ENRICO FERMI ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT

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DISCLAIMER

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PREFACE

PURPOSE

The purpose of this monthly report is to make available to the fast reactor program the current experience being gained from the Enrico Fermi Atomic Power Plant.

SCOPE

The scope of this report includes all phases of current operations and maintenance experience concerning the nuclear portion and related systems of the Enrico Fermi Atomic Power Plant.

Earlier Fermi experience in certain selected areas is being recorded in a series of technical reports, completed or in preparation by Atomic Power Development Associates, Inc. for the U. S. Atomic Energy Commission under Contract No. AT(11-1)-865, Project Agreement 15. This series of reports provides detailed information on the nuclear testing, machinery dome, steam generators, pumps, flowmeters, level detectors, sodium sampling, and development of the primary sodium system.

Items in the sections of this report are selected on the basis of their special significance during the month. Other items may be found in the monthly report submitted to the Atomic Energy Commission by Power Reactor Development Company in compliance with the requirements of provisional Operating License No. DPR-9, as amended.

BACKGROUND

The Fermi reactor achieved initial criticality on August 23, 1963. An extensive series of nuclear tests was conducted at power levels below 1 Mwt, through 1965. A high power (200 Mwt) license was issued on December 17, 1965, and operation in excess of 1 Mwt was initiated on December 29, 1965. In January, 1966, the power was raised in a series of steps to 20 Mwt; on April 1, 1966, power was first raised to 67 Mwt; and on July 8, 1966, operation at 100 Mwt was initiated. On October 5, 1966, fuel damage occurred during an approach to power. Since that time, the reactor has been shut down while the cause and extent of the damage are being investigated.

It is assumed that those reading this report have a general familiarity with the plant. As an aid to the reader, a perspective drawing of the plant was included at the back of APDA-CFE-1. In addition, a topical index appears at the end of APDA-CFE-17.

Since this report is intended to follow closely the current proceedings at the Fermi plant, it must necessarily be treated as preliminary information, subject to supersedence in the light of subsequent experience.

I. CURRENT EXPERIENCE SUMMARY

The chiseling and arc melting concepts are, at present, equally preferred as methods of screw head removal to free the remaining zirconium segments from the conical flow guide in the inlet plenum of the reactor core. A full-length chisel tool is being fabricated and will be tested in an improved mock-up of the reactor inlet plenum to demonstrate the feasibility of the chiseling concept. The development of the arc melting process has been encouraging to date, but additional experiments are required for the selection of optimum electrical conditions and geometry.

One approach being pursued to remove a zirconium segment that is free of the fastening screws is to hook a bottom corner, peel the segment off the conical flow guide and withdraw it through the 14-inch inlet line with a strengthened object-retrieval gripper.

An extensive program of viewing and photographing inside the reactor vessel disclosed no other foreign objects that could cause concern. A number of small metallic shavings were sighted in the inlet plenum and some were removed. The shavings were identified as 6061 aluminum, indicating that they likely resulted from interferences between the lighting and viewing tubes and the walls of the penetrations through the rotating shield plug.

The results of experimental work reasonably confirmed that the zirconium object removed last month could have blocked coolant flow to subassemblies M127 and M098 sufficiently to have caused the melting of fuel in the two subassemblies.

The bellows failures in two valves in the primary sodium service system were found to be localized intergranular attack at the inside surface of the bellows. The cause of the failures is not presently known.

II. INVESTIGATION OF ZIRCONIUM SEGMENTS

A. BACKGROUND

In January 1968, the object discovered in the core inlet plenum was identified to be one of six 0..040-inch-thick zirconium plates originally installed on the sloping sectors of the conical flow guide in the inlet plenum. Visual inspection and photography of the inlet plenum showed that the zirconium liner plate segment was missing from the northeast sector of the conical flow guide and that the zirconium segments were still in place in the other three sectors of the conical flow guide which were accessible for viewing. Two segments remain to be checked. Inasmuch as the object had evidently been torn loose from its installed position, and since it was discovered that a segment fastening screw was missing from another sector, it was decided that the remaining segments should be removed, with some considerations to be given to fixing the segments in place as an alternative measure.

The most feasible concept for removing the remaining zirconium segments involves three distinct operations for each segment

1. Removing the heads of the three zirconium fastening screws holding a segment in place

2. Detaching the zirconium segments from the surface of the conical flow guide

3. Removing the segments through the 14-inch sodium inlet pipe.

A number of concepts for screw head removal were considered, and of these, the milling, chiseling, and arc melting methods were believed to be the most feasible. Initial engineering effort was devoted to investigating the requirements for gaining access to the fastening screws. Preliminary tests were conducted to determine the energy required to chisel off a screw head, but high-speed movies of the tests indicated that test conditions were too conservative; hence, the resulting energy values in the range of 200 to 250 foot-pounds are probably higher than should be reasonably expected. The tests will be repeated under more realistic conditions. Previous information in the investigation of the zirconium segments is given in APDA-CFE-19 and 20.

B. APPROACHES FOR SCREW HEAD REMOVAL

Figure 4 of APDA-CFE-18 shows the zirconium segments installed on the sloping surface of the conical flow guide and the fastening screw-heads that must be removed to free the segments. One screw fastens the top of each triangular-shaped zirconium segment to the conical flow guide and two others fasten the bottom corners. From visual inspection and photography of the four accessible segments, it is known that one of the bottom screws in the southeast sector is missing.* Although the zirconium segment is missing from the northeast sector (the origin of the object removed in March), the three fastening screws are still in place on the cone in this sector.

1. Chiseling

The initial study showed that it is possible to obtain access to all of the fastening screws with a chisel tool inserted through available penetrations in the rotating shield plug and the core support plate. It was determined that despite the limited space in the inlet plenum for articulating a chisel tool, a single-jointed chisel tool could be addressed to each of the fastening screws at a sufficiently low angle of attack (the angle between the axis of the chisel and the plane of the screw head) to shear off the screw head. It appears that the angle of attack for the six upper screws would be in the range of 42° to 51° and for the lower screws, about 39°.

This method for removing screw heads appears to have considerable merit. Applying the chisel to the screw head from 40 feet above requires good viewing and lighting, but the chisel method offers the advantage of repeated tries at shearing off the screw head. No concept has yet been developed for recovery and removal of the severed screw heads but it is felt that this can be done.

In April. the engineering design of an articulated chisel tool was completed and fabrication started. Elements of this tool are the subject of a pending patent disclosure; therefore, illustration and description of the tool are omitted in this report. They will be given in a future report, however. The principle of energy transmission is as follows: A pneumatic system is the source of energy which drives a 1-1/2-inch diameter, 18-inch-long hammer down 18 inches on a solid rod of 38 feet effective length to deliver the chisel impact at the screw head. The bottom 2 feet of the chisel tool may be articulated up to 10° by rotation of an upper shaft; this is accomplished at a ball joint by action of mating eccentric face plates.

A full-length chisel tool of the same design but without the articulation feature is also being fabricated for initial mock-up testing to

* See APDA-CFE-18, Page 11

demonstrate feasibility. This tool incorporates the energy transmission linkage of the articulated tool but without the articulated joint. It is expected that this tool, being less complex, will be available much sooner than the articulated tool and can be used at an early date to demonstrate its feasibility.

2. Chiseling Tests

Tests to determine the energy requirements for chiseling off a screw head were repeated in April. Test conditions and procedures were much the same as those in the tests conducted last month* except that the support structure holding the zirconium plate and fastening screws was made more rigid to more closely simulate the rigidity of the conical flow guide in the reactor inlet plenum. Results of the March tests (200 to 250 foot-pounds) are believed to have been too high because of the significant deflection of the support structure at the chisel impact, as observed in high-speed movies of the tests. In the series of repeat tests, the chisel was applied to the screw heads at a 45° attack angle. Results indicate that about 100 foot-pounds of energy delivered to the chisel at impact will shear off a screw head in one blow.

3. Arc Melting

The arc melting method is another approach for removal of screw heads that has shown promise. Results of feasibility tests of this concept have been very encouraging. It is, therefore, intended to proceed with the development of this concept in parallel with efforts on the chisel method until a preference can be established, based upon test results.

In the arc melting process, the intense heat generated by an electric current arcing between an electrode and the screw head melts the screw head and blasts the debris aside. A 1/4-inch diameter, 3/4-inch-long thoriated tungsten pin imbedded in and extending from a copper terminal bolt has been the electrode used in early feasibility tests. The electrode is brought into light contact with the screw head and a circuit switch closed to start current flow. The heat generated at the light electrode contact starts melting the screw head, breaking the contact, and in turn producing the arc which demolishes the screw head. A timing device opens the circuit to terminate the arc. In the experimental work to date, 1500 to 2000 amps of current flow at 25 and at 110 volts a-c for 1/3 second has been effective in blasting away the screw head. It appears that the application of high power and minimum time of current flow are important considerations for melting away the screw head while minimizing the quantity of debris.

* APDA-CFE-20, Section III. B

It is felt that the feasibility of arc melting as a method of screw head removal has been satisfactorily demonstrated in open air but that more experimental work must be done under conditions more nearly simulating the in-reactor atmosphere and temperature. Further effort is required to determine optimum electrical conditions and the feasibility of the concept in the presence of sodium in an inert atmosphere. Methods of minimizing or collecting and removing the arc melting debris are to be investigated.

A tool which may be used to apply the electrode to the screw head is also the subject of a pending patent disclosure. The principle of the tool actuation is as follows: The electrode cable is carried in a nonarticulated tube and is trunnion-mounted near the bottom end so that when the cable is advanced the electrode may be pivoted to approach the screw head coaxially. It would be possible to add an articulating arm feature if needed to afford greater flexibility for positioning the electrode. The engineering design of the trunnion tool is presently underway.

4. Milling

Milling is another method for screw head removal that was considered worthy of investigation. In the test of one concept, a ball-shaped file of about the same diameter as the screw head successfully cut away a screw head when operated at 30,000 rpm. The resulting chip size was exceedingly small and, therefore, probably would cause no concern if left in the reactor. Another milling concept is a trepanning cutter which, operating in the manner of a cookie cutter, cuts into the zirconium plate around the screw head to free the plate from the fastening screw. A disadvantage evident in initial tests was the high axial force (300 pounds) required to cut through the zirconium plate around the screw head.

In view of the more advanced progress and promise in the chiseling and arc melting concepts, the milling method was assigned lower priority so that more effort could be concentrated on these two concepts.

C. FIXING ZIRCONIUM SEGMENTS IN PLACE

As an alternative to removing the remaining zirconium segments, the possibility of fixing the segments in place was considered. The feasibility of fixing in place was demonstrated in the use of an explosive propellant device to nail a zirconium plate securely to a 6-inch stainless steel pipe of 0.432-inch wall thickness. The device, which employs a cartridged powder charge as the propellant and fires a 3/16-inch diameter, 3/4-inch-long, bullet-nosed fastening pin, is commonly used to fasten construction materials.

No further investigation of fixing segments in place is planned at the present time because of the favorable progress made in developing concepts for segment removal.

D. APPROACHES FOR DETACHING ZIRCONIUM SEGMENTS

After the screw heads are removed, it is not likely that the zirconium segments will be free enough of the conical flow guide so that they can be immediately handled for removal in the same manner as the object removed last month. If the screw heads are chiseled off, metal peening will undoubtedly offer restraint to the segment; the arc melting process might leave some restraining debris. The present plan, therefore, is to have tools available to dislodge the segments from the surface of the conical flow guide. The concepts of two tools to serve this purpose are presently being investigated to determine their feasibility.

One tool is a shover device wherein a serrated surface of the tool exerts a downward force on the upper surface of the segment to shove the segment tangentially off the conical flow guide. The other tool is a hook device which would engage in the lower corner of the segment and peel the segment free. The latter device, termed a peeling tool, offers the additional advantage of sufficiently deforming the lower corners of the segments so that they may more easily be withdrawn through the 14-inch inlet pipe (see Sec. II. E). There is sufficient evidence to believe that a small space exists between the segment and the cone at each bottom corner to permit engaging the peeler tool. The conceptual design of these two tools has just been started.

E. PLANS FOR SEGMENT REMOVAL

It is planned to remove the five remaining zirconium segments through the 14-inch sodium inlet line of the No. 1 primary loop in the same manner in which the object was withdrawn last month. It is expected that the spine-type manipulator and the object-retrieval device will be used for this removal operation.*

A feasibility study indicates that it should be possible to withdraw the segments through the inlet line without difficulty. Although, in the worst condition, a 0.036-inch -wide interference could exist at the 10-inch ID shield ring in the inlet plenum nozzle (the most restrictive section of the inlet line), the use of the peeler tool to deform the segment and reasonable care in orienting the segment should ensure its unobstructed passage through the inlet line. In addition, it is planned to add strengthening features to the

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* See APDA-CFE-20, Sec. II

retrieval gripper to provide for a firmer grasp of each segment and for better control of orientation of each segment. The additional features planned for the retrieval gripper are presently in the early stages of conceptual design; they will be described in a future report.

To obtain a more firm grasp of the entire segment, alternate conceptual designs were made of a bridle to hold a segment during removal. These concepts employ cage and basket devices, in lieu of the retrieval gripper, which are controlled by cables through the object-retrieval device. The bridle concept has, however, been assigned a low priority in the program of development of tools and devices.

