UNITED STATES ATOMIC ENERGY COMMISSION

AECD-3435

A LOW COST EXPERIMENTAL NEUTRON CHAIN REACTOR

Part I

By
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Technical Information Service, Oak Ridge, Tennessee

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AECD-3435 is Part I of a report describing a low cost neutron chain reactor which was originally written in the spring of 1951 for declassification purposes. The design of the reactor described herein was declassified by the Atomic Energy Commission in March, 1952.

This report contains a description of the reactor and controls and appendices discussing health physics aspects, abnormal behavior, and corrosion problems. It is planned that Part II will contain neutron flux distribution curves, flux and heat transfer calculations, discussion of the results of a year's operating experience, and other matters germane to this reactor's construction and operation.

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LIST OF ILLUSTRATIONS

A LOW COST EXPERIMENTAL NEUTRON CHAIN REACTOR

William M. Breazeale

Recently, the Oak Ridge National Laboratory put into operation a low power reactor which, with minor changes to make it subject to declassification, should be suitable for installation in schools or other research institutions as a tool for experiments requiring a source of neutrons and as a laboratory device for educational purposes.

A chain reactor of the type described here is relatively so inexpensive that its construction is hardly a major enterprise. Since a low cost reactor which produces an experimentally significant neutron flux ought to be of wide interest, the following report describing such a facility has been prepared.

General Description

The reactor is a water cooled and moderated thermal neutron system using enriched uranium (greater than 90% U^{235}) for fuel. The fuel investment is 3 kilograms of U^{235} and is contained in parallel aluminum-uranium alloy plates. The entire reactor is suspended in a pool of water sufficiently deep for the water to serve as a shield. The active lattice* measures 12 in. by 12 in. by 24 in. high. Convection circulation of the pool water throught and around the reactor supplies ample cooling for operation at a nominal power level of 100 kilowatts. At this power level, the available neutron flux is about 10^{12} neutrons/cm^2/sec.

Exact cost figures as of the summer of 1950, as well as operating data, are available. The reactor and controls cost $58,400 and desirable auxiliary equipment another $36,100, for a total of $94,500. It is estimated that a suitable building and pool can be constructed for $125,000. The low cost of the reactor is partly the result of using the pool of water for moderating, shielding, and cooling.

As built at ORNL, this type of reactor is inexpensive, safe to operate, and easy to maintain. The fuel to moderator ratio is very nearly optimum and hence fissionable material is conserved. Its design is such that when used for instructional purposes it permits a student to perform a critical experiment, investigate neutron distributions, and (within limits) observe the effects of different loading geometries. If operated at full power, the flux is sufficiently high so that long irradiation times are not necessary for most experiments.

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*With water reflector. Better reflectors reduce the size of the active lattice and fuel investment.

The maximum power level at which the reactor is to be operated will depend in part on its intended use. In general, the flux at the surface of the reactor will be about 10^7 neutrons/sec/cm^2 when the total reactor power is 1 watt and, perhaps, twice this much at the center. This relation will vary somewhat with the loading. If the reactor is to be used primarily for educational purposes, then the level over long periods of time should not exceed a few hundred watts to prevent build-up of any substantial amount of radioactive fission products. Actually, 1 watt is sufficient to activate indium foils in the lattice, and 100 watts will permit considerable exploration in the water around the reactor. It is permissible to operate at 10 kw, or so, for short periods to observe the blue glow in the water around the reactor. This is a striking manifestation of what is known as Cerenkov radiation(1). Should it be desirable to use the reactor as a source of neutrons for experimental work, then a power level in the neighborhood of 100 kw is feasible. The corresponding flux density of 10^{12} neutrons/cm^2/sec compares favorably with that obtained in the large graphite reactors used for experimental purposes in this country.

Research in many unclassified fields can be carried on with the aid of this equipment. It is not the purpose of this report to discuss in detail any experimental programs. However, a few general subjects can be mentioned. Collimation of the neutron beam is accomplished easily with an empty pipe leading through the water to the surface of the reactor, and this beam will serve as a source of sufficient intensity for neutron diffraction equipment; or the beam can be used in conjunction with velocity selectors for relevant investigations. Chemical analysis by means of radioactivation is a rapidly expanding activity(2). This reactor will supply a neutron flux ample for satisfactory bombardment. There have been relatively few systematic radiation damage programs in the past, and much remains to be done. Radioactive isotopes with fairly intense, specific activities can be produced; the short-lived ones should be of especial interest. Investigation of short life β and γ activities resulting from slow neutron bombardment will continue to be a fruitful field for a considerable time. Many interesting programs covering the effect of radiations on biological processes can be planned.

On the more practical side, the reactor can be used as an aid in tracer work. This type of program can be illustrated by the scheme of determining the transfer of material from one gear to another with which it is meshed. The first gear is activated by neutron bombardment, run against the second one, and the material transferred determined by measuring the activity of the surface of the second gear. The AEC is interested in finding uses for radioactive fission products(3). These fission products can be obtained in suitable quantities by exposing a solution of a uranium salt to neutron bombardment in either a loop

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(1) Jordon, W. H., "Radiation from a Reactor", SCIENTIFIC AMERICAN, Vol. 185, No. 4, October, 1951.

(2) Leddicotte, G. W., and Reynolds, S. A., "Activation Analysis with the Oak Ridge Reactor", NUCLEONICS, Vol. 8, No. 3, March, 1951.

(3) "Problems in the Use of Fission Products", NUCLEONICS, Vol. 10, No. 1, January, 1952.

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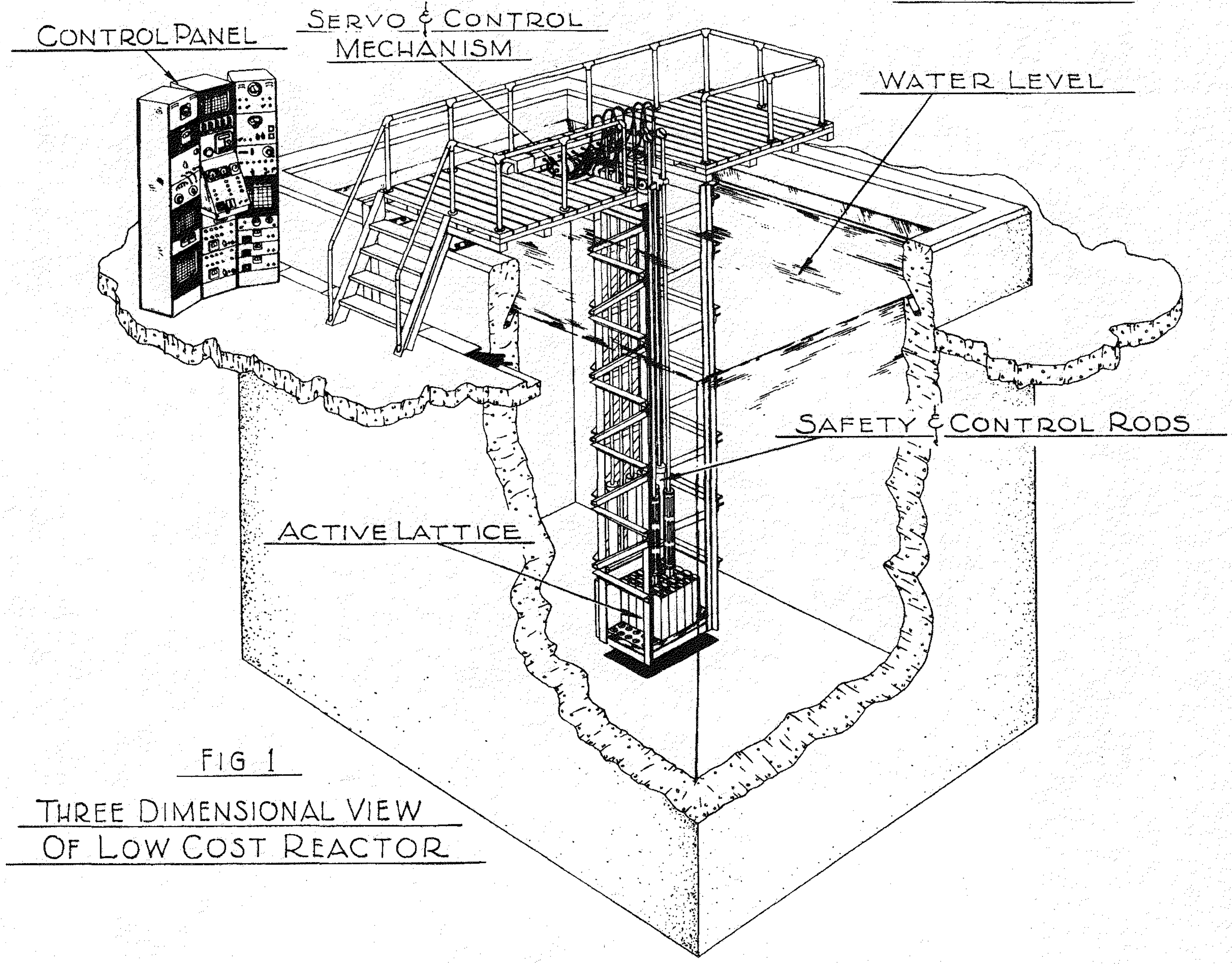

FIG 1

THREE DIMENSIONAL VIEW OF LOW COST REACTOR or a closed container. Many, many other radiation chemistry programs will benefit from the availability of the reactor. Perhaps, the best summation is to say that the work which is done with the reactor will be determined by the interests of the staff members of the establishment owning the reactor.

The Reactor

Figure 1 is an artist's conception of one arrangement of the reactor, pool, and control panel. The scheme of suspending the reactor in a pool of water provides a number of advantages. Besides cooling and moderating the pile, the water supplies a foolproof shield for personnel. Furthermore, should the shield (water) become contaminated, it can be replaced by simply draining and refilling the pool.

The reactor is an assembly of removable fuel elements placed on end in a 2S aluminum grid. A photograph of one element is shown in Figure 2 and details in Figure 3. The elements are 3 in. by 3 in. square and about 30 in. long. The active section is made up of 4 or 5 flat aluminum "sandwiches" 3 in. by 24 in. by 0.100 in. thick. These consist of a sheet of aluminum-uranium alloy sandwiched between two 2S aluminum plates. This assembly is hot rolled into a solid plate of the proper dimensions and is sufficiently tight so that fission products cannot escape. The plates are clad with a thin layer of 72S aluminum; the latter corrodes preferentially. A conical end box which fits into matching holes in the bottom grid is welded to the bottom of the active section. These elements can be made at ORNL at a present cost of about $120 each including overhead, but exclusive of the cost of enriched material.

Figure 4 is a picture of one design of the bottom grid. There are 54 holes for fuel elements in an array 9 holes by 6 holes. The reactor for which this grid was made had a beryllium oxide reflector on four sides. The photograph shows one portion of the reflector, which is a flat aluminum can containing cold-pressed BeO blocks, in its normal position covering the back row of 9 holes. The electromagnets which support the control and safety rods appear in the foreground and above the reflector are the aluminum cans which contain the ion chambers. The grid has many more holes than are required to hold the number of fuel elements necessary for criticality, but the design permits a variety of loading geometries. The holes through the grid for the fuel elements are 5 in. deep and hold the elements with sufficient rigidity so that no support at the top is required. The spacing between the fuel elements is sufficient to permit insertion of thin foils for determining neutron flux distributions.

The reactor is provided with two boron-lead shim-safety rods made by properly combining a mixture of lead and boral in an oval aluminum can 1 in. by 2-1/2 in. by 26 in. long. These rods travel in special fuel elements which have longitudinal holes 1-1/2 in. by 3 in. (Fig. 5). The rods, having almost the density of lead, fall with nearly the acceleration of gravity. An iron armature is fastened to the top of each rod, and this in turn is suspended from an electromagnet which can be raised or lowered with the aid of a small electric motor. Figure 6 is a picture of one rod. Each rod is equivalent to between 4 and 5 percent $\delta k/k$, depending on the loading. One control or

PHOTO 7610

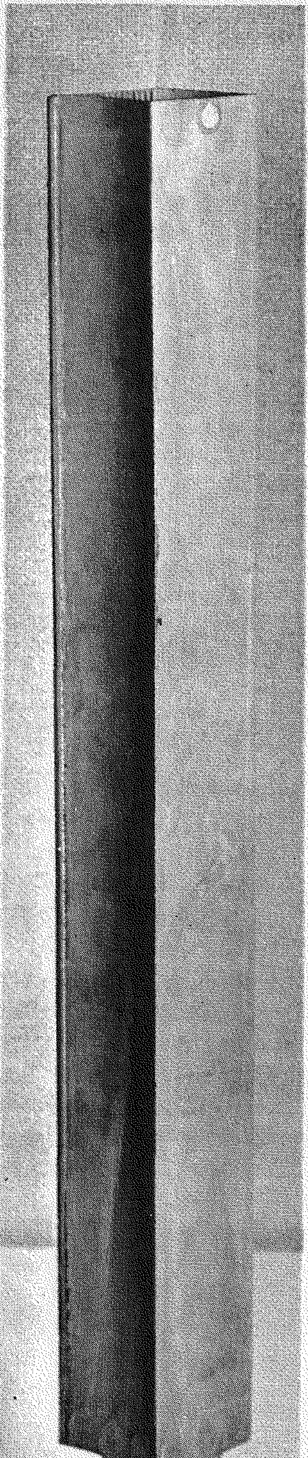

FIGURE 2. FUEL ELEMENT

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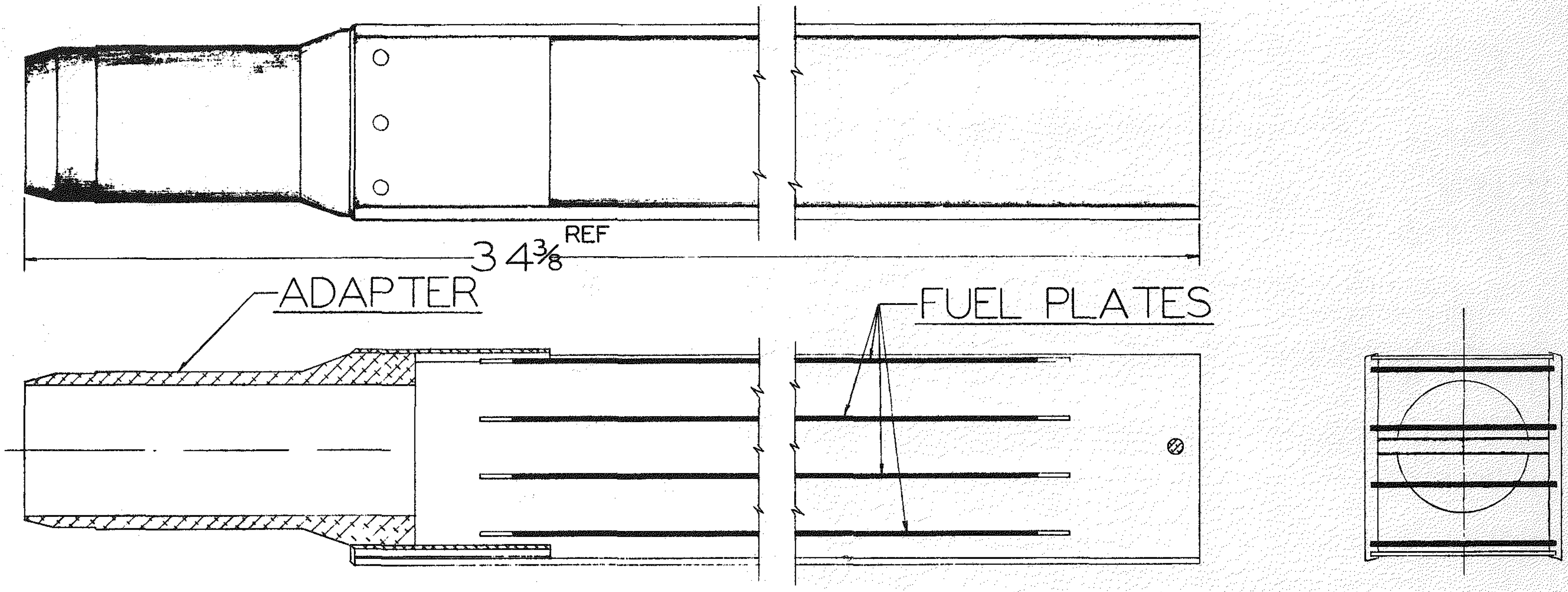
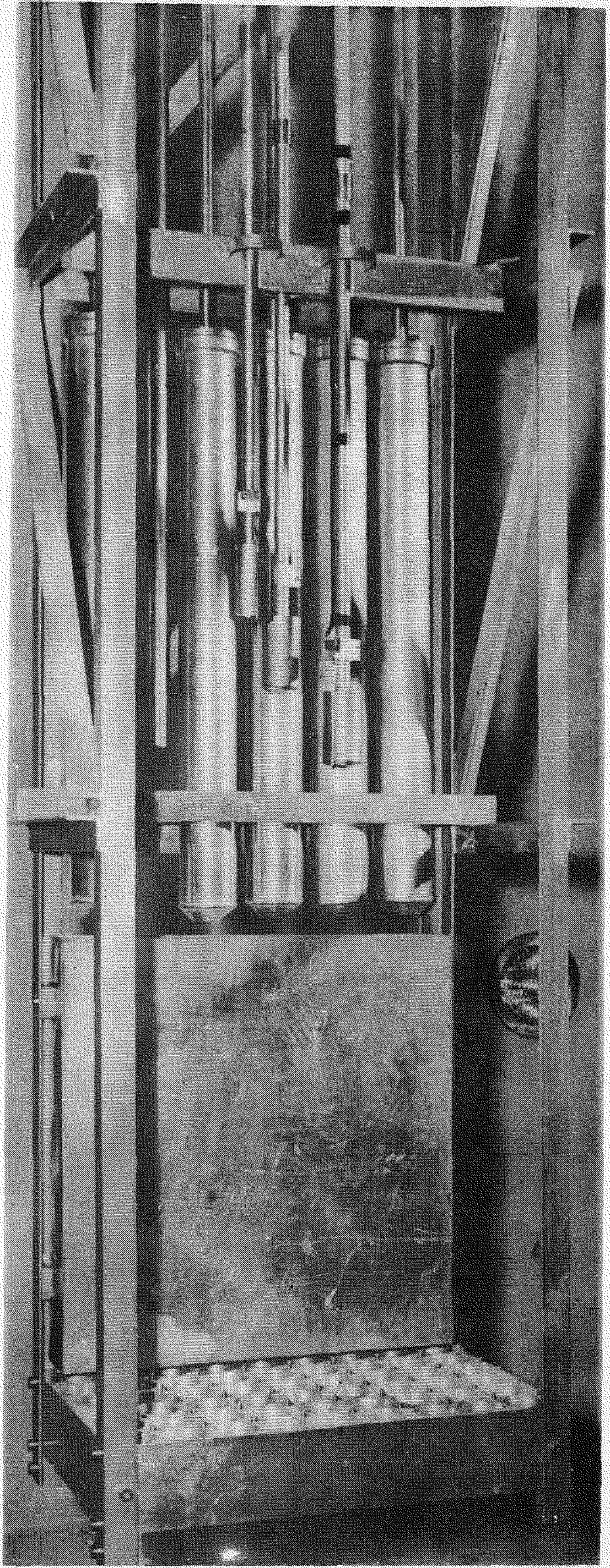


