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ORNL-1868
C-84 – Reactors-Special Features of Aircraft Reactors

AEC RESEARCH AND DEVELOPMENT REPORT

DISASSEMBLY AND POSTOPERATIVE EXAMINATION OF THE AIRCRAFT REACTOR EXPERIMENT

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OAK RIDGE NATIONAL LABORATORY
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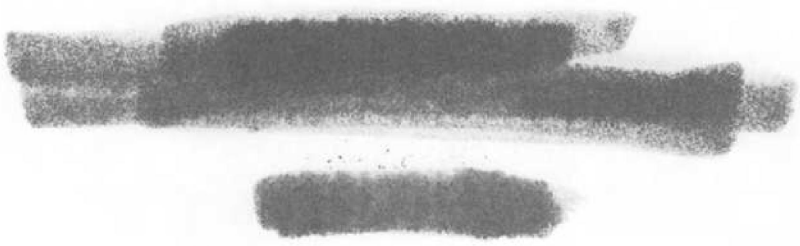

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DISASSEMBLY AND POSTOPERATIVE EXAMINATION OF THE AIRCRAFT REACTOR EXPERIMENT

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INTRODUCTION

The Aircraft Reactor Experiment (ARE) was successfully concluded in November of 1954, and a detailed report of the operation was published the following year.[1] At that time it was thought that an extensive examination of the reactor and system components after disassembly was warranted. It was realized, of course, that the level of radioactivity of the components would necessitate extensive delays in the examinations.

Since examination of a few critical ARE samples showed nothing unexpected, much of the planned hot-cell inspection was postponed and complete examination of all but a few specimens was indefinitely suspended. The few examinations that were completed are described in this report, along with a description of the disassembly of the ARE system. Diagrams of the fuel system, sodium system, and off-gas system are presented in Figs. 1, 2, and 3 for reference use in visualizing the disassembly process.

[1] W. B. Cottrell *et al.*, *Operation of the Aircraft Reactor Experiment*, ORNL-1845 (Aug. 22, 1955).

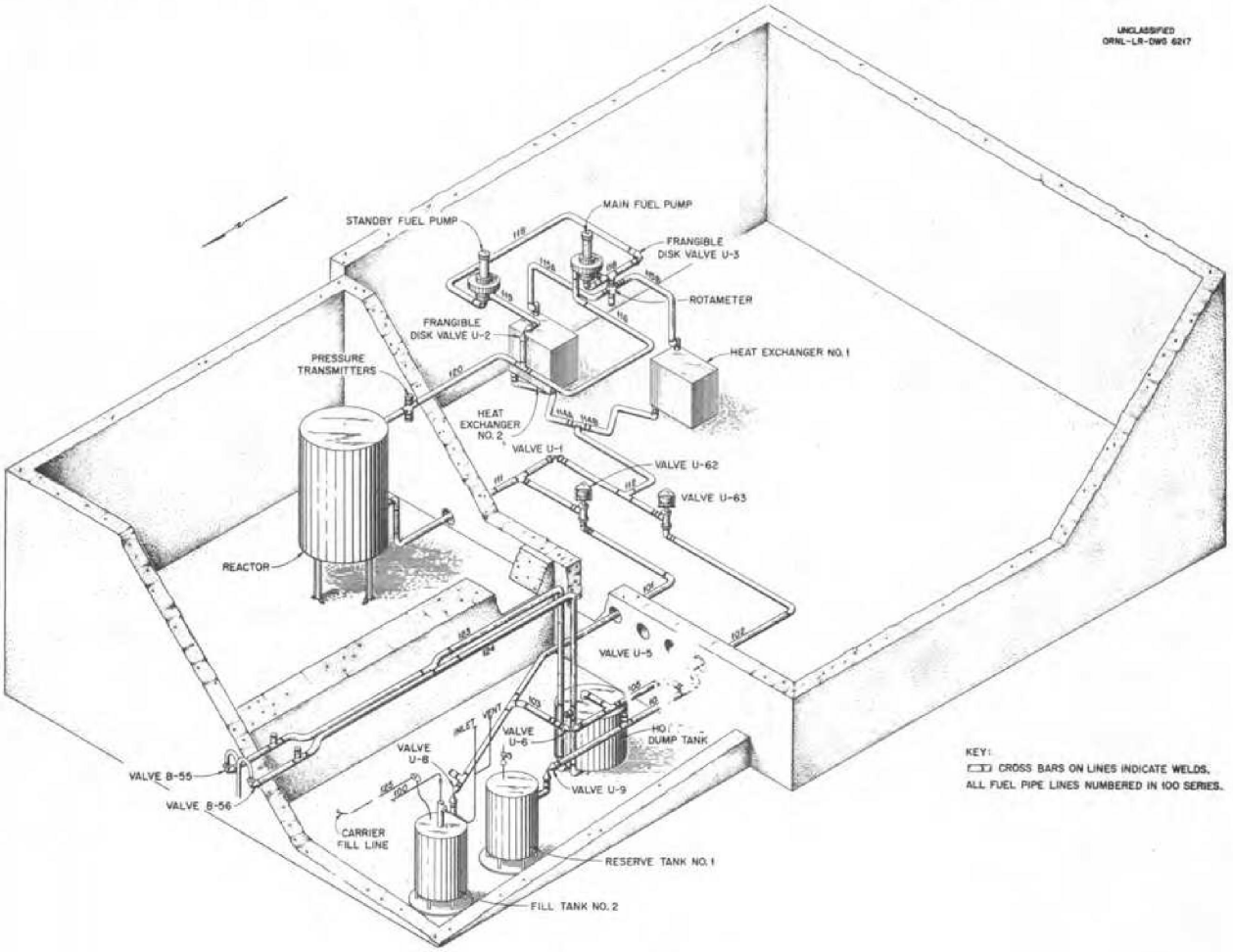

Fig. 1. Diagram of ARE Fuel System.

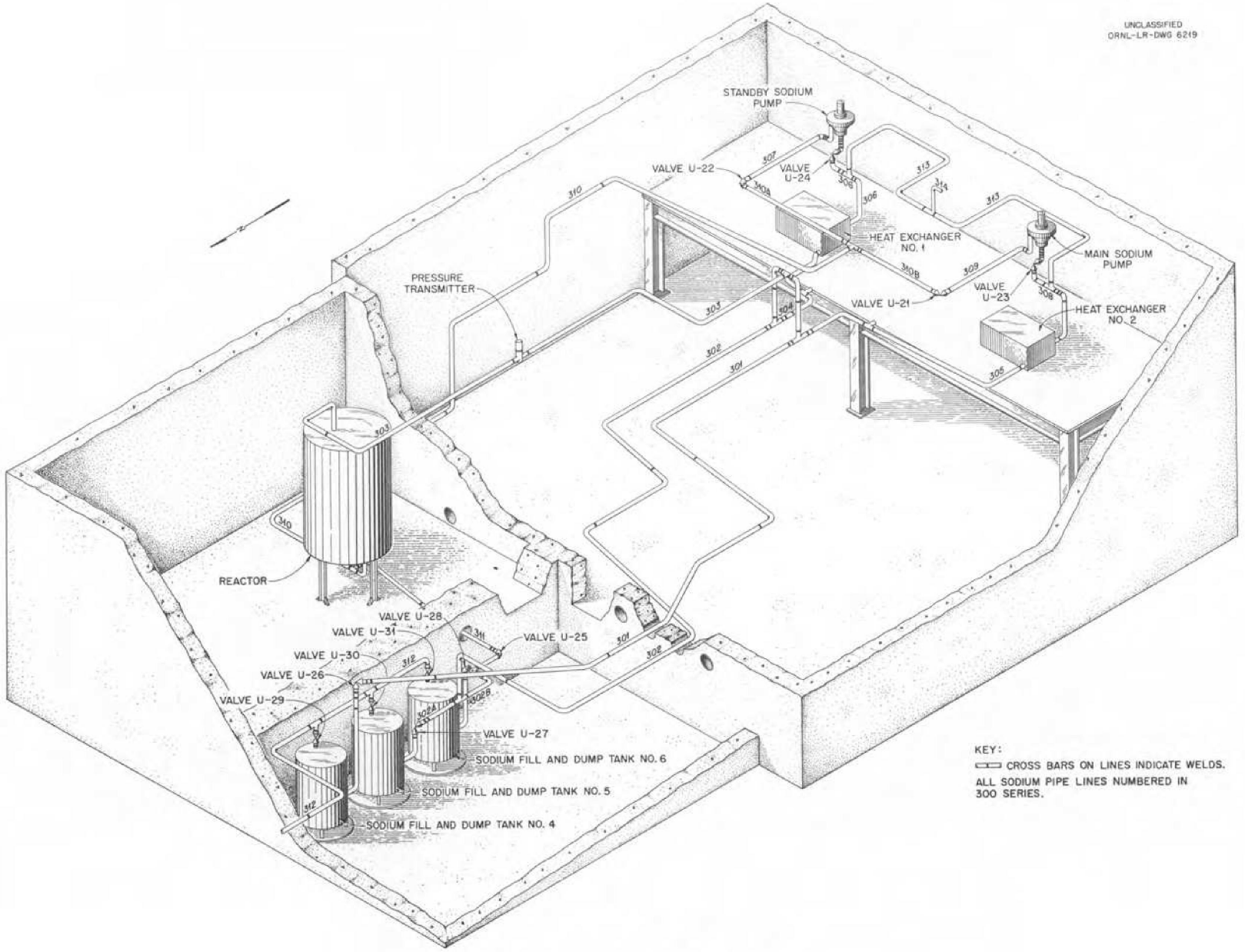

Fig. 2. Diagram of ARE Sodium System.

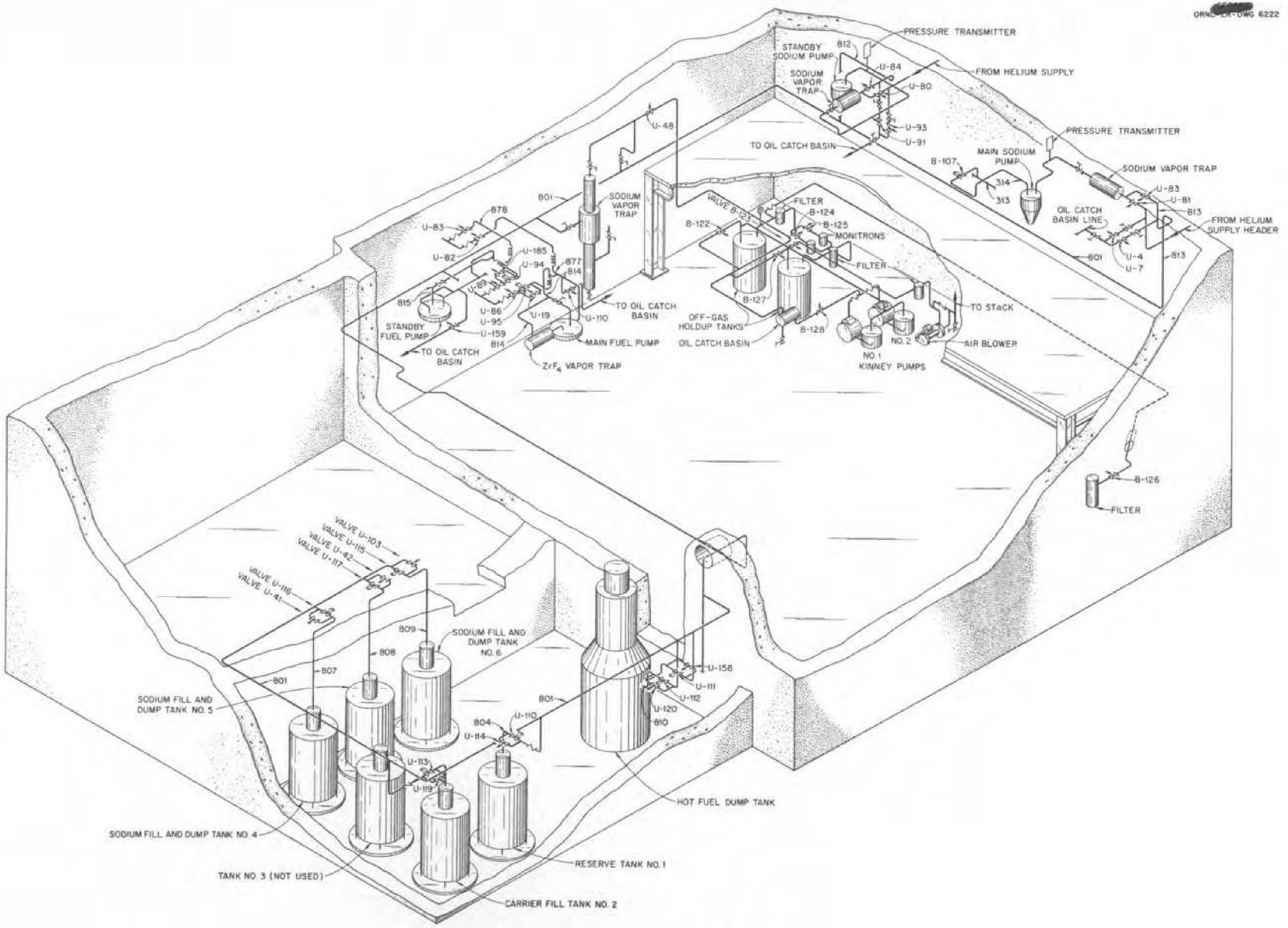

Fig. 3. Diagram of ARE Off-Gas System.

CHRONOLOGY FROM FINAL SCRAM TO DISASSEMBLY

The nuclear operation of the ARE was concluded at 8:04 PM, Friday, November 12, 1954, upon insertion of the scram rods. Circulation of the fuel and the sodium was continued, however, until the next day in order to allow the afterheat to decay before dumping the fuel into the fuel dump tank (and the sodium into the sodium dump tanks).

During the period from shutdown of the reactor until the fuel was dumped it was necessary for the operating personnel to wear gas masks for several hours because of the level of the airborne activity. As was previously noted,[1] a gas leak, which permitted gaseous fission-product activity to be released to the cell, had existed at least since the start of high-power operation. The exact location of this leak has never been determined, although it was known to be from the gas volume above the main fuel pump. While it was subsequently shown that the diaphragm of the main fuel pump pressure transmitter was ruptured, it is believed that this occurred during the early morning of November 13 and that prior to that time the activity had been leaking from one or more of the potential leaks in or around the main fuel pump (spark plugs, seals, fill lines, vent lines, etc.). In order to keep these gases from leaking from the cell into occupied areas of the building (the cell proved to be quite porous), it was necessary to maintain the cell at a subatmospheric pressure (–2 to –4 in. H_2O) by the use of a jet air pump which pumped around 100 cfm of air from the cell. This air, with its activity, was discharged some 1000 ft south of the ARE building,[1] but the negligible wind velocity that existed at times during the period after shutdown was not sufficient to cause adequate dispersion of this activity. (Prior to this period the wind had dispersed the activity.)

On the morning of Saturday, November 13, the fuel was transferred into the fuel dump tank. Since the fuel tubes in the reactor would not have drained completely if the fuel had merely been allowed to drain from the system by gravity, the fuel was forced out of the system by pressurizing carrier material into one end of the system and draining the other end of the system into the dump tank. The fuel in the dump tank was thus diluted by this flush material. The dilution ratio was approximately 1 to 1.

In order to determine whether the fuel in all six parallel reactor passages was removed, the flush carrier was heated to a temperature which was 100°F hotter than the fuel system so that the passage of the carrier through each of the six parallel reactor tubes could be observed by thermocouples on these tubes. Although the individual recorder charts showing temperature differentials across each of the six reactor tubes indicated that one tube (No. 4) did not clear, the multipoint temperature record of all tubes indicated that the carrier had passed through all tubes. The individual temperature recorder on tube No. 4 was subsequently found to be inoperative.

After the fuel was dumped, the increased level of the airborne activity caused all but a few of the operating personnel who were wearing gas masks to evacuate the building for about 1 hr. The gas used in pressuring the fuel system during the dumping operation was discharged to the stack, but the activity level in the building rose because the wind at the time was such that the activity descended and entered the ventilators on the top of the building.

Following the completion of the removal of the fuel on Saturday, November 13, the sodium was drained into the sodium drain tanks. In this instance gravity drainage was sufficient and was readily effected.

Evidence that all the fuel had been removed from the reactor was obtained the following Monday by a test in which the three control rods were withdrawn from the reactor and no increase in background count was observed. It was of especial interest that there was no measurable afterheat in the fuel; however, the expected amount of afterheat was only a small fraction of the electrical heat on the tank. Subsequent analysis of the fuel indicated the activity to be in reasonable agreement with expectations.

DISASSEMBLY OF THE REACTOR AND AUXILIARY SYSTEMS

The postoperative examination of the ARE system started on December 10, 1954, with the taking of a fuel sample from the dump tank while the fuel was still molten. Disassembly of the reactor and the auxiliary systems then proceeded as radiation levels permitted. The primary objective of this work was, of course, the obtaining of samples for metallurgical, chemical, and physical examination. It was also expected that in the dismantling of the system much equipment could be salvaged while the cells were being prepared for the modifications required for the forthcoming Aircraft Reactor Test.

Radiation surveys were used as a basis for planning the disassembly sequence and techniques. The radiation decay curves obtained from data taken daily at five monitoring points are shown in Fig. 4.

AUXILIARY SHIELDING AND SAFETY PRECAUTIONS

In order to separate the fuel circuit from the sodium circuit for the purpose of dismantling the sodium system, two flat lead shields, 6 ft high, 4 ft wide, and 2 in. thick, were suspended on beams which ran the width of the heat exchanger cell. For work in the higher radiation fields associated with the fuel system, a lead box was built that was 2 in. thick on four sides and the bottom. One end of the bottom had a 6-in.-wide slot, and one side had an 8-in.-square slot, 9 in. off the bottom, through which personnel worked. The building crane was used for moving the flat shields and the box.

In order to offset the fire hazard associated with the sodium system, all water lines were cut from the water manifold in the basement below the heat exchanger cell, and no sodium lines were cut with flame or arc torches. All sodium lines were cut with hack saws, and all cut lines were immediately sealed with several layers of masking tape. Fire-fighting equipment was available at all times.

DISMANTLING OF THE SODIUM SYSTEM

The work of dismantling the sodium system, shown in Fig. 2, was started on January 18, 1955, at which time the radiation level was down to 30 to 250 mr/hr with the lead shields hanging between the sodium system and the fuel system.

The main sodium pump was removed first, and it was found that the rotary element had a radiation reading of 12 mr/hr at contact with the impeller. After the rotary assembly had been cleaned in a bath of 50% kerosene and 50% methyl alcohol, the impeller was removed and submitted for examination.

The standby sodium pump was removed as a unit with 6-in. stubs left on all lines so that the pump could be salvaged for use in other experiments. At the time it was removed, this pump, which had not operated during nuclear operation of the reactor, had a radiation reading of 1 mr/hr at contact with the bowl and the top flange.

The sodium-to-helium and the helium-to-water heat exchangers were removed next. The water that had remained in the heat exchangers was first drained into containers and removed from the cell. The sodium lines were then cut with a hand hack saw and the helium blower duct was cut loose with a cutting torch. The exchangers were thus removable as complete units. After the insulation and electric heaters had been stripped from the exchangers, neither showed a radiation reading on the outside. The ends of the cut sodium lines read 2 mr/hr at contact.

After the removal of the sodium system pumps and heat exchangers, it was found that the radiation field had increased to about 600 mr/hr. The equipment had helped to shield the area from the fuel system radiation.

