ENRICO FERMI ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT APRIL 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 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

Early in April the siphon line for draining sodium was installed in the reactor vessel, and on April 5 the vessel was drained to an elevation two inches below the lower support plate. The sodium was drained into storage tanks No. 2 and No. 3 in the primary sodium storage building. The sodium level detectors in the primary system were recalibrated prior to the drain, and calibration checks made during the drain showed that the calibration accuracy was within ± 3 inches.

When the holddown mechanism inspection tool was removed from the mockup facility after testing, the aluminum post was found to be distorted at the elevation of the four Quartzline lamps. Apparently, the distortion was due to thermal effects resulting from the 6 kw of thermal and radiant heat from the lights. The aluminum post was replaced by an insulated steel post and the holddown inspection tool was returned to the mockup facility for heating tests with dummy subassemblies placed around it to simulate the core cavity. The tests, which were completed by the end of the month, showed that the new post was not susceptible to thermal distortion and that adjacent subassemblies would not experience excessive heating in the reactor.

Feedwater flow tests were completed on the No. 1 steam generator to determine the pressure drop across the steam generator as a result of installation of pressure drop orifices in the water tubes. The measured pressure drop agreed well with predicted values. Three isolation tests were also performed on the No. 1 unit. Nitrogen pressure tests following each thermal shock showed no evidence of leaks. The tube sides of the No. 2 and No. 3 units remained pressurized with nitrogen.

The erection of the new fuel transfer facility in the reactor building was completed during the month and painting of the facility was begun prior to testing.

The paint was removed from the inside of the No. 2 equipment decay tank, which is being modified as an auxiliary fuel storage facility. Installation of insulation inside the tank was nearly completed. The false floor and sodium drip pan also were installed.

Installation of the IBM series 1800 computer portion of the reactor malfunction detection analyzer was completed. Testing by IBM field engineers revealed only a few minor faults, which were corrected. Signal conditioning work on the required input signals continued throughout the month.

Preparations were begun for modifications to the plugging indicator circuit in the cold-trap room of the primary sodium service system. Piping changes are planned, an electromagnetic pump will be installed in series with the plugging indicator, and a hydrogen detector will be installed in series with the sodium sample coil. To allow longer periods of worker access to the cold-trap room during the modifications, a temporary lead shield was installed partially around the cold trap.

The outer radial blanket support plate orifice damaged in the original attempts to photograph the holddown fingers was replaced using an orifice removal tool inserted through the rotating shield plug.

The No. 1 safety rod, which was found to be stuck in its lower guide tube during an attempted rod transfer last month, was successfully removed from the guide tube and transferred to the No. 3 safety rod position using a special tool inserted through the rotating shield plug.

Investigations of the high torque which developed on the No. 1 primary sodium pump last month revealed that normal torque exists during vertical adjustment of the shaft through its allowable operating range. However, the slip rings of the rotor were found badly grooved due to arc-over resulting from oil leakage, and the rotor was sent out for repair to the slip rings.

Replacement of the primary shield tank boot seal found to be defective in June, 1968, was begun. The old seal was cut out and removed and installation of the upper clamps for the new seal on the primary shield tank wall was begun.

II. OPERATIONS

A. STATUS OF REACTOR SODIUM AND GAS SYSTEMS

Following the sodium drain of the primary system at the beginning of the month, the temperature of the argon-filled primary system was maintained between 365 F and 380 F by the pipe and vessel induction and resistance heaters. All three loops of the secondary sodium system remained filled with sodium at approximately 350 F. Feedwater flow and isolation tests were conducted on the No. 1 steam generator; however, water was not added to the No. 2 and No. 3 units and their water tubes remained pressurized with nitrogen at 20 psig. 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. During replacement of the rubber boot seal located between the shield tank and machinery deck at the top of the reactor vessel, some air was mixed with the nitrogen atmosphere of the primary shield tank as indicated by an increase in the dew point of the shield atmosphere.

B. PRIMARY SYSTEM SODIUM DRAIN

In March the reactor vessel was refilled with sodium to allow fuel subassembly movement using the offset handling mechanism operating in sodium to achieve the core configuration required for insertion of the hold-down mechanism inspection tool (see APDA-CFE-31, Sec. III. B. and III. D). This month the sodium siphon drain line was reinstalled and the vessel was partially drained of sodium in anticipation of insertion of the holddown mechanism inspection tool for use in the subsequent photographic survey of the holddown fingers.

1. Draining Operations

On April 5 the reactor vessel was partially drained of sodium down to an elevation of 559 feet 9 inches, which corresponds to an elevation two inches below the lower fuel support plate. A complete plenum drain was not required for insertion of the holddown mechanism inspection tool and subsequent photographic survey. The drain was performed in essentially the same manner as the earlier partial drain made in June, 1967 (see APDA-CFE-11, Sec. II). It was also essentially identical to the two previous full drains (see APDA-CFE-13, Sec. II. C. and APDA-CFE-27, Sec. III. C) made in August 1967 and November 1968 with two exceptions: (1) a temporary rubber backup seal for the rotating shield plug Klotz seal was not used because the reactor vessel argon cover gas pressure had to be raised to only 2-1/4 psig for the partial drain (10 psig is required for a full drain), which

is considerably less than the 5 psig maximum pressure limitation of the Klocure seal, and (2) the two-inch carbon steel siphon pipe was inserted through the rotating plug and support plate holes in the N01-N02 core position, rather than in the P00-P00 core center position required for connection with the hollow central column permanently installed in the lower plenum for a complete drain.

Prior to the drain, the primary system sodium temperature was lowered from 450 F to 350 F and the nitrogen-to-water ventilation heat exchanger in the primary storage tank room was reactivated and leak-tested. No leaks were found. The ventilation blower bearings and room lighting bulbs used for sodium leak detection via mirror viewing were replaced before returning the unit to service. The primary sodium storage building was then resealed and the oxygen content reduced to less than 5% by purging with nitrogen to eliminate the possibility of a fire in the event of a sodium leak. The electrical leads to the primary pump main and pony motors were also disconnected to prevent any possibility of pump operation with the vessel drained. This completed the preparation and precautionary measures.

The initial portion of the drain was made using the overflow pump. This reduced the original sodium level from 580 feet 11 inches in all primary system components to 575 feet 0 inches in the reactor vessel and 574 feet 0 inches in the intermediate heat exchangers and pump tanks (see APDA-CFE-27, Figure 7). Approximately 6 hours of overflow pump operation was required for this portion of the drain. The sodium level in the overflow tank was then left at 574 feet 0 inches elevation and the sodium in the reactor vessel further lowered to 559 feet 9 inches elevation by siphoning action through the siphon pipe and temporary drain lines which discharged to the primary sodium storage tanks through the plant siphon drain system. Approximately 8-1/2 hours of siphoning operation was required to reduce the sodium in the vessel to this level. The drained sodium was stored in the No. 2 and No. 3 sodium storage tanks and was maintained there at a temperature of about 325 F.

2. Calibration Check of Level Detectors

The sodium level detectors in the primary system, which are located in the reactor vessel, overflow tank, and three primary pump tanks, were recalibrated prior to the sodium drain and their calibrations checked during the course of the drain. A differential pressure-type level detection system is used in which (1) a pressure detector is located in the gas volume of each tank to measure the cover gas pressure; (2) another pressure detector is located in the liquid volume of the tanks to measure the sum of the cover gas pressure and sodium head above it; and (3) these two pressures are fed into a differential transmitter which subtracts the cover gas pressure, giving the actual head or level of sodium. A Moore-type pressure detection system, utilizing pneumatic signals throughout is used for all primary system level

detection except for the overflow tank. The overflow tank uses a Taylor system and is completely electrical.