F. MODIFICATION OF REACTOR VESSEL MOCK-UP

Work was started this month to improve the existing inlet pipe and reactor vessel mock-up* to more closely simulate the conditions that will exist during the intended operations to remove the remaining zirconium segments. The intent is to establish mock-up conditions that are as realistic as possible for the testing of all tools and devices fabricated for the segment removal program as complete systems. The most significant improvements to the existing mock-up are as follows:

1. The remaining half will be added to the mock-up so that lighting and viewing conditions in the inlet plenum of the reactor core may be simulated.

2. A stainless steel conical flow guide with zirconium segments fastened by screws in five sectors will be installed. The complete screw head removal and segment dislodging operations can then be tried out.

3. An elevated steel platform will be erected and features added to simulate the top of the rotating shield plug, accessible penetrations for tools and devices, and the obstructing upper structures of reactor mechanisms and other plug-top mechanisms.

G. ASSIGNMENT OF APDA ENGINEERING DESIGN GROUP AT FERMI

In April, the APDA Engineering Design Group, assigned to the zirconium segment removal program, was assembled at the Enrico Fermi Atomic Power Plant. This measure was undertaken to improve communications with plant operating and maintenance personnel and to afford engineering design personnel the opportunity to have direct access to the physical environment during design and mock-up testing.

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* See page 15 of APDA-CFE-14 and Figure 17 of APDA-CFE-16

III. INSPECTION OF INTERIOR OF REACTOR VESSEL

Immediately after the removal of the first zirconium segment (the object) from the reactor vessel, a thorough borescope inspection of the interior of the reactor vessel was carried out in a search for other loose objects and, generally, to look for any abnormal conditions. There was extensive viewing in the upper reactor compartment and in the inlet plenum to the reactor core. A total of 270 photographs was taken, of which those shown in this section are representative. Details of the inspection are given below.

A. VIEWING AND PHOTOGRAPHING THE UPPER REACTOR COMPARTMENT

Figures 1 through 7 are representative of photographs obtained of details in the upper reactor compartment and of the support plate. There were no loose or foreign objects sighted in the upper reactor compartment, nor was there any evidence of damage except the condition of the orifice retainer spring fingers in lattice position NO7-POO. As can be seen in Figure 7, several of the orifice retainer springs in NO7-POO appear to be bent, presumably resulting from insertion of the spherical mirror support post (see inset photograph) in this position.* Viewing revealed no evidence of erosion of core subassembly seating surfaces due to sodium leakage around subassembly nozzles. Photographs such as shown in Figure 6 were not sharp enough to confirm this, however.

The holddown and sweep** mechanisms and the OHM appeared to be in normal condition; in appearance, all mechanisms were typically light gray with light brown stains. It is intended to remove the sweep mechanism in the near future to permit additional access for tools and devices for removal of the zirconium segment in the north sector. The inspection uncovered no evidence of any condition that would suggest difficulties after reinstallation of the sweep mechanism in the reactor.

Attention was directed to the exposed walls of the lower safety rod guide tube in position NO3-POO and core subassembly M094 in position NO3-PO1 because of apparent discolorations that were observed on each during an inspection in September, 1967. These stains, located at the top of each element, were still in evidence, and similar stains, greenish in color, were sighted at the bottom of each item.

