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COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT
ENRICO FERMI ATOMIC POWER PLANT

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Enclosure: Isometric, Cut-away Drawing of the Enrico
 Fermi Atomic Power Plant

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 of 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. Electrical production using nuclear heat was initiated on August 5, 1966.

The scope of the first two or three monthly reports will not be limited to items occurring since August 1, 1966, 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 is included at the back of this report.

Note

Since this report is intended to follow closely the current proceedings at the Fermi plant, it must necessarily be a preliminary information document, subject to supersedence in light of subsequent experience.

SECTION I
CURRENT EXPERIENCE SUMMARY

July 1966

The reactor was operated full time or part time on ten days during July: two days at 67 Mwt, four days at 100 Mwt and four days at less than 1 Mwt. Scheduled testing at 67 Mwt, 2-loop operation, was completed except for oscillator tests with automatic feedwater control. The 100 Mwt operation was with all three primary and secondary sodium loops in service.

Four of the nuclear test procedures to be conducted at 100 Mwt were completed: oscillator tests, a heat balance measurement, a safety rod drop test and a power coefficient measurement. The reactor was operated on only one day, at less than 1 Mwt, in the latter half of the month. During this time, the No. 2 steam generator was out of service for replacement of a leaking water manifold gasket. Piping modifications were also made at this time to increase the capacity of the steam dump lines so that the reactor can be operated at 200 Mwt, if desired, without passing any steam through the turbine. See Page 30, Paragraph D.

Difficulty with steam generator stability was encountered as the reactor power was increased and the corresponding steam generator conditions reached the region where superheated steam was becoming available. Superheat was obtained by decreasing the steam pressure or increasing the steam generator sodium outlet temperatures. Two scrams occurred as a result of this lack of stability, which may be due to the light loading of the steam generators at this point. Revisions in the control and operating methods are in process. See Page 18, Paragraph D.

August 1966

The reactor was operated full time or part time on ten days during August: four days at 100 Mwt, four days at 67 Mwt and one day at less than 1 Mwt. Pulse, noise and setback tests were made at 100 Mwt, then the reactor was operated for 60 hours from August 5 through 7 at 100 Mwt and generated 22 Mwe for 53 hours. This represents the first electrical output from the reactor plant. The operation was very stable. The control rods were twice withdrawn slightly to compensate for the reactivity loss due to burnup.

One purpose of the 60-hour run was to obtain information on numerous plant systems and equipment including: shield temperatures; sodium and gas purity; equipment temperatures and component movements, radiation levels and gas leakage. Results

were generally as expected. The 100 Mwt run was terminated twelve hours ahead of schedule because of a high hydrogen level in the No. 1 steam generator, caused by leaks at the water tube sheet. Leak repair and installation of thermocouples for water and steam temperature distribution measurements kept the unit out of service the rest of the month. The thermocouples are stainless steel sheathed, magnesium oxide insulated, tip-grounded iron-constantan.

Two-loop, 67 Mwt operation late in the month checked out the steam generator sodium temperature detector modifications providing signals to the automatic feedwater control system. These modifications will be explained in detail in the September report. Operation was satisfactory except for oscillations in Loop 3 feedwater flow control. Revised feedwater control programs were run to investigate the controllability of the rate of change of steam temperature upon superheating and desuperheating the steam generators. In addition, the reactor was operated with maximum sodium flow through the blanket subassemblies and it was determined that the temperature differential between the transfer rotor container top head and the upper reactor vessel shell was substantially reduced. This differential has been exceeding expectations. See Page 17.

SECTION II

DATA AND STATISTICS

A. Plant Operations

1. Reactor Operating Statistics

	<u>July</u>	<u>August</u>	<u>Cumulative through August</u>
Reactor Start-ups	14	7	394
Scheduled Scrams (Includes usual test scram prior to reactor start-up)	22	13	699
Unscheduled Scrams	2***	1**	52
Maximum Reactor Power, Mwt	114*	114*	114*
Integrated Reactor Power, Mwt - days	146*	439*	853*
Hours of Reactor Operation	46	160	2067
Maximum Burnup in Central Core Subassembly, a/o	-	-	0.06
Maximum Electrical Output, Gross Mwe	-	22	22
Integrated Electrical Output, Gross kwh		1,160,000	1,160,000

* Indicated values, actual values 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. See Page 12.

