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OPERATION OF THE ORNL GRAPHITE REACTOR AND THE LOW-INTENSITY TEST REACTOR – 1954

A. F. Rupp and J. A. Cox

OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE NUCLEAR COMPANY
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ORNL GRAPHITE REACTOR

The ORNL Graphite Reactor in conjunction with the LITR continued to operate very successfully throughout the year; no major trouble of any kind was encountered. By operating the two reactors with a single crew, large economies were effected; the saving for the fiscal year 1954 is estimated to be about $150,000.

The bonded slugs canned at Y-12 by the lead-dip process,[1] charged to the Graphite Reactor in 1952, showed a total of only four ruptures during 1954, compared with 14 found in 1953. The large decline in the number of ruptures is thought to be due to several changes in operating procedure put into effect early in the fall of 1953. Fast startups were stopped to eliminate the possibility of the uranium expanding faster than the aluminum jacket and thus rupturing it. Also, the practice of operating for 1 hr or so at 4400 kw immediately after startup, so that temperature equilibrium could be reached sooner, was stopped. The four ruptures in 1954 appear to have occurred in pairs, as indicated in Table 1.

Apparently the slugs that were not beta transformed, and which were charged into the reactor before it was learned that the beta transformation was not complete, have been increasing in length. This was determined (by R. E. Adams, Metallurgy Division) by measuring slugs which had been discharged from the reactor into the storage canal. Results of measurements on slugs from two such channels are given in Table 2.

[1]E. J. Boyle, ORNL-1275 (Secret).

TABLE 1. DATA ON BONDED SLUGS THAT RUPTURED IN 1954

Rupture No.	Channel	Date	Days in Reactor	Approximate Temperature (°C)	Position from W End of Row	Remarks
126	1961	7-19	826	238	19	100% beta transformed
127	1263	7-19	819	153	5	100% beta transformed
128	2069	10-23	935	235	17	Completely oxidized; from Lot 113 (nontransformed)
129	1674	10-25	931	215	31	From Lot 115; 27.3% beta transformed

TABLE 2. MEASUREMENTS ON SLUGS DISCHARGED FROM CHANNELS 2069 AND 1674

	Number of Slugs		Average Increase in Length (in.)
Slugs from Channel 2069, nontransformed	1		0.5
	6		0.25
	13		0.2
		Av for 40 slugs	0.151
Slugs from Channel 1674, 27.3% beta transformed	1		0.21
	9		0.1
		Av for 50 slugs	0.56

Both these channels had ruptures, and both contained slugs which were not completely beta transformed. The results from two channels containing beta-transformed slugs indicate that these slugs are not growing nearly so much as the nontransformed slugs.

The average growth in channel No. 1961 for 18 slugs was 0.0085 in., and the maximum slug growth in three slugs was 0.013, 0.012, and 0.010 in. The average growth in channel No. 1263 for 52 slugs was 0.011 in., with maximum growth observed on two slugs, 0.040 and 0.030 in.; no other slug grew more than 0.020 in., and the minimum was 0.001 in. All these slugs were 100% beta transformed.

The conclusion that the untransformed rows are expanding in length has been further borne out by measurements made on a number of slug rows in the reactor. These have been measured from time to time by putting a steel tape through the channel. The average increase in length of ten rows containing beta-transformed slugs and varying in length from 41 to 54 slugs was 0.3 in., while the average increase in length of nine untransformed rows (all of 41-slug length) was 3.1 in. The average increase in length of six partially transformed slug rows, varying in length from 45 to 51 slugs, was 2.8 in. If the growth of the nontransformed slugs continues, it may be necessary to discharge considerable numbers of slugs from the reactor.

The slug inventory as of December 31, 1954, is shown in Table 3.

An automatic controller was installed in September to operate the No. 5 shim rod, thus eliminating the manual operation which had been used for almost 11 years. The controller has operated with little difficulty; however, several mechanical changes had to be made in the shim rod system. The controller is now operated from a neutron chamber formerly attached to No. 1 galvanometer in hole 40; the galvanometer is now attached to a chamber in hole 37.

In one instance, the No. 6 shim rod failed to go in when the reactor scrammed. This apparently was caused by a faulty solenoid valve in the system, but it has not yet been determined exactly why the valve failed to operate.

The control system now comprises safety rods in holes 7, 8, and 9, shim rods in holes 5 and 6, and regulating rods in holes 1 and 2. Safety rods 8 and 9 are in the No. 1 safety circuit, while the two shim rods and safety rod No. 7 are in the No. 2 circuit. Following are the trips for the No. 1 circuit: control desk switch, south door switch, water level in storage tank, three safeties (scram from amplifiers), water flow at hole 12, water temperature at hole 12, core hole monitron, north door switch, kilowatt (thermopile) recorder, and differential air pressure across graphite. The following trips are on the No. 2 circuit: control desk switch, No. 63 accumulator, No. 54 accumulator, inlet air recorder, hole 51 water flow, and metal temperature recorder.

TABLE 3. SLUG INVENTORY AS OF DECEMBER 31, 1954

	Number of Slugs	Weight of Uranium (kg)
In Reactor		
Bonded slugs	41,900	49,050.235
Experimental bonded slugs (1866)	15	16.400
Donut slugs	21	19.184
Enriched (Hole 51)		(0.112)*
Samples		0.317
Total, Normal U		49,086.136
In Vault		
Bonded slugs	9,765	
Donut slugs	35	

*Enriched uranium.

During the year, the scram on the stack gas activity was removed, since experience had shown that this device is unnecessary because the only high air activity ever noted had come from LITR experiments rather than from the Graphite Reactor. The high stack activity recorder was rearranged to actuate an alarm. The scrams from hole 12 were added to the safety circuits so that it would be possible to safely irradiate flammable liquids boiling at a temperature as low as 40°C.

The canal continued to be a major source of radioactive contamination and radiation; therefore working time was generally limited to about 1 hr per man per day. The canal water in the main section usually has a radioactive contamination of 2000 to 3000 counts/min/ml, measured with well-type gamma scintillation counter. Recently an analysis showed the radioactivity to be about half Co^{60} and half Cs^{137}. This indicates that some of the uranium stored in the canal is exposed to the canal water and that fission products are being dissolved. It was previously known that the Co^{60} was coming from exposed metallic cobalt in the canal. Most of the uranium in the canal was discharged in 1952, and the short-lived fission products have decayed to a low level.

