LA-5570-PR

Progress Report

Special Distribution
Issued: April 1974

Applied Nuclear Data Research and Development Quarterly Progress Report

October 1 through December 31, 1973

Edited by

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UNITED STATES
ATOMIC ENERGY COMMISSION
CONTRACT W-7405-ENG. 36

This report presents the status of the Applied Nuclear Data Research and Development Program. The first report in this series, unclassified, is:

LA-5546-PR

In the interest of prompt distribution, this progress report was not edited by the Technical Information staff.

This work performed under the joint auspices of the U.S. Atomic Energy Commission's Divisions of Military Applications, Reactor Research and Development, and Controlled Thermonuclear Reactors, as well as the Defense Nuclear Agency of the Department of Defense, and the National Aeronautics and Space Administration.

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Items I, II, III, IV, V, and X include work for DRRD. Items I, IV, X, XI, and XII include work for DMA. Items VIII, IX, and X include work for DNA. Items VI and VII include work for DCTR. Item XI includes work for NASA. Item XIII was prepared on a fee basis for Pacific Northwest Laboratories of Battelle Memorial Institute.

APPLIED NUCLEAR DATA RESEARCH AND DEVELOPMENT

Edited by

G. M. Hale, D. R. Harris, and R. E. MacFarlane

ABSTRACT

This report present progress in provision of nuclear data for nuclear design applications. The work described here is carried out through the LASL Applied Nuclear Data Group and covers the period October 1 through December 31, 1973. The topical content of this report is summarized in the Contents.

I. MULTIGROUP PROCESSING CODE, MINX (P. D. Soran, R. E. MacFarlane, R. J. LaBauve, and D. R. Harris)

MINX is a modular code system for computation of multigroup nuclear data from evaluated nuclear data in Evaluated Nuclear Data File (ENDF) format. Version I of the code is nearly operational. Development of Version II has begun. It will eliminate the known deficiencies of the previous version and provide expanded capabilities for the MINX cross-section processing system. Progress this quarter will be described under five headings: CCCC Format Output, Auxiliary Codes, Code Structure, Unresolved Resonance Algorithms, and Transport Cross Sections. The current status of MINX was recently reported at the Seminar on Codes for Nuclear Data Processing held at NEA-CPL, Ispra, Italy.[1]

A. CCCC Format Output

The Version II ISOTXS[2] and BRKOXS[2] output data files from MINX were completed during the last quarter. The ISOTXS file contains infinite dilution cross sections and transfer matrices at some specified temperature (usually the temperature is 0°K). The BRKOXS file contains self-shielding and temperature-dependent data. These data are presented as Bondarenko factors. By proper utilization of the ISOTXS and BRKOXS libraries, one can obtain pseudo-composition and temperature dependent cross sections. At present MINX outputs the ISOTXS and BRKOXS files as binary libraries, each on separate output units.

B. Auxiliary Codes

During the last quarter four auxiliary codes to MINX have been or are being developed. These are the BINX, CINX, LINX, and TINX codes and all employ the ISOTXS or BRKOXS data files as input at this stage of development. The BINX (BCD Interpretation of Nuclear X-Sections) code produces BCD records according to CCCC[3] specifications from binary records. The converse is also true, e.g., BINX can write binary records using input BCD records. BINX is required so that BCD MINX libraries may be distributed nationwide. Once a non-CDC user has the BCD MINX libraries, he can convert to the more efficient (in an input/output sense) binary MINX libraries by using BINX.

The CINX (Collapsed Interpretation of Nuclear X-Sections) code permits a zero dimensional collapse of the binary MINX libraries to a smaller subset of the original cross section set. Since the MINX library will be produced using the standard Cross Section Evaluation Working Group (CSEWG) structure[4] (239 groups) many users may want to perform a group collapse. CINX will permit a user input P_n flux and then collapse to a specified subset of the original set. At present there are no provisions for transport approximations other than specified by Version II of CCCC. This condition will change in the next two quarters by the addition of various transport cross-section approximations. CINX is not fully implemented and is still in the design and "debugging" stage, but should be ready for distribution with the MINX library. The LINX (Library Interpretation of Nuclear X-Sections) code permits the merging of additional ISOTXS and BRKOXS files onto master libraries. Because of the structure of ENDF and optimum computer turnaround time, it was found that MINX works best when it processes only one or a few isotopes per job submission. Consequently, only one or a few ISOTXS or BRKOXS files are written. The LINX code was developed to enable the user to work from one ISOTXS or BRKOXS library. The LINX code is fully implemented.

At present the only output processor in MINX is the CCCC interface file. For the short term, the TINX (TD-6 Interpretation of Nuclear X-Sections) code has been developed. The TINX code uses the ISOTXS data file as input and produces output cross sections as specified by TD-6.[5] Presently, only transfer scattering matrices [elastic, inelastic, (n,2n), and total transfer], absorption, nu times fission, and total cross sections are processed. During the coming quarter, individual cross sections [(n,γ), (n,α), etc.] will be processed as specified by TD-6.[5]

C. Code Structure

A new structure has been developed for MINX Version II using the "Top-Down Design" procedure recommended by the Los Alamos Scientific Laboratory. The result is simple and more readable than the previous structure. It also provides for the treatment of unresolved resonances, a feature which has proved difficult to implement in Version I. As shown in Fig. 1, the new structure has four overlays with well-defined purposes. The first three overlays are now in operation and producing an intermediate temperature dependent tape in a non-standard ENDF/B type format. This is called the "PENDFA" tape and is an input to the ETOPL system for producing pointwise library tapes for the continuous energy Monte Carlo codes.

D. Unresolved Resonance Algorithms

Because of the modular organization of the Version II structure, it is possible to use any unresolved resonance treatment without changing the structure of the code. The treatments which have been proposed or considered in the past are the ETOX-MC[2] treatment,[6] the J* method,[7] and the probability table method.[8] Each of these methods has its advantages and disadvantages and the method of choice is not clear. Therefore, it was decided to do a detailed study and comparison based on common data drawn from MINX. During this quarter, a new statistical method based on the ETOX approach was developed. A second treatment based on the probability table was also implemented. Both of these approaches consider the effects of many neighboring resonances, thereby correcting the main deficiency of the ETOX method.

E. Transport Cross-Sections

At the direction of the MINX/SPHINX code development group, a study of transport approximations has been started. The main concern is the transport cross section for diffusion theory, but transport corrections for Sn codes are also of interest.

