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APDA-CFE-26

MASTER

ENRICO FERMI ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

**COMPILATION OF CURRENT TECHNICAL
EXPERIENCE AT**

ENRICO FERMI ATOMIC POWER PLANT

OCTOBER 1968

R. H. Costello

**United States Atomic Energy Commission
Contract No. AT (11-1) - 865
Project Agreement No. 15**

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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 65 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-23.

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

It was demonstrated in two complete removal operations in the reactor mock-up that a zirconium segment can be dislodged from the conical flow guide, manipulated in the inlet plenum, and removed through the 14-inch sodium inlet line. It is intended to dislodge the remaining segments from the conical flow guide in the mock-up and remove them also through the 14-inch line to demonstrate that all segments can be removed from the reactor with the available tools and devices. Damage to the object-retrieval device and the viewing lamp during segment removal test operations in the mock-up emphasized the value of mock-up testing in revealing unforeseen difficulties in the use of tools and equipment to be used in the reactor. These devices were easily repaired following modifications to eliminate the source of the difficulty.

A new concept for vacuum cleaning any accumulation of debris from the plenum is being developed. The new concept places the vacuum source exterior to the reactor to provide access and to assure adequate capacity; in a previous concept, a blower-motor unit was to be inserted in the inlet plenum, but the advantages of easy access and availability for maintenance prompted the redesign of the vacuum system.

The three primary sodium pumps were operated at refueling speed during the month and primary system temperatures were maintained at 350 F to 375 F during the first half of the month. During the latter half of the month the primary system temperatures were increased to 440 F to provide secondary sodium temperatures as high as possible to permit a thermal shock test of the No. 2 steam generator. Three successive shock tests were performed on the No. 2 steam generator by suddenly venting the water side of the unit and in this way reducing the temperature of the water tubes. This test confirmed the integrity of the unit under conditions of severe and rapidly occurring thermal gradients between the water and sodium spaces. A preliminary hydraulic test on the No. 3 steam generator with orifice tubes installed indicated that the water tube pressure drop across the unit is somewhat less than had been predicted. The leaking manifold gaskets were replaced on the No. 2 steam generator, and similar work was nearing completion on the No. 3 steam generator at the end of October. Three leaking tubes in the No. 1 steam generator were plug-welded.

The sweep mechanism was removed from the reactor and stored in a special cask in a decay tank in the reactor containment building. The sweep was removed to permit insertion of tools and equipment into the reactor for the inspection and removal of the zirconium segment on the north sector of the conical flow guide. The presence of the sweep mechanism had previously prohibited access to this sector.

II. INVESTIGATION OF ZIRCONIUM SEGMENTS

A. BACKGROUND

The deformed sheet of zirconium 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. Details of the removal of the first segment are given in APDA-CFE-20. The present plan is to dislodge the remaining segments from the conical flow guide and remove them through the 14-inch sodium inlet pipe of the No. 1 loop in a similar manner.

An arc-melt tool was developed to melt the heads of the fastening zirconium screws* to free the segments from the conical flow guide. This tool, designed to operate through a 6-inch penetration in the rotating shield plug and holes in the support plates, has been functionally tested in a mock-up of the reactor vessel and is now ready for a demonstration of the complete segment dislodging and removal operation in the mock-up. During this demonstration, the reactor viewing conditions are to be simulated as closely as possible. It was demonstrated that the arc-melt tool has adequate access to all fastening screws on the conical flow guide. Tools to dislodge the segments after arc-melting the fastening screws have been developed and are available, if needed.

A new electrohydraulic control system (see APDA-CFE-23, page 8) for the spine-type manipulator** has been assembled and is ready for segment-removal operations in the vessel mock-up. The spine-type manipulator was operated through a 6-inch penetration in the rotating plug and support plate holes to manipulate the first segment for removal via the 14-inch inlet pipe.

A new football shaped gripper body for the object retrieval device,** designed to grasp a segment in the inlet plenum and remove it through the 14-inch inlet pipe, has been fabricated and is ready for test in the mock-up. The gripper body, now constructed of aluminum, is a redesign of the original body structure for improved control of movements in the inlet plenum when the gripper is extended from the retrieval hose. Previous information on the investigation of the zirconium segments is given in APDA-CFE-20 through 25.

* See Section II of APDA-CFE-21 through 25 for previous information about the arc-melt method.

** See Section II of APDA-CFE-15 through 19 for previous information about the spine-type manipulator and the object-retrieval device.

B. DEMONSTRATION OF SEGMENT REMOVAL IN REACTOR MOCK-UP

1. Melting Screwheads

In October it was demonstrated in two complete removal operations in the reactor mock-up that a zirconium segment can be dislodged from the conical flow guide, manipulated in the inlet plenum, and removed through the 14-inch sodium inlet line. The freed segments were located on the conical flow guide (see Figure 1) in positions which are the farthest removed from the inlet nozzle of the No. 1 loop. It was, therefore, required that they be maneuvered under the perforated deflector plates (see APDA-CFE-14, p. 16) to put them in position to be grasped by the retrieval gripper for removal. A close-up of this football shaped retrieval gripper is given in Figure 2.