*** Due to steam generator instability experienced when breaking into superheat. See Page 18.

2. Daily Operation Curve Sheets

(See the following three fold-out pages)

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

FERMI DAILY REACTOR PLANT OPERATION DURING THE MONTH OF AUG 1966

SHEET NUMBER TWO

SHEET NUMBER THREE

NOTE: VALUES ARE AVERAGE OF ALL LOOPS IN SERVICE

SECTION II
DATA AND STATISTICS

A. Plant Operation

3. Reactor Loading

No changes were made during August at any of the positions in the reactor core.

The diagram on the following page shows the reactor loading at the end of August 1966.

REACTOR LOADING

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 "CP"

LATTICE: 103 CORES

34 IRB
486 ORB

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 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 347 Inless steel wrapper tube
- ☐ Handling head short
- ☐ Test flow subassembly (S-400)
- ☐ Slugs previously used in a test subassembly
- ☐ "CP" slugs

SECTION IIA

4. Fuel Handling Operations

	Cumulative Totals as of September* 1964	July* 1966
Reactor Holddown Mechanism, Cycles	372	526
Reactor Rotating Plug, Movements	4254	4982
Reactor Offset Handling Mechanism		
Azimuth Movements	6428	7188
Elevation Movements	6905	7834
Units Handled	337	632
Reactor Transfer Rotor - Movements	373	567
Units Handled	286	-
Cask Car - Movements	315	474
- Units Handled	282	-
Fuel & Repair Bldg Transfer Rotor		
Movements	267	287
Units Handled	266	-
Steam Cleaning Machine,		
Cleaning Operations	Approx 200	10**
Fuel Cut-up Machine,		
Cuts Made	Approx 10	Approx 10

* Dates selected on basis of best available data.

** Machine substantially modified after September 1964.

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

<u>August 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	50	43	188	65
Maximum Plugging Temperature - F	below 240	300	350	360
Minimum Plugging Temperature - F	below 220	270	225	265

b. Primary Sodium Chemical Analysis* - after 3 days @ 100 Mwt

Carbon 42,54,56,94	Carbonate Carbon	1,2
Oxygen 6,6,7	Hydroxide Hydrogen	1,1,1
Nickel below .3	Nonhydroxide Hydrogen	.1,.1,.2
Iron .8		
Chromium below .1	Note: Values are in ppm	

c. Primary System Cover Gas Analyses

	<u>Reactor Cover Gas (Argon)</u>	<u>Primary Shield Tank Atmosphere (Nitrogen)</u>
	<u>ppm by Volume</u>	<u>ppm by Volume</u>
Oxygen	Below 25	45
Carbon Monoxide	Below 10	Below 10
Carbon Dioxide	Below 10	88
Hydrogen	Below 8	5
Helium	Below 4	Below 4
Methane	Below 10	Below 10
N ₂ O	-	Below 10
NO ₂	-	Below 1
Water	-	-
Argon	Remainder	4,750
Nitrogen	2,350	Remainder
Dew Point	-	Below -75 F
Analysis Date	August 26	August 9

* One sodium sample coil was analyzed at several different points along its length to provide the separate readings indicated.

d. Sodium Activity - Maximum Valves for August

Primary System*	4.7 Millicuries Na-24/cc Na.
Secondary System	Below instrument min sensitivity of 2.5×10^{-4} Microcuries Na-24/cc Na

e. Plant Inert Gas Consumption

8-10-66 to 9-12-66	Argon Cu Ft	Nitrogen Cu Ft
Reactor Bldg & Pri System	27,000	213,000
Secondary System	113,000	55,000
Other	25,000	97,000
Total	165,000	365,000

B. Plant Repairs and Modifications

1. Primary and Secondary Sodium Pumps

The vertically-mounted seal lubricant pump on the No.2 primary sodium pump was replaced with a horizontal pump. A similar modification was made earlier on the No. 3 pump, and the No. 1 pump had a horizontal seal oil pump as original equipment. The vertical pumps required excessive maintenance.

