APDA CFE-9

MONTHLY REPORT No. 9 - APRIL 1967

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT

AEC CONTRACT No. AT (11-1)-865
PROJECT AGREEMENT No. 15

SPONSOR: DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY, UNITED STATES ATOMIC ENERGY COMMISSION

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DISCLAIMER

REPORT No. 9 - APRIL 1967

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

Items A and B in Section II are usually reported each month; items in the other sections 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 one megawatt thermal, 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 this time the reactor has been shut down while the cause and extent of the damage are being investigated.

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 Report No. 1.

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.

SECTION I

CURRENT EXPERIENCE SUMMARY

The unloading of subassemblies from the reactor vessel lattice, for viewing and removal of the stuck-together pair of subassemblies was completed. Eleven subassemblies in the core and one outer radial blanket (ORB) subassembly were removed to permit future viewing alongside the stuck pair (M098 and M127). Two ORB subassemblies were taken out of the lattice to complete the opening up of an area into which the pair will be swung to avoid interferences during the removal operation.

Subassembly M091 was placed in a shipping pot at the Fuel and Repair Building (FARB) transfer tank and the pot was loaded into the multipurpose shipping cask and sent to the Battelle Memorial Institute hot lab. The shipping pot was not filled with sodium. This was the first operating experience with the pot, special transfer equipment and the cask; all performed satisfactorily. However, difficulty was experienced later under sodium at the reactor vessel exit port with a gripper of the same design as the one used to transfer the M091 pot. The gripper would not disengage the shipping pot in which subassembly M140 is to be shipped. The gripper and pot were removed from the exit port and taken to the FARB glove box in a transfer cask specially modified for this operation. It was necessary to disassemble the gripper to remove it from the pot.

All parts of the fuel inspection facility are on hand or scheduled for early May delivery. Assembly of the facility on the reactor building operating floor was about 30 percent completed at the date of this report.

Work commenced on piping and ventilation modifications necessary to permit draining of the reactor vessel sodium for examination and removal of the stuck pair. The special removal tool was placed on order for late August delivery. New, improved sodium level detection probes were installed in the primary sodium storage tanks in preparation for the sodium drain operation.

SECTION II

PLANT OPERATIONS

A. Reactor Unloading

Twelve additional subassemblies, located in a north, northwest direction from the bonded pair, were removed in April. It is planned to lower the borescope and a light source into this space to examine the stuck-together pair (M098 and M127).

Of the 12 subassemblies removed, 10 were core subassemblies, one was an IRB and one an ORB subassembly. Three core subassemblies were moved to other vacant core positions, six were transferred to storage positions at the outside of the reactor lattice and one core subassembly, together with the IRB and ORB subassemblies, was placed in the transfer rotor.

Two additional outer radial blanket (ORB) subassemblies, M942 and M972, were removed from the reactor from locations NO7-PO6 and NO7-PO7 and placed in the transfer rotor. Fifteen ORB subassemblies had been removed from this area in March. The removal of the additional two subassemblies completes the opening up of an area to permit the extraction of the stuck pair by swinging them into this area as described on Page 18 of Report No. 6.

The locations where the 14 subassemblies were removed are shown on the diagram on Page 10. The diagram also shows the lattice positions where 9 of the 14 subassemblies were relocated. The chart of subassembly locations on Page 11 summarizes the present locations of the various type subassemblies which were in the reactor at the time of the October 5, 1966, fuel damage incident. At present there are 22 vacant positions in the outer radial blanket and 25 in the core, including the No. 3 safety rod position. No further reactor unloading is planned in the immediate future.

The following listing summarizes the reactor unloading moves made during April 1967.

Subassy No.	Type	Former Location	Present Location
M972	ORB	NO7-PO7	Transfer Rotor
M942	ORB	NO7-PO6	Transfer Rotor
S733	ORB	NO7-NO2	Transfer Rotor
M008	Core	NO6-POO	NO9-N11
M359	IRB	NO6-NO1	Transfer Rotor

Subassy No.	Type	Former Location	Present Location
M152	Core	NO6-NO2	NO3-PO6
M138	Core	NO5-PO1	PO2-NO1
M128	Core	NO5-POO	P11-NO9
M132	Core	NO5-NO1	NO1-PO3
M162	Core	NO5-NO2	Transfer Rotor
M110	Core	NO4-PO1	N11-PO9
M100	Core	NO4-POO	P11-PO9
M103	Core	NO4-NO1	PO9-N11
M070	Core	NO3-PO1	NO9-P11

B. Sodium and Gas Systems Performance

1. Sodium Cold Traps and Plugging Indicators

April Operating Data	Primary System	Secondary System Loop 1	Loop 2	Loop 3
Cold Trap Operation (Hours)	121	*	627	4 (0001 to 0400 on 4-1-67)
Maximum Plugging Temperature - F	Below 220	*	Below 220	240
Minimum Plugging Temperature - F	Below 220	*	Below 220	Below 220

* Sodium drained for steam generator tube sheet rewelding.

2. Primary System Cover Gas Analysis

	Reactor Cover Gas (Argon) ppm by Volume	Primary Shield Tank Atmosphere (Nitrogen) ppm by Volume
Oxygen	Below 25	270*
Carbon Monoxide	Below 10	Below 10
Carbon Dioxide	Below 10	70
Hydrogen	4**	5
Helium	Below 4	Below 4
Methane	Below 10	Below 10
N_2O	Not measured	Below 10
Argon	Remainder	13,000
Nitrogen	1920	Remainder
Dew Point	Not Measured	Below -40 F
Sample Date	April 14, 1967	April 14, 1967

* Technical Specifications state 1000 ppm maximum
** 10 ppm is the recommended maximum

3. Primary System Gas Activity

Location	Sample Date	Concentration (microcuries/cc)
Reactor Cover Gas	Apr 14	5.5×10^{-6}
Reactor Cover Gas	Apr 27	3.2×10^{-6}
Primary Shield Tank	Apr 14	6.1×10^{-7}
Primary Shield Tank	Apr 27	Below Background

4. Primary Sodium Chemical Analysis - April 1967

Oxygen	11, 12, 12	Iron	0.5
Carbon	85, 85, 94, 127 130, 135, 154	Nickel	Below 0.1
		Chromium	Below 0.1
*Hydroxide Hydrogen	1.1, 1.2, 1.3		
*Non-Hydroxide Hydrogen	0.3, 0.3, 0.3		

* 1.3 ppm recommended maximum for total hydrogen

Note: Values are in ppm by weight. One sodium sample coil was analyzed at several different points along its length to provide the separate readings indicated.

C. Shipment of Subassembly M091

Subassembly M091 was loaded into the new multipurpose shipping cask in early April but shipment was delayed several weeks awaiting license approval of the new cask. The approval was received on the morning of April 26 and the truck shipment departed within the hour, arriving at the Battelle Memorial Institute hot lab in Columbus, Ohio, the same day.

The shipment was escorted by plant health physics personnel. The maximum radiation level at the surface of the shielded shipping cask was 0.17 mr/hr. Subassembly M091 was contained inside a sealed shipping pot which was in turn sealed inside the shipping cask.

As reported in Section II E 2 of Report No. 8, M091 has been sent to Battelle for examination because it had been observed to have an outlet sodium temperature rise (delta T) approximately 40 percent higher than predicted during July-August 1966 reactor operation and also on October 5, 1966. At the Battelle hot lab the fuel pins will be examined for plugging or damage. The Battelle hot lab had been inerted with nitrogen because the subassembly was not steam cleaned before shipment and therefore had a small amount of sodium on it. At the date of this report the shipping pot had been taken out of the cask and the subassembly removed from the pot. The exterior appeared to be in good condition. The subassembly was observed to be coated with a thin film of sodium oxide when it was first removed from the shipping pot. Examination is continuing and further results will be included in the May report.

