APDA-CFE-27

ENRICO FERMI ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT NOVEMBER 1968

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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 AEC Contracts 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 of damage is being eliminated and the reactor restored to the operating condition.

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-23.

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 demonstration of the removal of the remaining zirconium segments from the reactor mock-up was successfully completed. It was shown in the demonstration that all the fastening screwheads could be melted by the arc-melt tool to free the segments from the surface of the conical flow guide in the inlet plenum and that, after being dislodged from the flow guide surface, the segments could be handled by the spine-type manipulator, fed to the object retrieval device, and then removed through the 14-inch sodium inlet line.

The temperature of the primary sodium system, which had been maintained at 440 F for the thermal shock tests of the No. 2 steam generator, was decreased to 370 F for the reactor vessel drain. The reactor vessel was siphon-drained of sodium to below the bottom of the core inlet plenum and the sodium level in the No. 1 primary loop was lowered to an elevation four feet below the expected penetration in the 14-inch inlet line elbow for the removal of the remaining zirconium segments from the inlet plenum. The sodium in the riser pipe below the elbow was allowed to freeze, the pipe heaters and insulation were removed from the elbow, and a patch was removed from the secondary containment elbow. At the end of November, the primary elbow was exposed and ready to be penetrated for removal of the segments.

After the reactor was drained of sodium, a borescope inspection of reactor internals revealed that the segment was not in place on the south sector of the conical flow guide. This sector had not been accessible for viewing previously because of obstructions by subassemblies in the core lattice. The missing segment, observed to be bent and flattened, was found in the northeast sector in a position where it appeared to be clinging to the underside of the lower support plate. The segment fell to the floor of the inlet plenum when it was inadvertently touched by viewing equipment being inserted for viewing the segment from another angle. The four remaining segments were found to be in their expected positions on the conical flow guide.

The arc melt tool, along with viewing equipment, was inserted through penetrations in the rotating plug and the core support plate and was used to melt all the fastening screwheads. A dislodging tool was inserted through a rotating plug penetration and the support plate to free the segments from the cone. The present intent is to use the spine-type manipulator to feed the segments to the object retrieval device for removal through the penetration in the elbow in the 14-inch primary pipe.

A vacuum cleaning device for insertion into the inlet plenum through the 14-inch line is presently being fabricated, and the plant waste gas system was shown in a demonstration test to be satisfactory as a vacuum source.

A second hydraulic test of the No. 3 steam generator verified the results of the first preliminary test, which showed that the pressure drop through the 1200 tubes with flow-restricting orifices installed is in agreement with that predicted by calculation. The soft iron water manifold cover joint gaskets are to be replaced by copper gaskets because of water leaks to atmosphere from the manifolds. The copper gaskets, a long delivery item, has been ordered previously and are expected to be available shortly.

II. INVESTIGATION OF ZIRCONIUM SEGMENTS

A. BACKGROUND

The deformed sheet of zirconium removed from the reactor in March was one of six triangular-shaped segments originally installed on the sloping sectors of the conical flow guide in the inlet plenum. Details of the removal of the first segment are given in APDA-CFE-20. The present plan is to dislodge the remaining segments from the conical flow guide and remove them through the 14-inch sodium inlet pipe of the No. 1 loop in the same manner that the deformed segment was removed.

An arc-melt tool was developed to melt the heads of the fastening zirconium screws to free the remaining segments from the conical flow guide. Tools to dislodge the freed segments from the flow guide were also developed and are available if needed. The arc-melt and dis lodging tools were designed to operate through a 6-inch penetration in the rotating shield plug and through the 1-5/8-inch holes in the support plates. The spine-type manipulator* and the object retrieval device* used in the removal of the first segment in March have been modified for more precise control of these devices in the removal of the remaining segments.

Two segments were freed from the conical flow guide in the reactor mock-up and then removed through the 14-inch inlet line as part of a demonstration that the removal operation can be performed with the available tools and devices. Reactor viewing conditions were simulated as closely as possible in the mock-up. It is intended to remove the remaining three segments from the mock-up to complete the demonstration.

Core subassemblies were relocated in the reactor and the sweep mechanism was removed from the reactor to permit access to the north and south sectors of the conical flow guide for viewing and for removal of the zirconium segments from these sectors. Viewing of these sectors in the past has not been possible because of obstruction by fuel subassemblies located directly over the south sector and the presence of the core sweep mechansim in the 6-inch penetration required to view the north sector. Previous information on the investigation of the zirconium segments is given in APDA-CFE-20 through 26.

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* See Section II of APDA-CFE-15 through 19 for previous information on the spine-type manipulator and the object retrieval device.

B. MOCK-UP DEMONSTRATION OF SEGMENT REMOVAL COMPLETED

In November, segments were freed and dislodged from the north, south, and southeast sectors* of the conical flow guide in the mock-up and satisfactorily removed through the 14-inch inlet line. These operations completed the program started last month to demonstrate that all segments known or believed to be fixed to the conical flow guide in the reactor inlet plenum can be successfully removed from the reactor. It was shown that the arc-melt tool can melt all of the fastening screwheads to free the segments and that the segments can be fed to the object-retrieval device by the spine-type manipulator and removed through the 14-inch line. It was found that all segments could be dislodged from the conical flow guide in the mock-up by tapping with the arc-melt tool. The performance of the dislodging tool had been previously demonstrated.**

In the mock-up demonstration of segment removal, attention was directed to the improvement of techniques to facilitate removal operations in the reactor. For instance, emphasis was placed on investigating optimum lattice positions for insertion of the arc-melt tool and the viewing equipment to accomplish the melting of all screwheads with the least number of tool and viewing equipment changes. In addition, segment handling in the inlet plenum was the subject of testing to determine the most effective maneuvers in feeding the segment to the retrieval gripper. It was found that the penetration in the 14-inch elbow will have to be enlarged to 6 x 10-1/2 inches to accommodate the full-width segment.

C. SPINE DISCS REPLACED IN SPINE-TYPE MANIPULATOR

In the demonstration segment removal operations in the mock-up, it was found that the spine column of the spine-type manipulator was not responding to movements of the remote control toggle switch in the manner expected. As a result, the movements of the manipulator gripper were unpredictable, thus prolonging the time needed to grasp and transfer the segment to the retrieval gripper. An inspection of the tool revealed that the disc-on-disc bearing surfaces of the 56 carbon steel discs making up the spine column had become worn to the extent that the accumulative height of the stacked discs had decreased about 1 inch from the as-built dimension. The result was a loss in tension in the disc control cables which unfavorably affected the control of the spine column. The worn bearing surfaces of the discs were the result of the cumulative experiences of flexing the spine column in both the reactor and mock-up operations.

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* See APDA-CFE-26, Figure 1

** See APDA-CFE-24, page 7 and Figure 6

Demonstration of segment-removal operations were delayed while new discs of the same design were installed. This modification greatly improved control of the tool. Subsequently, the total operation was shown to require only 1-1/2 hours to grasp the segment in the mock-up, transfer it to the retrieval gripper and remove it through the 14-inch pipe. It is expected that more time will be required for removal of a segment from the reactor because of the necessity for step-wise withdrawal of the object-retrieval device from the reactor to allow the equipment to cool sufficiently for handling.

D. SEGMENT MISSING FROM SOUTH SECTOR OF CONICAL FLOW GUIDE IN REACTOR

1. Viewing of Inlet Plenum

It was discovered that the zirconium segment believed to have been fastened on the south sector of the conical flow guide in the reactor inlet plenum was not in place on the flow guide,nor was it in the vicinity of its original location. This discovery was made during a borescope inspection of features inside the reactor vessel (see Sec. IV. A) after the reactor was drained of sodium (see Sec. III. C) in preparation for removal of all of the remaining segments. Previously, both the north and south sectors had not been accessible for viewing through a borescope because of obstructions by the sweep mechanism and by subassemblies in the reactor lattice. Subassembly movements in September* cleared a path for viewing the south sector, and removal of the sweep mechanism in October** allowed viewing of the north sector.

The fastening screws were still in place on the south sector of the flow guide and also on the northeast sector, believed to be the initial position of the loose segment removed from the reactor in March 1968. The other four segments, including the one on the north sector, were observed to be in place and securely fastened. As reported in APDA-CFE-21, page 19, a bottom screw was missing from the southwest segment.

2. Location of Second Detached Segment

A thorough borescope inspection of the inlet plenum revealed the missing zirconium segment to be in the northeast sector in the corner of the flow inlet chamber to the core subassembly nozzles, in a position where it appeared to be clinging to the underside of the lower support plate. This

* See APDA-CFE-25, page 11
** See APDA-CFE-26, page 29

segment, like the first one removed, was observed to be bent. Its position is shown in Figure 1, a construction photograph of the core flow inlet chamber above the inlet plenum. It is not known what held the segment against the underside of the support plate. A possible explanation is that it wedged between the top edge of the thermal shielding and the bottom of the lower support plate, which rests on the seal rail to form a seal between the core inlet and the blanket inlet plena.

The segment was viewed through a borescope in different horizontal directions to determine its shape and its position with respect to the core lattice. As shown in Figure 2, it appears that the segment was fully under four support plate holes, i.e., NO4-NO3, NO4-NO4, NO5-NO3, and NO5-NO4, and partially under eight surrounding holes. In an attempt to insert viewing equipment through the upper support plate for a plan survey of the position and shape of the segment, the segment was inadvertently touched and fell to the floor of the inlet plenum. Figure 3 is a composite photograph of the bent segment resting on the plenum floor with the inlet from the No. 1 primary loop in the background.