II. EVALUATION OF NEUTRON-INDUCED REACTIONS ON ^{10}B AT LOW ENERGIES (G. M. Hale, P. G. Young, and R. A. Nisley)

The neutron-induced reactions on ^{10}B at low energies are of interest in the design of nuclear reactors and weapons since ^{10}B is a strong neutron absorber. Furthermore, the large, relatively structureless $^{10}B(n,\alpha)$ cross sections are widely used as standards on which other neutron cross sections are based.

Evaluated cross sections for the reactions $^{10}B(n,n)^{10}B$, $^{10}B(n,\alpha_0)^7Li$, $^{10}B(n,\alpha_1)^7Li$, $^{10}B(n,t)$, $^{10}B(n,p)$, and $^{10}B(n,n')$ below 2 MeV have been joined to an evaluation of the neutron cross sections between 2 and 20 MeV done last quarter. The full evaluated set has been submitted for inclusion in Version IV of ENDF.

The evaluated cross sections at low energies are based primarily on a detailed R-matrix analysis of the ^{11}B system which includes data from α-induced reactions on ^7Li in addition to those from n-induced reactions on ^{10}B. The interest in the (n,α_0) and (n,α_1) cross sections as standards at low energies motivated including the α-induced reactions in the hope of identifying broad structure evident in these reactions near the n-^{10}B threshold. Tentative assignments have been made for these levels. The R-matrix analysis is currently being extended to include data at higher energies, in addition to new (n,α_0) and (n,α_1) measurements by Friesenhahn at Intelcom Rad Tech and by Sealock at the University of Oregon.

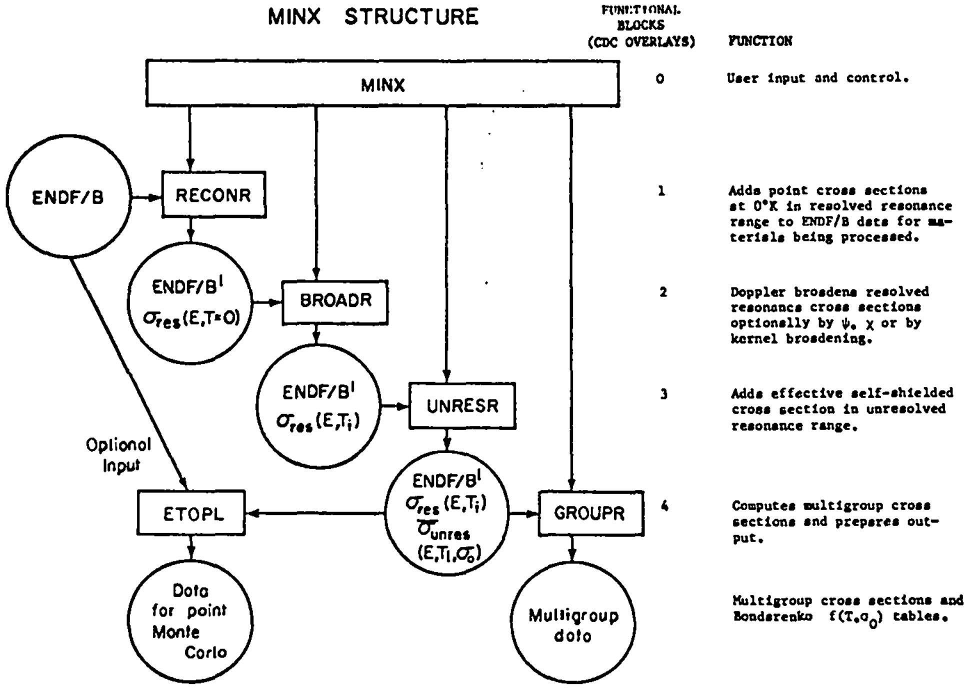

Fig. 1. Structure of MINX code illustrating functional blocks and data flow.

III. TESTING OF NUCLEAR DATA OF IMPORTANCE IN SHIELDING APPLICATIONS AGAINST INTEGRAL EXPERIMENT (D. W. Muir and R. J. LaBauve)

We are currently completing the final draft of the specifications for the CSEWG shielding data testing benchmark SDT8, concerning the ZPPR/FTR-2 shield experiment. Recent investigations have indicated the need for two corrections to the ZPPR experimental data. These are (1) the response of the thermoluminescent gamma-ray dosimeters to fast neutrons, and (2) the response of the ^{238}U fission detector to low energy neutrons (subthreshold fission).

We have reviewed the available experimental data on the sensitivity of thermoluminescent dosimeters to fast neutrons. This sensitivity, C(E), is conventionally expressed as the ratio of the TLD reader response for an exposure of 1 n/cm^2 (at energy E) to the response produced by 1 Rad of ^{60}Co gamma irradiation. We have decided to recommend the results of Wingate[9] in the benchmark specification. More recent measurements of C(E) have been both higher[10] and lower[11] than the Wingate data, in both cases by as much as a factor of two. Obviously, this area needs much more experimental investigation. Using the Wingate data, we estimate that about 12% of the TLD response in the Pu-fueled core is due to neutrons.

In addition, we have reviewed the available data on subthreshold fission of ^{238}U. In ENDF/B (MAT 1158) the fission cross section is zero below 100 keV. However, recent experimental data[12-14] establish rather convincingly that the fission cross section is tens of microbarns over the entire kilovolt range and much higher (up to 30 millibarns) over a few resonances. These features have been incorporated into an evaluation of this cross section over the range 0-100 keV. These data have been multigrouped and will be included in the benchmark specification report. Using these data, we calculate that, deep in the ZPPR/FTR-2 shield, almost half of the ^{238}U fissions in the fission counter traverses were due to neutrons with energies below 100 keV.

IV. ENDF/B FISSION PRODUCT PRODUCTION AND TRANSMUTATION (T. R. England)

The determination of fission product densities is essential for calculating decay heating and delayed phenomena pertinent to reactor safety and weapons design. LASL is participating in all aspects of preparing fission yield/cross section/decay data files for ENDF, and in preliminary (Phase 1) reviews of these files for Version IV of ENDF/B. The initial files, described in the last T-2 quarterly progress report, include data for more than 800 nuclides and are more extensive than originally anticticipated. It is expected that these files will be useful in a number of LASL applications including weapons systems design, shielding, waste disposal, etc.

Part of the Phase 1 review included a calculation of delayed neutron (d.n.) yields using two ENDF/B-IV preliminary nuclide yield compilations and neutron emission probabilities (Pn) for 37 nuclides. The nuclide yields are now being revised and an even-odd Z effect on yield dispersion for ^{239}Pu may be removed for ENDF/B-IV. There is external interest in the calculations reported here, but these will be revised subject to the final nuclide fission yields, and the differences between calculated and evaluated d.n. yields will be analyzed in detail.