* APDA-CFE-18, 19, 20
** APDA-CFE-4, Page 26

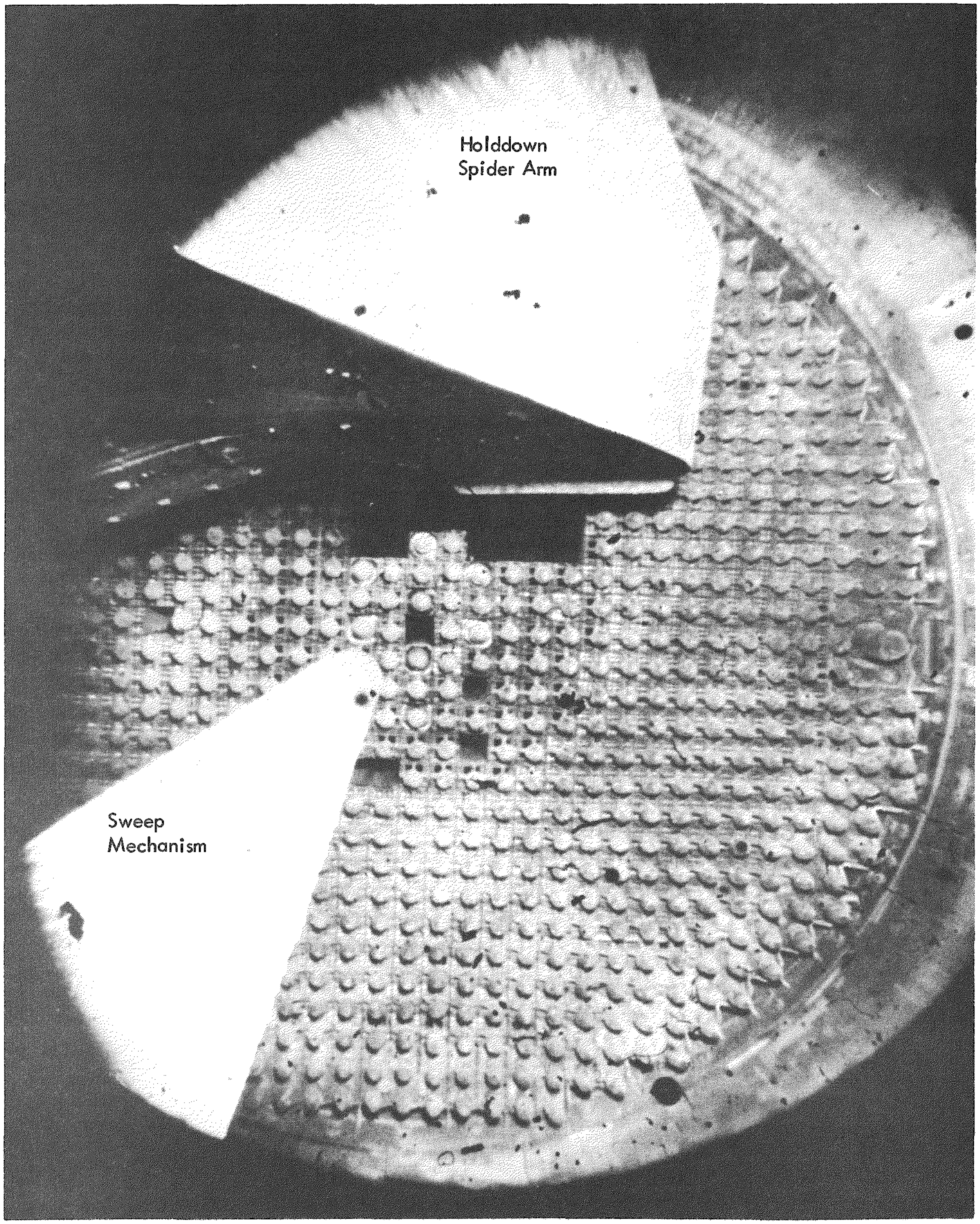

FIG. 1 REACTOR CORE AND BLANKET LATTICE

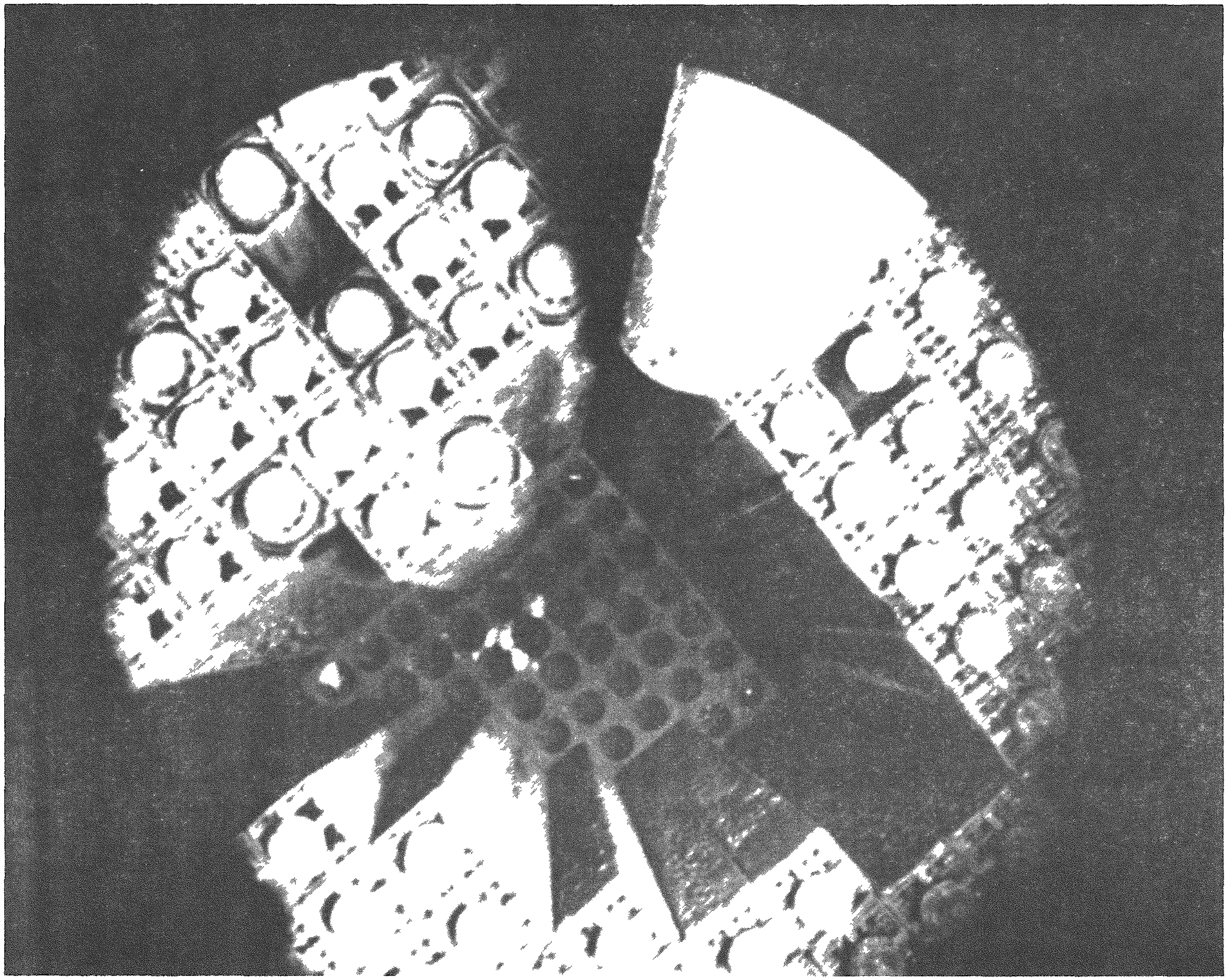

FIG. 2 CANYON IN EAST SIDE OF REACTOR CORE AND BLANKET LATTICE

FIG. 3 UNDERSIDE OF ROTATING SHIELD PLUG NEAR HOLDDOWN AND PENETRATION THROUGH ROTATING PLUG

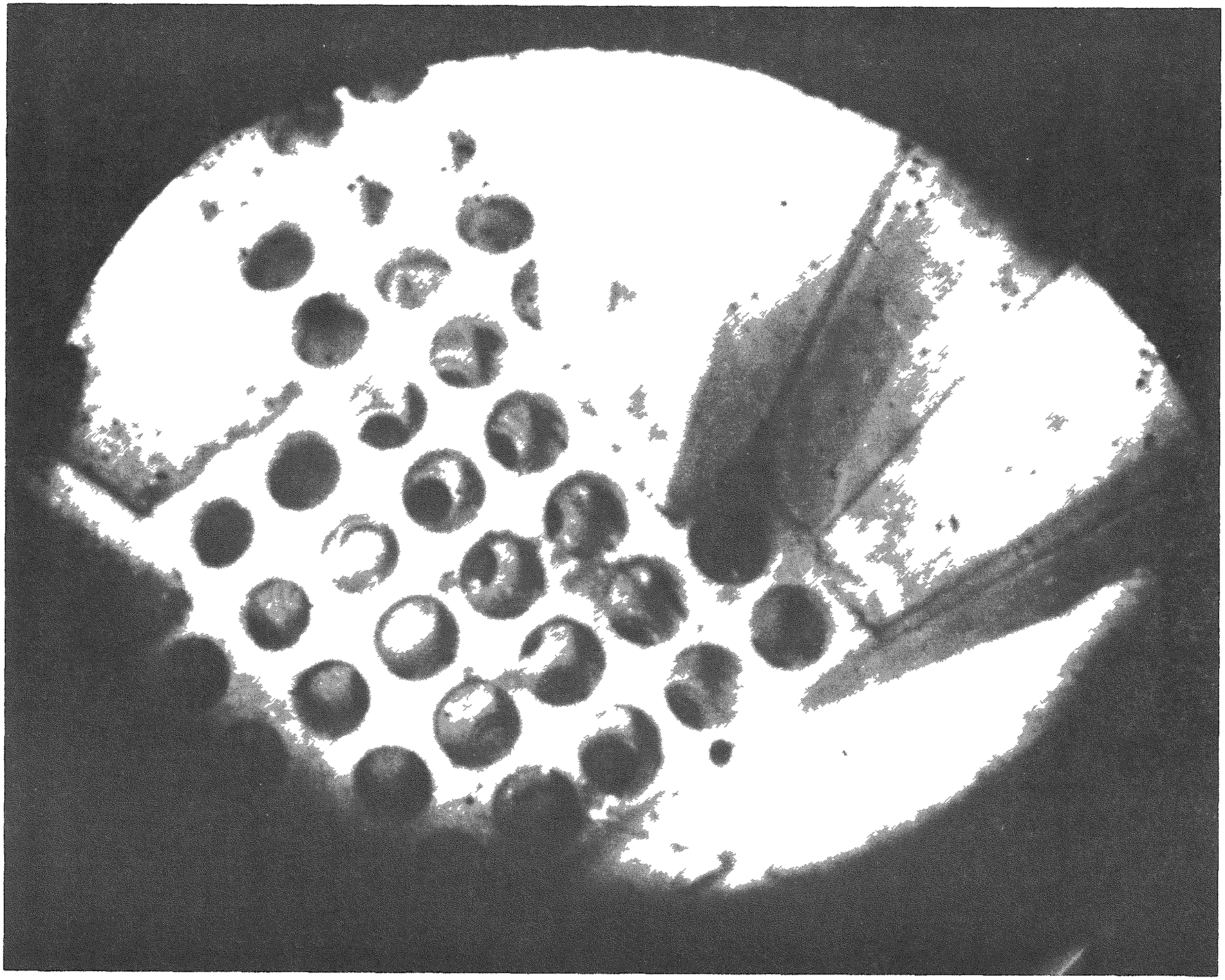

FIG. 4 CLOSE-UP OF SUPPORT PLATE

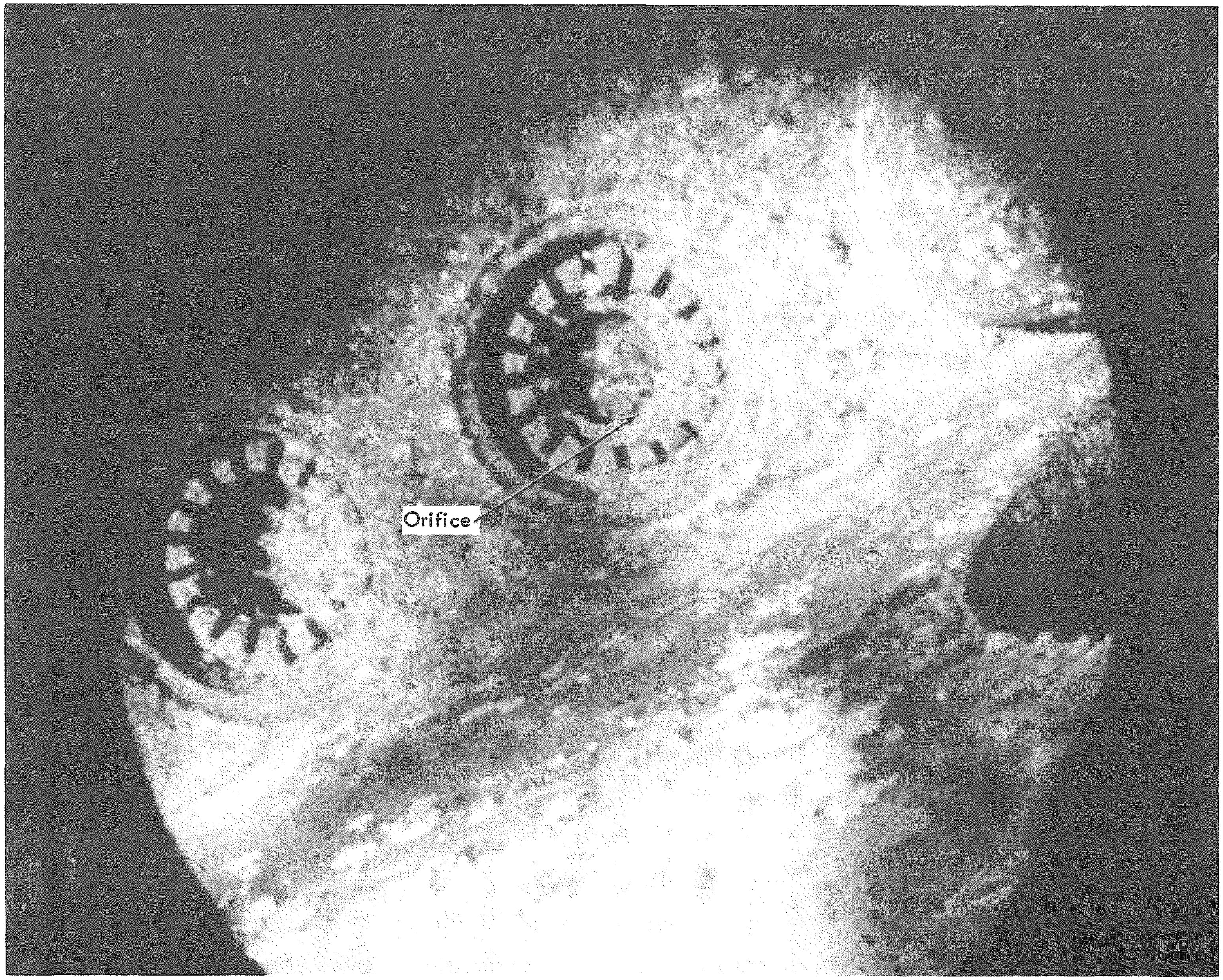

FIG. 5 ORIFICES IN BLANKET POSITIONS IN LOWER SUPPORT PLATE

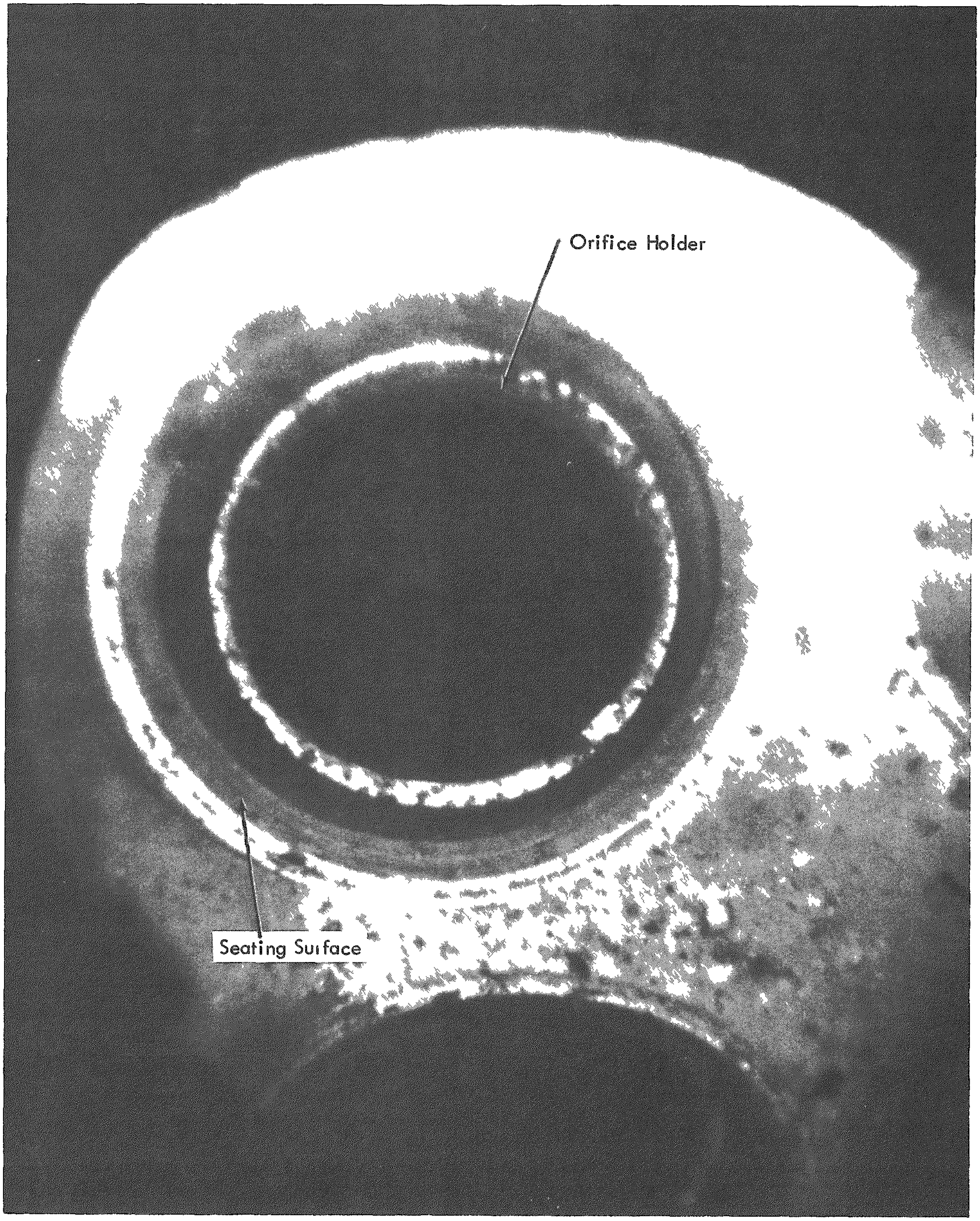

FIG. 6 SEATING SURFACE IN LOWER SUPPORT PLATE FOR CORE SUBASSEMBLY NOZZLE

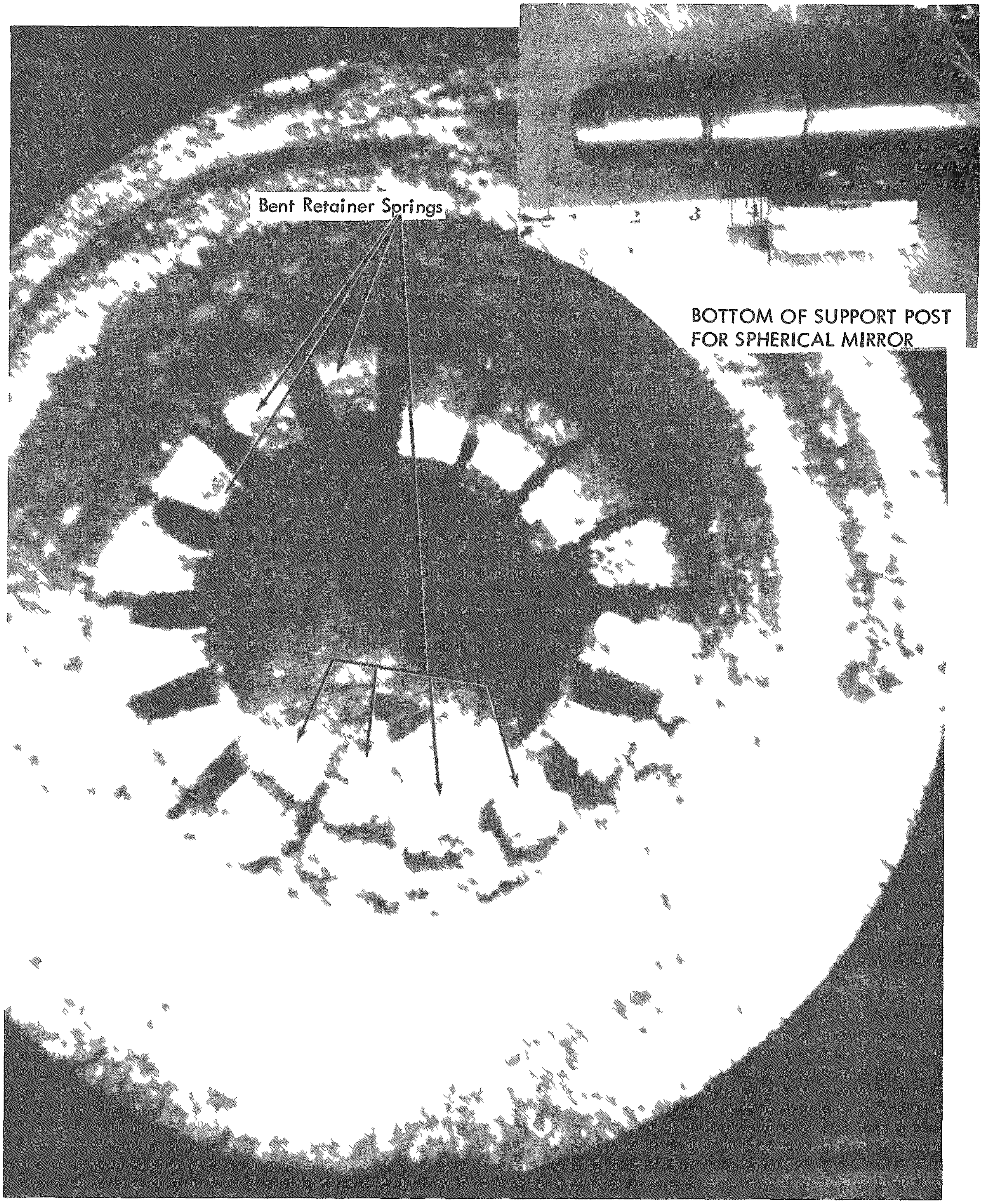

FIG. 7 BENT ORIFICE RETAINER SPRINGS IN BLANKET POSITION N07-P00 IN LOWER SUPPORT PLATE

It was thought possible that photographs of an array of subassembly handling heads could be analyzed to give an indication of gross distortion of any subassemblies. However, photographs taken to date, such as those in Figures 1 and 2, have not been clear enough to provide sufficient resolution to allow lowering of the HDM without a more detailed dimensional inspection of each subassembly.

