ENRICO FERMİ ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMİ ATOMIC POWER PLANT JANUARY 1968

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

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Atomic Power Development Associates, Inc.
with the Cooperation of
Power Reactor Development Company
and
The Detroit Edison Company

REPRODUCED FROM ORIGINAL

ISSUED: APRIL 1968

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PREFACE

PURPOSE

The purpose of this monthly report is to make available to the fast reactor program the current experience being gained from the Enrico Fermi Atomic Power Plant.

SCOPE

The scope of this report includes all phases of current operations and maintenance experience concerning the nuclear portion and related systems of the Enrico Fermi Atomic Power Plant.

Earlier Fermi experience in certain selected areas is being recorded in a series of technical reports completed or in preparation by Atomic Power Development Associates, Inc., for the U. S. Atomic Energy Commission under AEC Contract No. AT(11-1)-865, Project Agreement 15. This series of reports provides detailed information on the nuclear testing, machinery dome, steam generators, pumps, flowmeters, level detectors, sodium sampling, and development of the primary sodium system.

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

BACKGROUND

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

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

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

SECTION I

CURRENT EXPERIENCE SUMMARY

The object in the core inlet plenum was identified as one of six zirconium plate segments installed on the stainless steel conical flow guide during the final phase of reactor construction. Viewing of the conical flow guide confirmed that five zirconium segments were still in place and that one was missing. The present intention is still to remove the object through the 14-inch sodium inlet line. The other five segments will either be detached and removed or permanently fixed in place.

The sodium level in the No. 1 primary loop was lowered, and the riser portion of the 14-inch inlet line below the elbow was frozen to cool the elbow region. A patch was cut and removed from the secondary containment elbow in preparation for penetrating the primary pipe. Fabrication of a glove-box structure to maintain the atmosphere in the primary system during penetration of the primary pipe and removal of the object is under way. The spine-type manipulator, which is intended to operate through the core support plates and feed the object to the object retrieval device, is still being developed. The object retrieval device, which is intended to remove the object through the 14-inch line, is ready for final trial operation in the 14-inch pipe and inlet plenum mock-up.

Attempts were made to view the holddown fingers to determine whether or not the fingers were damaged. The new PRDC borescope was used in an effort to sight an image of the fingers in a spherical mirror installed in the reactor lattice. The attempts to view the fingers were unsuccessful because of many difficulties encountered.

SECTION II

INVESTIGATION OF THE OBJECT IN THE INLET PLENUM

A. BACKGROUND

An object, which was discovered in the reactor core inlet plenum in September, 1967, is believed to be the cause of the melting of two fuel sub-assemblies on October 5, 1966. Initial investigations and efforts directed toward removing the object have proceeded along two parallel paths, leading to (1) cutting up the object and removing the pieces through the core support plates and (2) retrieving the object intact through the 14-inch sodium inlet line.

A large number of photographs of the object were taken which aided in the development of cut-up and removal tools and devices. A stereoscopic analysis of selected photographs by Battelle Memorial Institute fixed the major dimensions of the object at 10.8 inches long and 4.8 inches wide on one side and 10.3 inches long and 4.7 inches wide on the other side. Attempts to obtain a sample of the object were unsuccessful.

In late December, preparations were made to remove the object through the 14-inch sodium inlet line. Emphasis on this removal method came about with the resolution of questions pertaining to penetrating the No. 1 primary sodium inlet line and subsequent restoration of system integrity.

Currently in the investigation, the important removal devices are the spine-type manipulator and the object retrieval device. The manipulator, operating through the support plates, will feed the object to the object retrieval device which, in turn, will remove the object through the 14-inch pipe. In late December, both of these devices were being developed and tested in the mock-up facility. At that time, tools were available to cut the object into small pieces for removal via the support plate.

Reports of the investigation of the object to date are given in APDA-CFE-14 through 17.

B. OBJECT IDENTIFIED

In late January, the object was identified as one of six 0.040-inch-thick zirconium plates originally installed on the sloping sectors of the conical flow guide in the reactor core inlet plenum. In Figure 1, the inlet plenum

mock-up is used to illustrate the location and appearance of the zirconium segments on the conical flow guide. Location of the segments is also shown in Figure 2.

A clue to the identity of the object came to light when a model of the probable original shape of the object ⁽¹⁾ (Figure 3) was laid against the surface of the conical flow guide in the inlet plenum mock-up. Then, while examining a construction photograph of the reactor core inlet plenum (Figure 4), the zirconium sheet was observed. In this photograph, a detail that appears to be a triangular liner plate fastened to the conical flow guide was noticed. This detail looks quite similar in shape and size to the model of the object.

The discovery of this similarity led to a search of records and files, as well as to a visual borescope inspection of the surfaces of the conical flow guide in the inlet plenum. These investigations turned up conclusive evidence that the object is one of the zirconium segments. It should be noted that drawings of the core inlet plenum were among those examined in detail early in the object investigation to determine whether the object could have been part of the reactor. The zirconium segments covering the conical flow guide were not shown on any of the drawings, however.

C. SUBSTANTIATING ZIRCONIUM PLATE INSTALLATION

Review of memoranda and construction logs has substantiated that zirconium liner plates were installed on the six sectors of the conical flow guide. It was indicated that the installation was an addition to the original design and occurred in the latter stages of reactor construction. The zirconium segments first were installed on the conical flow guide at a local machine shop and then removed at the construction site to permit a thorough cleaning of the parts. Construction logs revealed that the zirconium segments were reinstalled at the construction site and secured with zirconium screws at the three corners. It was specified in instructions to the installation crew and confirmed in the construction logs that each screw head was to be tack welded in two places to the zirconium plate.

