...NET NO. 50-22
formal file

Westinghouse

ELECTRIC CORPORATION

TESTING REACTOR

July 11, 1960

P.O. BOX 1075
PITTSBURGH 30, PA.

Mr. H. L. Price, Director
Division of Licensing and Regulation
U. S. Atomic Energy Commission
Washington 25, D. C.

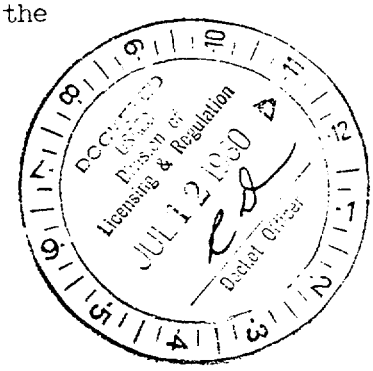
Dear Sir:

Subject: License No. TR-2
Docket 50-22

Transmitted herewith are forty (40) copies of a report, WTR-49, which contains a description, analysis and conclusions concerning the partial destruction of a fuel element in the Westinghouse Testing Reactor on April 3, 1960.

With reference to the Order sent with your letter of June 30, 1960 we wish to make the following comments concerning items 1, 2, and 3, page 1:

1. The direct cause of the partial melting of one fuel element can never be known with complete certainty. However, as indicated in WTR-49, considerable circumstantial evidence exists that a defective fuel tube was responsible. For example, calculations indicate that a heat transfer defect in the element, in the order of 1/2-inch in diameter, could under certain circumstances have caused the tube to melt. Recent examination of fuel tubes from the same lot as the injured one revealed that 34% of these tubes have one or more defects larger that 1/2-inch. Inspection of the failed element further indicates a peculiar pattern of element melting; one plausible reason for which is bonding failure at the end of the element. The majority of the defects noted in the recent inspection have been near the ends of the elements.



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In contrast, no evidence whatever has been found of inadequate coolant flow due to boiling or other causes. On the contrary, calculations from temperature measurements taken in a duplicate element symmetrical to the one which failed give good evidence that the flow and cooling were proper and as specified. Further calculations indicate that hypothesized reductions of 15% of the total coolant flow at the time of the incident would not have caused any trouble.

2. The presumption that "the incident might have been substantially minimized if the WTR reactor operators had been provided with specific detailed instructions relating to operation of the facility when a sudden change in reactivity occurs" is highly speculative. The evidence presented in WTR-49 is that the fuel element failed, melted, and completely blocked the coolant channel before the operating error occurred. Thus the probability of any additional release of fission products to the primary coolant as the result of lack of detailed operating instructions seems slight.

3. From the philosophical viewpoint of complete containment we agree that the present venting system has a design deficiency. We propose to modify the system to protect against releases of fission products to the atmosphere. The proposed method of modifying the vent system is being submitted by separate letter dated July 8, 1960.

We will continue with the metallurgical analysis of the failed fuel element as outlined in Appendix IV, WTR-49. However, it is believed that no further significant information relating to the safety of plant operation will result from this examination. We therefore request that Section I, WTR-49, be considered as providing the information requested by item 1, page 2, of your letter.

Certain corrective actions have been taken as a result of this incident. They are:

1. The initiation of a rigid inspection program of all cold fuel elements now on hand and currently being manufactured. Details of the new inspection requirements are presented in WTR-49, page 19. Fuel with defects larger than an equivalent diameter of approximately one-eighth (1/8)-inch will not be used. The presence of such defects will be determined by ultrasonic means or by any better means which may become available.

2. All members of the WTR Operations Department have been instructed on the hazards of fast negative reactivity changes. The WTR Operating Procedures P-107 have been revised to cover the operation of control rods subsequent to sudden changes in reactivity.

3. The 60 MW power escalation program will be modified to limit the amount of boiling in the core to a value below that permitted by License No. TR-2. At no time in the escalation program or the early 60 MW operating cycles will the boiling pattern be permitted to be more severe than the proven pattern of Figure 13, WTR-49, 8000 gpm case.

Based upon the analysis of and conclusions concerning the incident presented in WTR-49 and the proposed modifications to the vent system for the process water head and surge tanks described in WTR-51, we request your prompt written approval to modify the vent system and to load, start up and operate the WTR in accordance with the terms and provisions of License No. TR-2.

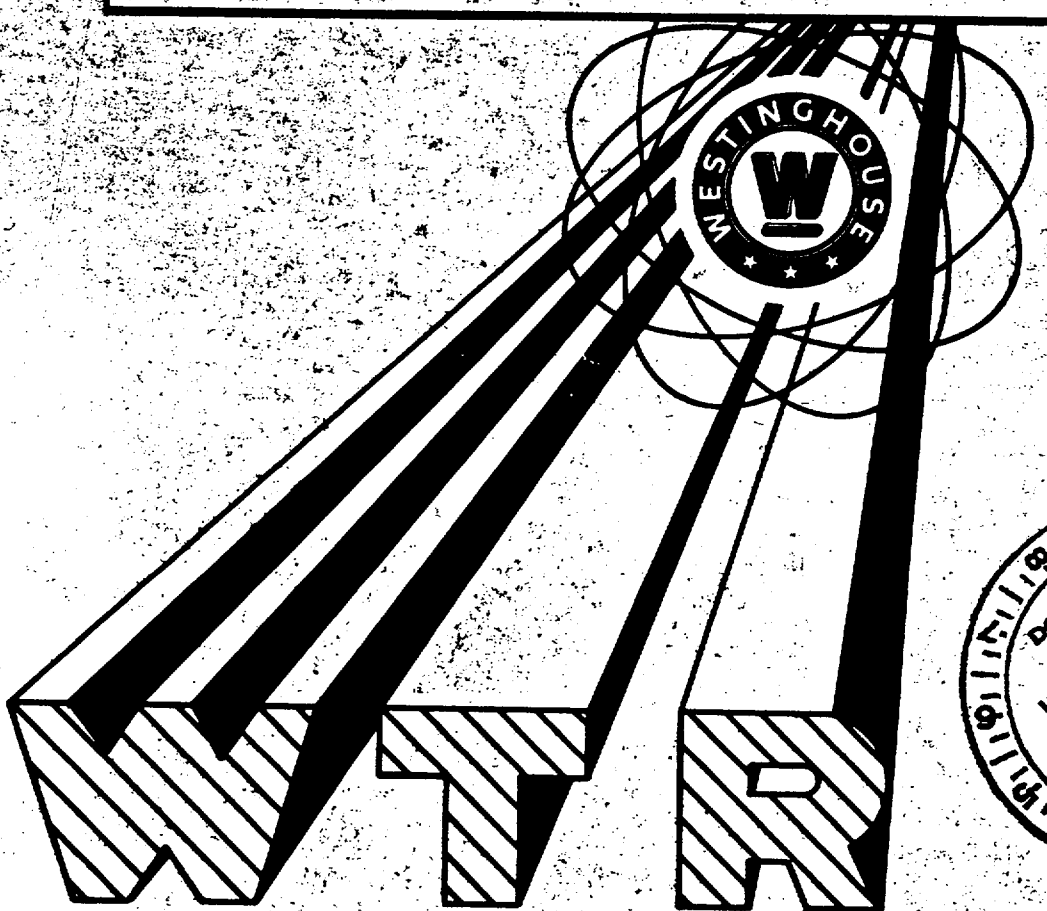
Sincerely yours,

E. T. Morris

E. T. Morris
General Manager

REPORT ON WTR FUEL ELEMENT FAILURE APRIL 3, 1960

JULY 7, 1960

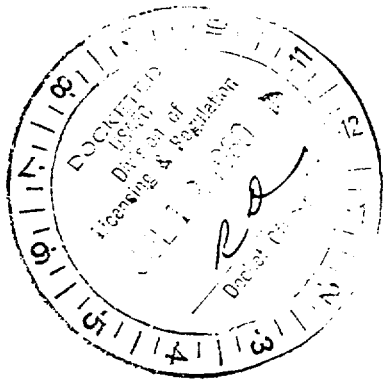


WESTINGHOUSE TESTING REACTOR
P. O. BOX 1075
PITTSBURGH 30, PENNSYLVANIA

REPORT ON
WTR FUEL ELEMENT
FAILURE APRIL 3, 1960

PREPARED
by

THE STAFF OF
THE WESTINGHOUSE TESTING REACTOR



— JULY 7, 1960 —

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF FIGURES

FUEL ELEMENT FAILURE
IN THE
WESTINGHOUSE TESTING REACTOR

A. Introduction

On April 3, 1960 at approximately 8:40 P.M. a fuel element failure occurred in the Westinghouse Testing Reactor, accompanied by a release of fission products to the primary coolant system and a discharge of some gaseous fission products to the atmosphere. The following report describes the occurrence in detail, including a description of the operating conditions, and the sequence of events before, during, and after the occurrence. A thermal and hydraulic analysis of the incident is presented together with an interpretation of the observed data and a determination of burn-out heat flux for the operating conditions. Also included are the results of the inspection of cold fuel elements on hand and the progress to date in the examination of the failed fuel element.

B. License Requirements for Power Escalation Program

The WTR was originally licensed to operate at a maximum power level of 20 megawatts (thermal) by License No. TR-2. This license was amended on January 8, 1960 to permit operation at a maximum of 60 megawatts (thermal) with the following restrictions:

1. "Westinghouse shall retain the bubble formation apparatus and the special detection channel described in the application in the reactor during the power escalation program until stable operation at 60 megawatts thermal power level has been established;

2. The ratio of the maximum heat flux in the reactor to the burnout heat flux shall never exceed one-half;

3. The reactor shall not be operated in such a way that the ratio of core steam void volume to core coolant volume exceeds one percent; and

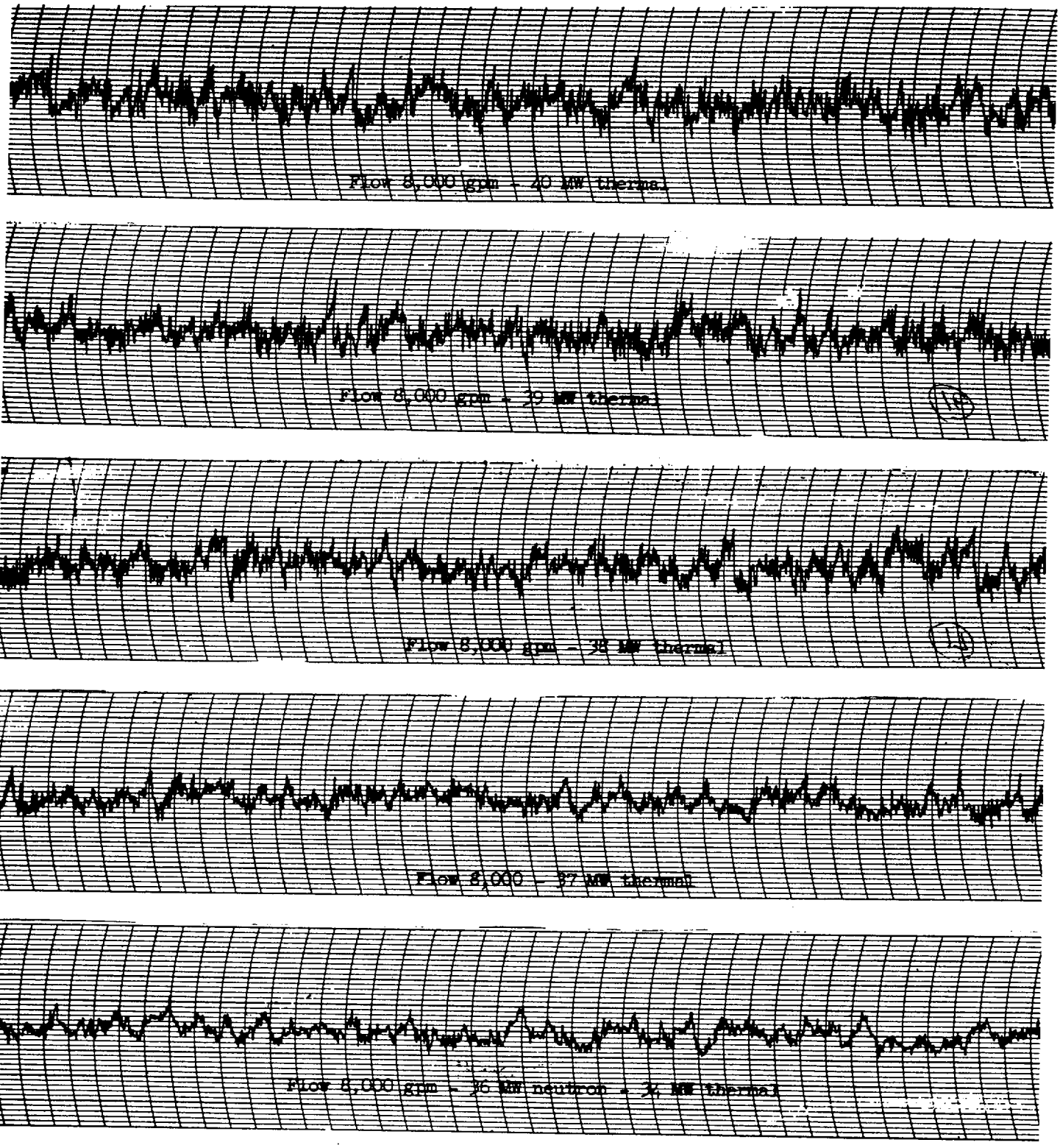
4. When the reactor is being operated with the automatic control system, the magnitude of boiling induced neutron level perturbations shall not exceed 5 percent or whatever lesser value is necessary to prevent erratic behavior of or oscillatory interaction between the boiling phenomenon, the reactor power level and the automatic control system."

The special bubble detection channel mentioned in the above restrictions is a sensitive ionization chamber connected to a fast Brush recorder. Its operation is described in detail in WTR-27, submitted to the Commission on November 11, 1959 in conjunction with Amendment No. 14 to License TR-2. A later modification to this apparatus provided a servo controlled dc level bucking voltage such that the recorder only indicated the ac variations in level.

This apparatus had been connected as required during many power escalation tests prior to April 3. A typical run is indicated in Figure 1. For reference purposes the scale calibrations on these runs are: one large block in amplitude equals 50 KW in power level; the chart speed is 1 cm/sec.; and these runs were at void percentages smaller than 0.1 percent. For comparison a typical "noise" trace reported for the ORR (CF-59-8-39) is shown as Figure 2.

The reactor had also been operating under another restriction specified in the application for license amendment. This restriction was that the bulk water temperature from an element in the first fuel ring of the core would not be permitted to exceed 220°F during the escalation steps.

To measure this temperature, a fuel element was initially instrumented with three aluminum clad fiber glass insulated chromel-alumel thermocouples projecting into the nozzle space below the element. These thermocouples were connected to a printing data-logger which was used to record the temperatures. The fuel element was placed in core position L-7-6, shown in Figure 3.

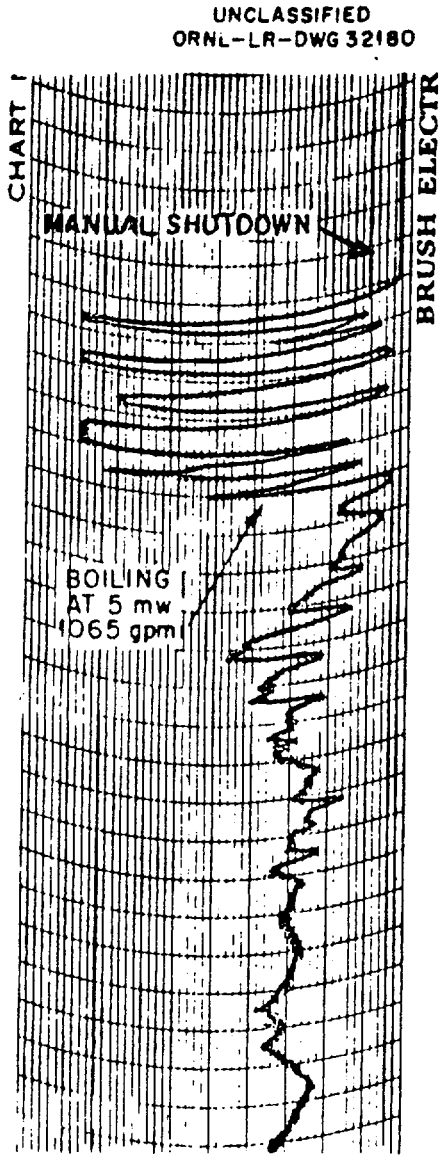


Noise Level vs Power

Constant Flow at 8000 gpm

Scale: 1 Large Block = 50 KW
Access Tube In

Figure 1



Tracing from Brush Recorder Showing Typical Onset of Boiling.

FIGURE

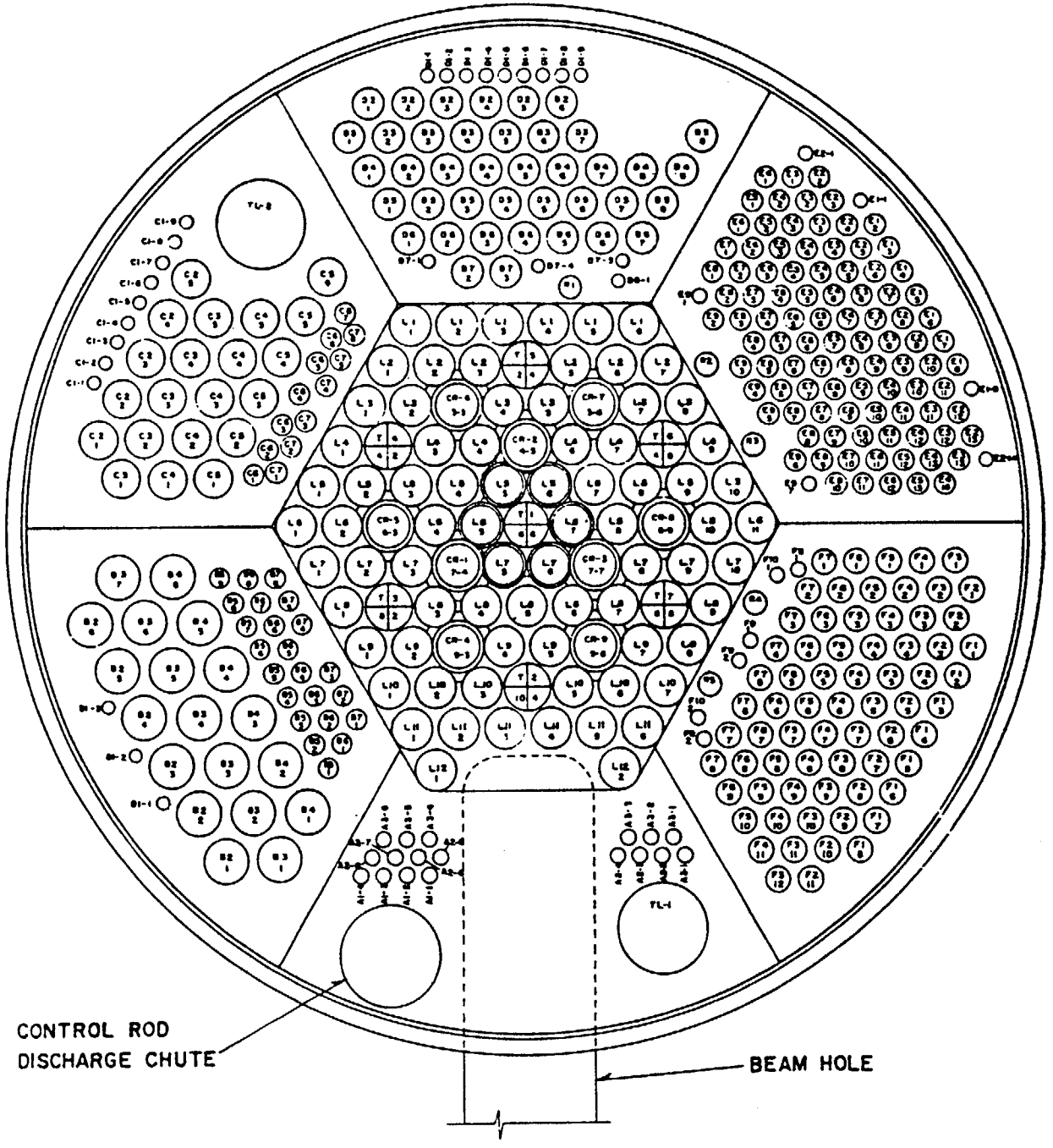


FIGURE 3.
CORE POSITION INDEX.

FUEL ELEMENTS

CONTROL RODS

HIGH PRESSURE THIMBLES

A,B,C,D,E & F HOLES ARE
REFLECTOR POSITIONS

R-HOLES RABBIT FACILITY

During the early stages of the power escalation program, the readings from these thermocouples agreed with those predicted theoretically from a consideration of coolant flow and power production parameters. However, within a few days, a drift upward was observed in the thermocouple readings which was not related to the gross power and flow values in the reactor. The thermocouple readings continued to drift upward and eventually two of them rose to a value considerably in excess of 220°F. (Investigation subsequent to the failure of the fuel element indicated that these symptoms were common for water leakage into aluminum clad thermocouples.)

At this point the thermocouples were assumed to have failed and the reactor was shut down while the thermocouples were replaced by four new thermocouples arranged in a similar geometry. The new thermocouples then read temperatures which agreed with predicted values.

With time the second set of thermocouples exhibited a similar drift and eventually indicated in excess of 220°F. Figure 4 shows this effect by plotting the increase in the ratio of the temperature rise across the instrumented fuel element to the temperature rise across the reactor vessel as a function of time. This ratio should remain constant independent of the reactor power and the coolant flow rates, and for the calculated conditions should be 1.95.

Power escalation was again interrupted and a third instrumented fuel element was constructed using stainless steel clad, magnesium oxide insulated thermocouples. These thermocouples were mounted as shown in Figure 5 with two thermocouples reading the bulk exit water temperatures, two reading the discharge water temperature in the channel between the sample basket and the first fuel tube, and the third pair reading the water discharge temperature between the inner and middle fuel tubes. This instrumented fuel element was inserted in reactor core position L-5-6. Although the second set of aluminum sheathed thermocouples were considered defective, they were not removed from the core and their

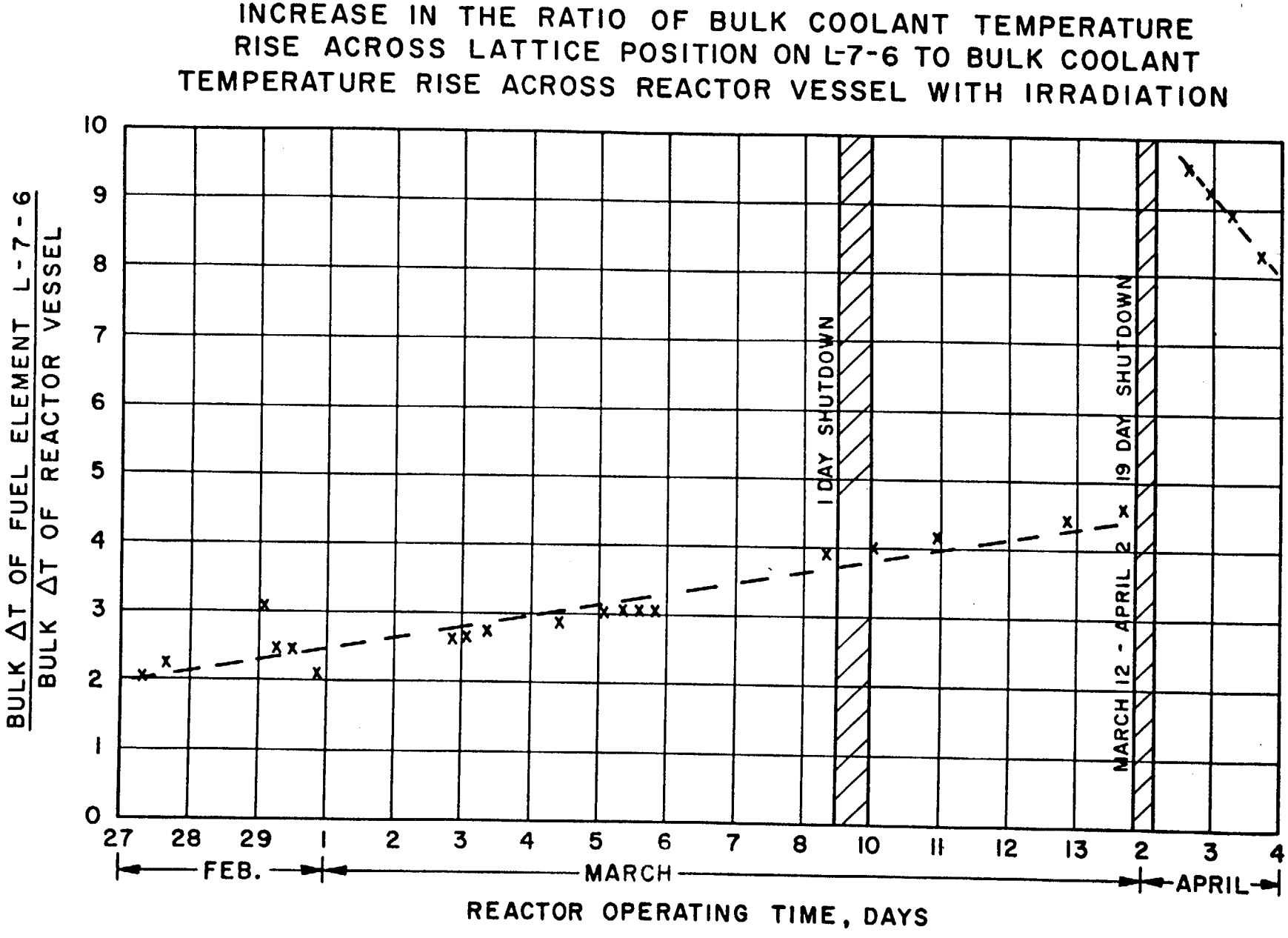
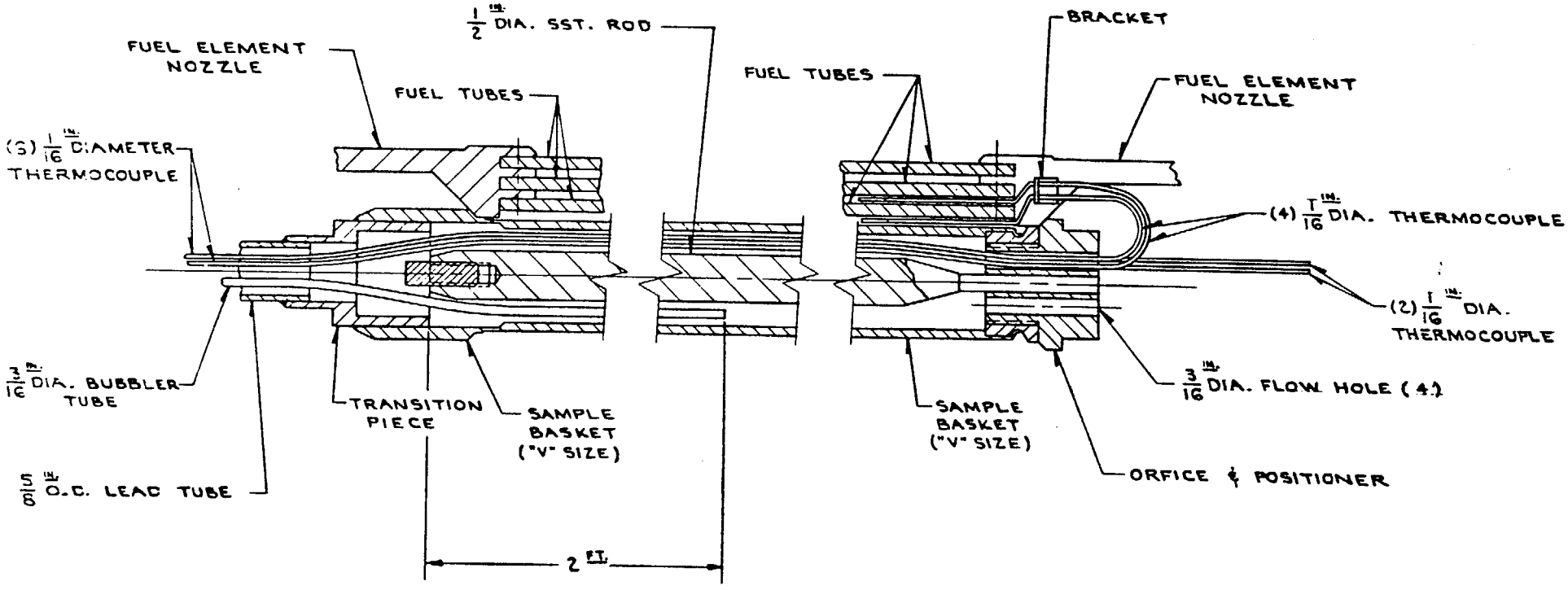

INCREASE IN THE RATIO OF BULK COOLANT TEMPERATURE RISE ACROSS LATTICE POSITION ON L-7-6 TO BULK COOLANT TEMPERATURE RISE ACROSS REACTOR VESSEL WITH IRRADIATION
BULK ΔT OF FUEL ELEMENT L-7-6 / BULK ΔT OF REACTOR VESSEL
0
1
2
3
4
5
6
7
8
9
10
1 DAY SHUTDOWN
MARCH 12 - APRIL 2 19 DAY SHUTDOWN
27 28 29 1 2 3 4 5 6 7 8 10 11 12 13 2 3 4
FEB.
MARCH
APRIL
REACTOR OPERATING TIME, DAYS

FIGURE 4



FUEL ELEMENT INSTRUMENTED WITH SIX STAINLESS STEEL THERMOCOUPLES

FIGURE 5

outputs were periodically read. A typical set of readings of all the thermocouples as taken on the night of April 3 is shown in Figure 6. In this figure the columns headed W, V, X, U are readings of the defective aluminum sheathed thermocouples and the columns headed #1 - #6 are for the stainless steel sheathed thermocouples. At 16:46 (4:46 P.M.) the reactor was operating at 40 MW with 15,000 gpm primary coolant flow. The reactor bulk water inlet temperature was 126°F. The thermocouples of the newly instrumented element indicated temperatures which agreed closely with predicted values.

C. Early Power Escalation Runs with Access Tubes

During the early power escalation runs five access tubes had been installed in the reactor. These access tubes were one inch diameter aluminum pipes sealed at the bottom end and filled with stagnant water under atmospheric pressure. The tubes entered the reactor vessel through one of the top access ports. These tubes had been used in reactor calibration experiments such as gamma heating and power calibration by foil activation. Three of these tubes entered the fuel elements in core positions L-5-6, L-5-8, and L-3-8 and two were in reflector positions E-8-5 and E-5-7. These access tube locations are shown in Figure 7.

Prior to undertaking the power escalation program, Test Specification T-Spec 5-1, was written to establish the values of power and flow for the various steps of the escalation program. This specification is included as Appendix I. The operating parameters were chosen to be consistent with the heat transfer work reported in WTR 25, also submitted with Amendment No. 14 to License No. TR-2. The total primary coolant flow for normal operation was presumed to be twice the core flow shown on Figure 5 of the above referenced report. This curve, including some of the experimental points previously obtained, is presented here as Figure 8. During the actual power escalation, using the bubble detector, the flow was reduced to 83 percent of this normal value.

INSTRUMENTED FUEL ELEMENT TEMPERATURES

SUNDAY APRIL 3, 1960

Time	Fuel Element W	V	X	U	Beam Hole	Inst. Fuel Element #4	#6	#2	#1	#5	#3	
16:46	270.	276.	295.	276.	.081.	082.	174.	161.	150.	150.	163.	179.
17:01	271.	275.	296.	274.	.082.	082.	175.	161.	151.	151.	164.	179.
17:16	270.	274.	295.	274.	.082.	082.	174.	160.	150.	150.	163.	179.
17:31	269.	274.	294.	274.	.082.	082.	175.	160.	150.	150.	163.	179.
17:46	269.	274.	298.	274.	.082.	082.	174.	160.	150.	150.	163.	179.
18:01	269.	274.	306.	274.	.081.	082.	174.	160.	150.	150.	163.	179.
18:16	268.	273.	302.	274.	.082.	082.	174.	160.	150.	150.	163.	178.
18:31	268.	273.	306.	274.	.082.	082.	174.	160.	150.	149.	163.	179.
18:46	268.	274.	306.	274.	.081.	082.	174.	160.	150.	150.	163.	179.
19:01	267.	273.	303.	274.	.082.	082.	174.	160.	150.	150.	163.	179.
19:16	223.	247.	254.	225.	.082.	082.	160.	149.	142.	141.	151.	164.
19:31	210.	214.	233.	211.	.082.	082.	153.	143.	136.	136.	145.	157.
19:46	208.	212.	230.	208.	.082.	082.	149.	140.	133.	134.	142.	154.
19:55	206.	210.	223.	206.	.082.	082.	148.	139.	131.	131.	141.	152.
19:57	206.	210.	223.	206.	.082.	082.	148.	140.	131.	131.	141.	153.
20:00	206.	210.	223.	205.	.082.	082.	147.	139.	130.	130.	140.	152.
20:02	217.	222.	238.	219.	.082.	083.	166.	152.	140.	140.	156.	174.
20:04	234.	244.	260.	240.	.082.	082.	200.	176.	156.	157.	184.	206.
20:07	250.	262.	276.	260.	.082.	082.	218.	191.	165.	168.	201.	224.
20:09	256.	267.	282.	262.	.082.	083.	219.	192.	164.	168.	201.	224.
20:11	251.	266.	275.	259.	.082.	083.	218.	191.	164.	166.	200.	221.
20:13	248.	259.	268.	254.	.082.	083.	218.	189.	162.	164.	195.	219.
20:16	244.	262.	268.	251.	.082.	083.	214.	186.	161.	164.	196.	217.
20:18	241.	256.	266.	248.	.082.	083.	212.	186.	160.	162.	194.	216.
20:20	241.	255.	262.	247.	.082.	083.	212.	186.	160.	162.	195.	220.
20:22	266.	280.	290.	272.	.082.	083.	233.	200.	168.	173.	212.	237.
20:25	281.	293.	306.	286.	.082.	083.	238.	204.	172.	176.	214.	240.
20:27	274.	290.	301.	283.	.082.	083.	233.	200.	168.	173.	212.	236.
20:29	274.	289.	298.	279.	.082.	083.	232.	201.	169.	173.	211.	235.
20:31	271.	284.	296.	280.	.082.	082.	233.	202.	169.	173.	210.	235.
20:34	279.	294.	306.	288.	.082.	083.	249.	211.	174.	179.	212.	227.
20:36	184.	189.	202.	185.	.082.	082.	173.	157.	141.	143.	164.	180.
20:38	285.	295.	311.	287.	.082.	083.	245.	210.	175.	180.	220.	245.
20:40	240.	203.	187.	156.	.082.	082.	117.	116.	114.	114.	116.	117.
20:43	118.	118.	129.	117.	.082.	083.	114.	114.	113.	113.	114.	114.
20:45	116.	116.	124.	116.	.083.	083.	113.	114.	112.	113.	113.	113.
20:47	116.	116.	124.	115.	.082.	083.	113.	113.	113.	113.	114.	114.
20:49	116.	116.	123.	116.	.082.	083.	114.	114.	112.	113.	114.	114.
20:52	115.	116.	122.	115.	.082.	083.	113.	114.	113.	113.	114.	114.
20:54	115.	115.	122.	115.	.082.	083.	113.	113.	113.	113.	114.	113.
20:56	116.	116.	122.	115.	.082.	083.	113.	113.	113.	113.	113.	114.
20:58	114.	114.	120.	113.	.082.	083.	112.	112.	111.	111.	111.	112.
21:01	112.	112.	117.	111.	.082.	083.	110.	110.	109.	109.	110.	110.
21:03	110.	110.	114.	110.	.082.	083.	108.	108.	106.	107.	108.	108.
21:05	107.	107.	112.	107.	.082.	083.	105.	106.	105.	105.	105.	105.
21:07	106.	106.	110.	106.	.082.	083.	105.	104.	104.	104.	105.	105.
21:10	104.	104.	106.	103.	.083.	084.	102.	102.	102.	102.	102.	102.
21:12	103.	103.	106.	102.	.082.	083.	102.	102.	101.	102.	102.	102.
21:14	102.	102.	105.	102.	.082.	083.	100.	100.	100.	100.	100.	100.
21:16	101.	100.	102.	099.	.082.	083.	098.	098.	098.	098.	098.	098.
21:19	101.	100.	102.	099.	.082.	083.	100.	100.	099.	099.	101.	101.
21:21	102.	102.	104.	102.	.082.	083.	101.	100.	099.	099.	100.	100.
21:23	102.	102.	104.	101.	.082.	082.	101.	100.	099.	099.	100.	100.

