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NP-20047(Suppl. 1)

RETIREMENT OF THE ENRICO FERMI ATOMIC POWER PLANT

Supplement 1

Date Published—October 1975

Power Reactor Development Company
Newport, Michigan (USA)

UNITED STATES ENERGY RESEARCH & DEVELOPMENT ADMINISTRATION
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SUPPLEMENT
TO

RETIREMENT OF THE ENRICO FERMI ATOMIC POWER PLANT

POWER REACTOR DEVELOPMENT COMPANY

OCTOBER, 1975

ABSTRACT

Supplement 1 to Report NP-20047 describes the decommissioning activities performed by Power Reactor Development Company (PRDC) from March 1974 through October 1975 relating to the retirement of the Enrico Fermi Atomic Power Plant (EFAPP or Fermi-1). The major decommissioning tasks undertaken and reported are the following:

1. Disposal of blanket subassemblies
2. Disposal of radioactive sodium
3. Removal and disposition of primary and auxiliary secondary system components
4. Removal and disposal of miscellaneous contaminated equipment and materials
5. Removal and disposition of radioactivity monitors
6. Use of carbon dioxide gas (CO_2) as a sodium passivating agent in decommissioning
7. Closing and sealing contaminated areas
8. Decontamination and painting of pools.

FOREWORD

Report NP-20047 describes in detail the decommissioning activities performed by Power Reactor Development Company (PRDC) through March 1974 relating to the retirement of the Enrico Fermi Atomic Power Plant (EFAPP or Fermi-1). This supplement contains a description of the subsequent decommissioning activities performed through October 1975, at which time such activities were virtually completed.

No attempt is made herein to describe the equipment and systems discussed in this supplement because it is assumed that the reader has a general familiarity with both the design of the plant and the decommissioning activities previously reported. Should such information be needed, reference can be made to Report NP-20047 and/or to the EFAPP Technical Information and Hazards Summary Report, as amended. Report NP-20047 is available from the U. S. Energy Research and Development Administration, Office of Public Affairs, Technical Information Center, Oak Ridge, Tennessee.

Questions concerning decommissioning activities, current status of the plant, or the availability of plant components, as well as the availability of prints or copies of slides contained in the photographic history of decommissioning, should be referred to the Director, Generation Engineering Department, The Detroit Edison Company, 2000 Second Street, Detroit, Michigan 48226.

EXECUTIVE SUMMARY

Decommissioning of the Fermi-1 plant was initiated on November 2 , 1972, when the PRDC Executive Committee made the difficult decision to discontinue the Fermi-1 project because of its inability to fund fully a proposed 6-year oxide core program.

Retirement of the plant was accomplished through provisions of 10CFR 50.59, "Authorization of Changes, Tests and Experiments," and through a series of technical specification changes where required. The plan consisted of the following major elements:

1. Shipping all fuel and blanket elements offsite. Both were accepted by the AEC under the reprocessing provisions of 33 FR 30
2. Shipping all bulk nonradioactive sodium offsite and passivating the residual sodium
3. Disposing of all other contaminated or irradiated materials by shipping offsite except for material in restricted areas, access to which is limited to authorized personnel
4. Securing some of the reactor building electrical, instrumentation, piping, ventilation, personnel and equipment penetrations
5. Draining and sealing the primary system, comprised of the reactor vessel, primary sodium piping, primary shield tank, machinery dome, primary sodium service, and secondary sodium system out to welded pipe caps, and passivating the residual sodium therein
6. Revising the site boundary to a very limited area
7. Implementing a postretirement surveillance plan
8. Leaving the turbine generator, which is owned by The Detroit Edison Company, in place for continued service with steam supplied from an oil fired boiler.

The decommissioning program in accordance with the above plan was substantially completed by March 1974 when Report NP-20047, "Retirement of the Enrico Fermi Atomic Power Plant," was prepared. At that time, the principal items remaining to be done were (1) the disposition of the blanket elements and the primary sodium, (2) sealing the primary system, and (3) retirement of the fuel and repair building (FARB) systems.

This supplement describes the decommissioning activities performed from March 1974 through October 1975, at which time the program was virtually completed. The major activities undertaken during this period and reported are as follows:

1. Disposal of blanket subassemblies by segmenting some of them and shipping all the active portion to the Idaho Chemical Processing Plant and shipping the nozzles for burial elsewhere

2. Disposition of radioactive sodium by retaining it on the plant site until the Clinch River Breeder Reactor Plant is in a position to use it

3. Removal and disposition of primary and auxiliary secondary system components by donation, sale, or burial offsite

4. Removal and disposal of miscellaneous contaminated equipment and materials by decontamination and sale or by segmenting and burial offsite

5. Removal and disposition of radioactivity monitors by donation or sale

6. Use of carbon dioxide gas as a sodium passivating agent in decommissioning to prevent buildup of hydrogen in tanks and pipes due to the reaction of sodium with moisture in the atmosphere

7. Closing and sealing contaminated areas to contain residual contamination within these areas

8. Decontamination and painting of pools to eliminate removable surface contamination.

With respect to surveillance of the retired plant, the moisture detectors in the biological shield wall cavity in the reactor building and in the FARB hot sump were modified to respond to water level, while the moisture monitor in the lower reactor building remains intact.

Environmental sampling will be done as scheduled in the current plant Technical Specifications approved by the AEC in July 1974. Background samples include those from raw city water, river water, and river sediment. PRDC management believes that the sampling should be discontinued no later than 1 year after liquid waste discharges cease and certainly following the removal of the primary sodium from the site. Liquid waste discharges from the plant were not detectable by environmental monitoring during plant operation and decommissioning. Gaseous waste discharges were discontinued in October 1973, and air and precipitation monitoring was discontinued following the approval of a Technical Specification change in July 1974.

With the decommissioning essentially completed and funds provided for the future disposal of sodium, the total cost is now expected to be $6,940,000, excluding surveillance. An increase in blanket disposal costs and a decrease in sodium disposal costs partially offset each other. Completion of decommissioning activities took longer than estimated, but the additional expense was offset by savings made from the rapid reduction of work forces as personnel could no longer be effectively used, with the final cleanup in 1975 being accomplished by a very small group. Retiring buildings and equipment as soon as they were not needed resulted in substantial savings in property tax, insurance, and other expenses. The total cost of decommissioning has remained essentially as it was budgeted in 1973. An escrow fund is being established to cover the surveillance and custodianship of the site.

Most of APDA's Technical files and some PRDC technical files were transferred to Overseas Advisory Associates, Inc. The remaining files, including the PRDC and APDA corporate files and the decommissioning history slide file, are being consolidated in the former upper fuel vault in the FARB. Drawing files remain intact on the first floor of the office building for future reference as needed. The Detroit Edison Company will become custodian of these files when the EFAPP site is returned to Edison's control at the end of 1975.

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LIST OF ILLUSTRATIONS

LIST OF ABBREVIATIONS

AB	Axial Blanket
AEC	U. S. Atomic Energy Commission, divided into ERDA and NRC in 1975
ANL	Argonne National Laboratory
APDA	Atomic Power Development Associates, Inc
Ci	Curie(s)
CRBRP	Clinch River Breeder Reactor Project
DLT	Dry Loading Tunnel
d/m	Disintegrations per minute
EFAPP	Enrico Fermi Atomic Power Plant
ERDA	Energy Research and Development Administration
F	Degrees Fahrenheit
FARB	Fuel and Repair Building
FBC	Fuel Basket Container, terminology used in Appendixes IV and VI for Subassembly Basket Container, which was designed to accept fuel
Fermi-1	Enrico Fermi Atomic Power Plant
Fermi-2	Detroit Edison's Fermi-2 Project, having a boiling water reactor as the heat source
FTF	Fuel Transport Facility
ft^2	Square foot (feet)
ft-lb	foot-pound(s)
gm	gram
HEDL	Hanford Engineering Development Laboratory
HFEF	Hot Fuel Examination Facility (ANL/Idaho Div)
hr	Hour
IBM	International Business Machines Co
ICPP	Idaho Chemical Processing Plant
IHX	Intermediate Heat Exchanger

IRB	Inner Radial Blanket
LMEC	Liquid Metal Engineering Center
MPC	Maximum Permissible Concentration
μCi	Microcurie(s)
mCi	Millicurie(s)
mm	Millimeter(s)
mr	Milliroentgen(s)
ORB	Outer Radial Blanket
PCV	Pressure Control Valve
PMC	Project Management Corporation
PNC	Power Reactor and Nuclear Fuel Development Corporation (Japan)
PRDC	Power Reactor Development Company
PST	Primary Shield Tank
R	Roentgen(s)
RE	Radiation Element
w	watt(s)

1.0 DECOMMISSIONING ACTIVITIES

1.1 SUPPLEMENTAL DECOMMISSIONING SCHEDULE

A schedule for the major decommissioning activities performed since March 1974 is given in Table 1.

1.2 DISPOSAL OF BLANKET SUBASSEMBLIES

1.2.1 Development of Disposal Plan

The former U.S. Atomic Energy Commission (AEC), divided in 1975 into the Energy Research and Development Administration (ERDA) and the Nuclear Regulatory Commission (NRC), agreed to accept ownership of the Fermi-1 blanket material under the hypothetical reprocessing provision of 33 FR 30. The estimated cost at the time the contract was negotiated was $1.2 million. The contract, however, had escalation provisions which included a significant dependency on the basic inorganic chemical index which, in turn, is strongly dependent on the price of oil. Mainly because of significant increases in this index, the final payment made to ERDA on June 27, 1975, was $1,594,073.

Subsequent to the general agreement for ERDA to accept the Fermi-1 blanket material and for PRDC to deliver it to the Idaho Chemical Processing Plant (ICPP), several meetings were held to discuss where and how it was to be stored, and in what containers and casks it would be delivered. The disposal and shipment of the blanket, as described in the following section, was accomplished in accordance with the verbal agreement and the specific terms of the ERDA/PRDC contract.

1.2.2 Blanket Segments

Disposal of all blanket subassemblies, some of which were purposely segmented, was accomplished by shipment to ICPP. This involved 962 subassemblies or segments containing a total of 6524 grams of Pu-239 which were shipped in 14 cask loadings between December 17, 1974, and April 18, 1975, as shown in Table 2. Of the 962 subassemblies or segments, 318 were uncut outer radial blankets, 168 were cut outer radial blankets, 73 were cut inner radial blankets, 202 were upper axial blankets, 132 were cut lower axial blankets, and 69 were uncut lower axial blankets. The term "cut" indicates that the nozzle was removed from the subassembly.

In addition to the items mentioned above, the shipments also included (1) one high power oscillator rod, (2) 80 loose axial blanket pins contained in five 30-inch-square cans that had been heliarc welded shut in the pool just above the water's surface, and (3) 14 loose blanket pins in a 30-gallon drum. The drum, a field-fabricated shipping cask, contained lead brick shielding on the bottom supporting a 6-inch-diameter pipe surrounded by concrete. The 80 axial blanket pins and the other 14 pins were released when misalignment of the cutting machine head resulted in cutting through the pin support grids of several axial blanket segments, as well as when segments were dropped to the floor of the cutup pool, as described on page 27. They were retrieved from the pool floor using a field-fabricated mechanical coil spring retriever. Prior to canning, they were stored in an underwater basket hanging from the pool wall.

All blanket shipments were made using two Philadelphia Electric

TABLE 1 SUPPLEMENTAL DECOMMISSIONING SCHEDULE

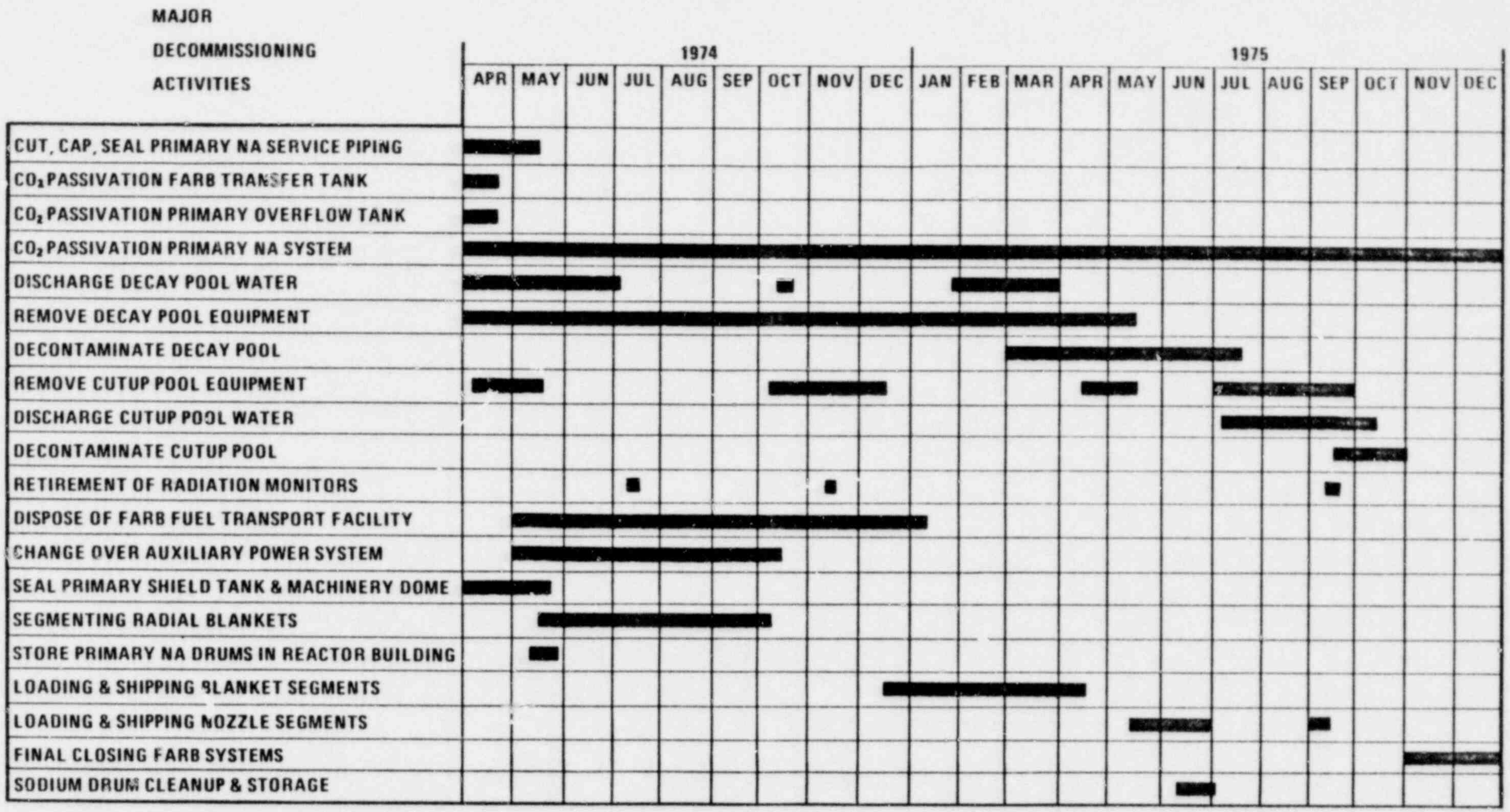


TABLE 2

BLANKET SHIPMENTS

Shipment	Date	Uncut ORB	Cut ORB	IRB	UAB	LAB	Uncut LAB
1	12/17/74	27	20	-	-	-	-
2	1/3/75	27	20	-	-	-	-
3	1/10/75	27	20	-	-	-	-
4	1/23/75	27	20	-	-	-	-
5	1/24/75	27	20	-	-	-	-
6	2/5/75	27	20	-	-	-	-
7	2/7/75	27	20	-	-	-	-
8	2/17/75	27	-	20	-	-	-
9	2/24/75	27	-	20	-	-	-
10	3/10/75	27	-	20	-	-	-
11	3/14/75	27	7	13	-	-	-
12	3/27/75	21	21	-	-	-	-
13	4/10/75	-	-	-	165	53	12
14	4/18/75	-	-	-	37	79	57
		318	168	73	202	132	69

Cut = Nozzle removed
ORB = Outer Radial Blanket
IRB = Inner Radial Blanket
UAB = Upper Axial Blanket
LAB = Lower Axial Blanket

Company Model PB-1 shipping casks, one of which is owned by the General Atomic Company. To utilize these casks to handle Fermi-1 blanket subassemblies, the Columbus Ohio Laboratories of Battelle Memorial Institute designed 14 baskets and basket containers, which were then fabricated by the Central Ohio Welding Company under Battelle's supervision. A sketch of the subassembly basket and basket container inside the cask is given in Figure 1. The subassembly baskets were constructed of carbon steel gridwork, sized to fit into a cylinder 25 inches O.D. x 158 inches long, containing 37 individual storage holes. The basket containers were fabricated under rigid specifications and quality assurance provisions and were made of Type 316 stainless steel, 1/4 inch thick in the walls and with a 3/4-inch-thick top and bottom.

Prior to the first loading for shipment, approximately five dry runs were made to develop handling procedures and to establish time requirements. These runs essentially were complete simulations except that no subassemblies were loaded and no welding was done. Immediately preceding each of the 14 loadings, a dimensional check for fit was made by placing an empty basket into the basket container and then placing the container into the cask.

Before cutting the nozzles from some of the radial blanket subassemblies, required prior to shipment, the subassemblies had to be removed from the stack and put into an upright position with the one-piece gripper. Then each subassembly could be transferred to the cutup machine with the tong tool (see Figure 15) for segmenting. Subsequently, using the tong tool, the blanket segment could be transferred to the shipping basket and the nozzle segment transferred to a storage basket.

The first step in loading a cask for shipment required placing an empty basket into the cutup pool and loading it with subassemblies or segments. Because of an unloading procedure at ICPP*, it was necessary to ensure that subassemblies or segments could not move more than 1/2 inch axially. For this purpose, a 2-inch-diameter, carbon steel mill tube of appropriate length weighing approximately 0.735 lb/ft was placed over each subassembly or segment as a spacer. The length of the spacers varied from about 5 to 62 inches, depending on the stacking arrangement below the spacers. The spacers were field cut to fit, using a 30-ft measuring pole inserted into each basket storage hole and referenced to an elevation mark on the pool bridge crane. A loaded basket was then raised from the pool and allowed to dry overnight. Radiation levels up to 50 R/hr at the basket surface necessitated isolation of the building and rigid health physics controls on personnel while the basket was unshielded. Crane operators worked behind portable concrete shielding panels using mirrors to observe the transfer of baskets from the cutup pool to the cask loading area. After a loaded basket was lowered into the container in a cask, a lid was placed on the basket container. A rotating lead welding shield was then placed on top of the cask to permit welding the lid in what otherwise would have been a high radiation area. A certified welder using certified welding rod then seal-welded the container. A helium mass spectrometer leak test of the container weld followed, prior to closing the cask for shipment. A detailed procedure for loading blanket subassemblies and segments for shipment is given in Appendix IV.

Actual photographs of some of the equipment and loading operations described in the preceding paragraphs are shown in Figures 2 through 8.

*For loading at the Fermi plant, the top of the cask is as shown in Figure 1. At ICPP the bottom of the cask shown in Figure 1 becomes the top of the cask for unloading.

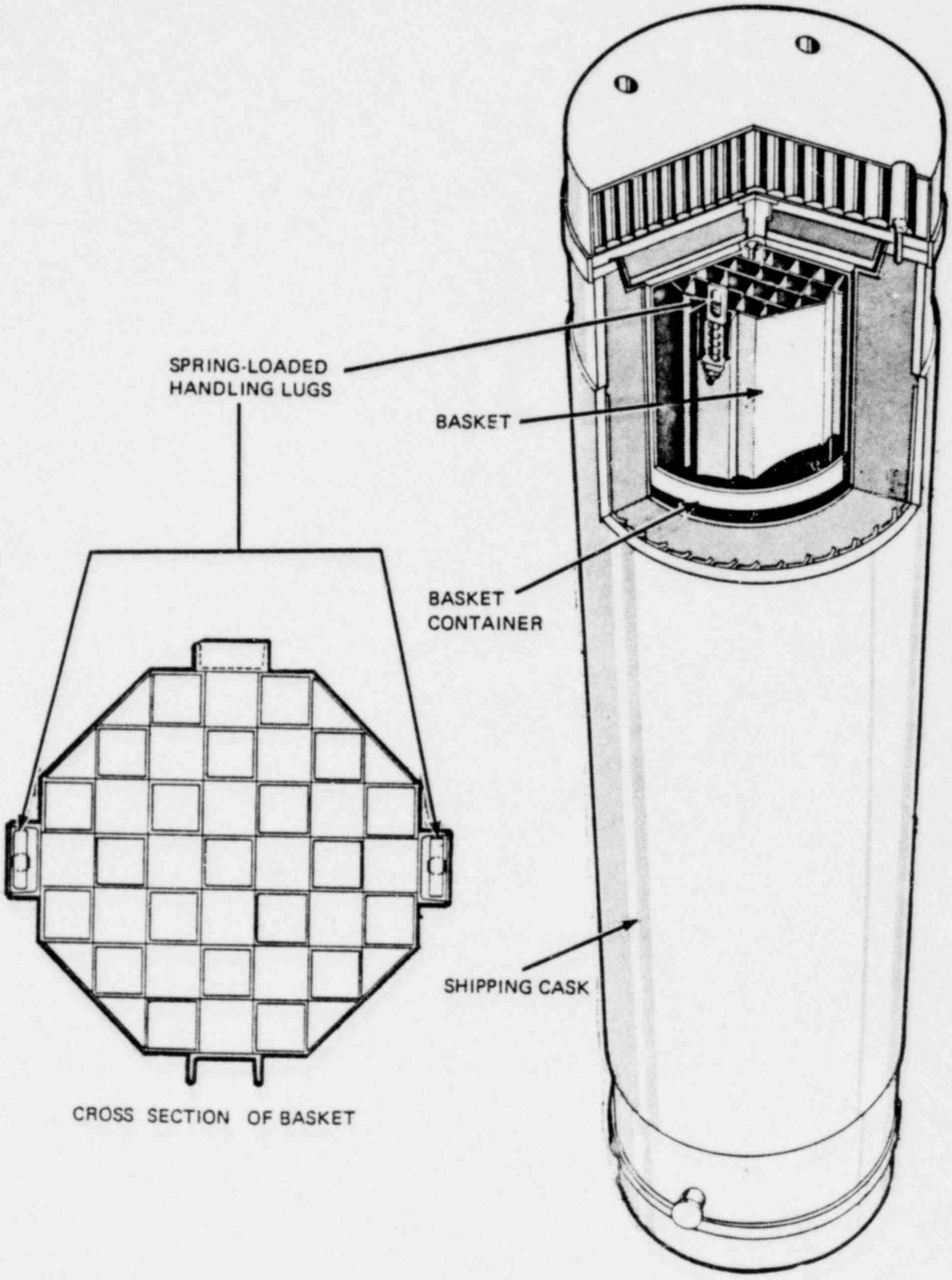


FIG. 1 SUBASSEMBLY BASKET, CONTAINER, AND SHIPPING CASK

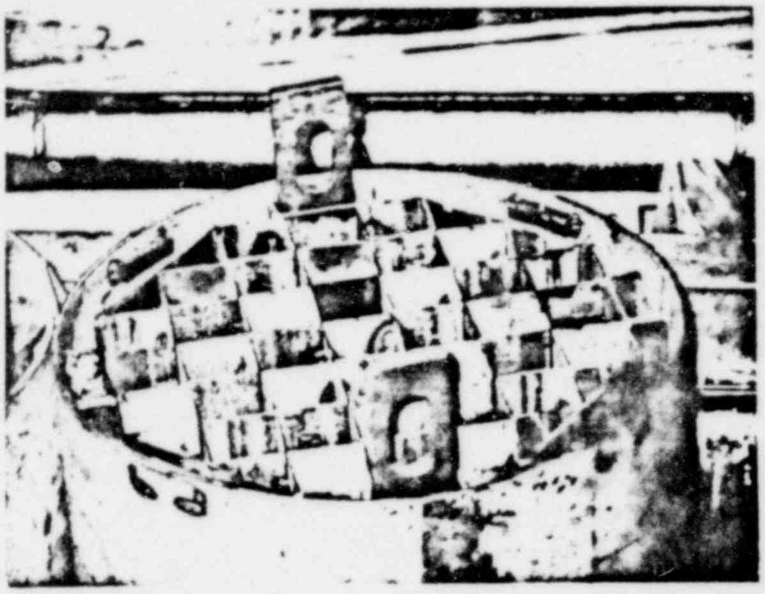


FIG. 2 SUBASSEMBLY BASKET IN CONTAINER

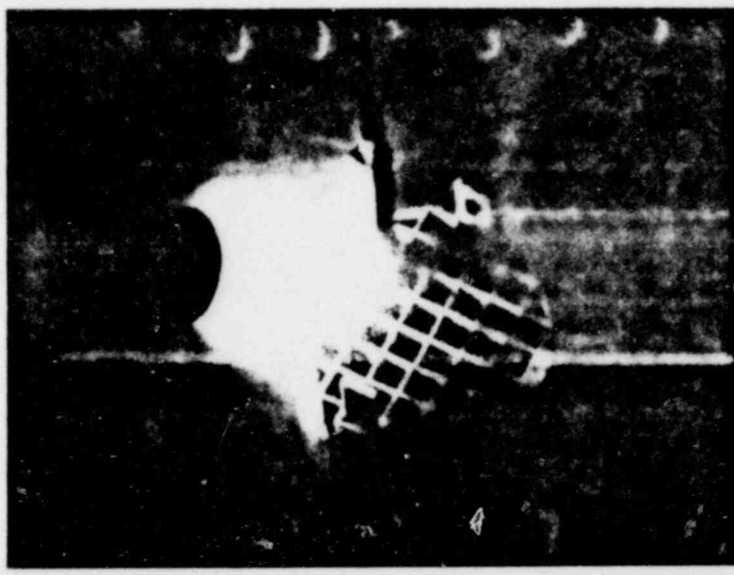


FIG. 3 INSERTING "CUT" BLANKET SEGMENT INTO BASKET USING THE HYDRAULIC FLUSHER (BAG) TOOL (UNDERWATER)

