IN-1126

NUCLEAR TECHNOLOGY BRANCHES QUARTERLY REPORT

April 1 - June 30, 1967

IDAHO NUCLEAR CORPORATION
NATIONAL REACTOR TESTING STATION
IDAHO FALLS, IDAHO

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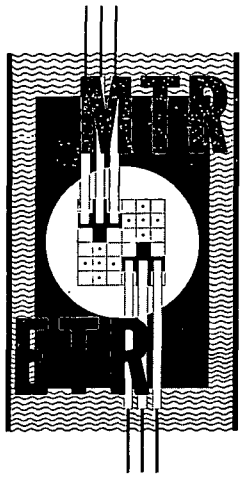

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D. R. deBoisblanc
MANAGER
NUCLEAR AND CHEMICAL
TECHNOLOGY DIVISION

W. C. Francis
MANAGER
REACTOR ENGINEERING

R. G. Fluharty
MANAGER
NUCLEAR TECHNOLOGY

IDAHO NUCLEAR CORPORATION

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SUMMARY

Irradiation studies to determine the errors in measuring fission rates and fuel element powers in ATRC have been completed. The contributions of wire positioning, fuel element positioning, beta counter precision, and wire nonuniformity to measuring errors are discussed, and the error values reported.

Conditions responsible for systematic errors in the boron assaying of ETR and ATR fuel elements are discussed, and a summary of assayed boron contents is reported.

Further results of ETRC measurements to examine the effects of rod travel on midplane fission rates are reported.

Experiments to determine ARMF regulating rod transient time have led to the adoption of new operating procedures to improve reactor operating efficiency.

The results are given for fast-chopper cross section measurements which have been continued on Pm-147. Newly determined activation cross sections of Am-243 to 10.1-hr, Am-244 and to both isomers are reported. Results also are tabulated for recently determined energies and absolute intensities of 2.1-day Np-238 gamma rays. Additional information is provided on the yields of I-135 of the common fissile isotopes relative to the yields of U-235.

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I. REACTOR ENGINEERING

1. CRITICAL FACILITIES
(E. E. Burdick)

1.1 Errors in ATRC Fission-Rate Measurements (J. L. Durney)

A series of ATRC irradiations has been completed to determine the errors associated with measurements of fission rates and fuel element powers in ATRC. In general, the errors (one standard deviation) were found to be rather small. The error in measuring the fuel element power is less than 1 percent; for fission-rate measurements in the fuel elements, the error varies (depending on the position in the element) from about 0.5 to 4 percent; and for fission-rate measurements in typical experiment positions, the error is about 1 percent. As a by-product of obtaining these data, the uniformity of the fission-rate monitors was shown to be in error by no more than 1.6 percent, the outer-shim critical position to be reproducible within 0.2 degree, and the MTR automatic beta counter to vary by no more than 0.8 percent.

The errors were determined from data obtained in 10 identical ATRC runs in which the power division was approximately 40-50-60. Fission-rate measurements were made at the horizontal midplane in fuel positions 31 through 35 (see core diagram in Figure I-1) and near the horizontal midplane of typical experiment positions. Normalization of the data from each run was based on the total power generated in these five fuel elements. The five fuel positions and the power division were chosen to produce the most severe fission-rate gradients normally expected and, hence, the largest errors. Also, it is assumed that errors for measurements in other fuel elements can be estimated from these data.

In each of the 10 runs, the No. 6 neck shims and NE-4 and SW-4 neck shims were withdrawn. The outer shims around the low-power lobes (E-12, N-34, W-12, S-34) were withdrawn to 25.0 degrees, and the outer shims around the high-power lobes (W-34, N-12, E-34, S-12) were then used to establish the critical condition. In 10 runs the average position of these four outer shim pairs at critical varied only 0.2 degree (which corresponds to less than 0.025$ in reactivity).

Fission-rate measurements are made in the ATRC to provide information on power distributions, fission-rate peaking for hot-spot calculations, and experiment fission-rate distributions for sponsors[a]. These data are routinely obtained using quarter-inch-long 40-mil uranium-aluminum wires. To determine the fuel element power, 17 of these wires are irradiated at the horizontal midplane of each fuel element. If the fission-rate peaking value at the edge of the fuel in plate 19 is desired (the core hot spot generally occurs in plate 19), an additional eight wires are included in water channel 19. (Four of these wires are used in duplicating the fission-rate measurement at each side of channel 19 by placing one wire above and one below the two wires routinely

[a] See Reference 1 for a detailed description of the techniques used in these measurements.

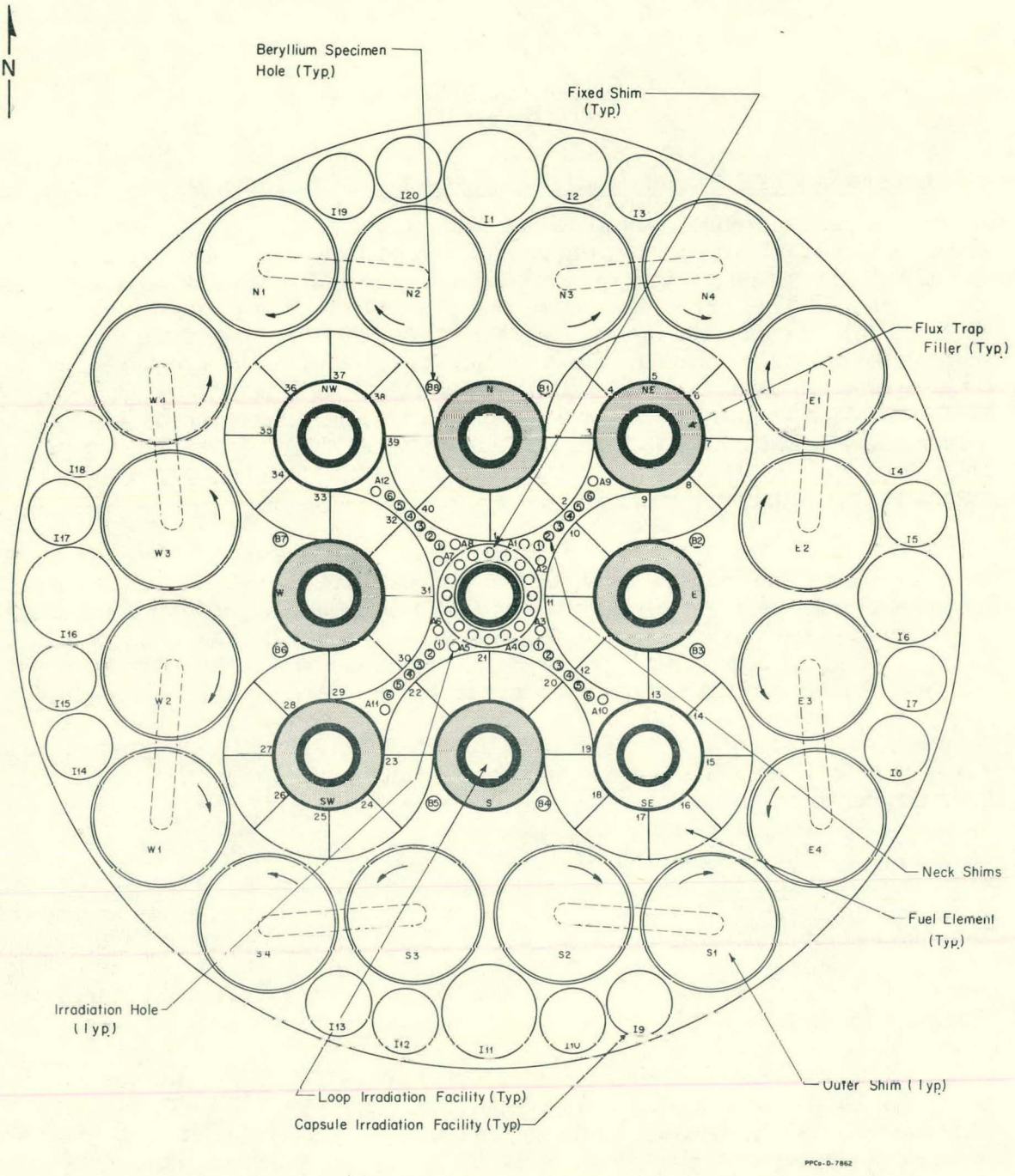


Fig. I-1 Cross-sectional diagram of ATRC core.

used at these points.) The errors in the fission-rate measurements at these 25 points (17 for power and 8 for fission-rate peaking) in fuel positions 31 through 35 are shown in Figures I-2 through -6. These data generally show that the error in measuring fission rates near the central region of a fuel element is about 1 percent. However, near the edges of a fuel element where the fission-rate gradients are much more severe, the error is as much as 4 percent. This difference is undoubtedly due to slight wire positioning errors which result in more significant errors in the severe gradients and to the fuel element positioning which will be mentioned later.