D. VIEWING AND PHOTOGRAPHING THE CONICAL FLOW GUIDE

Immediately after the identity of the object was established, the accessible sectors of the conical flow guide were viewed and photographed through the new PRDC borescope (see Section IV.A). The borescope, along with a light source, was inserted through one of the 6-inch penetrations in the rotating shield plug and through the available holes in the core support plates.

(1) APDA-CFE-17, Section II. C.

FIG. 1 INLET PLENUM MOCK-UP

FIG. 2 MELTDOWN SECTION IN LOWER REACTOR VESSEL

FIG. 3 MODEL OF OBJECT AND TEMPLATE

FIG. 4 CONSTRUCTION PHOTOGRAPH OF INLET PLENUM SHOWING ZIRCONIUM SEGMENTS INSTALLED

Only four of the six sectors of the conical flow guide were easily accessible for viewing and photographing. These were the southeast, northeast, northwest, and southwest sectors. The presence of the reactor-sweep mechanism ⁽²⁾ prevented viewing of the north sector, and subassemblies remaining in the core interfered with viewing of the south sector. Both sectors will be viewed at a later time.

The view of the northeast sector of the conical flow guide revealed that the zirconium plate was missing but that the three zirconium screws for securing the plate to the conical surface were still in place. (See Figure 5.) With the zirconium plate missing, a vent hole through the conical flow guide near the apex could be observed. It is evident that hydraulic forces in the core inlet plenum ripped the zirconium liner plate from the screws holding it on the conical surface and battered it into the shape in which it was first discovered in September, 1967.

In the southwest sector, the zirconium plate was still in place, but the bottom screw on the south side was missing. Figure 6 shows the hole left by the missing screw, along with two areas of tack-welded metal for securing the screw to the zirconium plate. Figure 7 shows one of the round-head screws in place, tack welded in two places to the zirconium plate. Early records infer that the retaining screws may be 1/4-inch in diameter.

The zirconium plates and retaining screws in the southeast and northwest sectors were observed to be in place. (See Figure 8.)

E. REMOVAL PLANS

1. Zirconium Plate Segments

The present intention is to go ahead with the plan to remove the damaged zirconium segment (the object) through the 14-inch line. This will provide a demonstration of the feasibility of removing the other segments in a similar manner. In addition, knowledge of the characteristics of the damaged segment will be useful in devising means of freeing the others from the cone surface. Once removed, it is planned to use the object or a model of it in a flow test to demonstrate that it is capable of blocking a sufficient number of coolant channels to produce the fuel melting that occurred on October 5, 1966.

In addition to investigating removal methods for the five other zirconium segments, methods of providing more secure fastening of the segments to the cone will be investigated, as will methods for preventing the blockage of flow to subassemblies.

(2) APDA-CFE-4, page 26.

FIG. 5 SECTOR OF CONICAL FLOW GUIDE WITH ZIRCONIUM SEGMENT MISSING

FIG. 6 ENLARGED VIEW OF HOLE IN ZIRCONIUM SEGMENT LEFT BY MISSING SCREW

FIG. 7 ENLARGED VIEW OF ZIRCONIUM SCREW TACK WELDED TO ZIRCONIUM SEGMENT

FIG. 8 SECTOR OF CONICAL FLOW GUIDE WITH ZIRCONIUM SEGMENT AND SCREWS IN PLACE

2. Access Through the 14-Inch Sodium Inlet Line

During January, the development of tools and devices for removing the object through the 14-inch sodium inlet line continued (see Section II. F), as did work on the preparations for penetrating the elbow in the 14-inch line immediately outside the secondary containment wall. The sodium level in the riser pipe below the elbow to be penetrated was lowered 6 feet, and the pipe heating was turned off (see Section III. A. 3) to cool the sodium in the riser pipe. Both the lowering of the sodium level and the turning off of the pipe heating were done to provide reasonable comfortable temperatures for the object removal operation through the elbow.

Heating wire and insulation were removed from the elbow, and a 16-inch by 14-inch ellipsoid patch was removed by grinding from the 20-inch diameter secondary containment pipe elbow. Figure 9 shows the elbow before removal of the insulation, and Figure 10 shows where the patch was cut from the containment, exposing the primary pipe elbow. Before removing the containment patch, it was necessary to cut away a small part of the hanger bracket that overlapped the patch area. The hanger bracket was then satisfactorily reinforced by welding.

It was not necessary to provide forced cooling in the annulus between the primary and secondary pipes to cool the riser pipe below the elbow to be penetrated. Elimination of pipe heating was sufficient.

The preliminary procedures for cutting into the 14-inch line elbow⁽³⁾ were reviewed. Questions were raised, particularly with respect to the preparation of the sodium-exposed inside surface of the pipe elbow for weld closure. In addition to resolving these questions, the following work will have to be completed prior to cutting into the primary pipe elbow: the development of the removal devices; the functional testing of the removal devices in the reactor mock-up;⁽⁴⁾ and the fabrication of a glove-box structure for handling the tools and the object-removal device at the pipe elbow while containing the primary system atmosphere.

