ENRICO FERMI ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT 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 United States Atomic Energy Commission under AEC 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 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-17.

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

I. CURRENT EXPERIENCE SUMMARY

Parallel efforts are continuing on the chisel and arc-melt methods for removal of the zirconium screw heads which fasten the zirconium segments to the conical flow guide in the reactor core inlet plenum. A full-length chisel tool, without the articulation feature, successfully sheared off screw heads in feasibility tests. Feasibility tests of a full-length chisel tool with articulation will take place when fabrication of this tool is completed.

Experimental work has shown that the most favorable conditions for the arc-melt process are initial electrical conditions of 1500 amps, 100 volts open-circuit d-c, with the screw head used as the positive electrode. These conditions produce almost no splatter of debris during melting of the screw head. An additional arcing with lower current puddles the top of the remaining screw shank, firmly fastening it to the surface of the conical flow guide, as demonstrated by torque tests which resulted in shearing of the shank before breaking loose from the flow guide.

Effort has been initiated to develop a concept of a vacuum-cleaning device to remove any debris which might result from this operation. Fabrication has started on four types of tools to dislodge the zirconium segments from the conical flow guide. Each of these tools would function in a different manner to afford flexibility in dislodging the segments. The design of the light source used in previous operations in the inlet plenum is being modified with the goal of extending the operating life of the lamp.

The No. 3 secondary loop and steam generator were filled with sodium. The steam generator was successfully pressure-tested for leaks. All three secondary sodium loops are now filled and sodium circulation is being maintained in each loop at a low flow rate.

The introduction of an optical collimating technique for dimensionally gauging subassemblies in the exit port inspection facility to determine subassembly straightness has produced encouraging results. Data repeatability to within 0.0025 inch has been achieved. The acceptance tests of the fuel and repair building (FARB) section of the new fuel transfer facility (FTF) was completed. The facility accomplished the handling of a sodium-filled pot out of the FARB transfer tank, movement of the transport cask a short distance toward the reactor building, and return of the pot to the transfer tank. Twenty successive cycles of this operation were carried out. One significant difficulty, a light-to-moderate dripping of sodium on the transfer tank exit port seal, was experienced. It is believed that this difficulty can be remedied by a design change.

II. INVESTIGATION OF ZIRCONIUM SEGMENTS

A. BACKGROUND

The deformed sheet of zirconium (UFO) removed from the reactor in March was one of six triangular-shaped segments originally installed on the sloping sectors of the conical flow guide in the inlet plenum to the reactor core. Details of the removal operation are given in APDA-CFE-20. It was decided to remove the remaining zirconium segments while yet giving consideration to fixing the segments in place as a possible alternative. The plan for removal is to free a segment from the three fastening screws, dislodge it from the surface of the conical flow guide, and withdraw the segment through the 14-inch sodium inlet pipe of the No. 1 loop in the same manner that the deformed zirconium object was removed.

Design and development of tools to remove the heads of fastening screws and to dislodge the freed segments was started. The preferred methods of screw head removal are (1) by chiseling and (2) by melting away the screw head by means of an arc melting process. Tests demonstrated the feasibility of both methods. It was decided to develop both concepts in parallel efforts to avoid long delays if difficulties with either method should arise later in the program. A study of means to dislodge a segment from the conical flow guide when the screw heads are removed resulted in the concept of a tool to hook and pry loose the bottom corners of a segment and pull it free of the conical flow guide surface. Another dislodging tool concept is that of a shover tool with a serrated face to push the segment tangentially off the surface of the conical flow guide.

Other conceptual efforts were directed toward refining the control of the spine-type manipulator and strengthening and improving the structural features of the object-retrieval device, based on experience with these devices during removal of the zirconium object. It is expected that one of the dislodging tool concepts can serve to deform the segments to sufficiently reduce the width of each if required to ensure the unobstructed withdrawal of the segments through the 14-inch line by means of the object-retrieval device.

Modification and enlargement of the reactor vessel mock-up was started to provide detailed simulation of the geometry and operating conditions for testing the tools and devices under development for removal of the zirconium segments. Previous information on the investigation of the zirconium segments is given in APDA-CFE-20 and 21.