B. VIEWING AND PHOTOGRAPHING THE CORE INLET PLENUM

1. Zirconium Screws and Acorn Nuts

The borescope inspection in the core inlet plenum confirmed that the zirconium screws fastening the zirconium segments on the conical flow guide were in place in the northeast, northwest, southeast,and southwest sectors. The lone exception was one bottom screw missing from the southwest sector. As had been discovered earlier,* the zirconium segment was missing from the northeast sector but the three screws were still in place. Viewing of the screws in the north and south sectors is still not possible because of interference by the sweep mechanism and core subassemblies remaining in the support plate.

All but two of the acorn nuts fastening the batten strips which cover the joints between sections of the zirconium liner were viewed and found to be properly in place. The positions of the innermost nuts immediately north and south of the center of the plenum were not accessible for viewing because of sweep mechanism and subassembly interference, as indicated in Figure 8.

2. Discovery and Removal of Small UFOs

Detailed borescope inspection and photography in the core inlet plenum indicated that there are no other large, loose objects in this area; however, a number of very small objects (UFOs) were sighted on the floor of the inlet plenum. Some of these were of sufficient size to be removed by the spine-type manipulator used in handling the zirconium object.** Eight of these UFOs, found in locations shown in Figure 8, were identified by number, investigated, and photographed. The zirconium object removed last month was was identified as UFO No. 1. Details of the newly-discovered UFOs are as follows:

- UFOs No. 2, 4, and 7 were curly pieces of metal similar to lathe shavings, about 1/2 inch by 1/4 inch and smaller. UFOs

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* APDA-CFE-18, Page 11
** APDA-CFE-20, Section II

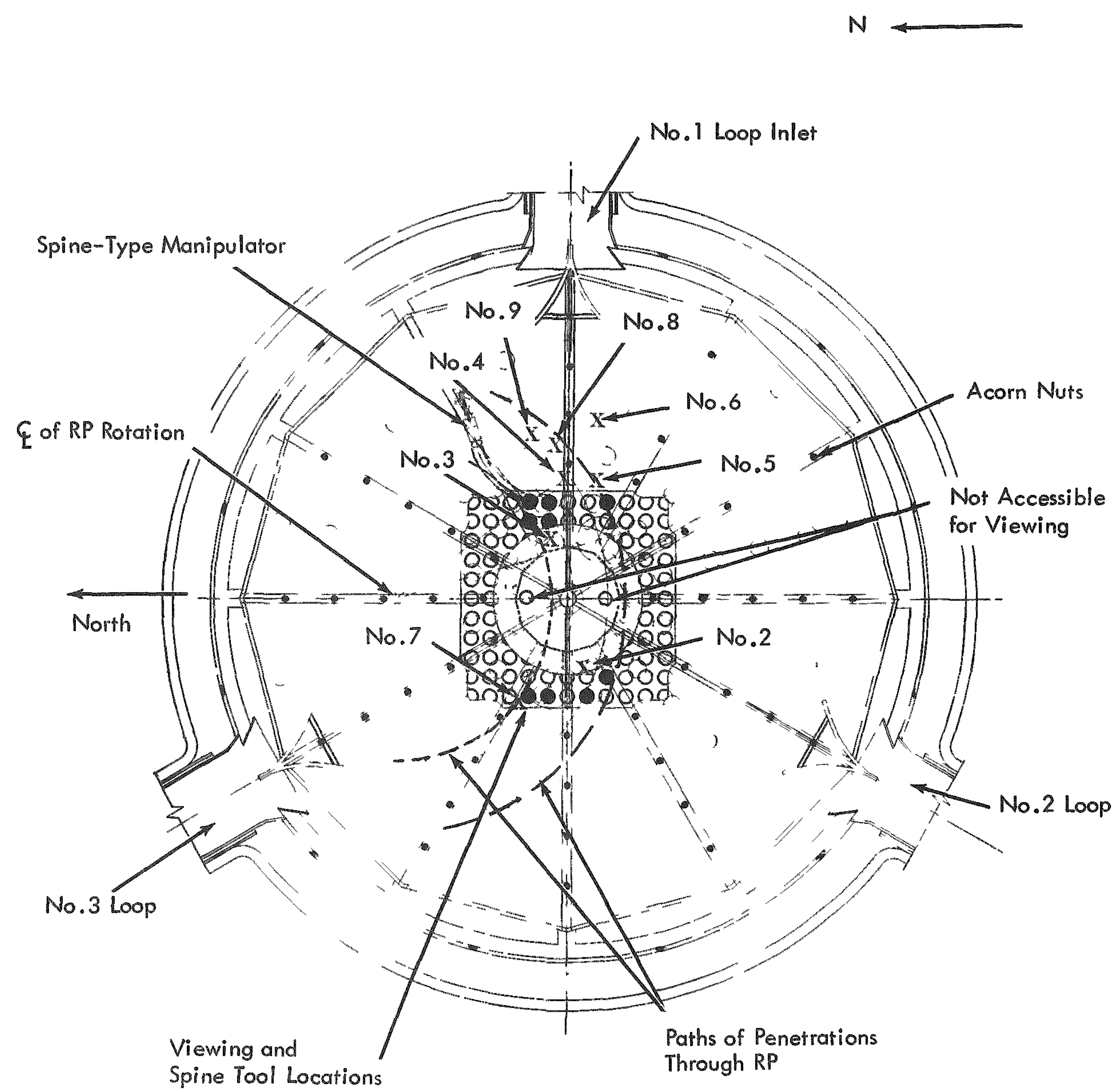

FIG. 8 LOCATIONS OF SMALL UFOs DISCOVERED IN INLET PLENUM

No. 2 (Figure 9) and 4 were removed from the reactor and determined by analysis to be Type 6061 aluminum.

- UFOs No. 3 and 8 were of charred paper texture. Both were removed from the reactor; analysis is as yet incomplete.
- UFO No. 5 initially had the appearance of a 1/4-inch screw, 1/4 inch long. Efforts to remove this UFO were unsuccessful, since it tended to blend into the residual sodium on the floor of the inlet plenum when moved by the spine-type manipulator.
- UFO No. 6 was an object estimated to be less than 1 inch long. It was grasped by the spine-tool gripper (Figure 10) but was lost as the tool was taken out of the reactor.
- UFO No. 9 appeared to be a very small cylindrical object. No attempt was made to remove it.

Other metallic pieces smaller in size than those removed were observed but no attempt was made to remove them. It is felt that these pieces represent no hazard to the future operation of the reactor.

It is interesting to note that, as can be seen in Figure 8, all the UFOs were found in the path of the rotating plug penetrations. From a subsequent inspection of the borescope, it appears likely that the aluminum UFOs resulted from interferences between the Type 6061 aluminum borescope and light tubes and the walls of the rotating shield plug penetrations or the penetration seals.

3. Cleanliness of the Inlet Plenum

It was observed that the south, southwest, and northwest sectors of the inlet plenum were quite clean, (see Figure 11 showing the loop 3 inlet area) except under the core in the southwest and northwest sectors. These latter two areas and most of the east side of the plenum were dirty, as shown in Figure 12, probably due to the extensive removal operations and tool activity. It is believed that the crud in these dirty areas is the result of solidified sodium and sodium oxide dropping from the sides and bottom of the rotating plug and the penetrations through the plug utilized for the recovery operations. Sodium fill, heat-up, and cold-trapping of the primary system, planned for the near future, are expected to restore the former state of sodium purity and primary system cleanliness.

