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**NUCLEAR TECHNOLOGY BRANCHES
QUARTERLY REPORT**

October 1 - December 31, 1966

IDAHO NUCLEAR CORPORATION
NATIONAL REACTOR TESTING STATION
IDAHO FALLS, IDAHO

U. S. ATOMIC ENERGY COMMISSION

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SUMMARY

Self-shielding effects of boron carbide particles of various sizes have been studied. A comparison of values obtained by (a) reactivity measurements in the ARMF and (b) first-flight transport calculations indicates that results obtained by the two methods agree well for particle sizes smaller than 85 microns. The self-shielding factor, f_p , for -325 mesh-size particles now being used in ETR and ATR fuel elements is 0.94.

Insufficient fuel-plate to side-plate engagement was apparently the cause of premature failure of two ATR dummy fuel assemblies during the hydraulic proof test; whereas, inadequate swaging technique resulted in the failure of one fueled ATR fuel assembly during a similar test. Four other ATR dummy assemblies successfully withstood the hydraulic proof test although one was marginal in its performance.

The fuel core width tolerances for the ATR fuel plates have been relaxed as a result of further investigations.

A review of the allowable dogboning in MTR fuel assemblies has been completed, and the results are reported.

Improvements in the response time of commercially available, sheathed, Type K thermocouples have been obtained by remaking the hot junctions using copper and silver rather than stainless steel.

Post-operation measurements of the channel thicknesses of ETR fuel elements were made and are reported. Significant changes in these dimensions were observed.

The results are given of an irradiation test program conducted for the Argonne Advanced Research Reactor. In this test, stainless steel clad plates fueled with UO_2 were irradiated in the G-12 pressurized water loop of the ETR at heat fluxes of 0.6 to 1.1×10^6 Btu/hr-ft² to exposures of up to 14.6×10^{20} fissions/cc. Post-irradiation examinations indicated that the samples were sound, that the fuel core swelling rates were lower than predicted, and that the failures produced by post-irradiation annealing were in good agreement with failure-no-failure data produced during irradiation testing by others on similar samples.

Compatibility studies were made on tantalum, graphite, and Al_2O_3 with inter-metallic fuel (UAl_x , U_3Si , or UCu_5) systems for the purpose of selecting a suitable heat transfer medium. Graphite appears to be most suitable.

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I. REACTOR ENGINEERING

1. CRITICAL FACILITIES

(E. E. Burdick)

1.1 Self-Shielding in Boron Carbide Particles (E. F. Aber, E. E. Burdick)

Self-shielding in boron carbide particles is of special interest now that ETR and ATR fuel elements are being fabricated by powder metallurgical techniques. The particle size being used is -325 mesh. Theory predicts that particles as large as 325 mesh (44 microns) experience self-shielding effects. The effects of boron self-shielding are particularly significant in (a) boron assaying of fuel elements and (b) excess reactivity determinations (burnup response). Since the boron content of ETR and ATR fuel elements is assayed by reactivity measurements, the assayed content can be in error by an amount inversely proportional to the self-shielding factor if corrections are not made. On the other hand, boron self-shielding can be used to an advantage in reducing excess reactivity variations as a function of burnup.

Self-shielding in boron carbide particles has been evaluated both by an experimental program in the ARMF^[1] and by first-flight transport calculations. Results of these evaluations, presented below, show that the two methods closely agree for particle sizes smaller than 85 microns (170 mesh) and that the -325 mesh particles used in the experimental samples have a self-shielding factor of 0.94.

1.11 Experimental Program. Self-shielding factors were determined experimentally from ARMF reactivity measurements on a series of small, high-purity aluminum sample plates which contained boron in the form of boron carbide. An unborated control plate was made from the same high-purity aluminum and by the same techniques. The plates were fabricated by powder metallurgical techniques^[2] and were identical in design to fuel plates, except that they contained no fuel; ie, the plate was comprised of a core containing the boron, cladding, and a "picture frame" edging. Seventeen plates, each containing 13 mg of natural boron, were studied. The seventeen plates were divided into four groups, according to the size of the carbide particles. The different particle sizes were obtained by screening the boron carbide into mesh size intervals of -100 +120, -170 +200, -270 +325, and -325.

The reactivity measurements were made in the corner of the ARMF-I central water hole in a position which placed the sample plates parallel to and 0.57 inch from the ARMF fuel elements. The net reactivity effect due to boron was obtained by subtracting the reactivity effect of the unborated control plate. The results of these measurements, shown in Table I-1, clearly indicate from the spread of 26 percent in the reactivity values that self-shielding does exist. It should also be noted from the table that the standard deviation of the reactivity effects of the plates within a given group is small (<1 percent of the mean reactivity). This shows that control of the amount of boron carbide in the plates was very good and increases the confidence in the results of this study.

TABLE I-1

REACTIVITY AND SELF-SHIELDING
FACTORS FOR BORON CARBIDE PARTICLES

Mesh Sizes	Number of Plates Measured	Mean Reactivity Due to Boron (μk)[a]	Standard Deviation (μk)	Self-Shielding Factor (f_p)	
				Experimental	Theoretical
-100 +120	4	- 88.8[b]	0.64	0.74	0.68
-170 +200	5	- 94.1	0.66	0.79	0.79
-270 +325	4	-101.4	0.38	0.85	0.86
-325	4	-111.3	0.52	0.94	----

[a] $\mu k = 10^{-6} \Delta k/k$.

[b] A fifth plate was an outlier at -85.31 μk . Measurement precision based on eight measurements on a single plate is 0.11 μk .

The reactivity effect of the boron in the ARMF is expressed analytically as

$$\rho = f_p(\tau_p) f_b(n, \tau_p) n \Sigma_p V_p W_p V_c \quad (1)$$

where

$f_p(\tau)$ = particle self-shielding factor

$f_b(n, \tau_p)$ = bulk self-shielding factor resulting from particles shielding other particles

τ_p = a function of particle size and absorption cross section, defined in following sections

n = particle number density (cm^{-3})

Σ_p = particle macroscopic absorption cross section (cm^{-1})

V_p = particle mean volume (cm^3)

W_p = infinitely dilute statistical weight for the poison material (ρ/cm^2 , where ρ is reactivity)

V_c = volume of the plate core (cm^3).

Values for $f_p(\tau_p)$ can be obtained from this expression if all other values are known. Values for $f_b(\tau_p, n)$ are obtained by calculation as described in the section below. The value for $n \Sigma_p V_p W_p V_c$ is obtained by determining the reactivity effect of a plate containing the same amount of boron in the form of

infinitesimal particles where $f_p(0)$ is equal to unity. This reactivity was determined by fitting the three smallest reactivity values of Table I-1 to a second order polynomial function of τ_p . The largest reactivity value (that for -325 mesh) was not used because it was felt the mean particle size was not known as accurately. The mean particle size for the other three plates was assumed to be the median dimension of the screen openings. The results of this fit are

$$n \Sigma_p V_p W_p V_c = \frac{\rho}{f_p(\tau_p) f_b(\tau_{p,n})} = -123.77 \mu k \quad (2)$$

Using this reactivity value, the measured reactivities, and values of $f_b(n, \tau_p)$ in Equation (1), particle self-shielding values can be calculated. Values for the four particle sizes are listed in Table I-1 and plotted in Figure I-1 along with theoretical values.

Fig. I-1 Calculated and measured particle self-shielding factor, f_p , for plates containing boron carbide.

It is seen from the data that there is excellent agreement between the experimental and theoretical values for sizes smaller than 170 mesh. No reasons for the disagreement for sizes larger than this (as large as eight percent for -100 +120 mesh) can be advanced at this time. The self-shielding factor of 0.94 for -325 mesh particles is of particular interest since this is the size presently being used in ETR and ATR fuel elements. From Figure I-1 it is possible to obtain the effective diameter of the -325 mesh particles by reading the value from the curve which corresponds to an f_p of 0.94. This value is 18 microns, which is slightly smaller than the median dimensions of -325 +00 mesh.

1.2 Theoretical Calculations

Self-shielding factors are derived by Meghrebian and Holmes[3] for one-velocity neutrons in absorbing regions in a uniform isotropic neutron field.

TABLE I-2

THEORETICAL SELF-SHIELDING FACTORS

Mesh Size	Mean Diameter (cm)	$\tau_p = D \Sigma_p / 3$	f_p Equation (6)	$\tau_b = t \Sigma_b$	f_b Equation (10)
-100 +120	0.0137	0.378	0.678	0.0126	0.967
-170 +200	0.0081	0.223	0.788	0.0147	0.962
-270 +325	0.0049	0.134	0.864	0.0161	0.959

Defining,

$$f_p = \frac{\text{average flux throughout volume of a particle}}{\text{value of flux at position of particle before it was introduced}} .$$

A solution of the first-flight transport model for spherical particles gives

$$f_p = \frac{1 - \alpha}{\bar{s} \Sigma_p} \quad (3)$$

where

$$\alpha = \frac{8}{9\bar{s}^2 \Sigma_p^2} \left[1 - \left(\frac{3\bar{s} \Sigma_p}{2} + 1 \right) \exp \left(\frac{-3\bar{s} \Sigma_p}{2} \right) \right] \quad (4)$$

α = fraction of all neutrons incident upon the surface of a particle of material which pass through the particle without being absorbed

$\bar{s}$ = average path length of neutrons through the volume of the particle (with the particle nonabsorbing, cm)

Σ_p = macroscopic absorption cross section of the particle (cm^{-1}).

The average path length, $\bar{s}$, in a spherical particle is given by the formula

$$\bar{s} = \frac{4 V_p}{S_p} \quad (5)$$

where

V_p = volume of the particle (cm^3)

S_p = surface area of the particle (cm^2).

Combining Equations (3) and (4),

$$f_p = \frac{1}{\bar{s} \Sigma_p} \left\{ 1 - \frac{8}{9\bar{s}^2 \Sigma_p^2} \left[1 - \left(\frac{3\bar{s} \Sigma_p}{2} + 1 \right) \exp \left(\frac{-3\bar{s} \Sigma_p}{2} \right) \right] \right\} .$$

Letting $\tau_p = \frac{\bar{s} \Sigma_p}{2}$ to aid in plotting,

$$f_p = \frac{1}{\tau_p} \left[\frac{\tau_p^2}{3} - \frac{1}{9} + \left(\frac{1}{9} + \frac{\tau_p}{3} \right) \exp(-3\tau_p) \right] . \quad (6)$$

Further shielding occurs when the particles have become sufficiently numerous to shield one another from incident neutrons. This bulk self-shielding factor can be calculated as before except that the mass is now the entire core of the sample plate, assumed to be an infinite slab. Again defining,

$$f_b = \frac{\text{average flux throughout volume of core}}{\text{value of flux at position of core before it was introduced}}$$

A solution of the first-flight transport model gives

$$f_b = \frac{1 - \alpha}{\bar{s} \Sigma_b} \quad (7)$$

where

$$\alpha = \exp\left(-\frac{\bar{s} \Sigma_b}{2}\right) - \frac{\bar{s} \Sigma_b}{2} \exp\left(-\frac{\bar{s} \Sigma_b}{2}\right) + \frac{\bar{s}^2 \Sigma_b^2}{4} E_1\left(\frac{\bar{s} \Sigma_b}{2}\right) \quad (8)$$

Σ_b = macroscopic absorption cross section of the sample plate core (cm⁻¹)

= $f_p n \Sigma_p V_p$ (where n is the number density of particles, cm⁻³, and V_p is the average volume of a particle, cm³)

$E_1(x)$ = exponential integral as defined in Reference 4.

Combining Equations (7) and (8),

$$f_b = \frac{1}{\bar{s} \Sigma_b} \left\{ 1 - \exp\left(-\frac{\bar{s} \Sigma_b}{2}\right) + \frac{\bar{s} \Sigma_b}{2} \exp\left(-\frac{\bar{s} \Sigma_b}{2}\right) - \frac{\bar{s}^2 \Sigma_b^2}{4} E_1\left(\frac{\bar{s} \Sigma_b}{2}\right) \right\} \quad (9)$$

Letting $\tau_b = \frac{\bar{s} \Sigma_b}{2}$,

$$f_b = \frac{1}{2\tau_b} \left[1 - \exp(-\tau_b) + \tau_b \exp(-\tau_b) - \tau_b^2 E_1(\tau_b) \right] \quad (10)$$

Self-shielding factors can be calculated using the equation above and values of τ obtained from the following equations.

Spherical geometry (B₄C particles):

$$\tau_p = \frac{\bar{s} \Sigma_p}{2} = \frac{4 V_p}{S_p} \frac{\Sigma_p}{2}$$

$$\begin{aligned}
&= 4 \left(\frac{\frac{4}{3} \pi r^3}{4 \pi r^2} \right) \frac{\Sigma_p}{2} \\
&= \frac{2 \Sigma_p r}{3} = \frac{D \Sigma_p}{3} .
\end{aligned} \tag{11}$$

Slab geometry:

$$\begin{aligned}
\tau_b &= \frac{\bar{s} \Sigma_b}{2} = \frac{4 V_b \Sigma_b}{S_b} \frac{1}{2} \\
&= t \Sigma_b
\end{aligned}$$

where t is the core thickness (cm).

