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COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT
ENRICO FERMI ATOMIC POWER PLANT

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

Approved

Charles E. Branyan
Fermi Project Engineer
APDA

Author: J. F. McCarthy
Commonwealth Associates Inc.
Jackson, Michigan

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PREFACE

PURPOSE

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

SCOPE

The scope of this report includes all phases of current nuclear operating and maintenance experience at the Enrico Fermi Atomic Power Plant.

Earlier Fermi experience in certain selected areas is being recorded in a series of technical reports completed or in preparation by Atomic Power Development Associates Inc. for the US Atomic Energy Commission under AEC Contract No. AT (11-1) -865, Project Agreement 15. This series of reports provides detailed information on the nuclear testing, 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, 1966, but will also include selected 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.

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

On the fifth day of October the reactor was manually scrammed from 29 Mwt, during an approach to power, when the staff member present in the control room noted several significant abnormalities. The reactor control rods appeared to be withdrawn unusually far, two core subassembly outlet thermocouples were reading higher than normal, alarms were received from the radiation detectors which monitor the reactor building ventilation exhaust outlets - automatically isolating the building, and the reactor cover gas fission product detector indicated a sharp rise in count rate.

The reactor had been placed in operation the previous day at a power of less than 1 Mwt and maintained overnight at this low power to help raise the primary sodium temperature to the normal value of 517 F for power start-up. There had been no prior reactor operation in October.

It has been concluded that one or more core sub-assemblies were damaged by overheating, resulting in some fuel melting with attendant release of fission products. The reason for the overheating is not known at this time, nor is there as yet a prime suspect as to the cause. The reactor will not be operated at a power level greater than 1 Kwt until the extent of the damage is determined and the condition corrected.

The gaseous fission products entered the primary system cover gas and were released into the reactor building through very small and ordinarily insignificant leaks in the primary cover gas system. The aforementioned automatic isolation prevented the building ventilation system from exhausting any significant amount of radioactivity to the waste gas stack. Other area radiation monitors within the building confirmed the fission product release. However, the activity levels were not dangerously high and no significant plant personnel exposure occurred.

The building was entered by plant personnel two days after the incident. Analysis of samples taken inside the building showed little or no surface contamination. Controlled release of the reactor building atmosphere to the waste gas stack was started and access restrictions were removed the following week.

Section I

Various samples of the primary system sodium and argon cover gas were taken and analyzed for fission product activity. In most cases there was uncertainty as to whether a representative sample of the system had been obtained. After two weeks, 3/4 of the primary cover gas fission product activity had been transferred to waste gas storage tank No. 1. Best estimates as to the extent of the fission product release are felt to have been obtained by an absolute activity analysis at Argonne National Laboratory of a sample from this tank. Based on Kr-85 analysis, the data indicate complete melting of one-half of one average core subassembly, assuming 100 percent transfer of the Kr-85 from this subassembly into the primary argon cover gas. If a lesser percent of the Kr-85 gas is assumed to have been released, as many as two subassemblies may have completely failed, or fractions of a correspondingly larger number.

An analysis has been made of the change in the reactivity of the core. A check made on October 7, with the safety rods withdrawn to their first subcritical stop showed a 22 cent loss of reactivity, plus or minus 2 cents. This agrees closely with the loss determined by analysis of the reactor power versus control rod positions during the October 5th operation.

Tests were conducted on October 21, 1966, to determine if there were any measurable changes in the primary sodium flow characteristics, or unusual sounds issuing from the reactor. It was concluded that no significant flow changes exist, nor were any abnormal audio signals detected.

At the end of the month, preparations were being completed for raising the core hold-down mechanism and planning was underway for use of the sweep mechanism and offset handling mechanism for probing the subassemblies. Planning is in the early stages for removal and inspection of the subassemblies.

See Section III for further details regarding the reactor core damage.

On the weekend of October 1-3, the annual leak rate test of the reactor building was successfully conducted as described in Section VI.

During the month a proof test run was started on the steam cleaning machine - see Section II A.4.

Section I

Safety rod No. 89, which had been removed in September from core position P02-P02, was removed from the reactor and visually examined at the cask car glove box in the Fuel and Repair Building. Nothing abnormal was noted. See Section II B.2.

SECTION II
DATA AND STATISTICS

A. Plant Operations

1. <u>Reactor Operating Statistics</u>	<u>October</u>	<u>Cumulative through October</u>
Reactor Start-ups	1	406
Scheduled Scrams (Includes usual test scram prior to reactor start-up)	1	718
Unscheduled Scrams	1**	56
Maximum Reactor Power, Mwt	34*	114*
Integrated Reactor Power, Mwt - days	1*	855*
Hours of Reactor Operation	17	2096
Maximum Burn-up 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 approximately 10 percent lower, based on heat balance data. Further data to be taken and analyzed before final correction values are established.