The main pump shaft mechanical seals (face seals for the argon gas) are to be replaced with new design seals which are on order and overdue. Lack of concentricity has accelerated the wear on the present design seals, causing frequent replacement, which is difficult and requires removal of the pump from service. Average seal life has been only six months. The new design will first be proof-tested on one of the three pumps.

When the Fluorolube lubricant leaks past the seal it flows onto a slinger ring and down into a catch chamber. However, probably some lubricant has been introduced into the sodium, perhaps by volatilizing out of the chamber. More reliable, redesigned seals would minimize this contamination.

On the whole, pump performance continues to be very good.

* By calculation

2. Reactor Safety Systems

A calibration check of the nuclear portion of the reactor safety system has recently been accomplished, with the exception of the verification of the mixer drivers which is to be done during the next scheduled shutdown. The process portion of the safety system will be checked in about six months.

An unscheduled scram occurred on August 31 at low power due to delatching of one safety rod. The cause is suspected to be poor holding magnet contact. Further detailed information will be furnished in the September report.

3. Instrumentation

Radiation monitors continue to require a large amount of maintenance and repairs. Data on instrument repairs will be presented in the report for September.

4. Fuel Handling Equipment

a. Offset Handling Mechanism

During fuel handling, the OHM failed to stop at an indicated 0000 elevation and was stopped manually by the operator at 999.8. The gripping mechanism should have bottomed on the reactor subassembly handling head at elevation 0000. It has been concluded that the fault was at the latch rod locating bearing which is in the topworks of the OHM, inside the machinery dome. The top end of the latch rod is held at a fixed relative elevation by this bearing, but is permitted by the bearing to rotate when the OHM arm is rotating.

The rod is held in firm contact with the ball bearing by a retainer. The retainer is threaded onto the latch rod and a jam nut is provided--intended to keep the retainer from unscrewing. However, there is a spacer piece between the nut and the retainer; a cap on this spacer rubs against limit switch S-29 whenever the OHM is rotated while carrying a subassembly. Unexpectedly large scuffing forces between the cap and the switch permitted the cap and spacer to rotate, loosening the locknut and permitting the retainer to unscrew, thus developing play at the bearing. This had the effect of shortening the latch rod, meaning the gripping mechanism at the lower end had to go further to engage the subassembly handling head. This resulted in the false elevation reading.

The retainer was reset and dowelled to the latch rod, the OHM then operated properly.

b. Cask Car

After leak tests, the cask car acceptance tests were completed by transferring 21 finned pots in and out of the cask with no major difficulties. A further test using a simulated 2.4 kw decay heat subassembly showed the maximum finned pot sodium temperature to be 538 F when the cask car cooling was shut down. Disassembly for cleaning and inspection revealed a heavy uranium oxide coating which had built up on the uranium ball valve, preventing optimum performance. The valve is located under the cask and serves as its "gate" during fuel transfer operations. It has been cleaned.

A new fuel transfer facility to replace the cask car is on order from Atomics International with mid-1967 delivery expected.

c. Cut-up Machine

After an extended period of inactivity this machine has been taken out of the cut-up pool and is being reactivated to receive, in the near future, fuel elements from the reactor for removal of the axial blanket ends before shipping to Battelle Memorial Institute in Columbus, Ohio, for inspection of the core pins. The chip collection system has been improved, deteriorated hydraulic hoses and flooded limit switches replaced and an operational check has been performed successfully, with the machine out of the pool.

5. Steam Generators

The tabulation on Page 13A lists in detail the tests, repairs and modifications of the steam generators during the period of May-August 1966. Also noted are the dates when the steam generators were in operation and the power level during operation.

