ENRICO FERMİ ATOMIC POWER PLANT**CURRENT EXPERIENCE SERIES**~~CERTIFIED PRICES~~

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RELEASED FOR ANNOUNCEMENT**IN NUCLEAR SCIENCE ABSTRACTS****COMPILATION OF CURRENT TECHNICAL
EXPERIENCE AT
ENRICO FERMİ ATOMIC POWER PLANT**

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Atomic Power Development Associates, Inc.

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MONTHLY REPORT No. 2 - SEPTEMBER 1966

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

Prepared by
> ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
DETROIT, MICHIGAN
with the cooperation of
> POWER REACTOR DEVELOPMENT COMPANY
and
> THE DETROIT EDISON COMPANY

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PREFACE

PURPOSE

The purpose of this monthly report is to make available to the fast reactor program the current experience being gained from the Enrico Fermi Atomic Power Plant.

SCOPE

The scope of this report includes all phases of current 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 the Atomic Power Development Associates Inc. for the US Atomic Energy Commission under AEC Contract No. AT (11-1) - 865, Project Agreement No. 15. This series of reports provides detailed information on the nuclear testing, rod drives, machinery dome, fuel subassemblies, fuel handling, steam generators, pumps, flowmeters, level detectors, sodium sampling and development of the primary sodium system.

The items 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.

The scope of the first several monthly reports will not be limited to items occurring since August 1, but will also include several earlier items of experience dating back as far as April 1, 1966.

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.

NOTE

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

There was only one brief period of reactor operation at power above 1 Mwt during the month of September because of safety rod drive difficulties and unexpected minor steam generator problems. These have been resolved as reported in Sections IIB3 and IIB6.

The reactor was operated twice at low power for oscillator tests and for a base reactivity measurement. A low power scram trace was also obtained.

Analysis continued on the temperature difference between the upper reactor vessel wall and the top of the transfer rotor container. Preliminary results obtained from Franklin Institute indicate that the wall stresses, resulting from this temperature difference, will not be excessive with a sodium temperature rise through the reactor core of 125 F. It is expected that further study will permit a much higher temperature rise through the reactor core.

The reactor rotating shield plug showed some evidence of interference after a two-month period during which no plug rotation took place. It is suspected that condensed sodium deposits were the cause and that these were worn away by the action of the plug. With continued rotation of the plug its drive motor current returned to normal.

SECTION II
DATA AND STATISTICS

A. Plant Operations

1. Reactor Operating Statistics

	<u>September</u>	<u>Cumulative through September</u>
Reactor Start-ups	11	405
Scheduled Scrams (Includes usual test scram prior to reactor start-up)	18	717
Unscheduled Scrams	3**	55
Maximum Reactor Power, Mwt	26*	114*
Integrated Reactor Power, Mwt - days	1*	854*
Hours of Reactor Operation	12	2079
Maximum Burnup in Central Core Subassembly, a/o	-	0.06
Maximum Electrical Output, Gross Mwe	-	22
Integrated Electrical Output, Gross Kwhr	-	1,160,000

* Indicated values - actual values are 10 percent lower, based on heat balance data. Further data to be taken and analyzed before final correction values are established.

** No. 2 Safety rod delatched. Poor holding magnet contact. See Page 19.

2. Daily Operation Curve Sheets

See the following two fold-out pages. The reactor produced no steam during September and reactor power above 1 Mwt occurred only briefly on one day, when there was a scram from 26 Mwt during an approach to a planned 67 Mwt operation, see Page 19. Therefore, steam and power curves are not included this month.

The primary sodium flows are indicated values - actual values are approximately 10 percent lower based on heat balance data. Further data will be taken and analyzed before final correction values are established.

FERMI DAILY REACTOR PLANT OPERATION DURING THE MONTH OF SEPT. 1966 SHEET NUMBER ONE

FERMI DAILY REACTOR PLANT OPERATION DURING THE MONTH OF SEPT 1966

SHEET NUMBER TWO

3. Reactor Loading

The following is a brief summary of recent changes in the reactor loading:

a. May 1966 - Removed two subassemblies (M097 and M154) near the core center line and interchanged them with inner radial blanket subassemblies M304 and M330 from reactor storage positions, to reduce core reactivity during an extended reactor shutdown for steam generator repairs.

b. June 1966 - Replaced the two core subassemblies noted above and resumed reactor operation at power.

c. July 1966 - Before commencing 100 Mwt reactor operation, one subassembly, M099, was removed from core position N01-P01 and a new subassembly inserted in its place. The M099 subassembly was placed at a storage position in the reactor for possible future examination.

d. September 1966 -

(1) Safety rods No. 83 and 89, from positions No. 6 and 3 respectively, were ~~inter~~changed to evaluate indications of interference (see Section II B 3). Four adjacent fuel subassemblies were removed from around each rod to permit access to the safety rods and then were replaced in their former locations after the rods were interchanged.