The arc-melt tool was operated from the station on top of the mock-up of the rotating shield plug and positioned against the fastening screwheads with the aid of viewing and lighting equipment in the same manner as is anticipated in the reactor. The tool successfully melted the zirconium screwheads to free the segments from the stainless steel conical flow guide. One objective of the mock-up test was to determine the minimum number of moves of the tool from one shield plug penetration to another to obtain access to all of the screwheads. After the screwheads were melted, it was found that a slight nudge of the arc-melt tool against a segment caused the segment to fall clear of the conical flow guide surface and onto the floor of the mock-up inlet plenum. Since it may not be as easy to dislodge a segment in the reactor because of the possibility of a zirconium-to-stainless steel bond under inert gas conditions, the use of the available dislodging tools may be needed for this purpose.

2. Removing Segments

The two segments dislodged from the conical flow guide were pushed along the floor of the inlet plenum mock-up and under the deflector plates by the bottom end of the viewing lamp to position them in the area where they could be transferred or delivered to the retrieval gripper. Figures 3 through 8 show the sequence of transferring a segment to the retrieval gripper and removing it through the 14-inch inlet line.

The present intention is to free and dislodge the remaining segments from the conical flow guide in the mock-up and remove them also through the 14-inch pipe to demonstrate that all segments can be removed from the reactor with the available tools and devices.

3. Spine-Type Manipulator Control Difficulty

Erratic control of the spine-type manipulator was encountered when operating the device in the inlet plenum mock-up. This difficulty did

023668

FIG. 1 ZIRCONIUM SEGMENT REMOVAL OPERATIONS IN MOCK-UP OF INLET PLENUM

**FIG. 2 NEW FOOTBALL-SHAPED GRIPPER
FOR OBJECT RETRIEVAL DEVICE**

023868

FIG. 3 ZIRCONIUM SEGMENT TRANSFERRED TO RETRIEVAL GRIPPER BY SPINE-TYPE MANIPULATOR IN MOCK-UP OF INLET PLENUM

FIG. 4 SEGMENT RELEASED BY SPINE-TYPE MANIPULATOR

023968

024068

FIG. 5 RETRIEVAL GRIPPER PARTIALLY WITHDRAWN INTO FLEXIBLE RETRIEVAL HOSE

**FIG. 6 SEGMENT ENTERING RETRIEVAL HOSE
IN MOCK- UP OF INLET PLENUM**

024168

**FIG. 7 RETRIEVAL HOSE BEING WITHDRAWN
FROM INLET PLENUM**

024268

**FIG. 8 RETRIEVAL HOSE WITH SEGMENT BEING
WITHDRAWN FROM MOCK-UP OF INLET
PLENUM**

024368

not prevent the manipulator from handling the segment but did add considerably to the time required to carry out the operation. The spine column moved in a jerky fashion, due to faulty timing of the sequence of actuations of the electrical control components regulating the hydraulic pressure to the pistons which move the spine disk control cables. The jerky motion was corrected by eliminating the restraining force on the relaxing cables when the hydraulic power is applied to the opposite cable or cables to flex the spine column.

Figure 9 shows the hydraulic piston housing attached to the top of the manipulator, and the control toggle conveniently located near the bore-scope. The special power unit which is the source of hydraulic pressure for the control cable actuating pistons is shown in Figure 10. The rotatable gripper for the spine manipulator is shown in Figures 11 and 12. This gripper is rotated by a ratchet device manipulated by the cable which opens and closes the gripper jaws. This cable extends through the manipulator housing and is operated on top of the rotating plug. The gripper can be rotated only when the gripper jaws are open.

4. Damaged Flexible Retrieval Hose and Viewing Lamp

Damage to the flexible retrieval hose, shown in Figure 13, was discovered during the segment removal operations in the mock-up when the first segment was pulled through the penetration in the 14-inch pipe elbow. The end of the hose was partially torn from the metal block to which it is welded and which secures the handling cable to the retrieval hose. It appears that the hose-to-block weld was weakened during fabrication when the weld joining the two pieces was ground smooth. A 6-inch-long sleeve inserted in the retrieval hose was welded to both the metal block and the end of the flexible retrieval hose and additionally secured to the hose by four screws to reinforce the hose-to-block connection. It was still possible to pull the retrieval hose out of the 14-inch pipe despite the weakened cable-to-hose connection because the segment, firmly grasped by the retrieval gripper and thus being pulled by the handling cable, was wedged into the bottom of the hose and pushed the hose up the 14-inch pipe.

The discovery of the above damage emphasizes the value of mock-up testing in revealing unforeseen difficulties in the use of tools and equipment to be used in the reactor. Satisfactory operations in the mock-up has instilled confidence in removing the segments from the reactor.