#4 2 Channel
#6 Bulk Water
#2 Basket Channel
#1 Basket Channel
#5 Bulk Water
#3 2 Channel

Reduce Power to 30 MW

1950 Reduced Flow Started

P.C. Flow at 5250 (8 P.M.)

17 MW

20:44 Scram Reactor

Figure 6

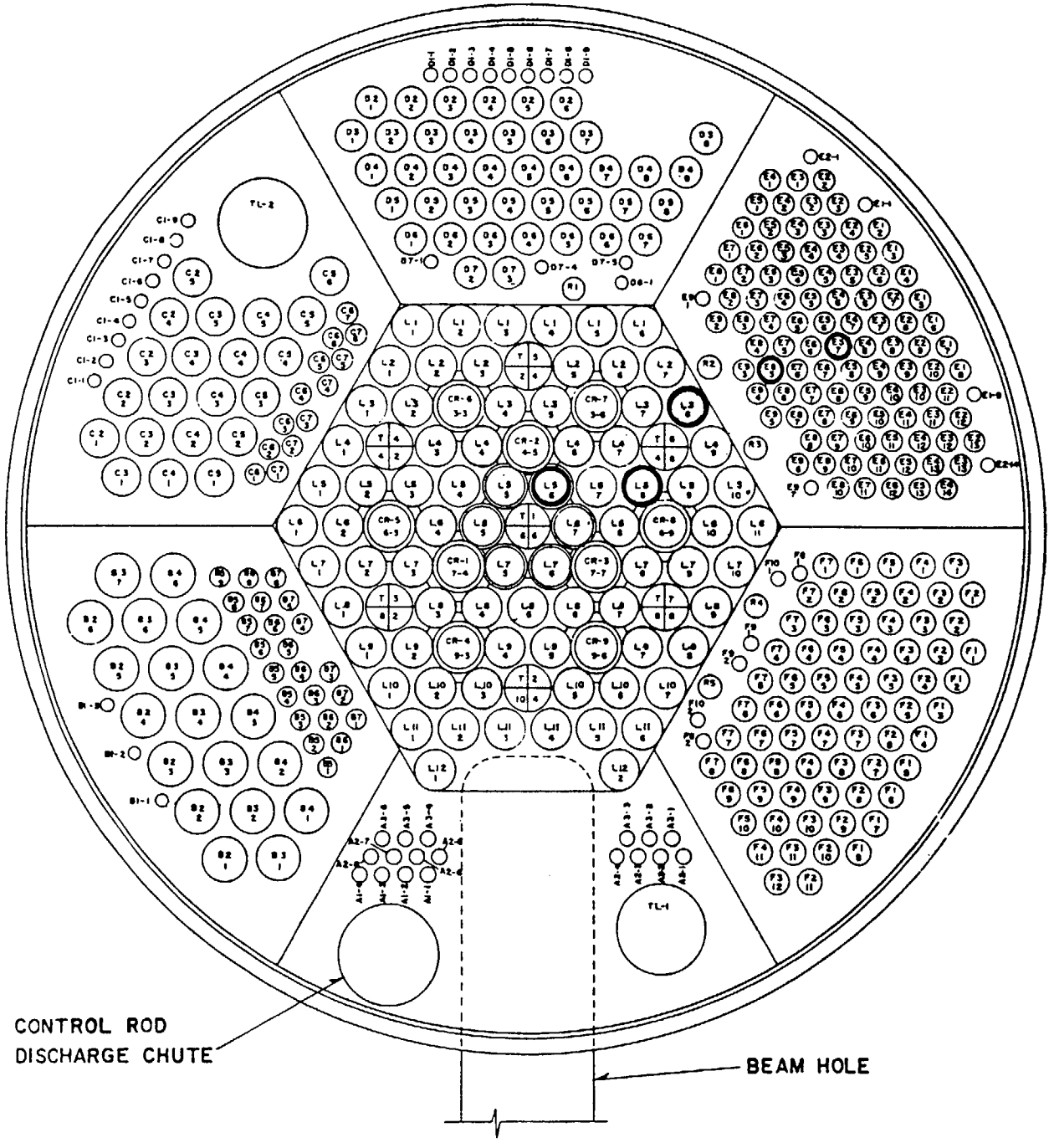


FIGURE 7
ACCESS TUBE POSITIONS

ACCESS TUBES

FUEL ELEMENTS

CONTROL RODS

HIGH PRESSURE THIMBLES

A,B,C,D,E & F HOLES ARE
REFLECTOR POSITIONS

R-HOLES RABBIT FACILITY

POWER TO INITIATE
BOILING VS. CORE FLOW

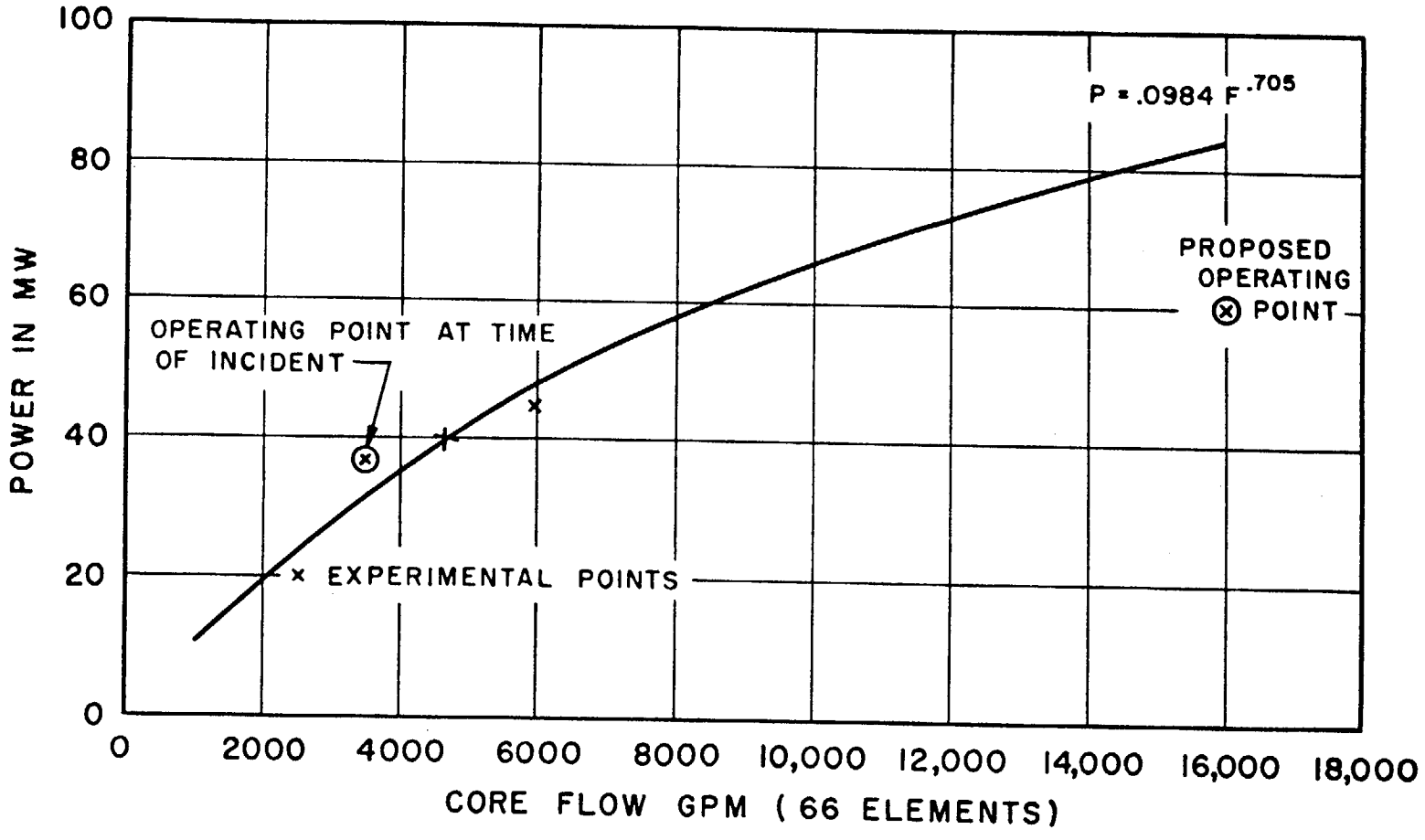


FIGURE 8

The power escalation program proceeded on the following schedule:

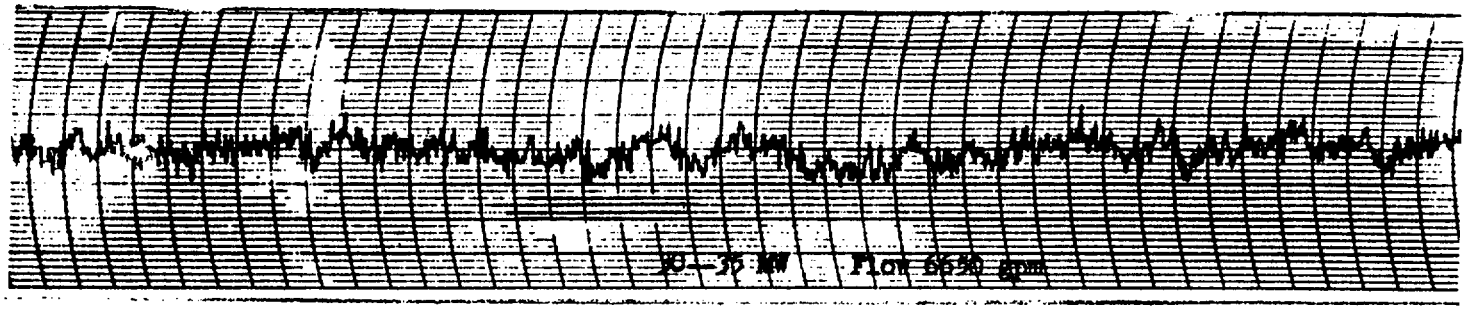
Power Increase	Date
20 - 25	Feb. 21, 1960
25 - 30	Feb. 24, 1960
30 - 35	Feb. 27, 1960
35 - 40	Feb. 29, 1960
40 - 45	Mar. 7, 1960

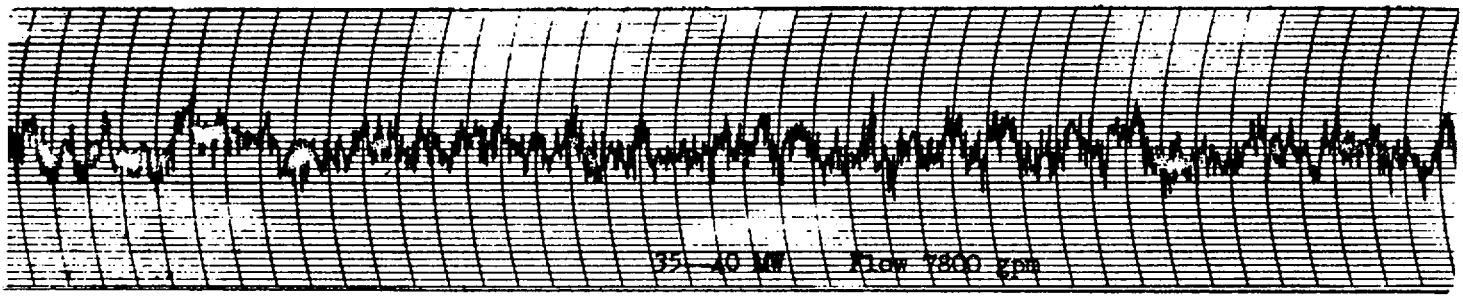
A portion of the "bubble detector" record for the last three steps is given in Figure 9.

As the program progressed and the power level was increased, boiling noise was indicated at slightly higher flows than were predicted by Figure 8. At this time, however, it was established that boiling was occurring in the access tubes, and not distributed over the core.

Boiling in the access tube in position L-5-8 was substantiated by traversing a thermocouple axially along the tube from the bottom to the top of the core region and observing the temperature. The resulting temperature versus position curve is shown as Figure 10. A maximum temperature of 248°F was measured in the region of maximum power production in the reactor. Saturation temperature of water under atmospheric pressure and 22 feet of head is 240°F. Consequently boiling in this tube took place over several inches of length.

It was believed that boiling in the access tubes was obscuring the detection of the initiation of boiling in the fuel channels and a decision was made to remove these tubes at the next shutdown. Meanwhile, several runs had been made using the boiling detector, and the system had been calibrated as to amount of void corresponding to noise pattern and amplitude of noise in terms of power output. Representative results of these runs are shown in Figures 11 through 14.





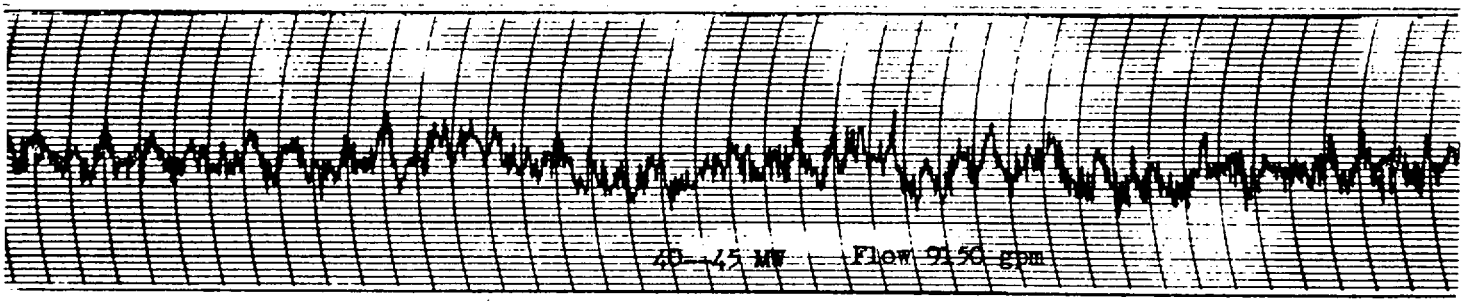
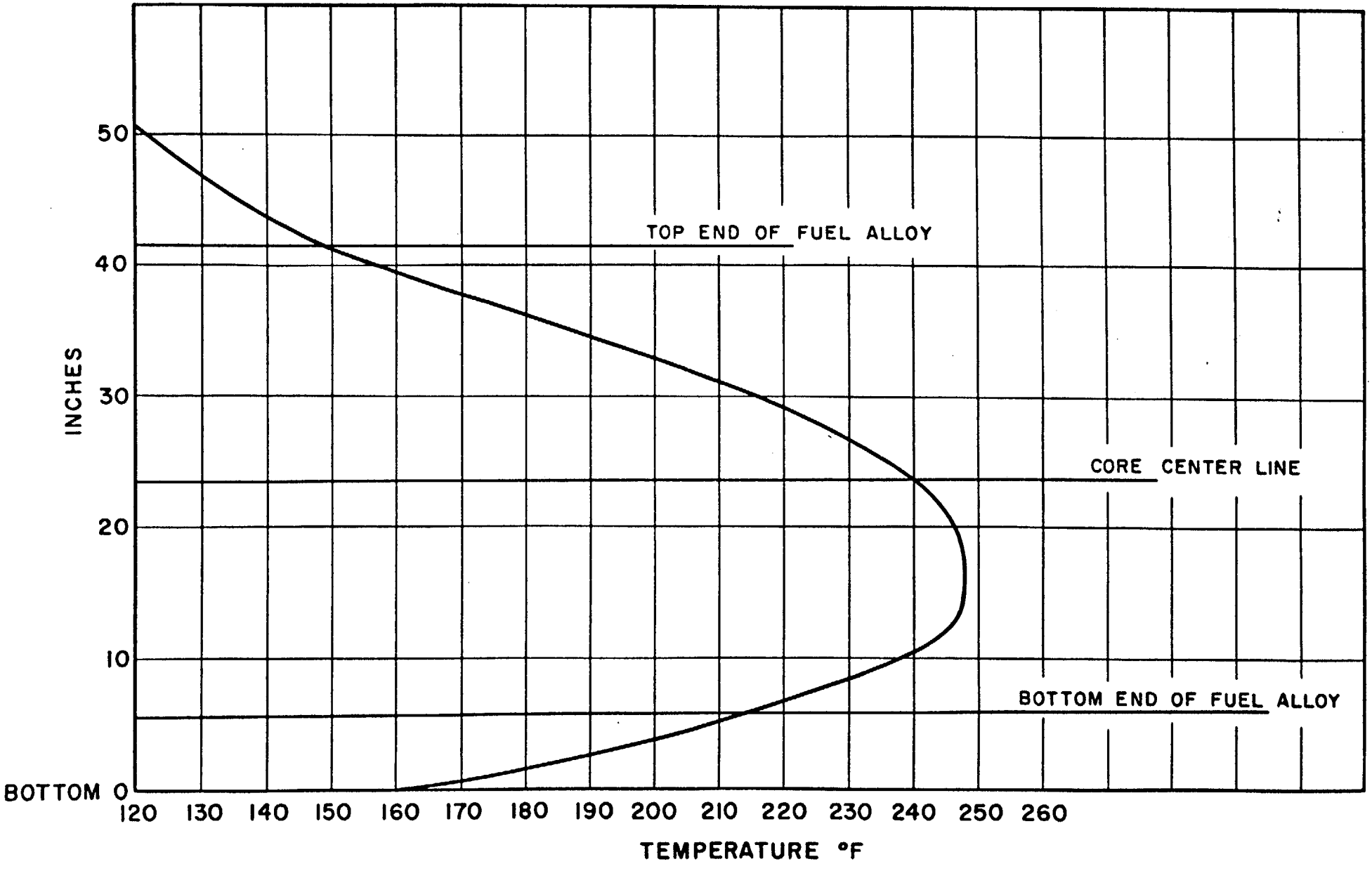


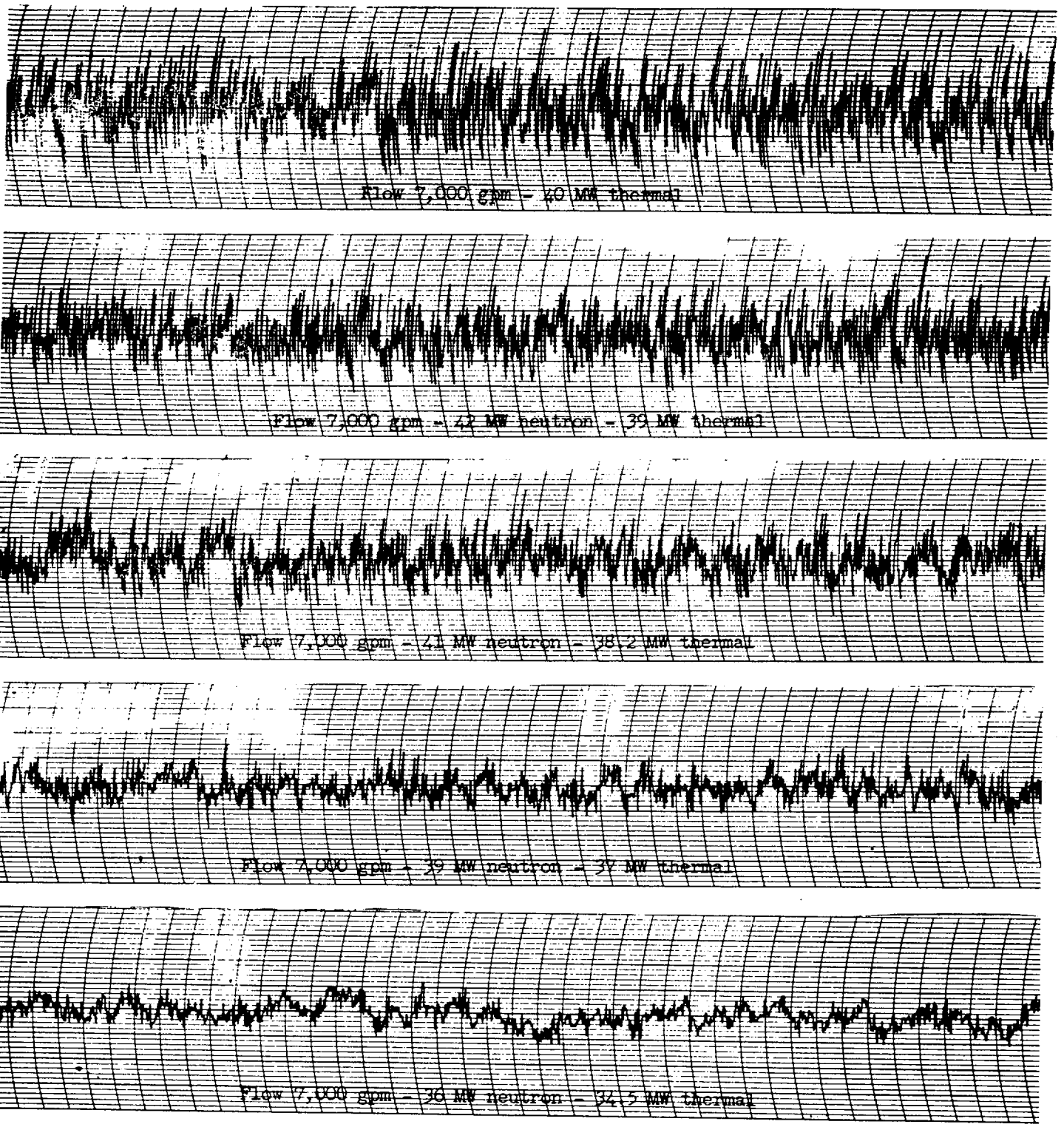
FIGURE 9

"Bubble Trace" During Power Increase

FIGURE 10

TEMPERATURE PROFILE L5-8
ACCESS TUBE 40MW 11000 GPM

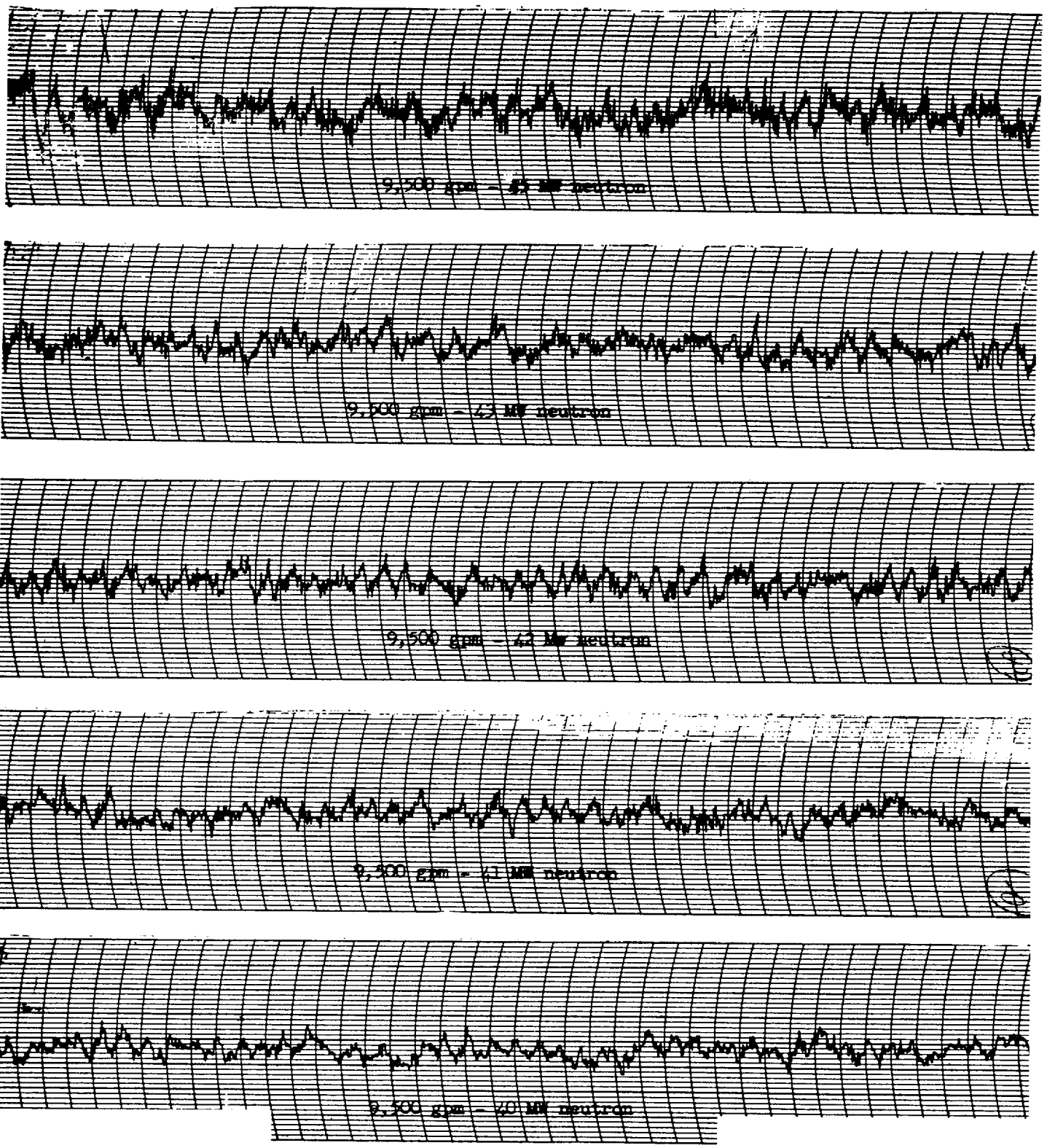




Noise Level as a Function of Power Output
Constant Flow at 7000 gpm
Scale: 1 Large Block = 50 KW

Access Tubes In

Figure 11



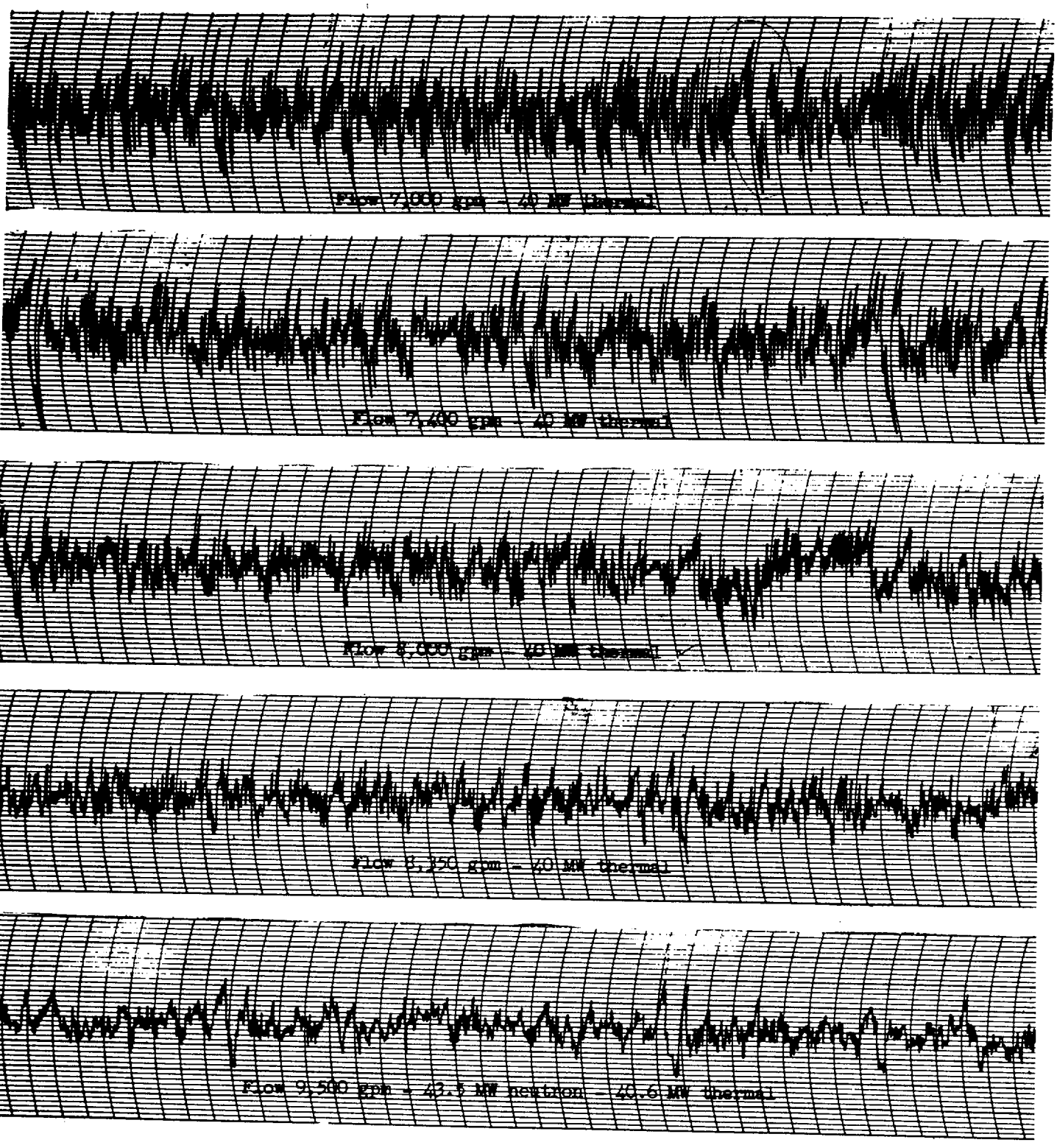
Noise Level as a Function of Power

Constant Flow at 9500 gpm

Scale: 1 Large Block = 50 KW

Access Tube In

Figure 12



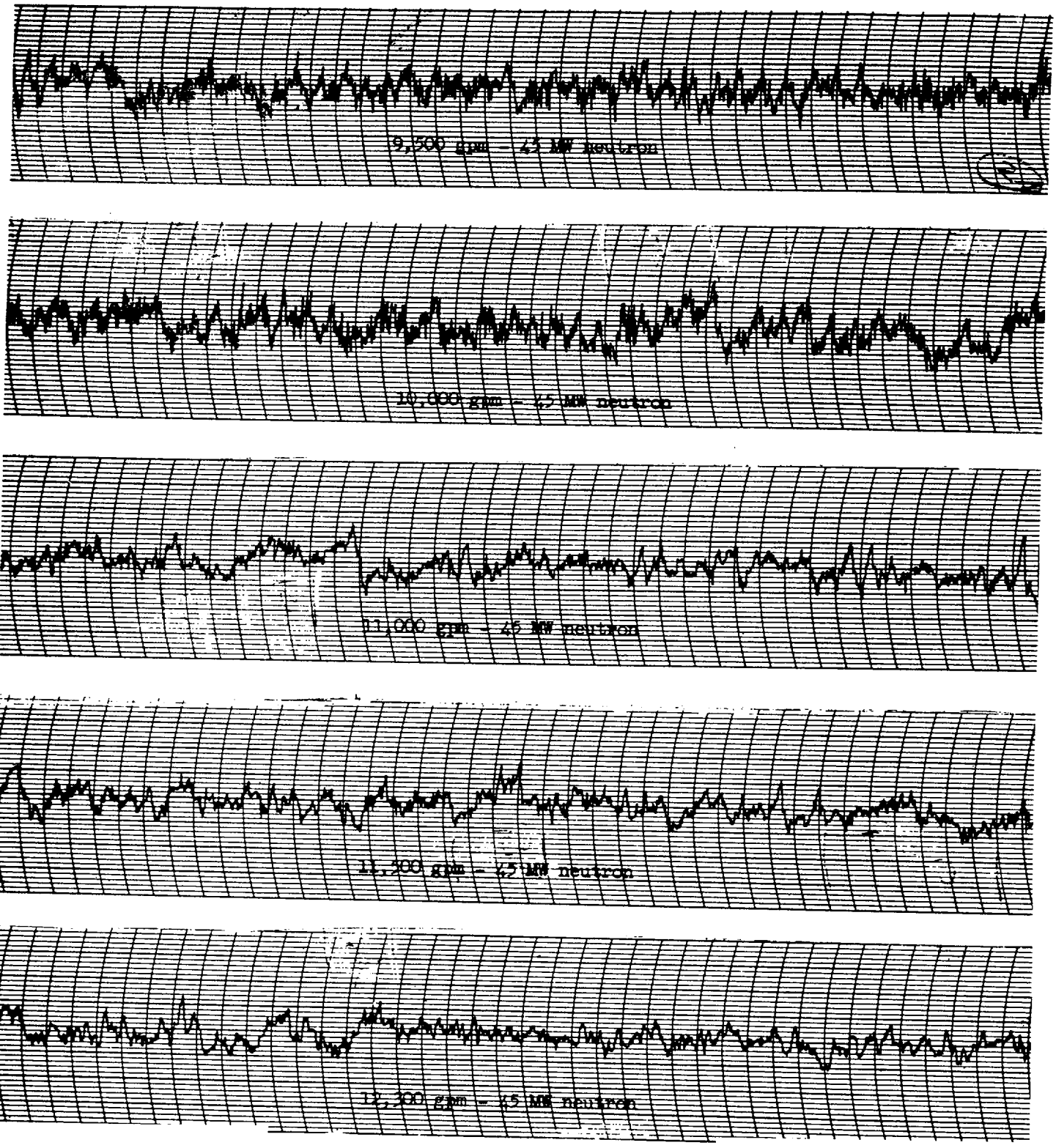
Noise Level as a Function of Flow

Constant Power 40 MW

Scale: 1 Large Block = 50 KW

Access Tubes In

Figure 13



Noise Level as a Function of Flow

Constant Power 45 MW

Scale: 1 Large Block = 50 KW
Access Tube In

Figure 14

D. Reactor Modifications

The reactor was shut down from March 14 to April 2 to make changes in the control assembly, the control magnets, and the control rod drives. These changes were required for operation at higher coolant flow rates needed at power levels above 45 MW. A detailed description of these changes is contained in WTR 35, submitted to the Commission with our letter dated February 16, 1960. At the same time the access tubes described above were removed from the reactor.

The reactor was scheduled to be shut down for an extended period at midnight on April 3. It was started up on April 2 for a two-day run to check out the new equipment that had been installed, to determine the effect of removing the access tubes, and to recalibrate the boiling detector at low coolant flow rates. Test Specification, T-Spec. 6-1 was issued to cover testing of the new equipment and T-Spec 5-2 delineated the low flow boiling experiment. A copy of the latter specification is included as Appendix II.

E. Reactor Core Loading - April 2nd Run

For this two-day run the reactor core loading in grams of fuel in each fuel element and the contents of the irradiation volume of each fuel element are shown as Figures 15 and 16. The fuel element which subsequently failed was in core position L-6-5, and contained 199 grams of fuel, indicating that it was a new element.

