A major purpose of the Technical Information Center is to provide the broadest dissemination possible of information contained in DOE's Research and Development Reports to business, industry, the academic community, and federal, state and local governments.

Although a small portion of this report is not reproducible, it is being made available to expedite the availability of information on the research discussed herein.
TITLE: DELAYED NEUTRON SPECTRA BY DECAY GROUP FOR FISSIONING SYSTEMS FROM $^{227}$Th THROUGH $^{255}$Fm

AUTHOR(S): Tal madge R. England, T-2
Michelle C. Brady, ORNL (former T-2 collaborator)


DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

By acceptance of this article the publisher recognizes that the U.S. Government retains a nonexclusive, royalty-free license to publish or reproduce the published form of this contribution or to allow others to do so, for U.S. Government purposes.

The Los Alamos National Laboratory requests that the publisher identify this article as work performed under the auspices of the U.S. Department of Energy.
DELAYED NEUTRON SPECTRA BY DECAY GROUP FOR FISSIONING SYSTEMS FROM $^{227}$Th THROUGH $^{255}$Fm

T. R. England and M. C. Brady*
Los Alamos National Laboratory
Los Alamos, New Mexico, USA

ABSTRACT

The quality and quantity of delayed neutron precursor data has greatly improved over the past approximately 15 years. Supplementation of the data with model calculations and the use of models to extend the number of precursors to 271 is now practical. These data, along with improved fission-product parameters, permit direct calculations of aggregate behavior for many fissioning nuclides. The results can even by approximated using a few (usually six) temporal groups. This paper summarizes an extensive four-year effort to provide a complete set of evaluated data and emphasizes its use to generate the temporal approximations; precursor data and group values are intended for inclusion in ENDF/B-VI.

INTRODUCTION

Most applications of delayed neutrons use an approximate temporal group representation of measured aggregate data. 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 improved. Such data are still inadequate for use in calculations of aggregate behavior from individual precursors. We have evaluated the measured data, supplemented it with model spectra for completeness, and added model values for probable unmeasured precursors. The resulting extensive data base will be incorporated into ENDF/B-VI.

This paper can only briefly describe the data base. We will concentrate on its use to produce aggregate delayed neutron yields, halflives, and spectra in the classical six-group representation for 43 fissioning systems from $^{227}$Th to $^{255}$Fm. Group data are also required in ENDF/B-VI. Comments and observations made concerning the use of more than six time groups are included. The application of the data in both its explicit and reduced (six temporal group representation) forms in the point reactor kinetics equations are also discussed. Results from beta-effective calculations in a simple Godiva-type system are presented, but this paper will concentrate on the precursor data base and emphasize the temporal group representations.

* M. C. Brady, presently with Nuclear Engineering Applications Dept., Oak Ridge National Laboratory.
I'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 evaluation for ENDF/B-VI. 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 and evaluated. The evaluation also provides a fit to the parameters in the systematic Herrmann-Kraitz equation used to predict the unmeasured Pn's.

Spectra for 34 precursors have been measured. Thirty of these were found to be inadequate in the measured energy range and had to be supplemented with nuclear models. The same models were used to estimate the spectra for the remaining 237 precursors. Two models were used. The BETA code was used to extend 30 measured spectra. Otherwise we used a modified evaporation model, described in Ref. 11, because it has the virtue of producing the shape of typical spectra without the need to know, as does the BETA code, energy levels, spins, and parities of precursors and daughters; such data are unknown for most short-lived nuclides. Figure 1 shows a comparison of measured and model results.

![Fig. 1. Delayed neutron spectra for nuclide $^{84}$Rb.](image-url)
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 at reactor shutdown 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 complete detail when Ref. 12 is published.

REDUCTION OF DATA INTO DECAY AND SPECTRA 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. The delayed neutron activity curves, as illustrated in Fig. 2, can be approximated mathematically as a sum of N exponentials representing N-time-groups:

$$n_d(t) = \sum_{i=1}^{N} A_i \cdot e^{-\lambda_i t}.$$  \hfill (1)

![Fig. 2. Total \(n_d\)'s for eight fuels vs time (pulse).](image-url)
The parameters $A_i$ and $\lambda_i$ in Eq. (1) are determined using a nonlinear, least-squares fitting code. The data represents a pulse irradiation, implying that $A_i$ is the product of the group decay constant, $\lambda_i$, and the group yield per fission, $a_i$.

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.

