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TITLE: DELAYED NEUTRON SPECTRA BY DECAY GROUP FOR FISSIONING SYSTEMS FROM ^{227}TH THROUGH ^{255}FM

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DELAYED NEUTRON SPECTRA BY DECAY GROUP FOR FISSIONING SYSTEMS FROM ^{227}Th THROUGH ^{255}Fm

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I. INTRODUCTION

Most applications of delayed neutrons use an approximate temporal group representation of measured aggregate data.[1] Such data have been limited to the few fissioning nuclides that have aggregate measurements, and even these have inadequate or no spectral measurements.

Improvements in the experimental techniques of isotope separation and neutron spectroscopy have made the study of delayed neutron emission from individual precursor nuclides more practical and productive over the past fifteen or so years. The quantity and quality of the delayed neutron emission probabilities and particularly the neutron emission spectra for the individual nuclides have been greatly inproved.

This paper will briefly review a recently completed evaluation of precursor data that comprises the largest single set of such data to date, and will also describe how the precursor data has been used to produce delayed neutron yields, halflives, and spectra in the classical six-group representation for 43 fissioning systems from ^{227}Th to ^{255}Fm. Comments and observations made concerning the use of more than six time groups will also be included. The application of the data in both its explicit and reduced (six temporal group representation) forms in the point reactor kinetics equations will also be discussed. Results from beta-effective calculations in a simple Godiva-type system will be presented, but the paper will concentrate on the data base and its few-group representations.

II. PRECURSOR DATA BASE

Based on energetics, approximately 271 fission products should be delayed neutron precursors. Only a brief description of the types and sources of data for precursors, including fission-product yields, can be given here, along with a summary of the relative importance of the experimental data *vs* that provided by various model calculations.

Fission-product yields are based on a preliminary evalution for ENDF/B-VI.[2] This comprises data at one or more neutron incident energies [denoted as thermal (T), fast (F), and high (H) and for spontaneous fission (S)] for 34 fissioning nuclides. Forty-three cases are included in this paper for 28 fissioning nuclides. Most of the yields are based on models.

Emission probabilities (Pn values) for 85 precursors have been measured[3] and evaluated.[4] The evaluation also proides a fit to the parameters in the systematic Herrmann-Kratz equation[5] used to preduct the unmeasured Pn's.

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Spectra for 34 precursors have been measured.[6-8] Thirty of these were found to be inadequate in the measured energy range and had to be supplemented with nuclear models.[9,10] The same models were used to estimate the spectra for the remaining 237 precursors. The simple count of nuclides having measured data is misleading. The 85 having measured Pn values account for 80% or more of the total emission rate and the 34 having measured spectra account for 67% or more of the total. These contributions depend on the fissioning nuclide; e.g., for ^{235}U thermal fission, the respective contributions are 96% and 84%.

The largest effort in the evaluation was directed at model estimates of unmeasured spectra and the expansion of the incomplete measured spectra. This effort and the models are summarized in Refs. 9 and 11; it will be described in more detail in the final paper and in complete detail when Ref. 12 is published.

III. REDUCTION OF DATA INTO DECAY AND SPECTRAL GROUPS

The use of a few temporal groups to represent the behavior of a large, unknown number of precursors started with aggregate experiments. It is still a convenient approximation for use in applications and can be duplicated from aggregate calculations of the individual precursors.

The fission product depletion code, CINDER-10, was used to calculate the activities of all precursor nuclides for various cooling times (to 300 seconds) following a prompt irradiation in each of the fissioning systems. These nuclide activities were folded in with the evaluated emission probabilities to produce aggregate delayed neutron emission values. A nonlinear least-squares analysis was performed on these data to produce a sum of exponentials representing the delayed neutron groups.

