MASTER

ENRICO FERMI ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

**COMPILATION OF CURRENT TECHNICAL
EXPERIENCE AT**

ENRICO FERMI ATOMIC POWER PLANT

JUNE 1968

R. H. Costello

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

Atomic Power Development Associates, Inc.

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PREPARED BY

Atomic Power Development Associates, Inc.
with the Cooperation of
Power Reactor Development Company
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The Detroit Edison 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 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 development of tools and devices for removing the remaining zirconium segments from the reactor vessel continued in June. Parallel effort continued on the chisel and arc-melt methods for removal of the zirconium screw heads which fasten the zirconium segments to the conical flow guide in the reactor core inlet plenum.

Full-length tools for both of these methods are designed to employ the same articulation feature for positioning the chisel and the arc-melt electrode at the screw heads. The articulated joint was tested in a sodium vapor atmosphere at 350 F and found to function satisfactorily. In demonstrations on a test plate, a full-length chisel tool sheared off screw heads heated to 350 F but left protruding screw shanks and burrs which could leave the zirconium segments still somewhat restrained on the conical flow guide. Fabrication of the full-length arc-melt tool was started in June.

It is expected that the spine-type manipulator used in the removal of the first zirconium segment will be used in the program to remove the remaining segments. A new electrohydraulic-type control is being developed to permit direct control, and hence precise movement, of the device by the person viewing operations in the inlet plenum through the borescope. Rotation of the spine-tool gripper to make the tool more versatile is another feature under development. The mock-up structures simulating the full-diameter inlet plenum and the top deck of the rotating shield plug were completed in June and ready for testing of tools and devices.

The primary system was filled with 30,000 gallons of sodium from the No. 2 and No. 3 primary storage tanks in preparation for fuel movement to facilitate access through the core support plate and to provide for clean-up of the primary system. After sodium filling, the holddown mechanism (HDM) was lowered to the point where the HDM fingers were 2-3/4 inches above the subassembly handling heads as a barrier to the unlikely lifting of subassemblies out of the support plates by the abnormal sodium flow conditions resulting from the partially unloaded core. The HDM fingers were not engaged with the subassembly handling heads because of the possible misalignment which could result from the free standing subassemblies in the partially unloaded core and the concern regarding the alignment of the fingers.

Sodium was circulated in the primary system and its temperature increased by heat input from the three primary pumps, the reactor vessel and pipe heaters in the primary system, and component heaters in the secondary system. Sodium flow in the primary system was 6.7×10^6 pounds per hour. Plant specifications limit primary system flow to 7×10^6 pounds

per hour with the HDM in the up position. All three primary pumps were operated together at low speed to develop the head necessary for pump bearing lubrication in view of the 32 vacant positions in the core.

Primary system temperature was increased in steps of about 50 F above the 350 F filling temperature, and primary sodium was cold-trapped at each 50 F plateau until plugging temperatures were reduced to 220 F or lower. At a primary system temperature of 563 F, the temperature of the secondary shield wall reached 190 F. Because this approached the transient limiting temperature of the wall, it was decided to decrease primary system temperature. This limiting temperature was reached because of the extensive use of external heaters necessitated by the inability to operate the primary pumps at full flow. It was intended to raise primary system temperature to a maximum 600 F for clean-up operations. This 600 F maximum was deemed to be satisfactory for clean-up while still within the maximum allowable stress conditions at the 14-inch pipe nozzle of the No. 3 loop at the reactor vessel.

During the first part of June, plugging temperatures were not too clearly defined when obtained by the conventional method of diverting 0.5 gallon per minute (gpm) to the plugging indicator from the 60 to 80 gpm high pressure flow to the cold trap. Subsequently, plugging temperatures were obtained satisfactorily by an alternate mode in which a minimum pressure drop was established across the plugging indicator. In this mode the normal cold trap flow was bypassed back to the primary system and low flow at low pressure was delivered to the plugging indicator.

When the primary system temperature was at 535 F, excessive nitrogen leakage through the primary shield tank expansion seal (boot seal) was discovered. It was no longer possible to hold a positive nitrogen pressure of 8 inches water column in the primary shield tank, despite efforts to patch the defective areas of the seal; therefore, a nitrogen flow-through atmosphere was maintained in the primary shield tank. A method of replacing the old seal is presently being studied.

