APDA-CFE-31

ENRICO FERMI ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT MARCH 1969

R.E. Mueller

Assistant Project Engineer

United States Atomic Energy Commission

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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

and

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 U. S. Atomic Energy Commission under AEC Contracts No. AT (11-1)-865, Project Agreement 15. This series of reports provides detailed information on the nuclear testing, machinery dome, steam generators, pumps, flowmeters, level detectors, sodium sampling and development of the primary sodium system.

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

BACKGROUND

The Fermi reactor achieved initial criticality on August 23, 1963. An extensive series of nuclear tests was conducted at power levels below 1 Mwt, through 1965. A high power (200 Mwt) license was issued on December 17, 1965, and operation in excess of 1 Mwt was initiated on December 29, 1965. In January 1966, the power was raised in a series of steps to 20 Mwt. On April 1, 1966, power was first raised to 67 Mwt and on July 8, 1966, operation at 100 Mwt was initiated. On October 5, 1966, fuel damage occurred during an approach to power. Since that time the reactor has been shut down while the cause 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-28.

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

During March, the patch in the secondary containment pipe at the 14-inch line penetration was welded back in place. Several holes were drilled in the patch to allow inspection of the integrity of the primary pipe patch after sodium refill. The primary system was then refilled with sodium and fuel handling operations were conducted to achieve the core configuration required for insertion of the holddown mechanism inspection tool.

During the fuel handling operations, an attempt was made to transfer the No. 1 safety rod to a vacant safety rod position; however, there were indications that the safety rod guide tube lifted from the core support plate with the safety rod. The safety rod and guide tube were subsequently returned to their original position.

Following sodium refill, high torque developed on the No. 1 primary pump during pony motor operation. The pump is being disassembled to investigate the cause of the malfunction.

Cold-trapping of the primary, secondary, and fuel and repair building transfer tank sodium systems was conducted periodically to maintain all plugging temperatures well below the bulk sodium temperatures.

Sodium circulation in the steam generators was continued throughout the month. There was no water in the units. Periodic water tube nitrogen leak tests indicate no increase occurred in the previously observed small leakage detected in the No. 1 unit last month.

The erection of the new fuel transfer facility in the reactor building is about 35 percent completed with the support columns for the bridge and trolley structure now in place.

The piping for the three above-ground waste gas lines, which are replacing underground lines, has been completed and the new lines have been put into service.

The auxiliary fuel storage facility loading port and rotor shaft holes through the shield plug of the No. 2 equipment decay tank have been completed. The paint is now being removed from the inside walls of the decay tank.

Shipment has been received of an IBM digital computer for use in an on-line reactor malfunction detection system. Installation of the computer in the reactor control room has begun.

The steam-cleaning chamber gripper which malfunctioned last month has been repaired. The cause of its failure was foreign material on the cam operating shaft. The safety rod which had been caught in the gripper was returned to the sodium-filled transfer tank in the fuel and repair building. A leak in the steam line to the steam-cleaning chamber was also repaired.

Inspection of the debris trap in the converted arc-melt tool vacuum cleaner revealed that several pieces of weld debris were collected in it during the lower plenum vacuuming program.

The provisional operating license for the plant has been extended by the AEC to June 30, 1970.

II. SECONDARY CONTAINMENT PATCH REPLACED AT 14-INCH LINE PENETRATION

Last month, the patch in the 14-inch primary line penetration was replaced and its integrity checked by X-ray, dye-penetrant, mass-spectrographic, and ultrasonic examination (see APDA-CFE-30, Sec. II. D). Welding of the secondary containment pipe patch was begun on March 11 and completed March 19 (Figures 1 through 4). Interferences caused by the re-installation of a pipe hanger attached to the secondary containment pipe prevented mass-spectrographic testing of the patch weld; however, the final weld was checked by dye-penetrant and ultrasonic examinations. Four 2-inch-diameter holes were drilled in the patch to allow visual examination of the integrity of the primary pipe patch following sodium refill. The insulation and induction heating coils were re-installed over the secondary containment patch to avoid creation of a cold spot at this location. The viewing holes were not covered. An inspection at the end of the month, following sodium refill (see Section III. B), revealed no evidence of sodium leakage through the primary pipe patch weld. The viewing holes will be plug-welded when the inspection is completed, and the plug welds checked by dye-penetrant and ultrasonic examination.

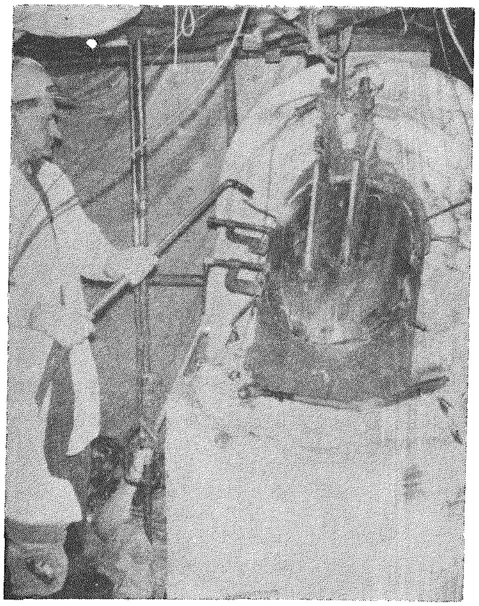

FIG. 1 HEAT-FORMING CONTAINMENT PATCH TO CUT-OUT

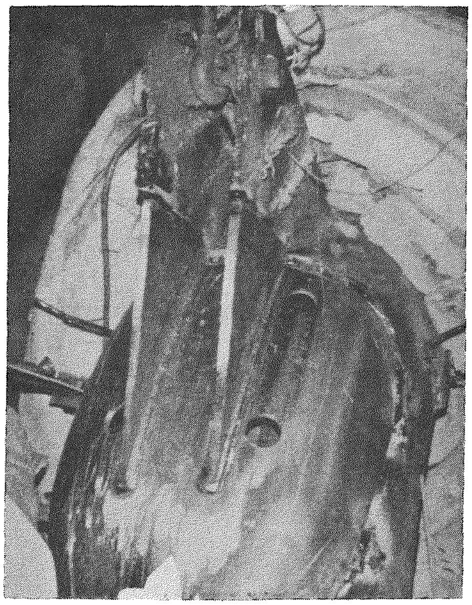

FIG. 2 CLOSE-UP OF VIEWING HOLES IN PATCH

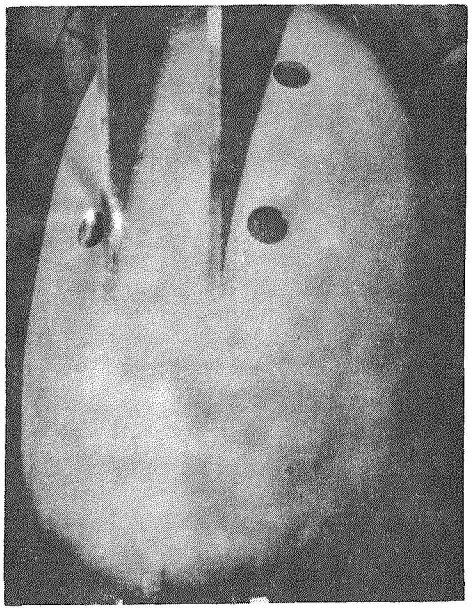

FIG. 3 DYE-PENETRANT TEST OF PATCH

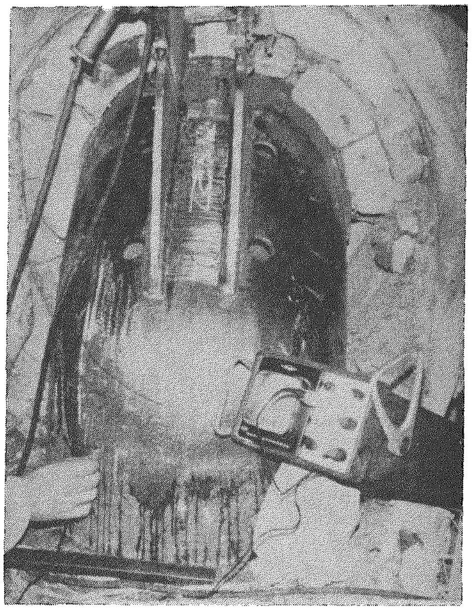

FIG. 4 ULTRASONIC TEST OF PATCH

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III. OPERATIONS

A. STATUS OF REACTOR SODIUM AND GAS SYSTEMS

The primary system was refilled with sodium at the end of the month following the closing of the 14-inch pipe secondary containment penetration. The sodium temperature was raised from 350 F to 450 F for cold-trapping. Intermittent cold-trapping of the secondary and transfer tank sodium systems was also performed. Twenty-six subassemblies were relocated within the reactor lattice to form the core cavity required for installation of the hold-down mechanism inspection tool.

