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ENDF/B-5 FISSION PRODUCT
CROSS SECTION EVALUATIONS

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ENDF/B-5 FISSION PRODUCT CROSS SECTION EVALUATIONS*

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ABSTRACT

Cross section evaluations have been made for the 196 fission product nuclides on the ENDF/B-5 data files. Most of the evaluations involve updating the capture cross sections of the important absorbers for fast and thermal reactor systems. This included updating thermal values, resonance integrals, resonance parameter sets and fast capture cross sections. For the fast capture results generalized least-squares calculations were made using the computer code FERRET. Input for these cross section adjustments included nuclear models calculations and both integral and differential experimental data results. The differential cross sections and their uncertainties were obtained from the CSIRS library. Integral measurement results came from CFRMF and STEK Assemblies 500, 1000, 2000, 3000, 4000. Comparisons of these evaluations with recent capture measurements will be presented.

Introduction

The ENDF/B-5 fission product file consists of 877 nuclide evaluations. Of these, 196 have full cross section data sets which contain values for total, elastic, inelastic, capture and angular distributions for the energy range 10^{-5} eV to 20 MeV. Table I lists the 196 nuclides. The file is basically an update of ENDF/B-4, where most of the changes occur for the fast and thermal capture cross sections. These changes involve the use of important new evaluation techniques and recent experimental results from both integral and differential measurements.

In this paper we present typical results illustrating these new techniques where use was made of a generalized least-squares adjustment procedure^{1/} to obtain a nominal cross section curve and uncertainty information

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Table I. Fission Product Nuclides with Cross Section Data for ENDF/B-5.

1.	32	Ge	72	51.	43	Tc	99	101.	51	Sb	125	151.	60	Nd	142
2.	32	Ge	73	52.	44	Ru	96	102.	51	Sb	126	152.	60	Nd	143
3.	32	Ge	74	53.	44	Ru	98	103.	52	Te	120	153.	60	Nd	144
4.	32	Ge	76	54.	44	Ru	99	104.	52	Te	122	154.	60	Nd	145
5.	33	As	75	55.	44	Ru	100	105.	52	Te	123	155.	60	Nd	146
6.	34	Se	75	56.	44	Ru	101	106.	52	Te	124	156.	60	Nd	147
7.	34	Se	76	57.	44	Ru	102	107.	52	Te	125	157.	60	Nd	148
8.	34	Se	77	58.	44	Ru	103	108.	52	Te	126	158.	60	Nd	150
9.	34	Se	78	59.	44	Ru	104	109.	52	Te	127M	159.	61	Pm	147
10.	34	Se	80	60.	44	Ru	105	110.	52	Te	128	160.	61	Pm	148
11.	34	Se	82	61.	44	Ru	106	111.	52	Te	129M	161.	61	Pm	148M
12.	35	Br	79	62.	45	Rh	103	112.	52	Te	130	162.	61	Pm	149
13.	35	Br	81	63.	45	Rh	105	113.	52	Te	132	163.	61	Pm	151
14.	36	Kr	78	64.	46	Pd	102	114.	53	I	127	164.	62	Sm	144
15.	36	Kr	80	65.	46	Pd	104	115.	53	I	129	165.	62	Sm	147
16.	36	Kr	82	66.	46	Pd	105	116.	53	I	130	166.	62	Sm	148
17.	36	Kr	83	67.	46	Pd	106	117.	53	I	131	167.	62	Sm	149
18.	36	Kr	84	68.	46	Pd	107	118.	53	I	135	168.	62	Sm	150
19.	36	Kr	85	69.	46	Pd	108	119.	54	Xe	124	169.	62	Sm	151
20.	36	Kr	86	70.	46	Pd	110	120.	54	Xe	126	170.	62	Sm	152
21.	37	Rb	85	71.	47	Ag	107	121.	54	Xe	128	171.	62	Sm	153
22.	37	Rb	86	72.	47	Ag	109	122.	54	Xe	129	172.	62	Sm	154
23.	37	Rb	87	73.	47	Ag	111	123.	54	Xe	130	173.	63	Eu	151
24.	38	Sr	84	74.	48	Cd	106	124.	54	Xe	131	174.	63	Eu	152
25.	38	Sr	86	75.	48	Cd	108	125.	54	Xe	132	175.	63	Eu	153
26.	38	Sr	87	76.	48	Cd	110	126.	54	Xe	133	176.	63	Eu	154
27.	38	Sr	88	77.	48	Cd	111	127.	54	Xe	134	177.	63	Eu	155
28.	38	Sr	89	78.	48	Cd	112	128.	54	Xe	135	178.	63	Eu	156
29.	38	Sr	90	79.	48	Cd	113	129.	54	Xe	136	179.	63	Eu	157
30.	39	Y	89	80.	48	Cd	114	130.	55	Cs	133	180.	64	Gd	152
31.	39	Y	90	81.	48	Cd	115M	131.	55	Cs	134	181.	64	Gd	154
32.	39	Y	91	82.	48	Cd	116	132.	55	Cs	135	182.	64	Gd	155
33.	40	Zr	90	83.	49	In	113	133.	55	Cs	136	183.	64	Gd	156
34.	40	Zr	91	84.	49	In	115	134.	55	Cs	137	184.	64	Gd	157
35.	40	Zr	92	85.	50	Sn	112	135.	56	Ba	134	185.	64	Gd	158
36.	40	Zr	93	86.	50	Sn	114	136.	56	Ba	135	186.	64	Gd	160
37.	40	Zr	94	87.	50	Sn	115	137.	56	Ba	136	187.	65	Tb	159
38.	40	Zr	95	88.	50	Sn	116	138.	56	Ba	137	188.	65	Tb	160
39.	40	Zr	96	89.	50	Sn	117	139.	56	Ba	138	189.	66	Dy	160
40.	41	Nb	93	90.	50	Sn	118	140.	56	Ba	140	190.	66	Dy	161
41.	41	Nb	94	91.	50	Sn	119	141.	57	La	139	191.	66	Dy	162
42.	41	Nb	95	92.	50	Sn	120	142.	57	La	140	192.	66	Dy	163
43.	42	Mo	92	93.	50	Sn	122	143.	58	Ce	140	193.	66	Dy	164
44.	42	Mo	94	94.	50	Sn	123	144.	58	Ce	141	194.	67	Ho	165
45.	42	Mo	95	95.	50	Sn	124	145.	58	Ce	142	195.	68	Er	166
46.	42	Mo	96	96.	50	Sn	125	146.	58	Ce	143	196.	68	Er	167
47.	42	Mo	97	97.	50	Sn	126	147.	58	Ce	144				
48.	42	Mo	98	98.	51	Sb	121	148.	59	Pr	141				
49.	42	Mo	99	99.	51	Sb	123	149.	59	Pr	142				
50.	42	Mo	100	100.	51	Sb	124	150.	50	Pr	143				

