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COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT
ENRICO FERMI ATOMIC POWER PLANT

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PREFACE

PURPOSE

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

SCOPE

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

Earlier Fermi experience in certain selected areas is being recorded in a series of technical reports completed or in preparation by Atomic Power Development Associates Inc. for the US Atomic Energy Commission under AEC Contract No. AT(11-1)-865, Project Agreement 15. This series of reports provides detailed information on the nuclear testing, rod drives, machinery dome, fuel subassemblies, fuel handling, reactor building ventilation, steam generators, pumps, flow-meters, level detectors, sodium sampling and development of the primary sodium system.

The items in Section II A & B are usually reported each month; items in the other sections are selected on the basis of their special significance during the month. Other items may be found in the monthly report submitted to the Atomic Energy Commission by Power Reactor Development Company in compliance with the requirements of Provisional Operating License No. DPR-9, as amended.

BACKGROUND

The Fermi Reactor achieved initial criticality on August 23, 1963. An extensive series of nuclear tests was conducted at power levels below one megawatt thermal, through 1965. A high power (200 Mwt) license was issued on December 17, 1965, and operation in excess of 1 Mwt was initiated on December 29, 1965. In January 1966, the power was raised in a series of steps to 20 Mwt. On April 1, 1966, power was first raised to 67 Mwt and on July 8, 1966, operation at 100 Mwt was initiated. On October 5, 1966, fuel damage occurred during an approach to power, since this time the reactor has been shut down while the cause and extent of the damage are being investigated.

It is assumed that those reading this report have a general familiarity with the plant. As an aid to the reader a perspective drawing of the plant was included at the back of Report No. 1.

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

SECTION I

CURRENT EXPERIENCE SUMMARY

The reactor continued to remain shut down as a result of the fuel damage sustained on October 5, 1966. Three additional subassemblies were removed from core positions adjoining subassemblies M098 and M127. The latter two subassemblies resisted a second attempt to separate one from the other. All the remaining core subassemblies, plus 18 outer radial blanket subassemblies, were then checked by lifting 15 inches to demonstrate that they could be removed from the support plates. As a result of removing the subassemblies around M127 and M098, it was observed that the required lifting forces had decreased in the rows where they had previously been higher than average.

A preliminary procedure was developed for removal of subassemblies M098 and M127 as a pair from the core. The sodium level in the reactor will be lowered and viewing equipment will be utilized to facilitate lifting the pair from the support plates and depositing them in a special pot at the transfer rotor. A special, articulated transfer tool is being designed for insertion through an access penetration in the rotating shield plug of the reactor vessel. This tool will perform a function similar to that normally performed by the reactor offset handling mechanism (OHM). The OHM is designed to accommodate only one subassembly at a time, and hence cannot completely withdraw the paired subassemblies.

As the reactor is drained, the sodium will be transferred to the primary sodium storage tanks in a shielded area of the Primary Sodium Service Building. The atmosphere in the room will be inerted with nitrogen to preclude the hazard of a radioactive sodium fire in the unlikely event of a sodium leak. Should a primary sodium fire occur despite all precautions, the fire would not result in exceeding the maximum permissible radionuclide concentrations at the site boundary but high exposure levels in the vicinity of the storage building would prevail.

A radiation survey was conducted in the area below the operating floor in the Reactor Building and in the cold trap cell of the Primary Sodium Service Building. Radiation levels were generally 1-2 mr/hr in the below floor area and 15-40 mr/hr in the cold trap cell.

Inspection of subassembly M099 at the Battelle Memorial Institute hot laboratory continued during the month. Several fuel pins were sectioned and metallurgically examined for hydriding of the zirconium cladding. Two of four pins

inspected indicated an unexpectedly high degree of hydriding - although not at a high enough level to cause spalling or blistering of the cladding. There was no external evidence of cladding damage on any of the pins. Previous examinations of test subassemblies removed from the reactor had shown the hydriding of the zirconium cladding to be negligible. Investigation of the hydriding is continuing.

The inlet strainer of core subassembly M091 was examined using a self-illuminated borescope mounted beneath the dry loading port of the steam cleaning machine in the Fuel and Repair Building. Subassembly M091 had been observed to have a high outlet sodium temperature prior to and during the October 5 run. It was removed from sodium into an argon atmosphere for visual inspection and immediately returned to the sodium. The strainer appeared to be entirely clean, with no evidence of blockage.

