MONTHLY REPORT NO. 7 - FEBRUARY 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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REPORT NO. 7 - FEBRUARY 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 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.

The items in Section II A & B 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

Subassembly M156 was removed from the reactor, steam cleaned, cut up and shipped to the Battelle Memorial Institute hot lab for examination, with particular interest in a determination of the extent of hydriding of the fuel pin zirconium cladding. Micrographic examination of the cladding had not been completed at the date of this report, but general examination has revealed nothing abnormal. Inspection of subassembly M099 at BMI has been completed. Except for two fuel pins with higher than expected zirconium hydriding, no significant abnormalities were noted.

Partial unloading of the reactor continued during February. A space will be cleared in the reactor to permit the removal of subassemblies M127 and M098 as a pair. Spaces to store the subassemblies which must be removed are being vacated by removing 12 dummy subassemblies from the periphery of the outer radial blanket and other accessory items from the reactor storage positions, from the reactor transfer rotor and from the Fuel and Repair Building transfer tank.

A final and unsuccessful attempt was made to separate stuck subassemblies M127 and M098, using the offset handling mechanism. The OHM was reconverted from manual to electric powered operation to expedite clearing the area in the reactor.

Work commenced to prepare steam generator No. 1 for rewelding of the water manifold tube-to-tube sheet joints by a new method. Preliminary inspection before the rewelding showed two leaking welds and several welds with flaws. A machine-rotated tungsten inert gas welding head will be inserted 3/4 inch into the tubes and will make a fusion weld through the wall of the tube penetrating into the adjacent tube sheet.

SECTION II

PLANT OPERATIONS

A. Reactor Unloading

The diagram on the following page shows the extent of the reactor unloading at the end of February 1967. Subassemblies M367, M145 and M113, were removed from core positions N05-P04, N05-P02 and N02-P04 respectively. M367 and M145 were diagonally adjacent to subassembly M127 and subassembly M113 had a corner adjoining subassembly M098. With the removal of M367, M145 and M113, the only item still adjacent to the M127-M098 bonded pair was the No. 1 safety rod lower guide tube which touches the northwest corner of M098. The guide tube will be left in place because it cannot be removed without the removal of M098.

Subassembly M156 was removed from core position P02-N01 and transferred to the Fuel and Repair Building. It was steam cleaned, sectioned in the cut-up pool and shipped to Battelle Memorial Institute where it was examined in the hot cell. See pages 12 and 22 for further details.

Three outer radial blanket (ORB) dummy subassemblies were removed from positions N14-N02, N03-N14 and P02-N14 in the outer row of the radial blanket of the reactor. The spaces they occupied will receive subassemblies from the area of the reactor which must be vacated to permit the removal of subassemblies M127 and M098. (See Section III C3 of report No. 6.) The three dummy subassemblies were placed temporarily in the transfer rotor; they will subsequently be moved to the Fuel and Repair Building. Fuel subassembly M158 was moved from the transfer rotor to reactor storage position N11-N09.

ORB subassembly S-512 was moved from position N06-P04 to the transfer tank. ORB subassembly S-550 was moved from position N09-P09 to position N13-P03. These moves were made as part of the above-mentioned program to open up an area to permit the removal of M127 and M098.

The chart on page 8 summarizes the changes in the reactor loading during February and the reasons for the changes.

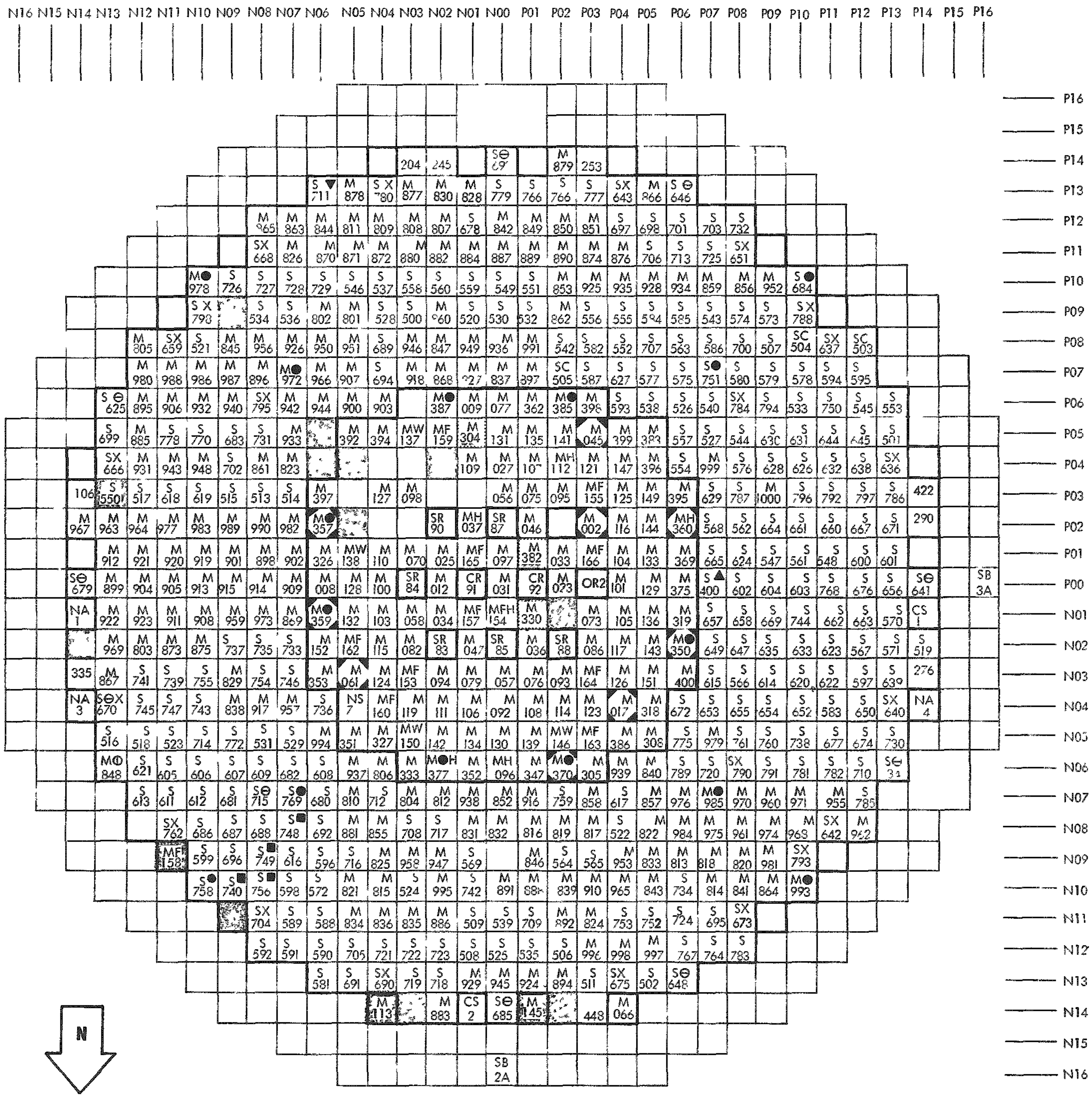
KEY:

- 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
- X 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



REACTOR LOADING 2-28-67

CHANGES IN REACTOR LOADING DURING FEBRUARY 1967

Date	Subassy No.	From Position	To Position	Reason for Move	Remarks
2-1-67	M367	N05-P04	TRC**	To clear area around M127-M098	-
2-1-67	M145	N05-P02	Storage*	To clear area around M127-M098	-
2-1-67	M113	N02-P04	Storage*	To clear area around M127-M098	Seemed to drop 1/2-inch during latching
2-1-67	M156	P02-N01	Cleaned, Shipped to BMI Hot lab	To examine in lab, especially for clad hydriding	Examination incomplete - No visual abnormalities noted.
2-1-67	M158	TRC**	N11-N09	Make space available in TRC	Will be stored at N11-N09
2-6-67	74 (dummy)	N14-N02	TRC**	Make space available for other subassemblies	Will be transferred to fuel and repair building and cleaned
2-6-67	468 (dummy)	N03-N14	TRC**	Make space available for other subassemblies	Will be transferred to fuel and repair building and cleaned
2-6-67	455 (dummy)	P02-N14	TRC**	Make space available for other subassemblies	Will be transferred to fuel and repair building and cleaned
2-9-67	S-512	N06-P04	Transfer Tank	To clear area to swing M127-M098 during removal	-
2-13-67	S-550	N09-P09	N13-P03	To clear area to swing M127-M098 during removal	100-lb extra seating force reqd
2-13-67	M-941	N06-P05	TRC**	To clear area to swing M127-M098 during removal	-

* Storage at outside of outer radial blanket ** Transfer rotor container

A 100-pound extra pushing force was required to complete seat subassembly S-550 in position N13-P03. This was because of hang-up as the upper portion of the support nozzle entered the upper support plate. See the diagram on page 10. Subassembly S-510 had previously exhibited an interference upon insertion in this position, as reported in Section II A of report No. 6. A 150-pound extra pushing force had failed to move S-510 the final one-inch of travel.

S-510 was transferred to the cask car from the transfer rotor container. It was let down out of the cask car into the glove box (in the Fuel and Repair Building) and examined by direct vision through a viewing window. At the upper shoulder on the support nozzle a small raised gall mark 1/16-inch wide by 1/8-inch high was observed.

The radiation level at the 3/16-inch thick glass viewport, as the subassembly moved past the viewport, was 1.5 R/hr maximum and 300 mr/hr average. The radiation level with the detector in contact with the subassembly handling head was 2.5 R/hr.

S-510 is at present stored in the FARB transfer tank. The galled area will be stoned smooth and the subassembly eventually returned to the reactor.