The following is a brief description of the experiments that were in the core:

L-6-5 (Failed Element) Nickel Wire

The experiment contained in this element was a fast neutron flux monitoring assembly. It consisted of a set of seven hairlike nickel wires, each separately encapsulated in a quartz capsule and

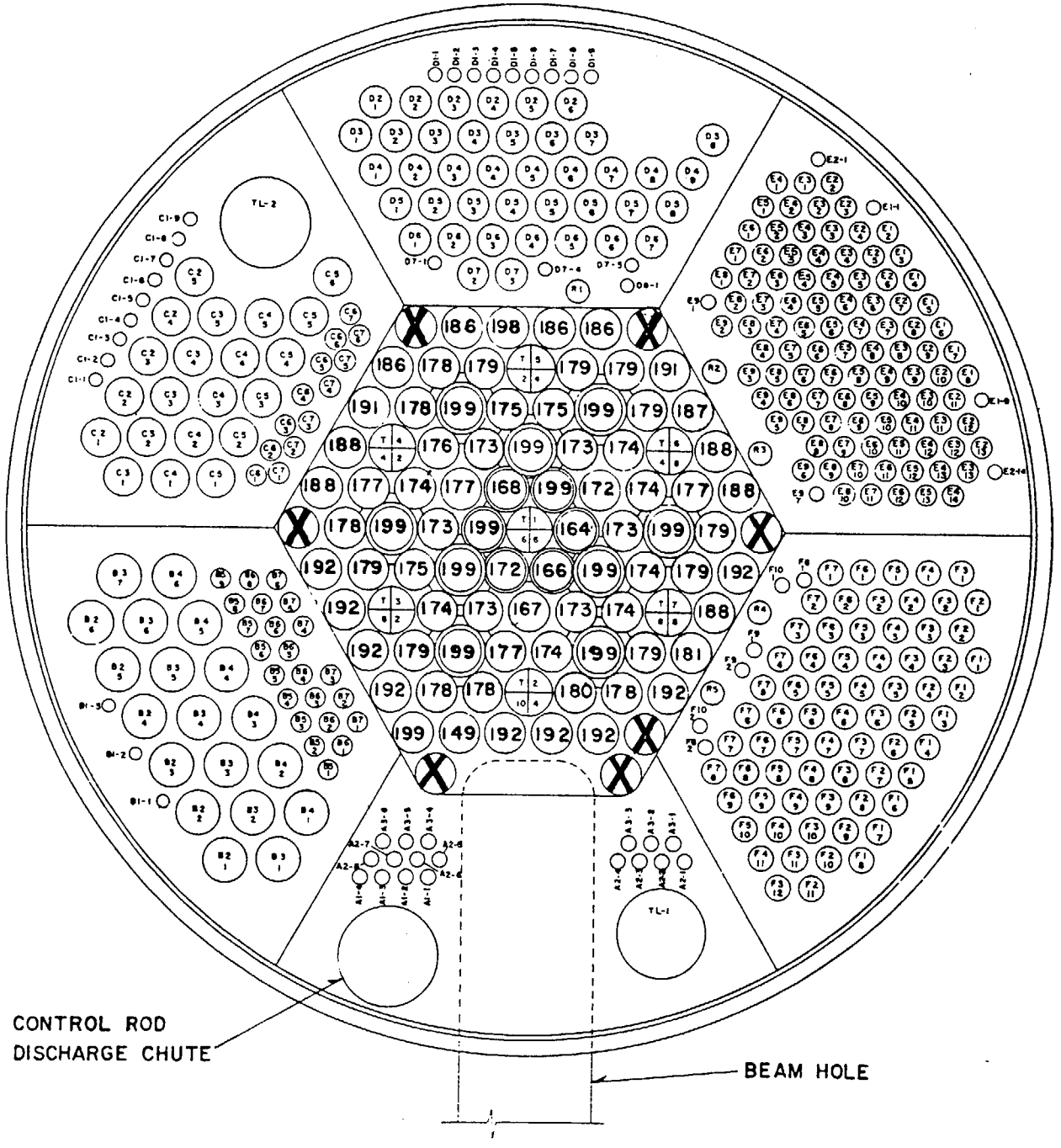


FIGURE 15

U-235 CONTENT OF FUEL ELEMENTS (GRAMS)

PLUGGED

FUEL ELEMENTS

CONTROL RODS

HIGH PRESSURE THIMBLES

A,B,C,D,E & F HOLES ARE REFLECTOR POSITIONS

R-HOLES RABBIT FACILITY

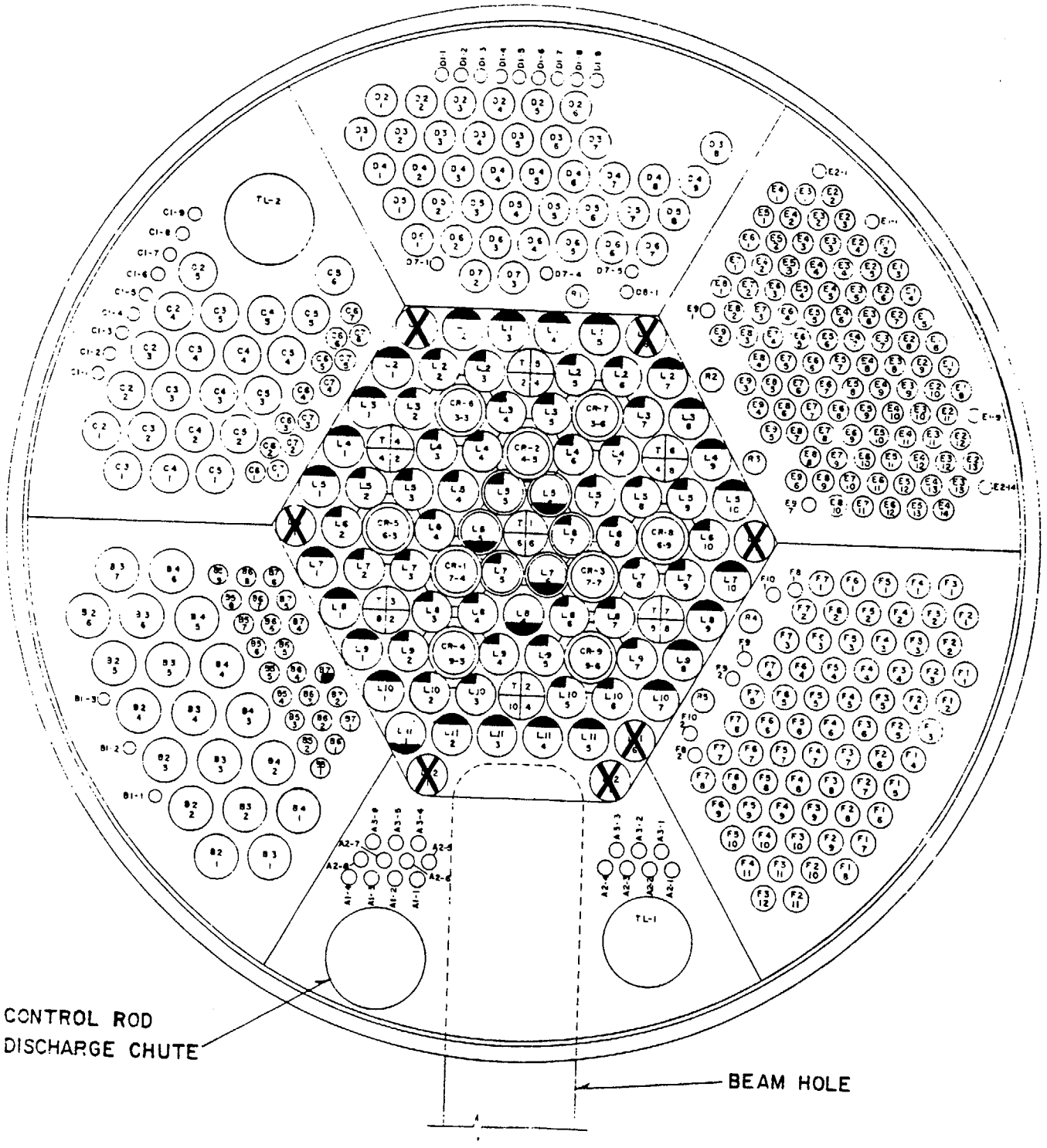


FIGURE 16

CONTENTS OF IRRADIATION VOLUMES

- SOLID ALUMINUM "V" BASKET
- STD. "V" BASKET WITH COBALT
- EXPERIMENT
- PLUGGED
- THERMOELECTRIC EXPERIMENT
- FUEL ELEMENTS
- CONTROL RODS
- HIGH PRESSURE THIMBLES

A,B,C,D,E & F HOLES ARE REFLECTOR POSITIONS

R-HOLES RABBIT FACILITY

held in a recesses in a 3/8-inch diameter aluminum rod. The rod was encased in a 1/2-inch O.D. aluminum tube with 1/8-inch weep holes drilled through the wall at 4-inch intervals. This assembly was centered in a standard irradiation V-basket which in turn was placed inside the fuel element in the normal manner. A flow orifice at the bottom of the V-basket limited the flow to that required for cooling the rod.

L-8-5 Battelle Experiment

This experiment was substituted for a standard V-basket. It consisted of an instrumented stainless steel capsule containing stainless steel samples imbedded in aluminum serving as a heat sink and heat transfer medium. The capsule was located approximately at the axial power peak. The space above the capsule was occupied by a stainless steel lead tube containing thermocouple wires. The space below the capsule was occupied by a stainless steel tube of the same diameter as the capsule and containing weep holes to eliminate dead water space. The capsule assembly was the same diameter as a standard V-basket.

L-11-1 Thermionic Experiment

This experiment consisted of an instrumented stainless steel capsule occupying a standard irradiation W-basket in a two-tube fuel element. It contained a small U-235 fueled cesium thermionic converter inside an assembly of rings of thermoelectric material. The space above the capsule was occupied by a stainless steel tube containing the electrical leads. The leads permitted both temperature and power output measurements. The space below the capsule was occupied by a stainless steel flux depressor.

B-7-3 Thermoelectric Experiment

This experiment was contained in an instrumented stainless steel capsule, located in the B reflector segment. It consisted of a gamma heated assembly of thermoelectric material. The assembly was placed in a V-basket in the referenced reflector position.

L-5-6 Fuel Element Instrumented with Stainless Clad Thermocouples

This experiment is described in Section B of this report.

L-7-6 Fuel Element Instrumented with Aluminum Clad Thermocouples

This experiment is also described in Section B.

Cobalt Filled V-Baskets

Those elements containing cobalt are shown in Figure 16. Each cobalt assembly produced a macroscopic absorption area of approximately 1.2 cm^2/inch of length.

Aluminum Mandrels

Those elements containing aluminum held a solid aluminum bar of the same diameter and external configuration as a V-basket. The positions of these elements are also shown in Figure 16.

In-Core Thimble Positions

1. The center thimble position was loaded as shown in Figure 17.
2. Five of the peripheral thimble positions were loaded as shown in Figure 18.
3. The remaining position (To-4-8) was occupied by a high pressure thimble containing control rod material coupons cooled with a boric acid solution.

F. Reactor Operation - April 2nd Run

Reactor startup began on April 2 and criticality was achieved at 7:10 A.M. The power level was gradually increased and reached 40 MW at 1:44 P.M. The following conditions were recorded in the reactor log at approximately that time:

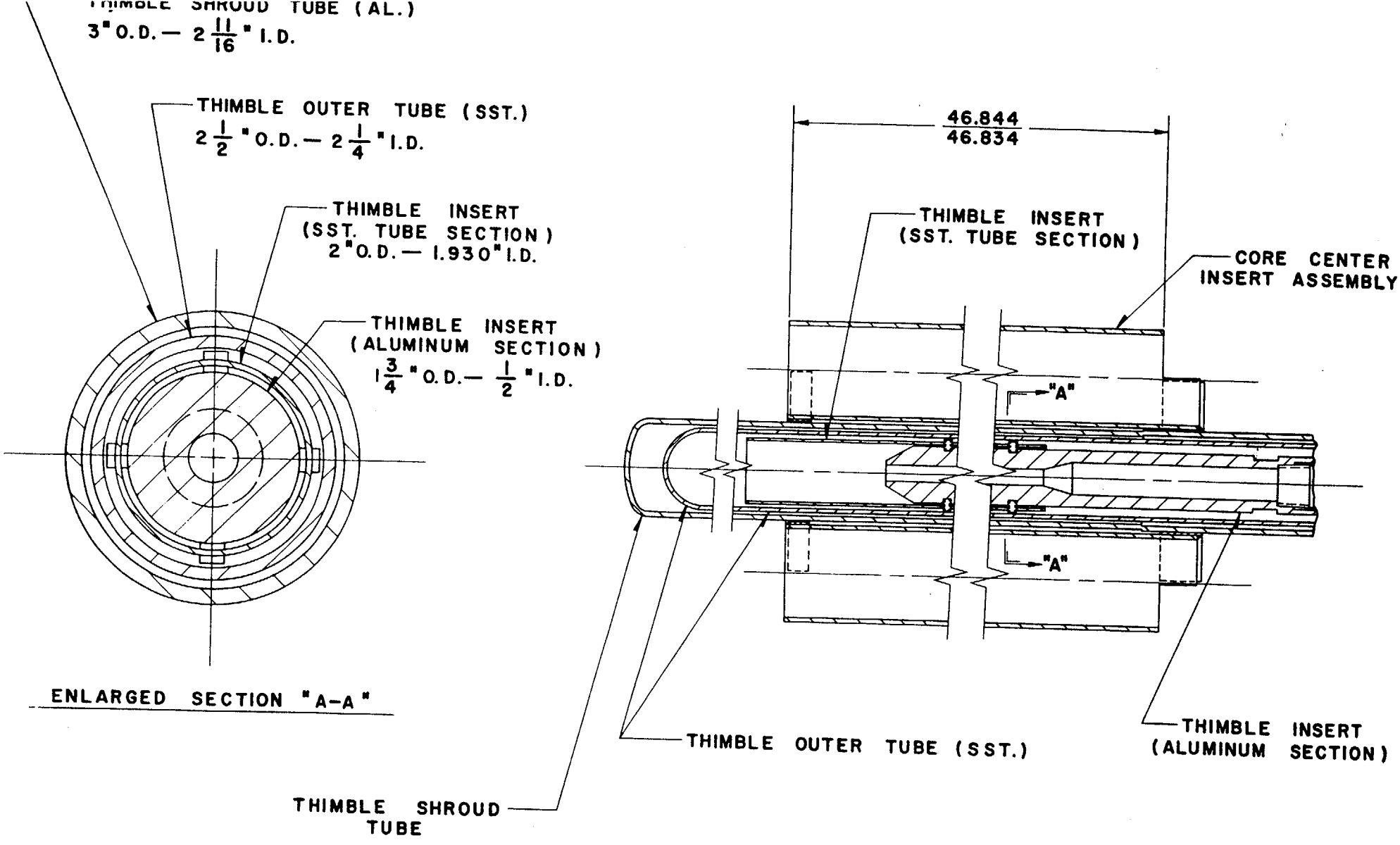
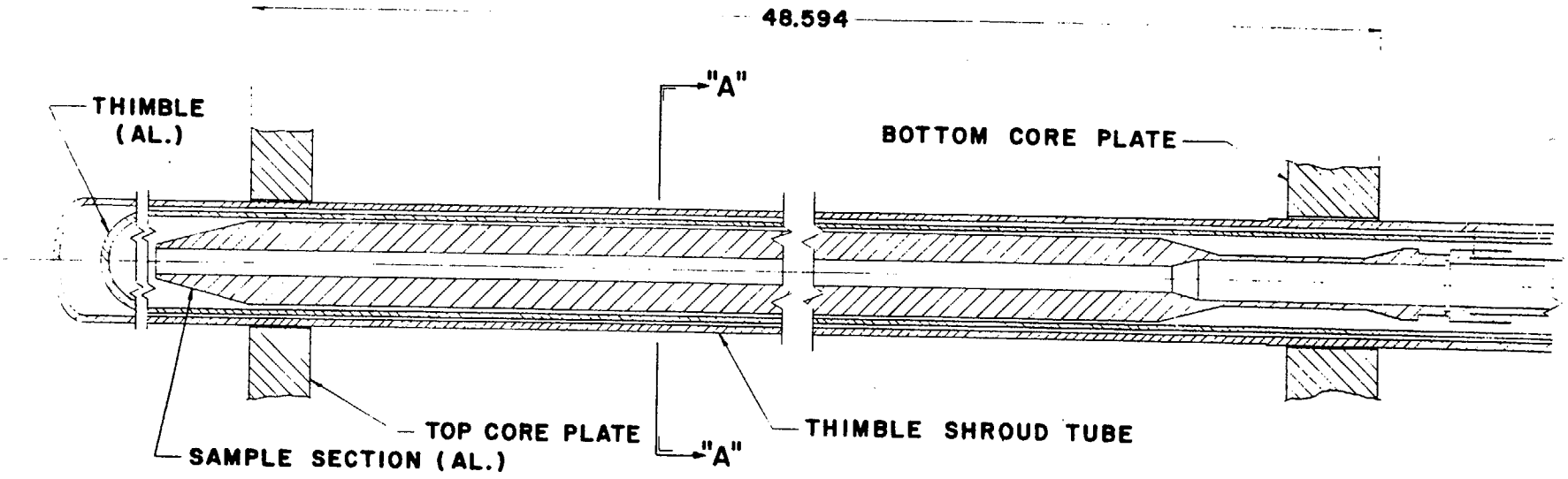
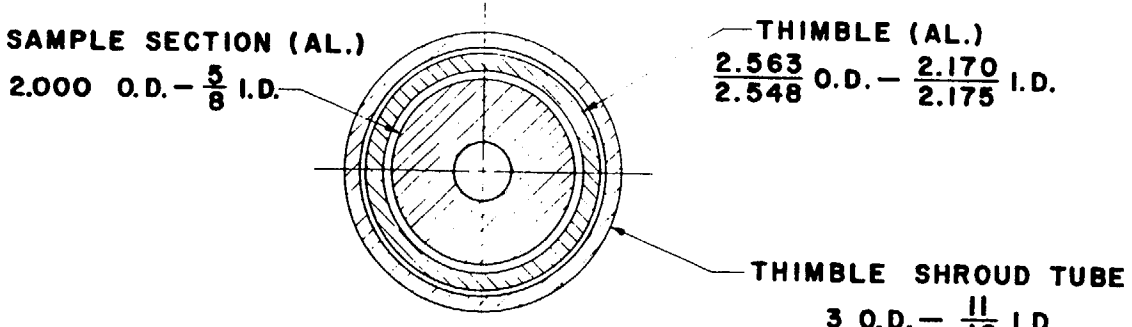


FIGURE 17
CROSS SECTION VIEW OF
CENTER TEST THIMBLE POSITION OF W.T.R.

FIGURE 18
CROSS SECTION VIEW OF OFF-CENTER TEST THIMBLE POSITIONS IN W T R (TYP.)





SECTION "A-A"

Reactor Vessel Temperatures and Pressure	T_{in} 125.2°F	T_{out} 143.8°F
	P_{in} 105 psi	P_{out} 83 psi
Primary Coolant Flow	15,000 gpm	
Reactor Power	Nuclear 40 MW - Thermal 40 MW	

Control Rod Positions:

Rod No.	#1	#2	#3	#4	#5	#6	#7	#8	#9
% Withdrawn	50	50	50	49	48	48	48	48	48

A set of data taken from the fuel element (in location L-5-6) instrumented with the stainless steel thermocouples was:

Fuel Element Bulk Coolant Discharge Temperature	--	164°F - 167°F
First Coolant Channel Discharge Temperature	--	153°F - 156°F
Second Coolant Channel Discharge Temperature	--	183°F - 175°F

The defective aluminum sheathed thermocouples in the other instrumented fuel element (in location L-7-6) were recorded as reading -- 296 - 297 - 319 - 290°F

With two-thirds of the total flow passing through the core and a radial peak to average power production of 1.33, the expected temperatures compared with the measured ones were:

	Computed	Measured
ΔT across reactor vessel	18.5°F	18°F
ΔT across position L-5-6	37 °F	36-39°F
ΔT across first channel	25 °F	25-28°F
ΔT across second channel	47 °F	47-55°F

The reactor was maintained at a power level of 40 MW except for a reduction in power at 9:15 hours, April 3, due to test loop trouble. This trouble was inconsequential and the reactor was returned to 40 MW at 10:01 hours. At approximately 19:00 hours on April 3, the reactor power was reduced to 30 MW in preparation for the test to be

conducted as outlined in T-Spec 5-2 (Appendix II). The alarms, cut-back and scram points were reset in accordance with these requirements. At about 20:00 hours primary coolant flow was gradually reduced to 5,250 gpm.

Figure 19 is a copy of the nuclear power recorder chart covering this and the following time interval. As can be observed from this chart, the reduction in flow was accompanied by a slight reduction in power caused by the temperature coefficient. This power dip was compensated for by the automatic control system. Prior and subsequent to the reduction in flow, the boiling detector record was observed and samples of these records are shown in Figure 20.

At 20:20 hours the power level was raised to approximately 35 MW and allowed to settle to approximately 34 MW as measured by the nuclear power instrumentation. Thermal power calculations were performed prior to and subsequent to raising the power level using both the reactor flow and core ΔT, and the reactor flow and the instrumented fuel element ΔT. In addition, after the expected delay, the thermal power was displayed by the thermal power recorder. These data are presented below:

Nuclear Power	Core ΔT x Flow	Thermal Power Instrumented Fuel Element ΔT x Flow	Thermal Power Recorder
30 MW	32.9 MW	29 MW	30 MW
35 MW	37.8 MW	36 MW	[illegible] MW

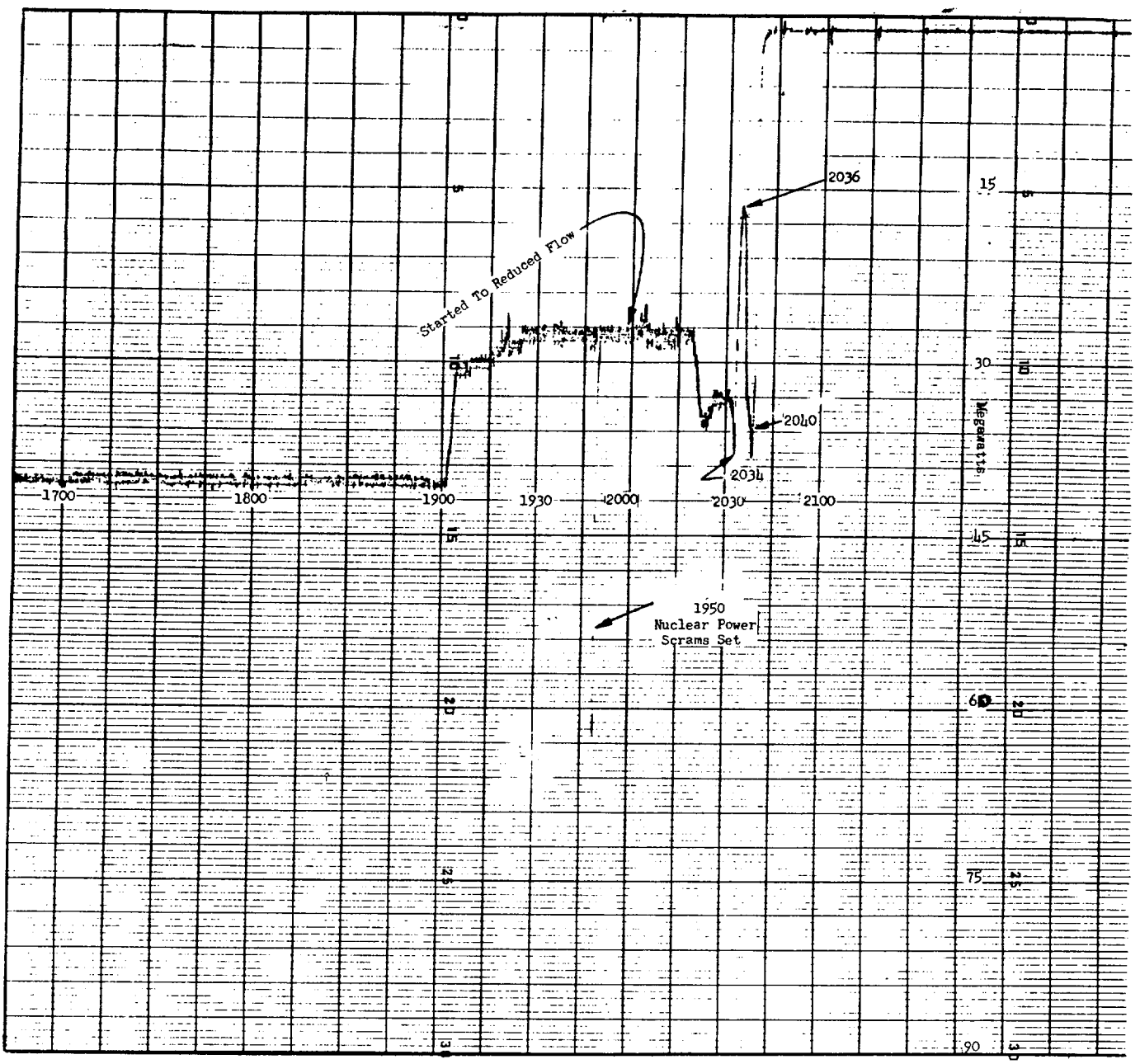
Instructions were then given to the reactor operator by the shift supervisor to increase the power level to 40 MW. To increase the power level, the automatic control system called for additional rod withdrawal. All nine control rods were banked at about 62% at this time. Control rod No. 9 which was on automatic control was, like the other rods, in a low differential worth region of its travel and shortly reached 85% withdrawn. Automatic control was then manually switched to rod No. 8 which also was withdrawn to 85% and the automatic control was manually

switched to rod No. 7. About this time (approximately 20:35 on Figure 19) it was observed that the power level was falling and the operator, under instructions of the shift supervisor manually withdrew rods No. 1, No. 2, No. 3 in turn, each 2% in travel. This movement, together with reactivity added by the automatic control system, returned the reactor to approximately 37 MW. Just before the reactor reached 37 MW, the demineralized water monitor channel alarmed. This alarm was acknowledged and almost immediately thereafter, several other alarms indicated high radiation levels in the various monitored areas. At approximately 20:40 the power demand set point was reduced followed immediately by manual reactor cutback and at 20:44 the reactor was manually scrammed. It was suspected and later confirmed that a fission break had occurred and that the accumulation of fission products in the head tank was producing radiation levels in the plant areas sufficiently high to produce alarms. The plant was evacuated and the immediate subsequent actions are described in Section H.

G. Other Observations During the Fuel Element Failure

The pertinent operational data collected during the time of the fuel element failure consists of the Neutron Level Chart, Figure 19; the thermocouple measured temperatures of the fuel element water passages recorded on the Data Logger, Figure 6, and the Brush Recorder trace from the bubble detector, Figure 21.

The power reduction shown in Figure 19 at 20:34 is believed to have occurred as a result of a decrease in reactivity caused by the fuel element failure meltdown and subsequent blockage of the coolant channel. The blockage is presumed to have voided the water channels by the production of steam and bulk boiling in the failed element. A consideration of previously measured void coefficients for an element in this position would indicate a loss of 0.3 to 0.6% reactivity if all channels were voided. This reactivity loss



NEUTRON POWER LEVEL RECORDER CHART

NIGHT OF APRIL 3, 1960

FIGURE 19

cannot be explained by simple bulk boiling with open channels at the top and bottom because the reactivity would then have been reinserted when boiling ceased at the reduced power level at 20:36. In addition the overall reactor temperature coefficient must have added approximately 0.18% reactivity as soon as the power level was reduced to 17 MW.

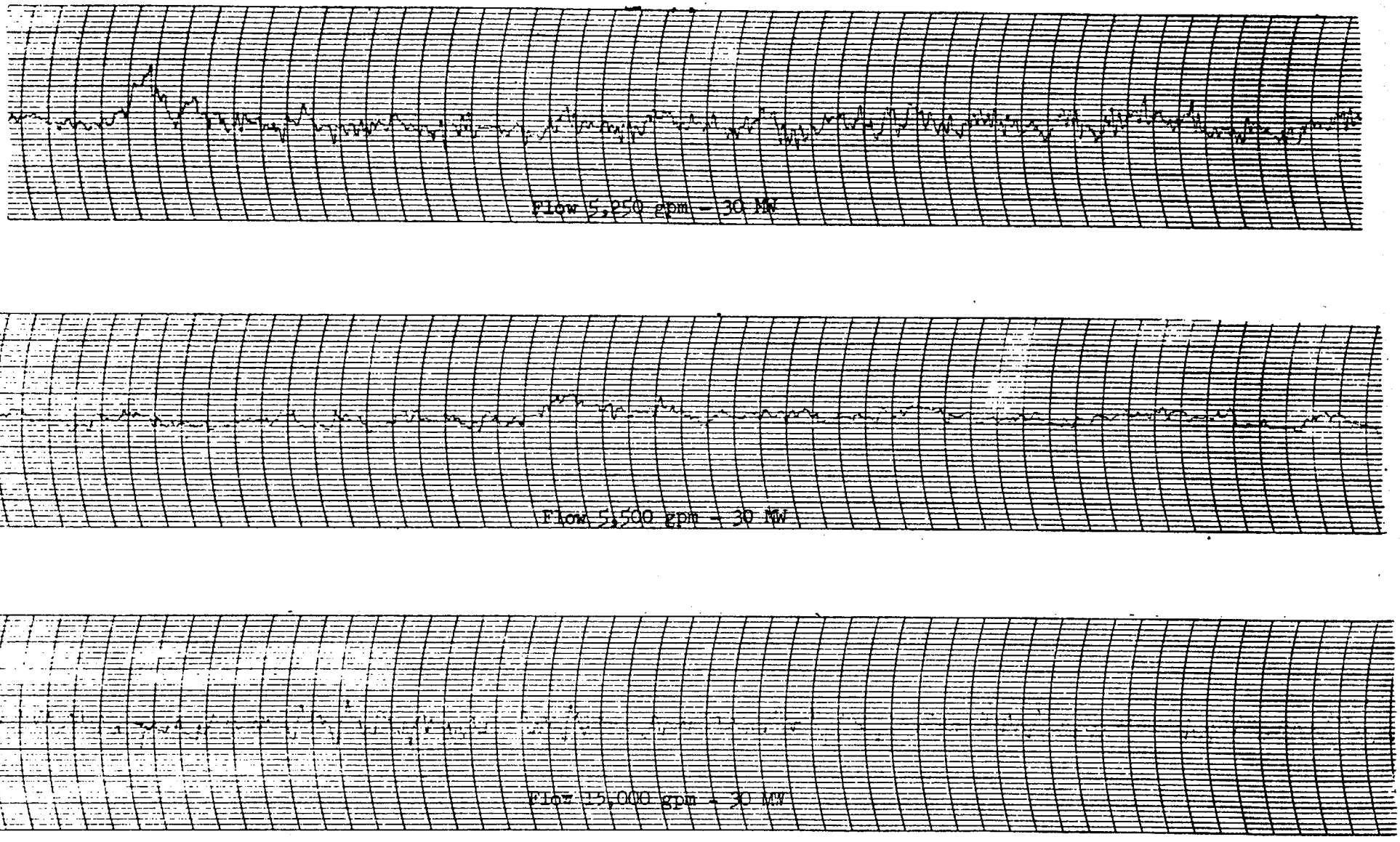
Another possible source of permanent reactivity loss is displacement of the fuel contained in the failed section of the element. An upper bound on this effect is that the worth of a fuel element in the failed position is 0.9%. This number obviously is a gross overestimate of what could have happened since the fuel element was not completely displaced from its core position. Later observation of the failed fuel element indicated a considerable amount of burnt up debris was lost from the element. That portion of the element which was severely damaged was worth about 0.6 - 0.7% in reactivity but only a small portion of this worth was lost. The total reactivity added by means of the control rod withdrawals previously described is approximately 0.6% Δk. Thus, the reactivity changes caused by the voiding theory and possibly by the loss of a small amount of fuel are consistent.

It is believed that the element failed at approximately the same time as it voided and later examination confirmed the melting and permanently blocking of the channel. The element remained voided after the reduction in power because of a lack of water about the blocked portion and because of the presence of steam above the blockage.

Figure 6 is the reproduction of the Data Logger Chart giving thermocouple temperatures. The temperatures of the stainless steel thermocouples are presented in the columns headed #1 to #6. On the basis of these thermocouple readings the reactor power was 28 MW at 20:18; 35 MW at 20:22; 36 MW at 20:25; 35 MW at 20:29:, 38 MW at 20:34; 19 MW at 20:36, and 38 MW at 20:38. The chart is formed by a typewriter traverse taking about two min./line including reset. The readings at

20:34 appear to be different possibly because of typewriter delay during the fast power reduction. The fact that the calculated power from the thermocouple data checks closely with overall power measurements confirms that this element, in a symmetrical position to the burnt out one, received the anticipated coolant flow.

Figure 21 is a reproduction of the boiling detector Brush Recorder trace during the time interval under discussion. If time zero is taken at 20:34 corresponding to the first peak in power level of Figure 19, then the reduction in power level caused by the failure and voiding appears to have occurred in about eleven seconds. That a permanent block was established can be seen in that over the next three minutes the noise level remained approximately constant. Then as rod motion forced the power level back up, the boiling noise pattern increased in amplitude, but didn't quite return to the before failure amplitude when the cutback was initiated. An independent observation indicated that an alarm on top of the reactor, actuated by the radiation from the head tank went off at approximately 240 seconds on Figure 21. This further confirms the other deduction that the element failed at about 20:34 at the first power peak rather than at 20:40 the second peak in power.



