ENRICO FERMI ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

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

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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 was recorded in a series of technical reports completed by Atomic Power Development Associates, Inc., for the U.S. Atomic Energy Commission under AEC Contracts No. AT (11-1)-865, Project Agreement 15. These reports provided 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 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 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 shutdown 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, which is the final report of the series.

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 reactor vessel remained partially drained of sodium during June, while the photographic survey of the holddown mechanism was completed. Detailed analysis of the photographs obtained show that the holddown mechanism was not damaged by the incident and that it can be safely lowered onto the new core after it is loaded.

Inspection of the pressure drop orifices in the No. 1 steam generator following isolation tests showed, as expected from the hydraulic tests conducted last month, that several orifices were raised off their seats. The orifices were being held up by small particles of scale. The No. 2 steam generator remained inerted with nitrogen on the water tube side during the entire month. Retorquing of the steam manifold cover plate on the No. 3 steam generator failed to eliminate the gasket leak encountered last month; the water side of the unit was subsequently vented, purged, and filled with nitrogen to await replacement of the defective steam manifold gasket.

Checkout of the new fuel transport facility in the reactor building continued following installation of the floor valve over the exit port in the reactor building. Alignment of the gripper cask to the transfer cask and of the transfer cask to the floor valve has been made; qualification testing of the fuel transport equipment is now in progress.

Erection of the auxiliary fuel storage facility in the No. 2 equipment decay tank in the reactor building is nearly complete. The manually operated worm gear drive and indexing equipment for the storage rotor were installed, and the heater and thermocouple cable leads which pass through the hollow rotor drive shaft were connected at the junction box located on top of the rotor shaft. Installation of the purge gas control station for the facility, the major work remaining on the facility, is about fifty per cent complete.

Construction of the temporary simulator for the malfunction detection analyzer (MDA) was completed; operational checkout and debugging of the device was also completed. Installation of the delayed neutron detector system which will be later connected to the malfunction detection analyzer is complete.

Work continued on the modifications being made to the primary sodium purity monitoring system in the cold trap room of the sodium service building. The new electromagnetic pump for the plugging indicator and flowmeter for the hydrogen detector have been installed. Work is now in progress to re-route the outlet piping from the plugging indicator so that it is no longer in parallel with the cold trap.

Leak rate testing of the new boot seal for the primary shield tank at 2 psig revealed the leakage to be larger than desired. Modifications were made to some of the clamp ring segments of the upper clamp ring located on the primary shield tank wall to correct the problem; the measured leakage is now within acceptable limits.

Shipment was received of a guide tube removal tool, intended for use in removing and replacing the No. 1 safety rod lower guide tube presently in the core. Mock-up testing of the tool has begun. Although some minor difficulties have been encountered in the use of the tool, the tool is nearly ready for use in the reactor.

II. OPERATIONS

A. STATUS OF REACTOR SODIUM AND GAS SYSTEMS

The primary system remained partially drained of sodium throughout the month for completion of the holddown photographic survey and in anticipation of replacement of the No. 1 safety rod lower guide tube. The temperature of the primary system was maintained between 350 F and 380 F by the pipe and vessel induction and resistance heaters.

Loops No. 2 and 3 of the secondary sodium system remained filled with sodium at approximately 350 F. Loop No. 1 was drained of sodium prior to opening the No. 1 steam generator for inspection of the pressure drop orifices in the water header.

No cold trapping of the primary, secondary, or fuel and repair building transfer tank sodium systems was performed. No movement of subassemblies was made during the month.

The below-floor area of the reactor containment building and the primary cold trap room atmosphere remained purged with air throughout the month. The purity levels of the primary shield tank nitrogen atmosphere have returned to their normal values following the shield tank boot seal replacement.

B. GAS SYSTEMS PERFORMANCE

Since the last reported data (APDA-CFE-33), the following primary system cover gas activity and gas analysis information have been obtained.

TABLE 1 - PRIMARY SYSTEM COVER GAS ACTIVITY

<u>Location</u>	<u>Sample Date</u>	<u>Gross Beta Activity, $\mu\text{Ci}/\text{cm}^3$</u>
Reactor Cover Gas	6-5-69	3.97×10^{-6}
Reactor Cover Gas	6-11-69	1.49×10^{-6}
Reactor Cover Gas	6-18-69	1.32×10^{-6}
Reactor Cover Gas	6-25-69	1.49×10^{-6}
Primary Shield Tank	6-4-69	2.38×10^{-8}
Primary Shield Tank	6-25-69	5.50×10^{-8}

TABLE 2 - PRIMARY SYSTEM COVER GAS ANALYSIS

	<u>Reactor Cover Gas (Argon), ppm by volume*</u>	<u>Primary Shield Tank Atmosphere (Nitrogen), ppm by volume*</u>
Oxygen	Below 25	25**
Carbon Monoxide	Below 10	Below 10
Carbon Dioxide	20	20**
Hydrogen	Below 4***	15
Helium	Below 4	Below 4
Methane	Below 10	Below 10
Nitrous Oxide	←————Not Measured————→	
Nitrogen	570	Remainder
Argon	Remainder	Not Measured
Dew Point	Not Measured	-38 F*****
Shield Graphite Temperature	312 F	312 F

* Sample date was June 11, 1969.

** Technical Specifications state 1000 ppm maximum during reactor operation and during shutdown whenever primary system temperatures exceed 400 F.

*** The recommended maximum for reactor operation is 10 ppm.

**** Technical Specifications state less than -35 F, except during maintenance when values as large as +80 F are allowed if primary system temperatures are less than 400 F.