in the form of a covariance matrix which linked energy points. This procedure involves calculations which use the finite element representation of the FERRET /2/ data adjustment code. Also presented in this paper will be a study of thermal reactor absorption sensitivity to capture cross section changes in important fission products.

FAST CAPTURE

Table II lists the 25 most important fission product absorbers in a fast reactor core with their percent contribution to total absorption. Emphasis for ENDF/B-5 has been on updating the capture cross sections of these top absorbers plus an additional 50 more. Typical results of these evaluations are shown in Figures 1-14, where the "Adjusted" curves will be used for ENDF/B-5. Also shown is the data used as input to the FERRET code. This includes both differential and integral data and their uncertainties. Also inputted to the calculation was an "a priori" description which combined multi-group average cross sections obtained from resonance parameters for the resolved resonance region with "smooth" average cross section from ENDF/B-4 for the higher energy region. The resonance parameters used were from ENDF/B-4 or BNL-325 /3/ as indicated. The histogram or multi-group cross section description in the resonance region is required for the FERRET least-squares calculation because following the exact resonance structure takes too many points for the standard computer calculations, especially for the covariance matrix part.

The integral data came from reaction rate measurements in STEK Assemblies 500, 1000, 2000, 3000, 4000 /4/ and CFRMF /5/. ENDF/B-4 evaluations also used earlier CFRMF measurements which helped normalize nuclear models calculations made with the Hauser-Feshbach computer code NCAP /6/.

This type of approach which quantitatively combines three evaluation sources, diverges somewhat from the usual ENDF/B philosophy which is to rely essentially on differential results with some adjustment due to integral results. This method also differs from the Japanese and European procedures in which either only differential or only integral values are exclusively used. For individual fission products, however, it is felt that this approach has substantial promise since many of the problems associated with the use of integral data are greatly simplified in dealing with a single nuclide. As can be seen in Table II, for most of the nuclides which have experimental measurements, the differential data is discrepant. Consequently, it is hoped that use of the integral results will supplement our information and help resolve these discrepancies. Another advantage of the procedure is that it provides quantitative uncertainty information in the form of covariance matrices which can be easily translated into ENDF/B formats. Obviously there are problems associated with using integral data such as self shielding effects, mixtures of isotopes for the same element, flux and adjoint flux determinations and uncertainty assignments and these will have to be examined carefully to have a viable method.

Figure 1 shows results for the important absorber Sm149. As can be seen from the curves, significant changes occur in going from ENDF/B-4 to ENDF/B-5. The ENDF/B-4 result was made with nuclear model calculations where no previous experimental data existed in 1973. Also shown in Figure 1 are "Adjusted" cross sections using the FERRET code, comparing various results when only integral data is used or differential data is used. For the Sm149 case, the results are consistent with each other. For some of the other cases shown in Figures 5-14 this does not necessarily occur.

Table II. Twenty five most "important" fission product absorbers in fast reactor cores. Also shown are fast integral and differential experiments which have published results as of November 1979. The CFRMF, STEK and differential data were used in the HEDL-PETTEN evaluations for ENDF/B-5.