MONTH	STEAM GENERATOR 1	STEAM GENERATOR 2	STEAM GENERATOR 3
May 1966	1. 567 pitted tube-to-tube sheet welds repaired by Microbrazed overlay. 2. 331 tube welds repaired by TIG welds using Cr-Mo filler rod. 3. All inner row tubes in water tube sheet counterbored and TIG rewelded with Cr-Mo filler. 4. Four tubes plugged. -----(No operation above 1 mwt during May)-----	1. Bubble test showed 2 weld leaks at water tube sheet. 2. DP check showed 164 doubtful welds - 11 with flaws. 3. 179 tube welds repaired by counterboring & TIG rewelding with Cr-Mo filler. -----(No operation above 1 mwt during May)-----	In service
June	-----(Operation at 67 mwt on June 23, 24, 25, 28 & 30 - Loops 1 & 2)----- During operation cover gas H ₂ <250 ppm----- During shutdown cover gas H ₂ 1000 ppm-----	 <250 ppm <2000 ppm	1. Replaced leaking stm manifold cover gasket. 2. Passed N ₂ leak test.
July	-----(Operation at 67 mwt on July 1 and 5 - Loops 1 & 2)----- -----(Operation at 100 mwt on July 8, 11, 12 & 14 - Loops 1, 2 & 3)----- In service	Replacing leaking water manifold cover gasket and DP checked water tube sheet - no cracks, etc.	In service
August	-----(Operation at 100 mwt on August 1, 3, 5, 6 & 7 - Loops 1, 2 & 3)----- 1. Cover gas H ₂ <1000 ppm at start of 3-day run. 2. Cover gas H ₂ 6500 ppm at end of 3-day run. 3. Leak rate tested at 450 cc N ₂ /day. 4. Bubble test - 2 weld leaks at water tube sheet, outer & inner rows - counterbored & TIG rewelded with Cr-Mo filler; also tube wall leak 1/4" below top face of tube sheet - plugged. Cumulative total of plugged tubes is 15. 5. DP check showed 29 cracks where welds had been repaired in May with Microbrazed overlay. Overlay removed and TIG rewelded with Cr-Mo filler. Additional surface cracks in the Microbrazed overlay repaired by grinding down to good metal. 6. Bubble test and DP check showed no steam tube sheet leaks. 7. Installed 41 TCs to check temp distribution. Will relate to flow distrib for possible correlation to flow instability. 6 TCs at bottom of downcomers; others in water & steam manifolds.	1. Cover gas H ₂ approx 2000 ppm all during 3-day run. 2. Passed N ₂ leak test. 3. Refilled with water - H ₂ increased, vented water. 4. Passed second N ₂ test. 5. Back in service on August 20. --(Reactor at 67 mwt on August 23, 24, 25 & 31 - Loops 2 & 3)-- ABBREVIATIONS: Cr-Mo - 2-1/4% chromium - 1% molybdenum steel DP - Dye penetrant H ₂ - Hydrogen evolved from Na-H ₂ O reaction N ₂ - Leak check measuring rate of loss of 600 psi nitrogen from tube side TIG - Tungsten electrode, inert gas blanketing TC - Thermocouples	In service

SECTION II - DATA AND STATISTICS
B.5 Repairs and Modifications to Steam Generators

SECTION III

NUCLEAR TEST PROGRAM

A. Power Calibration at 100 Mwt

Heat balance test results at 100 Mwt show that the neutron instrumentation used to measure reactor power is reading too high by approximately 12 percent. The instrumentation consists of an intermediate range ion chamber installed in a spare instrument guide tube, with a Keithley linear micro-microammeter reading the ion chamber output current. This power level instrumentation is separate from the neutron detectors used for reactor control. The reading from this instrumentation is commonly termed the Keithley reading.

A total of five runs was taken at 100 and 67 Mwt. When corrected using the best available heat balance, there was close agreement as to the amount of error in the present Keithley calibration. The present calibration was based on Nuclear Test Procedure No. 2 and was made several years ago at a power level of less than 1 Mwt. An error of about 10 percent was recently found in the calculations on which the present calibration has been based. Furthermore, the calibration of the ion chamber may have shifted due to its having been repositioned.

Power level readings will be based on reducing the indicated Keithley power level by 12 percent, but additional calibration at 100 Mwt, 3-loop operation is planned to further check the calibration.

B. Power Coefficient of Reactivity at 100 Mwt

Previous testing in February at 20 Mwt resulted in a 3-loop power coefficient of $-0.20\text{¢/Mwt} \pm 25$ percent. The large tolerance resulted from the uncertainty of measurement accuracy because of possible power or flow calibration errors.