B. APPROACHES FOR SCREW HEAD REMOVAL

1. Chisel Tool

An articulated chisel tool designed for insertion through an available 6-inch-diameter penetration in the rotating shield plug and holes in the support plate to deliver a chisel impact at a screw head was described in Section II. B. 1 of APDA-CFE-21. The bottom two feet of this full length tool can be articulated through an angle of 10 degrees in a single plane to permit addressing the chisel to all screw heads at effective angles of attack. Fabrication of this tool is expected to be completed in June, at which time a feasibility test will be conducted to demonstrate that the full-length articulated chisel tool can shear off a screw head. It is also planned to test the functioning of the articulated joint in a hot sodium vapor environment similar to that which will exist in the reactor inlet plenum during segment removal. A final functional test of the articulated chisel tool in the reactor vessel mock-up will be required to give assurance that the tool can shear off screw heads in the reactor.

The feasibility of the same full length chisel tool, but without the articulated joint, was demonstrated this month. In several trial tests it was found that two to three blows were required to shear off a screw head. A 65-psig pneumatic pressure produced a hammer velocity of 20 feet per second at the moment of impact. It is expected that very little energy will be absorbed by the articulated joint.

Figures 1, 2, and 3 show an accessory that was fixed to the chisel head to trap and retain the sheared-off screw head; this concept requires refinement. It appears that the chisel recoils enough from impact to allow the sheared screw head to bounce free of the trap.

2. Development of Arc-Melt Concept

The feasibility of the arc-melt concept for the removal of screw heads was demonstrated in earlier tests. Previous information about this concept is given in APDA-CFE-21, Section II. B. 3.

Experimental work was conducted in May to determine the optimum electrical conditions for melting away a screw head while keeping the resulting debris at a minimum. The best results to date have been obtained with initial electrical conditions of 1500 amps, 100 volts open-circuit d-c, with the screw head used as the positive electrode and the arc maintained for one-half second. In all experiments, the tungsten electrode was directed coaxially at the screw head. It was also found that an inert atmosphere is a favorable influence in limiting free debris. In an inert atmosphere much of the debris welds firmly to the surrounding surface of the zirconium plate. An improvement in minimizing and containing debris has been achieved;

**FIG. 1 CHISEL TOOL AND DISASSEMBLED
SCREW HEAD CATCHER**

**FIG. 2 SCREW HEAD CATCHER ASSEMBLED
ON CHISEL TOOL**

**FIG. 3 MOCK-UP OF CHISEL TOOL IN
POSITION ON A SCREW HEAD**

however, additional experimental work is planned in an effort to further reduce or eliminate free debris.

Although the above electrical process appears to securely fasten the screw shank into the conical flow guide, techniques to more securely fasten the shank to the guide have been investigated to protect against the shank's working loose. An additional arc at a lower current of about 700 amps, when applied for one-half second, melts and puddles the top of the screw shank, firmly fixing it to the stainless steel conical flow guide.

The direction in which the electrode approaches the screw head is a consideration that is to be investigated further. In tests to date the electrode has been directed coaxially at the screw head. It is expected that an angular or slant approach of the electrode to the screw head can be used for convenience in applying a tool to the screw head from the top of the rotating shield plug.

3. Removal of Debris by Vacuum-Cleaning Device

Effort has been initiated to develop a vacuum-cleaning device which could be used following screw head removal to remove debris from the inlet plenum if, during tests in the mock-up, it is shown that debris is produced. The vacuum-cleaning device would be totally inserted into the inlet plenum to avoid disturbing the positive pressure conditions in the reactor (4 inches water column). This vacuum-cleaning device would be inserted by means of the flexible duct of the object-retrieval device used in the removal of the zirconium object in March and would be inserted into the inlet plenum to the reactor core through the 14-inch sodium inlet line.

The vacuum unit is a three-stage blower actuated by a 0.29-horsepower, 400-cycle, 3-phase motor and is designed to draw 47 cfm at a 14.5 psia back pressure. The blower and motor would be enclosed in a trolley-mounted housing and be preceded by a spun-glass filter. The vacuum-cleaning device would have the capability of withdrawing debris of the size of a screw head into a collection chamber in the housing ahead of the filter.

The engineering design of the vacuum-cleaning device is presently underway. The blower-motor unit has been procured and is expected to be functionally tested in a sodium vapor environment shortly. Based on studies of the debris resulting from the tests in the mock-up, it will be determined if such a device is needed.