NUCLEAR NOISE LEVEL AT 30 MW DURING FLOW REDUCTION
FROM 15,000 GPM TO 5,250 GPM

FIGURE 20

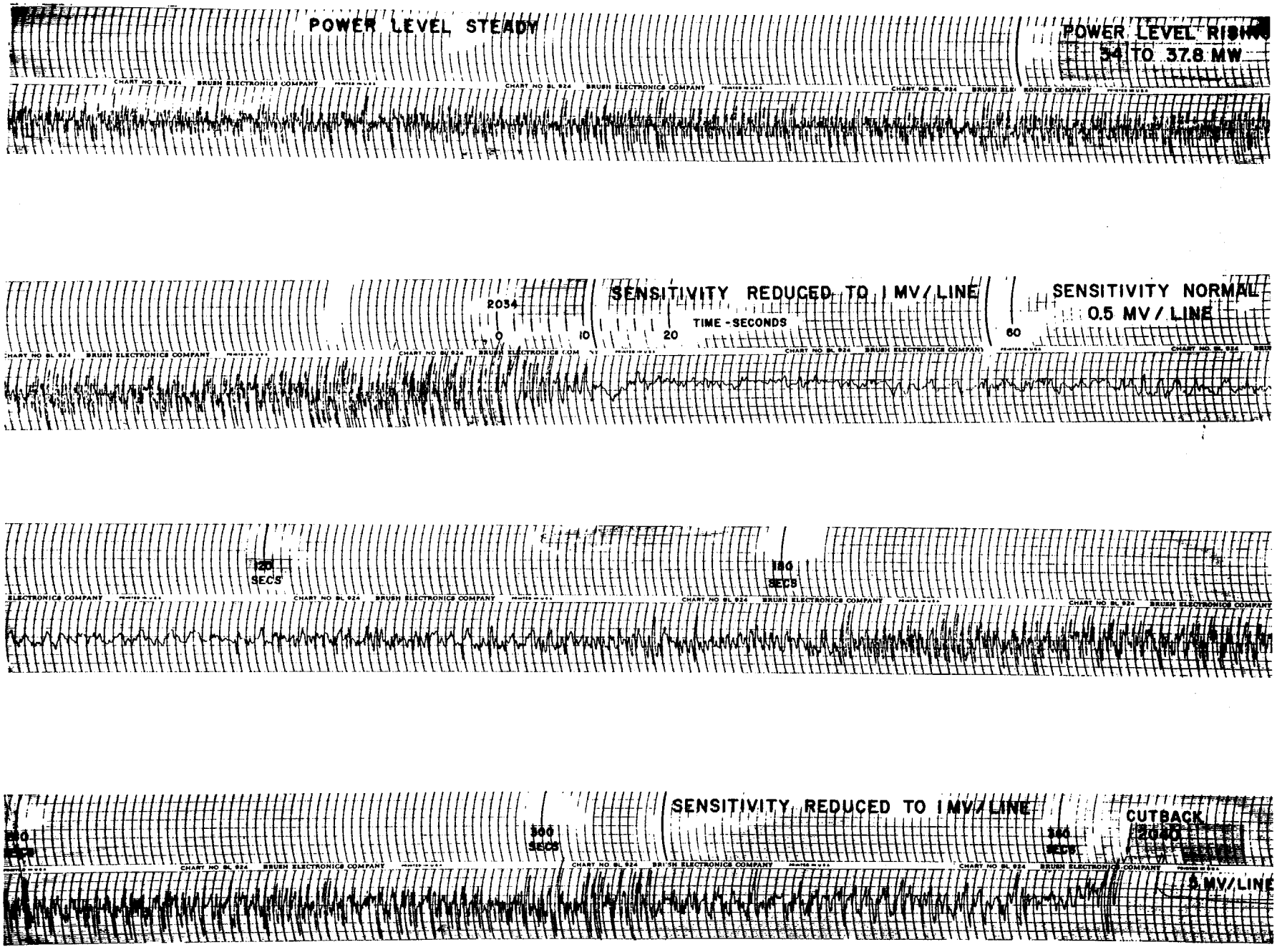


FIG. 21

BOILING DETECTOR RECORD OF APRIL 3, 1960 DURING FUEL ELEMENT FAILURE

H. Events Immediately Following Incident

The following paragraphs are quoted from WTR-TO-R752 reporting the incident to the Atomic Energy Commission, the next morning:

"Immediately following scram, request for evacuation of the reactor top was initiated on the Femco system. As all radiation monitoring instruments continued rising, the signal for general evacuation was sounded. Operations and Health Physics personnel remained a short time to secure plant and continue survey but were also ordered to leave the plant when levels continued rising rapidly. One Health Physics person remainded on continuous duty using self-reading dosimeters to limit his exposure. The assembly point was the guardhouse at the entrance to the WTR property but was changed to Seubert House, approximately one-third of a mile southeast, as radiation levels continued to rise.

The primary coolant system was left in operation and high pressure loop No. 1 was placed on cool down; the reactor shell ventilation system switched to recirculate when activated by stack and reactor monitors for gas and particulate material. The surge tank vent blower was left running to prevent possible blowback of fission material into the process area and was turned off at sometime between 9:00 and 9:15 p.m. At that time the primary coolant system was also placed on shutdown flow."

An outline of the major activities in the plant on a day by day basis for the next eight weeks is presented as Appendix III. It will be recognized that a large number of side issues had to be dealt with in order to pursue the main line efforts of determining the cause of the failure, getting the plant decontaminated, and the reactor back into operation. Problems such as water storage and radiation protection occupied a considerable effort and the solution to these type problems governed the pace of the main activities. Some of these problems will be described in detail in Section L of this report and in the Appendices.

In an attempt to determine the cause and possible effects of the incident the next several days work was directed towards reducing the activity in the primary loop sufficiently to be able to remove the head of the pressure vessel and examine the core. The principal method used was that of ion exchange. The main primary loop bypass demineralizer

was without resin at the time of the incident. Resin was obtained and circulation started. The flow in the main loop originally was the shutdown flow of a thousand gallons per minute and this flow was soon increased to 4,000 gpm to obtain degassing. A program of water sample analysis was initiated and revealed initial activity levels of 3 to 5 μc/ml of which approximately half of this activity appeared to be caused by dissolved Xenon 133. This dissolved Xenon was purged in the recirculation process by degassing through the surge tank to the head tank vent. The purge blower was operated intermittently and activity release values were set at the maximum permissible concentration for Xenon 133 as measured by the head tank monitor. Advantage of the release point height of 250 feet for prevailing wind conditions was taken using Sutton's equation.

In addition to recirculation, water was exited from the primary loop through home-made barrel demineralizers. The demineralizers, consisting of a 6 inch pipe filled with resin, were shielded in 55 gallon drums surrounded by ilmenite concrete. Approximately 100 of these ion exchangers were made up and their usage indicated in Figure 22. The discharge from these ion exchangers was passed through the bubble cap tower of the waste disposal system evaporator to permit further degassing. This water was then discharged to the main retention basin at an activity of approximately 10^{-2} to 10^{-3} μc/ml of mixed fission products. A small amount of new clean water was added to the reactor. Table I indicates the water activity measurements during the first few days of this combined treatment. A substantial reduction in activity was made in excess of the early radioactive decay.

On April 9 the reactor head was raised one foot for examination and radiation survey. The following radiation levels were observed: 1 r/hr gamma at 6 inches, 3-5 rem/hr beta at 6 inches, 200 mrem beta-gamma at four feet. The head was replaced pending construction of beta shields and to prepare washing and decontamination equipment. Curved bus windowshields were used as beta shields, and a system of car-wash

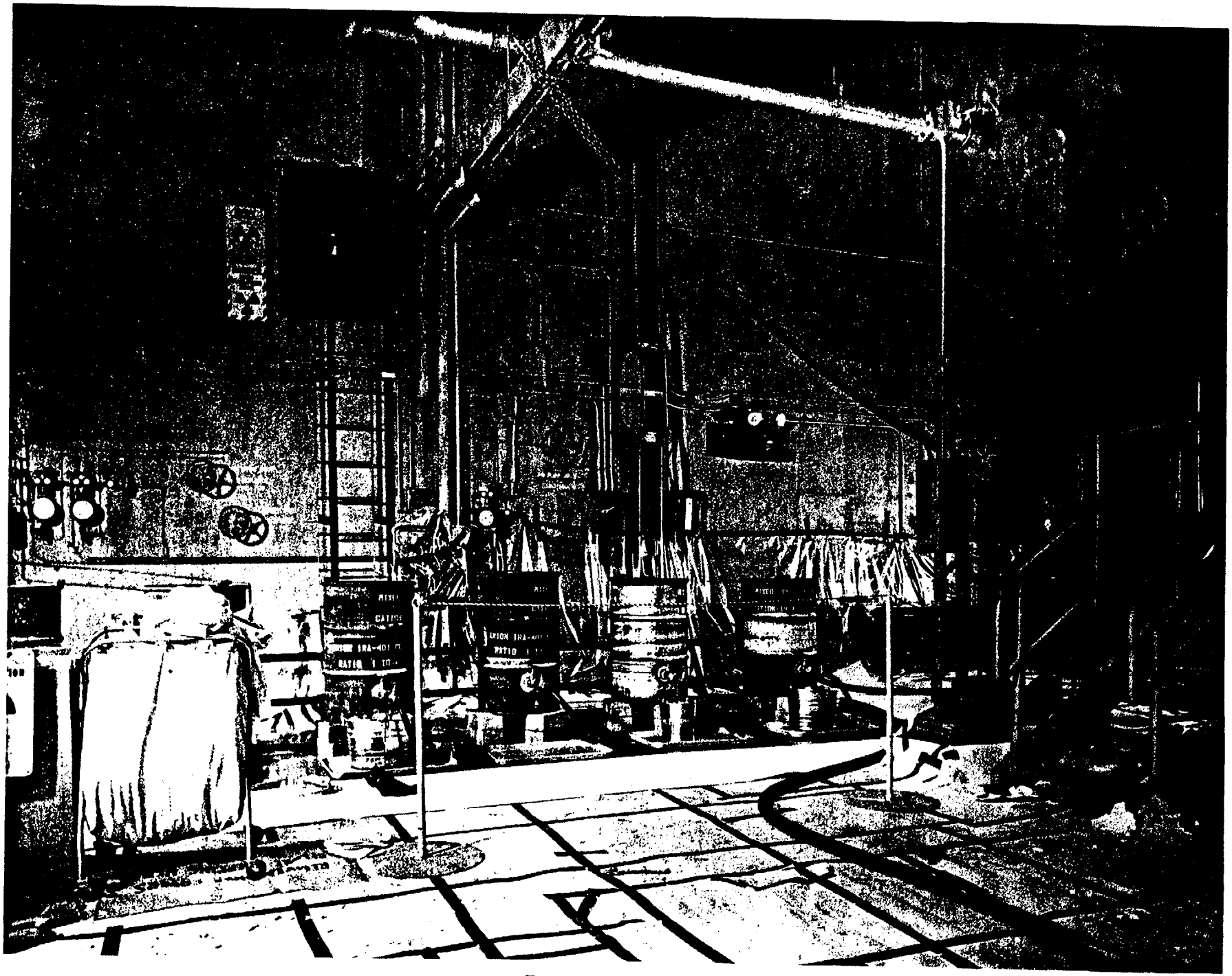


FIGURE 22

BARREL DEMINERALIZERS IN OPERATING AREA

TABLE I

Water Activity First Few Days Following Fuel Element Rupture

		Counts/5 sec/milliter
April 4, 1960		
Fission product detector	9:30 a.m.	557,898
Head Tank	9:30 a.m.	681,406
Head Tank	12:00 noon	663,756
April 5, 1960		
P.C. Ion Exchange inlet	2:00 p.m.	650,216
P.C. Ion Exchange outlet	2:00 p.m.	25,626
P.C. Ion Exchange inlet	9:15 p.m.	575,213
April 6, 1960		
P.C. Ion Exchange inlet	10:00 a.m.	659,800
P.C. Ion Exchange outlet	10:00 a.m.	78,000
P.C. Ion Exchange inlet	8:00 p.m.	545,000
P.C. Ion Exchange outlet	8:00 p.m.	83,000
April 7, 1960		
P.C. Ion Exchange inlet	9:30 a.m.	258,000
P.C. Ion Exchange outlet	9:30 a.m.	5,500
Retention tank		900
P.C. Ion Exchange inlet	12:00 noon	213,000
P.C. Ion Exchange outlet	12:00 noon	25,000
P.C. Ion Exchange inlet	3:45 p.m.	208,000
P.C. Ion Exchange outlet	3:45 p.m.	4,500
Retention tank	3:45 p.m.	1,000
P.C. Ion Exchange inlet	8:15 p.m.	108,000
P.C. Ion Exchange outlet	8:15 p.m.	700
Retention tank	8:15 p.m.	200
April 8, 1960		
P.C. Ion Exchange inlet	0030 a.m.	113,000
P.C. Ion Exchange outlet	0030 a.m.	2,300
Retention tank	0030 a.m.	1,000
P.C. Ion Exchange inlet	4:30 a.m.	106,000
P.C. Ion Exchange outlet	4:30 a.m.	1,500
Retention tank	4:30 a.m.	1,300
Reactor vessel - 0' depth sample	6:30 p.m.	50,200
Reactor vessel - 10' depth sample	6:30 p.m.	70,106
April 9, 1960		
P.C. Ion Exchange inlet	10:00 a.m.	84,600
P.C. Ion Exchange outlet	10:00 a.m.	3,450

brushes hooked up for continuous scrubbing during the raising of the head. On April 11 the head was removed and decontaminated on external surfaces as shown in Figure 23. General radiation level in the vicinity of the reactor head was 1 rem/hr beta-gamma at one foot.

In order to permit visual observation of the core and to begin unloading the core, a 3 inch thick iron shielding platform was constructed. This reduced the radiation level to 35 mr/hr gamma at the working level. Figure 23a shows a photograph of the core taken on April 11. No visible damage was apparent at the time.

Fuel unloading then began with elements being removed first from the outside of the core, working towards the middle. Some elements stuck slightly and were removed by a hoist with a 350 lb. removal force limitation on a hydraulic scale. Following removal of all fuel elements but one, which could not easily be dislodged within the above force limitation, all the control rods and their fuel element followers were removed.

Upon examination, all fuel elements thus removed from the core appeared discolored but without apparent physical damage. The stuck element was finally removed by a 500 lb. force and only the upper third of the element came loose. This section, after it was removed from the core in a special cask is shown in Figure 24. Several futile attempts were made to push up the lower portion of the failed element from the core. The bottom end of the shroud tube appeared to be solidly plugged and through the water had a cast-iron-like appearance. Finally a core drill type hole saw was fabricated to remove this section from the element shroud. The drilling operation began on April 21 and terminated on April 25 when the final portion of the damaged element was removed in the core drill. Figure 25 shows the remains of this portion of the element after it was extracted from the drill. In this figure some of the center section has also been removed. This drilling operation greatly increased the vessel water and airborne activity levels. Visual examination of the shroud holes and a later check with a sizing tool indicated that the core structure had not been damaged.

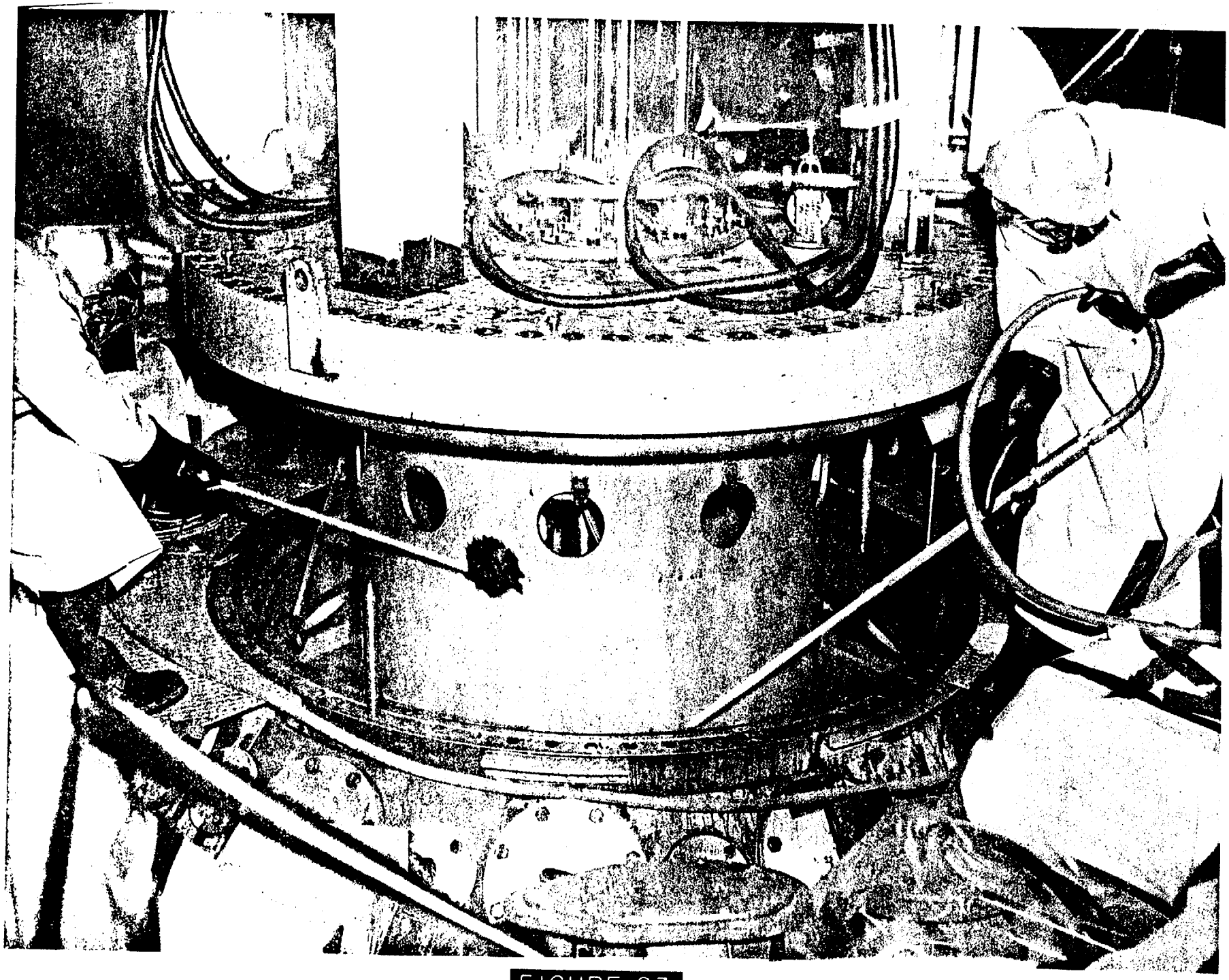


FIGURE 23
HEAD REMOVAL AND DECONTAMINATION

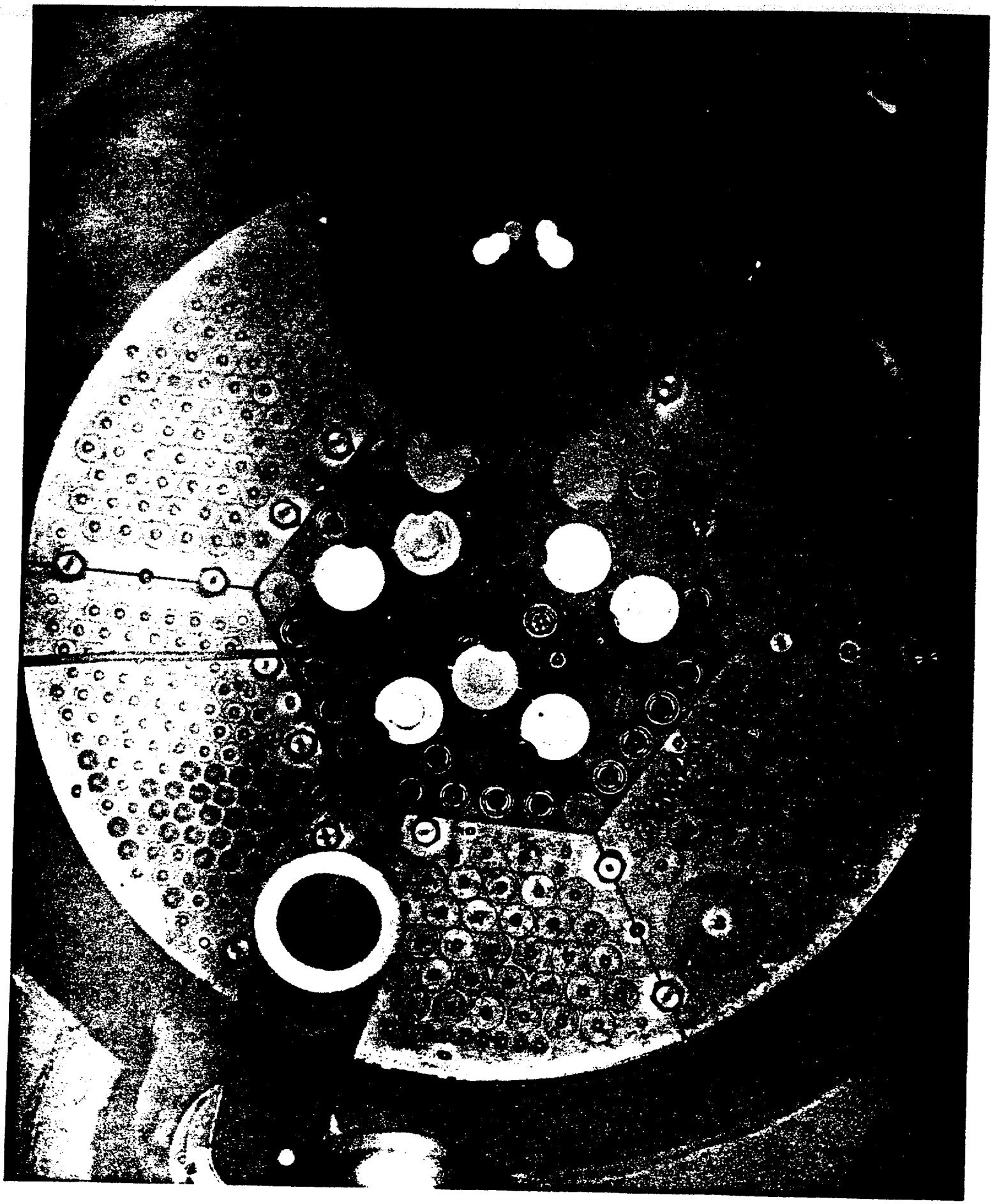


Figure 23a

View of Core - April 11, 1960

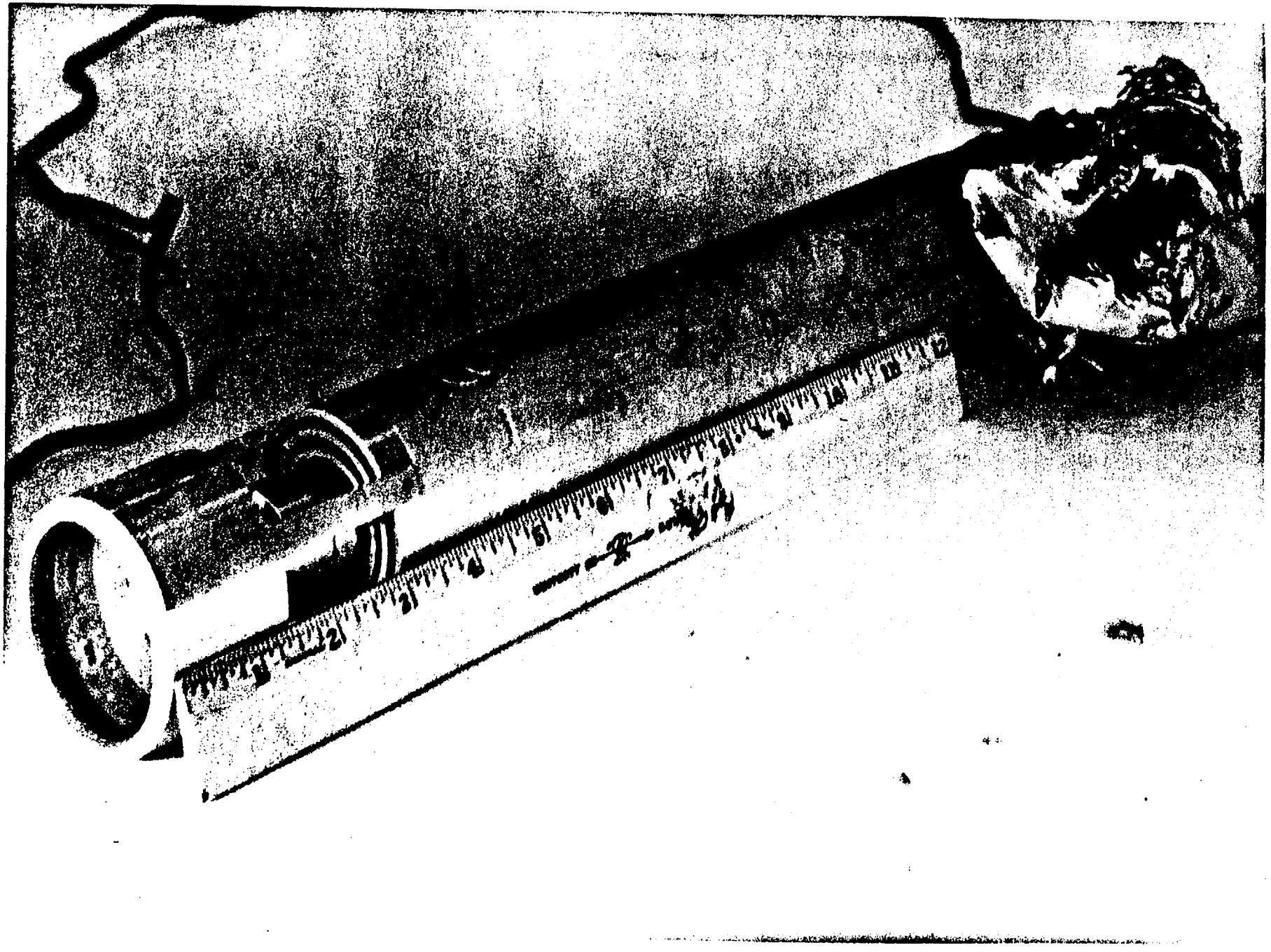


FIGURE 24
VIEW OF UPPER SECTION OF FAILED FUEL ELEMENT

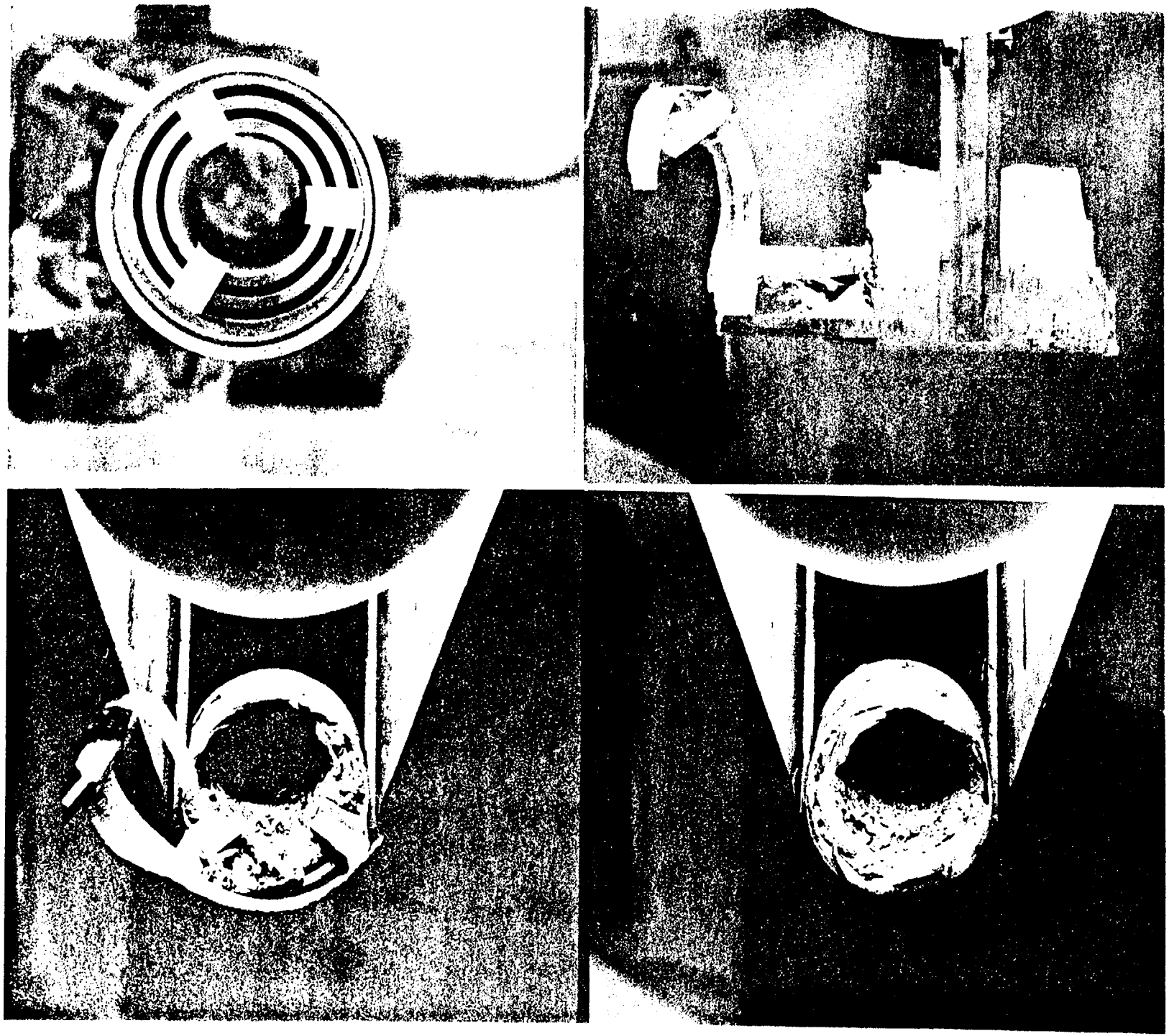


FIGURE 25
VIEW OF LOWER SECTION OF FAILED FUEL ELEMENT
WHEN EXTRACTED FROM HOLE SAW

I. Visual Observations of Failed Fuel Element

Both the upper and lower portion of the failed element were removed to the hot cells. Photographs were taken through the cell windows and through a stereomicroscope. Some of the photographs are shown in Figures 26 to 32. The appearance of the fragments is indicative of the violence and suddenness with which the melting took place. Some evidence of poor bonding between the aluminum cladding and the meat is suggested in many of the photographs, particularly in Figure 31.

Figure 33 is the "V" basket lock originally located at the upper end of the fuel element. Deposits of molten aluminum can be seen on the lower end of the lock. As the element was disassembled it was found that the innermost fuel tube was melted higher up than were the outer two tubes. Figure 34 shows an exploded view sketch of the three tubes.

A series of cuts were taken on these tubes. A rather surprising fact was observed that the alloy had run out between the cladding of the center tube all the way to the top portion of the meat. Figure 35 shows photographs of a section taken across this tube approximately 5-1/2 inches from its end. (The top 3 inches are dead aluminum). Although this cut was taken with a hacksaw and is as yet unpolished, it is quite evident that most of the alloy has run out. In extermal appearance the section seemed sound. Figure 35a is an enlargement of the lower right hand photo of Fig. 35. The small holes observed suggest inhomogeneities in the fuel or poor bonding. The other two tubes when sectioned in the same place were solid. The pattern of the melted run out in the one tube is curious and is presented in Figure 36. The funnel shaped appearance seems to indicate that the tube was hotter at the top than at the bottom, a phenomenon which might have been caused by poor bonding at the element top.