(2) Safety rods No. 83 and 89 were returned to their original positions, the adjacent subassemblies again being removed as described above in (1).

(3) During recent operation at 100 Mwt core subassemblies M091 and M140 showed higher outlet coolant temperatures than the calculated values, while the MF158 temperature was somewhat lower than calculated. Analysis of the outlet temperatures is in process and the results will be presented in the October report. Subassemblies M091, M140 and MF158 were located at positions N03-P05, N05-N02 and P01-P01 respectively. The temperatures are measured by thermocouples in the holddown fingers which rest on the subassembly handling heads. Approximately one fourth of the core subassemblies are instrumented with thermocouples. The differences could not be explained by the core power distribution which has been obtained by foil irradiation measurements. To determine whether the subassemblies themselves, their position in the core or the thermocouples were responsible for the differences, they were interchanged with other subassemblies located at symmetrical, or nearly symmetrical core positions, MW137, MF162 and M066 respectively. The latter were located at positions N01-P05, N04-P04 and P01-N01. Outlet temperatures will be noted at these six positions during the next power run and compared with the former readings.

APDA Materials Surveillance Subassembly MS1 (P01-P03) was interchanged with core subassembly M075 (N01-P03), located at a similar core position, so that it would be located at a position which is monitored by a thermocouple.

(4) Safety rod No. 89 was removed from position No. 3 in the reactor core and stored in the transfer rotor. It will be examined for possible damage to its handling head which may explain the apparent slight interference between the head and the mechanisms in the rod extension. Reactor operation will continue with the No. 3 safety rod position vacant.

(5) The diagrams on the following pages show the reactor loading after each of the above four changes, described in items (1), (2), (3) and (4). The key to the rod positions near the core center is:

SR#1	SR#2	SR#3	
SR#4	CR(Reg.)	CR(Shim)	Oscil. Rod
SR#5	SR#6	SR#7	

4. Fuel Handling Operations

	<u>Total Movements</u> <u>7/19/66 to 9/28/66</u>	<u>Cumulative</u> <u>Totals</u> <u>as of 9/28/66</u>
Reactor Hold-down Mechanism	12*	538*
Reactor Rotating Plug	169	5,151
Reactor Offset Handling Mech.		
Azimuth Movements	164	7,352
Elevation Movements	148	7,982

* One count per complete up-and-down cycle.

REACTOR LOADING 9-16-66

KEY:

CR Central 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 "CP"
 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 Syloer Division, Sylvania Electric Products Co.

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

Units shown without prefix are dummy outer radial blanket subassemblies.

- Oversize Nozzle Unit
- "F" Subassembly (Contains fuel pins with high iron plus nickel, high carbon or zirconium content.)
- "W" Subassembly (Contains fuel pins with high iron plus nickel content.)
- Blanket slugs have high carbon content. (APDA Surveillance Program Unit)
- Stringering in blanket slugs
- Large grain blanket material (Hash)
- Larger than normal spacing between the blanket elements and the support grid
- Type 347 stainless steel wrapper tube
- Handling head short
- Test flow subassembly (S-400)
- Slugs previously used in a test subassembly
- "CP" slugs
- SPACES WHERE SUBASSEMBLIES WERE MOVED

N16 N15 N14 N13 N12 N11 N10 N09 N08 N07 N06 N05 N04 N03 N02 N01 P00 P01 P02 P03 P04 P05 P06 P07 P08 P09 P10 P11 P12 P13 P14 P15 P16

REACTOR LOADING 9-22-66

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 Syloor Division, Sylvania Electric Products Co.

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

Units shown without prefix are dummy outer radial blanket subassemblies.

- ☒ Oversize Nozzle Unit
- ☐ "F" Subassembly (Contains fuel pins with high iron plus nickel, high carbon or zirconium content.)
- ☐ "W" Subassembly (Contains fuel pins with high iron plus nickel content.)
- ☐ Blanket slugs have high carbon content. (APDA Surveillance Program Unit)
- ☐ Stringering in blanket slugs
- ☐ Large grain blanket material (Hash)
- ☐ Larger than normal spacing between the blanket elements and the support grid
- ☐ Type 317 stainless steel wrapper tube
- ☐ Handling head short
- ☐ Test flow subassembly (S-400)
- ☐ Slugs previously used in a test subassembly
- ☐ "CP" slugs

SPACES WHERE SUBASSEMBLIES WERE MOVED

1N16:1N15 N14 N13 N12 N11 N10 N09 N08 N07 N06 N05 N04 N03 N02 N01 P00P P P 4 P05 P06 P07 P08 9 P09 P10 P11 P12 P13 P14 P15 P16