At 100 Mwt it was possible to more precisely calibrate power and flow using heat balance data obtainable at this higher power. Correction of the observed power coefficient of -0.162¢/Mwt , for power and flow discrepancies, results in a value of -0.185¢/Mwt , with an accuracy of ± 7 percent.

This value was below the minimum permissible power coefficient of -0.20¢/Mwt originally included in the Technical Specifications. An analysis was made showing that a power coefficient as low as -0.015¢/mw does not introduce any new safety considerations. A request, with supporting data, was made to the AEC, Division of Reactor Licensing, to change the minimum value of the power coefficient to -0.15¢/Mwt . This request was granted on August 16, 1966.

C. Oscillator Tests at 100 Mwt

1. Test Results

Oscillator tests have been run at 100 mwt, with all three coolant loops in operation, on both manual and automatic reactor control with manual feedwater control.

On manual control data were obtained at 19 frequencies between 0.005 and 5.0 cps; on automatic control 12 frequencies were investigated between 0.02 and 5.0 cps. At each frequency the amplitude of the reactor transfer function at 100 mwt has been divided by the amplitude at zero power, and the ratio has been defined as gamma.

Preliminary results on manual control show that the maximum value of gamma observed was 1.003, which is sufficiently low to permit the power to be increased to 133 mw. The results also satisfy one of the criteria for increasing power to 200 mwt - the other must be satisfied by tests at 133 mwt.

Preliminary results with automatic control show that gamma exceeds 1.0 throughout the approximate range of 0.1 to 0.4 cps, with a maximum value of 1.14 at 0.16 cps. Since there is a phase margin for stability of 55 degrees, it is concluded that the reactor is stable under automatic reactor control at 100 mwt and there is no indication that it will be unstable at higher powers.

2. Oscillator Rod

The high power oscillator rod was installed in the core in a safety rod position late in 1965 just prior to the approach-to-power program. It contains seven boron carbide pins with a total peak-to-peak reactivity amplitude of approximately 2¢, which has provided satisfactory data at all power levels. The rod and drive have performed well in the reactor throughout this period.

D. Reactivity Pulse and Noise Analyses at 100 Mwt

These tests have been completed at 100 mwt. The pulse test consists of positive and/or negative ramp changes in reactivity and measurement of the reactor power response. A typical pulse has been a ramp of 2¢ reactivity inserted over periods of 24 to 100 seconds. Results to date at 100 mwt and lower power levels indicate that the transfer function obtained in this manner is in line with that obtained from the oscillator tests.

The noise analysis consists of tape recording the following five input signals to the reactor control system: reactor power, reactor inlet temperature, reactor outlet temperature, core

SECTION III

outlet temperature and primary sodium flow. It is hoped to be able to relate the analysis of these test data to the oscillator tests, but no conclusions have been reached as yet.

SECTION IV

PLANT TEST PROGRAM

A. IHX Thermal Efficiency

The heat transfer coefficient of the three intermediate heat exchangers has been determined by a system heat balance at 100 mwt to correspond to a UA value of 3.6×10^6 Btu/hr-F for loop 1, 3.0 for loop 2 and 3.0 for loop 3, where A is approximately 6600 sq ft.

Based on the design data used for the Hazards Summary Report, these values should be 7.3×10^6 Btu/hr-F. Using an updated U value of 1300 Btu/hr-ft²-F, the best present theoretical UA value for the IHX is 8.7×10^6 Btu/hr-F. This is a theoretical maximum value since it does not include allowances for fouling, flow distribution, etc.

No explanation is offered at this time as to the difference between the values obtained and the theoretical values. The heat balance data are estimated to have an accuracy of ± 5 percent. Possible reasons for the lower coefficient are:

1. Mal-distribution of flow on the primary or shell side.
2. Bypassing the internal seals on either the primary or secondary side of the heat exchanger.
3. Fouling of the heat transfer surfaces.
4. Use, in the calculations, of heat transfer information which may not be applicable.

