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QUARTERLY REPORT
October 1-December 31, 1963

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**PHILLIPS
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ATOMIC ENERGY DIVISION

**NATIONAL REACTOR TESTING STATION
US ATOMIC ENERGY COMMISSION**

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MATERIALS TESTING REACTOR-ENGINEERING TEST REACTOR TECHNICAL BRANCHES QUARTERLY REPORT

OCTOBER 1 - DECEMBER 31, 1963

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ERRATA FOR IDO-16977

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The caption for Figure II-12 should read:

Fig. II-12 $S(\kappa^2, \beta)$ vs β for H_2O on Nelkin Model with $M_H' = 18.2$.

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The caption for Figure II-16 should read:

Fig. II-16 Partial differential scattering cross section of CD_4 .

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SUMMARY

Reactor fuel material consisting of U_3Si particles embedded in an aluminum matrix have been examined with an electron microprobe to determine reaction between fuel particles and matrix. Partial reaction was observed with no observable tendency for the uranium or silicon to diffuse outward into the matrix.

A reevaluation of techniques used in ARMF reactivity measurements included (1) study of temperature control of canal water, (2) effect of rod and sample positions on accuracy of the measurements, and (3) an analysis of reactivity measurements where self-shielding is involved.

Resonance absorption integrals are reported for natural dysprosium and hafnium for their major isotopes.

Expanded investigations of possible Xe-135 oscillations in the ATR indicate stability against any reasonable perturbation.

Calculations with the AIM-6 diffusion code indicate that reactivity measurements in the ARMF cadmium tube need to be corrected for non-1/E flux and for an adjoint flux that varies with energy.

The Reactor Physics Section continued the practice of calculating fuel loadings for MTR Operations during 1963. A discussion of the year's activity and the results of the calculations are presented.

A third large sample of Pa^{233} was chemically separated from irradiated thorium and prepared in a form for use in the MTR Fast Chopper to determine the total cross section as a function of energy. The sample consisted of 787.6 mg of $Pa_2^{233}O_5$ (14,000 Curies); the cross section at thermal energy was measured as 56^{+3}_{-5} barns.

Measurements of η (the number of neutrons emitted per neutron absorbed) for U^{233} and Pu^{239} were completed for incident neutron energies of 0.057 eV; the values obtained were 2.296 and 2.041 for U^{233} and Pu^{239} , respectively.

The half-life of the longer Ho^{166} isomer was determined to be 1200 ± 150 years, based upon mass analysis of a highly irradiated Ho sample coupled with measurement of the absolute disintegration rate of the Ho^{166} .

The Placzek first moment condition has been calculated for the available scattering law data to see if any of the data obey it. The elastic scattering from powdered beryllium has been measured and shows the expected structure. The scattering profiles from gaseous, liquid, and solid ammonia show the effect of rotations, hindered rotations, and crystalline binding.

Better fits to the water scattering data were obtained with a theory that assumed that five types of water molecules existed, each type in cluster.

A program has been written for the scattering of slow neutrons from CH₄ and CD₄. The CD₄ results show the pronounced effects of coherent scattering. Mathematical analysis shows that in the one eV and above range Doppler broadening is insensitive to the model used.

Work on the Low Energy Nuclear Physics Program has included an attempt to observe gamma-ray spectra of Yb-178 and Gd-162. No evidence of these nuclides was indicated. Branching ratios and internal conversion coefficients for the 145-keV transition in the decay of Ce-141 are reported. The thermal-neutron activation cross section for Mn-54 has been measured and a value reported.

The August 1963 TRIGA Proof Test has indicated the effects of neutron and gamma dose rates on the induced electrical transients in various types of pressure transducers. The preliminary results demonstrate the importance of using proper circuitry and the difficulty of extrapolating the results to other radiation fields. Some of the types tested are suitable for "near-core" measurements in the Spert and STEP destructive tests.

An analog computer study of the In-Pile Fatigue Test temperature control of the test sample has resulted in a controller system design capable of maintaining the sample temperature within $\pm 5^{\circ}\text{C}$. Control is achieved by two heaters in the temperature controlling gas stream. The heaters in turn are controlled by two 3-mode controllers in a two loop feedback system.

A fuel plate spacing gage has been developed which operates on the eddy current loading of a sensing coil by the fuel plate. It measures spacings of 68 to 90 mils over a 50-inch length.

Substantial progress has been made in the writing of new 7040 programs and in the conversion of existing 7090 codes. The 650 Simulator is in operation and a number of programs have been put into service on it.

MTR-ETR TECHNICAL BRANCHES QUARTERLY REPORT
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I. REACTOR PHYSICS AND ENGINEERING

1. REACTOR FUELS AND MATERIALS DEVELOPMENT

(W. C. Francis)

1.1 Microprobe Study of the Reaction of U₃Si Fuel Particles with an Aluminum Matrix at 600°C (W. F. Zelezny, G. W. Gibson, M. J. Graber, B. Carlson)

1.11 Introduction. Attention has been turned to the possible use of several intermetallic compounds of uranium (eg, UAl₂, UAl₃, UAl₄, and U₃Si) as a nuclear fuel. Among the advantages which these intermetallic compounds appear to offer, in comparison to the uranium oxides, are: higher uranium density (U₃Si), higher thermal conductivity, and low neutron absorption combined with reasonably high melting points. The behavior of these fuels under irradiation at elevated temperatures will determine whether their apparent advantages can be capitalized upon. The present study was undertaken to determine what effects exposure to elevated temperatures would have on one of these fuels, U₃Si particles dispersed in an aluminum matrix.

1.12 Materials. This study was carried out using sample fuel plates manufactured by the Sylcor Division of Sylvania Electric Products, Inc., according to the following procedure [1]. The plates had APM claddings (20 mils thick) and APM frames. The 47 weight percent U₃Si aluminum powder cores were eight mils in thickness. The U₃Si was produced by arc melting uranium and silicon metal. This arc melted material was homogenized at 850°C for one week. The U₃Si was then ground with a mortar and pestle to minus 100 mesh powder, but resisted further grinding to a finer particle size due to its ductility. Compacts of this material and 1100 aluminum powder were pressed at 45 tsi in a 3/4- by 1-3/4-inch die. The green compacts were outgassed at 600°C, cleaned by wire brushing and assembled with a picture frame and cover plate for pressure bonding. After pressure bonding the plates were hot-rolled to the finished size.

1.13 Experimental. Strips were cut from manufactured plates in the "as-received" condition, were sealed in glass tubes, and were annealed for periods of 24, 90, and 163 hours at 600°C. The specimens were sectioned, mounted, and polished for examination. The principal mode of examination used a Materials Analysis Co. MAC Model 400 electron probe microanalyzer. X-ray diffraction also was used.

1.14 Observations and Results. Metallographic observation at lower magnification (eg. 50 x) of polished, unetched cross sections of the U₃Si fuel plates in the as-received condition revealed the moderately coarse-grained fuel particles shown in Figure I-1.

At higher magnification the larger fuel particles showed two regions (see Figure I-2A): a dark, central area surrounded by an outer region which was lighter in appearance. This appearance is suggestive of a diffusion or "reaction" zone, involving the fuel particle and the aluminum matrix. The appearance of the smaller particles was identical with that of the outer region of the larger particles (Figure I-2A).

The results of one-dimensional microprobe scans (giving essentially the concentration profiles of U, Si, and Al across the particle) are shown in Figures

Fig. I-1 Photomicrograph of cross section of as-received U_3Si -Al fuel plate.

I-2B, C, and D. It is of particular interest to note that within each region there seems to be little or no variation in concentration. Similar results are indicated by the two-dimensional scans. Figure I-2E, F, G, and H, are of the same region with the signals being displayed on the screen of a cathode ray oscilloscope. The specimen current presentation of Figure I-2E is essentially the negative of Figure I-2A, showing the region of the highest uranium concentration as white and the region of intermediate uranium concentration as gray, while the aluminum matrix appears black. It should be noted that the smaller particles evident in Figure I-2A and E exhibit an appearance identical with that of the outer region of the large particle (which may be taken as an indication that the regions were of similar composition). This could be accounted for by assuming that the particles were originally U_3Si , and had been altered by the inward diffusion of Al from the matrix. The amount of diffusion was the same in each case, thus, resulting in altered regions having the same final composition.

Careful observation of Figure I-2F shows the same uranium concentration distribution already indicated in Figure I-2B; that is, a higher uranium concentration in the central region of the particle, a lower concentration in the outer region, and a completely negligible (at least below the limits of detection by the microprobe) concentration of uranium in the aluminum matrix.

The observations mentioned so far are qualitative or, at best, semi-quantitative in nature. Quantitative measurements of the uranium, silicon, and aluminum concentrations in the two regions of the fuel particles were obtained by making timed counts of the $U\ M\alpha$, $Si\ K\alpha$, and $Al\ K\alpha$ X-ray radiations while the electron beam of the microprobe was held stationary with respect to the specimen. These counts were compared with the counts obtained under identical excitation conditions from the pure elements uranium, silicon, and aluminum

Fig. I-2. As-received U_3Si -Al fuel plate.

(A) Photomicrograph (500x) showing a partially reacted large particle of U_3Si and a completely reacted smaller particle. The darker region is the unchanged U_3Si , the regions of intermediate gray represent the "reaction zone", and the lightest area is the aluminum matrix. The straight, black marks are the contamination left by the electron beam of the microprobe. Polished, unetched.

Strip-chart records: (B) $U M\alpha$, (C) $Si K\alpha$, and (D) $Al K\alpha$ x-ray signal response during probe traverse of specimen.

Photographs of oscilloscope screen showing: (E) specimen current, and (F) $U M\alpha$, (G) $Si K\alpha$, and (H) $Al K\alpha$ x-ray response during two-dimensional probe scans of same specimen area as shown in (A).

with the appropriate absorption corrections being made to place them on a quantitative basis.

These quantitative analyses by means of the microprobe indicate a composition in the neighborhood of 95.2 weight percent uranium and 4.8 weight percent silicon for the central region of the large fuel particles. This compares with 96.2 weight percent uranium and 3.8 weight percent silicon for U_3Si of exactly stoichiometric composition. That is to say, the indications of the microprobe analysis are that the U_3Si particles used as starting material were slightly higher in silicon content than stoichiometric U_3Si .

The outer regions (the "reaction zones") of the fuel particles yielded analyses in the neighborhood of 70.5 weight percent uranium, 25 weight percent aluminum, and 4.5 weight percent silicon. The composition for stoichiometric UAl_3 is 74.6 weight percent uranium and 25.4 weight percent aluminum, while that for stoichiometric UAl_4 is 68.8 weight percent uranium and 31.2 weight percent aluminum. Thus, the composition of the region containing uranium, silicon, and aluminum is much closer to the proportions of UAl_3 than UAl_4 . This point will be discussed further in connection with x-ray diffraction data from both the as-received and annealed fuel plates.

1.15 Annealed Fuel Plates. Figure I-3A shows a photomicrograph of a typical fuel particle in a fuel plate which had been annealed for 90 hours at 600°C. Comparison with Figure I-2A will show that the entire particle now has the same appearance as the smaller particles and the outer regions of the larger particles in the as-received fuel plate.

The one-dimensional scans of the particle, Figures I-3B, C, and D, show that the composition is essentially uniform throughout the particle.

Quantitative analyses of the fuel plates annealed at 600°C for 24 hours, for 90 hours, and for 163 hours showed that the composition of the fuel particles in all cases was, within experimental error, that listed above for the "reaction zone" in the as-received fuel particles, and that heating beyond the 24 hour period did not produce any noticeable change in composition.

1.16 X-ray Diffraction Data. X-ray diffraction of the as-received and of all the heated specimens revealed the presence of UAl_3 , plus a relatively weak diffraction pattern of an unidentified phase (or phases). The unidentified diffraction pattern was slightly sharper in the sample heated for 163 hours. Since the composition of the fuel particles in this sample remained unchanged from the compositions of the samples heated for 24 and 90 hours, it appears that the more pronounced diffraction pattern from the new, unidentified phase could be due to the growth of the individual crystallite size (accompanied by a decrease in the number of crystallites) with continued heating of the new phase without an increase in the percentage present. The diffraction pattern of UAl_4 was not observed in any of the annealed specimens.

1.17 Conclusions. The above observations are consistent with the following picture of events taking place during heating at 600°C in a system of U_3Si particles embedded in an aluminum matrix:

- (1) Aluminum readily diffuses from the aluminum matrix into the U_3Si particle.

Fig. 1-3. U_3Si -Al fuel plate after annealing for 90 hours at $600^\circ C$.

(A) Photomicrograph (500x) showing the completely reacted U_3Si particles. The very light area surrounding the particles is the aluminum matrix, while the speckled area to the right is the cladding. Polished, unetched.

Strip-chart records: (B) U $M\alpha$, (C) Si $K\alpha$, and (D) Al $K\alpha$ x-ray signal response during probe traverse of particle.

- (2) The altered fuel particle resulting from the inward diffusion of aluminum consists of UAl_3 plus an unidentified phase (or phases).
- (3) The altered fuel particle is stable (possibly due to the presence of silicon [*]) since extended periods of heating at 600°C do not produce additional inward diffusion of aluminum to convert the UAl_3 to UAl_4 .
- (4) No diffusion of uranium or silicon from the fuel particle outward into the aluminum matrix is observed.

2. REACTOR EXPERIMENTS (E. Fast)

2.1 Resonance Integral Measurements (J. J. Scoville, E. Fast)

2.11 Dysprosium. Tentative values of the infinitely dilute resonance integrals for natural dysprosium and its principal isotopes are given in Table I-1. The measurements were obtained in ARMF-II (except where indicated otherwise) where the response to resonance absorption was approximately four times greater than in the RMF and ARMF-I. The values are considered tentative pending further study of the interpretation of reactivity effects and results of measurements on improved gold standards. The samples were Dy_2O_3 diluted in aluminum powder, of hollow cylindrical geometry (0.86-inch diameter and 0.030-inch wall thickness) and were measured under a 0.020-inch-thick cylindrical cadmium shield. Details of procedure and interpretation of measurements are given elsewhere [2].

Little previous work appears to have been reported on dysprosium. Two values obtained by activation of Dy-164 were reported [3] in 1960. Calculated values were reported more recently by ANL [4] which are recorded in Table I-1 for comparison. The Dy-164 bound level calculation agrees well with our measurements. However, in every other case the calculated values are significantly lower indicating uncertainties in parameters and unresolved resonances. Our calculated values did not consider bound levels.

Three independent determinations of the resonance integral of natural dysprosium are given. The RMF-ARMF-I value is a revised value obtained by correcting an inadvertent error, in a previously reported result [2, 5]. A third value was obtained from the results of measurement of the principal isotopes, neglecting contributions from Dy-156 and Dy-158 which should be negligible.

[*] When silicon is added to uranium-aluminum alloys, in amounts of 1.5 percent or more, the peritectic reaction which normally occurs at 750°C is suppressed and UAl_3 rather than UAl_4 is retained as the stable phase: W. C. Thurber and R. J. Beaver, "Silicon-Modified Uranium-Aluminum Alloys for Foreign Reactor Applications," Nuclear Metallurgy, Volume V, AIME 1958.

TABLE I-1
DYSPROSIUM RESONANCE INTEGRALS

Isotope	Resonance Integrals, barns		
	Measured	Calculated	Literature [3, 4]
160	1160 $\pm$ 130	----	-----
161	1670 $\pm$ 167	1044	947.8 (calc.)
162	3320 $\pm$ 400	1284	2610.0 (calc.)
163	1960 $\pm$ 176	1236	1244.8 (calc.)
164	377 $\pm$ 34	1269	382.1 (calc.)
			406.0 (calc.)
			420 $\pm$ 50 (act.)
			480 $\pm$ 33 (act.)
<u>Natural</u>			
RMF & ARMF-I	1810 $\pm$ 440		
ARMF-II	1970 $\pm$ 180		
From Isotopes above		1240	1264 (calc.)
Weighted Average	1840 $\pm$ 180		

2.12 Hafnium. Resonance integrals of hafnium and hafnium isotopes as Hf_2O_3 in aluminum, obtained by the same procedure, are given in Table I-2. The large uncertainties are due primarily to the low purity of the available separated isotopes. Within these uncertainties a reasonable agreement with calculated values was obtained. By a modification of the procedure of Lane, Nordheim, and Sampson [6], a correction due to particle self-shielding was calculated for Hf-178. A similar correction appeared to be negligible for the other samples.

2.2 Diffusion Theory Studies of the ARMF Neutron Flux and Adjoint Function (W. K. Foell)

Before the results of reactivity measurements in the Advanced Reactivity Measurement Facility (ARMF) can be correctly interpreted, it is necessary to obtain a good estimate of the spatial and energy dependence of both the real and adjoint flux functions. In order to obtain these functions, a cylindrical

TABLE I-2
HAFNIUM RESONANCE INTEGRALS

<u>Isotopes</u>	<u>Resonance Integrals in barns</u>		
	<u>Measured</u>	<u>Calculated</u>	<u>Literature [3]</u>
176	(≈ 900)	----	-----
177	8090 ± 1800	7340	7476 (calc.)
178	1610 ± 300	1810	2999 (calc.)
179	500 ± 200	397	429.7 (calc.)
180	(≈ 18)	28	33.3 (calc.)
			21.8 (act.)
<u>Natural</u>			
From Isotopes	(2045)	2050	2258 (calc.)
ARMF-I	2150 ± 480	----	2900 (abs.)
ARMF-II	2390 ± 250		
Weighted Ave.	$2340 \pm \approx 250$		≈ 2000 (abs.)

representation of the ARMF was studied by means of the AIM-6, one-dimensional diffusion code [7].