The fission rates obtained at the standard 17 points were used to calculate the fuel element powers in positions 31 through 35 for each of the 10 runs. These powers, normalized on the basis of the total power in the five elements, are shown in Table I-1 along with the calculated errors. The errors in measuring these powers are generally very small (< 1 percent).

In addition to calculating the errors outlined above, the data were used to compute the errors shown in Figure I-7 for fission-rate peaking values in water channel 19. This fission-rate peaking, used in determining the peaking in fuel plate 19, is obtained from a graph of the azimuthal fission-rate profile in

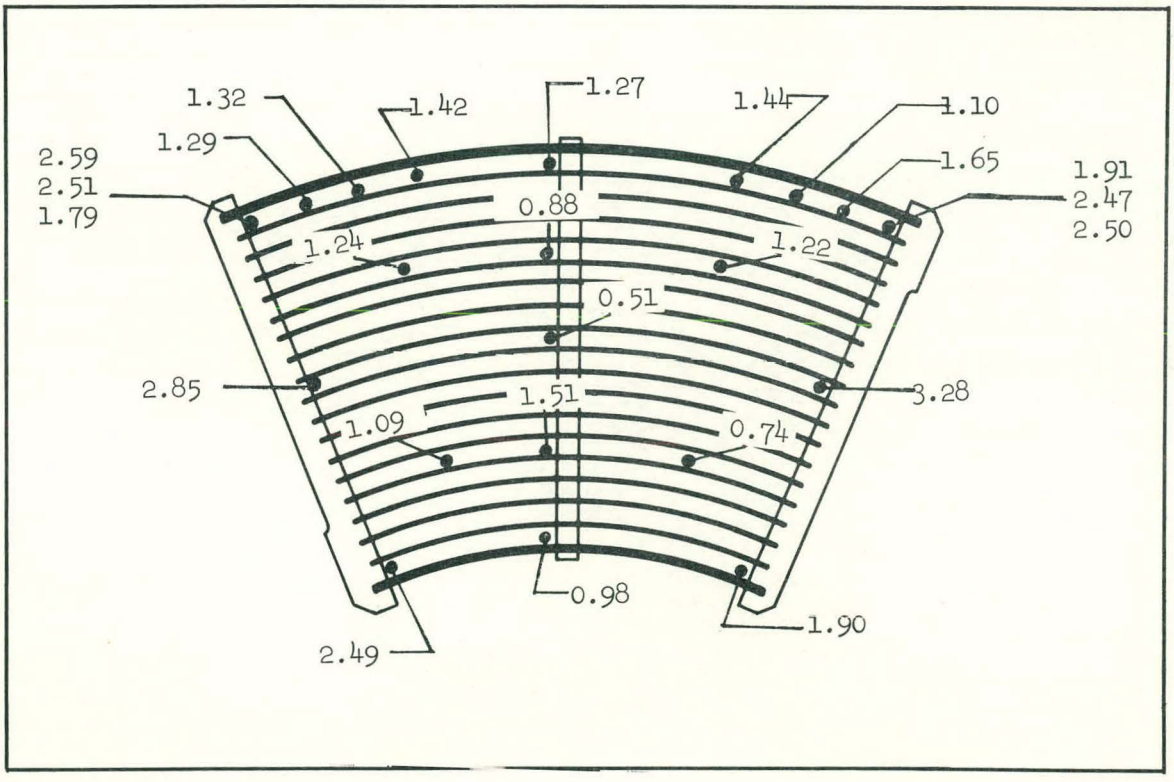


Fig. I-2 Fission-rate measurement errors (as fractional standard deviations in percent) at standard monitoring positions for ATRC fuel position 31.

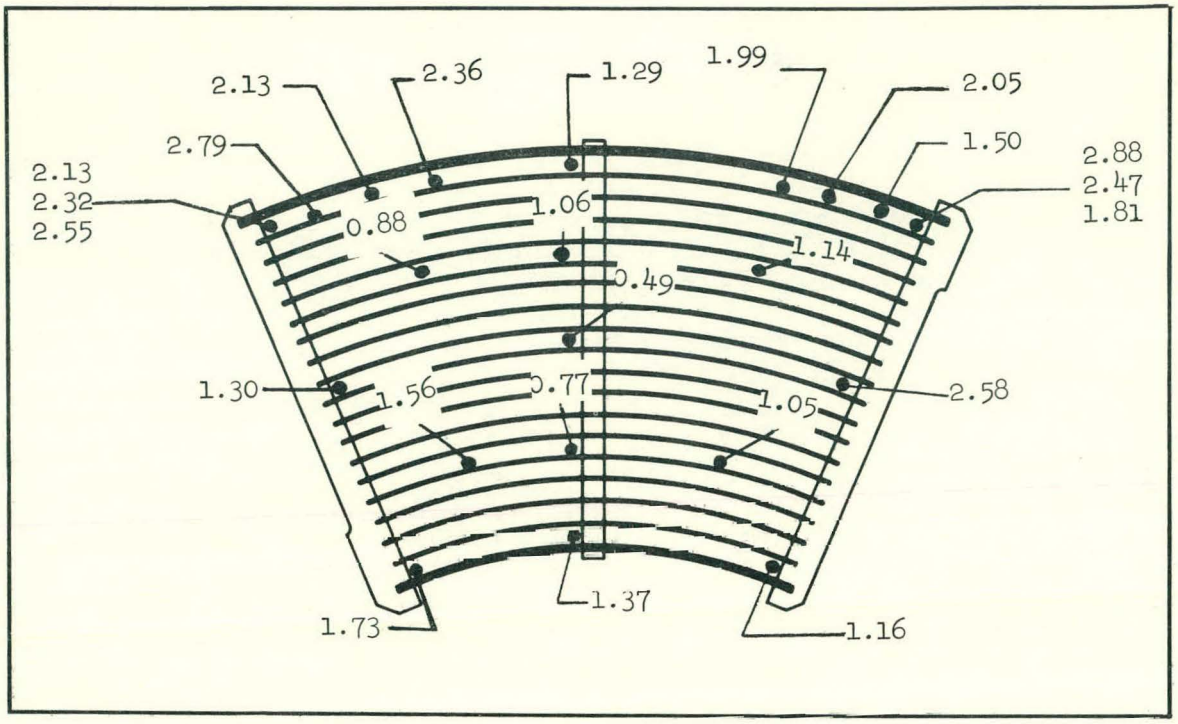


Fig. I-3 Fission-rate measurement errors (as fractional standard deviations in percent) at standard monitoring positions for ATRC fuel position 32.

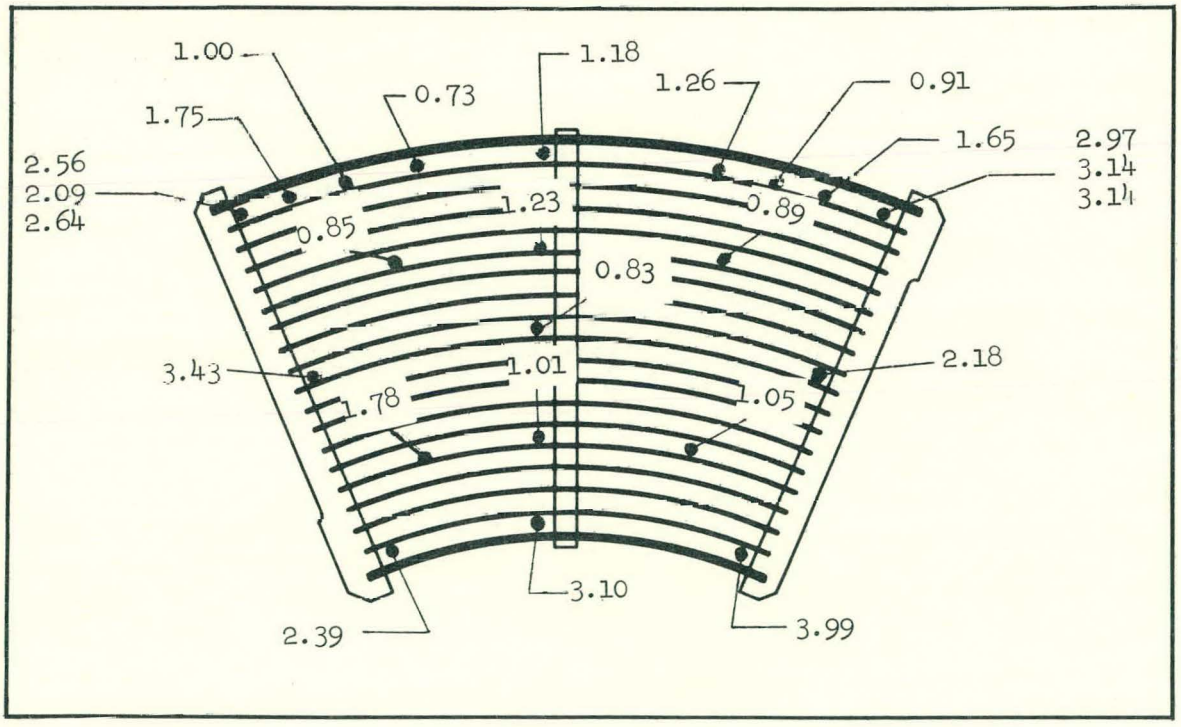


Fig. I-4 Fission-rate measurement errors (as fractional standard deviations in percent) at standard monitoring positions for ATRC fuel position 33.

