ENRICO FERMI ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT DECEMBER 1968

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224

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TABLE OF CONTENTS

<u>Section</u>	<u>Page</u>
LIST OF ILLUSTRATIONS	v
PREFACE	vii
I. CURRENT EXPERIENCE SUMMARY	3
II. INVESTIGATION OF ZIRCONIUM SEGMENTS	5
A. BACKGROUND	5
B. DEMONSTRATION REMOVAL OF SECOND LOOSE SEGMENT IN MOCK-UP	6
C. PENETRATING THE PRIMARY PIPE ELBOW	6
D. REMOVAL OF THE FIVE REMAINING SEGMENTS FROM REACTOR	7
E. SCREW SHANKS WELDED IN CONICAL FLOW GUIDE	13
F. SCREW REMOVED FROM SOUTH SECTOR OF CONICAL FLOW GUIDE	15
G. DEVELOPMENT OF VACUUM CLEANING SYSTEM	15
1. Test of Vacuum Device in Mock-Up	15
2. Test of Waste Gas System as Vacuum Source	20
III. OPERATIONS ..	23
A. GENERAL	23
1. Primary System	23
2. Secondary System and Steam Generators	23
B. TESTS OF THE NO. 3 STEAM GENERATOR	24
1. Thermal Shock Test	24
2. Hydraulic Test	26
C. GAS SYSTEMS PERFORMANCE	26
1. Primary System Gas Activity	26
2. Primary System Cover Gas Activity	27
IV. SPECIAL INVESTIGATIONS	29
A. PRIMARY PIPE ELBOW WELDS RADIOGRAPHED	29
B. INSPECTION OF DAMAGED SUBASSEMBLY M007	29
V. MAINTENANCE	33
A. REPAIR OF PLUGGING INDICATOR LINE	33
B. PIPE REPLACEMENT IN WASTE GAS SYSTEM	35
C. HEATER EXPERIENCES	35

LIST OF ILLUSTRATIONS

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
1.	Primary Elbow Patch Partially Cut	8
2.	Glove Box Erected at Primary Elbow	8
3.	Chiseling Patch at Primary Elbow	8
4.	Penetration Through Primary Elbow	9
5.	Primary Elbow Patch with Locating Bracket	9
6.	Cover Plate over Elbow Penetration	9
7.	Control for Object-Retrieval Device	10
8.	Inserting Object-Retrieval Device Through Glove Box	10
9.	Operating Spine-Type Manipulator	10
10.	Plan of Core Inlet Plenum	12
11.	No. 4 Segment (NW Sector) Removed from Reactor	14
12.	Second Detached Segment Removed from Reactor	14
13.	Second Detached Segment, Apex Down	14
14.	Closeup of Second Detached Segment	14
15.	Screw Shank Before Welding	16
16.	Screw Shank After Welding	17
17.	Loose Screw Before Removal	18
18.	Cup-Like Gripper for Spine-Type Manipulator	19
19.	Loose Screw Removed from Reactor	19
20.	Vacuum Cleaner-Separator Disassembled	21

LIST OF ILLUSTRATIONS (Cont'd)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
21.	Another View of Vacuum Cleaner-Separator	21
22.	Vacuum Hose in Mock-Up of Inlet Plenum	21
23.	Decay of Pressure and Saturation Temperature of No. 3 Steam Generator	25
24.	Section of Core Lattice on October 5, 1966 Showing Area of Damaged Fuel	30
25.	Schematic of Plugging Indicator System for Transfer Tank	34

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 this report.

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

The primary pipe elbow in the 14-inch sodium inlet line was penetrated early in December for insertion of the object-retrieval device into the reactor core inlet plenum to remove the five remaining zirconium segments from the reactor. The spine-type manipulator inserted through the holes in the core support plates picked up the loose segments from the floor of the inlet plenum and transferred them to the gripper of the object-retrieval device for removal. Four of the segments were gripped at or near the apex by the retrieval gripper for withdrawal, a procedure demonstrated in mock-up testing to be the most effective for unobstructed passage of the full-width segment through the inlet pipe. The second loose segment which was flattened and found loosely fastened to the underside of the lower support plate after the reactor vessel drain in November, was gripped in the manner demonstrated to be most effective in a special mock-up removal test using a model of this segment. The last two of the five segments were removed from the reactor on December 16.

The arc-melt tool was reinserted through the support plates into the south side of the inlet plenum to securely weld in place the zirconium screw shanks remaining in the stainless steel cone after screwhead melting. The lower left screw in the south sector, where the segment was found missing in November, was discovered to be loose. It was unscrewed from its hole by rubbing with the arc-melt tool and removed from the reactor by the spine-type manipulator equipped with cup-like gripper jaws. It is hoped that a match-up of tackwelds on the screwhead and on one of the two loose segments can identify the loose plates with their respective sectors of origin. It is expected that welding of the remaining screw shanks will be completed in January.

Assembly of the vacuum cleaner designed for insertion through the 14-inch pipe to remove debris from the inlet plenum was completed. The results of mock-up tests of the device indicated that considerable improvement in control is required. It is believed that a control cable of different design will provide the desired improvement. A second test of the waste gas system confirmed its adequacy as a vacuum source for cleaning the inlet plenum.

The No. 3 steam generator was subjected to thermal shock tests, and subsequent pressure tests confirmed the continuing integrity of the unit under conditions of rapidly occurring thermal gradients between the water and sodium spaces. The results of a hydraulic test of the No. 3 steam generator indicated that the ball-tube orifices had reseated properly after being lifted

by the pressure differential caused by the water flashing into steam upon venting of water header to atmosphere in the thermal shock tests.

The butt welds at both ends of the 14-inch primary pipe elbow penetrated for segment removal were X-rayed, and examination of radiographs showed that no weld deterioration had occurred during the previous 8 years of sodium service. The removal of insulation and heaters in this area and the penetration through the secondary containment provided the opportunity to check the condition of typical pipe welds in the primary system.

II. INVESTIGATION OF ZIRCONIUM SEGMENTS

A. BACKGROUND

The deformed sheet of zirconium removed from the reactor in March 1968 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.

Tools were developed for insertion through access ports in the rotating shield plug and holes in the support plates to melt the screwheads fastening the remaining segments to the flow guide, and to dislodge the segments from the flow guide, if necessary. Improvements were incorporated in tools previously used to handle the first segment in the inlet plenum and to remove it through the 14-inch sodium inlet line. An extensive program in the reactor mock-up demonstrated the capability of the special tools to free the remaining segments from the flow guide, dislodge them, and then remove them through the 14-inch inlet line.