Sodium lines 304 through 309, 313, and 314 were then removed in as long lengths as possible and sealed at the cut ends from air and moisture. Valves in the lines were left as installed. Radiation levels on these lines were 2 to 10 mr/hr. During the removal of these sodium lines the flat lead shields were adequate to protect personnel from the radiation from the fuel system, as shown in Fig. 5. The remainder of the sodium piping ran adjacent to the fuel system, however, and, in places, over and under the fuel piping, and therefore the lead box described above had to be brought into use. Sodium lines 301 and 302 were removed by personnel working within the lead box. To make the line cutting job easier, the heavy gage stainless steel annulus can surrounding the sodium line was cut by using an electric arc before the sodium line was cut

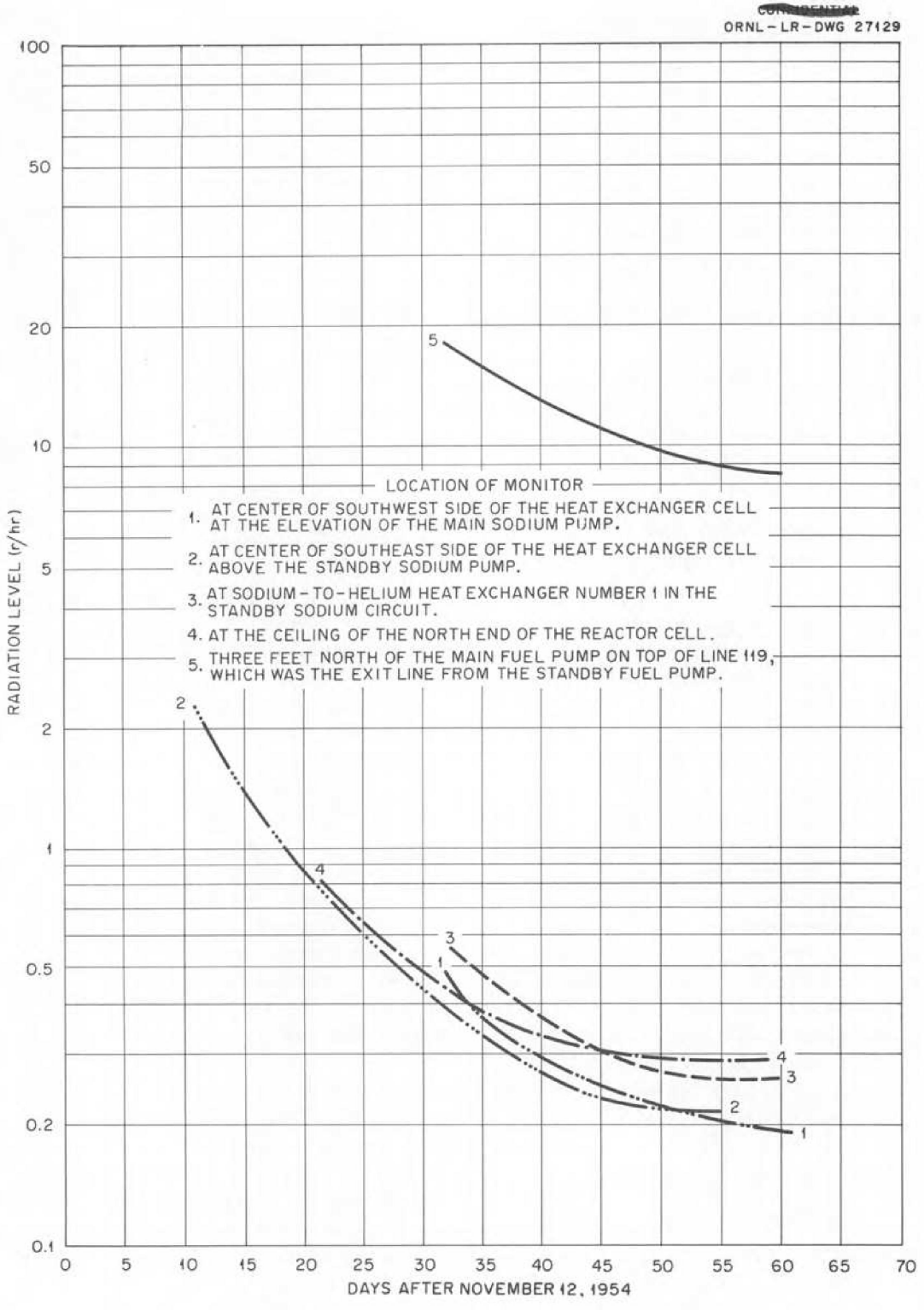

Fig. 4. Radiation Decay Curves for Five Locations in ARE System.

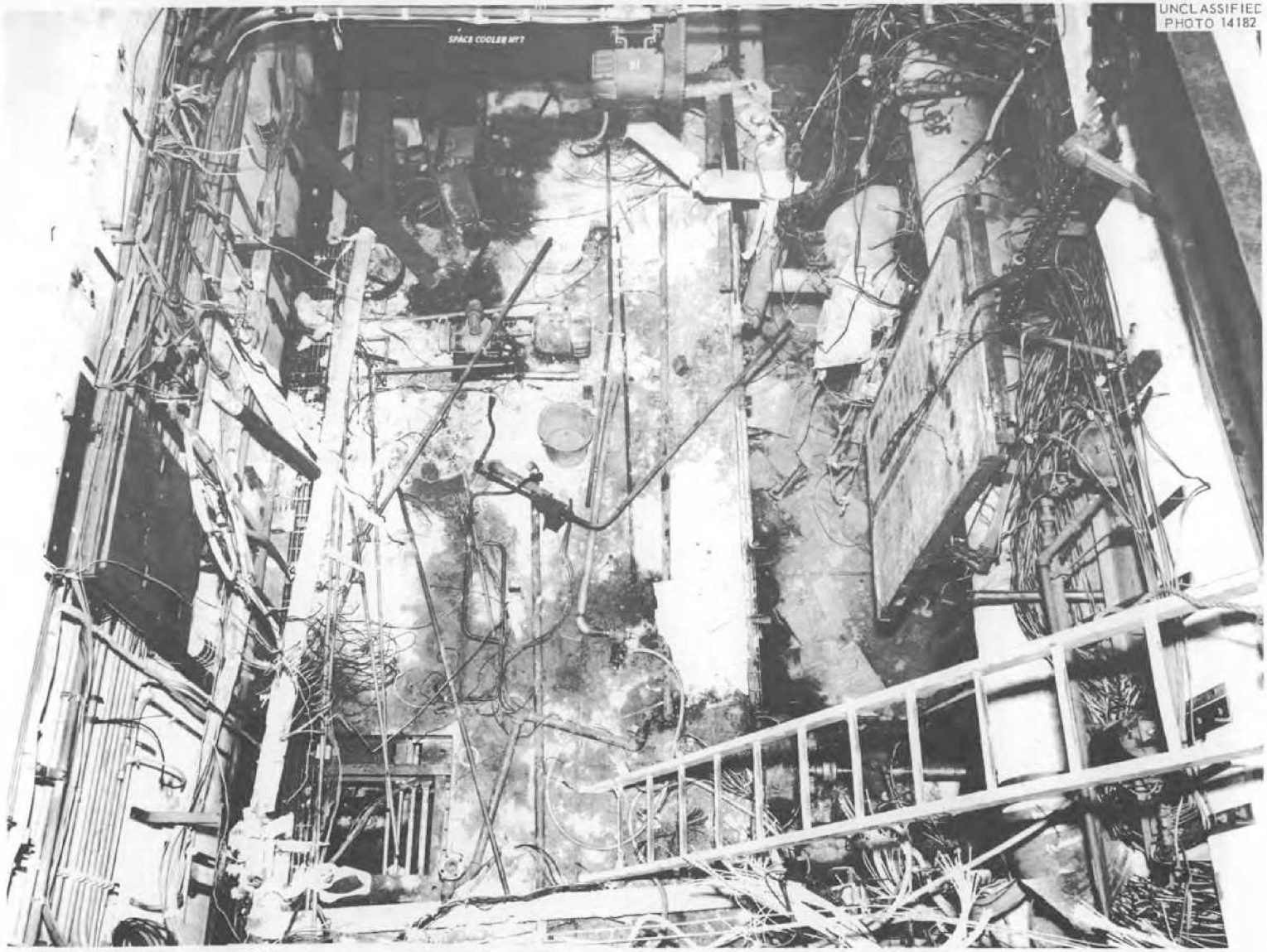

Fig. 5. Heat Exchanger Pit After Removal of Sodium System Pumps and Heat Exchangers and Lines. Note lead shield used to protect workers from fuel system radiation.

with a hack saw. The radiation level at the open ends of lines 301 and 302, which were cut where they entered the tank pit wall, was only 2 mr/hr, but the radiation field in that region outside the lead box was 700 to 1000 mr/hr.

Sodium lines 303 and 310, the lines to and from the reactor, were removed next. Because of the position of line 303, it was necessary to sever it where it entered the reactor cell wall by using an electric arc. The cutting operation caused a sodium fire, which was extinguished with Metal-X extinguishers. A small amount of air activity, 2 divisions on the 2K scale of the air monitor, was observed as a consequence of the sodium fire. Both line 303 and line 310, which was cut with a hack saw by a craftsman working from the lead box, showed radiation levels of 2 mr/hr at contact with the ends of the lines.

Auxiliary equipment was removed as necessary or convenient. Most of these items, such as pump drive motors and lubricating oil systems, were salvaged for further use. The radiation level of the main sodium-pump lubricating system was 2 mr/hr, but this was found to be surface contamination. The oil showed no radiation.

DISMANTLING OF THE FUEL SYSTEM

A complete radiation survey of the fuel system, made on February 14, 1955, showed 75 r/hr at contact with the insulation on the rotameter of the main fuel pump, 55 r/hr at contact with insulation on the dead leg on the bottom of the pump bowl, and 12 r/hr at contact with the top flange. These high radiation levels indicated the advisability of removing the high radiation sources first, where possible.

The components of the fuel system are illustrated in Fig. 1, and a photograph of the cell containing the fuel pumps is presented in Fig. 6. The disassembly work was started with the main fuel pump which was causing a high radiation field. Removal of the rotary assembly caused a small burst of air activity that cleared in 2 to 3 min. The radiation level of the rotary assembly was 20 r/hr at 5 ft. After removal of the rotary assembly, the lines to the pump were cut and sealed and the pump was lifted out of the cell, as shown in Fig. 7. The radiation level of the main fuel pump bowl was 900 mr/hr at 5 ft. The standby fuel pump was removed in a similar manner. This pump was not used during operation at power, and therefore the radiation level on the bottom of the pump bowl was only 16 mr/hr. This pump was salvaged for further use.

The fuel-to-helium and helium-to-water heat exchangers were removed next. The radiation levels of both exchangers were 10 r/hr at 6 in. In order to remove these exchangers intact, the water lines and the helium ducts were cut. The electric wiring which ran in large bundles across the heat exchangers was removed by hooking onto the bundles with the overhead crane and pulling the wires out of the cell. The fuel lines to the heat exchangers were cut by a craftsman working from the lead box with a hack saw. The insulation and the electric heaters were removed from the heat exchangers after they were removed from the cell, and heat exchanger No. 2 then had a radiation level of 10 r/hr at 12 in.

During removal of the pumps and heat exchangers, most of the fuel lines were cut at one end. The open ends of these fuel lines had radiation readings of 10 r/hr at 3 in.

Lines in the standby fuel circuit were free of fuel and showed only surface contamination of 2 mr/hr. The reactor supply and discharge lines were cut with hack saws by craftsmen working within the lead box. With the removal of the fuel lines, all equipment installed in the heat exchanger cell had been removed.

DISMANTLING OF THE REACTOR

On March 10, 1955, the concrete pit plugs were removed from the top of the reactor pit and the concrete blocks which lined the walls and bottom of the reactor cell were lifted out to provide space for the lead box. A portable grinder with a flexible shaft and an extension handle was mounted on the side of the lead box. The craftsman could operate the grinder from inside the box by using the extension handle.

The insulation was stripped from the reactor and the sodium and fuel lines that were in the reactor cell. The helium manifold and the control rod chamber (Fig. 8) were then removed. The helium manifold and the shim and regulating rod chambers had radiation readings of 200 mr/hr at contact.

The portable grinder was then used to cut the fuel lines as close as possible to the reactor can. These lines had radiation readings of 10 r/hr at 3 in. when removed. The grinding

Fig. 6. Heat Exchanger Pit Showing Components of Fuel System Before Dismantling Operation.

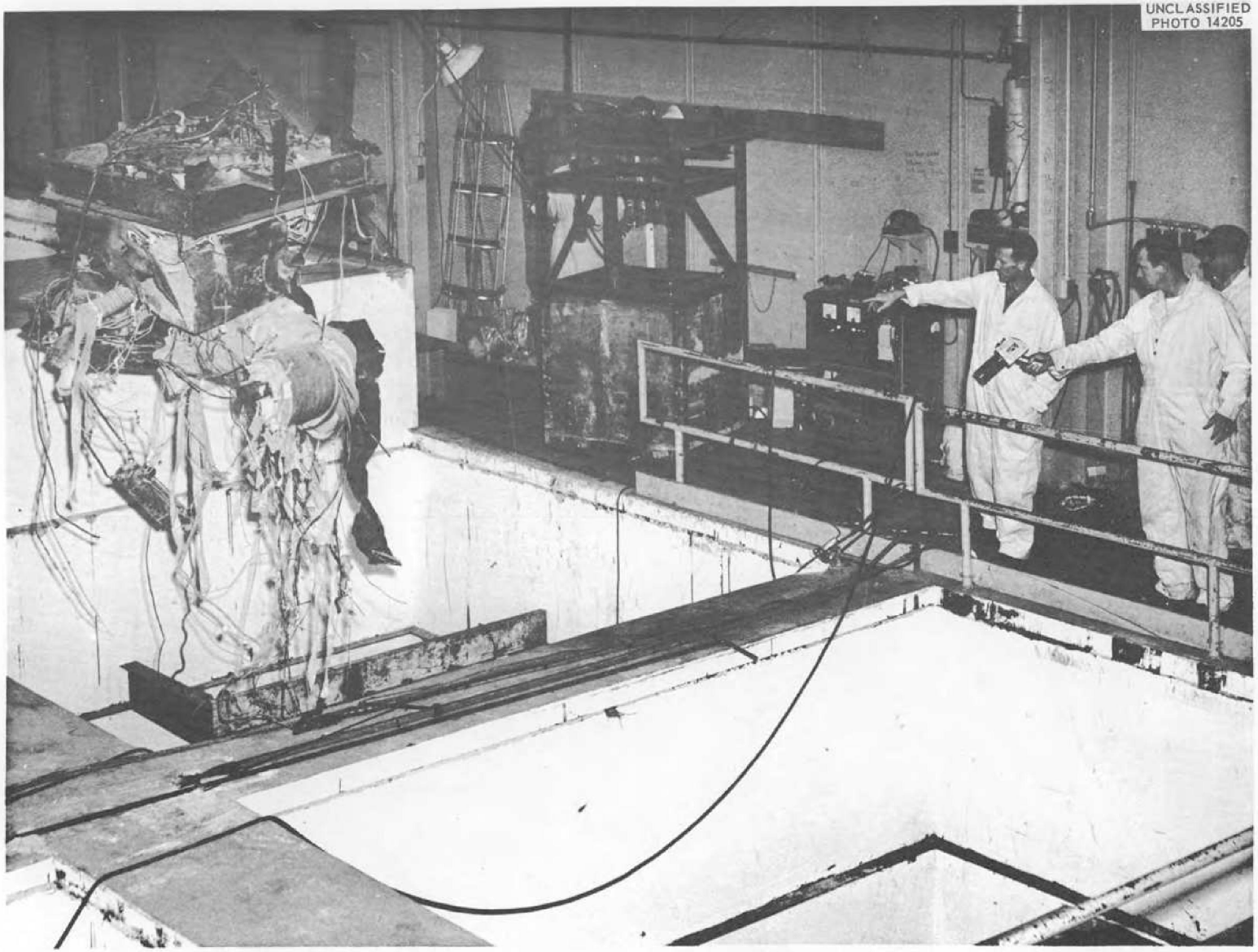

Fig. 7. Main Fuel Pump Being Lifted from Cell.

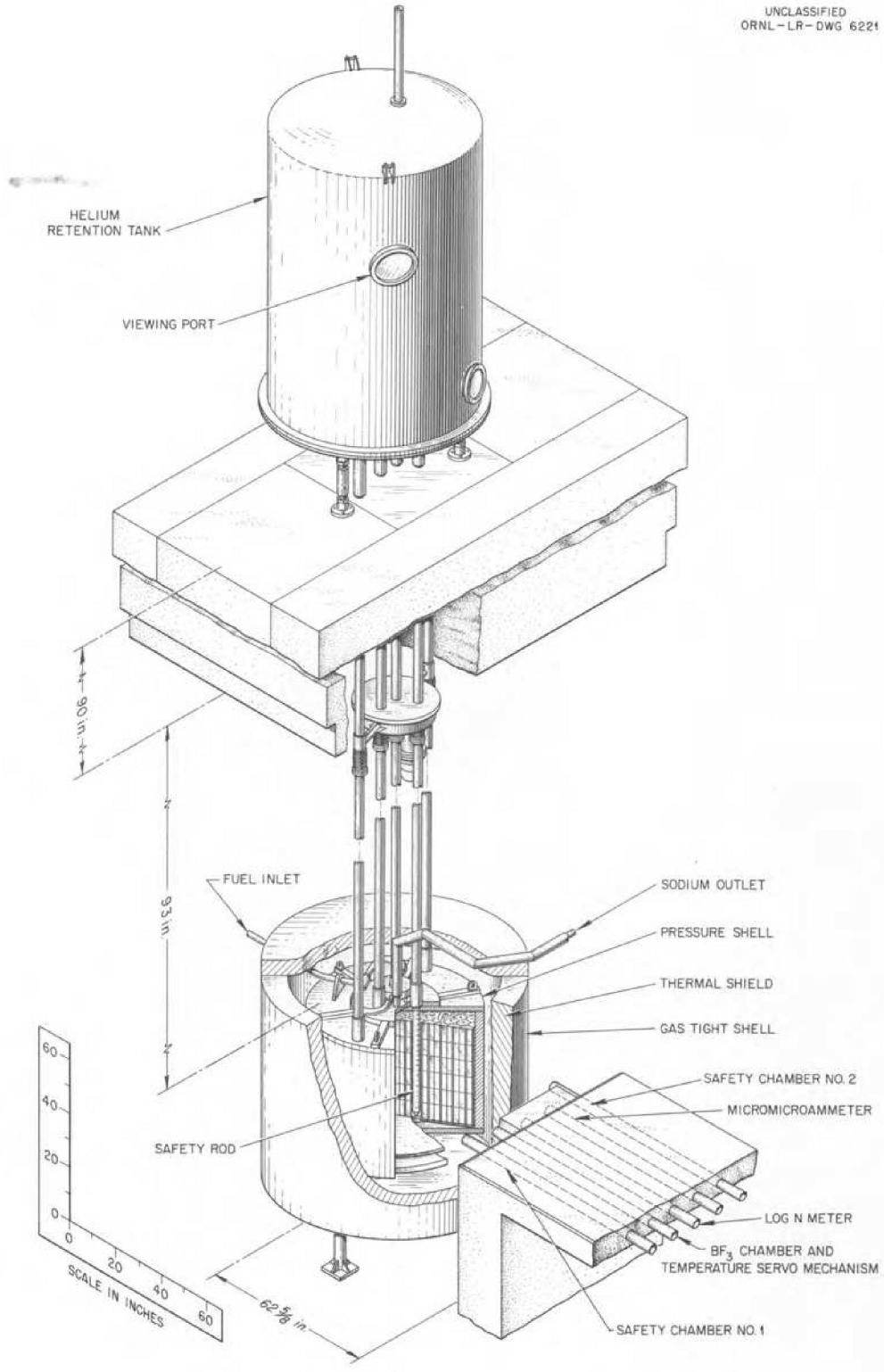

Fig. 8. Diagram of Reactor.

operation generated so much heat that it was necessary, as before, to cut the sodium lines with a hack saw. The sodium lines had radiation readings of 300 mr/hr at contact.

The gas-tight shell surrounding the reactor was then cut in quarter sections by using an electric arc (Fig. 9). The can had a radiation level of 200 mr/hr at contact. The exposed reactor pressure shell showed 6 r/hr at 6 in.

Special disassembly tools were fabricated for dismantling the reactor. This equipment included a steel tank with driven rollers for rotating the reactor and a hydraulic-powered cutoff grinder with a 14-in.-dia abrasive wheel. The reactor

Fig. 9. Gas-Tight Shell Being Removed from Reactor.

is shown in Fig. 10 positioned in the tank for grinding, and the cutting side of the grinder, which was designed to be used under liquid, is shown in Fig. 11. With this equipment, a period of five 8-hr days was required to cut the 2-in.-thick pressure shell. Upon removal of the shell, a radiation check was made. The top of the reactor showed a radiation level of 75 r/hr at contact and the side showed 37 r/hr.

