preferable to carry out extra experiments so as to avoid performing them for the entire range of nuclei, while still obtaining the data required for reactor design.

When determining the content of fissionable materials in spent fuel and when calculating reactor kinetics, it is essential to possess data on the relative delayed neutron yields also for such materials for which experimental determination of the data is difficult because of high background or prompt neutrons from spontaneous fission, or because only a small amount of material is available. Furthermore, there is clearly no point in making measurements for all materials if some of the data can be predicted with sufficient accuracy on the basis of measurements of a selected number of them. It is therefore of interest to systematize the existing data and seek to establish the relationships between the yields and the mass and charge of the fissionable materials so as to be able to predict the data that are unavailable, though still required for reactor design.

In the coordinate system Z_F, N_F (charge and number of neutrons of the fissioning compound nucleus) shown in Fig. 1 the black dots indicate the nuclei for which the decrease in delayed neutrons activity has been measured and parametrization carried out, i.e. for which the relative yields of the groups and their half-lives have been determined by the method of least squares. Clearly, it is not for these parameters - which have no physical meaning since they represent weighted mean yields from several pure precursors - that we should establish the systematics, but for the precursors themselves. It should be noted that for the purpose of systematizing and predicting delayed neutron yields from any precursor it is rather the cumulative than the independent yields of precursors that we have to consider, since the independent yields involve also the beta decay features of fragment nuclei, while the cumulative yields are associated with the fission process only. The link between

the delayed neutron yield y_i from the given i-th precursor, its cumulative yield Q_i and the delayed neutron emission probability of a given fragment, P_{ni}, is given by the relationship

$$y_i = Q_i \, P_{ni} \qquad (1)$$

The procedure for solving the given problem is as follows: delayed neutrons are emitted by groups of light and heavy fragments; among the former we find the isotopes of bromine, rubidium and so on, while the second group contains those of iodine, caesium and so forth. The distribution of the cumulative isotope fragment yields can be satisfactorily described by a Gaussian of definite width [1]. If we can determine the dependence of the position of the maximum of this Gaussian, A_p, on the charge and the number of neutrons of the fissioning compound nucleus, for example for bromine isotopes, we can calculate the yield of any of the Br-isotopes and use Eq.(1) to find the relative delayed neutron yields. The position A_p for rubidium isotopes can be found from the linear dependence between the fragment charge and A_p for the known value $K = \partial Z_p / \partial A$ [2]. We can thus find the relative yields of delayed neutrons from all light fragments. To determine the same yields from heavy fragments we have to apply the procedure of calculating A_p for complementarily-charged light and heavy fragments using the mean number of prompt neutrons, as described in Ref.[2]. In this way we can calculate the relative cumulative yields of all precursor fragments, and use Eq.(1) to determine the relative delayed neutron yields from all individual precursors, and then find, from their known half-lives, the decrease in the delayed neutron activity for materials whose measurements are either difficult to make or are not worth making if they can be reasonably accurately predicted.

Let us consider how we can determine the ratio of cumulative yields for all bromine isotope precursors if we know the yields ratio for only two of them, for example:

$$Q\,(^{88}Br) \;/\; Q\,(^{87}Br)$$

This ratio can be found by analysing the decrease in activity of the delayed neutrons by the method described in Ref.[3]. In effect, the analysis by a least squares method indicates in this case a split in the yield of the second group of delayed neutrons into yields from ^{137}I and ^{88}Br. We should make the reservation that the yields of these long-lived precursors can be determined more accurately than the remainder due to their large integral contribution to the activity decrease. Assuming that the distribution of cumulative precursor yields at Z = const tallies satisfactorily with the Gaussian [1], we get the following relationship for the bromine isotope precursors:

$$\ln\,(Q_i/Q_o) \;=\; -\,i/\sigma_z^2 \;\left\{\,(A_o - A_p) + i/2\,\right\} \qquad (2)$$

where

Q_o is the cumulative yield of bromine isotope fragment with mass $A_o = 87$;

Q_i is the cumulative yield of bromine isotope fragment with mass $A_o + i (i = 1,2,3 \ldots)$;

A_p is the most probable mass for a given fragment charge (in this instance at $Z = 35$);

σ_z is the distribution width for the cumulative isotope yields.

It can be seen from Eq. (2) that with known values of Q_i/Q_o and σ_z we can find any Q_i/Q_o, i.e. the ratios of the precursor yields $Q(^{89}Br)/Q(^{87}Br)$, $Q(^{90}Br)/Q(^{87}Br)$, and so on. We should point out that since σ_z has a uniquely defined relationship with the distribution width of the prompt fission neutrons [2], and the latter is the same for all fissionable nuclei from thorium up to californium [4], σ_z does not depend on the nucleonic composition of the fissionable nuclei and can be taken in all cases as being the same.

It follows from Eq. (2) that

$$A_p = A_o + 0.5 + \sigma_z^2 \ln (Q_1/Q_o) \qquad (3)$$

Hence determination of the relationship $A_p (Z_F, N_F)$ is reduced to determining it for the quantity $\ln(Q_1/Q_o)$.

Using the method described earlier [3], we determined the ratio Q_1/Q_o from an analysis of the decrease in activity, which we calculated by computer from the parameters obtained from experimental data of other authors using the method of least squares [5]. We took from Ref. [5] all the data for fission induced by fission-spectrum neutrons, and for thermal neutron fission of ^{241}Pu; the data for ^{242}Pu were estimated.

Figures 2 and 3 show the dependence of Q_1/Q_o for the fission of uranium and plutonium isotopes. One can clearly see a linear dependece (on a semi-logarithmic scale) on the number of neutrons in the fissioning compound nucleus, Eq. (4), from where a linear dependence of A_p follows.

$$\ln (Q_1/Q_o) = \ln A_{po} (Z_F) + k (Z_f) \cdot N_f \qquad (4)$$

A deviation from the estimated data is observable in the case of ^{242}Pu.

Thus, the calculation of A_p for bromine isotopes in order to predict their relative yields from fission of uranium and plutonium isotopes for which there are no direct experimental data can be made from Eq. (3). According to our calculations

$$\ln(Q_1/Q_o) = -(32.811 \pm 0.011) + (0.2295 \pm 0.0076) N_F \text{ for uranium isotopes,}$$

and

$$\ln(Q_1/Q_o) = -(72.879 \pm 0.010) + (0.501 \pm 0.018) N_F \text{ for plutonium isotopes.}$$

An indication of the accuracy in each case is the deviation of the value calculated on the basis of these equations from the value A_p found directly

from decomposition of the decay curves. They differ within the limits of 0.02 - 0.05 mass units.

Having found that A_p for bromine isotopes is a linear function of the number of neutrons N_F in the fissioning nucleus, it is of interest to determine the form of this dependence on the charge of the nucleus Z_F and to determine the corresponding coefficients; the problem would then be completely solved. In accordance with Eq.(4), in order to do this we should determine the dependence $A_{po}(Z_F)$ and $k(Z_F)$. Since we have only two points, or to be more exact two straight lines, we do not have enough information to determine even the type of function involved. In Ref.[6], the authors derive Z_p for A = 136 (caesium isotope) for fission of ^{227}Th. Applying the relationships derived by us in Ref.[2], we can find A_p for caesium isotopes and for the complementarily charged bromine isotopes. Unfortunately, the accuracy here is not very high, compared with the accuracy attained for uranium and plutonium isotopes, since the value of $\bar{\nu}$ used is only known approximately ($\bar{\nu} \simeq 1$). If we now assume that for the fission of thorium isotopes there is likewise a linear dependence of A_p on the number of neutrons in the compound nucleus, we can find $k(Z_F = 90)$ and the corresponding values of A_{po}.

$$\ln (Q_1/Q_o) = -19{,}6985 + 0{,}13957\ N_F.$$

It is clear from Fig. 4 that the slope of the linear dependence $A_p(Z_F)$ on N_F increases with Z_F. The simplest assumption, seeing that $\ln A_{po}(Z_F)$ and $k(Z_F)$ do not fit into a linear dependence (Fig.5), is that for the three materials with Z_F = 90, 92 and 94, these values can be described by a parabola, i.e.

$$\ln A_{po}(Z_F) = aZ_F^2 + BZ_F + c$$

$$k(Z_F) = pZ_F^2 + qZ_F + r$$

Calculation gives

$$\ln A_{po}(Z_F) = -3.3695(Z_F^2 - 180.054\ Z_F + 8110.73)$$

$$k(Z_F) = 0.022684(Z_F^2 - 180.018\ Z_F + 8107.74) \qquad (5)$$

In this way we can find the position of A_p for bromine isotopes in the case of any fissionable isotope ranging from Z_F = 90 to Z_F = 94 for different values of Z_F and N_F. The accuracy attained in solving this problem can be improved by obtaining more reliable data for ^{227}Th.

Unfortunately, further extrapolation - calculation of A_p for ^{252}Cf - produces a result differing substantially from the true one ($\sim$ 25%). On the other hand, we have merely to change $k(Z_F)$ by only 4% to obtain the correct result for the same value of A_{po}.

The analysis made demonstrates that there is no need to measure the yields of delayed neutrons from the entire range of fissionable isotopes, even if more data are required. It shows for which ones new measurements would be of value in order, first, to obtain the data necessary for calculating all the others, and, second, to study the physical systematics of the variation in cumulative fragment yields as a function of the charge and mass of the fissioning nucleus. For the purpose of systematics it would be very useful to study the delayed neutron yields from ^{227}Th and one other californium isotope. This "miniprogramme" would permit calculation of the relative delayed neutron yields for all the isotopes of fissionable materials from thorium to californium.

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FIGURE CAPTIONS

Fig. 1 Fissionable nuclei for which the decrease in activity of the delayed neutrons has been measured

Fig. 2 Dependence of Q_1/Q_o and Q_2/Q_o on N_F for uranium isotopes

Fig. 3 Dependence of " " " for plutonium isotopes

Fig. 4 Dependence of $k(Z_F)$

Fig. 5 Dependence of $k(Z_F)$ and $\ln A_{po}(Z_F)$

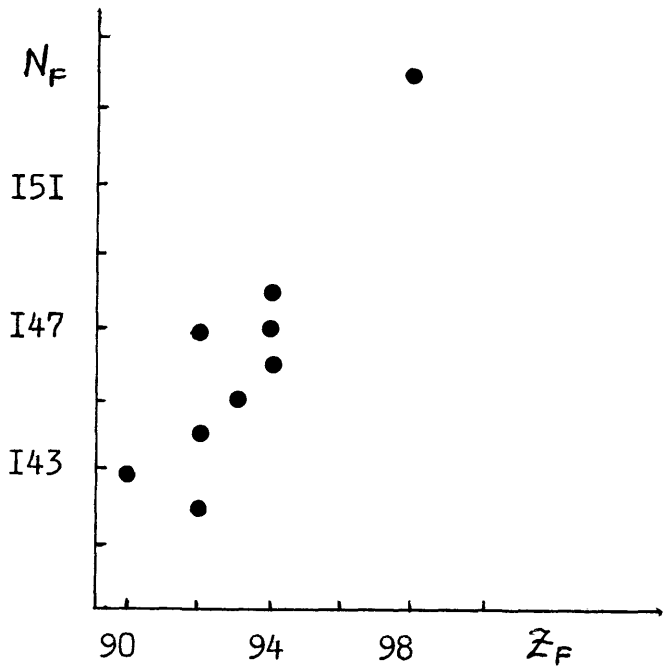

Fig. 1 Fissionable nuclei for which the decrease in activity of delayed neutrons has been measured