### Table I

**Delayed Neutron Six-Group Parameters**

<table>
<thead>
<tr>
<th>Fission Nuclide</th>
<th>Group 1</th>
<th>Group 2</th>
<th>Group 3</th>
<th>Group 4</th>
<th>Group 5</th>
<th>Group 6</th>
</tr>
</thead>
<tbody>
<tr>
<td>Th227T</td>
<td>0.1027</td>
<td>0.2182</td>
<td>0.1304</td>
<td>0.3555</td>
<td>0.1647</td>
<td>0.0284</td>
</tr>
<tr>
<td>alpha</td>
<td>0.0128</td>
<td>0.0354</td>
<td>0.1098</td>
<td>0.2677</td>
<td>0.5022</td>
<td>2.0956</td>
</tr>
<tr>
<td>lambda</td>
<td>0.0067</td>
<td>0.1907</td>
<td>0.1297</td>
<td>0.3887</td>
<td>0.1729</td>
<td>0.0312</td>
</tr>
<tr>
<td>Th229T</td>
<td>0.0128</td>
<td>0.0350</td>
<td>0.1123</td>
<td>0.2760</td>
<td>0.4950</td>
<td>2.0456</td>
</tr>
<tr>
<td>alpha</td>
<td>0.0364</td>
<td>0.1259</td>
<td>0.1501</td>
<td>0.4406</td>
<td>0.1663</td>
<td>0.0808</td>
</tr>
<tr>
<td>lambda</td>
<td>0.0131</td>
<td>0.0350</td>
<td>0.1272</td>
<td>0.3287</td>
<td>0.9100</td>
<td>2.8203</td>
</tr>
<tr>
<td>Th232F</td>
<td>0.0326</td>
<td>0.0997</td>
<td>0.1431</td>
<td>0.5062</td>
<td>0.1336</td>
<td>0.0848</td>
</tr>
<tr>
<td>alpha</td>
<td>0.0130</td>
<td>0.0350</td>
<td>0.1307</td>
<td>0.3274</td>
<td>0.9638</td>
<td>3.1667</td>
</tr>
<tr>
<td>lambda</td>
<td>0.0826</td>
<td>0.2230</td>
<td>0.1608</td>
<td>0.3885</td>
<td>0.1050</td>
<td>0.0401</td>
</tr>
<tr>
<td>Pa231F</td>
<td>0.0129</td>
<td>0.0347</td>
<td>0.1150</td>
<td>0.2856</td>
<td>0.6706</td>
<td>2.3111</td>
</tr>
<tr>
<td>alpha</td>
<td>0.1360</td>
<td>0.2745</td>
<td>0.1509</td>
<td>0.3052</td>
<td>0.1007</td>
<td>0.0326</td>
</tr>
<tr>
<td>lambda</td>
<td>0.0128</td>
<td>0.0350</td>
<td>0.1073</td>
<td>0.2557</td>
<td>0.6626</td>
<td>2.0254</td>
</tr>
<tr>
<td>U232T</td>
<td>0.0674</td>
<td>0.1927</td>
<td>0.1383</td>
<td>0.2798</td>
<td>0.1128</td>
<td>0.2091</td>
</tr>
<tr>
<td>alpha</td>
<td>0.0129</td>
<td>0.0333</td>
<td>0.1163</td>
<td>0.2933</td>
<td>0.7943</td>
<td>2.3751</td>
</tr>
<tr>
<td>lambda</td>
<td>0.0859</td>
<td>0.2292</td>
<td>0.1781</td>
<td>0.3516</td>
<td>0.1142</td>
<td>0.0409</td>
</tr>
<tr>
<td>U233F</td>
<td>0.0129</td>
<td>0.0347</td>
<td>0.1193</td>
<td>0.2862</td>
<td>0.7877</td>
<td>2.4417</td>
</tr>
<tr>
<td>alpha</td>
<td>0.0900</td>
<td>0.2007</td>
<td>0.1912</td>
<td>0.3684</td>
<td>0.1090</td>
<td>0.0405</td>
</tr>
<tr>
<td>lambda</td>
<td>0.0128</td>
<td>0.0378</td>
<td>0.1271</td>
<td>0.2981</td>
<td>0.8543</td>
<td>2.5314</td>
</tr>
<tr>
<td>U234F</td>
<td>0.0550</td>
<td>0.1964</td>
<td>0.1803</td>
<td>0.3877</td>
<td>0.1324</td>
<td>0.0482</td>
</tr>
<tr>
<td>alpha</td>
<td>0.0131</td>
<td>0.0337</td>
<td>0.1210</td>
<td>0.2952</td>
<td>0.8136</td>
<td>2.5721</td>
</tr>
<tr>
<td>lambda</td>
<td>0.0808</td>
<td>0.1880</td>
<td>0.1791</td>
<td>0.3888</td>
<td>0.1212</td>
<td>0.0420</td>
</tr>
<tr>
<td>U233H</td>
<td>0.0128</td>
<td>0.0364</td>
<td>0.1256</td>
<td>0.2981</td>
<td>0.8475</td>
<td>2.5696</td>
</tr>
<tr>
<td>alpha</td>
<td>0.0380</td>
<td>0.1918</td>
<td>0.1538</td>
<td>0.3431</td>
<td>0.1744</td>
<td>0.0890</td>
</tr>
<tr>
<td>lambda</td>
<td>0.0133</td>
<td>0.0325</td>
<td>0.1219</td>
<td>0.3169</td>
<td>0.9886</td>
<td>2.9544</td>
</tr>
<tr>
<td>U235F</td>
<td>0.0350</td>
<td>0.1807</td>
<td>0.1725</td>
<td>0.3868</td>
<td>0.1586</td>
<td>0.0664</td>
</tr>
<tr>
<td>alpha</td>
<td>0.0133</td>
<td>0.0327</td>
<td>0.1208</td>
<td>0.3028</td>
<td>0.8495</td>
<td>2.8530</td>
</tr>
<tr>
<td>lambda</td>
<td>0.0458</td>
<td>0.1688</td>
<td>0.1769</td>
<td>0.4079</td>
<td>0.1411</td>
<td>0.0595</td>
</tr>
<tr>
<td>U235H</td>
<td>0.0311</td>
<td>0.0356</td>
<td>0.1246</td>
<td>0.2962</td>
<td>0.8260</td>
<td>2.6575</td>
</tr>
<tr>
<td>alpha</td>
<td>0.0130</td>
<td>0.0322</td>
<td>0.1202</td>
<td>0.3113</td>
<td>0.8794</td>
<td>2.8405</td>
</tr>
<tr>
<td>lambda</td>
<td>0.0438</td>
<td>0.1540</td>
<td>0.1719</td>
<td>0.4018</td>
<td>0.1578</td>
<td>0.0707</td>
</tr>
<tr>
<td>U236H</td>
<td>0.0131</td>
<td>0.0333</td>
<td>0.1252</td>
<td>0.3030</td>
<td>0.8802</td>
<td>2.8167</td>
</tr>
<tr>
<td>alpha</td>
<td>0.0178</td>
<td>0.1477</td>
<td>0.1445</td>
<td>0.3864</td>
<td>0.2095</td>
<td>0.0941</td>
</tr>
<tr>
<td>lambda</td>
<td>0.0138</td>
<td>0.0316</td>
<td>0.1211</td>
<td>0.3162</td>
<td>0.9073</td>
<td>3.0368</td>
</tr>
</tbody>
</table>
TABLE I (Cont.)