The reactor vessel was drained of sodium in November, and the sodium level in the No. 1 primary loop was lowered in preparation for removal of the remaining zirconium segments. The sodium remaining in the riser pipe below the 14-inch pipe elbow to be penetrated was cooled, and a patch was removed from the secondary containment elbow.

The zirconium segment originally installed on the south sector of the flow guide was missing from that position. The segment presumed to be from this south sector was found in a bent and flattened condition in the northeast corner of the core flow inlet chamber, in a position where it appeared to be fastened to the underside of the lower support plate. The segment fell to the floor of the inlet plenum when it was inadvertently touched by viewing equipment inserted into the reactor. It is felt that the segment was not up against the support plate during the inspection of the inlet plenum following the removal of the first segment, although this cannot be stated with certainty. It is hoped that evidence can be obtained to identify the two loose segments with their original positions on the conical flow guide, and to indicate which of the two, or if both, caused the subassembly flow blockage.

All fastening screwheads were melted to free the remaining zirconium segments from the flow guide surface; and by means of a dislodging tool, the segments were pushed off the flow guide onto the plenum floor. It is intended to remove a patch from the primary pipe elbow and to insert the special

devices into the reactor for removal of the remaining segments through the 14-inch inlet line. Previous information on the investigation of the zirconium segments is given in APDA-CFE-20 through 27.

B. DEMONSTRATION OF REMOVAL OF SECOND LOOSE SEGMENT IN MOCK-UP

Part of the plan for removal of the remaining segments in the reactor inlet plenum is for the gripper of the object-retrieval device to grasp the segment at the apex so that the long sides of the segment may be aimed into the 14-inch pipe nozzle for unobstructed passage past the 10-inch flow orifice and through the inlet line. This method for segment removal was demonstrated to be effective in mock-up testing.

The second loose segment discovered last month posed a problem, however, because its apex was found to be bent back, and the preformed bent sides appeared to be flattened. The dimensions of the segment appeared to be sufficiently great that it might have to be forced through the 10-inch inlet orifice for withdrawal. A model of the misshapen segment was fabricated from knowledge of its present shape obtained from viewing and photography in the reactor, and the model was used in the reactor mock-up to determine the optimum method of grasping it for removal through the 14-inch inlet pipe. Tests in the mock-up indicated that this segment should be grasped at the flattened side near the lower left screw hole. It was shown that the segment is thus oriented for removal so that its least dimension is maintained perpendicular to the axis of the pipe for maximum clearance at the 10-inch orifice in the pipe nozzle which is the most restrictive part of the inlet pipe.

C. PENETRATING THE PRIMARY PIPE ELBOW

In November, a 16-inch by 14-inch ellipsoidal patch was removed from the secondary containment elbow for the 14-inch sodium inlet line to the reactor core inlet plenum, exposing the primary elbow to be penetrated. The location of these elbows with respect to the inlet plenum is shown in Figure 1 of APDA-CFE-20. The secondary containment elbow patch was the same one removed for withdrawal of the first zirconium segment in March, 1968. The remaining five segments, loose on the floor of the inlet plenum after dislodging from the conical flow guide in November, were in position for removal through the 14-inch line.

The primary pipe elbow was penetrated by cutting a 6-inch by 10-1/2-inch patch which included the existing patch (removed for withdrawal of the first loose segment in March 1968) and an additional 3-1/2 inches of metal at the top to accommodate the full-width segments. To facilitate the handling of tools, the initial cutting of the patch was done with the primary elbow exposed to the air atmosphere in the below-floor area of the containment

building. The final removal of the patch was accomplished in a glove box sealed around the primary elbow penetration to maintain the primary system argon atmosphere.

The procedure for removing the patch was essentially the same as that for the patch removal in March.* A small pinhole penetration appeared during the formation of the U-type weld prep prepared for rewelding the patch after segment removal, and this was immediately sealed with tape. When the weld prep was completed, the entire profile of the patch was covered with tape, pending installation of the glove box and final chiseling to penetrate the elbow. The maximum measured radiation during the initial cutting was 1 to 2 mr/hr. Figure 1 shows the patch profile sealed with tape after the initial cut and weld prep. The cross brace welded to the patch was installed to prevent the patch from falling inside the pipe and to locate the patch for rewelding after segment removal. The glove box supported by the containment elbow is shown in Figure 2 and Figure 3, where the last 1/16 inch of metal is being penetrated by chisel to free the patch. When the elbow was penetrated (Figure 4), the measured radiation level in the glove box was in the range of 10 to 20 mr/hr, gamma plus beta. Figure 5 shows the patch removed and Figure 6 the cover plate for covering the elbow penetration during unattended periods between segment removal operations in the inlet plenum. The inside surface of the elbow appeared to be clean and free of sodium.

D. REMOVAL OF THE FIVE REMAINING SEGMENTS FROM THE REACTOR

1. Removal

The five remaining zirconium segments were removed from the reactor core inlet plenum during the first half of December. Each segment was handled in the inlet plenum by the spine-type manipulator inserted with the viewing equipment through core support plate holes, and transferred to the object retrieval device for removal through the 14-inch inlet line. Figure 7 is a composite photograph showing the control end of the object-retrieval device in the below-floor, tent-enclosed work area and the control cable extending through the glove box containing the penetration in the 14-inch primary pipe elbow. The flexible retrieval hose is shown being inserted through the glove box in Figure 8, and in Figure 9 movements of the spine-type manipulator are being accomplished with the control toggle switch while viewing through a borescope at the top of the rotating shield plug.

Removal operations were first carried out in the north half of the inlet plenum for the removal of segments No. 2, 4, and 5, lying loose

*APDA-CFE-20, Page 6

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**FIG. 1 PRIMARY ELBOW PATCH
PARTIALLY CUT**

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**FIG. 2 GLOVE BOX ERECTED AT
PRIMARY ELBOW**

072169

FIG. 3 CHISELING PATCH AT PRIMARY ELBOW

072269

**FIG. 4 PENETRATION THROUGH
PRIMARY ELBOW**

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**FIG. 5 PRIMARY ELBOW PATCH WITH
LOCATING BRACKET**

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**FIG. 6 COVER PLATE OVER ELBOW
PENETRATION**

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FIG. 7 CONTROL FOR OBJECT-RETRIEVAL DEVICE

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**FIG. 8 INSERTING OBJECT-RETRIEVAL
DEVICE THROUGH GLOVE BOX**

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**FIG. 9 OPERATING SPINE-TYPE
MANIPULATOR**

on the plenum floor in that area. For the purpose of convenient identification, the segments are numbered as shown in Figure 10, according to positions of origin. Segment No. 2 is the second loose segment, presumed to have been originally fastened on the south sector of the conical flow guide and recovered in the northeast sector of the inlet plenum. Segment No. 6 is the first loose segment, removed in March 1968 and presumed to have originated on the northeast sector of the flow guide.