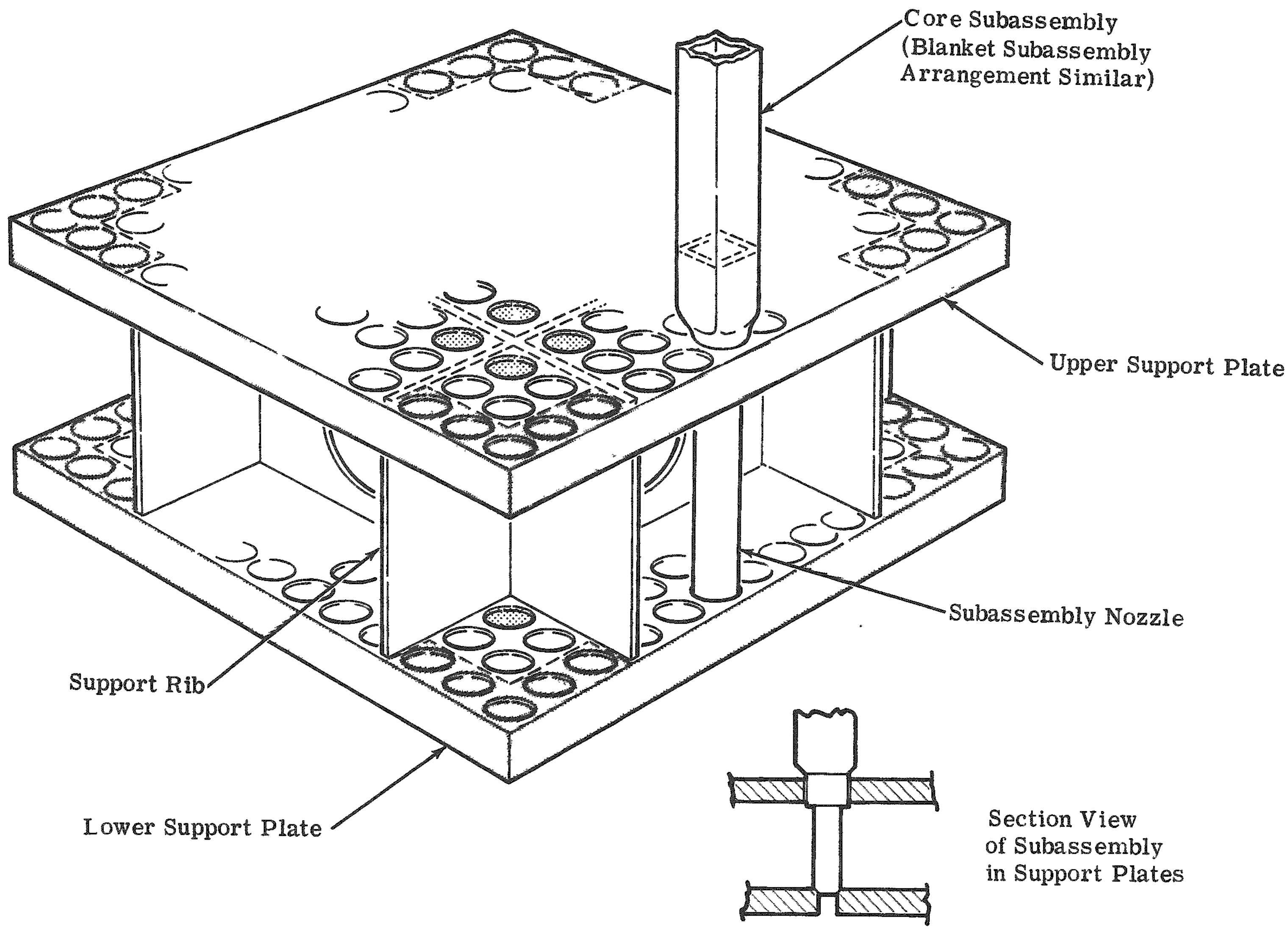
B. Sodium and Gas Systems Performance

1. Sodium Cold Traps and Plugging Indicators

February Operating Data	Primary Systems	Secondary System		
		Loop 1	Loop 2	Loop 3
Cold Trap Operation	81 hr	70 hr	119 hr	0 hr
Maximum Plugging Temperature	220 F	275 F	280 F	285 F
Minimum Plugging Temperature	220 F	220 F	220 F	235 F

2. Primary Cover Gas Activity

Location	Sample Date	Concentration
Reactor	2-17-67	1.24×10^{-5} Microcuries/cc



SUPPORT PLATE AND RIB STRUCTURE

3. Primary System Cover Gas Analysis

	Reactor Cover Gas (Argon) ppm by Volume	Primary Shield Tank Atmosphere (Nitrogen) ppm by Volume
Dew Point	Not Measured	Below -50 F
Oxygen	Below 25	25*
Carbon Monoxide	Below 10	Below 10
Carbon Dioxide	Below 10	Be 25
Hydrogen	4 **	5
Helium	Below 4	Below 4
Methane	Below 10	Below 10
N_2O	Not Measured	Below 10
NO_2	Not Measured	Below 1
Argon	Balance	Not Measured
Nitrogen	1650	Balance
Sample Date	2-17-67	2-17-67

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

4. Primary Sodium Chemical Analysis

Date Sample Taken	2-17-67
Oxygen	5,5,6,6
*** Hydroxide Hydrogen	1.1, 0.9
*** Nonhydroxide Hydrogen	0.2, 0.2
Nickel	Below 0.1
Iron	0.4
Chromium	Below 0.2
Carbon	153,60,66,98,62,102,82

*** 1.3 ppm recommended maximum for total hydrogen

NOTE: Values are in ppm by weight. One sodium gallon sample was analyzed at several different points to provide the separate readings indicated.

5. Secondary Sodium Chemical Analysis

	Loop 1	Loop 2	Loop 3
Oxygen -as Na_2O	8	18	27
-as NaOH	16	36	54
Hydroxide Hydrogen	1.2	2.0	3.2
Nonhydroxide Hydrogen	0.2	0.3	0.4

Values are in ppm by weight.

6. Primary Sodium Impurities

The curve sheet on page 14 shows the variation of several impurities (carbon, oxygen, hydrogen and iron) in the primary sodium during the period from May 1965, through August 1966. Also shown are the periods of cold trap operation and of reactor operation and the intervals when the sodium temperature was above the normal shutdown temperature of approximately 500 F.

C. Cleaning, Cutting and Shipping M156

Based on the results of the inspection of subassembly M099 for hydriding of the fuel pin cladding, subassembly M156 was selected for removal from the reactor for shipment to the Battelle Memorial Institute (BMI) hot laboratory for a similar inspection, and other tests. See Section IV for further details on the inspection of M099 and M156. M099 had been removed from the reactor in early July prior to the 100 Mwt operation, while M156 had been in the reactor since the initial loading.

Subassembly M156 was selected because it contains fuel pins fabricated from a variety of uranium alloy billets and heat treatment lots. These pins are representative of 19 furnace lots which are also identified with other subassemblies of special interest (M127, NA2, M097, M165, M122 and M140).

Subassembly M156 was steam cleaned and placed in a leaker can in the cut-up pool. A water sample from the leaker can taken about 40 hours after the insertion of the subassembly had an activity of 4×10^{-2} microcuries/cc. Another sample taken 24 hours later had very close to the same activity. The leaker can samples were 30 times higher in activity than those obtained from M099; a factor of ten maximum increase had been expected.

The subassembly was subsequently removed from the leaker can and examined externally with underwater TV cameras. No abnormalities were observed. The radiation level was measured while the subassembly was in the pool. At a point alongside the subassembly 6 inches above the top of the core and attenuated by 3.5 inches of water and 0.3 inch of steel, the detector reading was 100 R/hr.