It is hoped that the approximately 57 tons of uranium slugs now stored in the canal will be processed by the Metal Recovery Plant so that it will be possible to clean out the canal and to remove all pieces of bare cobalt that are exposed to the water. Also, a program is under way to paint the walls and walkway of the canal to reduce the contamination which soaks into the bare concrete from the canal water. At some later date, if the radioactive materials have been cleared out sufficiently, it is planned to concentrate all the remaining active material in the deep pit and to paint the canal itself.

For approximately four years a cation demineralizer has been used to process the water that is discharged from the canal to the settling basin. At the same time, the cooling water from holes 19 and 51 (demineralized water prepared at the water demineralization plant) is sent to the canal to serve as a makeup. Two anion columns now have been added to the system so that the water passing through the demineralizers will be nonacidic and can be returned to the canal. Also, a head tank has been installed on the top of the Graphite Reactor so that the water from the demineralizer can be pumped to the head tank, from which it will flow to the holes requiring water cooling. This water can then be returned to the canal, so that the water system will be a closed circuit with no purge at all except when the demineralizers are being regenerated. This will eliminate a load on the process waste system of approximately 500,000 gal per month.

Installation of the anion demineralizers is now practically complete, and operation has begun but the head tank is not being used and the effluents from the demineralizer are still being sent to the settling basin. Data on the system are given in Appendix I.

No changes occurred in the mattress plates during the year. The north plate was last changed on November 24, 1952, and the south plate on December 1, 1952. The plates have lasted this well because slug discharges have been quite limited; the weekly discharge of 16 slugs used for producing radioactive iodine is the only regular slug discharge operation.

The two 900-hp fans which draw cooling air through the reactor operated well, but there was considerable trouble from lightning during the summer. Lightning struck the fan house three times: on June 1, on July 22, and again on August 28. In the first two occurrences the damage was limited to the starting cubicle; but the last time, one of the 900-hp motor windings was burned out. As a result, several major electrical supply companies were asked for advice and were essentially unanimous in recommending that lightning arresters and capacitors be installed near the motors. This equipment is now on order and is expected to be installed in February or March 1955. The motor which was burned out was repaired at K-25, and a new motor is on order so that there will always be at least one spare motor on hand.

Table 4 gives a comparison of the pressure-drop data on the reactor exit air filters for December 1953 and for December 1954.

Several filter changes were made during 1954; they are shown in Table 5, together with changes that were made in previous years.

Enough paper filters are on hand to complete changing cells 1 and 2 when needed. The cost of the paper filters is approximately $50 each, and 200 are required to change the four cells, making the total cost of paper filters for a change approximately $10,000. However, since the paper filters last approximately three and one-half years, the yearly cost is only about $3000.

TABLE 4. PRESSURE-DROP DATA

	Pressure Drop (in. water gage)		
	Glass Wool	CWS No. 6	Total Across House
12-31-54	1.6	1.3	4.3
12-31-53	2.7	3.0	6.8
Clean filters	1.1	1.3	3.3

TABLE 5. CHANGES MADE IN FILTERS DURING 1954 AND PREVIOUS YEARS

Cell No.	Type of Filter	Date Changed	Date Previously Changed
1	Glass wool	May 8, 1950	
	Paper	May 25, 1951	
2	Glass wool	April 9, 1953	
	Paper	May 25, 1951	
3	Glass wool	May 10, 1954	October 23, 1950
	Paper	June 7, 1954	May 3, 1951
4	Glass wool	August 16, 1954	April 26, 1949
	Paper	July 6, 1954	May 10, 1951

LOW INTENSITY TEST REACTOR

The LITR operated at a power level of 3 Mw throughout 1954 without major difficulty. The downtime was somewhat high (19.4%), but this was due to long shutdowns required by research programs when important experiments were being installed or removed. The downtime for operational reasons was only 5.5%.

The presence of Np^{239} in the LITR water has not been explained, although further investigation was done during the year. A check was made for the presence of U^{239}, which was expected to be found if uranium were exposed to the water. Since none was found, this seems to indicate that neptunium is selectively diffusing through the fuel plate cladding; however, this has not been proved.

Major improvements were made in the water system when a new 30-ft^3 mixed-bed column was installed in September. The previous demineralizer system consisted of two 3-ft^3 mixed-bed columns which were operated alternately, one being regenerated while the other one was in service. The previous system did not have sufficient capacity to keep the radioactivity in the circulating water as low as desired, and therefore radiation around the water system was initially high on shutdown day. The radioactivity in the water was usually 200,000 to 300,000 counts/min/ml, and tools used inside the reactor tank during shutdown emerged with considerable contamination. The small 3-ft^3 columns were also unsatisfactory because they could not be regenerated unless the resin was removed from the column, and, since the radiation from the columns was more than 10 r/hr at 1 ft, this was a very difficult operation.

It was found that the anion resin in the mixed-bed column became damaged because of radiation, and it was decided to install cation columns before the new mixed-bed column so that most of the radioactivity would be removed by the cation resin, leaving the new mixed-bed column only the remaining cations and the anionic constituents to remove. This arrangement has worked very satisfactorily, and the quality of the circulating water has been much improved. The system is now operated with the flow rate through the demineralizer at about 15 gpm, and the radioactivity in the reactor water is now generally below 50,000 counts/min/ml. The life of the mixed-bed columns between regenerations has been about six weeks. Radiation from the open reactor tank has been reduced from 100 to about 30 mr/hr, and all the tank surfaces exposed to the water are much less radioactive than they were formerly.

Corrosion studies are under way, but as yet sufficient data are not available to compare the corrosion rates before and after installation of the new mixed-bed column. However, comparisons were made recently of corrosion rates in the reactor lattice, where the samples are exposed to a high radiation field, and approximately 15 ft above the lattice in the reactor tank, where the samples receive very little irradiation. The results are shown in Table 6.

