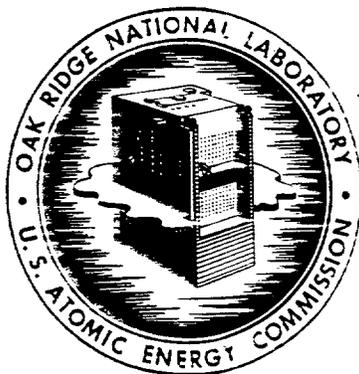


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REACTOR OPERATIONS AND RADIOACTIVE  
WASTES OPERATIONS QUARTERLY REPORT  
JULY-SEPTEMBER 1959



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REACTOR OPERATIONS  
AND RADIOACTIVE WASTES OPERATIONS  
QUARTERLY REPORT  
July - September, 1959

By

J. A. Cox

Prepared from reports by:

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DATE ISSUED

DEC 30 1959

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OAK RIDGE NATIONAL LABORATORY  
Oak Ridge, Tennessee  
operated by  
UNION CARBIDE CORPORATION  
for the  
U. S. ATOMIC ENERGY COMMISSION

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REACTOR OPERATIONS  
AND RADIOACTIVE WASTES OPERATIONS  
QUARTERLY REPORT

Summary

For most of the summer, the ORR was operated at 16 instead of 20 Mw because of a deficiency in the cooling system. Engineering has been completed and directive approval requested for 30 Mw of new cooling tower capacity. The operating time increased to 80.78% and the number of false shutdowns decreased. Heat transfer in the ORR core and tank appears to be satisfactory at 30 Mw, but additional investigations appear necessary for higher power. Additional standby pumping capacity for power failure conditions will also probably be necessary. Measurements were made in the ORR hot cell of the time at which irradiated fuel elements could be safely stored in air instead of water.

Two new vertical holes have been successfully drilled in the Graphite Reactor to provide more facilities for experimenters. Approval for the graphite annealing was received September 9 and the directive September 16. Work is starting on this project and it is expected to anneal next summer.

The waste treatment plant for radioactive liquid waste effectively removed 99% of a release of  $\text{Pm}^{147}$  as well as 80-90% of most other radioactive contaminants.

1. OAK RIDGE RESEARCH REACTOR

1.1. Operations

W. R. Casto

Operations

The ORR was operated at 16 Mw after July 18. Prior to this date, the operating level was raised to 20 Mw when the outside air temperatures were less than 82°F and reduced when air temperature exceeded 82°F. Experimenters generally preferred a constant power rather than an intermittently high power. Operating data are given in Table 1.1. Total operating time for this quarter was 1783.652 hours.

TABLE 1.1. ORR OPERATIONS

Period July 1, 1959, through September 30, 1959

	This Quarter	Last Quarter	Year to Date
Total energy, Mwd	1195.1	1261.2	3771.7
Average power, Mw operating hour	16.08	19.2	18.08
Time operating, %	80.78	72.0	76.43
Reactor water radioactivity, c/m/ml (av)	24,330	30,348	27,684
Pool water radioactivity, c/m/ml (av)	433	593	565
Reactor water resistivity, ohm-cm (av)	780,666	880,000	813,000
Pool water resistivity, ohm-cm (av)	901,600	855,000	902,000
Research samples	14	20	46
Radioisotope samples	129	96	289

Minor changes in the core configuration were necessitated by research

program changes. Figure 1.1 indicates the configuration on September 30, 1959.

Cycles of operation during this period are shown in Table 1.2.

TABLE 1.2. CYCLES OF OPERATION

Cycle No.	Date Begun	Date Ended	Accumulated Energy (Mwd)
13	In Progress	July 12	332.8
14	July 17	August 9	347.9
15	August 14	September 7	362.4
16	September 11	In Progress	307.1

Figures 1.2, 1.3, and 1.4 indicate shim rod positions versus the operating time during cycles 13, 14, and 15, respectively.

#### Radioactivity Release

On September 23 a significant amount of air activity was detected, and this required the evacuation of the ORR building for about 20 minutes. The source of the activity was  $A^{41}$  leaking into the building from the warm-drain system while the HB-3 liner was being purged with air. The valve which normally isolates the off-gas header from the warm drain was not fully closed. An analysis of the incident by Health Physics indicated that no significant exposures were received.

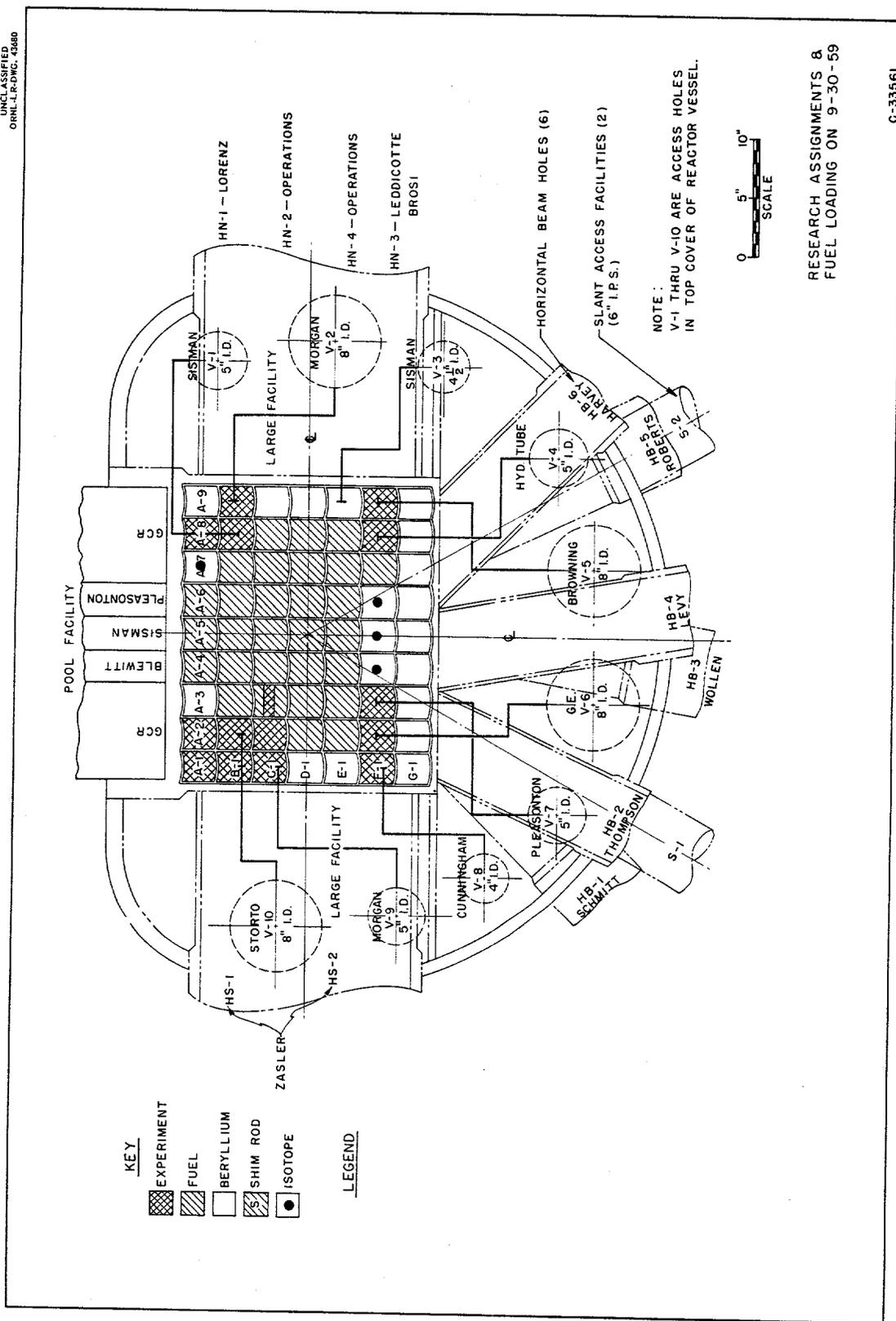
#### Shutdowns

Three major shutdowns occurred during this quarter.

July 12 through July 17. Flux mapping was done. Xenon build-up after shutdown and fuel addition tests were made.

August 9 through August 14. Experiments to determine the equilibrium

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RESEARCH ASSIGNMENTS &  
FUEL LOADING ON 9-30-59

C-33561

Fig. 1.1. ORNL Research Reactor Lattice Pattern and Experiment Locations.

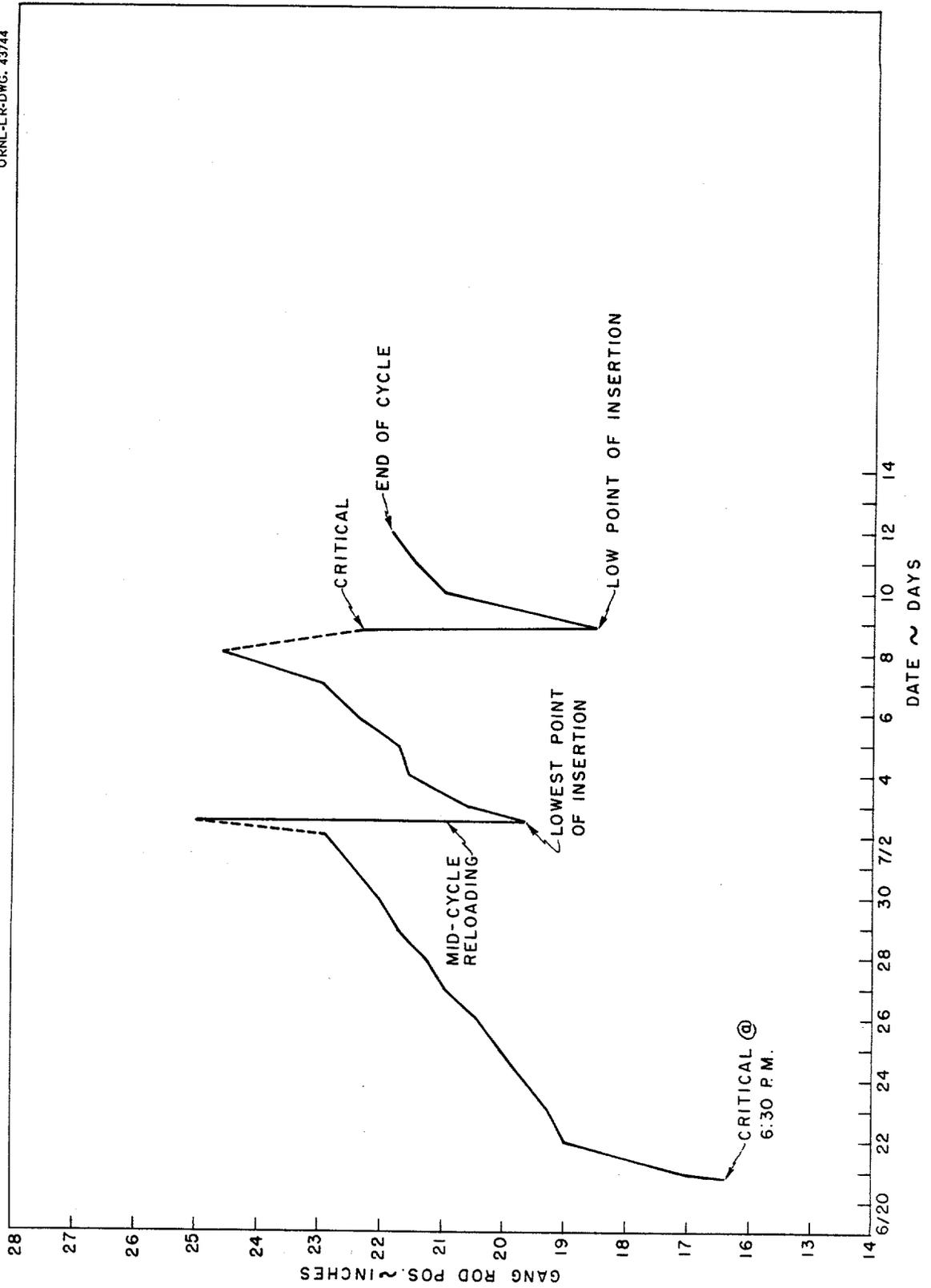


Fig. 1.2. ORR Shim Rod Positions vs Time in Cycle. Cycle XIII Operation.

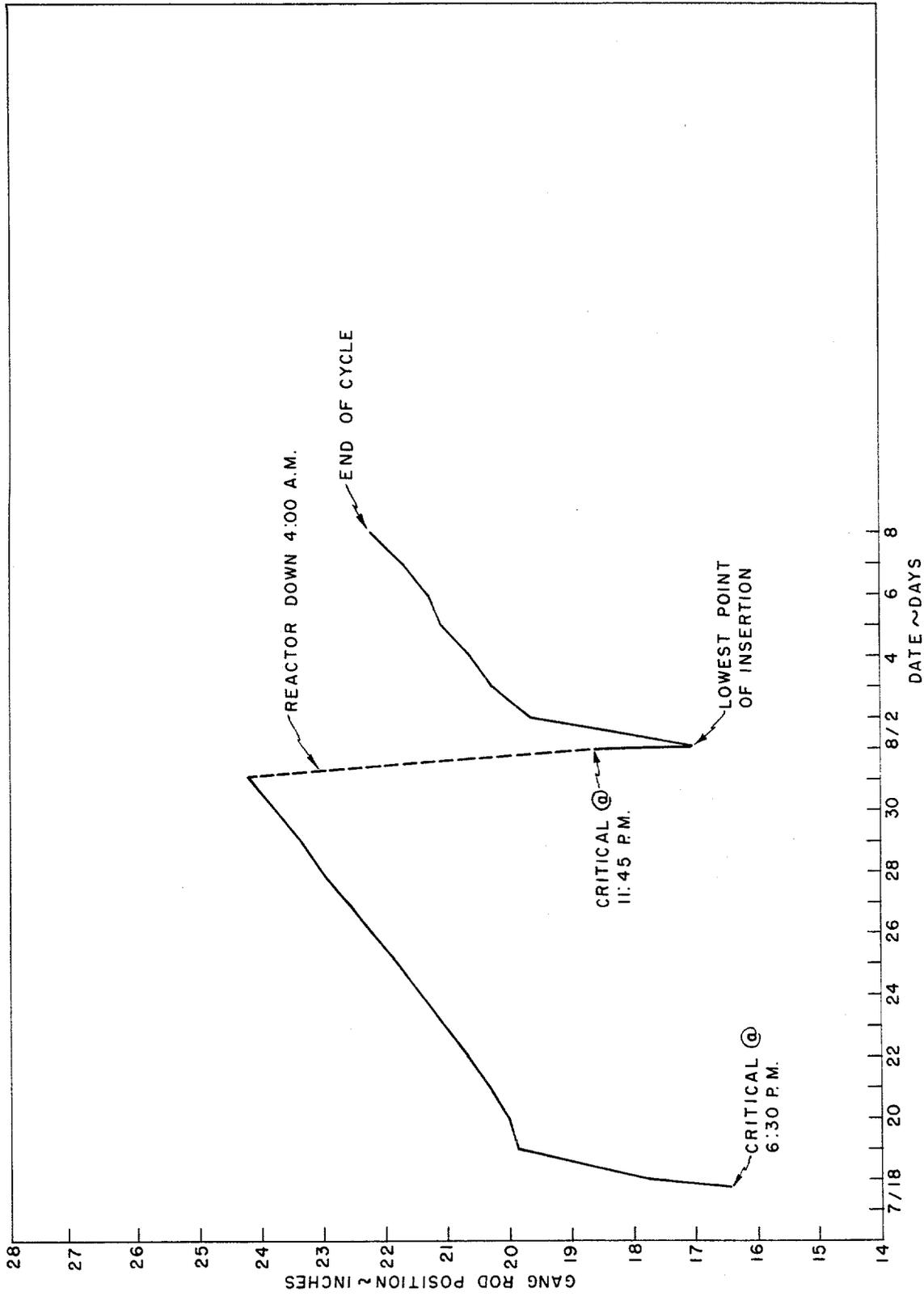


Fig. 1.3. ORR Shim Rod Positions vs Time in Cycle, Cycle XIV Operation.

