ENRICO FERMI ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT FEBRUARY 1969

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TABLE OF CONTENTS

<u>Section</u>	<u>Page</u>
LIST OF ILLUSTRATIONS	v
PREFACE	1
I. CURRENT EXPERIENCE SUMMARY	3
II. RESTORATION OF PRIMARY SYSTEM FOR SODIUM REFILL	5
A. BACKGROUND	5
B. COMPLETION OF LOWER PLENUM VACUUMING	5
C. PLENUM INSPECTION	6
D. PATCH REPLACED IN 14-INCH LINE PENETRATION ..	6
III. MOCKUP TESTS OF HOLDDOWN MECHANISM INSPECTION TOOL	13
A. BACKGROUND	13
B. HOLDDOWN MECHANISM INSPECTION TOOL	14
C. MOCKUP TESTS	19
IV. SPECIAL INVESTIGATIONS	21
A. NEUTRON DETECTOR TUBE COOLING CHECK	21
B. TRANSFER TANK PLUGGING INDICATOR RETURNED TO SERVICE	21
V. OPERATIONS	23
A. STATUS OF REACTOR SODIUM AND GAS SYSTEMS	23
B. GAS SYSTEMS PERFORMANCE	23
C. STEAM GENERATORS	24
1. Steam Generator Water Manifold Cover Gasket Replacements Completed	24
2. Secondary System Sodium Refill and Cold Trapping...	25
3. Steam Generator Leak Test	25
D. RADIOACTIVE WASTE DISPOSAL	26
VI. PLANT MAINTENANCE	27
A. REPLACEMENT OF WASTE GAS LINES	27
B. REPLACEMENT OF ARGON GAS SEAL FOR NO. 3 PRIMARY SODIUM PUMP	30
C. REMOVAL OF MOISTURE INDICATOR IN PRIMARY COLD TRAP ROOM	30

TABLE OF CONTENTS (Cont'd)

<u>Section</u>	<u>Page</u>
D. SAFETY ROD HANG-UP IN STEAM CLEANING CHAMBER GRIPPER	31
VII. ERECTION OF AUXILIARY FUEL STORAGE FACILITY INSIDE REACTOR BUILDING	35
A. BACKGROUND	35
B. GENERAL DESCRIPTION OF THE FACILITY	35
C. SHIELD PLUG CORING	36
VIII. ERECTION OF NEW FUEL TRANSFER FACILITY IN REACTOR BUILDING	43
A. BACKGROUND	43
B. EXIT PORT MODIFICATIONS	45
C. ERECTION OF SUPPORT STRUCTURES	46

LIST OF ILLUSTRATIONS

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
1.	Zirconium Pan in Mockup Facility Showing Method of Construction	7
2.	Photograph of Typical Zirconium Pan Sector in Reactor Following Vacuuming	7
3.	Initial 14-Inch Line Patch Weld Working Through Glove Box	9
4.	Completed Root-Pass Weld	9
5.	Final Weld (After Repairs)	9
6.	Final Weld Viewed Through Secondary Containment	10
7.	Vacuum Adapter Box Used for Mass Spectrographic Leak-Testing	10
8.	Holddown Mechanism	14
9.	Holddown Inspection Tool in Mockup Facility	15
10.	Mirror and Post	16
11.	Light Source with Protective Screen and Pyrex Tube Removed ..	16
12.	Holddown Inspection Tool in Mockup Showing Umbilical Cord Layout	17
13.	Close-Up of Holddown Mechanism Inspection Tool in Mockup	17
14.	Routing of New Waste Gas Lines	28
15.	Fuel and Repair Building	32
16.	Using Mirror to View Defective Gripper over Temporary Shield Wall in Dry Loading Tunnel	33
17.	Gripper After Delatchment from Safety Rod	33

LIST OF ILLUSTRATIONS (Cont'd)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
18.	Auxiliary Fuel Storage Facility	36
19.	Oxygen Lance Used for Burning Holes Through Steel	38
20.	Close-Up of Oxygen Lance	38
21.	View of No. 2 Equipment Decay Tank Shield Plugs Showing Location of Loading Port and Rotor Shaft Penetrations (Cover of Inner Plug Removed)	38
22.	Close-Up of Loading Ports Before Concrete Coring	39
23.	Diamond-Tipped Coring Tool	39
24.	Coring Machine During Operation	39
25.	Lifting Out Partial Core	40
26.	Core Lifted Out	40
27.	Stacked Cores from One Access Port Penetration	40
28.	Completed Stepped Loading Port — Bottom 7 Inches of Steel Still Intact	40
29.	New Fuel Transfer Facility in Fuel and Repair Building.....	44
30.	Reactor Building Exit Port with Shielding in Floor Removed	44
31.	Old Exit Port Shield Seal Flange.....	44
32.	New Steel Flange	44
33.	Exit Port Reassembled	47
34.	Drilling Holes for Anchor Bolts of FTF Support Columns	47
35.	Close-Up of Drilling Apparatus	47
36.	Anchor Bolts in Place	47

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-28.

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

During February restoration work in the primary system was continued in preparation for sodium refill. Vacuuming of the lower plenum was completed; very little debris was found. Inspection indicates that the plenum floor is clean and the zirconium pan is in good condition. The patch was replaced on the penetration in the 14-inch inlet line. Examination indicates the original integrity of the pipe has been restored.

Testing of a tool to inspect the fingers of the holddown mechanism has been completed. Tests in the mockup facility determined the reactor subassembly configuration required for insertion of the tool and verified that adequate lighting and mirror image photographs of the holddown fingers could be obtained. Pressure checks made on the neutron detector tube cooling system to verify the modifications of last month indicate a large improvement in neutron detector cooling gas flow has been achieved.

The transfer tank plugging indicator has been returned to service following the repair of last month. Cold-trapping of the transfer tank sodium reduced the plugging temperature from 330 F to 250 F.

Replacement of the water manifold cover gaskets on the steam generators was completed. The three secondary system loops were refilled with sodium and cold-trapped to reduce plugging temperatures to less than 250 F. Nitrogen leak tests on the steam generators indicate no water tube leaks in the No. 2 and No. 3 units. The No. 1 unit has a small leak of about 100 cm³ of nitrogen per day at 22 psig pressure differential.

Work is nearly completed on replacement of three underground waste gas lines and one underground liquid waste line, all of which have suffered corrosion. The replacement lines are being rerouted above ground in the overhead pipe racks.

Hang-up of a safety rod in the gripper of the steam cleaning chamber has occurred. The safety rod, one which had been removed from the reactor 2-1/2 years ago, was being transferred to a new finned pot prior to reinsertion in the reactor. The gripper fingers cannot be delatched and the rod has been left in the steam-cleaning chamber while provisions for repairs are being made.

Preparations for installation of the reactor building portion of the new fuel transfer facility equipment have begun. This equipment is similar to that previously installed in the fuel and repair building.

Construction has begun on an auxiliary fuel storage facility, which is to be located inside an equipment decay tank in the reactor building. The facility will be used to store a portion of the first core when it is removed from the reactor prior to loading a new core for reactor startup later this year.