WATER DEMINERALIZATION

Considerable trouble was encountered in the water demineralization plant in 1954. The resin particles in the cation columns were found to be breaking up so that a large amount of sludge-like fines accumulated throughout the bed and eventually made it impossible for water to flow through it. After lengthy negotiations with the manufacturer, arrangements were made to supply, free of charge, 55 ft^3 of a more stable type of resin to fill one of the cation columns, leaving the resin removed from this column to be used as makeup for the second column. It is believed that about one-third of the resin in the cation columns has been lost during backwashing operations to remove the fines.

In September, because of a demand for a large amount of demineralized water by the Bulk Shielding Reactor, 5 ft^3 of anion resin was added to each of the anion columns, and the capacity of the plant was markedly increased. Approximately three days were required to produce 140,000 gal in addition to normal production. Following this, the capacity of the columns gradually declined until, during December, the anion columns were producing only about one-third of the rated capacity.

HYDROGEN LIQUEFACTION

About 135 liters of liquid hydrogen was produced during the year. Because of increased demands for this material, a new liquefier is being built and is expected to be ready early next year.

REACTOR INFORMATION

Considerable interest has been shown in having a list of general information about the Graphite Reactor and the LITR. These data are now being compiled, but it will be some time before the section on various nuclear constants, etc., is finished. Therefore, some of the data now available are included in Appendixes I, II, and III for those who may have need for the data at the present time. Only a small amount of the data is classified (LITR Cooling System), but this is now undergoing declassification, and it is hoped that the final information handbook on these two reactors will be unclassified.

TABLE 6. CORROSION OF SAMPLES IN RELATION TO RADIATION

Sample	Material	Exposure (days)	Corrosion Rate (mpy)	Position
420-67	Aluminum	42	0.3	Upper tank out of flux
420-69	Aluminum	42	0.24	C-53-B-24 flux approx 1×10^{13} neutrons/cm^2/sec
420-52	1015 Steel	27	24.8	C-57-B-1 flux approx 1×10^{13} neutrons/cm^2/sec
420-53	1015 Steel	175	8.6	C-57-B-20 flux approx 1×10^{13} neutrons/cm^2/sec
420-71	1015 Steel	42	4.5	Upper tank out of flux

Appendix I

GRAPHITE REACTOR INFORMATION

LATTICE DATA

Size of graphite	24 ft long × 24 ft wide × $24\frac{1}{3}$ ft high
Number of fuel channels	1248
Number of 1.6-in.-dia holes through central part of core	4
Type of fuel	Normal uranium
Dimension of aluminum-clad fuel element (slug)	$4\frac{1}{8}$ in long × 1.19 in. in diameter
Weight of uranium metal per slug	1175 g
Thickness of aluminum cladding	0.035 in.
Number of fuel channels containing fuel	821
24-slug rows 10	
41-slug rows 51	
45-slug rows 23	
47-slug rows 115	
51-slug rows 79	
54-slug rows 543	
Number of slugs in reactor	Approx 41,000
Approximate reactivity absorbed by experiments and radioisotope samples	180 inhr
Maximum temperature of slug jacket	Approx 250°C
Maximum thermal neutron flux at 3.5 Mw (from CP-2602 and flux measurements by G. W. Leddicotte)	1.1×10^{12} neutrons/cm^2/sec
Cadmium difference (from memo by E. J. Murphy)	20
Cadmium ratio (Ag-Cd) (from NNES IV-5)	1
Average gamma photon energy (from ORNL-129)	0.9 Mev
Graphite	
Diffusion length	48.0–50.3 cm (from NNES IV-5)
Density of graphite blocks	1.61 g/cc (from NNES IV-5)
Maximum gamma photon flux (calculated from data in ORNL-129)	5.13×10^{11} γ/cm^2/sec
Maximum gamma dose rate (calculated from data in ORNL-129)	9.2×10^5 r/hr
Ratio of gamma flux to thermal neutron flux (calculated from ORNL-129)	0.50
Reactor power level	Approx 3.5 Mw
Maximum graphite temperature (hole 13)	Approx 170°C
Average temperature differential between inlet air and graphite in hole 20	Approx 90°C
Average metal temperature at hottest region	Approx 250°C
Average neutron flux (based upon power level and amount of fuel)	5.0×10^{11} neutrons/cm^2/sec

Ratio of maximum thermal neutron flux to average thermal neutron flux (CP-2602)	2.24
Volume of active portion of reactor	5,933.4 ft^3
Average power density inside active portion	587.5 w/ft^3
Operating metal temperature coefficient (BNL-152)	Approx 0.29 inhr/°C central temperature
Operating graphite temperature coefficient (BNL-152)	0.77 inhr/°C central temperature
Barometric pressure coefficient (BNL-152)	0.38 inhr/mm Hg
Xenon poisoning coefficient	25 inhr/1000 kev

EXPERIMENTAL FACILITIES

Number of 4-in. square openings	56
Number of horizontal through-holes (4 in. sq)	22
Number of horizontal half-holes (4 in. sq)	6
Number of vertical holes (4 in. sq)	6
Number of 4-in. square horizontal through-holes used for control and shutdown rods	4
Number of 4-in. square horizontal openings for experiments and target exposures	47
Number of 4-in. square vertical holes used for shutdown rods	3
Number of 4-in. square vertical holes for experiments and target exposures	3
Other experiment facilities	14
Approximate amount of reactivity taken up by experiments and radioisotope samples	180 inhr

LIST OF HOLES AND MAXIMUM THERMAL NEUTRON FLUXES

Hole No.	Maximum Flux at 3500 kw (Neutrons/cm^2/sec) and Other Information	Source
1	Regulating rod	
2	Regulating rod	
3	8.5×10^{11}	Estimated from CP-2602
4	8.5×10^{11}	Estimated from CP-2602
5	Shim rod	
6	Shim rod	
7	Safety rod	
8	Safety rod	
9	Safety rod	
10	$\sim 9 \times 10^{11}$	Estimated from hole 12 flux
11	8.9×10^{11}	Estimated from hole 12 flux
12	8.9×10^{11}	G. W. Leddicotte
13	6.4×10^{11}	CP-2602
14	8.5×10^{11}	CP-2602

