27
25 for NTIS

MASTER

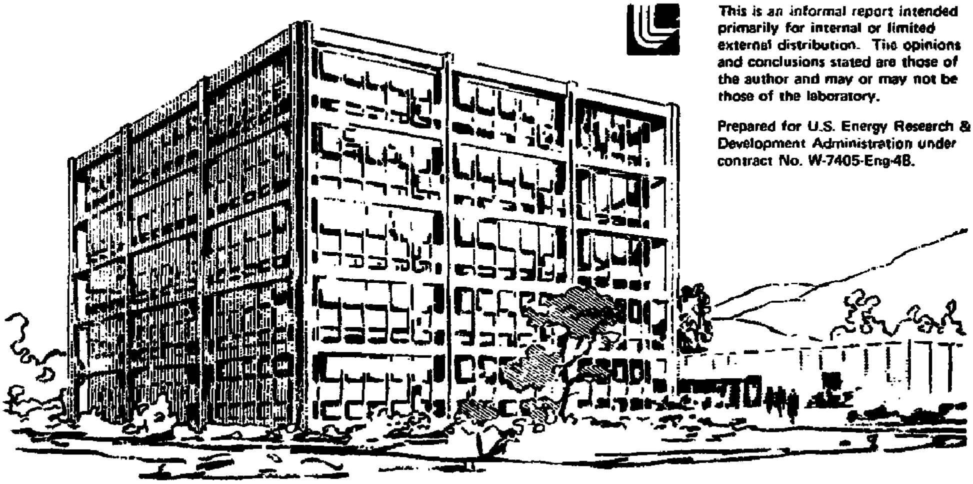
UCID- 17370

Lawrence Livermore Laboratory

MEASUREMENT OF THE NEUTRON-INDUCED
FISSION CROSS SECTION OF ^{237}Np RELATIVE TO ^{235}U
FROM 0.02 TO 30 MeV

J. W. Behrens, J. W. Magana, and J. C. Browne

January 19, 1977



This is an informal report intended primarily for internal or limited external distribution. The opinions and conclusions stated are those of the author and may or may not be those of the laboratory.

Prepared for U.S. Energy Research & Development Administration under contract No. W-7405-Eng-48.

MEASUREMENT OF THE NEUTRON-INDUCED FISSION CROSS SECTION OF ^{237}Np RELATIVE TO ^{235}U FROM 0.02 TO 30 MeV

ABSTRACT

Continuing our fission cross-section ratio studies at Lawrence Livermore Laboratory, we have measured the $^{237}Np/^{235}U$ fission cross-section ratio from 0.02 to 30 MeV. Using the threshold method, we obtained a value of 1.294 ± 0.019 for the average cross-section ratio in the interval from 1.75 to 4.00 MeV.

PRELIMINARY RESULTS

We measured the fission cross section of ^{237}Np relative to that of ^{235}U, using ionization fission chambers at the Lawrence Livermore Laboratory's 100-MeV electron linear accelerator. The time-of-flight technique was used to measure the cross-section ratio as a function of neutron energy over the energy range from 0.001 to 30 MeV. Using the threshold method,[1,2] we obtained a value of 1.294 ± 0.019 for the average cross-section ratio in the interval from 1.75 to 4.00 MeV. We conducted the measurement at the 15.7-meter time-of-flight station.

Figures 1-4 show and Table 1 lists our preliminary data for the $^{237}Np/^{235}U$ ratio from 0.02 to 30 MeV. The lines shown in Figs. 1-3 were obtained by using files of evaluated fission cross sections. The ENDL[3] evaluation is shown in Fig. 1, and the ENDF/B-IV[4] evaluation is shown in Figs. 2 and 3. In Figs. 3 and 4, our data are compared with the measurements of White *et al.*,[5] White and Warner,[6] and Stein *et al.*[7]

The Cross Section Evaluation Working Group (CSEWG), responsible for the upcoming ENDF/B-V evaluations, requested this brief report. We plan a more complete and formal presentation of this measurement.