Resulting self-shielding factors (f_p and f_b) along with related information are listed in Table I-2. It will be noted that the value of f_b is quite constant. This is significant when it is remembered that f_b was used in the experimental determination of f_p . The constancy of f_b places negligible dependence of the experimental values of f_p on theoretical values of f_b . Since boron has a cross section which is proportional to $1/V$ of the neutron flux, no flux spectrum correction is needed to apply these experimental or theoretical results to various reactor conditions.

2. ENGINEERING EXPERIMENTS

(V. A. Walker)

2.1 ATR Fuel Element Hydraulic Testing (M. L. Griobcnow, J. A. Hendricks)

The first two dummy fuel assemblies, fabricated by the current fuel fabricator, failed below the normal hydraulic proof test of 120 percent nominal flow and 350°F. Inspection of two assemblies showed faulty swaging procedures to be the cause of failure. The arc lengths of the fuel plates were too short and the plates were bottomed in the side plate slot that was swaged first. This caused the entire tolerance to be on one side, reducing the extent of side plate engagement on this side below the value required for consistently acceptable joint strengths. The problem is shown in Figure I-2 which is a photograph of a section taken from fuel assembly DA8D. The first live assembly, A1D, was fabricated with the fuel plates centered between the side plates and both sides swaged simultaneously. A1D was inspected at the NRTS, and it appeared that the plates were indeed centered. However, the swage joint in Plate 19 appeared to run off the side plate, probably due to operator error. The assembly was subjected to a proof test (120 percent of nominal flow at 350°F) and failed after seven hours operation. The failure occurred when Plate 19 slipped out of the swage joint from the top down to the first vent and broke the comb (see Figure I-3). The motion of Plate 19 produced hydraulic forces that buckled Plates 16, 17, and 18 as shown in Figure I-4.

Fig. I-2 ATR fuel assembly DA8D showing inadequate side plate engagement.

Fig. I-3 Assembly A1D after hydraulic failure.

The four remaining vented dummy fuel assemblies, fabricated by this vendor, were proof tested. There were no catastrophic failures, but Plate 17, in DA10D, loosened resulting in unacceptable shifts in Channels 17 and 18. The remaining three fuel assemblies were judged to be acceptable, although the hydraulic forces tended to induce a permanent set in the fuel assemblies. The proof tests are run with a nominal 0.108-inch Channel 1 and 0.175-inch Channel 20. The difference between these external channels results in a pressure gradient tending to bow the assembly toward Channel 20. A typical, permanent set is shown in Figure I-5. Previous analysis^[5] indicates that the bowing does not constitute a significant increase in operating hazards.

It appears that consistently acceptable fuel assemblies can be fabricated only by maintaining 0.055- to 0.060-inch minimum side plate engagement for Plates 2 through 18 and 0.080- to 0.090-inch minimum engagement for Plates 1 and 19. It is also necessary to maintain constant surveillance of the location and depth of penetration of the swaging wheel.

The five special fuel assemblies fabricated at Oak Ridge for the ATR power approach program, were received, inspected, and hydraulically proof tested. HP-1 through -4 withstood the proof test, but HP-5 failed. Oak Ridge has been requested to fabricate a replacement assembly.

Fig. I-4 Internal distortion in AID assembly.

2.2 ATR Fuel Core Tolerance Relaxation

The possibility of a fuel width tolerance relaxation, beyond the ± 0.050 inch currently specified, that would still allow an assembly to operate in any reactor position has been investigated using the current ATRC data and PDQ nuclear calculations. Any increase in the positive tolerance would increase the operating hazard; however, it was ascertained that a relaxation to $\begin{matrix} + 0.050 \\ - 0.100 \end{matrix}$ inch on Plates 1-17 and -19 would not change the hot spot if the tolerance on Plate 18 was changed to $\begin{matrix} + 0.000 \\ - 0.100 \end{matrix}$ inch. Since such a change should reduce fuel plate rejection rates and lower procurement costs, the new tolerances have been incorporated into the specifications.

Because the axial temperature profile peaks 14 to 18 inches below the midplane with both ends operating cooler, the tolerances may be further relaxed in the end sections. It is not practical to determine top and bottom ends from radiographs; therefore, the degree of tolerance relaxation was determined from calculations at the bottom of the fuel assembly where conditions are more severe. These relaxations have been transmitted to the current fuel fabricator and will be incorporated into the next set of specifications and drawings.

POST FLOW TEST MEASUREMENTS MINUS ORIGINAL MEASUREMENTS
FOR FUEL ASSEMBLY DA11D

Fig. I-5 Fuel element bowing.

2.3 MTR Dogbone Analysis

Several MTR fuel assemblies have been received with what appears to be excessive dogboning. The extent of dogboning allowable in the various core positions was evaluated using the best available flux data. Assuming no change in the vertical fuel distribution, due to previous depletion, the fuel concentration allowable in the dogbone area is shown in Table I-3.

The effect of nonuniform depletion prior to moving an assembly into a hotter core position was evaluated assuming the assembly was depleted in a position with an effective $\sigma_a \phi$ in the dogbone, 1.3 times the average vertical value. This is lower than the values actually encountered in the external positions and conservatively allows for increased self-shielding in the thicker dogbone area. The depression factor due to the nonuniform depletion is shown in Figure I-6. The initial fuel in the dogbone that can be tolerated in any position is given by the value from Table I-3 divided by the value from Figure I-6.

Fig. I-6 Dogbone depression factor.

TABLE I-3
FUEL DOGBONE
ALLOWABLE IN MTR

Fuel Position	Nominal Fuel in Dogbone (percent)
11	205
12	140
13	113
14	115
15	108
16	102
17	98
18	118
19	156
21	187
23	115
25	108
27	92
29	156
31	222
32	162
33	124
34	108
35	108
36	102
37	97
38	111
39	170

2.4 Development of Rapid-Response Thermocouples (E. H. Porter)

Techniques were developed for the fabrication of fast-response thermocouples. Since the application is that of fuelplate midplane temperature sensing,

the procedures used for installing these thermocouples require that they be 0.0625-inch (nominal) OD, stainless steel sheathed, grounded (integral junction) Type K. Dynamic response measurements were made on thermocouples with as-received, commercially fabricated, hot junctions installed in sample plates. This was found to be from 73 to 109 msec for 50 percent of a step input. The hot junctions were then remade using copper and silver (pure and in alloy form), and the response was measured at from 8 to 48 msec. The test hot junction thickness varied from 0.0037 to 0.030 inch, producing the variation in response time. The hot junction end is not in contact with reactor primary coolant.

2.5 ETR Fuel Element Channel-Width Measurements (E. H. Porter)

The channels in four 500-gram (nominal) ETR fuel elements irradiated for all of Cycle 83 and part of Cycle 84 were measured to obtain data on possible fuel plate deformation. The elements measured were E-175-D, -157-D, -155-D, and -153-D. The percent power generation measured in ETRC in each element was as follows:

	<u>Core Power (percent)</u>
E-175-D	2.20
E-157-D	2.50
E-155-D	2.36
E-153-D	2.66

During Cycle 84 midcycle shutdown, the elements were discharged and the channels were probed again with a mechanical device after about ten days decay.

Each element was oriented in a vertical position near the canal wall, with the corner grid locating (polarizing) pin in the upper left-hand corner relative to the person taking the measurements. Traversing was from bottom to top, and readings were taken at three-inch intervals, starting at 1-7/8 inches and ending at 37-1/8 inches. Traverses x, y, and z were made at a position 1/4 inch from the left side plate, immediately to the right of the comb and 1/4 inch in from the right side plate, respectively.

Element E-175-D was remeasured in selected channels to verify earlier measurements. The results of the measurements are shown in Figure I-7, Channels 1 through 18.

Measurements on elements E-157-D, -155-D, and -153-D are given in Figures I-8, -9, and -10, respectively.

3. METALLURGICAL DEVELOPMENT

(G. W. Gibson)

3.1 Fuel Stability: UO₂ in Stainless Steel (M. Zukor, M. J. Graber)

3.11 Introduction. During ETR Cycles 80 through 82, a total of six 0.041-inch-thick UO₂-SS sample fuel plates were irradiated in the G-12 pressurized

Fig. I-7 Post irradiation channel thickness measurements on element E-175-D.

Fig. 1-7 Post irradiation channel thickness measurements on element E-175-D.

Fig. I-8 Post irradiation channel thickness measurements on element E-157-D.

Fig. I-8 Post irradiation channel thickness measurements on element E-157-D.

Fig. I-9 Post irradiation channel thickness measurements on element E-155-D.

Fig. I-9 Post irradiation channel thickness measurements on element E-155-D.

Fig. I-10 Post irradiation channel thickness measurements on element E-153-D.

Fig. I-10 Post irradiation channel thickness measurements on element E-153-D.

water loop of the Engineering Test Reactor. These plates were fabricated by Battelle Memorial Institute (BMI) with low-density UO_2 fuel for the Argonne Advanced Research Reactor (AARR) irradiation test program. The fuel plates were 1.25 x 5.75 inches containing a 0.031-inch-thick core clad with 0.005-inch 347 stainless steel. The fuel cores consisted of 37 wt% UO_2 dispersed in a 304 stainless steel matrix.

During ETR Cycle 80, three of these plates, A-29-927, -928, and -929, containing approximately 6.20 g U-235 and 0.026 g natural boron in the form of zirconium diboride (ZrB_2) were irradiated in Tier 1 of the G-12 sample assembly for 4443 megawatt days (MWd) of reactor operation. For Cycle 81 these plates were switched to Tier 2, the lower 4-1/2 inches of which were shrouded with hafnium to reduce the neutron flux. These three samples were irradiated in Cycles 80 and 81 for a total of 8911 MWd of reactor operation.

During Cycles 81 and 82, fuel plates A-29-936, -938, and -939 were irradiated in Tier 1 receiving 8778 MWd of reactor operation. Plates A-29-936 and -938 contained 0.041 and 0.0397 g boron, respectively, in the form of ZrB_2 ; whereas, sample A-29-939 contained 6 wt% boro-silicate glass beads containing 1.13 wt% boric oxide, B_2O_3 .

The fuel burnup, heat fluxes, and plate temperatures for the above fuel plates are shown in Table I-4. Fuel burnup was determined from gamma scans made on the irradiated plates and from isotopic burnup analysis. Plates A-29-927, -928, and -929 exhibited a uniform burnup profile; whereas, plates A-29-936, -938, and -939 showed a sharp burnup gradient down the sample. Based on the burnup analysis, plate heat fluxes and plate surface and center-line temperatures, averaged over the two cycles of irradiation, were calculated.

3.12 Uniform Swelling. Swelling in irradiated fuel plates is a result of fission product formation. The fission products have a volume greater than the volume of the fissioned uranium atom. Two fission products are formed per fission, and their combined volume is approximately four times that of the original uranium atom^[6]. For theoretically dense fuel, each 1×10^{20} fissions/cc of core produces a core swelling of 0.638 percent^[6] due to fission product accumulation. For less than theoretically dense fuel, the fission products will first fill in the fuel voids before significant plate swelling occurs. If boron is present as a burnable poison, additional swelling is produced due to the $\text{B-10}(n,\alpha)\text{Li-7}$ reaction. The lithium plus helium atoms occupy almost five times the volume of the B-10 atom.

In fuel plates exhibiting a sharp burnup profile, such as in plates A-29-936, -938, and -939, fuel voids in the lower burnup regions may not be completely filled, so little swelling occurs; whereas in the higher fuel burnup regions, the voids are filled and additional fission products cause greater swelling. This leads to nonuniform swelling along the plate. The measured or "uniform" core swelling is the average of the core swelling changes over the entire fuel plate. The core swelling is obtained from plate density measurements taken before and after irradiation, from which data the core volume can be determined. The as-fabricated UO_2 fuel density can also be determined from the density measurements. The as-fabricated UO_2 densities, with the exception of plate A-29-938, all agree within 1.2 percent of the average value of 9.01 g/cc. These plates contain 16.8 ± 1.3 percent fuel particle voids. Plate

TABLE I-4

UO₂-SS FUEL PLATE IRRADIATION DATA

Plate Number	U-235 (g)	MWd	Burnup (percent)	Fissions/cc of Core	Heat Flux (Btu/hr-ft ²)	Average T Surface (°C)	Average T Centerline (°C)
A-29-927	6.19	8911	23.2	12.0 x 10 ²⁰	0.93 x 10 ⁶	127	193
A-29-928	6.22	8911	21.3	11.0	0.86	124	185
A-29-929	6.19	8911	23.9	12.5	0.98	130	198
A-29-936	6.188	8778	----	----	----	---	---
Top	----	----	15.3	8.0 x 10 ²⁰	0.62 x 10 ⁶	115	160
Middle	----	----	19.2	10.1	0.78	122	177
Bottom	----	----	26.4	13.8	1.07	133	209
Average	----	----	20.1	10.5	0.82	123	180
A-29-938	6.15	8778	----	----	----	---	---
Top	----	----	14.9	7.8 x 10 ²⁰	0.60 x 10 ⁶	114	156
Middle	----	----	21.6	11.3	0.87	125	186
Bottom	----	----	27.8	14.6	1.12	135	214
Average	----	----	21.8	11.5	0.88	115	177
A-29-939	5.755	8778	----	----	----	---	---
Top	----	----	15.4	7.7 x 10 ²⁰	0.58 x 10 ⁶	114	155
Middle	----	----	20.0	10.1	0.77	121	175
Bottom	----	----	27.0	13.5	1.02	132	204
Average	----	----	21.1	10.5	0.80	122	178

A-29-938 was overheated during fabrication causing stainless steel to diffuse into the UO₂ fuel voids. This resulted in densification of the fuel, increasing the UO₂ density to 9.42 g/cc.