B. PRIMARY SYSTEM SODIUM REFILL

The primary system was refilled with sodium on March 22 following replacement of the patch in the 14-inch containment line and use of the pipe heaters to thaw the sodium that was frozen in the lines. The reactor had been drained of sodium since November, 1968 (see APDA-CFE-27, Sec. III. C). The reason for refilling the system at this time was to permit subassembly relocations to be made with the offset handling mechanism operating in sodium for installation of the holddown mechanism inspection tool (see APDA-CFE-30, Sec. III). The reactor will be drained again following completion of the fuel handling operations.

The reactor was filled with sodium in the same manner as used for the earlier fill in June, 1968 (see APDA-CFE-23, Sec. III. A. 2). The sodium stored in the No. 2 and No. 3 primary storage tanks was heated to 350 F from its storage temperature of 300 to 325 F, the argon cover gas in the storage tanks was pressurized to 12 psig, and sodium was forced into the primary system through the recirculating line (see APDA-CFE-23, Figure 6). It entered the reactor vessel via the No. 1 intermediate heat exchanger and No. 1 pump tank through the 30-inch outlet line of Loop 1. Sodium levels in the three loops were then equalized through the IHX drain lines and the reactor vessel. Cover gas displaced from the primary system was discharged through the waste gas system. A total of about 30,000 gallons of sodium was transferred in 6 hours.

Following the sodium transfer, with the loop sodium at 350 F, the primary system plugging temperature was 220 F. In preparation for cold-trapping, the system temperature was raised from 350 F to 450 F using the heat input from the secondary pumps, the vessel and pipe heating. The plugging temperature rose to 260 F. The main primary pump motors were not energized at the time and the primary pump pony motors had to be used to achieve uniform mixing in the system. The lack of primary pump heat input resulted in a slower than normal rate of heat-up. Cold-trapping was further delayed when it was found that the inlet valve of the primary cold trap economizer would not function. The nitrogen atmosphere of the cold-trap room was subsequently purged with air to allow access to check the valve. Upon entry it was found that a pin was missing from the valve reach rod, allowing the reach rod to become uncoupled from the valve stem. The missing pin was found on the floor of the cold-trap room and was returned to the valve stem coupling. Following valve repair, the primary sodium was cold-trapped for two days at the end of the month. This reduced the sodium plugging temperature from 260 F to 215 F.

C. PRIMARY SODIUM PUMP MALFUNCTION

Following primary system sodium refill and heat-up, the three primary sodium pumps were run intermittently on their pony motors. The No. 1 pump was run several times, after which it developed high torque and could not be restarted. It could not be rotated by hand with 400 ft-lb of torque, an amount four times larger than the normal torque required to turn the pump shaft. The pump is being disassembled for investigation. Preliminary investigation with the motor uncoupled from the pump shaft indicates that the pump can be turned with normal force when the shaft is raised through its normal operating range. This indicates that the trouble probably is not in the pump itself, but somewhere in the motor, clutch, or gear mechanisms.

D. FUEL HANDLING

1. Transfer Operations

Following sodium refill, an extensive fuel handling program was conducted March 24 through 27. The purpose of the program was to achieve the core configuration required for insertion of the holddown mechanism inspection tool and subsequent holddown finger photographic survey. The principal subassembly movements made involved relocations in the core of subassemblies moved in September 1968 to open up the lattice for the zirconium segment removal program (see APDA-CFE-25, Sec. III. C).

Twenty-six subassemblies were relocated within the reactor: Thirteen core subassemblies already in the core were relocated to new core positions; six outer radial blanket (ORB) subassemblies were relocated to different ORB positions; two inner radial blanket (IRB) subassemblies were removed from storage in outer peripheral storage positions and placed in the core lattice; one IRB subassembly was relocated within the core lattice, and two core subassemblies were moved from the transfer rotor container (TRC) to the core lattice, while two other core subassemblies were transferred from the core to the TRC.

Figure 1 of APDA-CFE-25 shows the reactor loading configuration prior to the loading changes. The reactor configuration after the loading program is shown in Figure 5, and the subassembly movements are summarized in Table 1. As seen in Figure 5, subassemblies were arranged so that a wedge shaped core cavity was created for insertion of the holddown inspection tool. The inspection tool will be inserted in lattice position N07-P00. The umbilical cord cable path through the cavity will be along row P03 of the lattice. Subassemblies have been positioned around the safety rod guide tubes adjacent to the path to provide lateral support and guard against the possibility of their bending if interferences develop during the cable layout.

TABLE 1 - SUBASSEMBLY MOVEMENTS DURING MARCH

Subassembly No.	Type	Position Changes From	Position Changes To
M878	ORB	N08-P01 (ORB)	N09-P07 (ORB)
M870	ORB	N08-P00 (ORB)	N09-P06 (ORB)
M942	ORB	N08-N01 (ORB)	N08-P07 (ORB)
M997	ORB	N08-N02 (ORB)	N08-P06 (ORB)
M915	ORB	N09-P00 (ORB)	N07-P07 (ORB)
M863	ORB	N09-N01 (ORB)	N07-P06 (ORB)
M095	Core	P02-P03 (Core)	P01-P04 (Core)
M133	Core	N02-P03 (Core)	N01-P04 (Core)
M166F	Core	N06-P02 (Core)	P04-P03 (Core)
M125F	Core	N06-P01 (Core)	P05-P02 (Core)
M092	Core	N06-N03 (Core)	P05-P01 (Core)
M144	Core	N06-P00 (Core)	P04-P01 (Core)
M155F	Core	N06-N02 (Core)	P03-P01 (Core)
M047	Core	TR-03 (TRC)	P02-P01 (Core)
M036	Core	TR-11 (TRC)	P01-P01 (Core)
M119	Core	N03-N04 (Core)	P00-P01 (Core)
M076	Core	N02-N06 (Core)	N01-P02 (Core)
M002	Core	N06-N01 (Core)	TR-11 (TRC)
M137W	Core	N05-P00 (Core)	TR-03 (TRC)
M056	Core	N04-P04 (Core)	P00-P04 (Core)
M107	Core	N03-P04 (Core)	N01-P01 (Core)
M046	Core	P04-P05 (Core)	P01-P02 (Core)
M164F	Core	N01-N06 (Core)	N02-P01 (Core)
M382	IRB	N14-P01 (Blanket storage)	P00-P00 (Core)
M330	IRB	N14-P04 (Blanket storage)	P02-P00 (Core)
M304	IRB	N01-P05 (Core)	N02-P00 (Core)
SR90	Safety Rod	Attempted move from N02-P02 to P02-P02, but transfer was not completed.	

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.E. 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

Units 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)

Stringering 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

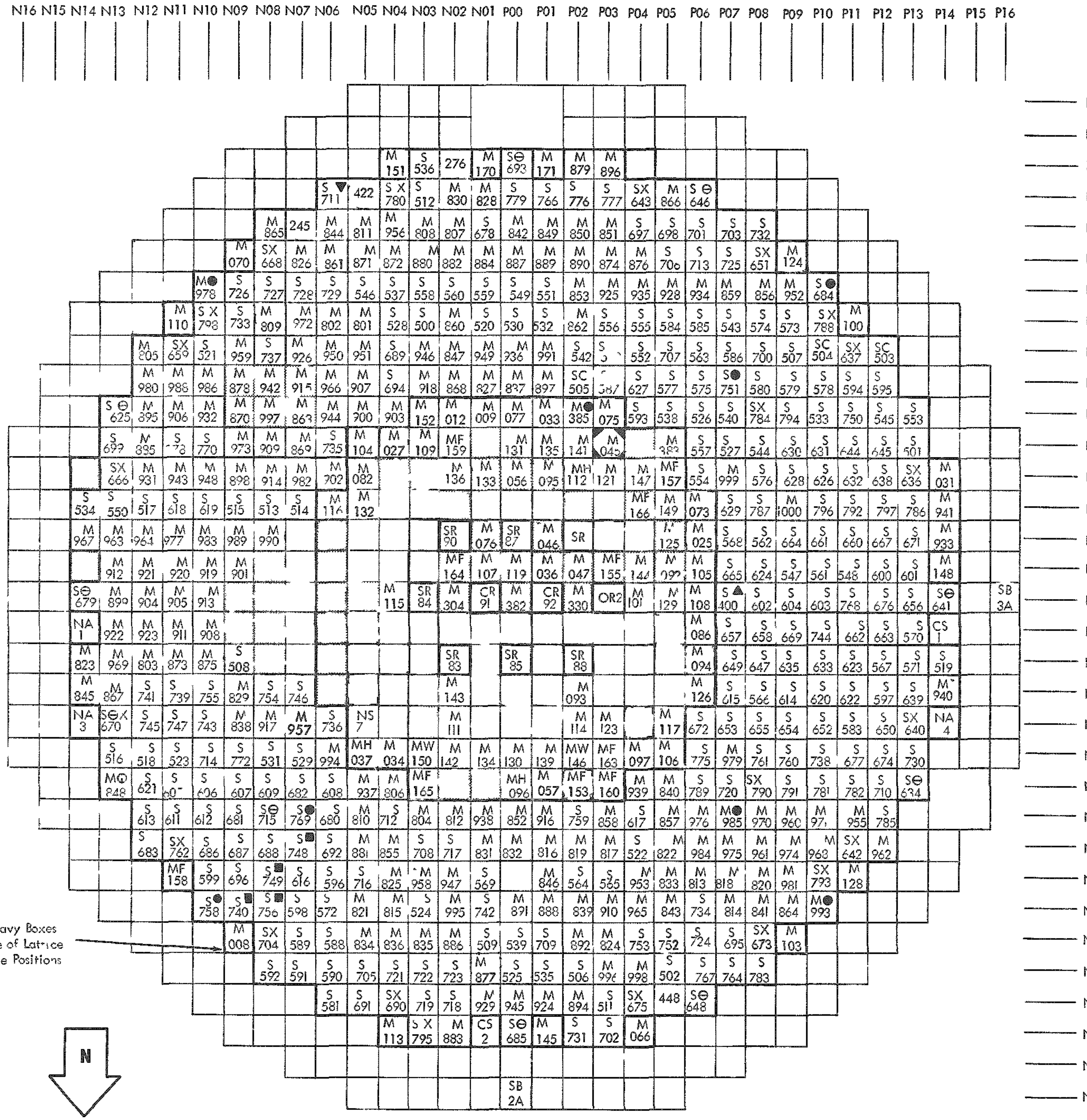

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FIG. 5 - CORE CONFIGURATION ON MARCH 27, 1969

