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DESIGN-DEVELOPMENT AND OPERATION OF THE EXPERIMENTAL BOILING-WATER REACTOR (EBWR) FACILITY 1955-1967

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EBWR Decontamination and
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Plant Facilities and Services Division

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ABSTRACT

The **E**xperimental **B**oiling-**W**ater **R**eactor (EBWR) was designed, built, and operated to provide experience and engineering data that would demonstrate the feasibility of the direct-cycle, boiling-water reactor and be applicable to improved, larger nuclear power stations; and was based on information obtained in the first test boiling-water reactors, the BORAX series. EBWR initially produced 20 MW(t), 5 MW(e); later modified and upgraded, as described and illustrated, it was operated at up to 100 MW(t). The facility fulfilled its primary mission -- demonstrating the practicality of the direct-boiling concept -- and, in fact, was the prototype of some of the first commercial plants and of reactor programs in some other countries. After successful completion of the Water-Cooled Reactor Program, EBWR was utilized in the joint Argonne-Hanford Plutonium Recycle Program to develop data for the utilization of plutonium as a fuel in light-water thermal systems. Final shutdown of the EBWR facility followed the termination of the latter program.

CHRONOLOGY

The Predecessor BORAX Series

The first test boiling-water reactor, BORAX I (**Bo**iling **Rea**ctor **Ex**periment), built at Argonne National Laboratory's Idaho site, proved that a boiling-water reactor is self-regulating and inherently safe. The BORAX I was deliberately destroyed in 1954, to demonstrate that extraordinary measures were necessary to produce a damaging reaction.

BORAX II, the next development in boiling-water reactors, was a larger and pressurized version of BORAX I, constructed nearby in 1955.[1] A turbine generator was added to BORAX II the same year, and it was thereafter known as BORAX III. On the night of July 17, 1955, the Argonne staff, using temporary equipment, produced enough steam-generated electricity with BORAX III to light the town of Arco, Idaho.

BORAX IV was designed and put in operation in 1956, primarily to test the suitability of a new type of fuel element[2] and determine whether the thermodynamic behavior of that fuel would give results directly applicable to the EBWR (**E**xperimental **B**oiling-**W**ater **R**eactor).

The design of EBWR was based on information derived from the BORAX experiments. These experiments resulted in a reactor that was built for steady-power generation, to demonstrate the feasibility and safety of large-scale nuclear power plants using uranium fuels low in U-235.[3,4]

EBWR Goals and Accomplishments

The Experimental Boiling-Water Reactor was conceived as a means of providing experience and engineering data that would be useful in the design of improved and larger nuclear power stations, and of demonstrating the feasibility of the direct-cycle boiling-water reactor. It was designed to produce 20 megawatts of thermal power [20 MW(t)] and five megawatts of electrical power [5 MW(e)]; this size was considered large enough that the information obtained could be extrapolated to a central station power reactor, and small enough to minimize the costs of its construction and operation. It was built in 19 months: ground was broken in May 1955; the plant was operational in December 1956. Figure 1 shows the facility as originally constructed.

The energy provided to the ANL electrical grid by the experimental reactor was used on site. By agreement between the Laboratory and the local utility company (Commonwealth Edison) the EBWR could supply the Laboratory with all but 200 kW of electrical energy. The EBWR was run in parallel with the utility, so that the effect of the network characteristics on the generator, turbine, and reactor could be observed and factored into the design of future nuclear power plants. [Some of the first commercial plants were patterned after the EBWR, as were the reactor programs of some other countries.] The Laboratory load was approximately 4.5 MW of electrical energy. The plant operated through 1961 at the 20-MW(t) power level.

Experiments at power levels ranging from 20 to 40 MW(t)[5] indicated that stable operation could be achieved up to 66 MW(t) with the initial 4-ft diameter core. This was demonstrated by a short-term operation at 61.7 MW(t); but the higher limit was precluded by the feed water pumps, which were operating at maximum capacity. During that run, detailed studies were made of EBWR stability through a series of transfer function measurements relating flux or power level to reactivity input. Reactor and generator operating levels during the years of 1957 through 1959 are shown in the same order in Figures 2-4 (from Reference 5).

Analyses and projection of these data to a core of 5-ft diameter indicated that, with some modification of the core structure and pressure vessel internals, and with additional heat-removal equipment, EBWR could operate at or near 100 MW(t).

The facility was shut down and upgraded for operation at 100 MW(t).[6,7] The original 5-MW turbine was retained and the additional heat was dissipated through air-cooled heat exchangers. In addition to the system changes, the core was extensively instrumented. EBWR as modified for 100-MW operation is shown in Figure 5. After completion of the plant additions and modifications, the reactor went operational in 1962, and was tested for stability at successive power level increases up to 70 MW(t). Following careful analyses of the data, the reactor was operated successfully at 100 MW(t). It was operated at an average lower level of approximately 75 MW.[8,9]

After completion of the EBWR Water-Cooled Reactor Program, the AEC Division of Reactor Development requested that Argonne use the EBWR facility in the Plutonium Recycle Program. A joint Argonne-Hanford program evolved for the purpose of obtaining information useful for the utilization of plutonium as a fuel in light-water thermal systems.[10-13] The final shutdown of the EBWR facility followed the termination of the Plutonium Recycle Program in July 1967.

Although the EBWR plant was not large, compared with most modern nuclear power plants, the experimental facility did achieve its primary objective: to demonstrate the practicality of direct-cycle boiling-water reactors. It was primarily the "direct boiling"--the generation of steam within the reactor vessel and fed directly to the turbine--that initially distinguished the EBWR reactor from other types of power reactors.

During EBWR operations at up to 100 MW(t) and its shutdowns, a number of other technical points of interest were observed and/or investigated. These included:

Radiation levels, throughout the facility and at the various components.

The use of boric acid as additional poison in the core.

Locations of accumulated corrosion products.

Fission gas release.

Personnel work within a reactor vessel in modifying and adding internal structures and in maintenance operations.

DATA FACT SHEET: EBWR OPERATION

Activation Period,* Calendar Days	**3,920**
Average MW(t) Power over Activation Period	**5.38**
Gross Thermal Power Generation, MW-h	**506,563**
Gross Electrical Power Generation, MW-h	**56,367**
Effective Full-Power (5.38-MW) Hours of Operation	**94,157**

(From Refs. 8 and 11)

* For EBWR, an experimental facility, activation period encompasses date of reactor start-up through date operations were terminated before mothballing, and includes all necessary shutdowns for inserting/removing experimental apparatus, periodic checkouts and maintenance, and all facility modifications required for the various programs.

OPERATIONAL DETAILS

A. 20-MW (THERMAL)

Steam-and-Condensate Cycle

A cutaway pictorial view of the EBWR reactor and its components is shown in Figure 6.

During the nuclear reaction, the fissioning of uranium atoms converted part of the water within the core to steam. The core was comprised of zirconium-clad uranium fuel elements, hafnium and borated-steel control rods, and demineralized water. The saturated 600-psig (489^{0}F) steam was fed to a turbine generator.

Spent steam went to a condenser, from where the condensate was returned to the reactor pressure vessel. Water circulated through the reactor core by natural convection.

Direct steam flow between the reactor and the turbine eliminated the need for costly heat exchangers; thus, the reactor and the turbine operated at the same steam pressure. All the equipment in the plant was designed for continuous remote-control operation and incorporated means to prevent the escape of radioactive water or contaminated material within the containment building. The steel containment building was the tertiary containment for the fuel.

Components

Pressure Vessel--The reactor proper was located within this vessel of 7-ft inside diameter and 23-ft internal height, with carbon steel walls 2-3/8 inches thick. Its inner surfaces were clad with 0.109-in. stainless steel sheet. The vessel was surrounded by a heavily-shielded concrete monolith, such that the top of the reactor was just below the main floor of the power plant (ground level). The vessel was supported at its upper end by springs resting on a steel frame that was itself fastened to steel columns. When the reactor was operating, the heat generated by the fission process raised the water temperature to 489^{o}F, the saturation temperature of steam at 600 psig. A boron-stainless steel thermal shield was installed between the core and the wall of the reactor vessel to minimize the effect of heat generated in the reactor vessel by neutron bombardment. Figure 7 is a diagram of the EBWR pressure vessel and its internals.

