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QUARTERLY TECHNICAL PROGRESS REPORT
LMFBR PHYSICS PROGRAMS
JANUARY-MARCH 1972

AEC Research and Development Report

Atomics International
North American Rockwell

P.O. Box 309
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JANUARY-MARCH 1972

The preceding Progress Report was AI-AEC-13019

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Program: General Reactor Physics

AEC Task: 4 - Cross-Section Analysis

Project Manager: H. A. Morewitz

Reporting Period: January-March 1972

189A: 13493

General Order: 06524

Subaccount: 13130

Category: 04-40-01-03.1

Principal Investigators: H. Alter, P. F. Rose, R. S. Hubner, G. Gigas

I. PROJECT OBJECTIVES

The objectives of this project include the evaluation and maintenance of an up-to-date set of basic nuclear data, the production and evaluation of multigroup constants, and the improvement of present-day methods of neutronic calculations, as related to microscopic and macroscopic nuclear data. To accomplish these objectives, existing experimental and theoretical information on nuclear data will be surveyed, analyzed, and compiled. Automated methods will be provided to periodically update and evaluate the compiled data, and nuclear model calculations will be developed and used to help fill gaps in measured data. An automated system will be developed and maintained for the ready production of multigroup constants. Spectra-generating techniques for production of realistic multigroup constants will be developed, and incorporated into the system. The data and methods produced are needed to improve reliability for calculating advanced reactor concepts. In particular, the realization of these project objectives will be utilized in ensuring that the LMFBR design incorporates adequate margins for safety and economy. The accomplishment of the stated project objectives is related to successful implementation of the LMFBR Program Plan in the areas of acquisition of nuclear data (Task 9-1), cross-section evaluation (Task 9-1-2), and compilation and distribution of evaluated data (Subtask 9-1-2.2).

II. MAJOR ACCOMPLISHMENTS DURING FISCAL YEAR 1972

An operational version of SCORE III A,⁽¹⁾ checked out and currently available, provides a number of advantages for the least-squares fitting of resonance data. These include: continuity of effort and thinking is provided by practically instant turn-around time; results can be easily compared and immediately improved upon

by the interactive mode of operation; and SCORE provides an ability to select different groups of neighboring resonances and, by allowing for rapid changes in resonance energies and total line widths, provides an intuitive insight into the sensitivity of the parameters obtained to the choices made. SCORE III A has proved to be an excellent tool for interactive data fitting and in its present form will perform required analyses for narrow energy bands within a few minutes of real time computer-scope use.

III. PROGRESS DURING REPORT PERIOD

A. ENDF/B DATA EVALUATION

The evaluation of neutron cross section data for the isotopes of tungsten progressed in several areas.

Evaluation of the capture cross sections were completed for natural tungsten, and data for the isotopes have been compiled and renormalized to new standards where appropriate. Three to six experiments contain data in the unresolved resonance region for each isotope. The data of the various experiments appear to differ in a systematic as well as in a statistical manner. The following technique has been pursued in order to determine appropriate unresolved resonance parameters:

- 1) An optimized uncertainty weighted parameter fit was made for each set of experimental data. Where there were few experimental points, unresolved parameters were set equal to values obtained from resolved resonance parameters. The URF code⁽¹⁾ was modified to perform these calculations.
- 2) The capture cross sections resulting from the fit in Step 1, above, were calculated on a fixed energy grid within a range applicable to each experiment. The average uncertainty weighted cross section curve was calculated along with the error expectation.
- 3) The fitting procedure outlined in Step 1 was repeated for the isotopic data generated in Step 2.
- 4) The natural element cross sections were then compared to the abundance weighted isotopic cross sections obtained in Step 3.

At this point adjustments were made to the calculations in previous steps in order to bring the isotopic data into better agreement with the natural element.

The above procedure accommodates rather well the problems of systematic errors, measurements over only part of the energy range, and uneven distributions of measurement energies.

The (n,2n) cross sections have been evaluated using the semi-empirical technique of S. Pearlstein.⁽²⁾ A fit for W^{182} was consistent with two experimental data points⁽³⁾ but a fit to W^{186} compared poorly to a measurement reported by Druzhinin.⁽⁴⁾ The calculations were re-computed with an effective threshold for the (n,3n) reaction 1 Mev higher than that calculated from mass tables. This improved the fit considerably for W^{186} without affecting the W^{182} results.