P16
 P15
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 P02
 P01
 P00
 N01
 N02
 N03
 N04
 N05
 N06
 N07
 N08
 N09
 N10
 N11
 N12
 N13
 N14
 N15
 N16

REACTOR LOADING 9-23-66

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 Syloor 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 Nossle Unit
- "F" Subassembly (Contains fuel pins with high iron plus nickel, high carbon or zirconium content.)
- "W" Subassembly (Contains fuel pins with high iron plus nickel content.)
- Blanket slugs have high carbon content. (APDA Surveillance Program Unit)
- Stringering in blanket slugs
- Large grain blanket material (Hash)
- Larger than normal spacing between the blanket elements and the support grid
- Type 3147 stainless steel wrapper tube
- Handling head short
- Test flow subassembly (S-400)
- Slugs previously used in a test subassembly
- "CP" slugs

SPACES WHERE SUBASSEMBLIES WERE MOVED

P16
 P15
 P14
 P13
 P12
 P11
 P10
 P09
 P08
 P07
 P06
 P05
 P04
 P03
 P02
 P01
 P00
 N01
 N02
 N03
 N04
 N05
 N06
 N07
 N08
 N09
 N10
 N11
 N12
 N13
 N14
 N15
 N16

REACTOR LOADING 9-27-66

KEY:

CR Control Rod No. 430169-
 SR Safety Rod No. 430192-
 OR Oscillator Rod
 CS Core Shim Subassembly
 CF Core Foil Subassembly
 HF Blanket Foil Subassembly
 UT 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 Syloor 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.

- Overwise Nozzle Unit
- "F" Subassembly (Contains fuel pins with high iron plus nickel, high carbon or zirconium content.)
- "W" Subassembly (Contains fuel pins with high iron plus nickel content.)
- Blanket slugs have high carbon content. (APDA Surveillance Program Unit)
- Stringering in blanket slugs
- Large grain blanket material (Hash)
- Larger than normal spacing between the blanket elements and the support grid
- Type 347 stainless steel wrapper tube
- Handling head short
- Test flow subassembly (S-400)
- Slugs previously used in a test subassembly
- "CP" slugs

SPACES WHERE SUBASSEMBLIES WERE MOVED

5. Sodium and Gas Systems Performancea. Sodium Cold Traps and Plugging Indicators

<u>September Operating Data</u>	<u>Primary System</u>	<u>Secondary System</u>		
		<u>Loop 1</u>	<u>Loop 2</u>	<u>Loop 3</u>
Hours of Cold Trap Operation	None	134	86	60
Maximum Plugging Temperature - F	Below 220	410	340	460
Minimum Plugging Temperature - F	Below 220	275	210	220

b. Sodium Activity - Maximum Values for September

Primary System*	10 microcuries Na-24/cc Na.
Secondary System	Below instrument min sensitivity of 2.5×10^{-4} microcuries Na-24/cc Na.

c. Primary System Cover Gas Analysis

	<u>Reactor Cover Gas (Argon) ppm by Volume</u>	<u>Primary Shield Tank Atmosphere (Nitrogen) ppm by Volume</u>
Oxygen	below 25	25
Carbon Monoxide	below 10	below 10
Carbon Dioxide	below 10	50
Hydrogen	4	5
Helium	below 4	below 4
Methane	below 10	below 10
N ₂ O	-	below 10
NO ₂	-	below 1
Argon	remainder	not measured
Nitrogen	1900	remainder
Dew Point	-	below - 75 F
Analysis Date	Sept 9	Sept 9

* By Calculation

d. Plant Inert Gas Consumption - Sept. 1966

	<u>Argon Cu Ft</u>	<u>Nitrogen Cu Ft</u>
Reactor Building and Primary System	12,000	273,000
Secondary System	110,000	-
Other	43,000	114,000
Total	165,000	387,000

SECTION II
DATA AND STATISTICS

B. PLANT REPAIRS AND MODIFICATIONS

1. Primary Sodium Pumps

General maintenance and repair of the No. 1, 2, and 3 pump liquid rheostats were performed several months ago. All rubber liners on the electrolyte cell walls were replaced because they had deteriorated, resulting in a failure of one of the liners and a resultant arc-over. Spacing between the stationary electrodes and the cell walls was increased by removing 1/4-inch of material from the electrodes. This will decrease the dielectric stress on the cell liners at this point. All electrical resistance blocks were replaced.

In early May 1966, the No. 1 pump was taken out of service for major repairs, 14 months after its last similar shutdown. Principal items were replacement of the pump shaft oil face seals and the shaft thrust bearings; both items have required frequent replacement in the past.

