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OPERATION OF THE ORNL GRAPHITE REACTOR AND THE LOW-INTENSITY TEST REACTOR – 1955

LITR FLUX TRAVERSES

A. F. Rupp
J. A. Cox

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OPERATION OF THE ORNL GRAPHITE REACTOR AND THE LOW-INTENSITY TEST REACTOR – 1955

LITR FLUX TRAVERSES

A. F. Rupp and J. A. Cox

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OPERATION OF THE ORNL GRAPHITE REACTOR AND THE LOW-INTENSITY TEST REACTOR – 1955

ORNL GRAPHITE REACTOR

Summary

The ORNL Graphite Reactor operated very well during 1955. The downtime was low, only 8.6%.

The fuel in the bonded slugs did not perform as well in 1955 as in 1954. Much of the trouble was undoubtedly due to growth of slugs which were not beta-transformed. It is known that some slugs had grown over ½ in.

The automatic control system installed in 1954 continued to operate satisfactorily. The cooling system gave minor trouble when one of the 900-hp fan motors had to be replaced because of shorts in the rotor.

The high radiation in the canal was the largest source of trouble. Approximately 55 tons of slugs discharged from the reactor in 1952 was sent to the Metal Recovery Plant. Enough slugs had ruptured, due to their long exposure in the canal and reactor, to badly contaminate of water. Most of the contamination was removed by the end of the year, but the radioactivity which had soaked into the canal walls was enough to give high radiation fields. A solution to this problem was being sought at the end of the year.

A study is under way on the possibility of increasing the flux of the ORNL Graphite Reactor by reloading with enriched uranium fuel. Results of the study appear to be promising.

Downtime Analysis

An analysis of the downtime for the ORNL Graphite Reactor is presented in Table 1.

Performance of Reactor Fuel

The performance of the bonded-slug fuel was not so satisfactory in 1955 as in 1954; 19 ruptured slugs were located, compared with only 4 in 1954. The number and type of slugs that have ruptured since 1944 are given in Table 2. A 1954–1955 inventory of the slugs in the ORNL Graphite Reactor is given in Table 3.

Approximately 42,000 bonded slugs are in the reactor, most of which were charged in 1952. All are silicon bonded and 42 bonded slugs have ruptured since they were charged in mid-1952. Including slugs discharged along with ruptures, approximately 45,000 of the bonded slugs have been charged to the reactor with a rupture frequency of about 0.1%.

During the fabrication of the bonded slugs a beta heat treatment was used to reduce growth of the uranium in the reactor, which had occurred because

TABLE 1. ANALYSIS OF ORNL GRAPHITE REACTOR DOWNTIME

	Number	Duration (hr)
Scheduled Shutdowns		
Regular	52	615.7
Special for research	20	62.2
Special for operations	17	56.2
Subtotal	89	734.1
Unscheduled Shutdowns		
Operations	3	2.6
Instrument Department	2	0.6
Research	3	0.8
Maintenance	2	0.7
Equipment failures		
Instrument	17	7.4
Others	10	7.2
Subtotal	37	19.3
Total	126	753.4

[redacted]

TABLE 2. SLUG RUPTURE SINCE 1944

Year	Number of Ruptured Slugs
Bonded	
1955	19
1954	4
1953	14
1952	5
Unbonded	
1953	4
1952*	8
1951	15
1950	13
1949	14
1948	14
1947	13
1946	6
1945	14
1944	4

*Most of the reactor was loaded with bonded slugs in 1952.

TABLE 3. INVENTORY OF FUEL FOR GRAPHITE REACTOR

	Number of Slugs	
	12/31/55	12/31/54
In Reactor		
Bonded slugs	41,844	41,900
Donut slugs	21	21
Experimental bonded slugs (hole 1866)	15	15
In Vault		
Donut bonded slugs	35	35
Bonded slugs	8,322	9,765
In Canal		
Bonded slugs	681	~1,400
Unbonded	3	~43,400

of thermal cycling. Faulty equipment, discovered after part of the slugs had been charged to the reactor, resulted in a considerable number of nontransformed slugs being in the reactor. For the past several years, measurements have been made by the Metallurgy Division on the growth of slugs which were discharged.[1] Also, the slug rows still in the reactor have been measured in order to follow the increase in slug length. All data obtained indicate that the beta-transformed slugs grow very slowly. The average growth rate has been less than 0.010 in. per year.

In the nontransformed slugs the average growth rate has been about 0.070 in. per year; some slugs have grown more than 0.5 in. (original slug length, 4.123 in.) in less than two and one half years. Of 92 slugs measured, 55 grew more than 0.1 in. after more than two years in the reactor and 11 grew more than 0.3 in. Individually, the growth rate of the partially transformed slugs was erratic, but slug rows in the reactor showed a growth rate almost as high as that of the nontransformed rows.

The rupture rate increased markedly in 1955, and it is assumed that the growth of the slugs has accounted for at least part of the increase. In thermal cycling tests, two of four slugs growing more than 0.4 in. ruptured.

Of the 19 ruptures in 1955, 11 were from slug lots which were nontransformed or only partially transformed, and individual slug lots have shown high rupture rates. The number of ruptures of partially transformed and nontransformed slugs, by lots, is compared with that of the transformed slugs in Table 4, which indicates a high rupture frequency among certain lots of untransformed or partially transformed slugs. Data on the bonded slug ruptures that occurred in 1955 are given in Table 5.

Further measurements on undamaged slugs discharged along with ruptured slugs have been discontinued, since the rate of growth of the different kinds of slugs appears to be established.

In July a slug rupture was located by means of the temperature plots kept on all thermocouple slugs. The temperature of this slug had risen, for about a month previously, out of proportion to that of other thermocouple slugs. The ruptured slug was actually three slugs past the thermocouple slug. This suggests that another method should be used for detecting ruptures in rows containing thermocouples if the plugs cannot be removed for visual inspection. At present, these channels are monitored each week by sucking an air sample from each thermocouple hole through an ion chamber

[1] R. E. Adams, *Summary of Work Done to Observe the Behavior of the Silicon Bonded Slugs in the X-10 Graphite Reactor*, ORNL CF-55-12-8 (Dec. 2, 1955).

TABLE 4. COMPARISON OF NONTRANSFORMED AND PARTIALLY TRANSFORMED SLUG RUPTURES WITH TRANSFORMED SLUG RUPTURES

Nontransformed and Partially Transformed Slugs		Beta-Transformed Slugs	
Lot No.	Number of Ruptures	Lot No.	Number of Ruptures
110	5	127	2
113	4	129	3
114	1	130	3
115	5	135	1
116	1	138	2
118	1	136	1
119	1	6137	1
154	3	145	1
155	1	149	1
		168	1
		169	1
		6206	1
		6213	2

while the reactor is operating at 50 kw. This procedure, called "scanning," is somewhat objectionable because of radiation from the sampling tube, which must be pulled out of the scanner hole while the reactor is operating.

Several ruptured slugs in thermocouple rows have now been located by the temperature plots, which were started when it was noted that the temperature of other slugs changed rapidly in a channel in which a slug was beginning to rupture.

Control System

Several times during the past year one of the hydraulic shim rods failed to operate during a scram, but in no case did both rods fail to operate. Numerous examinations of the rods were made, and, although the exact reason for failure has not been determined, the trouble appeared to be in the solenoid valves.

The automatic controller was installed permanently in the panel board in March, and the power supply and the wiring to the chamber in hole 40 were reworked.

The controller gave some trouble during May because of a faulty relay. It proved necessary to obtain a special relay with proper characteristics in order to repair the instrument.

The blockout system was revised and a new panel built so that only certain combinations of safety devices could be blocked out at a time. With the old panel, it was possible to block out any number of safety devices. A number of scram switches were installed at various locations around the ORNL Graphite Reactor so that it could be scrammed from each floor.

A program for repairing and replacing the old control wiring of the reactor, made necessary by the long period that the wiring had been in use, was started in August and was still in progress at the end of the year. The wiring diagram is being checked and brought up to date.

Cooling System

Capacitors and lightning arresters were installed on each of the two 900-hp motors, and no trouble from lightning occurred during the year. Two 900-hp motors are now on hand as spares, one which was purchased and one which was repaired. Since the motor needing repairs could not be sent to the factory for overhaul without removal of the stator coils and the rotor, which cannot be completely decontaminated, it was decided that it would be cheaper to have the repair work done at K-25 and to buy a new motor.

Fan motor No. 2 was changed on August 1; installation was completed on the evening of August 3. Disassembly of the motor that was removed showed that two copper bus bars in the rotor had broken. Considerable arcing had occurred, and the metal was fused at one of the breaks.

The south mattress plate was changed November 7.

TABLE 5. GRAPHITE REACTOR BONDED SLUG RUPTURES IN 1955

Lot No.	Rupture No.	Row No.	Number from West End of Row	Date	Days in Reactor	Temperature (°C)	Beta-Transfarmed (%)	Remarks
154	130	1576	16	2/22	1016	175	75	Average growth of 15 slugs discharged with the rupture, 0.050 in.; indicated by probe
136	131	2276	2	3/19	1055	136	100	11 slugs discharged with the rupture found to have an average increase in length af 0.011 in.; maximum, 0.046 in.; minimum, 0.003 in.; indicated by probe
155	132	2061	12	3/21	1043	210	66	13 slugs measured for increase in length; average, 0.037 in.; maximum, 0.128 in.; minimum, 0.004 in.; found by inspection
129	133	1561	22	3/28	1064	205	100	Measurements not been completed; found by inspection
154	134	2060	12	5/31	1114	200	75	Probes slightly higher prior to rupture
206	135	1878	30	6/13	1014	185	100	Almost completely oxidized; found by visual inspection
110	136	1569	19	6/15	1170	210	0	Indicated by probes
115	137	1873	13	6/20	1169	200	27	Found by visual inspection
115	138	1663	20	6/20	1169	245	27	Temperature rise of thermacouple slug, 19th from west side
116	139	1874	20	7/18	1197	204	57	Found by visual inspection
110	140	1672	12	7/24	1209	272	0	Indicated by thermocouple on 15th slug from west end
113	141	1772	11	7/30	1215	215	0	Indicated by probes
115	142	1472	27	8/8	1218	220–266	27	Thermocouple row; indicated by high temperatures; 19th slug was thermocouple slug
168	143	0964	29	9/6	1191	159	100	Found by visual inspection
169	144	0974	14	9/12	1197	159	100	Found by visual inspection
119	145	2274	21	9/19	1253	215	77.8	Found by visual inspection
115	146	1873	4	10/10	281	200	27	Located by visual inspection
130	147	1164	27	10/17	1267	220	100	Located by visual inspection
129	148	2667	23	12/5	1323	180	100	Found by visual inspection

Filters

No changes of filters occurred in 1955. The filter pressure-drop vs time curves are shown in Fig. 1. These curves extend over about six years' operation. The last filter changes occurred as follows:

Cell No.	Date Changed	
	Glass Wool Filter	Paper Filter
1	5/8/50	5/25/51
2	4/9/53	5/25/51
3	6/7/54	5/10/54
4	8/16/54	7/6/54

A comparison of the filter house pressure drops with those of a year ago and with clean filters is given in Table 6.

Inlet filters were changed in 1955 as follows: first cell, 3/28 and 4/4; second cell, 10/10 and 10/17; third cell, 12/12 and 12/27. These filters were changed more frequently than normal because of a high dust load from the ORR excavation.

Canal

Several changes have been made in the anion columns in the canal that have been converted from the old sand filters, and so far no acceptable revision has been found which will permit the columns to be properly rinsed after regeneration in a reasonable amount of time. Because of other projects, not much time could be devoted to this job, but it is hoped that it will soon be possible to try a gravel bed for a resin support.

Painting of the canal walls and walkways was completed in January, and housekeeping in the canal has been much easier, since contamination, which previously soaked into the floor or wall, can now be washed off.

An electromagnet was built from discarded parts of LITR magnets to pick up pieces of radioactive cobalt which had fallen on the canal floor. This permitted the radioactivity in the section west of

TABLE 6. PRESSURE-DROP DATA

	Pressure Drop (in. water gage)		
	Glass Wool	CWS No. 6	Total Across Filter House
12/31/55	2.3	1.6	5.3
12/31/54	1.6	1.3	4.3
Clean filters	1.1	1.3	3.3

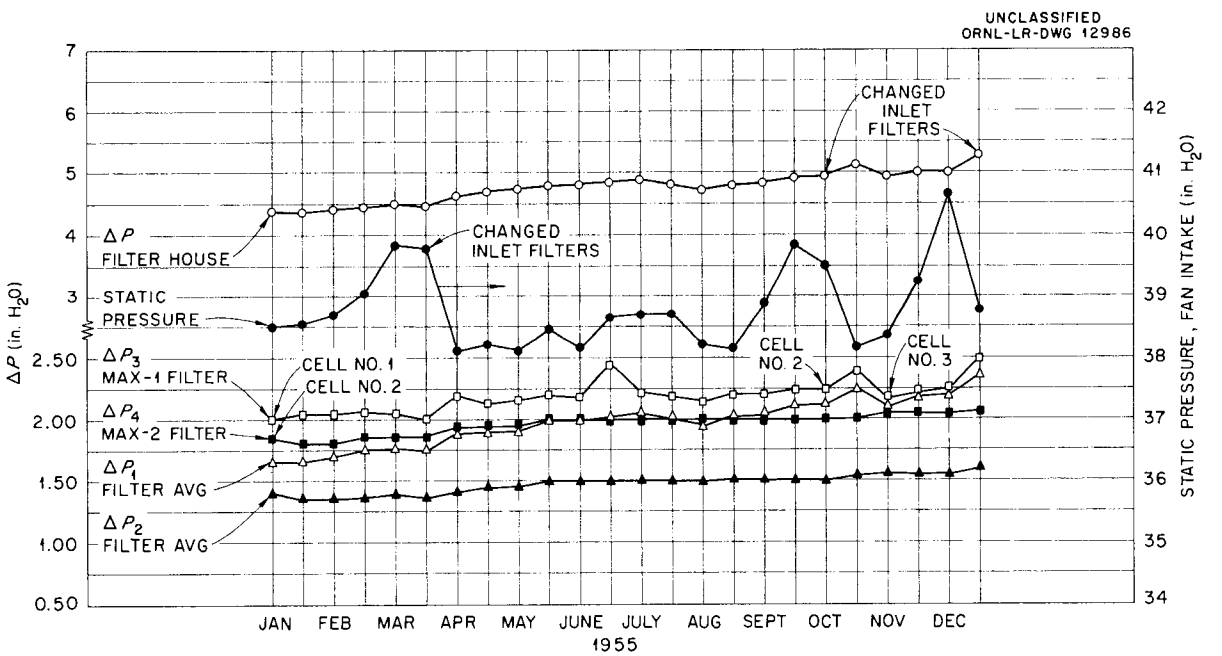

Fig. 1. Filter Pressure-Drop vs Time Curves.

the canal dam to be reduced from about 10,000 to approximately 4,000 counts/min/ml.