3. Segment Bent and Flattened

A further borescope inspection of the segment showed details of its appearance. The apex was bent toward the back side, in the opposite direction from the bend of the first segment removed and opposite the direction of bend that would be expected if sodium flow had wrenched it loose from the bottom screws and bent it back on the fastened apex. The sides of the segment, bent at installation to form with the corners between the conical flow guide and the baffle plates, appeared to be flattened, and 1/3 of the small bent tab which formed over the bottom edge of the flow guide was missing. The segment appeared to have a broad expanse of flat surface, making it more likely to have been the object which blocked the flow to several of the subassemblies on October 5, 1966, than the first segment removed.

It is not known at this time when this segment tore loose from the flow guide nor whether it came from the northeast or south sectors. The segment was not seen in the position where found, nor in any other part of the inlet plenum during the thorough borescope inspection of the inlet plenum following removal of the first segment. There is the possibility that it was either still in position on the cone or lodged in one of the loop inlet nozzles, inaccessible for viewing, at the time of the borescope inspection. It is hoped that evidence can be obtained to identify the segment with one of the uncovered sectors and thus possibly help to establish some of its history. Radioactive measurements of the segments are also planned to help establish the history.

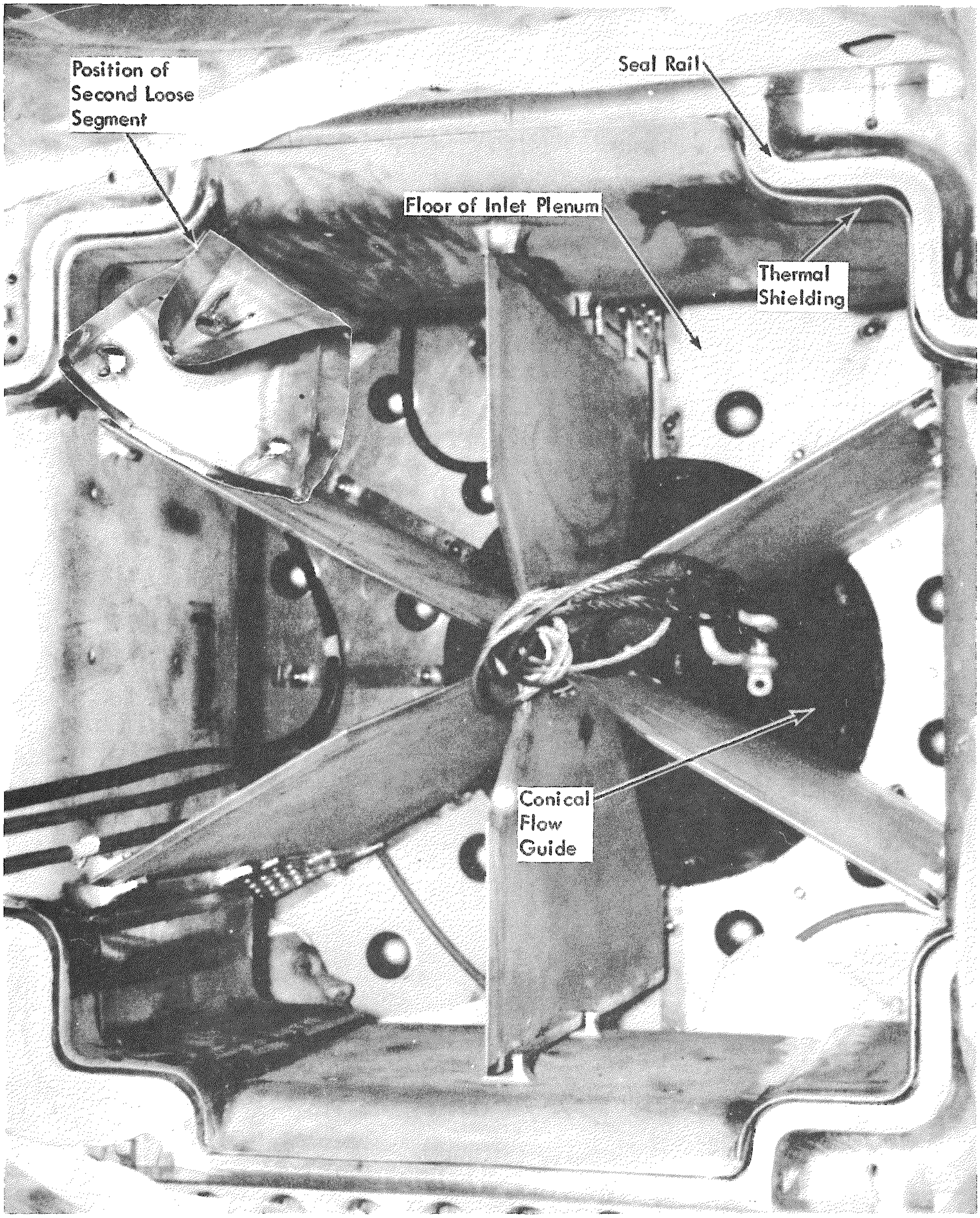

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FIG. 1 CORE FLOW INLET CHAMBER IN CONSTRUCTION STAGE

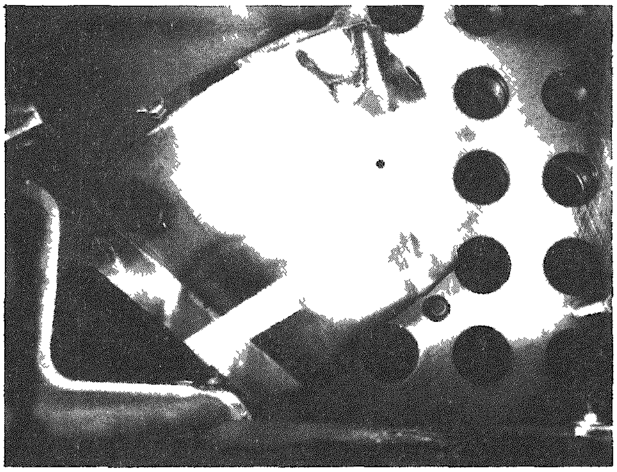

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FIG. 2 SEGMENT HELD TO UNDERSIDE OF LOWER SUPPORT PLATE IN MOCK-UP

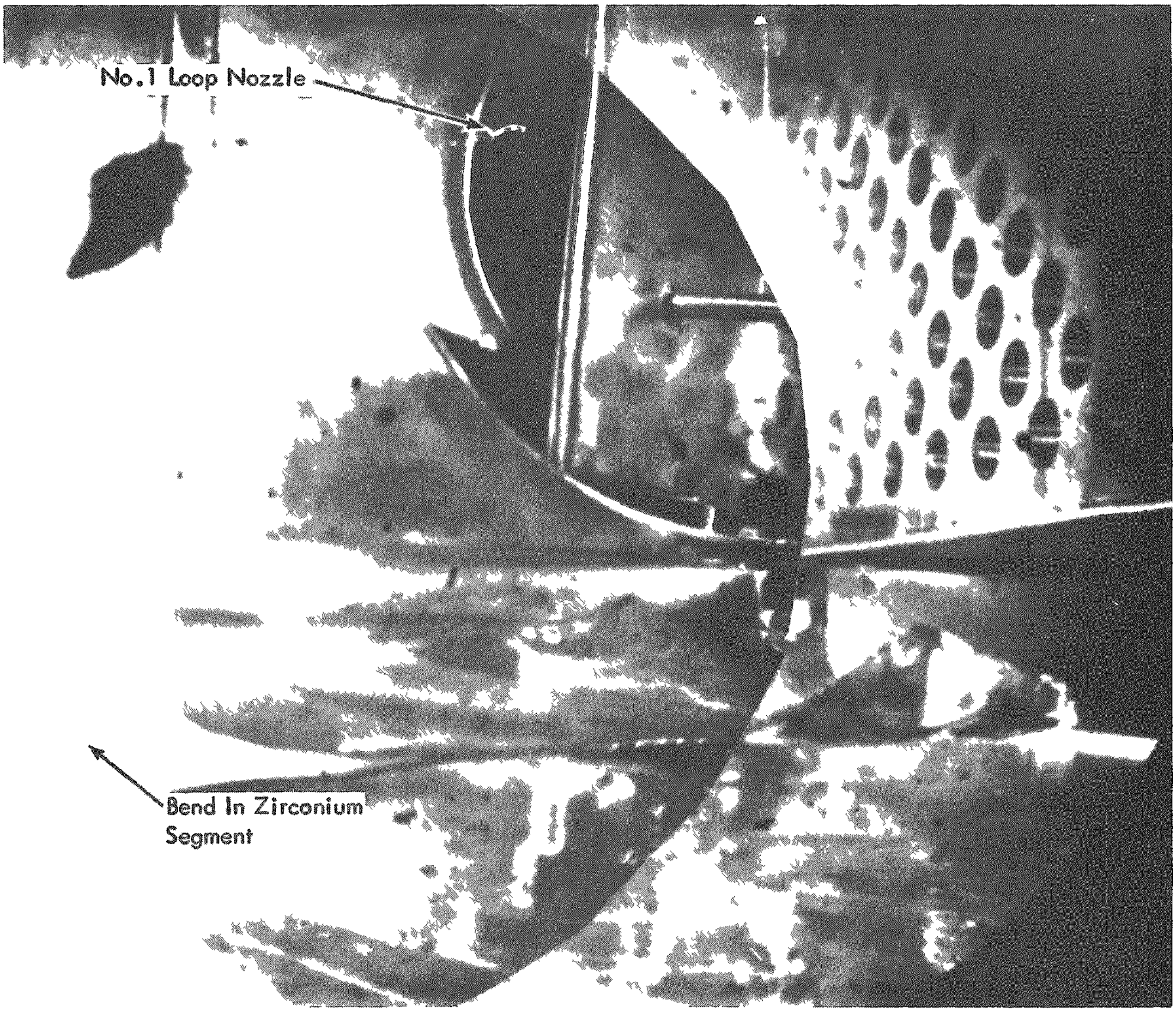

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FIG. 3 BENT SEGMENT ON FLOOR OF REACTOR CORE INLET PLENUM IN NORTHEAST SECTOR

E. SCREWHEADS MELTED IN REACTOR

The functioning of the arc melt tool having been satisfactorily demonstrated in mock-up testing, this tool together with two borescopes and the lamp were inserted through a penetration in the rotating shield plug and the holes in the support plate to begin the task of melting the screwheads fastening the remaining four segments onto the conical flow guide. With the arc-melt tool and the viewing equipment inserted through the first plug penetration, the screwheads fastening the segments on the southeast and southwest sectors were melted. The tools were then transferred to another plug penetration and the screwheads in the north and northwest sectors were melted. The screwheads in the northeast sector, believed to have been the original location of the first segment removed, were also melted.