** Reactor manually scrammed from 29 Mwt after several abnormal operating conditions were noted. See page 22.

SECTION II
DATA AND STATISTICS

A. Plant Operations

2. Daily Operation Curve Sheets

See the following two fold-out pages. The reactor produced no steam during October and reactor power above 1 Mwt occurred only briefly on October 5th, when there was a scram from 29 Mwt during an approach to a planned 67 Mwt operation, see page 22. 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 OCT 1966

FERMI DAILY REACTOR PLANT OPERATION DURING THE MONTH OF OCT 1966

SECTION II
DATA AND STATISTICS

A. Plant Operations

3. Reactor Loading

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

The diagram on the following page shows the loading existing from September 27, 1966, through the month of October.

N16 N15 N14 N13 N12 N11 N10 N09 N08 N07 N06 N05 N04 N03 N02 N01 N00 P01 P02 P03 P04 P05 P06 P07 P08 P09 P10 P11 P12 P13 P14 P15 P16

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" 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)
- Stringer 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

REACTOR LOADING 9-27-66

SECTION II

DATA AND STATISTICS

A. Plant Operations

4. Fuel Handling Operations

No reactor subassemblies were moved during October. However, acceptance testing of the steam cleaning machine in the Fuel and Repair Building was started, using dummy subassemblies. The preoperational test and demonstration program for the steam cleaning machine includes cleaning the sodium from twenty subassemblies to demonstrate that the machine is capable of handling irradiated subassemblies. This demonstration is in support of the license application for the use of this facility for irradiated fuel.

The test subassemblies are removed from the sodium-filled transfer tank, lifted into the cleaning chamber, cleaned with argon and steam, rinsed with water, dried by means of evacuating and purging with argon, and lowered out of the cleaning chamber for inspection.

After seven cleaning cycles, it was observed that the gripper of the hoist, when lowered into the transfer tank, failed to engage the handling head on the next subassembly. At present the gripper has been removed from the machine for inspection of its internals.

5. Sodium and Gas Systems Performance

a. Sodium Cold Traps and Plugging Indicators

<u>October Operating Data</u>	<u>Primary System</u>	<u>Secondary System</u>		
		<u>Loop 1</u>	<u>Loop 2</u>	<u>Loop 3</u>
Cold Trap Operation (hours)	16	16	39	34
Maximum Plugging Temperature - F	220	295	360	285 below
Minimum Plugging Temperature - F	220	230	220	220

b. Sodium Activity - Measured values for October.

Primary System	41 microcuries/cc Na (10/6/66)*
Secondary System	Below instrument min sensitivity of 2.5×10^{-4} microcuries/cc Na

c. Primary System Cover Gas Chemical Analysis

The reactor cover gas was too radioactive for chemical analysis using the normal facilities. The primary shield tank atmosphere was not analyzed because of more urgent analysis work.

d. Plant Inert Gas Consumption - October 1966

	<u>Argon</u> <u>Cu Ft</u>	<u>Nitrogen</u> <u>Cu Ft</u>
Reactor Building and Primary System	15,000	310,000
Secondary System	23,000	103,000
Other	97,000	207,000
Total	135,000	620,000

e. Primary Sodium Chemical Analysis

Sampling Dates	October 3	October 6
Oxygen	6,6,7,7,7	8,8,8,9,
Carbon	37,53,75,99	74,75,81,93,
**Nonhydroxide Hydrogen	0.1, 0.1	0.2, 0.3
**Hydroxide Hydrogen	1.2, 1.3	1.2, 1.3
Nickel	Not Measured	0.2, 0.3
Iron	Not Measured	below 0.2
Chromium	Not Measured	below 0.2

** 1.3 ppm is recommended maximum total hydrogen value.

Note: Values shown are for one sample analyzed at several points, in ppm by weight.

* Gross Beta Activity. Unusually high due to the presence of fission products.

f. Primary Sodium System

Subsequent to the fission product release on October 5, the decision was made not to commence cold trapping of the primary sodium at that time. The cold trap had been in operation at the time of the reactor scram on October 5, but circulation was stopped about two hours after the scram. On October 6, the primary system plugging temperature was less than 220 F (very good). It was necessary to operate the cold trap for eight hours on two occasions after October 5th to obtain representative samples of the reactor primary sodium.

The primary purpose in cold trapping would be to attempt to remove the solid fission products from the sodium and to getter those fission products which tend to plate out in the primary system. The decision not to cold trap was based on doubts as to its effectiveness in removing fission products, on the fact that there seemed to be no urgency to begin this operation, and on the desire not to make the cold trap any more radioactive than it was since it may be necessary to gain access to the cold trap cell.