The detectors were recalibrated before the drain by adjusting their readings to agree with the sodium level at the elevation of the overflow lines in each of the sodium tanks. The differential signal voltages between cover gas sensors and liquid volume sensors were calculated on the basis of the cover gas pressure, sodium head, sodium density and other parameters. The reactor vessel level detectors, which are the only ones having temperature compensation, also had temperature effects factored into their calibration.

The calibrations were checked during the drain by comparing the readings with the known elevations of piping which was uncovered during the drain; i. e., when the sodium was drained past the level of a pipe opening, the sodium flow indicators would indicate cessation of flow and the actual sodium level could thus be deduced. The result of the calibration check showed that all primary system level detectors read satisfactorily. The overflow tank detector did read slightly high (about 10 inches), but the three pump tank and reactor vessel indicators all read within ± 3 inches of the actual sodium level at the normal cover gas pressure of 4 inches WCG. Similar accuracy was obtained at the elevated cover gas pressure (63 inches WCG) used for the siphon drain with the exception of the level detector in the No. 1 pump tank, which gave poor readings. It has been suspected for some time that the upper cover gas pressure sensor in this tank has a frozen bellows, giving rise to erroneous readings at pressures significantly different than the normal cover gas pressure, which is close to atmospheric.

C. STEAM GENERATOR PERFORMANCE

The No. 1 steam generator was filled with water early in April and feedwater flow pressure drop tests and thermal shock isolation tests were performed on the unit. The measured pressure drop across the steam generator agreed very well with the predicted value, and no increase in water-tube-side to sodium-shell-side gas leakage was detected by nitrogen leak test measurements made after the thermal shock tests. The tube sides of the No. 2 and No. 3 units remained filled with nitrogen throughout the month. Tests on these units were performed earlier and are reported in APDA-CFE-25, -27, and -28.

Prior to filling the No. 1 unit with water, the special test instrumentation installed in the water heater and downcomer tubes was connected to the Sanborn 650 recorders and the isolation valve system of the steam generator was checked. All systems were found to be functioning properly, and the unit was filled with water near the middle of the month. The water for filling the steam generator was obtained from the No. 4 heater, where plant condensate was heated under pressure to 350 F. The water-filling

procedure included back-pressurizing from the steam outlet to lift the water tube orifices and relieve the nitrogen trapped in the annulus around the orifice tube in each water tube (see APDA-CFE-12, Figure on p 29). Circulation at a pressure of 925 psig was then established by a plant boiler feed pump through the steam generator and main steam bypass line back to the No. 4 heater. Water flow rates in the range of 120,000 to 210,000 lb/hr were desired as conditions for the subsequent hydraulic tests.

Feedwater flow tests were conducted on April 24 to determine the pressure drop across the steam generator resulting from the earlier installation of the poppet-type pressure drop orifices in the water tubes (see APDA-CFE-12, Sec. IV.A). Water flows of 120,000 lb/hr to 210,000 lb/hr were used during the test, the latter flow being the nominal value for 200 Mwt conditions. The system temperature was 345 F (isothermal), and the water pressure in the steam generator was 925 psig with a pump discharge pressure of 940 to 960 psig. The measured pressure drop across the steam generator varied from 6.8 psi to 19.8 psi at the two extreme flows, which is only 2% smaller than the pressure drop predicted by calculations.

Following the hydraulic tests, three successive thermal shock tests were performed on April 26, 27, and 28 by isolating and venting the No. 1 steam generator to verify the integrity of the unit under conditions of rapidly occurring thermal gradients between the water and sodium spaces. In the tests, water at 960 to 1110 psig pressure was used in the tubes and sodium was circulated on the shell side. The steam generator was held at an isothermal temperature of approximately 400 F prior to the tests. It was then isolated from the feedwater system and the water side of the unit was quickly vented to the atmosphere. Figure 18 of APDA-CFE-26 is a flow diagram for a thermal shock test.

The initial conditions for the thermal shock tests were as shown in Table 1.

TABLE 1 - INITIAL CONDITIONS FOR THERMAL SHOCK TESTS OF NO. 1 STEAM GENERATOR*

	<u>Test No. 1</u>	<u>Test No. 3</u>
Water Flow, lb/hr	No Flow**	No Flow**
Water Pressure, psig	960	1110
Water Temperature, F	400	405
Sodium Flow, lb/hr	2.8×10^6	2.8×10^6
Sodium Temperature, F	400	405

* Data were not obtained for the second isolation test.

** The water in the tubes was stagnant with only natural circulation flow.

The No. 1 steam generator is the only one of the three units to which special instrumentation has been added for measuring the temperature gradients experienced during isolation tests. The thermal transient data from the April tests are presently being analyzed.

Following each thermal shock test, the water side of the unit was filled with nitrogen at 600 psig and nitrogen leak tests were performed on the water tubes in the same manner as done in February and March of this year (see APDA-CFE-30, Sec. V.C. 3 and APDA-CFE-31, Sec. III. E). There was no evidence of new leaks in the unit; the measured leakages of approximately 120 cm³/day (at a 20 psig pressure differential) were comparable to the leakages measured in February and March.

Since the steam generator modifications of last year, all three steam generators have now been isolation-tested, and all except unit No. 2 have been hydraulic-tested. The results to date have been very satisfactory. Feedwater flow-hydraulic tests are scheduled to be performed on the No. 2 unit soon; they also will be repeated on the No. 1 unit next month to determine whether the poppet-type pressure drop orifices all reseated following the isolation test made this month.

No cold-trapping was performed on any of the three secondary sodium loops in April. In Loops No. 2 and No. 3, where no steam generator testing was conducted, the bulk sodium temperature was maintained at approximately 350 F to 360 F throughout the entire month; the plugging temperature held steady at 275 F in each loop throughout the month. In Loop No. 1 the bulk sodium temperature was held at 350 F, except for the three days of isolation tests, when it was raised to approximately 400 F, and the sodium plugging temperature fluctuated between 260 F and 280 F during the month.

D. GAS SYSTEM PERFORMANCE

Tables 2 and 3 show the primary system cover gas activity and gas analysis information obtained since the last reported data (APDA-CFE-31).

TABLE 2 - PRIMARY SYSTEM COVER GAS ACTIVITY

<u>Location</u>	<u>Sample Data*</u>	<u>Gross Beta Activity, $\mu\text{Ci}/\text{cm}^3$</u>
Reactor Cover Gas	4-2-69	1.61×10^{-6}
Reactor Cover Gas	4-9-69	2.23×10^{-6}
Reactor Cover Gas	4-17-69	1.59×10^{-6}
Reactor Cover Gas	4-22-69	1.24×10^{-6}
Reactor Cover Gas	4-30-69	1.86×10^{-6}
Primary Shield Tank	4-2-69	3.05×10^{-8}
Primary Shield Tank	4-17-69	3.99×10^{-8}
Primary Shield Tank	4-22-69	2.99×10^{-8}

* Sodium drain performed on 4-5-69.

