ENRICO FERMI ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

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R. H. Costello
Assistant Project Engineer

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APPROVED

Charles E. Branyan

C. E. Branyan
Fermi Project Engineer
APDA

PREPARED BY

Atomic Power Development Associates, Inc.
with the Cooperation of
Power Reactor Development Company
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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 tests that the arc-melt tool can be satisfactorily articulated, and can melt zirconium screwheads in a 350 F argon and sodium vapor environment such as will be experienced in the reactor. The arc-melt tool is to be used to melt the fastening screwheads to free the zirconium segments from the conical flow guide in the reactor inlet plenum.

The new electrohydraulic control system for the spine-type manipulator was assembled and made ready for testing in the reactor mock-up. A new football-shaped housing for the retrieval-device gripper is being constructed of aluminum. It is expected that this new housing, which is to replace a steel structure with ball rollers, will afford better control of the retrieval gripper when it is extended out of the flexible retrieval hose in the inlet plenum to grip a freed segment.

It was demonstrated in the mock-up that an undistorted zirconium segment could be removed through the 14-inch inlet line by means of the object-retrieval device and associated gripper, even though it is slightly wider than the shield ring in the 14-inch line.

Thirty-five core subassemblies were relocated to open up core positions for insertion of tools and viewing equipment for the segment removal program. Most of these core subassemblies were placed in IRB positions for temporary storage. Thirty-one IRBs, most of them displaced by the relocation of core subassemblies, and three core subassemblies that had been found to be grossly distorted were removed from the reactor. All except one of the core subassemblies were steam-cleaned and stored in the cut-up and decay pools.

The steam generators were operationally tested to determine the integrity of the water tubes. Leaks were discovered in No. 1 steam generator in two tube-to-tubesheet welds at the water manifold and in a tube about 10 feet below the steam tubesheet. There was no evidence of tube leaks in the No. 2 and 3 steam generators. Water and steam manifold gaskets will be replaced on all steam generators because of leaks to atmosphere from the cover plate joints.

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 in the sloping sectors of the conical flow guide. Details of the removal of the first segment are given in APDA-CFE-20. The present plan is to remove each of the remaining zirconium segments and withdraw them through the 14-inch sodium inlet pipe of the No. 1 loop in the same manner that the deformed object was withdrawn.

Tools for removing the heads of the zirconium fastening screws to free the segments from the conical flow guide were developed, and on the basis of preliminary tests emphasis was placed on the arc-melt method^{*} for screwhead removal. A full-length arc-melt tool was fabricated and used in the mock-up of the reactor vessel and the simulated top structure of the rotating shield plug to confirm satisfactory tool access to all fastening screw positions and to demonstrate that the tool can melt away screw-heads.

Tests in the reactor vessel mock-up indicated that of the several dislodging tools developed, the tire iron tool^{**} would be the most effective for prying the freed segments loose from the conical flow guide. This tool and any of the other dislodging tools could be used, if necessary, to deform the segments to ensure unobstructed removal through the 14-inch inlet line.

A new electrohydraulic control system was developed for the spine-type manipulator^{***} used in the handling and removal of the first segment and intended to function similarly in the removal of the remaining segments. The gripper for the object-retrieval device^{***} also used in the removal of the first segment was modified to provide greater versatility of the gripper in the inlet plenum. In subsequent tests in the mock-up, it was shown that additional modification is needed to provide better control of gripper movement in the inlet plenum. Previous information on the investigation of the zirconium segments is given in APDA-CFE-20 through 24.

* See Section II of APDA-CFE-21 through 24 for previous information about the arc melt method.

** See APDA-CFE-24, Figures 6, 7 and 8.

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

B. TESTING ARC-MELT TOOL IN SIMULATED REACTOR ATMOSPHERE

It was demonstrated in a test in an atmosphere simulating reactor conditions that the full-length arc-melt tool can be satisfactorily manipulated and can melt away screwheads in the reactor environment. The test was conducted in a chamber containing argon and sodium vapor at 350 F and sized to accommodate the full-length arc-melt tool and a metal plate structure with sample screwheads. Viewing ports in the lower section of the test chamber permitted observation of the screwhead positions and the articulating arm of the tool.

In a preliminary test in the chamber, the tool was exposed to argon at 350 F for four days and then articulation of the arm and the melt process were tried. A screwhead was satisfactorily melted away, but the articulating arm could not be flexed through the full design movement. The arm could be swung through an arc of only 3 to 5 degrees instead of the designed 10 degrees. After the articulating arm mechanism was disassembled for inspection, a slight galling of the mating surfaces of the ball-and-socket joint was observed. There were tiny dirt particles and what appeared to be burnt oil deposits on interior surfaces of the joint mechanism, apparently the result of the fabrication process. All parts of the mechanism were thoroughly cleaned and the gall marks were sanded smooth.