C. STEAM GENERATOR PERFORMANCE

Feedwater pressure drop tests performed on the No. 1 steam generator during May indicated that not all of the poppet-type orifices in the water tubes had reseated following the isolation tests made in April (see APDA-CFE-33, Sec. II. B). On June 20, the No. 1 secondary sodium loop was drained of sodium and the steam generator was allowed to cool down in preparation for the removal of the water manifold cover and inspection of the poppet-type orifices. The unit was dismantled on June 23. Inspection revealed that six orifices (out of a total of 1200) on the south of the unit were being held raised off their seats because of small particles of scale lodged between the orifices and the downcomer tubes, as shown in Figure 1. The number of orifices found raised was consistent with the hydraulic pressure drop data obtained last month. Six additional orifices, also on the south side of the unit, were found with dirt particles caught in the central orifice hole, as shown in Figure 2. The particles were removed and will be further identified in laboratory examinations. While the steam generator was dismantled, the opportunity was taken to conduct a shell metal thickness survey on the area of the steam generator shell adjacent to the water tube previously found failed and repaired by plug-welding in October 1968 following detection of a relatively large leak (see APDA-CFE-25, Sec. III. G. 3). This was done to determine whether the leak had produced shell wastage from any resultant sodium-water reaction. The wall thickness was checked by ultrasonic measurements, and the results of the test show no metal wastage in the area checked.

It is planned that bubble tests of the No. 1 steam generator will be conducted next month, while it is dismantled, in an attempt to locate the source of the small leakage observed in the leak tests made over the past several months (see APDA-CFE-31, Sec. III. E).

The No. 2 steam generator remained vented of water and inerted with nitrogen at 20 psig on the tube side, and filled with 350 F sodium on the shell side, throughout the month.

The No. 3 steam generator was refilled with water at 600 psig on June 4 to determine whether the leakage observed last month at the steam manifold cover plate had been eliminated following retorquing of the cover plate head bolts (see APDA-CFE-33, Sec. II B); however, steam leakage occurred and the water side of the unit was subsequently vented, purged, and filled with nitrogen. Because retorquing failed to eliminate the gasket leak, the steam manifold gasket will be replaced with a new gasket after repairs to the No. 1 steam generator are completed. In the meantime, the unit is being kept filled with 350 F sodium on the shell side and inerted with nitrogen at 20 psig on the tube side.

No cold trapping was performed on any of the three secondary sodium loops during June. The plugging temperature in loop 1 fluctuated between

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**FIG. 1 SCALE PARTICLE CAUGHT UNDER ORIFICE IN
WATER TUBE OF NO. 1 STEAM GENERATOR**

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**FIG. 2 DIRT PARTICLE LODGED IN ORIFICE HOLE
OF NO. 1 STEAM GENERATOR**

270 F and 290 F until the time of the drain; the bulk sodium temperature during this time was maintained at approximately 350 F. The plugging temperatures of loops 2 and 3 varied between 260 F and 290 F and 265 F and 280 F, respectively, throughout the month with the bulk sodium temperature being maintained at approximately 350 F.

D. FUEL AND REPAIR BUILDING SODIUM SYSTEM

No fuel-handling or steam-cleaning operations were performed in the fuel and repair building during the month of June. No cold-trapping of the transfer tank sodium system was conducted. The sodium plugging temperature was 272 F at the beginning of the month and rose to 274 F by the end of the month. The bulk sodium temperature in the system was maintained at approximately 450 F throughout the month.

E. RADIOACTIVE WASTE DISPOSAL

During June, there was no liquid waste discharged from the plant to the north lagoon. The volume of liquid waste generated was small and could be adequately retained in the hold-up storage tanks.

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

Data are available from an environmental radiation survey made during the four-week period from March 20 through April 17, 1969. A total of 84 environmental samples were collected and analyzed during this period and a summary of the activity levels found is as follows:

- The gross alpha and gross beta activity levels of airborne dust samples obtained from the reactor site area and from areas away from the reactor (to indicate the vicinity background) were found to be the same in both locations - approximately $2.5 \times 10^{-15} \mu\text{Ci}/\text{cm}^3$ for the alpha activity in both cases and $15 \times 10^{-14} \mu\text{Ci}/\text{cm}^3$ for the beta activity, thus indicating no airborne activity above background on the plant site. The alpha activity levels are essentially identical to those obtained in similar measurements made in 1967 and the beta levels are about double the 1967 values.
- Measurements of precipitation samples taken from the reactor site area and from locations away from the reactor site also showed no activity levels above background on site, the gross alpha and gross beta activity levels in both locations being approximately 0.30 mCi/mile² and 12.4 mCi/mile², respectively. The alpha values are essentially the same as those obtained in 1967 and the beta activities are about double similar 1967 values.

- The specific activity of water samples taken from lakes and creeks near the plant site and from the drinking water supply of neighboring cities showed no significant change in gross beta activity from similar measurements made in 1967, the activity levels in both cases being approximately 8×10^{-9} $\mu\text{Ci/ml}$ and 3×10^{-9} $\mu\text{Ci/ml}$, respectively.
- No significant change in the gross beta and gamma activity levels was found in samples of Lake Eire fish, when compared to previously analyzed samples.

F. NEUTRON SOURCE SHIPMENT

A new radioactive antimony rod for the antimony-beryllium neutron source used for reactor start-up was delivered to the plant in June. It has been irradiated in the General Electric Test Reactor (GETR) located at Vallecitos, California. The antimony rod is approximately 0.7 inch in diameter by 25 inches long and is sealed inside a tantalum can with a helium bond. The antimony portion of the source fits inside a hollow beryllium cylinder (2.4 kg beryllium) contained inside a standard core subassembly wrapper can. When assembled in the core, the source produces neutrons by means of the 1.8 Mev (γ, n) threshold reaction in beryllium, initiated by the 2.1 Mev decay gammas from the antimony-124 isotope in the antimony rod.