The melting operations proceeded satisfactorily. Most of the operations required only one application of the arc-melt tool; in a few instances, a second application was required.

After the fastening screwheads were melted, each segment was tapped with the arc-melt tool in the hope that it would fall free of the flow guide surface. None did. A dislodging tool was then inserted through the rotating plug and support plate holes, articulated to pry the segments at the apexes in the manner shown in Figure 6 of APDA-CFE-24. This technique easily dislodged the segments from the surface of the conical flow guide. The program is to use the spine-type manipulator and, when access through the 14-inch inlet line of the No. 1 loop has been obtained, the object-retrieval device for removal of the 5 free segments from the inlet plenum in the same manner that the first segment was removed.

F. PATCH REMOVED FROM SECONDARY CONTAINMENT ELBOW

Work proceeded to provide access through the 14-inch inlet pipe of the No. 1 primary loop for insertion of the object-retrieval device into the core inlet plenum. It is intended to again penetrate the elbow at the top of the riser pipe in the discharge line from the No. 1 primary pump. The location of the elbow, which is about 35 linear feet from the inlet nozzle of the No. 1 loop, is shown in Figure 1 of APDA-CFE-20.

The sodium level in the riser pipe was lowered to a point four feet below the anticipated lower lip of the primary pipe penetration, and heaters were turned off to allow the sodium in the riser pipe to freeze to provide reasonable comfort for work at the elbow. The scaffold was re-erected at the elbow and induction heating wire and insulation removed to expose the 20-inch diameter secondary containment pipe elbow for penetration. Heating circuits were again modified to retain heating on the 6-inch inlet pipe to the blanket plenum.* The same 16-inch by 14-inch ellipsoidal patch that was cut out for the removal of the first segment was removed by grinding. Figure 4 shows the penetration in the secondary containment elbow, exposing the patch weld in the primary elbow. Figure 5 is a close-up of the primary patch weld. The primary pipe elbow is to be penetrated by cutting at the weld to remove a patch including the existing patch and an additional 3-1/2 inches of metal at the top. The new patch is expected to be 6 inches by 10-1/2 inches to accommodate the full-width segment.

G. DEVELOPMENT OF VACUUM CLEANING SYSTEM FOR INLET PLENUM

1. Components of Vacuum Line Being Fabricated

Figure 6 shows the concept of a vacuum cleaner-separator for insertion through the 14-inch inlet pipe to remove debris from the core inlet plenum. The design of this vacuum cleaner has been completed and the components are presently being procured and fabricated. Preliminary tests (see Sec. II, G-2) have given assurance that the plant waste gas system can be used as a vacuum source.

The separator chamber shown in Figure 6 is designed to remove and retain solid material and particulate matter by a double change in gas flow direction and impingement on stainless steel wool. Positioning the silicone rubber vacuum hose in the inlet plenum is accomplished by advancing or retracting the separator chamber inside a flexible aluminum carrier hose and by rotating the separator internal element to which the silicone rubber hose is clamped. Previous mock-up tests showed that the spine-type manipulator, when extended toward the outer area of the inlet plenum, was incapable of accurately positioning the silicone rubber hose.

The flexible carrier hose, which contains the separator chamber, is externally maneuvered by means of a control cable. It is intended to functionally test this vacuum system in the reactor mock-up.

2. Demonstration of Waste Gas System as Vacuum Source

The results of a preliminary test of the plant waste gas system gave assurance that this system would be a satisfactory source of vacuum for vacuum cleaning the inlet plenum. A 3/4-inch-diameter hose with a temporary separator chamber attached was connected to the waste gas system at a maintenance connection above the operating floor in the reactor building. A line from this maintenance connection, along with lines from

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* See APDA-CFE-18, Figure 18

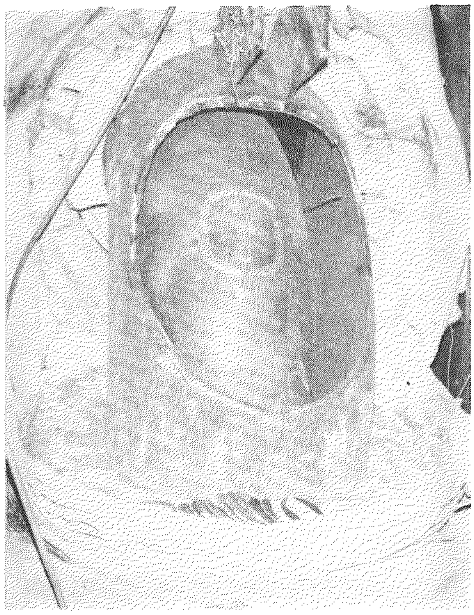

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FIG. 4 PATCH REMOVED FROM SECONDARY CONTAINMENT ELBOW IN 14-INCH INLET LINE

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FIG. 5 PATCH WELDED IN PRIMARY PIPE ELBOW PRIOR TO REMOVAL FOR WITHDRAWAL OF REMAINING SEGMENTS

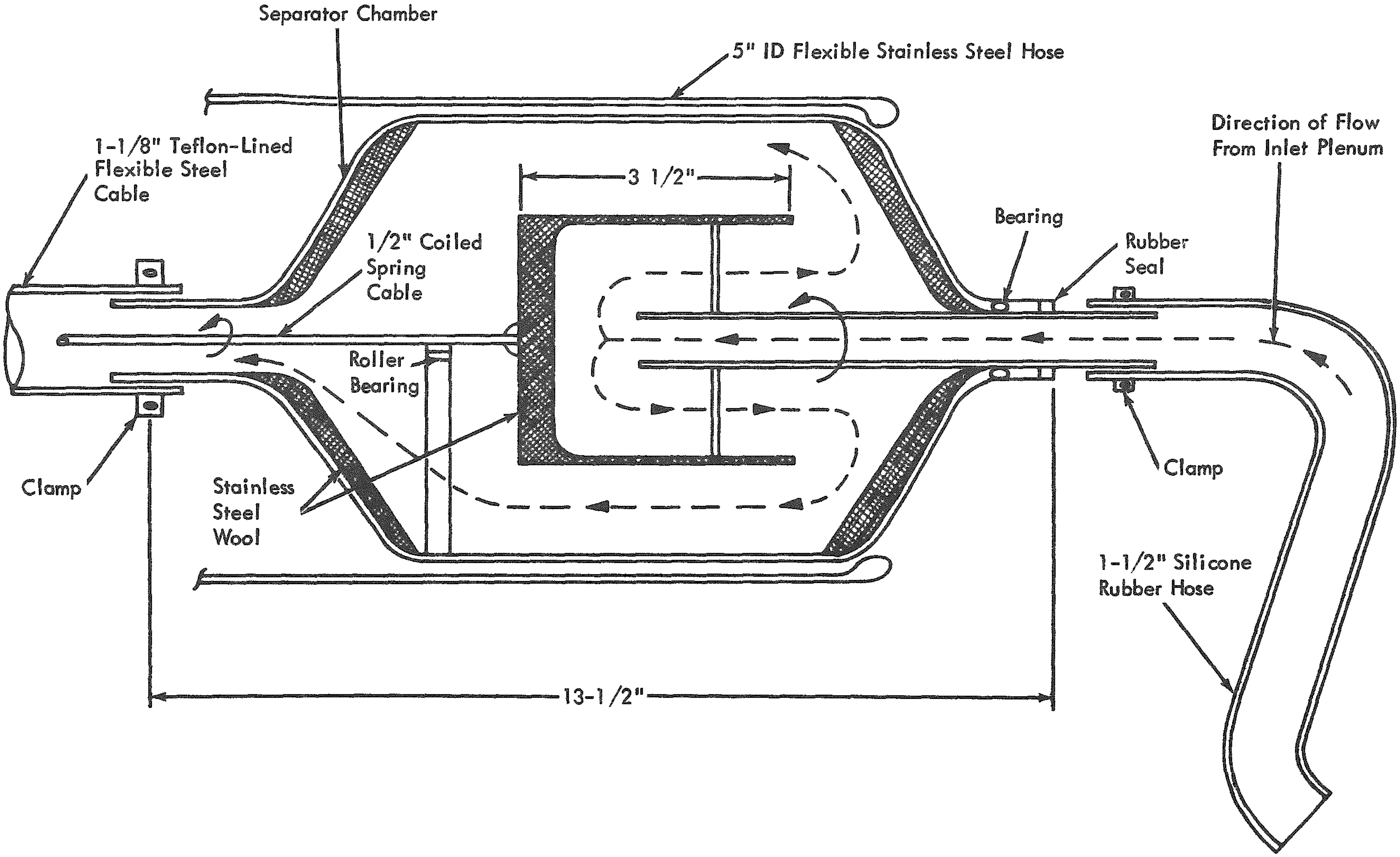

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FIG. 6 VACUUM LINE WITH SEPARATOR CHAMBER the reactor seals, discharges to the waste gas collection header shown in Figure 15 in APDA-CFE-22. The waste gas header discharges through absolute filters to an evacuated waste gas storage tank. It was demonstrated that the vacuum line located on the operating floor could pick up small particle debris. The debris was shown to be captured in the temporary separator chamber.