FUEL ASSEMBLY
LOW COST REACTOR
FIG. 3

FIGURE 4. GRID AND BeO REFLECTOR

Photo 7290



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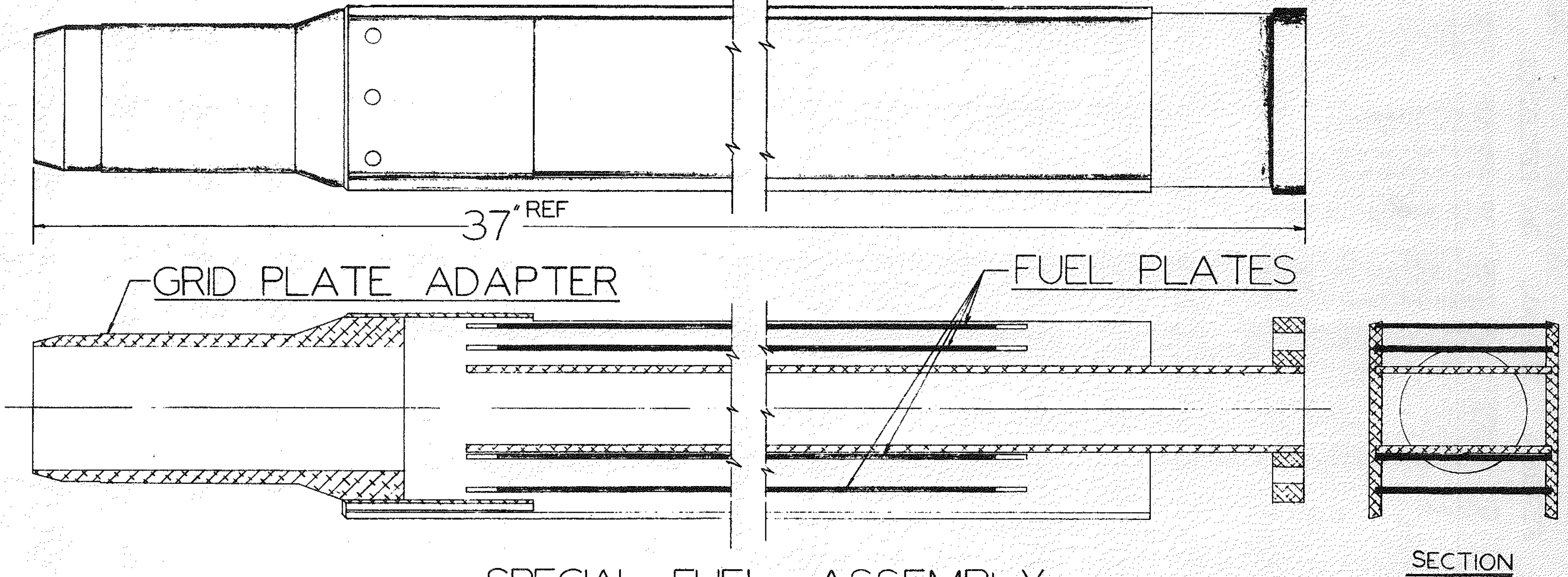


SPECIAL FUEL ASSEMBLY
LOW COST REACTOR
FIG. 5

FIGURE 6. SAFETY ROD PHOTO 7753

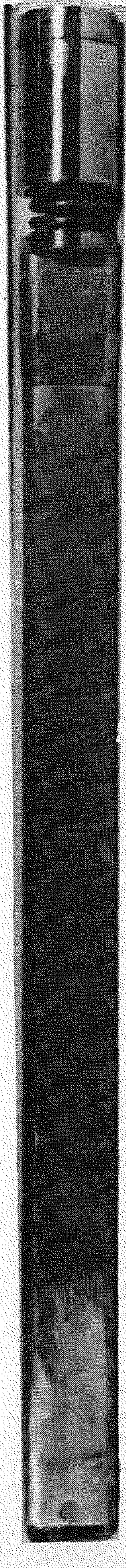
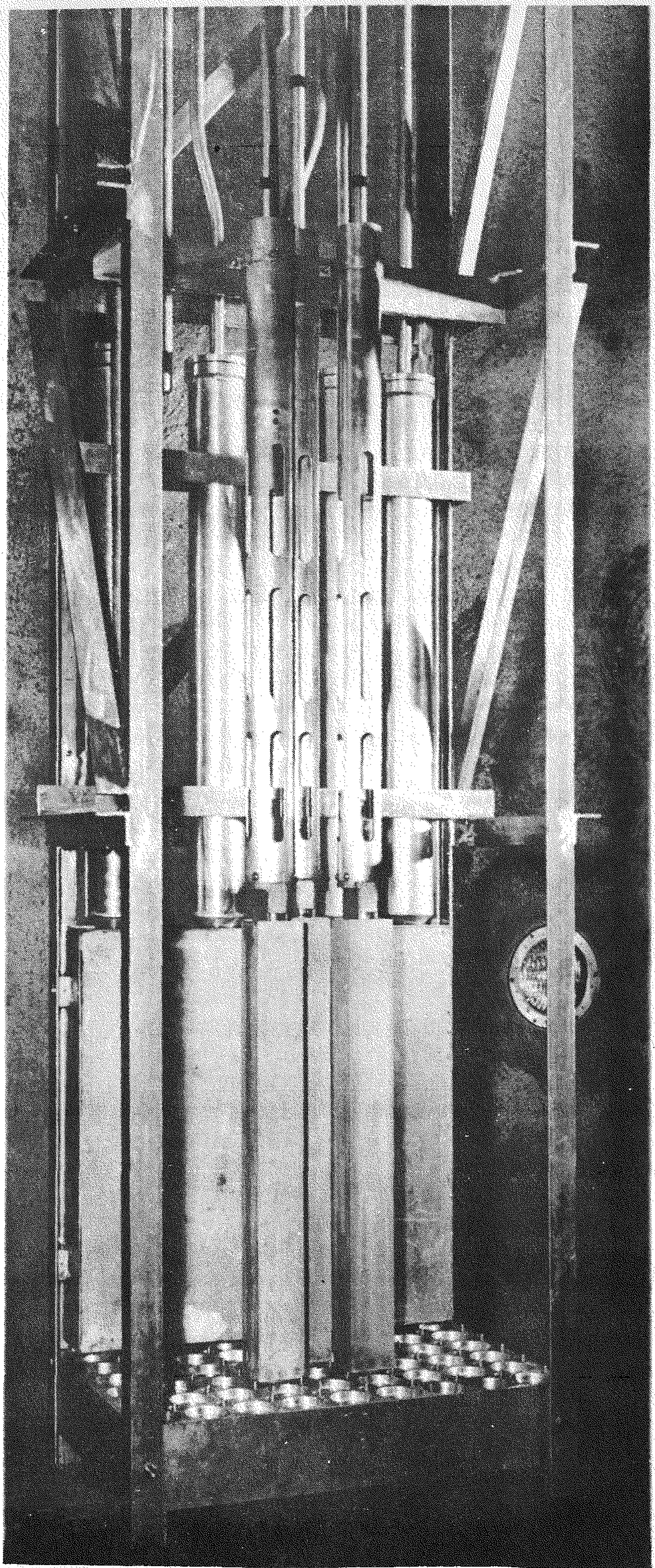


FIGURE 7. REACTOR REACTOR CONTROL PANEL

PHOTO 7415



regulating rod of the same dimensions and construction as the shim-safety rods, but containing only 8 gms of boron, is provided. It is equivalent to about 0.8 percent δ k/k.

Figure 7 shows the cylindrical tubes mounted on top of the special fuel elements which serve to guide the electromagnets and armatures. A hydraulic buffer and spring decelerate the rods at the end of their fall. The safety circuits are described in another section; it is sufficient to say here that "scramming"* the pile is accomplished by reducing the current through the magnets so that the rods fall by gravity.

The design of the fuel elements is based on the requirement that the active lattice dissipate some 100 kw by convection cooling with an ample margin of safety. This and nuclear considerations dictate 4 or 5 aluminum plates per element and result in an aluminum to water ratio of about 0.25. With water reflector on all sides and the fuel elements arranged in a 4 by 4 square lattice, calculations indicate that 2750 gms of U^{235} or 170 gms per fuel element are required. Such an amount is not far from the minimum critical mass for a reactor of this type.

Three of the central elements, as shown in Figure 8, are of the special design illustrated in Figure 4 and contain the safety and control rods. In addition, it is suggested that four fractional fuel assemblies of 20, 40, 60, and 80 percent, respectively, of the normal U^{235} content be provided. This enables one to load the reactor so that possible excess k is held to a small value. The total U^{235} requirement for the facility is then about 3 kilograms.

A good reflector placed around the active lattice will reduce the fuel requirement and improve the flux distribution in the core. It will also reduce the available flux at the surface of the reactor. A satisfactory scheme used on the reactor here is to provide a number of aluminum cans of the same outline as a fuel element filled with cold-pressed beryllium oxide bricks. It is calculated that 3 in. of BeO on the four vertical sides will reduce the size of the active lattice to an array 3 elements by 4 elements and the critical mass to about 2 kilograms. Thirty cans of BeO will be sufficient to surround this lattice. From an educational viewpoint, it is desirable that both types of reactors, i.e., water reflected and (partially) BeO reflected, be available. The student can then examine the effect of the reflector on the flux patterns. The flux at the outer surface of the reflector (the BeO) will, of course, be less than at the surface of the active lattice when the pool water serves as the reflector.

The problem of canning samples and instruments and placing them under water near the reactor may seem formidable at first, but experience in the facility at ORNL shows that no real difficulties are encountered. The various chambers and, when necessary, their associated preamplifiers are sealed in aluminum cans and the necessary conductors brought to the surface through rigid conduit or a flexible Tygon sheath. The instruments can be positioned

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*i.e., shutting the pile down as quickly as possible.

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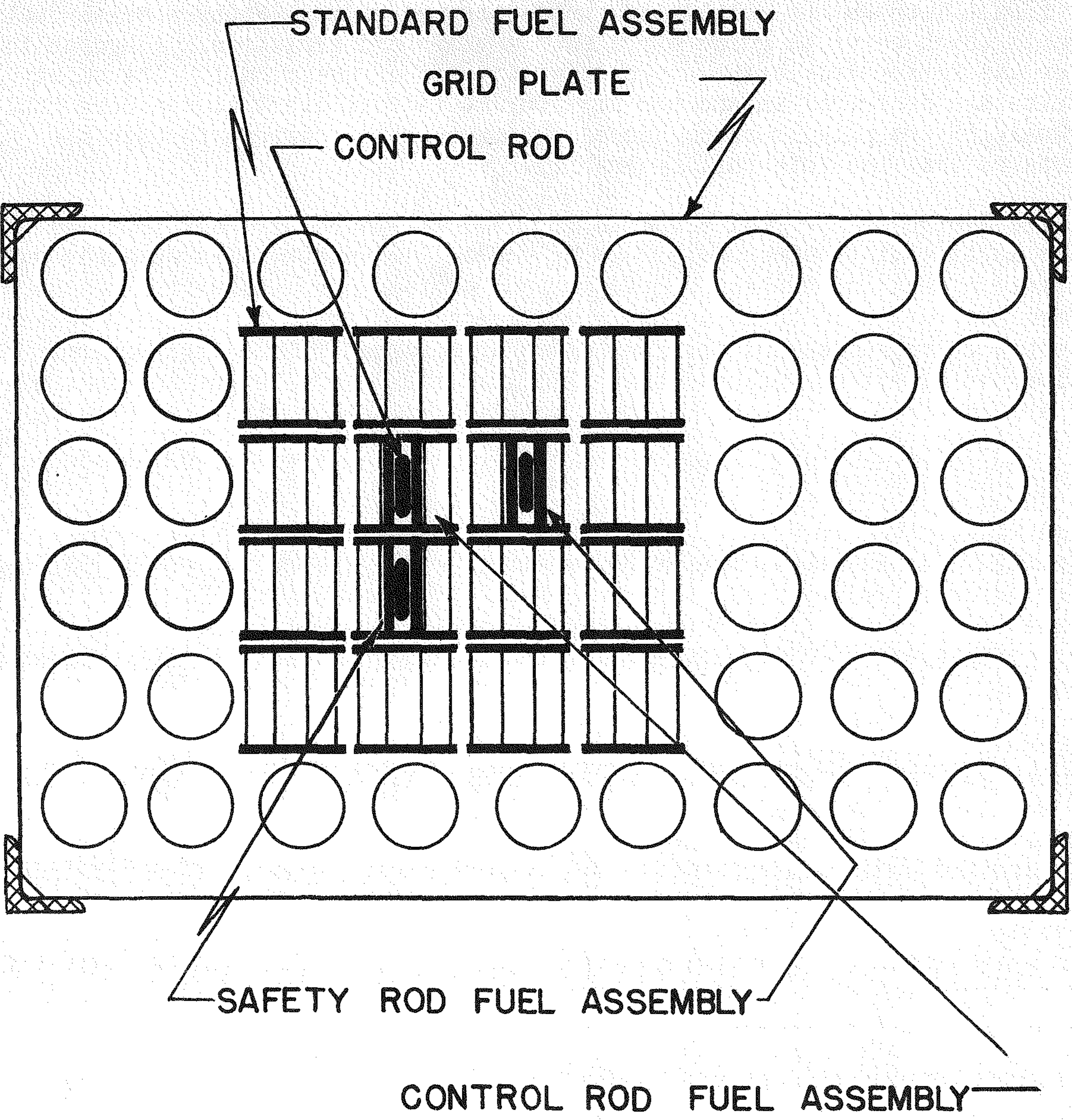