The next task was the removal of fuel tubes and samples of the BeO moderator blocks. The fuel tubes and the BeO blocks were encased in a stainless steel can with welded seams, shown in Fig. 12, and the small sodium tubes

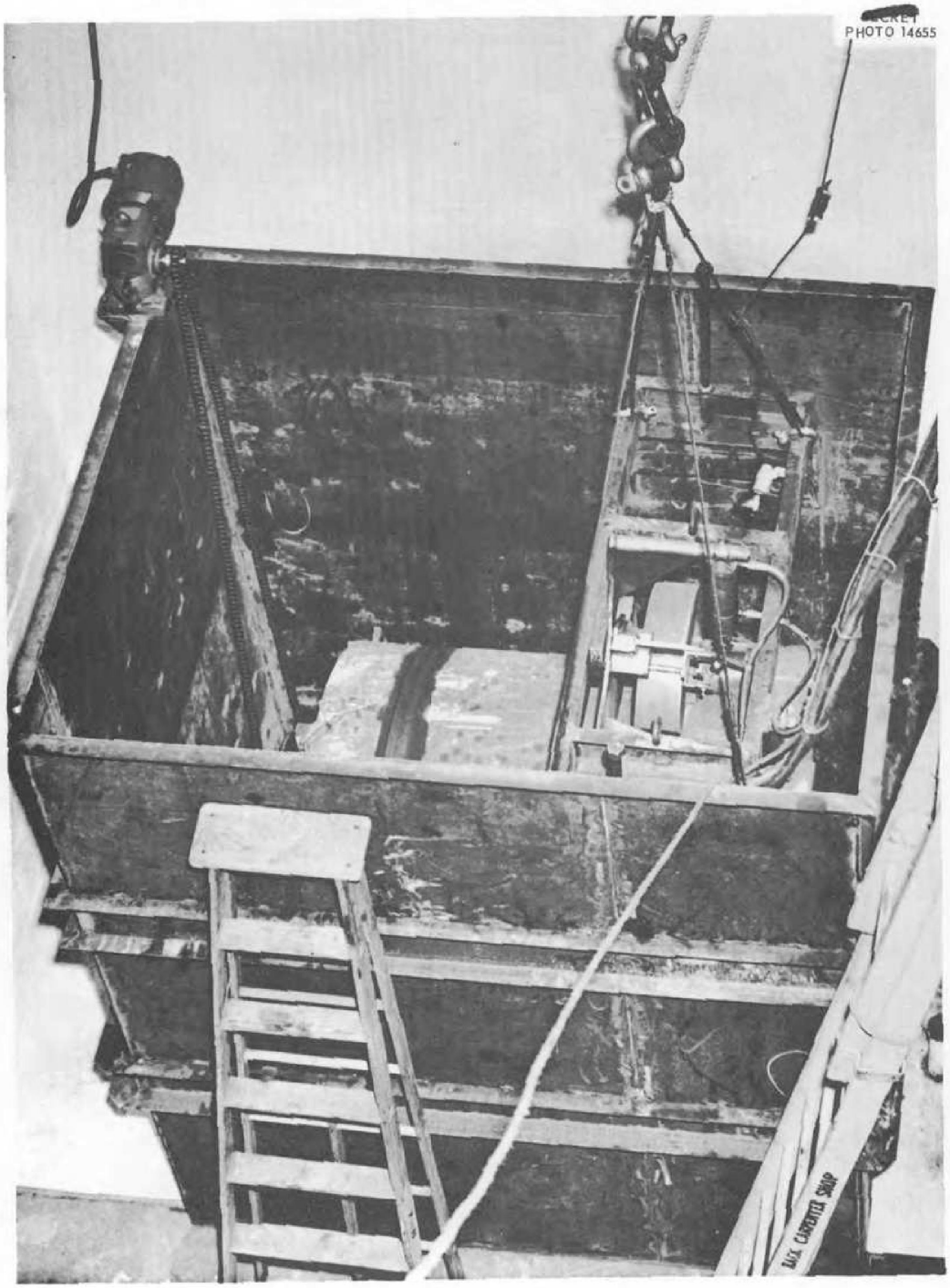

Fig. 10. Reactor Positioned in Disassembly Tank for Grinding.

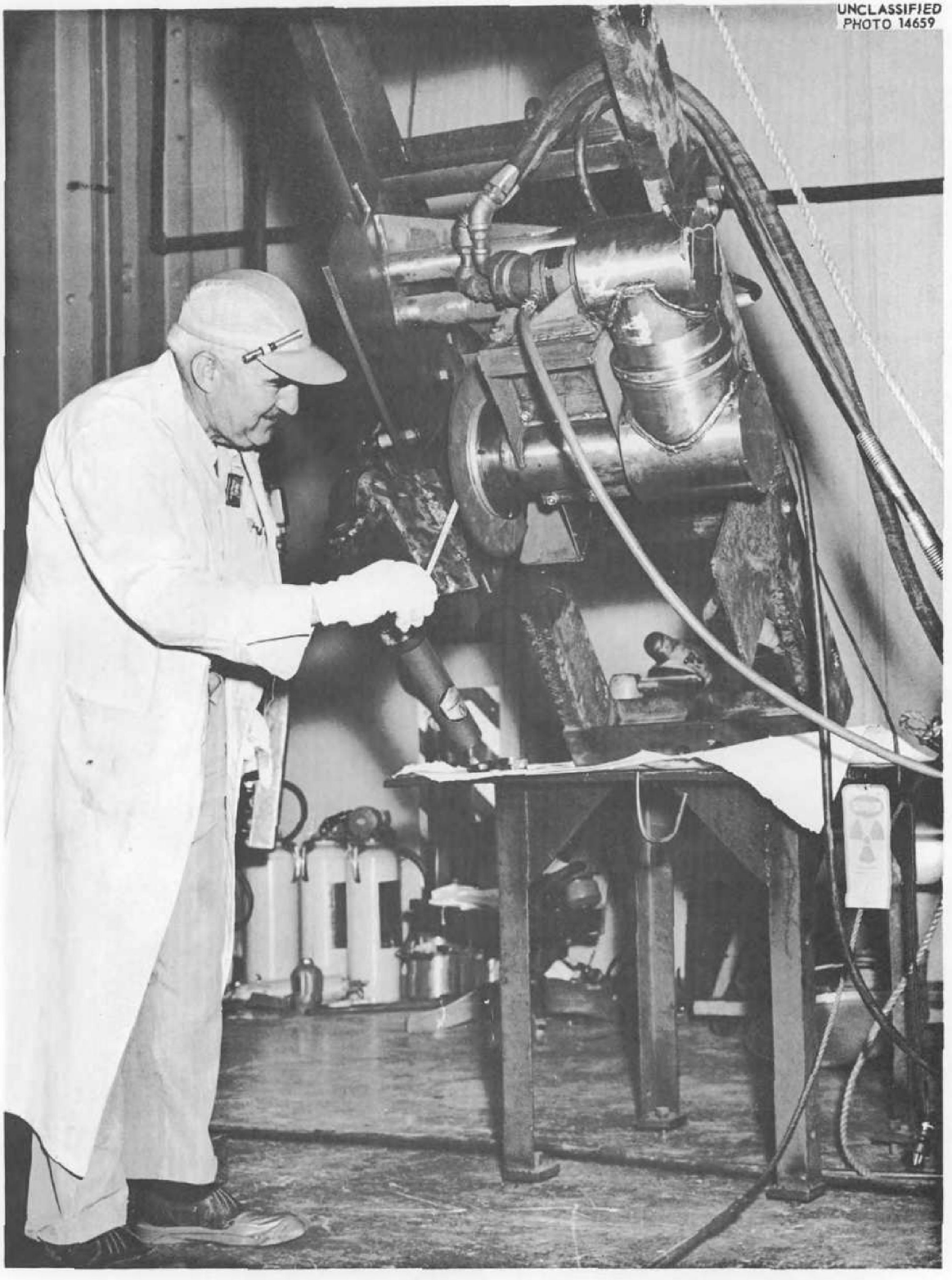

Fig. 11. Cutting Side of Grinder for Dismantling the Reactor.

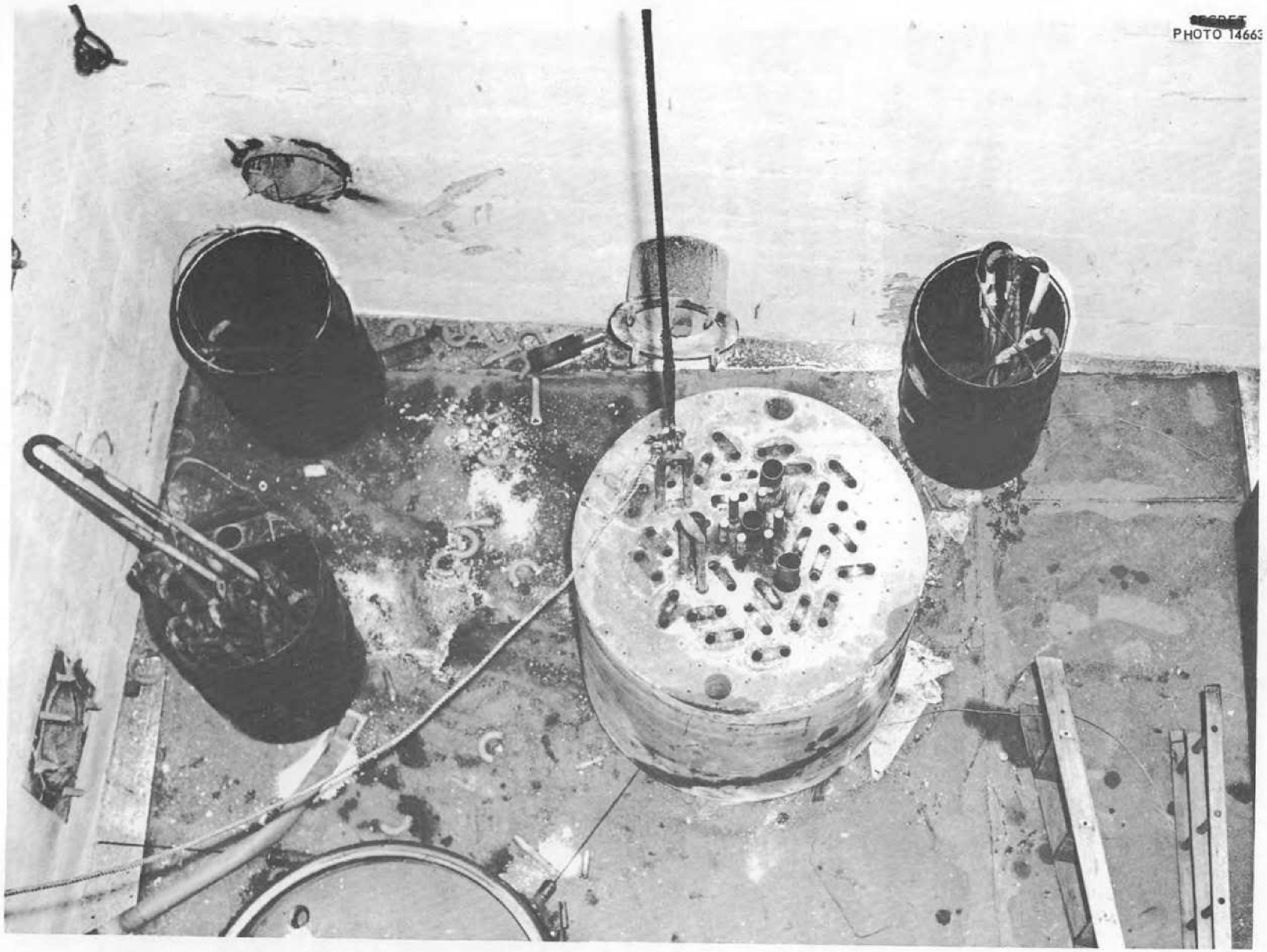

Fig. 12. Top of Reactor After Removal of Pressure Shell and Some of the Sodium and Fuel Tubes.

that penetrated the BeO blocks were welded to the top and bottom of the can. In order to free the top of the can, the welds holding the sodium tubes were drilled out and the outer edge of the can was cut. The reactor was then placed in a horizontal position and the fuel tube bends were cut off flush with the bottom of the can. After removal of the bottom tube bends, the reactor was set upright and the crane was used to pull out the fuel tubes and to place them in drums, as shown in Fig. 12.

With all the fuel tubes removed, only the BeO blocks remained in the can. The required samples of BeO were obtained, and the can containing the BeO blocks was taken to the storage area. The radiation reading at that time was 4.5 r/hr at 2 ft. The reactor cell was then cleared of all contaminated material left from disassembly of the reactor.

DISMANTLING OF THE FILL AND DRAIN TANKS

The fuel dump tank was removed first because it presented the only radiation in the tank pit, which also contained three sodium tanks and three fuel-carrier tanks. (One of the fuel-carrier tanks was not used.) A radiation survey showed 18 r/hr at contact with the top of the tank, 990 r/hr in one of the cooling tubes through the tank, and 84 r/hr at contact with insulation on the side of the tank.

In order to get the lead box into position for disassembly work, it was necessary to remove steel framework, a space cooler, and the stack at the top of the tank. With the lead box in position, the fuel lines were cut with the grinder and then taped to prevent loss of material and the spread of contamination. The tank was then removed to a storage area. After the dump tank and fuel lines had been removed, the radiation level in the pit was 20 to 50 mr/hr.

The lines to the three fuel carrier tanks and the three sodium tanks were then cut, and these tanks were removed from the pits with insulation and heaters intact. After removal of insulation and heaters, the sodium tanks were found to be free of radiation. One of the fuel carrier tanks, however, had a radiation level of 100 mr/hr and was therefore sent to storage. The sodium and the fuel carrier were salvaged from the non-radioactive tanks.

The tank pit was then cleared of small gas lines, valves, and wiring, and the dismantling of the tank pit was complete. General cleaning of the entire area then completed the disassembly of the ARE reactor and auxiliary systems.

SAMPLES TAKEN DURING DISASSEMBLY

Many samples were taken during disassembly of the ARE, and much re-usable equipment was salvaged. The samples taken are listed in Table 1, which also gives the type of examination requested and the status of the samples.

Sodium system samples were taken after the components had been removed from the cells. After all the samples had been taken, the remaining equipment that could not be salvaged was cleaned of sodium by immersing it in water until the sodium-water reaction was complete. Most of the equipment cleaned in this manner was later disposed of in the burial ground because of radiation levels of about 2 mr/hr.

Fuel system and reactor samples were taken in the reactor pit after the reactor had been disassembled and removed, except the sample from fuel supply line No. 120, which was taken before dismantling was started. After removal of the reactor, a horizontal cutoff saw was set up in the reactor pit for cutting the samples, which were moved to the reactor pit from the radioactive storage facility for sampling. All work involved in obtaining the fuel system and reactor samples had to be done from within the lead box with the use of improvised long-handled equipment, the portable grinder used for disassembly work, and the cutoff saw.

A sample of the fuel was taken from the fuel dump tank while the fuel was still molten. In order to obtain the sample, a hole had to be drilled in the top of the dump tank. Since the tank was at a temperature of between 1200 and 1300°F and the radiation level was high, the drilling had to be done from atop the cell plugs. A pipe extension which would reach to the top of the tank was adapted to the drill chuck, and the drill bit was welded to the pipe. Several

Table 1. List of Samples Taken from the ARE During Disassembly

Sample No.	Description	Type of Examination Requested	Status of Examination
		Sodium System Samples	
S1	Section of moderator coolant return line 303	Metallographic	Sample to be examined
S2	Section of moderator coolant supply line 310	Metallographic	Sample to be examined
S3	Section of main Na pump impeller	Metallographic	Sample to be examined
S4	Seat from sodium valve U-23	Stereophotographic	Results of examination given in this report
S5	Plunger from sodium valve U-23	Stereophotographic	Results of examination given in this report
S6	Bellows from sodium valve U-23	Metallographic	Results of examination given in this report
S7	Bellows from sodium pressure transmitter in line 303	Metallographic	Sample to be examined
S8	Section from cold dead leg No. 931	Metallographic	Request rescinded, CF-56-6-24; sample to be disposed of
S9	Section from cold dead leg No. 932	Metallographic	Request rescinded, CF-56-6-24; sample to be disposed of
	Sample of scale in main Na pump	Spectrographic analysis	Spec Lab Report No. 3450, and this report
	Section of line 308 to main Na pump	Activation analysis	CF-55-10-35, and this report
	Section of line 305 to Na pressure transmitter No. 2	Activation analysis	CF-55-10-35, and this report
	Sample of sodium	Activation analysis	CF-55-10-35, and this report
		Fuel System Samples	
F1	Section of fuel supply line 120	Metallographic and activation	CF-55-2-58, CF-55-2-36, and this report
F2	Section of fuel return line 111	Metallographic	Request rescinded, CF-56-6-24; sample to be disposed of
F3	Section of main fuel-pump impeller	Metallographic	Request rescinded, CF-56-6-24; sample to be disposed of
F4-1	Main fuel-pump bowl	Visual	Sample examined visually and found to be in good condition; sample to be disposed of
F4-2	Main fuel-pump bottom baffle plate	Visual	Sample examined visually and found to be in good condition; sample to be disposed of
F4-3	Main fuel-pump socket-weld joint	Visual	Sample examined visually and found to be in good condition; sample to be disposed of
F4-4	Main fuel-pump flange outside bowl	Visual	Sample examined visually and found to be in good condition; sample to be disposed of
F4-5	Main fuel-pump spark plug	Visual	Sample examined visually and found to be in good condition; sample to be disposed of
F4-6	Main fuel-pump channel	Visual	Sample examined visually and found to be in good condition; sample to be disposed of
F5	Section of inlet to fuel heat exchanger No. 2	Metallographic	Results of examination given in this report

Table 1. (continued)

Sample No.	Description	Type of Examination Requested	Status of Examination
		Fuel System Samples	
F6	Section of inlet to fuel heat exchanger No. 2	Metallographic	Request suspended, CF-56-6-24; sample to be held in storage
F7	Section of outlet to fuel heat exchanger No. 2	Metallographic	Request suspended, CF-56-6-24; sample to be held in storage
F8	Section of outlet to fuel heat exchanger No. 2	Metallographic	Request suspended, CF-56-6-24; sample to be held in storage
F9	Section of middle of bends in heat exchanger No. 2	Metallographic	Results of examination given in this report
F10	Section of middle of bends in heat exchanger No. 2	Metallographic	Request suspended, CF-56-6-24; sample to be held in storage
F11	Seat from fuel valve U-1	Stereophotographic	Results of examination given in this report
F12	Plunger from fuel valve U-1	Stereophotographic	Results of examination given in this report
F13	Bellows from fuel valve U-1	Stereophotographic	Request suspended, CF-56-6-24; sample to be held in storage
F14	Seat from fuel valve U-2	Stereophotographic	Request suspended, CF-56-6-24; sample to be held in storage
F15	Plunger from fuel valve U-?	Stereophotographic	Request suspended, CF-56-5-24; sample to be held in storage
F16	Bellows from fuel valve U-2	Stereophotographic	Request suspended, CF-56-6-24; sample to be held in storage
F17	Tapered plug from main fuel rotameter	Metallographic	Request rescinded, CF-56-6-24; sample to be disposed of
	Sample of fuel from dump tank	Activation analysis	CF-55-1-128 (results analyzed in CF-55-2-36), and this report
	Hot fuel dump tank with fuel	Fuel to be reprocessed	See ANP Quar. Prog. Reps.
		Reactor Samples	
R1	Section of serpentine fuel tube close to pressure shell	Metallographic	Request rescinded, CF-56-6-24; sample to be disposed of
R2	Section of serpentine at $\frac{1}{2}$ radius	Metallographic	Request rescinded, CF-56-6-24; sample to be disposed of
R3	Section of serpentine in center of core	Metallographic	Results of examination given in this report
R4	Serpentine bend from region near outer surface	Metallographic	Request rescinded, CF-56-6-24; sample to be disposed of
R5	Socket weld joint near center	Metallographic	Request rescinded, CF-56-6-24; sample to be disposed of
R6	Socket weld joint near outer surface	Metallographic	Request rescinded, CF-56-6-24; sample to be disposed of
R7	Serpentine bend from center of reactor	Metallographic	Results of examination given in this report
R8	Sample of pressure shell wall	Metallographic	Results of examination given in this report

Table 1. (continued)

Sample No.	Description	Type of Examination Requested	Status of Examination
		Reactor Samples	
R9	BeO block from outer region of core	Metallographic	CF-56-6-113, and this report
R10	BeO block from central region of core	Metallographic	CF-56-6-113, and this report
R11	BeO block cut prior to installation	Metallographic	CF-56-6-113, and this report
R12	BeO block cut prior to installation	Metallographic	CF-56-6-113, and this report
	Reactor inlet mainfold	Metallographic	Sample in storage to be disposed of
	Reactor outlet manifold	Metallographic	Sample in storage to be disposed of
		Miscellaneous Samples	
M1	U-belts from main fuel pumps	Visual	CF-55-7-27, and this report
M2	O-ring from main fuel pumps	Visual	CF-55-7-27, and this report
M3	Oil sample from main fuel pump	Visual plus viscosity check	CF-55-7-27, and this report
M4	Oil sample from standby Na pump	Visual plus viscosity check	CF-55-7-27, and this report
M5	Oil sample helium blower	Visual	CF-55-7-27, and this report
M6	Electric wire from main fuel pump region	Visual	CF-55-7-27, and this report
M7	Thermocouple wire from main fuel pump region	Visual	CF-55-7-27, and this report
M8	Thermal insulation from main fuel pump region	Visual	CF-55-7-27, and this report
M9	Concrete from wall near main fuel pump	Activation analysis	Wall since decontaminated
	Rubber diaphragm from H_2O valve B139	Visual plus viscosity check	CF-55-7-27, and this report
	Thermal insulation wrapping paste on line 303	Visual	CF-55-7-27, and this report
	Thermal insulation on reactor	Visual	CF-55-7-27, and this report
	Thermocouple lead wire at reactor	Visual	CF-55-7-27, and this report
	Braided thermocouple wire on top of reactor	Visual	CF-55-7-27, and this report
	Main fuel pump vapor trap and lines	Visual	Results of examination given in this report
	Standby fuel dump tank pressure transmitter 7	Performance test and visual	Results of examination given in this report
	Main fuel dump tank pressure transmitter 6	Performance test and visual	Results of examination given in this report
	Hot fuel dump tank pressure transmitter 10	Performance test and visual	Not examined; sample to be disposed of

Table 1. (continued)

Sample No.	Description	Type of Examination Requested	Status of Examination
		Miscellaneous Samples	
	Main fuel pump vent solenoid U-19	Leak test and visual	Results of examination given in this report
	Main fuel pump vent solenoid U-86	Leak test and visual	Results of examination given in this report
	Main fuel pump He supply solenoid U-82	Leak test and visual	Results of examination given in this report
	Standby fuel pump vent solenoid U-185	Leak test and visual	Results of examination given in this report
	Main fuel blower He supply solenoid B-233	Leak test and visual	Results of examination given in this report
	Pit-activity-monitor isolation solenoid B-418	Leak test and visual	Results of examination given in this report
	Off-gas system monitron inlet solenoid B-124	Leak test and visual	Results of examination given in this report
	Off-gas system monitron inlet solenoid B-125	Leak test and visual	Results of examination given in this report
	Standby fuel pump vent solenoid U-89	Leak test and visual	Results of examination given in this report
	Standby fuel pump emergency vent solenoid U-159	Leak test and visual	Results of examination given in this report
	Main fuel pump emergency vent solenoid U-10	Leak test and visual	Results of examination given in this report
	Sample of inlet emergency off-gas line	Activation analysis	CF-55-2-36, and this report
	Sample of off-gases from cell	Activation analysis	Undocumented memo 12-22-55, results described in CF-55-2-36

drill bits, made of different metals, were tried before the hole was finally completed. To remove the fuel sample, a 3/8-in.-dia Inconel tube was passed through the opening in the tank, and a small vacuum pump was used to pull fuel into the tube. The tube was then removed from the tank and allowed to cool. A section of the tube containing fuel was then cut to provide the desired sample.