A temporary vacuum line extending from the waste gas maintenance connection to a fitting at one of the 6-inch penetrations through the rotating plug was installed to determine the response of the primary cover gas pressure control system to maintain a positive cover gas pressure in the drained reactor vessel during the vacuuming operation. A gas flow of about 28 cfm was measured on a flowmeter installed in the temporary vacuum line when the waste gas system was opened to the temporary line. The cover gas pressure control system responded satisfactorily to maintain a positive cover gas pressure between 1/2 inch and 1 inch of water column. It is believed that the vacuum system will provide the desired 50 cfm when the vacuum tool is connected to the waste gas system at a 2-inch fitting to be installed at a below-floor location, rather than through the restrictive 3/4-inch above-floor piping used in this preliminary test. Additional demonstration test work is planned.

III. OPERATIONS

A. GAS SYSTEMS PERFORMANCE

Since the last data were reported,* the following primary system gas activity and gas analysis have been obtained:

1. Primary System Cover Gas Activity

Location	Sample Date	Gross Beta Concentration $\mu Ci/cm^3$
Reactor Cover Gas	11-1-68	1.7×10^{-6}
Reactor Cover Gas	11-15-68	1.6×10^{-6}
Reactor Cover Gas	11-21-68	1.1×10^{-6}
Reactor Cover Gas	11-22-68	6.6×10^{-7}
Reactor Cover Gas	11-25-68	2.4×10^{-6}
Reactor Cover Gas	11-29-68	1.4×10^{-6}
Primary Shield Tank	11-1-68	2.0×10^{-8}
Primary Shield Tank	11-7-68	3.2×10^{-8}
Primary Shield Tank	11-15-68	2.7×10^{-8}
Primary Shield Tank	11-25-68	3.1×10^{-8}
Primary Shield Tank	11-29-68	2.5×10^{-8}

* APDA-CFE-26

2. Primary System Cover Gas Analysis

	Reactor Cover Gas (Argon), ppm by volume*	Primary Shield Tank Atmosphere (Nitrogen) ppm by volume**
Oxygen	Below 25	90***
Carbon Monoxide	Below 10	Below 10
Carbon Dioxide	10	10
Hydrogen	Below 4****	Below 2.5
Helium	Below 4	Below 4
Methane	Below 10	Below 10
Nitrous Oxide	------------- Not Measured -------------	
Argon	Remainder	Not Measured
Nitrogen	1040	Remainder
Dew Point	Not Measured	-45 F

* Sample date for reactor cover gas was 11-6-68
** Sample date for primary shield tank atmosphere was 11-15-68
*** Technical Specifications state 1000 ppm maximum
**** 10 ppm is the recommended maximum for reactor operations

B. GENERAL

1. Primary Sodium System

The temperature of the primary sodium system had been maintained at 440 F to provide a high secondary sodium temperature in the No. 2 steam generator as an initial condition for the thermal shock tests of this unit (see APDA-CFE-26, page 21). After the thermal shock test, the primary system temperature was decreased to 370 F by de-energizing some of the reactor vessel, pipe, and component heaters in preparation for draining the reactor vessel and lowering the sodium level in the No. 1 primary loop for inspection of the interior of the reactor vessel and the removal of the zirconium segments from the core inlet plenum. The decrease in primary system temperature, accomplished in about 4 days, provided minimum temperature conditions for segment removal operations in the inlet plenum.

The three primary sodium pumps were operated at refueling speed. Care was taken to maintain the sodium flow rate at less than the maximum 7×10^6 lb/hr specified in the Plant Technical Specifications for the condition with the holddown in the UP position. The holddown has been in the UP position since it was raised subsequent to the fuel melting incident, pending inspection of the holddown fingers and inspection or replacement of the fuel and inner radial blanket subassemblies with new subassemblies. The syphon drain line was inserted into the reactor on November 8, and the primary pump main motors were shut down prior to lowering the syphon line down through the support plate into the inlet plenum. Pump operation was continued at pony motor speed until sodium drain. The primary pump main motor electrical leads were disconnected to ensure that the pump main motors could not be started while the syphon drain line was in the reactor, while yet permitting operation of the secondary sodium pumps. Normally, interlocks prevent operation of the secondary pumps when the primary pumps are not running.

The November operating data are as follows:

	Primary System	Secondary System*	
		Loop 2	Loop 3
Cold Trap Operation, hr	34	19	28
Maximum Plugging Temperature, F	<220	325	280
Minimum Plugging Temperature, F	<220	<220	<220

* The No. 1 secondary loop was drained of sodium during November.

2. Secondary Sodium System

Sodium remained drained from the No. 1 secondary loop and steam generator during November. The three leaking tubes in the No. 1 steam generator had been plug-welded* and the water and steam manifold covers remained off for the replacement of leaking gaskets** and the reinstallation of test thermocouples (see APDA-CFE-17, page 25).

* See APDA-26, page 39.

** See Section VI. B for information on the replacement of water and steam manifold gaskets on all steam generators.

After the thermal shock tests of the No. 2 steam generator and the subsequent 600 psig nitrogen pressure test to verify the continued integrity of the unit,* the water tubes were purged with nitrogen for a period of 6 days to dry out the interior of the tubes. It is intended to drain sodium from the No. 2 secondary loop in preparation for again replacing the water manifold gasket on the No. 2 steam generator. Both the water and steam manifold gaskets were replaced in October because of previously discovered leaks from the manifolds to atmoshpere. The water-to-air manifold joint leaked again during the thermal shock tests. The sodium in the No. 2 secondary loop was cold-trapped for 19 hours in November to reduce the plugging temperature from 325 F to less than 220 F.

The No. 3 steam generator was reassembled after the installation of replacement annealed soft iron gaskets in both the water and steam manifold cover joints. The No. 3 secondary loop was filled with sodium and, after a 600 psig nitrogen pressure check which confirmed the integrity of the No. 3 steam generator, the steam generator was filled with water in preparation for an hydraulic test of the unit (see Section III, D). There were no indications of leaks in the steam generator as evidenced by the observation of no change in the 5 ppm hydrogen content in the sodium-side cover gas after water fill. After the No. 3 secondary loop was filled with sodium, the sodium was cold-trapped for 28 hours and the plugging temperature was reduced from 280 F to less than 220 F.

C. REACTOR VESSEL DRAIN

1. General

The reactor vessel was drained to below the bottom of the core inlet plenum and the sodium level in the No. 1 primary loop was lowered to an elevation of four feet below the expected penetration in the 14-inch inlet line elbow for removal of the remaining zirconium segments from the inlet plenum. The drain procedure was the same as that carried out in August 1967 for the inspection of the reactor vessel interior.** In June 1967, the sodium level in the reactor vessel was lowered to 1 inch below the top support plate*** for inspection of the damaged subassemblies M098 and M127.

In the manner of the August 1967 drain operation, the sodium level in the reactor vessel and the No. 1 primary loop was lowered from the operating level at an elevation of 580 feet 11 inches to an elevation of 575 feet (see Figure 7), at which point no further spill-over from the reactor vessel through the 30-inch discharge pipe was possible. The drop in

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* See APDA-CFE-26, page 21.
** See APDA-CFE-13, page 8.
*** See APDA-CFE-11, page 6.