The assembled tool was then exposed to the hot argon and sodium vapor environment for another four-day period, and again the articulating arm was flexed and several screwheads were melted. All of these operations were successful. The articulating arm assembly was again disassembled; this time, it was found to be in good condition.

C. REMOVING UNDEFORMED SEGMENT THROUGH MOCK-UP OF 14-INCH LINE

It was demonstrated in a mock-up test that a segment can be removed through the 14-inch inlet line without being deformed by a dislodging tool. Although there is a slight condition of interference at the 10-inch-ID shield ring in the inlet plenum nozzle,** the demonstration removal gave assurance that the segment can be maneuvered through the shield ring and that only a minimal extra force will be required to wedge it through. The present plan is to grasp the apex of the segment with the object-retrieval gripper so that as the segment is pulled partly into the flexible removal hose, it thus becomes aligned in the most favorable position for easy removal through the 14-inch line. As was shown in the mock-up test, the

* APDA-CFE-21, Figure 7

** APDA-CFE-21, Page 9

sides of the segment are expected to deform slightly during passage through 10-inch-ID shield ring.

The object-retrieval gripper used in the above segment removal demonstration was the one shown as a wood model in Figure 2 of APDA-CFE-23. A newly-designed body structure for the gripper is presently being fabricated of aluminum and is shaped like a football for reduction of weight and improved control of the device when extended beyond the flexible retrieval hose in the inlet plenum. The same gripper mechanism with its strong bite is being retained.

D. ASSEMBLING CONTROL SYSTEM FOR SPINE-TYPE MANIPULATOR

During September, elements of the new electrohydraulic control system for the spine-type manipulator were assembled and electric and hydraulic settings and adjustments were made to complete the control package for the tool. A description of the control system is given in Section II. D of APDA-CFE-23. It is intended that motion of a single lever control will operate the manipulator through a penetration in the rotating shield plug and holes in the support plates to handle a freed segment and feed it to the object-retrieval gripper for removal through the 14-inch pipe.

The manipulator tool used in the removal of the first segment was completely disassembled, cleaned, inspected, and reassembled to make it ready for tests in the vessel mock-up.

III. OPERATIONS

A. SODIUM AND GAS SYSTEMS PERFORMANCE

Since the last data reported,* the following primary system sodium and cover gas analyses and sodium and cover gas activity have been obtained.

1. Primary Sodium Chemical Analysis (ppm by weight)

Oxygen	15, 16, 17, 18	Iron	2.2, 1.8, 1.8, 3.8
Carbon	32, 34, 40, 57, 91	Nickel	3.1, 1.3, 0.8, 1.2
			Below
		Chromium	0.3, 0.1, 0.3, 0.9

Note: A sample coil was taken from the sampling station on September 15 and analyzed at several different points in the coil to obtain the separate readings indicated. This sample was taken in a nickel coil to eliminate the possibility of sample contamination from the usual Type 304 stainless steel sample tubes. As noted in Section III.C.1 of APDA-CFE-24, the high metal values and the scatter of data reported for the sample taken in July were suspected to have been caused by contamination from the stainless steel sampling tube. The above September values indicate that this suspicion was probably correct; however, the possibility that the July values are truly representative of the sodium condition at that time cannot be ruled out.

2. Primary System Activity

<u>Location</u>	<u>Sample Date</u>	<u>Analysis Date</u>	<u>Specific Activity, μCi/g</u>
Sample Coil from Sampling Station	July 12	July 17	9.7×10^{-3}

* The last sodium activity data are given in APDA-CFE-23. The last sodium chemical analysis data and cover gas activity and chemical analysis data are given in APDA-CFE-24.

3. Primary System Gas Activity

<u>Location</u>	<u>Sample Date</u>	<u>Gross Beta Concentration, $\mu\text{Ci}/\text{cm}^3$</u>
Reactor Cover Gas	9-6-68	3.1×10^{-6}
Reactor Cover Gas	9-13-68	3.2×10^{-6}
Reactor Cover Gas	9-26-68	3.6×10^{-6}
Primary Shield Tank	9-6-68	3.9×10^{-8}
Primary Shield Tank	9-13-68	4.2×10^{-8}
Primary Shield Tank	9-26-68	3.4×10^{-8}

4. 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	135**
Carbon Monoxide	Below 10	Below 10
Carbon Dioxide	20	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	1600	Remainder
Dew Point	Not Measured	-45 F

B. MAINTAINING PRIMARY SYSTEM SODIUM AT 375 F FOR SUBASSEMBLY TRANSFERS

The sodium in the primary system was maintained at a temperature of 375 F during subassembly movements in the reactor and the transfer of subassemblies from the reactor in September (see Sections C and D). The holddown mechanism was raised to the up position just before the beginning of subassembly movements in August and was kept in that position through September, after being kept in the partially-down position as a backup should the abnormal flow conditions resulting from pump operation at refueling speeds and the existence of large void areas in the core cause the subassemblies to float. All three primary sodium pumps were operated in September