In addition to the sodium system, fuel system, and reactor samples, numerous samples were taken from other parts of the various systems during the dismantling operations. The samples taken are described in Table 1.

RESULTS OF EXAMINATIONS OF SAMPLES

BERYLLIUM OXIDE BLOCKS[2]

Beryllium oxide blocks were taken for examination from the outer region, the central region, and the core region of the ARE. The stacking arrangement may be seen in Fig. 13, which shows the top of the reactor during assembly. The blocks surrounding the serpentine fuel tubes were split to facilitate assembly, but the blocks with small holes for sodium-coolant tubes were not cut. The small spaces between the blocks were filled with slowly moving sodium.

The blocks that were removed are shown in Figs. 14, 15, and 16. It was found that draining the reactor had left the surfaces of the BeO blocks essentially free of sodium. Since it was not necessary to strip sodium from the blocks it could be assumed that any damage that was found during the initial examination had occurred during operation or was present in the as-fabricated material. The post-test handling could not have caused further damage.

The block shown in Fig. 14, which had surrounded a fuel tube in the central region of the reactor, has many visible cracks and one half of the block had fractured. Previous tests of the as-received blocks had revealed that nearly all the blocks had at least one crack and some of the blocks had several cracks.[3] Comparison with photographs of the as-received blocks showed that slight erosion of the edges occurred during reactor operation. The presence of small flakes of BeO adhering to the surface of the block indicated some spalling.

A BeO block that surrounded a fuel tube in the outer region of the reactor is shown in Fig. 15. The droplets visible on the surface are sodium hydroxide produced from sodium that remained after removal from the reactor. There were no complete fractures of this block, but many cracks are visible.

Three cut blocks from the core are shown in Fig. 16. Two of the three blocks fractured during operation of the ARE; however, no cracks are visible.

Only the BeO block taken from the outer region of the core (Fig. 15) was examined to determine the depth of penetration of the sodium. The dark areas in the transverse sections shown in Figs. 17 and 18 indicate high sodium concentrations in porous material, whereas the light areas, which were not penetrated by sodium, are dense material. The dark lines indicate sodium in cracks and crevices. As may be seen, the core of the block is more porous than the periphery. Cracks did not propagate through the dense material. In general, the BeO blocks withstood the operation of the ARE reasonably well.

The remainder of the blocks were stored in place in the reactor. When final disposition of the reactor was to be made in October 1957, it was found that, as a result of atmospheric action on the adsorbed sodium, the blocks had deteriorated to the point where they had lost their structural strength. Since the blocks could serve no further useful purpose, they were buried along with the reactor. The results of attempts to remove blocks for examination prior to the disposal decision are shown in Figs. 19 and 20, and the reactor may be seen in Figs. 21, 22, and 23 just prior to burial.

[2]Material abstracted from a report by R. J. Gray and E. L. Long, Jr., *Examination of BeO Blocks from the ARE*, ORNL CF-56-6-113 (June 18, 1956).

[3]L. M. Doney, *Structure of BeO Block with the $1\frac{1}{8}$ in. Central Hole*, ORNL CF-52-11-146 (Nov. 17, 1952).

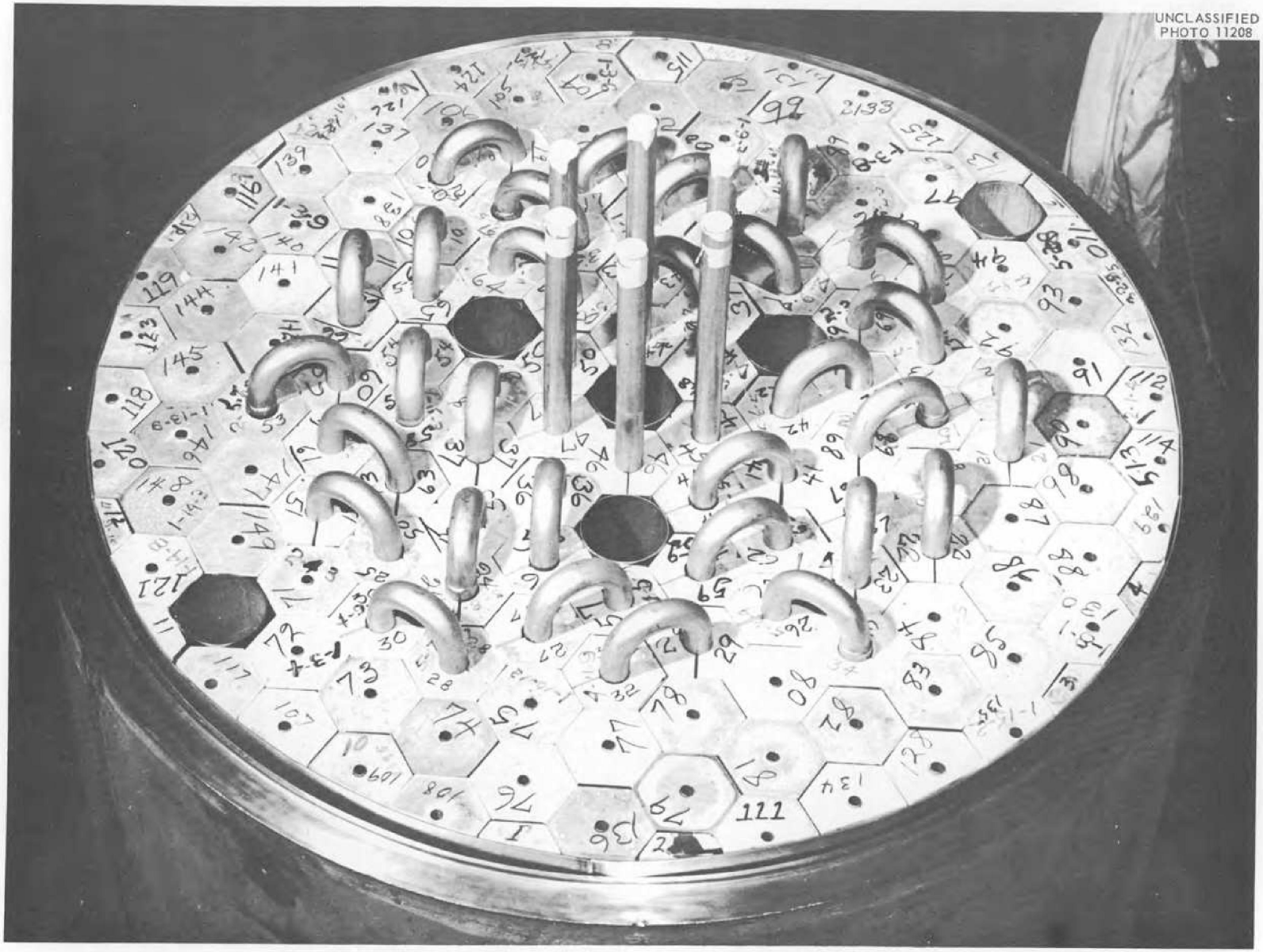

Fig. 13. ARE Reactor Core During Assembly. (Secret with caption)

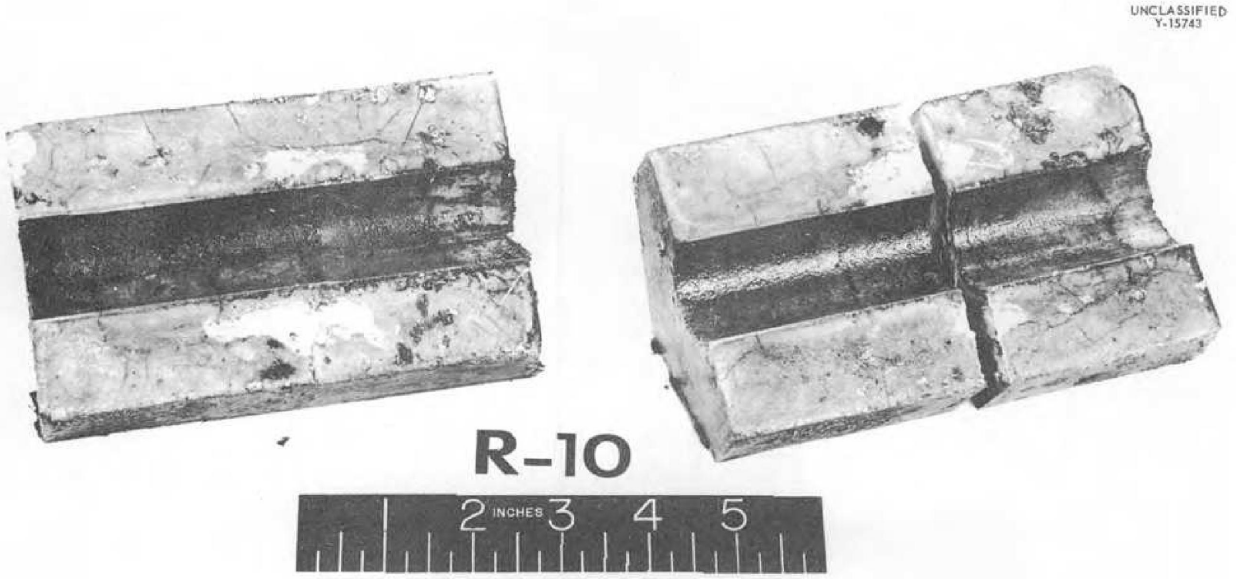

Fig. 14. BeO Block Which Had Surrounded a Fuel Tube in the Central Region of the Reactor.

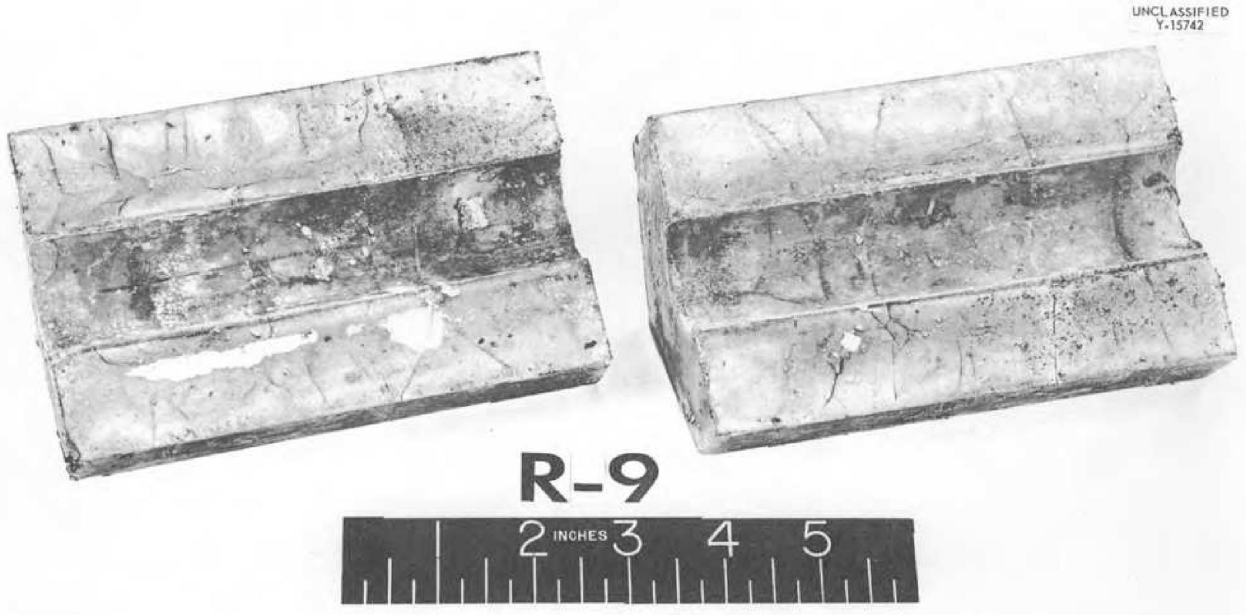

Fig. 15. BeO Block Which Had Surrounded a Fuel Tube in the Outer Region of the Reactor.

Fig. 16. Three Cut BeO Blocks from Reactor Core.

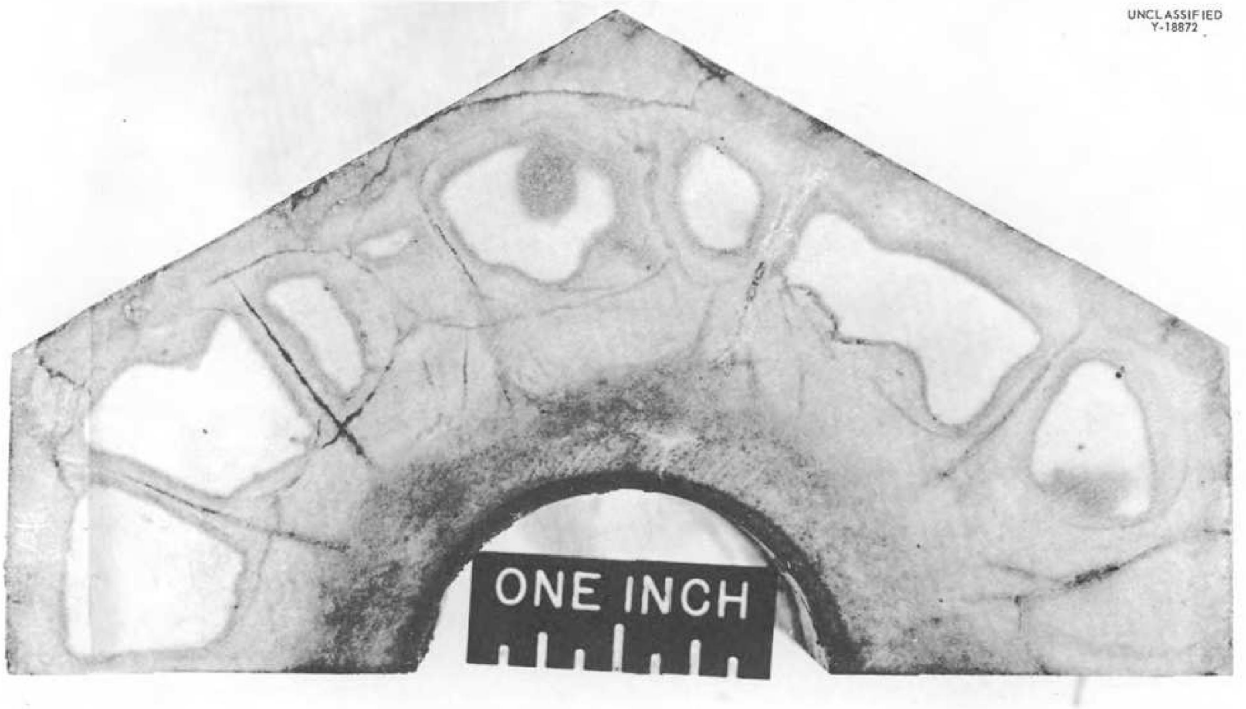

Fig. 17. Transverse Section of Block Shown in Fig. 15.

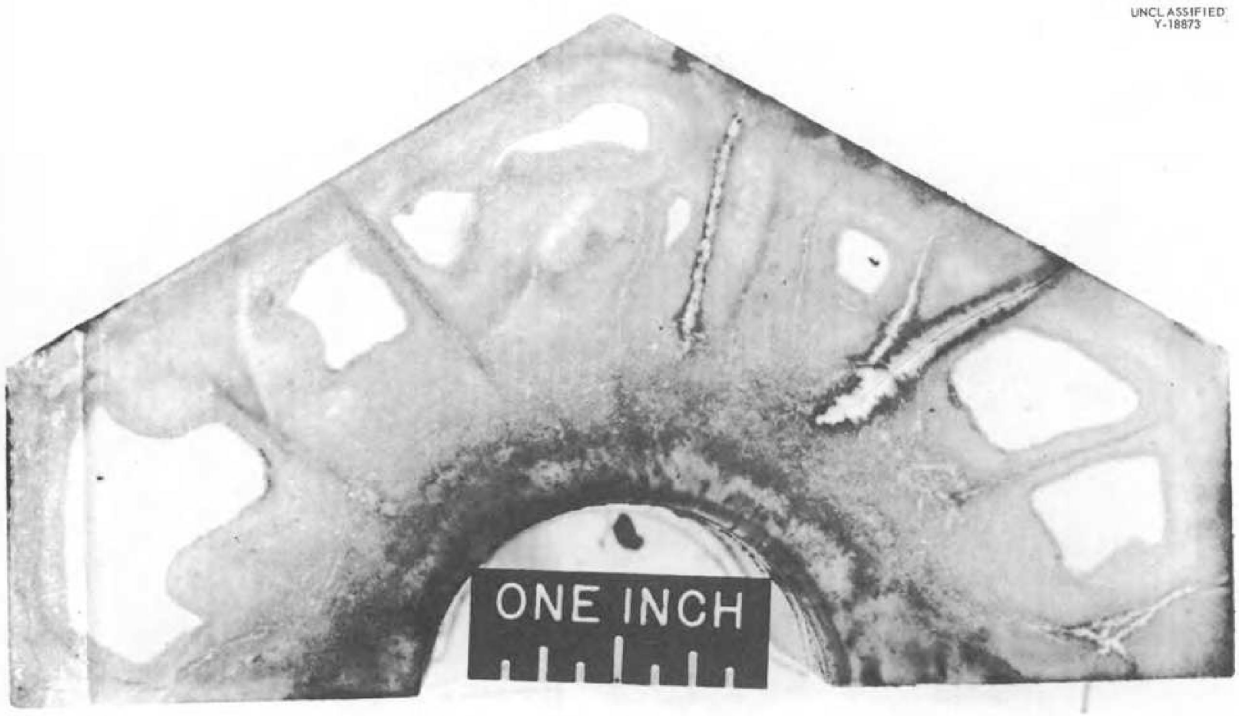

Fig. 18. Another Transverse Section of Block Shown in Fig. 15.

