NRL REACTOR OPERATIONS REPORT, 1 JULY 1968 - 30 JUNE 1969

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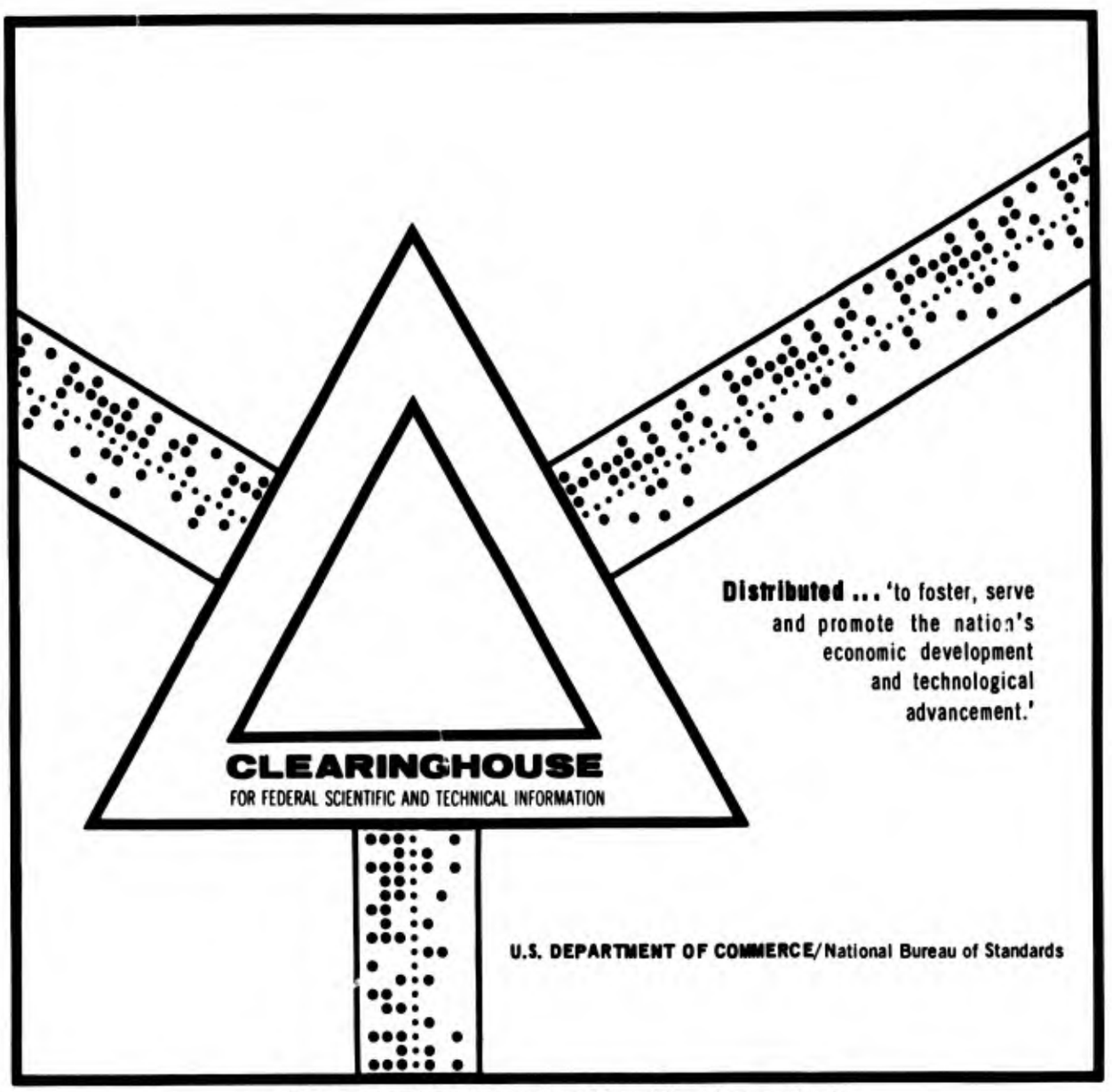
Washington, D. C.

September 1969



NRL Memorandum Report 2048

NRL Reactor Operations Report 1 July 1968 to 30 June 1969

E. H. BEBBS, K. W. MARLOW, R. H. VOGT

Reactors Branch
Nuclear Physics Division

September 1969

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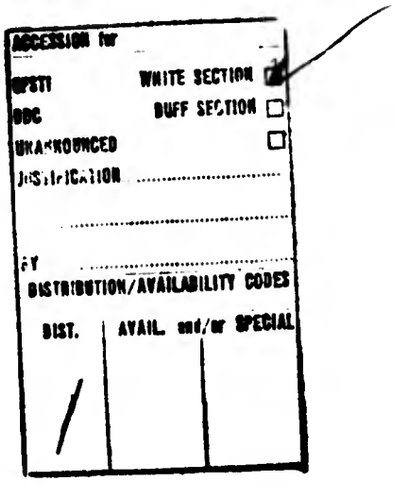
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ABSTRACT

This report provides a summary of reactor operations: operational data as well as pertinent information regarding maintenance, modifications, tests and services during the period 1 July 1968 to 30 June 1969. The current status of reactor utilization by in-house and outside users is discussed.