The UO₂ fuel density, fuel void fraction, and measured core swelling are shown in Table I-5. The measured core swelling is the amount of swelling caused by fission product accumulation plus that caused by the B-10 reaction less the amount accommodated by fuel and fabrication voids.

The effect of boron on the swelling behavior of the fuel plates can be calculated. For fuel burnup in excess of approximately 15 percent, essentially all of the B-10 is converted to He-4 + Li-7. In plates A-29-927, -928, and -929, containing 26 mg natural boron, 19.78 percent of which is B-10, approximately 11 percent of the measured swelling is due to the B-10 reaction. In sample A-29-936, which contained 41 mg natural boron, approximately 17 percent of the observed swelling was due to helium and lithium formation.

Plate A-29-938 contained 39.7 mg natural boron. However, this plate, which was overheated during fabrication resulting in fuel densification, exhibited a higher core swelling than the other fuel plates, 5.18 versus 3.2 percent. The effect of the B-10 induced swelling is fixed by the amount of boron present. Due to the higher swelling in plate A-29-938, the swelling due to the B-10 reaction amounted to only 10.6 percent of the measured core swelling.

TABLE I-5

UO₂-SS MEASURED CORE SWELLING

Plate Number	As-Fabricated UO ₂ Fuel Density (g/cc)	UO ₂ Void Fraction (percent)	Average Fissions of Core x 10 ²⁰ (cc)	Initial Core Volume (cc)	Measured ΔV (cc)	Measured Core Swelling $\Delta V/V \times 100$ (percent)
A-29-927	8.87	18.1	12.0	2.584	0.085	3.27
A-29-928	9.04	16.5	11.0	2.577	0.076	2.92
A-29-929	8.99	17.0	12.5	2.571	0.088	3.40
A-29-936	9.12	15.8	10.5	2.562	0.084	3.25
A-29-938	9.42	13.0	11.5	2.547	0.132	5.18
A-29-939	9.05	16.4	10.5	2.502	0.073	2.90

Plate A-29-939 contained only 1.3 mg boron in the form of glass beads. This small amount of boron did not contribute to the core swelling.

The contribution that the B-10 reaction makes to the overall measured core swelling can be subtracted out, leaving the swelling due to fission products. This is shown in Table I-6. Also shown, for comparison, is the expected swelling due to fission product accumulation for theoretically dense UO₂. This value is based on 6.38 percent core swelling for each 10²¹ fissions/cc of core. Fission product swelling averages approximately 40 percent of that for theoretically dense UO₂. This is brought about by the presence of fabrication voids and fuel porosity in the fuel particles, which accommodates fission products. Thus, the utilization of low density fuels can be beneficial to fuel stability.

TABLE I-6

EFFECT OF B-10 ON CORE SWELLING

Plate Number	Boron (mg)	Swelling Due to Li-7 + He-4 (cc)	Percent of Measured Swelling Due to Li + He (percent)	Swelling Due to U-235 Fission Products (percent)	Theoretically Expected Core Swelling from Dense UO ₂ (percent)	Percent Theoretical Swelling Due to Fission Products (percent)
A-29-927	26.0	0.0092	10.8	2.93	7.54	38.9
A-29-928	26.0	0.0092	12.1	2.59	6.97	37.2
A-29-929	26.0	0.0092	10.4	3.06	7.85	39.0
A-29-936	41.0	0.0145	17.3	2.71	6.59	41.1
A-29-938	39.7	0.0140	10.6	4.63	7.34	63.1
A-29-939	1.3	0.005	0.6	2.90	6.59	44.0

The observed core swelling versus average fissions/cc of core for plates A-29-927, -928, -929, -936, and -939 is plotted in Figure I-11. For the above UO₂-SS plates after $\approx 9 \times 10^{20}$ fissions/cc, all the fuel particle voids (on the average) are filled in with fission products. Additional fission products cannot be accommodated by fuel particle voids, and the additional swelling is proportional to swelling produced in theoretically dense fuel. The swelling curve should then exhibit a "dog-leg": a very low swelling rate up to $\approx 9 \times 10^{20}$ fissions/cc

Fig. I-11 UO₂-SS fuel core swelling.

with a break in the line, the swelling curve becoming parallel to that for theoretically dense fuel. This is shown in dotted lines (Figure I-11). This is not observed in a plot of the data because of (a) the small number of plates tested, (b) the narrow burnup range of the plates tested (10 to 12 $\times 10^{20}$ fissions/cc), and (c) the nonuniform swelling caused by the sharp burnup gradients in some of the plates.

3.13 Blister Testing. Blistering of irradiated fuel plates is due to the accumulation of fission product gases (Xe and Kr) in the fuel particle voids. As the gases accumulate, the internal pressure of the gas inside the voids increases. When the internal gas pressure exceeds the creep strength of the material, the void expands. This expansion relieves the pressure in the void. As more gas diffuses into the void, the internal pressure again exceeds that of the creep strength of the matrix and yielding again occurs. In-pile, the formation of a small blister reduces the thermal conductivity, increasing the fuel temperature, which enhances gas diffusion. As the void expands, interconnecting may occur, increasing the fission gas supply available for blistering. In fuel plates containing boron, helium gas is produced by the B-10(n, α)Li-7 reaction. Helium can diffuse through the matrix, collecting in the voids and increasing the internal pressure in the void. In an analysis of the blister gas from a 42 wt% UO₂ aluminum-clad fuel plate containing 10.4 mg B₄C (8.1 mg natural B), 41 percent of the gas found in the blister was helium^[7].

The irradiated UO₂-SS fuel plates were cut longitudinally in half, with half of each plate used for out-of-pile blister testing. The plates were placed in a furnace at $399 \pm 6^\circ\text{C}$ ($750 \pm 10^\circ\text{F}$) for 30 minutes, the temperature being increased in 28°C (50°F) increments until blistering or accelerated swelling occurred. From the burnup profile and position of the blister, the fuel burnup at the blister site could be ascertained.

Plates A-29-927, -928, and -929 all blistered at $454 \pm 6^\circ\text{C}$ ($850 \pm 10^\circ\text{F}$). Two blisters formed on plate A-29-929, both occurring at 454°C (850°F). Blistering occurs at or near points of maximum burnup, which for these plates is from 11 to 13×10^{20} fissions/cc. All of these plates contained 26 mg boron, which would produce 14.0 ml of helium at standard temperature and pressure(STP).

Plate A-29-936 containing 41 mg boron blistered twice at $427 \pm 6^\circ\text{C}$ ($800 \pm 10^\circ\text{F}$) at 13.2 and 13.8×10^{20} fissions/cc.

Plate A-29-938, which was overheated during fabrication, blistered at 399°C (750°F) at 14.6×10^{20} fissions/cc. However, blister testing was initiated at this temperature, and it is possible that this plate would have blistered at a lower temperature.

In plate A-29-939, which contained only a minor amount of boron (1.3 mg) in the form of glass beads, accelerated swelling was initiated at 538°C (1000°F), though blisters did not appear even at 677°C (1250°F).

Figure I-12 compares the blister anneal points for the UO_2 -SS plates with the data of D. L. Keller, Nucleonics, June 1961. Keller's data are for 25 to 30 wt% UO_2 in stainless steel, based on capsule irradiations and blistering occurring in-pile during irradiation. No boron was present in these samples. Plate center-line temperatures usually exceeded 427°C (800°F) during irradiation. From his data, Keller arrived at two curves. The first is the temperature below which,

Fig. I-12 Comparison of failures produced in UO_2 -SS plates by post-irradiation anneal tests with those produced during irradiation testing.

for a given burnup, no blistering occurs; the second is the temperature above which all the samples blistered. In between these two curves is a region approximately 143°C (290°F) in width in which a portion of the samples blistered. The A-15 and -16 composition UO₂-SS plates previously irradiated during ETR Cycles 70 and 71 and containing no boron all fall within this central blistering region[8]. The A-29 plates, which contain 26 to 41 mg boron, all blistered farther to the left, around the no-blister-blistering boundary curve. Plates A-29-927 and -928 blistered just outside of the blistering curve, in the no-blister region. Plate A-29-939, which contained only 1.3 mg natural boron in the form of glass beads, blistered farther to the right, near the center of the some-blister region, as would be expected of a plate without boron.

From Figure I-12 a comparison of the blistering temperatures of the AARR fuel plates with and without boron indicates that the presence of boron, which forms helium under irradiation, significantly reduces the blister temperature. In plates containing 26 mg natural boron, blister temperature lowering amounts to approximately 56°C (100°F). For 41 mg of boron, approximately 83°C (150°F) lowering of the blister temperature is observed. Thus, for each 10 mg of natural boron burned or 5.4 ml (at STP) of helium formed, the blister temperature is lowered by approximately 22°C (40°F) in the UO₂-SS sample fuel plates containing ZrB₂.

3.2 Radiograph Standards (K. C. Sumpter)

An investigation is currently underway to determine the applicability and feasibility of using radiographic standards as a means of measuring fuel homogeneity in ATR fuel plates. These standards were fabricated by powder metallurgy techniques similar to the technique employed for making resonance integral samples[8]. Basically, the process consists of blending Al powder with certain amounts of UAl₃ powder and then compacting. The green compact is put into a restraining Al picture frame, heated to 500°C, and rolled with a 20 percent reduction per pass to a total 8:1 reduction.

Three different fuel loadings were used based on maximum, average, and minimum ATR fuel concentrations. As shown in Table I-7, four particle-size

TABLE I-7
PARTICLE SIZES AND FUEL LOADINGS USED
FOR FABRICATING UAl₃-Al RADIOGRAPHY STANDARDS

Particle Size				
Range (μ)	Maximum (μ)	gU/in. ² /Standard Number		
106.68	149.86	0.375/10	0.431/11	0.491/12
25.40	149.86	0.375/7	0.431/8	0.491/9
15.23	88.90	0.375/4	0.431/5	0.491/6
5.08	43.18	0.375/1	0.431/2	0.491/3

ranges were used in addition to the three fuel concentrations mentioned previously. All other factors such as fabrication techniques, radiography conditions, and film density measurement techniques were held constant and contributed toward the final radiograph density variance.

Each sample was radiographed at least twice, and 15 density readings were taken per sample per radiograph. The results are depicted graphically in Figure I-13. Note that, although the density readings for a particular sample are different between radiographs, the difference between samples with different fuel loadings on the same radiograph remains fairly constant.

Further work is needed to adjust fuel loadings in the standards and to verify their use with actual ATR fuel plates.

Fig. 1-13 Film density differences of 12 radiograph standards containing UAl_3 -Al with 4 particle size ranges and 3 different fuel loadings.

4. MATERIALS RESEARCH

(J. M. Beeston)

4.1 Compatibility of Intermetallic Fuels with Additional Heat-Transfer Media (R. A. Moen, W. F. Zelezny)

Studies have previously been reported[5,9] on the selection of a heat transfer medium for use in subsequent swelling and gas release experiments with uranium intermetallic compounds. Because of the undesirable results of prior tests with the liquid-metal heat-transfer media (lead, bismuth, tin), concepts involving encapsulating the fuel pellets in a refractory metal such as tantalum or placing the unclad fuel pellet in a particulate medium such as Al_2O_3 or graphite were investigated.

Several quartz capsules, containing various combinations of materials, were evacuated and sealed. These combinations were heated according to the following schedule:

Ta - Al_2O_3	21 days at 800°C
Ta - UAl_3	21 days at 800°C
Ta - U_3Si	21 days at 800°C
Ta - UCu_5	21 days at 800°C
UAl_3 - Al_2O_3	21 days at 800°C
U_3Si - Al_2O_3	21 days at 800°C
UCu_5 - Al_2O_3	21 days at 800°C
UAl_3 - graphite	48 hours at 600°C
U_3Si - graphite	48 hours at 600°C
UCu_5 - graphite	48 hours at 600°C

Tantalum was not attacked by the U_3Si and UCu_5 under these conditions. UAl_3 appeared to be compatible with the Al_2O_3 as expected.

A thin layer of oxidation product formed on the surface of the tantalum in contact with the Al_2O_3 . The tantalum metal beneath the oxide layer contained platelets of a second-phase material; the structural details shown in Figure I-14 are typical of the structure found. The platelets and twin-like structure that appears beneath the platelets are undoubtedly associated with oxygen pickup by the metal. It has been found[10] that tantalum metal at elevated temperatures will absorb appreciable quantities of oxygen before a compound is formed.