Radiation in the canal apparently comes mostly from the canal walls, which have soaked up considerable amounts of radioactivity from the water.

In order to test the amount of radioactivity that had soaked into the concrete of the canal wall, three consecutive layers, each $\frac{1}{8}$ in. thick, were chipped off the canal wall. Comparative gamma measurements on the chips gave the following results: first $\frac{1}{8}$-in. layer, 200 mr/hr; second $\frac{1}{8}$-in. layer, 20 mr/hr; third $\frac{1}{8}$-in. layer, 2 mr/hr. It was thought that it might be feasible to chip away about $\frac{3}{8}$ in. of the canal wall and replace with fresh concrete, which could be painted or coated to prevent water soaking into the concrete. However, studies of the working time available when the water is drained from the canal indicated that this procedure would be impractical because the radiation would be several roentgens per hour in the working area, limiting the working time to only a few minutes.

During the approximately three years that the last discharged graphite fuel loading had been stored in the canal, it was impossible to clean the canal floor of silt. In loading the slugs for shipment to metal recovery, the silt was stirred up, causing the water to become very murky. In order to continue loading, it became necessary to start purging the canal at a rate of about 500,000 gal per week. This also had the beneficial effect of removing large amounts of radioactivity from the canal. Radioactivity in the canal water was found to be approximately one-half Cs^{137} and one-half Co^{60}. The cesium was released from ruptured slugs which had corroded during their three-year storage in the canal and in many cases had become completely oxidized. The Co^{60} was released from pieces of radioactive cobalt which had been stored in the canal and had dropped onto the canal floor.

As the slugs were removed from the canal and the floor became available for cleanup work, the slit and the uranium oxide were pumped out to the settling basin, and the relative amounts of cesium and cobalt changed appreciably. At the end of the year, when the cleanup of the canal was almost complete, the cobalt accounted for approximately 79% of the activity in the canal water and the cesium for only about 15%. Examples of analyses are shown in Table 7.

The total radioactivity in the canal water changed markedly as the cleanup program progressed. Before the program was started, the radioactivity had varied between 4000 and 5000 counts/min/ml (taken with a well-type gamma scintillation counter). At the height of the program the activity in the water was as great as 8000 counts/min/ml, owing largely, it is believed, to increased fission products stirred up with the oxide from ruptured slugs. As the oxide and silt were pumped out, the activity decreased until, at the end of the year, the total activity in the water was only about 500 counts/min/ml, most of which was Co^{60}.

Even though the radioactivity in the canal water had been decreased and the slugs had been removed, the radiation in the working areas was not reduced appreciably, because the canal walls had soaked up radioactivity from the water. Several methods of painting the canal walls were under investigation at the end of the year, but the necessity of storing large quantities of irradiated thorium forced postponement of this program. In order to reduce radiation in the working areas, it is now planned to make lead or iron shields which can be

TABLE 7. CANAL-WATER ACTIVITY

	January 1955		December 1955	
	Activity		Activity	
	Counts/min/ml	Per Cent	Counts/min/ml	Per Cent
Gross gamma	4000*	100	572	100
Cs^{137} gamma		50	83	~15
Co^{60} gamma		50	450	~79
Gross beta			170	

*Counted in a well-type gamma scintillation counter.

hung along the canal walls and will extend about 1 ft under water. These lead shields will cover the walls for about 10 ft on either side of the canal adjacent to the area where most of the work is done. Listed below are background radiation readings taken with a paper shell ion chamber:

	Activity (mr/hr)	
	January 1955	December 1955
Average in walkway	32	27
Over water at walls	150	125

The uranium slugs stored in the canal since discharge from the reactor in 1952 were found to be rather badly corroded. Approximately 2% of the slugs were badly ruptured and oxidized. The oxide was pumped out to the settling basin.

New Facilities

A new shield was installed at the vertical irradiation facility at hole 10. Hole 12, previously occupied by this facility, was turned over to the Solid State Division for low-temperature work.

Miscellaneous Service Work

An irradiated thorium slug was sliced into twenty-one $\frac{1}{8}$-in. wafers for North American Aviation, Inc., in March. This was done with a special abrasive cutting wheel installed under approximately 6 ft of water in the canal.

In August, an irradiated bonded slug was cut into 15 slices for the Chemical Technology Division.

Five slices were cut from an irradiated uranium slug for the Chemical Technology Division in October.

Experiments

The total neutron emission from three Sb-Be neutron sources was measured with the use of an $MnSO_4$ solution with the source suspended in the center of the bath. The activation of the manganese was measured and from this the neutron emission was estimated after allowance was made for neutron leakage from the bath. All three sources gave approximately 6×10^{16} neutrons/sec at time of discharge from the Graphite Reactor. One of the sources was shipped to the Bureau of Standards for calibration, and their measurement was approximately 6% lower than the measurement at ORNL.

Channel 2568 was discharged early in October and a boron-coated thermopile installed a few inches away from hole 12. This permitted the reactor to be operated from a galvanometer attached to the thermopile so that the neutron flux in hole 12 could be held constant during special experiments. In normal operation, the reactor is controlled from a neutron chamber in hole 40, and this results in the neutron flux in other parts of the reactor varying somewhat as the control rods are moved.

Increasing the Flux of the Graphite Reactor

At the end of the year a study was being made on the possibility of increasing the neutron flux in the Graphite Reactor by replacing the natural uranium slugs with enriched uranium fuel. Results thus far have been encouraging, and it appears feasible to increase the power by a factor of 2 to 3 and the neutron flux by a factor of 10 to 15.

Estimates of the amount of enriched fuel needed in reloading and operating the reactor for a year are in the order of 40 to 60 kg based upon a loading shaped approximately like a cylinder, about $16\frac{1}{2}$ ft in diameter and 16 ft long.

Some preliminary experiments on a mocked-up graphite channel 8 ft long have been made with several proposed fuel element shapes. The shape which appears best at this time is one formed from an MTR-type plate rolled into a cylinder $1\frac{1}{4}$ in. in diameter and 2 ft long. This is favored over a fin-type element because of simplicity, a better Al-U ratio, and ease of unloading. The inside diameter ($1\frac{3}{8}$ in.) of the bridge tubes through which the fuel must be inserted does not allow enough area for a finned element (with enough area to be superior to the cylindrical type) to be passed through. Also, it appears that the cylindrical element can be unloaded from the east face, which would permit the elements to be rearranged. The ones in the center having the most burnup could be removed and those on the ends replaced so that quite uniform burnup could be achieved.

The possibility of cooling the air with water sprays in the west plenum chamber to reduce its volume and thus reduce the pressure loss in the ducts is under study. When the reactor is operating at higher power, the cooling air will be heated to a higher temperature than formerly and thus have a larger volume. Enlargement of the ducts is also being studied, since approximately two-thirds of the pressure drop is caused by the inlet and exit ducts.

The power at which the enriched loading can be operated will be limited by the temperature of the fuel. It is believed that the fuel can be operated at a maximum temperature of 350°C, and, based upon the heat transfer obtained with natural fuel, an initial power of about 6 Mw can be obtained with the present cooling system. With modifications to the ducts it may be possible to operate at 10 Mw.

Research Reactor Personnel Training

Personnel training for the Research Reactor was begun in 1955. All the supervisors expected to be directly associated with operation have been sent to the MTR for a week to study operation at high power levels, and four additional operators have been hired so that they will be adequately trained when initial operation of the ORR begins.

LOW-INTENSITY TEST REACTOR (LITR)

Summary

The LITR operated very satisfactorily in 1955. No major troubles developed, and the usage for research increased steadily. More facilities were installed for radioisotope production, and at the end of the year a project was under way to make a smaller grid assembly, thus exposing more of the lattice for experiments. Another project was under way to provide a third vertical facility which will not have to be removed each time the top plug is taken off. The cycle time for major shutdowns was increased to three weeks, which will cause less interference with research and operations.

The downtime was high because of over 1000 hr of scheduled and 150 hr of unscheduled shutdowns for research. The utility and versatility of the LITR were well demonstrated by the 149 special scheduled shutdowns for research which enabled experimenters to obtain important data.

A test of remote control of the LITR from the Graphite Reactor control room was begun in October and was still under way at the end of the year. The results appear very promising to date.

A large number of scrams, apparently originating in the period amplifier or period system, was one of the worst troubles encountered. However, no single cause for the spurious scrams could be found, and they continued to occur at varying frequencies throughout the year.

Improvements continued in the demineralizer system, and the amount of Np^{239} in the water decreased by a factor of about 6. This appears to indicate that the Np^{239} was being formed from natural uranium contamination in the reactor and that the source has been partly removed.

Downtime Analysis

An analysis of the shutdowns which occurred at the LITR during 1955 is given in Table 8.

Fuel Usage in the LITR

In 1955, 12 fuel elements were added to the LITR core and 11 were removed. The increase in the lattice was for the purpose of increasing the flux in hole HB-4. The use of 195- to 200-g elements has reduced the frequency at which fuel must be added from once every two weeks to once a month. At the end of the year the lattice contained 3354 g of U^{235}.

Table 9 shows the initial weights of U^{235} in elements added and the depleted weights of U^{235} in the elements removed. The burnup, calculated at 1.26 g/Mwd, includes the fission loss and the conversion to U^{236}. In most elements removed, the burnup was the usual 35%, but on the shim rod the burnup was 55%. A discussion of the method of calculating burnup is shown in Appendix B.

Experimental Usage of LITR

The utility of the LITR was demonstrated by the large number of shutdowns for the various research projects. During 1955 there were 149 scheduled shutdowns for research. In a reactor having a higher xenon growth after shutdown, it would have been impossible to shut down so often, since it would have been necessary to change some of the fuel or wait several days before starting up again. Practically all the shutdowns for research were for the purpose of taking data of the type requiring that the reactor power be reduced to a low level for several hours. Such data could not have been taken in a less flexible reactor.

During 1955, 100 experiments and target exposures were conducted in the 13 research facilities in the LITR. In addition, approximately 800 target exposures were made for research and for the radioisotope production program in the pneumatic tubes and core exposure facilities.

TABLE 8. ANALYSIS OF LITR DOWNTIME

	Number	Duration (hr)
Scheduled Shutdowns		
Regular	39	471.3
Special for research	149	1032.8
Special for operations	3	8.8
Subtotal		1512.9
Unscheduled Shutdowns		
Operations	1	0.1
Instrument Department	6	8.9
Janitorial (accidental)	1	0.5
Research	4	5.0
Instrument failures		
Research	30	133.8
Operations	58	22.4
Other equipment failure		
Research	10	16.8
Operations	8	16.6
Subtotal		204.1
Total		1717.0

The major troubles encountered during the year were malfunction of instruments, fission gas release from experiments, and radioactive dust encountered when experiments were inserted in the beam holes or were removed.

A short discussion of each experiment facility is given in the following sections, and a summary of the shutdowns is given in Table 10. In some cases the shutdowns coincide, so that the true down time of the reactor is not indicated. The intention is to indicate the shutdowns required for each facility; the trouble shutdowns do not coincide.

Horizontal Beam Hole HB-1. – The HB-1 facility, which contains a chopper-type neutron velocity selector, was not used beyond the first part of the year because of personnel changes. Another factor which interfered with its operation has been the work done at HB-2, which necessitated covering the HB-1 instruments for about two months to protect them from dust. By the end of the year new personnel had been obtained to operate the facility, and work to reactivate it had begun. Total usage time for the facility was less than 60 days.

Horizontal Beam Hole HB-2. – This facility was used for special classified work until June 14; installation of an HRP fuel loop occupied the balance of the year. The total usage time was 83 days.

TABLE 9. FUEL USAGE IN THE LITR IN 1955

Fuel Piece	Weight of U^{235} (g)		
	Original	Depleted	Burnup
Fuel Elements Removed			
L-20	168.250	107.531	60.719
L-17	168.030	108.015	60.015
L-37	168.250	102.870	65.380
L-38	168.250	104.948	63.302
L-36	168.280	106.755	61.525
L-34	168.260	100.904	67.356
L-30	168.140	100.269	67.871
L-47	199.720	128.761	70.959
L-41	168.711	94.565	74.146
G-6*	165.550	155.224	10.326
Total	1711.441	1109.842	601.599
Av	171.144	110.984	60.160
Shim rod L-29-S	130.622	59.359	71.263
Fuel Elements Added			
L-59	195.550		
L-65	200.460		
L-64	200.460		
L-60	195.940		
L-58	195.550		
L-57	194.990		
L-56	194.990		
L-55	194.990		
L-54	194.480		
L-53	199.790		
G-6*	165.550		
Total	2132.750		
Av	193.886		
Shim rod L-23-S	131.580		

*An experimental element for the Geneva Reactor.

Horizontal Beam Hole HB-3. – A total of eight radiation damage and stress corrosion tests were conducted, and one flux calibration was made in HB-3 with little trouble. Only one reactor setback, which was of short duration, resulted from the experiments, and that was due to instrument failure. The total usage time was 212 days.

Horizontal Beam Hole HB-4. – Five HRP loop runs and one flux measurement were made during the year. During the loop operations 31 planned

TABLE 10. ANALYSIS OF LITR SHUTDOWNS

Facility	Number of Experiments	Days in Use	Per Cent of Time Used	Special Planned Shutdowns	Average Length of Planned Shutdowns (hr)	Trouble Shutdowns	Average Length of Trouble Shutdowns (hr)	Short Setbacks
HB-1	1	<60	<16.4	0	0.0	0	0.0	0
HB-2	4	83	22.7	3	3.1	4	25.6	2
HB-3	9	212	58.1	0	0.0	0	0.0	1
HB-4	6	220	60.3	31	12.2	13	23.0	30
HB-5	7	137	37.5	28	3.0	10	3.9	1
HB-6	10	253	69.3	37	3.3	4	3.3	0
C-28	6	250	68.5					
C-38	2	55	15.1					
C-42	6	173	47.4	1	5.7	1	0.4	1
C-44	28	180	49.3	15	0.8	5	0.8	1
C-46	6	52	14.2	0	0.0	0	0.0	0
C-48	10	101	27.7	1	4.0	6	12.4	1
V-2	5	13	3.6	0	0.0	3	1.2	0

shutdowns, totaling 377.7 hr, 13 trouble shutdowns, totaling 298.9 hr, and 30 short reactor setbacks were experienced. Most of the setbacks and trouble shutdowns were due to instrument failures. Construction of the pressurizer line to one of the loops caused considerable trouble; 265 hr of reactor downtime was required between October 24 and November 3 in attempts to alleviate the difficulty.

After a number of air contamination incidents occurred, the external shielding was partially removed and the drain system inspected. The hot drain installation was found to be incomplete, some openings being covered only with pieces of plywood.

The total usage time was 220 days.

Horizontal Beam Hole HB-5. – Six rocking-bomb HRP fuel corrosion tests and one flux measurement were made during the year. Twenty-eight planned shutdowns, totaling 84.3 hr, ten trouble shutdowns, totaling 38.9 hr, and one setback occurred during this time. The trouble shutdowns were due primarily to bad instruments, thermocouple failures, and water leaks from the coolant lines. The setback was due to instrument trouble. One shutdown, however, was caused by actual trouble with an experiment when the test capsule ruptured.