PROBLEM STATUS

This is an interim report; work on this problem is continuing.

AUTHORIZATION

NRL Problems 76H01-01
76H01-14
Project RR 002-06-41-5855
RR 002-06-41-5000

NRL Reactor Operations Report
1 July 1968 to 30 June 1969

I. INTRODUCTION

This is the fifth report written to cover operation and utilization of the NRL Research Reactor. The first three were informal reports emphasizing operations and covered the period 1 January 1966 to 31 December 1967. NRL Memo Report 1955, covering the period 1 January 1968 to 30 June 1968, increased the emphasis on utilization. The present report provides a summary of operations and utilization for the NRL Research Reactor for the period 1 July 1968 to 30 June 1969.

II. BRANCH ORGANIZATION

The organizational structure of the Reactors Branch during the reporting period is as shown below.

Reactors Branch Personnel
7650 K. W. Marlow, Head
B. L. Eckert, Secretary

7651 Radioactivity Section	7652 Operations
K. W. Marlow, Acting Head	E. H. Bebbs, Head
R. H. Vogt	G. L. McCroskey[2]
S. M. Lombard[1]	L. A. Harris
D. M. Shores	G. W. Dobbs
G. E. Holloway	N. J. Farmer

[1] Lt. Lombard, an Army Officer assigned to NRL, came on board in April 1969.

[2] Mr. McCroskey is the Reactor Facility Engineer, but is administratively in the Engineering Consultant Group (Code 7603).

III. REACTOR OPERATIONS

A. Operational Summary

During the period of this report the reactor has been operated exclusively in the partially graphite reflected mode, generally at its maximum licensed power of one megawatt.

The average operating time has been approximately 36 hours per week, with 33 hours at full power. Except for several protracted periods of shut down for maintenance, the normal operating schedule has been 8 hours per day, Monday through Thursday, and 4 hours on Friday.

A compilation of pertinent operational data for the reporting period is given in the table below.

TABLE 1

Operational Data for the Period
1 July 1968 - 30 June 1969, Inclusive

Total Hours of Operation	1595.50
Cumulative Energy Production	1396.350 MWH
Burn Up	74.007 g (U^{235}) 62.836 g (Net Total U)
Ending Inventory of SNM for Period Ending 30 June 1969*	9293.98 g (U^{235}) 10111.38 g (Total U)
Reactor Start-Up, Total No.	244
Normal Shutdown, Total No.	224
No.of Manual Scrams	1
No. of Unscheduled Scrams	19

*These values include approximately 5 kg of elemental uranium in spent fuel elements and fission chambers, with an average of about 90 wt percent U^{235}.

The unscheduled scrams noted in Table 1 were due to the following causes: Thirteen were caused by component failure or malfunction in control and safety circuits; two were induced by the bridge gamma-ray monitor when the trip level was not properly reset following the morning checkouts; two were caused by the drop of single safety rods when the drop level settings of their respective magnet current amplifiers were set slightly too low; one was a simulated period scram when the start-up fission chamber was being driven in during a reduction in power from 1 MW to 1 KW to allow beam port experimental changes; one was due to loss of coolant flow when the primary pump was inadvertently turned off during low power operation.

A contract between NRL and the United Nuclear Corporation (UNC) for the fabrication of thirty-five fuel elements was fulfilled with the delivery in two partial shipments of elements to NRL on 24 July and 24 October 1968. The new batch of fuel consisted of twenty-nine standard elements (nominal 140 g), and six of the control type containing approximately 70 grams each. The total contained uranium in the new fuel was 4802.47 g, 93.15 wt percent of which was U^{235}. Following an extensive inspection regime on each new fuel element the batch was accepted as having met the contract specifications in late November.