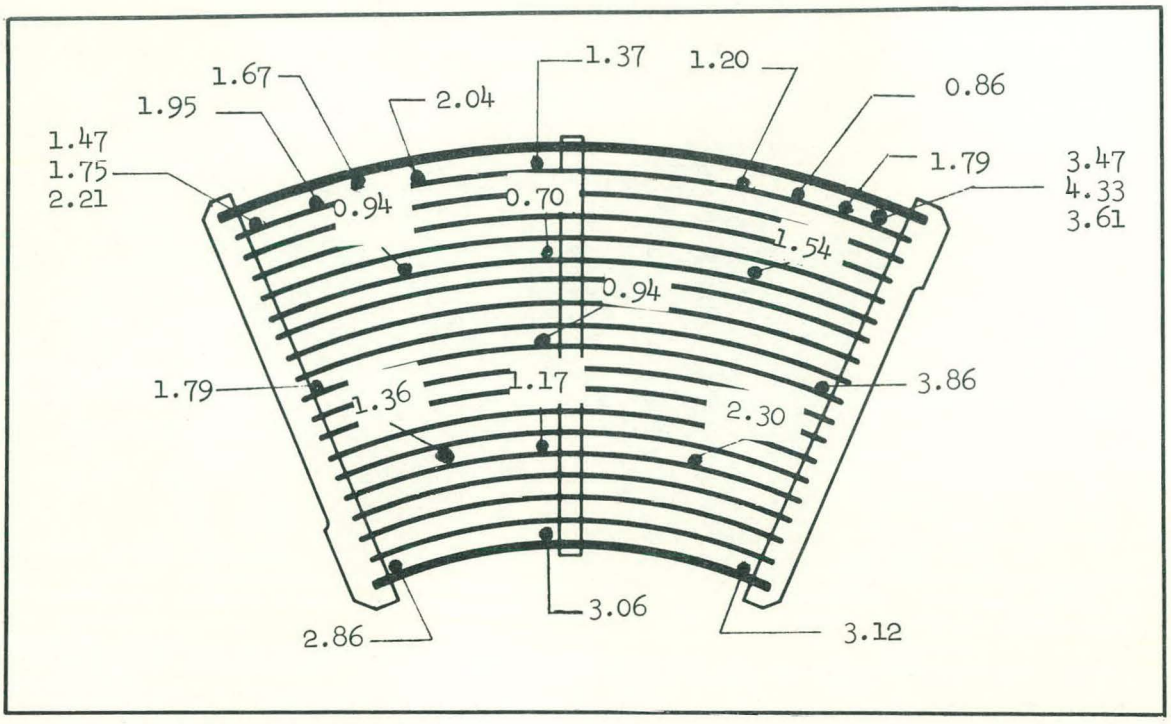


Fig. I-5 Fission-rate measurement errors (as fractional standard deviations in percent) at standard monitoring positions for ATRC fuel position 34.

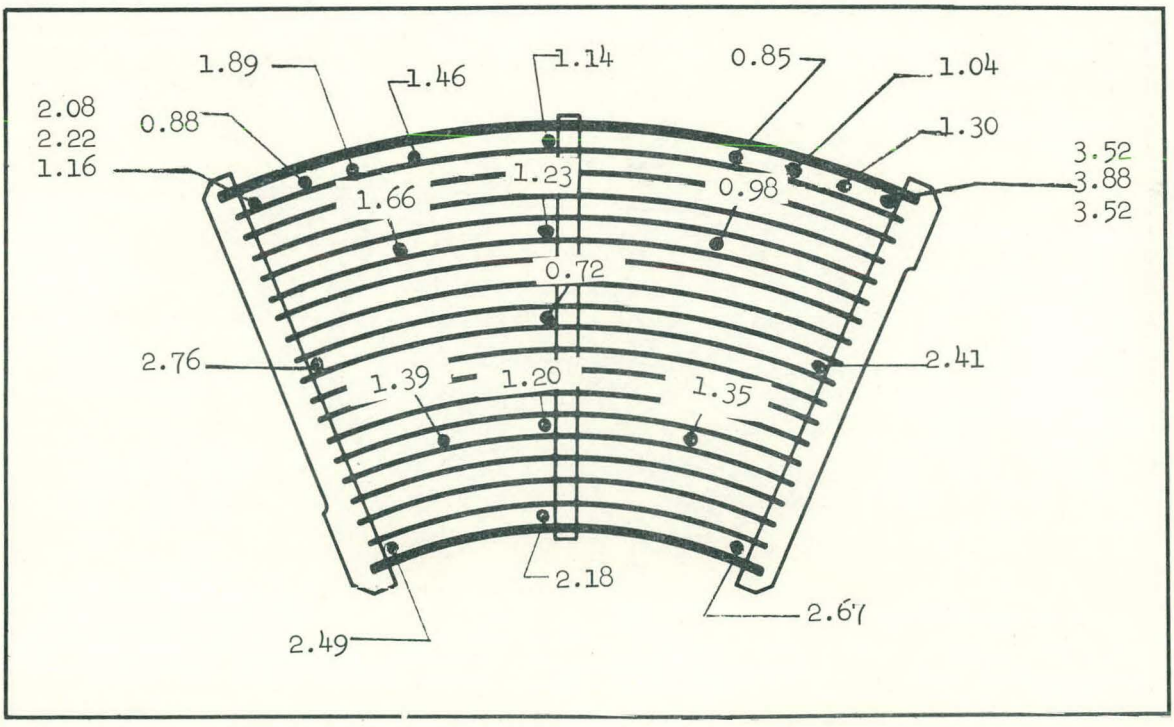


Fig. I-6 Fission-rate measurement errors (as fractional standard deviations in percent) at standard monitoring positions for ATRC fuel position 35.

TABLE I-1

NORMALIZED FUEL ELEMENT POWERS WITH ERRORS
DERIVED FROM TEN ATRC RUNS

Run No.	Fuel Element Power (Watts)				
	Position 31	Position 32	Position 33	Position 34	Position 35
1	5.75	6.63	7.06	6.28	5.62
2	5.79	6.65	7.00	6.31	5.61
3	5.78	6.66	6.97	6.28	5.65
4	5.81	6.66	7.03	6.35	5.64
5	5.75	6.62	6.96	6.19	5.53
6	5.68	6.62	6.93	6.23	5.63
7	5.78	6.64	6.96	6.37	5.57
8	5.86	6.65	6.96	6.30	5.62
9	5.76	6.60	7.08	6.23	5.68
10	5.78	6.62	7.06	6.22	5.65
Fractional Standard Deviation	0.73%	0.32%	0.76%	0.90%	0.78%

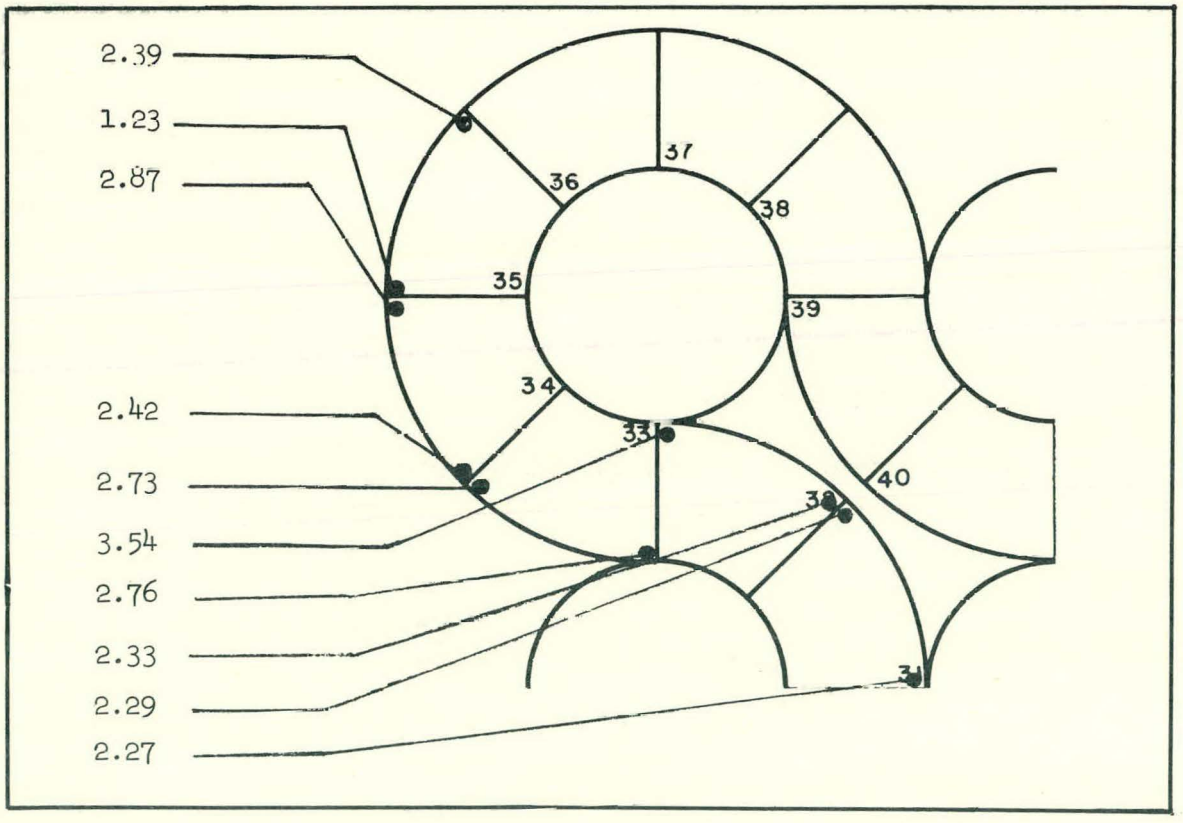


Fig. I-7 ATRC quarter-core diagram showing errors (in percent) for measuring fission-rate peaking in water channel 19 at the nominal edge of the fuel in plate 19.

