ENRICO FERMI ATOMIC POWER PLANT

CURRENT EXPERIENCE SERIES

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT ENRICO FERMI ATOMIC POWER PLANT JULY - AUGUST 1968

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and
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PREFACE

PURPOSE

The purpose of this 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 operations and maintenance experience concerning the nuclear portion and related systems of 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 U. S. Atomic Energy Commission under AEC Contracts No. AT(11-1)-865, Project Agreement 15. This series of reports provides detailed information on the nuclear testing, machinery dome, steam generators, pumps, flowmeters, level detectors, sodium sampling and development of the primary sodium system.

Items in the sections of this report 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 1 Mwt, 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 that time the reactor has been shut down while the cause of damage is being eliminated and the reactor restored to the operating condition.

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 APDA-CFE-1. In addition, a topical index appears at the end of APDA-CFE-23.

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.

I. CURRENT EXPERIENCE SUMMARY

The development of tools and devices for removing the remaining zirconium segments from the reactor vessel continued through July and August.

There is sufficient confidence in the arc-melt process for removing the screwheads fastening the segments to the conical flow guide that emphasis has now been placed on the development of the arc-melt tool and development of the chisel tool has been assigned a lower priority. It has been demonstrated in the reactor mock-up that the arc-melt tool can be positioned at all screwheads holding the zirconium segments in place and can melt the screwheads to free the segments, after which a slight nudge by the tire iron or similar tool easily dislodges the segment from the surface of the conical flow guide.

The new electrohydraulic control system and the rotatable gripper for the spine-type manipulator are still in the development stage and must yet be tested in the reactor vessel mock-up. In a mock-up test, the new retrieval gripper for withdrawing the segments through the 14-inch line was found to need modification and a design change is presently being made.

Sodium in the primary system at a temperature of 500 F was cold-trapped to a plugging temperature of less than 220 F. After primary system clean-up, the system temperature was reduced to 350 F and an extensive fuel handling program was started to remove core subassemblies from the support plate positions needed for insertion of tools and devices for the zirconium segment removal program. The core subassemblies to be re-located will be stored in inner radial blanket (IRB) subassembly positions in the reactor lattice and in the transfer rotor. Thirty-one displaced IRB subassemblies will be removed from the reactor, steam-cleaned, and stored in the decay pool.

Three core subassemblies presently stored in the transfer rotor were found to be distorted when inspected in the exit port inspection facility. These subassemblies, which were adjacent to damaged subassemblies M098, M127, M140, and M122 during the incident of October 5, 1966, are to be removed from the reactor during the IRB transfer program and temporarily stored in the transfer tank in the Fuel and Repair Building (FARB) prior to steam-cleaning and storage in the decay pool.

II. INVESTIGATION OF ZIRCONIUM SEGMENTS

A. BACKGROUND

The deformed sheet of zirconium removed from the reactor in March was one of six triangular-shaped segments originally installed on the sloping sectors of the conical flow guide in the inlet plenum to the reactor core. Details of the removal of the first segment are given in APDA-CFE-20. The present plan is to dislodge each of the remaining zirconium segments and withdraw them through the 14-inch sodium inlet pipe of the No. 1 loop in the same manner that the deformed object was removed.

The development of tools to remove the heads of screws fastening the segments on the conical flow guide and to dislodge the segments from the flow guide is in progress. The control and some of the structural features of the devices intended for handling and removing the remaining zirconium segments through the 14-inch pipe are being modified for greater flexibility on the basis of previous experience with these devices in handling and removing the first zirconium segment. The full-diameter mock-up of the reactor vessel and the simulated top structure of the rotating shield plug are complete and ready for the functional testing of the tools and devices intended for use in the segment removal program. Previous information on the investigation of the zirconium segments is given in APDA-CFE-20, 21, 22, 23.