Figure 14 shows the articulated lamp used in the mock-up testing and intended for use in the reactor. In another instance of the advantages of mock-up testing, the viewing lamp screen expanded more than the surrounding structure, apparently because of the concentration of lamp heat, and buckled beyond the confines of the housing. Upon withdrawal through a hole in the lower support plate, the screen crumpled and caused the lamp to jam in the lower support plate. This condition is shown in Figures 15 and 16. A new screen

Control
Toggle

FIG. 9 BORESCOPE VIEWING IN MOCK-UP OF INLET PLENUM

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**FIG. 10 HYDRAULIC POWER UNIT FOR CONTROL
OF SPINE-TYPE MANIPULATOR**

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024668

**FIG. 11 ROTATABLE GRIPPER FOR
SPINE-TYPE MANIPULATOR**

FIG. 12 ROTATABLE GRIPPER DISASSEMBLED

024768

024868

**FIG. 13 RETRIEVAL HOSE DAMAGED
DURING MOCK-UP TESTING**

025268

FIG. 14 ARTICULATED LAMP

**FIG. 15 LAMP SCREEN DAMAGED IN MOCK-UP
SUPPORT PLATE**

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**FIG. 16 CRUMPLED LAMP SCREEN
AFTER REMOVAL FROM
MOCK-UP**

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FIG. 17 GUARD RODS OVER LAMP SCREEN

025168

was installed and reinforced by three 1/8-inch-diameter rods as shown in Figure 17. This new screen is constructed of finer mesh than that used in previous viewing operations. The finer mesh screen permits improved light emission.

C. NEW CONCEPT FOR VACUUM-CLEANING INLET PLENUM

A new concept for vacuum cleaning any accumulation of debris from the segment removal operations in the inlet plenum is presently being developed. The purpose of a vacuum cleaning system would be to remove any debris from the inlet plenum. Effort had previously been initiated to develop a vacuum cleaning device that would be totally inserted in the inlet plenum (see APDA-CFE-22, p. 8). However, it was decided to redesign the device to keep the source of vacuum external to the primary system for easy access and maintenance. This decision was aided by the demonstration of a commercial vacuum unit over a 40-foot rise with an air velocity of approximately 50 cfm.

The present scheme is to insert heat-resistant vacuum tubing into the inlet plenum through the 14-inch inlet line. The end of the vacuum tubing would be maneuvered to cover the plenum floor by use of the spine-type manipulator or by means of a guide cable rigid enough to be handled from outside the 14-inch pipe elbow. The vacuum source could be a commercially available vacuum cleaner located outside the reactor or, preferably, the plant waste gas system, which offers the advantage of convenient disposal of contaminated gas from the reactor. In a mock-up test using a 1-horsepower commercial vacuum cleaner and a plastic hose of 15/16-inch inside diameter, sample screws were easily pulled into the vacuum cleaner through a more tortuous recovery path than is anticipated in the reactor. The gas flow during this trial was estimated to be 50 to 60 cfm, which is within the demonstrated capacity of the waste gas system of 79 scfm.

In further developmental work, it is expected that the following will be investigated:

- a. The capacity of various components of the waste gas system
- b. The capacity and response of the reactor argon gas make-up system to ensure maintaining a constant positive reactor cover gas pressure if the waste gas system is used as the vacuum source. Vacuum cleaning operations may be either constant or intermittent
- c. The ease of maneuvering the end of the vacuum tubing by the spine-type manipulator
- d. The concept of a rigid cable arrangement as an alternate method of maneuvering the vacuum tubing in the inlet plenum.

- e. The concept of a separator device in the vacuum system to capture screws and weld debris, and to be located either in the inlet plenum or outside the reactor.

D. VACUUM FUSION ANALYSIS OF FIRST ZIRCONIUM SEGMENT

As stated in APDA-CFE-20, page 16, a sample of the tab* taken from the first zirconium segment was metallographically estimated to contain 400 ppm hydrogen. This determination was made by The Detroit Edison Company Engineering Research Department, who also estimated that the hydrogen content was 70 ppm in a specimen believed to be of the original material. The hardness of the specimen of original material is 169 Knoop; the hardness of the specimen taken from the segment was 155, 146 and 147 Knoop as determined in three tests.** These tests indicate a very slight softening effect on the segment due to its experience in the reactor, despite a significant increase in hydride content which might reasonably be expected to have caused an increase in hardness. The configuration of the zirconium segment taken from the reactor in March is evidence of good ductility.

Two samples of the same tab taken from the first segment and samples of two other parts of the segment were analyzed at Battelle Memorial Institute for hydrogen and oxygen content by the vacuum fusion method. Following is a tabulation of the results of these analyses.

TABLE 1 - VACUUM FUSION ANALYSIS OF SEGMENT SAMPLES

<u>Sample Location on Segment</u>	<u>Content, ppm by weight</u>	
	<u>Hydrogen ($\pm$ 3)</u>	<u>Oxygen ($\pm$ 35)</u>
First Tab	725	2640
Second Tab*	445	3300
In area of bolt hole at apex of segment	740	1500
Center of Segment	145	1350

* This sample was taken from the same location as the sample that the Detroit Edison Company Engineering Research Department metallographically estimated to contain 400 ppm hydrogen.