* Sample date for gas analysis was 9-6-68
 * Technical specifications state 1000 ppm maximum
 * 10 ppm is the recommended maximum for reactor operations

except during periods of repairs to the No. 1 primary pump oil seal (August 16 to September 12) and to the liquid rheostats for No. 2 and 3 primary pumps (September 17 and 25). During pumping periods, the three primary pumps were operated simultaneously in consideration of pump bearing lubrication.* All three primary pumps were operated at refueling speed to maintain the primary sodium flow rate at less than the 7×10^6 lb/hr specified in the Plant Technical Specification as maximum for the condition with the holddown mechanism in the up position.

The September operating data are as follows:

	<u>Primary System</u>	<u>Secondary System</u>		
		<u>Loop 1</u>	<u>Loop 2</u>	<u>Loop 3</u>
Cold Trap Operation, hr	27	23	9	
Maximum Plugging Temperature, F	230	350	330	Below 240
Minimum Plugging Temperature, F	Below 220	320	260	Below 240

When an attempt was made to cold-trap the No. 1 secondary system sodium in the No. 2 secondary cold trap, a flow of only 10 cubic feet per minute (cfm) was the maximum that could be obtained through the cold trap; normal flow is 60 to 70 cfm. The No. 1 secondary sodium flow was then re-directed to the No. 1 cold trap and the operation carried out. The No. 2 cold trap was heated to a temperature in the range 600 F to 650 F by means of external wall heaters and heaters on adjacent sodium piping for a period of about four days before normal flow was re-established. It is believed that the sustained heat-up redistributed the accumulated sodium oxide in the cold trap to permit flow through the cold trap.

C. REACTOR SUBASSEMBLY MOVEMENTS

In September, 35 core subassemblies, 6 IRB subassemblies, and 5 ORB subassemblies were moved to different positions in the reactor lattice and into the transfer rotor. These moves, along with the transfer of subassemblies (mostly IRBs) out of the reactor for steam-cleaning and temporary storage in the cut-up and decay pools,** complete the extensive fuel-handling program undertaken to open up the required support plate positions for insertion of tools and devices into the inlet plenum for the zirconium segment removal program.*** The fuel-handling program was started in August, as reported in Section III of APDA-CFE-24.

* See page 16 of APDA-CFE-23

** See Section III. D and III. E

*** APDA-CFE-24, Figure 1

The subassembly relocations made during September are shown in Figure 1 and are summarized in Table 1.

The core subassemblies were relocated in IRB positions, in the transfer rotor, one in storage position N04-P14, and in other positions in the core lattice, where it is anticipated that there will be no interference with the insertion of tools and devices for the segment-removal program. At the end of September there were ten core subassemblies in the transfer rotor, seven as a result of the above moves and three from past moves. The latter three subassemblies included M162F, one of a group relocated in April 1967 to permit viewing and extraction of the stuck pair (M098 and M127) and M079 and M058 removed from the reactor lattice in August and December 1967, respectively, to clear the area for holdown finger inspection. Those subassemblies deposited in transfer rotor (TR) positions 11 and 12 were placed in large-diameter pots with shim pads (0.3-inch thick for No. 11 and 0.5 for No. 12) welded to the underside of the pot flanges to compensate for the deformation of the TR plate caused when melted subassembly M098 was dropped during removal efforts.

The subassembly movements were accomplished with the offset handling mechanism (OHM) and the rotating shield plug, operated in the automatic mode. It was frequently necessary to manually adjust plug and OHM positions for those moves where there were less than three camming surfaces available for subassembly insertion and this was often the case because of the complexity of the subassembly rearrangement. It is unusual to insert subassemblies into positions with fewer than three camming surfaces. To facilitate such moves, precamming was often accomplished by insertion in intermediate positions or, where possible, by selecting lift-out positions with favorable subassembly orientation. The rotating plug and OHM azimuth calibration data were found to be accurate.

On occasion it was necessary to shut down the primary pumps to accomplish fuel movements. This was necessary because in some cases there were sufficient flow forces to prevent the incoming subassembly from camming into its proper position.

D. TRANSFER OF SUBASSEMBLIES FROM REACTOR

1. Transfer Operations

By the end of September, 31 IRB and 3 core subassemblies had been removed from the reactor and transported to the transfer tank in the FARB in the multipurpose shipping cask. The subassemblies were then

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

KEY:

CR Control Rod No. 430169-
SR Safety Rod No. 430192-
OR Oscillator Rod
CS Core Shim Subassembly
CF Core Foil Subassembly
BF Blanket Foil Subassembly
CT Coarse Filter, Take-apart, Dummy Core Subassembly
NOTE: Dummy Core Subassemblies in the reactor meet 'Core A' core subassembly specifications and bear the suffix "CF"

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

M 001 - M 206 Core Subassemblies
M 301 - M 400 Inner Radial Blanket Subassemblies
S 500 - S 798 Outer Radial Blanket Subassemblies
M 801 - M 1000 Outer Radial Blanket Subassemblies

Unit shown without prefix are dummy outer radial blanket subassemblies.