The results in Table I indicate that the ENDF/B fission product data are in good agreement with the evaluated thermal delayed neutron yields (2-9% in column 2), but all fast neutron and 14 MeV neutron induced yields are very low (30-40%). This appears to be due both to the charge dispersion model and some mass yields. To some extent there may be a bias toward thermal d.n. yields in that a few reported experimental Pn values may have been inferred using thermal nuclide yields. However, the ENDF/B-IV nuclide yields used in Table I are a new evaluation from the Pn values in use, and incorporate an even-odd Z effect (which generally changes the independent yields by ± 25%).

No column in Table I incorporates the final ENDF/B-IV values and nuclide yields. Column 2 used the latest yields available (November) but other Phase 1 reviews of selected mass chains indicated errors in the November yields. The chains where errors were found involved nuclides important to

TABLE I

CALCULATED TOTAL DELAYED NEUTRON YIELDS PER 100 FISSIONS USING PRELIMINARY ENDF/B PRECURSOR YIELDS

Fission	ν_d Pn[1] Yields[4]	ν_d Pn[1] Yields[3]	ν_d Pn[1] Yields[4]
$^{235}U+n_{th}$	1.809	1.818	1.662*
$^{233}U+n_{th}$	0.762	0.685	
$^{239}Pu+n_{th}$	0.602	0.470	
$^{241}Pu+n_{th}$	0.868	0.872	
$^{235}U+n_f$	1.075	1.102	
$^{239}Pu+n_f$	0.413	0.431	
$^{238}U+n_f$	2.585	2.672	
$^{232}Th+n_f$	3.383	3.190	
$^{235}U+n_{HE}$	0.613	0.526	
$^{238}U+n_{HE}$	1.501	1.481	

1. AERE-R-6993,2/72.
2. Atomic Data & Nuclear Data Tables V12, No. 2.
3. ENDF/B-IV November 73 Yield Listing.
4. ENDF/B-IV October 73 Preliminary Yield Tape

*This value is in remarkable agreement with the preliminary evaluation by Cox of 1.668 ± 0.07.

neutron absorption and did not include an examination of chains containing delayed neutron precursors; there may be no significant errors in the precursor chains.

Column 3 used the earlier (October) ENDF/B-IV yield tape and a later evaluation of Pn's which only became available during the week prior to the Brookhaven National Laboratory meeting. The small value of ν_d for $^{239}Pu+n_{th}$ in this column may be altered when the even-odd Z effect is removed in the final ENDF/B-IV yield tape.

The "final" ENDF/B-IV yield tape will be available in late January. It will incorporate corrections from various Phase 1 reviews, changes in the magnitudes of the even-odd Z effect, and changes in the ZP(A) values used in the yield vs. charge dispersion model. A code has been prepared and these calculations will be repeated using the later data when it is available.

The third meeting of the ENDF Decay Data Task Force for preparation of ENDF/B fission product files took place at BNL on December 11, 1973. LASL contributions related to the work already described included a T-2 outline for processing these ENDF/B-IV data and calculational needs for users, a joint CNC-11/T-2 proposal for yield measurements directed at improvement in the yield vs charge dispersion model, and a T-2/P-2/P-5 proposal for beta and gamma decay heat measurements.

On December 12, 1973, the responsibility of the ENDF/B Decay Data Task Force was expanded to include non-fission product decay data (C. W. Reich, Aerojet, Chairman) and delayed neutron data (T. R. England, Chairman). These and 11 other task force subcommittees constitute the current activity of the CSEWG Fission Product Committee (Chaired by R. E. Schenter, Hanford Engineering Development Lab). LASL has been requested to participate in 9 of the 13 subcommittees and this will be carried out as support permits.

V. COMPARISON OF SUMMATION CALCULATIONS FOR DECAY HEATING IN FISSION REACTORS WITH AN EVALUATION FOR AEC REGULATORY AND THE ANS 5.1 PROPOSED STANDARD FOR REACTOR DESIGN (T. R. England and D. R. Harris)

The most critical cooling interval for emergency core cooling systems (ECCS) in pressurized water reactors is 10 to 1000 seconds following a loss of coolant in core. A proposed ANSI/ANS Standard uses the recommendation of Ref. 15 which, in the $0 \rightarrow 10^3$ second interval, is based on the 1958 recommendation of Ref. 16. In Ref. 17 more recent experimental data are evaluated. Ref. 17 will be used in Phase II testing of ENDF/B-IV files. Both Refs. 15 and 17 apply only to ^{235}U thermal fission and assume (a) no neutron absorption in the fission products, and (b) infinite irradiation of ^{235}U with constant fuel replenishment. To the extent that these assumptions are valid, that is, so long as neutron absorption can be ignored, the heating for finite irradiation can be readily obtained from the infinite irradiation values for heating vs. cooling time.

In Refs. 18 and 19, the heating from other fuels (^{233}U, ^{239}Pu, ^{238}U, ^{232}Th) are compared to ^{235}U using summation calculations and all coupling systematics including neutron absorption. The effects on ^{235}U heating of various irradiation conditions (flux levels, spectrum, irradiation times, etc.) are also compared. These studies used only basic nuclide parameters (yields, halflives, β and γ decay energies, cross sections, etc.). The basic intent was a comparison of fuels and irradiation effects; fuel depletion was permitted which complicated comparison with evaluations.

For comparison with the proposed ANS Standard and the more recent experimental evaluation by Perry et al.[17] all earlier summation calculations have been redone but without fuel depletion. Input data errors discovered in an intense search by Shure and others were corrected. Those errors, listed in Ref. 20, plus a correction in a CINDER code generally reduce the heating by $0 \rightarrow 8\%$ in the $0 \rightarrow 1000$-second interval.

On Figs. 2, 3, and 4, the ^{235}U calculations for a 10,000 hr. irradiation, no depletion, and a constant (10^{13}) flux are compared to the infinite irradiation evaluations. Calculations were made for cooling times beginning at 10 seconds and extending to 10 years. Calculations for cooling times of less than 10 seconds were not made. The data compilation of Ref. 18 used only those measured nuclide parameters available through 1966 and, in some important cases, through 1969.

It is evident that for cooling times between ~30 and 1000 seconds the summation calculations are generally in better agreement with the more recent Perry evaluation than the ANS Standard for β^-, γ, and total heating. At 10 seconds the calculations for total heating are ~8% below Perry. The integrated heating to 1000 seconds would appear to equal or slightly exceed Perry's evaluation.