Segments No. 4 and 5 were nudged from their respective sectors of origin into the northeast sector for transfer to the object-retrieval device for removal from the reactor. The removal of these two segments proceeded satisfactorily and in the same manner as demonstrated in the mock-up, except that it was necessary for the retrieval gripper to grasp segment No. 5 at the side of the apex because of a minor distortion of the apex caused by the dislodging tool used to shove this segment off the surface of the conical flow guide. The bent segment No. 2 was handled and removed from the inlet plenum in the manner demonstrated in mock-up testing to be the most effective for this segment. When segment No. 2, grasped by the retrieval gripper and jammed into the end of the flexible retrieval hose, approached the No. 1 loop nozzle during withdrawal, it was found necessary to rotate the retrieval hose to line up the least dimension of the segment for unobstructed passage through the inlet line.

The spine-type manipulator and the viewing equipment were inserted through a single 6-inch penetration through the rotating shield plug for all operations in the north half of the inlet plenum. Rotation of the shield plug lined up the tools with the various combinations of support plate holes over the working areas. For transferring the segments to the retrieval gripper, extended out of the flexible retrieval hose into the northeast sector of the inlet plenum, the manipulator was inserted through support plate hole N03-N03, the articulated lamp through N03-N02 and the two borescopes, for vertical and horizontal viewing, through N04-N02 and N04-N03.

The manipulator tool and the viewing equipment were then inserted through another rotating plug penetration for access to the south half of the inlet plenum for handling and removal of segments No. 1 and No. 3. Some minor difficulty was experienced in maneuvering both the spine-type manipulator and the object-retrieval device. When the latter device was pushed through the 14-inch line, it tended to either hang up on the central flow divider at the entrance to the inlet plenum or to inadvertently enter the northeast area. Several attempts were required to maneuver the device into the southeast sector of the plenum. On one occasion, when the spine-type manipulator was releasing the No. 3 segment after moving it into position for transfer to the retrieval gripper, residual tension in the spine disc stack caused the spine gripper to jump, flipping the No. 3 segment out of viewing range. It was necessary to handle the segment again to position it in the desired location for transfer to the retrieval gripper. The tool and viewing

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FIG. 10 PLAN OF CORE INLET PLENUM

equipment positions for transferring the segments to the retrieval gripper in the southeast quadrant were N05-P02, N04-P01, N05-P01 and N04-P02. Segments No. 1 and No. 3 were withdrawn on December 16, completing the removal operation.

2. Radiation Levels

After removal from the reactor, each of the five zirconium segments was sealed in a plastic bag, deposited in a metal container and temporarily stored in the health physics building pending alcohol and ultrasonic cleaning. Immediately after removal, and while still contained in the glove box at the primary pipe elbow, the activity level of each of the segments was measured. The levels measured at the outside surface of the glove box were as follows:

Segment No. 1	150 mr/hr, beta
Segment No. 2	>200 mr/hr, beta
Segment No. 3	120 mr/hr, beta
Segment No. 4	190 mr/hr, beta
Segment No. 5	160 mr/hr, beta

The activity level of the No. 6 segment, removed in March, 1968, was measured as 200 mr/hr, beta, and 15 mr/hr, gamma, measured also at the outside surface of the glove box containing the segment.

Figure 11 shows segment No. 4 before cleaning. This was the first of the segments removed and is typical in appearance of all segments removed except segment No. 2, the bent segment shown in Figures 12, 13 and 14. This segment is considerably flatter than segment No. 6, the other loose segment, and it is, therefore, suspected that segment No. 2 caused the plugging which resulted in fuel melting on October 5, 1966. When segment No. 2 was viewed in the reactor, one-third of the tab which folded over the bottom edge of the conical flow guide surface was observed to be missing. Following removal from the reactor, it was observed that an additional third of the tab was missing.

E. SCREW SHANKS WELDED IN CONICAL FLOW GUIDE

After the removal of the zirconium segments from the reactor, the arc-melt tool was reinserted into the reactor inlet plenum through the core support plate holes to permanently secure the remaining shanks of the zirconium fastening screws in the conical flow guide by welding. The screwheads in all sectors except the south sector (presumed origin of the loose segment No. 2) had been melted in November to free the segments for removal. In December, the screw shanks in the southeast and southwest sectors were welded to the flow guide and two of the screwheads in the south sector were melted and the shanks welded. The remaining screw in the

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**FIG. 11 NO. 4 SEGMENT (NW SECTOR)
REMOVED FROM REACTOR**

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**FIG. 12 SECOND DETACHED SEGMENT
REMOVED FROM REACTOR**

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**FIG. 13 SECOND DETACHED SEGMENT,
APEX DOWN**

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**FIG. 14 CLOSEUP OF SECOND DETACHED
SEGMENT**

south sector was found to be loose enough to permit it to be turned when rubbed with the arc-melt tool. It was, therefore, decided to remove the screw intact from the reactor (see Sec. II. F) if possible, and use it to help identify the two loose segments with their original positions on the conical flow guide.

The screw shank welding process is similar to the arc-melting process except that the applied current is about 700 amps, one-half that for arc melting. Both processes are of the same time duration, about one-half second. Figure 15 shows a typical screw shank after melting of the screwhead and before shank welding. A typical welded shank is shown in Figure 16.

There was very little debris observed on the south side of the conical surface after arc melting and shank welding in that area. A tool developed as a segment-dislodging tool was available and therefore used to scrape the small amount of weld splatter from the south side of the cone onto the plenum floor where it could be picked up by a vacuum cleaning system. The scraper tool, designated as a shover tool at the time of its development and shown in Figure 10 in APDA-CFE-24, was inserted through the core support plates along with the viewing equipment. The tools and viewing equipment will be moved next into the north half of the inlet plenum to weld the screw shanks in that area.

F. SCREW REMOVED FROM SOUTH SECTOR OF CONICAL FLOW GUIDE

The lower left screw on the south sector of the conical flow guide was discovered to be loose in its hole when rubbed by the arc-melt tool during the shank welding process. With continued rubbing motion by the arc-melt tool, the screw was turned 5-1/2 times and backed out of its hole. The screw fell to the plenum floor and was removed from the reactor by the spine-type manipulator fitted with a special gripper device.

Figure 17 is a close-up of the screw still engaged in the cone. The special retrieval gripper with cup-like jaws is shown in Figure 18. Figure 19 shows the screw after removal from the reactor. It is hoped that the tackweld positions on the screw will match mating tackwelds on one of the two loose segments and thus identify the loose segments with their location of origin on the cone.