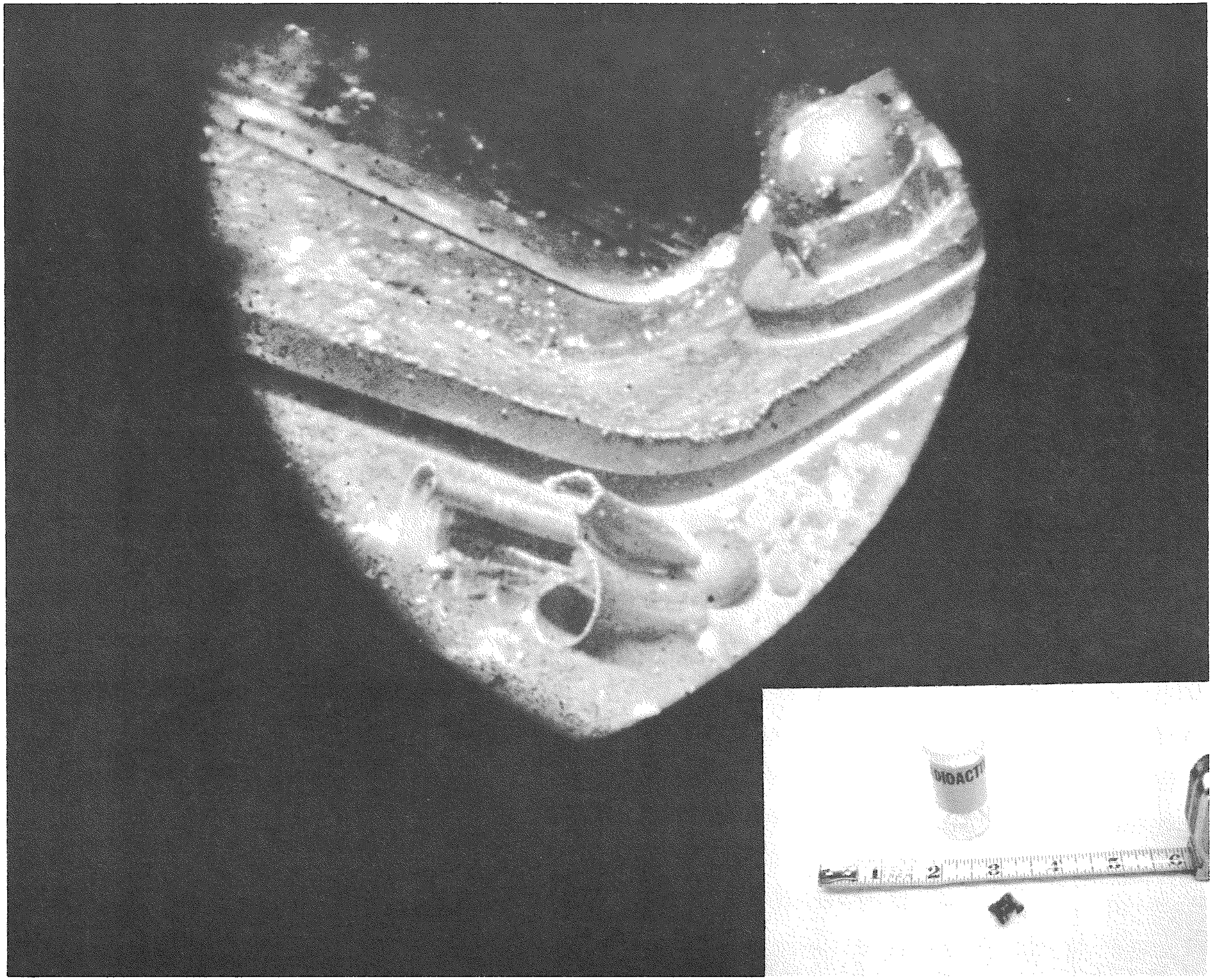

FIG. 9 UFO NO. 2 IN INLET PLENUM

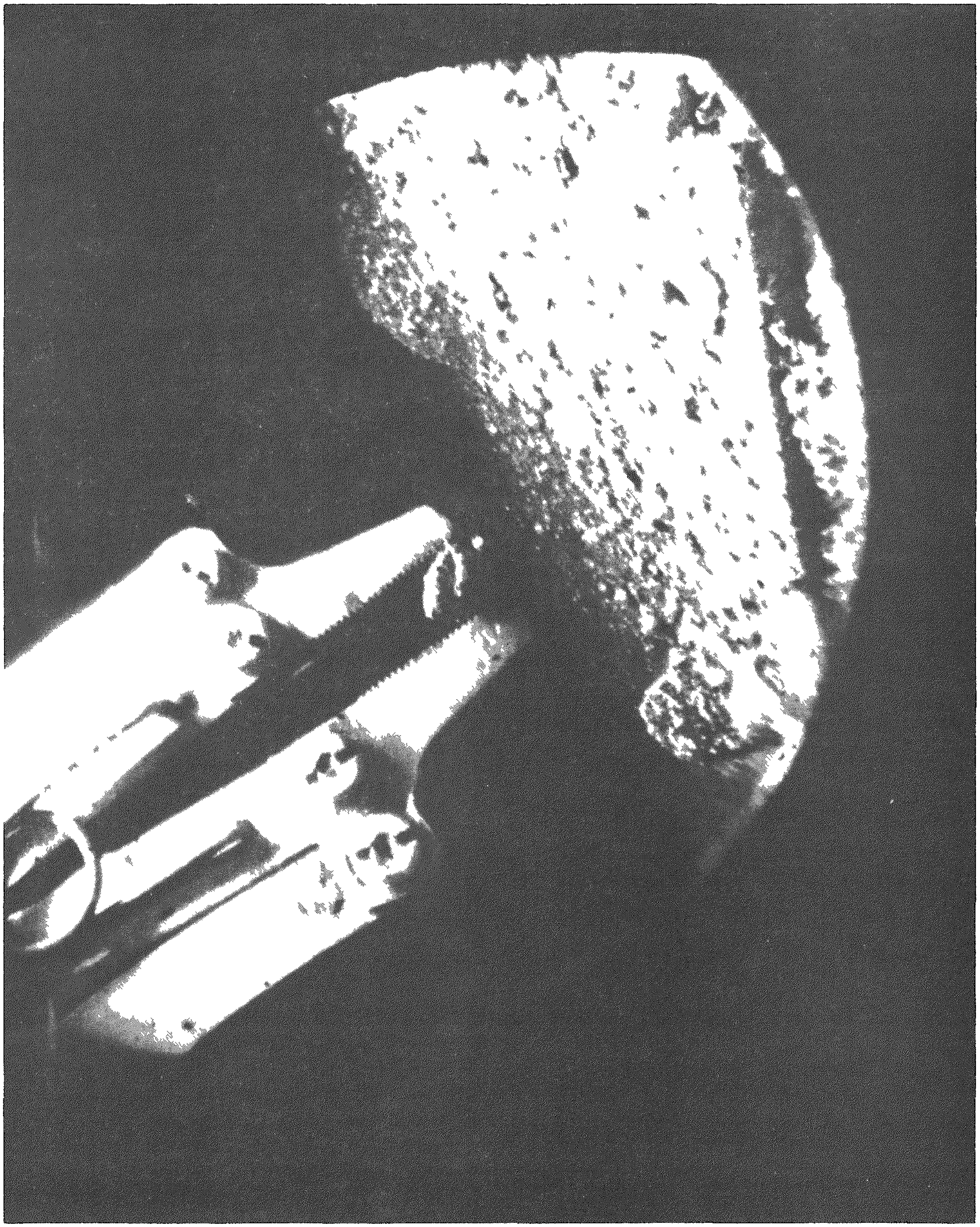

FIG. 10 UFO NO. 6 GRASPED BY SPINE TOOL GRIPPER IN INLET PLENUM

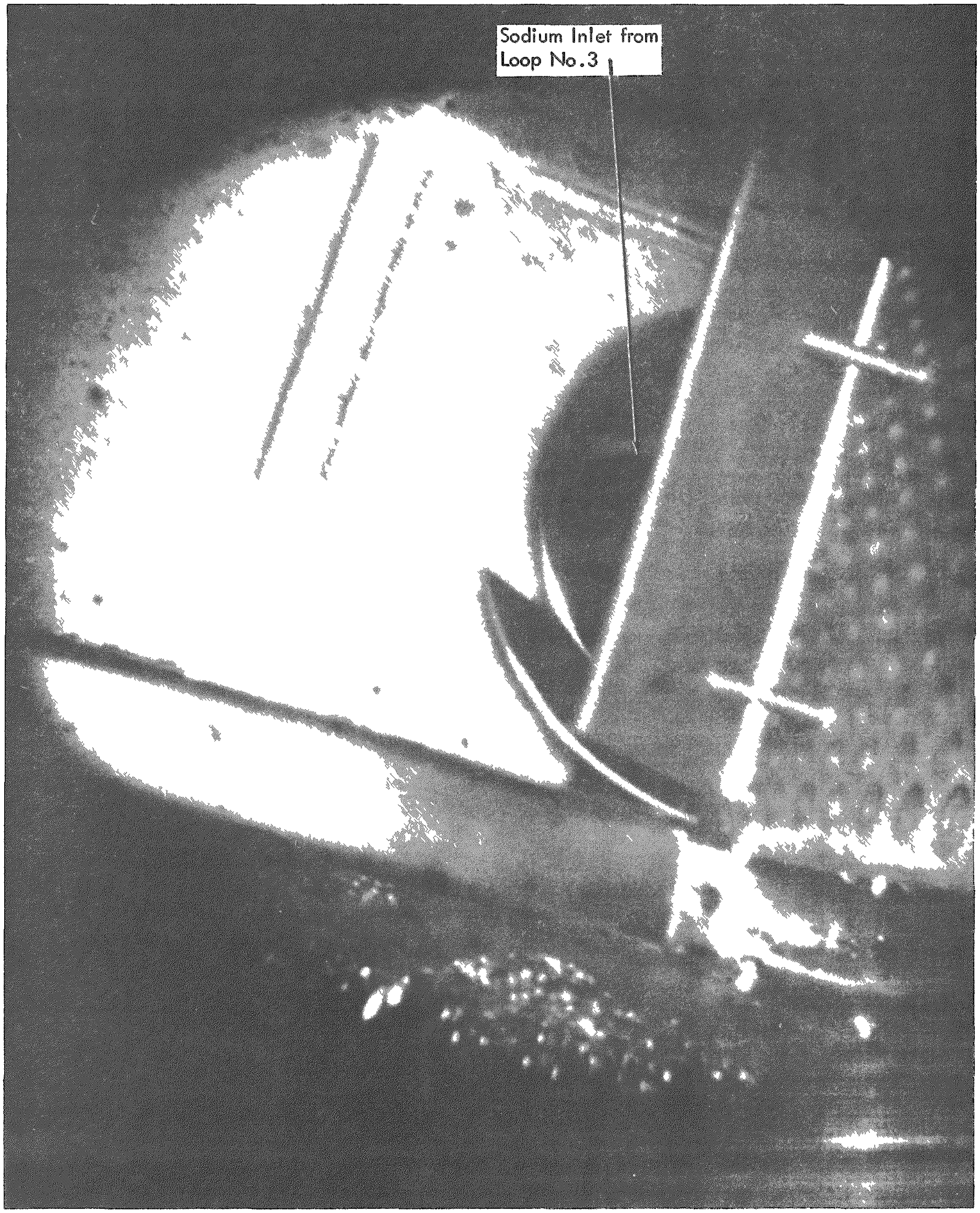

FIG. 11 CLEAN WEST SIDE OF INLET PLENUM

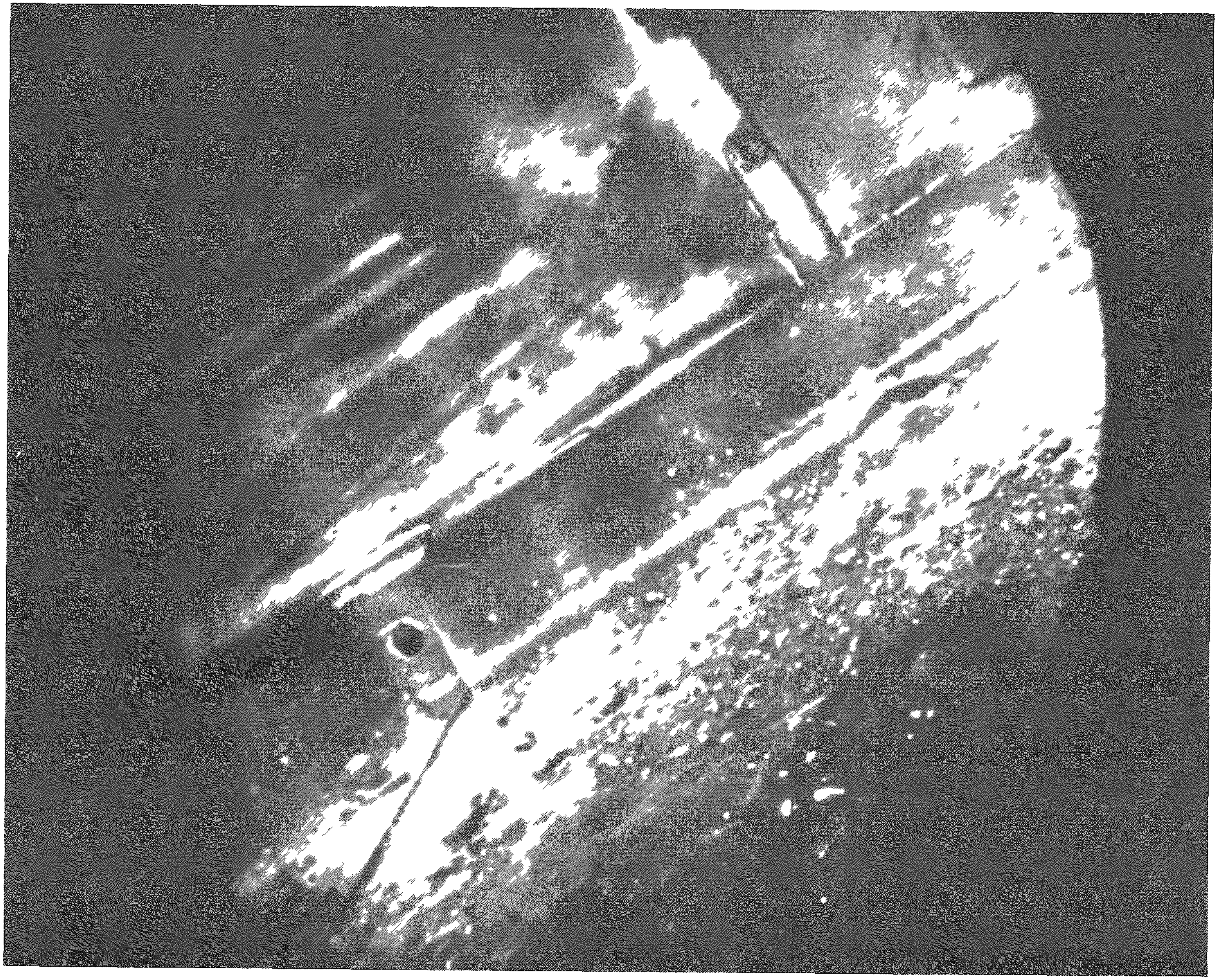

FIG. 12 CRUD IN EAST SIDE OF INLET PLENUM

C. DIFFERENCE BETWEEN VIEWING AND PHOTOGRAPHING

There is usually a notable difference between the image viewed in the reactor through the borescope and that photographed. The eye is able to adjust to bring into proper focus the entire image within the field of view. The camera, on the other hand, is focused on the center of the image, resulting in loss of clarity and distortion at the outer edges of the field.

IV. OPERATIONS

A. REACTOR SUBASSEMBLY MOVEMENTS

As shown in the reactor loading diagram (Figure 13), one core subassembly was moved to a different position in the reactor during April. Core subassembly M082 was moved from PO5-PO1 to NO5-PO4 for the purpose of opening up an additional support plate position for insertion of the spine-type manipulator with the borescope and light source for removal of the small foreign objects sighted on the west side of the inlet plenum.

The offset handling mechanism and the rotating shield plug were operated in the automatic mode for this subassembly move. The intended destination of the subassembly was lattice position NO3-PO5, but after a number of unsuccessful attempts to insert it at that position, the subassembly was deposited in alternate position NO5-PO4. There were only two adjacent surfaces available for camming the subassembly into the intended position NO3-PO5, and, in such a case, it is not unusual to encounter difficulty in inserting a subassembly into a lattice position. Frequently, a subassembly is precammed in another position where more camming surfaces are available to aid in final insertion. No difficulty was experienced in inserting the subassembly into position NO5-PO4.

B. MELTING SODIUM IN THE 14-INCH SODIUM INLET LINE OF LOOP NO. 1

In preparation for the planned sodium fill and clean-up operations in the primary sodium system, the sodium in the 14-inch inlet line of loop No. 1 was melted and brought to a temperature of about 350 F. In Section III.A of APDA-CFE-18, it was shown how the sodium level was lowered in this line back to near the pump discharge and the pipe cooled to provide a reasonable measure of comfort to plant personnel at the top elbow in zirconium object removal operations.

Insulation and induction heaters were reinstalled on the elbow and heating circuits restored to their arrangement prior to the modification for cooling the elbow. Melting of the sodium was accomplished in a sequential manner, beginning in the area of the elbow (the gas space above the sodium level) and working back toward the pump discharge. During the period of sodium melting, pipe heaters were alternately energized and de-energized to provide successive periods of heating and soaking to preclude pipe stress difficulties. After about two days of melting and heat-up, the temperature of the inlet line reached a steady 350 F.

KEY:

CR Control Rod No. 430169-
SR Safety Rod No. 430192-
OR Oscillator Rod
CS Core Shim Subassembly
CF Core Foil Subassembly
BF Blanket Foil Subassembly
CT Coarse Filter, Take-apart, Dummy Core Subassembly
NOTE: Dummy Core subassemblies in the reactor meet "Core A" core subassembly specifications and bear the suffix "CF"

NA Sodium Worth Subassembly
NS Neutron Source
TIT Temporary Instrument Thimble
MS APDA Materials Surveillance Subassembly
M Subassembly Manufactured by D.E. Makepeace Co.
S Subassembly Manufactured by Sylcor Division, Sylvania Electric Products Co.

M 001 - M 206 Core Subassemblies
M 301 - M 400 Inner Radial Blanket Subassemblies
S 500 - S 798 Outer Radial Blanket Subassemblies
M 801 - M 1000 Outer Radial Blanket Subassemblies

Units shown without prefix are dummy outer radial blanket subassemblies.