II. RESTORATION OF PRIMARY SYSTEM FOR SODIUM REFILL

A. BACKGROUND*

In January, the zirconium screw shanks on the north half of the conical flow guide were welded permanently in place and the cone surface scraped of any loose weld material. The converted arc-melt tool vacuum cleaner was then installed through the rotating shield plug, and the north-west sector of the plenum was vacuumed. The plenum vacuuming was completed this month and the patch on the 14-inch inlet line restored in preparation for sodium refill.

B. COMPLETION OF LOWER PLENUM VACUUMING

Vacuuming of the plenum in the vicinity of the flow cone using the converted arc melt tool vacuum cleaner was completed on February 10. The operation went smoothly. The time to vacuum the remaining sectors was accelerated by reducing the extent of the photographic survey. All debris of any substance in the vicinity of the flow cone was picked up. As expected, very little weld debris was seen. Another stain was discovered, however, on the wall of the reactor vessel similar to that found earlier on the plenum floor near the flow cone.** It is not known what causes these stains; they have no apparent thickness. A combined visual-photographic inspection of the plenum shows all areas to be free of debris. Only sodium droplets are present. This is also true in the area outboard from the flow cone, around the plenum periphery; therefore, no vacuuming was performed using the alternate vacuum cleaner designed for installation through the 14-inch inlet line penetration.***

When the floor collection tank (see APDA-CFE-29, Sec. III. B) was opened following completion of vacuuming, the only debris found consisted of a number of pea-size pieces of metal and some flattened metal pieces about 1/2-inch long and 1/4-inch wide. Both apparently consisted of weld slag from the earlier arc-melting process. This debris, along with a small quantity of sodium fluff material collected, was put in plastic bags for future examination. Inspection of the fiber glass filter in the tank revealed only sodium oxide deposits in the form of fine dust. The radiation level from the floor tank was 0.01 mr/hr with the top cover on and 2.5 mr/hr with the cover

* APDA-CFE-29

** APDA-CFE-29, Sec. III, C.

*** APDA-CFE-28, Sec. II, G.1

removed. The debris trap (see APDA-CFE-29, Sec. III, B) in the vacuum cleaner will be opened for inspection in the near future.

When the vacuum cleaner was taken out of the reactor, it was discovered that the articulated arm on its lower end had become unscrewed from the upper portion of the tool and was being held in place by the cable attached to the flexible metal hose extension located inside the articulated arm.

C. PLENUM INSPECTION

A visual inspection of the lower plenum was made following vacuuming to determine the condition of the zirconium pan which lines the floor of the plenum. The pan, shown in Figure 1, is constructed from twelve 1/8-inch-thick, pie-shaped zirconium sheets formed to the plenum floor. Approximately a 3/8-inch thermal expansion gap exists between individual sheets, and zirconium batten strips are placed over the gaps, overlapping each adjacent zirconium sheet. Stainless steel bolts affixed to the underlying stainless steel shielding extend up through the batten strips to secure the zirconium sheets. Acorn nuts made of zirconium are placed over the tops of the bolts and secured to the bolts by stainless steel wire passing through holes in the nuts and bolts. Visual and photographic inspection (Figure 2) of this construction in the lower plenum has revealed all acorn nuts and batten strips to be in place (see APDA-CFE-29, Figure 13).

An inspection was also made of the area immediately underneath the lower support plates, where the second loose zirconium segment was found (see APDA-CFE-27, page 7). There was some question originally as to what was holding the plate up in this position. Review of the design details and visual inspection revealed there is a ledge in this area formed by the interlocking thermal shield plates of the inlet plenum into which the zirconium segment could have become lodged.

D. PATCH REPLACED IN 14-INCH LINE PENETRATION

Following detailed inspection of the lower inlet plenum, the patch was replaced in the 14-inch inlet line elbow. Since access to the lower plenum is not expected to be required again, welding of the patch began in the middle of the month. Some weld anomalies were detected which required grinding and rewelding to obtain a satisfactory weld. The work was completed at the end of the month.

The welding was done under unusual conditions. Figure 3 shows the root pass weld being made using a glove box and working through the secondary pipe containment. The glove box was removed for the final closure weld, making access somewhat easier; nevertheless, it was a difficult weld to make

**FIG. 1 ZIRCONIUM PAN IN MOCKUP FACILITY
SHOWING METHOD OF CONSTRUCTION**

**FIG. 2 TYPICAL PHOTOGRAPH OF ZIRCONIUM PAN
SECTOR IN REACTOR FOLLOWING VACUUMING**

because the penetration was located on the pipe elbow inside the secondary containment and the weld was oblong on the pipe surface.

To ensure that the original integrity of the pipe had been regained, an extensive program of weld inspection was conducted. The root pass weld was X-rayed by placing the radiation source behind the pipe, in the annulus between the primary pipe and the secondary containment, and placing the radiographic film over the weld on the front of the pipe. Radiographs were then taken through the double layer thickness of pipe wall. Figure 4 shows the root pass weld after completion. The root pass weld passed radiograph inspection on the first attempt and no repair was required; however, difficulty was encountered in obtaining radiographic approval of the final closure weld. Shadows appeared on the early radiographs, indicating lack of penetration. The affected areas had to be ground out twice and rewelded before satisfactory radiographs were obtained.

Figures 5 and 6 show the closure weld finally accepted as satisfactory. A horizontal weld mark across the upper half of the patch, seen in Figures 4 and 5, is the weldment left from the welding of the original (smaller) patch last year, following removal of the first bent zirconium segment from the reactor.* The new larger penetration was made by cutting along three sides of the old patch, leaving only the horizontal crescent shown.

In addition to radiography, the final weld was inspected by dye-penetrant, mass-spectrographic, and ultrasonic examination. The mass-spectrographic helium leak test was conducted by introducing helium into the reactor vessel and sealing a vacuum box on the outside of the pipe over the patch. Some difficulty was experienced in fitting the mass-spectrographic vacuum box over the patch because of the irregularity of the patch from the normal pipe contour. The vacuum adapter box had to be altered to fit the pipe contour and Silastic rubber used to seal the larger gaps. Figure 7 shows the box in place over the patch. A calibrated leak located on the adapter box gave a leak detection sensitivity of approximately 3×10^{-11} cubic feet per hour. No readable leakage was obtained in the test.

Ultrasonic inspection of the patch weld was made because it offers the possibility of detecting internal cracks and inclusions which cannot be found by the other methods. A 2.25-megacycle, 45° angle beam-type SM-2 transducer 1/8 inch wide by 1/2 inch long was used. A test weld having a known defect was used in the laboratory to calibrate the device. A milled notch 60 mils deep gave full-scale deflection of the oscilloscope. The patch

* APDA-CFE-20, page 22 and APDA-CFE-27, Figure 5

**FIG. 3 INITIAL 14 INCH LINE
PATCH WELD WORKING
THROUGH GLOVE BOX**

FIG. 4 COMPLETED ROOT-PASS WELD

FIG. 5 FINAL WELD (AFTER REPAIRS)

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**FIG. 6 FINAL WELD VIEWED THROUGH
SECONDARY CONTAINMENT**

**FIG. 7 VACUUM ADAPTER BOX USED FOR MASS
SPECTROGRAPHIC LEAK-TESTING**

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weld on the 14-inch line was tested from all directions by passing the crystal around the weld periphery, both inside and outside the weld ring. No defects were seen on the oscilloscope pattern.