SECTION II

PLANT OPERATIONS

A. REACTOR UNLOADING

The diagram on the following page shows the extent of the reactor unloading at the end of January 1967. Three additional core subassemblies and one outer radial blanket subassembly were removed from the reactor during January. They were placed in the reactor vessel transfer rotor. The three core subassemblies were removed from positions immediately adjacent to M098 and M127, the two subassemblies that are stuck together. The three were removed to establish that no additional subassemblies were fused to M098 or M127, and to gain clearance around the fused pair to aid in their future inspection and removal. All subassemblies with sides adjoining M098 or M127 have been removed, three subassemblies with touching corners will also be removed.

The following is a listing of the changes in reactor loading made during January 1967.

<u>Date Removed</u>	<u>Subassy No.</u>	<u>Former Position</u>	<u>Remarks</u>
1/4/67	M087	NO3-PO2	Gave indications of being bent or warped. Placed in an oversized transfer pot
1/4/67	M007	NO2-PO3	No abnormal indications. Placed in a normal transfer pot
1/12/67	M118	NO4-PO2	Handling head was 3/8 inches low, had been 0.1 inch high in December. Lifted and resealed adjacent subassemblies, M118 then 0.1 inch low. Gave indications during pulling of being bent or warped. Put into oversized transfer pot.
1/16/67	S-510	N13-PO3	Raised 15 inches, would not reseat, stopped 1.2 inches above normal. Had resealed normally when raised less than 3/4 inch in December. Second attempt gave same result, therefore removed S-510 from the blanket.

Outer radial blanket (ORB) subassembly S-510 was removed when it was found that it could not be easily reinserted into the reactor lattice following a lift operation to 15 inches to verify that it was free. This difficulty is probably the result of galling between the subassembly nozzle

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)

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

Locations Where Changes Were Made ^d

IRB'S Replacing Core Subassemblies

REACTOR LOADING 1-31-67

and the lower support plate and not related to the October 5 fuel damage. The nozzles of the ORB subassemblies were not nitrided as were the core and IRB subassembly nozzles since the total amount of movement in and out of the blanket is more limited than in the core. The nominal clearance between an ORB subassembly nozzle and the upper support plate is 10 mils on the diameter.

B. SODIUM AND GAS SYSTEMS PERFORMANCE

1. Sodium Cold Traps and Plugging Indicators

<u>January 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)	290	0	0	0
Maximum Plugging Temperature - F	below 210	275	285	240
Minimum Plugging Temperature - F	below 210	250	240	220

2. 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	45 *
Carbon Monoxide	below 10	below 10
Carbon Dioxide	10	20
Hydrogen	below 4 **	2.5
Helium	below 4	below 4
Methane	below 10	below 10
N ₂ O	not measured	below 10
NO ₂	not measured	below 1
Argon	remainder	not measured
Nitrogen	1440	remainder
Analysis Date	January 6	January 6

* Technical Specifications state 1000 ppm maximum

**10 ppm recommended maximum

3. Primary Cover Gas Activity

<u>Location</u>	<u>Sample Date</u>	<u>Concentration</u>
Reactor	1/20/67	2.24×10^{-5} microcuries/cc

C. COUNT RATE INCREASE

At about 1300 hours on January 10, 1967, the source range count rate recorders in the main control room showed a rise in the shutdown count rate. Channel 1 increased by a factor of two and Channel 2 increased by a factor of approximately three. The increase took about five minutes to reach equilibrium and the recorders remained at the higher levels until 0410 hours the following morning, when they abruptly returned to their previous count rate levels. One hour counts were taken and logged from the control room scalars during the 1300-0410 time interval; they were irregular and reached peaks of four to twenty times normal. The fission product detector showed no increase in count rate and the reactor had approximately a \$12 shutdown margin, therefore it was concluded that there was no nuclear cause.

Discriminator curve data taken at the safety system cabinet during the interval showed little or no change from normal operational data. Analysis of the electronic systems indicated that the reason for the source range count rate increase was a noise source, common to both the Channel 1 and Channel 2 recorders, which entered this circuitry in the vicinity of the scale-of-two modules.

