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SOME ASPECTS OF THE WTR & SL-I ACCIDENTS

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ABSTRACT

Some Aspects of the WTR and SL-1 Accidents

The Westinghouse Testing Reactor (WTR), a privately owned 60 megawatt tank type reactor, underwent a fuel element failure on April 3, 1960. A meltdown of one fuel element occurred causing the spread of fission products through the reactor cooling system. There were no casualties or overexposures. The cause of the accident could not be established beyond a reasonable doubt, but the possible cause appeared to be a cladding failure at a bonding defect.

The Stationary Low Power Reactor No. 1 (SL-1), a 3 megawatt prototype reactor, underwent a nuclear excursion at the U. S. Atomic Energy Commission's National Reactor Testing Station (NRTS), Idaho, on January 3, 1961. The three military operators on duty at the time received fatal injuries and the core experienced severe damage. Large amounts of radioactivity were released inside the reactor building; however, release of radioactivity from the building to the atmosphere was slight. This was the first fatal reactor accident in the history of reactor operation in the United States. Prior to the accident, the reactor had operated for 931 megawatt days, approximately 40 percent of its core life.

Primary efforts subsequent to the SL-1 accident consisted of removal of the victims from the reactor building, determination of the nuclear status of the reactor, and analysis of the cause of the accident, including dismantling of the facility.

Since the cause of the SL-1 accident was not known, work on the dismantling and decontamination of the reactor building had to proceed slowly lest some important evidence might be overlooked. The high radiation levels inside the reactor building also played an important part in slowing up the recovery operations. By the end of November 1961, the pressure vessel with the SL-1 core had been removed from the reactor building and transported 40 miles to a large hot cell on the testing station previously used to disassemble large experimental reactors. In the hot cell a more detailed examination of the disarranged core proceeded and is still underway.

Table of Contents

Some Aspects of the WTR and SL-1 Accidents

I. Introduction

Recent reactor accidents in the United States at the Westinghouse Testing Reactor (WTR) and the Stationary Low Power Reactor No. 1 (SL-1) are discussed in this report. The WTR accident occurred on April 3, 1960 and the SL-1 accident occurred on January 3, 1961. This paper briefly describes the facilities and the events relevant to the accidents, with a brief discussion and analysis of the damage incurred. Some of the pertinent implications as to reactor safety are discussed.

The SL-1 was operated by Combustion Engineering, Incorporated for the U. S. Atomic Energy Commission at the AEC National Reactor Testing Station, Idaho. The WTR is owned by the Westinghouse Electric Corporation, Pennsylvania.

The two accidents described in this report are of greatly different proportions. The SL-1 accident was much more serious than the WTR accident. The three crew members on duty at the SL-1 were fatally injured and the recovery of the SL-1 reactor was economically infeasible from a program standpoint. On the other hand, no one received an overexposure of radiation as a result of the accident at the WTR and the facility was readily returned to operation.* Both accidents, of course, were thoroughly investigated. The SL-1 investigation is still continuing and probably will continue until midsummer.

*The Westinghouse Electric Corporation announced in March 1962 that it was terminating the operation of the WTR because of lack of customer demand.

The aspects of the accidents discussed in this report, selected as those of direct interest to the nuclear power industry, are excerpted from reports, already published, by various committees, boards and other persons directly associated with investigation of these accidents.

II. Westinghouse Testing Reactor

A. Background

The Westinghouse Testing Reactor (WTR) is located on an 830 acre tract, approximately 20 miles southeast of Greater Pittsburgh. The surrounding land area usage is predominantly farming (Figure 1).

The WTR is a pressurized, tank type, light water-cooled and moderated reactor. The primary function of the WTR is to test reactor materials and components. The reactor was designed for 60 megawatt (MW) power operation, although it was originally licensed and operated at 20 MW. The primary coolant system is a recirculating loop in which water flows from the reactor vessel to a surge tank from which it is pumped through heat exchangers to an elevated head tank, 250 feet above the ground. From the head tank, water flows by gravity back to the reactor vessel.

Each fuel assembly has 200 grams of highly enriched uranium fuel as aluminum-uranium alloy in the walls of three long concentric cylinders around a central aluminum mandrel tube in which small canned specimens can be irradiated. The uranium-aluminum alloy is aluminum clad: cladding thickness is 36 mils; the meat, 52 mils. The fuel tubes or cylinders are 44 inches long and the outside diameter of the fuel assembly is 2.5 inches. Orifices at both ends distribute the coolant flow through the channels within the assembly and provide some of the static pressure required on the fuel assemblies to prevent boiling at the hot spots.

Figure 2 shows the relationship of the reactor core to the reactor vessel. A plan view of the core barrel is shown in Figure 3. The inner hexagon contains fuel elements, test loops, and control rods, while the outer segments are used for experimental purposes.

At the time of the accident, the reactor had been operated up to 45 MW and studies were underway to determine the effect of incipient boiling on reactor stability in anticipation of 60 MW operation. As an initial part of the experiment, tests were conducted to study the effects of bubbling helium through the core. When the accident occurred, a program was underway to operate the reactor at incipient boiling by reduction in the primary coolant flow, observing the formation of steam bubbles using the same recorders previously tested during the helium bubbling experiment.

B. WTR Accident

On April 3, 1960, the reactor had been operating at a steady state at 40 MW with a primary coolant flow of 15,000 gallons per minute. In preparation for carrying out the reduced flow experiment, reactor power was reduced to 30 MW and appropriate reactor safety circuits were reset to permit reduction of flow to 5,000 gallons per minute. During the experiment, it was intended to raise the power level gradually, with continuous monitoring of the bubble measuring recorders, until a power level of 45 MW was reached or until boiling was observed.

The reactor flow was reduced gradually to 5,250 gallons per minute. At 8:20 p.m. the reactor power was increased to 37 MW (calculated). A recording of power levels observed is shown on Figure 4. At 8:33 p.m. the power demand was adjusted to raise the power to 40 MW. At 8:35 p.m. the power level began to drop rapidly, going down to 17 MW over a period of about two minutes for no apparent reason. During this period, the control rod on automatic control withdrew to its upper limit. The other control rods were withdrawn on manual control in order to maintain power, and on reaching approximately 17 MW, power level started to increase on approximately a 60-second period or greater. The power returned to approximately 38 MW. At 8:40 p.m. the radiation detector monitoring the demineralized water supply alarmed. This was followed by further alarms from other radiation monitors within a minute. Power was lowered to 15 MW and at 8:44 p.m. the reactor was scrammed manually as radiation levels continued to rise.

Immediately following reactor scram, personnel on the reactor top were evacuated to the control room. As radiation levels continued to rise on all monitoring channels, a general evacuation was begun to a remote location on the site. Certain operating and health physics personnel remained for a short time to secure the plant and to continue survey work, but were evacuated due to the high radiation levels.

The reactor primary coolant system was left in operation and one of the high pressure test loops set for cooldown. Activation of the stack gas and particulate monitors (located in the Process Building) by external radiation caused automatic recirculation of the vapor container ventilation system. The surge tank vent blower, which sweeps air from the surge tank to the top of the head tank where it is discharged, was left in operation to prevent possible blowback of fission product material into the process area. To prevent further releases of material, personnel returned to the plant to shut down this blower.

The initial radiation survey indicated that gross fission product contamination of the primary coolant system had occurred. The highest reading of 40 roentgen/hr was taken at the head tank downcomer at ground level.

C. WTR Recovery Operations

The major effort was to determine the cause of the failure, get the plant decontaminated and the reactor back into operation. Such problems as water storage and radiation protection occupied a considerable effort and the solution to these type problems governed the pace of the main activities.

By April 9, decontamination efforts had proceeded sufficiently so that the reactor head was raised one foot for examination and radiation survey. Since the radiation levels close to the head were approximately one roentgen/hr, the head was replaced pending construction of shields and to prepare washing and decontamination equipment.
A system of car-wash brushes was hooked up for continuous scrubbing during the raising of the head (Figure 5). A three-inch thick iron shielding platform was constructed to permit visual observation of

the core and to begin unloading the core. On April 11, the head was removed. Figure 6 shows a photograph of the core taken on this date. No visible damage was apparent at this time.

Fuel unloading then began with elements being removed first from the outside of the core, working towards the middle. Some elements stuck slightly and were removed by a hoist with a 350 pound force limitation. Following removal of all fuel elements but one, which could not be dislodged within the above force limitation, all the control rods and their fuel element followers were removed.

Upon examination, all fuel elements thus removed from the core appeared discolored but without apparent physical damage. The stuck element was finally removed by a 500 pound force and only the upper third of the element came loose (Figure 7). The bottom end of the shroud tube appeared to be solidly plugged. Finally by using a specially fabricated core drill, the final portion of the damaged element was removed. A visual examination of the shroud holes and a later check with a sizing tool indicated that the core structure had not been damaged.

D. Accident Analysis

The power reduction, shown in Figure 4 is believed to have occurred as a result of a decrease in reactivity caused by the fuel element failure meltdown and subsequent blockage of the coolant channels. Production of steam and bulk boiling in the blocked element voided the water channels. It was calculated that reactivity loss by voiding the water channels and possibly by the loss of a small amount of fuel is consistent with the reactivity change which caused the power loss from 38 MW to 17 MW.

A close examination of the trace made by a boiling detector Brush recorder, being used during the reduced-flow experiment in progress at the time of the accident, confirmed that the element failed prior to the power loss.

Visual observations of the failed fuel element disclosed some evidence of poor bonding; hence a program was instituted to reinspect the unused cold fuel elements on hand. A mechanical inspection of these elements disclosed many small deviations from specifications and a few elements with serious bows in the tubes or with visible blisters. An ultrasonic inspection revealed dozens of imperfections (Figure 8). The defects ranged in size from a few thousandths of an inch to greater than 1 inch in diameter.

An experimental check was made to determine whether defects grew upon temperature cycling. No significant change in size or number of the defects was noted.

Thermal hydraulic analysis, using pertinent heat transfer information (reactor power at 38 MW; coolant flow rate, 5250 gallons per minute; etc.), applicable to the reactor when the fuel element failed, revealed that a burnout type failure of a good element did not occur. The heat transfer calculations indicated, however, a bonding defect greater than 1/2 inch in diameter could account for the fuel element failure.

Investigation by the AEC's Division of Licensing and Regulation indicated that either or both of two factors played a major role in the WTR accident: (1) inadequate coolant flow under conditions existing at the time, or (2) defective metallurgical bonding in the fuel element. Further detailed calculations by the WTR staff indicated that the cause of the failure could not be established beyond a reasonable doubt.

E. Conclusions

A fuel element failure at the WTR on April 3, 1960, resulted in the spread of gross fission products throughout the reactor primary coolant system. The cause of the failure was not established beyond reasonable doubt, but it may be assumed that a normal fuel element operating under the conditions at the time of the accident would not have failed. A strong possibility exists that the failed element was not normal and possibly had a defect greater than 1/2 inch in diameter.

The rapid and spontaneous decrease in power was not recognized by the reactor operator or supervisor as being abnormal. The recovery of the specified power was not consistent with safety of operations. Apparently the fuel element failed prior to the power loss and, therefore, the following increase in power by direct withdrawal of the control rods only aggravated the situation.

Subsequent to the accident, approximately 100 cold fuel elements from the same batch as the ruptured fuel element were reinspected. The results of the reinspection disclosed dozens of defects.

Rigorous inspections cannot be done without adding costs to the fabrication of fuel elements; however, these additional costs are rather insignificant when compared to accident recovery costs.

III. Stationary Low Power Reactor No. 1 (SL-1)

A. Background

The SL-1 was a direct cycle, natural recirculation boiling water reactor designed for 3000 KW gross thermal capacity and was capable of producing 200 KW net of electricity and 1.3 million BTU per hour for space heat. Work on this plant started in 1955 in response to a Department of Defense request for a small nuclear power plant. The requirement was based on the need to develop such a plant for future use at remote military installations.

Site work began in the fall of 1956; plant construction started in 1957 and initial criticality was achieved in August 1958.

Argonne National Laboratory, the prime contractor, performed the initial criticality and startup tests and successfully completed a 500-hour, full power plant performance test in December 1958. In February 1959, the permanent operator (Combustion Engineering, Inc.) assumed responsibility for the operation of the SL-1. After startup the SL-1 was used to furnish operating experience, develop plant performance characteristics, obtain core burnup data, train military personnel in plant maintenance and operation, and test improved components planned for use in subsequent reactors of this type.

The SL-1 site is located at the National Reactor Testing Station about 3/4 mile north of Route 20 (Figure 9). Site facilities consisted of the reactor building, an adjoining support building which contained the control room, and miscellaneous service buildings (Figure 10).

The majority of the plant equipment was located in a cylindrical steel reactor building 38-1/2 feet in diameter having an over-all height of 48 feet. This building was made of steel plate, most of which had a thickness of 1/4 inch. Access to the building was provided by ordinary doors. The building was not a pressure-type containment shell as would have been used for reactors located in populated areas. Nevertheless, the building was able to contain most of the radioactive particles released by the explosion.

The building was erected on dummy support piles to simulate the type of construction that would be used in the Arctic, in the permafrost area, where the whole structure would be supported by piles (Figure 11). The reactor vessel, fuel storage wells, and demineralizers were located in the lower third of the building and shielded with gravel. Gravel was used because this was a material that was readily available at the remote sites where location of such reactors was planned. A recirculating, air-cooler condenser was located in the upper third of the building. The middle third of the building contained the turbine generator, feedwater equipment, and shielding blocks located around the reactor pressure vessel head. These shielding blocks were movable by an overhead crane, permitting access to the pressure vessel head and control rod drive mechanisms.

The reactor core was located near the bottom of the vessel; above was the chimney section formed by the control rod shrouds (Figure 12). Each of the five control rods was connected to a vertical extension rod and a rack which was driven by a pinion gear in the control drive mechanism located on the head. Each pinion gear was driven by a horizontal shaft which extended through a pressure seal in the housing of the drive mechanism and through surrounding shielding blocks to a motor located on the outside. Over the head of the vessel was a sheet metal enclosure filled with metal punchings, gravel and boric oxide to provide shielding. A top shield cap rested on the side shielding blocks.

The core structure was built for a capacity of 59 fuel assemblies, one source assembly, and 9 control rods of which 5 were cruciform rods and 4 T rods. The core in use, however, had 40 fuel elements and was controlled by 5 cruciform rods. The control rods were made of 60 mil thick cadmium, mechanically clad with 80 mils of aluminum. They had an over-all span of 14-1/4 inches and an effective length of 32 inches (Figure 13). The 40 fuel assemblies were composed of 9 fuel plates each (Figure 14). The plates were 120 mils thick consisting of a 50 mil uranium-aluminum alloy "meat" and 35 mils of X-8001 aluminum cladding. The meat was 25.8 inches long and 3.5 inches wide. The water gap between fuel plates was 310 mils. The initial loading of the 40 assembly core was highly enriched and contained 14 kilograms of U-235.

On each of the 16 fuel assemblies in the center of the core (Figure 15), a full length burnable poison strip was spot welded to one side plate, (shown by dashed lines) and a half length strip to the other side plate (shown by solid lines). The remainder of the fuel assemblies had a full length strip only on one side plate. The strips were aluminum-nickel, containing Boron-10. The half length strips were 21 mils thick, and the full length strips, 26 mils thick. The core contained a total of 23 gms of B-10 as burnable poison.

The fuel was calculated to provide about 15 percent excess reactivity (Figure 16). The burnable poison was calculated to provide negative reactivity of 11.2 percent. Reactivity of about 10 percent was expected to be burned out in 4-1/2 years at normal power operation. The fission products were expected to provide additional negative reactivity of up to 2 percent over core life. The combined excess reactivity (or the reactivity held down by the rods) was calculated to be 3 percent at beginning of life, rising to over 3-1/2 percent in just under one year, then decreasing gradually yielding a calculated life of over 3 years.

At the time of the accident, the SL-1 had been in operation for over 2 years. The reactor had produced 931.5 megawatt days of thermal energy which was approximately 40 percent of the design life of the core.

B. SL-1 Accident

On December 23, 1960, the reactor was shut down for routine maintenance, instrument calibration, installation of auxiliary system valves, minor plant modifications, and installation of flux wires in the core. During the period December 27-30, 1960, the maintenance, calibration, and modification work was performed. Work on the installation of the flux wires started after midnight on the morning of January 3, 1961. This work involved moving the shielding blocks back from the reactor, raising the water level to the top of the reactor vessel, removing selected control drive mechanisms, and inserting the 44 flux wires into predesignated water channels within the fuel assemblies. The flux wires were aluminum, containing cobalt-aluminum alloy slugs, and were to be used to measure flux distribution within the core as part of an investigation of reactor core power distribution. By 4 p.m. on January 3, 1961, installation of the flux wires was completed. The three-man, 4 to 12 p.m. shift on January 3, 1961, was directed to pump the water down to the normal operating level, install the shield plugs in the top head of the reactor pressure vessel (around the control rod extensions), reassemble the control rod drive mechanism, replace the shield blocks, and connect the motors in preparation for resuming operations the following morning.

The operating log disclosed that the crew had pumped the water down to a level 2-1/2 feet below the reactor head. The recovery evidence obtained so far indicates that the crew had installed all the shield plugs and was completing the reassembly or "hook up" of the central rod when the accident occurred at 9:01 p.m. The three crew members on duty, working in the reactor room, received fatal injuries from the explosion. Two crew members died instantly; the third, a few hours later.

C. SL-1 Recovery Operations

The post-accident SL-1 investigation and dismantling operations, which are expected to be completed by midsummer 1962, consisted of three phases. Phase I (January 3-9, 1961) included the emergency operations mainly concerned with the recovery of the victims from the SL-1 reactor building. Efforts also were made to determine whether a nuclear excursion had taken place and the status of the reactor facility. Phase II (January 9, 1961 through April 21, 1961) consisted basically of efforts to determine the gross extent of the accident and the nuclear status of the SL-1 reactor. Phase III (April 21, 1961 to midsummer 1962) consisted of the detailed investigation of damaged reactor components and effects of the excursion in an effort to determine the cause of the accident. The cleanup and dismantling operations of the SL-1 site also took place during this phase.

1. Phase I (January 3-9, 1961)

   During the emergency operations, it was determined that a neutron excursion had taken place. The following are some of the analytical results which supported this conclusion:

   a. Bare gold foils from a Hurst dosimeter which was located near the entrance to the operating floor indicated a neutron exposure of 1.2×10^8 thermal neutrons per cm^2.

   b. A brass screw taken from a cigarette lighter indicated a neutron exposure of 9.3×10^9 thermal neutrons per cm^2.

   c. A brass watch band buckle indicated a neutron exposure of 1.8×10^{10} thermal neutrons per cm^2.

   d. Gold jewelry indicated a neutron exposure of 9×10^9 thermal neutrons per cm^2.

   e. Analysis of samples taken from the clothing of the victims indicated the presence of uranium and strontium--quantitative analysis of these samples yielded a yttrium-21 activity of 2.4×10^4 d/m/ml.

f. Soil samples from within the area, clothing samples from personnel that entered the reactor room, and air samples from the control room all exhibited a gross fission product spectrum.

Photographs (Figures 17 and 18) were taken of the reactor head area to assist in the recovery of the third victim. The photographer, permitted to enter the building for only 30 seconds, took these photographs in a 500-1000 roentgen/hr field. Figure 17 shows that the metal cover of the pressure vessel head shield was forced upward and the metal punchings and gravel forced out covering the floor area in the foreground. Control rod racks are protruding from nozzles 1 and 7 (see Figure 19 for nozzle positions) and are about 1/2 foot further out than they would normally be during a shutdown. Across the top of the head is a shield plug with a portion of the control rod extension shaft still in this plug, later identified as the No. 9 shield plug. Figure 18 shows the various control rod drive components which had not yet been assembled.

Other photographs taken during this phase of operation indicate physical damage, other than to the pressure vessel and core, was confined to the area directly above the reactor. Tools lying on the shielding blocks were essentially unmoved and only one light located directly above the reactor head was broken.

As mentioned previously, the radiation levels in the vicinity of the reactor head were 500-1000 roentgen/hr and at the building walls the levels were approximately 100 roentgen/hr.

2. Phase II (January 9-April 21, 1961)

With emergency operations completed, no one was allowed to enter the reactor building due to the high radiation fields and because the nuclear status of the reactor had not been determined. It was not then known whether or not water was in the vessel, whether portions of the reactor fuel were precariously balanced and might be dislodged into another nuclear configuration, etc. Hence all penetrations into the reactor room were accomplished remotely.

For these remote penetrations, several devices were used which disclosed valuable though not always conclusive information.

A mockup of the reactor building, reactor head, vessel, etc. was constructed whereby the recovery crews could practice the intricate manipulations required to handle photographic and television cameras and associated lighting in order to view the reactor head area and inside the pressure vessel. Also various probes were used to measure the radiation fields in the reactor building and inside the pressure vessel, the temperature over the reactor head and core, and the water level in the pressure vessel.

The specially shielded crane with a movable boom used throughout the remote operations is shown in Figure 20 performing an entry in the reactor building. Photographs of the various cameras and probes used during this phase are shown in Figures 21 through 26.

Figure 27, a frame from the first movies taken directly over the reactor head, indicated that six nozzles were open to the atmosphere and that nozzle No. 8 appeared to be free of any obstructions. Hence most of the remote penetrations into the pressure vessel were made through No. 8. Although the television shots were not too clear, valuable information was obtained as to the condition of the core. Photographs taken of the core using a Minox miniature camera added significantly to our knowledge of the condition of the core (Figure 28).

On April 15, 1961, the shielded miniature camera assembly was used in conjunction with a chemical probe which reached within 3 inches of the bottom of the pressure vessel (Figure 28). The probe gave no indication that water was present in the vessel and hence the reactor was declared nuclearly safe as long as the core remained unmoderated.

Aside from determining the nuclear status of the reactor, significant information was obtained from the numerous photographs, movies, etc. taken inside the pressure vessel. It was determined that the four outside control rods (1, 3, 5, and 7) were essentially in place and that the central rod, No. 9, had been ejected upward and was lying across the top of the core. These observations clearly indicated that the core and core structure were severely damaged.

3. Phase III (April 21 to Present)

With the nuclear status of the SL-1 reactor known, the recovery operations could proceed more deliberately. By the end of April, radiation levels within the reactor building had decayed to approximately 200 roentgen/hr. The primary objective of this phase was to determine the cause of the accident. Complete photographic and radiation surveys were a necessity before removing debris and reactor components from the reactor building. As these surveys progressed, some of the reactor components (excluding those inside the pressure vessel) were removed from the building. Limited personnel access to the reactor building was eventually allowed when the radiation fields became better known. A hole cut into the side of the reactor building at the fan room level (above the reactor room) permitted access to that area for completion of surveys of the interior of the building.

Careful examination of the photographs taken and the debris recovered from the reactor room led us to believe that the pressure vessel as a whole might have been physically dislocated upward as a result of the nuclear excursion. The most notable evidence which supported this belief was the presence of block insulation lying on the reactor room floor (Figure 29). This insulation was originally wrapped around the pressure vessel and held in place by a 1/4 inch steel jacket. The most likely explanation to account for such large pieces of insulation on the reactor room floor was that the vessel must have been forced upward. Early in November 1961, a trial lift of the pressure vessel confirmed that the vessel had indeed been projected up by the explosion, shearing the steam nozzle and other pipes (Figure 30), and had then fallen back approximately into its normal position.

Before the pressure vessel was lifted, a 2-1/2 inch hole was drilled into the side of the reactor building and through the wall of the pressure vessel at a level below the core. Through this hole, photographs were taken, using a boroscope, which disclosed severe damage to the lower core structure (Figure 31). Also, four of the five control followers were identified, confirming that the four outside rods were essentially fully inserted into the damaged core.

From June through November 1961, cleanup operations proceeded rather slowly since water or any other moderating material could not be used to decontaminate the interior of the reactor building. Vacuum cleaners, a remotely controlled electromagnet, and manual labor (on a rapid, large scale turnover rate to avoid overexposure to any individual) were the techniques used to remove the debris from the reactor building. The radiation levels within the building were substantially reduced using these techniques and by placing several thousand pounds of steel and lead sheet and lead shot over the reactor head.