that channel. In working with the graphs from the 10 runs, some inconsistencies in the fission rates were found near the side plates. Figures I-8 through -12 are comparisons of channel 19 data chosen to demonstrate the greatest differences noted. These differences also are generally substantiated by data from adjacent fuel elements. Although it has not been confirmed, these differences are primarily attributed to changes in the amount of water at the side plates caused by variations in fuel element positioning. Prior measurements have shown that a change from 0.072 to 0.145 inch in the water gap between side plates of adjacent elements can lead to an increase of about 7 percent in the channel 19 fission-rate peaking.

Fission-rate measurements also were performed and analyzed for errors in typical experiment positions using the monitor described above. The errors were measured at the horizontal midplane in the NW flux trap and in experiment positions H15 (NW side of center flux trap baffle, B7, and A12. The magnitude of the errors ranged from 0.75 to 1.45 percent.

The fission-rate errors discussed thus far are composite errors. It is generally supposed that the primary contributions to these errors arise from wire positioning in the fuel element or experiment position, fuel element positioning, beta counter errors, and wire nonuniformity. Of these four contributions, qualitative indications of the first two have been shown above, and quantitative values for the latter two are discussed below.

The MTR automatic dual-channel beta counting system is used to obtain the activity of the irradiated uranium-aluminum wires. The error contributed by this counter was determined by recounting 140 wires (20 wires from seven runs) from the series of measurements. Some of the wires were recounted on the same channel of the dual-channel system while the remaining wires were recounted on the other channel. The fractional standard deviation of these data (140 pairs of data) was 0.76 percent.

Data concerning the uniformity of the wires were obtained somewhat incidentally from the 10 runs. As pointed out above, there were three wires irradiated at both edges of channel 19 in each of five fuel positions. Since these wires were at nearly the same location they were essentially in the same neutron flux. Due to the proximity of side plate vents, however, the data from one wire of the three at the left-hand side of each element (as oriented in Figures I-2 through -6) were slightly perturbed. As a result, only the data from the wires at the right-hand side were used. The fractional standard deviation obtained for these data is 1.57 percent. However, due to the difficulty of actually placing three wires in precisely the same neutron flux, this figure undoubtedly also reflects some positioning error in addition to the counting error. It is therefore assumed that the error contributed by the wire nonuniformity is only a part of this number. This conclusion is in good agreement with results obtained earlier[2].

1.2 Systematic Errors in Boron Content Assays of ETR and ATR Fuel Elements

(E. E. Burdick, J. W. Henscheid, A. D. Mackley)

In order to measure the boron content of each ETR and ATR fuel element, a nondestructive assay is performed in the ETRC or ATRC on individual fuel elements. The boron contents are determined by relating the reactivity effect of each element in a standard fuel element position to a reactivity-effect versus boron-content calibration curve for that fuel element position. The

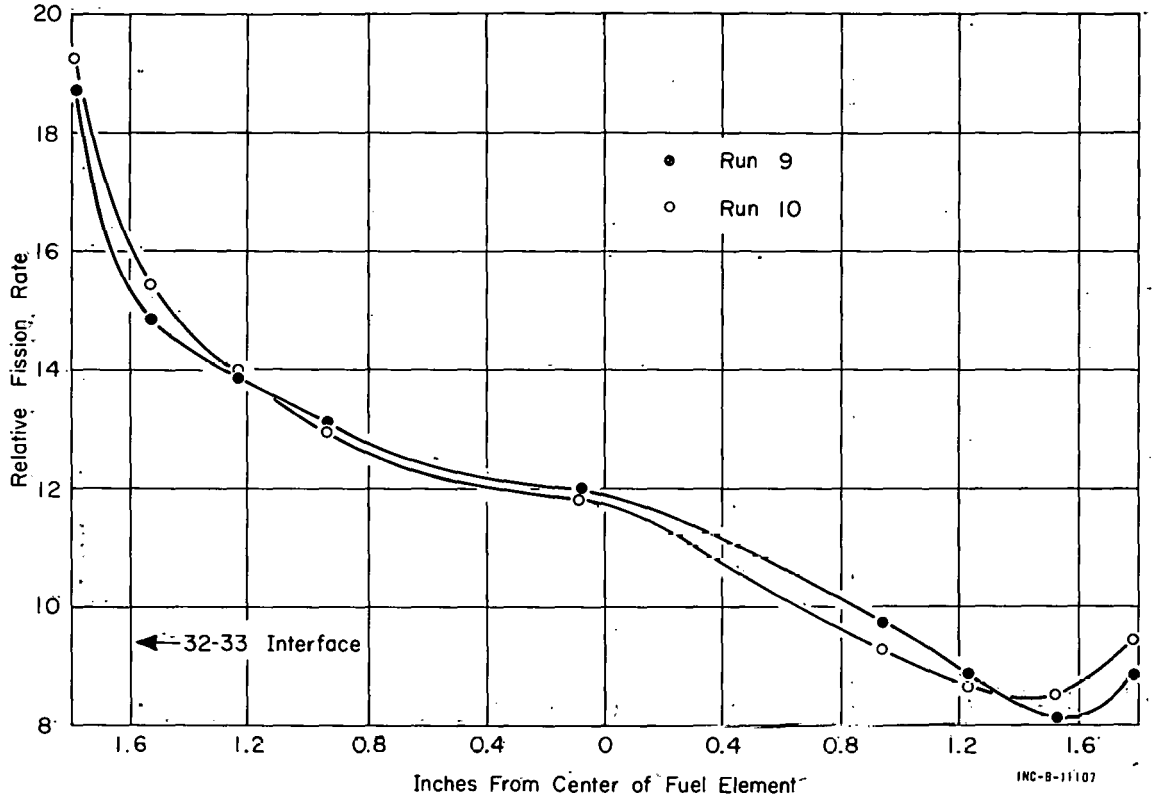


Fig. I-8 Comparison of azimuthal fission-rate profiles in water channel 19 position 32.

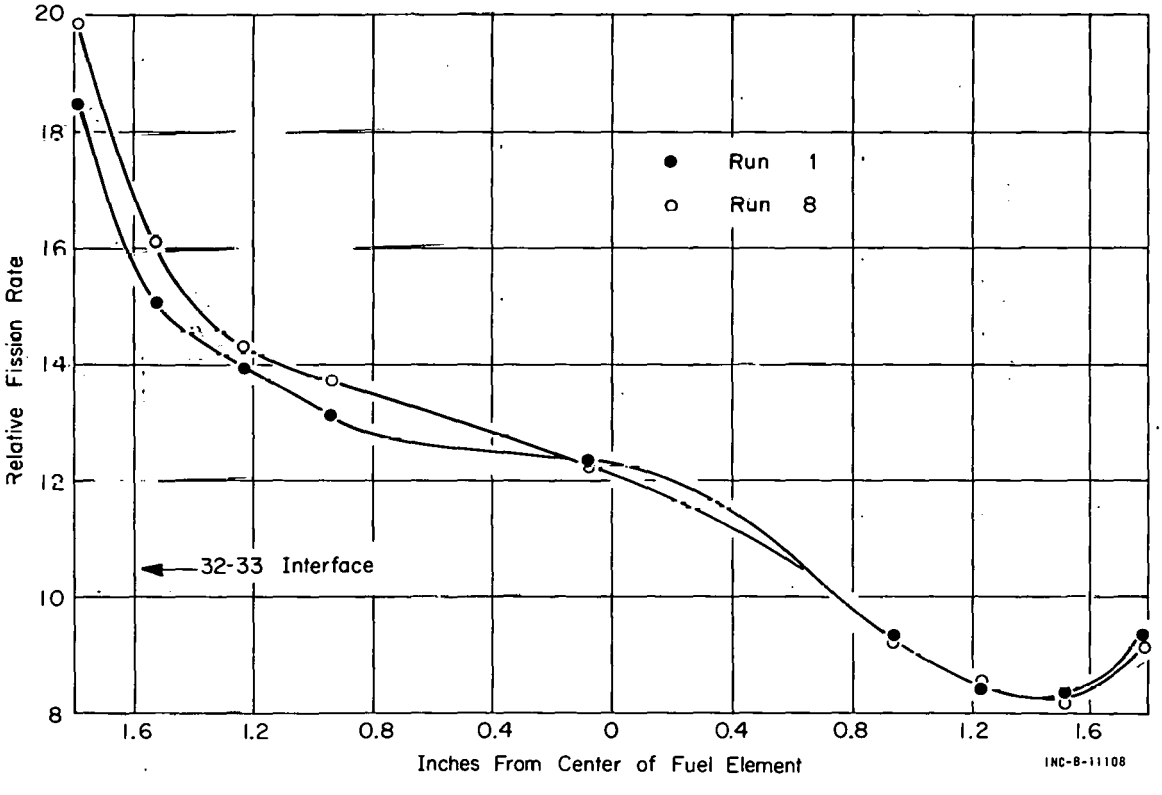


Fig. I-9 Comparison of azimuthal fission-rate profiles in water channel 19 position 32.