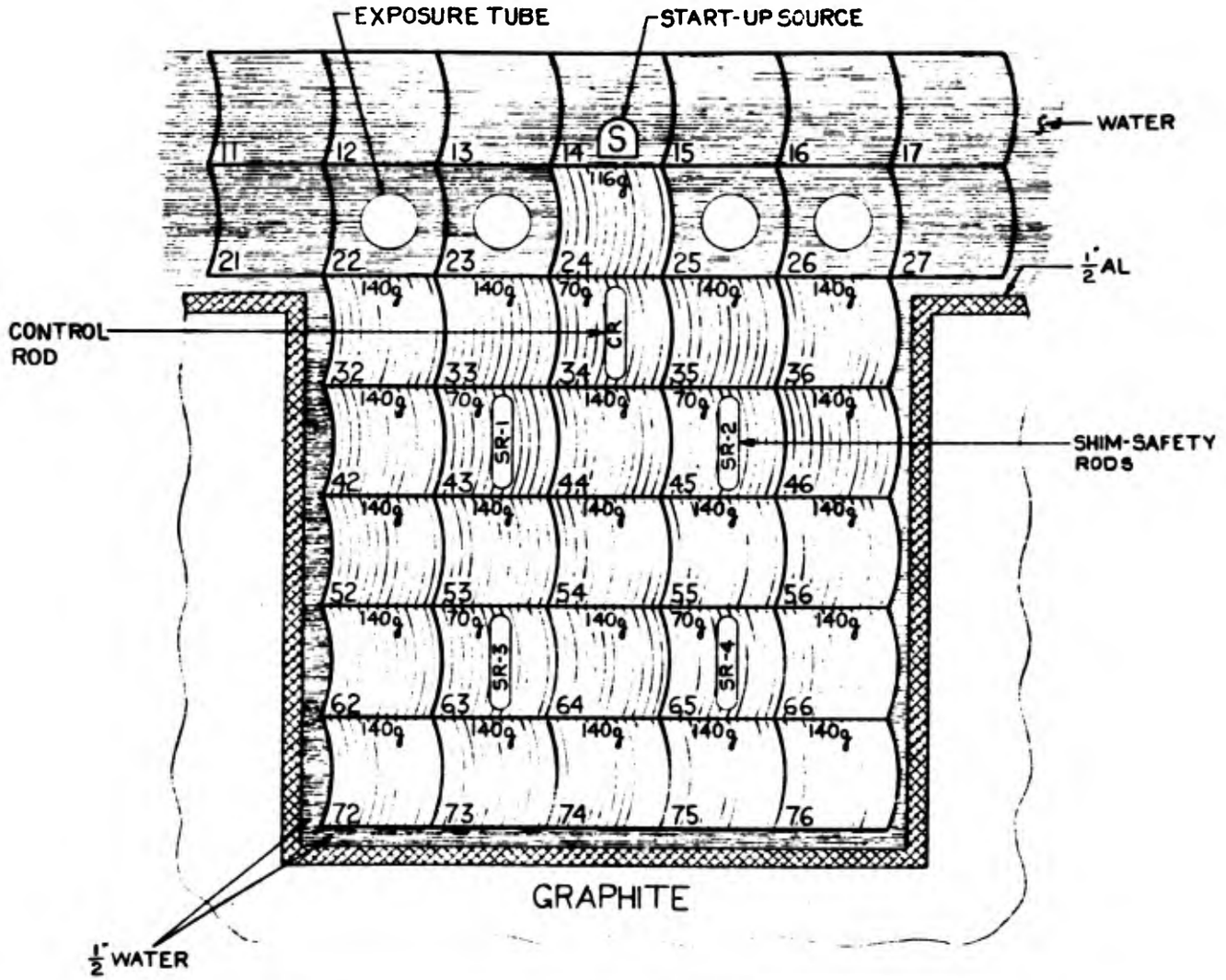
On 24 January 1969 the reactor core made up of Babcock and Wilcox (B&W) fuel elements was unloaded, preparatory to performing critical experiments using the new (UNC) fuel elements.

During the weekend of 25-26 January, critical experiments were performed, with criticality being achieved at 11:34 a.m. on the 26th. Reactivity measurements were made on a total of thirteen core configurations including one (loading No. 308), which had an associated reactivity (ρ_{ex}) of approximately 5% above that of a similar cold clean "just critical" configuration. The value of excess reactivity, 5%, is the maximum permissible under the current AEC license for the NRL Reactor.

Extensive reactivity measurements were made on the core selected for initial power operation (Core No. 309). These measurements included the determination of the individual and total worths of the exposure tubes in their normal positions in the fuel grid, as well as a detailed calibration of the shim-safety and control rods. Core loading No. 309, the core presently in use is shown in Fig 1. The negative reactivity worths of the individual exposure tubes are also given.

For the UNC core loadings the locations of safety rod No. 1A (SR-1A) and safety rod No. 3A (SR-3A) were reciprocally changed from those which they occupied in the previous B&W loadings, i.e., SR-1A, in grid position 43 in the B&W cores, was moved to grid position 63 for the UNC core; with SR-3A being moved from position 63 to position 43 for the UNC core. SR-2A and SR-4A were similarly interchanged at positions 45 and 65.

The calibration of the four safety rods to determine their differential and total worths showed good agreement with earlier calibrations



CORE REACTIVITY = 3.05 % (WITH NO EXPOSURE TUBES) | MASS U^{235} = 3263.10g

WORTH OF EXPOSURE TUBE (%)	-0.02	-0.04	-0.04	-0.02
GRID LOCATION	22	23	25	26

Fig. 1 Reactor Core Loading No. 309

of rods in the same grid locations and for similar core configurations. The total worth of each of the indicated safety rods in grid locations 43 and 45 was approximately 2.5% while for those in grid positions 63 and 65 (a region where the fuel is graphite reflected on three sides) the worth of each was approximately 3.0%. These measurements indicated that there has been no significant degradation of the worth of the rods due to burn-up of B^{10} in the included B_4C in the safety rods; i.e., the safety rods had not become "grey" to thermal neutrons.

B. Maintenance

During the period of this report the condition of the safety rods was checked on four occasions; twice by visual techniques, and twice by electronic determination of the individual rod drop times. The former method tested for any change in lateral dimensions of the rods by determining the clearance of each rod in precision go-no-go gauges. The exterior walls, welds, armatures, and shock absorbers were also inspected using an underwater telescope and mirror. There has been no indication of change in dimension of any rod nor in the appearance of exterior walls or welds. There has, however, been significant pitting and discoloration of the central areas of several of the safety rod armatures.