<table>
<thead>
<tr>
<th>Element</th>
<th>alpha</th>
<th>lambda</th>
</tr>
</thead>
<tbody>
<tr>
<td>U-238F</td>
<td>0.0139</td>
<td>0.1128</td>
</tr>
<tr>
<td>U-238H</td>
<td>0.0195</td>
<td>0.1184</td>
</tr>
<tr>
<td>Np-237F</td>
<td>0.0400</td>
<td>0.2162</td>
</tr>
<tr>
<td>Pu-239F</td>
<td>0.0216</td>
<td>0.1845</td>
</tr>
<tr>
<td>Pu-239F</td>
<td>0.0377</td>
<td>0.2390</td>
</tr>
<tr>
<td>Pu-239F</td>
<td>0.0306</td>
<td>0.2623</td>
</tr>
<tr>
<td>Pu-239F</td>
<td>0.0363</td>
<td>0.2364</td>
</tr>
<tr>
<td>Pu-239F</td>
<td>0.0678</td>
<td>0.1847</td>
</tr>
<tr>
<td>Pu-240F</td>
<td>0.0320</td>
<td>0.2529</td>
</tr>
<tr>
<td>Pu-240F</td>
<td>0.0133</td>
<td>0.1136</td>
</tr>
<tr>
<td>Pu-240F</td>
<td>0.0534</td>
<td>0.1812</td>
</tr>
<tr>
<td>Pu-241T</td>
<td>0.0167</td>
<td>0.2404</td>
</tr>
<tr>
<td>Pu-241T</td>
<td>0.0180</td>
<td>0.2243</td>
</tr>
<tr>
<td>Pu-241T</td>
<td>0.0136</td>
<td>0.1136</td>
</tr>
<tr>
<td>Am-241T</td>
<td>0.0305</td>
<td>0.2760</td>
</tr>
<tr>
<td>Am-241T</td>
<td>0.0133</td>
<td>0.1136</td>
</tr>
<tr>
<td>Am-241H</td>
<td>0.0355</td>
<td>0.2540</td>
</tr>
<tr>
<td>Am-241H</td>
<td>0.0133</td>
<td>0.1136</td>
</tr>
<tr>
<td>Am-242m</td>
<td>0.0247</td>
<td>0.2659</td>
</tr>
<tr>
<td>Am-243F</td>
<td>0.0234</td>
<td>0.2945</td>
</tr>
<tr>
<td>Cm-242F</td>
<td>0.0763</td>
<td>0.2547</td>
</tr>
<tr>
<td>Cm-245T</td>
<td>0.0222</td>
<td>0.1788</td>
</tr>
<tr>
<td>Cf-249T</td>
<td>0.0246</td>
<td>0.3919</td>
</tr>
<tr>
<td>Cf-251T</td>
<td>0.0055</td>
<td>0.3587</td>
</tr>
<tr>
<td>Cf-252S</td>
<td>0.0124</td>
<td>0.3052</td>
</tr>
<tr>
<td>Es-254T</td>
<td>0.0073</td>
<td>0.3148</td>
</tr>
<tr>
<td>Fm-255T</td>
<td>0.0060</td>
<td>0.4856</td>
</tr>
</tbody>
</table>