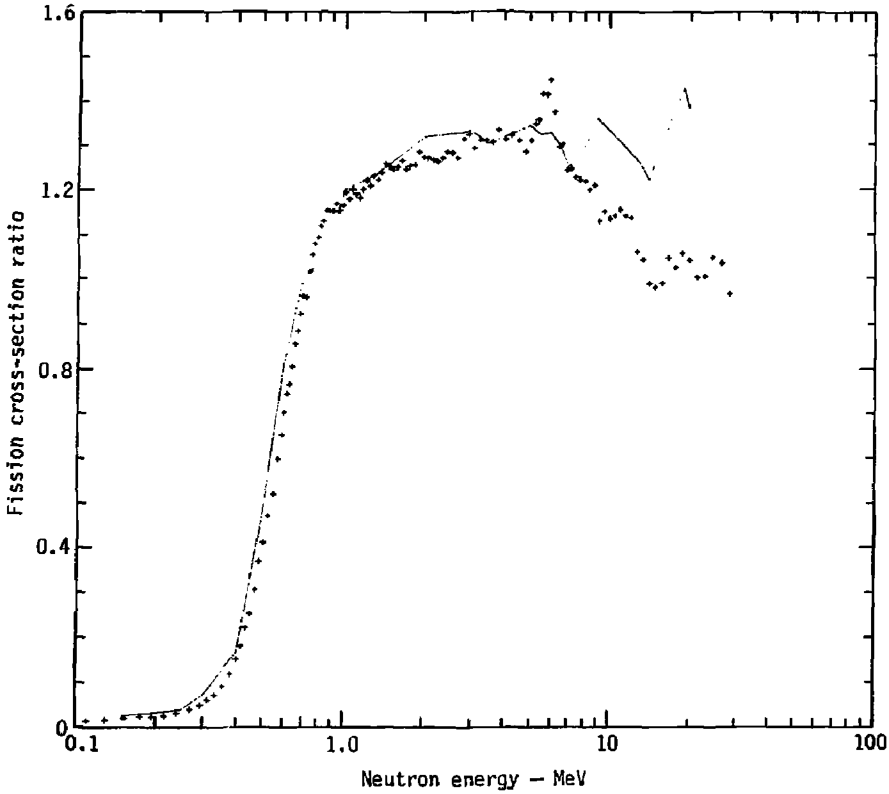


Fig. 1. Ratio of the ^{237}Np to ^{235}U fission cross sections in the energy range from 0.1 to 30 MeV. Our preliminary work is given by +. The line denotes the $^{237}Np/^{235}U$ ratio obtained by using the ENDL evaluated fission cross-section files.