Metallic tantalum exposed to powdered UAl_3 at 800°C for 21 days showed a duplex reaction zone about 10 to 12 microns thick. The structure is shown in Figure I-15 (under both brightfield and polarized light illumination). The columnar structure exhibited by the reaction zone was similar to NiAl_3 in

Fig. I-14 Polished cross section of Tantalum Metal heated with Al_2O_3 showing surface oxide film and platelets beneath the surface.

Fig. I-15 Surface reaction zone produced by heating Tantalum Metal in powdered UAl_3 for 21 days at 800°C .

appearance. A very thin layer of material (about one micron or less in thickness) at the extreme outer edge of the reaction zone contained only tantalum and uranium. This is illustrated by the tracings of the microprobe chart recordings reproduced in Figure I-16.

Fig. I-16 Chart recordings of microprobe scan across reaction zone on Tantalum Metal exposed to UAl_3 at 800°C for 21 days.

The sample of UCu_5 heated with the Al_2O_3 formed a peripheral reaction zone to a maximum depth of about 125 microns. The U_3Si exposed to Al_2O_3 developed superficial porosity to a depth of about 50 microns, but microprobe examination failed to reveal any change in composition of the porous region.

Of the three materials heated with the graphite, the UAl_3 and U_3Si structures did not appear altered. A slight alteration of the surface of the UCu_5 particles was evident, but it was believed to be associated with the grinding process used to prepare the particles rather than the heating of the material in graphite.

Based on the results of these tests, graphite appears to be the best material to date for use as a heat transfer medium for these candidate fuels, UAl_3 , U_3Si , and UCu_5 . Reactions in the Ta-UAl_3 and $\text{UCu}_5\text{-Al}_2\text{O}_3$ systems and development of superficial porosity in the $\text{U}_3\text{Si-Al}_2\text{O}_3$ system rule out the use of these material combinations in any in-reactor experiment at and above the test temperature of 800°C .

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II. NUCLEAR TECHNOLOGY

1. CROSS SECTIONS

1.1 U-232 and Th-228 Total Cross Sections (O. D. Simpson, M. S. Moore)

A paper describing the U-232 total neutron cross section measurements has been submitted to the Journal of Nuclear Science and Engineering. A preliminary publication of these data was given in the fourth quarter of the Nuclear Technology Branches Quarterly Report, IDO-17081 (1964). The final data are shown in Figures II-1 and -2. The multilevel resonance parameters are listed in Table II-1, and the solid line shown in Figures II-1 and -2 represents a two-channel multilevel fit to the data below 30 eV. The dashed line below 30 eV represents a one-channel multilevel fit to the data. It was concluded that a two-channel multilevel fit was necessary to describe the data adequately.

The Th-228 resonance at 1.9 eV was also reported in IDO-17081. Since this time, a sufficient quantity of U-232 has decayed to produce enough Th-228 so that another level at 7.5 eV could be identified as belonging to Th-228. These data are shown in Figure II-3, and the resonance parameters are as follows: $E_0 = 7.55 \pm 0.02$ eV, $\Gamma_n^0 = 0.44 \pm 0.15$ mV, and $\Gamma_\gamma = 70 \pm 30$ mV. These data, including the level at 1.9 eV, will appear as a companion paper to the U-232 described above.

1.2 Pu-238 Total Cross Sections (T. E. Young)

The Pu-238 total neutron cross section from 0.008 to 6500 eV is ready for publication. Figure II-4 shows the results of the total cross section measurements of Pu-238 from 2 to 5000 eV. The solid line below 200 eV is a Breit-Wigner single level fit to the data produced by the shape fit parameters from Table II-2. Table II-3 gives the parameters obtained by combining shape fit parameters from Table II-2 with the results of area analysis of resonances below 200 eV. The value of the s-wave neutron strength function has been determined using the parameters from Table II-3, and independently by using the average resonance cross section between 1000 and 5000 eV. This analysis gives $(\bar{\Gamma}_n^0/D) = (1.1 \pm 0.2) \times 10^{-4}$. Using parameters of individual resonances from Table II-3 below 200 eV and average parameters above 200 eV, one obtains 164 ± 15 barns as the resonance absorption integral of Pu-238 for neutrons of energies greater than 0.5 eV.

1.3 Scattering Cross Section Measurements (M. S. Moore, F. B. Simpson, L. G. Miller, INC; R. W. Hockenbury, RPI, Troy, New York)

Scattering cross sections have been measured for the three fissile nuclides, U-233, U-235, and Pu-239, for neutron energies between 1 and 100 eV. The RPI linear accelerator was used as the pulsed neutron source. The electron bursts from the accelerator were allowed to strike a water-cooled tantalum target, and neutrons, from the (γ, n) reaction in tantalum, were scattered from a polyethylene moderator at an angle of 135 degrees. This moderator-source geometry is indicated in Figure II-5; it has been found to have the advantage of reducing the effects of the gamma flash in a detector system at 90 degrees to the electron beam. The electron bursts were 0.25 μ sec wide, with energy of 70 to 80 MeV and peak currents of 2.5 to 3.0 amperes. The accelerator was pulsed at a

Fig. II-1 The total neutron cross section of U-232 from 0.01 to 1.0 eV. The solid line represents a multilevel two-channel fit to the data.

consisted of 24 one-inch-diameter $^{10}\text{BF}_3$ proportional tubes six inches long. Since electronic time delay circuitry was used between banks of $^{10}\text{BF}_3$ tubes to compensate for differences in flight path, the resolution at higher neutron energies was governed primarily by the jitter time of the counters, 0.6 μsec . The transmission of the sample was also measured by a zero-degree detector located the same distance from the sample (0.31 meters) as the last scattered-neutron detector bank on each side. A fourth detector, at 9.2 meters from the moderator, served both as a flux monitor and as an open-beam detector for sample transmission calculations from which the total cross section could be determined. Both the flux monitor and the transmission detector consisted of one-inch-diameter BF_3 tubes, depleted to 11 percent ^{10}B in order to keep the counting rate from these detectors comparable to that from the scattered-neutron detectors. Each detector was isolated from the surroundings by boron carbide shielding and from the fissile sample by $^{10}\text{B}_4\text{C}$ filters which could be cycled into and out of the beam. These filters, permitting elastically scattered neutrons to be distinguished from fission neutrons, were essentially "black" to neutrons below 20 eV and could be used up to 100 eV. In order to minimize background effects, the collimation system and drift tube were lined with B_4C and then surrounded externally with high-density concrete blocks. It was considered to be especially important to avoid all hydrogenous material in or near the beam, so that the moderator was the only hydrogenous source of slow neutrons which could reach the scattering sample. Overlap neutrons (neutrons detected during a given counting period but originating from an electron burst associated with a prior counting period) were eliminated by interposing a sheet of Cd, 0.040 inch thick, between the moderator and the detector system.

The signals from each of the four detectors were routed to a different quadrant of a 4096-channel time-of-flight analyzer. Background runs, taken with various combinations of $^{10}\text{B}_4\text{C}$ filters in the beam, were alternated with data runs at reasonably short intervals (10 minutes to 1 hour). While the filters were

repetition rate of 480 pulses/sec. The average power of the delivered beam was restricted to 25 kW in order to avoid damaging the target.

The detector system is also shown in Figure II-5. The sample, a metal plate of fissile material, was placed in the neutron beam at an angle of 45 degrees and at a flight path of 11.0 meters from the moderator. Scattered-neutron detectors were placed at 90 and 270 degrees to the incident beam (on either side of the sample). The neutron energy region of interest lay below 100 eV, since the corrections for prompt fission neutron detection became inadequate at higher energies. Therefore, no attempt was made in this experiment to obtain data at angles other than 90 degrees or to distinguish between s-wave and higher angular momentum resonances. Each of the two scattered-neutron detectors

Fig. II-2 The total neutron cross section of U-232 from 1 to 10,000 eV. The solid line below 30 eV is a two-channel multilevel fit to the data; the dashed line below 30 eV is a one-channel multilevel fit. The dashed line above 30 eV is an "eyeball" fit to the data. The data in the regions of 2 and 7.5 eV were omitted because of a Th-228 contaminant.

TABLE II-1

U-232 NEUTRON RESONANCE PARAMETERS
FOR A MULTILEVEL TWO-CHANNEL FIT[a]

E_o (eV)	Γ_n^o (mV)	Γ_{f1} (mV)[b]	Γ_{f2} (mV)[b]	Γ_y (mV)[c]
-0.600	0.108	-47	---	47
5.985	0.60	24	---	47
12.73	1.81	---	262	47
21.10	0.39	---	-500	47
23.87	1.00	80	---	47
27.75	0.45	---	-150	47

[a] Since the resonance parameters obtained from a multilevel analysis are not unique, it is impossible to assign errors to the resonance parameters. The quantity Γ_{f1} and Γ_{f2} are the fission widths associated with the two fission channels.

[b] Fission widths are always positive; the sign is used to define the type of interference. Resonances having fission widths of the same sign interfere destructively between resonances, while resonances with opposite signs interfere constructively.

[c] Assumed value.

changing, the analyzer memory was stored on magnetic tape, and summing of the runs was done while the accelerator was not operating.

Scattering data were obtained for U-233, U-235, and Pu-239. The known large resonances were observed for all three isotopes. Since the U-233 and Pu-239 samples were clad with Cu and Ni foils, respectively, scattering cross sections were also measured for Cu and Ni blanks and for a sample of natural W as a standard. It was hoped that the cross sections of the fissile isotopes might be determined relative to Cu and Ni, but this proved not to be possible for Pu-239. In this case, scattering from the Ni cladding proved to be a factor of three larger than off-resonance scattering from the Pu-239 sample. The Pu-239 was required to be the standard sample, and the off-resonance or potential scattering cross section was normalized to the value of 10.3 barns recommended by Uttley^[1]. The U-233 and U-235 scattering cross sections were then calculated using the counter efficiencies required by the Pu-239 normalization, and were found to agree reasonably well with previous measurements made below 20 eV^[2,3,4].

Various corrections were applied in obtaining scattering cross sections from the raw scattering data. Dead-time corrections, which depended upon the sum of the pulses from all four detectors, were kept below five percent. Backgrounds were treated as having three components: Static backgrounds were negligible except when the U-233 sample was in place; this sample had a gamma activity

Fig. II-3 The total neutron cross section of Th-228 times the square root of the neutron energy in the region of 7.5 eV. The solid line is a single-level Breit-Wigner fit to the data.

in excess of 15 R/hr on contact, which induced an appreciable amount of gamma pile-up into the scattered-neutron detectors. Quasi-static backgrounds (defined as those backgrounds which are related to the burst, but are smoothly varying) were monitored by $^{10}\text{B}_4\text{C}$ and Cd filters during each run. Backgrounds were also measured in separate runs with "notch" filters of Mn and Na to give the shape of the background at selected points above which the $^{10}\text{B}_4\text{C}$ filter was effective. The fission neutron background produced in the sample was measured by interposing $^{10}\text{B}_4\text{C}$ filters between the fissile sample and the scattered-neutron and transmission detectors. These filters were not adequate to stop scattered neutrons much above 100 eV, and were the primary limitation on the energy region covered. Normalization of background and data runs was done by comparison of the flux

Fig. II-4 The total neutron cross section of Pu-238 from 2 to 5,000 eV. The solid line below 200 eV is a single-level Breit-Wigner fit to the data.

monitor to correct for possible changes in the accelerator running conditions. A correction was also made for neutrons scattered into the transmission detector by assuming that the time-of-flight spectrum of such "background" neutrons was identical to that observed by the scattered neutron detector at 270 degrees. A

TABLE II-2

Pu-238 NEUTRON
RESONANCE PARAMETERS
(from shape fit)

E_o (eV)	Γ_n^o (mV)	Γ_γ (mV)
-10.0	5.0	45
- 0.4	0.545	45
2.885	0.045	40
9.975	0.067	45
18.562	0.78	45
59.8	0.17	38
70.1	0.27	38
83.0	2.0	38
110.0	0.7	38
113.8	1.2	38
118.6	3.0	38
122.5	2.3	38
140.0	0.15	38
151.0	1.10	38
164.0	0.20	38
171.2	5.0	38
182.5	2.4	38
192.0	3.5	38
202.0	0.4	38
209.0	0.4	38
220.0	4.5	38

correction was also found to be necessary to the flux monitor, to account for an effective change of efficiency due to the closer proximity of B₄C shielding material around it. This effect is related to the detector-scattered background^[5], consisting of neutrons scattered by the detector and re-scattered into the detector by surrounding material.

The lower energy data obtained for the scattering cross section of Pu-239 are shown in Figure II-6. The solid line drawn through the data points represents

TABLE II-3

SINGLE-LEVEL BREIT-WIGNER
RESONANCE PARAMETERS
OF Pu-238 FROM SHAPE
FITTING AND AREA ANALYSIS
OF MTR FAST CHOPPER DATA

E_o (eV)	Γ_γ (mV)[a]	Γ_n^o [mV/(eV) ^{1/2}]
-10.0	45	5.00
- 0.4	45	0.540
2.885	38 ± 2	0.044 ± 0.002
9.975	37 ± 6	0.066 ± 0.004
18.562	39 ± 6	0.81 ± 0.06
59.8	38[b]	0.20 ± 0.03
70.1	38[b]	0.30 ± 0.04
83.0	38[b]	2.10 ± 0.21
110.0	38[b]	0.50 ± 0.15
113.8	38[b]	0.97 ± 0.20
118.6	38[b]	2.93 ± 0.50
122.5	38[b]	2.41 ± 0.40
151.0	38[b]	1.10 ± 0.30
171.2	38[b]	5.00 ± 1.25
182.5	38[b]	2.40 ± 0.60
192.0	38[b]	3.50 ± 0.90

[a] The values shown for Γ_γ include other partial widths which might occur, specifically Γ_f .