Any continuing mistrust of summation calculations should be dispelled by these results. The calculations are well within the assigned evaluation uncertainties. On Figs. 2-4, the error bars are taken as only ± 10% about the Perry evaluation, however the assigned uncertainties are 10 to 15% for Perry and > 20% for the ANS Standard.

Unlike Ref. 18, the new ENDF/B-IV files have been assembled and checked by a large task force and incorporate more (primarily short-lived) nuclides and improved yields. These files are expected to hold for calculated heating considerably below 10 seconds. Because there are large discrepancies in the experimental data used in the evaluations and because even the 10-15% uncertainty assigned in Perry's evaluation are costly in conservative ECCS designs, benchmark measurements of β and γ heating will be needed

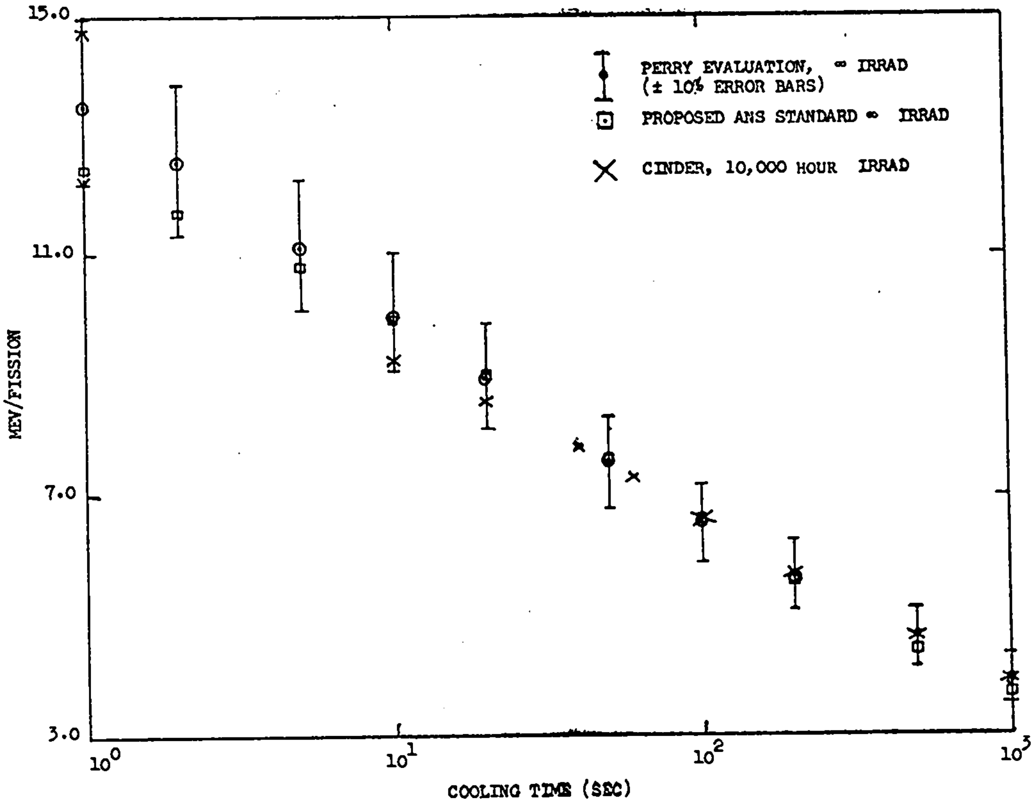

Fig. 2. Total ($\beta^-+\gamma$) heating vs cooling time ^{235}U thermal fission.

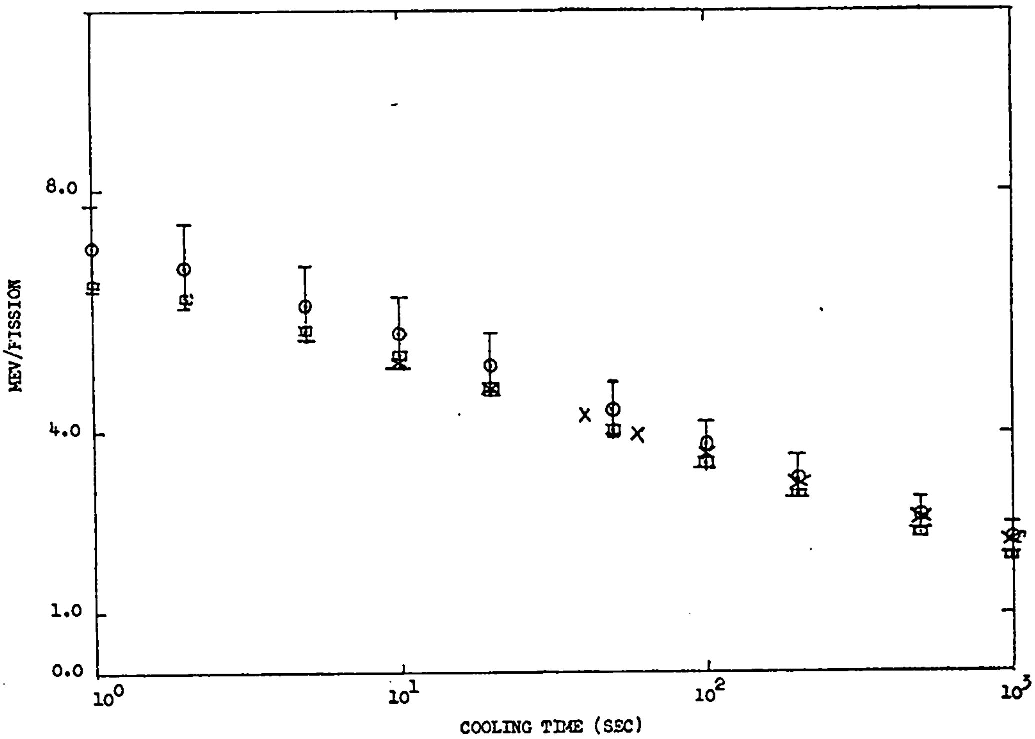

Fig. 3. Gamma heating vs cooling time.

