ENRICO FERMI ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

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

R. H. Costello
Assistant Project Engineer

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APPROVED Charles E. Branyan
C. E. Branyan
Fermi Project Engineer
APDA

PREPARED BY
Atomic Power Development Associates, Inc.
with the Cooperation of
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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 plant 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 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 patch was removed from the primary pipe elbow in the 14-inch sodium inlet line, the object-retrieval device was inserted through the elbow penetration and the spine-type manipulator through the support plates to begin operations to remove the object from the reactor core inlet plenum. Equipment troubles and tool maneuvering difficulties hampered efforts to stuff the object into the retrieval duct. The object was too wide to fit completely into the retrieval duct; therefore, the retrieval gripper was modified to ensure that it would firmly grasp the object.

About 2-1/2 weeks after the patch was removed from the elbow, the object was withdrawn through the 14-inch inlet line. Metallographic examination was used to estimate the amount of hydriding that had occurred (about 400 ppm). Spectral analysis of a sample of the object verified that it was zirconium. Circular score marks observed on the surface of the object suggest that it had been forced upward against the underside of the lower support plate holes.

The patch was rewelded in the elbow to restore the integrity of the primary sodium system. Engineering efforts were continued in the program to remove the five remaining zirconium segments. Consideration of the approaches for screw removal was narrowed to the machining, chiseling, and arc melting methods. Preliminary metal-working tests were performed to determine energy requirements of these methods.

The spherical mirror assembly, originally installed in the reactor for viewing of holddown fingers, was removed from the top support plate, where it had fallen after being broken off its support post.

II. REMOVAL OF OBJECT FROM INLET PLENUM

A. BACKGROUND

The object was discovered in the inlet plenum of the reactor core on September 11, 1967. It is believed to have caused the melting of two fuel subassemblies on October 5, 1966, when it was lifted by the sodium flow to a position against the underside of the lower support plate. Initial investigations and effort directed toward removing the object proceeded along two parallel paths: (1) cutting the object into pieces and removing the pieces through the core support plates and (2) retrieving the object intact through the 14-inch sodium inlet line. Cut-up and removal tools and devices were developed, aided by a large number of photographs of the object. Attempts to obtain a sample of the object to establish its identification were unsuccessful.

In late December, efforts were concentrated on preparations to remove the object intact through the 14-inch sodium inlet line. The 14-inch pipe elbow to be penetrated and the surrounding work area were cooled, and a patch was cut and removed from the secondary containment to permit access to the primary pipe elbow. Mock-up tests demonstrated that two special devices, an all-purpose manipulator tool and a retrieval device, were needed to remove the object through the 14-inch sodium inlet line. By the end of February, these proven removal devices were available for use in the reactor, and the stage was set for penetration of the 14-inch sodium inlet line and removal of the object.

In late January, the object was tentatively identified as one of six 0.040-inch-thick zirconium plates originally installed on the sloping sectors of the conical flow guide in the reactor core inlet plenum. Various approaches were investigated for the removal of the remaining five zirconium segments from the flow guide surfaces, with emphasis on first removing the fastening screws. Engineering effort continues to be directed toward removal of the remaining five zirconium segments, with some consideration given to fixing them in place.

Details of the investigation of the object and the five remaining zirconium segments are given in previous reports starting with APDA-CFE-14.

B. PENETRATING THE 14-INCH SODIUM INLET LINE

On March 1 work was started on the task of penetrating a primary pipe elbow in the 14-inch sodium inlet line to provide access for inserting the object-retrieval device* into the reactor core inlet plenum. The plan was to operate the spine-type manipulator* through the core support plates to feed the object to the retrieval device which, in turn, would remove the object through the 14-inch line. Figure 1 depicts the use of the spine tool to grasp the object and to present it to the object-retrieval gripper prior to removing the object from the reactor. The primary pipe elbow penetrated is located immediately outside the secondary shield wall, and is about 35 feet from the inlet plenum. Details of penetrating the elbow are given below.

1. Removal of Patch from the Elbow

The penetration of the primary pipe elbow was accomplished by cutting a 6-inch by 7-inch patch from the outside radius of the elbow. During the initial cutting, the work was carried out under normal atmosphere to facilitate the handling of tools. Final penetration of the pipe was made following installation of the glove box** to maintain the primary system argon atmosphere.

A 1/8-inch-thick nylon grinding wheel was used to make the first cut to a depth of 1/4 inch in the 3/8-inch-thick wall of the elbow. In a subsequent cut, a 1/4-inch-thick Carboloy rotary file outlined the patch with a U-type weld prep and left about 1/32 inch of metal. Several small pinholes that appeared were temporarily covered by Silastic cement applied liberally to the cut as shown on Figure 2. Figure 3 shows the locating plates welded to the patch for the purposes of preventing the patch from falling inside the pipe and for locating the patch for rewelding after object removal. The glove box*** was then clamped to the adapter which enclosed the elbow patch, and after the glove box was purged with inert gas and checked for leaks, the remaining metal in the cut was chiseled through to free the patch. When the patch was removed, it was observed that the internal surfaces of the pipe and elbow that could be seen appeared to be in the as-installed condition and did not appear to be wetted with sodium. Figure 4 shows the glove box installed and the object-retrieval device cable inserted through it.

2. Radiation and Contaminant Monitoring

The radiation level at the surface of the primary pipe elbow before cutting was 0.8 mr per hour. The contaminant content in the air in the tent-enclosed work area was 7.95×10^{-11} microcurie per cm^3. The maximum permissible concentration of krypton-85, the major isotope in the cover gas, is 1×10^{-5} microcurie per cm^3. When the first pinhole appeared during

* See Section II of APDA-CFE-15 through 19 for pervious information.
** See APDA-CFE-18, 19, Section II.
*** See APDA-CFE-18, page 15, for a description of the glove box.