2. Secondary Sodium Pumps

The secondary sodium pumps are currently running virtually maintenance free. The only significant repair item in the recent past has been the replacement of a Thyratron vacuum tube on one of the variable speed magnetic couplings. There are two of these tubes in the power supply to the couplings of each of the three secondary pumps--this was the first time tube replacement has been necessary.

3. Reactor Safety Systems

After operating since January without difficulty, the No. 2 and 3 safety rods failed to operate satisfactorily in September. Key to rod positions near the core center is:

SR#1	SR#2	SR#3	
SR#4	CR(Reg.)	CR(Shim)	Oscil. Rod
SR#5	SR#6	SR#7	

The No. 2 rod delatched on four different occasions, within a few days, scrambling the reactor each time. One scram was from 26 Mwt (during an approach to 67 Mwt power), the other scrams were from less than 1 Mwt.

Recording instrument traces were made of the latch magnet breakaway current, and analysis of the trace revealed a peculiarity. Inspection of the magnet and its armature revealed a 0.016 inch misalignment of the mating surfaces whereas the design value is 0.002 inch.

It was observed that the magnet could be manually tapped in or out of alignment. The magnet is held too tightly for free self-alignment but is not tight enough to prevent a shift in alignment by applying a light lateral force. Before magnet realignment, breakaway current traces taken with the safety rod at the full up position showed the initiation of armature movement at 35 milliseconds with the main breakaway at 118 milliseconds. After realignment, the initiation of armature movement occurred at 50-53 milliseconds with the main breakaway occurring at 110 milliseconds.

It has been theorized that the armature and magnet faces are not parallel when they meet, but are forced into a parallel position by the pull of the magnet. Then, as the magnet face attraction decreases (during the current decrease following a scram), the faces pull back into their normal (nonparallel) positions before the main breakaway.

After realignment and refacing of the magnet, the No. 2 safety rod operated without further difficulty and the breakaway current was increased. Further study and a possible redesign is planned on the magnet-armature alignment problem to improve contact between the two surfaces.

During the same period when the No. 2 safety rod problem occurred, the No. 3 safety rod drive extension encountered interference sufficient to slip the drive clutch when re-engaging safety rod No. 89 after a rod drop. Instrument recordings were made from strain gauges attached to the No. 3 rod drive. The interference at first occurred between the cocking tube and the lower gripping surface of the safety rod handling head washer. The amount of interference varied widely from test to test; about half the time there was no interference, sometimes slight and other times greater interference. At no time did the interference completely prevent full rundown of the gripper.

The safety rods in the No. 3 and 6 positions were then interchanged and the strain gauge tests repeated. No further cocking tube interference was noted at either position but at the No. 6 position interference was observed intermittently between the latching fingers and the top portion of the handling head washer. The other safety rods were tested and no interference was observed. Since the problem shifted to the No. 6 position when the safety rod No. 89 was transferred there, it was assumed that the rod itself is probably at fault rather than the extension.

The safety rods in the No. 3 and No. 6 locations were first returned again to their original positions, then the No. 89 rod from position No. 3 was removed from the reactor core as reported on Page 12.

4. Instrumentation

a. Steam Generator Sodium Outlet Thermocouples

Major modifications to the steam generator sodium outlet temperature instrumentation and control system were recently completed. The sodium temperature measurement at the steam generator outlet is made with the TE-301 sensing elements. If the measurement is not the same as the desired value (maintained constant at approximately 517 F), feedwater flow control valves are adjusted to restore the steam generator sodium outlet temperature.

The original TE-301 resistance temperature detectors were installed close to the steam generator sodium outlets where it is felt that sodium streams at different temperatures are not adequately mixed, causing wide variations in the temperature measurement. In order to promote better mixing, these elements have been replaced by Leeds & Northrup iron-constantan thermocouples installed in wells at a point about 20 feet and two 90 degree bends downstream. As a result, bias and noise have been reduced. Four adjacent thermocouples were installed in each loop. Two are for feedwater flow control, one for the interlock circuits and one is a spare. The thermocouples have a faster response time than the original RTDs.

Bailey transmitters were installed to receive the direct current output from the thermocouples and to transmit a proportional pneumatic signal to the feedwater flow control system. They replaced electric-pneumatic transmitters which received their input from the RTDs. Tubing modifications were made to the TE-301 selective circuitry and to the emergency and main feedwater control systems. The emergency cooling set point control was changed to provide a single adjuster for all three loops; the temperature bias set point controls were changed to provide individual set point controls for each steam generator sodium outlet temperature.

Reactor operations at power to adjust the automatic feedwater control system were performed during two runs at 67 Mwt. Performance data were obtained at power, during the approach to power, and during power reduction. A few calibration errors and equipment malfunctions were discovered and corrected. The automatic feedwater control of loop No. 2 is working satisfactorily, some final adjustments are to be made on loop No. 3, and loop No. 1 has not yet been available for testing.

b. Radiation Monitors

During the months of July, August and September 1966, there was a total of 14 malfunctions associated with the plant radiation monitors. The listing on page 24 describes the difficulties.