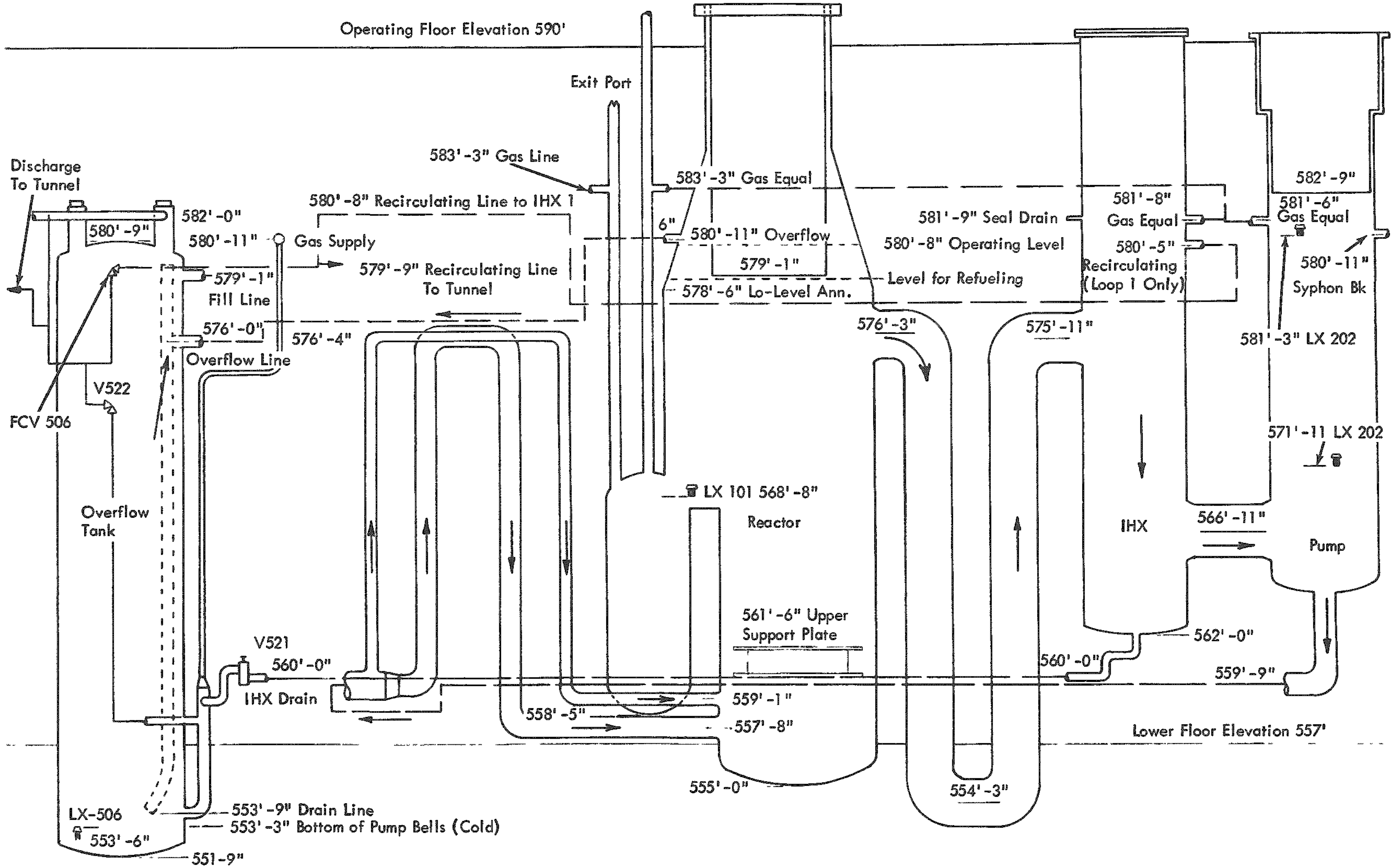

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FIG. 7 PRIMARY SODIUM SYSTEM ELEVATION LAYOUT sodium level was accomplished by alternate draining through the No. 1 IHX drain line to the overflow tank and the transfer by an overflow pump to the No. 2 and 3 storage tanks. The sodium level in the No. 1 primary loop was further decreased to an elevation of 572 feet by the same alternate drain-and-pump process. The balance of the sodium in the reactor vessel was transferred to the No. 2 and No. 3 storage tanks by siphoning action through a temporarily installed siphon pipe and temporary drain line which discharged through the plant siphon drain system (see Figure 8).

2. Preparations for Drain

Details of the preparations for the drain operation appear in APDA-CFE-6, 9, and 10, pages 23, 20 and 29, respectively. The temporary rubber back-up seal* for the plug klozure seal was fastened in place to span the outside gap between the plug and the top part of the vessel to allow pressure equilization across the klozure seal with the reactor vessel at 10 psig for draining. The 2-inch carbon steel siphon pipe was inserted through a plug penetration and support plate holes in the P00-P00 position to mate with the hollow central column permanently installed in the lower plenum for complete drain (see APDA-CFE-13, page 10). Most of the temporary drain line which passes through a manhole penetration in the operating floor to tie into the plant siphon line was left in place after the August 1967 drain. This 2-inch carbon steel line was extended to connect with the siphon pipe. The heater, thermocouple, and insulation coverage were completed. The photograph on page 9 of APDA-CFR-11 shows the insulated drain line. The storage tank room was inerted with nitrogen to eliminate the possibility of a fire in the event of a sodium leak, and the electrical leads to the three primary pump main and pony motors were disconnected to complete the preparations and precautionary measures.

3. Drain Details

The temperature of the primary sodium system was reduced to 370 F for the drain operation. In the initial drain of the No. 1 primary loop, 1640 gallons of sodium were transferred to the overflow tank and to the No. 2 and 3 storage tanks in 5 hours and 20 minutes. An additional eleven hours and 15 minutes was required to siphon drain 2060 gallons of sodium from the reactor vessel.

To accomplish the siphon drain, the argon cover gas pressure in the reactor vessel was established at 10 psig (it is normally about 1 to 4 inches water column) and the pressure in the storage tanks was maintained at 1 to 2 inches of mercury, absolute. The interior of the klozure seal was pressurized with argon at 12 psig and the space between the klozure

* See APDA-CFE-10, page 30

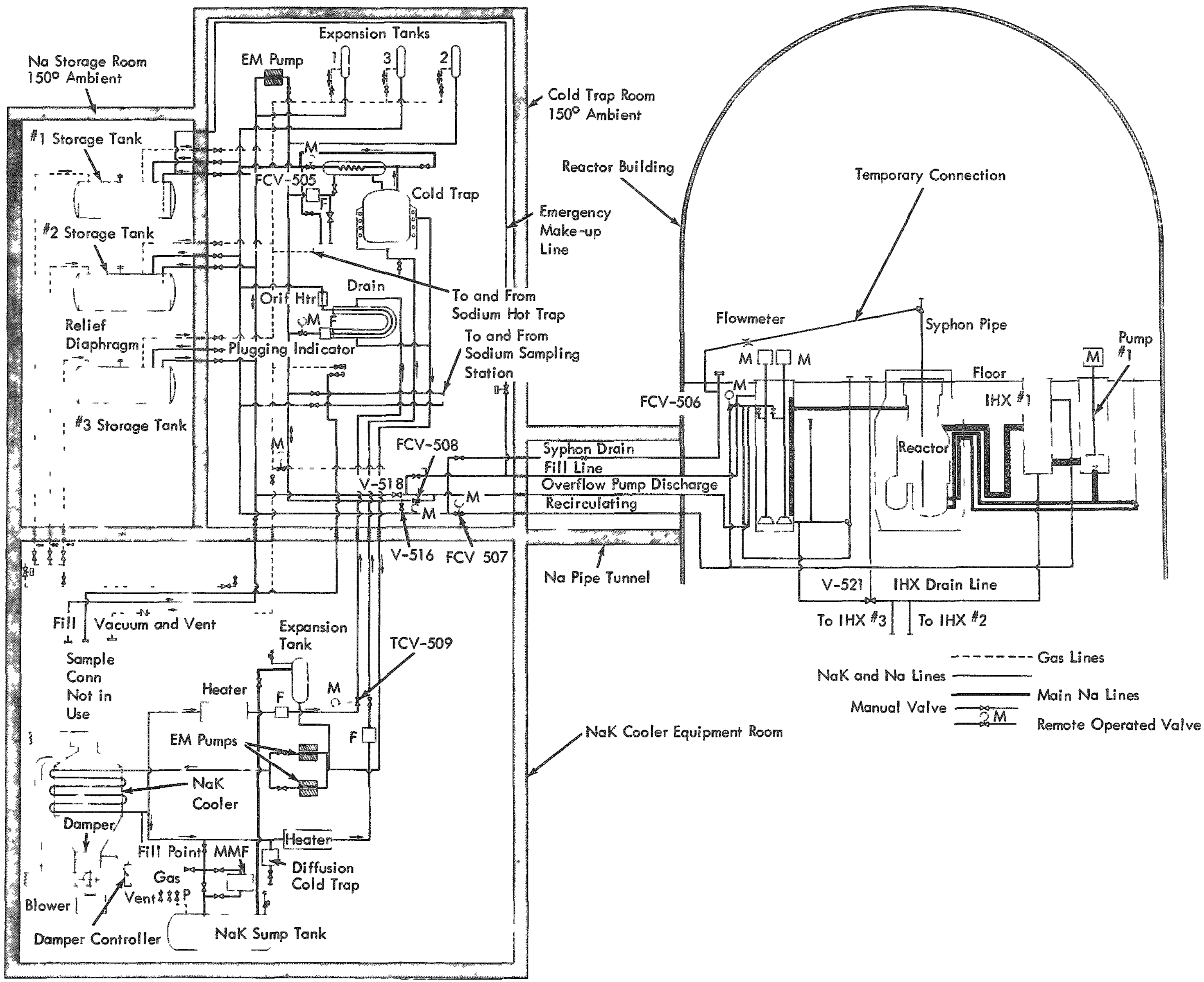

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FIG. 8 PRIMARY SODIUM SERVICE SYSTEM FLOW DIAGRAM

seal and the back-up seal at 10-psig to compensate for the 10-psig vessel cover gas pressure. Thus, the klozure seal experienced a near-normal pressure differential.

D. HYDRAULIC TEST OF NO. 3 STEAM GENERATOR

A second hydraulic test of the No. 3 steam generator was performed to verify the results of the first test* conducted last month. The second test was carried out after the replacement of manifold gaskets and the subsequent sodium and water fill and circulation.

The purpose of the test was to determine the new flow-pressure drop characteristics of the steam generator due to the installation of the flow orifices in the 1200 water tubes (see APDA-CFE-21, page 44). Before the orifice installation, the pressure drops across the unit were determined to be low at low heat loads, resulting in pressure imbalances between tubes and flow instabilities.

The first test, which was a preliminary one because precision instrmentation for measuring pressure drop across the steam generator was not then available, indicated that the actual pressure drops across the unit at various flows are less than those predicted by calculations made at that time. A review and subsequent refinement of the calculations resulted in a revised prediction of the flow-ΔP characteristics to be expected. Results of the preliminary test in October are in agreement with the revised prediction as shown in Figure 9.

In the second test, precision instrumentation for measuring pressure differential was used. Inlet feedwater conditions were 350 F temperature and pressures in the range of 930 to 960 psig. Flows ranged between 27,000 lb/hr and 210,000 lb/hr, the latter being the flow characteristic of 200 Mwt conditions. The test results are given in Figure 9 and are shown to be in agreement with the results of the first test and with the revised prediction of flow-ΔP characteristics.

It is intended next to perform a thermal shock test on No. 3 steam generator in a similar manner to the test of the No. 2 unit reported in APDA-CFE-26, page 21.

* See APDA-CFE-26, page 25.

