ENRICO FERMI ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT

FEBRUARY 1968

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U. S. Atomic Energy Commission
Contract No. AT(11-1)-865,
Project Agreement No. 15

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and
The Detroit Edison Company

ISSUED: MAY 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 Contract No. AT(11-1)-865, Project Agreement 15. This series of reports provides detailed information on the nuclear testing, machinery dome, steam generators, pumps, flowmeters, level detectors, sodium sampling, and development of the primary sodium system.

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

BACKGROUND

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

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

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

SECTION I

CURRENT EXPERIENCE SUMMARY

The development of special devices for removing the object from the reactor through the 14-inch sodium inlet line was completed. Testing in the 14-inch pipe and reactor mock-up demonstrated that the special devices could remove the object from the reactor core inlet plenum through the 14-inch pipe. Preparations were completed by the end of the month for proceeding with the penetration of the 14-inch pipe elbow of the No. 1 primary loop and the removal of the object.

Various approaches for removal of the remaining five zirconium segments from the conical flow guide were investigated, with emphasis on first removing the fastening screws. The organization of the engineering effort for removal of the remaining five zirconium segments was initiated.

Spherical mirror images of a few of the holddown fingers were photographed, but the pictures were of poor quality and as a result, were inadequate to establish the condition of the fingers. Insufficient illumination and difficulty in properly orienting the mirror were the principal obstacles to a successful observation. In an attempt to remove the spherical mirror assembly from the reactor, the mirror and its mounting broke off the support post and fell to the top support plate.

Preoperational testing of the exit port inspection facility was continued with effort being concentrated on the gripper, hoist, and associated mechanisms. The need for modifications was shown. Erection and assembly of the FARB section of the new fuel transfer facility was started.

The results of metallographic examinations of subassembly specimens at BMI indicated that significant coolant flow blockage occurred in subassembly M122, which adjoined the melted subassemblies M098 and M127 in the reactor.

SECTION II

INVESTIGATION OF OBJECT IN INLET PLENUM

A. BACKGROUND

The object discovered in the reactor core inlet plenum on September 11, 1967 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 support plate. Initial investigation and effort directed at removing the object have proceeded along two parallel paths: (1) cutting up the object and removing the pieces through the core support plates, and (2) retrieving the object intact through the 14-inch sodium inlet line. The development of cut-up and removal tools and devices was undertaken, aided by a large number of photographs of the object. Attempts to obtain a sample of the object were unsuccessful, and it was decided after a number of attempts that a sample was not worth further effort.

In late December, effort was concentrated on preparations to remove the object intact through the 14-inch sodium inlet line. Questions pertaining to penetrating the No. 1 primary sodium inlet line and subsequent restoration of primary system integrity were satisfactorily resolved.

In preparation for the object-removal operation, 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. Development of an object-retrieval device for removing the object through the 14-inch line was completed. The all-purpose, spine-type manipulator, designed to operate through the core support plates and feed the object to the object-retrieval device, approached the final stages of development.

In late January, the object was tentatively identified as being one of six 0.040-inch-thick zirconium plates which were originally installed on the sloping sectors of the conical flow guide in the reactor core inlet plenum. Effort directed toward removal of the object will continue and consideration will be given to either detaching and removing the remaining five zirconium segments or permanently fixing them in place.

Details of the investigation of the object are given in previous reports starting with APDA-CFE-14.

B. DEVELOPMENT OF TOOLS AND DEVICES

1. Spine-Type Manipulator *

Additional functional testing of the spine-type manipulator in the reactor mock-up** revealed the need for modification of the control mechanism for the device. There was evidence that some of the small components in the control mechanism would not be sturdy enough to withstand the manual forces required at the control head to flex the spine disk control cables 40 feet below in the reactor inlet plenum. These components were small gear trains and clutches that were designed to give the control mechanism flexibility in operating the disk control cables either singly or in pairs. The gears and clutches were necessarily small for space consideration.

A modified design eliminated the gear trains and clutches, providing instead for direct drive of each cable rack-and-pinion gear individually through a hex knob attached to the pinion. Figures 1 and 2 show the manipulator control head before and after the above modification. The control handwheels which engage with the hex knobs for operating the cable rack-and-pinion gears were prompted by the limited operating space around the control head. The proximity of the manipulator control to the upper end of the borescope is shown in Figure 2.

After the above modification was accomplished, the spine-type manipulator was used with the object-retrieval device in trial object-removal operations in the 14-inch pipe and reactor mock-up. Details of the mock-up trials are given in Section II. C.

2. Object-Retrieval Device ***

As reported in APDA-CFE-18, the object-retrieval device intended for removing the object through the 14-inch sodium inlet pipe was assembled at the end of January and was ready for trial operations in the pipe and reactor mock-up.

3. Articulated Lamp ****

It is intended that the articulated lamp will provide illumination for object-removal operations in the reactor core inlet plenum after completion of the development of the spine-type manipulator and successful demonstrations of object removal in the 14-inch pipe and reactor mock-up.

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* See Section II of APDA-CFE-15 through 18 for previous information.
** See page 15 of APDA-CFE-14 and Figure 17 of APDA-CFE-16.
*** See Section II of APDA-CFE-15, 16, 17, 18 for previous information.
**** See Section II. F. 3 of APDA-CFE-18 for previous information.

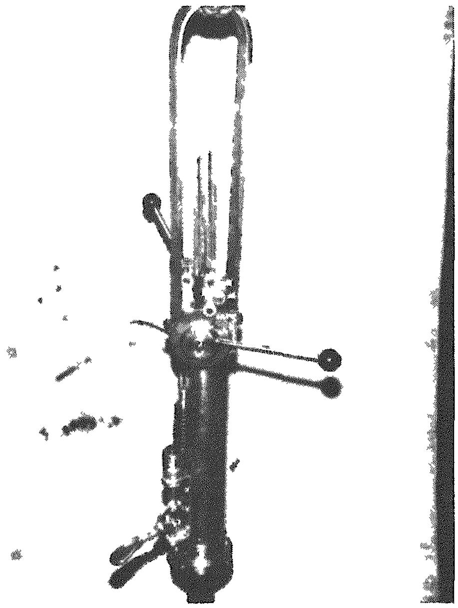

FIG.1 CONTROL MECHANISM FOR SPINE-TYPE MANIPULATOR BEFORE MODIFICATION

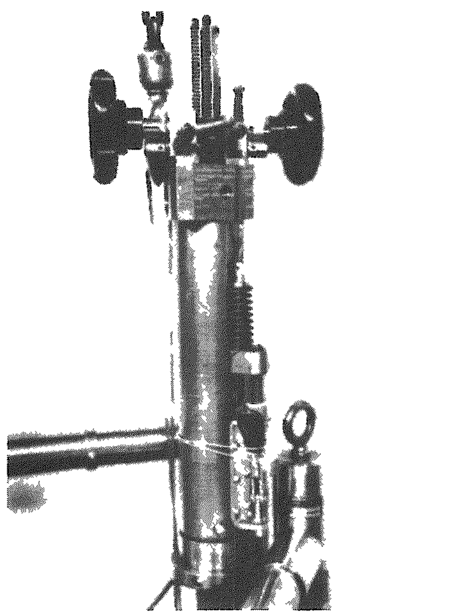

FIG.2 MODIFIED CONTROL MECHANISM

In the meantime, the articulated lamp was used in the holddown finger inspection in February (see Section III). Experience with the lamp during finger inspection prompted a modification of the lamp structure. The modification is described in Section III. A.