It was also intended during the fuel handling program that safety rod No. 90, the No. 1 safety rod located in lattice position N02-P02, would be transferred to the empty No. 3 safety rod position P02-P02. However, when this was attempted, difficulties were encountered (see Section III. D. 2) and the rod was left in its original position.

The subassembly movements were made using the offset handling mechanism (OHM) and the rotating shield plug, operated in the automatic mode. The holddown mechanism (HDM) was in the up position. It was sometimes necessary to also make small manual adjustments in the plug and OHM azimuth positions, because during most moves there were less than the three adjacent subassembly camming services available to guide the subassemblies into the support plates during subassembly insertion. Insofar as possible the sequence of the loading program was planned to make optimum use of the camming surfaces. In no case did subassemblies fail to cam after minor adjustments were made in the plug or OHM positions.

Following completion of the fuel handling program, the sodium siphon line for draining the reactor vessel was installed. The siphon line was positioned 2 inches below the lower core support plate, and sodium will be drained down to that elevation. The welding, insulation, and heating of the siphon line had not been completed by the end of March. It is anticipated that this work will be completed the first week in April and the drain will follow.

2. Safety Rod Transfer Difficulties

On March 24, early in the fuel loading program, an attempt was made to transfer safety rod No. 90 from position N02-P02 to P02-P02 using the OHM; however, an interference encountered during OHM rotation indicated that the lower safety rod guide tube had probably been picked up together with the safety rod. The safety rod and guide tube were subsequently returned to their original position. The possibility of this occurring was anticipated and precautions were being taken at the time to rotate the OHM by hand to allow detection of any interference and to avoid damage if it occurred.

The design of a safety rod and its associated guide tube is shown in Figure 6. The safety rod is 68 inches long, whereas the lower guide tube, which separates from the upper guide tube when the holddown mechanism is raised, is approximately 91 inches long, slightly less than the 96-inch length of a subassembly; consequently, when withdrawing a safety rod from the reactor, the OHM carrying elevation is only 70 inches compared to the 98 inches used for a subassembly transfer. If a lower guide tube were stuck to a safety rod during transfer, it would extend down into the core almost two feet below the elevation of the surrounding subassembly handling heads. In the attempted safety rod transfer, calculations had previously been made of the positions at which interferences with subassemblies would occur if the guide tube was

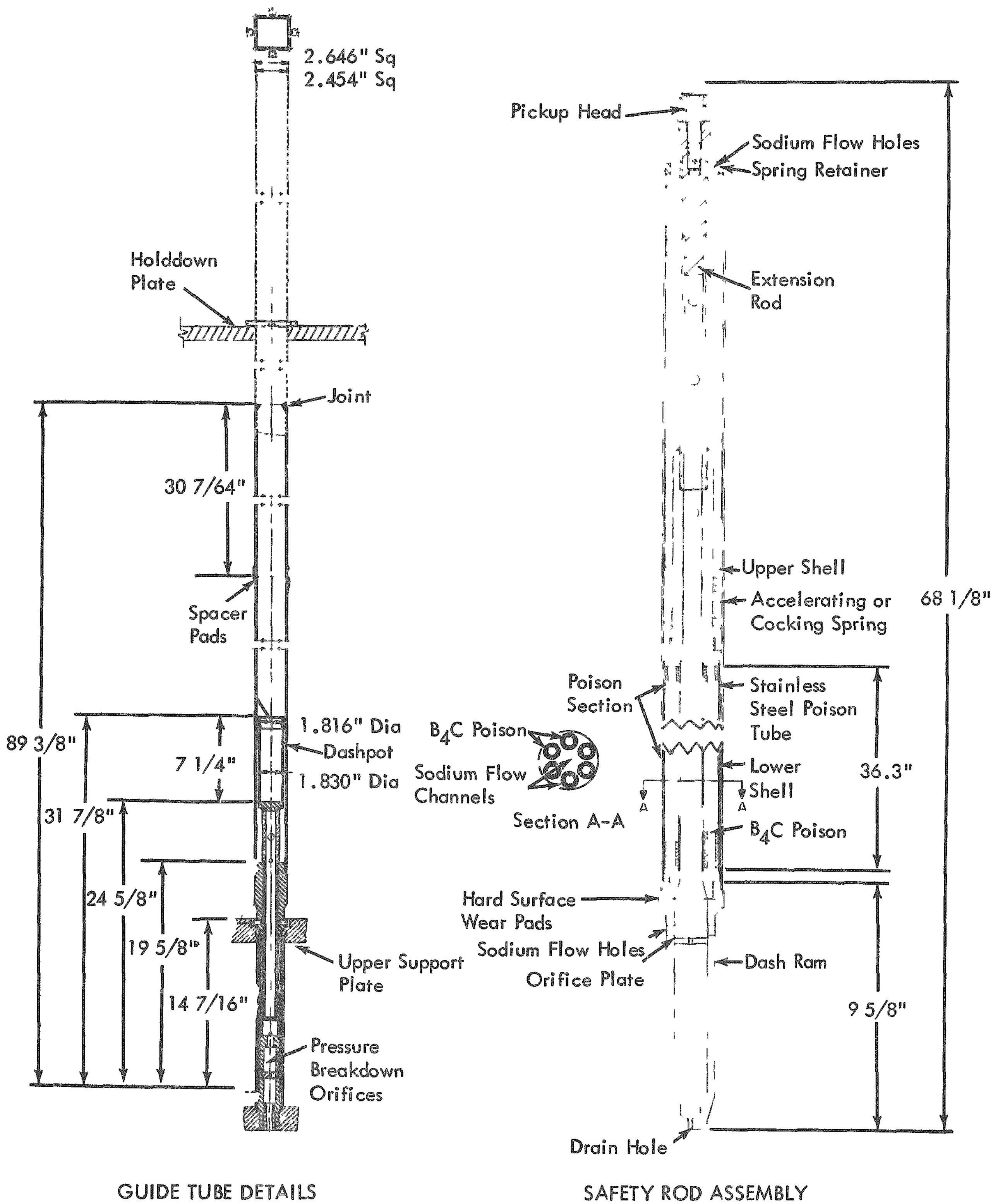

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FIG. 6 SAFETY ROD AND GUIDE TUBE

being carried with the safety rod. The OHM was rotated slowly by hand in two directions after the safety rod pick-up, and in each case movement was obstructed at the predicted locations. Although the vessel was filled with sodium and visual observations could not be made, this appeared to be rather positive indication that the lower guide tube had been lifted out with the safety rod. The safety rod and guide tube were subsequently lowered back into their original position N02-P02 and left latched in the OHM at a 2-inch elevation overnight. The rod and guide tube were fully deposited in the support plates the next day and the OHM delatched from the safety rod. Position indicators on the OHM showed that the rod delatched at the proper elevation.

All safety rods had been picked up with their drives and individually test scrammed shortly after the incident to verify that they were free (see APDA-CFE-3, Sec. III. C). However, since the No. 1 guide tube was in the vicinity of the melted subassemblies, it was felt to be prudent to again inspect the guide tube for straightness at this time during the rod withdrawal. Although the No. 1 rod had been withdrawn normally in the earlier tests, the tests were made with the HDM in the down position. The positive holddown restraint provided by the upper guide tube bearing against the lower guide tube may have been sufficient to straighten a slightly bent guide tube; it at least provided a counterrestraint on the lower guide tube during rod withdrawal.