The Core--This component, inside the pressure vessel, consisted of fuel assemblies and control rods fitted into a support shroud structure that was bolted to the bottom of the vessel. The control rod guide and the top shroud structure that could accommodate up to 148 fuel assemblies within the 5-ft diameter support plate are shown in Figure 8. The first core loading, approximately 4-ft diameter, consisted of 114 assemblies, of which 106 contained enriched (1.44% U-235) uranium and the remaining eight contained natural uranium. The core contained a total of 6.1 tons of uranium, averaging 1.4% U-235. Dummy assemblies were used to fill the core to the 5-ft diameter. The space between the core and vessel wall was the downcomer space for water circulation. The shroud structure served a threefold purpose: holding the upper ends of the fuel assemblies; providing additional riser height above the assemblies, to enhance natural circulation; and serving as the guide for the nine movable control rods within the core.

Fuel Assemblies--Each fuel assembly was 77-5/8 in. long and 3-3/4 in. square. An assembly was comprised of a lower locating fitting, six fuel plates, two side plates, and a top fitting.

The fuel plates were uranium-zirconium-niobium alloy sheets, manufactured in two thicknesses (0.214 in. and 0.279 in.) and two types (normal 0.72% and 1.44% U-235). The uranium alloy was clad with Zircaloy-2 sheet. The six fuel plates were arranged parallel to one another, with water channel space between adjacent plates. Perforations in the side plates permitted increase in the length of fuel plates (creep) caused by irradiation. Dummy fuel assemblies were made of 0.0625-in. thick aluminum-nickel alloy sheet. A dummy fuel assembly and a cutaway view of its integral holddown assembly (which enables the unit to withstand the expulsive force) are shown in Figure 9.

Control Rods--Nine control rods made of neutron absorbing metal were used to control rate of change of the nuclear chain reaction. Hafnium was used as the absorbing material in five of the rods, which were located in the most effective (central) position. The other four, corner rods, were made from stainless steel containing 2% boron. Figure 10 depicts the Hafnium-Zircaloy 2 rod, and Figure 11, the boron-stainless steel rod. The chain reaction was started by raising the absorbing material out of the active region. The power level was regulated by adjusting the position of the rods.

The control rods were installed through the top of the vessel, but operated from the bottom through labyrinth seals on the lower ends of thimbles beneath the pressure vessel. During operation, the absorber section of the rod was driven up out of the core, followed by a Zircaloy-2 follower section that replaced the absorber in the guide channel. Release of the rods and rapid reinsertion for reactor shutdown took less than one second from full-out position.

Shielding--The shielding around the reactor vessel was designed for operation at 40 MW(t), using heavy water as the moderator. The heavy water operation option was never used in EBWR.

Fuel Handling System--Irradiated fuel assemblies were transferred in a lead-shielded cask from the reactor to the water-filled storage well. Removable plugs in eccentric ports in the reactor top shield plug afforded access to each fuel assembly that could be raised into the cask. A leak-proof door on the bottom of the cask was then closed, so that the cask could have been filled with water if it was necessary to cool the fuel during the transfer. The storage well to which the units were transferred was a water-filled pit, 25 ft deep, adjacent to the reactor. There, the units could be cooled until ready for reprocessing.

Turbine-Generator--The turbine was of conventional design, except with respect to the shaft seals and provision for collecting condensate leakage. The generator was also essentially of conventional design; it generated 5 MW of electricity at 4160 volts.

Condenser--This component was constructed with double tube sheets and divided water boxes, as a precaution against the escape of radioactivity from the system, as well as against the possible dilution of heavy water with light water if heavy water operation were used in the plant in later years. The tubes were aluminum, to minimize radioactivity induced in corrosion and erosion particles. In other respects, the condenser was of conventional design. Two pumps circulated approximately 14,000 gpm of cooling water to a cooling tower outside the containment building.

System Bypass Control System--The reactor was ordinarily operated at constant power and steam-flow rate. In order to minimize necessity to regulate reactor power in response to fluctuations in electric power demands, power fluctuations were controlled by means of an automatic steam control valve that opened and closed to maintain the desired power level.

Desuperheater--The steam from the bypass valve moved through a desuperheater before entering the condenser. The desuperheater, which extended across the condenser below the condenser tubes and above the water level in the divided hot well, was a steel pipe of 14-in. diameter, fitted with two water spray nozzles. Usually, when both halves of the condenser were in operation, the spray system was not used. The condensate water cooled the steam and prevented occurrence of high temperatures in the shell and tube surfaces. If the condenser were operated with cooling water flowing in only half the tubes, however, entry of superheated steam into the drained half could stress and damage the uncooled tubes. The automatic operation of the desuperheater spray valves prevented such damage. In addition to the automatic control, electrical initiation of the spray valves was provided for use at the operator's discretion.

Relief and Safety Valves--To protect the reactor vessel in the event of accidental overpressure, a relief valve and two safety valves were piped to discharge into the desuperheater. These valves were integral parts of the closed steam system. If any of them opened, the reactor would automatically shut down.

Water Purification System--Before feed water (condensate from the condenser hotwell) entered the reactor, it passed through disposable cotton filters, for removal of corrosion and erosion particles. For economic reasons, not all materials in the steam/feed water cycle were corrosion resistant. Two parallel full-flow filters were provided: normally, one was in service and the other on standby. Contaminated (radioactive) filter cartridges were removed to either the cave building for cartridge replacement or to waste management for disposal.

The continuous removal of corrosion products from the water in the reactor vessel was the function of the reactor purification system. That all-stainless steel system removed water from the bottom of the reactor vessel through one of four 6-in. recirculation lines and delivered it, after cooling, to the ion exchange columns. The purified water was pumped back to the reactor vessel with the feed water.

Air Drying and Fluid Recovery System--This system served a threefold purpose:

(1) minimize in-leakage to the steam and condensate system,

(2) recover most out-leakage of radioactive gases and vapors, and

(3) minimize contamination and loss of heavy water and return this in-leakage to the steam system. (The heavy water was never used.)

The system was designed to hold the loss of working fluid for the entire plant to less than two pounds per day.

Steam Dryer/Emergency Cooler--This unit was installed in the main steam system, between the reactor and the turbine. During normal operation of the reactor, the unit's prime function was to dry the steam leaving the reactor, as well as to minimize radioactive carryover. During an emergency, it could cool reactor water by using water from a 15,000-gallon overhead storage.

Control

The entire EBWR Facility--comprising both the steam plant and the reactor--was operated by remote control. This required an intricate system of instruments, alarm signals, and control mechanisms, with all communication elements located in the control room adjacent to, but outside the power plant building.

From a station at the console panel in the control room, the operator maintained constant vigilance over the various meters and signals that showed conditions throughout the plant, and appropriate adjustments were made as needed. For safe steady operation, automatic controls were provided for many of the variables. The apparatus in the control room were connected to components in the plant by means of approximately 1200 electrical cables, passed through gas-tight ports in the containment shell.

Structure

The containment structure that housed the power plant was a steel shell, half underground, 80 ft in diameter, 120 ft in height (60 ft below, 60 ft above ground), providing protection against the escape of radioactivity to the atmosphere in the event of damage to any part of the system. The shell was capable of withstanding 15 psi internal pressure, and retaining that amount of pressure for a considerable time when sealed. The service building, adjacent to the shell, housed the remote control equipment, offices, and certain auxiliary equipment.

The power plant building contained most equipment essential to operation, including the 15,000-gal water storage tank, suspended under the dome of the structure. The stored water was to be used for cooling the reactor water in the event of an emergency. Whenever the water level in the storage tank dropped, additional water was added automatically. Where it was necessary to pass cables, pipes, etc. through the walls of the power plant building, pressure-tight seals were installed. Personnel access was through double-door air locks. If radioactivity were detected in the building atmosphere, the ventilating system was automatically sealed.

The steel shell was protected by a thick concrete liner, or missile shield, against possible damage by flying fragments from any rotating machinery that might break apart while operating.

B. 100-MW (THERMAL) OPERATION

Experiments at design power level, and at higher power levels, which were part of the EBWR program, led to the prediction of stable operation at power levels as high as 66 MW(t) with the initial 4-ft diameter core.

On March 20, 1958, the reactor was operated successfully at 61.7 MW(t). At that point,the power was limited by the fact that the feed water pumps were operating at maximum capacity. Analysis of the data obtained indicated that a 5-ft diameter core would be stable at an operating level of 100 MW(t).