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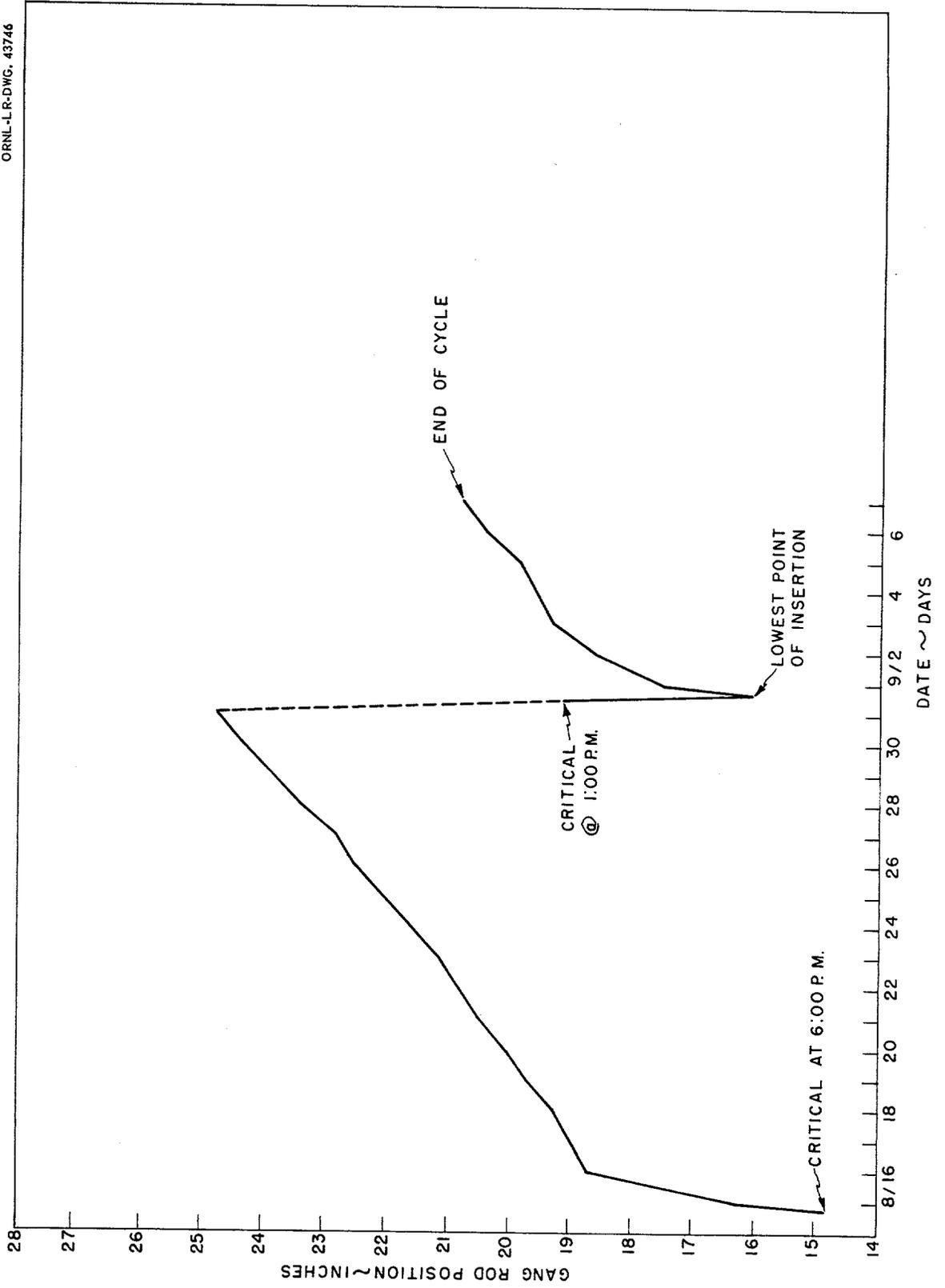


Fig. 1.4. ORR Shim Rod Positions vs Time in Cycle. Cycle XV Operation.

temperature of irradiated fuel elements in still air were performed in the hot cell. Reactivity worth measurements were made by replacing depleted fuel elements with new 200-gram elements. Beam-hole piping modifications were made.

September 7 through September 11. The first fuel loop to be operated in the ORR was removed during the September shutdown. The loop, an aqueous homogeneous type, had operated successfully since May. The fuel was removed from the reactor prior to removing the loop. This reduced the radiation level considerably and made it possible to use an LITR loop shield. Some difficulty was encountered in pulling the loop into the shield because of the close fit between the loop and the shield.

Table 1.3 lists the unscheduled shutdowns which occurred this quarter.

TABLE 1.3. UNSCHEDULED SHUTDOWNS AT THE ORR

Date	Duration (hr)	Remarks
7-1-59	0.200	Dropped #4 rod, reason unknown.
7-1-59	0.183	Dropped #3 rod, reason unknown.
7-8-59	7.667	Scrammed by northeast gamma chamber; reloaded core because of xenon.
7-9-59	0.267	Scrammed by southeast gamma chamber.
7-11-59	0.183	Dropped #3 rod.
7-17-59	0.217	HN-1 setback below $N_L$ . REED group changed set point on instrument.
7-28-59	0.283	Servo follower instrument at HN-1 gave spike upscale and scrammed reactor.
8-5-59	0.333	Scrammed due to loss of power on G.E.'s F-2, reason unknown.
9-12-59	0.200	G.E. experiment thermocouple failed. The recorder "up scale" burn-out caused a setback.
	9.533	TOTAL

Table 1.4 gives the analysis of the causes of shutdowns for this quarter.

TABLE 1.4. ANALYSIS OF ORR SHUTDOWNS

Description	Number	Down Time (hr)
Scheduled Shutdowns		
Regular, end of cycle	3	373.616
Regular, midcycle	3	41.199
Research	<u>0</u>	<u>0</u>
Subtotal	6	414.815
Unscheduled Shutdowns		
Instrument failure, reactor controls	7	8.833
Instrument failure, research	3	0.700
Power failure	<u>0</u>	<u>0</u>
Subtotal	9	9.533
TOTAL	15	424.348

Water System

Flow-decay tests were rerun on the reactor water system on July 16. The data obtained shows a decay rate comparable to that obtained on 10-11-58 as reported in ORNL CF 58-12-147. Figure 1.5 shows the results of each test.

Beam-Hole Modifications

A special collimator plug, Figure 1.6a, was installed in HB-6 during the September shutdown.

Some difficulty has been encountered with the water piping embedded in one of the beam-hole plugs. An original beam-hole plug was modified, Figure 1.6b, to permit filling and draining of the collimator without affecting the operation of the reactor. The embedded aluminum line which supplies

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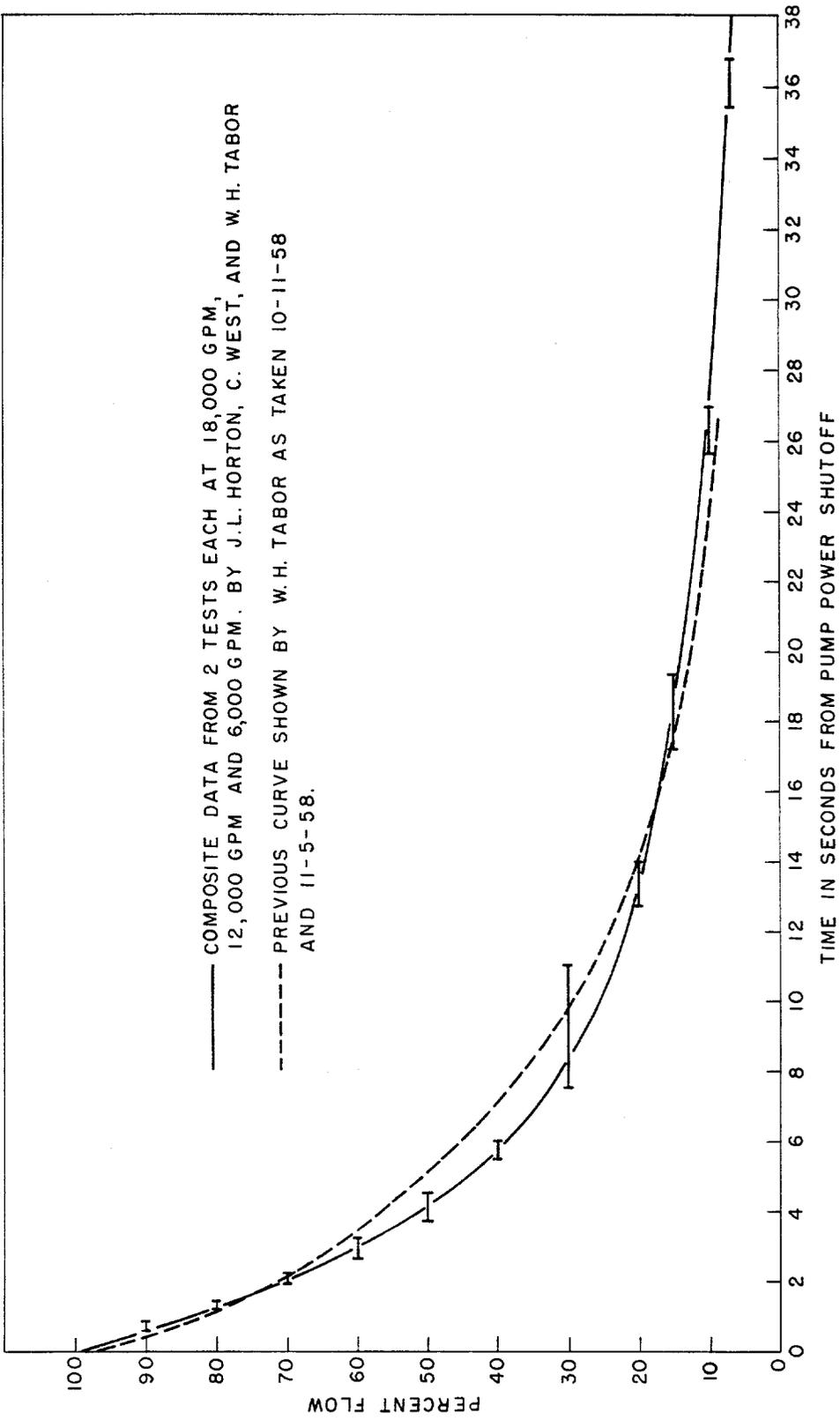


Fig. 1.5. ORR Main Cooling Water Flow Decay Tests.

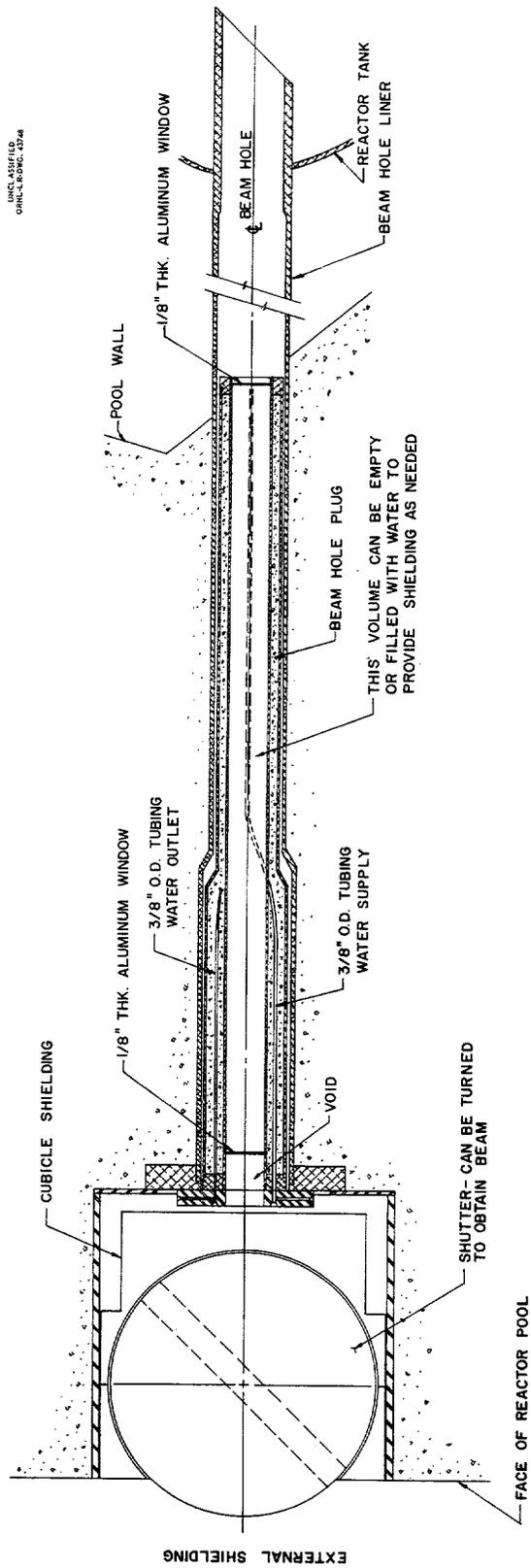


Fig. 1.6a. Section Through Beam Hole No. 6 and Modified Plug.

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THIS VOLUME CAN BE EMPTY,  
OR FILLED WITH WATER TO  
PROVIDE SHIELDING AS NEEDED.

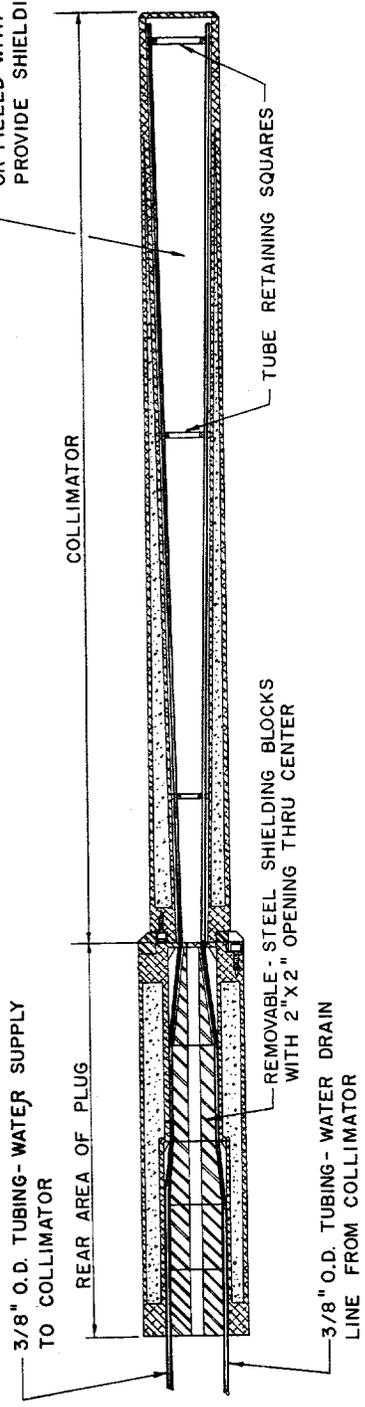


Fig. 1.6b. Typical Section Through Modified Beam Hole Plug.

cooling water to the liner became plugged. Efforts to remove the restriction in the line failed. After the concrete had been carefully chipped away from the tube, several holes were found in it. This can be seen in Figure 1.7. An investigation to determine the cause is under way.

#### Reactor Controls

The modifications to the ORR servo control system have been completed. It is now possible for the servo system to pull the shim rod in the "preferred" position when the following conditions are met: (1) the servo limit switches are in the withdraw limit, (2) the demand switch is in the raise mode, and (3) the servo is turned on. With this change, fast reactor startups can be effected more readily.

In an effort to eliminate spurious rod drops, three changes were made.

1. The minimum distance between the magnet and the keeper, just before the ball mechanism releases, was adjusted to 0.167 in. on each shim-rod drive system.
2. Two amplifiers were connected in parallel for each magnet.
3. New magnets identical to the prototype installed on #4 rod were installed on rods #3, #5, and #6.

These three changes have eliminated the dropping of the shim rods.

Gamma chambers, installed in the pool adjacent to the reactor vessel last March, failed and were removed. These were required to safeguard the reactor safety system against local neutron flux changes caused by filling and draining the beam holes. Replacement chambers were installed at the southeast and northeast corners of the reactor vessel.