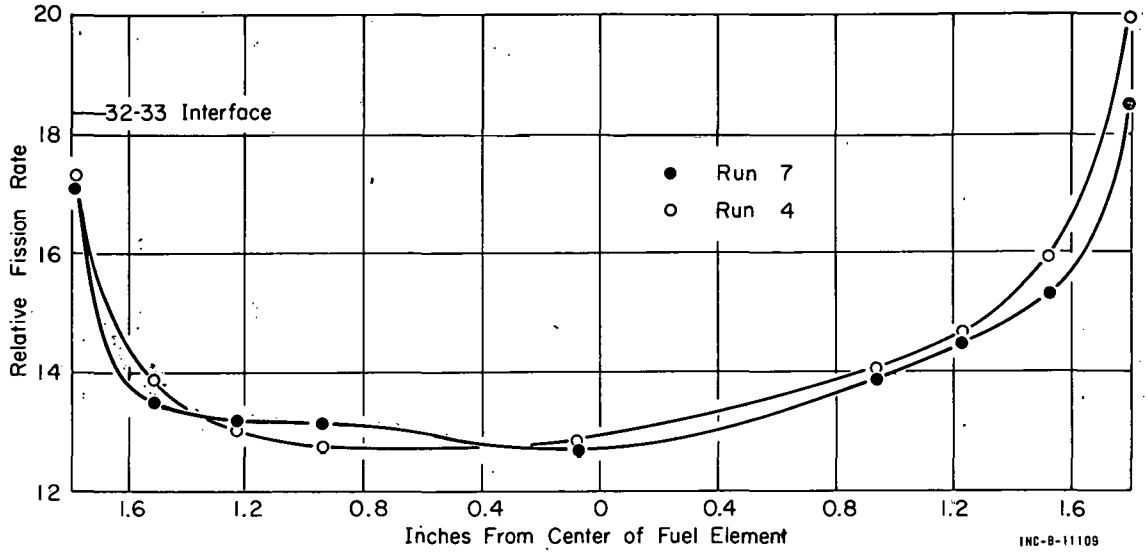


Fig. I-10 Comparison of azimuthal fission-rate profiles in water channel 19 position 33.

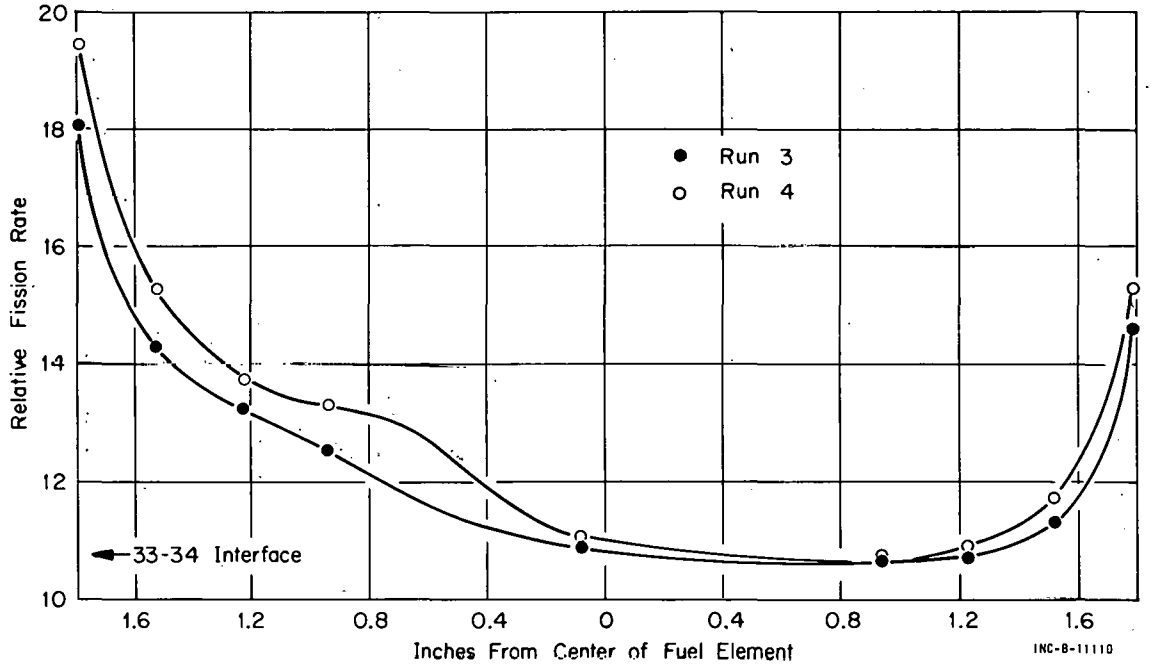


Fig. I-11 Comparison of azimuthal fission-rate profiles in water channel 19 position 34.

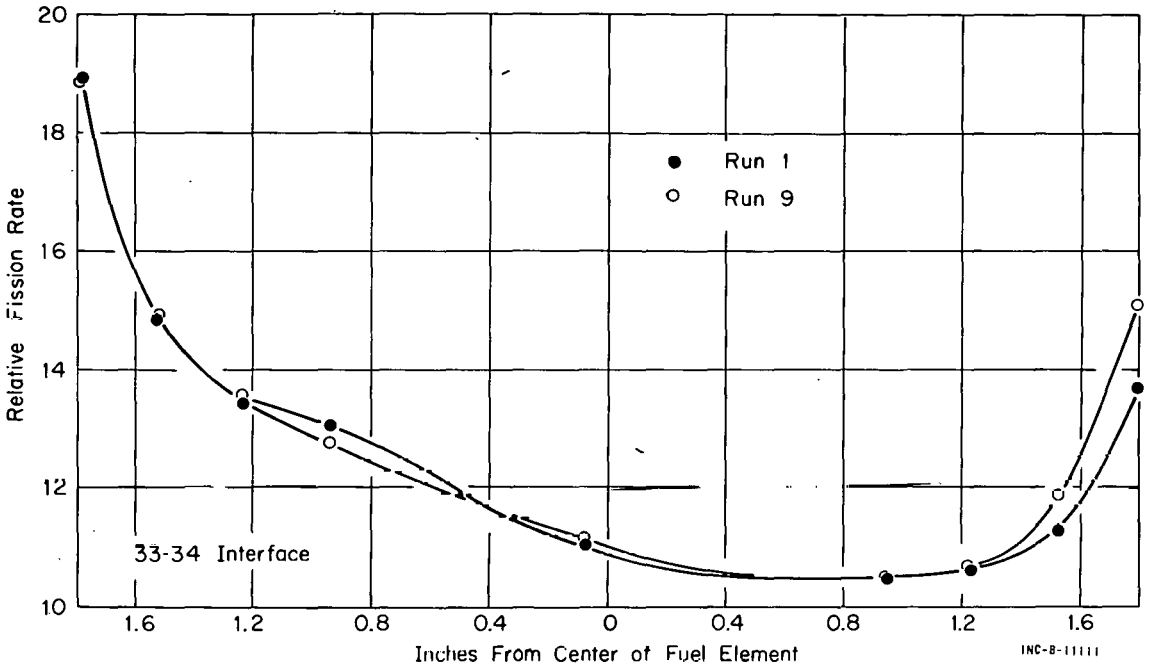


Fig. I-12 Comparison of azimuthal fission-rate profiles in water channel 19 position 34.

calibration curve is constructed from reactivity measurements with several different precisely known amounts of boron in plastic tapes inserted in a special nonborated fuel element. The borated plastic tapes are the same length as the fuel plates and are uniformly distributed in the water channels in order to simulate the boron distribution in a standard borated element. The assay includes a correction for known U-235 content differences (obtained from fabricator-supplied U-235 contents) between the standard and the fuel elements being assayed.