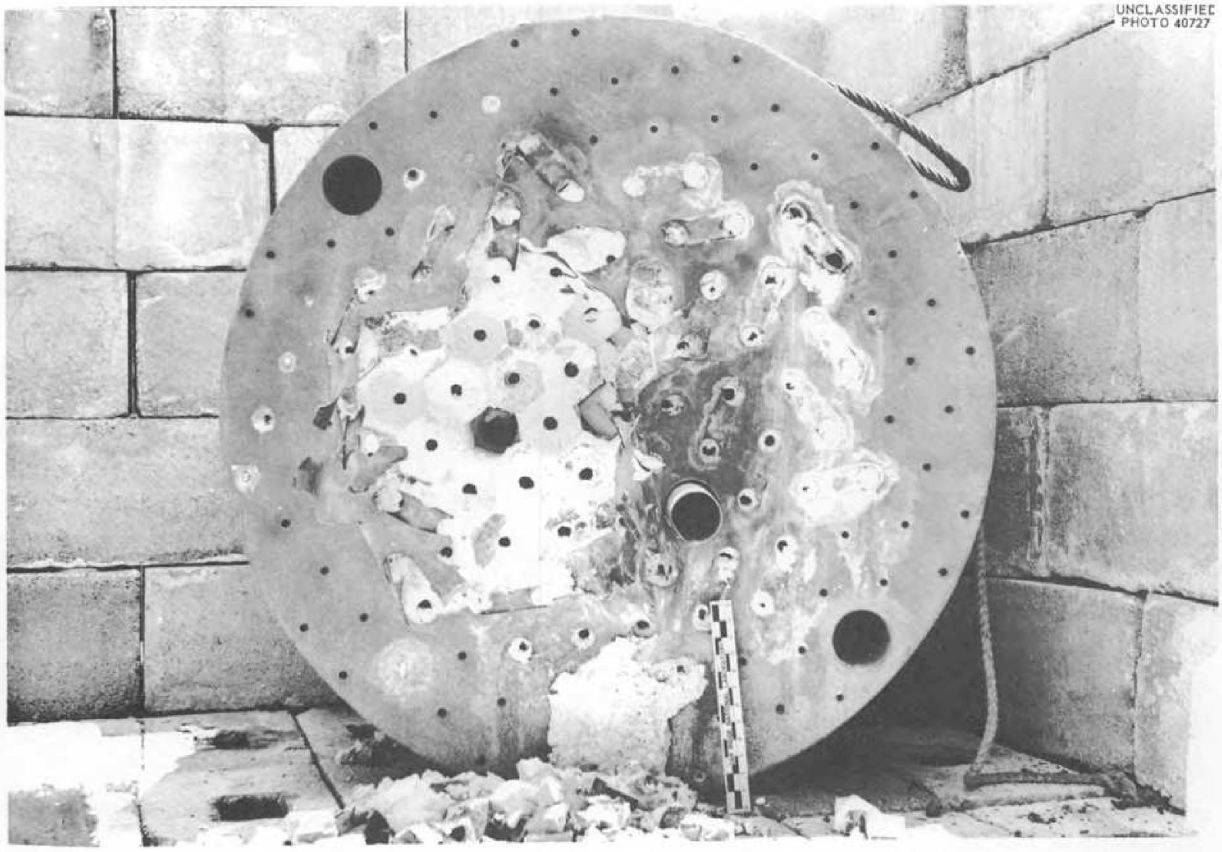

Fig. 19. Condition of Reactor When Removed from Storage in October 1957. The broken pieces of BeO blocks on the floor resulted from attempts to remove blocks for examination.

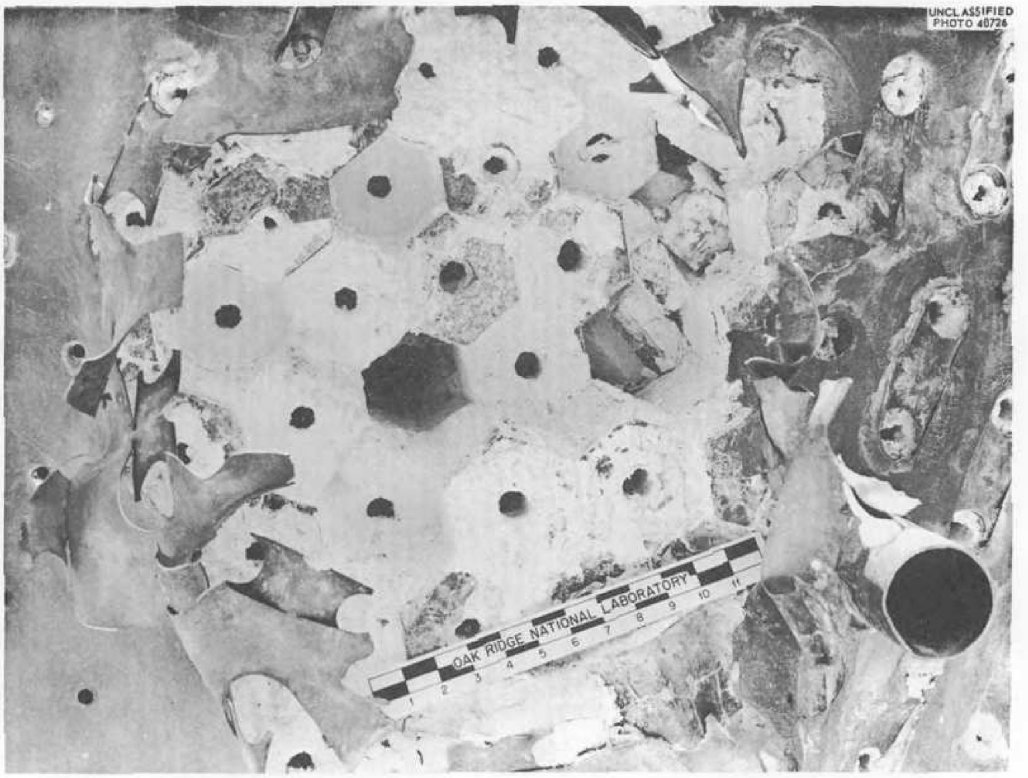

Fig. 20. View of Deteriorated BeO Blocks in Reactor.

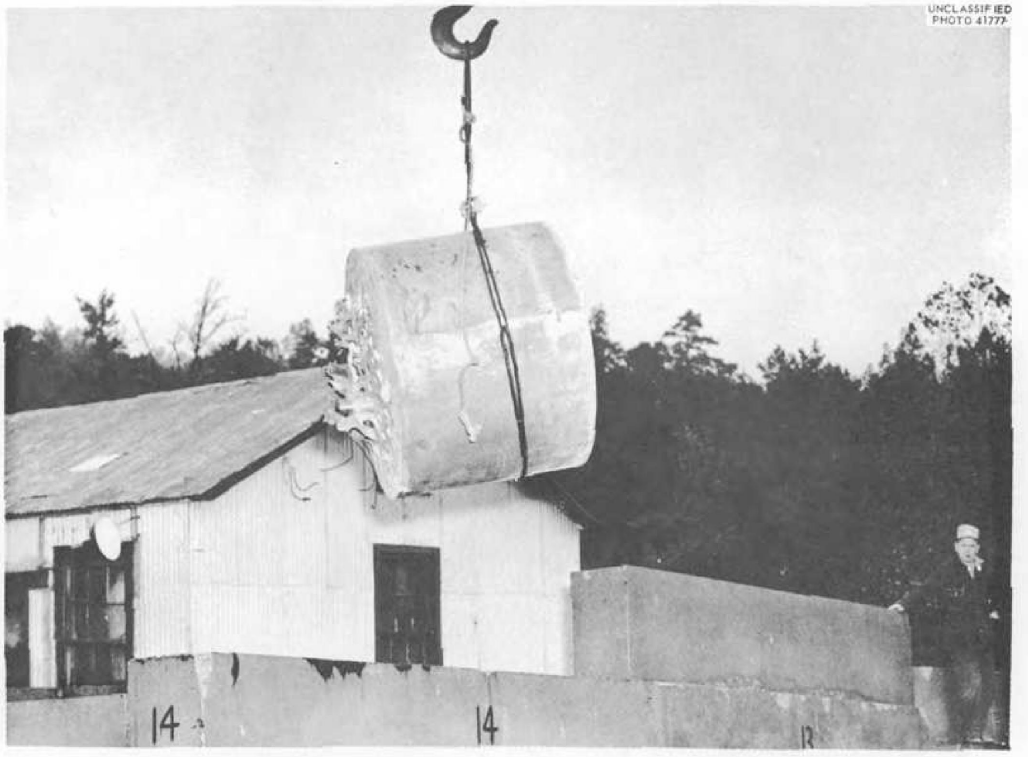

Fig. 21. Reactor Being Removed from Storage for Transfer to Burial Ground.

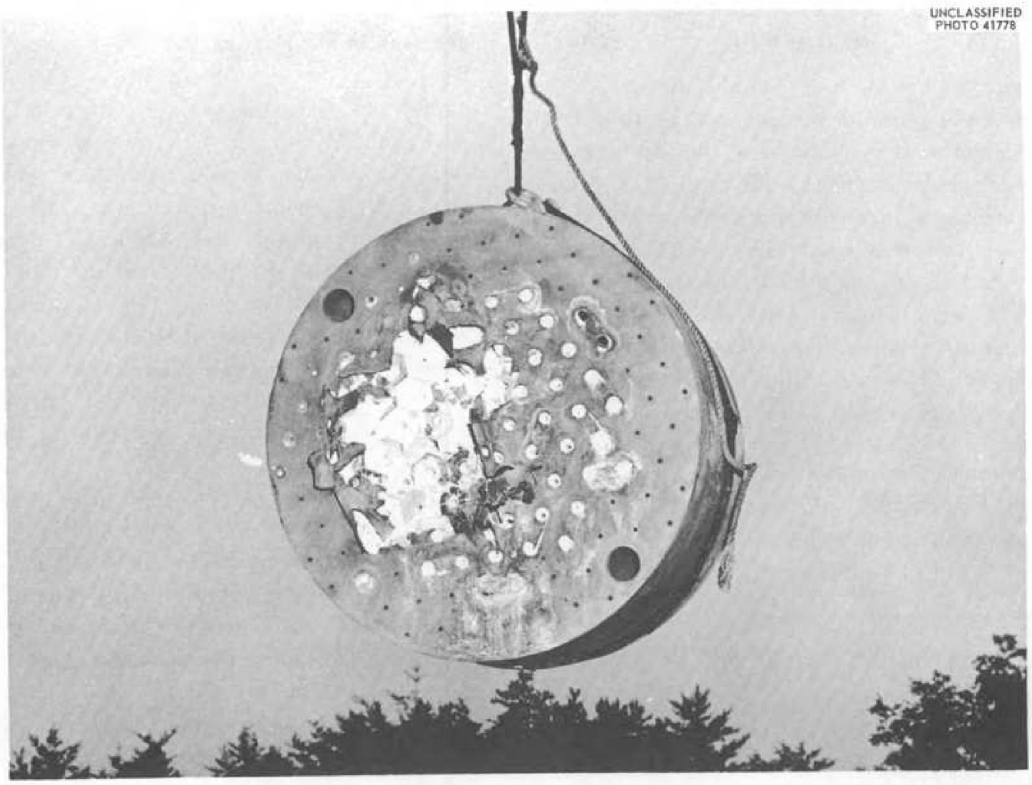

Fig. 22. Top View of Reactor During Removal from Storage.

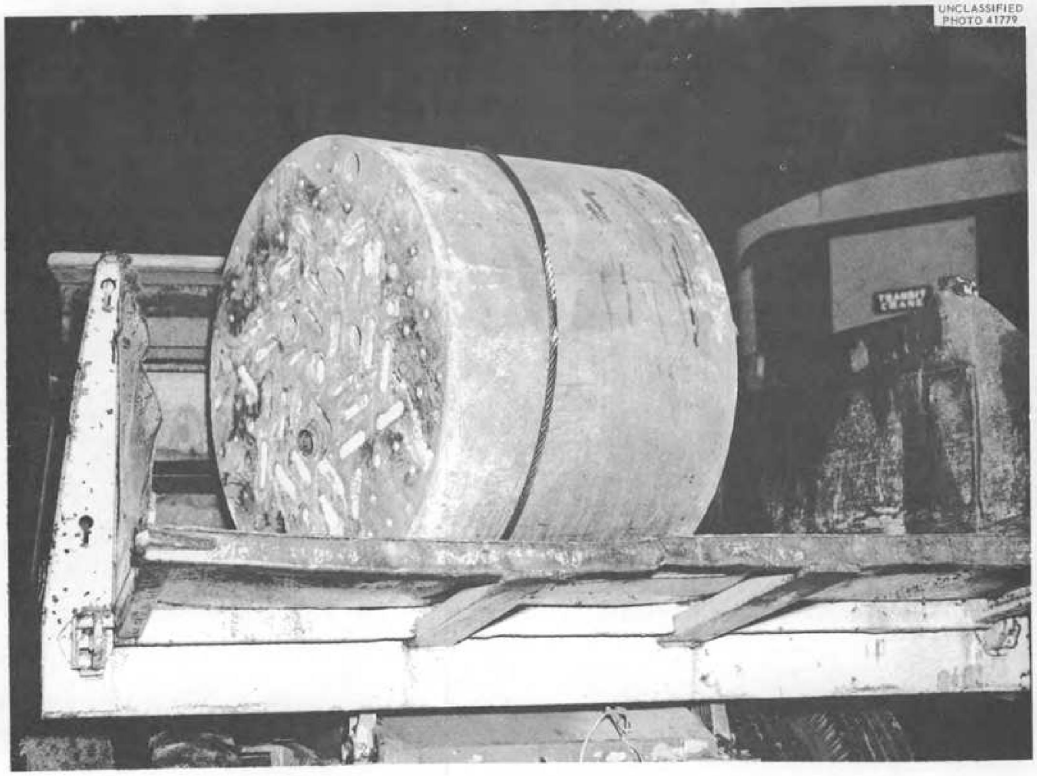

Fig. 23. Bottom View of Reactor Ready for Transfer to Burial Ground.

STRUCTURAL MATERIALS AND VALVE COMPONENTS

Specimen F1 from fuel supply line 120 was found[4] to have general surface attack to a depth of 1 to 2 mils (Fig. 24), and the surface was rough, as shown in Figs. 25 and 26. Stereophotographs were taken of the seats and plungers from valves U-1 and U-23 (specimens F11, F12, S4, and S5).[5] As shown in Figs. 27 and 28, there were dark deposits on the Stellite seat and plunger from valve U-1. The Stellite plunger from valve U-23 was scored, as shown in Fig. 29; the seat of valve U-23 is shown in Fig. 30.

Metallographic examinations were completed of specimens R3, R7, R8, F5, F9, and S6.[6] Specimen R8, a section from the pressure shell wall, including a weld, is shown in Fig. 31. A crack may be seen at the weld junction, and there are several voids in the weld area. A section from a serpentine bend in a fuel tube in the center of the reactor, specimen R7, is shown in Figs. 32 and 33. There was subsurface void formation on the interior wall to a maximum depth of 3.5 mils, but the penetration was not uniform. Some parts of the wall showed deeper and more dense penetration than others. The outer wall of the serpentine bend, which was in contact with sodium, showed what appeared to be a mass transfer deposit (Fig. 34) in addition to some subsurface void formation. The deposit had plated on the wall to a maximum thickness of 1 mil.

Specimen R3, which was also taken from a serpentine bend in a fuel tube in the center of the core, also showed some subsurface void formation; however, the density and depth of penetration were not so great as for specimen R7. The areas of attack were localized, and some areas of the wall showed no attack, as may be seen in Figs. 35 and 36. No deposit similar to that found on specimen R7 was noted on the exterior wall.

[4] M. J. Feldman, *ARE — Line 120 (Inner Surface)*, ORNL CF-55-2-58 (Feb. 11, 1955).

[5] A. E. Richt *et al.*, *ANP Quar. Prog. Rep. June 30, 1957*, ORNL-2340, p 267.

[6] A. E. Richt *et al.*, *ANP Quar. Prog. Rep. Sept. 30, 1957*, ORNL-2387 (in press).

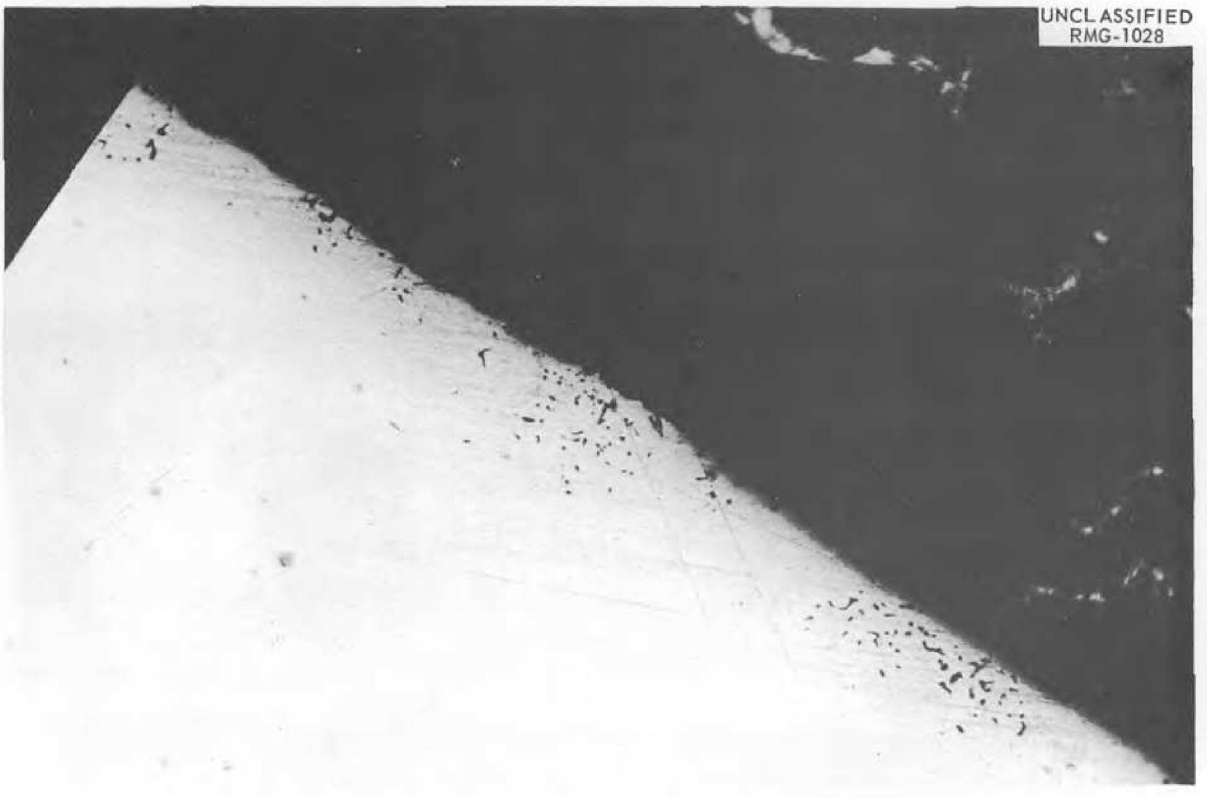

Fig. 24. Inner Surface of Fuel Supply Line 120 (Specimen F1). Unetched. 250X.

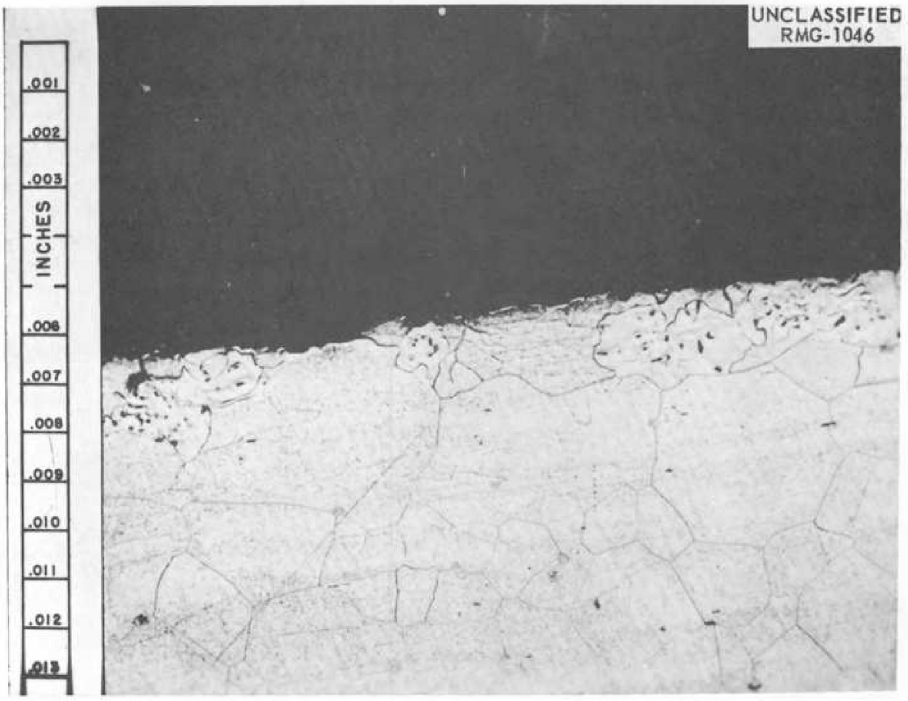

Fig. 25. Etched Section from Specimen F1. Etchant: electrolytic oxalic acid (10%). 250X.