The attempted transfer of the safety rod from guide tube No. 1 to guide tube No. 3 was not related to the fuel handling requirements for the holddown mechanism inspection program. The transfer was being made now, as a matter of convenience, because it is planned that the No. 1 lower guide tube will be replaced by a new tube at a later date to allow examination of the original guide tube. Therefore, following the attempted safety rod transfer, fuel handling to complete the core configuration for the HDM inspection was completed normally. The reactor will next be drained and the HDM inspection program will proceed as planned. In the meantime, a tool is being developed which can be inserted through the rotating shield plug for use in removing the rod from its guide tube. The rod stripping tool is being designed so that a differential pull can be applied between the rod and guide tube, i. e., such that a downward force can be exerted on the guide tube while the rod is simultaneously pulled upward. If the rod can be removed in this manner with a lifting force of less than 150 lb, it will be transferred to the No. 3 guide tube using the tool.* Replacement of the No. 1 lower guide tube will then be made as originally planned. Alternately, if the lifting force required to strip the rod is more than 150 lb, which could indicate rod damage as well, the rod and guide tube will be removed together using the OHM and be placed in the transfer rotor container.

* The weight of a safety rod in sodium is approximately 35 pounds.

E. STEAM GENERATOR PERFORMANCE

Secondary sodium circulation through the three steam generator units was continued throughout March, with no water in the tubes. The sodium temperature was maintained at approximately 400 F and fairly high flow rates of the order of 3.3×10^6 lb/hr were used. The purpose of this was to determine whether any increase would occur in the small gas leakage observed last month in the No. 1 unit (see APDA-CFE-30, Sec. V.C.3). Two leak tests of the No. 1 unit were made during the month with nitrogen at 650 psig in the water tubes. In both cases, measurements of the nitrogen in the argon cover gas over the sodium shell side of the unit indicate that the leakage is about 150 cm^3 per day of nitrogen (adjusted to 22 psig pressure differential). This leakage compares favorably with the 100 cm^3 per day leakage measured at 450 psig last month and indicates no increase in leakage has occurred. The No. 2 and No. 3 units remain tight.

The measured leak rate in the No. 1 unit is extremely small - it is equivalent to approximately a 3-mil-diameter hole in a tube sheet. If the unit were opened up to look for the leak, it is possible that the leaker tube or tubes would not be found due to the low leak rate. The hole may even be self-plugging: consequently, all units will be filled with water in the near future, and the leakage in the No. 1 unit will be monitored for any change in leakage rate using the hydrogen detectors. Following evaluation of these results, a decision will be made as to what further thermal shock and hydraulic tests will be conducted.

Throughout the month, the bulk sodium temperature in the secondary sodium systems was maintained at approximately 450 F in Loop 1 and between 350 F and 400 F in Loops 2 and 3; the plugging temperatures of Loops 1 and 2 fluctuated between 250 F and 275 F and no cold-trapping was done on either loop. The plugging temperature in Loop 3 rose to 315 F early in the month, and cold-trapping was conducted for 210 hours to reduce it to 230 F on March 12. The plugging temperature in Loop 3 has since risen to about 275 F, but no further cold-trapping has been performed.

F. GAS SYSTEMS PERFORMANCE

Since the last reported data (see APDA-CFE-30), the following primary system cover gas activity and gas analysis information have been obtained.

TABLE 2 - PRIMARY SYSTEM COVER GAS ACTIVITY

Location	Sample Date*	Gross Beta Activity, $\mu Ci/cm^3$
Reactor Cover Gas	3-6-69	1.57×10^{-6}
Reactor Cover Gas	3-10-69	3.70×10^{-7}
Reactor Cover Gas	3-14-69	1.73×10^{-6}
Reactor Cover Gas	3-20-69	2.75×10^{-7}
Reactor Cover Gas	3-27-69	2.75×10^{-7}
Primary Shield Tank	3-6-69	2.80×10^{-8}
Primary Shield Tank	3-14-69	4.70×10^{-8}
Primary Shield Tank	3-20-69	3.80×10^{-8}
Primary Shield Tank	3-27-69	2.55×10^{-8}

* Sodium Fill on March 22, 1969.

TABLE 3 - PRIMARY SYSTEM COVER GAS ANALYSIS

	Reactor Cover Gas (Argon), ppm by Volume		Primary Shield Tank Atmosphere (Nitrogen), ppm by Volume	
	Before Sodium Fill*	After Sodium Fill**	Before Sodium Fill*	After Sodium Fill**
Oxygen	Below 25	Below 25	600***	75***
Carbon Monoxide	Below 10	Not Measured	Below 10	Not Measured
Carbon Dioxide	90	Not Measured	15	Not Measured
Hydrogen	Below 4****	Below 4****	Below 2.5	Below 2.5
Helium	270	Below 8	Below 4	Below 8
Methane	Below 10	Not Measured	Below 10	Not Measured
Nitrous Oxide	← Not Measured →			
Nitrogen	440	950	← Remainder →	
Argon	← Remainder →		← Not Measured →	
Dew Point	← Not Measured →		-65 F	-55 F

* Sample taken March 13, 1969.
** Sample taken March 27, 1969.
*** Technical Specifications state 1000 ppm maximum.
**** The recommended maximum for reactor operation is 10 ppm.

G. RADIOACTIVE WASTE DISPOSAL

During March, 5,327 gallons of liquid waste was discharged to the north lagoon. The maximum specific activity in the liquid waste tanks was approximately 7.1×10^{-5} $\mu Ci/cm^3$. The total activity discharged to the lagoon was approximately 11 millicuries. All effluent released after dilution with the circulating pump discharge was below maximum permissible concentration (MPC).

The solenoid-operated isolation valves for the three 2500-gallon liquid waste storage tanks and the metering pump lift tank were also repaired this month. Leakage had been occurring between tanks. Since the repairs, argon leakage and waste liquid transfer between the tanks has been eliminated.

Approximately 1×10^9 cubic feet of gaseous effluent was discharged through the plant stack during March. The concentration of activity was very small and well below MPC after dilution in the stack blower and in the atmosphere.

Additional activity measurements of the water in the cutup and decay pools in the fuel and repair building gave maximum specific activities of 5×10^{-7} $\mu Ci/cm^3$ and 3.6×10^{-6} $\mu Ci/cm^3$, respectively.

H. FUEL AND REPAIR BUILDING SODIUM COLD-TRAPPING

Cold trapping of the fuel and repair building transfer tank sodium system was begun in February following repair of the transfer tank plugging indicator (see APDA-CFE-30, Sec. IV.B). By the end of February, the plugging temperature had been reduced from 330 F to 250 F. Intermittent cold trap operation was continued through the first two weeks of this month, resulting in further lowering of the plugging temperature from 250 F to 215 F. At this point, cold trap operation was stopped. Since then the plugging temperature has risen to approximately 260 F, the main source of the contamination being air introduced into the system during calibration tests of the fuel transfer facility in the fuel and repair building (see Sec. IV.A). However, no further cold trapping has been performed because a bellows failure which occurred last month at a valve in a bypass line of the cold-trap system has become progressively worse (see APDA-CFE-30, Sec. IV.B). The bellows is now frozen and the valve cannot be stroked to operate the cold trap system. No further cold trapping will be done until a replacement bellows is installed. The bulk sodium temperature of the transfer tank sodium system is presently being maintained at approximately 415 F.

The automatic temperature control system for the fuel and repair building cold trap, which had been out of service for some time, is now operational. Earlier problems of instability have been corrected by tuning of the control system and by elimination of a damper operated interlock.

IV. STATUS OF PLANT MODIFICATIONS UNDER CONSTRUCTION

A. FUEL TRANSFER FACILITY

Early in March, the drilling of holes was completed in the concrete operating floor of the reactor building. The holes, begun last month (see APDA-CFE-30, Sec. VIII. C), are for placement of anchor bolts for the steel support columns of the new fuel transfer facility (FTF). Twenty-eight 1-1/4-inch-diameter, 12-inch-deep holes were drilled (see Figures 34 and 35 of APDA-CFE-30). When the anchor bolts were inserted in the holes and their cinch wedges expanded (Figures 7 and 8) it was found that the bolts would not hold the 400 ft-lb of torque called for in the design without shifting in their holes. Molten lead was therefore poured down the holes around the bolts. After securing in this manner, over 500 ft-lb of torque could be contained. The erection of the four structural steel support columns has since been completed (Figure 9). With their erection, the new fuel handling facility in the reactor building is about 50 percent completed. It is anticipated that the bridge and trolley structure and the gripper and plug casks will be erected next month.

Calibration work on the companion FTF previously erected in the fuel and repair building was conducted this month; however, problems with the gripper and hoist mechanisms prevented the actual pick-up of a subassembly finned pot from the transfer tank. Calibration of the gripper elevation indicators and load cells was therefore made with the gripper empty. There was considerable difference between the remote elevation readout at the control panel, the local readout at the gripper cask, and the actual elevation measured with a tape measure. Modifications to correct this are in progress.

B. AUXILIARY FUEL STORAGE FACILITY

Last month the coring and burning of the two 12-inch-diameter holes for the two loading ports and the 6-inch-diameter hole for the rotor shaft of the auxiliary fuel storage facility (AFSF) had been completed through the shield plug of equipment decay tank No. 2 down to the elevation of the 7-inch-thick layer of steel plates which line the bottom of the plug (see APDA-CFE-30, Sec. VII. C). This month the holes through the bottom layer of steel were completed using the oxygen lance, as shown in Figures 10 and 11. The shield plugs have since been removed from the decay tank and are temporarily located on the operating floor while work progresses on removing the chemically resistant paint from the inside of the decay tank by means of steel shot blasting (Figure 12). This is being done to prevent the possibility of offgasing occurring during the use of the tank as a high temperature fuel storage facility. The paint removal has proven to be difficult because the paint is so firmly attached.