- Oversize Nozzle Unit
- "F" Subassembly (Contains fuel pins with high iron plus nickel, high carbon or zirconium content.)
- "W" Subassembly (Contains fuel pins with high iron plus nickel content.)
- Blanket slugs have high carbon content. (APDA Surveillance Program Unit)
- Stranger in Blanket Slugs
- Large Grain Blanket Material (Hash)
- Larger Than Normal Spacing Between the Blanket Elements and the Support Grid
- Type 347 Stainless Steel Wrapper Tube
- Handling Head Short
- Test Flow Subassembly (S-400)
- Slugs Previously Used in a Test Subassembly
- "CP" Slugs
- Locations Where Changes Were Made

Note: Heavy Boxes or Outside of Lattice are Storage Position

FIG. 1 REACTOR LOADING ON 9-30-68

TABLE 1 - SUBASSEMBLY MOVES DURING SEPTEMBER

No.	Type	Position		Reason for Change
		From	To	
M333	IRB	N03-N06	N05-N02	For canning M137 W and M115 into po- sition. These 4 IRBs were subse- quently removed from the reactor.
M351	IRB	N05-N05	N04-N02	
M386	IRB	P04-N05	N05-N01	
M305	IRB	P03-N06	N04-N01	
M115	Core	N04-N03	N04-P00	To open up core support plate positions for zirconium seg- ment removal program
M137W	Core	N03-N03	N05-P00	
M037H	Core	N01-P02	N05-N05	
M165F	Core	N01-P01	N03-N06	
M097	Core	P00-P01	P04-N05	
M160F	Core	N04-N04	P03-N06	
M061	Core	N05-N03	TR-07	
M094	Core	N03-P01	P06-N02	
M025	Core	N02-P01	P06-P02	
M105	Core	P04-N01	P06-P01	
M117	Core	P04-N02	P05-N04	
M126	Core	P04-N03	P06-N03	
M017	Core	P04-N04	TR-02	
M051	Core	P05-N03	N04-P14	
M073	Core	P03-N01	P06-P03	
M086	Core	P03-N02	P06-N01	
M164F	Core	P03-N03	N01-N06	
M153F	Core	N01-N03	P02-N06	To open up core support plate positions for zirconium segment removal program.
M057	Core	P00-N03	P01-N06	
M076	Core	P01-N03	N02-N06	
M106	Core	N01-N04	P05-N05	
M157F	Core	N01-N01	P05-P04	
M092	Core	P00-N04	N06-N03	
M108	Core	P01-N04	P06-P00	
M034	Core	N02-N01	N04-N05	
*M382	IRB	P01-P01	N14-P01	To open up positions for viewing equipment for inspection of the damaged orifice re- tainer fingers in N07-P00** and for hold- down finger*** inspection.
*M330	IRB	P01-N01	N14-P04	
M154	Core	P00-N01	TR-10	
M075	Core	P01-P03	P03-P06	
M023	Core	P02-P00	TR-04	
M046	Core	P01-P02	P04-P05	
M033	Core	P02-P01	P01-P06	
M012	Core	N02-P00	N02-P06	
M138W	Core	P02-N01	TR-09	
M047	Core	N01-N02	TR-03	
M036	Core	P01-N02	TR-11	
M957	ORB	N07-P02	N07-N04	
S502	ORB	N07-P01	P05-N12	
M861	ORB	N07-P00	N06-P11	
M809	ORB	N07-N01	N08-P09	
M877	ORB	N07-N02	N01-N12	

* These two IRBs, along with M304 remaining in core position N01-P05, were inserted in the core lattice after the incident of October 5, 1966 to ensure a noncritical array even if all control and safety rods were withdrawn and reactor temperature reduced to below the freezing point of sodium. The IRBs were not in the reactor lattice during the incident of October 5.

** APDA-CFE-21, Page 11 and Figure 7

*** See APDA-CFE-13 through 19 and 24 for previous information about holddown finger inspection.

steam-cleaned and deposited in the decay and cut-up pools for temporary storage.^{*} One of the subassemblies, IRB M326, was removed from the reactor in August^{**} and the remainder in September.

Four of the IRBs removed, including M326, had been located in the transfer rotor and the remaining 27 in the reactor lattice. Removal of the latter group created space in the IRB rows for core subassemblies displaced to clear the core area for insertion of tools and devices for the segment-removal program. For future reactor operations, replacement IRBs will be drawn from an adequate number of new IRBs presently in new subassembly storage. The three core subassemblies removed were those in the transfer rotor that had been found to be distorted.^{***} Table 2 summarizes the subassembly transfers from the reactor in September.