M156 was cut into three segments using the underwater cut-up machine in the FARB. The segments were loaded into the shipping cask underwater; the cask was removed from the cut-up pool and loaded onto a truck and transported to the hot cell at the BMI laboratory.

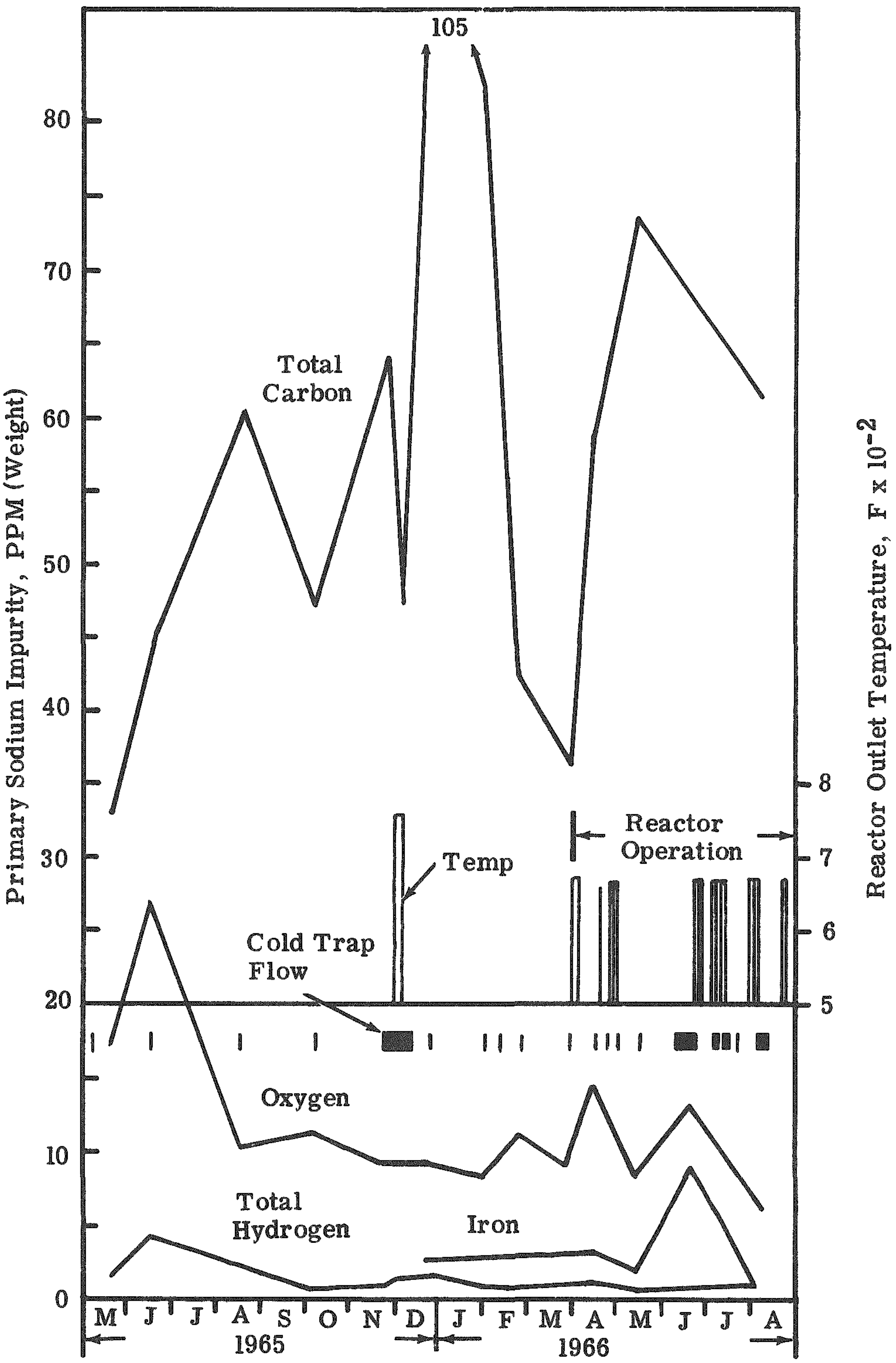
The cleaning, cutting and shipping of M156 followed closely the pattern established for subassembly M099, which was described in Section II B of report No. 5. There were no difficulties with either subassembly.

D. Radiation Level of Stool Inserts

Special stool inserts had been installed in drainable finned pots in the reactor transfer rotor. These inserts were in the transfer rotor at the time that the October 5th fuel damage occurred. The stools are thin-walled stainless steel tubes three inches in diameter and three feet long. They are designed to accept the offset handling mechanism calibration tool.

The stools were removed to make the finned pots available for handling the subassemblies which will be removed from the reactor vessel. The removal took place at the glove box in the FARB.

The radiation level of the stools one inch from the surface was 150 mr/hr, due chiefly to fission product plate-out. The high activity of the stools was somewhat unexpected since the most recent sodium sample coil had an activity of only 3 mr/hr. Sodium on the walls of the stools was analyzed and it was found that the gross sodium activity was twice that measured from the sample coil.



PRIMARY SODIUM IMPURITIES

SECTION III

PLANT REPAIRS - CORE INVESTIGATION

A. Relift Survey of Subassemblies

Twenty-one core and inner radial blanket subassemblies were rechecked by lifting 3/4-inches and measuring the load as shown on the diagram on the following page. The survey was made to re-evaluate some previously measured high lifting forces. All positions checked had normal lifting forces, but the initial latch elevation of subassembly M137 (at N03-P05) was 1/2-inch lower than normal. A second attempt resulted in latching at the normal elevation. The reason for this occurrence is not known at this time. This also occurred with S/A M118 at N04-P02, as reported on page 7 of report No. 6.