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.
Having determined the six-group parameters for each fissioning nuclide, the next logical step was to calculate a consistent set of six-group spectra.

Equation (1) is the mathematical representation of delayed neutron activity following a fission pulse. The physical representation using the individual fission-product precursor data is (ignoring coupling)

\[ n_d(t) = \sum_{j} \lambda_j P_{nj} Y_{1j} e^{-\lambda_j t}, \]  

where \( Y_{1j} \) is the direct fission yield of nuclide \( j \) and \( P_{nj} \) is its emission probability.

Both Eqs. (1) and (2) describe the delayed neutron activity per fission following a pulse and should be equivalent. In the present evaluation, it is required that

\[ A_i e^{-\lambda_i t} = \sum_{k} f_{k,i} \lambda_k P_{nk} Y_{1k} e^{-\lambda_k t}, \]  

where the subscript \( i \) represents mathematical group \( i \), the summation is over all precursors contributing to group \( i \), and \( f_{k,i} \) is the fraction of delayed neutrons produced by precursor \( k \) that contribute to group \( i \). It is assumed that the fission-product precursor may contribute to either or both of the adjacent mathematical groups determined by the decay constants. [Previous analyses of individual precursor data, with respect to its reduction to the six-group representation, simply assigned each precursor to a particular group based on either half-life or \( \lambda \) bounds.\( ^6 \text{--} ^{14} \)] The sum of \( f_{k,i} \) for any \( k \) must be unity. These fractions were determined by requiring the least-squares error

\[ \int_0^\infty \left\{ \lambda_k e^{-\lambda_k t} \left[ f_{k,i} \lambda_i e^{-\lambda_i t} + (1 - f_{k,i}) \lambda_{i+1} e^{-\lambda_{i+1} t} \right] \right\} \ dt \]  

to be a minimum.\( ^{13,15} \) A modification of Newton's method was used to return the minimum of Eq. (4). The equilibrium group spectra were then found as

\[ \Phi_i(E) = \sum_k f_{k,i} YC_k P_{nk} \Phi_k(E), \]  

where \( \Phi_k(E) \) is the normalized delayed neutron spectrum of precursor \( k \) and \( YC_k \) is the cumulative yield from fission for precursor \( k \). The normalized six-group spectra for \( ^{235}U \) fast and thermal fission are given in Figs. 3-8 over a 1-MeV energy range in comparison with the six-group spectra taken from ENDF/B-V for \( ^{235}U \). Differences between our thermal and fast spectra are much less than those with the ENDF/B-V evaluation. All spectra are normalized such that the integral over 1 MeV is unity in these comparisons.