In September it was found that some of the required safety interlocks had been removed from the instrumentation. They were restored before further operation of the facility was permitted. The total usage time was 137 days.

On three occasions serious air contamination in the LITR West Experiment room occurred when the capsules in HB-5 and HB-6 were bled to the small hold tank in use at the time. A hold tank with a large off-gas vent is to be installed to prevent a recurrence of such air contamination.

Horizontal Beam Hole HB-6. – Nine rocking-bomb HRP fuel corrosion studies and one flux measurement were made; there were 37 planned shutdowns, totaling 122.6 hr, and 4 trouble shutdowns, totaling 13.3 hr. Three of the trouble shutdowns were due to instrument failures, and the other one was due to heater failure. The total usage time was 253 days.

Core Position C-42. – Five experiments were conducted in core position C-42 by the General Electric Company to determine the amount of fission product released from a predamaged fuel element. A neutron flux measurement was done in December. The facility was in use for a total of 173 days. The period from July until December was spent in revising the external components.

One planned shutdown (5.7 hr), one trouble shutdown (0.4 hr), and one short setback occurred.

Core Position C-44. – The Chemical Technology Division conducted 28 experiments in core position C-44. Eight were cross-section determinations, and twenty were pressurized capsule exposures of uranium solutions or thorium slurries. Fifteen planned shutdowns (totaling 12.7 hr), five trouble shutdowns (totaling 4.1 hr), and one short setback occurred. There was one special shutdown of 164 hr for cutting a side access hole for the C-44 position. The total operating time for the facility was 180 days.

The principal troubles during operation were due to malfunction of instrumentation and leaks in the capillary pressure tube connections.

Core Position C-46. – Very little usage was made of this facility for vertical loops; four radiation damage tests were done by the Solid State Division. The total time that the facility was in use was 52 days.

Core Position C-48. – This facility was used intermittently for special classified work during most of the year. Between July 26 and October 25, five miscellaneous samples were exposed by different groups. On October 25 the facility was assigned to a ceramics testing group, and two samples were run before the end of the year. One of the samples began releasing fission gases after startup and required an 0.8-hr shutdown for removal.

The total time that C-48 was in use was 101 days.

Low-Flux Slant Hole V-2. – V-2 is a low-flux neutron facility normally assigned to the General Electric Company for miscellaneous materials testing not requiring reactor downtime for insertion and removal. In addition, this facility was used by the Health Physics Division for operating five low-temperature radiation-damage studies, two of which were reinsertions of the same test following successive repairs. Three trouble shutdowns, totaling 3.5 hr, were caused by the low-temperature work.

Special Irradiation Facilities. – A radiation-damage test of gaskets was made in an aluminum can set upon the stacked beryllium, with pressure leads coming out through the south manhole cover in the top plug. The facility was operated for 42 days.

Six miscellaneous samples were exposed in the hollow fuel element in position C-28 for a total usage time of 250 days. In September, this facility was released to the Operations Division by the Solid State Division.

Two samples were exposed by the Solid State Division in the hollow fuel element in position C-38 for a total usage time of 50 days.

Exposure of Small Samples in the LITR

The pneumatic tubes and lattice stringers have been used at an increasing rate. From 1952 through 1955 there were 1029 irradiations for the radioisotope program and 798 irradiations for the research program, making a total of 1827.

During 1955, irradiations totaled approximately 800, or 43% of the total for the past four years. The rate of use of irradiation facilities has also increased markedly.

New Facilities

In March it became necessary to install a third magnesium stringer, holding approximately 70 samples, in the LITR core because the two other stringers were almost completely full. A facility for irradiating samples up to about $1\frac{1}{8}$ in. in diameter was installed in the unused regulating rod hole on the west side of the lattice. This will be convenient for irradiating such items as Sb-Be neutron sources. The neutron flux measurement indicated a maximum flux of about 2×10^{13} neutrons/cm^2·sec.

An additional hole was cut in the side of tank A at the LITR so that a flexible tube could be connected to an experiment in the lattice similar to the ones in the lattice positions C-42 and C-48. This will permit considerably more flexible operation of the experiment.

Design work is proceeding on a small upper grid assembly for the LITR which will leave the fourth and fifth rows of the lattice free for experiments. The usefulness of the lattice will thus be increased considerably, 18 lattice positions being accessible instead of the present 4. Also, experiments can be left in the lattice instead of having to be removed whenever the grid is removed.

Local Control System

A very compact drive mechanism was designed and was installed on the fission chamber, permitting the chamber to be moved close to the reactor tank at startup and then withdrawn to prevent burnout of the chamber.

An interesting calibration of the control rods was made by using the growth of xenon after shutdown to calibrate shim rod No. 2. The reactor was maintained at a very low power, and the change in rod position was related to the calculated xenon growth curve. Based on this calibration, rods 2 and 3 were found to be approximately equal in effectiveness, while rod 1 was approximately one-half as effective as either of the other two. These data were obtained in order to check the effect of inserting approximately 130 g of U^{235} in lattice position C-48 (adjacent to the fuel). The maximum effect on reactivity was approximately 0.14% $\Delta k/k$, and the greatest effect was observed when the material was approximately 4 in. above the center line of the lattice.

An unusual failure in the control system occurred on May 17 after a setback. While the power was at N_L (1% of full power), the reactor power suddenly began to rise. The trouble was finally traced to a faulty selsyn transmitter which had acted as a motor to drive the log potentiometer upscale, causing the regulating rod to be withdrawn. In an effort to reduce the scrams in the control circuits the period scram circuit was modified so that it would be less sensitive above N_L. After this was done, on October 21, there were fewer scrams.

Remote Control System

The remote control test equipment was put into operation in October, with all operations being done from the remote control circuit in the Graphite Reactor control room, except during startup or periods when the Graphite Reactor was shut down. It has been the practice for an operator to remain in the LITR control room at all times to observe operation and to record any incidents which require his participation; however, it has never been necessary for him to take over the controls or do anything that could not have been done on a routine patrol basis. Therefore it is believed that the LITR can be safely and economically operated by remote control. In some respects, operations should be even safer, since much more information will be available to supervision from a central control room than from separate control rooms (see Appendix A for list of instruments and drawing numbers for the LITR remote control board).

LITR Demineralizer System

Prior to installation of the present mixed-resin-bed demineralizer system, a small mixed-resin-bed unit containing approximately 3 ft^3 of resin was installed in a bypass loop (which handles water at a rate of ~3 gpm) of the reactor cooling system. The resin mixture consisted of 40% IR-120 and 60% IRA-410. In addition to the use of the demineralizer as a water purifier, the reactor cooling system was periodically purged with demineralized water to keep the specific resistance of the water above the minimum limit of 150,000 ohm-cm. The performance of the original small unit is indicated in Table 11.

TABLE 11. PERFORMANCE DATA OF MIXED-RESIN-BED UNIT

	Inlet	Exit
pH	6.1	6.1
Specific resistance (ohm-cm)	3.5×10^5	1.4×10^6
Radioactivity (counts/min/ml)*	1.37×10^5	9.2×10^3
Radiation level above reactor tank, reactor not operating	100 mr/hr	

*Samples were counted in a well-type scintillation counter in which an aluminum absorber is used. The counter efficiency is 44% for Na^{24} and 54% for Np^{239}, which, together, accounted for about 85% of the gross activity.

The LITR process water system consists of a 4000-gal-capacity reactor tank, a 4000-gal-capacity seal or surge tank, pumps, air coolers and piping; the total volume of the water system is approximately 10,000 gal. The LITR presents an especially difficult water purification problem because of large amounts of steel surfaces exposed to the water. These are in the reactor tanks, valves, pump housing, fittings, and some pipe. Originally most of the steel surfaces were painted with a plastic paint. The air coolers and part of the piping are aluminum; the demineralizer system, the pump shafts, and the impellers are of stainless steel. An important component of the water system is a clarification filter, which is installed in a bypass loop through which water is bypassed at the rate of about 200 gpm. Without the filter the water becomes cloudy from suspended floc. In the process water system there are about 3800 ft^2 of

exposed aluminum and about 700 ft^2 of exposed steel. A detailed estimate is given below:

	Area (ft^2)
Aluminum	
Fluid cooler and manifold lines	2563
8-in. lines	515
D tank of reactor	27
Fuel assemblies and shim rods	490
Grid assembly, support casting, and exit line in reactor tank	130
Total	3725 ± 10%
Steel	
Filter housing	15
Strainer	5
Valves (4, 6, and 8 in.)	25
Lines (24 in. dia)	610
Reactor tanks, based on half the total surface being exposed	20
Total	675 ± 10%
Portions of seal tank, which was given a coat of plastic paint, that might be exposed	250 ± 10%

In September 1954, a larger demineralizer system was placed in service to replace the smaller unit. It consists of three major components: a mixed-resin-bed unit containing approximately 18 ft^3 of IRA-410 anion resin (Nalcite SAR has been used interchangeably, depending on supply) and approximately 12 ft^3 of IR-120 cation resin (Nalcite HCR has been used interchangeably, depending on supply); two cation-bed units, each containing approximately 2.5 ft^3 of IR-120 resin. Only one cation unit was used at a time, the other remaining in standby.

The small cation columns were used with the mixed-bed column to measure their effectiveness in keeping radioactivity away from the anion resin in the mixed bed. Previous experience indicated that anion resin loses its effectiveness rather rapidly in a mixed-bed column holding large amounts of radioactivity. Since the small columns were available, it was possible to add them to the system with minimum expense.

The normal flow through the demineralizer system is as follows. The inlet water, taken from the discharge side of the fluid cooler, is passed through a filter, one cation column, the mixed-bed unit, a second filter, and then back into the reactor cooling system; the flow rate is 15 gpm.

In March 1955, one anion column was installed in series between the cation column and the mixed-bed unit. This installation was made in order to protect the anion resin in the mixed-bed unit from damage by radioactive anions such as fission-product iodine. Since regeneration is more difficult for a mixed-bed unit than for a single-bed unit, it was desirable to protect the mixed-bed resin. Also, it was more difficult to shield the mixed-bed column than a small anion bed.

The performance of the system before and after the addition of the single anion unit is given in Table 12. Typical analyses of the reactor cooling water are given in Table 13.

When the effluent water resistance of the mixed-bed unit drops to or near to the resistance of the reactor cooling water, the unit is considered depleted. It is then isolated from the system and regenerated. The average life per cycle is approximately five weeks, and the average number of gallons passing through the unit is approximately 560,000 per cycle.

The frequency of regenerations of the single beds of both anion and cation resins is determined by their decontamination factor. On a routine basis, samples of their effluent are counted for gross radioactivity. With freshly regenerated resin the radioactivity in water passing through the columns is reduced by a factor of approximately 20 for the cation resin and by a factor of approximately 2 for the anion resin. The resin beds are removed from service when their effectiveness decreases approximately 50%. The average life per regeneration of the cation resin is about four weeks, while the life of the anion resin is about six weeks.

Data on the operational history of the anion and cation units are presented below:

	Anion Unit	Cation Unit
Date of installation	3/24/55	North 9/15/54 South 10/24/54
Number of regenerations	8	16
Average life of resin	~4 months	~7 months
Number of resin replacements (due to resin damage)	2	3
Average life of resin per regeneration	~6 weeks	~4 weeks

In March 1955, after about six months of operation, regenerations of the mixed-bed unit were

TABLE 12. PERFORMANCE DATA ON LITR COOLING WATER BEFORE AND AFTER ADDITION OF SINGLE ANION UNIT

	Before		After	
	Inlet	Exit	Inlet	Exit
pH	6.2	5.9	6.0	6.1
Specific resistance (ohm-cm)	5.5×10^5	1.7×10^6	6.2×10^5	1.7×10^6
Radioactivity (counts/min/ml)[a]	5.3×10^4	2.8×10^3	4.3×10^4	2×10^3
Average life of mixed-resin bed per regeneration[b] (days)	39		37	
Number of regenerations	6		6	
Amount of resin (IRA-410) added to mixed bed[c] (ft^3)	5		18	
Radiation level at surface of mixed bed (mr/hr)	400		25	
Radiation level over reactor tank, with reactor not operating and top plug removed (mr/hr)	50		40	

[a]Samples were counted on a well-type scintillation counter in which an aluminum absorber was used. The counter efficiency is 44% for Na^{24} and 54% for Np^{239}; the sum of the two activities is approximately 85% of the gross activity.

[b]The decrease in average life is probably due to damage to the anion resin prior to the installation of the single anion unit.

[c]The replacement of the 18 ft^3 of IRA-410 resin in the mixed bed was probably due to radiation damage prior to the anion unit installation. Additional service will be necessary to demonstrate this.

unsuccessful. After about 5 ft^3 of anion resin was added to the mixed-bed column, the column performed normally. This addition was necessitated by the loss of resin during the backwash step of regeneration. There may also have been some loss of capacity because of the damage of the resin by radioactivity. However, in October 1955, after about 12 months of operation, the cycle time began to shorten, and difficulty was again encountered in obtaining a successful regeneration. At this time, the entire quantity of anion resin was replaced, and then the regenerations and performance were again normal.

In April 1955, after about six months of operation of the single cation unit, difficulty was encountered in maintaining normal flow. After examination, the resin was found to contain a large amount of "fines" (chips from resin beads). The cause of this has not been established; however, in demineralizers for plant process water, this type of cation resin has been found to be similarly damaged, although the time in which the damage occurred was considerably shorter. To restore normal operation, the damaged resin was replaced.

Difficulty of a similar nature was experienced with the single anion unit. In July 1955, after about four months of operation, the resin had to be replaced because it was impossible to maintain normal flow rates. Inspection showed the presence of "fines" similar to those found in the cation unit. The cause of resin damage has not been definitely established; however, it is believed that the high radioactivity has greatly increased the damage rate.

The present system is a very great improvement over previous systems. The reduction in radioactivity of the water lowered the radiation field around the equipment, and most of the contamination received by personnel working with tools in the water of the reactor tank was eliminated. The high specific resistance maintained in the water

greatly reduced the corrosion rate – an important factor in the life of the LITR with its large exposed steel surfaces. More time will be required to evaluate the anion column, since it is difficult to determine the damage that was sustained by the mixed bed before the anion column was installed.

To further improve the system it would be necessary only to increase the size of the cation and anion units so that more water could be put through.

Np^{239} in Process Water

A striking reduction in the amount of Np^{239} in the process water took place. In the past year, the ratio of Np^{239} to Na^{24} has decreased by a factor of about 6. This represents a decrease in water activity of about 50%; this reduction has been indicated as shown in Table 14. It appears that the source of most of the Np^{239}, apparently a contaminated fuel element, has been removed.