B. SCREWHEAD REMOVAL TOOLS

In July, primary emphasis was placed on the arc-melt method* for screwhead removal. An alternate method, the chisel process,** has been assigned a lower priority. Fabrication of the full-length arc-melt tool was completed and the tool used in the reactor mock-up, confirming satisfactory tool access to all fastening screw positions on the conical flow guide. Figure 1 shows the core support plate positions that must be available for insertion of tools and viewing and lighting equipment to accomplish the dislodging of the remaining zirconium segments and handling them in the inlet plenum. The swing positions indicated are those that must be vacant to permit swinging the lower ends of tools and devices into adjacent positions that are slightly offset from the 6-inch penetration through the rotating plug.

* See Section II of APDA-CFE-21, 22, and 23 for previous information about the arc-melt method.

** See Section II of APDA-CFE-20 through 23 for previous information about the chisel process.

Note:

T = Tool SP = Spine Manipulator
S = Scope SW = Swing Path for Tools
L = Light

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FIG. 1 LATTICE POSITIONS NEEDED FOR TOOLS AND DEVICES
FOR ZIRCONIUM SEGMENT REMOVAL PROGRAM

In a functional test, the arc-melt tool successfully melted away the three screwheads from each of two of the zirconium segments fixed to the conical flow guide in the mock-up, and the two segments fell away from the flow guide with minimal dislodging effort. The craters that were left after arc-melting are as shown in Figure 2, and a dislodged segment is shown in Figure 3. A commercially available stud welder is the power source for the arc-melt tool. The necessity for applying a second arc to assure that the remaining screw shank is welded to the conical flow guide and for collecting any resultant debris is to be investigated.

After application of the arc-melt tool to one of the screwheads, part of the head remained, and this small fragment held the segment securely on the flow guide. It was concluded that a repeat arc-melt operation should be carried out if the entire screwhead is not removed at the first try.

Figures 4 and 5 show the arc-melt tool inserted through the mock-up core support plates along with the lighting and viewing equipment. The latter devices are lifted above the inlet plenum during the arc-melt operations. It is intended that the full-length tool will be completely contained in a 350 F argon test atmosphere to verify the functioning of the articulating joint and the arc-melt operation in the drained reactor environment.

C. SEGMENT DISLODGING TOOLS*

It now appears that the tire iron tool may be the most commonly used of the available segment dislodging tools. Tests of all the tools in the mock-up have demonstrated that the tire iron tool is the easiest to apply for dislodging the segment (Figure 6) and, if necessary, for deforming the segment before removal of the screwheads (Figure 7). A close-up of this tool is shown in Figure 8 (the lower tool) along with another design of tire iron tool. Other tools that are available for deforming and dislodging the zirconium segments are shown in Figures 9 and 10. Figure 11 shows a dislodging tool and the viewing and lighting devices inserted through the mock-up support plates for access to the zirconium segments on the conical flow guide in the inlet plenum.

D. HANDLING AND RETRIEVAL DEVICES**

The new electrohydraulic control system for the spine-type manipulator and the new rotatable gripper are in the process of development. These features are expected to provide the precise control and versatility of the gripper necessary for operating the manipulator through the rotating plug and the support plate holes to handle the freed segments for removal through the 14-inch inlet line.

* See Section II of APDA-CFE-21, 22, and 23.

** See Section II of APDA-CFE-15 through 19 for previous information about the spine-type manipulator and the object-retrieval device.

**FIG. 2 ARC-MELT TOOL AND CRATER LEFT
AFTER SCREWHEAD REMOVAL**

**FIG. 3 ZIRCONIUM SEGMENT AFTER
REMOVAL OF SCREWHEADS
BY ARC-MELT PROCESS**

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FIG. 4 ARC-MELT TOOL APPLIED TO ZIRCONIUM SCREWHEAD IN MOCK-UP OF INLET PLENUM

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**FIG. 5 ARC-MELT TOOL CLUSTERED WITH DUMMY SUBASSEMBLIES
IN MOCK-UP OF SUPPORT PLATE**

**FIG. 6 TIRE IRON TOOL DISLODGING
ZIRCONIUM SEGMENT IN
REACTOR MOCK-UP**

**FIG. 7 DEFORMING SEGMENT IN REACTOR
VESSEL MOCK-UP**

FIG. 8 TWO DESIGNS OF TIRE IRON TOOL

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**FIG. 9 EDGE-CURLER TOOL DEFORMING ZIRCONIUM
SEGMENT IN REACTOR MOCK-UP**