The initial calculations for $^{235}U(F)$, $^{238}U(F)$, and $^{239}Pu(F)$ were performed using three, six, nine and twelve groups. Increasing the number of groups from six to nine resulted in a significant improvement in the fit; however, the results from point kinetics calculations using both the six- and nine-group fits for prompt changes in reactivity did not reveal any significant differences. [12,13]

Based on these results and the general acceptance of a six-group representation, the fits for the remaining 40 fissioning systems were performed only for six-groups. Table I presents the normalized group abundances and decay constants for all 43 fissioning systems.

Having determined the six-group parameters for each fissioning nuclide, the next logical step was to calculate a consistent set of six-group spectra. Rather than assign each precursor to a particular group based on halflife bounds,[6,14] the individual precursors were allowed to contribute delayed neutrons to the two groups whose decay constants were closest to that of the precursor nuclide. The fraction contributed to each of the adjacent groups was determined by minimizing the error introduced by the few-group approximation relative to the explicit precursor notation.[13,15] These fractions, Pn values, and fission yields can be applied to any number of available energy groups to produce the six temporal group spectra. Currently, we use a constant energy bin width of 10 keV for up to 300 groups. In some applications[11] we have used the same bin structure to > 8.5 MeV, but these calculations did not use temporal groups.

IV. APPLICATIONS OF THE GROUP DATA

The accuracy of the six-group parameters is difficult to quantify as it is influenced by not only the uncertainties included in the basic data that was used in calculating the delayed neutron activity curves (i.e., the direct fission yields, halflives, and emission probabilities) but also by the uncertainty introduced by the least-squares fit to that data. Likewise, the calculation of the uncertainties for the group spectra is not straightforward because of the method used to calculate the fractional contribution from each precursor to the various groups.

A reasonable check on the group abundances and decay constants would be to use them in a point kinetics calculation. These calculations were performed for step changes in reactivity in a $^{235}U(F)$ system and the results are given in Fig. 1. The point kinetics equations were modified[13,16] for the calculation using the explicit precursor data. A total of 386 nuclides were required in that calculation to include the 271 precursors and their parents. Agreement of the fitted six-groups with the explicit calculations is very good. The results using the ENDF/B-V six-group parameters is also given for comparison; the number of delayed neutrons produced is constant for all cases.

Rossi-alpha (β_{eff}/generation time) calculations for Godiva were also made using ENDF/B-V, the new six-group spectra and total spectra using 80 energy groups. Results will be presented.

V. SUMMARY

Six-group parameters and spectra have been calculated for 43 fissioning systems that are consistent with the explicit precursor results, although some disagreement with the ENDF/B-V is observed. The major improvement has been in the delayed neutron group spectra and data produced for unmeasured systems. ENDF/B-V contains six-group spectra for only seven fissioning nuclides and those are in a very coarse bin structure that extends only to about 1.2 MeV, whereas the present group spectra cover 28 fissioning nuclides in a fine 10-keV energy bin structure and extend to 3.0 MeV (the maximum range of the experimental data for any precursor). The normalized spectra, and the group constants to a lesser degree, appear to be nearly independent of the incident neutron energy and therefore the ν_d data recommended for inclusion in ENDF/B-VI contains only one set of group constants and spectra (usually that of the fast system) for each of the 28 fissioning nuclides. The delayed neutron yields are a function of incident neutron energy, especially for high energy (14-MeV) fission and are given as they were in ENDF/B-V. The actual values for ν_d recommended for ENDF/B-VI, but not the abundances or spectra, are those taken from the previous ENDF/B-V evaluation[17] or newly evaluated measurements. Nuclides with no reported measurement of ν_d in the literature were assigned the calculated values.