Nuclide	Percent Contribution to Total Absorption	EBR-2	CFRMF	STEK	Differen- tial Data*	Discrepant Diff. Data
1. Pd 105	9.9	-	-	✓	M,S, ..	-
2. Tc 99	8.6	-	✓	✓	S,..	✓
3. Ru 101	7.7	-	-	✓	M,S	✓
4. Pd 107	6.2	-	-	-	-	no data
5. Rh 103	5.5	-	✓	✓	M,S,..	-
6. Cs 133	4.9	-	✓	✓	..	✓
7. Pm 147	3.5	-	✓	-	-	no data
8. Sm 149	3.4	✓	-	✓	S	one data set
9. Nd 145	3.4	✓	-	-	S	one data set
10. Ru 102	3.3	-	✓	✓	M,S,..	✓
11. Cs 135	3.0	-	-	-	-	no data
12. Mo 97	2.9	-	-	✓	Mu,..	-
13. Ag 109	2.7	-	✓	✓	..	✓
14. Ru 106	2.3	-	-	-	-	no data
15. Nd 143	2.3	✓	-	-	Mu	one data set
16. Xe 131	1.9	-	-	-	-	no data
17. Sm 151	1.9	-	-	✓	-	no data
18. Mo 95	1.5	-	-	✓	Mu,..	-
19. Ru 104	1.3	-	✓	✓	M,S	✓
20. Eu 153	1.3	-	✓	-	..	-
21. Mo 98	1.2	-	✓	✓	Mu,..	✓
22. Ce 144	1.1	-	-	-	-	no data
23. I 129	1.0	-	✓	✓	-	no data
24. Mo 100	0.9	-	✓	✓	Mu,..	✓
25. Pr 141	<u>0.9</u>	-	✓	✓	..	✓
82.6						

* M=Macklin (79)ORL, S=Shaw(75)RPI, Mu=Musgrove(77)AUA;
 ..=Additional differential data

Error estimates are outputted from the FERRET code and Figure 2 shows fractional uncertainty versus energy for the Sm149 case, where the final uncertainty changes from about 60% to 10% (1σ) in going from ENDF/B-4 to ENDF/B-5. Figures 3 and 4 show the cases of adjusted curves using STEK measurements compared to recent integral results at EBR-II adjusted by Anderl et al., /7/ for Sm149 and Sm147. The full energy region (1 eV - 10 MeV) evaluation is shown in Figure 5 for Mo97. Comparisons to recent differential measurements by Macklin, et al., /8/ is shown in Figures 6-10 for Ru101, Ru102, Ru104, Pd104, and Pd108. Finally, results for four of the top twenty-five fission product absorbers (Tc99, Rh103, Ag109 and Cs133) are given in Figures 11-14.

FIGURE 1. SM 149 CROSS SECTION EVALUATIONS. ENDF/B4+BNL - APRIORI.

FIGURE 2. SM 149 UNCERTAINTY EVALUATION. ENDF/B4+BNL - APRIORI.

FIGURE 3. SM 149 CROSS SECTION EVALUATIONS. ENDF/B4+BNL - APRIORI.

FIGURE 4. SM 147 CROSS SECTION EVALUATION. ENDF/B4+BNL - APRIORI.

FIGURE 5. MO 97 CROSS SECTION EVALUATION. ENDF/B4 - APRIORI.

FIGURE 6. RU 101 CROSS SECTION EVALUATION. ENDF/B4 - APRIORI.

FIGURE 7. RU 102 CROSS SECTION EVALUATION. ENDF/B4 - APRIORI.

FIGURE 8. RU 104 CROSS SECTION EVALUATION. ENDF/B4 - APRIORI.

FIGURE 9. PD 104 CROSS SECTION EVALUATION. ENDF/B4 - APRIORI.

FIGURE 10. PD 108 CROSS SECTION EVALUATION. ENDF/B4.

FIGURE 11. Tc 99 EVALUATION. ENDF/B4+BNL - APRIORI.

FIGURE 12. RH 103 EVALUATION. ENDF/B4+BNL - APRIORI.

FIGURE 13. Ag 109 EVALUATION. ENDF/B4+BNL - APRIORI.

FIGURE 14. Cs 133 EVALUATION. ENDF/B4 - APRIORI.

THERMAL AND RESONANCE CAPTURE

Table III gives the results of calculations of ENDF/B-4 resonance integrals ($E_{\text{cut}}=0.5$, $T=0^\circ\text{K}$) and thermal cross sections for the major fission product absorbers in thermal reactors. These will form the basis of the ENDF/B-5 evaluations. Also shown in Table III are the values given in BNL-325 /3/ which for most cases are identical or very close to, the ENDF/B-4 results. Most of the ENDF/B-5 "196 nuclides" with cross section data will have "File 2" resonance parameter descriptions. The "File 3" smooth background cross section will then be adjusted consistent with experimental results to give measured resonance integrals (within 1σ) and measured thermal cross sections. The main source of experimental resonance integrals and thermal values was BNL-325 which has almost identical agreement with the "Chart of the Nuclides". For those cases which BNL-325 did not list measured values, other sources of data were from Clayton /9/, Ribon and Krebs /10/, Pope and Story /11/, and Kirouac /12/.

Appendix A provides thermal reactor absorption sensitivity analysis.

Table III. Resonance Integrals and Thermal Cross Sections of Major EP Absorbers in Thermal Reactors.