C. OBJECT-REMOVAL DEMONSTRATIONS IN REACTOR MOCK-UP

1. Demonstrating Object Removal

It was demonstrated in several trial object-removal operations in the 14-inch pipe and reactor mock-up that the spine-type manipulator and the object-retrieval device could function together to remove the object from the reactor core inlet plenum through the 14-inch sodium inlet pipe. To the largest extent possible, these demonstrations constituted a dress rehearsal for the forthcoming object-removal operations in the reactor.

In the mock-up demonstrations, operation of the manipulator controls and viewing of movements in the inlet plenum were done in the same manner as will be done on top of the reactor rotating shield plug. The limited area that exists on top of the rotating plug for operating the manipulator controls and for viewing was simulated in the mock-up. Figure 3 shows the manipulator clustered with the PRDC borescope* (for vertical viewing) and the Sunscope** (for horizontal viewing), with a plywood wall erected around the manipulator control head and viewing tubes to simulate the limitations of the work area. The full-length manipulator and scopes were used in the mock-up demonstrations.

The object-retrieval device was inserted into the mock-up inlet plenum through the penetration in the 14-inch pipe elbow mock-up via the glove box,*** which is intended to provide an inert gas seal. This part of the demonstration revealed deficiencies in the glove box structure and the containment for the retrieval device cable. Details are given in Section II. C. 3, below.

The capability of the spine-type manipulator was shown to be satisfactory; i. e., the gripper was able to reach out and grasp a metal model of the object purposely positioned in a limited access location under one of the baffle plates in the mock-up inlet plenum. Figure 4 shows the model of the object in the grasp of the manipulator gripper, about to be fed to the gripper in the object-retrieval duct.

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

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

*** See Section II. E. 3 of APDA-CFE-18 for a description of the glove box.

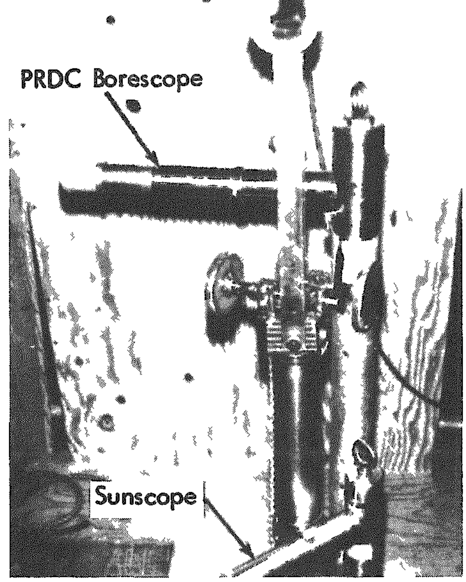


FIG.3 MOCK-UP OF WORK AREA AROUND SPINE-TYPE MANIPULATOR CONTROLS AND SCOPES

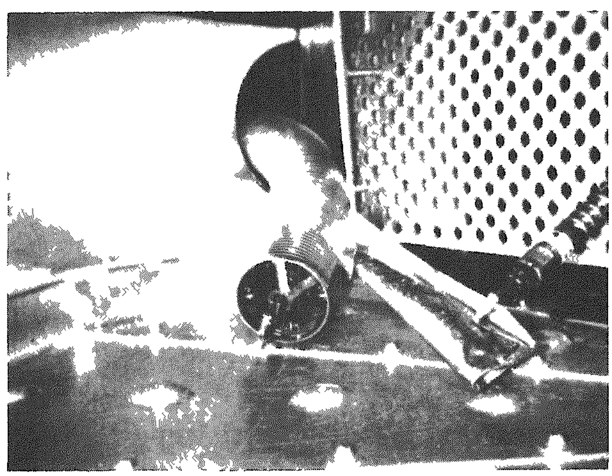


FIG.4 MANIPULATOR FEEDING OBJECT TO OBJECT-RETRIEVAL DEVICE IN MOCK-UP

2. Practice Operation of Spine-Type Manipulator

After the object-removal demonstrations, additional time was spent practicing operation of the manipulator. Familiarity with the manipulator control movements is required to produce the desired responsive movements of the gripper, and is expected to facilitate the removal of the object from the reactor.

3. Glove Box Modifications

The transparent bellows which makes up the body of the glove box was originally fabricated from three doughnut rings of Teflon plastic. The edges of adjacent rings were fused together by flaming to produce an accordion-like bellows structure. One end ring was clamped to the primary pipe adapter and the other end ring to the aluminum face plate. In the object-removal demonstrations, the fused seams tore open in many places. As a result, doughnut rings of polyvinyl chloride plastic were substituted and the adjacent edges were fused together by a heated roller device. Figure 5 shows the object-retrieval duct passing through the glove box in a mock-up demonstration.

The plastic containment bag for the 40-foot-long object-retrieval device was designed to telescope, each section inside the adjacent one, as the retrieval device moves through the elbow penetration. In the mock-up demonstrations, the bag crumpled instead of telescoping, and a new containment bag of larger diameter was fabricated to permit the retrieval cable and control mechanism to move through the entire length of the bag.

At the end of February, modifications to the glove box were completed and the spine-type manipulator and object-retrieval device were ready for use in the reactor. All preparations were completed in readiness for proceeding with the penetration of the 14-inch pipe elbow of the No. 1 primary loop and the object-removal operation.