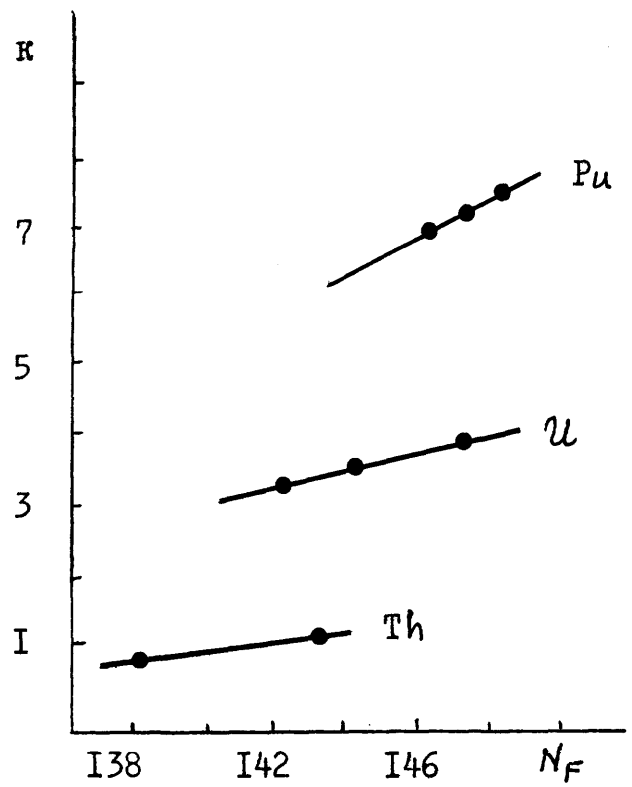

Fig. 4 Dependence of $k(Z_F)$

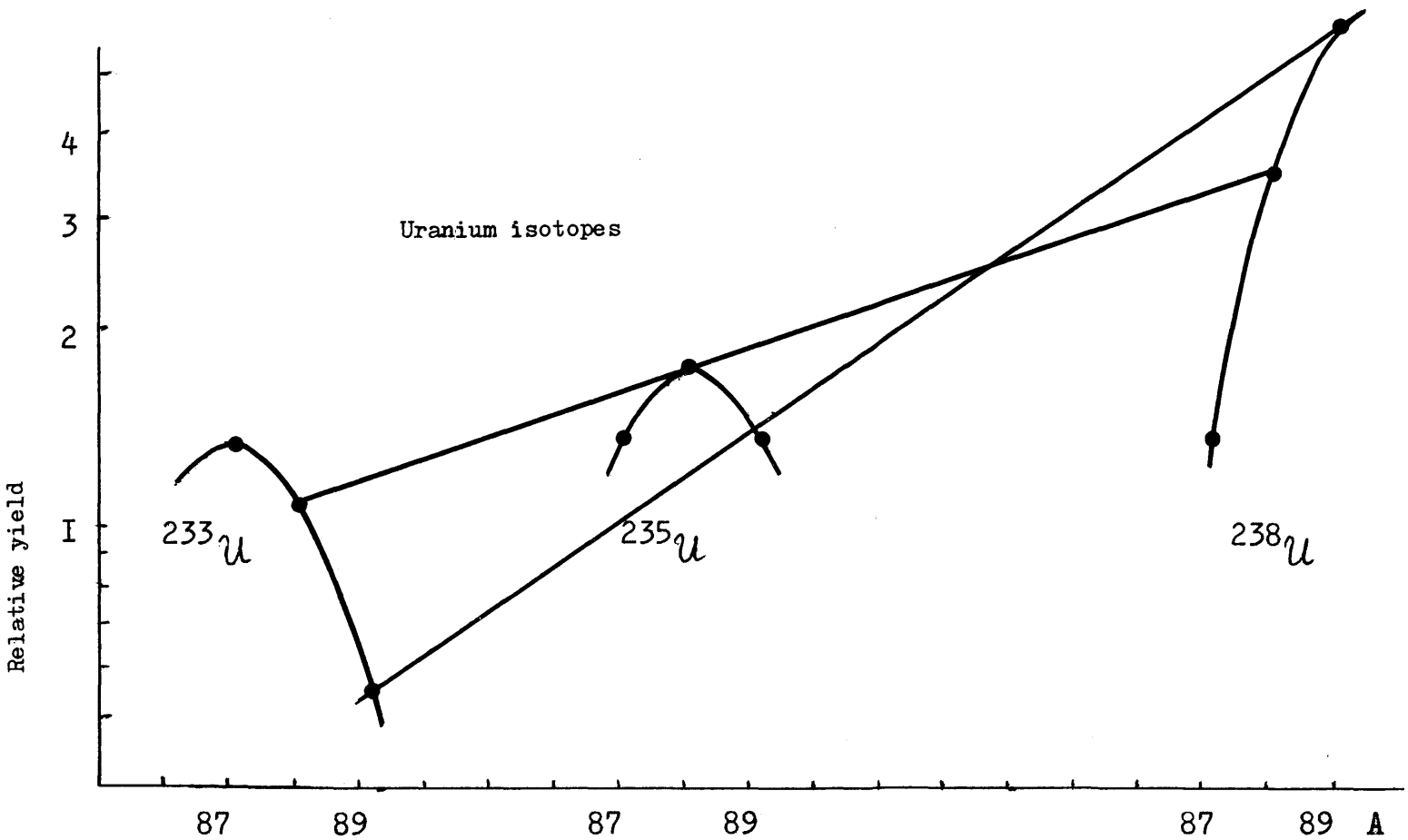

Fig. 2 Dependence of Q_1/Q_o and Q_2/Q_o on N_F for uranium isotopes

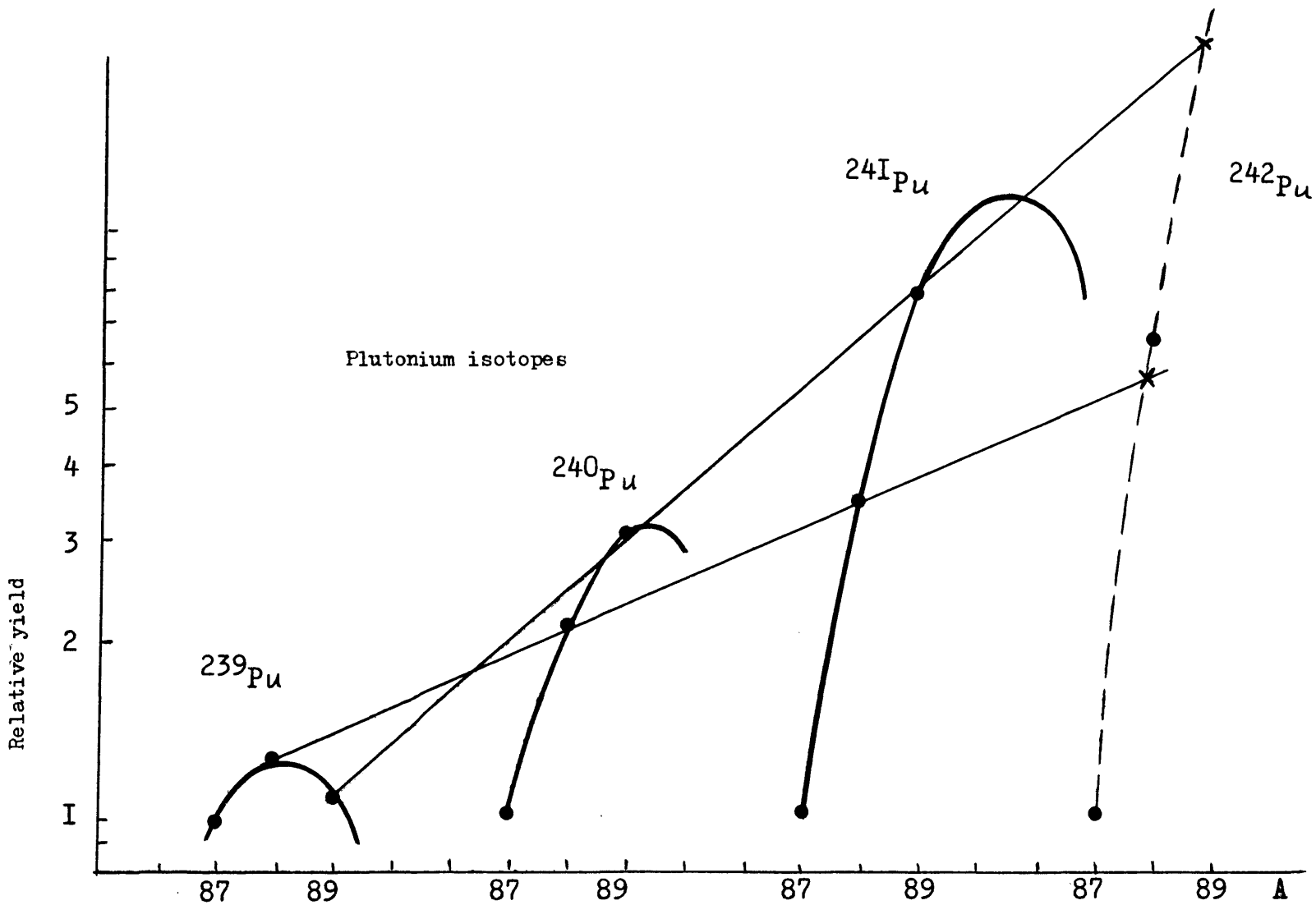

Fig. 3 Dependence of Q_1/Q_o and Q_2/Q_o on N_F for plutonium isotopes

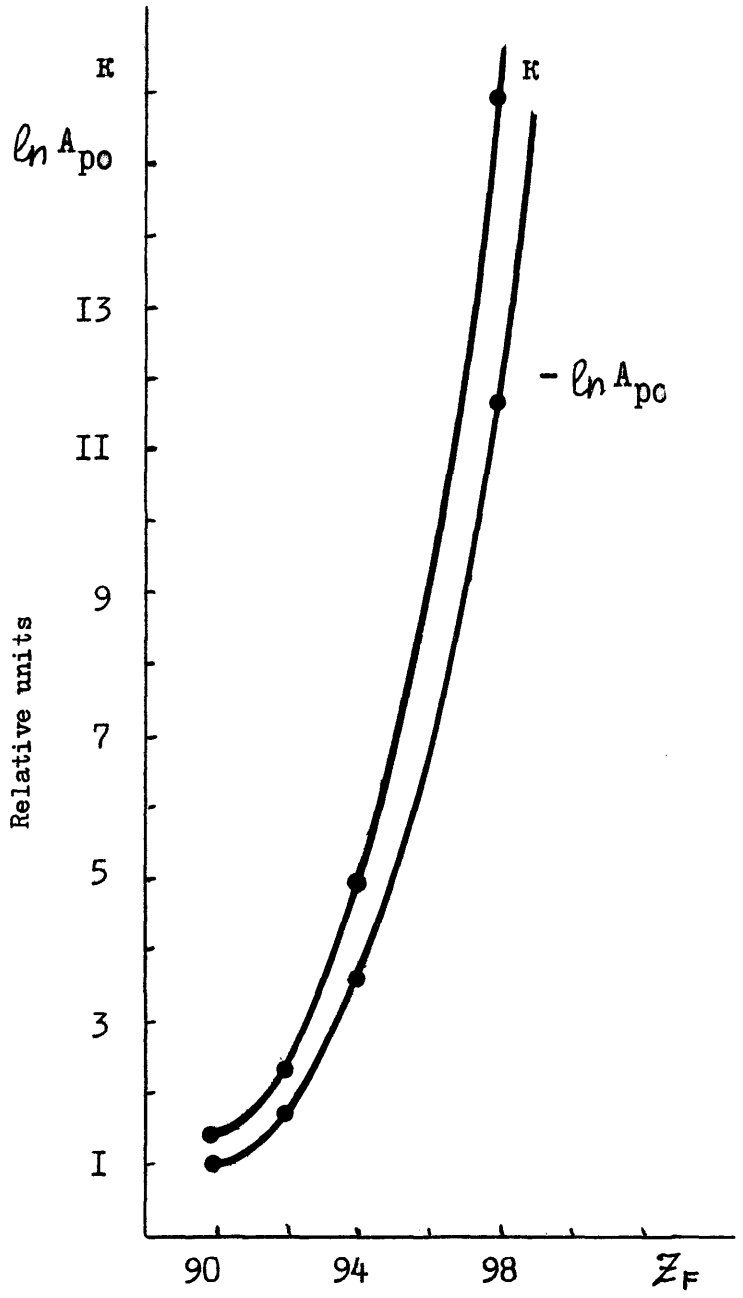

Fig. 5 Dependence of $k(Z_F)$ and $\ln A_{po}(Z_F)$

Contribution to Review Paper 14

THE FRENCH PROGRAMME ON THE DETERMINATION OF THE EFFECT OF FISSION PRODUCTS IN FAST REACTORS

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INTRODUCTION

Among the main neutron characteristics studied by reactor physics, those associated with neutron evolution as a function of fuel irradiation play a basic role in the optimization of designs, fuel cycles and operation [1, 2]. The activities undertaken in this field by the Commissariat à l'Energie Atomique (CEA) are closely linked with the development of the French nuclear programme, especially the construction of 1200- and 1800-MW(e) fast reactors of the Super-Phénix type [3]. The increase in power from Phénix to Super-Phénix and in the duration of the operating cycle, and the optimization of doubling time now being investigated, have enhanced the importance of the parameters associated with neutron evolution, particularly with reactivity loss due to fission products.

In a 1200-MW(e) reactor, for example, fission products account for 75% and the variation in heavy isotope composition for 25% of the reactivity loss in a cycle of 308 days ($\simeq$2.5% $\Delta k/k$). This justifies the work undertaken in France on the calculation methods, evaluation and adjustment of fission products. An accuracy of the order of $\pm$7% (2σ) is required for the total effect of fission products.

The approach which has been followed at the CEA since 1973 for the study of reactor evolution is identical with that used to treat the other parameters of reactor physics [4]. The accuracy required for fission products should be attained first by adjusting the fission product cross-sections separately and then by regrouping them (pseudo-fission product) for final readjustment by irradiated fuel measurements.

In addition, the problems associated with the fuel cycle (transport, reprocessing and fabrication) and waste treatment call for determination of the nature and quantity of the fission products present in fuels as a function of irradiation and of cooling time.

I. STEPS CHOSEN

The starting point for the studies is a set of calculation methods and evaluated neutron cross-sections. The set of methods and data used for fast reactors is called the CARNAVAL code. We then determine the main sources of uncertainty in the macroscopic parameters which are due to the uncertainties in basic data. Lastly, we measure the integral parameters which can be used to adjust the cross-sections and to calculate with the required accuracy the power plant characteristics depending on reactor evolution.

Subsequently, an overall check of the data set (fission products, heavy nuclei, actinides, etc.) by investigations in power reactors may be envisaged.

The CEA study on fission products was carried out according to this general scheme and consisted of the following steps:

Step 1

Establishment of a reference code (188 fission products) based only on evaluated data and theoretical methods.

The adjustment scheme chosen made it necessary at first to have a fairly complete set of cross-sections for the individual fission products, so that the pseudo-fission product cross-section could be described correctly as a function of energy (the pseudo product cross-section is obtained by weighting the capture cross-sections of the individual isotopes by their fission yield, account being taken of a variety of phenomena - neutron capture, radioactive decay, migration in the fuel - which modify their accumulation during irradiation).

An initial file of 188 isotopes was therefore established on the basis of the French evaluation by P. Ribon [5] for the 22 most important isotopes, supplemented by the Australian evaluation by J.L. Cook [6]. This reference file was subsequently updated when more recent evaluations became available.

Step 2

Study of sensitivities during the evolution. This study guides the choice of the most important individual fission products. It uses the reference code which is supposed to be sufficient at this stage for choosing a programme of specific integral experiments.

Step 3

Adjustment of the individual fission product cross-sections. The preceding step contributed to the choice of integral experiments. The energy range covered by these experiments on each individual fission product is extended. The main fission products are adjusted, each one separately, by the corresponding experiments. The adjustment of the individual products essentially corrects the shape and partly the value of the capture cross-section as a function of energy. The final adjustment of the cross-section is obtained at the final step 5.

Step 4

Generation of pseudo-fission products, defined as the sum of the individual fission products weighted by their effective yields. Considering the preceding adjustment, the correct energy variation is preserved for the pseudo-fission product by condensing the individual adjusted fission products. The condensation is performed with the effective yields for each fissionable nucleus by distinguishing the contributions of ^{235}U, ^{239}Pu and ^{241}Pu.

Step 5

Readjustment of the lumped capture cross-section of pseudo-fission products. Such readjustment is envisaged by some integral experiments performed on fuel irradiated in Phênix and Rapsodie under conditions as close as possible to reality. This will enable us to reconstitute the absolute level of the lumped cross-section. These integral experiments represent the basis for the whole fission product programme.

Remarks on the reference code

The initial sensitivity studies were carried out on the basis of the CEA internal reference file (Ribon + Cook). As a result, the 188 fission products were classified into four groups of unequal importance:

Group 1: The most important (81.3% of capture in Super-Phênix), i.e. about twenty fission products considered to deserve priority;

Group 2: Less important (14.4% of capture), i.e. about twenty fission products which can be substantially improved but do not deserve priority;

Group 3: Fission products of little importance (2.5% of capture) comprising approximately another twenty for which no integral measurement seems necessary but whose evaluations can however be improved in future;

Group 4: The remainder of about 120 fission products which have a low capture effect and whose reference evaluation is sufficient [5, 6].

The first adjustment to the available integral measurements for the first group was found to be of large amplitude and revealed the importance of the initial file and the integral results. For this reason, a new reference code, based on a more recent and reliable evaluation (CNEN/CEA collaboration), and new French integral measurements (Phênix-Profil, Ermine-Masurca) have since been performed.

The new code now covers groups 1, 2 and 3 (about sixty fission products) which have been fully re-evaluated, while most isotopes of the first two groups have, in addition, been adjusted (about forty fission products) to more than 300 integral measurements of individual fission products in the STEK, FRO, CFRMF, Ermine, Masurca and Phênix reactors.

Remarks on readjustment step (5)

Readjustment by "global" experiments is a specific feature of the French programme. In comparison with the programmes which are based exclusively on measurements of individual isotopes, this programme is, in our opinion, more realistic for the following reasons:

- In practice, only knowledge of total fission product capture in a reactor is important;
- In order to go over to the total effect of fission products, we are not compelled to involve quantities such as fission yields and accumulation parameters (gas migration), which are still not well known;

- The effect of fission products is measured under conditions typical of a reactor fuel, with respect to both spectrum and self-shielding;
- We avoid measuring numerous individual fission products which are of little importance, rare, costly and difficult to obtain in the pure state.

At present, we are moving over to step (5).

II. EVALUATIONS

A special effort was undertaken in this field (full evaluations covering the entire energy range: E_{th}-15 MeV). When the work is completed at the end of 1978, the final set (about sixty fission products) of groups 1, 2 and 3 will be available. While initially, until 1975, only the facilities available at the CEA [5] were used, the subsequent activities were carried out in co-operation with the CNEN within the general framework of the CNEN/CEA agreement on fast neutrons. As a result of this co-operation, the methods were brought in line [7], the codes were unified and the work was shared.

The first phase has now been completed, i.e. the first set of CNEN-CEA evaluations [8, 9] has been carried out for groups 1, 2 and 3 on the basis of only microscopic data.

The second phase, which will end with the publication of the final set, has started. It consists of re-examination of the first CNEN-CEA set in order to take into account both the most recent microscopic data and the tendencies indicated by the various local systematics, together with those derived from the above-mentioned integral experiments. It is to be noted that in general the two types of tendency are consistent.

III. MEASUREMENT TECHNIQUES

These are the standard techniques used in the Minerve and Masurca critical assemblies and in the Rapsodie and Phênix power reactors.

In regard to fission products, the programmes include:

- Measurements by oscillation of samples of separated fission products;
- Measurements by activation of detectors made from a pure fission-product isotope;

- Measurements by irradiation of deposits of separated fission products in Phênix (the capture rates can be deduced from the variation of the fission product contents);
- Measurements by oscillation of irradiated fuels to obtain the overall fission-product capture [10].

These techniques involve:

- Measurements by non-destructive methods: gamma spectrometry, calorimetry and spontaneous neutron emission;
- Destructive analyses [11] used essentially to give absolute values: mass-spectrometry, chemical analyses and alpha-spectrometry.

These measurements were carried out under the fast reactor programme in the Rapsodie core assembly and similar work is now in progress in Phênix.

IV. THE METHOD OF ADJUSTMENT OF INDIVIDUAL FISSION PRODUCTS: GENERATION OF PSEUDO-FISSION PRODUCTS

The integral experiment results are used for statistical adjustment of the capture cross-sections of the individual fission products in the reference code. The adjustment method is of the BARRACA type [4], i.e. it uses a small number (4) of energy groups. The adjustment result is redistributed among the 25 groups of the code. The shape of the reference evaluation is taken into account by means of correlation between groups. The capture level is modified until negligible residues are obtained (minimum Ki2). The calculation of the uncertainty in the adjustment is derived directly from the experimental uncertainties. The file thus adjusted is used as input for the calculation of reactor evolution in order to generate the pseudo-fission products. During these calculations, account is taken of:

- the heavy nuclei present and their evolution;
- fission product half-lives, yields, captures and migrations;
- the operating diagram including shut-downs;
- the spectra of the reactor zones (core 1, core 2, blankets).

The resulting pseudo-fission products are different with respect to the generating heavy nucleus (^{235}U, ^{238}U, ^{239}Pu, ^{241}Pu) and describe, for 25 groups,

the following reactions: capture, elastic, inelastic, total and transport. By definition a pseudo-fission product, which aims to represent the total effect of all the fission products present, depends on the design and operating diagram of the reactor. The pseudo-fission products to be included in version IV of the CARNAVAL code correspond to Super-Phênix in equilibrium.

V. THE EXPERIMENTS USED FOR INDIVIDUAL FISSION PRODUCTS

For the adjustment of the capture cross-sections of individual fission-product isotopes, we interpreted a fair number of foreign experiments. In the case of the most important products, they were supplemented by an appropriate experimental programme in the characteristic lattices of fast power reactors.

The capture measurements can be obtained by oscillation, activation or irradiation, as appropriate.

(1) Foreign experiments

Initially, our programme included oscillation experiments in the thermal-fast coupled assembly STEK in five cores of different spectra. This experimental programme [12] was interpreted under an RCN-CEA agreement.

We took into consideration the majority of these experiments, which were devoted to the important fission products (groups 1 and 2) and for which the five cores covered an extended spectrum range. In addition there were a smaller number of oscillation experiments in the FRO reactor [13], measured in three different cores. We supplemented them by activation experiments in the CFRMF thermal-fast coupled assembly of the ARMF II reactor [14] for some isotopes of group 1.

The above accounts for about 250 experiments distributed unequally between groups 1 and 2. Their results are of low accuracy, varying between 10 and 30%. In the case of the oscillations (STEK + FRO) this is due to the small mass of the samples, the uncertainty in their self-shielding and the parasitic signal from flux depression. In the case of activations (CFRMF), the sample spectrum is affected by uncertainties.

(2) CEA's own programme

Three types of measurement were used, depending on the nuclear characteristics and the possibilities of supply of the different separated fission products.

(a) Oscillation measurements in Masurca and in the thermal-fast coupled assembly Ermine of the Minerve reactor. This type of measurement was limited to products of which a mass of a few grams could be obtained in order to attain the desired accuracy. This method requires corrections for self-shielding and flux depression in order to arrive at the capture effect and has an accuracy of up to ±10%.