FIG. 10 SHOVER TOOL

**FIG. 11 DISLODGING TOOL AND VIEWING
EQUIPMENT INSERTED THROUGH
MOCK-UP OF SUPPORT PLATES**

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It was demonstrated that the wood model of the gripper * for the object-retrieval device could roll in and out of the flexible hose in the inlet plenum mock-up satisfactorily. When the actual steel gripper device became available and was test maneuvered in the mock-up through the flexible hose inserted through the 14-inch inlet line, it was found that the device spun on the plenum deck and would not properly respond to movements of the control cable. It was also noted that the device galled the plenum deck during maneuvering. It is intended to modify the design using aluminum as the material of construction, eliminating the two rings of ball rollers, and shaping the device in the form of a football.

* See APDA-CFE-23, Figure 2.

III. OPERATIONS

A. COOLING PRIMARY SYSTEM

At the beginning of July, the temperature of the primary system was being held at 500 F and had just been decreased from a maximum of 563 F by reduction of pipe and equipment heating and operation of the cold trap. As stated in APDA-CFE-23, Section III. A. 4, the intent had been to raise the primary system temperature to 600 F and to conduct clean-up operations at that temperature. However, primary system heat-up was discontinued at 563 F, because of excessive temperatures developing in the secondary shield wall concrete,* and primary system temperature was decreased to 500 F for clean-up operations. Shield wall temperatures, which had reached about 190 F in local areas (40 F in excess of the plant-imposed limit) decreased as primary system temperature was reduced; leveled off at 175 to 180 F at primary system temperature of 500 F; and dropped below the 150 F plant-imposed limit when primary system temperature decreased to 350 F at the end of July. The limit of 150 F was established on the basis of not exceeding 200 F in the event of the loss of below-floor cooling for a 24-hour period to prohibit dehydration of the concrete.

The primary system temperature remained at 500 F during the first half of July, and intermittent cold-trapping was performed as necessary to achieve a plugging temperature of less than 220 F. Primary system temperature was then further decreased to 350 F, where it remained during August when an extensive fuel handling program was started.** There was no cold-trapping during August; the plugging temperature of primary system sodium remained below 240 F. Primary sodium flow with three pumps operating at minimum speed was about 6.4×10^6 lb/hr, which is less than the 7×10^6 lb/hr limit imposed by plant technical specifications with the holddown mechanism in the up position.

July and August operating data were as follows:

* APDA-CFE-23, Section III. A. 5

** See Section III. D

<u>July 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, hr	157	26	89	0
Maximum Plugging Temperature, F	240	310	290	260
Minimum Plugging Temperature, F	220	220	220	225

<u>August Operating Data</u>				
Cold Trap Operation, hr	0	23	0	0
Maximum Plugging Temperature, F	240	260	275	235
Minimum Plugging Temperature, F	220	<220	<240	230

B. TEST OF SWEEP MECHANISM

The operation of the sweep mechanism was checked in 500 F sodium in the reactor vessel to verify that the mechanism is still operable after its long storage in a sodium vapor-argon atmosphere in the reactor vessel. The mechanism, which is installed through one of the 6-inch penetrations through the rotating plug, has a horizontal arm at its lower end which traverses a fan-shaped area and sweeps just above the top of the subassembly handling heads. It is normally operated immediately after the holddown mechanism is raised from the full down position and prior to shield plug rotation to ensure that no subassemblies have been pulled out of the lattice with the holddown column by adhering to the holddown fingers.

The sweep arm was lowered manually and the torque required to rotate was noted. The mechanism was found to operate satisfactorily. A build-up of oxide deposits in the mechanism sufficient to obstruct free motion would have been reflected by higher than normal torque requirements to lower the sweep arm. Preparations are presently under way for removal and temporary storage of the sweep mechanism. Sweep removal is necessary to afford access for viewing equipment and tools for removal of the zirconium segment from the north sector of the conical flow guide in the reactor inlet plenum. Preparations for the sweep removal will be described in a future CFE report.