C. APPROACHES FOR DETACHING ZIRCONIUM SEGMENTS

In May, the designs of four dislodging tools were completed and fabrication started. These tools which include two tools described in a previous report,* are intended to deform and detach the remaining zirconium segments from the conical flow guide. After the deformed segments are detached, it is expected that they will be handled by the spine-type manipulator and the object-retrieval device and removed from the reactor through the 14-inch sodium inlet line.

Figure 4 shows the plow tool designed for use while the segment is still fastened down by the screws. This tool hooks into the opening at a bottom corner of the segment and when pulled up deforms the segment to produce an opening between the bent edge of the segment and the radial baffle plate (see Figure 8). The tire iron tool, shown in Figure 5, can then be inserted into the opening between the segment and the baffle plate to bend the segment inward and further reduce its width.

The edge-curler tool (see Figure 6), a refinement of the peeler tool described in Section II. D. of APDA-CFE-21, also may be applied to a still-fastened segment. This tool hooks into each of the bottom corners of the segment and curls the segment inward, reducing the width of the segment at its widest part. After the screw heads are removed, the shover tool (see Figure 7), or any of the above tools which can hook the segment, may be used to dislodge the segment from the surface of the conical flow guide. It is intended that all of the above dislodging tools will be tested in the reactor mock-up to establish their feasibility and application.

D. MODIFIED VIEWING LAMP

The Quartzline lamp used previously to illuminate the inlet plenum and other internal areas of the reactor vessel offers the disadvantage of a short life due to burnout and occasional failure of the quartz envelope. In the environment of the drained but still hot (350 F) reactor vessel, sodium vapor and droplets react chemically with the quartz envelope to change the physical properties of the quartz. This chemical reaction is temperature dependent. The result is a continued reduction in cooling of the lamp envelope and reduced light transmission from the quartz envelope to the point of lamp burnout or failure of the envelope.

A recently completed modified lamp design is expected to provide a much longer lamp life, thereby reducing interruptions to viewing inside the reactor. In the modified design, a quartz tube encloses the quartz envelope, forming a 3/8-inch annulus through which a 2 cfm clean inert argon purge

* APDA-CFE-21, Section II. D.

FIG. 4 PLOW TOOL FOR DISLODGING ZIRCONIUM SEGMENTS

FIG. 5 TIRE IRON TOOL FOR DISLODGING ZIRCONIUM SEGMENTS

FIG. 6 EDGE-CURLER TOOL FOR DISLODGING ZIRCONIUM SEGMENTS

FIG. 7 SHOVER TOOL FOR DISLODGING ZIRCONIUM SEGMENTS

**FIG. 8 ZIRCONIUM SEGMENTS FIXED ON CONICAL FLOW
GUIDE IN MOCK-UP OF INLET PLENUM**

is maintained for cooling. The quartz tube protects the quartz envelope from chemical attack, and the argon purge cools the lamp and also the quartz tube, retarding the reaction of sodium vapor with the quartz tube. It is planned that such a modified lamp assembly, which is expected to be received shortly, will be life-tested in a sodium vapor environment.

E. MODIFICATION OF REACTOR MOCK-UP

The modification of the inlet pipe and reactor vessel mock-up continued during May. Improvements being incorporated into the mock-up are noted in APDA-CFE-21, Section II. F. Additional features pertaining to segment removal operations include a platform at the penetrated 14-inch elbow and a new metal upper and lower core support plate structure which has all holes of precise dimensions, accurately located. The platform at the 14-inch elbow will provide a work area of a size to simulate that available at the primary pipe elbow for manipulation of the object-retrieval device.

Another feature of the mock-up will provide a facility for testing a new approach to provide illumination of the holddown fingers. Inadequate lighting was one of the factors contributing to the difficulties experienced in previous attempts to inspect holddown fingers by the spherical mirror method.* In the new approach, it is intended to include four 1500-watt lamps in the lower part of the mirror support post. The question to be resolved by mock-up testing is whether such a lamp arrangement will provide sufficient lighting of the holddown fingers. It is expected that the lamps will be tested in a structure simulating the exterior walls of subassemblies forming the empty lattice cavity on the west side of the reactor (see the reactor loading diagram in Figure 9). The test will be described in a future report.

* See APDA-CFE-19, p. 15.