The (n,p), (n,np), and (n, α) excitation functions were obtained for the isotopes of tungsten. Experimental data for W^{186} (n,p) Ta^{186} and W^{186} (n,np) Ta^{185} were taken primarily from Barry, Coleman, Hawker, and Perkin.⁽⁵⁾ The shape of the curves for the other isotopes of tungsten were made similar to W^{186} , inasmuch as little experimental data are available for them. Coulomb energy barrier penetration shifts for the threshold were calculated as prolate charged ellipsoids after the method of Lamm⁽⁶⁾ with corrections for diffuseness and exchange effects. The isotopes were assumed to be stably deformed and the equilibrium deformation values were used in a computed calculation of the Coulomb energy barrier. The reaction energy balance and threshold energies were calculated for the tungsten isotopes for the (n,p) and (n,np) as well as (n, α), (n,2n), and (n,3n) reactions. The 1971 Atomic Mass Evaluation of A. H. Wapstra and N. B. Gove⁽⁷⁾ was used. Direct evaluation from the Atomic Mass Table, Part I of the above reference, is in close agreement with the Nuclear-Reaction and Separation Energies, Part II of the reference, except for W^{183} (n,p) where the direct calculation is higher. At 14 Mev the evaluation for W^{183} (n,p) is some 20% lower than the prediction of Levkovskii's semi-empirical formula. The reaction cross sections for the (n,p), (n,np), and the (n, α) reaction are shown in Figures 1 through 3 (data sources are given at the end of this Task Report).

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Figure 1. Comparison of Calculated and Measured Cross Section, Tungsten (n,p)

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Figure 2. Comparison of Calculated and Measured Cross Section, Tungsten (n,np)

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Figure 3. Comparison of Calculated and Measured Cross Section, Tungsten (n,α)

The nuclear code JUPITOR was received from Brookhaven National Laboratory. Test cases for checkout on the CDC-7600 computer were run and changes to correct and update the BNL version are underway. Data for a coupled-channel model calculation were assembled for the odd-even nuclide W^{183} in two approximations: as a deformed nucleus and as a vibrational nucleus excited to multiple-phonon states, each calculation with two assumed level schemes. For the vibrational calculation with a spherical nucleus, the low lying states were pure one- and two-phonon cores. The strength of the quadrupole vibrations have been determined as the "partial deformations" for exciting levels according to the distorted-wave analysis model of Bassel, Satcher, Drisko, and Rost.⁽⁹⁾

Inelastic continuum particle emission and gamma-ray transmission coefficients were calculated for a test case of tungsten excited to its continuum region and de-excited by a gamma-ray cascade with the possibility of neutron competition (fission excluded). The output transmission coefficients were used in a program (COMNUC) to analyze the compound nucleus interactions. A comparison calculation of first excited state coupling (2+ level) vs strong coupling including the second excited state (4+ level) for various cross sections for the even isotopes of tungsten was carried out. In general, at the energy considered (1 Mev) for tungsten, the fit to the experimental data was considered better with 2+ only coupling.

A search was conducted to update the experimental data library for europium-151 and -153 including resonance parameters, reaction cross sections, and energy level diagrams.

B. ENDF/B VERSION 3 DATA - PHASE II CALCULATIONS

All 8 Version 3 ENDF/B (final) tapes received from BNL were copied onto the CHIPSTORE at Lawrence Berkeley Laboratory Computing Center. Backup CHIPSTORE data sets were prepared for all data sets. In addition, the Version 3 data files were copied onto tape on the S/360 at the North American Rockwell Computing Center.

The ENDF/B data processing codes RIGEL-3, ETOE-3, MERMC2, and MC² were copied onto the CHIPSTORE at Berkeley.

TABLE 1
PARTICIPATION

Assembly	AI	ANC	ANL	BNL	B&W	GE	GGA	HEDL	LASL	LLL	ORNL	WARD	Total
Jezebel									X	X	X		3
Vera 11A					X			X		X			3
ZPR-3-48		X	X	X			X						4
Zebra 3		X			X			X					3
Godiva				X					X	X	X		4
Vera 1B		X							X	X		X	4
ZPR-3-6F			X					X	X				3
ZPR-3-11	X					X	X						3
ZPR-3-12					X				X		X	X	4
Zebra-2	X		X				X						3
ZPPR-2	X			X		X						X	4
ZPR-6-7	X			X		X						X	4
ZPR-3-56B			X				X	X			X		4
SEFOR	<u>X</u>	<u>X</u>	<u>X</u>	<u>X</u>	<u>X</u>	<u>X</u>	<u>—</u>	<u>X</u>	<u>—</u>	<u>—</u>	<u>X</u>	<u>X</u>	<u>9</u>
Total	5	4	5	5	4	4	4	5	5	4	5	5	

Work was begun on the Phase II testing of the ENDF/B Version 3 (final) data. The CSEWG benchmark assemblies assigned to AI were ZPR-3-11, ZEBRA-2, ZPPR-2, ZPR-6-7, and SEFOR. It was recognized that certain Version 3 data would be unacceptable to ETOE-3; this required that changes be made to the data and the code. ENDF/B Version 3 materials Be and Fe were modified to remove tabular distributions for inelastic and $n,2n$ reactions. These were replaced by temperatures for evaporation spectra. Also, Fe was modified by removing the 4th isotope specification in the resonance file. Program ETOE-3 was modified to allow the processing of more than 1000 points for the 1st inelastic level in Fe.