FIGURE 27

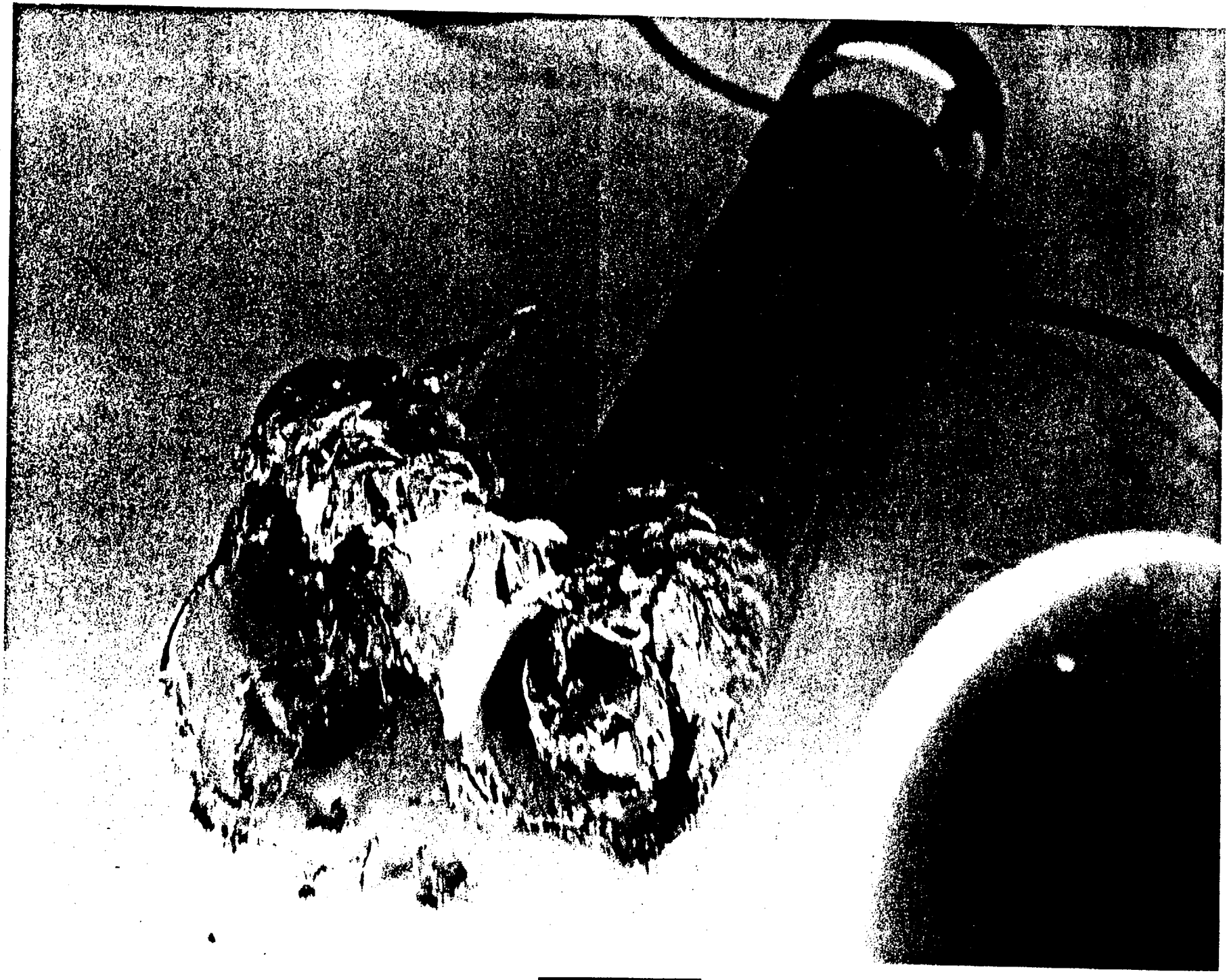


FIGURE 28

FIGURE 29

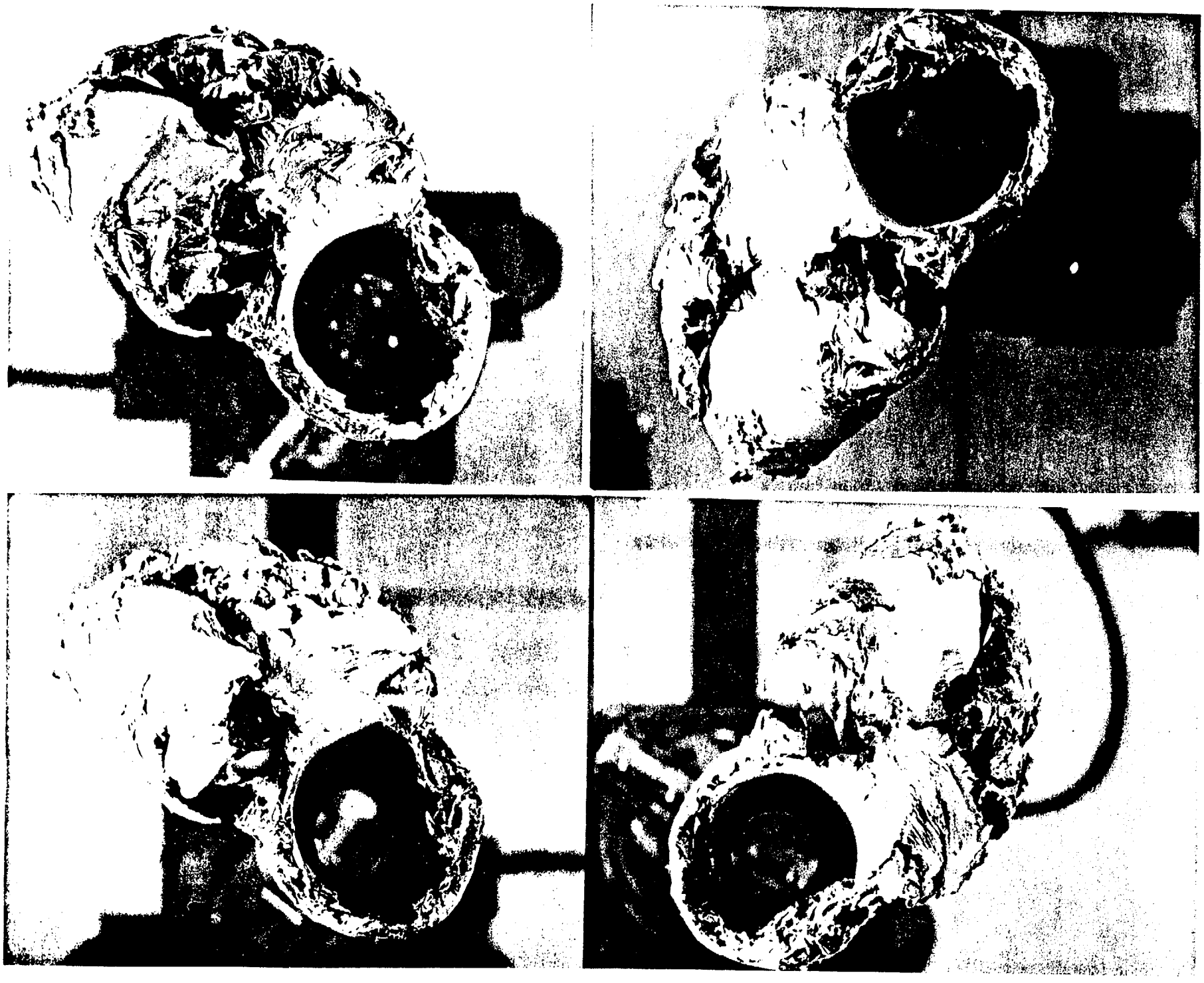


FIGURE 30

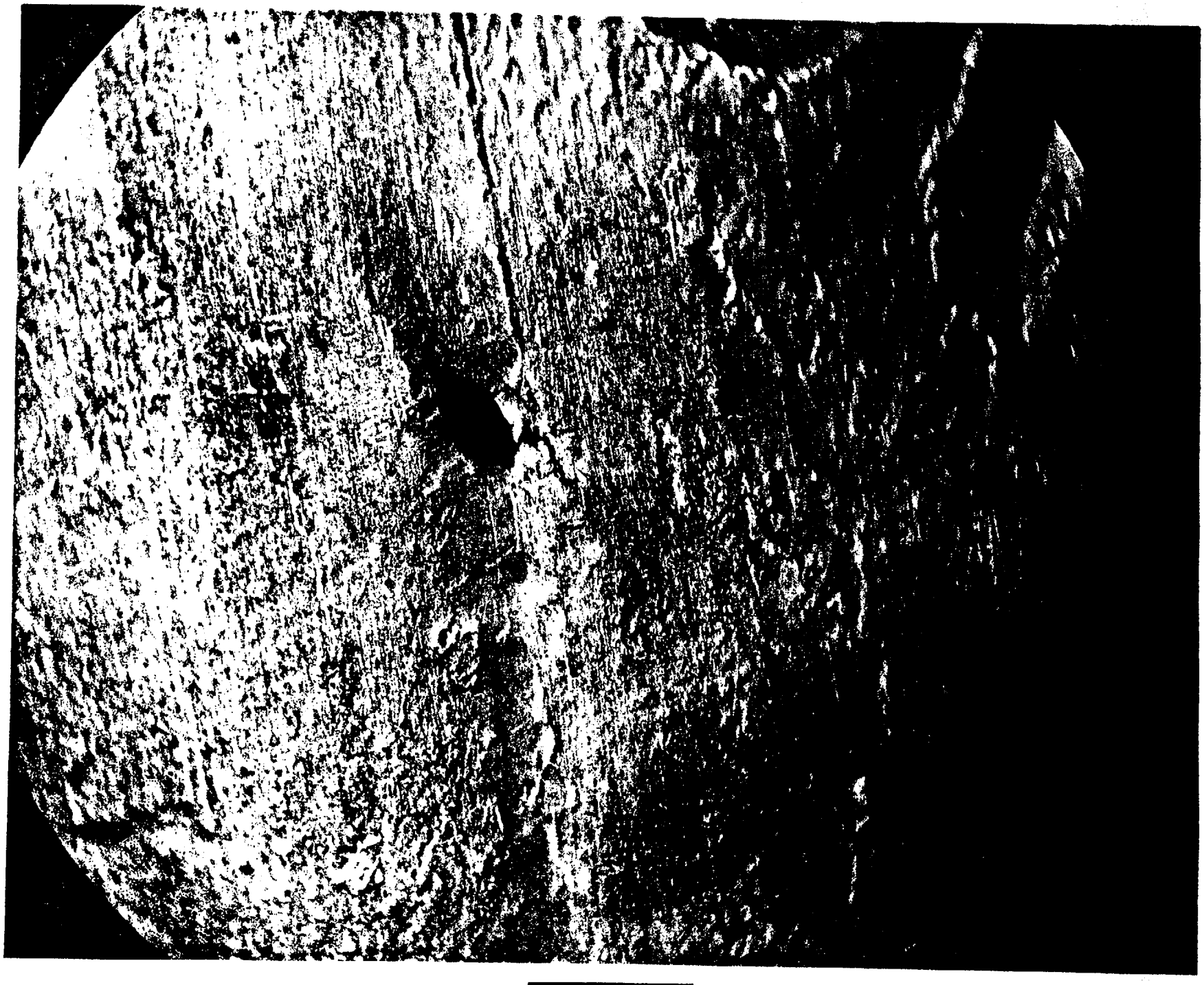


FIGURE 31

FIGURE 32

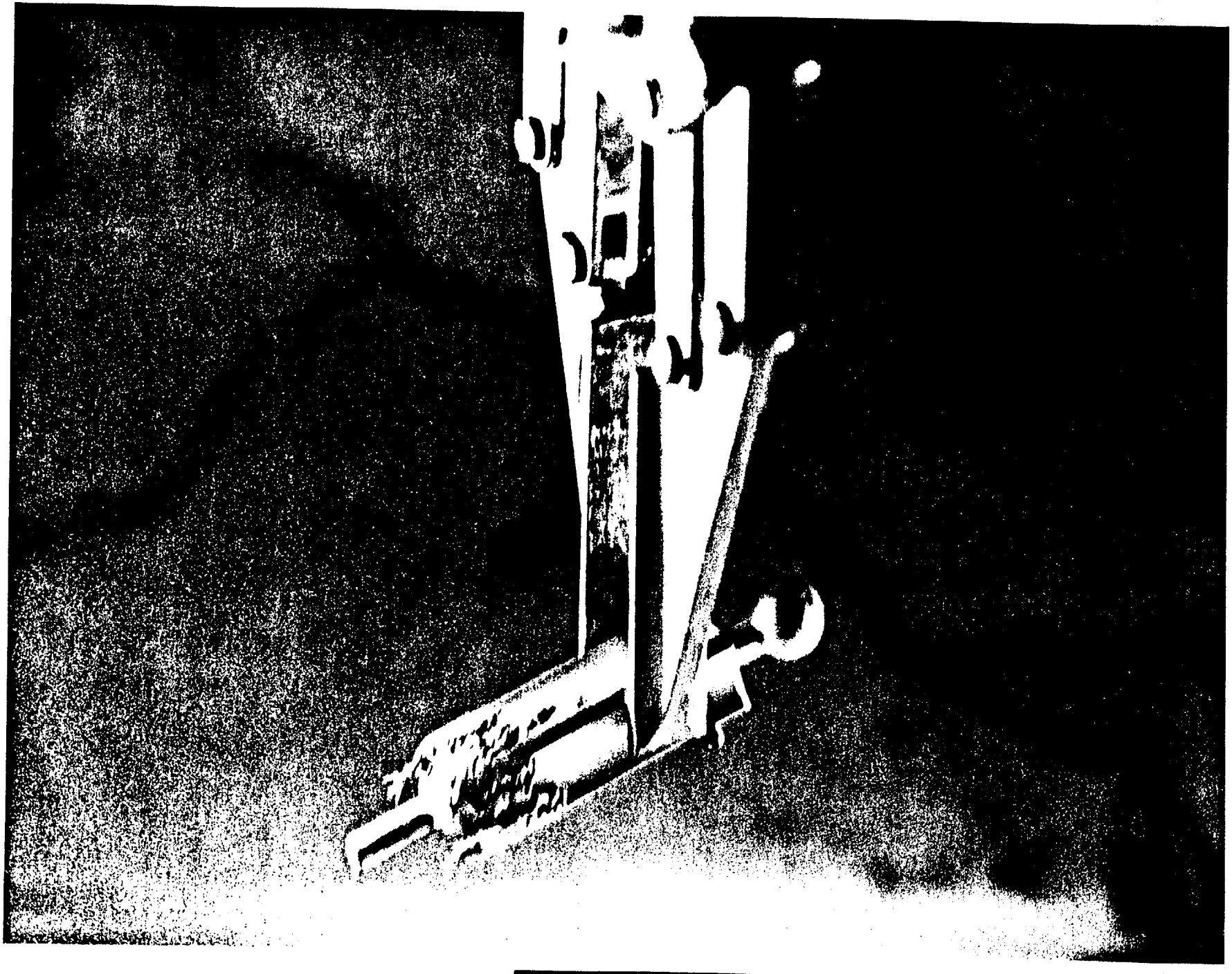


FIGURE 33
V-BASKET LOCK

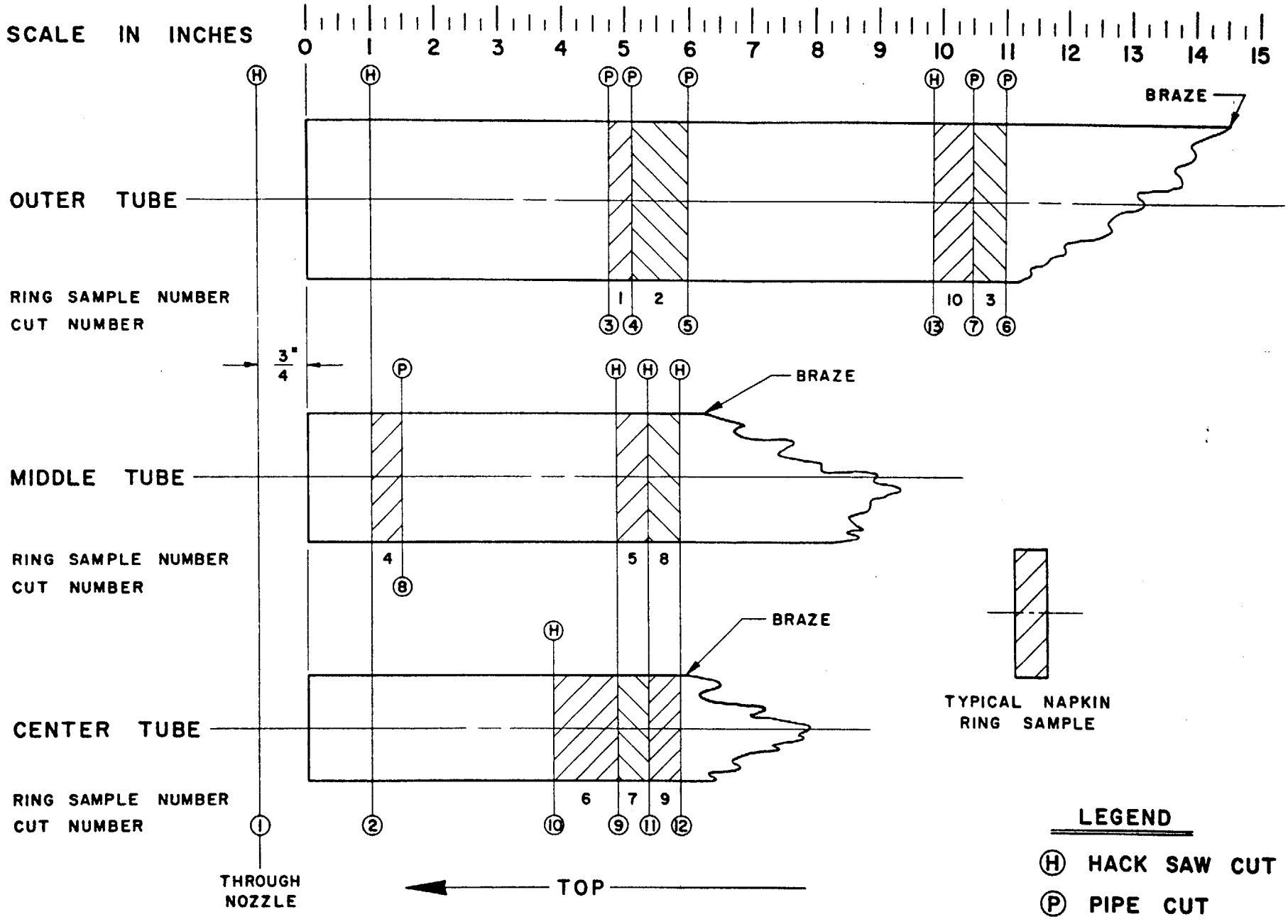


FIGURE 34

VIEW OF FAILED ELEMENT SHOWING SECTIONING

FIGURE 35
FOUR VIEWS OF SECTION 5 ½" FROM END

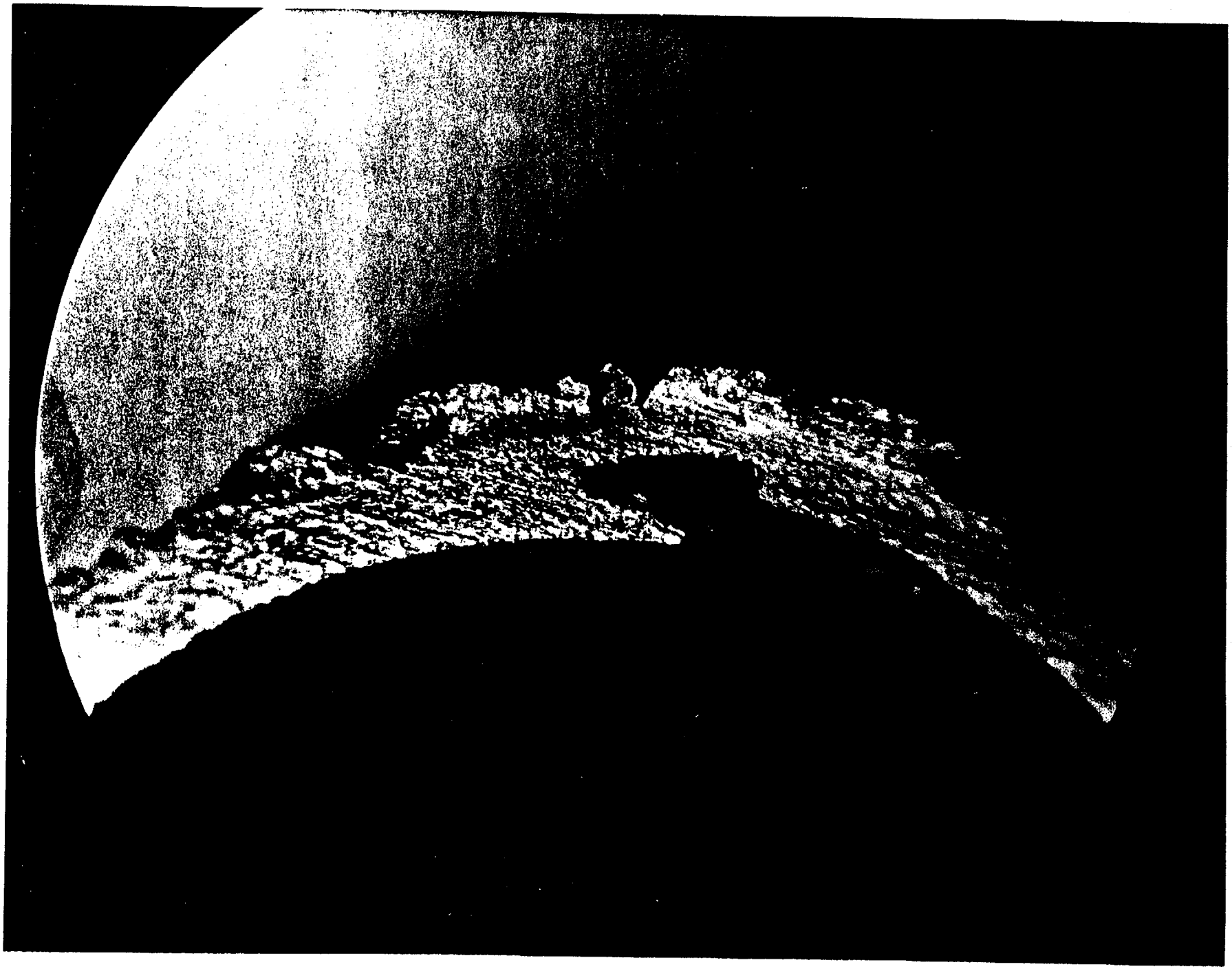
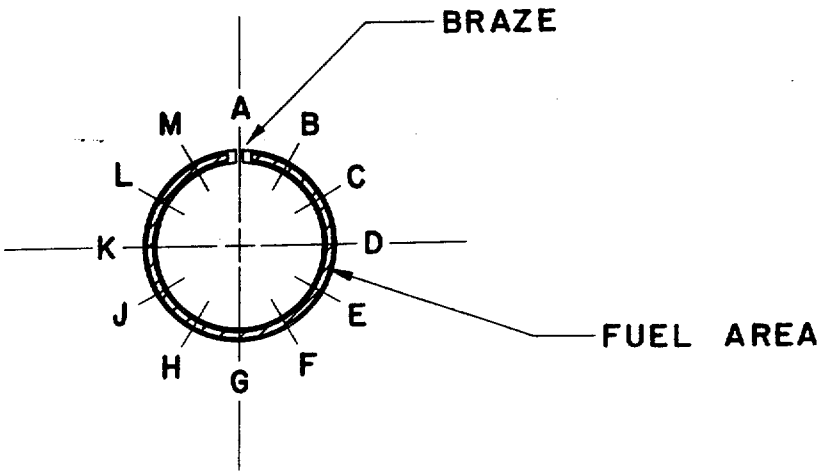


FIGURE 35A
ENLARGED VIEW OF DEFECT PHOTO



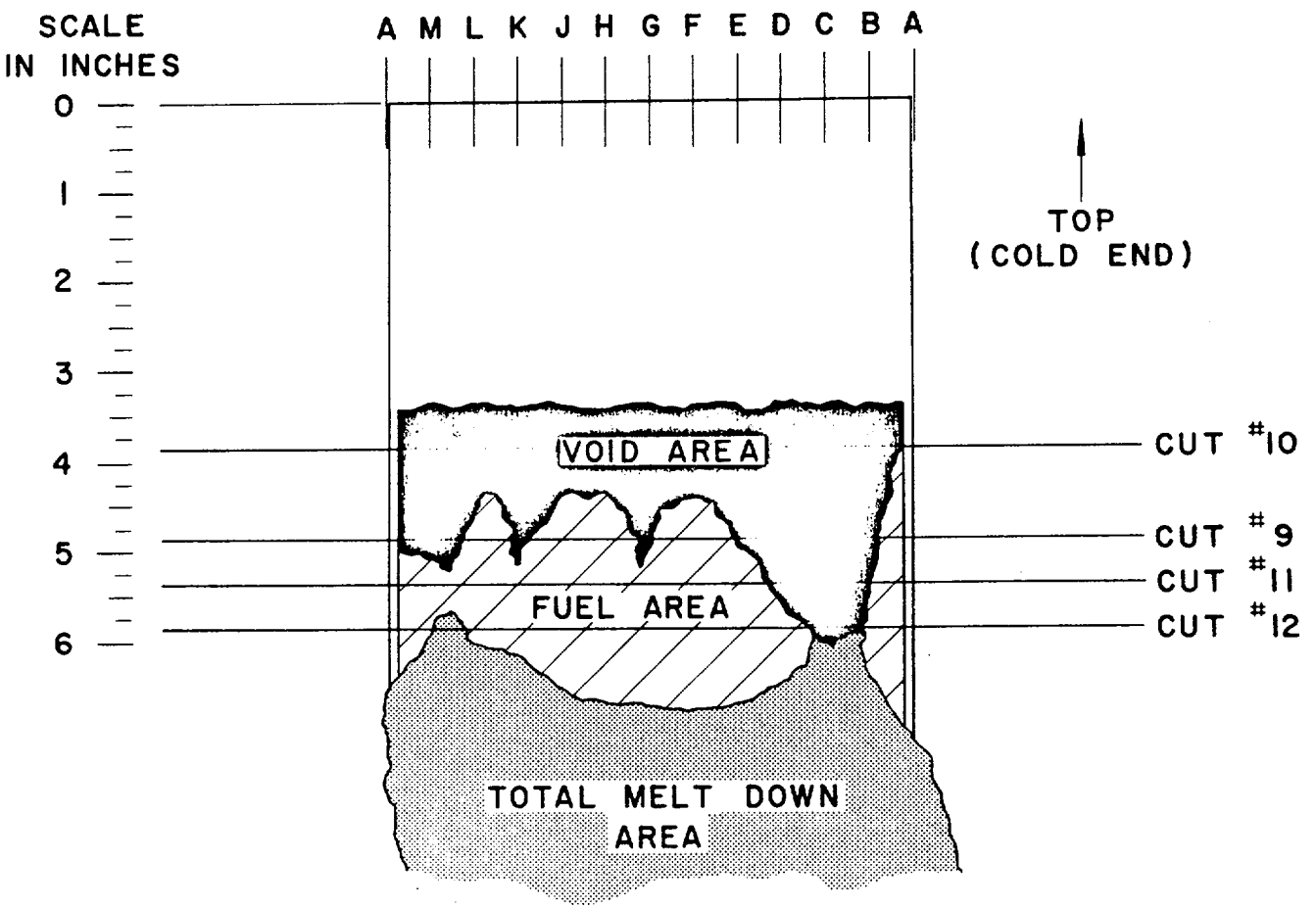


FIGURE 36

PATTERN OF MELTED RUNOUT (CENTER TUBE)

It is planned to continue post-mortem metallurgical and chemical analysis of the failed element as facilities become available. A complete program outline has been proposed with the assistance of ORNL and is presented here as Appendix IV.

J. New Fuel Element Examinations

Because of the appearance of the damaged fuel and the results of analyses presented in Section K, a program was instituted to reinspect the unused cold fuel elements on hand. Approximately 100 cold elements were available from the batch of the ruptured fuel element, and 80 more elements from another source were available from the critical experiment. Of these latter elements only two had sufficiently small residual activity to permit full examination. The instructions to the Westinghouse Atomic Fuel Department in Cheswick, Pennsylvania, who performed the reinspection were as follows:

1. Each element will be completely disassembled.
2. It will be given a thorough visual inspection for the following:
   a. Quality of braze and its conformance to specifications.
   b. Pits, scratches, and other surface imperfections.
   The visual examinations will include both the inner and outer surfaces of the tubes and cover every square inch of surface. Inside examination is to be made with a borescope. Depth of surface imperfections is to be made with a measuring microscope on outer surfaces and by casting a replica of imperfections on inside surfaces. The replica is then to be measured with the microscope.
3. The tubes will then be given a complete dimensional check for conformance to specifications. This will include a measurement of maximum and minimum outside diameters at three positions along the tube and spot checking of inside diameters where OD's indicate abnormalities. The dimensional inspection will also include a determination of bowing, utilizing surface plate and feeler gauges.

4. Each tube will be given an ultrasonic test over its complete area to detect defects and imperfect bonding. Sensitivity of the ultrasonic test is first to be established with samples having known or simulated defects.
5. The ultrasonic test is then to be followed by a cleaning in hot detergent solution and a thorough hot-water rinse. The elements are then to be reassembled utilizing the original assembly tools. The reassembly is to be followed by another detergent cleaning and hot-water rinse.
6. In addition to the above listed tests on all elements, radiographs will be made of the two ends of all tubes from approximately one dozen elements. The radiographs are to show the end configuration of the fuel alloy and give an indication of the extent of "dogboning", if any.
7. In addition, approximately one dozen elements will be scanned by scintillation technique to detect any fuel non-homogeneity.
8. Following air-drying and cooling, the acceptable elements are to be packaged in polyethylene with a packet of desiccant and the polyethylene heat-sealed.

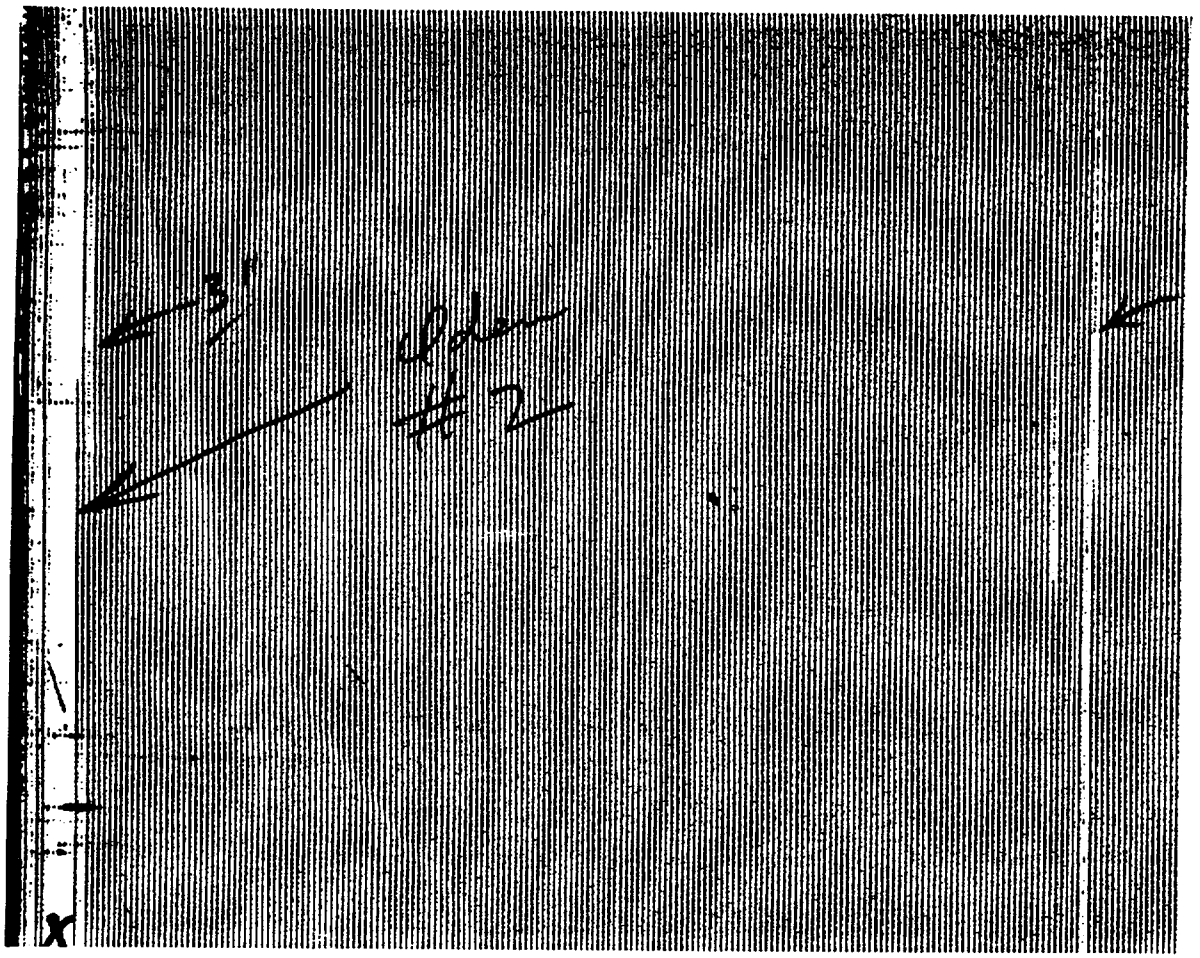
[illegible] key inspection, and one which was not previously used was the ultrasonic test, Item 5. This ultrasonic test consists of scanning each fuel tube by a sharp ultrasonic beam 0.093" in diameter. A mechanical traverse whereby the tube is fed through the beam in a spiral is set up in which the pitch of the spiral is also 0.093". In this way the entire surface is scanned. A record is produced on a chart which represents transmission through areas where no defect exists, and the chart line is interrupted whenever a defect interferes with normal transmission. Defects approximately 0.015" in diameter can be located, and means are available for determining interior defects against surface scratches.

A complete tabulation of the results of the mechanical inspection will be found in Appendix V. These results indicate many small deviations from specifications and a few elements were found with serious bows in the tubes or with visible blisters.

The complete results of the ultrasonic inspection are also tabulated in Appendix V. These results indicated a range of defects from perfect tubes to dozens of imperfections. The defects ranged in size from a few thousandths of an inch to greater than 1-inch in diameter.

To confirm the ultrasonic inspection method several tubes were sectioned at typical indicated flaw points in the ultrasonic record. Some of these records and the photographs of the sectioned flaws are indicated in Figures 37 and 38. The pictures by no means indicate the worst cases, but as indicated are representative. In each case of a suspected flaw the ultrasonic technique proved infallable and sectioning always produced the defect. All types of defects were discovered including poor bonding, cracks in the fuel, foreign inclusions, and voids. The conclusion reached is that the inspection requirements originally specified were not sufficiently rigid. In view of new tightened specifications this particular batch of fuel was of questionable quality, with over 1464 defects having been found, and with 133 of them over 1/2-inch in size in a sampling of 237 tubes.

In an effort to determine what size defect might be acceptable, new heat transfer calculations were made by computing machine. The problem that was set up provided a temperature profile of an element section that contained thermal insulating voids of various sizes at the boundary of the meat and the cladding. The heat flux could then be obtained over any given surface area. The results of these calculations are shown in Figure 39 which indicates the relative increase in heat flux as a function of defect size. Illustratively, a 1/4-inch defect would create a hot spot increase of 28% and a 1/2-inch defect of 61%. It will be recognized that an infinite



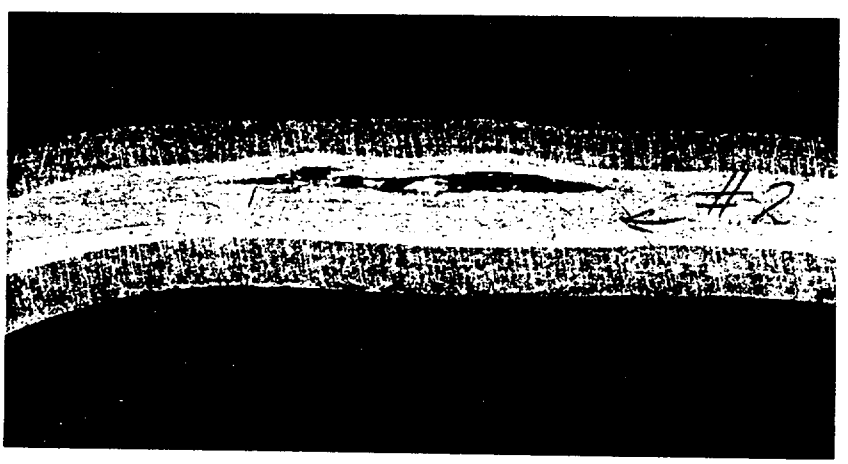
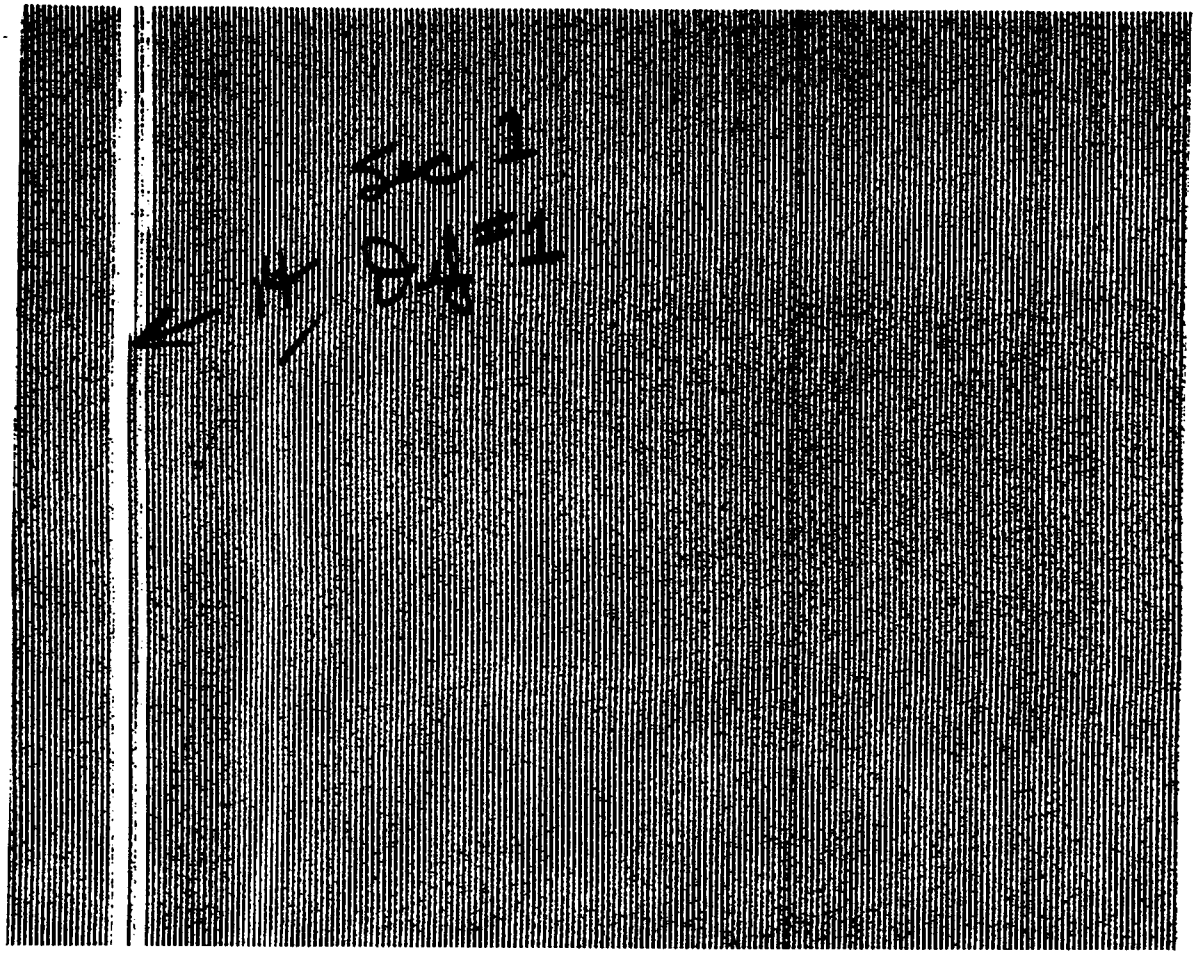


FIGURE 37
TYPICAL ULTRASONIC TRACE WITH DEFECT PHOTO



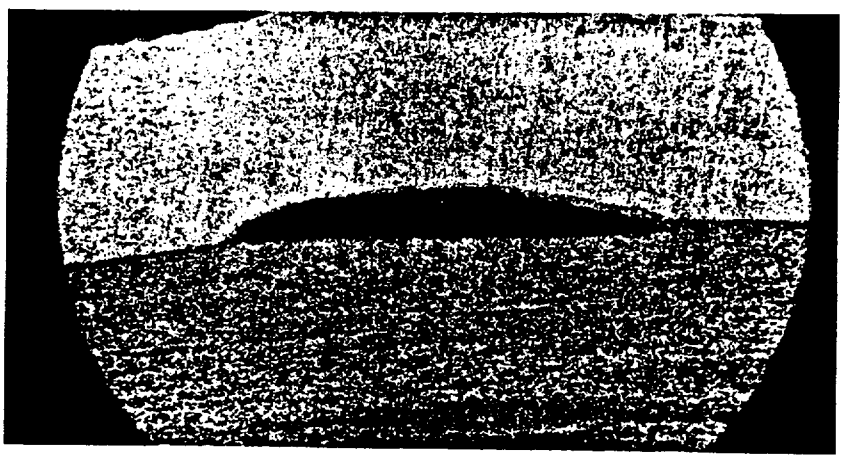


FIGURE 38

TYPICAL ULTRASONIC TRACE WITH DEFECT PHOTO

DEFECT HOT CHANNEL FACTOR INCREASE
AS A FUNCTION OF DEFECT SIZE

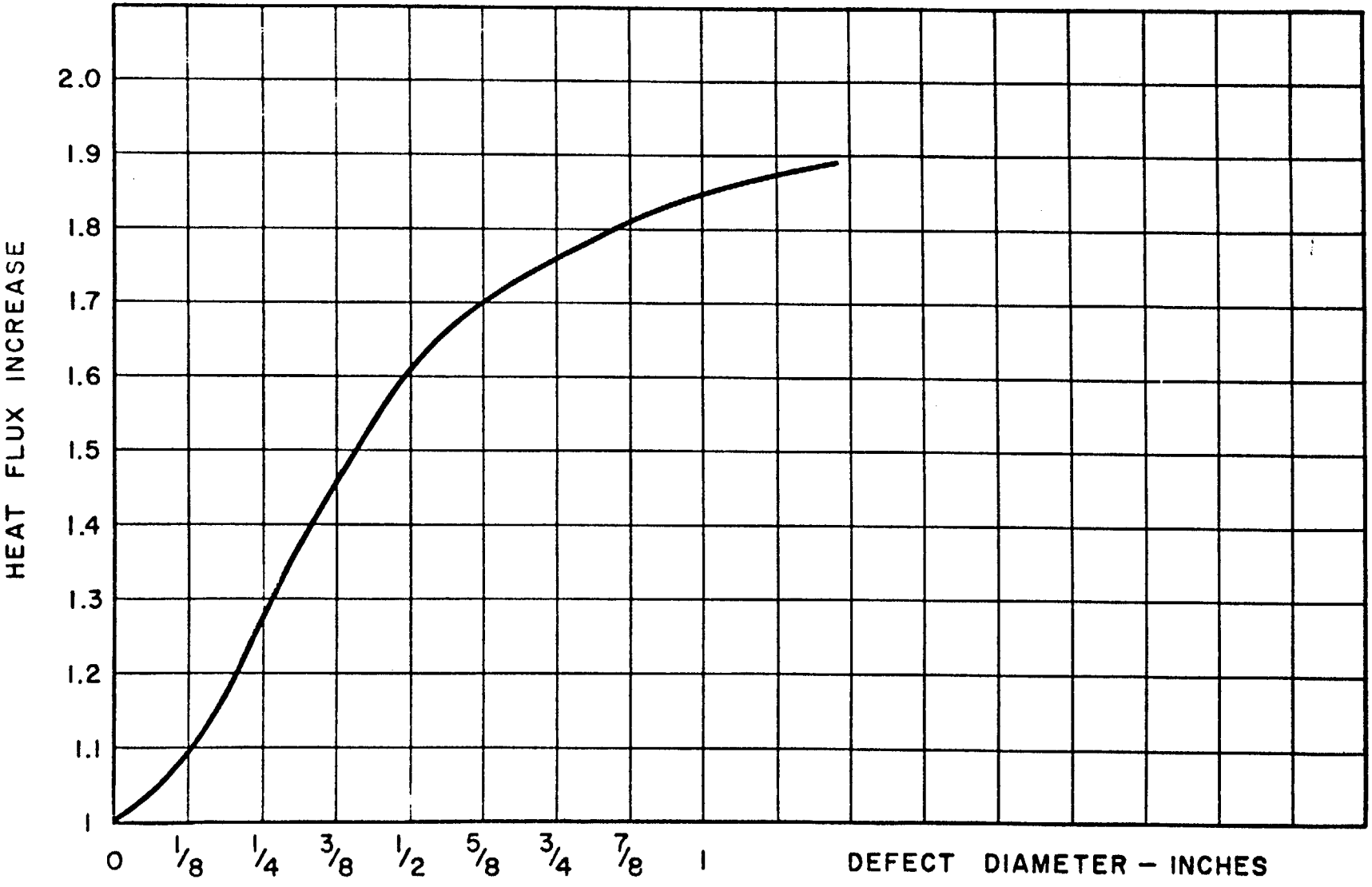


FIGURE 39

defect in bonding on one side will produce a 100% increase in heat flux through the other side. Thus, for practical purposes, a 1-inch diameter defect may be regarded as infinite. Specifications for future new fuel will call for no defects greater than approximately 1/8-inch as determined by ultrasonic methods and thus providing only an approximate 10% increase in heat flux.