III. OPERATIONS

A. REACTOR SUBASSEMBLY MOVEMENT

As shown on the reactor loading diagram (Figure 9), one outer radial blanket subassembly (ORB) was moved to a different position in the reactor lattice in May. ORB subassembly M957 was moved from position N07-N04 to N07-P00 to ensure that the potentially damaged orifice retainer in the lower support plate hole in N07-P00 will not be lifted out of the support plate by sodium flow during the forthcoming primary system cleanup operations. Some of the orifice retainer spring fingers were observed to be bent when viewed and photographed in April.* It is presumed that the damage was caused by insertion of the spherical mirror support post for the attempts to inspect the holddown fingers.**

The offset handling mechanism (OHM) and the rotating shield plug (RP) were operated in the automatic mode for the subassembly move. The insertion of the subassembly into the new lattice position required a slight manual adjustment of the OHM in rotation; otherwise, no difficulties were encountered. The subassembly was properly inserted into position, but inasmuch as there are no camming surfaces adjacent to position N07-P00, there is no assurance as to orientation of the subassembly. The reactor vessel continued in the drained condition during this subassembly move.

B. FILLING THE NO. 3 SECONDARY SODIUM SYSTEM

The No. 3 secondary sodium system, which includes the No. 3 steam generator shell, was filled with sodium from the No. 3 secondary storage tank. Following a successful 600-psig nitrogen pressure test on the tube or water side that verified the integrity between the tube and shell sides of the No. 3 steam generator (see Section VI. A), the No. 3 secondary sodium loop was ready for operation. Orifices were installed at the water header end of all tubes of No. 1, 2, and 3 steam generators to prevent feedwater flow instabilities at part load,*** and sodium has been circulating at a low rate of flow in Nos. 1 and 2 secondary loops since December 1967 and January 1968, respectively. It is expected that all three secondary loops will contribute heat to the primary system during the forthcoming primary system refill and clean-up operations.

* APDA-CFE-21, Section III. B

** APDA-CFE-18, 19 and 20.

*** See APDA-CFE-16, Figures 22 and 23.

FIG. 9 REACTOR LOADING ON 5-31-68

The sodium fill operation for No. 3 secondary loop was accomplished in accordance with established operating procedures with details being essentially the same as those for the filling of the No. 1 and 2 secondary sodium loops.* The initial sodium plugging temperature of the No. 3 secondary loop was determined to be about 275 F. Cold-trapping operations were not started immediately because of the temporary unavailability of the No. 3 secondary sodium pump. The main motor drives for this pump and for No. 3 primary sodium pump are energized by closing the same circuit breaker, and the primary sodium pump cannot be operated until the reactor vessel and the No. 1 primary loop are refilled with sodium. In the meantime, the No. 3 secondary loop sodium was circulated at a flow rate of 600,000 pounds per hour by the No. 3 secondary sodium pump driven by the pony motor. It is expected that clean-up of the No. 3 secondary loop sodium will be accomplished by cold-trapping after the primary system is filled and sodium circulation established.

C. GAS SYSTEMS PERFORMANCE

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

1. Primary System Gas Activity

<u>Location</u>	<u>Sample Date</u>	<u>Gross Beta Concentration, microcuries/cm³</u>
Reactor Cover Gas	5-2-68	2.2×10^{-6}
Reactor Cover Gas	5-6-68	5.38×10^{-6}
Reactor Cover Gas	5-10-68	2.86×10^{-6}
Reactor Cover Gas	5-17-68	5.87×10^{-6}
Reactor Cover Gas	5-24-68	3.15×10^{-6}
Reactor Cover Gas	5-29-68	2.48×10^{-6}
Primary Shield Tank	5-6-68	2.5×10^{-8}
Primary Shield Tank	5-10-68	1.67×10^{-8}
Primary Shield Tank	5-17-68	2.66×10^{-8}
Primary Shield Tank	5-24-68	2.5×10^{-8}
Primary Shield Tank	5-29-68	2.49×10^{-8}

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

** APDA-CFE-21

2. 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	45**
Carbon Monoxide	Below 10	Below 10
Carbon Dioxide	10	10
Hydrogen	Below 4***	Below 2.5
Helium	Below 4	Below 4
Methane	Below 10	Below 10
Argon	Remainder	7000
Nitrogen	2720	Remainder
Dew Point	Not Measured	-55 F

* Sample date was 5-29-68.