3. Glove Box at the 14-Inch Line Elbow

Figures 11 and 12 illustrate the concept of a glove box and removal-tool containment that will be firmly secured to the 14-inch pipe to enclose the elbow penetration. This structure will permit the maneuvering of the lengthy and cumbersome object-removal device while maintaining an inert gas seal around the elbow penetration.

(3) APDA-CFE-17, Section II. E. 3.

(4) APDA-CFE-16, page 24.

**FIG. 9 ELBOW TO BE PENETRATED — BEFORE
INSULATION REMOVED**

**FIG. 10 PATCH REMOVED FROM SECONDARY
CONTAINMENT PIPE**

FIG. 11 GLOVE BOX FOR 14-INCH PIPE ELBOW PENETRATION

FIG. 12 GLOVE BOX AND CONTAINMENT FOR OBJECT RETRIEVAL DEVICE

A steel adapter, which will be adjusted and secured to the containment elbow, will be shaped to fit the external contour of the primary pipe elbow to enclose the penetration. A silastic sealant will be applied to seal the adapter-elbow joint. An aluminum faceplate, which will be connected to the adapter by a transparent plastic bellows, will contain two glove penetrations and a third penetration for admitting tools to the glove box and the object-removal device to the interior of the pipe. A polyvinyl chloride containment for the removal device will be secured to the faceplate and will enclose the third penetration.

Fabrication of the glove box is currently under way. Details are subject to change, depending on experience gained when the structure is used for testing the removal devices in the reactor mock-up.

F. DEVELOPMENT OF TOOLS AND DEVICES

1. Spine-Type Manipulator ⁽⁵⁾

Tests conducted in the cold reactor mock-up have indicated that the use of lighter weight aluminum spine disks does not significantly improve flexibility of the tool; therefore, it was decided to retain the carbon steel disks. There was also concern that aluminum bearing on aluminum in a hot atmosphere might result in galling. To compromise, the steel disks were made somewhat lighter in weight by reducing peripheral thickness. The original disk thickness at the center or bearing surface was retained, however, as was the original outer diameter. Fabrication of the control mechanism, which will provide control for all four disk control cables, was completed. The mechanism used in the initial tests in the reactor mock-up provided control for only two of the four disk cables. This two-cable control mechanism was intended only as a temporary device to expedite testing of the tool. It did demonstrate the need for four-cable control to obtain effective control of the spine manipulator.

At the end of the month, these modifications had been accomplished. Reassembly of parts was started in preparation for another functional test in the reactor mock-up.

2. Object Retrieval Device

By the end of January, the object retrieval device ⁽⁶⁾ was assembled and ready for testing in the pipe and reactor mock-up. (See Figures 13 and 14 for photographs of the object retrieval device and its control mechanism.) The flexible duct, which is designed to receive and retain the

(5) APDA-CFE-15, 16, 17, Section II.

(6) APDA-CFE-15, 16, 17, Section II.

FIG. 13 OBJECT RETRIEVAL DEVICE AND CONTROL MECHANISM

FIG. 14 CLOSE-UP OF CONTROL MECHANISM

object for removal, contains a trolley-mounted gripper and spring fingers at its forward end to retain the object. The duct is 5 feet long and has a 4-3/4-inch minimum diameter at the opening. A cone-shaped adapter is fitted to the other end of the duct to prevent the duct from hanging up on any shoulder or obstruction in the reactor or pipe during removal. The duct is manipulated by a 40-foot length of 1-1/4-inch-diameter helical spring cable. Inside this cable is a run of 7/16-inch-diameter helical spring cable which can extend, retract, and rotate the trolley-mounted gripper in the flexible duct. A 1/8-inch push-pull cable operates inside the 7/16-inch cable to actuate the gripper. Prior to assembly, the flexible duct was degreased by exposure to hot solvent vapors. This was done to prevent any heat-solidified impurities from impairing the flexibility of the duct.

3. Articulated Lamp

A new articulated lamp is now available as a light source for the object-removal operation. (See Figure 15.) The lamp-tube structure uses the design features of the mechanical arm manipulator.⁽⁷⁾ The lamp is located at the end of the articulated arm. This design of lamp structure is an improvement over that previously used⁽⁸⁾ in that it allows the lamp arm to be positively held in the set position and does not require that the lamp touch down on the inlet plenum deck or on an object in the reactor. In addition, the new design protects the lamp socket from an accumulation of sodium oxide, which had been the cause of many previous lamp failures.

4. Right-Angle Hinged Gripper

Fabrication of the right-angle hinged gripper⁽⁹⁾ was completed in January. This gripper, which is shown in Figure 16, is a short-radius manipulator tool that is designed to operate through the holes in the support plate. It is not expected to be needed for the removal of the object but will be available as an alternate tool.

(7) APDA-CFE-16, page 18.

(8) APDA-CFE-16, page 12.

(9) APDA-CFE-16, page 19; APDA-CFE-17, Section II. D. 3.

FIG. 15 NEW ARTICULATED LAMP

FIG. 16 RIGHT-ANGLE HINGED GRIPPER

SECTION III

OPERATIONS

A. PREPARATIONS FOR COOLING THE 14-INCH LINE ELBOW FOR OBJECT REMOVAL

Sodium in the No. 1 primary loop was drained from the existing elevation of 573 feet to an elevation of 567 feet, and the sodium remaining in the riser pipe below the 14-inch pipe elbow to be penetrated for object removal was cooled. (See Figure 17 for an elevation layout of the primary sodium system.) The purpose of the draining and cooling was to lower the sodium level away from the working area and, as indicated in Section II. E. 2 , to cool the elbow for the object-removal operation. After the draining was completed, the sodium level was about 9 feet below the elbow.