5. Fuel Handling Equipment

- a. The fuel handling equipment key interlock system was successfully tested several months ago.
- b. A calibration gauge for the offset handling mechanism was inserted through the exit port into the reactor transfer rotor, but was withdrawn before any measurements were made because of difficulties with its special hoist gripper. When the gripper was delatched from the gauge and raised out of the transfer rotor, it was discovered the gauge had remained with the gripper. After the exit port valve below the hoist mechanism was closed, an attempt was made to close the gripper fingers. This resulted in the gauge dropping off of the gripper and onto the top of the closed gate valve. Inspection of the gripper and gauge showed a heavy coating of sodium oxide. Since the gripper delatch mechanism is spring actuated, it is thought that the gripper fingers did not fully open due to the presence of the excessive sodium oxide. It is thought that oxide on the sodium surface in the exit pipe resulted from a brief exposure of the exit port atmosphere to air during installation of the calibration gauge's special hoist.

The gauge is designed to measure any small discrepancy in alignment between the OHM and a mating subassembly. Data will be obtained at various core and blanket positions by use of the gauge and the alignment at the remaining positions will be calculated from the data. The gripper is being modified to eliminate the previously observed problems. The latching cable is being replaced with a mechanical latch rod so that the latching position will be observable. The exit port sodium surface will be inspected and cleaned if necessary.

- c. The cask car was used in September to carry a special safety rod transfer pot to the reactor exit port and to lower it into the transfer rotor. The pot was used in the interchange of the safety rods as reported in IIB.3. above. A photo of the cask car, taken in the passageway between the reactor building and the fuel and repair building during the operation, is shown on Page 33.
- d. The steam cleaning machine received a steam, argon and water cycle check. Five simulated cleaning cycles were run without actual sodium. The results were satisfactory except for the drying cycle where three gas leaks prevented achievement of the high vacuum needed for good drying. The leaks were repaired. Further tests are being conducted using post-cleaning argon purge cycles to aid in drying the cleaning chamber.

6. Steam Generators

Steam Generator No. 1 - The new test thermocouple penetrations were not leak-tight at the threaded joint between the steam generator wall and the adapter which seals the penetration. After venting the water, the leaks were repaired by welding the threads and the unit was returned to service.

Steam Generator No. 2 - Further testing was conducted to determine the source of a small leak. The sodium was drained and gas pressure applied to the shell side with the water tube sheet flooded. This revealed one pin-hole leak at a tube-to-tube sheet weld. The leak was repaired by welding but a stress crack developed, as a result of the weld, extending into the ligament of the tube sheet. The crack was ground out, rewelded with preheating, leak tested and found to be tight.

A dye penetrant check of the tube sheet showed eight other faulty tube welds, which were counterbored and rewelded. These eight welds had been made by the manufacturer. A leak test with the tube sheet flooded revealed two of the eight repaired welds still leaking, plus one additional weld. These later welds were repaired and retested. The steam generator is now in the process of being reheated and refilled for operating service. All welds were made using an inert gas shielded arc, tungsten electrode and 2-1/4% chromium, 1% molybdenum filler rod.

Steam Generator No. 3 - This unit was out of service during the last half of the month for replacement of a leaking water manifold gasket. When the replacement gasket was leak tested with nitrogen it was discovered that it had not seated properly. The gasket was replaced the second time and the unit returned to service.

RADIATION MONITOR MALFUNCTIONS

July through September 1966

<u>Type of Failure</u>	<u>Number of Failures</u>	<u>Number of Monitors Involved</u>	<u>Type of Monitors Involved</u>
Filter paper drive mechanism	4	3	Air sampling - Particulate radioactivity (3)
Compressors or vacuum pumps (which draw the gas sample through the detector)	4	4	Air sampling - Particulates (2) and gaseous (2)
Broken gas or water sampling lines	2	2	Air sampling (1) Water sampling (1)
Detector or circuit failures	4	4	Air sampling - gaseous activity (1), area gamma monitors (3)
Totals	14	10*	

* Total allows for three monitors which were involved in two types of failures.

SECTION III

NUCLEAR TEST PROGRAM

A. SETBACK TESTS AT 100 MWT - 3 LOOPS OPERATING

The function of the setback system is to correct abnormal reactor operation before it becomes abnormal enough to cause a scram.

The first setback test was initiated by a manual power ramp at 34 Mwt/minute, starting from 55 Mwt. The setback occurred at the 75 Mwt power range trip point rather than at the 620 F reactor outlet sodium temperature set point. The total number of automatic setbacks on this test before reset was two.