[b] Assumed from consideration of Γ_γ 's for 2.885, 9.975, and 18.562 eV resonances.

Fig. II-5 Experimental arrangement. Electrons from the RPI linear accelerator produce fast neutrons which are scattered from a moderator at 135 degrees. Four detectors are used: two scattering detectors, D1 and D2, at 270 and 90 degrees to the beam; the transmission detector, D3; and the flux monitor, D4. Filters of $^{10}\text{B}_4\text{C}$ are used to distinguish between scattered neutrons and fission neutrons and to measure fast neutron background.

Fig. II-6 The scattering cross section of Pu-239 below 56 eV. The solid curve represents a calculation of the Pu-239 scattering cross section, with resolution broadening, based on presently accepted parameters [6] and spin assignments [7].

a calculated curve which utilizes the Pu-239 resonance parameters obtained by Bollinger [6] and the spin assignments made by Sauter and Bowman [7]. Resolution broadening of the calculated curve was done linearly, preserving the area in cross section. Below 40 eV, the agreement is reasonably good, except that the resonance peaks invariably lie below that which would be expected from resonance parameters. Above 40 eV, the calculated curve is consistently higher than the data. The discrepancies at lower energies can probably be attributed to sample thickness effects. The cross sections were calculated from the raw data by a technique based on that described by Moore and Simpson [4]. This technique involves the sample transmission and the total cross section, both of which were measured in this experiment. The resolution and the sample thickness were such that the measured total cross section was considerably lower than the accepted values on the resonance peaks, leading to a low value of the scattering cross section as well.

Several conclusions can be drawn from these data in their present preliminary form. First, the spin assignments made by Sauter and Bowman [7] for resonances below 30 eV are generally confirmed. The size of the interference dip associated with the resonances at 7.8, 10.9, 11.9, and 14.7 shows that these are associated with $J = 1$. The relative sizes show the resonances at 17.6, 22.2, and 26.2 are also $J = 1$ resonances. The resonance at 15.5 appears to be more nearly consistent with $J = 1$ than with $J = 0$, although neither assignment describes the data well. At energies above 40 eV, spin assignments cannot be made until the basic discrepancy between data and parameters is resolved. Analysis of these data and of the data obtained on U-233 and U-235 is continuing.

2. NUCLEAR CHEMISTRY

2.1 Cross Section of Pm-147 (J. W. Coddington, R. L. Tromp, D. K. Oestreich)

The composition of the purified promethium used for the cross section samples has been determined and is given in Table II-4. The Sm content was calculated by area analysis of the Sm-147 resonance at 3.4 eV, using the resonance parameters obtained by Pattenden^[8]. The amounts of Ag, Al, Cr, and Zn were determined by activation analysis. The amounts of Ca and Mg were found by emission spectrographic analysis of the raffinate from a Pm extraction.

A very thin D₂O solution sample of Pm-147 has been made and is being run in order to obtain better information on the 5.37 eV resonance. Preliminary data indicate that σ_0 is $\approx 50,000$ barns at this resonance. Area analyses have been made on a number of Pm-147 resonances, and the resulting resonance parameters are given in Table II-5.

TABLE II-4

COMPOSITION OF
Pm-147 CHOPPER SAMPLE
(In weight percent of
metal constituents at
time of separation)

Pm-147	99.07
Sm	0.55
Ag	0.052
Al	0.05
Ca	0.10
Cr	0.02
Mg	0.10
Zn	0.06

TABLE II-5

PARAMETERS FOR CERTAIN
Pm-147 RESONANCES

E_0 (eV)	Γ (mV)	Γ_n (mV)	Γ_n^0 (nV)	σ_0 (barns)
6.58	97	1.67	0.651	3,425
6.91	61	4.85	1.85	15,061
15.11	69	1.599	0.410	2,002
19.50	74	4.744	1.086	4,349

2.2 Cross Section Sample Production (J. R. Berreth, J. W. Coddington, D. K. Oestreich, R. P. Schuman, S. D. Reeder)

A total of four Am-243 samples of varying thicknesses have been prepared and are ready for chopper measurements. A thin D₂O solution sample of Tc-99 has been prepared by dissolving Tc metal in DNO₃. Mass analysis of a sample of Cm that had been purified by an α hydroxy isobutyrate cation column showed that the curium used for the Cm-244 cross section had an atomic composition of 0.03 percent Cm-243, 95.26 percent Cm-244, 1.66 percent Cm-245, 3.12 percent Cm-246, 0.05 percent Cm-247, and 0.02 percent Cm-248. Some enriched U-236 that had been depleted in U-235 by long irradiation in the MTR has been chemically purified and converted to U₃O₈. The atomic composition of the uranium is as follows: 0.04 percent U-234, 0.003 percent U-235, 97.5 percent U-236, and 2.4 percent U-238. A lithium-drifted gamma spectrum of the uranium shows 49.4 and 112.8 keV gamma rays due to U-236 and 53.2 and 121.0 keV gamma rays due to U-234 and Th X-rays.

2.3 Decay of Cm-246 and Cm-248 (R. P. Schuman)

Alpha spectra have been taken on curium milked from californium. The half-lives of the Cm-246 and Cm-248 were determined from the yields of the curium isotopes and the amounts of the californium parents. The alpha energies of the two major alpha groups of each curium isotope were determined relative to Cm-244, 5.802 and 5.760 MeV; Am-241, 5.482, 5.439, and 5.386 MeV; Pu-240, 5.160 and 5.116 MeV; and Pu-242, 4.898 and 4.854 MeV[9]. The californium half-lives used in calculating the curium half-lives were Cf-252, 2.48 yr (α), and Cf-250, 11 yr. The results were Cm-248, half-life for α -decay, 4.3×10^5 yr, alpha energies, 5.071 (81 percent), 5.029 (19 percent) and Cm-246, half-life 5.8×10^3 yr, alpha energies, 5.379 (78 percent) and 5.336 (22 percent). An alpha spectrum is shown in Figure II-7.

Fig. II-7 Alpha spectrum for curium daughters milked from californium.

2.4 Analytical Chemistry (S. D. Reeder, R. J. Gehrke)

Preliminary results have been computed for the amounts of certain radio-nuclides which had been absorbed on a small, mixed bed, ion exchange resin column through which the MTR primary coolant water had been passed. The counting was accomplished with a lithium-drifted germanium detector on several samples representing different volumes of process water. The data reported below are for the first sample which represents a decay of 1.78 hours after sampling. The results are compared below to those obtained by routine radio-chemical methods on the same water sample:

Nuclide	Germanium Detector	Radiochemistry
I-131	1.82×10^3 d/min/ml	1.85×10^3 d/min/ml
I-133	7.12×10^3	7.6×10^3
Ba-139	9.41×10^3	9.55×10^3
Ba-140	3.12×10^3	2.56×10^3
Sr-92	5.62×10^3	5.67×10^3
Na-24	4.48×10^4	4.45×10^4
Cd-115	2.18×10^4	2.07×10^4

Although the results are in excellent agreement, it is believed that analysis of other data chosen for optimal decay times to enhance the spectrum of a particular nuclide will result in higher accuracy. The prospects for determining radioactive nuclides in the process water without radiochemistry look very promising.

2.5 Alpha Spectroscopy (R. N. Chanda, R. A. Deal)

In conjunction with the compilation of an alpha spectrum catalog^[10,11], special attention has been given to arriving at optimum counting conditions and obtaining the best possible α -resolutions for commercially available surface barrier detectors. Chambers and preamplifiers in use at the time in this laboratory were found to be inadequate. Consequently, a new chamber was built which allowed for sample insertion with the detector in constant vacuum and also for an internal FET preamplifier. Using this preamplifier^[12] and chamber, alpha resolutions of < 14 keV FWHM were obtainable for room-temperature operation. However, the system was extremely count-rate sensitive; ie, resolutions deteriorated as the count rate increased beyond 10 c/sec. The original preamplifier, which was AC coupled, has been replaced with a DC-coupled preamplifier and the count-rate dependence sharply reduced. Alpha resolutions of ≈ 13.8 keV are again obtainable at room temperature. However, when cooled to 0°C , the resolution drops to 12.8 keV (see Figure II-8). Catalog spectra are now being taken under the latter conditions.

2.6 Sample Preparation Technique (R. A. Deal, R. N. Chanda)

Several methods for preparing high resolution alpha samples are currently in use, viz, vacuum sublimation, electrospraying, and electroplating. While these methods are quite adequate, in most cases they are relatively complex in procedure and time consuming. We have found that a 0.25 M solution of TTA (thenyltrifluoro-acetone) in benzene leaves no discernible residue when evaporated and flamed on a polished platinum plate. Therefore, we are now preparing alpha samples by first extracting the nuclide in question into the TTA-benzene solution and then evaporating $\approx 50\lambda$ of this solution into a platinum-counting plate. Alpha resolutions obtainable by this method are at least equal to, if not better than, other methods we have employed. In addition, there are several distinct advantages to using this method: (a) the procedure is rapid and requires no apparatus other than a small transfer pipette, (b) the yield is always 100 percent, and (c) TTA is also a very selective extractant for the actinides.

Fig. II-8 Alpha particle spectrum of U-234 from source prepared by electrodeposition on a platinum plate. Energies are given in MeV with relative intensities in parentheses.

3. PHYSICS

3.1 Decay of In-115m (E. L. Murri, L. D. McIsaac, R. G. Helmer)

Previous studies^[13] of the decay of In-115m (4.5 hours) have indicated only the presence of an isomeric gamma-ray transition of 336 keV. In the present study, a 497-keV transition has also been observed. This indicates the existence of a beta-decay branch to an excited state in Sn-115.

For this study, a sample of Cd-115 was produced by neutron irradiation of normal cadmium. After allowing the sample to decay for one day, the indium was removed by several co-precipitations on $\text{Fe}(\text{OH})_3$ with ammonia. After two hours, the In-115m was milked from the Cd by another $\text{Fe}(\text{OH})_3$ precipitation. This In was purified by repeated $\text{Fe}(\text{OH})_3$ co-precipitations by solvent extraction of InBr_3 with diethyl ether and by selective elution from anion resin using HCl ^[14].

Two gamma-ray spectra for In-115m are shown in Figures II-9 and -10. These clearly indicate the presence of a gamma ray at 497 keV. From successive spectra, it was shown that both gamma rays have the same half-life. In order to determine the gamma-ray energies, a spectrum of In-115m, Hg-203, and Au-198 was measured using a Ge(Li) detector system. The resulting energies were (336.25 ± 0.10) keV and (497.3 ± 0.5) keV with relative photon intensities of 1 and 1.3×10^{-3} . In the calculation of relative transition intensities shown in the decay scheme, the total internal-conversion coefficient of the 336-keV transition was taken to be 1.0.

The proposed decay schemes of In-115m and In-115 and a partial decay scheme^[15] of Sb-115 are shown in Figure II-11. The data other than those in the previous paragraph are from References 13 and 15.

The spins of the ground states Sn-115 and In-115 and that of In-115m have been measured^[16]. These spins and parities are in agreement with the expectations of the shell model. The spin of Sb-115 is presumed^[15] to be the same as Sb-121 (ie, $5/2$). The Sb-115 data^[15] indicate that the 497-keV level in Sn-115 should have $I^\pi = 3/2^+$ or $5/2^+$. From the observation of the decay from

Fig. II-9 Gamma-ray spectrum of In-115m measured on a 3- x 3-in. NaI(Tl) detector with the source 10 cm from the detector.

Fig. II-10 Gamma-ray spectrum of In-115m obtained with a lithium-drifted germanium detector ($2.5 \text{ cm}^2 \times 0.8 \text{ cm}$) and a 4096-channel analyzer. The full-width-at-half-maximum is about 1.1 keV at 333 keV.

Fig. II-11 Decay schemes of In-115m and In-115 and a portion of the decay scheme of Sb-115. The decay scheme in Reference 13 has been modified to include the beta-decay branch to the excited state in Sn-115 and the reported gamma-ray transition energies. The beta-ray intensities are in percent of decays for each decaying level, and the energies are in keV.

In-115m with a log ft value of 7.2, it is then concluded that this state must be $3/2^+$.

The calculations of Kisslinger and Sorensen[17] indicate that the wave functions of the ground states of Sb-115 and Sn-115 and the first $3/2^+$ excited state in Sn-115 are expected to be predominantly those of the corresponding shell model states, ie, $d_{5/2}$, $s_{1/2}$, and $d_{3/2}$, respectively. The low log ft value for the positron transition[15] would support this conclusion, since the beta decay to any vibrational components in the $3/2^+$ state would probably be much slower.