Before the No. 1 primary loop draining operation, the drain pipes from the No. 2 and 3 IHX's were frozen to isolate primary loops No. 2 and 3 from the common drain header between all three IHX's and the overflow tank. This was done to allow the selective partial draining of the No. 1 primary loop. The extent of the draining was limited in order to retain sodium in the short 30-inch crossover pipe between the No. 1 pump and the IHX tank. Details of the above operations are given below.

1. Freezing the No. 2 and 3 IHX Drain Lines

A single heating circuit on the 2-inch-diameter drain line from each of the No. 2 and 3 primary loops was de-energized to freeze the sodium in each line, thus forming a freeze seal between the No. 2 and 3 primary loops and the common IHX drain header. Drain-line temperatures were reduced to about 150 F in somewhat less than 24 hours. The freeze seal extended for about 40 feet along each drain pipe.

2. Draining the No. 1 Primary Sodium Loop

Before draining of the No. 1 primary sodium loop was started, it was determined by calculation that 280 cubic feet of sodium would have to be transferred from the No. 1 primary loop and 240 cubic feet transferred from the overflow tank to lower the sodium level in the loop to an elevation of 567 feet. It was ascertained that No. 3 primary storage tank could contain this additional volume of sodium.

The new elevation of 567 feet is the level at which the 30-inch crossover pipe between the pump and IHX tanks will be half-full of sodium.

FIG. 17 PRIMARY SODIUM SYSTEM ELEVATION LAYOUT

(See Figure 17.) This condition is advisable for effective equalization of the temperature between the tanks. The deflection of one tank relative to the other due to thermal gradients must be less than 0.016-inch. This criterion takes into account the structural stresses at the 30-inch pipe-tank joints. The deflections actually produced in the operation were 0.006 inch and less, as measured by the temperatures on the pump and IHX support structures.

The draining operation was carried out in accordance with an established procedure. Alternately, sodium was pumped from the overflow tank into the No. 3 primary storage tank and then sodium was drained by gravity from the No. 1 primary loop into the overflow tank until the sodium levels equalized between the No. 1 primary loop and overflow tank. The level indicator in the overflow tank was used to measure the sodium level in the No. 1 loop. These operations were accomplished in several successive steps for effective control of the sodium level in the No. 1 loop.

Operation of control valve V521 in the common IHX drain header controlled the draining of the No. 1 primary loop. When V521 was in the open position, the sodium levels were equalized between the overflow tank and the loop. As noted previously, primary loops No. 2 and 3 were isolated by the freeze seals in the IHX drain lines in those loops. A total of 520 cubic feet of sodium was pumped from the overflow tank to the No. 3 primary storage tank, and a total of 280 cubic feet of sodium was drained into the overflow tank from the No. 1 primary loop.

After the sodium level in No. 1 primary system was stabilized at the 567-foot elevation, the overflow tank high-level alarm system, normally set to signal a high sodium level at 571 feet, 6 inches, was reset to signal a high sodium level at 567 feet, 9 inches. Two conditions that would cause the high-level alarm to signal are as follows: (1) a tube leak in No. 1 IHX, releasing secondary system sodium to No. 1 primary loop; or (2) the thawing of the IHX drain lines in primary loops No. 2 and 3, releasing sodium into the overflow tank and the No. 1 primary loop.

In the unlikely event of either of the above conditions occurring, it would be practically impossible for sodium to reach the penetrated elbow. With reasonable haste, the secondary loop could be drained to a secondary storage tank. If the frozen IHX drain lines thawed to permit draining of the No. 2 and 3 primary loops thus equalizing the sodium levels in all three primary loops and the overflow tank, the maximum elevation that could be attained would be 570 feet, or 5 feet below the open elbow.

3. Cooling Sodium in the Riser Pipe to the 14-Inch Elbow

Sodium was cooled in the 14-inch riser pipe below the elbow to be penetrated and in the lower horizontal run of the 14-inch pipe as far back toward the pump discharge as the 16-14-6-inch tee (the source of

sodium coolant flow to the reactor outer radial blanket inlet plenum). As shown in Figure 18, cooling of the sodium was accomplished by de-energizing the induction heating circuits on the 14-inch line from the 16-14-6-inch tee to the secondary shield wall. The induction heating on the syphon breaker line was also removed. Heating on the 6-inch line, which feeds the radial blanket inlet plenum, was continued.

Modification of the heater circuit was required so that the 6-inch line could remain heated. Resistor banks having a resistance equal to that of circuit 12-10 were substituted for that circuit, which was disconnected. In addition to disconnecting circuit 12-10, its windings were removed in the area of the elbow to be penetrated. This modification made it possible to re-energize circuit 12-14 for heating the 6-inch line without overloading the circuit.

The temperatures obtained from the unheated section of the 14-inch line are shown in Figure 18. The temperature of the elbow to be penetrated was 115 F, and near the 6-inch tee, the temperature was 160 F. The next nearest temperature point recorded was 348 F on the pump tank, 20 feet removed from the 160 F freeze point. It was observed that in the cooling process, cooling was accelerated when the patch was removed from the secondary containment pipe elbow. This was probably due to a chimney effect created in the annulus between the primary and secondary containment pipes when the elbow at the top of the secondary pipe was penetrated. Temperature data indicated that the sodium was frozen for a distance of about 19 feet back up the 14-inch line. These data also revealed that freezing occurred about four days after the elimination of heating.