An a-c motor driven a-c generator provides the power supply to the safety system with a battery-powered d-c driver automatically taking over in case of a station power failure. This M-G set also powers the control room scalars which receive their signals from the scale-of-two modules, therefore it is a likely source of the noise. In the event of a recurrence of the high count rate, the M-G set will be carefully checked for any departure from normal performance.

D. RADIATION LEVELS

1. Reactor Building Below Floor Area

Before entering the below floor area, an exploratory radiation level survey of the area was made from inside equipment decay tank No. 1. This tank is accessible from the operating floor by removal of its floor shield plug. This does not disturb the below-floor nitrogen atmosphere since the decay tank walls form the air-nitrogen seal, with the nitrogen outside the tank. The ten foot diameter tank is located in the southwest part of the reactor building, approximately five feet south of the No. 3 intermediate heat exchanger (IHX). The bottom of the 32 foot deep tank is within about two feet of the reactor building lower floor. A vertical traverse made at the tank wall immediately adjacent to the No. 3 IHX indicated that the below floor activity at this location ranged from 0.04 to 0.5 mr/hr, with the highest value about 19 feet above the bottom of the decay tank.

The below floor atmosphere was then purged from nitrogen to air and this area was entered for further radiation monitoring. A vertical traverse was made on the south and west sides of the No. 3 IHX with the monitor touching the outside of the vessel insulation. The readings ranged from 0.05 mr/hr at an elevation 6 feet above the floor to 6 mr/hr at the No. 3 IHX drain line. A level of 2 mr/hr was observed at 12 feet and 20 feet above the floor.

Based on the analysis of krypton-85 in the cover gas taken shortly after the operation on October 5, it was concluded that the extent of melting was equivalent to about one complete subassembly. This conclusion was based on the assumption that all the krypton would be released to the cover gas during melting. The present radiation level of 2 mr/hr from the outside surface of the IHX, which could be entirely due to the sodium-22 activity, gives the maximum value for the amount of gamma-emitting fission products that could have escaped along with the krypton-85 during fuel melting.

Assuming uniform plate-out of these nongaseous gamma-emitting fission products and based on a radiation level of 2 mr/hr, it can be concluded that slightly less than one percent of the nongaseous gamma-emitting fission products was released during the melting if the equivalent of one subassembly was involved. This would result in a fission product concentration of slightly less than three-fourths of a microcurie per square inch of steel surface in the reactor. This is approximately the concentration of fission products detected on the can wall of subassembly M099. See Page 24.

The overflow tank and several of the primary sodium pipes were also surveyed, with the readings ranging from 1 to 3 mr/hr. A more complete survey of the area is planned to determine whether any concentrations of radioactive material can be located in the external primary system.

2. Cold Trap Cell

Radiation levels were measured at two locations inside the cold trap cell by means of an access port built into the north wall of the cell. Prior to cold trapping, the reading immediately inside the north wall was 14 mr/hr, while a level of 40 mr/hr was observed at a point 9 feet inside the wall (beneath the cold trap economizer). After 36 hours of cold trapping at 90 gpm sodium flow, the respective readings were 14 and 35 mr/hr. No significant change in these readings was observed after an additional 72 hours of cold trapping.

E. 1966 REACTOR OPERATIONS

The bargraphs on Pages 13 and 14 show the reactor power operation day-by-day from January 1, 1966, through October 5, 1966. Also noted on the graphs are the following data:

1. Primary coolant loops which were in service
2. Reactor control mode-automatic, manual or combined
3. Dates on which plant test data were taken
4. Dates on which nuclear test data were taken
5. Dates of scrams
6. Special operating conditions and the dates on which they applied
7. Fuel handling sequences, by date.

A note on Page 13 describes additional plant test data taken on a continuous basis.

F. 1966 REACTOR SCRAMS

The tabulation on Page 15 presents detailed information on the eighteen significant reactor scrams which occurred in 1966 up to and including October 5. Fourteen of these scrams were unscheduled and four were scheduled. All normal shutdowns of the reactor are terminated by a scram from low power. Since these scrams do not cause thermal transients nor indicate any malfunction, they are not included in this tabulation.