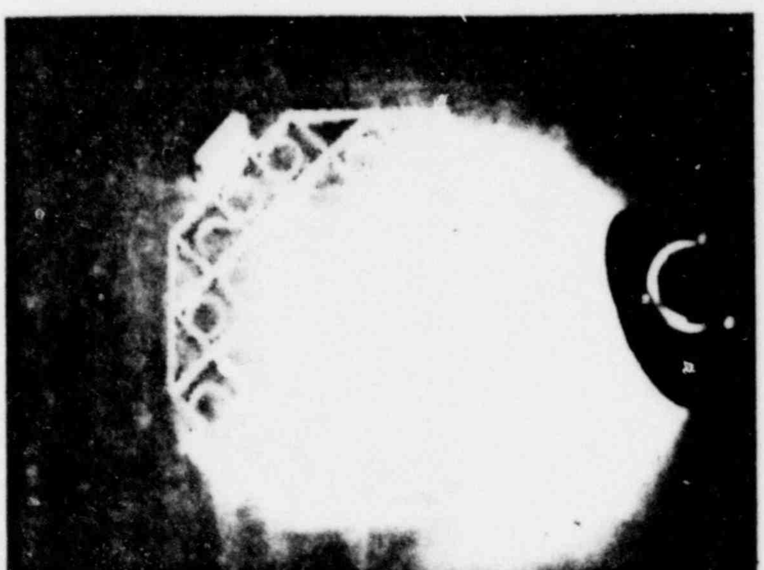


FIG. 4 BASKET WITH SPACER TUBES VISIBLE (UNDERWATER)

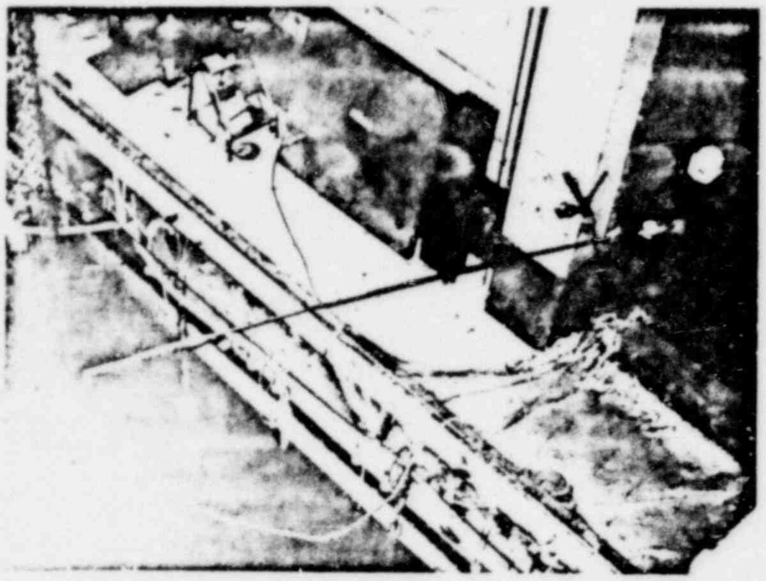


FIG. 5 HEALTH-PHYSICS PERSONNEL MONITORING REMOVAL OF LOADED BASKET FROM POOL USING A TELESCOPING TELTECTOR (EBERLINE INSTRUMENT CO MODEL 6112B)

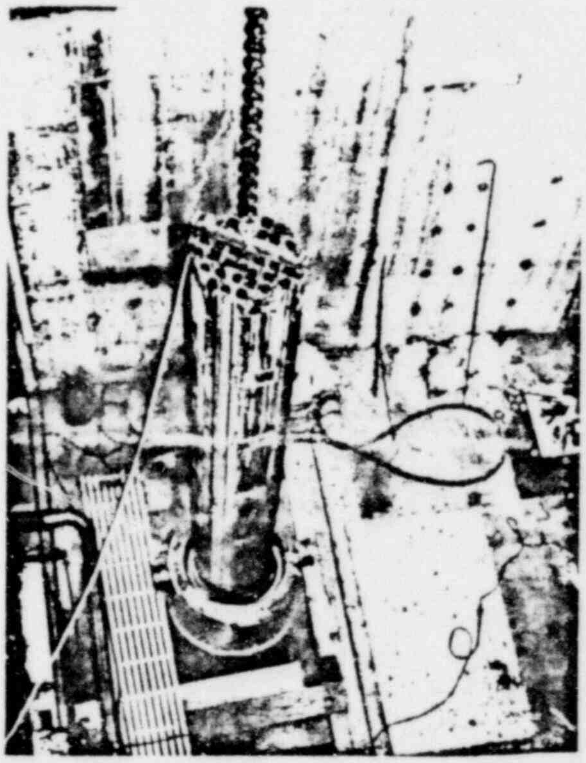


FIG. 6 LOADED BASKET TRANSFERRED FROM CUTUP POOL TO CASK

FIG. 7 SEAL-WELDING THE BASKET CONTAINER IN CASK

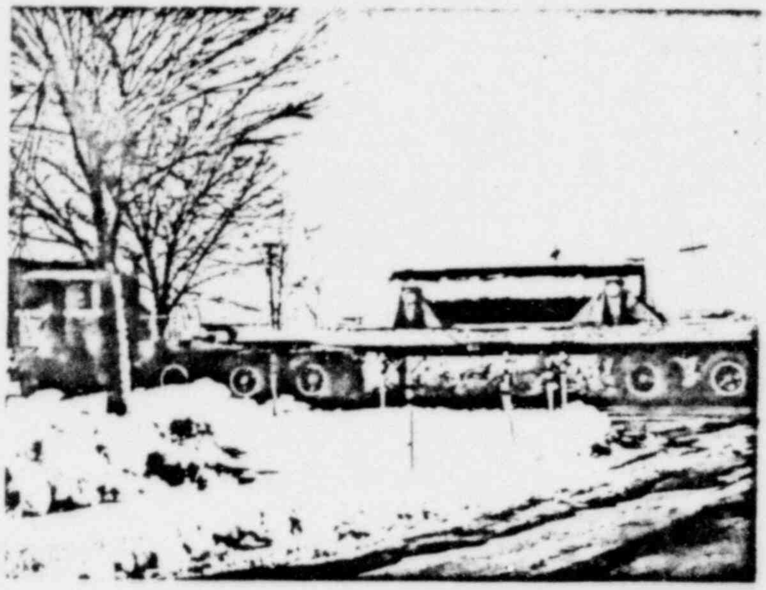


FIG. 8 SHIPPING CASK LEAVING FERMI PLANT

FIG. 9 BLANKET SEGMENT SPILL ON CUTUP POOL FLOOR (UNDERWATER)

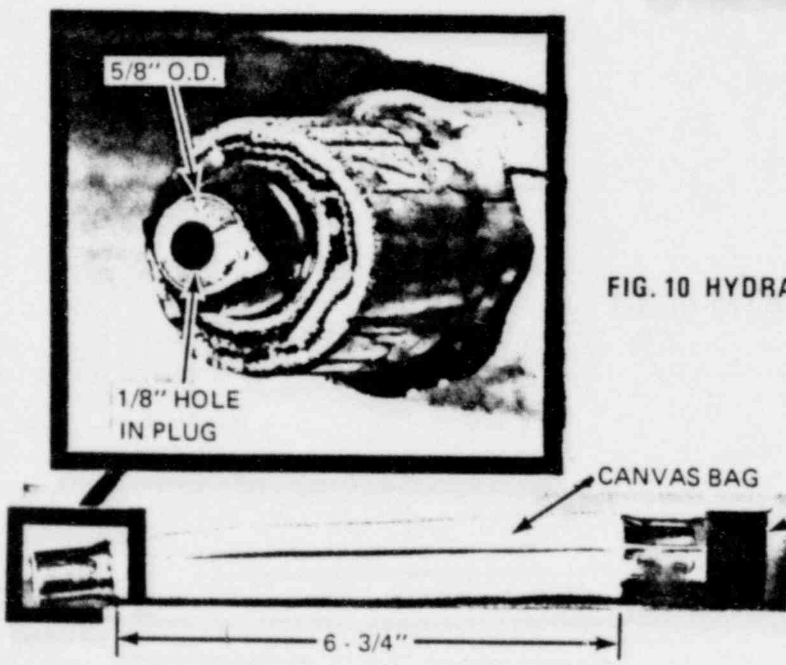


FIG. 10 HYDRAULIC FLUSHER - BAG TOOL

Two of the 14 shipments consisted mostly of axial blanket segments. Two hundred one (201) were lower axial blanket segments which were more difficult to load because they had no handling head or other suitable gripping feature. In the process of loading these headless segments, the bottom supports of the basket failed, dropping several segments to the pool floor. The basket was unloaded by inverting and dumping the remaining segments to the pool floor, resulting in some axial blanket pins becoming jarred loose and having to be separately canned as noted previously. A view of the segment spill on the pool floor is shown in Figure 9.

The bottom supports of the damaged basket were replaced with much sturdier support members. A second basket for the same service was similarly modified before use. A unique handling tool was subsequently devised to grip and gently lower each lower axial blanket segment into the basket. This tool is an adaptation of a commercially available device known as a hydraulic flusher, which consists of a special cloth bag normally used by plumbers for blocking back flow in a pipe by expanding the bag with water pressure. Water discharged from the bag flows forward under full pressure to clear the pipe of obstructions. In the Fermi-1 application the tool was modified by almost closing the outlet hole in the bag, as shown in Figure 10. It was then mounted on a long pipe connected to a water supply by a flexible hose. The pipe served as a handle for transferring segments underwater after the deflated bag was inserted between pins or in the nozzle of the axial blanket section and expanded with water pressure to grip the unit. Thus, by gripping segments internally, they could be lowered into a basket to the desired elevation before releasing. Closing off the water supply released the unit. Previously, the segments were gripped on the outside, necessitating release near the top of the basket and resulting in drops of up to several feet to reach the desired elevation. The original supports across the bottom of the basket had been designed to carry adequate weight but not to absorb the impact of dropping. The problem was not evident earlier because the upper axial blankets were loaded first, taking advantage of their handling heads for gripping and lowering them into shipping baskets.

No problems other than the resulting loose pins and several days' delay resulted from the basket damage. There were no radiation exposure incidents, and contamination levels were small. Several pins which were not located by the time the last shipment was made to Idaho were shipped to Nuclear Engineering Company's Sheffield, Illinois, burial site on September 23, 1975.

1.2.3 Nozzle Segments

In order to permit loading a greater quantity of blanket and fuel material into each shipping cask, thereby reducing the number of required shipments, it was decided to cut the nozzles off many subassemblies and to ship the nozzles separately. An inventory of the 379 nozzles prepared for shipment in seven containers is as follows: 132 from the lower axial blanket section of core subassemblies, 168 from outer radial blanket subassemblies, and 79 from inner radial blanket subassemblies. The first six containers, which also included five empty wrapper cans and three each of lower and upper sections cut from sodium worth assemblies* having no axial blanket pins, were shipped to the Nuclear Engineering Company in Morehead, Kentucky, in June 1975. The seventh basket was temporarily retained for additional use, as noted on page 28.

*special subassemblies for sodium physics measurements

Disposal of the nozzles was accomplished by transferring them from a large field-fabricated basket in the cutup pool to the seven field-designed and field-fabricated smaller baskets for shipment. The small baskets were made of expanded metal reinforced with flat bar stock and had a rod across the top for lifting. These baskets, approximately 39 inches high and 23 inches in diameter, were designed for use in the Nuclear Engineering Company's B-3 cask liner.

In accordance with the established loading program, nozzle baskets were submerged in the cutup pool and were loaded with approximately 57 nozzle assemblies each. Then B-3 cask liners (also called nozzle basket containers) were placed beside the pool. Each basket was raised from the pool, allowed to dry, and then placed into a cask liner. The liner and basket assembly were then transferred temporarily to the FARB cold trap room for short-term storage. There each cask liner cover was installed using a special tool with an 18-foot handle to reduce personnel exposure. A 2000-pound-test aircraft cable secured to the lifting lugs of each liner was used for transferring the loaded liners to the shipping casks. Photographs of these operations or the equipment used are shown in Figures 11, 12, and 13.

Three principal tools were used for the underwater handling of nozzles: (1) the tong tool previously used to handle core subassemblies; (2) an S-shaped hook that is inserted into the nozzle; and (3) the hydraulic flushers described in Section 1.2.2. Photographs of these tools in use, as well as several sketches, are given in Figures 14, 15, and 16.

The seventh nozzle basket was stored in the FARB cold trap room until the metallic chips were recovered from the floor of the cutup pool. Virtually all the chips were retrieved by vacuuming the pool floor while it was under a sufficient depth of water to serve as shielding. These chips were placed in the last B-3 cask liner and, with the seventh basket, were shipped to Morehead, Kentucky, on September 4, 1975. A diaphragm pump with a long suction hose was used to pick up the chips. The pump discharged into the top of a 30-gallon carbon steel drum having a 30 x 30 mesh stainless steel filter and having 1/2-inch drain holes in its sides and bottom. A few additional chips were observed on the floor of the pool after the water level was lowered to about 1 foot. These chips were retrieved with the diaphragm pump discharging into a bag filter. The filter bag, which was placed in a 55-gallon drum with other contaminated material, was shipped to Morehead, Kentucky, on October 20, 1975, in Nuclear Engineering Company's B-2 cask.

1.3 DISPOSAL OF IRRADIATED OR CONTAMINATED MATERIAL AND EQUIPMENT

1.3.1 Liquid Waste

During April, May, and June 1974 there were 4.359×10^5 liters of liquid waste with a total activity of approximately 50.0 millicuries dis charged to the north lagoon. From July 1, 1974 through June 30, 1975 there were 1.274×10^5 liters of liquid waste with a total activity of approximately 141.2 millicuries discharged to the north lagoon. All effluents released to the environment, after dilution with the circulating pump discharge, were below 1/30 of MPC.

The decay pool contained approximately 118,000 gallons of radioactive water. This was disposed of through the FARB liquid waste discharge system. Most of the liquid was discharged at 10 gallons per minute.

FIG. 11 INSTALLING NOZZLE BASKET IN CUT-UP POOL (UNDERWATER)

FIG. 12 NOZZLE CONTAINER FOR B-3 SHIPPING CASK

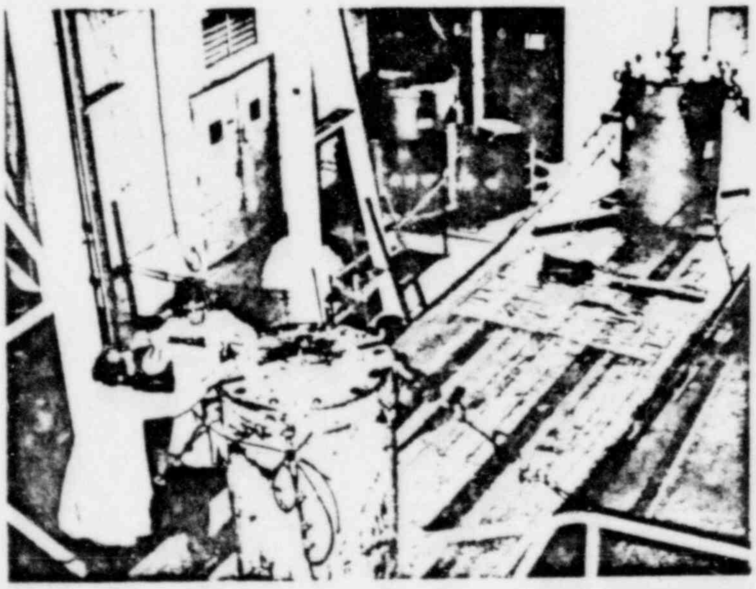


FIG. 13 NOZZLE SHIPPING CASKS (B-3/DOT-6058)

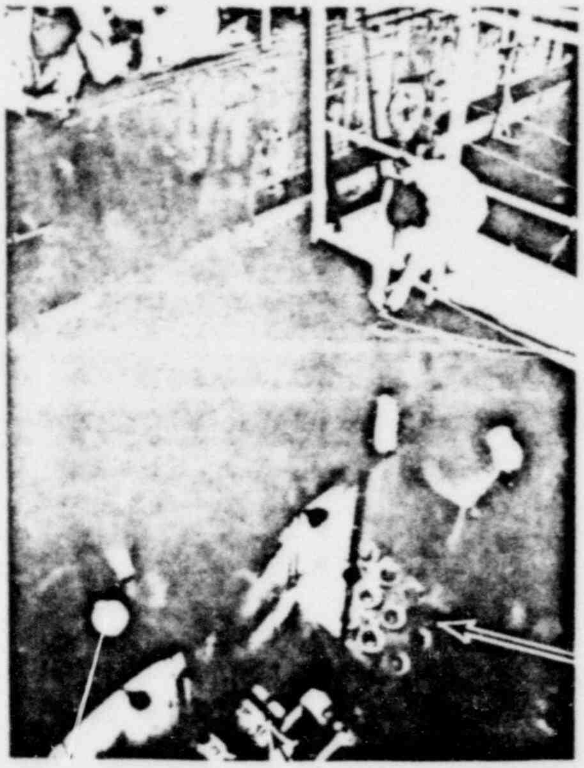
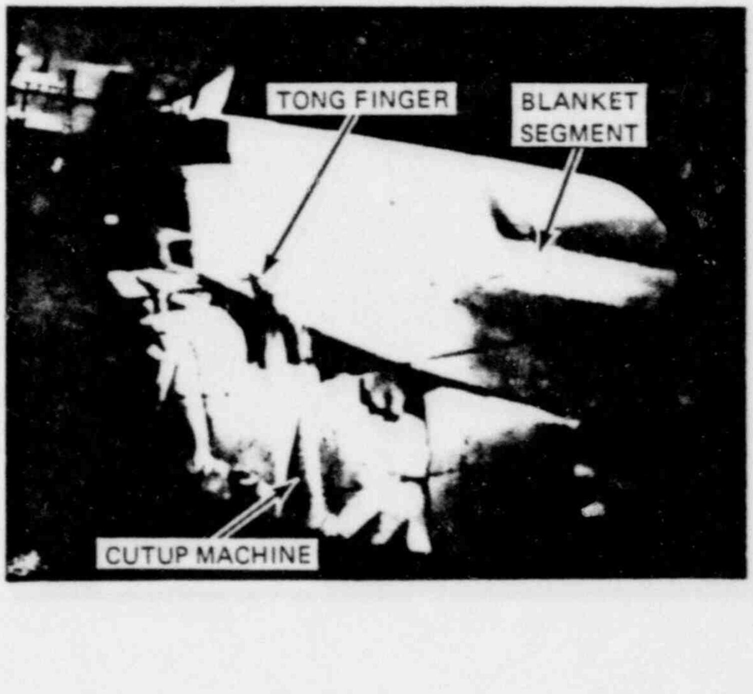
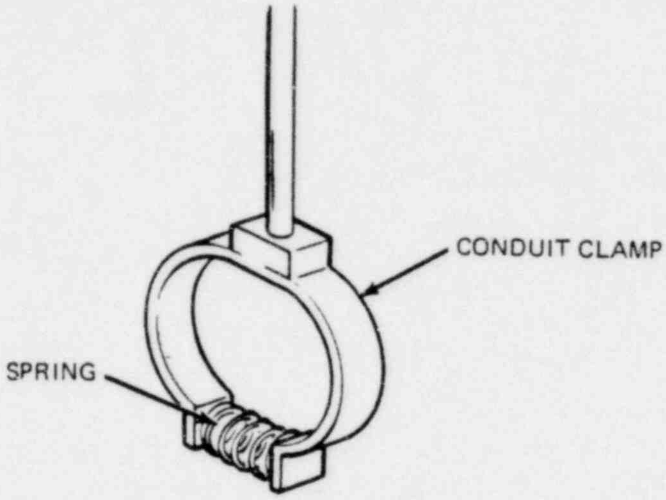


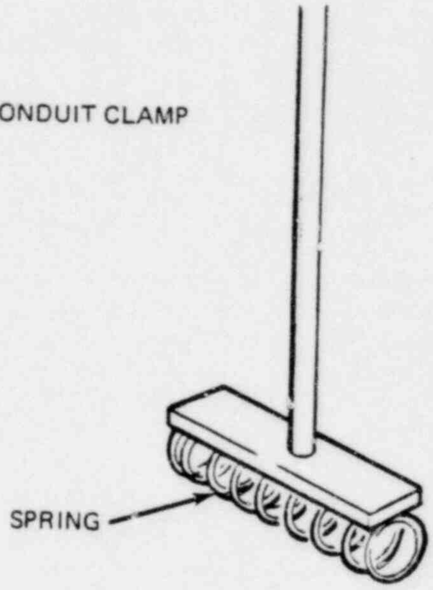
FIG. 14 TONG TOOL BEING USED TO TRANSFER SUBASSEMBLY (UNDERWATER)

FIG. 15 TV PICTURE OF TONG TOOL BEING USED TO RAISE BLANKET SEGMENT (UNDERWATER)

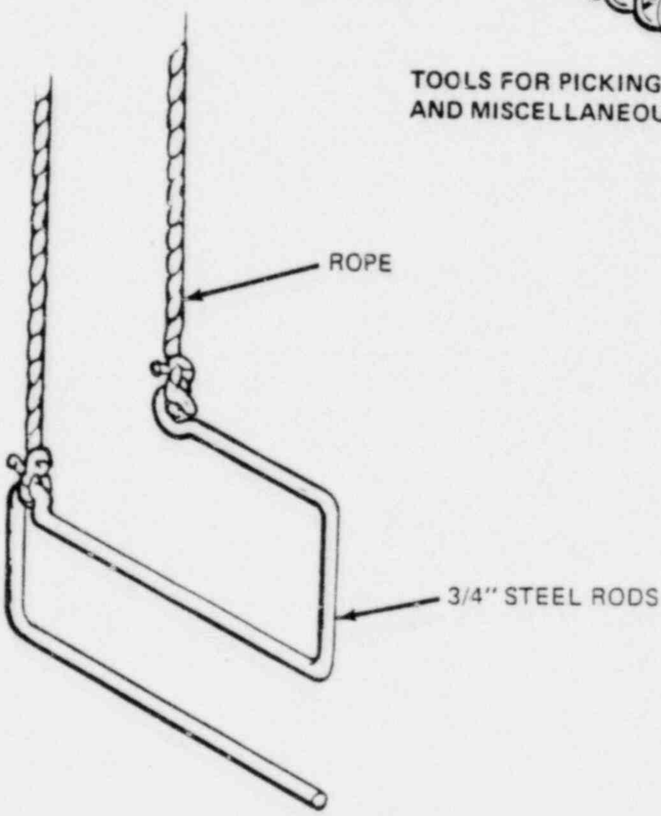




TOOL FOR PICKING UP BLANKET PINS
AND MISCELLANEOUS SMALL OBJECTS



TOOLS FOR PICKING UP STRAINERS
AND MISCELLANEOUS SMALL OBJECTS



S HOOK FOR LIFTING CUT LOWER SEGMENTS

FIG. 16 EXAMPLES OF FIELD DESIGNED AND FABRICATED MISCELLANEOUS TOOLS

The bottom 11 inches (3500 gallons) was discharged at 1/4 gallon per minute because of the higher level of dissolved activity.

The cutup pool contained approximately 83,000 gallons of radioactive water that was all disposed of at 10 gallons per minute from the liquid waste discharge system. Because of the extensive efforts to filter the particulate matter from the cutup pool, it was not necessary to lower the discharge rate when the water level approached the bottom of the pool. The filtering effort is described in Section 1.8.2.2.

1.3.2 Equipment Sold or Scrapped

1.3.2.1 Underwater Cutup Machine

The cutup machine, shown in Figures 17 and [illegible], was removed from the cutup pool, washed over the pool surface with high-pressure demineralized water, and hung over the pool until it was dry. A health physics survey indicated an activity reading under the inner cutting head of 2 R/hr at 1 centimeter. Because the machine could not economically be decontaminated, it was prepared for shipment as contaminated material by wrapping it in plastic and boxing it in a specially prepared shipping container.

Because this machine represented a unique concept for underwater segmenting, a special effort was made to find a use for it. It was shipped in July 1975 to the Idaho Chemical Processing Plant (ICPP), together with the control console, electrical switchgear, oil pump and reservoir, hydraulic hoses, and several cutting blades. The machine was given to ICPP, however, the expense of shipping it to Idaho and part of expense of the partially completed new blades were borne by that organization. The tipping fixture and chip collector were buried as contaminated waste because there was no use for them.

1.3.2.2 Shipping Casks

Shipping casks PRDC-1 and PRDC-2 were sold to Argonne National Laboratory, Idaho Division in May 1975 for $60,000 f.o.b. the Fermi-1 plant. Each, cask, including the mounting skid and impact limiters, weighed approximately 36,000 pounds. Fabrication costs for the 2 casks, including the later addition of impact limiters and a copper-cadmium liner for PRDC-1, totaled approximately $140,000. These casks are to be modified for use in Argonne's Hot Fuel Examination Facility (HFEF) in Idaho.

1.3.2.3 Miscellaneous Items

Other miscellaneous items sold, donated, or scrapped are listed below. These items were decontaminated where necessary and shipped "as is." The first five items listed were donated to Westinghouse Hanford Company, which paid for packaging and shipping, as well as a nominal value for the salvageable lead in both exit port floor valves.

- NaK heat exchanger and miscellaneous piping for primary sodium service system
- Four 3-inch sodium valves from the primary cold trap system
- Subassembly gripper for steam cleaning machine

FIG. 17 CUTUP MACHINE IN CUTUP POOL (UNDERWATER)

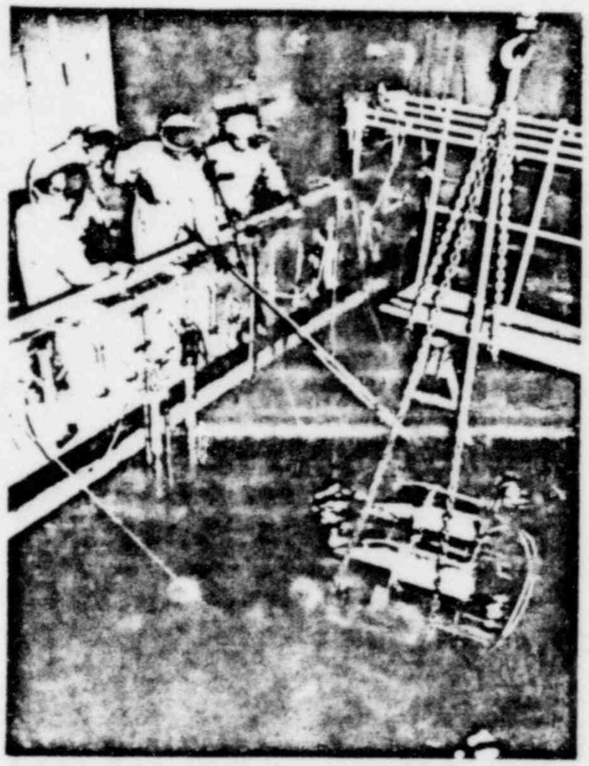


FIG. 18 DECONTAMINATING CUTUP MACHINE USING HIGH-PRESSURE WATER IN THE CUTUP POOL

FIG. 19 CONTAINERS AWAITING USE FOR PACKAGING CONTAMINATED SCRAP

- Reactor exit port floor valve for fuel transport facility (FTF)
- FARB transfer tank exit port floor valve for FTF
- Transfer pots for fuel and special components
- Finned pot support rack and shield assembly for dry loading tunnel (DLT)
- Subassembly dimensional gaging fixture
- Two data loggers for fuel handling activities

1.3.3 Components and Scrap Buried

The items listed below were buried in Nuclear Engineering Company's facilities at either Morehead, Kentucky or Beatty, Nevada. In most cases these items were drained, washed with demineralized water, segmented, wrapped in plastic, and put into a plastic-lined box. Photographs of packaged contaminated scrap ready for shipment are shown in Figures 19 and 20.