FIG. 8

ONE SUGGESTED LOADING PATTERN
LOW COST REACTOR

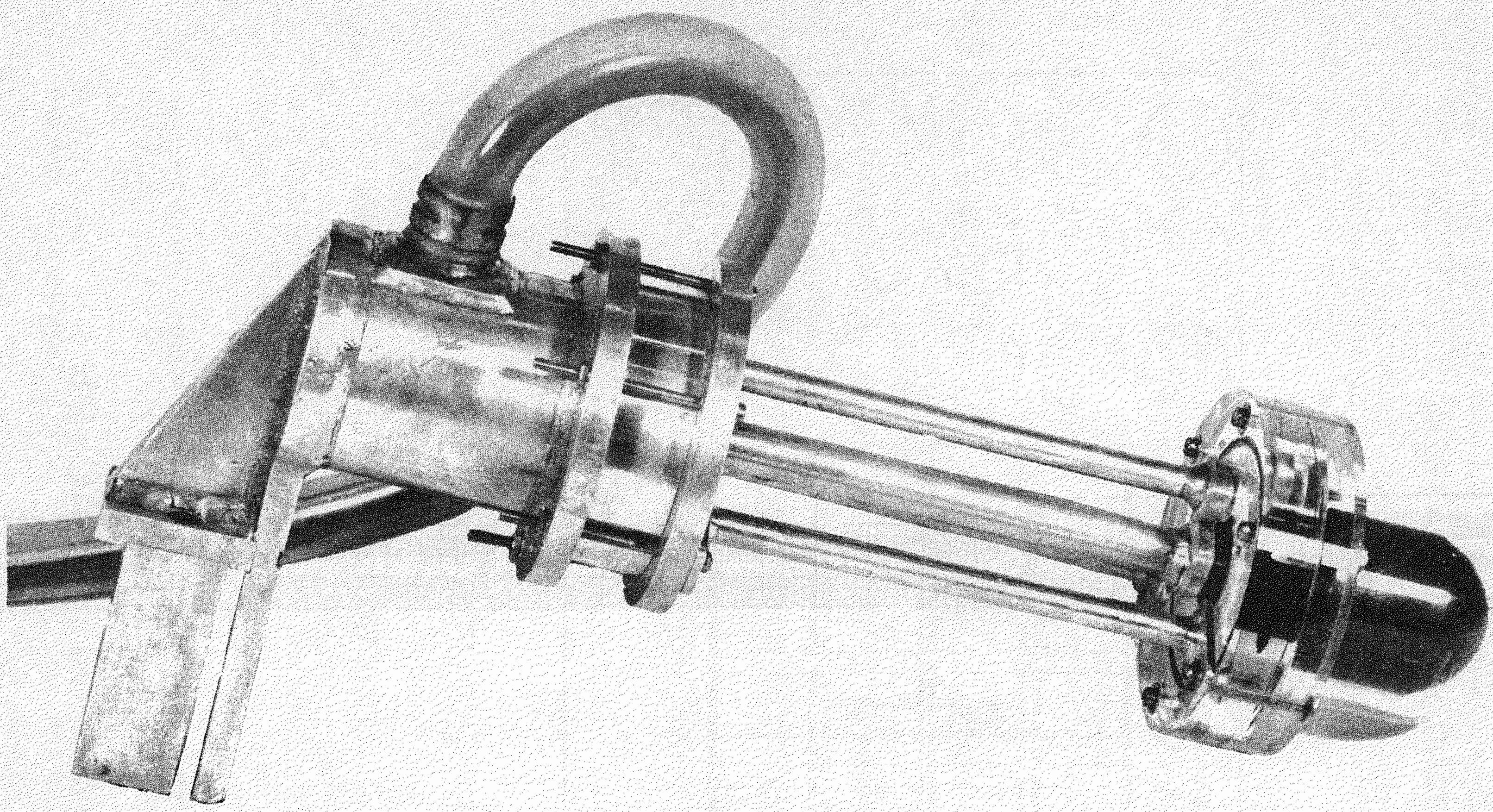

PHOTO 7757

FIG. 9 - IONIZATION CHAMBER AND PREAMPLIFIER FOR UNDERWATER USE

by a support which is fastened either to the reactor bridge or to another bridge across the pool. Reactor control chambers with rigid conduit are shown in Figure 4 and an ionization chamber with preamplifier and flexible cable in Figure 9. If necessary, more elaborate equipment can be designed should problems such as irradiation and examination of short-life isotopes demand it.

Empty, closed-end pipes, with the closed end placed against the surface of the reactor, will provide the conventional beam holes. The pipes can be vertical with the upper ends extending above the water or horizontal with the ends extending through the pool wall. Since a strong beam of neutrons and gamma rays will emerge from these pipes, they must be plugged carefully when the reactor is operating. A stack of dry graphite will thermalize the neutrons from the reactor. Two and one-half feet of graphite provides the maximum number of thermal neutrons, but more is required to afford complete thermalization.

The pool size again will depend on the particular use contemplated for the reactor. A suggested size is 14 ft by 18 ft by 22 ft deep. The top of the active lattice should be 16-1/2 ft below the surface of the water to afford protection to operating personnel for long-time operation at 100 kw. This depth of water attenuates the gammas sufficiently so that a person standing next to the pool will receive less than 60 mr in 8 hours. The water also attenuates the neutron flux sufficiently to keep the concrete pool walls from becoming seriously activated, and is more than deep enough to protect personnel.

If the pool is built above ground, then sufficient thickness of poured concrete, or concrete blocks, must be supplied on the sides so that any gamma ray from the reactor passes through a mass of water plus concrete equal to 16-1/2 ft of water. If the pool is approximately the dimensions given above, then the water plus concrete will provide ample neutron shielding.

The heat capacity of the pool is sufficiently great so that intermittent operation at 100 kw or less is permissible without external cooling. The size suggested above contains about 45,000 gallons and the average rise in temperature of the water, with the reactor running at 100 kw, is 1°F per hour. If continuous operation at 100 kw is desired, then arrangements must be made to remove 6,000 BTU per minute, either by adding cold water (with corrosion inhibitor) or circulating the pool water through a heat exchanger. Assuming a change in temperature of the cooling stream of 20°F, a rate of flow of 40 gallons per minute is required.

The building housing the reactor and pool should be substantially constructed. There should be sufficient head-room over the pool to lift the frame supporting the reactor, and an adequate number of exits at ground level should be provided. Suitable office space should also be supplied. The building housing the facility at ORNL has a bay 70 ft by 35 ft by 38 ft high, containing a pool 40 ft by 20 ft by 20 ft deep, plus 3000 sq ft of office and shop space. It is of steel frame construction covered with H. H. Robertson "Q" siding. The building, pool, lighting, plumbing, etc. (but without the reactor and equipment) cost $137,000 in the summer of 1950. A more modest building should suffice for the uses contemplated for the low cost reactor under discussion in this report.

In the present design, the reactor is suspended by an aluminum framework from a pair of 12 in. I-beams laid across the pool parapets (Fig. 1). The I-beams have transverse bracing and are covered with a wooden platform. The racks containing the control equipment are placed beside the pool. Experience at ORNL shows no need for a separate control room. The supports for the framework are cantilevered out from one of the I-beams and the framework made completely open on one side. This enables an operator standing on the platform to remove or replace fuel elements in the reactor with the aid of an especially designed long handled tool without draining the water from the tank. A lightweight (1-1/2 ton) overhead hoist and bridge crane is very desirable as an aid in assembling the reactor, removing fuel elements, etc.

After the fuel elements have been in operation for a long time at high power, the long-life fission products will have accumulated to the point where the elements will not "cool" down in a reasonable time. The elements can be removed from the pool only inside a protective shield. A lead "coffin" with walls 4 in. thick will suffice as protection for one element a few days after shutdown from long time reactor operation at 100 kw. Such a shield weighs a little less than a ton.

The AEC has facilities for reprocessing these aluminum alloy fuel elements without any pretreatment. Recovery of the uranium need not be a problem to the institution owning the reactor.

Control and Safety Circuits

The control and safety circuits for the low cost reactor can be the same as those now in use at ORNL. These circuits have been extensively tested here, and their performance has been perfectly reliable. A discussion of the circuits follows, and a list of drawings describing instrument and circuit details can be found in Appendix III.

Over the operating range this is essentially a constant temperature reactor. Hence, the temperature coefficient, while negative, does not vary sufficiently to stabilize the reactor, and it is recommended that a servo control be supplied to hold the power level constant. Details of a servo system incorporating the usual amplifier, a 2-phase AC servo motor geared to the regulating rod drive, etc., are available from the Control Group at ORNL.

A block diagram of the circuits for controlling and observing the operation of the reactor is shown in Figure 10. The first instrument is a 4 in. differential chamber* which supplies a Leeds and Northrup Model 2430D galvanometer. There are no electron tubes in this circuit, and the galvanometer reading is always directly proportional to the chamber current. The range is

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*This instrument contains two chambers, one of which is boron coated. Both respond to gammas, but only the boron coated one to neutrons. The outputs are connected in opposition and thus, when the compensation is properly adjusted, the net output is a function only of the neutron flux. A report covering this chamber is forthcoming.

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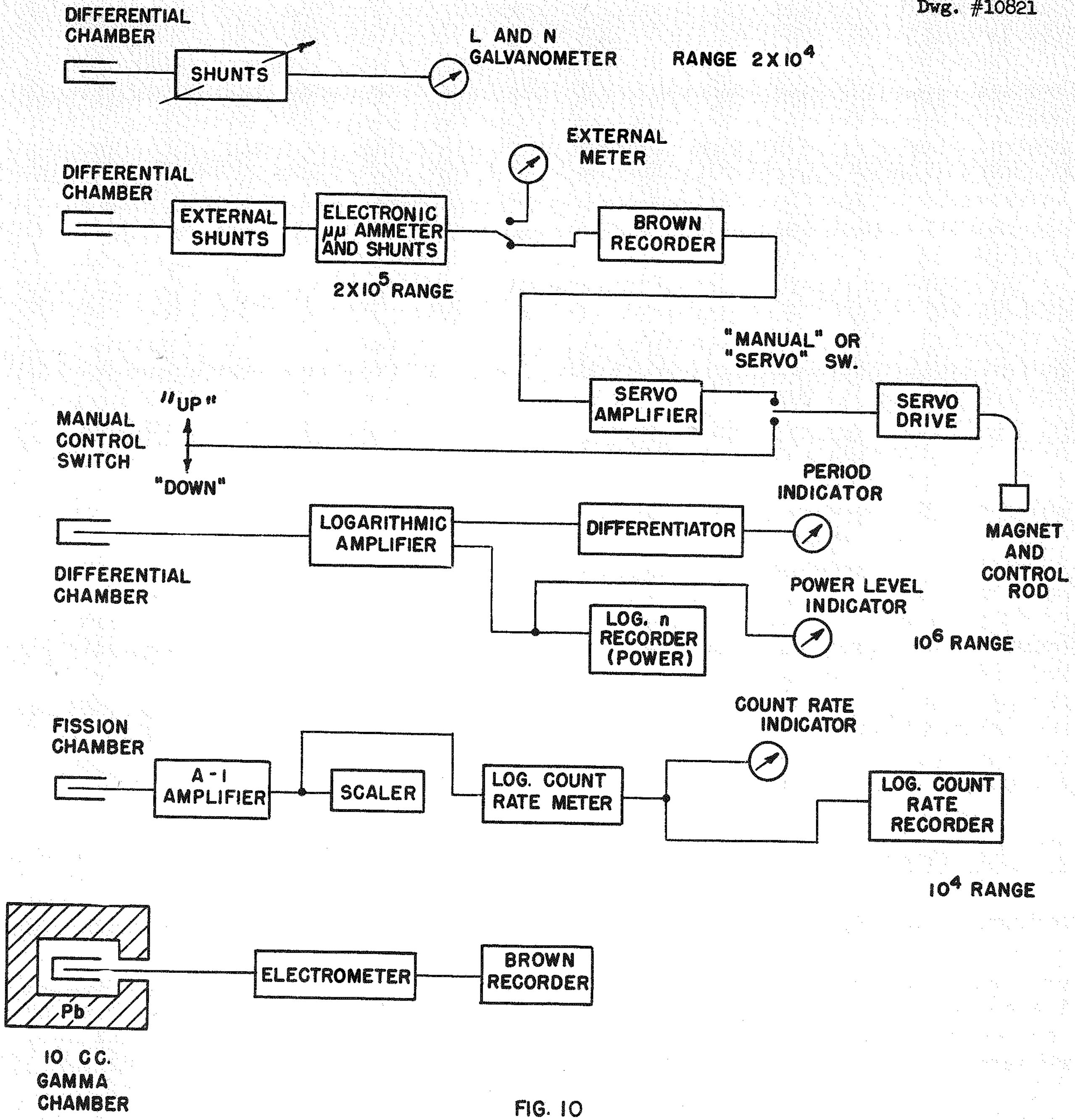

FIG. 10

CONTROL CIRCUITS

about 2 x 10^4 and the relation between the galvanometer indication and the reactor flux depends on the location of the chamber. Another differential chamber is connected to a Leeds and Northrup Model 9836A electronic micro-micro ammeter. With the aid of shunts, the range is extended to 2 x 10^5. The micro-micro ammeter controls a Brown Recorder which in turn drives the servo amplifier. An extra slidewire has been added to this recorder, and the position of the contact on the slidewire determines the power level which the servo will seek in controlling the pile. A third differential chamber supplies a logarithmic amplifier. The output of this amplifier is differentiated to give a pile period indication. The output also controls a Brown Recorder from which a continuous record (without changing shunts) of the power level over a range of 10^6 is available. This is the so-called Log N indicator.

To provide indication during start-up, a U^{235} fission chamber, amplifier, scaler and register, and log count rate meter and recorder are provided. A neutron source of sufficient strength to supply about 3 counts per second with the rods down is placed in the reactor to facilitate start-up. This circuit has a range of 10^4 (10,000 counts maximum), and both the micro-micro ammeter and the Log N circuits are responding before this level is reached. An electric motor drive raises the fission chamber to keep it from being activated at high reactor powers.

Should the experimental program require an accurate knowledge of the pile gamma level, then a small gamma chamber, shielded from the capture gammas in the water, must be provided. A 10 cc graphite chamber placed against one face of the reactor and shielded from the capture gammas in the water by a lead half cylinder will supply a current very nearly proportional to the gamma ray level in the reactor.

The philosophy of the safety system is that it should "fail safe", i.e., the safety rods must fall if the power is cut off or if major circuit trouble develops. In furtherance of this idea, vacuum tubes are used throughout in preference to gas tubes or relays. The amplifiers are monitored at several points, and lights warn the operator of an abnormal condition. A block diagram of the system is shown in Figure 11. The "safety" chambers are 3 in. parallel plate, boron coated, ionization chambers which supply a current proportional to the neutron plus gamma level. This current flows through a high resistance which is across the input of a preamplifier consisting of a single stage cathode follower. The output of the preamplifier feeds a DC amplifier (the "sigma" amplifier) which is the source of the signal for operating the safety circuits. As shown in the block diagram, both of these amplifiers are connected to a common point called the "sigma bus".

The electromagnets which support the safety and control rods draw exciting current from separate magnet amplifiers. The inputs to these amplifiers are controlled by the voltage on the "sigma bus". As this voltage is increased, the current through the magnets decreases. Thus, the result of an increase in neutron flux is to reduce the magnet current, and the circuit is adjusted to release the rods when the flux reaches a predetermined level. Grounding the "sigma bus" will also cause the rods to drop. Adjustment is obtained by changing the preamplifier input resistance or by moving the safety chambers in the

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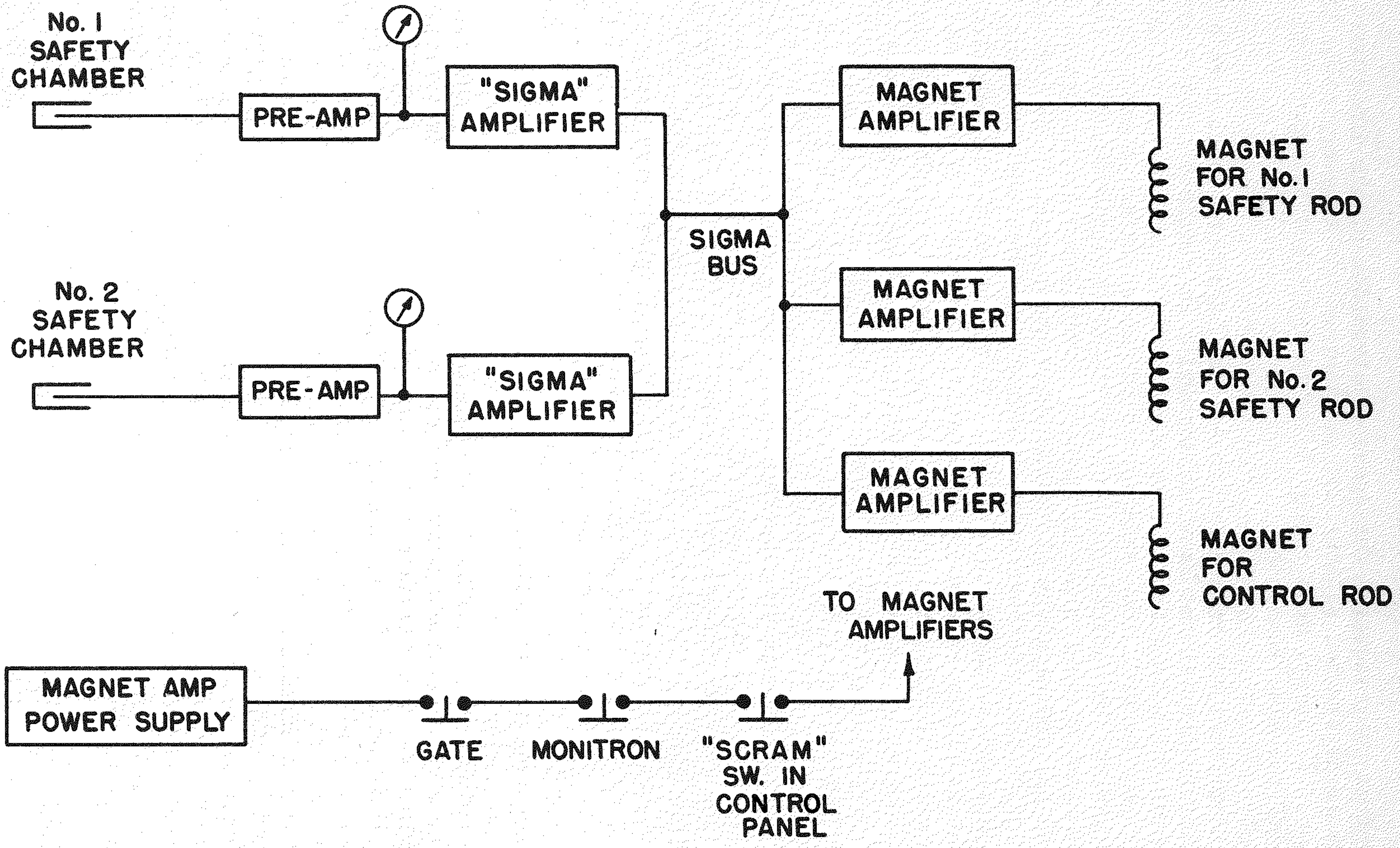


FIGURE 11

SAFETY CIRCUITS

water away from the reactor. The speed of operation of the electronic part of the circuit is determined by the time constant of the chamber, cable, and input resistor. Actually, most of the delay in operation is associated with the inductance of the electromagnets.