Calculations have indicated that it would be possible to cool the 14-inch line adequately without forced cooling by de-energizing heaters. However, if it had been found necessary to achieve still further cooling, another patch could have been taken from the secondary pipe elbow and air could have been directed into the new hole from the below-floor cooling system.

4. Experience With Freezing and Thawing Sodium in Pipes and Containers

There has been considerable favorable experience gained at the Enrico Fermi Atomic Power Plant in freezing and thawing sodium in pipes and containers. This experience gives confidence that the frozen sodium in the 14-inch discharge line from the No. 1 primary pump can be liquified without causing unfavorable stress conditions in the 14-inch pipe.

The freezing-thawing experience has been with sodium in pipes up to the nominal 3-inch size and with sodium in tanks. When possible, the general procedure followed has been to apply the greater heat at the

FIG. 18 ARRANGEMENT OF HEATING CIRCUITS AND CIRCUIT MODIFICATION FOR COOLING 14-INCH ELBOW

sodium surface in contact with a gas volume so that melting will start at the sodium-gas interface, thus permitting the sodium to expand into the gas volume. Heat is then applied to melt the sodium in successive layers, each in contact with molten sodium. This allows sodium expansion through the molten region into the gas space.

When it is not possible to control the application of heat to start the sodium melting at the sodium-gas interface, the practice has been to melt the sodium by alternate heating and heat-soaking periods. Heat is first applied to the sodium mass for a short period, and then the heaters are de-energized; this permits the heat that was applied to soak through the mass of sodium. This method of melting requires close attention to temperatures throughout the sodium to be melted. Temperatures tend to rise together, and melting of the entire mass of sodium occurs at about the same time, thereby permitting expansion into an adjoining gas volume or through an adjoining molten sodium mass.

Some instances in which sodium has been frozen in pipes or other containment and subsequently melted without difficulty are listed in the following:

- Sodium was frozen in the IHX tube sheet seal leak detector piping after the pipe heaters failed.
- Sodium was frozen in the reactor syphon line to permit cutting into the line for connecting the reactor drain line.
- Sodium was frozen in the discharge line from the No. 1 primary storage tank to permit cutting into the line for connecting a make-up sodium line.
- Sodium freeze seals were created in the primary sodium service system to permit cutting out valves for seal bellows replacement.
- Sodium was frozen in IHX drain lines to permit isolation and repair of the drain control valve.
- Sodium was frozen in the primary service system cold trap to permit isolation and replacement of valve seal bellows.
- Tank cars arrived on the plant site containing frozen sodium. The sodium was melted in the tank cars before being transferred to the plant systems.

B. FILLING THE NO. 2 SECONDARY SODIUM SYSTEM

The No. 2 secondary sodium system, which includes the No. 2 steam generator shell, was filled with sodium from the No. 2 secondary storage tank. This operation was in preparation for a 600 psig nitrogen pressure test (see Section VII. B) to verify the integrity between the tube and shell sides of the unit subsequent to the completion of the internal bore weld repair.

The steam generator shell was filled with sodium to establish a shell condition similar to those for succeeding leak tests. Results of the succeeding tests, then, will serve as one indication of any changes in integrity between tube and shell sides of the unit in the course of operation of the steam generator.

The sodium-fill operation and the following cold-trapping operation were accomplished in accordance with established operating procedures. The details are essentially the same as those used for the filling of the No. 1 secondary sodium system and cold trapping.⁽¹⁰⁾

C. GAS SYSTEMS PERFORMANCE

Since the last data were reported,⁽¹¹⁾ the following primary system gas and cover gas analyses have been made:

1. Primary System Gas Activity

<u>Location</u>	<u>Sample Date</u>	<u>Gross Beta Concentration, microcuries/cc</u>
Reactor Cover Gas	1-5-68	1.3×10^{-6}
Reactor Cover Gas	1-12-68	6.6×10^{-6}
Reactor Cover Gas	1-17-68	3.5×10^{-6}
Primary Shield Tank	1-19-68	Not Detected
Primary Shield Tank	1-22-68	Not Detected
Reactor Cover Gas	1-22-68	1.98×10^{-6}
Primary Shield Tank	1-26-68	Not Detected
Reactor Cover Gas	1-29-68	2.48×10^{-6}
Primary Shield Tank	1-29-68	4.83×10^{-8}

(10) APDA-CFE-17, Section III. B.

(11) APDA-CFE-17.

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	160**
Carbon Monoxide	Below 10	Not Measured
Carbon Dioxide	Below 10	Not Measured
Hydrogen	Below 4***	2.5
Helium	Below 4	Below 4
Methane	Below 10	Not Measured
Nitrous Oxide	Not Measured	Not Measured
Argon	Remainder	Not Measured
Nitrogen	4800	
Dew Point	Not Measured	-50 F
Sample Date	January 29, 1968	January 5, 1968

* Cover plate seals for PST machinery deck penetrations
were replaced in January.

** Technical specifications state 1000 ppm maximum

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

SECTION IV

INSPECTION OF HOLDDOWN FINGERS

Although calculations have indicated that the holddown fingers would not be deformed by the subassemblies containing molten fuel, ⁽¹²⁾ it is believed prudent to inspect the fingers to verify those calculations. The southeast quadrant of the reactor, where the bent subassemblies M140, M122, M127, and M098 were located, is of greatest interest; therefore, the present plan is to inspect the holddown fingers in this quadrant first. If this inspection verifies that there is no finger damage in this quadrant, it is unlikely that any further inspection will be made. If finger damage should be detected, a decision will have to be made about the extent of additional inspection. It is intended that holddown finger inspection will be accomplished by viewing and photographing the image of the fingers in a spherical mirror inserted in the reactor.