1.3.3.1 Resin Tanks

There were nine contaminated resin tanks installed in the cutup pool for removing contamination from the water in either the cutup or decay pool. These tanks were mounted in 15 feet of water in a rack at the south end of the pool. Each tank, containing about 3 cubic feet of contaminated resin, was connected to the ion exchange system by flexible metal hoses with quick disconnects. These tanks were removed from the system, as illustrated in Figure 21, and were connected to a 35-psi air supply for dewatering. Weighing before and after demonstrated that the tanks had been successfully dewatered for shipment. The tanks, still containing resin, were then wrapped in plastic for shipment in Nuclear Engineering Company's B-2 cask to Morehead, Kentucky, on July 9, 1975. Figure 22 shows two of the resin tanks being lowered into the shipping container located on a flat bed truck.

1.3.3.2 High-Power Oscillator Rod

The high-power oscillator rod, which contained the isotope B-10 in the form of the carbide (B_4C), was stored in the cutup pool after removal from the reactor in September 1971. After ICPP agreed to accept the rod, it was sent there in blanket shipment Number 12 on March 27, 1975. Because the rod was highly radioactive (400 R/hr on contact), it was placed into a center storage position of the basket to obtain maximum shielding.

1.3.3.3 Miscellaneous Items

In the second quarter of 1974, three shipments of radioactive waste were made: two totaling 1332 cubic feet went to Nuclear Engineering Company's Beatty, Nevada, site, and one totaling 460 cubic feet went to the Morehead, Kentucky, site for burial. In the same quarter, one shipment of sodium-filled piping containing approximately 52 microcuries of Na-22 was sent to Atomics International at its request and at its expense for removal, packaging, and delivery.

On October 1, 1974, 1123 cubic feet of contaminated material and equipment, containing 2.4 mCi of Cs-137 and Co-60, were shipped to Morehead, Kentucky. Included in the shipment were the cutup sections of the

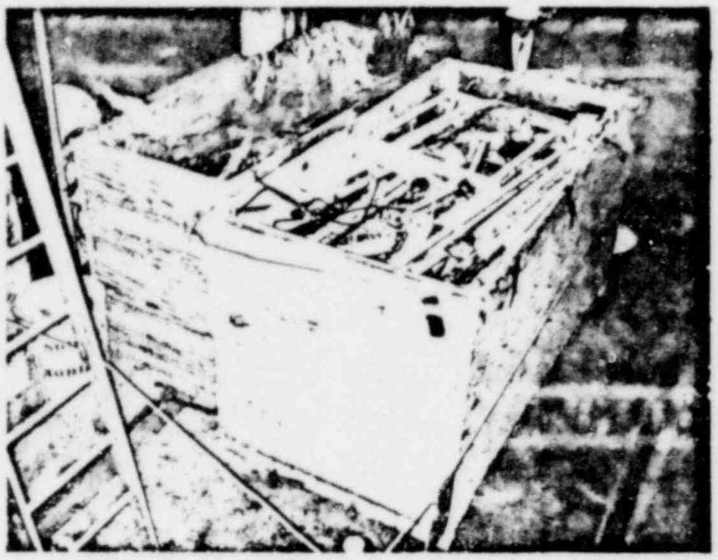


FIG. 20 CONTAMINATED SCRAP BEING BOXED FOR SHIPMENT

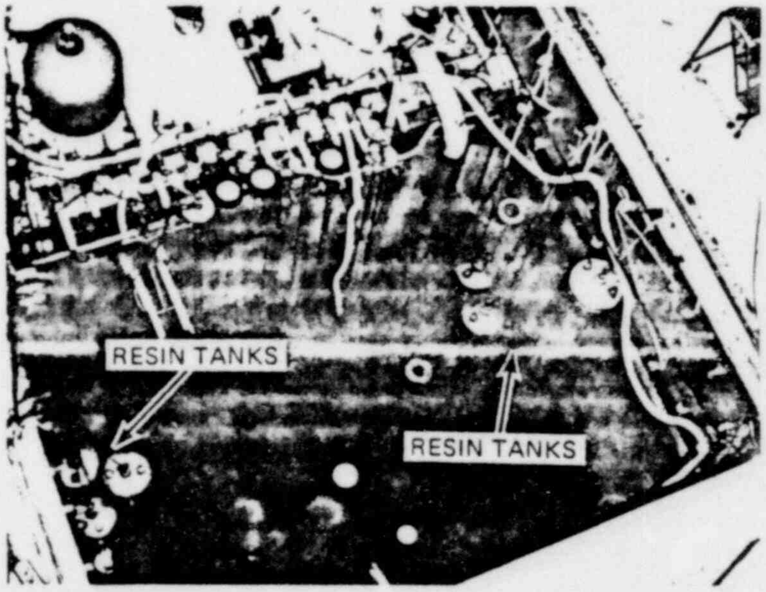


FIG. 21 REMOVAL OF RESIN TANKS FROM CUTUP POOL (SIX TANKS SHOWN HANGING AFTER DEWATERING)

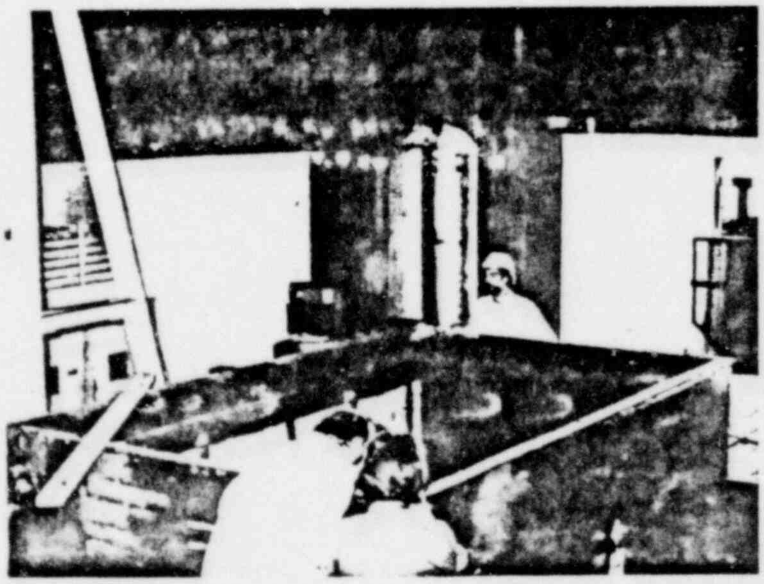


FIG. 22 TWO RESIN TANKS BEING LOWERED INTO B-2 (DOT 6144) SHIPPING CASK

decay pool storage racks, the cutup machine chip collector, the flow guard installation tool of the exit port inspection facility, the glove box for flow guarding subassemblies in the dry loading tunnel, and miscellaneous scrap piping.

On May 8, 1975, 613.9 cubic feet of rad waste in 56 containers (boxes, barrels, and piping), incorporating a total of 210 mCi of Cs-137, Co-60, Sr-90 and Mn-54, were shipped to Morehead, Kentucky.

Three shipments of cut subassembly nozzles (see Section 1.2.3) in 6 separate casks, each containing between 121 and 166 curies of the isotopes of Co-60, Mn-54, Cs-137, and Sr-90, were shipped to Morehead, Kentucky, on June 20th, 24th, and 26th, 1975. A shipment consisting of one 30-gallon steel drum containing uranium blanket pins having 1.54 grams of Pu-239 and one 55-gallon steel drum containing sodium metal, soda ash, and sodium hydroxide was sent to Nuclear Engineering Company's Sheffield, Illinois, burial site on September 23, 1975. The shipment contained 106 mCi of Na-22, Cs-137, Co-60, Mn-54, U-238, Sr-90, and Pu-239.

A listing of significant components that were buried, which were not previously mentioned, follows:

- Remote manual underwater handling tools for cutup pool
- Cutup pool dewatering station
- Cutup pool floor support rack for shipping casks
- Cutup pool underwater gaging fixture
- Bridge crane grapples and stabilizers for cutup and decay pools
- Three transfer carts with support tubes: (1) dry loading tunnel, (2) decay pool to cutup pool, and (3) steam cleaning chamber to cutup pool
- Support rack for blanket segment basket
- Racks for fuel segment storage
- Racks for resin tank storage
- Leaker cans and racks
- FARB cold trap and hot trap components listed on page 6.28 of Report NP-20047
- Sodium melting facility
- Two plugs for guide tube access penetrations through the rotating shield plug
- FTF cask heater and liner assembly
- Two nonirradiated beryllium sections of neutron sources
- Transfer pots for fuel and special components

- Several subassembly finned pots
- Several dummy subassemblies
- Core foil subassembly hardware
- Core shim subassembly hardware

1.4 DISPOSAL OF RADIOACTIVE SODIUM

1.4.1 Drum Cleaning and Resealing

An inspection of the drums containing frozen sodium in the reactor building on March 26, 1975, revealed some sodium reaction products around the bungholes of some drums; and, in most cases, sodium was present in the female threads of the bungs. It is highly likely that sodium was deposited in these threads during the filling operation. Subsequently, sodium reaction with air after diffusing or leaking past the gasket seal caused reaction products to accumulate around the bungs.

The following corrective action was applied to all 630 55-gallon drums for long-term storage. With bungs intact, sodium reaction products were removed from the drum tops by scraping and vacuuming. The bungs were removed (and replaced with precleaned and regasketed units, as described in Appendix II) for cleaning both male and female threads with steel wool and moist towels. The used bung gaskets were removed and replaced, and the threads were wrapped with "Grafoil" carbon thread-sealant tape.* After both bungs were replaced, the top of each drum was cleaned again with steel wool and moist towels. The tops were allowed to dry thoroughly before an alkyd white paint containing no zinc or chlorides was applied around the bungs to form a secondary seal. Cleaning and resealing operations are depicted in Figures 23 and 24.

The drums were handled four at a time on a pallet using the containment building crane, a pallet lifting fixture, and a forklift truck. They were transferred temporarily from the reactor building to the trestleway in groups of about 32 for the cleaning operation and were returned before another group was moved out of the building. This procedure reduced the radiation exposure to personnel working on the drums. All personnel involved with drum cleaning wore face shields and gloves. During the steel wool cleaning, a breathing mask was required. Dosimeters were used at all times, and health physics personnel monitored the overall operation. The drum cleanup and storage program, which took 3 weeks, was started on June 10 and completed on July 2, 1975.

1.4.2 Sodium Storage

The complete inventory of about 70,000 gallons of primary sodium consists of about 38,000 gallons stored frozen in the three primary sodium storage tanks located in the sodium building and about 32,000 gallons stored frozen in 55-gallon drums in the reactor building. Figure 25 shows the 3-tier storage of some of the 630 sodium-filled drums in the reactor building. Both storage locations contain dry air atmospheres, are not heated, and are subject to controlled access. Storage conditions and the scheduled maintenance program were reviewed and approved by the PRDC Review Committee on April 17, 1975. The areas are inspected periodically for evidence of sodium leakage in conformance with the Technical Specifications. The tritium content of the sodium was 8.5×10^{-5} μCi/gm in September 1975, which is very close to the detectable limit of 5×10^{-5} μCi/gm.

*Crane Packing Company, Morton Grove, Illinois

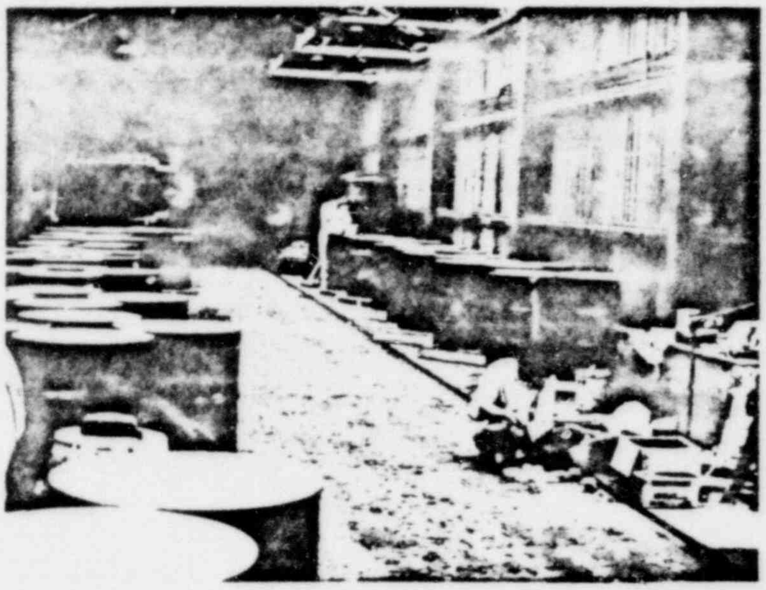


FIG. 23 REBUNGING AND PAINTING OPERATIONS IN TRESTLEWAY

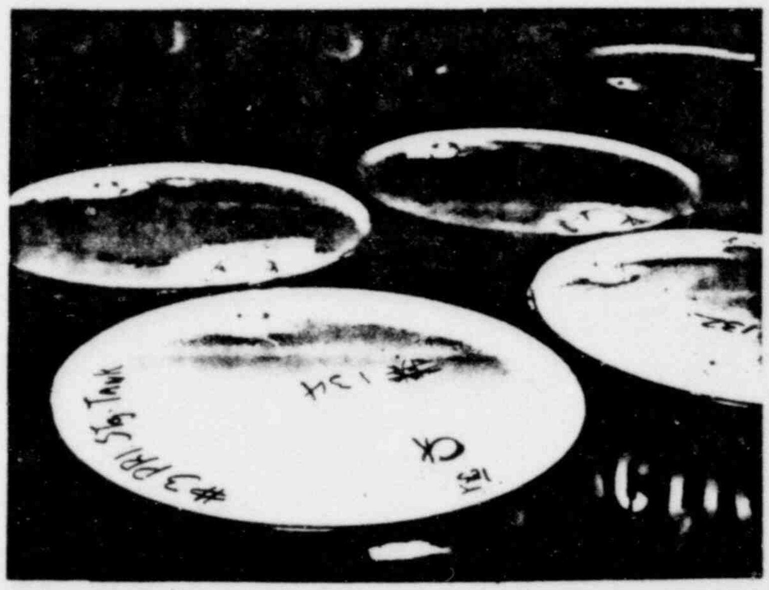


FIG. 24 DRUM TOP LABELING AFTER CLEANING, REBUNGING, AND PAINTING

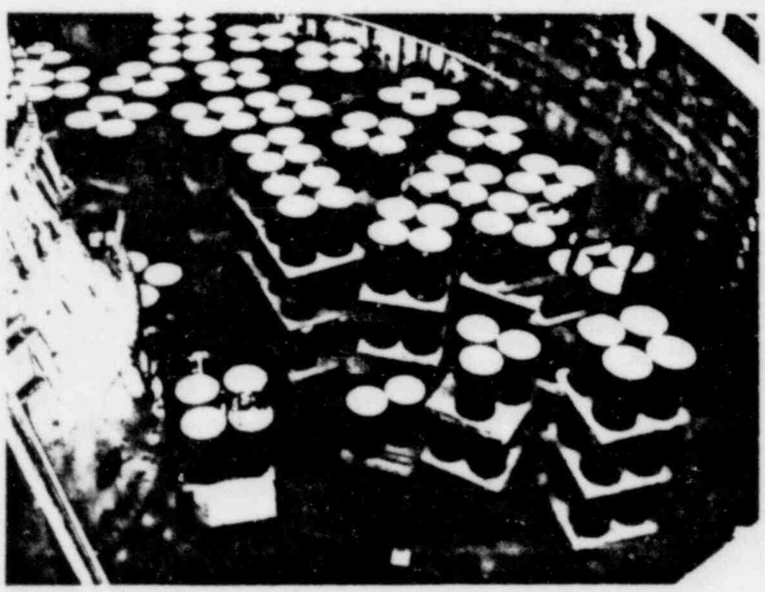


FIG. 25 THREE-TIER STORAGE OF REFURBISHED DRUMS IN REACTOR BUILDING

Project Management Corporation, on behalf of the Clinch River Breeder Reactor Project (CRBRP), agreed to accept, on delivery, full ownership of the primary sodium when needed for use in the plant in 6 to 10 years. In the meantime, all the sodium will remain stored at the Fermi-1 plant, initially under PRDC's supervision and subsequently under Detroit Edison's supervision. Funds to provide for insurance and surveillance, as well as to cover the cost of drumming and shipping the sodium, will be placed in an escrow account by PRDC for use by Detroit Edison as site custodian when PRDC is dissolved.

1.4.3 Monitoring Procedure

The primary sodium storage tanks in the sodium building and the 55-gallon drums of sodium in the reactor building are visually inspected weekly for evidence of leakage, the results being recorded and filed at the plant site in compliance with the Technical Specifications. A smear survey of the drum storage area in the reactor building is performed quarterly, and the results also are recorded and filed.

1.4.4 Fire Fighting Equipment

The reactor building is essentially fireproof, dry, and weatherproof. An existing smoke detection system was left in service. The system alarm is in the plant main control room.

The only access to the locked reactor building is through the personnel air lock. However, emergency exit from the building is possible through the one-way escape lock. Some maintained fire extinguishers are stored in the adjacent clean building ready for immediate transfer to the personnel air lock. The reactor building can be completely sealed from outside air by covering a single 6-inch penetration outside the building, which is in use to permit the building to breathe.

1.5 RETIREMENT OF PRIMARY SODIUM SYSTEM

1.5.1 Equipment Disposal

1.5.1.1 Primary Pumps

Inquiries with respect to obtaining the Fermi-1 primary pumps were received from ERDA in Germantown, Maryland, from Argonne National Laboratory (ANL), and from Liquid Metal Engineering Center (LMEC). In all cases, the pumps would be used for large component test loops. The pumps have a surface layer of contaminated sodium (2.048×10^{-2} μCi/gm of Na -- gross Beta count on 2-24-73) which can be removed without a great deal of difficulty at any future date. The sodium is in the form of sodium carbonate from the CO_2 atmosphere being maintained in the primary system. It is feasible to remove both the pump and pump tank from each loop provided that the nozzle from the tank to the intermediate heat exchanger (IHX) is cut off rather close to the tanks. At the present time, it is thought that the pumps would be donated to any prospective user provided that the user pays all costs associated with removal of the units and with restoring the integrity of the primary system. Openings in the floor would have to be covered with steel plate or grating that would be strong enough to permit a loaded forklift to be driven over them. The pumps will remain in place until a use for them develops. All three primary sodium pump main drive motors, motor barrels, shaft seal housings, and motor speed control liquid rheostats were sold.

1.5.1.2 Overflow Pumps

Some informal inquiries were received with respect to the availability of the overflow pumps, motors, and couplings. If a use for the overflow pumps develops, it is likely that the transfer of ownership could be handled in the same manner and under the same conditions as those stated in Section 1.5.1.1. The motors and couplings will be sold either as part of the total package or separately should a need for those specific components develop at any future date.

1.5.2 System Sealing

The seal-welding of the upper and lower section of the machinery dome, as well as the access door, was completed. A photograph of the completely seal-welded machinery dome is shown in Figure 26.

The sealing of all primary sodium piping penetrations of the primary shield tank below floor, as described in Section 6.6.3 and Figures 6.50 through 6.54 of Report NP-20047, will not be done except that the argon lines used for testing the bellows were capped.

When the 6-inch overflow line between the reactor vessel and the primary overflow tank was cut and capped, the overflow tank was separated from the primary system. Subsequently, the two individual "systems" were purged separately to accomplish a CO_2 passivation of the residual sodium deposits. When passivation of the sodium in the overflow tank was completed, the 4-inch gas supply and pressure equalizing line in the top of the tank was cut to encourage continued oxidation of the sodium deposits and to prevent any unexpected buildup of hydrogen as a result of moisture leaking into a sealed tank. The overflow tank will remain open indefinitely to the air atmosphere of the lower reactor building through the 4-inch line.

The breech in this 4-inch line is the only remaining opening in what was formerly the primary sodium system. The gastight containment system established in 1974 remains sealed and will remain pressurized with CO_2 indefinitely to slowly convert the residual sodium to sodium carbonate. This system consists of the primary sodium system without the overflow tank, the primary shield tank, the machinery dome, and parts of the primary sodium service and secondary sodium systems isolated within the reactor building.

It was originally expected that the pumps associated with the overflow tank would be removed and that the tank would be sealed. Because the tank is open to the air atmosphere, a cover required for sealing the pump tank flanges, as shown in Figure 6.68 of Report NP-20047, is not needed and will not be installed.

1.6 SEALING OF REACTOR BUILDING PENETRATIONS

The sealing of the reactor building penetrations was accomplished as described in Report NP-20047 with the following exceptions:

- 30" Ventilation -- These 4 penetrations (Insert No. 2 of Table 6.7 of Report NP-20047) were to be capped with welded covers in the lower reactor building. Instead, they will be left intact and sealed by an interconnected external ductwork and cooling system, which was also left intact.

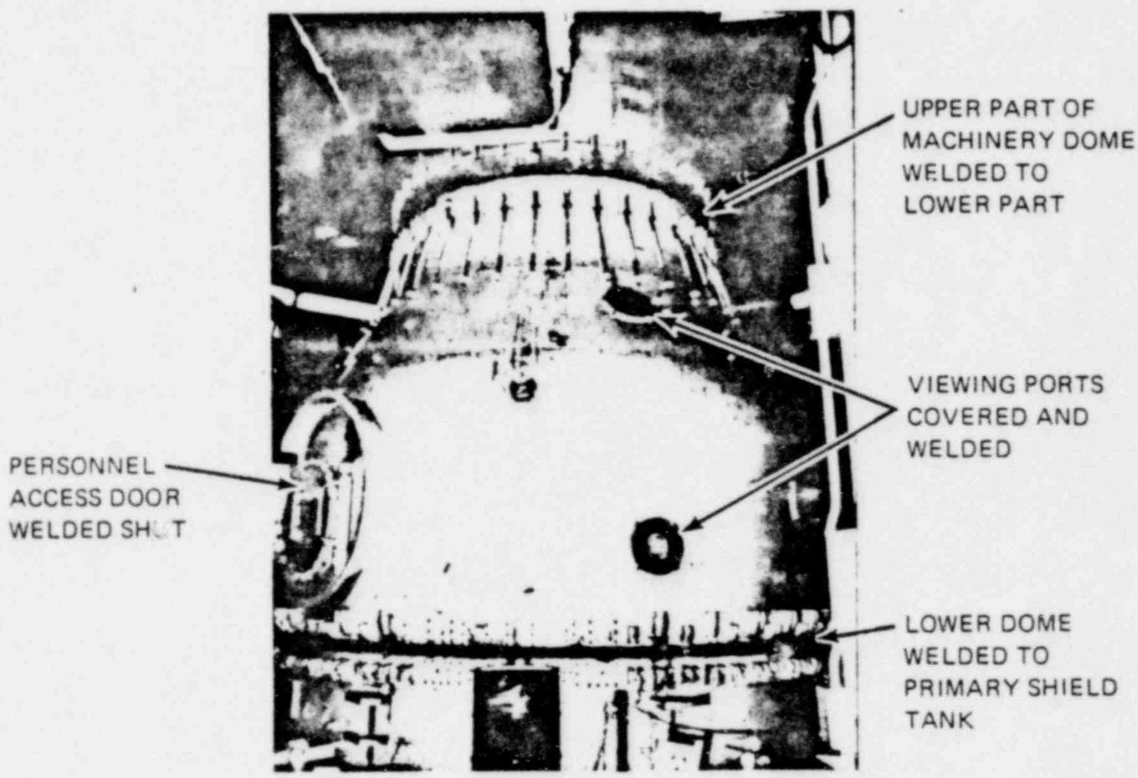


FIG. 26 MACHINERY DOME COMPLETELY SEAL-WELDED

- 10" Spare -- These 2 penetrations (Insert No. 10 of Table 6.7 of Report NP-20047) were to be modified to permit the building to breathe. Instead, they will be left intact with the original welded covers.

- 6" Regulation Supply and Exhaust -- One of the two penetrations (Insert No. 6 of Table 6.7 of Report NP-20047), which were to be left intact, was opened inside the building by blocking open 6-inch valve PCV-1305. A 2½-inch maintenance purge line to the same penetration near PCV-1305 was blocked closed. The 6-inch penetration line is covered with an absolute filter that will remain intact. The line was cut off a few inches past the outside of the building and covered with a grating to prevent birds or other animals from entering.

1.7 DISPOSAL OF SECONDARY SODIUM SYSTEM COMPONENTS

1.7.1 Sodium Pumps

Both remaining secondary sodium pumps (see Section 6.7.3 of Report NP-20047), magnetic couplings, and pump drive motors were removed from the system and sold.

1.7.2 Boiler Feed Pumps

All three steam generator boiler feed pumps, couplings, and drives were removed from the system and sold.

1.8 RETIREMENT OF FARB SYSTEMS

1.8.1 Radioactive Liquid Waste System

A significant change has been made in plans, as stated in Section 6.8.10 of Report NP-20047, for the way that the liquid waste system will be left. Because it is impractical to provide an alternate floor drain to an alternate sump for use in case water enters any of the rooms from pipe failures, heavy storms or from any other source, it is now planned to leave the sump, liquid waste system, and liquid waste discharge radiation monitor RE-742 in condition to be operated. Contaminated drains in the building will be labelled. Contaminated drains from the health physics building will be sealed The sump and sump pumps will be left so that accumulated water will be pumped automatically into the 7500-gallon (MK-15) liquid waste storage tank. The liquid waste system would be reactivated for a short time to dispose of this water should it be necessary. No room flooding has ever occurred, and none is expected in the future; however, equipment must be left in place to handle such an event. Such water would be disposed of in accordance with Technical Specifications even though it should only be very slightly radioactive as a result of flowing through contaminated piping and into and out of a contaminated sump and storage tanks. Thus, the liquid waste pipes will not be cut and capped as earlier planned to isolate the tanks. Most of the contamination in the liquid waste system is Co-60, which will decay to negligible amounts in 10 half-lives or in about 50 years.