C. SODIUM AND GAS SYSTEMS PERFORMANCE

Since the last data reported,* the following primary system sodium and cover gas analyses and gas activity have been obtained:

1. Primary Sodium Chemical Analysis

Oxygen	12, 13, 14	Iron	13, 27.4, 2.9, 12
Carbon	29, 38, 40	Nickel	1.9, 4.6, 0.4, 2.1
**Hydroxide Hydrogen	1.7, 1.8	Chromium	3.4, 9.2, 0.8, 3.8
**Nonhydroxide Hydrogen	0.2, 0.3		

Note: Values are in ppm by weight. A sample coil was taken from the sampling station on July 12 after primary system clean-up and was analyzed at several different points to obtain the separate readings indicated.

The wide variation in values for iron, nickel, and chromium should be noted. The third set of determinations (2.9, 0.4 and 0.8) is considered normal; the other sets of determinations vary from normal, yet the individual elements remain in about the same proportion as found in stainless steel. It is possible that there was pick-up of stainless steel in varying amounts at the different sampling points, or that the samples were contaminated at the time the 304 stainless steel sample tube was cut. It is intended that another sample for analysis be taken shortly but in a nickel tube to eliminate the uncertainty of contamination.

* APDA-CFE-23

** 1.3 ppm recommended maximum total hydrogen for reactor operation.

2. Primary System Cover Gas Analysis

	Reactor Cover Gas (Argon), ppm by volume		Primary Shield Tank Atmosphere (Nitrogen), ppm by volume	
	July 25	August 30	July 19	August 8
Oxygen	Below 25	Below 25	100*	100*
Carbon Monoxide	Below 10	Below 10	Below 10	Below 10
Carbon Dioxide	25	Below 10	25	25
Hydrogen**	Below 4	Below 4	Below 2.5	Below 2.5
Helium	Below 4	Below 4	Below 4	Below 4
Methane	Below 10	Below 10	Below 10	Below 10
Nitrous Oxide	← Not Measured →			
Argon	← Remainder →		90	3600
Nitrogen	4720	1620	← Remainder →	
Dew Point	← Not Measured →		-45	-55

3. Primary System Gas Activity

<u>Location</u>	<u>Sample Date</u>	<u>Gross Beta Concentration, microcuries/cm³</u>
Reactor Cover Gas	7-3-68	1.5×10^{-5}
Reactor Cover Gas	7-11-68	9.5×10^{-6}
Reactor Cover Gas	7-19-68	1.2×10^{-5}
Reactor Cover Gas	7-25-68	8.7×10^{-6}
Reactor Cover Gas	8-2-68	9.1×10^{-6}
Reactor Cover Gas	8-9-68	7.9×10^{-6}
Reactor Cover Gas	8-16-68	7.9×10^{-6}
Reactor Cover Gas	8-23-68	5.7×10^{-6}
Reactor Cover Gas	8-30-68	4.1×10^{-6}
Primary Shield Tank	7-3-68	2.8×10^{-8}
Primary Shield Tank	7-11-68	4.0×10^{-8}
Primary Shield Tank	7-19-68	2.5×10^{-8}
Primary Shield Tank	7-25-68	8.0×10^{-8}
Primary Shield Tank	8-2-68	1.5×10^{-8}
Primary Shield Tank	8-9-68	3.5×10^{-8}
Primary Shield Tank	8-16-68	2.4×10^{-8}
Primary Shield Tank	8-23-68	3.5×10^{-8}
Primary Shield Tank	8-30-68	3.5×10^{-8}

* Technical specifications state 1000 ppm maximum.

** 10 ppm is the recommended maximum for reactor operations.

D. REACTOR SUBASSEMBLY MOVEMENTS

By the end of August, ten core subassemblies and eleven outer radial blanket (ORB) subassemblies had been moved to different positions in the reactor lattice. One IRB subassembly was moved from a transfer rotor position to the transfer tank in the FARB. These moves were the beginning of an extensive fuel handling program to open up the required support plate positions for insertion of tools and devices into the inlet plenum for the zirconium segment removal program (see Figure 1).