Three ORB subassemblies and four dummy ORBs were transferred from the FARB transfer tank to ORB positions in the reactor lattice. These transfers were made to make space available in the transfer tank for storage of some of the present core subassemblies when reactor operation is resumed. It is intended that a new core will be loaded in the reactor for resumption of nuclear operation and that the existing core subassemblies will be stored in reactor storage positions, the transfer rotor, the transfer tank and in a special storage facility located in No. 2 decay tank. It is expected that those of the used core found by inspection to be in good condition will be reinstalled in the reactor as individual replacements for new core spent subassemblies.

Table 3 summarizes the ORB transfers to the reactor vessel in September.

The subassemblies listed in Table 3 had been removed from the reactor in March 1967 to vacate an area in the reactor lattice for removal of the damaged subassemblies, M127 and M098.

2. Transfer Difficulties

Subassemblies were transferred between the reactor and the fuel and repair building (FARB) transfer tank in the multipurpose cask^{****} which had been recently modified to include a lower valve assembly.^{*****} The cask was transported between the reactor building and the FARB on a lowboy trolley

* See Section III. E

** APDA-CFE-24, Page 19

*** APDA-CFE-24, Page 25

**** APDA-CFE-5, Pages 27-28

***** APDA-CFE-24, Figure 14

TABLE 2 - SUBASSEMBLIES TRANSFERRED
FROM REACTOR DURING SEPTEMBER

<u>Subassembly Number</u>	<u>Type</u>	<u>From Reactor Position</u>
M357	IRB	TR-08
M397	IRB	P04-P14
M392	IRB	N14-P04
M386	IRB	N05-N01
M367	IRB	TR-07
M351	IRB	N04-N02
M305	IRB	N04-N01
M360H	IRB	P06-P02
M350	IRB	P06-N02
M394	IRB	N04-P14
M333	IRB	N05-N02
M318	IRB	P05-N04
M369	IRB	P06-P01
M400	IRB	P06-N03
M370	IRB	P02-N06
M352	IRB	N01-N06
M395	IRB	P06-P03
M319	IRB	P06-N01
M377H	IRB	N02-N06
M347	IRB	P01-N06
M308	IRB	P05-N05
M396	IRB	P05-P04
M375	IRB	P06-P00
M353	IRB	N06-N03
M359	IRB	TR-04
M327	IRB	N04-N05
M387	IRB	N02-N06
M362	IRB	P01-P06
M398	IRB	P03-P06
M399	IRB	P04-P05
M118	Core	TR-11
M087	Core	TR-09
M007	Core	TR-10

* These IRBs had been relocated in the reactor lattice for camming purposes; see Section III. C.

TABLE 3 - ORB TRANSFERS TO REACTOR VESSEL IN SEPTEMBER

<u>Subassembly Number</u>	<u>Type</u>	<u>To Reactor Position</u>
M926	ORB	N07-P08
422	Dummy ORB	N05-P13
276	Dummy ORB	N02-P14
S512	ORB	N03-P13
M956	ORB	N04-P12
245	Dummy ORB	N07-P12
448	Dummy ORB	P05-N13

structure that formerly was part of the old cask car chassis. As reported in APDA-CFE-24, the cask car was dismantled and usable items reclaimed for other uses.

Figure 2 shows the cask being lowered onto the valve stand which seals the top of the reactor exit port. A silicone rubber gasket is the seal between the cask and the valve stand. In the FARB the cask is sealed to the floor valve over the transfer tank by Silastic cement. Before the cask was opened to either the exit port or the transfer tank for subassembly transfer, it was thoroughly purged with argon gas to minimize air contamination of the reactor and transfer tank atmospheres.

The difficulty in latching the cask gripper on the subassembly pot for the first transfer, as described in Section III. E. of APDA-CFE-24, continued through the next eight transfers. Considering the freedom of swing of the 2-1/2-inch-diameter gripper extension rod in the 9-inch-ID exit pipe, it became a matter of developing the technique of vertically aligning the gripper and the extension rod by feel to lower and latch the gripper on a pot. The proper handling technique appeared to have been improved by about the ninth transfer and thereafter there was no significant difficulty in latching pots. Figure 3 shows the gripper extension rod being lowered into the reactor exit tube; Figure 4, a demonstration of the gripper fingers about to latch a pot. When the gripper is immersed in sodium for deposit or pick-up of a pot and subassembly in the transfer rotor, a downward gas flow through the extension rod is maintained to build up a sufficient pressure to prevent sodium from entering the extension rod.