* APDA-CFE-20, Figure 12

** This range of Knoop hardness values is the same as the 76 B Rockwell value reported in APDA-CFE-20, p. 22.

It is believed that the zirconium segment was affected very little by the hydrogen from the sodium, based on the above analysis for the specimen from the center of the segment, which is considered most representative of the bulk of the segment. This analysis result is in general agreement with observations of 150 to 300 ppm hydrogen in the zirconium cladding on the fuel pins from subassembly M156, inspected soon after the incident of October 5, 1966 (see APDA-CFE-8, p. 17).

The other three samples can be expected to show high hydrogen content because they are from locations affected by overheating from welding and torch-forming during initial installation. It is possible to obtain hydrogenation during the welding of zirconium if the atmosphere is not inert and if sufficient moisture is present. It is also possible that welding in a somewhat oxidizing atmosphere might make the oxide film less resistant to hydrogen permeation from the sodium than an oxide layer formed at a lower temperature.

III. OPERATIONS

A. GAS SYSTEMS PERFORMANCE

Since the last data were reported (APDA-CFE-25), the following primary system gas activity and gas analyses have been obtained:

1. Primary System Cover Gas Activity

<u>Location</u>	<u>Sample Date</u>	<u>Gross Beta Concentration, microcuries/cm³</u>
Reactor Cover Gas	10-11-68	4.1×10^{-6}
Reactor Cover Gas	10-18-68	1.5×10^{-5}
Reactor Cover Gas	10-25-68	7.5×10^{-7}
Primary Shield Tank	10-4-68	3.5×10^{-8}
Primary Shield Tank	10-11-68	3.5×10^{-8}
Primary Shield Tank	10-18-68	1.0×10^{-7}
Primary Shield Tank	10-25-68	3.2×10^{-8}

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	90**
Carbon Monoxide	Below 10	Below 10
Carbon Dioxide	20	30
Hydrogen	Below 4***	Below 2.5
Helium	Below 4	Below 4
Methane	Below 10	Below 10
Nitrous Oxide	Not Measured	Not Measured
Argon	Remainder	2300
Nitrogen	1000	Remainder
Dew Point	Not Measured	- 45 F

* Sample date was 10-25-68

** Technical Specifications state 1000 ppm maximum

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

B. TEMPERATURE OF PRIMARY SYSTEM SODIUM INCREASED TO 440 F FOR THERMAL SHOCK TEST OF NO. 2 STEAM GENERATOR

The temperature of the primary sodium system had been maintained in the range of 350 F to 375 F during the fuel handling and steam generator test programs in September (see III. G of APDA-CFE-25) and continued through the first three weeks in October. The primary system temperature was increased from 350 F to 440 F during the last week in October to provide a high secondary sodium temperature in the No. 2 steam generator as an initial condition for a thermal shock test of this unit (see Sec. III. C. 1). Heat transfer from the primary system to the No. 2 secondary loop, and hence the No. 2 steam generator, is achieved in the No. 2 intermediate heat exchanger (IHX). The primary system temperature increase was started on October 23 and an equilibrium temperature of 440 F with secondary sodium and water flow through the No. 2 steam generator was reached by about the end of the month.

The three primary sodium pumps were operated simultaneously and at refueling speed to maintain the primary sodium flow rate at less than the maximum 7×10^6 lb/hr specified in the Plant Technical Specifications for the condition with the holddown mechanism in the UP position. The holddown mechanism, which had been raised to the full UP position prior to the extensive fuel handling program in September, had been maintained in that position pending inspection of the HDM fingers and the complete replacement of the fuel and inner radial blanket subassemblies with new subassemblies. Because of the low pump flow, the pumping power was not sufficient to raise the primary system temperature to 440 F. Additional heat input was provided by the operation of external heaters on the reactor vessel and on the primary pipe and components, and by operation of the No. 2 secondary pump at high speed.

A sudden and abnormal increase in flow occurred in the No. 1 primary loop on October 8. The No. 1 pump liquid rheostat control unit was inspected to determine the cause, but nothing unusual was found. The total indicated flow did not exceed the maximum specified flow of 7×10^6 lb/hr. The flow meter recorder chart for the 14-inch sodium inlet line of the No. 3 primary loop continued to show an undesirable wide trace. This condition, which has been occurring during the circulation of sodium in the primary system, is described on page 20 of APDA-CFE-1.