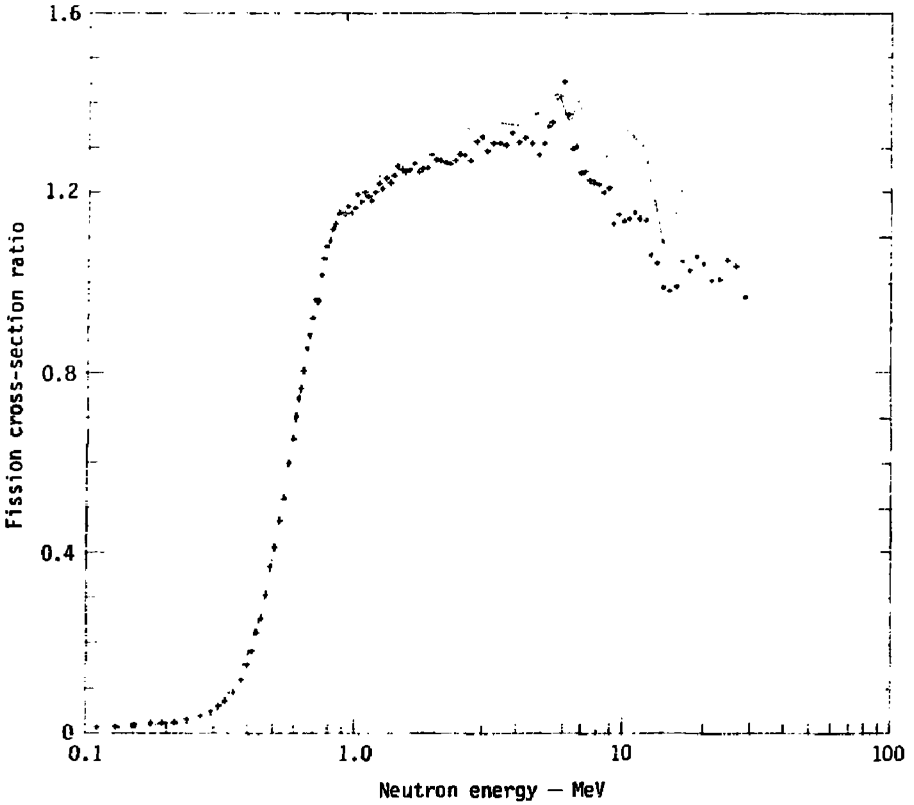


Fig. 2. Ratio of the ^{237}Np to ^{235}U fission cross sections in the energy range from 0.1 to 30 MeV. Our preliminary work is given by +. The line denotes the $^{237}Np/^{235}U$ ratio obtained by using the ENDF/B-IV evaluated fission cross-section files.

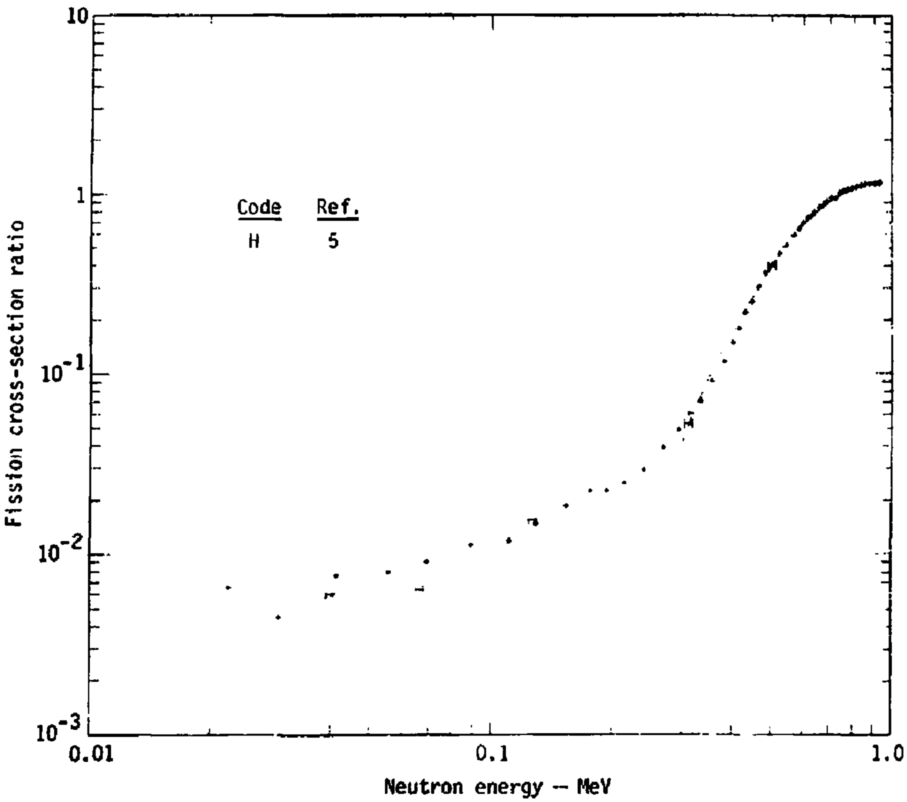


Fig. 3. Ratio of the ^{237}Np to ^{235}U fission cross sections in the energy range from 0.02 to 0.9 MeV. Our preliminary work is given by +. A letter code indicates the work of others. The line denotes the ^{237}Np/^{235}U ratio obtained by using the ENDF/B-IV evaluated fission cross-section files.