The cask, pot and gripper (which seals the top of the pot) will be decontaminated and are scheduled to be returned promptly to Fermi. The cask next will be loaded with subassembly M140 for shipment to Battelle, probably in May. Subassembly M140 had an outlet sodium temperature rise (delta T) on October 5th approximately three times higher than predicted. Plans are to load M140 into the cask at the reactor vessel exit port.

All equipment used in loading M091 into the shipping cask performed satisfactorily. The transfer was made from the transfer tank in the Fuel and Repair Building. The cask was positioned on the special support platform which was located over the transfer tank access tube near the center of the FARB operating floor (see Pages 31 and 32 of Report No. 5).

A special shipping pot, designed to rest in the transfer tank with the top above the sodium level, was lowered through the shipping cask into an empty position in the transfer tank rotor. The pot was rotated under a viewing window in the steam cleaning cell and it was verified that no sodium had spilled into the pot. Subassembly M091 was temporarily raised out of the sodium in the transfer tank and into the steam cleaning machine where it was allowed to drip drain. The pot was then indexed under the machine, M091 was lowered into the pot and the pot was indexed under the cask. A special gripper, that could be clamped and sealed to the top of the pot, was used to raise M091 and the pot into the cask. The top and bottom openings of the cask were then sealed thus providing a sealed argon atmosphere with a minimum of sodium within the pot. The upper left photograph on Page 14 shows the gripper-pot engagement being tested. In the upper right photograph the cask is pivoting on to the shipping cradle. A view of the various parts of the gripper is shown in the lower photograph.

KEY:

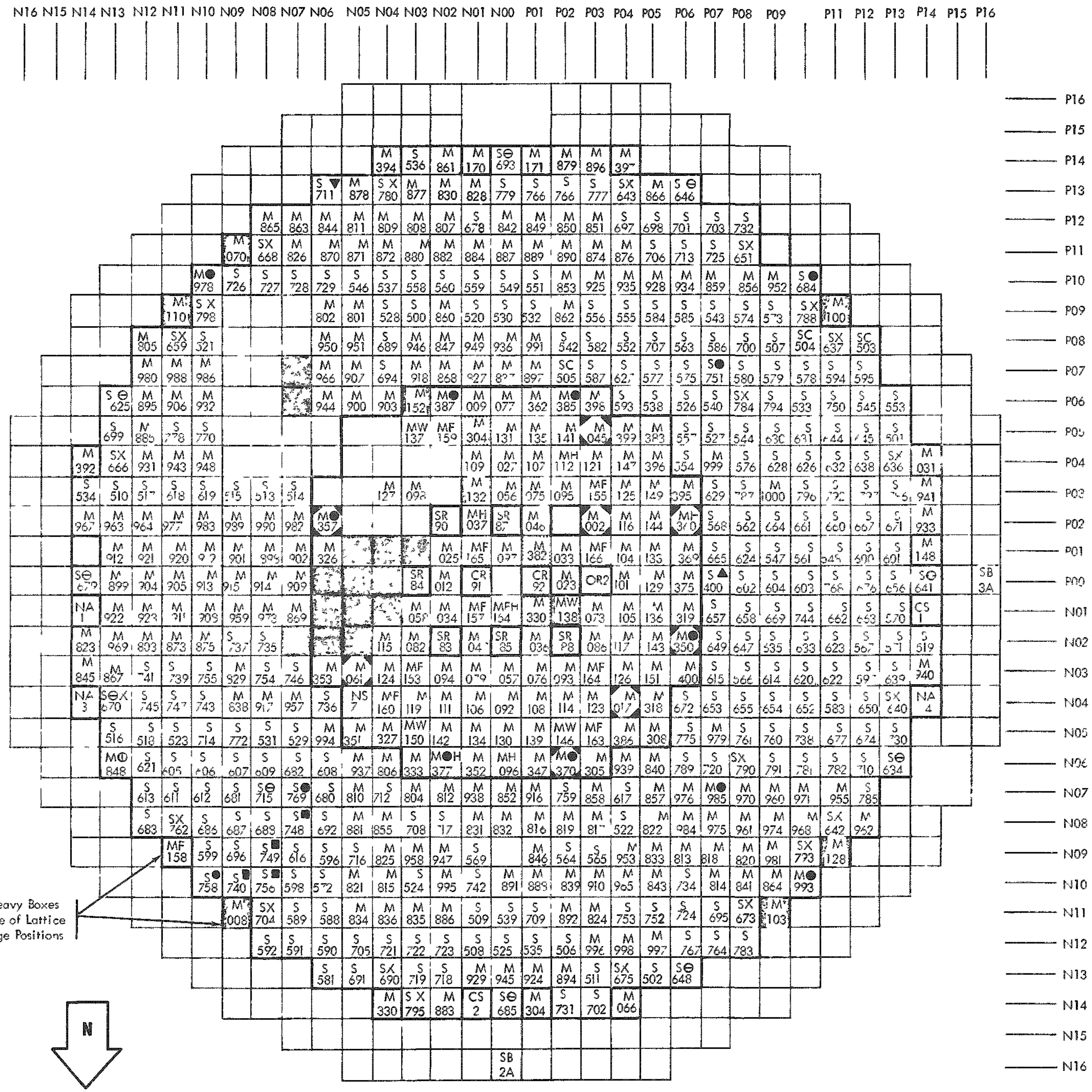
- CR Control Rod No. 430169-
- SR Safety Rod No. 430192-
- OR Oscillator Rod
- CS Core Shim Subassembly
- CF Core Foil Subassembly
- BF Blanket Foil Subassembly
- CT Coarse Filter, Take-apart, Dummy Core Subassembly
  NOTE: Dummy Core subassemblies in the reactor meet "Core A" core subassembly specifications and bear the suffix "CF"
- NA Sodium Worth Subassembly
- NS Neutron Source
- TIT Temporary Instrument Thimble
- MS APDA Materials Surveillance Subassembly
- M Subassembly Manufactured by D.E. Makepeace Co.
- S Subassembly Manufactured by Sylcor Division, Sylvania Electric Products Co.

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

Units shown without prefix are dummy outer radial blanket subassemblies.

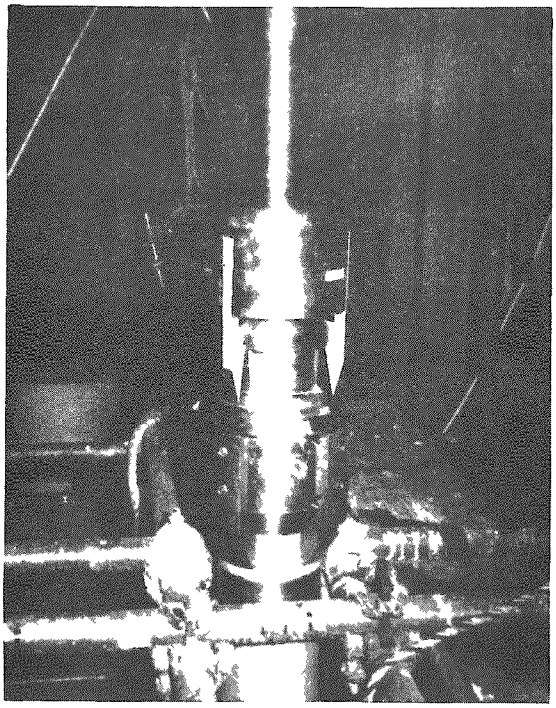
- Oversize Nozzle Unit
- F "F" Subassembly (Contains fuel pins with high iron plus nickel, high carbon or Zirconium content.)
- W "W" Subassembly (Contains fuel pins with high iron plus nickel content.)
- ● Blanket slugs have high carbon content. (APDA Surveillance Program Unit)
- ■ Stringering in Blanket Slugs
- < Large Grain Blanket Material (Hash)
- Φ Larger Than Normal Spacing Between the Blanket Elements and the Support Grid
- ⊖ Type 347 Stainless Steel Wrapper Tube
- H Handling Head Short
- ▲ Test Flow Subassembly (S-400)
- ▼ Slugs Previously Used in a Test Subassembly
- C "CP" Slugs
- Locations Where Changes Were Made
- IRB'S Replacing Core Subassemblies