Oversize Nozzle Unit
F "F" Subassembly (Contains fuel pins with high iron plus nickel, high carbon or zirconium content.)
W "W" Subassembly (Contains fuel pins with high iron plus nickel content.)
Blanket slugs have high carbon content. (APDA Surveillance Program Unit)
Stringering in Blanket Slugs
X Large Grain Blanket Material (Hash)
Larger Than Normal Spacing Between the Blanket Elements and the Support Grid
Θ Type 347 Stainless Steel Wrapper Tube
H Handling Head Short
Test Flow Subassembly (S-400)
Slugs Previously Used in a Test Subassembly
C "CP" Slugs
Locations Where Changes Were Made

Note: Heavy Boxes at Outside of Lattice are Storage Positions

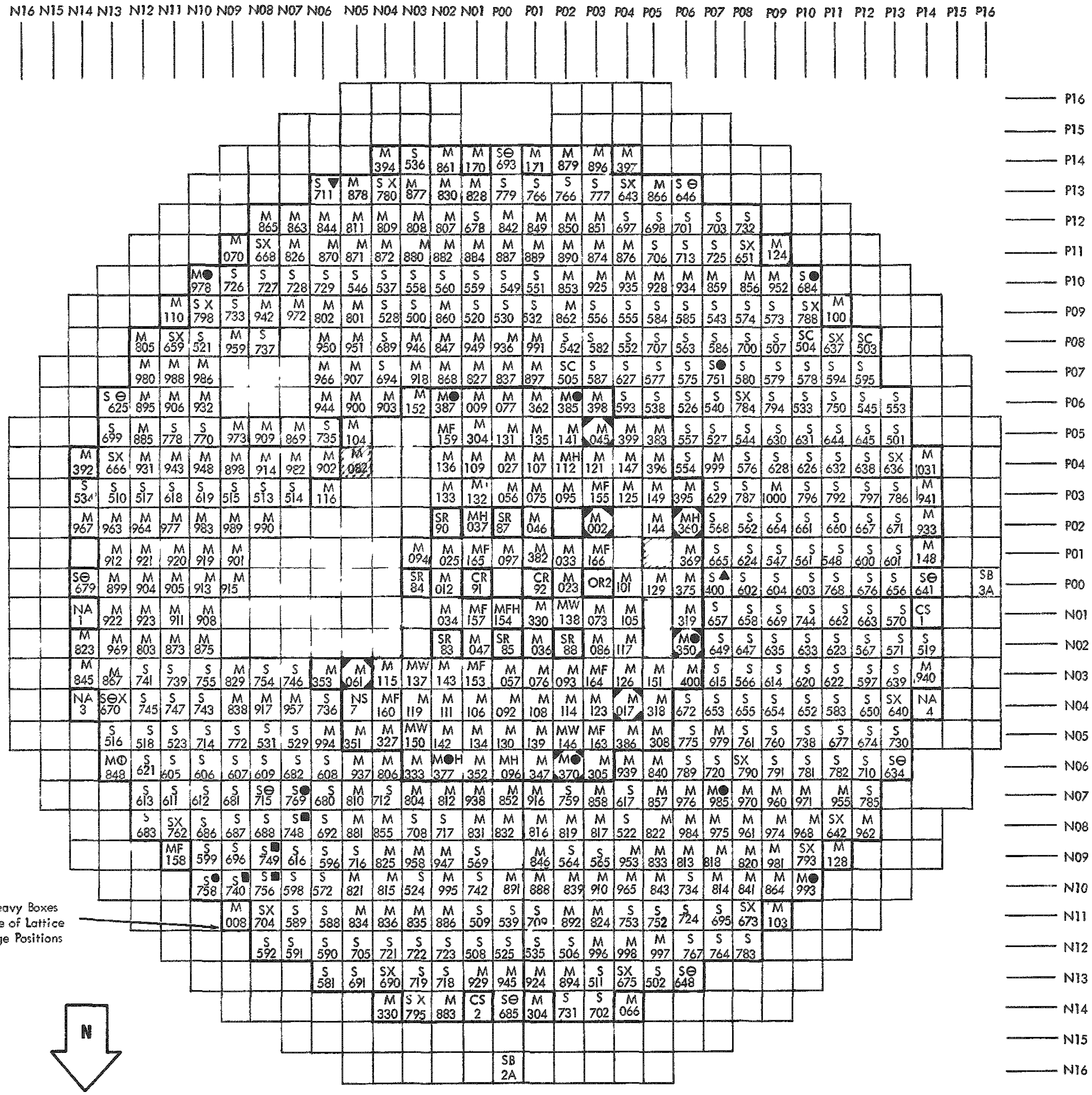

FIG. 13 REACTOR LOADING ON 4-30-68

C. GAS SYSTEMS PERFORMANCE

Since the last data were reported*, the following primary system gas activity and cover gas analyses have been made:

1. Primary System Gas Activity

Location	Sample Date	Gross Beta Concentration, microcuries/cm^3
Reactor Cover Gas	4-5-68	9.9×10^{-7}
Primary Shield Tank	4-11-68	Not Detected
Reactor Cover Gas	4-11-68	2.0×10^{-8}
Primary Shield Tank	4-19-68	3.5×10^{-8}
Reactor Cover Gas	4-19-68	5.2×10^{-6}
Primary Shield Tank	4-25-68	2.5×10^{-8}
Reactor Cover Gas	4-25-68	2.5×10^{-6}

2. Primary System Cover Gas Analysis

	Reactor Cover Gas (Argon), ppm by volume*	Primary Shield Tank Atmosphere (Nitrogen), ppm by volume*
Oxygen	Below 25	80**
Carbon Monoxide	Below 10	Below 10
Carbon Dioxide	25	25
Hydrogen	Below 4***	Below 2.5
Helium	Below 4	Below 4
Methane	Below 10	Below 10
Nitrous Oxide	Not Measured	Not Measured
Argon	Remainder	0.7
Nitrogen	2000	Remainder
Dew Point	Not Measured	-55 F

* Sample dates for reactor cover gas and primary shield tank atmosphere are 4-25-68 and 4-6-68, respectively

** Technical specifications state 1000 ppm maximum

*** 10 ppm is the recommended maximum for reactor operations

* APDA-CFE-20

V. SPECIAL INVESTIGATIONS

A. SUBASSEMBLY FLOW BLOCKAGE TEST

1. Melting of Two Subassemblies by Blockage of Coolant Flow

A recent study indicated that the melting of subassemblies M127 and M098 during the incident of October 5, 1966, would have required almost complete blockage of coolant flow to the subassemblies. The study cited 95% blockage of M098 and 96% blockage of M127 as required conditions to initiate melting at the power levels of operation on that date. An experimental program was therefore undertaken to confirm the liklihood that it was indeed the first zirconium object in its battered shape that caused flow blockage and, hence, meltdown of the subassemblies.

Early photographs of the object in the reactor revealed a flat surface that could have covered a number of support plate holes and prevented coolant flow to the subassemblies. The faint circular score marks discovered on the surface of the object* when it was removed from the reactor added support to the belief that the object had caused flow blockage. A model of the object, fabricated to dimensions determined from analysis of photographs in the inlet plenum, was used in hydraulic flow tests to confirm the flow blockage theory. The actual object also was used following its removal from the reactor.

2. Hydraulic Test Loop Simulating Reactor Flow Conditions

An available hydraulic test loop was adapted to include a windowed chamber which simulated the core inlet plenum and contained a 12-hole mock-up support plate with individual discharge pipes. The model of the object was located at the bottom of the mock-up support plate aligned in positions believed to be practical for preventing flow. Figure 14 is a photograph of the hydraulic loop and equipment; Figure 15 is a schematic of the test loop, and Figure 16 shows the actual object, following its removal from the reactor, against the bottom of the mock-up support plate. Reactor flow conditions of 100 gpm through each support plate opening and a pressure differential of 22 to 23 psi were established as initial test conditions. It was possible to accurately measure the flow through individual channels to determine the degree of blockage.