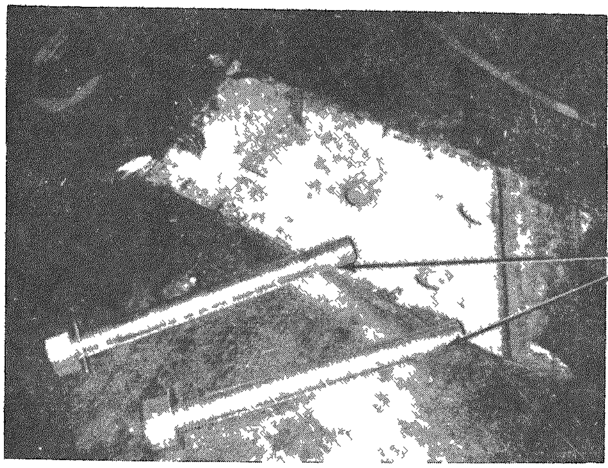

FIG. 7 ANCHOR BOLTS FOR FUEL TRANSFER FACILITY

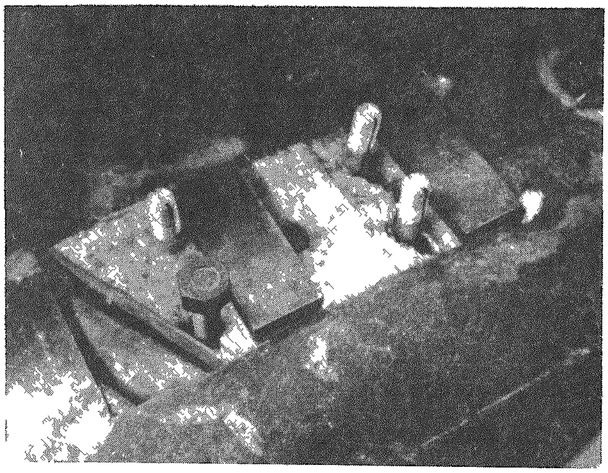

FIG. 8 ANCHOR BOLTS IN PLACE

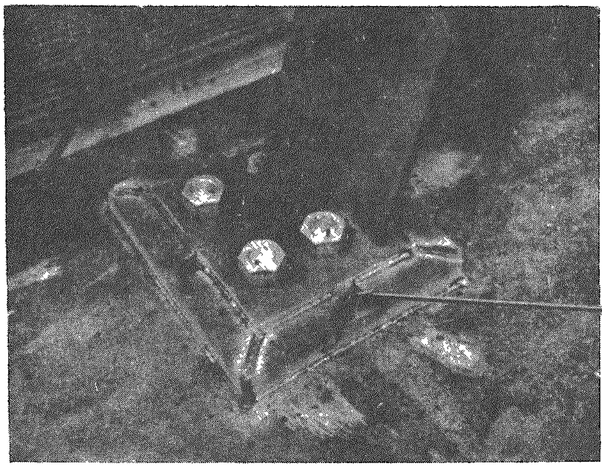

FIG. 9 SUPPORT COLUMN ERECTED (BEFORE GROUTING)

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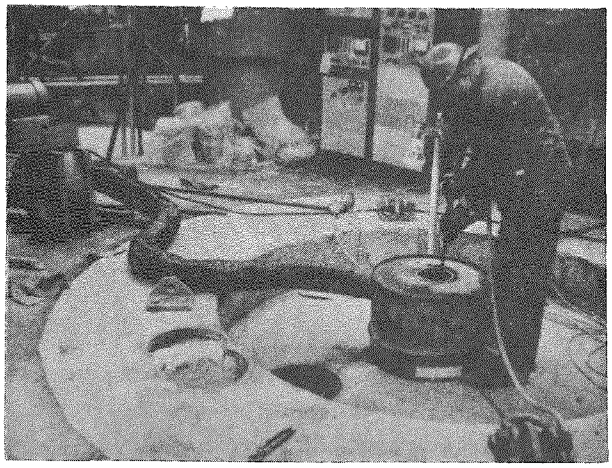

FIG. 10 USE OF OXYGEN LANCE TO BURN HOLE FOR ROTOR SHAFT OF AFSF

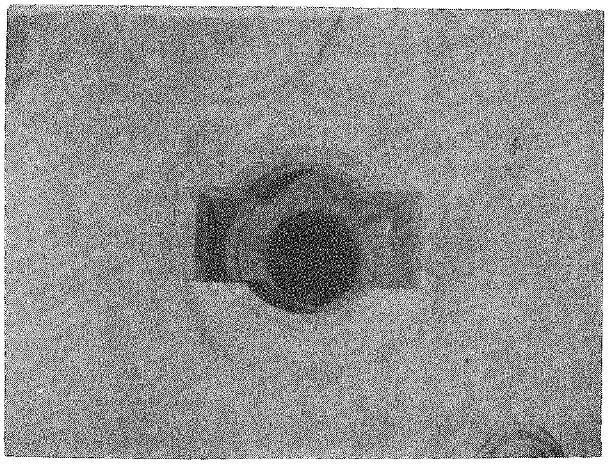

FIG. 11 SEVEN INCHES OF STEEL PENETRATED AT BOTTOM OF PLUG

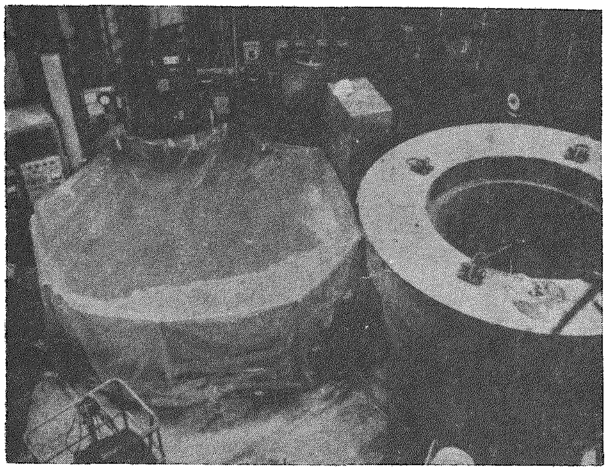

FIG. 12 STEEL SHOT BLASTING OF PAINT FROM DECAY TANK

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Use of liquid paint removers was attempted, but they were found to be unsatisfactory. One week has been spent on the job and the work is about 40% complete.

C. MALFUNCTION DETECTION SYSTEM

1. Background

An assessment of the events of the October 5, 1966 incident indicates that signals from the installed reactor instrumentation contained evidence that core conditions were abnormal, even at low power levels, during the approach to power.* If a detailed and rapid interpretation of output signals could have been made, the malfunction might have been detected before fuel melting occurred. In any event, the consequences of the fuel melting incident probably could have been minimized.

As a result of this experience, it was decided to install a malfunction detection system (MDS) at the Fermi plant to provide more refined interpretations of reactor output signals than heretofore possible, to ensure that such interpretations are made, and make results immediately available to the operator. The principal intent of the MDS is to allow the plant operators to make early detection of incidents, or hopefully the malfunctions leading to them, of a type similar to the October 5 incident; i.e., the incidents being considered are those characterized by slowly occurring malfunctions which do not constitute an immediate safety problem. Examples would be reactor startup with blockage of coolant flow channels, gradual crud buildup in subassembly coolant passages, fuel pin cladding failures, local fuel pin melting, and other similar malfunctions. The MDS is not intended for fast occurring incidents such as sudden flow blockage during full power operation.

2. Description of MDS

The principal component of the MDS is the malfunction detection analyzer (MDA). It consists of an on-line digital computer used to analyze reactor temperature, power, flow, and control rod position signals, among others. From these data, four principal calculations will be made: (1) reactor reactivity balances will be made, and (2) temperature rise across subassemblies will be calculated. Comparison with predicted values will then allow the determination of whether any reactivity or temperature anomalies exist. Then (3) fission product activity in the reactor cover gas will be analyzed using the signal from the existing fission product precipitation detector; and (4) the sodium coolant will be analyzed for fission products using the signal from a delayed neutron detector system soon to be installed.

The analyses will be made at frequent periodic intervals on a real time basis. The computer will generate an error signal based on comparison between the predicted and actual values of each of the four parameters.

* "Report on the Fuel Melting Incident in the EFAPP on October 5, 1966," APDA-233.

If any of the anomalies are outside of prescribed limits, the computer will notify the reactor operator of the discrepancy by means of audible and visual alarms. The frequency of the calculations will be automatically increased if any anomalies are detected.

Besides serving its primary function as a malfunction detection analyzer, the computer will also be used for routine analysis of over one hundred other plant instrumentation signals. These calculations will be done on a shared-time basis when the computer is not being used as an MDA. The supplemental calculations will be accomplished through a time sharing system which will enable priorities to be assigned to each of the calculations. The calculations will then be done in order of their priorities as time allows. The supplemental calculations will involve general plant performance and system monitoring analyses, such as burnup and heat balance determinations, analysis of nuclear and plant test data, and other special data analyses to support efficient plant operation.