The secondary containment pipe patch is scheduled to be welded back in place next month. It is planned that several temporary holes will be drilled in it to allow a final visual inspection of the primary pipe patch following sodium refill. The patch area, except for the holes, will be covered with a removable layer of insulation and the induction heating wires replaced to avoid creation of a cold spot at this location. Following inspection, the holes in the secondary containment will be plug-welded and the entire patch checked by dye-penetrant and ultrasonic examination.

III. MOCKUP TESTS OF HOLDDOWN MECHANISM INSPECTION TOOL

A. BACKGROUND

The reactor holddown mechanism, shown in Figure 8, is located inside the reactor vessel above the core and is supported by the vessel rotating shield plug. The three alignment spider arms of the holddown rest on supports in the outer blanket. The central column has a mounting plate containing holddown fingers which engage with the subassembly handling heads in the high-pressure region. The device provides holddown restraint to prevent subassemblies in the high-pressure region from floating up out of the core. It also cams in on the subassembly handling heads in such a way that, when it is lowered, the core is gathered together tightly and the subassemblies slightly prebowed.

A holddown mechanism inspection tool has been built to view the holddown fingers before reloading new fuel. Inspection of the fingers is desired because it is known that larger than normal forces on some fingers occurred at the time of the October 5, 1966 incident (see APDA-CFE-5, Sec. III, and CFE-6, Sec. II). Postincident observations revealed several rows and columns of subassemblies in the lattice had suffered abnormal bending stresses with some subassemblies being found permanently bent upon removal from the reactor (APDA-CFE-7, Sec. III, C, and APDA-CFE-11, Sec. IV, B). Although calculations show that the holddown fingers are sufficiently strong to resist deformation from the bent subassemblies, a dimensional check to verify the calculations is desirable.

Previous attempts were made to observe and photograph the condition of the holddown fingers shortly after the incident (APDA-CFE-19, page 13). Although the earlier inspection tool was similar to the new device recently designed and built, it had significant deficiencies and the earlier survey was not successful. The earlier tool consisted of a post set in one of the support plate positions. A spherical mirror set on top of the post reflected images of the holddown fingers to a borescope located in an adjacent rotating plug penetration position. Pictures of the fingers were taken through the borescope. Lighting of the holddown fingers was achieved using a 1500-watt Quartzline light placed in the vessel through an adjacent plug penetration.

Difficulties were encountered in the operation of this tool because the light was projected on the holddown fingers in a horizontal plane and shadows arose. Also, the early version of the light was not purged to keep sodium vapor away from the lamp surface, and many light failures occurred.

FIG. 8 HOLDDOWN MECHANISM (SECTIONAL ELEVATION)

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Lastly, the spherical mirror was not remotely adjustable and accurate orientation was not possible prior to its insetion into the reactor vessel.

The new holddown mechanism inspection tool has been designed to overcome the earlier difficulties. It has a self-contained light source which shines on the holddown fingers from below, thus eliminating shadows. Also, the light source is purged to minimize light failures. An improved method of applying silver to the spherical quartz mirror will hopefully eliminate much of the fogging encountered with the old mirror. Finally, the new tool uses a motor-driven, articulated mirror. This will allow remote positioning of the mirror to obtain the desired field of view.

The inspection tool is at present undergoing proof-testing in the mockup facility. Following completion of the mockup tests and reactor vessel sodium refill, fuel movements to allow location of the inspection tool will be made. The vessel will then be drained, the inspection tool inserted, and the holddown fingers photographed through a borescope focusing on the mirror.

B. HOLDDOWN MECHANISM INSPECTION TOOL

Photographs of the holddown mechanism inspection tool taken in the mockup facility are shown in Figure 9 through 13. Basically, the tool consists of a long cylindrical aluminum post, supporting the mirror and light source, which can be inserted in a blanket lattice position. The motor-driven, rotating mirror is mounted on top the post. Power and purging lines are supplied through a flexible, metal-sheathed umbilical cord attached to a U-shaped hinged bail located at the top of the post over the mirror. The umbilical cord is enclosed in a rigid outer metal sheath for insertion in the reactor.

The tool will be inserted in the reactor in lattice position N07-P00. This position is located in the first row of the outer radial blanket on the reactor east-west centerline. Prior to insertion, a wedge shaped sector of subassemblies will be removed from around the position to provide room for insertion and operation of the tool (Figure 13). The size and shape of the core cavity is determined by the optical line-of-sight requirements between the holddown fingers, mirror, and borescope, and the path used for laying the umbilical cord out on top the core. The device will be inserted through the 6-inch penetration in the No. 3 rotating shield plug. The borescope with camera attachment will be installed in the adjacent No. 4 6-inch penetration. While the post is being lowered into position in the support plates, the 3-inch-diameter umbilical cord will remain fully enclosed inside its outer sheath. Once in position, the outer sheath will be raised approximately 20 feet, exposing the flexible stainless steel umbilical core. The plug will then be rotated away from its position over the post. The resultant drag forces on the core will lower it along the predetermined path in the core cavity, and on top of the subassembly handling heads. The

**FIG. 9 HOLDDOWN MECHANISM
INSPECTION TOOL IN
MOCKUP FACILITY**

FIG. 10 MIRROR AND POST

**FIG. 11 LIGHT SOURCE WITH
PROTECTIVE SCREEN AND
PYREX TUBE REMOVED**

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Dummy Holddown
Finger Block

**FIG. 12 HOLDDOWN MECHANISM
INSPECTION TOOL IN MOCKUP
SHOWING UMBILICAL CORD LAYOUT**

**FIG. 13 CLOSE-UP OF HOLDDOWN MECHANISM
INSPECTION TOOL IN MOCKUP**

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bail will be automatically lowered out of the field of view during this operation.

In position, the power cables for the lights and mirror motor pass down one side of the bail, and the argon gas cooling line for the lights passes down the other side (Figure 10). The umbilical cord will be laid out on top of subassembly handling heads over a portion of its path. To avoid the possibility of the upper safety rod guide tubes hitting the hose and becoming damaged during plug rotations, the core and safety rod guide tubes will be viewed through the borescope to ensure that no interferences exist.

In position, the viewing mirror is at an elevation slightly below the top of the surrounding subassembly handling heads. The mirror support is connected to the drive motor located inside a motor housing on the post. The design is such that the azimuth and pitch positions of the mirror can be changed independently using the single drive motor. This design provides much greater flexibility in photographing the holddown fingers than existed previously. Following plug rotation to a new viewing position, the desired field of view can be obtained by making adjustments in the mirror position. Only a relatively few plug positions will be needed for full view coverage of the holddown fingers.