The intent is to open up positions in the IRB row in the reactor lattice and positions in the transfer rotor to provide temporary storage for core subassemblies removed from the core lattice to accommodate segment-removal tools and devices. Displaced IRBs will be removed from the reactor to the FARB transfer tank, steam-cleaned, and temporarily stored under water in the decay pool. It is expected that 31 IRBs will be removed from the reactor to provide in-sodium storage for core subassemblies. For future reactor operations, replacement IRBs will be drawn from an adequate number of new IRBs presently in new subassembly storage.

Table 1 summarizes the subassembly relocations made during August.

The subassembly moves, shown on the loading diagram (Figure 12), were accomplished with the offset handling mechanism and the rotating shield plug operated in the automatic mode. The ORBs were moved to fill the area on the east side of the reactor lattice vacated in August 1967 for the holddown finger inspection program,* to provide camming surfaces for storage of core subassemblies in IRB positions in row N06. Where possible, throughout all subassembly moves, the lift-out or insertion positions were chosen so that the subassemblies were oriented in the offset handling mechanism for easy camming into the new insertion position. For other subassembly relocations, it was necessary to precam the subassembly in an intermediate position before insertion in the intended position. This was necessary because many times only two camming surfaces were available, rather than the preferred three or four.

E. TRANSFER OF IRB SUBASSEMBLY TO TRANSFER TANK

IRB subassembly M326 was removed from the reactor transfer rotor in late August and transported to the transfer tank in the FARB in the multi-purpose shipping cask** fabricated by Battelle Memorial Institute. This was the first of 31 IRB transfers to be carried out to provide storage for core

* APDA-CFE-18 and 19

** See APDA-CFE-5, pages 27 and 28, for a description of the shipping cask.

TABLE 1 - SUBASSEMBLY MOVES DURING AUGUST

<u>Subassembly Number</u>	<u>Type</u>	<u>Position</u>		<u>Reason for Change</u>
		<u>From</u>	<u>To</u>	
S508	ORB	N01-N12	N09-N02	To provide camming surfaces for inserting core subassem- blies into IRB row N06
M863	ORB	N07-P12	N09-N01	
M997	ORB	P05-N12	N08-N02	
M877	ORB	N03-P13	N07-N02	
M957	ORB	N07-P00	N07-P02	
M942	ORB	N08-P09	N08-N01	
M809	ORB	N04-P12	N07-N01	
M870	ORB	N06-P11	N08-P00	
M861	ORB	N02-P14	N07-P00	
M878	ORB	N05-P13	N08-P01	
S502	ORB	P05-N13	N07-P01	
M109	Core	N01-P04	N03-P05	To open up core support plate po- sitions for zirconium segment removal program
M107	Core	P01-P04	N03-P04	
M027	Core	P00-P04	N04-P05	
M132	Core	N01-P03	N05-P03	
M056	Core	P00-P03	N04-P04	
M155F*	Core	P03-P03	N06-N02	
M002	Core	P03-P02	N06-N01	
M166F*	Core	P03-P01	N06-P02	
M125	Core	P04-P03	N06-P01	
M144	Core	P05-P02	N06-P00	
M326	IRB	TR01	FARB Transfer Tank	To provide core subassembly storage

* See Figure 12 for explanation of F subassemblies.

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

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 Sylvania 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

Note: Heavy Boxes at Outside of Lattice are Storage Positions

FIG. 12 REACTOR LOADING ON 8-31-68

subassemblies. The exit port inspection facility was moved from its position over the exit port prior to this transfer. The shipping cask was used because the reactor exit port section of the new fuel transfer facility is not yet available and the old cask car has been disassembled for disposal.* The shipping cask, shown in Figure 13, includes a new lower valve assembly which seals to the top of both the reactor exit port and to the floor valve over the transfer tank, replacing the plastic bag used for sealing purposes for previous subassembly transfers. Two views of this new valve assembly are shown in Figures 14 and 15. The valve assembly serves as part of the cask shielding also.