1.8.2 Pools

Initial retirement activities relating to the decay and cutup pools are described in Section 6.8.14 of Report NP-20047.

1.8.2.1 Decay Pool

Following removal of all fuel and water from the decay pool in March 1975, there remained the task of disposing of all contaminated pool equipment and subsequent decontamination of the pool itself. The first phase of equipment disposal was the removal of all portable hardware. This consisted of all subassembly pots, control rod pots, and neutron source pots that were contaminated on the inside. These pots were wrapped and boxed for shipment to the Morehead, Kentucky, site. Next, the heat exchanger coolers were removed from the pool walls and were decontaminated and scrapped. The grapple assembly on the pool crane was segmented, wrapped, and boxed for burial. The pool cart and track assembly and all fuel storage racks, shown in Figure 27, were removed, wrapped, and boxed for burial.

After all equipment was removed from the decay pool, the pool walls were scrubbed with a detergent and soapy water using scrub brushes. The final wash consisted of a 10 percent nitric acid solution and a demineralized water rinse. Protective clothing worn by personnel consisted of plastic coveralls, rubber gloves, plastic shoe covers, and respirators. Personnel exposure was monitored by health physics technicians at all times.

Subsequent to the final cleaning and drying of the decay pool, a 20-mil-thick layer of Cooks Spray Booth Shield White strippable paint was applied in several coats to all surfaces of the decay pool walls, floor, and tunnel, as indicated in Figure 28. Approximately 50 gallons of paint were applied using hand rollers. Personnel wore coveralls and oxygen breathing masks and worked from a hanging cage. Additional air horns were provided for ventilation during the painting process. The decay pool room cleanup started in May and was completed July 23, 1975.

1.8.2.2 Cutup Pool

After the last of the blanket was shipped on April 18, 1975, the nozzles were disposed of as described in Section 1.2.3. Concurrently, the leaker cans were removed from the pool, drained, and prepared for burial by wrapping in plastic and sealing in wooden boxes. The following decommissioning activities occurred in the order given below. The phrase "prepared for burial," as used below, means rinsed over the pool with demineralized water, dried in air, wrapped in plastic, and placed in a plastic-lined box.

- Storage racks 2, 3, and 4 were removed and prepared for burial.
- The resin tanks and cutup machine were removed and shipped, as described in Section 1.3.2.
- Pool water was clarified for maximum underwater viewing by passing it through the Cuno micron filter system.
- Loose pins and other contaminated scrap that could be seen on the pool floor were removed and prepared for burial.
- Leaker can racks were removed, rinsed, segmented with a portable electric band saw, and prepared for burial.
- Concurrently, tools no longer needed in the pool were removed, segmented, and prepared for burial.

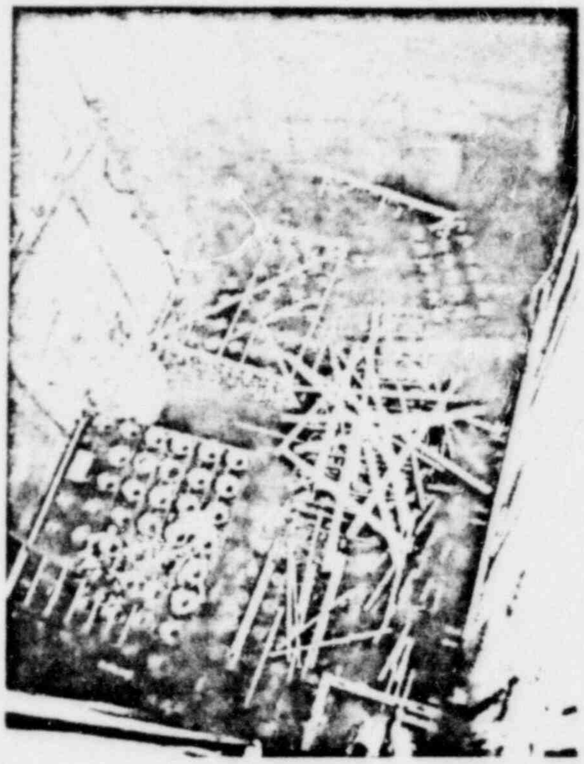


FIG. 27 DECAY POOL STORAGE RACKS BEING CUT APART

FIG. 28 ROLLER PAINTING THE DECAY POOL AND TUNNEL BETWEEN POOLS

- The water in the pool was drained to a depth of about 3 feet, a level necessary to shield the remaining irradiated waste. As this was being done, suspended rust was removed by circulating pool water through the Cuno micron filter using an impeller pump and a submersible sump pump in a two-lift operation, as shown in Figures 29 and 30.

- The irradiated waste, principally chips from the segmenting of fuel and blanket subassemblies, were removed and disposed of as described in Section 1.2.3 and shown in Figure 31. When all the chips were removed from the pool, the water level was lowered to about 1 inch using the two-pump system and micron filter mentioned above.

- A wet/dry, air-powered, vacuum pickup positioned on top of a 55-gallon drum was used to vacuum the residue from the pool floor and to remove the final inch of water. Water and residue drained through a valve in the bottom of this drum into a cloth bag filter housed in another drum. Cloth bags containing the residue were wrapped in plastic and stored in 55-gallon drums for disposal. The filtered liquid drained into the liquid waste system.

- Piping, filter system, pumps, resin tank racks, and pool cart were removed, segmented as necessary, and prepared for burial.

- The pool was cleaned and painted using the same procedure and material that was employed for the decay pool, as reported in Section 1.8.2.1.

Personnel exposure was monitored by health physics technicians during all dismantling activities.

1.8.3 Decontamination Chamber

The decontamination chamber is described in Section 6.4.3.4 and in Figure 6.27 of Report NP-20047. This facility was never used because the jet pressure of the decontaminating fluid from the nozzles was insufficient to decontaminate components. Instead, all sodium was melted and drained from equipment. Then the equipment was generally packaged and shipped as dry waste. Because in most cases it was more economical to bury than to decontaminate, less material was decontaminated than originally planned.

Each item slated for disposal was rinsed to remove loose surface contamination and surveyed for contamination level. Then a decision was made whether to decontaminate it for salvage or to package it as contaminated waste for burial. In general, items with radiation or contamination levels above MPC were logged and sealed for offsite disposal at Morehead, Kentucky, and all items with radiation or contamination levels below MPC were sold as scrap.

1.8.4 Steam Cleaning Facility

The steam cleaning facility, discussed in Section 6.8.13 of Report NP-20047, was secured essentially intact but isolated from services and made inaccessible by direct entry. No attempt was made to decontaminate any equipment contained within the isolated chamber. All demineralized water, steam, and argon supply lines to the chamber were cut and capped. All electrical power to the facility was disconnected at breaker cabinets. All access tube valves were closed, and the access ports to the manual valve actuators

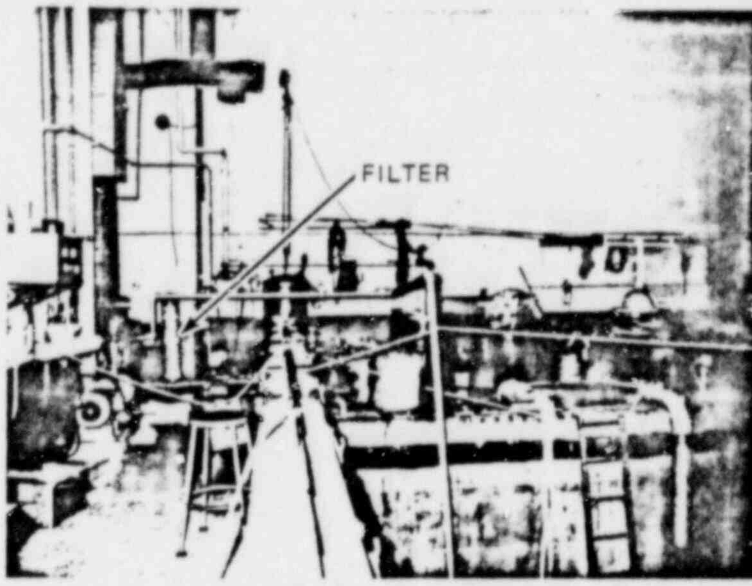


FIG. 29 CUNO MICRON FILTER USED HERE TO REMOVE RUST FROM CUTUP POOL WATER

FIG. 30 SECOND-STAGE LIFT FOR REMOVAL OF RUST AND WATER FROM CUTUP POOL

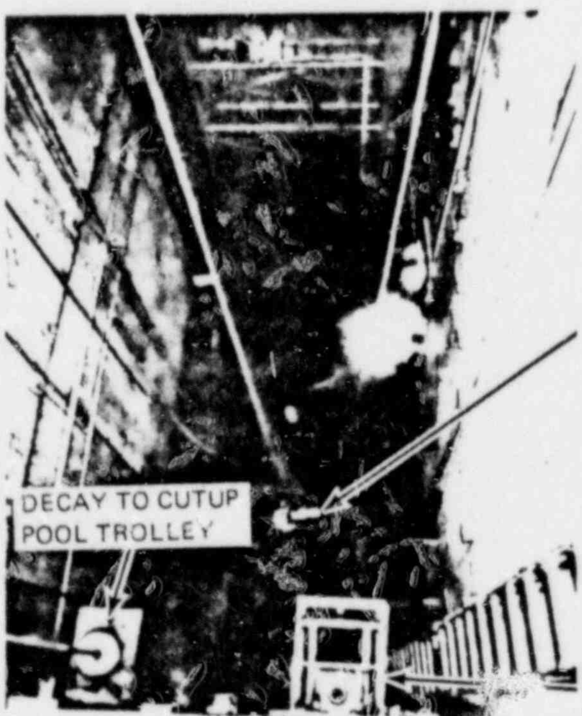


FIG. 31 PUMPING WATER AND CHIPS FROM FLOOR OF CUTUP POOL USING BELLOWS PUMPS

were closed with steel plugs tack-welded in place. The substructure exhaust ducts to the chamber were left intact and valved shut to prevent spread of contamination. The chamber floor drain to the dry loading tunnel was left intact. The chamber equipment access plug remains in place and sealed with a metal strip around the annulus. The manhole access plug remains in place, its removal possible only by means of the overhead crane. The cleaning machine drive and transfer tank drive mounted on top of the chamber were electrically disconnected but otherwise left intact.

The dry loading tunnel for new fuel transfer to the cleaning chamber was emptied of all mobile and portable equipment and then decontaminated. The tunnel floor drain to the hot sump drain system was left intact. The tunnel manhole and equipment access from the receiving room was closed with a hinged steel door resting in a rubber seal. A former access hole through the center of the door for loading fuel was closed with a welded plate, as were the door handle holes.

1.8.5 CO_2 Passivation of Overflow Tank

When Report NP-20047 was written, it was expected that the overflow tank would be sealed and purged with CO_2 cover gas from the same supply and control system as that for the reactor vessel. Because the tank was well drained, containing only a small heel of sodium, it was decided that it would not be necessary to protect it with an inert gas and that opening it to ambient air would be practical. Appendix VIII to Report NP-20047 mentions that opening the tank to air was being considered, although this was not reflected in Section 6.6.7 on page 6.39 of the report.

After an initial week of CO_2 pressure (from the CO_2 supply to the primary system) on the tank, the 4-inch gas supply and pressure equalizing line to the tank was opened through which the tank could breathe. Two sample lines were installed through the cut 4-inch line, one ending near the top of the tank and one ending well down within the tank through which air samples could be drawn to measure hydrogen level. The passivation and monitoring procedure for the overflow and transfer tanks is given in Appendix VIII of Report NP-20047, and the passivation procedure to be used for the primary sodium storage tanks is given in Appendix III in this supplement. Testing with a local hydrogen analyzer initially was done twice each shift and then gradually reduced in frequency over a 5-month period. To supplement the local hydrogen analyzer, grab samples were taken periodically from the tank and analyzed by The Detroit Edison Company's Engineering Research Department. Data from these analyses are shown in Table A3-1. At no time did the hydrogen concentration reach as much as 10 percent of the lower explosive limit of 4 percent hydrogen in air. Therefore, sampling was discontinued. The remaining low level of activity present in the residual sodium can decay over a period of years (without the need to monitor the cover gas pressure as is the case with the CO_2 supply to the primary system) until complete dismantling of the plant becomes appropriate.

1.8.6 Miscellaneous Equipment and Services

For long-term retention of the Fuel and Repair Building, either for use or vacant, the following miscellaneous closing details were done:

The floors and any remaining equipment in all areas associated with the steam cleaning and liquid waste systems were decontaminated, except where these areas could be totally secured from direct access. In those exceptions, the areas are being identified with information tags or signs denoting internal contamination.

All pipes, valves, and stationary equipment with no external contamination that are located in accessible areas not previously mentioned for disposal in this report were left intact. These items are being identified with permanent metal or plastic signs requiring health physics protection where needed should any future work be done on them. No basement room manholes were plugged or otherwise sealed.

The building heating system was secured but otherwise left intact. Steam, condensate, water, and compressed air lines were valved closed, drained, some of them cut and capped, but otherwise they were left intact. The building ventilation system ductwork was left intact. The inside roof drains were capped, and outside roof drains were installed.

The building nonradioactive sump was left intact and was adapted to automatically discharge to the plant condenser discharge canal on a signal of high level. The former "hot" sump was left intact so that water can be pumped to the waste liquid storage tanks and held for metered discharge in the future. Power supplies for the sump pumps, alarm panels, and general lighting will be left intact.

1.9 RETIREMENT OF AUXILIARY SYSTEMS

1.9.1 Fuel Transport Facility

Initial retirement activities relating to the Fuel Transport Facility (FTF) were described in Section 6.8.2 of Report NP-20047.

Disposal of the FTF transport car and the FARB gripper and plug casks in January 1975 necessitated some component dismantling for removal of depleted uranium shielding and subsequent decontamination or burial of any contaminated components. Of particular interest was removal of 5700 pounds of uranium from the front of the transport car cask. Access to the uranium required opening the top of the fuel cask shield cavity. The uranium shield had been fabricated in the form of two adjacent stacks of seven blocks per stack, each block being plated with a metal not identified on drawings. The outer stack was held together with vertical tie rods and was secured to the frontal plate with setscrews. Each block contained a bolthole for attaching an eyebolt. Uranium dust was vacuumed off each block, and all boltholes required retapping before securing eyebolts. Dust had accumulated within the shield cavity because the metal plating on all 14 blocks had cracked from thermal cycling of the uranium during operational heating and cooling of the cask. By attaching plastic bags at the top of the cask and lifting through the bags, single uranium blocks were raised and bagged for transfer to a shipping drum. The maximum radiation level measured at the surface of a loaded drum was 3 mr/hr. All frontal and some base plates were cut away from the cask to gain access for final decontamination of the shield cavity and removal of contaminated lead shot which was also present in the cavity. Sandpaper, wire brushes, soap and water, paper towels, and nitric acid were used to decontaminate the salvageable components.

Also dismantled to remove uranium shielding were the upper and lower cask valves and the cask shielding gates of the transport cask. About 500 pounds of depleted uranium in 2 valve disc slides were removed from the cask valve bodies. The 2 shield gates weighed 2750 pounds and measured 4 mr/hr on contact. The cask finned liner, which could not be decontaminated, was cut out of the cask. The liner and all depleted uranium were shipped to Morehead, Kentucky, for burial. All other components of the transport car and FARB casks were decontaminated for salvage, including the gripper cask and valve, the drip pan housings for the gripper cask and transport car cask, and the transfer tank exit port plug and plug cask. Components sold or donated to Hanford Engineering Development Laboratory (HEDL) for decontamination and other studies included the gripper within the FARB gripper cask, the gripper hoist drive, the plug cask hoist, the reactor exit port and transfer tank exit port floor valves. Both gripper cask control consoles were sold to Atomics International. The reactor building gripper cask and plug cask (with exit port shield plug contained) remain intact in the building. Those items do not contain uranium but are internally contaminated with radioactive primary sodium residues.

1.9.2 Cooling System for Primary Sodium Storage Tank Room

Section 6.8.11 of Report NP-20047 is updated in this paragraph. The nitrogen atmosphere in the sodium storage tank room was replaced with air. The room door was bolted closed, and subsequent access is permitted only for the performance of routine inspection for evidence of sodium leakage, as required by the Technical Specifications. The nitrogen-to-water cooling system for the room atmosphere was dismantled, and the blower and heat exchanger equipment was scrapped.

1.9.3 Station and Emergency Power Supply System

The retirement of the station and emergency power supply system was discussed in Section 6.8.16 of Report NP-20047. All 480-V and 4800-V AC buses were removed from the building with the exception of the 480-V AC control building bus. This bus is used to supply all remaining power requirements and is reconnected to The Detroit Edison Company's 480-V AC House Service Bus #82. The PRDC in-plant communications system will be left in service. All retired PRDC bus cabinets, power cable, heating transformers, and electrical switchgear were sold.

1.9.4 Plant Simulator

The operators' training console (simulator) was sold to a private concern. The analog computer was cannibalized by Detroit Edison's Engineering Research Department, and the remains were scrapped. The 8-channel Sanborn recorders were sold to Detroit Edison. The instructors' console was cannibalized and the remains also were scrapped.

1.9.5 Malfunction Detection Analyzer

The malfunction detection analyzer, installed in 1969 as an early warning safety device following the core melting incident, consisted essentially of a series of signal conditioners installed throughout the reactor plant, together with an IBM digital computer with a line printer, data storage, and program storage. The computer portion, leased to PRDC, was returned to IBM. The signal conditioning instrumentation was left intact.

1.9.6 Radiation Monitors

A system of fixed monitors to detect and measure radiation on a continuous basis was installed in the plant. This system included area-gamma, gaseous, particulate, and water monitors. With one exception, all fixed monitors were retired as described in the following paragraphs.

1.9.6.1 Area Gamma

Most area gamma monitors were a part of a multiple unit system where several detectors were power fed from a single control and high voltage panel. All detectors were removed for salvage, and all service feeds were disconnected at the control panels.

Single channel units, including high voltage and power supply readout equipment and detectors, were located in the same area of the decay and cutup pool rooms. Those units were disconnected from their 110-V system, and the control/detector units were removed for salvage.

1.9.6.2 Gaseous and Particulate

Most gaseous and particulate monitors were of the constant-air-type in which the detector, shielding, electronics, pump, and compressor are all located in a single console. In those cases, a continuous sample of the material being monitored was brought to the console and returned to the main flow through small shielded pipes. All those monitors were disconnected from their 110-V power supplies and control room alarms and then donated to Detroit Edison's Fermi-2 project.

Some of the gaseous monitors were pipe monitors in which the detector and shielding are located on the outside of the pipe being monitored, with the remainder of the equipment located elsewhere. The shielding was removed from the detectors, and all detectors and electronics for those monitors were disconnected from their power supplies and control room alarms. This equipment also was donated to the Fermi-2 project.

1.9.6.3 Water

The water monitor (RE-742) controlling the discharge from the waste liquid tank room will remain intact to be utilized as needed in the event of building flooding, as discussed in Section 1.8.1. Two other off-line type of water monitors were disconnected from their power supplies and control room alarms for salvage of the detectors, shielding material, and electronics. Both water line supplies to the detectors were valved out.

1.9.7 Other Miscellaneous Equipment

Other miscellaneous, uncontaminated, major plant equipment sold or scrapped included the following items:

- Two fuel shipping casks (See Section 1.3.2.2), cask stand, and valve assembly sold to Argonne National Laboratory
- All hydrogen-type water coolers from the decay pool
- Finned pot gripper for steam cleaning machine
- Three hydrocarbon analyzers for the primary inert gas system
- All portable health physics survey meters.

1.10 SURVEILLANCE OF RETIRED PLANT

The second paragraph of Section 6.10.4, which appears on page 6.52 of Report NP-20047 is in error and is corrected in the following paragraphs.

Environmental sampling will be done as scheduled in the current plant Technical Specifications approved by the AEC in July of 1974. There are two sampling regimes. Regime I applies for 90 days after any liquid waste discharge. Regime II applies beyond 90 days after the last radioactive discharge. Regime I specifies the sampling of river and lake water on a weekly basis; raw city water every 4 weeks; and lagoon water, lagoon sediment, and river sediment every 26 weeks. Regime II specifies the same sampling program but the frequency for all samples is reduced to once every 6 months. Background samples include those from raw city water, river water, and river sediment.

PRDC management felt that all sampling could be discontinued once liquid waste discharges are discontinued. Waste discharges from the plant always were so well-diluted during plant operation and decommissioning activities that variation, as indicated in background samples, always masked small differences introduced by the plant. If waste discharges were not detectable during those periods, it is almost impossible for any small leakage from the plant in the future to be significant enough for detection because only small quantities of radioactive materials remain on site. All 560,000 pounds of primary sodium on site contained less than 2 curies of activity as of July 1, 1975. All other activity, except for small amounts of surface contamination, is in the form of induced activity in the stainless steel reactor vessel and its internals. It is highly unlikely that any of this activity could measurably affect nearby waters or their sediments. It is believed that the sampling should be discontinued no later than one year after liquid waste discharges cease and certainly following removal of the primary sodium from the site.

Gaseous waste discharges were discontinued in October 1973, and air and precipitation monitoring was discontinued following approval of a Technical Specification Change in July 1974.

The moisture detector in the biological shield wall cavity in the reactor building was modified to respond to water level by the insertion of copper probes 3 inches into the floor drain. A similar modification was made to the moisture detector in the FARB hot sump. The moisture monitor in the lower reactor building remains intact.

2.0 STATUS OF RETIRED PLANT

This section will update Section 8.0 of Report NP-20047 to reflect current decisions on how the plant will be left at the completion of decommissioning. All changes mentioned are consistent with the general philosophy and plan proposed to the AEC in September 1973 except in those cases where the Commission was notified otherwise.

The most significant change concerns the disposition of the primary sodium. Rather than ship it off site for immediate disposal as originally planned, it was donated to the Clinch River Breeder Reactor Project, to be delivered on demand sometime between 1981 and 1985. Details are given in Sections 1.4.2 and 3.0 of this supplement.

As stated earlier, most contaminated items were cut, boxed, and shipped to Morehead, Kentucky, for burial. It was deemed impractical to decontaminate many items, particularly those having difficult-to-reach surfaces, such as the inside of piping and transfer pots.

The decay pool in the Fuel and Repair Building (FARB) was drained and partially decontaminated. Because some areas of the pool walls retained removable surface contamination levels as high as 50,000 d/m/100 cm^2, particularly in corners and at attachments to the pool liners, the walls were coated with a thick layer of strippable paint. The cutup pool was drained and painted in a like manner. Room walls and the surfaces of piping and equipment above the pool in both rooms were decontaminated to less than 500 d/m/100 cm^2.

The large concrete plug was used for closure of the steam cleaning chamber room. It would require a deliberate effort with the building crane to gain entrance to the chamber. The plug will be labeled to show that the room inside is contaminated.

Radiation and residual surface contamination levels in various areas of the plant are given in Table 3.

As described in detail in Section 1.8.1, the liquid waste and sump pump system will be deactivated but left intact so that water may be pumped into the FARB liquid waste storage tanks or other tanks and held for discharge later. The only drains that will be plugged are the radioactive drains from the health physics building.

Most of APDA s Technical files and some PRDC technical files were transferred to Overseas Advisory Associates, Inc. The remaining files, including the PRDC and APDA corporate files, are being consolidated in the former upper fuel vault in the FARB. This room is a dry, secure area with a single, securely-locked entrance door. The Detroit Edison Company will become custodian of these files when the EFAPP site is returned to Edison's control late in 1975. Drawing files remain intact on the first floor of the office building for future reference as needed.

TABLE 3

RADIATION AND SURFACE CONTAMINATION LEVELS

Location	Radiation Levels mr/hr Max	Radiation Levels mr/hr Avg	Contamination Levels d/m/ft^2 Max	Contamination Levels d/m/ft^2 Avg	Date
Fuel and Repair Building					
Repair Pit	1	.02	150	<100	11/19/75
Decontamination Facility	0.03	0.01	400	<100	11/19/75
Dry Loading Tunnel	6	0.4	250	<100	7/14/75
Steam Cleaning Chamber	10	2	140,000	44,000	2/8/74
Decay Pool and Room	20*	0.03	18,000*	<100	12/1/75
Cutup Pool and Room	40**	0.03	2,200**	150	12/1/75
Mechanical Equipment Room	2	0.1	8,300***	<100	12/1/75
Control and Receiving Room	0.3	0.01	183	<100	11/21/75
Cask Car Maintenance Pit	0.3	0.01	<100	<100	11/19/75
Fan Room	1	0.01	110	<100	11/19/75
Unloading Pit	0.03	0.02	<100	<100	11/25/75
Lower Fuel Vault	0.02	0.01	<100	<100	11/19/75
Upper Fuel Vault	0.3	0.01	<100	<100	11/21/75
Transfer Tank Room	7	0.5	<100	<100	2/6/74
Pool Sump (1)	0.03	0.02	<100	<100	11/18/75
Hot Sump Pit (2)	120	2	800	170	11/4/75
Clean Shop	0.02	0.01	<100	<100	11/24/75
Cold Trap Room	0.03	0.02	126	<100	11/6/75
North Waste Tank Room	100	3	<100	<100	11/19/75
South Waste Tank Room	60	1	<100	<100	11/19/[illegible]
Reactor Building					
Biological Shield Wall Annulus	0.05	0.02	<100	<100	11/18/75
Below Floor Area	2	0.2	<100	<100	10/8/75
Outside Auxiliary Fuel Storage Facility	0.1	0.1	<100	<100	5/7/74
Operating Floor	30	7	200	<100	10/8/75
Machinery Dome	1.5	1	105	<100	4/24/75
Secondary Shield Wall Cavity	15	0.1	<100	<100	10/8/75
Reactor Building Anti-Contamination Building	0.05	0.03	<100	<100	11/7/75
Cask Car Trestleway (3)	70	0.05	<100	<100	11/21/75
Waste Gas Compressor Room	0.02	0.01	<100	<100	11/17/75
Waste Gas Valve Room	0.02	0.02	<100	<100	11/17/75
Primary Sodium Cold Trap Room	3	0.05	108	<100	11/5/75
Primary Sodium Storage Room	6	3	<100	<100	11/5/75
Primary Sodium Service System Valve Room	0.3	0.01	<100	<100	11/17/75
East Sodium Gallery	0.02	0.02	<100	<100	10/9/75
West Sodium Gallery	0.02	0.02	<100	<100	10/9/75
Fission Products Detector Building	0.03	0.02	<100	<100	5/16/74
Inert Gas Tunnel	0.1	0.02	<100	<100	11/26/75

(1) Above waterline. Does not include below waterline
(2) Does not include sump below waterline
(3) 70 mr/hr is on rad waste drum temporarily stored there

TABLE 3

RADIATION AND SURFACE CONTAMINATION LEVELS

Location	Radiation Levels mr/hr Max	Avg	Contamination Levels $d/m/ft^2$ Max	Avg	Date
Vent Building	0.03	0.02	<100	<100	11/17/75
Vent Building Equipment Pit	0.02	0.01	<100	<100	11/17/75
Health Physics Building					
Office and Lab	0.5	0.02	<100	<100	11/12/75
Locker Room	0.03	0.02	<100	<100	11/12/75
First Aid Rooms	0.02	0.02	<100	<100	10/17/75
Chem Lab	0.02	0.02	<100	<100	11/12/75

*(1) 20/mr hr on decay pool tunnel to cutup pool wall support beam. Whole pool covered with strippable paint

(2) 18,000 dpm is on remnants of chain for bridge grapple inside bridge trolley cabinet which is sealed and labeled

**(1) 40/mr due to material trapped under alignment pads on bottom of cutup pool. Pads sealed with metal putty and whole pool covered with strippable paint

(2) 2200 dpm is on boom crane which is sealed in plastic and labeled

***8300 dpm is on liquid waste pump which is to be decontaminated and covered with plastic when finished using it

3.0 COST OF RETIREMENT

Several events occurred since the preparation of Section 9.0 of Report NP-20047 was completed in February 1974 that altered the then projected cost of decommissioning. The largest increase in cost resulted from the very rapid escalation of indexes used to figure the hypothetical costs for reprocessing the blanket. The blanket was accepted by Energy Research and Development Administration (ERDA, for temporary storage at the Idaho Chemical Processing Plant under the spent fuel chemical processing and conversion provision of the Atomic Energy Act of 1954, as given in 33 FR 30. On page 6.7 of Report NP-20047, these acceptance costs were estimated at $1.2 million. The actual payment made to ERDA in June of 1975 was $1,594,000. One of the indexes used to compute the payment was the basic inorganic chemical index, which is significantly affected by the price of oil. This index escalated by more than 50 percent from February 1974 to May 1975.