TABLE 13. TYPICAL ANALYSES OF REACTOR COOLING WATER

Impurity (ppm)	System A[a]		System B[b]			Present System[c]		
	10/53	3/54	10/54	12/54	2/55	7/55	9/55	11/55
Al	0.32	0.2	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05
Fe	<1	0.05	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05
Be	0.44	0.28	<0.05	0.2	<0.05	<0.05	<0.05	<0.05
Cu			<0.05	<0.05	<0.05	<0.05	<0.05	<0.05
Ca	<0.2	<0.05				0.085	<0.05	0.03
Na	0.2	0.30				0.075	0.064	0.05
SO_4	<1	2.0	<0.05	<0.05	<3	<0.05	<0.05	<0.05
Cl	1.7	3.3	<1	<1	<1	1.47	<1	<0.03
NO_3						0.28	<0.05	0.29
H_2O_2		22.0				6.8	3.4	3.74
Dissolved O_2						9.0–9.2	6.4	
CO_2		2.0	1	3.29	4	2.0	1.5	1.0
Total solids	3.2	8.4	5.2	2.7	6.6	3.8	3.6	2.2
Specific resistance (ohm-cm)	4×10^5	2.67×10^5	4.1×10^5	2.31×10^5	3.02×10^5	3.99×10^5	6.84×10^5	6.86×10^5

[a] 3-gpm demineralizer, mixed-bed unit.

[b] 15-gpm demineralizer, mixed-bed unit plus cation column.

[c] 15-gpm demineralizer, mixed-bed unit and cation column, with one anion column installed in series between them.

TABLE 14. COMPARISON OF Np^{239} WITH Na^{24} ACTIVITY IN LITR PROCESS WATER

Flow through process water system: 1200 gpm

	Gross Activity (counts/min/ml)	Na^{24} (%)	Np^{239} (%)	Flow Through Demineralizer (gpm)
12/27/55	2 to 3×10^4	58	17	12 to 14
9/23/54	4 to 6×10^4	32	55.3	12 to 15
4/1/54	1.5 to 2.0×10^5	28	59	3
3/24/53	2.4×10^5*	27	58	3

*Adjusted to 3000 kw; the power was increased on September 2, 1953, from 1500 to 3000 kw.

Appendix A

LITR REMOTE CONTROL INSTRUMENTATION

LIST OF INSTRUMENTS

Annunciator

1. LITR recorders
2. Off-gas vacuum
3. Electronic instruments
4. Remote dc
5. HB-1
6. HB-2
7. HB-3
8. HB-4
9. HB-5
10. HB-6
11. C-44
12. Top plug No. 2
13. C-48
14. East room activity
15. West room activity
16. Top room activity
17. Reactor water
18. East room scram
19. West room scram
20. Seal tank level

Recorders

1. No. 1 safety
2. Regulating rod position
3. Water outlet temperature
4. Utility
5. Water flow
6. Water differential temperature

Meters

1. Monitors
2. Reactor period
3. Fission chamber (counts per sec)
4. No. 1 safety
5. Remote dc
6. Reactor power (% of fuel power)
7. No. 3 safety
8. No. 2 safety
9. West fan ammeter
10. East fan ammeter
11. North pump ammeter
12. South pump ammeter
13. Reactor tank pressure
14. Seal tank level
15. Exhaust vacuum
16. Spare
17. No. 1 shim rod selsyn
18. No. 2 shim rod selsyn
19. No. 3 shim rod selsyn
20. Log potentiometer selsyn

Switches and Other Equipment

1. Switches and indicating lights for complete control of individual shim rods
2. Servo demand switch and servo vernier switch
3. Fan control switches
4. Scram switch and scram reset button
5. Setback light
6. N_L and N_F lights from log potentiometer

INSTRUMENTATION DRAWINGS

RC4-1-4	Block Diagram Control Instruments
RC4-1-5	Block Diagram Process Instruments
RC4-1-6A	Interconnection Diagram
RC4-1-6B	Interconnection Diagram
RC4-1-7	Elementary Diagram Fan Control
RC4-1-8	LITR Control Elementary
RC4-1-9	LITR Control Elementary Diagram of Auxiliary Safety Circuits
RC4-2-1	Console X-10
RC4-2-2	Panels
RC4-2-3	Synchro Mounts and Dial Cover Details
RC4-2-4	LITR Console Panels and Details
RC4-2-5	Drilling Details for Panels 3 and 4
RC4-2-6	Shim Rod Synchro Mounts for Top Plug
RC4-2-7	Control Cubicle Internal Wiring
RC4-2-9	LITR Control Auxiliary Panel and Log Potentiometer Junction Boxes
RC4-2-10	Terminal Box Wiring Diagram
RC4-2-11	LITR Control Relay Panel Wiring Diagram
RC4-2-12	LITR Control Console Panel Wiring Diagram
RC4-2-13	LITR Top Plug Terminal Box and Equipment Wiring
RC4-3-1	Relay Junction Box Wiring and Elementary
RC4-3-2	Fan Control Wiring Diagram
RC4-3-3	Fan and Pump Current Box Wiring Diagram
RC4-6-2	Boxes

QUESTIONS UNDER CONSIDERATION CONCERNING FURTHER INSTRUMENTATION

1. Connection of gas flow to alarm
2. Connection of ion chamber voltages to alarm
3. Reading of magnet currents at the remote desk
4. Remote reading of the radioactivity of the water
5. Determination of the maximum power change in either direction that will be taken care of at the remote desk
6. Inclusion in the general annunciator panel of positions for log N, period, etc., to give a remote indication other than merely "Electronic Instrument"
7. Inclusion of a smoke detector for the control room to detect possible instrument fires

Appendix B

LITR FUEL LOADING PATTERN AND METHOD OF DETERMINING THERMAL FLUX DISTRIBUTION TO BE USED IN BURNUP CALCULATIONS

On October 26, 1954, the normal LITR fuel loading pattern was altered to meet increased flux requirements at HB-5. The lattice modification involved the following changes:

1. Three fuel pieces were transferred from positions 11, 21, and 31 to positions 45, 55, and 56 (see Figs. B.1 and B.2).
2. Isotope-target-material trays were transferred from positions 53 and 55 to positions 29 and 39.
3. Beryllium pieces, replaced by the fuel and isotope-target-material trays, were transferred to the positions that were vacated.

The thermal flux in HB-5 was increased from 7.6×10^{12} neutrons/cm^2·sec, determined while the hole was empty, to 1.3×10^{13} neutrons/cm^2·sec with an experiment in place.

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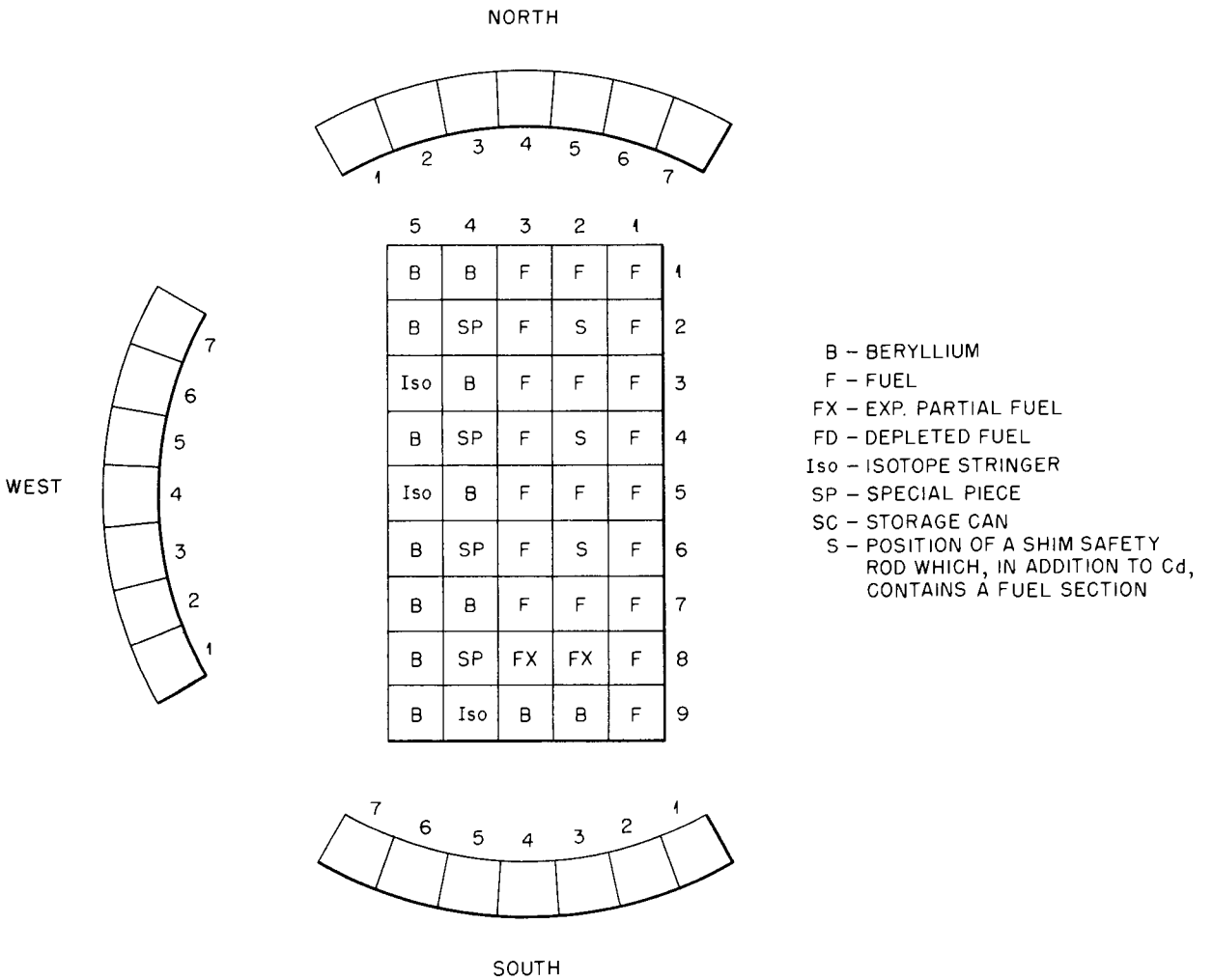

Fig. B.1. LITR Fuel Loading Pattern Prior to the Change of October 1954.

The modification of the loading pattern necessitated a subsequent re-evaluation of the neutron flux distribution for purposes of fuel burnup calculations. Curves for the neutron flux distribution were plotted from data obtained by chopping into 1-in. segments 24-in. (the length of the fuel section) stainless steel strips exposed for 30 min between the plates of 14 fuel pieces and making radiation measurements of the pieces with a mica window counter (see Table B.1). Points for the flux in 14 fuel elements were measured, and the values in the remaining fuel elements were determined by interpolation and extrapolation.

Figures B.3 through B.5 show plots of the average flux per element from north to south through fuel rows 1, 2, and 3; Fig. B.6 shows the plot of the flux from east to west through fuel positions 15, 25, 35, 45, and 55; and Fig. B.7 gives the plot of the flux diagonally through fuel positions 12, 23, 34, 45, and 56. The measured points of the curves were normalized to the average flux for the core as determined by burnup calculations (see Table B.2).

Figures B.8 through B.18 are plots of the counts on the stainless steel segments intended to show the vertical flux distribution through each of the 14 measured elements.

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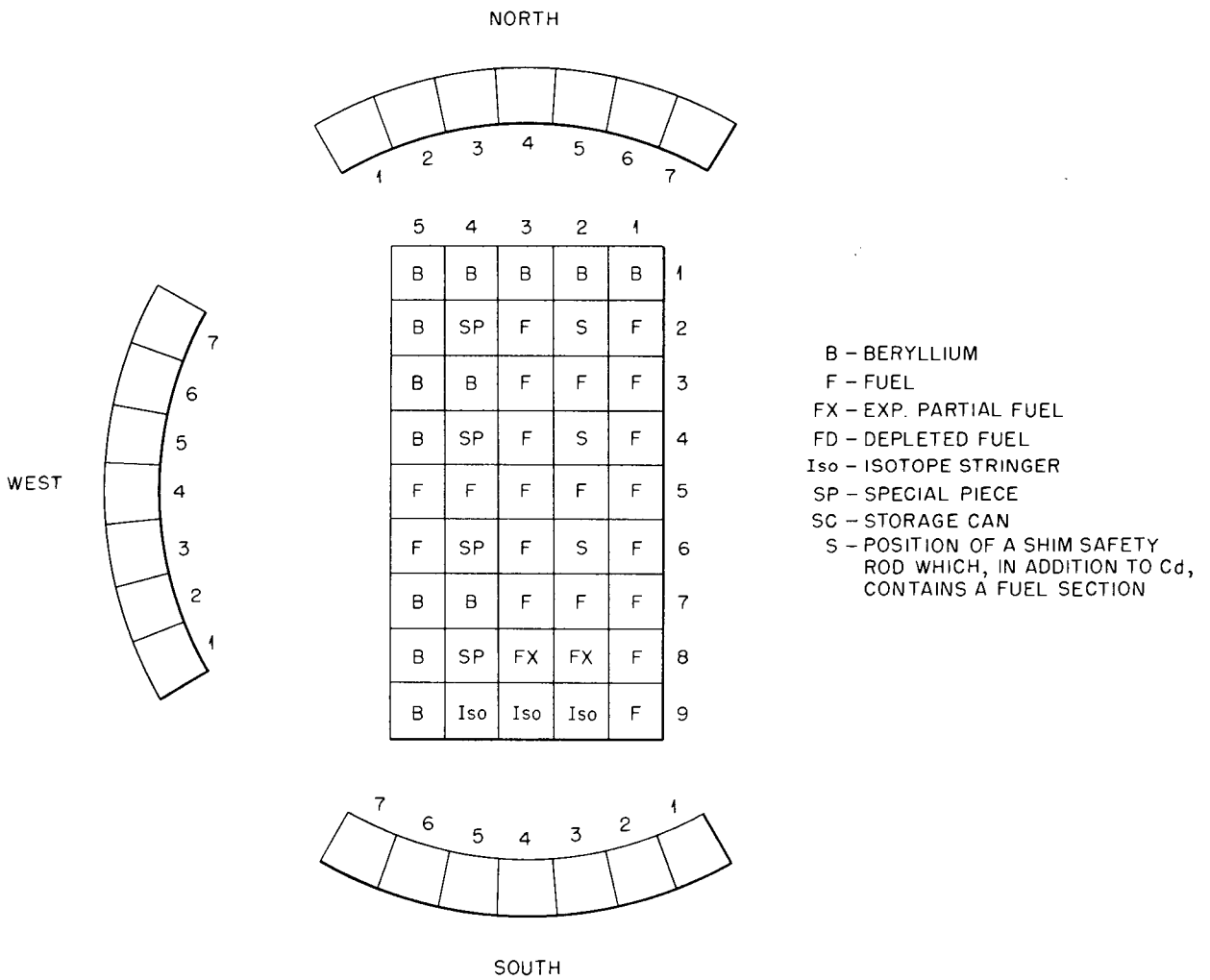

Fig. B.2. LITR Fuel Loading Pattern After Change of October 26, 1954.