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TABLE I

DELAYED NEUTRON SIX-GROUP PARAMETERS

Fission Nuclide		Group 1	2	3	4	5	6
Th227T	alpha	0.1027	0.2182	0.1304	0.3555	0.1647	0.0284
	lambda	0.0128	0.0354	0.1098	0.2677	0.5022	2.0956
Th229T	alpha	0.0867	0.1907	0.1297	0.3887	0.1729	0.0312
	lambda	0.0128	0.0350	0.1123	0.2760	0.4950	2.0456
Th232F	alpha	0.0364	0.1259	0.1501	0.4406	0.1663	0.0808
	lambda	0.0131	0.0350	0.1272	0.3287	0.9100	2.8203
Th232H	alpha	0.0326	0.0997	0.1431	0.5062	0.1336	0.0848
	lambda	0.0130	0.0350	0.1307	0.3274	0.9638	3.1667
Pa231F	alpha	0.0826	0.2230	0.1608	0.3885	0.1050	0.0401
	lambda	0.0129	0.0347	0.1150	0.2856	0.6706	2.3111
U232T	alpha	0.1360	0.2745	0.1509	0.3052	0.1007	0.0326
	lambda	0.0128	0.0350	0.1073	0.2557	0.6626	2.0254
U233T	alpha	0.0674	0.1927	0.1383	0.2798	0.1128	0.2091
	lambda	0.0129	0.0333	0.1163	0.2933	0.7943	2.3751
U233F	alpha	0.0859	0.2292	0.1781	0.3516	0.1142	0.0409
	lambda	0.0129	0.0347	0.1193	0.2862	0.7877	2.4417
U233H	alpha	0.0900	0.2007	0.1912	0.3684	0.1090	0.0405
	lambda	0.0128	0.0378	0.1271	0.2981	0.8543	2.5314
U234F	alpha	0.0550	0.1964	0.1803	0.3877	0.1324	0.0482
	lambda	0.0131	0.0337	0.1210	0.2952	0.8136	2.5721
U234H	alpha	0.0808	0.1880	0.1791	0.3888	0.1212	0.0420
	lambda	0.0128	0.0364	0.1256	0.2981	0.8475	2.5696
U235T	alpha	0.0380	0.1918	0.1638	0.3431	0.1744	0.0890
	lambda	0.0133	0.0325	0.1219	0.3169	0.9886	2.9544
U235F	alpha	0.0350	0.1807	0.1725	0.3868	0.1586	0.0664
	lambda	0.0133	0.0327	0.1208	0.3028	0.8495	2.8530
U235H	alpha	0.0458	0.1688	0.1769	0.4079	0.1411	0.0595
	lambda	0.0131	0.0356	0.1246	0.2962	0.8260	2.6575
U236F	alpha	0.0302	0.1722	0.1619	0.3841	0.1775	0.0741
	lambda	0.0134	0.0322	0.1202	0.3113	0.8794	2.8405
U236H	alpha	0.0438	0.1540	0.1719	0.4018	0.1578	0.0707
	lambda	0.0131	0.0333	0.1252	0.3030	0.8802	2.8167
U237F	alpha	0.0178	0.1477	0.1445	0.3864	0.2095	0.0941
	lambda	0.0138	0.0316	0.1211	0.3162	0.9073	3.0368
U238F	alpha	0.0139	0.1128	0.1310	0.3851	0.2540	0.1031
	lambda	0.0136	0.0313	0.1233	0.3237	0.9060	3.0487
U238H	alpha	0.0195	0.1184	0.1490	0.3978	0.2081	0.1072
	lambda	0.0135	0.0320	0.1214	0.3142	0.9109	3.0196
Np237F	alpha	0.0400	0.2162	0.1558	0.3633	0.1659	0.0589
	lambda	0.0133	0.0316	0.1168	0.3006	0.8667	2.7600
Np237H	alpha	0.0326	0.1571	0.1589	0.3929	0.1789	0.0796
	lambda	0.0133	0.0322	0.1211	0.2933	0.8841	2.7922
Np238F	alpha	0.0216	0.1845	0.1519	0.3760	0.1861	0.0798
	lambda	0.0136	0.0308	0.1189	0.3077	0.8988	2.9676
Pu238F	alpha	0.0377	0.2390	0.1577	0.3562	0.1590	0.0504
	lambda	0.0133	0.0312	0.1162	0.2888	0.8561	2.7138
Pu239T	alpha	0.0306	0.2623	0.1828	0.3283	0.1482	0.0479
	lambda	0.0133	0.0301	0.1135	0.2953	0.8537	2.6224
Pu239F	alpha	0.0363	0.2364	0.1789	0.3267	0.1702	0.0515
	lambda	0.0133	0.0309	0.1134	0.2925	0.8575	2.7297