#### Gamma Facility

The ORR pipe chase provides a gamma exposure facility which is now

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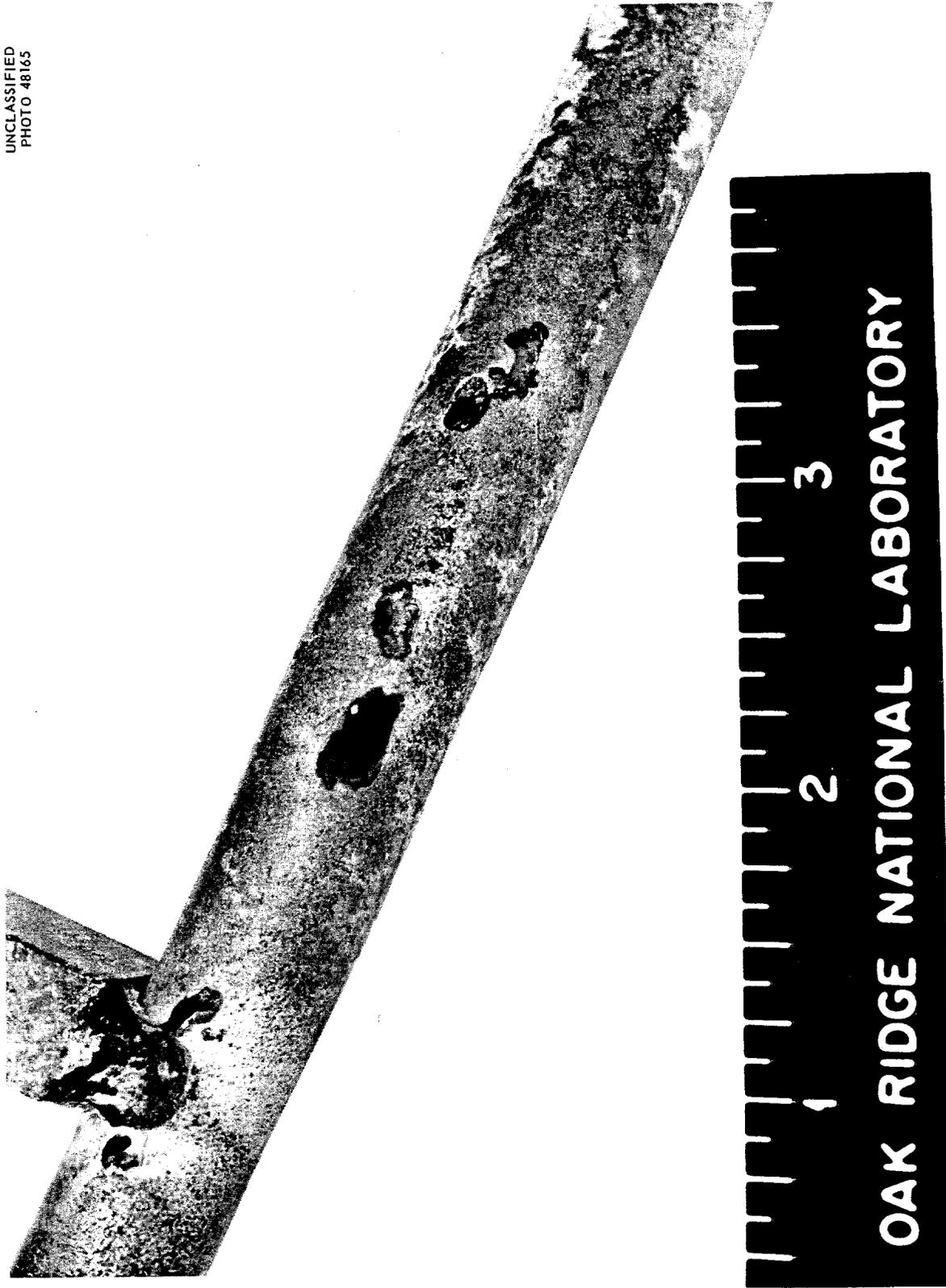


Fig. 1.7. Corroded Aluminum Piping.

available for use. Six gamma flux measurements made in this space using chemical dosimeters indicate a uniform field of  $2.2 \times 10^3$  r/hr  $\pm 10\%$  at 20 Mw.

TABLE 1.5. FACILITY ASSIGNMENT

Facility	Access Flange	Nature of Experiment	Division Sponsor
A-1, A-2	V-1	Pressurized water loop	Reactor Projects
A-8, C-2		UO <sub>2</sub> -ThO <sub>2</sub> capsule irradiations	Chemical Technology
B-1, B-2	V-10	Gas-cooled loop	Reactor Projects (GE)
B-8, E-9	V-1, V-3	Irradiation damage studies	Solid State
B-9	V-2	Gas-cooled loop	Solid State
C-1	V-9	Fuel tests	Solid State
C-3		Irradiation damage studies	Solid State
F-1	V-8	Solid fuel studies	Metallurgy
F-2	V-6	Fuel tests	Reactor Projects (GE)
F-3	V-7	Ne <sup>23</sup>	Physics
F-4, F-5, F-6, A-7		Radioisotope production	Isotopes
F-8		Hydraulic rabbit	Operations
F-9	V-5	Fused-salt loop	Reactor Chemistry
P-1, 2, 3, 7, 8, and 9		EGCR capsule irradiations	Reactor Projects
P-4		Radiation damage	Solid State
P-5		Radiation damage	Solid State
P-6		He <sup>6</sup>	Physics
HN-1		Homogeneous fuel loop	REED
HN-3		Pneumatic rabbit for activation analysis	Chemistry-Analytical Chemistry
HB-1		Magnetic analysis of fission fragments	Physics
HB-2		Neutron spectrometer	Solid State
HB-3		Neutron spectrometer	Physics

TABLE 1.5. (continued)

Facility	Access Flange	Nature of Experiment	Division Sponsor
HB-4		Neutron spectrometer	Chemistry
HB-5		Neutron spectrometer	Physics
HB-6		Time-of-flight spectrometer	Physics
HN-2, HN-4			Operations
HS-1, HS-2			Operations

## 1.2. Heat Transfer in the ORR Core at Elevated Powers

J. F. Wett, Jr.

Studies were made on heat transfer in the ORR at powers higher than 20 Mw. The following equation for wall temperature was derived.

$$T_w = T_i + 8540 \frac{P}{F} + \frac{133}{b} F^{-0.8} \left( \frac{q}{A} \right)$$

where

$T_i$  = inlet temperature,

$P$  = reactor power in megawatts,

$F$  = reactor flow in gpm,

$\frac{q}{A}$  = heat flux,

$b$  = factor dependent upon film temperature.

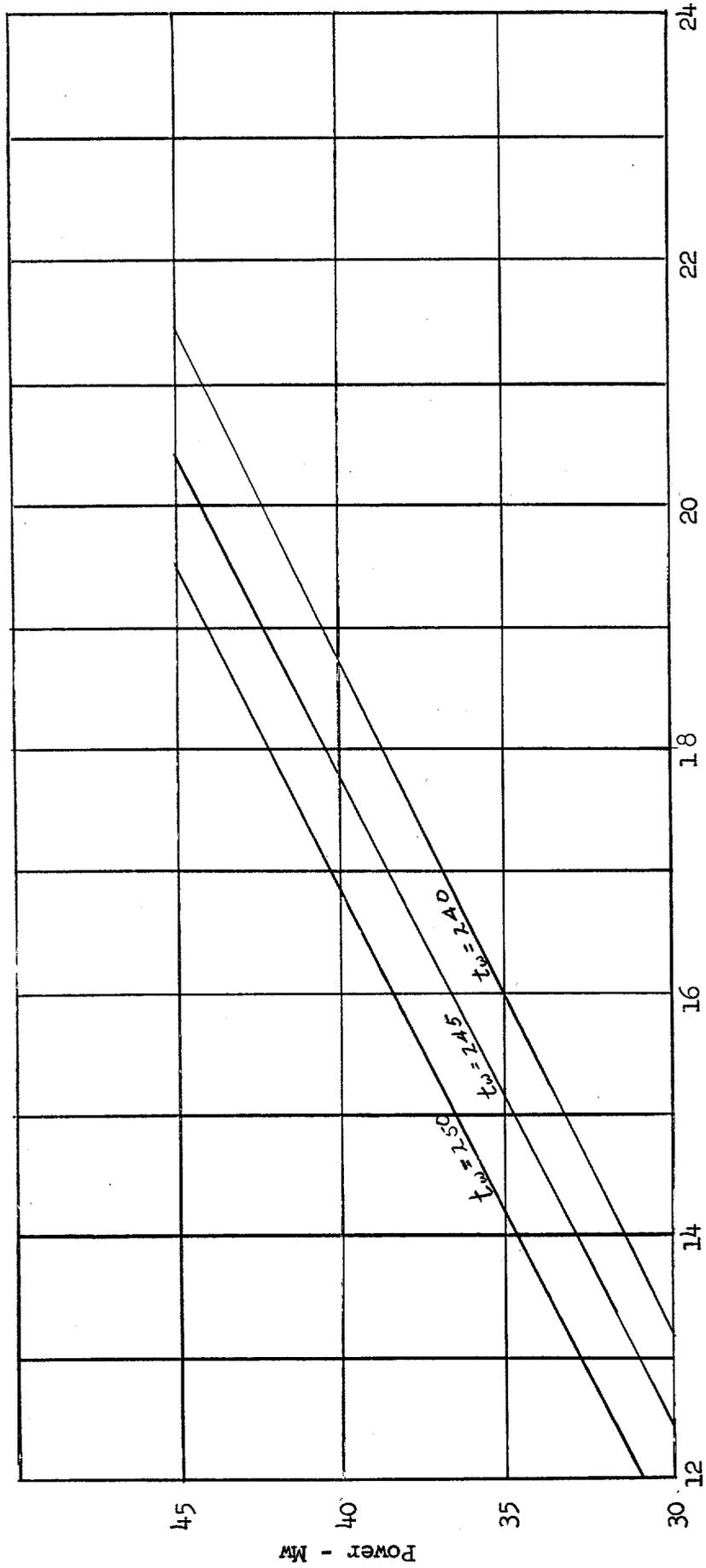
Figure 1.8 shows the maximum power levels which can be reached for various flow rates and plate surface temperatures if the inlet temperature is 110°F. Figure 1.9 shows the same thing as Figure 1.8 but with an inlet temperature of 115°F.

Examining a flow rate of 18,000 gpm, one sees that with an inlet temperature of 115°F and a plate temperature of 240°F one can attain a power level of 36.6 Mw. With an inlet temperature of 110°F and a plate temperature of 245°F, one can attain a power level of 40.6 Mw. Thus, by decreasing the inlet temperature 5°F and raising the maximum allowable plate surface temperature 5°F, one can raise the power level 4 Mw. It is further indicated that at a power of 30 Mw one can operate with a maximum plate temperature of 240°F and an inlet temperature of 115°F at a flow rate of ~14,000 gpm.

It was found that for  $T_i = 110$  and  $T_w = 240$  the maximum allowable heat flux for 45 Mw is  $8.4 \times 10^5$  BTU/hr·ft<sup>2</sup>, and for  $T_i = 110$  and  $T_w = 245$ . The

FIGURE 1.8. REACTOR POWER VERSUS REACTOR COOLANT FLOW

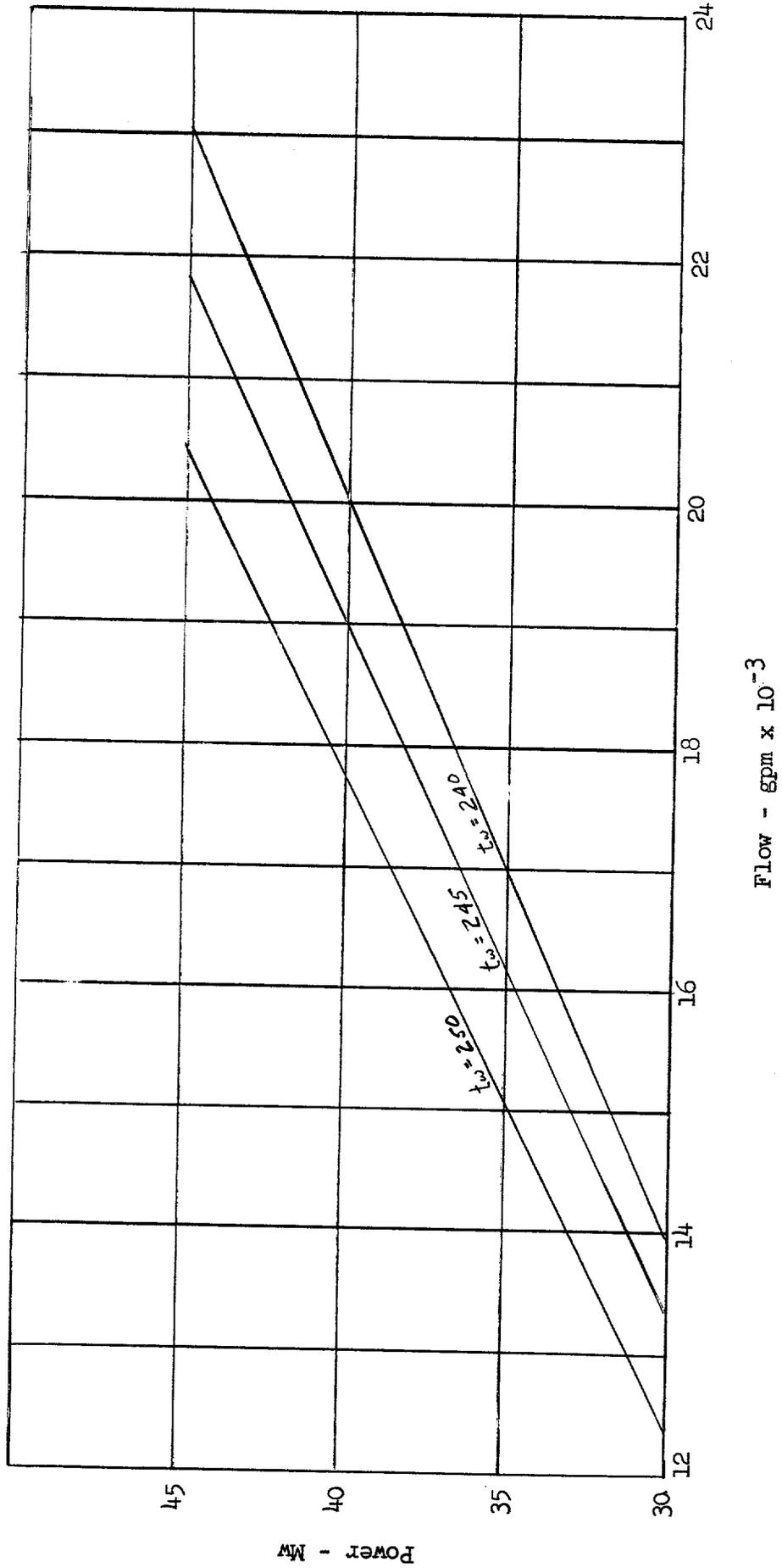
Inlet Temperature 110°F



Flow - gpm x 10<sup>-3</sup>

Figure 1.9. REACTOR POWER VERSUS REACTOR COOLANT FLOW

Inlet Temperature 115°F



Flow - gpm x 10<sup>-3</sup>

maximum allowable heat flux for 45 Mw is  $8.8 \times 10^5$  BTU/hr·ft<sup>2</sup>. At 45 Mw the average heat flux will be  $\sim 3 \times 10^5$  BTU/hr·ft. Thus, by keeping the peak-to-average thermal flux below 2.8 one can hope to attain 45 Mw with an 18,000 gpm flow rate.

A more detailed account of this work will be presented in a report to be issued in the near future.

### 1.3. Fuel Element Temperatures in Stagnant Air

J. F. Wett, Jr.

An investigation of maximum surface temperatures of irradiated ORR fuel elements cooled by stagnant air has been initiated by the Operations Division. Parameters to be varied include irradiation time, flux, fuel loading, and decay time.

To date, seven different elements have been raised into the south hot cell of the ORR. Knowing the history of the element and using Blomeke and Todd's data, one can calculate the total fission product power of the element. Such calculations were made. The plot of maximum temperature attained 16 inches from the top of the fuel section versus total fission product power is shown in Figure 1.10.

One element was raised a number of times to check temperature decay. This is shown in Figure 1.11.

Future plans call for decreasing decay time, making temperature traverses, doing dynamic calorimetry work, and attempting to develop an empirical correlation for temperature.

This work is reported in a preliminary report, CF 59-8-103.

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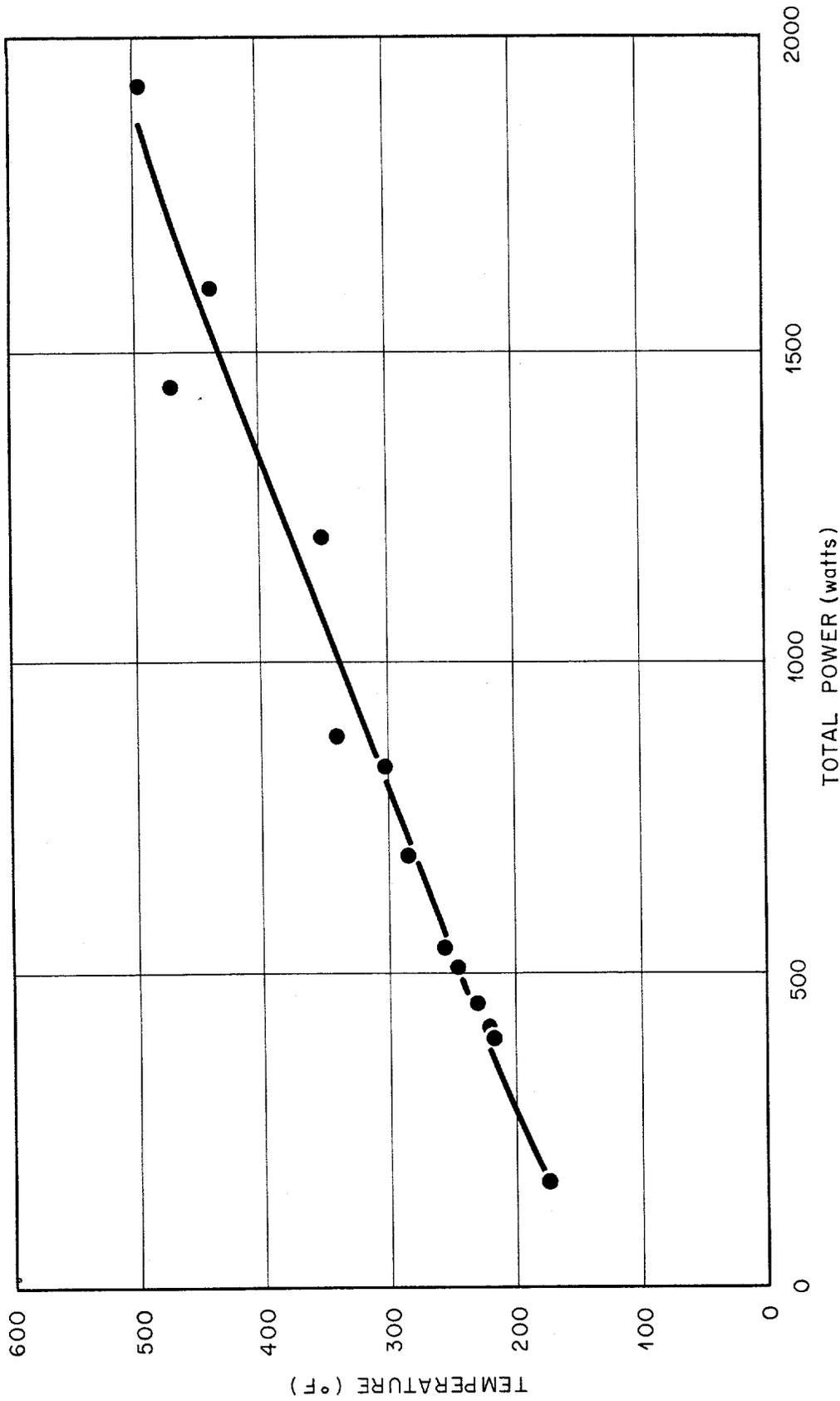


Fig. 1.10. Maximum Temperature at 16-in. Probe.