SHEET 2 OF 2

REACTOR OPERATIONS

Sequence Number	Date Time	Power Level (MWT)	Sens. or U (H or S)	Cause or Reason for Scram	Primary Control Flow & Balance Summary				Primary Control Flow After Scram				Max. React. Core Power (MW)	"Before" Scram Temp (°F)	Scram Temp (°F)	Time Duration ΔT (M ² ·s ^{1/2})	ΔT (°F)	Time ΔT (M ² ·s ^{1/2})
					10 ³ LB _{min}				10 ³ LB _{min}									
					L ¹	L ²	L ³	L ⁴	L ¹	L ²	L ³	L ⁴						
1	Jan. 5 1958	0.3-1.0	U	Unknown Transition in Primary System	0.1 2.0	0.38 2.6	0.37 2.6	1.1 7.7	0.1 2.0	0.38 2.6	0.37 2.6	1.1 7.7						
2	Jan. 6 1958	~2.0	U	Control Rod Malfunction	0.38 2.6	0.30 2.6	0.27 2.6	1.1 7.7	0.38 2.6	0.30 2.6	0.27 2.6	1.1 7.7						
3	Jan. 8 1958	0.2-0.4	U	Unknown Transient in Steam System	0.40 2.7	0.37 2.6	0.35 2.6	1.17 7.9	0.39 2.6	0.35 2.6	0.37 2.6	1.14 7.5						
4	Jan. 21 1959	13.0	S*	For NTP 20 (Flow Manually Reduced 3 Loop After Test)	0.30 2.0	0.25 2.0	0.20 2.0	0.61 4.1	0.44 2.9	0.41 2.9	0.41 2.9	1.34 8.5						
5	Jan. 24 1959	14.5	U	Safety Rod Dripped-Rod Holding Magnet Contact	0.29 2.2	0.25 2.0	0.22 2.0	0.83 5.9	0.29 2.2	0.25 2.0	0.22 2.0	0.82 4.9						
6	Feb. 2 1960	20.0	S*	For NTP 28	0.4 2.5	0.4 2.5	0.4 2.5	1.2 7.7	0.4 2.5	0.4 2.5	0.4 2.5	1.2 7.7						
7	Feb. 4 1960	<1.0	U	Facility Operation of Neutron Detectors	0.39 2.6	0.37 2.6	0.29 2.1	1.18 7.5	0.39 2.6	0.37 2.6	0.29 2.1	1.16 7.5						
8	Apr. 27 1961	<1.0	U	High For After Temp Single Detector Shutdown	0.42 3.0	0.42 3.0	0.42 3.0	1.2 7.7	0.42 3.0	0.42 3.0	0.42 3.0	1.10 6.7						
9	June 23 1960	0.8	S	Test Compliance With Technical Specifications	0.25 2.6	0.22 2.6	0.21 2.6	0.72 4.7	0.25 2.6	0.22 2.6	0.21 2.6	0.72 4.7						
10	June 23 1960	0.5-0.8	U	Loss of Primary Coolant Due to High Feed Water Temperature	0.31 2.6	0.22 2.6	0.21 2.6	0.72 4.7	0.31 2.6	0.22 2.6	0.21 2.6	0.72 4.7	11.5 POD-PO	735	815	580	235	27.1
11	July 1 1960	74.0	S*	For NTP 28	0.31 2.6	0.22 2.6	0.21 2.6	0.72 4.7	0.31 2.6	0.22 2.6	0.21 2.6	0.72 4.7	11.5 POD-PO	735	815	580	235	27.1
12	July 5 1960	74.0	U	Loss of Primary Coolant Due to High Feedwater Temperature	0.31 2.6	0.22 2.6	0.21 2.6	0.72 4.7	0.31 2.6	0.22 2.6	0.21 2.6	0.72 4.7	11.5 POD-PO	735	815	580	235	27.1
13	July 8 1960	110.0	U	Loss of Primary Coolant Due to High Feedwater Temperature	0.30 2.6	0.21 2.6	0.20 2.6	0.72 4.7	0.30 2.6	0.21 2.6	0.20 2.6	0.72 4.7	11.5 POD-PO	735	815	580	235	27.1
14	Aug. 31 1960	5.0	U	Safety Rod No. 2 Dripped-Rod Holding Magnet Contact	0.44 2.7	0.44 2.7	0.44 2.7	1.2 7.7	0.44 2.7	0.44 2.7	0.44 2.7	1.32 8.6	No Data Available for TR 154					
15	Sept. 2 1960	~6.5	U	As Above	0.30 2.6	0.30 2.6	0.30 2.6	0.72 4.7	0.30 2.6	0.30 2.6	0.30 2.6	0.72 4.7	No Data Available for TR 154					
16	Sept. 6 1960	22.0	U	As Above	0.29 2.6	0.30 2.6	0.32 2.6	0.91 5.6	0.29 2.6	0.30 2.6	0.32 2.6	0.91 5.6	No Data Available for TR 154					
17	Oct. 17 1960	~	U	As Above	0.29 2.6	0.30 2.6	0.32 2.6	0.91 5.6	0.29 2.6	0.30 2.6	0.32 2.6	0.91 5.6	No Data Available for TR 154					
18	Oct. 25 1960	22.0	U	As Above	0.30 2.6	0.30 2.6	0.30 2.6	0.72 4.7	0.30 2.6	0.30 2.6	0.30 2.6	0.72 4.7	No Data Available for TR 154					