Using the method outlined above, the group-one spectrum shown in Fig. 3, has three contributing precursor nuclides. The precursor \(^{87}\text{Br}\) contributes 100% of its delayed neutrons to group one, as is expected; however, two additional precursors, \(^{137}\text{I}\) and \(^{141}\text{Cs}\) contribute about 20% of their delayed neutrons to group one. This result allows the group one spectrum to change for different fissioning systems, as suggested by ENDF/B-V data.
Precursor and group spectra are in energy bins of 10 keV for up to 300 groups. In some applications, we have used the same bin structure to >8.5 MeV, but these calculations did not use temporal groups.

Fig. 3. Group 1 normalized $v_d$ spectra $^{235}$U.

Fig. 4. Group 2 normalized $v_d$ spectra $^{235}$U.
Fig. 5. Group 3 normalized $\nu_d$ spectra $^{235}\text{U}$.

Fig. 6. Group 4 normalized $\nu_d$ spectra $^{235}\text{U}$. 
Fig. 7. Group 5 normalized $v_d$ spectra $^{235}$U.

Fig. 8. Group 6 normalized $v_d$ spectra $^{235}$U.
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, half-lives, 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 $^{235}\text{U}(F)$ system and the results are given in Fig 9. 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.

![Graph showing neutron density over time for various reactivity changes.](image)

**Fig. 9.** Calculated neutron density following step reactivity ($\rho$) inputs for $^{235}\text{U}$ fast fission.
Rossi-alpha ($\beta_{\text{eff}}$/generation time) calculations were performed to insure that the group spectra produce results which are consistent with those using the aggregate spectra derived from the individual precursor data. The one-dimensional transport theory code, ONEDANT\textsuperscript{18} was used to model a Godiva reactor (a bare sphere of enriched $^{235}$U metal) and to calculate the fluxes and adjoint fluxes needed for the perturbation calculation. This was accomplished with a modified version of the PERT-V code\textsuperscript{19} which uses first order perturbation theory based on the multigroup diffusion model and calculates the effective delayed neutron fraction, $\beta_{\text{eff}}$, and the mean generation time. The PERT-V code allows the user to input either a single delayed neutron spectrum or individual group spectra, the results for Rossi-alpha, $\alpha_{\text{DC}}$, calculated using each of these options are given below:

- Aggregate Spectrum: $-1.1185 \times 10^6$ s\textsuperscript{-1}
- Six-Group Spectra: $-1.1123 \times 10^6$ s\textsuperscript{-1}

The perturbation calculations were performed using the six-group decay constants and fractional abundances for $^{235}$U(F) from Table I, and a total delayed neutron yield of 0.0167 neutrons per fission, which is the value recommended in ENDF/B-V.

The measured value of $\alpha_{\text{DC}}$ noted in the Cross Section Evaluation Working Group Benchmark Specifications (BNL 19302), Fast Reactor Benchmark No. 5 is $-(1.11 \pm 0.02) \times 10^6$ s\textsuperscript{-1}. The ratio of the six-group result to the experiment is 1.0020 and that using the aggregate spectrum is 1.0216. These results are in excellent agreement and provide an adequate support of the methods used to derive the six-group spectra.

**SUMMARY**

An evaluated library for 271 delayed neutron precursors has been completed. 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 in data produced for unmeasured systems. ENDF/B-V contains six group spectra for only seven fissioning nuclides and these 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 $\bar{v}_q$ data recommended for inclusion in ENDF/B-V are contain 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 $\bar{v}_q$ recommended for ENDF/B-V, but not the abundances or spectra, are those taken from the previous ENDF/B-V evaluation\textsuperscript{17} or newly evaluated measurements. Nuclides with no reported measurement of $\bar{v}_q$ in the literature were assigned the calculated values. We have presented some results related to $\beta_{\text{eff}}$ and point kinetics; these were primarily used to check the validity of the precursor and group data.

**ACKNOWLEDGMENTS**

We greatly appreciate the direct assistance and communications with F. M. Mann, R. E. Schenter, C. W. Maynard, W. B. Wilson, R. T. Perry, T. Parish, R. J. E. Bauve, E. D. Arthur, D. C. George, G. Rudstam, and K. -L. Kratz.
REFERENCES


17. SAMSON A. COX, "Delayed Neutron Data-Review and Evaluation, Argonne National Laboratory report ANL/NDM-5 (April 1974). [For ENDF/B-V, these data were updated, but unpublished, by R. E. Kaiser and S. G. Carpenter, Argonne Nat. Lab.-West]