One significant decrease in decommissioning costs resulted from the use of an alternate plan for sodium disposition developed with the cooperation of ERDA, the Nuclear Regulatory Commission (NRC), and The Detroit Edison Company. With ERDA's assistance, Project Management Corporation (PMC) agreed to accept the sodium for the Clinch River Breeder Reactor; NRC agreed to permit on-site storage of the sodium for up to 10 years; and Detroit Edison agreed to permit the sodium to be stored on-site and become its custodian. As a result, a contract was signed between PRDC and PMC under which the primary sodium was donated to the Clinch River Breeder Reactor Project (CRBRP), and PMC is obligated to accept the material on delivery in 6 to 10 years as indicated in Section 1.4.2. Mutual advantages were derived from this arrangement: PRDC saved over $500,000 ($700,000 minus insurance and monitoring costs) of processing and disposal costs, and PMC will save a like amount (less extra handling costs) in not having to purchase over 500,000 pounds of reactor grade sodium.

Table 9.1 of Report NP-20047 has been updated in Table 4 of this supplement to show current cost estimates, and Table 5 has been added to show the source of these funds. Of the $6.9 million total spent on decommissioning, $3.8 million was paid to the AEC as follows:

$1.8 million	for core reprocessing
1.6 million	for blanket reprocessing
.4 million	for use charges during decommissioning
$3.8 million*	

It is of interest to note that in decommissioning a commercial plant, the $3.8 million paid to the AEC would be charged against plant operations as part of a fuel account rather than as a part of decommissioning expenses.

Salvage sales provided a significant income principally because some of the plant buildings were useful to The Detroit Edison Company for continued operation of the turbine-generator using steam from the conventional oil-fired boiler located adjacent to the reactor plant. The PRDC warehouse proved to be of good value for the Fermi-2 project.

*An added $380,000 of use charges incurred in the period July 1 through November 30, 1972 were also paid to the AEC after December 1, 1972.

In Report NP-20047, the projected cost for completion of decommissioning activities was given as $2,685,000, making a total estimated cost of $7,085,000, excluding provisions for surveillance. With the project essentially completed and funds provided for the future disposal of sodium, the total cost is now predicted to be $6,940,000, excluding surveillance. The increase in blanket disposal costs and the decrease in sodium disposal costs partially offset each other. Completion of decommissioning activities took longer than estimated, but the additional expense was offset by savings made from the rapid reduction of work forces as personnel could no longer be effectively used, with the final cleanup in 1975 being accomplished by a very small group. Retiring buildings and equipment as soon as they were not needed resulted in substantial savings in property tax, insurance, and other expenses. The total cost of decommissioning has remained essentially as it was budgeted in 1973.

TABLE 4

DECOMMISSIONING EXPENSES THROUGH OCTOBER 31, 1975

1.	Core Fuel Processing Includes transferring subassemblies from reactor and other storage areas to FARB, steam-cleaning, underwater segmenting, loading in casks and shipping to Savannah River; also includes material and fabrication cost of No. 2 shipping cask and modifications to No. 1 cask.	$ 418,000
2.	AEC Core Fuel Processing Includes basic processing and conversion charges, processing and conversion losses, and use charges during the processing period.	1,783,000
3.	Blanket Subassembly Processing Includes transferring subassemblies from reactor and other storage areas to FARB, steam-cleaning and storage in cutup pool.	67,000
4.	Blanket Subassembly Processing for Disposal at Idaho Includes licensing and rental of two casks, design and purchase of special basket and container assemblies, round trip shipping charges, and cutting and disposal of some nozzles.	386,000
5.	AEC (ERDA) Blanket Subassembly Processing Payment Blanket accepted by ERDA under reprocessing provisions of 33 FR 30.	1,594,000
6.	Sodium and Cold Trap Disposal Includes transferring primary sodium from all systems to storage tanks, constructing a sodium-barreling facility and the dismantling and removal of the primary cold trap for shipment to Beatty, Nevada. Includes $75,000 allowance to drum and ship primary sodium to PMC in 1981-1985.	250,000
7.	Sodium Piping and Contaminated Equipment Disposal Includes cutting and sealing of pipes and equipment, decontaminating equipment and packaging solid waste for burial.	450,000
8.	Plant and Administrative Expenses Includes plant and administrative expenses, nuclear insurance, property tax, regulatory charges and AEC use charges. Amount is reduced by $370,000 of interest received on invested funds during the decommissioning period.	1,992,000
	Total Cost of Decommissioning	$6,940,000
9.	Provision for Surveillance Includes anticipated NELIA refunds of $325,000 over the next 10 years to be used for this purpose.	540,000
	Total Cost of Decommissioning including Surveillance Fund	$7,480,000

TABLE 5

SOURCE OF DECOMMISSIONING FUNDS

Cash and Commitments as of December 1, 1972	$4,114,000
Member Company Contributions 1974 & 1975	2,809,000
Contributions from APDA	42,000
Revenue from Salvage Sales	555,000
Anticipated NELIA Rebates, 1975 - 1985	325,000
	$7,845,000
Use charges, oxide core, and other operating costs paid after December 1, 1972 from the above funds. (Net of $103,000 of fuel handling and equipment charges prior to December 1, 1972)	-365,000
Funds Available for Decommissioning and Surveillance	$7,480,000

4.0 ACKNOWLEDGEMENTS

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APPENDIX I

PROCEDURE FOR REMELTING STORED PRIMARY SODIUM FOR DRUMMING

The primary sodium tanks, consisting of three 15,000-gallon tanks with heaters and level probes, provide the facility to store radioactive sodium removed from the primary coolant system. The tanks are equipped with gas and sodium piping to permit sodium to be transferred.

A very detailed procedure for remelting the sodium in the storage tanks is contained in the EFAPP technical files. A generalized procedure is given in this appendix because the type of steps taken should be of more interest and applicability to other plants than the details.

The first step in the remelting sequence is to ascertain that all lines to be used are essentially free of sodium. After last use, these lines should have been blown free of sodium. If such action was not taken or if sodium collected in the low point of a line, heating should start in a gas space to allow for expansion. The steps to be taken are:

1. The gas lines should be heated first by turning on the appropriate panel breakers. The operation of the breakers and the increases in temperature should be recorded on the breaker log sheet and the temperature recording log sheet. The resistance of each heating circuit is adjusted to allow continuous operation; however, if the temperature does not level off at 700 F or less, a manually-controlled "on-off" operation may be necessary.

2. When the temperature of the gas lines reaches at least 250 F, the lines should be checked to determine whether they are clear. This is done by checking the pressure gauges installed at the storage tank gas manifold located just above the drum loading enclosure. A propane torch may be used, if necessary, to heat and clear the gas piping at the gas manifold. To ensure that the gas lines are clear, a small amount of gas can be added to each storage tank, which can be detected by the increased reading on the pressure gauges on each tank.

3. When the gas lines are clear, start heating the storage tanks and connected sodium lines in the following sequence:

   a. Start heating with some top heaters, with intermittent operation of the heaters so that the temperatures will equalize. The temperature rise should be checked and recorded on the heatup log sheets, which are designed to receive data in the sequence in which the heating should take place.

   b. As the storage tank temperature increases, the heaters can be kept on for longer periods, and more heaters can be turned on if needed. The tank temperature can be raised to about 375 F and kept at that point. All areas should be checked for sodium leaks during heating.

   c. The connected lines can be heated either while the storage tanks are heating or when the tanks are at 375 F by turning on the apprioriate breakers. The heating should proceed from the storage gas space outward, with the temperature of the first section being at least 250 F before prcoeeding to heat the next section of pipe. Data on heating should be recorded on the heatup log sheets. Before any sodium is transferred, all parts of the transfer route must reach at least 250 F, a temperature which may take several days to reach in the tanks and connected lines.

d. When one or more tanks and appropriate connected lines reach at least 250 F, the line to the drum-filling station should be heated by activating the powerstats and power breakers. Record the thermocouple readouts from a potentiometer or the readouts from a pyrometer on the heatup log sheets. Eight hours should be allowed to bring this portion of the system to operating temperature.

APPENDIX II

PROCEDURE FOR RESEALING PRIMARY SODIUM STORAGE DRUMS

1. With bungs intact, remove sodium reaction products from drum top by scraping and vacuuming.
2. Remove bungs one at a time for cleaning and resealing.*
3. Clean bung threads with steel wool and moist towels. Check for thoroughness of cleanup with pH indicator paper.
4. Replace bung gasket, apply carbon thread seal to bung plugs, insert bung and tighten.
5. After both bungs have been cleaned and replaced, continue cleaning top surface of drum with steel wool and moist towels. Check thoroughness of cleanup with pH indicator paper. Wipe dry.
6. Allow drum to dry thoroughly.
7. Apply paint to top surface and bung with a brush. Dab paint around bung to form secondary gas seal.
8. Re-inspect drums for other leaks. Set leaker drums aside.
9. Seal leaker drums in overpack drums.

*Spare bungs, precleaned and with new gaskets and Grafoil carbon tape, were available to immediately replace those removed as soon as the female bung threads were clean.

APPENDIX III

PROCEDURE FOR CO_2 PASSIVATION OF SODIUM

1.0 FARB OVERFLOW AND TRANSFER TANK AND PRIMARY SYSTEM OVERFLOW TANK

The procedure for the initial passivation of the residual sodium in the FARB Transfer Tank and Overflow Tank and the Primary System Overflow Tank using CO_2 were described in Report NP-20047. After the initial passivation with CO_2, these vessels were allowed to breathe ambient air as a means of continuing passivation of the residual sodium. Vapor space hydrogen concentrations of these vessels as a function of time after continuous exposure to ambient air are presented in Table A3-1.

2.0 PRIMARY SODIUM TANKS

The procedure for the passivation of the residual sodium in the primary system sodium storage tanks after they are emptied sometime in the 1981-1985 period is as follows:

> On removal of sodium from the primary sodium tanks, a nitrogen cover gas is maintained until the remaining heel of sodium is removed by a drain line installed in the bottom of these tanks, the sodium drained, and passivation by CO_2 begun.
>
> Passivation of the residual sodium is accomplished using the same procedure as that used to passivate the transfer tank, i.e., applying a CO_2 blanket before opening the tanks open to atmosphere and monitoring the hydrogen generation until its concentration levels out to an acceptable value.
>
> After the primary sodium in these tanks is shipped offsite, a nitrogen cover gas will be maintained on these tanks and inspections continued until the tanks are passivated.

This procedure was adopted based on the past success of passivating the FARB Transfer Tank and Overflow Tank and the Primary System Overflow Tank. The 50 percent caustic dissolution technique was not used because of the difficulties Fike Chemical Company experienced when contacting a drum of frozen sodium with a 50 percent caustic solution.

TABLE A3-1

HYDROGEN CONCENTRATIONS IN THE FARB TRANSFER TANK AND OVERFLOW TANK AND THE PRIMARY SYSTEM OVERFLOW TANK AFTER INITIATION OF AMBIENT AIR* PASSIVATION

	Hydrogen Concentration, Percent**	
Date	FARB Transfer Tank and Overflow Tank	Primary Sodium System Overflow Tank
4/8/74	Opened to ambient atmosphere	
4/26/74		Opened to ambient atmosphere
5/14/74	0.095	0.15
6/20/74	0.05	0.02
9/30/74	0.22	0.27
12/17/74	0.026	0.006
4/9/75	0.37	0.18
9/23/75	0.34	0.26

*Tanks breathe ambient air. No forced convection.

**Determined from grab samples analyzed at Detroit Edison's Engineering Research Department on a gas chromatograph.

APPENDIX IV

PROCEDURE FOR SHIPMENT OF THE IRRADIATED BLANKET

Shipping Cask

The cask is equipped with closures at both ends to accommodate a practical method of unloading at the Idaho Chemical Processing Plant (ICPP). For unloading at ICPP, the top of the cask is designated as the small-shield-plug end cover and can be readily identified as such. For loading at Fermi-1, the top of the cask is designated as the end without the small shielding plug in the center of the cover.

Arrival of the Cask. (Inspection on arrival of cask. Fill out and sign inspection form.)

* 1. Release cask from vehicle mounting cradle.

2. Remove impact limiters from both ends by removing cover bolts on each end.

* 3. Vent any pressure in cask. (Health physics to connect a particulate filter on vent.)

4. Attach lifting device to top cover of cask with 2 bolts, and tighten the 2 bolts to 100 ft-lb.

* 5. Remove shield plug from the center of the bottom cover of cask by removing 4 cap screws.

6. Attach crane hook to lifting yoke and position over top end of cask.

7. Attach lifting yoke to trunnions at top of cask and test with partial weight of cask.

8. Lift cask from trailer and position the cask on the support plate in the maintenance pit.

9. Disengage yoke from cask. Place yoke in storage area.

10. Attach cask cover lifting device to crane hook.

11. Remove 6 bolts from top cover of cask and loosen the remaining 2 bolts to finger tightness.

* 12. Lift cask cover from cask and place in storage area.

At this point, the shipping cask is ready to receive the fuel basket container (FBC). The FBC can be placed in the shipping cask with the basket inside or the basket may have already been removed from the FBC for loading in the cutup pool.

Fuel Basket Container and Basket

The fuel basket container and basket should arrive with the basket inside

*Radiation survey required at this point and results recorded.

the container. The container will have a special shipping cover with lifting lugs. This shipping cover will be bolted to the fuel basket container. The permanent FBC cover will arrive with the FBC. These covers and FBC will be bench marked for proper fit. Also, each shipping container will be stamped on both ends with a number 1, 2, etc.

Arrival of FBC and Basket. (Fill out and sign Form #2, Inspection and Arrival of FBC and Basket.)

1. If FBC is not to be used for a day or two, cover it with plastic sheet to keep it clean.

Loading FBC into Shipping Cask. (Complete Form #2, Inspection and Arrival of FBC and Basket.)

1. Lift the FBC using the temporary shipping cover equipped with the lifting ring. Position FBC over shipping cask. Lower FBC into shipping cask to ensure full insertion. When fully inserted, the top should be 1/2 inch below the lower "step" in cask wall.

2. Lift the FBC, using the temporary shipping cover equipped with the lifting ring, and set it on blocks.

3. From the bottom of the FBC screw the 1-1/2 inch-diameter steel support rod into the bottom of the FBC. Use Moly-Cote or equivalent lubricant on threads of support rod.

4. Lift and insert FBC into cask cavity until it bottoms against steel support rod. Top of FBC should be about 1.0 inch above the top of the cask. This FBC must be the same one that has been tested for fit. (Arrived with the basket that is (or will be) loaded for shipment.)

5. Remove the 4 bolts from the temporary shipping cover and remove the cover. The temporary shipping cover with the lifting eyes will remain at the plant. The other temporary shipping covers will be returned with the truck for subsequent FBC shipments from Central Ohio Welding Company, Columbus, Ohio.

Loading of Basket. (Check to ascertain that orientation marks are fully visible and for basket cleanliness before insertion into pool.)

1. Attach crane hook to fuel basket lifting device and attach lifting device to basket lifting blocks.

2. Raise basket to position over cutup pool.

3. Lower basket into cutup pool to loading position.

4. Disengage basket lifting device from lifting blocks.

5. Place basket lifting device in storage area.

6. Load basket per loading instructions (Form #3 will be used to record loading). Record type and number of subassemblies in each cell and check to ascertain that all cells are filled to within 1 inch of top of basket and that none extend above the top of the basket. Each basket will have a loading log -- loading 1, 2, 3,-----14, etc.

7. After basket loading is completed, lower basket lifting device into cutup pool and attach lifting device to basket lifting blocks.

* 8. Barricade the area prior to raising basket for drying and initiate interlock defeat order to bypass the radiation evacuation alarm (RE-753).

* 9. Raise basket above pool for drying (operator must be behind radiation shield). The basket will remain in this position overnight. Dry is defined as no drippage from basket onto pool surface and no wetting of observable surfaces.

* 10. After drying, position the basket over the shipping cask.

Loading Fuel Basket into Fuel Basket Container. (Care must be taken to ensure proper orientation of basket to FBC. Health physics personnel to monitor each step of this operation and record the results of radiation and contamination readings on the appropriate forms.)

1. Lower fuel basket into fuel basket container. Be sure that the orientation marks placed on FBC and basket, Form #2, Item #4 and #5 line up for proper fit before lowering basket. Be sure that basket is fully inserted into FBC. Check fully the insertion orientation mark (Form #2, Item #5) using the scope.

2. Disengage lifting device from basket lifting blocks.

    a. Check lip of FBC for contamination. Decontaminate as necessary.

    b. Sign off interlock defeat order and restore RE-753.

3. Install fuel basket container cover. Be sure orientation marks on FBC and cover line up and that cover seats properly. This will place the small plug of cover in the proper position over an empty portion of the FBC.

4. Install radiation shield for welding cover in place.

5. Make cover seal welds on FBC, as specified in the Safety Analysis Report prepared by Battelle Memorial Institute. Fill out welding form (Form #5).

6. Install leak check tube for leak checking weld.

7. Pressurize FBC to 5 psig with helium. Record results.

8. Run a mass spectrometer check of weld on FBC (sniffing technique). Record results (must be $<1 \times 10^{-4}$ atm cc/sec).

    a. Lower pressure on FBC to one atmosphere through particulate filter.

*Radiation survey required at this point and results recorded.

9. If result of mass spectrometer check is satisfactory, remove leak check tube, insert and tighten plug. If not, reweld.

10. Seal-weld plug.

11. Install leak check adapter and leak check plug weld. Record results.

12. If result of mass spectrometer test on plug is satisfactory, remove welding (radiation) shield from top of FBC. If not, reweld.

13. Attach cask yoke to crane and engage top trunnion of cask.

14. Slowly lift cask to remove 1-1/2 inch-diameter support rod from bottom end of cask and set on blocks.

15. Remove support rod, install shield plug, and finger tighten the 4 cap screws.

16. Lift cask and move to position over support plate at pool side.

17. Lower cask onto support plate.

18. Check cask cover gasket. Replace or repair as necessary.

19. Replace cask cover on cask.

20. Replace 4 bolts on quadrants and tighten to 10 ft-lb torque with wrench.

21. Check cask for contamination.

22. Replace remaining cover bolts and tighten all bolts on cover to 100 ft-lb.

23. Load cask onto trailer and secure it to cradle.

24. Tighten shield plug (small plug on bottom) cap screws to 50 ft-lb.

25. Install impact limiters and torque bolts to 100 ft-lb.

26. Pressure check cask with air at 3 to 5 psig. Observe pressure for 10 minutes to check for cask leakage. Record results on Form #6, Item #5.

27. Seal all cask closures with safety wire.

28. After final inspection for contamination, the cask is ready for shipment. Fill out final Inspection Form #6.

FORM #1

PRDC SHIPMENT OF FERMI-I IRRADIATED BLANKET
FUEL SUBASSEMBLIES TO ICPP IN IDAHO

Inspection on Arrival of Shipping Cask

Inspector's Initials

Freight Bill No. ____________

Cask No. ____________

Date and Time ____________

1. Visual Inspection of Cask for Damage ____________

2. Visual Inspection of Transport-Vehicle-Mounting Cradle and Trailer for Damage ____________

3. Monitor External Radiation Level. Record Here and on Form #4, Item #2

   Radiation Level @ Surface ________ Mr/Hr
   @ One Meter ________ Mr/Hr

   Smear Test ________ α dpm/100 cm^2
   ________ β dpm/100 cm^2 (H.P.) ____________

4. Check Cask Pressure Gauge and Vent.
   (H.P. to attach a particulate filter on vent.)

   Pressure Release Particulate Filter ________ dpm
   (H.P.) ____________

5. Do Not Break Seal Wire on Trunnions That are Sealed. ____________

6. Remove Road Dirt from Cask and Mounting Cradle. (By flushing outside of cask with plant water.) ____________

Remarks: ____________________

____

Inspected By ____________________
(Supervisor)

____
(Health Physics)

PRDC SHIPMENT OF FERMI-I IRRADIATED BLANKET
FUEL SUBASSEMBLIES TO ICPP IN IDAHO

Inspection on Arrival of Fuel Basket Container (FBC) and Basket

Inspector's Initials

Freight Bill No. ____________

FBC No. ____________

Date and Time ____________

1. Visual Inspection of FBC and Cover for Damage. Mark number of FBC on bottom. ____________
2. Check to see that FBC and Permanent Cover are Bench Marked and Numbered for Proper Identification and Orientation. ____________
3. Remove Shipping Cover (4 Bolts) for Visual Inspection of FBC Interior and Basket. Lift Basket out of FBC. (Check for dirt, foreign material, oil, etc. Clean if necessary.) This will also check the basket for full withdrawal and full insertion. Replace basket in FBC. ____________
4. Lift Permanent Cover and Place on FBC to Check for Proper Orientation and Fit of Permanent Cover to FBC. Another Orientation Requirement is to Ensure that the Test Plug on the Permanent Cover is Positioned Over an Empty Portion of the Fuel Basket Container. Remove Permanent Cover. ____________
5. Mark Basket with Number Stamped on FBC. Also Mark Basket to FBC for Proper Orientation and Full Basket Insertion Check Mark. ____________
6. Replace Shipping Cover on FBC and Bolt Down. (Use Temporary Shipping Cover with Lifting Rings.) ____________
7. Place FBC and Permanent Cover in Storage Area (or proceed to Section C, Step 1 of Procedure). ____________

Inspected By ____________

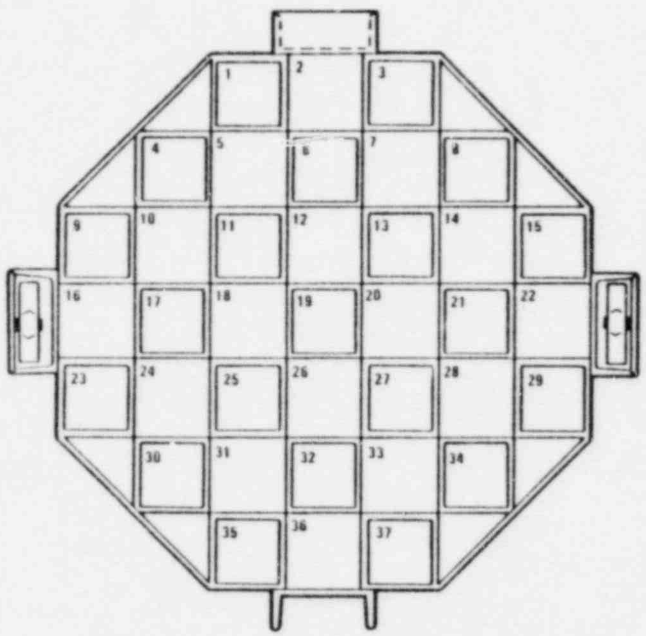
Date and Time ____________

FORM NO. 3

PRDC Shipment of Fermi-1 Irradiated Blanket Fu
Subassemblies to ICPP in Idaho

Basket Loading History for Shipment No.

Basket loading verified by



SKETCH OF CROSS SECTION OF BASKET

LOADING CODE:

ORB - OUTER RADIAL BLANKET
IRB - INNER RADIAL BLANKET
UAB - UPPER AXIAL BLANKET
LAB - LOWER AXIAL BLANKET
ORB U - OUTER RADIAL BLANKET (UNCUT)
LAB U - LOWER AXIAL BLANKET (UNCUT)

CELL NO.	LOADING CODE AND/OR SERIAL NO. AND SPACER TUBING LENGTH							
1								
2								
3								
4								
5								
6								
7								
8								
9								
10								
11								
12								
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FORM #4

PRDC SHIPMENT OF FERMI-I IRRADIATED BLANKET
FUEL SUBASSEMBLIES TO ICPP IN IDAHO

Radiation History for Shipment No. _____

1. Cask No. _______________

   FBC No. ________________

2. Radiation Survey of Shipping Cask (Arrival)

   Radiation Level @ Surface ______ Mr/Hr ($<$200 Mr/Hr)
   @ One Meter ______ Mr/Hr ($<$ 10 Mr/Hr)

   Smear Test _________ α dpm/100 cm^2 ($<$220 dpm/100 cm^2)
   _________ β dpm/100 cm^2 ($<$2200 dpm/100 cm^2)

   Pressure Release Particulate Filter _________ dpm

3. Radiation Survey -- Shipping Cask Interior

Top Plug	Bottom Plug	Cask	
_______	__________	_________	α dpm/100 cm^2
_______	__________	_________	β dpm/100 cm^2
_______	__________	_________	Mr/Hr @ 6 in.