* Scram initiated by instrument

1966 HISTORY OF SCRAMS AT FERMI

SECTION III

PLANT REPAIRS - CORE INVESTIGATION

A. LIFTING TEST

Following the removal in January of three more core subassemblies (see Page 7), the remaining subassemblies were lifted 15 inches to demonstrate that the inlet nozzles can be removed through the upper support plate. All remaining core and inner radial blanket subassemblies and eighteen ORB subassemblies were checked in this manner and found to be free. There was no difficulty except for ORB subassembly S-510, see Page 7 for details on this subassembly. Ten of the ORBs were raised 15 inches and eight (after S-510 galled) were raised only 1 inch. Additional unsuccessful attempts were made to separate subassemblies M127 and M098 using the OHM in the manner described in Section III F of Report No. 5.

The diagram on Page 19 gives the initial pulling forces required to start lifting the subassemblies. Similar diagrams were included in Section III of Report No. 5, showing pulling forces before and after removal of subassembly M140 from core position NO4-PO4. M140 was the first subassembly to be removed. Comparison of the diagrams shows that the previously observed tight rows appear to be essentially normal now that a total of eight subassemblies have been removed from the core. For a description of the manner in which the forces were measured see Section II B8 of Report No. 4.

B. SUBASSEMBLY DIMENSIONAL CHECK FIXTURE

A special fixture for dimensionally checking the materials surveillance subassembly prior to reloading into the reactor was received at the plant during January. The fixture has been set up temporarily on the operating floor of the Fuel and Repair Building for prechecking and testing. After acceptance it will be lowered into the cut-up pool where it will be utilized to perform several dimensional checks on the materials surveillance subassembly MS-1 (removed from the reactor on December 29, 1966) prior to disassembly and removal of test specimens. It is also expected that this gauging fixture will be useful to check any additional core subassemblies removed from the reactor after they have been steam cleaned and before they are segmented. IRB subassembly M393, removed from the core and steam cleaned in December, is awaiting dimensional inspection.

The fixture is over thirty feet in height. It will rest on the bottom of the cut-up pool and extend above the surface for operator access. Because of its large size it has been constructed in four sections to facilitate handling. The fixture was fabricated almost entirely from Type 304 stainless steel. A photograph of the fixture is shown on Page 20.

C. REMOVAL EQUIPMENT FOR SUBASSEMBLIES M098 AND M127

1. General

Since M098 (NO3-PO3) and M127 (NO4-PO3) are apparently stuck together, a special tool will be required to remove them as a pair from the reactor because the offset handling mechanism can withdraw only one subassembly at a time, as described in Section III F of Report No. 5. Design work was initiated on a tool to lift M098 and M127 from the core and place them in the transfer rotor. The design of related equipment, such as lighting and viewing, was also started in January. It is estimated that six months will be required to design, fabricate, install and check out this equipment.

2. Location of Equipment

Access to the reactor core will be gained through existing access holes in the reactor vessel rotating shield plug. Consideration is being given to installation of in-vessel lighting through the access hole on the plug center line. It is planned to insert a small diameter tube containing a 1500 watt Quartzline bulb mounted on the end in a protective cage with a half circle reflector for direction control.

There are three lower guide tube access holes in the rotating shield plug. These holes are approximately six inches in diameter and are normally used for installing or removing the safety and control rod lower guide tubes which occupy ten positions near the center of the core. The core sweep mechanism is presently located in one hole; the transfer tool will be installed in the second hole and the remaining hole will contain viewing equipment. Present plans are to obtain a corescope from the AEC for viewing. The sodium will be drained, as described in Section III D, at least to below the level of the subassembly handling heads. The drain line will be installed through one of the abovementioned holes. It will be necessary to remove the machinery dome, which encloses the top of the reactor, in order to install the removal equipment.