The silvering method used for the new spherical mirror is expected to make the mirror less susceptible to sodium vapor fogging than the earlier mirror. However, because of experiences with sodium drippage from vessel surfaces in the segment-removal and vacuum-cleaning program, there is concern that the mirror may become spotted with liquid sodium. If it cannot be shaken off by mirror rotation and tilting, the mirror will in all likelihood require resurfacing unless the photographic surveys can be completed in a relatively short period of time. The improved design of the new inspection tool affords hope that this can be done. The presence of a few sodium droplets on the mirror surface should not significantly affect the quality of the images obtained, however. Fuzzy images can probably be eliminated using the variable pitch and rotation of the mirror.

The post contains its own light source (Figure 11) located near the middle and around the periphery of the post. The light source consists of four 1500 watt Quartzline lamps, surface purged by means of the argon line which enters the motor housing at the top of the post from the umbilical cord. Holes in the bottom of the motor housing allow gas to flow to the light located below. A Pyrex tube around the lights aids in guiding the gas flow. The gas then passes out the bottom of the post. The lights and Pyrex tube are contained inside a wire mesh blanket for protection against breakage, and to contain any pieces if breakage does occur. The holddown finger lighting is dependent on light reflected upward from the walls of the core cavity. The light available for picture taking should be much better than with the previous device; shadows will be eliminated and lamp failures will be minimized.

Photographs taken through the borescope will be used to determine if any holddown fingers are out of alignment. The photographs will give elliptical images of the fingers in the field of view. The square array of hold-down fingers will appear as curved rows in the photograph (see APDA-CFE-26, Figure 24). The photograph is analyzed graphically in the following manner: A curved mesh is laid out over the photograph, each finger being located at the intersection of mesh-lines; tangents to the fingers at the points where the mesh lines intersect are drawn to obtain the curvature of each finger in a row in the image on the photograph; the apparent curvature of each finger in a row versus finger position is then plotted. If the fingers are properly aligned, the change in curvature between fingers will be a smooth, uniformly varying curve. If a finger is out of alignment, a sharp discontinuity occurs in the plot. The minimum amount of finger distortion which can be detected by this method was determined in tests using a mockup of the holddown mechanism inspection tool (see APDA-CFE-26, page 29). Alignment verification to within $\pm 1/32$ of an inch was indicated. A resolution of approximately $\pm 1/16$ of an inch is required to determine whether any significant displacement of the holddown fingers has occurred.

C. MOCKUP TESTS

During February, the holddown mechanism inspection tool was tested in the mockup facility. The tests resulted in several minor modifications in the purging system and mirror design; however, no major design changes were made. The device was determined to be mechanically sound in regard to its post, lighting, mirror and umbilical cord design, and was accepted for use in the reactor.

The majority of the tests involved determining the shape and size of the core cavity for optimum hose layout and holdown finger viewing. The tests verified that all four quadrants of the holddown finger array could be seen adequately with the core cavity chosen. The plug azimuth and mirror rotation positions required for full coverage were determined. Although sub-assembly distortions occurred under only one quadrant of the holddown finger array, all four quadrants will be photographed.

In photographic tests, two sets of photographs were taken through the borescope. In the first, pictures of a dummy holddown finger block were obtained. The block was painted to simulate the poor reflectivity existing in the reactor. In the other photographs lined paper was used in place of the dummy holddown finger block. The photographs of the dummy holddown fingers have not yet been fully analyzed; however, good clarity photographs were obtained which verified the adequacy of the inspection tool optics and lighting under the test conditions.

The grid photographs were taken to provide a rectangular subject as a standard of reference for verifying the shape of the pin position versus curvature curve for perfect pin alignment. The photograph will be analyzed by plotting the curvature of the grid lines on the photograph as a function of the pseudo pin positions.

Consideration was also given to the development of a mathematical method by which the curved images on the photographs could be translated to the linear rectangular coordinate system of the actual finger array. Investigation showed that the focal lengths of the camera, borescope and other parameters required as input in the analysis were not well known. Analytical analysis of the holddown finger photographs has therefore been abandoned in favor of the graphical method, which eliminates the need for data on the characteristics of the mirror, borescope, and other dimensional parameters.

All scheduled mockup testing of the holddown mechanism finger inspection tool has been completed. The principal work remaining involves completing the evaluations of the photographs of the dummy finger array and reference grid.

IV. SPECIAL INVESTIGATIONS

A. NEUTRON DETECTOR TUBE COOLING CHECK

Repairs were made last month on the neutron detector tubes located in the graphite shield to eliminate the loss of cooling gas from the tubes (see APDA-CFE-29, Sec. V. A). To determine whether the repairs were successful, pressure checks were made this month in the neutron detector tube cooling circuit. A water manometer was placed in the 4-inch coolant supply line at the tube inlet header, and a pressure of 8 inches WCG was measured. Prior to repairs only 4 inches WCG pressure had been measured.

Pressure measurements were not repeated at the ventilation pipe which exhausts from the outlet header of the detector tube cooling circuit; however, a significant gas flow out the ventilation pipe can be detected. Before repairs, no perceptible gas flow from the pipe could be detected. This also indicates considerable improvement in detector tube cooling.

Meaningful information cannot be obtained at present from temperature measurements made in the detector tubes with the cooling system turned off and on. The reactor vessel is still drained of sodium and is being maintained at a temperature of approximately 320 F; consequently, graphite temperatures in the vicinity of the detector tubes are only of the order of 100 F, approximately the same temperature as that of the cooling gas. The primary system will have to be filled with sodium and a system temperature of 500 to 600 F achieved before meaningful temperature measurements can be made to determine the effectiveness of the cooling changes.

B. TRANSFER TANK PLUGGING INDICATOR RETURNED TO SERVICE

During January (see APDA-CFE-29, Sec. V. C.2) attempts were made to re-establish sodium flow through the transfer tank plugging indicator circuit in the fuel and repair building following replacement of a distorted line in the circuit's electromagnetic pump. They were unsuccessful, and further investigation revealed the plugging orifice flow plate was plugged with foreign deposits which were not readily identifiable. Deposits were also flushed from the sodium outlet line upstream from the orifice, the restrictions were removed, and by the end of the month the circuit was being reassembled with special argon test lines installed on either side of the orifice location.

A sodium plugging temperature of about 330 F was measured initially for the transfer tank system with the bulk sodium temperature at 425 F. Cold-trapping operations for 16 hours reduced the plugging temperature to

below 300 F; at the end of 24 hours, it had been reduced to 250 F. Operation of the plugging indicator and cold trap continued to be satisfactory until the last week of the month, when a leak was detected at a valve in a bypass line of the cold trap system. Bellows failure was indicated, and removal of the leak detector showed the presence of sodium and confirmed failure of the bellows. Until a replacement bellows is received, cold-trapping and operation of the valve will be continued as conditions permit. The leak is small, and operation with a small leak in the valve body is possible because leakage to the air atmosphere of the fuel and repair building cold trap room is prevented by the nitrogen-filled secondary containment around the piping. Although leakage down the valve stem would normally be prevented by the backup freeze seal maintained near ambient temperature (see APDA-CFE-11), the freeze seal is being kept at elevated temperature to allow operation of the valve.

The failed bellows was placed in service in 1966 and, at the time of failure, had been stroked over 100 times.