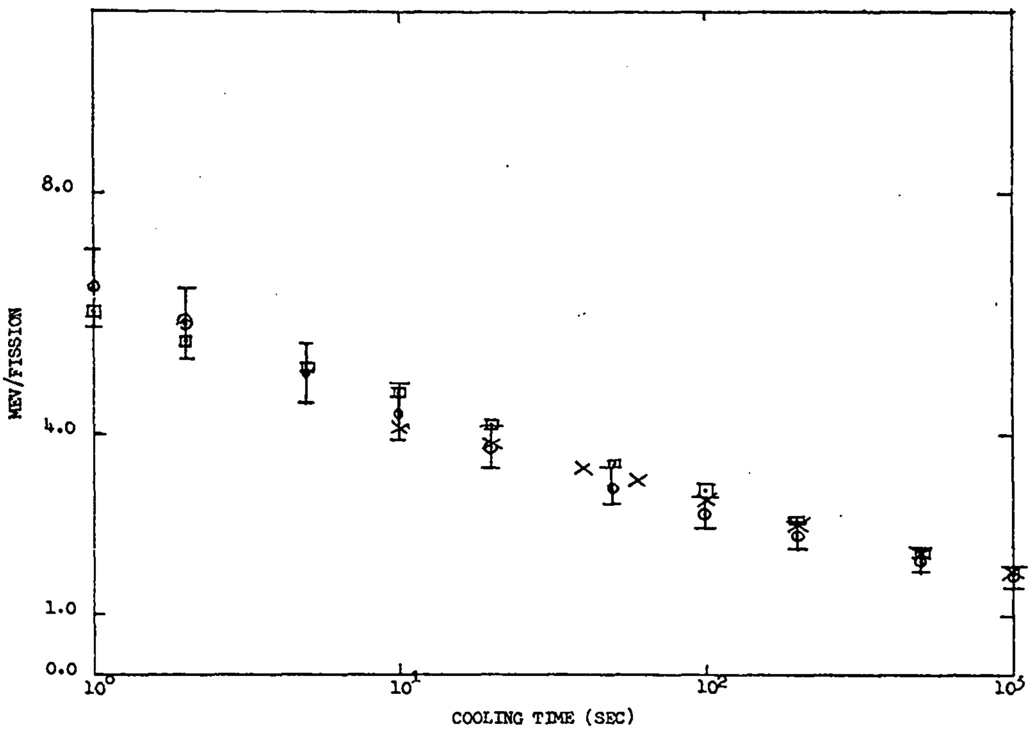

Fig. 4. Beta heating vs cooling time.

for any final judgment of the ENDF/B-IV file. One expects that these new data will ultimately permit use of summation calculations for each fuel and power reactor history rather than the present use of the ^{235}U thermal evaluations for infinite irradiations + 20%.

Finally, one should note that the fraction of heating by γ's exceeds the β heating in these and earlier calculations of Refs. 18 and 19 and in Perry's evaluation. (This is also true to a lesser extent in the proposed ANS Standard, but not by the source[16] referenced by the Standard.) The difference could be of significance in ECCS design.

A. Selected ^{235}U Heating Comparison for Other Irradiations

Using the ANS Proposed Standard as a basis for comparison, Fig. 5 shows the effect of various irradiation times in % on heating vs the cooling times of 60 → 10^5 seconds. Neutron absorption is permitted by the fission products, but the flux and fuel density are held constant.

It is noted that for a small time interval, even a 1000-hour irradiation exceeds the infinite irradiation standard.

The effects of neutron absorption can be seen by comparing the two 10,000-hour irradiation curves. The increase in heating due to long irradiation times implicitly contains the effect of neutron absorption and the buildup of long half lived nuclides.

B. Fuel Comparisons

In Fig. 6 the total heating of ^{238}U, ^{239}Pu, ^{233}U, and ^{232}Th are compared with the calculated ^{235}Y results. All results use the same fission rate, flux level and no fuel depletion. In the 10-10^3 second interval, only ^{232}Th exceeds ^{235}U. This would justify the use of ^{235}U heating for all fuels, except ^{232}Th, in conservative ECCS design. However, such judgment should be reserved pending use of ENDF/B-IV data.

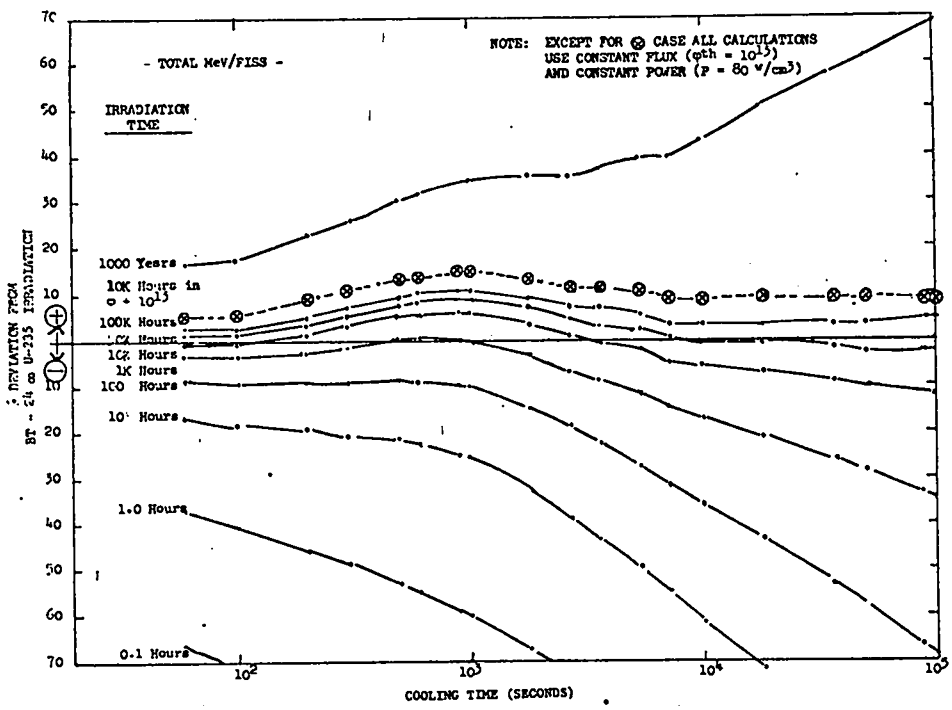

Fig. 5. Heating comparisons of ^{235}U thermal fission with ANS 5-1 ∞ irradiation standard for various irradiation times; total β + γ heating.