The drop time measurement is designed to reveal any evidence of swelling or binding of a safety rod, as it is dropped into its control fuel element from the rod's fully withdrawn position. Any binding would be reflected as a substantial increase in the

measured drop time. A tabulation of the two most recent drop time measurements is shown in the last two columns of Table 2. For sake of comparison, several representative measurements made over the past 2½ years are also included in Table 2. It should be noted that it is characteristic of newly installed magnets or new magnet-armature combinations, for the associated drop times to be slightly longer than normal. This is due to the inherently longer release times of the non-pitted and "crud"-free contacting surfaces of the new magnet armature combinations. This is illustrated in the measurements of 16 December 1966, just prior to which all new magnets and armatures had been installed.

The present set of shim-safety rods, although showing no significant degradation to date in shut-down capability, nor any apparent exterior physical damage, has been in service in the reactor since 1961. There is some evidence that with long time operation with this type of rod, even at the relatively low power of one megawatt, there is the possibility of embrittlement of the stainless steel shell, with risk of eventual fracture. With the foregoing possibility in mind, a new set of four stainless steel clad boron-carbide safety rods, identical to the present set, has been ordered. These rods are currently being fabricated and tested to the same specifications as the present rods by the Engineering Services Division. Delivery of the completed rods is expected early in September, 1969.

TABLE 2

Safety Rod Drop Times

The following rod drop times were obtained from photographs of a calibrated oscilloscope sweep triggered by the collapsing magnetic field when the magnet current is cut off. The time is measured from the beginning of the sweep to the first vertical deflection derived from an electro-mechanical transducer mounted on the control element hold-down mechanism.

Date	12-16-66	5-26-67	1-8-68	5-31-68	1-3-69	4-25-69
Safety Rod No.	Drop Time (m sec)	Drop Time (m sec)	Drop Time (m sec)	Drop Time (m sec)	Drop Time (m sec)	Drop Time (m sec)
1	425	420	420	420	425	420
2	460	430	440	440	475*	460
3	530	510	525	500	500	495
4	525	510	480	450	460	465

*A new magnet was installed on No. 2 safety rod drive on 14 November 1968.

The reactor was shut down from 28 October to 3 December 1968 for repair of a water leak in the east end of the reactor pool. The rate of loss of demineralized water from the pool, progressively becoming worse, had reached a value of greater than 1000 gallons per day just prior to the shut-down. Since the leak occurred in the floor area of the recess in the main shield where the reactor core is normally located, the materials surrounding the leak were quite radioactive. With five two-inch thick steel shielding plates covering most of the recess and a heavy steel weldment in the lower portion, local dose rates greater than 500 mrem/hr were encountered in the immediate area of the leak. By the rotation of personnel, including several volunteers from other NPD Branches, the old material around the leak was chipped away and the leak repaired with 3M(EC801) sealer compound.

In addition to repairing the leak, other maintenance projects were conducted during the shut-down period, these included: (a) repainting of the east end of the pool with Amercoat No. 23 Paint, (b) checking and cleaning all inpool piping, replacing gaskets as required, (c) checking the flapper valve ball switch and installing a new flapper valve trip line (d) coating of aluminum components with Iridite, (e) repairing of the check valve in the 8-inch emergency fill water line; and (f) removal of the underwater pool wall lights (a suspected source of traces of copper in the pool water) and covering their openings with stainless steel plates.

Other major maintenance projects completed during the reporting period were: the replacement of the ion exchange resins in the pool water mixed-bed demineralizer in November 1968; and the replacement of the materials in the pool water filter (graded sands and gravels) in February 1969.

A work request has been submitted to the Public Works Division to have the west end of the reactor pool repainted during the first quarter of FY1970. It has been proposed that one section of the west-end pool walls be coated with Plasite No. 7155 a relatively new type coating, and the other walls coated with Amercoat No. 23 paints, in order to test the relative merits of the two coatings. Plasite No. 7155, an epoxy-based coating has been tested extensively at ORNL where it was shown that the coating maintained its integrity in demineralized water at elevated temperatures, and in the presense of high radiation fields. Recently Babcock and Wilcox ran tests on Plasite 7155 at their Alliance, Ohio facilities in reactor containment vessels with excellent results. In addition, a coated test concrete block has been suspended in the NRL Reactor pool over a period of approximately two years with no apparent degradation in the coating.

In order to provide a proper basis for comparison of the two coatings, the same pre-painting preparation of the wall and floor surfaces will be made i.e., all walls and floor surfaces will be sand-blasted to remove all traces of the present coating. The proprietary procurement of the service of Industrial Coatings Inc. of Baltimore, Maryland has

been requested for the sandblasting and the rather specialized application of the Plasite coating. It has been requested that Public Works Division painters be allowed to apply the Amercoat No. 23 Paints.