Note: Heavy Boxes at Outside of Lattice are Storage Positions

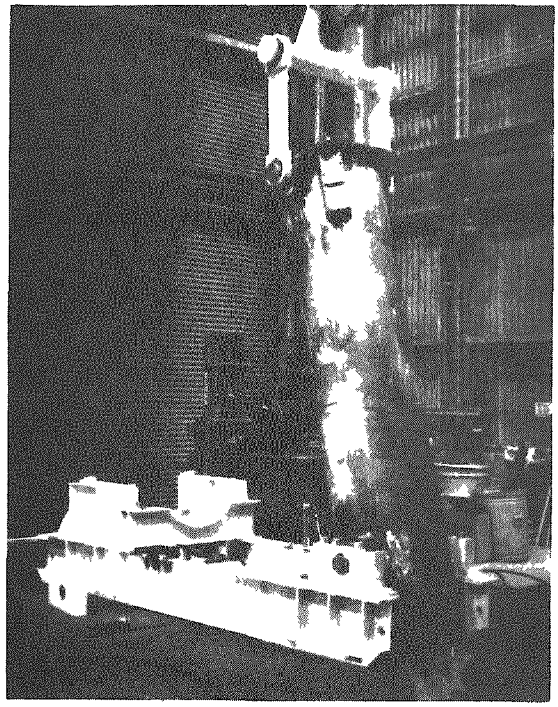


REACTOR LOADING 4-30-67

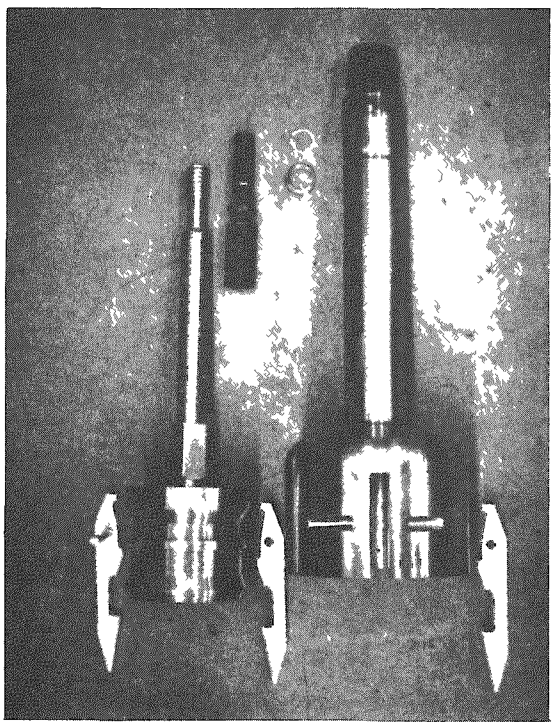
At the date of this report the gripper had been cleaned ultrasonically and it will next be disassembled to determine the cause of failure. Design modifications will be made to prevent unscrewing of the gripper shaft joints.



TEST OF GRIPPER ENGAGEMENT AND SEAL ON MO 91 POT



MULTIPURPOSE SHIPPING CASK PIVOTING ONTO ITS CRADLE



EXPLODED VIEW OF THE INTERNAL PARTS OF THE SHIPPING POT GRIPPER

SHIPMENT OF SUBASSEMBLY MO 91

SECTION III

FUEL INSPECTION FACILITY

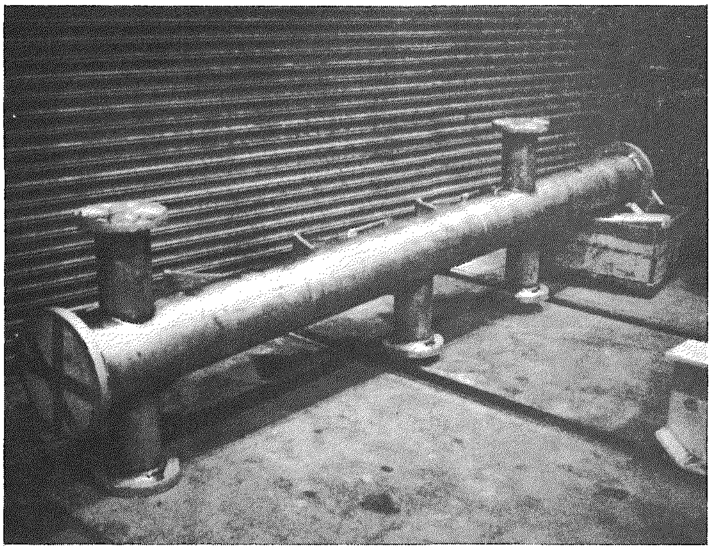
The electrical resistance heaters and lagging have been installed on the inspection chamber. The center photograph on Page 16 shows the band type heaters and part of the lagging in place. Insulation of the chamber has been completed as shown in the bottom photograph on Page 16. The top photograph on the same page shows the inspection chamber as received from the fabricator. During the month of May the chamber will be positioned on the support stand and the windows, lighting and valves are to be installed on the chamber. Field wiring of the power control panel has been completed. The subassembly gas flow pressure drop test equipment is on hand and early May delivery is expected for the periscope, dimensional check apparatus and the subassembly stabilizer. All other parts of the facility have been delivered.

The support stand is completed and in place over the exit port. The lower left photograph on Page 17 shows the stand ready to be positioned over the exit port. The stand was used to support the shipping cask during the loading of subassembly M091 as described on Page 9. The stand had previously been given a 50,000 pound static load test to demonstrate its strength.

A view of the shield housing is shown on Page 17. It is not planned to completely install the housing and lead block shielding until preoperational testing of the facility is finished. It will be possible easily to remove the inspection chamber, housing and shielding as a single unit from the support stand over the exit port, using the reactor building overhead crane.

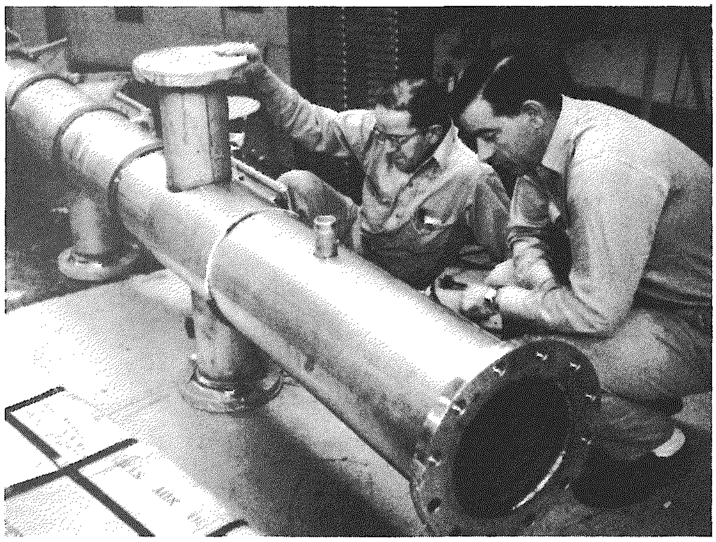
The subassembly hoist mechanism will be mounted on top of the inspection chamber. The hoist mechanism had previously been constructed for other lifting operations at the exit port. Several modifications have been completed to improve its performance. The hoist has been mounted in the repair pit in the Fuel and Repair Building and is ready for preoperational testing. See Page 17 for a view of the hoist at the repair pit with the hoist control panel.