See Page 21 for a layout diagram of the transfer operations.

3. The Removal Tool

The paired subassemblies will be lifted and transferred by means of an articulated tool. See Page 22 for a drawing showing how the tool will be installed in the rotating shield plug. The rotating shield plug will be positioned so that the vertical centerline of the access hole containing the removal tool is at the position of closest approach to the common centerline of M098 and M127; approximately 4-inch displacement will exist between the two centerlines at this point. Since this horizontal offset is small, the tool in this position will be capable of exerting a 10,000 lb vertical lifting force on the subassemblies, in the event that they should be stuck in the support plates. The gripping mechanism at the bottom of the tool's articulated arm will provide engagement for both sub-assembly handling heads.

The carrying arm will then be extended part way toward the horizontal position, after the subassemblies have been raised sufficiently to clear the upper support plate. The arm will rotate upward as a result of a toggling action brought about by movement of a linkage within the main shaft of the tool. At the same time the arm is lifting, the tool will pivot and the shield plug will rotate so that the pair remains in almost the same horizontal position.

An additional 22 subassemblies located radially outward from M098 and M127 will have previously been removed so that the pair can next be moved into this area to clear the OHM upper stabilizer arm. This will be accomplished by first pivoting the removal tool to swing the pair about two feet towards the east, and then further extending the arm towards the horizontal position to move the pair approximately two feet towards the south.

At this point the shaft of the tool will be pulled up through the plug to fully withdraw the pair from the lattice. By rotating the plug and the shaft of the tool, the gripper can finally be positioned over a special pot in the transfer rotor; it will be a 43-inch horizontal span from the centerline of the main shaft of the tool to the centerline of the pot.

4. Shipping

A special transfer pot with a 7-inch inside diameter will be located in the transfer rotor to receive the fused pair. The multipurpose shipping cask will be positioned over the exit port in the manner described in Section V.B of Report No. 5. The operating shaft which lifts the pot into the cask will be fitted with a special gripper capable of engaging and sealing the special pot. The shipping cask will require new inside rails for supporting the special pot.

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

Net Pounds - Corrected For Buoyancy

Type Subassembly	Weight - Lbs (In Air)
Core	121
IRB	165
ORB	161

* Lifting Forces Not Measured
On Pair That Are Fused Together

SUBASSEMBLY LIFTING FORCES AFTER EIGHT SUBASSEMBLIES WERE REMOVED FROM THE CORE

COMPOSITE PHOTOGRAPH OF SUBASSEMBLY DIMENSIONAL CHECK FIXTURE

REACTOR PLAN SHOWING LATTICE POSITIONS AND PLUG PENETRATION TRAVERSES

ELEVATION OF REACTOR VESSEL

D. PROPOSED DRAINING OF THE REACTOR VESSEL

The use of the special tool for the removal of sub-assemblies M098 and M127 requires visual observation of the subassembly handling heads and will necessitate lowering the sodium in the reactor vessel at least to a level just below the handling heads. In order to examine visually the outside condition of the subassemblies prior to removal, the sodium will probably be completely removed from the vessel. A temporary line will be extended from the vessel to a drain line connection in the reactor building operating floor. The sodium will travel up out of the vessel, down through the floor penetration, out of the lower reactor building, through the sodium tunnel, into and through the cold trap cell and into the storage tanks in the Primary Sodium Service Building. Approximately 12 psig argon cover gas pressure, together with a vacuum on the storage tanks, will be utilized to transfer the sodium out of the reactor vessel.

The room in the sodium service building where the storage tanks are located is normally cooled with outside air. Before the radioactive sodium from the reactor vessel is returned to the tanks, the atmosphere in the room will be changed to nitrogen to eliminate the possibility of a fire in the event of a sodium leak. Such a fire would present an inhalation hazard in the immediate vicinity of the building chiefly because of Sr-90. There is no external radiation hazard from the sodium in transferring it or working near it.