C. Modifications to the Facility

A modification of the south tangential beam port shielding plug increased the size of the aperture of the through-hole, with a resultant increase in the fluence of neutrons at the beam port exit. Having originally a constant $\frac{1}{2}$ inch diameter, the through hole in the modified plug sections varies from $2\frac{1}{4}$ inches in diameter close to the core to a $1\frac{1}{8}$ inch diameter in the outer sections of the plug. A drawing of the assembled plug is shown in Fig 2.

A vertical exposure facility in the reactor thermal column with accesss from the mezzanine level has been described previously (NRL Report 6704). Recent modifications to the facility to improve its utility and safety consist of: (a) the addition of a 3 in. o.d., 80 in. long aluminum liner; (b) additional shielding in the form of a massive masonite and lead plug, 6 in. of borated epoxy resin, and a 0.032 in. thick sheet of cadmium and; (c) two rotatable shutters, one of lead and the other of lithiated epoxy resin. The new assembly is shown in Fig 3.

A permanent installation was completed in October 1968 to replace the temporary apparatus (fire hose and gasoline powered pump), which had been used earlier to discharge reactor effluent

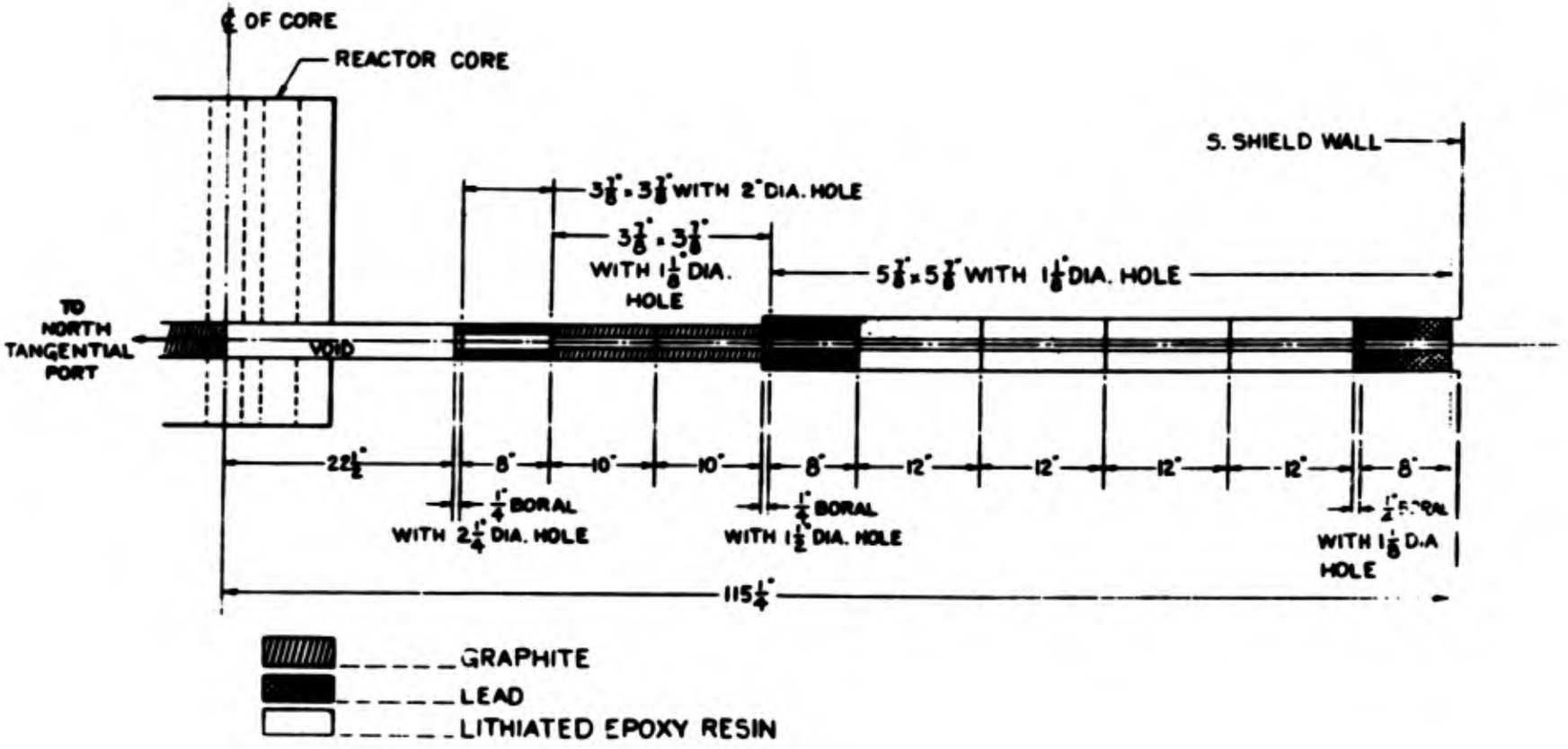


Fig. 3 Tangential Beam Port Plug

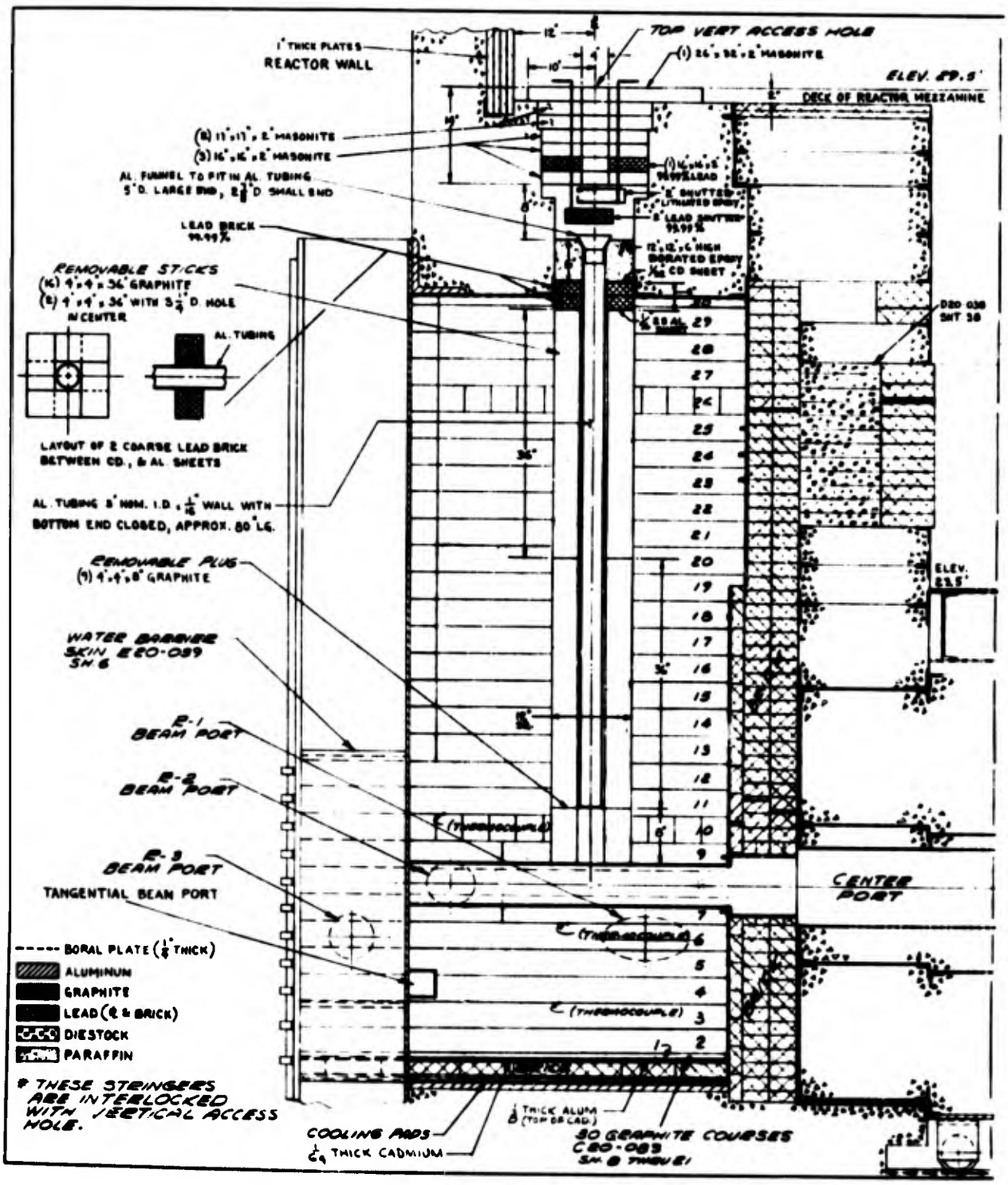


Fig. 3 Cross Section of Reactor Shield, Showing Thermal Column and Exposure Tube