TABLE I (continued)

Pu239H	alpha	0.0678	0.1847	0.1553	0.3685	0.1750	0.0487
	lambda	0.0129	0.0353	0.1215	0.2885	0.8486	2.5587
Pu240F	alpha	0.0320	0.2529	0.1508	0.3301	0.1795	0.0547
	lambda	0.0133	0.0305	0.1152	0.2974	0.8477	2.8796
Pu240H	alpha	0.0534	0.1812	0.1533	0.3715	0.1849	0.0558
	lambda	0.0130	0.0329	0.1191	0.2918	0.8462	2.7080
Pu241T	alpha	0.0167	0.2404	0.1474	0.3430	0.1898	0.0627
	lambda	0.0137	0.0299	0.1136	0.3078	0.8569	3.0800
Pu241F	alpha	0.0180	0.2243	0.1426	0.3493	0.1976	0.0682
	lambda	0.0136	0.0300	0.1167	0.3069	0.8701	3.0028
Pu242F	alpha	0.0196	0.2314	0.1256	0.3262	0.2255	0.0716
	lambda	0.0136	0.0302	0.1154	0.3042	0.8272	3.1372
Am241T	alpha	0.0305	0.2760	0.1531	0.3122	0.1825	0.0457
	lambda	0.0133	0.0300	0.1145	0.2949	0.8818	2.6879
Am241F	alpha	0.0355	0.2540	0.1563	0.3364	0.1724	0.0454
	lambda	0.0133	0.0308	0.1130	0.2868	0.8654	2.6430
Am241H	alpha	0.0740	0.1757	0.1754	0.3589	0.1783	0.0377
	lambda	0.0129	0.0346	0.1267	0.3051	0.9536	3.3205
Am242mT	alpha	0.0247	0.2659	0.1512	0.3337	0.1756	0.0489
	lambda	0.0135	0.0301	0.1152	0.2994	0.8646	2.8107
Am243F	alpha	0.0234	0.2945	0.1537	0.3148	0.1656	0.0480
	lambda	0.0135	0.0298	0.1138	0.2986	0.8820	2.8111
Cm242F	alpha	0.0763	0.2847	0.1419	0.2833	0.1763	0.0375
	lambda	0.0130	0.0312	0.1129	0.2783	0.8710	2.1969
Cm245T	alpha	0.0222	0.1788	0.1672	0.3706	0.2054	0.0559
	lambda	0.0134	0.0307	0.1130	0.3001	0.8340	2.7686
Cf249T	alpha	0.0246	0.3919	0.1349	0.2598	0.1614	0.0273
	lambda	0.0135	0.0294	0.1053	0.2930	0.8475	2.4698
Cf251T	alpha	0.0055	0.3587	0.1736	0.2693	0.1688	0.0242
	lambda	0.0157	0.0288	0.1077	0.3246	0.8837	2.6314
Cf252S	alpha	0.0124	0.3052	0.1813	0.2992	0.1729	0.0290
	lambda	0.0136	0.0291	0.1068	0.3024	0.8173	2.6159
Es254T	alpha	0.0073	0.3148	0.1547	0.2788	0.2010	0.0435
	lambda	0.0194	0.0289	0.1048	0.3185	0.8332	2.7238
Fm255T	alpha	0.0060	0.4856	0.1766	0.1940	0.1160	0.0218
	lambda	0.0149	0.0287	0.1027	0.3130	0.8072	2.5768

In this table T, F, and H, refer to fission neutron incident energies of thermal, fast, and high energy and S refers to spontaneous fission.