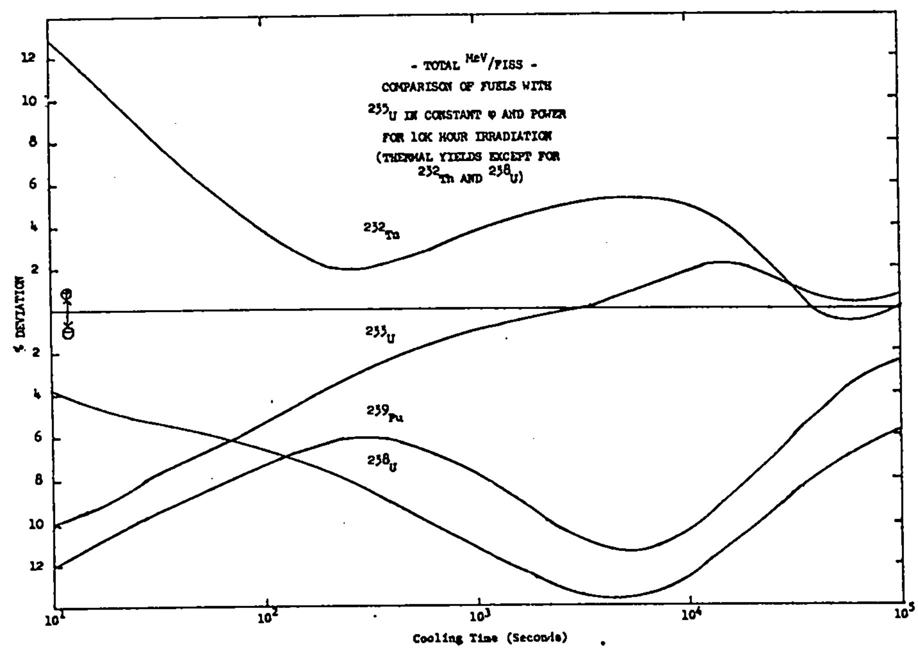

Fig. 6. Comparison of fuels with ^{235}U in constant ϕ and power for 10K hour irradiation (thermal yields except for ^{232}Th and ^{238}U).

VI. EFFECT OF Be(n,2n) MULTIGROUP TREATMENT ON THETA-PINCH BLANKET NUCLEONICS (P. D. Soran, D. J. Dudziak [T-1], and D. W. Muir)

An important material in the reference theta-pinch reactor is the neutron multiplier, Be, which makes possible a thin blanket with an adequate breeding ratio. The blanket multiplication is due primarily to the Be(n,2n) reaction, so the proper multigroup processing treatment of this cross section is of considerable importance. In the Version III of ENDF[21] (MAT 1154)[22] the energy distribution, g (E,E') of the secondary neutrons from this reaction is given as a tabulated function of the secondary energy, E', rather than as an analytic function. Further, the angular distribution, $P(E,\mu)$, is given as a tabulated function of μ, the cosine of the scattering angle.

Earlier cross sections were calculated using the ETOG[23] code. ETOG cannot process the (n,2n) secondary distributions in MAT 1154. Because of this limitation, secondary energy distributions were treated by using evaporation model data extracted from ENDF-II (MAT 1006).[24] The angular distributions were approximated by ETOG as isotropic in the laboratory system.

The development of the MINX[1,25] code has permitted the use of the more accurate (n,2n) secondary distribution data. This study used P_3 data processed by ETOG (Case A)[26] as well as the same data set in Case A with the exception of Be (MAT 1154) cross sections processed by MINX (Case B). All neutronic transport calculations were performed with the DTF-IV discrete ordinates code[27] using S_4 angular quadrature in a one-dimensional cylindrical geometry illustrated in Ref. 28. A uniform volumetric source of 14-MeV neutrons was placed in the plasma region.

Selected results showing the effect of the Be (n,2n) secondary distribution on integral quantities such as tritium breeding, neutron energy deposition and displacements per atom are present in Table II. These results are for a modification of the RTPR with a 3-cm thick Be region. The table illustrates two important trends; first a softer neutron spectrum for Case B, and second an angular effect. The former is illustrated in line 1 of the table. Since the ^{7}Li(n,n'α)T reaction has a high threshold (~ 2.5 MeV) only the high energy flux is effective and consequently there is a higher tritium production for Case A than for Case B. The softer spectrum is further exemplified by the total tritium breeding ratio. The softer spectrum in Case B can be explained by the difference in the Be(n,2n) models. In Case A the average energy of the secondary neutrons is ~ 7 MeV whereas in Case B the average secondary energy is ~ 4 MeV. Note that the displacements per atom (dpa) in the niobium wall are less for Case B than for Case A. This is due primarily to the angular distribution of the Be(n,2n) secondary neutrons. In case A the angular distribution is isotropic where in Case B the angular distribution is forward peaked. consequently, fewer secondary (n,2n) neutrons return to the niobium wall in Case B.

VII. NUCLEAR DATA FOR THE CONTROLLED FUSION PROGRAM (D. W. Muir, L. Stewart, R. J. LaBauve, and R. E. Seamon [TD-6])

Multigroup cross sections for ^{19}F have been added to the LASL/CTR neutron/photon transport cross section library. The neutron interaction data were obtained from the Los Alamos Master Data File (SID 515) and processed with EVXS.[29] Preliminary photon production cross sections were obtained by using the energy-balance method. Photon interaction cross sections were obtained using GAMLEG.[30] Other recent additions to the library include revised ^{9}Be neutron interaction cross sections (see Sec. VI) and transport corrected cross sections for D.

A study of the effects of thermal broadening of the D-T fusion neutron peak[31] has been extended to treat more realistic cross-section energy-dependence near threshold. Among the reactions studied is ^{9}Be(n,p)^{9}Li, which is of interest since Be is a major constituent of the Reference Theta-Pinch Reactor.[28] ^{9}Li beta decays with a half-life of 175 ms, with 75% of the decays resulting in delayed neutron emission. The high threshold (14.26 MeV) and distinctive decay mode make this reaction an interesting candidate for plasma-temperature diagnostics. Recent experimental data[32,33] for the (n,p) reaction cross section were used to create an ENDF/B-type file which was processed into a peak-averaged cross section using the MINX processing code, for 11 different plasma temperatures ranging from 3 to 0 keV. The numerical results can be fit by a smooth curve of the form

$$\ell n\sigma = a + (b + \frac{c}{\sqrt{T}})\ \ell nT \quad .$$

TABLE II

Selected Neutronic Parameters in a RTPR as a Function of Be(n,2n) Models

	Parameter	Case A	Case B	% Difference
1.	$^7Li(n,n'\alpha)T$ (reactions/plasma neutron)	0.1790	0.1568	12.4
2.	Total tritium breeding ratio	1.084[a]	1.020[a]	5.9
3.	Be(n,2n) (reactions/plasma neutron)	0.4332	0.3490	19.4
4.	Nb(n,2n) in first wall (reactions/plasma neutron)	1.286-2	1.262-2	1.9
5.	Nb dpa/year in first wall at 2 MW/m^2	37.31	32.28	13.5
6.	C dpa/year max. in graphite at 2 MW/m^2	11.84	9.88	16.6
7.	Cu parasitic absorption (reactions/plasma neutron)	0.1105	0.0958	13.3
8.	C parasitic absorption (reactions/plasma neutron)	0.0883	0.0840	4.9
9.	Total neutron flux at center of Be region at 2 MW/m^2 ($m^{-2}s^{-1}$)	8.526+18	7.832+18	8.1
	a. 3.01 MeV to 12.2 MeV	1.077+18	6.909+17	35.8
	b. 202 keV to 3.01 MeV	2.596+18	1.949+18	24.9
10.	Neutron heating (kerma) in Be region (MW/m)	2.305	1.955	15.2
11.	Neutron energy deposition (kerma) per plasma neutron (MeV)	16.372	14.595	10.9

[a] Not including resonance self-shielding, which is estimated to increase the breeding ratio by ~7%.