TABLE B.1. RADIATION READINGS (counts/min/g) ON SEGMENTS OF STAINLESS STEEL STRIPS

Segment Number	Core Positions						
	C-13	C-36	C-32	C-15	C-23	C-34	C-35
1*	2,771	2,784	2,148	2,718	2,851	3,308	3,277
2	2,339	2,711	2,140	2,490	2,752	3,128	3,063
3	2,512	2,898	2,373	2,658	2,863	3,329	3,451
4	2,767	3,265	2,610	2,965	3,358	3,861	3,606
5	3,001	3,726	2,957	3,024	3,729	4,135	4,200
6	3,436	4,012	3,246	3,242	4,221	4,538	4,461
7	3,402	4,246	2,058	3,437	4,469	4,855	4,553
8	3,627	4,626	3,669	3,543	4,613	5,446	4,769
9	3,746	4,822	3,870	3,689	4,888	5,415	5,296
10	3,883	5,091	3,971	3,845	5,071	5,404	5,604
11	4,125	5,234	4,145	4,044	5,357	5,717	5,423
12	4,326	5,442	4,203	4,215	5,510	5,853	5,948
13	4,302	5,466	4,329	4,561	5,498	5,892	6,418
14	4,431	5,472	4,261	5,073	5,556	5,863	6,167
15	4,474	5,428	4,180	5,157	5,573	5,775	6,095
16	4,301	5,420	4,207	4,989	5,409	5,822	6,077
17	4,267	5,179	3,957	4,965	5,182	5,627	5,657
18	4,009	4,924	3,754	4,904	5,134	5,439	5,823
19	3,785	4,862	3,552	4,571	4,592	5,127	5,226
20	3,461	4,267	3,305	4,167	4,338	4,737	4,796
21	3,191	3,959	2,964	3,723	3,994	4,253	4,309
22	3,274	2,117	2,649	3,305	3,404	4,105	3,779
23	2,291	3,181	2,849	2,908	2,914	4,178	3,277
24*	2,085	2,736	2,045	2,620	2,665	2,934	3,175
Total	83,806	101,868	79,450	90,813	103,941	114,441	114,450
Average	3,492	4,245	3,310	3,784	4,331	4,768	4,769
	C-55	C-19	C-27	C-17	C-25	C-28	C-56
1*	2,113	1,172	2,875	2,332	3,315	1,615	2,604
2	1,564	1,132	2,258	2,085	3,357	1,734	2,155
3	1,744	1,261	2,250	2,138	3,395	1,911	2,097
4	1,962	1,363	2,484	2,236	3,960	2,070	2,228
5	1,959	1,578	2,957	2,574	4,381	2,219	2,541
6	2,066	1,678	3,202	2,807	4,972	2,372	3,103
7	2,068	1,797	3,195	2,897	2,765	2,492	3,062
8	2,218	1,887	3,488	3,065	5,543	2,593	2,661
9	2,272	1,841	3,921	3,202	5,839	2,666	2,691
10	2,337	1,807	4,045	3,258	6,052	2,792	3,292
11	2,479	1,777	3,898	3,426	6,453	2,989	3,259
12	2,659	1,831	4,008	3,627	6,258	3,040	3,575
13	2,910	1,979	4,412	3,811	6,222	3,050	4,096
14	3,063	1,945	4,965	3,934	6,367	2,910	3,775
15	3,380	2,037	4,588	3,970	6,352	2,914	4,608
16	3,301	2,008	4,127	3,926	6,153	2,674	5,369
17	3,107	2,000	3,916	3,991	5,915	2,658	4,328

TABLE B.1 (continued)

Segment Number	Core Positions						
	C-55	C-19	C-27	C-17	C-25	C-28	C-56
18	3,044	1,866	3,738	3,744	5,975	2,682	5,320
19	2,916	1,861	3,573	3,511	5,281	2,167	3,991
20	2,930	1,749	3,229	3,312	5,084	1,988	3,911
21	2,533	1,647	3,037	3,112	4,767	1,736	2,817
22	2,304	1,676	2,963	2,918	4,357	1,527	2,327
23	2,034	1,359	2,505	2,475	6,436	1,427	2,102
24*	1,822	1,442	2,535	2,273	3,156	1,253	1,974
Total	58,785	40,693	82,169	74,624	122,355	55,479	77,886
Average	2,449	1,696	3,424	3,109	5,098	2,312	2,852**

*Segment 1 was positioned at the top of the 24-in. fuel section and segment 24 at the bottom.

**An adjusted average, to correct for obvious inaccuracies in the vertical flux traverse (see Fig. B.10).

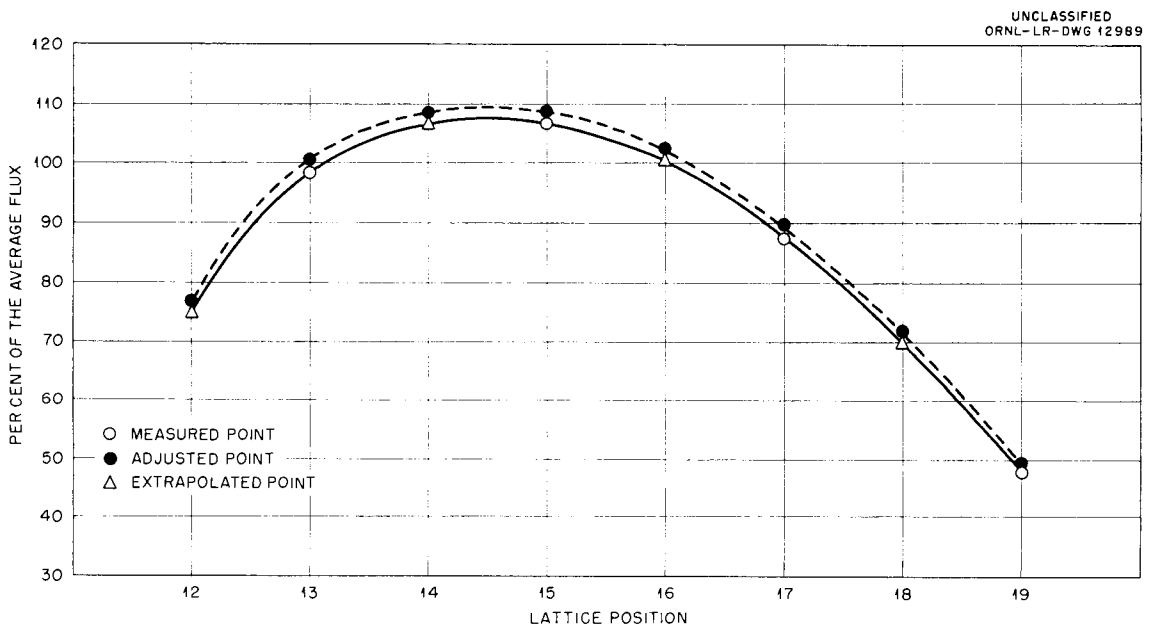

Fig. B.3. Plot of the Thermal Neutron Flux Per Element Through Fuel Row No. 1.

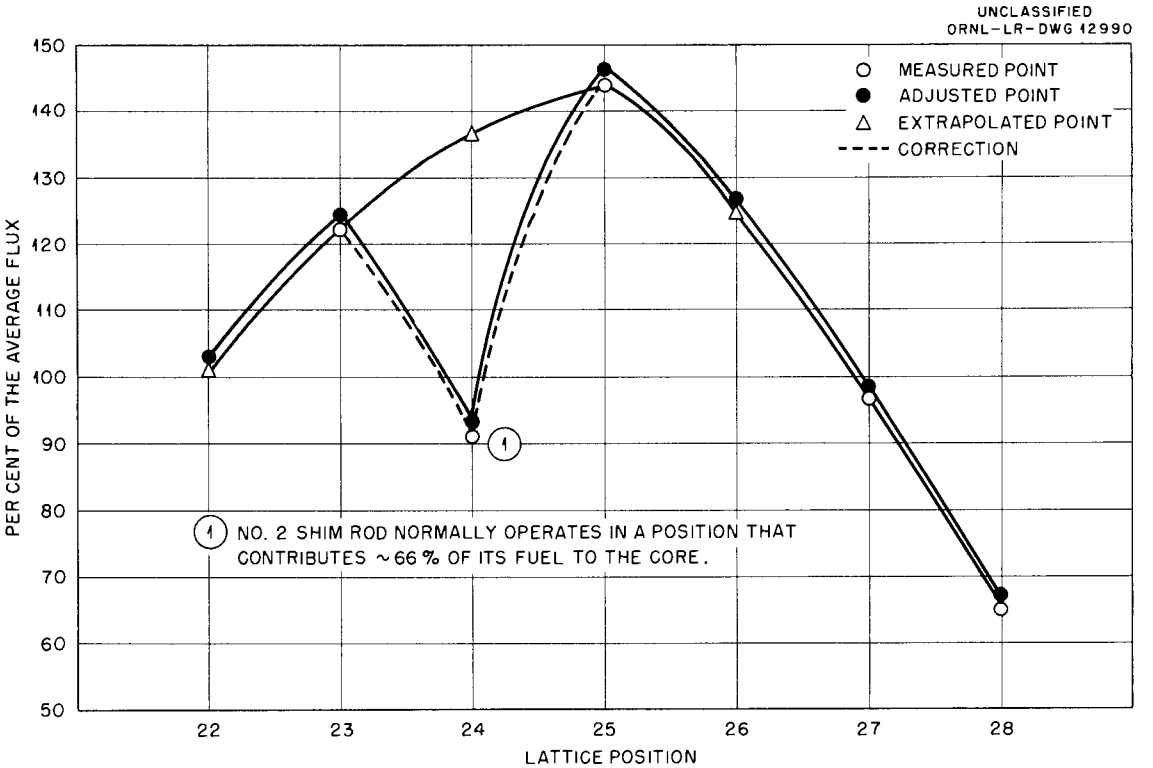

Fig. B.4. Plot of the Thermal Neutron Flux Per Element Through Fuel Row No. 2.

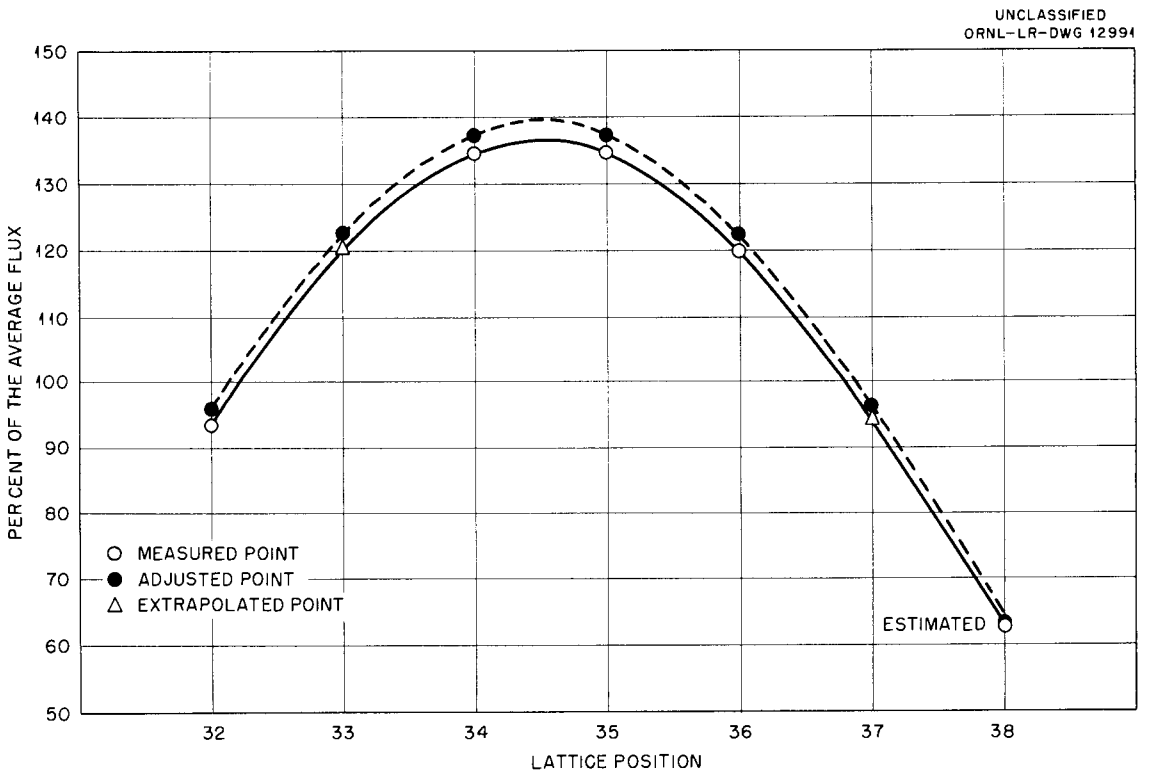

Fig. B.5. Plot of the Thermal Neutron Flux Per Element Through Fuel Row No. 3.

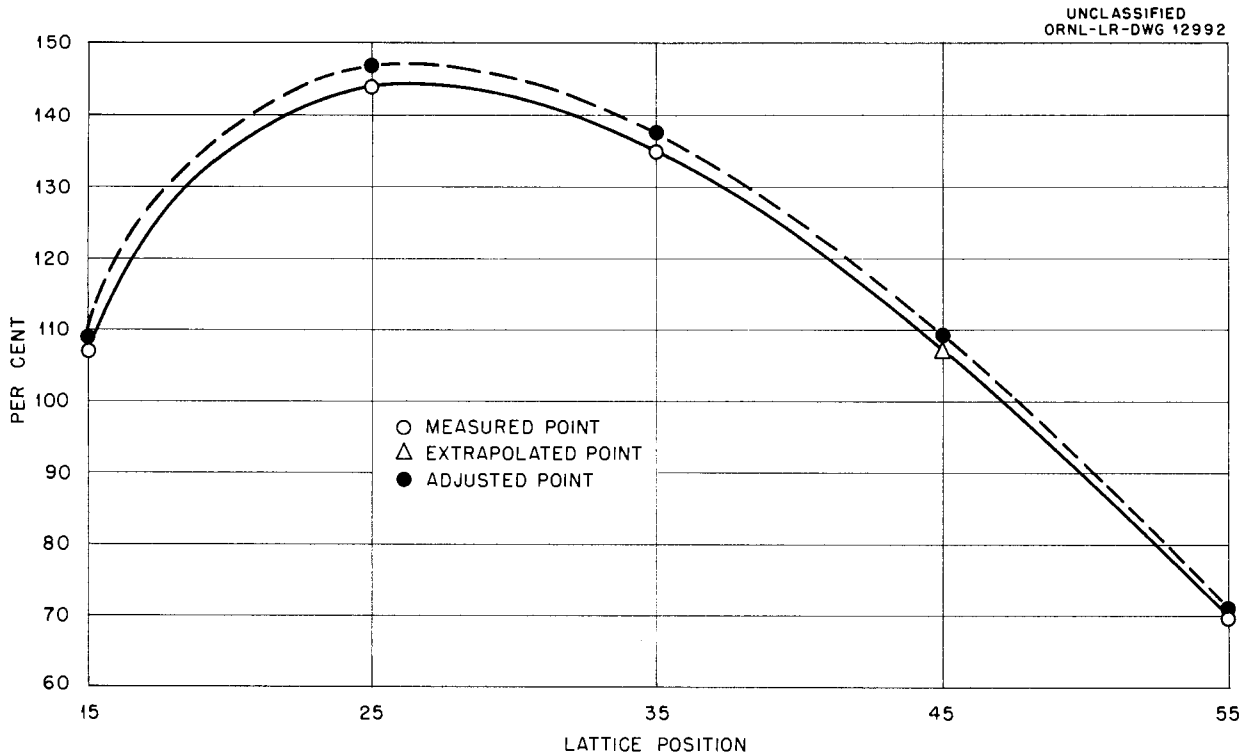

Fig. B.6. Plot of the Thermal Neutron Flux Per Element Through Fuel Positions 15, 25, 35, 45, and 55.