from the hold-up tank to a sanitary sewer trunk line. In this connection, more than 400 feet of 8 in. underground piping was installed between the hold-up tank adjacent to Building 43 and sanitary sewer manhole No. 11 located at the rear of Building 12. Existing electric sump pumps and check valves associated with the original hold-up tank installation are used in the new discharge system. Reactor facility liquid effluent, until early in 1968 was discharged from the hold-up tank directly into the river. The background for the discontinuance of this practice was discussed in the previous Reactors Branch Operations Report (NRL Memo Report 1955).

The initiation of a work request for the installation of three radiochemical fume hoods (including one perchloric acid hood) in Room 117 of Building 71R was discussed in the previous Operations Report. The radiochemistry laboratory, upon its completion, will be occupied by U. S. Geological Survey personnel. The status of this project at the end of the reporting period is as follows: (a) two 47 in. wide radiochemical fume hoods and accessories have been obtained from NRL surplus, and one 71 in. wide perchloric acid hood has been purchased; (b) the necessary engineering drawings have been made and approved; (c) the exhaust blowers, filters, a fume scrubber for the perchloric acid ductwork, and other required materials have been ordered; (d) a supplementary work request for the installation of a 5 ft. partition and a chemical sink in Room 117 has been

made, with the provision that this project be coordinated with the work of the original request. Actual work on the installation is expected to commence in late August or early September 1969.