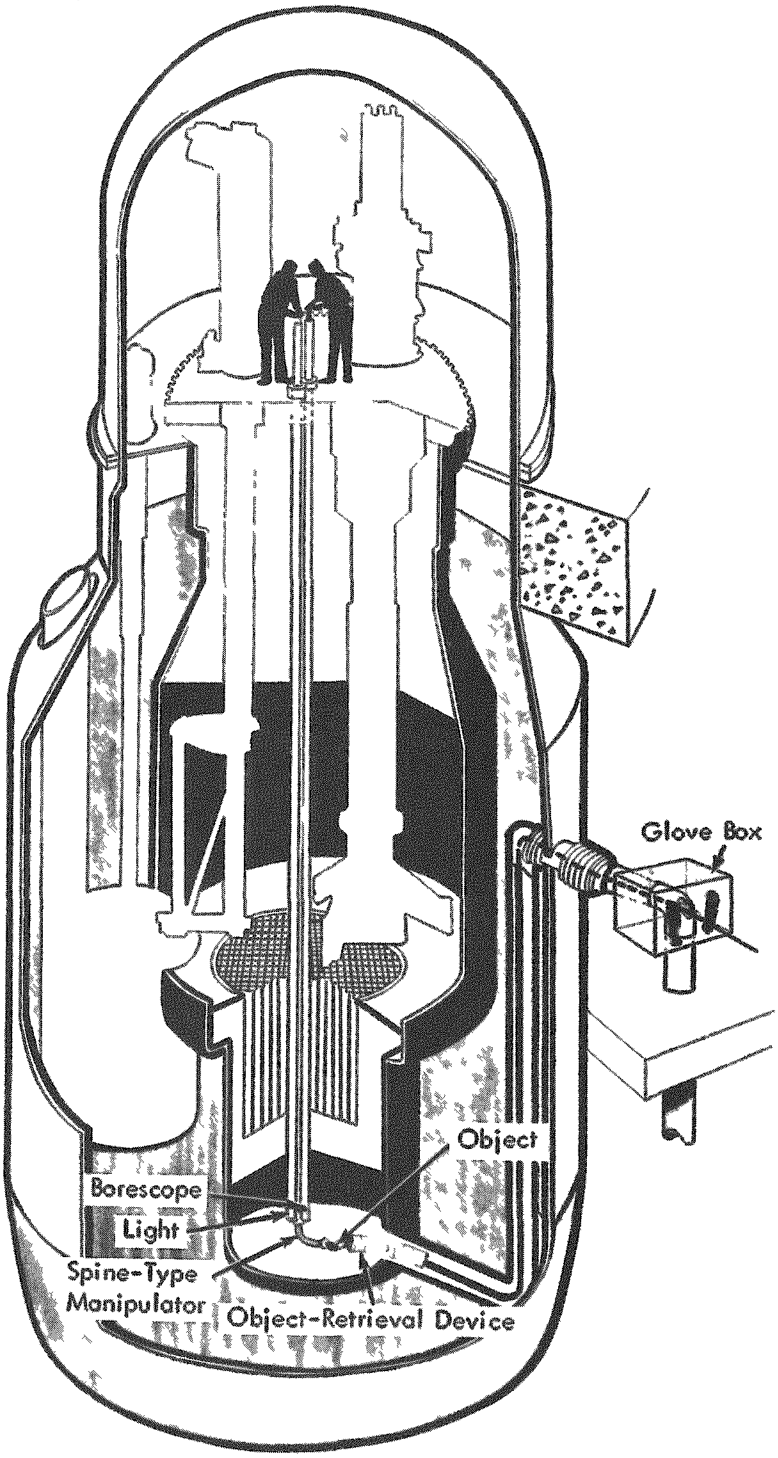

FIG. 1 REMOVING OBJECT FROM REACTOR CORE INLET PLENUM

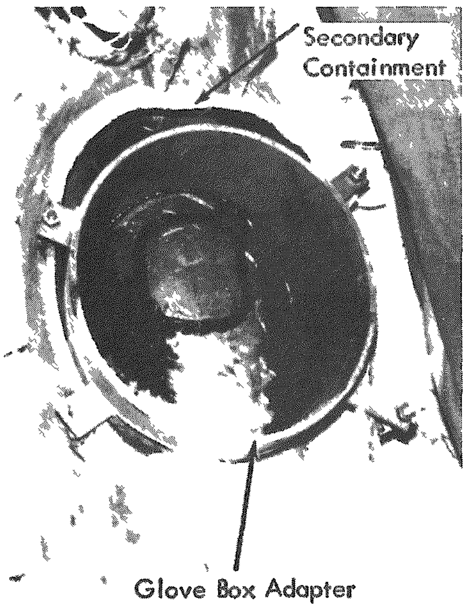

FIG.2 SILASTIC SEALANT ON PARTIALLY GROUND OUT PATCH ON PRIMARY ELBOW

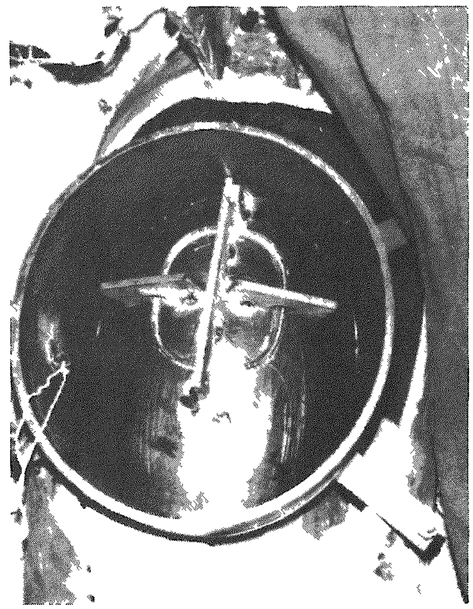

FIG.3 LOCATING PLATES WELDED TO PATCH

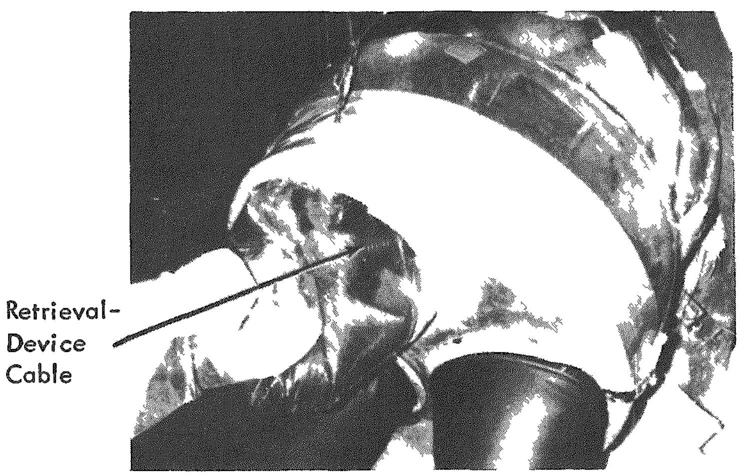

FIG.4 GLOVE BOX ENCLOSING PENETRATION THROUGH PRIMARY ELBOW

cutting of the elbow patch, a radiation level of 2 mr per hour was measured at the cut. As more pinholes appeared, the radiation level rose to 13 mr per hour at the cut. When the partial cut was covered with Silastic cement and tape, a level of 1 mr per hour was measured at the tape.

After the removal of the patch, radiation levels up to 150 mr per hour, principally beta radiation, were measured at the elbow penetration and at the surface of the plastic-enclosed patch. Tests of sample smears from the patch in a gamma spectrometer showed the major contaminants to be cesium-137, a gamma emitter and strontium-90, a beta emitter. The contaminant content in the atmosphere in the work area after removal of the patch was 1.04×10^{-9} microcurie per cm^3. Dosimeters worn by personnel in the work area around the 14-inch pipe elbow showed that the maximum cumulative radiation exposure for any one person during an 8-hour period was 0.01 mr.

3. Object Grasped by Spine-Type Manipulator

Before the breakthrough into the primary pipe was made, the spine-type manipulator was inserted through the support plates into the core inlet plenum to grasp the object in readiness for feeding into the object-retrieval device. Figure 5 shows the spine tool gripper in the reactor inlet plenum nearly in position to grasp the object. The PRDC borescope* and the Sunscope** provided vertical and horizontal viewing and the new articulated lamp*** provided the lighting. The four devices were inserted through lattice positions NO4-NO1, NO4-NO2, NO5-NO1, and NO5-NO2 (see Figure 18) in the northeast sector of the reactor, where the object was located and where it was first discovered. Figure 6 shows the positions of the object and the above devices in relation to the inlet to the plenum from the 14-inch inlet line of the No. 1 loop.

4. Inserting the Object-Retrieval Device

The object-retrieval device was inserted through the elbow penetration and into the 14-inch pipe immediately after the elbow patch was removed. Figure 7 shows the retrieval device cable passing through the elbow penetration. The entire retrieval device was enclosed in a polyvinyl chloride plastic containment bag which, after being purged with inert gas, was connected to the tool port in the glove box aluminum faceplate.

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* See Section IV.A of APDA-CFE-18