Several samples were taken of the primary system sodium. Results of the chemical analysis of two of the samples are given on page 15. On another sample which was analyzed for fission product activity, considerable plating out of the fission products onto the sample tube wall was observed. However, this effect was reduced in later samples. Because of the unknown amount of fission products plated out in the primary sodium system the sodium activity did not provide an accurate means for determining the total amount of fission products released. In the first sample the predominant fission product activity was found to be due to radionuclides of barium and lanthanum, which are very soluble in sodium. The sample coil removed on October 10 had a radiation level of 100 mr/hr at one inch on that date.

g. Primary Cover Gas Activity

The following tabulation gives the cover gas activity on October 14 and 17, at various locations in the recirculating gas system and in the waste gas storage tank. The reported activity was obtained by a vibrating reed electrometer measurement of the sample flask. The electrometer was calibrated by measuring a standard flask filled with argon containing a known amount of Kr-85. The measurement includes uncertainties as to whether the sample is representative, whether the calibration of the electrometer is correct and the level of contribution from isotopes other than Xe-133, such as Xe-131 and Kr-85.

	<u>Xe-133 Activity</u> <u>(Microcuries/cc)</u>		<u>Vol at</u> <u>Sample</u>	<u>Volume</u> <u>Corrected</u>	<u>Total</u> <u>Activity</u>
	<u>10/14</u>	<u>10/17</u>	<u>Location</u>	<u>For Press</u>	<u>10/17</u>
<u>Location</u>			<u>(Cu Ft)</u>	<u>(Cu Ft)</u>	<u>(Curies)</u>
No. 1 Pump Tank	0.8	0.31	50	50	0.22
No. 3 Pump Tank	0.42	0.28	50	50	0.19
Reactor Vessel	0.45	0.10	50	50	0.07
Overflow Tank	0.42	0.27	550	550	2.1
Holdup and Vacuum Tanks	-	0.44	1,000	1,400	8.5
No. 1 Waste Tank	-	0.41	2,500	8,160	34
					45.1

Two one-liter gas samples from waste gas Tank No. 1 were sent to Argonne National Laboratory for analysis. They reported a Kr-85 density of 5×10^{10} atoms/cc of argon. Since Kr-85 is a long lived isotope (10 year half-life) its buildup has been essentially linear with accumulation of burnup. It was calculated based on the fast fission yield of U-235 that an average pin at the current burnup level should contain 3.4×10^{17} atoms of Kr-85. Five $\times 10^{10}$ atoms/cc in 1.36×10^4 scf in the waste tank and 3.4×10^{17} atoms/pin gives 71 average pins melted if all the Kr-85 is released and 80 percent of it is in the waste tank. Even if partial release of Kr-85 is assumed it means that only the pins in the equivalent of 1 to 2 subassemblies have melted. Reference literature data show up to 100 percent release of noble gasses in less than 10 minutes if burnup has reached 400-500 Mwd/tonne (burnup in the affected area is approximately 400 Mwd/tonne).

SECTION II

DATA AND STATISTICS

B. Plant Repairs and Modifications

1. Primary and Secondary Sodium Pumps

There were no significant items of maintenance on these pumps during October.

2. Reactor Safety Systems

Safety rod No. 89 which had been removed from the reactor and placed in the transfer rotor container, as noted in the September report, was moved to the Fuel and Repair Building by the cask car. This safety rod had at times shown interference when being engaged by its extension. It was let down out of the cask car into the cask car gripper glove box and examined from a distance of six inches by direct vision through a viewing window. Its entire length was inspected and there was no evidence of nicks, deposits, misalignment or wrapper tube bowing. It appeared that the safety rod was in as good condition as when originally put into the reactor and could be reinstalled in the reactor, although this is not immediately planned.

The rod appeared to be approximately 90 percent wetted with sodium. The radiation field at the viewing station was 12 R/hr six inches from the rod.

After the reactor core damage on October 5, each safety rod was raised individually and scrambled by dropping from the full out position with the rest of the safety rods remaining fully inserted in the core. All rods functioned normally.

3. Instrumentation

As noted in the September report, the fission product detector had been adjusted at low reactor power and was to be checked during October at higher power levels. Concurrent with the reactor operation on October 5, 1966, a test was underway to calibrate the fission product detector and to evaluate its operation. At approximately 1509, a high radiation alarm from the monitor in the fission product detector building sounded and the fission product detector system was isolated. The following is a summary of the data from this instrument.

Between the hours of 0730 and 0943 when the reactor was operated at 1 Mwt, the fission product detector was reading between 250 and 400 cpm with perhaps a slight rising trend. Soak time on the tape was set at 3 minutes. At 0943, the tape advance was stopped for 43 minutes to observe the decay of activity on the tape. Following this, the tape was advanced, once, placing the section of tape which had soaked in the precipitator to a position opposite the detector. The count rate was recorded at 1580 cpm. Immediately following this, the tape was given a double advance to measure a zero soak time section. This was done at 1036 hours and the count rate was observed to be 200 cpm. At 1043 hours the normal 3 minute soak time was resumed and the count rate was 390 cpm, increasing to about 500 cpm (at about 1120 hours) which was the full scale reading. At 1213 hours the next higher scale was selected and the count rate was still at 500 cpm following which it slowly increased, reaching 550 cpm at 1346 hours. All of these readings were taken with the reactor at 1 Mwt. They may be compared with readings taken at the same calibration settings on September 29, 1966, as follows:

Time: 1100 hours, 9/29/66
Reactor Power: 0.24 Mwt
Count Rate: 50 cpm

Time: 1200 - 1300 hours, 9/29/66
Reactor Power: 60 - 70 watts
Count Rate: 50.5 cpm

Time: 1610 hours, 9/29/66
Reactor Power: 0.87 Mwt
Count Rate: 190 cpm

At 1347 hours the person doing the calibration was informed that the reactor power was being increased and the chart was marked accordingly. No increase in count rate was observed on the fission product detector until 1359 hours at which time it increased from the 550 cpm reading to a maximum of 790 cpm which occurred at 1403 hours. Following this, the count rate decreased and settled out at approximately 650 cpm (apparently corresponding to the return of reactor power to 2 Mwt).

Between the hours 1414 and 1451, the count rate increased from 680 to 1470 as reactor power was increased. During this time the chart was marked at a reported reactor power of 14 Mwt. The time of notation was 1444 hours and the count rate was 1350 cpm. At 1451 hours, the normal tape advance was

interrupted and a double tape advance was made to obtain a zero soak time measurement. The count rate was 750 cpm increasing to 1000 cpm during the seven minute observation. The fact that this count rate increased appeared to be somewhat unusual. However, the next reading which was taken on a section of tape which had soaked for seven minutes appeared as expected at 3030 cpm or slightly over twice the count rate of a three minute soak specimen. This measurement was taken starting at 1458 and ending at 1501 hours.

The normal three minute advance was resumed at 1502 hours and the first tape specimen at this time was found to be off scale on the 5000 cpm range. The 50,000 cpm scale and the 500,000 cpm scale were selected successively with full scale observation on each. A gain check was made to verify that the instrument was performing properly. The gain check indicated no fault in the channel.

At 1642 hours, following the reactor scram at 1520, the discriminator setting was increased sufficiently to permit an on-scale record to be made of the radiation level. The level was found to be increasing and finally reached a peak at 1651 hours, after which it gradually decreased. No count rate measurement is given because the instrument was in an uncalibrated state and could serve only as a trend monitor.

Between 1710 and 1719 hours, a differential discriminator curve was made to determine whether energy peaks could be found. One large peak was observed at a discriminator setting of 970 on a 0 to 1000 scale.

A count rate scale multiplier with a factor of twenty was actuated at 1728 hours (this multiplier had not been used prior to this time). The discriminator was returned to normal and the count rate was measured at this time at 1,240,000 cpm. The following count rates and times were measured:

2034 hours:	208,000 cpm
2241 hours:	113,000 cpm
1255 hours:	15,100 cpm
(on 10/6/66)	

4. Fuel Handling Equipment

The fuel transfer cask car was disassembled during October for locating and repairing leaks in the argon gas system.

The several compartments were isolated and tested individually. Leaks were found in the ball valve, in the liner plate, in the north argon blower shaft sleeve, in the damper operator shaft bellows, in the heat exchanger bellows and in a thermocouple penetration. After the leaks are repaired, the car will be reassembled and given an overall leak rate check and a mechanical check-out.

5. Steam Generators

There were no significant items of maintenance on any of the steam generators during October. Normal operation was experienced during the operation to 34 Mwt on October 5th.

SECTION III

NUCLEAR TEST PROGRAM

Fuel Damage on October 5, 1966

A. Reactor Operation

On Tuesday afternoon, October 4, 1966, two of the steam generators were filled with water at a system sodium temperature of approximately 460 F and the reactor was brought critical, starting at 2000 hours. The reactivity increment between the first and second stops and between the second stop and full out positions of the safety rods appeared to give a very good critical rod prediction. Upon withdrawal of the No. 6 safety rod, the rod extension drive motor apparently stopped occasionally because of a loose electrical connection and this resulted in a longer than normal time for the rod to be withdrawn.

The reactor was operated throughout the night with the three primary pumps in operation, and the reactor power was kept below 1 Mwt. The energy input to the sodium from the pumps resulted in a rise of system temperature at a rate of about 12 degrees an hour. At 0345, Wednesday morning, a check was made of the correlation between the rod positions and the previously measured reactivity corresponding to the sodium temperature. There was good correlation between these two values.

At 0800 on Wednesday, October 5, the No. 2 primary pump was shut off and the other two pumps were each operated at a flow rate of about 3.5×10^6 lb/hr. The reactor continued to operate at 1 Mwt; during this time difficulty was experienced with valve FCV-608-1. This valve is an emergency steam dump valve; it was found that an air leak had developed in the control system to the valve, preventing it from functioning properly. After repairs, the valve was put back in service but the reactor was held at 1 Mwt while the valve was not operating properly.