Hole No.	Maximum Flux at 3500 kw (Neutrons/cm^2/sec) and Other Information	Source
15	9.6×10^{11}	CP-2602
16	8.5×10^{11}	CP-2602
17	6.3×10^{11}	CP-2602
18	6.3×10^{11}	CP-2602
19	8.5×10^{11}	CP-2602
20	9.6×10^{11}	CP-2602
21	8.5×10^{11}	CP-2602
22	6.3×10^{11}	CP-2602 and G. W. Leddicotte
30	1.1×10^{9}*	J. D. Knox
31	Blocked by H-beam air seal	
32	4×10^{8}	Estimated from hole 30 flux
33	4×10^{8}	Estimated from hole 30 flux
34	1.1×10^{9}	Estimated from hole 30 flux
35	Blocked by H-beam air seal	
36	4×10^{8}	Estimated from hole 30 flux
37	4×10^{8}	Estimated from hole 30 flux
40	2×10^{9} at inner edge of concrete shield	Estimated from hole 43 flux
41	3.3×10^{9} at inner edge of concrete shield	Estimated from hole 43 flux
42	3.3×10^{9} at inner edge of concrete shield	Estimated from hole 43 flux
43	3.3×10^{9} at inner edge of concrete shield	J. D. Knox
44	2×10^{9} at inner edge of concrete shield	Estimated from hole 43 flux
45	2×10^{9} at inner edge of concrete shield	Estimated from hole 43 flux
46	Periscope hole	
47	Periscope hole	
50	4.3×10^{11}	CP-2602
51	7.5×10^{11}	CP-2602
52	4.3×10^{11}	CP-2602
53	5.1×10^{11}	CP-2602
54	9.7×10^{11}	CP-2602

*9 ft from outside surface of concrete shield.

Hole No.	Maximum Flux at 3500 kw (Neutrons/cm^2/sec) and Other Information	Source
55	5.5×10^{11}	CP-2602
56	3.6×10^{11}	CP-2602
57	6.3×10^{11}	Estimated from CP-2602
58	3.6×10^{11}	CP-2602
59	5.5×10^{11}	CP-2602
60	9.6×10^{11}	CP-2602
61	5.7×10^{11}	CP-2602
62	1.1×10^{12}	CP-2602
West core hole	5×10^9 at inner end of hole 12,000 r/hr gamma at inner edge of hole	Memo from E. J. Murphy
East animal tunnel	5.7×10^8 in Al cart Mn-Cd ratio 73 in Al cart Gamma 3400 r/hr 4.29×10^8 in Al cart	C. D. Cagle and J. B. Trice J. O. Arterburn, General Electric Co.
West animal tunnel	In fission cart: Thermal neutrons, 6×10^8 Fast neutrons, 5×10^7 Out of fission cart: Thermal neutrons, 1.3×10^9 Fast neutrons, 1.3×10^4	J. S. Cheka
Inclined animal tunnel	1×10^9 In-Cd ratio, 100,000	G. Stapleton
Thermal column	7.2×10^7 In-Cd ratio $\sim 10^6$ Gamma 80 r/hr	G. McCammon in ORNL CF-49-11-327 Memo from E. J. Murphy
Neutron converter hole 1867 (1768 and 1968 are similar)	Neutron fluxes: $(nv)_{\text{thermal}} = 3.8 \times 10^{11}$ $(nv)_{\text{thermal}} = 4.0 \times 10^{11}$ >1 Mev $= 1.6 \times 10^{11}$ >2 Mev $= 6.5 \times 10^{10}$ >4 Mev $= 1.4 \times 10^{10}$ >6 Mev $= 3.5 \times 10^9$ Maximum sample size, $11 \times \frac{1}{2}$ in.	J. B. Trice in ORNL-1359
Pneumatic tube in hole 2079	5×10^{11} (See drawing No. C-11945 for capsule)	Calculated from CP-2602
Pneumatic tubes in hole 22	6.3×10^{11} (See drawings No. C-12901 and A-6045 for capsules)	G. W. Leddicotte
A, B, C, and D	1.1×10^{12}	Estimated from hole 62; flux data in CP-2602

INTENSITY OF TYPICAL COLLIMATED BEAMS FROM GRAPHITE REACTOR

Facility		Work Done By
Hole 51-S	The scattered 1.2 Å (~0.07 ev) neutrons through a 2.43-cm^2 collimator set at an angle from the primary beam give a beam strength of 6×10^4 neutrons/cm^2/sec, assuming a 2% efficiency for the detector; the scatterer is a single crystal of lead; power level is 3500 kw	M. K. Wilkinson, Feb. 1955
Hole 17-N	From a 1-in.-dia collimator, a thermal neutron beam strength of 1.0×10^7 neutrons/cm^2/sec was measured where the beam was $1\frac{5}{16}$ in. in diameter; power level was 3700 kw	F. Pleasanton, Aug. 1951

MAXIMUM TEMPERATURE AT VARIOUS LOCATIONS

In fuel channels – approximately 2 in. west of center

Hole 13	Max. ~120% of Graphite temperature at center of hole 20
Hole 14	Max. ~104% of Graphite temperature at center of hole 20
Hole 16	Max. ~65% of Graphite temperature at center of hole 20

Graphite temperature (hole 20) ~90°C above inlet air temperature

SHIELDING DATA

Thickness of concrete shield: 7 ft 0 in.

Composition of shield: 1-ft-thick retaining walls inside and outside, with 5 ft of special barytes ($BaSO_4$)-haydite aggregate in the center

Reference: NNES IV-5

Data on shield efficiency: in ORNL 737

CONTROL SYSTEM

Originally the control system included 4 safety rods (vertical), 4 shim rods (horizontal), 2 regulating rods (horizontal), and 2 boron steel shot tubes (vertical). Of these, 1 shot tube, 2 shim rods, and 1 safety rod have been removed, and the other shot tube is scheduled to be removed soon.

The safety rods have been increased in size (ORNL CF-49-4-135), and one of them has been removed. This increase in size resulted in an increase in effectiveness from 105 to approximately 185 inhr.

The two shim rods are worth approximately 235 inhr each and are composed of boron steel ($1\frac{1}{2}$% boron) rods $1\frac{3}{4}$ in. square.

The two regulating rods are worth about 180 and 210 inhr, respectively (BNL-22).

The safety rods hang vertically from cables wound on drums. In the event of a power failure or a scram signal, the drums are released, allowing the rods to fall into the reactor in about 1 sec.