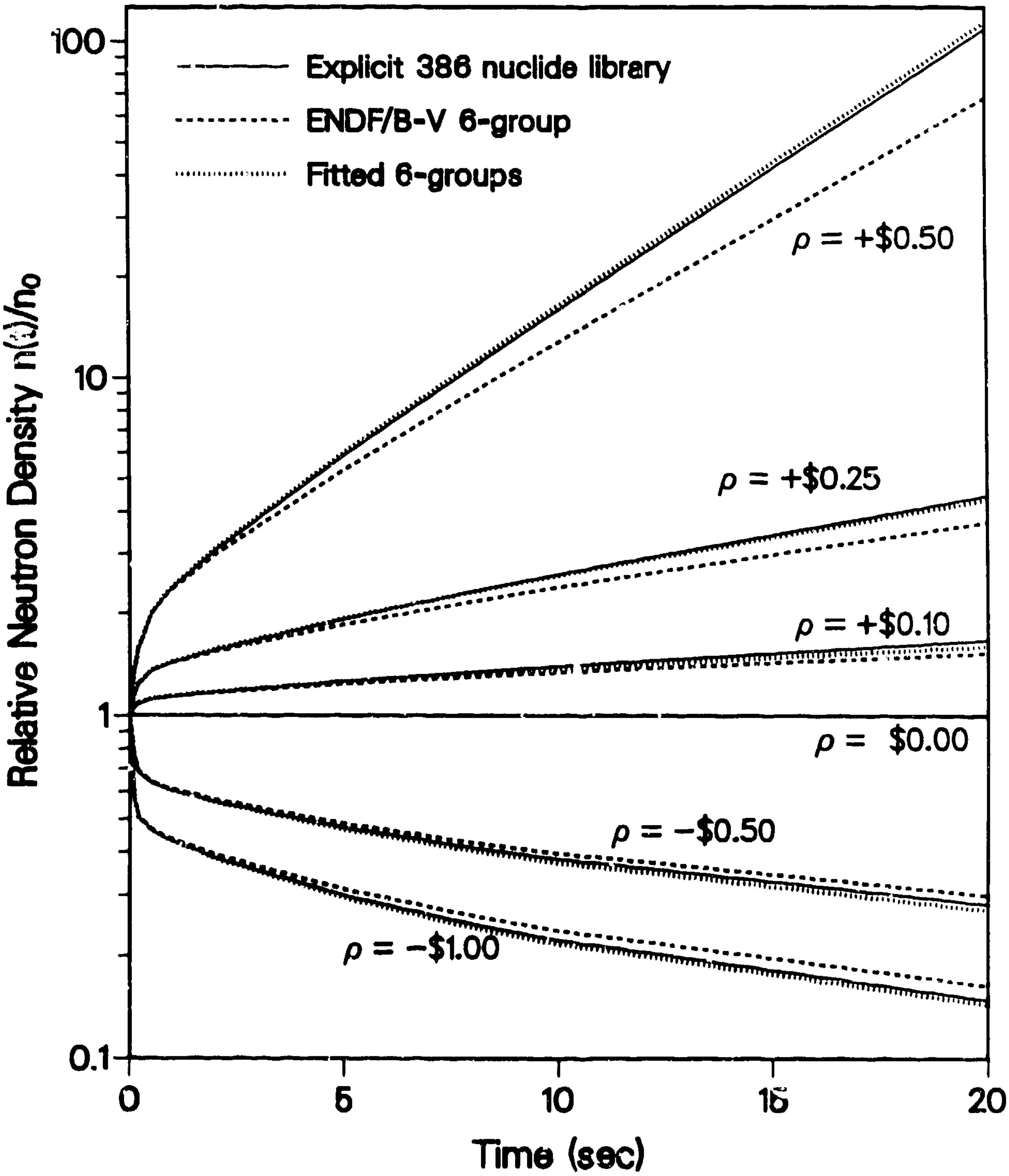

Figure 1. Calculated neutron density following step reactivity (ρ) inputs for ^{235}U fast fission.