TABLE 3 - PRIMARY SYSTEM COVER GAS ANALYSIS

	Reactor Cover Gas (Argon), ppm by volume		Primary Shield Tank Atmosphere (Nitrogen), ppm by volume	
	Before Sodium Drain*	After Sodium Drain**	Before Boot Seal Repair**	During Boot Seal Repair***
Oxygen	Below 25	Below 25	250+	10.5%+
Carbon Monoxide	Below 10	Below 10	Below 10	Below 10
Carbon Dioxide	20	25	20+	2.0+
Hydrogen	Below 4++	Below 4++	Below 2.5	Below 2.5
Helium	Below 8	Below 8	Below 8	Below 8
Methane	Below 10	Below 10	Below 10	Below 10
Nitrous Oxide	← Not Measured →			
Nitrogen	720	1270	← Remainder →	
Argon	← Remainder →		Not Measured	0.21%
Dew Point	← Not Measured →		-50F+++	+15F+++
Shield Graphite Temperature	Not Measured	320F	320F	310F

* Sample taken April 3, 1969.

** Sample taken April 15, 1969.

*** Sample taken April 30, 1969.

+ Technical Specifications state 1000 ppm maximum during reactor operation and during shutdown whenever primary system temperatures exceed 400 F (see Sec. V.E).

++ 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 (see Sec. V.E.).

E. 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 April. No cold-trapping of the transfer tank sodium system was conducted because of a valve in a bypass line of the cold trap system which froze last month following bellows failure and is out of service until a replacement bellows is installed (see APDA-CFE-31, Sec. III.H). The sodium plugging temperature was 250 F at the beginning of the month and slowly rose to 270 F by the end of the month. The bulk sodium temperature in the system is being maintained between 450 F and 465 F.

F. RADIOACTIVE WASTE DISPOSAL

During April, 5,187 gallons of liquid waste was discharged through the metering pump to the North Lagoon. The maximum specific activity in the two liquid waste tanks in the fuel and repair building was approximately $3.3 \times 10^{-5} \mu\text{Ci}/\text{cm}^3$ and $1.3 \times 10^{-3} \mu\text{Ci}/\text{cm}^3$, respectively. The total activity discharged to the lagoon was approximately 65 μCi . All effluent released after dilution with the circulating pump discharge was below maximum permissible concentration (MPC).

Approximately $1 \times 10^9 \text{ ft}^3$ of gaseous effluent was discharged through the plant waste gas stack during April. The concentration of activity was very small, well below the MPC after dilution in the stack blower and in the atmosphere.

III. HOLDDOWN INSPECTION TOOL FOUND DAMAGED

Tests of the holddown mechanism inspection tool in the mock-up facility were conducted during March (see APDA-CFE-31, Sec. VI.A). When the inspection tool was removed from the mock-up facility this month, following the vessel sodium drain, distortion of the post of the mirror assembly was discovered.

A. DISTORTION OF HOLDDOWN INSPECTION TOOL POST

Figures 1 through 3 are photographs of the holddown mechanism inspection tool taken after its removal from the mock-up facility on April 7, 1969. The distortion found in the aluminum post used to support the light compartment, motor assembly, and rotating mirror is evident. The offset in the post due to bending was approximately two degrees, occurring in the area directly opposite the four 1500-watt Quartzline lights (the blackened area on the photographs). The warpage was apparently due to the 6 kw of thermal and radiant energy resulting from the lighting attached to the post.

The inspection tool was subsequently returned to the mock-up facility on April 21 for additional investigations of its heating effect on the post and on the surrounding subassemblies which will make up the core cavity during use of the tool in the reactor. Before returning the inspection tool to the mock-up, several design modifications were made to minimize heating effects. The aluminum post was first replaced with an annealed Type 304 stainless steel post having a yield strength of 18×10^3 psi at the 1000 F estimated operating temperature of the post in the reactor, compared to 2×10^3 psi for aluminum at the same temperature. Also, to reflect heat from the post, a 1/8-inch thickness of Fiberfrax insulation was added around the post at the elevation of the lights, and the insulation was covered in turn with 5 mils of stainless steel metal sheath shim stock, as shown in Figure 4.

B. ADDITIONAL MOCK-UP TESTS PERFORMED

The modified holddown mechanism inspection tool was tested in the mock-up facility for its heating effect on the mirror post and surrounding subassemblies. A photograph of the equipment arrangement in the mock-up facility during the test is shown in Figure 5. The purpose of the test was twofold: (1) to measure the maximum temperature gradients across nearby subassemblies under simulated conditions during use in the reactor and (2) to determine if any thermal distortion of either the post or adjacent subassemblies occurred as a result of heating from the lights.

083969

**FIG. 1 HOLDDOWN INSPECTION
TOOL AFTER REMOVAL
FROM MOCK-UP – LIGHT
BASKET STILL IN PLACE**

084069

**FIG. 2 VIEW OF BENT POST
WITH LIGHT BASKET
REMOVED**

084169

Location of Distortion

**FIG. 3 CLOSE-UP OF WARPED
ALUMINUM POST AT
ELEVATION OF LIGHTS**

084269

**FIG. 4 NEW STEEL POST WITH
INSULATION INBOARD
OF LIGHTS**

084369

Location of Inspection Tool
During Tests

**FIG. 5 ARRANGEMENT FOR HEATING
TESTS OF HOLDDOWN
INSPECTION TOOL IN MOCK-UP**

To conduct the test, 22 dummy subassemblies (empty subassembly wrapper cans having no pads) were placed in the mock-up of the core support plate and arranged as shown in Figure 5. The complete core cavity was not mocked up, but only those subassemblies closest to the inspection tool were duplicated. This resulted in an array consisting of an inner row of 10 subassemblies, backed up by an outer row of 12 subassemblies. The subassemblies were all gauged and dimensionally checked prior to installation in the mock-up facility so that any resultant thermal distortion could be observed. An adaptor sleeve had to be added to the post of the inspection tool to allow its placement in the mock-up of the core support plates since the intended position in the reactor vessel is in an outer radial blanket position having a smaller diameter than core positions.

Twenty thermocouples were then spot-welded on the outside surface of the wrapper cans of the 10 dummy subassemblies in the inner row closest to the inspection tool. They were placed at an elevation corresponding to the midplane of the Quartzline lamps to allow measurement of the maximum radial temperature gradient across the subassemblies. Another thermocouple also was added at a similar elevation on the mirror post of the inspection tool behind the layer of insulation. The thermocouples were all connected to the multipoint switches of a Speedomax high-temperature indicator and recorder. Finally, an airflow of 2 cfm was established through the purge line to the lights of the inspection tool for the tests to simulate the 3 cfm of argon flow which will be used during in-reactor operation of the assembly.

A series of three measurements was made in the mock-up tests in which the mirror post and adjacent subassemblies were exposed to the 6 kw of radiant and conductive heating resulting from light operation for continuous periods of four minutes, five minutes, and six minutes. This range of exposures bracketed the anticipated maximum exposures that will be used during the photographic surveys in the reactor and, therefore, covered the range of maximum temperatures expected. Thermocouple readings were taken at one-minute intervals during each test, and the lights were turned off between tests to allow the subassemblies and inspection tool to cool to ambient temperature before conducting the next test. Dimensional measurements to determine subassembly straightness were made on the hottest subassemblies after each run, and detailed dimensional checks were made on all subassemblies after the final run.