D. APPROACHES TO REMOVAL OF ZIRCONIUM SEGMENTS

The principal design effort during the month was the consideration of methods for detaching the five remaining segments from the conical flow guide and removing them from the reactor. In a preliminary investigation, a number of concepts were considered for removing all of the fastening screws, or just enough to free the segments from the surfaces of the conical flow guide. The screw removal concepts considered were

1. Grind off the tack weld and unscrew the screw

2. Grind off the screw head

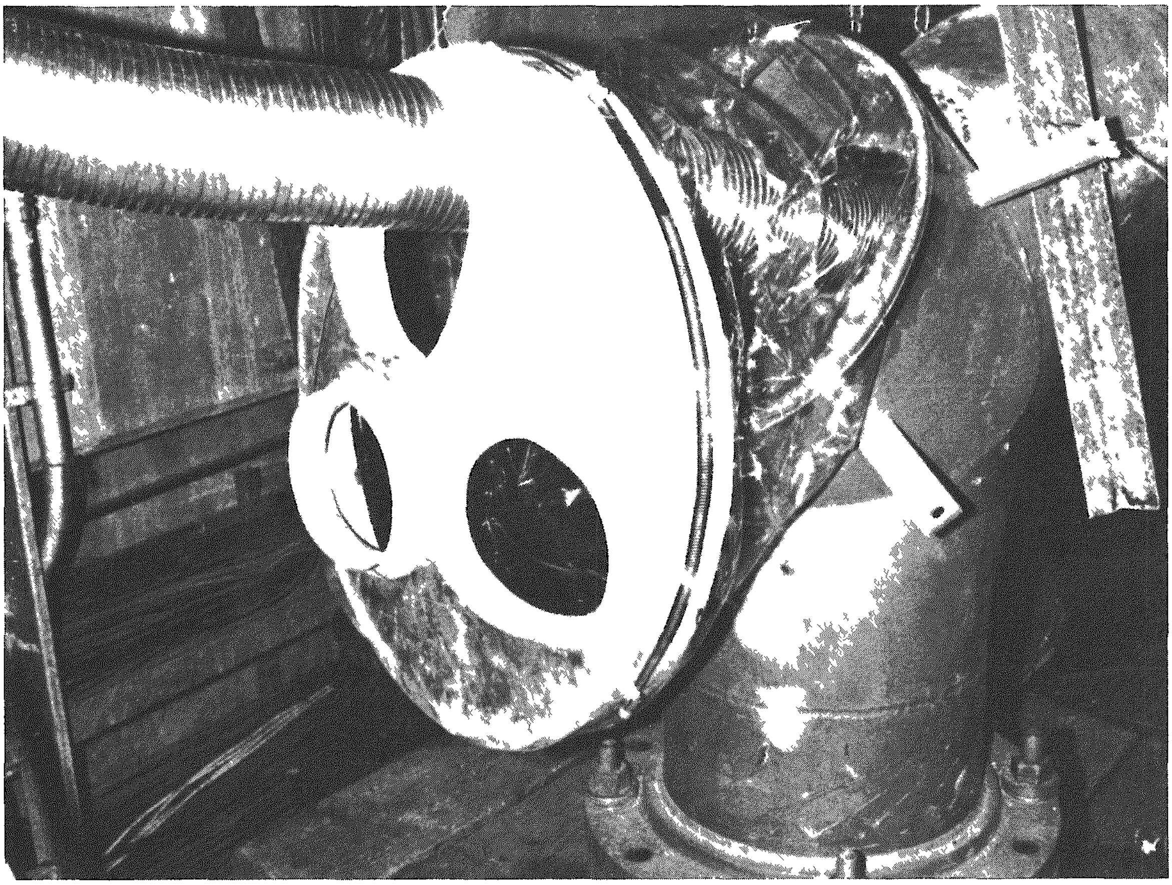


FIG.5 OBJECT-RETRIEVAL DEVICE PENETRATING GLOVE BOX IN 14-INCH ELBOW MOCK-UP

3. Unscrew the screw, shearing the welds

4. Cut off the screw head with a pincer tool

5. Chisel off the screw head by single or repetitive impact

6. Drill or mill out the screw head

7. Arc-melt the screw head

8. Melt the screw head by laser bursts

9. Remove the screw head by spark discharge method

10. Reduce the screw head to powder by impingement of hot hydrogen gas.

Items 2, 5, and 6 were selected as methods which will be given further consideration, although some of the other methods are believed to have merit.

Other concepts for detachment of the zirconium segments involved direct detachment of the segments by use of a gripper with a sufficient grasp to rip the segment free of the fastening screws or by application of a diaphragm-type suction device to lift the segment free. All detachment concepts require access through the support plate holes for insertion of articulated tools and a manipulator into the core inlet plenum.

Several concepts were considered for removing the zirconium segments from the reactor vessel, most of which make use of the original cut-up and removal tools and devices developed for removal of the object.

In February, the organization of engineering effort was initiated, with design emphasis on screw removal, particularly by the chiseling and milling methods.

SECTION III

INSPECTION OF HOLDDOWN FINGERS

A. EQUIPMENT MODIFICATIONS

Previous information regarding inspection of the holddown fingers can be found in APDA-CFE-13 and 18. As reported in APDA-CFE-18, initial attempts to view the spherical mirror images of the holddown fingers through the PRDC borescope were unsuccessful due to difficulties encountered in handling the mirror and support post assembly in the reactor.

Prior to a second attempt at viewing and photographing the holddown fingers to inspect for damage, equipment modifications were undertaken. These were modifications to the mirror and support post and to the insertion device, primarily to facilitate handling the mirror in the reactor. The mirror was reduced from 5-3/4 inches to 5 inches in diameter and then resurfaced because of damage incurred, apparently by abrasion, during a removal from the reactor.

As reported in Section II. B. 3, the new articulated lamp tube structure was modified as a result of experience during holddown inspection. The lamp tube was designed for insertion through one of the 6-inch penetrations in the rotating shield plug to illuminate the interior of the reactor vessel. The original 2-1/2-inch-long internal coupling sleeves, which joined the four tube sections comprising the full length tube structure, did not give sufficient rigidity to the structure and were replaced by 6-inch-long sleeves to increase rigidity.

The lamp tube is constructed in sections to permit leaving the lamp partially inserted into the reactor chamber while the rotating plug is rotated. The removal of exposed tube sections eliminates tube interference with a stationary overhead cable tray as the rotating plug is rotated. Thus, time-consuming lamp withdrawal and reinsertion are avoided.