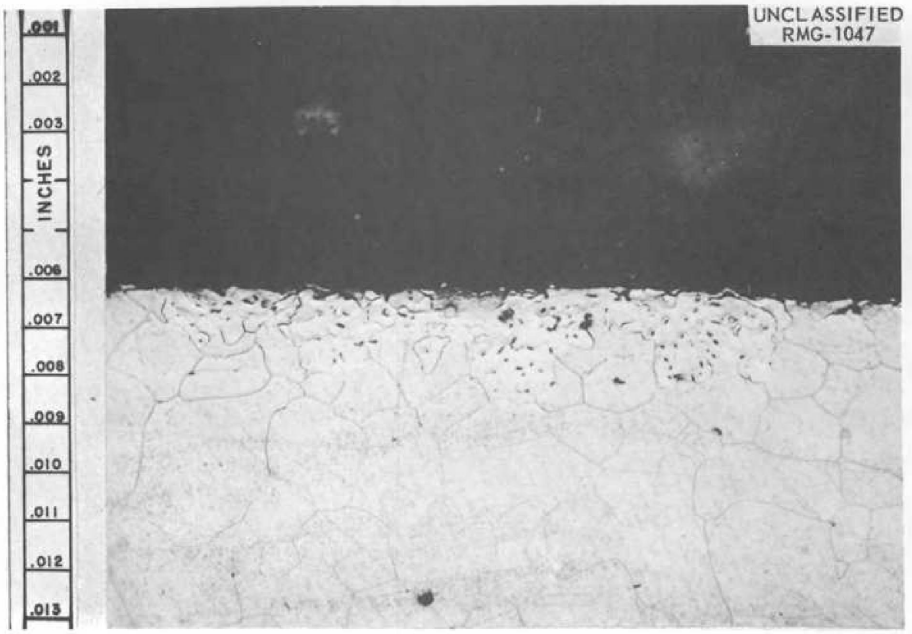

Fig. 26. Another Section from Specimen F1. Etchant: electrolytic oxalic acid (10%). 250X.

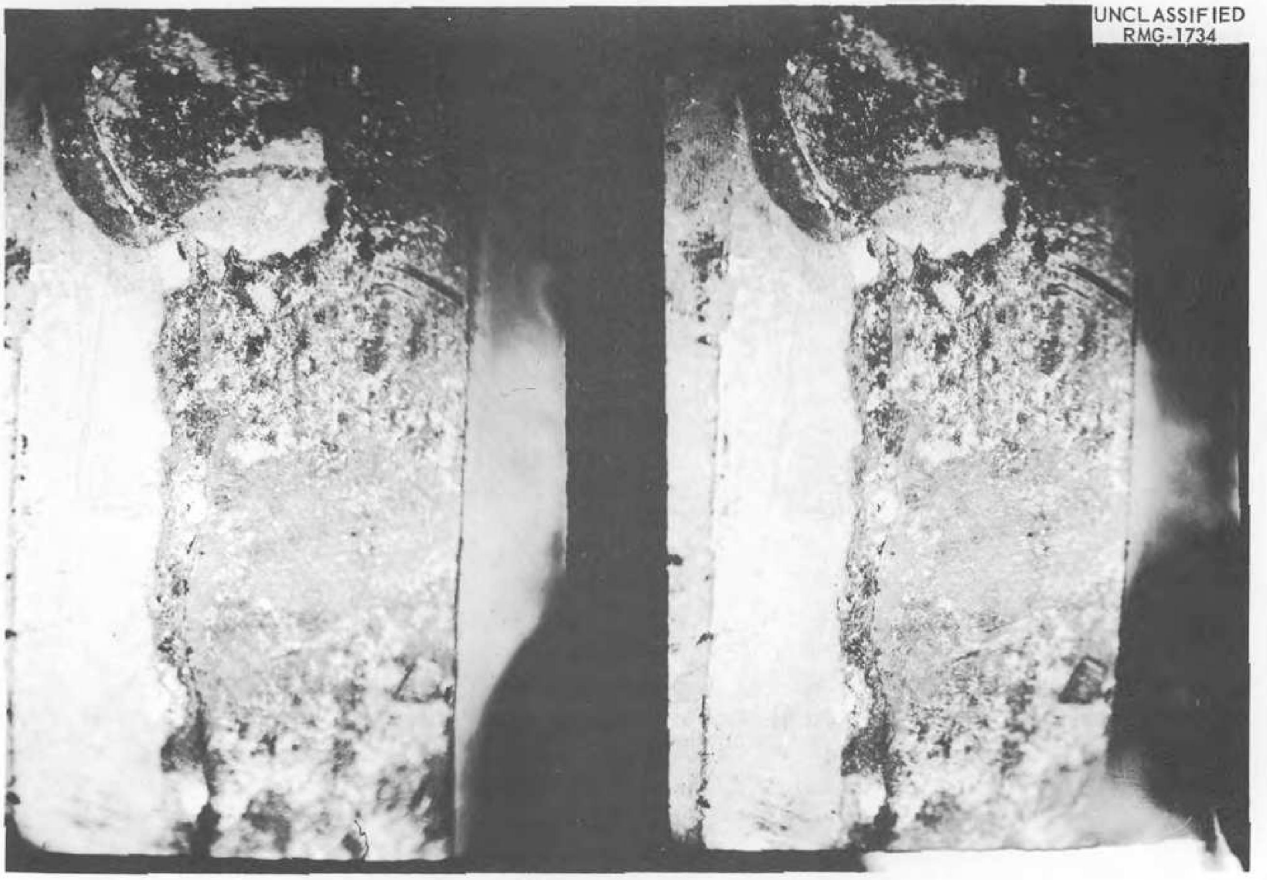

Fig. 27. Plunger from Valve U-1. 5X.

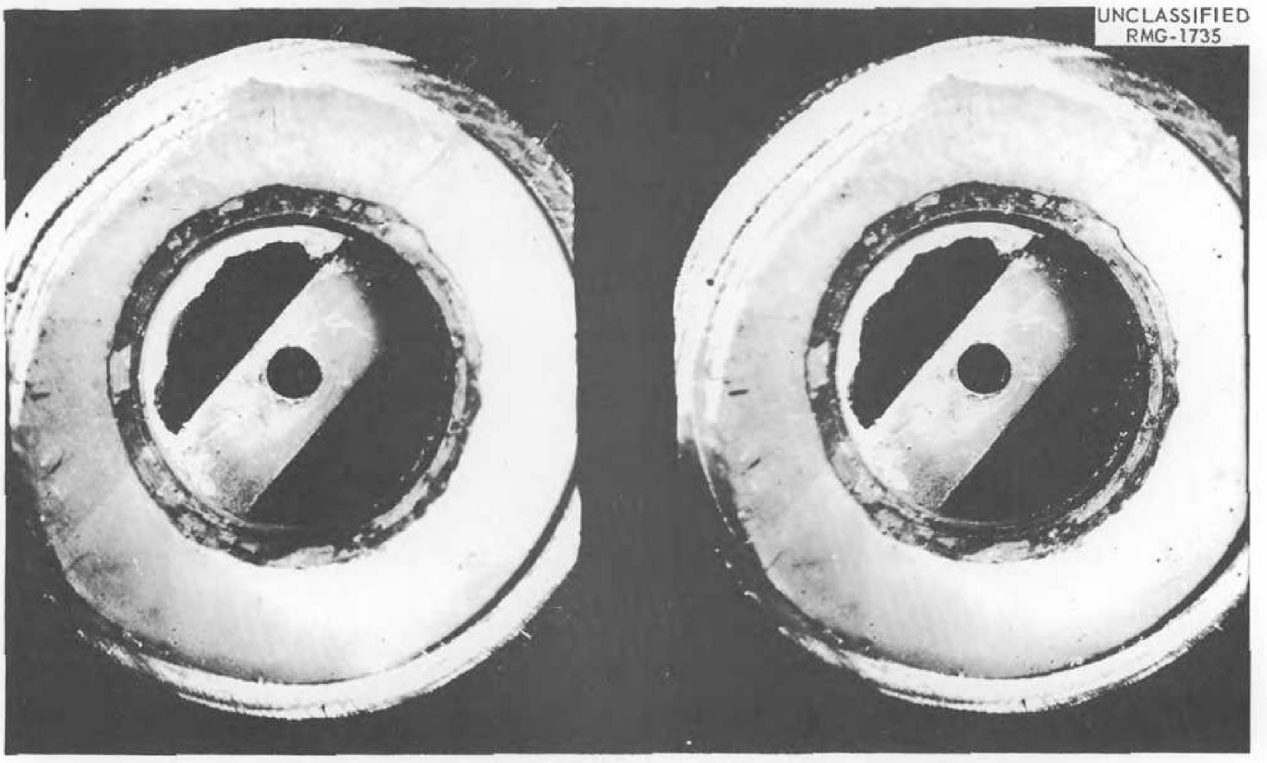

Fig. 28. Seat from Valve U-1. ½X.

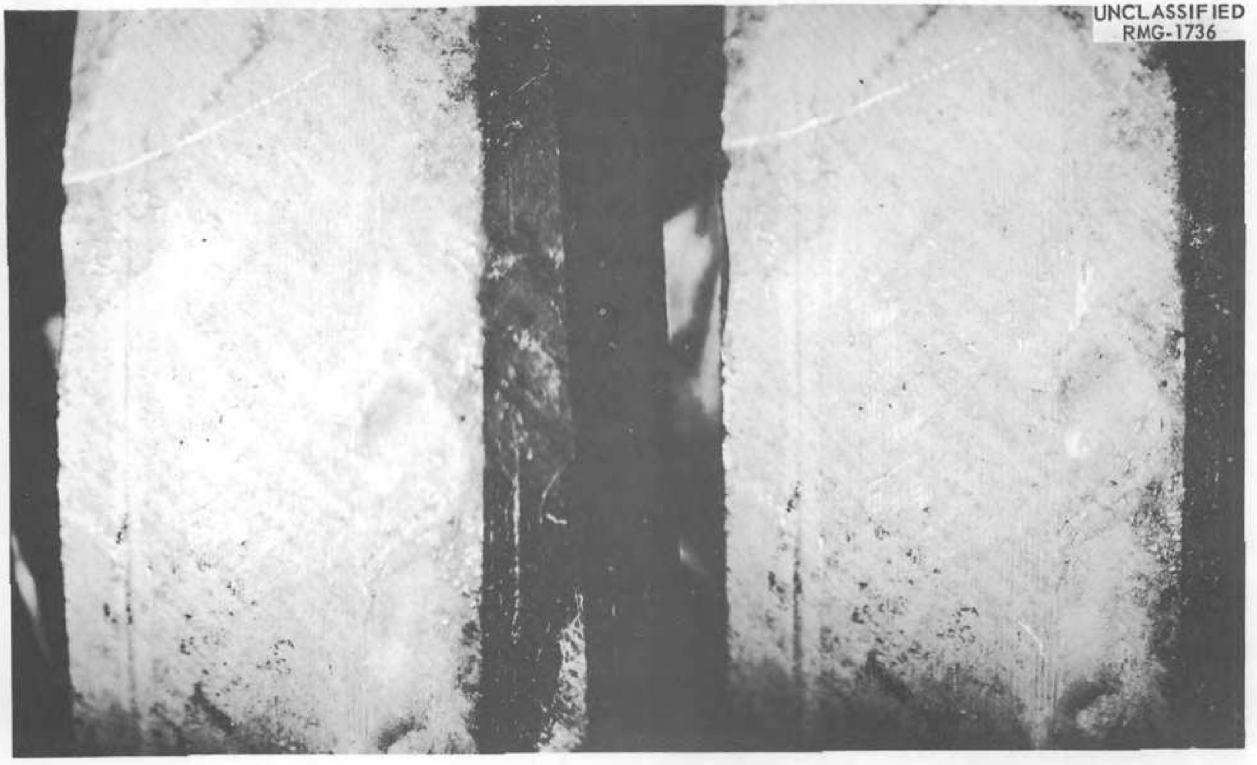

Fig. 29. Plunger from Valve U-23. 5X.

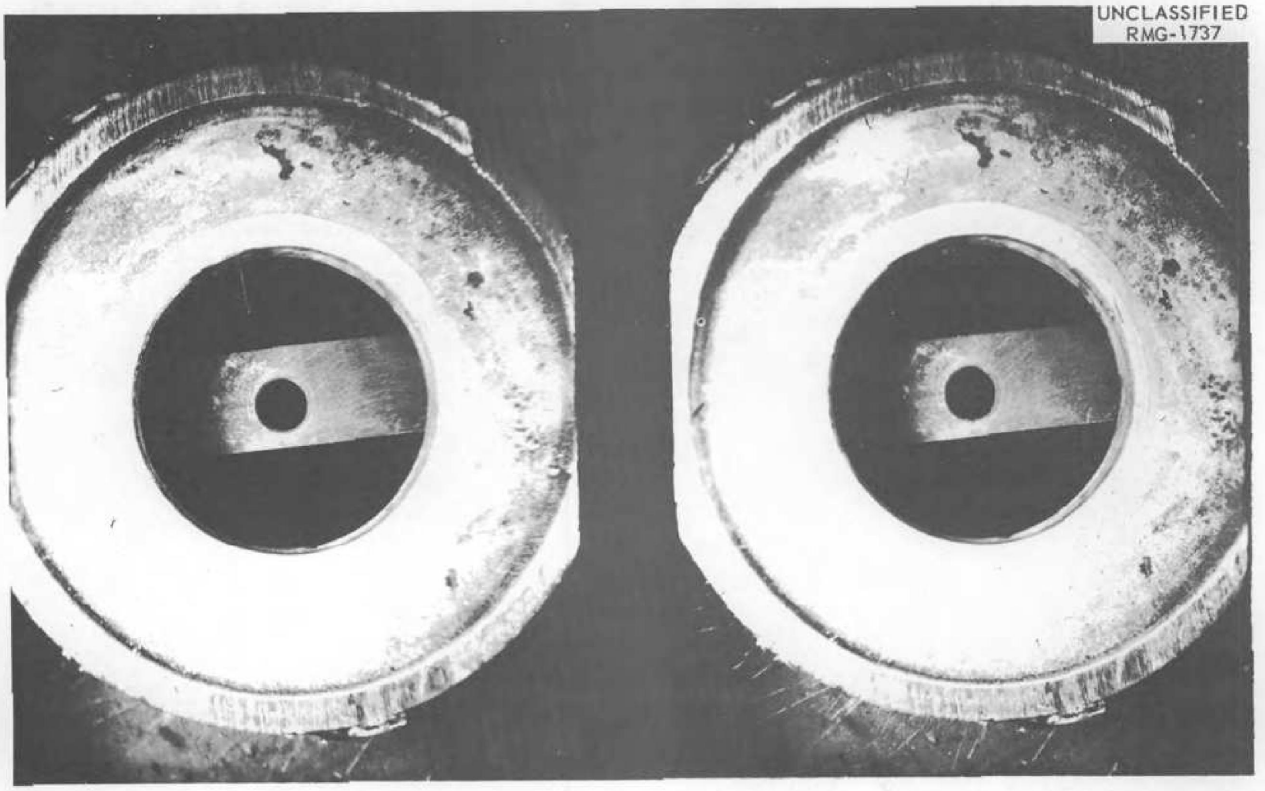

Fig. 30. Seat from Valve U-23. ½X.

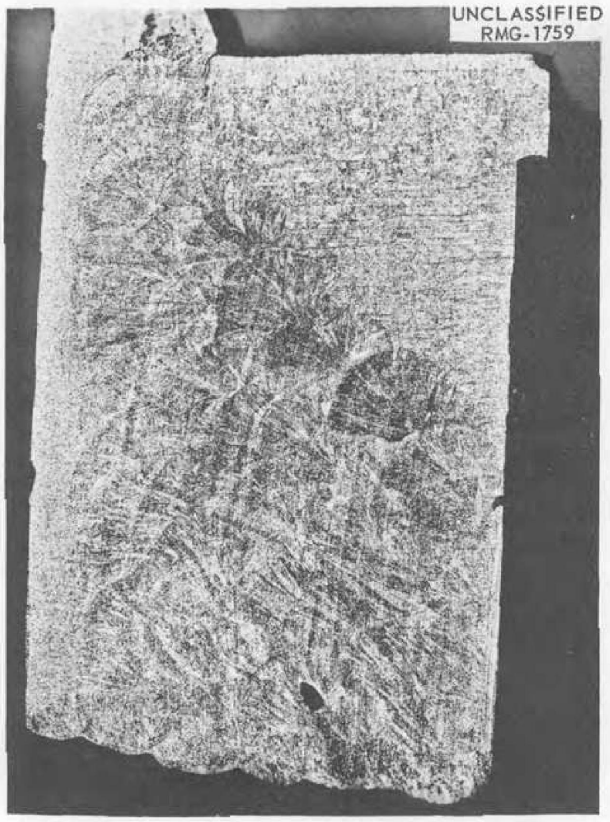

Fig. 31. Section from ARE Pressure Shell, Including a Weld. 5X. (Secret with caption)

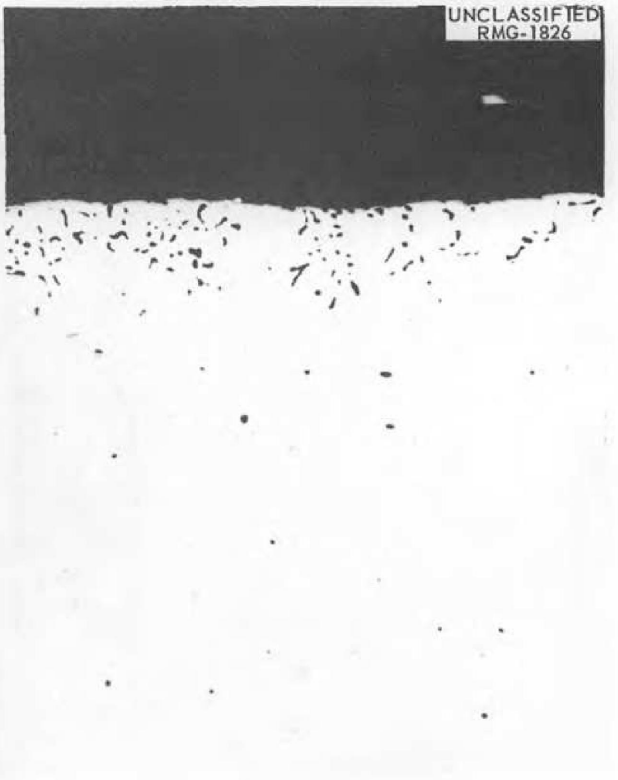

Fig. 32. Section from a Serpentine Bend in a Fuel Tube in the ARE Core, Specimen R7. As polished. 250X. (Secret with caption)

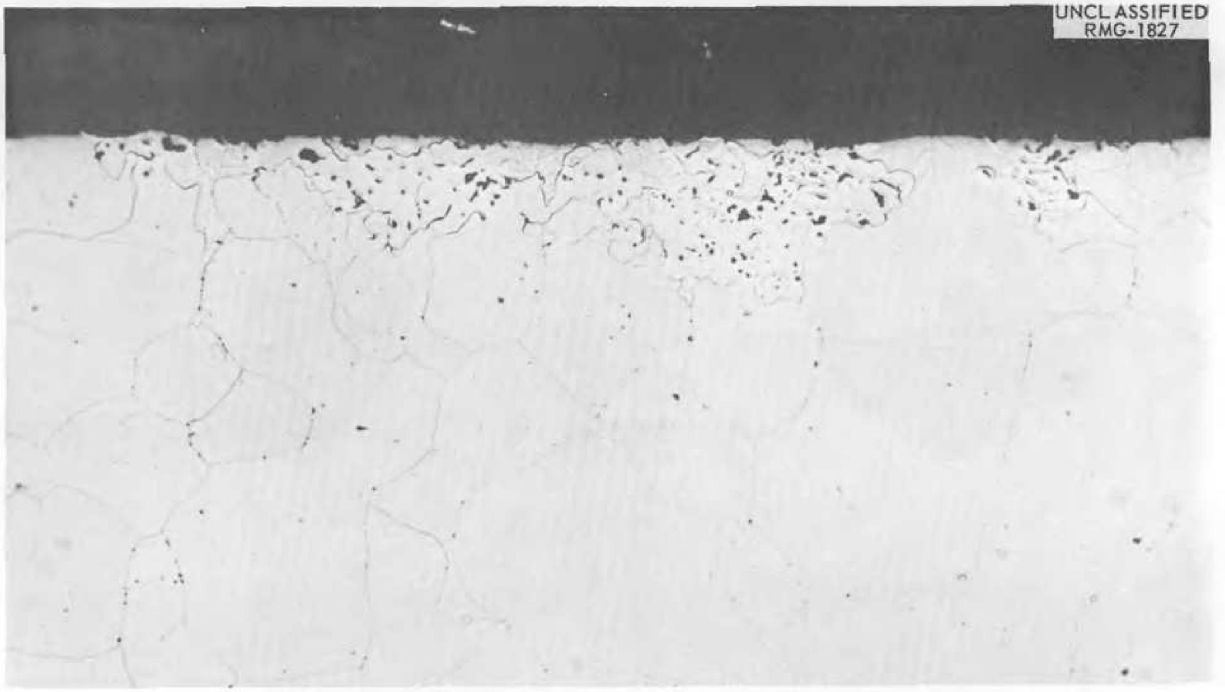

Fig. 33. Section Shown in Fig. 32 After Etching. 250X. (Secret with caption)