There were some instances when gas leakage through the extension rod joints prevented the build-up of sufficient pressure and sodium

APDA-CFE-24, Figure 13

**FIG. 2 MULTIPURPOSE CASK BEING LOWERED
ON EXIT PORT VALVE STAND**

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**FIG. 3 COUPLING SECTIONS OF
GRIPPER EXTENSION ROD**

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Gripper
Fingers

**FIG. 4 GRIPPER LATCHING ON
SUBASSEMBLY POT**

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was admitted into the extension rod. The resulting build-up of frozen sodium and oxide deposits in the lowest threaded extension rod joint made it difficult to disconnect the section, thus creating time-consuming delays in the operation. The extension rod was made up in three sections because of the lack of head room in the reactor building and in the FARB to accommodate a single 35-foot extension rod above the cask. Sodium sometimes oozed through the lowest joint as the sections were being disconnected when raising a pot and subassembly into the cask. The joint was enclosed in a plastic bag containing an inert atmosphere to prevent the outbreak of sodium fires as the sections were disconnected.

With continued usage, the gripper became caked with sodium oxide which interfered with but did not prevent latching of the gripper and raising of the gripper through the access pipes. Two gripper assemblies were used interchangeably throughout the transfer program to permit cleaning one when the need was indicated by the above condition.

3. Radiation Levels

With one exception, the maximum radiation level observed during the transfer of all of the subassemblies was 150 mr/hr, and this was measured at the surface of the lower valve housing on the cask with the valve open and a subassembly passing through the valve. When core subassembly M118, the first of three core subassemblies lifted out of the reactor exit port, passed through the lower cask valve, a radiation level of 2 r/hr was measured at the valve stand, and building radiation monitors triggered a Class 1 radiation alarm (local area only -- evacuate that area). As M118 passed into the cask during the alarm, the radiation level at the cask was 150 mr/hr. Lead brick shielding was placed around the lower cask valve and the valve stand over the reactor exit port to prevent recurrence of this incident.

The subassembly transfer operation extended through the entire month of September. Considerable time was taken to accomplish the first 8 of the 41 transfers; but as handling techniques were improved, 2 transfers per day and, at the end of the month, even 3 per day were accomplished.

F. STEAM-CLEANING SUBASSEMBLIES

All of the IRBs and two of the core subassemblies removed from the reactor were taken out of the FARB transfer tank and steam-cleaned. The IRBs were deposited in the decay pool for temporary storage and the core subassemblies were placed in leaker cans in the cut-up pool. At the end of September, core subassembly M118 was still in the transfer tank. It is intended that this subassembly will also be steam-cleaned.

Steam-cleaning of the 33 subassemblies proceeded satisfactorily and without incident, except for the failure of two electrical relays which function

in the normal steam-cleaning process. It was necessary to interrupt the operation for a short period to replace the first of the faulty relays. The proper functioning of this relay was necessary to provide continuity of cooling argon flow through the cleaning chamber while the chamber was moved from the transfer tank access tube to its cleaning position over the cut-up pool. The other faulty relay automatically operates a valve to admit high steam flow in the cleaning cycle. The low steam flow was maintained by manual operation for a longer period than the normal 22 minutes and the process continued. The faulty relay was replaced after the completion of the cleaning cycle. The operating cycle for each subassembly from the transfer tank to the cut-up pool lasts about one hour.

F. COLD-TRAPPING TRANSFER TANK SODIUM

The plugging temperature of the FARB transfer tank sodium increased to 425 F, probably because of impurities introduced during the transfer of subassemblies. Sodium oxide could have been formed on the subassemblies by air leakage during transfer in the multipurpose cask and depositing in the transfer tank.

As the plugging temperature increased toward the 425 F value, the sodium temperature in the transfer tank was increased to 500 F from 400 F and steam-cleaning operations were interrupted to prevent cold-shocking the subassemblies with the relatively cool steam. Cold-trapping of the transfer tank sodium was started and maintained continuously for one week. Plugging temperatures could not be obtained after the last reported value of 425 F because sodium flow to the plugging indicator stopped and could not be re-established. The normal means of determining the results of cold-trapping were, therefore, not available.

At the beginning of cold-trapping, a flow of only 8 to 10 gpm could be obtained, indicating a substantial quantity of impurities in the transfer tank sodium; the normal flow rate is 20 gpm. As the cold-trapping operation proceeded, the cold trap temperature was reduced to 250 F and at the end of the week's operation it was observed that sodium flow to the cold trap increased to the normal 20 gpm. This was considered to be satisfactory evidence that the plugging temperature had been sufficiently reduced to allow resumption of fuel transfer operations. Cold-trapping was stopped, the temperature of sodium in the transfer tank was reduced to 400 F, and steam-cleaning of subassemblies was resumed. The plugged condition in the plugging indicator circuit is to be investigated.