** Technical specifications state 1000 ppm maximum.

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

IV. MAJOR EQUIPMENT

A. EXIT PORT INSPECTION FACILITY

Experimental and developmental work conducted in May demonstrated that use of an optical collimating technique would improve the accuracy of the dimensional gauging of subassemblies positioned in the chamber of the exit port inspection facility. The collimating technique ensures that subassembly measurements are made with the transit line of sight normal to the viewing window, thus eliminating errors due to refraction in the viewing windows. As stated in Section VI. A. of APDA-CFE-21, this collimating technique was recommended by the suppliers of the optical equipment, as an approach to refining the transit measuring method for dimensionally gauging subassemblies. Previous attempts at gauging dummy subassemblies to determine the capability of the exit port inspection facility to examine reactor subassemblies for bowing resulted in unacceptable scatter in the data. Earlier information about the exit port inspection facility has been given in previous reports.*

1. Sighting Normal to Viewing Window

The transit line of sight is established normal to the viewing window by sighting on a precision optical mirror having a surface flatness within 4×10^{-6} inch fixed exactly parallel to the viewing window. The coincidence of horizontal and vertical crosshairs located in the transit telescope, with mirror images of these crosshairs, fix the transit line of sight normal to the plane of the mirror and hence the viewing window to within an angle of 5 seconds.

In the above manner, the transit line of sight is made normal to the middle window. However, the lines of sight must, of necessity, be tilted downward and upward for viewing through the top and bottom windows respectively, because the subassembly wrapper can, when in the inspection position in the inspection chamber, does not extend fully up to the upper window nor fully down to the lower window. The transit lines of sight are, therefore, made normal to the top and bottom windows in the horizontal plane only, still an improvement in the technique of dimensionally gauging subassemblies. Figure 10 shows the transit and mirror at the upper window.

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

FIG. 10 TRANSIT ON PRECISION TRACKS AND COLLIMATING MIRROR

2. Precision Transit Tracks

The transit, after being accurately positioned in the above manner, traverses the width of the viewing window (with the precision mirror removed), sighting in turn on the taut wire benchmark and on measuring points on a subassembly positioned in the chamber of the inspection facility. The transit traverses the window on newly-installed precision lathe tracks which hold the line of sight normal to the plane of the window throughout the traverse. The new lathe tracks replace the originally installed transit rails which had been found to be bowed 0.008 inch in the horizontal plane. The bowed transit rails were a contributing cause of the scatter in the previous data.

The transit and lathe track assembly moves vertically on two upright posts to each of the three viewing windows. The same mirror collimating technique establishes normal lines of sight at each window.

3. New Method of Measurement

In the new method of measurement, an indicator bar which moves laterally with the transit as it traverses the viewing window indicates data on a stationary vernier device. This new measuring method, which ensures that measurements are taken in the same horizontal plane across the viewing window, replaces the practice of dipping the transit to read data on a stationary scale at each measuring point, thus introducing error due to possible vertical shift of measuring points.

The results of preliminary gauging of a dummy subassembly with the new lathe tracks installed and employing the collimating technique and the new vernier measuring method were encouraging. Repeatability of data to within 0.0025 inch was obtained. Further investigation is planned to determine whether a pattern of magnification exists for each viewing window that might be factored into observed dimensional data. The results of previous dimensional gauging tests of the inspection facility indicated a tendency of the facility to magnify dimensions. It is expected that a standard subassembly-shaped bar of precisely known dimensions will be the subject of this investigation, as well as a continuing determination of the dimensional gauging capability of the exit port inspection facility.

B. FUEL TRANSFER FACILITY

The preoperational tests of the FARB section of the new fuel transfer facility were completed in May, as were the acceptance tests. The purpose of the tests and a description of the facility and other information are given

in previous reports.* Except as noted below for the acceptance tests, no significant difficulties were encountered in conducting these tests.

1. Twenty-Cycle Demonstration

The acceptance tests consisted of a twenty-cycle demonstration of the facility handling a sodium-filled pot out of the FARB transfer tank, transporting it in the transport cask a short distance in the direction of the containment building and then returning it to the transfer tank. There was no subassembly in the pot; otherwise, operations were carried out in the same manner as is expected for handling subassemblies out of and depositing them in the transfer tank. Several days were required to accomplish the twenty-cycle demonstration. The transport and gripper casks were heated in advance of the tests performed in a continuous manner during a particular day's run and maintained at operating temperature throughout the day's program. Each cycle lasted 80 minutes.