II. INVESTIGATION OF ZIRCONIUM SEGMENTS

A. BACKGROUND

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

The development of tools to remove the heads of fastening screws is in progress. The concepts of screw head removal by chiseling and by the arc-melt process are being developed in parallel efforts to avoid long delays if difficulties with either method should arise later in the program. Four tools are being fabricated to dislodge a segment from the conical flow guide after screw heads are removed, and to deform a segment if that is required to ensure its unobstructed withdrawal through the 14-inch pipe. Other conceptual efforts have been directed toward refining the control features and improving the functioning and structural features of the devices for handling the segments in the inlet plenum and for removing them through the 14-inch pipe.

Previous information on the investigation of the zirconium segments is given in APDA-CFE-20, 21, and 22.

B. SCREW HEAD REMOVAL TOOLS

1. Chisel Tool

In June, fabrication of a full-length chisel tool with an articulated joint was completed. This tool is designed to remove screw heads which fasten the zirconium segments on the surface of the conical flow guide in the reactor inlet plenum. Feasibility tests demonstrated that this tool could shear off screw heads at room temperature and at the anticipated inlet plenum temperature of 350 F. The screw heads sheared off were representative of those on the conical flow guide and require various angles of attack by the chisel. Articulation of the tool through angles of up to 7 degrees from the axis of the tool was required in positioning the chisel to all of the screw heads in the feasibility test; however, the shearing operations

did not produce clean cuts. There was a tendency in some cases to leave slightly protruding screw shanks, peened-over shanks, and burred shanks, which could offer restraint to dislodging the segments from the surface of the conical flow guide.

Figure 1 shows an accessory designed to trap screw heads after they are sheared off by the chisel. This accessory, an integral part of the chisel tool head, offers a spring restraint to ensure retention of the sheared-off screw head. This screw head catcher, which is an improvement over that described in Section II. B. 1 of APDA-CFE-22, functioned satisfactorily in the feasibility tests of the chisel tool.

2. Arc-Melt Tool

The engineering design of a full-length arc-melt tool was completed and fabrication of the tool was started. The arc-melt method is another approach to the removal of the fastening screw heads. The arc-melt tool, which, like the chisel tool, is designed for insertion through the holes in the core support plate via the 6-inch penetration in the rotating shield plug for access to the screw heads, will employ the same design of articulating joint used on the chisel tool. Another feature of the tool provides for advancing the electrode 7/8-inch to afford flexibility in positioning the electrode relative to the screw head to be melted.

The articulating joint was exposed to sodium vapor for a week to determine the effects of long exposure in the atmosphere that exists in the inlet plenum. After the exposure, this feature of the arc-melt tool continued to be capable of full articulation through the designed 10-degree angle. The ball-and-socket surfaces of the articulating joint were initially coated with a molybdenum disulphide lubricant. It is intended that both the full-length chisel and arc-melt tools will be tested in the reactor mock-up.

C. SEGMENT-DISLODGING TOOLS

The fabrication of an assortment of segment dislodging tools^{*} is currently under way. These include the plow, tire iron, edge curler and shover tools designed to dislodge the zirconium segments from the surface of the conical flow guide after removal of the fastening screw heads. It is intended that all of these dislodging tools will be manipulated from the top of the rotating plug by means of a spring cable inserted through the holes in the core support plates via a 6-inch penetration in the rotating plug. Spring cables of 3/4 inch, 1 inch, and 1-1/4 inch have been provided. The specific size will depend upon the degree of flexibility required for access to the various zirconium segments as demonstrated in the mock-up.

^{*} APDA-CFE-22, Section II. C

FIG. 1 CHISEL TOOL AND SCREW HEAD CATCHER

D. SPINE-TYPE MANIPULATOR

It is expected that the spine-type manipulator* used in the removal of the first zirconium segment in March will also be used in the program to remove the remaining segments. This manipulator is operated through a rotating plug penetration and holes in the core support plates to handle a freed segment for removal through the 14-inch pipe. Based on previous experience with the tool, a completely new type of control and a modification of the gripper are currently under development.

— The new type control is an electrohydraulic system which offers the advantage of direct control of the device by the person viewing operations in the inlet plenum through the borescope, essential for accomplishing the small, precise movements so often required of the tool. During the removal of the first zirconium segment, the manipulator was manually controlled at the head of the tool by one person in response to instructions from another person viewing through the borescope.

Movement of the spine-type manipulator, and hence positioning of the gripper in the inlet plenum, continues to be accomplished by flexing and relaxing the four control cables which penetrate the outer periphery of the stacked disks. In the new control system being developed, a double-acting hydraulic power piston actuates each control cable through a connecting rod and chain. The present intention is to experiment with a single lever control to produce manipulator movements. Positioning the control lever would actuate an electrical system to operate solenoid valves which would direct hydraulic pressures to the power pistons. The design of the control system provides effective control of the manipulator disks by applying a restraining force to the relaxing control cables while powering the flexing cable or cables.