The primary purpose of these calculations was to study the lethargy dependence of the real and adjoint fluxes inside the cadmium tube in an ARMF lattice position. Resonance integral measurements are normally made in this location. Core diagrams of the ARMF-I and ARMF-II are shown in Figures I-4A and I-4B. Even though the lattice position is off center in the ARMF-I, it was felt that a one-dimensional cylindrical model of the reactor with the cadmium tube located in the center would provide a reasonable estimate of the fluxes. It is clear, however, that a one-dimensional model more closely corresponds to the symmetrical ARMF-II loading. There are two-dimensional diffusion codes available which would allow a more accurate geometrical description of these reactors but none of them provides for the detailed group structure and adjoint flux capability of AIM-6.

The geometrical model of the reactor consists of four concentric cylindrical regions as shown in Figure I-4C. The central lead dioxide region, which represents a dummy sample, is surrounded successively by the cadmium tube, fuel, and water reflector regions. The fuel region area was made equal to that of the ARMF-I.

Fig. I-4A ARMF-I core diagram.

Fig. I-4B ARMF-II core diagram.

Fig. I-4C One-dimensional model of ARMF core for AIM-6 calculation.

The calculation utilized 17 fast groups and one thermal group. The group widths, which are multiples of 0.25 lethargy units, were chosen so as to give the highest resolution at the lower end of the energy spectrum where changes in the flux distribution were expected to be the greatest. The GAM-I [8] code was used to calculate the constants for the first 17 groups. The thermal constants were obtained from SOFOCATE [9]. No self-shielding corrections were introduced into the group constants.

The AIM-6 output gives directly, at each spatial mesh point, the average adjoint flux, $\bar{\phi}^+$ in each group and the integrated real flux $\bar{\phi}\Delta u$ in each group. The real fluxes were divided by the proper group widths Δu in order to convert them into a distribution function. The fluxes at the center of the cadmium tube are presented in lethargy space in Figure I-5. Figure I-6 shows the weighting function $\bar{\phi} \bar{\phi}^+$. Since a $1/E$ flux distribution corresponds to a constant flux in lethargy space, it is apparent that the flux deviates quite drastically from an asymptotic $1/E$ behavior. It would seem that this decrease at low energies is due primarily to the influence of the cadmium tube. The adjoint flux also shows the same rapid decrease at energies of a few electron volts. These results

Fig. I-5 Fluxes at center of ARMF cadmium tube.

Fig. I-6 Weighting function at center of ARMF cadmium tube.

indicate that the proper interpretation of reactivity measurements performed inside the cadmium tube must include some corrections to account for both the non- $1/E$ behavior of the real flux and the lack of constancy of the adjoint flux.

2.3 Xenon Induced Instability Studies in the ATR Core (G. M. Sandquist)

The stability of the axial power distribution in the ATR during possible xenon-135 induced instabilities is currently under investigation. Previous investigations [10, 11, 12, 13] of the linearized kinetic equations, found from first order perturbation theory approximations, have indicated axial stability for small perturbations. To expand these previous investigations, the pertinent, nonlinear kinetic equations, employing one dimensional, one group diffusion theory, have been simulated on a digital computer (IBM 7040). On the basis of this simulation the ATR is stable against any reasonable perturbation of the axial power distribution which may be encountered. This conclusion of axial stability has been corroborated by various analytical techniques imposed upon the same nonlinear equations.

Presently, digital computer simulation is being extended to two group diffusion theory. Also, an experimental investigation of axial stability utilizing the ATR Critical Facility is planned employing the following scheme: If the axial flux distribution in a lobe is perturbed by some poison distribution which simulates xenon, then the experimentally determined, perturbed axial flux distribution can serve as input to the digital computer which will determine an appropriate time step change in the xenon distribution. This computed distribution of xenon can be simulated by the poison and the procedure repeated until the axial stability of the flux distribution is determined.

2.4 Evaluation of ARMF Measurement Techniques (D. W. Knight)

As part of a continuing program to gain a better understanding of the problems associated with ARMF measurements, studies have been made in temperature control, calibration, and interpretation of reactivity measurements. A description of the program and results are given below.

2.41 Temperature Control of ARMF Canal (I. E. Stepan, D. W. Knight). It has previously been shown that the effect of temperature on reactivity measurements in the RMF [14] and ARMF [15] reactors is significant, even for changes of a fraction of a degree. For example, a fuel sample worth $500 \mu k$ may change $4 \mu k$ per $^{\circ}C$. Furthermore, the direction and magnitude of the change is a function of its position in the reactor and type of sample. As a result it is best to control the water temperature at some convenient value.

Before the canal water temperature was controlled the range of temperature drift from season to season was about $7^{\circ}C$ and as much as $0.5^{\circ}C$ per day in the 32,000 gallon ARMF canal. It was found that by mixing with two stirrers, judicious adjustment of purge water flow rate and number of submerged lights used, the temperature could be controlled at the desired value. Using these procedures the temperature has been controlled during the past 9 months at $18.00 \pm 0.005^{\circ}C$ during operating periods. Temperatures outside these limits are experienced when the reactor is unattended since control is manual. However, drifts during reactor down periods can be quickly corrected to $18.00^{\circ}C$ during reactor check-out and start-up by either the addition of steam or increasing cold purge water. An additional demineralized water line with a pressure reducing valve and rotometer has been added to allow fine control of inlet water flow rate.

2.42 Regulating Rod Calibration (D. W. Knight). The ARMF regulating rod calibration is a relationship between the regulating rod orientation and reactivity, $\rho = \Delta k/k_{eff}$, of the reactor, assuming the reactor is critical with the regulating rod at its inner limit.

A brief description of the experimental [16] and analytical [17] techniques used to calibrate the ARMF regulating rod is presented here. In general, the experimental technique consists of taking a series of regulating rod position readings with a calibration sample in and out of the reactor. The critical position of the reactor is adjusted by moving the shim so that the sets of in and out measurements are nearly uniformly spaced over all rod positions. The analytical method [15] consists of using a least squares fit technique to obtain a functional relationship between the assumed constant, but unknown, reactivity worth of the calibration sample and the regulating rod position. The resultant curve is normalized to read in actual values of reactivity by the use of period measurements and the inhour equation.

The above method is used in lieu of a calibration based strictly on period measurements because of the exceedingly long time required to obtain enough measurements for a statistically accurate description of the shape of the calibration curve.

The present studies were made to determine the adequacy of the technique used (a) for samples measured at any portion of the regulating rod span and (b) for samples measured in experimental positions other than the one used in obtaining the calibration data.

(1) Effect of Regulating and Shim Rod Positions (E. F. Aber and W. K. Foell). As the control rods in a reactor change positions the flux distribution changes and, thus, the reactivity worth of a sample in a given position changes. In the ARMF regulating rod calibration, it is assumed that the rod position changes are small enough that their effect on the sample worth is negligible. To test this assumption in ARMF-II the sample reactivity worth was measured by the reactor period method at several rod positions. The AIPMAN 7090 Computer codes [18] were used for the data analysis. The waiting time (time required for the reactor period to approach within a small fraction of the stable reactor period) was determined from Toppel's calculations [19]. The counting equipment which consisted of a BF₃ chamber, an A-8 amplifier, a preset timer (10 sec) and print-out system was checked for statistical purity by applying the Riedel test [20]. Periods were shown to be reproducible to within about 27 sec or 0.36 μ k standard deviation for a period of 2700 sec and reactivity of -35.25 μ k.

The measurement results are given in Table I-3.

TABLE I-3

REACTIVITY MEASUREMENT RESULTS AND ESTIMATED STANDARD DEVIATIONS

Regulating Rod Position (%)	Reactivity by Period Measurement		
	Reactivity (μ k)	Number of Measurements	Estimated Std. Dev. (μ k)
85	-35.32	4	0.37
49	-35.23	3	0.22
68	-35.20	5	0.47

These results show that with a confidence level of 90 percent the reactivity worth of the sample does not change more than 1.00 μ k for an assumed Student "t" statistical significance level of 5 percent.

(2) Effect of Measurement Position (D. W. Knight, E. Fast). The assumption to be checked was that the calibration curve obtained from measurements of a sample in one measuring position is adequate for measurements in other measuring positions. Using the calibration method described above, two calibration curves were obtained from measurements on a sample in two different measuring positions (center and lattice position E-2, see Figure I-7), in the ARMF-I. The sample worth was determined from both calibrations in both positions, producing four sets of data. These data are summarized in Table I-4 and representative curves are shown in Figure I-8.

It is seen from Table I-4 that, when the sample measuring position and calibration data correspond, the standard deviation is small, as would be expected since the program forces a fit to these data. However, this was shown to be a valid procedure in Part (1) of this work. When the calibration data are not of the measuring position the standard deviation is larger. In Figure I-8 it can

be seen that the sample worth varies systematically as a function of the portion of the regulating rod span used for the measurement, and that for this size sample ($73 \mu k$) the deviation is approximately $1 \mu k$ near the limits. These results show that using only one calibration for both positions introduces only small errors because most experiments use a fraction of the span of the regulating rod and rely primarily on standard calibration samples rather than absolute reactivity values from the regulating rod. However, if a precision in the absolute value of reactivity consistent with reproducibility of measurements is to be realized, separate calibration data would be needed.

Fig. I-7 ARMF-I core loading diagram.

TABLE I-4

COMPARISON OF RESULTS OF ARMF MEASURING POSITION CALIBRATIONS

Measuring Position	Calibration Data Used	Standard Deviation, μk ($10^{-6} \Delta k/k$)
Lattice	Lattice	0.27
Lattice	Center	1.12
Center	Center	0.24
Center	Lattice	3.22

Fig. I-8 Reactivity comparisons between ARMF-I measuring position calibrations.

2.43 Analysis of Reactivity Measurements (D. A. Millsap, R. G. Nisle).

(1) Introduction. One of the principal problems associated with the analysis of reactivity measurements results from the fact that, in general, reactivity responses are not linearly additive. This is due to the perturbation that the sample itself imposes on the flux. Such perturbation or "self-shielding" is primarily associated with neutron absorption. If a sample is predominantly a $1/v$ absorber, the calibrating procedure (relating reactivity and total sample cross section) is relatively simple. A plot of reactivity, ρ , vs cross section, ΣV , obtained from a set of $1/v$ absorbers should be sufficient to calibrate for any $1/v$ absorber regardless of self-shielding and the resulting nonlinearity of the curve. Interpretation becomes more complicated when a reactivity response is due to more than one nuclear effect as in the case of samples containing mixtures of absorbers and fissionable materials. The number of standard samples necessary to cover even a two component system adequately soon becomes prohibitively large. A theoretical understanding can greatly reduce the number of standards needed to interpret such measurements. An ARMF experimental program is now in progress which is designed to check a simple theoretical approach to this problem. The theory and assumptions made are outlined below along with a few preliminary experimental results.

The theoretical model presented here emphasizes the interpretation of reactivity measurements for samples containing mixtures of pure absorbers with fissionable materials in highly thermalized fluxes. These materials are referred to as "poisons" and "fuels", respectively. The principal objective of the approach used is to find an acceptable procedure for relating theoretical expressions for the condition of "infinite dilution" with empirically determined fuel and poison calibration curves. The calibration samples are assumed to have quantities of absorbing materials present sufficient to cause a significant flux perturbation. This implies a nonlinear reactivity response.

(2) Derivation of the Reactivity Equations.

1. Assumptions

- (a) At infinite dilution the reactivity effects of neutron absorption and neutron production are linearly additive.
- (b) The fast neutrons produced in the sample all escape from the sample; furthermore, the reactivity of the escaping fast neutrons is in direct linear proportion to their total number.
- (c) The reactivity effects due to changes in scattering or moderation can be neglected.
- (d) The nonlinearity of reactivity response as a function of sample content (whether due to "self-shielding" or other perturbation) can be related to the total absorption area of the sample, ie, $n v$ atoms times $\hat{\sigma}$ cm²/atom, regardless of the type of absorber, whether it be fuel or poison.

The assumptions above are, of course, oversimplifications. However, the limitations should be minimal if their applicability is restricted to near $1/v$ absorbers in predominantly thermal fluxes. This allows the use of a one-group absorption model.

2. The Model

Near infinite dilution the reactivity of a poison can be represented by the equation

$$\rho = -\bar{W}_p \sum_p \lambda_p V. \quad (1)$$

The meaning of various symbols are given in Table I-5.

TABLE I-5

DEFINITION OF SYMBOLS

<u>Principal Symbols</u>	<u>Definition of Symbols</u>
A, K	Characteristic parameters of an empirical fit to the reactivity calibration measurements
N	Number of nuclei per cm ³
V	Volume of the active material of sample in cm ³
$\bar{W}$	The proportionality constant related to the average statistical weight for a particular nuclear effect
η	Average number of neutrons produced per neutron absorbed in a fuel
ν	Average number of neutrons produced per fission
ρ	Reactivity, $\Delta k/k_{eff}$
Σ	Effective macroscopic cross section in cm ⁻¹
σ	Effective microscopic cross section in cm ²
<u>Subscripts</u>	
a	Absorption in a pure fuel
F	Fuel
f	Fission
p	Absorption in a pure poison
T	Total

As the quantity of absorber is increased self-shielding occurs. According to 1-(d) Equation (1) can be written

$$\rho = -\bar{W}_p \hat{\Sigma}_p V G(\hat{\Sigma}_p V) \quad (2)$$

where $G(\hat{\Sigma}_p V)$ is any self-shielding function that can be shown to be applicable. $G(\hat{\Sigma}_p V)$ must satisfy the condition

$$\lim_{\hat{\Sigma}_p \rightarrow 0} G(\hat{\Sigma}_p V) = 1. \quad (3)$$

Thus, Equation (2) approaches Equation (1) as $\hat{\Sigma}_p V$ approaches zero. A similar pair of equations may be written for a pure fuel.

$$\rho = -\bar{W}_a \hat{\Sigma}_a V + \nu \bar{W}_f \hat{\Sigma}_f V \quad (4)$$

and

$$\rho = [-\bar{W}_a \hat{\Sigma}_a V + \nu \bar{W}_f \hat{\Sigma}_f V] G(\hat{\Sigma}_a V) \quad (5)$$

Poison and fuel calibrations can be obtained by taking reactivity measurements on sets of standard samples whose cross section ($\hat{\Sigma}_p V$) and ($\hat{\Sigma}_a V$) cover the range of interest. In both cases, it has been found that the relation

$$\rho = A(1 - e^{-K\Sigma V}) \quad (6)$$

gives a good approximation to the experimental data for values of Σ not too large.

The constants A and K in Equation (6) may be found by evaluating the derivative ($d\rho/d(\hat{\Sigma}V)$) at $\hat{\Sigma}V = 0$. The results are, for the case of Equation (1),

$$A K_p = -\bar{W}_p \quad (7)$$

and for Equation (4)

$$A K_f = -\bar{W}_a + \eta \bar{W}_f \quad (8)$$

where the substitution $\eta \hat{\Sigma}_a = \nu \hat{\Sigma}_f$, has been made in the latter expression.

Now, if the fuel cross section, as a function of energy, does not differ too strikingly from that of the absorber used for the poison calibration, then the substitution $\bar{W}_a = \bar{W}_p$ should be a good approximation. By substitution of Equations (7) and (8) into (2) and (5), respectively, one has

$$G(\hat{\Sigma}_p V) = \frac{1 - \exp(-K_p \hat{\Sigma}_p V)}{K_p \hat{\Sigma}_p V} \quad (9)$$

and

$$G(\hat{\Sigma}_a V) = \frac{1 - \exp(-K_F \hat{\Sigma}_a V)}{K_F \hat{\Sigma}_a V} \quad (10)$$

According to assumption 1-(d) there is no reason for distinguishing between the self-shielding functions for fuel and poison. $\hat{\Sigma}_p$ of Equation (9) and $\hat{\Sigma}_a$ of Equation (10) might just as well be replaced by some $\hat{\Sigma}_T$ representing the total absorption cross section in the sample, including both fuel and poison. All that remains to make the self-shielding functions equal is that $K_p = K_F$. The general empirically determined self-shielding function becomes, therefore,

$$G(\Sigma_T V) = \frac{1 - \exp(-K \hat{\Sigma}_T V)}{K \hat{\Sigma}_T V} \quad (11)$$

where

$$\hat{\Sigma}_T = \hat{\Sigma}_p + \hat{\Sigma}_a.$$

The above results can be used to derive a general equation representing mixtures of fuel and poison. By assumption 1-(a) one has

$$\rho = -\bar{W}_p (\hat{\Sigma}_p V) + (-\bar{W}_p + \eta \bar{W}_f) (\hat{\Sigma}_a V). \quad (12)$$

When self-shielding is present, Equation (12) may be written

$$\rho = [-\bar{W}_p (\hat{\Sigma}_T V) + \eta \bar{W}_f (\hat{\Sigma}_a V)] G(\hat{\Sigma}_T V) \quad (13)$$

which, using Equation (11), becomes

$$\rho = \left[-\frac{\bar{W}_p}{K} + \eta \frac{\bar{W}_f}{K} \frac{\Sigma_a}{\Sigma_T} \right] [1 - \exp(-K \Sigma_T V)]. \quad (14)$$

The discussion has attempted to demonstrate that the empirical expressions for fuel and poison calibration data can be combined with the theoretical equations one would use at infinite dilution. The result, Equation (14), should be correct for near $1/v$ absorbers in highly thermalized fluxes; as a matter of fact it

should hold in any reactor measuring position where K_p is found equal to K_F for reasonable estimates of $\hat{\Sigma}_p$ and $\hat{\Sigma}_f$.