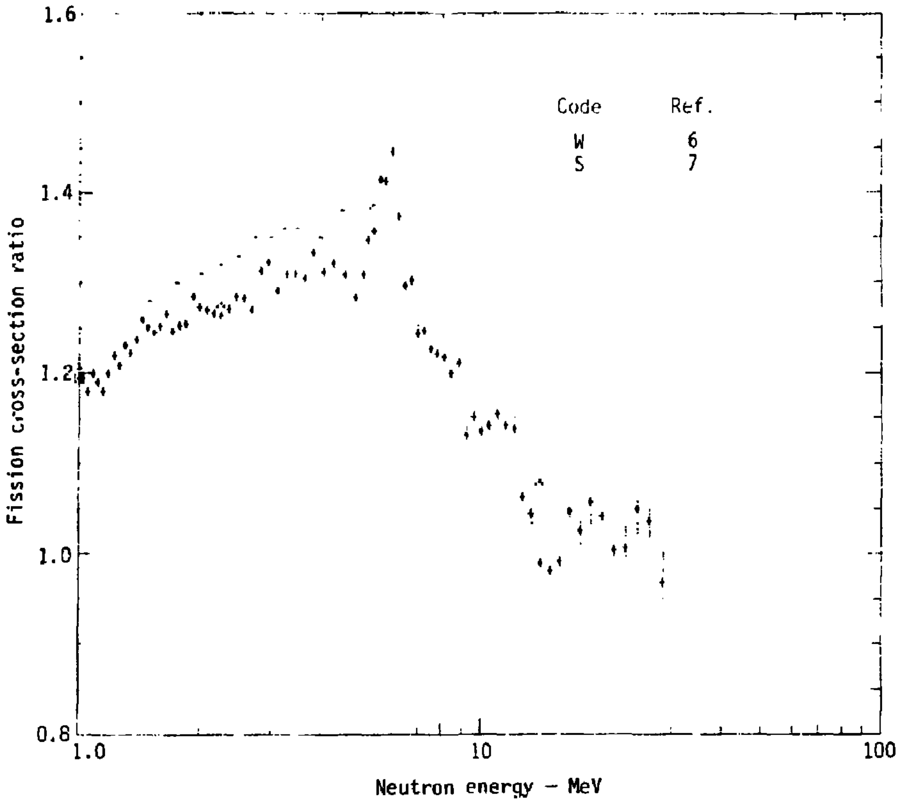


Fig. 4. Ratio of the ^{237}Np to ^{235}U fission cross sections in the energy range from 1 to 30 MeV. Our preliminary work is given by +. The statistical error bars, representing one standard deviation, are shown for each point. A letter code indicates the work of others.

Table 1. Fission cross-section ratio of ^{237}Np to ^{235}U.

Neutron energy (MeV)	Ratio	Statistical uncertainty[a] (%)	Neutron energy (MeV)	Ratio	Statistical uncertainty[a] (%)
0.02215	0.0066	±18.2	0.6765	0.8829	0.9
0.02955	0.0045	23.7	0.6925	0.9194	0.9
0.04139	0.0076	11.7	0.7090	0.9616	0.8
0.05565	0.0080	11.1	0.7261	0.9577	0.9
0.06969	0.0092	9.1	0.7439	1.017	0.8
0.08980	0.0113	6.9	0.7623	1.053	0.8
0.1112	0.0120	7.4	0.7815	1.078	0.8
0.1300	0.0148	5.5	0.8013	1.092	0.8
0.1540	0.0187	4.6	0.8219	1.118	0.8
0.1767	0.0226	4.9	0.8434	1.129	0.7
0.1948	0.0226	4.6	0.8656	1.153	0.7
0.2157	0.0250	4.1	0.8888	1.153	0.7
0.2402	0.0294	3.4	0.9129	1.152	0.7
0.2691	0.0389	2.7	0.9380	1.169	0.7
0.2944	0.0489	3.1	0.9642	1.154	0.7
0.3132	0.0606	2.6	0.9915	1.164	0.7
0.3340	0.0706	2.2	1.020	1.194	0.7
0.3569	0.0919	2.1	1.050	1.179	0.7
0.3822	0.1181	1.6	1.081	1.199	0.6
0.4030	0.1510	2.0	1.113	1.189	0.6
0.4178	0.1802	1.9	1.147	1.180	0.6
0.4334	0.2223	1.7	1.182	1.199	0.6
0.4500	0.2531	1.5	1.219	1.219	0.6
0.4675	0.3070	1.2	1.258	1.209	0.6
0.4861	0.3592	1.1	1.299	1.231	0.6
0.5058	0.4121	1.0	1.342	1.222	0.6
0.5267	0.4704	1.0	1.387	1.237	0.6
0.5489	0.5188	0.9	1.434	1.259	0.6
0.5726	0.5983	0.8	1.484	1.251	0.6
0.5913	0.6508	1.1	1.536	1.244	0.6
0.6044	0.7006	1.0	1.591	1.252	0.6
0.6178	0.7422	1.0	1.650	1.265	0.7
0.6318	0.7637	1.0	1.711	1.245	0.7
0.6462	0.8030	1.0	1.776	1.252	0.7
0.6611	0.8536	0.9	1.845	1.254	0.7

Table 1 (cont.)