** See pages 9, 11, and 21 of APDA-CFE-13 for information about the Sunscope.

*** APDA-CFE-18, 19.

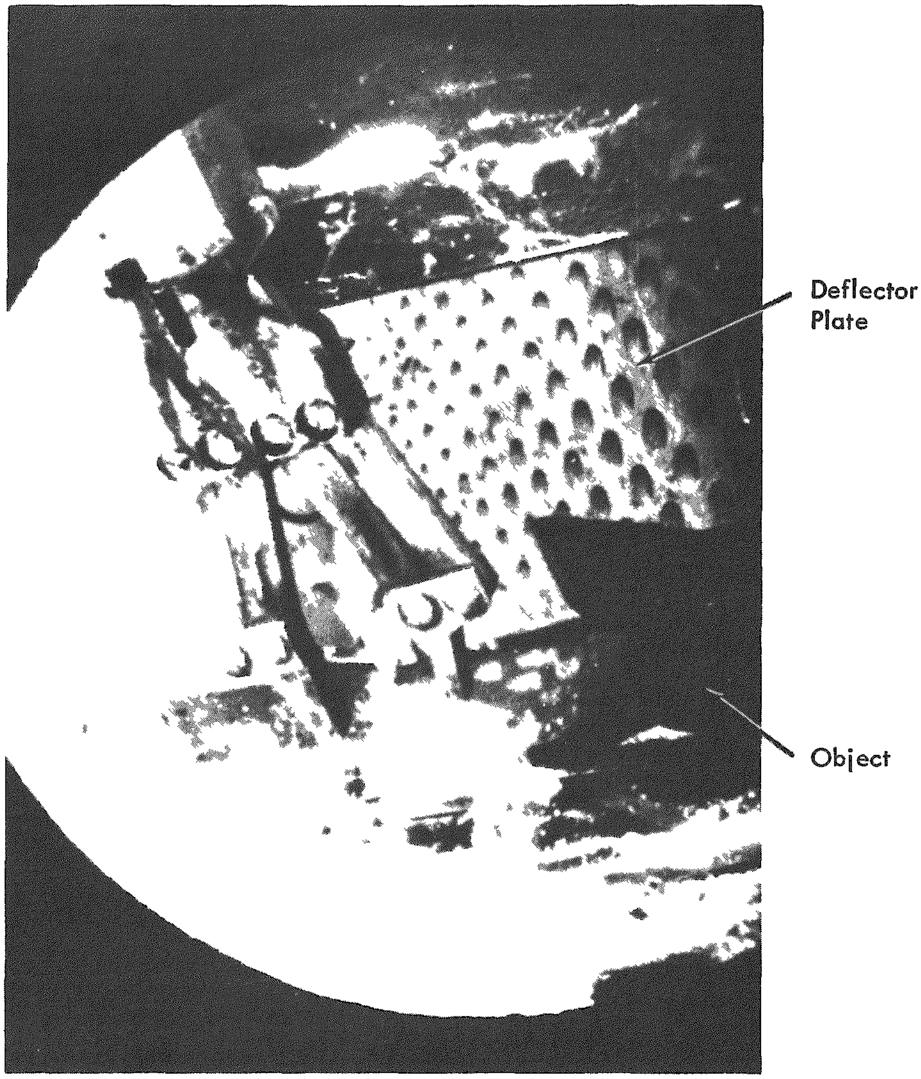

FIG.5 SPINE-TOOL GRIPPER REACHING TO GRASP OBJECT IN INLET PLENUM

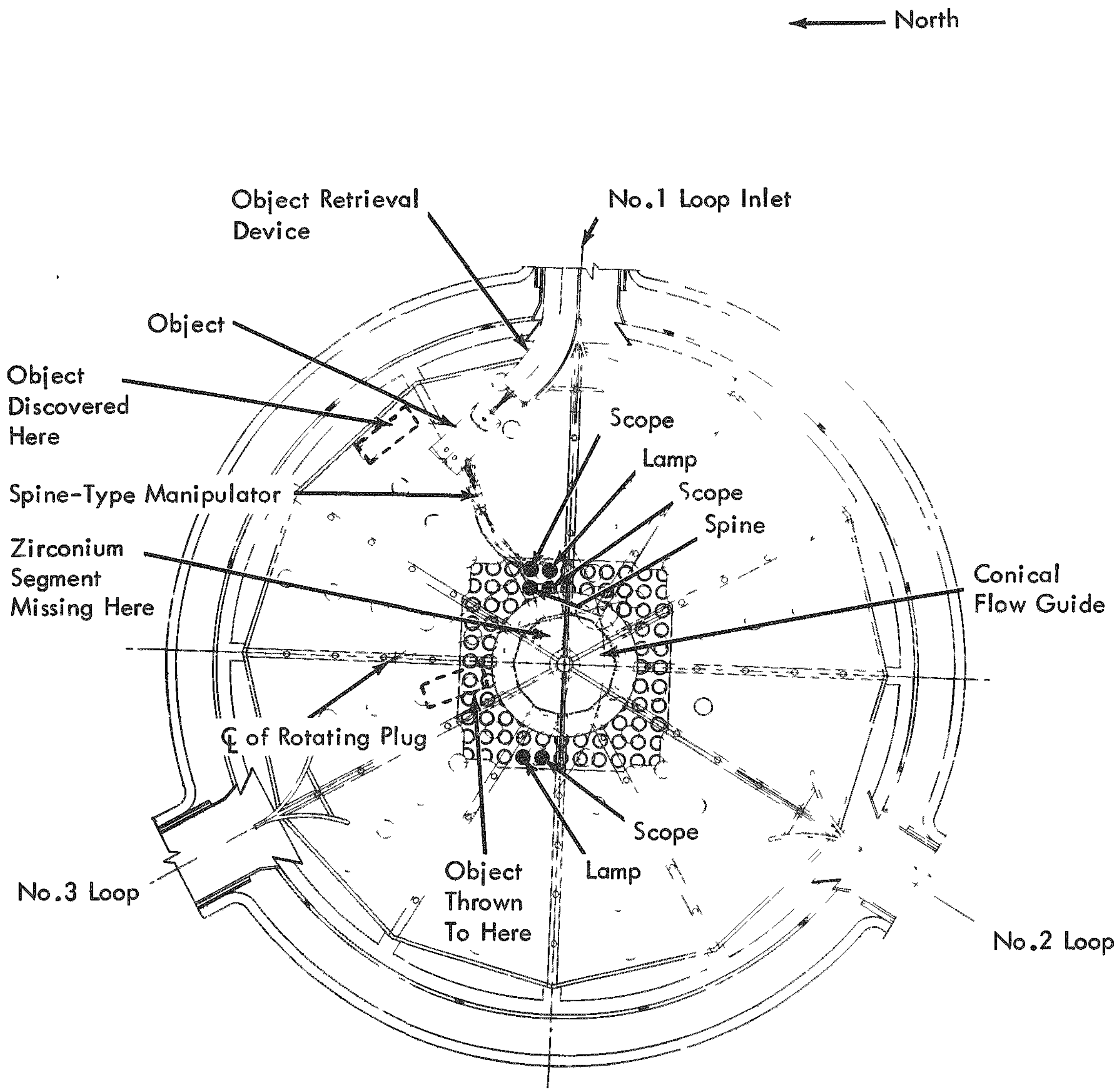

FIG.6 OBJECT-REMOVAL OPERATIONS

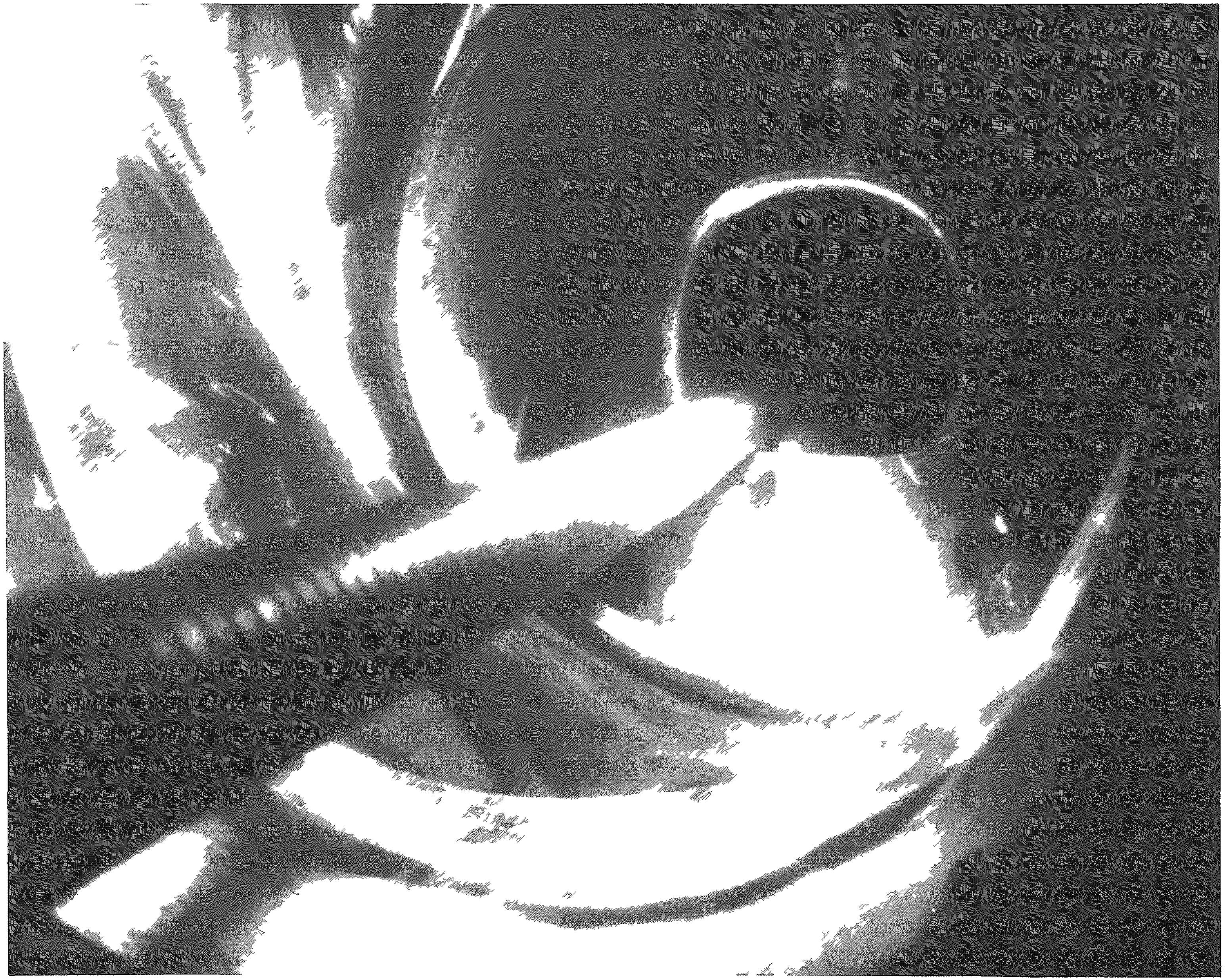

FIG.7 OBJECT-RETRIEVAL DEVICE CABLE INSERTED THROUGH PRIMARY ELBOW PENETRATION

The retrieval device was maneuvered through the 14-inch pipe without difficulty until the leading edge of the flexible duct reached the shield ring in the reactor inlet nozzle, as determined by the length of cable payed out. After several hours of maneuvering, the retrieval device failed to move past the shield ring, and it was, therefore, withdrawn from the inlet pipe. The front few inches of the flexible duct was found to be sharply bent, apparently enough for the duct to make contact with the protruding shield ring and prevent entry of the device through the inlet nozzle and into the plenum. It is possible that the duct was inserted too rapidly and that it rammed the shield ring, causing the sharp bend at the end of the duct. The retrieval device was withdrawn at a rate of 5 to 8 feet every 20 minutes to permit cooling and avoid damage to the polyvinyl chloride containment bag because of excessive temperature. The device, when withdrawn, was kept in the containment bag in an inert atmosphere by snaking the bag and tool cable back off the elevated platform adjoining the inlet pipe to the bottom floor of the containment building. The elbow penetration was temporarily sealed by a Cranite patch firmly taped to the elbow.

C. OBJECT-RETRIEVAL OPERATION IN INLET PLENUM

The object-retrieval device was successfully maneuvered into the inlet plenum on the second try, and the spine-type manipulator fed the object to the retrieval gripper. It was hoped that the removal of the object would take only a few hours and that the patch could be immediately rewelded into the primary pipe elbow. Unfortunately, due to two unsuccessful retrieval attempts and intervening equipment troubles and corrective measures, it was not until March 22, seventeen days after the patch was removed from the primary pipe elbow, that the object was finally withdrawn from the reactor.

1. First Attempt to Retrieve the Object

In the first retrieval attempt, the retrieval duct lay in a large arc with a tilted gripper trolley and an off-center gripper. The entry of the object into the duct, then, was obstructed by the end of the duct and the object slipped from the grasp of the gripper. Before another attempt could be made, the viewing equipment developed trouble: the lamp burned out and the scope lenses became blurred from sodium particulate and condensation, necessitating disassembly and clean-up of the device; leaks developed at the fused joints of the glove box bellows, and the bellows structure was replaced by a one-piece sheet of the same polyvinyl chloride plastic. During this and other periods of inactivity in the reactor, the retrieval device was withdrawn into its containment bag, and the elbow penetration was sealed by a temporary Cranite patch.

2. Second Attempt to Retrieve the Object

In the second retrieval attempt, it became apparent that the object was too wide to fit easily into the retrieval duct; i. e., only about two-thirds of the object could be pulled into the duct and then the object slid from the grasp of the gripper and fell out of the duct. This too was evidence that the retrieval gripper could not be depended upon to maintain a firm grasp on the object.

Further maneuvering in an attempt to force the duct over the object created additional difficulties. The duct was pushed further into the plenum, backing the object against the conical flow guide. When it was observed that the duct was beginning to hump and bend, the device was retracted slightly to straighten out the bends. As the duct started to straighten out, it twisted instead, throwing the object over into the northwest sector of the plenum, out of range of the light and spine-type manipulator (see Figure 6). It was then necessary to enter different support plate holes to locate the object and move it back to the northeast sector. Since there were only two accessible support plate holes in the northwest quadrant, PO5-NO1 and PO5-NO2, a scope was inserted in one position and the old articulated lamp (see Figure 8) in the other. A 1/2-inch-diameter, 5-inch-long rod was welded to the end of the lamp structure* so that this device could also serve to move the object. The object was successfully batted back in hockey stick fashion into the northeast sector of the plenum and without benefit of viewing, because the lamp failed just prior to the operation.

3. Modification of Retrieval Gripper

The retrieval gripper was replaced by another Ameray gripper of modified design to ensure a secure grasp of the object, inasmuch as it was realized that only part of the object could be lodged in the retrieval duct during removal. As can be seen in Figure 9, the jaw opening was increased from 1-1/4 inches to 2-5/8 inches and three fang-like prongs were welded to the jaw faces, two on one jaw and one on the other. At the same time, the spring fingers at the end of the retrieval duct were replaced by another set of stiffer construction. The finger thickness was increased from 0.011 inch to 0.015 inch and the finger width was increased from 1/2 inch to 9/16 inch. The new set of spring fingers, which it was hoped would assist in holding the object firmly, also are shown in Figure 9.

* See Section II. B of APDA-CFE-16.