By late November 1961, all preparations for removal of the pressure vessel, with the core left inside as it was, were completed. On November 29, 1961, the pressure vessel was successfully removed from the reactor building and transported in a large concrete shipping cask to a large disassembly hot cell located 40 miles north of the SL-1 site (Figure 32).

In January 1962, preliminary hot cell examination of the pressure vessel and core disclosed that the vessel was not ruptured but was bulged about 4 inches in diameter just below the head flange and was bulged about 1 inch above and below the core (Figure 33). The reactor head nozzles were also found to be bulged. The pressure vessel flange was so distorted that the head could not be raised off the head bolts after the nuts had been removed. It was necessary to force the head upward using wedges. After the reactor head was removed, it was clear that the central rod, within its own shroud, was entirely out of and above the core. The rod with shroud was lying approximately 45 degrees to the horizontal across the top of the core (Figure 34). When the central rod and shroud were removed, it was quite evident that the center of the core suffered severe melting and destruction (Figure 35).

Dismantling the reactor building and decontamination of the SL-1 site proceeded quickly with the major sources of radiation removed. At the present time, the building components, the gravel shield and most of the equipment in the building are being buried at a site approximately 1/4 mile from the SL-1 site. The remaining buildings on the site are being restored for future use.

IV. Pre-accident Condition of the SL-1 Reactor

A. General

This section is concerned with certain circumstances and conditions of the SL-1 reactor which are relevant to discussions of the accident. There is no evidence to indicate that any of these circumstances had a direct relationship to the SL-1 accident. Each of the factors mentioned has a logical explanation as to why it existed, though there has been debate as to whether some of these circumstances and conditions should have existed. Factors underlying various design features, conditions, procedures, etc., include such intangibles as operating and design philosophy, engineering judgment, state-of-the-art of reactor development at the time, budgetary and programming considerations, administrative procedures and organization.

B. SL-1 Core Design

1. Reactivity Worth of the Central Control Rod

With the reactor at ambient temperature and pressure and with the four outside rods fully inserted, the reactor could be made critical by the withdrawal of the central rod alone.

The central rod (No. 9) critical position, measured early in the core life, was 19.2 inches at 83°F; in February 1960 this position was 16.1 inches at 83°F. In September 1960 the position was measured at 14.3 inches at a temperature of 106°F. In November 1960 additional cadmium was added to the core which decreased the core reactivity by about 1 percent as indicated by the change in rod bank position. This presumably would have also raised the critical position of rod 9 a slight amount, but this was not measured.

For remote site applications, it is necessary to keep the size and weight of the reactor to a minimum in order to minimize transportation and installation costs. This requirement made it necessary to optimize for compactness, efficiency, and reliability. The SL-1 reactor was designed to accommodate 59 fuel elements, one source assembly, and 9 control rods. However, during the initial zero power experiments, it was evident that a 40 element, 5 rod core would adequately meet the basic design criteria of 3 megawatt thermal (MWt) operation with a 3-year core life. It was this deliberate effort to minimize the size of the core which gave the central rod an abnormally large reactivity worth.

2. Boron Burnable Poison

In order to obtain a 3-year core life at 3 MWt, burnable poison was required to compensate for the heavy loading of uranium 235. Attempts to include this poison in the aluminum-uranium fuel matrix proved unsuccessful. As was done in Boiling Reactor Experiment No. 3 (BORAX III) where boron strips were used to assist rod control, boron strips were fusion welded to one or both side plates of designated fuel assemblies. The flexibility of this method proved to be very useful since the final boron loading could be readily changed during the zero power experiments which immediately preceded full power operation.

During the fabrication of these strips, the aluminum-boron meat was placed in an aluminum jacket. Pressing and rolling were calculated to result in a 2 mil clad. Strips were then cut from large rolled sheets leaving the meat on the edges exposed and, subsequently, these strips were fusion welded to the fuel assemblies.

In the operation of the SL-1 reactor, there had been considerable concern that swelling of the aluminum fuel elements might occur as a function of irradiation damage. A schedule for the removal and inspection of selected fuel elements was established to check for fuel element swelling. During such an inspection in August 1959, the aluminum fuel elements were in good condition, but there were indications that the boron strips were bowing. During a similar inspection in August 1960, the boron strips had bowed between the weld joints and had wedged the elements tightly together. Much force was required to remove one of the center fuel elements. On one element it was found that boron side strips had bowed up to 170 mils between the welds. This element, photographed under water above the core, is shown in Figure 36. On another element, both the half length and full length boron strips were missing when removed. Portions of these strips plus a loose boron strip from an adjacent fuel element were subsequently recovered from the core. The appearance of these strips, in comparison with an unirradiated strip, is shown in Figure 37.

Prior to finding the corroded boron strips, it was noted that the operating rod positions were deviating from those that had been predicted analytically by the window shade technique (Figure 38). It has been calculated that over the core life the rod bank positions would first move in, then level off, then move out. Actually, the bank positions were moving in but at a faster rate than expected, indicating a more rapid gain of reactivity then expected. This could have been caused by the loss of some of the boron.

In September and October 1960, an experimental and analytical program was conducted to investigate the reactivity gain and the corrosion of the burnable poison. As a result, additional shutdown margin was provided by the addition of 60 mil cadmium strips in two of the T slots. It was estimated that this increased the shutdown capability of the reactor by approximately 1 percent reactivity.

Except for additional boron burnup, there is no information which indicates that the condition of the boron strips changed during November and December 1960; hence, the above was supposedly the approximate burnable poison status of the core at the time of the accident.

C. SL-1 Control Rod Drive Mechanisms

1. Performance of the SL-1 Control Rod Drive Mechanisms

The control rods were driven by a rack and pinion mechanism located in a pressure housing (also called a "bell housing") on the head of the reactor vessel as shown in Figure 39. The control rod blades were guided by shrouds within the core. At a ball joint, the blades were connected to vertical control rod extensions and racks. The racks meshed with a pinion gear. The horizontal pinion shaft penetrated the wall of the thimble through a rotating seal and was driven by a motor through a gear-box and magnetic clutch. By de-energizing the clutch coil, the pinion was released from the motor, and the rod could then fall by gravity, with the rack and pinion gear "free-wheeling." Any friction in the seal on the horizontal pinion shaft would tend to impede the fall of the rod. An auxiliary clutch permitted the motor to drive the released rod downward and, if necessary, prevented upward rod motion after release.

A detailed investigation of the SL-1 operating logs disclosed that the SL-1 control rod drive mechanisms performed a total of 4300 movements. In 98 percent of these cases, the mechanism operated satisfactorily. In 84 instances, or 2 percent of these cases, one or another of the 5 mechanisms operated in a less than satisfactory manner. Forty-six instances were noted where a rod did not fall freely in a scram and required the mechanical drive to assist or drive the rod in. During November and December 1960, 33 instances of sluggish or sticking performance were experienced.

The 84 instances mentioned above include instances (1) when a control rod did not meet specified minimum drop time requirements during "free" fall, (2) when a power assist from the drive assembly was necessary to enable a control rod to reach its zero position, or (3) when it was not possible to withdraw a rod prior to startup.

Cases of unsatisfactory performance occurred in a sporadic and erratic manner. Because of the erratic operation, it is difficult to indicate any mechanism which by itself could have caused sticking to occur. In a few of the sticking instances noted, it was known that crud accumulation around the rotating seals and pinion bearings was the cause. Other instances can be attributed to other mechanism problems; however, the cause of the majority of the instances was not identified.

The SL-1 Board of Investigation considered several other possible causes of control rod sticking, but found no evidence for any one cause. Among these was the possibility that the control rod shrouds may have closed in on the blades, because of bowing of the boron strips resulting in crowding of the fuel elements against the shrouds, adding to the friction of the system; crud accumulation within the shrouds may have caused the erratic performance of the control rods.

Very few incidents of rod sticking were formally reported. There had been some trouble with the control rod mechanisms from the beginning, and the crew was accustomed to slight rod irregularities. The increasing frequency of difficulties just prior to the accident were not reported to the AEC.

2. Other Design Considerations

a. 17-4 PH Steel

The use of 17-4 PH steel in the fabrication of some of the SL-1 control rod drive components was consistent with the state-of-the-art at the time. The control rod racks recovered from the SL-1 reactor building subsequent to the accident show many surface cracks. In other reactors, it has recently been found that 17-4 PH steel can only be used in reactor components if it is fabricated and processed through carefully controlled heat treatments and manufacturing procedures. Otherwise, progressive stress cracking leading to eventual failure might occur. This was not known when the SL-1 was constructed. Some of the components in SL-1 showed stress cracking (Figures 40 and 41), but so far there is no evidence any cracks had progressed to the point of failure.

b. Manual Movement of Rods During Disassembly and Assembly

During the disassembly and assembly of the SL-1 control rod drive mechanisms, it was necessary to move manually the control rod blades within the core. As noted earlier, each control rod in the core is connected by a long, upward projecting control rod extension to a rack and pinion gear drive located on the top of the reactor vessel head. The rack and pinion gear is inside of a tall, bolted-on bell housing. The horizontal drive shaft from the pinion gear to its drive motor outside the bell housing extends through a rotating reactor pressure seal in the wall of the housing.

When an SL-1 drive mechanism was disassembled, all the drive components were removed from the reactor head and, hence, access to the core was possible. This situation is shown in a pre-accident photograph, Figure 42, with only the control rod rack protruding through the reactor nozzle.

In the reassembly of these mechanisms, the shield plug was lowered into the reactor head nozzle over the rack. The pinion support and spring housings were then lowered over the rack and bolted to the shield plug. A lifting tool was attached to the threaded end of the top of the rack, down inside the spring housing a few inches. At this point the rack and, hence, the control rod were lifted so that a "C" clamp could be attached to hold the rod in a raised position. Very explicit instructions had been given to all operators that this manual raising of a rod should not exceed 4 inches. However, the operator was expected to exercise judgement estimating this height. There was no position stop; it was possible for the operator to raise the rod higher--even to complete withdrawal.

With the "C" clamp on the rack, the lifting tool was removed and a washer and nut were placed on the rack. This nut and washer acted against the spring to hold the rod in the zero operating position and to absorb the force of scrams. The lifting tool was again attached and the rod lifted to free the "C" clamp. The rod was then lowered to the spring. This point in the reassembly is shown in Figure 43. Figure 44 shows the cadmium overlap in the active core at various positions during this reassembly procedure.

Based on the last measurement of critical position of the center rod, there should have been at least a 12-inch margin between criticality and the position to which the center rod is normally raised to during this operation.

D. SL-1 Operating and Maintenance Procedures

Prior to the accident, the SL-1 control rod drive disassembly and assembly procedure was considered routine by all concerned and had been done many times. Hence, a reactor engineer was not scheduled to be present while this procedure was performed on the night of the accident. The written procedure for the disassembly and assembly of the SL-1 control rod drive mechanisms did not have a precautionary note to indicate the danger involved in withdrawing the central rod, but this procedure and the administrative precautions relating thereto were well covered in the training of all operators. The Board of Investigation found that all reactor operators were well aware of the danger associated with this procedure.

The established procedures did not require a crew member to be in the control room during maintenance on the reactor. The SL-1 control and nuclear instrumentation was adequate. However, at the time of the accident, the recorders associated with the nuclear instrumentation (with few exceptions) were turned off. The operating procedures did not require that all recorders be turned on. The constant air monitoring system was on. However, this system would not have responded to the difficulties within the reactor.

E. Reactor Safety Reviews and Inspections

During the operation of the SL-1, many safety reviews and inspections were performed by groups directly associated with the operations and programming. However, most formal inspections were concerned with health physics, radiation protection, and industrial safety problems. No nuclear or reactor engineers were included on the formal inspection teams. Hence, the reactor safety aspect was not adequately covered.

There were only two truly independent over-all safety reviews of the SL-1 facility including the reactor. The first review was made by the AEC's Division of Licensing and Regulation and the Advisory Committee on Reactor Safeguards. This review was made prior to the initial operation of the SL-1. The second review was accomplished by an independent group from the operating contractor's organization at the time this firm assumed operational responsibility for the SL-1 in February 1959.

V. Probable Initiation and Course of the SL-1 Accident

The investigation into the cause of the accident is still underway by the SL-1 Board of Investigation. The final report by the Board will probably be completed this summer. The Board has released several interim reports; the latest on April 3, 1962 follows:

"A meeting of the Board of Investigation on the SL-1 reactor incident of January 3, 1961, was held on March 7, 1962. The purpose of this meeting was to review evidence which has been brought to light since the Board's last report of May 1961. During this period, the reactor has been moved to a large "hot cell" and partially disassembled to facilitate careful detailed viewing and study of each component and bit of evidence which might bear upon the cause of the incident.

"The Board finds it is not in a position to submit a final report but does wish to reaffirm the conclusions reached in its report of May 10, 1961. A great deal of additional evidence has been developed since that report, touching particularly on conclusion (H)*. While the Board has not made a complete review or study of all the new evidence, it finds none which appears to change its conclusions materially, but rather finds further support for those conclusions.

*Conclusion H in the May report states: "At this time it is not possible to identify completely or with certainty the causes of the incident. The most likely immediate cause of the explosion appears to have been a nuclear excursion resulting from unusually rapid and extensive motion of the central control rod. As yet there is no evidence to support any of several other conceivable initiating mechanisms."

"The following observations are based, in large part, on information obtained by the General Electric Company during the recovery and disassembly of the SL-1 reactor vessel and core over the past several months under a Commission contract administered by the Idaho Operations Office:

1. When the explosion occurred, the reactor core was destroyed and a pressure wave or water hammer followed which apparently trapped the central control rod (No. 9) within its shroud at a 20 inch, plus or minus 1/2 inch, withdrawn position.
2. The radial dislocation of the core components indicates that the explosion emanated from the center axis of the core or that part of the core controlled by the central rod.
3. Severe meltdown of the center and lower portions of the central fuel elements was experienced.
4. Preliminary flux wire measurements from wires which were in the core at the time of the incident indicate that the magnitude of the energy released from the resulting nuclear excursion was sufficient to cause the observed damage and effects.
5. Direct measurements of the critical position of the central rod with the core in a cold condition were few; however, from an analysis of the history of the SL-1 core, it appears that the critical position was between 14 inches to 16 inches. Hence, with the known reactivity worth of the rod, its withdrawal to 20 inches appears sufficient to cause the effects observed.

6. Evidence accumulated so far from within the reactor vessel points to no self-propagating metal-water reaction or any other type of chemical explosion.

7. All the observed damage to the reactor building, vessel and core can be reasonably accounted for as a result of the withdrawal of the No. 9 (central) control rod.

"The reason for the withdrawn position of the central control rod is unknown. It is a principal and final objective of the Board to find this reason, if possible, or a reasonable hypothesis of the withdrawing mechanism, and to report other evidence of value to reactor safety through a detailed evaluation and analysis of the reactor core and reactor components. A final report will be written upon conclusion of this work."

The conclusions in this interim report are based on the following most probable sequence of events, believed to have occurred during the course of the accident, and which, at the present time, reasonably explains all of the observed damage.

It is believed that the SL-1 accident was caused by the rapid withdrawal of the central control rod (No. 9) above its critical position, 14 to 16 inches, to a position of approximately 20 inches, thus taking the core above prompt critical.

This nuclear condition rapidly increased the fuel plate temperature to a point near or above melting. The simultaneous generation of steam throughout the center of the core produced a relatively large steam void and high pressures in the core in the order of 500 pounds per square inch. Consequently, the core experienced considerable damage by the expansion of this steam and by the high pressures. At this time, the central rod was probably seized by the shroud surrounding it at about a 20 inch withdrawn position. The 500 psi steam pressure apparently forced a slug of water upward from the general zone of the core. This water slug was accelerated by the steam and was suddenly stopped by the reactor vessel head, causing a high pressure, water hammer phenomenon with pressures probably as high as 10,000 psi. The forces generated by the decelerating water slug collapsed all the shield plug housing extension tubes (Figures 27 and 39) and deformed the reactor vessel wall (Figure 33). Additionally, the momentum of the water slug was transferred to the reactor vessel, imparting a vertical motion to the vessel itself and the shield plugs, which were not bolted to the vessel head. The vessel was projected upward sufficiently to shear the steam nozzle and water lines and to expel onto the operating room floor whole blocks of insulation which originally surrounded the vessel. Subsequently the vessel fell back approximately to its original position.

It has been calculated that the energy released was about 300 megawatt seconds.

Bibliography

For further details concerning the WTR and SL-1, the reader is referred to the following published documents:

Final Safety Report for the Westinghouse Testing Reactor, WCAP-369 (Rev.), August 7, 1958; Westinghouse Electric Corporation, Atomic Power Department.

Catlin, Robert J., Health Physics Aspects of the Westinghouse Testing Reactor Fuel Element Meltdown, WTR-55, June 1960.

Report on WTR Fuel Element Failure April 3, 1960, WTR-49, July 7, 1960; prepared by the Staff of the Westinghouse Testing Reactor.

Hazard Summary Report on the Argonne Low Power Reactor (ALPR), ANL-5744, published November 1958; Argonne National Laboratory.

SL-1 Accident, U. S. Atomic Energy Commission Investigation Board Report, May 1960.

SL-1 Reactor Accident on January 3, 1961 (Interim Report), IDO-19300, May 15, 1961; Combustion Engineering, Incorporated.

Radiation Safety and Regulation, Hearings before the Joint Committee on Atomic Energy, Congress of the United States, 87th Congress, June 12, 13, 14 and 15, 1961.

SL-1 Recovery Operations January 3 thru May 20, 1961, IDO-19301, June 30, 1961; Combustion Engineering, Incorporated.

Nuclear Incident at the SL-1 Reactor, IDO-19302, issued January 1962; U. S. Atomic Energy Commission, Idaho Operations Office.

List of Figures

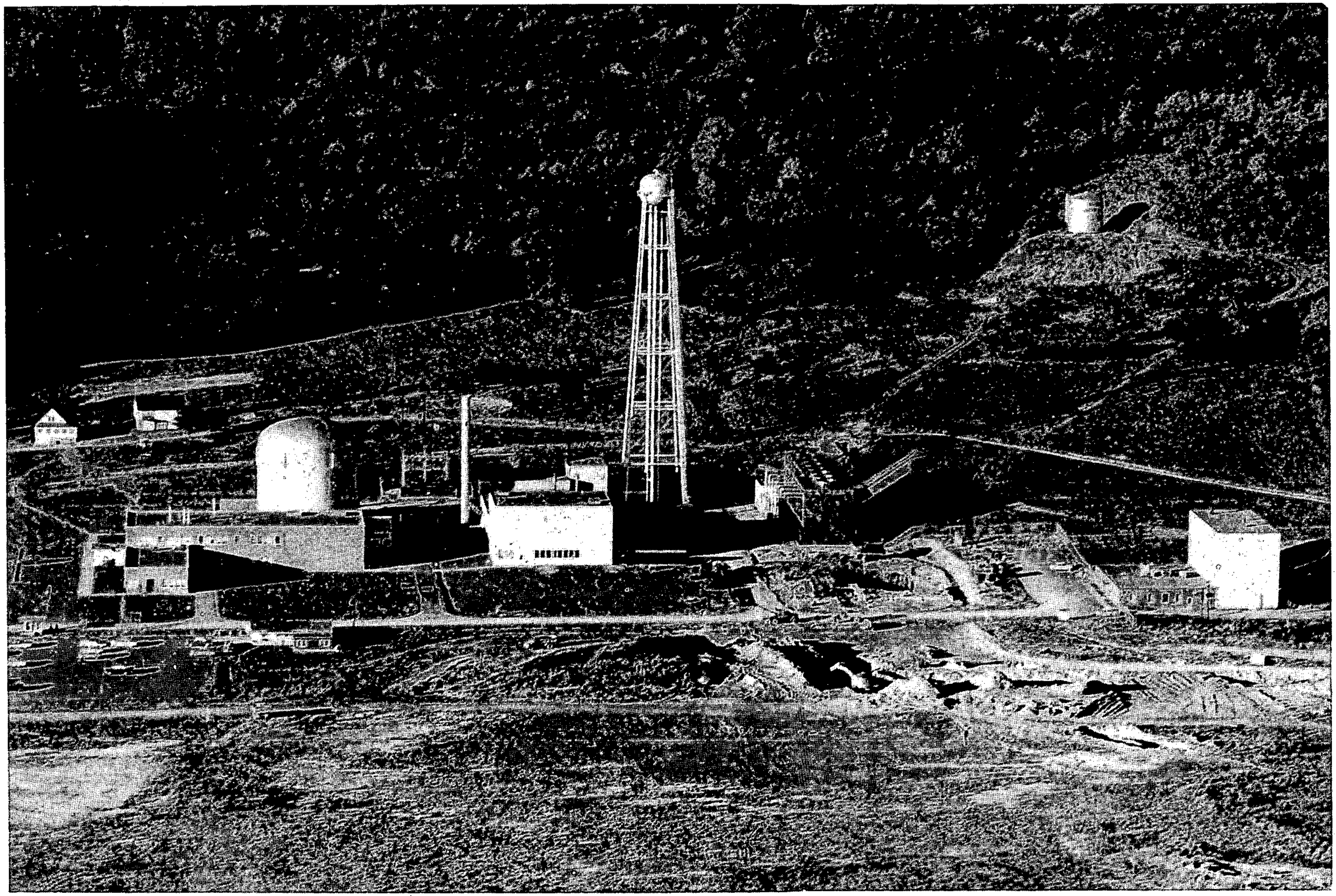


Figure 1 Westinghouse Testing Reactor Located 20 Miles Southeast of Greater Pittsburgh