The nitrogen will be recirculated within the primary sodium storage room, passing through cooling coils to remove the heat developed in the room by the setting losses from the tanks. The coils will contain refrigerant 12 or refrigerant 22 which will carry the heat to condensing units to be located outside the room. The storage room has 30-inch thick concrete walls and ceiling which are adequate for shielding. The room has very few penetrations and will be reasonably leak-tight. The room was designed to be inerted if this became necessary.

SECTION IV

OTHER INVESTIGATIONS

A. INSPECTION OF SUBASSEMBLY M099

Examination of subassembly M099 at the Battelle Memorial Institute hot lab at Columbus, Ohio, continued during January - see Section IV A of Report No. 5 for the previous history of M099. Results obtained during January are as follows:

1. The nozzle spring rate and free length were within specifications and indicated no significant relaxation due to irradiation.

2. The radiation level of the core section was as follows:

<u>Distance (Inches)</u>	<u>Radiation Level R/hr</u>
12	400
18	180
24	100
36	44
48	25

3. A very small quantity of a black deposit was removed from the lower nozzle inner sleeve, and is being analyzed. It has been determined that it is not carbonaceous.

4. Twenty core pins were checked for length. The average change in length was -0.10 percent, which is about as predicted. Core pins previously checked had a -0.16 percent change in length but had not been exposed to a sufficiently high fission rate to effect any significant phase retransformation. Core pin diameter was essentially unchanged.

5. The fuel pin spacing appeared to be normal. After the fuel pin bundle was removed from the wrapper can it was observed to be bowed by 50 mils. Since the wrapper can was not bowed, this slight distortion is thought to be of no special significance. See Page 27 for a photograph of the bundle.

6. The surface of the subassembly just above the nozzle and on the inside surface was analyzed for fission product plate-out by washing and then etching, with the following results:

<u>Fission Product</u>	<u>Concentration₂ microcuries/in. with etch depth of 0.65 mils</u>	<u>Concentration₂ microcuries/in. with additional 0.3 mils removed</u>
Ce-141 & 144	0.2	0.09
I-133	0.03	Not Detected
Ru-103	0.04	Not Detected
Zr-95 & Nb-95	0.7	0.3

Note: These results were normalized for base metal neutron induced radionucleides.

7. Four pins were removed from the fuel bundle and metallographically examined by sectioning, polishing and etching. The zirconium cladding on two of the pins appeared by photomicrographic inspection to have picked up about 2500 ppm of hydrogen, an unusually large amount. The third pin indicated 450 ppm hydriding of the cladding and the fourth pin 150 ppm hydriding. The first three pins all were produced from the same uranium billet and were heat treated in the same furnace lot. The original control samples of this billet and furnace lot showed less than 100 ppm hydriding in the as-built condition.

Previous cladding tests of fuel pin specimens in dry gas in the presence of hydrogen showed the five mil cladding began to fail when it had become hydrided to about 10,000 ppm. The method of failure was blistering, spalling and flaking. None of the 144 fuel pins in M099 showed any of these effects.

Earlier examinations of three proof test subassemblies and one sodium worth subassembly, all of which had been in the reactor, had shown the zirconium increased in hydrogen content by about 150 ppm. Hydriding tests have also been conducted by The Detroit Edison Company Engineering Research Laboratory and were discussed on Page 28 of Report No. 2.

Three more fuel pins will be removed from the bundle and checked for hydriding. It is also planned to remove the cladding from a pin section and determine the hydrogen content by vacuum fusion analysis.

B. EXAMINATION OF M091 WITH BORESCOPE

Subassembly M091 was removed from the reactor in December and taken to the Fuel and Repair Building (FARB). M091 had been observed to have outlet sodium temperature readings approximately 40 percent higher than predicted during July-August reactor operation and also on October 5th.

Section V. C. of Report No. 5 describes the method adopted for examining the inlet strainer of M091 by means of a borescope mounted under the steam cleaning machine dry loading port. Plugging of this strainer is a potential cause of fuel overheating. However, observation of the strainer at the end of January showed it was intact and clean. The subassembly was not steam cleaned prior to the examination, and was returned to sodium following observation.

C. POSSIBLE STEAM GENERATOR MODIFICATIONS

Steam generator modifications to improve reliability and performance are being considered. With the reactor down for investigation and repair, there is time available for changes to the steam generators. Discussions have been held with manufacturers concerning field modifications, shop rework or replacement units.

The two areas being most critically evaluated are:

1. The tube-to-tube sheet welds.
2. Pressure drop between the water and steam manifolds, and the pressure drop within the steam manifold.