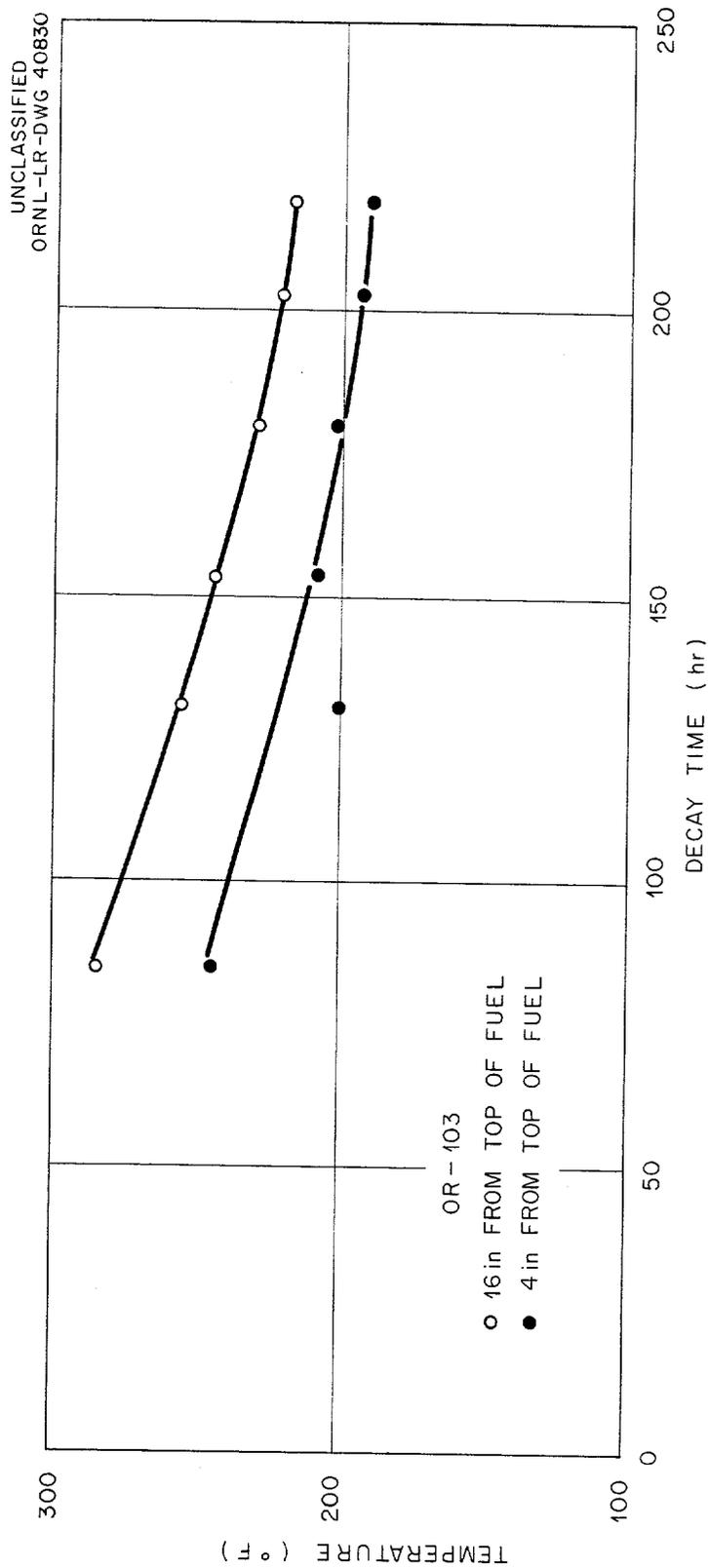


Fig. 1.11. Maximum Temperature at 16 Inches.

1.4. Fission Product Distribution in ORR Fuel Elements

A. L. Colomb

Experimental Set-Up

To measure the fission product activity along a fuel element without distortion and with a good resolution, two small graphite ionization chambers were built. These chambers were calibrated against a Victoreen model 131 integrating ionization chamber. Their geometrical characteristics and sensitivities are given in Table 1.6.

TABLE 1.6. SIZE AND SENSITIVITY  
OF THE GRAPHITE IONIZATION CHAMBERS

Chamber Designation	Size of Sensitive Volume	Sensitivity in $\frac{r}{\text{amp/hr}}$ Corrected to 22°C and 760 mm Hg
GICS	Diameter = 0.2 in. Length = 0.2 in.	$s = (8.26 \pm 0.4) \times 10^{13}$
GICL	Diameter = 0.2 in. Length = 0.4 in.	$s = (4.49 \pm 0.2) \times 10^{13}$

Figure 1.12 shows that, with these small chambers, distortion of the dose rate distribution is negligible. Curve 1 was measured with an ionization chamber 1 in. in diameter and 2.5 in. long. Curves 2 and 3 were measured with GICL and GICS respectively. The fuel element used was OR-121. These curves show that in order to measure the fission product distribution in a fuel element it is absolutely necessary to work with a chamber as small as possible and to keep the chamber in contact with the fuel element.

Measurements of the Fission Product Distribution Along Fuel Elements

The fission product activity was measured for the following elements:

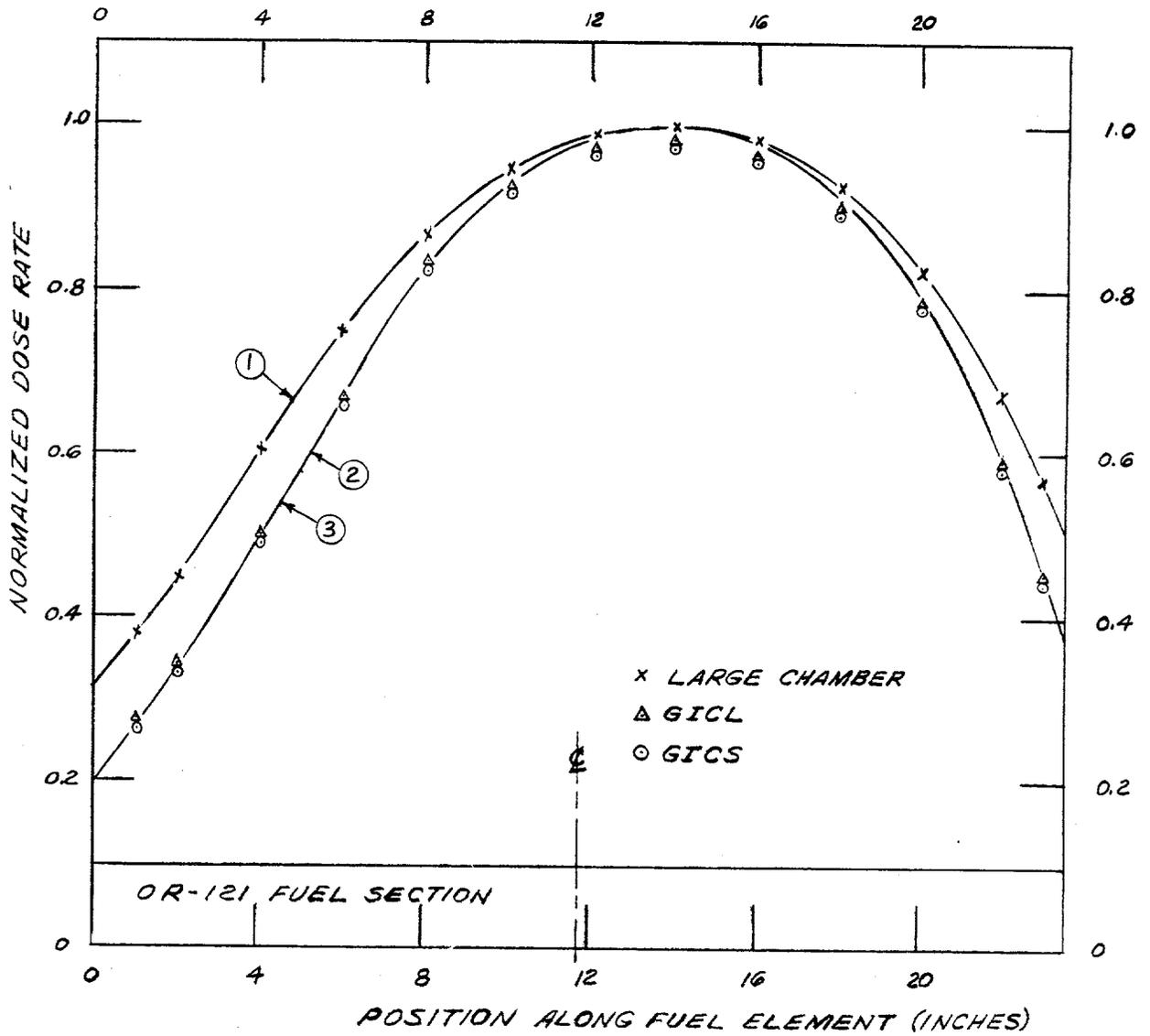


Fig. 1.12. Distortion in the Dose Rate Distribution Due To Chamber Size.

OR-158, OR-159, OR-150, and OR-121. Table 1.7 summarizes the irradiation history of these elements. The cooling time given in Table 1.7 is the time elapsed between the end of a given irradiation and the measurement. The U-235 consumption is obtained from the reactor power, the irradiation time duration, and the reactor position factor.

TABLE 1.7. IRRADIATION HISTORY OF THE MEASURED FUEL ELEMENTS

Element No.	Irradiation Time in Days	Power Mw	Position in Reactor	Cooling Time in days	U-235 Consumption in grams
OR-158	20	16	C-7	23	16
OR-159	20	16	E-7	23	18
OR-150	10	16	E-7	45	9
	20	16	D-7	23	18
OR-121	14	20	D-1	137	4
	21	20	D-3	113	25
	11	20	B-5	84	10
	14	16	C-6	63	11
	4	16	D-2	53	2
	10	16	D-5	25	7

The normalized fission product distributions for these four elements are shown in Figure 1.13 through 1.16. It is possible to see from these figures that the fission product distribution is not symmetrical about the element horizontal center plane. This is not surprising because the reactor flux is not symmetrical about its central plane, and some of the elements that have received more than one irradiation were not inverted. Fuel elements should, if possible, be inverted from one irradiation to the other. This not only prolongs their

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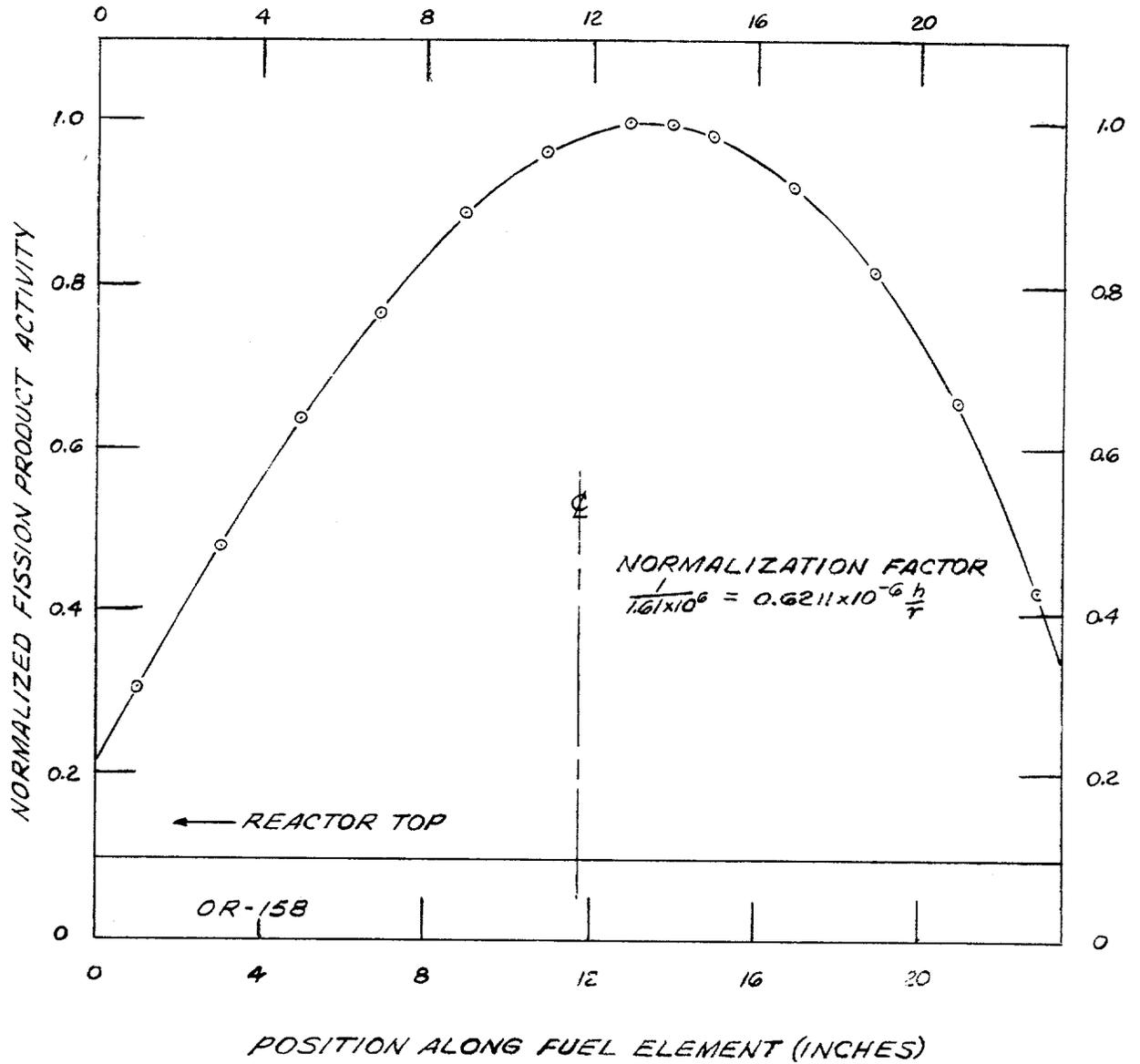


Fig. 1.13. Fission Product Distribution in Fuel Element OR-158.