3. Experimental Program

An experimental program to test the theory outlined above is in progress. The procedure is as follows:

(1) Geometrically identical samples of boron and of U-235 of various concentrations are to be measured in a center position of the ARMF and the results fitted to equations of the form (6).

(2) Assuming K_p and K_F are equal for reasonable values of $\hat{\Sigma}_p$ and $\hat{\Sigma}_f$ a set of boron-U235 mixtures will be measured in the same position, and the data fitted to Equation (14). Parameters $\bar{W}_p$, $\bar{W}_f$, and K will be calculated.

(3) Extension to other materials and geometries will follow.

The samples being used are of the Sample Fuel Plate design described in IDO-16574. A special ARMF-I plate holder is in use which permits measurement in any one of 19 positions. Four representative positions have been chosen for the measurements which locations range from the center of the water hole to an extreme corner position.

Preliminary results have given some confirmation of the basic model. However, additional reactivity measurements and content analyses will be needed before final results can be reported. The results of Table I-6 are given as tentative values by way of illustration. Measurements include two positions near the center and two near a corner of the center water hole in the ARMF. The uncertainties reflect simply the degree to which the measured reactivity values fit the empirical exponential function. There are uncertainties in other quantities involved such as in sample composition or the value of effective cross section which are probably as great as those indicated in the table. Thus, within these uncertainties it is not clear that K_p and K_F are significantly different. However, the corner positions having an appreciably harder flux as well as a flux gradient would tend to amplify differences in the absorption weighting factors, $\bar{W}_p$ and $\bar{W}_a$, which were assumed to be identical.

TABLE I-6
EXPERIMENTAL VALUES OF K_p AND K_F

ARMF-I Experimental Position	$K_p \text{ cm}^{-2}$	$K_F \text{ cm}^{-2}$
Center-5	0.275 ± 0.011	0.266 ± 0.015
Center-10	0.271 ± 0.012	0.275 ± 0.015
Corner-7	0.259 ± 0.005	0.287 ± 0.011
Corner-9	0.257 ± 0.006	0.284 ± 0.012

3. REACTOR PHYSICS

3.1 Summary of Results from MTR Fuel Loading Calculations During 1963 (B. L. Rushton, L. Smith)

The Reactor Physics Section continued the practice of calculating fuel loading for MTR Operations during 1963. The fuel loadings were calculated following the procedures given in IDO-16661 [21]. A discussion of the year's activity and the results of the calculations will be presented here.

Fuel loadings were calculated for MTR cycles 185 through 201. An initial charge (A-type) was prepared for each cycle and emergency charges (B, C, ..., type) were furnished as required to complete the three week cycle. Thirty-two charges were used during the year.

The experimental charge life was obtained for each charge whose final fuel shim rod position was in the range of the extrapolation curve given in Reference 21. This curve gives the MWd remaining versus the average fuel shim rod position. Figure I-9 shows the comparison of the calculated charge life with the experimental charge life. Figure I-10 shows the ratio of the calculated life to the experimental life. There was good agreement throughout the year.

Fig. I-9 Comparison of calculated charge life with experimental charge life.

Charge 200-A was a heavy loading which required approval from the MTR Safeguard Committee. MTR Operations requested a loading that would run the MTR for approximately 725 MWd. A 723 MWd loading was prepared. With this loading the reactor went critical with an average fuel shim rod position of

Fig. I-10 Ratio of calculated charge life to experimental charge life.

15.86 inches. The reactor had accumulated 724 MWd when the fuel shim rods reached the 30.0-inch level.

Charges that are fully utilized, ie, charges that are in the reactor until the fuel shim rods have been withdrawn to the 30.0-inch level, give a means for checking the accuracy of the extrapolation curve in Reference 21. In all cases, the charge life predicted from the curve was in excellent agreement with the actual life.

In conclusion, the procedures for calculating MTR fuel loadings were very successful during 1963. Close attention was given to the comparison of calculated life with experimental life, and the "c" constants were adjusted on the basis of these comparisons. It is felt that these adjustments lend much to the success of the fuel loading procedure.

4. REFERENCES

1. G. W. Gibson and W. C. Francis, Annual Progress Report on Fuel Element Development for FY 1962, IDO-16799, August 1962.
2. E. Fast and J. J. Scoville, "Resonance Absorption of Dysprosium and Tungsten" in MTR-ETR Technical Branches Quarterly Report First Quarter 1962, IDO-16781 (1962) pp 2-3.
3. A. E. McCarthy et al, "Neutron Resonance Integral and Age Data", pp6-8 in ANL Reactor Physics Constants Center Newsletter No. 1 (1961).

4. P. J. Persiani et al, Neutron Resonance Integral and Age Data, ANL Reactor Physics Constants Newsletter No. 10 (1963).
5. J. J. Scoville et al, "Resonance Absorption Integrals of Dysprosium and Tungsten", Trans. Am. Nucl. Soc., 5, November 1962, pp 377-378.
6. R. K. Lane et al, "Resonance Absorption in Materials with Grain Structure", Nuc. and Sci. Eng., 14, 1962, pp 390-396.
7. H. P. Flatt and D. C. Baller, "The AIM-6 Code", NAA Internal Memorandum (1961) describing modifications to the AIM-5 Code; See H. P. Flatt and D. C. Baller, AIM-5 A Multigroup One-Dimensional Diffusion Equation Code, NAA-SR-4694, March 1, 1960.
8. G. D. Joanou and J. S. Dudek, Gam-1: A Consistent P_1 , Multigroup Code for the Calculation of Fast Neutron Spectra and Multigroup Constants GA-1850, June 1961.
9. R. J. Wagner, 650-Sofocate - A Code to Determine Thermal Constants on the IBM-650, IDO-16758, March 1962.
10. R. S. Wick, Space- and Time-Dependent Flux Oscillations (And Instability) in Thermal Reactors Due to Nonuniform Formation and Depletion of Xenon, WAPD-TM-138, August 1958.
11. D. Randall and D. S. St. John, "Xenon Spatial Oscillations", Nucleonics, 16, March 1958, pp 82-86.
12. R. S. Marsden et al, Reactor Physics Studies for the Final Conceptual Design of the Advanced Test Reactor, IDO-16668, March 1961.
13. S. W. Spetz, Advanced Test Reactor. Axial Xenon Stability, Study S-R-122, TID-17290, November 1962.
14. E. Fast, Temperature Coefficients of the Reactivity Measurement Facility, IDO-16772, July 1962.
15. J. W. Rogers et al, "Temperature Effects in the ARMF" in MTR-ETR Technical Branches Quarterly Report Third Quarter 1962, IDO-16827, (1962) pp 3-5.
16. E. Fast et al, Calibration of RMF Control Elements, IDO-16610, Parts I, II, III, and IV, August 1960.
17. D. P. Gibson, "Calibration of the ARMF Regulating Rod" in MTR-ETR Technical Branches Quarterly Report Fourth Quarter 1960, IDO-16665, (1961) pp 51-53.
18. H. J. Richardson, "AIPMAN I, II, AND III", NAA Internal Memorandum, (January 1962).
19. B. J. Toppel, "Sources of Error in Reactivity Determinations by Means of Asymptotic Period Measurements", Nuc. Sci. Eng., 5, February 1959, pp 88-98.

20. O. Riedel, "Statistical Purity in Nuclear Counting", Nucleonics, 12, June 1954, pp 65-67.
21. R. W. Goin, Standard MTR Fuel Loading Procedures, IDO-16661, January 1961.

II. NUCLEAR PHYSICS

1. CROSS SECTIONS PROGRAM (M. S. Moore)

1.1 Fast Chopper

1.11 Pa-233 Total Cross Section (F. B. Simpson and Fast Chopper Group). Total neutron cross section measurements have been taken on two different separated Pa-233 samples. The first sample contained approximately 410 barns/atom and the second had a thickness of approximately 300 barns/atom. The two samples were prepared from two different irradiations and chemical separations over a period of about three months. The cross section has been investigated from 0.01 to 1000 eV.

A plot of the total neutron cross section as a function of energy between 0.01 to 1.0 eV is shown in Figure II-1. These data give a total cross section at thermal energy of 56^{+3}_{-5} barns. During the accumulation of these data, a gold sample was included in each run as an operating standard. The equipment was personally monitored during each run for further assurance of proper operation.

After the data in the thermal energy region were corrected for the contaminants present in the sample, which included the U-233 produced from the decay of Pa-233, the results of the data from the two independent samples agreed very well. There is still a possibility of small amounts of other contaminants present in the samples which have not been corrected. The two small resonances observed in this energy region may be due to contaminants.

The data above 1.0 eV are being analyzed and will be presented in a future progress report. These data will give good resonance parameters up to 20 eV.

1.2 Crystal Spectrometer

1.21 Mn Bath Eta Measurements (J. R. Smith, S. D. Reeder). The manganese bath experiment for the measurement of eta of U-233 and Pu-239 completed a successful series of irradiations during November. Irradiations of the same type produced results with a standard error in the vicinity of 0.1 percent. The improved results are due principally to three factors:

(1) The technique of counting was changed. Formerly an attempt was made to set the discriminator level by the position of the 0.85 MeV gamma ray photopeak. This was an unsatisfactory procedure. Gain and bias changes in the electronics system and scintillator had a large effect on counting rate stability

Fig. II-1 Total neutron cross sections of Pa-233 from 0.01 to 1.0 eV.

because of the differences in position of the features of the spectra from the monitor foil and solution sample, due to γ -ray degradation in the latter. A decisive improvement was realized in both stability and counting rate after converting to the use of a low noise amplifier biased just above the photomultiplier noise level, and counting everything above that point.

(2) The counting system was moved away from the reactor area to a region of more constant background. Since the low-level counting technique also yields higher background rates, a constant background is especially important.

(3) A mechanical neutron filter was used to eliminate order effects. This is a considerably more efficient and satisfying technique than the use of gadolinium filters for order discrimination, which was used previously.

The data from this past set of irradiations were all taken at 0.057 eV incident neutron energy. Not all of the corrections are in their final form. Using preliminary values for these obtains $\eta = 2.296$ for U-233 and $\eta = 2.041$ for Pu-239 at this energy. These values will change somewhat as final values for the corrections are agreed upon, but probably by somewhat less than one percent.

2. NUCLEAR CHEMISTRY (W. H. Burgus)

2.1 Preparation of Chemically Separated Pa-233 for Cross Section Measurements (J. W. Coddington, J. R. Berreth, R. A. Deal, R. P. Schuman, W. H. Burgus, H. G. Miller)

A third, large, chemically separated Pa-233 sample was prepared during the quarter for total thermal cross section measurements with the MTR fast chopper. The same isolation box and equipment that produced the previous samples [1] were used again, but several modifications and replacements were needed before the system was adequate for processing work. Radiation embrittlement had affected polypropylene tubing, nylon tubing fittings, glass tubing, and rubber stoppers; wherever these components were subjected to stress, replacement was required. With the aid of specially designed wrenches, all the necessary maintenance was accomplished remotely, including replacement of Swagelok tubing fittings, modification of filter entry ports, etc. Despite the high radiation doses to which the fritted glass filters had been subjected, their usefulness was unaffected, and they continued to function perfectly throughout the run.

Sixteen pieces of thorium metal (280 grams total) in the form of 2-x 5-x 0.15-cm plates were irradiated in a high flux position (K-7-NE) for one cycle in the ETR. After less than one day's cooling these were removed from their aluminum irradiation capsules in the Hot Alpha Cave and dissolved in incrementally-added 13M HNO_3 - 0.1M HF to produce a solution about 0.5M in Th^{+4} . An analysis of the dissolver solution made by sampling and absolute γ counting indicated that ≈ 2.0 grams of Pa-233 had been formed during irradiation.

Protactinium was first separated from solution by co-precipitation with MnO_2 formed in situ by adding first KMnO_4 , then $\text{Mn}(\text{NO}_3)_2$, and digesting at an elevated temperature. The MnO_2 precipitate was filtered on the first fritted-

glass filter. A sample of the filtrate showed ≈ 200 mg Pa-233 content; this was discharged to waste as representing only ≈ 10 percent of the total Pa-233. The Pa-bearing MnO_2 was then dissolved in $6\text{M H}_2\text{SO}_4 + \text{H}_2\text{O}_2$, diluted, and returned to the dissolver, where addition of KMnO_4 produced a second MnO_2 precipitate. This was filtered as before, and the filtrate sampled. Analysis showed that ≈ 30 percent of the remaining Pa-233 was in the filtrate; this was considered an intolerably high loss. Much of the second MnO_2 precipitate had redissolved through chemical reaction with radiolytic products during a three-hour delay prior to filtration while a leaking transfer line to the filter was being repaired. Consequently, the filtrate was returned to the dissolver and another MnO_2 precipitate formed. This time the filtration resulted in adequate Pa recovery, as observed by a mobile radiation monitor inside the processing cell. The dissolver was washed out with 0.05M HNO_3 ; the solution subsequently served to wash the precipitate.

The second Pa- MnO_2 precipitate was dissolved as before in $6\text{M H}_2\text{SO}_4 + \text{H}_2\text{O}_2$, diluted, and again recycled to the dissolver for an additional MnO_2 precipitation. Filtration of this third Pa- MnO_2 precipitate again produced adequate Pa recovery. The precipitate was washed well with 0.05M HNO_3 , and then the Mn was leached away from the hydrated Pa_2O_5 by slow treatment with $0.5\text{M HNO}_3 - 0.5\% \text{H}_2\text{O}_2$. This gave a good separation from Mn without appreciable loss of Pa. The protactinium left on the filter was then dissolved by prolonged digestion with $6\text{M H}_2\text{SO}_4$. The resulting solution was diluted to $\approx 1.5 \text{M H}_2\text{SO}_4$ in preparation for an iodate precipitation. At this point the pure Pa-233-sulfate was contained in 195 ml of clear, bubbling solution. Analysis showed that 940 mg of Pa-233 product had been recovered.

After reserving a small fraction for analysis, excess HIO_3 was added to precipitate Pa-iodate. The voluminous, white precipitate was filtered on a Millipore filter and washed with dilute H_2SO_4 containing some HIO_3 . It was then transferred to a porcelain crucible for drying and ignition. No external heat was required for drying; the glowing precipitate generated sufficient heat not only to dry itself, but to cause decomposition with evolution of iodine. The material was finally ignited at 800°C to assure complete decomposition of iodate to oxide. At this point 808.4 mg of white Pa_2O_5 were recovered. This is about the upper limit of sample that can be accommodated in the chopper sample container.

The chopper sample consisted of two 30 wt% $\text{Pa}_2\text{O}_5 - \text{Al}$ compacts and contained a total of 787.6 mg of $\text{Pa}_2^{233}\text{O}_5$. It was used to check the thermal and resonance cross sections obtained with the previous samples.

Removal of a clean sample (free of surface contamination) from the Cave presented a problem because of the extensive contamination present in the isolation box. In order to avoid contaminating the chopper block, into which the sample was loaded, a small Lucite transfer box was constructed which fit into the transfer port of the Cave. This box was equipped with a simple magnetic manipulator to load the sample into the chopper block, and the chopper block into the chopper cask. An alcohol trap was included to clean the contaminated sample as it came from the large isolation box of the Cave. With this equipment, contamination of the chopper block was kept to a minimum, and it was unnecessary to remove the sample to another hot cell for washing and loading into the chopper block and cask.

Because the second large separated Pa-233 chopper sample (isolated last quarter) showed a resonance at 0.09 eV due to an impurity (most likely Sm-149), a rare earth radiochemical analysis was made on an aliquot of the solution from which the Pa sample had been isolated. The rare earth activity was carried on NdF_3 (five precipitations made), Pa was further separated from rare earths using a Dowex 1 anion column, and finally $\text{Nd}_2(\text{C}_2\text{O}_4)_3$ was precipitated and counted. The only rare earth activity detectable in the final sample was ^{141}Ce which was present to an extent of $\approx 6 \times 10^{-6}$ of the original Pa-233 in the chopper sample. On the basis of the small amount of ^{141}Ce observed, plus the absence of any other countable rare earth activity, it appears that the second Pa sample could not have contained sufficient Sm-149 from fission products to exhibit a detectable resonance of Sm-149. The contamination observed must therefore result from some other source.