4. Radiation Survey -- Fuel Basket Removed From Pool

   Radiation Level--Pool Room (Exterior) ___________ Mr/Hr
   FARB (Exterior) ______________ Mr/Hr
   Other: ______________________

   Plant Supervisor Notified of Results: _______________________

   Date ________________ Time ______________ Surveyor ___________________

5. Radiation Survey -- Before and During Loading of Fuel Basket into Fuel Basket Container (FBC)

   Working Area __________Mr/Hr Working Time _______________

   Particulate Sample No. ___________ _________ x 10^{-}___ μCi/cc

   Other: _______________________________________

   Date ________________ Time ______________ Surveyor ___________________

Radiation History for Shipment No. ______

6. Smear Survey of FBC Lip ______________ α dpm/100 cm^2

______________ β dpm/100 cm^2

7. Radiation Survey -- Placement of Welding Shield

Work Area ____________ Mr/Hr Working Time ____________ (Min.)

Remarks:

Date ________________ Time ________________ Surveyor ________________

8. Radiation Survey -- Welding of FBC Cover

Work Area ____________ Mr/Hr Working Time ____________ (Min.) (Hrs.)

Extremities ____________ Mr/Hr

9. Radiation Survey -- Mass Spec. Test

Work Area ____________ Mr/Hr Working Time ____________ (Min.) (Hrs.)

10. Radiation Survey -- Installation of Cask Cover - Plug

Work Area ____________ Mr/Hr Working Time ____________ (Min.)

Remarks:

Date ________________ Time ________________ Surveyor ________________

11. Radiation Survey -- Shipping Cask (Repair Pit)

Smear Survey ________________ α dpm/100 cm^2

________________ β dpm/100 cm^2

12. Radiation Survey and Release Data -- Shipping Cask

Radiation Level @ Surface ____________ Mr/Hr ($<$ 200 Mr/Hr)

@ One Meter ____________ Mr/Hr ($<$ 10 Mr/Hr)

Smear Test ________________ α dpm/100 cm^2 ($<$ 220 dpm/100 cm^2)

________________ β dpm/100 cm^2 ($<$2220 dpm/100 cm^2)

Date ________________ Time ________________ Surveyor ________________

PRDC SHIPMENT OF FERMI-I IRRADIATED BLANKET
FUEL SUBASSEMBLIES TO ICPP IN IDAHO

Cover Seal Weld and Helium Mass Spectrometer Leak Detector

Date ______________________

Fuel Basket Container # ____________________

Welder's Name ______________________________

Welding Process: arc welding 1/4" fillet

Material Spec: 1/4" 316 SS to 3/4" 316 SS cover

Welding Rod: 5/32, Titanium, AC-DC Type 316-16 certified

Welder's Qualification Record Attached

Welding performed as per BMI SAR Appendix 4-F ______________________

Mass Spectrometer Leak Test:

Date of Test ______________________

Leak check of fuel basket container # __________ was made as per appendix 4F, Q.A. Section 12.2 of Safety Analysis Report (SAR) for Fermi Blanket Fuel Subassemblies.

The helium overpressure of the FBC was approximately 5.0 psig. Probing speed was about 1 ft/minute and probe distance from weld surface was no more than 1/16 inch.

The apparent maximum leak for the above conditions was about __________ atm cc/sec. Therefore, it is reasonably certain that a leak of 1×10^{-4} atm cc/sec. does not exist.

Leakage on plug weld was about ______________ atm cc/sec. Therefore, it is reasonably certain that a leak of 1×10^{-4} atm/cc sec. does not exist.

____

PRDC SHIPMENT OF FERMI-I IRRADIATED BLANKET FUEL SUBASSEMBLIES TO ICPP IN IDAHO

Power Reactor Development Company
P. O. Box 725
Monroe, Michigan 48161

Idaho Chemical Processing Plant
National Reactor Testing Station
Idaho

Date ____________________

Shipment # ____________________ Loading # ____________________

Cask # ____________________ Trailer # ____________________

Truck # ____________________ Driver ____________________

1. Cask mounted on trailer with I.D. tag "up." Visual inspection of cask for damage. ____________

2. Visual inspection that both impact limiters are in place and properly bolted. ____________

3. Visual inspection that trunnion covers are in place and scaled with lockwires. ____________

4. Visual inspection that cask is secured to mounting cradle of trailer. ____________

5. Pressure check cask 3 to 5 psig. Observe pressure for 10 minutes. Record Pressure. (Should be 3 to 5 psig; if not, action taken to correct leakage and results rechecked.) ____________

6. All radiation warning signs properly placed and readable. ____________

Inspected and Recorded by:

____

____
Health Physics

APPENDIX V

SAFETY ANALYSIS OF BLANKET SHIPPING CASKS

The attached sheets, which describe the casks used for the shipment of the Fermi-l blanket, are excerpted from Addendum No. 4 to "Safety Analysis for the Shipment of Irradiated Fermi Blanket Fuel Subassemblies in the Whitehead & Kales Shipping Casks Models PB-1 and PB-1 (1972)," August 14, 1974, Battelle Columbus Laboratories.

The original safety analysis report, dated January 15, 1970, provides the safeguard evaluation of the design and use of the Model PB-1 casks for shipping irradiated Peach Bottom No. 1 fuel elements to the Idaho Chemical Processing Plant (ICPP). Addendum No. 4 provides the safeguard evaluation of the design and use of the Model PB-1 casks for transporting irradiated Fermi-1 blanket from the Fermi-1 plant at Monroe, Michigan, to ICPP. To minimize the overall time required to ship all blanket elements, two identical Whitehead & Kales shipping casks, identified as PB-1 and PB-1 (1972), were utilized. The PB-1 cask is owned by Philadelphia Electric Company, and PB-1 (1972) is owned by Gulf Atomic Company.

Each shipment of blanket elements was contained within a sealed fuel basket container (FBC) placed within the shipping cask. A given shipment consisted of either outer radial, inner radial, upper axial, or lower axial blanket elements, or mixtures thereof.

All shipments were made in accordance with Atomic Energy Commission (AEC) and Department of Transportation (DOT) regulations, 10-CFR-Part 71 and 49-CFR-Parts 171-179, respectively, for transporting large quantities of Fissile Class I radioactive materials by motor vehicle assigned for the sole use of the Power Reactor Development Company and were unloaded from the motor vehicle by the ICPP personnel.

The drawings mentioned in the excerpt are large drawings which are not attached hereto. They were prepared, numbered, and retained by Battelle Columbus Laboratories. Two sketches, one showing the basket and basket container and the other showing the basket and basket container inside the PB-1 cask, are included in the excerpt.

EXCERPTED FROM

ADDENDUM NUMBER 4
to
SAFETY ANALYSIS
for

THE SHIPMENT OF IRRADIATED
FERMI BLANKET FUEL SUBASSEMBLIES
IN THE WHITEHEAD & KALES SHIPPING
CASKS MODELS PB-1 AND PB-2

from

BATTELLE
Columbus Laboratories

August 14, 1974

PACKAGE DESCRIPTION

Description of Cask

Cask Design

A design layout of the W&K Model No. PB-1 and PB-1(1972) shipping casks for transporting irradiated Peach Bottom No. 1 reactor fuel elements is shown in Drawing 9123-0001. The two casks which are identical have an empty scale weight of 57,050 lb compared to a calculated empty weight of 58,460 lb which was used as the design basis for the analysis of the original SAR. The maximum loaded cask weight for the Fermi blanket shipments based on the actual empty scale weight will be 67,050 lb compared to a calculated loaded weight of 62,800 lb as used in the original SAR. The gross payload weight for the Fermi shipments is approximately 10,000 lb which includes the fuel basket container (FBC), the basket, and the blanket elements. The cask has an outer diameter of 42.5 in., an overall length of 191.1 in., including impact limiters, and a width across the trunnions of 50.0 in. The cask internal cavity is 26.0 in. in diameter and 159.0 in. long.

The cylindrical cask body is constructed with a 0.25-in., 304 stainless steel cavity liner, a maximum of 6.25-in. chemical lead, a 1.50-in. mild-steel outer shell, and a 0.25-in., 304 stainless steel overlay. The cavity liner is seam welded and polished to a No. 3 finish. It is welded at both ends to offset cones which form cavities for the end closures. The lead thickness is 5.25 in. from the bottom of the cavity to 24.5 in. above the bottom; it is 6.25 in. thick from 24.5 in. above the bottom to 134.5 in. above the bottom; and it is 5.25 in. thick over the remainder of the length. Since lead shrinks upon solidification, a patented fin arrangement* is used to attach the lead to the inside of the outer shell. The fins bridge the gap between the lead and outer shell and enhance the transfer of heat. A venting device** for the lead cavity prevents a buildup of excessive pressure from either moisture or lead during a fire-temperature excursion. The stepped outer shell is constructed by welding three coaxial, formed and welded, mild-steel cylinders. The overlay is welded to the outer shell at the end of each cylinder, and at cutouts around each

*Edward Lead Company - U.S. Patent No. 3,005,105 (1958)
**Edward Lead Company - U.S. Patent No. 3,466,444 (1969)

trunnion. It is spaced from the outer shell by 1/16-in. spot welded spacers on the outer shell and serves as a heat shield to inhibit lead melting during the hypothetical fire-temperature excursion. A pressure relief plug in the overlay shell prevents a buildup of excessive pressure from moisture.

The end covers have 4.00 in. of chemical lead sandwiched between two 1.50-in. 304 stainless steel plates. An impact limiter is attached to each end which also serves as a heat shield. The covers are tapered to allow easier alignment during closure.

Guide pins provide final alignment of the cover bolt holes with tapped holes in the ends of the cask body. Twelve 1.25-in.-diameter ASTM A325 cadmium-plated steel bolts secure each cover. A silicone-rubber O-ring gasket seal is used between the cask seat at each end to provide secondary containment of the cask contents.

The basket grid assembly contains 37 positions within which the Fermi blanket subassemblies are stacked. A sketch of the basket is shown in Figure 1; construction details are shown in Drawing 000-000-553. Essentially, the basket is constructed of a series of 21 - 3 x 3 x 0.083-in.-thick wall square steel tubes arranged and welded together such that 37 grid positions are formed as shown. Sufficient structural members are welded to the outside to make the assembly free standing and capable of supporting the blanket assemblies without failure. The bottom support structure consists of two 1/4-in.-diameter round bars welded to the bottom of each square tube as shown. Two spring loaded lifting devices are welded to the sides of the basket so that the lifting yoke can be attached and detached from the basket remotely. Spacers welded to the outside of the basket limit the free motion of the basket within the FBC during transit. The basket contains no materials intended as nonfissile neutron absorbers or moderators.

The basket grid assembly is contained within a sealed and helium leak checked Type 316 stainless steel fuel basket container (FBC) which has dimensions of 25.5-in. OD x 25.0 in. ID x 158.5-in. overall length. The top and bottom covers are 0.75 in. thick and are heli-arc welded* to the 1/4-in.-thick sidewall of this container. A threaded hole in the base of both covers accommodates a lifting eye which is compatible to the handling system at ICPP. A sketch showing both the basket and FBC inside the PB-1 cask is shown in Figure 2.

All heat rejection is accomplished by conduction through the cask walls and radiation and convection from the cylindrical wall of the cask. The cask is designed to operate either wet or dry; however, all shipments are planned to be dry with air as the only heat transfer medium from the contents to the cavity liner of the cask. The basket will be wet loaded in the pool, raised out of the water and drained, dried, and inserted into the FBC. The top cover is then seal-welded closed and leak checked to provide containment for the blanket elements.

Four 8.0-in.-diameter lifting and pivoting trunnions are welded to the outer shell. An 0.5-in.-thick 304 stainless steel patch plate is welded to the outer shell at each trunnion for added strength. The trunnions permit changing the cask from the horizontal to the vertical position and vice versa with a minimum effort. They also serve as a convenient method of attaching the cask to the transport-vehicle-mounting cradle. The trunnions are hollow and provide protective housings for the drain valves, flush valve, pressure gage, pressure-relief valve, and valve exhaust filter. (The drain system will not be used

*Editor's Note: Stick welding was performed instead of heli-arc welding.

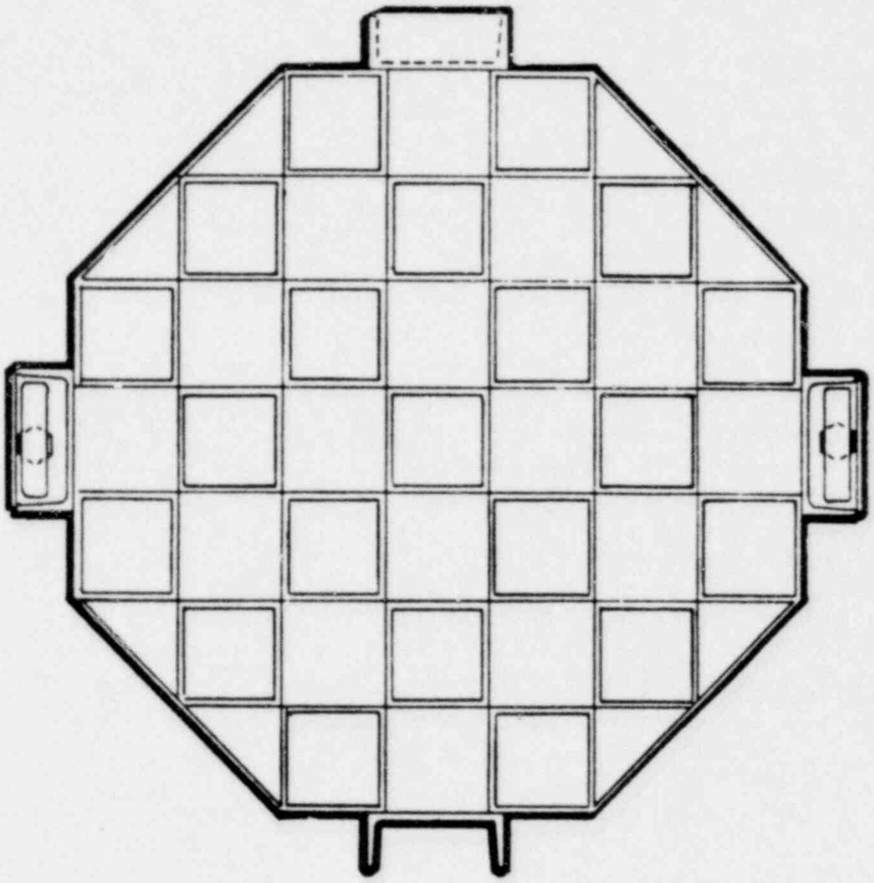


FIG. 1 SKETCH OF CROSS SECTION OF BASKET

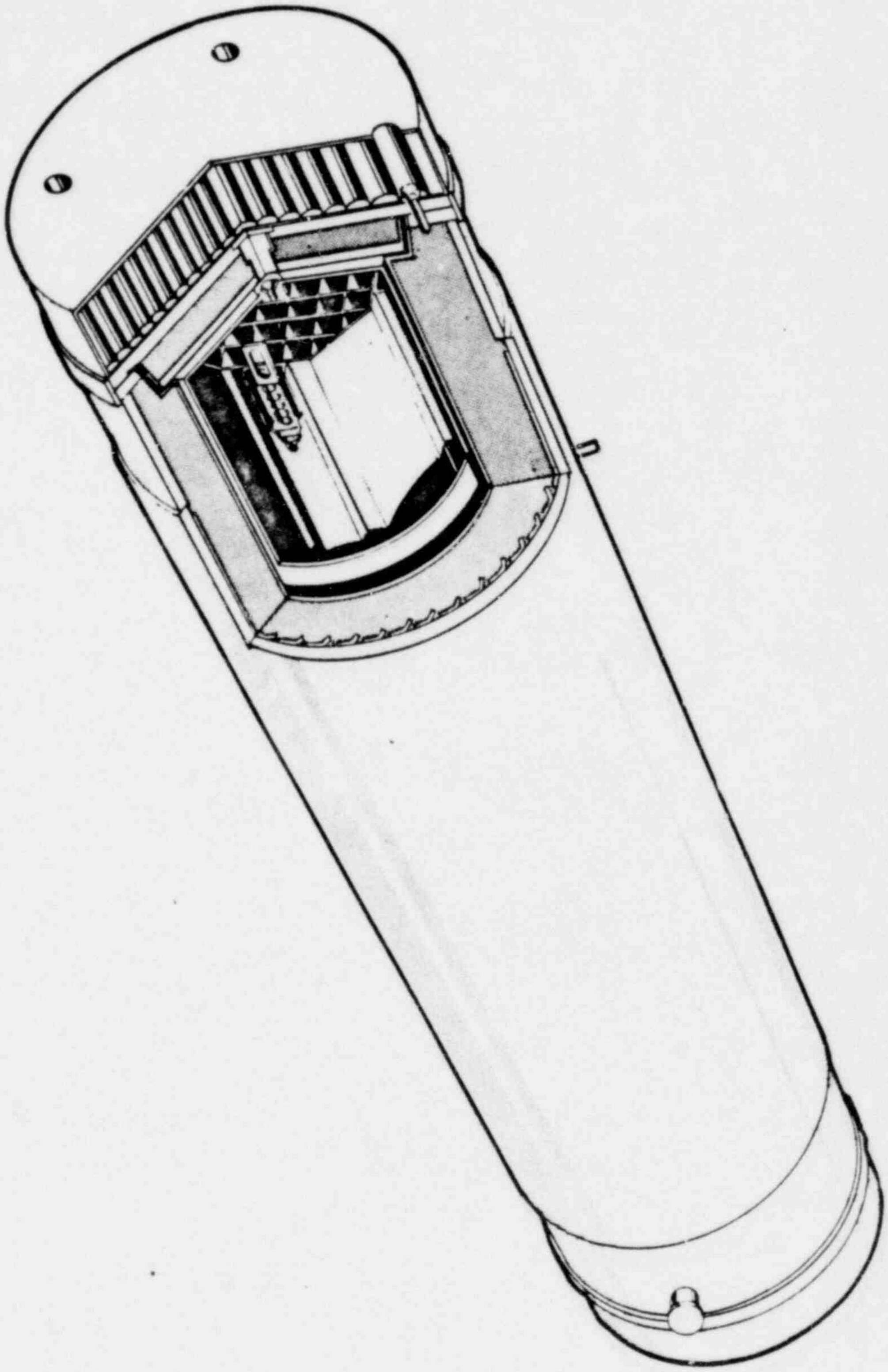


FIG. 2 SKETCH OF BASKET AND FUEL BASKET CONTAINER
INSIDE PB-1 CASK

since basket and contents will be dried prior to insertion into FBC.) The pressure-relief valve is set at a pressure of 100 psig.

The cask is mounted horizontally and handled during transport in a structural steel cradle that is designed to spread the load. Trunnion sockets on the vehicle-mounting-cradle support and secure the cask trunnions. They allow rotation of the cask from the vertical to the horizontal shipping position. One set of trunnion sockets is adjustable to accommodate changes in the length of the cask due to temperature changes. Pads on the vehicle-mounting cradle provide additional support to the cask when it is in the horizontal position.

An impact limiter is attached to each end cover with four of the twelve 1.25-in. cover bolts in order to limit the impact load on the fuel canisters after an accidental 30-ft drop. The impact limiters are constructed by welding a bundle of 2-1/2-in. nominal diameter x 13 gage mechanical tubing between 1/4-in. 304 stainless steel plates and enclosing the bundle with a 1/16-in. 304 stainless steel shell as shown on Drawing 9123-0001. A 4-in.-long skirt fits over the cask for added resistance to radial motion in a corner drop.

Weight of the cask and contents are given in Table 1.

TABLE 1. WHITEHEAD & KALES MODEL NO. PB-1 CASK WEIGHT

Component	Weight, lb	
	Calculated	Actual
Cask empty (includes covers & impact limiters) (Limiters at 630 lb ea.) (Covers at 1995 lb ea.)		57,050
Fuel basket container (FBC)	1,050	
Basket assembly	1,250	
Fuel subassemblies (maximum load)	7,700	
Total weight:	67,050 lb	

Construction Specification

Construction of the casks was performed in accordance with specifications and procedures outlined in the original SAR.

Description of Cask Contents

Cask Contents

In accordance with the requirements of paragraph 71.22(b) of 10-CFR-71 Subpart B, the materials planned for shipment in the Whitehead & Kales Model No. PB-1 cask are described as follows:

(1) Identification and Maximum Radioactivity of Radioactive Constituents

The maximum quantity of radionuclides which could ever be present in any one shipment is 8651 curies. This was determined from data for each blanket element supplied by PRDC.

(2) Identification and Maximum Quantities of Fissile Constituents

The cask filled with 47 highest burnup Fermi radial blanket subassemblies will contain the following fissile constituents:

Pu-239	1660 gms
U-235	9100 gms
(0.35% enriched)	
(U-238	2.6×10^6 gms)

(3) Chemical and Physical Form

The package will contain up to 47 irradiated radial blanket subassemblies or 227 upper and lower axial blanket subassemblies sealed in a fuel basket container. They will be spaced and supported by the basket as shown in Figure 1.

Details of the blanket subassemblies and the fuel basket container are presented in the next section.

(4) Extent of Reflection, Neutron Absorbers, and Moderator Ratio

Reflection, absorption, and moderation characteristics of this package contents are summarized as follows:

Extent of Reflections	Maximum
Nonfissile neutron absorbers assumed	None
Atomic ratio of moderator to fissile constituents	NA (there is no moderator material in the blanket subassemblies)

(5) Maximum Weight

The maximum weight of the package contents, excluding the fuel-element basket and fuel basket container is 7700 lb.

(6) Maximum Amount of Decay Heat

As indicated in the original SAR, the cask and fuel-element basket were designed to handle a decay heat of 12,690 BTU/hr or 3715 w. While the basket design and fuel basket container for the Fermi shipments are different, the maximum possible decay heat in any given shipment consisting of 47 radial blanket subassemblies is less than 99 watts. This is based on a decay heat of 2.1 w for the hottest radial subassembly (#336) and assuming that all 47 subassemblies in any one shipment have the same decay heat. Actually there are only 5 radial subassemblies which have a decay heat this high, the rest are all lower. Therefore, a conservative approach has been used for the maximum decay heat in a given shipment of radial blanket subassemblies. Similarly, the maximum decay heat for the maximum load of upper and lower axial blanket subassemblies was calculated to be 41 watts, which is even lower than for the maximum radial blanket. Thus, decay heat is not a problem in any of the proposed shipments.

APPENDIX VI

REVISED FIGURES FROM REPORT NP-20047

Some of the illustrations in Report NP-20047 contain inaccuracies. In each case where the inaccuracy is deemed to be significant and/or was not adequately covered in the text of this supplement, a revised illustration is included in this appendix. The revised illustrations, still identified by the figure numbers and page numbers used in Report NP-20047, are tabulated below.