G. DEVELOPMENT OF VACUUM CLEANING SYSTEM

1. Test of Vacuum Cleaner in Mock-Up

Components of the vacuum cleaner described in Section II. G. 1 of APDA-CFE-27 were assembled, and the device was tested in the reactor

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FIG. 15 SCREW SHANK BEFORE WELDING

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FIG. 16 SCREW SHANK AFTER WELDING

FIG. 17 LOOSE SCREW BEFORE REMOVAL

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**FIG. 18 CUP-LIKE GRIPPER FOR SPINE-TYPE
MANIPULATOR**

073769

**FIG. 19 LOOSE SCREW REMOVED FROM
REACTOR**

mock-up for control and mobility in the inlet plenum and to demonstrate its capability to remove and contain debris. The vacuum cleaner was designed for insertion through the 14-inch inlet line in a 5-inch-ID, flexible steel carrier hose. Figures 20 and 21 show the partially assembled components of the vacuum cleaner-separator, and in Figure 22 the silicone rubber vacuum hose is extended on the floor of the inlet plenum mock-up.

In the mock-up test of the vacuum cleaner using a high capacity commercial unit as the vacuum source, it was found that the cleaner did pick up and retain debris but it appeared to be limited in range of pick-up to a small area immediately in front of the vacuum hose. In tests of ability to cover the plenum area, the vacuum hose could be placed in all areas of the plenum except immediately in front of the inlet nozzle, its entry point, and in the space under the conical flow guide diametrically opposite the entry nozzle. Movement of the vacuum hose in response to rotation of the control cable was highly erratic. The hose lay motionless through several turns of the control cable; and, when it did move, it whipped uncontrollably instead of rolling over smoothly. In addition, with a part of the vacuum hose beyond viewing range, there was the possibility that that section could coil up, unseen by the observer, and make it difficult to extract the device.

The test was terminated pending procurement of a control cable of different design which it is felt will greatly improve the control of the device by eliminating cable windup.

2. Test of Waste Gas System as Vacuum Source

In a preliminary test of the waste gas system as a vacuum source in November, a gas flow of 28 cfm was obtained when the waste gas system was opened to the drained reactor vessel through a restrictive 3/4-inch temporary vacuum line. The reactor cover gas differential pressure was satisfactorily maintained between positive 1/2 and 1 inch of water column gage. In December, after the installation of a 2-inch connecting vacuum line, the flow test was repeated with the 2-inch line opened to reactor building atmosphere, and flow rates in the range of 42 to 50 cfm were obtained. These results confirmed the adequacy of the waste gas system as a vacuum source.

The new 2-inch line was connected to the below-floor waste gas line which discharges to the waste gas collection header shown in Figure 15 of APDA-CFE-22. Before the test, one of the waste gas storage tanks was pumped down to about 27 inches of mercury vacuum to serve as the vacuum source. During the test, the vacuum pump was operated intermittently to maintain the tank at 23 to 27 inches of mercury vacuum. Instantaneous flows in excess of 100 cfm were obtained at first applications of vacuum; however, for continuous operation the flows were in the range of 42 to 50 cfm.

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**FIG. 20 VACUUM CLEANER-SEPARATOR
DISASSEMBLED**

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**FIG. 21 ANOTHER VIEW OF VACUUM
CLEANER-SEPARATOR**

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**FIG. 22 VACUUM HOSE IN MOCK-UP OF
INLET PLENUM**

III. OPERATION

A. GENERAL

1. Primary Sodium System

Throughout December, the reactor vessel was maintained in the drained condition and the sodium level in the No. 1 primary loop was kept at an elevation four feet below the penetration in the 14-inch pipe elbow for the segment removal operations in the core inlet plenum. Sodium in the pump discharge riser pipe below the elbow was frozen to permit working at the elbow. Sodium temperature in the No. 2 and No. 3 primary loops and the remainder of sodium in the No. 1 loop was maintained at about 350 F. The short 30-inch crossover pipe between the pump and IHX tanks in the No. 1 loop was full of sodium, ensuring adequate temperature equalization between the two tanks in consideration of stress conditions at the crossover pipe nozzles.

2. Secondary Sodium System and Steam Generators

Sodium has been drained from the No. 1 secondary sodium loop since September, following the discovery of three water-to-sodium leaks in the No. 1 steam generator (see APDA-CFE-25, page 22). The three water tubes have since been plug-welded. The water manifold cover is presently disassembled from the steam generator pending installation of a copper gasket which is expected to be accomplished shortly. Manifold-to-atmosphere leaks occurred on all steam generators during operation of the units in September and replacement annealed soft iron gaskets in the water manifold joints of the No. 2 and No. 3 steam generators were unsatisfactory in subsequent operations of these units. As a result, copper gaskets are to be installed in the water manifold joints of all steam generators.

Sodium was drained from the No. 2 secondary loop in December in preparation for manifold gasket replacement on the No. 2 steam generator. The No. 2 unit has been subjected to thermal shock tests in October to verify the integrity of the unit.

The No. 3 steam generator underwent thermal shock tests verifying its integrity (see Section III. B. 1) and a subsequent hydraulic flow test was performed to establish that the ball-tube orifices in the water tubes had reseated properly (Section III. B. 2). Sodium was drained from the No. 3 secondary loop in late December for manifold gasket replacement on the No. 3 steam generator. There was no cold trapping of sodium in the secondary sodium loops in December.

B. TESTS OF THE NO. 3 STEAM GENERATOR

1. Thermal Shock Tests

Three successive thermal shock tests were performed on the No. 3 steam generator to verify the integrity of the unit under conditions of rapidly occurring thermal gradients between the water and sodium spaces. The steam generator was thermally shocked by isolation and venting the water side to atmosphere, and checked for integrity in the same manner as the test of the No. 2 steam generator described in APDA-CFE-26, page 24.

The initial conditions for the thermal shock tests are given in the following table.

TABLE I - INITIAL CONDITIONS FOR THERMAL SHOCK TESTS

	<u>Test No. 1</u>	<u>Test No. 2</u>	<u>Test No. 3</u>
Water flow, lb/hr	< 1000	> 1000	< 1000
Water pressure, psia	925	920	920
Sodium flow, lb/hr $\times 10^6$	2.88	2.92	2.84
Sodium temperature, F	← 410 isothermal →		

The No. 3 steam generator was held at a 410 F isothermal temperature prior to each of the tests. It should be noted that the sodium temperature in the No. 3 secondary loop was obtained primarily from heat input from heaters on the No. 3 loop piping and components and with much less heat input from the primary system than in the thermal shock tests of the No. 2 steam generator.