The antimony source rod was irradiated in the GETR for one 28-day cycle at 50 Mwt. Eight similar gamma-emitting antimony rods for the neutron source have been previously acquired since the time of the initial reactor start-up in 1963. Periodic replacement of the source is required because antimony-124 decays with a 60-day half-life and source regeneration is negligible at low power levels. Following irradiation of the new source, it was cooled for approximately six weeks to allow the shortlived, high-level antimony-122 and tantalum-183 activities to decay. The source then was dimensionally inspected, leak-tested, and ultrasonically cleaned before shipment. A Fermi-type handling head was installed and it was shipped to the Fermi plant in the PRDC shipping cask. At the time of shipment the source contained a total activity of 19,200 curies, approximately equally divided between antimony-124 and tantalum-182. The source is presently stored in the shipping cask at the plant. It will subsequently be moved into the cut-up pool in the fuel and repair building for additional leak testing, and from there to the transfer tank prior to insertion in the reactor following sodium refill of the primary system.

III. HOLDDOWN MECHANISM PHOTOGRAPHIC SURVEY

A. SURVEY OF SOUTHWEST SECTOR

Following completion of the holddown mechanism inspection tool modifications initiated at the end of May, the inspection tool was reinserted in the reactor on June 12 to photograph the holddown fingers in the southwest quadrant near the core edge, the only location for which complete photographs were not obtained in the May survey (see APDA-CFE-33, Sec. III). To make the survey, the Sunscope and one articulated viewing light were installed in the No. 5 rotating plug access penetration, the Mark II and Mark III borescopes plus another articulated light were installed in the No. 3 access penetration, and the inspection tool with camera attachment was inserted in the No. 4 access penetration. The plug was rotated to the 119° azimuth position required to view the southwest sector.

Similar to the surveys made in May, the photographs of the holddown fingers in the southwest sector were handicapped by poor lighting. This resulted in difficulties in focusing the camera, long exposure times, and pictures of less than the desired quality. However, the photographs still contained sufficient contrast and clarity to allow identification of the finger locations and accurate analysis of finger alignment.

Near the end of the photographic survey (the survey had been essentially completed) the spherical mirror became caught between the arms of the bail located on top of the inspection tool post and could not be rotated in either direction. The shield plug was subsequently rotated to retract the inspection tool service hose from its laid-out position in the core cavity in preparation for removal of the tool from the reactor. At this time it was observed through one of the borescopes that the top portion of the argon purge hose to the light chamber was severed and hanging loose from the bail. The purge hose breakage had probably occurred earlier during the initial service hose layout and was probably the reason for the poor lighting experienced during the picture taking.

When the inspection tool was withdrawn from the reactor through the rotating shield plug penetration, a 7-inch-long piece of the rubber purge hose was torn from the mirror post. The torn piece of hose dropped between the handling heads of the outer radial blanket subassemblies located adjacent to lattice position N11-P03 (see APDA-CFE-31, Figure 5). The first attempt to retrieve the piece of broken hose from the reactor using the spine tool (see APDA-CFE-15, p 18) was unsuccessful when the hose dropped out of the jaws of the gripper and came to rest on the upper fuel support plate. It was successfully retrieved on the second effort. Match-up of the severed ends of the hose showed that no pieces had been left behind in the reactor.

B. CONCLUSIONS FROM ANALYSIS OF PHOTOGRAPHS

The holddown finger photographs obtained in May and June was analyzed for finger alignment (one finger relative to the surrounding fingers) using the graphical technique described in APDA-CFE-30, p 19. Sufficient photographs were found of adequate clarity to allow accurate analysis of the complete holddown finger array. Analysis of the photographs showed that no finger bending or distortion exists to within the $\pm 1/16$ -inch test tolerances, which are well within the maximum allowable finger misalignment of $\pm 1/8$ inch. The finger over melted subassembly M098, the finger in the southeast quadrant considered most likely to have been bent during the incident, was found from the average of a number of measurements to be only 0.006 inch west and 0.012 inch north of its nominal position. The largest single measurements of apparent finger displacement in the southeast quadrant were 0.031 inch displacements in the northerly direction for each of the fingers over subassembly positions N03-P01 and N02-P04 (see APDA-CFE-31, Figure 5). The average displacement for both of these fingers was only 0.009 inch north, however. The largest single measurement of apparent finger displacement anywhere in the core was 0.038 inch west on the finger over position P05-P00 in the southwest quadrant, the average of the measurements for this finger being 0.015 inch west.

On the basis of the holddown finger photographic analyses, it has been concluded that the holddown mechanism was not damaged by the fuel melting incident and that it can be safely lowered following reloading of the core. Confirmation that the control and safety rod guide tubes are straight will be gained from an inspection of the guide tube adjacent to the fuel melting region (see Sec. V. B) and from mechanical tests of the rods before the resumption of reactor operation.

IV. STATUS OF PLANT MODIFICATIONS UNDER CONSTRUCTION

A. FUEL TRANSFER FACILITY

Assembly of the new fuel transport facility (see APDA-CFE-33, Sec. IV. A) in the reactor building is complete and final qualification testing of the equipment is now in progress. This month the new shield-sealing flange (see APDA-CFE-30, Sec. VIII. B) for the exit port in the reactor building was set permanently in place in the floor around the exit port (Figure 3) and the base plates for the exit port floor valve were grouted into the operating floor and sealed in place (Figure 4).