Nuclide	Resonance Integrals (barns)		Thermal Capture Cross Sections (barns)	
	ENDF/B-4*	BNL-325	ENDF/B-4	BNL-325
Xe135	7645.	7634.	2.64×10^6	$(2.65 \pm 0.20) \times 10^6$
Sm149	3200.	3183.	41190.	$41000. \pm 2000.$
Pm147	2283.	$2300. \pm 350.$	182.	$181. \pm 7.$
Xe131	879.	870.	90.	$90. \pm 10.$
Cs133	381.	$415. \pm 15.$	29.5	31.5 ± 1.5
Nd143	205.	140.	325.	$325. \pm 10.$
Sm151	3357.	$3300. \pm 700.$	15000.	$15000. \pm 1200.$
Rh103	1050.	$1100. \pm 50.$	148.	$150. \pm 5.$
Tc99	354.	$340. \pm 35.$	19.	$19. \pm 2.$
Nd145	226.	$240. \pm 35.$	42.	$42. \pm 2.$
Sm152	3008.	$3000. \pm 200.$	206.	$206. \pm 6.$
Pm148M	3608.	3600.	10600.	$22000. \pm 2500.$
Mo95	113.	$105. \pm 7.$	14.5	14.5 ± 0.5
Eu153	1590.	$1635. \pm 200.$	453.	$390. \pm 30.$
Kr83	192.	230.	208.	$200. \pm 30.$
Rh105	15850.	$15800. \pm 1500$	16000.	$16000. \pm 1500.$
Sm150	320.	$310. \pm 15.$	102.	$102. \pm 5.$
Pm148	40010.	40000.	2000.	$2000. \pm 1000$
Cs135	61.6	62.	8.7	8.7 ± 0.5
Ru101	95.2	85.	3.1	3.1 ± 0.9
Eu155	1856.	--	4040.	$4040. \pm 125.$
La139	13.0	12.2 ± 0.6	9.0	9.0 ± 0.3
Pr141	19.4	14.1 ± 0.2	11.5	11.5 ± 0.3
Eu154	2700.	--	1500.	$1500. \pm 400.$
Pr143	190.	$190. \pm 25.$	89.	$89. \pm 10.$
Xe133	356.	--	190.	$190. \pm 90.$
Cd113	405.	--	19880.	$19910. \pm 300.$
Nd146	3.3	3.2 ± 0.5	1.4	1.4 ± 0.1
Cs134	213.	--	140.	$140. \pm 12.$
Mo97	16.1	$13. \pm 3.$	2.2	2.2 ± 0.7

*ECUT=0.5 eV, T=0°K

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APPENDIX A

FISSION-PRODUCT ABSORPTION IN THERMAL REACTORS: MAJOR CONTRIBUTORS AND CROSS-SECTION SENSITIVITIES

All sensitivity data in this appendix are based on ENDF/B-IV; the ENDF/B-V files were not complete in time for a processing and sensitivity study before this meeting. However for thermal reactors, the cross-section changes are not likely to be significant to aggregate absorption buildup, and the sensitivities in this appendix can be used to predict the results of specific cross-section changes between Version IV and V.

The cross sections used in this study are listed in Refs. 1 and 2. These were first processed using NJOY/3/ in 154 energy groups and then spectrum weighted into the 4-group structure defined in Table A-I using a typical mid-life spectrum for LWR's. The function is illustrated in Fig. A-1 and listed in Ref. 1. A detailed listing of the 4-group cross sections will be available for this meeting, and differences in σ_{2200} and RI between Versions IV and V noted.

The depletion and buildup calculations also require decay and yield data. Decay parameters are listed in Ref. 4; ENDF/B-IV yields are essentially those listed in Ref. 5, except for minor changes in independent yields that are of no significance to this study. The basic depletion calculations use a 3.3 w/% U-235 enriched fuel at a constant power level of 30 Mw/MT for a total burnup of 33 Gwd/MT. Power and fission-product production was from the constantly changing production of Pu-239 and concomitant depletion of U-235 and U-238. The content and 4-group macroscopic absorption of each fission product was calculated at 33 times in increments of 800 h (1 Gwd/MT) out to 26 400 h (approximately 3 yr). A large number of such calculations were required for the sensitivity studies.

All parameters affecting the major nuclides contributing to the aggregate absorption were examined./6/ Space limits this appendix to a condensed summary of the results at two times in reactor life and to cross-section data. However, those readers interested in fission-product absorption in thermal reactors should remember that, depending on the nuclide, cross sections are not always the most important parameter. For example, at approximately 20 Gwd/MT depletion, the yields, branching fractions, and decay constants account for 26 of the largest 40 sensitivities. Furthermore, those nuclides having very large cross sections such as Sm-149 have relatively small cross-section sensitivities in LWR's. The reason for this behavior is simply that the products are both produced and depleted, and therefore those stable nuclides having large cross sections tend to reach an equilibrium macroscopic value determined only by the production (yield) rate; at constant power, the cross section determines the time of equilibrium, not the ultimate absorption rate. It is also true that an increase in the cross sections of some nuclides (e.g., Nd-147) can decrease the aggregate absorption, an effect due to the reduction of the decay branching to daughters having large cross sections. These statements are not intended to reduce the importance of cross sections. In fast reactors and in transient phenomena in LWR's, the cross sections are of dominate importance.

IMPORTANT NUCLIDE ABSORBERS IN LWR's

Table A-II lists 64 nuclides that contribute >0.1% to the aggregate thermal, epithermal, or total absorption at some time during the 33 Gwd/MT depletion. The specific contribution at 33 times of 10 of the most important contributors is listed in Table A-III.

Fig. A-1
Flux weighting function.