The power piston chambers are contained in a housing which traps any leakage of hydraulic fluid and drains it to a separate receptacle. Seals prevent the escape of cover gas to this housing or to the atmosphere. For convenience in inserting or removing the tool from the reactor, the power piston package is disconnected from the tool at a specially-designed set of disconnect links.

The gripper is to be modified by the addition of features for rotation of the gripper body to provide versatility of the manipulator in the inlet plenum.

* See APDA-CFE-15 through 19, Section II, for previous information about the spine-type manipulator.

E. OBJECT-RETRIEVAL DEVICE

The object-retrieval device* inserted through the 14-inch pipe to retrieve the first zirconium segment is intended for use in the current program to remove the remaining segments from the reactor. A new design of the retrieval trolley and gripper is being developed because, from the known dimensions of a segment, it is recognized that the segment cannot be wholly contained in the flexible duct during removal; therefore, the segment must be grasped firmly. Figure 2 shows a wood model of the trolley and gripper assembly. The gripper mechanism provides a mechanical advantage such that the three teeth built into the jaws, by design, will penetrate through the 0.040-inch-thick zirconium plate.

The two sets of ball rollers in the body of the trolley are intended for movement of the trolley in and out of the retrieval duct in the inlet plenum to facilitate grasping a segment. The antenna-like lead-in rods direct the segment into the gripper jaws. It is expected that use of the model in the reactor mock-up will provide information useful in the development of the working model.

F. REACTOR MOCK-UP

Figures 3 and 4 are views of the reactor vessel mock-up and the mock-up of the top of the rotating plug where tools are inserted into the reactor. In Figure 4, the simulated mechanisms in the vicinity of the RP penetrations may be observed. Figure 5 shows the platform sized to simulate the working area at the 14-inch pipe elbow to be again penetrated for removal of the remaining zirconium segments. The mock-up structures were completed in June.

* See APDA-CFE-15 through 19, Section II for previous information about the object-retrieval device.

**FIG. 2 WOOD MODEL OF TROLLEY AND GRIPPER FOR
OBJECT-RETRIEVAL DEVICE**

FIG. 3 MOCK-UP OF REACTOR AND 14-INCH PIPE

**FIG. 4 MOCK-UP OF TOP OF
ROTATING SHIELD PLUG**

**FIG. 5 MOCK-UP OF PLATFORM AT
14-INCH ELBOW PENETRATION**

III. OPERATIONS

A. CLEAN-UP OF PRIMARY SODIUM SYSTEM

The presence of impurities and the general condition of the interior of the reactor vessel observed in April * and during the investigation of the first zirconium segment prompted sodium filling and clean-up operations in the primary sodium system. The need for clean-up is largely the result of the extensive maneuvering of tools and devices in the inlet plenum which culminated in the removal of the deformed zirconium segment in March.** The reactor vessel has been maintained in a completely drained condition since August, 1967, and the piping loops, IHX's, and pumps in a partially drained condition since January, 1968. The present period of inactivity in the reactor vessel during development of tools and devices to detach the remaining zirconium segments and remove them from the reactor presented the opportunity to refill the primary system for clean-up and the necessary fuel handling to provide access to the remaining zirconium segments.

1. Preparations For Filling The Primary System

The following steps were taken in preparation for filling the primary system for clean-up operations:

- a. The sodium in the 14-inch inlet line and in the drain lines from the No. 1, 2, and 3 IHX's was sequentially thawed and the line temperatures increased to 350 F. The details of thawing the inlet line are given in APDA-CFE-21, page 27.
- b. A level probe was installed in one of the available 6-inch penetrations through the rotating shield plug and a lead glass window in the other available penetration. This level probe was intended as back-up for the normal high level detection instrumentation in the event of a plugged overflow line.
- c. The electrical power leads were connected to the primary pump main motors. The leads had been disconnected in accordance with plant specifications, which specify this measure when the holddown is not in the down position.
- d. About 250 cu ft of sodium was pumped from the overflow tank to the primary system through the No. 1 IHX (see Figure

* APDA-CFE-21, Section III. B

** APDA-CFE-20, Section II

6). The sodium level in the overflow tank was thus lowered from 13-1/2 feet to 7 feet to accommodate overflow from the reactor vessel indicating the end of the fill operation.