To operate the reactor at that level, it was necessary to modify the facility in a number of ways, so as to handle the increased steam load. The major changes and additions that were made are described in the following sections.

Reactor Vessel Modifications

The 6-in. diameter steam outlet was enlarged to 10-in. diameter. Two 1-1/2-in. gage nozzles were added. A 6-in. condensate return line with a 4-in. pipe distribution ring was also added.

The steam lines outside the reactor vessel were replaced with piping of equal diameter, and a third 2-in. instrument nozzle was installed in the stud ring forging.

Changes Within the Reactor Vessel

The initial 4-ft high metallic fuel plate assemblies were supplemented with 32 highly enriched rod-type ceramic pellet assemblies of 4-ft active length. This made a core 5 ft in diameter.

Nine new control rods featured a 5-ft absorber length and fueled followers, so that future operations with a core 5 ft high could be handled.

The 4-ft lead-screw control rod drives were replaced with rack-and-pinion type drive mechanisms designed for 5-ft travel.

A conical core riser and its extension were fitted on the 5-ft diameter core shroud, to permit a better driving force for natural circulation, help collapse entrained steam bubbles, and reduce pressure drop.

A steam duct was welded to the steam outlet nozzle in such a way that steam was collected from the highest point in the reactor vessel.

A chemical control system using boric acid was installed in the plant, primarily to provide an 8-rod shutdown, which was a requirement for rod drop tests, maintenance, and the possibility of rod failure. It also provided an additional margin of safety beyond the 9-rod shutdown.

Turbine Plant

Although the turbine plant was kept intact, many additions and modifications associated with the reboiler plant were made within the existing facility. Five major items installed in the containment building to meet reboiler plant and higher-power operation requirements were:

2 additional full-flow feed water filters
1 deaerator
1 subcooler
2 additional reboiler plant feed water pumps
1 large-capacity instrument air compressor

A 16-ft control panel was installed in the control room to accommodate the requirements of the reboiler plant and the operating controls for the newly-installed equipment for 100-MW operation.

Heat-Dissipating Equipment

The 20 MW(t)-capacity plant was modified to handle an additional heat load of 80 MW(t). This was accomplished by adding a reboiler plant and air cooling fans to dissipate the heat. No change was made to the 5-MW(e) output of the turbine generator, which used 20 MW of thermal power.

The turbine plant was to be operated in essentially the same manner as originally designed, and the reboiler plant was operated in parallel, to accept the increased heat output.

An underground piping tunnel was built to connect the EBWR containment building with the reboiler building. Figure 12 is a flowsheet of the modified facility.

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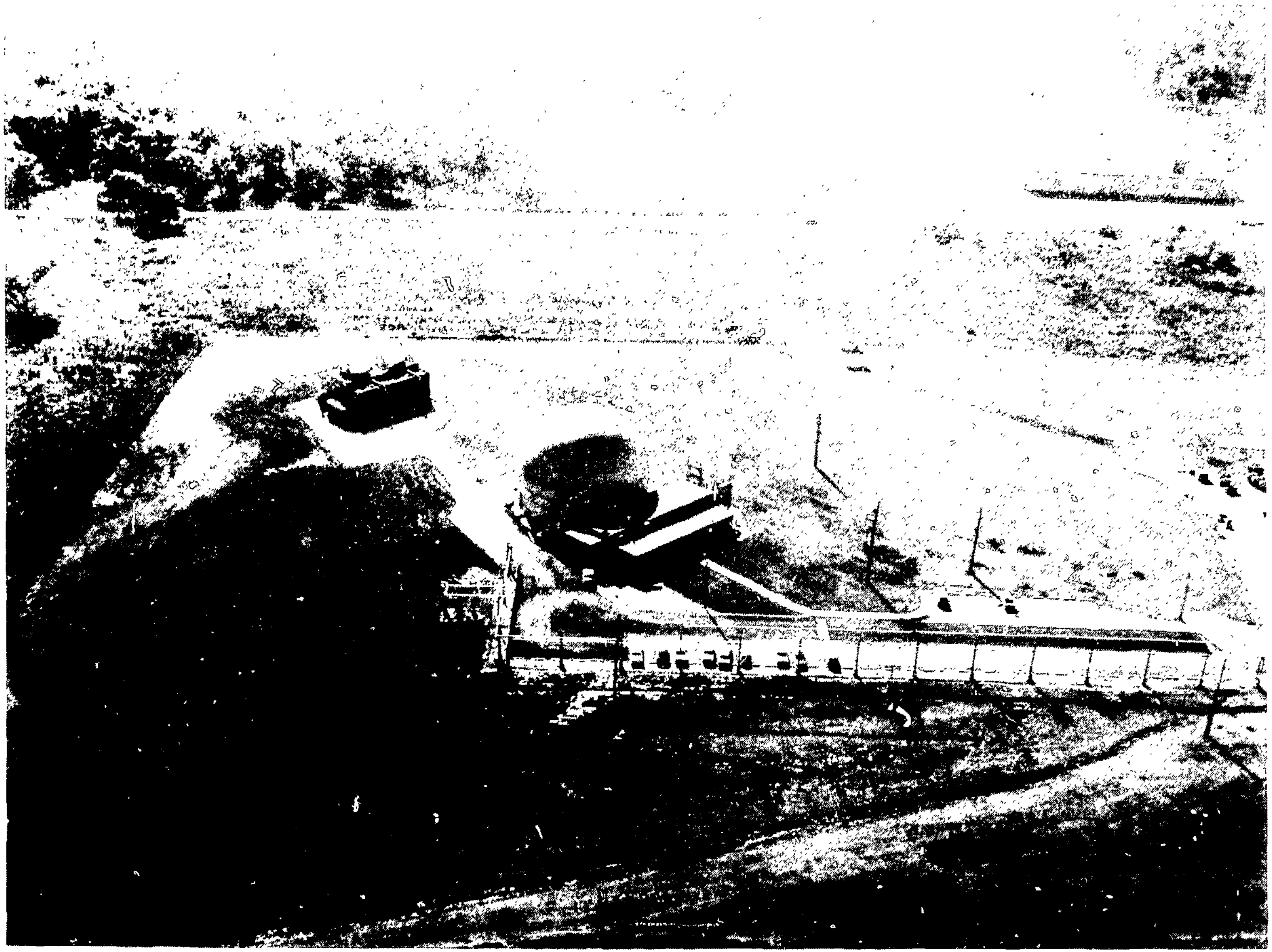
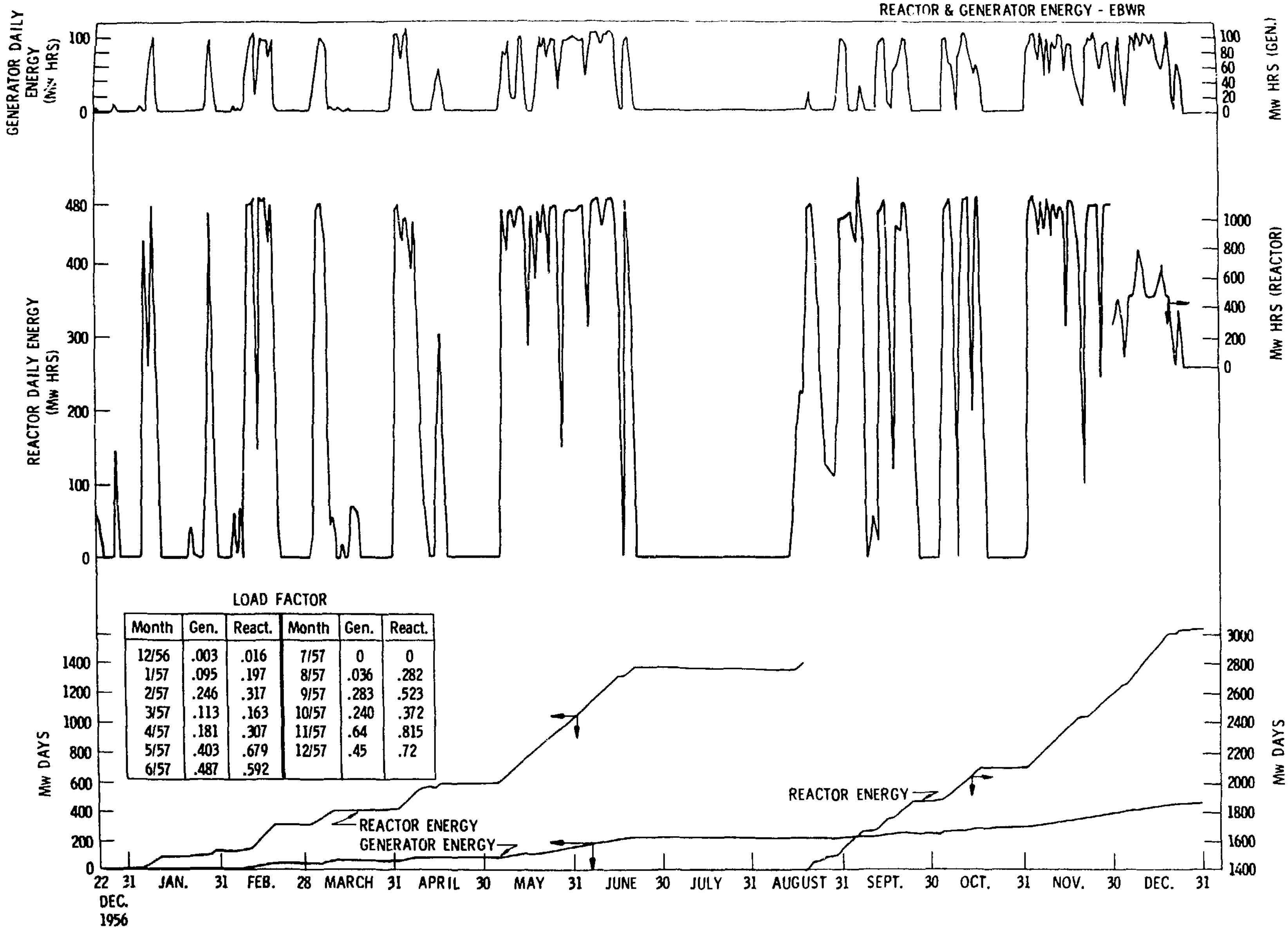