The test results show a variance in subassembly can wall temperature depending on the distance from the light source and the exposure time. The temperature gradient measured across subassemblies varied from 125 F for the coolest subassembly for a four-minute exposure, to 240 F for the hottest subassembly for a six-minute exposure. The maximum subassembly wall temperature measured was 365 F. The wall temperature on the side of the subassembly away from the light source did not increase significantly in any of the tests (about 15 F maximum), whereas the rate of temperature

rise of the wall nearest the light source was fairly constant at 60 F per minute in the case of the hottest subassembly for the first four minutes of exposure. After four minutes, it leveled off markedly and did not increase significantly in the last two minutes of exposure. The maximum temperature measured on the mirror post behind the light source was 820 F. The effect of the ambient temperature difference between the reactor and the mock-up facility, approximately 350 F versus 70 F in the mock-up, is expected to result in about a 300-F temperature increase in all temperatures measured in the mock-up. However, the increased ambient temperature in the reactor vessel is not expected to significantly affect the temperature gradients as measured in the mock-up facility.

The dimensional check of the subassemblies following completion of the tests showed no abnormal bowing or thermal distortion of the subassemblies or mirror post as a result of the heating tests; therefore, the modified holddown mechanism inspection tool has been determined to be satisfactory for use in the reactor.

IV. STATUS OF PLANT MODIFICATIONS UNDER CONSTRUCTION

A. FUEL TRANSFER FACILITY

Erection of the new fuel transfer facility in the reactor building was completed this month (see APDA-CFE-30, Sec. VIII and APDA-CFE-31, Sec. IV.A). Figures 6, 7, and 8 are photographs showing the placement of the gripper and plug casks on the bridge and trolley structure in the reactor building. Figure 9 is a photograph of the completed facility with the transfer cask in place. The total time for erection following the placement of the bridge and trolley support columns last month was about three weeks. The control console has also been put in place, the various piping and electrical connections to the facility and control console are now being completed, and the entire facility is being painted white. Engineering personnel from Atomics International, designer of the facility, will begin final checking and calibration of the facility early next month.

A new emergency argon cooling line also has been completed to the companion equipment previously erected in the fuel and repair building. The line is intended for use with the fuel transport system if a fuel subassembly becomes stuck in either the transfer cask or exit port pipe in the fuel and repair building. The original emergency cooling line, which entered the top of the gripper cask and forced gas through the gripper pulley housing, through small holes into the transfer cask, and then down the exit pipe into the transfer tank, was found to give inadequate flow due to high back pressure resulting from the small capacity of the transfer tank discharge to the waste gas system in the fuel and repair building. The new emergency cooling line is routed via a 150-foot run between the pump room of the building recirculating argon system to the exit port valve pit. A temporary quick-disconnect hose connection will be used between the valve pit and fittings on the side of the transfer cask for use in recirculating emergency argon cooling gas through the transfer cask and transfer tank, to the pump room and back to the transport cask. In this way, pressure buildup in the transfer tank will be eliminated.

B. AUXILIARY FUEL STORAGE FACILITY

During April, work progressed on the conversion of equipment decay tank No. 2 in the reactor building to an auxiliary fuel storage facility (APDA-CFE-30, Figure 19): removal of the chemical-resistant paint from inside the tank, begun last month, was completed (APDA-CFE-31, Sec. IV.B); the tank was insulated; the false floor and drip pan were installed inside the tank; preparations were completed for the fuel storage rotor installation; and steel liners were fitted inside the access port holes penetrating the shield plug of the decay tank.

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FIG. 6 GRIPPER CASK FOR FUEL TRANSFER FACILITY IN SHIPPING FRAME

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FIG. 7 GRIPPER CASK BEING LIFTED TO BRIDGE AND TROLLEY STRUCTURE

Gripper Cask
Bridge and Trolley Structure

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FIG. 8 GRIPPER AND PLUG CASKS INSTALLED ON BRIDGE AND TROLLEY

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FIG. 9 COMPLETE FACILITY WITH TRANSPORT CASK IN FOREGROUND

Figure 10 shows the false floor and sodium drip pan installed 15-1/2 feet down from the operating floor level inside the 36-foot-deep decay tank. The false floor divides the decay tank into upper and lower chambers, with the fuel storage area being located in the upper chamber. The 9-foot, 4-inch diameter steel floor was constructed from two 5/16-inch-thick steel plates welded to the top and bottom of 4-inch steel ribs. A 5-inch I beam provides support beneath the floor ribs, and the whole assembly rests on six supports welded around the circumference of the decay tank wall. The sodium drip pan sits on top of the storage tank floor and is formed from two semicircular sections having an 8-inch-high wall at their outer periphery. The two semicircles were welded together at their diameters along a common 5-inch-high wall. In addition, a smaller-diameter, 5-inch-high inner circular wall is welded to the center of the pan to prevent any sodium deposited on the pan from reaching the central region, over which the heaters for the facility will be supported. Sodium leak detectors consisting of electrical conductors activated by contact with sodium will be mounted at various locations on the drip pan.

A 4-inch thickness of Johns-Mansville spun glass 1000 F Fiberfrax insulation was installed on the bottom and sides of the decay tank (see Figure 11). It was selected on the basis of its rigidity and minimum offgasing properties under the high-temperature conditions of the auxiliary fuel storage facility. The insulation, installed in blocks, was attached to the steel surface of the decay tank by means of mounting studs welded to the tank; a protective layer of 5-mil-thick corrugated aluminum lagging was installed over the insulation. The bottom surface of the tank shield plug assembly will be similarly insulated next month.

Figures 10 and 11 also show the three support brackets installed on the wall of the decay tank for supporting the fuel storage rotor. These brackets will provide temporary support for the rotor during facility construction and can be used to support the weight of the fully loaded rotor should maintenance be required on the mechanisms of the above-floor portion of the rotor shaft during operation of the facility. The storage rotor, shown in Figure 12, installed on the support brackets for trial fit-up this month, contains two concentric circular rows of 4.84-inch-diameter holes in a rotor plate for holding fuel storage pots. Twenty storage positions are provided in the inner row and 40 in the outer row, each row being located under one of the two access loading ports provided in the decay tank shield plug. The rotor plate is of 1/2-inch-thick steel construction and is reinforced underneath by two circumferential members and eight radial members to give it a rigid, box-like construction. A hub assembly is centrally located on top of the rotor plate for use in connecting the rotor plate to the rotor shaft.

Figure 13 shows the outer shield plug of the decay tank raised for fitting the stepped steel liner inside the 12-inch-diameter access port drilled

Block of
Insulation

Walls of
Drip Pan

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FIG. 10 FALSE FLOOR AND DRIP PAN OF THE AUXILIARY FUEL STORAGE FACILITY

Support
Bracket
for Storage
Rotor

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FIG. 11 INSULATION PARTIALLY INSTALLED INSIDE DECAY TANK

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FIG. 12 AUXILIARY FUEL STORAGE ROTOR ASSEMBLY

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FIG. 13 FITTING OF ACCESS PORT LINER IN OUTER SHIELD PLUG

through the plug. Solid carbon steel shield plugs will be placed in the access ports when fuel handling is not being performed.