There was no cold-trapping of primary sodium in October. The No. 1 secondary loop was drained of sodium during October for the replacement of manifold gaskets and the investigation of three tube leaks (see III. G of APDA-CFE-22). The No. 2 secondary loop was drained of sodium for most of the month for replacement of manifold gaskets on the No. 2 steam generator; the loop was then filled with sodium in preparation for the thermal shock tests of No. 2 steam generator, which were performed at the end of the month. Sodium in the No. 2 secondary loop was cold-trapped for 14 hours in October, and

plugging temperature was reduced from 340 F to less than 225 F. The No. 3 secondary loop was drained of sodium early in the month, after the preliminary flow test on No. 3 steam generator (see Sec. III. C. 2) for replacement of manifold gaskets. There was no cold-trapping of sodium in the No. 1 and No. 3 secondary loops in October.

C. STEAM GENERATOR TESTS

1. Thermal Shock Tests of No. 2 Steam Generator

Three successive thermal shock tests were performed on the No. 2 steam generator to verify the integrity of the unit under conditions of severe and rapidly occurring thermal gradients between the water and sodium spaces. In the test, water at 735 psig to 920 psig pressure was circulated through the tubes and sodium was circulated on the shell side. The steam generator was held at a 430 F isothermal temperature prior to the test. The steam generator was then isolated from the feedwater system and the water side of the unit was quickly vented to the atmosphere. Figure 18 is a flow diagram of the thermal shock test.

The initial conditions for the thermal shock tests are as follows:

TABLE 2 - INITIAL CONDITIONS FOR THERMAL SHOCK TESTS

	<u>Test No. 1</u>	<u>Test No. 2</u>
Water flow, lb/hr	<1000	<1000
Water pressure, psig	735	920
Sodium flow, lb/hr	4×10^6	3.6×10^6
Sodium temperature, inlet to steam generator, F	430	430
Sodium temperature, outlet from steam generator, F	430	430

Data were not obtained for the third thermal shock test.

When the water side of the steam generator was vented to the atmosphere, the resulting reduction in pressure on the water side, measured in the steam outlet line from the unit, indicated a decrease in saturation temperature of 217 F in 58 and 68 seconds, as shown in Figure 19. The temperature in the sodium outlet line from the steam generator increased slightly and then decreased to 377 F at 80 seconds after isolation. Sodium inlet temperature

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FIG. 18 FLOW DIAGRAM FOR THERMAL SHOCK TEST OF NO. 2 STEAM GENERATOR

FIG. 19 DECAY OF PRESSURE AND SATURATION TEMPERATURE DURING SHOCK TEST OF NO. 2 STEAM GENERATOR

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remained constant throughout the thermal shock test. Definitive temperature gradients at the tube sheets, and hence the actual degree of thermal shock in these areas, is unknown because temperature instrumentation specifically for this purpose is not available on No. 2 or No. 3 steam generator. No. 2 secondary pump was automatically stopped at the isolation. Sodium flow decayed to 0 gpm at 90 seconds after isolation.

The steam generator was thermally shocked by the isolation operation which would normally be initiated in the event of a gross tube leak causing a sodium-water reaction in the steam generator. The isolation is initiated in the plant control room by depressing the isolation button which triggers operations to contain and minimize the sodium-water reaction. The steps in the sequence are as follows:

- a. close valves in the water inlet and steam outlet lines to isolate the unit from the external water system
- b. open vent valves to quickly release water from the tubes to atmosphere through a 6-inch-diameter vent pipe
- c. stop the secondary pump to contain the reaction in the steam generator shell.

After each thermal shock test, the water tubes were pressurized with nitrogen at 600 psig and the argon cover gas on the sodium-containing shell side was monitored by a chromatograph for a significant increase in nitrogen content, which would be indicative of tube or tubesheet leaks or leaks in the tube-to-tubesheet welds. The internal bore welds at the water manifold which were repaired a year ago (see APDA-CFE-15, p. 28) were of particular concern. There was no significant increase in nitrogen or hydrogen content in the cover gas observed after the thermal shock tests, as shown in Table 3, indicating that no tube-to-shell leak had developed.

TABLE 3 - NITROGEN AND HYDROGEN CONTENT
IN NO. 2 STEAM GENERATOR COVER
GAS AFTER THERMAL SHOCK TEST

	<u>Chromatograph Readings</u>	
	<u>N₂, ppm</u>	<u>H₂, ppm</u>
Before first thermal shock test	{ 8 100	71 76
After first thermal shock test	{ 11 95 90	42 100 94
After third thermal shock test	90	40

2. Preliminary Hydraulic Test of No. 3 Steam Generator

A preliminary hydraulic test was performed on the No. 3 steam generator prior to venting water and draining sodium from the unit for replacement of manifold cover gaskets (see Sec. V.A.). The purpose of the test was to determine the new flow-pressure drop characteristics of the steam generator due to the installation of the flow orifices in the 1200 water tubes (see APDA-CFE-21, p. 44). It has been predicted by calculation that with the flow restrictions added, the pressure drop between the water and steam manifolds should be approximately 35 psi at 200 Mwt. It is known that the pressure drop before the orifice installation has been negative at low loads, resulting in flow instabilities.