Program RIGEL-3 was used to merge selected materials from ENDF/B Data Sets 301, 302, 303, and 304 and place them on Data Set 351. Also, a merge of selected materials from ENDF/B Data Sets 305 and 308 was done; the result was placed on Data Set 352. Program RIGEL-3 was used, again, to convert these data sets from Mode 3 (BCD, standard arrangement) to Mode 2 (binary, alternate arrangement) for use in ETOE-3. Three ETOE-3 runs were made on each of these data sets using temperature specifications of 300, 677, and 1365°K. The two resulting MC² libraries at each temperature were then merged using program MERMC2. Since U²³⁶ was not available on ENDF/B Version 3, the Version 2 data from another MC² library was merged onto the 300°K MC² library using MERMC2. All of the libraries, both intermediate and final, were saved in the CHIPSTORE. All of the above data manipulation was accomplished without a single parity error; however, other problems arise when using the CHIPSTORE. These problems result in some delay but in no loss of information or ability to process the data as is the case when tape parity errors occur.

C. CSEWG PARTICIPATION

With the release of the ENDF/B Series 300 data files, the data testing subcommittee of CSEWG had to concern itself with fully establishing the validity of these data. To meet this requirement a participation matrix, which would equitably share the burdens of this task, was devised and distributed, Table 1. One sees from this matrix that each of the CSEWG benchmark assemblies will be calculated three times at a minimum and that no participant has more than

five assemblies to calculate. Results of the calculations are to be reported at the May 1972 CSEWG meeting.

D. TESTING OF ENDF/B MULTIGROUP PROCESSING CODES

In the last report in this series, AI-AEC-13019, several results for the GEDANKEN calculations were presented. During this report period, complete results for these calculations had been sent to the several participants. All participants were requested to examine the results and forward comments concerning their own degree of comparison with other participant results. They were also asked to investigate their discrepant results and determine if the discrepancies were reasonable or errors. To date no replies have been received and a letter urging a response is being prepared.

E. TASK FORCE REVIEW OF MICROSCOPIC CROSS SECTION DATA FOR NEUTRON DOSIMETRY

Atomics International is participating in a review of microscopic cross section data for neutron dosimetry by reviewing cross section data for the n,γ and n,α reactions in Cu^{63} . Two evaluated data sets, ENDF/B and the SAND library, for these reactions have been graphically compared (Figure 4) and significant differences are evident. Experimental data for these reactions are being accumulated to help resolve the differences.

IV. EVALUATION OF EFFORT TO DATE

The completion of the cross section data evaluation for the tungsten isotopes will provide users of the ENDF/B data file an improved set of information.

The processing of experimental data for tungsten and its isotopes, although completed, does not provide adequate definition of the partial cross sections above 4 Mev.

Additional effort has been required in order to define theoretical cross sections where experimental data is clearly lacking. Semi-empirical techniques have yielded the n,γ ; n,p ; n,np ; and n,α as well as the $n,2n$ and $n,3n$ excitation functions. These evaluations are consistent with the few available experimental data points for these reactions. The level of confidence in the prescribed curves remains marginal until additional corroborative experimental data become available.

Figure 4. Comparison of ENDF/B and SAND Data for Various Energy Ranges

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Participants in the GEDANKEN calculations have not been responsive in examining and commenting on the results of these calculations. Additional urging is required.

The successful implementation of the ENDF/B Version 3 data files and the necessary processing codes will now permit progress in the Phase II calculations and analyses. Thus validation of the ENDF/B-3 data files as useful for LMFBR Design application can be rapidly accomplished.

V. NEXT REPORT PERIOD ACTIVITIES

The evaluation, for the ENDF/B-4 file of the tungsten and europium isotopes will be continued. Primary emphasis will be placed on obtaining the inelastic cross sections in the Mev range, and constructing the final ENDF/B data files for the tungsten isotopes. The evaluation of europium cross sections using the JUPITOR code will be completed. Europium-151 and -153 are odd-odd isotopes not directly amenable to conventional calculation; furthermore Eu^{151} and Eu^{153} are somewhat dissimilar nuclides. The levels of Eu^{153} may in part be analyzed as rotational structure, whereas Eu^{151} possesses no rotational structure and must be otherwise analyzed. The paucity of experimental data will force an extensive reliance on the theoretical calculation.

The JUPITOR code will be updated to remove a few programming errors as determined by H. Rebel and G. W. Schweimer at the Karlsruhe Laboratory.

Phase II calculations for the CSEWG benchmark assemblies assigned to Atomics International will be completed.