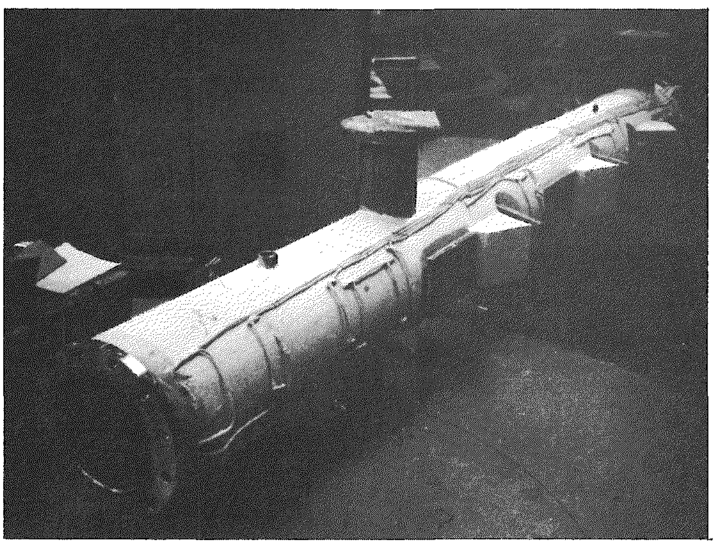
Further description of the inspection facility was given on Pages 25-27 of Report No. 5.



VIEW OF THE CHAMBER
AS-RECEIVED ASSEMBLY
IS ALL TP 304 STAINLESS

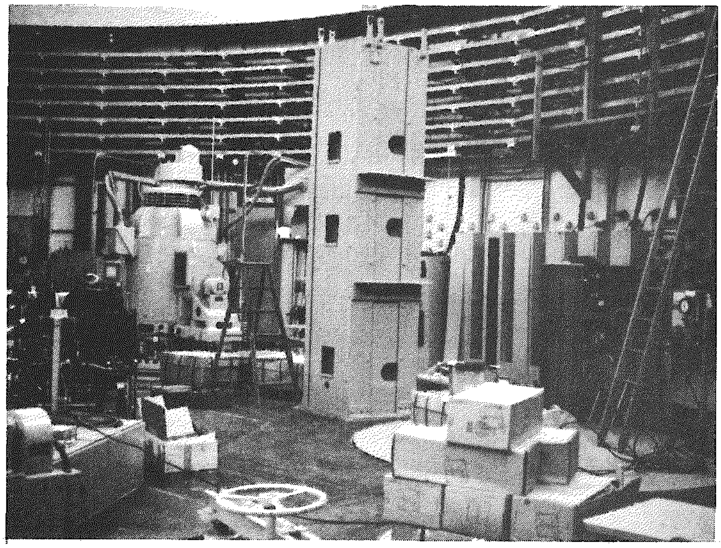
INSTALLING STAINLESS
LAGGING OVER THE
ELECTRICAL HEATER
ELEMENTS



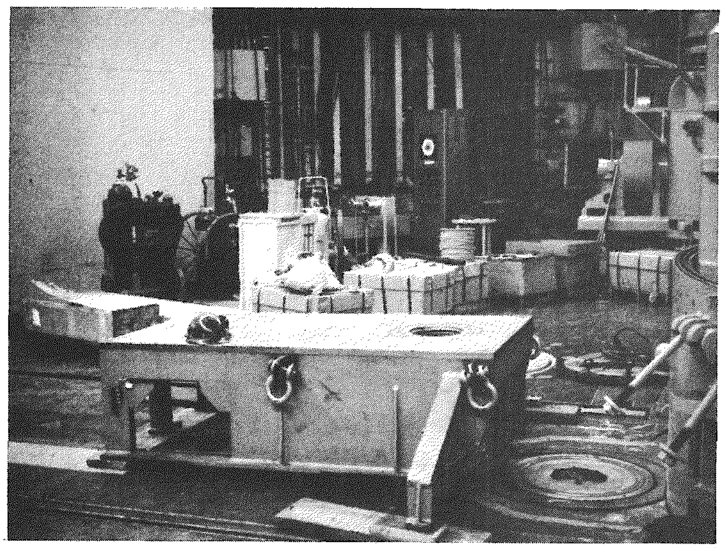


CHAMBER INSULATED
AND WIRED

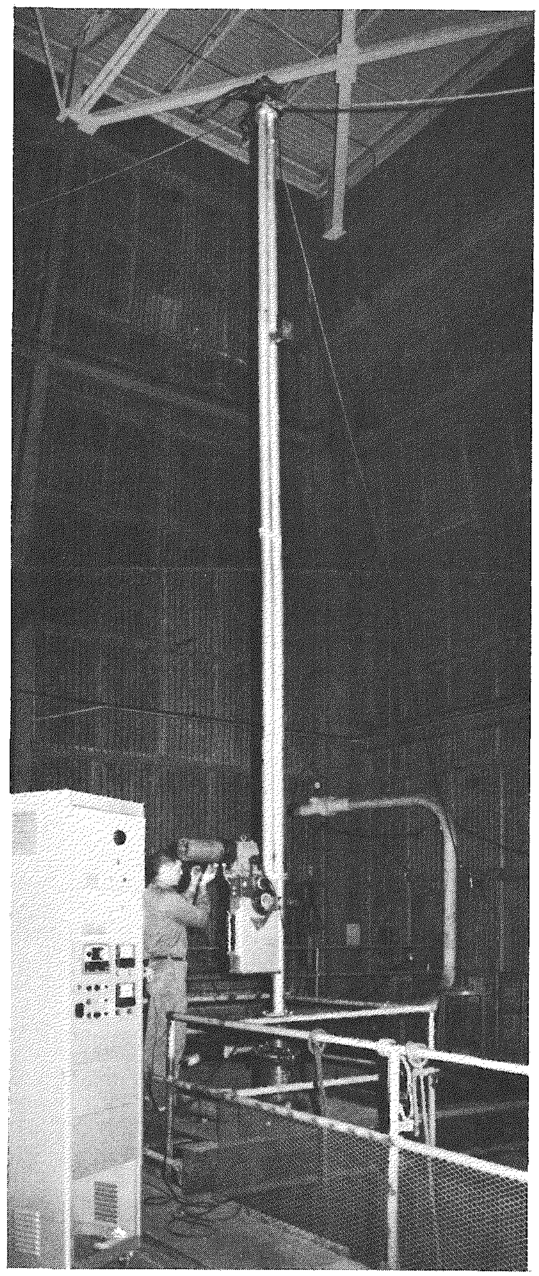
INSPECTION CHAMBER



SHIELDING SUPPORT HOUSING AND MISCELLANEOUS BOXED MATERIAL FOR THE INSPECTION FACILITY



FACILITY SUPPORT TABLE READY FOR INSTALLATION AT THE EXIT PORT



COMPOSITE PHOTOGRAPH OF HOIST MECHANISM TEST INSTALLATION IN THE FARB

INSPECTION FACILITY EQUIPMENT

SECTION IV

STATUS OF THE PAIRED SUBASSEMBLY REMOVAL PROGRAM

A. Pair Removal Tool

A special tool will be required to remove the subassemblies that are joined together (M098 and M127) from the reactor lattice, as described in Section III C of Report No. 6. Design of this tool has been completed and the tool was placed on order for delivery in late August.