- e. The temperatures of the No. 2 and 3 primary storage tanks and service system piping were adjusted to 350 F.

2. Filling The Primary System

The primary system was filled with sodium from No. 2 and 3 primary storage tanks by pressurizing the argon cover gas in the storage tanks. The fill route is shown in Figure 6. Sodium entered the primary system through the recirculating line and the No. 1 IHX and from there filled the No. 1 pump tank and the reactor vessel via the 30-inch outlet line. Sodium levels in the three primary loops were equalized through the IHX drain lines and the reactor vessel.

The filling operation, proceeding at the rate of about 100 gallons per minute, was accomplished in 7-1/2 hours. About 30,000 gallons of sodium was transferred to the primary system from the No. 2 and No. 3 primary storage tanks. Cover gas pressure in the storage tanks was maintained at 12 psig during the fill, and primary system cover gas at 3 inches water column gauge. Cover gas displaced from the primary system was discharged to the waste gas system.

The sodium level in the reactor vessel was observed through the lead glass window installed in one of the 6-inch penetrations through the rotating shield plug. When the sodium level covered the core subassembly handling heads, the filling operation was temporarily halted. The surface of the sodium was viewed to determine the feasibility of using the sodium surface as a mirror to reflect an image of the holddown fingers for borescope viewing. The observation of continuous surface ripples discouraged this approach.

A minor difficulty experienced during filling was the leaking of the overflow pump check valves, which allowed sodium to enter the overflow tank and raise the tank level. Intermittent operation of an overflow pump was required to return sodium to the primary system so as to maintain the original low level in the overflow tank. The completion of filling was evidenced by a sudden rise in level in the overflow tank due to spill-over from the reactor vessel.

3. Special System Conditions

The following special system conditions were imposed during the clean-up operations:

FIG. 6 PRIMARY SODIUM SERVICE SYSTEM FLOW DIAGRAM

- a. The holddown mechanism was lowered, but only to an elevation 2-3/4 inches above the full down position because of the presently unknown condition of the tops of all subassemblies in the core. In this position the holddown spider legs were aligned on the support columns and the fingers were poised to prevent the unlikely lifting of core subassemblies out of the support plates.
- b. Primary pump speed was limited to that which would produce a sodium flow rate less than 7×10^6 pounds per hour (refueling flow), the flow rate specified in the Plant Technical Specifications for the condition with the holddown in the up position.
- c. It was required that all three primary pumps be operated simultaneously to develop sufficient pump head with the core configuration having 32 vacant positions. Insufficient pump head would cause a lack of bearing lubrication. The throttle valves in the 6-inch lines to the blanket plenum were closed to aid in developing pump head.

4. Primary System Heat-Up

A primary system temperature of 600 F was established as the maximum allowable for clean-up operations to prevent overstressing the 14-inch pipe-to-reactor vessel nozzle of No. 3 loop. It is known that the stress conditions are satisfactory at 600 F, but no attempt was made to verify that stress conditions would be acceptable at higher temperatures, since it was determined that 600 F would be acceptable for sodium clean-up.

Heat-up of the primary system was accomplished by power input to the primary and secondary pumps and by use of component and pipe heaters in both the primary and secondary systems.

5. Cold Trapping

The intent was to raise primary system temperature from the 350 F filling temperature to the established maximum temperature of 600 F in 50 F steps, and at each 50 F plateau to cold-trap the primary sodium until a plugging temperature of 220 F or below was obtained. This plan was followed to a large extent, except that the highest primary system temperature reached was 563 F. Temperatures up to 190 F (40 F above a given plant limit) were observed in the concrete secondary shield wall; therefore, it was decided to decrease primary system temperature to correct the shield wall condition. The excessive temperatures existed primarily at the top of the north wall and were probably due to proximity to external inductive and resistance heaters. The 150 F limit for shield wall temperatures is being re-evaluated.

The primary system flow was maintained at 6.3×10^6 pounds per hour with three primary pumps operating. Sodium flow to the cold trap was 60 to 80 gpm throughout the clean-up operations. The cold trap was operated about 184 hours during June. During most of this time the cold trap was in continuous operation. A sodium sample (see Sec. II.D.2) representative of the preclean-up condition of the primary sodium was obtained at a primary system temperature of 440 F. Sodium had been circulating in the primary system for four days and the cold trap had been in operation for only one hour when the sample was taken. It is expected that another sample of primary system sodium will be taken at the end of the clean-up operations.