The shim rods are driven for normal insertion, and all removals are by an oil gear pump acting on a hydraulic piston. The withdrawal rate is approximately 1 in./sec. For emergency insertion, such as a scram signal or power failure, an independent hydraulic power source is used to drive the rods into the reactor in approximately 4 sec.

In 1954, one of the shim rods was arranged to be automatically controlled by a Leeds & Northrup position-adjust-type controller. The scram feature overrides the controller in the event of a shutdown.

The two regulating rods are driven by electric motors and do not go into the reactor automatically during a scram.

The automatic portion of the safety system is worth approximately

3 safety rods, each at 185	=	555 inhr
2 shim rods, each at 235	=	470 inhr
Total		1025 inhr

The two regulating rods are worth 180 + 210 inhr, making a total of 390 inhr.

With a normal loading the clean, cold reactor at startup may have an excess reactivity of as much as 250 inhr. The usual excess in the control rods while operating at temperature and xenon equilibrium is 60 to 100 inhr.

COOLING SYSTEM

(NNES IV-5)

Coolant – air

Rate of coolant flow – approx 120,000 cfm

Typical pressure values (in. of H_2O)

Across inlet filters	−1.0
Pitot static (inlet duct)	−8.8
Pressure drop across reactor	16.0
Suction pressure at rear of reactor	−26.0
Pressure drop across filter house	4.4
Static suction pressure on fan side of filter house	−38.5
Suction pressure of fan	−43.0
Discharge pressure of fan	8

Air temperatures (typical)

- Outlet air temperatures average about 60°C above outside air temperature
- The reactor acts as a heat sink, giving up heat at night and storing heat during the daytime

Fans – two 900-hp centrifugal compressors

Filter house (ORNL-1417)

Glass wool filters

- 1 layer of American Air Filter Corp. FG No. 50 ½ in. thick
- 1 layer of American Air Filter Corp. FG No. 25 ½ in. thick
- 4 cells containing 30 frames of 5 pockets, 4 × 30 × 5 = 600 pockets
- 4 pads/pockets in 2 sections, 1 layer of FG 25 and 1 layer of FG 50 per section
- Size 2 × 3 ft

Paper filters

- Composition AEC No. 1 (Revised Spec) paper
- Number – 4 cells containing 10 frames, each containing 5 units = 200 units
- Size – each unit is 2 ft × 2 ft × 11½ in. containing pleated filter paper

Number of changes necessary from November 1949 to December 1954

- 7 changes of No. 1 cells
- 6 changes of No. 2 cells

Inlet filters – American Air Filter Corp. Airmat-type PI-24-108 frames loaded with $\frac{3}{32}$ in. Type G Filter Media. Dimension of each frame – 2 ft × 2 ft × 6 in. Media pleated to give about 28 ft^2/frame.

Activity in discharged air – approximately 500 curies A^{41}/day

This radioisotope has a 1.8 hr half-life and decays so rapidly that it represents no hazard.

Stack – 200 ft high

CANAL

The canal at the Graphite Reactor has become quite radioactive (~4000 counts/min/ml counted with a well-type scintillation counter) due to the presence of considerable radioactive cobalt and irradiated uranium exposed to the water. About half of the activity is due to Co^{60} and half to Cs^{137}. Most of the uranium has been stored in the canal for several years.

It has been the practice to purge the canal at a rate of about 10 gpm and to discharge the water through cation exchange resin columns so that most of the activity would be removed and concentrated on the resin. When the resin was regenerated, the radioactivity contained in the acid regenerating solution was sent to the hot waste tanks.

In order to utilize the demineralized water, anion-exchange resin columns have been installed so that the effluent water will have a pH between 5 and 6 and can be used for cooling experiments in the reactor. After cooling the experiments the water is returned to the canal, thus making a closed cycle except for the regenerating solutions from the ion columns. It is expected that this will reduce the amount of waste water from about 400,000 gal/month to about 20,000.

Following are specifications on the new system.

Head Tank	
Structural material	Stainless steel
Capacity	1800 gal
Approximate height above canal level	50 ft
Pumps	
Number	2
Make	Fairbanks-Morse
Model	FIA16F4
Capacity	50 gpm at 100 ft head
Anion Units	
Number	2
Structural material	Stainless steel
Volume of column	16 ft^3
Volume of resin	10 ft^3
Type of resin	Nalcite SAR
Regenerant	4% Sodium hydroxide
Regenerant volume	250 gal
Backwash rate	10 gpm
Regeneration rate	5 gpm
Rinse volume	1000 gal
Rinse rate	5 gpm
Throughput capacity	Approx 50,000 gal
Throughput rate	10 to 20 gpm
Influent pH	3 to 4.5
Effluent pH	4.5 to 6
Influent specific resistance	6000 to 8000 ohm-cm
Effluent specific resistance	100,000 to 150,000 ohm-cm
Influent	200 to 400 counts/min/ml (practically all activity is removed by cation resin)

Effluent	100 to 400 counts/min/ml
Shielding	2 in. lead (These units originally were used for sand filters. The shielding needed on anion units is much less.)

Engineering drawings D19857, D19858, D11534, D11575, D20440, C21557, D11535, and D11574 show details of the system.

Filters

Number	2
Manufacturer	Commercial Filter Corp.
Model	WS-26T-10-4F
Shielding	12 in. barytes concrete block

Ion-Exchange Columns

Cation Units

Number of units	2
Structural material	Stainless steel
Volume of column	24 ft^3
Volume of resin	15 ft^3
Type of resin	Nalcite HCR
Regenerant	5% nitric acid
Regenerant volume	150 gal
Backwash rate	10 gpm
Regeneration rate	5 gpm
Rinse volume	700 gal
Rinse rate	5 gpm
Throughput capacity	Approx 30,000 gal
Throughput rate	10 to 20 gpm
Influent pH	6 to 8
Effluent pH	3 to 4.5
Influent specific resistance	6000 to 8000 ohm-cm
Effluent specific resistance	6000 to 8000 ohm-cm
Influent	10,000 to 15,000 counts/min/ml
Effluent	200 to 400 counts/min/ml
Shielding	2 in. lead