C. MALFUNCTION DETECTION ANALYZER

Erection of the special support platform and installation of the malfunction detection analyzer (MDA) computer system in the main control room was completed during April (see APDA-CFE-31, Sec. IV.C). The power supply feed to the MDA from the motor-generator set formerly used for the plant simulator was completed. Following the equipment installation and power feed transfer, the equipment was turned over to IBM field engineers for interconnection of the various components and internal calibration and checkout of the computer. Only a few minor faults were found which required correction; these included a defective printed circuit board in the computer main frame and an incorrectly wired interconnecting cable from the main frame to the line printer. The latter fault resulted in damage to a fan motor in the line printer. Following correction of these problems, diagnostic programs were run and completed with good results. Calibration checks of the analog input voltage signals showed them to be well within the manufacturer's specifications. In addition, a test was made on the performance of the computer during power interruption to the motor-generator set. The test showed that with the computer set on a disc-reading operation, power interruptions as long as 8 seconds did not interrupt the operation of the computer. This was an unexpected result. Although the motor-generator set contains a voltage regulator to maintain constant voltage as the flywheel slows down during power interruption, it was believed that the corresponding frequency change would affect operation of the computer. However, the computer is apparently not as sensitive to frequency changes as originally thought.

Work also continued on obtaining the signal conditioners required for the analog inputs to the MDA computer system (see APDA-CFE-31, Sec. IV.C.4). The circuit designs have been completed and components ordered for the necessary safety system signal conditioners. Since these signals are part of the safety system, they cannot be used directly as input to the MDA but must be taken from the output of the safety system. Components have also been ordered for the power range signal conditioners and the signal conditioners for the cover gas fission product detector system. In these two cases, the output voltages must be rescaled when range changes are made to maintain the signals to the MDA in the millivolt range.

A temporary signal simulator is being built for use with the malfunction detection analyzer to aid in the development and verification of the various reactor analysis programs, and for use in operator training prior to construction of a larger permanent signal simulator. The temporary simulator will allow simulation of as many as 25 analog signals, in addition to 16 process-interrupt and 16 contact-operate digital output signals.

D. PRIMARY SODIUM COLD-TRAP ROOM MODIFICATIONS

1. Background

Work has begun to modify the plugging indicator circuit in the primary sodium service system with the expectation of increasing the accuracy of determining plugging temperatures. It is believed that with the present arrangement, shown in Figure 17 of APDA-CFE-21, operation of the cold trap perturbs the flow to the plugging indicator and influences the indicated plugging temperature. It is also believed that the present mode of operation, using high pressure with a high resistance in series with the indicator, affects the plugging temperature because of the variable pressure drop created across the orifice.

The planned modification involves a change in the piping to return the plugging indicator sodium discharge stream back to the inlet header rather than to the cold-trap outlet header, thus eliminating the influence of cold-trap operation. The installation of an electromagnetic pump in series with the plugging indicator and improved operating controls are also expected to produce a more desirable constant pressure drop across the orifice for determining plugging temperatures. It is planned to further modify the primary sodium service system to provide for the addition of a flowmeter and in-line hydrogen detector in series with the sodium sample coil.

2. Erection of Temporary Shield Around Cold Trap

In preparation for the plugging indicator modification work in the primary cold-trap room, a temporary sheet lead shadow shield has been erected part way around the primary cold trap. This was done to allow a longer working time in the room for workmen during the modifications. Estimates of the shielding necessary for the protection of working personnel were obtained from a radiation survey made in the cold-trap room in November, 1968 (see APDA-CFE-27, Sec. IV.C). The survey showed that the radiation levels at waist-high elevation one foot from the west side of the cold trap (the side facing the interior of the room) due to fission products deposited in the cold trap varied from approximately 500 to 1000 mr/hr. It was desired to place sufficient shielding around the cold trap to limit the maximum exposure of workers to 100 mr/week, i. e., to limit the average exposure to 2-1/2 mr/hr for a 40-hour work week.

Prior to beginning the modifications, the recirculating nitrogen atmosphere normally used as a cooling medium in the cold-trap room was replaced by outside air drawn into the room by the room blower through a temporary opening in the blower housing. The air was discharged through the cold-trap room and tunnel to the atmosphere through a temporary open manhole located in the top of the absolute filter vault (see APDA-CFE-21, Fig. 21). Access to the room was gained through a manhole in an opening

in the north side of the cold-trap room in the sodium building. The access manhole is shown in Figure 14, together with the six rolls of 1/8-inch-thick, 4-1/2-foot-wide, 10-1/2-foot-long lead sheet used for the temporary shield. Access to the room was under health physics supervision. In addition, a barrier rope and gate were installed outside the room opposite the manhole to mark off all areas in which the radiation level exceeded 5 mr/hr with the manhole open. Continuous health physics monitoring was maintained inside the room during the shield construction period.

The lead sheet shielding material was cut into sections and placed on the west side of the cold trap in successive layers until the radiation level measured at the location of the plugging indicator was reduced to 2-1/2 mr/hr. The lead sections were temporarily held in place during construction by clamps attached to two permanent I beams located adjacent to the cold trap on its northwest and southwest corners (see Figure 15). A slot was also cut in the floor around the west side of the cold trap and the lead sheets slipped into it. This required that a section of the drip pan around the cold trap above the slot be cut out first. The final shield configuration was banded together using metal bands as shown in Figures 16 and 17. The final lead thickness was approximately 2 inches at the bottom of the cold trap and 1 inch at the top, and the shield extended over about a quarter of the circumference of the cold trap.

Rolls of
Lead
Sheet

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FIG. 14 MANHOLE TO COLD-TRAP ROOM

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**FIG. 15 COLD-TRAP SHIELDING
TEMPORARILY CLAMPED
TO I BEAMS**

Drip Pan
Partially
Removed

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**FIG. 16 COMPLETED SHIELDING
BONDED TOGETHER**

Slot
Cut
in
Floor

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**FIG. 17 SIDE VIEW OF COMPLETED
SHIELD AROUND COLD TRAP**

V. PLANT MAINTENANCE

A. OUTER RADIAL BLANKET ORIFICE REPLACED

On April 11 a damaged orifice in the lower support plate was removed and replaced with a new orifice in outer radial blanket position N07-P00. This is the position in which the holddown mechanism inspection tool will be inserted for the photographic survey of the holddown fingers. In April 1968, the original orifice was found to have several bent orifice retainer springs, the damage apparently resulting from the insertion of the original spherical mirror support post in this position during the early attempts to photograph the holddown mechanism (see APDA-CFE-21, Sec. III.A and Fig. 7).

The orifice replacement was made using an existing orifice removal tool. Although the tool had been used in 1959 for installing orifices while working inside the reactor vessel during the reactor construction period, it had not been operated previously through the rotating shield plug. For this reason the use of the tool was checked in the mock-up facility prior to the orifice replacement; it operated smoothly there as well as in the subsequent reactor replacement.

The orifice replacement tool is a dual-purpose device consisting essentially of a long pole, having approximately the same outer diameter as a subassembly nozzle, to which different attachments can be added for orifice removal and insertion. To remove the damaged orifice, a cup-like attachment was affixed to the lower end of the tool. The tool was then lowered through the rotating shield plug and upper support plate, using the visual aid afforded by the borescope, until it contacted the outer surfaces of the orifice holder retainer spring fingers in the lower support plate. Further downward motion compressed the fingers, causing them to disengage from the orifice retainer ring and causing the orifice holder to be trapped in the mouth of the tool's cup-like attachment. The orifice was then removed from the reactor by withdrawing the tool and captured orifice holder up through the rotating shield plug penetration. The radiation level at the surface of the removed stainless steel orifice was approximately 200 mr/hr.