2. Facility Components

Figure 11 shows the transport cask moving into position to form a seal over the floor valve that seals the transfer tank. The gripper cask is in the background above the transport cask. Double rubber O-rings seal between the casks and between the transport cask and the floor valve.

In Figure 12, the gripper is in the lowered position showing how it reaches through the transport cask. Figure 13 is a view of the top of the transport cask showing the internal cooling fins that mesh with the external fins on the pot. A full length annulus surrounds the fins in the cask to accommodate the flow of cooling (for spent subassemblies) or heating gas recirculated by a blower contained in the cask structure. This gas recirculation system maintained a temperature of 320 F at the top of the cask with higher temperatures toward the bottom or inlet end of the annulus. Due to high heat loss, 9 hours' heat-up time was required for the transport cask. The same high heat loss was probably accountable for the too frequent on-off operation of the transport cask heater controls during the acceptance tests.

Figure 14 shows a drip pan cask under the transport cask. The function of the drip pan cask is to collect sodium drippings from the transport cask at the conclusion of a subassembly handling operation. At the end of the 20-cycle demonstration only 5 ounces of sodium was collected in the drip pan cask. This amount was much less than expected.

* APDA-CFE-10, 19, 20 and 21.

**FIG. 11 TRANSPORT CASK APPROACHING
POSITION OVER FLOOR VALVE
TO TRANSFER TANK IN FARB**

FIG. 12 GRIPPER LOWERED FROM GRIPPER CASK

FIG. 13 COOLING FINS IN TRANSPORT CASK

FIG. 14 DRIP PAN CASK UNDER TRANSPORT CASK

3. Sodium on O-Ring Seal

A significant fault observed during the acceptance tests was the accumulation, though rather small, of sodium or sodium oxide on the upward-facing rubber O-rings that form the floor valve-to-exit port seal. After a pot was lifted into the transport cask through the floor valve, the lower cask valve was closed as quickly as possible, but apparently not quickly enough to prevent sodium dropping on the exposed seal rings. The inert gas purge of the transport cask before the pot lifting operation was thought to possibly blow oxide particles around, some landing on the O-ring seals. A design modification that would provide for closing the valve against a bottom-facing O-ring seal is thought to be worthy of consideration. It is presently planned to test-operate the fuel transfer facility further for the purpose of training plant personnel.

V. INSPECTIONS OF SUBASSEMBLIES

Metallographic replicas of transverse sections of fuel pins taken from subassemblies M099, M091, M122 and M140 were examined at the Engineering Research Department of The Detroit Edison Company. The purpose of the examination was to determine the extent of phase transformation in the fuel alloy and hydriding in the zirconium cladding. Preliminary observations made at Battelle Memorial Institute were reported in APDA-CFE-19. Subassemblies M122 and M140 were adjacent to the melted subassemblies M098 and M127 in a four-subassembly cluster. Earlier results of inspections of the above subassemblies are given in previous reports as follows:

Subassembly M099	APDA-CFE-5, 6, 7, 16, 19
Subassembly M091	APDA-CFE-9, 10, 11, 19
Subassemblies M122 and M140	APDA-CFE-11, 12, 15, 16, 18, 19

A. INSPECTION OF SUBASSEMBLY M099

Transverse Section, Fourth Grid from the Top - There is 30% to 90% fuel alloy transformation to the alpha-plus-delta phase from the original gamma phase; the average transformation is 75%. The hydride content in the cladding is estimated to be 600 ppm in one sample and the average about 150 ppm.

B. INSPECTION OF SUBASSEMBLY M091

Transverse Section, Fourth Grid from the Top - Transformation of fuel alloy is estimated to be between 40% and 80% with the average about 60%. The hydrogen content in the cladding is estimated to be less than 200 ppm.

Transverse Section, Tenth Grid from the Top - Transformation of the fuel alloy to the alpha-plus-delta phase is between 40% and 80%, and the average about 60%. The hydride content in the cladding is estimated to be less than 100 ppm. Although about 50% of the sample replicas were unacceptable for observation, a representative sampling of the subassembly was obtained.

C. INSPECTION OF SUBASSEMBLY M122

Transverse Section, Fourth Grid from the Top - The fuel alloy appears to be entirely in the gamma phase, indicating that this section had experienced a temperature in excess of 1100 F in the reactor. As concluded

from the preliminary observations of this sample at BMI,* 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 cladding is intact on all fuel pins but the hydride content is in the range of about 200 ppm to 1000 ppm. The pins with higher hydride content in the cladding appear also to have thicker diffusion zones between the fuel and the cladding, indicating they had seen excessive temperatures.