Figure	Page	Subject
6.68	6.135	Overflow Tank Pump Shaft Seal
6.69	6.137	Overflow Tank Fill Line Seal*
6.70	6.139	Overflow Tank Gas Supply Line, Equalizing Line, Sodium Fill and Discharge Line, Overflow Line, and Pressure Detector Tube Seals
6.82	6.155	Final Boundary of Contaminated Area

*Retitled Primary Sodium Service System Piping Seals

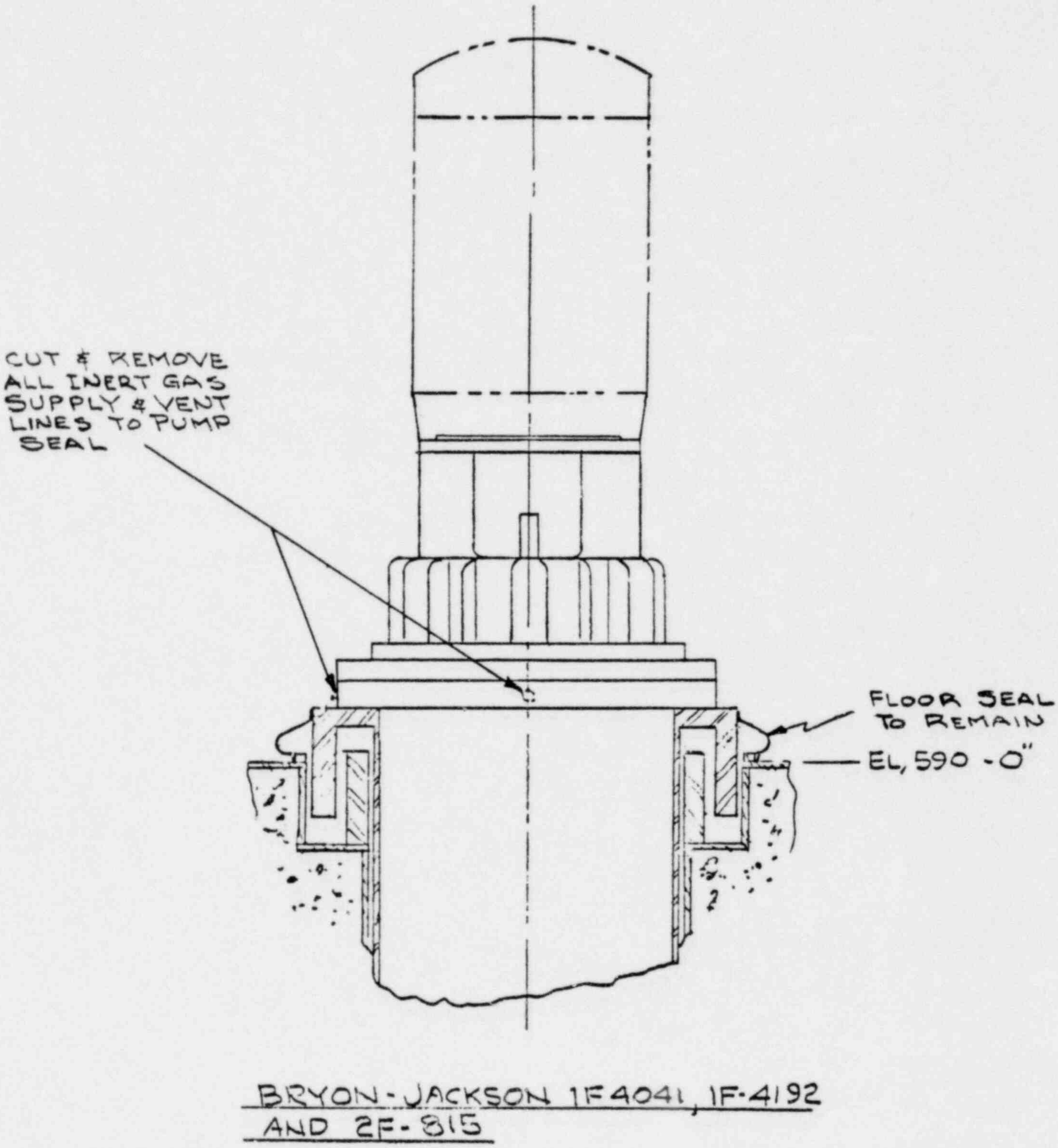


FIG. 6.68 OVERFLOW TANK PUMP SHAFT SEAL

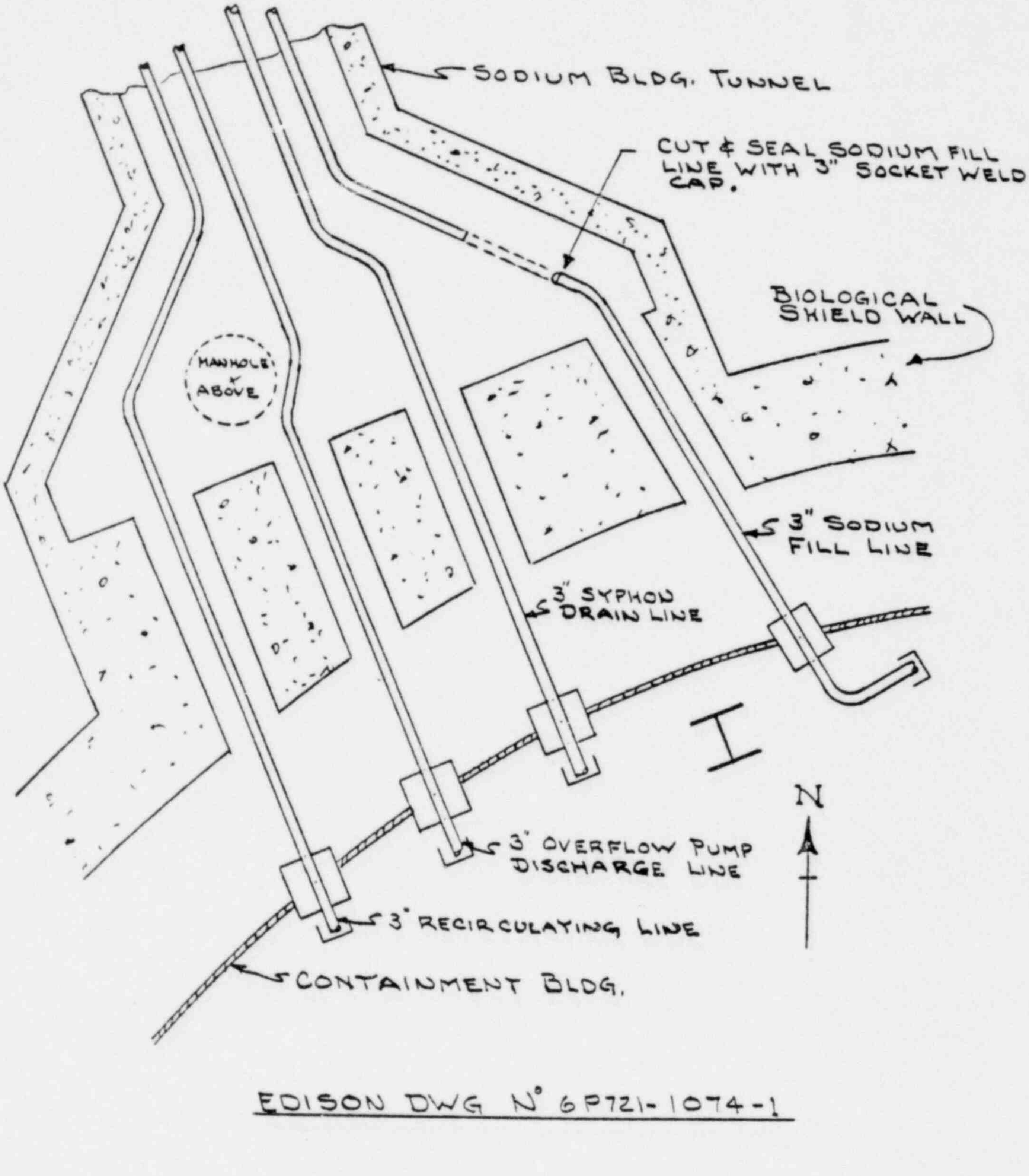


FIG. 6.69 PRIMARY SODIUM SERVICE SYSTEM PIPING SEALS

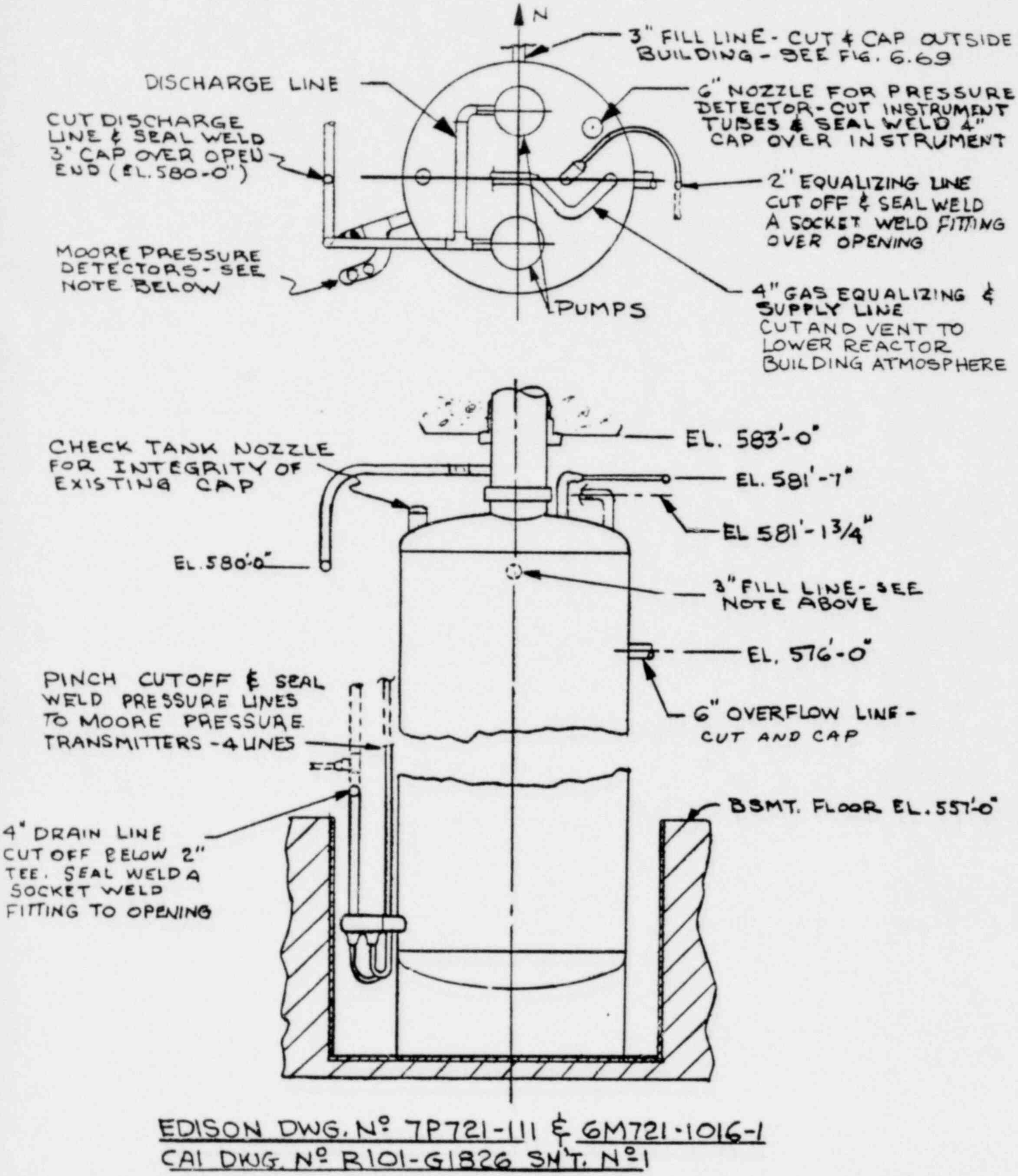
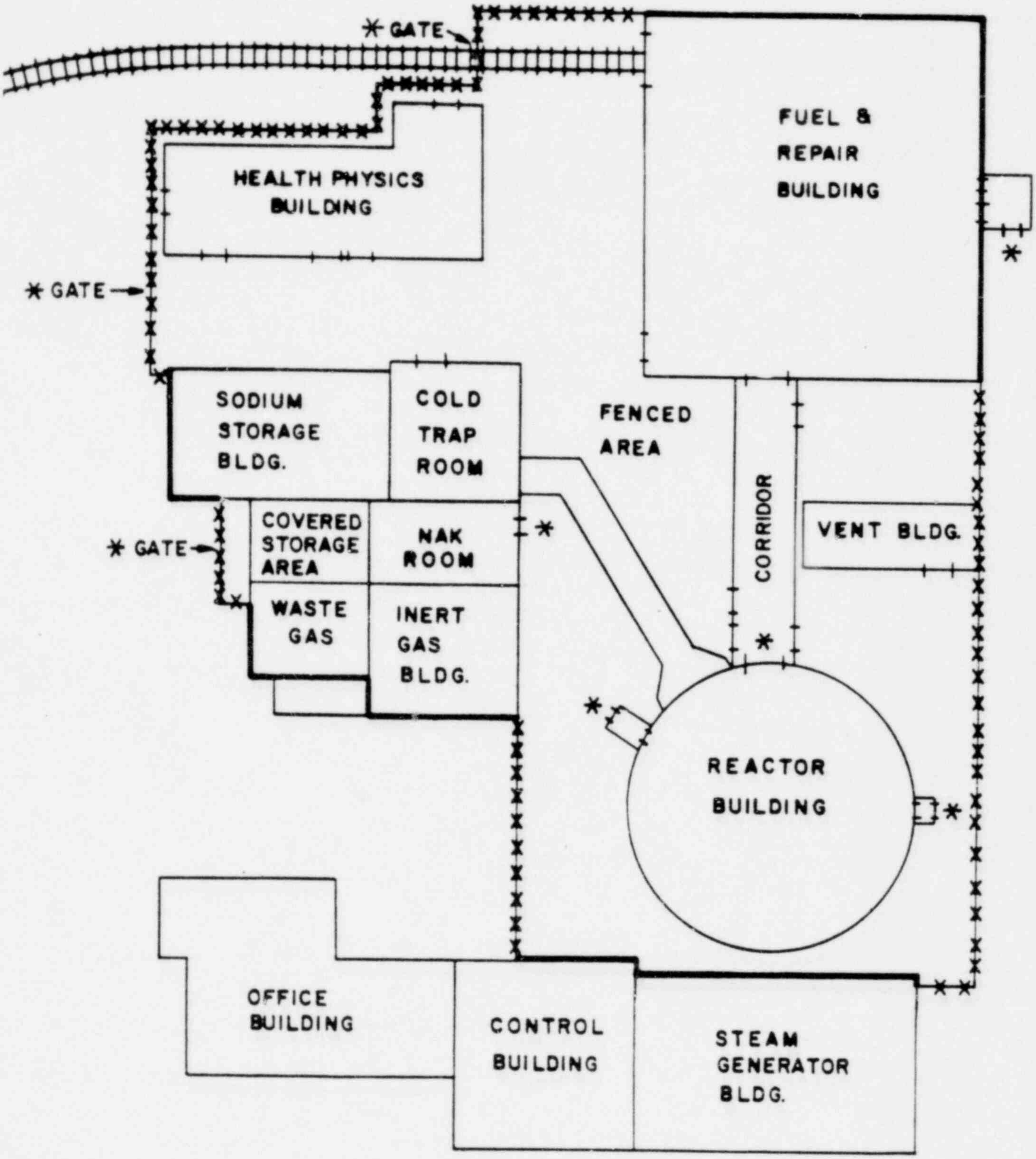


FIG. 6.70 OVERFLOW TANK GAS SUPPLY LINE, EQUALIZING LINE, SODIUM FILL AND DISCHARGE LINE, OVERFLOW LINE, AND PRESSURE DETECTOR TUBE SEALS



* THESE DOORS TO BE LOCKED OR PERMANENTLY CLOSED.

FIG. 6.82 FINAL BOUNDARY OF CONTAMINATED AREA

APPENDIX VII

PHOTOGRAPHIC HISTORY

A photographic record, in the form of a slide file, was made of the decommissioning activities described in Report NP-20047 and in this supplement. The photographs appearing in this report are examples of the 1029, 35-mm colored slides comprising the photographic history. A complete set of the slides is on file with the plant records, and the index to this file is reproduced on the following pages.

DECOMMISSIONING SLIDE INDEX

TRAY NO. 1

1. FARB FTF gripper cask and floor valve.
2. FARB FTF gripper cask stacked over transport car.
3. FARB FTF gripper cask stacked over transport car.
4. FARB cleaning machine and process system control consoles
5. FTF transport car in trestleway.
6. FTF transport car coming into reactor building.
7. Reactor building FTF gripper cask stacked over transport car at exit port.
8. Reactor building FTF gripper cask stacked over transport car at exit port.
9. Machinery dome access door; FTF transport car in background.
10. Reactor building FTF control console.
11. FTF transport car and plug cask at reactor exit port.
12. FTF gripper cask over reactor exit port.
13. FTF plug cask over reactor exit port.
14. FTF gripper cask over reactor exit port.
15. AFSF floor valve before dismantling.
16. Underwater gaging fixture dismantled and wrapped for contaminated waste storage.
17. Multiplication Tests: Cask loading.
18. Multiplication Tests: Tool handling from temporary pool bridge.
19. Multiplication Tests: Tool handling from temporary pool bridge.
20. Multiplication Tests: Cask loading.
21. Multiplication Tests: Core transfer.
22. Multiplication Tests: Tool handling from temporary pool bridge.
23. Multiplication Tests: Data taking.
24. IRB radiation exposure test.
25. IRB suspended for exposure test.
26. Subassembly locations in pool.
27. New segment storage rack.
28. New segment storage rack.
29. Pool bridge crane controls.
30. Subassembly storage in decay pool.
31. Subassembly storage in decay pool.
32. Installing Subassembly in tipping fixture.
33. Subassembly installed in tipping fixture.
34. Subassembly No. 203 in tipping fixture.
35. Lowering Subassembly tipping fixture.
36. Subassembly No. 203 before cutting.
37. Core subassembly before cutting.
38. Operating the cut-up machine controls.
39. Cut-up machine controls.
40. Cut-up pool floor during cutting operation.
41. Cutting of last (No. 203) core subassembly: Rateick, O'Mara Wagner, Gray, Elward
42. Cutting No. 203: Wagner
43. Cutting No. 203: Rateick, O'Mara , Wagner, Gray, Elward, Betterly
44. Cutting No. 203: Rateick, Wagner
45 TV screen: blade approaches subassembly
46. Upper segment cut begins

TRAY NO. 1 (Continued)

47. TV screen: blade begins cutting.
48. Subassembly No. 203 in cut-up machine.
49. Upper segment cut begins.
50. TV screen: blade cutting
51. Subassembly No. 203 upper segment cut through
52. Upper segment cut through.
53. Lower segment cutting begins.
54. TV screen: blade cutting lower segment.
55. Lower segment cutting.
56. Lower segment of No. 203 cut through.
57. Picking up segment transfer tool.
58. Transferring a segment.
59. Picking up tong tool.
60. Using tong tool off temporary bridge.
61. Segment transfer operation from bridge.
62. Tong tool used to pick up No. 203 upper segment.
63. Tong tool grips No. 203 upper segment.
64. Tong tool grips subassembly upper segment.
65. Tong tool picks No. 203 upper segment.
66. Tong tool transfers segment.
67. Tong tool segment transfer to racks.
68. Tong tool transfers segment to rack.
69. Tong tool transfers segment to rack.
70. TV screen of segment transfer to rack.
71. Tong tool transfer of segment to racks.
72. Upper segment removed.
73. Core segment gripped.
74. Core No. 203 segment raised.
75. Core segment raised.
76. Tong tool raises core segment.
77. Transfer operations viewed on TV.
78. TV screen: tongs grip core segment.
79. TV screen: tong tool raises core segment.
80. Transfer operations off temporary bridge.
81. Transfer tool carries core segment to rack.
82. Core segment No. 203 deposited in segment rack.
83. Core segment No. 203 deposited in segment rack.
84. Core segment deposited in segment rack.
85. Control for releasing segment tong tool.
86. Core segment deposited in segment rack.
87. Transferring core segment to storage rack.
88. Segment storage rack.
89. Segment storage rack filled with segments.
90. Viewing transfers with spotting scope.
91. Core storage rack No. 1.
92. Core storage rack No. 1 loaded.
93. Core segment withdrawn from leaker can.
94. Problem subassemblies stored in leaker cans.
95. Core No. 203 nozzle segment remains in machine.

TRAY NO. 1 (Continued)

96. Nozzle segment being raised.
97. Nozzle segment No. 203 being raised.
98. Nozzle segment being transferred.
99. Nozzle segment being transferred.
100. Segment transfer, cutup machine to rack.
101. TV screen used to verify subassembly number.
102. TV picture of subassembly No.
103. Nozzle segment being inserted in rack.
104. Tong tool over stored segments.
105. TV picture of stored nozzle segments.
106. Cutup pool floor equipment.
107. Bridge crane grapple over leaker cans.
108. Bridge crane grapple connected to leaker can.
109. Removing cutup pool roof panels.
110. TV camera repairs.
111. Raising cut-up machine for repairs.
112. Raising cut-up machine for repairs.
113. " " " " "
114. " " " " "
115. " " " " "
116. " " " " "
117. Retrieving broken blade pieces.
118. Cutup machine top - one cutting head removed.
119. Cutup machine stored on pool room floor.
120. Cutup machine stored on pool room floor.
121. Cutup machine blades: three designs.
122. Cutup machine blades: three designs.
123. Raising empty cask.
124. Raising empty cask.
125. Raising empty cask.
126. Cleaning shipping cask.
127. Transferring empty shipping cask.
128. Transferring empty shipping cask.
129. Transferring empty shipping cask.
130. Lowering cask into pool room.
131. Lowering cask into pool room.
132. Cut-up pool room.
133. Empty cask on pool room floor.
134. Removing empty cask plug.
135. Surveying empty cask plug.
136. Cu-Cd liner in shipping cask.
137. Raising cask from pool room floor.
138. Raising cask from pool room floor.
139. Lowering empty cask into cut-up pool.
140. Lowering empty cask into cut-up pool.

DECOMMISSIONING SLIDE INDEX

TRAY NO. 2

1. Lowering empty cask into cutup pool
2. Lowering empty cask into cutup pool
3. Empty cask lowered to cutup pool floor
4. Raising core segment from segment storage rack
5. Raising core segment from segment storage rack
6. Lowering core segment into shipping cask
7. Lowering core segment into shipping cask
8. Lowering core segment into shipping cask
9. Core tool moves to remove core segment from segment storage rack
10. " " " " " " " " " "
11. Core segment raised from segment storage rack
12. Core segment moved toward shipping cask
13. Core segment dropped on top of cask (#L017)
14. Core segment dropped on top of cask (#L017)
15. Segment storage on cutup pool floor
16. Tong tool moved to engage segment
17. Tong tool over cask and segment L017
18. Retrieving core segment L017
19. Tong tool over cask and segment L017
20. " " " " " " "
21. " " " " " " "
22. Installing L017 in segment storage rack
23. Six core segments stored in cask
24. Segment L017 in segment storage rack (wrapper without pin)
25. Six core segments stored in cask
26. Raising core segment from leaker can
27. Six core segments stored in cask
28. Loading core segment into cask
29. Six core segments stored in cask
30. Core segment deposited in cask
31. Cask plug picked up
32. Cask plug moved over pool water
33. Installing plug in shipping cask
34. Installing plug in shipping cask
35. Raising loaded shipping cask from pool
36. " " " " " "
37. " " " " " "
38. " " " " " "
39. " " " " " "
40. " " " " " "
41. Draining shipping cask water
42. Sampling shipping cask water
43. Raising shipping cask from cutup pool
44. Cask set on pool room floor to change lifting fixtures
45. " " " " " " " " " "
46. Lifting chain removed from cask
47. Lifting fixture installed on cask

TRAY NO. 2 (Continued)

48. Cask raised for transfer
49. Cask raised for transfer
50. Initially loaded No. 2 cask in shipping skid
51. Upper side impact limiter installed on cask
52. Lower site impact limiter installed on cask
53. Top impact limiter installed on cask
54. Bottom impact limiter installed on cask
55. No. 2 cask secured on skid for transfer
56. Cask moved out of pool room
57. Cask moved out of pool room
58. Loaded cask lowered into skid pivots
59. Loaded cask lowered into skid pivots
60. Cask lowered into shipping skid
61. Cask lowered into shipping skid
62. Cask lowered into shipping skid
63. Installing upper side impact limiter on cask
64. Installing lower side impact limiter on cask
65. Installing lower side impact limiter on cask
66. Installing lower side impact limiter on cask
67. Installing top impact limiter on cask
68. Installing upper side impact limiter on cask
69. All impact limiters installed on cask
70. Cask secured in shipping skid
71. Cask and skid placed on truck bed
72. Cask secured in shipping skid
73. " " " " "
74. " " " " "
75. " " " " "
76. " " " " "
77. Empty shipping skid on truck bed
78. Cask and skid secured to truck bed
79. Cask and skid secured to truck bed
80. Installing upper side impact limiter on cask
81. First core shipment leaving FARB
82. First core shipment passing by plant
83. First core shipment leaves plant site
84. Trucker prepares to hook up to trailer with cask
85. Trucker prepares to leave FARB with cask
86. Trucker prepares to leave FARB with cask
87. " " " " " " "
88. Trucker rolls with core shipment
89. " " " " "
90. " " " " "
91. " " " " "
92. " " " " "
93. " " " " "
94. Trucker leaves plant site with core load
95. Trucker leaves plant site with core load
96. Both shipping casks in pool room

TRAY NO. 2 (Continued)

97. Subassembly L071 deposited in dry tunnel pot shielded in access tube floor
98. Cleaning machine drive before dismantling
99. Cleaning machine drive being removed
100. Cleaning machine drive removed
101. Cleaning machine drive removed
102. Cleaning machine drive shaft removed
103. Cleaning machine cable drum inspected
104. Cleaning machine cable drum inspected
105. Cleaning chamber manhole
106. Cleaning machine subassembly gripper removed
107. Lowering cleaning machine gripper through room manhole
108. Cleaning machine gripper placed in access tube "saddle"
109. Cleaning machine gripper placed in access tube "saddle"
110. Upper half of cleaning machine - assembled
111. Pot gripper suspended
112. Machinery dome air lock shipped out
113. Sweep removal container removed from storage
114. Lower third of maintenance container removed
115. Lower third of maintenance container removed
116. " " " " " "
117. " " " " " "
118. " " " " " "
119. " " " " " "
120. FARB repair pot emptied of equipment
121. No. 1 throttle valve linkage removed
122. Throttle valve shield plugs discarded
123. AFSF floor valve being removed
124. AFSF fuel port plugs removed
125. FARB sodium cold trap room equipment
126. FARB sodium cold trap room equipment
127. Fuel pot sodium deposits being scrapped in maintenance pit under FTF
128. " " " " " " " " " " "
129. Pot TPG-5 in lathe for ORB stool
130. Pot TPG-5 in lathe for ORB stool
131. Drawing transfer tank sodium sample
132. Manual gripper for raising subassembly in transfer tank
133. Manual gripper suspended over transfer tank
134. Manual gripper installed in transfer tank to pick up ORB No. 902
135. Alcohol cleaning reactor FTF gripper
136. Alcohol cleaning reactor FTF gripper
137. Alcohol cleaning FARB FTF gripper
138. East sodium gallery area
139. 12-inch secondary sodium north outlet in east sodium gallery
140. Loop 1-12" secondary sodium north inlet - east sodium gallery

DECOMMISSIONING SLIDE INDEX

TRAY NO. 3

1. Loop 1-12" secondary sodium north inlet insul. removed in east sodium gallery
2. Loop 1-12" secondary sodium north outlet insulation removed
3. Loop 1-12" secondary sodium south outlet insulation removed
4. Cutting 4" hole in Loop 1-12" secondary sodium north inlet
5. Cutting 4" hole in Loop 3-18" secondary sodium outlet header in west sodium gallery
6. Drilling loop 1-12" secondary sodium north inlet for equalizing pressure
7. Pressure equalizing line between loop 1-12" north inlet and outlet
8. Loop 1-12" north outlet cut and taped
9. Pipe ballon for 12" secondary sodium lines
10. Pipe balloon installed in Loop 1-18" outlet header in east sodium gallery
11. Inflating pipe balloon in Loop 1-18" outlet
12. " " " " " " "
13. " " " " " " "
14. " " " " " " "
15. " " " " " " "
16. Loop 1-18" secondary sodium outlet header balloon installed
17. Loop 1-12" north inlet being cut with guillotine
18. " " " " " " " "
19. " " " " " " " "
20. " " " " " " " "
21. " " " " " " " "
22. Guillotine saw on Loop 1-12" south outlet
23. Lathe-type cutter on Loop 1-12" south inlet
24. Loop 1-12" north inlet segment removed
25. Loop 1-12" north inlet segment removed
26. Loop 1-12" north inlet segment removed, showing pipe balloon
27. Loop 1-12" north inlet capped and dye checked
28. Loop 1-12" north inlet capped and dye checked
29. Loop 2-12" north inlet capped
30. Loop 2-12" north outlet capped
31. Primary cold trap room freon compressor/condenser removed for salvage
32. NaK room entrance before enclosing
33. NaK room entrance before enclosing
34. NaK room entrance during equipment dismantling
35. Cold trap room evaporator dismantled
36. Cold trap room blower housing dismantled
37. Cold trap room blower housing dismantled
38. NaK room entrance during equipment dismantling
39. Cold trap room blower housing being removed
40. NaK room after removal of C.T.R. equipment
41. NaK room entrance ready for concrete floor
42. Dismantling sodium unloading station
43. Dismantling sodium unloading station
44. " " " "

TRAY NO. 3 (Continued)