V. OPERATIONS

A. STATUS OF REACTOR SODIUM AND GAS SYSTEMS

The primary system remained drained of sodium throughout the month to allow completion of vacuuming inside the reactor vessel and closing of the 14-inch pipe penetration. During this time, the temperature of the argon-filled primary system was maintained at approximately 320 F by the pipe and vessel induction and resistance heaters. The secondary system was refilled with 350 F sodium following completion of the steam generator manifold gasket replacements and secondary system cold-trapping was performed. No movement of subassemblies was made during the month. Sodium activity was not measured.

B. GAS SYSTEMS PERFORMANCE

Since the last reported data (APDA-CFE-29), the following primary system cover gas activity and gas analysis information have been obtained.

Primary System Cover Gas Activity

<u>Location</u>	<u>Sample Date</u>	<u>Gross Beta Activity, $\mu\text{Ci}/\text{cm}^3$</u>
Reactor Cover Gas	2-6-69	1.50×10^{-6}
Reactor Cover Gas	2-13-69	1.11×10^{-6}
Reactor Cover Gas	2-20-69	1.61×10^{-6}
Reactor Cover Gas	2-27-69	1.78×10^{-6}
Primary Shield Tank	2-6-69	1.00×10^{-7}
Primary Shield Tank	2-13-69	3.50×10^{-8}
Primary Shield Tank	2-20-69	3.50×10^{-8}
Primary Shield Tank	2-27-69	3.66×10^{-8}

Primary System Cover Gas Analysis *

	<u>Reactor Cover Gas (Argon), ppm by Volume</u>	<u>Primary Shield Tank Atmosphere (Nitrogen), ppm by Volume</u>
Oxygen	Below 25	300 **
Carbon Monoxide	Below 10	Below 10
Carbon Dioxide	25	10
Hydrogen	Below 4 ***	Below 2.5
Helium	Below 4	Below 4
Methane	Below 10	Below 10
Nitrous Oxide	←———— Not Measured ———→	
Nitrogen	240	Remainder
Argon	Remainder	Not Measured
Dew Point	Not Measured	-55 F

1 Sample taken February 20, 1969.

2 Technical Specifications state 1000 ppm maximum.

3 The recommended maximum for reactor operation is 10 ppm.

C. STEAM GENERATORS

1. Steam Generator Water Manifold Cover
Gasket Replacements Completed

Reassembly work on the three steam generators was completed by the middle of February with new copper gaskets installed on the water manifold covers in place of the soft iron gaskets. Copper was the originally specified material for this application.

As reported earlier (see APDA-CFE-29, Sec. V.B), all orifices added in the water tubes of units No. 2 and No. 3 were found to be properly seated after the isolation tests; therefore, no further thermal shock tests are planned for these two units. Due to the tube leaks found in unit No. 1 during September, this unit has not been thermal-shock-tested. This will be done at a later date.

Hydraulic tests were conducted on unit No. 3 last year to verify the predicted feedwater flow and pressure drop characteristics following installation of the water tube orifices.* These tests indicated that the actual pressure drop across the unit at various flows were quite a bit less than predicted by calculation. Preliminary reviews now indicate that application of the correct flow equation will give calculated results in good agreement with the earlier test results. Pressure drop tests are scheduled on units No. 1 and No. 2 in the near future to verify their characteristics.

2. Secondary System Sodium Refill and Cold Trapping

Following replacement of the manifold covers, the three secondary sodium system loops were heated and refilled with 350 F sodium. The manifold covers were then tightened by retorquing the cover plate bolts. This work was completed during the third week in February. The steam generator cover gas systems were purged to reduce hydrogen and nitrogen levels and the sodium in Loops 1 and 3 was cold-trapped. In Loop 1, 176 hours of cold trapping reduced the sodium plugging temperature from 335 F to 220 F; in Loop No. 3, 36 hours of cold-trap operation reduced the sodium plugging temperature from 265 F to below 220 F. Loop No. 2 was not cold-trapped since its plugging temperature was already below 250 F. The secondary sodium system loops were heated to 450 F for the cold-trapping operations.

3. Steam Generator Leak Test

In preparation for filling the units with water, all three steam generators were leak-tested with nitrogen at 450 psig in the water tubes. There was no evidence of tube leaks in the No. 2 and No. 3 units; however, nitrogen was measured in a gas chromatograph sample taken from the argon cover gas over the sodium shell side of unit No. 1. The indicated leakage is small, amounting to about 100 cm³ of nitrogen per day at 22 psig. Before the three tubes were repaired on this unit in October, 1968, the leakage had been of the order of several thousand cm³ per day. The leakage has, therefore, been greatly reduced. Since the earlier tube leaks had been repaired by plug welding and the repaired tubes had been determined to be leaktight by bubble tests made in November, the new leaks must be from other tubes. Sodium will be circulated through the unit for several weeks to determine if the leakage increases. A decision will then be made as to whether the unit will be opened up to look for the leak, or whether it will be refilled with water. The current leak rate is thought to be marginal for localization by available detection methods.

* APDA-CFE's-26, pp 25; APDA-CFE-27, pp 24, and APDA-CFE-28, Sec. III, B. 2.

D. RADIOACTIVE WASTE DISPOSAL

During February, 3,642 gallons of liquid waste was discharged to the north lagoon. The specific activity in the liquid waste tank varied between $2 \times 10^{-3} \mu\text{Ci}/\text{cm}^3$ and $2 \times 10^{-4} \mu\text{Ci}/\text{cm}^3$. The total activity discharged to the lagoon was approximately 3.2 millicuries. All effluent released after dilution with circulating pump discharge was below maximum permissible concentration (MPC).

Approximately 6.5×10^8 cubic feet of gaseous effluent was discharged through the plant stack during February. The concentration of activities with half-lives greater than 8 days was less than $7.5 \times 10^{-6} \mu\text{Ci}/\text{cm}^3$ at the waste gas stack outlet. The concentration of all isotopes was less than $3 \times 10^{-5} \mu\text{Ci}/\text{cm}^3$ at the stack outlet. After dilution with the atmosphere, all activity was reduced to much less than MPC.

The maximum activity measured in the cutup pool in the fuel and repair building was about $7 \times 10^{-7} \mu\text{Ci}/\text{cm}^3$.

VI. PLANT MAINTENANCE

A. REPLACEMENT OF WASTE GAS LINES

A program is underway to replace several underground waste gas lines with new above-ground lines. The old lines have either failed or are believed near failure due to corrosion. Numerous corrosion failures of buried waste lines have occurred over the past several years, requiring replacement of the original lines with above-ground lines. The work now in progress involves three waste gas lines. A schematic drawing of the old lines and their new routing is shown in Figure 14.