Figure 1. The Experimental Boiling-Water Reactor facility as originally constructed



Month	Gen.	React.	Month	Gen.	React.
12/56	.003	.016	7/57	0	0
1/57	.095	.197	8/57	.036	.282
2/57	.246	.317	9/57	.283	.523
3/57	.113	.163	10/57	.240	.372
4/57	.181	.307	11/57	.64	.815
5/57	.403	.679	12/57	.45	.72
6/57	.487	.592			

Figure 2. Reactor and generator operating levels by month, 1957

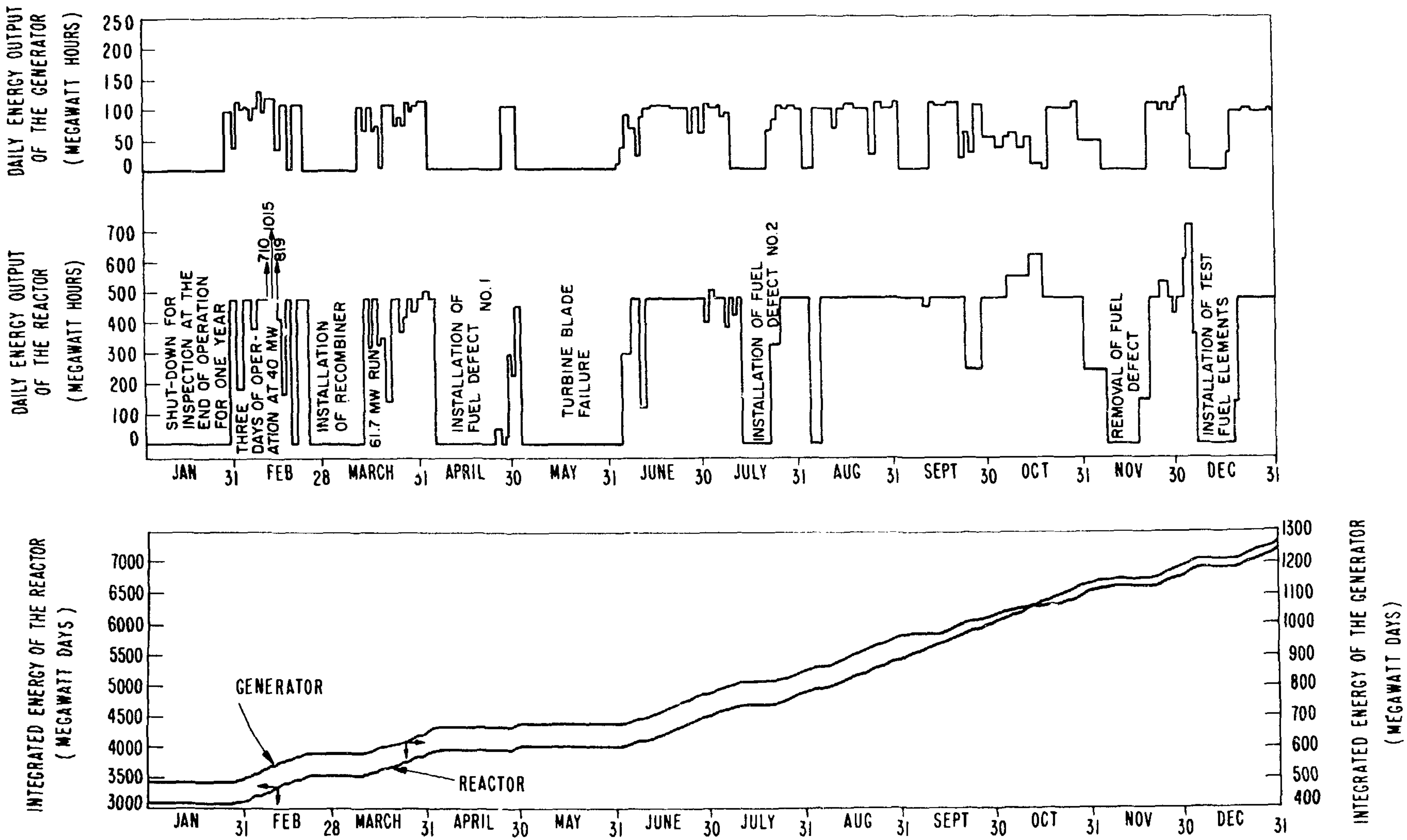


Figure 3. Reactor and generator operating levels by month, 1958

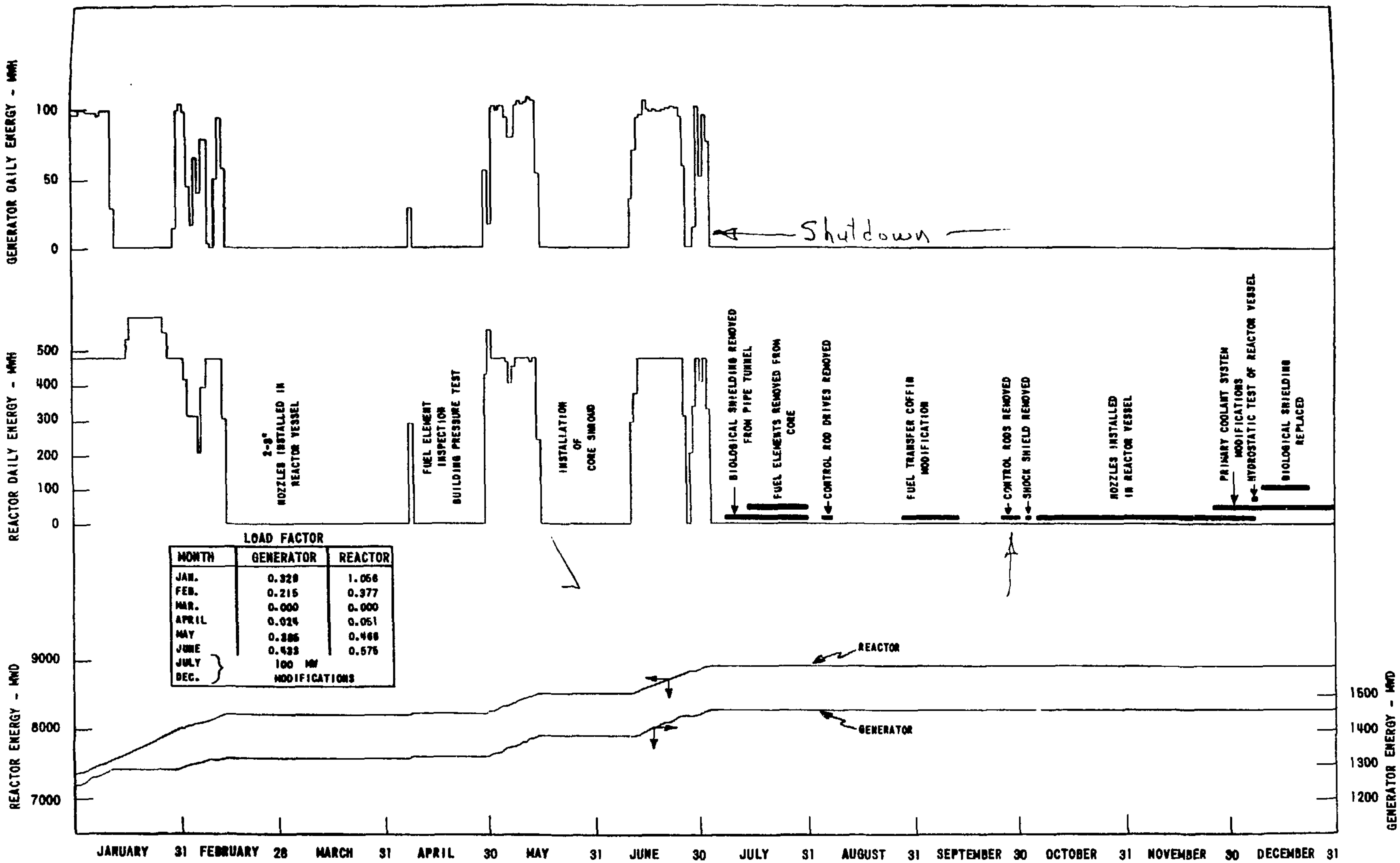