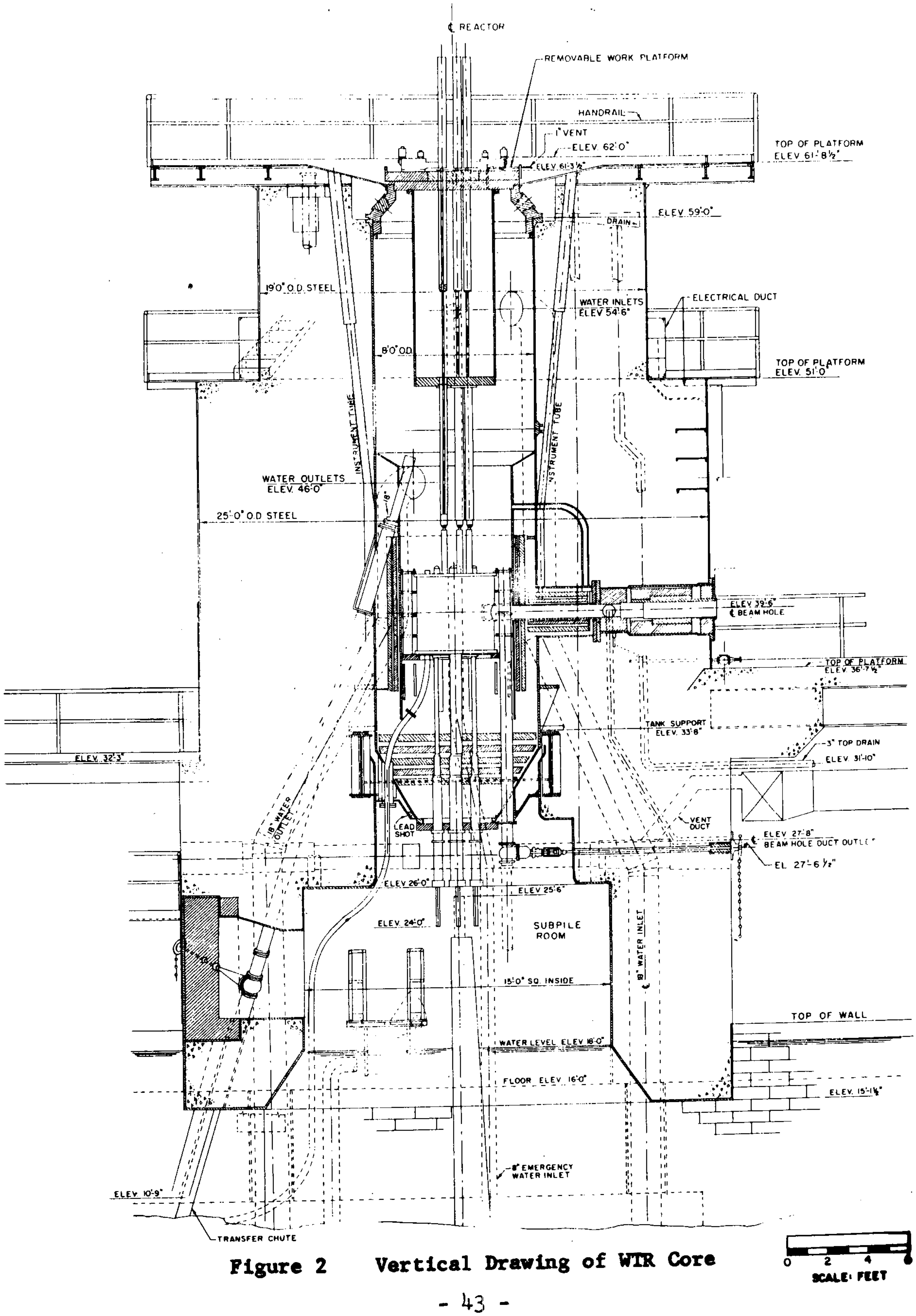


Figure 2 Vertical Drawing of WTR Core

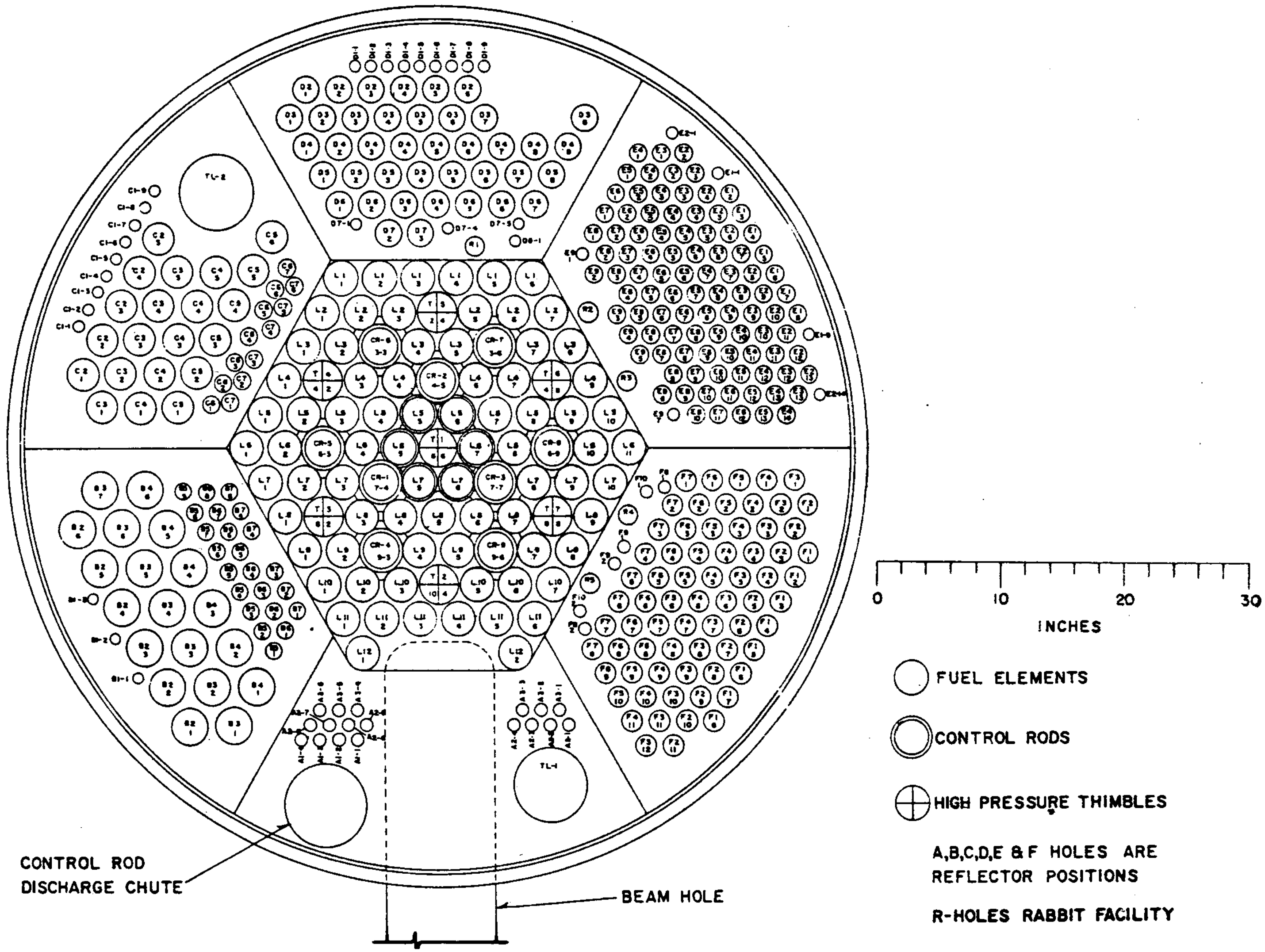


Figure 3 Plan View of WTR Core

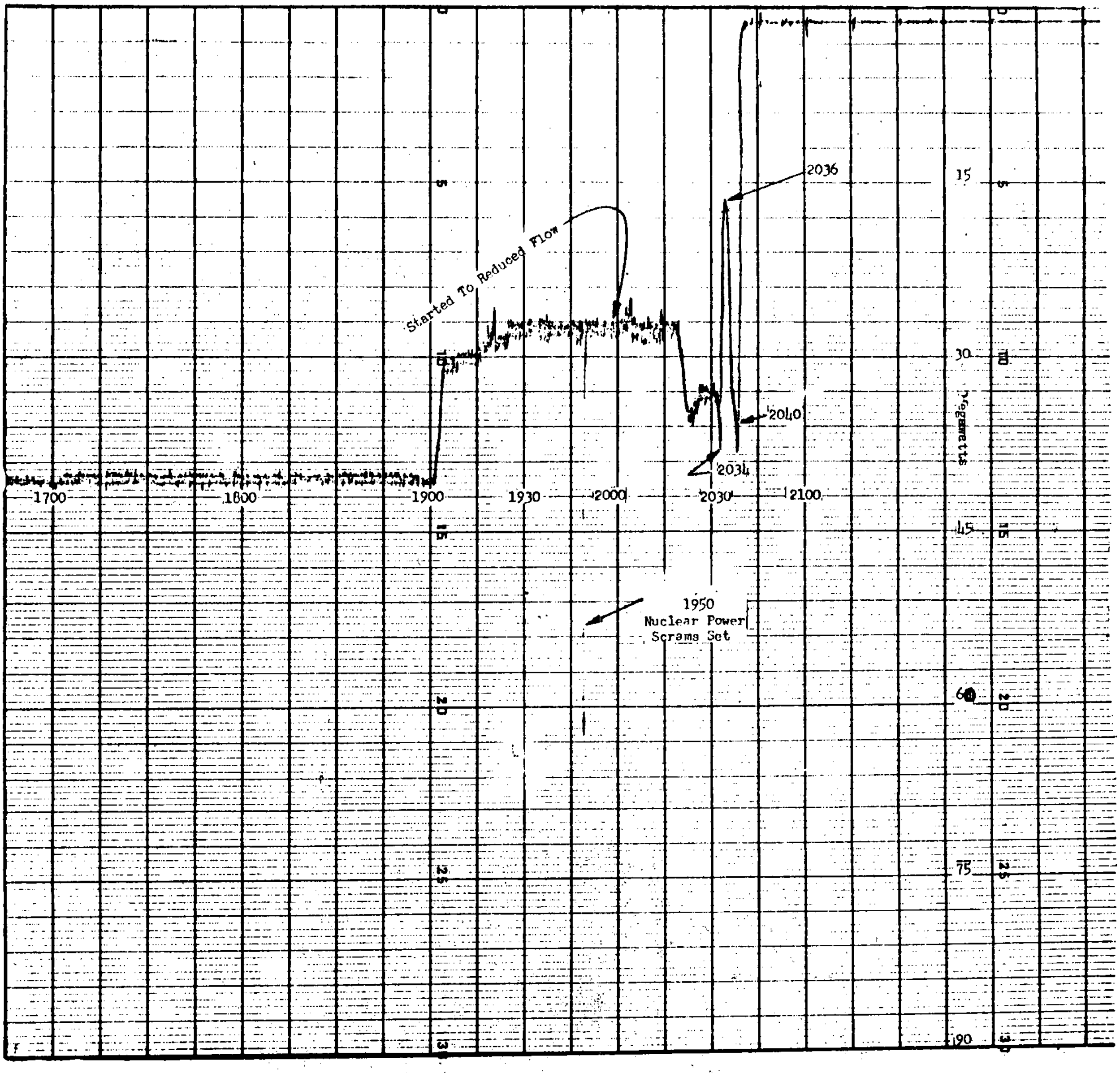


**Figure 4 WTR Neutron Power Level Recorder Chart
Night of April 3, 1960**

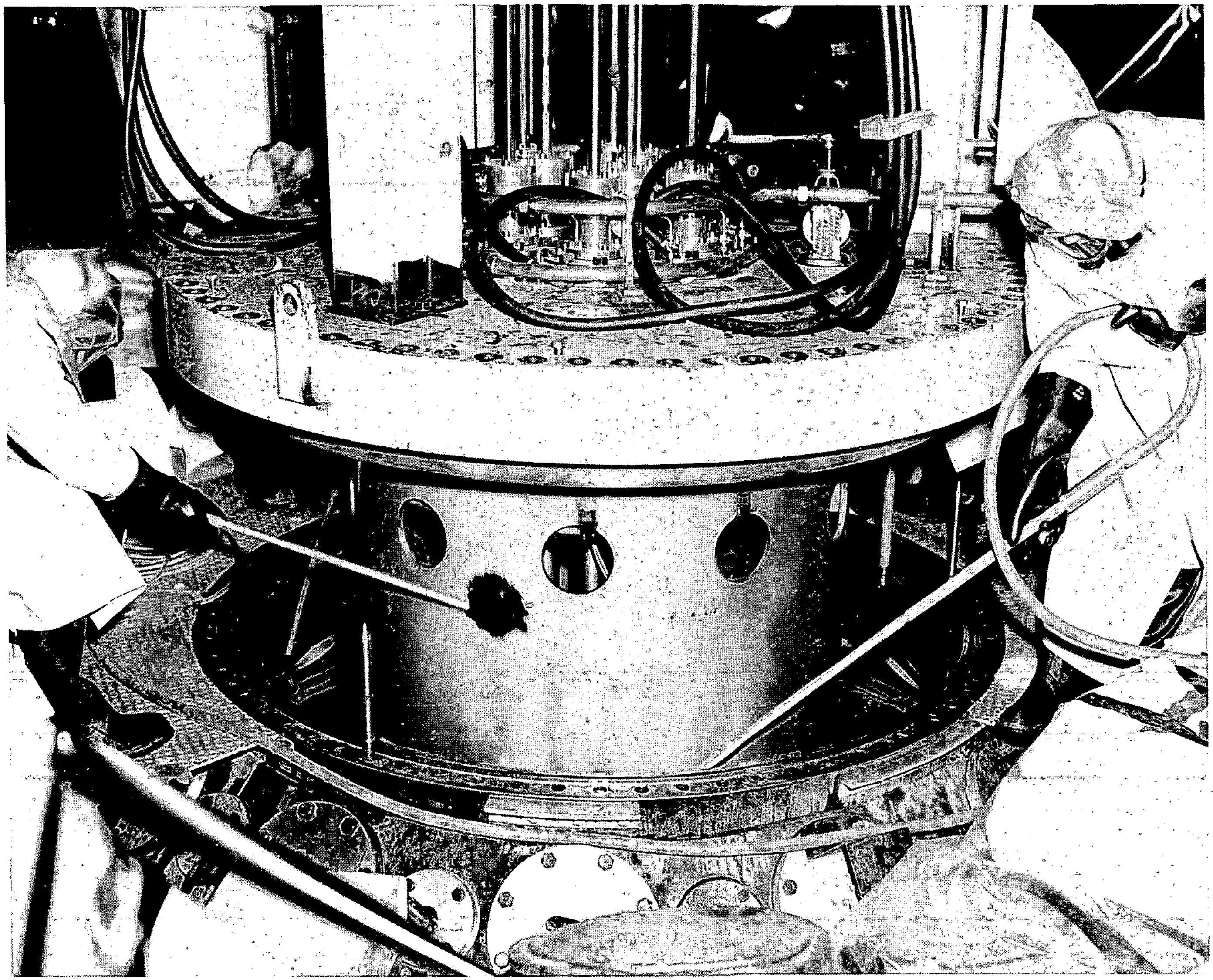


Figure 5 WTR Reactor Vessel Head Removal and Decontamination

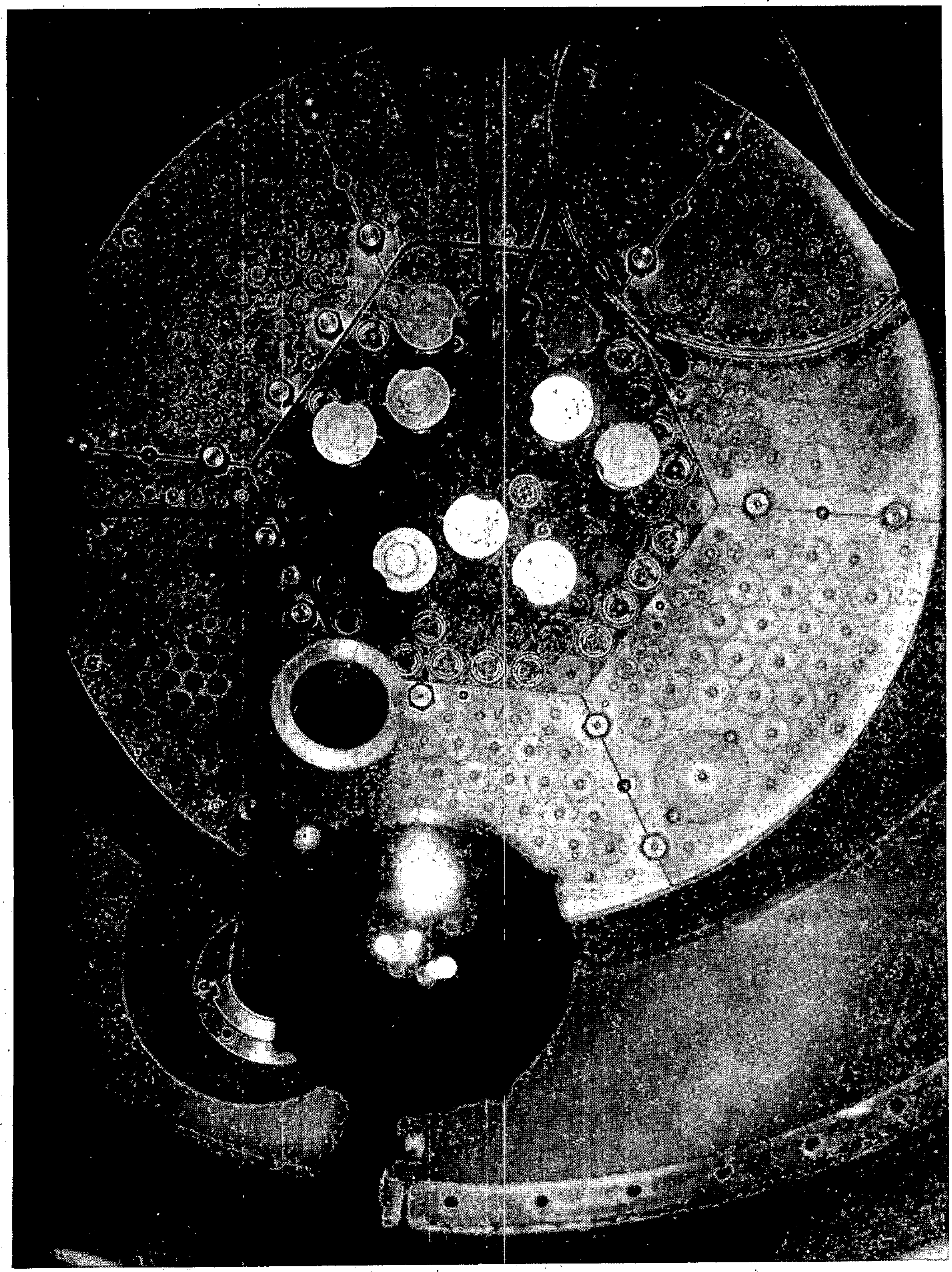


Figure 6 View of WTR Core - April 11, 1960

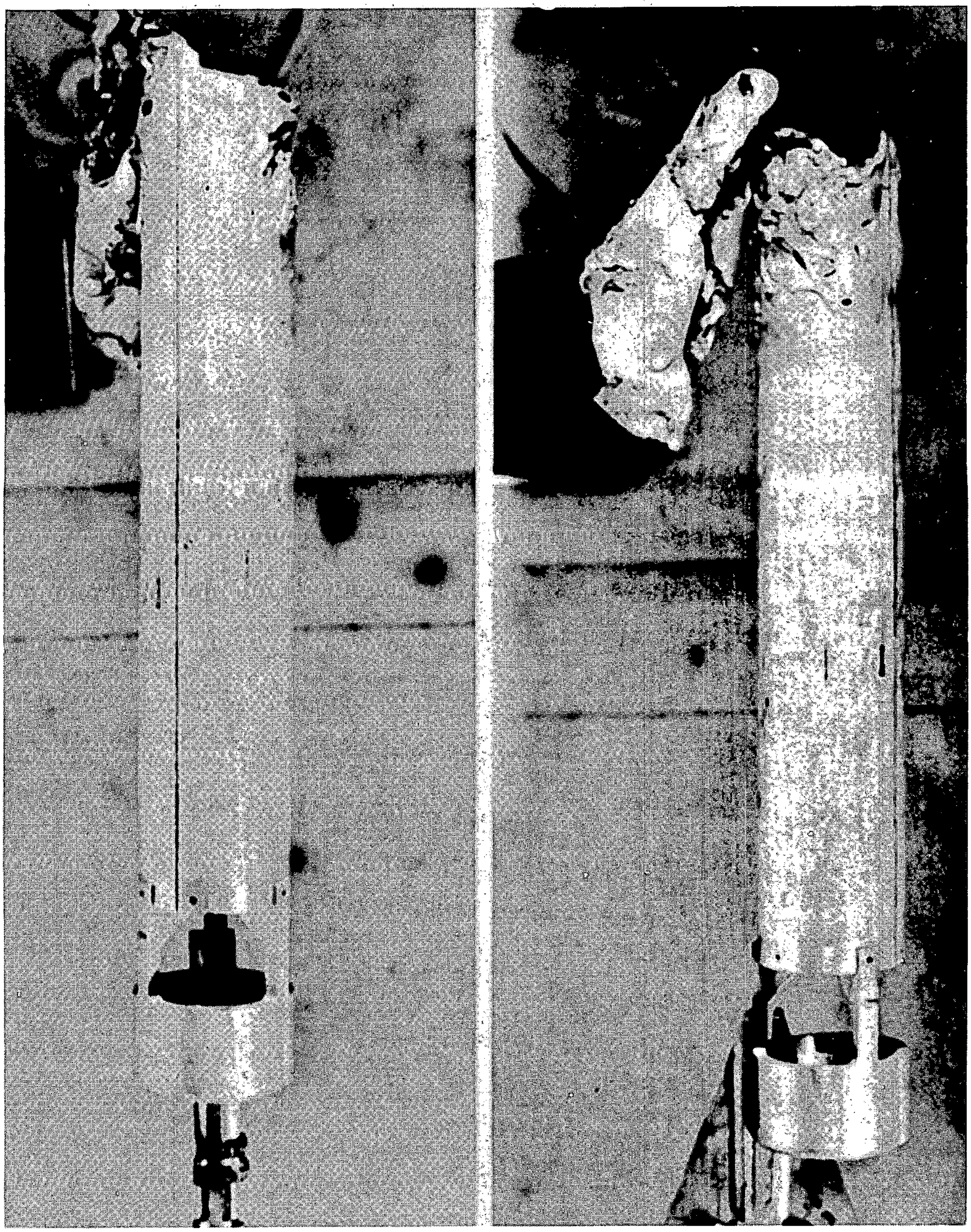


Figure 7 Damaged WTR Fuel Element

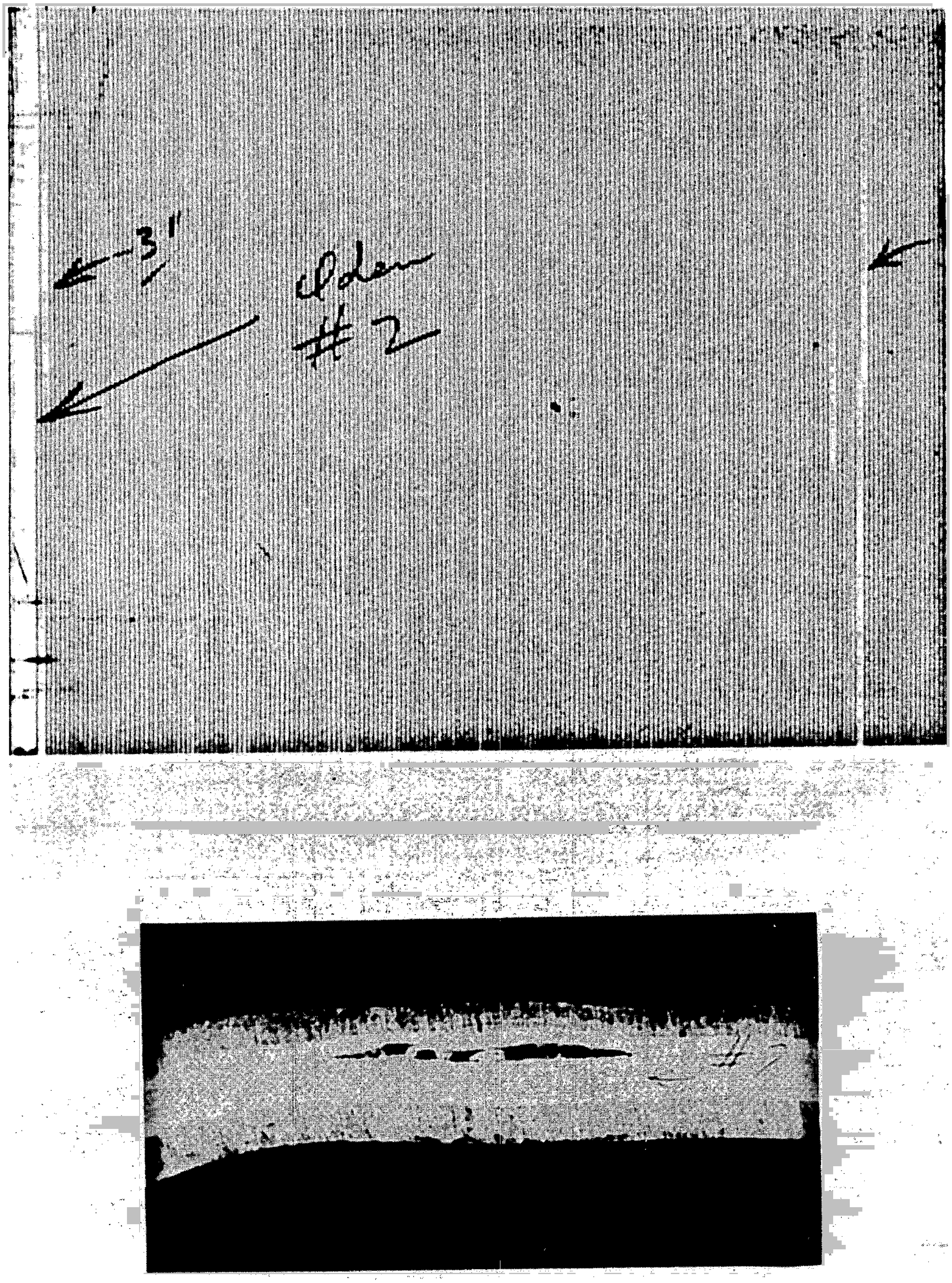


Figure 8 WTR-Typical Ultrasonic Trace with Defect Photo

NATIONAL REACTOR TESTING STATION

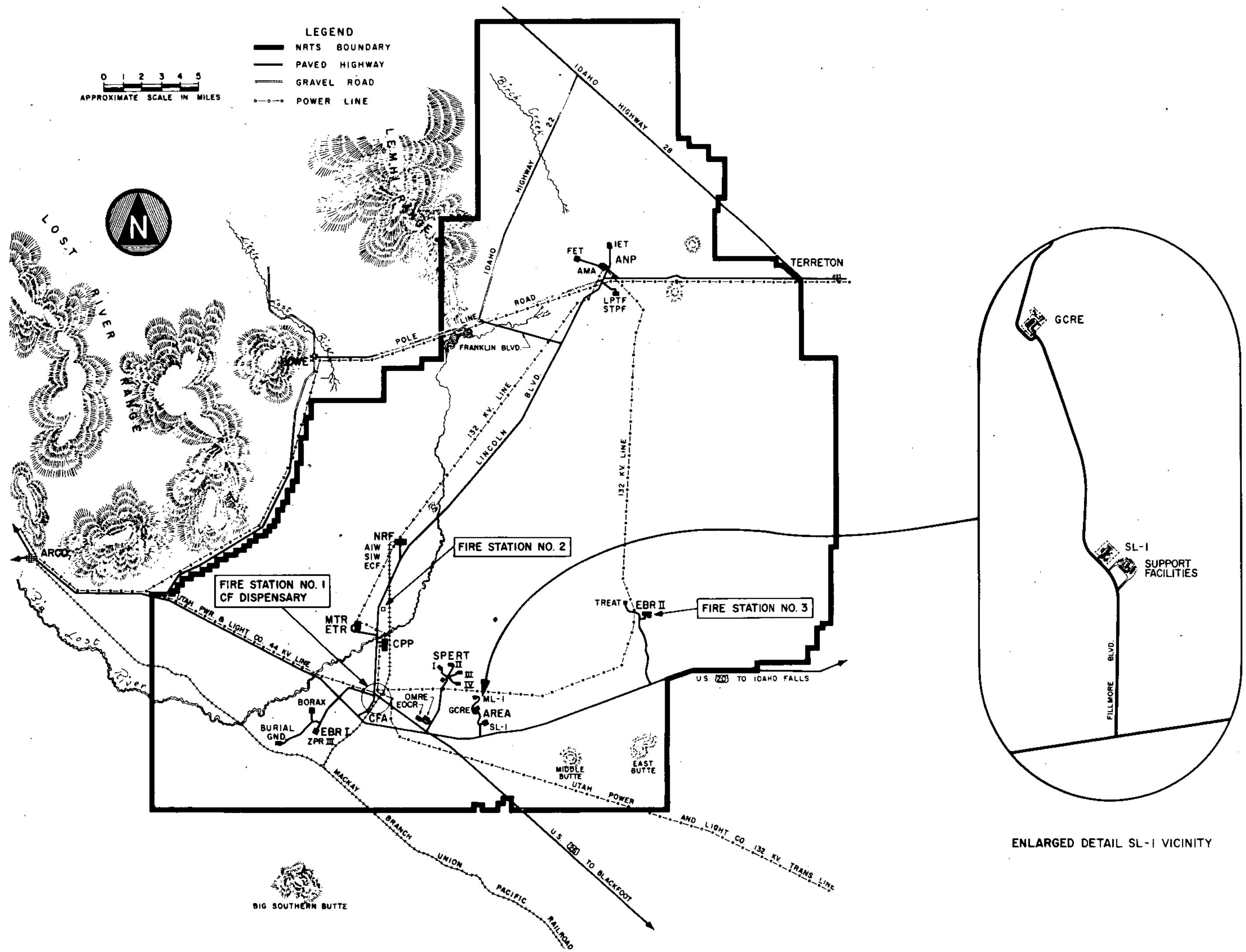