2.2 Preparation of Radioactive Samples for Chopper Measurements (R. P. Schuman, R. A. Deal, J. R. Berreth)

Experiments have begun which will result in production samples of Tl-204, Ir-192, Tm-170, Tb-160, Eu-154, Eu-152, and Cs-134 for chopper cross section measurements. The samples will be produced by irradiating the stable elements in the MTR or ETR to produce sufficient concentrations of the capture nuclides for chopper measurements.

In order to plan for optimum irradiation times, calculations have been made of the compositions of irradiated samples of Eu, Tb, Tm, Ir, and Tl. The nuclear data assumed for the calculations are given in Table II-1. Plots of the calculated compositions vs time are given in Figures II-2 through II-6.

Before irradiation, the target elements will be fabricated into the geometric shapes that will be used in the chopper and then placed in sealed containers. Three samples will be irradiated, one will be used as a thin chopper sample, a second will be used together with the first as a thick sample, and the third as a monitor for analysis after irradiation. Thallium, thulium, and terbium will be irradiated as metal, iridium as a compact of Ir and Al powders, europium as a compact of Eu_2O_3 and Al powders, and cesium as CsF. Initially, the natural elements will be irradiated.

2.3 Holmium-166 Half-Life Measurement (K. T. Faler)

All measurements were completed on the holmium sample separated last quarter for the purpose of half-life determination. Results were as follows:

Weight:	$143 \pm 3 \mu\text{g}$ as Ho_2O_3
Disintegration rate:	$5.5 \times 10^3 \text{ dps} \pm 10 \text{ percent}$
Mass of 166:	$0.066 \pm 0.001 \text{ atom percent}$

The measurement of the disintegration rate assumes conversion coefficients for the 184 and 280 keV gamma transitions of 0.33 and 0.08, respectively. These are the theoretically calculated values. It is further assumed that the number of gamma-transitions per disintegration is correct as shown in the decay scheme of Cline and Reich [2]. Estimated limits of error on this number are based on counting statistics, spectrum separation, and counting efficiency. The quoted error is believed to be conservative.

TABLE II-1

NUCLEAR DATA USED FOR CALCULATING THE
COMPOSITION OF IRRADIATED SAMPLES

Nuclide	Half-Life	Pile Cross Section Barns	
Tl-203	stable	11	[a]
Tl-204	3.9 yr	10	[b]
Tl-205	stable	0.11	[a]
Tl-206	4.20 min	0	[b]
Pb-204	stable	0.7	[a]
Pb-206	stable	0.03	[c]
Ir-191	stable	1000	[c]
Ir-192	74 d	700	[a]
Ir-193	stable	130	[a]
Ir-194	19 hr	0	[b]
Pt-192	stable	5	[b]
Pt-194	stable	1.2	[a]
Tm-169	stable	127	[a]
Tm-170	129 d	150	[a]
Tm-171	1.9 yr	30	[b]
Tm-172	64 hr	0	[b]
Tb-159	stable	46	[a]
Tb-160	73 d	525	[a]
Tb-161	7.1 d	0	[b]
Dy-160	stable	100	[b]
Dy-161	stable	580	[c]
Eu-151	stable	1400 [a] to 9.2 hr	
		6300 [a] to 12.7 hr	
Eu-152	9.2 hr	0	[a]
Eu-152	12.7 yr	5500	[a]
Eu-153	stable	450	[a]
Eu-154	16 yr	1500	[a]
Eu-155	1.7 yr	14000	[a]
Eu-156	15.4 d	500	[b]
Sm-152	stable	225	[a]
Gd-152	stable	225	[b]
Gd-155	stable	61000	[a]
Gd-156	stable	500	[b]

[a] From BNL 325

[b] Estimated

[c] From G. E. Chart of Nuclides (1961)

Fig. II-2 Calculated composition of natural europium irradiated at 4×10^{14} flux.

Fig. II-3 Calculated composition of natural thallium irradiated at 5×10^{14} flux.

Fig. II-4 Calculated composition of natural iridium irradiated at 4×10^{14} flux.

Fig. II-5 Calculated composition of thulium irradiated at 4×10^{14} flux.

Calculation of the half-life of this long-lived isomer gives a value of 1200 ± 150 years. The literature value previously reported is >30 years [3]. A half-life value of 90,000 yr for this isotope was previously reported [4]. However, this report must now be retracted since the mass analysis of the sample used in obtaining the half-life has been found grossly in error.

3. INELASTIC SCATTERING (R. M. Brugger)

3.1 The Placzek First Moment Condition (R. M. Brugger)

The Placzek first moment condition $\langle \beta^1 \rangle$ in the scattering law notation is

$$\langle \beta^1 \rangle = \int_0^\infty 2\beta \sinh \beta/2 S(\alpha, \beta) d\beta = \alpha$$

Fig. II-6 Calculated composition of terbium irradiated at 4×10^{14} flux.

data that are too limited in their range of β to include all of the contributions. Finding explanations for values that are high is difficult. Possible explanations are (a) the theory is not complete, (b) multiple scattering is distorting the data or (c) coherent scattering is not being properly treated. Experiments are now being started to study these possible sources of error.

where $\beta = \frac{\hbar\omega}{K_B T}$ is the energy transfer

in dimensionless units, $\alpha = \frac{\hbar^2 k^2}{2M K_B T}$

is the momentum transfer squared in dimensionless units and $S(\alpha, \beta)$ is the total scattering law. In an attempt to understand this moment condition, values of $\langle \beta^1 \rangle / \alpha$ have been calculated for different values of α for the existing $S(\alpha, \beta)$ data. These are shown in Figure II-7. In these calculations values of M and the bound atom cross section have been used such that $\langle \beta^1 \rangle / \alpha$ would be a minimum.

The predicted value of $\langle \beta^1 \rangle / \alpha$ is one, but the data for none of the samples give this value over a range of α 's. Values that are low can be explained by

Fig. II-7 Values of the Placzek first moment condition $\langle \beta^1 \rangle$ divided by the function of the momentum change squared, for some of the existing scattering law data.

3.2 Elastic Scattering from Powdered Beryllium (R. E. Schmunk)

In Figure II-8 are shown the reduced partial differential cross sections for elastic scattering from powder beryllium. Because the scattering peaks which are centered at $\epsilon = 0$ closely approximate the beam monitor peaks in width, the number of counts was summed over all channels in the peak to obtain the elastic cross sections. The structure which is observed in the data is due to Bragg reflection by the different crystallographic planes in the sample. It should be noted that all Bragg scattering from a given set of planes ($h\ k\ i\ l$) contributes to the scattering at a given value of κ as noted in Table II-2. Bragg scattering does not occur for κ values less than $3.175\text{\AA}^{-1}$, and data to the left of this value in the figure represent multiple Bragg scattering.

Fig. II-8 Reduced partial differential cross section for elastic scattering from powder beryllium. The various crystallographic planes contribute to the scattering at the kappa values indicated by the slash marks at the bottom of the figure and as noted in the accompanying table.

TABLE II-2

KAPPA VALUES FOR WHICH THE
CRYSTALLOGRAPHIC PLANES IN BERYLLIUM CONTRIBUTE
TO THE REDUCED PARTIAL DIFFERENTIAL CROSS SECTION OF FIGURE II-8

$(h\ k\ i\ l)$	$\kappa\ (\text{\AA})^{-1}$	$(h\ k\ i\ l)$	$\kappa\ (\text{\AA})^{-1}$
10 $\bar{1}$ 0	3.1744	2022	7.2530
0002	3.5070	20 $\bar{2}$ 3	8.2449
10 $\bar{1}$ 1	3.6276	12 $\bar{3}$ 1	8.5796
10 $\bar{1}$ 2	4.7300	11 $\bar{2}$ 4	8.9120
11 $\bar{2}$ 0	5.4980	12 $\bar{3}$ 3	9.9100
10 $\bar{1}$ 3	6.1437	30 $\bar{3}$ 2	10.148
2020	6.3487	22 $\bar{4}$ 2	11.541
11 $\bar{2}$ 2	6.5212	32 $\bar{5}$ 1	13.947
2021	6.5865		

3.3 Background and Resolution Corrections (Y. D. Harker)

In comparing cold methane gas ($T = 125^\circ\text{K}$) data with Griffing's Discrete Rotation Theory [5], the experimental points fall below the theory for momentum

transfers less than $5(\text{\AA})^{-1}$. This discrepancy was thought to be a result of angular and energy resolution or improper background subtractions. Correction of the data for improper background subtraction brought the theory and experiment into agreement for energy transfers less than 0.003 eV and momentum transfers greater than $3(\text{\AA})^{-1}$. For energy transfers less than 0.003 eV and for momentum transfers less than $3(\text{\AA})^{-1}$, energy and angular resolution have the greatest effect. Work is now in progress to include energy resolution into Griffing's theory for comparison with the data.

For energy transfers greater than 0.003 eV and less than 0.008 eV and for momentum transfer between $2(\text{\AA})^{-1}$ and $4(\text{\AA})^{-1}$, the background corrected data are still slightly below (≈ 10 percent) the theory. The removal of resolution effects in this region is not considered to provide the necessary correction to eliminate the difference, and no further corrections to the data are anticipated. Other causes for this difference between theory and data are being sought.

3.4 Removing Internal Molecular Effects from the Liquid Methane Data (Y. D. Harker)

The inelastic scattering from liquid and gaseous methane is composed of contributions from three types of excitations; translational excitations, rotational excitations, and vibrational excitations. The liquid and gaseous methane data indicate that the contributions to the scattering from vibrational and rotational excitations are the same but that the contributions from the translational excitations have changed in going from the gaseous to the liquid state. Work is now in progress to remove the rotational and vibrational contributions from the two sets of data so that the translational contributions can be compared. This is being accomplished in the following manner; from the theory of Zemach and Glauber [6], the differential scattering cross section is given by

$$\frac{d^2\sigma}{d\epsilon d\Omega} = \sum_{i,f} P_i (2\pi)^{-1} \sum_{v,v'} \alpha_{vv'} \int_{-\infty}^{\infty} \frac{k}{k_0} e^{-i\epsilon t} \langle f, i/v, v' \rangle dt \quad (1)$$

where i denotes the initial state, f denotes the final state, P_i is the probability that the molecule is in the i^{th} initial state, v denotes a particular scatterer in the scattering system, v' denotes another scatterer in the same system, $\alpha_{vv'} = A_v A_{v'} + \delta_{vv'} C_v C_{v'}$. The bound coherent and incoherent scattering lengths of the v^{th} scatterer are denoted by A_v and C_v , respectively. The incident and final momentum numbers of the scattered neutron are denoted by k_0 and k , respectively. The energy transfer to the neutron is denoted by ϵ . There are three modes of freedom for the molecule: that is, translational, rotational, and vibrational excitations. Here the term translational excitations encompasses the complex motion of the molecule as a result of external molecular fields and it is not restricted to translation in the ideal gas sense. If these modes are completely independent, then $\langle fi/vv' \rangle = \langle \text{Translation} \rangle \langle \text{Rotation} \rangle \langle \text{Vibration} \rangle$. The translational term is for the molecule as a whole; the rotational and vibrational terms depend upon the particular scatterers in the molecule and upon the molecular geometry. With this in mind, one can write

$$\sum_{i,f} P_i \sum_{v,v'} \alpha_{vv'} \langle fi/vv' \rangle = \langle \text{Translation} \rangle \langle \text{Rotational-vibration} \rangle \quad (2)$$

Upon examination of Equation (1), one notices that $\frac{k_o}{k} \frac{d^2\sigma}{d\epsilon d\Omega}$ is a Fourier transform of $\sum_{i,f} \sum_{v,v'} \alpha_{vv'} \langle fi/vv' \rangle$. So, if one performs the inverse transform

on the cross section data, $\sum_{i,f} \sum_{v,v'} \alpha_{vv'} \langle fi/vv' \rangle$ is obtained. The rotational-

vibrational term of Equation (2) has been computed for methane gas [5] and the data indicate that it is the same in the liquid. Therefore, the translational term

is obtained by dividing $\sum_{i,f} \sum_{v,v'} \alpha_{vv'} \langle fi/vv' \rangle$, obtained by Fourier transforming

the data, by the rotational-vibrational term obtained from theory. The advantage of this method is that one isolates the effects caused by going from the gaseous to the liquid state.

3.5 Scattering of Neutrons from Ammonia in its Three States (K. A. Strong, J. I. Petz)

Neutron scattering measurements at a scattering angle of 90° have been made with samples of solid, liquid, and gaseous ammonia at the MTR cold neutron velocity selector. Differential cross section distributions for the three samples are shown in Figure II-9. In the solid state, the rotational and translational motions of the NH_3 molecule have been frozen out and the distribution shows only an elastic peak. The liquid state distribution clearly shows effects of both molecular rotations and translations, the energy gain peak being the contribution of rotations, and the quasi-elastic peak that for translations. In the gaseous state distribution the effects of translational and rotational motions have merged into a single energy gain peak centered around 1.8 Å.

3.6 First Moment of the Energy Transfer in Liquid Sodium (P. D. Randolph)

Corrections for multiple scattering have been made on the sodium data in an attempt to find the cause of the discrepancy in the measured first moment of the energy transfer [7]. The correction is made by using an expansion of the

Fig. II-9 Cold neutron inelastic scattering cross sections for ammonia at a scattering angle of 90° .

experimental scattering law in powers of α .

$$S(\alpha, \beta) = A_0(\beta) \alpha + A_2(\beta) \alpha^2 + \dots$$

The first term in a theoretical expansion of $S(\alpha, \beta)$ should be proportional to α , and multiple scattering is expected to be roughly isotropic, ie, independent of α . The assumption is made, therefore, that the term $A_0(\beta)$ in the experimental scattering law can be attributed to multiple scattering. The results of this correction to the experimental values of the ratio of the observed to the theoretical first moment are shown in Figure II-10. If there were no discrepancy, the ratio

$$\frac{\langle \beta^1 \rangle}{\alpha} \text{ should be one for all } \alpha. \text{ It is}$$

seen that the correction has a large effect at small α , and may be an over correction. For $\alpha \gtrsim 0.01$, however, the correction is generally too small and becomes negligible at the larger α . Thus, in spite of this multiple scattering correction, the first moments are still larger than predicted. Whether the correction obtained over or under estimates the multiple scattering cannot be determined. The process used however is a reasonable one and is all that can be done at present.

Fig. II-10 Ratios of observed to calculated zeroth and first moments of the energy transfer for liquid sodium at 100, 150, and 200°.

4. THEORETICAL PHYSICS (H. L. McMurry)

4.1 Computational Program for Slow Neutron Scattering (H. L. McMurry, L. J. Gannon)

The computation program for calculating the scattering of slow neutrons by atoms bound harmonically to molecular aggregates [8] has been made more efficient by modifications which are more effective in restricting calculations to only significant terms. The program has been found very useful for treating systems under conditions where relatively few vibrational modes are excited. Its efficiency and accuracy for calculating scattering at higher neutron energies remains to be tested. This computational program will be described fully in a forthcoming IDO report.

4.2 Slow Neutron Scattering by Propane (H. L. McMurry)

Discrepancies have been noted between the observed and calculated scattering of slow neutrons by propane [9, 10]. The nature of these is indicated in

Figure II-11 which compares calculated and observed $S(\kappa^2, \beta)$ vs κ^2 curves. It was surmised that the use of an incorrect barrier height for the torsional motions of the CH_3 groups might be a cause of the discrepancies [10].

To test this hypothesis, calculations have been made assuming barrier heights of 3000 and 4500 calories per mol. The $S(\kappa^2, \beta)$ function was found to be insensitive to the assumed barrier height. Therefore, other causes must be sought for the discrepancies. The possibility that the Krieger-Nelkin method of averaging over orientation [11] is at fault will be tested after the computing program is altered to permit numerical averaging over orientation.

Fig. II-11 $S(\kappa^2, \beta)$ vs β for propane at 20°C.

4.3 Slow Neutron Scattering by Water (H. L. McMurry)

Attempts have been made to fit recent data obtained by Haywood [12] by means of calculations based on an extension of Nelkin's model for water [13]. Instead of assuming a single torsional frequency of 0.06 eV for the water molecules, several vibration frequencies have been assumed for describing hindered rotations and translations. An additional flexibility has been explored through altering the amplitude factors $\bar{\delta}_{\nu\tau}$ which are associated with the motion of atom ν during excitation of the τ normal mode [14]. The $\bar{\delta}_{\nu\tau}$ have dimensions of reciprocal masses and in the computational program modes of zero frequency (like free translations or rotations) have $\bar{\delta}_{\nu\tau}$ values associated with them such that the effective mass M_ν of the scattering atom is given by the following sum over these modes

$$\frac{1}{M_\nu} = \frac{1}{3} \sum_{\lambda} \bar{\delta}_{\nu\lambda} \quad (1)$$

So far it has not been possible to get a single set of low frequencies and associated $\bar{\delta}_{\nu\tau}$ values which make the calculated $S(\kappa^2, \beta)$ vs β curves agree reasonably with the observed data at all β values. For example, the Nelkin model with $M_H = 18$ gives a curve for $\beta = 0.2$ which peaks at too low κ^2 (Figure II-12). This can be corrected by choosing the $\bar{\delta}_{\nu\tau}$ values so M_H is about 80, but discrepancies persist in the magnitudes and shapes (Figure II-13). Haywood's $\rho(\beta)$ vs β curve was used to associate $\bar{\delta}_{\nu\tau}$ values with E_τ values by assuming

$$\int_{\beta_i}^{\beta_j} \rho(\beta) d\beta \bar{\delta}_{H\tau} \quad (2)$$

Here $\bar{\delta}_{H\tau}$ is associated with a $\beta_\tau = E_\tau/kT$ which is midway between β_i and β_j on Haywood's curve. Only the $\rho(\beta)$ vs β out to $\beta = 7.0$ was used. This corresponds with omitting the part due to the internal frequencies of 0.2, 0.45, and 0.46 eV.