45. Roof steel and loading ramp for sodium barreling
46. Roof sheeting for sodium storage "greenhouse"
47. Sodium barreling greenhouse roof and floor completed
48. Sodium barreling greenhouse frontal sheeting
49. Sodium barreling greenhouse frontal sheeting completed
50. Sodium barreling greenhouse roof completed
51. Sodium barreling empty drum storage
52. Sodium barreling empty drum storage
53. Secondary sodium V 800-2 removed
54. Installing sodium barreling station fill valve
55. Handling empty sodium drums
56. Handling empty sodium drums
57. Pressure testing empty sodium drum
58. Pressure tested drums: failed
59. Sodium unloading station ready for sodium filling
60. Sodium unloading station ready for sodium filling
61. Truck trailer at barrel loading ramp
62. Handling sodium-filled drums at fill pit
63. Drum top with fill probes installed
64. Sodium fill station with drum in fill pot
65. Fill station corridor to "greenhouse" storage
66. Secondary sodium fill operations
67. Secondary sodium filled drum storage
68. Handling sodium-filled drum at fill pit
69. "Greenhouse" storage of secondary sodium
70. Secondary sodium fill operations
71. Secondary sodium fill operations
72. Secondary sodium fill operations: sodium smoking
73. Handling sodium-filled drum at fill pit
74. Handling sodium-filled drum at fill pit
75. Removing high level fill probe from drum
76. High level fill probe in position; sodium smoking
77. "Greenhouse" storage of secondary sodium
78. " " " " "
79. " " " " "
80. Truck trailer at barrel loading ramp
81. Moving sodium-filled drums to trailer
82. " " " " " "
83. " " " " " "
84. " " " " " "
85. " " " " " "
86. " " " " " "
87. " " " " " "
88. Truck trailer half filled with sodium drums
89. Secondary sodium fill operations at fill pit
90. Secondary sodium cold trap (CT) cooling ductwork removed to install drain line

TRAY NO. 3 (Continued)

91. Secondary sodium cold trap drain installed
92. Primary cold trap room north wall dismantling
93. " " " " " " "
94. " " " " " " "
95. " " " " " " "
96. " " " " " " "
97. " " " " " " "
98. Primary cold trap room north wall steel liner cut away
99. Primary cold trap room contents exposed
100. Primary cold trap room contents exposed
101. Primary cold trap room north wall temporarily enclosed
102. Primary CTR leak detection floor dismantling
103. Primary CTR leak detection floor dismantling
104. Primary CTR leak detection floor dismantled
105. Primary cold trap lead shielding arrives
106. Primary cold trap lead shielding being installed
107. Primary cold trap lead shielding installed
108. Primary cold trap lead shielding installed
109. Primary cold trap NaK lines stripped of insulation
110. Primary hot trap stripped of insulation
111. Primary cold trap NaK lines cut
112. Res. sodium fill system line cut in cold trap room
113. Primary cold trap NaK lines cut
114. Primary cold trap room equipment insulation removed
115. Primary hot trap and economizer stripped
116. Primary cold trap sodium lines cut
117. Primary cold trap sodium lines cut
118. Primary cold trap free standing and shielded
119. Primary cold trap free standing and shielded
120. Primary cold trap room equipment dismantling
121. Primary hot trap and economizer stripped of heaters
122. Primary plugging indicator, EM pump and flowmeter stripped
123. 100 GPM EM pump stripped
124. Primary sodium sample station
125. Primary sodium sample station vault
126. Primary sodium sample coil exposed
127. Primary hot trap flowmeter piping cut
128. " " " " " "
129. " " " " " "
130. " " " " " "
131. Primary hot trap and economizer removed
132. Primary P.I., EM pump and flowmeter stripped
133. Primary cold trap room equipment dismantling
134. Primary hot trap and economizer stored in FARB cold trap room
135. Primary cold trap lifting tackle installed

TRAY NO. 3 (Continued)

136. Primary cold trap greased rails installed
137. Primary cold trap greased rails installed
138. Primary cold trap moved across rails
139. Primary cold trap moved across rails
140. Primary cold trap moved to north access door

TRAY NO. 4

1. Primary cold trap moved to north door
2. Primary cold trap moved to north door
3. Primary cold trap lifted with mobile crane
4. " " " " " " "
5. " " " " " " "
6. " " " " " " "
7. " " " " " " "
8. " " " " " " "
9. Primary cold trap room after cold trap removal
10. Primary cold trap placed on flatbed trailer
11. Primary cold trap placed on flatbed trailer
12. Primary cold trap shielded in concrete-filled conduit
13. Primary cold trap room north wall restored
14. Primary cold trap room equipment dismantling
15. Primary cold trap room dismantled piping removed
16. Na-2 fuel pins loaded into birdcage
17. Canister for fuel segments without anchor pin
18. Canister for fuel segments without anchor pin
19. BMI canister for fuel segments
20. Loop 3 IHX support skirt being cut up for access to shell bottom
21. Loop 3 IHX support skirt cut open for access to shell bottom
22. Loop 3 IHX support skirt cut open for access
23. Loop 2 IHX support skirt cut open for access
24. Loop 2 IHX shell bottom drilled and rewelded
25. Loop 3 IHX sheel bottom drilled and rewelded
26. IHX air drill and bits for bundle penetration
27. Reactor building vent pins removed from vent building
28. Typical flexible hose sodium syphon line
29. Loop 2-30" pipe syphon line header
30. Loop 2-30" pipe containment cut for syphon
31. Loop 2-30" pipe syphon line header
32. Syphon header around Loop 1 - IHX to loops 1 and 2 30" pipe syphons
33. Loop 3 syphon header connecting to recirculating line near overflow tank
34. " " " " " " " " " " "
35. Loop 1-30" pipe syphon being connected to syphon header
36. Loop 1-30" pipe syphon being connected to syphon header
37. Loop 1 and 2-30" pipe syphon header connected to recirculating line, by-passing No. 1 IHX
38. Loop 1 and 2-30" pipe syphon header connected to recirculating line, by-passing No. 1 IHX
39. Loops 1 and 2-30" pipe syphon header, bare
40. Loop 3 syphon header connecting to recirculating line
41. Loop 3-30" line syphon pipe insulated
42. Loop 2-30" line syphon pipe, bare

TRAY NO. 4 (Continued)

43. Loop 1-30" line syphon pipe and header connection
44. Loop 2-30" line syphon pipe containment penetration
45. " " " " " " "
46. Loop 1-30" line syphon pipe and header connection
47. Loop 2-30" line syphon pipe, bare
48. Loop 2-30" line new syphon pipe inserted beside original syphon
49. Loop 1-30" line new syphon pipe inserted beside original syphon
50. Loop 2-30" line new syphon pipe connected to original header
51. Loop 1-30" line new syphon pipe connected to original header
52. Loop 1-30" line new syphon pipe connected to original header, close up
53. Fission products detector precipitator removed
54. Removing the machinery domes
55. " " " "
56. " " " "
57. " " " "
58. " " " " One of these
59. " " " "
60. " " " "
61. " " " "
62. " " " "
63. " " " "
64. " " " "
65. " " " "
66. Reactor machinery exposed after dome removal
67. Sweep mechanism shielding removed
68. Raising sweep mechanism
69. Raising sweep mechanism
70. Sweep mechanism blocked 12 feet up
71. Sweep column capped and seal welded
72. No. 1 primary sodium pump (PSP) motor removed
73. Removing No. 2 PSP motor
74. " " " " "
75. " " " " "
76. " " " " "
77. No. 2 PSP motor barrel exposed
78. No. 2 PSP shift seal cap welded
79. No. 3 PSP motor before dismantling
80. No. 2 PSP shaft cap welded
81. PRDC-APDA files stored in cafeteria
82. Exit port adapter for draining AFSF pots
83. Removing AFSF pots
84. " " "
85. " " "
86. " " "
87. Inverting AFSF pots at exit port
88. " " " " " "
89. " " " " " "
90. " " " " " "

TRAY NO. 4 (Continued)

91. Original AFSF pot adapter in use
92. Heating AFSF pot; new adapter in use
93. Modified AFSF pot support fixture
94. Modified AFSF pot adapter
95. Inverted AFSF pot at exit port
96. Spill reservoir around AFSF pot adapter
97. Heating inverted AFSF pot at exit port
98. " " " " " " "
99. " " " " " " "
100. " " " " " " "
101. " " " " " " "
102. Inverting AFSF pot at exit port
103. Inverting AFSF pot at exit port
104. Heating inverted AFSF pot at exit port
105. Emptied AFSF pots piled for decontamination
106. Decontaminating AFSF pot exterior
107. Clean shop alcohol fire
108. Clean shop alcohol fire
109. Removing sodium pots during shop fire
110. Clean shop after alcohol fire
111. Clean shop barriers after alcohol fire
112. Clean shop interior after alcohol fire
113. Clean shop interior after alcohol fire
114. " " " " " "
115. " " " " " "
116. Clean shop cyclone separator after fire
117. Clean shop interior after alcohol fire
118. Clean shop interior after alcohol fire
119. Clean shop ultrasonic cleaner after fire
120. " " " " " "
121. " " " " " "
122. " " " " " "
123. Clean shop interior after alcohol fire
124. Clean shop ultrasonic cleaner after fire
125. " " " " " "
126. " " " " " "
127. " " " " " "
128. Clean shop cleanup after alcohol fire
129. Clean shop power and heating controls after fire
130. Clean shop hard hat and clock after fire
131. Clean shop cleanup after fire
132. " " " " "
133. " " " " "
134. Clean shop painted after fire

TRAY NO. 4 (Continued)

135. Clean shop ultrasonic cleaning tank restored
136. No. 1 steam generator (SG) decommissioned
137. Capping No. 1 SG steam manifold nozzles
138. SG building roof opened for SG removal
139. No. 1 SG west sodium inlet cut and capped
140. No. 1 SG bottom sodium outlet being cut

TRAY NO. 5

1. Bottom of No. 2 Steam Generator
2. Bottom of No. 2 Steam Generator
3. Mobile crane being assembled for SG removal
4. No. 1 SG sodium outlet cut and capped
5. No. 1 SG east sodium inlet cut and capped
6. No. 1 SG before removal
7. Mobile crane being assembled for SG removal
8. " " " " " " "
9. " " " " " " "
10. No. 1 steam generator before removal
11. " " " " "
12. " " " " "
13. " " " " "
14. " " " " "
15. No. 1 steam generator being raised
16. " " " " "
17. " " " " "
18. " " " " "
19. " " " " "
20. " " " " "
21. No. 1 steam generator coming through building roof
22. " " " " " " "
23. " " " " " " "
24. No. 1 steam generator being lowered to ground
25. " " " " " " "
26. " " " " " " "
27. " " " " " " "
28. " " " " " " "
29. " " " " " " "
30. " " " " " " "
31. No. 1 steam generator being installed in shipping fixture
32. " " " " " " " "
33. " " " " " " " "
34. " " " " " " " "
35. " " " " " " " "
36. " " " " " " " "
37. " " " " " " " "
38. " " " " " " " "
39. " " " " " " " "
40. " " " " " " " "
41. " " " " " " " "
42. " " " " " " " "
43. No. 1 steam generator being loaded for shipment
44. " " " " " " "
45. " " " " " " "
46. " " " " " " "
47. " " " " " " "
48. " " " " " " "
49. " " " " " " "

TRAY NO. 5 (Continued)

50. Secondary Loop 1 with Steam Generator removed
51. " " " " " "
52. FARB waste gas (WG) stack before dismantling
53. FARB waste gas stack breeching cut
54. FARB waste gas stack rigged for dismantling
55. " " " " " " "
56. " " " " " " "
57. " " " " " " "
58. FARB waste gas stack guy wires uncoupled
59. FARB waste gas stack rigging inspection
60. Mobile crane for WG stack removal
61. FARB WG stack guy wires uncoupled
62. FARB WG stack raised
63. " " " "
64. " " " "
65. " " " "
66. FARB WG stack moved south for laydown
67. " " " " " " "
68. FARB WG stack moved south for laydown
69. FARB WG stack being laid to ground
70. " " " " " " "
71. " " " " " " "
72. " " " " " " "
73. FARB WG stack laid on ground
74. " " " " " "
75. " " " " " "
76. " " " " " "
77. " " " " " "
78. " " " " " "
79. " " " " " "
80. FARB with WG stack removed
81. FARB WG stack foundation pad
82. FARB WG stack being segmented and shipped
83. Reactor vessel syphon drain line installed
84. " " " " " "
85. Reactor syphon line flowmeter
86. Reactor syphon line insulated
87. Reactor syphon line insulated
88. Reactor syphon line insulated
89. Reactor syphon line insulated
90. Primary sodium fill station
91. Primary sodium fill station
92. Primary sodium barrel storage in "Greenhouse"
93. Primary sodium barrel handling outside "Greenhouse"
94. Primary sodium barrel storage in trestleway
95. Primary sodium barrel storage in trestleway
96. Primary sodium barrel handling in trestleway
97. Primary sodium empty barrel storage outdoors
98. Primary sodium barrel storage in trestleway

TRAY NO. 5 (Continued)

99. Primary sodium barrel storage in reactor building
100. " " " " " " "
101. " " " " " " "
102. Primary sodium storage in trestleway
103. Primary sodium storage in trestleway
104. FTF transport car in trestleway
105. Sodium melting facility tank
106. Sodium melting facility and instrumentation
107. Sodium melting facility and instrumentation
108. Sodium melting facility, lower half
109. Sodium melting facility, upper half
110. Unloading pot AP-1 from sodium melting facility
111. " " " " " " "
112. " " " " " " "
113. " " " " " " "
114. Decontamination facility erection
115. " " "
116. " " "
117. Decontamination facility installed and grouted
118. Decontamination facility services installation
119. Decontamination facility services installation
120. " " " "
121. " " " "
122. Decontamination facility mixing tank
123. Decontamination facility exhaust duct installed
124. Decontamination facility spray nozzles installed
125. Decontamination facility completed
126. Decontamination facility completed
127. FARB control room
128. FARB control room
129. FARB mechanical equipment room
130. FARB waste storage tank room
131. Conduits, inserts and canisters for irradiated element shipment
132. Shipping conduits with lead-shielded inserts for shipping neutron sources
133. Shipping conduits for shipping neutron sources
134. Shipping canisters for irradiated elements
135. Shipping canisters being filled with concrete
136. Shipping canisters being filled with concrete
137. Shipping canisters being filled with concrete
138. Shipping canisters being filled with concrete
139. Shipping canisters ready for loading in pool room
140. Loading neutron source in shipping conduit

DECOMMISSIONING SLIDE INDEX

TRAY NO. 6

1. Shipping containers: neutron source being loaded
2. Shipping containers: neutron source being loaded
3. Neutron source installed in shipping container
4. Shipping containers ready for loading in pool room
5. Shipping containers ready for loading in pool room
6. Measuring radioactivity of loaded neutron source
7. Pouring concrete over loaded neutron source
8. " " " " " "
9. " " " " " "
10. Loading irradiated dummy subassembly in shipping container
11. " " " " " " "
12. " " " " " " "
13. Transfer rotor container (TRC) syphon drain line
14. TRC syphon drain line at exit port
15. TRC syphon drain line at exit port
16. TRC syphon drain line at exit port
17. Reactor vessel syphon pipe raised
18. Reactor vessel syphon pipe installed
19. Reactor support plate after final sodium drain
20. Reactor support plate after final sodium drain
21. Viewing down exit port for sodium level
22. Reactor plenum syphon line in Loop 2-6" pipe
23. Reactor plenum syphon line in Loop 2-6" pipe, insulated
24. Reactor plenum syphon line routed across operating floor
25. FARB overflow tank syphon line installed
26. FARB overflow tank syphon line installed
27. FARB overflow tank syphon line insulated
28. " " " " " "
29. " " " " " "
30. " " " " " "
31. " " " " " "
32. " " " " " "
33. FARB overflow/transfer tank syphon insulated
34. New transfer tank syphon line through rupture disc
35. " " " " " " " "
36. " " " " " " " "
37. FARB sodium syphon line through NaK Room
38. Primary cold trap NaK sump drain
39. Clean argon system NaK bubbler drain
40. Neutron detector rack leads cut
41. Neutron detector rack leads cut (No. 4)
42. Neutron detector rack removed from tube (No. 3)
43. Neutron detector dismantled

TRAY NO. 6 (Continued)

44. Neutron detector dismantled
45. Neutron detector racks dismantled
46. Neutron detector racks dismantled
47. Blanket subassembly's stacked on cutup pool floor
48. Blanket subassembly's stacked on cutup pool floor
49. Primary sodium pump liquid rheostat retired
50. Machinery dome observation ports capped
51. PST electrical and service line penetrations capped
52. PST electrical cables cut
53. PST electrical penetrations capped
54. PST 4" exhaust penetration capped
55. Reactor exit port stripped and capped
56. Reactor exit port adapters removed
57. Reactor exit port capped and isolated
58. Reactor exit port covered
59. PST below floor instrument penetrations capped
60. Emergency argon line to exit port below floor; line and shielding cut away at PST
61. Emergency argon line to exit port below floor; line and shielding cut away at PST
62. Emergency argon line to exit port below floor; line and shielding cut away at PST; emergency line capped
63. Neutron counter tubes below floor cut
64. Neutron counter tubes below floor cut and capped
65. Neutron counter tubes below floor cut and capped
66. Neutron counter tubes below floor cut and capped
67. Neutron counter tubes below floor cut and capped
68. Machinery dome seal welded
69. Machinery dome seal welded
70. Machinery dome seal welded
71. No. 2-30" U-bend sodium syphon line installation
72. No. 2-30" U-bend sodium syphon line installation
73. Controlled area for cut sodium pipe inspection
74. Controlled area for cut sodium pipe inspection
75. Cutting and containing contaminated scrap
76. Cutting and containing contaminated scrap
77. Sodium-filled pipe cut and taped for boxing
78. Contaminated scrap ready for shipment
79. No. 1 secondary sodium pump (SSP) motor removed
80. No. 1 secondary sodium pump sodium discharge stripped
81. No. 1 SSP sodium discharge line cut
82. No. 1 SSP with motor removed
83. No. 1 SSP rigged for lifting
84. No. 1 SSP being raised
85. " " " "
86. " " " "
87. " " " "
88. " " " "
89. " " " "

TRAY NO. 6 (Continued)

90. No. 1 SSP boxed for shipment
91. No. 1 SSP sodium inlet and outlet after pump removal
92. Loop 1 - PSP 16" discharge drain to overflow tank
93. No. 1 - IHX drain line cut and capped
94. No. 2 IHX drain line drain to overflow tank
95. No. 2 IHX drain line drain to overflow tank
96. No. 2 PSP 16" discharge drain to overflow tank
97. No. 3 IHX drain line drain to overflow tank
98. V521 bypass for IHX drain line drain to overflow tank
99. Cut and capped 2" drainline downstream of overflow tank more pressure transmitters
100. Overflow tank drain line added
101. Overflow tank drain line and water monitor in tank sump pit
102. Overflow tank 6" overflow line cut and capped
103. Overflow tank gas equalizing line cut and capped
104. Overflow tank discharge to recirculating line cut and capped
105. FARB cold trap room emptied
106. FARB cold trap removed
107. FARB cold trap economizer removed
108. FARB cold trap boxed for shipment
109. FARB cold trap boxed for shipment
110. FARB cold trap boxed for shipment
111. Emergency diesel generator before dismantling
112. Emergency diesel being removed
113. Emergency diesel being removed
114. Emergency diesel generator being removed
115. Emergency diesel generator being removed
116. Overflow tank discharge cut and capped
117. Cut and capped 2" drainline downstream of overflow tank more pressure transmitters
118. Loop 3 - IHX drain line cut and capped
119. Loop 1 - IHX drain line cut and capped
120. Loop 2 - IHX drain line cut and capped
121. Loop 2 - IHX drain line cut and capped
122. Loop 2 - PSP 16" discharge drain cut and capped
123. Loop 2 - 30" U-bend syphon nozzle capped
124. PST electrical penetrations capped
125. Fission products sample line from reactor capped
126. Fission products sample line from reactor capped
127. Recirculating line cut and capped in LRB
128. Discharge line cut and capped in LRB
129. Syphon line cut and capped in LRB
130. Loop 2 - 6" instrument cluster standpipe capped
131. Loop 2 - 6" throttle valve capped
132. Loop 2 - 6" throttle valve capped
133. PRDC control battery being dismantled

TRAY NO. 6 (Continued)

134. PRDC control battery being dismantled
135. PRDC control batteries removed
136. Essential bus 3-unit MG set removed
137. Decay pool drained ten feet down
138. East boundary fence at vent building
139. East boundary fence at fission products detector building
140. Decontaminating cutup machine with high pressure water blasting

DECOMMISSIONING SLIDE INDEX

TRAY NO. 7

1. Decontaminating cutup machine using high pressure water blasting
2. Decontaminating cutup machine using high pressure water blasting
3. Decontaminating cutup machine using high pressure water blasting
4. Decontaminating cutup machine using high pressure water blasting
5. Decontaminating cutup machine using high pressure water blasting
6. Decontaminating cutup machine using high pressure water blasting
7. High pressure water blasting equipment
8. High pressure water blasting equipment
9. Decay pool half drained
10. Cutting up the decay pool racks
11. Cutting up the decay pool racks
12. Loaded canisters of contaminated scrap loaded at decontamination facility
13. FTF gripper dismantled in FARB
14. FTF gripper diamantled in FARB
15. Machinery dome instrument penetrations seal welded
16. Machinery dome access door seal welded
17. Machinery dome sealing complete
18. Machinery dome sealing complete
19. Primary sodium drums stored in reactor building
20. Primary sodium drums stored in reactor building
21. Reactor building personnel lock secured
22. PRDC heating transformers removed
23. PRDC buses dismantled
24. Primary sodium storage tank No. 1
25. Primary sodium storage room N_2-H_2O
26. Sodium building secured
27. Reactor building secured
28. Sodium "greenhouse" secured with locked gate and fence
29. West perimeter fence complete
30. West perimeter fence complete at Health Physics Building
31. Handling empty cask and fuel basket
32. Handling empty cask and fuel basket
33. Handling empty cask and fuel basket
34. Handling empty cask and fuel basket
35. Handling empty cask and fuel basket
36. Empty cask with basket container
37. Empty basket containers in FARB
38. Empty cask on truck
39. H.P. survey of empty cask
40. Removing empty cask from truck

DECOMMISSIONING SLIDE INDEX

TRAY NO. 7 (Continued)

41. Removing empty cask from truck
42. " " " "
43. " " " "
44. Cask welding shield installed
45. Subassembly transfer -- rack to basket
46. " " " "
47. " " " "
48. " " " "
49. " " " "
50. " " " "
51. " " " "
52. " " " "
53. " " " "
54. " " " "
55. " " " "
56. Tool handling for subassembly transfer
57. Fuel basket with spacer tubes visible
58. Subassembly transfer to fuel basket
59. " " "
60. " " "
61. Spacer tube transfer to fuel basket
62. " " "
63. " " "
64. Loaded fuel basket transfer -- pool to cask
65. " " " "
66. " " " "
67. " " " "
68. " " " "
69. " " " "
70. " " " "
71. " " " "
72. Cask lid seal installed
73. Basket lid installed
74. Cask lid installed
75. Cask lid installed
76. Cask top surveyed
77. Cask lid welded
78. Cask lid welded
79. Fuel basket in container
80. Cask weld shield positioned
81. Cask lid welded
82. Cask lid welded
83. Cask lid welded
84. Pool floor equipment
85. " "
86. " "
87. " "
88. Subassemblies transferred -- floor to racks
89. Subassemblies transferred -- floor to racks
90. Subassemblies transfer hook

DECOMMISSIONING SLIDE INDEX

TRAY NO. 7 (Continued)

91. Subassemblies transferred -- floor to racks
92. " " "
93. Subassembly transferred -- floor to racks
94. Raising fuel basket from container
95. Raising fuel basket from container
96. Raising fuel basket from container
97. Installing basket in pool
98. " " "
99. " " "
100. " " "
101. " " "
102. " " "
103. " " "
104. " " "
105. Securing loaded cask for lifting
106. " " " "
107. " " " "
108. Raising loaded cask
109. " " "
110. " " "
111. " " "
112. Installing loaded cask on trailer
113. " " " "
114. Torquing cask bolts
115. Installing cask lid center plug
116. Installing cask end impact limiters
117. Installing cask end impact limiters
118. Cask loaded on trailer for shipment
119. Cask loaded on trailer for shipment
120. Raising loaded fuel basket from pool
121. Surveying loaded fuel basket over pool
122. Raising loaded fuel basket over pool
123. Loaded cask leaves site in storm
124. Mass spec testing of cask top weld
125. " " " "
126. " " " "
127. " " " "
128. Subassembly segment spill on pool floor
129. " " " "
130. " " " "
131. Empty decay pool
132. Empty decay pool
133. FARB receiving room stripped of equipment
134. Pressurized water bag used for segment pickup
135. " " " "
136. " " " "
137. " " " "
138. " " " "
139. " " " "
140. " " " "

DECOMMISSIONING SLIDE INDEX

TRAY NO. 8

1. Na Drum Rebunging Operation in Trestleway
2. Drum Pallet Lifting Fixture
3. Painting drum tops after rebungin
4. New drum bungs and seals
5. Na drum rebunging operation in trestleway
6. Painted drum tops
7. Rebunging Na drums
8. Sodium drum pallets stacked in Reactor Bldg.
9. " " " "
10. " " " "
11. Contaminated scrap in shipping boxes
12. B3 nozzle cask on trailer
13. Nozzle cask shipping container (liner)
14. Fuel basket support structure over pool
15. Nozzle cask open
16. Irradiated nozzle container being raised from vault
17. " " " " "
18. " " " over shipping casks
19. " " " " "
20. Nozzle cask being covered
21. Nozzle cask being secured
22. Nozzle cask being covered
23. Vendor tractor loaded with nozzle casks
24. Cutup pool resin tank on floor
25. Decay pool drained
26. Decay pool walls painted
27. Decay pool bridge crane stabilizer removed
28. FTF gripper cask structural steel vacated
29. Dismantled resin tanks hang in cutup pool
30. " " " "
31. " " " "
32. Decay pool walls painted
33. " " " "
34. " " " "
35. " " " "
36. Open B2 shipping cask for resin tanks
37. " " " "
38. Contaminated scrap in shipping box
39. Resin tank over B2 shipping cask
40. Nozzle casks being secured
41. Cutup pool drained to one foot level
42. Cutup pool draining and pumping equipment
43. " " " "
44. Decay pool painted
45. " " "
46. " " "
47. Lower fuel vault vacated
48. Cutup pool resin tanks on floor
49. Maintenance shops foundations cleared