Construction of neutron cross section library for use with the VIM Monte Carlo code will continue. Upon successful completion of this task, the VIM code will be used to calculate several CSEWG benchmark assemblies.

The efforts associated with the testing of ENDF/B shielding data will continue. Shielding benchmarks will be established and the methodology for their calculation advanced.

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DATA SOURCES FOR FIGURES 1 THROUGH 3

(n, α)

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(n,p)

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(n,n,p)

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Program: Reactor Physics		
AEC Task: 8 - Theory and Computation for Reactor Analysis and Design		
Project Manager: H. A. Morewitz		
Reporting Period: January-March 1972	189A: 13239	
General Order: 09003	Subaccount: 14320	Category:

Principal Investigators: L. B. Levitt, R. C. Lewis, J. Spanier

I. PROJECT OBJECTIVES

The basic objective of this project is to advance the state of the art of Monte Carlo calculations to the analysis of fast reactors and fast critical assemblies. This development is needed, so that experimental programs, as well as design studies, may be optimized through the use of computer programs capable of describing complicated geometric configurations while calculating the effects of localized perturbations. These objectives will manifest themselves in the ultimate development of fast reactor criticality codes and specialized codes to treat perturbations due to changes in temperature, density, and material composition. The objectives of this project are related to Tasks 9-3-3 and 9-4-5 outlined in WASH-1109, "LMFBR Program Plan, Volume 9, Physics."

II. MAJOR ACCOMPLISHMENTS DURING FISCAL YEAR 1972

The accuracy of the cross section probability table representation for cross sections in the unresolved resonance region was further established by testing the procedures for both fissile (Pu^{239}) and fertile (U^{238}) nuclides.

III. PROGRESS DURING REPORT PERIOD

Coding of the preliminary version of the source free, infinite medium, pure scattering flux program IMPS was completed. IMPS calculates adjoint scattering kernel matrices and infinite medium fluxes which will be used in the adjoint VIM code. The scattering kernels generated by the IMPS code conform to the requirement of implicit normalization when integrated over all upper energies, thus permitting unit weights to be used at each scattering collision in the adjoint

Monte Carlo random walks. They will be prepared in tabular or, if possible, analytical form for each distinct composition present in a critical assembly and form the basis for the adjoint Monte Carlo sampling procedures employed in VIM. The IMPS code is presently being checked out. The best procedure to deal with Placzek type oscillations of the infinite medium flux caused by the arbitrary insertion of a unit source at 10 Mev is now under investigation. One possibility is to add a $1/E$ slowing-down source to the flux between $E_{\max}$ (10 Mev) and $\alpha E_{\max}$. This procedure was carried out and the results are being studied. Another possibility would be to assume the 10 Mev cross sections apply above 10 Mev as well and compute the fluxes in such a way as to include scatterings from these energies down to energies below 10 Mev.

The IMPS code appears to be generating correct infinite medium fluxes for medium weight isotopes such as sodium and carbon. Difficulties in the calculation of infinite medium fluxes for heavy isotopes using the IMPS code have been traced to a need for an ultra-fine energy grid. As a result of these difficulties, a Monte Carlo version for the calculation of infinite medium fluxes and scattering kernels for a pure scattering medium has been written. Convergence should be rapid for heavy isotopes due to the large number of collisions per history and the small energy loss per collision. The extreme fineness of the grid required to perform the necessary integrations analytically would increase not only the storage requirements but the computation time to where it could easily exceed that required to obtain converged results by Monte Carlo. The situation is analogous to that encountered in developing the MCL code for calculating mean chord lengths, wherein it was found that an analytically determined set of source coordinates would have to be much larger than one obtained by random sampling, since convergence in the latter case was very rapid. The problem with IMPS is not to obtain unusually high accuracy but to avoid an accumulation of consistent error as the integration proceeds. The Monte Carlo approach avoids such a buildup of error, replacing it with statistical fluctuation which should damp out rapidly. Preliminary indications of computation time show 50,000 collisions processed in 3.1 sec. This would lead to well-converged results in about 10 min of computer time.

Coding of the adjoint Monte Carlo version of the VIM code continues. The main flow program, including weight modifications at boundary crossings, and a subroutine which computes these weight factors has been programmed.

The adjoint Monte Carlo Code STRIFE is being used as a skeleton model for implementation of the adjoint Monte Carlo techniques to be used in VIM. STRIFE was written to compare the performance of the transformed adjoint method described in AI-AEC-12774 with that of an analog adjoint method, as well as to explore the range of problems in which an adjoint method may be preferable to a direct method. Geometrically, it solves a two-region infinite slab lattice. All facets of adjoint Monte Carlo sampling can be tested within this framework as development proceeds, since the geometric aspects of Monte Carlo are the same in both direct and adjoint sampling schemes. Points of similarity as well as differences between STRIFE and the new adjoint version of VIM are being pinpointed as an aid in formulating the latter.