B. PHOTOGRAPHING HOLDDOWN FINGERS

In order to view the holddown fingers in the southeast quadrant, it has been determined by calculation that the borescope must sight on the mirror along the line shown in Figure 6. The mirror, in lattice position NO7-PO0 and tilted to 32° from the horizontal, is oriented toward the southeast quadrant. (An imaginary plane intersecting the outermost point on the spherical surface and the lowest point on the spherical surface makes an angle of 79° with the east-west lattice datum line.) A bead was spot-welded on the mirror support post to assist in visual-

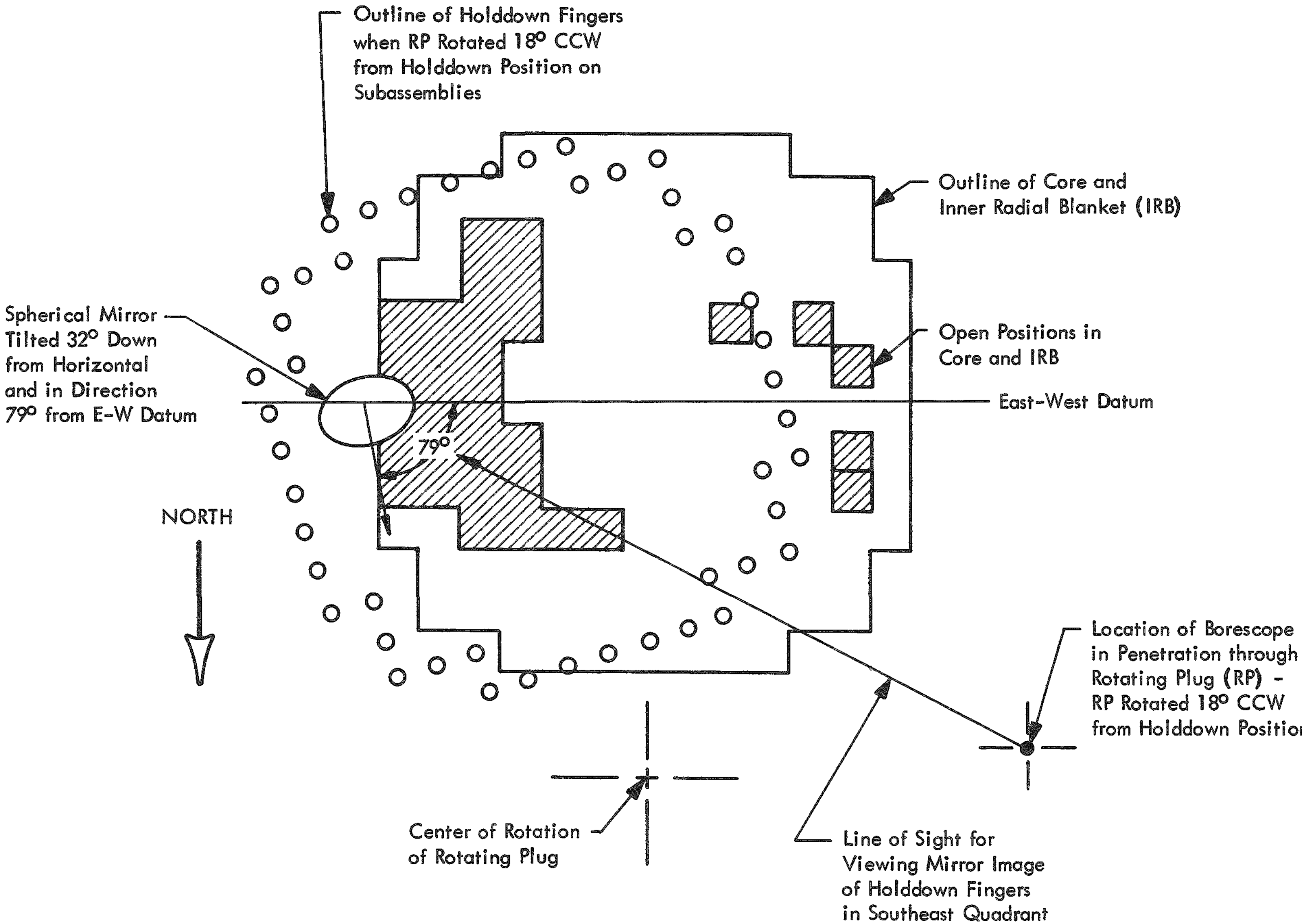


FIG.6 VIEWING SPHERICAL MIRROR IMAGE OF HOLDDOWN FINGERS IN SOUTHEAST QUADRANT

ly orienting the mirror during installation. The outermost point on the spherical mirror surface is 14-3/16 inches below the bottom of the holddown fingers. The borescope, with an 8^{o} field of view and a 23^{o} angle of view below the horizontal, is located with its objective 6-1/2 inches above the bottom of the fingers.

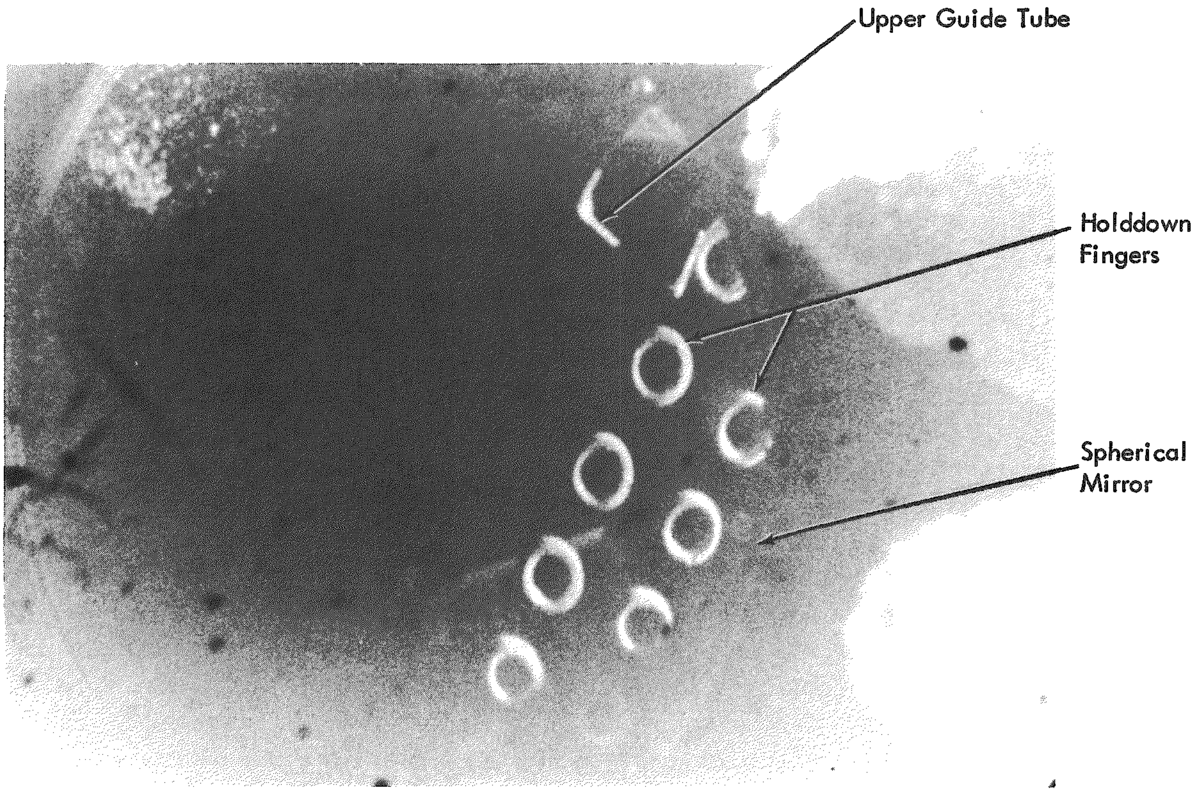
On the first try at viewing and photographing, the lamp source was directed at the mirror. The light reflection in the mirror obscured the images of the fingers. The lamp was then directed at the fingers and an image of five fingers in one row and three in a second row was viewed and photographed with a 15-minute exposure using Kodak Tri-X pan film. After the mirror azimuth was slightly changed in an attempt to pick up all fingers in the quadrant, an image of only three fingers in a single row was visible. Photographs of both of the above images are shown in Figure 7. The pictures obtained in both attempts were of poor quality,as can be seen in Figure 7. The exact position of the holddown fingers could not be determined, and the lighting was found to be inadequate to satisfactorily illuminate the fingers. A combination of the angle and the distance of the light from the fingers, plus the lack of the expected degree of reflectivity, resulted in insufficient light for suitable photography.