At the end of June, the primary system temperature was about 500 F and the plugging temperature had been less than 220 F for several days. The intent is to soak the system at this temperature for an additional period and cold-trap as necessary to maintain the plugging temperature below 220 F, and then to reduce system temperature to 350 F.

6. Plugged Temperature Control Valve

The plugging indicator was in operation on automatic cycling throughout the clean-up period. In June there was an interruption of plugging indicator operation, due apparently to malfunction of temperature control valve TCV 509 (Figure 6) in the NaK cooling system for the plugging indicator. Lack of NaK flow prevented cooling the plugging indicator and hence its operation. It was concluded that oxide had accumulated at TCV 509 and obstructed NaK flow as a result of the long period of inactivity.

The NaK in the cooling system was drained to a sump tank and impurities were removed by filtering and diffusion cold-trapping. After the application of heat to the body of TCV 509 and the clean-up of the NaK, the NaK system was restored to normal operation.

7. Two Modes of Plugging Indicator Runs

In the beginning stages of primary system clean-up, plugging runs, carried out during cold-trapping operations, were accomplished by throttling FCV 504 (see Figure 6) to produce the required 1/2-gpm flow through the plugging indicator. In some instances this mode of operation resulted in questionable interpretations of the chart records indicating the plugging temperatures. Figure 7 is a representative record of a plugging run performed in the above mode. It is possible that the high pressure drop across the plugging indicator erodes the formation of plugs at the plugging orifice and causes faulty results.

Another mode of operation, which is based on the principle of the same 1/2-gpm flow but with very little pressure drop across the plugging indicator, was then tried. In this second mode of operation, the cold trap

FIG. 7 PLUGGING INDICATOR RECORDER CHART WITH COLD TRAP IN OPERATION

is shut down and a 1/2-gpm flow to the plugging indicator at low pressure is accomplished by control of the large volume of flow in the bypass circuit to the No. 1 IHX. Figure 8 is a record of a typical plugging run in this low pressure drop mode.

B. LEAKS THROUGH PRIMARY SHIELD TANK EXPANSION SEAL

Excessive nitrogen leakage from the primary shield tank through the expansion seal was discovered during clean-up of the primary system when system temperature was at 535 F. It was observed subsequently that the rubber seal was extensively cracked. The expansion seal, referred to as the boot seal, is a 10-inch-wide, 5/8-inch-thick, styrene-butadiene rubber belt that rings the inner periphery of the primary shield tank (PST) and seals between the PST and the machinery platform at the top of the reactor vessel. Figure 9 shows the location of the boot seal. The machinery platform, which forms the roof of the primary shield tank, is sealed to the top flange of the reactor vessel by a metal tubular ring with 3-inch-OD, 0.065-inch-thick cross section. As shown in Figure 9, this tubular ring seal, called the omega seal, is split and welded to the adjacent surfaces.

The boot seal was designed for an operating life of 15 years at 200 F, with the provision that for operation at each 5 F above 200 F, the belt life would be shortened by 1-1/2 years. The temperature in the vicinity of the boot seal during the clean-up operation was observed to be 240 F. It is felt that this excessive temperature resulted from the extensive use of external heat input necessitated by the inability to operate the primary pumps at full flow and hence a restriction on the heat input from mechanical sources. It is known that previous periods of operation also have required similar use of resistive heating; hence, it is not surprising that failure of the boot seal occurred.

Attention was directed to the boot seal leakage when excessive consumption of nitrogen was observed and positive nitrogen pressure in the primary shield tank could not be maintained. The machinery dome was reinstalled on the PST to verify leakage through the seal and to contain the PST nitrogen atmosphere, if required. The oxygen content inside the dome decreased with a corresponding pressure increase, indicating that the leakage was indeed through the machinery platform seals.

A borescope inspection of the boot seal revealed extensive longitudinal cracking of the rubber seal material. The failed area was covered with plastic and tape, the PST nitrogen system was temporarily changed from a pressure control to a flow-through system, and primary system heat-up continued for clean-up. At the 535 F primary system temperature the leak rate through the boot seal was measured to be 29 cfm with a positive pressure of 1 inch water column in the PST. When system temperature was temporarily reduced to about 500 F, the leak rate decreased considerably,

**FIG. 8 PLUGGING INDICATOR RECORDER CHART—LOW PRESSURE DROP
MODE—NO COLD TRAP OPERATION**

FIG. 9 EXPANSION JOINT (BOOT SEAL) BETWEEN PRIMARY SHIELD TANK AND MACHINERY PLATFORM

indicating that at higher temperatures the upward movement of the reactor vessel with respect to the PST was flexing the boot seal and opening up the cracks in it.