3.2 A Study of the Level Structure of Sm-148 Excited in the Decay of Eu-148 (J. E. Cline)

Investigation has been made on the decay of 54-day Eu-148, which decays by electron capture to Sm-148 with a decay energy in excess of 3 MeV. Studies have been made using NaI(Tl) and Ge(Li) gamma-ray spectrometers and Si(Li), Si(Au), and 180-degree permanent-magnet electron spectrometers. Coincidence measurements have been carried out using the gamma-ray spectrometers and a 256- x 256-channel multi-parameter analyzer. Nearly 100 of the 121 transitions identified in this work have been placed in a consistent decay scheme on the basis of the coincidence measurements and "energy-fitting" techniques. Multipolarity assignments have been made to many of the gamma rays on the basis of the conversion-coefficient measurements. Spins and parities have been made to many of the levels on the basis of these and previous measurements. A brief discussion of the possible interpretations is given.

Sources of Eu-148 were produced by a (p,2n) reaction on samples of Sm_2O_3 , enriched to 99+ percent in Sm-149. Irradiations were performed using the 22-MeV proton beam of the ORNL production cyclotron. Samples were purified by rare-earth chemistry followed by an ion-exchange column separation with alpha-hydroxyisobutyric acid as the eluting agent. A slight contaminant of 24-day Eu-147 was observed in early runs on the samples. After one year, the presence of 120-day Eu-149 became evident. No other Eu activity or any other rare-earth activity became apparent even after two years of use of the sample. Samples for gamma-ray counting were made by evaporating the material onto "scotch-tape" backing. Sources for conversion-electron studies were made by electroplating from solution onto 0.15-mm-diameter platinum wire.

In Figure II-12 is shown a gamma-ray spectrum of Eu-148 taken with a 2.5-cm² x 8-mm Ge(Li) spectrometer. The Ge(Li) spectrometer incorporated a preamplifier which was internal to the cryostat. This system, previously reported[12,18], yielded resolutions (FWHM) of 0.75 keV at 100 keV and 1.7

Fig. II-12 Gamma-ray spectrum of Eu-148 taken with a 2.5-cm² x 8-mm Ge(Li) spectrometer. The counts in the first 2100 channels have been increased by a factor of 10 so that the last 2000 channels could be shown on the same plot.

keV at 1 MeV. With these resolution characteristics, 121 transitions were identified as being associated with the decay of Eu-148. Energy calibration of the 4096-channel spectrometer was accomplished through internal calibration using sources of Gd-153, Zn-65, Cs-137, Y-88, and the Sm X-rays. The values used for the energies of the transitions in these sources are, respectively, 97.43 and 103.18 keV for Gd-153, 1115.5 for Zn-65, 661.60 for Cs-137, 898.01 and 1836.08 for Y-88, and 40.12 for K_{α} in Sm. The photopeak detection efficiency, as a function of gamma-ray energy, has been determined for the Ge(Li) spectrometer by using sources calibrated with the NaI(Tl) spectrometer in use at this laboratory for several years. From this efficiency curve and an integration under each of the peaks shown in Figure II-12, relative intensities were obtained for all of the transitions observed. Absolute intensities were obtained by assuming that all decays proceeded through the level at 550 keV, an assumption which appeared justified by the coincidence measurements.

Conversion-electron spectra were taken using a 1-mm-thick Si(Au), a 2-mm-thick Si(Li), and 180-degree permanent-magnet electron spectrometers. A spectrum of the conversion electrons taken with the Si(Au) spectrometer is shown in Figure II-13. The resolution (FWHM) of this system was 1.5 keV at 100 keV and 2.3 keV at 1 MeV. An efficiency curve for this spectrometer was measured by using calibrated sources of Cd-109, Ce-139, Mn-54, In-113m, and Bi-207. Relative intensities for the conversion electrons were obtained using this curve. Absolute intensities were obtained by assuming that the 550 keV and the 630 keV are pure E2 transitions, an assumption justified by previous authors [19,20]. Internal-conversion electrons were also observed photographically with 180-degree magnetic spectrographs. These data were used to identify the low-energy transitions that were present as well as to determine energies and some electron intensities. The intensities were determined using a method reported by Slátiš [21], where it is assumed that

$$\log I_0/I = C \log (E + 1)$$

where E is the exposure (in electrons/cm²), I_0 is the light intensity at any unexposed portion of the photographic plate, I is the intensity at any particular exposed portion, and C is a constant to be determined from different exposure times.

The data from the gamma-ray and conversion-electron measurements are summarized in Tables II-6 and -7. Table II-6 lists the gamma-ray energies and intensities and the uncertainties which have been assigned to these values. Although these uncertainties are rather subjective in nature, it is felt that they are reasonably conservative. The 22.41-, 67.63-, and 215.9-keV transitions were observed only with the magnetic spectrographs; therefore, no intensities can be given for these gamma rays. The doublet nature of the peak at 414 keV was established through coincidence measurements and through a comparison of the relative intensities of the 414- and 432-keV transitions as observed in the decay of Eu-148 and Pm-148m. These two transitions have been shown, in the decay of Pm-148m, to depopulate the same level in Sm-148, a level which is also excited in the decay of Eu-148.

Table II-7 lists the transition energies, electron energies, electron assignments, electron intensities, measured conversion coefficients, and the most probable multipolarity assignments for the transitions. Included also are the

Fig. II-13 Conversion-electron spectrum of Eu-148 taken with a 1-cm² x 1-mm Si(Au) spectrometer. The energies and the assignments of the peaks are included in this plot.

TABLE II-6

LIST OF ENERGIES AND ABSOLUTE INTENSITIES FOR
THE GAMMA RAYS OBSERVED IN THE DECAY OF Eu-148

(The estimated uncertainties in the values have been included. Although these estimates are subjective, it is felt that they represent a realistic appraisal of the uncertainties involved in the techniques which were used.)

E_Y (keV)	I_Y (percent of β^- decays)	E_Y	I_Y	E_Y	I_Y
22.41 $\pm$ 0.10	---	654.3 $\pm$ 0.1	1.31 $\pm$ 0.15	1146.9 $\pm$ 0.1	1.9 $\pm$ 0.2
67.63 $\pm$ 0.10	---	657.0 $\pm$ 0.5	0.17 $\pm$ 0.06	1151.7 $\pm$ 0.5	0.08 $\pm$ 0.03
98.4 $\pm$ 0.10	0.09 $\pm$ 0.03	667.7 $\pm$ 0.5	0.13 $\pm$ 0.05	1156.4 $\pm$ 0.5	0.04 $\pm$ 0.02
116.3 $\pm$ 0.2	0.10 $\pm$ 0.03	669.9 $\pm$ 0.2	0.34 $\pm$ 0.08	1165.3 $\pm$ 0.5	0.08 $\pm$ 0.03
121.18 $\pm$ 0.12	0.006 $\pm$ 0.003	675.4 $\pm$ 0.5	0.08 $\pm$ 0.03	1180.5 $\pm$ 0.5	0.3 $\pm$ 0.08
166.2 $\pm$ 0.3	0.05 $\pm$ 0.02	683.2 $\pm$ 0.1	1.06 $\pm$ 0.10	1183.3 $\pm$ 0.1	1.7 $\pm$ 0.2
182.8 $\pm$ 0.2	0.11 $\pm$ 0.3	690.7 $\pm$ 0.5	0.08 $\pm$ 0.03	1194.1 $\pm$ 0.5	0.13 $\pm$ 0.04
189.6 $\pm$ 0.2	0.10 $\pm$ 0.3	714.8 $\pm$ 0.1	1.54 $\pm$ 0.12	1207.4 $\pm$ 0.2	0.59 $\pm$ 0.12
215.9 $\pm$ 0.2	---	719.6 $\pm$ 0.3	0.11 $\pm$ 0.03	1219.7 $\pm$ 0.5	0.08 $\pm$ 0.03
241.5 $\pm$ 0.1	1.04 $\pm$ 0.10	725.7 $\pm$ 0.1	12.2 $\pm$ 1.0	1222.0 $\pm$ 0.5	0.09 $\pm$ 0.03
243.6 $\pm$ 0.2	0.23 $\pm$ 0.05	734.8 $\pm$ 1.0	0.06 $\pm$ 0.03	1230.0 $\pm$ 0.5	0.03 $\pm$ 0.02
252.3 $\pm$ 0.3	0.05 $\pm$ 0.02	756.6 $\pm$ 0.5	0.19 $\pm$ 0.05	1236.6 $\pm$ 0.2	0.39 $\pm$ 0.09
279.1 $\pm$ 0.5	0.06 $\pm$ 0.02	770.4 $\pm$ 0.2	0.38 $\pm$ 0.07	1267.1 $\pm$ 0.5	0.09 $\pm$ 0.03
287.9 $\pm$ 0.2	0.22 $\pm$ 0.05	780.2 $\pm$ 0.5	0.08 $\pm$ 0.03	1276.5 $\pm$ 0.5	0.07 $\pm$ 0.03
296.0 $\pm$ 0.3	---	799.2 $\pm$ 0.2	0.34 $\pm$ 0.07	1309.8 $\pm$ 0.2	0.46 $\pm$ 0.09
310.0 $\pm$ 0.3	0.22 $\pm$ 0.05	870.0 $\pm$ 0.1	4.9 $\pm$ 0.5	1328.5 $\pm$ 0.2	1.22 $\pm$ 0.15
311.4 $\pm$ 0.1	1.14 $\pm$ 0.10	895.8 $\pm$ 0.1	0.55 $\pm$ 0.10	1344.6 $\pm$ 0.1	3.3 $\pm$ 0.3
319.1 $\pm$ 0.3	0.05 $\pm$ 0.02	903.9 $\pm$ 0.2	0.34 $\pm$ 0.07	1353.6 $\pm$ 0.2	0.47 $\pm$ 0.09
377.5 $\pm$ 0.5	---	906.9 $\pm$ 0.3	0.15 $\pm$ 0.04	1362.6 $\pm$ 0.2	0.55 $\pm$ 0.10
413.9 $\pm$ 0.1	18.6 $\pm$ 0.5	915.3 $\pm$ 0.1	2.4 $\pm$ 0.2	1409.0 $\pm$ 0.6	0.10 $\pm$ 0.03
413.9 $\pm$ 0.1	18.6 $\pm$ 0.5	924.9 $\pm$ 0.2	0.27 $\pm$ 0.06	1454.3 $\pm$ 0.3	0.25 $\pm$ 0.07
432.6 $\pm$ 0.1	2.79 $\pm$ 0.20	930.5 $\pm$ 0.1	2.4 $\pm$ 0.2	1460.5 $\pm$ 0.2	1.10 $\pm$ 0.12
437.2 $\pm$ 0.3		938.2 $\pm$ 1.0	0.11 $\pm$ 0.04	1493.2 $\pm$ 0.6	0.07 $\pm$ 0.03
468.4 $\pm$ 0.3	0.49 $\pm$ 0.08	949.7 $\pm$ 0.2	0.21 $\pm$ 0.06	1503.0 $\pm$ 0.3	0.13 $\pm$ 0.04
481.2 $\pm$ 0.5	0.16 $\pm$ 0.05	964.2 $\pm$ 0.4	0.2 $\pm$ 0.07	1512.0 $\pm$ 0.5	0.04 $\pm$ 0.02
495.2 $\pm$ 0.3	0.25 $\pm$ 0.06	967.3 $\pm$ 0.1	2.9 $\pm$ 0.3	1521.9 $\pm$ 0.3	0.11 $\pm$ 0.03
501.3 $\pm$ 0.1	0.75 $\pm$ 0.13	980.0 $\pm$ 0.5	0.13 $\pm$ 0.04	1536.1 $\pm$ 0.3	0.08 $\pm$ 0.03
505.2 $\pm$ 0.5	0.28 $\pm$ 0.07	989.7 $\pm$ 0.2	0.39 $\pm$ 0.08	1543.1 $\pm$ 0.2	0.71 $\pm$ 0.13
516.7 $\pm$ 0.3	0.40 $\pm$ 0.10	1013.9 $\pm$ 0.2	0.41 $\pm$ 0.08	1547.3 $\pm$ 0.4	0.08 $\pm$ 0.03
528.5 $\pm$ 0.3	0.19 $\pm$ 0.05	1034.1 $\pm$ 0.1	7.9 $\pm$ 0.7	1551.7 $\pm$ 0.5	0.02 $\pm$ 0.01
550.2 $\pm$ 0.5	99.0 $\pm$ 0	1043.9 $\pm$ 1.0	0.04 $\pm$ 0.02	1560.7 $\pm$ 0.2	0.87 $\pm$ 0.10
553.2 $\pm$ 0.1	17.1 $\pm$ 0.5	1047.5 $\pm$ 0.3	0.20 $\pm$ 0.07	1565.0 $\pm$ 0.6	0.03 $\pm$ 0.02
571.9 $\pm$ 0.05	9.1 $\pm$ 0.3	1066.8 $\pm$ 0.2	0.29 $\pm$ 0.08	1621.5 $\pm$ 0.1	4.6 $\pm$ 0.5
587.6 $\pm$ 0.5	0.08 $\pm$ 0.03	1069.2 $\pm$ 0.4	0.22 $\pm$ 0.08	1635.3 $\pm$ 0.3	0.18 $\pm$ 0.4
590.1 $\pm$ 0.2	0.32 $\pm$ 0.08	1082.2 $\pm$ 0.5	0.17 $\pm$ 0.07	1650.4 $\pm$ 0.1	3.7 $\pm$ 0.5
595.2 $\pm$ 0.3	0.20 $\pm$ 0.06	1089.3 $\pm$ 0.5	0.20 $\pm$ 0.07	1664.5 $\pm$ 0.5	0.08 $\pm$ 0.03
599.5 $\pm$ 0.2	0.51 $\pm$ 0.10	1097.5 $\pm$ 0.7	0.10 $\pm$ 0.04	1677.8 $\pm$ 0.1	0.42 $\pm$ 0.08
602.6 $\pm$ 0.3	0.23 $\pm$ 0.06	1104.3 $\pm$ 0.2	0.45 $\pm$ 0.09	1748.1 $\pm$ 0.5	0.02 $\pm$ 0.01
611.26 $\pm$ 0.05	19.3 $\pm$ 0.5	1107.9 $\pm$ 0.3	0.10 $\pm$ 0.03	1776.9 $\pm$ 0.3	0.04 $\pm$ 0.02
620.0 $\pm$ 0.1	1.0 $\pm$ 0.1	1113.8 $\pm$ 0.3	0.13 $\pm$ 0.04	1940.0 $\pm$ 0.5	0.05 $\pm$ 0.03
629.9 $\pm$ 0.05	70.9 $\pm$ 3.0	1126.9 $\pm$ 0.5	0.10 $\pm$ 0.03	1974.2 $\pm$ 0.5	0.03 $\pm$ 0.02
---	---	---	---	2173.2 $\pm$ 0.3	0.22 $\pm$ 0.06