Lines 21 and 22, which are 2-inch-diameter carbon-steel lines, are presently routed together inside a deeply buried 8-inch-diameter, asphalt-coated, carbon-steel secondary containment pipe running between the waste gas building and the fuel and repair building. Line 21 originally carried waste gas from the fuel and repair building to a valve pit located near the stack, and from there to the waste gas building header. The new line is being routed approximately 8 feet above ground in the existing yard pipe trestle on the opposite side of the fuel and repair building from the original line; i. e., along the south and east walls of the fuel and repair building. It will tie into the old Line 21 at the northeast corner of the fuel and repair building just inside its east wall. Line 22 carries waste gas from the storage tanks in the waste gas building to the valve pit near the stack. At this point it leaves the containment pipe, runs underground to the stack, and rises vertically into the stack breeching. The section of Line 22 from the valve pit to stack had to be repaired in May of last year because of leaks (see APDA-CFE-22, page 31 and Figures 16 and 17). In December 1968 this same section of line was replaced with a temporary above-ground line as a result of flow interruptions caused by plugging of the stack diffuser with dirt (see APDA-28, page 35). The section of Line 22 from the valve pit to the waste gas building can still be used to make discharges to the stack, however.

The replacement for Line 22, presently being fabricated, is being routed above ground over the same route as the replacement for Line 21. It will enter the east wall of the fuel and repair building and exit through the north wall to the diffuser in the stack breeching. At the waste gas building, both new lines are being tied into the existing lines below ground

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FIG. 14 ROUTING OF NEW WASTE GAS LINES

at the point where the old lines enter the inert gas tunnel which runs between the reactor building and waste gas building.

The maximum dose rate measured on the surface of old Line 22 during previous reactor operation was 0.2 mr/hr. Radiation monitoring equipment installed in the waste gas building prevents Line 22 from handling gas with a specific activity greater than $1 \times 10^{-3} \mu\text{Ci/cc}$, which corresponds to the 0.2 mr/hr surface dose rate seen. This dose rate is below the plant tolerance level of 0.75 mr/hr and will permit rerouting of this line above ground. The maximum dose rate measured on the surface of old Line 21 during previous operation was 0.5 mr/hr; however, its exhaust is not presently controlled by a radiation monitor, and higher activities would be possible if higher burnup fuel were handled in the fuel and repair building. A radiation monitor will therefore be installed on the waste gas receiver in the fuel and repair building to similarly protect Line 21 and allow its rerouting above ground.

The third waste gas line being replaced is Line 26, the ventilation line from the sodium building cold-trap room to the compressor room of the waste gas building. It presently runs underground beneath the waste gas building and is inaccessible. Although it may not have failed, it was taken out of service last year as a precautionary measure due to the failure of a number of similar lines along this route intended for purging radioactive atmospheres from the gas and sodium building rooms. The other purge lines have been taken out of service. Line 26 is being replaced, since it provides pressure relief for the nitrogen atmosphere of the cold-trap room in the event of a large sodium spill. The replacement line for Line 26 is being routed above ground, replacing a 30-foot section of the old line. Its route is from the NaK room in the sodium building, around the walls of the waste gas building, to the compressor room. It will be tied into the existing line in both locations.

The new lines are all constructed of 2-inch-diameter Schedule 40 seamless carbon steel pipe. A total of 72 welds must be made in the new lines. All welds are being liquid-penetrant-examined, radiographed, and with the exception of closure welds, mass-spectrometer-tested. The completed lines will also be pneumatically tested at 100 psig. The specifications for the welds require no defects.

Difficulty has been encountered in making the welds. Of the first 36 welds, 15 had to be rejected after radiography due to the rigid requirements. The high rejection rate was mainly due to roughness on the inside of the weld; however, the welding did improve, and by the end of the month, approximately 75% of the new gas lines had been erected.

In February, work was also completed on rerouting the liquid waste line between the health physics building chemical laboratory and liquid waste

process system in the fuel and repair building. It had been found to be corroded in May of last year* and replacement work was begun in October. The new pipe line is constructed of 3-inch-diameter polyvinyl and was re-routed so that only 40 feet of the original 150 foot long run is buried. The remainder of the line is routed through the fuel and repair building where it can be periodically inspected. The underground portion of the run is contained in a 6-inch-diameter carbon-steel pipe. It is polyethylene coated and grounded to a magnesium rod to prevent pipe deterioration from electrolysis. The completed 3-inch line was given a 5-psig hydrostatic pressure test, and the buried portion of the 6-inch containment pipe was pneumatically tested at 5 psig to ensure leak-tightness. The void between the inner and outer pipes is also fitted with a drain line to permit periodic inspection of containment pipe integrity.

B. REPLACEMENT OF THE ARGON GAS SEAL FOR NO. 3 PRIMARY SODIUM PUMP

During February, the No. 3 primary sodium pump, which is shut down, was inspected for routine maintenance. Both sets of pump shaft argon face seals were found to be grooved, requiring replacement. Several rubber O-ring seals were also replaced together with the seal oil pump seals.

Pump No. 3 had been in service a total of 39 months since its last shutdown for major repairs; however, for 14 of these months it was shut down because no sodium was in the reactor. It is generally recognized that replacement of the pump seals is necessary about every two years.

C. REMOVAL OF MOISTURE INDICATOR IN PRIMARY COLD-TRAP ROOM

The moisture indicator in the primary sodium-cold trap room in the sodium service building has been removed from service. This instrument has had a history of limited reliability owing to its complexity and the highly sensitive nature of the detecting cells. A Honeywell dew point indicator, similar to those now being used in the lower reactor building and primary sodium storage vault, will be installed for moisture indication and alarm. A viewing window will also be added to the blower box to permit visual inspection of the refrigeration coils for frost buildup.

*APDA-CFE-22, page 33

D. SAFETY ROD HANG-UP IN STEAM-CLEANING CHAMBER GRIPPER

During the third week in February, a safety rod finned pot for the new fuel transfer facility was installed in the transfer tank in the fuel and repair building (see Figure 15). This was done in anticipation of transferring a stored safety rod from its location in an old finned pot to the new pot for return to the reactor. The stored safety rod had been removed from the reactor for inspection in September of 1966, following several malfunctions of the rod drive latch mechanism during reactor operation (see APDA-CFE-2, pages 12 and 21).

Following withdrawal of the safety rod from the old pot up into the steam cleaning chamber, the transfer tank storage rack was rotated to position the new pot under the loading port. However, the cleaning machine gripper would not delatch from the safety rod when it was lowered into the new pot. Several attempts were made to deposit the rod in the new pot, and a final effort was made to return it to its original pot. None of these were successful. The rod was finally withdrawn into the steam-cleaning chamber, where it was secured in a suspended position in the gripper in the 350 F argon atmosphere of the chamber.

The two locations available for inspecting the malfunctioning gripper were either in the water decay pool or dry loading tunnel. Since it is possible to gain direct access to the gripper in the dry loading tunnel, this location was chosen. In preparation for the inspection, a spare safety rod transfer pot was placed on a temporary stand in the dry loading tunnel. A lead brick shield wall, similar to that used earlier for gripper inspection in August, 1968 (see APDA-CFE-24, Figures 19 and 20), was then erected in front of the pot in the tunnel. The safety rod was steam-cleaned and subsequently deposited in the pot. Radiation measurements showed the dose rate to be greater than 2 r/hr behind the shield wall near the rod, and 10 mr/hr in the work area in front of the wall.