Plant Test Procedure B.12 will be conducted to learn more about the IHX thermal performances. In this test flows in one loop will be varied over a wide range with only small flow changes in the other two loops. This flow variation will be done in loops 1 and 2 in separate runs.

B. Reactor Vessel Temperatures

Thermocouple measurements during 67 and 100 mwt operation have shown certain areas on the reactor vessel shell where the temperatures between nearby locations exceed predictions. Franklin Institute has been retained by APDA to provide design stress analyses at the following locations: 1) Junction of Transfer Rotor Container and Upper Reactor Vessel, 2) 6" Overflow

Line Nozzle and Upper Reactor Vessel and 3) 30-inch Outlet Nozzles (this is a transient temperature occurring during a scram).

The TRC-URV temperature differential had been 60-70 F during 67 and 100 Mwt operation. This was reduced by one half when the 6-inch throttle valves were opened fully. The predicted temperature rise was 30 F. It is planned to install a thermocouple probe in the transfer rotor container to measure the vertical temperature distribution in the sodium.

The 6-inch nozzle was predicted to be 100 F higher than the vessel temperature, at 100 Mwt it is 155 F higher. However, the thermocouples are on the outside of the leak detector skirt. The actual wall temperature differences are expected to be less. The sodium transient temperature at the outlet nozzles during a 67 Mwt scram was about 40 F/sec during one two-second period. It is felt that the baffling prevented the nozzles from exceeding the 16 F/sec design limit during this brief interval.

C. Reactor Core Outlet Temperatures

Preliminary analysis of data taken during the 100 Mwt operation indicates that the sodium temperature rise through the core is about 7 percent higher than the predicted 125 F. This is the difference between the sodium temperature entering and leaving the core subassemblies, measured at the core hold down mechanism fingers. Possible reasons, not yet verified, are: 1) Too high a proportion of the flow diverted to the blanket subassemblies or 2) excessive clearance between lower support plate and core subassembly inlet nozzles permitting bypass flow around the outside of the subassemblies. 3) Imperfect mixing of coolant in the axial blanket so that the thermocouples are "seeing" the warmer sodium flowing up the center of the subassembly rather than a mixture of this sodium and the cooler sodium flowing up the outer regions.

D. Steam Generator Instability

As mentioned on Page 1 the reactor scrammed automatically twice in July due to steam generator instability. The instability in both cases resulted in extreme oscillations in steam generator sodium outlet temperatures, having a maximum peak-to-peak amplitude of 100 F, which tripped the pumps and initiated the scrams. One scram, from 67 Mwt-2 loops, resulted while decreasing the steam pressure to approach the turbine operating conditions. The other scram, from 100 Mwt-3 loops, was due to a lag in the manual feedwater flow control while going up in power, which caused the sodium temperatures to overshoot. A detailed study of the problem is being made and the results will be reported when available.

SECTION V

OTHER OPERATING EXPERIENCE

A. Loss of 480 V Power Supply

On June 29, 1966, at 6:00 pm, the power supply to the reactor plant 4800/480 volt transformer banks failed as a result of simultaneous grounding of two underground feeds. The 480 volt buses provide the normal supply to the reactor plant except for the main sodium pump motors and sodium heating elements. See the diagram on page 22 for details.

The primary sides of the transformers have ungrounded delta windings, each phase being 2770 volts aboveground. One phase of one of the underground feeds had previously shown sporadic grounding as indicated by the ground lights. This does not result in a current flow to ground or in any way affect system operation, but merely grounds the affected phase while raising the other two phases to 4800 v above ground. On June 29, a ground in one phase of one feed occurred and the resulting voltage boost probably triggered a weak spot in another phase in the other feed, causing it to break down and ground at the same instant, resulting in a phase-to-phase short through ground which opened both 4800 v supply breakers by means of their overcurrent relays and destroyed the cables.

The 250 kw standby diesel generator automatically started, synchronized and carried its assigned loads on the diesel bus. The station battery automatically energized the standby d-c motors and they momentarily carried the a-c generators, which supply the plant essential bus and the safety bus until the normal a-c motor drivers were re-energized by the diesel. The reactor was not in operation at the time. The 4800 v primary pump motors were tripped automatically by loss of power to their liquid rheostat 480 v control circuits.