Time-consuming difficulty was encountered in trying to latch the special cask gripper on the subassembly pot in the transfer rotor. The gripper, located at the bottom of three extension rod sections which penetrated the top of the cask, had to be raised and lowered several times and realigned in the center of the cask before it finally gripped the pot properly. The subassembly was transported in an extra-large-diameter drainable pot designed to accommodate a bent subassembly. The highest radiation observed during the transfer of the subassembly was 100 mr at the surface of the cask, while the lower cask valve was open and a subassembly was being raised into the cask.

* See Section VI. A.

**FIG. 13 MULTIPURPOSE CASK FOR TRANSFERRING
SUBASSEMBLIES FROM REACTOR**

FIG. 14 LOWER VALVE ASSEMBLY

**FIG. 15 BOTTOM SIDE OF LOWER VALVE
ASSEMBLY**

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IV. SUBASSEMBLY INSPECTIONS

A. INSPECTION OF SUBASSEMBLIES IN TRANSFER ROTOR

Two IRB and six core subassemblies positioned in the transfer rotor were inspected in the exit port inspection facility prior to removal of the facility for transfer of subassemblies out of the reactor. Two other IRBs in the transfer rotor were not inspected because they were not near the melted subassemblies at the time of the incident of October 5, 1966, and will be removed from the reactor for disposal as part of the IRB transfer program.

Table 2 shows the type of inspections carried out and the location of the subassemblies with respect to the damaged subassemblies M127, M098, M140 and M122 at the time of the incident:

TABLE 2 - SUBASSEMBLY INSPECTIONS

<u>Subassembly Number</u>	<u>Type</u>	<u>Visual Inspection</u>	<u>Quick Scan</u>	<u>Full Dimensional Scan</u>	<u>Location</u>
M357	IRB	X			2 rows from damaged cluster
M367	IRB	X	X	X	Adjacent to cluster
M007	Core	X		X	} Adjacent to cluster
M118	Core	X	X		
M087	Core	X	X	X	
M058	Core	X	X	X	4 rows from cluster
M162F	Core	X	X	X	5 rows from cluster
M079	Core	X	X	X	7 rows from cluster
M359	IRB	} Not Inspected			} 3 and 6 rows from damaged cluster
M326	IRB				

By means of the inspection methods used, core subassemblies M007, M118, and M087 were found to be definitely distorted, M087 5/8-inch and M118 in excess of 7/8 inch. M007 was determined by visual inspection to be distorted, and this was verified by the preliminary results of the full dimensional scan which indicated about 1/4-inch distortion. These three subassemblies had experienced an excessive lateral temperature differential of 240 F to 270 F (less than 50 F is normal) and therefore were expected to be distorted. It was decided that these subassemblies will be removed from the reactor as part of the IRB transfer program and temporarily stored in the transfer tank. There was no immediate indication of distortion of any of the other subassemblies inspected; however, the results of the full dimensional scan are to be analyzed for more precise determinations of subassembly straightness.

The quick scan method is an inspection for gross distortion. With the periscope located at one of the viewing windows of the inspection facility, the subassembly is raised and lowered and dimensional data obtained.

B. MODIFICATIONS TO EXIT PORT INSPECTION FACILITY

Design changes were incorporated into the exit port inspection facility as a result of mechanical difficulties found during the operational test of the facility.* These modifications were made prior to the above inspection of subassemblies.

The upper four corners of the gripper tube and the edges in the bottom opening of the rotational plate were chamfered for better lead-in for guiding the gripper tube when the gripper raises a subassembly into the inspection chamber. The penetration through the inspection chamber for the yoke actuator rod was elongated in the vertical direction and tilted upward to give sufficient flexibility to the lower gripper latch rod (inserted through the yoke actuator rod) to engage the gripper latching and delatching slot in the gripper tube. The yoke is the lower guide fixture for the gripper tube as the gripper raises or lowers the subassembly through the lower part of the inspection chamber. The yoke actuator is now capable of moving vertically through a 1-1/2-inch range inside the inspection chamber, and the latch rod through a 1/2-inch vertical range. A plastic bellows seals the yoke rod penetration, and a double O ring and gas seal confines the annulus between the latch and yoke actuator rods.