Figure 4. Reactor and generator operating levels by month, 1959

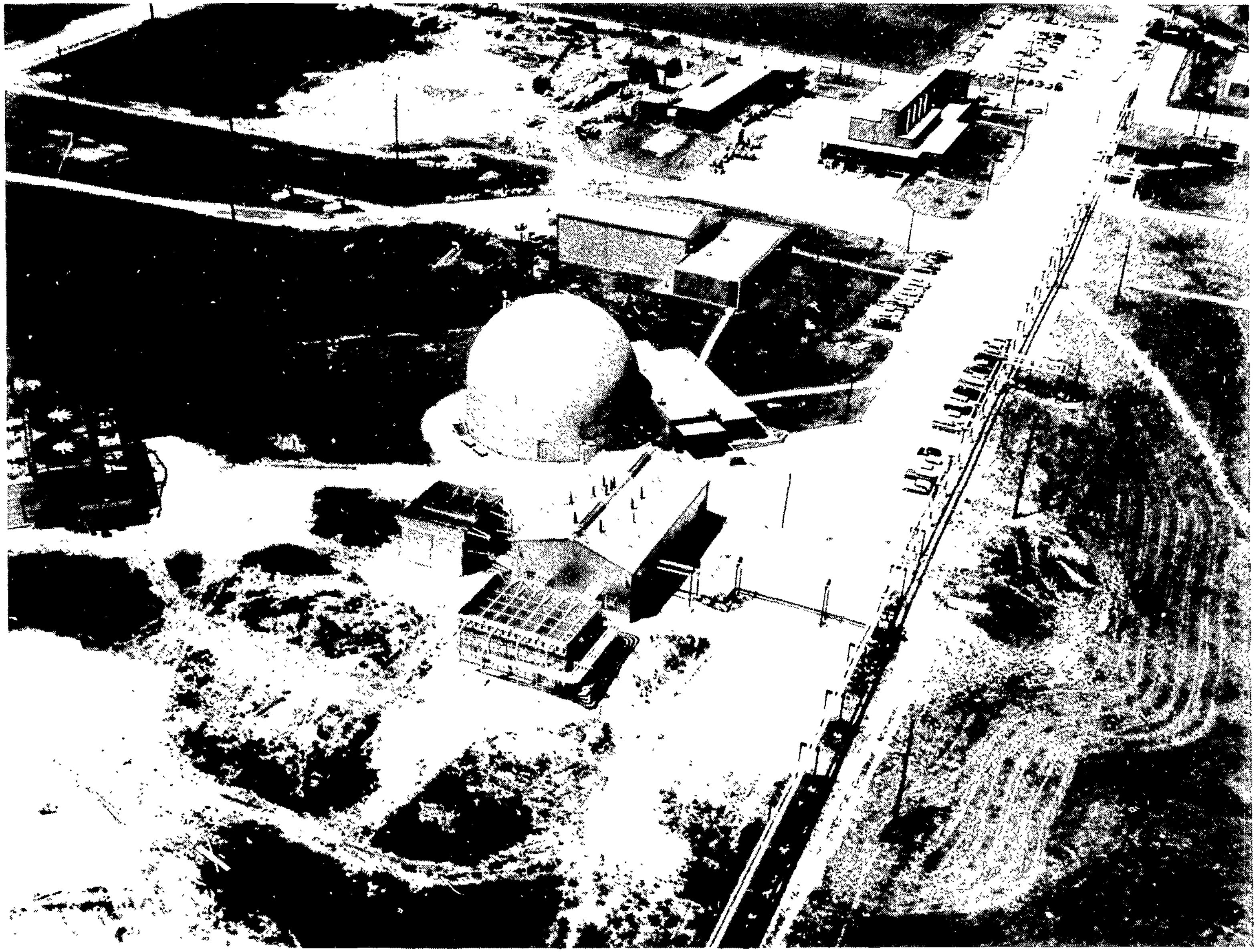


Figure 5. The EBWR as modified for 100-MW operation

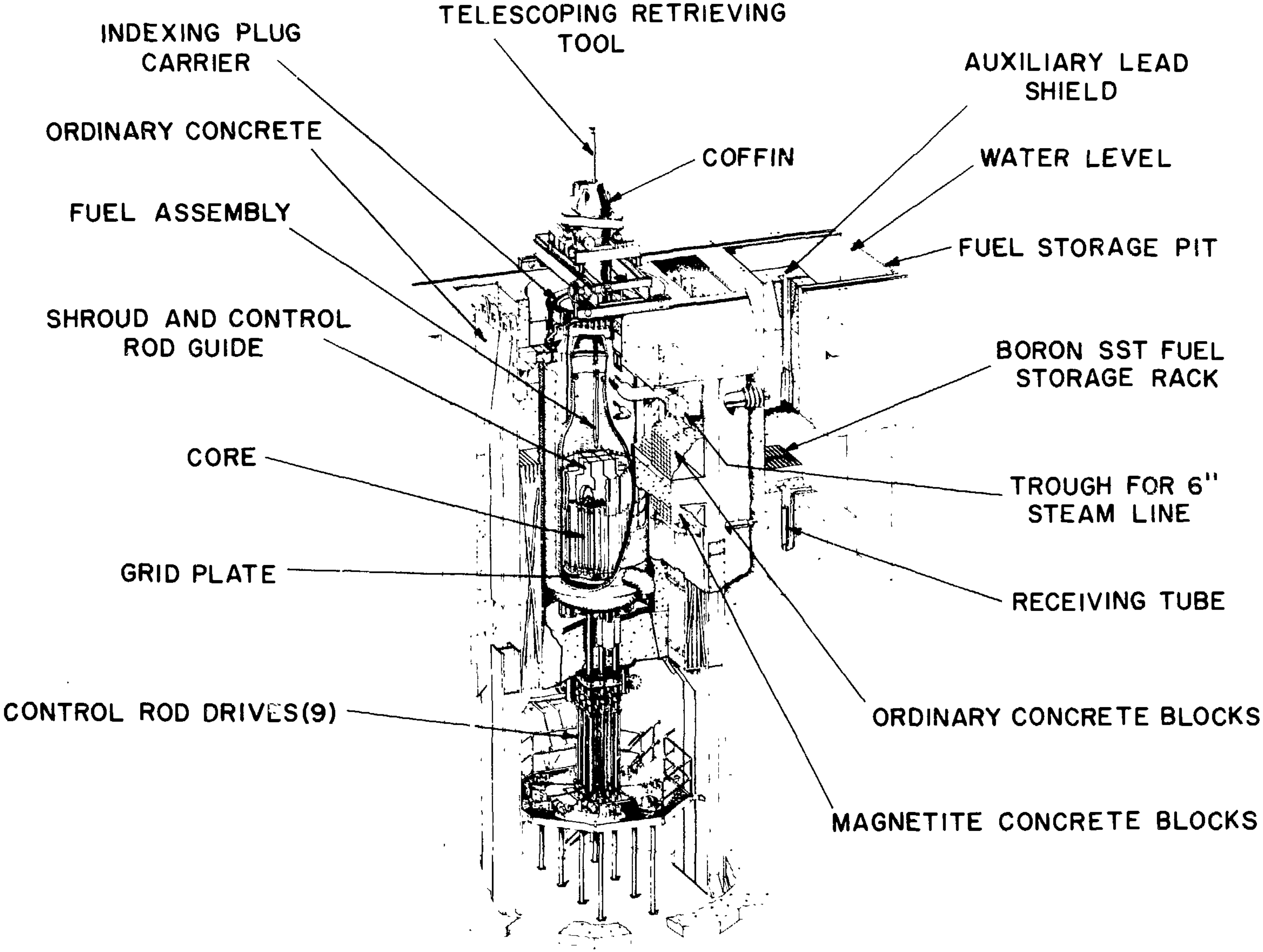


Figure 6. Cutaway pictorial view of the EBWR reactor and its components

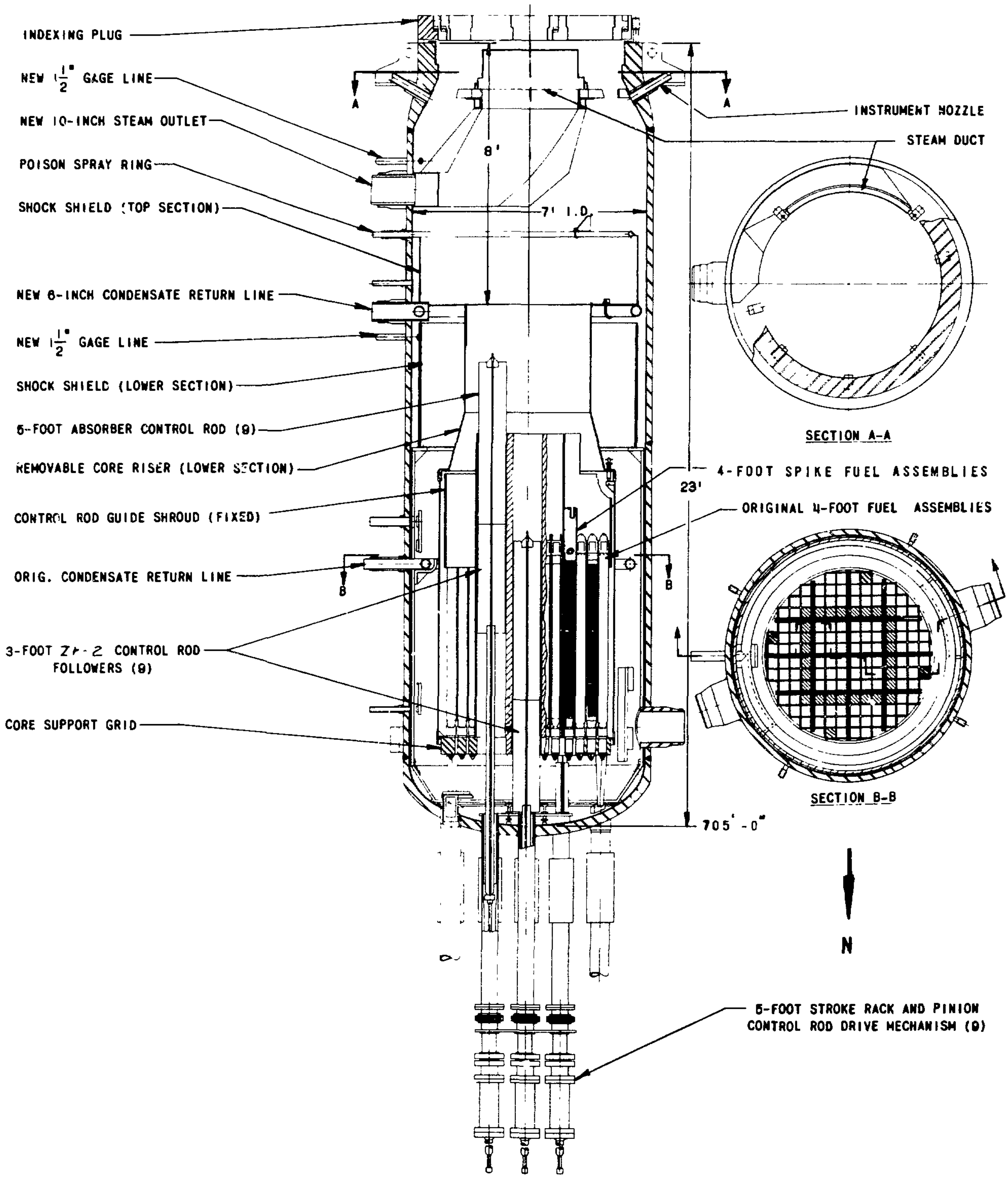