The second setback test was initiated by an automatic power ramp at 1.6 Mwt/minute. The power range trip point was raised to 220 Mwt so that the setback occurred from the 620 F reactor outlet trip point. The initial temperature was 605 F. The total number of setbacks on this test before reset was four.

The power reduction was 26 Mwt/setback and the temperature reduction was 27 F/minute. The setback system automatically inserts the regulating control rod at a fixed velocity for a specified time interval and in steps, or setbacks, until the initiating signal (power or temperature) has been reduced to an acceptable level.

The results of these tests are satisfactory and agree with those predicted on the reactor simulator.

B. SAFETY ROD DROP TESTS AND LOW POWER SCRAM TRACE

The safety rod drop tests were made in July at a power of 800 Kwt. The neutron flux decay signals were recorded on magnetic tape from which reactivity vs time curves were prepared. Each rod was dropped individually, then all rods were dropped simultaneously, finally all rods except #4 were dropped simultaneously.

The results were satisfactory and compare well with the last rod drop test. Rod worths were computed to be \$1.25 - \$1.48 each, \$8.40 for seven rods and \$7.80 for six rods. Lag time (time from magnet delatch to 5 percent reactivity reduction) averaged 0.32 seconds. Insertion time (from 5 percent to 95 percent reactivity reduction) averaged 0.37 seconds.

The low power scram trace is similar to the safety rod drop tests with the following exceptions:

1. The test is automatically actuated from the scram logic circuitry, rather than manually actuated.

2. Time is computed from the moment of actuation rather than from magnet delatch.
3. Power vs time is plotted rather than reactivity vs time.

The low power scram trace made at the end of September has a time to 10 percent power somewhat longer than the previous test but was well within the one second limit. This test was made with 6 safety rods, the earlier test was with 7 rods.

C. BASE REACTIVITY MEASUREMENTS

During the low power test program from late 1963 until late 1965, a small but gradual increase in the reactor core reactivity has been observed. After each period when the primary sodium temperature was raised, from the normal 500-520 F to 600-740 F, there was a marked reactivity increase.

The total increase of 50 inhours, over the two-year period prior to operation above 1 Mwt, is thought to be due to two causes: (a) Contraction of the fuel pins due to phase retransformation of the uranium - molybdenum alloy, and (b) hydriding of the fuel pin zirconium cladding. A core subassembly was removed from the reactor and by examination and calculation, the above two mechanisms appeared to account for the reactivity change.

Since operation above 1 Mwt began in late December 1965, there has been a decrease, presumably resulting from burnup. There have been two exceptions: a 1.6 inhour increase after 5 days isothermal operation at 650 F primary sodium temperature and a 6.1 inhour increase due to build-up of Pu-239 from the decay of Np-239. This latter increase was noted between a measurement taken immediately after the 3-day run at 100 Mwt in August 1966, and a measurement taken two weeks later. The net effect since December 1965 has been a reduction of 33 inhours in reactivity. This is a greater amount than that predicted to result from the calculated burnup - possibly due to a reversed fuel phase transformation.

As a result of the fuel relocations during September (detailed on page 11), a base reactivity measurement made at the end of September showed a further decrease of approximately 6 inhours. Present reactor core base reactivity is approximately 11 inhours above the late 1963 value.

For further information refer to ANS-100, page 45, or to page 14 in Report 2/4, "Recent Operating Experience of the Enrico Fermi Atomic Power Plant" given at the BNES May 1966 London Conference on Fast Breeder Reactors.

SECTION IV

PLANT TEST PROGRAM

A. Reactor Vessel Temperatures

As reported in August, there are three areas on the reactor vessel shell or nozzles where the temperature differences between nearby locations exceed predictions. The main emphasis during September was placed on analysis of the temperature difference between the top of the transfer rotor container and the upper reactor vessel wall.

Based on readings from the nearest thermocouples to the area of interest, which are four feet apart, a predicted temperature profile of the area was determined by Franklin Institute. Their preliminary calculations showed that the maximum vessel stress, with a 125 F sodium temperature increase through the core, is 10 percent lower than that allowed in the ASME code. This conservative analysis permits reactor operation at 100 Mwt with 3 loops.

The boundary conditions used for the study, which establish the temperature profile and stresses, will require verification and additional test data. The test program could not proceed during September because the safety rod difficulties and unexpected minor steam generator problems precluded extensive reactor operation. The program has been formulated and accepted-- it consists of fully instrumented tests at higher than normal sodium flows to the blanket subassemblies, rotation of the transfer rotor to circulate the transfer rotor container sodium, and installation of a probe in the transfer rotor container to measure the vertical temperature distribution in the sodium. The probe was placed on order, for October delivery.