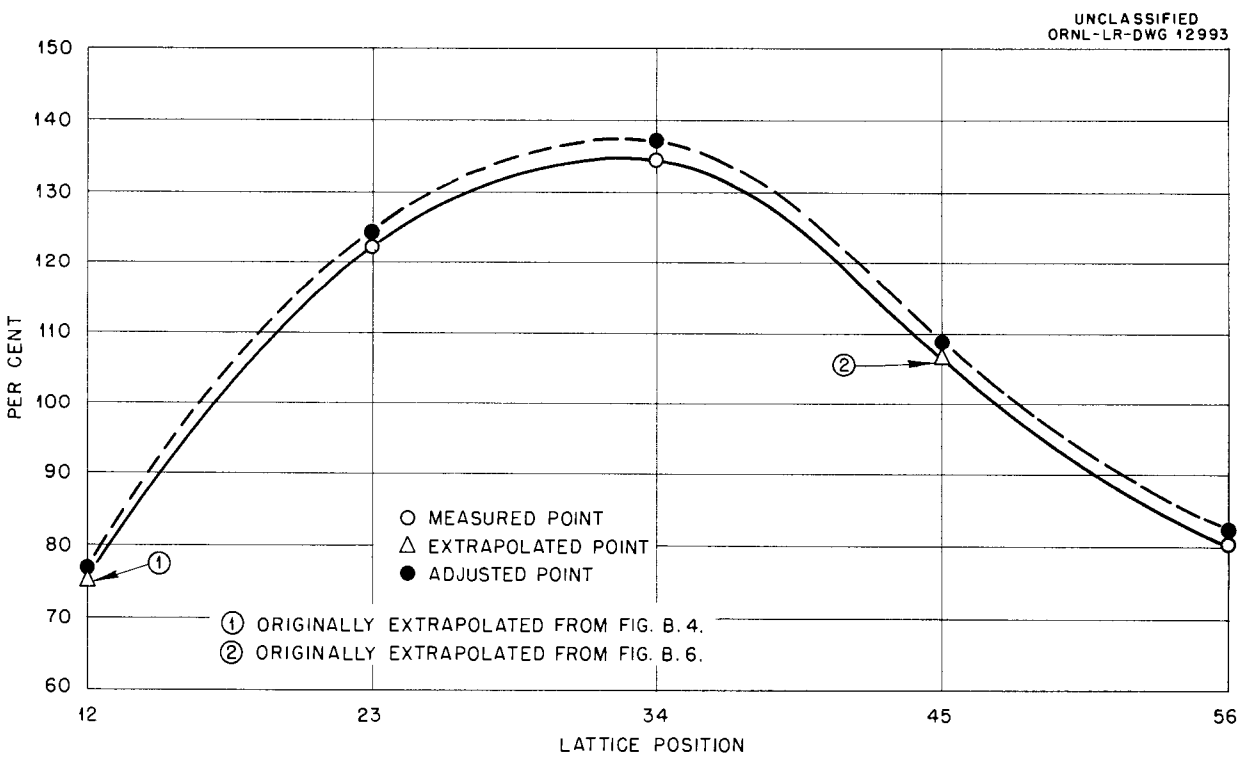

Fig. B.7. Plot of the Thermal Neutron Flux Per Element Through Fuel Positions 12, 23, 34, 45, and 56.

TABLE B.2. DATA FOR FLUX CURVES

Core position or fuel number	Average counts per gram of stainless steel segments	Average counts per gram expressed as per cent of the average for the entire core	Extrapolated and interpolated figures (from Figs. B.3–B.7) representing average counts per gram expressed as per cent of the average for the entire core	Percentages reconverted to counts per gram based on a core average of 3546 counts	Corrected percentages based on a corrected average count on core from column 5	Position factor
38		62.66*		2,222	63.91	0.64
12			75.00	2,660	76.50	0.76
13	3,492	98.48		3,492	100.43	1.00
14			106.50	3,776	108.60	1.09
15	3,784	106.71		3,784	108.83	1.09
16			100.40	3,560	102.39	1.02
17	3,109	87.68		3,109	89.42	0.89
18			70.00	2,482	71.38	0.71
19	1,696	47.83		1,696	48.81	0.49
22			100.80	3,574	102.79	1.03
23	4,331	122.14		4,331	124.56	1.25
24			91.20**	3,234	93.01	0.93
25	5,098	143.77		5,098	146.62	1.47
26			124.50	4,415	126.98	1.27
27	3,424	96.55		3,424	98.48	0.99
28	2,312	65.20		2,312	66.49	0.67
32	3,310	93.34		3,310	95.20	0.95
33			120.00	4,255	122.38	1.22
34	4,768	134.46		4,768	137.13	1.37
35	4,769	134.49		4,769	137.16	1.37
36	4,245	119.71		4,245	122.09	1.22
37			93.80	3,326	95.66	0.96
45			106.70	3,784	108.83	1.09
55	2,449	69.06		2,449	70.43	0.70
56	2,852	80.43		2,852	82.02	0.82
Total	49,639			86,927		
Average	3,546			3,477		

*Estimated to coincide with the flux of another half fuel piece in C-28.

**Represents 66% of the actual interpolated value, because the shim rod in this position normally contributes approximately 66% of its fuel to the core.

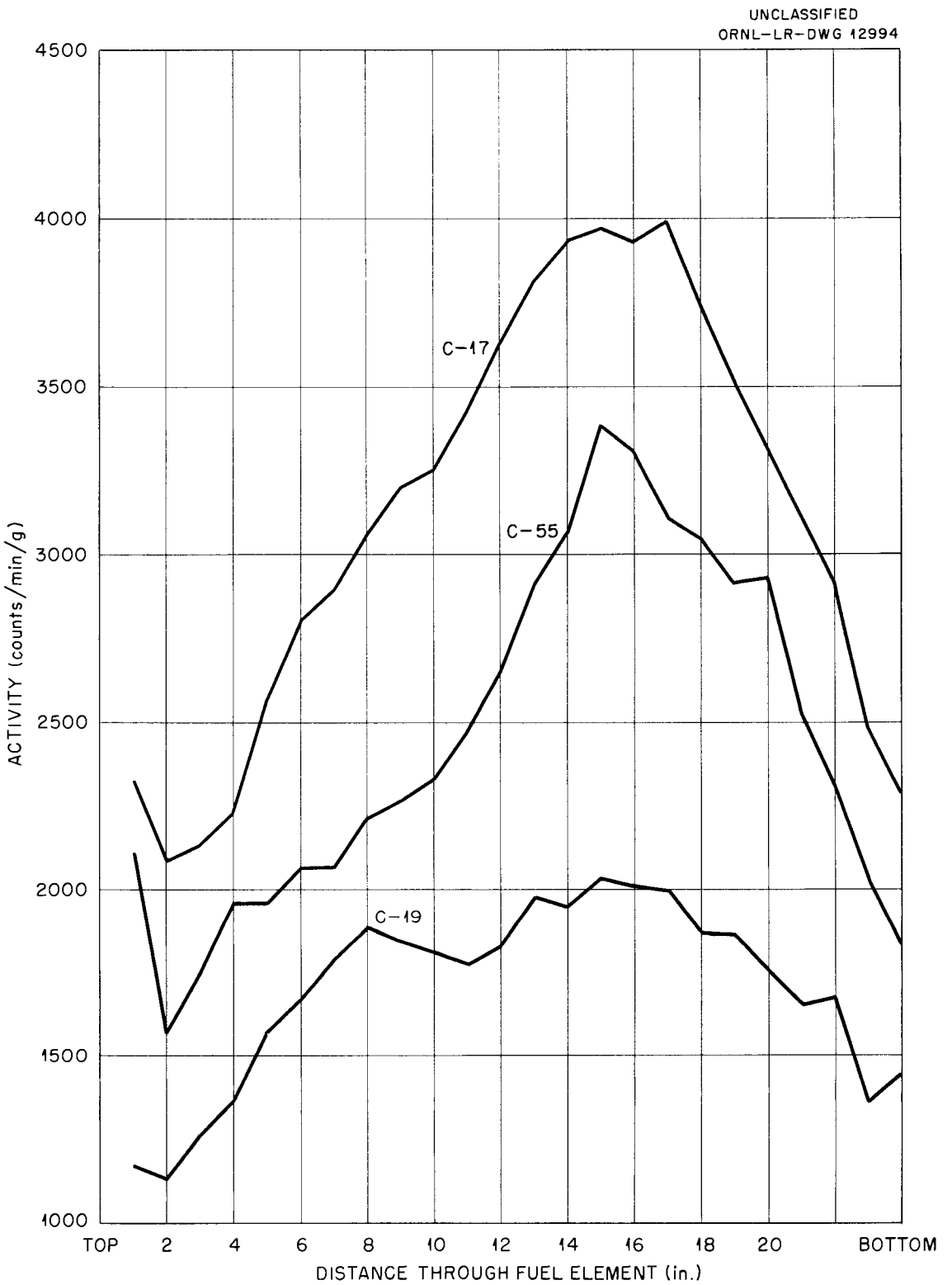

Fig. B.8. Vertical Traverses Through Fuel Elements C-17, C-55, and C-19.

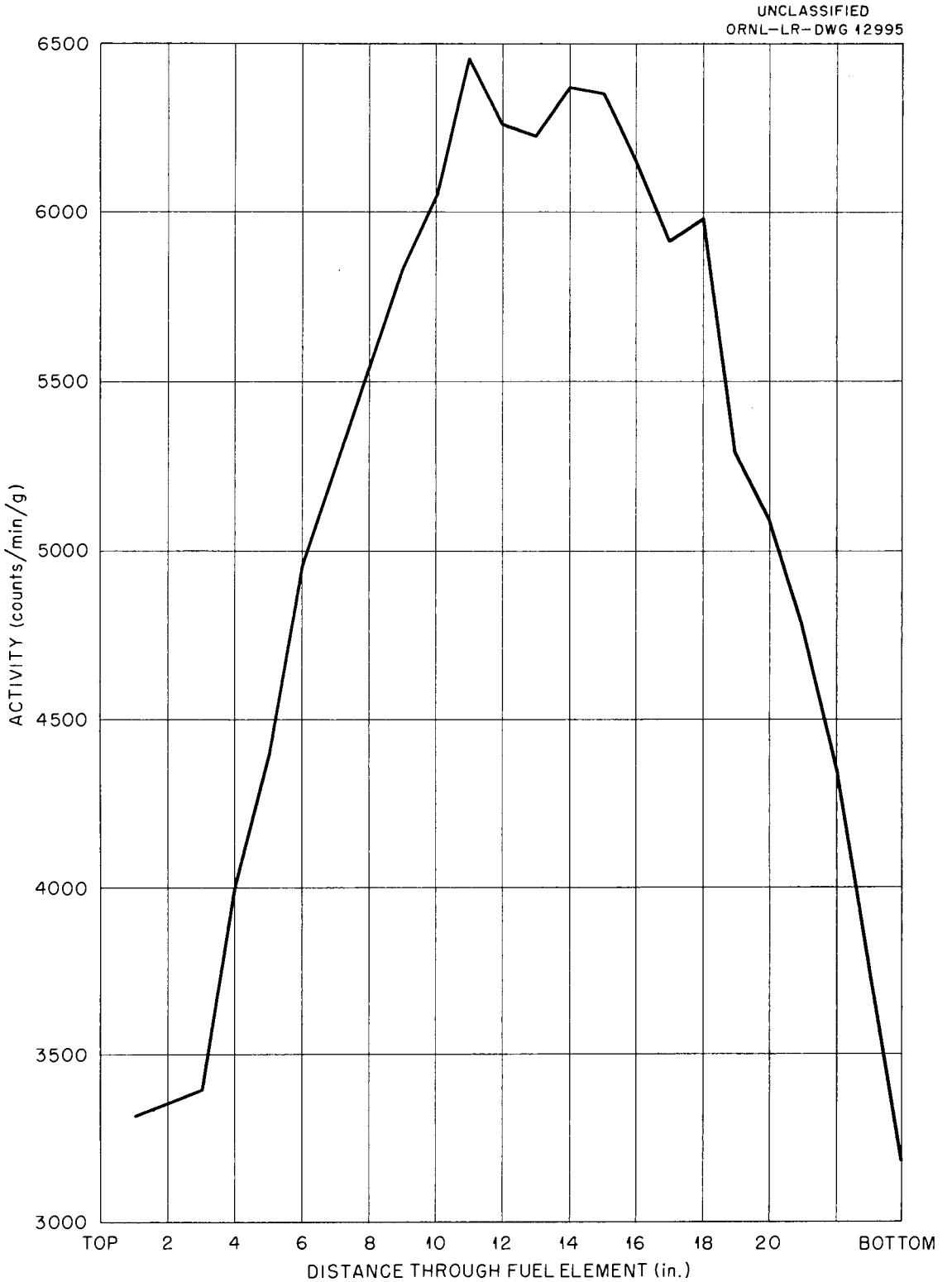

Fig. B.9. Vertical Traverse Through Fuel Element C-25.