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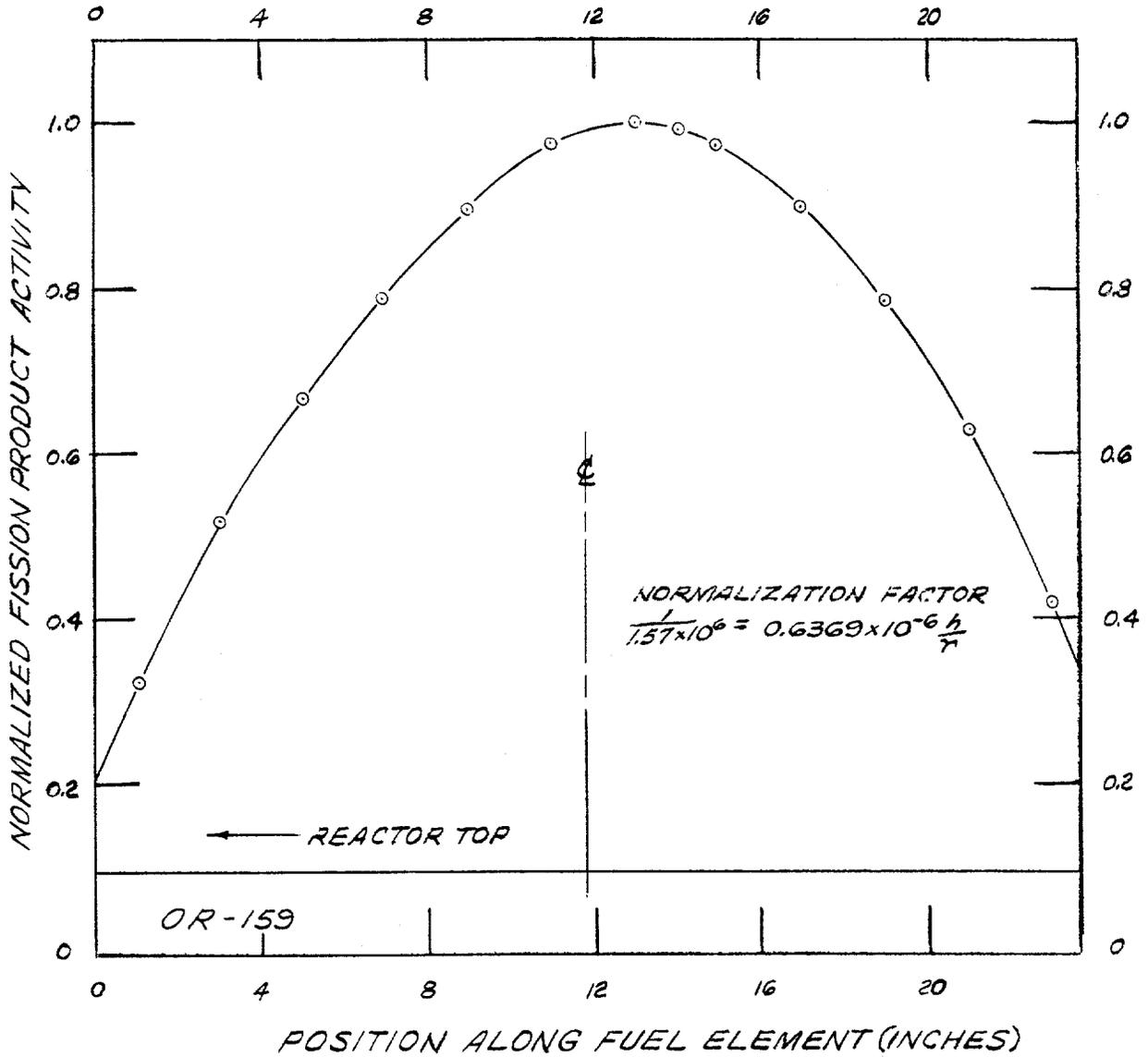


Fig. 1.14. Fission Product Distribution in Fuel Element OR-159.

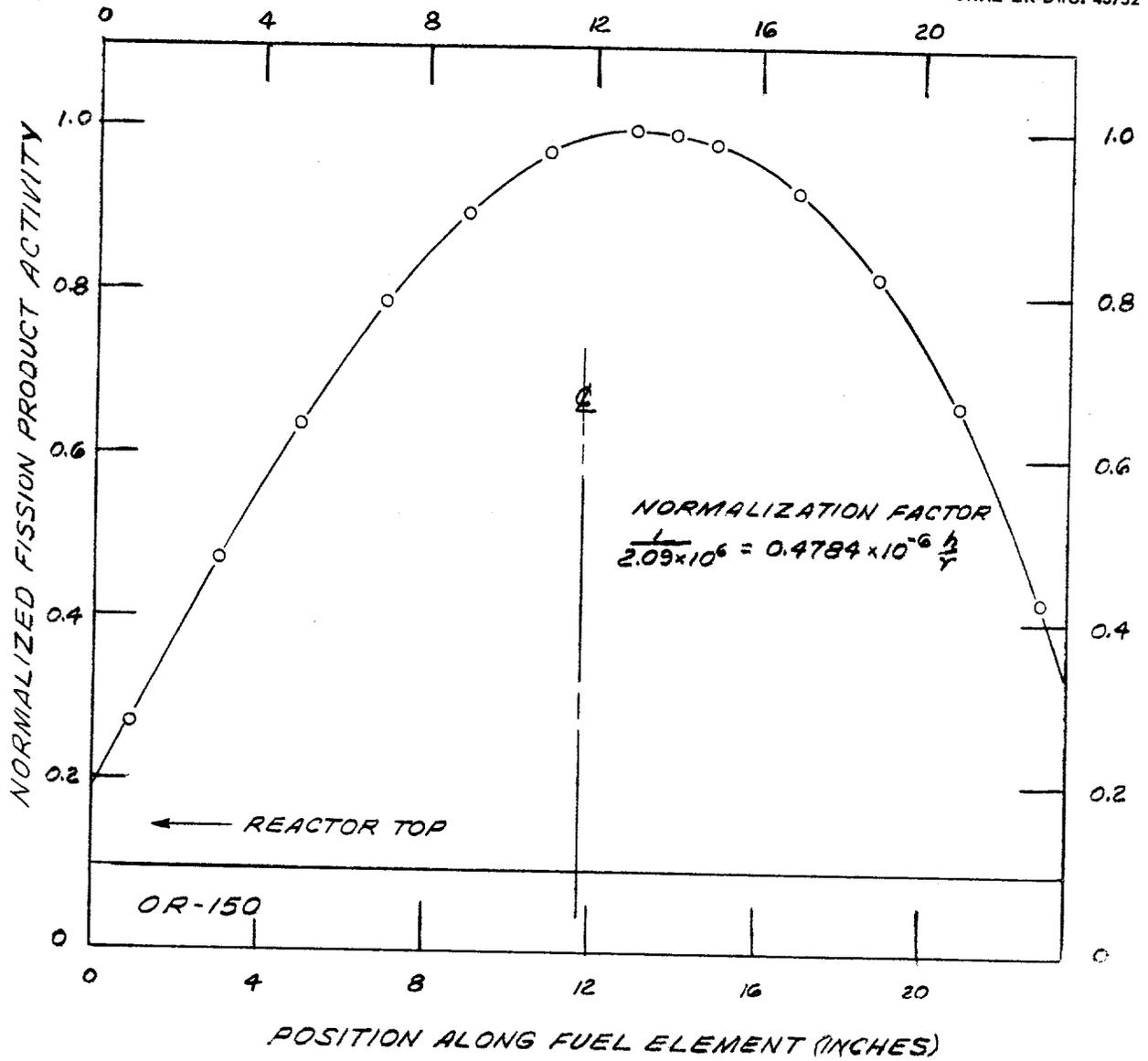


Fig. 1.15. Fission Product Distribution in Fuel Element OR-150.

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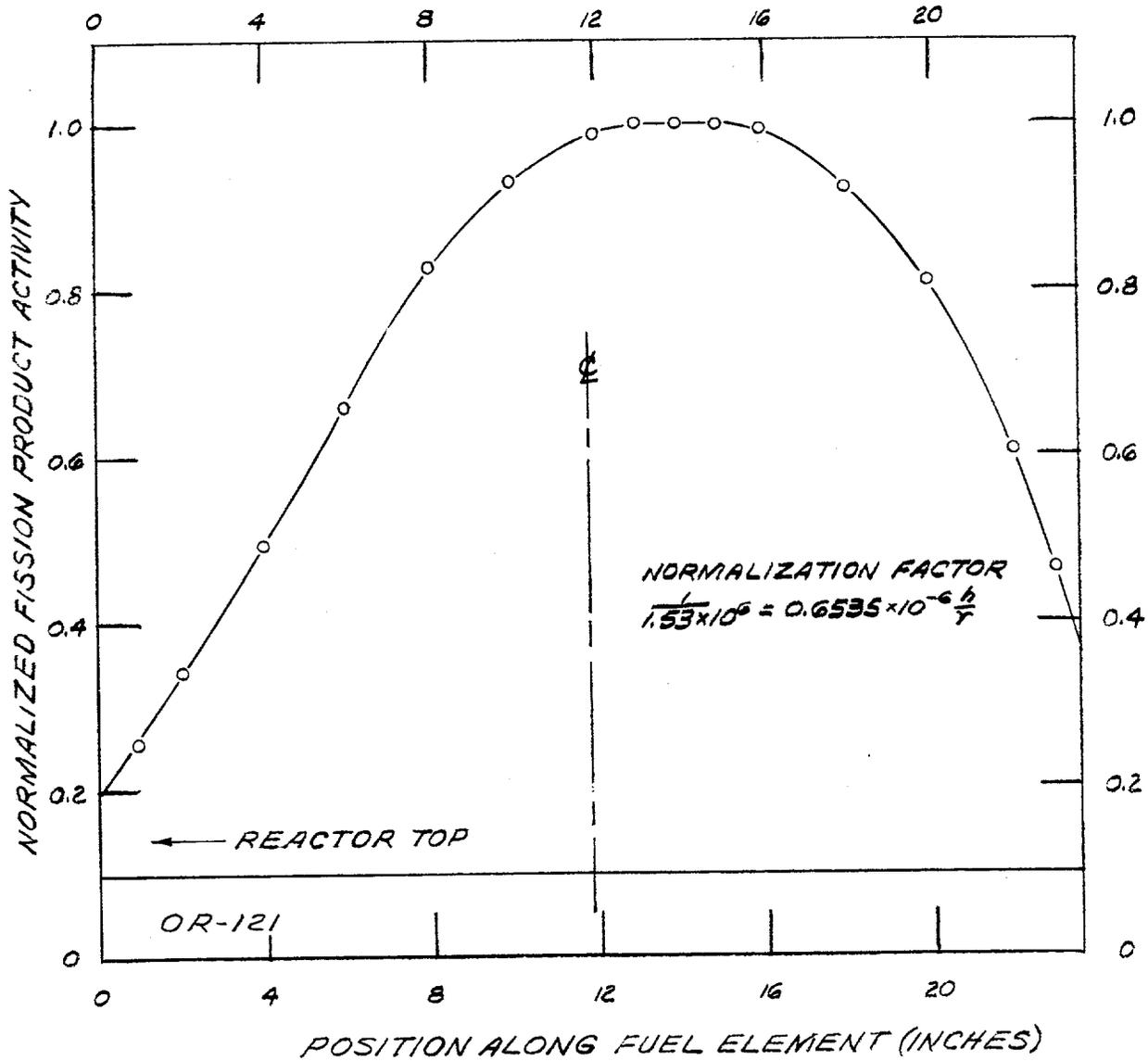


Fig. 1.16. Fission Product Distribution in Fuel Element OR-121.

life (this effect is not very important, however, being smaller than 5%) but also helps keep the maximum flux as close as possible to the reactor horizontal center plane (important in getting as great a flux as possible in the experiment facilities).

Fission product distribution measurements can be used to determine if the calculated values for U-235 consumption are correct. The value of the integral over the distribution curve is proportional to the total amount of fission products accumulated in the fuel element. This is, in turn, proportional to the total amount of U-235 consumed.

Table 1.8 shows the value of these integrals over the normalized distributions; the third column contains the value of the second multiplied by the inverse normalization factor.

TABLE 1.8. INTEGRALS OVER FISSION PRODUCT DISTRIBUTIONS

Element No.	Integral of Normalized Distribution	Integral of Distribution (in.)(r) hr
OR-158	17.6	$2.84 \times 10^7$
OR-159	17.7	$2.78 \times 10^7$
OR-150	17.6	$3.69 \times 10^7$
OR-121	17.6	$2.69 \times 10^7$

It is interesting to notice that the values of the integrals over the normalized distributions are, within the experimental errors, independent of the number of irradiations given to the element. This indicates that the shape of the U-235 consumption distribution along a given element is independent of the initial U-235 concentration or, in other words, that the flux is always inversely proportional to the U-235 concentration.

It is now possible to check if the U-235 consumption calculated in the conventional manner is right. A relation between the amount of U-235 burned and the value of the integral over the fission product activity distribution can be obtained from the fuel elements OR-158 and OR-159 that had had only one irradiation. For twenty-three days cooling time between the end of the irradiation and the time of the activity measurement this relation is:

$$1 \text{ gram U-235 consumption corresponds to } 1.65 \times 10^6 \frac{(\text{in.})(r)}{\text{hr}} .$$

Table 1.9 presents the contribution of each irradiation to the total integral. Cooling times were all reduced to twenty-three days in order to be able to use the relation gained from elements OR-158 and OR-159. Fission product decay data were taken from Perkins and King.<sup>1</sup>

TABLE 1.9. CONTRIBUTION FROM EACH IRRADIATION TO THE TOTAL AMOUNT OF FISSION PRODUCTS

Element No.	U-235 Consumption (gr)	Decay from 23 d. to Total Cooling Time	Fraction of Integral in $\frac{(\text{in.})(r)}{\text{hr}}$
OR-150	9	0.50	$7.45 \times 10^6$
	18	1	$2.97 \times 10^7$
TOTAL			$3.72 \times 10^7$
OR-121	4	0.134	$0.88 \times 10^6$
	25	0.192	$7.92 \times 10^6$
	10	0.225	$3.72 \times 10^6$
	11	0.337	$6.13 \times 10^6$
	2	0.340	$1.12 \times 10^6$
	7	0.912	$1.05 \times 10^7$
TOTAL			$3.03 \times 10^7$

<sup>1</sup>J. F. Perkins and R. W. King, "Energy Release from the Decay of Fission Products," Nucl. Sci. and Eng., 3, 726-746 (1958).

It is interesting to see that the totals obtained on Table 1.9 correspond within 12% to that of Table 1.8. This means that the total consumption of U-235 calculated by the conventional way is right within these limits of error. These results also show that the fission product activity measurement represents a good method to determine experimentally the U-235 consumption in fuel elements. This method could certainly be made quite accurate. Calibration could be done by measuring the neutron flux in a given new element at the beginning and at the end of an irradiation in order to compute accurately the amount of U-235 burned. Then gamma activity measurements at different times after the end of irradiation will give the relation between grams of U-235 burned and gamma activity as a function of cooling time.

An advantage of this method is that, knowing the distribution of U-235 and fission products in the fuel element, it is possible to compute the variation of the infinite multiplication factor and the macroscopic absorption cross-section along the element. The knowledge of these quantities can be of great help if some flux deformation (flattening or pushing the flux toward an experiment) are to be done. Figure 1.17 shows the variation of the infinite multiplication factor and the macroscopic absorption cross-section along fuel element OR-121.

These curves were obtained by the following calculation.

For a new element with 200 grams U-235

$$\sum_a^0 = \sum_{H_2O + Al} + \sum_a^u = \sum_{H_2O + Al} + N_{25}^0 \sigma_{25}$$

where  $N_{25}^0$  is the atomic density of U-235 before irradiation.

After an irradiation of time  $t$  at flux  $\phi$ ,

$$\sum_a^t = \sum_{H_2O + Al} + N_{25}^0 \sigma_{25} e^{-\sigma_{25}\phi t} + N_{F.P} \sigma_{F.P} + N_{Sm} \sigma_{Sm}$$

where  $N_{F.P}$  and  $N_{Sm}$  are the atomic densities of fission products and samarium respectively.

$$N_{FP} = \frac{2 N_{25}^0}{1 + \alpha_{25}} (1 - e^{-\sigma_{25}\phi t})$$

and at samarium saturation<sup>(2)</sup>

$$N_{Sm} = 3.21 \times 10^{-4} N_{25}^0 (1 - e^{-\sigma_{25}\phi t})$$

The following cross section values were used:

$$\sigma_{25} = 593 \text{ b}$$

$$\sigma_F = 502 \text{ b}$$

$$\sigma_{Al} = 0.204 \text{ b}$$

$$\sigma_{H_2O} = 0.586 \text{ b}$$

$$\sigma_{FP} = 38 \text{ b}$$

$$\sigma_{Sm} = 5.3 \times 10^4 \text{ b}$$

The infinite multiplication factor for a new element is:

$$k_{\infty}^0 = \frac{\gamma \sum_f}{\sum_a^0}$$

and after an irradiation of time  $t$  at flux  $\phi$

$$k = \gamma \frac{N_{25}^0 \sigma_F e^{-\sigma_{25}\phi t}}{\sum_a^t}$$

<sup>(2)</sup>S. Glasstone, M. C. Edlund, "The Elements of Nuclear Reactor Theory,"  
D. Van Nostrand Company, Inc., Princeton, N. J., (1952): 338-339.