B. Primary Inert Gas System Integrity

Integrity tests were conducted on this system. The system consists of an argon blanket gas over the sodium in the reactor, pumps, intermediate heat exchangers and overflow tank, together with gas compressors and storage tanks in the inert gas building with interconnecting piping and valves. The argon becomes activated when the reactor is at high power operation.

The system was given "soak" tests of 19 and 24 hours duration, during which gains in gas inventory were reported. An additional test was performed in which all parts of the system were operated by changing the gas pressure control set

points. There appeared to be no detectable out-leakage when the system is inactive, and a leakage of approximately 2 scfh when the system was kept in operation by frequent changes in the control set points. The leakage appears to be the summation of several small undetermined leaks. The system is normally inactive.

A later test was run in August during the three-day 100 Mwt operation to determine the primary inert gas leakage from the reactor into the machinery dome. The dome is mounted atop the reactor to collect this leakage, thus preventing its spread throughout the upper reactor building. By measuring the argon activity in the reactor cover gas and in the machinery dome atmosphere it was estimated that this leakage is 3/4 scfh.

C. ZIRCONIUM HYDRIDING

Four tests of zirconium clad reactor core fuel pin segments have been conducted at the Detroit Edison Company engineering laboratory. Three tests were with 750 F sodium and one with 830 F sodium, each test included four specimens. The largest rate of zirconium hydriding was during a 14-day test at 750 F with hydrogen being readded to the sodium to maintain the concentration from 1 to 5 ppm; smallest rate was during a 14-day test at 830 F, during which no hydrogen was added, and in which the hydrogen concentration was in the range of 1 to 2 ppm. In the former case the hydriding rate was 30 to 714 ppm/day, assuming linear rate for the test period. It was negligible in the 830 F test.

Over the past year the total hydrogen in the primary sodium has been maintained at 1.0 ± 0.3 ppm. This includes hydroxide hydrogen and nonhydroxide hydrogen. This hydrogen level is felt to be acceptable. See the summary sheet of recent sodium chemical analyses. Section IV D, for the hydrogen content of recent plant sodium samples.

D. SODIUM SAMPLING

Approximately once a month a sample is taken of the plant primary sodium and is sent to the Atomic Power Development Associates sodium laboratory for chemical analysis. Recently the analysis was expanded to include an atomic absorption spectrometry test for iron, chromium and nickel - elements of interest with respect to stainless steel corrosion.

The Atomic Power Development Associates sodium laboratory has recently been expanded to provide facilities for analysis of radioactive sodium and the plant sodium sampling station will be modified to permit removal and transport of radioactive sodium samples.

SECTION IV

The following table summarizes the analyses of four recent sodium samples:

Sample Coil No.	28	29	30	32
Sampling Date	3/29/66	4/14/66	5/1/66	8/8/66
Carbon, ppm	30,31,32 39,47,36*	36,38,42,45 63,126,77*	34,40,66 104,123,73*	42,54,56 94,61*
Carbonate Carbon, ppm	1,1,2	1,2,2	2,2	1,2
Hydroxide Hydrogen, ppm	.8,.9	.8,.9	.6,.6,.7	1,1,1
Nonhydroxide Hydrogen, ppm	.1,.2	.3,.3	.1,.1,.1	.1,.1,.2
Oxygen	9,9	12,14,14	7,7,8	6,6,7
Nickel	-	.1	.2	below .3
Iron	-	3.2	1.9	.8
Chromium	-	below .3	below .3	below .1

NOTE: Each sodium sample coil was analyzed at several different points along its length to provide the separate readings indicated.

* Average

SECTION V

OTHER OPERATING EXPERIENCE

A. ROTATING SHIELD PLUG INCREASED FRICTION

During fuel handling operations in September to interchange two safety rods, the rotating shield plug motor drive tripped out on overload on the first and third rotational movements. A check of the motor current showed the starting current to be normal but the running current was high. However, the running current decreased with additional movements and approached normal.

The resistance to plug rotation, as evidenced by the high motor current, is probably due to a buildup of condensed sodium vapor at the plug annulus. Two months had elapsed since the plug was last moved and during this time the reactor was operated considerably--raising the sodium temperature and the amount of sodium vapor in the cover gas.

Repeated rotations of the plug appear to wear away the interferences; no further action is planned.

B. FISSION PRODUCT DETECTOR AND COVER GAS ANALYZER

These instruments continuously sample a side stream of the primary system cover gas taken at the reactor or at the No. 1 or 3 pumps. In the past there has been considerable difficulty with plugging of the sample lines. At present the fission product detector is operated each time the reactor is in operation and the reading noted on a continuous recorder, with an alarm for a high reading. During the 3-day run at 100 Mwt the instrument did not perform properly, apparently due to the lack of an optimum pulse height discriminator setting. It has been adjusted and will receive further checking during continued operation at higher power levels. Background count has been determined and with the present adjustments the instrument is capable of detecting an increase in fission product level from reactor operation at power levels as low as 250 kwt.