An alternate method of sealing is presently being studied. Access to the PST for removal of the failed belt and installation of a new seal is extremely difficult because of the subsequent addition of lead shot in the space under the machinery deck area, where access to the seal had been provided.

C. CLEAN-UP OF THE SECONDARY SODIUM LOOPS

Immediately after the refill and start of clean-up of the primary system, sodium in the three secondary loops was cold-trapped for the periods tabulated below, and the plugging temperatures reduced to 220 F. The initial plugging temperatures are given in the same tabulation.

JUNE OPERATING DATA

	<u>Primary System</u>	<u>Secondary System</u>		
		<u>Loop 1</u>	<u>Loop 2</u>	<u>Loop 3</u>
Cold Trap Operation, hr	184	13	12	14
Maximum Plugging Temperature, F	420	305	275	280
Minimum Plugging Temperature, F	220	220	220	220

D. SODIUM AND GAS SYSTEM PERFORMANCE

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

1. Primary Sodium Activity

<u>Location</u>	<u>Sample Date</u>	<u>Analysis Date</u>	<u>Specific Activity, microcuries/g</u>
Sample Coil from Sampling Station	June 11	June 28	8×10^{-3}

* The last sodium activity and analysis data are given in APDA-CFE-16.
The last gas activity and analysis data are given in APDA-CFE-22.

2. Primary Sodium Chemical Analysis

Oxygen	14, 15, 17	Iron	6.1, 8.4
Carbon	27, 33, 34	Nickel	0.9, 1.3
*Hydroxide Hydrogen	1.7, 1.7	Chromium	1.9, 2.2
*Nonhydroxide Hydrogen	0.2, 0.3		

* 1.3 ppm recommended maximum for total hydrogen for reactor operation.

Note: Values are in ppm by weight. The sample was taken from the sampling station on June 11, before primary system clean-up and was analyzed at several different points to obtain the separate readings indicated.

3. Primary System Cover Gas Activity

<u>Location</u>	<u>Sample Date</u>	<u>Gross Beta Concentration, microcuries/cm³</u>
Reactor Cover Gas	6-7-68	1.7×10^{-6}
Reactor Cover Gas	6-19-68	2.3×10^{-5}
Reactor Cover Gas	6-21-68	2.1×10^{-5}
Reactor Cover Gas	6-28-68	2.0×10^{-5}
Primary Shield Tank	6-7-68	2.4×10^{-7}
Primary Shield Tank	6-19-68	3.5×10^{-8}
Primary Shield Tank	6-28-68	1.5×10^{-9}

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	45**
Carbon Monoxide	Below 10	Below 10
Carbon Dioxide	20	20
Hydrogen	Below 4***	Below 2.5
Helium	Below 4	Below 4
Methane	Below 10	Below 10
Nitrous Oxide	Not Measured	Not Measured
Argon	Remainder	600
Nitrogen	4320	Remainder
Dew Point	Not Measured	-55 F

* Sample dates for reactor cover gas and primary shield tank atmosphere are 6-7-68 and 6-19-68, respectively.

** Technical specifications state 1000 ppm maximum.

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

IV. SPECIAL INVESTIGATIONS

A. INSPECTION OF SUBASSEMBLY M122

Microhardness measurements were taken on the wrapper can of subassembly M122 adjacent to the region of the large hole in subassembly M098 (third grid from the bottom) and where apparent molten fuel alloy had been deposited. Results of these microhardness measurements have indicated little, if any, diffusion of the adherent fuel alloy into the stainless steel wrapper can, confirming the results of a previous metallographic examination of a transverse section of M122 reported in APDA-CFE-18, page 41. Subassemblies M122 and M140 were adjacent to the melted subassemblies M098 and M127 in a four-subassembly cluster at the time of the incident of October 5, 1966. Earlier results of inspections of subassembly M122 are given in APDA-CFE-15, 16, 18, 19, and 22.

The very small amount of particulate matter found in various areas in M122 was analyzed by X-ray fluorescence and emission spectrography to check the chemical analysis reported in APDA-CFE-16, page 30. Results of the analyses by X-ray fluorescence and emission spectrography are given in Table I. The particulate was obtained by alcohol- and water-washing the various locations, shown in Table I, and filtering the solutions. The purpose of the analyses was to determine if meltdown debris from melted subassemblies had been carried into other subassemblies in the reactor.