When the steam generator was isolated and the water side vented to atmosphere, the resulting reduction in water side pressure, measured in the steam outlet line from the unit, indicated a decrease in saturation temperature of 197 F in 76 seconds as shown in Figure 23, which shows the pressure and saturation temperature decay in Test No. 1, typical of the results of all three tests. The temperature in the sodium outlet line from the steam generator remained constant for 18 seconds then decreased to 350 F at 210 seconds after isolation. The sodium inlet temperature remained constant throughout the thermal shock test. The No. 3 secondary sodium pump was automatically stopped at the time of isolation and sodium flow dropped to 0.16×10^6 lb/hr in 95 seconds, decreasing more slowly to 0 at 220 seconds after isolation. Definitive temperature gradients at the tube sheets are unknown because temperature instrumentation specifically for this purpose is available only on No. 1 steam generator.

074169

FIG. 23 DECAY OF PRESSURE AND SATURATION TEMPERATURE DURING THERMAL SHOCK TEST OF NO. 3 STEAM GENERATOR

2. Hydraulic Test

A hydraulic test of the No. 3 steam generator was performed in December to check that the ball-tube orifices in the 1200 water tubes had reseated properly after the thermal shock tests. As can be seen in Figure 18 of APDA-CFE-26, the water flow reverses as the steam generator is vented in the isolation operation, and the ball-tube orifices are lifted to increase the water release path. It was felt that results of the hydraulic flow test as compared with results of the November test (see APDA-CFE-27, Sec. III. D) would be indicative of the condition of the orifices. Details of the tube orifices are given in APDA-CFE-12, page 27.

In a series of test runs, inlet feedwater was at a temperature of 342 F and in the pressure range of 925 to 935 psig. Flows were varied between 51,000 lb/hr and 212,000 lb/hr, corresponding to 200 Mw conditions. The measured pressure differentials are in agreement with the results of the November hydraulic test, indicating that most or all of the ball-tube orifices had reseated properly. The orifice condition will be confirmed when the water manifold cover is removed for gasket replacement.

C. GAS SYSTEMS PERFORMANCE

Since the last data were reported (APDA-CFE-27), the following primary system gas activity and gas analysis have been obtained:

1. Primary System Cover Gas Activity

<u>Location</u>	<u>Sample Date</u>	<u>Gross Beta Concentration, $\mu\text{Ci}/\text{Cm}^3$</u>
Reactor Cover Gas	12-6-68	1.4×10^{-6}
Reactor Cover Gas	12-13-68	1.7×10^{-6}
Reactor Cover Gas	12-20-68	1.5×10^{-6}
Reactor Cover Gas	12-27-68	1.8×10^{-6}
Primary Shield Tank	12-13-68	2.7×10^{-8}
Primary Shield Tank	12-20-68	3.5×10^{-8}
Primary Shield Tank	12-27-68	3.8×10^{-8}

2. Primary System Cover Gas Analysis

	Reactor Cover Gas (Argon), <u>ppm by volume*</u>	Primary Shield Tank Atmosphere (Nitrogen), <u>ppm by volume*</u>
Oxygen	Below 25	90**
Carbon Monoxide	Below 10	Below 10
Carbon Dioxide	15	10
Hydrogen	Below 4***	Below 2.5
Helium	Below 4	Below 4
Methane	Below 10	Below 10
Nitrous Oxide	← Not Measured →	
Argon	Remainder	Not Measured
Nitrogen	200	Remainder
Dew Point	Not Measured	-55 F

* Sample date for reactor cover gas and primary shield tank atmosphere was 12-20-68.

** Tehcnical Specifications state 1000 ppm maximum.

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

IV. SPECIAL INVESTIGATIONS

A. PRIMARY PIPE ELBOW WELDS RADIOGRAPHED

The two butt weld joints which connect the 14-inch pipe elbow with the adjoining straight sections of pipe in the sodium inlet line of the No. 1 primary loop were radiographed in December to determine the quality of the welds after exposure to high temperature and radioactive sodium for eight years. The opportunity for this investigation arose when both the secondary containment and the primary pipe elbow were penetrated for removal of the zirconium segments. It was felt that the findings would be indicative of the condition of welds generally in the primary system piping.

The welds were radiographed through a single thickness of pipe using an iridium-192 point source. All but 6 inches of the periphery of the weld was accessible for inspection. The joint with the downstream or horizontal section of pipe penetrating the secondary shield wall is a field weld made at the time of loop erection. This weld was shown to contain no significant voids or other imperfections. The weld joining the primary pipe elbow with the riser pipe from the primary pump discharge is a shop weld. Radiographs of this weld showed some small porosity and inclusions but not outside specifications. The weld was determined to be acceptable in accordance with Section 3 of the ASME Boiler and Pressure Vessel Code (Nuclear Vessels). Comparison with radiographs of the as-built weld indicated that there has been no deterioration of this shop weld.

B. INSPECTION OF DAMAGED SUBASSEMBLY M007

Core subassembly M007 was inspected for distortion in the cut-up pool in the fuel and repair building in the same manner as the inspection of core subassemblies M118 and M087 reported in Section IV. B of APDA-CFE-27. As shown in Figure 24, subassemblies M007, M118, and M087 were adjacent to the cluster of damaged subassemblies M127, M098, M140, and M122 in the core lattice at the time of the incident of October 5, 1966. The three subassemblies had been stored in the reactor vessel transfer rotor and were transferred from the reactor to the fuel and repair building along with 31 inner radial blanket (IRB) subassemblies in September.* The three subassemblies were inspected in the fuel inspection facility just prior to transfer and were found to be distorted.**

* APDA-CFE-25, page 12.

** APDA-CFE-25, page 25, and APDA-CFE-25, page 27.

074569

FIG. 24 SECTION OF CORE LATTICE ON OCTOBER 5, 1966, SHOWING AREA OF DAMAGED FUEL

After being steam cleaned and deposited in the cut-up pool in the fuel and repair building, M007 was checked for distortion by the plumb bob method. It was observed that in one plane of the subassembly wrapper tube, the bow, starting 2 inches above the middle of the spacer pad, was $21/64$ inch at the top of the wrapper tube and $3/8$ inch at the top of the handling head. In the adjacent plane of the wrapper tube, the bow started 3 inches above the middle of the spacer pad and was $1/8$ inch at the top of the handling head.

V. MAINTENANCE

A. REPAIR OF PLUGGING INDICATOR LINE

As reported in APDA-CFE-25, page 20, the plugging temperature of the sodium in the transfer tank in the fuel and repair building increased to 425 F at a system temperature of 500 F during the transfer of subassemblies from the reactor to the transfer tank. With the sodium in that condition, the plugging indicator circuit became plugged and thus the normal means for determining the results of subsequent cold-trapping was not available. Access to the transfer tank room for investigation of the plugged condition of the plugging indicator circuit was delayed until all of the irradiated subassemblies were removed from the transfer tank for steam cleaning and deposit in the cut-up and decay pools.