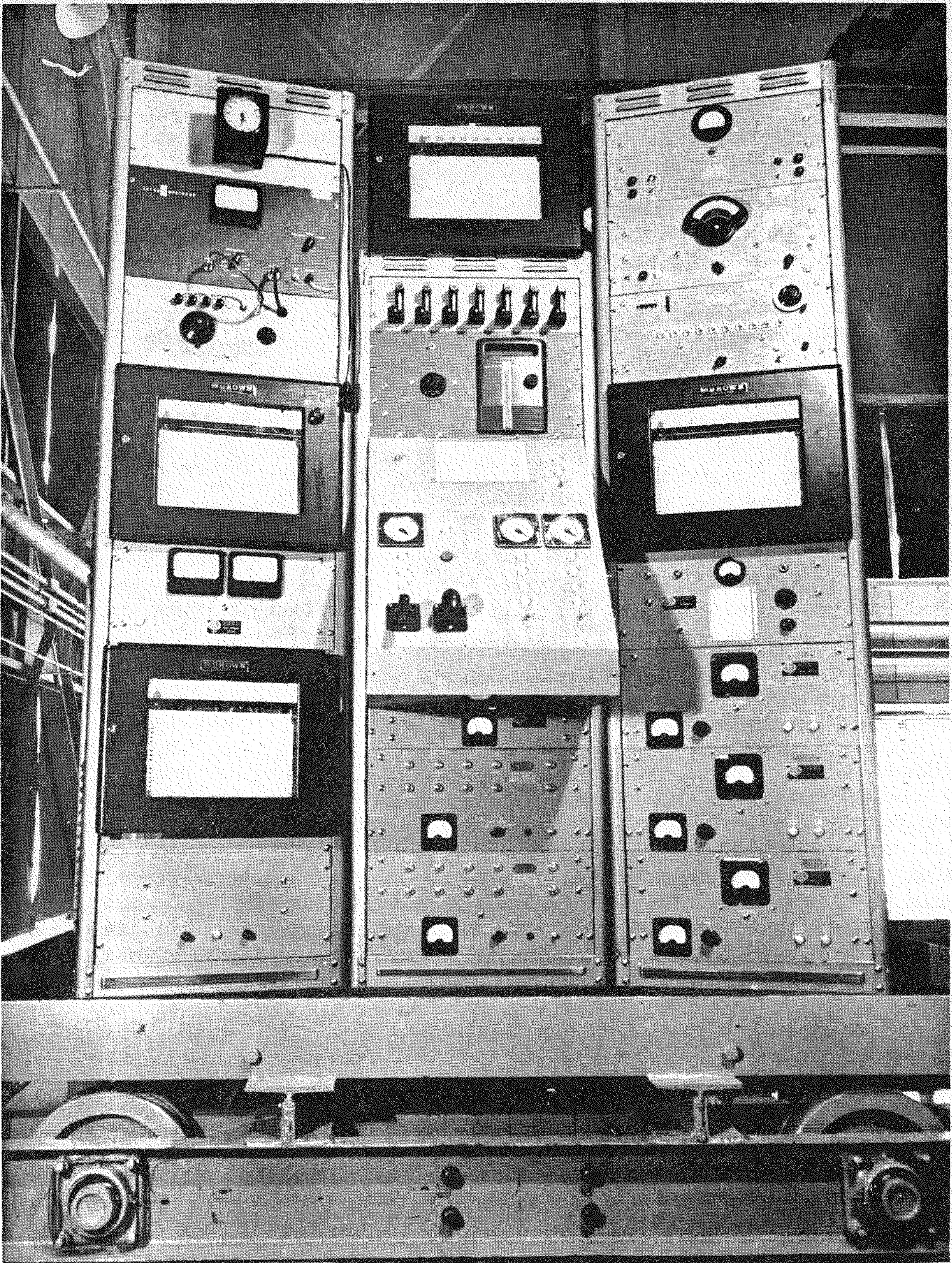
The safety rods are raised by three-phase AC motors which have a definite speed and insure against too rapid withdrawal. It requires about 3 minutes to raise the rods completely, which rate corresponds to an average change in k of 0.02 percent per second. The safety rods cannot be raised unless the control rod is all the way in, and the control rod cannot be operated until the safety rods are at least three-fourths withdrawn. This last 6 in. of travel is the "shim range" and indicated by lights on the control panel. Instructions to the operator require that the reactor be operated only when the shim rods are withdrawn at least three-fourths of the way. This insures that a negative δ k of 6 to 8 percent is always obtainable by dropping the rods. If fractional fuel assemblies are available, the reactor can be loaded to conform with this requirement. It is, of course, possible for the operator to disregard instructions deliberately and operate the reactor with a larger amount of possible excess δ k, but the interlocking described will prevent him from doing so inadvertently. An interlock, operated by the pointer of the Brown recorder in the count rate circuit, keeps the operator from raising the safety rods unless the count rate is more than two per second. This prevents start-up unless the source and the fission chamber are in place. In addition, the magnet current will be interrupted if the gate to the reactor bridge is opened, if a monitron fixed on the bridge shows more than a few mr/hr of gamma rays emerging from the water, or if one of the "scram" buttons located on the control panel and on the walls near the four corners of the pool is operated. When the magnet current circuit is completed, "Reactor On" signs located near the pool and above all the entrances are lighted automatically.

All the electronic equipment for operating and monitoring the reactor is contained in five relay racks which can be located next to the bridge. Three of these (Fig. 12) form the control panel, and the other two contain power supplies and the amplifiers for the control rod servo. The operating panel (the section with sloping front) has two switches for raising or lowering the safety rods, selsyn indicators to show their locations, a switch to raise or lower the fission chamber, a "manual" or "servo" switch to activate the automatic control system, and a switch to operate the control rod when the circuit is in the "manual" position. The "scram" button, in addition to dropping the rods, resets the safety circuit after it has been tripped by opening the gate on the bridge or operating one of the other scram buttons. The control circuits are interlocked to prevent the person in charge from inadvertently operating the pile in a dangerous manner.

The electromagnets which support the safety and control rods are made from one piece of Armco soft iron in an iron-clad design. The exciting coil has 4800 turns of #30 copper wire and is impregnated under vacuum after assembly in the magnet with Irvington Varnish and Insulator Company's Harvel "Oil Stop". This forms a waterproof and shock absorbing bulk insulator around the coil. Tests in the ORNL graphite pile indicate that this material is stable under

FIGURE 12

PHOTO 7750



REACTOR CONTROL PANEL

neutron and gamma ray radiations. As an added precaution against moisture, all joints are painted with Glyptal.

The magnets were designed to support the rods with an air gap of 0.005 in. when 30 milliamperes exciting current flows in the windings. The air gap is obtained by crowning the face of the armature. This crowning also makes perfect alignment between magnet and armature faces unnecessary, and hence no universal joint is supplied with magnet or armature. Tests indicate that the release time, with 50 percent more current than is required to hold the control rod, is of the order of 40 milliseconds. Since the exciting current is an inverse function of the reactor power level, the actual release time after the reactor passes the scram level is much less than this. The release time can be sharply reduced by laminating the magnetic circuit. This entails appréciable extra expense and, if fractional fuel elements are available so that the reactor can be loaded in such a way that it cannot reach a fast period, is not justified.

Provision is made for supplying inert gas (argon, nitrogen, and carbon dioxide) to the control chambers. The flow must be monitored so that the operator can be assured that the gas supply is not interrupted. This is accomplished by connecting two gas lines to each chamber and inserting a flowmeter in the outgoing line.

Abnormal Operation

The kinetic behavior of this reactor after an arbitrary stepwise increase in k of 2 percent above prompt critical has been studied by H. C. Claiborne, H. F. Poppendiek, and M. C. Edlung*. The value of 2 percent was chosen to bring these calculations in line with calculations of abnormal behavior of other reactors; we cannot imagine how this would occur in actual practice.

The calculations were based on the following conditions:

1. Initial water temperature 68°F and steady state reactor power of 1 kilowatt.
2. Rate of rise of power after supercritical condition is reached is proportional to $e^{100\,\theta}$ where θ is time in seconds.
3. When some 10 percent of the moderator has been expelled by steam, the reactor becomes subcritical and remains so until the steam condenses.

They find that, if all heat transferred to the water after the fuel plate surface reaches boiling temperature converts water into steam, the critical conditions will persist for 0.127 second. At that time, sufficient moderator will be expelled to make the reactor subcritical. The maximum fuel plate temperature (at the center of the plate) is 390°F. Because of lack of information on transient boiling heat transfer and rate of formation of steam bubbles, one cannot be sure that the reactor will follow this prediction.

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*See Appendix II, ORNL memo Central Files No. 51-8-176.

Delay in the production of steam bubbles will allow the plates to rise to a higher temperature. A relevant experiment has been performed recently by M. Untermeyer* at Argonne National Laboratory. Heavy electrical pulses were sent through thin-walled aluminum tubing immersed in a tank of water at room temperature. Ten, twenty, and thirty millisecond pulses resulting in heat fluxes of about 50 cal/cm^2-sec failed to melt the aluminum, but vaporized and ejected the water from the interior of the tube. It appeared that the delay in formation of the steam bubbles might be as little as 30 milliseconds from the time the power was applied. Our calculations indicate heat fluxes during "runaway" of the same order as those measured by Dr. Untermeyer, but his conditions are not identical with ours and results of his experiment should not be taken as positive indication that the fuel plates will not melt when the reactor is suddenly made 2 percent prompt critical.

Further analysis based on the assumption that boiling makes the reactor subcritical before the plates are damaged indicates that the reactor will oscillate with an average power level somewhat less than 200 kilowatts.

If the stepwise increase in prompt reactivity is substantially more than 2 percent, the fuel plates probably will melt during the first cycle. This is because of the extremely rapid rise in the rate of heat generation in the fuel plates. Most of the fission products thus released will be dissolved in the water.

It is difficult for us to envisage how an instantaneous increase in reactivity can be obtained. It is possible, however, to think of a condition where the safety rods have been (intentionally?) jammed with the reactor on a rising period, but less than prompt critical. If the rate of rise is sufficiently slow to permit the water between the plates to be warmed uniformly, the negative temperature coefficient may stabilize the reactor before the boiling point is reached. This coefficient is approximately 0.0075 percent per °F, so a 150°F rise in the exit water temperature, relative to the inlet temperature, will take care of 1/2 percent of excess reactivity. If the excess reactivity is greater than this, or the rate of rise too fast, the water will boil before the reactor reaches 500 kilowatts and the reactor will probably oscillate as described before. If the calculations indicating oscillation is at an average power level of 200 kilowatts are correct, personnel standing near the pool will not receive an overdose. No measurements have been made of the oxygen activity, O^{16} (n,p)N^{16}, but calculations of the rate at which the activated water diffuses to the surface do not vitiate the above conclusions.

Life of Fuel Elements

Fuel (U^{235}) is consumed at a rate of about one gram per 24,000 kilowatt hours of operation. Presumably, the fuel elements would be reprocessed when 5 to 10 percent of the fuel has been consumed. In other words, the maximum life is a little less than one megawatt year. At power levels no greater than 100 kilowatts, the effect of poisoning due to fission products can be neglected.

Unless care is taken, the life of the fuel element may be determined by corrosion. At ORNL, it has been found that the addition of 50 parts per million

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*ANL-WPB-60

by weight of sodium chromate to the process (drinking) water used in the pool inhibits this pitting*. An an added precaution, all the fuel elements are alodized by a hot dip process patented by the American Chemical Paint Company. In essence, this treatment deposits a thin aluminum oxide-chromate coating on the surface of the fuel element. The alternative to treating the water and the fuel elements is to provide a supply of demineralized water.

Costs

Exact costs as of the summer of 1950 were determined from ORNL records. Breakdown of costs pertinent to the reactor are as follows:

20 Fuel Elements at $120 each (exclusive of cost of enriched uranium)	$ 2,400
Reactor Assembly; Labor, Overhead, and Materials (Motors, magnets, grid superstructure, etc.)	28,000
Electronic Circuits; Labor, Overhead, and Materials (Chambers, circuits, recording instruments, etc.)**	28,000
Total: Reactor and Controls	$58,400

In addition, the following equipment is very desirable:

BeO Reflector (30 elements)	$17,000
Servo Automatic Control	2,600
Spares (Chambers and Electronic Equipment)	9,000
Health Physics Instruments (See Appendix I)	7,500
Total	$36,100
Grand Total	$94,500

The cost of the building and pool will depend on the design, materials, difficulty of excavating, etc. Possibly, the most convenient location for this facility is on the side of a hill or a bank. Our estimate of the cost of a

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*See Appendix IV

**This complete equipment is now available commercially at approximately this cost. However, the cost will be greater if switchboard mounting, console operation, etc., is desired.

building with a pool 14 ft by 18 ft by 22 ft deep, a bay 28 ft by 40 ft and 30 ft high containing the pool and reactor, and 2500 sq ft of laboratory space is $125,000. This is the bare minimum. To these costs must be added the expenses of additional facilities necessary for whatever experimental program is desired.

The relative inexpensiveness of the reactor described in this report is the result of several design considerations. Among these are the facts that we are able to employ convection cooling, that the same water is used for shielding as well as cooling, and that the gaseous fission products are confined to the fuel assemblies, removing the need for a stack. Also, because there is negligible "poisoning" to deal with, the control and safety circuits can be relatively simple.

Were this type reactor to be designed for operation in the megawatt region, the costs would go up by, perhaps, a factor of ten. Forced circulation of coolant must be supplied, more shielding is necessary, control and safety circuits must be expanded to take care of the excess reactivity necessary to overcome poisoning, etc. The installation then becomes a major engineering task.

Acknowledgements

As mentioned earlier, the reactor and control system described is a slightly modified version of a reactor now installed at ORNL. A large number of people from the ORNL staff contributed to the design and construction of the facility presently operating. A few of the major contributors are listed below.

The original design was investigated by E. P. Wigner, A. M. Weinberg, H. Soodak, E. Greuling, and others. Critical experiments to confirm the design were carried out by M. M. Mann. The control and safety circuits were proposed by H. W. Newson and P. R. Bell, and designed and built by W. H. Jordon, T. E. Cole, J. E. Owens, E. P. Epler, and others. The design upon which this report is largely based was developed by W. R. Gall, who was aided by operational investigations made by S. E. Beall.

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APPENDIX I

Health Physics Instruments for the Low Cost Reactor Installation

by

T. H. J. Burnett - Health Physics Division, ORNL

Purpose

The primary objective for health physics radiation instrumentation is to facilitate, and insure as far as possible, safe operation and experimental usage of the reactor. The particular aspect of Health Physics concern is adequate protection of personnel against over-exposure to radiation. It is necessary to be able to determine accurately the intensities of several different hazardous types of radiation at different sections of the area of training pile operations.

The usage of health physics instruments for making experimental measurements or that of experimental instruments for health physics measurements is not improper in principle, but has certain disadvantages which tend to be offset by practical considerations of economy, often paramount to schools. These disadvantages are considered later under "Responsibilities".