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ORNL-LR-DWG 8313

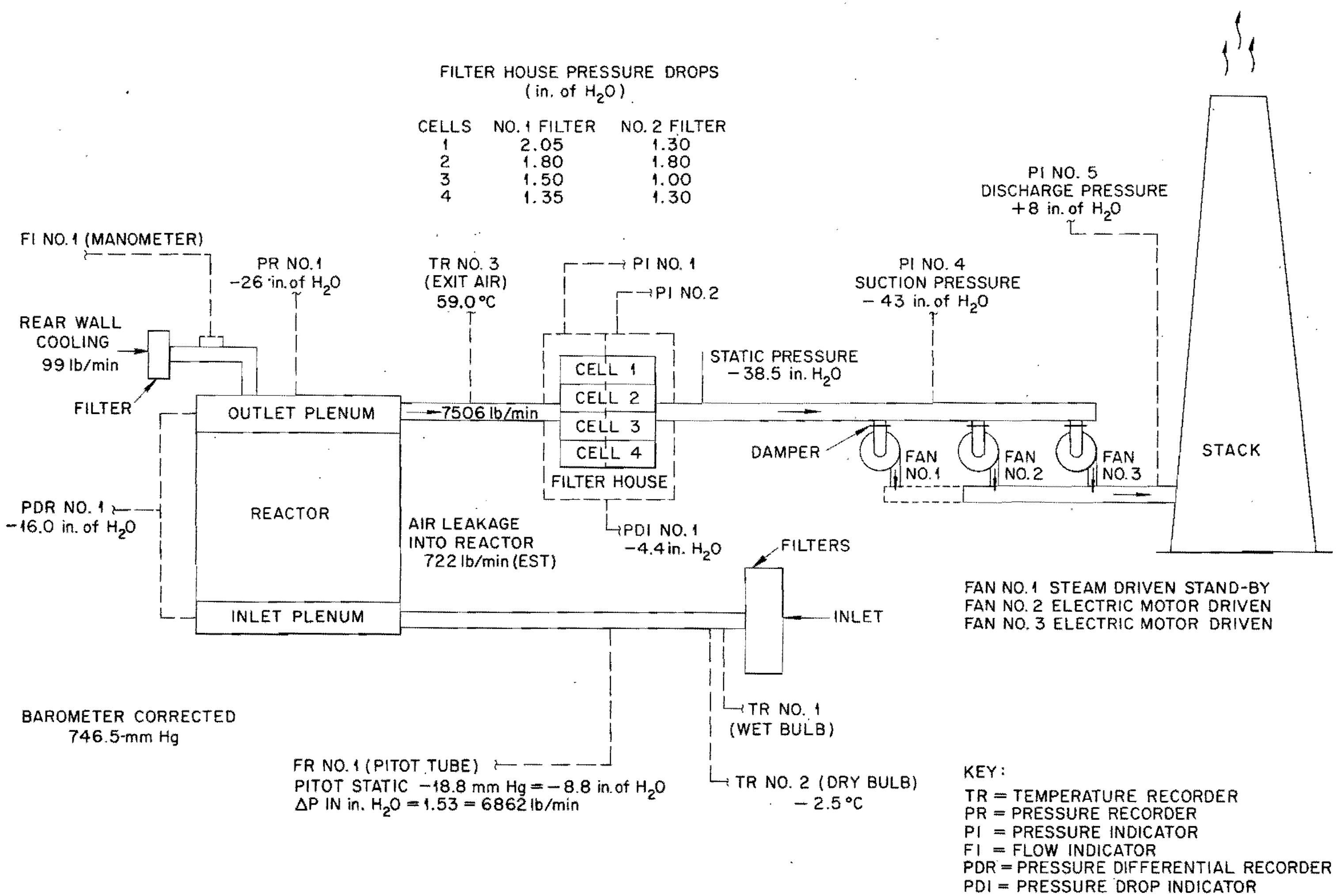

Diagram of Air Flow with Pressures and Temperatures Recorded on 1-25-55 —6:30 PM TO 7:30 PM.

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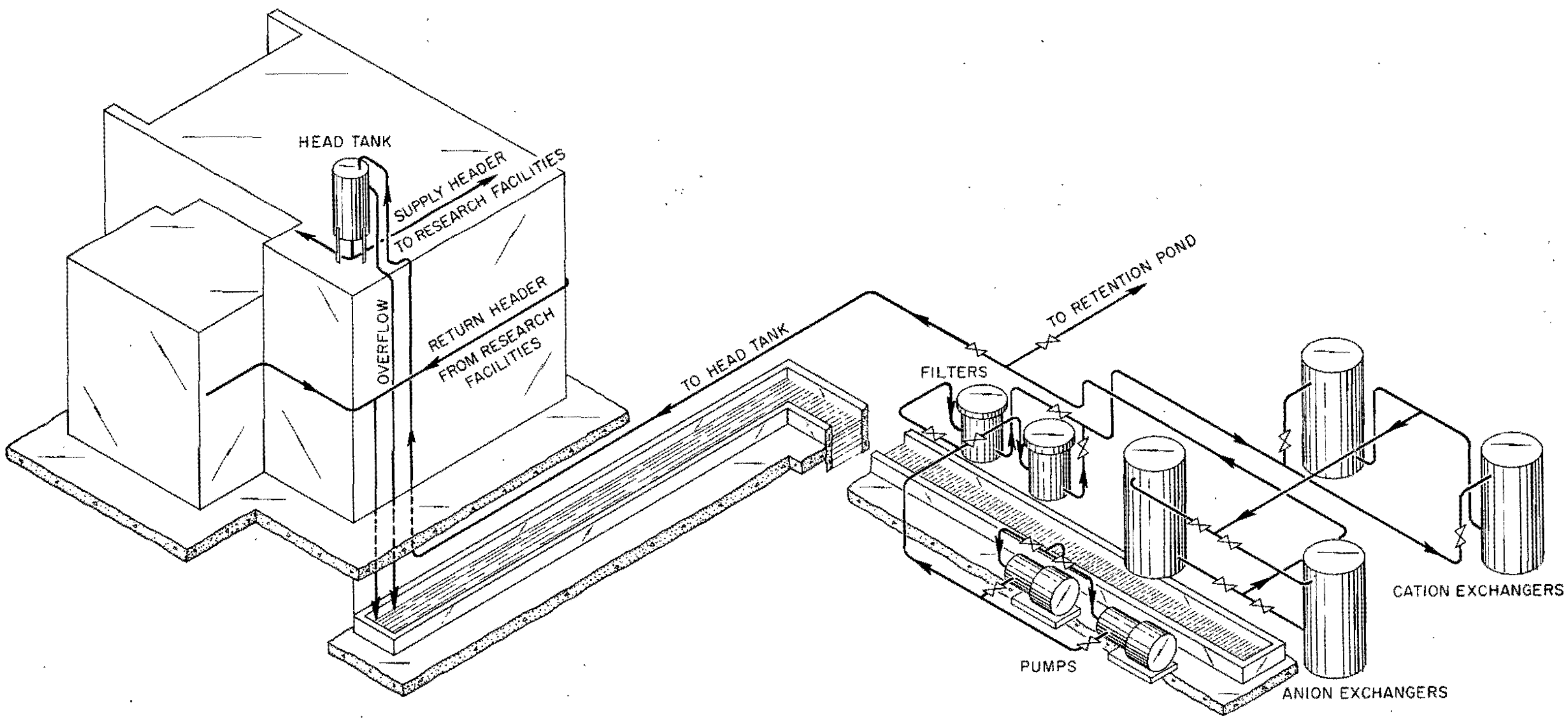
ORNL-LR-DWG 8314