Fig. II-12 $S(\kappa^2, \beta)$ vs β for H_2O on Nelkin Model with $M_H = 83.3$.

Fig. II-13 $S(\kappa^2, \beta)$ vs β for H_2O with $M_H = 83.3$.

The $\bar{\delta}_{H\tau}$ for these are known from theoretical calculations and Equation (1) yields $\sum_{\tau} \bar{\delta}_{H\tau} = 1.59$ for the lower modes. The equation for the $\bar{\delta}_{H\tau}$ is then

$$\frac{1.59 \int_0^{\beta} \rho(\beta) d\beta}{\int_0^{\beta} \rho(\beta) d\beta} = \bar{\delta}_{H\tau} \quad (3)$$

The $S(\kappa^2, \beta)$ or β curves calculated from those parameters are shown in Figure II-14.

The best agreement was obtained by assuming the water molecules exist in different environments. Figure II-15 shows the result of a least squares fit

Fig. II-14 $S(\kappa^2, \beta)$ vs β for H_2O with parameters determined from Haywood's $\rho(\beta)$ function.

Fig. II-15 $S(\kappa^2, \beta)$ vs β for H_2O with mixture of molecules to give least squares fit to Haywood's data.

using the mixture of these three types and two other types of molecules. The least squares fit was based on minimizing the sum of squares between observed and calculated of $S(\kappa^2, \beta)$ at eleven κ^2 and β values. The E_τ and $\bar{\delta}_{H\tau}$ values used in all of these cases are shown in Table II-3.

TABLE II-3

PARAMETERS GIVING LEAST SQUARES FIT TO $S(\kappa^2, \beta)$
% OF COMPONENT M_H , E_τ , AND $\bar{\delta}_{H\tau}$ VALUES

1	150	E_τ	0	0.025	0.198	0.46				
		$\bar{\delta}_{H\tau}$	0.02	1.57	0.46	0.95				
49	150	E_τ	0.0	0.02	0.03	0.06	0.08	0.10	0.198	0.46
		$\bar{\delta}_{H\tau}$	0.02	0.03	0.015	0.475	0.475	0.475	0.470	0.95
34 [a]	100	E_τ	0.0	0.015	0.025	0.045	0.0725	0.125	0.198	0.46
		$\bar{\delta}_{H\tau}$	0.03	0.07	0.06	0.33	0.50	0.60	0.46	0.95
14 [b]	18.2	E_τ	0.0	0.06	0.198	0.46				
		$\bar{\delta}_{H\tau}$	0.165	1.425	0.4637	0.9463				
2 [c]	83	E_τ	0.0	0.02	0.03	0.06	0.08	0.10	0.198	0.46
		$\bar{\delta}_{H\tau}$	0.036	0.024	0.105	0.475	0.475	0.475	0.46	0.95

[a] Based on Haywood $\rho(\beta)$ vs β

[b] Nelkin Model

[c] Mass was chosen to make $S(\kappa^2, \beta)$ vs κ^2 peak at the right κ^2 when $\beta = 0.2$

4.4 Slow Neutron Scattering Calculations Liquid CH₄ and Gaseous CD₄ (G. W. Griffing)

An IBM 7040 program was written which permits calculation of the slow neutron scattering by liquid methane or CD₄ on the assumption that vibrational states of low frequency can be associated with vibrations within an aggregate of CH₄ molecules. The program is quite versatile and can be used to compute the scattering by gaseous CH₄ and CD₄.

Computations have been made on gaseous CD₄ using bound coherent and incoherent scattering lengths of 0.65×10^{-12} cm and -0.41×10^{-12} cm, respectively. Since these are of comparable magnitude the interference effects are very pronounced as can be seen in Figure II-16.

The total cross section of the deuteron is an order of magnitude smaller than that for the proton which means that experimental study of scattering by gaseous CD₄ presents difficulties. However, features of greatest interest can be observed with incident neutrons of 0.1 eV energy and scattering angles

greater than 15° . This presents the possibility that the scattering by liquid CD_4 will be about like that for a gas of the same density and the same temperature. This is because the binding energy of CD_4 molecule to its parent liquid is only about 0.081 eV. Gas model calculations for CH_4 under these conditions agree well with the experiments [15].

4.5 Doppler Effect in Neutron Cross Sections (A. W. Solbrig, Jr.)

In the interpretation of experimental results on total cross section, fission cross section, and total scattering cross section of U-233, U-235, Pu-239, and Pu-241, it is necessary to consider the Doppler effect, or the effect of the velocity distribution of target nuclei. The theory of this effect was reviewed briefly in order to answer the following questions: (a) What form of the Doppler broadening integral is in principle correct? (b) What form of the Doppler broadening integral should be used for analysis or interpretation of experimental results? On the basis of the classical, field-free transport equation, the form of the integral which is correct in principle may be written

$$\sigma^*(s) = \frac{1}{N} \int_{\text{all } \underline{u}} |\underline{s} - \underline{u}| \sigma(|\underline{s} - \underline{u}|) f(\underline{u}) d\underline{u}$$

where $\sigma^*(s)$ = the broadened cross section (fission, absorption, or scattering cross section, or any sum of these cross sections), s = velocity of incident neutron in the laboratory, $\underline{u}$ = velocity of target nucleus in the laboratory, $f(\underline{u})$ = density of target nuclei (assumed to depend on the magnitude but not on

the direction of $\underline{u}$), and $N = \int_{\text{all } \underline{u}} f(\underline{u}) d\underline{u}$. Several forms of the aforementioned

exact integral were written for the special case of Maxwellian $f(\underline{u})$. The decision as to what form of the integral should be used for analysis of experiments may depend on many factors, including convenience, computer time, accuracy desired, and details of the experiments. For most experiments which do not involve light nuclei or exceedingly low energies, it is concluded tentatively that almost any plausible approximation is sufficiently accurate. This conclusion is based on some brief numerical calculations and on the results of other workers. In particular, the approximate form now in use by the Cross Sections Section [16] is probably adequate. It might be worthwhile to check this conclusion by a few numerical test calculations with the exact formula.

Fig. II-16 Partial differential scattering cross section of D_2O .

5. LOW-ENERGY NUCLEAR PHYSICS - DECAY SCHEME STUDIES

(R. L. Heath, C. W. Reich)

5.1 Attempts to Observe Gamma-Ray Spectra of Yb-178 and Gd-162 (R. G. Helmer, L. D. McIsaac)

As reported previously [17], an attempt is being made to study the level structure in odd-odd nuclei by means of the radioactive decay of even-even nuclides. Attempts to observe the gamma-ray spectra associated with the decay of Yb-178 and Gd-162 are discussed herein. These activities should be produced by (2n, γ) reactions on the stable isotopes Yb-176 and Gd-160, respectively.

Yb-178 Experiments

No information has been published concerning the properties of the decay of Yb-178. Although the half-life is not known, it might be expected to be considerably longer than that of Yb-177, which is 1.9 hr.

A 1.5 mg sample of ytterbium oxide enriched in Yb-176 (97.8 percent Yb-176) was irradiated for ten hours in a flux of 2.5×10^{14} n/cm²-sec. The source was allowed to decay for twelve hours so that the rate of production of Lu-177 from the decay of 1.9-hr Yb-177 would be small. The rare-earth elements were then separated by means of an ion-exchange column. The ytterbium fraction was collected and re-separated about 28 hours after the irradiation. Since the lutetium fraction of the second separation still contained Lu-177, a third separation was carried out on the ytterbium fraction about 38 hours after the irradiation.

If Yb-178 had a sufficiently long half-life and had been produced in sufficient quantity, detection of some Lu-178 (16 min) or Lu-178m (30 min) could be expected in the lutetium fraction of either the second or third separations. Neither was observed. This method is more sensitive than looking for gamma rays from Yb-178 due to the Yb-175 and Yb-169 which are present. The experiment was repeated and the same results obtained.

Another sample was irradiated for fifteen days in a flux of 4.9×10^{14} n/cm²-sec. After decaying for four months, the only observable activity was Yb-169 (32 day).

No conclusions can be reached as to properties of Yb-178. The failure to observe radiations from it could result from either a small activation cross section or a half-life short compared to one day or long compared to one month.

Gd-162 Experiment

Some previous attempts to observe the radiations from the decay of Gd-162 have been reported [18]. Although these attempts were unsuccessful, it was suggested that Gd-162 has a long half-life (> 1 yr).

Two samples, each containing 20 mg of enriched Gd-160 (99 percent Gd-160), were irradiated in a flux of 5×10^{14} n/cm²-sec. One sample was irradiated for 75 days and the other for 180 days. After allowing the sample to decay for a month or more, the rare-earth elements were separated by ion-exchange

column techniques. The activities identified were Sc-46, Yb-169, Tb-160, Gd-153, Eu-156, Ce-139, and Ce-141. In the gadolinium fractions, no radiations other than those of Gd-153 (200 day) were observed.

A sample of enriched Dy-164 was irradiated in an attempt to produce Gd-162 by the sequential reaction: $\text{Dy-164}(n,\gamma)\text{Dy-165}(n,\alpha)\text{Gd-162}$. Since the probability for neutron capture is very high, it was thought that this might produce more Gd-162 than the double-neutron capture method. A 10 mg sample was irradiated for 18 days in a thermal flux of 8×10^{14} n/cm²-sec and a fast flux (>1 MeV) of about 2×10^{14} n/cm²-sec. After decaying for two months, the sample was purified by ion-exchange techniques. The activities identified were Yb-175, Yb-169, Tm-170, Ho-166m, Dy-166 and Ho-166, and Tb-160. No gadolinium activities were observed.

5.2 Branching Ratios and Internal Conversion Coefficients for the 145-keV Transition Associated with the Decay of Ce-141 (D. F. Crouch, L. D. McIsaac)

The branching ratio for the 145-keV gamma ray resulting from the decay of Ce-141 has been measured using an extrapolation technique developed by Campion et al for 4π beta-gamma coincidence data [19]. In addition, the K-shell internal-conversion coefficient and the K/L+M ratio have been measured by scintillation and solid-state detectors, respectively. This branching ratio is required for a knowledge of the peak-to-total ratios for low-energy gamma rays.

5.21 Experimental Procedure for Measuring 145-keV Gamma-Ray Branching Ratio. High specific activity Ce-141 was prepared by irradiating Ce in the VH-2 position of MTR. After a sufficient cooling period to permit removal of shortlived Ce isotopes, the Ce was purified by standard rare-earth chemistry. The purity of the resulting Ce-141 was checked by 4π beta counter half-life measurements and by examination of the gamma-ray spectrum.

An important difficulty associated with 4π beta-gamma coincidence counting of Ce-141 is that approximately 30 percent of the betas go directly to Pr-141 ground state. Therefore, the beta-counting efficiency for these betas cannot be determined through beta-gamma coincidence counting, and the absolute beta disintegration rate cannot be calculated using the usual 4π beta-gamma coincidence counter equations [20]. By varying the counting efficiency for beta rays that lead to excited states of Pr-141, a plot may be obtained for beta counting efficiency versus 4π beta counting rate which can be extrapolated linearly to an absolute disintegration rate at 100 percent efficiency. Theoretical and experimental justifications for this technique are given in Reference 19.

Beta counting efficiencies of the 4π beta-gamma sources were varied by adding different amounts of inactive lanthanum carrier to Ce-141 aliquots before drying on VYNS films. Mass of sources varied from approximately zero to 7.5 mg. Activities of the sources were made as nearly equal as practical by careful micropipetting. The pipetting was checked by computer least-squares fitting of the associated gamma-ray photopeaks and was found to vary $\pm 1/2$ percent from the average. The discriminator of the amplifier of the gamma-ray detector was set below the Pr-141 x-ray in order to reduce errors due to counting of conversion electrons in the beta counter [21]. After 4π beta-gamma coincidence counting, the absolute gamma-ray emission rates of the sources were determined using a standard NaI(Tl) gamma-ray scintillation spectrometer.

The emission rates of the 145-keV gamma ray were calculated from the areas of Gaussian curves which were fitted to the gamma ray photopeak, using a least-squares technique. The gamma-ray emission rate is then

$$N_{\gamma} = \frac{N_p}{p E_t A}$$

where N_p is the full energy peak counting rate, p is the peak-to-total ratio, E_t is the total absolute efficiency, and A is the gamma-ray transmission of the beta absorber.

The 145-keV gamma-ray branching ratio N_{γ}/N_D was calculated by dividing the gamma-ray emission rate by the absolute beta rate. The resulting value is 0.454 ± 0.023 .

5.22 K-Shell Internal-Conversion Coefficient and K/L+M Ratio. Because of widely varying experimental results [22] for the conversion coefficients associated with the 145-keV transition in Pr-141, it was felt desirable to remeasure them at this laboratory.

The K-shell coefficient of the 145-keV transition in Pr-141 was measured with a NaI(Tl) gamma-ray scintillation spectrometer. The Ce-141 source was mounted on 0.00015 inch aluminized Mylar tape and analyzed in a low back-scatter geometry to reduce the back-scattering contribution to the photopeak. The area of the photopeak was calculated by fitting a Gaussian to that portion of the photopeak above the backscatter contribution. The approximate Compton response of the crystal for the 145-keV gamma ray was obtained by critically absorbing the x-ray with Cd.

The intensity of the x-ray was corrected for the Compton response, for the escape of iodine K x-rays from the surface of the crystal, and for x-ray absorption in the packing surrounding the crystal. A further correction was made for the K-shell fluorescent yield.

The final x-ray and gamma-ray intensities were corrected for total absolute efficiencies and beta absorber absorption. The K x-ray intensity to gamma-ray intensity ratio was found to be 0.39 ± 0.04 .

The K/L+M ratio was measured using a silicon surface-barrier detector having a 300-micron depletion depth. The areas under the K and the L+M electron peaks of the resulting spectrum were calculated after correcting for the Ce-141 beta radiation and, in the case of the K-electron peak, for the L+M-electron contribution. The ratio of the area of the peaks was found to be 5.3 ± 0.8 , the greatest uncertainty being that of the baseline corrections.

From these data, the total internal-conversion coefficient is 0.46 ± 0.04 . This value together with the gamma- and beta-emission rates yield a branching ratio of 0.66 ± 0.05 for the 145-keV transition in Pr-141.

5.3 The Thermal-Neutron Capture Cross Section of Mn-54 (C. H. Hogg, L. D. Weber)

Considerable interest has been shown recently in the use of the Fe-54(n,p) Mn-54 threshold reaction as a fast-neutron flux monitor. The advantage of this monitor is that the reaction is suitable for irradiation times up to about 600 days. However, some isotopes used as fast flux monitors have large thermal-neutron cross sections [23] which limit their use in long irradiations. An experiment was performed to determine the thermal-neutron cross section of Mn-54 from the measured burnup when irradiated in a high neutron flux.

Samples of very high specific activity Mn-54 were prepared by carrier-free chemical separation from irradiated Fe. The decay rate of each sample was then determined by measurement of the gamma-ray spectrum on a 512-channel pulse-height analyzer. The samples were then irradiated in a thermal-neutron flux of $\approx 5 \times 10^{14}$ n/cm²-sec for periods of 64 and 144 days. The activity of the samples was then redetermined after removal from the reactor. No loss of Mn-54 was detected that could not be attributed to decay.

The 64-day irradiation indicated that the Mn-54 thermal-neutron cross section is less than 30 barns. This result was previously reported in Reference 24. The longer 144-day irradiation indicated that the upper limit for the activation cross section could be reduced to 10 barns.

6. REFERENCES

1. J. W. Coddling et al, "Preparation of Chemically Separated Pa-233 for Cross Section Measurements", MTR-ETR Technical Branches Quarterly Report, July 1 - September 30, 1963, IDO-16932 February 1964 pp 19-22.
2. J. E. Cline and C. W. Reich, "Investigation of High-Spin State in Er-166", PHYS. REV., 129 (1963) pp 2152-2164.
3. P. Strominger et al, "Table of Isotopes", REV. MOD. PHYS., 30 (1958) pp 585-904.
4. B. Keisch and K. T. Faler, "Half-Life Determination of Ho-166", MTR-ETR Technical Branches Quarterly Report, April 1 - June 30, 1962, IDO-16805 (September 1962) p 10.
5. G. W. Griffing, "Scattering of Slow Neutrons by Gaseous Methane", Proceedings Series: Inelastic Scattering of Neutrons in Solids and Liquids, Vol. I, International Atomic Energy Agency, 1963, pp 435-450.
6. A. C. Zemach and R. J. Glauber, "Dynamics of Neutron Scattering by Molecules", PHYS. REV. 101 (1959) pp 118-129.
7. MTR-ETR Technical Branches Quarterly Report, October 1 - December 31, 1962, IDO-16857 (March 1963) pp 20-22.