* See APDA-CFE-20, Section II. D. 4

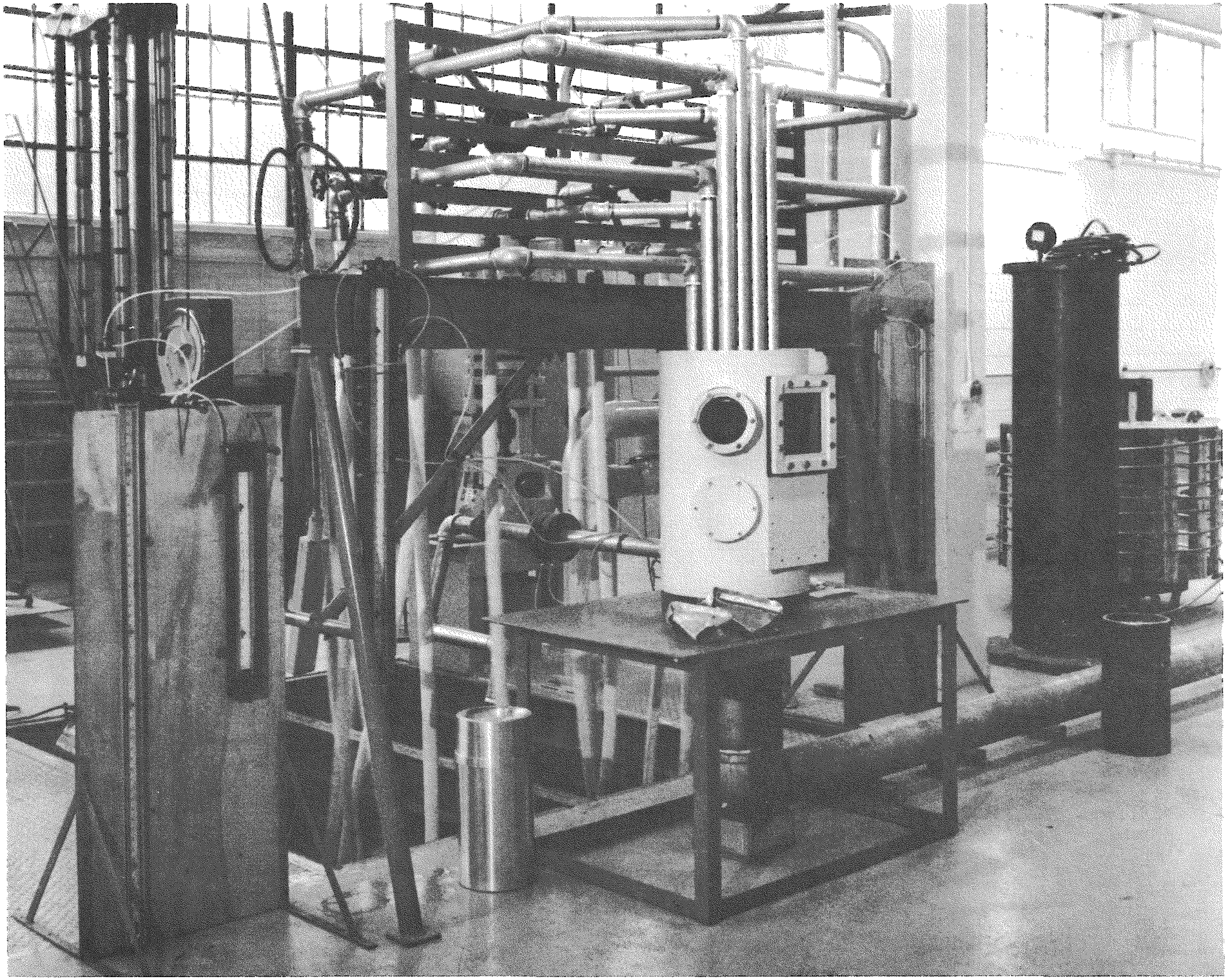

FIG. 14 HYDRAULIC LOOP AND TEST CHAMBER FOR SUBASSEMBLY FLOW BLOCKAGE TESTS

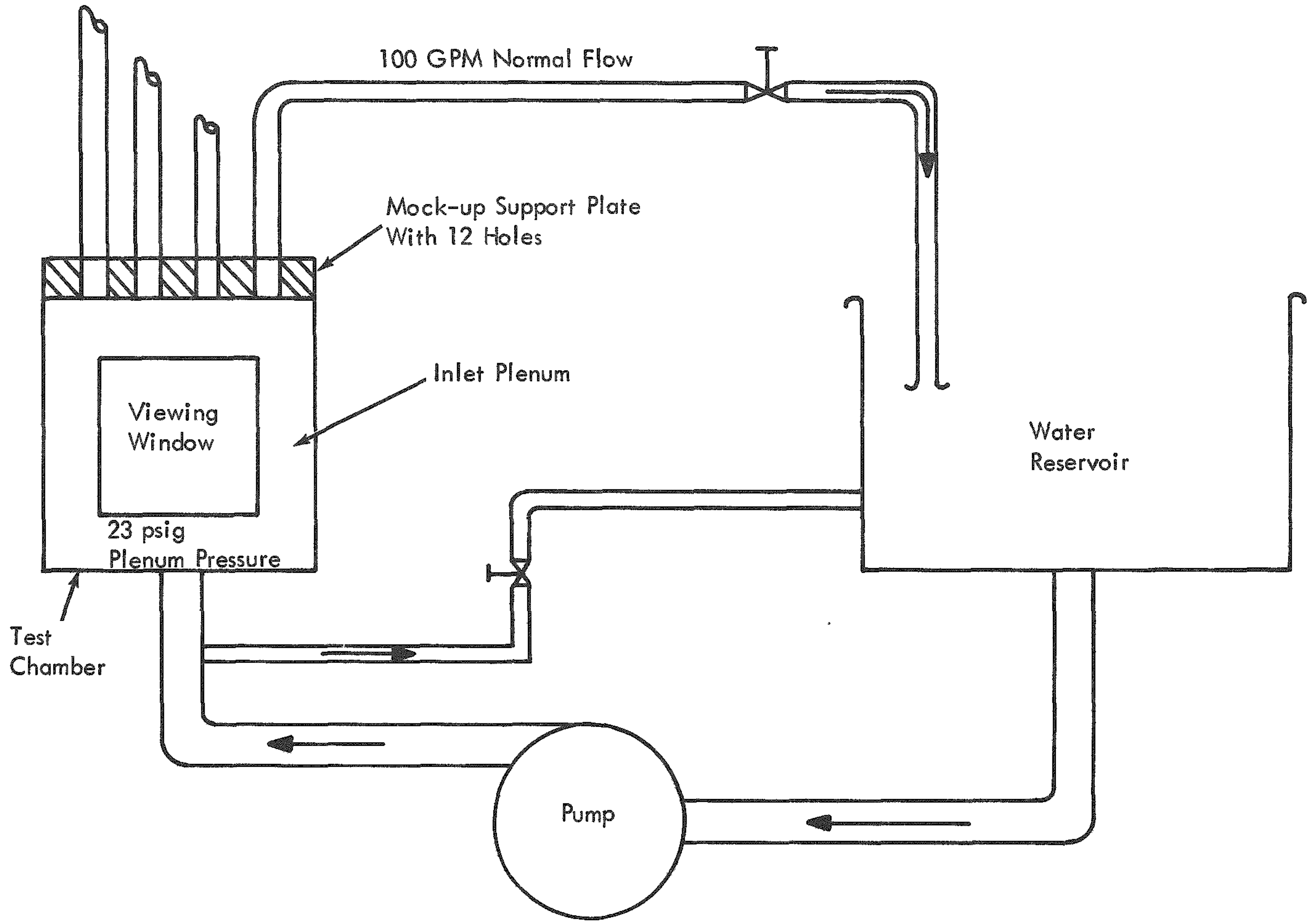

FIG. 15 SKETCH OF HYDRAULIC LOOP AND TEST CHAMBER FOR SUBASSEMBLY FLOW BLOCKAGE TESTS

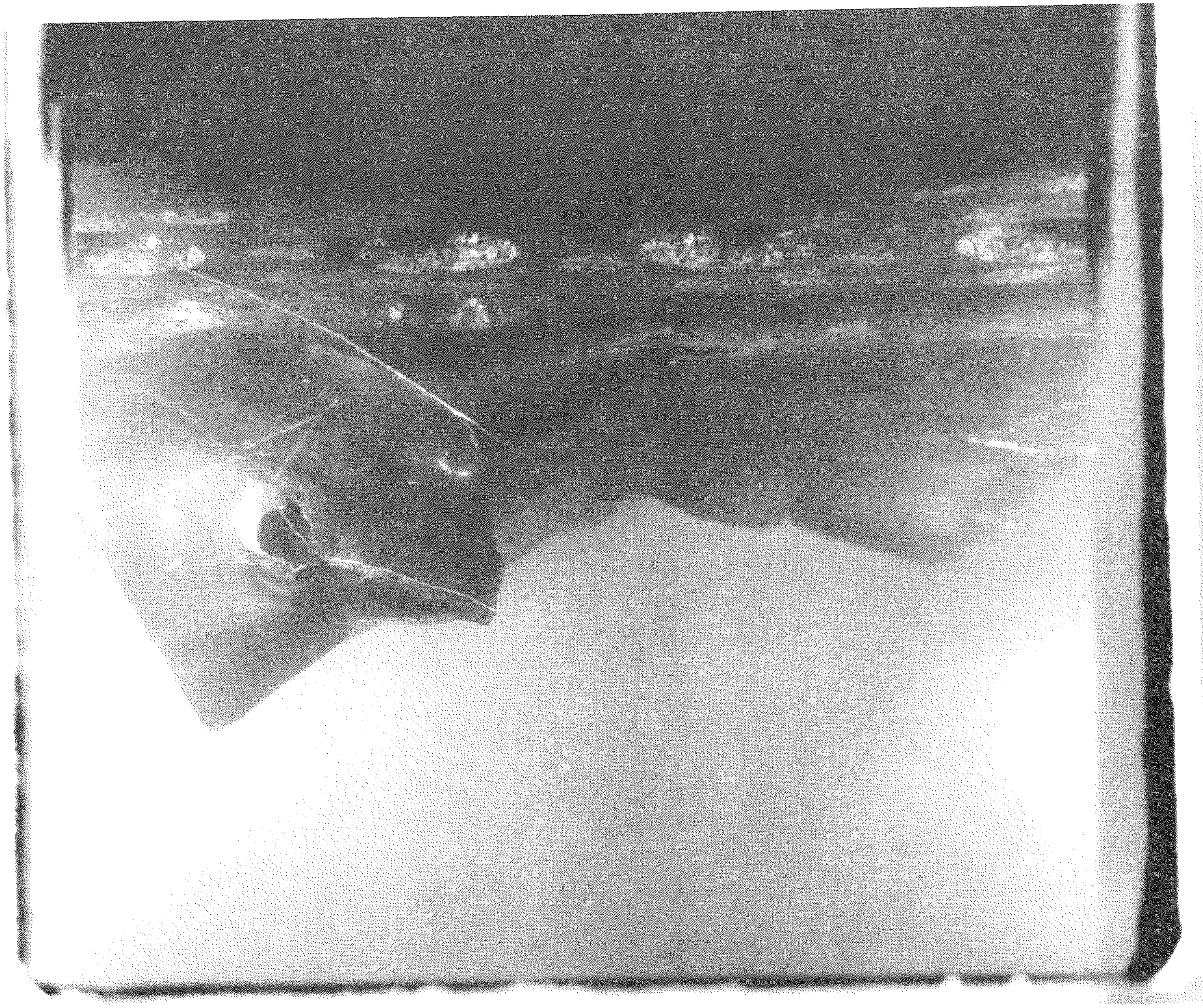

FIG. 16 ZIRCONIUM OBJECT HELD AGAINST UNDERSIDE OF MOCK-UP BOTTOM SUPPORT PLATE IN FLOW BLOCKAGE TEST CHAMBER

In initial tests, the model of the object made from photographs was used. When the zirconium object was removed from the reactor and cleaned and decontaminated, it was used also. For subsequent tests involving deformation of the object, a model was made to dimensions of the object.

3. Verification of Zirconium Object Blocking Coolant Flow

Results of tests of the initial model produced from the photographs of the UFO indicated flow blockage values lower than those predicted for melting. When an accurate model of the actual object was used following removal from the reactor, the results were better but still short of the predictions. In one test using the model of the actual UFO, flow blockages of 86% and 81% in adjacent holes was obtained; and in another try, 65% and 72%. When this same model was somewhat flattened, the blockage values became 97% and 97%. When the actual object was tested, successive results of 88% and 28%, and 90% and 30% were obtained on adjacent holes. The plenum pressure was then increased to 29 psig to better simulate the properties of the zirconium at reactor inlet temperatures of approximately 500 F. Under this condition, the blockages were determined to be 92% and 30%.

Even though the estimated blockages of 95% and 96% were not attained under what were thought to be realistic conditions, the experimental results were encouraging enough to provide a reasonable verification that the zironcium object was responsible for blocking coolant flow to the extent that would cause fuel melting in subassemblies M098 and M127. It is possible that the object was more favorably shaped for blocking coolant flow at the time of the incident than it was following removal from the reactor. The object had to be handled very roughly during the removal operations and could easily have been distorted from its original shape.

B. NONFISSIONABLE MATERIALS SURVEILLANCE PROGRAM

Electrical resistance measurements of the remaining ten tensile specimens irradiated in MS-1, the materials surveillance subassembly in the reactor, and removed from MS-1 last month were completed in April. Electrical resistance measurements of the first 12 specimens, which are Inconel X-750 and representative of subassembly spring materials, were completed in March. The twenty-two tensile specimens had been irradiated to a maximum neutron exposure of about 1.2×10^{21} nvt (E >0. 1 Mev). Previous information about the surveillance program is given in APDA-CFE-20, Section VII.

The ten specimens tested in April are Types 304 and 347 stainless steel, equally divided, and are representative of reactor vessel and core support plate material, respectively. These specimens were subjected to fractions of the maximum flux ranging between 0.15 for the specimen positioned 23.4 inches above the core midplane and 0.99 for the specimen 1.5 inches below the core midplane. The changes between preirradiated and irradiated values of electrical resistance are very small and are essentially within the range of experimental error. It may be said, therefore, that for these stainless steel specimens, there was no significant change in electrical resistance due to irradiation.

It is intended that the twenty-two surveillance specimens will next be the subjects of physical testing such as tensile testing and measurement of hardness.

VI. MAJOR EQUIPMENT

A. EXIT PORT INSPECTION FACILITY

1. Plans to Examine Reactor Subassemblies for Bowing

Efforts are being concentrated on that phase of preoperational testing to determine the dimensional gauging capability of the exit port inspection facility. The present expectation is to utilize the facility to examine reactor subassemblies for bowing, especially those in the southeast quadrant that were near the melted and damaged subassemblies at the time of the incident of October 5, 1966. Several mechanical problems described in Section V.A of APDA-CFE-19 must be resolved. The design effort necessary for the required modifications is presently underway. Earlier information about the exit port inspection facility has been given in previous reports.*

2. Testing Dimensional Gauging Capability

As reported in Section VI of APDA-CFE-20, the results of a dimensional survey of a dummy core subassembly containing depleted uranium fuel pins in the exit port inspection facility compared favorably with dimensional data obtained on the gauging fixture in the Fuel and Repair building (FARB). As reported further, a dimensional survey of an empty dummy subassembly, however, produced scattered results. Another dimensional survey of the dummy subassembly, this time at a temperature of about 350 F, again produced data significantly different from that obtained from the FARB gauging fixture and also inconclusive and scattered. A dimensional check of the dummy subassembly on a precision surface plate supported the validity of the data obtained from the FARB gauging fixture; therefore, attention was directed toward details of the inspection facility in a search for influencing factors.

Representatives of Keuffel and Esser Co., vendors of the optical equipment, when consulted about the dimensional gauging problem, concurred in the use of the taut vertical wire as a benchmark for measurements, but suggested a refinement in the transit measuring method. A collimating technique was recommended to ensure that all measurements be made with the transit line of sight normal to the viewing window to eliminate errors due to refraction through the windows. It is intended to demonstrate the collimating technique and to determine if improvements in dimensional gauging will result.

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* APDA-CFE-5, 8, 9, 10, 16 through 20

B. FUEL TRANSFER FACILITY

In April, the qualification tests of the FARB section of the new fuel transfer facility were satisfactorily completed. In these tests the initial or one-time conditions and settings of equipment, controls and instrumentation were established. A description of the facility and other information are given in previous reports.*

The preoperational tests, the second of three series of tests, were started in April to proof-demonstrate the routine preparatory steps to be accomplished in advance of each operation of the facility. Heaters in the transport and gripper casks were energized and the specified temperature of 320 F in the transport cask and 500 F in the gripper cask were attained. The casks were purged with inert argon gas, and then a positive argon pressure of 4-1/2 inches water column was maintained. An argon pressure of 2 psig was applied as a buffer to shaft seals, valve gate seals, and inner face seals. No difficulties have been experienced to date.

The third and final series of tests is the fuel transfer facility is the acceptance series or proof demonstrations of the proper functioning of the facility. It is expected that the acceptance tests will be completed next month.

* APDA-CFE-10, 19, and 20

VII. MAINTENANCE

A. REPAIR OF FAILED SEAL BELLOWS OF VALVES V516 AND V502-1

1. Access to Valve Locations

In April, the cold trap room in the primary sodium service building was penetrated for access to valves V516 and V502-1 in the primary sodium service system for investigation and repair of indicated sodium leaks through the valve bellows seal. Leak detector alarms had indicated a leak in V502-1 in December, 1967 and V516 in January, 1968. The location of the two valves is shown in Figure 17, a schematic diagram of the primary sodium service system. Valve V502-1 is in the discharge line from No. 1 storage tank, and V516 permits the flow in the primary sodium service system to bypass the cold trap, hot trap, plugging indicator, and the sample coil.

Inasmuch as the primary sodium service system was not in operation and a nitrogen atmosphere was being maintained in the cold trap room, it was decided at the time the leaks were observed to delay investigation and repair of the suspected leaks and combine it with maintenance and repair of the cold trap room cooling system.*

Access to the cold trap room was obtained by removing the blocks of concrete forming a 6-foot-thick shield wall and cutting a hole in the inner wall steel liner.** A preliminary radiation check through the wall monitoring port into the area had revealed that the radiation level just inside the wall was 25 mr per hour. There was no evidence of free sodium in the room. Radiation levels of 40 mr per hour in the center of the room and a maximum 1800 mr per hour at the cold trap were observed.