The insertion of the new orifice, made of Inconel X, was accomplished by attaching a second device on the end of the tool. The insertion attachment contained a number of spring fingers which were inserted inside the new orifice holder and held it firmly on the end of the tool. Following insertion of the tool and attached orifice down through the shield plug and upper support plate to the point of contact with the orifice retainer ring in the lower support plate position N07-P00, further downward motion engaged the orifice retainer fingers in the retainer ring. The orifice could be felt to snap in

place. The tool was then removed from the orifice by a single upward pull, which caused the spring fingers of the orifice to deflect radially inward, leaving the orifice holder in its final position. Positive evidence that the orifice had been fully inserted and that the orifice holder was fully secured was determined from the fact that the insertion tool could be pulled free of the holder.

B. SAFETY ROD REMOVED FROM NO. 1 GUIDE TUBE

The safety rod which was found stuck inside the No. 1 lower safety rod guide tube last month (see APDA-CFE-31, Sec. III. D. 2) could not be transferred as intended to the empty No. 3 lower safety rod guide tube in the normal manner using the offset handling mechanism. The rod was removed from the No. 1 guide tube and transferred to the new position this month using a specially designed rod-stripping tool, shown photographically in Figure 18. The tool consists of an outer housing tube containing an inner gripper mechanism. The gripper assembly, which has fingers for grasping and withdrawing the safety rod up inside the housing tube, was an existing gripper fabricated in 1961 and was originally used to remove dummy safety rods from the guide tubes during the preoperational testing of the reactor at a time when the entire core had been loaded with dummy subassemblies and dummy safety rods for tight core and alignment tests. The outer housing tube added around the gripper has flanges on its lower face which were used to restrain the lower guide tube while the gripper pulled the safety rod upward. A dynamotor was also added on the gripper to ensure that the predetermined maximum allowable lifting force of 150 pounds was not exceeded during the attempted rod withdrawal (see APDA-CFE-31, Sec. III. D. 2).

The rod-stripping tool was inserted into the reactor through one of the rotating plug access penetrations on April 11, following the orifice replacement. The stuck safety rod was removed from the No. 1 position N02-P02, the plug was rotated, and the rod was deposited in the vacant No. 3 position P02-P02 without difficulty. No measurable lifting force in excess of the eighty-five pound frictional force of the tool packing gland seal between the two components plus the thirty-five pound weight of the safety rod could be detected when the safety rod was removed from the guide tube. This indicated that the restriction holding the safety rod inside the guide tube was small.

C. INVESTIGATION OF HIGH TORQUE ON NO. 1 PRIMARY SODIUM PUMP

The shaft binding problem observed last month in the No. 1 primary sodium pump is still under investigation (see APDA-CFE-31, Sec. III. C). The source of the trouble is not yet known; however, it has been found that although the turning torque varies noticeably when adjusting the motor and pump shaft vertically through their allowable operating range, at certain

FIG. 18 TOOL USED TO REMOVE STUCK SAFETY ROD FROM GUIDE TUBE

FIG. 20 DAMAGED CONTROL TRANSFORMER IN NEUTRON DETECTOR COOLING FAN CIRCUIT

FIG. 19 GROOVED SLIP RINGS ON ROTOR OF NO. 1 PRIMARY PUMP

elevations the shaft turns freely and is well within the normal operating torque (approximately 100 ft-lb). Meanwhile, inspection of the drive motor during pump disassembly has revealed a leaky gasket in the pony motor reduction gear oil reservoir mounted on top of the main motor housing. This apparently allowed oil to leak down and deposit on the main motor windings. When the motor was disassembled for cleaning and gasket replacement, the rotor slip rings were found to be badly grooved and scarred from apparent numerous arc-overs (see Figure 19). It is believed that such an arc-over occurred on October 8, 1968, when a sudden and unexpected flow increase and change in pump tank sodium level were observed for this loop (see APDA-CFE-20, p 20, and APDA-CFE-28, p 26). The accumulation of dirt in the oil on the insulation between the slip rings over an extended period of time had apparently resulted in sufficient surface conductivity that arc-over occurred, causing the pump motor to speed up to its maximum speed. Definite evidence of melting can be seen on the slip rings in comparison with the mechanical type of wear normally seen. The rotor of the No. 1 primary pump has since been removed for repair and machining of the slip rings. Following return of the rotor, the reason for the high pump shaft torque will be further investigated during reassembly of the pump.

D. REPAIR OF NEUTRON DETECTOR COOLING FAN

On April 5 a control transformer located in the ventilation building in the blower control circuit of the neutron detector cooling fan was found burned out as a result of a short circuit due to corroded terminals (see Figure 20). The motor leads to the cooling fan located outside the building were also found damaged at a terminal on the fan housing. Detector cooling was not required at the time and the circuit was shut down and temporarily taken out of service during replacement of the damaged control transformer, corroded terminal screws, and damaged section of motor lead cable.

E. REPLACEMENT OF PRIMARY SHIELD TANK BOOT SEAL

1. Background

Work has begun on replacing the primary shield tank (PST) boot seal, from which excessive PST nitrogen cover gas leakage was found to be occurring in June, 1968 (see APDA-CFE-23, Sec. III. B). The original boot seal was a 10-inch-wide, 5/8-inch-thick, 64-foot-long, canvas-web-reinforced styrene-butadiene rubber belt that circled the inner periphery of the primary shield tank and effected a seal between the PST and machinery deck platform attached to the top of the reactor vessel (see APDA-CFE-23, Figure 9). The boot seal, designed for 200 F operation, provided sufficient slack to accommodate a 2-inch-vertical and 1/2-inch-radial differential thermal expansion of the reactor vessel relative to the PST during reactor approach to power and operation.

A borescope inspection of the boot seal, made last year in the narrow annulus where it is located, revealed the failure to be due to extensive longitudinal cracking of the rubber seal material. It is believed that the cracking was caused by excessive temperature gradients and overheating in the vicinity of the seal since the October 5, 1966, incident. Extensive use of external heat input to the reactor from resistance heaters installed on the vessel and shield has been made periodically during this time due to the inability to operate the primary pumps at full flow and obtain the required heat input from mechanical sources.

The failed seal area has been temporarily sealed with plastic and tape since the time of its discovery last year, and the PST nitrogen cover gas system was changed from a pressure control system to a flow-through system to minimize leakage until repairs could be made.

2. New Boot Seal

The new boot seal, an improved design which is thinner and much more flexible than the old seal, is made of 1/4-inch-thick fiber-glass-reinforced silicone rubber. The new seal has a maximum operating temperature limitation of 350 F compared to 200 F for the old seal and is designed for a maximum pressure differential of 6 psi. The 1/4-inch-thick seal was manufactured in several layers; the required 64-foot length was attained by splicing the seal together from six shorter lengths using a combination of cementing, sewing, and covering with Silastic compound to effect the joints. The final closure joint will be made in the same manner in the field during the seal installation. The method of seal installation to be used will allow the new boot seal to be more easily maintained than the old seal.