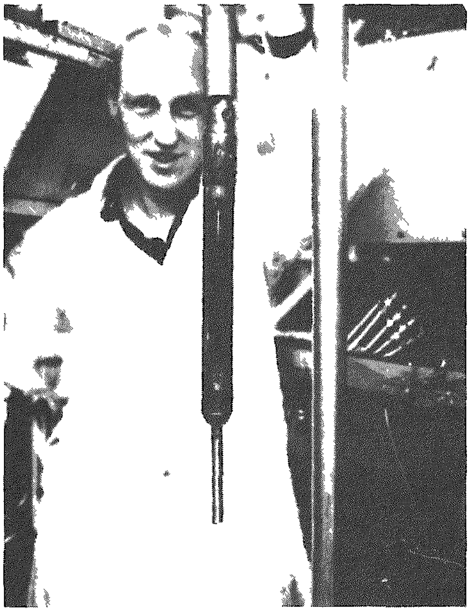

FIG.8 FIVE-INCH LENGTH OF ROD WELDED TO OLD ARTICULATED LAMP TO MANEUVER OBJECT

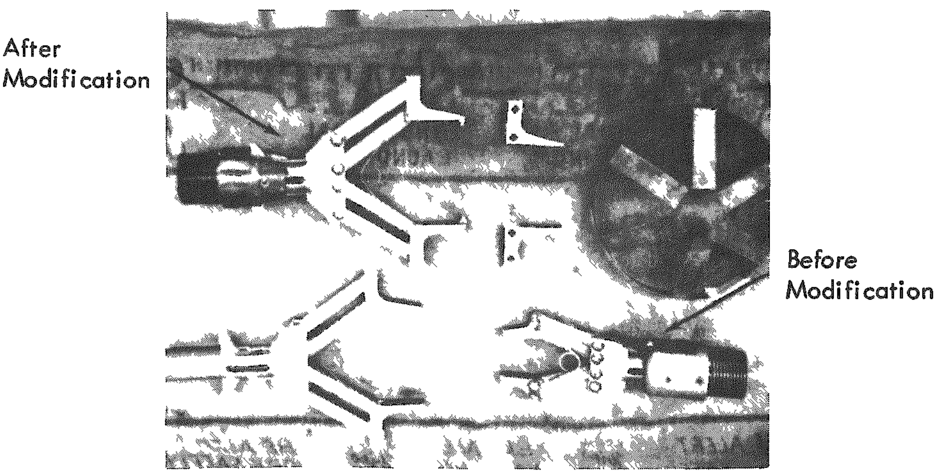

FIG.9 RETRIEVAL GRIPPER AND SPRING FINGERS MODIFIED

4. Removal of the Object

The new retrieval gripper and a modified manner of actuating the gripper, which first takes the slack out of the gripper-actuating cable and ensures a stronger bite, are believed to be the principal reasons for success in the third attempt at removal of the object. Simultaneous tugging on the nose of the object with the retrieval gripper and pushing with the spine-tool gripper in the plenum succeeded in forcing all but about 2-1/2 inches of the object's length into the retrieval duct. Figure 10 shows the spine-tool gripper feeding the object to the retrieval duct. The duct was then withdrawn through the 14-inch pipe at the rate of about 5 feet every 20 minutes. When the object appeared in the glove box, radiation levels of 200 mr per hour beta and 15 mr per hour gamma were measured a few inches from the glove box surface. When the object was withdrawn into the retrieval device containment bag, it was released from the retrieval gripper which had a secure bite on the object. The object, contained in a plastic bag in an inert atmosphere, was placed in a 5-gallon steel container and then transferred to the health physics laboratory for photographing, decontamination, and general examination.

D. PHOTOGRAPHING AND EXAMINATING OF THE OBJECT

1. Photographing the Object

Figure 11 shows three photographs which are representative of many photographs taken of the object before cleaning. Comparison should be made of these photographs with those shown in Figures 2, 3, 4, and 5 of APDA-CFE-16, which were taken of the object in the inlet plenum in November 1967. Figures 12, 13, and 14 are representative of the many photographs taken of the object after cleaning.

2. Gamma Analysis of Deposit on the Object

Before the object was cleaned, a sample of the deposit on it was removed for analysis in the gamma spectrometer. Results showed the major radioactive isotope to be cesium-137. After being cleaned with alcohol and then steam, and after being decontaminated in an ultrasonic cleaner, the radiation level on the surface of the object was 2 mr per hour.

3. Metallurgical Analysis

A 1-1/2-inch length of a tab, removed from the object (see Figure 12) for metallurgical analysis, was verified by spark analysis to be zirconium. When etched and examined for hydriding, the sample of the object was found to contain 400 ppm hydrogen, in a preliminary analysis that is subject to verification by further analysis at Battelle Memorial Institute. The result