At 1345 on Wednesday, October 5, an increase in the reactor power level was begun. When the power reached 5 Mwt at 1400 hours, it was found that the east boiler feed pump was not being properly controlled by the control system; therefore, the power was reduced to 2 Mwt and a switch was made to the center boiler feed pump. At about 1420, the rise in power was once again started and continued up to 8 Mwt where there was a brief hold, then the increase continued to 13 Mwt at 1445 when another brief hold took place to get the center boiler feed pump on automatic control. The increase in power was resumed

and, at about 1500, the reactor operator observed variations in the automatic control system wherein the dn/dt signal became erratic. This problem had been experienced often in the past and was felt to be noise pickup in the control system. The reactor was put on manual control and the dn/dt meter observed until the apparent noise disappeared. There were no indications of instability or of a nuclear transient. The reactor was once again put on automatic control and the increase in power resumed. At approximately 1505, the feedwater control was put on automatic. The reactor operator once again observed variations on the dn/dt signal.

At this time it was noted by the staff member present that the control rods were at a higher than normal position for this reactor power. The shim and regulating rods were observed to be at a withdrawn elevation of approximately nine inches each. It has since been determined that their combined excess withdrawal, after correcting for inlet temperature and power level, was approximately six inches, or three inches on each rod. A check of the core outlet temperatures showed that the sodium outlet temperatures from subassemblies M-140 and M-098 were running abnormally high at 715 F and 695 F, with a bulk reactor outlet temperature of about 600 F. Reactor power at this time was 34 Mwt as measured by the Keithley micromicroammeter. At 1509 hours there were alarms from the radiation monitors in the upper reactor building ventilation exhaust ducts. The building was automatically isolated; no one was inside at this time. A Class I radiation emergency was announced. This is the class of radiation emergency which is lowest in severity, being restricted to a specific locality of the plant. The detector in the fission product detector building also exceeded its set point, isolating the fission product detector system. Reactor power was manually reduced over the next few minutes but no improvement in any of the readings was noticed. At 1520 hours the reactor was manually scrammed from 29 Mwt and the Keithley recording showed that all six safety rods dropped promptly.

B. Radiation Measurements

1. The table on page 25 lists the readings recorded by the permanently installed radiation detectors in the reactor building, in the building ventilation exhaust ducts, in the waste gas stack and in the fission product detector building. Readings were taken at the time of the emergency, for several hours following and the next morning.

2. Immediately after the emergency was announced, Health Physics personnel obtained the following data:

a. At the 3 inch exhaust line from the machinery dome external to the reactor building, the direct radiation level was 0.02 to 0.05 mr/hr at 1 cm.

b. At the 6 inch exhaust line from the upper reactor building, external to the reactor building the direct radiation level was 0.08 to 0.12 mr/hr at 1 cm.

c. Measurements at the reactor building exterior indicated direct radiation levels of 2 to 3 mr/hr at 1 cm, except opposite the No. 2 primary sodium pump where the reading was 9 mr/hr. This was the pump that was not in operation and it is postulated that its gas seal, not being oil-flooded as when in operation, may have permitted more leakage than the seals on the other two operating pumps.

3. The following are the results of radiation surveys taken between 1545 and 1600 hours.

East side of the reactor building, 0.4-0.5 mr/hr
West side of the reactor building, up to 1 mr/hr
Near water treatment plant, 0.2-0.09 mt/hr.
South of water plant, 0.02-0.05 mr/hr
Top of stairs to fuel & repair building, 0.05 mr/hr.

4. At 1650 hours a primary cover gas sample was taken through the hydrocarbon analyzer taps downstream of the Corblin compressors, and was analyzed using a 250 cc chamber. The activity level at the exterior of the chamber was 1 mr/hr at 1 cm. The gas was analyzed on a 1024 channel gamma ray spectrometer and a number of gamma peaks indicative of fission products were identified.

5. At approximately 1530 hours the activity at the steel lid on the manhole (shield plug removed) of the fission product detector building was above 200 mr/hr at 1 cm. At 0845 on October 6, the activity at this location was 1.5 mr/hr and the activity at the reactor building exterior was 0.07 to 0.3 mr/hr. In the early afternoon, smear samples were taken on top of the fission product detector building and from the reactor building exhaust blower. These showed beta counts of 13 cpm and 108 cpm respectively, with no alpha activity. At this same time samples were also taken and analyzed from the above floor region of the reactor building and from the primary cover gas system through the fission product detector lines. The activity for the upper reactor building was 5.36×10^{-6} microcuries/cc based on Xe-133 in air, and the cover gas activity was 20 microcuries/cc based on Xe-133 in argon. In both cases the spectrometer indicated Xe-133 and Xe-135 to be the major contributors.

6. See page 19 for information on the fission product detector readings, and page 16 for further information concerning the cover gas fission product activity.