Canal Demineralizer

Appendix II

LITR INFORMATION

LATTICE DATA

Size of active core	9 in. wide × 24 in. high × 24 in. long
Approximate weight of U^{235} in core	3400 g
Average thermal neutron flux in core	2.2×10^{13} neutrons/cm^2/sec
Normal power level	3000 kw
Number of fuel elements in core	22 (2 are approx half-elements)
Number of shim-safety rods in core	3
Weight of U^{235} per element	200 g (original weight)

Until October 25, 1954, the lattice arrangement was generally a slab $3 \times 8 + 1 = 25$ elements (including three shim rods). Occasionally another element was placed in front of one of the holes on the west side but this placement was generally for short periods. On the above date three elements were removed from the north end of the lattice and placed in front of HB-5 in lattice positions 45, 55, and 56.

The reflector is composed of beryllium blocks $2 \times 2 \times 8$ in. stacked around the four sides of the core leaving the top and bottom water-reflected. The beryllium has a minimum thickness of 8 in. and the total weight is 2500 lb.

For details of fuel elements see:

Drawing TD 1913	Fuel element
Drawing DRP 277	Fuel element
Drawing TD 1925	Fuel element
Drawing TD 1908	Shim-safety rod

SHIELDING DATA

The shield is composed of unmortared concrete block with a minimum thickness of 10 ft 7 in.

Radiation measurements made by R. L. Clark (Feb. 2, 1955) while the LITR was operating at 3000 kw showed the following:

East Side	~5 ft north of East-West centerline and 3 ft above the center of the lattice	1 mr/hr
West Side	~4 ft south of East-West centerline and on the horizontal centerline	6 mr/hr

Radiation leakage around the beam hole plugs which have about 1/8 in. clearance in the 8 in. section and 1/4 in. in the 7 in. section is as much as 1 r/hr at the top and external shielding is therefore required.

CONTROL SYSTEM

Number of shim-safety rods in core	3
Number of ionization chamber holes	6
Number of regulating rods	1

ORNL 963 describes the control instrumentation at the MTR which is similar in most respects to that at the LITR. Major differences are:

3 shim-safety rods instead of 4 (worth ~8% $\Delta k/k$ each)
1 regulating rod instead of 2
No Cd-Be control rods

Following is a list of alarms and scrams on the Reactor Control system:

Alarms

- Temperature differential between inlet and outlet water (20°F)
- Reactor water temperature (125°F)
- Reactor water activity (80% of arbitrary scale)
- Seal tank level (low – 1 in., high – 7 in.)
- Off-gas vacuum (low)

Alarms and Scrams

- Low water flow
- High water pressure in tank
- Both bypass and exit valve closed
- Reactor period
- Reactor power (2 channels)
- Experimental holes (each experiment may have several alarms or scrams)

EXPERIMENTAL FACILITIES

Number of beam holes (See drawing No. E-12859 for typical beam hole)	6
Number of pneumatic tubes (See drawing No. 13139 for capsule)	2
Number of low-flux slant holes	4

LITR FLUX MEASUREMENTS

Facility	Flux Adjusted to 3000 kw Power Level	Done By	Date of Measurement
HB-1		None made	
HB-2	3×10^{13}	J. B. Trice, ORNL-1359	8-52
HB-4	7.8×10^{12}	G. Jenks	11-16-53
HB-5	7.6×10^{12}	F. Sweeton	3-6-53
HB-6			
C-42	3.1×10^{13}	General Electric Co., ANP	1-21-53
C-44	3.3×10^{13}	W. Brooksbank	10-9-52
C-46	3.4×10^{13} 4.3×10^{13}	W. Willis M. T. Robinson	6-12-53 10-19-53
C-48	3.4×10^{13}	M. T. Robinson	10-19-53
C-38	nv above 50 kv = 3×10^{13}	M. Wittels	12-16-53
C-28		None made	
V-1	4×10^{11}	G. Leddicotte	8-52
V-2	5×10^{11}	G. Leddicotte	8-52
V-3	2.4×10^{11}	G. Leddicotte	8-52
V-4	2.6×10^{11}	G. Leddicotte	8-52
HR-1 and 2	1.1×10^{13}	J. B. Trice	9-15-52

Nuclear heating in HB-2 (adjusted to 3000 kw power level) from ORNL-1606, work done by J. B. Trice.

Material	Watts/gram	Distance from Inner End of Hole to Reduce Heating by Factor of 10
Aluminum	0.36	12 in.
Copper	0.42	12 in.
347 Stainless steel	0.40	13 in.
Solid lead	0.36	13 in.
Nylon	0.36	14 in.

Nuclear heating in lattice position C-44 (adjusted to 3000 kw power level) from ORNL-1214, work done by J. B. Trice.