B. Access Holes in the Rotating Shield Plug

The shielding was removed from around the top portion of the reactor sweep mechanism, which occupies one of the three closely spaced access holes in the rotating shield plug. The shielding, and a part of the sweep mechanism topworks, were removed to avoid interferences with the special pair removal tool and auxiliary equipment which will be inserted in the other two holes.

The access holes have shield plugs installed beneath their cover plates. The cover plates will be removed and flanged adapter pieces will be placed over the holes. The shield plugs will be drawn up into containers mounted on the adapters. The adapters and containers are on order and are scheduled for delivery in May.

Special sealing glands will be placed on the adapters after the shield plugs have been removed. The glands will mount and seal the equipment for draining and viewing of the interior of the reactor vessel. The sealing glands are scheduled to be delivered in June.

C. Viewing and Lighting Equipment

1. General

The corescope, borescope and a lead glass viewing window have been borrowed from the U.S. Atomic Energy Commission under a lease arrangement. This equipment, now on hand at Fermi, had been used by Atomics International for investigation of the damaged core at the SRE reactor.

The corescope and borescope can be adapted for either 90 degree or straight ahead viewing. Both instruments are being modified by adding cooling gas sheath tubes. Argon gas at a maximum flow of 40 scfm will pass down the annulus between the outside of the instrument and the sheath tube. Thermocouples will measure the temperature at the lower optics; the borescope is limited to 350 F and the corescope to 400 F. The cooling system is designed for an ambient temperature of 500 F inside the reactor vessel. The cooling gas sheath tubes are scheduled to be delivered in June.

2. Corescope

The 2-1/2-inch diameter corescope was designed and built by Atomics International. It is a one-piece instrument and will extend into the reactor vessel as far as the top of the subassembly handling heads. A separate light source, described in Section IV C 4, will provide illumination for the corescope.

The corescope will be used for panoramic viewing of the reactor lattice, for viewing of the bonded pair as it is removed and for viewing to assist in the insertion of the borescope through the lower support plate into the core inlet plenum.

3. Borescope

The 1-inch diameter borescope is assembled in 6-foot sections. An additional 6-foot section has been ordered from the original manufacturer, National Electric Instrument Company. The additional section will permit the instrument to reach down into the core inlet plenum.

There is a 45 watt light at the bottom of the borescope. This light source will be used during close-up scanning of the bonded pair. The borescope will also be used, with a separate light source, to examine the core inlet plenum and to check the lower nozzles of M098 and M127 as they are lifted to ensure that both nozzles are lifting.

4. Light Tube

The light tube has an outside diameter of 1-5/8-inches, the same OD as that of the cooling gas sheath around the borescope. This permits interchangability of the light and borescope in the sealing glands at the top of the rotating shield plug access holes. The light tube is scheduled for delivery in late May.

A 1500 watt Quartzline bulb is mounted at the lower end of the light tube. A stainless steel wire screen protects the bulb and a reflector provides control of the direction of the lighting. Cooling of the light is not required.

5. Viewing Window

Provision has been made for installation of a lead glass viewing window in a sealing gland at the top of one of the access holes in the rotating shield plug. The view will be restricted to a limited area beneath the access hole.

D. Mock-Up of the Removal Operation

It has been decided to build a full sized plywood mock-up of the vacated area of the reactor core adjoining M098 and M127, to test removal clearances. The mock-up will be installed in the bottom of the repair pit in the Fuel and Repair building. A pair of dummy subassemblies will be used to check out the operation of the removal tool. Provision will be made for simulating the rotation of the reactor shield plug. The lighting and viewing equipment will also be tested with the mock-up model.

E. Shipping Pot for M098-M127

Following the removal of the paired subassemblies (M098 and M127) from the core by the special tool, they will be deposited in a shipping pot located in the transfer rotor. The pot will have an extra large inside diameter (7-inches) to accommodate both subassemblies at the same time. The multipurpose shipping cask will be located over the exit port and the pot will be lifted into the cask by means of a special gripper at the bottom of a lifting rod. The gripper will be disconnected from the rod and remain with the pot to form the top plug. The shipping cask requires special inside rails to support the special pot. The pot, gripper and rails are scheduled to be delivered in August.

F. Reactor Vessel Sodium Drain Operation

It will be necessary to transfer the sodium from the reactor vessel to the storage tanks in order to examine and remove M098 and M127. See Page 23 of Report No. 6 for further details on this operation.

A 2-inch carbon steel syphon pipe will be inserted into the reactor vessel through one of the access holes in the rotating shield plug. Smaller extension pieces at the bottom of the 2-inch pipe will permit removal of the sodium from below the lower support plate (the core inlet plenum).

Approximately 12 psig argon cover gas pressure, together with a vacuum on the storage tanks, will be utilized to transfer the sodium. It is planned to transfer the sodium at a temperature of 350 F - the operation will require approximately 24 hours.

The elastomeric Klozure seal on the rotating shield plug has been in service seven years and because of aging may not withstand the abnormally high cover gas pressure required to transfer the sodium out of the reactor vessel. A static backup seal has been designed. It consists of a rubber gasket around the outside diameter of the rotating plug, held in place by a heavy steel band. The gasket spans the gap between the plug and the top part of the vessel.

Design of a temporary drain line connection from the rotating shield plug to a tie-in point just below the operating floor was completed in April and fabrication was started. The line has been relocated to pass through the operating floor at a manhole penetration rather than through the pipe penetration originally intended for use during the drain operation. The unheated pipe penetration had been plugged with frozen sodium and reopened (see Page 36 of Report No. 4). However, after further plugging occurred it was decided to go through the manhole so that the pipe can be heated where it passes through the floor. This pipe is 2-inch carbon steel, insulated and induction heated.

G. Inerting and Cooling of the Storage Tank Room

Work was started in April on a recirculating cooling gas system which will be necessary when the primary sodium storage tank room is inerted with nitrogen. The atmosphere in the room is being converted to nitrogen to prevent burning in the event of a leak of the radioactive sodium which will be stored in two of the three 15,000 gallon tanks. The nitrogen cooling system will utilize an existing fan and distribution ductwork which presently supply outside air for cooling the room. The nitrogen will be passed through a water cooled coil to remove the heat developed in the room by the setting losses from the tanks. Approximately 80 gpm of lake water will be required to cool the nitrogen. Leaktight, welded ductwork will conduct the nitrogen to and from the cooling coil which will be located outside the storage tank room.

The inside dimensions of the room are 64 feet long, 35 feet wide and 19 feet high. The walls and ceiling are 30-inch thick concrete. The room has very few penetrations and should be reasonably leaktight; it is estimated that after the initial purge 2 scfm of nitrogen will maintain sufficiently high nitrogen purity. An oil filled U-tube will provide pressure or vacuum relief. Cooling of the room is necessary to prevent overheating of the concrete. The cooling system is designed to limit the maximum temperature inside the room to 150 F.

The cooling coil has been delivered. The base slab, outlet drain piping and water supply piping were installed in April. The ductwork is scheduled for early May delivery.

SECTION V

SPECIAL INVESTIGATIONS

A. Final Analysis Results on Subassembly M156

Page 22 of Report No. 7 and Page 17 of Report No. 8 give earlier analysis results on the inspection of subassembly M156 at the Battelle Memorial Institute hot lab. The final results listed below were obtained in April.