Neutron energy (MeV)	Ratio	Statistical uncertainty[a] (%)	Neutron energy (MeV)	Ratio	Statistical uncertainty[a] (%)
1.918	1.285	0.7	7.214	1.247	1.1
1.995	1.273	0.7	7.497	1.227	1.1
2.077	1.270	0.7	7.796	1.221	1.1
2.164	1.266	0.7	8.114	1.218	1.1
2.257	1.263	0.8	8.453	1.199	1.1
2.356	1.271	0.8	8.812	1.210	1.2
2.461	1.285	0.8	9.196	1.130	1.3
2.574	1.282	0.8	9.605	1.150	1.3
2.695	1.270	0.8	10.04	1.136	1.4
2.824	1.313	0.9	10.51	1.141	1.5
2.963	1.323	0.9	11.01	1.155	1.6
3.112	1.291	0.9	11.55	1.142	1.7
3.274	1.309	0.9	12.13	1.138	1.7
3.448	1.310	1.0	12.75	1.061	1.8
3.636	1.305	1.0	13.43	1.044	1.8
3.840	1.333	1.0	14.16	0.9897	1.9
4.062	1.312	1.0	14.95	0.9817	2.0
4.304	1.321	1.0	15.81	0.9908	2.1
4.568	1.309	1.0	16.74	1.047	2.2
4.857	1.284	1.0	17.77	1.025	2.3
5.093	1.309	1.3	18.89	1.057	2.5
5.259	1.348	1.3	20.12	1.041	2.7
5.434	1.357	1.3	21.48	1.004	2.8
5.618	1.415	1.3	22.98	1.006	3.0
5.812	1.413	1.2	24.65	1.049	3.2
6.015	1.446	1.2	26.51	1.036	3.4
6.230	1.374	1.1	28.59	0.9681	3.7
6.456	1.296	1.1	30.93	1.047	3.8
6.695	1.302	1.1	33.57	1.028	4.1
6.947	1.244	1.1			

[a]This indicates a counting error expressed as one standard deviation. Total errors may be estimated by combining the normalization error of 1.5% and the estimated overall systematic error of 1% with the counting errors in the table.

REFERENCES

1. J. W. Behrens, G. W. Carlson, and R. W. Bauer, "Neutron-Induced Fission Cross Sections of ^{233}U, ^{234}U, ^{236}U, and ^{238}U with respect to ^{235}U," in [illegible] (National Bureau of Standards, Washington, D.C., 1975), vol. 2, p. 591.
2. J. W. Behrens and G. W. Carlson, "Measurements of Neutron-Induced Fission Cross-Section Ratios Involving Isotopes of Uranium and Plutonium," in [illegible], Argonne National Laboratory, Ill., Rept. ANL-76-90 (1976), p. 47.
3. Evaluated Nuclear Data Library, August 1976. This evaluation originates at the Lawrence Livermore Laboratory.
4. Evaluated Nuclear Data File/Format B - Version IV. This evaluation originates at the Brookhaven National Laboratory, Upton, N.Y.
5. P. H. White, J. G. Hodgkinson, and G. J. Wall, in [illegible] *and Chemistry of* [illegible] (International Atomic Energy Agency, Vienna, 1965), vol. 1, p. 219.
6. P. H. White and G. P. Warner, [illegible], 671, (1967).
7. W. E. Stein, R. K. Smith, and H. L. Smith, in [illegible] *Section Techn.* (U.S. Government Printing Office, Washington, D.C., 1968). vol. 2, p. 627.

JWB/ck/mla

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161
Price: Printed Copy $: Microfiche $3.00

Page Range	Domestic Price	Page Range	Domestic Price
001–025	$ 3.50	326-350	10.00
026–050	4.00	351-375	10.50
051–075	4.50	376-400	10.75
076–100	5.00	401–425	11.00
101–125	5.50	426-450	11.75
126–150	6.00	451-475	12.00
151–175	6.75	476-500	12.50
176–200	7.50	501–525	12.75
201–225	7.75	526-550	13.00
226–250	8.00	551-575	13.50
251–275	9.00	576-600	13.75
276–300	9.25	601-up	*
301–325	9.75		

*Add $2.50 for each additional 100 page increment from 601 to 1,000 pages; add $4.50 for each additional 100 page increment over 1,000 pages.