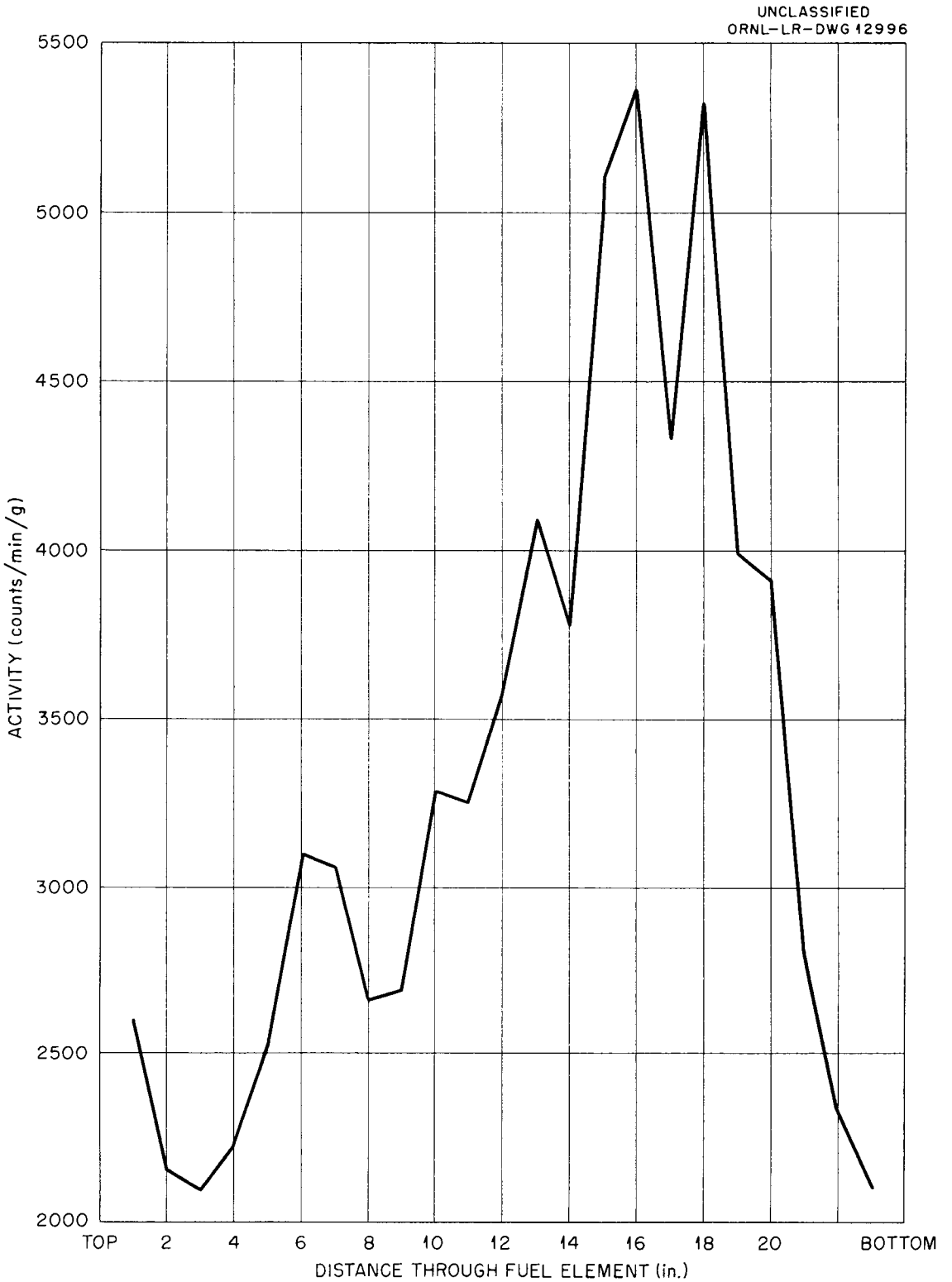

Fig. B.10. Vertical Traverse Through Fuel Element C-56.

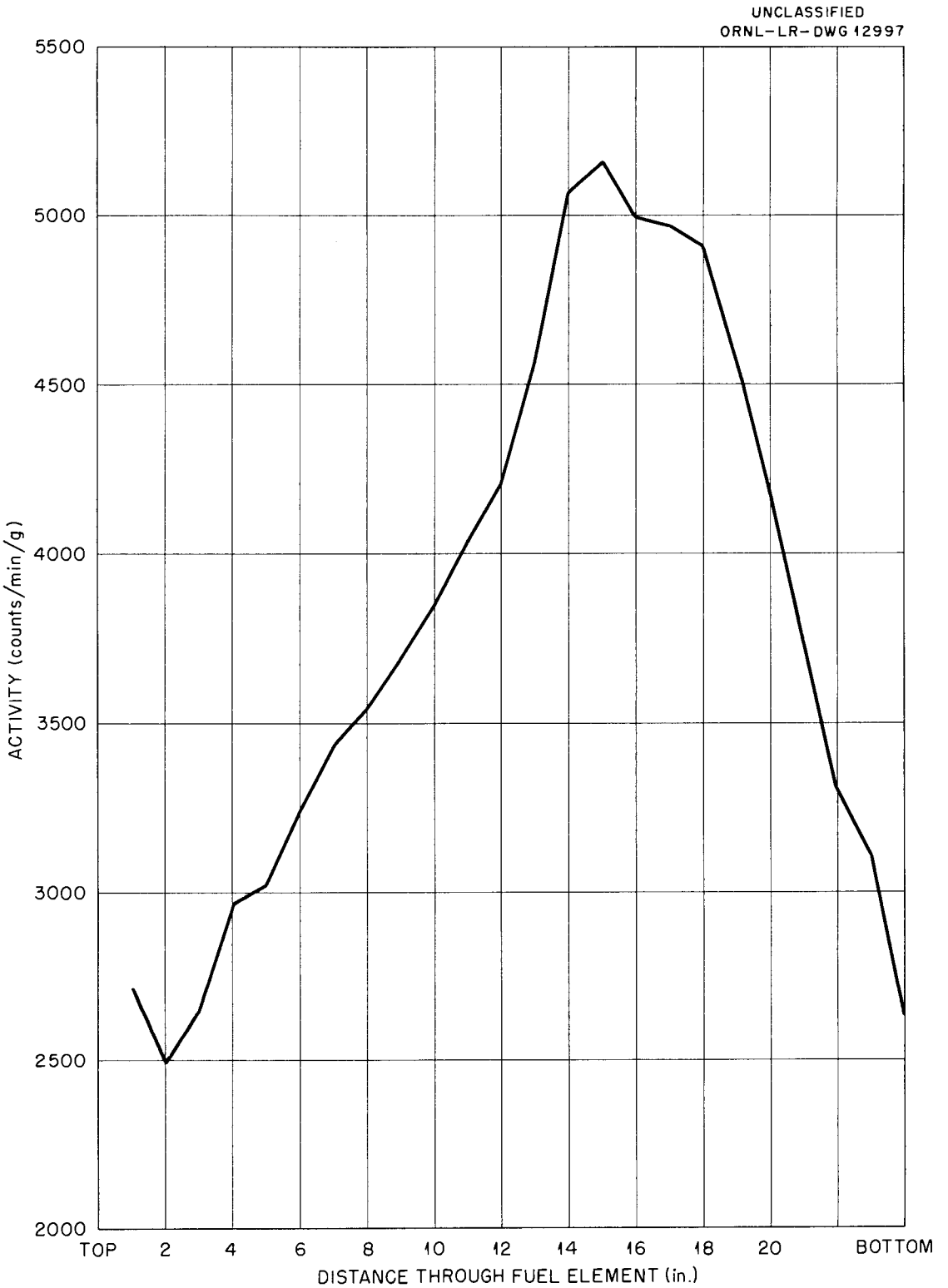

Fig. B.11. Vertical Traverse Through Fuel Element C-15.

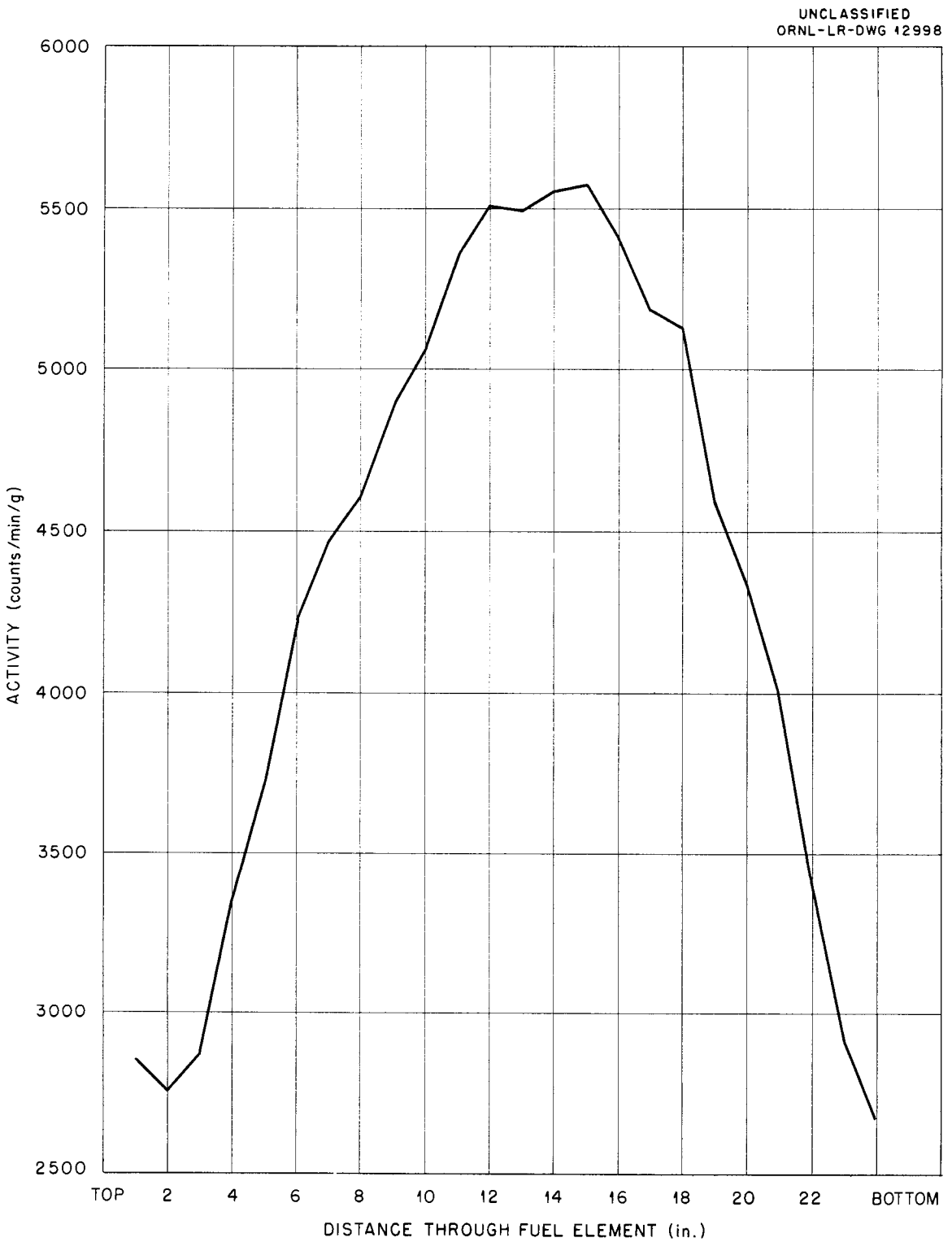

Fig. B.12. Vertical Traverse Through Fuel Element C-23.

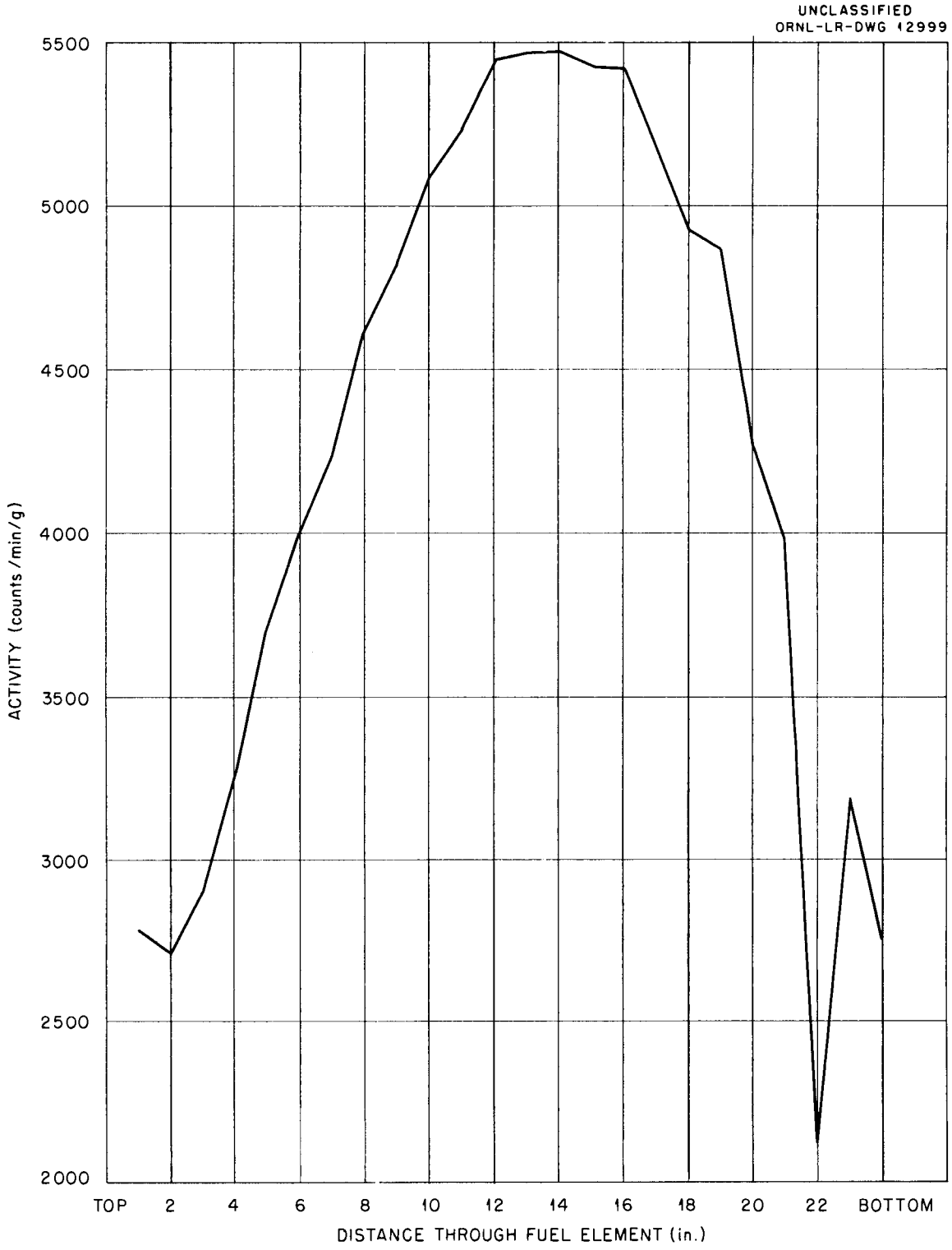

Fig. B.13. Vertical Traverse Through Fuel Element C-36.