The chain systematics of each of the 64 nuclides was examined to determine the most important parameters (291) affecting their total absorption, and sensitivities were calculated./6/

SENSITIVITIES

Thermal and epithermal cross sections were individually increased by 10%, one at a time, and the change in aggregate absorption calculated. The tabulated sensitivity, S , of the aggregate absorption rate, R_T , was calculated from the change, ΔR_T , where R_T is a sum over all groups and nuclides.

$$S \equiv \frac{\Delta R_T / R_T}{\Delta X / X}, \quad \text{---(A-1)}$$

where ΔX is the 10% increase in either a thermal or epithermal (resonance) cross section X . Values of S thus represent the direct change in the aggregate absorption per unit change in one of these cross sections. Values of $S < 10^{-5}$ are listed as ~ 0 .

Table A-IV lists the resulting sensitivities at two depletion times, 5 GWd/MT and 33 GWd/MT (4000 and 26 400 h). A total of 78 nuclides were examined; that is, the list includes sensitivities for cross sections coupled to the initial 64 in Table A-II. An ordered list of the largest 15 sensitivities at both depletion times is given in Table A-V.

Sensitivities do not show the actual importance of the major contributors. Tables A-VI and A-VII show the percent contribution to the total, thermal, and resonance absorption, ordered on the total absorption for the major contributors.

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TABLE A-I

4-GROUP ENERGY STRUCTURE AND FLUX RATIOS

<u>Group</u>	<u>Energy Boundary eV</u>	<u>Lethargy Width</u>	<u>LWR Flux Ratios to Thermal*</u>
	10^7		
1	8.2085×10^5	2.5	1.4759
2	5.5308×10^3	5.0	2.3401
3	6.2506×10^{-1}	9.088	1.7833
4	1.0×10^{-5}	8.412	1.0

* The degree of spectrum hardness used is indicated by the average value, 0.554018, of a unit $1/v$ cross section at 0.0253 eV.

TABLE A-II

NUCLIDES CONTRIBUTING GREATER THAN 0.1% OF THERMAL, EPITHERMAL
OR TOTAL FISSION-PRODUCT ABSORPTION RATE AT SOME TIME DURING FUEL HISTORY
(to 33,000 MWd/MT)

Z	SYM	Thermal	Epithermal	Total
36	Kr	⁸³ Kr	⁸³ Kr	⁸³ Kr
40	Zr	⁹³ Zr	^{91,93,95,96} Zr	^{91,93} Zr
41	Nb		⁹⁵ Nb	
42	Mo	^{95,97} Mo	^{95,97,98,99,100} Mo	^{95,97,98} Mo
43	Tc	⁹⁹ Tc	⁹⁹ Tc	⁹⁹ Tc
44	Ru	¹⁰¹ Ru	^{101,102,103,104} Ru	^{101,102,103} Ru
45	Rh	^{103,105} Rh	^{103,105} Rh	^{103,105} Rh
46	Pd	^{105,107,108} Pd	^{104,105,107,108} Pd	^{105,107,108} Pd
47	Ag	¹⁰⁹ Ag	¹⁰⁹ Ag	¹⁰⁹ Ag
48	Cd	¹¹³ Cd		¹¹³ Cd
49	In		¹¹⁵ In	
53	I	¹²⁹ I	^{127,129} I	¹²⁹ I
54	Xe	^{131,133,135} Xe	^{131,133,135} Xe	^{131,133,135} Xe
55	Cs	^{133,134,135} Cs	^{133,134,135} Cs	^{133,134,135} Cs
56	Ba		¹⁴⁰ Ba	
57	La	¹³⁹ La	^{139,140} La	¹³⁹ La
58	Ce	¹⁴¹ Ce	^{141,143} Ce	¹⁴¹ Ce
59	Pr	^{141,143} Pr	^{141,143} Pr	^{141,143} Pr
60	Nd	^{143,144,145} Nd	^{143,144,145,147,148} Nd	^{143,144,145,147,148} Nd
61	Pm	^{147,148,148m,149} Pm	^{147,148,148m,149,151} Pm	^{147,148,148m,149} Pm
62	Sm	^{147,149,150,151,152} Sm	^{147,148,149,150,151,152,153} Sm	^{147,149,150,151,152} Sm
63	Eu	^{153,154,155,156} Eu	^{153,154,155} Eu	^{153,154,155,156} Eu
64	Gd	^{155,157} Gd	¹⁵⁶ Gd	¹⁵⁷ Gd