An additional check was made on a typical fuel tube to determine if any of the defects grew upon temperature cycling. The fuel tube was first inspected ultrasonically and then placed in an autoclave. The temperature in the autoclave was cycled between 100°F and 400°F for 50 cycles. The tube was then removed and reinspected ultrasonically. No significant change in size or number of the defects was noted.

K. Thermal and Hydraulic Analysis

This section contains pertinent heat transfer information applicable to the reactor when the fuel element failed. The results are tabulated below and additional comments where required are referenced as superscripts to similar numbers in Appendix VI. Maximum heat flux is obtained in the following manner:

Reactor Power = $1.30 \times 10^8 \frac{BTU}{hr.}$		38 MW
Primary Coolant System Flow Rate		5250 GPM
Reactor Vessel Inlet Temperature		108°F
Reactor Vessel Outlet Temperature		158°F
Number of Fuel Elements - including control rods - 78		
Total Heat Transfer Area - 9 C.R. assumed 60% inserted into core[1]		680 $ft.^2$
Average Heat Flux		191,000 $\frac{BTU}{hr\text{-}ft^2}$
Neutron Flux Peaking Factors[2]		
Nuclear Peak to Average Radial	1.50	
Nuclear Peak to Average Axial	1.76	
Fuel Alloy Distribution	1.05	
Local Peaking	1.15	
Total -	3.20	
Maximum Heat Flux		610,000 $\frac{BTU}{hr\text{-}ft^2}$

Burnout heat flux is calculated below:

Coolant Flow Through Core - 2/3 of total P.C. flow	3500 GPM
Average Coolant Flow per Fuel Element	45 GPM
Flow Area per Fuel Element[3]	.01597 ft.^{2}
Average Coolant Velocity Through the Core[4]	6.26 ft./sec.
Coolant Mass Flow Rate	1.37×10^6 lb/hr-ft^2
Coolant Channel Width	0.188 in. = 0.0157 ft.
Pressure at Exit of Core	112 psia
Saturation Temperature for 112 psia	336°F

Coolant Velocity through the instrumented channels of the fuel element in L-5-6 based on coolant temperature rise with:

Radial Hot Channel Factor of 1.3[5]	6.12 ft/sec
Radial Hot Channel Factor of 1.5	7.06 ft/sec
Maximum Fuel Surface Temperature[6]	384°F
$T_{wall} - T_{sat}$	48°F
Burnout Heat Flux from DP-355 (Mirshak et. al.)[7]	$1.98 \times 10^6 \frac{BTU}{hr.ft.^2}$
Burnout Heat Flux from Jens & Lottes[8]	$2.21 \times 10^6 \frac{BTU}{hr.ft.^2}$

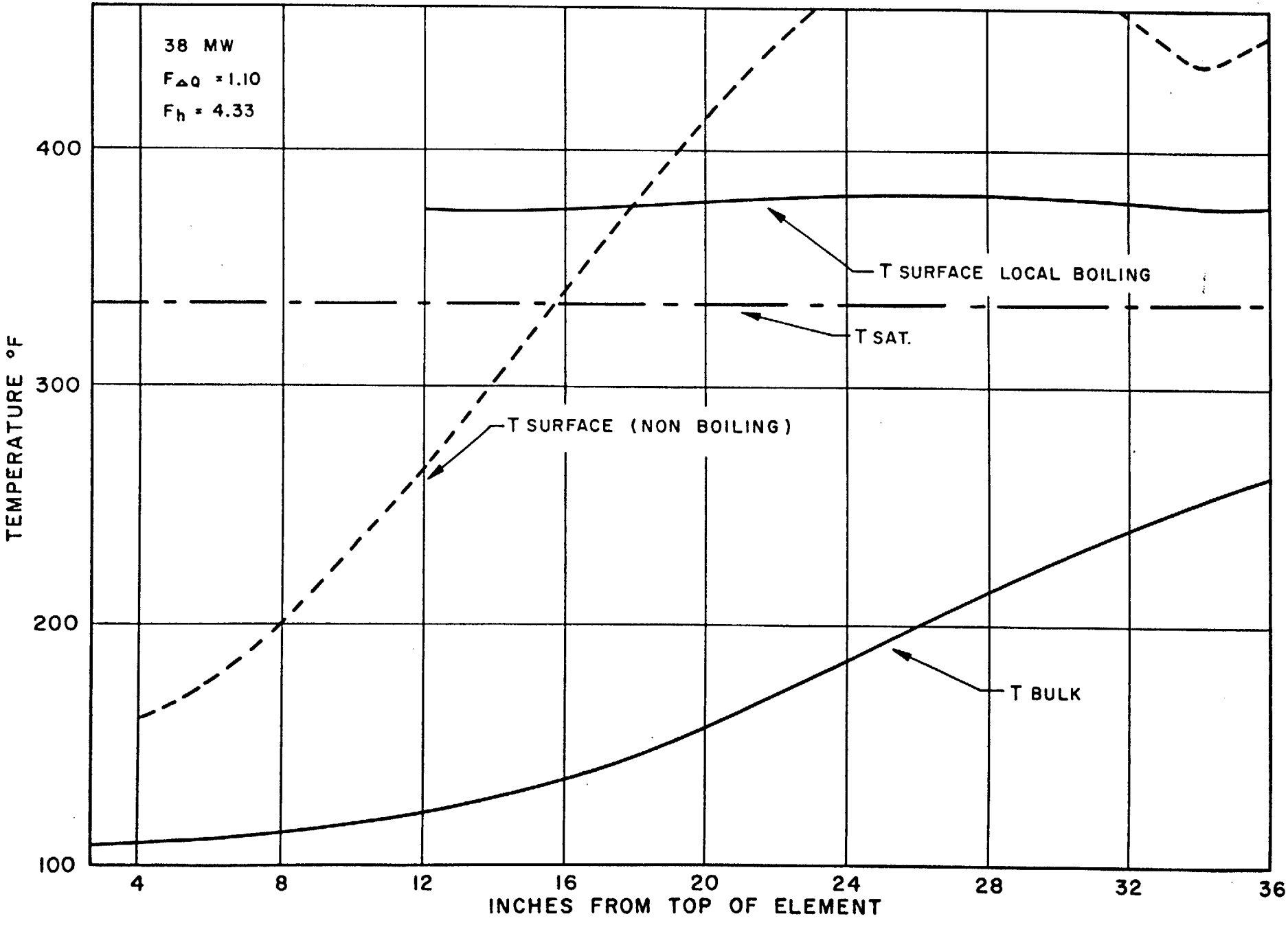
The conservative value of the maximum heat flux calculated previously was 610,000 BTU/hr. - ft.^{2}, or the ratio of the burnout heat flux as predicted by the best fit of the Mirshak et.al. data, to the maximum heat flux is 3.25. A conservative correlation factor of 0.60 might be applied to account for the spread in the Mirshak data points. This factor will decrease the burnout ratio to 1.95.

Figure 40 indicates the bulk coolant temperature and the fuel surface temperature for the hot channel of a fuel element in the core position of the failed fuel element. The initial cold critical axial flux distribution given in WTR-25 was used in deriving this curve. The bulk coolant temperature rise was taken as 1.10 times the bulk coolant temperature rise measured during the experiment. This number includes a power distribution factor across a fuel element of 1.04, a

FIGURE 40

LOCAL BOILING ZONE

COOLANT VELOCITY 6.2 fps



fuel content tolerance per fuel plate of 1.01 and a channel to channel coolant velocity variation of 1.05. These factors are those considered in WTR-25. A hot channel factor F_ϕ of 4.33 was applied to the average heat flux of 191,000 BTU/hr-ft.^{2} to obtain the film temperature rise at the axial maximum heat flux.

The following factors were taken from WTR-25 to obtain the value of F_ϕ:

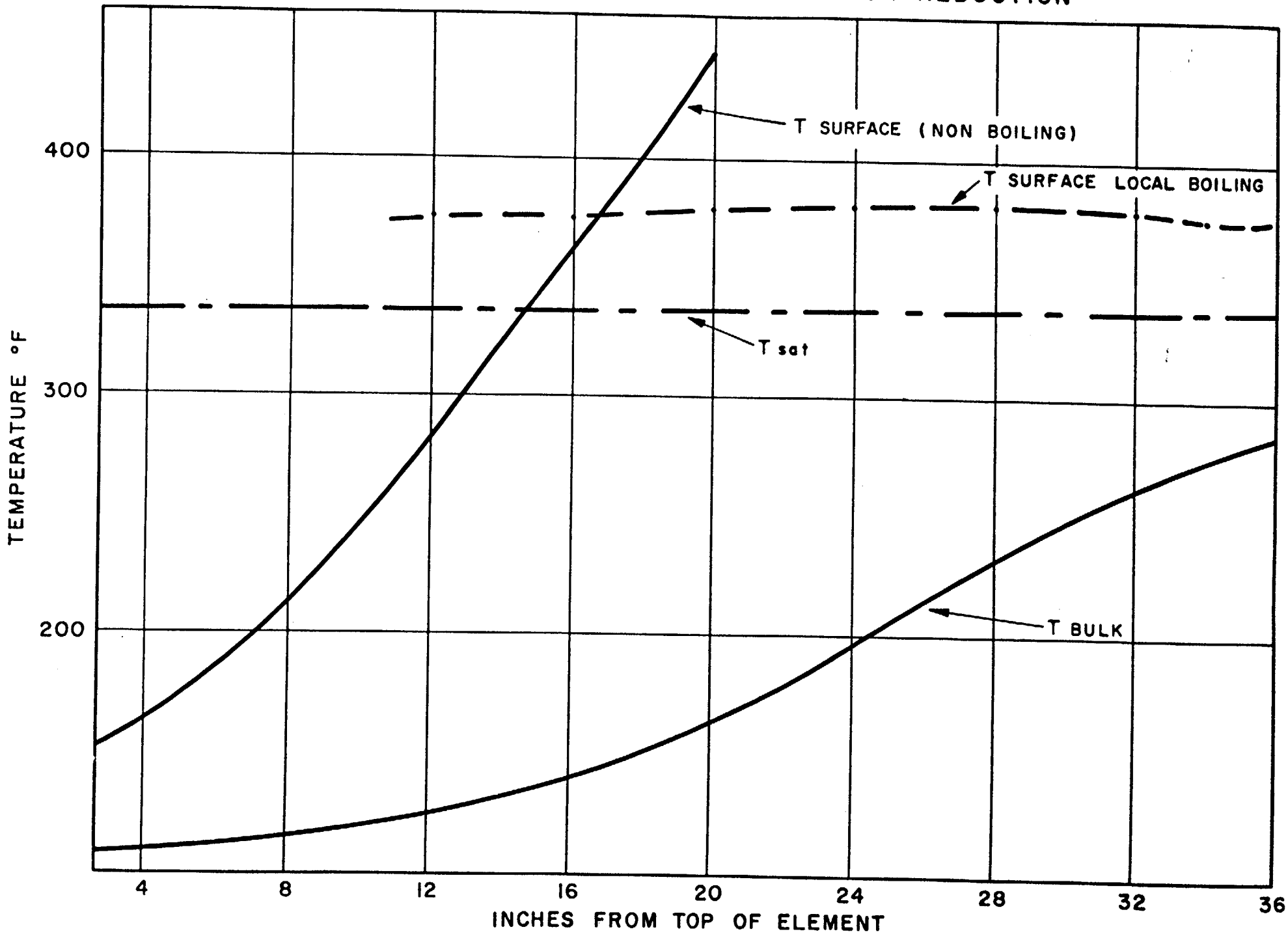
Radial max. to average power production	1.30
Axial max. to average power production	1.76
Fraction of power generated in fuel plates	0.95
Local reduction in coolant velocity due to fuel tube bowing	1.14
Local variation in fuel content of fuel plates	1.05
Power distribution across diameter of fuel element	1.04
Fuel content allowance per fuel plate	1.01
Local power peaking of unknown origin	1.10
Variation in coolant velocity	1.15
Correlation factor for computing film coefficient	1.25
Total	4.33

The Jens and Lottes[6] correlation for fuel surface temperature quoted above was used in the region of local boiling. A total hot channel factor of 2.75 was used in this case which includes a local power peaking of 1.20 in addition to the nuclear power factors.

Figure 41 indicates the bulk coolant temperature and the fuel surface temperature for the hot channel with an assumed 15% additional

FIGURE 41

LOCAL BOILING ZONE WITH 15% FLOW REDUCTION



reduction of flow. The bulk coolant temperature rise increases to 170°F and the discharge temperature to 278°F. The hot channel factor for film temperature rise increases by 1.12.

Comparing Figure 41 with Figure 40 local boiling occurs over a greater length of the fuel element with the reduced flow. This creates an increase pressure drop whose effect is not considered significant. Using the data reported by J. B. Reynolds in ANL-5178, it can be shown that the increased pressure drop caused by local boiling is not adequate to account for even the assumed 15% reduction in flow. Additional flow restrictions must be postulated to produce a dangerous condition.

Using a flow velocity of 6.26 ft/sec. and a less conservative hot channel factor for film temperature rise of 3.45, boiling would be expected at 30 MW. There is some indication from the boiling detector that boiling did begin at 30 MW as the primary coolant flow was reduced from 5500 gpm to 5250 gpm. This effect can be seen from Figure 20.

Additional heat transfer work has been presented in ORNL-CF-60-5-33 which contains the results of a meeting at WTR April 28. It is concluded in this report, and the results given above confirm, that-- "an analysis made with the best data presently available and with pessimistic estimates of all pertinent factors indicates that a boiling - burnout type failure of a good fuel element probably did not occur."

All of these analyses were conducted before the examination of the cold fuel elements described in Section J. From this previous section, a bonding defect between a 1/2-inch in size and infinity must be regarded conservatively as doubling the maximum heat flux. Using the 0.60 correlation factor with the Mirshak et. al. expression the ratio of burnout heat flux to maximum heat flux was seen to be 1.95. Thus a bonding defect in excess of 1/2-inch diameter could account for the fuel element failure.

L. Other Information and Problems Associated with the Incident

During the aftermath of the incident, a number of non-routine situations arose some of which may be of interest. Most of this information is in the Health Physics area, and includes such items as initial radiation surveys, environmental surveys, radiation protection, primary system decontamination, waste disposal problems, and handling of huge quantities of radioactive water. This information is presented in Appendix VII.

M. Conclusions

A fuel element failure occurred in the WTR on April 3, 1960. The effects of this failure were not measurable off the WTR site. The cause of the failure cannot be established beyond reasonable doubt, but it is reasonable to believe that a normal fuel element operating under the specified test conditions would not have failed. No information has been found that indicates that the stated operating conditions were not being met. A strong possibility exists that the failed element was not normal. Fuel element specifications and inspection in the past have been too lax and rigid control will be required in the future.

APPENDIX I

Raising Power Level From 20 MW to 60 MW

The reactor power is to be raised from an operating level of 20 MW to 60 MW in incremental steps of 5 MW. A set of operating parameters has been established for each power level and will be adjusted during the program, as the need arises. If plant limitations are exceeded, the program will stop until remedial action is taken. In general, the program consists of establishing that the reactor can be operated safely with a given set of conditions and then operating under these conditions for several days.

The criteria for safe operation have been established in the WTR License Application Amendment No. 14 and supplementary information issued to the AEC and in Amendment No. 1 to the WTR Operating License. They are:

1. First fuel ring bulk water outlet temperature. $< 220°F$
2. Inlet primary coolant temperature. $< 140°F$
3. Nuclear power level oscillation controlled with the automatic control system. $< \pm 5\%$
4. Radiation level at accessible portions of the face of the biological shield. < 1 mr/hr
5. Radiation level at accessible points in the primary loop. < 1 mr/hr
6. Boiling will be detected by the "bubble experiment" described in WTR-SS-TA-258.
7. Less than 1% voids will occur in the moderator due to boiling.
8. Maximum heat flux will not exceed one-half of burnout heat flux.

Table I lists the pertinent plant parameters establishing the safety of operating at a given power level. These conditions are obtained by establishing the required flow for a given power with the reactor at a power level 5 MW below that desired. A reference noise level should then be established on the bubble detection equipment. The reactor power should then be raised to the specified level. During this time particular attention should be given to the bubble detection equipment. The power should be reduced to the starting point if boiling is detected. If boiling is not detected, the reactor should be operated with these conditions for 15 minutes or until boiling is noted. At the end of this period the flow should be increased to the value listed in Table II, the special nuclear channel of the bubble equipment shut down, and operation continued at that power level for the period given in the schedule, Table III. Two exceptions will occur. The boiling check will be made prior to the 12-day run at 40 MW and a second reduced flow experiment will be carried out after 2-day operation at 60 MW.

Secondary coolant parameters and cooling tower operation are left to the judgement of the reactor operator. Standard records should be maintained of the temperatures and the flow rate, but the actual values are not important if the required primary coolant conditions are maintained.

Radiation survey of the entire primary system will be required during the program to insure against excessive radiation levels.

Based on a void coefficient of -0.14% reactivity/% voids in the moderator, the loss of .14% reactivity will indicate that a prescribed limitation has been exceeded. This reactivity is equivalent to 3% motion of the peripheral control rod in its most sensitive position. This motion can also be caused by a core temperature increase of about 15°F. If this rod motion is suddenly required during an increase in power level or subsequently, the reactor power level should be decreased to 10 MW and the

incident reported to Scientific Support for appraisal. It is assumed that temperature can be held constant to within ± 5°F.

The bulk outlet temperature of the first fuel ring will be measured by installing thermocouples in a "V" basket inserted in core position L-7-6.

Boiling will be detected by the "bubble detection" equipment described in WTR-SS-TA-258. A description of this equipment and its operation will be contained in an appendix to this test specification to be written after the installation is made and checked out.

TABLE I

Power MW	P.C. Flow GPM	R.V. ΔT °F	T_{in} R.V. °F	T_{out} R.V. °F	ΔT Alarm °F	Power Level Cutback MW	Power Level Scram MW	Low Flow Cutback GPM	Low Flow Scram GPM
20	6650	21	135	156	25	25	30	5650	5000
25	6650	26	130	156	31	28	30	5650	5000
30	6650	32	125	157	38	35	40	5650	5000
35	6650	37	120	157	43	40	45	5650	5000
40	7800	36	120	156	42	45	50	6650	5850
45	9150	34	120	154	40	50	55	7750	6850
50	10600	33	125	158	39	55	65	9050	7950
55	12300	31	125	156	36	60	70	10500	9200
60	14100	30	125	155	35	65	75	12000	10600
SPECIAL RUN									
60	13000	32	125	157	17	65	75	11000	9750

TABLE II

Power MW	P.C. Flow GPM	R.V. ΔT °F	T_{in} R.V. °F	T_{out} R.V. °F	ΔT Alarm °F	Power Level Cutback MW	Power Level Scram MW	Low Flow Cutback GPM	Low Flow Scram GPM
20	8000	18	140	158	21	25	30	6800	6000
25	8000	22	135	157	26	28	30	6800	6000
30	8000	26	130	156	31	35	40	6800	6000
35	8000	30	130	160	35	40	45	6800	6000
40	9400	30	130	160	35	45	50	8000	7050
45	11000	28	130	158	33	50	55	9350	8250
50	12800	27	130	157	32	55	65	10900	9600
55	14800	26	130	156	30	60	70	12600	11100
60	17000	25	130	155	29	65	75	14500	12750

TABLE III

Schedule for Increase in Power

February 18	20 MW	1 day
February 19 - 20	25 MW	2 days
February 21 - 22	30 MW	2 days
February 23 - 24	35 MW	2 days
February 25 - March 1	40 MW	7 days
March 11 - 23	40 MW	(480 MWD)
March 26 - 27	45 MW	2 days
March 28 - 29	50 MW	2 days
March 30 - 31	55 MW	2 days
April 1 - 2	60 MW	2 days

APPENDIX 11

Power Escalation Program
Local Boiling In The Core

Several changes have been made in the reactor core instrumentation which should aid in the detection of the beginning of local boiling. They are:

(a) The removal of the core access tube.

(b) Insertion of the helium bubble into a fuel element V basket for the production of bubble in the P.C. flow.

(c) Instrumentation of the two (2) fuel element channels in addition to the bulk outlet of the element.

(d) Direct measurement of control rod position.

The reactor has previously operated at 40 MW with 7,000 gpm P.C. flow and an inlet temperature of 125°F with existence of only mild disburbance on the special nuclear instrumentation. Since then it has been established that there was some boiling in the core access tube installed in the core.

For the present test, the reactor will be brought to equilibrium at 30 MW with a flow of 5,000 gpm and a core inlet temperature (heat exchanger outlet temperature) of 130°F. A reference set of data will be taken. The reactor power will be raised successively to 35 MW, 38 MW, 40 MW, 42 MW and 44 MW. Automatic control will be accomplished by one rod during the entire test. Data to be taken at each power level is: T_{in} R.V.; T_{out} R.V.; T_{out} heat exchanger; position of control rod on automatic control; temperature of the six (6) thermocouples in the instrumented fuel element, and the compensated power trace with the special nuclear instrumentation.

Scram and Cutback

Setting	-	Low Flow	Cutback	4,000 gpm
			Scram	3,000 gpm
		Power - Nuclear		
			Cutback	55 MW
			Scram	60 MW
T_{out} R.V.	-	Keep less than 180°F.		
T_{in} R.V.	-	T_{out} of heat exchanger		130°F
L. P. Thimble	-	Alarm - Set to 200°F		
T_{out} of fuel element		- Less than 250°F		
Reactor ΔT		Cutback 76°F		

APPENDIX III

Outline of Main Line
Actions Taken Since Incident On April 3, 1960

Date	
4/3	20.40 hrs. Radiation monitor alarms. 20.44 hrs. Reactor scrammed.
4/4	Survey of radiation levels throughout offices and plant. P.C. shutdown flow system in operation to cool off reactor.
4/5	Clean up operations commenced: 1. Ventilation purge of surge tank etc. to remove gas. 2. Flow of P. C. water through ion exchanger (new resin). 3. Clearing of loose equipment etc. from vicinity of reactor head.
4/6	Gas and airborne contamination cleared sufficiently to permit reversion to normal ventilation for vapor container. Revented to normal P. C. flow at 4000 gpm to purge system and degas. Additional Barrel Ion Exchangers installed to hasten cleanup of P. C. water.
4/7	Loops isolated in Subpile Room and H.P. thimble purged. Reactor vessel purged in 20 minutes at 550 gpm with normal DW purge system. Fuel handling equipment checked. Reverted to P.C. shutdown flow. All loose contaminated equipment removed from Reactor Head platform to Trucklock.
4/8	Preparations made for reactor upper head removal. All outside of reactor vessel and platforms covered with paper/polyethylene/paper to minimize spread of contamination. Temporary support beams installed across canal in case head was too hot for Drydock.

4/9 Head lifted a short distance for examination and radiation survey. Replaced as being too radioactive pending construction of beta shields and preparation of washing equipment, etc.
Retention basin closed off due to high water level.
Reactor vessel access ports removed to study decontamination procedure.

4/10 Continued preparations for head removal.
Prepared P.C. poison system for immediate use.

4/11 Completed preparations for head removal. Removed head, lifting slowly and decontamination outer surfaces of barrel etc., by scrubbing as it came up. Placed in drydock and sheeted up with ploye thylene to contain loose contamination. Commenced unloading fuel (12 elements removed to canal).

4/12 Continued core unloading (6 more elements) but trouble with discharge mechanism stopped work.
Fuel shuttle mechanism repaired.
Checked freedom, etc., of elements in core.
Installed barrel ion exchanger system to purify canal water.
Continued removal of fuel from core.

4/13 Continued removal of fuel from core - 10 elements and experiments remaining.
Worked on removing experiments.

4/14 Continued work as stated in 4/13.

4/15 Commenced unloading control rods. Accidental release of thimble loader spilled P.C. water over Subpile Room and vapor container floor - high contamination.
Cease all work to decomtaminate vapor container, etc.

4/16 Decontamination of floors, etc.

4/17 Decontamination of floors, etc.

4/18 Continued removal of control rods.

4/19 Completed removal of control rods.
Upper portion of damage fuel element removed and laid on reflector segments to await cask for removal.

4/20 Removal of remaining experiments and upper portion of damaged element in cask to bottom of canal.

Barrel ion exchanger put into operation to purify water in reactor vessel.

Tools being fabricated to drill out remainder of damaged element.

4/21 Drilling operations on remainder of element - hole saw driven by electric drill.

4/22 Attempted to remove remainder of element - tools not satisfactory. New tools obtained.

4/23 Continued work started on 4/22.

4/24 Continued work started on 4/22.

4/25 Final portion of damaged element removed from hole jammed in tool. Hole probed with dummy element and found clear.

Reamer run through shroud hole to clean any protrusions.

4/26 Filter and suction cleaning rig assembled at reactor top to attempt to pick up loose pieces of element.

4/27 Suction cleaning operation continued with only moderate success.

4/28 Suction cleaning rig secured.

Reactor head replaced and connections made to drain on lower head in order to drain vessel. Drain hose taken to waste line in annex area.

4/29 Operated alternate flush and drain cycle on reactor vessel. Some success in removing pieces of active material. Finally closed off drain and refilled vessel - reverted to shutdown coolant flow with heat exchanger bypasses open.

Loose water, etc., in pipe tunnel sump being pumped out to give access.

4/30 Temporary tank farm at retention basin now available - waste water transferred to tanks from retention basin, giving additional waste capacity.

Reactor head bolted down for reversion to P.C. main flow of [illegible] gpm to attempt to shake loose deposits of crud.

Idle was an attempt to clear hot spots and drive crud into surge tank.

5/1/60 Broken thermocouple connection on main P.C. piping caused loss of some water into pipe tunnel sump. System drained down for repair (water now sufficiently clean to permit this.)

5/2 Restarted main P.C. flow and increased to 14,000 gpm for several hours. Then shut down and drained vessel to see if the levels of radiation had dropped. No useful change observed. It was decided to try effect of maximum flow.
Restarted P. C. main circulation.

5/3 Worked up to P.C. flow of 20,000 gpm. Subsequently reduced P.C. circulation to 10,000 gpm for a short time, stopped, and started to drain vessel through hose and valve on top loop room. Hose fractured.

5/4 Hose system repaired and vessel drained down.

5/5 Washed out empty reactor vessel with D.W. through vent valve. No appreciable change in radiation levels.
Started periodic flushing to waste line using canal water supplied by emergency diesel pump.
Inserted dummy thimble in No. 6 hole to eliminate gamma beam.
Connected heat exchanger drain line to canal drain and commenced addition of nitric acid to heat exchangers.

5/6 Continued period flushing of reactor vessel.
Continued chemical decontamination of heat exchangers.
Commenced clean up of vapor container to permit tiling of floor to cover and hold in contamination which cannot be removed from concrete.

5/7 Continued work as stated in 5/6.

5/8 Continued as stated in 5/6.
Rigged suction cleaner and filter for 36" P.C. line in pipe tunnel sump. Broke into P.C. line and welded in stub for insertion of hoses. This was an attempt to remove very hot crud from bottom of line.

5/9 Started recirculating water in new cleaner rig completed (36" P.C. line).
Continued other operations as of 5/6.

5/10 Continued work as stated in 5/9.

5/11 Ceased flushing of reactor vessel and rigged dual pump, filter, strainer and catch tank system with bypasses, etc. for installation over canal, connected to reactor vessel lower head drain. This rig was intended to flush vessel without the collection of water in the vessel.

5/12 Operating recirculation cleaning rig with some success in collecting crud but no useful change in Subpile Room radiation level.

Commenced cold fuel element inspection program at AFD, Cheswick.

5/13 Replaced strainer and filter in recirculation system - new units submerged in canal so that a higher radiation level could be accepted.

5/14 Continued recirculation of reactor vessel.

5/15 Reactor vessel flushed with canal water using emergency pump. Drained back to canal.

Recirculation system in operation.

5/16 Continued flushing of vessel as indicated on the 15th.

Plastic coating of vapor container floor not satisfactory due to smooth surface.

Sandblasting improved adhesion but caused airborne contamination.

5/17 Commenced preparations for tiling vapor container floor (thermoplastic tiles).

Added Versene as decontaminant to reactor vessel recirculation system.

5/18 Drained out decontaminant from reactor vessel and recirculation rig and flushed through.

Rig disconnected and moved due to high radiation levels.

Secured 36" P.C. line filter system.

Prepared for thimble and shroud removal to facilitate decontamination of vessel and to check crud level.

5/19 No. 5 thimble and shroud removed and 4" drain line attached to shroud tube hole in lower head. This drain was 16' long and arranged with a coarse strainer at the bottom, six feet below

5/19 (Cont.) water level, in the hope that use of the emergency pump would wash pieces of crud out of the vessel.

A rudimentary periscope was used to inspect inside of vessel below biological shields - negative result.

5/20 Commenced flushing vessel using emergency pump. No appreciable change of radiation level observed in Subpile Room. Operation suspended and commenced untolting of remaining shroud tubes preparatory to removal.

5/21 Radial spray nozzles inserted through No. 5 shroud tube hole in lower head. Lowerportion of vessel flushed with this rig. Negative result.

Removed spray nozzles. No. 5 hole plugged.

5/22 Removed No. 3 thimble and shroud and plugged hole with stainless steel plug. Operating further barrel ion exchangers on canal water.

5/23 No. 2 thimble and shroud removed and hole plugged.

No. 7 thimble and shroud removed and hole plugged.

5/24 Removed No. 6 shroud and plug (dummy) and plugged shroud hole.

Removed No. 4 thimble and shroud.

5/25 Removed No. 1 (central) thimble and shroud. Removal of all thimbles did not lower radiation levels in Subpile Room, but rather levels in general rose somewhat. All available personnel utilized in bolting up plugs.

5/26 Installation of rig (manifold and piping) for chemical decontamination of upper surface of lower head of vessel

5/27 continued preparations as for 26th. Further inspection of upper side of lower head by periscope revealed at least two pieces of burnt out element. Preparations begun for mechanical removal. Chemical attack postponed.

5/28 Vacuum cleaning of particles and flushing continued. Hot spot

5/29 under head in Subpile Room reduced from ~50 r/hr to 4 r/hr

5/30

5/30 Radiation levels in Subpile Room generally acceptable.

This log terminated.

APPENDIX IV

Outline of Post Mortem Examination Of Element B-62

I. Visual Examination and Photographs

A. General

1. Interior.
2. Exterior.
3. Determination of relative damage to all three tubes.

B. Macrophotographs

1. Unaffected area.
2. Discolored area.
3. Melted-down area.
4. Transition bands.
5. Braze.
6. Holes, cracks, exposed edges.