(b) Activation measurements in the thermal-fast coupled assembly Ermine. This method directly gives the capture effect, but its application is limited to isotopes whose capture products are radioactive and can be measured by gamma-spectrometry. It has an accuracy of up to ±5%.

(c) Irradiation: since we have an irradiation facility like Phénix, this method can be used for rapidly attaining the necessary fluence and is of particular interest in that it has a neutron spectrum in the desired range. It requires only small quantities of material, and, moreover, mass spectrometric analyses give a very high accuracy for the integral capture cross-sections (3-4%). A particular feature of this method is that it enables us to obtain the isotopes of a chain of successive captures with very satisfactory accuracies.

Although they are less numerous (about fifty) and limited mainly to group 1, the measurements performed under this particular programme are nevertheless very valuable:

- they are obtained from three independent sources;
- they have an accuracy higher than 10% in general;
- they exactly cover the energy range of the reactor.

Applying these measurements, we have been able to increase accuracy (by about 30%).

Figure 1 gives the basic characteristics of the four Minerve and Masurca cores constituted for these measurements. Changes in enrichment are obtained by "discharge" [4].

The spectrum range covered by the set of lattices studied is shown as a function of the spectrum index r in Fig. 2 [4].

All these measurements, both French and foreign, were interpreted as consistently as possible, especially as regards error estimation.

A summary of the measurements indicating the experimental precision for each technique will be found in Table I, which includes the first 24 fission products to which the French programme makes a substantial contribution.

VI. EXPERIMENTS FOR UNSEPARATED FISSION PRODUCTS

These experimental results serve as reference for the whole CEA fission product programme. The total effect of the fission products was measured by oscillation of irradiated fast reactor fuels in the thermal-fast coupled assembly Ermine.

The fission product effect is obtained by measuring the difference in reactivity between irradiated and unirradiated fuels of the same initial content and by subtracting the effects due to the disappearance of the fuel during irradiation.

After oscillations, the fuels are analysed chemically in order to determine their compositions with accuracy.

The experimental programme carried out in France can be divided into two parts:

(1) The preliminary experiment in 1973, during which the Rapsodie and Fortissimo fuels (UO_2-PuO_2, 20-25% Pu, U with 60 and 80% ^{235}U) were oscillated in a metallic uranium lattice of the characteristic Super-Phênix spectrum (Lattice R3) [10].

The main purpose of this experiment was to develop the oscillation and chemical analysis techniques for highly irradiated fuels.

(2) The main experiment in 1975-76, when we oscillated a pure ^{235}U fuel (PROFI U) irradiated in Rapsodie and a Phênix fuel after three cycles of irradiation in order to measure the effect of the fission products of ^{235}U and plutonium separately.

These measurements were performed in three oxide-fuel lattices - one characteristic uranium lattice of Super-Phênix (lattice RONA 3) and two characteristic plutonium lattices of Super-Phênix (ZONA 3) and Phênix (ZONA 1).

All these measurements are summarized in Table II.

VII. RESULTS OBTAINED IN THE FISSION PRODUCT PROGRAMME

VII.1. Classification of the principal isotopes separated

Below we give the 20 main isotopes for which measurements were carried out, i.e. Group 1, in order of importance as a percentage of total pseudo-fission product capture:

^{105}Pd	9.9	^{149}Sm	5.9	^{133}Cs	3.0	^{104}Ru	2.1
^{101}Ru	8.9	^{151}Sm	5.9	^{109}Ag	2.6	^{95}Mo	1.3
^{103}Rh	7.3	^{147}Pm	4.1	^{102}Ru	2.4	^{100}Mo	1.3
^{99}Tc	7.3	^{97}Mo	3.4	^{153}Eu	2.2	^{141}Pr	1.3
^{107}Pd	6.4	^{145}Nd	3.1	^{143}Nd	2.1	^{98}Mo	1.0
						Total	81.3%

These isotopes are therefore placed at the top of the classification diagram given in Fig. 3. However, there is no measurement for three isotopes which, because of their importance, should have been included in Group 1:

^{135}Cs	(2.9%)
^{103}Ru */	(2.5%)
^{155}Eu	(1.1%)

These three are unstable radioactive isotopes which are difficult to obtain.

Of the 40 included in groups 2 and 3, 24 less important isotopes were also measured; they are: ^{93}Zr, ^{108}Pd, ^{106}Pd, ^{52}Sm, ^{146}Nd, ^{139}La, ^{148}Nd, ^{147}Sm, ^{150}Nd, ^{142}Ce, ^{96}Zr, ^{144}Nd, ^{91}Zr, ^{154}Sm, ^{111}Cd, ^{150}Sm, ^{157}Gd, ^{159}Tb, ^{156}Gd, ^{104}Pd, ^{92}Zr, ^{115}In, ^{128}Te, ^{110}Pd.

These measurements were obtained mostly from the STEK and CFRMF experiments. The contribution of all these fission products to the total effect lies between 1‰ and 1% and their partial sum amounts to 8.5% of the total.

Lastly, the non-measured fission products account for only 10% of the capture signal.

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*/ Impossible to measure directly (half-life 4 days)

VII.2. The capture cross-sections of the principal isotopes separated

The mean fission-product capture level in a fast spectrum is of the order of 0.2 b with a very large scatter because certain fission products (samarium and europium) have mean captures of 1-4 b. The capture cross-sections of the isotopes measured, accounting for 90% of the overall capture, were adjusted in the spectrum range of interest to fast power reactors.

The comparison between the very first non-adjusted file (Ribon + Cook) and the present adjusted file is given in Fig. 4. The changes are large - the cross-sections changed by a factor of 2 in one direction or the other, particularly in the case of palladium and ruthenium. Because of compensations, these large changes are absorbed in the pseudo-fission product mixture.

On the basis of the most recent re-evaluated file (CNEN-CEA), these deviations were systematically reduced. In particular, the new evaluations of the following isotopes show clearly better agreement with the integral measurements: ^{105}Pd, ^{101}Ru, ^{107}Pd, ^{149}Sm and ^{151}Sm.

This observation suggests that the three important isotopes ^{135}Cs, ^{103}Ru and ^{155}Eu, for which there are no measurements, are now more reliable than before.

Table 3 gives a list of the individual isotopes which have been adjusted together with their capture cross-section accuracy after adjustment and the contribution of these uncertainties to the uncertainty in the overall fission-product effect.

The sum of these contributions of the adjusted fission products represents 77% of the overall accuracy attained.

The accuracies of the results are obtained by taking into account the evaluation precision, the measurement precision, the uncertainties of self-shielding and of the flux depression signal, and the adjustment precision. The calculation of each of these terms takes into account a quadratic combination of the errors, partly statistical and partly systematic, whenever this distinction is possible. The contribution to the resulting uncertainty appears to be relatively large for the following isotopes: ^{105}Pd, ^{99}Tc, ^{107}Pd and ^{151}Sm.

Additional integral measurements for these isotopes will be valuable. The most accurate technique is irradiation, when it can be carried out. It

will be applied in 1978 (especially for ^{151}Sm and ^{107}Pd) in the Phênix Profil 2 experiment, which is in preparation. Table IV shows in detail the improvement achieved in the absolute value as well as accuracy by the Phênix Profil 1 irradiations.

VII.3. Creation of pseudo-fission products

On the basis of the adjusted file for the individual fission products, pseudo-fission products were created by weighting them with the cumulative yields which are calculated under typical operating conditions of a Super-Phênix-type power reactor.

The evaluation of the uncertainties of the pseudo product takes several factors into account, for the pseudo-fission products are sensitive not only to capture but also to fission yields, migrations, operating diagrams, reactor design and, lastly, evolution parameters. This fact was utilized in a study of the various causes of error [12], the conclusions of which are summarized below:

Uncertainties due to all cross-sections: combined results for 188 fission products of which 44 main ones are adjusted and the rest evaluated (CNEN-CEA + Cook)

$\pm 9\%$

Uncertainties due to yields: combination of uncertainties of direct and of cumulative yields for 60 products obtained directly from fission, and influence of these uncertainties on the effective yields for 188 fission products

$\pm 8\%$

Uncertainties due to fission-product migration: only gaseous fission products (Br, Kr, I and Xe) were considered in this study

$\pm 3\%$

Uncertainties associated with description of the state of the reactor: regarding the Super-Phênix project, this includes various uncertainties of reactor evolution and operation

$\pm 2\%$

The combination of all these more or less related error sources is evaluated at

$\pm 15\%$

The pseudo-product mixture corresponding to the total fissionable nuclei in the case of a 1200-MW(e) reactor in equilibrium (half-life), as obtained from the file of adjusted individual fission product data condensed into one energy group, is of the order of 500 mb/fission. The accuracy is of the order of ±75 mb/fission.

VII.4. The capture cross-section of pseudo-fission products from irradiated fuel measurements

Measurement of the total effect of fission products by comparison of unirradiated and irradiated fuels requires that not only highly irradiated fuels should be available in order to furnish a sufficient amount of fission products but also - considering the effect of fission products in relation to the effects of fissionable isotopes in this spectrum range - that:

- high-precision reactivity effect measurements should be carried out by oscillations;
- the effects of the principal fissionable isotopes which disappeared during irradiation should be measured with precision; and
- the compositions of unirradiated and irradiated fuels should be known by chemical analysis with an accuracy of the order of 0.1% with respect to the amounts of uranium and plutonium.

Using the experimental facility of the Minerve reactor [10], it was possible to attain the required accuracy for the reactivity effect measurements by the oscillation method.

After some technological difficulties with the first few irradiated samples, the chemical analyses gave the expected accuracies for the amounts of uranium and plutonium contained in the unirradiated and irradiated fuels.

The accuracies obtained for the total fission-product effect are ±9% from the highly irradiated Rapsodie fuels (burn-up of about 10%) and ±13% from the Phénix fuels whose burn-up did not exceed 3%.

The initial Phénix results, which are being analysed, indicate that the present set of individual fission products gives a correct accuracy for the overall pseudo-fission product effect, within the experimental uncertainties, but has a tendency to lower the pseudo-fission product capture for both ^{235}U and ^{239}Pu. Detailed analysis of these results is still in progress, and if

the observed tendency is confirmed it will be taken into account subsequently in the overall final readjustment of the pseudo product created from the individual fission products.

CONCLUSION

The activities on the problems of fission product evolution carried out within the general framework of the French nuclear programme since 1973 at the CEA are justified by the need to optimize the fuel cycle.

As regards Super-Phênix, the uncertainty associated with fission products is of the order of ±400 per hundred thousand at the present state of progress of our work. The activities will continue in four directions:

- Obtaining new evaluations: altogether 64 fission products evaluated, representing 98% of the total effect;
- Taking account of new fission yield data libraries, mainly for ^{240}Pu;
- Integral capture measurements for certain isotopes by irradiation in Phênix (^{107}Pd, ^{151}Sm, ^{153}Eu, ^{143}Nd and ^{93}Zr) or by activation in Masurca and Ermine (^{147}Pm, ^{135}Cs, ^{155}Eu and ^{152}Sm). Measurements of other isotopes by oscillations is also foreseen (^{109}Ag, ^{105}Pd, ^{101}Ru, ^{99}Tc and ^{103}Rh);
- In particular, oscillations of irradiated fuels: pure Pu samples from Rapsodie and from Phênix fuel after six cycles of irradiation.

These measurements are expected to improve the precision for the pseudo-fission products and to give a final uncertainty of the order of ±300 per hundred thousand for fission products.

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Table I. Measurements on separated fission products: number of experiments used and experimental accuracy (%)

NOMBRE D'EXPERIENCES UTILISEES	OSCILLATIONS				ACTIVATIONS				IRRADIATIONS		TOTAL DES MESURES
	STEK	FRO	ERMINE MASURCA		C F R M F		ERMINE		PHENIX		
	n	n	n	± ε	n	± ε	n	± ε	n	± ε	
Pd 105	5		3	10.					1	4.0	9
Ru 101	5	3		9.					1	3.3	9
Rh 103	5	3	6	8.	1	20.					15
Tc 99	5	3	6	10.	1	20.					15
Pd 107	5			15.							5
Sm 149	5	3		10.					1	3.3	9
Sm 151	5			50.					1	9.9	6
Pm 147	5	3	2	13.	1	25.				.	11
Mo 97	5	3		10.					1	4.4	9
Nd 145	5			13.					1	4.0	6
Cs 133	5		5	14.	1	20.			1	4.0	12
Ag 109	5		2	14.	1	20.					8
Ru 102	5	3	3	25.	1	30.	1	5.0			13
Eu 153	5		2	10.	1	20.					8
Nd 143	5			13.							5
Ru 104	5	3		17.	1	20.	2	3.3			11
Mo 95	5	3		14.					1	4.3	9
Mo 100	5			17.	1	16.	1	5.9			7
P 141	5		4	17.	1	28.	1	5.9			11
Zr 93	5			17.							5
Mo 98	5			17.	1	21.	2	3.8			8
Pd 108	5			17.			1	6.6			6
Pd 106	5			80.					1	9.0	6
Sm 147	5		4	10.							9

Table II. Measurements of the overall effect of fission products in irradiated fuels

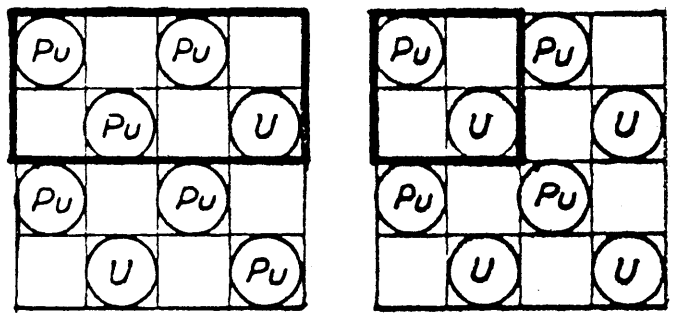
	Experimental lattices	Type of irradiated fuels measured	Number of oscillated samples	Mass of fission products per sample
Preliminary experiment	R 3	Rapsodie	1	20 g
		Rapsodie-Aurore	1	18 g
		Rapsodie-Fortissimo	1	10 g
Main experiment	RONA 3	Rapsodie-Profi U	2	10 g
		Phenix 3 cycles	2	4,5 g
	ZONA 3	Rapsodie-Profi U	1	10 g
		Phenix 3 cycles	1	4,5 g
	ZONA 1	Rapsodie-Profi U	2	10 g
		Phenix 3 cycles	1	4,5 g

Table III. Contribution of adjusted fission products to the overall accuracy

Isotope	Precision 2σ (%)	Part erreur globale (%)	Isotope	Precision 2σ (%)	Part erreur globale (%)
Pd105	± 9.2	6.6	Zr 93	± 18.0	1.0
Ru101	8.8	5.6	Pd108	10.5	0.8
Rh103	8.3	4.4	Pd106	17.5	1.2
Tc 99	9.8	6.5	Sm152	18.3	1.0
Pd107	14.6	6.4	Nd146	15.0	0.5
Sm149	8.9	3.8	La139	23.0	0.7
Sm151	19.0	7.4	Nd148	13.0	0.4
Pm147	12.3	3.4	Sm147	11.8	0.4
Mo 97	9.4	2.4	Nd150	29.0	0.8
Nd145	10.8	2.3	Ce142	23.0	0.6
Cs133	10.1	3.1	Zr 96	20.0	0.2
Ag109	13.3	2.5	Nd144	16.5	0.2
Ru102	12.0	2.1	Zr 91	13.0	0.2
Eu153	9.1	1.5	Sm154	20.0	0.2
Nd143	12.3	1.9	Cd111	22.0	0.8
Ru104	11.3	1.7	Sm150	12.0	0.2
Mo 95	10.0	1.1	Gd157	20.0	0.4
Mo100	12.2	1.1	Tb159	8.0	0.1
Pr141	13.1	1.2	Gd156	16.0	0.3
Mo 98	12.0	0.8	Pd104	16.0	0.1
			Zr 92	22.0	0.5
			In115	39.0	0.2
			Te128	39.0	0.2
			Pd110	21.0	0.2
	TOTAL	65.8%		TOTAL	11.2%

Table IV. Use of Phenix Profil 1 irradiations

Isotope	Effect on adjusted σc, %	Accuracy (%)	
		without Phenix	with Phenix
Pd105	+ 1,0 ± 4,0	10	9
Ru101	- 4,4 ± 3,3	10	9
Sm149	- 2,7 ± 3,3	10	9
Sm151	- 9,3 ± 9,9	50	19
Mo 97	+ 11,1 ± 4,4	12	9
Nd145	- 4,2 ± 4,0	13	11
Cs133	+ 6,3 ± 4,0	14	10
Mo 95	+ 14,3 ± 4,3	15	11
Pd106	+ 22,7 ± 9,0	80	17



ZONA 1 (E ∿ 18%) ZONA 3 (E ∿ 12%)