Herein are discussed both permanently mounted instruments and the portable instruments used by the personnel responsible for health physics. The usage of these should be part of a routine for standardized surveys, a practice which will insure (1) frequent check of the proper operations of the instruments (2) practice and familiarity in the usage of these instruments (3) recognition of any contamination or radiation hazard otherwise overlooked or unknown.

Requirements

The radiation detection system may include some or all of the following as components: primary detecting elements, amplifiers, indicators, alarms, and recorders. These must be designed so that they will operate under all normal conditions, made up so as to thwart the tampering impulses of students, and designed to "fail safe". Alarm or scram circuits should be activated if the instrument is not in proper operating condition.

The indications of radiation intensity must be positive, unambiguous, and directly intelligible in terms of permissible dosage rate or exposure, consistently reproducible, and reliable. A record of the variation of radiation intensity with time for all the locations of concern is required, and for the fixed instruments can be prepared automatically.

The operating area and environs should be monitored by instruments equipped with alarms which will operate whenever the radiation exceeds the safe maximum.

Economy of construction, simplicity of operation, and ease of maintenance are all even more important for school use instruments than ordinary commercial requirements.

Location

While one cannot specify precise sites, the following will in general apply. Neutron intensities will be present solely in the vicinity of the pile and tank itself, hence one monitron with boron lined chamber should be installed so that the chamber is between the pile operator and the pile. Being sensitive both to gamma and to slow neutrons, it should alarm when the radiation exceeds tolerance and be interlocked to scram the pile at some small multiple of tolerance determined by operating time expected. A second such monitron should be positioned at the next closest populated area of observers, workers, lab or office personnel, etc. This provides an added safeguard in event of failure or proper operation by the first.

Contamination by leakage and/or induced activity in impurities of the tank water can most economically be monitored by routine sampling and counting.

For surveying for fast neutrons during certain experiments, one of the hydrogen recoil proportional counters as developed by G. S. Hurst (ORNL-930) is recommended.

For slow neutron surveys during operations a pair of electroscopes are probably both simplest and cheapest. One has a boron coated chamber, and the other can be used for quantitative gamma measurements also.

A beta-gamma ion chamber survey meter (cutie pie) will be very convenient for quick gamma measurements during operations and for evaluating the hazards from irradiated samples.

A beta-gamma probe type instrument using a G. M. tube will be highly worthwhile for scanning and the detection of contamination of hands, body, clothing, floors, equipment, etc., chiefly from spills of irradiated material and contaminated tank water.

To insure against undetected leakage of alpha emitting materials, a stable alpha poppy (as in ORNL-602) is recommended and should be used for part of the routine surveys necessary.

Suitable film badges and pocket meters must be worn at all times.

Most of these instruments may be obtained from commercial suppliers. Some of them can be constructed easily in a school laboratory. The ORNL Health Physics Division and the Instrument Division will be glad to advise on specific problems.

Responsibilities

While everyone connected in anyway with reactor experiments must have a strong sense of joint and individual responsibility for his own and others' safety, it is mandatory that one person explicitly have the chief health physics responsibility and proper associated authority. Such a one cannot also have his prime job that of experimentation.

Hence arises one difficulty from having a single set of instruments for the dual usage of radiation safety personnel and experimenters, the portable instruments are prone to be in simultaneous demand. Further difficulties appear when the responsibility for breakage, repair, and misuse is to be fixed, and again when the chore of calibration is to be performed.

A duplication of at least the beta-gamma ion chamber and probe instrument is necessary, and may be desirable as well in the case of other portable instruments so that experiments need not be delayed or cut short by instrument failure. Similar reasoning will justify a spare monitron for standby.

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APPENDIX II

Transient Thermal Behavior of Low Cost Reactor When Prompt Critical

by

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Summary

This memorandum is concerned with a numerical heat transfer analysis of a water cooled aluminum alloy reactor. The boundary conditions imposed are that the fuel plate heat source suddenly increases exponentially with time. After about 10 percent steam voids are generated in the moderator, the reactor becomes subcritical. Upon the collapse of the steam bubbles, the process repeats itself with some modification because of the higher initial temperatures.

The differential heat transfer equations for the system were transformed to finite difference equations and the temperature solutions were evaluated numerically.

Two specific analyses were undertaken. Analysis No. 1 was characterized by the initiation of boiling as soon as the wall-coolant interface temperature reached the boiling temperature. Also, all heat subsequently transferred produced steam bubbles which did not collapse. In analysis No. 2, boiling heat transfer occurred in the presence of sufficient subcooling so that 10 percent steam voids were never formed before the melting temperature of the fuel plate was reached.

Introduction

This group was requested to predict the transient temperature behavior of a fuel element of the low cost reactor under the abnormal condition that the reactor is suddenly made supercritical by two percent in excess of prompt critical. This value has been used in similar calculations of the abnormal behavior of other types of reactors. It is postulated that the power density rises at an exponential rate given by $e^{100\,\theta}$, where θ is time in seconds. When about 10 percent of the moderator has been replaced by the generated steam, the reactor becomes subcritical.

The heat generation term then falls to a low value, rising again after the steam has condensed. The semi-periodic or transient temperature history within the reactor under such conditions was desired.

The cooling heat transfer mechanisms at the fuel plate-water interface control the thermal structure within the reactor. Cooling is initially achieved by non boiling convection circulation. After the wall surface temperature has attained the boiling temperature, cooling may be achieved by boiling heat transfer if no superheating occurs and if bubbles can be formed under the rapid transient conditions that characterize the problem under consideration. Boiling research studies (reference 1 at end of this appendix) have indicated that it is possible to obtain superheats of the order of 100°F for subcooled boiling systems. The time required for a bubble to grow under transient heating conditions is not known. High speed photography studies relative to bubble behavior under steady state heat transfer conditions (reference 2) indicated that a bubble will grow and collapse in about 0.001 seconds; bubble growths under transient conditions could possibly be much slower. Except in some specific cases, it is generally not known how much vapor volume exists in a boiling system characterized by certain surface and bulk fluid temperatures. The amount of vapor present is not simply related to the heat transfer rate in a subcooled system because of the continuous steam condensation in the relatively cool coolant. Although significant vapor formation will probably accompany the boiling-cooling mechanism, the question is, might there be a sufficient delay in bubble formation such that the melting temperature of the fuel plate is reached before cooling becomes effective?

In view of these unknowns the problem under consideration can really be bounded only by limiting cases. The optimistic limiting case is that boiling begins as soon as the wall-coolant temperature reaches the boiling temperature and all heat subsequently transferred produces steam bubbles which do not collapse. The pessimistic limiting case is that boiling never occurs (before the fuel plates melt) because of the high degree of superheat.

Two heat transfer analyses are presented here. Analysis No. 1 was made for the optimistic case described above. Analysis No. 2 was made for the intermediate case of boiling after the boiling temperature is obtained but without the formation of 10 percent vapor because of subcooling.

Fuel Assembly Description

A typical fuel assembly unit consists of four parallel aluminum-uranium fuel plates spaced three-fourths of an inch from center to center. Each fuel plate is 24 inches long and three inches wide. It is composed of a sandwich of an 80 percent aluminum - 20 percent uranium alloy (0.060 inches thick) between two 0.020 inch plates of aluminum. The fuel elements are cooled by convective water flow.

Heat Transfer Equations

In view of the large length to thickness ratio of the fuel plate, the heat flow is essentially unidirectional (radially from within the source region out through the aluminum layers to the coolant).

The differential equation describing transient conduction can be represented by the following finite difference equation (reference 3). It assumes that time is measured in hours rather than seconds.

$$t_{n, k+1} = \frac{t_{n+1, k} + t_{n-1, k}}{2} + \frac{G(\theta,x)\Delta\theta}{C_p \gamma} \qquad (1)$$

where,

$$\frac{2a\Delta\theta}{\Delta x^2} = 1$$

t = temperature at any point, °F

a = thermal diffusivity, ft^2/hr

$\Delta\theta$ = time, increment, hr

Δx = distance increment, ft

x = lateral distance from fuel centerline, ft

$G(\theta,x)$ = heat generation function, Btu/hr ft^3

C_p = heat capacity, Btu/lb °F

γ = density, lb/ft^3

n refers to the nth distance increment

k refers to the kth time increment

The heat teneration term for a standard initial power level of one kilowatt in a reactor of 12 fuel elements and a two percent excess reactivity above prompt critical is,

$$G(\theta,x) = \frac{1 \times 3413}{4 \times 12 \times 3 \times 24 \times 0.06 \times 5.787 \times 10^{-4}} e^{360{,}000\,\theta}$$

$$= 28{,}430\, e^{360{,}000\,\theta} \qquad (2)$$

It is postulated that $G(\theta,x) = 0$ in the aluminum layers on either side of the fuel.

The boundary conditions are as follows: The heat transferred across the fuel layer centerline is zero. This fact can be represented in finite difference form as follows:

$$\left(\frac{\Delta t}{\Delta x}\right)_{x = 0} = 0 \qquad (3)$$

The heat flow from the uranium alloy into the aluminum layers may be expressed in terms of the corresponding thermal conductivities and the temperature gradients at the two sides of the interface. This equation expressed in finite difference form is

$$t_{n,k} = \frac{\dfrac{k_u \overline{\Delta x_a}}{k_a \overline{\Delta x_u}} \; t_{n-1,\,k} + t_{n+1,\,k}}{1 + \dfrac{k_u \overline{\Delta x_a}}{k_a \overline{\Delta x_u}}} \qquad (4)$$

where, u refers to the uranium alloy,

a refers to the aluminum,

k = thermal conductivity, Btu/hr ft^2 (^oF/ft),

x = distance increment, ft.

Similarly, the equation for the temperature at the aluminum-water interface is

$$t_{n,k} = \frac{\dfrac{h \overline{\Delta x_a}}{k_a} \; t_{f,k} + t_{n-1,\,k}}{1 + \dfrac{h \overline{\Delta x_a}}{k_a}} \qquad (5)$$

where, h, heat transfer coefficient, Btu/hr ft^2 ^oF

t_f, mixed mean water temperature, ^oF

The value of the heat transfer coefficient, h, depends upon the cooling mechanism. It was postulated that laminar flow existed between the fuel plates previous to the initiation of boiling(1). It can be shown that the heat transfer coefficient for such a system under uniform wall flux conditions is (see Appendix 1)

$$h = \frac{35}{17} \frac{k_f}{r_o} \qquad (6)$$

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(1) Actually, the velocity profile is somewhat more complex than the parabola. However, since the major temperature changes occur during the period when boiling is the transfer mechanism, this simplification is justified.

where, k_f = thermal conductivity of water, Btu/hr ft^2 (°F/ft)

r_o = half the distance between fuel plates, ft.

When subcooled boiling takes place, the heat transfer coefficient can be expressed as (reference 4)

$$h = \frac{0.074\,(t_{i,\,k} - t_{sat})^{3.86}}{(t_{i,\,k} - t_{f,\,k})} \tag{7}$$

where t_i is the aluminum-water interface temperature and t_{sat} is the saturation temperature. This equation represents experimental heat transfer data for steady state subcooled boiling in annuli for water flows in the range of 1 to 10 ft/sec.

Numerical Solution Procedure

The numerical method of solution of the equations presented above can be found in reference 3. The initial temperature distribution in the reactor was considered to be 68°F. The boiling temperature for the specified system was 233°F. The physical properties of uranium and the uranium alloy are given in Appendix 2.

Results and Discussion

Analysis No. 1 (boiling begins as soon as the wall-coolant temperature reaches the boiling temperature and all heat subsequently transferred produces steam bubbles which do not collapse) indicated that after 0.127 seconds, 10 percent steam was produced in the coolant passage. The maximum fuel plate temperature (at the uranium alloy center) at that time was 391°F.

Analysis No. 2 (boiling occurs but subcooling prevents the formation of the 10 percent vapor before melting temperatures are attained) indicated that after approximately 0.14 seconds the uranium alloy center temperature will attain the melting temperature. The lateral temperature distribution after 0.139 seconds is shown in Figure 1; the time temperature histories of the fuel plate at the centerline, the alloy-aluminum interface, and the aluminum-water interface are shown in Figure 2.

In order that a more satisfactory heat transfer analysis be conducted, it will be necessary to obtain fundamental information on the transient boiling heat transfer outlines in previous paragraphs.

Some transient boiling experiments have been conducted by S. Untermyer (reference 5). A tube which was filled with water was heated exponentially with time; high speed photographs of water ejection upon the initiation of boiling were obtained. Apparently, however, insufficient general information on bubble density and lag in bubble formation was obtained.

The exponent of the heat generation function is not a constant, as used in the calculations, but is actually a function of density. This simplification is on the pessimistic side because the reactivity decreases as the temperature of the moderator increases. An increase of 10°F in the average water temperature will reduce the exponent from 360,000 θ to 349,000 θ. It was considered unnecessary to complicate the analysis by including this effect.

The numerical analyses which have been conducted have revealed the existence of high temperature gradients within the thin fuel plates. It is suggested that thermal stress calculations be made to determine the magnitude of the stresses within the fuel elements.

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Heat Transfer Coefficient for Laminar Flow Between Parallel Plates with Uniform Heat Flux

The case of laminar flow between parallel plates with uniform heat flux from the walls is a good approximation to the actual system.

The radial heat transferred, dq, to an elemental strip of water, a distance r from the centerline, is

$$d^2q = u\ w\ dr\ \gamma\ C_p \left(\frac{\partial t}{\partial z}\right) dz \qquad (8)$$

The total heat transferred is

$$dq_t = w\ \gamma\ U\ r_o\ C_p \left(\frac{\partial t}{\partial z}\right)_{mean} dz \qquad (9)$$

where dq_t = heat transferred at the wall, Btu/hr.

U = average velocity, ft/hr.

u = velocity at any point, ft/hr.

w = fuel plate width, ft.

r_o = one-half the distance between plates, ft.

z = distance along length of fuel plate, ft.

For uniform flux beyond the entrance region,

$$\frac{\partial t}{\partial z} = \left(\frac{\partial t}{\partial z}\right)_{mean} \qquad (10)$$

For laminar flow between parallel plates,

$$u = \frac{3}{2} U \left(1 - \frac{r^2}{r_o^2}\right) \qquad (11)$$

Combining equations (8), (9), (10), and (11), and integrating between the limits of q and r,

$$dq = \frac{dq_t}{2} \left(\frac{3r}{r_o} - \frac{r^3}{r_o^3}\right) \qquad (12)$$

at any point,

$$\frac{dq}{dA} = - k_f \frac{\partial t}{\partial r} \tag{13}$$

where $dA = wdz = \text{constant}$ (14)

Combining equations (12 and 13) and integrating between the limits of t and r,

$$t - t_w = - \frac{1}{k_f} \left(\frac{dq_t}{dA}\right) \left(\frac{3r^2}{4r_o} - \frac{r^4}{8r_o^3} - \frac{5r_o}{8}\right) \tag{15}$$

where t_w = surface temperature, °F

The mixed mean temperature, t_f, is defined by

$$t_f = \frac{\int_o^{r_o} u\, t\, w\, dr}{U \int_o^{r_o} w\, dr} = \frac{3}{2r_o} \int_o^{r_o} t \left(1 - \frac{r^2}{r_o^2}\right) dr \tag{16}$$

Combining equations (15) and (16) and integrating,

$$t_f = - \frac{17}{35} \frac{dq_t}{dA} \frac{r_o}{k_f} + t_w \tag{17}$$

The heat transfer coefficient, h, is defined by

$$h = \frac{1}{(t_w - t_f)} \frac{dq_t}{dA} \tag{18}$$

Substituting for t_f,

$$h = \frac{35\, k_f}{17 r_o} \tag{19}$$

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5. S. Untermyer, Personal Communication, August 14, 1951