* Mechanical difficulties were reported in APDA-CFE-19, page 21.

See APDA-CFE-5, 8, 9, 10, 16 through 22 for other information about the exit port inspection facility.

C. CHECKOUT OF EXIT PORT INSPECTION FACILITY WITH STANDARD SUBASSEMBLY GAUGE

A preliminary checkout of the exit port inspection facility was performed using a standard subassembly-shaped bar of precisely known dimensions. The collimating technique * was applied to the subassembly inspection procedure. Following this, a dummy subassembly of known bow was dimensionally checked in the inspection chamber. The results of both inspections indicated that with the present degree of measuring skills the facility should be capable of detecting subassembly distortion in excess of 0.050 inch. The principal inaccuracy appears to result from difficulty in obtaining repeatability of taut wire bench marks for dimensional measurements. Considering only a single distorted subassembly in the support plate, it has been calculated that a combination of distortion of a subassembly of 0.086 inch and the out-of-plumb conditions of the support plate and holddown fingers could result in failure of the holddown finger to engage the handling head in the worst position in the reactor support plate. This allowable distortion number will be determined for the case of the interaction of many subassemblies in the support plate in order to arrive at the allowable subassembly distortion.

* APDA-CFE-22, page 21.

V. SPECIAL INVESTIGATIONS

A. TEST OF LIGHTING FOR HOLDDOWN FINGER INSPECTION

Figure 16 shows the mock-up of a cavity in the subassembly lattice that was used in a preliminary test of a new method of lighting for inspection of holddown fingers in the reactor. The inner walls of the cavity were prepared in a manner to reproduce the reflective conditions of the subassembly wrapper can walls in the reactor. The difficulties encountered in an attempt to view and photograph the holddown fingers in February are related in APDA-CFE-19. Figure 17 shows the wire-mesh-enclosed compartment designed to contain four 1500-watt lamps clustered around the mirror support post.

In this test, the mirror and support post and a borescope were inserted into the cavity mock-up, and a cardboard mock-up of the holddown fingers (circles drawn on the cardboard) was suspended above the cavity. With this arrangement sealed from external light, many photographs of the simulated fingers were taken using Kodak Tri-X pan film at varying exposure time. The results, though yet inconclusive, were encouraging. For 5- to 10-second film exposure time, the lighting appeared adequate for image resolution down to 1/16 inch, considered to be about the upper limit for determination of holddown finger condition. It is felt that further refinement in photographic technique can be attained. It is expected that a more precise replica of the holddown fingers which is now being constructed will be used in further tests.

B. REPLACEMENT FOR PRIMARY SHIELD TANK EXPANSION SEAL

Design effort is under way to produce a new seal to contain the primary shield tank nitrogen atmosphere between the machinery platform and the shield tank wall. The new seal is intended to replace the expansion seal (boot seal) which leaked during primary system heat-up for clean-up operations in June.² The material presently being considered for the replacement seal is 1/8-inch-thick silicon rubber with fiber glass reinforcement. It is intended that the old seal will be cut and removed from anchoring positions beneath the machinery deck and the new belt seal installed between these same anchoring positions and the shield tank wall at a higher, more accessible elevation. Securing the new seal at a higher elevation on the shield wall is expected to eliminate tensioning the seal material as the machinery deck rises with respect to the primary shield tank wall when reactor vessel temperature increases.

² APDA-CFE-23, Section III. B.