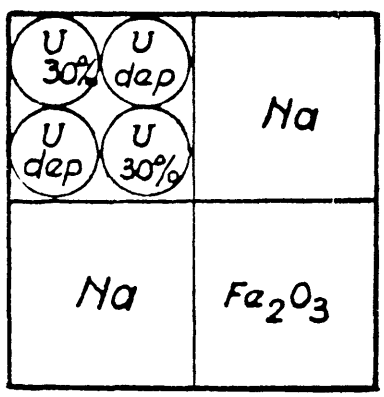
Pu PuO_2/UO_2

U UO_2 Naturel

Sodium

Cellule de base

R3



RONA 3

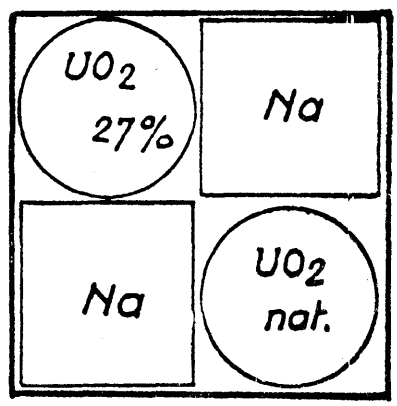


Fig. 1

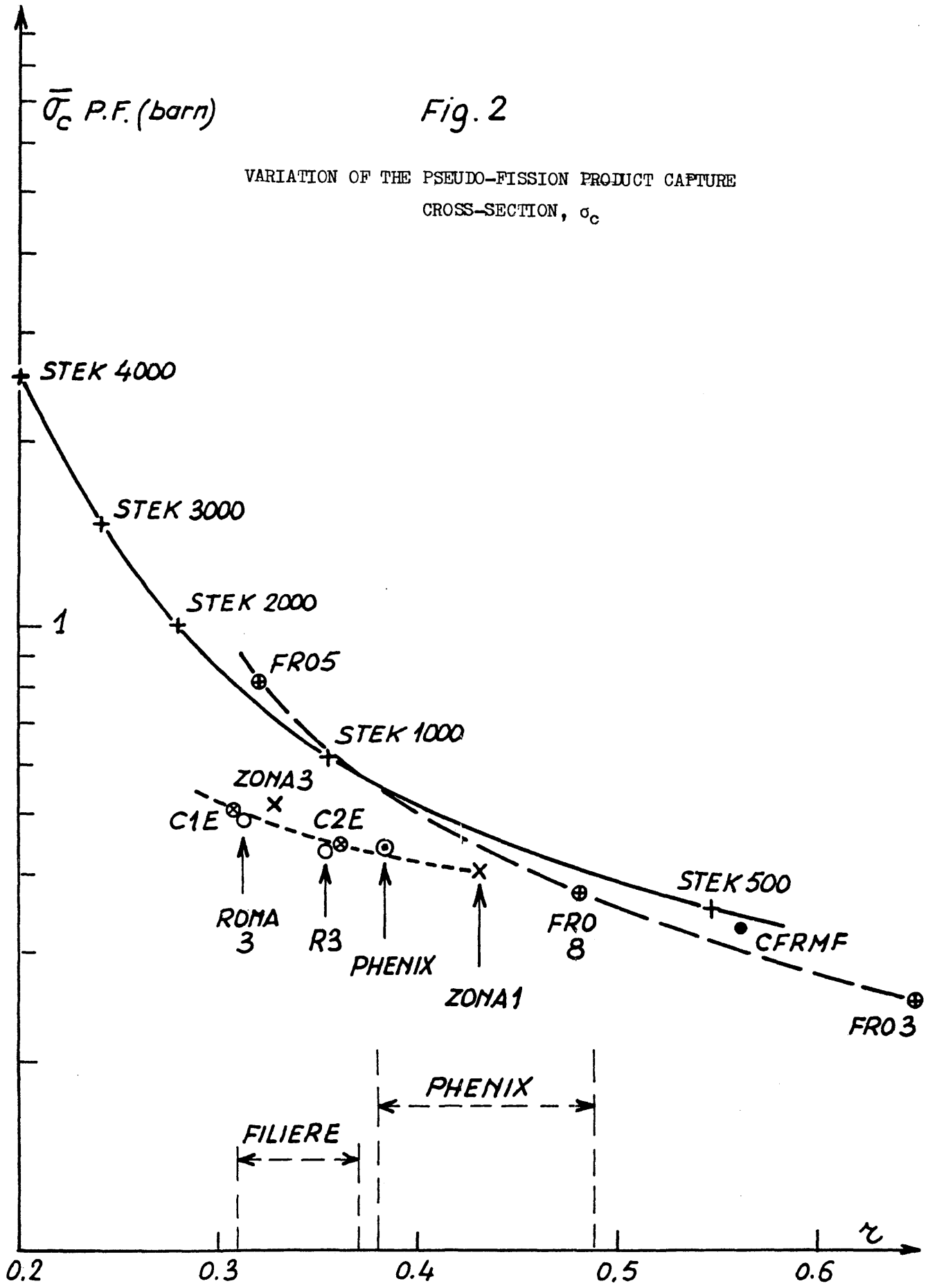


Fig. 2

VARIATION OF THE PSEUDO-FISSION PRODUCT CAPTURE CROSS-SECTION, σ_c

CLASSIFICATION OF THE 43 MOST IMPORTANT FISSION PRODUCTS

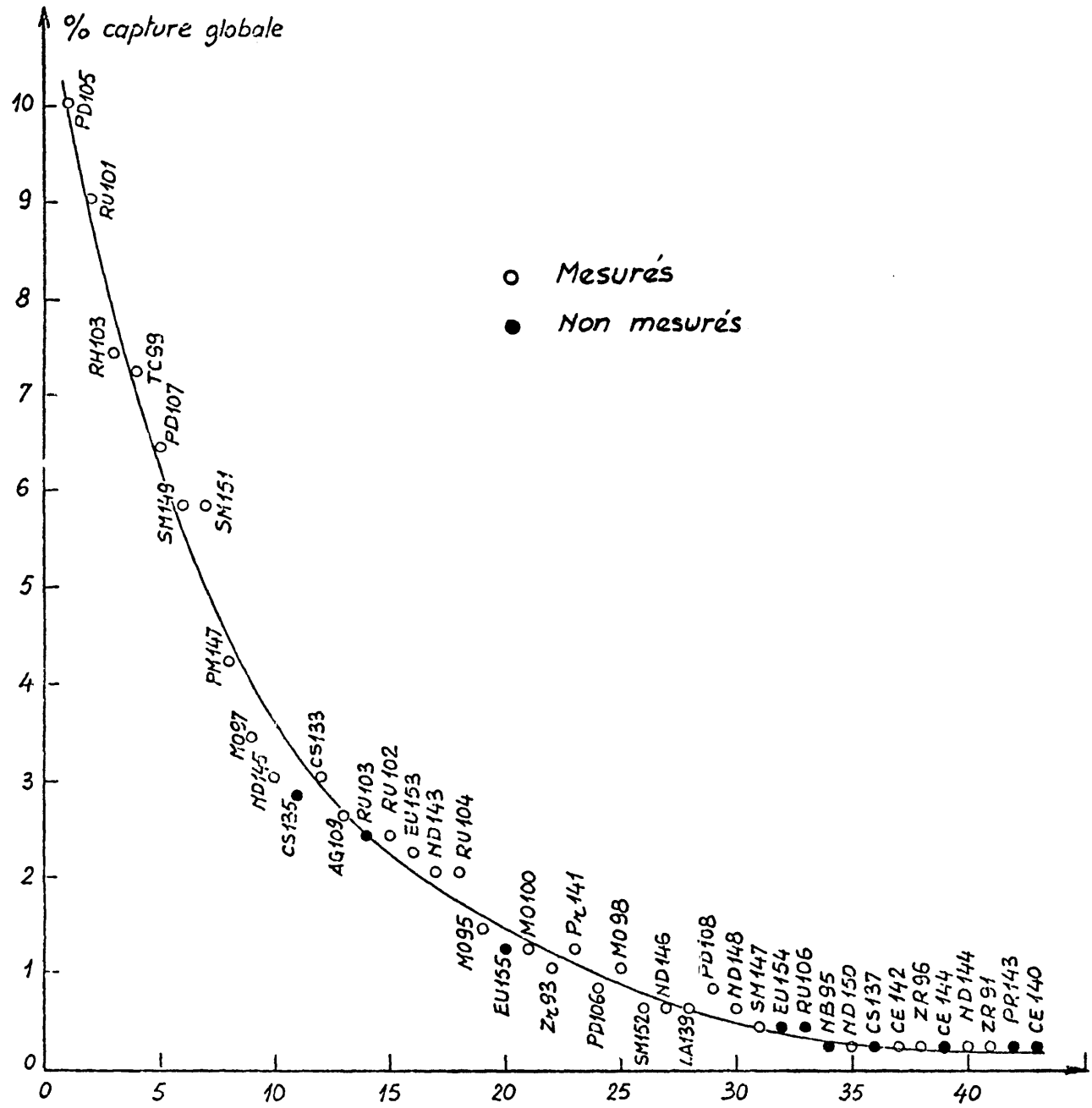


Fig. 3

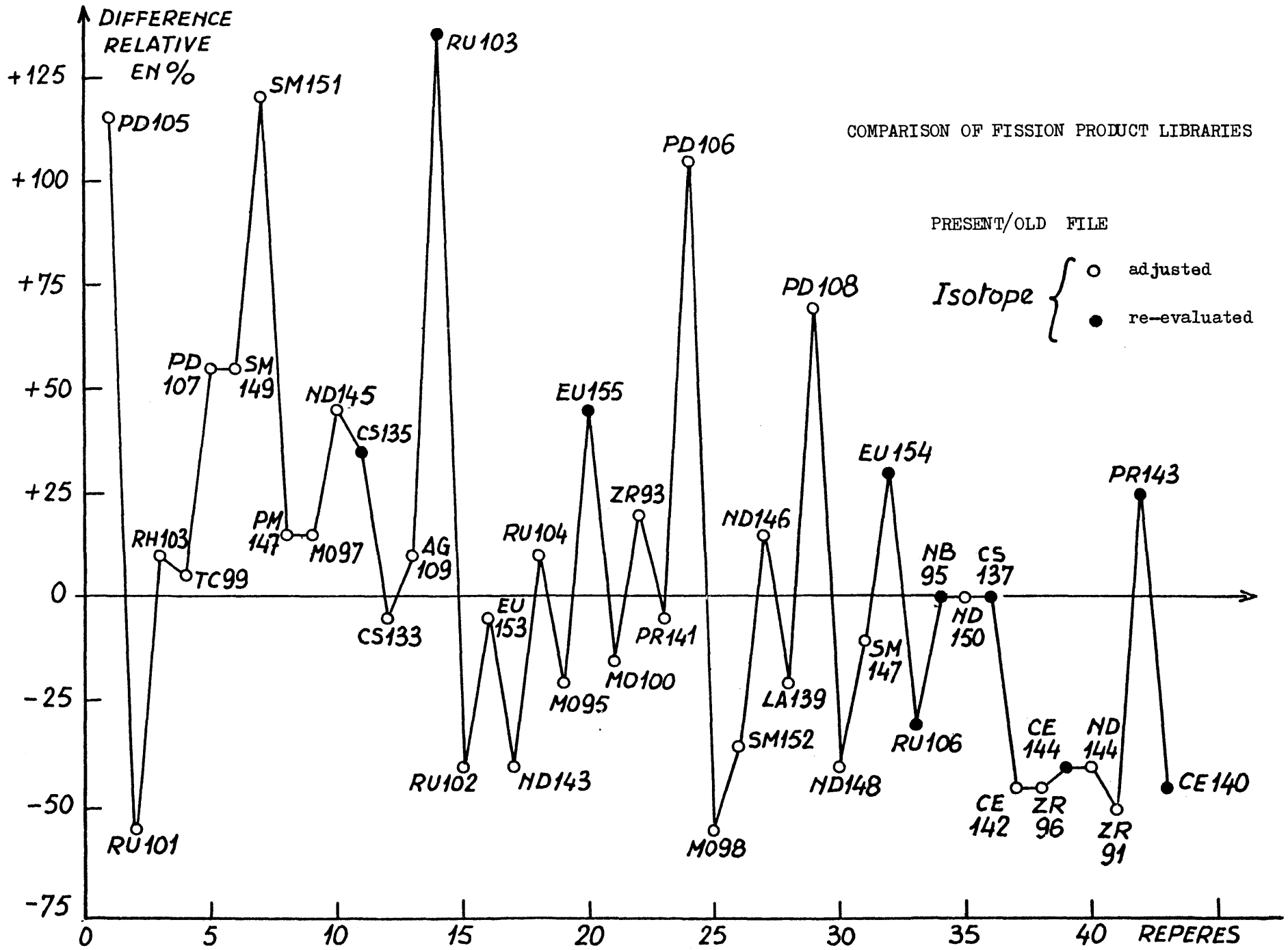

COMPARISON OF FISSION PRODUCT LIBRARIES
PRESENT/OLD FILE
Isotope
adjusted
re-evaluated
DIFFERENCE RELATIVE EN %
+125
+100
+75
+50
+25
0
-25
-50
-75
0
5
10
15
20
25
30
35
40
REPERES
PD105
RU101
SM151
PD 107
SM 149
RH103
TC99
ND145
CS135
PM 147
M097
CS133
RU103
AG 109
RU102
EU 153
ND143
RU104
EU155
M095
ZR93
M0100
PR141
PD106
M098
SM152
ND146
LA139
PD108
ND148
EU154
SM 147
RU106
NB 95
CS 137
ND 150
CE 142
CE 144
ZR 96
ND 144
ZR 91
PR143
CE140

Fig. 4

Contribution to Review Paper 15

FISSION-PRODUCT BETA AND GAMMA DECAY-ENERGY RELEASE RATES FOR THE TIME PERIOD 2 - 14000 SEC FOLLOWING THERMAL-NEUTRON FISSION OF ^{235}U*

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1. INTRODUCTION

The decay heat energy-release rate from thermal-neutron fission of ^{235}U can be obtained by two complementary methods. The first is based on measurements of the total beta and gamma power released; the second consists of cumulating by calculation the individual energies released by each one of the fission products. The second method, a large calculational problem, has the merit that once it is proven to reproduce measurements satisfactorily, it can be applied to more complex situations in a reactor taking into account for example reactor power variations or the creation of other fissionable isotopes as a function of reactor operation.

Since 1974 four new measurements of fission-product decay power have been performed in the United States: (1) a total-absorption nuclear radiation measurement at Intelcom Radiation Technology, San Diego,[1] (2) a calorimetric measurement at Los Alamos Scientific Laboratory,[2] (3) a second calorimetric measurement using different techniques at the University of California, Berkeley,[3] and (4) the present measurements at ORNL, consisting of separate differential energy measurements of beta and gamma rays. The first three experiments mentioned provide integral energy-release data; the present experiment provides, in addition, spectral distributions which can be used for computing e.g. high-energy gamma-ray leakage. Furthermore, the spectral distributions provide a more stringent test of the validity of the "microscopic" calculational approach.

2. EXPERIMENTAL METHOD

The total beta and gamma energy release rate from fission product decay following thermal neutron fission of the fuel element ^{235}U was obtained by measuring separately the energy release rates for each component. That is, one set of data was obtained for gamma energy release using a large NaI gamma-ray detector, and another set was obtained for beta energy release using a plastic scintillation beta-ray detector. The Oak Ridge Research Reactor (ORR) was used to provide the thermal neutrons. Small samples of ^{235}U were irradiated for short periods of time and were rapidly recovered following irradiation.

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* Research performed at Oak Ridge National Laboratory, operated by Union Carbide Corporation, under the sponsorship of the U.S. Nuclear Regulatory Commission.

3. DATA REDUCTION

The first step in data reduction is the summing of equivalent pulse-height data and the subtraction of background (in the case of gamma-ray data) or else the subtraction of "magnet-up" data (in the case of beta-ray data). The next step is to bin the pulse-height data into predetermined gamma- (or beta-) ray energy bins. These data are unfolded using FERD.[4] The unfolded spectra are divided by n_f to yield normalized spectra. An example of a gamma-ray spectrum is shown in Fig. 1, and an example of a beta-ray spectrum is shown in Fig. 2. Also shown on these figures are spectral data previously reported in the literature,[5-7] and also calculated spectra. The calculated gamma spectrum used the Oak Ridge code ORIGEN;[8] the calculated beta spectrum was provided by England and Stamatelatos.[9]

The next step is to integrate the differential energy spectra, i.e. $N(E_\beta)$ vs E_β and $N(E_\gamma)$ vs E_γ,

$$N_{tot} = \int_{E_{min}}^{8\ MeV} N(E)dE \tag{2}$$

$$E_{tot} = \int_{E_{min}}^{8\ MeV} E\ N(E)dE \tag{3}$$

for beta- and gamma-ray data separately, where (experimental) E_{min} = 0.05 MeV for gamma rays and 0.16 MeV for beta rays. Several corrections were applied to the integral data N_{tot} and E_{tot} which were not applied to the differential data. These included estimates of energy emitted in gamma rays having E_γ < 0.05 MeV and beta rays having E_β < 0.16 MeV.

4. THE INTEGRAL DATA

The present energy-integral data may be used to obtain the function f(t) which is defined as the rate of energy release t seconds following an instantaneous pulse of fission, and has units of (MeV/sec)/fission. This function is easy to estimate from our integral energy data if the integrals are divided by the counting time, and the time t is taken to be

$$t = t_{wait} + 0.5\ (t_{irrad} + t_{count}). \tag{4}$$

Data were obtained by detecting and measuring individual events for several irradiation times (t_{irrad}), waiting times following the end of the irradiation (t_{wait}), counting times starting at the end of the waiting time (t_{count}), and particle energies (E_β or E_γ). The resulting energy spectra were integrated over particle energy to obtain the energy release for each component for each utilized combination of t_{irrad}, t_{wait}, and t_{count}.

The experimental method calls for short irradiations (compared to decay time prior to beginning the observation), nominally 1, 10, and 100 sec; small samples, $\sim$ 1, 5, and 10 μgm of U_3O_8 enriched to 93.5% in the isotope ^{235}U; and spectral measurements for specified, and short, counting intervals. Except for the shortest waiting time intervals, the experiment closely approximates differential energy release rates for time t after a pulse of fissions.