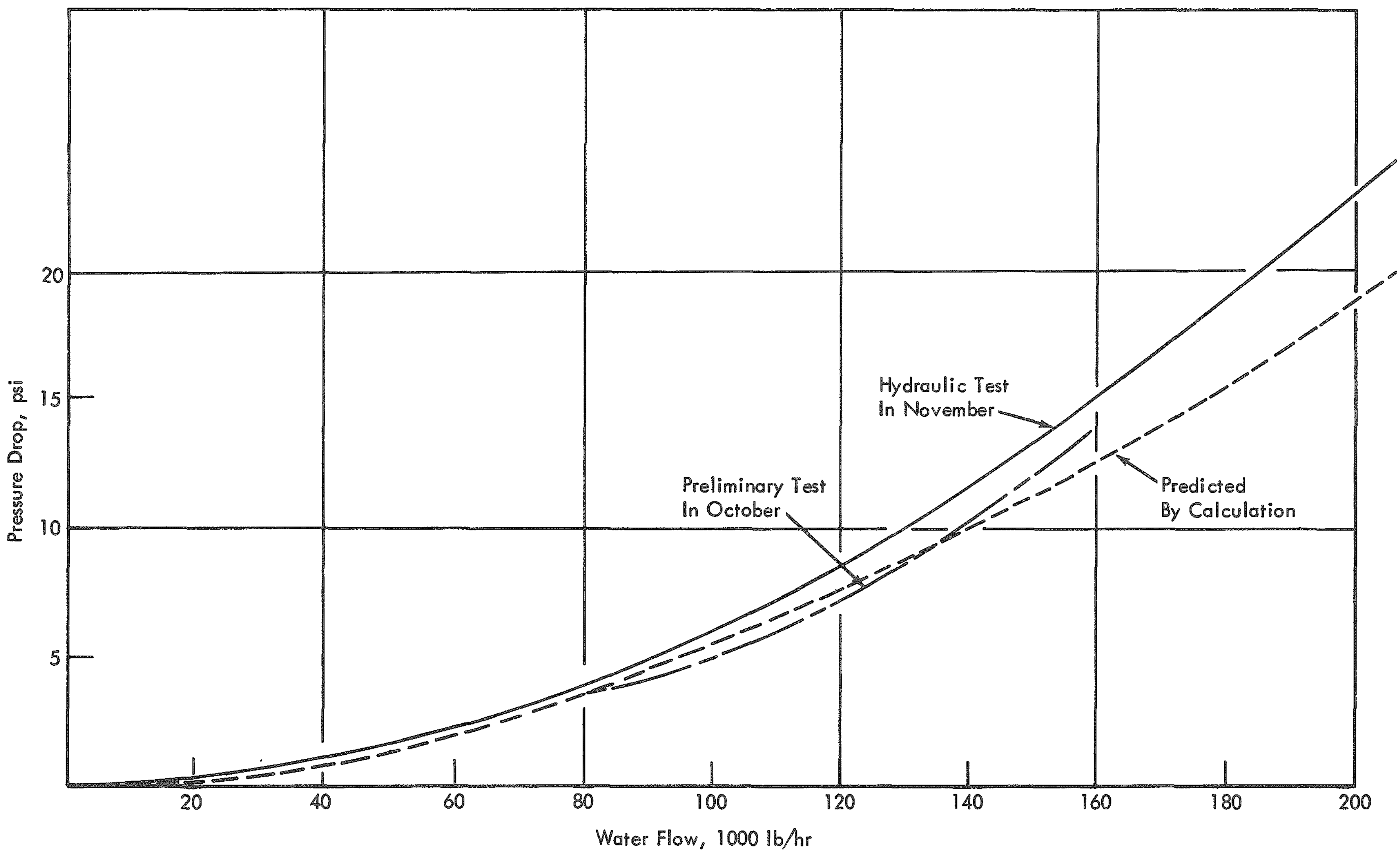

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FIG. 9 FLOW-ΔP CHARACTERISTICS OF THE NO. 3 STEAM GENERATOR WITH TUBE ORIFICES INSTALLED

IV. SPECIAL INVESTIGATIONS

A. INSPECTION OF REACTOR INTERNALS

Features in the interior of the reactor vessel were inspected with a borescope after the siphon drain of the vessel and just before work was started to free the zirconium segments from the cone for removal from the inlet plenum. The outer radial blanket orifice in lattice position N07-P00 was viewed and it was verified that several of the spring fingers are bent. This condition, shown in Figure 7 of APDA-CFE-21, was discovered in April, 1968 during an inspection of the reactor interior following removal of the first zirconium segment. The damage was caused by the bottom of the mirror post bearing on the orifice during the first attempt to photograph the underside of the holddown plate. It is planned to remove and replace this damaged orifice using the same type tools, with extensions, that were used in the initial installation. An outer radial blanket subassembly was inserted in this lattice position during sodium circulation in recent months to ensure that the damaged orifice would not become dislodged from its position.

The exposed wall surfaces of the lower safety rod guide tubes in positions N02-P02, N03-P00, and N02-N02, which are close to the subassembly meltdown region, were visually inspected and found to be free of any evidence of an abnormal condition. Greenish stains had been observed on the surface of the guide tube in position N03-P00 in the April inspection. Apparently, the washing action of the sodium removed the stains.

In the lower plenum, the innermost acorn nuts immediately north and south of the center of the plenum were found in place. These nuts, which are among the many fastening the batten strips covering the joints between sections of the zirconium liner in the inlet plenum, were not accessible for viewing previously because of obstructions by subassemblies and the sweep mechanism. All other acorn nuts were viewed during the April inspection.

The borescope inspection of the south sector of the conical flow guide revealed that the zirconium segment was not attached to the flow guide surface; however, the three fastening screws were still in place. The screw known to have been missing from the southwest sector was not located. It was observed that the lower plenum was considerably cleaner than when viewed in April, probably due to the washing action of circulated sodium during the subsequent refill and clean-up period.

B. INSPECTION OF DAMAGED SUBASSEMBLIES M118 AND M087

In September, core subassemblies M118, M087, and M007 were transferred from the reactor to the Fuel and Repair Building, along with 31 IRB subassemblies.* M118, M087, and M007 were adjacent to the cluster of damaged subassemblies M127, M098, M140 and M122 in the core lattice at the time of the incident of October 5, 1966. These three fuel subassemblies which had been stored in the reactor vessel transfer rotor were inspected in the fuel inspection facility just prior to transfer and were found to be distorted.**

After being steam cleaned and deposited in the cut-up pool in the Fuel and Repair Building, M118 and M087 were again checked for distortion. For this inspection the subassembly was laid horizontally on a 1/4-inch-thick sheet of lead on the material surveillance subassembly loading tray beneath the surface of the water in the cut-up pool. At successive points along the length of the subassembly, a pointed plumb bob was dropped to make indentations in the lead sheet, thus reproducing the outline of the subassembly. The plumb bob was allowed to fall through a tube held against a vertical face of the subassembly wrapper tube at the successive points.

The bow in M118 started at about 10-1/4 inches below the spacer pad and, in one plane of the subassembly wrapper tube, was measured as a 1-inch displacement of the handling head from a centerline through the nozzle. The bow was 7/64 inch at the spacer pad. In the adjacent plane of the wapper tube, the maximum bow was 1/8 inch at the top of the handling head. Subassembly M087 was observed to be straight in one plane while in the other plane distortion started 2 inches below the spacer pad and was a maximum of 0.6 inch at the top of the handling head. The bow was 35/64 inch at the top of the wrapper tube.

C. RADIATION SURVEY IN COLD TRAP ROOM

After entry into the cold trap room to investigate an indicated leak in the bellows of valve V502-1,*** a radiation survey was conducted in the cold-trap room. Figure 10 shows the radiation levels at various points on the outside surface of the cold trap and at about waist elevation elsewhere in the cold-trap room inapproximate locations as shown.

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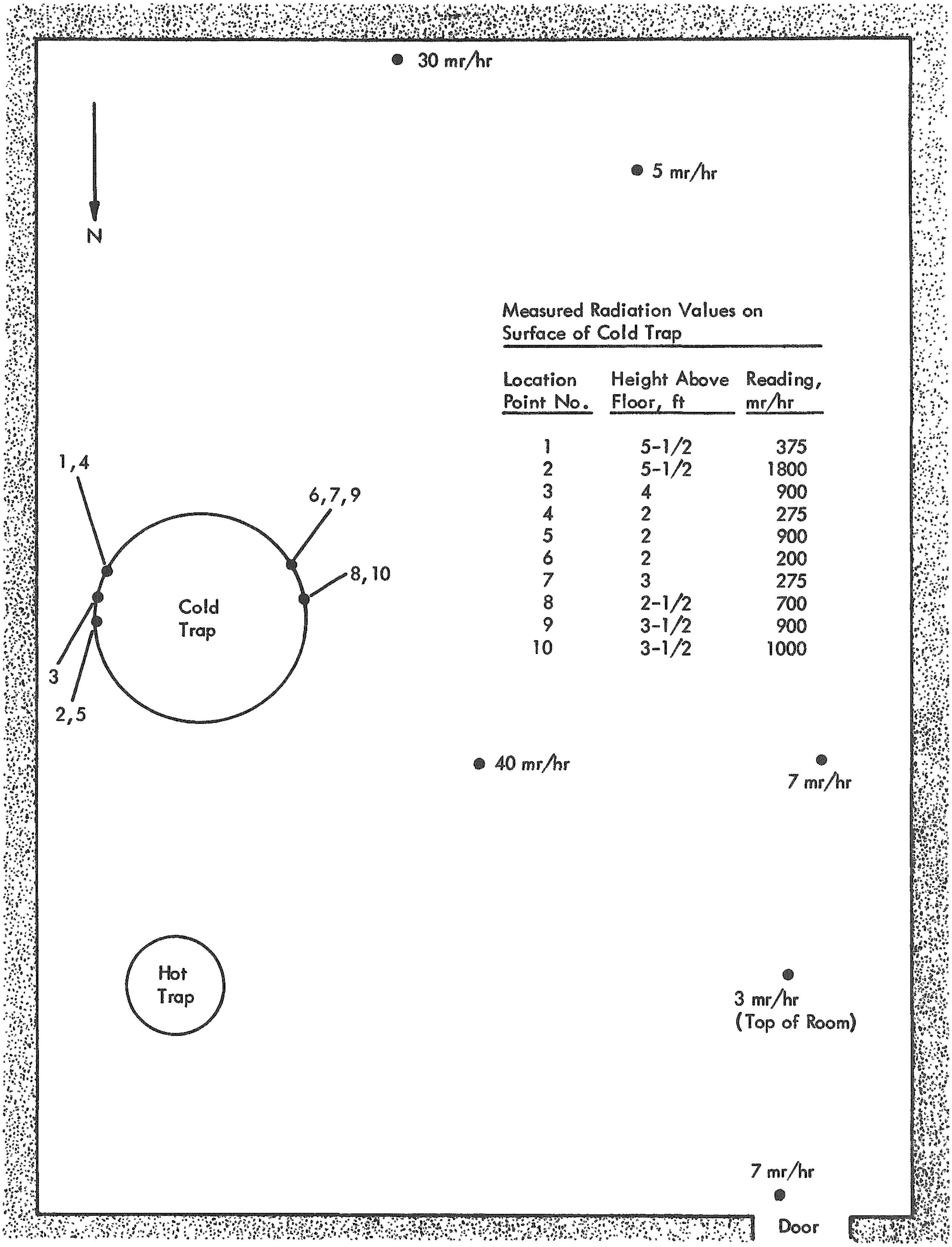
* APDA-CFE-25, page 12.