Fig. 34. Mass Transfer Deposit on Outer Wall of Fuel Tube at Serpentine Bend. This surface was in contact with sodium during ARE operation. As polished. 500X. ([redacted] with caption)

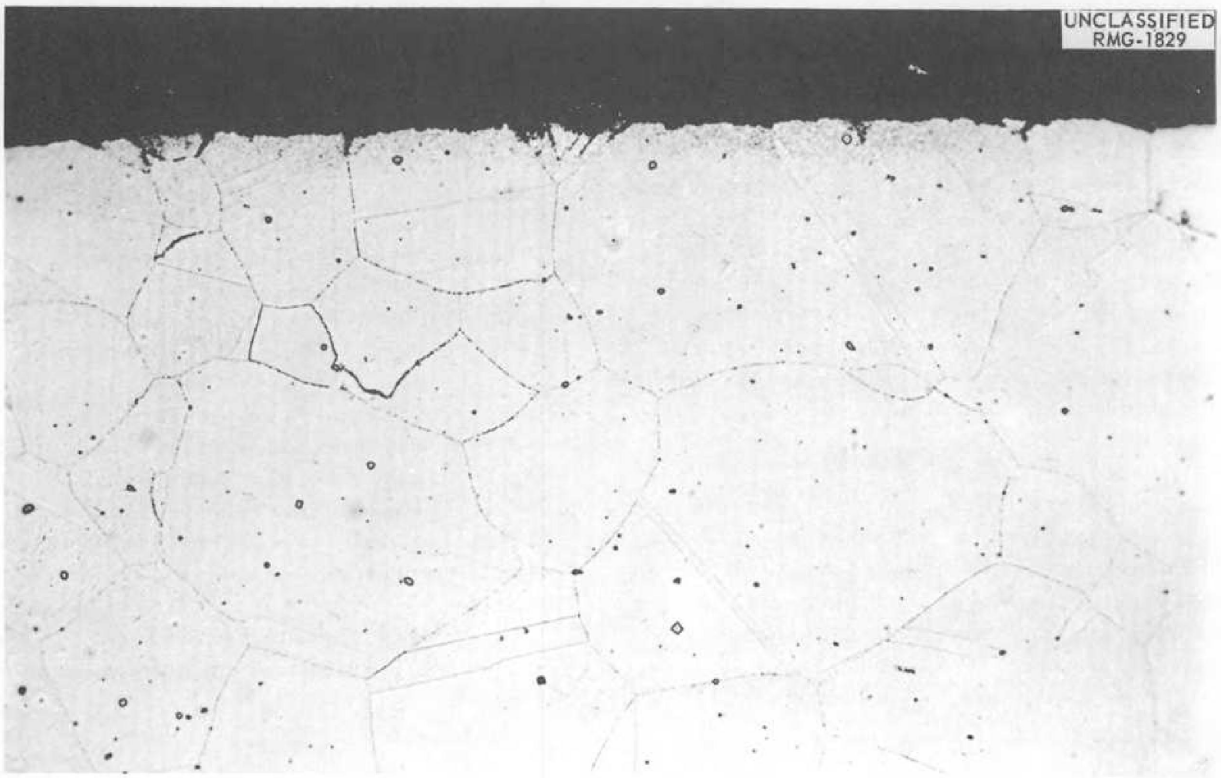

Fig. 35. Section of Specimen R3 Taken from Serpentine Bend in Fuel Tube in ARE Core. Etched. 250X. ([redacted] with caption)

Fig. 36. Another Section of Specimen Shown in Fig. 35. Etched. 250X. ([redacted] with caption)

Specimen F5, which was taken from the inlet of a fuel-to-helium heat exchanger, was cast in epoxy resin so that the fins on the tube would not be damaged during cutting. The fin-to-tube wall junction is shown in Fig. 37. (In the ARE fuel-to-helium heat exchangers the fins were helical strips placed in grooves on the tubes. The strips were not brazed to the tubes.) The interior wall of the tube showed subsurface voids to a depth of 4 mils, as shown in Figs. 38 and 39. Specimen F9, which was taken from the middle of a bend in a fuel-to-helium heat exchanger, also showed subsurface voids to a depth of 4 mils, as shown in Figs. 40 and 41. Specimen S6, the bottom bellows from sodium valve U-23, was also cast in epoxy resin. No cracks in the bellows folds were found, and only a slight roughening of the inside surface was noted, as shown in Fig. 42.

NONMETALLIC MATERIALS

Several nonmetallic specimens taken from the ARE were examined for radiation damage effects.[7] Viscosity measurements were made on the oil specimens, but all other specimens were examined visually and compared with unirradiated specimens. No changes that could be attributed to irradiation were noted in any of the materials. The specimens examined included thermal insulation, electrical insulation, rubber, and oil.

SOLENOID VALVES[8]

The solenoid valves removed from the ARE were examined in simulated service tests and were found to be completely leaktight. The valve seats were found to be in poor condition as a result of dirt and filings that were deposited during the disassembly process. The valves examined are listed below:

Valve No.	Description
U-10	Main fuel pump emergency vent valve
U-19	Main fuel pump vent valve
U-82	Main fuel pump helium supply valve
U-83	Standby fuel pump helium supply valve
U-86	Main fuel pump vent valve
U-89	Standby fuel pump vent valve
U-159	Standby fuel pump emergency vent valve
U-185	Standby fuel pump vent valve
B-124	Off-gas system monitron inlet valve
B-125	Off-gas system monitron inlet valve
B-418	Pit activity-monitor isolation valve

Radiation measurements showed only valves U-10 and U-85 to be contaminated, and the activity was slight.

[7] O. Sisman, *Radiation Damage to ARE Nonmetallic Materials*, ORNL CF-55-7-27 (July 7, 1955).

[8] Results of these examinations were reported by R. G. Affel on May 13, 1955.

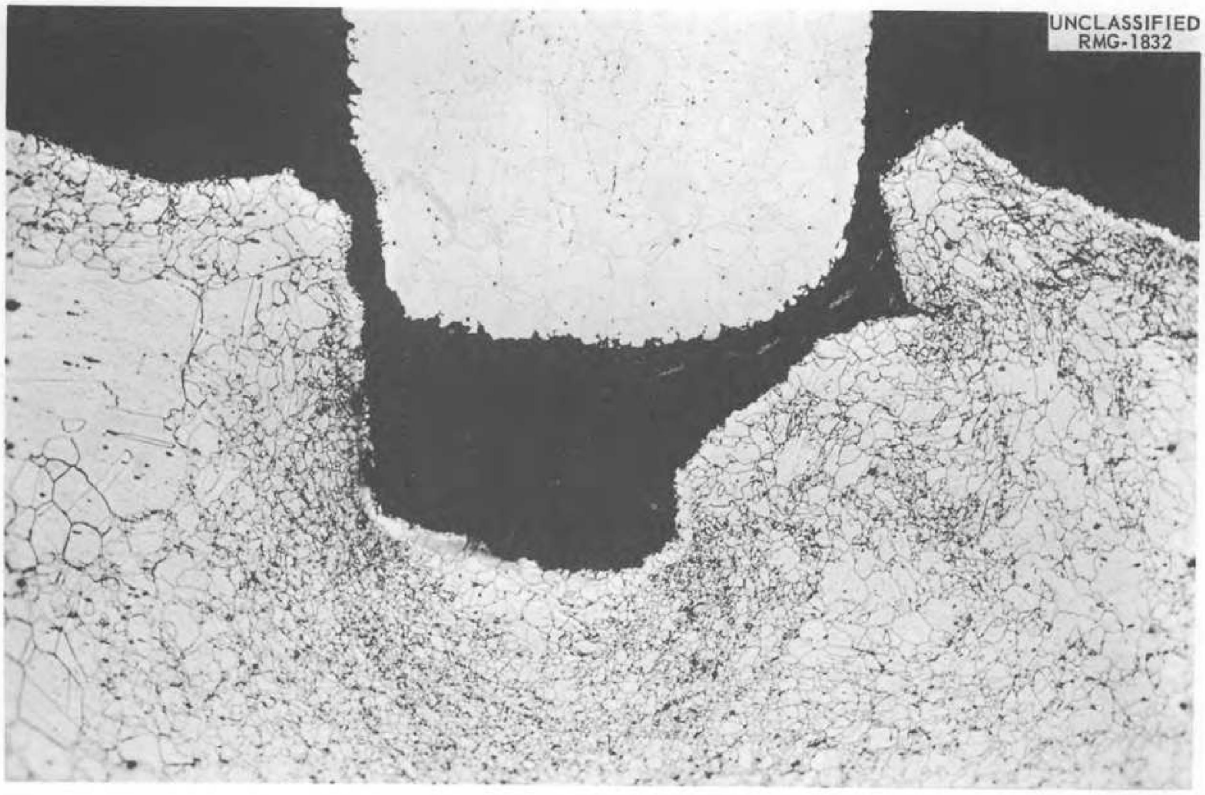

Fig. 37. Section of Specimen F5 Showing Fin-to-Tube Wall Junction in ARE Fuel-to-Helium Heat Exchanger. Fins were wound helically in grooves and were not brazed to the tubes. Etched. 100X. ([redacted] with caption)

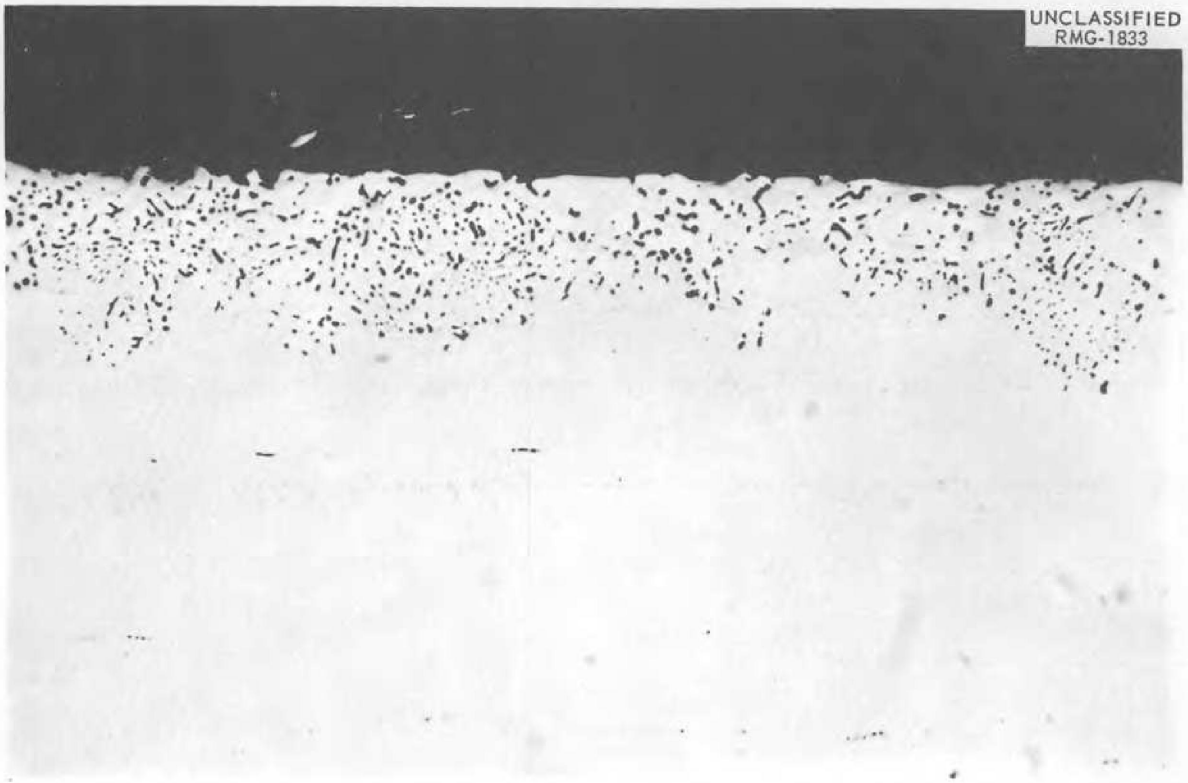

Fig. 38. Interior Wall of Tube Shown in Fig. 37. As polished. 250X. ([redacted] with caption)

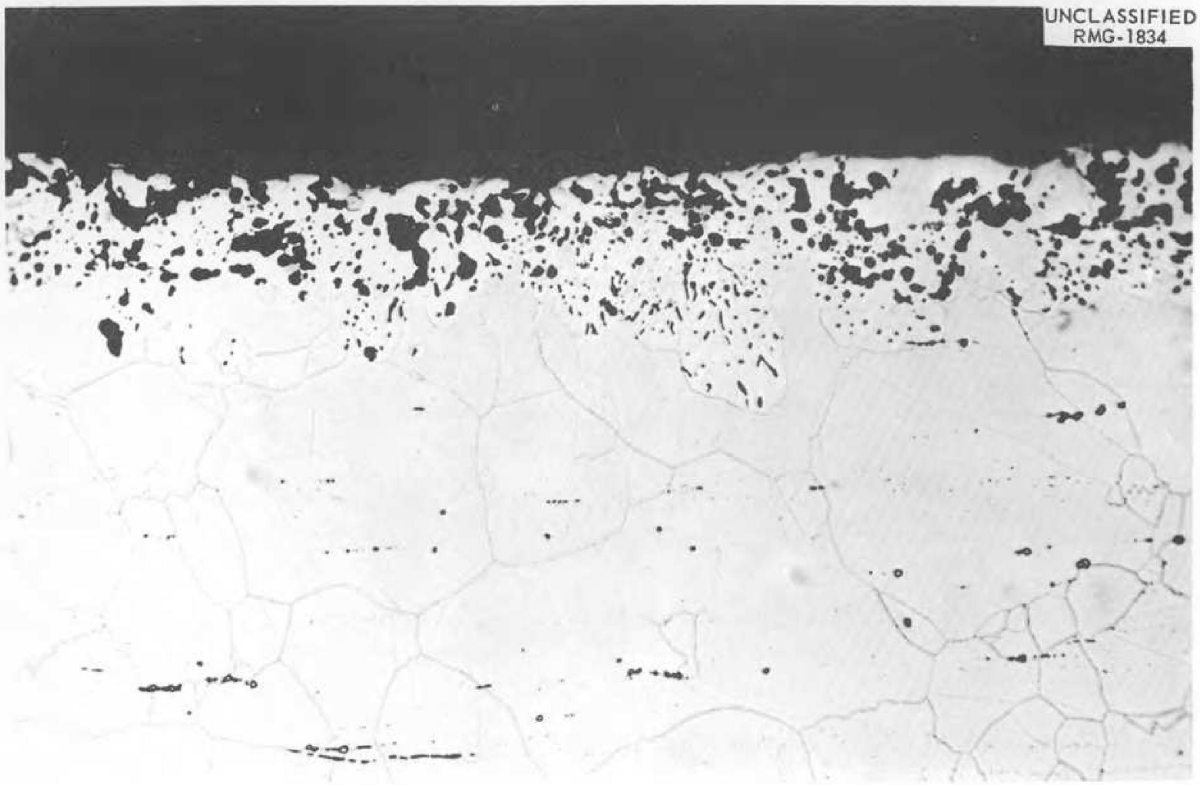

Fig. 39. Section Shown in Fig. 38 After Etching. 250X. ([redacted] with caption)

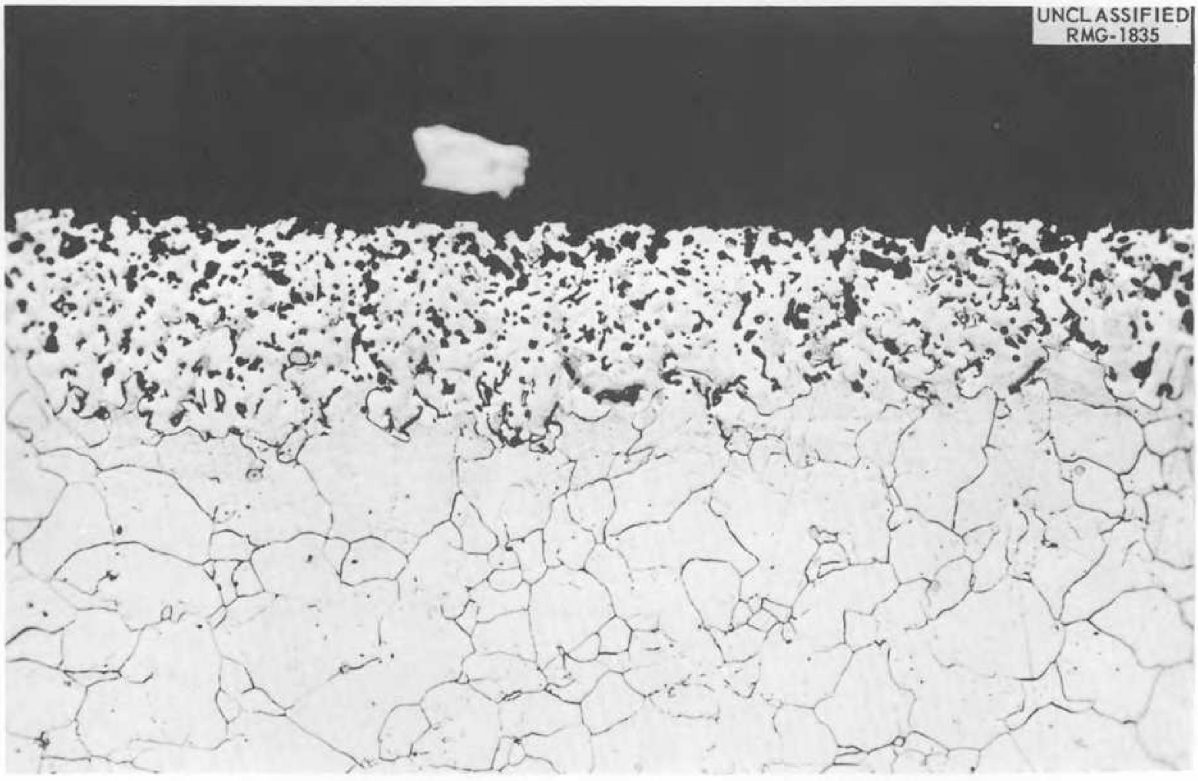

Fig. 40. Section of Specimen F9 Taken from Middle of a Bend in an ARE Fuel-to-Helium Heat Exchanger. As polished. 250X. ([redacted] with caption)