The cover gas analyzer is a companion instrument which cyclically records hydrogen, oxygen, nitrogen, methane and carbon monoxide. This instrument has been performing well for the past two years. It is intended primarily to check for offgassing of the graphite blocks in the reactor rotating shield plug.

Grab samples of the reactor cover gas are also taken periodically and analyzed in the laboratory at Fermi (see Page 17). A new, shielded sampling station has recently been placed in service. It is located above the fission product detector building--formerly the operator had to enter the building to take a sample.

C. Shielding Tests

Radiation measurements for the first stage of the plant radiation survey program were taken during the 3-day run at 100 Mwt. The sodium activity was at 90 percent of its saturation level at the time of the measurements. Neutron and gamma readings were taken over the entire reactor building floor--no radiation was found in excess of the background value of 0.1 mr/hr. The background level with the reactor at 800 kw was previously determined to be less than the minimum instrument sensitivity of 0.01 mr/hr.

The increase in the reactor building above-floor activity is due to leakage through the sodium cover gas seals. Above-floor air particulate samples gave activities near 8×10^{-9} microcuries/cc, due to rubidium-88. Rb-88 is a daughter product of Kr-88, part of which is formed from decay of Br-88 and the remainder of which is a direct fission product. The maximum permissible concentration in air of Rb-88, for a 40 hr/week exposure, is 1×10^{-6} microcuries/cc.

Samples were also taken of the air inside the reactor machinery dome and of the sodium cover gas. The samples contained both argon-41 and other radioactive isotopes. The cover gas sample activity was 0.106 microcuries/cc after 24 hours of reactor operation, while the machinery dome air sample averaged about 8×10^{-6} microcuries/cc (assuming all the activity was due to A-41).

Gamma measurements were also made in areas external to the reactor building containing systems connected to the reactor building. Time did not permit a survey of the gas building and tunnel; this check will be made during future high power operation. All external readings were below 0.06 mr/hr except for three locations outside of the primary sodium service system cold trap cell. Sodium was circulating through this system during the 100 Mwt run. The highest measurement, 2R/hr, was at the sodium sampling station above the cell. A lead brick wall has been installed to shield a streaming path from the cell. The other two values above 0.06 were 0.3 mr/hr outside the bricked up access door to the cell and 3.0 mr/hr on the sodium storage room side of the piping labyrinth leading into the cell.

SECTION VI

MAINTENANCE AND ADDITIONS

A. ELECTRICAL INDUCTION HEATING

Overall performance of this electrical induction sodium heating system has been very good. Almost all the equipment and piping containing sodium is induction heated with 120 volt, 60 cycle current. The induction loops are placed on the outer surface of the heat insulation where they are accessible for adjustment and repair. The heat insulation also protects this copper wire from the high sodium temperatures at the pipe or vessel wall. Even so, the wiring becomes hot and is therefore electrically insulated with asbestos, which also resists radiation damage.

Several months ago when the reactor building below floor atmosphere was temporarily converted to air, repairs were made to seven sodium heating circuits in this area. In some cases the leads to the circuits had failed due to overheating or mechanical abuse during construction. Better quality lead wire is on order. In one case the induction heating wire itself failed because its location in a hot, congested area caused overheating. There are approximately 330 induction heating circuits in the plant.

B. MACHINERY DOME AIR LOCK

A large, square, gas-tight metal box was recently brought into the reactor building and is being fitted to the machinery dome entrance hatch, as shown in the photos on Page 33. Its purpose is to enable maintenance of an argon atmosphere in the machinery dome during possible future replacement or repairs of the reactor cover gas seals. Normally the machinery dome atmosphere is air, at negative pressure relative to the upper reactor building and also relative to the reactor argon cover gas. This air is vented to the waste gas stack, with isolation effected by quick closing valves in the event of an excessive radiation level.

The cover gas seals within the machinery dome, especially the rotating plug rubber seal (called the Klosure seal), are at present doing a good job of maintaining the desired degree of leak tightness, as reported in Section IV B.

After leak tests of the air lock have been completed and the unit otherwise checked out, it will be removed and stored outside the containment building. Maintenance personnel entering the argon atmosphere of the machinery dome through the air lock would breathe through masks supplied from air hoses or self-contained air packs.

SECTION VI

Maintenance Air Lock
for the Reactor Machinery
Dome - Side View
Looking Northwest

Machinery Air Lock
for the Reactor Machinery
Dome - Upper Corner View
Looking West

Cask Car in the
Corridor En route
to the Reactor Building