Figure 9 National Reactor Testing Station

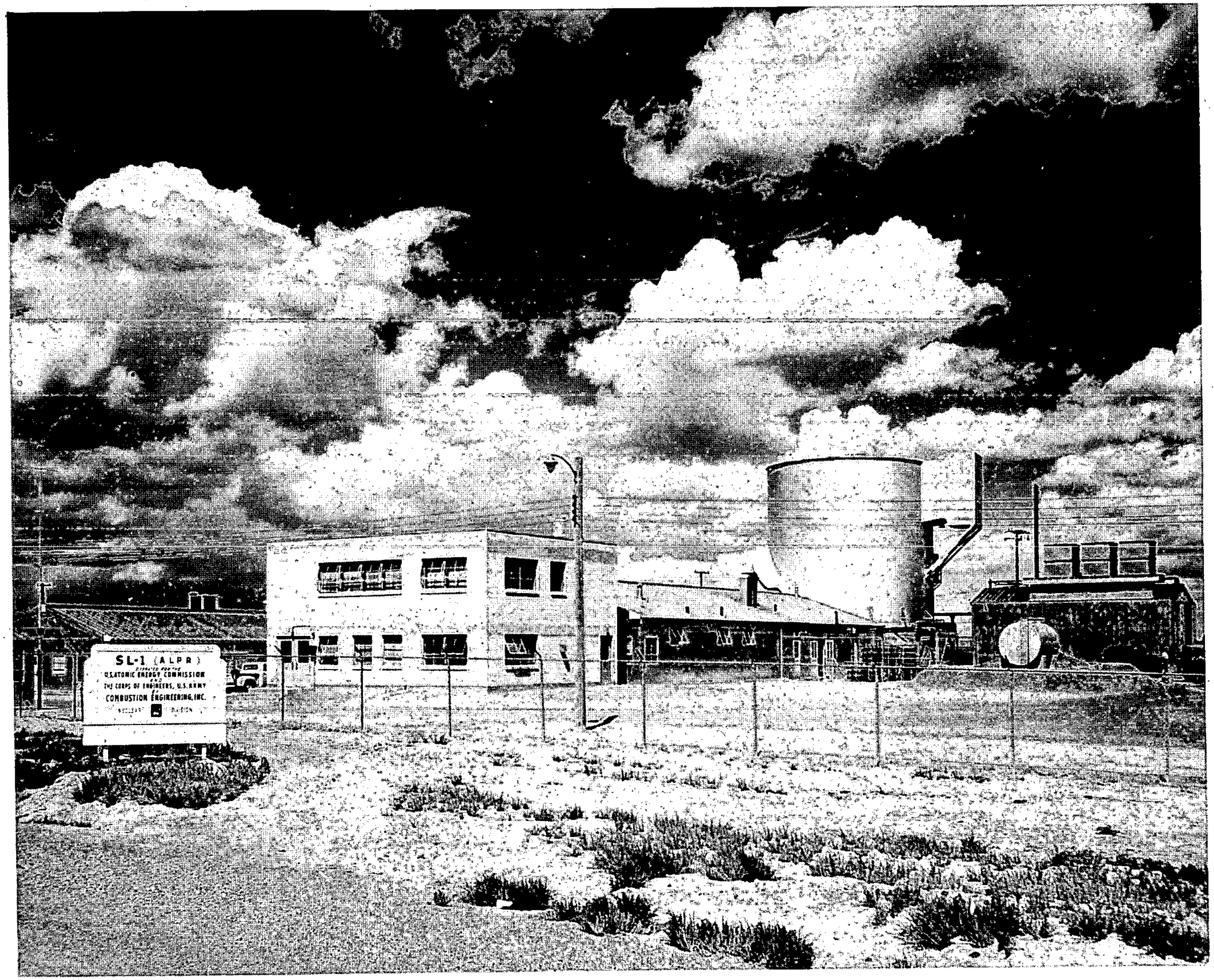


Figure 10 General View - SL-1 Facility

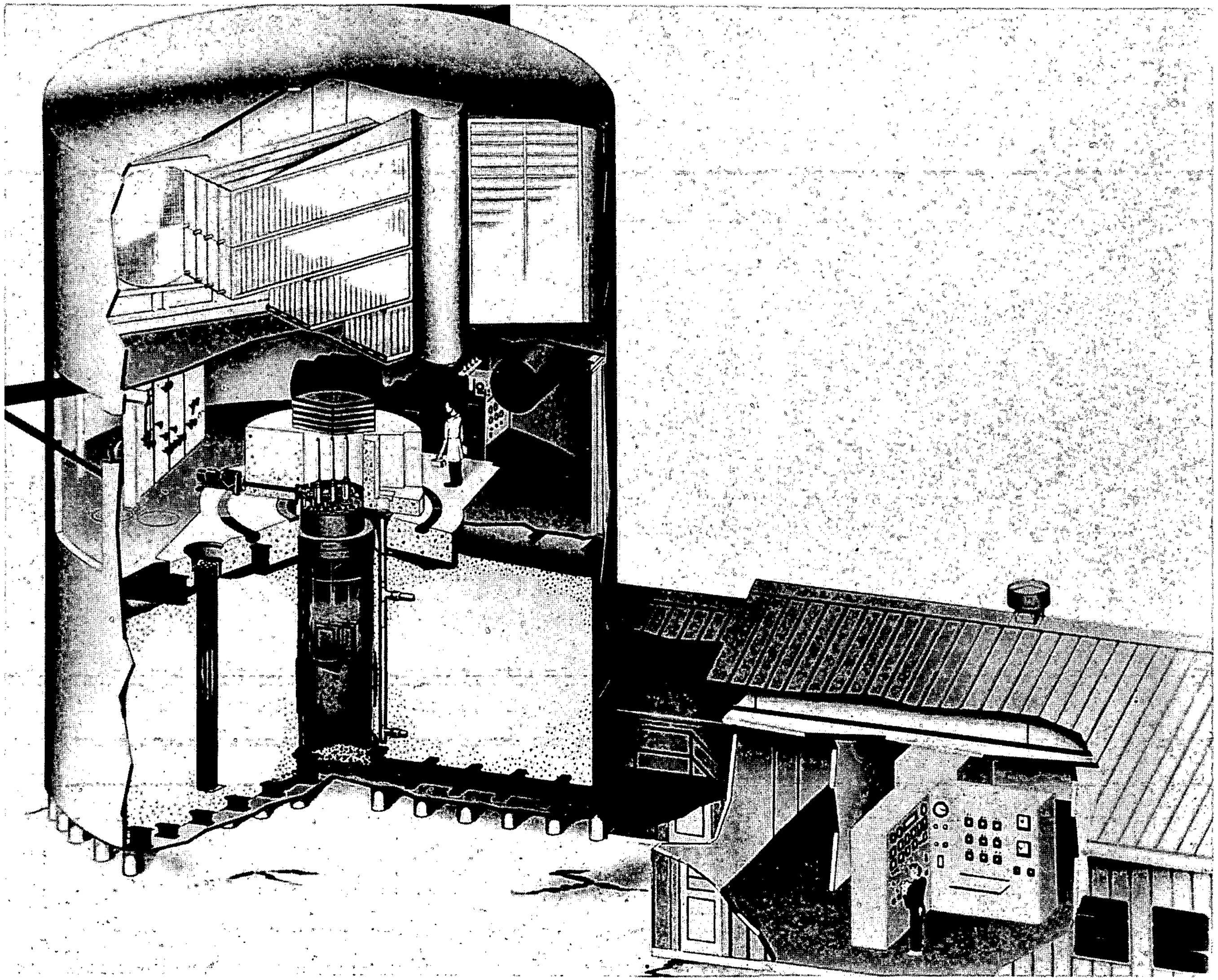


Figure 11 SL-1 Plant Perspective

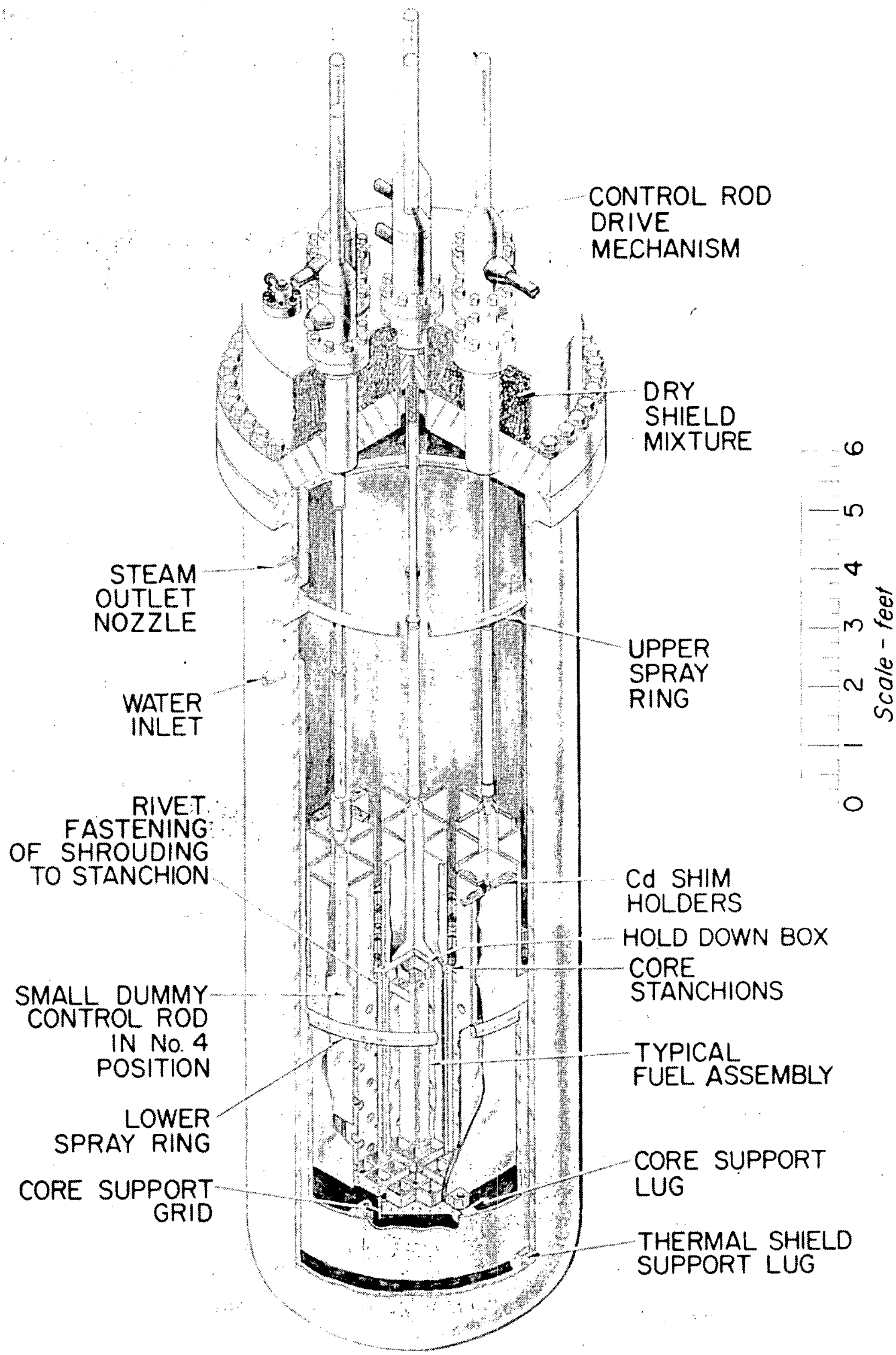


Figure 12 SL-1 Reactor Perspective

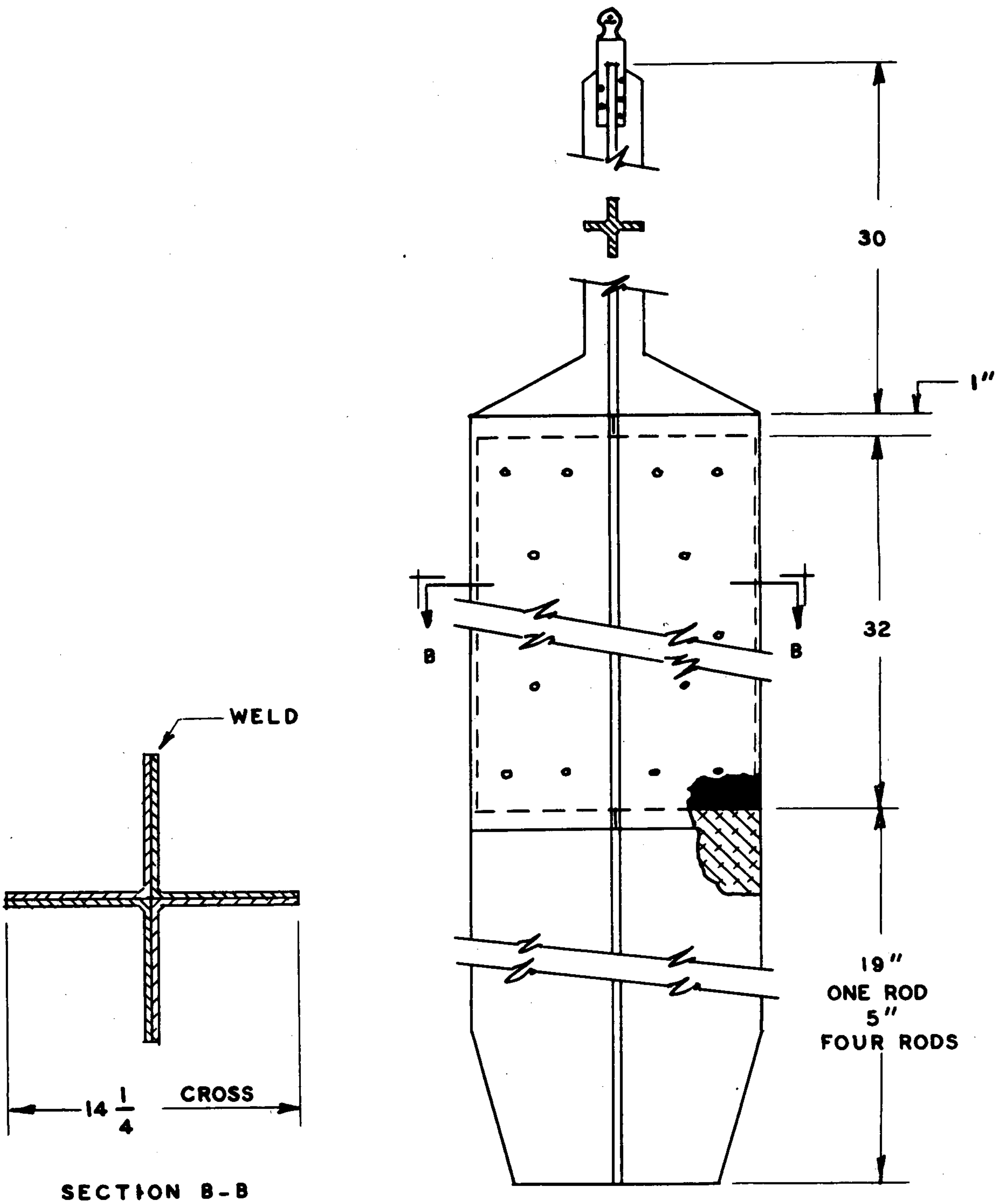


Figure 13 Cross Type Control Rod

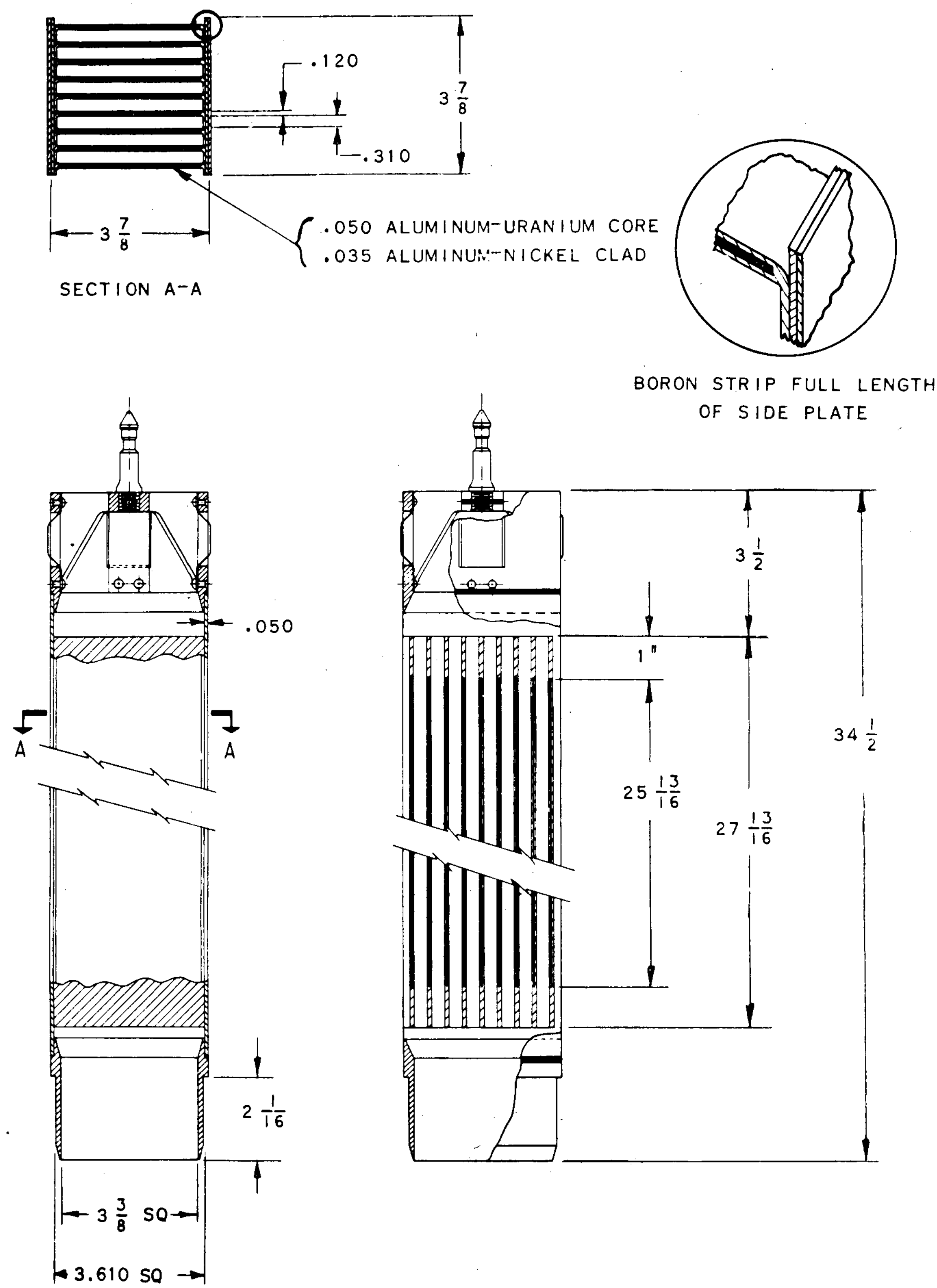
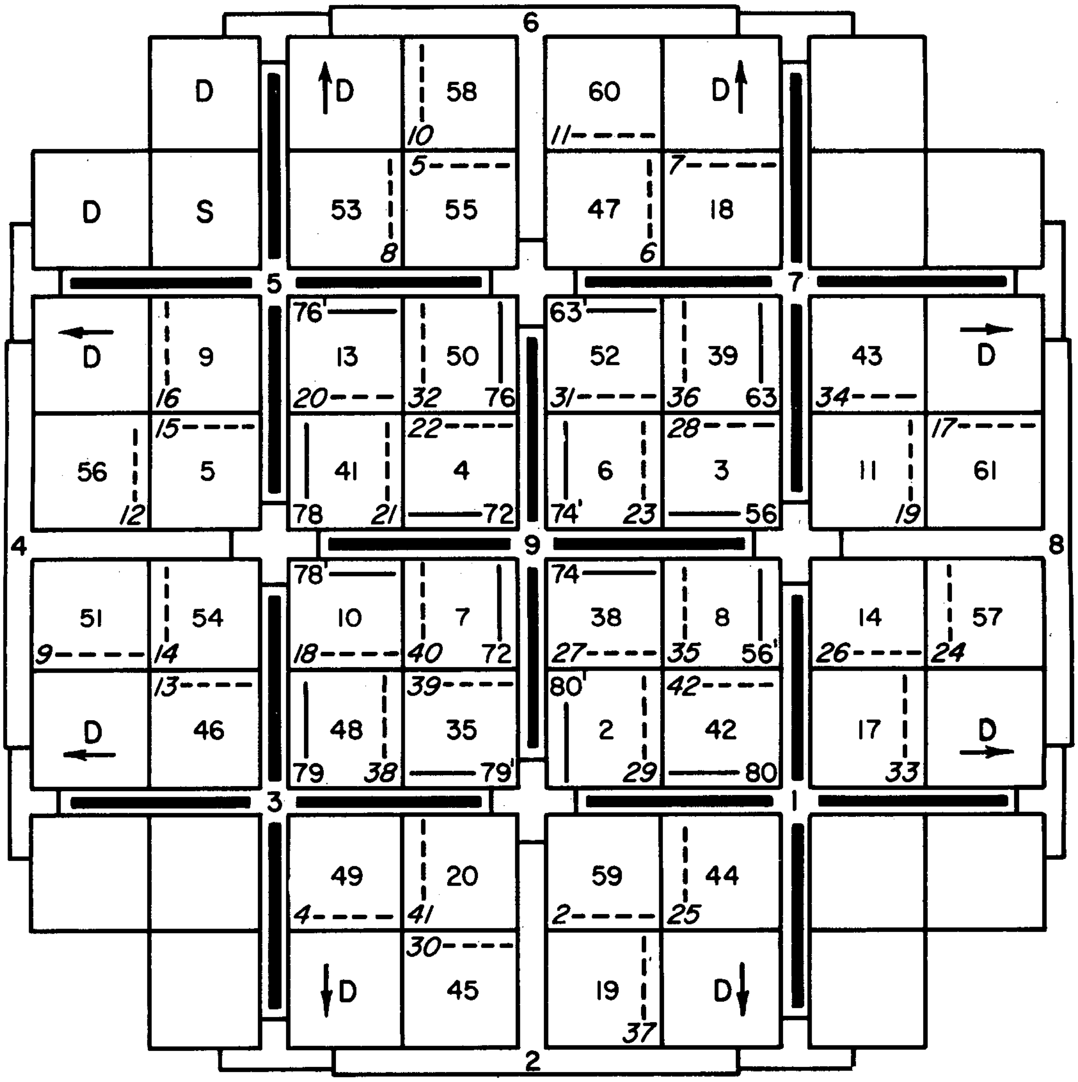
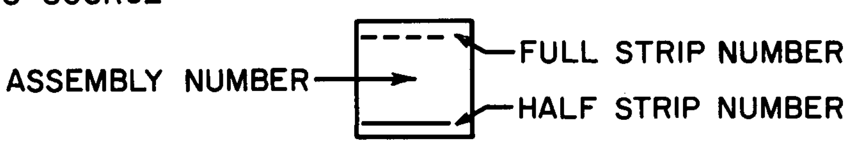


Figure 14 Fuel Element



KEY:

FULL STRIP OF BORON – DASH LINE
HALF STRIP OF BORON – SOLID LINE
D – DUMMY ELEMENT
S – SOURCE



The position of the full and dotted lines indicate the orientation of the assembly and the position of the boron within a cell of four assemblies.