** See APDA-CFE-24, page 25 and APDA-CFE-25, page 27 for results of this inspection.

*** See APDA-CFE-26, page 40.



Measured Radiation Values on Surface of Cold Trap

Location Point No.	Height Above Floor, ft	Reading, mr/hr
1	5-1/2	375
2	5-1/2	1800
3	4	900
4	2	275
5	2	900
6	2	200
7	3	275
8	2-1/2	700
9	3-1/2	900
10	3-1/2	1000

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FIG. 10 RADIATION SURVEY OF COLD TRAP ROOM

V. HISTORY OF PRIMARY SODIUM VALVE BELLOWS FAILURES

During the life of the plant to date, there have been nine failures of bellows-type sodium valves. The bellows form a seal between the valve body and the moveable valve stem to maintain positive integrity of the sodium system in which the valve is installed. The bellows valve also has a freeze-seal feature which is the elongated valve stem housing maintained near ambient temperature. Any sodium leakage from the interior of the bellows would freeze in the stem housing and prevent the release of sodium to the atmosphere. Figure 11 shows the bellows and freeze seal features.

Table I lists the bellows failures by valve number, the year in which failure occurred, and comments on the nature of failure. The comments are based on metallographic examination of the failed bellows. All of the 500 series valves are in the primary sodium service system and their locations are shown in Figure 12. There are about 30 primary sodium bellows valves. Valve V840 is in the secondary sodium system and FCV 509 is in the NaK cooling system for the primary sodium cold trap. The most recent bellows failures, in valves V502-1 and V516, were reported in APDA-CFE-21, page 39.

A review of the bellows failures to date raised concern for those failures, namely in valves FCV 506, FCV 508, V502-1 and V516, where cracks were initiated from the inside or gas side of the bellows and where carburization was found on the inside surface. It was concluded that the cause of failure in these instances existed prior to the installation of the valve in the sodium system. From what is known of fabrication procedures, it was further concluded that carburizing contamination of the bellows could have occurred during fabrication which could lead to the type of failures experienced. Figure 13 shows a failure in the bellows from valve V516. Photomicrographs showing failures in V516 and V502-1 are given in Figure 14. Presently a program is underway to prepare new specifications for the fabrication of bellows for sodium valve service. These specifications will require ultrasonic cleaning which is not now included in the fabrication procedures.

TABLE 1 - HISTORY OF SODIUM VALVE BELLOWS FAILURES

Valve No.	Year Failed	Comments on Failure
V 840	1961	Fatigue failure, possibly by vibration; no evidence of carburization in area of failure; in service 2 weeks
FCV 506	1961	Intergranular fatigue cracks not typical of Type 321 stainless steel bellows material but could be expected with the high carburization found. Carburization was evident on both inside and outside surfaces. In service 8 months
FCV 508	1961	Comments the same as for FCV 506 except that cracks appear to have been initiated from inside surface. In service about 1 year.
FCV 505	1962	Fatigue failure; cracks not intergranular; only slightly carburized; in service 33 months
V 514	1962	Mechanical abrasion; in service 39 months
FCV 506	1965	Crack initiated at inside wall; working of crack edges indicated cracks extended through bellows wall before service; in service 4 years and 4 months since last failure, above
TCV 509	1967	Valve returned to vendor for repair; in service 8 years
V 502-1	1968	Crack initiated from inside bellows; it is likely that crack was produced when bellows was formed and that stress relief eliminated evidence of cause; in service 8-1/2 years
V 516	1968	Same comment as for V502-1; in service 8-1/2 years

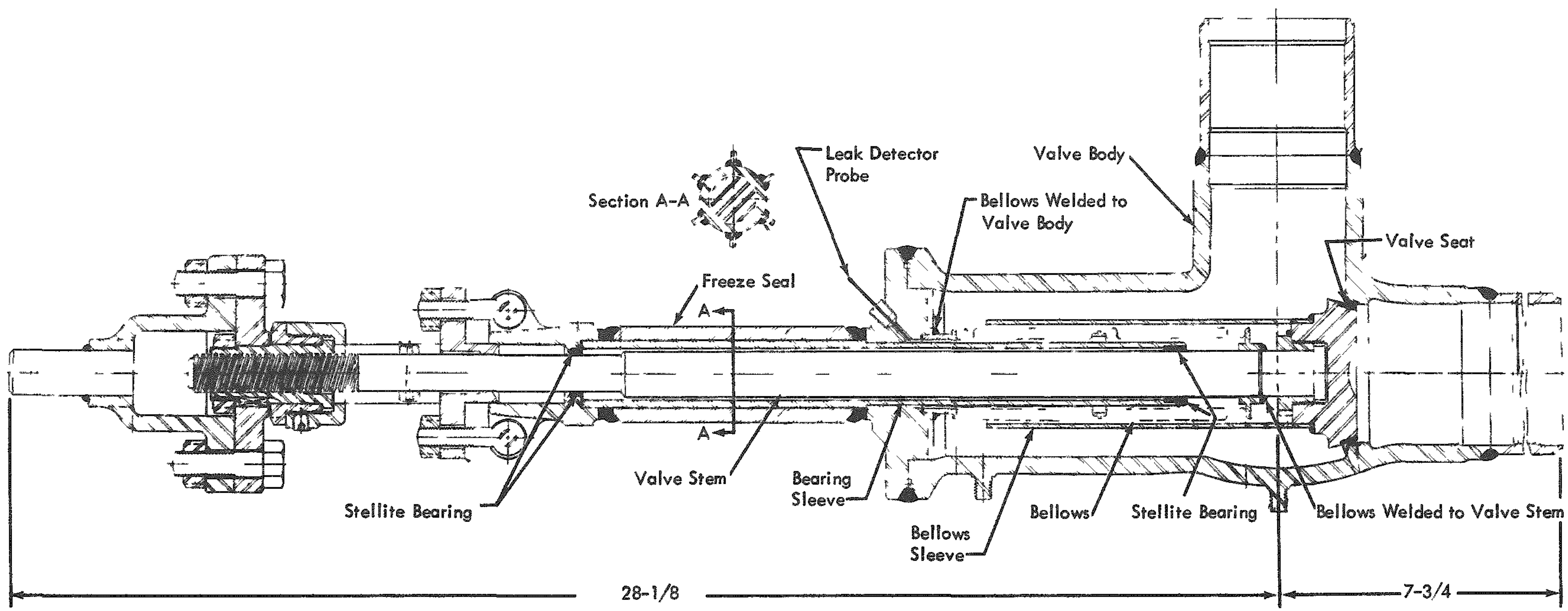

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FIG. 11 DETAIL SKETCH OF TYPICAL PRIMARY SODIUM SYSTEM BELLOWS SEAL VALVE