Table I indicates the high values for zinc and zirconium at the top grid but lower values at the bottom grid. It would be reasonable to expect that the bottom grid would have more meltdown particulate than the top grid if the material was pumped through the subassembly. It is possible, however, that the swollen upper end caps of some of the fuel pins in M122 contributed to the deposit of zirconium on the upper grid dimples. The reason for the high zinc value is not known at the present time. It should be noted also that the emission spectrographic analysis of particulate from all grids produced much lower zinc and zirconium values, but somewhat higher uranium values.

B. STUDY OF SUBASSEMBLY MELTDOWN

An earlier study* of the conditions attendant on the melting of subassemblies M127 and M098 during the incident of October 5, 1966 cited 95% blockage of M098 and 96% blockage of M127 as required conditions to initiate

* APDA-CFE-21, Section V. A. 1.

TABLE I - RESULTS OF ANALYSES OF PARTICULATE
RESIDUE ON SUBASSEMBLY M122 BY X-RAY
FLUORESCENCE AND EMISSION SPECTROGRAPHY

	Analysis, % by Weight					Emission Spectrography, All Grids
	X-Ray Fluorescence*					
	Top Grid	Bottom Grid	All Grids (1)	Grids (2)	Wrapper Can	
Uranium	≤0.5	≤0.1	≤0.1	≤2.0	≤0.5	3.7
Molybdenum	≤1.0	≤0.1	≤0.5	≤1.0	≤1.0	0.04
Niobium	≤0.5	≤0.1	≤0.1	≤0.1	≤3.0	
Zirconium	25	3.5	5.0	6.0	2.0	0.37
Copper	7.0	1.0	0.2	7.0	0.2	0.19
Nickel	3.0	6.0	6.0	6.0	5.0	3.74
Iron	44.5	70.0	68.0	66.0	65.0	74.70
Chromium	5.0	14.5	12.5	7.0	11.0	7.47
Magnesium						0.37
Cadmium	1.3	0.6	0.5	0.6	0.74	2.0
Titanium	0.1	0.3	1.0		0.8	0.19
Zinc	10.0	2.0	0.8	2.0	3.0	0.4
Lead	0.25	0.3	0.5		2.0	1.87
Silicon						1.87
Manganese	0.5	0.7	1.0	1.7	1.0	0.04
Tin						0.37
Aluminum						3.74
Silver						0.19

* The estimated accuracy for emission spectrography is $\pm 50\%$;
for X-ray, $\pm 100\%$.

melting at the power levels of operation on that date. These findings prompted the flow blockage test to establish that the first zirconium segment, loose in the inlet plenum in a battered condition, could sufficiently block coolant flow to the subassemblies to cause melting. A review of the earlier study was then conducted in light of the results of (1) computer calculations of temperature distribution in M098 and M127, (2) metallurgical examinations of M122* and (3) the flow blockage test. All of this information became available after the earlier study was made.

From the study review, it was concluded that the best estimate of flow blockage was 97% for both M127 and M098, based on temperature distribution in the subassemblies. Phase transformation at the midplane of M122, indicating that temperatures at this plane exceeded 1100 F, suggests 95% flow blockage in M122. The earlier study estimated 90% blockage. However, the absence of melting in M122 and the location of the molten zone in M098 (adjacent to M122) indicate 93% blockage in M122 as a better estimate. There was no subsequent evidence to alter the estimate of 70% blockage for M140, which in the earlier study was based on the core ΔT measured on the date of the incident.

The hydraulic tests showed that the battered zirconium segment could have caused 97% flow blockage in M127 and M097 when flattened. It is possible that the segment may have been flatter at the time of the incident and then was further distorted when handled in the inlet plenum. Simultaneous blockage of 95% in M122 and 70% in M140 was not confirmed by the hydraulic tests; however, additional flattening likely would have shown this to be the case.

* APDA-CFE-22, Section V. C.

V. MAINTENANCE

A. FAILURE OF CIRCUIT BREAKER FOR VENTILATION EQUIPMENT

Early in June, a flashover in the ventilation building power distribution cabinet (Figure 10) interrupted the power feed to the cabinet bus, which in turn interrupted the power feed to the following areas:

1. Equipment in the ventilation system and in the heating and cooling system for the above-floor area in the containment building
2. Equipment in the ventilation system for the below-floor area in the containment building
3. The blower furnishing cooling nitrogen gas for the neutron detectors
4. The compressor furnishing reactor cover gas flow to the fission product detectors
5. Lighting in the ventilation building.