In January, the fabrication of equipment for the inspection of the holddown fingers was complete, and the equipment was ready for insertion into the reactor. Significant details of both the equipment and the in-reactor experience during the month are given below. Access to the reactor compartment for this inspection program was possible at this time because the object removal program (see Section II) was centered on the development of removal tools and there was little work being done in the reactor.

A. PRDC BORESCOPE

The Sunscope and the new PRDC-designed borescope were the items of viewing equipment selected for the holddown finger inspection program. Information about the Sunscope may be found in a previous report. ⁽¹³⁾ The PRDC borescope was constructed to provide improved viewing in the reactor. Its principal advantage over previously available scopes (Sunscope, Core-scope, Atomics International borescope) is that it contains only five relay lenses, one-third of the number in the Sunscope, and can be expected to provide a clearer image for viewing and for photographing. Construction of the borescope was completed in January.

The PRDC borescope is 2-1/2 inches in diameter and has several interchangeable sections for varying its length, field of view, and angular

(12) APDA-CFE-13, page 13

(13) APDA-CFE-13, page 9.

direction of view. The relay lenses are noncemented doublets, some of air-spaced optical glass and the remainder of direct-contact optical glass.

B. TESTS OF VIEWING EQUIPMENT

The available scopes were tested under mock-up conditions to determine their respective capabilities for use in inspecting the holddown fingers. A cardboard mock-up of the holddown plate and fingers, the spherical mirror to be used, and a given scope were set up in positions and at distances comparable to those that would exist in the reactor. In a test of the PRDC borescope, it was determined that the mirror filled an 8° field of view and that images 1/16-inch apart could easily be distinguished. In a similar test of the Corescope, images separated by 1/8-inch could not be distinguished. In another test, the superiority of the PRDC borescope over the Sunscope in giving image resolution was demonstrated.

The tests confirmed that the PRDC borescope has the capability for adequately viewing the holddown fingers as reflected in the spherical mirror and established the optimum field of the view adapter (8° objective). Therefore, the PRDC borescope will be used to view the mirror image of the holddown fingers. The Sunscope, with its wider field of view, will be used for viewing the mirror insertion and alignment operation.

C. SPHERICAL MIRROR

The spherical mirror is a 5-3/4-inch-diameter piece of ground quartz, 15/16-inch thick. The convex mirror surface has a 10-inch radius of curvature. The surface is coated with aluminum that is protected by a thin coating of silicon dioxide. A retaining ring secures the mirror to the support post which, when inserted in the reactor support plate, locates the mirror at an elevation 4 inches below the top of the subassembly handling heads. An eccentric ball-and-socket fixture at the top of the mirror support post provides for fitting the mirror to the proper angle for viewing the holddown fingers. Figure 19 shows the mirror and support post.

An insertion device at the bottom end of a 27-foot-long steel tube engages with the mirror post and inserts the mirror into the reactor through one of the 6-inch-diameter guide tube penetrations in the rotating shield plug. Figure 20 shows the insertion tool engaged with the mirror post, and Figure 21 shows a close-up of the insertion device.

D. TRIAL INSERTION OF THE MIRROR INTO THE REACTOR

A trial insertion and removal of the mirror assembly was conducted in the reactor vessel to uncover any unforeseen difficulties in the operation. A stainless steel mirror was used in place of the quartz mirror. The insertion tool deposited the mirror assembly in lattice position N07-P00 (see Fig-

FIG. 19 SPHERICAL MIRROR AND SUPPORT POST

**INSERTION TOOL ENGAGED WITH
MIRROR POST**

**FIG. 21 CLOSE-UP OF MODIFIED
INSERTION DEVICE**

ure 22), and without delatching, removed the assembly from the reactor through one of the 6-inch penetrations in the rotating shield plug.

Interference was encountered at the bottom of the rotating plug as the large-diameter wire guide cage over the mirror was being maneuvered into the access penetration in the rotating plug during removal. As a result, the wire guide cage was modified to provide better lead-in to the rotating plug penetration. The eccentric ball-and-socket fixture could not be depended on to rigidly hold the set mirror angle as the mirror assembly passed through the penetration in the rotating plug. Three interchangeable fixed mirror supports for holding the mirror at angles of 24° , 32° , and 45° from the horizontal were fabricated to substitute for the ball-and-socket fixture.

E. ATTEMPTS TO VIEW THE HOLDDOWN FINGERS

Attempts were made to view the holddown fingers through the PRDC borescope equipped to provide an 8° field of view. The PRDC borescope, the Sunscope for wider range viewing, and the light source were installed in access ports in the rotating plug approximately over the P09-N08 lattice position. Shown in Figure 22 is the mirror in lattice position N07-P00, as well as the line of sight between the borescope and the mirror. The fixed-angular mirror support set the mirror at an angle to view the holddown fingers in the southeast quadrant. In previous reports⁽¹⁴⁾ are listed the changes in the core loading arrangement to accommodate both the insertion of the mirror assembly and the viewing of the holddown fingers.