In a series of test runs with inlet feedwater at temperature and pressure conditions of 350 F and 905 psig to 920 psig, flows in the range 56,000 to 166,000 lb/hr (the flow for 200 Mw conditions is 212,000 lb/hr) were obtained. Precision instrumentation for measuring pressure drop across the steam generator was not available. The indications were that the measured pressure drops obtained from pressure gages installed at the inlet and outlet of the steam generator were less than those predicted by calculation as shown in Table 4. It is intended to repeat the hydraulic test of No. 3 steam generator after the manifold gaskets are replaced and the integrity of the unit confirmed by thermal shock tests.

TABLE 4 - PRESSURE DROP ACROSS
NO. 3 STEAM GENERATOR WITH
ORIFICES INSTALLED IN WATER TUBES

<u>Water Flow, lb/hr</u>	<u>Measured ΔP, psi</u>	<u>Predicted ΔP, psi</u>
56,000	5	3
75,000	5	6
129,000	8	15
150,000	12	18
166,000	15	23

D. STEAM CLEANING CORE SUBASSEMBLY M118

Core subassembly M118, transferred from the reactor to the transfer tank in the fuel and repair building in September, along with 31 IRB and 2 other core subassemblies,* was removed from the transfer tank, steam cleaned, and deposited in the cut-up pool. This subassembly was left in the transfer tank until the completion of the other transfer and steam cleaning operations because it was stuck in a standard diameter pot. It was distorted;

* APDA-CFE-25, page 12

and when lowered into a pot by the OHM, it jammed in the pot. Special handling to remove it from the pot for steam cleaning was required, since the gripper in the steam cleaning machine could not lift the subassembly without lifting the pot as well.

M118 was lifted from the normal-diameter pot by means of a manual gripper operated through an outer pipe which rested on the pot flange to serve as a holddown device. Figure 20 shows the holddown pipe and Figure 21 the plastic bag sealing the pipe to the floor valve and the gripper rod to the pipe. The lifting force required to extract the subassembly was about 160 pounds greater than the weight of the subassembly. The weight of the holddown pipe was sufficient to withstand the total lifting force. M118 was deposited in a large-diameter pot in the transfer tank and then removed for steam cleaning and deposited in the cut-up pool. The steam cleaning operation proceeded satisfactorily.

The subassembly was visually examined in the cut-up pool by means of underwater television and a monocular for close up viewing. There was considerable distortion in one direction. The upper part of the subassembly was bowed, starting at the spacer pads. Score marks were observed on one corner of the wrapper can from the spacer pads to the top, possibly the result of being lifted out of the normal-diameter pot. It is intended to measure the amount of the subassembly bow in the decay pool.

**FIG. 20 HOLDDOWN PIPE TO RESTRAIN POT FOR
REMOVAL OF WEDGED CORE SUBASSEMBLY
M118 IN TRANSFER TANK**

025368

**FIG. 21 GRIPPER ROD AND HOLDDOWN
PIPE SEALED TO FLOOR VALVE**

025468

IV. SPECIAL INVESTIGATIONS

A. MOCK-UP DEMONSTRATION OF HOLDDOWN INSPECTION

It was demonstrated in photographing a mirror image of a mock-up of the holddown fingers that satisfactory image resolution could be obtained. In an earlier mock-up test (see APDA-CFE-24, p. 29), it was determined that the arrangement of four 1500-watt lamps clustered around the mirror support post in the subassembly lattice cavity provided sufficient lighting for inspection of the holddown fingers. The difficulties encountered in attempting to view and photograph the holddown fingers in the reactor are described in APDA-CFE-19.

The same lattice cavity mock-up was used in this test (see Figure 22). A realistic replica of the holddown fingers, shown in Figure 23, which also includes the lattice cavity with the mirror inserted, replaces the cardboard mock-up used in the earlier test. The lattice cavity and the mock-up holddown fingers reproduce the reflective conditions in the reactor.

The mirror images of the holddown fingers and the upper guide tubes shown in Figures 24 and 25 are representative of the photography in the mock-up. The image resolution is good; image resolution of 1/16 inch is considered the upper limit for determination of holddown finger condition. Note the focusing conditions in Figures 24 and 25; in Figure 24 the foreground image is sharp and the background is out of focus and in Figure 23, image clarity in the foreground is sacrificed in the interest of obtaining background detail.

Additional mock-up testing is planned to check out possible schemes for tilting and revolving the mirror in place in the lattice cavity. It is expected that manipulating the mirror will be done by either mechanical push-pull cables or by a motor drive. After the mirror post is inserted and the rotating plug revolved to position the holddown fingers over the mirror, the electrical lamp leads and the push-pull cables or the electrical motor leads contained in a duct would lie along the tops of subassembly handling heads.