1. It was observed that there had been no significant changes in the lower nozzle spring rate or free length.

2. The inside surface of the subassembly wrapper can, just above the inlet nozzle, was analyzed for fission product plate-out by a radiochemical analysis. The surface was cleaned with a brush and water and then etched to several depths with the following results:

	Concentration - microcuries/sq in			
Fission Product	0.26 Mils Removed	Additional 0.39 Mils Removed	Additional 0.31 Mils Removed	Additional 0.34 Mils Removed
Sr-89 & 90	0.08	0.0009	ND	ND
Zr-95 & Nb-95	0.014	Below 0.01	Below 0.01	Below 0.01
Cs-137	0.003	0.003	0.003	0.003
Sb-124	Detectable	ND	ND	ND
Ru-103	Below 0.01	Below 0.01	Below 0.01	Below 0.01
Ce-141	0.005	0.005	0.005	0.005

Note: ND - not detected. These results were normalized for base metal neutron-induced radionucleides. The concentrations listed are only for the removal increment indicated.

B. Nondestructive Hydriding Test

Investigation was made of a technique using a low frequency magnetic coil to nondestructively test fuel pins for severe hydriding of the zirconium clad. Depleted uranium fuel pins which had previously been hydrided to various levels ranging from 250 ppm to approximately 5000 ppm hydrogen were tested with the new technique. The results appear promising for measuring and/or detecting localized areas of zirconium hydriding existing above 1000 ppm. At lower concentrations the technique is insensitive.

A check was made of 130 enriched fuel pins from the lightly irradiated sodium worth subassembly NA-2. Sixteen pins gave readings that indicated excessive hydriding at various locations along their length. Two of these pins were sent to the Battelle Memorial Institute hot lab for metallographic examination to determine if these pins actually do contain 1000 ppm of zirconium hydriding in the clad. The metallographic appearance did not confirm the nondestructive observation. There appeared to be only about 200 ppm hydrogen in the clad. The reason for this discrepancy is not presently known.

C. Reactor Vessel Flow Analysis

An investigation has been concluded to determine if coolant maldistribution due to 2-loop flow could have been responsible for the October 5, 1966, fuel damage. An analysis was made of each core and IRB subassembly temperature rise as measured by thermocouples at approximately one fourth of the core and IRB positions (see Section III E of Report No. 3 for further thermocouple details). The data were normalized to account for the reactor power level, the core and IRB coolant flow rate and the position of the subassembly. A large number of runs were analyzed at various power levels and flows from January to October 1966. Comparison was made between 3-loop operation and operation on Loops 1-2, 2-3 or 1-3. The normalized subassembly delta-T readings should reveal areas of coolant flow variations in the core. No significant flow patterns or trends were observed for any mode of operation.

The possibility was considered that one or more subassemblies had been grossly mis-seated in the lower support plate, permitting flow by-pass. It was concluded that if a subassembly lower nozzle somehow became misaligned so that it caught on the top of the support plate instead of seating properly, the coolant flow through the subassembly would be reduced by approximately 7 percent — an insufficient amount to cause fuel damage.

An analysis was made to determine the relative importance of the various parts of the primary system in affecting the coolant flow distribution in the reactor core. The flow resistances for the Fermi primary system are as follows:

Location	Flow Resistance, Delta-P (psi) 3 Loop Flow (8.86x10^6 lb/hr)	2 Loop Flow (Oct. 5 - 5.78x10^6 lb/hr)
Reactor Outlet to 14 & 6-inch pipe	3.0	2.9
14-inch piping	1.5	1.4
High pressure plenum	2.2	1.0
Core & axial blankets	46.0	20.3
Hold-down & upper plenum	3.0	1.3
Total system	55.7	26.9

The analysis shows that only the core region pressure drop has a significant effect on the total system resistance. Very large perturbations or unbalances would have to occur in the remaining parts of the primary system to give any significant flow maldistribution in the core region. Hence, flow maldistribution in the inlet plenum is considered to be highly unlikely.

D. Sodium Particulate Analysis

An analysis had been made in March of a one gallon sample of the Fermi primary system sodium (see Section III B of Report No. 8). In April a comparison was made between this sample and a sample of the sodium in the as-supplied condition as received from the vendor in 1960. With two exceptions, the same types of particulate matter were observed in both samples. The carbon content in the as-supplied sodium was twice as high as in the recent sample.

Neutron activation analysis at the General Atomics Laboratory of the residue obtained from filtering the gallon primary sodium sample showed the presence of aluminum, magnesium, sodium, mercury, zinc and cadmium. The mercury is thought to remain from the mercury amalgamation procedure used to separate the sodium.

SECTION VI

MAINTENANCE

A. No. 1 Steam Generator Tube-To-Tube Sheet Weld Repairs

By the end of April, 1150 of the 1200 water manifold tube-to-tube sheet joints had been rewelded in the manner described on Pages 23-25 of Report No. 7 and Pages 20-21 of Report No. 8. During April the rewelding proceeded on a two shift basis. The photographs on Page 28 show portions of the top of the water manifold tube sheet.

Rewelding of the tubes at the No. 1 steam generator water manifold tube sheet is scheduled to be completed in May. The welds will be leak tested, thermally shocked and leak tested a second time.

B. Instrumentation

1. Storage Tank Sodium Level Probes

The level detectors in the three 10-foot diameter sodium storage tanks were supplemented with the addition of a new type of sodium level probe system. Each of the tanks has had two float-type level detectors, one set to indicate low level and the other high level. The float-type detectors were not completely reliable — the level probes were added to improve the reliability before the primary system sodium is transferred to the storage tanks, as further described on Page 20.

Level detector probes were added to the tanks at 18, 6 and 4-inches from the top. Filling of the tank is terminated when the sodium level reaches the 6-inch probe level. The improved probe system is similar to the more conventional shorted type digital probe system utilizing a probe consisting of a single copper conductor in an Inconel sheath. Both systems utilize a continuously shorted circuit and when sodium contacts the probe, an electrical path through the sodium and tank wall is produced which is of less resistance than the initial short. This difference is measured by a sensitive relay and circuitry which energize an indicating light on the relay cabinet located in an adjoining room.

The improved system makes use of an ordinary stainless steel sheathed, mineral insulated, iron-constantan thermocouple which is mounted in the top of the tank with a Swagelok connector. A thermocouple is used because it serves as a readily available sheathed probe with two insulated conductors instead of one. The added conductor permits a four terminal approach in sensing the voltage drop across the sheath which results in a very large ratio (50:1) of output voltage before and after sodium contact. The resistance of the thermocouple wires and copper extension leads have virtually no effect on the sensitivity of the system. Further improvements were obtained by adding a step-up transformer and a solid state electronic relay for sensing the output signal from the probe. A diagram of the probe installation is shown on Page 29 and a schematic of the electronic relay may be seen on Page 30.