For the gamma-ray data, background was measured by running an empty sample carrier. Some of the background was time dependent, corresponding to activity (e.g. ^{24}Na) picked up by the sample carrier in the movement of the sample to and from the reactor. Some of the background was time independent, consisting primarily of ^{41}Ar decay plus a low-energy contribution.

For the beta-ray detector, two samples had to be run, one with no magnetic deflection, measuring $(\beta + \gamma)$. The second sample was measured with magnetic deflection, yielding (γ) only. Background from activity on the sample carrier was nearly eliminated by tight collimation. From these data the data handling includes subtraction of "magnet-up" data from "magnet-down" data. This operation is equivalent to

$$(\beta) = (\beta + \gamma) - (\gamma) \quad . \tag{1}$$

Following the irradiation and measurements the samples were "cooled" for varying periods and then counted for characteristic gamma rays associated with decay of ^{97}Zr, ^{99}Mo, and ^{132}Te. These measurements were performed in a separate low-background counting area using Ge(Li) detectors. These data were used to determine the number of fissions, n_f, for each sample.

For a given irradiation time and sample size, a single irradiation and measurement does not provide enough data for the statistical accuracy desired. Hence, many identical irradiations and measurements were required. During the course of this experiment several hundred irradiations were required. The software was designed to provide an efficient and reproducible method of controlling and monitoring each irradiation and subsequent beta- or gamma-ray counting, and also to be simple to use.

The difference between our estimate and an "exact" value is small if t_{wait} is > ($t_{irrad} + t_{count}$), becoming larger for short waiting times compared to irradiation and count times, but the correction is < 5% for our shortest values of t.

The function f(t) decreases approximately as t^{-1}. Hence, for presentation purposes it has become common practice to illustrate the "pulse" function as t f(t). This presentation has the advantage of expressing the t f(t) axis on linear graphics, but the disadvantage of increasing the error in presentation if there is any error in t. Acknowledging this defect, our beta-ray data are presented in Fig. 3 in this format for comparison with summation calculation[10] and with other data[7,11,12] sets, including those for which the earlier data were obtained only in an integral format. The present data agree very well with the earlier data for t between 15 and 500 sec, but not so well with the calculation in this time interval. The present data are 5 to 7% larger than calculation for t between 1000 and 5000 sec, and in reasonable agreement for longer times.

Our gamma-ray data are presented in similar format in Fig. 4. Our data are smaller than the other experimental data,[5,6,13] although within assigned uncertainties of previous ORNL measurements.[5] The comparison of calculation[10] with the present data is good for t between 20 and 1000 sec and for t > 5000 sec. There is a disagreement for t < 10 sec which may be indicative of incomplete information in the basic data files. The disagreement for t between 1000 and 5000 sec is rather unexpected, and suggests that further study is needed in this time interval. These figures (3 and 4) emphasize the differences between our data and calculated values for a pulse of fissions. The differences are much less obvious for extended irradiations.

5. THE DIFFERENTIAL DATA

Although of modest energy resolution the differential spectra should prove to be useful for evaluation of individual nuclide contributions to the total decay power calculations. The calculation shown in Fig. 1 utilized contributions only from the 180 nuclides in the ENDF/B-IV data file[14] having tabulated gamma ray energies and yields, and did not include any contribution from the > 500 nuclides in the file having only an average gamma energy in the file. However, integral f(t) curves shown in Figs. 3 and 4 and the beta-ray calculation in Fig. 2 include contributions from all nuclides in the file. The calculation of a gamma-ray spectrum using ORIGEN may include contributions from all isotopes in the file; an example is shown in Fig. 5. The most obvious difference between calculation and experiment is for $E_\gamma \sim 1.14$ MeV, and has been traced to the contribution to the calculation due to 14-min ^{143}La. This nuclide is not among the 180 nuclides in the ENDF/B-IV data file

having tabulated gamma ray energies and yields. There is little solid experimental information on the decay of ^{143}La (see Ref. 15); such information as exists suggests that ^{143}La contributes less to the gamma-ray decay power and more to the beta-ray decay power than currently given in the ENDF/B-IV data file. These differences are small, 1 or 2% of the total, but tend to reduce the discrepancies shown in Figs. 3 and 4. The spectra obtained by the present program should make it easier to ascertain which data in the files may warrant further study.

ACKNOWLEDGMENTS

We express our appreciation for the careful analytical work of J. F. Emery and K. J. Northcutt and the contributions to the experimental measurements of R. M. Freestone and H. Weaver.

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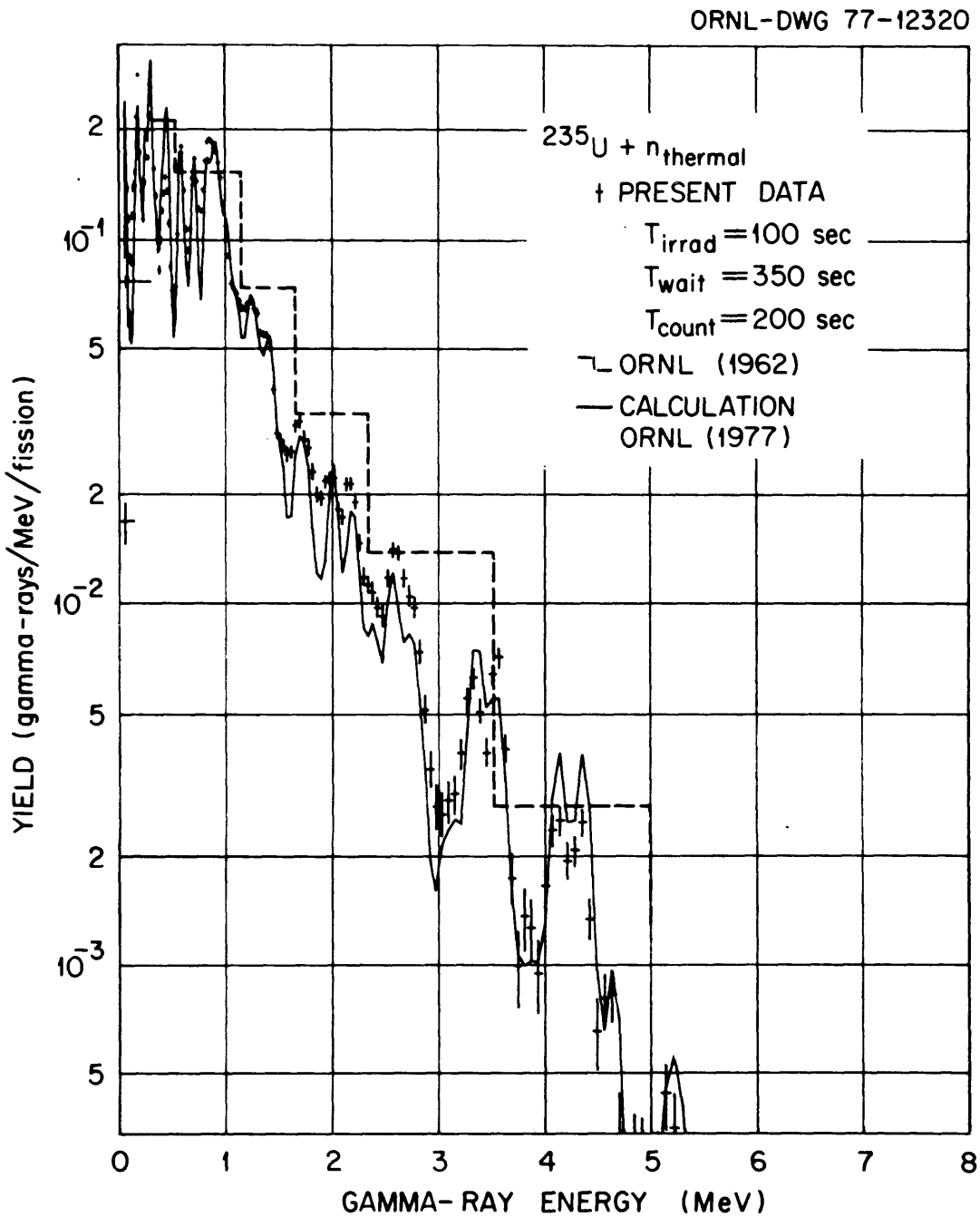

Fig. 1. Spectrum of gamma rays due to thermal-neutron fission of ^{235}U. The histogram represents earlier ORNL data (Ref. 5). The solid line is a calculated spectrum using ORIGEN (Ref. 8).

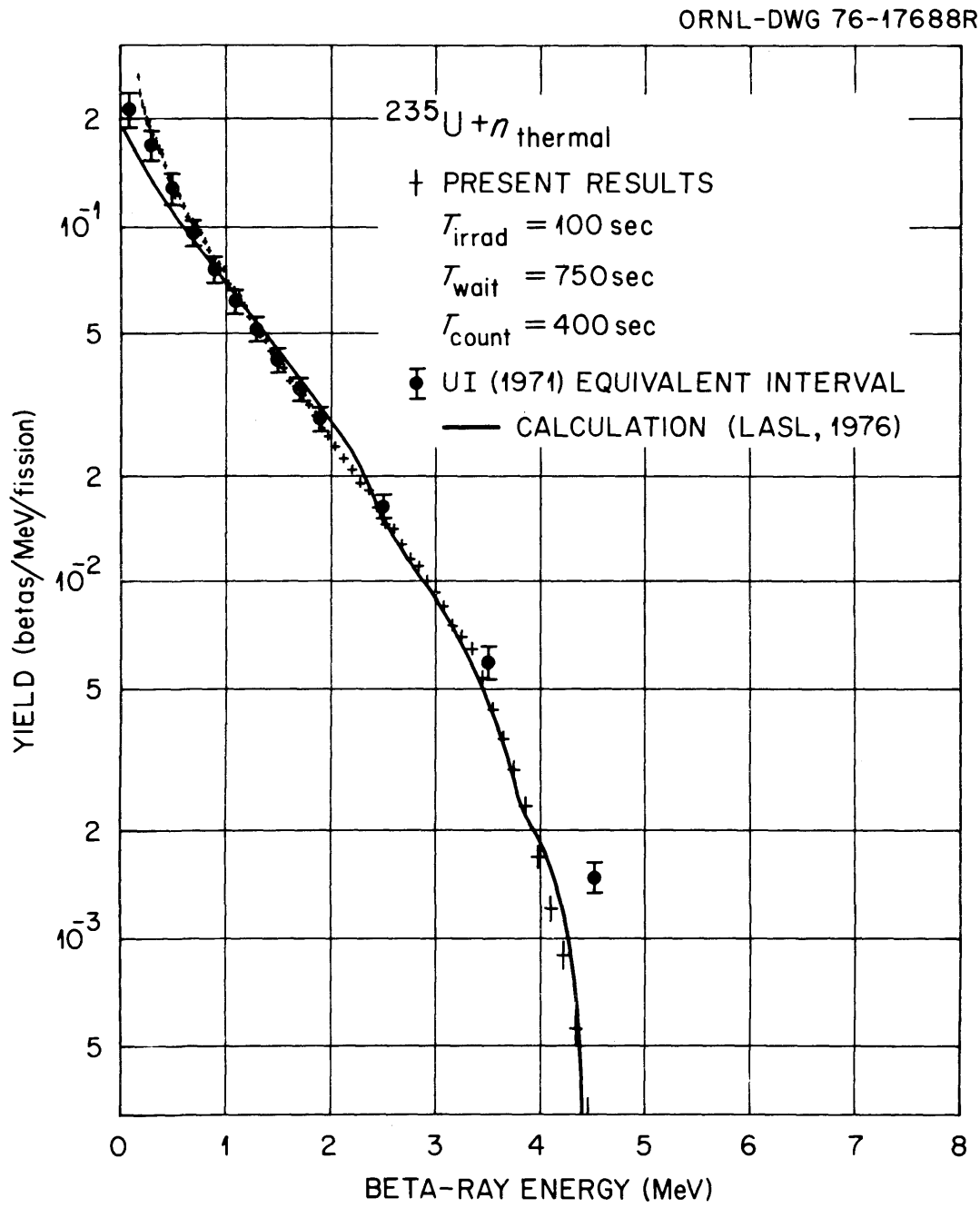

Fig. 2. Spectrum of beta rays due to thermal-neutron fission of ^{235}U. The solid points are the data of Tsoulfanidis et al. (Ref. 7) and the calculations are the work of England and Stamatelatos (Ref. 9). The irradiation time, waiting time, and counting time intervals are given in the legend.

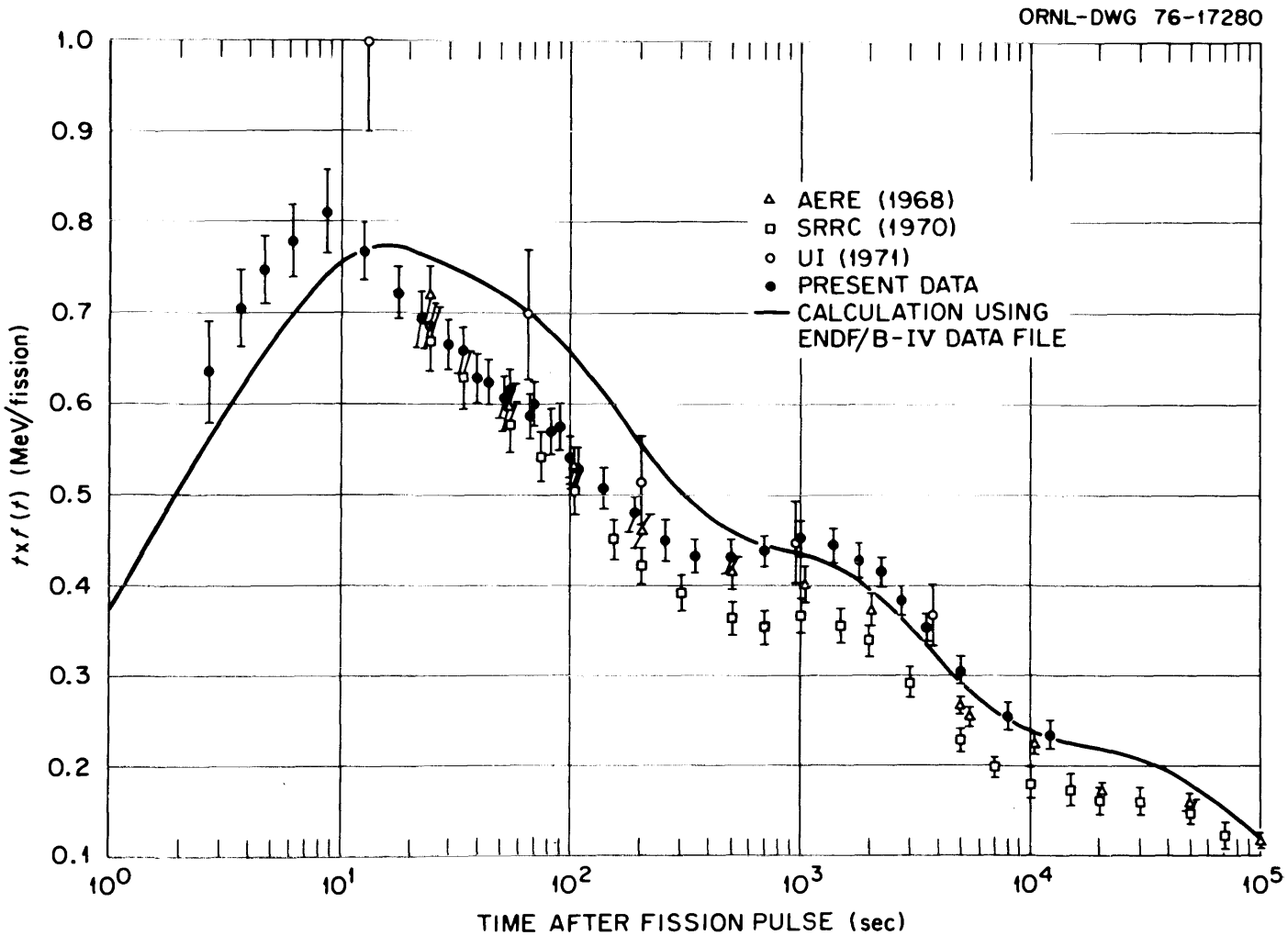

Fig. 3. Beta energy emission rate following an instantaneous pulse of thermal-neutron fissions of ^{235}U. The abscissa, t, is the time after a pulse of fissions. The ordinate is a quantity derived by obtaining f(t) and then multiplying it by t. The units are a contraction of $\frac{\text{MeV/sec}}{\text{fission}}$ x sec. The solid circles represent the present data as described in the text. The open triangles are data of McNair et al. (Ref. 11), the open squares of MacMahon et al., (Ref. 12), and the open cricles of Tsoulfanidis et al., (Ref. 7). The calculation was carried out by R. Schenter (Ref. 10) and the ENDF/B-IV data file (Ref. 14).