It was decided to halt the current holddown finger inspection program and direct effort toward establishing an improved method of determining the condition of the fingers.

C. MIRROR BROKEN OFF SUPPORT POST

In an attempt to remove the miror from the reactor, the insertion tool was lowered to engage the mirror support post, but instead the post was inadvertently tilted in the upper support plate. In another try, the insertion tool engaged the tilted post but not securely enough. As the lamp arm was swung over to align the post with the insertion device, the post slipped out of the insertion device and fell free into an empty lattice cavity. The mirror assembly, however, struck some subassemblies and broke off at the top of the support post.

The support post was retrieved from the reactor by the insertion tool after first being set upright in the support plate by means of a special tool. It was then observed that the mirror had fallen into a small lattice cavity, mirror side up, over reactor positions NO3-PO4 and NO4-PO3 (see Figure 8). When viewed in this position, the mirror appeared to be cracked, though still intact. Figure 9 shows a close-up of the mirror resting on the support plate. It was discovered that the break was in a weld in a solid shaft connecting the mirror tilt plate assembly with the top of the support post. Retrieval of the mirror will be delayed until after the object is removed from the reactor core inlet plenum.



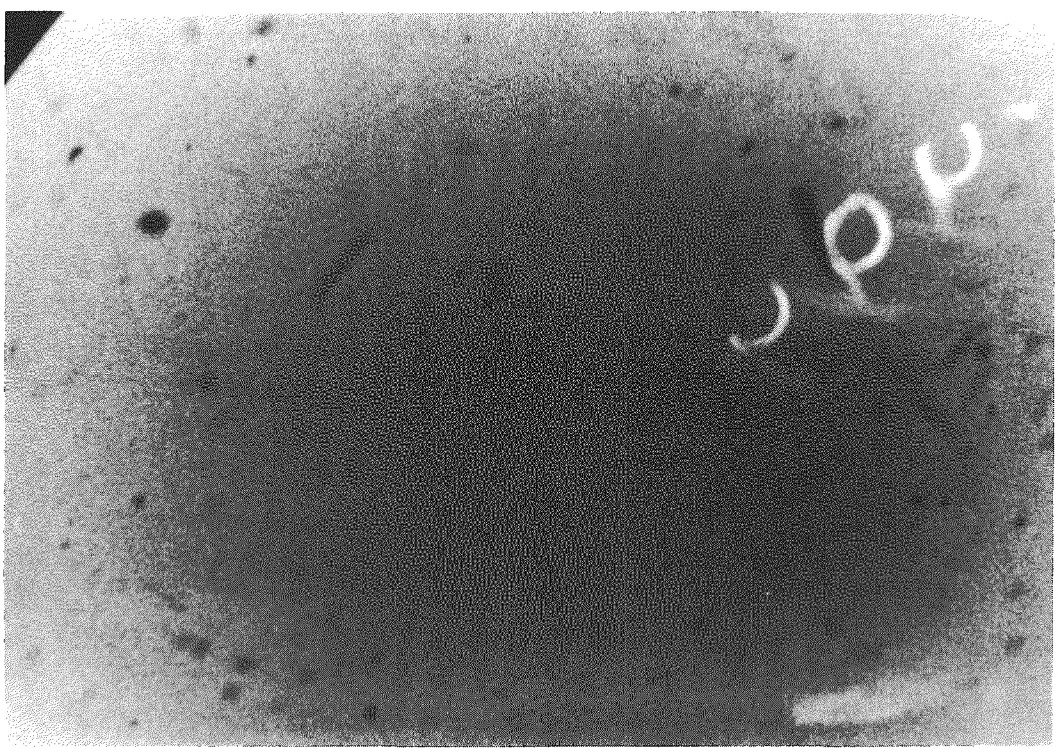


FIG.7 IMAGES OF HOLDDOWN FINGERS IN SPHERICAL MIRROR

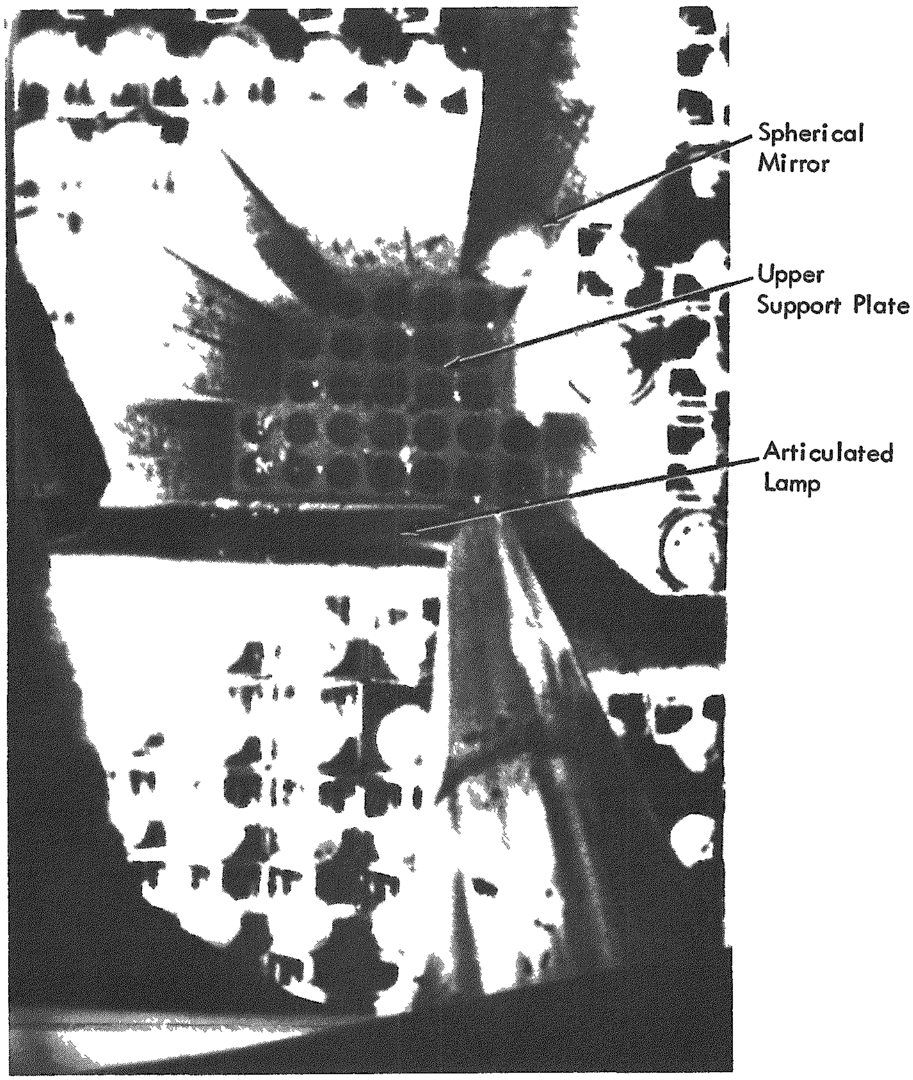


FIG.8 SPHERICAL MIRROR NESTED IN CAVITY IN REACTOR CORE LATTICE

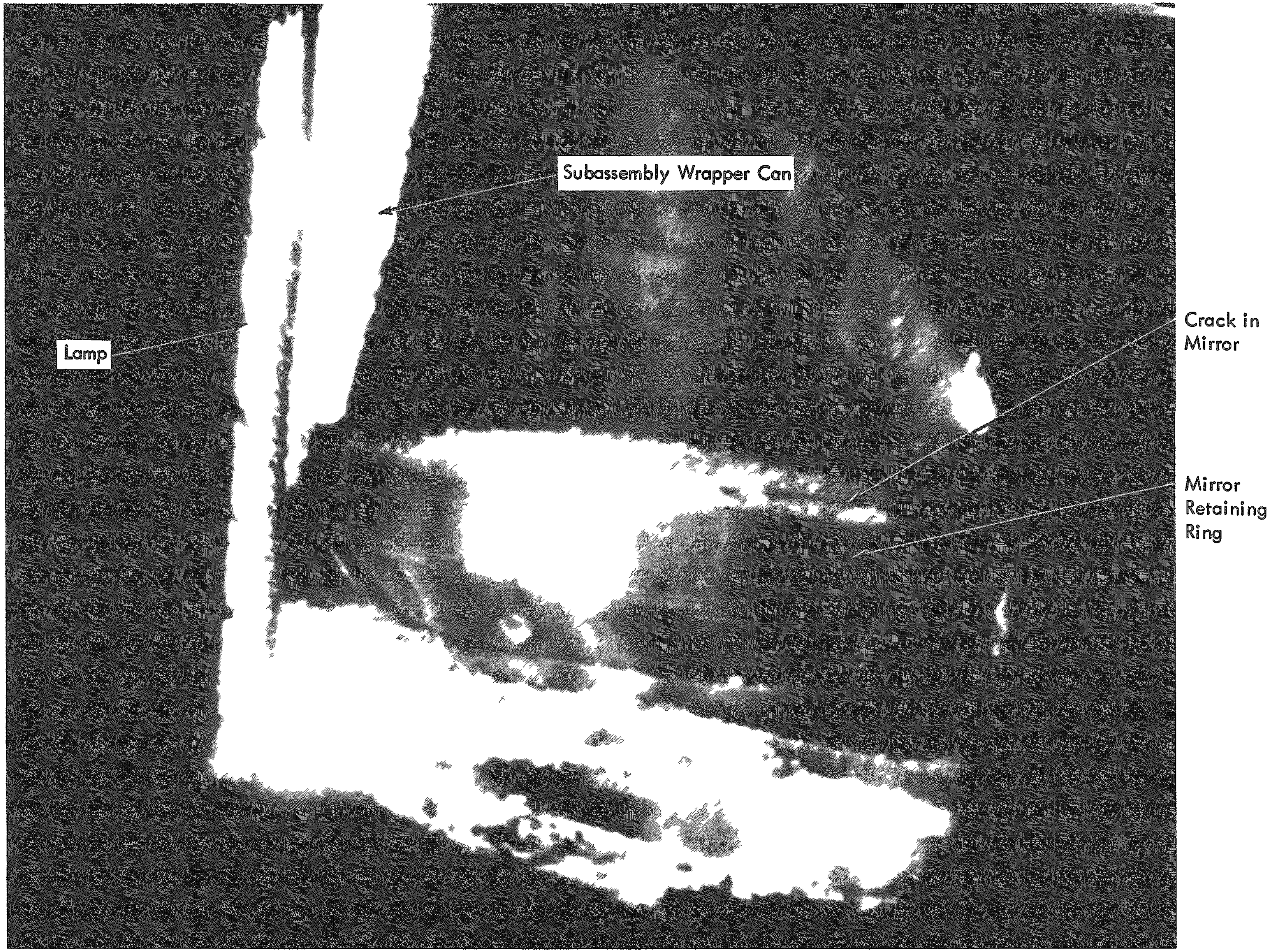


FIG.9 CLOSE-UP OF SPHERICAL MIRROR RESTING ON UPPER SUPPORT PLATE

SECTION IV

GAS SYSTEMS PERFORMANCE

Since the last data reported (APDA-CFE-18), the following primary system gas activity and analyses have been determined:

A. PRIMARY SYSTEM GAS ACTIVITY

Location	Sample Data	Gross Beta Concentration, microcuries/ cm^3
Primary Shield Tank	2-1-68	3.33×10^{-8}
Reactor Cover Gas	2-5-68	6.4×10^{-6}
Primary Shield Tank	2-9-68	2.67×10^{-8}
Reactor Cover Gas	2-12-68	4.5×10^{-6}
Primary Shield Tank	2-16-68	3.0×10^{-8}
Reactor Cover Gas	2-19-68	3.0×10^{-6}
Primary Shield Tank	2-23-68	3.0×10^{-8}
Primary Shield Tank	2-29-68	3.0×10^{-8}
Reactor Cover Gas	2-29-68	3.64×10^{-6}

B. PRIMARY SYSTEM COVER GAS ANALYSIS

	Reactor Cover Gas (Argon), ppm by volume*	Primary Shield Tank Atmosphere (Nitrogen), ppm by Volume*
Oxygen	Below 25	240**
Carbon Monoxide	Below 10	Below 10
Carbon Dioxide	10	30
Hydrogen	Below 4***	Below 2.5
Helium	Below 4	Below 4
Methane	Below 10	Below 10
Nitrous Oxide	Not Measured	Not Measured
Argon	Remainder	Not Measured
Nitrogen	4860	Remainder
Dew Point	Not Measured	-50 F

* The sample dates for reactor cover gas and primary shield tank atmosphere are 2-5-68 and 2-16-68, respectively.

** Technical specifications state 1000 ppm maximum.

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

SECTION V

MAJOR EQUIPMENT

A. EXIT PORT INSPECTION FACILITY

Preoperational testing of the exit port inspection facility continued. Effort was concentrated on testing the functioning and control of the gripper and hoist mechanism and associated components and mechanisms. Earlier information has been given in previous reports*.

Additional features were incorporated into the system of hoist controls. An interlock was provided to stop upward gripper movement short of contact of the gripper head with the yoke, the lower gripper tube aligning fixture, when the yoke is still engaged with the gripper tube. With the gripper head approaching the yoke, the yoke may be disengaged because the gripper tube is now passing through the upper aligning fixture (the rotational plate). During upward movement of the hoist tube, difficulty was experienced in moving the hoist tube into the rotational plate because of minimal clearance and the lack of a lead-in feature. Other control features were added to automatically reduce the speed of gripper travel as the gripper tube approaches its up limit (as it nears the hoist sheave), down limit (gripper nearing a subassembly handling head in the transfer rotor), and as the gripper tube is about to enter the rotational plate.

When the gripper was lowered onto each of two subassemblies in the transfer rotor, it was found that the lower actuator wrench penetration through the chamber wall was about one to two inches too low, preventing wrench rod engagement with the gripper tube to open or close the gripper fingers. The gripper tube was subjected to exit port temperature (350 F) for periods sufficient to produce thermal growth of the tube. It may have been this thermal growth which put the tube slot out of reach of the actuator wrench. The design provided for a maximum actuator rod-to-guide tube slot misalignment of 3/16 inch. Temporary relocation of the actuator rod penetration in the chamber is possible and will be made to permit a survey of subassembly elevations in the transfer rotor. This survey will determine the proper location of the actuator rod.

The ability to open and close the gripper fingers with both the upper and lower actuator wrenches was demonstrated; but, for a single operation

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* APDA-CFE-5, 8, 9, 10, 16, 17, 18

of a wrench, there was uncertainty as to whether the gripper was being opened or closed. There was no observable indication of gripper actuation.

When the exit port, and hence argon cover gas, was opened to the inspection chamber, fogging of the lower inspection window was observed, probably by sodium vapor from the exit port. It remains to be seen how seriously this condition may affect subassembly inspections, or how it may increase the time for inspections due to possible frequent window cleaning.

B. FUEL TRANSFER FACILITY

Components and structural steel for the new fuel transfer facility arrived at the Fermi plant in February; construction and assembly of the facility was started under the supervision of personnel furnished by the vendor, North American-Rockwell. A description of the fuel transfer facility is given in APDA-CFE-10, and the facility is shown pictorially in Figures 10 and 11.

The new fuel transfer facility replaces the existing cask car, which has required an exorbitant amount of maintenance. Both facilities were designed to transport a shielded fuel cask between the transfer port in the fuel and repair building (FARB) and the reactor exit port in the containment building. At the present time, only the FARB section of the transfer facility is being erected and assembled. Delivery of the containment building section of the facility which has been delayed because the gripper cask was accidentally damaged during preparation for shipment from the vendor's plant, is not expected until the latter part of the year.

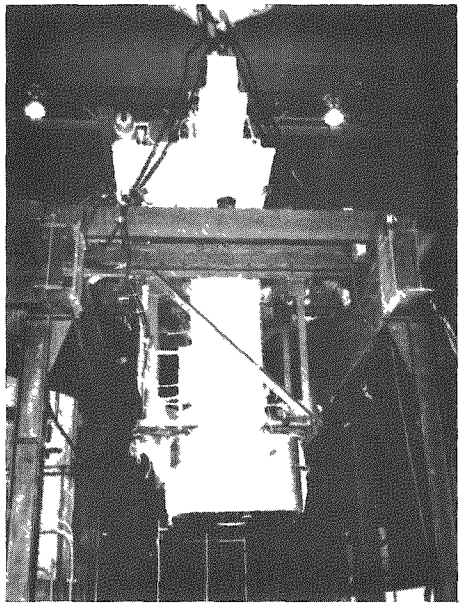

FIG.10 FARB SECTION OF FUEL TRANSFER FACILITY SHOWING GRIPPER CASK AND SUPPORT STRUCTURE

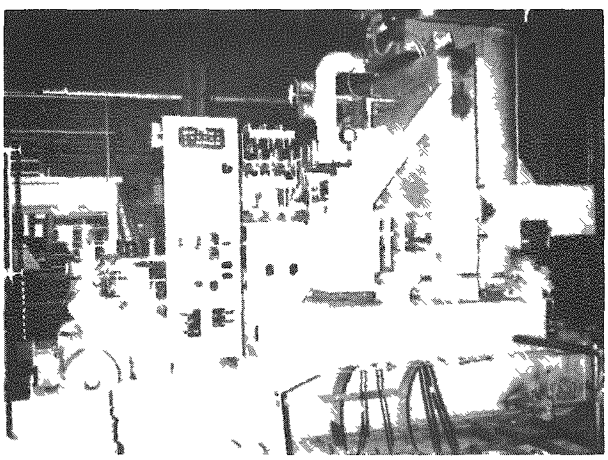

FIG.11 FUEL TRANSFER FACILITY TRANSPORT CAR

SECTION VI

SUBASSEMBLY INSPECTIONS AT BMI

Observations were made at Battelle Memorial Institute to determine the extent of phase transformation in the fuel alloy and hydriding in the zirconium cladding in transverse sections of fuel pins taken from subassemblies M099, M091, M122 and M140. Subassemblies M122 and M140 were adjacent to the melted subassemblies M098 and M127 in a four-subassembly cluster. Where possible, the specimens were directly viewed in the hot cell, under magnifications up to 250 times. Where direct viewing was impractical, replicas of the specimens were observed. The results of these observations, given below, are preliminary. An examination of the replicas of all specimens will be conducted at The Detroit Edison Company's Engineering Research Department before final conclusions are drawn.