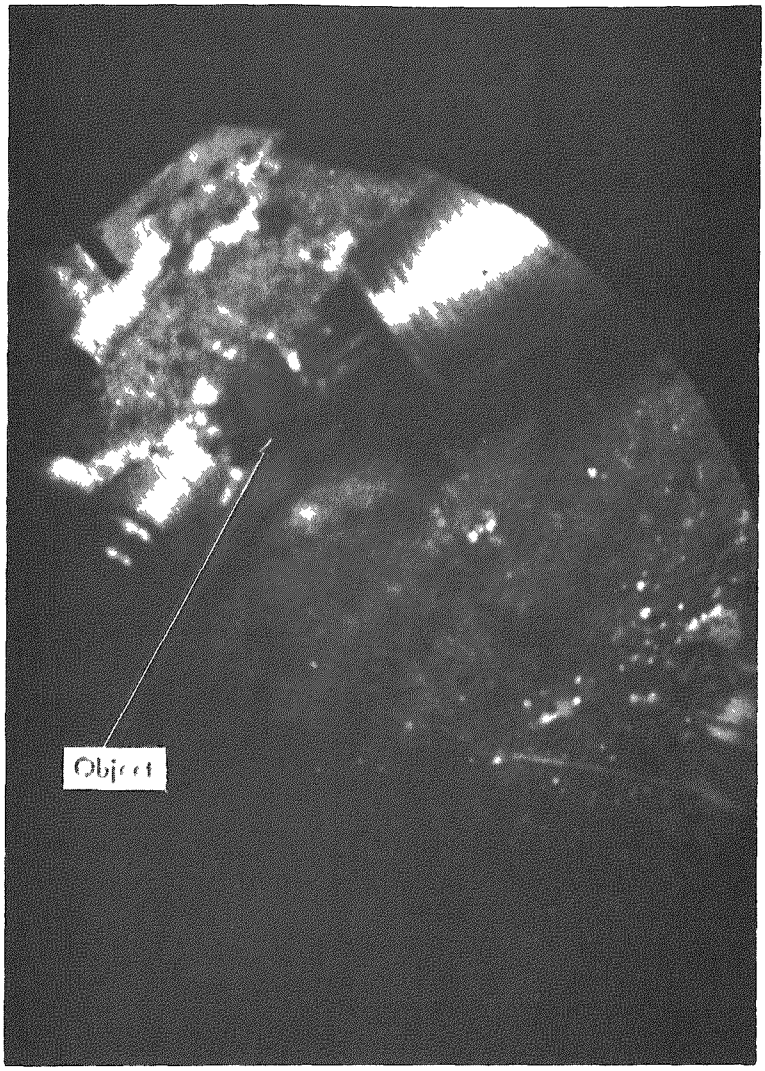

FIG.10 SPINE-TOOL GRIPPER FEEDING OBJECT TO RETRIEVAL GRIPPER IN INLET PLENUM

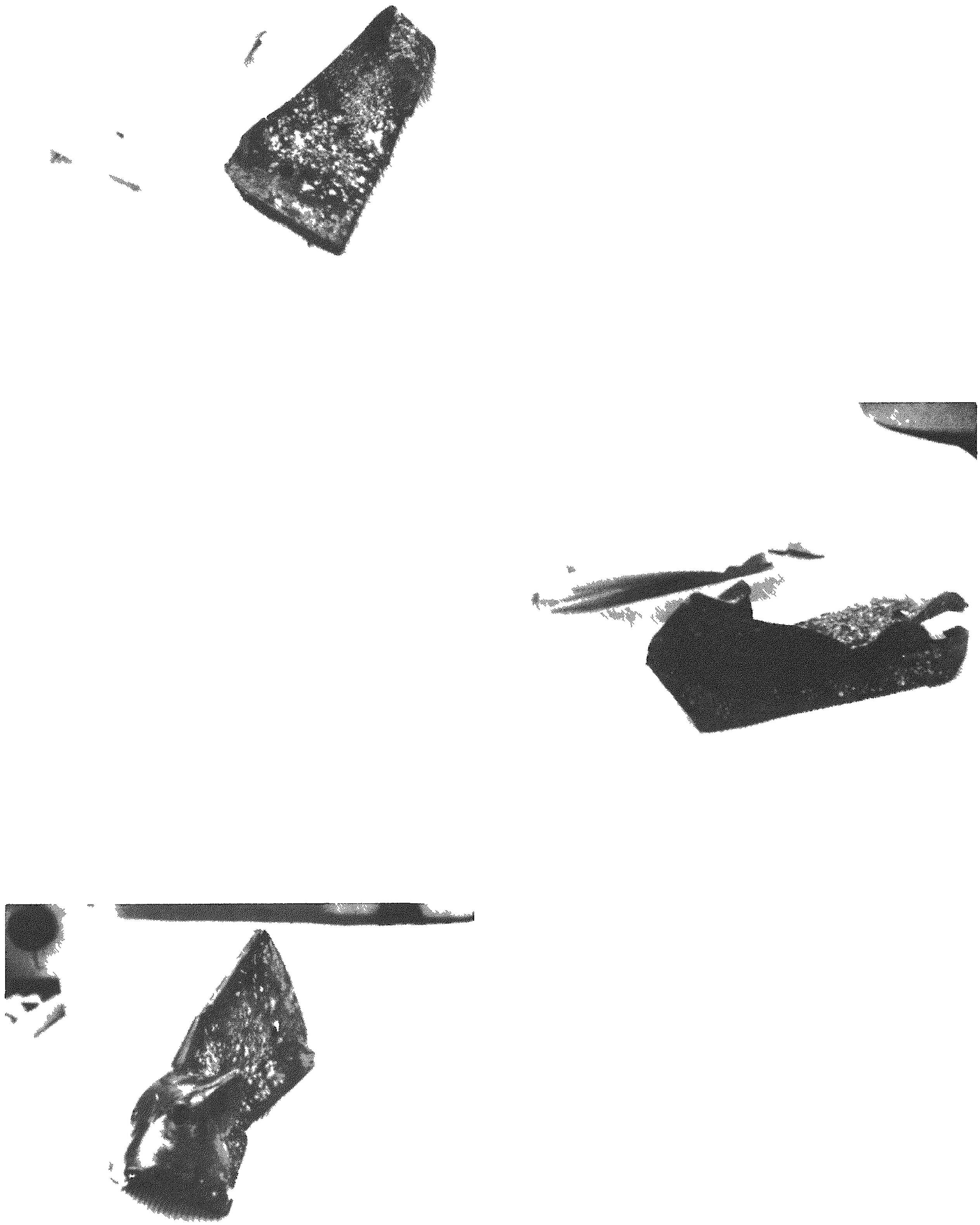

FIG.11 THREE VIEWS OF RETRIEVED OBJECT BEFORE CLEANING

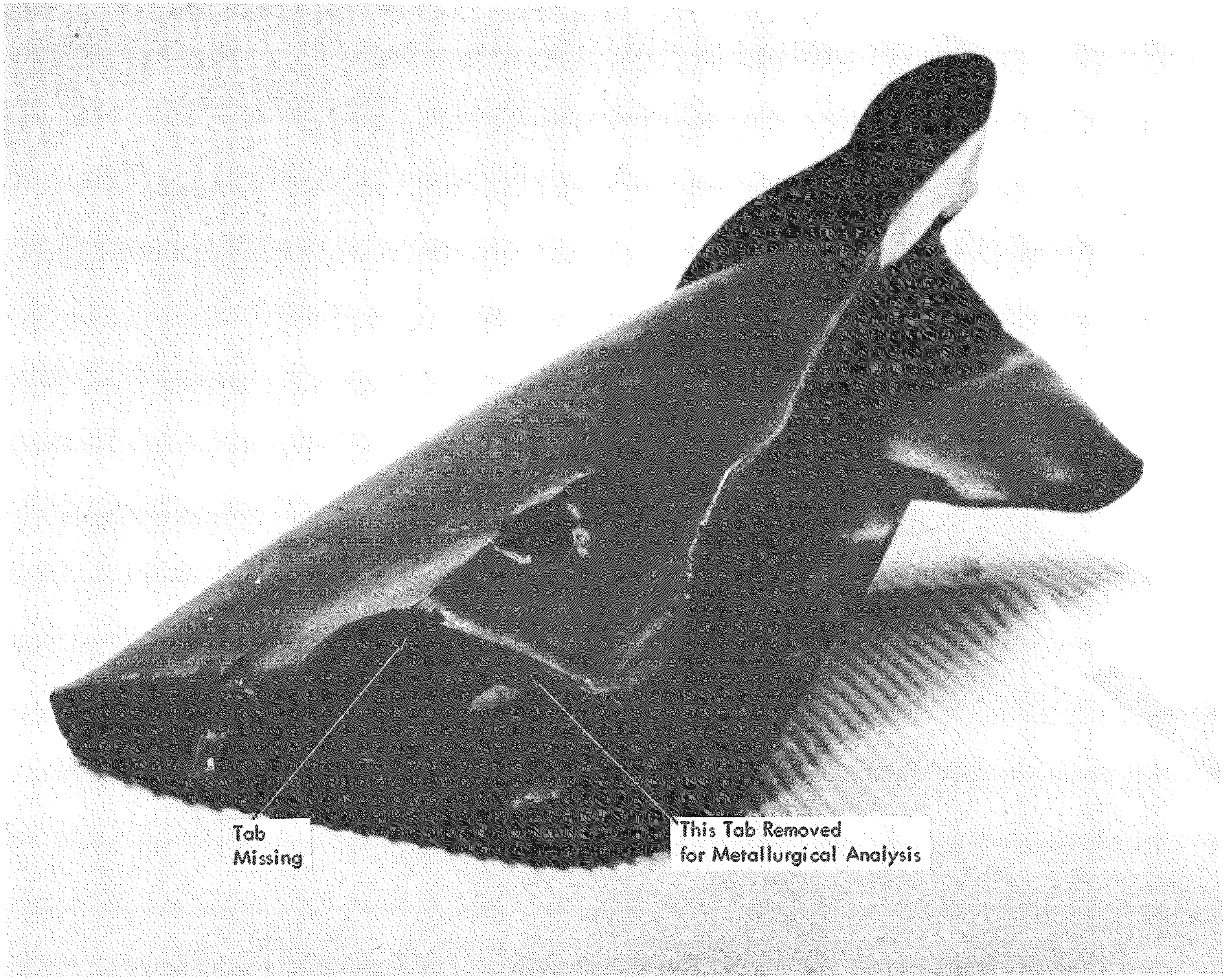

FIG.12 FLAT SIDE OF CLEANED OBJECT

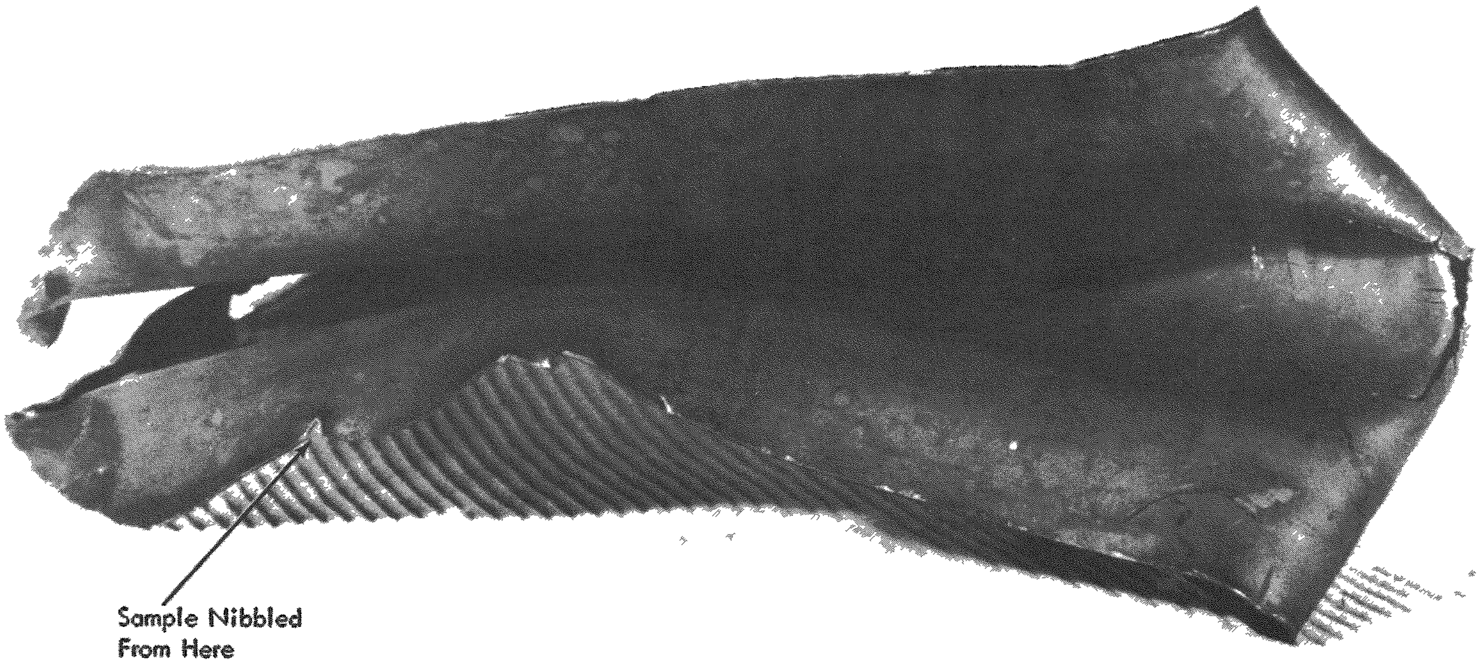

FIG.13 FOLD OF THE CLEANED OBJECT

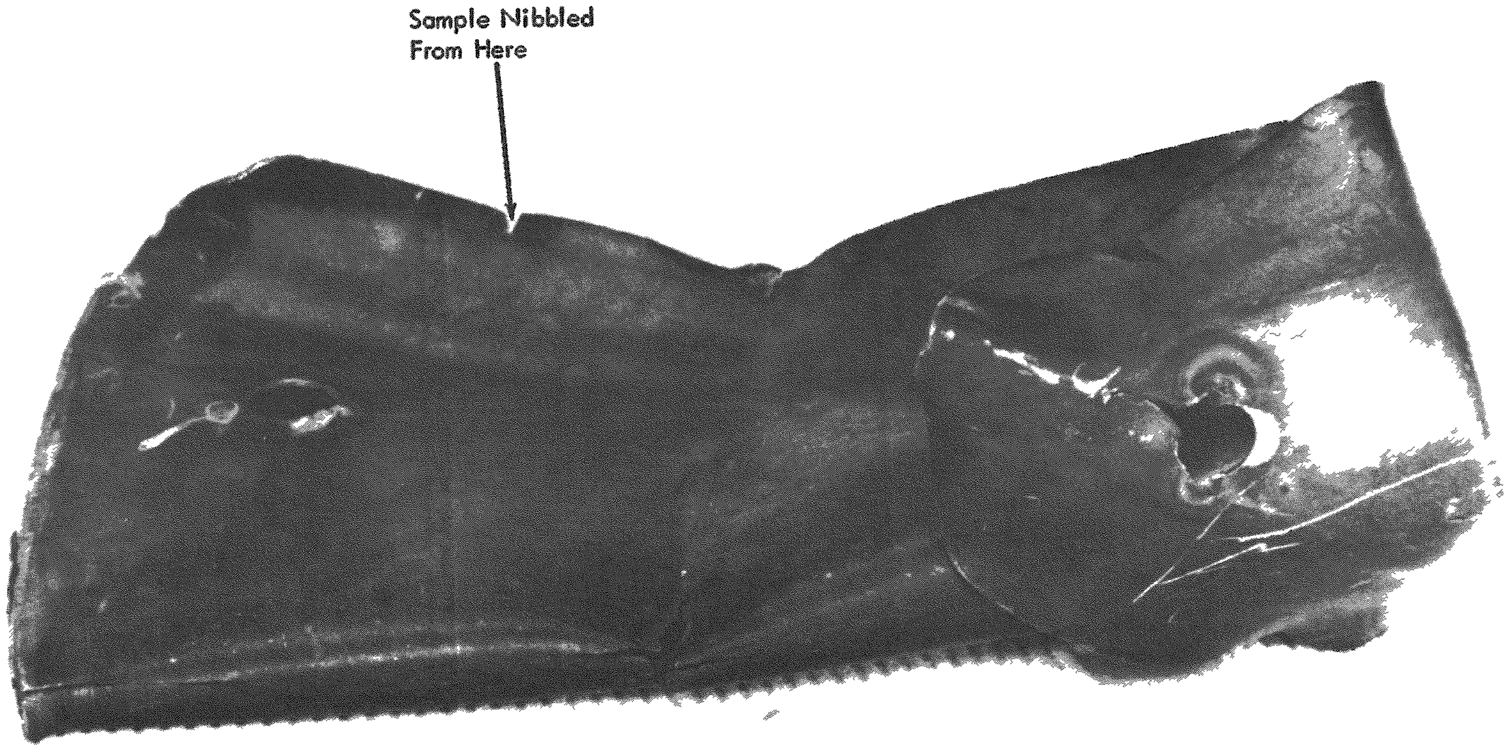

FIG.14 BENT SIDE OF CLEANED OBJECT

of a hardness test was reported as 76 "B" Rockwell. In a bending test to determine the degree of embrittlement, it was shown that the sample would bend about 130 degrees at room temperature before cracking.

4. Visual Examination

Visual examination of the object after clean-up and decontamination revealed the following circumstances:

1. A 2-inch-long tab was missing from the wide end (see Figure 12).

2. Faint circular score marks were found on the flat sections of both sides of the object, indicating that the object may have hammered against the support plate holes.

3. Measurement of the screw holes indicated that the diameter of the holes was 3/8 inch.

4. The condition of the screw holes suggested that the object in its original shape had been wrenched loose from the conical flow guide. Surface marks around the holes clearly showed where screws had been tack-welded to the object.

5. A V-shaped notch on the edge of the object was evidence of the sample nipped by the hawk-bill nibbler in December, 1967.* The V-shaped notch is shown in Figures 13 and 14.

6. The major dimensions of the object were determined to be 10-1/8 inches long, 5-7/8 inches wide, and 4 inches deep. These measurements compare well with dimensions estimated based on analysis of photographs taken in the reactor,** except that the actual width is significantly greater than the earlier estimate and explains why the object could not be pulled completely into the retrieval duct.

E. RESTORATION OF PRIMARY SYSTEM INTEGRITY

1. Rewelding the Elbow Patch

The primary system integrity was restored by rewelding the patch in the elbow. The rewelding and inspection of the weld were accom-

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* APDA-CFE-17, Section II. B.

** APDA-CFE-16, page 14, and APDA-CFE-17, page 8.

plished in accordance with the ASME Boiler and Pressure Vessel Code, Section IX, Welding Qualifications, and Section III, paragraph UW-51, Technique for Radiographic Examination of Welded Joints. In preparation for the rewelding, the edge of the patch was built up with Type 308 stainless steel weld metal to replace the metal removed by the original cutting, a U-type weld prep was formed, and the edge was ground back enough to allow a 3/32-inch gap around the patch when fitted in the elbow penetration flush with the surface of the elbow. Burrs were removed from the edge of the elbow penetration by stroking with a file in the outward direction. Both the inside and outside surfaces of the elbow within a few inches of the penetration were cleaned by applying "no oxide" emery cloth.

For rewelding, the patch was positioned in the elbow penetration by the locating plates and held fast by four equally-spaced tack welds. The 1/8-inch-deep root pass was then applied, using 1/16-inch, 2%-thoriated tungsten electrodes, inert gas, and Type 308 stainless steel filler rods. The welding was done through the glove box, using a glass window in the tool port in the aluminum faceplate for viewing.

2. <u>Radiographing the Root Pass</u>

The quality of the root pass of the elbow patch weld was determined by radiographing the weld through the opposite walls of both the primary and secondary containment elbows, using an iridium-192 point source. Three areas of insufficient weld penetration were shown to exist, and two successive weld repairs were required to obtain a root pass of good quality. The elbow patch weld was completed using tungsten, inert gas, and Type 308 stainless steel filler rods. Another radiograph showed the final weld to be of acceptable quality. The secondary containment elbow was closed by butt-welding the original patch flush with the elbow surface.

3. <u>Welding Difficulties</u>

The following are difficulties experienced in making the root pass of the primary elbow weld:

1. The depth of the glove box did not afford the welder a close enough view while making such a critical weld.

2. The 10-inch-diameter viewing window did not permit a sufficient range in the angle of view.

3. The locating plates obstructed the view of the welder.