TABULATION OF RADIATION DETECTOR READINGS

Radiation Detector	Location	Alarm Setting	Time of Emergency	October 5, 1966 Time				Oct 6, 1966
				1607	1727	1738	1828	Time 0855 am
RE-706	6-inch exhaust pipe from above floor re- actor bldg	200	10^4			$2.3-2.5 \times 10^4$		200
RE-706A	Common 6-inch exhaust pipe from above floor and machinery dome	400			1.9×10^3 (from chart)		900	900
RE-707	3-inch exhaust pipe from machinery dome	270	2×10^3			$7-8 \times 10^3$		10^3
RE-708	Near personnel door in reactor bldg	1.5 mr/hr	10 mr/hr at 3:40 pm		5-6 mr/hr		3 mr/hr	1 mr/hr
RE-709	Near personnel door in reactor bldg	580	5×10^3	1.5×10^3	10^4		6×10^3	2×10^3
RE-710	Near personnel door in reactor bldg	2400		2.9×10^4	5.5×10^5		8×10^5	1×10^6
RE-710D	Near personnel door in reactor bldg	2400		10	40		120	800-900
RE-745-1	FARB sample in stack	740	5×10^2					
RE-746-2	FARB sample in stack	Not Available	3×10^2					
RE-760	Samples fission pro- ducts detector bldg atmosphere	400				500-700		200

Note: Values are in counts per minute except where otherwise noted.

C. Core Reactivity

A subcritical reactivity check made on October 7, showed a loss of approximately 22 cents from the October 4 value. An analysis of the October 5 data on control rod position compared to reactor power showed a reactivity loss of approximately 26 cents. Curve sheets on pages 28, 29 and 30 present the data on the power, rod position and reactivity. The greater than normal reactivity loss appears to have begun at a reactor power of about 10-15 Mwt. Melting at 10-15 Mwt requires essentially a fully blocked subassembly.

Calculations indicate that a completely blocked subassembly would melt near its center first, progressing both radially and axially as the power is gradually increased. Fuel motion would be mostly radial allowing a large fraction of the fuel to melt with little effect on reactivity. With complete subassembly flow blockage and 2 loops operating at a reactor power of 34 Mwt, it is predicted that 93 percent of the fuel in a central subassembly would melt, varying from 33 percent at the bottom to 100 percent in the center to 85 percent at the top. The can wall at the center could reach eutectic formation temperatures and thus might be penetrated. An average subassembly would be 75 percent melted, about 90 percent melted in the center plane. An edge subassembly would be 29 percent melted, about 45 percent at the center.

The following model was used to evaluate the safety of the present core configuration. A reasonable reassembly model to postulate the maximum potential reactivity gain subsequent to and as a result of a meltdown, would assume that normal (40 percent) average fuel density exists at the subassembly vertical center, though all the fuel is near the wall. Then the upper and lower volumes of crust would be jarred loose in pieces, filling the center void space. Because of the irregularity of the pieces it can be assumed that the maximum density is 60 percent in the space now occupied by the crust pieces. In a center subassembly, 40 percent of the volume near the vertical center is occupied (before the crust shifts) by what was melted fuel, and this increases to 76 percent when the other 60 percent is filled by 60 percent dense material. Space occupied by clad is ignored so that the 76 percent maximum density is an overestimate of the extent of compaction.

The reactivity values associated with the above are:

<u>Collapsed Density</u>	<u>Reactivity Gain For Number of Central Subassemblies Involved</u>			
	<u>3</u>	<u>7</u>	<u>14</u>	<u>21</u>
100%	\$0.8	\$1.6	\$2.9	\$4.2
66%	0.7	1.4	2.5	3.6
50%	0.6	1.1	1.9	2.7

It can be seen that even if as many as 21 central subassemblies are involved and the postulated subsequent collapse results in 100 percent central density, safety rod shutdown reactivity (\$7.8) cannot be overridden.

It was therefore concluded that the core is at present safe and no reasonable event could result in criticality. Nevertheless, reactivity has and will be watched closely. A daily check is made of the source range count rate, using the high sensitivity neutron detector rack No. 4 in parallel with rack No. 3.

REACTIVITY VARIATION ON OCTOBER 5, 1966

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REACTOR POWER AND ROD POSITIONS ON OCTOBER 5, 1966

APPARENT REACTIVITY LOSS ON OCTOBER 5, 1966

D. Flow Tests

On October 21, primary sodium flow tests were conducted with the reactor shut down, to determine a head-flow characteristic curve for the primary system. The purpose was to determine if any change had occurred in the system pressure drop characteristic, which would indicate that considerable plugging of core subassemblies had occurred. The results showed good agreement with the system characteristics previously measured on April 7, 1966, at the same pump speeds - 300, 420, 500 and 550 rpm. There was no significant change in core pressure drop after taking into account the tolerances on the instrumentation accuracy. The test accuracy indicates that no more than 4 to 6 subassemblies could be completely plugged. The system head data were obtained from permanently installed pressure detectors at the pump outlets, while flows were obtained from permanently-installed electromagnetic flowmeters in the 6-inch and 14-inch reactor inlet lines.

After obtaining the abovementioned head-flow data, the pump speeds were reduced to 300 rpm and all three pumps tripped simultaneously to observe the coast-down performance. The No. 3 pump flow vs time curve had a small change in slope at about 50 percent flow. However, this pump and the others had previously shown a similar characteristic. The pony motors were then energized in sequence, followed by the main motors. A base reading on the pressure and flow instruments was recorded at the zero flow to assist in their calibration.