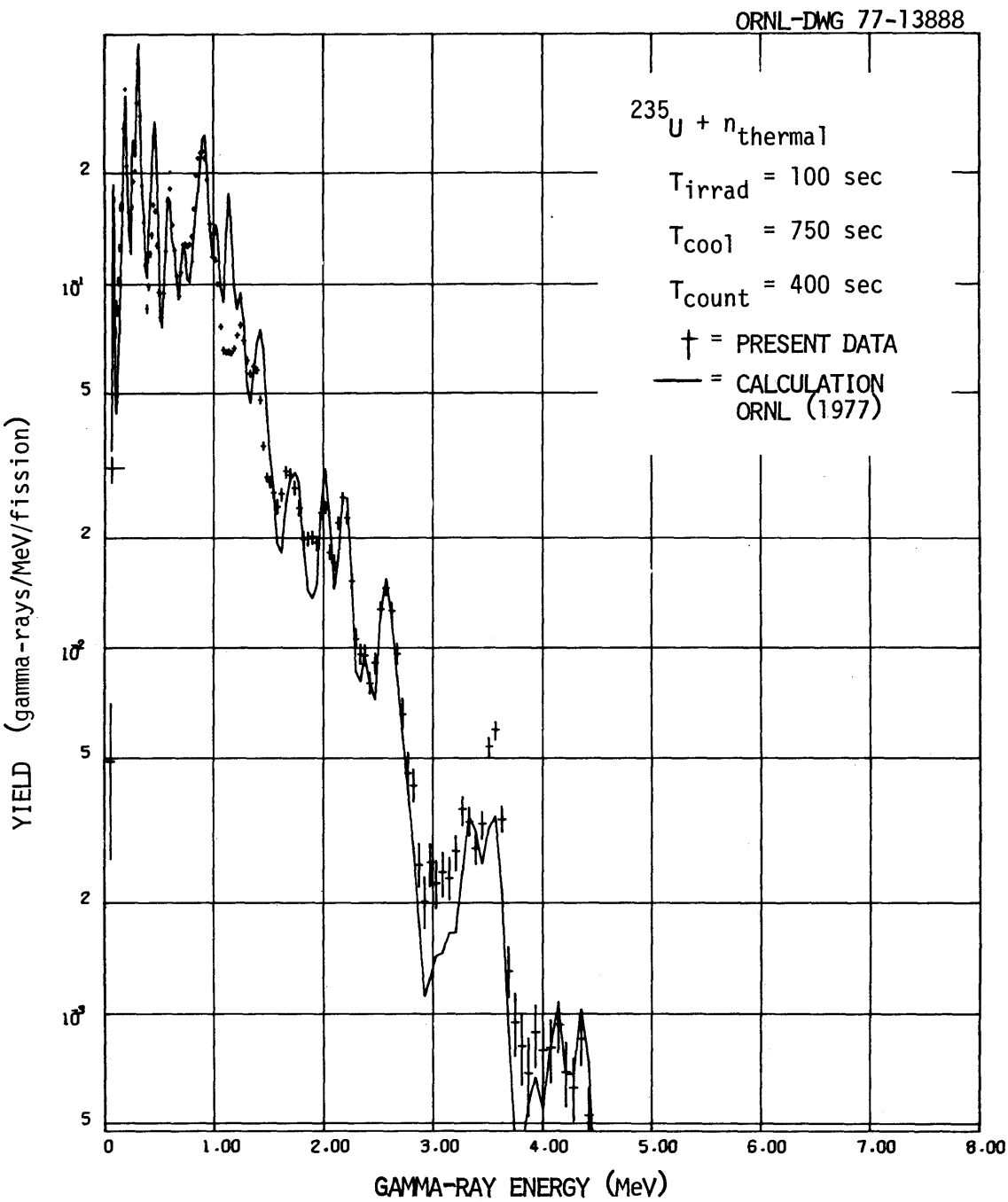

Fig. 5. Spectral distribution of gamma rays following thermal-neutron irradiation of ^{235}U. Irradiation, waiting, and counting intervals are given in the legend. The calculation was performed using ORIGEN (Ref. 8) and the data in the ENDF/B-IV file (Ref. 14). Almost all of the calculated data shown are due to isotopes having tabulated gamma ray energies and yields. The notable exception is the contribution due to ^{143}La, which is calculated as if the average energy, given in the ENDF/B-IV file as $\bar{E}_\gamma$ = 1.14 MeV, were due to a monoenergetic gamma-ray transition having a 100% branching ratio.

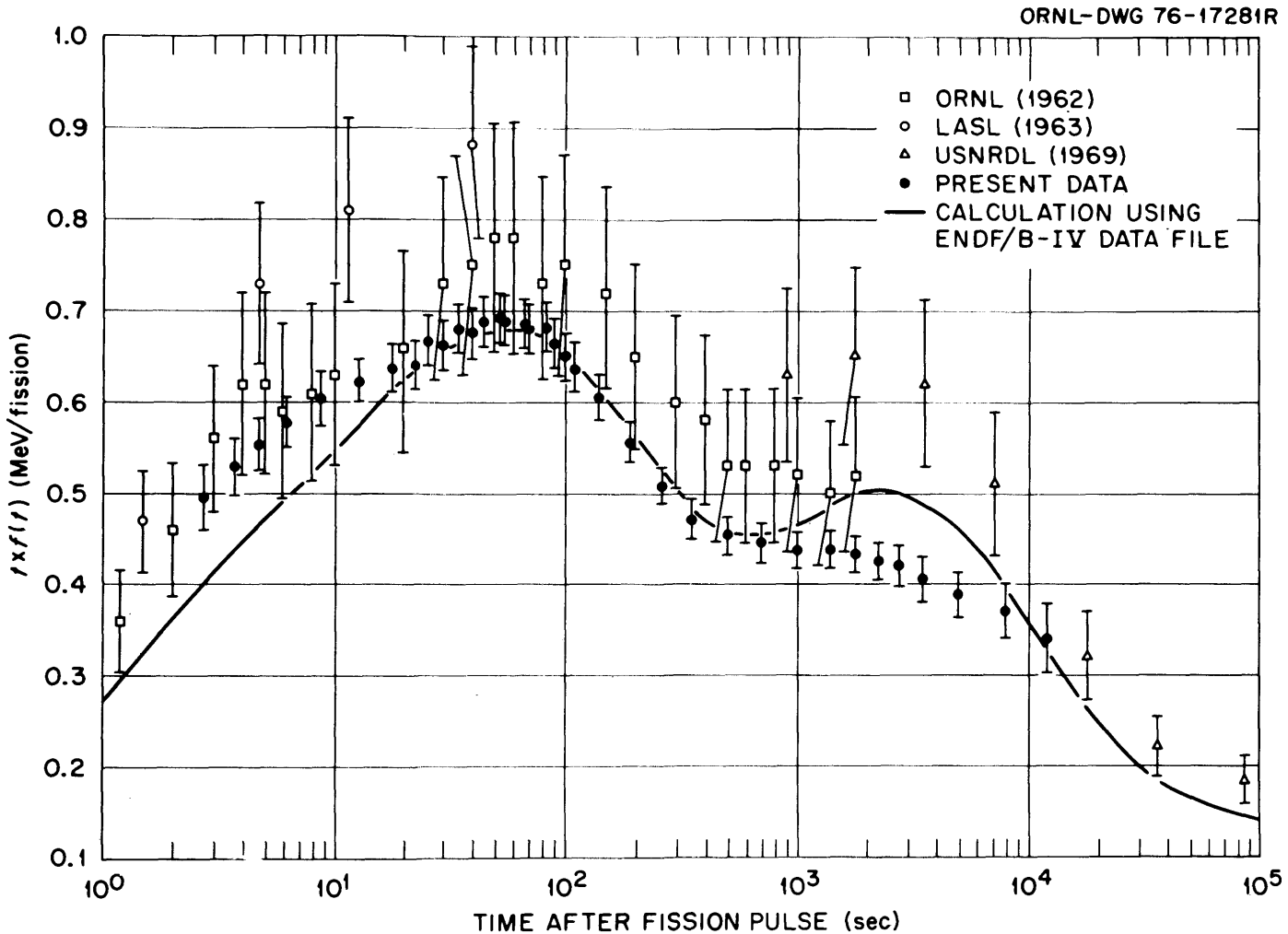

Fig. 4. Photon energy emission rate for thermal-neutron fission of ^{235}U. The abscissa, t, is the time after a pulse of fissions. The ordinate is a quantity derived by obtaining f(t) and then multiplying it by t. The units are a contraction of $\frac{\text{MeV/sec}}{\text{fission}}$ x sec. The solid circles represent the present data as described in the text. The open squares are the data of Peelle et al., (Ref. 5), open circles represent data of Fisher and Engle, (Ref. 6), and the open triangles are data of Bunney and Sam (Ref. 13). The calculation was carried out by R. Schenter (Ref. 10) and the ENDF/B-IV data file (Ref. 14).

Contribution to Review Paper 15

DECAY HEAT MEASUREMENTS BY FAST-RESPONSE BOIL-OFF CALORIMETRY

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ABSTRACT

A liquid-helium boil-off calorimeter with a 1-s time constant was used to measure ^{235}U fission product decay heat at times between 10 and 10^5 s following a 2 x 10^4 s thermal neutron irradiation. The uncertainty in the data was $\sim$ 2% (1 σ) except at the shortest cooling times, where it rose to $\sim$ 4%. Similar measurements on ^{239}Pu and ^{233}U in progress.

We report calorimetric measurements of fission product decay heat, with emphasis on short cooling time. This work is part of a program sponsored by the U. S. Nuclear Regulatory Commission to provide better values and reduced uncertainties for the decay heat source term, for use in reactor safety evaluations, and, in particular, for the analysis of the Loss-of-Coolant Accident (LOCA).

Calorimetric measurements have the advantage that they are relatively simple and straightforward, and thus present a minimum opportunity for the introduction of systematic errors. Their disadvantages are that they provide only integral information (total decay heat, not β and γ spectra), and that, in general, they tend to have long time constants.

Our measurements were carried out with a cryogenic boil-off calorimeter. In this technique, decay heat is used to evaporate a cryogen, in our case liquid helium, and the rate at which boil-off gas is evolved is measured with a fast-response flowmeter. Since the heat of vaporization of helium is known, the amount of decay heat may be calculated from the boil-off rate. Direct calibration by means of electrical heating provides a check on these calculations.

Although the idea of boil-off calorimetry is an old one, we chose it because it offered an opportunity to develop a calorimeter with a short thermal time constant. By operating nearly isothermally, and by taking

advantage of the large reduction in the heat capacity of solids which occurs at the temperature of liquid helium (4 K), we were able to achieve a time constant of less than one second in a calorimeter which contained a 52-kg radiation absorber.

Samples of ^{235}U, ^{239}Pu, and ^{233}U, weighing $\sim$ 60 mg and encased in cladding to retain gaseous fission products, were irradiated for 2×10^4 s in a constant thermal-neutron flux of 3×10^{13} n/cm^2 s. They were then transferred to the calorimeter in $\sim$ 1 s, and had reached the temperature of liquid helium after an addition 3 s. Decay heat data were obtained for cooling times between 10 and 10^5 s. After the calorimetric measurements were completed, the number of fissions in the samples was determined radiochemically.

Corrections to the data were made for the leakage of gamma radiation from the calorimeter (3% maximum) and for the initial transient (3.4% maximum). The overall uncertainty (1 σ) in the experimental data is estimated to be $\sim$ 2% except at the shortest cooling times, where it rises to $\sim$ 4% The measurements on ^{235}U have been completed, and those on ^{239}Pu and ^{233}U are in progress.

The experimental data for a 2×10^4 s thermal neutron irradiation of ^{235}U agree within the errors with summation calculations based on the ENDF/B-IV data base. The experimental data were extended to the case of an infinite irradiation by adding the calculated decay heat due to those fissions which took place before the experimental irradiation interval of 2×10^4 s. The extended data for infinite irradiation are in good agreement with the summation calculation.

Both the extended experimental data and the summation calculations are significantly ($\sim$ 7%) below the current ANS Decay Heat Standard at short cooling times. This standard, plus 20%, is currently used in reactor safety evaluations.

A preliminary report of our decay heat measurements was presented at the Fourth Water Reactor Safety Information Meeting in September, 1976. A complete description of these measurements is given in The Los Alamos Scientific Laboratory Report LA-NUREG-6713, July 1977 (in publication).

The details of the experiment and its results are indicated in the following Figures and in the Table.

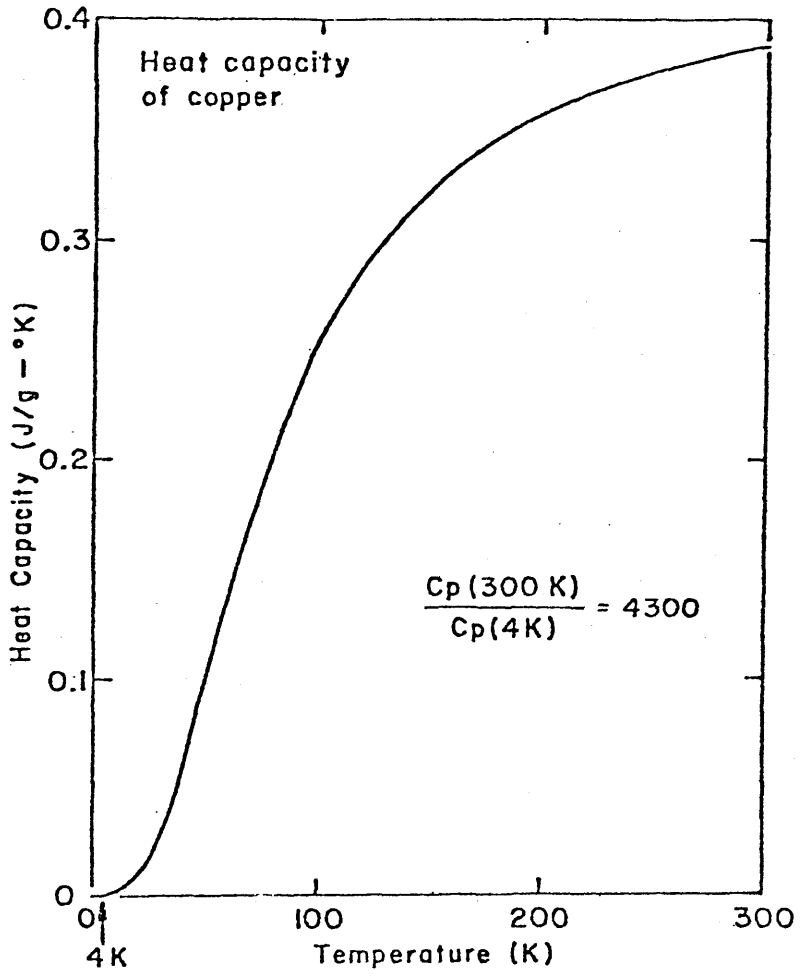

Fig. 1. Heat capacity of copper as a function of the absolute temperature. By operating the calorimeter at 4 K, the heat capacity of the 52 kg copper absorber was minimized. The temperature was stabilized by a liquid helium bath boiling at nearly constant pressure in a reservoir within the copper absorber. The absorbed decay heat evaporated liquid helium from the reservoir and a hot-film anemometer-type mass flowmeter measured the evolution of the boil-off gas. The flowmeter had a time constant of $\sim$ 1 ms. The pressure drop across the flowmeter at maximum flow ($\sim$ 1 liter/s) was $\sim$ 1 torr. This pressure change corresponds to a temperature change of $\sim$ 1.7 mK in the helium reservoir. Under these conditions, there is a minimum storage of thermal energy in the calorimeter, and a fast response time is possible. The high thermal conductivity of the very pure, well-annealed copper used for the absorber also contributed to the short time constant.

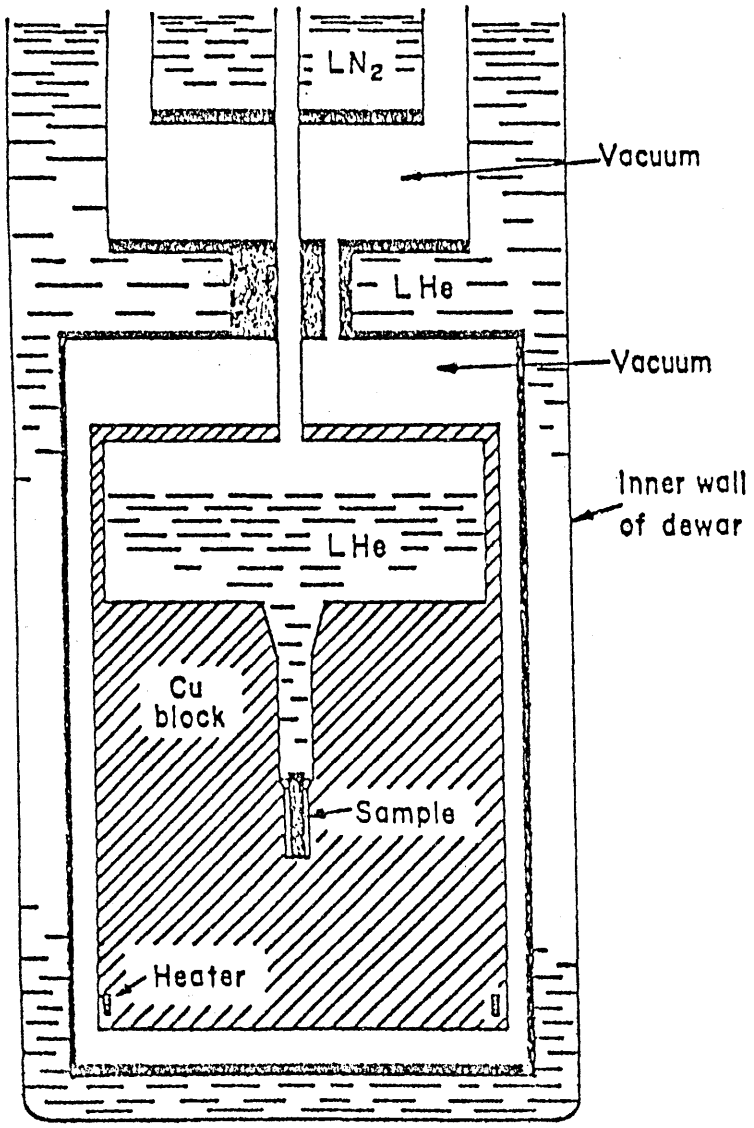

Fig. 2. Essential features of the calorimeter. The copper block, which absorbed > 97% of the radiation energy emitted by the sample, was 177.8 mm in diameter, 298.5 mm high, and weighed 52.008 kg. The reservoir in the top of the block was 3/4 filled with liquid helium. The block was suspended in vacuum by a 10-mm-i.d. thin-wall stainless steel tube which conducted the boil-off gas to a flowmeter at room temperature. The tube was also used to insert and remove samples and to transfer the liquid helium to the reservoir. A heater in the base of the block was used for calibration and testing. To prevent heat leak, a copper vacuum jacket, immersed in an outer liquid helium bath, surrounded the calorimeter block. A commercial liquid-nitrogen-jacketed dewar contained the entire assembly. All electrical leads reaching the calorimeter block passed through the outer helium bath, which intercepted heat conducted along them from warmer regions. The outer helium bath was kept 25mK above the temperature of the helium in the inner reservoir to prevent the boil-off gas from condensing on the tube walls after it left the reservoir. The boil-off gas was raised to room temperature in a controlled manner to minimize the effect of changes of temperature gradients along the gas-transport tube with boil-off gas flow rate.