The direction of the side plates of the dummy elements are shown by an arrow.

Figure 15 SL-1 Loading for 40 Element Core

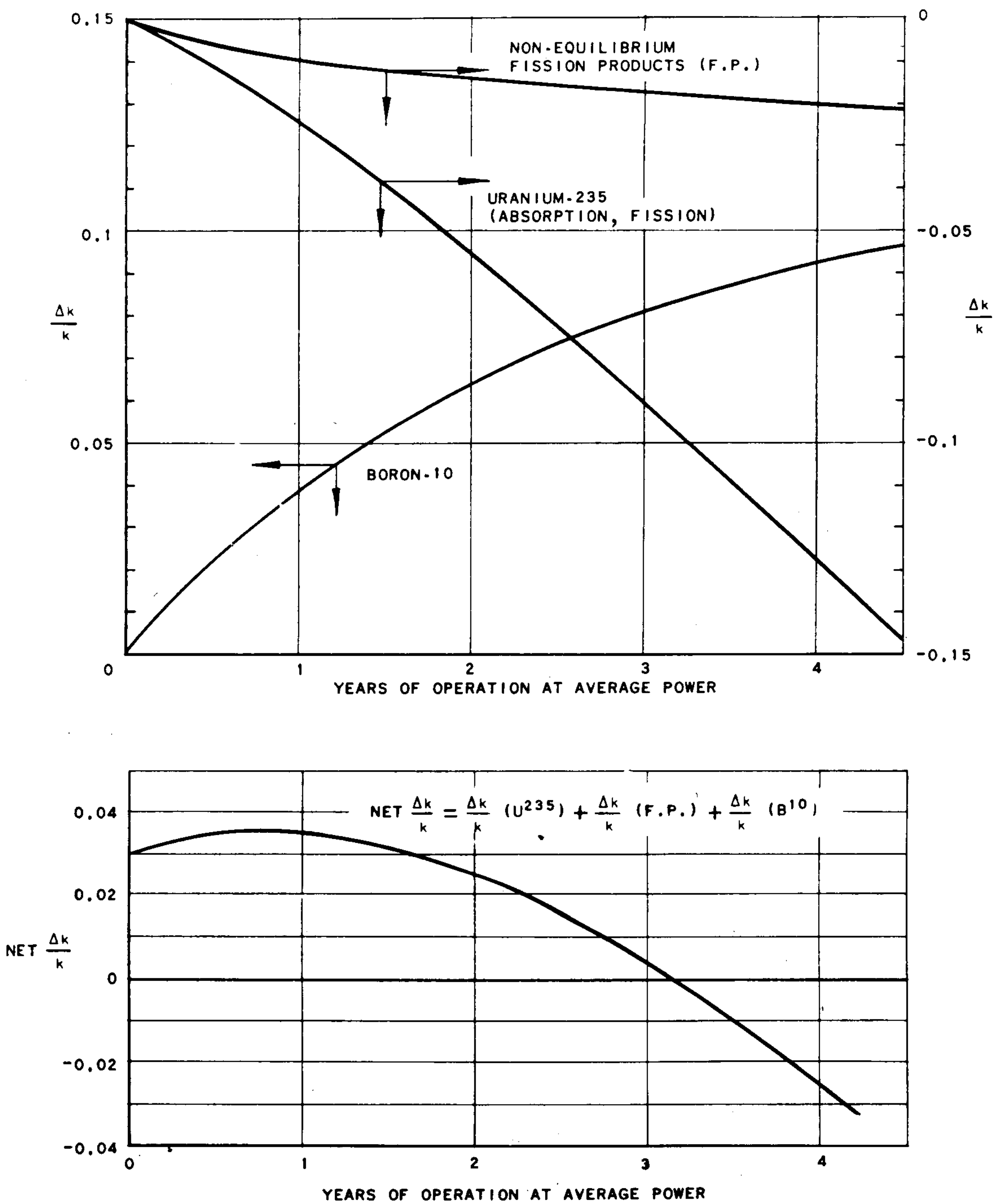


Figure 16 Reactivity Variation During SL-1 Core Lifetime

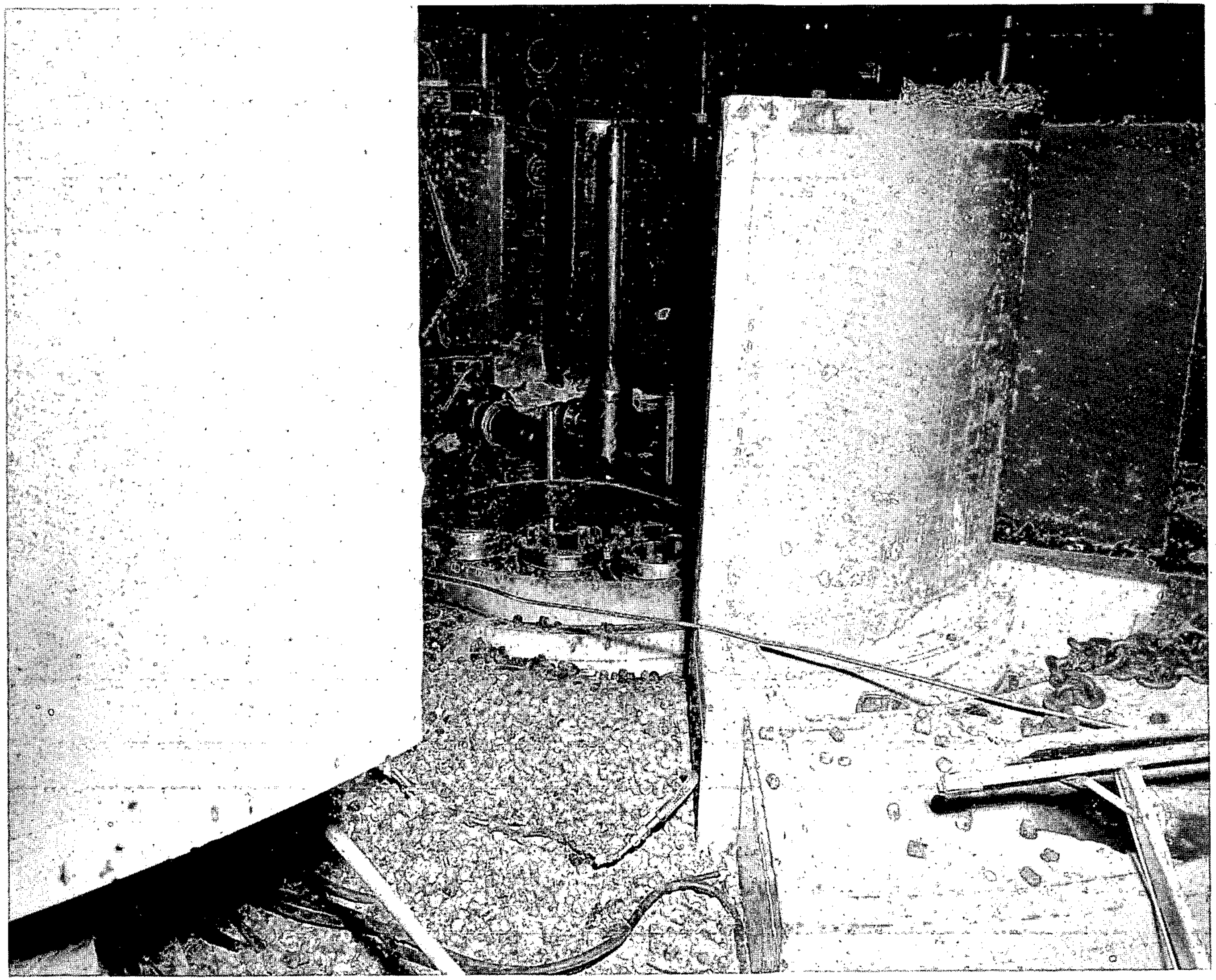


Figure 17 View of SL-1 Reactor Head Area After Accident

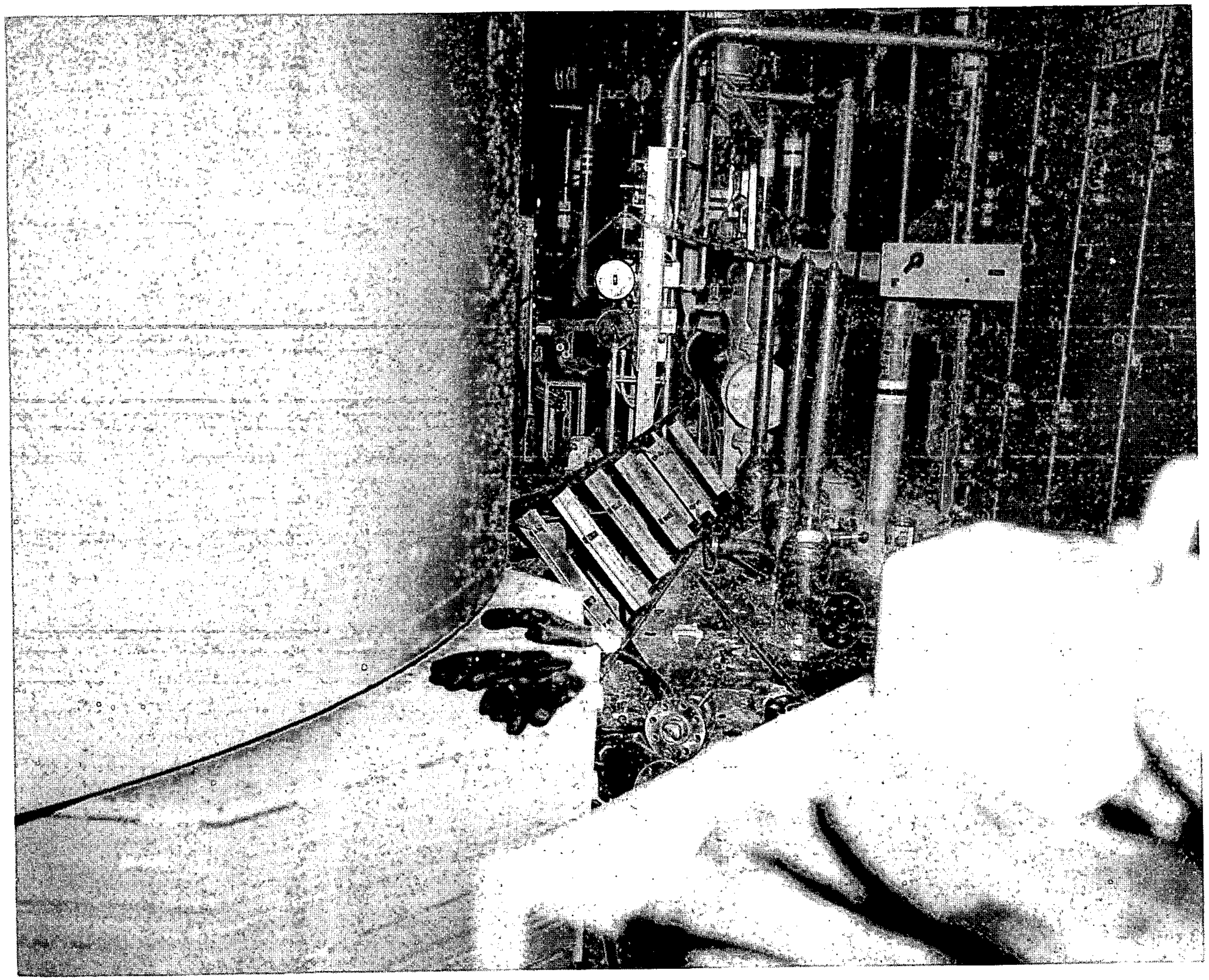


Figure 18 View of SL-1 Reactor Room After Accident

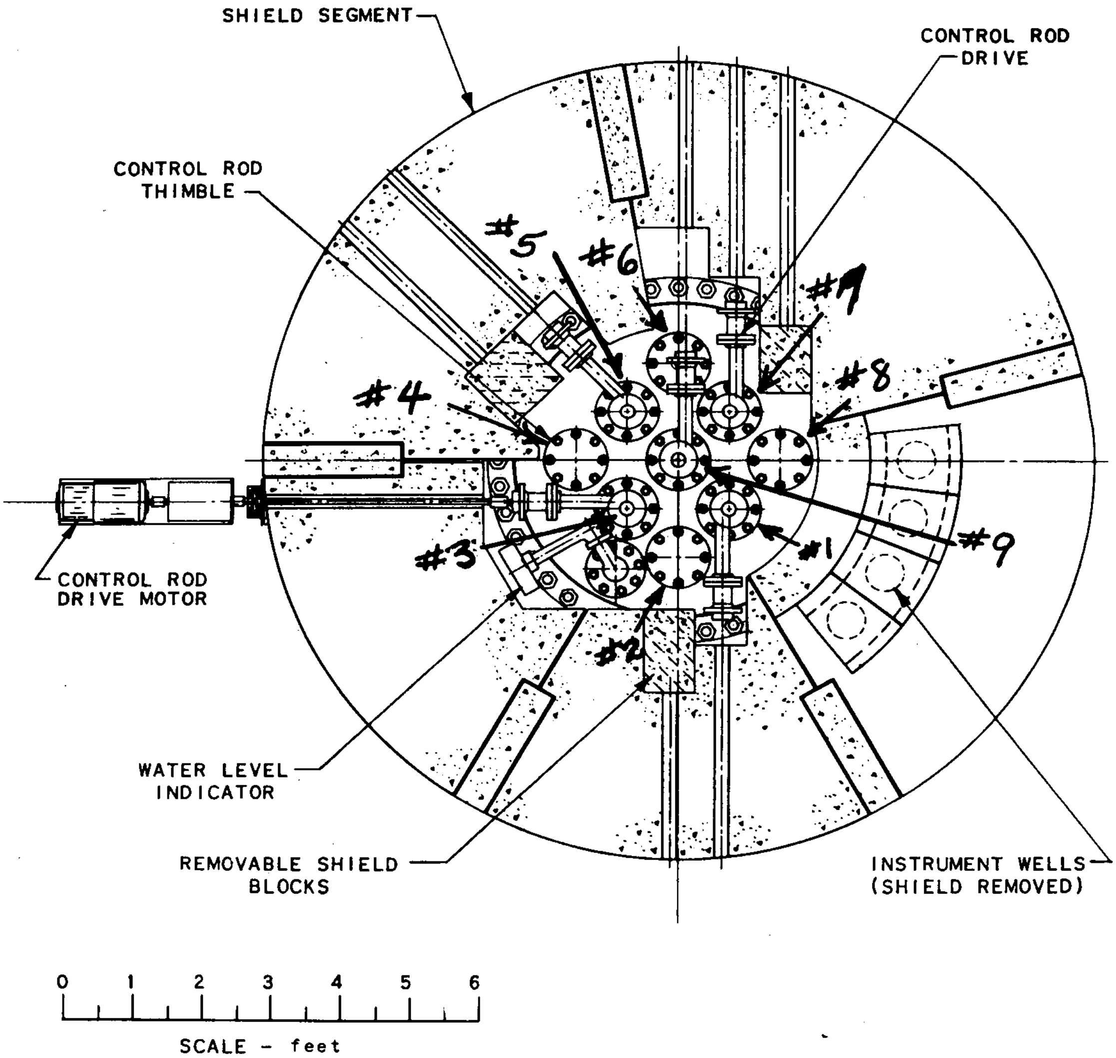


Figure 19 Control Rod Drives and Top Shielding

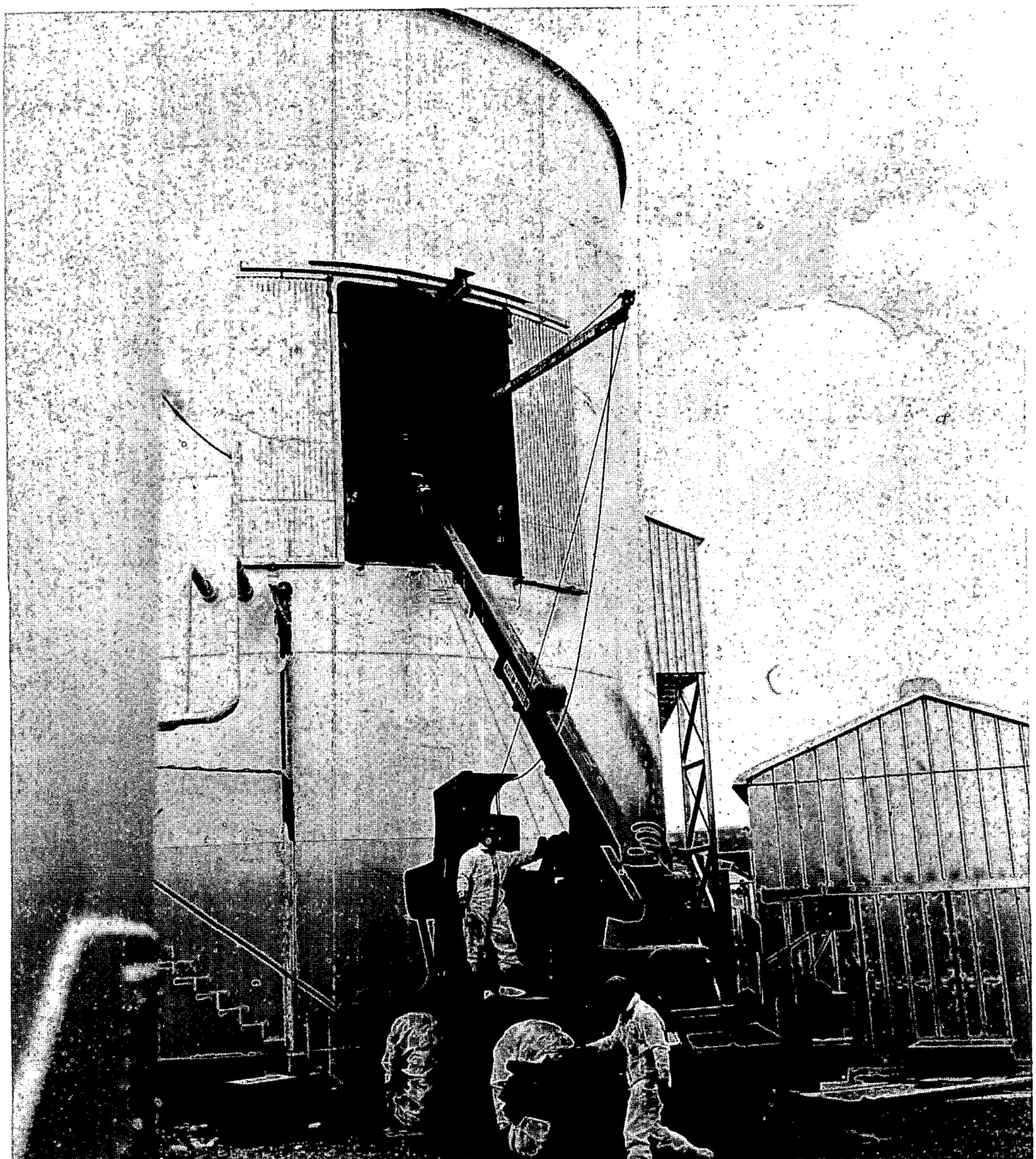


Figure 20 Performing Entry into the SL-1 Reactor Building

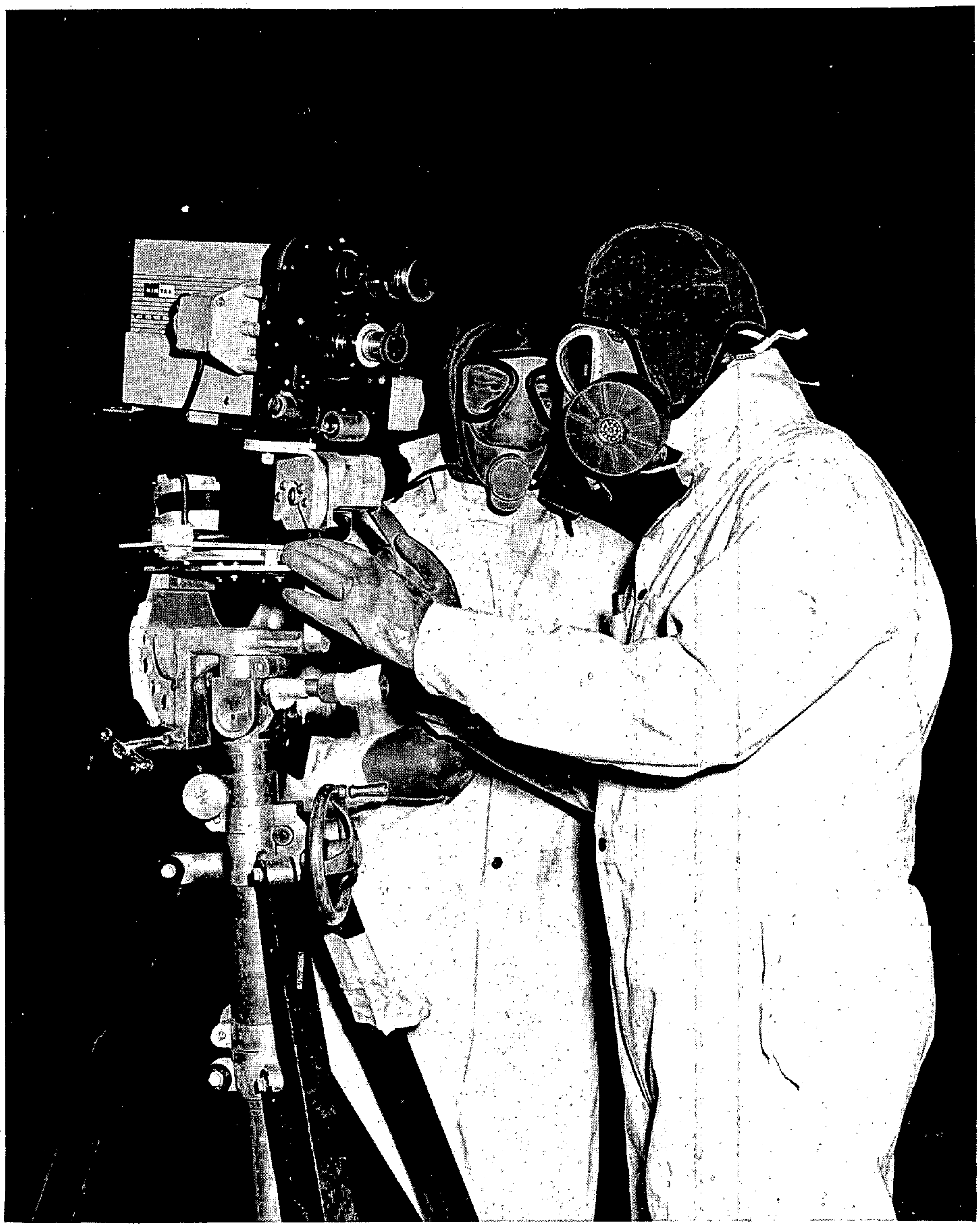


Figure 21 Recovery Crew Adjusting TV Camera

Figure 22 Minox Camera and Shielding Assembly

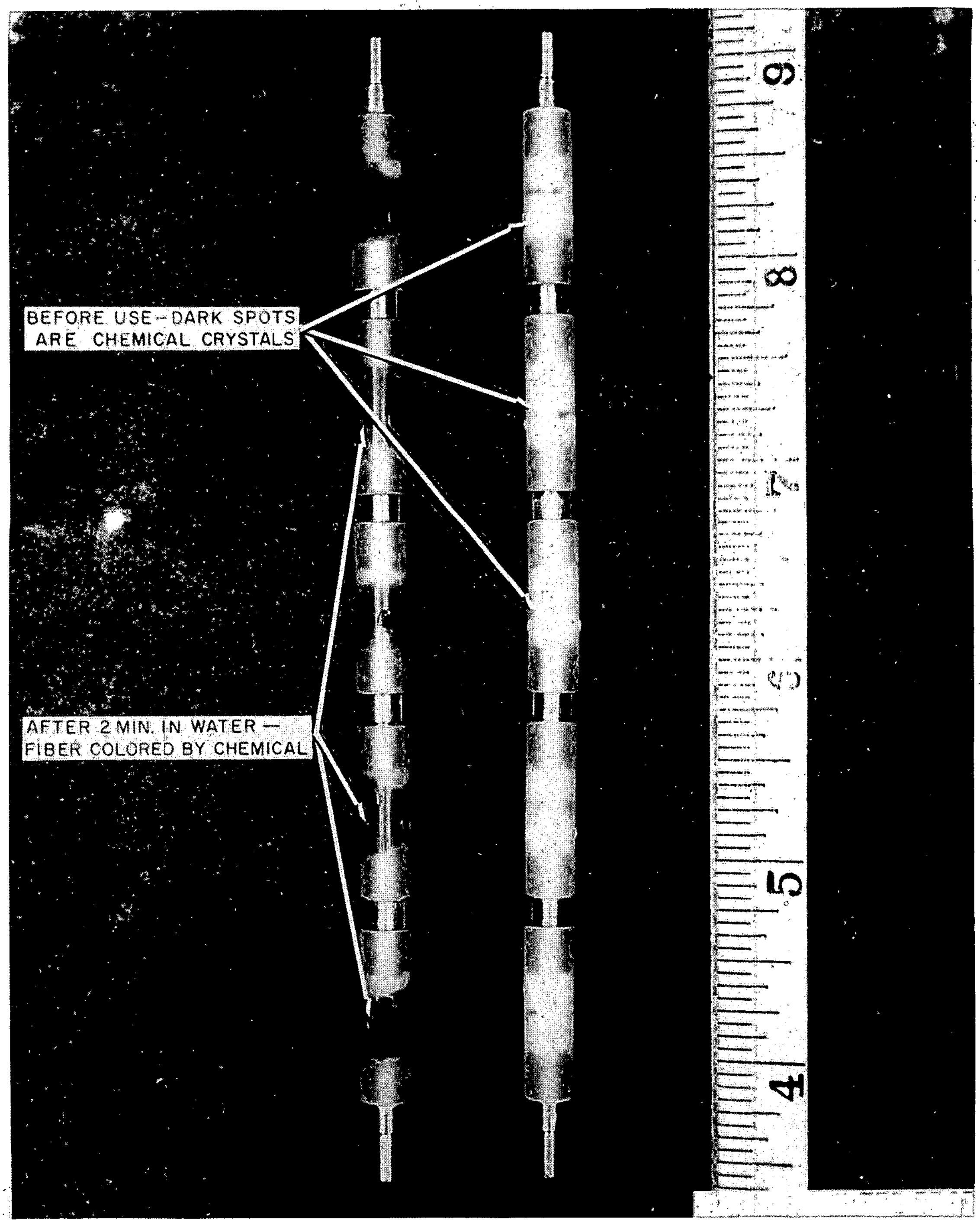