A conduit box on the front of the plug cask, which was found to interface with the above-floor extension of the wall of the primary shield tank adjacent to the exit port, was repositioned to eliminate the interference. The six-ton floor valve, which is filled with blocks of depleted uranium for shielding purposes, was then set on the base plates and aligned to the exit port as shown in Figures 5 and 6. Finally, the fuel transfer cask was fitted to the floor valve and aligned and the gripper cask was set on top of the transfer cask and its guide wheels and lifting jacks aligned. During the qualification testing of the completed transport facility to date, only a few minor faults--a broken limit switch housing on the gripper cask and an incorrectly assembled plug cask top limit switch--have been found.

B. AUXILIARY FUEL STORAGE FACILITY

Erection of the auxiliary fuel storage facility (APDA-CFE-30, Figure 19) is essentially complete except for final gas line connections and mechanical testing of the facility. Figure 7 shows the manually operated worm gear drive assembly that was added at the top of the rotor shaft to turn the fuel storage rotor. The drive assembly turns the storage rotor through a drive sprocket at very low angular rates, corresponding to less than 1 rpm. As the rotor turns, a second series of worm gears operates a mechanical digital-type counter that will be calibrated to indicate the rotational position of the rotor to an accuracy of $\pm 3/8$ -inch following initial calibration and checkout. After assembly of the drive mechanism on the rotor shaft, the rotor was raised off the three temporary support brackets inside the decay tank by turning a large, heavy-duty nut which is threaded and fitted to the uppermost portion of the rotor shaft. The rotor was raised 1-3/8 inches to its normal operating position. At this elevation an approximate 2-inch clearance exists between the top of a subassembly storage pot and the insulation added on the bottom of the decay tank shield plug. The heavy-duty circular nut at the top of the rotor shaft bears the weight of the storage rotor and transmits its weight to a spherical roller thrust bearing. The weight is then transferred to the structural members of the decay tank walls and finally to the tank support structure.

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**FIG. 3 SHIELD SEALING FLANGE SET
IN FLOOR AROUND EXIT PORT**

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**FIG. 4 BASE PLATES FOR FLOOR VALVE
SET IN PLACE**

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**FIG. 5 FLOOR VALVE BEING LOWERED
ONTO BASE PLATES**

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FIG. 6 FLOOR VALVE SET IN PLACE

Figure 8 shows the heater cable leads from the heater stack, the thermocouple leads, and the 1/4-inch argon purity sampling line being brought up from the storage facility through the center of the hollow rotor shaft. Figures 9 and 10 show the termination of the heater and thermocouple leads at the junction box which was mounted on a plate attached to the rotor shaft above the floor level. Flexible cables were used to allow the shaft to rotate. The thermocouple leads from the junction box were connected to a multipoint temperature recorder located just above the fuel storage facility.

The present work effort on the fuel storage facility is devoted to installing an argon purge and gas control station on top of the plug adjacent to the junction box. High-purity argon from the plant inert gas system enters the storage tank at its base through an existing 1/2-inch line. The gas flows from the lower to upper chamber of the storage facility through sleeves provided in the false floor and is exhausted from the storage facility through an existing 3/4-inch line that leaves the storage area just below the shield plug. The gas then passes through the reactor building operating floor to a valve pit located adjacent to the plug.

Previously a temporary quick connect-disconnect line was used, when required, to make a connection between the valve pit and an existing waste gas line located in the containment building floor for exhausting the decay tank atmosphere to the waste gas disposal system; however, for storage facility use, the argon station is being added so that more adequate pressure control over the tank atmosphere can be maintained and to facilitate purging of the tank. An existing 2-inch line which penetrates the shield plug near its outer boundary is fitted for pressure sensing and the pressure inside the tank will be maintained at 2 inches WCG above atmospheric pressure by an automatic pressure regulating valve. This is being done to prevent leakage of above-floor air into the facility. The system will exhaust to the waste gas stack through a line to be added from the argon purge station around the wall of the containment building which will connect with the building's ventilation exhaust system. By the first of July the erection of the purge station should be complete. The facility should then be ready for the mechanical qualification tests required before used fuel can be moved into the storage facility.

C. MALFUNCTION DETECTION ANALYZER

Construction of the temporary signal simulator (see APDA-CFE-33, Sec. IV. C) for the malfunction detection analyzer (MDA) was completed in June. The simulator, which provides complete simulation of the anomalous reactivity calculation and partial simulation of the subassembly outlet temperature monitoring calculation (see APDA-CFE-31, Sec. IV. C. 3), was mounted on the control room wall behind the MDA computer in the MDA operating area. It will also simulate various process-interrupt signals and contact-operate output signals. The cables between the temporary simulator and the computer for simulator input to the computer were routed down through the floor of the control room to the relay room, through a switchbox, and

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**FIG. 7 WORM GEAR DRIVE AND INDEXING
EQUIPMENT FOR STORAGE ROTOR**

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**FIG. 8 CABLE PENETRATIONS THROUGH
HOLLOW ROTOR SHAFT**

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**FIG. 9 TERMINATION OF LEADS AT
JUNCTION BOX**

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FIG. 10 CLOSE-UP OF JUNCTION BOX

back up to the computer. This arrangement was used to allow the computer to be run during reactor operation using either the signals from the permanent simulator (see APDA-CFE-32, Sec. IV.C) to be installed later in place of the temporary simulator) or the actual reactor output signals. Signal conditioners were installed in the simulator to condition the pot signals to the form required for use in the computer. Use of the temporary simulator in conjunction with the MDA computer is expected to be a valuable tool for verifying and making refinements to the MDA computer program.