Attempts made to view the holddown fingers in January were unsuccessful. In the first attempt, the mirror, when viewed through the borescope, was found to be flaked, apparently as a result of abrasions incurred during insertion. In a second try, the mirror was not correctly oriented for viewing. During attempts to relatch the insertion tool on the mirror support post to swivel the mirror into the proper position for viewing, the mirror post was inadvertently unseated and tilted. A wire sling was then required to remove the mirror from the reactor because the mirror post was moved out of the reach of the insertion tool. The mirror, when being removed, was again damaged.

Contributing to the mirror-handling difficulties was the inability to adequately view the latching of the insertion tool with the mirror post. The wire cage structure and the umbrella guard at the end of the insertion tool obstructed the view of the latching position on the mirror post.

(14) APDA-CFE-13, 17.

N16 N15 N14 N13 N12 N11 N10 N09 N08 N07 N06 N05 N04 N03 N02 N01 P00 P01 P02 P03 P04 P05 P06 P07 P08 P09 P10 P11 P12 P13 P14 P15 P16

KEY

CR Control Rod No. 430169-
 SR Safety Rod No. 430192-
 OR Oscillator Rod
 CS Core Shim Subassembly
 CF Core Foil Subassembly
 BF Blanket Foil Subassembly
 CT Core Filter, Take-apart, Dummy Core Subassembly
 NOTE Dummy Core subassemblies in the reactor meet
 Core Assembly specifications and bear the
 identification CF

NA Sodium Worth Subassembly
 NS Neutron Source
 TIT Temporary Insertion Thimble
 MS APDA Materials Surveillance Subassembly
 M Subassembly Manufactured by D.E. Makepeace Co.
 S Subassembly Manufactured by Sylvania Division,
 Sylvania Electric Products Co.

M 001 - M 205 Core Subassemblies
 M 301 - M 400 Inner Radial Blanket Subassembly
 S 500 - S 798 Outer Radial Blanket Subassembly
 M 801 - M 1000 Outer Radial Blanket Subassembly

Units without prefix are dummy outer radial blanket subassembly

- Oversize Nozzle Unit
- F' Subassembly (Contains fuel pins with high iron plus nickel, high carbon or Zirconium content.)
- W' Subassembly (Contains fuel pins with high iron plus nickel content.)
- BL Blanket slugs have high carbon content (APDA Surveillance Program Unit)
- Blanket Slugs
- Large Grain Blanket Material (Hafnium)
- Large Than Normal Spacing Between the Blanket Elements and the Support Grid
- Type 347 Stainless Steel Wrapper Tube
- Handling Head Short
- Test Flow Subassembly (S-100)
- Slugs Previously Used in a Test Subassembly
- CP' Slugs

Note: Heavy Boxes
 indicate Flat or
 other special

FIG. 22 REACTOR LOADING 1-31-68

SECTION V

EXIT PORT INSPECTION FACILITY

In January, modifications were made in the exit port inspection facility to correct deficiencies discovered during preoperational testing of the facility. Earlier information about this facility has been given in previous reports. (15)

Ground quartz plates were installed in the inspection chamber, replacing all the Pyrex windows. The Pyrex glass appeared to have been introducing dimensional inaccuracies.

The upper actuator rod port in the hoist housing was relocated one foot higher on the housing. This was done to permit engagement of the gripper actuator rod with the gripper assembly to latch and delatch the gripper on a subassembly handling head when the subassembly is supported in the inspection position.

The taut vertical wire used as a base in dimensional and distortion inspections of subassemblies was relocated within the inspection chamber to eliminate a slight bowing of the wire. This bowing was introduced by the wire coming in contact with the back wall of the inspection chamber. Extensions were added to the upper and lower wire mounting fixtures to hold the wire slightly away from the inspection chamber wall, thus ensuring straightness.

(15) APDA-CFE-5, 8, 9, 10, 16, 17

SECTION VI
SPECIAL INVESTIGATIONS

A. RADIATION LEVELS IN THE REACTOR

A survey of radiation levels in the reactor compartment was conducted in January. A detector monitored gamma radiation levels at approximately one-foot intervals as it was withdrawn vertically from the top of the upper support plate. The detector was installed through a guide-tube penetration in the rotating shield plug over lattice position N05-P01 (see Figure 22). Beta radiation was eliminated by the detector housing and by an aluminum tube which enclosed the detector.

Results of the radiation survey are given in Table 1. The highest recorded level of radiation was 1550 roentgens per hour, which was observed at about the midplane of the core region.

B. SUBASSEMBLY INSPECTIONS AT BMI

In January, metallographic samples of subassemblies M122, M140, M098, and M127 were viewed in the hot cell at Battelle Memorial Institute. The majority of samples were viewed in the as-polished condition. M098 and M127 are the two melted subassemblies, and M122 and M140 were adjacent to M098 and M127 in a four-subassembly cluster in the reactor. Earlier inspection results of these four subassemblies have been described in previous reports. (16)

Preliminary observations of the metallographic samples of subassemblies M122, M140, M098, and M127 are given below.

1. M122 - Upper-End Cap Bulging

Two fuel pins were examined for indications of the cause of upper-end cap bulging on nine fuel pins in M122 observed in the November, 1967, inspection. One of the pins examined had been located in the outside row adjacent to subassembly M098. The other pin examined had been located in the next row inward.

The bulging appeared to be caused by movement of the cap over a thicker section of the fuel pin. A possible mechanism may have been thermal bowing of the outside fuel pins, which could have caused the adjacent

(16) APDA-CFE-11 through 17.