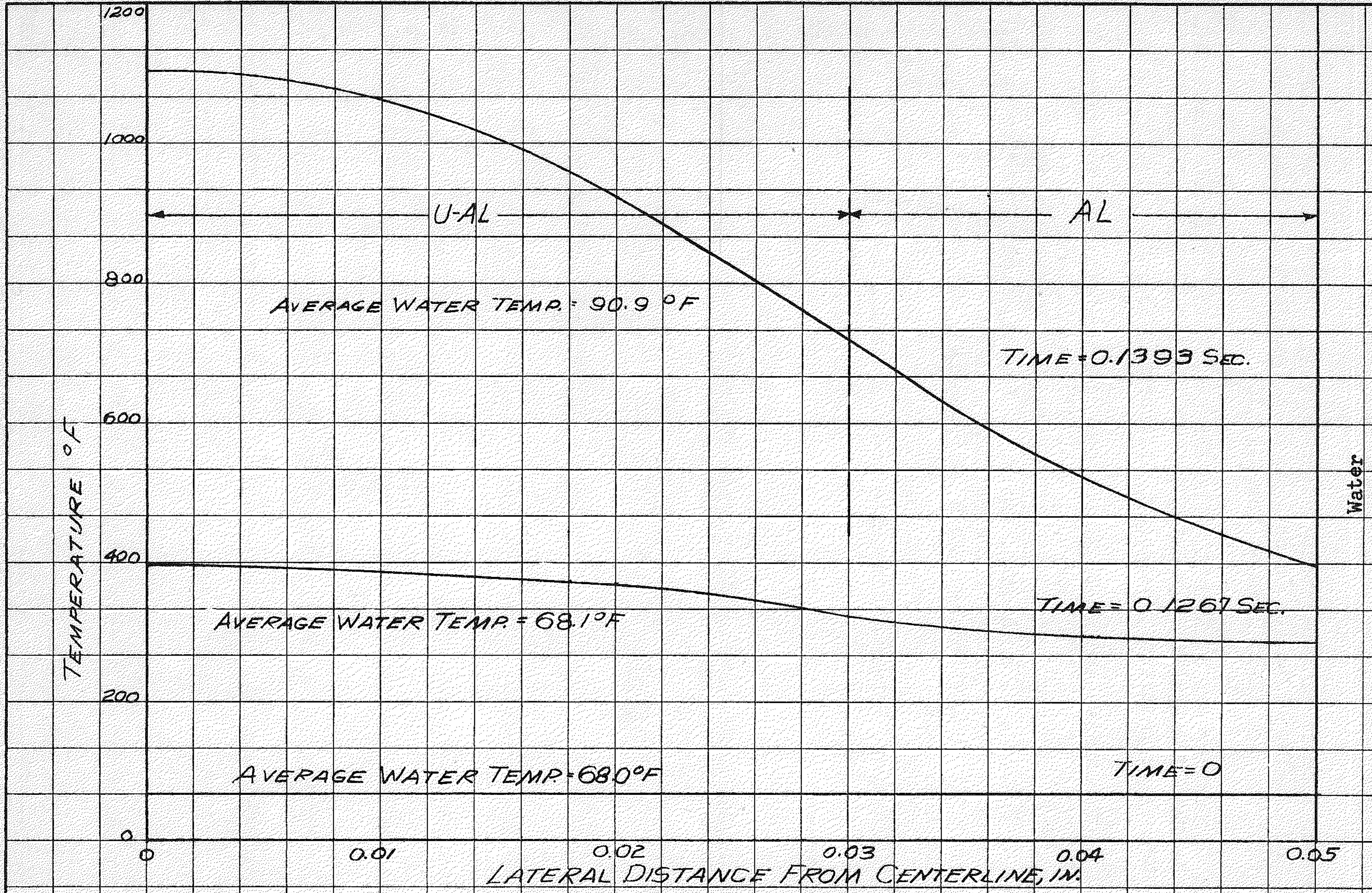

FIGURE 13

LATERAL TEMPERATURE DISTRIBUTION OF FUEL PLATE

Drawing #12449
Unclassified

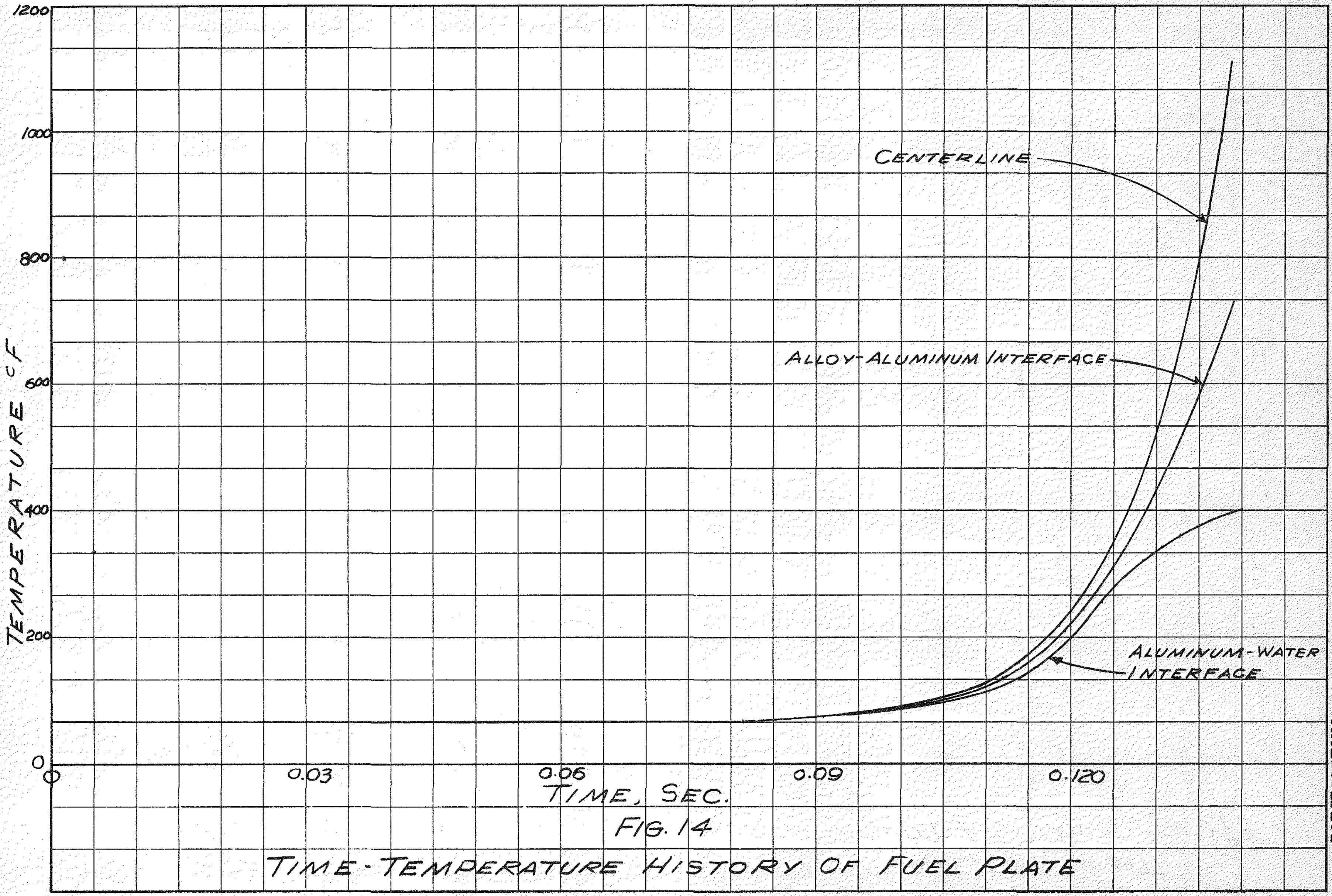

FIG. 14

TIME-TEMPERATURE HISTORY OF FUEL PLATE

Drawing #12450
Unclassified

THERMAL CONDUCTIVITY OF ALLOY (22.7% U-77.3% AL) FROM M-3477
HEAT CAPACITY OF ALLOY COMPUTED FROM WEIGHTED AVG. OF VALUES FOR U & AL.
HEAT CAPACITY OF U FROM MDDC-1499

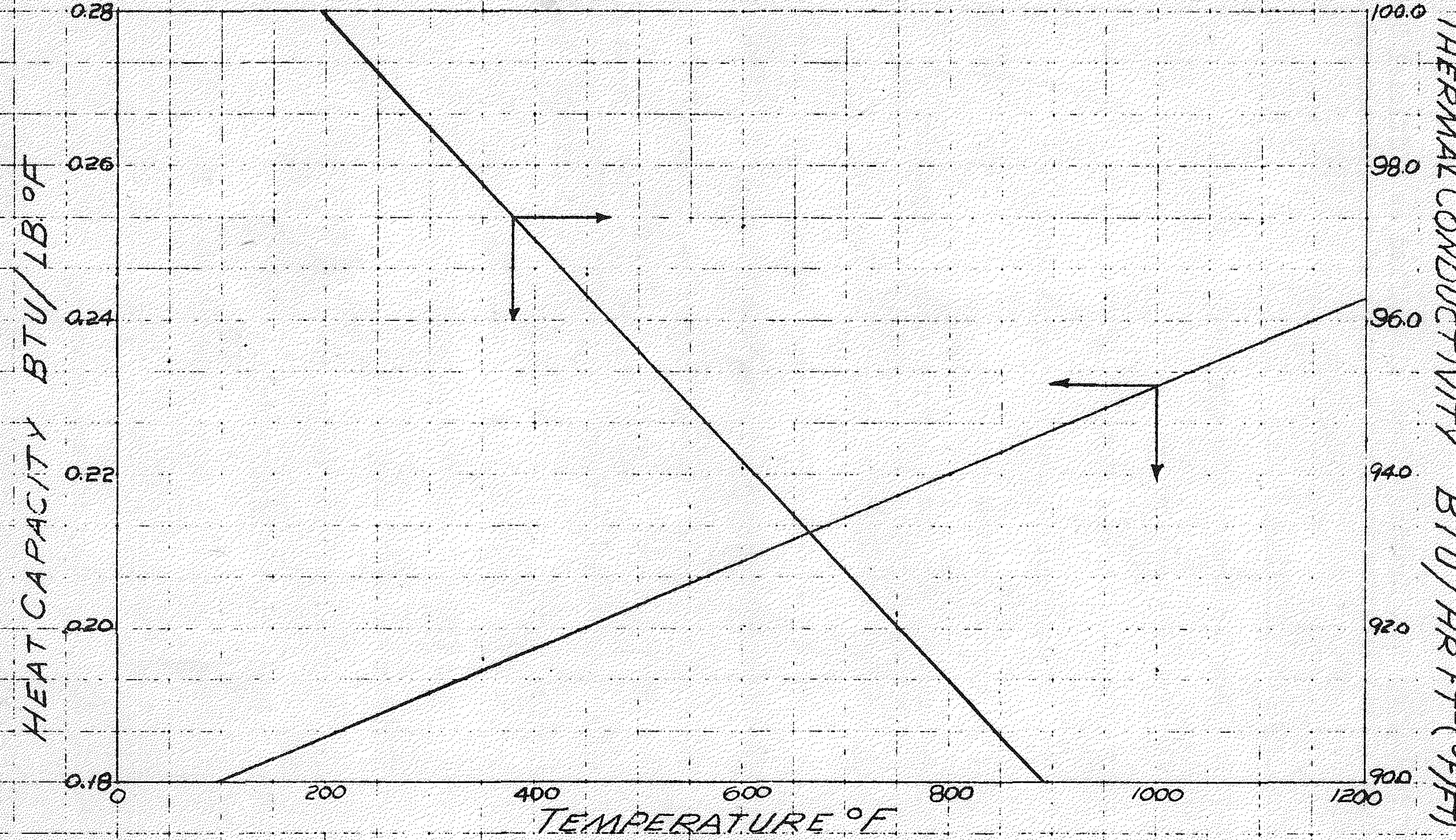

FIGURE 15

PHYSICAL PROPERTIES OF 20% U-80% AL. ALLOY

Drawing #12451
Unclassified

APPENDIX III

The following ORNL drawings (some classified as "Official Use Only" and some unclassified) describe the main features of the reactor and control circuits.

1. Reactor and Reactor Bridge

E-7201	Reactor Bridge Assembly
D-7202	Grid Plate
D-7208	Control Rod and Safety Rod
D-7209	Lifting Magnet-Assembly and Details
D-7211	Reactor Suspension Frame
D-7212	Waterproof Containers for Counter Chambers
D-7213	Safety Rod Shock Absorber
D-7214	Reactor Bridge Frame
D-7215	Reactor Bridge Superstructure
D-7216	Motor Drive for Reactor Controls - Assembly
D-7217	Motor Drive for Reactor Controls - Detail Sheet #1
D-7218	Motor Drive for Reactor Controls - Detail Sheet #2
D-7221	Reactor Assembly Tool - Assembly
D-7222	Reactor Assembly Tool - Details
A-7223	Molded Block (BeO)
A-7224	Section of Drawn Tubing for Cans
D-8238	Power Plan - Reactor Bridge
D-8239	Power Details - Reactor Bridge Sheet 1
D-8240	Power Details - Reactor Bridge Sheet 2
D-8170	Fuel Assembly Storage Rack

2. Reactor Control Circuits and Instruments

Q-1045-1	4 in. Compensated Ion Chamber
Q-975-1B	Neutron Chamber 3 in. PCP Model 2
Q-1092-1	Fission Chamber
Q-1058-1	Gamma Ion Chamber
Q-541-A-E	A1 Amplifier and A1A Preamplifier
Q-834-1	1024 Scaler (modified)
Q-751 A	Log Count Rate Meter
Q-826-1	Low Drift Electrometer
Q-995-1	Power Supply for Compensated Ion Chamber
Q-915-1	Log N Amplifier (Pile Period Meter)
Q-947-1	Sigma Amplifier and Safety Preamplifier (modified)
Q-889-1	Magnet Amplifier
Q-1054	Control System Circuits

3. Instruments

Q-369	Neutron Counter 1" O.D. - 8" Long
Q-804	Neutron Chamber 2" O.D. - 12-1/2" Long
Q-846-4	Fission Chamber 1/2" Dia.
Q-846-5	Fission Chamber 1/2" Dia.
Q-1059	Fission Chamber 3" Dia.
Q-961	Proportional Counter (Brass Case 1" O.D. x 13-5/8")
Q-1058	50 cc. Ion Chamber
Q-1057	900 cc. Ion Chamber
Q-954	Watertight Housings

APPENDIX IV

Corrosion Studies on Aluminum Clad Reactor Fuel Element

by

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Abstract

The investigation of corrosion on aluminum clad reactor fuel elements in filtered water is reported.

The study included two basic types of protection: Element pretreatment, by either anodizing or alodizing; and solution control, using nitric acid to maintain a pH of 5.5 to 6.5 or the addition of 60 ppm sodium chromate as an inhibitor.

No pretreatment was effective in stopping pitting attack for more than four months as a maximum. Control of the pH tended to reduce pitting attack but did not stop it. However the addition of sodium chromate was found to stop all pitting attack and at the same time reduce the overall corrosion rate to essentially zero.

Introduction

In August of 1950 the Corrosion Group was asked to investigate the effect of filtered water as a corrosive media on aluminum clad fuel elements. Plans had originally called for demineralized water but the additional cost involved seemed excessive. Consequently the investigation of filtered water was begun. The need for some passivation treatment or protective measure was known before the tests were started and various alternatives were included in the original group.

The construction and start-up schedule of the subject reactor was so far advanced at the time of the decision to use filtered water that only two week corrosion tests were available when it was decided to use alodizing as the pretreatment for added protection. The inherent risk involved in using short term tests as a basis for long term exposures was realized and consequently the tests were continued for a period of some months. Both the short term and the longer term tests are reported here.

Test Procedure

Corrosion Media

Filtered water as supplied to the Laboratory was used in these tests. Table I contains an average analysis together with the maximum and minimum values obtained from analysis of this water over a five month period.

In addition to these analyses certain components were also checked in the water at the end of one month exposure when the water was changed. Results of these analyses are listed in Table II. The only significant figures from these data are the marked rise in aluminum content in the final solutions in which unprotected 2S aluminum and 2S-72S aluminum were exposed. The reason for the similar pick up in aluminum during the first two months of exposure in the Na_2CrO_4 treated water is probably due to a slight attack involved in establishing the passive surface.

In only three tests were additions made to the water; these were Na_2CrO_4 - 60 ppm, K_2CrO_4 - 60 ppm, and HNO_3 sufficient to lower and maintain the pH in the 5.5 to 6.5 range.