Operational checkout and debugging of the temporary simulator was completed this month. The anomalous reactivity calculation was performed on the MDA computer using the simulator input signals and the results were displayed on the line printer. Although the simulator and anomalous reactivity program appeared to be operating satisfactorily, the temperature conversion program used to extract temperature values from the thermocouple millivolt signals was found to be working improperly. This was corrected and the temperature conversion program is now operational.

The electrical circuits which convert the output of the MDA computer to a form compatible with the storage display oscilloscope have also been completed and will be tested in the near future. Upon completion of verification and preoperational testing, the MDA installation will be ready for use.

D. DELAYED NEUTRON DETECTOR

Assembly of the delayed neutron detector system (see APDA-CFE-33, Sec. IV. E.) was completed during June. The cables from the four Reuter-Stokes proportional detectors located below the reactor building floor were secured and sealed in place in a clamp plate set in the bottom of the cavity penetration manhole in the operating floor. The manhole cavity was then filled with lead shot as shown in Figure 11. The preamplifiers for the detectors were mounted on a support bracket located adjacent to the manhole and the cables were connected to the preamplifiers as shown in Figure 11. A support frame for the associated delayed neutron detector nuclear instrument module bin was also fabricated and erected on the operating floor and the bin containing the pulse amplifiers, discriminators, and count rate meters for each of the two operating channels was assembled. Chart recorders for the two channels will be located in the reactor control room following operational check out of the system electronics by Reuter-Stokes personnel. The delayed neutron detector system will be calibrated when reactor operation is resumed. When successful operation of the system is demonstrated the system will be connected to the MDA.

E. PRIMARY SODIUM SYSTEM COLD-TRAP ROOM MODIFICATIONS

The modifications being made in the primary sodium cold-trap room to improve the efficiency of the sodium purity monitoring system (see

116169

**FIG. 11 COMPLETED FLOOR PENETRATION OF THE CABLES
FOR THE DELAYED NEUTRON DETECTOR**

116269

**FIG. 12 ELECTROMAGNETIC PUMP FOR PLUGGING
INDICATOR CIRCUIT IN PRIMARY COLD TRAP ROOM**

APDA-CFE-33, Sec. IV. D) were about 75% complete by the end of June. The new electromagnetic pump for the plugging indicator circuit was installed together with the new flowmeter for the hydrogen detector in the sodium sample coil inlet line. The principal work remaining is rerouting of the plugging indicator outlet piping to change the circuit so that it no longer operates in parallel with the primary cold trap. Figure 12 shows the piping upstream from the plugging indicator with the electromagnetic pump installed. Figure 13 shows the sodium sample coil inlet line before flowmeter installation and Figure 14 shows the line with the flowmeter installed. Figure 15 shows the outlet piping from the plugging indicator that still remains to be rerouted. The radiation level in the work area in front of the cold trap and temporary lead shadow shield has been approximately 6 mr/hr during the modifications, in good agreement with expectations.

F. VOLTAGE REGULATING TRANSFORMER FOR POWER RANGE SIGNAL AMPLIFIERS

The voltage regulating transformer ordered (see APDA-CFE-33, Sec. IV. B.) for the amplifiers in the automatic reactor control system dn/dt (rate of change of power) circuit to prevent line voltage fluctuations from affecting control system readability and operability was installed this month. Tests show a very pronounced reduction in the noise level at the output of the four amplifiers in the system as a result of the change.

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**FIG. 13 SAMPLE COIL INLET
LINE BEFORE FLOWMETER
INSTALLATION**

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**FIG. 14 SAMPLE COIL INLET
LINE WITH FLOWMETER
INSTALLED**

116569

Plugging Indicator

Flowmeter

**FIG. 15 OUTLET PIPING FROM
PLUGGING INDICATOR BEFORE
REROUTING**

V. PLANT MAINTENANCE

A. LEAK TEST OF NEW PRIMARY SHIELD TANK BOOT SEAL

Installation of the new primary shield tank boot seal was completed last month and the shield tank atmosphere was purged with nitrogen to a dew point of -40 F. A preliminary leak rate test conducted at a pressure of 8 inches WCG showed no significant shield tank leakage (see APDA-CFE-33, Sec. V. A).

Purging of the shield tank with dry nitrogen at 20 cfm was continued in June until the dew point reached -45 F, following which the annual shield tank leak rate test at 2 psig was conducted on June 17. The measured leak rate was 2568 scfd, which is considered excessive.

To permit identification of the source of the leakage, the primary shield tank was subsequently repressurized to 2 psig and soap bubble tests of the boot seal were made. Two rather large blowing leaks and a number of small leaks were found at locations along the upper clamp ring of the new boot seal where it is attached to the primary shield tank wall above floor level (see APDA-CFE-33, Figures 26 and 30). The leaks were found to be caused by weldments and other irregularities on the inner surface of the shield tank wall that prevented adequate compression of the boot seal lip by the holding clamps. The upper clamp ring segments in the leak areas were removed and the leaks were repaired by modifications which allowed the boot seal to more closely follow the contours of the shield tank wall. The surface irregularities on the inside wall of the shield tank were ground smooth, and narrow 1/8-inch-thick metal straps were installed beneath some of the upper clamp ring segments to bridge the 1/4-inch-wide gap which normally exists between segments, thus increasing compression between clamps in the areas where the original boot seal compression was inadequate. The smaller leaks were stopped by cementing 1/8-inch-thick strips of Silastic rubber to the sealing faces of the boot seal so as to increase the compression under the clamps and also by simply tightening the clamp ring bolts. To aid in boot seal repair, the primary shield tank atmosphere was alternately pressurized to 2 psig and depressurized to 1/2 inch WCG during the course of the repairs to enable additional leak-checking, identification, and repairs to be made. Following the repair of all known leaks, a final 2 psig leak rate test was conducted on June 27. The measured leak rate was 612 scfd, which is acceptable, and the boot seal installation has been accepted as satisfactory.