Figure 7. Diagram of the EBWR pressure vessel and its internals

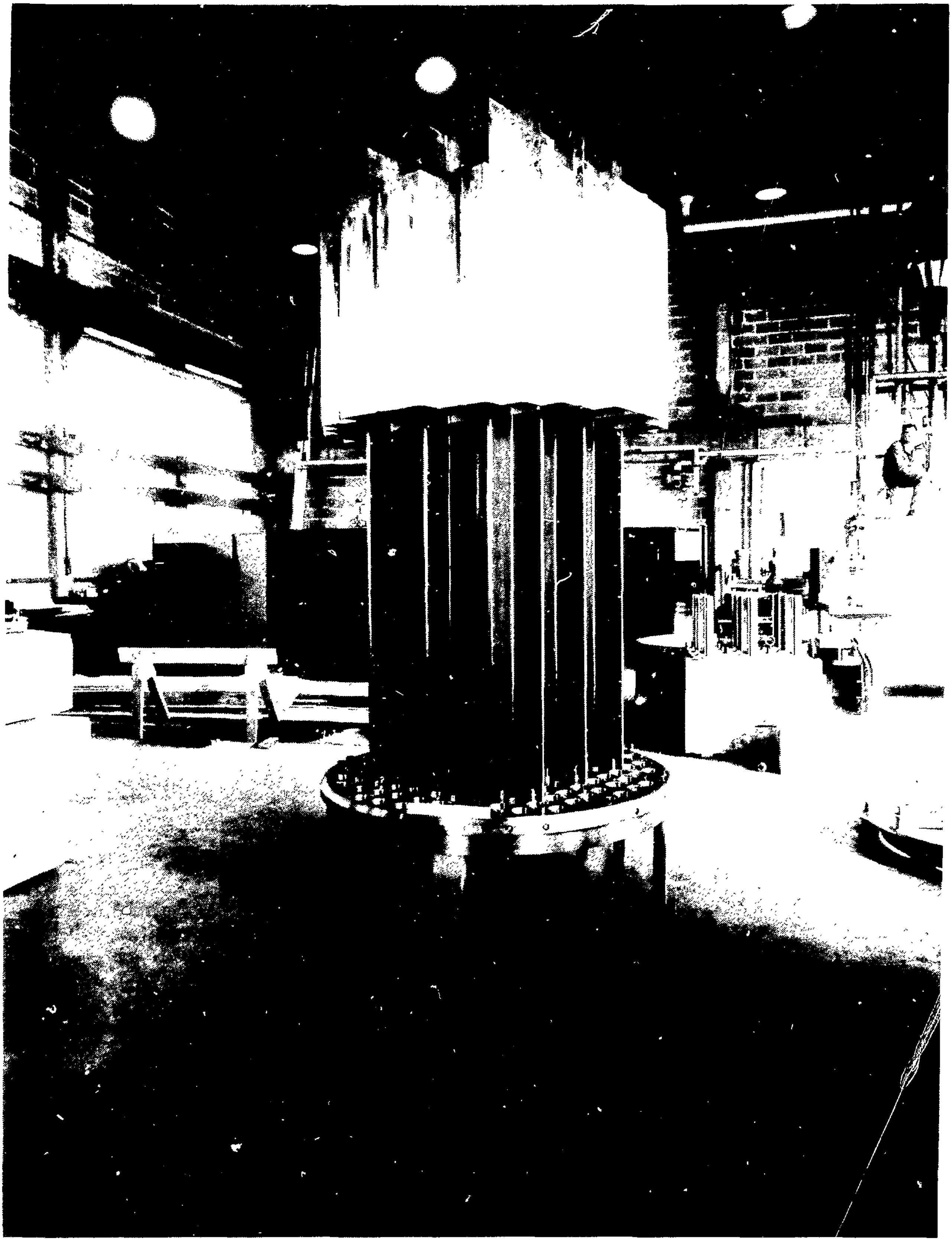


Figure 8. The control rod guide and top shroud structure

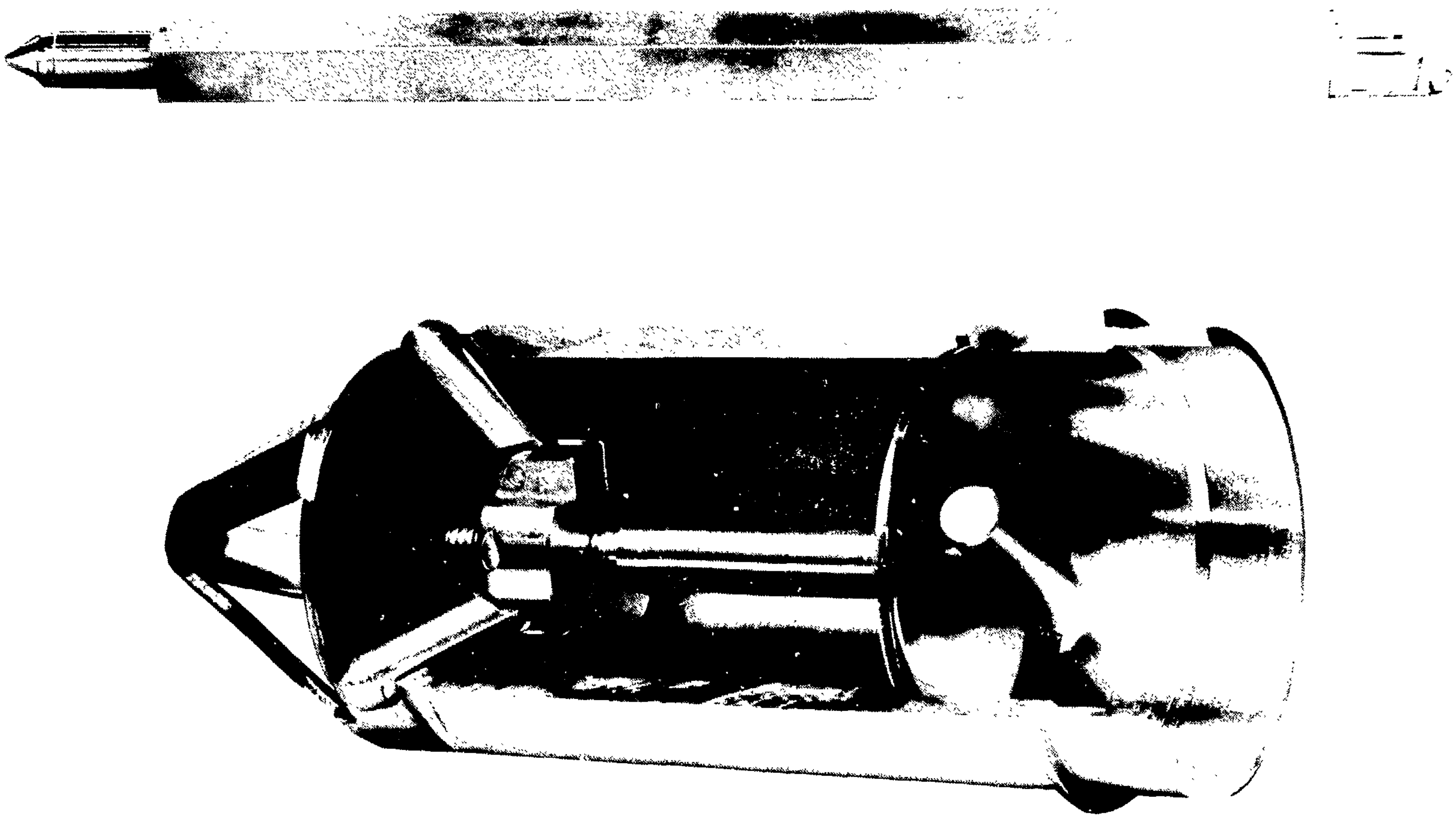


Figure 9. Dummy fuel assembly (top) and (below) its integral holddown assembly