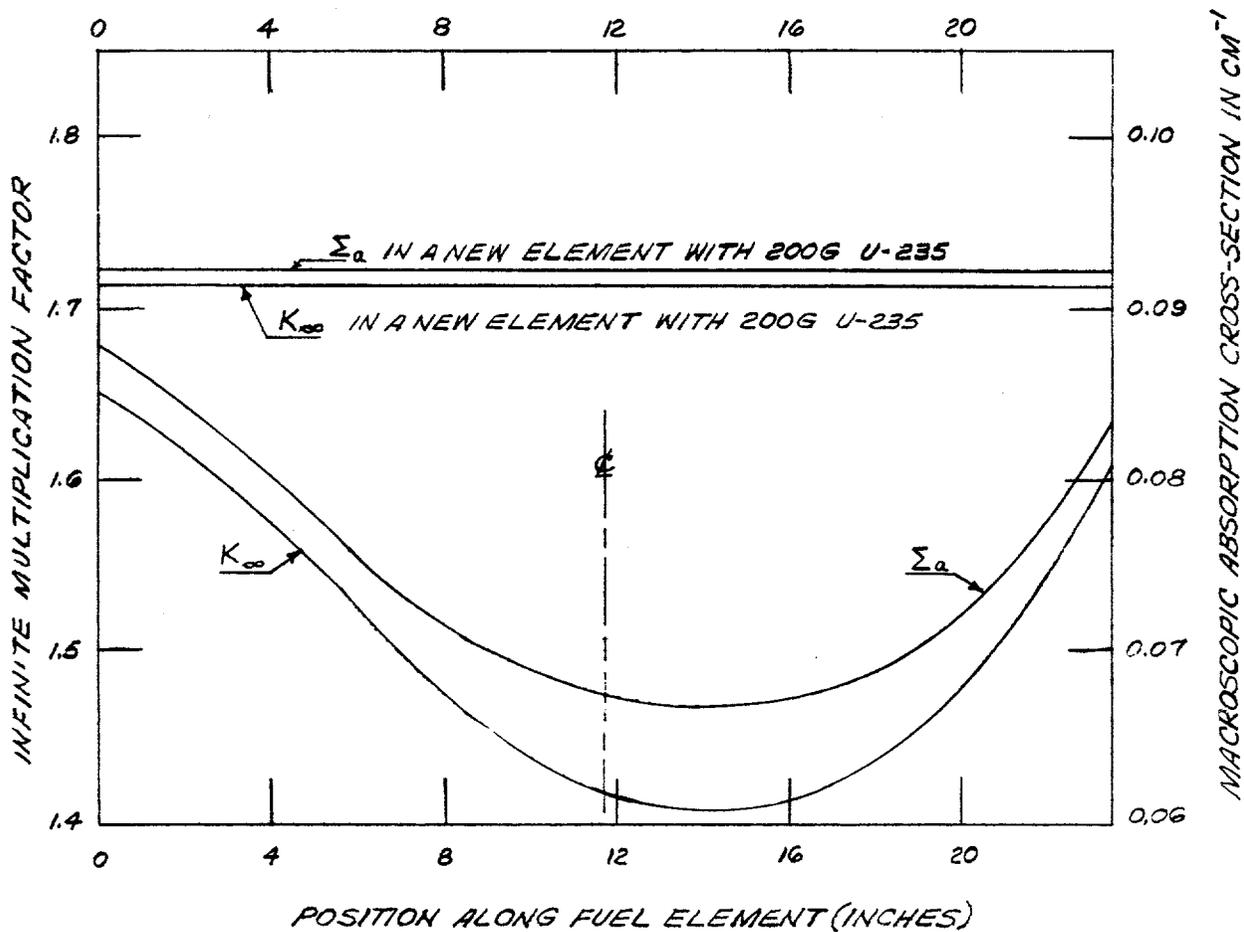


Fig. 1.17. Infinite Multiplication Factor and Macroscopic Absorption Cross-Section Variation Along Fuel Element OR-121.

### Measurement of the Fission Product Distribution Along Shim Rods

The fission product activity was measured along two shim rods (i.e., OR-2-S and OR-6-S) by the same method that was described previously. The results were found to be almost the same for the two shim rods, so only rod OR-2-S will be discussed here.

Shim rod OR-2-S was in the reactor in position D-6 from 11-4-58 to 2-23-59 and in B-6 from 2-28-59 to 6-16-59. The total amount of U-235 burned, as computed by the conventional way, was 85 grams. The fission product distribution measured is plotted on Figure 1.18. It is interesting to notice that most of the fission products are accumulated in the upper half of the rod. This is due to the fact that this section is always in the reactor; whereas, the lower section is out of the reactor at the beginning of an irradiation cycle and slowly moves inward during the cycle.

Using the same methods as in the fuel element measurements it is now possible to obtain the macroscopic absorption and the infinite multiplication variation along the shim rod. These results are shown in Figure 1.19. It is striking to see that, at the minimum,  $k_{\infty}$  is only equal to 0.4 and that even the average of  $k_{\infty}$  over the length of the element is smaller than 1. The consequence of this is that the fuel section of this shim rod does not add anything appreciable to its control capacity. It is also interesting to note that this rod was burned down to 35% of its original U-235 loading although the normal fuel elements are never burned down to less than 50% of their original U-235 content.

In order to obtain an idea of what maximum total burnup should be allowed in the fuel section of ORR shim rods, the U-235 concentration, the macroscopic absorption cross-section, and the infinite multiplication were calculated as

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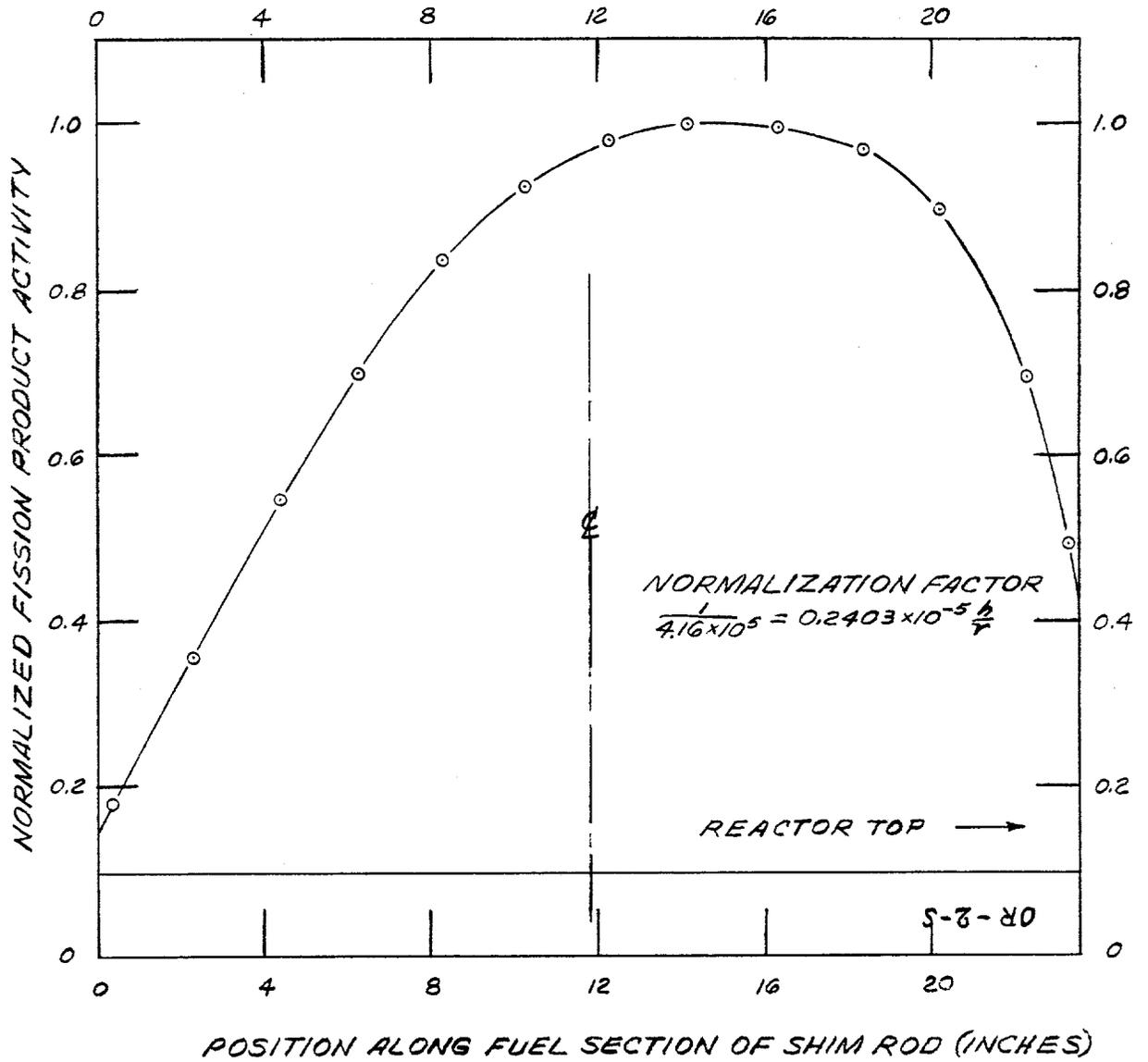


Fig. 1.18. Fission Product Distribution Along the Fuel Section of Shim Rod OR-2-S.

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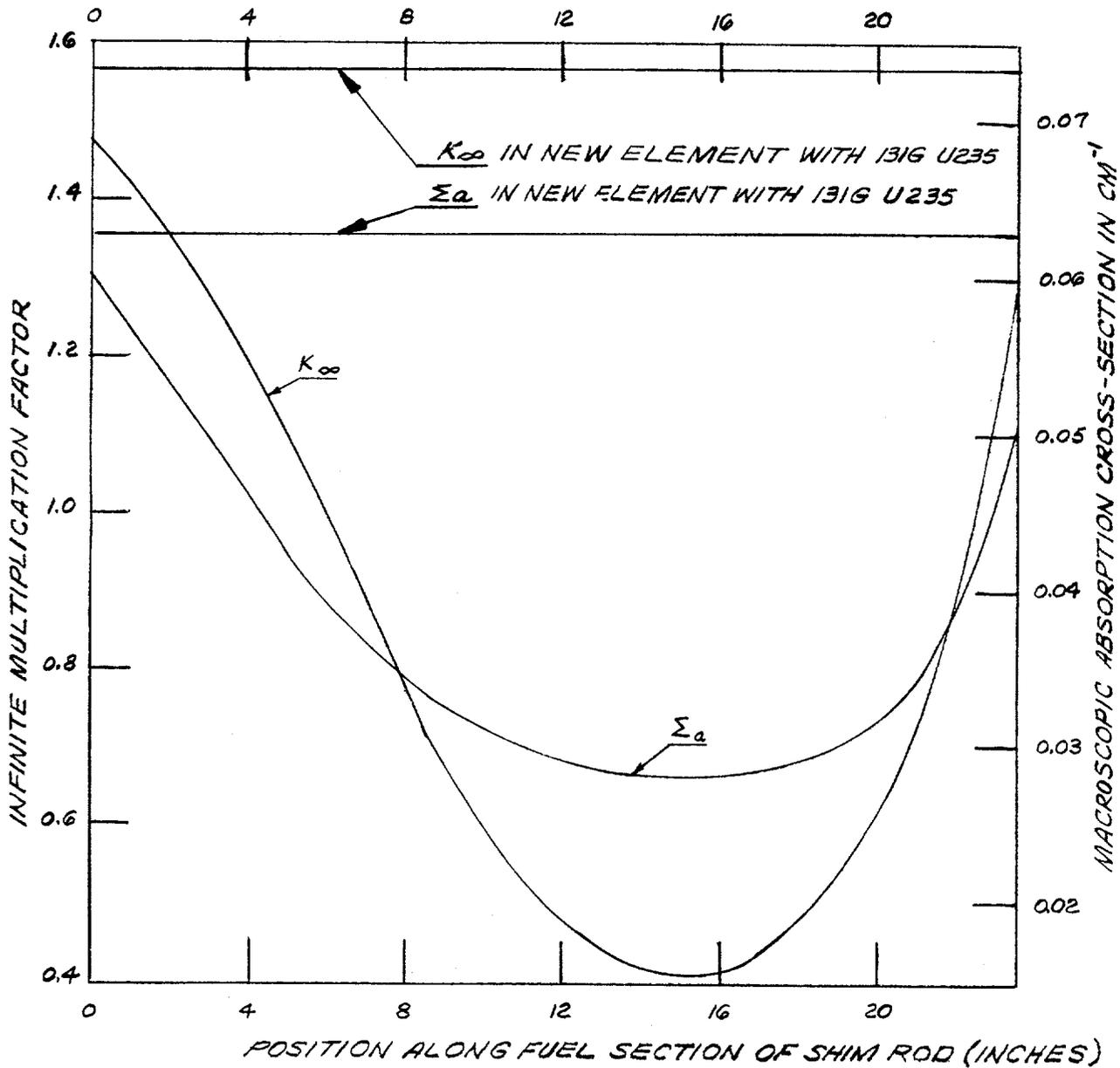


Fig. 1.19. Infinite Multiplication Factor and Macroscopic Absorption Cross-Section Variation Along the Fuel Section of Shim Rod OR-2-S.

functions of the total burnup. These quantities are plotted against position along the fuel section in Figures 1.20, 1.21. and 1.22. It can be seen from these results that a maximum allowable burnup of 40 to 50 grams is an adequate value. This means that if the advantages given by the fuel section (minimizing of flux peaking and more control per rod) are to be kept the shim rods should be changed every 4 to 5 operation cycles.

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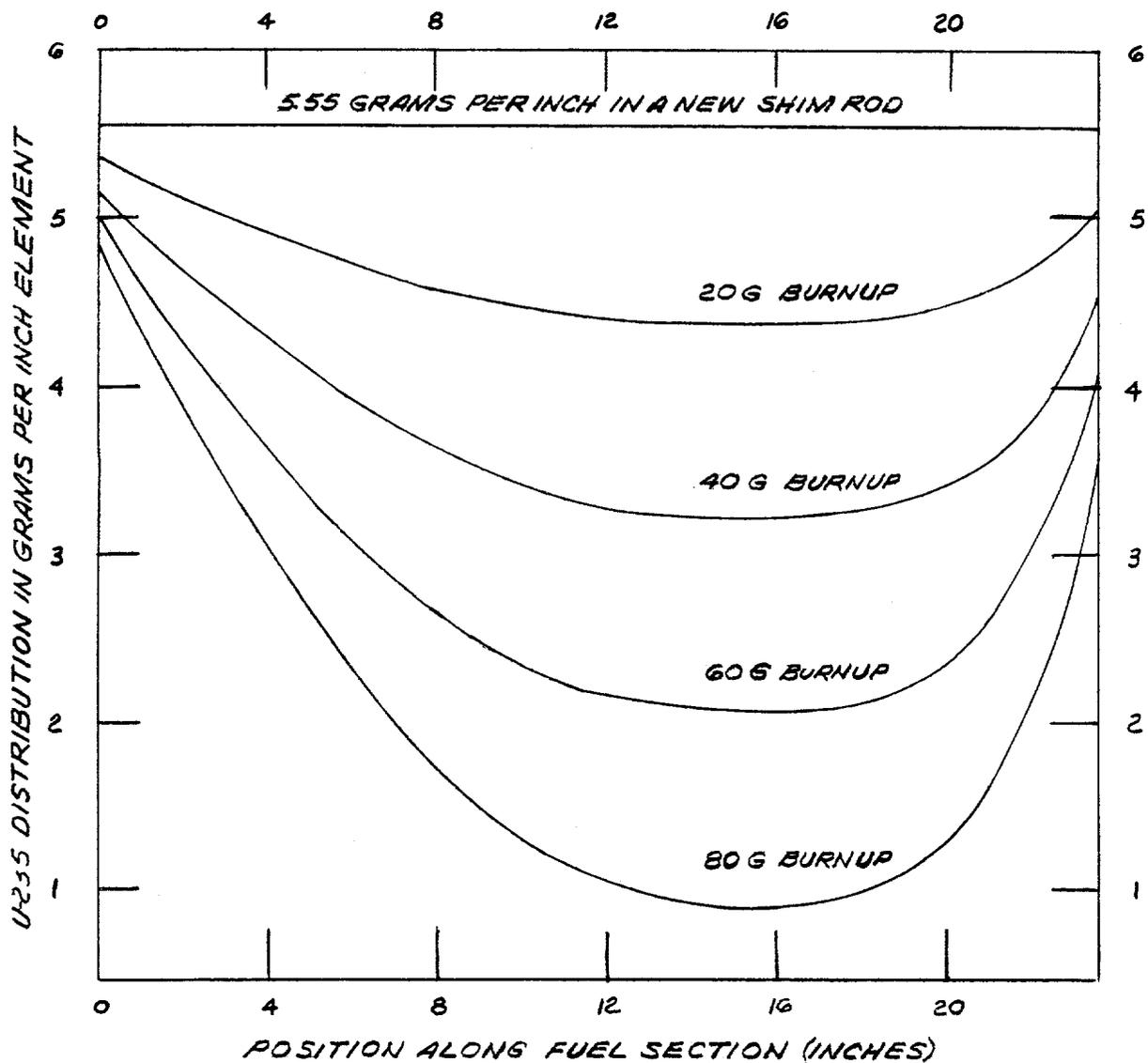


Fig. 1.20.  $U^{235}$  Concentration Along the Fuel Section of an ORR Shim Rod as a Function of Total Burnup.