The observation was made during the boot seal leak repair that the pressure cycling performed apparently caused an increase in the moisture

content of the primary shield tank atmosphere, i. e., the dew point of the shield tank atmosphere measured at the sampling station increased noticeably following a pressure cycle. It is thought that the pressure cycling may have forced high moisture containing gas out of areas normally inaccessible to the purge stream. Because of this, it is planned that additional pressure cycling and purging of the primary shield tank atmosphere between 2 psig and 1/10 psig will be conducted to remove any moisture which remains similarly trapped.

B. PREPARATIONS FOR REPLACEMENT OF NO. 1 SAFETY ROD LOWER GUIDE TUBE

1. Background

Following completion of the holddown mechanism inspection, the principal work effort inside the reactor vessel has been directed toward preparations to replace the No. 1 safety rod lower guide tube with a new guide tube. Replacement and inspection of the original No. 1 guide tube located in the core lattice position N02-P02 (see APDA-CFE-31, Figure 5) is considered a necessary precaution before resuming reactor operation since it was diagonally adjacent to one of the melted fuel subassemblies at the time of the October 5, 1966, incident. It is also located directly adjacent to a row of subassemblies found to have undergone abnormal thermal bending stresses as a result of the incident (see APDA-CFE-5, Sec. III). Although in-reactor inspections of the guide tubes made with a borescope in September 1967 and November 1968 (see APDA-CFE-14, Sec. III and APDA-CFE-27, Sec. IV. A) had disclosed no evidence of damage to any of the guide tubes, the possibility that some thermal deformation might have occurred to the guide tube adjacent to the subassemblies that had melted fuel cannot be entirely discounted without removing the guide tube for detailed analysis. Some suspicion that the No. 1 lower guide tube may be distorted was found in March of this year (see APDA-CFE-31, Sec. III. D. 3) when, during an attempt to transfer the safety rod and vacate the No. 1 guide tube using the offset handling mechanism, the No. 1 guide tube was inadvertently raised with the safety rod and the safety rod had to be removed from the guide tube using a special hand tool inserted through the rotating shield plug (see APDA-CFE-32, Sec. V. B). The most likely reason for the sticking condition is the possibility of a small amount of oxide or surface tension in the sodium-filled dash pot.

Although an existing guide tube removal tool is available at the plant for removal of guide tubes through the rotating shield plug, it is designed for nonradioactive maintenance. It will not be used for the No. 1 guide tube removal because of the difficulties involved in adapting the tool for use in removing highly radioactive guide tubes. A new guide tube removal and replacement tool has therefore been designed, built, and is presently undergoing testing in a mock-up facility at the plant.

2. New Guide Tube Removal Tool

The new guide tube removal and replacement tool is designed to be used through the rotating shield plug to insert an adapter, the upper portion of which consists of a subassembly handling head, inside the No. 1 safety rod lower guide tube in core lattice position N02-P02. This will allow the guide tube to be removed from the lattice to the transfer rotor container using the offset handling mechanism and permit its subsequent removal from the transfer rotor to the fuel and repair building using the normal fuel transfer equipment in the same manner as a fuel subassembly. The removed guide tube will then be visually inspected and dimensionally checked under water in the cut-up pool in the fuel and repair building. To insert the new lower guide tube in the vacated lattice position, an identical handling head adapter will be installed in the new guide tube, the guide tube will be lowered into the reactor through the shield plug on the end of the tool, and the adapter will be removed from the guide tube and reactor using the tool.

The guide tube removal tool and special handling head adapter are shown in photographs taken in the mock-up facility in Figures 16 through 22. Although replacement of the No. 1 guide tube will be made with no sodium in the reactor, the tool is designed for use with the reactor either drained or filled with sodium. The adapter (see Figure 16) consists of a 15-inch-long shaft on top of which the handling head is located. A 1/4-inch-diameter wire coil spring is wound around the shaft below the handling head. An Acme thread and two sets of splines are located between the handling head and spring sections of the adapter. The splines and thread engage with similar mating surfaces on three separate concentric tubes contained inside the housing of the removal tool and are used in securing the adapter inside the tool and in winding up or relaxing the spring to cause a decrease or increase in spring overall diameter, respectively.

The removal tool consists of four concentric tubes (see Figure 17), each about 27 feet long. The outer housing tube is fitted with a square guide at its lower end, which rests on top of the guide tube and restrains the guide tube from turning under rotational forces during installation and removal of the adapter, when the tool is inserted in the reactor (see Figure 18). The innermost tube screws to the Acme thread of the adapter and is used to retract and hold the adapter inside the tool. One of the two remaining intermediate tubes meshes with one set of splines on the adapter and its rotation controls the wind-up and relaxation of the adapter spring. The other intermediate tube engages with the other set of adapter splines and is used to restrain the adapter and hold it steady while winding the spring and unscrewing the inner tube from the adapter. The removal tool is equipped with an argon purge line to prevent contamination of the reactor atmosphere during its use.