The cables are Buna-N insulated, metallic sheathed and neoprene jacketed. The feeder for each transformer bank consists of one buried electrical conduit enclosing three separate power cables, one for each phase, each rated at 4800 v.

Partial power was restored in six hours, all 480 v busses were energized (at reduced loading) in twelve hours, and full power was available in twenty-four hours. Both faulted cables were replaced, plus the cables of the other two phases in each feeder.

Principal item of inconvenience was the freezing of the sodium cold trap piping at the Fuel and Repair Building transfer tank. This auxiliary piping is heated by the 480 v system. It was thawed and back in service within 24 hours of the power failure.

To improve the reactor plant power supply reliability, it is planned to make revisions in the 4800 v and 480 v circuits. These will be reported when completed.

B. Electromagnetic Flowmeters

1. EM Flowmeter on Loop 3 Fourteen Inch Reactor Inlet

In the primary sodium system there is one electromagnetic flowmeter in each of the three 14-inch lines supplying the reactor vessel high pressure plenum and one in each of the three 6-inch lines supplying the low pressure plenum. One of these six flowmeters, the 14-inch on Loop 3, has an undesirably wide band in the trace it makes on its recorder chart in the main control room.

Chart 6 on Page 25 shows that this band has a maximum width of 0.8×10^6 pounds sodium flow per hour at a flow of approximately 3.0×10^6 lb/hr. Average band width is $0.3 - 0.4 \times 10^6$ lb/hr for this instrument at this flow. The band width decreases with decreasing flow. The five other flowmeters do not have this wide fluctuation in their recorder traces. (See Chart 2 and Chart 4, Pages 23 and 24, and the 6-inch flowmeter on Chart 6.)

Tests were run which proved that the electrical signals were not being distorted in the circuit between the Loop 3, 14-inch sodium flow sensing element and the recorder. It was also established that the recorder was not at fault. Attempts were made to improve the flowmeter's output signal by lengthening and broadening the pole shoes of the permanent magnet. This had no effect.

The cause of the wide band on the flowmeter appears to be sodium flow turbulence in the 14-inch line, caused by locating this flowmeter too close downstream from the branch point off the 16-inch sodium return line, as shown on the sketch on Page 26. Commonly accepted engineering practice is to, where possible, locate flowmeters (any type) at least 15 pipe diameters downstream of any change in flow direction, but in this case the distance is only eight diameters. The piping layouts show that the Loop 3, 14-inch flowmeter has a flow inlet configuration which is much less desirable than any of the other flowmeters. This was recognized early in the plant design but other considerations forced the existing arrangement. The wide band, though undesirable, is acceptable.

2. Other Flowmeter Information

Electromagnetic flowmeters were chosen because, despite their accuracy limitations, they were commonly accepted as the best means of measuring sodium flow.

Each of the three secondary sodium loops also has an EM flowmeter. These are in 18" lines. There is no problem with the flow signals from these flowmeters.

All the main loop EM flowmeters had been calibrated from the sodium pump predicted performance curves. The reactor blanket EM flowmeters were calibrated from the pump and throttle valve curves. Recent calculations made from heat balance data at 100 mwt have led to the conclusion that the primary sodium flow is actually about 10 percent lower than the EM flowmeter average readings. The secondary sodium flow and the feedwater flow have on the average less than 1 percent error; however, a short while ago the secondary sodium flowmeter readings had been recalibrated by a 20 percent reduction in the indicated reading.

FERMI REACTOR PLANT POWER SUPPLY

SIMPLIFIED ONE LINE DIAGRAM

LOOP 3

PRIMARY SODIUM PIPING

PUMP TO REACTOR

SECTION VI

MAINTENANCE AND ADDITIONS

A. Recirculating Gas Compressors

Three diaphragm-type gas compressors, made in France by Corblin, recirculate the argon cover gas over the primary sodium system. Purpose of the recirculation is to store the gas when there is an excess and return it as needed, thus conserving argon consumption. Flexible, oil-pulsed metal diaphragms are the compressing element - they are leak-tight and expose the gas to no lubricants - see sketch next page. Each compressor can handle 25 cfm at 10 psia inlet and 40 psig outlet - too great a range for centrifugal compressors.