Transverse Section, Tenth Grid from the Top - Observations and conclusion with respect to retransformation to the gamma phase are the same as those for the transverse section, fourth grid from the top. The hydride content in the cladding is estimated to be between 150 and 600 ppm with the average about 350 ppm. The cladding is intact on all pins.

D. INSPECTION OF SUBASSEMBLY M140

Transverse Section, Fourth Grid from the Top - Transformation of the fuel alloy to the alpha-plus-delta phase varies between 30% and 70%, with the average about 45%. The pins in the center of the fuel bundle are somewhat more transformed. It is estimated that there is little hydriding of the cladding (less than 150 ppm).

Transverse Section, Tenth Grid from the Top - Transformation of the fuel alloy is the same as at the fourth grid from the top. The maximum hydrogen content in the cladding is less than 200 ppm.

* APDA-CFE-19, Section VI. C.

VI. MAINTENANCE

A. LEAK TEST OF THE NO. 3 STEAM GENERATOR

After the No. 3 steam generator was filled with sodium (see Section III. B), a gas pressure test was performed to verify the integrity between the tube and shell sides of the unit. There was no water in the tubes. The tube side was pressurized with nitrogen at 600 psig for a 24-hour test period. There was no measurable leakage, as evidenced by the lack of detectable increase in nitrogen content in the argon cover gas above the shell side sodium. The chromatograph method was used to determine the nitrogen content in the argon cover gas.

B. LEAKS IN RADIOACTIVE WASTE GAS LINE

Leaks were discovered in a section of pipe in the radioactive waste gas disposal system near the point in the system where waste gas at a low level of activity is diluted with a large quantity of air and released to the atmosphere via the stack. Attention was attracted to the leaks by the eruption of gas and mud from the ground surface above the location of the leaks. The section of leaking pipe was isolated from the waste gas system, and then replaced by a new section of pipe as a temporary measure.

1. Description of Waste Gas Disposal System

A diagram of the waste gas disposal system is shown in Figure 15. Waste gas that may contain radioactive materials is discharged into the waste gas collection header. From this header, the waste gas, if at a sufficiently low level of activity, is diluted with air and discharged to atmosphere through a 200-foot-high stack. All waste gas with an activity of $1 \times 10^{-3} \mu\text{Ci}/\text{cm}^3$ or more is automatically shunted to decay tanks, where it remains until its activity has decayed to that level or less. It is then discharged to atmosphere through the waste gas stack after being diluted with air. System controls regulate discharge from the waste gas system to maintain an activity level of $2 \times 10^{-6} \mu\text{Ci}/\text{cm}^3$ or less at the mouth of the stack. Waste gas discharge to the stack is automatically shut off if the flow of diluting air is less than 20,000 cubic feet per minute. With stack mouth concentrations limited to $2 \times 10^{-6} \mu\text{Ci}/\text{cm}^3$, the concentration at the nearest boundary of the exclusion area is reduced to $2 \times 10^{-8} \mu\text{Ci}/\text{cm}^3$, which is the maximum permissible concentration for a specially chosen mixture of noble fission gases and irradiated argon.

FIG. 15 RADIOACTIVE WASTE GAS DISPOSAL SYSTEM

2. Repair of Leaks

The leaks were discovered in a 2-inch-diameter, carbon-steel section of vertical pipe that enters the air plenum upstream from the air dilution fan at the base of the waste gas stack. Figure 16 shows the general area around the leaking pipe at the northwest corner of the fuel and repair building. Figure 17 is a close-up of the replacement section of pipe protected with Bitumastic and a layer of tar paper. This vertical line, rising from a below-ground, 8-inch-diameter carbon steel containment pipe, was found to be corroded through at several points about four feet below ground level. Corrosion protection was not evident on the vertical line. The containment pipe also contained another waste gas line over its full burial route. The two lines within the containment pipe were checked at one point by opening the containment and appeared to be in good condition. However, the 8-inch containment pipe is badly corroded; therefore, it is felt that more extensive repair or modification is required. A permanent repair method is under study.