3. Description of MDA

Manual, analogue, and digital systems were considered for use as the malfunction detection analyzer in the MDS. A digital MDA system was chosen for reasons of accuracy, reliability, and much greater flexibility over the other systems. The computer chosen for the MDA is an IBM Series 1800 computer. It is being leased from IBM. The computer is very flexible and can be easily expanded as the needs of the plant dictate. It has 24 K core storage and 4 μ sec cycle time. All programming is done in Fortran. It can scan data signals at the rate of 100 points per second, and a total of 124 different signals can be accommodated. The computer has seven major auxiliary components: the processer controller, analogue input, disc storage unit, card reader/punch, line printer, keyboard, and scope (Figure 13). Initially, the computer will receive close to 100 signals from the reactor, coolant loops, and auxiliary equipment. These signals can be displayed in graphic or tabular form on command by the operator. The computer will also log much of this information automatically at designated intervals by means of the line printer. The programs for the malfunction and plant performance calculations will be permanently stored in the computer and automatically called in off the magnetic disc as required.

The main malfunction detection computation programs are those relating to anomalous reactivity and subassembly temperature rise calculations. In the anomalous reactivity calculations, a reactivity balance will be made approximately every three seconds using the power, flow, rod position, and core inlet temperature signals. From these data individual reactivities will be calculated as a function of time for the measured reactivity given by the rod position and for each of the expected reactivities - kinetic reactivity, inlet temperature, burnup, and feedback reactivity. The difference between the measured and expected reactivities will then give the anomalous reactivity.

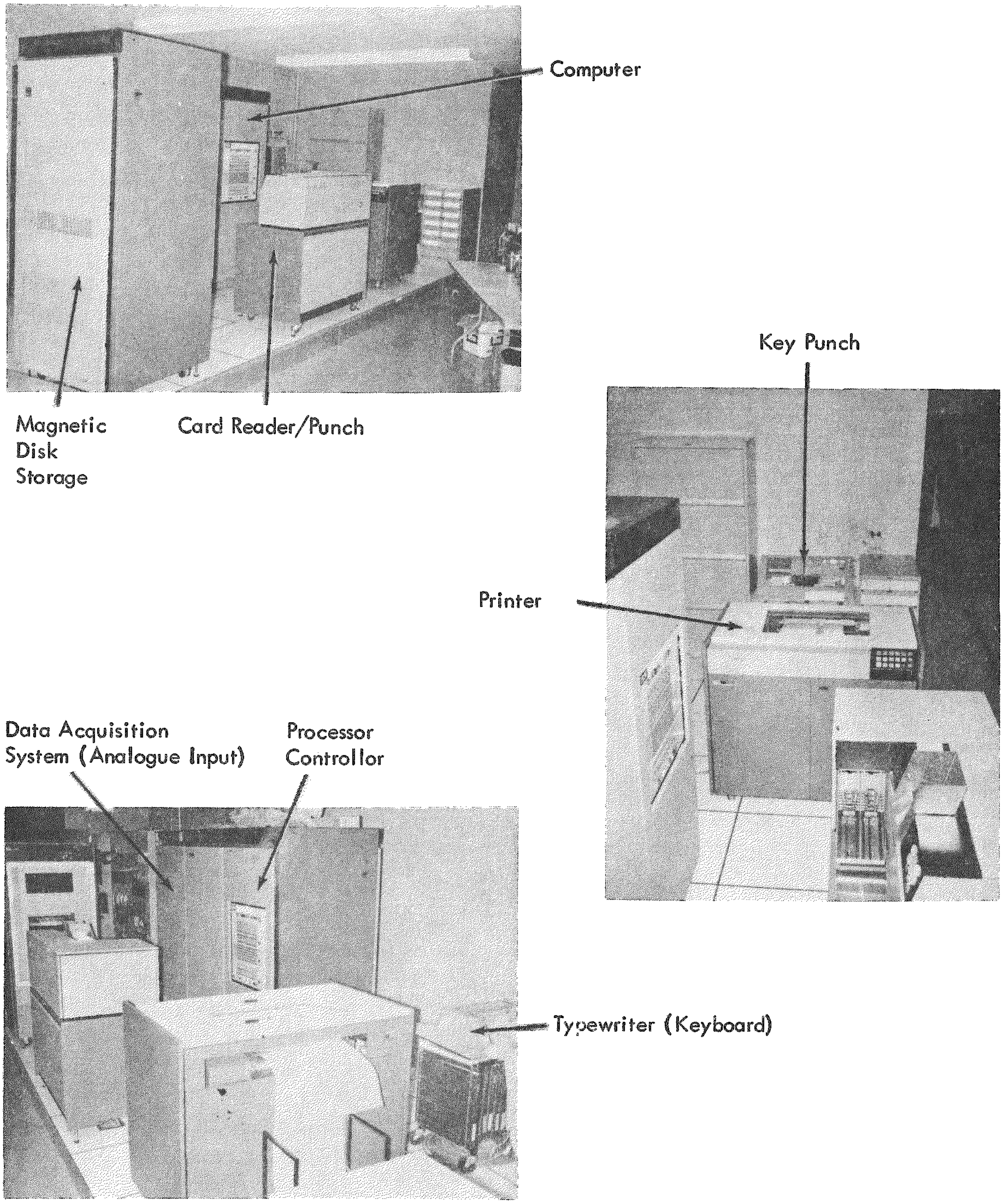

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FIG. 13 MDA COMPONENTS AFTER FLOOR ERECTION

It will be an indication of core anomalies caused by overheating, changes in core feedback characteristics, or slowly progressive fuel melting.

Program checks have been made using the data from the October 5 incident. The calculations indicate that reactivity anomalies as small as two or three cents can be detected using the data available from the plant instrumentation, and using the reactor rod, temperature, power and flow coefficient calibrations.

The temperature rise across subassemblies during reactor operation will be calculated using the inlet and outlet temperature signals received from the 25 percent of the core subassemblies which are instrumented. These values will then be compared with predicted rises, calculated from the reactor power and flow signals, to obtain anomalous temperature rise error signals. The subassembly outlet temperature calculations will be repeated approximately every 10 seconds. Program checks indicate that temperature discrepancies as small as 2 F may be detected using the plant instrumentation signals and the core power distribution determined during low-power nuclear testing.

The cover gas and coolant fission product activity checks will be calculated at less frequent intervals than either of the two checks above.

4. Installation of MDA

The MDA computer arrived at the plant this month and is presently being installed. It is being erected in the main control room on a special raised platform made of aluminum and steel. It is located along the east wall of the control room behind the main control panel. Construction of the platform is about 80% complete. The cut-outs for the power and signal cable feeds to the computer cabinets remain to be completed. The power requirements for the computer are rather large, amounting to about 8 kw. The motor-generator set presently in the switchgear room, which was formerly used to supply power to the plant simulator, is being used for the MDA power supply. The simulator will be transferred to the plant diesel bus.

During the past several months, various modifications have been made in the plant instrumentation circuitry to condition the instrument output signals for use in the MDA. The MDA requires an analogue signal input in the millivolt range. Conditioning is required because some of the signals needed are presently digital, others are pneumatic, and still others are in the volt range. The core thermocouple temperature signals are being transferred from the multipoint pen recorders and connected directly to the MDA data acquisition multiplexer system. In some cases recorder signals cannot be fed directly to the MDA and constant potential slide wires are being installed for retransmission of their data to the MDA data acquisition system.

To date, the safety rod delatch and control rod position signal conditioners have been designed and built. The circuit designs have also been completed and components ordered for many of the other signals. The safety rod delatch signal conditioners will provide a signal to the MDA within about 5 milliseconds from the time of actuation of the delatch light switches. The signals will be used in the MDA for evaluating safety rod performance during automatic scram. They will also serve to identify the malfunctioning rod in the event of a single rod drop, which, at-power is presently obscured by the automatic -dn/dt scram of the remaining rods.

The design of the control rod position signal conditioners required rebuilding the divider network which drives the rod position recorder. Operational amplifiers have been added to reduce the loading on the position transmitters. The control rod signal conditioning network assembly has been mounted and connected and preliminary testing indicates that an overall accuracy of better than 0.2% can be expected over the full range of rod travel. For the time being, approximate calibration settings have been made in order to minimize the amount of rod operation required during final calibration.

The programming work for the four principal malfunction detection calculations has been completed and demonstration runs were made on an IBM Series 1800 computer at another installation. Simulated analogue input voltages were used for the instrumentation signals to verify the adaptability of the programs. It is planned that a signal simulator will also be added to the computer at the plant to allow checking of the programs on the plant computer prior to reactor startup and any subsequent program changes.