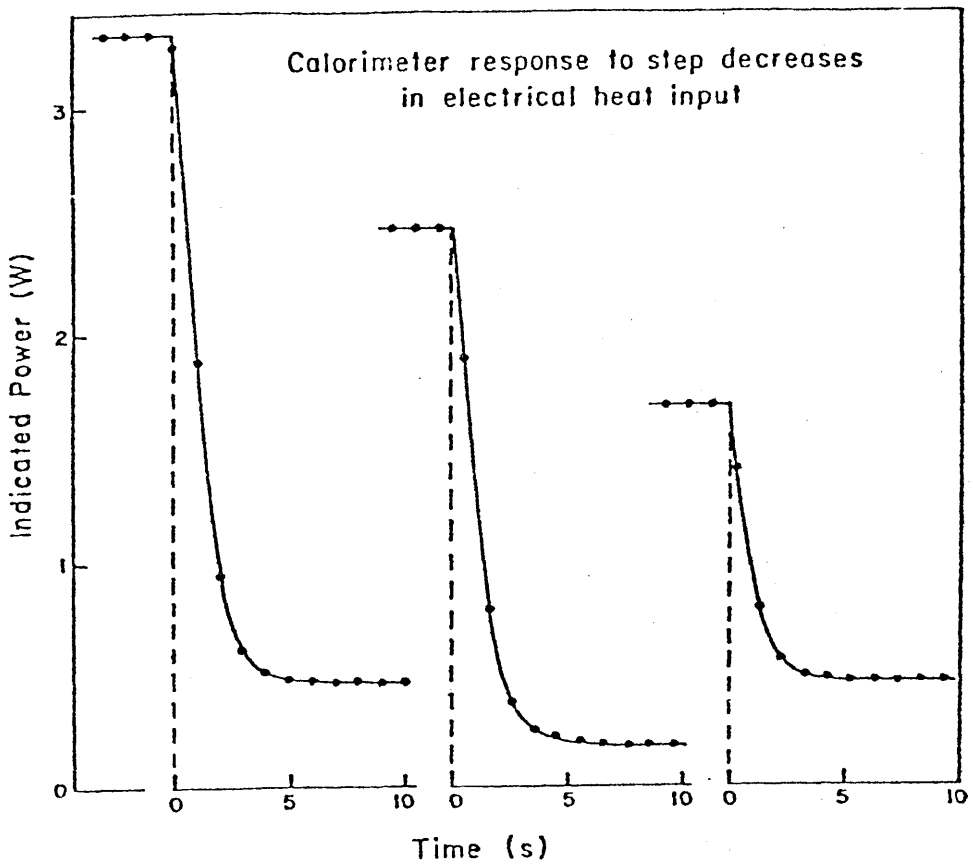

Fig. 3. Calorimeter response to step changes in electrical input. Step changes in electric heat input to the calorimeter were supplied via the electric heater embedded in the copper absorber block. For changes between power levels that were both > 70 mW, the calorimeter response was well represented by a single exponential function. Ten measurements that included both increases and decreases in input power yielded a value of 0.85 ± 0.09 s for the exponential response time. Below 70 mW, the anemometer flowmeter was no longer accurate. For powers below 70 mW, the boil-off flow rate was measured using an integrating volumetric flowmeter and a stopwatch. This caused no problems, since the decay heat was changing very slowly by the time ($\sim 3 \times 10^4$ s) it had fallen to 70 mW. The electric heater was also used to determine an absolute calibration curve for the calorimeter. The calibration curve so obtained was in excellent agreement with one obtained from absolute gas-flow measurements using the well-known heat of vaporization of helium. The final calibration curve used in the determination of the ^{235}U decay heat was a combination of the curves obtained by the two independent methods. In an additional measurement, the heat leak into the copper block was determined to be $\leq$ 20 μW under the conditions of the experiment.

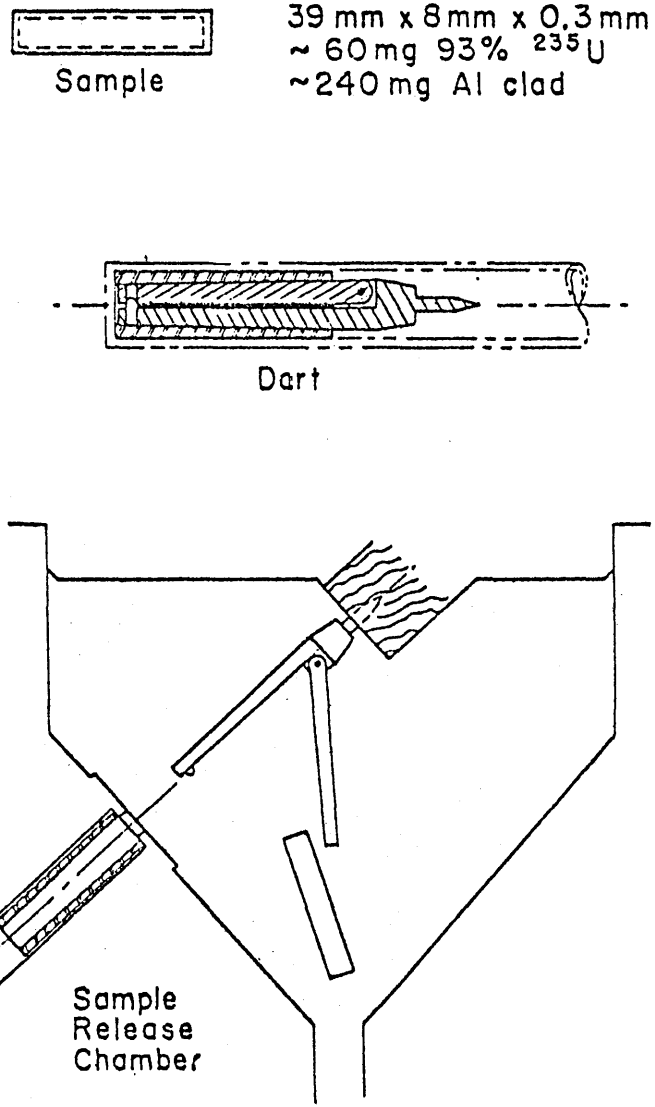

Fig. 4. Uranium - 235 sample and the system used to transport it from the irradiation position to the calorimeter. During irradiation, the sample was held between the hinged halves of an aluminum dart, which was clamped together by a tapered steel sleeve. After irradiation, the dart assembly was blown into the sample release chamber, where the sleeve was stripped off and the dart opened as it hit and stuck in a wooden target. Upon impact, the sample was released, then fell into the calorimeter through a ball valve that was opened at the correct time. The sample reached the liquid helium in the calorimeter $\sim$ 1.5 s after the start of the ejection. It reached the temperature of the liquid helium $\sim$ 3 s later. The ball valve remained open until the puff of gas from the sample cool-down had escaped; then it was closed, forcing the boil-off gas to pass through the flow-meter. During irradiation, helium gas flowed over the sample and through a charcoal trap near a NaI(Tl) radiation detector, which gave a prompt and sensitive indication of any escape of fission gases. Any samples that leaked were rejected. During the calorimetric measurements, the samples were at 4 K and all fission products were solids.

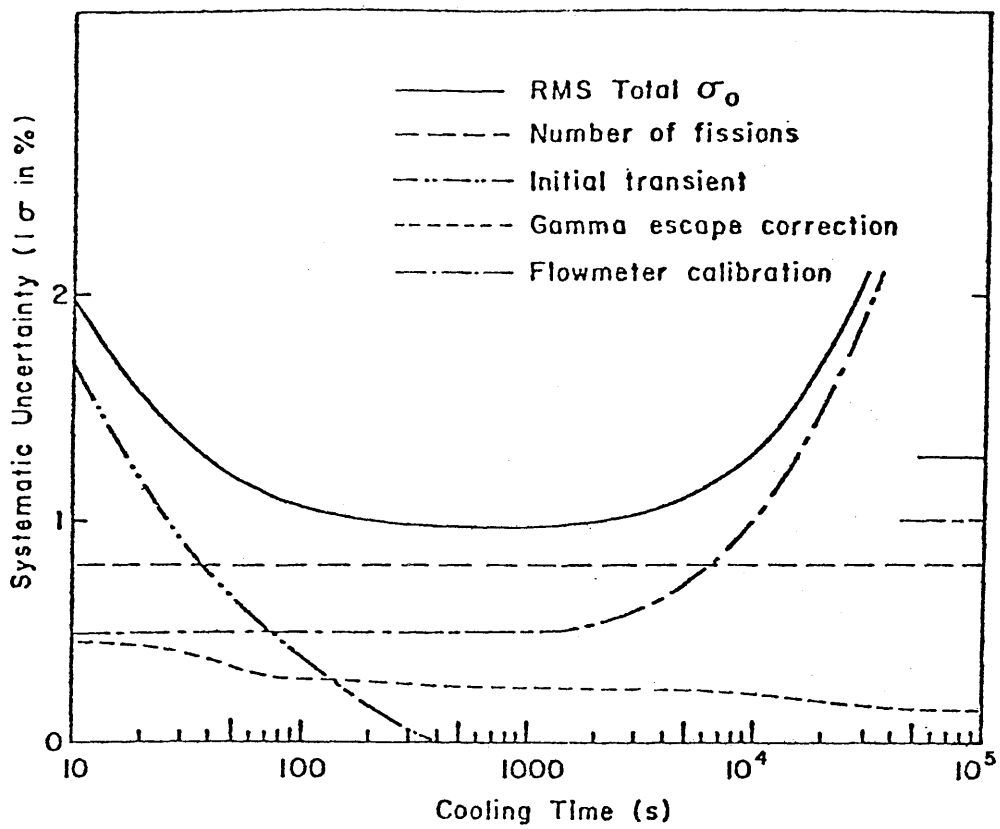

Fig. 6. Systematic uncertainties (those that do not contribute to the scatter in the data points at a given cooling time). The uncertainty in the number of fissions was estimated from an analysis of the calibration method used and from consistancy checks with other Laboratories. The initial-transient correction consisted of (1) subtraction of the signal from an irradiated aluminum dummy sample and (2) an analytic correction based on the 0.85 s time constant of the calorimeter. The total initial-transient correction was assigned an uncertainty of 50% of its value. The gamma escape correction was assumed known to 15% of its value at the shortest cooling times, and to 10% of its value for cooling times > 70 s. The flowmeter calibration curve uncertainty was estimated from the accuracy of the calibration standards and from the scatter of the calibration points around the fitted calibration curve. Flow measurements for t > 60 000 s were made using an integrating volumetric flowmeter. Its uncertainty is shown by a separate part of the curve. Because they are uncorrelated, the total systematic uncertainty, σ_o, was taken to be the RMS sum of the individual systematic uncertainties.

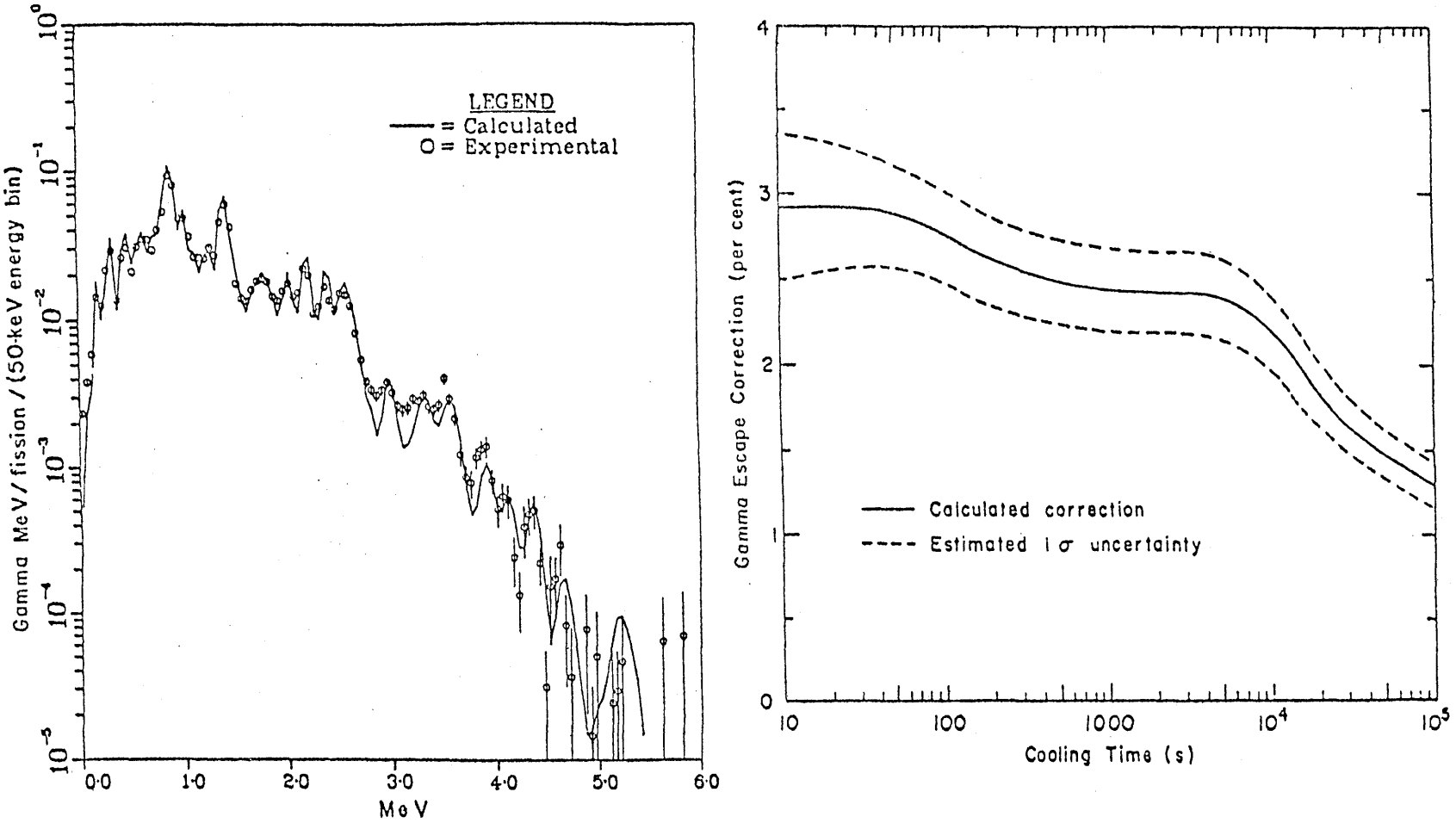

Fig. 5. Fission-product gamma spectrum (left) measured at a mean cooling time of 660 s after a 2×10^4 s ^{235}U irradiation. The solid curve was obtained from the ENDF/B-IV data base using a summation calculation. The correction for gamma escape from the calorimeter (right) was obtained from ten measured spectra using a Monte Carlo gamma transport calculation.