Mirrors were used for viewing as attempts were made to delatch the gripper from the rod (Figure 16). The gripper did not delatch on the first attempt but did delatch on the second (Figure 17). It was subsequently relatched; however, 15 further attempts to delatch the gripper by normal operation were unsuccessful. The rod was returned to the argon-filled steam-cleaning chamber, where it is presently secured at ambient temperature awaiting further corrective measures.

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FIG. 15 FUEL AND REPAIR BUILDING (SECTIONAL ELEVATION)

**FIG. 16 USING MIRROR TO VIEW DEFECTIVE GRIPPER OVER
TEMPORARY SHIELD WALL IN DRY LOADING TUNNEL**

**FIG. 17 GRIPPER AFTER DELATCHMENT
FROM SAFETY ROD**

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VI. ERECTION OF AUXILIARY FUEL STORAGE FACILITY INSIDE THE REACTOR BUILDING

A. BACKGROUND

It is planned that the fuel now in the reactor will be removed and replaced by new fuel prior to operation of the reactor. The used fuel, which has experienced negligible burnup, will be stored on site for subsequent inspection and requalification. The present facilities at the plant for storing intact used fuel in sodium and an inert atmosphere, which is required if the fuel is to be reused, comprises 57 positions: 22 in the fuel and repair building transfer tank, 24 in-reactor storage positions, and 11 positions in the transfer rotor. The 11 positions in the transfer rotor are reserved for fuel handling operations, leaving only 46 positions available for subassembly storage. There are 91 core subassemblies yet to be removed from the reactor and placed in storage, leaving a deficit of 45 storage positions. A new storage facility is therefore being built to accommodate up to 60 subassemblies in heated sodium and provide the extra storage capacity needed.

B. GENERAL DESCRIPTION OF THE FACILITY

The Auxiliary Fuel Storage Facility (AFSF), for use in storing irradiated Core A subassemblies of low decay heat, will be located below floor in the reactor building in equipment decay tank No. 2. This tank, the largest of the four decay tanks in the reactor building, was originally intended for use in storing large radioactive components removed from the primary system. It is located on the east side of the containment building and is approximately 10 feet in diameter and 36 feet high. It extends through the 5-foot-thick concrete biological shield floor to the floor of the lower reactor building, and has the advantage that it is located inside containment and provides sufficient space and shielding to accommodate the required number of subassemblies; however, it requires a reasonable amount of modification.

The required modifications to the tank (see Figure 18) include installing (1) a false floor approximately 14 feet below operating floor level, (2) a storage rotor with two concentric circular rows of holes that will accept 60 subassemblies in sodium-filled capped steel pots, (3) a rotor shaft through the center of the decay tank shield plug and two fuel access ports over the rotor holes at the periphery of the plug, (4) an inert argon gas atmosphere, (5) insulation, and (6) a heating system suspended on and below the storage rotor. In addition, the associated controls for maintaining temperature and pressure are provided. A hand-operated drive mounted on the reactor building floor turns the rotor shaft. The capped pots will

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FIG.18 ELEVATION OF AUXILIARY FUEL STORAGE FACILITY

be stored at approximately 350 F and the argon atmosphere maintained at a slight positive pressure (~ 2 inches of water above barometric) to prevent air inleakage.

Subassemblies will be transferred from the reactor in unsealed storage pots using the multipurpose shipping cask (see APDA-CFE-5, pp 27 and 28 and APDA-CFE-24, p 22) and building crane. The cask will be set on top of a shielded floor valve placed over one of the loading ports in the AFSF. The storage pot and its contained subassembly will be lowered into the support rack using a pot gripper assembly inserted through the top of the shipping cask.

The gripper with its shaft extension is supported by a chain fall located on the overhead building crane. The pot gripper assembly will be used to remove the sealing caps from empty pots in the AFSF and place them on pots containing subassemblies. Empty storage pots will be removed from the AFSF and transported to the reactor transfer rotor container in a reverse manner. A special adapter sleeve is being added to the transfer rotor container to accommodate the storage pots, since they have a smaller overall diameter than the normal finned pots used for transporting subassemblies having high decay heat.

C. SHIELD PLUG CORING

During February, holes were made through the decay tank shield plug to accommodate the rotor shaft and two loading ports.

The 5-foot-thick shield plug consists of two concentric plugs: a 7-foot-diameter inner plug having a 1-foot-thick cover and a 12-foot-diameter outer plug which supports the inner plug. This arrangement was adopted for compatibility with the original remote maintenance function of the decay tank. Each plug consists of a cylindrical steel shell filled with concrete and reinforcing gussets and wire mesh are located within the concrete. An additional 7-inch-thick layer of steel plate is located along the bottom of both the inner and outer plugs.

The 6-inch-diameter rotor shaft hole and one of the two 12-inch-diameter stepped loading port holes were bored through the inner shield plug. The second loading port hole penetrated the outer shield plug. An oxygen lance was used to burn the holes through the steel portions of the shield plugs and a diamond drill coring device was used for the holes in the concrete. The oxygen lance is shown in Figures 19 and 20. Photographs of the coring operation are shown in Figures 21 through 28.

The oxygen lance, which has been available commercially for about two years, consists of a 1/2-inch-diameter iron tube filled with metal rods. The top of the tube is heated to incandescence with a torch, after which oxygen is supplied. The tube then burns and is rapidly consumed as the lance cuts. Tests made prior to using the lance showed that 7

**FIG. 19 OXYGEN LANCE USED FOR
BURNING HOLES THROUGH STEEL**

FIG. 20 CLOSE-UP OF OXYGEN LANCE

**FIG. 21 VIEW OF NO-2 EQUIPMENT DECAY
TANK SHIELD PLUGS SHOWING
LOCATION OF LOADING PORT AND
ROTOR SHAFT PENETRATIONS
(UPPER SECTION OF INNER PLUG
REMOVED)**

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**FIG. 22 CLOSE-UP OF LOADING
PORTS BEFORE CONCRETE
CORING**

FIG. 23 DIAMOND-TIPPED CORING TOOL

**FIG. 24 CORING MACHINE DURING
OPERATION**

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FIG. 25 LIFTING OUT PARTIAL CORE

FIG. 26 CORE LIFTED OUT

**FIG. 27 STACKED CORES FROM
ONE ACCESS PORT
PENETRATION**

**FIG. 28 COMPLETED STEPPED
LOADING PORT—BOTTOM
7 INCHES OF STEEL STILL
INTACT**

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inches of steel could be penetrated in about 45 seconds and a 2-inch-diameter hole was burned 1 foot deep in a concrete block in about 2 minutes. Surprisingly, the outer surface of the concrete block was cold to the touch after the hole was completed. The tests indicate that the oxygen lance may be a fast and economical method of penetrating thick shield walls and floors. It has worked very satisfactorily in cutting the holes through the steel of the shield plug.

The concrete coring using diamond drills had to be done dry, since water is not allowed inside the reactor containment building. Drilling dry with diamonds using compressed air instead of water for cooling presents problems. Adequate cooling is hard to achieve. Only one of the contractors contacted could do the work dry; all others required a water jet of five to six gallons per minute for cooling. Concrete cores were removed from the holes periodically at approximately every two feet of depth, by driving a wedge alongside them and breaking them off with a sledge hammer. A metal strap wrapped around the cores was then used in lifting them out of the holes.