Figure 23 Chemical Water Probe (Section)

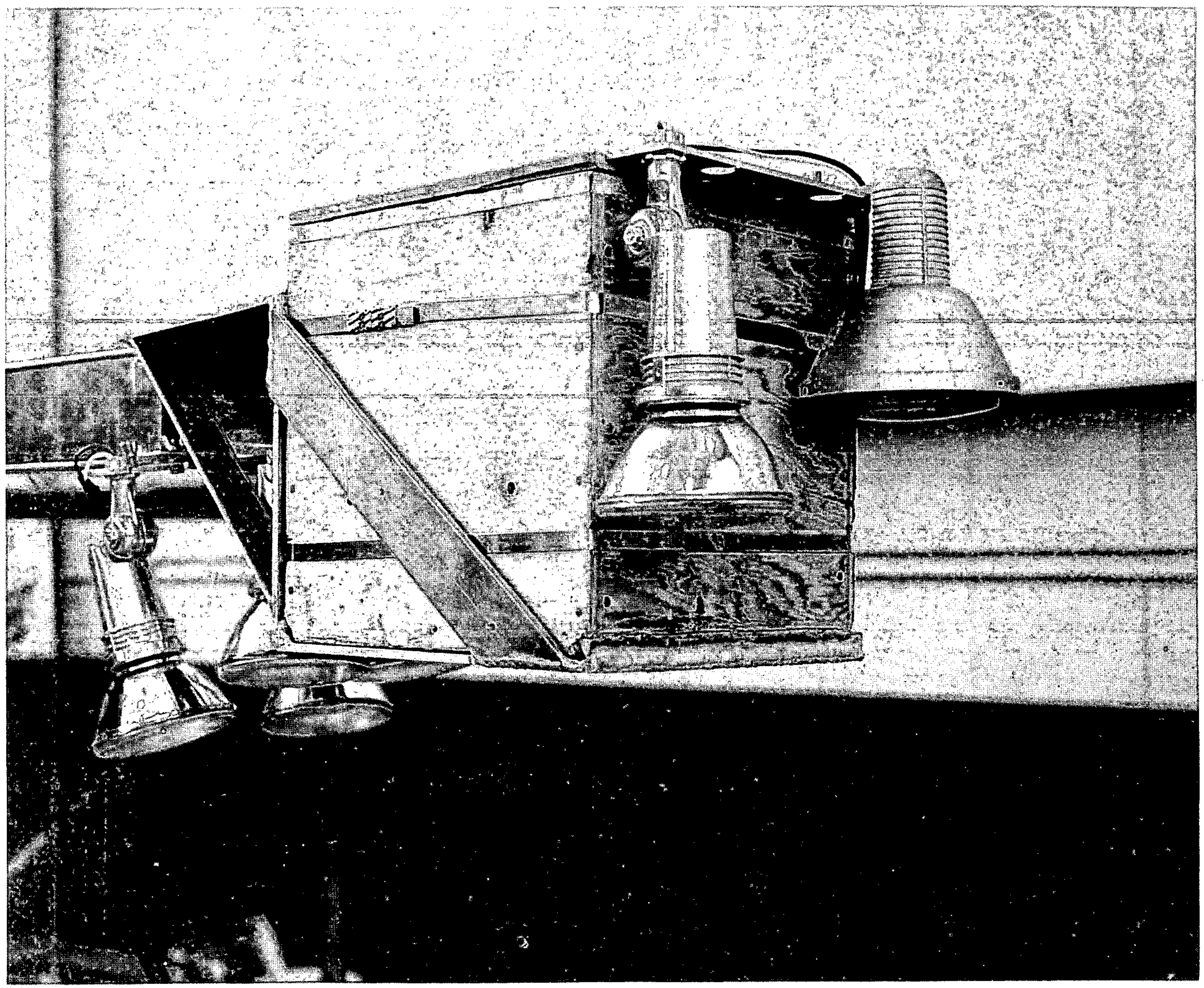


Figure 24 Shielded Movie Camera Mounted on Crane

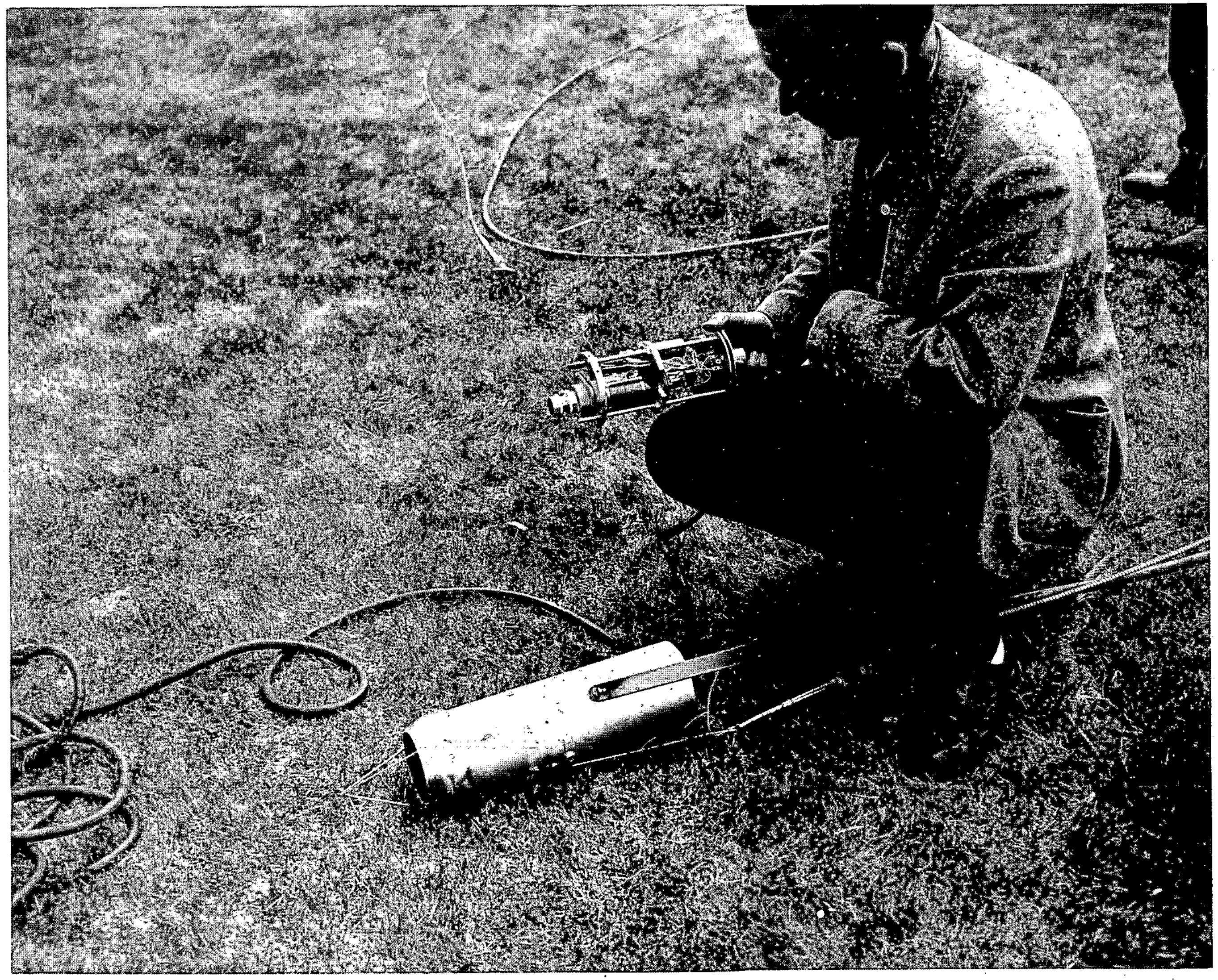


Figure 25 Special Barrel Mounted Television Camera

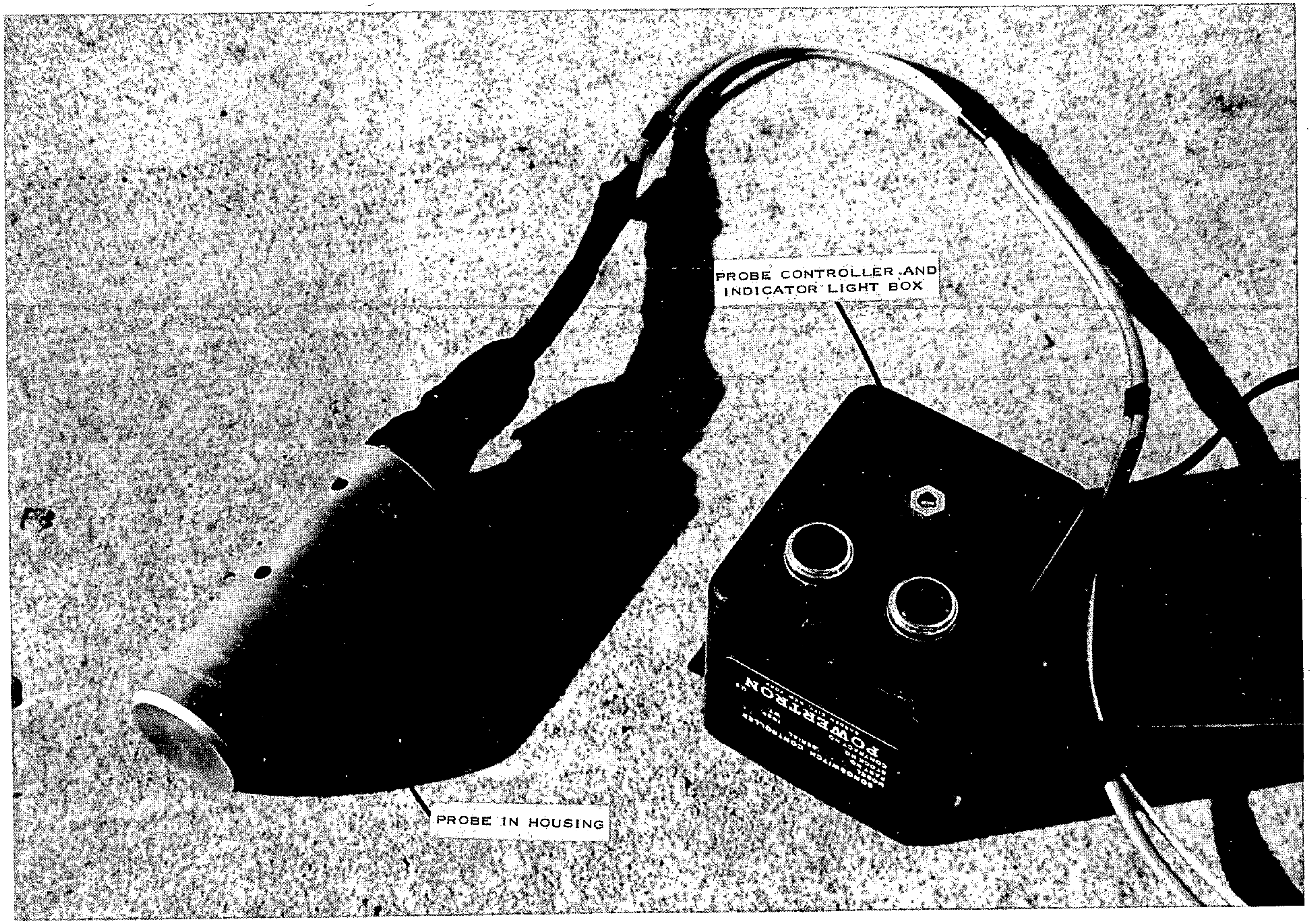


Figure 26 Ultrasonic Probe and Housing for Water Detection

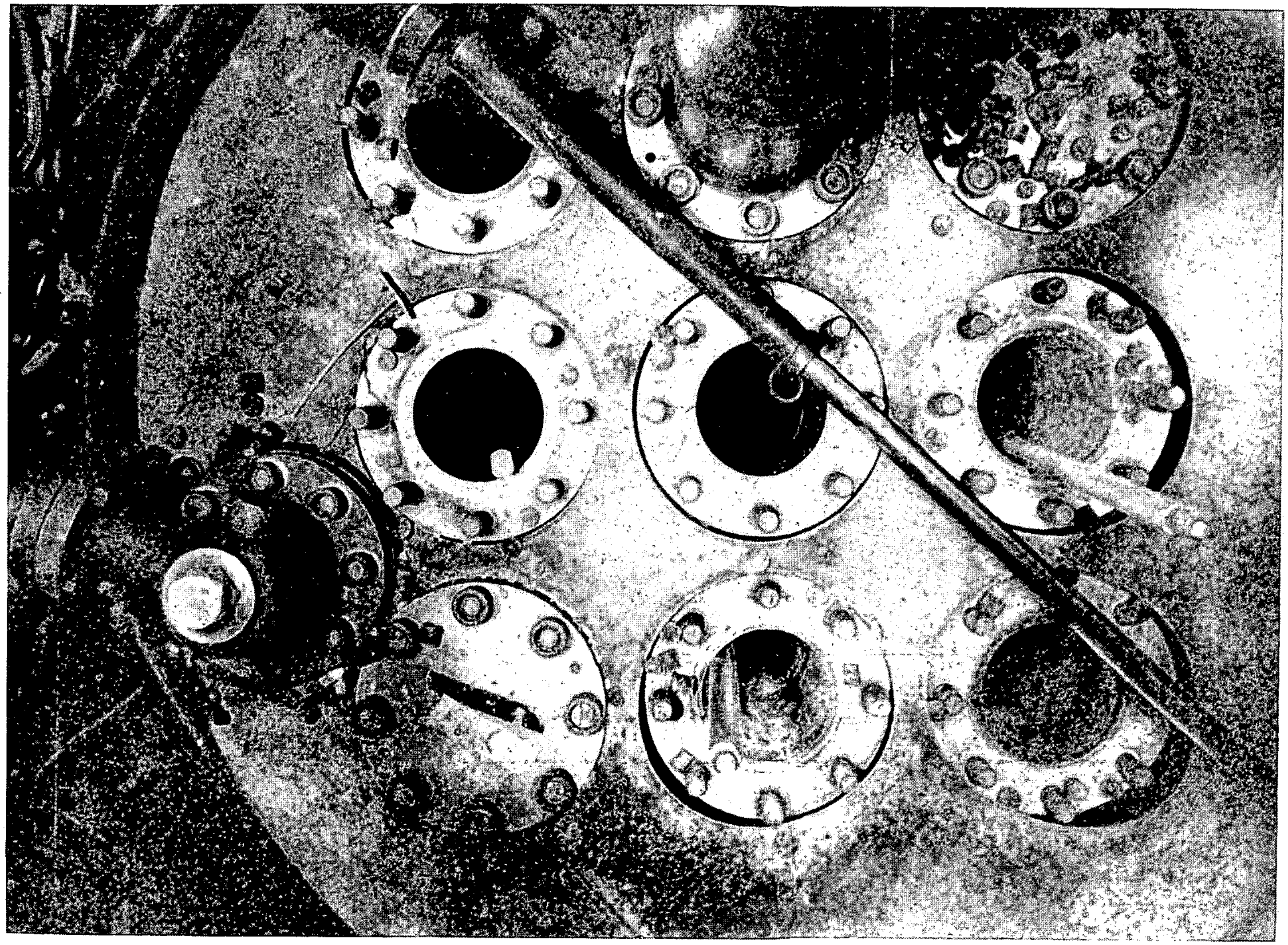


Figure 27 View of SL-1 Reactor Vessel Head After Incident

Figure 28 Photographic Evidence of Chemical Probe Penetrating Core Structure Through Control Rod Shroud No. 8

Figure 29 Pieces of Block Insulation on SL-1 Reactor Operating Floor

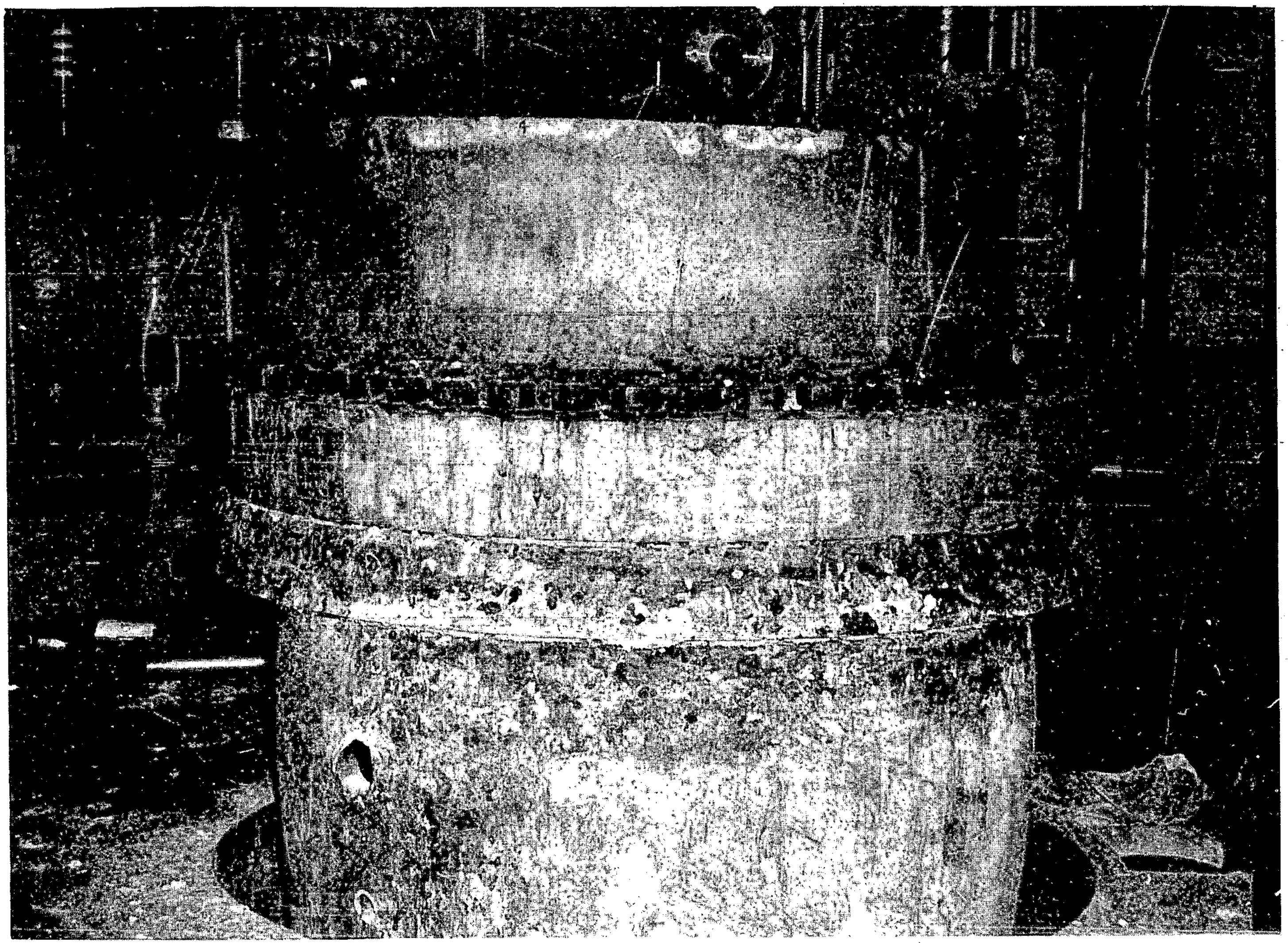


Figure 30 Trial Lift of Pressure Vessel Showing Distorted Flange, Bulged Vessel and Sheared Steam Line