Procedures for terminal tallying in the adjoint mode for a fission source are being developed. These procedures will actually dictate the best method for storing a spatial fixed source of fission neutrons generated in a direct Monte Carlo calculation. Since such a source may involve as many as 50,000 sets of (X, Y, Z) coordinates, a method of packing and indexing is being developed to permit storage in the high speed memory. Present thinking is to set up a spatial matrix fine enough to assume all particles whose coordinates lie in a particular matrix element may be started at the center of the element. Storage will be kept to a minimum by packing matrix identification indices and source densities in integer form in a single computer word and then storing them linearly, eliminating those with density zero from the array. In the adjoint tallying, this matrix will be searched in an optimized way to determine whether the particle could terminate at the coordinates of the current collision point.

The random number generator for the CDC 7600 has been modified so that naive correlation can be used in the VIM codes. Checkout of this modification of the random number generator has been successfully completed.

A paper describing the probability table method for the unresolved resonance region has been written and submitted to the Fourth International Conference on Reactor Shielding to be held in October of this year.

Checkout of new capabilities which were added to VIM has been completed. These include: (1) rectangular cell capability, (2) ability to use an external

source produced by the ANISN transport code, (3) ability to perform an uncollided flux calculation using an external source, and (4) angular biasing of the source for purposes of investigating streaming through sodium channels.

IV. EVALUATION OF EFFORT TO DATE

Development of the adjoint Monte Carlo version of VIM and of the IMPS code in both numerical and Monte Carlo form is making steady progress. An understanding of the energy dependent behavior of the scattering kernels is emerging from the results to date, and should lead to an efficient scheme for preparing the matrices to be used by VIM.

The generalization of the Levitt-Spanier method of adjoint Monte Carlo sampling to include all types of scattering with energy dependent cross sections in all regions has been essentially achieved in the outline of an adjoint Monte Carlo version of VIM. It is not clear that a working code and an adjoint library generator (IMPS) can be developed wholly within the framework of this method. The use of Monte Carlo in the IMPS code will greatly expedite the development of the adjoint VIM code by shortening the development time for library preparation. Thus, calculation of central reactivity worths and control rod worths in a heterogeneous representation using full point cross section detail is that much closer to realization.

V. NEXT REPORT PERIOD ACTIVITIES

Development of the code to generate infinite medium spectra and integrated scattering matrices will be completed. Development of the adjoint Monte Carlo version of VIM will continue. A journal article on the Probability Table Method will be submitted to Nuclear Science and Engineering.

Program: Civilian Power, LMFBR

AEC Task: 9 - Fast Reactor Physics Experiments

Project Manager: H. A. Morewitz

Reporting Period: January-March 1972

189A: 13482

General Order: 09004

Subaccount: 14610

Category: 04-40-01-02.1

Principal Investigators: T. H. Springer, R. J. Tuttle, L. A. Mountford,
J. M. Otter

I. PROJECT OBJECTIVES

The objectives of this project are to: (1) determine reactivity, temperature, and Doppler coefficients of various materials in a variety of fast-reactor spectra; (2) investigate effects due to different geometric arrangements of such materials, including the variation of the surface-to-mass ratio; (3) provide experimental results for checking theoretical predictions; and (4) develop improved experimental and theoretical methods for treatment of problems in the field of fast reactor physics.

II. MAJOR ACCOMPLISHMENTS DURING FISCAL YEAR 1972

Measurements of the reactivity worth of nearly full-size control rod mockups, placed in the sodium voided pin-type test region of Core 20, were completed during the first quarterly period. Different material types and configurations were investigated. In the multipin-type design, both Ta metal and B_4C powder were used to obtain an intercomparison of worths under identical surface area conditions. Using Ta metal, two different geometrical arrangements were studied, one a multipin cluster and another a solid cylinder. These reactivity worth measurements were conducted as a function of axial position in a 32-in.-high UO_2 -fueled (20% U^{235}), pin-type core, abutted on each end by a 12-in.-high depleted- UO_2 -fueled, pin-type blanket.

Generalized response functions for thermoluminescent dosimeter (TLD) detectors imbedded in any material for which the density, atomic number and mass, mean ionization and excitation potential, and photon interaction cross sections can be specified were generated for use in the analysis of TLD measurements. These

measurements are used to predict the heating rates which would take place in the same type of material located in the operating LMFBR which the critical assembly is simulating.

The fuel in each of the twelve standard, 271-pin, sodium-voided, hexagonal fuel elements that make up the test region of Core 20 was unloaded, the fuel elements were filled with sodium, the fuel was re-inserted, and criticality for the sodium-filled version (Core 20N) of an LMFBR was achieved. The fraction of the available volume that can be occupied by sodium in each element was, in all cases, in excess of 0.92.