G. OPERATION OF STEAM GENERATORS

1. Preparing for Steam Generator Tests

In September, preparations were made for conducting tests of all three steam generators to confirm the integrity of the recent tube-to-

tube sheet weld repairs under isothermal conditions and under conditions of severe thermal gradient between the water and sodium sides. In addition, the intent is to carry out hydraulic tests to verify the predicted feedwater flow and pressure drop characteristics due to the installation of the orifices in the water tubes." The integrity of all three steam generators had been verified by nitrogen pressure tests after each was filled with sodium and prior to water fill. In all tests, the water tubes were pressurized with nitrogen (no water in the tubes) at 600 psig and the shell side argon cover gas monitored on a chromatograph for any increase in nitrogen content as evidence of tube leakage. In all cases there was no significant increase in nitrogen content over a 24-hour test period.

All three steam generators were filled with water in September; No. 1 at the beginning of the month, No. 2 at the middle of the month, and No. 3 near the end of the month. The source of water for filling the steam generators was No. 4 heater where plant condensate was heated under pressure to 350 F. The water-filling procedure included back pressurizing from the steam outlet to lift the water tube orifices to relieve the nitrogen trapped in the annulus around the orifice tube in each water tube. Circulation at a pressure of 600 psig was established by a plant boiler feed pump through the steam generator and main steam bypass line back to No. 4 heater. It was intended to obtain water flow rates in the range of 10,000 to 212,000 pounds per hour as conditions for hydraulic tests.

2. Manifold Leaks

Within a day after circulation was established in each of the steam generators at 600 psig, leakage from the water manifold cover plate joint to air occurred. It was observed that there was similar leakage from the steam manifold of No. 1 steam generator. Water pressure was not maintained long enough in No. 2 and 3 steam generators to determine the integrity of these steam manifold cover plate joints. Attempts to stop the leaks through the cover plate joints by increasing the tightening torque on the cover plate bolts were not successful. The torque recommended by the steam generator manufacturer is 1,000 ft-lb for the water manifold cover plate bolts and 1,200 ft-lb for those fastening the steam manifold cover plate. The torque values were increased to 1,225 and 1,500 ft-lb, respectively. The leaks were discovered first in No. 1 steam generator; therefore, the increased torque was applied to No. 2 and No. 3 steam generators before these units were filled with water. The water was vented from No. 1 steam generator to retorque the cover plate bolts. A nitrogen pressure test of the unit prior to water refill indicated that a tube leak had developed (see Sec. III.G.3). Consequently, the unit was not refilled with water and hence it was not known if the cover plate joint leaks were stopped

on this unit. The persistence of water manifold leaks from the No. 2 and 3 steam generators was sufficient evidence that the manifold-to-air leak problem was common to all three steam generators. It was, therefore, decided that the suspect cover plate gaskets would have to be replaced.

The water manifold gasket from No. 1 steam generator and replacement gaskets for all units were checked for hardness. It was found that these supposedly soft iron gaskets had not been annealed. The measured hardness value was Rockwell 50B; it should have been Rockwell 30B. Replacement gaskets are to be properly annealed and installed on all steam generators.

3. Tube-to-Shell Leaks in No. 1 Steam Generator

The tests in September indicated tube-to-shell leakage in No. 1 steam generator. There was no evidence of tube leaks in No. 2 and 3 steam generators. Immediately after No. 1 steam generator was filled initially with water, the hydrogen content in the argon cover gas over the sodium shell side increased rapidly to 5100 ppm; the nitrogen content increased to a value in excess of 2500 ppm. These values, measured by chromatograph sampling of the cover gas, declined and leveled off at about 500 ppm and 65 ppm, respectively, until the water was vented to retorque the cover plate bolts as noted in the preceding sections.

In advance of a second water refill of the No. 1 steam generator when the water tubes were pressurized to 300 psig with nitrogen to prevent water flashing, it was observed that the nitrogen pressure in the tube decreased rapidly, reaching 70 psig in less than four hours. Simultaneously, the measured nitrogen in the argon cover gas increased to a value in excess of 2500 ppm. This was evidence of additional tube leakage since the initial water fill. There had been no decrease in pressure in the tubes observed in the nitrogen pressure test prior to the initial water fill.

The measured contents of hydrogen and nitrogen in the cover gas over the shell side sodium in the No. 2 and 3 steam generators were less than 20 ppm while water was being circulated through these units.

Sodium was drained from the No. 1 steam generator and, after the manifold covers were removed, a bubble test was applied to locate the tube leaks. In this test, the tubes are filled with water to the tubesheet level and the shell side pressurized with argon at 35 psia. Bubbles indicated leaks from two tube-to-tubesheet, internal-bore welds and a leak in the outer row of steam tubes that was subsequently determined to be in the vertical section of tubing 10 to 10-1/2 feet below the steam tube sheet, as shown in Figure 5. It is expected that the leaking tubes will be plug-welded.

The method for determining the location of this leak will be described in a future report.