TABLE 1 - RADIATION LEVELS IN THE REACTOR

<u>Depth Below Top of Rotating Plug Guide Tube Penetration, feet-inches</u>		<u>Roentgens per Hour</u>
6-5		0.93
10-5		1.63
12-0	Bottom of Rotating Plug	
12-5		4.89
14-5		5.89
15-5		9.69
16-5		12.8
17-5		17.4
18-5		27.7
19-5		42.6
20-5		66.6
21-5		96.9
22-5		143
23-4	Top of Handling Heads	
23-5		194
24-5		215
25-5		853
26-5	Midplane of Core Region	1550
27-5		1320
28-5		515
29-9	Top of Support Plate	221

end caps to make contact with the wall of the wrapper can, thus pushing the caps away from the fuel pins. Observations of the fuel pins revealed that the caps were not bulged uniformly around the circumference. There was no evidence of corrosion under the end caps to suggest moisture as a mechanism for moving the end caps away from the fuel pins.

The upper end caps and the fuel cladding near the caps appeared to be excessively hydrided (approximately 4000 ppm in both cases). However, there was considerable variation of hydriding along the length of the end cap.

2. M122 and M140 - Alloy on the Outside Surface of the Wrapper Can

In previous inspections, deposits believed to be fuel alloy were observed on the outside surfaces of the stainless steel wrapper cans of subassemblies M122 and M140. The fuel alloy was thought to have come from either or both of the melted subassemblies M098 and M127. Transverse sections from subassemblies M122 and M140, which included the alloy deposits, were metallographically examined in the as-polished condition. The section from M122 had been taken at an elevation corresponding to the large hole in the bottom of the core of subassembly M098. The section from M140 had been taken from a location several inches below the spacer pads.

The metallographic examination revealed that there was no appreciable diffusion of the fuel alloy into the 347 stainless steel wrapper can of either subassembly. Rather, the alloy seems to have adhered to the wrapper can similar to a well-tinned solder coating. Results of the above metallographic examination will have to be confirmed by observation of etched specimens.

The maximum thickness of the alloy deposit on M122 was 0.105 inch, and on M140, 0.005 inch. At the present time there is no suitable explanation for the large thickness of the deposit on M122. It must be noted, however, that a hole in the M098 wrapper can matched the alloy deposit on M122.

3. M098 - Examination of U-Mo Blanket Alloy

A transverse section of an upper axial blanket rod from M098 was taken from the lower region of the rod in the vicinity of the lower-end cap weld. In a previous inspection, this end cap weld had appeared to be damaged.

With the stainless steel cladding stripped from the rod, the blanket alloy rod was observed to have a plastic-like surface appearance. The diameter of the alloy slug was optically measured to be 0.363 inch. The original diameter was 0.395 inch. From observation of the alloy slug in the polished condition, it was impossible to determine whether the decrease in diameter was caused by a corrosion or a melting mechanism. An examination of the alloy slug in the etched condition may reveal the cause.

4. M098 and M127 - Fuel Pins Exhibiting Large Voids

Longitudinal sections of fuel pins from subassemblies M098 and M127 were taken in an area where large voids had been observed in the fuel alloy. These voids had previously been observed in longitudinal samples taken from M098 and M127. The void areas were located at about the third

grid from the top and slightly below. Extensive melting had occurred just below the void region, but above it the fuel pins were intact.

Metallographic examination of the sections in the polished condition showed that even where the fuel in the pins appeared solid, extensive spherical voids existed. The voids were smaller near the top and increased in size toward the bottom of the section. The clad was intact and did not appear to be bulged significantly. The structural appearance of the zirconium clad suggested hydriding, although extensive diffusion of uranium alloy into the zirconium could have occurred.

SECTION VII
MAINTENANCE

A. PRIMARY SHIELD TANK PRESSURE DECAY TESTS

Two successive 12-hour primary shield tank pressure decay tests, each from 2 psig, were conducted. The results of the tests indicated that leakage was occurring in excess of 3500 standard cubic feet per day (scfd). The maximum allowable leakage during reactor operation is 2400 scfd. This is a routine annual test. The leak rate measured in January, 1967, was 1340 scfd. (17)

Several large leaks were discovered around the bolted cover plates over the instrument and valve access penetrations in the machinery deck of the primary shield tank. The leaks were discovered audibly and by bubble testing. To correct the leakage, the O-ring seals between the penetration cover plates and the machinery deck were replaced. The replaced O-ring seals were the original ones installed, and had been in service about eight years.

Another pressure decay test will be conducted to ensure that the seal replacement has satisfactorily reduced the leak rate. If the leakage has not been sufficiently reduced, other corrective measures will be taken.

B. LEAK TEST OF THE NO. 2 STEAM GENERATOR

After the No. 2 steam generator was filled with sodium (see Section II-B), a gas pressure test was performed to verify the integrity between the tube and shell sides of the unit. The pressure test procedure employed was identical to that used for pressure testing the No. 1 steam generator in December. (18) There was no measureable leakage.

C. PREPARATIONS FOR TUBE-TO-TUBE SHEET
WELD REPAIRS ON THE NO. 3 STEAM GENERATOR

By the end of January, preparations for tube-to-tube sheet weld repairs at the water inlet header of the No. 3 steam generator were complete. The preparation of the tubes for weld repairs was the same as those described on page 24 of APDA-CFE-7. The weld repairs are expected to begin shortly.

(17) APDA-CFE-6, page 29.

(18) APDA-CFE-17, Section VI.A.2.