An evaluation of all currently available data pertaining to delayed neutron parameters was completed. The recommended values for the absolute delayed neutron yields are given in Table 2.

TABLE 2
RECOMMENDED ABSOLUTE DELAYED
NEUTRON YIELDS

Isotope	Yield (delayed neutrons/fission)
Th ²³²	0.0545 ± 0.0021 (+3.9%)
U ²³³	0.00697 ± 0.00016 (±2.3%)
U ²³⁵	0.01682 ± 0.00039 (±2.3%)
U ²³⁸	0.0438 ± 0.0013 (±2.9%)
Pu ²³⁹	0.00658 ± 0.00011 (±1.6%)
Pu ²⁴⁰	0.0092 ± 0.0010 (±10.9%)
Pu ²⁴¹	0.0165 ± 0.0016 (±9.8%)
Pu ²⁴²	0.0155 ± 0.0050 (±31.4%)

A 2-dimensional diffusion-theory calculation (2DB code) was carried out for Core 20 which contained a sodium-voided version of the pin-type test region. The results of this calculation indicate that 2-D diffusion theory calculations will probably be adequate for analytically predicting the worths of the various full-scale, pin-type Ta and B₄C control rods whose worths were measured as a function of axial position along the axis of the test region.

III. PROGRESS DURING REPORT PERIOD

During this quarterly period, the filling with sodium of all twelve of the standard, 271-pin, hexagonal fuel elements making up the test region of Core 20 was completed. The elements were re-loaded with fuel and inserted into the core. Adjustments were made in the loading of the buffer region (both the transition and square-matrix portions) in order to match as closely as possible the material atom densities in the pin-type test region. Among other core changes, plate-type (2 in. by 2 in. by 0.25 in. and 0.50 in.), sodium-filled cans were added to the square-matrix portion of the buffer, and aluminum tubes were inserted into the transition region to occupy the nonstandard geometrical spaces. The completely loaded core (Core 20N) is shown in Figure 5, where a test element with a hexagonal through-tube (~ 2 in. across flats) is seen in the central hexagonal matrix space. The transition region, which is that volume outside of the seven fully hexagonal spaces and inside of the 9 by 9 square-drawer matrix space, also contains some 2 in. by 2 in. fuel and structural material placed in special stainless-steel drawer inserts.

Core 20N, whose pin-type test region simulates a sodium-cooled FBR, was made critical by adjustments in the thermal driver zone, and various operational parameters were measured and analyzed during this reporting period. A preliminary series of measurements of the reactivity worth of a large Ta control rod was carried out.

One additional analysis relative to the chemical purity of the B_4C material that was used in the 18-pin full-size control rod experiments previously conducted in the sodium-voided version (Core 20) of this core was carried out. The oxygen content of Samples 1 and 2 (See Quarterly Technical Progress Report, LMFBR Physics Programs, October-December 1971, AI-AEC-13019, Page 30) was found to average 805 and 1140 ppm, respectively.

B. DATA ANALYSIS

A re-analysis of the TLD measurements in Cores 17A and 20 (see Annual Technical Progress Report, LMFBR Physics Programs, GFY, AI-AEC-13005, p. 45) was conducted, using various assumed, but realistic, gamma ray spectra to which the TLD detectors were exposed. The measured heating rate was found

Figure 5. Stationary Half of Core 20N (Sodium-Filled)

to vary as much as 30%, depending upon which spectrum was judged to be correct. The results of this re-analysis are given in Table 3, where a range of values is listed to account for this spectrum dependence. In one case, the calculation and experiment are normalized on the basis of a calculated and measured fission rate. In the other, the normalization is based upon a calculated and measured capture rate in Ta.

TABLE 3
HEATING RATES IN CORES 17A AND 20
Heating Rate (ergs/sec-gm/fissions/sec-cm³ x 10⁻⁶)

	Core Gamma		Capture Gamma	Total
Fission Rate Normalization	Calculated	0.81	3.48	4.29
	Measured	1.69 to 2.43	2.59 to 3.66	4.23 to 6.09
Ta Absorption Rate Normalization	Calculated	1.96	6.52	8.48
	Measured	3.86 to 5.57	5.80 to 8.38	9.66 to 13.95

C. ANALYTICAL STUDIES

The technique for carrying out an evaluation of measured neutron spectra vis-a-vis calculated spectra was reviewed and various shortcomings were discussed. The technique currently underway consists of collapsing the "point-wise" data obtained in proton-recoil spectrometry into an appropriate set of energy groups (26, for example) that correspond to a calculated spectrum. The "measured" spectrum is then used in a perturbation calculation, along with a calculated adjoint flux and a calculated normalization integral. The calculated worths of selected samples are generated and compared to analytical reactivity worth values based on the calculated spectrum. The method appears to be adequate as a means for testing the correctness of the central real flux as obtained by analytical means.