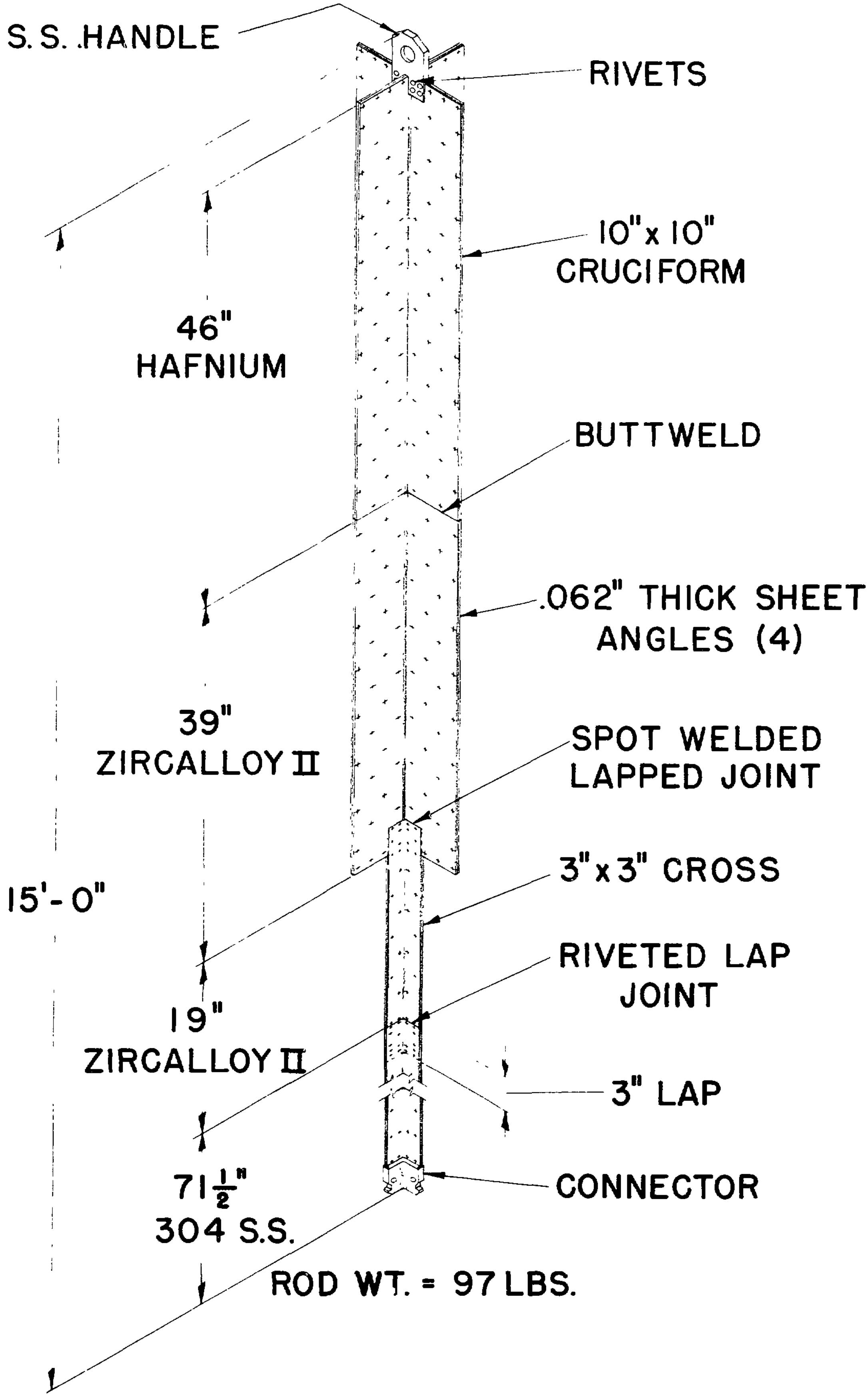


Figure 10, Hafnium-Zircaloy 2 control rod

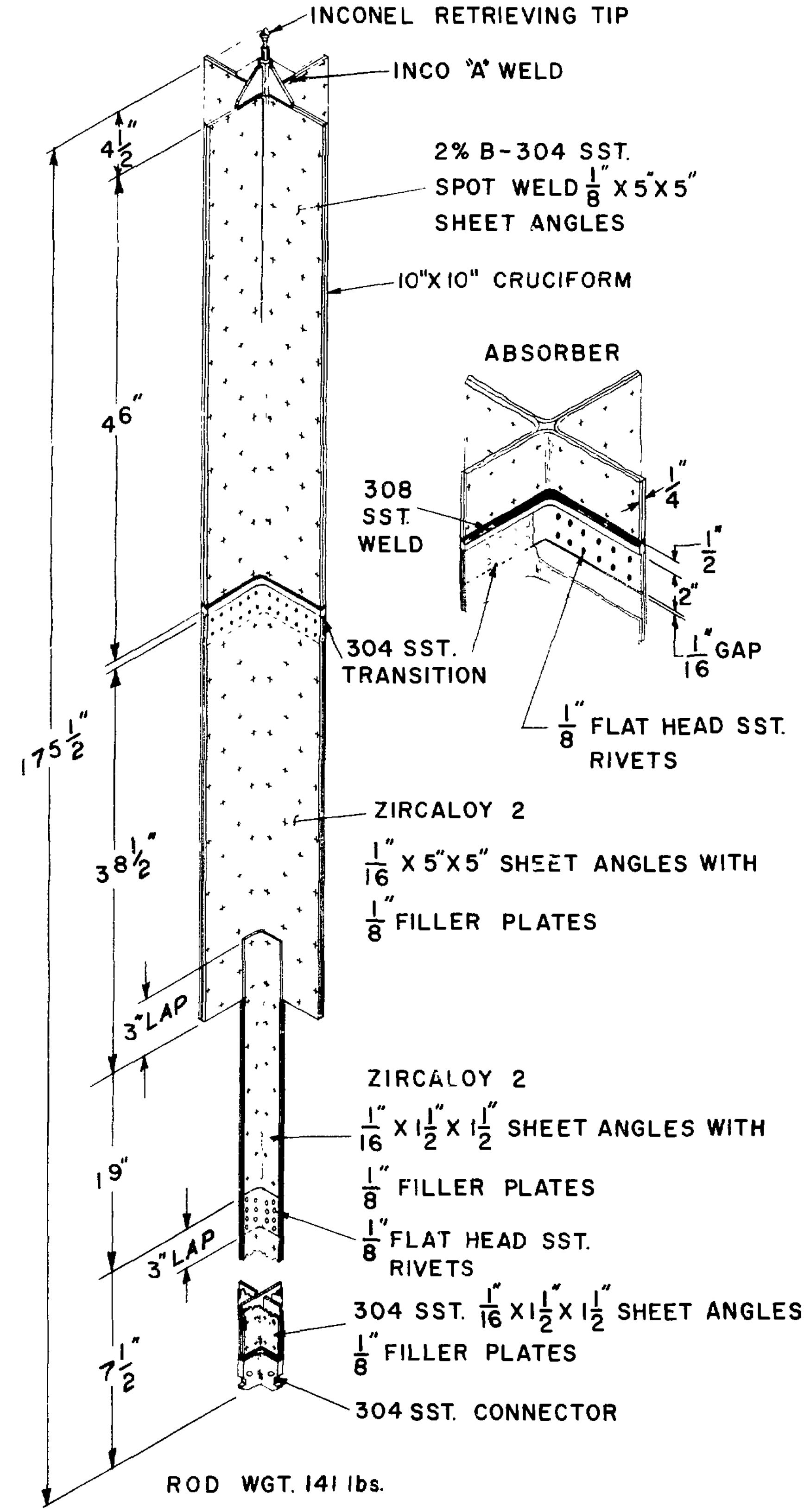


Figure 11. Boron-stainless steel control rod

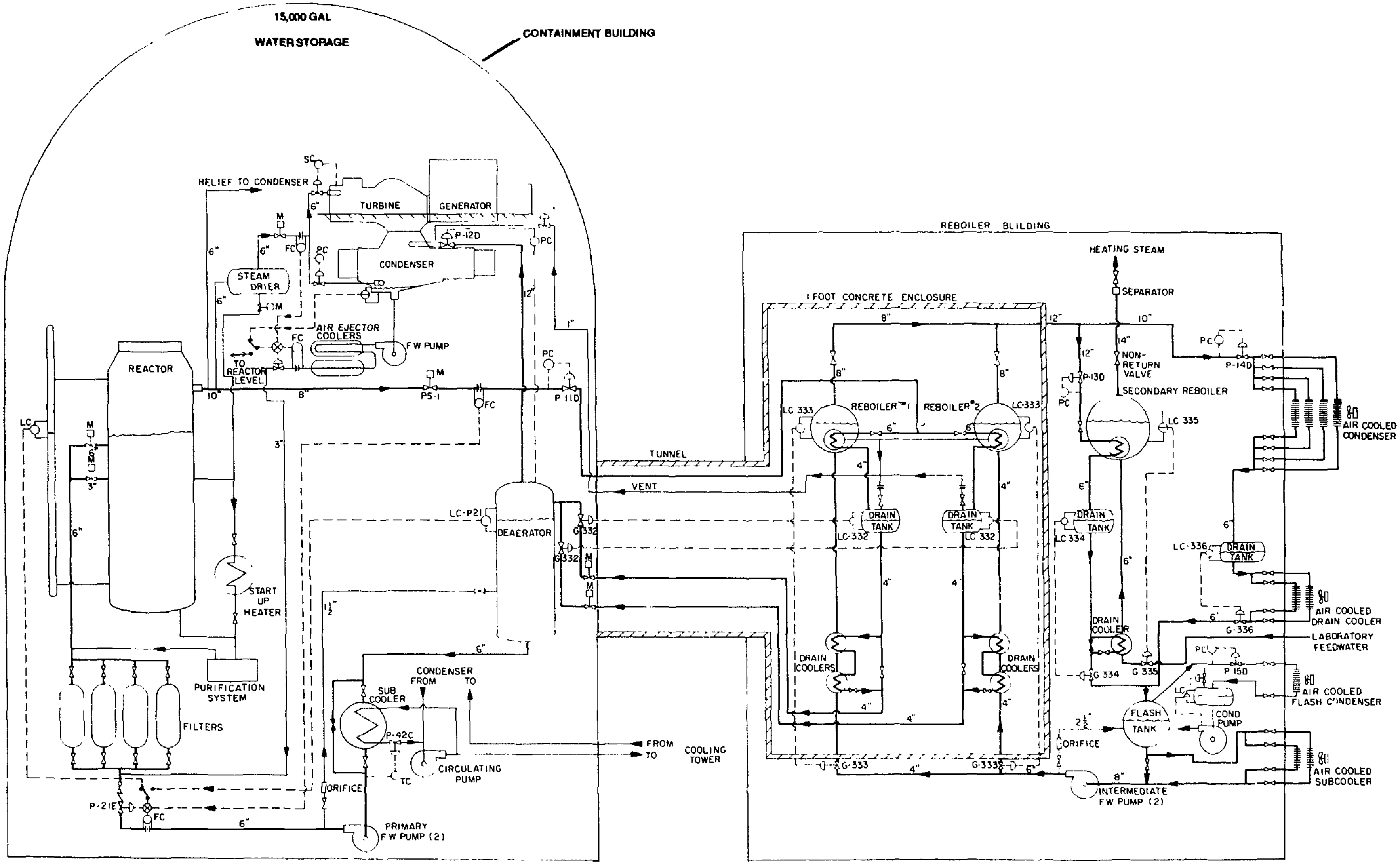


Figure 12. Flowsheet of the modified EBWR facility