To remove a guide tube from the reactor the following sequence of steps is followed:

116669

FIG. 16 HANDLING HEAD ADAPTER FOR GRIDE TUBE REMOVAL TOOL

116769

FIG. 17 TOP PORTION OF GUIDE TUBE REMOVAL TOOL SUPPORTED FROM BRIDGE IN FUEL AND REPAIR BUILDING

116869

FIG. 18 ADAPTER PARTIALLY EXTENDED OUT OF GUIDE TUBE REMOVAL TOOL

116969

FIG. 19 GUIDE TUBE REMOVAL TOOL MATED WITH GUIDE TUBE IN MOCKUP FACILITY

117069

**FIG. 20 ADAPTER DEPOSITED IN DUMMY
GUIDE TUBE IN MOCKUP FACILITY**

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**FIG. 21 SIMULATED REMOVAL OF GUIDE
TUBE USING OFFSET HANDLING
MECHANISM GRIPPER**

117269

**FIG. 22 INSERTION OF NEW GUIDE TUBE
INTO VACATED LATTICE POSITION**

- The adapter is installed on the tool by screwing it into the innermost tube.
- The adapter spring is wound up, thus extending its length and reducing its overall diameter (across the adapter shaft), and the adapter is withdrawn inside the tool housing and locked in place with setscrews.
- The tool and contained adapter are hung from the building overhead crane and inserted into a standpipe erected over the guide tube access penetration in the rotating shield plug (Figure 17).
- Following purging, the standpipe valve is opened and the tool is lowered into the reactor over the guide tube to be removed until the tool housing guide comes to rest on the upper surface of the lower guide tube (Figures 18 and 19).
- The adapter is unlocked and lowered into the guide tube until the shoulder of the adapter contacts the inner guide tube.
- The adapter spring is unwound, thus releasing the tension and increasing the spring diameter, until a firm friction fit of the adapter inside the guide tube is attained.
- The tool is unscrewed and disengaged from the adapter and withdrawn from the reactor (Figure 20).
- The guide tube is removed from the reactor using the normal fuel handling system (Figure 21).

To insert a new lower guide tube into the vacated lattice position, an identical adapter is first installed on the tool and inserted inside the new lower guide tube by means of first winding and then unwinding the adapter spring. Following this, the tool and attached guide tube are lowered into the reactor through the standpipe using the overhead crane, the guide tube is seated in the support plates (Figure 22), and the spring is rewound to release the adapter from the guide tube. Finally, the adapter is withdrawn from the guide tube into the tool and the tool is removed from the reactor.

During replacement of the No. 1 safety rod lower guide tube with the reactor drained, the removal tool will be located in the No. 3 rotating plug access penetration positioned by means of plug rotation over the No. 1 lower guide tube. Two borescopes, one for horizontal viewing and one for vertical viewing, and an articulated viewing light will be installed in the adjacent No. 4 access penetration for use in observing the operations and facilitating the guide tube removal and replacement.

3. Mock-Up Tests

The guide tube removal tool was delivered to the plant by the manufacturer the third week in June and assembled for mock-up acceptance testing to demonstrate its adequacy and to verify the planned operating procedures for in-reactor guide tube replacement. During assembly of the tool some difficulties with galling of component parts was encountered because of the close clearances and numerous stainless steel to stainless steel material interfaces of the four concentric tubes. Similar problems of galling have been previously encountered and are apparently inherent in other very long tools having little strength when laid horizontally which have been used at the Fermi plant over the years. The galling problems were eliminated by suitable grinding and polishing.

To simulate the reactor conditions for acceptance testing of the tool, a mock-up facility was constructed in the equipment repair pit in the fuel and repair building. The core support plate from the mock-up facility¹ used for the zirconium segment removal studies was transferred from the steam generator building to the fuel and repair building for use in the guide tube removal tool equipment tests. A cluster of eight adjacent dummy sub-assemblies was placed around a dummy guide tube in the mock-up support plate to simulate the reactor array during the replacement and the removal tool was suspended at the required elevation from the bridge above the repair pit. The series of photographs in Figures 17 through 22 shows the removal tool being used in the mock-up facility to study the simulated guide tube replacement.

In the case of safety rod removal using the offset handling mechanism, four adjacent subassemblies must first be removed from around the guide tube because the safety rod handling head elevation is below that of the surrounding subassemblies. Eight adjacent subassemblies were used in the guide tube mock-up facility, however, because the installed elevation of the adapter handling head is the same as those on surrounding subassemblies and the offset handling mechanism gripper can be accommodated without subassembly removal. The presence of adjacent subassemblies is desirable, since they can be used as camming surfaces when installing the new guide tube in the lattice and thereby compensate for the certain amount of sway inevitable with the new guide tube as it is lowered into the reactor on the end of the tool.

As a result of the mock-up tests to date, some modifications of the tool are indicated to correct problems of binding and sticking encountered between the innermost tube, which holds the adapter, and the adjacent intermediate tube when inserting the adapter into the dummy guide tube. The problem is believed to be caused by interferences between the innermost tube and the setscrews used to hold the adjacent tube steady while lowering the center tube. To correct the problem, modifications are planned which

will change the manner in which the innermost tube is supported. Other than this difficulty, no major problems in the operation of the guide tube removal tool equipment have been found in the mock-up tests.

C. SODIUM PUMP MAINTENANCE

High voltage insulation tests were conducted on the No. 2 and No. 3 primary sodium pump motors and on all three secondary sodium pump motors during June. The 4800-volt and secondary cables of all five pumps were also Meggered. No faults were found. The No. 1 primary sodium pump motor was not tested because it is disassembled for repairs (see APDA-CFE-33, Sec. V. B).

The No. 1 secondary sodium pump seal oil system was also drained for inspection, cleaning, and leak-testing of the seal oil cooler. Similar maintenance will be performed on the other two secondary sodium pumps to ensure that all three pumps have adequate coolant capacity for 200 Mwt conditions once reactor operation is resumed. A suspected leak in the No. 2 secondary sodium pump seal oil cooler initiated the current investigation of these systems. The inspection will be completed next month.