4. The limited penetration in the secondary containment elbow impeded the maneuvering of the welding torch.

III. INVESTIGATION OF ZIRCONIUM SEGMENTS

A. APPROACHES FOR SCREW REMOVAL

A previous report (APDA-CFE-19, Sec. II. D) outlined a number of possible concepts for removing the screws which fasten the zirconium segments to the conical flow guide in the reactor core inlet plenum. Screw head removal appears to be the most feasible method for freeing the segments from the conical flow guide so that they may be removed from the reactor. Background information on this investigation may be found in APDA-CFE-18 and 19.

In March, the approaches for screw removal were narrowed to considerations of the machining and chiseling methods and to the possibility of melting and blasting off the screw head by the arc discharge process. Initial engineering efforts were devoted to an investigation of the requirements for gaining access to the three screws fastening each of the zirconium segments. The most convenient access penetrations through the support plates, the optimum angles for applying removal tools to the sloping surfaces, and the need for articulated mechanisms were subjects for study. Attention was given to setting up tests to demonstrate the effectiveness of the arc discharge process in removing the screws.

B. MACHINING AND CHISELING TESTS

In efforts paralleling the above engineering investigations, preliminary machining and chiseling tests were conducted to determine the energy requirements for these methods of screw removal. In one test, a zirconium screw head was machined off by a hand-applied 2200 rpm conical rotary grinder in about 10 minutes; the applied force was estimated to be about 15 pounds. In another test, a 9/16-inch hole cutter operating at 492 rpm trepanned a section of 0.060-inch zirconium plate in 3 to 4 minutes with an applied force of 10 pounds. In this instance, the cutter freed the plate from the fastening screw tack-welded to it. It is felt that a force of 3 pounds could do the same job but would require a longer cutting time.

The chiseling tests employed a free-fall weight drop as the force applied to the chisel in attempts to knock off the head of a zirconium screw fastening a 0.060-inch zirconium plate to a test structure. Just before impact, the test structure was torched to a temperature of 360 F. With the chisel directed at angles of 23° and 51° from the plane of the zirconium plate, about 200 to 250 foot-pounds of energy knocked off the screw. The angle of chisel attack appeared to make no difference. Screw head removal was accomplished in only half of the total number of tries, however.

Subsequent observation of films of the chiseling tests revealed significant vertical deflection of the supporting test structure. Thus, the validity of chiseling tests is questionable. It is planned to repeat the tests using a more rigid support structure.

IV. REMOVAL OF SPHERICAL MIRROR FROM THE REACTOR

As reported in Section III. C of APDA-CFE-19, the spherical mirror used in the attempts to view and photograph the holddown fingers to determine the condition of the fingers broke off its support post and fell onto the top support plate. Figure 8 in APDA-CFE-19 shows the mirror nested in a small canyon over lattice positions NO4-PO4, NO4-PO3, NO3-PO4, and NO3-PO3. *

Figure 15 shows the bulb-snatcher retrieval tool designed to slip over and grasp the mirror and remove it from the reactor. The tool is offset from its handling tube to permit reaching out about one lattice position from the closest rotating plug penetration. From observation of the photograph in Figure 8 of APDA-CFE-19, which shows the mirror nestled against a subassembly nozzle sleeve and under the wrapper can overhang, it appeared that there was not the required clearance for the retrieval tool to slip over the mirror. Mock-up tests confirmed the lack of sufficient clearance. It was, therefore, necessary to move five adjacent subassemblies to open up the canyon to obtain access to the mirror. The five lattice positions vacated were NO3-PO5, NO4-PO5, NO5-PO5, NO5-PO4, and NO5-PO3. *

The mirror was removed from the reactor without difficulty and, after being cleaned and decontaminated in the ultrasonic cleaner, was disposed of to radioactive waste, since it was no longer usable. Figure 16 shows the retrieval tool with the mirror withdrawn into an inert chamber above the rotating plug penetration. The mirror after cleaning is shown in Figure 17. The surface of the mirror was cracked and extensively marred, but the piece was still intact.

* See the reactor loading diagram, Figure 18.