B. REMOVAL OF SWEEP MECHANISM

The sweep mechanism was removed from the reactor to permit insertion of viewing equipment and tools through the penetration in the rotating shield plug vacated by the sweep. The removal was accomplished by use of an unshielded container in which many of the removal operations were carried out from remote locations. Access through this penetration is required for

**FIG. 22 MOCK-UP OF CAVITY IN SUBASSEMBLY
LATTICE FOR INSPECTION OF
HOLDDOWN FINGERS**

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**FIG. 23 MOCK-UP OF HOLDDOWN
FINGERS**

Mirror

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**FIG. 24 MIRROR IMAGE OF MOCK-UP HOLDDOWN
FINGERS — FOREGROUND IN FOCUS**

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**FIG. 25 MIRROR IMAGE OF MOCK-UP HOLDDOWN
FINGERS – BACKGROUND IN FOCUS**

viewing and removal of the zirconium segment from the north sector of the conical flow guide in the inlet plenum. The sweep was stored temporarily in a decay tank after removal from the reactor.

1. Tests of Equipment and Controls

The cask and associated hardware and mechanisms for removing the sweep from the reactor and temporarily storing it in a decay tank are described and illustrated in Section IV.A of APDA-CFE-25. This equipment, along with remote control systems, was assembled on the plant site and subjected to tests in the interest of personnel safety and the protection of the sweep mechanism and hardware on top of the rotating plug in the vicinity of the removal operation. Controls designed for operation of the sweep hoist mechanism and for operation of special valves at the bottom of the cask and in the standpipe sealed to the penetration in the rotating plug were set up for actuation at a remote station near the building crane controls in the containment building. Automatic controls provide for stopping the hoist motor in event of cable overload (at 2000 pounds the sweep weighs about 1600 pounds), slack wire, hoist in UP position, and when the sweep weight is automatically transferred to the cask structure.

Figure 26 shows the bottom of the cask and its special sealing valve; Figure 27 shows the standpipe sealed at the top of the rotating plug; and Figure 28 shows the cask being lowered to seal with the standpipe. The conical bottom section of the cask seals against a Viton ring seal in the standpipe with about 600 pounds of the cask weight bearing on the ring seal. A switch mechanism which senses cask position in the standpipe actuates a signal indicating when the building crane supporting the cask should be stopped. The building crane retains the balance of the weight of the cask structure. Figure 29 shows the cask sealed to the standpipe.

Prior to its use, all electrical connections were checked and controls tested by operating the equipment using dummy loads. The sweep lifting cable, consisting of 1/4-inch nylon-covered aircraft cable with a specified 4500-pound yield strength and a 7400-pound breaking point, was operated with a 2600-pound load. In addition, four 10-foot sections from the same cable reel were destructively tested to verify the 7400-pound breaking point. A leak rate test of the cask verified its satisfactory integrity, revealing a leak rate of 0.1 cu ft/hr from a positive argon pressure of 10 inches water column, the intended atmosphere during storage of the sweep.

2. Removal and Storage

Before the cask structure was sealed to the standpipe, a latching adapter was screwed on the top of the sweep column at the top of the rotating plug, providing the means for latching the hoist gripper on the sweep. Before

**FIG. 26 SEAL VALVE AT BOTTOM OF
SPECIAL CASK FOR REMOVAL
OF SWEEP MECHANISM FROM
REACTOR**

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**FIG. 27 SWEEP REMOVAL STANDPIPE
SEALED TO PENETRATION
THROUGH ROTATING PLUG**

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**FIG. 28 SWEEP-REMOVAL CASK BEING
LOWERED ON STANDPIPE**

026168

**FIG. 29 SWEEP-REMOVAL CASK
SEALED TO REACTOR**

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the cask was opened to the reactor atmosphere, it was evacuated and purged with argon, reducing the oxygen content to 45 ppm.

As the sweep was raised, load indications on the load-sensing device on the building crane cable supporting the cask were evidence that an accumulation of sodium oxide deposits on the sweep column was scraping against the inside wall of the rotating shield plug. The sweep was alternately lowered and raised several times; and as it was being inched up in this manner, the hoist drive motor tripped out on overload several times. When the top of the sweep was about 3 feet into the cask, the overload trip was reset for 4300 pounds, well within the 7400-pound breaking load.

The hoist motor drive again tripped out on overload when the top of the sweep was 15 feet into the cask and about 10 feet below its full UP position in the cask. Further upward movement of the sweep column was prevented by an interference determined by investigation to be between the folded sweep arm and a bronze sleeve bushing at the top of the penetration through the rotating plug. This bushing had been installed at an earlier date as a radial bearing for manual rotation of the sweep. When the motorized drive was installed, it was no longer needed. It was necessary to redeposit the sweep in the reactor and place the cask structure in the decay tank to obtain access to the bushing to remove it. Other hardware at the top of the rotating plug was modified to prevent further obstruction. The cask structure and associated components were reinstalled and the sweep withdrawn into the cask. There were two instances of interference which tripped out the hoist motor drive on overload during this second withdrawal but these obstructions were overcome by lowering the sweep and moving the cask slightly.

The cask structure was deposited in a decay tank for storage, and a seal cover installed over the hoist mechanisms. A constant argon gas flow to the cask and the hoist compartment was established to maintain a positive pressure of 10 inches water column in these spaces.