8. H. L. McMurry and L. J. Gannon, "Calculation of Slow-Neutron Scattering for Complex Molecules", TRANS. AM. NUCL. SOC., 6 (November 1963) pp 280-281.
9. K. A. Strong et al, "Scattering of Slow Neutrons from Propane Gas", PHYS. REV., 125 (1962) pp 933-937.
10. H. L. McMurry et al, "Evaluation of Techniques for Computing Partial Differential Scattering Cross Sections", NUCL. SCI. AND ENG., 15 (1963) pp 438-450.
11. T. J. Krieger and M. S. Nelkin, "Slow-Neutron Scattering by Molecules", PHYS. REV., 106 (1957) pp 290-295.
12. B. C. Haywood and T. M. Thorson, EANDC (Can)-12, March 1962. The Comparisons in Figures 12-15 use more recent data which Dr. Haywood has kindly supplied. The main difference is that at the $\beta = 0.2$ curve peaks at $\kappa^2 = 14$ while the old data peaked near $\kappa^2 = 5$.
13. Mark Nelkin, "Scattering of Slow Neutrons by Water", PHYS. REV., 119 (1960) pp 741-746.
14. H. L. McMurry, "Calculation of Partial Differential Scattering Cross Sections for Slow Neutrons", NUCL. SCI. AND ENG., 15 (1963) pp 429-437.
15. Bull. Am. Phys. Soc. 8, 478 (1963).
16. O. D. Simpson and N. H. Marshall, Doppler Broadening of the Low Energy Total Neutron Cross Section of Pu-241, IDO-16679 (August 1961).
17. R. G. Helmer, "Level Structure of Odd-Odd Nuclei", MTR-ETR Technical Branches Quarterly Report Second Quarter 1963, IDO-16917 (1962) p 19.
18. K. T. Faler and J. M. Hollander, "Heavy Isotopes of Gadolinium and Terbium" in Chemistry Division Quarterly Report (for) June, July, August 1959, UCRL-39 50 (1957) p 98.
19. P. J. Campion et al, "The Efficiency Tracing Technique for Eliminating Self-Absorption Errors in 4 Pi Beta Counting", revised, Intern. J. Appl. Radiation Isotopes, 8, November 1960, p 8-19.
20. P. J. Campion, "The Standardization of Radioisotopes by the Beta-Gamma Coincidence Method Using High Efficiency Detectors", Intern. J. Appl. Radiation Isotopes, revised, 4, June 1958, pp 232-248.
21. P. J. Campion, The Absolute Standardization of Radioisotopes by the Beta-Gamma Coincidence Method Using High Efficiency Detectors, CRP-746, January 1958.
22. C. H. Hogg and L. D. Weber, "Thermal Neutron Cross Section of the Co-58 Isomers and the Effect on Fast Flux Measurements", Trans. Am. Nucl. Soc., 4, November 1961, p 271.

23. C. H. Hogg and L. D. Weber, "Fast Neutron Dosimetry at the MTR-ETR Site", pp 133-140 in Symposium on Radiation Effects and Neutron Dosimetry (1962).

III. INSTRUMENT DEVELOPMENT

1. DETECTOR DEVELOPMENT

(S. D. Anderson)

1.1 Transient Radiation Sensitivity of Pressure Sensors -- Preliminary Results of the August 1963 TRIGA Proof Test (F. D. Terry, R. L. Kindred)

The August 1963 TRIGA Proof Test was conducted to determine the transient radiation sensitivities of the types of pressure sensors slated for use in the STEP and Spert destructive tests. The test was conducted in the three experimental locations of General Atomic's TRIGA "Mark F" Reactor which are tabulated (with their corresponding radiation fields) in Table III-1. The data in Table III-1 assume a 2000 megawatt nuclear pulse with ten milliseconds between the half-power points. The preliminary results presented in this report have not been normalized to the 2000 megawatt pulse, and because of variations in peak reactor power, may be in error by ± 30 percent. A few selected test samples were shielded with cadmium which reduced the thermal neutron flux by a factor of approximately 10^3 and increased the gamma dose rate by a factor of approximately 2.0 in the F ring, 1.4 in the Number 1 tube, and 2.0 in the Number 4 tube.

TABLE III-1

FLUX AND DOSE IN TEST LOCATIONS

Experimental Location	Peak Dose Rate			Dose Per Pulse		
	Neutron		Gamma	Neutron		Gamma
	Fast > 10 keV	Thermal < 0.4 eV	rads/1000	Fast nvt [a]	Thermal nvt	rads
F-ring	2.0×10^{16}	3.4×10^{16}	4.0×10^7	2.5×10^{14}	4.1×10^{14}	4.8×10^5
#1 tube	1.3×10^{15}	1.2×10^{15}	6.3×10^6	1.6×10^{13}	1.4×10^{13}	7.6×10^4
#4 tube	9.0×10^{13}	5.6×10^{14}	7.4×10^5	1.1×10^{12}	6.7×10^{12}	8.9×10^3
[a] 2.5×10^{-9} rads/n-cm ²						

The preliminary results of the Proof Test are briefly summarized as follows:

(1) Kaman Variable Impedance Pressure Transducers. The radiation induced spurious signals from the Kaman transducers in the F-ring ranged from +34 to +41 percent of the rated transducer full scale output. Removing the dummy arm from the radiation field decreased the spurious signal by a factor of two, indicating that the signals from the active and dummy arms add instead of cancel [*].

[*] The transducers have since been modified by the manufacturer such that the signals do cancel. This reduced the spurious signals to less than one percent FS.

Adding a cadmium shield to the transducer reduced the spurious signals by a factor of approximately two.

(2) Omega Variable Capacitance Pressure Transducers. The two Omega transducers, when tested in the F-ring location, produced spurious signals of -58 and -22 percent full scale and +34 percent full scale. (Double polarity signals have previously been observed from several other types of transducers.) The ionization of the air dielectric of the variable capacitor is probably responsible for the large output.

(3) Norwood Bonded Strain Gage Pressure Transducers. Model 114 Norwood transducers produced positive spurious signals of approximately +0.4 to +1.1 percent of full scale in the F-ring. The two Model 112 Norwood transducers tested produced spurious signals with opposite polarities (+0.6 and -1.3 percent full scale in the F-ring).

Model 104 Norwood transducers produced negative spurious signals of approximately -6 to -11 percent full scale in the F-ring. This is approximately one decade higher than for Model 112 and 114 transducers. The postulated reason for the larger signals from the Model 104 transducers was that, when being tested in the standard circuit configuration (ie, an electrical bridge circuit with the two strain gages in the Model 104 transducer representing the two arms of the electrical bridge adjacent to the positive signal lead), the radiation induced currents in the two strain gages and their lead wires did not cancel. This does not occur in Model 112 and 114 transducers since they contain four strain gages in a balanced electrical bridge which does cause the radiation induced currents to cancel. In order to check the postulate, one of the Model 104 transducers was wired into the recording console in such a way that the two strain gages in the transducer represented the two arms of the electrical bridge which were adjacent to the positive power supply lead. This would cause all radiation induced currents to cancel as in a balanced four arm transducer. The modified circuit produced a spurious signal of +0.5 percent full scale as compared to -10 percent for the standard two arm circuit and +0.4 to +1.1 percent for the Model 114 four arm transducers, thus verifying the postulate. In the case of one Model 104 transducer in the standard two arm circuit, cadmium shielding caused the spurious signal to change from -6 to +1.6 percent full scale.

(4) Crystal-Type Pressure Sensors. The results of tests on several types of crystal pressure sensors, including pressure transducers, time-of-arrival gages, hydrophones, and microphones are tabulated in Table III-2. These results show an extremely wide range in the transient radiation sensitivities of the sensors but definitely indicate that some types are usable for making near-core pressure measurements in the Spert and STEP Destructive Tests.

The spurious signals produced by crystal pressure sensors appear to be highly dependent upon neutron to gamma ratio (some

TABLE III-2

TRANSIENT RADIATION SENSITIVITIES OF CRYSTAL TYPE PRESSURE SENSORS

Pressure Sensor					Maximum Transient Radiation Sensitivity in % full scale		
Type Sensor	Type Crystal	Mfg.	Model	Range	F-ring	#1 Tube	#4 Tube
Pressure Transducer	Quartz	Kistler	601A	5K psi		+ 40%	- 18%
Pressure Transducer	Quartz	Kistler	603	5K psi		+ 150%	- 31%
Pressure Transducer	Quartz	Kistler	617	30K psi	+ 13%	+ 55%	
Pressure Transducer	Quartz	Kistler	617A	70K psi	+ 12%	- 9%	- 4.3%
Pressure Transducer	Quartz	Kistler	701A	5K psi		+ 15%	
Pressure Transducer	Tourmaline	Susquehanna	ST-4	2K psi		- 210%	+ 480%
Pressure Transducer	Tourmaline	Susquehanna	ST-4	10K psi	+ 930%	- 200%	- 70%
Hydrophone	Tourmaline	Crystal Research	$\frac{1}{4}$ " Simplex	150K psi	+ 11%	- 4.6%	+ 2.4%
Hydrophone	Tourmaline	Crystal Research	$\frac{1}{4}$ " C-5	150K psi	+ 4.2%	+ 0.57%	
Hydrophone	Tourmaline	Crystal Research	$\frac{1}{2}$ " C-5	150K psi		- 0.069%	
Time-of-Arrival Gage	Barium Titanate	Atlantic Research	BD-15	0.5K psi		- 160%	+ 2.8%
Microphone	ADP	Massa	M-213	0.2K psi		+ 82%	+ 18%

exhibiting polarity changes for a 70 percent change in neutron to gamma ratio).

In conclusion, the preliminary results of the Proof Test demonstrate that several types of pressure sensors (including some crystal types) have sufficiently low transient radiation sensitivities to permit their use for near-core pressure measurements in the impending Spert and STEP Destructive Tests. However, the results also indicate that care should be used in (a) selecting the proper types of circuitry to be used with the transducers; and (b) extrapolating the results of these tests to other radiation fields because of the strong dependence of the spurious signals produced by some pressure sensors upon neutron to gamma ratio.

1.2 ATR Fuel Plate Deflection Test (G. E. McGlothlin)

In the ATR fuel element design, a pressure differential across each of the 19 fuel plates subjects these plates to transverse deflections with corresponding lateral bending stresses. This effect is more pronounced on the two outer plates and to a less severe degree on the remaining plates. The stability of the fuel plates, under transverse loading, is indicated by the magnitude of transverse deflection caused by the pressure differential. In theory, both transverse plate deflection and lateral bending stress are linear functions of the pressure differential. Transverse deflection may thus be obtained from the lateral bending stress. The bending stress may be measured utilizing strain gages mounted laterally on the fuel element, yielding a correlation between lateral strain and transverse plate deflection. To this end, a correlation study of lateral strain and transverse plate deflection was performed on fuel elements Numbers 1 and 19 of an ATR fuel subassembly. This was done by applying pressure internally to the subassembly, with all 19 plates in place, and noting transverse plate deflection and lateral strain which was measured with dial gages and strain gages, respectively. These parameters were measured both at room temperature and 350°F.

Figure III-1 shows a typical fuel plate instrumented with strain gages. The 49.5-inch length fuel plates have an inside radius of curvature ranging from 3.015 inches for plate Numbers 1 to 5.349 inches for plate Number 19. Flat plate theory was used to approximately predict the experimental data. It was found that theory derived for a flat plate with all edges fixed, ie, clamped, more nearly predicted the experimental transverse deflection and lateral strain, the agreement being within 27 percent. The results of a statistical analysis performed on the limited set of data taken showed that, using calibration constants derived in the laboratory tests, transverse deflection may be obtained from measured lateral strain to within an error of 10 percent with a confidence limit of 0.95. Tests on plates Numbers 2 through 18 are currently in progress.

1.3 Fuel Plate Resistance Thermometer (R. H. Meservey, S. D. Anderson)

An interest exists in checking the time-temperature history of the data obtained from thermocouples which are being used in Spert reactors. A device was needed which could be used to make in-core temperature measurements which would be independent of such thermocouple effects as time lag and the "fin effect". Surface attached thermocouples measure the temperature at a point on the outside of the fuel plate cladding and hence, using thermocouple data alone, there is some doubt as to the exact time at which the plate temperature started to rise as well as the magnitude of the temperature within the fuel plate.

A fuel plate resistance thermometer has been designed and fabricated for use in the investigation of possible thermocouple errors. This device utilizes the principle that as the temperature of ordinary metals increases, their resistance also increases.

Basically the apparatus consists of a power supply that furnishes a large current (about 50 amps) which flows through a fuel plate. Voltage taps are

Fig. III-1 Fuel plate instrumented with strain gages.

attached at five-inch intervals along the face of the plate as well as across a current shunt. Since the resistance between adjacent taps is of the order of 50 micro-ohms, shunting effects of reactor water may be neglected. Voltages across each pair of taps are recorded. In this manner the current through the plate and voltage changes across various sections of the plate are known. From this the change in resistance in these plate sections can be computed.

The resistance of the fuel plate may be related to its temperature by the following expression: $R = R_0(1 + \alpha T)$, where R is the fuel plate composite resistance which occurs at the plate bulk temperature T , R_0 is the resistance at 0°C , and α is the fuel plate composite temperature coefficient of resistance.

Figure III-2 is a schematic of the circuit used in the calibration and testing of the resistance thermometer.

After being calibrated to 300°C in an oven at Spert the instrumented fuel plate was placed in a modified fuel can (to insure electrical insulation from the rest of the core) and installed in Spert IV. In the test series the voltages across each of the four, five-inch segments of the instrumented fuel plate were recorded. Thus, the time-temperature history in each five-inch section of the plate could be compared with those given by nearby thermocouples. Particular attention was given to the time at which the thermocouples first detected a temperature change in the core. It appears that thermocouples are capable of giving an accurate time-temperature history of the reactor core during long period (10 to 20 msec) transients such as those from which data were obtained. The results show that a fuel plate resistance thermometer can be a useful tool in the investigation of transient temperatures within reactor cores. The device is reported more completely in a forthcoming report [1].

Fig. III-2 Fuel plate resistance thermometer control circuit.

2. INSTRUMENTATION ANALYSIS (Ned Wilde)

2.1 In-Pile Fatigue Test Temperature Control Analysis (F. K. Hyer, J. W. Sielinsky)

The analog computer study of the temperature control system for the In-Pile Fatigue Test has been completed. The temperature control system configuration is shown in Figure III-3. The heat transfer characteristics of the specimen were simulated by breaking the specimen down axially into 5 modes. Temperature of the first upstream mode was taken as the controlled temperature, T_m . Test conditions require that this temperature vary not more than $\pm 5^\circ\text{C}$ from the set-point temperature, T_{sp} , for a 100 percent reactor-power reduction. The out-pile heater is the primary element for temperature control and is located in the gas stream, W_g , external to the reactor vessel.

Fig. III-3 In-Pile Fatigue Test specimen temperature control system.

This location results in significant transport lag of temperature changes being carried in the gas stream. This heater is driven with a power amplifier which in turn is controlled by a 3-mode controller, TRC-1. Since this heater must be capable of maintaining a gas inlet to outlet temperature differential in the order of several hundred degrees, the time constants of this loop are such that TRC-1 does not realize the compensation necessary to obtain the desired control. Therefore a second heater is located immediately upstream of the specimen and is driven by a power amplifier and 3-mode controller, TRC-2. The temperature error, T_e , is fed to TRC-1 and TRC-2 simultaneously. Since the in-pile heater is designed to correct for the fast transients only, it can be designed with smaller time constants than the out-pile heater. The computer study indicated that the portion of the control loop containing the temperature measurement, TRC-1, and the power amplifier to the in-pile heater must not have a dominate time constant greater than 0.5 seconds.

A root-locus study of this type of system is being made to generalize the time-response that can be obtained by using a 3-mode controller as the compensating device. This study is based upon step disturbances since the actual nature of the disturbances are not known at this time.

2.2 Analog Computer Development (A. W. Hauf)

The construction of a printer control unit has been completed. This unit is completely designed with solid state components in the digital circuitry. It allows information from the digital voltmeter or the timer to be printed on paper tape along with all locations from the analog cross-bar switch. The

printed information contains sign and decimal from the digital voltmeter plus frequency and time from the timer. The purpose of this unit is to speed up data taking and improve the ease with which the Analog Computer may be used.

3. INSTRUMENT COMPONENT AND SYSTEMS DEVELOPMENT

(T. J. Boland)

3.1 Laboratory Fatigue Test Instrumentation (L. G. Cloud, R. H. Brown)

The readout system for the laboratory fatigue tester has been designed, constructed, and placed in operation. The purpose of this system is to condition and record the electrical signals associated with the specimen fatigue test. These signals and the defining symbols are stress, $S(t)$; strain, $\epsilon(t)$; and applied signal, $A(t)$.

The system is diagrammatically shown in Figure III-4.

The operation of the readout instrumentation will be described utilizing Figure III-4 as a reference.

The applied signal $A(t)$ will operate a hydraulic system which will apply a cyclic load to a specific test specimen.