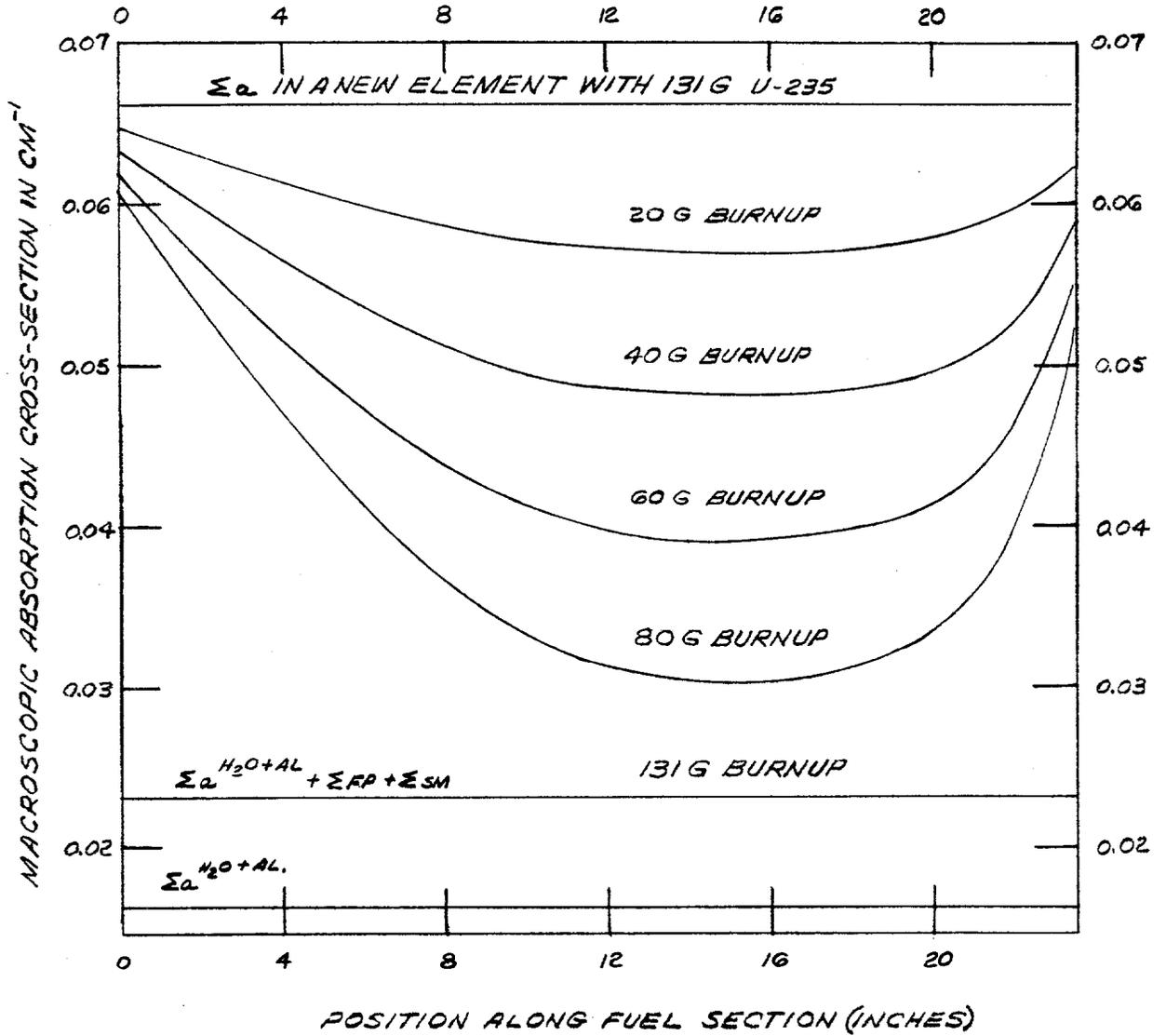


Fig. 1.21. Macroscopic Absorption Cross Section Along the Fuel Section of an ORR Shim Rod as a Function of Total Burnup.

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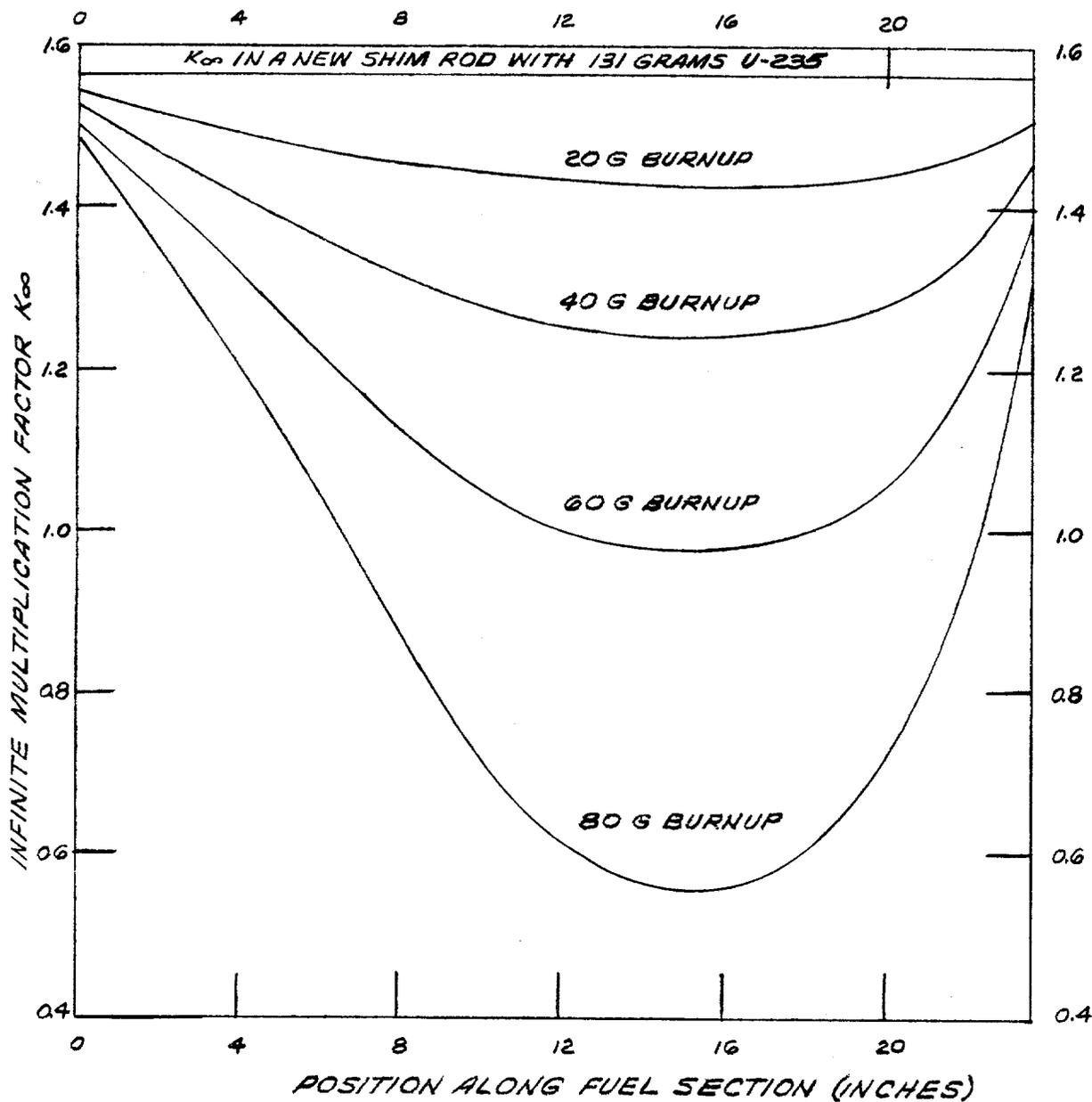


Fig. 1.22. Infinite Multiplication Factor Along the Fuel Section of an ORR Shim Rod as a Function of Total Burnup.

## 1.5. Core Measurements in the ORR

C. D. Cagle

The ORR is fueled with elements which nominally contain 200 grams of  $U^{235}$  evenly distributed along the active length when new. These are used until approximately 70 grams of the  $U^{235}$  has been depleted which requires from seven to nine cycles, depending upon the position in the core. During this depletion the element acquires a very unfavorable distribution of the remaining fuel and of the fission-product poisons resulting from the burnup. The remaining fuel is concentrated in the less important ends, and the poisons are concentrated in the important central region.

In order to determine the effect of this distribution of fuel and poisons upon reactivity, a comparison was made by replacing elements burned down from 200 grams to about 150 grams of  $U^{235}$  with a new element containing 150 grams of  $U^{235}$ . The new 150-gram element increased the reactivity by about 0.5%  $\Delta k/k$  or the equivalent of approximately 50 grams of  $U^{235}$ , indicating that a 200-gram element burned down to 150 grams is worth the same as a new 100-gram element as far as reactivity is concerned.

During this series of measurements, a check of the group shim-rod calibration was made by replacing the new 150-gram element with a new 200-gram element. This measurement, although not conclusive since it was made in only one core position, indicated that the group shim rod calibration has not changed appreciably from that obtained during the initial startup measurements. Measurements which have been made by replacing burned down elements with a new 200-gram element gave an erroneously low value for the rods since the equivalent of up to 100 grams of fuel was being added to the core although

the actual amount was only about 50 grams. Further measurements are to be made.

An attempt was made to obtain an empirical relation to predetermine the approximate group rod displacement caused by fuel additions in the various positions in the core. Such relations have been developed in the past but have become inapplicable due to the difference in fuel distributions in new and used fuel elements. It was found that the following relation applies if a fuel element with greater than 40 grams burnup is replaced with a new 200-gram element:

$$(\Delta r)_{\text{gang}} = 0.012 \times M_i \frac{\bar{\phi}_i}{\bar{\phi}_c}$$

where

$(\Delta r)_{\text{gang}}$  = displacement (in inches) of the shim rods moved groupwise,

$\Delta M_i$  = the weight (in grams) of the fuel added to position,  $i$ , by the change, and

$\frac{\bar{\phi}_i}{\bar{\phi}_c}$  = the ratio of the average flux in the core position to the average flux of the core.

If the displaced element has no more than about 10 grams burnup, the following relation seems to hold:

$$(\Delta r)_{\text{gang}} = 0.022 \times M_i \times \frac{\bar{\phi}_i}{\bar{\phi}_c} .$$

Further measurements are planned to develop a more general relationship of these quantities.

## 1.6. Corrosion Studies

L. E. Stanford

In a water-cooled reactor system constructed primarily of aluminum, the degree of water purity is important not only in influencing the induced radioactivity which must be dealt with in the cooling system, but also in minimizing the rate of aluminum corrosion. A number of investigations have been made of the influence of various ions on the corrosion of aluminum in water; and some of these indicate that the corrosion rate is substantially increased by the addition of trace amounts of copper ions to a water system. Unfortunately, very little data have been obtained for systems utilizing high purity water such as that used in the ORR or LITR water systems. However, the expense of extensive corrosion in a reactor system is so great that the use of copper, or copper alloys, in the water systems has been severely restricted at the LITR and completely prohibited at the ORR. Consequently, the cost of reactor and experiment components is very high and sometimes prohibitive. It is, therefore, desirable to establish more realistic criteria for use in the quality control of materials to be used in reactor water systems. Accordingly, tests are being set up to study the corrosion of aluminum in ORR water containing copper ions, metallic copper, metallic tin, and tinned copper with exposed copper surfaces.

A test stand for some of these tests has been fabricated and is presently being installed. The test-stand water system is constructed of glass, stainless steel, and aluminum. Reactor water from the ORR facility pump discharge will flow through the test specimen containers at a rate which provides a complete volume change approximately every six minutes. Discharge water from the system will be released to the drains. Sampling taps have been provided at appropriate points so that water samples may be taken for routine water analysis.

2. ORNL GRAPHITE REACTOR

2.1. Operations

W. R. Casto

Operations

The operating data for the ORNL Graphite Reactor are given in Table 2.1.

TABLE 2.1. GRAPHITE REACTOR OPERATIONS

Period July 1, 1959, through September 30, 1959

	This Quarter	Last Quarter	Year to Date
Total energy, Mwd	280.8	275.1	837.0
Average power, kw/operating hour	3419	3417	3425
Time operating, %	89.2	88.5	89.5
Exit air filters, $\Delta p$ , in. $H_2O$ (av)	3.48	3.88	3.77
Canal water radioactivity, c/m/ml (av)	715	578	671
Research samples	287	280	843
Radioisotope samples	218	300	920

Drilling New Experiment Holes

The first hole drilled to determine the feasibility of adding more research or operational facilities was successfully completed on July 10, 1959. This is a stepped hole, through the concrete shield, of 8-in. diameter the first four feet and 6-in. diameter the remaining three feet. In the graphite, the hole was drilled to a depth of 13 ft 4 in. with a diameter of three inches. This facility is located 5 ft 4 1/4 in. south of the east-west center line and 2 ft 11 5/8 in. west of the north-south center line.

A second hole was started August 10, 1959, at a position one foot south of the east-west center line and 4 ft 2 in. west of the north-south center line.

Slug Ruptures

Slug rupture data are given in Table 2.2.

TABLE 2.2. SLUG RUPTURES

Rupture Number	Date	Row No.	Number From West End of Row	Lot	Remarks
264	7-6-59	2164	21	117	Found by visual inspection
265	7-20-59	1364	Unknown	124	Found by visual inspection
266	7-27-59	1160	15	181	Found by visual inspection
267	7-27-59	1860	19	129	Found by visual inspection
268	8-3-59	1673	24	205	Found by visual inspection
269	8-24-59	1474	10	141	Found by visual inspection
270	8-24-50	1469	13	111	Found by visual inspection
271	8-31-59	1264	31	155	Found by visual inspection
272	8-31-59	2171	10	193	Found by visual inspection
273	9-14-59	2174	33	143	Found by visual inspection

Reactor Controls

On September 28, 1959, the PAT control unit for automatic reactor power control was replaced with a unit identical to that used at the BSF. Formerly, the signal to the control unit came from the recorder. Now the signal comes directly to the control unit from the chamber.

TABLE 2.3. FACILITY EMPLOYMENT

Facility	Nature of Experiment	Division Sponsor
3		Operations
4	Service irradiation facility	Isotopes
10	Water-cooled irradiation facility	Isotopes & Research
11		Chemical Technology
12	Cryostat	Solid State
13		Operations
14	Standard unit irradiation facility	Isotopes
15	Electrical component testing	Reactor Projects (GE)
16	Graphite thermocouple (annealing)	Operations
17-S		Operations
17-N	Cu crystal test	Solid State
18	Service irradiation facility	Operations
19	Hydraulic rabbit tube	Solid State
20	Electrical component testing	Reactor Projects (GE)
21	Service irradiation facility	Operations
22	Pneumatic tube	Isotopes & Chemistry
30		Solid State
38-S		Reactor Projects (GE)
50-S	Neutron spectrometer	Physics
50-N	Cryostat	Solid State
51-S	Neutron spectrometer	Physics
51-N	Enriched converter	Solid State
52-S	Neutron collimator	Physics
52-N	Cryostat	Solid State
53-N	Graphite temperature thermocouple	Operations
54-N		Operations
55-N		Operations

TABLE 2.3. Continued

Facility	Nature of Experiment	Division Sponsor
56-N	Fast pneumatic tube	Operations
56-S	Oscillator	Reactor Projects (GE)
57-N	Neutron beam collimator	ORSORT
57-S		Operations
58-N		Operations
58-S	Neutron beam collimator	Chemistry
59-S	Neutron beam collimator	Physics
60	Electronic component testing	Reactor Projects (GE)
61	Electronic component testing	Reactor Projects (GE)
A	Sample irradiation facility	Solid State
B	Sample irradiation facility	Isotopes
C	Sample irradiation facility	Operations
D	Sample irradiation facility	Solid State
1768	Sample irradiation facility	Operations
1867	Sample irradiation facility	Solid State
1968	Sample irradiation facility	Solid State
2568	Thermopile	Solid State
12 fuel channels	Fission iodine product	Isotopes
Core hole	Shielding facility	Neutron Physics
Thermal column	Sample irradiation facility	Neutron Physics
East animal tunnel	Sample irradiation facility	Operations
West animal tunnel	Sample irradiation facility	Isotopes and Chemical Technology
Slant animal tunnel	Sample irradiation facility	Research

## 2.2 Graphite Reactor Annealing

L. E. Stanford

A directive dated September 16, 1959, was received from the AEC authorizing work in preparation for the annealing. The directive further authorizes an 11-month extension relative to the previously authorized completion date of June 30, 1959, and a total expenditure of funds in the amount of \$88,000 as requested.

Work orders covering all phases of the system revisions needed for the annealing have been written and processed. Preparation for changes in the concrete ducts is expected to begin about November 1, 1959; and all work is expected to be completed by not later than May 1, 1960.

### 3. LOW-INTENSITY TEST REACTOR

#### 3.1. Operations

W. R. Casto

#### Operations

The operating data for the LITR are given in Table 3.1.