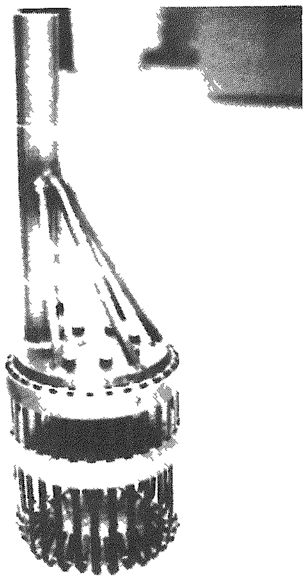

FIG.15 MIRROR RETRIEVAL TOOL

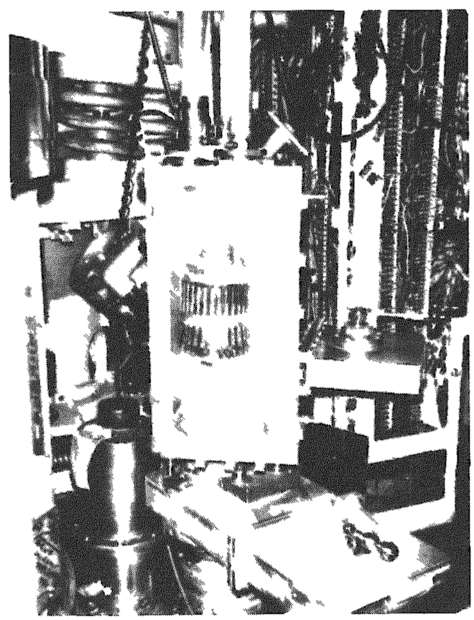

FIG.16 RETRIEVAL TOOL WITH MIRROR IN INERT CHAMBER AFTER REMOVAL FROM REACTOR

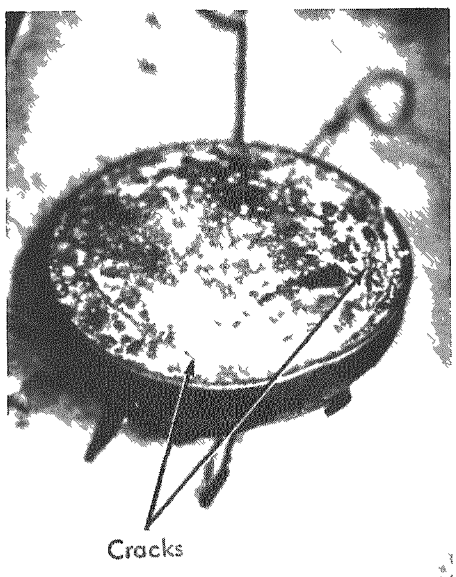

FIG.17 MIRROR AFTER REMOVAL FROM REACTOR

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)

Φ Larger Than Normal Spacing Between the Blanket Elements and the Support Grid

⊖ 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

Locations for Viewing Core Inlet Plenum

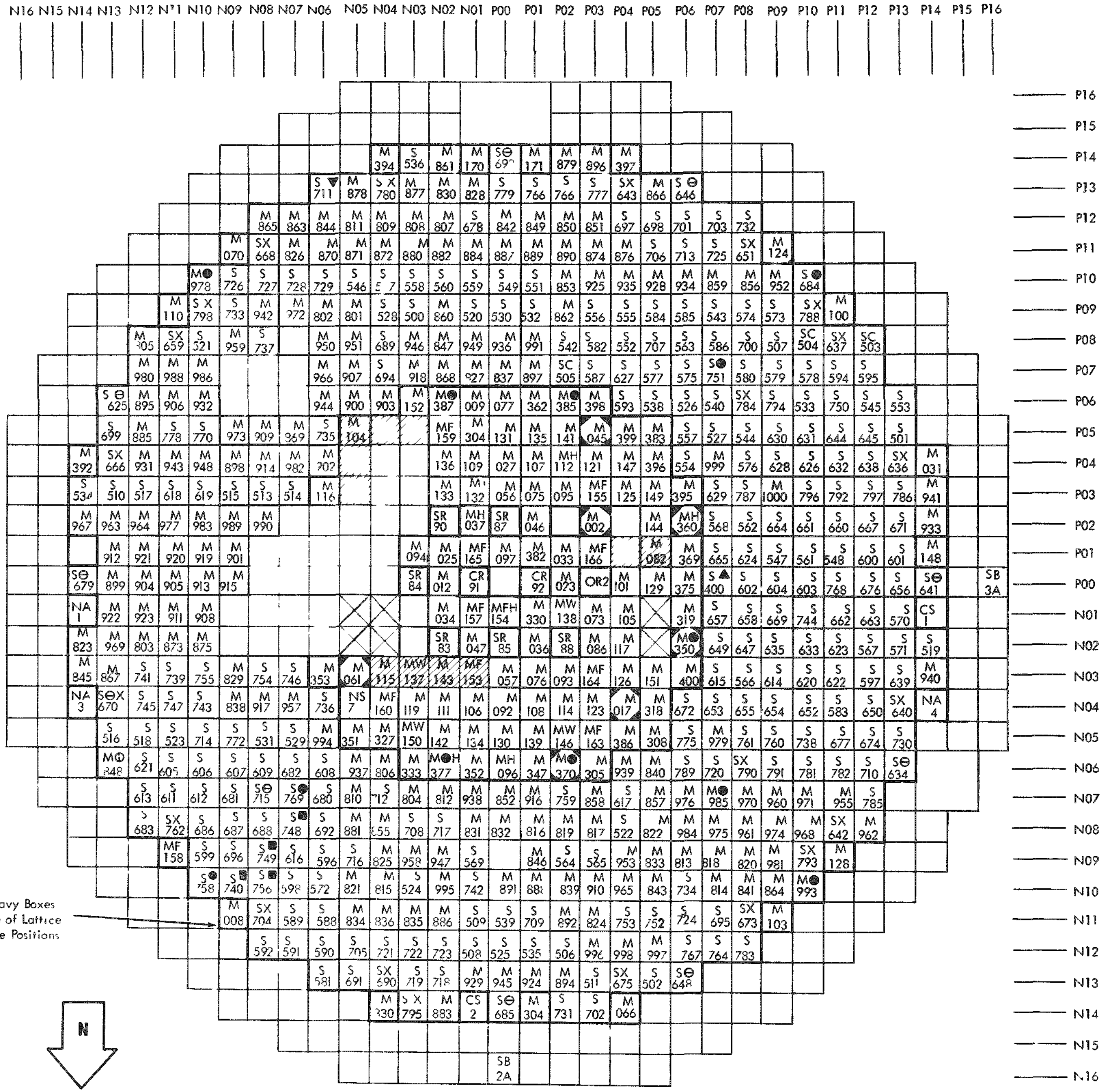

FIG.18 REACTOR LOADING ON 3-29-68

V. OPERATIONS

A. REACTOR SUBASSEMBLY MOVEMENTS

As shown in Figure 18, six core subassemblies were moved to different positions in the reactor during March. Five of these moves were made to open up an area to facilitate removal of the spherical mirror, as mentioned in the preceding section. The sixth move was made in preparation for inspecting and photographing the inlet plenum. Details of this inspection will be given in a future report.

The following table summarizes the subassembly relocations made during March:

Subassembly No.	Type	Position From	Position To	Reason for Change
M153 F*	Core	NO5-PO3	NO1-NO3	Removal of Spherical Mirror
M143	Core	NO5-PO5	NO2-NO3	
M082	Core	NO4-PO5	PO5-PO1	
M115	Core	NO5-PO4	NO4-NO3	
M137 W*	Core	NO3-PO5	NO3-NO3	
M104	Core	PO4-PO1	NO5-PO5	Inlet Plenum Inspection

* See Figure 18 for expanation of F and W subassemblies.

The offset handling mechanism and the rotating shield plug were operated in the automatic mode for the subassembly moves. It was recognized that there was a risk, though not a great one, of galling the OHM bearing surfaces in operating the component with the reactor drained of sodium. It was felt, however, that if there was any sodium oxide on bearing surfaces, the surfaces would be self-polished. As a precaution, the OHM was operated at slow speed in elevation when transferring the first subassembly; but, since there were no troubles encountered, the component was operated at normal speed when moving the other subassemblies.

There was difficulty experienced in attempting to insert subassembly M082 in its new lattice position. In a number of attempts to lower the subassembly into position, the OHM tripped out from overload, apparently because the subassembly was not properly aligned with the support plate hole. Slight changes in rotation of the rotating plug and OHM finally brought about proper alignment of the subassembly, and it was inserted into the lattice.

B. GAS SYSTEMS PERFORMANCE

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

1. Primary System Gas Activity

Location	Sample Date	Gross Beta Concentration, microcuries/cm^3
Reactor Cover Gas	3-4-68	7.2×10^{-6}
Reactor Cover Gas	3-4-68	4.4×10^{-6}
Primary Shield Tank	3-6-68	3.0×10^{-8}
Reactor Cover Gas	3-14-68	2.6×10^{-6}
Primary Shield Tank	3-15-68	1.7×10^{-8}
Primary Shield Tank	3-22-68	2.5×10^{-8}
Reactor Cover Gas	3-22-68	1.04×10^{-6}
Reactor Cover Gas	3-29-68	7.45×10^{-7}
Primary Shield Tank	3-29-68	2.0×10^{-8}

2. Primary System Cover Gas Analysis

	Reactor Cover Gas (Argon),* ppm by volume	Primary Shield Tank Atmosphere (Nitrogen),* ppm by volume
Oxygen	Below 25	60**
Carbon Monoxide	Below 10	Below 10
Carbon Dioxide	20	55
Hydrogen	Below 4***	Below 2.5
Helium	Below 4	Below 4
Methane	Below 10	Below 10
Argon	Remainder	Not Measured
Nitrogen	824	Remainder
Dew Point	Not Measured	-60 F

* Sample date was 3-22-68.
** Technical specifications state 1000 ppm maximum.
*** 10 ppm is the recommended maximum for reactor operation.

* APDA-CFE-19.

VI. MAJOR EQUIPMENT

A. EXIT PORT INSPECTION FACILITY

Preoperational testing of the exit port inspection facility continued in March. Efforts were directed toward improving mechanical features and determining the capability of the facility to detect subassembly bowing. Earlier information has been given in previous reports.*

A survey of the elevation of handling heads of subassemblies in transfer rotor (TR) positions was made to determine the range of vertical adjustment that must be designed into the chamber wall penetration to permit the lower actuator wrench to open and close the gripper on subassemblies in the TR positions. The difficulties that prompted this survey are described in APDA-CFE-19. For the purpose of the survey, the actuator wrench penetration through the chamber wall was temporarily modified.

The gripper was latched on two subassemblies, raised each of them 8 feet, and redeposited them in the TR. In other TR positions, because of actuator wrench limitations, the gripper merely opened on the handling heads or partially latched, it being assumed that full gripper actuation and lifting of the subassembly could be accomplished. There were no subassemblies in TR positions 1 and 12. The No. 12 position had been deformed about 1/2 inch low as a result of the melted subassembly M098 dropping into the TR during efforts to remove the subassembly for shipment to Battelle Memorial Institute.**

The survey showed that the range of elevations of the 10 subassemblies in the TR was 0.6 inch. Since the exit port tube and the inspection chamber were at the same temperature, about 420 F, the influence of temperature may be neglected. Only subassembly length and pot depth could affect the survey results. The survey results for TR positions 6 through 11 were in agreement with results of the TR elevation survey after the incident of dropping subassembly M098.