If T is expressed in keV and σ in μb, a good fit is obtained for a = -4.854, b = 1.794, and c = 3.734. Although the cross section σ is small (2 μb at 5 keV), the slope $\frac{d(\ell n\sigma)}{d(\ell nT)}$ is fairly large (2.1 at 5 keV). Thus, at this energy T can be determined with a statistical uncertainty of 8% with fewer than 35 counts from a neutron detector.

Critical reviews of those ENDF/B evaluations prepared at Los Alamos which are important to CTR applications are underway in preparation for the establishment of a national CTR evaluated neutron library. These reviews are needed in order to identify areas where further work is required and to establish reasonable estimates of the errors currently placed on important data files.

VIII. NEW EVALUATIONS OF NEUTRON-INDUCED GAMMA-RAY PRODUCTION DATA FOR Ag and Sn (P. G. Young, R. J. LaBauve, and D. M. McClellan)

New evaluations of gamma-ray production cross sections and secondary energy spectra for neutron-induced reactions on natural Ag and Sn have been performed. The results are in the ENDF/B format and cover the neutron energy range from 10^{-5} eV to 20 MeV. The secondary gamma-ray spectra and cross sections for both materials at neutron energies in the MeV region are based on measurements by Dickens et al.,[34] roughly extrapolated to $E_\gamma = 0$ using an empirical formula by Howerton.[35] The spectra in the eV region were obtained from measurements by Orphan et al.[36] The radiative capture cross section for Ag was calculated from ENDF/B-III resonance parameters; the capture cross section for Sn was estimated from very limited experimental data.

The evaluations have been transmitted to the Defense Nuclear Agency cross section library maintained by the Radiation Shielding Information Center at Oak Ridge National Laboratory.

IX. RE-EVALUATION OF SEVERAL NEUTRON-INDUCED REACTIONS ON ^{27}Al (D. G. Foster, Jr., P. G. Young, and D. M. McClellan)

Extensive revisions have been made to our earlier evaluation[37] of the neutron-induced cross sections of ^{27}Al. The most important changes are

1. The radiative capture gamma-ray spectrum for thermal neutrons incident was revised to better agree with recent measurements.
2. Improved (n,p) and (n,α) cross sections, which we provided earlier to the Standards Subcommittee of CSEWG, were incorporated into the evaluation.
3. The gamma-ray production cross sections and energy spectra were revised to include extensive

new measurements by Orphan and Hoot[38] and Dickens et al.[39]

4. The (n,n') cross sections to highly excited states in ^{27}Al were revised to be more consistent with measurements by Kammerdiener.[40] In addition, anisotropic (n,n') angular distributions based on experimental data were incorporated.
5. The neutron total cross section was completely re-evaluated, primarily to take advantage of a recent measurement by Perey et al.[41] The cross section below a few keV is about 11% lower than in our previous evaluation, and the fine structure from there up to about 0.5 MeV has been completely revised. Much false structure has been eliminated and the remaining structure has been corrected for experimental energy resolution. The fine structure above 3 MeV has been preserved instead of deliberately being oversmoothed, as it was in the previous evaluation. The new evaluation has an estimated uncertainty of 1% from 0.15 to 20 MeV.

The new evaluation has been provided to the Defense Nuclear Agency cross-section library at Oak Ridge as MAT 4135, MOD 3. The data are being included in Version IV of ENDF/B.

X. MISCELLANEOUS EVALUATION ACTIVITIES (D. G. Foster, Jr., R. E. Hunter [TD-2], R. J. LaBauve, W. E. Stein [P-DOR], L. Stewart, and P. G. Young)

The ENDF/B-III evaluation for neutrons incident on ^{7}Li was modified to include a 1972 evaluation of the neutron total cross section by Foster. Both the neutron and gamma production evaluations were also extended to 20 MeV.

Explicit (isotropic) angular distributions for the (n,2n), (n,3n), and (n,np) reactions and neutron energy distributions for the (n,np) reaction were added to the Atomics International ENDF/B-IV evaluations for ^{182}W, ^{183}W, ^{187}W, and ^{186}W.

Phase 1 reviews were performed for the ^{51}V and ^{181}Ta evaluations that have been submitted for inclusion in Version IV of ENDF/B.

The fast neutron fission cross section for ^{237}Np evaluated by Stein was submitted for Version IV of ENDF/B.

In cooperation with various CSEWG organizations effort is underway to provide new evaluated data files for U-235, U-238, Pu-239, and Pu-240; these will extend to 20 MeV incident neutron energy, and they will include gamma-production files. Except for U-238, many of the fast-neutron cross sections were evaluated at LASL along with the complete gamma-production files. We are currently assembling the U-235 file and checking the data for consistency through various checking and processing codes before transmission to BNL. These files will be consistent with newly evaluated thermal constants provided by the CSEWG Thermal Task Force and they will also contain delayed neutron yields and spectra. Except for U-238, the files contain contributions from high-energy "direct" inelastic scattering with anisotropic angular distributions and LASL has been asked to provide similar data for this isotope. Except for Pu-240, these files will also contain fission product yields provided by the CSEWG Fission Product Subcommittee.

XI. MEDIUM AND LOW ENERGY CROSS SECTION LIBRARY (D. R. Harris, D. G. Foster, Jr., and W. B. Wilson [Texas A & M])

A program is under way to improve the accuracy and cost-effectiveness of medium and low energy particle and photon transport calculations by preparing combined medium and low energy nuclear data libraries. The ENDF/B library of nuclear data for incident neutrons and photons below 20 MeV is augmented by nuclear data at higher incident energies (< 3.5 GeV) and for other particles. Required cross sections are determined from nuclear model calculations and from experimental data. The nuclear models which have thus far been used include the intranuclear cascade and evaporation models for reactions, together with the diffraction and optical models for elastic scattering. The fission reaction channel has thus far not been included, a serious omission for heavy targets.[42] During the past quarter we have been reviewing the computation tools that have been employed in this development. An incorrect energy balance calculation in the intranuclear cascade module has been identified and is being corrected. This error is most pronounced at high energies.