3. Radiation Monitoring

A radiation monitor was fixed to the 5-foot-high standpipe at about 2 feet above the deck of the rotating plug to give indications of radiation levels as the sweep column was raised from the reactor into the cask. The first measurement above background was 0.5 mr/hr observed when the top of the sweep column had been lifted about 4 feet into the cask. When the bearing bushing at the top of the plug penetration prevented further upward movement of the sweep column and the top of the column was 15 feet into the cask, the observed radiation level was 4 r/hr.

The radiation level at the bottom of a bearing pipe which enclosed the interfering bushing was 80 mr/hr. The bearing pipe was temporarily

removed for elimination of the bushing. When this was done, particulate contamination of a level of 3000 counts per minute (cpm) was released in the immediate area on the rotating plug deck. Further work was discontinued while the area was cleaned to reduce the contamination level to less than 100 cpm.

With the sweep in the near fully UP position in the cask, the radiation level at the surface of the cask was 14 r/hr, the highest level observed during the operation. At this time the radiation level at the personnel air lock was 3 mr/hr.

After the sweep was placed in storage, a radiation level of 500 mr/hr was measured at the outer wall of the decay tank 8 feet above the floor of the below-floor area. At the base of the below-floor personnel entry ladder, the radiation level was 2 mr/hr, which is safe for personnel passage below-floor for access to the 14-inch pipe elbow of the No. 1 primary loop.

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V. MAINTENANCE

A. REPLACING MANIFOLD GASKETS ON STEAM GENERATORS

The cover gaskets for the water and steam manifolds of the No. 2 steam generator were replaced in October, and the replacement of gaskets on No. 3 steam generator was near completion at the end of the month. It is intended to replace manifold gaskets on No. 1 steam generator also, following investigation of the tube leaks (see APDA-CFE-25, p.22). As reported in APDA-CFE-25, p.21, manifold-to-atmosphere leakage was observed on all units during test operation in September. The leakage continued despite increasing the tightening torque on the cover plate bolts. It was found that the gaskets in the No. 1 steam generator and the replacement gaskets were not annealed as specified. The replacement gaskets were annealed before being installed.

B. LOCATING TUBE LEAKS IN NO.1 STEAM GENERATOR

Bubble tests conducted on No. 1 steam generator in September revealed leaks in three tubes. Two of the leaks appeared to be in the internal bore welds at the water tubesheet as evidenced by bubbles rising from between the tube walls and the orifice sleeve inserts. A borescope inspection of these suspected welds after water was drained from the steam generator and the sleeve inserts removed, showed porosity of the welds. The third leak appeared to be somewhere in the convolution region of the tubes, inasmuch as gas bubbles appeared only in the steam manifold.

A special technique was employed to fix the location of the third leak: It was reasoned that gas bubbles released to the atmosphere through a water-filled tube would produce a sound pattern differing from that of a steady unobstructed flow of gas and change in sound pattern, as water level in the steam generator tube is varied, could fix the location of the leak with respect to the known water level.

With the steam generator drained of sodium, a 35 psig argon gas pressure was established in the steam generator shell and a throttled needle valve was connected through a short length of plastic tubing to the leaking tube at the steam manifold. A sonic detector held in contact with the needle valve to pick up the sounds of the escaping gas transmitted the sound pattern to a frequency spectrum analyzer through an amplifier. The needle valve was well insulated to minimize the effects of external noises. The water level in the tube was lowered by syphoning from the water manifold end and, as the level dropped below the leak, the sonic detector system distinguished between the intermittent bursts of gas bubbles and the steady flow of the

unobstructed gas. This technique was applied several times, raising and lowering the water level in the tube. By measuring the water level, it was established that the leak was 10 to 10 1/2 feet below the steam tube sheet. The cause and exact location of the leak, whether at a bend, in a straight section, or at a support location, remains unknown.

All three leaking tubes were plug-welded.

C. FAULTY LEAK DETECTOR IN PRIMARY SODIUM
BELLOWS VALVE 502-1

The cold trap room in the primary sodium service building was penetrated for an investigation of a possible leak in the seal bellows of valve V502-1 in the discharge line from the No. 1 storage tank (see APDA-CFE-21, Fig. 17). At the time of filling the primary system in June, a signal from the valve bellows leak detector was obtained. Examination of an inspection port revealed that there had been no sodium leakage through the bellows; the leak detector was found to be shorted, giving a false indication of a leak. The faulty leak detector was replaced. The seal bellows for this valve had leaked previously and, as a result, was replaced in April (APDA-CFE-21, p. 39).

The recirculating nitrogen system was purged with air and an air atmosphere established in the cold trap room before entering the room. The sodium lines on both sides of the valve were frozen for maintenance of the valve. The activity in the region of the valve was 3 mr/hr. Maintenance personnel spent no longer than a total of 1/2 hour investigating the trouble and replacing the leak detector. The cold trap room was resealed after the investigation and the nitrogen atmosphere reestablished.

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