Test Materials - Types of Aluminum

Three classes of aluminum are involved in the make up of the fuel assemblies. These classes together with typical composition are:

Type	Composition %				
	Al	Zn	Si	Cu	All Others
2S	99 +	0.10 max	0.25	0.20 max	0.05
72S	Bal	1.0	-	0.1 max	0.05
Brazing alloy	Bal	0.10 max	11.5	0.20 max	0.05

TABLE I

Filtered Water Analysis From Five Samples Taken at One Month Intervals From August Through December 1950

Concentration of All Constituents in Sample	Units	Ave.	Max.	Min.
pH	pH	8.0	8.3	7.4
ppm. Alkalinity as $CaCO_3$	ppm	5.0	10.0	0.0
M.O. Alkalinity as $CaCO_3$	ppm	98.0	106.0	95.0
Specific Resistance, 25°C	ohm-cm x 10^3	6.92	8.26	4.95
Soap Hardness as $CaCO_3$ detn.	ppm	112.0	120.0	106.0
Soap Hardness as $CaCO_3$ calc.	ppm	101.0	117.0	90.0
Dissolved CO_2	ppm	3.0	6.0	0
Dissolved Solids	ppm	138.0	164.0	112.0
Non-Volatile Solids	ppm	80.0	131.0	35.0
SiO_2	ppm	5.3	9.2	2.7
Fe	ppm	0.02	0.08	0.01
Al	ppm	0.08	0.14	0.02
Cu	ppm	0.02		
Ni	ppm	0.02		
Cr	ppm	0.02		
Ca	ppm	30.0	34.0	27.8
Mg	ppm	7.2	8.6	4.7
Na	ppm	9.6	14.0	6.6
SO_4	ppm	25.1	39.6	5.4
Cl	ppm	4.2	5.0	3.3
CO_3	ppm	58.2	63.6	56.4
HCO_3	ppm	118.8	129.3	114.7
NO_3	ppm	1.1	2.3	0.4
PO_4	ppm	0.8	1.7	0.2
F	ppm	0.02		

TABLE II

Comparative Analysis Before and After 30-Day Intervals of Corrosion Testing

Sample	pH	Cl ppm	Al ppm	Resistivity ohms x 10^3
Initial	8.3	5.0	.09	8.26
a* - 1F	8.2	4.4	0.3	7.69
b - 1F	8.2	3.8	2.1	6.0
c - 1F	8.3	4.2	0.24	6.96
d - 1F	8.4	3.9	0.23	6.94
e - 1F	8.5	4.2	0.5	7.80
Initial	8.3	4.4	.14	8.26
a - 2F	8.1	4.3	0.3	7.69
b - 2F	8.2	4.5	1.9	5.31
c - 2F	8.4	3.9	0.29	7.08
d - 2F	8.3	3.9	0.2	6.94
e - 2F	8.5	3.9	0.6	8.13
Initial	7.7	4.6	0.1	7.84
a - 3F	8.3	4.3	1.2	7.8
b - 3F	8.4	4.2	0.25	4.93
c - 3F	8.1	2.4	0.23	6.94
d - 3F	8.3	4.8	0.33	7.17
e - 3F	8.4	3.8	1.5	6.72
Initial	7.4	3.3	0.7	4.95
a - 4F	8.0	4.4	1.0	4.95
b - 4F	8.0	4.4	0.15	3.41
c - 4F	7.6	3.7	0.3	5.17
d - 4F	8.0	2.8	0.24	4.4
e - 4F	7.9	4.4	0.64	5.28
Initial	7.7	3.8	0.02	5.31
a - 5F	8.2	4.3	0.7	4.76
b - 5F	8.0	2.2	0.32	3.54
c - 5F	8.2	2.9	0.16	5.07
d - 5F	8.0	5.7	0.05	4.96
e - 5F	8.1	4.2	1.4	5.31

*a - 2S Aluminum
b - 2S Aluminum (Na_2CrO_4)
c - 2S Aluminum (Standard Alodized)
d - 2S Aluminum (Anodized)
e - 2S-72S Aluminum
1F = First month final solution
2F = Second month final solution, etc.

Pretreatments

In all, three classes of pretreatment for enhanced corrosion resistance were tested. These were Anodizing, Standard Alodine*, and Hot-Dry Alodine*.

Anodizing was the straight anodic treatment of 2S aluminum sheet. The sheared edges of the coupons were protected with a high temperature wax. The protection is from the extra thick oxide coat that is formed by the anodic treatment.

The alodized samples were of two different varieties; simple circular samples of 2S aluminum were given both the Hot-Dry and Standard Alodizing procedures, while a mock assembly involving two sheets of 72S clad 2S aluminum brazed together with 11.5% silicon alloy were only hot-dry alodized. These samples were alodized by Mr. R. P. Spruance of the American Chemical Paint Company. The protective coating is a mixed aluminum oxide-chromate-phosphate type of film.

Samples marked 72S-2S refer to roll bonded sheets of 72S aluminum on one side and 2S aluminum on the other side. Thus both types were exposed to the water during these tests.

Test Operation

The actual testing consisted of suspending a dried and weighed specimen of known area on a glass hook so that it was totally immersed in four liters of filtered water in a covered three-neck, five-liter distilling flask.

Samples were removed monthly, dried, weighed and photographed before being returned to test. The water was changed at the inspection period.

The defilmed sample weights were obtained only at the end of the test period. This defilming process involves the use of a 5% by volume phosphoric acid - 4% by weight chromic acid solution at 50°C for 20 minutes. The principal purpose of the defilming process was to make possible an accurate count and investigation of the pits.

Results of Initial Two-Week Tests

As the time schedule involved did not permit a thorough investigation before some protective treatment was decided upon, the original tests were only two weeks in duration.

The results of these tests are given in Table III.

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*Alodine - Patented process of the American Chemical Paint Company.

TABLE III

Two-Week Tests in Filtered Water at Room Temperature

Sample	Inhibitor	Weight Change (MDM*)	Pit Count Per Sq. Cm.	Maximum Pit Depth, mils
2S Al	---	- 14.8	0.15	1.0
2S Al	---	- 9.2	Neg	Neg
2S Al	Na_2CrO_4 60 ppm	+ 3.0	Neg	Neg
2S Al	Na_2CrO_4 60 ppm	+ 3.0	Neg	Neg
Std. Alodized 2S Al	---	+ 3.6	Neg	Neg
Std. Alodized 2S Al	---	Neg	Neg	Neg
2S-72S Al	---	- 47.2	0.2	2.6
2S-72S Al	---	- 43.6	0.25	6

* Milligrams per square decimeter per month

The untreated 2S aluminum and the 2S-72S bonded aluminum were stained dark gray on all surfaces. The major pitting attack on the bonded material was in the 72S aluminum; however, one pit did occur on the 2S aluminum side.

In view of these results and from the reported improvement in protection provided by the Hot-Dry Alodine process, it was decided to use this type of protection, but also, to continue the tests to determine, if possible, the life expectancy. In addition the tests with the remaining samples were continued for comparison.

Results of Tests of Longer Duration

Table IV gives the results of the continuation of tests as previously reported (Results of Initial Two-Week Tests) and all new tests which were started and run for various lengths of time. All tests were tested at least long enough to prove their worth.

Since the prime criterion for this application is pitting which might penetrate to the core and release the highly active fission products, it can readily be seen that no pretreatment offered satisfactory protection for long periods of time. A better idea of what this type of corrosion means may be obtained from the accompanying photographs, Figures 1 through 6.

TABLE IV

Results of Corrosion Tests on Various Aluminum Samples Exposed to Filtered Water at Room Temperature

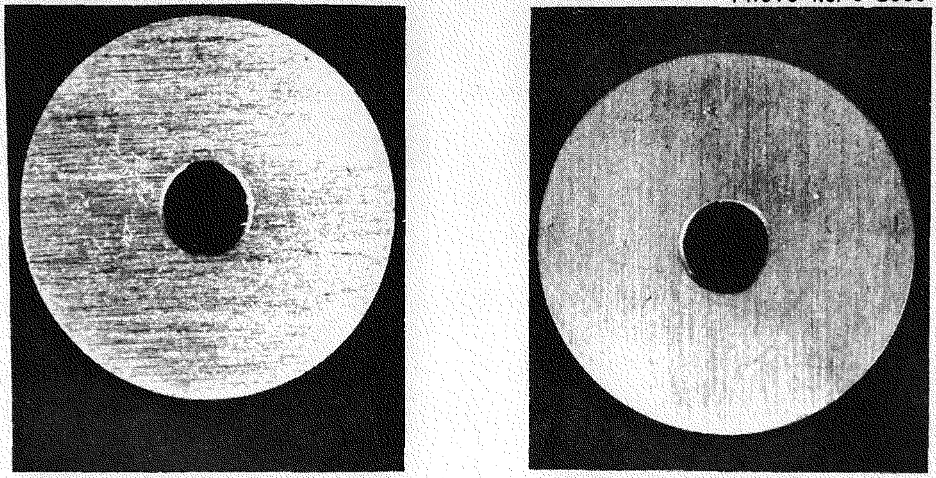
Sample	Weight Change (MDM)						Defilmed
	1 month	2 months	3 months	4 months	5 months	6 months	
2S Al	+ 8.8	+ 15.1	+ 12.4	+ 31.2	+ 26.4		- 32.0
2S Al	+ 7.3	+ 11.4	+ 13.0	+ 27.5	+ 18.3		- 27.8
2S-72S	+ 2.3	+ 23.4	+ 58.8	+ 91.9	+ 79.6		- 54.7
2S-72S	+ 5.1	+ 287.3	+ 37.4	+ 75.6	+ 69.0		- 53.7
2S Anodized	- 9.3	- 4.2	- 7.7	- 5.6	- 4.65	- 4.1	
2S Anodized	- 13.4	- 12.5	- 13.6	- 9.8	- 7.9	- 5.7	
2S (Na_2CrO_4)	+ 0.8	+ 2.2	+ 0.7	+ 0.0	+ 0.1	- 1.1	+ 0.3
2S (Na_2CrO_4)	+ 3.2	+ 3.4	+ 1.3	+ 0.7	+ 0.3	- 0.5	+ 0.2
2S (K_2CrO_4)	- 0.5	- 2.5					- 1.5
2S (K_2CrO_4)	- 2.4	- 2.9					- 2.5
2S Std. Alodized	- 3.0	- 0.9	- 1.1	+ 0.9	+ 2.9	- 6.6	
2S Std. Alodized	- 3.7	- 1.2	+ 0.3	+ 0.2	- 5.2	- 7.1	
Mock Assembly	+ 7.7	+ 5.6	+ 7.0				
Mock Assembly	+ 6.1	+ 5.0	+ 1.9				
2S Hot Dry	+ 0.5	- 1.5					
Alodized at X-10	+ 3.9	- 0.8					
2S (HNO_3)	+ 5.9						
2S (HNO_3)	+ 2.5						
2S-72S (HNO_3)	+ 2.0						
2S-72S (HNO_3)	+ 3.8						

TABLE IV (CONTINUED)

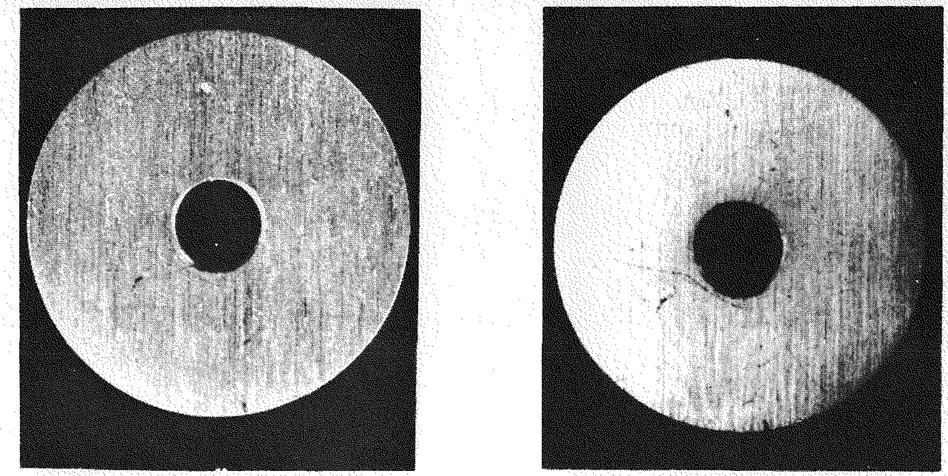
Results of Corrosion Tests on Various Aluminum Samples Exposed to Filtered Water at Room Temperature

	Number of Pits Per Sample							
Sample	1 month	2 months	3 months	4 months	5 months	6 months	Max. Pit Depth (mils)	Max. Pit Dia. (mils)
2S Al	1	8	12	12	12		46	40
2S Al	2	8	12	12	12		38	80
2S-72S	1	4	5	6	6		19	120
2S-72S	1	1	2	4	6		19	80
2S Anodized	0	0	0	0	3	16	1	20
2S Anodized	0	0	0	0	2	38	1	4
2S (Na_2CrO_4)	0	0	0	0	0	0	Nil	Nil
2S (Na_2CrO_4)	0	0	0	0	0	0	Nil	Nil
2S (K_2CrO_4)	0	0					Nil	Nil
2S (K_2CrO_4)	0	0					Nil	Nil
2S Std. Alodized	0	0	0	8	8	9	40	20
2S Std. Alodized	0	1	5	5	9	15	34	40
Mock Assembly			5				7	40
Mock Assembly			0				Neg	Neg
2S Hot Dry	1	5					7	40
Alodized at X-10	1	4					18	20
2S (HNO_3)	4							
2S (HNO_3)	6							
2S-72S (HNO_3)	Numerous							
2S-72S (HNO_3)	Tiny Pits							

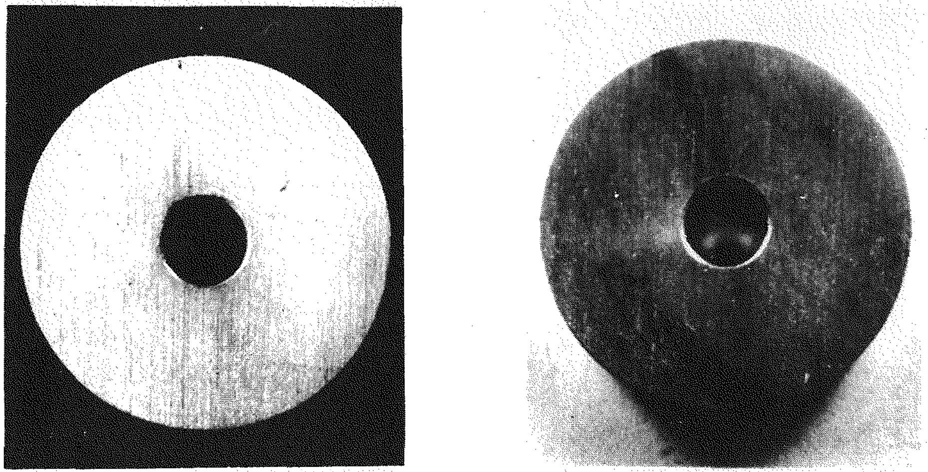
UNCLASSIFIED
PHOTO NO. 6-2500



I MONTH 2 MONTHS



3 MONTHS 4 MONTHS



5 MONTHS 6 MONTHS DEFILMED

FIGURE I. CORROSION OF UNPROTECTED 2 S ALUMINUM SAMPLES EXPOSED TO FILTERED WATER CONTAINING 60 ppm Na_2CrO_4 AT ROOM TEMPERATURE

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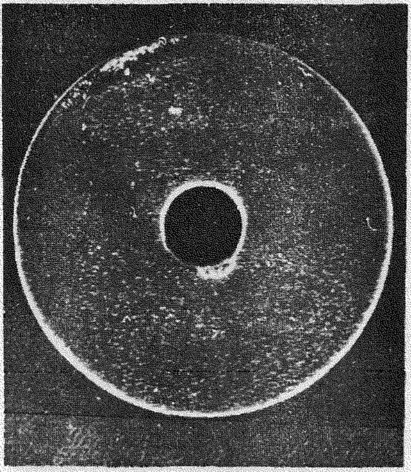
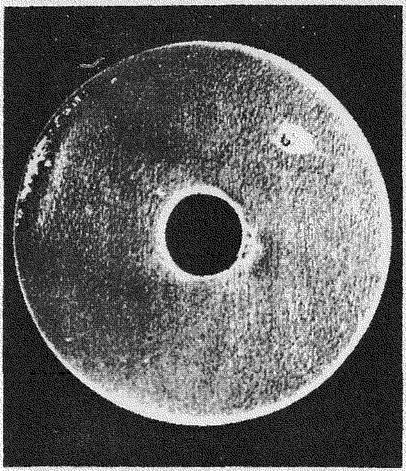
PHOTO NO. 6-2501



I MONTH



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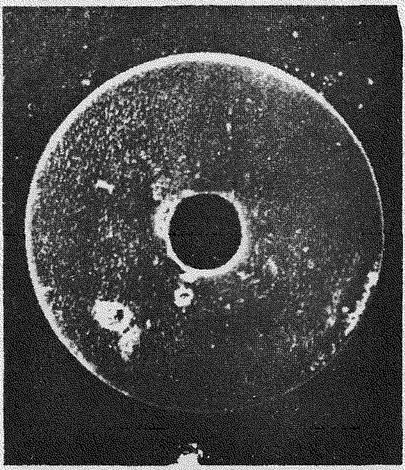