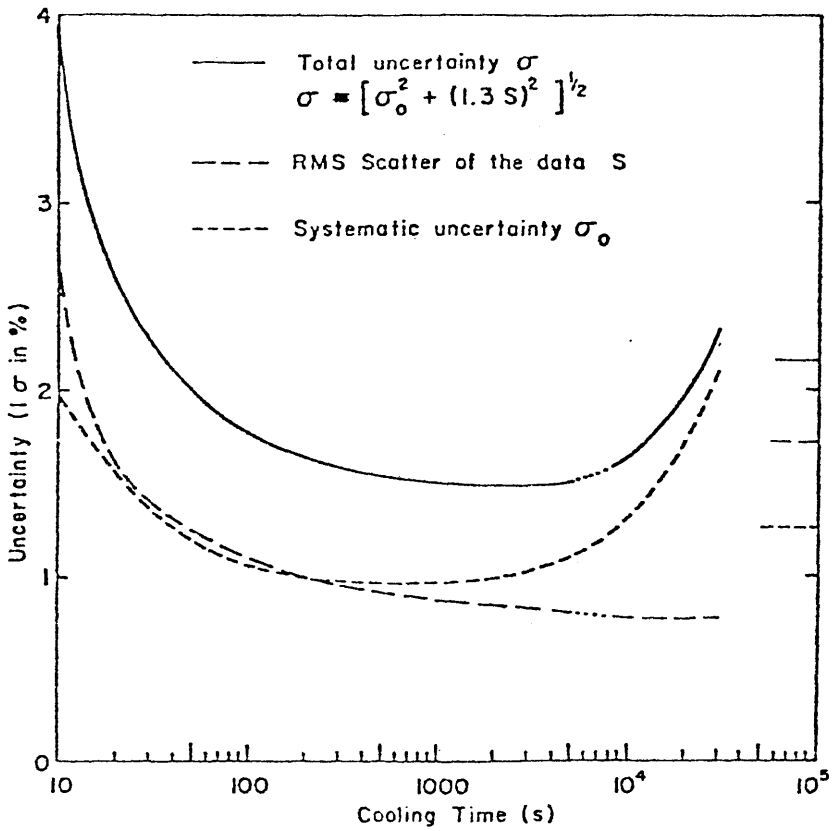

Fig. 7. Total uncertainty, σ, in the experimental data as estimated from the systematic uncertainty, σ_o, (Fig. 6.) and the RMS scatter, S, in the measurements made on different samples at the same cooling time. For most cooling times, data were obtained from three samples. When there were fewer than three data points, S was calculated by estimating the missing data points using nearby values from the same sample. Values of S larger than those shown on the curve were obtained for cooling times near 7000 s after the calorimeter was refilled with liquid helium. These times are indicated by dotted portions of the graph of S. The correct values were used in preparing Table I. To compensate for the small number of samples used (three), the experimental values of S were increased by a factor of 1.3 when the total uncertainty was calculated. All stated uncertainties are intended to represent one standard deviation of a normal distribution.

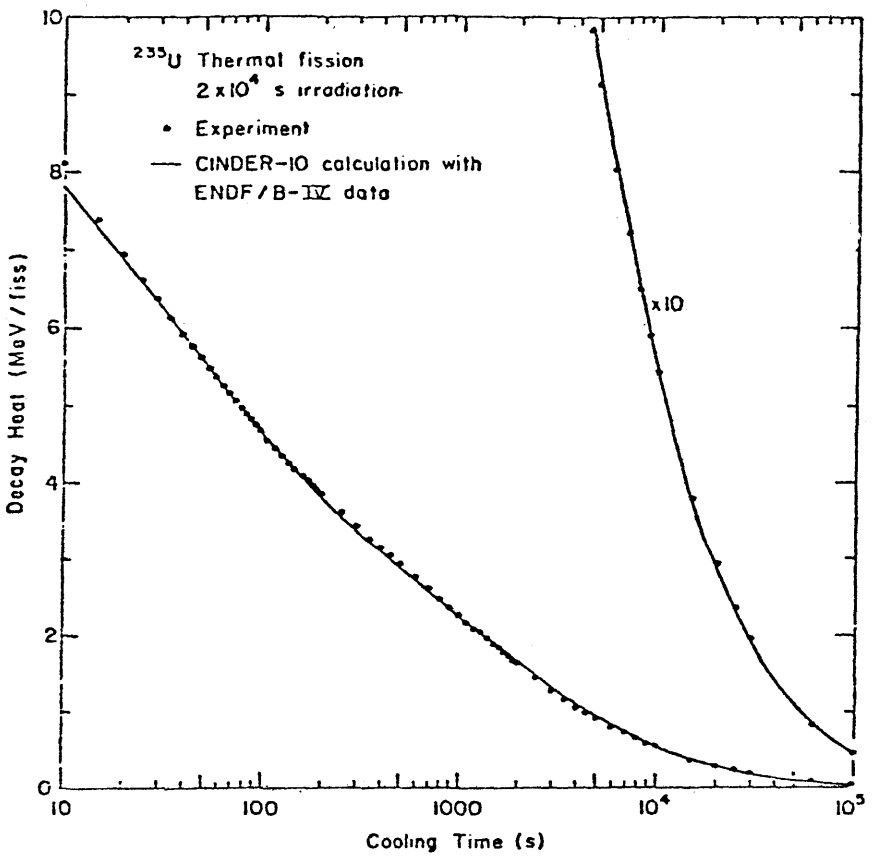

Fig. 8. Experimental and calculated decay heat for a 2 x 10^4 s irradiation of ^{235}U at constant thermal-neutron flux. The calculations used the ENDF/B-IV data base as listed and corrected by England and Schenter in Los Alamos Scientific Laboratory Report LA-6116-MS (ENDF-223). Measurements and calculations indicate that, under the conditions of the experiment, neutron capture in fission products, fission of ^{238}U, and production of ^{239}U and ^{239}Np were all negible. The calorimetric measurements were normalized to the fission rates in the samples determined by counting selected fission products subsequent to the calorimetric measurements. The isotopes ^{99}Mo and ^{140}Ba were beta-counted in chemically separated aliquots of the sample; ^{95}Zr, ^{99}Mo, ^{140}Ba-La, ^{141}Ce, and ^{147}Nd were gamma-counted in an unseparated aliquot. The fission determinations were calibrated against direct fission counting.

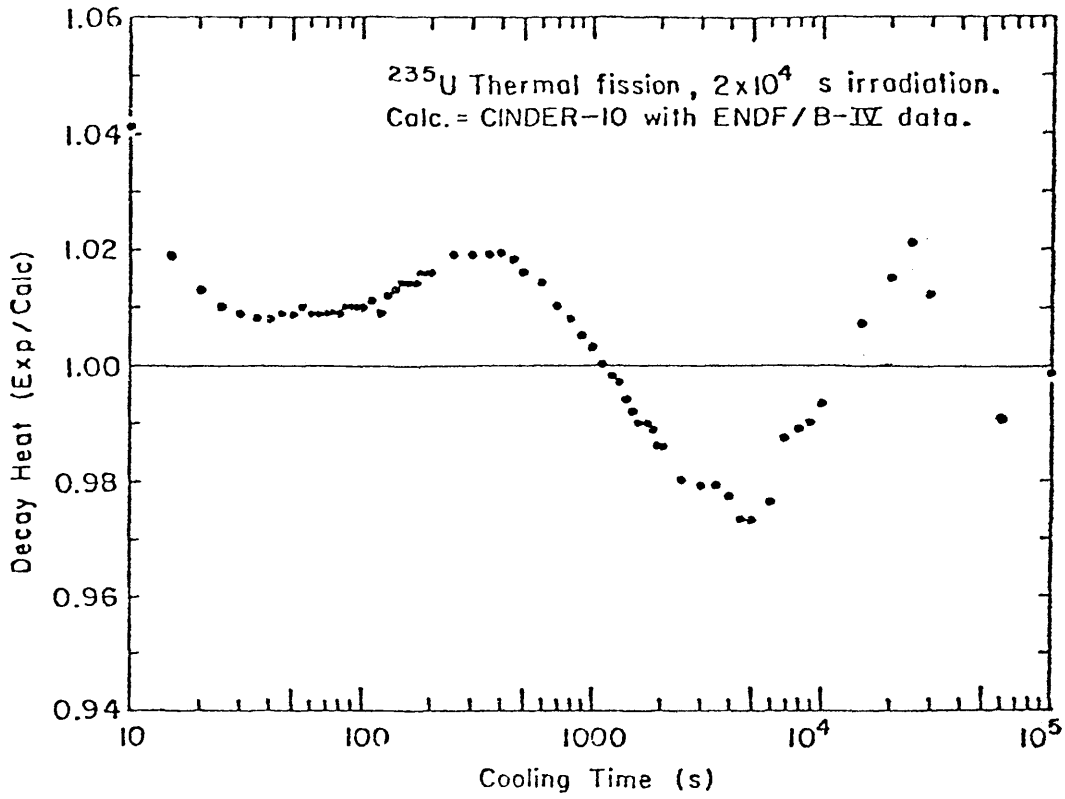

Fig. 9. Ratio of Experimental to calculated decay heat for a 2 x 10^4 s ^{235}U irradiation. To emphasize the small differences in the results shown in Fig. 8, the experimental data points have been divided by the summation calculation for the same cooling time and the ratios have been plotted on an expanded scale. The differences are well within the combined uncertainties for the experiment and the calculation.

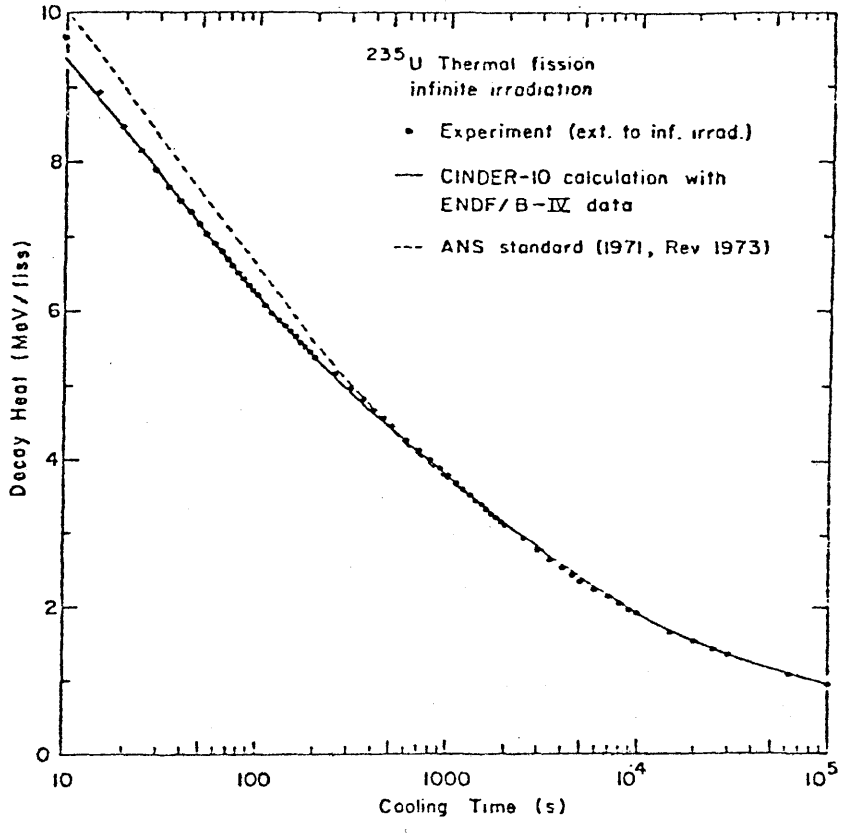

Fig. 10. Decay heat for infinite (10^{13} s) thermal-neutron irradiation of ^{235}U at constant fission rate without neutron capture in fission products. The experimental data points have been extended to infinite irradiation by the addition of summation calculations made using the ENDF/B-IV data base. The calculational contribution to the infinite-irradiation decay heat was 16% at t = 10 s, 51% at t = 2500 s, and 95% at t = 10^5 s. Also shown are the summation calculation and the present ANS Decay Heat Standard for infinite irradiation. It may be seen that the extended experimental data are in good agreement with the summation calculation, and are 7% below the standard at short cooling times. Moreover, the uncertainty associated with these measurements and calculations is significantly lower than that assigned to the standard. The ANS standard plus 20% is currently used for the decay heat source term in reactor safety evaluations.

TABLE

EXPERIMENTAL AND CALCULATED DECAY HEAT FOR THE PRODUCTS OF THERMAL FISSION OF ^{235}U.

	2×10^4 s Irradiation				Infinite Irradiation			
Cooling Time (s)	Experimental Decay Heat (MeV/fiss)	Experimental Uncertainty (1 σ in %)	Calculated Decay Heat using CINDER-10 and ENDF/B-IV (MeV/fiss)	Ratio Exp/Calc Decay Heat	Experimental Decay Heat Extended by CINDER-10 and ENDF/B-IV (MeV/fiss)	Experimental Uncertainty[a] (1 σ in %)	Calculated Decay Heat using CINDER-10 and ENDF/B-IV (MeV/fiss)	Ratio Exp/Calc Decay Heat
10	8.10	4.1	7.780	1.041	9.65	3.5	9.327	1.035
15	7.38	3.0	7.239	1.019	8.93	2.5	8.786	1.016
20	6.933	2.6	6.842	1.013	8.480	2.2	8.389	1.011
25	6.595	2.4	6.531	1.010	8.142	2.0		
30	6.335	2.3	6.276	1.009	7.882	1.9		
35	6.109	2.2	6.060	1.008	7.656	1.8		
40	5.920	2.1	5.873	1.008	7.467	1.7	7.420	1.006
45	5.758	2.1	5.709	1.009	7.305	1.7		
50	5.614	2.0	5.562	1.009	7.160	1.6		
55	5.481	2.0	5.429	1.010	7.027	1.6		
60	5.358	2.0	5.309	1.009	6.904	1.6	6.855	1.007
65	5.244	1.9	5.198	1.009	6.790	1.5		
70	5.141	1.9	5.097	1.009	6.687	1.5		
75	5.047	1.9	5.003	1.009	6.593	1.5		
80	4.958	1.8	4.915	1.009	6.504	1.5	6.461	1.007
85	4.881	1.8	4.834	1.010	6.427	1.4		
90	4.806	1.8	4.758	1.010	6.351	1.4		
95	4.734	1.8	4.686	1.010	6.279	1.4		
100	4.667	1.8	4.619	1.010	6.212	1.4	6.164	1.008
110	4.544	1.8	4.496	1.011	6.089	1.4		
120	4.426	1.7	4.385	1.009	5.971	1.4		
130	4.339	1.7	4.286	1.012	5.884	1.4		
140	4.251	1.7	4.195	1.013	5.795	1.4		
150	4.170	1.7	4.112	1.014	5.714	1.4	5.655	1.010
160	4.092	1.7	4.035	1.014	5.636	1.4		
170	4.021	1.7	3.964	1.014	5.565	1.3		
180	3.960	1.7	3.899	1.016	5.503	1.3		
190	3.899	1.6	3.838	1.016	5.442	1.3		
200	3.841	1.6	3.780	1.016	5.384	1.3	5.323	1.011
250	3.608	1.6	3.541	1.019	5.150	1.3		
300	3.419	1.6	3.355	1.019	4.960	1.3		
350	3.265	1.6	3.205	1.019	4.805	1.3		
400	3.135	1.6	3.078	1.019	4.673	1.3	4.616	1.012
450	3.022	1.6	2.969	1.018	4.559	1.3		
500	2.920	1.6	2.873	1.016	4.456	1.3		
600	2.746	1.5	2.709	1.014	4.280	1.2	4.243	1.009
700	2.598	1.5	2.572	1.010	4.130	1.2		
800	2.474	1.5	2.455	1.008	4.004	1.2	3.984	1.005
900	2.363	1.5	2.351	1.005	3.890	1.2		
1000	2.264	1.5	2.258	1.003	3.789	1.2	3.785	1.002

TABLE

(continued)

	2×10^4 s Irradiation				Infinite Irradiation			
Cooling Time (s)	Experimental Decay Heat (MeV/fiss)	Experimental Uncertainty (1 σ in %)	Calculated Decay Heat using CINDER-10 and ENDF/B-IV (MeV/fiss)	Ratio Exp/Calc Decay Heat	Experimental Decay Heat Extended by CINDER-10 and ENDF/B-Iv (MeV/fiss)	Experimental Uncertainty[a] (1 σ in %)	Decay Heat from CINDER-10 & ENDF/B-IV (MeV/fiss)	Ratio Exp/Calc Decay Heat
1100	2.173	1.5	2.174	1.000	3.696	1.2		
1200	2.093	1.5	2.098	0.998	3.614	1.2		
1300	2.020	1.5	2.027	0.997	3.539	1.2		
1400	1.950	1.5	1.962	0.994	3.467	1.2		
1500	1.886	1.5	1.901	0.992	3.401	1.2	3.416	0.996
1600	1.827	1.5	1.845	0.990	3.340	1.2		
1700	1.773	1.5	1.791	0.990	3.284	1.2		
1800	1.721	1.5	1.741	0.989	3.229	1.2		
1900	1.671	1.5	1.694	0.986	3.177	1.2		
2000	1.627	1.5	1.650	0.986	3.131	1.2	3.154	0.993
2500	1.431	1.5	1.460	0.980	2.925	1.3		
3000	1.283	1.5	1.311	0.979	2.768	1.3		
3500	1.166	1.5	1.191	0.979	2.641	1.3		
4000	1.067	1.5	1.092	0.977	2.533	1.3	2.557	0.991
4500	0,9808	1.5	1.008	0.973	2.438	1.3		
5000	0.9111	1.5	0.9362	0.973	2.360	1.3		
6000	0.7998	2.3	0.8198	0.976	2.231	1.5	2.251	0.991
7000	0.7195	2.0	0.7287	0.987	2.135	1.5		
8000	0.6480	1.6	0.6553	0.989	2.049	1.5	2.056	0.997
9000	0.5886	1.6	0.5948	0.990	1.975	1.5		
10000	0.5401	1.7	0.5440	0.993	1.912	1.5	1.916	0.998
15000	0.3803	1.8	0.3778	1.007	1.691	1.6	1.689	1.001
20000	0.2918	2.0	0.2874	1.015	1.552	1.7	1.547	1.003
25000	0.2359	2.2	0.2311	1.021	1.453	1.7		
30000	0.1947	2.3	0.1923	1.012	1.375	1.7		
62183	0.0823	2.2	0.0832	0.989	1.105	1.9		
100000	0.0454	2.2	0.0455	0.998	0.971	1.9	0.969	1.002

[a]Calculated on the assumptions that the calculated infinite irradiation decay heat curve has 2% uncertainty for $t_{cool} \geq 2 \times 10^4$ s, and that this uncertainty is uncorrelated with the experimental uncertainty.