Simple replacement of the old boot seal could not be made. Although the seal is fastened at its upper end by a readily accessible clamp ring attached to the floor of the machinery deck, the lower clamp ring is almost inaccessible from the top of the floor. It is attached to the primary shield tank wall approximately one foot deep in the narrow three-inch-wide space formed between the machinery deck and primary shield tank wall (see Figure 9 of APDA-CFE-23). It was originally intended that if the boot seal had to be replaced, access to the lower clamp ring bolts would be made through manholes built into the machinery deck floor adjacent to the annulus where the seal is located. The machinery deck compartment below the floor in this region was originally empty and would have allowed ready access to the lower bolts; however, during reactor construction this space was filled by steel shot and steel bars to provide extra shielding in this area, thus eliminating access to the lower bolts.

For this reason, a modification has been made in the manner in which the old boot seal will be removed and the new boot seal fastened in place (see Figure 21). The upper clamp ring will first be removed to allow

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FIG. 21 NEW BOOT SEAL BETWEEN PRIMARY SHIELD TANK AND MACHINERY PLATFORM

access to the old seal and the upper portion of the old seal will be removed from the annular space by cutting it longitudinally at an elevation just above the lower clamp ring around the entire 64-foot periphery of the primary shield tank wall. The lower clamp ring and lower portion of the old seal will be left in their present position and will not be removed. The upper clamp ring of the old boot seal will then be used as the lower clamp ring for the new boot seal, and a new upper clamp ring for the new seal will be installed on the inner surface of the primary shield tank wall above the lower clamp ring located at deck floor level. With this arrangement the new seal will be located entirely above the elevation of the deck structure and will be much easier to inspect and maintain than the old seal. Also, this design minimizes the possibility of excessive stresses being created in either the seal or reactor vessel because of an abnormally large upward thermal expansion of the vessel which, in the old design, could stretch the seal taut.

3. Precautions to Prevent Contamination of the Primary Shield Tank Atmosphere

Several precautions are being taken to limit the leakage of reactor building air and moisture into the nitrogen atmosphere of the primary shield tank during the boot seal replacement. The moisture content of the air in the reactor building will be kept low using dehumidifiers and only a small portion of the work area on the seal will be exposed to the atmosphere at any time. All open seal spaces except those where work is being performed will be filled with flexible foam rubber tubing. A slight positive pressure of the nitrogen inside the shield tank will also be maintained, and the tank will be continuously purged with a small flow of nitrogen during the boot seal replacement. The primary shield tank temperature will be kept below 400 F during the boot seal replacement. The tank will be fully purged when the seal replacement job is completed.

Although precautions are being taken to ensure that inleakage of air and moisture to the PST are kept to a minimum, a certain amount of inleakage is inevitable because of the nature of the work; consequently, limits have been set on the allowable oxygen and water vapor contamination levels in the PST during the work to avoid possible formation of corrosive agents or attack on the borated graphite shielding material. The revised technical specifications allow maintenance to be performed with air in the PST atmosphere if the temperature of the primary system (or shield graphite if the system is drained) is less than 400 F and the dew point of the PST atmosphere is less than +80 F. If the temperature exceeds 400 F during maintenance, oxygen and carbon dioxide maximum concentration limits of 1,000 ppm (less than 0.1 v/o) apply, and the dew point limit becomes -35 F.

Also, the PST gas supply during maintenance will be nitrogen at a flow rate of less than 200 cfm, or less than 200 cfh if the primary system temperature rises above 400 F. Following maintenance, the graphite will

be outgassed with nitrogen until the oxygen and carbon dioxide levels are less than 0.1 v/o and the dew point of the PST atmosphere is less than -35 F. During subsequent reactor operation the same limitations as those given above for temperatures above 400 F will apply for oxygen and carbon dioxide concentrations, dew point, and PST gas supply flow. Prior to the boot seal replacement the only limitation on the PST atmosphere was that oxygen levels had to be less than 1,000 ppm. The primary shield tank atmosphere will be sampled periodically during the boot seal replacement program to verify that these limits are being maintained.

4. Boot Seal Replacement

Replacement of the boot seal was begun on April 21. Prior to the actual cutting of the old seal and its removal, a nitrogen purge rate of 3 to 7 cfm was established through the PST to minimize the effects of reactor building air inleakage. The containment building blast fans were also operated to mix the containment building air with any nitrogen leakage from the PST. As a further precaution, the air in the work area was sampled continuously using the reactor building oxygen meter, and a flame lamp was positioned near the work area to provide indication of the quality of the air being breathed by the workers.

Following the unbolting and removal of the upper clamp rings for the old boot seal at deck level to provide clearance in the 3-inch-wide annulus for access to the old seal, the old seal was cut in half longitudinally and the upper half was removed. Various methods of cutting the seal were investigated during the past several months. Use of a milling tool was considered but was found to be unsatisfactory due to the inward curvature of the seal which prevented the tool from reaching the seal. Use of a tungsten wire to burn the seal was also considered but was found to be too slow and messy with much smoke being produced; furthermore, the tungsten wire was found to burn out quickly. A small jigsaw device to fit into the narrow annulus was next developed but was inefficient because the boot seal tended to move with the stroke of the saw blade, thus inhibiting the cutting action. The cutting tool finally decided upon and used was a specially shaped cold chisel, shown in Figure 22, having a curved steel blade 4 inches wide and one foot long, and driven by a small hydraulic jack located under the flange stiffening ribs of the PST, as shown in Figure 23. No dust or chips resulted from its use. The cut pieces of the 64-foot-long boot seal were removed from the annulus in sections using a cable noose and a come-along (see Figures 24 and 25). Following the boot seal removal, the annulus was temporarily sealed with tape and sections of flexible foam rubber tubing. It had been estimated that three weeks would be required to remove the old boot seal because of its inaccessibility, because of interferences at the machinery deck level (see Figures 26 and 27), the high heat, and the necessity to minimize air inleakage. However, the actual boot seal removal was completed in one week on April 26. The maximum dew point measured for the PST

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**FIG. 22 COLD CHISEL USED TO CUT OUT OLD
BOOT SEAL**

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FIG. 23 TWO VIEWS OF HYDRAULIC JACK USED TO DRIVE CHISEL

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**FIG. 24 REMOVAL OF PORTION OF
OLD BOOT SEAL FROM
ANNULAR SEAL SPACE**

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**FIG. 25 CUT SECTIONS OF OLD
BOOT SEAL**

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**FIG. 26 VIEW OF OBSTRUCTIONS
ENCOUNTERED DURING
BOOT SEAL REMOVAL**

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**FIG. 27 CABLES LIFTED TO PROVIDE
ACCESS TO BOOT SEAL**

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**FIG. 28 NEW UPPER CLAMP
INSTALLED ON PST WALL**

atmosphere during the removal operations was +35 F and the graphite temperatures did not exceed 300 F.

Work has now begun on installing the new upper clamps on the PST wall for use in fastening the upper end of the new boot seal (see Figure 28). A stud gun is being used to weld 1/2-inch studs around the periphery of the primary shield tank after which the new upper clamps will be slipped over the studs and fastening nuts attached. In several instances interferences with existing piping and junction boxes required that the clamp ring segments be cut out and shaped around the interferences.

VI. SPECIAL INVESTIGATIONS

A. LEAK TEST OF CONTAINMENT BUILDING PENETRATIONS

On April 3 the containment building electrical penetrations were leak-tested at 32 psig by the normal method (see APDA-CFE-7, p 25, and APDA-CFE-20, p 41) and the total leakage measured was 443 scfd. This leak rate was considered unsatisfactory based on comparison with previous results and on the fact that the total leakage from all containment building penetrations as determined from the most recent leak-test data was 1203 scfd, in excess of the 1,000 scfd limit given in the technical specifications.