As previously reported[3], neutron self-shielding in boron carbide particles can affect the boron assay of fuel elements. A more complete listing of the conditions which can affect the accuracy of the boron assay is as follows:

(1) Differences between the neutron flux in the coolant channel, where the boron tapes are placed in the nonborated standard, and the neutron flux in the fuel plate, where the fabricator places the boron

(2) Differences in particle self-shielding and particle agglomeration between amorphous boron in the tapes and the boron carbide particles used by the fabricator

(3) Boron impurities in the aluminum in both the standard nonborated element and the element being assayed

(4) Errors in the U-235 content of both the standard and the elements being assayed

(5) Errors in the boron content of the tapes.

Each of these conditions can cause the assayed boron content to be either higher or lower than the actual content. An analysis of all of these effects is not yet complete. However, a review of boron assay summaries (Table I-2) of elements received from two suppliers indicates that these effects are

TABLE I-2

SUMMARY OF ASSAYED BORON CONTENTS OF ETR AND ATR FUEL ELEMENTS

	ETR[a]				ATR[b]
Specified U-235 content (g)	500	500	500	400	975
Specified boron content (g)	3.8	4.0	4.5	2.8	4.13
Mean boron content[c] (Assayed tape equivalent)	3.92	3.92	4.56	2.56	4.14
Deviation of mean assayed boron content from specification (g)	+0.12	-0.08	+0.06	-0.24	+0.01
Minimum boron content (g)	3.61	3.85	4.54	2.42	3.96
Maximum boron content (g)	4.14	4.07	4.60	2.73	4.46
Spread (max-to-min)	0.53	0.22	0.06	0.31	0.50
Standard deviation (g)	0.16	0.10	0.03	0.12	0.10
Number of elements	17	5	5	33	212

[a] Fabricated by Atomics International.

[b] Fabricated by Sylcor.

[c] Precision of boron assay is 0.05 g.

small or tend to compensate -- probably both conditions exist. The mean deviation between the specified boron content and the mean assayed (tape equivalent) boron content is 0.1 g. This can be compared to a deviation of -0.24 g (nominal) that would be expected if six percent self-shielding effect existed in the boron carbide particles and none existed in the amorphous boron in the tapes. On the other hand, the mean assayed content of the 400-g elements has a significant deviation, indicating that there is a variation in these effects. This deviation may have resulted from fabrication difficulties.

The boron content control that has been achieved within the different groups of elements reported in Table I-2 also is of interest. The maximum spread is 0.5 g and the maximum standard deviation about the mean is 0.16 g. This demonstrates that the boron control needed for ETR flux stabilization can be achieved using powder-metallurgical fabrication methods.

1.3 Boron Content of ATR Fuel Elements (J. C. Tappendorf)

The natural boron contents of each of the 212 nominally 975-g (U-235) ATR fuel elements on Sylcor Contract C259 have been measured in the ATR Critical Facility. The contents range from 3.96 to 4.45 g per element with an average of 4.14 g. (The contract specification called for 5.28 g B_4C per element, which is equivalent to 4.13 g natural boron.) A histogram of the boron content distribution among the 212 elements is shown in Figure I-13.

The boron contents were determined in the manner described in Section I-1.2 above. Variations in fabricator-reported U-235 content were so small that corrections for these variations were unnecessary. No attempt has been made to correct for other possible systematic errors discussed previously. The precision of these assayed values is ±0.05 g (1 σ).

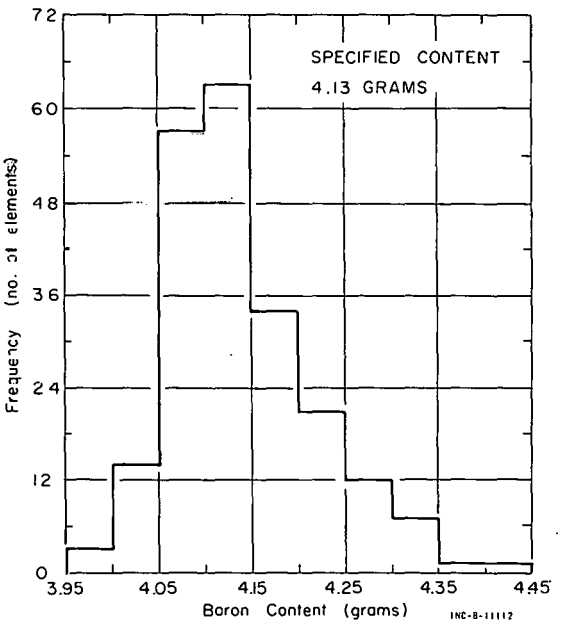


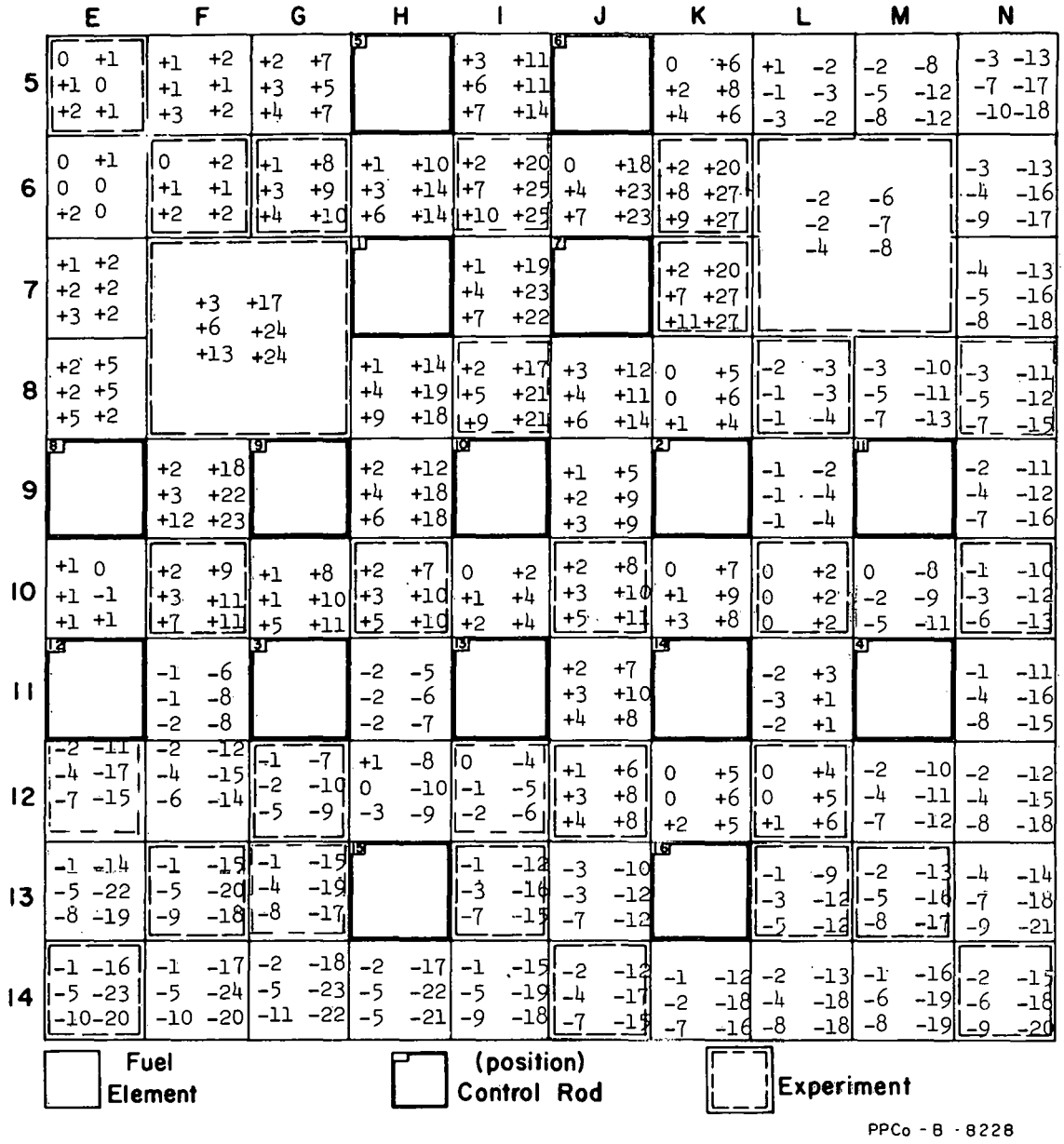
Fig. I-13 Frequency distribution of natural boron in the 212 nominally 975-g, ATR fuel elements on Sylcor Contract C-259.

1.4 ETRC Measurement of the Effects of Rod Travel on Fission Rate (R. J. Forrester, A. D. Mackley)

A major item of concern in stabilizing ETR fluxes has been the region of rod travel from the time xenon equilibrium is first reached until the end of cycle. The effects of 7-9-14 rod travel on vertical flux distribution were reported in Reference 4 where it is shown that rod travel about the midplane causes much smaller flux variations over the core as a whole than travel near the top or bottom of the core. The purpose of this section is to report in more detail the results of ETRC measurements taken to determine the effects of rod travel on fission rates at the midplane of the reactor.