When the transfer tank room was entered for investigation of the plugged system, there was no evidence of sodium leakage from the plugging indicator system. The measured radiation level at the surface of the electromagnetic (EM) pump in the system was 10 mr/hr. Figure 25 is a schematic diagram of the system. Heat-up of the system and gas blow-through from the test line connected near the orifice plate failed to clear the plug. The section of pipe passing through the EM pump, designed to be somewhat flattened, is the first restriction downstream of the transfer tank; therefore, the plug was believed to be at or a short distance upstream of the pump.

After the system was cooled, the pipe was cut at a location amply upstream of the pump and an oxide plug was found in the suspected area. The pipe was flushed with alcohol, first by an aspirating action, then by pumping. The alcohol was admitted through a 1/4-inch plug-probing tube and drained along with the dislodged oxide through a 3/4-inch drain hose connected to the cut pipe and surrounding the alcohol supply tube. The dislodged oxide and the drained alcohol were collected in a container cooled by dry ice and enclosed in an argon-purged plastic bag. The alcohol flush satisfactorily cleaned out the oxide plug which extended through the flowmeter downstream of the EM pump.

During the alcohol flush, the flattened section of pipe passing through the EM pump was found to be distorted, presumably by a stress condition caused by heat from the pump concentrated in this area of the plugged line, adding to the flow-restrictive condition of the pipe. The pipe between the EM pump and the flowmeter was cut, and the distorted section removed for straightening by hot and cold forming. The pipe section was

074269

FIG. 25 SCHEMATIC OF PLUGGING INDICATOR SYSTEM FOR TRANSFER TANK

reinstalled, and preparations are presently in progress to re-establish sodium flow.

B. PIPE REPLACEMENT IN WASTE GAS SYSTEM

In December, the perforated diffuser pipe which releases waste gas at a low level of activity into the high-volume air stream for discharge up the stack was found to be plugged with mud, believed to have leaked through corroded pipe in the underground line leading to the diffuser pipe. Figure 15 in APDA-CFE-22 shows the location of this area in the waste gas system. That report gives details of the failure and replacement of another section of waste gas pipe in the same area.

Discovery of the plugged condition of the diffuser pipe was the result of a trip-out of the vacuum pump in the waste gas system when pumping down a waste gas storage tank during the test of the vacuum cleaner for the reactor core inlet plenum. It was first confirmed that the cause of the trip-out was excessive back pressure. A gas flow was injected upstream from the pump and it was observed that pressure in excess of 15 psig, the trip-out point, built up in the piping downstream from the pump. The 2-inch diffuser pipe releases waste gas through 30 holes 1/8 inch in diameter; therefore, this pipe was the suspected location of the plug. The welded cap on the diffuser pipe which extends slightly above the air plenum was cut off and water and mud blew out, confirming the existence of the plug in the pipe. Pressure hold tests on individual sections of the waste gas system indicated that leaks existed in the below-ground pipe between the nest of flow control and safety control valves in the waste gas building and the diffuser pipe.

A temporary above-ground pipe was installed to replace the suspected leaking pipe. The replacement of underground lines in the waste gas system with above-ground pipe is expected to be accomplished in the near future.

C. HEATER EXPERIENCES

In mid-1968, all power supply cables between the circuit breaker panels and the terminal boards in the containment building, for induction heater coils on the primary sodium loops were replaced (see APDA-CFE-20, page 41). There had been a general deterioration of the insulation on the No. 8 Butyl-insulated cable due to crowding in the cable trays and up to 50% and higher overload current in nearly half the circuits. The overload conditions, the result of conflict between design philosophy and operating necessity, were relieved by such measures as connecting heater coils in series, and adding more turns to individual coils. The replacement supply cables are type THW PCV No. 6 wire.

The problems continued as failures of primary system heating coils occurred in the below floor area of the containment building. These failures

occurred particularly on the primary sodium pump and IHX tanks near pipe nozzles and at pipe elbows where the heater coils tend to be bunched at the inside elbow radius and hence are denied sufficient cooling. The burned-out coils were replaced and a program of extensive inspection of the No. 10 asbestos-insulated heating coils was started in the below-floor area. Heater wire in a deteriorated yet not a completely failed condition is spotted by the greenish discoloration of the wire insulation resulting from rapid oxidation of the wire. In most instances, only sections of a coil were replaced; in some cases, complete replacement of coils was required. The inspection and replacement program is continuing.

An investigation of the performance of the external resistance heaters on the reactor vessel indicated that there is little or no deterioration of heater circuits. The investigation was a preliminary check of heater performance in which circuit currents were measured and compared with measurements obtained in initial tests soon after construction of the plant.

INDEX

Items Appearing in APDA-CFE-1 through 28
(August 1966 through December 1968)

NOTE: All reference numbers are hyphenated. The digits preceding the hyphen refer to the report number; the digits following the hyphen refer to the page number.

Argon, Plant Consumption, 1-11, 2-18, 3-15, 14-19	Cut-Up Machine, 1-9, 1-13, 4-15 4-18, 5-11, 5-22, 7-13
Bonded Subassemblies (M098 and M127), 5-21, 6-7, 6-16, 6-17, 6-18, 7-15, 9-6, 9-18, 9-20, 11-12, 12-7	Drain, Reactor Vessel Sodium, 6-23, 9-20, 10-29, 11-6, 13-8, 27-20
Cask Car, 1-9, 1-13, 2-23, 3-20, 4-15, 4-18, 6-28, 8-14, 10-10, 24-31	Fill Secondary Sodium Loop, 17-17 18-29, 22-17
Cold Traps and Plug Indicators Cold Trap Experience, 3-16, 6-11, 10-20, 13-8, 17-18, 18-29, 23-16, 23-22, 24-15, 25-11, 25-20, 26-20 Monthly Operating Data, 1-10, 2-17, 3-14, 4-15, 5-12, 6-9, 7-9, 8-9, 9-7, 10-6, 23-22, 24-16, 25-11, 27-19 Plugging Indicator Maintenance, 28-33, Plugging Runs, 10-21, 13-8, 17-18, 23-16, 23-17, 23-22	Fill Primary System, 23-13 Filters, Sodium, 10-19 Fission Products, 2-30, 3-16, 3-18, 5-11, 6-25, 9-23, 20-6 Fixture, Underwater Gaging, 6-16, 10-11, 11-23, 14-28 Flowmeters, 1-20, 1-21, 10-29, 26-20, Freezing and Thawing Sodium Pipe, 18-26
Cold Trap Room Cooling System, 21-41	Fuel Pins Burnup, 1-3, 2-7, 2-26, 3-8 Cladding Test, 5-23 Hydriding, 2-26, 2-28, 6-25, 7-19, 8-17, 9-23, 17-23, 19-25, 22-29 Melting, 12-16, 12-20, 13-16, 13-17, 13-19, 15-21, 16-29 Phase Transformation, 19-25, 22-29
Compressors, Gas, 1-27	
Curves, Daily Operation Sodium Temperature, 1-4, 1-6, 2-9, 3-10, 4-9, 4-11 Sodium Flow, 1-5, 2-10, 3-11, 4-10 Steam Conditions, 1-6, 4-11 Reactor Power, 1-4, 4-9	