VI. SPECIAL INVESTIGATIONS

A. CONTAINMENT BUILDING INTEGRITY CHECK

1. Corrosion Inspection

The exterior surface of the below-ground portion of the reactor containment vessel was inspected this month for evidence of corrosion; no signs of corrosion were found. The 72-foot-diameter, 120-foot-high vessel has its lower 51 feet below the finished grade level. Inspection of the outside surface of the vessel below grade has not been made since the original reactor construction ten years ago. Access to the containment vessel was made through manholes located in the outer biological concrete shield wall, which is located below grade outside the containment building. The upper portion of the shield wall, which begins at grade level and extends 7 feet below grade, is a thick shield built flush against the containment building wall; however, the lower portion does not require as thick shielding and is positioned several feet away from the containment vessel wall, thus forming an access cavity between the shield and containment vessel which extends approximately 34 feet below grade. The portion of the containment vessel below this level is embedded in the building concrete foundation and cannot be inspected.

To prevent corrosion, the 1-inch-thick steel vessel was originally given a prime coat of red lead followed by a coating of asphalt. The concrete foundation is kept dry by eight drains located in the floor of the shield cavity outside the vessel which are connected to the sump in the steam generator building. In addition, copper water stops are installed at all construction joints in the biological shield wall concrete below grade level, and flashing is added around the outer surface of the containment vessel at grade elevation.

Besides inspection the external surface of the containment building below ground, all of the building piping penetrations and the eight floor drains inside the biological shield wall cavity were inspected. No evidence of corrosion of piping was found and all floor drains were found to be clear.

2. Isolation Tests

Tests of the radiation and pressure-activated isolation systems for the containment building were also made during June. The above-floor radiation detectors (full isolation) and below-floor radiation detectors (partial isolation) were both tripped in separate tests to verify that the various containment valves would close properly. The results were satisfactory; however, during the pressure isolation test of the upper reactor building ventilation system, a test trip of the building pressure-sensing element failed to trip the pressure recorder control contacts which activate the relays for

the valve closings. The fault was traced to a slide wire contact in the pressure recorder control that was found off the slide wire because a pen arm was not on track. Repairs were made and the pressure isolation tests were repeated with satisfactory results.

3. Vacuum Relief Valve Tests

The two vacuum relief valves located in 8-inch diameter penetrations on the north and south sides of the containment building were also tested in June. The valves are set to be fully open whenever the atmospheric pressure outside the building exceeds the above-floor pressure inside the building by 2 psi, the maximum external design pressure differential for the containment vessel. The principal occurrence being guarded against by the valves is a sodium-air reaction depletion of the oxygen in the building atmosphere and consequent reduced pressure inside the building (~3 psig less) upon return to normal temperatures following the reaction. Each valve is capable of introducing 3600 scfm into the building with a differential of 2 psi across the valve. Air flow is inward only, the valves resulting when the pressure is equalized. A tight seal exists whenever the pressure inside the building is equal to or greater than the atmospheric pressure. The valves are designed for a maximum internal pressure of 32 psig, the peak pressure estimated to result from a sodium fire in the reactor building.

The vacuum relief valves were first leak rate tested at 32 psig using special test head connections attached to the valves. Both had acceptable leak rates of less than 0.05 scfd. A torque wrench was then used on special test taps attached to the valves to measure the forces required to open the valves. In the case of the valve on the north side of the building, the torque required to unseat the valve disc was found to be 24 in-lb and the torque to fully lift open the valve was 90 in-lbs. On the south side of the building 12 in-lb was required to unseat the valve and 108 in-lb to fully open the valve. These values are consistent with full valve opening at the designed 2 psig pressure differential.

B. PRESSURE TESTS OF SAFETY ROD EXTENSION BELLOWS

The integrity of the bellows on the safety rod extensions, designed for an internal pressure of 10 psig, was tested by closing the valves in the argon inlet and outlet lines to the bellows and monitoring the pressure for one hour to determine whether any pressure loss occurred. No pressure drop was observed on any of the safety rod extension bellows.

C. ELEVATION LIMIT CHECK OF OFFSET HANDLING MECHANISM

The offset handling mechanism elevation limit switches were tested by raising the machine manually over the subassembly located in core lattice position N06-N04. The tests were made to verify that the mechanism still operated properly after a rather lengthy period of inactivity. The limit switches and interlocks were found to function satisfactorily.

D. SUBASSEMBLY FLOW GUARD STUDIES

The various concepts of flow guards (see APDA-CFE-33, Sec. VI. A) proposed for core and inner radial blanket subassemblies are still being studied. The design work is progressing well and the results of laboratory proof tests to date have been favorable. A decision on adopting a reference flow guard design will be made soon. Vibration, pressure drop, and transverse flow tests have been conducted on the designs and impact and sodium performance tests are scheduled. Review of the offset handling mechanism designs indicates that some reduction in the overall length of the flow guard from that originally anticipated will be required for it to accommodate flow guarded subassemblies. In addition, a change in the up-limit switch setting of the handling mechanism must be made. The gripper in the steam-cleaning chamber in the fuel and repair building will also have to be modified to provide more clearance between the bottom of the flow guarded subassembly and the lower chamber valve. The final major work item, besides the No. 1 safety rod lower guide tube replacement, which must be completed before the vessel can be refilled with sodium and fuel transfer operations begun is to process a flow guarded dummy subassembly and its associated longer finned pot through the complete fuel transfer facility, transfer rotor, and offset handling mechanism fuel handling cycle to verify that no interferences exist. The handling operations inside the reactor vessel during the tests will be observed with a borescope. The longer finned pot will be mocked up by modifying an existing finned pot.

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