All measurements were made with two uranium wires near the midpoint of each fuel element. (The standard deviation of measurements conducted in this manner has been determined to be 0.9 percent.) ETRC flux runs were made with rods 7-9-14 withdrawn approximately 1.5, 9.0, 13.8, 17.8, 22.0, 30.5, and 36 inches. Figure I-14 shows the flux variations caused by withdrawal of rods 7-9-14 from 1.5 inches to the remaining six different heights.

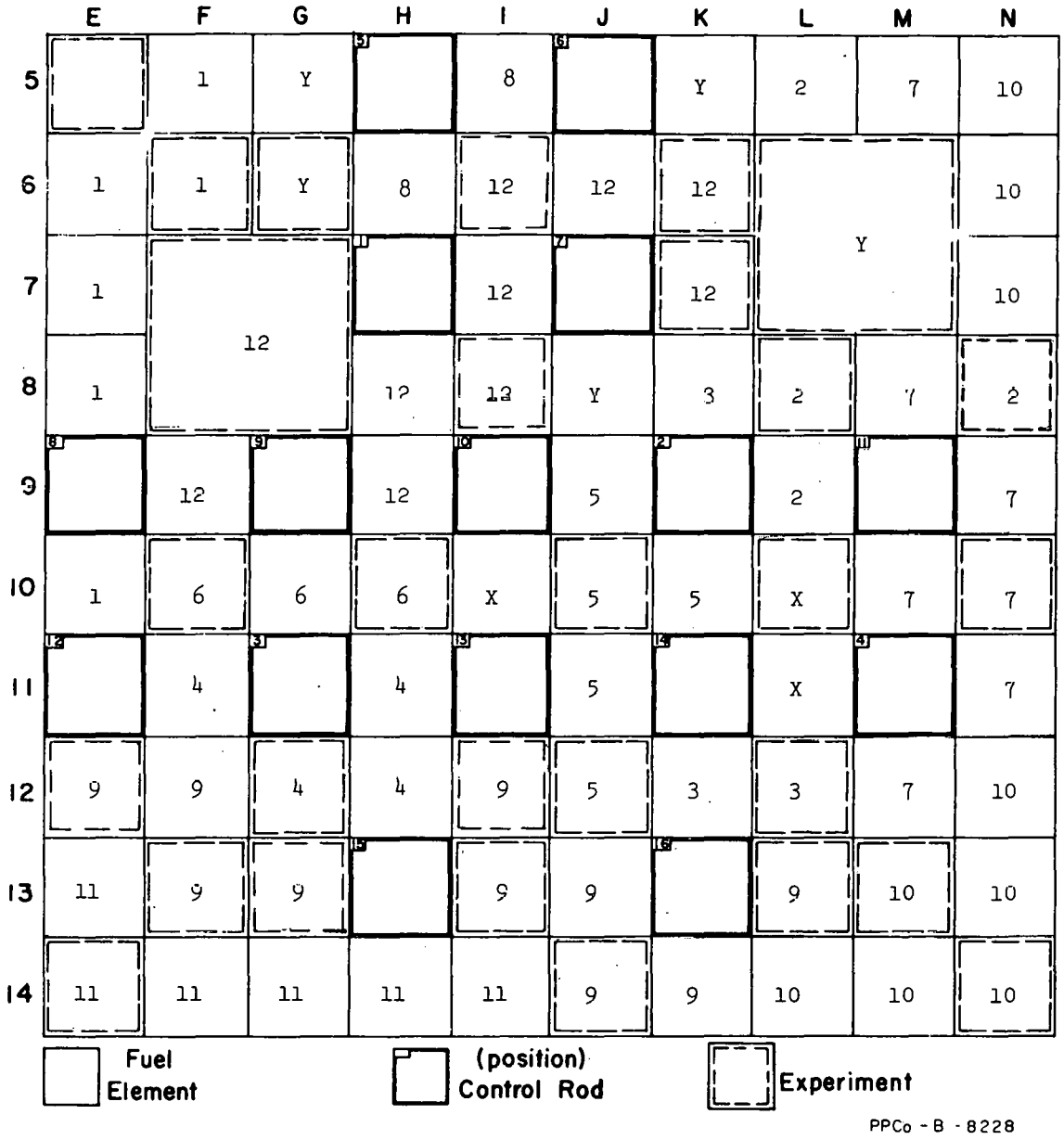
The fuel elements can be divided into groups which exhibit approximately the same flux variations. Figure I-15 shows these groups and Table I-3 shows the average flux variations within each group. The magnitude of the flux variations varies approximately as the group number. In positions located away from the reflector, the greatest flux variations occur next to rods 7 and 9. The smallest midplane variations are observed in positions next to the reflector in the NW corner of the core.



Entry	7-9-14 Rod Travel From	To
1st on left	1.5 in.	9.0 in.
2nd on left	1.5 in.	13.8 in.
3rd on left	1.5 in.	17.8 in.

Entry	7-9-14 Rod Travel From	To
1st on right	1.5 in.	22.0 in.
2nd on right	1.5 in.	30.5 in.
3rd on right	1.5 in.	36 in.

Fig. I-14 Percent variations in midplane fission rates due to withdrawal of rods 7-9-14 from 1.5 inches withdrawal.



The positions labeled "Y" have total flux variations of 6 to 10 percent as rods 7-9-14 are withdrawn from lower to upper limit but do not fall into any particular group.

The positions labeled "X" have small total flux variations (<4%) as rods 7-9-14 are withdrawn from lower to upper limit but do not fall into any particular group.

Fig. I-15 Cell groups, which show equivalent midplane flux or ENS variations due to withdrawal of rods 7-9-14.

TABLE I-3

AVERAGE FLUX VARIATION AT MIDPLANE IN VARIOUS GROUPS OF FUEL ELEMENTS AND EXPERIMENTS CAUSED BY WITHDRAWAL OF RODS 7, 9, AND 14 OVER SIX DIFFERENT VERTICAL REGIONS

Group Number	Range of Travel of Rods 7-9-14 (in.)					
	1.5 - 9.0	1.5 - 13.8	1.5 - 17.8	1.5 - 22.0	1.5 - 30.5	1.5 - 36.0
1	+0.9	+1.2	+2.7	+ 2.0	+ 1.3	+ 1.5
2	-0.7	-1.0	-1.1	- 2.3	- 3.3	- 3.3
3	0	0	+1.3	+ 4.7	+ 5.7	+ 5.0
4	-0.6	-1.2	-2.8	- 6.0	- 7.8	- 7.8
5	+1.5	+2.7	+4.2	+ 7.5	+ 9.5	+ 9.7
6	+1.7	+3.2	+5.7	+ 8.0	+10.3	+10.3
7	-1.8	-4.0	-6.9	- 9.9	-11.9	-13.4
8	+2.0	+4.5	+6.5	+10.5	+12.5	+14.0
9	-1.6	-3.6	-7.0	-12.0	-16.2	-14.9
10	-2.6	-5.3	-8.6	-13.6	-17.0	-18.4
11	-1.3	-5.0	-8.9	-16.2	-22.2	-20.0
12	+1.7	+5.2	+9.3	+17.5	+22.9	+22.8

These data show that the average flux variations at the midplane of the reactor are largest when rod movement occurs in this region. However, to minimize flux variations over the core as a whole, it will be necessary for the rod travel to occur near the midplane (see Reference 4).

1.5 ARMF Reactivity Measurement Investigation (D. W. Knight, E. F. Aber)

In a continuing effort to provide the best possible reactivity measurements for ARMF users, a series of measurements has been made investigating the effect on reactivity measurement efficiency of those items over which a reactor operator can exercise control. One of these items, reactor power, has previously been investigated and reported[5,6]. The present series of measurements deals with the effect of capsule worth, capsule withdrawal and insertion rate, and servo gain. The measurements indicated that all three items affect the reactor operating efficiency. However, the servo-gain setting used for the measurements was shown to be the most important. As a result of the measurements, procedures have been adopted at the ARMF which have improved the reactor operating efficiency.

The Advanced Reactivity Measurement Facilities, ARMF-I and -II, are low-power reactors designed to perform precise reactivity measurements. A servo-controlled regulating rod is the primary measuring scale of the ARMF. By means of a compensated ionization chamber and appropriate amplifiers, the regulating rod is controlled at a position which maintains the reactor critical and at a constant preset power level. The reactivity effect of a capsule is obtained from the regulating rod positions before and after a capsule insertion and a predetermined regulating rod calibration. The reactor operator inserts and removes the capsules from the reactor manually by using a long handling tool.

As previously stated, when a capsule is inserted or removed from the reactor, the regulating rod will seek a new equilibrium position. However, the new equilibrium or stable position is not achieved immediately. A period of time is required for the delayed neutron precursors throughout the core to come to a new equilibrium state as a result of the perturbation made on the core by the removal or insertion of a capsule. During this time the regulating rod is constantly changing position. When the reactor operator determines that the regulating rod position is stable, its position is measured. The position of the regulating rod is measured with and without the capsule in place. From a predetermined calibration curve the reactivity of the capsule may be determined.