TABLE A-III

CONTRIBUTIONS OF TEN MAJOR NUCLIDES TO THE TOTAL NEUTRON ABSORPTION
RATE OF FISSION PRODUCTS

Percent of Total Fission-Product Absorption Rate

Elapsed Time, hours	Burnup GWD/MT	⁹⁹ Tc	¹⁰³ Rh	¹³¹ Xe	¹³⁵ Xe	¹³³ Cs	¹⁴³ Nd	¹⁴⁷ Pm	¹⁴⁹ Sm	¹⁵¹ Sm	¹⁵² Sm	All Others
800	1	0.49	0.29	0.55	72.97	0.52	0.76	0.79	16.98	2.46	0.30	3.89
1600	2	0.94	0.88	1.21	65.87	1.12	1.97	1.82	16.05	3.79	0.64	5.71
2400	3	1.32	1.56	1.77	60.25	1.61	3.05	2.66	15.11	4.46	0.98	7.23
3200	4	1.64	2.24	2.24	55.60	2.03	3.96	3.32	14.33	4.77	1.29	8.58
4000	5	1.91	2.89	2.64	51.67	2.39	4.73	3.85	13.65	4.88	1.58	9.81
4800	6	2.16	3.49	3.00	48.28	2.70	5.39	4.27	13.04	4.88	1.84	10.95
5600	7	2.37	4.04	3.31	45.32	2.99	5.96	4.61	12.50	4.83	2.08	11.99
6400	8	2.57	4.55	3.59	42.69	3.24	6.46	4.88	12.01	4.75	2.29	12.97
7200	9	2.75	5.02	3.84	40.34	3.48	6.90	5.10	11.56	4.65	2.48	13.88
8000	10	2.91	5.45	4.07	38.21	3.69	7.29	5.27	11.14	4.55	2.65	14.77
8800	11	3.07	5.84	4.27	36.27	3.89	7.63	5.41	10.74	4.44	2.80	15.64
9600	12	3.21	6.21	4.46	34.50	4.07	7.94	5.51	10.38	4.34	2.94	16.44
10400	13	3.34	6.55	4.63	32.87	4.24	8.22	5.59	10.03	4.25	3.06	17.22
11200	14	3.46	6.87	4.79	31.36	4.39	8.47	5.64	9.71	4.16	3.17	17.98
12000	15	3.58	7.16	4.93	29.96	4.54	8.69	5.67	9.40	4.07	3.27	18.73
12800	16	3.69	7.43	5.06	28.65	4.68	8.89	5.69	9.11	3.99	3.36	19.47
13600	17	3.79	7.68	5.18	27.43	4.81	9.06	5.69	8.83	3.91	3.44	20.18
14400	18	3.89	7.92	5.29	26.28	4.94	9.22	5.67	8.57	3.84	3.51	20.87
15200	19	3.98	8.14	5.39	25.20	5.05	9.36	5.65	8.31	3.77	3.57	21.58
16000	20	4.07	8.34	5.48	24.19	5.17	9.48	5.61	8.07	3.71	3.62	22.26
16800	21	4.16	8.53	5.56	23.24	5.27	9.59	5.56	7.84	3.65	3.67	22.93
17600	22	4.24	8.70	5.64	22.34	5.37	9.68	5.51	7.62	3.60	3.71	23.59
18400	23	4.31	8.86	5.71	21.48	5.47	9.76	5.45	7.41	3.55	3.75	24.25
19200	24	4.39	9.00	5.77	20.68	5.56	9.82	5.38	7.21	3.50	3.78	24.91
20000	25	4.46	9.14	5.82	19.91	5.65	9.87	5.30	7.01	3.46	3.81	25.57
20800	26	4.53	9.26	5.87	19.19	5.73	9.91	5.23	6.82	3.42	3.83	26.21
21600	27	4.59	9.37	5.91	18.51	5.81	9.94	5.14	6.64	3.39	3.85	26.85
22400	28	4.66	9.47	5.94	17.86	5.89	9.96	5.06	6.47	3.35	3.86	27.48
23200	29	4.72	9.56	5.97	17.25	5.96	9.97	4.97	6.30	3.32	3.87	28.11
24000	30	4.78	9.64	6.00	16.66	6.03	9.96	4.88	6.14	3.30	3.88	28.73
24800	31	4.83	9.71	6.02	16.11	6.09	9.95	4.78	5.99	3.28	3.89	29.35
25600	32	4.89	9.77	6.03	15.59	6.16	9.93	4.69	5.84	3.26	3.89	29.95
26400	33	4.94	9.82	6.04	15.10	6.22	9.90	4.59	5.70	3.24	3.89	30.56