The transducer outputs, $S(t)$ and $\epsilon(t)$ are analog voltage signals. The pre-amplifiers are used to condition these analog signals such that the appropriate signal levels are available for efficient recording with the magnetic tape system.

The information recorded on the magnetic tape may then be played back and any required manipulations of the signals can be performed easily. For example, the stress-strain recording can be made by playing back the tape recording into an X-Y plotter. Note that, with the frequency scaling capabilities of the tape system, accurate stress-strain plots of the higher frequencies can be secured with an accurate electromechanical X-Y recorder.

The readout instrumentations for the laboratory fatigue tester will serve as a prototype for the forthcoming In-Pile Fatigue Tester.

3.2 A General Purpose Tape Perforator (E. P. Elkins, R. I. Little)

The use of digital computers as an aid in circuit and system design and analysis requires that data must be prepared in a form compatible with computer entry. Small amounts of information can be converted from printed, written, or visual form by use of a key punch. Where the data are generated at rates no greater than sixty characters per second, a tape perforator can quite economically

Fig. III-4 Readout system for the laboratory fatigue tester.

record the information. Higher rates require the use of magnetic tape, drums, discs, or even core buffers.

While perforated tape is satisfactory for many recording applications, whether the source be a digital voltmeter, a counter-timer, or a special piece of digital instrumentation, further economies can be gained if the use in any one application is limited and if the perforator logic can be adapted easily to many uses. The design of this General Purpose Perforator stressed such flexibility. Its logic can accept any four-line per digit code, up to eight digits per word and insert the record marks required for programming system compatibility in the NRTS 1401 computer.

Figure III-5 shows a block diagram of the perforator logic. Up to 32 BCD bits (eight decimal digits) feed through a patchboard to the input logic. Either polarity signals are allowed in 1-2-4-8, 1-2-4-2, 1-2-2-4 or 1-2-4-7 codes. Combinations of coding and polarity are permissible, as when recording time from one source and amplitude from another. The patchboard allows flexibility not only in logic and polarity but also in the use of input cabling. The logic also accepts a synchronizing (start) signal of either polarity and feeds back an inhibit signal which the source may use in order to maximize speed of operations. Following a synchronizing signal, the input logic scans (multiplexes) the inputs one digit at a time (only the desired digits) and transmits the digit information sequentially to the drive logic until a complete word is recorded. Periodically, at a spacing established by panel switching, an end of record signal interposes words. The punch drive logic calculates odd bit parity and provides punch actuator power to a Tally model 420 perforator.

Fig. III-5 A general purpose tape perforator block diagram.

The logic system employs solid state circuit modules and power supplies of proven reliability. In its initial application, it perforated over a mile of tape during a reactor noise analysis study.

3.3 Encoder Amplifier Test Set (R. I. Little)

The photoelectric shaft angle encoders used to precisely determine the positions of the regulating and shim rods in the ARMF reactors have provided more than their fair share of maintenance problems. A thorough investigation of this problem shows most of the trouble to be electric in nature: widely varying strobe lamp intensities, strobe lamp timing jitter, an extremely low signal to noise ratio, and temperature and age sensitivity in the binary digit amplifiers (of which each of the four encoders employ seventeen). The investigation also showed that most of the trouble can be averted by using amplifiers with the proper gain and bandwidth characteristics. The Encoder Amplifier Test Set was designed and constructed with two objectives in mind:

- (1) To aid in the design of replacement amplifiers.
- (2) To assist in repair and check-out of existing amplifiers until such time as the replacement amplifiers are available. The existing units have been failing at a rate of one or two per month, primarily due to transistor aging and attendant gain changes. Failures happen intermittently and often only a detailed statistical analysis reveals this type of failure. Poor accessibility makes replacement difficult and time consuming.

Figure III-6 shows the schematic of a test set which is packaged in a 4- x 6- x 8-inch standard electronic box. A simple astable multivibrator, running at approximately 30 cps, triggers a monostable multivibrator which generates a 25 microsecond negative pulse. This pulse feeds a transistor inverting amplifier

Fig. III-6 Encoder amplifier test set.

which produces a clean, $1/2$ microsecond rise and fall time positive pulse. The output network provides amplitude control, attenuation, and source impedance so that the amplifiers under test "see" a source quite similar to the photovoltaic semiconductors employed in the encoders. The power supply provides power to both the test set and the encoder amplifier module under test. An amplifier module (packaged in a wedge shaped case 2 x 1 x 1 inches) contains two complete and separate amplifiers and may be checked out by plugging it directly into the test set. The test set provides synchronization and amplifier output selection and termination to an oscilloscope. Extra test point connectors facilitate the use of the test set during design of the replacement amplifiers.

3.4 Fuel Plate Spacing Gage (B. G. Nelson, J. B. Colson)

A fuel element spacing gage is needed to determine fuel plate spacing before and after operational tests. Fuel plate spacing of 68 to 90 mils must be measured over a 50-inch length to within ± 1 mil.

The probe as designed is made up of 0.040-inch steel tubes spaced $3/8$ inch and are 64 inches long. A sense coil is mounted approximately 3 inches from one end. Electrical connection is made to the coil by means of wires through the center of the supporting tubes. Two eight foot lengths of RG-174/U connects the probe to the instrument (up to 25 feet has been used). The sense coil consists of 16 turns of 5 mil Bondeze wire flat-wound in a rectangular shape, $9/32 \times 1-1/4$ inch. Its inductance measures approximately 6 microhenries before mounting (Figure III-7).

The eddy current loading of the sense coil is determined by the position of a phosphor bronze strip mounted above the coil. The position of this strip

Fig. III-7 Fuel plate spacing gage.

is varied by the plate spacing. To allow measurement of curved plates, a small area in the center of the strip is raised 8 mils. The change of coil impedance with eddy current loading is measured with a bridge circuit and is read out on a meter which has been calibrated in mils. A two megacycle driving signal is connected to the bridge by means of a cathode follower to provide isolation and low impedance drive. The meter reading is not a linear function of fuel element spacing since the impedance of the sense coil does not change linearly with eddy current loading.

The probe is calibrated at spacings of 0.068, 0.078, 0.090, 0.100, 0.120, and 0.130 inch which are obtained from a test block. The test block is constructed of 0.050-inch aluminum plates mounted in slots milled in 1/2-inch thick side plates mounted on a 1/2-inch aluminum base plate.

4. REFERENCES

1. R. H. Meservey, Fuel Plate Resistance Thermometer; IDO-16965 (March 1964).

IV. APPLIED MATHEMATICS AND MACHINE COMPUTATIONS

1. IBM 7040 PROGRAM TO SIMULATE THE IBM 650 (A. V. Grimaud, D. H. Speas)

An efficient program for simulating the IBM 650 on the IBM 7040 was placed in service.

This program consists of two parts: (a) a 650 simulator package written by IBM for the 7040, and (b) a program for simulating the 650 input output control panels. These two programs were modified to compile together under the present operating system. The following schematic illustrates the loading sequence.

The SIMPU deck (Program Number 40.0087) will produce punched cards which are identical to what the 650 card output would have been. These cards must be listed on the 407 using the correct control panel. The 1403 printer listing will be 80-80 with part of the information missing.

The SIMPR deck (Program Number 40.0088) will produce 1403 printer output only (no cards). The listing will be equivalent to what the 407 listing of the output cards would have produced if they were listed on the correct control panel.

The 650 program deck consists of three parts: (a) two simulator control cards, (b) a set of control panel simulation control cards for IO, and (c) the 650 program deck. All three elements must be correctly assembled by someone familiar with the Simulator, the 650, the Lackland control panel subroutine, and the 7040. Since errors in these decks cause insurmountable operations trouble the procedure has been adopted that all 650 programs will be originally set up for simulation on the 7040 by the programming section (D. H. Speas or A. V.

Grimaud). After decks are once set up and labeled they are not to be altered by anyone. Address stops in the 650 programs which indicate problem completed should be replaced by a return to read another card. This will enable the 7040 to dump the buffer areas containing the last output.

The Data decks are identical to the 650 cards in most details.

The simulator package is working with complete accuracy. It is running at approximately three times the speed of the 650. Considerable difficulty is encountered if data are incorrect since there is little or no opportunity for manual operator intervention to correct troubles.

Following is a list of programs which have run, and are currently available for use with the 650 simulator.

	Modified Gaussian
	Gain Shift
	Velocity Selector
00.037	MUFT R Program
00.049	Polynomial Least Squares Fitting Program
00.066	Exponential Regression
00.067	Least Squares Fitting
00.111	MUFT D Blackness Coefficient
00.118	Nonlinear Least Squares Fitting
00.121	HEAT-1
10.013	Reactor Heat Flux
10.033	MTR Burnout
20.055	MTR Pricing
20.058	ETR Pricing

2. DPMIN -- IBM 7040 DOUBLE PRECISION MATRIX INVERSION
AND SOLUTION OF SIMULTANEOUS LINEAR EQUATIONS PROGRAM
(D. H. Gipson -- PP Co. Program Number 40.0227)

Many problems arise in matrix algebra where an inverse matrix or the solution of a set of simultaneous linear equations is to be found. This program was written to solve these problems with considerable accuracy.

The method [1] is basically a double pivotal Gauss-Jordan reduction method with the pivot terms being selected from the largest eligible elements of the matrix. The matrix is inverted and solution vectors formed in the same area as original matrices were stored.

The inverse matrix and solution vectors are computed using double precision arithmetic. The program also computes the identity matrix and the determinant of the original matrix.

3. SPUDNIC -- PROGRAM TO SOLVE FOR PARTITION OF URANIUM, DIBUTYL PHOSPHATE, AND NITRATE IN C-COLUMN (D. H. Gipson, H. T. Hahn)

The purpose of this code is to provide a numerical solution for the mass balance and equilibrium constant equations which describe the uranyl nitrate (UNH)--Nitric acid (HNO₃)--dibutyl phosphoric acid (HDBP)--tributyl phosphate (TBP)--AMSCO system. There is thus obtained a chemical description of each species and prediction of their distribution during the reextraction of uranium from a degraded TBP phase into immiscible dilute HNO₃.

The mass balance equations reduce to a system of simultaneous nonlinear equations with five unknown specie values. The method of solution consists essentially of linearizing and approximating the nonlinear equations by means of a truncated Taylor's series. The mass balance equations are written as

$$g_i(p_1, p_2, \dots, p_5) = 0 \quad i = 1, \dots, 5. \quad (1)$$

where $(p_i, \quad i = 1, 2, \dots, 5)$ are the unknown specie values.

If all terms of order higher than the first in the infinite Taylor's series are ignored, a formula is obtained which when applied to the mass balance equations gives

$$\begin{aligned} g_i(p_{1,1}, \dots, p_{5,1}) &\approx g_i(p_{1,0}, \dots, p_{5,0}) + G_1^i \Delta p_1 \\ &+ G_2^i \Delta p_2 + \dots + G_5^i \Delta p_5 = 0 \end{aligned} \quad (2)$$

for,

$$i = 1, 2, \dots, 5$$

where

$$G_j^i = \frac{\partial g_i(p_{1,0}, \dots, p_{5,0})}{\partial p_j} \quad (3)$$

and

$$\Delta p_j = p_{j,1} - p_{j,0} \quad j = 1, 2, \dots, 5. \quad (4)$$

Given initial estimates, $(p_{1,0}, \dots, p_{5,0})$, of the species, the system of equations, Equation (3), can be solved directly for the values Δp_j , ($j = 1, \dots, 5$), since they are linear in the system. Thus, new estimates of the parameters are obtained from the relationship

$$p_{j,1} = p_{j,0} + \lambda_m \Delta p_j \quad j = 1, \dots, 5. \quad (5)$$

where λ_m is a computed value used to speed up the process of finding the best values for the species. The process is repeated until, after k iterations, the sum

$$\Phi = \sum_{i=1}^5 \epsilon_i(p_{1,k}, \dots, p_{5,k}) \leq \epsilon$$

where ϵ is some prescribed number close to zero. The program is written in FORTRAN for the IBM 7090.

4. CALIBRATION OF THE ARMF-I AND ARMF-II REGULATING RODS (D. H. Gipson)

The recalibration of the ARMF-I and ARMF-II regulating rods was performed in November 1963. This calibration for both reactors will be referred to by a calibration code of zero which will be utilized in the new ARMF data analysis programs.

A least squares fit technique is used to obtain the functional relationship between relative reactivity and regulating rod position.

The method of finding a functional relationship is the following. Assuming the true function to be $\rho = f(r)$ where r is the position of the regulating rod and ρ is the relative reactivity, an approximating function, $\rho_n = F_n(r)$, is to be found, such that the functional difference $f(r) - F_n(r)$ is as small as possible for all values of r in some specified range of the regulating rod.

A small sample of poison or fuel with assumed constant relative reactivity is used to obtain paired readings of regulating rod position. The paired readings are the regulating rod positions with the sample in, and out of the reactor core, respectively, at different points over the space of the regulating rod.

Letting

$\Delta \rho$ = relative reactivity worth of sample,

r_2^j = regulating rod position with sample in,

r_1^j = regulating rod position with sample out,

then

$$\Delta\rho = f(r_2^j) - f(r_1^j)$$

for m pairs of readings, $j = 1, 2, \dots, m$.

From the approximating function,

$$\rho_n = F_n(r) = a_0 + a_1 G_1(r) + a_2 G_2(r) + \dots + a_n G_n(r)$$

the values of the coefficients a_i , ($i = 1, 2, \dots, n$), are to be found such that the function

$$\Delta\rho_n^j = F_n(r_2^j) - F_n(r_1^j)$$

is a good fit to the experimental values (r_2^j, r_1^j) and $\Delta\rho$.

The method of least squares is used to evaluate the coefficients where the function to be minimized is

$$S(a_1, a_2, \dots, a_n) = \sum_{j=1}^m (\Delta\rho_n^j - \Delta\rho)^2.$$

The value a_0 is determined by setting $\rho_n = 0$ at a specified regulating rod position.

The relative reactivity worth of the calibration sample $\Delta\rho$ is actually unknown until the calibration curve is calculated. An arbitrary value is assigned to the value $\Delta\rho$ to produce an unnormalized calibration curve. This curve is then normalized to the correct values of relative reactivity using the relative reactivity between two regulating rod positions found by period measurements.

4.1 ARMF-I

The ARMF-I regulating rod was calibrated in two core locations, the center and E-2 positions. The relative reactivity obtained by period measurements, 1183.22 μk from rod positions 122.42 to 1572.70, was used to normalize the curves to correct values of relative reactivity. There were 68 and 76 pairs, respectively, of experimental points (r_2^j, r_1^j) used in determining the calibration curves for the center and E-2 positions. The data were taken November 4, 1963.

The following trigonometric function was found to be the best fit for the two positions

$$\rho_4 = a_0 + \sum_{k=1}^4 a_k \sin k\theta + \sum_{k=1}^4 b_k \cos k\theta$$

where $\theta = \pi r/1600.0$

for $115 \leq r \leq 1540$.

Coefficients for Center Position

$$\begin{aligned} a_0 &= 549.02558 & b_1 &= 606.68792 \\ a_1 &= 106.21696 & b_2 &= 47.729649 \\ a_2 &= 4.5252387 & b_3 &= -13.807608 \\ a_3 &= -13.505504 & b_4 &= -3.8749316 \\ a_4 &= 0.29754210 \end{aligned}$$

Coefficients for E-2 Position

$$\begin{aligned} a_0 &= 554.98283 & b_1 &= 602.65339 \\ a_1 &= 100.64213 & b_2 &= 40.749383 \\ a_2 &= 9.3451538 & b_3 &= -10.299985 \\ a_3 &= -11.767069 & b_4 &= -3.3788066 \\ a_4 &= -0.94709891 \end{aligned}$$

4.2 ARMF-II

The relative reactivity obtained by period measurements, $264.70 \mu k$ from rod positions 650 to 950, was used to normalize the ARMF-II regulating rod calibration curve. There were 83 pairs of experimental points (r_2^j, r_1^j) used in determining the calibration curve. The last of the data was taken November 4, 1963.

The following trigonometric function was found to be the best fit.

$$\rho_3 = a_0 + \sum_{k=1}^3 a_k \sin k\theta + \sum_{k=1}^3 b_k \cos k\theta$$

where $\theta = \pi r/1600.0$

for $450 \leq r \leq 1500$.

Coefficients

$$\begin{aligned} a_0 &= 360.35408 & b_1 &= 398.32743 \\ a_1 &= 117.46679 & b_2 &= 35.195464 \end{aligned}$$

Coefficients (Continued)

$$a_2 = 26.287097$$

$$b_3 = -2.7778766$$

$$a_3 = -3.2379034$$

5. DOUBLE PRECISION ARITHMETIC SUBROUTINES

(L. J. Gannon, D. H. Speas -- PP Co. Program Number 40.0221)

Floating point arithmetic on the IBM 7040 computer uses the equivalent of eight significant digits. For many applications this is entirely adequate. However, in some types of work (notably matrix inversion) eight significant digits are not sufficient to prevent serious loss of significant digits. (Significant digits are lost when numbers of the same magnitude are subtracted.)

A double precision subroutine package has been written, so that a programmer using the MAP programming language can do a problem using double precision (16 digit) arithmetic. The subroutine package consists of two parts. The first part consists of a symbolic deck of the macro definition of the double precision op codes and must precede the programmers, MAP coding. The second part consists of the actual add, subtract, multiply, and divide subroutines which will normally be used as a binary deck.