To more efficiently use reactor operating time, it is desirable that the transient time or the time required for the regulating rod to reach its new equilibrium position be as small as possible.

Experience has shown that the reactor operator can increase considerably the percentage of actual measuring time during operation by a knowledge of the effects on operating efficiency of the following:

(1) The rate at which reactivity is inserted or removed from the reactor

(2) The size of the perturbation effect (reactivity worth of the capsule)

(3) The gain of the servo system.

Under most experimental conditions the reactor operator will have little, if any, control over item (2) -- the reactivity worth of capsules -- since the experimental program dictates the reactivity worth of capsules. However, a knowledge of the effect of capsule reactivity worth on operating efficiency may prove very useful. The reactor operator will be able to exercise a large measure of control over items (1) and (3) -- rate of reactivity insertion and gain of the servo system. In order to experimentally determine the effect of items (1), (2), and (3), two experiments have been completed.

The first experiment consisted of determining the transient time as a function of the reactivity worth of experimental capsules. Three capsules with different reactivity values were inserted in ARMF-II. The time required for the reactor to come into equilibrium was then determined from measurements made with the digital readout system. The range of times required for the regulating rod to reach equilibrium for the different capsules (five repeat measurements were made for each capsule) is presented in Table I-4. The time required for the regulating rod to reach its equilibrium position was determined visually from an inspection of an analog record of the regulating rod position. The spread in the data may possibly be explained by the fact that there may have been a difference in the rate the capsules were inserted.

Even though there is an overlap in the data, the data show a correlation between the reactivity worth of a sample and the transient time.

The second experiment consisted of determining the transient time as a function of withdrawal rate and of servo gain. The experimental equipment consisted of three pulleys of different diameters and a motor provided with a starting switch. The experiment was conducted by connecting a capsule to a

TABLE I-4

ARMF REGULATING ROD EQUILIBRIUM TIME AS FUNCTION OF CAPSULE REACTIVITY WORTH

Reactivity Worth of Capsule (10^{-6} $\Delta k/k$)	Time Required to Reach Equilibrium (sec)	
	Range	Average
428	100 - 150	125
290	80 - 120	100
130	60 - 100	80

pulley mounted on the motor shaft (this equipment was not available when the first series of experiments was conducted). At the time the motor was started, thus withdrawing the capsule, the digital readout system was started giving the regulating rod position as a function of time.

By using three pulleys, three different rates of capsule withdrawal were achieved. A run was made for each pulley using three different servo-gain settings, thus making a total of nine runs. Table I-5 presents the results of the experiments.

TABLE I-5

ARMF EQUILIBRIUM TIME AS FUNCTION OF CAPSULE WITHDRAWAL RATE AND SERVO GAIN

Pulley Diameter (in.)	Withdrawal Rate (in./sec)	Servo Gain Setting (volts/μamp)	Time to Equilibrium (sec)
0.25	3.5	316	140
0.25	3.5	100	80
0.25	3.5	10	>300
1.0	14.0	316	110
1.0	14.0	100	100
1.0	14.0	10	>300
3.0	41.9	316	140
3.0	41.9	100	130
3.0	41.9	10	>250

Table I-5 shows very clearly that the withdrawal rate and the servo-gain setting are very important. As a result of the experiments, reactivity measurements are now being made in approximately 15 percent less time than was previously required. The capsules are now inserted and withdrawn from the reactor

at about 3.5 inch/sec (at this rate the reactor stays in servo control at all times) and the servo gain is set at 100 volts/μamp. By knowing the relationship between reactor transient time and capsule reactivity worth, the reactor operators have been able to improve reactor operating efficiency. This is accomplished by not starting the measurement of the regulating rod position too soon or waiting too long.

Based on the experimental results, an engineering study is currently underway to determine if a mechanical means can be used to insert and remove capsules. By using this technique, capsules could be inserted or removed from the reactor at the most efficient rate at all times.

2. REFERENCES

1. J. L. Durney and N. C. Kaufman, Calculating Reactor Power from Activation Techniques as Applied to an Unusual Fuel Geometry (ATRC), IN-1047 (January 1967).

2. C. H. Hogg, Radiations Measurement Group (Private Communication).

3. E. F. Aber and E. E. Burdick, "Self-Shielding in Boron Carbide Particles", Nuclear Technology Branches Quarterly Report, October-December 31, 1966, IN-1083 (May 1967) pp 1-6.

4. R. J. Forrester, A. D. Mackley, E. E. Burdick, "ETRC Measurement of the Effects of Rod Travel on Power Distribution", Nuclear Technology Branches Quarterly Report, January-March, 1967, p. 6 IN-1117 (September 1967).

5. E. E. Burdick, E. Fast, D. W. Knight, The Advanced Reactivity Measurement Facilities, IDO-17005 (October 1964).

6. D. W. Knight, "ARMF-II Measurement Error as a Function of Operating Power Level", Nuclear Technology Branches Quarterly Report January 1-March 31, 1965, IDO-17104. pp 6-7.

II. NUCLEAR TECHNOLOGY
(R. G. Fluharty)

1. NUCLEAR CHEMISTRY
(R. P. Schuman)

1.1 Cross Section of Pm-147 (J. W. Codding, R. L. Tromp)

The fast-chopper cross section measurements on Pm-147 have continued, and additional resonance parameters have been obtained. The results obtained this quarter are given in Table II-1.

TABLE II-1

RESONANCE PARAMETERS FOR Pm-147

E_o (eV)	Γ (mV)	Γ_n (mV)	Γ_n^o (mV)
29.2	65.6	4.65	0.861
35.2	80.0	0.575	0.0969
38.0	90.0	40.0	6.49
45.5	97.0	33.4	4.95
48.0	100.0	31.8	4.59

1.2 Cross Section of Am-243 (J. R. Berreth)

A number of fast-chopper cross section measurements have been made on Am-243. Four samples of varying thicknesses have been made by pressing AmO_2 and Al powders. The data analyses have been started, and a number of new resonances have been seen.

1.3 Resonance Activation Integral Measurements (R. P. Schuman)

The activation cross sections of Am-243 to 10.1-hr Am-244 and to both isomers have been determined by irradiating Am-243 both inside a 1-mm thick Cd shield and unshielded. Gold and cobalt were used as flux monitors. The 10.1-hr Am-244 was determined by gamma counting, assuming abundances of 66.2 percent for the 744 keV γ and 27.6 percent for the 898 keV γ [1], the capture to both isomers by alpha counting Cm-244. The preliminary values of the resonance capture integral (including 1/v contribution) of Am-243 to 10.1-hr Am-244 is 111 barns, and to both isomers, 2160 barns. The thermal, subcadmium, activation cross section to 10.1-hr Am-244 is 5.9 barns.

The gamma counting of the 10.1-hr Am-244 with a calibrated germanium detector allowed the energies and the relative abundances of the gamma rays to be determined. Relative to 100 for the 744.1 keV γ, the photon abundances were 8 for the 99.3 keV, 27 for the 153.7 keV, 0.5 for the 205.4 keV, and 41 for the 898.2 keV.

The energies and the absolute intensities of the 2.1-day Np-238 gamma rays were determined by germanium gamma counting and alpha counting the Pu-238 decay product. The results are given in Table II-2.

TABLE II-2

ENERGIES AND ABUNDANCES OF SOME Np-238 GAMMA RAYS

Gamma Ray Energy, keV	Photons/100 disintegrations
883.3	0.8
924.2	2.4
985.5	23.0
1027.2	7.8
1029.7	18.0

2. REACTOR EXPERIMENTS

(E. Fast)

2.1 Relative Yields of I-135 in Fission (R. G. Nisle, I. E. Stepan)

In a previous report[2], the yields of I-135 of the common fissile isotopes were given relative to that of U-235. These were determined in a series of reactivity transient measurements of small samples in the ARMF-I and ARMF-II immediately following high flux irradiation in the MTR. Additional measurements have been made of U-233 because of adverse and questionable experimental conditions which existed during one of the previous measurements. The result obtained from three measurements which were acceptable experimentally is 0.825 ± 0.072, where the uncertainty quoted is the confidence interval at the 90 percent level. If the rejected value is retained, the result is 0.778 ± 0.117 which registers a marked increase in the uncertainty. The new value is higher than previously reported work (quoted in Reference 2) but is within the uncertainties given.

3. REFERENCES

1. R. G. Nisle and I. E. Stepan, "Relative Yields of I-135 in Fission", Nuclear Technology Branches Quarterly Report, July 1 - September 30, 1966, IN-1048, (1967) pp 56-58.

2. E. K. Hyde, J. Perlman, G. T. Seaborg, "The Nuclear Properties of the Heavy Elements", Detailed Radioactivity Properties, Vol. II, Prentice-Hall, New Jersey (1964).