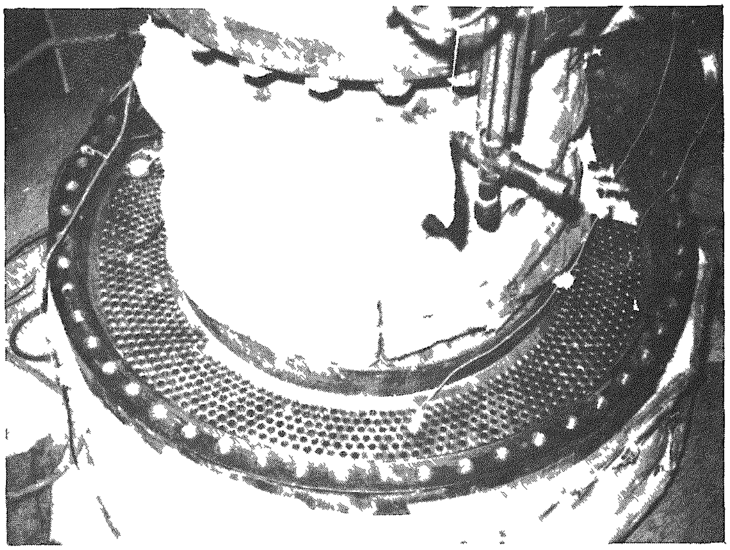
Storage Tank No. 1 contains a reserve supply of molten sodium designed for rapid transfer to the reactor vessel in case of a vessel leak into the primary shield tank. Because of this quantity of sodium in the No. 1 tank, the tank has a continuous level detector. This instrument is a Taylor differential pressure level detector utilizing two chemically sealed (Nak) pressure detectors and a differential force transducer to measure level and transmit a 0-20 mv d-c signal for a 0-100 percent level change. This continuous level detection system has operated satisfactorily.

2. Primary Shield Tank Pressure Control Failure

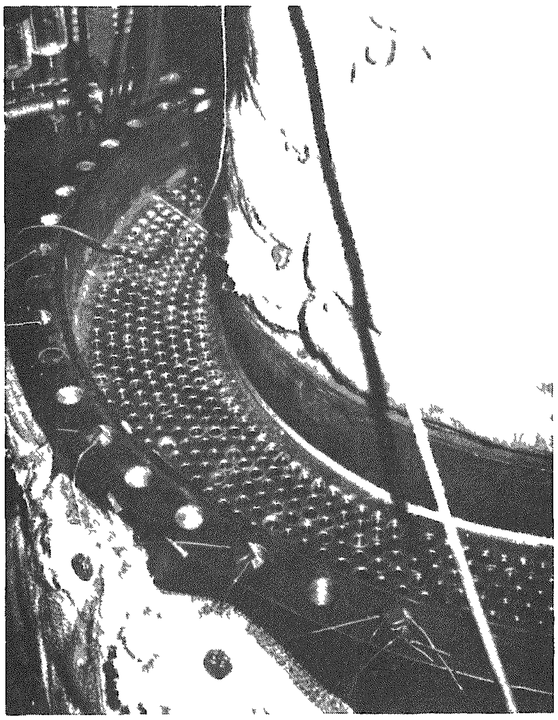
Failure of a solenoid valve on the primary shield tank (PST) pressure control system resulted in loss of control, accompanied by positive pressure swings. The backup pressure control system limited the pressure to about 1.3 psig. The nitrogen pressure in the PST is normally regulated to 8-inches of water above that of the lower reactor building.

The solenoid valve failed and blew a fuse in the electrical supply. This prevented switchover to the alternate sensing line because both the valve that failed, FCV 1370-3, and the alternate line valve FCV 1370-4 were fused in the same circuit. See Page 31 for a schematic diagram of the PST pressure control sensing lines.

The primary purpose of the switching valves is to permit use of the alternate sensing line in the event of a plug in either line. A new system of manually operated bellows-sealed valves will be installed to replace the solenoid valve group, since the sensing lines are now considered more reliable than the solenoid operated switching valves. The new valves will add only a slight delay in switching over to the alternate sensing line in the event of a failure. A gauge has been added at the PST control relay rack in the waste gas building to provide a readout in the backup system, in the event of a failure of the normal pressure control and display system.