As reported earlier, delivery has been made on General Electric Co. solid state control and safety instrumentation which will eventually replace the present start-up, log N, linear, and two safety level channels. Also, a solid state replacement for the present servo-amplifier has recently been purchased.

The interface electronic circuitry between the G. E. solid state instrumentation and the portion of the control and safety systems which will remain after the conversion, has been designed and, for the most part purchased or fabricated. Testing of the G. E. equipment and the interface will begin in the near future with the use of spare detectors, and simulated signals where necessary. Following the preliminary testing phase, the new channels will be tested in an arrangement in which a new solid state channel will parallel its corresponding existing channel without disturbing the present system.

The new instrumentation will not be incorporated into our control and safety system until prior approval has been obtained from the Division of Reactor Licensing of the Atomic Energy Commission.

D. Tests

In response to a request by the Atomic Energy Commission, a series of tests have been performed to determine the efficiency of the reactor facility emergency air clean-up system for particulate removal

and for removal of iodine vapor. In tests prior to the reporting period the efficiency of the air filtration system for particulate removal was measured by members of the Health Physics Staff. In their tests an aerosol of dioctyl phthalate (DOP) was generated upstream of the filter system. The concentrations of the aerosol upstream and downstream of the filter were measured by means of a forward light scattering photometer. With an average particle size of 0.7u, the lower filter unit transmitted 0.4% and the upper unit transmitted 0.01%. More refined tests are planned in the near future.

During the reporting period members of the Reactors Branch made measurements to determine the efficiency for iodine removal of the system. In the tests described above the effectiveness of the scrubber unit's prefilters and absolute filters (and to some degree the leakage around the filters) were measured. The iodine removal measurements, on the other hand, tested primarily the effectiveness of the trays of activated charcoal in the scrubber units for removing iodine vapor from the air stream in the clean-up system.

In the test, iodine vapor obtained by heating iodine crystals was introduced into an extension of the intake duct of a scrubber unit while that unit was in operation. Probes in the ductwork upstream and downstream of the filters enabled simultaneous sampling of the air at these points. These probes were connected to air samplers adjusted to the same measured air flow rate (~25 cu. ft per min).

To balance out any differences in sampling rates the air samplers were interchanged for successive measurements. Iodine samples from the air stream were collected on activated charcoal filters, the contents and geometry of which were maintained constant.

The amount of iodine collected in the charcoal was determined by neutron activation analysis utilizing the 441-keV gamma ray associated with the 25-minute beta decay of I^{128}. The intensity of the 441-keV gamma ray photopeak was measured using a $30cm^3$ Ge(Li) detector, a biased amplifier and a 512-channel pulse-height analyzer. The fraction of iodine transmitted through the system was computed as the ratio of the measured amounts of iodine collected simultaneously at the input and output of the scrubber unit. The result of six measurements showed that the fraction of iodine transmitted through either filter is no greater than 2×10^{-3}; very much better than is required by our license. A report on the Iodine Filter Tests is in the process of publication.

New Reactor Facility Emergency Procedures were issued on 24 December 1968 to replace those of 23 February 1960. The most important change was the inclusion of practice evacuation drills. The first drill under these procedures was held on 4 March 1969. Although each person involved performed his duties in accordance with existing procedures, the critique following the drill brought out the desirability of providing the reactor operator with explicit instructions for

checking certain exterior doors after leaving the building. Such changes were made, and the new emergency procedures were issued on 10 June 1969.

IV. REACTOR UTILIZATION

As noted in part III above, the reactor was shut down for maintenance for more than a month during this reporting period. This, of course, affected the total utilization of the reactor. The rate of use during operating periods continued at about the same level or slightly higher. Since the principal use of the reactor is for activation of samples, it is of interest to provide an overall summary of irradiations for the past year. This information is given in Table 3. To compare these data with previous years, the total samples and total irradiations for each of several recent years are given in Table 4. Besides the one month shut-down referred to above, a reduction in the NOS work and the transfer of NBS work to their own reactor account for a reduction in the number of irradiations. An increase in use by others have partially compensated for this reduction, however. It should also be pointed out that the number of irradiations is tending to approach a saturation point earlier than the number of samples. This results from a tendency for researchers to pack more samples into one container when they are faced with scheduling problems. An increase in available neutron exposure will be needed in the near future if the present trends continue. This could be obtained by operating for longer hours, an improvement in the exposure facilities and/or by an increase in power. An increase in power and the concomitant increase in flux presently appears

TABLE 3

Irradiations Performed 1 July 1968 to 30 June 1969,
Tabulated According to User Organization

Agency or Division	No. of Samples	No. of Irradiations
NPD	261	144
CHEM	33	12
OSD	300	109
SSD	104	28
ELEC	6	1
NBS	1110	191
NOS	246	50
IRS and POD	1683	244
FBI	1324	324
GS	5461	278
FDA	951	367
TOTAL	11479	1748

TABLE 4

Irradiations Performed 1 July 1964 to 30 June 1969,
Tabulated According to Fiscal Year

Fiscal Year	No. of Samples	No. of Irradiations
1965	2853	1624
1966	3885	1228
1967	7539	1912
1968	11938	2186
1969	11478	1748

to be the most attractive alternative and an investigation into the feasability of a power increase has been undertaken. Brief summaries of some of the work being performed may be found below and in previous reports.