B. Final Attempt to Free M127 and M098 Using the OHM

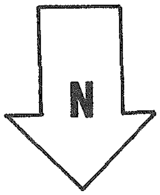
The offset handling mechanism (OHM) had been used several times in December and January in an effort to separate the bond between subassemblies M127 and M098, as described in Section III F of report No. 5. A 2800-pound lifting force was alternately applied to each subassembly after the last adjoining subassemblies had been removed as described on page 6. This pulling action attempted to strip off the other member of the pair at the OHM stabilizer foot; it was unsuccessful. No further attempts at separation are planned.

About 400 to 600 pounds force was required to lift the pair until the subassembly whose head was not engaged by the OHM gripper struck the foot. It was thought by some that there was indication of perhaps 0.1-inch relative motion between the pair. On the last OHM gripper engagement, there was a sudden drop of about 1/2-inch. Subassembly M113, transferred eariler in February, exhibited a similar unexplained 1/2-inch drop.

C. Removal of M140 from Regular Transfer Pot

When subassembly M140 was removed from the core in December, an abnormally high force was required to insert it into a regular transfer pot in the transfer rotor. This high force, estimated to be in the order of 800-pounds, was required to push the subassembly the final 17-inches into the pot. There is nominally a 1/8-inch clearance across the diagonals between a subassembly and the inside diameter of the pot. M140 is probably distorted; it had a high outlet temperature on and before October 5th.

	NO6	NO5	NO4	NO3	NO2	NO1	POO	PO1	PO2	PO3	PO4	PO5	PO6
PO6				Out									
PO5				103 (130)									
PO4		Out	Out	Out	Out		104 (126)		106 (134)				
PO3		Out			Out	Out							
PO2		Out	Out	Out	SR 1		2	102 (126)	Out				
PO1													
POO				4		R		S		O R			
NO1	132 (137)	105 (118)							Out				
NO2	96 (118)	107 (122)	105 (114)		6		7		8				
NO3	127 (137)	100 (126)	106 (114)	107 (118)									
NO4		Neut Source	96 (126)	103 (118)									
NO5		129 (145)	130 (141)	106 (118)	108 (118)								
NO6				126 (137)	132 (134)								



( ) Lifting Force Required In Mid-January Survey

Inner Radial Blanket Sub Ass'y.

Values Are Net Pounds - Corrected for Buoyancy

RECHECK OF LIFTING FORCES FOR 21 SUBASSEMBIES FEBRUARY 13, 1967

It had been anticipated that some difficulty might be experienced when attempting to withdraw M140 from the pot. On February 9th the subassembly was successfully moved to an oversized drainable pot at another position in the transfer rotor. The OHM engaged the subassembly handling head and raised M140 and the pot several inches until the pot struck the OHM stabilizer foot. Additional lifting force caused M140 to pull out of the pot. The lifting forces and travel were as follows:

Distance Lifted (inches)	Force Required (pounds)
1.0	230
1.9	560
2.8	785
4.1	850
6.0	860
8.8	865
9.4	570
16	205
17.5	Pot dropped off - onto transfer rotor

As a check, another subassembly was inserted in the transfer pot from which M140 had been removed. It went in easily, demonstrating that the previous binding was due to a distortion of subassembly M140 and not due to a distorted pot.

D. Reconversion of the OHM to Motorized Operation

At the end of February all sensitive operations had been completed which involved use of the OHM for locating, lifting or transferring reactor subassemblies. The drive mechanism had been manually powered during these operations; it was converted back to the electric motorized mode to facilitate removal of the 22 subassemblies which will be taken out to clear a path for the extraction of M127 and M098. The overload settings were adjusted lower than they had been previously.

The OHM proved to be a versatile gauge in helping to determine the extent of the October 5th reactor damage. The OHM and rotating plug mechanisms have reliably performed many hundreds of operations in the last four months.

The hollow, pressurized OHM crane tube was found to be leaking 4 cfh of argon gas. The leak was traced to a chip underneath a metal o-ring which provides the seal where the latch rod bellows is attached to the crane tube. A silastic rubber O-ring was installed and the leakage was reduced to zero.

SECTION IV

OTHER INVESTIGATIONS

A. Inspection of Subassembly M099

Examination of subassembly M099 at the Battelle Memorial Institute (BMI) hot lab was concluded in February. See Section IV A of report No. 5 and 6 for previous inspection results.

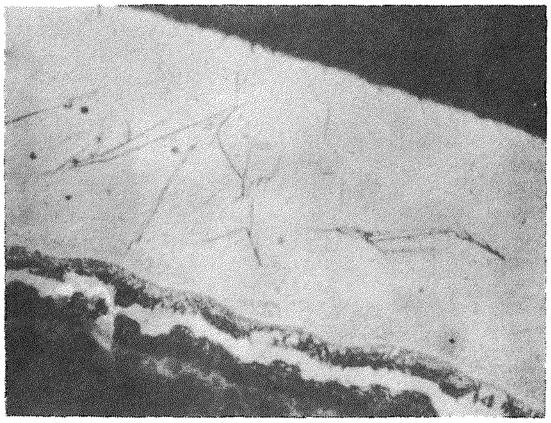
Fuel pin No. 18692 had been metallographically examined in January and the zirconium cladding appeared to have picked up about 2500 ppm of hydrogen. Sections adjacent to this specimen were removed, the cladding was heated to 1830 C and vacuum fusion analysis gave the following results:

Sample No.	H_2	N_2	CO	CO_2
1	700	23	below 7	nil
2	900	34	below 7	70

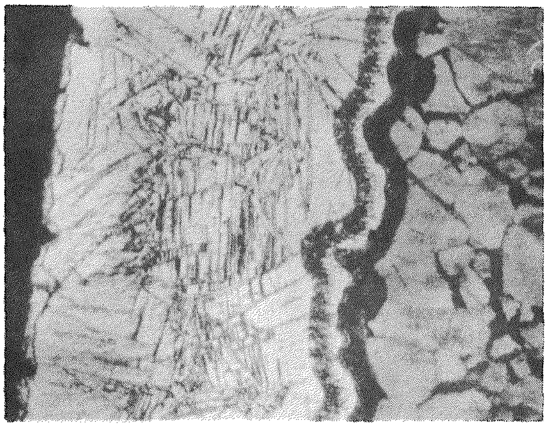
NOTE: Values are in ppm by weight

Three additional fuel pins were removed from the bundle and given a metallographic examination. These pins showed essentially no increase of hydrogen in the cladding. Thus, of the total of seven pins examined, two had hydrogen contents of approximately 2500 ppm, one had approximately 450 ppm and four had hydrogen content in the range of 150 to 200 ppm. All of these numbers are based on visual examination of photomicrographs. Two of the photomicrographs are shown on page 20. For comparison, two additional photomicrographs are shown on page 21 which were made before and after a depleted uranium pin had been deliberately hydrided in a hydrogen atmosphere in the laboratory. Again, the numbers given are based on visual examination rather than chemical analysis.

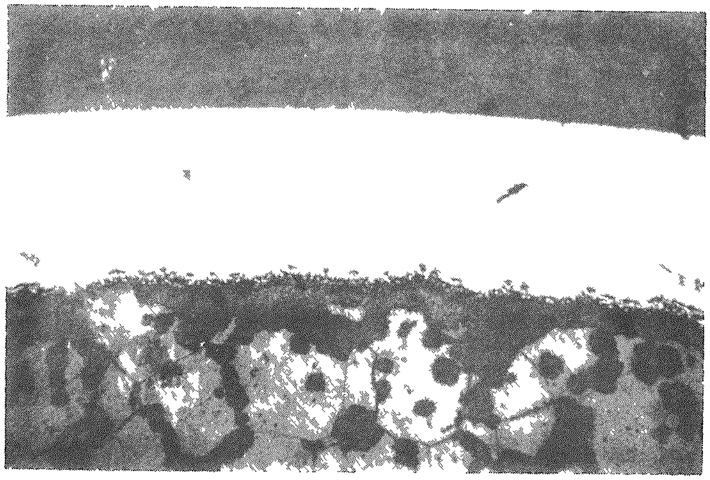
An examination has been made of the fabrication quality control metallographic samples from fuel pins contained in six core subassemblies. It was estimated that the as-built hydrogen content in the zirconium cladding of these pins is 70-200 ppm.



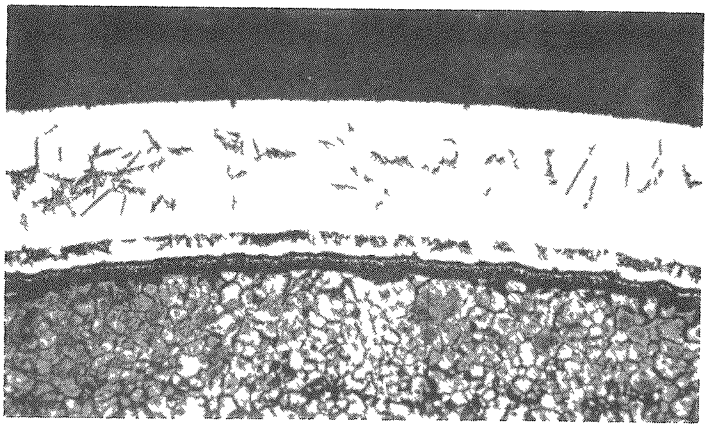
Microstructure of fuel pin No. 18699 removed from subassembly M099. Zirconium clad estimated to contain approximately 150 ppm hydrogen.



Microstructure of fuel pin No. 18692 removed from subassembly M099. Zirconium clad shows a metallographically estimated 2500 ppm hydriding, vacuum fusion analysis shows 700-900 ppm hydriding.



Clad Contains Approximately 70 ppm Hydrogen



Clad Contains Approximately 10,000 ppm Hydrogen
(50 percent Hydrided)

ZIRCONIUM HYDRIDING LABORATORY TESTS

Hardness measurements were made at 2-inch intervals along the inside surface of one of the sides of the wrapper tube. The calculated neutron exposure for M099 was 5.3×10^{20} nvt (total), 3.7×10^{20} nvt (E > 0.1 Mev) and 1.1×10^{20} nvt (E > 1 Mev). With this exposure it was estimated that the hardness increase would be about 10 Rockwell B numbers, the actual increase was only about one-half this amount. The maximum hardness was 88 Rockwell B at seven inches below the core centerline.

It was thought that the small black deposit removed from the lower nozzle sleeve was a compound of stainless steel. The size of the particles in the deposit ranged from 0.03 to 0.4 mils. The upper value is considerably smaller than the sub-assembly inlet nozzle strainer hole size of 40 mils.

B. Inspection of Subassembly M156

As described on page 12, M156 was removed from the reactor and sent to the BMI hot lab for examination. Approximately two-thirds of the scheduled inspections had been completed at the date of this report and had not revealed anything abnormal. The axial blankets and inlet strainer were examined with a borescope and appeared clean and in excellent condition. Two sides of the core wrapper can were removed and fuel pin spacing was checked and found to be normal. Fuel pin length and diameter were also checked but as yet have not been analyzed.

The March report will provide the results of the remaining inspections. It is planned to cut and mount all 140 fuel pins into one metallographic specimen and examine both ends of this specimen for hydriding of the zirconium cladding.

SECTION V

MAINTENANCE

A. No. 1 Steam Generator Tube Sheet Weld Repairs

1. General

During February work commenced in preparation for field rewelding of the No. 1 steam generator tube-to-tube sheet joints by a new method. Numerous leaks had developed in these joints during operation in 1966 and several repair techniques had been employed, as described on page 13A of report No. 1. These repairs involved welding or brazing between the top of the tubes and the face of the tube sheet, or slightly down from the face. The repairs did not assure against recurrence of the leaks.