Calculated and measured spectra for ZPR-6 Assemblies 6, 6A, and 7 and for ZPR-9 Assemblies 25 and 26 were reviewed in preparation for inclusion in this evaluation task. Evaluations of spectrum data for FCEL Cores 17, 17P, and 18 have already been completed in preliminary form.

A review of the literature and evaluation of all available delayed neutron data was completed during this reporting period. The review produced data from eleven separate experiments on yields for eight isotopes of direct interest in reactor physics. Additional data at high energies was also compiled. Evaluation of these measurements resulted in recommended values with lower estimated uncertainties than those currently used. The measurements involved sufficient variety in methods that the effect of systematic errors was reduced by combination of the separate values.

Both relative and absolute yield measurements were included in this analysis. The relative measurements avoided or reduced systematic errors involved in the calibration and energy dependence of neutron detectors. While the results of relative measurements were reported as yields relative to a single isotope chosen as a standard, for this analysis each set was converted to an interrelated set of ratios, with relative yields for each isotope measured with respect to every other isotope. This process reflected the symmetry in relative measurements and prevented the choice of standard values from unduly influencing the results of the evaluation.

All measured yields at low energies (0 to 4 Mev) were in reasonably good agreement and were consistent with an assumption that the yield is independent of energy in this range. At high energy (14 to 15 Mev), there were two groups of measurements, one group showing an increase of about a factor of two over the low energy yields, while the other group showed a decrease to about one-half the low energy yields. A calculation that was performed elsewhere and that was based on measured fission product yields supported the lower delayed neutron yields, as did the measurements of Krick and Evans (see Nucl. Sci. Eng., 47, 311, 1972) of LASL. The latter work covered the energy range up to about 7 Mev. Krick and Evans related the change in delayed neutron yields to the onset of the $(n, n'f)$ and $n, 2nf$ reactions.

On the basis of the former work, the variation of yield with energy was calculated for U^{235} and U^{238} at this laboratory. The calculation was based on assumed partial fission cross sections and the yield for the fissioning nucleus. In the case of U^{235} , the fissioning nucleus is U^{236*} , U^{235*} , or U^{234*} for first-, second-, and third-chance fission. Similarly, for U^{238} , the fissioning nucleus

is U^{239*} , U^{238*} , or U^{237*} . The calculated yields were in moderately good agreement with the measurements and showed a decrease at higher energies; modification of the assumed cross sections is expected to lead to better agreement.

Based on this detailed evaluation and review, a recommended set of values for absolute delayed neutron yields was derived and was shown in Table 2. The fractional uncertainties for Pu^{240} , Pu^{241} , and Pu^{242} are relatively large because only one measurement is available for each.

The preparation of input data for the initial 2DB calculation of Core 20 was completed. The calculation employed cross section sets in 10 energy groups which were obtained by collapsing, via the ANISN code, the standard 35 group cross section set used for 1-D diffusion calculations for Core 20. Flux-weighted cross section sets were obtained for each material of each region in radial and axial ANISN calculations, the axial cases being confined to the test region only. The cross section sets that were determined by axial ANISN case were used for all test region subregions, for the buffer region blanket, and for structural materials lying outside of the core and blanket. The cross section sets as determined by the radial ANISN case were used for all other regions. A total of 59 cross section sets were used in the 23 zones that consisted of different compositions. The mesh intervals constituted a 56 by 76 array (radial by axial), the number of mesh intervals being determined primarily by the spacing required in the thermal driver and polyethylene reflector zones where the neutron mean free path is short.

The selection of appropriate mesh spacings was based upon intercomparisons of CAESAR 1D diffusion calculations and ANISN 1D transport calculations (see Quarterly Technical Progress Report, LMFBR Physics Programs, October-December 1971, AI-AEC-13019). Spacings of 0.75 cm and 1.0 cm were used in the thermal driver and thermal reflector regions, respectively, and spacings of 1.0 cm (or less, if necessary to have at least three intervals per region) were chosen for the inner core regions. A spacing of 3.5 cm was assigned to the empty matrix regions outside of the radial reflector.

The value of LAST, the parameter that designates the size of the 2DB calculation, was 50K for the initial computer run. This version of 2DB has a maximum permissible value of LAST of 70K. The preliminary calculated value for

Figure 6. 18-Pin Sodium-Filled Control Rod and Follower

k_{eff} was 0.981, a value that compares favorably with 0.980 and 1.0206 for the ID CAESAR and ANISN calculations, respectively. The latter values are only approximations since they cannot correctly represent the strongly two-dimensional features of the problem. The experimental k_{eff} value was 0.9979 ± 0.0007 , including corrections for the difference between theoretical model and the as-built assembly. Currently, several cases involving the 50K problem size are expected to be run. These cases will give real and adjoint fluxes which can be compared to fission counter, source importance, and proton-recoil spectrum experiments. The real and adjoint fluxes will be used in a 2-D perturbation calculation to analytically determine the worths of several small samples whose worths were measured by the oscillator technique. In addition, these calculations will be used to obtain comparisons with the full-size control rod experiments that were reported in the July-September 1971 Quarterly Report (AI-AEC-13013).