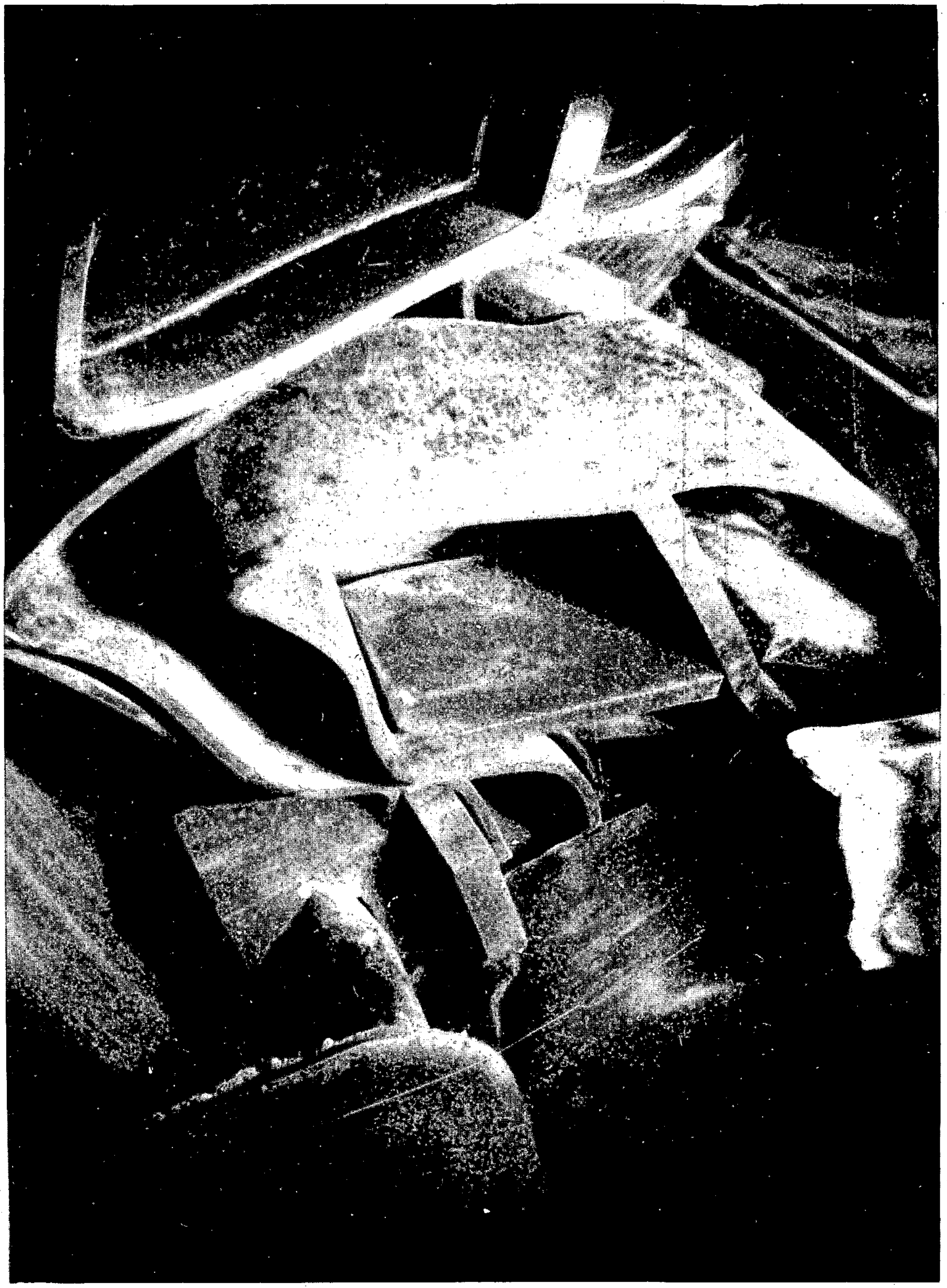


Figure 31 Underside of Core Showing Location Where Core Has Been Lifted from Support Bracket

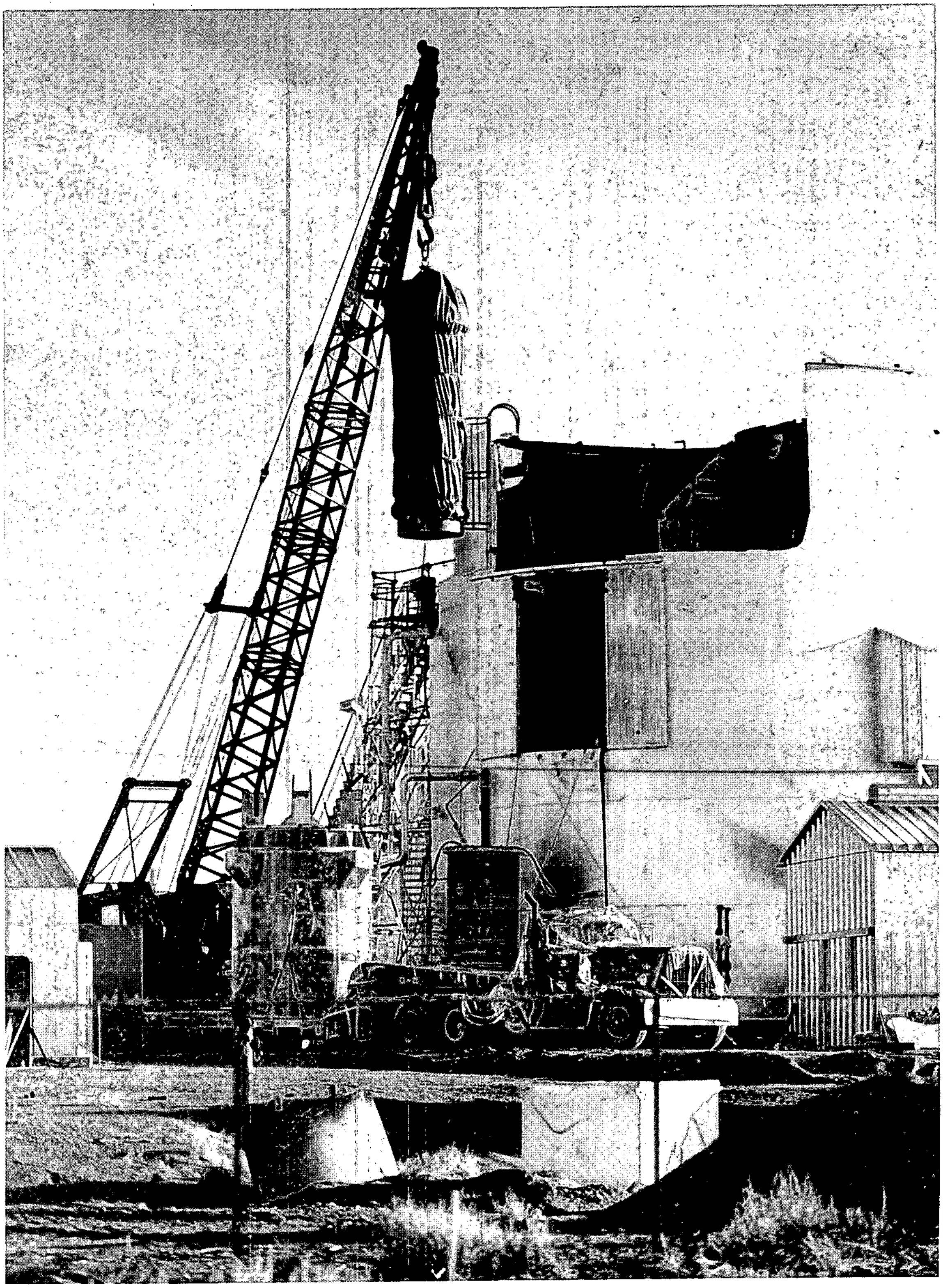


Figure 32 Pressure Vessel Being Removed from SL-1 Reactor Building

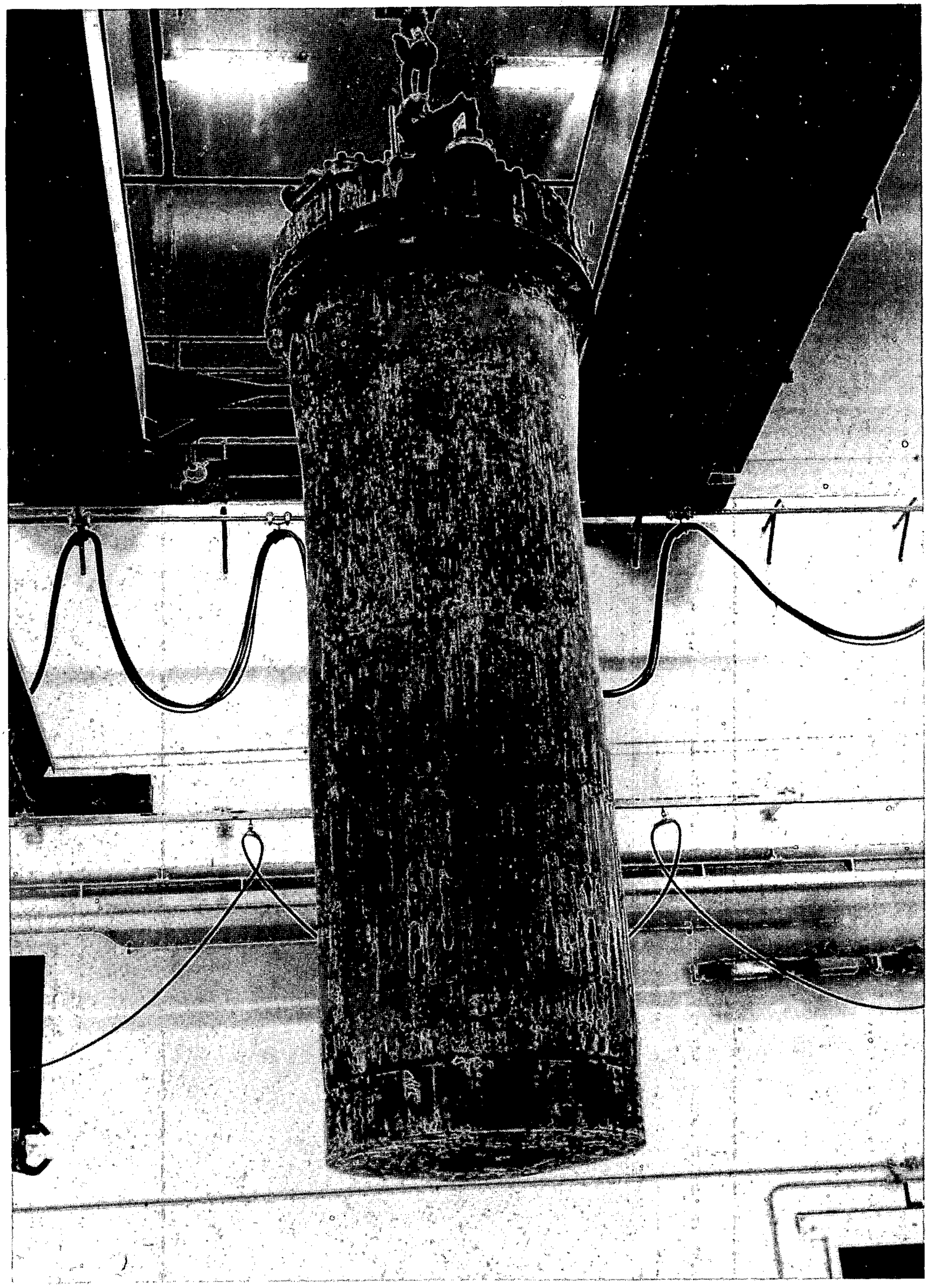


Figure 33 Outside of Pressure Vessel Before Bottom Skirt was Removed

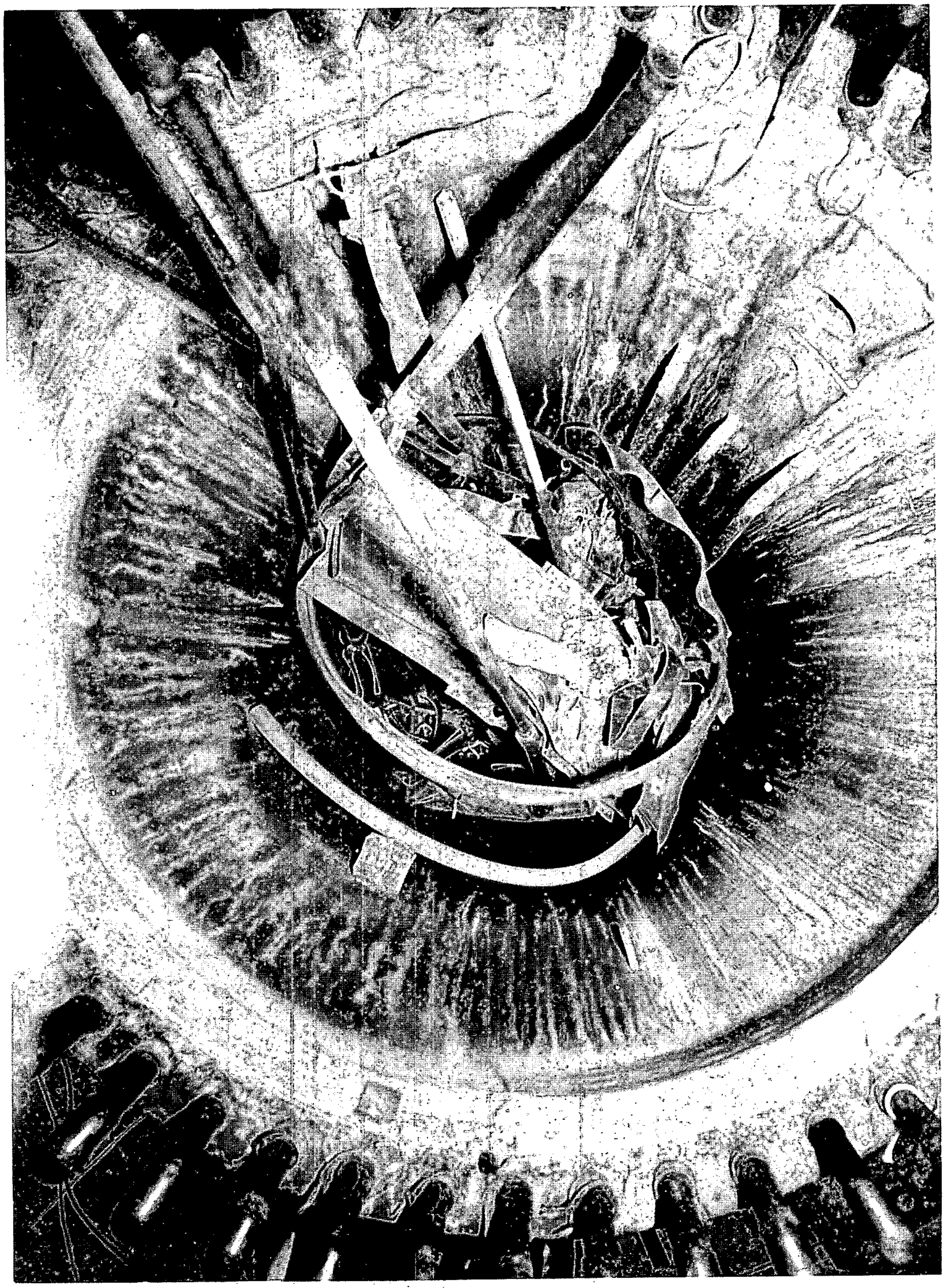


Figure 34 Inside of Pressure Vessel Just after Head was Removed - January 22, 1962

Figure 35   View of Core After the No. 9 Shroud was Removed

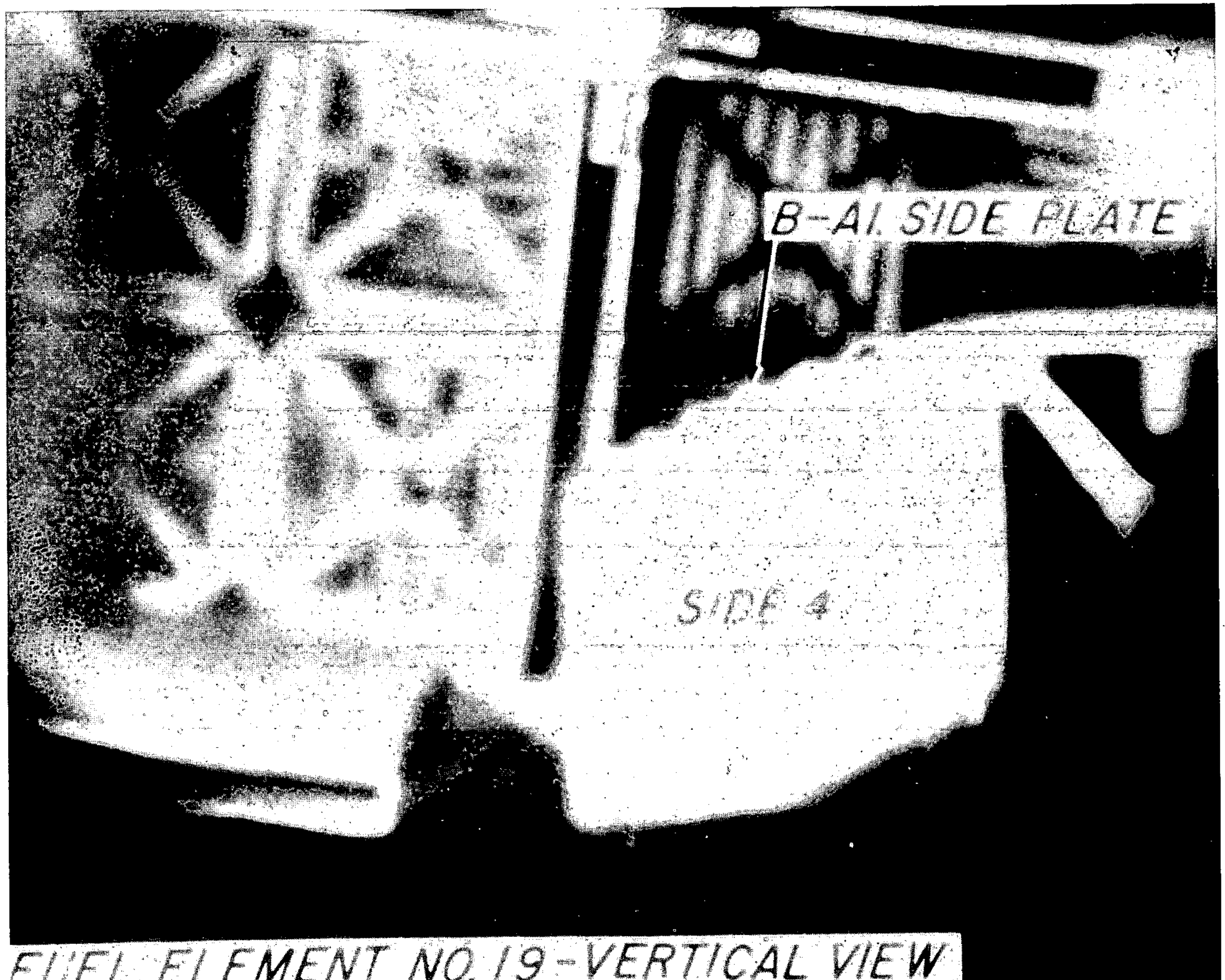


Figure 36 View of Boron Strip Bowing - August 1960

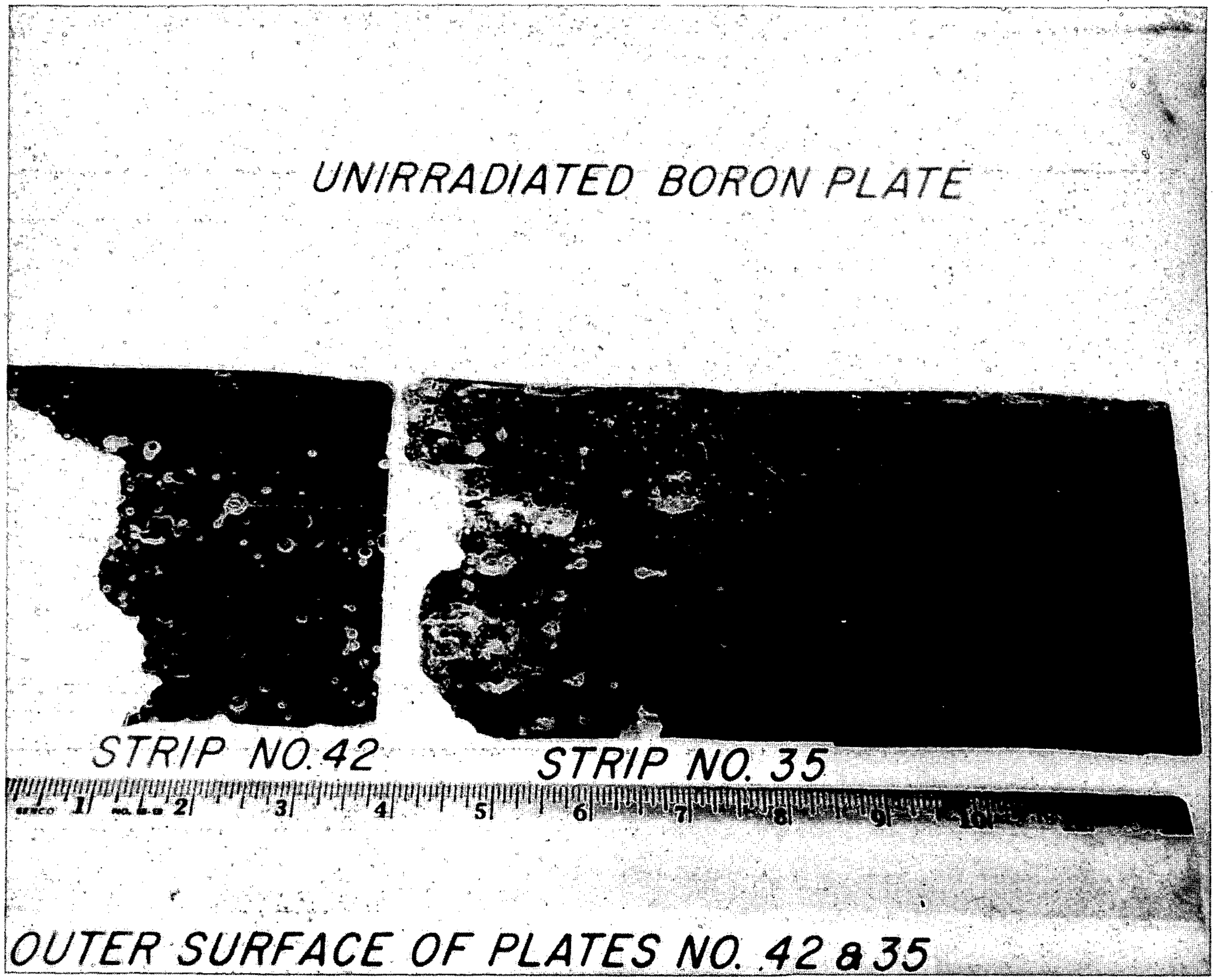


Figure 37 Comparison of Remaining Pieces of Boron-Aluminum Strips with Unirradiated Strip - August 1960