V. PLANT MAINTENANCE

A. NEW WASTE GAS LINES PUT INTO SERVICE

The conversion of buried waste gas piping to overhead piping was completed during March. As reported last month (see APDA-CFE-30, Sec. IV.A), two of the lines replaced were waste gas lines 21 and 22. These are the waste gas discharge lines between the fuel and repair building and the waste gas building, and between the waste gas building and waste gas stack, respectively. Line 26, the cold-trap room ventilation relief to waste gas, also was replaced. All weldments were completed and the lines put into service the last week in March. They satisfactorily passed mass spectrometer leak tests and pneumatic pressure tests. The new lines have worked satisfactorily since they were put into operation.

B. STEAM-CLEANING CHAMBER GRIPPER REPAIRED

In February, the steam cleaning chamber gripper failed to delatch from safety rod No. 89 during a transfer operation in the fuel and repair building (see APDA-CFE-30, Sec. VI. D). The cause of the malfunction could not be immediately determined after preliminary examination in the dry loading tunnel. The gripper and attached safety rod were subsequently withdrawn into the steam-cleaning chamber for temporary storage until further corrective measures could be taken.

The gripper was returned to the dry loading tunnel the first week in March. The safety rod was removed from the gripper by partially disassembling the gripper manually behind the temporary shield wall erected last month. The safety rod was then deposited in a transfer pot in the tunnel and the gripper removed from the tunnel for complete disassembly and detailed inspection (Figure 14).

The failure to delatch was determined to have been caused by a sticky cam operating shaft (Figures 15 and 16). The lift rod, which actuates the gripper fingers, was found stuck in the up position (gripper closed) and accounted for the inability to unlatch from the safety rod. Once the lift rod was freed, the gripper functioned satisfactorily. Foreign material from steam-cleaning operations is thought to have caused the problem. Gripper operation was restored by polishing the shaft and bearing surfaces of the components in the gripper and bushing using emery cloth and India stone. Following reassembly, the gripper was cycled numerous times using a dummy subassembly and dummy safety rod handling head (Figures 17 and 18). No failures to latch or delatch occurred. The gripper was reinstalled on the gripper cable, and the steam-cleaning machine restored to service

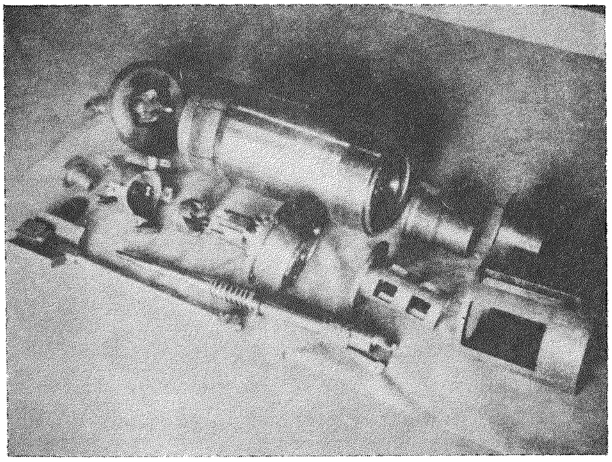

FIG. 14 STEAM-CLEANING CHAMBER GRIPPER DISASSEMBLED

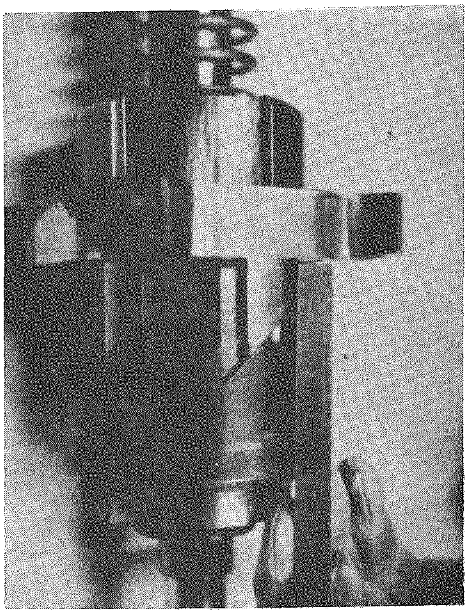

FIG. 15 GRIPPER CAMS CLOSED

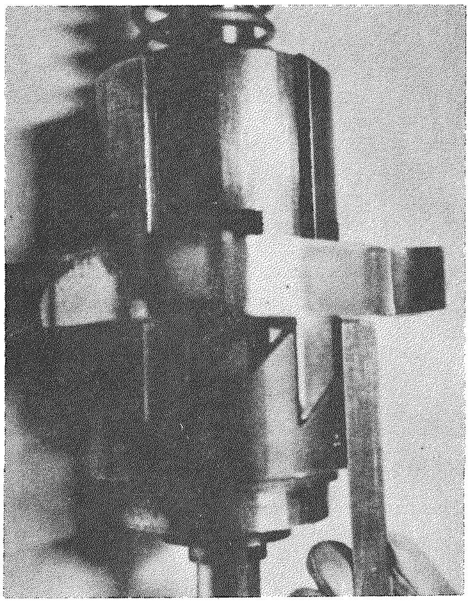

FIG. 16 GRIPPER CAMS OPEN

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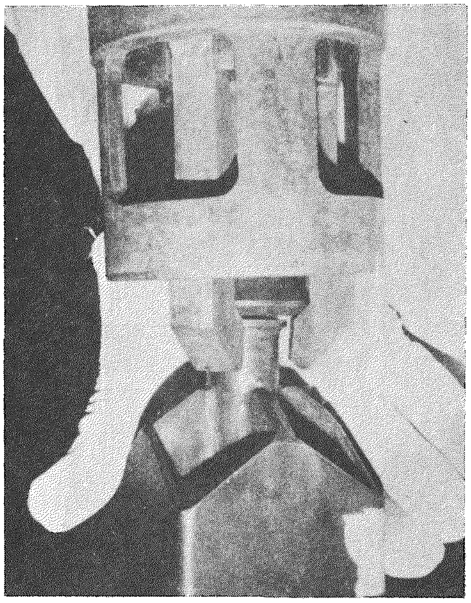

FIG. 17 TESTING GRIPPER ON DUMMY SUBASSEMBLY HANDLING HEAD

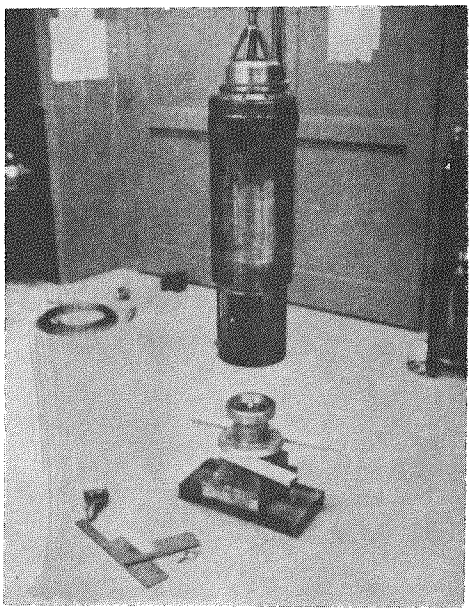

FIG. 18 TESTING GRIPPER ON DUMMY SAFETY ROD HANDLING HEAD (INSIDE DUMMY POT HEAD)

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on March 11, 1969. The safety rod was deposited back in the transfer tank in sodium in a new safety rod pot. No difficulties were observed with the operation of the gripper during the transfer. The safety rod will subsequently be returned to the reactor.

The steam-cleaning chamber gripper had successfully handled 34 subassemblies in September, 1968 prior to the occurrence of sticking (see APDA-CFE-25, Sec. III. D). However, since then, it had been idle for over 18 weeks. Because of this it is planned that a periodic maintenance program for the gripper will be established. The program will consist of cleaning the gripper components after a specified number of steam-cleaning cycles have been completed. Test pickup of dummy subassemblies will also be made before a new fuel handling program is initiated. This program is expected to minimize gripper malfunctions caused by gripper component stickage resulting from successive steam-cleaning operations.

C. REPAIR OF ROTATING JOINT IN PIPING TO STEAM-CLEANING CHAMBER

During the steam-cleaning of safety rod No. 89 last month following the gripper malfunction, a water alarm was received from the steam-cleaning chamber room. Upon inspection, the combination argon, water, and steam supply line which enters the top of the cleaning chamber was found to be leaking water at the Barco rotating joint used to permit rotation of the machine over the three access ports located in the steam-cleaning chamber room. The leak was repaired this month by tightening the sealing ring. Rotation of the machine had caused the ring to loosen and the setscrews on the ring were found to be backed off. The cut end setscrews previously used were replaced with point end setscrews, and backed up with lockscrews. No further leakage from the joint has been observed.