TABLE II-7

LIST OF CONVERSION-ELECTRON ENERGIES AND INTENSITIES
(Included in this table are the assignments of the electrons,
measured conversion coefficients, and multipolarity assignments.
Along with the multipolarity assignments are given the theoretical
conversion coefficients corresponding to these assignments.)

E_γ	E_e	Assign	$I_e(\text{Si})$	$I_e(\text{mag})$	α	Multipolarity Assignment	Reference 19
98.4	51.5	K	---	1.3×10^{-1}	1.4×10^{-1}	E2(1.15) M1(1.55)	---
98.4	91.1	L	---	2.3×10^{-2}	2.6×10^{-1}	E2(1.15) M1(1.55)	---
116.3	114.8	M	1.9×10^{-2}	2.0×10^{-2}	1.8×10^{-1}	E2($\approx 3.0 \times 10^{-1}$)	---
166.2	119.3	K	8.0×10^{-3}	3.0×10^{-2}	3.0×10^{-1}	E2(2.4×10^{-1})	---
189.6	142.7	K	2.5×10^{-2}	4.5×10^{-2}	2.5×10^{-1}	M1(2.4×10^{-1})	M1
241.5	194.6	K	1.2×10^{-1}	2.1×10^{-1}	1.6×10^{-1}	M1(1.3×10^{-1})	M1
241.5	234.2	L	1.2×10^{-2}	3.0×10^{-2}	2.0×10^{-2}	M1(1.7×10^{-2})	M1
287.9	241.0	K	1.9×10^{-2}	4.0×10^{-2}	1.0×10^{-1}	M1(7.6×10^{-2})	---
311.4	264.5	K	1.9×10^{-2}	2.0×10^{-2}	1.2×10^{-2}	E1(1.1×10^{-2})	E1
311.4	304.1	L	8.6×10^{-3}	2.0×10^{-2}	7.6×10^{-3}	E1(1.5×10^{-3})	E1
413.9	367.0	K	2.4×10^{-1}	2.8×10^{-1}	Two gamma rays of nearly equal intensity, one E1, one E2		E1, E2
413.9	406.6	L	3.2×10^{-2}	---			
413.9	412.4	M	9.5×10^{-3}	---			
432.6	385.7	K	2.4×10^{-2}	---	8.7×10^{-3}	E2(1.5×10^{-2})	E2
432.6	425.3	L	4.5×10^{-3}	---	1.6×10^{-3}	E2(2.8×10^{-2})	E2
468.4	421.5	K	2.6×10^{-2}	---	5.3×10^{-2}	M1(2.1×10^{-2})	---
550.2	503.3	K	9.9×10^{-1}	1.1×10^{-1}	8.6×10^{-3}	E2(8.1×10^{-3})[a]	E2
553.2	506.3	K	9.9×10^{-1}	1.1×10^{-1}	8.6×10^{-3}	E2(8.1×10^{-3})[a]	E2
553.2	543.0	L	1.3×10^{-1}	1.8×10^{-1}	1.1×10^{-3}	E2(1.4×10^{-3})	---
553.2	546.0	L	1.3×10^{-1}	1.8×10^{-1}	1.1×10^{-3}	E2(1.4×10^{-3})	---
553.2	549.0	M	5.2×10^{-2}	5.0×10^{-2}	4.5×10^{-4}	E2(1.4×10^{-3})	---
553.2	552.0	M	5.2×10^{-2}	5.0×10^{-2}	4.5×10^{-4}	E2(1.4×10^{-3})	---
571.9	525.0	K	1.9×10^{-2}	2.6×10^{-2}	2.4×10^{-3}	E1(2.6×10^{-3})	---
611.3	564.4	K	4.5×10^{-2}	5.0×10^{-2}	2.3×10^{-3}	E1(2.3×10^{-3})	E1
611.3	604.0	L	5.4×10^{-3}	5.0×10^{-3}	2.8×10^{-4}	E1(3.1×10^{-4})	E1
629.9	583.0	K	4.0×10^{-1}	4.2×10^{-1}	5.7×10^{-3}	E2(5.6×10^{-3})[a]	E2
629.9	622.5	L	5.4×10^{-2}	6.0×10^{-2}	8.0×10^{-4}	E2(9.8×10^{-4})	E2
629.9	628.4	M	1.6×10^{-2}	1.9×10^{-2}	2.5×10^{-4}	E2(9.8×10^{-4})	E2
654.3	607.4	K	1.1×10^{-2}	2.0×10^{-2}	1.0×10^{-2}	M1(9.0×10^{-3})	---
654.3	607.4	L	---	6.0×10^{-3}	3.0×10^{-3}	M1(3.0×10^{-3})	---
683.2	636.3	K	9.0×10^{-3}	1.2×10^{-2}	8.6×10^{-3}	M1(8.2×10^{-3})	---
714.8	667.9	K	1.1×10^{-2}	1.6×10^{-2}	7.5×10^{-3}	M1(7.4×10^{-3})	E1
725.7	678.8	K	4.9×10^{-2}	6.6×10^{-2}	4.7×10^{-3}	E2(4.2×10^{-3})	E2
725.7	718.4	L	3.7×10^{-3}	1.0×10^{-2}	5.0×10^{-4}	E2(6.8×10^{-4})	E2
725.7	725.7	M	9.2×10^{-4}	---	5.0×10^{-4}	E2(6.8×10^{-4})	E2
870.0	823.1	K	1.4×10^{-2}	1.8×10^{-2}	2.9×10^{-3}	E2(2.8×10^{-3})	E2
870.0	862.7	L	1.0×10^{-3}	---	2.0×10^{-3}	E2(4.3×10^{-4})	E2
895.9	848.8	K	1.2×10^{-3}	---	2.1×10^{-3}	E2(2.6×10^{-3})	---
915.3	868.4	K	5.1×10^{-3}	---	2.1×10^{-3}	E2(2.4×10^{-3})	E2
930.5	883.6	K	4.4×10^{-3}	---	1.9×10^{-3}	E2(2.3×10^{-3})	E2
967.3	920.4	K	5.7×10^{-3}	---	2.0×10^{-3}	E2(2.2×10^{-3})	E2

TABLE II-7 (contd.)

LIST OF CONVERSION-ELECTRON ENERGIES AND INTENSITIES
(Included in this table are the assignments of the electrons, measured conversion coefficients, and multipolarity assignments. Along with the multipolarity assignments are given the theoretical conversion coefficients corresponding to these assignments.)

E_γ	E_e	Assign	$I_e(\text{Si})$	$I_e(\text{mag})$	α	Multipolarity Assignment	Reference 19
1034.1	987.2	K	1.5×10^{-2}	---	1.9×10^{-3}	$E2(1.9 \times 10^{-3})$	E2
1034.1	1026.8	L	1.3×10^{-3}	---	1.6×10^{-4}	$E2(2.7 \times 10^{-4})$	E2
1146.9	1100.0	K	2.9×10^{-3}	$\approx 6.0 \times 10^{-3}$	1.6×10^{-3}	$E2(1.58 \times 10^{-3})$	E2
1183.3	1136.4	K	2.6×10^{-3}	---	1.5×10^{-3}	$E2(1.4 \times 10^{-3})$	E2
1344.6	1297.7	K	3.1×10^{-3}	$\approx 3.0 \times 10^{-3}$	9.4×10^{-4}	$E2(1.1 \times 10^{-3})$	E1

[a] Used for electron intensity normalization.

values of the conversion coefficients calculated by Sliv^[22] for these multipolarities and the multipolarities given by Baba et al^[20]. The only differences in the assignments in the present work and those reported by Baba occur in the assignment for the 714.8- and for the 1344.6-keV transitions. The reason for these discrepancies lies only in the gamma-ray intensities and not in the electron intensities. This is not surprising since accurate analysis of gamma-ray intensities would have been almost impossible without the resolution afforded by the germanium devices. Table II-7 includes the electron intensities derived from both the Si(Li) and magnetic spectrometers. The agreement between these two intensity determinations is felt to be indicative of the uncertainties to be associated with these values.

Gamma-gamma coincidence measurements were carried out using a 256- x 256-channel multi-parameter analyzer. Measurements were made with two NaI(Tl) detectors and also with one NaI(Tl) and one Ge(Li) detector. Because of the highly complex nature of this decay, only the latter type of measurements proved to be of much value. Because of counting-rate considerations, a two-Ge(Li) coincidence system was not used in this experiment. The Ge(Li) spectrometer used for these measurements was a 4-cm³ detector having a resolution of approximately 2.5 keV (FWHM) at 100 MeV and approximately 4 keV at 1 MeV. In the taking of these data, the detectors were operated at 180 degrees with respect to one another and the source was located between them at a distance of 10 cm from the front face of the 3- x 3-inch NaI(Tl) detector and 4 cm from the front face of the Ge(Li) detector. Wherever possible, the coincidence spectrum was analyzed by spectrum-stripping techniques for the intensities of the constituents. In most cases, the intensity analyses permitted a unique assignment for the order of the transitions. Careful analyses of the data taken coincident with the 611- and with the 630-keV gamma rays were made. These yielded considerable information as to which of the major transitions fed these transitions and with fed the 550-keV level directly. In many cases, the intensity information from these data enabled one to decide between gamma rays which were unresolved in the spectrum taken with the NaI(Tl) detector. The spectrum coincident with the 550-keV transition, except for contributions from coincidences with the 553-keV transition, was found to be identical with the singles spectrum. Data taken coincident with the 654-, 683-, 915-, 967-, 1034-, 1622-, and 1650-keV transitions

showed only the presence of the 550- and 630(611)-keV transitions, indicating that these transitions directly fed either the level at 1161 or the level at 1180 keV.

The proposed decay scheme for Eu-148 is presented in Figure II-14. Two categories of placement of transitions and levels in the scheme are shown. Transitions and levels represented by solid lines are those established either by

Fig. II-14 Proposed decay scheme for Eu-148. The levels and transitions shown as dashed lines represent those placed solely on the basis of energetics and have no corroborative coincidence results. Shown on the figure are gamma-ray energies and absolute intensities.

conclusive coincidence information or from the previously reported[19] decay of Pm-148 and Pm-148m. Those represented by dashed lines indicate placement of a transition where the energy difference between two known levels corresponds to an observed gamma ray within the reported uncertainties involved, or where three or more gamma rays could be placed with the establishment of a single new level. Of the 30 levels in this level scheme, only 7 are proposed by this latter technique. With over 120 gamma rays and 30 energy levels, it is likely that some of the transitions placed on the basis of energy differences are simply the result of random chance. Thus, the dashed transitions and levels are included in the scheme with some reservation.

Spins and parities which are shown on the proposed level scheme are those which have been assigned by other measurements[19,20] or through the few unambiguous multipolarity assignments made on the basis of the conversion-coefficient measurements. Those spin and parity assignments shown enclosed by brackets are made on the basis of decay systematics to lower lying levels of known spin and parity. Log-foot values for the electron-capture groups are included on the scheme. These values range, in general, from 8.5 to 9.5. These values are calculated assuming a Q of 3.11 MeV[20].

In the main, the proposed level scheme is consistent with that proposed by Baba et al[20]. With the exception of the many additional levels proposed above 2 MeV, and the more precise energies, the only difference lies in the parity assignment to the level at 1894.9 keV. From the present work, the 714.8- and the 1344.6-keV transitions have been identified as being E1 and E2, respectively. Thus, the level at 1894 keV, from whence these transitions originate, must have positive parity.