Material	Watts/gram	
Solid lead	0.51	Note: This figure is somewhat high because of dilution of power density by adding fuel in order to increase power level from 770 kw to 3000 kw. Adjusting for power dilution gives a value of 0.37 w/g.

For neutron flux spectrum in the pneumatic tubes, see Report No. ORNL-1261. Values listed must be multiplied by 2.75 to adjust for power level increase and power dilution. Cadmium differences should be nearly correct.

A dense, uncooled object in an experimental hole near the fuel can be expected to exceed a temperature of 400°C.

The lattice arrangement in the core during most of these measurements consisted of a slab loading $3 \times 8 + 1 = 25$ elements (including 3 shim rods). On Oct. 25, 1954 this was altered to give a $3 \times 7 + 1$ loading with three elements removed from the north end of the lattice and placed in lattice positions 45, 55, and 56 in front of HB-5 on the west side of the reactor.

Facility	Information	Work Done By
HR-1	Cadmium ratio – 3.2 nv (thermal) – 4.12×10^{12} neutrons/cm^2/sec nv (>0.1 ev < 1 Mev) – 1.05×10^{12} neutrons/cm^2/sec Note: These measurements were made at a power level of 770 kw with a smaller core loading than used at present. To correct to present conditions, multiply fluxes by 2.75. The cadmium ratio should be the same. Ratio of bare foil activation to cadmium-wrapped foil activation: Mn^{55} – 44.5 Na^{23} – 66.0 Cl^{37} – 73.3 V^{51} – 76.8 Al^{27} – 80.0 Reactor loading at time of measurement: Fuel and shim rod positions – 12, 13, 14, 15, 16, 17, 18, 22, 23, 24, 25, 26, 27, 28, 32, 33, 34, 35, 36, 37, and 38 Beryllium in other spaces	J. B. Trice, December 1951; quoted in ORNL-1261

Facility	Information	Work Done By
HB-2	nv (thermal) at 1000 kw – 1.1×10^{13} neutrons/cm^2/sec nv (>0.6 Mev) at 1000 kw – 7.0×10^{12} neutrons/cm^2/sec nv (>2.6 Mev) at 1000 kw – 2.0×10^{12} neutrons/cm^2/sec nv (>4.0 Mev) at 1000 kw – 7.0×10^{11} neutrons/cm^2/sec nv (>6.2 Mev) at 1000 kw – 2.5×10^{11} neutrons/cm^2/sec nv (>8.1 Mev) at 1000 kw – 7.4×10^{10} neutrons/cm^2/sec Ratio of bare foil activation to cadmium-wrapped foil activation: Mn^{55} – 19 Al^{27} – 26 Na^{23} – 30 Cl^{37} – 38 V^{51} – 41 Thermal neutron traverse in HB-2 containing partial graphite plug (1000 kw): Distance from Inner End — nv (thermal) 0 — 1.1×10^{13} neutrons/cm^2/sec 5 in. — 9.0×10^{12} neutrons/cm^2/sec 10 in. — 6.0×10^{12} neutrons/cm^2/sec The report contains spectrum curves and curves of fluxes vs distance from inner end of hole.	J. B. Trice, June and July 1952, reported in ORNL-1359
HB-1	The beam strength through the velocity selector collimator is 2×10^{12} neutrons/cm^2/sec for dE/E neutrons at 3000 kw. The inner end of the collimator is separated from fuel by 6 in. of beryllium.	E. C. Smith
HB-4	At power level of 1500 kw, the thermal neutron flux in a $1\frac{3}{8}$ in. beam measured at 8 in. from the opening at the shield face was 5.6×10^8 neutrons/cm^2/sec. A 6 in. layer of beryllium was between the fuel and the inner end of the collimator.	F. Pleasanton and H. L. Reynolds, June 1952
HB-2	A spectrum measurement was made and is reported in CF-53-12-51. Measurements were made with the BSF fast neutron spectrometer.	R. G. Cochran and K. M. Henry, June 1953

Appendix III

LITR INFORMATION

COOLING SYSTEM

Water coolant flow rate	1200 gpm
Water volume of system	~10,000 gal
Water temperature in core	~45°C
Maximum water temperature	120°F

Cooling is achieved by two Trane CG 77-40 air coolers plus a water-cooled heat exchanger

Major radioactive contaminants

Samples counted about 2 hr after coming from reactor in a well-type scintillation counter.

The Na^{24} is formed from sodium impurities and from (n,α) on Al	Na^{24} ~27%
The source of the Np^{239} is not definitely known but it may be diffusing from the fuel into the water	Np^{239} ~57%
Total activity	~50,000 counts/min/ml

Water quality

Cooling water specific resistance generally above	400,000 ohm-cm
pH maintained between	5.5 and 6.5

Water purification system – Quality control achieved by a bypass demineralizer with the following specifications:

Flow rate — 15 gpm

Resin columns

2 Cation columns — 2 ft^3 each cation resin

Regeneration – 10 lb H_2SO_4/ft^3 of resin as 5% solution put through column at rate of 1 gpm/ft^3 of resin

Rinse – 1 gpm/ft^3

Operation – Water is passed through the column at rate of 15 gpm either singly or in parallel

Efficiency – Radioactivity in the water is reduced by factor of 6–10

Monobed Unit

The monobed unit is preceded by cation columns to reduce the damage of the anion resin by radiation. Since the cation resin columns remove about 90% of the radioactivity at 15 gpm, the amount of radioactivity in the monobed column is much reduced.

Volume	55 ft^3
Anion resin	20.6 ft^3, Nalcite SAR
Cation resin	13.4 ft^3, Nalcite HCR

Regeneration of cation resin

Backwash and separation – 4–6 gpm/ft^2 for 30 min

5% H_2SO_4 at 1 gpm/ft^3

10 lb H_2SO_4/ft^3 of resin

Rinse – approximately 1 gpm/ft^3 for 4 hr

Regeneration of anion resin

- Backwash and separation – 4–6 gpm/ft^2 for 30 min
- 5% NaOH at 0.5 gpm/ft^3
- 5 lb $caustic/ft^3$ of resin
- Rinse – approximately 0.5 gpm/ft^3 for 4 hr

Effluent of monobed column

Specific resistance	$1–2 \times 10^6$ ohm-cm
pH	5.5 – 6.5