C. Bottom Nozzle

1. Interior.
2. Exterior.

II. Background Study and Evaluation

A. Fuel Fabrication

1. Fabrication history.
2. Inspection report.
3. Pre-irradiation photomicrographs (if available).
4. Drawings.

B. Irradiation History and Operating Conditions

1. Burnup.
2. Heat Flux.
3. Neutron Flux
4. Surface Temperature
5. Pressure.
6. Primary Coolant Flow Rate.
7. Power Level.

C. Other Data

1. Ni-foil experiment.
2. "V" basket drawings.
3. Other data as required.

III. Preliminary Measurements

A. Dose Rate

B. Dimensional Measurements

1. B-62 (where possible).
2. Adjacent fuel elements (B-19, B-38, A-86, A-93).

C. H_2O Sample

D. Multichannel Analyzer Readout Trace.

IV. Disassembly

A. Gross Sectioning

1. Fuel Tube Separation.
2. "Napkin-ring" preparation.
3. Longitudiual sectioning.
4. Additional photographs as required.

B. Sampling

1. Chemical Analysis Samples.
2. Metallography Samples.

V. Detailed Examination and Analysis

A. Chemical Analysis

1. Total U
2. Other Analyses as required.

B. Metallography

1. Comparison of Selected Area.
2. Bond Integrity.
3. Corrosion Evidence
4. Microhardness measurements (if required.)

VI. Reporting of Results

A. Interim Report

1. After Phase III.

B. Final Report

APPENDIX V

Tabulations of Cold Fuel Element Inspection Results

Table 5-1 indicates the tabulation of the results of the mechanical inspection. The symbols used are as follows:

S - Small tube

M - Medium tube

L - Large tube

X - Rejected, out of specification

A - Within specification

B - Out of original specification, but probably acceptable

* - Examined

\- - Not examined (usually because out of specification on other grounds)

A number
Viz., 1.618 - Actual dimension when out of specification tolerance

Tables 5-2, 5-3, and 5-4 are a tabulation by defect size of the results of the ultrasonic inspection. Some elements included in the mechanical inspection were not given the ultrasonic inspection because they were sectioned, or had obvious defects such as blisters.

Table 5-1
Tabulation of Results of Mechanical Inspection of Fuel Elements

Tube No.	Size	Surface I.D.	Surface O.D.	Braze I.D.	Braze O.D.	Bow	S. N. End Adj. to Seam	S. N. End 90°	Center Adj. to Seam	Center 90°	Opp. End Adj. to Seam	Opp. End 90°	Radiograph	Homogeneity	Comments
6-A14	S	X	-	-	-	-	-	-	-	-	-	-	-	-	
63-11	S					A	A	A	A	A	A	A	B	A	
63-14	S	*	B	*	B	A	A	A	A	A	A	A	B	A	
63-13	S	*	*	*	*	A	A	A	A	1.628	A	A	-	-	
43-5	S	*	B	*	B	A	A	A	A	A	A	A	B	A	
80-5	S	*	B	*	B	A	A	A	A	A	A	A	A	A	
69-13	S	X	-	-	-	-	-	-	-	-	-	-	-	-	
82-12	S	*	A	*	A	A	A	A	A	A	A	A	B	A	
81-6	S	*	A	*	A	A	A	A	A	A	A	A	B	A	
81-9	S	*	*	*	*	A	A	A	1.6265	A	A	A	-	-	
81-15	S	*	A	*	A	A	A	A	A	A	A	A	A	A	
43-13	S	*	*	X	*	A	1.618	A	A	A	1.618	A	-	-	
80-4	S	*	B	*	A	A	A	A	A	A	A	A	B	A	
408-8	S	-	-	X	X	-	-	-	-	-	-	-	-	-	
411-2	S	*	B	*	X	A	1.1615	A	A	A	A	A	-	-	
411-3	S	*	B	*	A	A	A	A	A	A	A	A	X	A	
65-3	S	*	B	*	A	A	A	A	A	A	A	A	B	A	Sectioned
69-7	S	*	*	*	*	A	A	A	A	A	1.615	1.615	-	-	
408-12	S	*	A	*	A	A	A	A	A	A	A	A	B	A	
2-A-3	S	*	*	X	X	A	A	A	A	A	A	A	-	-	
408-6	S	*	B	*	A	A	A	A	A	A	1.6175	A	-	-	
411-13	S	*	A	*	A	A	A	A	A	A	A	A	-	-	
63-12	S	*	-	*	-	A	A	A	A	A	A	A	-	-	
12-13	S	*	B	X	B	A	A	A	A	A	A	A	*	*	
43-2	S	-	-	X	X	-	-	-	-	-	-	-	-	-	
6-A-12	S	*	B	*	A	A	A	A	A	1.626	A	A	-	-	
12-15	S	*	B	*	A	A	1.616	A	A	A	A	A	-	-	
43-12	S	*	B	*	B	A	1.612	A	1.616	A	A	A	-	-	
63-9	S	*	A	*	B	A	A	A	A	A	A	A	-	-	
63-5	S	*	-	*	-	.019	A	A	1.627	1.626	A	A	-	-	
43-8	S	*	-	-	-	A	1.617	A	1.626	A	A	A	-	-	
11-7	S	*	B	*	A	A	A	A	1.626	A	A	1.618	-	-	
31-12	S	X	B	X	B	A	A	1.617	A	A	A	A	-	-	
A-3	S	*	A	X	X	A	A	1.618	A	1.616	A	1.617	-	-	) Critical experi-
A-32	S	*	A	X	B	.020	1.615	1.612	1.616	1.614	1.614	1.612	-	-	) ment elements

Table 5-1
Tabulation of Results of Mechanical
Inspection of Fuel Elements

Tube No.	Size	Surface I.D.	Surface O.D.	Braze I.D.	Braze O.D.	Bow	S. N. End Adj. to Seam	S. N. End 90°	Center Adj. to Seam	Center 90°	Opp. End Adj. to Seam	Opp. End 90°	Radiograph	Homogeneity	Comments
411-1	S	*	B	*	B	A	1.616	A	A	A	A	A	-	-	
87-3	S	*	B	*	B	A	A	A	1.626	A	A	1.618	-	-	Autoclaved
87-15	S	*	*	*	*	A	A	A	A	A	A	A	-	-	
80-7	S	*	A	*	A	A	A	A	A	A	A	A	-	-	
82-8	S	*	*	*	*	A	A	A	1.627	A	A	A	-	-	
411-11	S	*	A	*	A	A	A	A	A	A	A	A	-	-	
82-10	S	*	B	*	A	A	A	A	A	A	1.618	A	-	-	
64-8	S	*	B	*	A	A	1.617	A	A	A	A	A	-	-	
411-6	S	*	B	*	A	A	A	A	A	A	A	A	-	-	
65-12	S	*	A	*	B	A	A	A	A	1.626	A	A	-	-	
87-1	S	*	A	*	A	A	A	A	A	A	A	A	-	-	
81-5	S														
82-11	S														
81-14	S														
81-1	S	*	A	*	A	A	A	A	A	A	A	A	-	-	
80-13	S	*	A	*	A	A	A	A	A	A	A	A	-	-	
80-9	S														
82-3	S	*	A	*	A	A	A	A	A	A	A	A	-	-	
81-5	S	*	A	*	A	A	A	A	A	A	A	A	-	-	
408-9	S	*	*	*	*	A	A	A	A	A	A	A	-	-	
63-6	S	*	*	*	*	A	A	A	A	A	A	A	-	-	
411-10	S	*	*	*	*	A	A	A	A	A	A	A	-	-	
87-7	S	*	*	*	*	A	A	A	A	A	A	A	-	-	
408-7	S	*	*	*	*	A	A	A	A	A	A	A	-	-	
411-12	S	*	*	*	*	A	A	A	A	A	A	A	-	-	
408-2	S	*	A	*	A	A	A	A	A	A	A	A	-	-	
43-7	S	*	X	*	A	A	A	A	A	A	A	A			
408-3	S	*	*	*	*	A	A	A	1.625	1.624	A	A	-	-	
408-4	S														
87-14	S														
408-14	S	*	A	*	A	A	A	A	A	A	A	A	-	-	
19-8	M	-	X	X	-	-	-	-	-	-	-	-	-	-	
41-11	M	X	-	-	-	-	-	-	-	-	-	-	-	-	
56-1	M	*	B	*	B	A	A	A	2.065	A	A	A	-	-	
55-4	M	*	A	*	B	A	A	A	A	A	A	A	B	A	

Table 5-1
Tabulation of Results of Mechanical
Inspection of Fuel Elements

Tube		Surface		Braze			S. N. End		Center		Opp. End		Radio-	Homo-	
No.	Size	I.D.	O.D.	I.D.	O.D.	Bow	Adj. to Seam	90°	Adj. to Seam	90°	Adj. to Seam	90°	graph	geneity	Comments
51-6	M	*	*	B	B	A	A	A	A	2.065	A	A	-	-	
77-8	M	*	*	*	*	.033		2.082				2.0595	-	-	Bumped
68-1	M	*	B	*	A	A	A	A	A	A	A	A	B	A	
76-2	M	*	B	*	A	A	A	A	A	A	A	A	X	A	
20-3	M	-	-	X	X	-	-	-	-	-	-	-	-	-	
91-9	M	*	*	X	*	A	A	A	A	A	A	A	-	-	
92-1	M	*	B	*	A	A	A	A	A	A	A	A	B	A	
91-4	M	*	*	*	*	A	A	2.055	A	A	A	2.055	-	-	
90-8	M	*	B	*	A	A	A	A	A	A	A	A	A	A	
92-9	M	*	B	*	A	A	A	A	A	A	A	A	B	A	
409-9	M	*	A	*	B	A	A	A	A	A	A	A	B	A	
410-7	M	*	B	*	B	A	A	A	A	A	A	A	B	A	
58-6	M	*	B	*	*	A	A	A	A	A	A	A	B	A	Sectioned - Ultrasonic on Axial scan
78-10	M	*	A	*	A	A	A	A	A	A	A	A	A	A	
92-10	M	*	*	*	*	A	A	A	A	2.053	2.055	2.055	-	-	
410-6	M	*	A	*	B	A	A	A	A	A	A	A	B	A	
409-10	M	*	A	*	B	A	A	2.053	A	2.057	A	A	-	-	
404-3	M	*	A	*	B	A	A	A	A	A		2.055	-	-	
92-3	M	*	A	*	A	A	A	A	A	A	A	2.054	-	-	
409-11	M	*	A	*	B	A	A	2.055	A	A	A	A	-	-	
55-3	M	*	-	X	-	A	A	A	A	A	A	A	-	-	
56-9	M	*	A	*	B	A	A	A	A	A	A	A	-	-	
22-7	M	*	-	*	-	A	A	A	A	A	A	A	-	-	
56-5	M	*	A	*	X	A	2.063	A	2.063	A	A	A	-	-	
16-5	M	*	A	*	B	A	A	A	A	2.064	A	A	-	-	
34-9	M	*	B	*	B	A	A	A	A	A	2.055	A	-	-	
55-8	M	*	-	*	-	A	2.054	A	A	A	A	A	-	-	
51-10	M	*	A	*	A	A	A	A	A	A	A	A	-	-	
51-9	M	*	-	*	-	A	A	A	2.063	A	A	A	-	-	
56-4	M	*	B	*	B	A	A	A	A	A	A	A	-	-	
34-1	M	*	B	*	B	A	A	2.056	A	A	A	A	-	-	
9-3	M	*	*	*	*	A	A	A	A	A	A	A	-	-	
410-10	M	*	*	*	*	A	A	A	A	2.056	A	A	-	-	
404-10	M	*	*	*	*	A	2.055	2.056	A	A	A	2.055	-	-	
405-5	M	*	A	*	B	A	2.055	2.056	A	A	A	A	-	-	
70-9	M	*	B	X	B	A	A	A	A	A	2.056	2.055	-	-	

Table 5-1
Tabulation of Results of Mechanical
Inspection of Fuel Elements

Tube No.	Size	Surface I.D.	Surface O.D.	Braze I.D.	Braze O.D.	Bow	S. N. End Adj. to Seam	S. N. End 90°	Center Adj. to Seam	Center 90°	Opp. End Adj. to Seam	Opp. End 90°	Radio-graph	Homo-geneity	Comments
405-1	M	*	*	*	*	A	A	2.056	A	A	A	A	-	-	
405-8	M	*	B	*	B	A	A	A	2.063	2.056	A	2.055	-	-	
77-3	M	*	B	*	A	A	2.056	A	A	A	2.056	2.055	-	-	
A-3	M	*	X	*	B	-	-	-	-	-	-	-	-	-	)Critical experi-
A-32	M	*	B	X	X	.017	A	2.056	2.055	A	2.056	2.056	-	-	)ment elements
41-5	M	*	B	X	B	A	2.055	A	A	A	2.052	A	-	-	
405-9	M	*	B	*	B	A	2.055	2.056	A	A	A	A	-	-	
68-7	M	*	*	*	*	A	A	2.056	A	A	A	A	-	-	
56-11	M	*	*	*	*	A	A	A	A	A	A	A	-	-	
92-7	M	*	*	*	*	A	2.055	2.057	A	A	A	A	-	-	
404-11	M	*	*	*	*	A	A	A	A	A	2.055	2.057	-	-	
410-11	M	*	*	*	*	A	A	A	A	A	A	A	-	-	
405-11	M	*	*	*	*	A	A	A	A	A	A	A	-	-	
409-1	M	*	*	*	*	A	A	A	A	A	A	A	-	-	
76-10	M	*	A	X	A	A	A	A	A	A	A	A	-	-	Lack of Pene-
404-8	M	*	A	*	A	A	A	A	A	A	A	A	-	-	tration
90-1	M														
89-6	M	*	A	*	A	A	A	A	A	A	A	A	-	-	
405-3	M	*	A	*	A	A	A	A	A	A	A	A	-	-	
89-3	M														
89-4	M	*	A	*	A	A	A	A	A	A	2.057	2.055	-	-	
90-1	M	*	A	*	A	A	A	A	A	A	A	A	-	-	
409-2	M	*	A	*	A	A	A	A	A	A	A	A	-	-	
404-5	M														
77-10	M														
89-9	M														
69-11	M	*	X	*	A	A	A	A	A	A	A	A	-	-	
405-7	M														
706	M														
409-8	M	*	*	*	*	A	A	A	A	A	A	A	-	-	
A-92	L	*	A	*	B	A	A	A	A	A	A	A	B	A	
B-59	L	X	-	X	-	-	-	-	-	-	-	-	-	-	
B-65	L	*	A	*	B	A	A	A	A	A	A	A	B	A	
B-68	L	*	X	*	*	A	A	A	A	A	A	A	-	-	
B-70	L	X	-	X	-	-	-	-	-	-	-	-	-	-	

Table 5-1
Tabulation of Results of Mechanical
Inspection of Fuel Elements

Tube		Surface		Braze			S. N. End		Center		Opp. End				
No.	Size	I.D.	O.D.	I.D.	O.D.	Bow	Adj. to Seam	90°	Adj. to Seam	90°	Adj. to Seam	90°	Radio-graph	Homo-geneity	Comments
B-91	L	*	B	*	B	A	A	A	A	A	A	A	B	A	
B-92	L	-	-	-	-	-	A	A	2.5003	2.501	A	A	-	-	
C-5	L	*	A	*	A	A	A	A	A	A	A	A	B	A	
C-11	L	*	B	*	B	A	A	A	A	A	A	A	X	A	
C-22	L	*	B	*	A	A	A	A	A	A	A	A	B	A	
C-24	L	*	B	*	B	A	A	A	A	A	A	A	B	A	
C-33	L	*	B	*	A	A	A	A	A	A	A	A	B	A	
C-42	L	X	*	*	*	A	A	A	A	A	A	A	-	-	
C-58	L	*	B	*	B	A	A	A	A	A	A	A	B	A	
C-65	L	*	A	*	B	A	A	A	A	A	A	A	B	A	
C-76	L	*	A	*	B	A	A	A	A	A	A	A	X	A	
B-84	L	*	*	*	*	A	2.490	A	A	A	A	A	-	-	
B-97	L	X	*	X	*	A	A	A	A	A	A	A	-	-	
C-48	L	*	*	X	*	A	2.493	A	2.493	A	2.493	A	-	-	
C-78	L	*	B	*	B	A	A	A	A	A	A	A	X	A	
C-80	L	*	X	*	B	A	A	A	A	A	A	A	-	-	
C-29	L	*	B	*	B	A	A	A	A	A	A	A	-	-	
C-44	L	*	B	X	B	A	A	A	A	A	2.494	A	-	-	
C-64	L	*	B	X	B	A	A	A	A	A	A	A	-	-	
B-67	L	*	B	*	A	A	A	A	A	A	2.505	2.493	-	-	
B-76	L	*	A	*	A	A	A	A	A	A	A	A	-	-	
B-51	L	*	-	X	-	A	A	2.492	A	A	A	2.492	-	-	
B-55	L	*	B	X	B	A	A	2.492	A	A	2.494	2.494	-	-	
A-94	L	*	B	*	B	A	2.494	2.490	2.494	A	2.493	2.494	-	-	
B-28	L	X	B	X	B	A	A	A	A	A	A	A	-	-	
B-64	L	*	-	*	-	A	2.503	A	A	2.502	2.504	A	-	-	
B-71	L	*	A	*	B	A	A	A	A	A	A	A	-	-	
B-45	L	*	*	X		A	2.501	2.494	A	A	A	2.494	-	-	
B-61	L	-	X	-	-	-	-	-	-	-	-	-	-	-	
B-26	L	-	X	-	-	-	-	-	-	-	-	-	-	-	
B-42	L	*	B	*	A	A	A	2.492	A	A	A	2.494			
C-75	L	*	A	*	B	A	A	A	A	A	A	A	-	-	
C-43	L														
C-59	L	*	A	*	A	A	A	2.494	A	A	A	A	-	-	
C-20	L	-	X	-	A	-	-	-	-	-	-	-	-	-	

Table
Tabulation of Results of Mechanical Inspection of Fuel Elements

Tube		Surface		Braze			S. N. End		Center		Opp. End		Radio-	Homo-	
No.	Size	I.D.	O.D.	I.D.	O.D.	Bow	Adj. to Seam	90°	Adj. to Seam	90°	Adj. to Seam	90°	graph	geneity	Comments
C-27	L	X	A	*	A	A	A	2.493	A	A	A	A	-	-	
B-77	L	*	B	*	A	A	A	A	A	A	A	A	-	-	
C-69	L	X	B	*	B	A	A	2.492	A	A	2.493	A	-	-	
C-6	L	*	B	*	B	A	2.492	A	2.494	A	2.492	2.494	-	-	
B-96	L	-	X	X	-	-	-	-	-	-	-	-	-	-	
C-70	L	*	A	*	A	A	A	A	A	A	A	A	-	-	
A-3	L	*	B	*	B	.062	2.493	2.491	2.490	A	2.489	A	-	-	
A-32	L	*	B	X	X	.185	2.493	A	A	A	A	2.492	-	-	)Critical experi-
C-10	L	-	-	-	-	-	-	-	-	-	-	-	-	-	)ment elements
C-71	L	*	*	*	*	A	A	A	A	A	A	A	-	-	
C-68	L	*	*	X	*	A	A	A	A	A	A	A	-	-	
B-87	L														
C-25	L														
C-40	L														
C-72	L	*	*	*	*	A	A	A	A	A	A	A	-	-	
B-58	L														
C-79	L														
C-40	L														
C-45	L	*	A	*	A	A	A	A	A	A	A	A	-	-	
C-46	L	*	A	*	A	A	A	A	A	A	A	A	-	-	
C-31	L														
C-30	L	*	A	*	A	A	2.493	A	A	A	A	A	-	-	
C-49	L														
C-66	L	*	A	X	A	A	A	A	A	A	A	A	-	-	
C-41	L	X	A	X	A	A	A	A	A	A	A	A	-	-	Overpenetrated glob of braze & braze ground away
C-39	L														
C-37	L	*	A	*	A	A	A	A	A	A	A	A	-	-	
C-57	L														
C-60	L														
C-35	L	*	A	*	A	A	2.492	A	A	A	2.493	2.493			
C-34	L	*	A	*	A	A	A	A	A	2.502	A	A	-	-	
C-53	L														
B-82	L	*	B	*	B	A	2.493	2.494	A	A	2.493	2.492	-	-	
C-9	L	*	A	*	A	A	A	A	A	A	2.493	2.492	-	-	
C-26	L	B	A	B	B	A	A	A	A	A	A	A	-	-	
C-19	L	B	A	B	A	A	A	A	A	A	A	A	-	-	

Table 5-1
Tabulation of Results of Mechanical
Inspection of Fuel Elements

Tube		Surface		Braze			S. N. End		Center		Opp. End		Radio-	Homo-	
No.	Size	I.D.	O.D.	I.D.	O.D.	Bow	Adj. to Seam	90°	Adj. to Seam	90°	Adj. to Seam	90°	graph	geneity	Comments
C-23	L	*	B	*	B	A	A	A	A	A	A	A	-	-	
C-21	L	B	B	A	A	A	A	A	A	A	A	A	-	-	
C-3	L	B	B	A	A	A	A	A	A	A	A	A	-	-	
C-18	L	*	A	*	A	A	A	A	A	A	A	A	-	-	
B-88	L	B	B	B	B	A	A	A	A	A	A	A	-	-	
B-79	L	A	A	A	A	A	2.493	2.495	A	A	2.492	2.495	-	-	
C-13	L	*	B	*	B	A	A	A	A	A	A	A	-	-	
B-93	L	B	B	B	B	A	A	A	A	A	A	A	-	-	
B-87	L	B	A	B	A	A	A	A	A	A	A	A	-	-	
B-99	L	B	A	A	A	A	A	A	A	A	A	A	-	-	
C-25	L	A	A	A	A	A	A	A	A	A	A	A	-	-	
91-3	M	*	*	*	*	A	A	A	A	A	A	A	-	-	
91-2	M	*	*	*	*	A	A	A	A	A	A	A	-	-	
78-6	M	*	B	*	A	A	A	A	A	A	A	A	-	-	
91-10	M	*	B	*	B	A	A	A	A	A	A	A	-	-	
91-7	M	*	A	*	B	A	A	A	A	A	A	A	-	-	
90-11	M	*	*	*	*	A	A	A	A	A	A	A	-	-	
41-10	M	*	*	*	*	A	A	A	A	A	A	A	-	-	
70-7	M	*	B	*	B	A	A	A	A	A	A	A	-	-	
70-4	M	*	*	*	*	A	A	A	A	A	A	A	-	-	
78-3	M	*	*	*	*	A	A	A	A	A	2.055	2.055	-	-	
69-10	M	*	X	*	A	A	2.056	2.054	A	A	2.053	2.054	-	-	
69-9	M	*	B	*	B	A	A	A	A	A	A	A	-	-	
77-1	M	*	*	*	*	A	A	A	A	A	A	A	-	-	
89-5	M	*	*	*	*	A	A	A	A	A	A	A	-	-	
68-4	M	*	*	*	*	A	A	A	A	A	A	A	-	-	
68-8	M	X	B	*	B	A	A	A	A	A	A	A	-	-	
82-6	S	X	B	*	A	A	A	A	A	A	A	A	-	-	Blister
82-7	S	*	X	*	B	A	A	A	A	A	A	A	-	-	
80-11	S	*	A	*	A	A	A	A	A	A	A	A	-	-	
82-15	S	*	A	*	B	A	A	A	A	A	A	A	-	-	
80-8	S	*	A	*	B	A	A	A	A	A	A	A	-	-	
81-3	S	*	*	*	*	A	A	A	A	A	A	A	-	-	
64-15	S	*	*	*	*	A	A	A	1.615	A	A	A	-	-	
64-11	S	*	A	*	A	A	A	A	A	A	A	A	-	-	

Table 5-1
Tabulation of Results of Mechanical
Inspection of Fuel Elements

Tube		Surface		Braze			S. N. End		Center		Opp. End		Radio-	Homo-	
No.	Size	I.D.	O.D.	I.D.	O.D.	Bow	Adj. to Seam	90°	Adj. to Seam	90°	Adj. to Seam	90°	graph	geneity	Comments
80-15	S	*	*	*	*	.016	A	A	A	A	A	A	-	-	
65-11	S	*	X	*	*	A	A	A	A	A	A	A	-	-	
80-6	S	*	*	*	*	A	A	A	A	A	A	A	-	-	
82-9	S	*	B	*	A	A	A	A	A	A	A	A	-	-	
81-13	S	*	*	*	*	A	A	A	A	A	A	A	-	-	
37-2	S	*	*	*	*	A	A	A	A	A	1.615	1.617	-	-	
81-8	S	*	*	*	*	A	A	A	A	A	A	A	-	-	
80-14	S	X	*	*	*	.019	A	A	A	A	A	A	-	-	Blister

TABLE V--4

SMALL TUBES

Results of Ultrasonic Inspection of Fuel Elements
Defects Catalogued by Size

Elements	Total No. of Defects	No. of Defects less than 1/4" in any dimension	No. of Defects between 1/4" & 1/2" in any dimension	No. of Defects greater than 1/2" in any dimension	Comments
2A-3					Bad braze.
6A-12	21	15	6		
6A-14	17	11	6		Surface defects of questionable size and intensity.
11-7	2	1		1	Circumferential defect. 1 - 2".
12-13					Surface defects of questionable size and intensity.
12-15	2	1		1	
31-12					Surface defects of questionable size and intensity.
37-2	15	12	3		
43-2					Bad braze.
43-5	10	8	2		Circumferential defect.
43-7	4	3		1	Circumferential defect. 1 - 7/8".
43-8	5	5			
43-12	8	7	1		
43-13	5	4		1	Circumferential defect. 1 - 1-1/2".
57-2	Nil				
63-5	20	20			Plus extensive surface defects.

TABLE V--4 (cont'd.)

Elements	Total No. of Defects	No. of Defects less than 1/4" in any dimension	No. of Defects between 1/4" & 1/2" in any dimension	No. of Defects greater than 1/2" in any dimension	Comments
63-6	6	2	3	1	
63-9	Nil				
63-11	27	25	2		
63-12	21	18	1	2	Plus extensive surface defects.
63-13	> 25	24 +	1		
64-7	> 12	4 +	6	2	Circumferential defect. 1 - 1/2".
64-8	19	15	3	1	
64-11	> 22	> 20	1	1	
64-13	> 20	8 +	9	3	
64-15	> 53	> 50	3		
65-11					> 0.012" surface pits.
65-12	4	3	1		
69-7	Nil				
69-13	Nil				
80-4	8	8			
80-5	7	7			
80-6	3	3			
80-7	10	10			Extensive surface defects.
80-8	3	3			
80-9	1			1	
80-11	Nil				
80-13	1	1			
80-14					Surface blisters.
80-15					Bow out of specification.
81-1	1	1			
81-3	4	4			

TABLE V--4 (cont'd.)

Elements	Total No. of Defects	No of Defects less than 1/4" in any dimension	No. of Defects between 1/4" & 1/2" in any dimension	No. of Defects greater than 1/2" in any dimension	Comments
81-5	5	4	1		
81-6	> 12	12 +			
81-8	4	4			
81-9	2	2			
81-13	1	1			
81-14	2		2		
81-15	12	12			
82-3	4	3	1		
82-6					Surface blisters.
82-7					> 0.012" surface pits.
82-8	8	7	1		
82-9	7	7			
82-10	10	10			Plus extensive surface defects.
82-11	3	2		1	
82-12	7	5	2		
82-15	1	1			
87-1	> 12	2 +	7	3	
87-3	12	11		1	Plus extensive surface defects.
87-6	2	1	1		
87-7	6	2	4		
87-14	2	1		1	Circumferential defect. 1 - 1/2".
87-15	Nil				Large surface marking.
408-2	3	2	1		
408-4	1		1		
408-6	12	12			
408-7	10	7	2	1	Circumferential defect. 1 - 3/4".

TABLE V--4 (cont'd)

Elements	Total No. of Defects	No. of Defects less than 1/4" in any dimension	No. of Defects between 1/4" & 1/2" in any dimension	No. of Defects greater than 1/2" in any dimension	Comments
408-8	7	5		2	Circumferential defect. 1 - 1".
408-9	2		2		
408-12	12	11	1		
408-14	Nil				
411-1	8	6	2		
411-2	2		1	1	Circumferential defect. 1 - 1/2".
411-3	13	12	1		
411-6	4	3	1		Plus extensive surface defects.
411-10	4	1	3		Circumferential defects. 2 - 3/8".
411-11	8	5	1	2	Circumferential void at end.
411-12	4	3		1	
411-13	1	1			

TABLE 5--3

MEDIUM TUBES

Results of Ultrasonic Inspection of Fuel Elements
Defects Catalogued by Size

Element	Total No. of Defects	No. of Defects less than 1/4" in any dimension	No. of Defects between 1/4" & 1/2" in any dimension	No. of Defects greater than 1/2" in any dimension	Comments
9-3	4	4			Masked by surface defects.
16-5	2		1	1	Circumferential defect. Surface marking.
22-7	10	10			
34-1	1	1			
34-9	1	1			Few 1-line defects.
41-5	> 10	8 +	2		Masked by surface defect.
41-10	12	11	1		Circumferential defect.
41-11	1			1	
51-[illegible]	6	5		1	Circumferential defect. 1 - 1/2".
51-9	5	5			Masked by surface defect.
51-10	6	6			Masked by surface defect.
55-3	10	6	2	2	Circumferential defect.
55-4	12	10		2	Circumferential defect. 2 - 3/4".
55-8	10	6	2	2	Circumferential defect and surface defects.
56-1	1	1			
56-4					Masked by surface defects.
56-5	6	3	2	1	
56-9	5	5			Masked by surface defects.
68-1	10	8	1	1	Large circumferential defect.

TABLE V--3 (cont'd.)

Element	Total No. of Defects	No. of Defects less than 1/4" in any dimension	No. of Defects between 1/4" & 1/2" in any dimension	No. of Defects greater than 1/2" in any dimension	Comments
68-4	20	15	4	1	Circumferential defect.
68-7	> 12	10 +	1	1	Bad circumferential defect.
68-8	2	1		1	
69-9	18	15	2	1	Circumferential defect.
69-10	3	3			
69-11	3	1	2		Hairline imperfection length of tube.
70-4	10	9	1		Circumferential defect.
70-6	7	7			
70-7	6	5	1		
70-9	5	4		1	Circumferential defect.
73-10	2	2			
76-2	2	2			
76-10	2	2			
77-1	> 20	15 +	2	3	Circumferential defect.
77-3	14	12	1	1	Circumferential defect. Surface marking.
77-10	5	4	1		This tube has a hair line parallel to braze through center section of tube.
78-3	4	4			Appear to be surface defects.
78-6	> 22	15 +	5	2	Circumferential defect.
89-3	3		2	1	
89-4	2	1	1		
89-5	> 18	15 +	2	1	Circumferential defect.
89-6	6	5	1		
89-9	3	2	1		

TABLE V-5 (Cont'd.)

Element	Total No. of Defects	No. of Defects less than 1/4" in any dimension	No. of Defects between 1/4" & 1/2" in any dimension	No. of Defects greater than 1/2" in any dimension	Comments
90-1	6	3	3		Quite a few 1-line defects.
90-8	10	8	1	1	
90-11	7	5	2		Circumferential defect.
91-2	10	9	1		Circumferential defect.
91-3	> 15	11 +	3	1	Circumferential defect.
91-4	8	3	4	1	Circumferential defect. 1 - 1-1/4".
91-7	15	12	2	1	Circumferential defect.
91-9	5	3	1	1	Circumferential defect.
91-10	15	13	2		Circumferential defect.
92-3	3	3			
92-	7	3	2	2	Circumferential defect. 1 - 3/4" and 1 - 1/2".
92-7	2	2			
92-9	> 10	8 +	1	1	Large circumferential defect.
92-10	1	1			
404-3	4	3	1		
404-5	4	3	1		
404-8	1	1			
404-10	5	5			
404-11					3 - 1 line defects.
405-1	> 10	9 +	1		
405-3	2	1	1		
405-5	7	5		2	Circumferential defect.
405-7	7	6	1		
405-8	14	12	2		Circumferential defect.

TABLE V-3 (cont'd.)

Element	Total No. of Defects	No. of Defects less than 1/4" in any dimension	No. of Defects between 1/4" & 1/2" in any dimension	No. of Defects greater than 1/2" in any dimension	Comments
405-9	5	5			
405-11					1 - 1-line defect.
409-1	1	1			
409-2					Few 1-line defects.
409-9	> 5	5 +			
409-10	3	1	2		
409-11	2	2			
410-6	2	2			Circumferential defect.
410-10	> 8	6 +		2	Circumferential defect. 1 - greater than 1".
410-11	3	2	1		

TABLE 5--2

LARGE TUBES

Results of Ultrasonic Inspection of Fuel Elements
Defects Catalogued by Size

Element	Total No. of Defects	No. of Defects less than 1/4" in any dimension	No. of Defects Between 1/4" & 1/2" in any dimension	No. of Defects greater than 1/2" in any dimension	Comments
A-92	4	3		1	Circumferential defect. 1 - 0.650'.
A-94	16	5	10	1	Many small inclusions.
B-26	14	3	6	5	Many small inclusions.
B-28	Nil				None apparent.
B-42	Nil				None apparent
B-45	< 20		4 +	1 +	Many small voids masked by surface. 1 - 0.480" circumferential defect.
B-51	17	7	6	4	
B-55	9	2	4	3	Possibly large surface defect.
B-58	> 20	Many	Many	8	
B-59	3	3			Surface defects.
B-61	7	1	6		
B-64	14	4	6	4	Many small inclusions. 1 - 1-1/4" and 1 - 1" circumferential defects.
B-65	4	2	1	1	1 - 1.7" Circumferential defects.
B-67					Many strata inclusions (major)
B-68	9	4	2	3	2 - 1/2", 1 - large Circumferential defect.
B-70					

TABLE V--2 (cont'd.)

Element	Total No. of Defects	No. of Defects less than 1/4" in any dimension	No. of Defects between 1/4" & 1/2" in any dimension	No. of Defects greater than 1/2" in any dimension	Comments
B-71	6		6		1 - 0.400" Small voids masked by surface
B-76	15	8	6	1	Many small inclusions.
B-77	4	2	2		
B-79					Completely masked by surface tracks.
B-82	4	1	3		
B-87	6	3	1	2	
B-88	2	2			
B-91	~ 10	5	4	1	Masked by surface defects.
B-92	7	5	2		
B-93	5	3	2		
B-96	5	3	1	1	
B-97	13	7	4	2	Circumferential defect. 2 - 3/4".
B-99	3	3			
C-3	9	5	4		
C-5	> 10	Many	4		Masked by surface defects.
C-6	~ 12	4	~ 8		Circumferential defect. 1 - 0.480".
C-9	8	5	2	1	
C-11	4	1	3		
C-13	3	3			
C-18	1				Defect close to end.
C-19	2	1	1		
C-20	1	1			
C-21	3	2	1		

Element	Total No. of Defects	No. of Defects less than 1/4" in any dimension	No. of Defects between 1/4" & 1/2" in any dimension	No. of Defects greater than 1/2" in any dimension	Comments
C-22	5	3		2	
C-23	1	1			
C-24	4	3		1	Circumferential defect. 1 - 3/4".
C-25	2	1		1	
C-26	Nil				
C-27	2	1	1		Circumferential defect. 1 - 0.3".
C-29	5	2	3		
C-30	2	2			
C-31	1		1		
C-33	~ 6	~ 3	3		
C-34	3	2	1		
C-35	3	1	2		Many 1-line defects.
C-37	1	1			Possible bad surface condition.
C-39	1		1		Circumferential defect. 1 - 3/8".
C-40	8	5	1	2	Circumferential defect. 1 - 2-1/4" and 1 - 1-1/4".
C-41					A few 1-line defects.
C-42	7	3		4	Circumferential defect. 1 - large; 3 - 1-1/2".
C-43	2	1	1	1	Circumferential defect. 1 - 0.480".
C-44	1		1		
C-45	1	1			
C-46	2	1	1		
C-48					A few 1-line defects.