Earlier results of inspection of the above subassemblies have been given in previous reports as follows:

Subassembly M099	- APDA-CFE-5, 6, 7, 16
Subassembly M091	- APDA-CFE-9, 10, 11
Subassemblies M122 and M140	- APDA-CFE-11, 12, 15, 16, 18

A. INSPECTION OF SUBASSEMBLY M099

Transverse Sections, Fourth Grid from the Top - Specimens from 10 fuel pins were observed. There is 75% to 90% fuel alloy transformation to the alpha-plus-delta phase. One pin, with an original gamma grain size twice that of the other pins, showed 60% transformation. This is to be expected because transformation starts at the grain boundary and grows into the grain. The hydride content in the cladding is estimated to be less than 400 ppm.

B. INSPECTION OF SUBASSEMBLY M091

Transverse Sections, Fourth Grid from the Top - Specimens from 10 fuel pins were observed. Transformation of fuel alloy is greater than 75%. The hydride content in the cladding is estimated to be less than 200 ppm.

Transverse Sections, Tenth Grid from the Top - Specimens from 10 fuel pins were observed. Transformation of fuel alloy is greater than 80%. The hydride content in the cladding appears to be less than 200 ppm.

C. INSPECTION OF SUBASSEMBLY M122

Transverse Sections, Fourth Grid from the Top - The fuel alloy of all pins appears to be entirely in the gamma phase. The implication is that all pins at this cross section had reached temperatures above 1100 F. The basis of this conclusion is that there would have been greater than 75% fuel alloy transformation to the alpha-plus-delta phase as a result of the subassembly's thermal history prior to the October 1966 incident, but once a temperature above 1100 F was reached, there would have been a retransformation to the gamma phase that would be retained on cooling.

The hydrogen content in the cladding will be estimated in the near future, after the samples are etched and the microstructures replicated. Some pins appear to have larger-than-normal diffusion zones between the fuel and the cladding.

Transverse Sections, Tenth Grid from the Top - Observations and the conclusion of retransformation are the same as those for the transverse sections, fourth grid from the top. Based on examination of specimens from 12 pins, it appears that the cladding of some pins has a hydride content of about 600 ppm. In several other pins the content is only about 250 ppm. It is further concluded that significant flow blockage occurred in subassembly M122 as well as in M098 and M127, the melted subassemblies.

D. INSPECTION OF SUBASSEMBLY M140

Transverse Sections, Fourth and Tenth Grids from the Top - Transformation of fuel alloy in Subassembly M140 varied from pin to pin. The fuel alloy on pins near the wrapper can show less transformation to alpha-plus-delta phase than pins near the center of the fuel bundle. The extent of transformation varies between 25% and 70% from the outer rows of pins toward the center of the fuel bundle. From this observation, it does not appear that subassembly M140 experienced a large amount of coolant blockage, if any. Rather it appears that the temperatures in M140 were influenced by those in the adjacent, coolant-starved fuel subassemblies.

The zirconium cladding on the pins exhibiting partial transformation appeared to contain not more than 200 ppm hydrogen.

SECTION VII

MAINTENANCE

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

At the end of February, 564 of the 1200 tubes had been rewelded to the water manifold tubesheet. Details of the repair weld technique have been given in several earlier reports starting with APDA-CFE-7. The No. 3 steam generator rewelding, started in early February, is expected to be completed about the middle of March.

B. LEAK TESTS OF THE NO. 1 STEAM GENERATOR

On two occasions, a gas pressure test was performed to verify the integrity between the tube and shell sides of the No. 1 steam generator. On both occasions, the test procedure was identical with that for pressure-testing the No. 1 steam generator after the unit was filled with sodium in December*. In these two recent leak tests, there was no measurable leakage from the tubes containing nitrogen at 600 psig. Three successive leak tests verify that there has been no change from the tube-to-shell integrity determined immediately after the steam generator was filled with sodium.

C. PRIMARY SHIELD TANK PRESSURE DECAY TESTS

Primary shield tank pressure decay tests were repeated in February, after the replacement of O-ring seals between the penetration cover plates and the machinery deck over the primary shield tank. The replacement of the seals was undertaken after the discovery of excessive leakage from the primary shield tank in January.**

The repeat pressure decay tests indicated leakage of 2000 standard cubic feet per day (scfd) which is less than the maximum allowable of 2400 scfd for reactor operation. It is planned, however, to visually inspect the outer surface of the primary shield tank for possible sources of leaks in the near future if personnel access can be provided.

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* See Section VI. A. 2 of APDA-CFE-17.
** See Section VII. A of APDA-CFE-18.

D. LEAK TEST OF PERSONNEL AIR LOCK

In a recent leak test of the personnel air lock in the containment building, the leak rate was found to be 155 scfd. This leak rate was considered satisfactory, based 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.

In the leak test procedure, an air pressure of 32 psig was established in the personnel air lock and gross leak checks were made by the soap bubble method applied at both the inner and outer door seals and at the door penetration seals, the obvious source of leaks. The inner door isolates the air lock from the atmosphere within the containment building, and the outer door from the outside atmosphere. Since there were no gross leaks, the leak rate was then measured by the pressure decay method during a 24-hour test period. Compressed air was admitted to the personnel air lock through a penetration in the inner door to establish initial test pressure. A pressure indicator monitored air lock pressure through a second penetration in same door.

E. LEAK TEST OF EMERGENCY AIR LOCK

The emergency air lock in the containment building was leak-tested in the same manner as for the test of the personnel air lock. Gross leakage through the outer door was discovered and as a result, the leak test was discontinued and the seal for the door-opening handwheel shaft replaced. The leakage test will be resumed shortly.