Several methods are under study for improving the reliability of the tube-to-tube sheet welds. Orificing of the tubes is being considered to increase the pressure drop between the water and steam manifolds. An additional outlet nozzle in the steam manifold would decrease the pressure gradient in the manifold.

The steam generator modifications were still in the discussion stage of development during January. Further information will be reported in the February report.

SUBASSEMBLY M099 FUEL PIN BUNDLE
VIEWED THROUGH BMI HOT CELL WINDOW

SECTION V

MAINTENANCE

A. STEAM CLEANING MACHINE VALVE

The ten-inch shutter valve, at the bottom of the cleaning chamber of the steam cleaning machine in the Fuel and Repair Building (FARB), seized in the partially open position during operation. The valve had stuck several times previously but had been freed by flushing and manipulating. The valve is a double-disc pivoting-type gate valve that is actuated by an air cylinder connected to the valve stem by a crank.

The valve was removed from the steam cleaning machine, taken out of the shielded cleaning cell and decontaminated in the ultrasonic cleaner. The internals of the valve had an activity level of 70 mr/hr before decontamination.

The stem of the valve was hard chrome plated Type 304 stainless steel, turning in a 304 stainless steel bushing with asbestos-graphite packing. This combination had resulted in severe galling. See photographs on Page 30. A second area of galling was found in the bearing support pad of the valve plate carrier which had hard chrome plating on cast steel, sliding on a mating pad of type 304 stainless steel. The latter pad is an integral part of the housing (valve body).

The valve stem was replaced with an identical part made of hardened 440-C stainless steel. The stem bushing was bored out to receive a 440-C stainless bearing insert, with 8 mils clearance for the stem and with 1/16-inch deep wiper grooves on the inside diameter. The stem packing was replaced with Teflon. The galled bearing support pad on the valve plate carrier was repaired by machining off the old pad and building up a new pad with Stellite No. 5 overlay. Several other minor improvements in the valve construction were also made at this time. The valve was reinstalled and the cleaning machine returned to service.

The decontamination and repair of the valve was conducted in the new FARB machine shop. This shop was completed several months ago. It is located at the center of the FARB operating floor and was constructed over the north end of the large maintenance pit.

B. CASK CAR ELECTRICAL MALFUNCTION

The cask car gripper is routinely cleaned with alcohol and water in the FARB maintenance shop after each use of the cask car. This is accomplished by lowering the gripper into the maintenance glove box under the cask car, disconnecting the gripper from the hoist cable and transporting it to the maintenance shop.

Erratic operation of the gripper hoist occurred during the lowering operation in January, due to an electrical malfunction. The hoist position indicating lights operated improperly and the hoist motor action failed to follow the control signals, resulting in excess slack in the cables. This slack caused them to become fouled in the hoist compartment. One cable was damaged and had to be replaced.

The malfunction was traced to a quick disconnect coupling in the cask car hoist control cable. Several wires had been twisted and disconnected. The coupling was repaired.

C. PRIMARY SHIELD TANK PRESSURE DECAY TESTS

A pressure decay test from 2 psig was conducted and several large leaks were discovered, caused by loose cover plates over the surveillance tube penetrations in the machinery deck of the primary shield tank (PST). The plates were tightened and the test repeated, resulting in an initial leak rate of 1340 standard cubic feet per day (SCFD), decreasing as the pressure differential decreased. Although this leakage was well below the 2000 SCFD maximum stated in the plant Technical Specifications, it was approximately 500 SCFD higher than the value obtained in a test in October 1965.

To determine if the increase in leakage was through the machinery deck or through the PST below floor, a second test was run in which the machinery dome was pressurized and controlled at 2" of water less than the PST. During the pressure decay test, the machinery dome pressure was controlled at this differential by feeding or bleeding air from the machinery dome. The results of this test indicate that the entire increase in leakage is through the machinery deck and therefore it can be concluded that the integrity of the PST below the sodium level in the reactor remains satisfactory.

STEAM CLEANING CHAMBER
SHUTTER VALVE - LOWER
FLANGE, ACTUATING
SHAFT REMOVED

GALLED SURFACE
AT CENTER OF
ACTUATING SHAFT

VALVE UPPER FLANGE,
ACTUATING SHAFT
AND BUSHING