2. Procedure

Tube-to-tube sheet leaks have developed in the water manifolds of all three steam generators. However, the new welding method will be applied first to only the tubes in the water manifold of the No. 1 steam generator. The unit will then be thermally shocked by rapidly venting the water from the tubes, followed by leak testing and examination. If the results are satisfactory the joints on the No. 1 steam generator, steam manifold will also be rewelded. The actual weld time is 45 seconds per tube and there are 1200 tubes in each steam generator.

A newly-developed welding method will be tried. The welding machine will be rented on a weekly basis and a royalty paid for each weld. If repairs are made to steam generators No. 2 and 3 a new machine will be purchased.

3. Preparations to Date

The No. 1 secondary sodium loop was drained of sodium, the water manifold split cover was removed and the tube sheet and tubes were flooded with water. Nitrogen gas at 35 psig was applied to the shell side and two leaking tube welds were detected. The tube sheet was then oxide cleaned and a dye penetrant test showed seven other tube end welds that may be defective although they are not leaking. Four tube weld ends (including one of the two leakers) were removed from the tube sheet by trepanning and were sent to the Detroit Edison Engineering Research Laboratory for metallographic study to determine the nature of the weld failures. At the time of this report the results had not been reported. The welding machine will arrive in early March and the trial period will be four to six weeks. Welders will be trained and qualified on the job.

4. Description of Method

The repair method employs a machine-rotated tungsten inert gas welding head which is inserted into the steam generator tubes a distance of 3/4 inch. A fusion weld is made through the wall of the tube penetrating into the wall of the hole in the tube sheet. The electrode must be positioned within 5 mils accuracy, requiring counter boring of the tubes with a light cut about 1/16-inch deep to remove previous welding at the top of the tubes. The tubes will be tightly rolled into the tube sheet and wire brushed before the welding operation.

A recording ammeter on the welding machine will assist in maintaining quality control. The welds will be inspected by means of a borescope.

5. Shop Testing

Extensive shop testing has been performed using sample tubes and tube sheets. Successful welds were obtained even when mill scale and sodium hydroxide were deliberately introduced into the crevice between the tube and tube sheet.

B. Steam Cleaning Machine Valve

As reported in Section V A of report No. 6, the 10-inch shutter valve at the bottom of the steam cleaning machine was repaired during January. The valve had seized in the partially open position due in part to galling of the Type 304 stainless steel stem in its bushing of the same material. The parts were replaced with 440-C stainless steel parts hardened on the job, but the stem and bushing galled again after having been in service only a short time. Investigation showed that the parts had a hardness of about 35 Rockwell C; they were refinished, hardened at a heat treatment shop to 52 Rockwell C and reinstalled. No further difficulty was experienced as three additional dummy core filter subassemblies were cleaned. A total of 27 subassemblies have been steam cleaned as of the date of this report.

C. Inerting of Transfer Tank Room and Shroud

The transfer tank is located below the operating floor in the Fuel and Repair Building, as shown in the diagrams of Section V C of report No. 5. The tank has a cooling jacket for removal of decay heat from spent subassemblies stored in the tank.

Ductwork carries the cooling medium outside the tank room, to a water-cooled heat exchanger and fan, and back to the tank. Until recently, air has been the cooling medium in this recirculating system. During February the system was connected to the plant nitrogen supply and inerted with nitrogen controlled to a positive 1-inch of water pressure at the tank jacket. The nitrogen will prevent a sodium fire in the event of a leak from the tank into the jacket.

The tank room is cooled by ventilation air which is drawn from above the operating floor and discharged to the waste gas stack. If the tank should develop a major leak into the room, the pressure rise caused by the resulting sodium fire would initiate isolation of the normal room ventilation, opening of a relief line to the waste gas stack and rapid release of a standby bottled nitrogen supply to smother the fire. The standby system is being made operational and a lower capacity, manually controlled nitrogen supply from the plant header is being added to blanket a fire resulting from a smaller leak.