3. No Hazardous Release of Radioactivity

It is not known with any accuracy how long this waste gas line failure has existed; but it is extremely improbable that the failure existed for more than a short time prior to its discovery. All activity releases since November 1966 have been at very low levels. At the time the leak was discovered, gas stored in the decay tanks was sampled and found to contain 2.1×10^{-6} $\mu\text{Ci}/\text{cm}^3$, principally krypton-85. This is less than the occupational maximum permissible concentration (MPC) for krypton-85 and only seven times the nonoccupational MPC, i.e., 1×10^{-5} $\mu\text{Ci}/\text{cm}^3$ and 3×10^{-7} $\mu\text{Ci}/\text{cm}^3$, respectively. Such concentration would easily be diluted to less than the nonoccupational MPC before reaching the site boundaries, at a minimum distance of 1000 meters. This concentration is typical of releases in the past eighteen months, since the reactor has been shut down and very little steam-cleaning of subassemblies has been done.

Soil samples were taken in the area immediately surrounding the corroded pipe. No radioactivity above normal background could be detected.

C. LEAKS IN RADIOACTIVE LIQUID WASTE LINE

A Bitumastic-covered, 2-inch wrought iron pipe which carries liquid waste from the health physics (HP) building was found to have external corrosion and some internal corrosion, which resulted in complete penetration of the pipe in some places. The external corrosion resulted from local Bitumastic failure. This pipe is buried near the waste gas pipe in which leaks were found; therefore, it too was examined.

**FIG. 16 LEAKING WASTE GAS LINE
ENTERING AIR PLENUM**

**FIG. 17 REPLACEMENT WASTE GAS LINE
FROM CONTAINMENT PIPE**

1. Description of Liquid Waste Disposal System

Figure 18 shows a diagram of the liquid waste system. The major sources of liquid waste are the subassembly steam-cleaning chamber, miscellaneous low-level decontamination operations in the FARB (discharge to hot sump), and the hot laundry and laboratory sinks in the health physics building. Effluent from these sources is ultimately released to the turbine condenser discharge pipe where it is diluted by recirculating pump discharge water entering the lagoon. The effluent first empties into liquid waste tanks and from there is (1) discharged by steam ejector directly to the condenser discharge pipe or by a metering pump to the condenser discharge pipe, or (2) is held up for treatment by ion exchange units and then released through steam ejectors or the metering pump to the condenser discharge pipe. The method of release, which depends on the radioactivity of the liquid waste and whether one or two condenser recirculating pumps are operating, is selected so that the plant maximum permissible concentration (for nonoccupational exposure) in the lagoon will not exceed $8 \times 10^{-9} \mu\text{Ci}/\text{cm}^3$ when averaged over a period of a year.

2. Temporary Rerouting of Liquid Waste from Health Physics Building

A temporary plastic line was routed above ground to carry liquid waste from the health physics building to the hot sump in the FARB. It is planned to engineer and accomplish a more permanent repair. The corroded waste line was abandoned.

3. No Hazard from Leaks in Liquid Waste Line

Materials discharged into the HP sump contain only very low concentrations of radioactivity. Samples of radioactive materials used for instrument calibration as well as laundry wash water are disposed of through this system. The amount of radioactive material used for calibration and then discarded is extremely small, and the laundry wash water has never been contaminated except with very small amounts of sodium and cesium activities. The health physics building effluent is highly diluted by wash and rinse water and other nonradioactive waste liquids.

Water samples and soil samples taken adjacent to the pipe in an area of known leakage contained concentrations of $1.7 \times 10^{-7} \mu\text{Ci}/\text{cm}^3$ and $1.35 \times 10^{-7} \mu\text{Ci}/\text{mg}$, respectively. Water samples routinely examined at the plant site contain $1.2 \times 10^{-8} \mu\text{Ci}/\text{cm}^3$. Thus, there has been only a very small increase in radioactivity present in the soil immediately adjacent to the pipe.

FIG.18 LIQUID WASTE FLOW DIAGRAM

4. Other Buried Systems

There is only one other directly buried pipe which carries radioactive material at Fermi. This is the discharge line from the FARB liquid waste system to the condenser cooling water discharge line. This liquid waste line is made of stainless steel and is contained within a carbon steel containment pipe. There is no reason to believe that this line has failed. It is intended, however, that both the stainless steel and the secondary containment pipes be examined to determine if they are suitable for continued long-term use.