During the tests the source range count rate levels were followed closely to detect possible core reactivity changes. When at 550 rpm pump speed, there was an alarm when the source range period decreased to 9 seconds, occurring as the source range level recorder ended a standardization point. Similar short periods had occurred in the past. There were no other apparent changes in core reactivity during the course of tests, as measured by the other plant nuclear instruments.

Primary cover gas fission product detector alarms were received at 550 and 400 rpm pump speeds. The alarms were caused by hot spots on the continuous tape utilized in the detector. The spots were probably caused by fission products remaining on the tape from the October 5th release of fission products into the cover gas. These points on the tape travelled around and coincidentally arrived at the sensor during the pump tests.

The following additional data were also taken during the tests: pump motor currents, sodium levels, cover gas pressure, core outlet sodium temperatures, sodium temperatures entering and leaving the reactor, sodium pressure at the blanket inlet plenum and acoustical monitoring and recording. Playback of the sound tape from a pickup at the No. 1 safety rod extension revealed a clacking or rapping noise during pump coastdown. The frequency of the sound decreased and stopped as the flow came to zero. The source of the noise appears to be at or near the pumps with some evidence that bearings or seals are responsible.

E. Subassembly Outlet Thermocouple Reading

Core subassemblies in the Fermi reactor are not orificed and, therefore, the relative radial temperature distribution in the core is determined entirely by the neutron flux distribution. Under the two loop sodium flow conditions of 6.7×10^6 lbs per hour which existed on October 5 and with the reactor operating at 34 Mwt, the calculated sodium temperature rise in a central core subassembly was 91 F, and in an edge core subassembly was 44 F. The inlet temperature at that time was 535 F.

Iron-constantan thermocouples are installed in the holddown mechanism of the reactor to read the exit temperature from various core and blanket components. The location by subassembly position and the number of operable thermocouples at each given position, is shown in the diagram on page 35. Thermocouples installed above core or inner radial blanket subassemblies are located in the holddown fingers. Thermocouples installed above control or safety rods are located in the upper guide tubes. The method of installation is shown in the sketch on page 36.

Thermocouples to measure outer radial blanket exit temperatures are installed along two of the spider legs which locate and support the holddown mechanism.

On October 5, 1966, a total of 71 thermocouples were operable, consisting of:

- 19 core and IRB positions each fitted with a single thermocouple.

- 13 core and IRB positions each fitted with a pair of thermocouples.

- 4 control and safety rod positions each fitted with a pair of thermocouples.

- 5 control and safety rod positions each fitted with a single thermocouple.

- 13 ORB positions each fitted with a single thermocouple.

All control and safety rod positions, 23 percent of the core and IRB and 2 percent of the outer radial blanket positions are equipped with at least one thermocouple. During operation, four of the core subassembly thermocouples are used as input to the safety system, with three out of four coincidence of 120 percent of the normal 200 Mwt delta T required for scram. Thirty-seven of the other positions are automatically recorded on multipoint charts and any of the thermocouples can be read manually in the control room by use of a bridge if desired.

During the three-day run at 100 Mwt which took place over the weekend of August 5-7, 1966, abnormal readings were noted above three core subassemblies. The inlet temperature was in all cases 540 F. These measurements are summarized as follows:

<u>Subass'y No.</u>	<u>Position</u>	<u>Abnormal Meas. Outlet Temp.</u>	<u>Predicted Outlet Temp.</u>	<u>Max. Meas. Delta T Predicted Delta T</u>
M-140	NO5 NO2	757	668	1.69
M-091	NO3 PO5	695	647	1.45
M-158	PO1 PO1	720	731	0.94

Subassemblies M-158 and M-091 were in locations which had a pair of thermocouples and in each case one thermocouple read normally (as predicted) while the second was abnormal as shown above. Subassembly M-140 was in a location with a single thermocouple.

Prior to the October 5, 1966, operation these three subassemblies were interchanged with subassemblies under other thermocoupled positions in the core in order to determine whether the fault was with the thermocouples or whether an abnormal ΔT was actually being observed. The interchanges were made on the basis of approximately equal power density and thermocouple location. These moves were as follows:

<u>Subass'y No.</u>	<u>From</u>	<u>To</u>	<u>Interchanged with Subassembly No.</u>
M-140	NO5 NO2	NO4 PO4	M-162
M-091	NO3 PO5	NO1 PO5	M-137
M-158	PO1 PO1	PO1 NO1	M-066

During the October 5 operation it had been planned originally to read the outlet temperatures from the six interchanged subassemblies at various power levels up to the maximum of 67 Mwt. However, it had not been regarded as crucial that these readings be made since the subassemblies had already run

during the 100 Mwt operation with no evidence of damage and the high thermocouple readings in the original positions were strongly suspected of being false because of the normal readings from adjacent thermocouples. The delays due to the malfunction of steam dump valve FCV-608-1 led to a rearrangement of the program for the day and, while the thermocouple readings were recorded, they were not observed during the automatic start-up.