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FIG. 5 SKETCH OF STEAM GENERATOR

IV. SPECIAL INVESTIGATIONS

A. PREPARATIONS FOR REMOVAL OF SWEEP MECHANISM

It is planned to remove the sweep mechanism from the reactor to make room for insertion of tools and viewing and lighting equipment into the inlet plenum for the segment removal program. The sweep mechanism, shown in Figure 6, is presently installed through the rotating shield plug in the penetration which is needed for viewing of the north sector of the conical flow guide. This same penetration will be needed for insertion of tools and viewing equipment to remove the zirconium segment in this area.

The plan is to lift the sweep mechanism into a 25-foot-long, 6-inch-ID cask containing inert gas and to temporarily store the cask with the sweep in a decay tank. In anticipation of possible radiation from the sweep in the range of 5 to 50 r per hour at a distance of 1 foot, remote control of the removal and storage operation is planned in the containment building and from behind a temporarily erected shield wall if necessary. There will be no special shielding installed in or around the decay tank. For the period the sweep will be in storage, the only anticipated personnel activity below-floor will be for segment removal; and it has been determined that existing tanks and structures will provide adequate shielding.

Figure 7 shows the cask that was fabricated to contain the sweep and the 1-inch-thick plate that is to rest on a ledge in the decay tank to suspend the cask. A close-up of the hoist mechanism that rests on the suspension plate is shown in Figure 8. Argon from an external source will maintain the cask at a positive pressure of 10 inches of water during storage. Calculations based on a leak test of the cask indicates that the expected in-leakage of air by diffusion would be no greater than 0.5 standard cm³ per month, which is not considered to be objectionable. The cask is sealed at the top at the hoist cable penetration and at the bottom by a special valve.

B. ANALYSIS OF FULL DIMENSIONAL INSPECTION OF SUBASSEMBLIES

The results of the visual inspection and inspection by the quick scan method of subassemblies in the exit port inspection facility before the subassemblies located in the transfer rotor were removed from the reactor were reported in Section IV.A. of APDA-CFE-24. Most of the subassemblies were given full dimensional inspection at the same time and the results are shown in Table 4.

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FIG. 6 SWEEP MECHANISM PRIOR TO INITIAL INSTALLATION

FIG. 7 SWEEP REMOVAL CASK

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FIG. 8 CLOSE-UP OF HOIST MECHANISM

TABLE 4 - RESULT OF FULL-DIMENSIONAL INSPECTION

<u>Subassembly Number</u>	<u>Type</u>	<u>Reactor * Position</u>	<u>Measured Distortion, ** in.</u>
M367	IRB	P05-N04	0.01 0.07
M007	Core	N02-P03	0.29 0.15
M087	Core	N03-P02	0.74 0.41
M058	Core	N03-N01	0.15 0.12
M079	Core	N01-N03	0.24 0.23
M162F	Core	N05-N02	0.67*** 0.08

* At the time of the October 5, 1966 incident

** Distortion in adjacent planes

*** Considered invalid because of error in measurement

It is presently estimated that the above distortion values may be in error by 0.150 inch because of the preliminary techniques used and the difficulties in sighting on the edges of sodium-wetted subassemblies.

On the basis of the full dimensional inspection, M007 and M087 are considered to be distorted. M079, which was located remotely from the cluster of damaged subassemblies, is in doubtful condition and therefore must be considered possibly distorted. M058, M162F and M367 are believed to be undamaged. M007, M087, and M118, which were shown to be grossly distorted by the quick scan inspection, were removed from the reactor. The other core subassemblies that were inspected are still in the reactor and will be examined in greater detail at a later date.

V. DAMAGED SEAL IN FLOOR VALVE TO CUT-UP POOL

Difficulties encountered in the operation of the gripper for the steam-cleaning machine* led to the discovery of a damaged seal in the cleaning chamber floor valve to the cut-up pool. After a subassembly was deposited in the cut-up pool and the gripper was being lifted into the cleaning machine, the gripper hoist tripped out because of an indicated overload when the gripper reached the floor valve. The gripper weighs about 300 pounds and the hoist trips out at about 700 pounds. After alternate lowering and raising, the gripper cleared the valve.

The floor valve is the same as that shown in Figure 9 (this is actually the valve for the dry loading tunnel). Figure 10 shows the valve disassembled. The Buna rubber seal which seals the edges of the sliding valve plate was found deteriorated and torn (Figure 11) and was the cause of hanging up the cleaning machine gripper. There was no replacement seal immediately available; therefore, the valve was left in the open position to permit unobstructed passage of the gripper for deposit of the remaining cleaned subassemblies in the cut-up pool. A replacement seal will be installed when it is available. Access of personnel to the cleaning chamber for valve repairs was possible because the radiation level did not exceed background with no subassembly present.

* See Section III. E

**FIG. 9 FLOOR VALVE TO DRY
LOADING TUNNEL IN FARB**

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FIG. 10 VALVE PARTS DISASSEMBLED

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FIG. 11 DAMAGED VALVE SEAL

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