Most of the levels which have been identified in this experiment probably correspond to levels reported by Kenefick and Sheline[23] from (d,p) and (p,p') experiments and by Veje[24] from (d,d') experiments.

Interpretations of the level structure of Sm-148 have been made primarily by Baba et al[20], Bhatt[25], and Lipas[26]. Due to the incomplete and ambiguous nature of the information available on this nuclide, no attempt is made here to elaborate further on those possible interpretations. However, it should be pointed out that the level at 1173 keV included by Lipas[26] in his discussion of the two-phonon quadrupole-octupole excitations has been shown by the present study as well as by Baba[20] to have positive parity. Thus, it cannot be interpreted as a quadrupole-octupole state.

3.3 Precise Measurements of Primary Capture Gamma-Ray Energies Using a "Boot-Strap" Method (R. C. Greenwood)

A good resolution lithium-drifted germanium-detector system employing a 4096-channel analog-to-digital converter is capable, in principle, of providing measurements of gamma-ray energies with a precision of about 1 part in 10^4 over a wide range of energies. It has not, however, been possible to fully utilize this capability in capture gamma-ray spectroscopy because precise gamma-ray energy standards were not available above 3 MeV. To help remove this limitation, a series of measurements has been made to obtain precise values for the energies of selected primary capture gamma-ray lines. A "boot-strap" approach was used to make these measurements.

Briefly, in the "boot-strap" approach the neutron binding energy of the compound nucleus formed in the neutron capture process is obtained as a sum of the transition energies in a two- or three-step gamma-ray cascade from the capturing state to the ground state. In order to achieve the greatest precision, cascades were used where all of the transitions have double-escape peak energies less than 3 MeV. Once a value for the neutron binding energy is established, it is then possible to obtain equally precise values for the energy of a primary capture gamma ray emitted in direct transitions from the capturing state to low energy levels.

In order to measure gamma-ray line energies, calibration sources were recorded simultaneously with the capture gamma-ray spectra. The capture gamma-ray line energies (below 3 MeV) were obtained by making a least-squares straight-line fit to the peak positions of the calibration lines after making a correction for the nonlinearity of the detector system. A more complete discussion of this technique for measuring gamma-ray energies has been given in a number of references by Heath, Black, and Cline [18, 27, 28].

Beryllium and sodium were selected as samples for the first series of experiments because of the relative simplicity of the Be-10 and Na-24 compound nuclear decay schemes and because each has a strong primary capture gamma-ray transition to a low-lying state. The target samples were irradiated using an external neutron beam from the MTR; and prompt gamma rays were detected in a 2.5-cm² x 4-mm Ge(Li) detector, and in later measurements in a 10-cm³ coaxial detector.

A typical beryllium capture gamma-ray spectrum, using the annihilation radiation [29] and the hydrogen capture gamma-ray lines [30] as internal energy standards, is shown in Figure II-15. Lines not identified in this figure result from background. A total of seven separate beryllium capture gamma-ray spectra were measured. In four of these measurements, the 15-hour Na-24 lines were used as energy standards [29] instead of the hydrogen capture gamma-ray lines. The beryllium capture gamma-ray line energies shown in Table II-8 are an average of these seven runs. The decay scheme of the Be-10 compound nuclear state formed by thermal neutron capture in beryllium has been determined by Motz et al [31]. Based on their decay scheme, the sums of the transition energies of each of the two cascades suitable for use in determining the neutron binding energy of Be-10, are also shown in Table II-8. Since both of these cascades go directly to the ground state, we obtain

$$\text{Be-10 neutron binding energy} = 6811.9 \pm 0.4 \text{ keV} .$$

And thus, the energy of the primary, thermal neutron capture gamma ray emitted in the strong transition to the ground state of Be-10 is

$$6809.4 \pm 0.4 \text{ keV} .$$

The annihilation radiation and the 15-hour Na-24 activity, which is built up through the neutron irradiation of the target, were used as standards for the measurements of sodium capture gamma-ray energies. In making these energy measurements it was necessary to make a small correction (about 0.2 keV) to the energy of the calibration lines resulting from the 2753.92-keV Na-24 decay gamma rays because of the presence of the weaker 2752.3 keV sodium capture gamma rays. The decay scheme which was obtained during the course of the

Fig. II-15 A beryllium thermal neutron capture gamma-ray spectrum measured with a 2.5-cm² x 4-mm Ge(Li) detector.

TABLE II-8

PROMPT GAMMA-RAY TRANSITION
ENERGIES IN THE Be-9 (n, γ) REACTION AND THE
NEUTRON BINDING ENERGY VALUES DETERMINED FOR Be-10 BY SUMMING
TRANSITION ENERGIES OF TWO- AND THREE-STEP CASCADES OCCURRING
BETWEEN THE CAPTURING STATE AND THE GROUND STATE IN Be-10

Average Line Energy[a] (keV)	Gamma-Ray Energy (keV)	Nuclear Recoil Energy Correction (keV)	Transition Energy (keV)
853.5 $\pm$ 0.3	853.5 $\pm$ 0.3	0.04	853.5 $\pm$ 0.3
1567.9 $\pm$ 0.2 ₅	2589.9 $\pm$ 0.2 ₅	0.36	2590.3 $\pm$ 0.2 ₅
2345.4 $\pm$ 0.2	3367.4 $\pm$ 0.2	0.61	3368.0 $\pm$ 0.2
2421.3 $\pm$ 0.3	3443.3 $\pm$ 0.3	0.64	3443.9 $\pm$ 0.3

Be-10 Cascade Sums

$$3443.9 + 3368.0 = 6811.9 \pm 0.4$$

$$853.5 + 2590.3 + 3368.0 = 6811.8 \pm 0.5$$

[a] The error values shown here contain an allowance for the uncertainties in the energies of the calibration line.

present work for the Na-24 compound nuclear state formed by thermal neutron capture in sodium is shown in Figure II-16. Based on this decay scheme, 11 well defined two-step cascades were identified as occurring between the capturing state of Na-24 and levels at either 472.32, 563.31, or 1341.36 keV. These cascades are shown in Table II-9. The values obtained from each of these cascades for the Na-24 neutron binding energy are shown in the final column of this table. There is a very small spread, a total of 0.7 keV, in these individual binding energy values. From this data, we obtain

$$\text{Na-24 neutron binding energy} = 6959.3 \pm 0.4 \text{ keV} .$$

(This error assignment contains an allowance for the errors assigned to the standards and to the low energy Na-24 levels[32].) Thus, based on this Na-24 binding energy value, the energy of the primary capture gamma ray emitted in the strong transition to the second excited state of Na-24 is

$$6395.1 \pm 0.4 \text{ keV} .$$

When these Be-10 and Na-24 primary capture gamma-ray lines are themselves used as energy standards, excellent linear fits are still achieved to the energy calibration data, thus indicating that this Ge(Li) detector had a linear response over this energy range. From these data the energy of the principal gamma ray emitted in the decay of 7.35-sec N-16 was measured as

$$6128.9 \pm 0.4 \text{ keV} .$$

Fig. II-16 The decay scheme of the compound nuclear state of Na-24 produced as a result of thermal neutron capture in sodium.

TABLE II-9

VALUES FOR THE NEUTRON BINDING ENERGY OF
Na-24 OBTAINED BY SUMMING TRANSITION ENERGIES OF TWO-STEP
PROMPT GAMMA-RAY CASCADES OCCURRING BETWEEN THE CAPTURING
STATE AND LOW LYING LEVELS AT 472.32, 563.31, AND 1341.36 keV

Two-Step Cascades in Na-24 (keV)[a]	Na-24 Neutron Binding Energy Values (keV)
<u>Transitions to the 472.32-keV Level</u>	
3981.64 + 2505.4 = 6487.0 ± 0.3	6959.4 ± 0.3
2981.9 + 3504.60 = 6486.5 ± 0.4	6958.8 ± 0.4
2517.70 + 3969.40 = 6487.1 ± 0.4	6959.4 ± 0.4
<u>Transitions to the 563.31-keV Level</u>	
3981.64 + 2414.4 = 6396.0 ± 0.3	6959.4 ± 0.3
3587.53 + 2808.4 = 6395.9 ± 0.3	6959.2 ± 0.3
2517.70 + 3878.23 = 6395.9 ± 0.3	6959.2 ± 0.3
2208.5 + 4187.69 = 6396.2 ± 0.3	6959.5 ± 0.3
3370.2 + 3025.8 = 6396.0 ± 0.4	6959.3 ± 0.4
2981.9 + 3414.1 = 6396.0 ± 0.6	6959.3 ± 0.6
<u>Transitions to the 1341.36-keV Level</u>	
3587.53 + 2030.2 = 5617.7 ± 0.4	6959.1 ± 0.4
2208.5 + 3409.5 = 5618.0 ± 0.5	6959.4 ± 0.5

[a] The error values shown here contain an allowance for the uncertainties in the energies of the calibration lines.

(This line was present in many of the spectra as a contaminant.) The energies measured separately for this O-16 line from the beryllium and sodium capture gamma-ray spectra provide a check on the internal consistency of the present Be-10 and Na-24 data; these values are 6128.8 and 6128.9 keV, respectively.

In order to extend the range still further over which capture gamma-ray energy standards are available, measurements were also made on the nitrogen neutron capture gamma-ray spectrum, now using the sodium neutron capture gamma lines (together with the annihilation radiation, Co-60, and the hydrogen capture gamma-ray lines) as energy standards. Based on the decay scheme proposed by Motz et al^[31] for the compound nuclear state in N-15, the sums of the transition energies of the cascades which were found suitable for use in determining the neutron binding energy of N-14 are also shown in Table II-10. Since each of these cascades go directly to the ground state, we obtain

TABLE II-10

PROMPT GAMMA-RAY TRANSITION
ENERGIES IN THE N-14 (n, γ) REACTION AND
THE NEUTRON BINDING ENERGY VALUES DETERMINED FOR N-15 BY SUMMING
TRANSITION ENERGIES OF TWO- AND THREE-STEP CASCADES OCCURRING
BETWEEN THE CAPTURING STATE AND THE GROUND STATE

Average Line Energy[a] (keV)	Gamma-Ray Energy (keV)	Nuclear Recoil Energy Correction (keV)	Transition Energy (keV)
$1884.9 \pm 0.2_5$	$1884.9 \pm 0.2_5$	0.12	$1885.0 \pm 0.2_5$
$2518.0 \pm 0.2_5$	$2518.0 \pm 0.2_5$	0.23	$2518.2 \pm 0.2_5$
$2510.3 \pm 0.2_5$	$3532.3 \pm 0.2_5$	0.44	$3532.7 \pm 0.2_5$
$2656.1 \pm 0.2_5$	$3678.1 \pm 0.2_5$	0.48	$3678.6 \pm 0.2_5$
3487.2 ± 0.3	4509.2 ± 0.3	0.73	4509.9 ± 0.3
$4247.6 \pm 0.3_5$	$5269.6 \pm 0.3_5$	0.99	$5270.6 \pm 0.3_5$
$4276.3 \pm 0.3_5$	$5298.3 \pm 0.3_5$	1.00	$5299.3 \pm 0.3_5$
4511.5 ± 0.4	5533.5 ± 0.4	1.09	5534.6 ± 0.4
4540.6 ± 0.4	5562.6 ± 0.4	1.11	5563.7 ± 0.4
$5300.0 \pm 0.4_5$	$6322.0 \pm 0.4_5$	1.43	$6323.4 \pm 0.4_5$

N-15 Cascade Sums

$$3678.6 + 1885.0 + 5270.6 = 10,834.2 \pm 0.7$$

$$5563.7 + 5270.6 = 10,834.3 \pm 0.7$$

$$5534.6 + 5299.3 = 10,833.9 \pm 0.7$$

$$4509.9 + 6323.4 = 10,833.3 \pm 0.7$$

[a] The error values shown here contain an allowance for the uncertainties in the energies of the calibration lines.

N-15 neutron binding energy = $10,833.9 \pm 0.7$ keV.

And thus, the energy of the primary, thermal neutron capture gamma ray emitted in the strong transition to the ground state of N-15 is

$$10,829.7 \pm 0.7 \text{ keV.}$$

4. REACTOR EXPERIMENTS (E. Fast)

4.1 Resonance Absorption Integral of Lu-176 (J. J. Scoville, J. W. Rogers)

The infinitely dilute resonance absorption integral of Lu-176 has been measured in the ARMF-II cadmium-shielded facility. The experimental value is 1103 ± 34 barns. However, after correcting for the nonconstant weighting function of the reactor near the cadmium cut off energy, the integral above 0.5 eV becomes 1161 barns. No error is assigned to the corrected value since it includes the errors on the differential cross section as well as on a reactor transport calculation to determine the weighting function and cannot be explicitly evaluated. The correction is valid for a low energy limit on the resonance integral of 0.5 eV. A calculation of the resonance integral based on the same parameters as used in the correction yields a resonance integral of 901 barns. There do not appear to be any measurements in the literature for comparison to our experiment.

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