SENSITIVITY OF TOTAL LWR FISSION-PRODUCT ABSORPTION RATE TO THERMAL AND EPITHERMAL (RESONANCE) CROSS SECTIONS

Nuclide	← 5 Gwd/MT →		← 33 Gwd/MT →		Nuclide	← 5 Gwd/MT →		← 33 Gwd/MT →	
	(n,γ) Epithermal	(n,γ) Thermal	(n,γ) Epithermal	(n,γ) Thermal		(n,γ) Epithermal	(n,γ) Thermal	(n,γ) Epithermal	(n,γ) Thermal
Br-81	---	.00001	---	.00004	La-139	.00089	.00156	.00185	.00418
Kr-82	~0	~0	.00003	.00002	Ce-140	.00001	.00009	.00005	.00027
Kr-83	.00078	.00268	.00106	.00363	Ce-141	.00032	.00124	.00013	.00061
Sr-90	---	.00013	---	.00031	Pr-141	.00077	.00133	.00279	.00478
Zr-91	.00018	.00009	.00072	.00036	Ce-142	---	.00018	---	.00064
Zr-92	---	.00004	---	.00010	Pr-142	~0	~0	~0	~0
Zr-93	.00183	.00041	.00479	.00103	Nd-142	---	~0	---	.00020
Mo-95	.00224	.00081	.01431	.00619	Ce-143	.00003	.00001	.00001	~0
Mo-96	---	~0	---	.00002	Pr-143	.00104	.00162	.00020	.00029
Mo-97	.00068	.00036	.00244	.00097	Nd-143	.00491	.04026	.00672	.05337
Mo-98	.00039	.00002	.00126	.00007	Ce-144	---	.00012	---	.00015
Tc-99	.01562	.00307	.03628	.00706	Nd-144	.00006	.00012	.00080	.00153
Mo-100	.00026	.00003	.00074	.00011	Nd-145	.00737	.00432	.01560	.00917
Ru-100	---	.00001	---	.00036	Nd-146	.00009	.00013	.00049	.00072
Ru-101	.00421	.00044	.01189	.00122	Nd-147	.00084	.00021	~0	.00008
Ru-102	.00014	.00018	.00060	.00070	Pm-147	.04776	.01391	.03069	.00927
Ru-103	.00073	.00024	.00026	.00009	Sm-147	.00084	.00017	.00467	.00123
Rh-103	.01709	.01029	.03488	.02122	Nd-148	.00087	.00023	.00227	.00078
Ru-104	.00016	.00003	.00066	.00013	Pm-148	.00281	.00163	.00586	.00366
Rh-105	.00128	.00498	.00104	.00409	Pm-148m	.00022	.00697	.00021	.00649
Pd-105	.00114	.00064	.00627	.00250	Sm-148	.00006	.00001	.00107	.00029
Pd-106	.00001	~0	.00022	.00002	Pm-149	.00001	.00006	.00001	.00006
Pd-107	.00028	.00012	.00222	.00098	Sm-149	.00001	.00106	~0	.00028
Pd-108	.00064	.00009	.00548	.00094	Nd-150	.00019	.00004	.00112	.00023
Ag-109	.00153	.00036	.00718	.00167	Sm-150	.00181	.00478	.01118	.02939
Cd-112	~0	~0	.00003	.00002	Pm-151	.00001	.00002	~0	.00006
Cd-113	~0	.00004	~0	~0	Sm-151	.00178	.01462	.00039	.00330
Te-128	.00001	~0	.00003	.00001	Sm-152	.01242	.00299	.02945	.00735
I-129	.00022	.00060	.00081	.00183	Sm-153	.00010	.00004	~0	.00003
Xe-130	---	~0	---	.00003	Eu-153	.00318	.00286	.02056	.01721
Xe-131	.01908	.00612	.02766	.00906	Sm-154	.00006	.00002	.00028	.00012
Xe-132	.00006	.00006	.00026	.00022	Eu-154	.00043	.00064	.00712	.00822
Xe-133	---	.00156	---	.00050	Gd-154	~0	~0	.00014	.00015
Ce-133	.01861	.00612	.04667	.01298	Eu-155	.00084	.00193	.00038	.00290
Xe-134	.00006	.00009	.00014	.00028	Gd-155	~0	.00002	~0	.00001
Ce-134	.00016	.00032	.00268	.00542	Eu-156	~0	~0	~0	~0
I-135	~0	~0	~0	~0	Gd-156	.00006	~0	.00168	.00006
Xe-135	.00008	.03883	.00001	.02363	Eu-157	~0	~0	~0	~0
Ce-135	.00073	.00046	.00182	.00114	Gd-157	~0	~0	~0	.00001

Listed values are the fractional change in the aggregate total absorption rate (summed over all energies) per unit change in the thermal or epithermal cross section for the listed nuclides.

TABLE A-V

ORDERED LIST OF THE FIRST 15 MOST IMPORTANT
CROSS SECTION SENSITIVITIES AT 5 AND 33 GWd/MT

← 5 GWd/MT →			← 33 GWd/MT →		
Nuclide	X-sec type	Sensitivity	Nuclide	X-sec type	Sensitivity
Xe-135	Thermal	.13693	Nd-143	Thermal	.05337
Pm-147	Epithermal	.04775	Cs-133	Epithermal	.04657
Nd-147	Thermal	.04025	Tc-99	Epithermal	.03528
Xe-131	Epithermal	.01908	Rh-103	Epithermal	.03488
Cs-133	Epithermal	.01851	Pm-147	Epithermal	.03089
Rh-103	Epithermal	.01709	Sm-152	Epithermal	.02945
Tc-99	Epithermal	.01552	Sm-150	Thermal	.02839
Sn-151	Thermal	.01482	Xe-131	Epithermal	.02785
Pm-147	Thermal	.01391	Xe-135	Thermal	.02363
Sm-152	Epithermal	.01242	Rh-103	Thermal	.02122
Rh-103	Thermal	.01029	Eu-153	Epithermal	.02055
Nd-145	Epithermal	.00737	Eu-153	Thermal	.01721
Pm-148m	Thermal	.00697	Nd-145	Epithermal	.01580
Xe-131	Thermal	.00612	Mo-95	Epithermal	.01431
Cs-133	Thermal	.00512	Cs-133	Thermal	.01298

TABLE A

ORDERED LIST OF THE FIRST 15 MOST IMPORTANT
CROSS SECTION SENSITIVITIES AT 5 AND 33 GWd/MT

Nuclide	X-sec type	Sensitivity	Nuclide	X-sec type	Sensitivity
Xe-135	Thermal	.13693	Nd-143	Thermal	.05337
Pm-147	Epithermal	.04775	Cs-133	Epithermal	.04657
Nd-147	Thermal	.04025	Tc-99	Epithermal	.03528
Xe-131	Epithermal	.01908	Rh-103	Epithermal	.03488