TABLE 3.1. LITR OPERATIONS

Period July 1, 1959 through September 30, 1959

	This Quarter	Last Quarter	Year to Date
Total energy, Mwd	245.1	238.2	710.7
Average power, kw/operating hour	2,988	2,973	2,977
Time operating, %	89.2	88.0	87.4
Cooling water radioactivity, c/m/ml	44,500	49,800	46,500
Cooling water resistivity, ohm-cm	995,700	897,000	909,500
Fuel pieces charged	3	6*	12
Fuel pieces discharged	2	6*	11
Shim rods charged	0	1	2
Shim rods discharged	0	1	2

\*Includes 3 partial fuel elements

#### Experiment Facilities

Core position C-44 was transferred from the Chemical Technology Division to the Reactor Projects Division for use in the gas-cooled reactor program. The Reactor Projects Division now has seven core positions with instrumentation for six experiments.

The instrumentation for the Reactor Projects Division experiment in C-52

has been moved from Building 3004 to the space in the west room formerly occupied by a Chemical Technology Division experiment.

The liner in HB-5 accommodating three of REED's rocking autoclave experiments was removed. A new liner was installed in HB-5, making the experiments in HB-5 and HB-6 interchangeable. Each now accommodates a single rocking autoclave.

Radioactivity in Cooling Water

Table 3.2 gives some of the principal radioactivities found in LTR cooling water samples during the quarter.

TABLE 3.2. RADIOACTIVITIES IN LTR COOLING WATER

Date	Na <sup>24</sup>	Np	I <sup>131</sup>	I <sup>133</sup>	I <sup>135</sup>
8-12-59	$2.24 \times 10^4$	751	116	$1.0 \times 10^3$	$1.68 \times 10^3$
9-16-59	$2.14 \times 10^4$	764	120	$1.2 \times 10^3$	$1.8 \times 10^3$

Lattice Inventory

The lattice inventory is given in Table 3.3.

TABLE 3.3. LITER LATTICE INVENTORY AS OF SEPTEMBER 30, 1959

Position	Material	Weight (g)	Position	Material	Weight (g)									
51	F	131.277	41	Be		31	F (Fx)	93.657	21	F (Fx)	89.192	11	F	126.978
52	Be		42	Be (SP)		32	F	184.876	22	F (SR)	114.862	12	F	187.345
53	Is		43	Al (SP)		33	F	142.214	23	F	163.164	13	F	135.887
54	Be		44	Be (SP)		34	F	145.534	24	F (SR)	103.782	14	F	188.372
55	F	140.338	45	F (Fx)	82.372	35	F	160.293	25	F	198.320	15	F	132.138
56	F	143.156	46	Be (SP)		36	F	150.817	26	F (SR)	88.475	16	F	169.336
57	Al (SP)		47	Be (SP)		37	F	151.274	27	F	190.642	17	F	144.903
58	Al (SP)		48	Al (SP)		38	F (Fx)	124.079	28	Be (SP)		18	F	148.097
59	F	139.370	49	Be		39	Is		29	Is		19	F	157.621
											TOTAL	4,128.371		

Legend:

- F - Fuel
- Be - Beryllium
- Fx - Partial Fuel Element
- SR - Shim Rod
- SP - Special Piece
- Is - Isotope
- Al - Aluminum

Lattice Change

Figure 3.1 indicates a change in position 11 from beryllium to fuel. Fuel was placed in this position to improve the reactivity worth of No. 1 rod.

Corrosion Samples

The corrosion samples located in the "A" tank were inspected. None of the six samples showed any visible signs of corrosion. These are: two Al-sheathed boral, isolated in a plastic holder; one Al-sheathed Cd, not isolated; one Pb, isolated with plastic washers; one Pb with 2S Al backing plate, not isolated; one mechanically assembled dummy-fuel element section, isolated with plastic mounting strips.

Six additional samples were installed August 11, 1959. Three are 2S aluminum and three are 1030 grade carbon steel. All are in contact with aluminum.

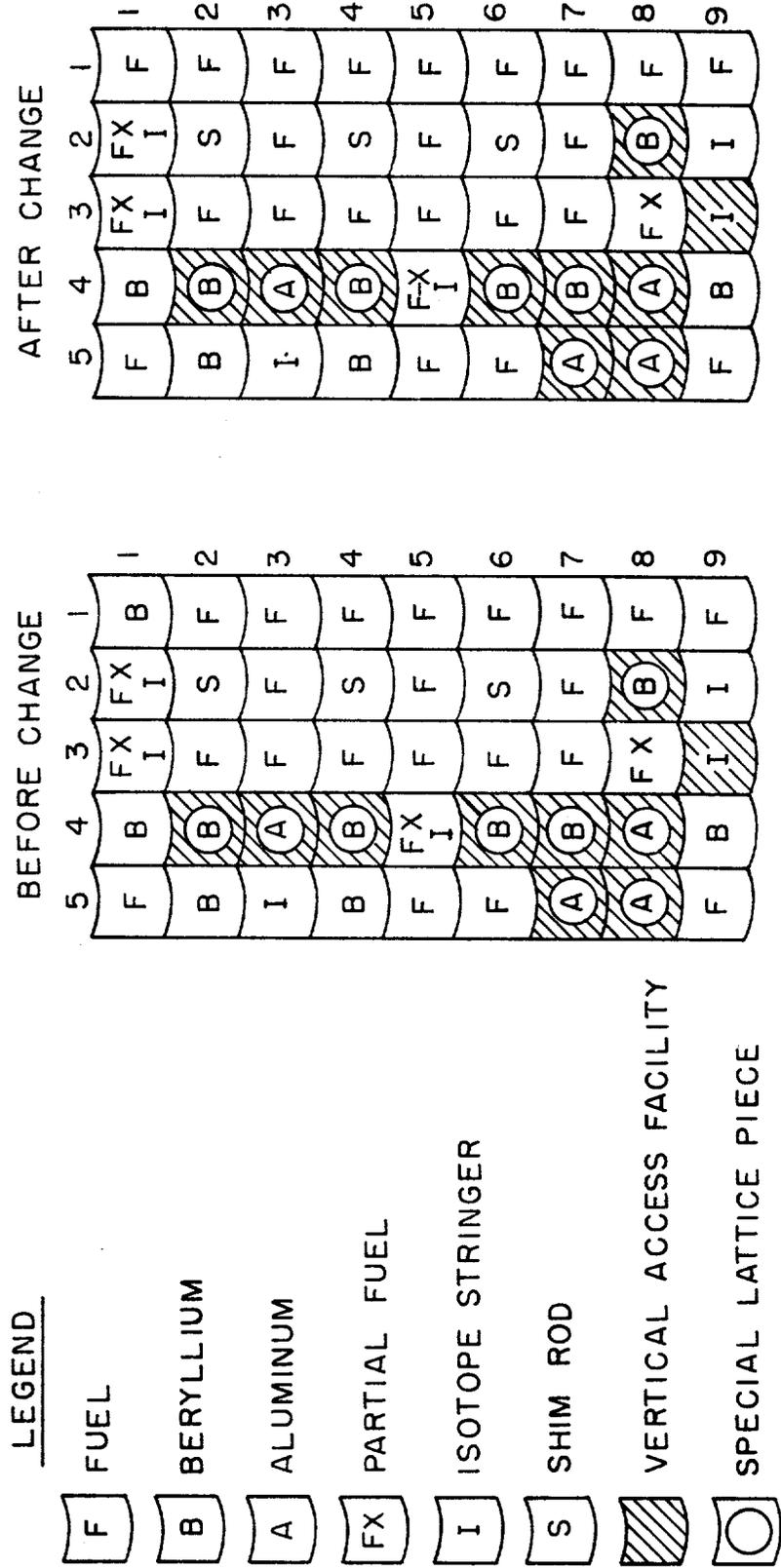


Fig. 3.1. LITR Lattice Configuration Change. Date of Change 9-22-59.

TABLE 3.4. LITR FACILITY EMPLOYMENT

Facility	Nature of Experiment	Division Sponsor
C-28, C-42, C-44, C-46, C-47, C-57, and C-58	Fuel test	Reactor Projects
C-48 and V-2	Fuel test	Reactor Projects (GE)
HB-1	Neutron beam	Chemistry
C-43	Thorium slurry	Chemical Technology
HB-2, HB-5, and HB-6	HRP solutions	REED
C-38	Crystal Damage	Solid State
1/4 C-39	Dry Tube for samples	Analytical Chemistry
3/4 C-39	Isotopes	Isotopes
1/5 P-tube	Pneumatic samples	Physics
3/4 P-tube	Isotopes	Isotopes
C-21, C-29, C-31, C-45, C-53, and regulating rod facility	Isotopes	Isotopes
HB-3	Radiation damage	Isotopes
HB-4		Operations

### 3.2. Comparison of the Worth of the LIIR Shim-Safety Rods

C. D. Cagle

In recent years the fuel loading in the LIIR core has been shifted southward to increase the neutron flux for experiments in the south half of the reactor. This was done both by fuel element placement and by keeping high-weight elements in the south half. This configuration decreased the worth of the northernmost (No. 1) shim-safety rod to about half the value of either of the other two rods.

Although this is not necessarily an unsafe condition, it was felt that the rod worths should be more nearly equalized. After the fuel concentration in the northern part of the core had been increased by loading higher weight elements in that region, a check of the effectiveness of the rods was completed on July 21 by comparison of the periods produced by moving the rods a short distance over the central portion of their travel where the worth is linear with respect to the distance moved. The No. 1 rod was worth 60% of the worth of the No. 2 rod, and the No. 3 rod was worth 90% of the worth of the No. 2 rod. This was but a slight improvement so a shift of the core configuration northward was begun in September by replacing the beryllium reflector piece in C-11 with a fuel element. The gain realized was again small. To increase the worth of the No. 1 rod significantly, full-sized fuel elements will have to be loaded into positions C-21 and C-31.

#### 4. LABORATORY FACILITIES

E. J. Witkowski

##### 4.1. Radioactive Waste Disposal

###### White Oak Creek Discharge to Clinch River

The Health Physics monitoring data for radioactive waste discharged to Clinch River during the last four quarters are listed in Table 4.1.

TABLE 4.1. CLINCH RIVER MONITORING DATA

	% of MPC* in Clinch River	
	Average for Quarter	Highest Weekly Discharge
July 1 - September 30, 1959	13	45
April 1 - June 30, 1959	44	159
January 1 - March 31, 1959	84	683
October 1 - December 31, 1958	48	400

\*The MPC in the Clinch River is the weighted average of the MPC values for occupational exposure of the individual radioisotopes as set forth by national and international committees on radiation protection. For prolonged exposure of a large population, it is recommended that the permissible levels be reduced by a factor of ten.

###### Discharges to White Oak Creek and Lagoons

A release of promethium-147 in the Isotopes area during the month of August was the only abnormal discharge of activity into the process waste system. As in the previous quarter, this activity is not reflected in the gross beta analyses shown in Table 4.2. Approximately 11 curies of promethium-147 was discharged to White Oak Creek after processing through the Process Waste Treatment Plant.

TABLE 4.2. DISCHARGES TO WHITE OAK CREEK AND LAGOONS

	This Quarter	Year to Date	Total for 1958
Process waste discharged to White Oak Creek, gal	99,880,000	251,690,000	233,340,000
Activity discharged to White Oak Creek, curies	30	88	92
Waste to lagoons, gal	1,163,000	2,884,000	3,157,000
Activity to lagoons, curies	166,000	202,000*	52,800
Total activity discharged to lagoons 2, 3, and 4 to date, curies	369,000		
Total volume discharged to lagoons 2, 3, and 4 to date, gal	14,587,000		

\*This total was reported 36,000 too high last quarter.

Process Waste Treatment Plant

The increase in throughput from 300 to 330 gpm, begun during the previous quarter, had no appreciable effect on the decontamination factor. Another 10% increase in throughput will be tried during the next quarter.

The plant was shut down for a period of 94 hours for general maintenance and for cleaning of equipment and basins.

The decontamination accomplished during the quarter is indicated in Table 4.3. The major activity, promethium-147, was discharged into the system in the Isotopes area.

TABLE 4.3. PROCESS WASTE TREATMENT PLANT OPERATING DATA

Contaminant	Activity (curies)			Percent Decontamination
	In	Out	Removed	
Sr <sup>90</sup>	29.6	55.0	24.6	83
Sr <sup>89</sup>	1.6	0.3	1.3	81
Cs <sup>137</sup>	24.2	3.0	21.2	88
Ru <sup>106</sup>	8.9*	0.6*	8.3*	93
Co <sup>60</sup>	1.4*	0.2*	1.2*	86
Pm <sup>147</sup>	1050.0**	11.2**	1039.0**	99

\*The activity handled during the month of September is not included in these figures because of meaningless analytical results caused by a mix up of samples.

\*\*These figures include only the Pm<sup>147</sup> processed during the month of August. Analysis for Pm<sup>147</sup> is usually not included unless it is known that a release of the material into the process waste system has occurred.

#### Miscellaneous Liquid Waste Disposal Items

The waste transfer line from the tank farm to the lagoons has been condemned by the Inspection Department because of extensive corrosion. Engineering work is now in progress for replacement of a large portion of the line where leaks may seriously affect operations of the Laboratory or result in the contamination of Clinch River.

Installation of a pump and pipe, connecting the outlet of the Process Waste Treatment Plant to the pipeline to the lagoons, has been started. This installation will permit pumping of process waste to the lagoons in the event of a major release of activity into the process waste system that could not be sufficiently decontaminated by the Process Waste Treatment Plant.

Excavation work has begun to increase the capacity of the equalization basin from 700,000 gal to approximately 1,000,000 gal. This will provide

more time for locating the sources of periodic accidental discharges into the process waste system. Since the Process Waste Treatment Plant can handle only one-half to two-thirds of the current volume of waste, the minimum time presently required for locating a source of discharge is only 12 hours before the waste must be sent directly to the creek. The expansion of the basin will more than double the present safety factor.

The three 170,000 gal chemical waste storage tanks, W-5, W-6, and W-8, have been provided with a ventilation system that removes any activity that may be released through openings in the float housings. Ventilation is provided from the cell ventilation duct serving the former Waste Evaporator Building, No. 3506.

Gas Disposal

The monitoring data for stacks 3020 and 3039 are given in Table 4.4.

Alarms have been installed, on the large turbines at both stacks, which alert the Steam Plant operators when the turbines are turned on. The purpose of the alarms is to prevent an accidental shutdown of the boilers by a sudden, unexpected increase in demand.

TABLE 4.4. STACK MONITORING DATA

	% of Maximum Permissible Operating Level*	
	Stack 3020	Stack 3039
Average daily discharges	11	2
Highest daily average	103	90

\*The maximum permissible operating level for air contamination has been arbitrarily established by Health Physics so as not to exceed more than 10% of the permissible exposure. The stack discharges are restricted to values which, in combined effect and under the worst conditions of dispersion, would not exceed at any point ground concentrations equal to the maximum permissible operating level.

#### 4.2. Dismantling Cells in Building 3026

The conversion of the facilities in Building 3026 into dismantling cells (as described and requested in the Preliminary Proposal No. 245, "Dismantling Cells for Power Reactor Development Experiments") is approximately 98% complete. Special equipment for the first dismantling job is now being installed in cell "A" and the storage cell. At present, no special equipment is being installed in cell B-1 or B-2. Equipment for the dismantling of the SRE fuel elements, the first phase of the PRFR Mechanical Phase Project, is being designed, fabricated, and installed. Design is approximately 90% complete, and fabrication is estimated as 52% complete with installation approximately 32% complete. December 1, 1959, is the target date for the first cold run on dismantling the SRE fuel elements.

Plans are being made to use cells B-1 and B-2 to dismantle a charcoal absorber for the Reactor Chemistry Division and to disassemble test capsules containing electronic components, irradiated at the Graphite Reactor, for General Electric.

Operating personnel spent time at the ORR hot cells and with the Chemical Technology Division familiarizing themselves with the special equipment to be used in dismantling the SRE fuel elements.

#### ORR Cells

These cells are being operated by the Laboratory Facilities Department with the aid of the Reactor Operations Department. Numerous alterations to the cells have been made and others initiated to improve the safety and ease of operation. Reactor Projects, Solid State, General Electric and Operations Divisions have been the main users.

The south cell has been more fully utilized than the north cell because

of easier access. The south cell has been scheduled for occupancy every day since the middle of July with the exception of one week when a hot off-gas line was being installed.

The cells have been operated on shifts at times when the work justified the schedule; e.g., disassembly of the MSR Pressurized Water Loop In-Pile Section and the study of irradiated ORR fuel element temperatures in air.

Planned alterations to the cells now in the design stage are: (1) addition of a semihot storage area south of the cells and (2) installation of a device to move the hot cell hoist hook east in the ORR pool under the cells to make loading of the cells easier.

#### 4.3. Liquid Hydrogen Dispensing Facility

Alterations to the facility were delayed by material deliveries. All of the material is now on hand, and the facility should be ready to operate some time in October 1959. The operating procedures have been changed as suggested by the experts at the Bureau of Standards Cryogenic Laboratory.

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