6. COMPLEX ARITHMETIC SUBROUTINES

(D. H. Speas -- PP Co. Program Number 40.0228)

There has been a need for a means of solving problems involving use of complex numbers, with the IBM 7040 computer.

A complex arithmetic routine package has been written and checked, so that a person can program in the MAP programming language, and utilize the complex arithmetic commands.

This package consists of two main parts. The first of which contains macro definitions of the complex arithmetic operations, and precedes the individual's main MAP code. The second part consists of the actual commands ie, add, subtract, multiply, and divide subroutines, which can be utilized as a binary deck.

7. CONVERSION OF THE ULCER CODE

(K. R. Dickey -- PP Co. Program Number 40.0024)

The ULCER (Upscattering Library Code for Examining Reactors) has been converted for use on the IBM 7040 computer. All features of the code that are available on the 7094 computer at Bartlesville are available with the 7040 version of the code.

The main problem in converting the ULCER code was in reducing the core storage requirements of the code [*]. Fortunately the library tape presently used by the code has been generated to only 23 groups. The 7094 ULCER code was written to accommodate a 40 group library tape. Therefore, the storage requirements of the code were reduced by lowering the number of groups the code could process.

8. IBM 7040 NONLINEAR MULTIPLE "MODIFIED
GAUSSIAN" LEAST SQUARES FITTING PROGRAM
(M. H. Putnam -- PP Co. Program Number 40.0001)

A program has been written in FORTRAN for the IBM 7040 which will perform a least squares fit to the function,

$$\bar{Y}_1 = \sum_{j=1}^J \bar{Y}_{1j} + \bar{Y}$$

where

$$\bar{Y}_{1j} = Y_{0j} \cdot e^{-\left(\frac{X_i - X_{0j}}{W_{0j}}\right)^2} \left[1 + \alpha_{1j}(X_i - X_{0j})^{M_1} + \alpha_{2j}(X_i - X_{0j})^{M_2} \right]$$

and

$$\bar{Y} = aX_i + b$$

Input to the program consists of pairs of numbers (Y_i , X_i) where the X_i are nonnegative integers. In the function M_1 and M_2 are even integers (given), Y_{0j} , X_{0j} , W_{0j} , α_{1j} , α_{2j} and a and b are parameters to be determined in the fit (except where they are constrained by options). The program permits the fit parameters to vary, to be held constant and, also, in case of W_{0j} , α_{1j} , and α_{2j} to be computed as functions of X_{0j} .

The program is capable of fitting a maximum of ten Gaussians. Fits can be made to the Gaussians only or a line may be added. The areas under the Gaussians as well as the errors in these areas are computed.

[*] This is necessary because of the increased size of the monitor program used on the 7040 when compared with the storage requirements of the 7094 Fortran II Monitor.

8.1 Method of Solution

The principle of least squares states that the best representation of the data under consideration is that which makes the sum of the squares a minimum.

$$S = \sum_{i=1}^n W_i \left[Y_i - \bar{Y}_i \right]^2 = \text{minimum}$$

The W_i term identifies the weights associated with each of the Y_i values.

To minimize the function, S , the function is differentiated with respect to each of the fit parameters (P_k). The derivatives are set equal to zero and the resulting set of simultaneous equations are solved using the Gauss Method.

$$\partial S / \partial P_k = -2 \sum_{i=1}^n W_i \left[Y_i - \bar{Y}_i \right] \left(\frac{\partial \bar{Y}_i}{\partial P_k} \right)$$

The Gauss Method is used in solving a set of equations when linearization is not feasible. It consists of linearizing the function in question with respect to each of the parameters by means of a truncated Taylor's series. Using initial estimates of the parameters to evaluate the coefficients of the expansion, new estimates are obtained. The process is repeated using the new estimates until the convergence criterion for each fit parameter is satisfied.

An IDO containing a more detailed discussion of this program will be issued during the fourth quarter of 1964.

9. AN IBM 1401 PAPER TAPE TO MAGNETIC TAPE PROGRAM

(G. A. Jayne -- PPCo File code 14.0052)

This program reads paper tape records, each up to 80 characters long, and writes them on magnetic tape in records 84 characters long. Zeros are written on the magnetic tape between the last character of the paper tape record and character 84. The 84th character is a digit which designates whether that particular record has no errors, has one or more parity errors, is of incorrect length, or is the last record. A data card specifies the length of paper tape, record to be checked, and supplies several output options.

10. AN IBM 7040 FORTRAN PROGRAM FOR THE ANALYSIS OF GAMMA SPECTRA

(G. A. Jayne -- PPCo File Code 40.0015)

It is desired to find the intensity of each nuclide known to be present in a gamma-emitting source. An existing 7040 program inspects the composite spectrum from the source and calculates these intensities with a least squares fitting method, holding the positions of the component photopeaks fixed. Since

these positions are not known exactly, it is believed that a better fit to the composite spectrum, and hence better answers for the intensities, can be obtained by allowing the locations of the photopeaks to vary, within reasonable limits. The SISYPHUS program is being written for this purpose. It shifts the position of one or more photopeaks, then calls a subroutine which generates new spectra shapes, given their photopeak positions. If the new shape or shapes give a better fit to the composite spectrum, in a least squares sense, then the new shapes and new photopeak location replace the old values, and the intensities are recalculated.

11. FIRST FLIGHT FINITE CYLINDER ESCAPE PROBABILITY (CEP)

(S. Tong, C. W. Berner -- PPCo Program number 40.0161)

A Monte Carlo program written in MAP for IBM 7040 to calculate the escape probability of neutrons in concentric finite cylinders is completed. The program has built-in "Student-t" statistical analysis for accuracy desired. The running time of the program is entirely a function of the number of cells, the desired accuracy, and the degree of confidence. The program processes approximately 10,000 cells per minute.

12. GENERAL ATOMIC MUFT (GAM)

(S. Tong, C. W. Berner -- PPCo Program number 40.0074)

GAM has been converted for IBM 7040 with the restriction that the total number of output groups is limited to seven or less.

13. SUBROUTINE FOR COMPUTING THE PSI AND CHI FUNCTIONS (PCF)

(S. Tong, C. W. Berner -- PPCo Program number 40.0162)

The resonance line shape functions, Psi and Chi, used for the Breit-Wigner cross section calculation in Monte Carlo have been programmed and checked. The method for the Psi and Chi calculations is a combination of Hermite-Gaussian and Trapezoid integrations. The time required to calculate the Psi and Chi functions is about one millisecond per point, which is believed to be about twice as fast as any other method known.

14. PROGRAM FOR PROCESSING TOTAL CROSS SECTION DATA

(N. H. Marshall -- PP Co. Program Number 40.0078)

The cross sections group is now using an early version of the program which requires the input data to be on cards. This version allows fitting (or

smoothing) [*] of the fast background data. Two major revisions have been started. The first, which is completed, will allow averaging of each individual run; the second, not yet completed will allow averaging of several runs together. Work is now in progress to complete this second revision. Work also is started towards converting the program to read magnetic tape input as this will be the standard form of the output data from the 4096 fast chopper.

15. TEST OF NUMERICAL INTEGRATION SUBROUTINES (L. A. Schmittroth)

Two 7040 programs were written to check and compare several numerical integration subroutines. Of the routines tested, ROMBRG, ROMCOS, ADSIM, and ADGAS, (the subroutines are filed under PP Co. File Numbers 40.0044, 40.0045, 40.0042, and 40.0043, respectively) have been described in a previous quarterly report [2]. ROMSN 1, 2, and 3 are described here.

The first program, named PKTEST, uses the Poisson Kernel:

$$P(\alpha, r, \theta) = \frac{(2\pi)^{-1} (1-r^2)}{1 - 2r \cos(\alpha-\theta) + r^2}$$

where

$$0 \leq r < 1, 0 \leq \alpha \leq \pi, -\pi \leq \theta \leq \pi.$$

This is a well-known function [3] which appears in the theory of Fourier Series, and elsewhere, and is especially well-suited to testing numerical integration subroutines. It is periodic of period 2π , and hence (for small r) will be favorable to trapezoidal integration. The integral over a period is always 1, regardless of the value of α or r . However, the most interesting property is that it approaches a "delta function" as r approaches 1:

$$\lim_{r \rightarrow 1} P(\alpha, r, \theta) = \begin{matrix} 0 & 0 \neq \alpha \\ +\infty & \theta = \alpha \end{matrix}$$

[*] Fitting of the fast background curve is accomplished by using a standard least square to a parabola in the regions where the backgrounds are flat, and using a Fourier smoothing technique where the backgrounds contain structure. This Fourier technique had been previously written for the IBM 650 and 7090, and has now been converted to the 7040 computer. This fitting scheme seems to do well except at the junction between the two fits. A procedure for smoothing out this junction remains to be developed.

$$\int_{-\pi}^{\pi} P(\alpha, r, \theta) d\theta = 1$$

$$\lim_{r \rightarrow 1} \int_{-\pi}^{\pi} f(\theta) P(\alpha, r, \theta) d\theta = f(\alpha).$$

A typical plot of $P(\alpha, r, \theta)$ is shown below:

where $P_M = P(\alpha, r, \alpha) = \frac{(2\pi)^{-1} (1+r)}{1-r}$ is the value of the maximum of the function. The width in radians at half-maximum is $\delta = 2 \cos^{-1} \left(\frac{4r^2 - r^2 - 1}{2r} \right)$.

As a parameter indicating width of the peak, $w = \delta/2\pi$ is taken which can be called the fractional width. In the program, w is read-in, and r computed. It is intuitively clear that as $w \rightarrow 0$ (ie, as $r \rightarrow 1$) the function becomes increasingly more difficult to integrate numerically; this is shown clearly in the results.

There are two other parameters read-in, α (the location of the peak) and ϵ (the relative error criterion). In numerical integration, the exact value, I , is never sought but rather the approximation A such that $|(A-I)/I| < \epsilon$. It is important to note that if a result is required with a relative error of ϵ , then any effort expended in computing an approximation with less error is a waste of computer time. In fact a routine which consistently uses excessive computer time to produce results which are too accurate should be discarded.

The routines tested [2] ROMBRG, ROMCOS, ADSIM, and ADGAS, and ROMSN 1, 2, and 3 automatically stop after a certain depth of subdivision has been reached. The fineness of subdivision is not necessarily uniform, and in fact the two routines ADSIM and ADGAS adapt the number of points used to the difficulty of integration over subintervals. In all of them, however, the strategy used is essentially as follows: If $|(A_n - A_{n-1})/A_{n-1}| < \epsilon$ then A_n is chosen as the final result. This is of course not equivalent to $|(A - I)/I| < \epsilon$ where I is the true value of the integral. However, it is gratifying to note that in all cases which have been run, the error committed has been less than the error allowed.

Some results of the test program, PKTEST, are as follows:

Case 1

Peak location: $\alpha = 0$

Peak width: $w = 0.5$

Error criterion: $\epsilon = 0.001$

	Number of functional evaluations	Actual error (1.0 - computed result)
ROMBRG	17	-0.00022851
ROMCOS	31	-0.00005391
ADSIM	31	0.00001379
ADGAS	42	0.00000122

Case 2

Peak location: $\alpha = 0$

Peak width: $w = 1$

Error criterion: $\epsilon = 0.001$

	Number of functional evaluations	Actual error (1.0 - computed result)
ROMBRG	33	0.00000083
ROMCOS	31	0.00000085
ADSIM	31	0.00001483
ADGAS	18	0.00003815

Case 3

Peak location: $\alpha = 0$

Peak width: $w = 0.01$

Error criterion: $\epsilon = 0.001$

	Number of functional evaluations	Actual error (1.0 - computed result)
ROMBRG	513	-0.00070243
ROMCOS	1023	-0.00025687
ADSIM	151	-0.00008599
ADGAS	210	0.00000941

Case 4

Peak location: $\alpha = 1$

Peak width: $w = 0.01$

Error criterion: $\epsilon = 0.001$

	Number of functional evaluations	Actual error (1.0 - computed result)
ROMBRG	513	0.00083359
ROMCOS	1023	0.00026517
ADSIM	163	0.00000130
ADGAS	186	0.00001722

Case 5

Peak location: $\alpha = 1.20$

Peak width: $w = 0.01$

Error criterion: $\epsilon = 0.001$

	Number of functional evaluations	Actual error (1.0 - computed result)
ROMBRG	513	0.00133111
ROMCOS	1023	-0.00014745

Case 5 (Continued)

	Number of functional evaluations	Actual error (1.0 - computed result)
ADSIM	175	0.00000478
ADGAS	174	0.00001468

Case 6

Peak location: $\alpha = 1.4$

Peak width: $w = 0.01$

Error criterion: $\epsilon = 0.001$

	Number of functional evaluations	Actual error (1.0 - computed result)
ROMBRG	513	0.00140990
ROMCOS	1023	0.00017019
ADSIM	187	-0.00003156
ADGAS	174	0.00002592

Case 7

Peak location: $\alpha = 1.59$

Peak width: $w = 0.01$

Error criterion: $\epsilon = 0.001$

	Number of functional evaluations	Actual error (1.0 - computed result)
ROMBRG	513	0.00102203
ROMCOS	1023	0.00018209
ADSIM	211	0.00000513
ADGAS	162	-0.00000539

A second test program was written for evaluating the integral

$$I = \int_{-1}^1 \frac{g(x) \, dx}{\sqrt{(1-x^2)(q^2-x^2)}} \quad \text{where } |q| < 1$$

and $g(x) = -x^3$ for $x \leq 0$, $g(x) = x$ for $x \geq 0$. Note that this integral has a $1/\sqrt{x}$ type singularity at both endpoints, and that as $q \rightarrow 1$, the integral diverges. In addition, $g(x)$, $g'(x)$, and $g''(x)$ are continuous, but $g'''(x)$ is discontinuous at the origin.

Because of the endpoint singularities, the numerical integration procedures must avoid use of endpoint values. There are three such procedures available on the 7040: ROMCOS, ROMSN, and ADGAS. The ROMSN method is described in detail in an article by Rustishauser [4]. He develops a formula:

$$T(h) = h[\alpha f(\beta h) + f(h) + \dots + f(a-h) + \frac{1}{2} f(a)]$$

in other words a modification of the trapezoidal rule whereby α and β are chosen so that the error law is

$$T(h) - \int_0^a \frac{\phi(x)}{\sqrt{x}} dx = c_1 h^2 + c_2 h^4 + \dots$$

for the special cases $\phi(x) = 1$ and $\phi(x) = x$. As is shown in the same article, this allows the Romberg transformation to be carried out. Three subroutines for the 7040 were programmed using the above method: ROMSN 1, 2, 3 for integrals with a square-root singularity at the left endpoint, right endpoint, and both endpoints, respectively. ROMSN 3 was used in the test program with excellent results.

In interpreting the results shown below one should remember that as $q \rightarrow 1$, the integral diverges and, hence, will be increasingly more difficult to integrate numerically.

Case 1

$$q = 10$$

$$\epsilon = 0.001$$

$$\text{Exact value} = 0.16727023$$

	Number of functional evaluations	Actual relative error (True - computed)/True
ROMCOS	511	-0.00092496
ROMSN3	25	-0.00000718
ADGAS	450	0.00029632

Case 2

$$q = 1.1$$

$$\epsilon = 0.001$$

Exact value = 2.6543600

	Number of functional evaluations	Actual relative error (True - computed)/True
ROMCOS	1023	-0.00062286
ROMSN3	43	-0.00003283
ADGAS	450	0.00040690

16. OPERATING SYSTEM FOR 1401-7040 (W. A. Hestir, E. D. Ritchie, L. L. Marsden)

16.1 1401 Multiple Utility Program

The standard 1401 utility programs for 7040 input and output as supplied by IBM are not simultaneous processing routines.

1401 programs which allow simultaneous processing on the 1401 have been written. Improved hardware has been ordered for the 1401 and the programs will be redone to utilize the increased hardware speeds.

16.2 7040 Operating System

The 7040 operating system is more general than the 7090 operating system being used in Bartlesville in that all types of data processing, eg, commercial and scientific, are under a single monitor system. The conversion of the nuclear codes library to the 7040 has progressed smoothly considering the differences in the "chain" feature and core storage available to the object program. We have written our own "chain" subroutine to maintain as much compatibility with the 7090 as possible.

The initial Version 1 of the 7040 operating system as received had many errors in it that have largely been removed by IBM. Operation under Version 3 is now underway with modifications, such as cost accounting, included. Each succeeding version should be more error free and easier to initiate.

17. REFERENCES

1. Alex Orden, "Matrix Inversion and Related Topics by Direct Methods", pp 39-55 of Mathematical Methods for Digital Computers, Wiley and Sons, Inc., New York, 1960.

2. MTR-ETR Technical Branches Quarterly Report, 3rd Qtr 1963, IDO-16932 (1963).
3. H. W. Carslaw, Introduction to the Theory of Fourier's Series and Integrals, 3rd Ed. New York: Dover, 1952.
4. H. Rustishauser, Ausdehnung des Rombergschen Prinzips, Numerische Mathematik 5, 1, 1963, p 48.

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