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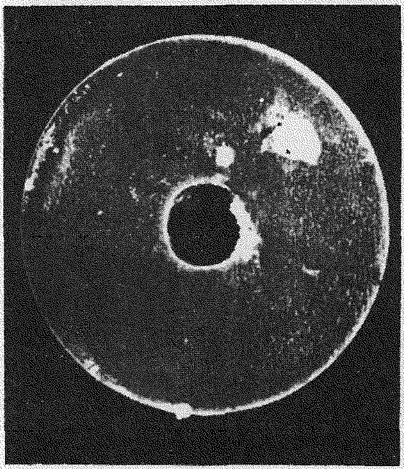
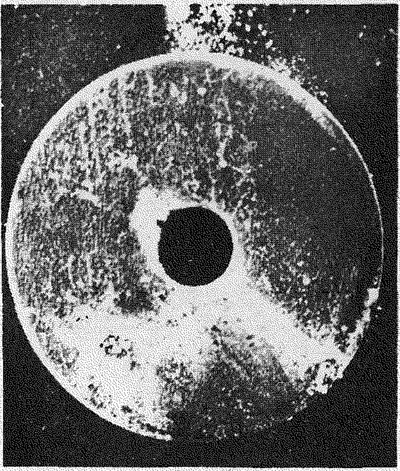
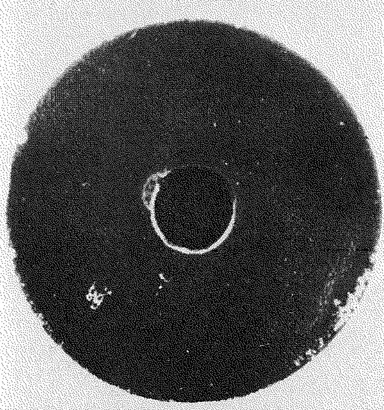


4 MONTHS



5 MONTHS



6 MONTHS
DEFILMED

FIGURE 2. CORROSION OF 2S ALUMINUM PRETREATED BY THE STANDARD ALODINE PROCESS AND EXPOSED TO FILTERED WATER AT ROOM TEMPERATURE

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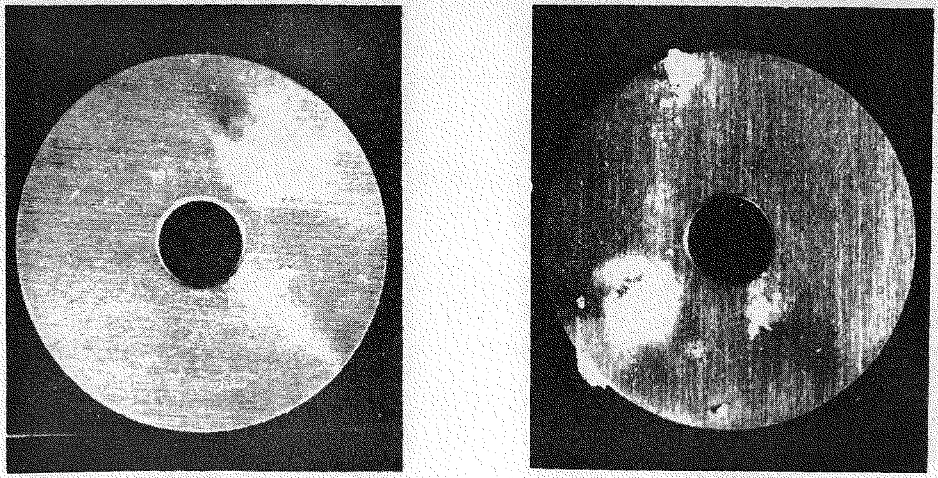
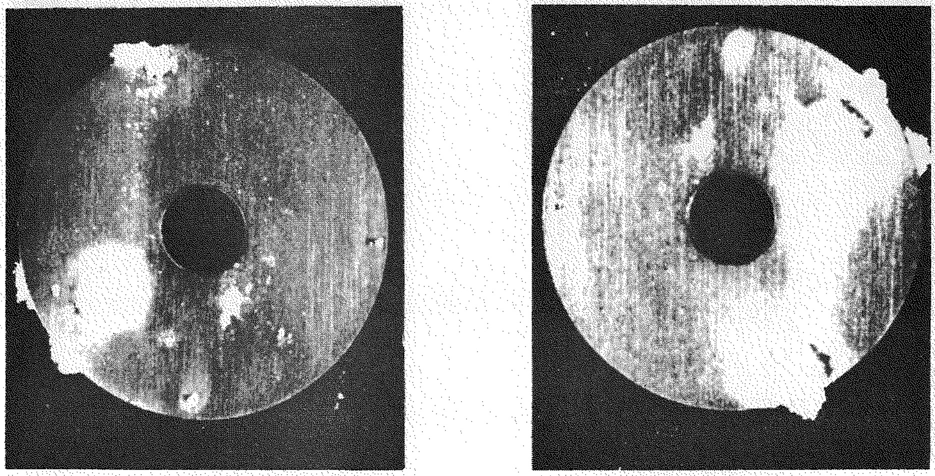
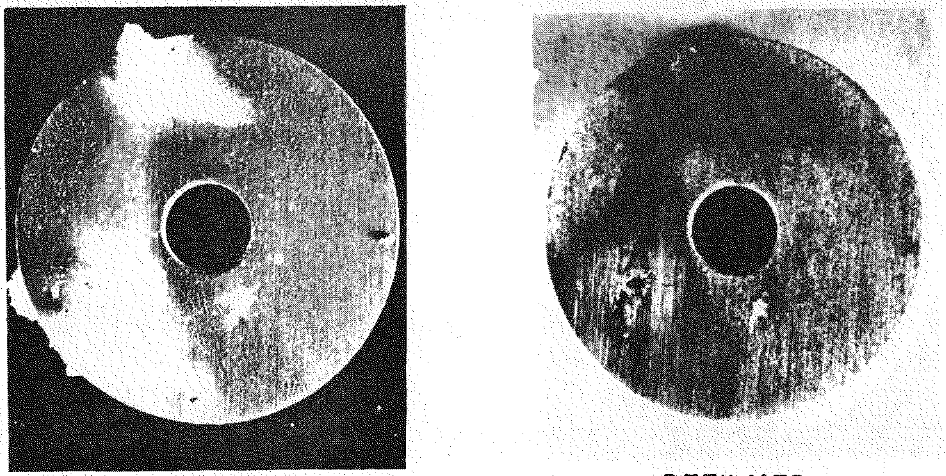


1 MONTH   2 MONTHS



3 MONTHS   4 MONTHS



5 MONTHS   DEFILMED

FIGURE 3. CORROSION OF UNPROTECTED 2S ALUMINUM SAMPLES EXPOSED TO FILTERED WATER AT ROOM TEMPERATURE

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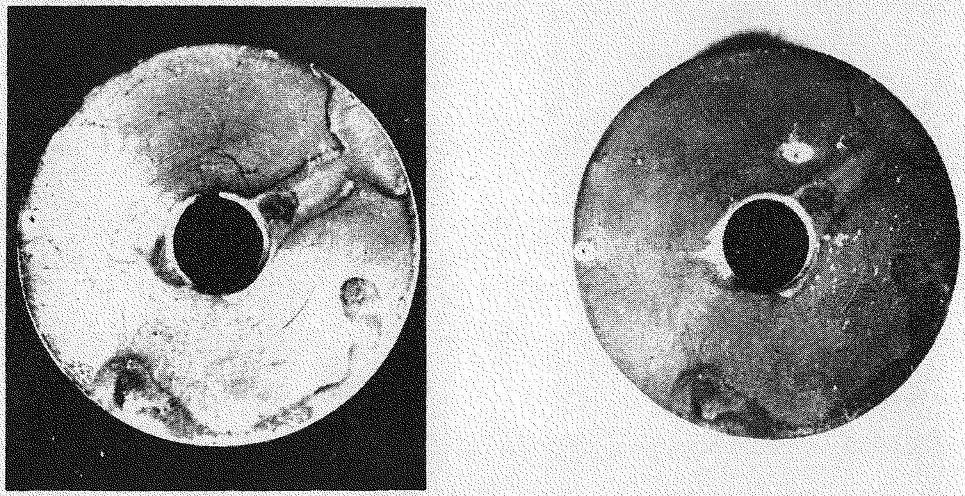
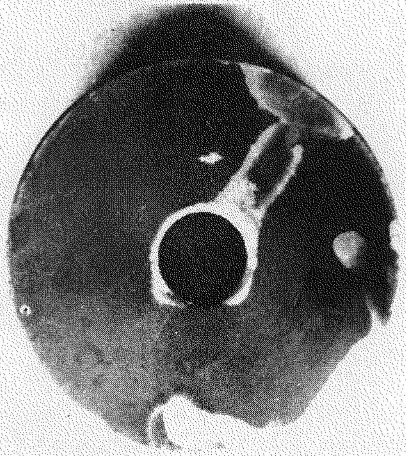


1 MONTH

2 MONTHS



DEFILMED

FIGURE 4. CORROSION OF 2S ALUMINUM PRETREATED BY THE HOT DRY ALODINE PROCESS AND EXPOSED TO FILTERED WATER AT ROOM TEMPERATURE

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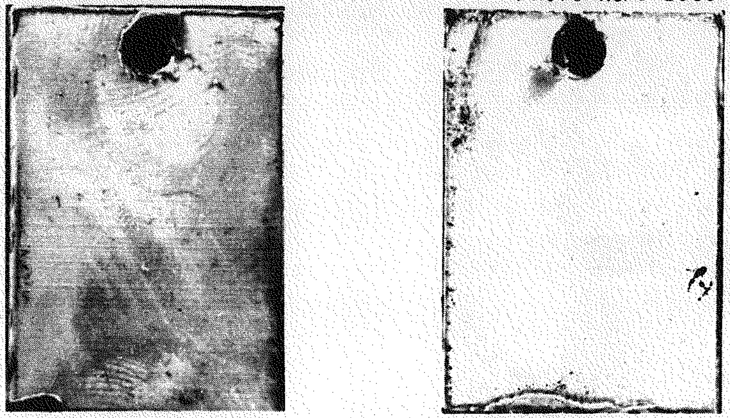
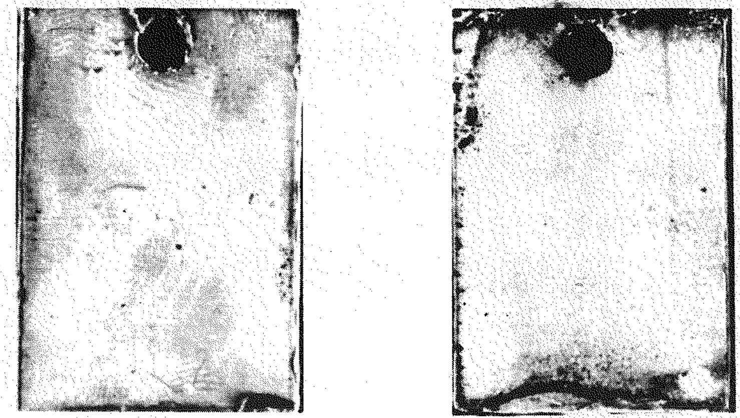
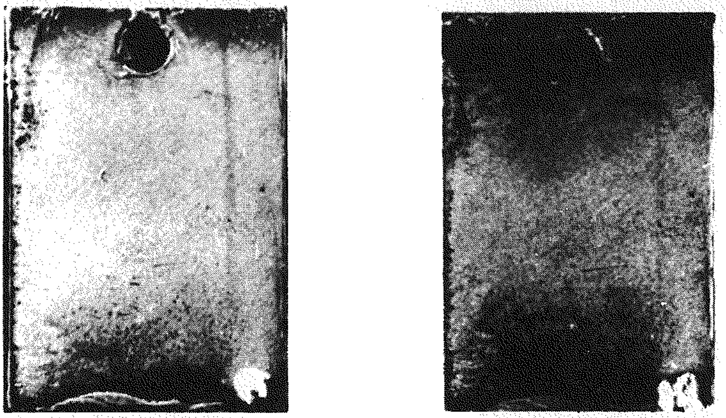


1 MONTH 2 MONTHS



3 MONTHS 4 MONTHS



5 MONTHS 6 MONTHS

FIGURE 5. CORROSION OF ANODIZED 2S ALUMINUM SAMPLES EXPOSED TO FILTERED WATER AT ROOM TEMPERATURE (EDGES WAXED)

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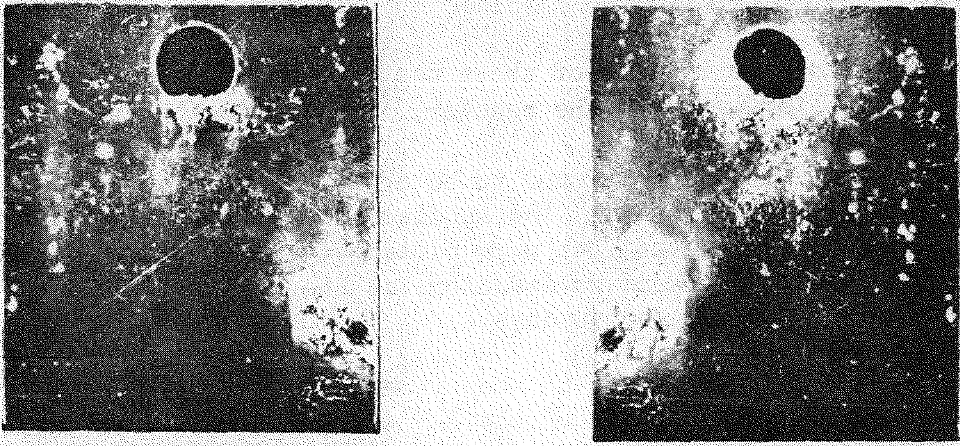
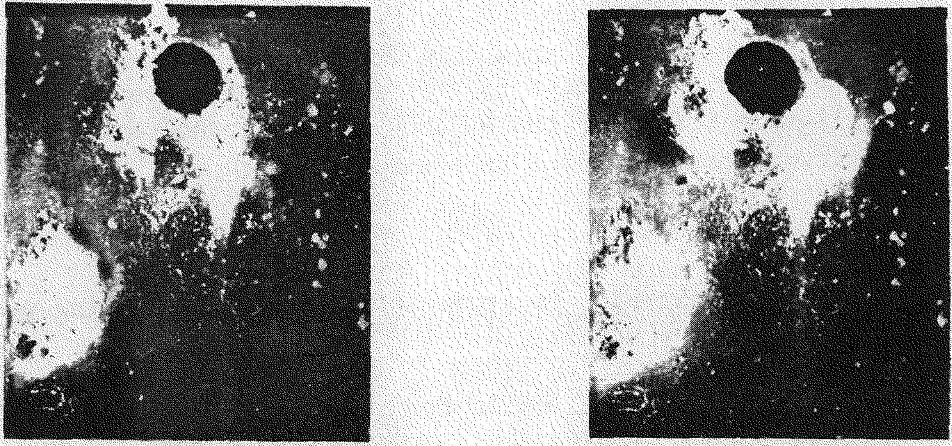
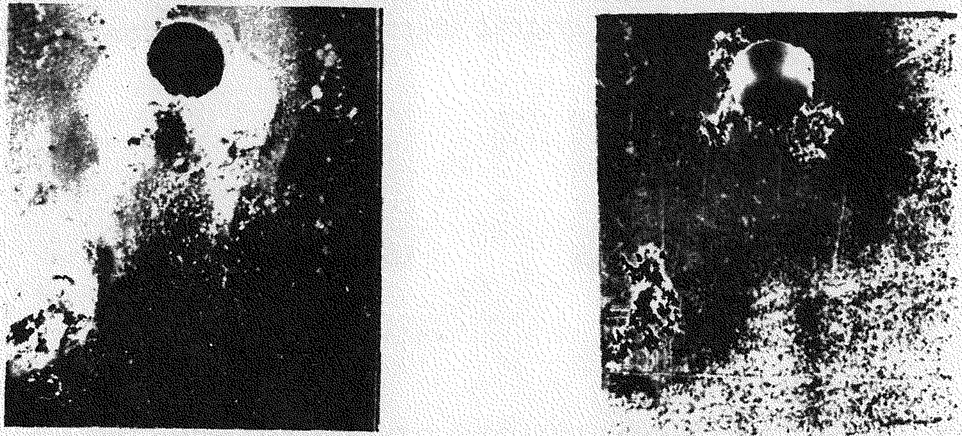
PHOTO NO. 6-2506



1 MONTH 2 MONTHS



3 MONTHS 4 MONTHS



5 MONTHS DEFILMED

FIGURE 6. CORROSION OF ANODIZED 2S ALUMINUM SAMPLES EXPOSED TO FILTERED WATER AT ROOM TEMPERATURE (EDGES WAXED)

Conclusions

On the basis of the results of these tests, it was decided to add 60 ppm Na_2CrO_4 to the cooling water in the reactor.

No pretreatment tested was found to be effective in preventing pit formation for more than four months. The anodized 2S aluminum samples showed no pits at the end of four months, but were pitted at the five month inspection. Alodized samples which appeared satisfactory at the end of two weeks, failed by pit formation between 60 and 90 days.

Although pit depth was shown to stop at the 2S-72S aluminum interface because of the sacrificial effect of the 72S aluminum, the accumulation of corrosion products was high. Such accumulation could effectively block the cooling channels, with subsequent formation of hot spots and consequent rapid corrosion.

Time and space did not permit investigation of lower concentrations of the sodium chromate inhibitor. Operation of the reactor for three months has shown that there is negligible effect of the Na_2CrO_4 on background activity, making consideration of lower concentrations unnecessary.

Acknowledgements

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