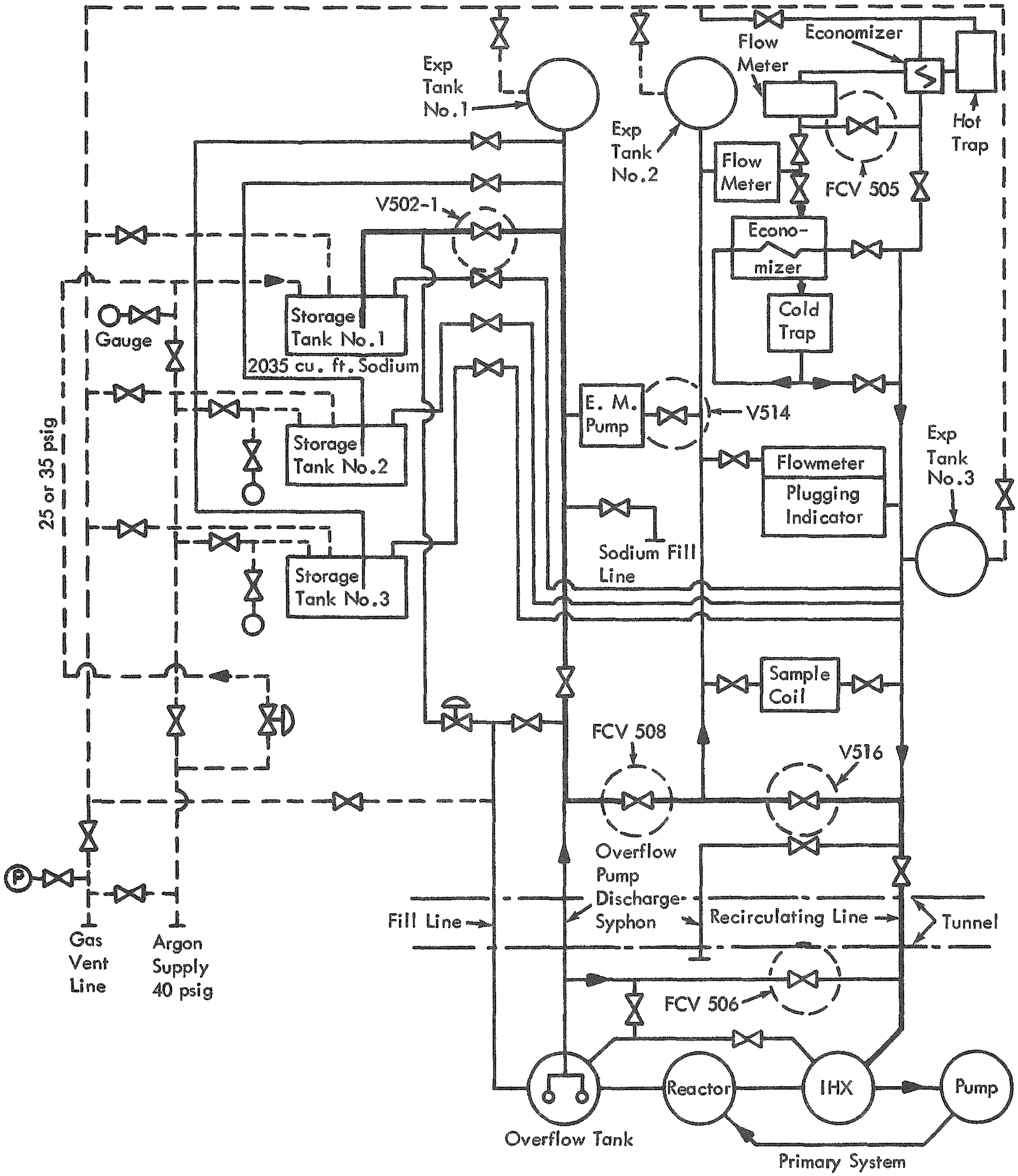

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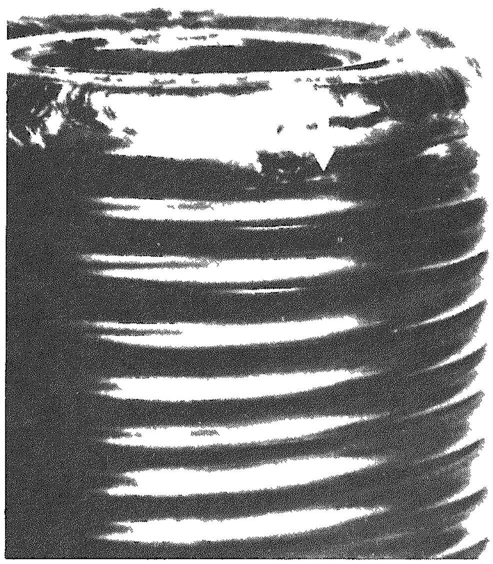
FIG. 12 SCHEMATIC DIAGRAM OF PRIMARY SODIUM SERVICE SYSTEM



Crack at Base of Convolution (Arrow)

2X

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FIG. 13 FAILURE IN BELLOWS FROM VALVE V516

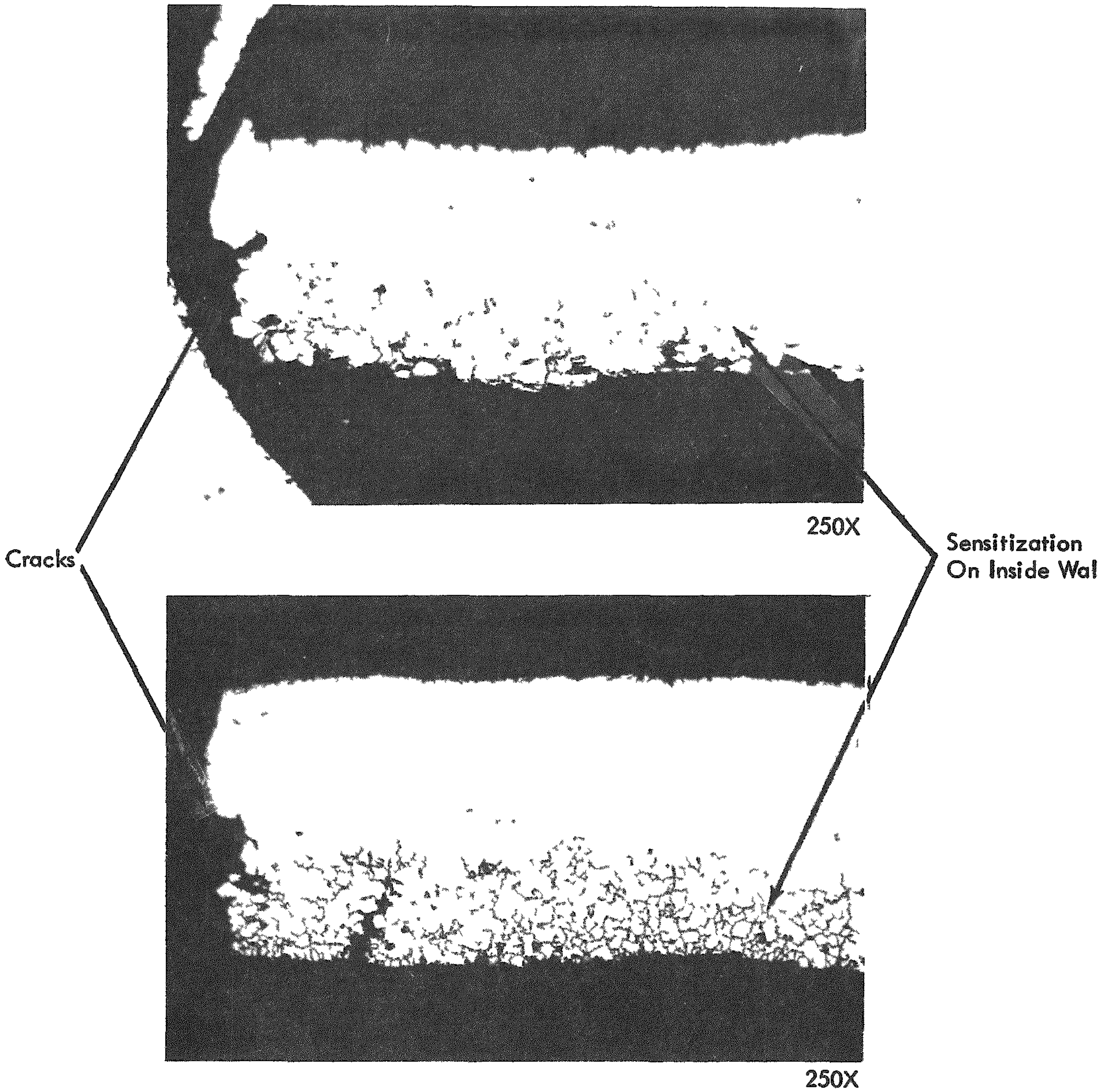

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FIG. 14 FAILURES IN BELLOWS FROM VALVES V516 (TOP) AND V502-1

VI. MAINTENANCE

A. BUBBLE LEAK TEST OF PLUG WELDED TUBES IN NO. 1 STEAM GENERATOR

As reported on page 22 of APDA-CFE-25, leaks were found in 3 of the 1200 tubes in the No. 1 steam generator in September. Two of the leaks were in the recently completed internal bore welds at the tube-to-tubesheet joints in the water manifold. The other leak was determined to be in a convolute section of a tube at a point 10 to 10-1/2 feet below the steam manifold. The leaks were eliminated by plug-welding both ends of the tubes.

In November, the integrity of the plug welds was leak-tested by the bubble method. The tube side of the steam generator was filled with water to levels slightly above the tubesheets and the shell side was pressurized with argon gas at 35 to 40 psig. There was no leakage through the plug welds as evidenced by the lack of gas bubbles rising from the tubes. There was no sodium in the steam generator.

B. REPLACEMENT OF STEAM GENERATOR MANIFOLD GASKETS

The water and steam manifold gaskets in the Nos. 2 and 3 steam generators were replaced in October and November to eliminate the manifold-to-atmosphere leaks that occurred when the units were operated with sodium and water for tests. The No. 1 steam generator also developed manifold leaks at that time, but there has been no gasket replacement as yet because repair effort on this unit has been concentrated on the tube leaks. The replacement gaskets for the Nos. 2 and 3 steam generators are of soft iron like the original and were annealed as specified. It was discovered that the original gaskets and the available replacements had not been annealed as specified.

During the thermal shock tests of the No. 2 steam generator at the end of October, a leak to atmosphere through the replacement gasket at the water manifold cover joint occurred. It is the present intention to replace the water manifold gaskets with copper, which is the originally specified material for this application. Soft iron was substituted because copper is a long delivery item. The replacement copper gaskets are expected to be available shortly.

The No. 2 steam generator is to be drained of sodium and disassembled for gasket replacement. A thermal shock test of the No. 3 unit will be conducted before dissembly and gasket replacement. The No. 1 steam generator will remain disassembled pending delivery of the copper gaskets; the test thermocouples which were removed for repairs on this unit are being reainstalled.