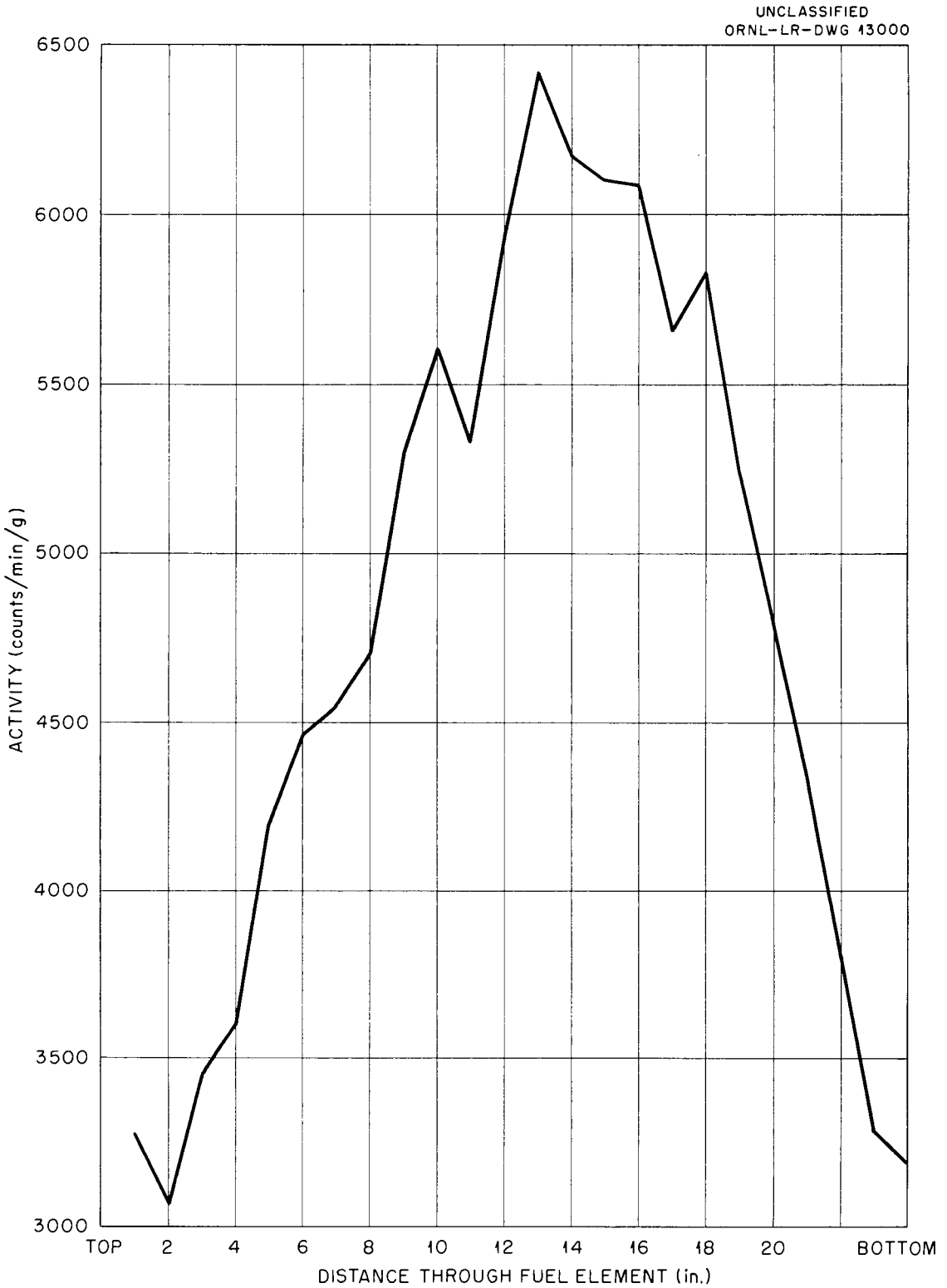

Fig. B. 14. Vertical Traverse Through Fuel Element C-35.

Fig. B.15. Vertical Traverses Through Fuel Elements C-34 and C-27.

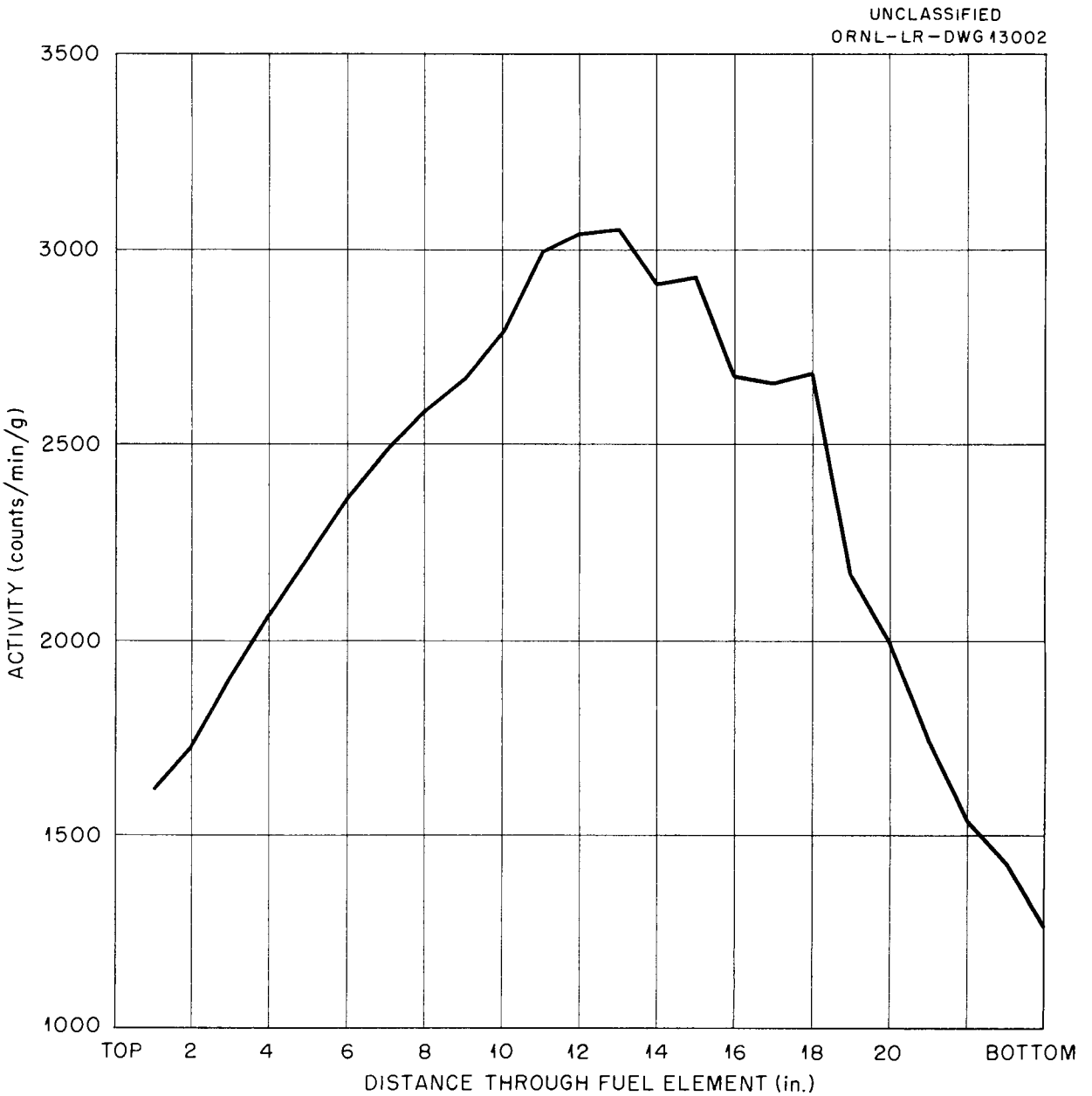

Fig. B.16. Vertical Traverse Through Fuel Element C-28.

[redacted]

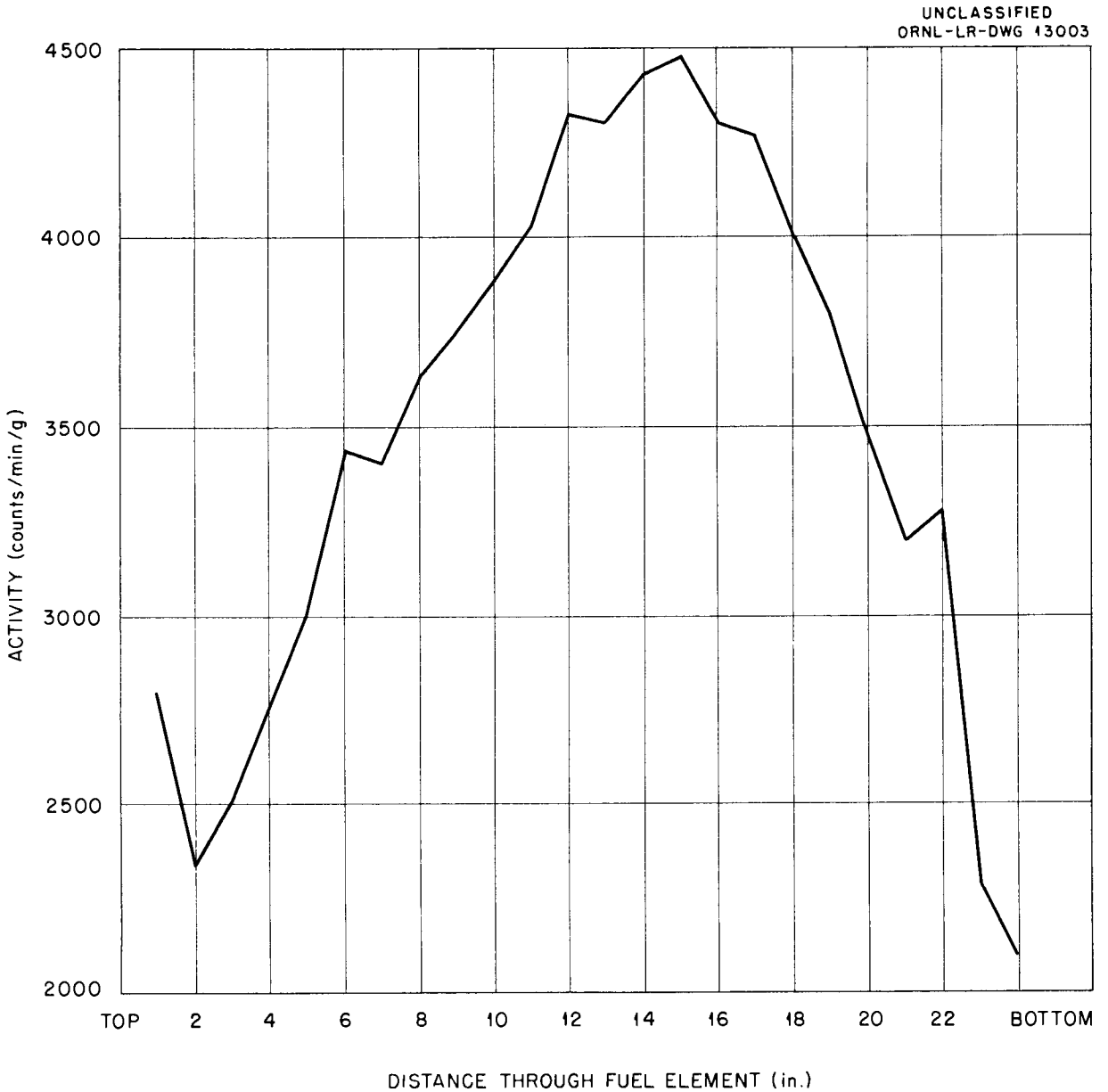

Fig. B.17. Vertical Traverse Through Fuel Element C-13.

[redacted]

[redacted]

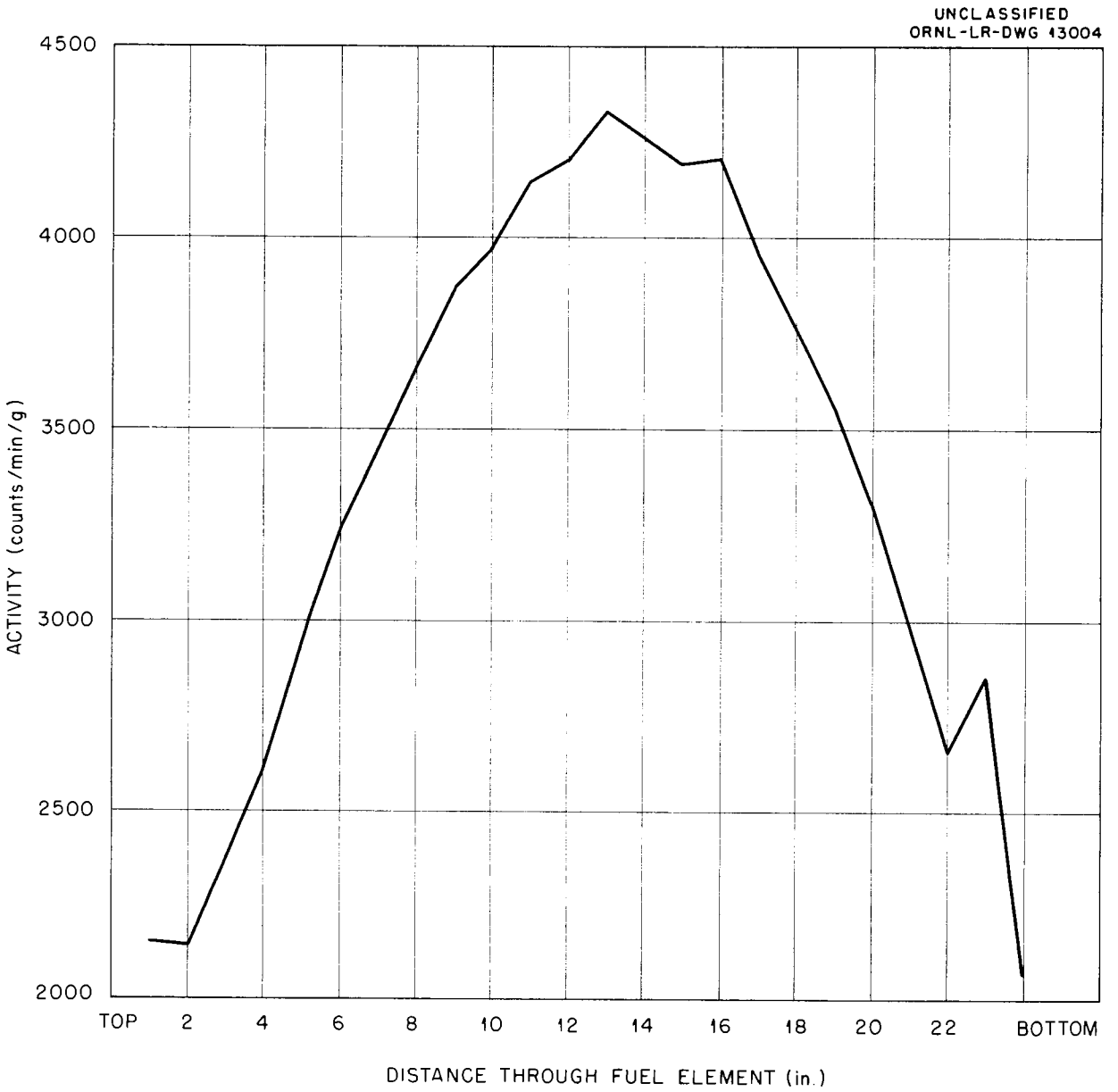

Fig. B.18. Vertical Traverse Through Fuel Element C-32.

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