Inspection revealed that one of the electrical cables had a damaged sheath at its penetration outside the containment building which was allowing excessive leakage (see Figure 29). To repair the sheath and effect a new seal, it was necessary to splice 10 feet of new cable through the penetration. Another leaking electrical penetration was also found; its leakage was reduced to an acceptable level by repotting the space between the gland nut and cable sheath with Weather Band sealing compound (see APDA-CFE-7, Figure on p 26). Following the repair work, the electrical penetrations were again leak-tested on April 21. The leakage rate on the two repaired banks was 18 scfd and the total measured leakage from all electrical penetrations was only 115 scfd. In addition, the total leak rate from all building penetrations was reduced to 875 scfd, well within the 1,000 scfd limit at 32 psig.

B. PRESSURE TEST ON FUEL AND REPAIR BUILDING WASTE GAS DISCHARGE LINE

Tests were made this month on the waste gas discharge line which runs from the fuel and repair building to the waste gas building to determine its flow-versus-pressure effect on the main waste gas collection header located in the waste gas building. Several occurrences have been noted in the past which indicate that the collection header pressure had been raised sufficiently during waste gas discharge from the fuel and repair building to cause a back-pressure effect on the gas chromatograph used to analyze reactor cover gas which also discharges to the waste gas collection header. This resulted in temporary malfunction of the instrument. The flow tests on the waste gas line were made as an aid in determining the maximum discharge rate which can be allowed from the fuel and repair building during normal operation of the waste gas system without affecting the operation of the gas chromatograph.

A simplified flow diagram of the fuel and repair building waste gas discharge to the waste gas collection header and the gas chromatograph

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**FIG. 29 DAMAGED SHEATH AT
CONTAINMENT BUILDING
ELECTRICAL PENETRATION**

discharge to the header is shown in Figure 30. Briefly, the system operates as follows: In the fuel and repair building, waste gas is pumped from surge tanks through a compressor to the waste gas receiver tanks and is then discharged into the waste gas system header through a pressure control valve set to automatically open at 37 psia; the waste gas collection header, which normally operates at 3 to 4 psia, is discharged through the waste gas compressor to either the waste gas stack or to the decay storage tanks located in the waste gas building.

The gas chromatograph discharges to the header from its location in the fission product detector building east of the containment building. It operates on a principle whereby the gaseous components of a process stream are separated differentially in an absorbent column, and thermal conductivity detectors (thermistors) are used to determine the concentration of the components as they are released at varying known times after separation from the absorption column. The reactor cover gas sample enters the fission product detector building through a bypass line, where it is split, part of it being distributed to the precipitation-type fission product detector which exhausts back to the reactor cover gas system, and the remainder being distributed as a process stream to the gas chromatograph which then discharges to the waste gas system. The gas flow through the analyzer is induced by the vacuum in the waste gas collection header. A back pressure control system consisting of a vacuum tank and pressure regulating valve is installed in the exhaust line between the chromatograph and waste gas collection header to maintain a constant pressure of 12 psia in the analyzer system. This is done because the pressure of the collection header normally fluctuates somewhat during operation and any pressure fluctuations at the analyzer are known to result in unreliable results. The earlier difficulties during operation of the gas chromatograph were thought to have resulted when discharges from the waste gas receivers in the fuel and repair building were larger than the waste gas compressor could handle. The consequent pressure build-up in the waste gas collection header might have then exceeded 12 psia, resulting in back pressure build-up behind the pressure regulating valve in the gas chromatograph exhaust line and in the analyzer tubing. Two tests were conducted to investigate this possibility further.

In the first test, the waste gas receiver tanks in the fuel and repair building were manually discharged from a pressure of 37 psia to the waste gas collection header with the building waste gas compressor turned off so as to avoid a continuous feed of new gas into the receiver tanks during the discharge. Pressure and flow measurements were made at various locations throughout the system during the 3-1/2 minute discharge time, and it was found that the pressure in the waste gas receivers rapidly decayed to a low value while the maximum pressure build-up in the waste gas collection header did not exceed 12 psia. The gas chromatograph was observed to operate normally with no malfunctions during the discharge. In the second test, the measurements were repeated with the waste gas compressor in the

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FIG. 30 FLOW DIAGRAM FOR FUEL AND REPAIR BUILDING AND GAS CHROMATOGRAPH EXHAUSTS TO WASTE GAS

fuel and repair building operating during the discharge so as to continuously add new gas to the waste gas receiver tanks. In this case it was found that the pressure in the receiver tanks did not drop as fast, or to as low a value, as in the earlier test. Also the maximum pressure build-up in the waste gas collection header was 13.5 psia. This resulted in back pressure on the gas chromatograph due to reversal of the pressure differential across the pressure control valve of its exhaust line. Indications of this were given by a 50% reduction in sample flow through the chromatograph and a consequent operating malfunction of the device. The flow disruption and malfunction of the chromatograph began approximately 3 minutes after the start of the discharge and lasted for about 2 minutes before returning to normal.

On this basis, it is believed that satisfactory gas chromatograph operation can be maintained if the pressure in the waste gas collection header can be held below 12 psia during waste gas discharges from the fuel and repair building. Ways in which this can be done are being studied; however, it appears feasible that the back pressure build-up in the waste gas collection header can be suitably limited by adjusting the controls on the pressure regulating valve which controls the discharge from the waste gas receiver tanks in the fuel and repair building so that the valve opens automatically at 30 psia instead of its present setting of 37 psia.

VII. MISCELLANEOUS ACTIVITIES

A. INSTALLATION OF ORIFICES IN INNER RADIAL BLANKET SUBASSEMBLIES

In anticipation of resumption of reactor operation later this year, the installation of flow orifices in the nozzles of thirty-five new inner radial blanket subassemblies has begun. The new subassemblies, acquired with the first core and presently in storage at the plant, will replace those in the reactor at the time of the incident. Some difficulties were encountered during early attempts to install the orifices due to the extra fine thread used on the orifice. This resulted in difficulty in engaging the first thread, high torque after thread engagement, and galling of the threads due to stainless steel-to-stainless steel material abrasion. The orifice plug threads have since been modified by rounding off the tops of the threads to achieve a small reduction in the major diameter of the thread. The modified orifices have been found much easier to install than the original orifices. To date orifices have been installed in only four of the thirty-five subassemblies because of the delay for redesign and machining of the orifices.

B. SIMULATOR TRAINING OF PLANT OPERATORS

Also in anticipation of reactor start-up, simulator training of the plant operating staff has been under way for the past several months. The plant simulator consists of an analog computer (Berkeley EASE Series 1000) located in the relay room of the main reactor control building. An attached control console, which is a duplicate of the actual control equipment used in the plant, allows the computer to be used as an effective operator training facility. The computer simulates the plant operating control system; and from an instructor's console, it is possible to insert malfunctions into the system without the operator's knowledge, thus enabling operators to become familiar with emergency operating procedures as well as normal start-up and shutdown procedures. Operator performance during simulator training is checked by recording instruments. A total of 180 man hours has been spent in start-up training on the plant simulator since the beginning of the year.

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