Fuel Pins (Continued)
 Size, 2-26, 6-24, 8-17, 10-17,
 16-31
 Upper-End Cap Bulging, 18-39

Fuel Transfer Facility, 10-10,
 19-22, 20-34, 21-38, 22-23

Galling, Metal Surfaces, 6-7, 6-28,
 7-9, 7-24, 9-12, 10-8

Heating, Pipe and Equipment, 2-32,
 10-9, 10-27, 16-37, 17-29, 20-41,
 28-35

Holddown Mechanism, 1-9, 2-12,
 4-15, 4-17, 4-20, 21-11, 23-16

Holddown Finger Inspection, 13-13,
 18-31, 19-6, 19-13, 20-27, 24-29,
 26-29

Hot Trap, Sodium, 10-20

Immersion Tests, Materials in
 Sodium, 10-18, 11-17, 11-20

Inspections, BMI Hot Cell,
 M099: 5-22, 6-24, 7-19, 16-31,
 19-25, 22-29
 M156: 7-22, 8-17, 9-23
 M091: 10-17, 11-15, 19-25, 22-29
 M140: 11-15, 12-16, 18-41, 19-26,
 22-30
 M127: 12-16, 13-16, 14-21, 15-20,
 16-29, 17-23, 18-41
 M098: 12-20, 13-19, 14-21, 15-21,
 16-29, 17-23, 18-41
 M122: 15-24, 16-29, 18-39, 19-26,
 22-29, 23-25
 Summary of Inspection Work, 16-31

Inspection Facility, Exit Port, 5-25,
 8-15, 9-15, 10-16, 12-27, 13-15,
 16-27, 17-21, 18-37, 19-21, 20-33,
 21-37, 22-21, 24-25, 24-26

Intermediate Heat Exchangers, 1-17,
 6-10, 6-11, 14-21

Level Indicators, Sodium, 8-22,
 9-26, 11-8

Machinery Dome, 2-28, 2-23, 8-11

Nitrogen, Plant Consumption, 1-11,
 2-18, 3-15, 11-7, 14-19

Nonfissionable Materials
 Surveillance Program, 20-37, 21-35

Offset Handling Mechanism
 Core Investigation, 2-22, 4-26,
 5-17, 5-18, 6-16, 7-15
 Operating Cycles, 1-9, 2-12, 4-15,
 4-17, 7-18
 Repairs, 1-12, 7-18

Piping
 Sodium, 1-20, 1-26, 4-36, 10-9,
 10-27, 10-29, 11-22, 18-23,
 19-5, 21-27, 23-16, 28-29
 Steam Dump, 1-30, 3-22

Plutonium Production, 5-13

Pool, Decay, 1-29, 8-24

Pool, Cut-Up, 1-29, 5-11

Power Supply, 1-19, 3-39, 6-10, 23-29

Primary Cover Gas
 Activity, 3-16, 3-24, 4-16a, 6-9, 7-9,
 8-10, 9-8, 10-17, 11-20, 14-18,
 16-35, 17-20, 18-29, 19-19, 20-32,
 21-29, 22-19, 23-23, 24-18,
 25-10, 26-19, 27-17, 28-26
 Cover Gas Analyzer, 2-30
 Leakage, 2-27, 11-19
 Monthly Chemical Analysis, 1-10,
 2-17, 2-31, 4-16, 5-13, 6-9,
 7-11, 8-10, 9-7, 10-6, 14-18,
 16-35, 17-20, 18-30, 19-19, 20-32,
 21-29, 22-20, 23-24, 24-18

Primary Cover Gas (Continued)

25-10, 26-19, 27-18, 28-27

Primary Shield Tank

Expansion Seal, 23-19, 24-29,

Leak Test, 6-29, 18-43, 19-27

Monthly Gas Chemical Analysis,

1-10, 2-17, 4-16, 5-13, 6-9,

7-11, 8-10, 9-7, 10-6, 14-18,

16-35, 17-20, 18-30, 19-19,

20-32, 21-29, 22-20, 23-24,

24-18, 25-10, 26-19, 27-18, 28-27

Pressure Control, 9-27

Wall Temperatures, 3-37

Pumps

Overflow, 8-24

Primary, 1-11, 2-19, 3-22, 3-31,

4-17, 5-23, 11-7, 23-16, 24-15,

25-10, 26-20, 27-19

Secondary, 1-11, 2-19, 10-28,

17-18

Radiation Monitoring, 2-22, 3-23

3-24, 3-25, 6-10, 6-24, 7-9, 7-13,

8-22, 8-23, 9-8, 9-12, 10-8, 10-17,

15-20, 17-26, 18-39, 20-6, 20-16,

24-22, 25-19, 26-36, 27-28, 28-7,

28-13

Reactor

Building Decontamination, 12-14,
13-13

Building Leak Tests, 3-39, 7-25,
15-20, 19-28, 20-41

Building Ventilation Equipment,
23-29, 23-31

Core Outlet Temperatures, 1-18,

2-11, 3-23, 3-32, 11-15, 12-25

Fuel Damage, 3-22, 14-6

Inspection of Vessel Interior, 12-12,
13-9, 14-17, 21-11, 27-27

Loading, 1-8, 2-11, 2-12, 3-12,

4-12, 5-8, 6-7, 7-6, 8-6, 9-6,

12-7, 13-6, 13-7, 14-8, 17-16,

18-35, 20-29, 21-28, 22-18, 24-19,

24-21, 25-11, 25-13

Reactor (Continued)