2. Repair of Valves

The existence of sodium in the valve body penetrations for the leak detector probes verified leaks through the bellows of both valves. This investigation was initiated following maintenance and repair of the cold trap room cooling system and the establishment of an air atmosphere in the cold trap room. After the valves were cooled and a freeze seal established on

* See Section VII. B

** See Figure 19

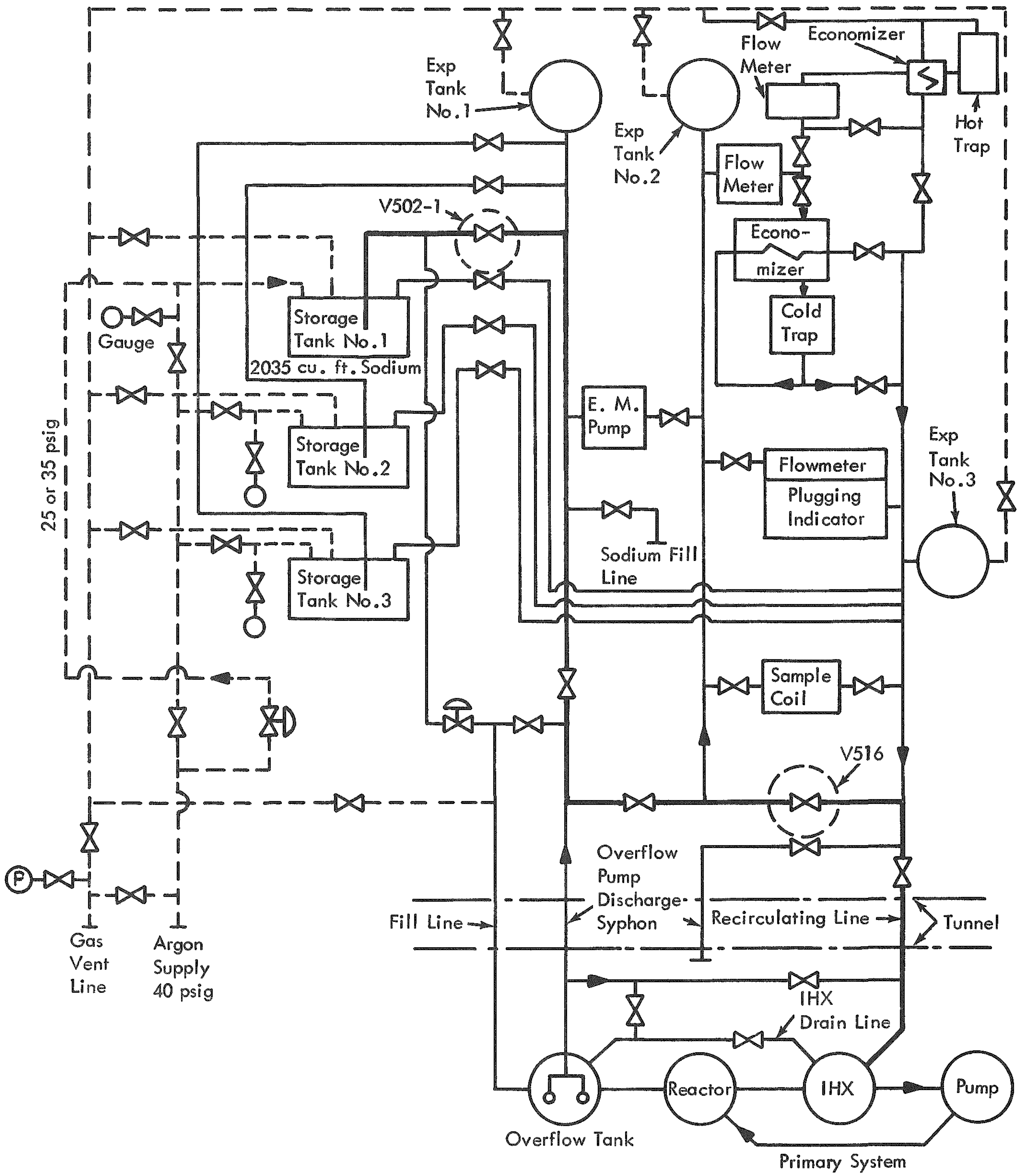

FIG. 17 SCHEMATIC DIAGRAM OF PRIMARY SODIUM SERVICE SYSTEM

either side of both valves, disassembly of the valves and cleaning of parts were accomplished in the same manner as for a previous bellows-sealed valve repair.* Dry ice freeze boxes enclosed the valves (see Figure 19).

A new bellows was installed on each valve and the valves reassembled ready for service. The cold trap room containment was reestablished and the internal nitrogen atmosphere restored.

3. Bellows Failures

A circumferential crack was observed at the bellows-to-seal plug weld on the bellows from V516 (Figure 20); there was no observable evidence of failure on the bellows from V502-1. A subsequent metallographic analysis of the bellows from V502-1 revealed many small longitudinal cracks within one convolution of the center weld between the two sections of bellows. The failures in both valves were found to be localized intergranular attack starting at the inside or gas side of the bellows. The failures were not the result of stress. Neither observation nor metallographic examination revealed any evidence of the cause of the failures.

B. MAINTENANCE AND REPAIR OF COOLING SYSTEM FOR COLD TRAP ROOM

Before the bellows valve repair (above) was undertaken, routine inspection and maintenance and repair of a freon-22 leak in the refrigerant cooled ventilation system for the cold trap room and sodium tunnel to the reactor building were completed. The function of the ventilation system, shown schematically in Figure 21, is to maintain a recirculated nitrogen atmosphere below 150 F in consideration of the structural properties of the concrete walls enclosing the cold trap room and the tunnel. The nitrogen atmosphere maintained at less than 5% oxygen and at a positive differential pressure of 0.5 inch water column minimizes the hazard of fire from sodium leaks.

Normal routine inspection and maintenance of the components of the ventilation system were conducted and the equipment was found to be generally in good condition. It had been two years since the previous inspection and maintenance period. The freon-22 leak was at a flange fitting at the discharge from the compressor. Replacement of the flange gasket and soft washers under the flange bolts and nuts appeared to stop the leaks.

During the maintenance and repair period, the recirculating nitrogen used as a cooling medium was replaced by outside air drawn in by the blower

* APDA-CFE-17, Page 26

FIG. 18 ENTERING PENETRATION THROUGH SHIELD WALL OF COLD TRAP ROOM IN SODIUM BUILDING

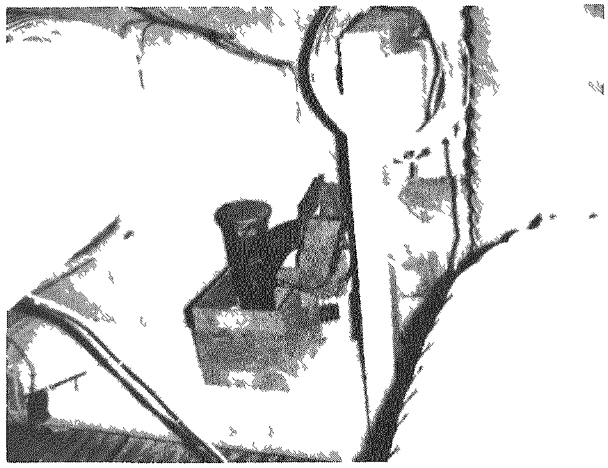

FIG. 19 DRY ICE FREEZE BOX INSTALLED AROUND BODY OF VALVE V516

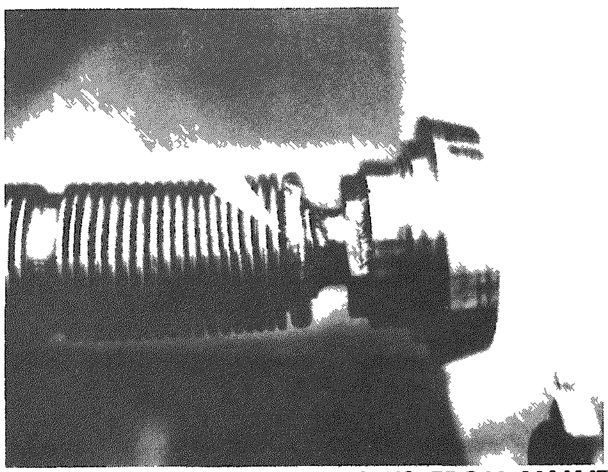

FIG. 20 CRACK IN SEAL BELLOWS FROM VALVE V516

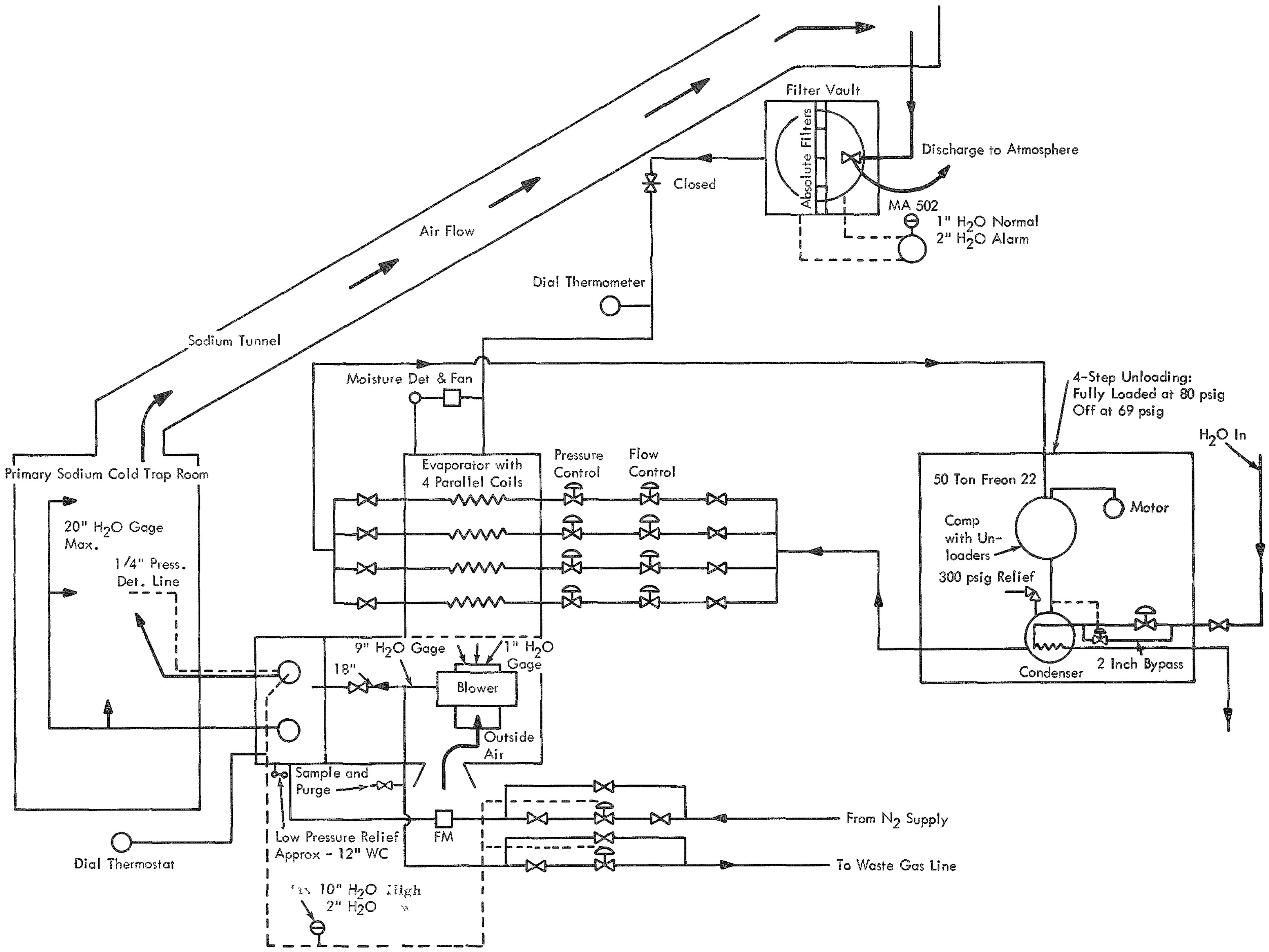

FIG. 21 SCHEMATIC OF VENTILATION SYSTEM FOR COLD TRAP ROOM AND TUNNEL

through a temporary opening in the blower housing and discharged through the cold trap room and the tunnel and out to atmosphere through a temporarily-opened manhole in the top of the absolute filter vault (Figure 21). The valve bellows repair (Section VII. A) was accomplished also during this period of air circulation. When the maintenance work was completed, nitrogen recirculation was re-established to restore the ventilation system to its normal operating condition.

C. INSTALLATION OF FEEDWATER FLOW ORIFICES IN THE NO. 3 STEAM GENERATOR

During April, feedwater flow orifices were installed in the 1200 tubes of the No. 3 steam generator. Details of the flow orifice and its installation were given in a previous report.* After the orifice sleeve inserts were rolled into the tubes at the water inlet header, the tube-to-tubesheet welds were leak-tested by the gas bubble method.** There were no leaks detected. This completes the program of tube-to-tubesheet weld repairs and installation of feedwater flow orifices in all three steam generators.

* APDA-CFE-12

** See APDA-CFE-15, Section IV. B