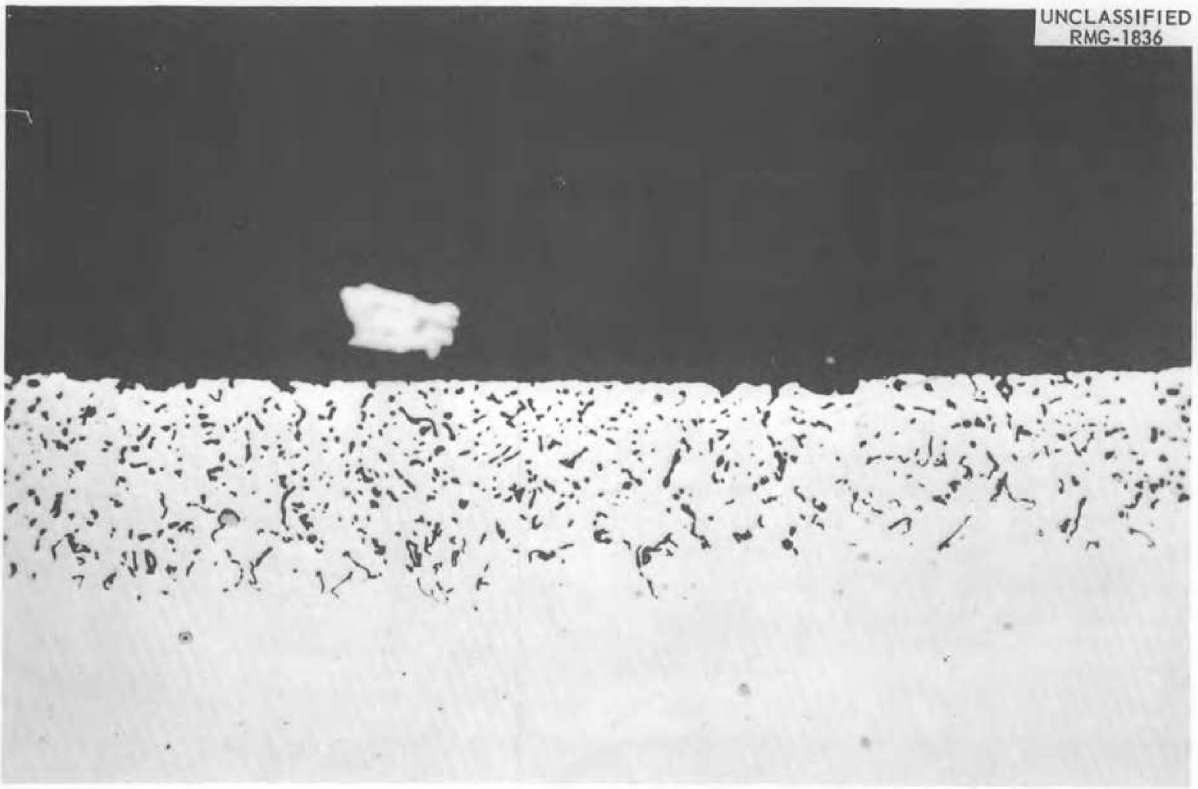

Fig. 41. Section Shown in Fig. 40 After Etching. 250X. ([redacted] with caption)

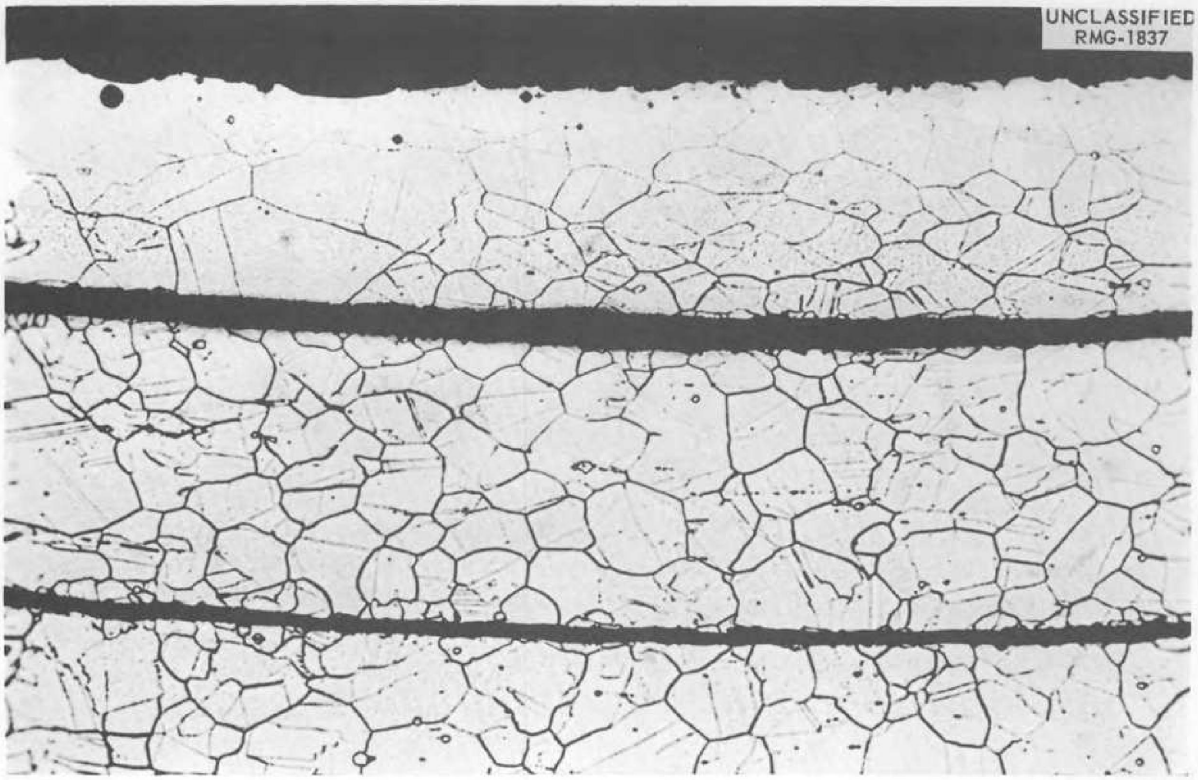

Fig. 42. Section of Specimen S6 Taken from the Bottom Bellows of Sodium Valve U-23. Etched. 250X. ([redacted] with caption)

FUEL PUMP PRESSURE TRANSMITTERS[9]

The main and standby fuel pump pressure transmitters (PXT-6 and PXT-7) were examined for calibration shifts, zero shift, radiation damage, and mechanical damage. The standby pump unit was in excellent condition, but the main pump unit was inoperable because of a failure of the glass-cloth pressure-sensing diaphragm, as shown in Fig. 43. Replacement of the diaphragm showed the unit to be otherwise undamaged. The ruptured diaphragm was not the source of the initial fission-gas leak in the ARE, inasmuch as several checks of the unit at the time the heat exchanger pits were closed indicated that the diaphragm was satisfactory. It is believed, however, that the subsequent rupture of the diaphragm was the source of the leak that developed after shutdown of the fuel system. As stated above, the fission-gas leaks that existed prior to shutdown were probably due to one or more of several potential leaks around the fuel pump.

[9]Results of these examinations were reported by R. G. Affel on February 18, 1957.

SCALE FROM MAIN SODIUM PUMP

An analysis[10] of scale removed from the sodium pump bowl showed the following:

Element	Quantity (wt %)
Ni	61.4
Cr	4.88
Fe	2.20
Si	9.44
Mg	0.01
Ca	0.8
Na	Bal

The silicon found in the scale is believed to be from a residue of the "Conklene" compound used to slush the system before it was filled with sodium. This cleaning mixture is 99% $Na_2O{\cdot}SiO_2{\cdot}5H_2O$.

[10]Spectroscopy Laboratory Report No. 3450.

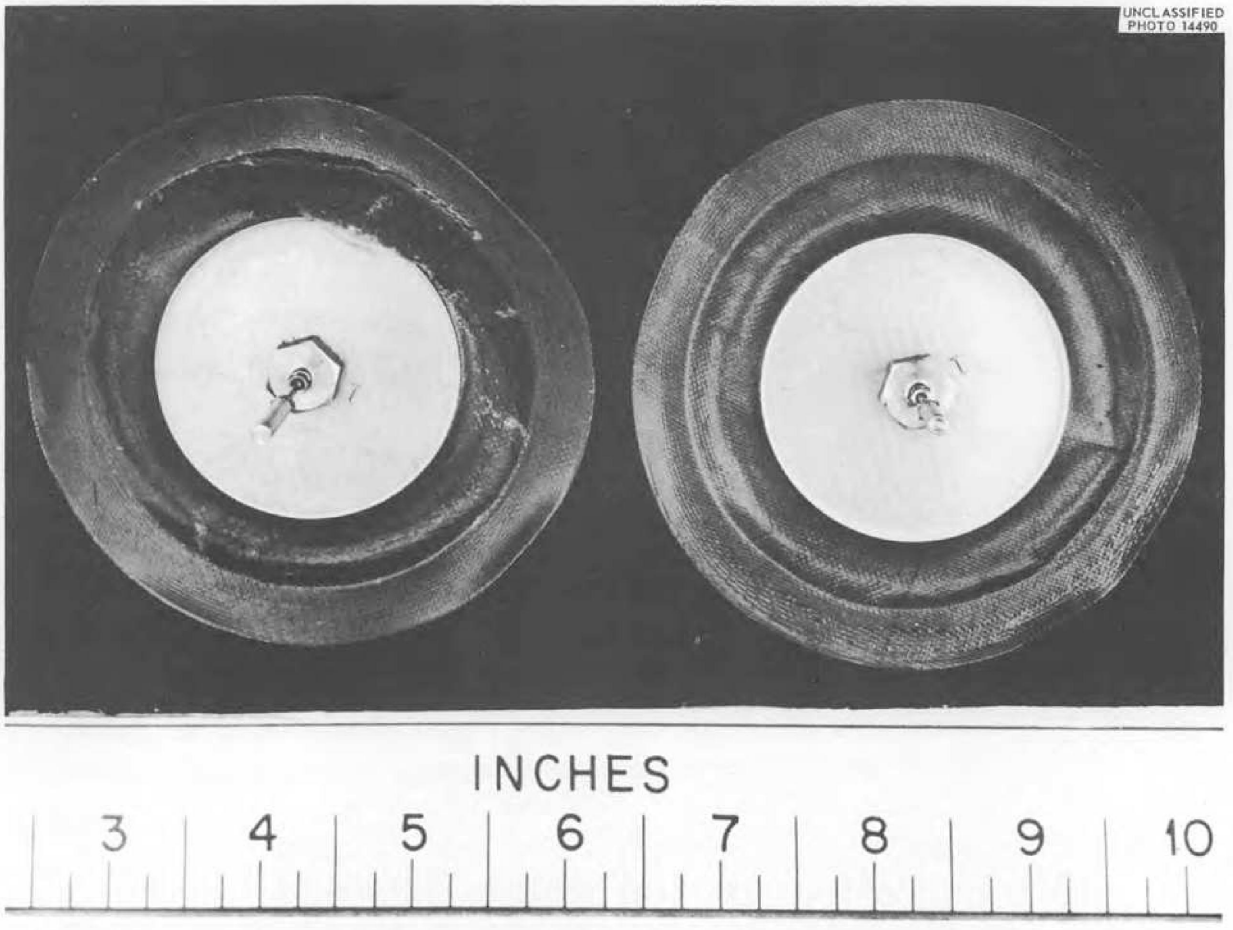

Fig. 43. Main and Standby Fuel Pump Pressure Transmitter Diaphragms. The ruptured diaphragm was removed from the main fuel pump pressure transmitter.

ACTIVATION ANALYSES OF SODIUM SYSTEM SAMPLES

Two sections of pipe containing residual sodium and a sample of sodium were examined for gamma activity after a decay time of 206 days, by using a sodium iodide gamma-ray spectrometer. The data obtained are presented in Table 2.

DISPOSITION OF FISSION-PRODUCT ACTIVITY[11]

Attempts have been made to determine the fate of several fission product nuclides in the ARE. The results leave much to be desired since no adequate plans were made before reactor operation for the study of this problem. Nevertheless, several interesting lines of research are suggested as a consequence of the investigations.

During operation, copious amounts of radioactive gas evolved from the reactor and escaped into the pit from a gas leak in the fuel pump. In an investigation of this gas by P. R. Bell *et al.*,[12] the presence of Xe^{135}, Xe^{138}, and Kr^{88} was revealed by their own or their descendants' gamma radiation. It was also observed that poisoning of the reactor was very much below the level expected if Xe^{135} were efficiently retained.[13] After shutdown of the ARE and dumping of the fuel, preliminary measurements indicated that the radioactivity of the dump tank was far below the expected level.[14] Therefore several samples taken from the ARE after shutdown were examined to determine what radioactive nuclides they contained.

A sample of pipe taken from the intake end of the emergency off-gas line for draining the pit was examined in a gamma spectrometer. The only activity clearly identified was due to Ru^{103}, which is characterized by a 0.50-Mev gamma ray and a 40-day half life. Chemical tests showed that this material was probably on the outside of the pipe.

A similar study was made of three small samples cut from the reactor fuel inlet line (line 120). Gamma spectrometry showed Ru^{103}, Ru^{106}, and Zr^{95}-Nb^{95} as the only identifiable activities. The following disintegration rates were observed at 62 days after shutdown of the ARE:

$$Ru^{103}\ (1.3 \pm 0.3) \times 10^{9}\ \text{dis/min/cm}^2\ ,$$

$$Zr^{95}\text{-}Nb^{95}\ (1.3 \pm 0.2) \times 10^{8}\ \text{dis/min/cm}^2\ .$$

[11] Material abstracted from a report by M. T. Robinson, S. A. Reynolds, and H. W. Wright, *The Fate of Certain Fission Products in the ARE*, ORNL CF-55-2-36 (Feb. 7, 1955).

[12] P. R. Bell *et al.*, *Measurement of Gamma Radiation from Off-Gases of ARE*, undocumented memorandum, Dec. 22, 1954.

[13] J. L. Meem and W. B. Cottrell, *Preliminary Report — Operation of the Aircraft Reactor Experiment*, ORNL CF-54-11-188 (Nov. 30, 1954).

[14] W. K. Ergen, personal communication.

Table 2. Gamma Activity in ARE Sodium System Samples

Sample Number	Description of Sample	Activity	Approximate Maximum Disintegrations per min per cm^3 of Na
1	10-in. section of 2-in. pipe to sodium pump containing about 10 cm^3 of sodium	Mn^{54}	7×10^3
		Zn^{65}	1×10^4
		Co^{60}	4×10^2
2	14-in. section of 1½-in. pipe to heat exchanger containing about 5 cm^3 of sodium	Mn^{54}	7×10^3
		Zn^{65}, Na^{22}, Co^{60} *	1×10^4
3	Sodium sample of about 50 cm^3	Na^{22}	2×10^2
		Mn^{54}	1×10^2
		Zn^{65}	2×10^2

*Activity indistinguishable; may be combination of one or more.

The expected ratio of disintegration rates for unsegregated fission products at this age is

$$Ru^{103}/Zr^{95}\text{-}Nb^{95} = 0.8 ,$$

whereas a value 10 was found. It is clear that some "plating" of ruthenium onto the walls of the fuel line had occurred.

The decay of radioactivity of a sample of solid ARE fuel taken as liquid from the dump tank was followed through the period from 31 to 81 days after ARE shutdown. The total activity of the sample was observed with the large ion chamber of the Radioisotopes Department. These results were combined with gamma spectra to yield both total photon emission rates and differential decay data. By observing gamma energy and half life, the following nuclides were identified:

Nuclide	Half Life	Gamma Energy (Mev)
Ba^{140}-La^{140}	13 days[15]	2.5, 1.60, 0.8, 0.5, 0.33
Ce^{141}	28 days	0.14
Zr^{95}-Nb^{95}	Very long[16]	0.8

No Ru^{103} or I^{131} was detected. It seems likely that the amount of the ruthenium present was relatively small. Because of the long delay before decay measurements were started, the detection of I^{131} was made difficult; however, the differential decay data were analyzed to estimate the specific activity of the sample:

Time After ARE Shutdown (days)	Specific Activity (curies/kg)	Average Gamma Energy (Mev)
31	16	0.96
79	3.5	0.73

At 79 days, the dose rate from the sample (0.0074 g) was measured with a calibrated "cutie pie." If the dose rate is assumed to be given by r/hr at 1 ft = $n\,CE$, where C is the number of curies in the sample and E is its energy, the results give $n = 8$. Since a value of 6 or 7 is usually assumed for n, it is apparent that the measurements of the specific activity of the fuel are essentially in agreement with the values obtained on small samples with a "cutie pie."

Another sample of fuel was used for various chemical and radiochemical analyses. The chemical results are given in Table 3. The uranium analysis was lower after ARE operation because of the use of barren fluorides to flush out the fuel system. The iron presumably resulted primarily from the drilling operation used to sample the salt.[17] Aliquots of the sample from the dump tank were analyzed for Sr^{89}, Zr^{95}, Ru^{103}, Cs^{136}, Cs^{137}, Ce^{141}, and La^{140}; all except lanthanum and the two cesium isotopes were determined by conventional radiochemical methods.

Table 3. Results of Chemical Analyses of ARE Fuel

Fuel Component	Before ARE High Power Operation*	In Dump Tank
U, wt %	13.59	5.97
Fe, ppm	25	140
Cr, ppm	420	250**
Ni, ppm	25	70**

*The U analysis was taken from the ARE Nuclear Log Book; the other results were obtained from W. R. Grimes, Jan. 4, 1955.
**Corrected to basis of undiluted fuel by multiplying by 13.59/5.97.

In order to estimate the efficiency of retention of typical fission products, the radiochemical analyses reported on ARE fuel were compared with similar results obtained on a sample of solid fluoride fuel (NaF-ZrF_4-UF_4, 8.5 wt % U) irradiated in hole 12 of the ORNL Graphite Reactor. The irradiation time approximately matched the high-power operating time of the ARE. The following ratios were then obtained between analyses reported for the ARE fuel and those reported for the standard sample (designated as MR-1):

[15] This mixture of nuclides was at transient steady state and decayed with the Ba^{140} half life. Most of the gamma rays were due to La^{140}.

[16] This mixture was still approaching equilibrium; the apparent half life was too long to measure with limited precision of the available equipment. Both isobars emit gamma rays at about 0.8 Mev.

[17] W. E. Browning and H. L. Hemphill, *Solid-State Semiann. Prog. Rep. Feb. 28, 1955*, ORNL-1851, p 18.

Nuclide	Ratio (ARE)/(MR-1)
Sr^{89}	6.1
Zr^{95}	3.3
Ru^{103}	1.5×10^{-4}
La^{140}	14
Ce^{141}	15

The ratios were corrected to the ends of the respective irradiations. Apparently Sr^{89}, a descendant of 2.6-min Kr^{89}, was reduced by a factor of about 2 below the expected level in the ARE, presumably due to partial escape of its noble gas ancestor. The low value of Zr^{95} has not been explained. It is of interest to point out that the amount of Zr^{95} reported radiochemically was in good agreement with the amount deduced from the decay data on solid fuel. It must also be mentioned that the amount of Zr^{95}-Nb^{95} found on the walls of the fuel circuit was apparently negligible compared with the amount found in the fuel (about 1 part in 2000 if the wall activity was taken to be uniform over the entire surface). The very low Ru^{103} value in the ARE sample may possibly have been due to a faulty analysis, but it was felt to be real. The agreement was fair between radiochemical results from the ARE dump-tank material for Ce^{141} and La^{140} and the results deduced from decay of the solid sample of ARE dump-tank material. The cerium and lanthanum results in the experiment and in the decay study appear to be in reasonable agreement with what would be expected from the ARE power history.

Gamma spectrometry was used to determine the ratio of amounts of Cs^{136} and Cs^{137} in the two samples. The former isotope is "shielded"; that is, it must be formed directly in fission since Xe^{136} is stable and has a very low thermal-neutron absorption cross section (0.15 barns). On the other hand, Cs^{137} is the daughter of 3.9-min Xe^{137}. Thus, a difference between the ratios of the amounts of these isotopes in the two samples was a measure of the escape from the ARE fuel of Xe^{137}. The results indicate that not over 20% of the Xe^{137} escaped from the fuel, and that, possibly, none did.

It has become customary[18,19] to describe the release of xenon from the fluoride fuels in terms of a quantity λ_p, defined by

$$\text{Rate of Xe escape} = \lambda_p \times \text{amount of Xe in fuel.}$$

From the ARE poisoning data it is roughly estimated that for Xe^{135}

$$\lambda_p = 5 \times 10^{-4}\ \text{sec}^{-1} .$$

This value is consistent with the observed behavior of Cs^{137} also. From the Sr^{89} data reported above it appears that krypton isotopes have larger values of λ_p than do xenon isotopes.

Several lines of investigation are suggested by the results reported here. In particular, the effects of ruthenium on the physical properties of Inconel and on corrosion by the fluoride fuels should be studied. If all Ru^{103} is removed from the fuel by the walls, and if the ruthenium "plate" is uniform over the entire reactor, the approximate rate of deposition of Ru^{103} is 0.7 (P/A) μ/hr, where P is the total reactor power in Mw and A is the surface area in cm^2.

[18] J. L. Meem, *The Xenon Problem in the ART*, ORNL CF-54-5-1 (May 3, 1954).

[19] M. T. Robinson, *Release on Xenon from Fluoride Fuels: Proposal for an Experimental Program*, ORNL CF-54-6-4 (June 2, 1954).

DISPOSAL OF DISASSEMBLED SYSTEM

The radioactive disassembly debris from the ARE will eventually be treated for recovery of unknown amounts of fuel believed to be adhering to these materials. The reactor and the BeO blocks have been buried, as described above. The fuel is awaiting reprocessing in the fused-salt fluoride-volatility process being developed.

The various samples not yet examined will be taken up as time permits and disposed of as appropriate at that time. Reports of subsequent examination will appear in periodic progress reports.

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