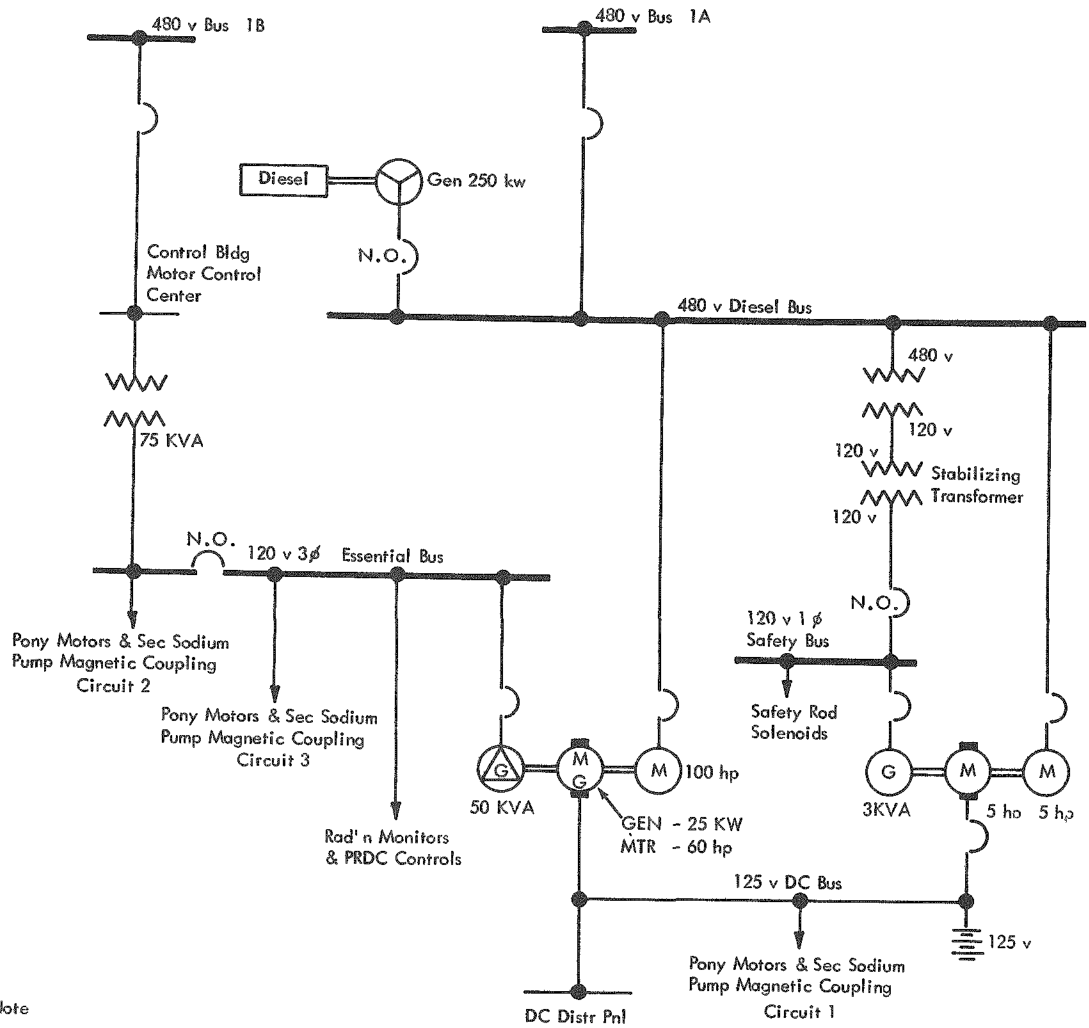
D. BURNED RELAY COIL FOR ESSENTIAL BUS CIRCUIT BREAKER REPLACED

On March 4, 1969, a burned closing coil was found in a closing relay to a circuit breaker in the a-c alternator feed to one of the two plant essential buses. Since the affected circuit breaker is normally energized and closed, the power supply to that section of the essential bus was not interrupted. However, repair of the defective relay required the bus to be de-energized for a short time. In addition, the defective relay blew the control fuses in the d-c feed to the affected circuit breaker and in the feed to the tie breaker (normally open) used to connect the two sections of the essential bus in emergencies. This prevented the automatic throwover of the first bus to the transformer feed of the second bus in the event of the failure of the a-c alternator to deliver power to the first bus.

The emergency auxiliary power system of the plant, used to carry loads classified as essential, is shown schematically in Figure 19. The essential bus is divided into two sections designated as A and B. Essential loads which must be capable of operating continuously upon loss of normal station power are supplied with emergency power derived from the station battery. The loads on essential bus A are in this category. Essential loads which can tolerate a temporary loss of power are supplied with emergency power from a diesel-driven alternator-generator connected to the 480 volt a-c diesel bus. Essential bus B is supplied from the 480 volt control building bus through a step-down transformer.

Essential bus A is normally supplied with power from a three unit motor-generator (MG) set. The 480-volt diesel bus, normally supplied from a 480 volt a-c distribution bus, feeds a synchronous a-c motor which in turn drives an a-c generator supplying essential bus A. The synchronous motor also drives a d-c generator charging the battery. On loss of power to the synchronous motor, its starter opens and the d-c generator automatically becomes a motor powered by the batteries. It continues to drive the a-c generator, maintaining a constant source of power to essential bus A. In the meantime, the diesel bus is re-energized within a minute by an automatically started diesel engine driving an a-c alternator supplying all loads connected to the diesel bus. With the power supply to the a-c motor thereby restored, the a-c motor starter is reclosed and normal operation is resumed. In the event the a-c alternator of the three unit MG set fails, cutting off power to essential bus A, an under voltage relay automatically activates sectionalizing breakers which isolate the alternator and tie essential bus B to essential bus A in less than half a second. The two essential buses can also be tied together manually when maintenance requires shutdown of the three unit MG set.

The burned relay closing coil and two 30-amp blown control fuses to the essential bus sectionalizing circuit breakers were found by the plant operators during routine plant patrol. The defective relay coil had been in service about 9 years. To allow repair of the defective relay, the MG set was manually shut down and a manual throwover of essential bus A from MG set feed to bus B transformer feed was made.



Note

All circuit breakers are normally closed
unless otherwise identified as normally open (N.O.)

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FIG. 19 EMERGENCY AUXILIARY POWER SYSTEM

VI. MISCELLANEOUS ACTIVITIES

A. ANALYSIS OF HOLDDOWN FINGER PHOTOGRAPHS TAKEN IN MOCK-UP FACILITY

Last month, during mock-up tests of the holddown mechanism inspection tool, photographs were taken of a dummy holddown finger block and a lined grid, using the equipment which will actually be used in the reactor (see APDA-CFE-30, Sec. III. C). The dummy holddown finger block photographs were analyzed this month. The graphical method was used in which the apparent curvature of each finger in a row in the photograph versus finger position is plotted (see APDA-CFE-30, Sec. III. B). A linear best-curve fit was made to the data points and the curvature of one finger was found to lie off the line by a large amount, indicating that the finger was displaced from its normal position. A 52-mil displacement was predicted, based on the amount of finger deviation from the linear rate of curvature change in the plot. Measurements on the dummy finger block gave an actual 60-mil displacement for the finger.

The lined grid photograph, made to verify the assumed linearity of the pin position versus curvature plot for perfect pin alignment, has not been analyzed yet; however, because of the close agreement between predicted finger displacement and actual finger displacement in the dummy holddown finger block photograph analysis using a linear approximation, it appears that linear curve fitting is valid.

B. EXAMINATION OF DEBRIS TRAP IN VACUUM CLEANER

The debris trap in the converted arc-melt tool vacuum cleaner (see APDA-CFE-29, Sec. III. B) used to vacuum the lower plenum following zirconium segment removal, was opened and examined this month. The debris found in the trap is shown in Figure 20. Six pieces of spherical and flattened metal about 1/4-inch in diameter were found in addition to several smaller pieces. The debris was identified as weld slag resulting from the arc melting of the zirconium screwheads and welding of the screw shanks on the flow cone (see Sec. II. E of APDA-CFE-27 and 28).

C. PROVISIONAL OPERATING LICENSE EXTENDED

The provisional operating license for the Enrico Fermi Atomic Power Plant has been extended by the AEC to June 30, 1970. The license extension was applied for November 21, 1968, the application amended on January 22, 1969, and the order extending the license became effective March 3, 1969.

FIG. 20 DEBRIS FROM DEBRIS TRAP OF VACUUM CLEANER

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Director

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(Box X, Oak Ridge, Tennessee)
F. L. Culler
J. H. Devan
D. Gardiner
J. White

Oak Ridge National Laboratory
(Box Y, Oak Ridge, Tennessee)
R. E. MacPherson, Jr.

Pacific Northwest Laboratory, BMI
E. Astley (5)

Power Reactor Development Company
(1911 First Street, Detroit, Mich. 48226)
A. S. Griswold

Southwest Atomic Energy Associates
(Box 1106, Shreveport, La 71102)
J. R. Welsh

United Nuclear Corporation
(Box 1583, New Haven, Conn)
A. Strasser (2)

Westinghouse Electric Corporation
(Box 158, Madison, Pa 15663)
J. C. R. Kelly, Jr. (2)

Westinghouse Electric Corporation
(Box 158, Madison, Pa 15663)
C. A. Anderson

USAEC-UKAEA Exchange
UKAEA
Reactor Group Headquarters
Risley, Warrington, Lancashire
England

J. Stephenson (12)

USAEC-EURATOM Exchange
EURATOM
53, Rue Belliard
Brussels 4, Belgium
A. deStordeur (10)

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Via Mazzini 2
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