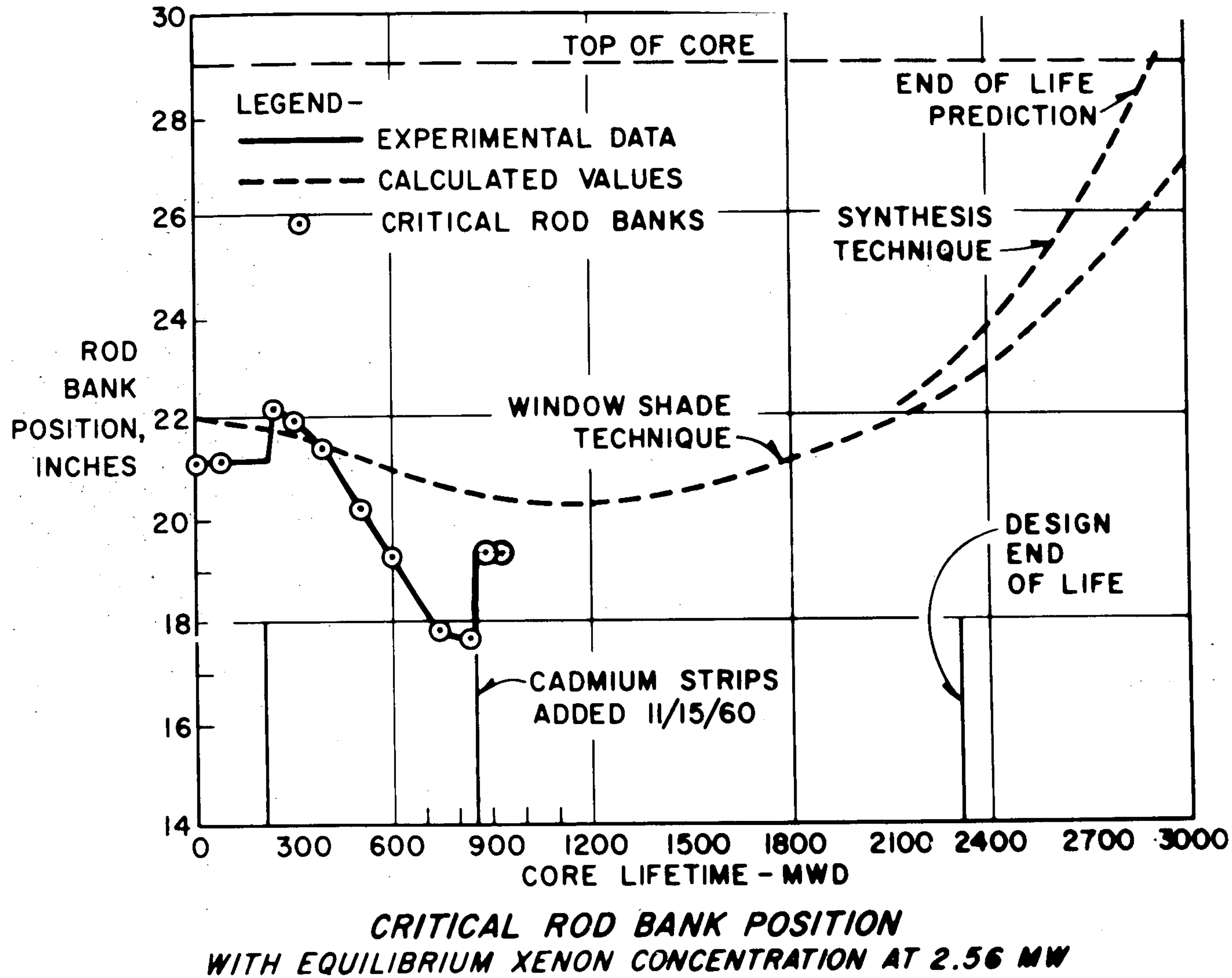


Figure 38 Critical Rod Bank Position with Equilibrium Xenon Concentration at 2.56 MW

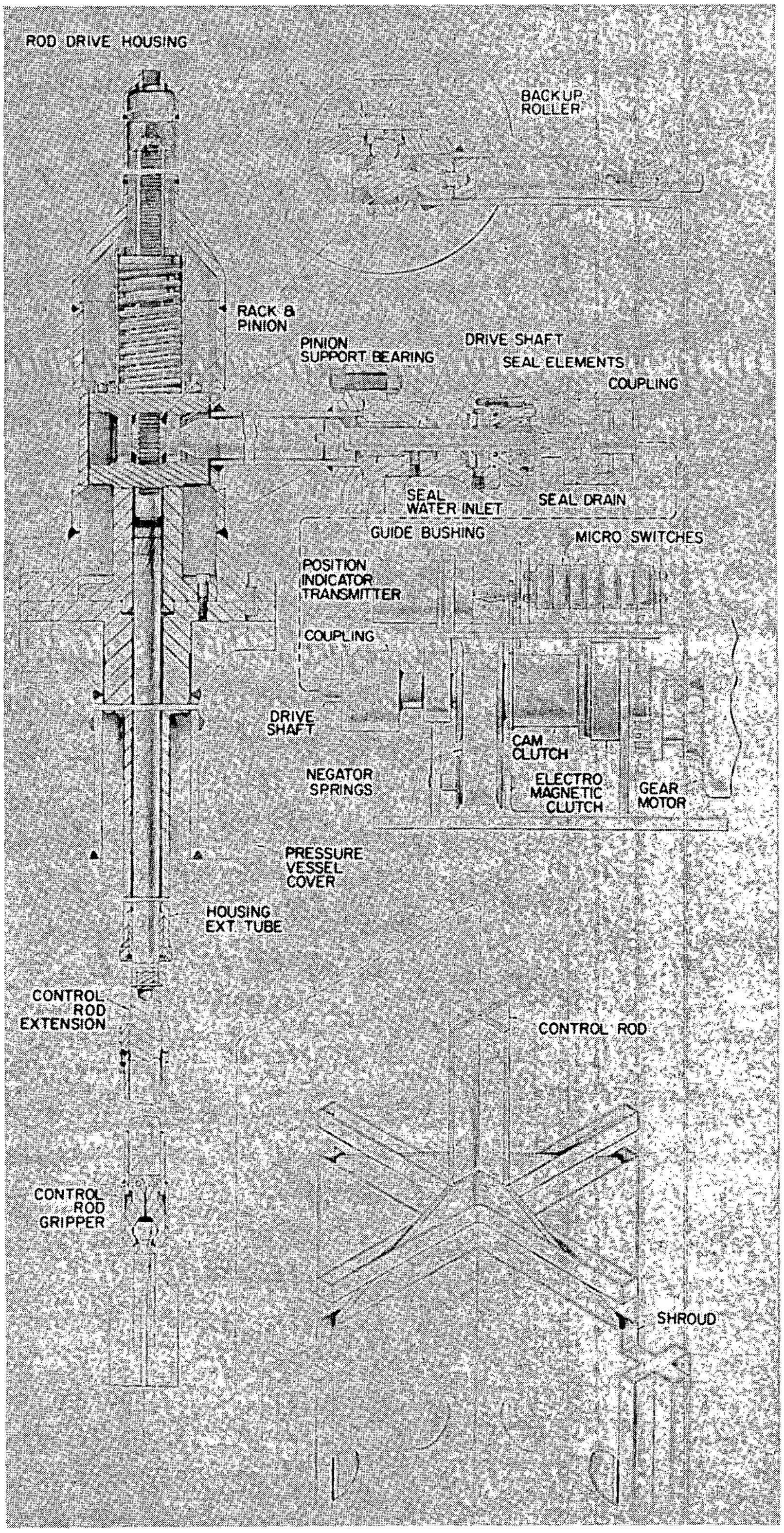


Figure 39 SL-1 Control Rod Drive Mechanism

Figure 40 Recovered Gear Rack Showing Transverse Stress Cracks on Flat of Rack

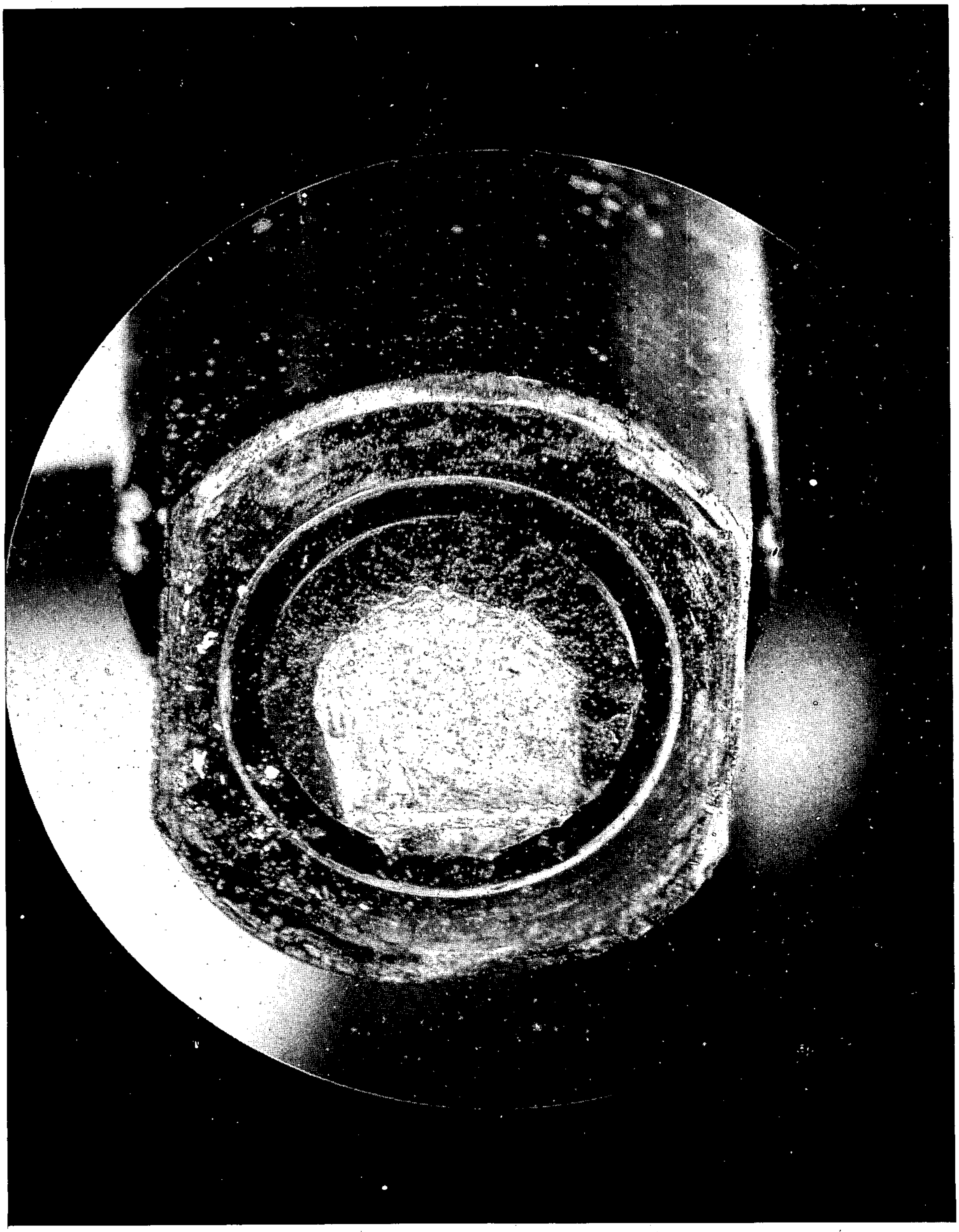


Figure 41 Recovered SL-1 Control Rod Extension Showing Fracture. Dark Outer Area of Fracture Caused by Pre-Accident Fatigue. Bright Area Caused by Impact Tensile.

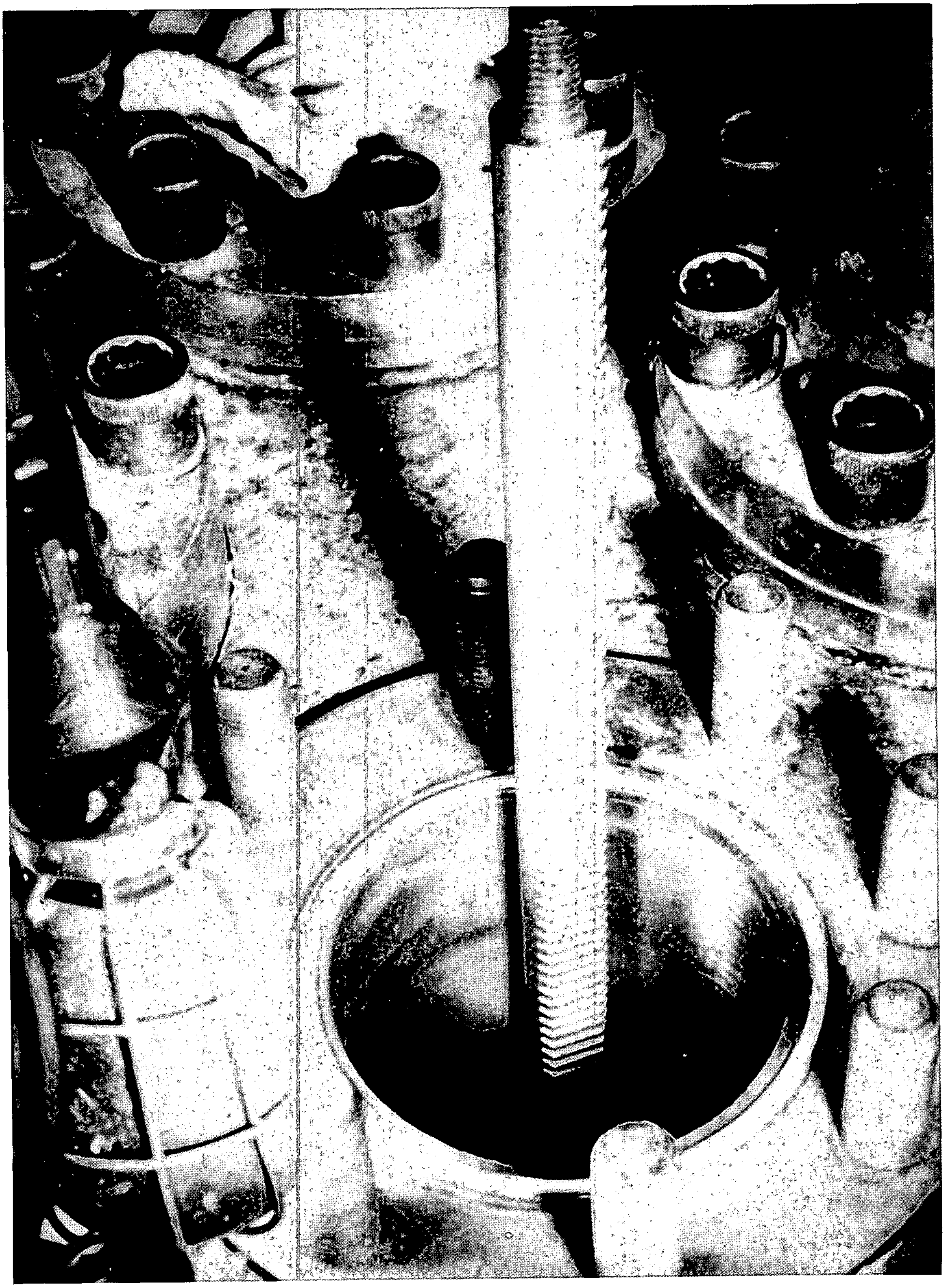


Figure 42 SL-1 Control Rod Drive Rack (Pre-accident photo)

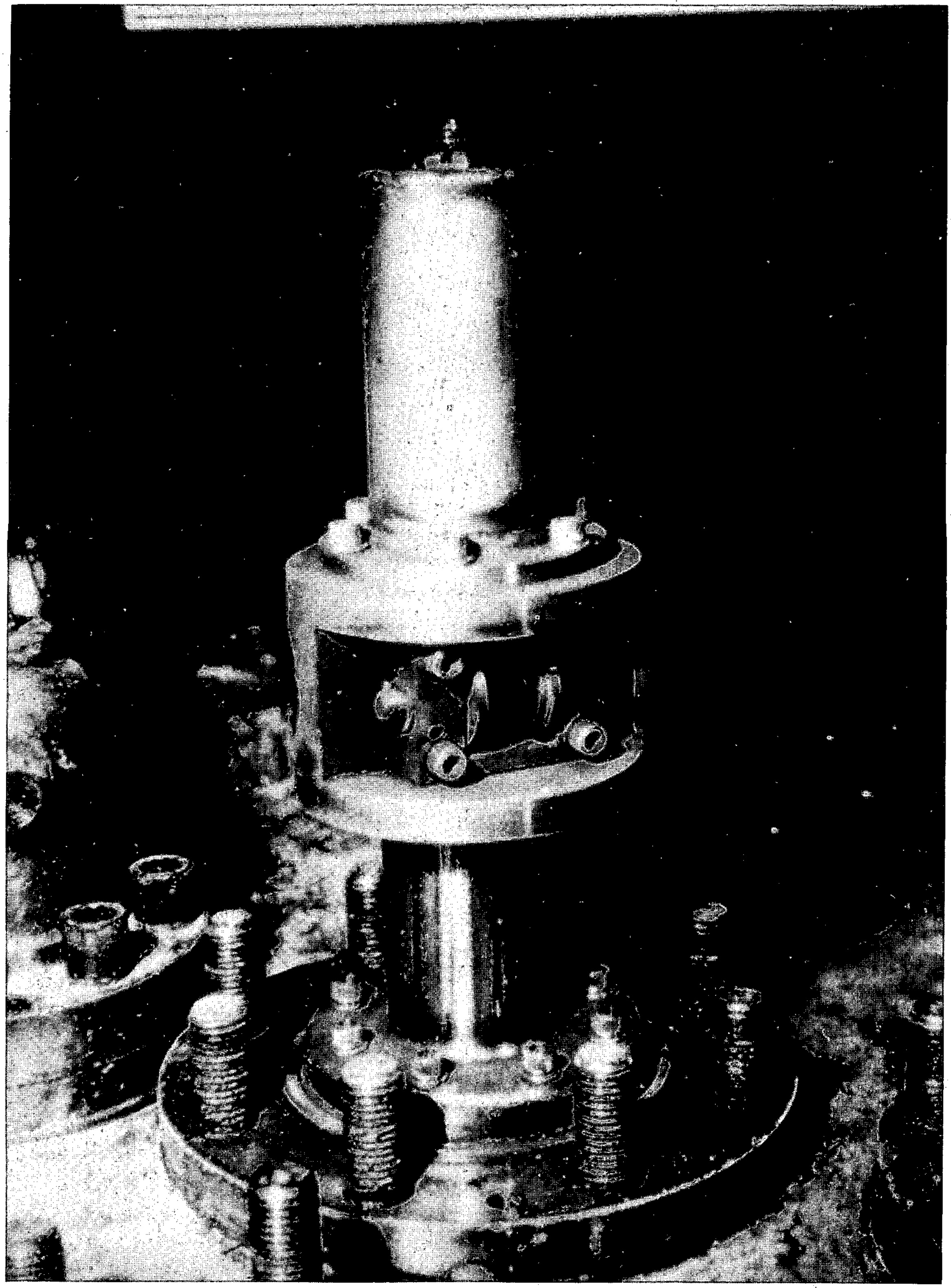


Figure 43 SL-1 Control Rod Mechanism (Pre-accident photo)

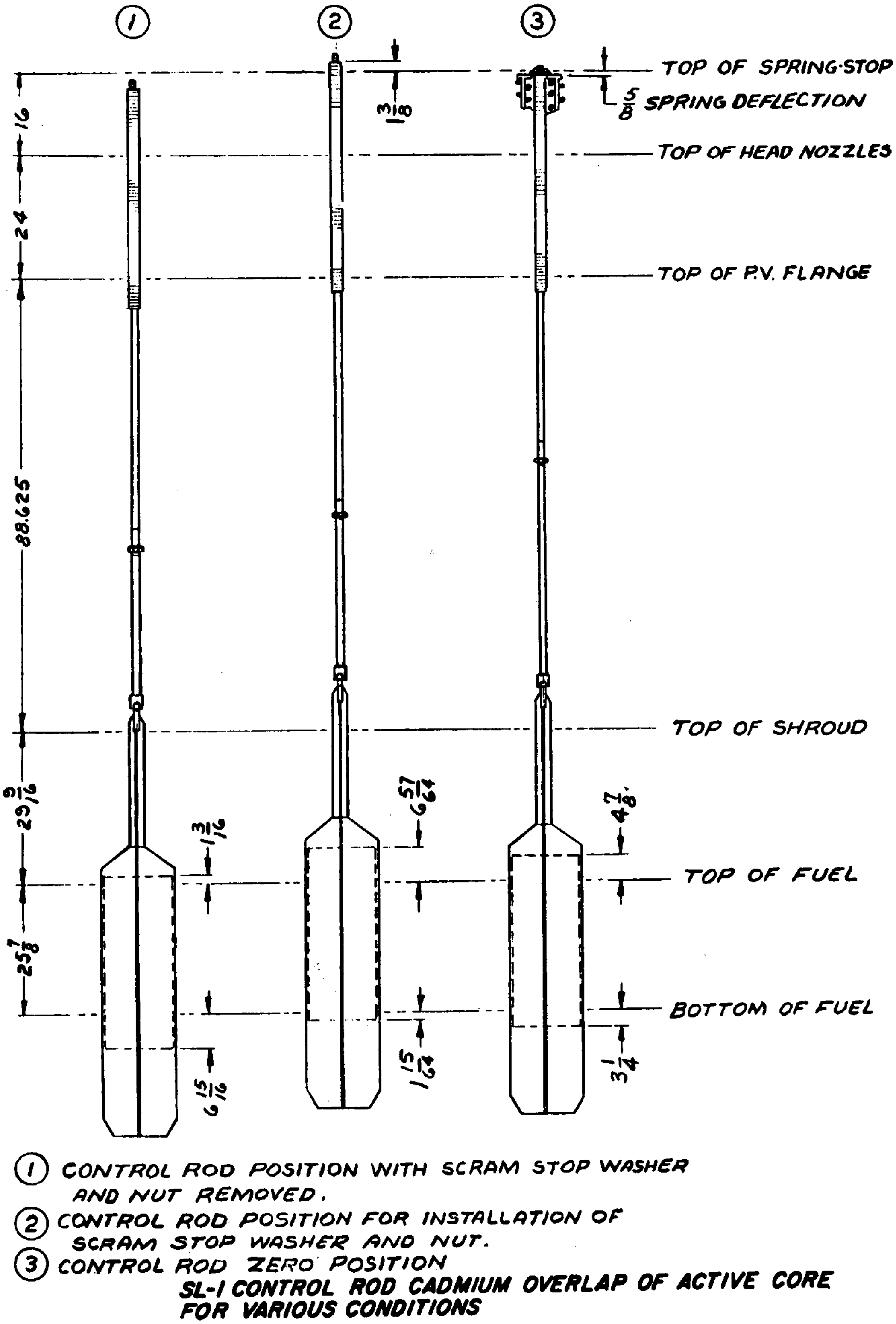


Figure 44 SL-1 Control Rod Cadmium Overlap in Active Core for Various Positions