D. PREPARATIONS FOR FORTHCOMING EXPERIMENTS

The fabrication of a nearly full-size, 18-pin, control-rod mockup that is capable of being filled with sodium in the region around and between pins was completed. This control rod assembly has a hexagonal housing which fits into the central hexagonal matrix space. Each of the 18 stainless-steel tubes (see Figure 6) was filled with 0.156-in.-diam Ta rodlets which, when clustered together, simulated an approximately 5/8-in.-diam by 34-in.-long Ta rod. The control rod mockup assembly was filled with sodium to 96% of the available volume for sodium. The control-rod mockup was joined to a hexagonal sodium-filled follower that moves into the core when the control-rod assembly is withdrawn. Thus, this control-rod mockup arrangement closely simulates the conditions that exist in an LMFBR: sodium occupies the control rod channel when the control rod is retracted.

A series of stainless-steel inserts that will be used, in the future, to fill the nonstandard-shaped spaces in the transition region of the core were fabricated. These inserts consist of 36-in.-long, stainless-steel tubes whose cross sections have shapes of: (1) a triangle with one right angle, (2) an isosceles triangle, (3) a 3/4-in.-diam circle, and (4) a 1/2-in. by 1-1/2-in. rectangle.

The U^{235} , Pu^{239} , and U^{nat} solid-state fission detectors were checked out and alpha counted.

IV. EVALUATION OF EFFORT TO DATE

The series of experiments dealing with the reactivity worths of nearly full-size control rod mockups as a function of axial position in the pin-type, UO_2 -fueled, sodium-voided test region of Core 20 was the first of this type in an LMFBR type of spectrum, and the technique appears to be very useful in evaluating various control-rod materials and designs, particularly on a relative basis. The worths of the rods are sufficiently small that measurements can be made by continuous-drive techniques; consequently, time-consuming fuel manipulations and problems concerning more difficult data interpretations are avoided. One of the major tasks remaining in this area is that of interrelating the results with other critical assemblies and normalizing the measurements to full-scale LMFBR control-rod designs. By the use of normalization integrals, which will be obtained by the 2-D calculational now underway, comparisons will be made with previous smaller-scale experiments in Core 17 and with the larger-scale static measurements in ZPPR.

Efforts were continued during this reporting period in the area of the measurement of heating rates in Ta by means of TLD detector techniques. Reasonably good correlations have been found between the measured dose rates in the FCEL reactor and ZPR-6; consequently, the method is consistent and holds promise as an effective means for determining heating rates in zero-power critical assemblies. Some fundamental improvements have been made on this program in generalizing the calculation of response functions as they apply to TLD measurements in critical assemblies. These response functions will be of general use to experimenters in the analysis of the experimental results.

The re-analysis that was discussed above and that takes into account the spectrum dependence of the TLD data interpretation indicates those areas in which improvements in calculational and experimental comparisons can be made. The absolute calibration of TLD detectors, a task which is currently underway on another program can potentially resolve present discrepancies and render the TLD detector methods a very versatile, rapid, and inexpensive way for absolute heating rate determinations.

The extensive and systematic evaluation of delayed neutron parameters has indicated that one of the major contributors to the so-called central reactivity

worth discrepancy has been the use of incorrect values for β_{eff} . The results of this evaluation imply that corrections of the order of 6 or 7% in the direction of generally improving the agreement between calculation and experiment are justified.

V. NEXT REPORT PERIOD ACTIVITIES

The measurements of the reactivity worth of the 18-pin, sodium-filled, Ta control-rod assembly will be completed along with identical measurements for a B_4C rod of the same type.

Spectrum characterization experiments consisting of selected small sample worth measurements, proton-recoil spectrum determinations, neutron importance traverses, and selected fission rate measurements will be undertaken and completed.

An experiment to obtain an order-of-magnitude value for the reactivity effects of progressive sodium voiding will be performed, and blanket optimization studies will be initiated.

In the area of analysis, a series of 2DB calculations will be completed in order to obtain an intercomparison with measurement of the reactivity worth of full-size control rod assemblies.

As a result of a budget reduction invoked during March, the previously scheduled work on the Fast Reactor Experiments Compilation and the spectrum evaluation tasks will be deferred until GFY 1973, and only a portion of the originally scheduled series of blanket optimization experiments will be undertaken.

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