Meltdown Pan, 17-23

Monthly Operating Data, 1-3, 2-7, 3-8

Oscillator Test, 1-15

Power, 1-3, 1-14, 2-7, 3-8, 3-22,

3-23, 3-29, 6-12

Power Coefficient of Reactivity, 1-14

Reactivity Measurements, 2-26,

3-26, 3-28, 3-20

Reactor Pulse and Noise Analyzer,
1-15

Safety System, 1-12, 2-19, 2-25,

3-18, 4-18, 6-10, 11-8, 15-28

Setback Tests, 2-25

Sweep Mechanism, 4-20, 6-17,

9-18, 21-11, 24-16, 25-25, 26-29

Transfer Rotor Movements, 1-9

Vessel Flow, 3-31, 5-13, 9-24,

23-16, 24-15, 25-11, 26-20, 27-19

Vessel Temperature, 1-17, 2-27, 10-9,

23-16, 24-15, 25-10, 26-20, 27-18

Rotating Plug, 1-9, 2-12, 2-20,

4-15, 6-17, 7-18, 9-18, 10-29

Scrams, 1-3, 1-12, 1-18, 2-7, 2-19,

2-25, 3-8, 3-23, 6-12

Shielding, 2-31, 3-37, 3-38, 23-16, 24-15

Shipping Casks, 5-12, 5-27, 7-13,

8-15, 9-8, 9-9, 9-20, 15-21, 24-19,

24-23, 25-15

Sodium

Activity, 1-11, 2-17, 3-15, 3-16,

4-16a, 5-13, 8-10, 14-19, 16-36,

23-22, 25-9

Monthly Chemical Analysis, 1-10,

3-15, 4-16, 5-12, 7-11, 7-12, 8-11,

9-8, 10-7, 14-19, 16-36, 23-23,

24-17, 25-9

Purity, 2-28, 2-29, 5-22, 7-12, 8-18,

9-25, 10-18, 10-19, 10-21

Steam Cleaning Machine, 1-9, 2-23,

3-14, 4-14, 4-15, 4-18, 5-8, 6-28,

7-24, 8-14, 8-23, 24-31, 25-19

Steam Generators

Maintenance, 1-13, 1-13A, 2-23,
6-26, 7-23, 8-20, 9-26, 10-28,
12-27, 13-21, 14-28, 15-28, 16-37,
17-25, 18-43, 19-27, 20-41, 21-44,
22-31, 25-21, 26-39, 27-19, 27-37,
28-23
Operation, 1-13, 1-13A, 1-18,
3-21, 4-20, 4-32, 25-20
Tests, 26-21, 26-25, 27-24, 28-24,
28-26

Subassemblies, Separation of M098
and M127, 12-7

Subassemblies, Steam Cleaning, 25-19,
26-25

Subassembly Flow Blockage
Study of, 21-31, 23-25
Test, 21-23

Subassembly Inspections, 24-25, 25-25,
27-28, 28-29

Subassembly Transfer, Between
Reactor and Transfer Tank
Core, 25-12
IRB, 24-19, 25-12
ORB, 25-12

Subassembly Transfers

M099: 4-14, 4-18, 5-8, 5-11, 5-12
M091: 5-10, 6-25, 8-15, 9-8
M140: 5-17, 7-15, 10-8
M156: 7-12
M127: 12-8
M098: 12-10
M122: 15-21

Tanks, Liquid Waste, 1-29, 5-11

Tank Room, Primary Sodium
Storage, 6-23, 9-21, 11-7

Thermocouples

Core Sodium Outlet, 3-23, 3-32,
11-17

Thermocouples (Continued)

Drain Line, 11-8
PST Wall, 3-37
Secondary Shield Wall, 3-37
Steam Generator Sodium Outlet, 2-21

Transfer Tank, FARB, 1-9, 7-24,
17-26

Transient Temperatures, Sodium
1-18, 4-32

Unidentified Foreign Object Investiga- tion (First Zirconium Segment)

Construction Logs and Memoranda,
18-6
Description, 14-6, 17-6
Discovery, 14-6
Draining and Cooling 14-Inch Inlet
Ling, 18-23
Experiences in Reactor Vessel,
16-5, 17-5
Glove Box at 14-Inch Line Elbow,
18-15, 19-10, 20-6
Identification, 14-11, 18-5
Location, 14-6, 14-9, 14-10, 15-6
Metallurgical Analysis, 20-16
Mock-Up, Reactor Vessel and 14-Inch
Line, 14-15, 14-16, 19-8
Object Maneuvering and Removal
Tools and Devices, 14-11, 14-14,
15-10, 15-14, 15-18, 15-19, 16-14,
16-20, 17-9, 18-19, 19-6, 20-9
Penetrating 14-Inch Elbow, 20-5
Radiation Monitoring, 20-6, 20-16
Removal Methods, 14-12, 15-12,
16-14, 18-15
Removal through 14-Inch Line,
15-13, 15-17, 16-19, 17-12, 18-11,
20-16
Rewelding 14-Inch Elbow, 20-22
Sampling, 16-5, 17-5, 20-16
Vacuum Fusion Analysis, 26-17
Viewing and Photography, 14-11,
15-6, 15-10, 16-5, 16-12, 16-24,
18-6, 19-6, 20-16
Visual Examination, 20-17

Valve Maintenance, 17-26, 21-39,
23-17, 25-29, 26-40, 27-31

Viewing Equipment

Borescope at Loading Tunnel, 5-28,
6-25

Reactor Vessel, 6-17, 8-6, 9-18,
11-6, 11-12, 13-9, 13-21, 14-12,
16-12, 16-24, 18-21, 18-31, 19-6,
19-13, 22-9

Waste Gas Line, 22-31, 28-35

Waste Gas System, Vacuum Source
for Cleaning Inlet Plenum, 26-16,
27-13, 28-20

Waste Liquid Line, 22-33, 23-32

Zirconium Segments Investigation

APDA Engineering Design Group.

Assignment at Fermi, 21-10

Construction Logs and Memoranda,
18-6

Detaching Methods, 19-10, 20-25,
21-9

Detaching Tools, 21-6, 21-9, 22-6,
22-9, 23-5, 24-5, 24-11, 25-6,
26-6, 27-6, 27-12, 28-15

Zirconium Segments Investigation (Continued)

Draining and Cooling 14-Inch Inlet
Line, 27-12, 27-24

Fixing in Place, 21-8

Glove Box at 14-Inch Line Elbow, 28-7
Identification, 18-5

Mock-Up, Reactor Vessel and 14-
Inch Line, 21-10, 22-15, 23-9, 24-9
25-6, 26-6, 26-16, 27-6, 28-6

Penetrating Primary Pipe Elbow, 28-6
Penetrating Secondary Containment,
27-13

Removal of Debris, 22-8

Removal through 14-Inch Line,
21-9, 25-6, 27-6, 28-6, 28-7

Second Detached Segment, 27-7,
28-6, 28-11

Screw Removed from Reactor, 28-15

Segment Maneuvering and Removal

Tools and Devices, 21-10, 23-8,
24-7, 25-7, 26-6, 27-6, 28-6,
28-7, 28-15

Vacuum Cleaner for Inlet Plenum,
26-16, 27-13, 28-15

Viewing and Photography, 22-9

Welding Screwshanks in Conical Flow
Guide, 28-13