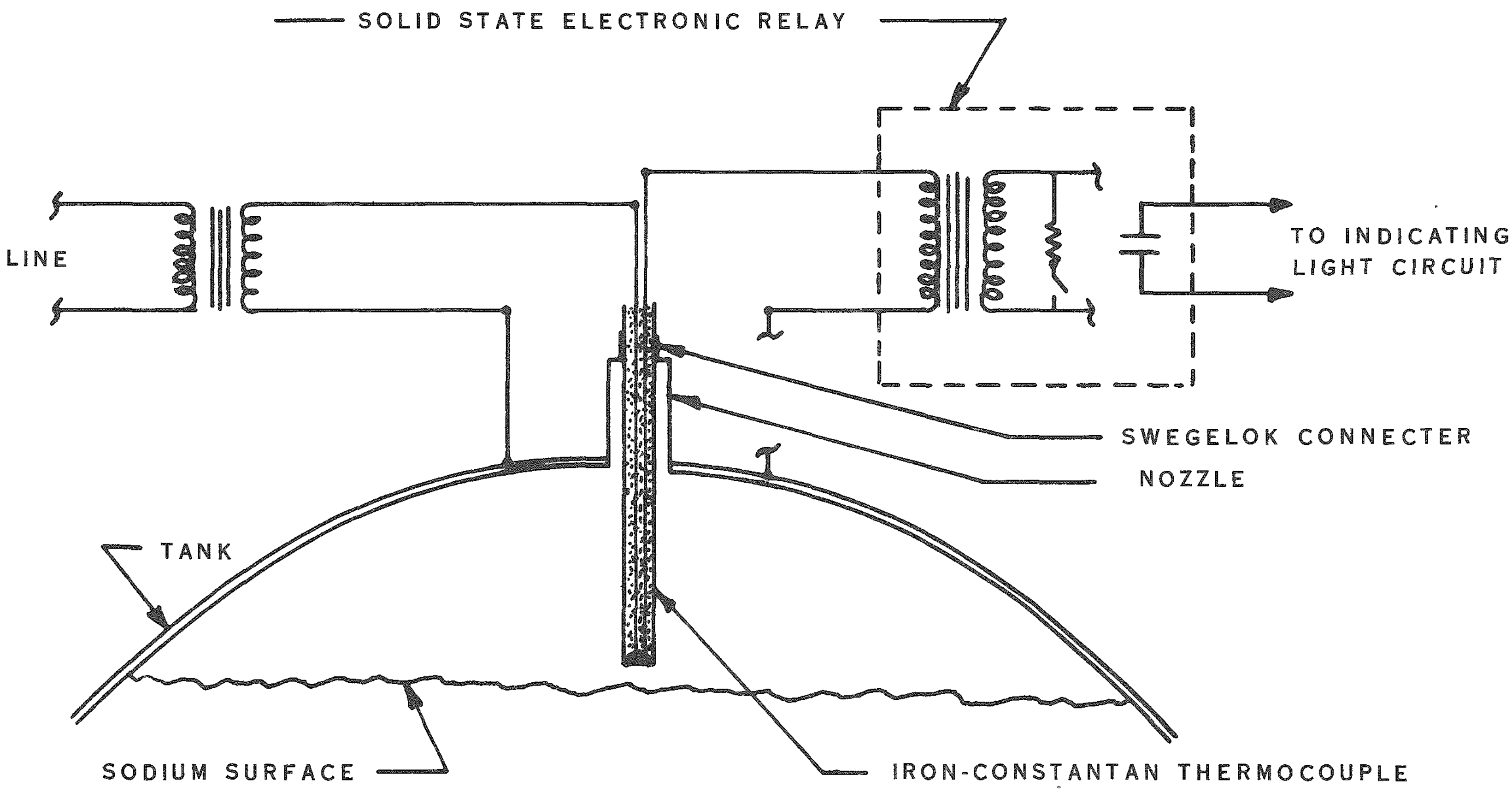


GENERAL VIEW OF THE WATER MANIFOLD
ON THE NO. 1 STEAM GENERATOR

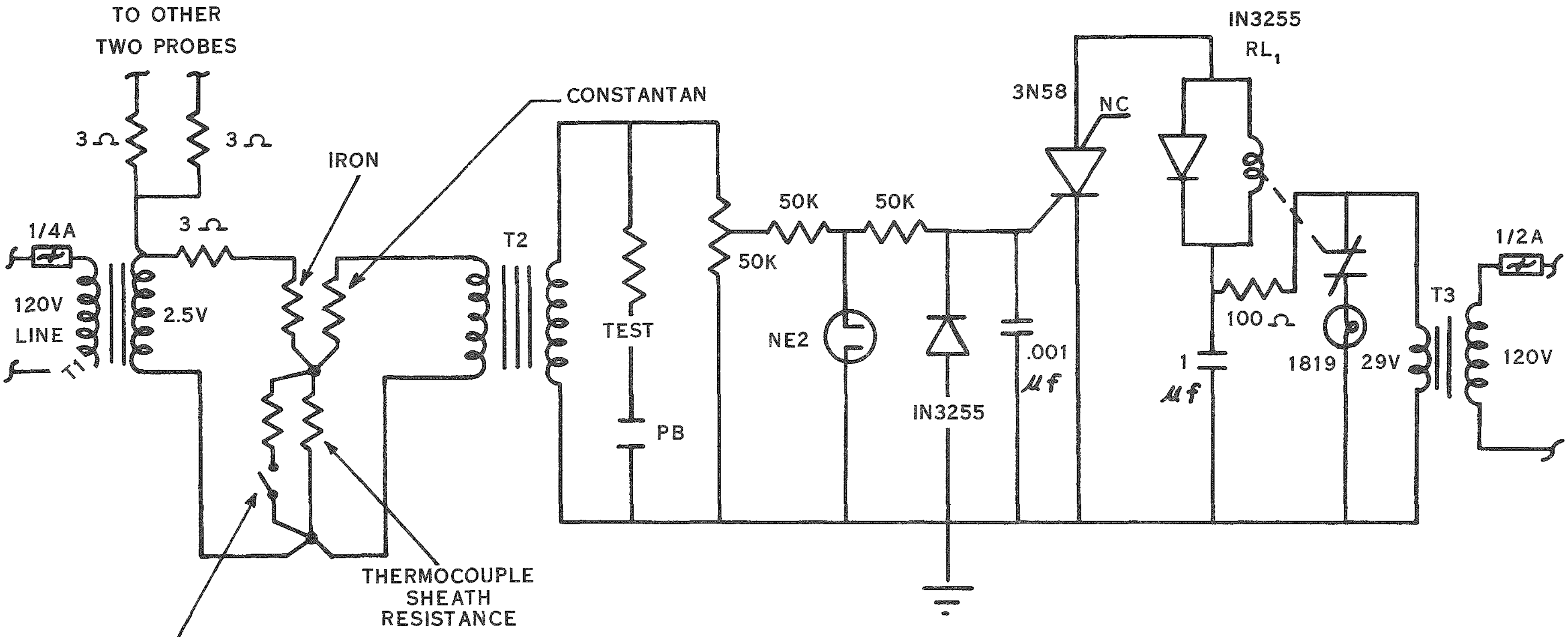


VIEW OF THE FACE
OF THE TUBE SHEET
AFTER THE REPAIR
WAS COMPLETED

STEAM GENERATOR TUBE - TO - TUBE SHEET WELDS

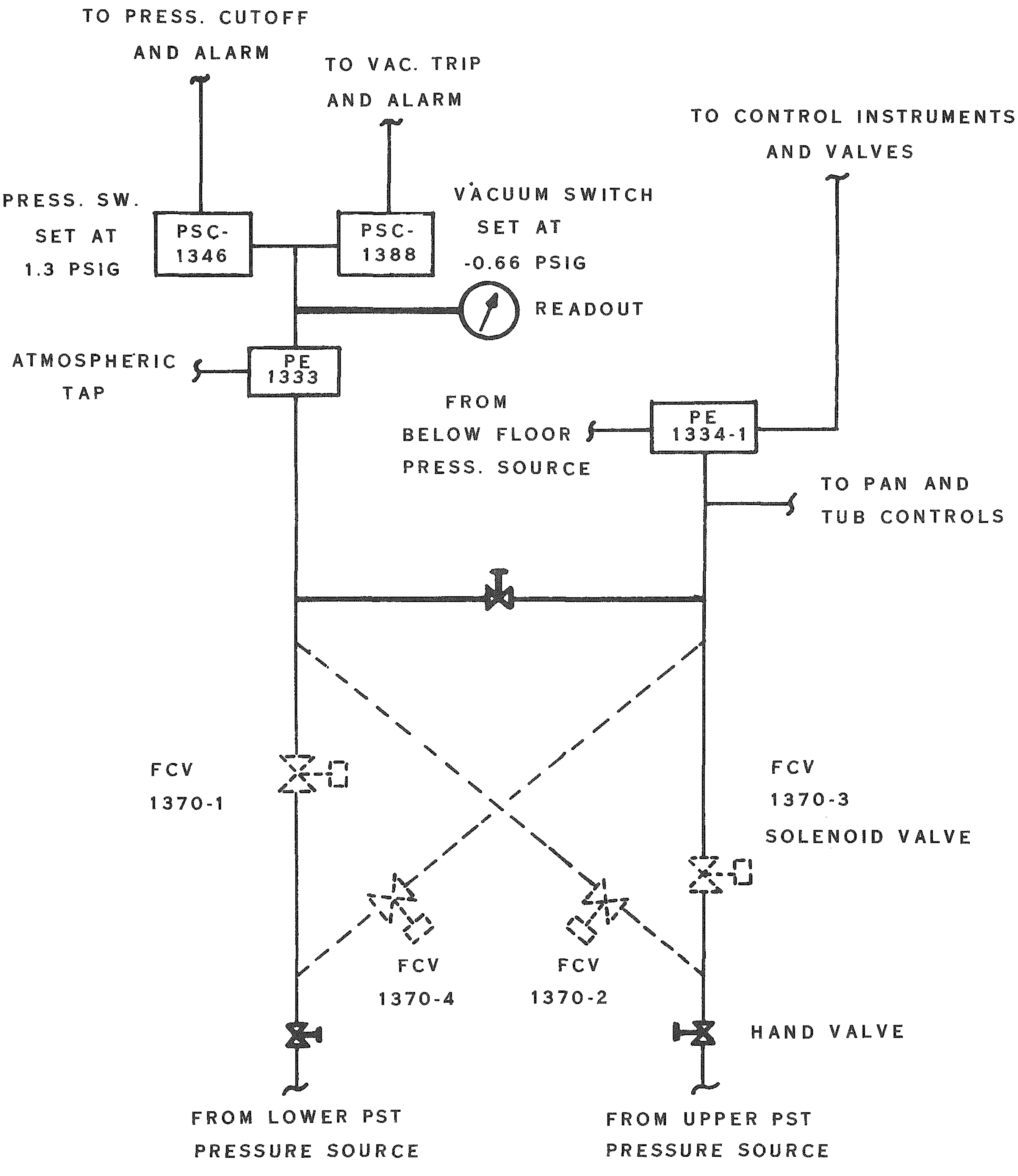


STORAGE TANK LEVEL PROBE



ITEM	DESCRIPTION
RL_1	SIGMA 42RO6. 1000 S - SIL
T_1	CHICAGO STANDARD (STANCOR) P6454. 117V TO 2.5V
T_2	CHICAGO STANDARD (STANCOR) A3327. 25K Ω TO 4 Ω
T_3	CHICAGO STANDARD (STANCOR) P8357. 25.2V TO 117V

LEVEL PROBE SYSTEM SCHEMATIC



CODE: DASHED-REMOVED: HEAVY-ADDED

PST PRESSURE CONTROL SYSTEM