Element	Total No. of Defects	No. of Defects less than 1/4" in any dimension	No. of Defects between 1/4" & 1/2" in any dimension	No. of Defects greater than 1/2" in any dimension	Comments
C-49					A few 1-line defects.
C-53	7	2	5		Many 1-line defects.
C-57	8	3	4	1	
C-58	1	1			
C-59	5	1	3	1	
C-60	3	1	2		
C-64	8	7	1		
C-65	1	1			
C-66	5	4		1	
C-68	12	8	3	1	
C-69	> 10	Many	7	3	Circumferential defect. 1 - 1.3".
C-70	6	4	1	1	Circumferential defect. 1 - 0.850
C-71	1	1			
C-72	4	4			
C-75	6	3	1	2	Circumferential defect. 1 - 1.0" and 1 - 1.1".
C-76	4	2	1	1	
C-78	11	3	6	2	
C-79	3	2	1		
C-80	> 11	Many	8	3	Circumferential defect. 1 - 1.000".
C-87	4	2		2	

APPENDIX VI

Details of Heat Transfer Calculations

1. a) The heat transfer area of a fuel element is 1313 in^2 or 9.12 ft^2.

   b) The individual fuel tubes have surface areas of 338.6 in^2, 437.7 in^2 and 536.7 in^2.

   c) The heat transfer areas heating the coolant channels are 155.2 in^2, 388.2 in^2, 487.1 in^2 and 282.5 in^2.

   d) 9.12 ft^2 per fuel element x (69 + 0.6 x 9) fuel elements = 680 ft^2 of heat transfer area.

2. The breakdown of neutron flux peaking factors are:

   a) Neutron Flux - peak to average radial.
   A value of 1.3 is given in WTR-21 and is consistent with the coolant temperature rise as measured in the instrumented fuel element. Other distributions indicate values as high as 1.45. The conservative value of 1.5 is used here for a hot spot factor to determine the maximum heat flux for comparison with burnout heat flux.

   b) Neutron Flux - peak to average axial.
   The axial peak to average of the distribution contained in WTR-21 is 1.76. This distribution is for the early cold critical position of the control rods and the 1.76 factor is greater than would be obtained for the control rods banked at 60%.

   c) Fuel distribution in fuel element (from WTR-21)

Fuel alloy area	1.02
Fuel alloy density	1.03

Local Power Peaking

Fuel element peak to average power distribution across fuel element	1.04

3. The coolant flow area of a fuel element is 2.30 in^2 or 0.01597 ft^2. The individual coolant channels have flow areas of 0.378 in^2, 0.507 in^2, 0.632 in^2, and 0.786 in^2.

4. Average coolant velocity through the core is:

$$\text{Velocity} = \frac{3500}{78 \times 60} \frac{\text{gal.}}{\text{sec.-fuel element}} \times \frac{\text{ft.}^3}{7.48 \text{ gal.}} \times \frac{\text{fuel element}}{0.01597 \text{ ft}^2 \text{ flow area}}$$

$$= 6.26 \text{ ft/sec.}$$

5. Coolant velocity through the instrumented channels of the fuel element in lattice position L-5-6.

$$\frac{388 \text{ in}^2 \text{ of Heat Transfer Area} \times 191{,}000 \text{ BTU/hr.ft.}^2 \times (\text{Radial HCF})}{1 \text{ BTU/lb } ^\circ\text{F} \times 61 \text{ lb/ft}^3 \times 0.508 \text{ in}^2 \text{ Flow} \times 3600 \text{ sec/hr} \times 141^\circ\text{F } \Delta T}$$

$$= 4.71 \text{ (HCF)}$$

$$4.71 \times 1.3 = 6.12 \text{ ft/sec}$$

$$4.71 \times 1.5 = 7.06 \text{ ft/sec}$$

The instrumented fuel element was in lattice position L-5-6. This position is identical to the position L-6-5, the location of the failed fuel element. Both fuel elements contained 200 gpm of U-235 and were surrounded by equivalent fuel elements.

6. An expression for the difference between the heat transfer surface temperature and the saturation temperature of the coolant, $(T_{wall} - T_{sat})$, for conditions of local boiling is given by Jens and Lottes in ANL-4627.

$$T_{wall} = \frac{1.9 \, \varphi^{.25}}{\exp(P/900)} + T_{sat}$$

$$= \frac{1.9 \, (28.3)}{1.13} + 336^\circ\text{F} = 47.6 + 336$$

$$= 384^\circ\text{F}$$

φ = heat flux in BTU/hr - ft^2

P = absolute pressure - psia

7. Heat flux at burnout has been correlated by S. Mirshak, et. al. and reported in DP-355.

$Q/A = 266{,}000\ (1 + 0.0365V)\ \ (1 + 0.00914\ T_s)\ \ (1 + 0.0131\ P)$

Q/A = heat flux in pcu/hr-ft^2

V = velocity of coolant in ft/sec. = 6.12 ft/sec.

T_s = sub-cooling in degree C ($T_{sat} - T_{bulk}$) = 40°C

P = pressure in psia = 112 psia

T_{sat} is saturation temperature of the coolant

T_{bulk} is bulk coolant temperature.

The temperature rise across the failed element is arbitrarily assumed to be 1.1 times the temperature rise across the instrumented fuel element. This value conservatively covers the hot channel factors. The conservative channel discharge temperature, T_{out}, is used rather than the bulk temperature at the region of maximum heat flux.

$T_{out} = 108°F + 1.1\ (141°F) = 263°F$

$T_s = 336°F - 263°F = 73°F = 40.5°C$

$Q/A = 266{,}000\ (1.223)\ \ (1.370)\ \ (2.457) = 1.10 \times 10^6$ pcu/hr-ft^2

$= 1.98 \times 10^6$ BTU/hr-ft^2

8. Burnout heat flux on the basis of the Jens and Lottes correlation is given below. This correlation was used in the original heat transfer work submitted to the AEC in WTR-25.

$$Q/A \times 10^{-6} = .817\left(\frac{G}{10^6}\right)^{.16} (T_{sat} - T_{bulk})^{.22}$$

$$= .817\ (1.052)\ \ (2.57) = 2.21$$

$Q/A = 2.21 \times 10^6$ BTU/hr.ft^2

where:

G = coolant mass flow rate - lb/hr-ft^2

APPENDIX VII

Health Physics and Other Problems Associated with the Incident

1. Radiation Surveys and Levels

Following the incident continuous monitoring of the radiation levels throughout the plant and vicinity was instituted. Initial radiation monitoring performed at 9:00 p.m. on April 3 shortly after the incident, gave the following levels:

Location	Radiation Level - $\beta\gamma$
Health Physics Office	2 mr/hr
Counting Room	5 mr/hr
Hallway Outside H.P. Office	5 mr/hr
South side of Reactor Service Building	50 mr/hr
Back of Hot Cells	200 mr/hr
2 meters from Head Tank Monitor (located in Process Building)	5 r/hr
Reactor Control Room	40 mr/hr
Main Gate to Exclusion Area	200 mr/hr
East Fence Exclusion Area	500 mr/hr
PC Pump Room	10 r/hr
Head Tank at Fence	4 r/hr
Head Tank Downcomer	40 r/hr
Film Badge Rack	2 mr/hr
Reactor Top	1 r/hr

In general, the radiation levels varied widely, because of shielding provided by various portions of the buildings on the site. The main source of activity was the primary system head tank. Additional surveys taken at 9:00 a.m. on the following day gave the following exposure levels:

Location	Radiation Level - βγ
Machine Shop	35 mr/hr
Boiler Room	35 mr/hr
Ventilation Room	60 mr/hr
Reactor Annex	40 mr/hr
Reactor Top	35 mr/hr to 4 r/hr
General Area at Foot of Head Tank	750 mr/hr
Primary Coolant Pump Rooms at Door	200 mr/hr
Pumps	Up to 5 r/hr at 6"

Subsequent periodic radiation surveys showed these levels to be decaying with an apparent decay time of $T^{-1.2}$.

Immediately following evacuation of the plant three groups with instruments started simultaneously to survey the surrounding countryside. One group drove northward three miles with an air sampler and geiger counter; another to the west a mile, and the third southwest approximately three miles. No detectable radiation levels above background were observed. Readings at the main road, 750 feet to the west of the head tank, gave the following radiation levels:

Time	Radiation Level
Immediately following incident	20 mr/hr
9:20 p.m.	6 mr/hr
12:05 a.m. (4/4)	5 mr/hr
1:05 a.m.	3.5 mr/hr

By 9:20 p.m. on April 3, air sampling stations had been established at seven locations within and around the site. By 2:30 a.m. on April 4, radiation levels had decreased to the point that personnel were permitted to return to certain areas of the plant which building shielding rendered inhabitable.

The first indication that a fission break had occurred in the reactor was given by the 27-channel radiation monitoring system, and almost all channels alarmed in rapid succession immediately following the incident. Nineteen of these channels have GM detectors, four are gamma scintillation detectors and four channels monitor the gaseous and particulate activity in the effluent air streams. The actual radiation levels prevalent in the area immediately following the incident varied considerably from those shown by the radiation monitoring system, due to the elevated nature of the radioactive source, and the shielding by various parts of the buildings. The fission product monitor was designed to indicate minute fission breaks in any of the fuel elements. However because certain valving operations had been performed to permit the primary coolant ion exchanger to be isolated from the primary coolant system, activity was not detected by this monitor until twenty minutes after the area monitors had alarmed. Off-gas released from the reactor system from the surge tank to the head tank and thence to the atmosphere was monitored by a Kanne chamber with the signal feeding a linear micro-micro ammeter. Because of the selection of operating scale, the initial burst of activity was not measured because of off-scale readings and some information was lost. Switching of this instrument to a less sensitive scale was delayed because of the high radiation levels surrounding this monitor, and it was difficult to determine whether or not the monitor readings were due to radiation levels internal to the chamber or external. Subsequent calculations showed that external radiation effects were

negligible compared with the radiation levels observed coming from the Kanne chamber itself and that for the known flow conditions, a maximum of 800 curies of radioactive gas consisting primarily of Xenon and Krypton were ultimately released from the head tank vent.

An estimate at this time of the quantity of radioactivity in the water and the head tank and the head tank downcomer showed about 4,000 curies to be present. As the head tank contained about 60,000 gallons as compared with the 120,000 gallons of water in the primary coolant system, a maximum 8,000 curies of activity had been released to the primary coolant apparently by the dissolution of a large fraction of one fuel element. Later calculations indicated that the total activity released to the primary system was about 5,000 curies, and the airborne activity released in the initial burst was 93 curies followed by an additional 168 curies before the blower was shut off. This discharge is well within the permissible 7 day averaged discharge.

An independent survey of environmental activity was undertaken by Nuclear Science and Engineering Corporation who had taken previous surveys of the same areas.

The following samples were taken in and about the Waltz Mill Site. 7 flypaper fallout samples, 9 water samples, 18 mud samples, 6 vegetation samples and 7 smear samples. The flypaper samples were processed for gross beta activity, the water samples for alpha, beta and uranium activity, the mud samples for alpha, beta and uranium activity, and the vegetation samples for gross beta, potassium 40, net beta, and uranium activity, and the smears for gross beta activity. Samples showed no release of activity and the results were compatible with background figures obtained on previous background surveys performed by NSEC during the preceding three years.

2. Exposures and Work Restrictions

Dosimetry problems involved during the system decontamination were of two types. First, in mixed beta gamma fields there was considerable variance between the beta to gamma ratios, varying with distance and because of the widespread sources of the radioactivity. In addition, there was considerable contamination of film badges and dosimeters so that at times it was necessary to cease operations until sufficient self-reading dosimeters could be decontaminated for use. Because of the inability to determine immediately beta exposures, it was necessary to use estimated beta-to-gamma exposure ratios which varied from a factor of 1-1 to 1-5 beta-to-gamma dose rate. Both remote reading and direct reading pencil dosimeters were used in addition to film badges, and in most cases personnel were required to wear at least two pencil dosimeters in addition to their film badge. Considerable operating advantage was obtained because of the one week cycle on film badge processing which provided rapid determinations of exposure in the highly variable beta gamma fields.

Bioassay samples were collected from all personnel present at the time of the incident. Later samples were taken from personnel involved in operations in airborne radioactivity areas or in decontamination operations involving high levels of contamination. All bioassay results were negative.

By April 13 it was apparent that work restrictions would become necessary in order to prevent excessive radiation exposures and the film badge limits set were those of the Federal Code of Regulations, Title 10,

Part 20 and of Regulation 433 of the Commonwealth of Pennsylvania. In general, it was attempted to keep average exposures to gamma radiation to the 300 mr per week level because of inability to determine how long the cleanup would last. Special authorization was given on certain jobs to approach the maximum legal limits. A working limit for pencil dosimeters was set according to the assumed ratios for beta-to gamma exposure, and restriction lists precluding radiation and contamination work were prepared on the basis of pencil exposures and corrected as film badge information was received. Accumulative 7-day pencil record exposures were kept on all personnel and corrected as indicated to conform with film badge results. Our film badge service supplier (Landauer) provided excellent 1-day service of all results which were telephoned to the WTR immediately. On April 28 the first personnel restriction list was issued and these lists were constantly revised. At one time, the list was revised four times in one day and a maximum of twenty-two personnel were on restriction at any one time. Restrictions ran from two days to six weeks Two technical overexposures occurred: One man received 950 mr gamma exposure when authorized to an exposure level of 900 mr. The overexposure was caused by variable radiation fields in the sub-pile room. In this case, the time limit on the job was determined by exposure to the lens of the eye and the employe was wearing his film badge and dosimeters attached to a surgeon's cap. In the second case, the employe was authorized to receive the permissible weekly exposure of 300 mrem gamma and a skin dose of 1800 mrem. He received an excess of 20 mrem again due to the highly variable beta fields.

All working groups in the plant participated in the high level radiation work. Some work was delayed due to the size of the restriction list, (at times as high as 20%). Additional assistance was rendered by the AEC Bettis plant, Westinghouse Atomic Power Department and Atomic Fuel Department, AEC personnel from various national laboratories, and contract and consultant personnel from NSEC, NUMEC, etc., raising the normal plant complement from 118 to over 200 personnel. Analysis of the radiation exposures received by personnel during the first twelve weeks of the incident gives the following results:

*Total Exposure mrem βγ	Number of People
0 - 300	106
300 - 600	39
600 - 900	16
900 - 1200	13
1200 - 1500	13
1500 - 1800	8
1800 - 2100	5

3. Decontamination Problems

Decontamination problems were encountered in many forms and encompassed the cleaning up of the primary loop to the decontamination of personnel.

a. Primary Loop Decontamination

The sawing out of the lower portion of the ruptured fuel element and the subsequent circulation of water spread radioactive debris in the form of metal particles throughout the primary loop.

*Does not include soft beta exposure.

These particles principally gathered in four areas. These were, the bottom of the reactor vessel, the surge tank, at bends in the piping and in the heat exchangers. In the first two instances the only reasonable solution was to physically remove the pieces, as some form of chemical incompatibility usually existed between gaskets, aluminum and stainless steel which prevented the use of chemicals to dissolve the particles.

The following table indicates the materials involved and their relative resistance to the dilute HF and HNO_3 which would be needed to dissolve fission products.

Material	Item	Resistance	
		HF	HNO_3
304-L S.S.	Piping	---	Good
304 S.S.	Valves	Poor	Good
F 304-L S.S.	Fittings	---	Good
316 S.S.	Pressure gauges, various other instruments	Good at 70°	Good
1100 Aluminum	Fuel Elements, weld filler	Poor	Poor
6061 Aluminum	Core plates, shroud tubes, etc.	Poor	Poor
3003 Aluminum	Surge Tank and Seal Tank	Poor	Poor
Phosphor Bronze	Bourdon Tubes in Pressure gauges	Fair	Poor
Asbestos, Teflon-Impregnated	Valve Packing	---	---
Asbestos, Neoprene-bonded	Gaskets	Good (Neoprene)	Depends on Conditions
Teflon	Valve Packing	Good	Good
Glass	Ph. electrodes, Conductivity Cell	Poor	Good

For large particle removal vacuum cleaning arrangements with high velocity air or water were successful. In the case of the reactor bottom head, the shroud tubes were removed, and an industrial vacuum cleaner was inserted in the holes and traversed about the bottom head. A fine mesh screen about 4-inches from the nozzle opening prevented particles from entering the cleaner. The hot nozzle was then dunked in water and pried off. This process was repeated numerous times and particles up to 2-inches long by 1/4-inch in diameter were removed by this method. Later a high velocity water flush through the seven exit holes created by shroud tube removal completely cleaned the bottom head of the tank of any remaining particles.

The surge tank cleanup used a different technique. Many particles lay on the bottom of the tank under about 6-inches of water. By sending a man into the tank and turning off all lights, it was possible to spot individual particles by their glow. The particles could then be picked up by long handled tweezers and placed in a shielded cask.

Particles at pipe bends at present constitute no operating problem as they can be easily shielded. Their removal has therefore been postponed.

The heat exchanger cleanup used chemical methods as these exchangers were made entirely of stainless steel and were impervious to strong

caustic or acid. Repeated rinses with caustic soda to dissolve the aluminum and 6 normal nitric acid to dissolve the fission products brought the level of the heat exchanger room down to workable values. Table 1 in conjunction with Figure 1 indicates the progress made in decontaminating using these chemicals.

For surface decontamination of the head, pipes and other stainless steel parts, water, versene, or nitric acid was used depending on the degree of decontamination. Attempts were made not to use chlorides or fluorides as it was desired to bypass the stress corrosion problem. In general 6 normal nitric acid rubbed on the surface with an absorbent pad would provide a factor of several hundred in decontamination. Extensive use was made of the prior literature and a listing of the pertinent references is made available as Appendix VIII of this report.

TABLE 1

Radiation Survey of Heat Exchanger Room

Time:		10:00 AM	10:00 AM	4:45 PM	3:30 PM	9:00 AM	1:30 PM	10:30 AM	3:00 PM
Date:	5/9	5/10	5/11	5/11	5/12	5/13	5/13	5/15	5/15
Location Number									
1.	5	5	5	5	3	< 1	< 1	< 1	< 1
2.	7	10	5	8	5	30	3	3	3
3.	25	20	17	30	17	30	9	7	6
4.	42-42-100	45-45-110	42-47-150	47-60-250	29-34-70	50-45-80	14-14-32	11-11-22	10-10-17
5.	50	33	35	40	35	50	13	14	11
6.	5R	3.3R	2.3R	2R	600	600	700	430	250
7.	2.2R	1.7R	1.5R	1.6R	1.3R	1.2R	1.5R	1.4R	125
8.	140	140	200	400	80	85	110	75	17
9.	700	700	125	500	450	600	900	470	100
10.	> 5R	> 5R	3.1R	3R	2.4R	2.3R	2.5R	2R	1.7R
11.	> 5R	3.8R	3.2R	3R	2.3R	3.3R	2.7R	2.4R	2.5R
12.	70	100	70	70	86	80	13	13	34
13.	120	90	70	70	32	60	9	12	12
14.	40	20	18	20	10	20	9	4	5
15.	5	6	4	6	4	1-1/2	2	1-1/2	1-1/2
16.	2	3	2	2	2	21	1	1	< 1
17.	250	250	240	250	120	300	200	33	25
18.	500	210	250	240	270	450	850	160	45
19.	60	40	35	50	20	40	50	6	8

All readings are MR, except where indicated.

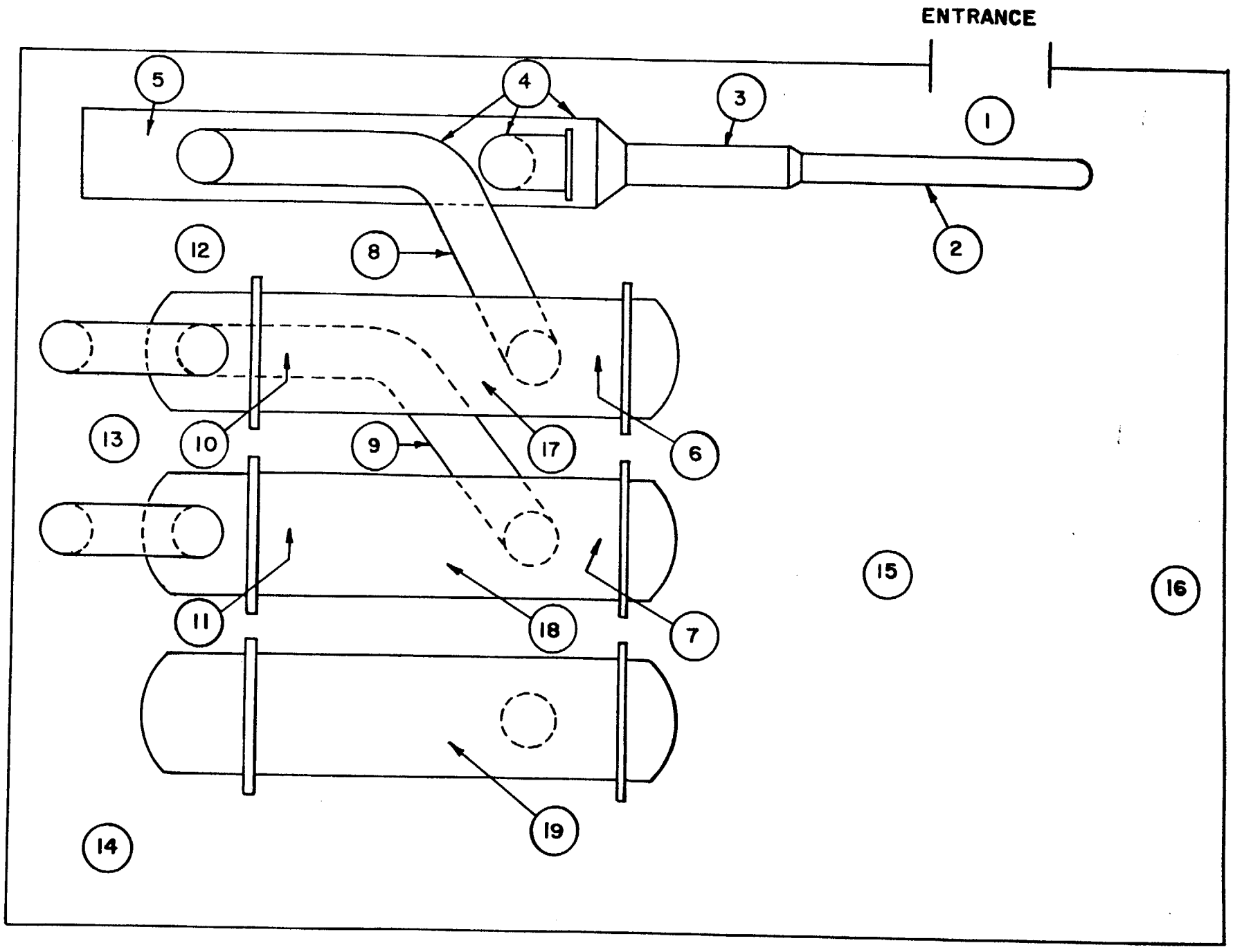


FIGURE I

HEAT EXCHANGER ROOM LAYOUT

b. Building Decontamination

Decontamination operations started at the reactor top which is composed of embossed metal deck plate that had been coated with a good grade of metal paint. Tools and equipment were first removed to a hot storage area and then waste paper and plastic were boxed for shipment. Dry waste material boxed for shipment averaged approximately 250, 4.5 cu. ft. boxes per month during April and May. The metal flooring, stair treads, and hand rails were dry vacuumed and scrubbed. The scrubbing was a hand procedure involving a Versene foam scrub with removal by absorbent pads using a one-wipe per pad surface technique to prevent cross-contamination. For surfaces difficult to decontaminate, chlorinated foaming cleanser (Comet) was used. Smearable contamination was reduced from 50,000 counts per minute beta-gamma to less than 100 counts per minute beta-gamma and the metal surfaces were then recoated with metal paint.

The concrete flooring of the reactor building and reactor building annex were decontaminated by dry vacuuming, followed by a scrub with chlorinated foaming cleanser, citric acid, and muriatic acid (HCl). Smearable contamination levels were reduced from 3,000 counts per minute beta-gamma to approximately 100 counts per minute beta-gamma. The floors were then coated with mastic and covered with a vinyl tile.

Asphalt roads which had become contaminated were decontaminated with a mixture of water and a commercially available Versene complex

solution (Radiac Wash). Areas which were grossly contaminated were vacuumed and then scrubbed. An attempt was made to keep the decontamination solution from flowing onto the nearby mud; where this was not achieved, mud was also drummed. Portions of the roadway where decontamination was ineffective were removed and drummed.

c. Decontamination of Tools

Decontamination of tools was performed with the following techniques: First, hand scrubbed with the following mixture: 20% Dowfax 5% Stephan (LDA), 10% Versene 100, 3% Glycerine, 2% Phosphoric acid. Second, decontamination by absorbent pads with citric and nitric acid was used on stainless steel surfaces. Third, ultrasonic decontamination with the Versene-Dowfax mixture and with citric acid. The last technique proved most useful on hand tools and the second method on large items which could not be introduced into the ultrasonic generator tank. The solution in this tank was changed every 2-3 batches to prevent cross-contamination. The problem of decontamination of tools was magnified perhaps ten-fold by the surplus of tools and equipment in areas which became contaminated during spills or were engulfed by spread of contamination areas.

d. Personnel Contamination and Decontamination Techniques

Many contamination cases resulted from improper fit of face masks and poor handling and removal techniques on the part of personnel. Generally, this was because of a lack of proper mask sizes to fit

all the individuals involved in decontamination work. Contamination of personal shoes resulted from frequent use of the wrong type of shoe covering. The use of light plastic shoe covers for heavy work where severe abrasion was encountered or from improper removal techniques resulted in many contamination problems. In some cases it was necessary for personnel to wear several pairs of shoe covers together with plastic bags, and to change shoe covers when moving from an area of high contamination to one of lower level. A summary of contamination cases encountered follows.

Area Contaminated	Number of Cases
Face, head, and neck	40
Hands and arms	80
Underwear and socks	50
Personal shoes	35

The following personnel decontamination techniques were needed.

Technique	Percentage
Soap and water	50%
Phi-So-Hex	25%
Turco Hand Cleaner	15%
$KMnO_4$ - $NaHSO_4$	10%

Most personnel were decontaminated to levels below background. In approximately 5-10 cases, personnel were sent home with contamination less than twice background. No cases which could not be decontaminated to less than twice background were encountered.

4. Contaminated Water Disposal

The initial capacity of the plant to contain contaminated water consisted of the following facilities: In the retention basins, 240,000

gallons; in the waste disposal tanks, 8,000 gallons; and in the reactor canal, 220,000 gallons for a total of 470,000 gallons. Of this capacity the canal was not available because in its normal operating condition it is filled. The normal plant discharge for waste had averaged 50,000 gallons per day (mostly secondary system blowdown water) of which not more than 500 gallons required reprocessing in the existing evaporator plant. The capacity of this evaporator was at a maximum 3 gallon per minute feed rate and an analysis of the storage needs made on April 12 showed the following potential water inventory on hand:

Location	Volume (gallons)
Retention Basin	240,000
Waste Disposal Tanks	8,000
Reactor Primary Coolant System	120,000
Canal	220,000
Reactor Purges	200,000
Decontamination Chemicals and Washes	50,000
TOTAL	838,000

It was therefore necessary to immediately acquire additional storage capacity and the following techniques were considered: First, the possibility of drumming waste; second, the transport of wastes to a dump site by means of railroad tank cars; third, procurement of a large capacity rubber tank, and fourth, procurement of steel storage tanks. All schemes proved impossible except the last one. For optimum delivery, it was decided to procure 20,000 gallon tanks, of which 20 were on order by April 12. This number subsequently rose to 30.

Larger tanks were considered but they originally could not be delivered in time. Finally, an additional 500,000 gallon tank was procured. The water inventory continued to increase and numerous problems were encountered in keeping water storage capacity abreast of the requirements for the decontamination operation. All available concrete pits and pipe tunnels were utilized for storage. The present contaminated water inventory is about 1,600,000 gallons of water. Signs were placed in all shower rooms and laboratories requesting the assistance of personnel in conserving water as these areas also were of necessity discharged to the waste holdup system. Spills, where they occurred, were collected in 55-gallon drums. Both concrete waste discharge lines to the retention basin were found to have failures in the line through which a considerable amount of in leakage from the water table occurred, thus adding to the disposal problem. To replace these lines, a 3-inch plastic discharge line was installed as a temporary expedient.

It became obvious that additional treatment capacity would be needed to process much waste which was above permissible dumping levels. Disposal by drumming was considered inadvisable due to a cost of well over $2,000,000. Ion exchange and filtration had been considered, but after ion exchange during which a df of 2×10^3 was achieved, the effluent activity still remained at about 10^{-3} μc/ml of mixed fission products. Laboratory tests showed this remaining material to be in the form of colloidal suspension and not easily filterable. Table II, III, IV and V indicate the results of some of the processes that were tried in the laboratory. Chemical separation by flocculation and co-precipitation

looked promising but was not pursued because of the large amount of time and space required for design and fabrication of such a system. Finally, it was decided to process these wastes in a large capacity evaporator and a used unit with a feed capacity of 2,000 gallons per hour was procured. This unit had previously been used for the preparation of evaporated and condensed milk at a creamery in Wisconsin. This system was designed and construction started in two weeks. Installation of the system is underway, and evaporator operation is expected to begin on July 15.

TABLE II

Retention Basin Water

Cleanup Processes

1.	Original sample - unfiltered	3×10^{-3} μc/ml
2.	Original sample - filtered	1.2×10^{-3} μc/ml
3.	Original sample - unfiltered - over silica gel	1.2×10^{-3} μc/ml
4.	Original sample - unfiltered - over steel wool	1.2×10^{-3} μc/ml
5.	Original sample - unfiltered - over silica gel and steel wool	0.70×10^{-3} μc/ml
6.	Original sample - filtered - over silica gel	0.60×10^{-3} μc/ml
7.	Original sample - filtered - over steel wool	0.90×10^{-3} μc/ml
8.	Original sample - filtered - over silica gel and steel wool	0.60×10^{-3} μc/ml
9.	Original sample - filtered - pH = 2.5 - over silica gel	1.7×10^{-3} μc/ml
10.	Original sample - filtered - pH = 2.5 - over steel wool	1.2×10^{-3} μc/ml
11.	Original sample - filtered - pH = 2.5 - over silica gel and steel wool	0.7×10^{-3} μc/ml
12.	Original sample - unfiltered - pH = 2.5 - over silica gel	1.6×10^{-3} μc/ml
13.	Original sample - unfiltered - pH = 2.5 - over steel wool	0.7×10^{-3} μc/ml
14.	Original sample - unfiltered - pH = 2.5 - over silica gel and steel wool	0.9×10^{-3} μc/ml
15.	Original sample - unfiltered - over H^+ - OH^- resin column	0.2×10^{-3} μc/ml
16.	Original sample - unfiltered - over H^+ - OH^- resin column and silica gel	0.15×10^{-3} μc/ml
17.	Original sample - unfiltered - over H^+ - OH^- resin column, silica gel and steel wool	0.18×10^{-3} μc/ml
18.	Original sample - filtered - over H^+ - OH^- resin column	0.11×10^{-3} μc/ml

19.	Original sample - filtered - over H^+ - OH^- resin column and silica gel	0.06×10^{-3} µc/ml
20.	Original sample - filtered - over H^+ - OH^- resin column, silica gel and steel wool	0.12×10^{-3} µc/ml
21.	Original sample - filtered - pH = 2.5 - over H^+ - OH^- resin column	0.01×10^{-3} µc/ml
22.	Original sample - filtered - pH = 2.5 - over H^+ - OH^- resin column and silica gel	0.02×10^{-3} µc/ml
23.	Original sample - filtered - pH = 2.5 - over H^+ - OH^- resin column, silica gel and steel wool	0.01×10^{-3} µc/ml
24.	Original sample - unfiltered - pH = 2.5 - over H^+ - OH^- resin column	0.01×10^{-3} µc/ml
25.	Original sample - unfiltered - pH = 2.5 - over H^+ - OH^- resin column and silica gel	0.01×10^{-3} µc/ml
26.	Original sample - unfiltered - pH = 2.5 - over H^+ - OH^- resin column, silica gel and steel wool	0.01×10^{-3} µc/ml

TABLE III

1.	Original sample	3×10^{-3} µc/ml
2.	Original sample - unfiltered - pH = 2.5 H^+ - OH^- resin (1:2)	1.4×10^{-5} µc/ml
3.	Original sample - unfiltered - pH = 2.5 - H^+ - OH^- resin (1:1)	4.1×10^{-6} µc/ml
4.	Same as (2) above but with flow across column reduced to half	4.05×10^{-6} µc/ml

TABLE IV

Retention Basin Water

1.	Original sample	3.47×10^{-6} µc/ml
2.	Original sample - filtered through a 100 mµ filter	2.30×10^{-6} µc/ml

TABLE IV (Cont'd.)

3.	Original sample - unfiltered - pH = 2.5 - H^+ - OH^- resin (1:2)	9.5×10^{-7} μc/ml
4.	Passing effluent from (3) over second H^+ - OH^- resin (1:2) after adjusting pH to 2.5	9.0×10^{-7} μc/ml
5.	Original sample - filtered - pH = 2.5 - over a double H^+ - OH^- resin column (1:2)	5×10^{-8} μc/ml

TABLE V

Retention Basin Water

1.	Original sample	3×10^{-3} μc/ml
2.	Original sample - filtered through a 100 mμ filter	0.75×10^{-3} μc/ml
3.	Original sample - filtered through #42 Whatman paper	0.70×10^{-3} μc/ml

APPENDIX VIII

DECONTAMINATION - SELECTED REFERENCES

This list includes references to the decontamination of equipment and various plant areas; decontamination of clothing and personnel are not considered.

1. A CONF.15/P/2024 Decontamination of low saline and low activity effluents of radiochemical industries. 11p.

2. APAE-43(Vol. II) Evaluation of chemical agents for nuclear reactor decontamination. Feb. 1959. 204p.
Caustic permanganate-rinse treatment; corrosion and metallurgical results...

3. CRR-836 (AECL-850) Contamination of the NRU in May 1958. 1959. 52 p.

4. HW-54509 Chemical decontamination of the internal surfaces of reactor coolant systems. C. M. Unruh. Mar. 1958. 10p.
A number of chemical reagents prepared especially for reactor plumbing decontamination were evaluated as to corrosiveness, cost, effectiveness, safe disposal. Turco-4306-B, composed of sulfamic acid, fluorides, chlorides, sulfonic base soap and wetting agent, was recommended for decontamination of aluminum and stainless steel surfaces, hand tools, and small parts.

5. HW-54735 Decontamination of aluminum-plutonium alloys. 1958. 4p.

6. HW-56001 Interim report on tests performed at the K reactors to effectively decontaminate the process piping by internal chemical flush. 13p.

7. HW-56001F Internal chemical decontamination of KW, KE, and B reactors, rear face piping, with Turco-4306-B. 1959. 11p.
"...exceedingly safe and effective..."

8. ANL-5002, rev. 2 Removal of slug rupture products from a water-cycling system. July 1953. (Secret Report)

9. CF-54-3-171 Decontamination of stainless steel. Mar. 1954.

10. HW-33710 Decontamination of surfaces - literature survey. 1954. (Secret Report)

11. HW-56865 Decontamination of stainless steel with acid persulfate solutions. July 1958.