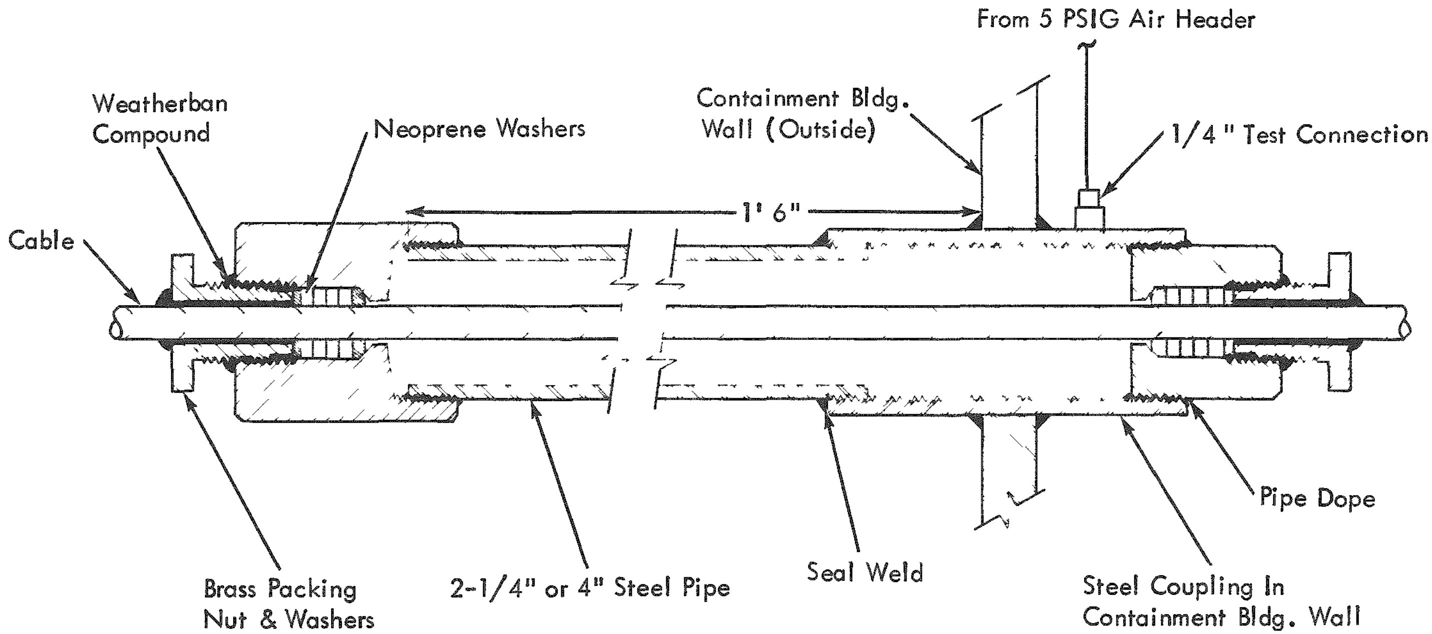
D. Reactor Building Penetration Leakage

The Reactor Building electrical penetrations are grouped in the south-southwest sector of the building, above the operating floor. The cluster of penetration nozzles is approximately 35-feet long and 15-feet high with an average spacing of about 10-inches on centers, for a total of around 600 penetrations. Approximately 350 are in use; the remainder have been weld capped. Those in use tend to be in clusters rather than scattered.

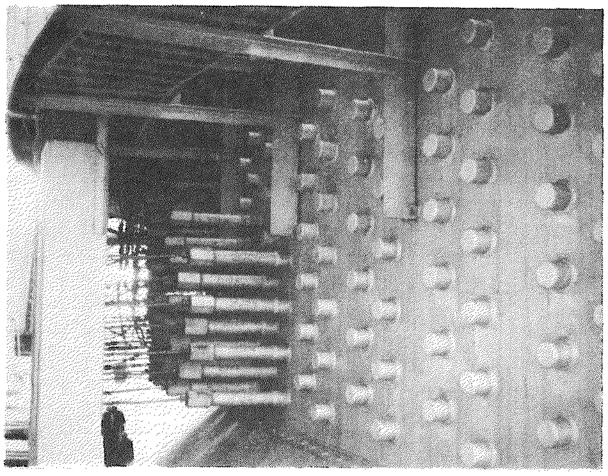
Details of the penetration are shown on the drawing on page 26. The cables vary in size from 1/2-inch to 2-inches outside diameter. The cables carry circuits at 4800 V, 480 V, 120 V and instrumentation voltages ranging down to less than one volt. Air leakage out of the reactor building through the cables is prevented by using gastight cables, or in the case of lead covered cables, by potting the ends.

Over the past year or two the cable seal original neoprene packing rings have been assisted by application of a back-up mastic (trade name Weatherban), applied with a caulking gun, which remains resilient and has given improved reliability. However, during extremely cold weather the outer seal appears to lose some of its resiliency and occasional leaks have appeared. These seals were recaulked. A header system supplies 5 psi air pressure continuously to each penetration. A flowmeter in the header is checked daily for indications of increased leakage. Each month the penetrations are tested at 32 psig; the average leakage is approximately 70 CFD.

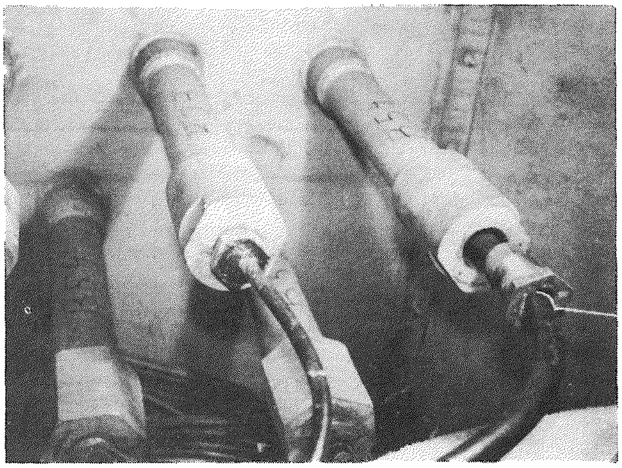
Photographs of the Reactor Building electrical penetrations are shown on page 27.



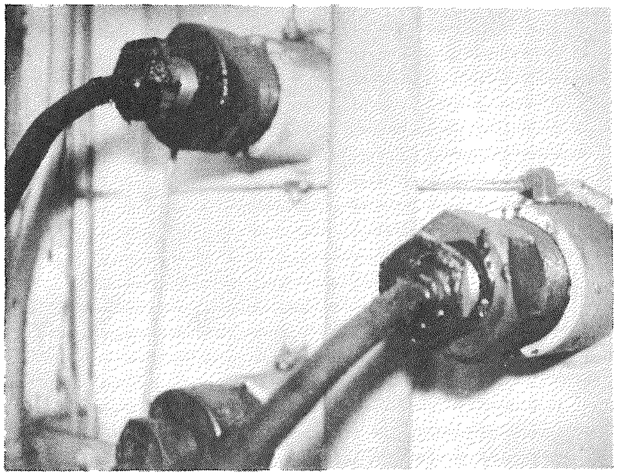
CABLE PENETRATION



EXTERIOR VIEW OF A PORTION OF THE REACTOR BUILDING ELECTRICAL PENETRATIONS. THOSE ON THE RIGHT ARE SPARES.



ELECTRICAL PENETRATIONS EXTENDING OUT FROM THE REACTOR BUILDING. PACKING NUT BEING INSTALLED IN THE PENETRATION ON THE RIGHT.



VIEW OF THE PENETRATIONS FROM INSIDE THE BUILDING. NOTE TUBING CONNECTION SUPPLYING 5 PSI AIR FOR CONTINUOUS LEAK MONITORING.