When the Assistant Reactor Engineer observed the abnormal withdrawal of the control rods at 34 Mwt, his first thought was that the core temperature was too high and he went to the rear of the control panel to observe the core outlet thermocouple recorder charts. At this time he saw that two subassembly positions were indicating an abnormally high outlet temperature. These were M-140, in position NO4 PO4, to which it had been moved and its diagonal neighbor, M098 in position NO3 PO3, which had never been moved. The five other subassemblies which had been moved were not noticeably abnormal.

The following table shows the status of the six interchanged subassemblies and M098 on October 5 at 34 Mwt when the inlet temperature was 535 F.

<u>Subass'y No.</u>	<u>Position</u>	<u>Max. Measured Outlet Temp.</u>	<u>Predicted Outlet Temp.</u>	<u>Max. Meas. Delta T Predicted Delta T</u>
M-140	NO4 PO4	715	589	3.3
M-091	NO1 PO5	607 $\pm$ 5	587	1.4
M-158	PO1 NO1	612 $\pm$ 5	620	1.0 approx
M-098	NO3 PO3	695	610	2.1
M-162	NO5 NO2	606 $\pm$ 4	599	1.0 approx
M-137	NO3 PO5	592 $\pm$ 4	585	1.1 approx
M-066	PO1 PO1	612 $\pm$ 5	618	1.0 approx

REACTOR OUTLET THERMOCOUPLE LOCATIONS

THERMOCOUPLE INSTALLATION IN HOLD-DOWN MECHANISM

SECTION IV
PLANT TEST PROGRAM

Shield Temperatures

Readings of thermocouples attached to the primary shield tank wall were taken at the 24th, 48th and 60th hour of the 100 Mwt test run in August. Analysis of the data has shown that the average temperature of the 19 thermocouples was 186 F after 60 hours. This is above the expected average of 145 F to 165 F but well below the limit of 430 F where wall to pipe interferences would occur. Analysis has not yet been completed for the data on the graphite shield temperatures within the primary shield tank; the results will be reported when available.

During the 100 Mwt 3-day test run in August, the temperature of the concrete secondary shield wall was also monitored by thermocouples imbedded in the wall. There was a 5 F to 15 F rise in the shield wall temperatures during the run. The wall temperature immediately below the operating floor, in the north-northwest portion of the building near the overflow tank, exceeded the target temperature of 150 F. The highest temperature recorded was 172 F. These temperatures are acceptable; however, they will require monitoring as power levels and duration of operation are increased.

SECTION V

OTHER OPERATING EXPERIENCE

Shielding Evaluation

The abovementioned secondary shield wall was also evaluated during the 100 Mwt reactor operation in August for its effectiveness as a neutron shield. Neutron flux measurements were made outside the shield wall at decay tank No. 4 which is directly south of the wall and very close to it. Penetration of the building operating floor for the measurements was achieved through the decay tank plug. Measurement of both fast and thermal neutrons was made at four different elevations.

A measure of the absolute neutron flux levels could not be obtained, but a reasonable extrapolation of the results indicates that the neutron levels outside the shield wall are at least an order of magnitude less than originally estimated, and the activation of the secondary sodium will be well below the original design limits.

SECTION VI
MAINTENANCE AND ADDITIONS

A. Building Pressure Test

From October 1, to October 3, the reactor building interior was held at a pressure of 2 psig and the leakage rate measured. The reference tank method of testing gave a resulting daily leak rate of less than 100 standard cubic feet of air per day, while the leak rate resulting from the absolute pressure method was zero. The most pessimistic interpretation of the data would indicate a leak rate of less than 400 SCFD. The allowable leakage is 600 SCFD at 2 psig.

B. Power Cable Replacement

On two days in the middle of October, the plant 4800 V bus ground alarm indicated that sporadic grounds were occurring in the 4800 V circuits. The various circuits were isolated and tested by applying high voltage direct current from a portable power supply. Two underground cables with weak insulation were found; both broke down at about 4000 V to ground. The buried conduits containing the cables were in each case filled with ground water--this is not unusual for buried power supplies.

The diagram on page 40 shows the location of the two weak cables. One cable serves as a bypass line around the PRDC 4800 V bus and is used only during maintenance on the bus; The other cable is a short feed from a splice box to the 4800 V side of sodium heating transformer No. 4. Both cables are normally at 2770 V above ground. There is a separate cable for each phase of the power supply, each cable rated at 4800 V. Both weak cables were replaced, plus the cables of the other two phases in each feeder.

There was no apparent damage to either cable. The cables were sent to the Detroit Edison Company Laboratories for further inspection. See page 19 of the August report for information on earlier failures of two other 4800 V power cables.

FERMI REACTOR PLANT POWER SUPPLY SIMPLIFIED ONE-LINE DIAGRAM