TABLE A-VI

SUMMARY OF MAJOR FISSION-PRODUCT TOTAL ABSORBERS AT 4000 HOURS

NUCLIDE	PERCENT OF TOTAL ABSORPTION (GROUPS 1-4)	PERCENT OF THERMAL ABSORPTION (GROUP 4)	PERCENT OF EPITHERMAL ABSORPTION (GROUP 3)	PERCENT OF RESONANCE ABSORPTION (GROUPS 1-3)
XE135	51.66877	62.25804	.11979	.11758
SM149	13.64641	16.28261	.82427	.81274
SM151	4.87625	5.28924	2.89994	2.86568
ND143	4.72617	5.06554	3.05532	3.07406
PM147	3.84727	1.03728	17.74753	17.52700
RH103	2.88576	1.29927	10.71796	10.60918
XE131	2.64281	.77094	11.94299	11.75554
CS133	2.38660	.61488	11.04344	11.01176
TC 99	1.91200	.37562	9.41367	9.39149
SM152	1.58060	.36688	7.61827	7.48928
PM148M	1.26102	1.47567	.21755	.21607
ND145	1.19667	.52806	4.47278	4.45166
RH105	.65107	.62619	.78595	.77220
EU153	.53709	.29189	1.73731	1.73080
RU101	.49444	.05294	2.52978	2.64382
SM150	.41197	.35620	.66846	.68346
EU155	.38497	.41294	.24903	.24883
KR 83	.34854	.32251	.46944	.47526
MO 95	.31201	.09792	1.34417	1.35424
PR143	.27672	.19653	.66838	.66711
XE133	.25333	.19356	.55340	.54433
PM148	.24959	.11543	.91820	.90268
CD113	.24338	.29297	.00195	.00195
ZR 93	.23937	.04997	1.15644	1.16140
LA139	.22821	.18864	.41419	.42088
PR141	.21542	.15979	.46206	.48621
AG109	.19883	.04410	.96633	.95210
GD157	.18819	.22641	.00215	.00213
PD105	.18198	.06566	.68771	.74826
CE141	.15961	.15063	.19272	.20334
ND147	.14174	.03452	.67001	.66372
MO 97	.13916	.04204	.52592	.61199
CS135	.11925	.05473	.43583	.43334
PM149	.10907	.11192	.09292	.09522
RU103	.10723	.03021	.46006	.48218
ALL OTHERS	1.17850	.51830	3.93406	4.39249

TABLE A-VII

SUMMARY OF MAJOR FISSION-PRODUCT TOTAL ABSORBERS AT 26400 HOURS

NUCLIDE	PERCENT OF TOTAL ABSORPTION (GROUPS 1-4)	PERCENT OF THERMAL ABSORPTION (GROUP 4)	PERCENT OF EPITHERMAL ABSORPTION (GROUP 3)	PERCENT OF RESONANCE ABSORPTION (GROUPS 1-3)
XE135	15.09836	26.77836	.01371	.01342
ND143	9.90337	15.62377	2.50825	2.51536
RH103	9.81659	6.50558	14.28416	14.09282
CS133	6.21505	2.35690	11.26709	11.19792
XE131	6.03893	2.59298	10.69177	10.48944
SM149	5.69759	10.00653	.13483	.13251
TC 99	4.93791	1.42788	9.52478	9.47118
PM147	4.59155	1.82217	8.29825	8.16826
SM152	3.89481	1.33069	7.35464	7.20642
SM151	3.23823	5.17014	.75449	.74313
EU153	3.21177	2.56920	4.07021	4.04165
ND145	2.88226	1.87209	4.22063	4.18692
EU154	2.23714	2.19512	2.31606	2.29140
EU155	2.18896	3.45605	.55475	.55249
MO 95	2.09987	.97005	3.54423	3.55906
PM148M	1.83657	3.16345	.12413	.12288
SM150	1.54143	1.96175	.97988	.99859
AG109	1.44728	.47249	2.75572	2.70625
RU101	1.42170	.22404	2.84980	2.96850
CS134	.91132	1.08947	.68118	.68123
PD105	.87408	.46420	1.29412	1.40344
SM147	.79702	.28738	1.47167	1.45524
PR141	.77187	.84276	.64865	.68031
KR 83	.66512	.90590	.35097	.35416
LA139	.61911	.75325	.44022	.44586
ZR 93	.60368	.18550	1.14261	1.14376
RH105	.58694	.83091	.27759	.27184
PD108	.51431	.13153	1.02206	1.00867
PM148	.49921	.33984	.71952	.70504
MO 97	.38887	.17290	.57576	.66779
PD107	.32190	.16308	.49077	.52702
CS135	.30709	.20745	.43972	.43578
I129	.28230	.33702	.19489	.21162
ND144	.22832	.24854	.17046	.20221
GD157	.21670	.38375	.00097	.00096
ND148	.14303	.05925	.22772	.25123
CD113	.13790	.24433	.00043	.00043
MO 98	.13324	.01024	.25971	.29209
RU102	.12987	.09197	.10409	.17882
ZR 91	.11696	.06208	.17006	.18784
XE133	.11005	.12376	.09418	.09234
PR143	.10586	.11066	.10017	.09965
EU156	.10472	.11108	.09860	.09651
ALL OTHERS	2.13118	1.34390	2.77646	3.14797