**FIG. 16 MOCK-UP OF CAVITY IN REACTOR
LATTICE FOR TEST OF LIGHTING FOR
HOLDDOWN FINGER INSPECTION**

**FIG. 17 MIRROR SUPPORT POST DESIGNED
TO HOLD LAMPS**

VI. MAINTENANCE

A. DISMANTLING OF OLD CASK CAR

The old fuel-handling cask car has been completely dismantled and all components containing depleted uranium have been loaded for rail shipment to National Lead Company, the purchaser, for recovery of the uranium. The components containing uranium include the main cask, the intermediate shield, the ball valve and seal flange assembly located under the cask for fuel transfer operations, and the access plug for the hoist compartment. Lead-bearing parts are to be sold as scrap. Usable items such as instrumentation and motors were reclaimed for use in other applications.

The cask car was retired from plant service because it had required an exorbitant amount of maintenance. The new fuel transfer facility is expected to be completely assembled and ready for fuel transfer service in early 1969.

B. STICKING OF GRIPPER LIFTING SUBASSEMBLY INTO STEAM CLEANING MACHINE

The gripper lifting ORB subassembly M987 became stuck in the steam-cleaning machine during a routine procedure for transfer between pots in the transfer tank. The ORB subassembly, which had been contained in a new design pot used only with the new fuel transfer facility currently in use for operator training on the new facility, was being transferred back to a conventional pot.

Figure 18 shows the position at which the gripper became stuck when raising M987 into the steam-cleaning chamber. In the transfer procedure, the subassembly is raised out of one pot completely into the cleaning chamber, and then lowered into the other pot. In the stuck position, the gripper and handling head had entered the cleaning chamber valve and the wrapper can extended downward into the access tube of the transfer tank.

The gripper with subassembly was dislodged by a 1200 pound upward pull; the normal gripper and subassembly load is about 510 pounds. The subassembly could not be delatched back in the transfer tank to free the gripper for investigation. It was then steam-cleaned, though this was not originally intended, and deposited in a shielded pot in the dry loading tunnel for investigation of the difficulties. Figure 19 shows the pot in the dry loading tunnel to hold the subassembly during the manual dismantling of the gripper. The lead shield around the pot and the gripper are shown in Figure 20. Figure 21 shows a close-up of the gripper.

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FIG. 18 SUBASSEMBLY STUCK IN STEAM CLEANING CHAMBER VALVE

**FIG. 19 POT FOR RECEIVING SUBASSEMBLY
IN DRY LOADING TUNNEL**

**FIG. 20 SHIELDED POT CONTAINING
SUBASSEMBLY**

FIGURE 21 CLOSE-UP OF GRIPPER

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The gripper push rod interlock (see Figure 18) indicator was found to be in the extended position, which would indicate a malfunctioning of the gripper actuating mechanism. An inspection of this actuating mechanism subsequent to dismantling indicated that the mechanism had functioned satisfactorily but that the interlock indicator linkage had failed. A part of the indicator mechanism had broken off and was missing.

The inability to delatch the subassembly in the transfer tank was due to the inadvertent functioning of a device within the gripper called the grapple. This device was designed to grip the handling head by slackening the gripper cable and was intended as a fool-proof way to grab and retain the handling head in event of gripper malfunction. Once the grapple engages, it cannot be disengaged remotely, thus necessitating a manual dismantling. Although effort was made to prevent it, slackening of the cable probably occurred.

Both the push rod interlock indicator and the grapple were removed from the cleaning machine gripper assembly, and M987 was grasped and deposited in the cut-up pool. A radiation value of 2.5 r/hr was observed at the subassembly behind the lead brick shield. At the outside surface of the shield the highest radiation was 18 mr/hr. The missing part of the push rod interlock indicator, a metal piece 1-1/2 inches long and about 3/8 inch in diameter, was not found. It is known by inspection that this piece is not lodged in the transfer tank floor valve, and by probing, that it did not drop in the pot in the transfer tank. It is believed to have dropped to the bottom of the sodium filled transfer tank.

ORB subassembly S510 was removed from the transfer tank, steam-cleaned, and deposited in the cut-up pool without incident. This subassembly could not be easily reinserted into the reactor lattice following a lift operation to 15 inches to verify that it was free after the incident of October 1966*. ORB S510 was removed from the reactor in January 1967.

*APDA-CFE-6, page 7

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