Although there have been maintenance problems with the three Corblin compressors in the past, these problems have been overcome and maintenance in the future should be purely a once-a-year routine inspection. There have been two diaphragm failures (cracks) on the compressors, but these were due to foreign materials entering the diaphragm gas space. Pre-filters have been installed to eliminate this problem. The oil detectors downstream of the compressors failed to operate when oil leaked into the discharge line through the cracked diaphragms. The addition of a highly volatile material to the oil now permits these detectors to function in case of a future break.

The compressors have each logged 750 hours total operating time to date. The diaphragms are rated for 2000 hours service. The principal cause of compressor operation is argon buffer seal inleakage₃ to the primary system. PRDC estimates this leakage at 300 ft³ per day.

Considerable difficulty had been encountered with the compressors losing their oil prime during extended shutdown periods. This problem was particularly acute since the operation of the compressors is rather infrequent during normal plant operation. Recently this problem has been overcome by the addition of automatic vent valves which open for 15 seconds at the starting of each compressor cycle. These valves automatically vent any air which has collected at the top of the oil side of the diaphragm and return it to the crank-case.

B. Draining Liquid Waste Tanks

Over a period of years the seven large liquid waste storage tanks in the Fuel and Repair Building had developed an accumulation of sludge in the bottom. The sludge is thought to have developed out of materials discharged from the Health Physics Building. The tanks also contained some radioactive tantalum. These particles were from the cladding of the neutron sources when they were transferred by the steam cleaning machine.

Tests showed that at times during draining lumps of the sludge lodged at the tank drain line radiation monitor and the radioactivity exceeded the monitor set point, stopping the draining operation. A temporary gravel, sand and charcoal filter bed has been installed to remove the sludge and tantalum and decrease the radioactivity before the effluent is released from the plant. It is working effectively. As a further check, the filtered effluent is gathered in a spare tank where it is sampled before final release. Draining has been completed on all but one tank. The filter will be stored and later shipped to a disposal facility.

C. Improving Clarity of Pool Water

Although the decay and cut-up pools have good underwater lighting, good viewing of objects at the bottom had become increasingly difficult. This was due in part to a gradual increase in finely suspended particulates and also to the small amount of lithium chromate in solution in the pools which changed the water from colorless to a definite yellowish hue. The color inherently cut down the light transmission.

The lithium chromate had been in the pools since the initial fill some five years ago. Its chemical properties were utilized to inhibit rusting of the nitrided surfaces of the core and inner blanket subassemblies in the demineralized pool water. Tests showed that the rusting was not as great a problem as originally assumed, allowing elimination of the lithium chromate.

It was determined that the suspended solids in the pool averaged only 1/2 micron in size. A special, fine pore cellulose filter was installed in the pool system and performed effectively. Tantalum activity from the cladding of the neutron sources formerly stored in the pool was reduced to one fifth the value before filtration. Removal of the particles will also decrease the tendency for algae or other bacteria to form in the pool, since they tend to cling to the particles. The filter will be left in place and the water filtered periodically.

Filtering and lithium chromate removal have improved the clarity of the water to permit good viewing into the pool.

D. Steam Dump Lines

The capacity of the steam dump line to the condenser was increased to improve the flexibility of operation of the reactor. The addition of a new 14-inch line in parallel with the existing 14-inch dump line will permit 200 Mwt operation of the reactor without passing any steam through the turbine, if plant conditions make this desirable. The condenser is capable of handling this reactor plant steam through the No. 4 (deaerating) feedwater heater, plus the full steam output of the peaking boiler from the turbine exhaust.

In addition to increasing the capacity of the dump lines, which convey steam from the No. 4 heater to the condenser, it is also necessary when dumping all steam at 67 Mwt or higher to augment the capacity of the main steam bypass line which connects the No. 4 heaters to the three steam generators. This is accomplished by routing part of the steam through a portion of the main steam line.

Enrico Fermi

KEY

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