Some difficulties were encountered with loose grout in the holes, which tended to rip the diamonds out of the tool and these areas had to be chipped out by hand. A problem also arose when drilling the hole for the outer 12-inch-diameter access port. An internal web of steel was encountered which interfered with the drill. The hole location had to be altered to avoid the web, since any steel which is encountered causes difficulty by overheating and shattering the diamonds.

By the end of the month, all holes had been burned and cored through the plug down to the elevation of the 7-inch-thick layer of steel plates at the bottom of the plug. The oxygen lance will be used next month to complete the holes through this portion of the plug.

VIII. ERECTION OF NEW FUEL TRANSFER FACILITY IN REACTOR BUILDING

A. BACKGROUND

The original fuel transfer facility used until the time of the meltdown accident consisted of a shielded fuel cask car which traveled on rails between the reactor building and the fuel and repair building. It carried new and spent fuel in and out of the reactor and storage. This single facility could accommodate eleven fuel subassemblies at a time. All required services were self-contained, i. e., the necessary heaters, inert gas purge system, sealing flanges, heat exchanger, hoist, gripper, exit port shield plug storage, and controls were all integrated into a single unit. For this reason, the basic design concept was attractive; however, the facility never worked satisfactorily and required continual maintenance.* The major problems encountered in its use were binding of the cask rotor assembly inside the lead shield compartment, erratic gripper operation, heater losses, hoist failure, gas leaks, seal failures, and sodium drippage.

Because of these deficiencies, a new fuel transport facility (FTF) was obtained.** The earlier problems are solved in the new facility by dividing the functions between independent and separate components. To achieve this, the new facility is designed to handle only one fuel subassembly at a time. The new facility has duplicate seal and gripper components in the fuel and repair building and reactor building, and the subassembly transfer cask shuttles between the two buildings. The erection of the fuel and repair building portion of the facility has been completed and is described in an earlier report.*** The reactor building construction, just underway, is essentially identical. It consists of six major components as follows (see Figure 29):

- A shielded floor valve, which sits on the operating floor over the fuel exit port pipe, and contains a motor-operated gate valve and has double O-ring seals at the bottom and top which seal the valve to the exit port and to the transfer cask positioned above it.

* Johnson, K. P., and Matte, J. III, "Operating Experience with Fermi Fuel Handling Cask Car", paper presented at 16th conference on Remote System Technology March 11-13, 1969, Idaho Falls, Idaho.

** Kirby, R., and Hallinan, G., "The Technical Evaluation of the Fermi Fuel Transfer Facility", *ibid.*

*** APDA-CFE-20, page 34.

Support Columns Bridge and Trolley Gripper Cask Transport Cask

FIG. 29 NEW FUEL TRANSFER FACILITY IN FUEL AND REPAIR BUILDING

FIG. 30 REACTOR BUILDING EXIT PORT WITH SHIELDING IN FLOOR REMOVED

FIG. 31 OLD EXIT PORT SHIELD SEAL FLANGE

FIG. 32 NEW STEEL FUEL FLANGE

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- A battery operated transport cask which travels on rails between the reactor building and fuel and repair building, sits on top of the floor valve during refueling, and can store a single sub-assembly.
- A gripper cask, containing a gripper assembly, which is positioned above the transport cask during refueling and is lowered through the transport cask and floor valve into the exit port, to grasp fuel stored in the transfer rotor container located below floor.
- A plug cask used to remove and store the exit port shield plug during refueling which has its own gripper and hoist mechanism and is positioned on top the floor valve prior to refueling.
- A bridge and motor-driven trolley structure supported 20 feet above the operating floor on support columns set in the floor on either side of the tracks, from which is suspended the plug and gripper casks allowing positioning of either over the exit port.
- A control console used to control all fuel transport equipment permanently installed in the building.*

The major difference between the fuel and repair building and reactor building installations is that in the fuel and repair building only the gripper cask is hung from the trolley; the plug cask is handled by the building crane and stored on a special pad when not in use.

B. EXIT PORT MODIFICATIONS

The exit port designs in the fuel and repair building and reactor building are slightly different. Because of this, exit port modifications to accommodate the new fuel transfer facility are necessary in the reactor building which were not required in the fuel and repair building.

In the fuel and repair building, the exit pipe through the concrete floor to the transfer tank is fixed permanently in place. Thermal expansions due to transfer tank sodium temperature changes are accommodated by movement of the transfer tank around the exit pipe; consequently, the floor valve is permanently installed over the exit pipe.

In the reactor building the exit pipe is attached to the reactor vessel transfer rotor container and moves with the vessel as a result of temperature changes. Significant vertical and horizontal pipe movements occur in the

* The movable transport cask has its own self-contained control panel.

approach to power and during reactor operation. Even during refueling, with the reactor shut down, exit pipe displacements from nominal alignment of the order of $\pm 1/2$ -inch horizontally and vertically must be accommodated. The cask car sealing flange of the old fuel transfer facility accommodated these displacements through a bellows arrangement. To accommodate the displacement using the floor valve of the new facility, clearance must be gained by making modifications in the combination shield and sealing flange set in the floor around the exit port. The floor valve must be removed from the exit port before operating the reactor.

Figure 30 is a photograph of the exit port in the reactor building with the original lead shield and sealing flange removed from around the exit pipe. Figure 31 shows the original flange and Figure 32 shows the replacement flange made of steel. Double O-ring seals on top of the flange seal the exit port to the floor valve. The replacement flange is made of steel rather than lead because of the additional shielding afforded by the lead floor valve in the new design. There is also a half annulus-shaped lead shielding ring (Figure 2) located in the floor around the sealing flange which requires no modification.

Changes in the purge gas piping (Figure 30) to the exit port are also being made to allow installation of the new flange. Strain gauges are being installed at the interface between the flange and floor valve to protect the exit pipe from damage in the event pipe movement occurs during refueling. The exit port modifications are now complete and the port has been re-assembled (Figure 32).

C. ERECTION OF SUPPORT STRUCTURES

During February, holes were drilled in the concrete operating floor to allow placement of the anchor bolts for the FTF bridge and trolley support columns (Figures 34 through 36). Twenty-eight 1-1/4-inch-diameter holes are required. The drilling has proceeded very slowly even though carbide-hardened bits are being used. Attempts were made to use the oxygen lance to blast the holes, but hole diameters as small as 1-1/4 inch could not be obtained. Difficulties were also encountered with a 6-inch-thick layer of weak grout, which made it necessary to increase the depth of all holes from 8-inches to 12-inches. Extra long anchor bolts had to be obtained. Lengthening of the holes has resulted in further difficulties due to interferences with structural steel in the floor; however, the drilling is nearly completed and the structural steel for the fuel transfer facility has been moved into the building in anticipation of its erection next month.

FIG. 33 EXIT PORT REASSEMBLED AFTER MODIFICATIONS

FIG. 34 DRILLING HOLES FOR ANCHOR BOLTS OF FTF SUPPORT COLUMNS

FIG. 35 CLOSE-UP OF DRILLING APPARATUS

FIG. 36 ANCHOR BOLTS IN PLACE

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