XII. NEW EVALUATION OF n-^{4}He SCATTERING CROSS SECTIONS (R. A. Nisley, G. M. Hale, P. G. Young)

An understanding of light element cross sections is important for nuclear weapons design. A

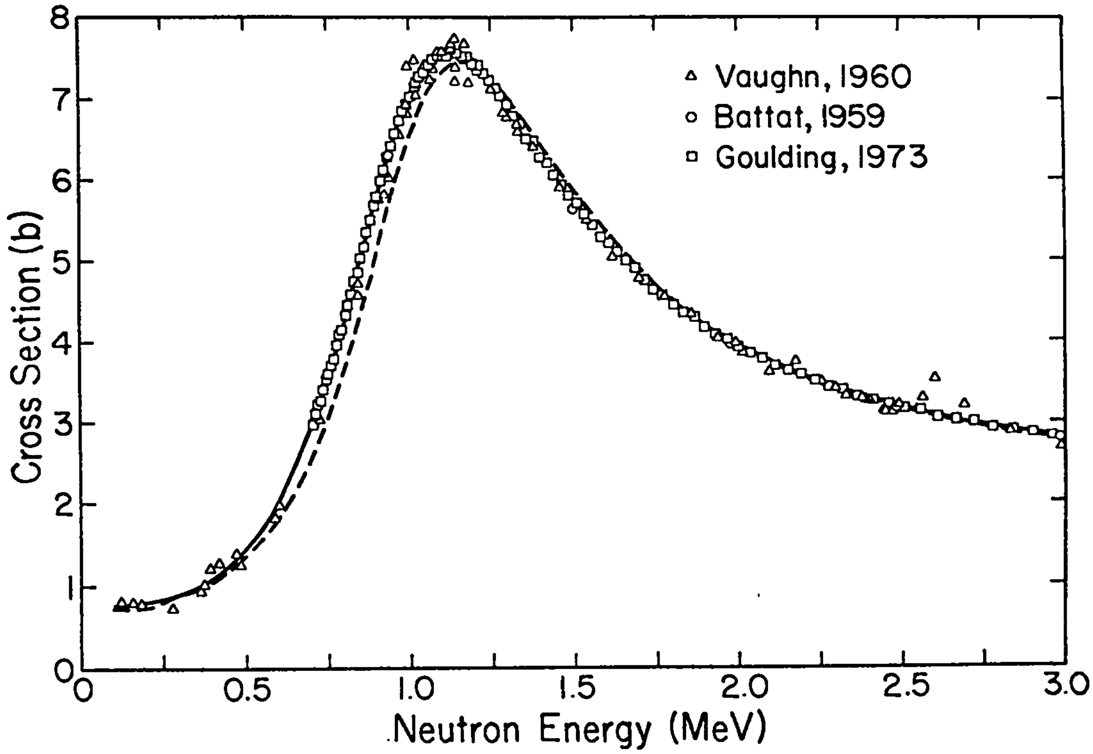

Fig. 7. Neutron total cross section of ^{4}He, The solid curve is the new evaluation and the dashed curve is a previous analysis.[43] The experimental data of Goulding, Battat, and Vaughn are from Refs. 45, 46, and 47, respectively.

new evaluation of n-^{4}He scattering has been completed for neutron energies between 10^{-5}eV and 20 MeV. The evaluation is based on a detailed R-matric analysis[43] that fits the n-^{4}He data simultaneously with generally more precise p-^{4}He scattering and polarization data, using a simple model to account for the charge difference. All the available n-^{4}He and p-^{4}He experimental data below 20 MeV were considered using a set of criteria based on statistical considerations to eliminate inconsistent data. A measure of the goodness of fit of the evaluation to the experimental data is the statistical parameter χ^2 per degree of freedom, which has an expected value of 1.0. Separate analyses of p-^{4}He and n-^{4}He measurements yielded fits having a χ^2 per degree of freedom of 0.99 in each case. The combined analysis of both systems, which included over 3000 data points, was accomplished with a χ^2 per degree of freedom of 1.07. The R-matrix parameters obtained by fitting these data have been used to predict observables with a high degree of accuracy.

The newly evaluated neutron total cross section (solid curve) is compared in Fig. 7 with an older analysis[44] (dashed curve) and with some of the available measurements.[45-47] One sees that the present evaluation differs significantly from the previous analysis below 2 MeV, and that it is indistinguishable from the measurements of Goulding et al.[45] above 0.7 MeV.

The new evaluation has been provided to the DNA cross section library at RSIC and to the National Neutron Cross Section Center (NNCSC) for inclusion in Version IV of ENDF/B.

XIII. TRANSMUTATION PROCESSING OF HIGH LEVEL WASTE (D. G. Foster, Jr., T. R. England, D. R. Harris, J. W. Healy [H-DO], H. S. Jordan [H-DO], E. A. Knapp [MP-3], K. D. Lathrop [T-1], R. E. MacFarlane, D. W. Muir, R. G. Shreffler [D-O], P. D. Soran, and R. F. Taschek [ADR])

We have reviewed a draft of a Battelle proposal[48] on transmutation processing of high level waste at the request of Pacific Northwest Laboratories. This document describes a preliminary study of transmutation concepts (accelerators, thermonuclear explosives, fission reactors, fusion reactors) for the management of fission-product and actinide waste generated in the nuclear-energy economy. We strongly support the principal conclusion of the study, i.e., that disposal of higher actinide nuclides by recycling in fission power reactors appears to be attractive with near-term, high-liklihood technology, and that major R & D emphasis is appropriate in that area. Effective, large-scale applications of this concept will depend on R & D which affects the economics of recycle utilization of the actinides including plutonium. With respect to disposal of fission-product nuclides, we feel that the work done to date may have discarded fission reactor transmutation of selected nuclei (after chemical separation) prematurely, whereas a more detailed study may demonstrate feasibility for special cases.

We support the PNL conclusion that transmutation both of actinides and of fission products in fusion reactors appears to be sufficiently attractive that a modest R & D effort is projected in this area. We feel, however, that the case for transmutation by particle accelerators is stronger than suggested and deserves an R & D effort to determine the utility of GeV proton bombardment with or without fission boosting. In this connection we describe a fast actinide burner, either stand-alone or driven by a proton accelerator, which appears to have economic potential.

Contrary to the PNL conclusion, we suggest that transmutation of high-level waste by neutrons from an underground nuclear explosion may be very promising, rather than infeasible.

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