The flashover accompanied a short circuit between two input terminals on the 480-volt, 3-phase, 100-amp circuit breaker feeding one of the refrigerant compressors for the below-floor cooling system. A loose bus link connection at one of the terminals started the incident. The loose connection caused excessive heating, which damaged the mounting plate bus link and terminal stud (see Figure 11) and charred the arc shield at the terminal, permitting current to arc over to an adjacent phase terminal. The arc caused the short circuit and tripped out the entire distribution cabinet.

Inasmuch as the damaged mounting plate carried four circuit breakers (see Figure 12) which would be inoperative during the repair period, it was necessary to make temporary connections to maintain power to the services affected. Power to the four services so affected was temporarily provided through other circuit breakers in the distribution cabinet that were already in use for other services and could safely accommodate the additional load.

The temporary connections required four hours to accomplish and the above-floor and below-floor cooling systems in the containment building were thus not in operation to cool the secondary shield wall at a time when the primary system was being heated for clean-up operations. At this time

**FIG. 10 POWER DISTRIBUTION CABINET
IN VENTILATION BUILDING**

**FIG. 11 DAMAGED MOUNTING PLATE
BUS LINK AND TERMINAL STUD**

**FIG. 12 REPLACEMENT CIRCUIT BREAKER
ON NEW MOUNTING PLATE**

primary system temperature was still only 350 F, and frequent monitoring gave assurance that the secondary shield wall temperatures were satisfactory.

At a later time, when the distribution cabinet was de-energized for installation of a new mounting plate and hardware and a new circuit breaker (Figure 12) to replace the damaged one, the containment building cooling systems were out of operation for about five hours, at a higher primary system temperature. There was no significant increase in temperatures in the secondary shield wall.

B. REPLACEMENT OF COOLING WATER PIPE FOR LOWER REACTOR BUILDING VENTILATION EQUIPMENT

The cooling water pipe servicing heat exchangers in the ventilation system for the below-floor compartment in the containment building was replaced because of major leaks discovered in a below-ground section of the pipe. The below-floor ventilation system consists of two separate duct loops with integral water-cooled heat exchangers and axial flow fans. Nitrogen from the below-floor compartment and from the area between the primary shield tank and the secondary shield wall is circulated through both duct loops, cooled in the heat exchangers, and returned to the below-floor area. The purpose of the ventilation system is to provide cooling to maintain temperatures below 150 F in the secondary shield wall and below 180 F in the operating floor. These temperature limitations assure that the shielding characteristics of the concrete in the secondary shield wall and in the operating floor are retained.

The heat exchangers along with the axial fans are located in sections of the duct loops outside the containment building. Each of the two heat exchangers is serviced by a 4-inch-diameter carbon steel pipe which carries cooling water from a general service water header. Observation of an excessive accumulation of water in the vicinity of the ventilation equipment outside the containment building led to an excavation in the area and the discovery of several corroded and leaking areas in the below-ground section of the cooling water supply line to the heat exchanger in the No. 2 ventilation loop. It was necessary to install a band patch over one large hole to stop a major gush of water. This cooling water pipe carried a protective covering and had been in service about 9 years.

A fire hose was connected between the general service water header and the No. 2 heat exchanger as a temporary measure to maintain cooling service, and the leaking pipe was abandoned. A new 4-inch-diameter carbon steel pipe line was then installed completely above ground, between the general service water source and the No. 2 heat exchanger. The cooling water line to the No. 1 heat exchanger, though appearing to be in good condition when uncovered along with the No. 2 loop pipe, was also replaced by new pipe in a similar manner.

The changeovers to the temporary fire hose and then to the permanent pipe were accomplished with very short interruptions in cooling service. The shutdowns were from one-half to one hour for each changeover. These short-duration interruptions of cooling were significant, inasmuch as clean-up of the primary system was in progress at this time, but produced no noticeable effects in secondary shield wall temperatures. It is intended that a rust inhibitor will be applied to the newly installed pipe and then heat insulation. No line heating will be required for winter operation because of the high velocity of water flow.

C. LIQUID WASTE LINE IN GOOD CONDITION

The below-ground section of the liquid waste line between the metering pump in the fuel and repair building (FARB) and the condenser cooling water discharge pipe was uncovered and found to be in good condition. This section of pipe, part of the liquid waste system described in Section VI. C and located as shown on the flow diagram in Figure 18 of APDA-CFE-22, was examined because of leaks discovered in the below-ground section of the liquid waste discharge line from the health physics building. The liquid waste discharge line from the metering pump is stainless steel pipe and is enclosed in a carbon steel secondary containment pipe. This secondary containment pipe appeared to be in good condition also.

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