A core subassembly containing depleted uranium fuel pins was dimensionally gauged in the inspection facility chamber and the results compared favorably with dimensional data obtained on an external gauging fixture. The external fixture had shown the subassembly to be straight within 0.015 inch. Both dimensional surveys were in agreement within 0.010 inch.

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* APDA-CFE-5, 8, 9, 10, 16 through 19.

** APDA-CFE-12, page 10.

When a dummy subassembly was inserted in the inspection chamber, however, the dimensional data obtained exceeded the known bow by 0.040 to 0.075 inch. The dummy subassembly was known to be bowed 0.125 inch. The dimensional data obtained for this bowed subassembly appears to show that the facility magnifies dimensions, but the scattering of results discourages attempts to establish a pattern. The dimensional survey work will continue and possible influencing factors will be investigated in an effort to establish the potential accuracy of this measurement.

Figures 19 and 20 show the transit set up in front of the center window for a dimensional survey, and a view of the subassembly through the center window.

B. FUEL TRANSFER FACILITY

Erection and assembly of the FARB section of the new fuel transfer facility was completed without difficulty in March. A description of the facility and other information are given in previous reports.*

Qualification tests are presently underway in which the initial or one-time conditions and settings of equipment, controls, instrumentation, etc. are established. In these tests, it is assured that individual components function as designed.

In the next series of tests, the preoperational tests, the routine preparatory steps such as inert gas purging, heating, etc. that must be carried out in advance of each operation of the facility will be checked out. It is expected that the preoperational tests will be initiated in April and completed in May.

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* APDA-CFE-10, 19.

FIG.19 TRANSIT VIEWING THROUGH CENTER WINDOW OF EXIT PORT INSPECTION FACILITY

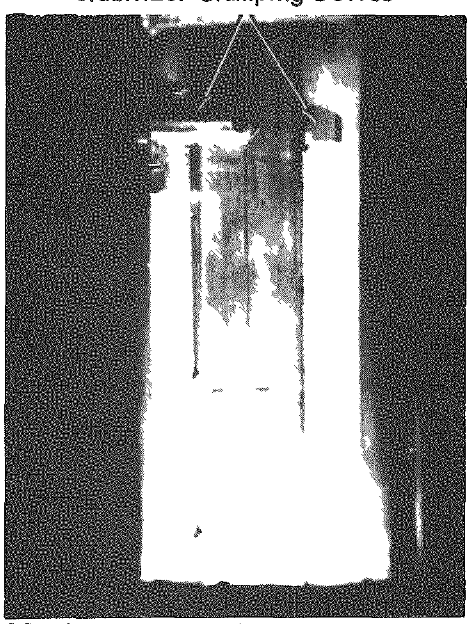

FIG.20 VIEW THROUGH CENTER WINDOW OF EXIT PORT INSPECTION FACILITY

VII. NONFISSIONABLE MATERIALS SURVEILLANCE PROGRAM

A. PROGRAM OBJECTIVE

The objective of the nonfissionable materials surveillance program is to study the effects of irradiation on the physical and mechanical properties of materials used in critical areas of the reactor. A total of 112 surveillance tensile specimens of Types 304 and 347 stainless steels and Inconel X-750 were originally installed in a special materials surveillance subassembly (MS-1), which, in turn, was installed in a reactor lattice position before the reactor power level exceeded 1 Mwt. The specimen materials are metals which are either used in some critical areas of the reactor or are of sufficiently close composition and history that they are representative of materials in other critical areas. The reactor applications and the materials under surveillance are as follows:

Reactor Application	Material
Core and Blanket Subassembly Wrapper Can	Types 304 and 347 Stainless Steel
Core and Blanket Subassembly Spring	Inconel X
Core Support Plate	Type 347 Stainless Steel
Safety and Operating Control Rods	Type 316 Stainless Steel and Boron Carbide
Guide Tubes	Type 316 Stainless Steel
Neutron Source	Beryllium and Tantalum
Reactor Vessel	Type 304 Stainless Steel

Subassembly MS-1 was installed in the reactor in lattice position PO1-PO3 on December 24, 1965, transferred to lattice position NO1-PO3 on September 22, 1966, and then removed from the reactor and stored in sodium in the FARB transfer tank on December 30, 1966. The surveillance specimens in MS-1 were irradiated to a maximum neutron exposure of 1.2×10^{21} nvt (E> 0.1 mev). Subassembly MS-1 remained in the transfer tank until March 1968, when it was removed, steam-cleaned, and disassembled in the cut-up pool.

B. CLEAN-UP AND DISASSEMBLY OF MS-1

The steam cleanup of MS-1 was the first time that a subassembly with a 0.125-inch-diameter orifice was steam-cleaned without difficulty. For disassembly, the subassembly was laid on a special tray in the cut-up pool and disassembled by long, remote-manipulated tools. Twenty-two surveillance tensil specimens and two temperature monitors were removed, and replacement tensile samples were inserted. At the same time, 64 tensile specimens belonging to French and German organizations were removed from subassembly MS-1 for shipment to the sponsors.

The subassembly was reassembled, dryed in the steam cleaning chamber, and stored in sodium in the transfer tank. Inasmuch as MS-1 was not dimensionally checked at this time, it must be checked some time prior to reinsertion in the reactor.

C. ELECTRICAL RESISTANCE TESTS

Of the 22 surveillance specimens removed from MS-1, 12 were Inconel X-750, 5 were Type 304 stainless steel (reactor vessel material), and 5 were Type 347 stainless steel (core support plate material). The Inconel X-750 specimens are products of four separate heat treatments: solution treated, singly aged, doubly aged, and a special aging treatment.

Electrical resistance measurements of the 12 Inconel X-750 specimens were made and compared with preirradiation data. Electrical resistance is used as a tool to indicate structural changes in material and changes in thermal conductivity. The activity level of these surveillance tensile specimens was less than 100 mr per hour at a distance of 1 foot; therefore, handling with long tongs afforded adequate personnel safety. By virtue of a measured decrease in electrical resistance of the solution treated specimen irradiated in the core, it appears from a preliminay analysis that this specimen has undergone radiation-induced aging. This must be confirmed by other mechanical and physical property measurements. The electrical resistance of all other Inconel X-750 specimens increased. No other conclusions have as yet been drawn.

It is expected that electrical resistance measurements of the 10 stainless steel tensile specimens removed from MS-1 will be made next month.

D. FRENCH AND GERMAN SPECIMENS

Thirty-two specimens from Commissariat A l'Energie Atomique (CEA) of France and thirty-two from Gesellschaft fur Kernforshung (GFK) of Germany were removed from MS-1 at the same time as the surveillance specimens. Irradiation of these specimens was carried out under contract with EURATOM. The GFK specimens were loaded into a special cask furnished by Battelle Memorial Institute and shipped by truck to BMI for testing. The CEA specimens were loaded into a cask (supplied by the French) which required the addition of lead shielding and lead shot in the cask cavity to reduce surface activity to an acceptable level for shipping. The cask will be sent to France when shipping details are resolved.

Twenty-one tensile specimens belonging to the United States Naval Research Laboratory, also installed originally in MS-1, remained in the subassembly when it was returned to the FARB transfer tank.

VIII. MAINTENANCE

A. TUBE-TO-TUBESHEET WELD REPAIRS ON THE NO. 3 STEAM GENERATOR

The rewelding of all 1200 tubes to the water manifold tubesheet of No. 3 steam generator was completed in March. Details of the repair weld technique have been related in several earlier reports starting with APDA-CFE-7.

After each tube was welded, the weld was inspected through a borescope. About 10% of the tubes required ten or more passes to obtain a satisfactory weld. After the total reweld job was completed, another borescope inspection of each weld was undertaken. The welds all appeared to be satisfactory.

The tube-to-tubesheet welds were then leak-tested by the bubble method and five leaking welds were detected. The five defective welds were repaired and then shown to be leaktight in a second bubble leak test.

B. LEAK TEST OF CONTAINMENT BUILDING ELECTRICAL PENETRATIONS

The containment building electrical penetrations were leak-tested at 32 psig and the leak rate measured as 218 scfd. This leak rate was considered satisfactory based on comparison with previous results and on the requirement that the total leakage from all containment building penetrations, as determined from the most recent leak tests, should not exceed 1000 scfd.

A description of the grouping of electrical penetrations and details of each penetration are given in APDA-CFE-7, page 25. In the leak test, a flowmeter measures the air leakage through a group of about 30 penetrations at a time.

C. REPLACEMENT OF POWER SUPPLY CABLES FOR INDUCTION HEATERS

There is an extensive program underway at the present time to replace all power supply cables, between the circuit breaker panels and the terminal boards in the containment building, induction heater coils on the primary sodium loops. This replacement program stems from the discovery of general deterioration of the insulation on the No. 8 Butyl-insulated cable crowded in cable trays. The insulation was hard and brittle, and in many places cracked open. The cable in 101 induction heating circuits is being replaced by about 12, 000 feet of type THW PCV No. 6 wire.

The deteriorated condition of the power supply cables is due to excessive heat from overloaded circuits for the last 8 to 9 years. Some of these cables, rated for 45 amps, have been carrying up to 88 amps during heating service. The feeder wires from the terminal boards to the induction heater terminals are rated for 65 amps and appear to be in satisfactory condition.

It was determined that nearly 25% of the primary system induction heating circuits had been subjected to up to 50% overload current. Another 20% of the primary system heating circuits had been carrying in excess of 50% overload current. The design of the primary system heating presumed the use of three adjoining circuits at one time and, further, the limiting of current in the three circuits by their mutual inductance. In operation, however, it was usually the case that the use of three adjoining circuits at one time was impractical because excessive heating or too-frequent on-off operation would result.

Cable replacement is being accomplished with a minimum of heater outage time. The new cable is run between the circuit breaker and the terminal board first, and then the substitution of new cable for the old is made at each terminal.

Measures are being taken to relieve heater circuit overloading where possible. To limit current, two or more induction coils are being connected in series and, in other instances, more turns are being added to existing coils.