A. In-House Programs

1. Nuclear Spectroscopy (K. W. Marlow and R. H. Vogt, Radioactivity Section, Reactors Branch)

In support of the main work of the section, the $4\pi\gamma$ ion-chamber was calibrated using sources which were in turn calibrated at the National Bureau of Standards. Along with other standard sources also obtained from NBS, it is then possible to determine the absolute efficiency as a function of energy for the two medium-size Ge(Li) detectors on hand. Much of this work has been done although the job is not yet complete. The efficiency is being determined for three different source-to-detector distances. Using some commercially-available parts, a Si(Li) detector system has been assembled for observing low-energy γ-ray transitions, x-rays and fluorescent x-rays. The resolution (full width at half-maximum) is 330 eV for the 6.4 keV Fe x-rays.

The read-search processor of the two-parameter pulse-height analyzer was modified to allow digital gate selection of one-channel width rather than multiples of eight channels.

During the sabbatical of K. W. Marlow at the Instituut voor Kernphysisch Onderzoek in Amsterdam, he worked on the decay of ^{73}Se. The write-up and analysis of data was completed after return to NRL and the article has been sent to the publisher.

2. Neutron Activation Analysis (K. W. Marlow, S. M. Lombard, and R. H. Vogt, Radioactivity Section, Reactors Branch).

Preliminary studies have begun on the study of trace elements in human hair. Experience is being gained on the effects of various washing procedures and radiation damage. It is planned to try to study trace elements as a function of distance from the root of the hair in order to see if the distributions of these elements would lead to an improved identification technique.

Tests were made for the X-Ray Optics Branch to determine the extent of neutron activation of Pb to which Na, Ca and Li had been added for hardening. As expected, Na provided most of the resultant radioactivity and the activation analysis result for Na content was in fair agreement with the amount introduced. This indicates that the Li was not effective in preferentially absorbing neutrons. Dr. C. V. Strain analyzed the data for this experiment.

In response to a request by Mr. T. A. Kovacina of the Chemistry Division, Co and Cr determinations were made for an organic material containing the Co and Cr as minor constituents (of the order of 1%). The determinations were very simple and for Co required only a few minutes, while the Cr determinations required a few hours due to the longer irradiation and counting times required. As in most cases, no chemical separations were required.

Because of a possible geological interest in the presence of Fe in various clays, a determination of Fe in zeolite was made to test the method. A concentration of 0.35% (by weight) was found. The method was simple, non-destructive, and quite sensitive since there were no important matrix interferences after a reasonable "cooling" time.

3. Neutron Radiography (R. H. Vogt and K. W. Marlow, Radioactivity Section, Reactors Branch).

Preliminary experiments in neutron radiographic techniques have been performed. Using a neutron beam from the thermal column and indium transfer foils, a resolution of approximately 0.02 inch was obtained for samples close to the foil. In order to improve the "depth of field", however, neutron collimators will be required. Materials for such collimators have been ordered. Neutron radiography is especially useful when the materials of interest include hydrogenous or other low-Z elements.

4. Capture of Polarized Neutrons (A. Stolovy, Reactors Branch and Linac Branch).

Using the L-1 beam port, measurements have been made of the circular polarization of the gamma rays following capture of thermal polarized neutrons by a Ni target. The transmission of the 9.00 MeV and 8.53 MeV gamma rays through magnetized iron was observed with NaI(Tl) detectors and two 512-channel analyzers. We have measured the percent change in the photopeak intensities upon

reversing the neutron beam polarization. Effects as large as 4% (about 3/4 of the theoretically-possible effect) have been observed with good counting statistics. As expected, data taken with unpolarized neutrons did not yield real effects. From these data, we conclude that the spin and parity of the ground state of Ni^{59} is $3/2^-$, and for the first excited state at 470 keV, it is $1/2^-$, in agreement with previous tentative assignments. Experiments are now in progress with a gold target.

5. Neutron Beam Measurements (R. H. Vogt and S. M. Lombard, Radioactivity Section, Reactors Branch).

It is of interest to know the thermal neutron current available for neutron beam experiments. Two available beam ports which are used for neutron radiography and neutron capture γ-ray experiments were particularly of interest. One such beam is the 2.8 cm diameter tangential hole and the other is the beam emerging from the exposure tube in the thermal column. Measurements were made with gold foils (49 mg cm^{-2} thickness) with and without 0.050-inch Cd covers. The gold activity was counted with an absolutely calibrated Ge(Li) detector. A thermal neutron absorption cross section of 98.8 barns was used to determine the neutron beam currents. The cadmium-ratio(R_{Cd}) is the ratio of the activities of bare and cadmium-covered gold foils. The results appear in Table 5.

TABLE 5

Neutron Currents and Cadmium Ratios

Facility	Diameter (cm)	Neutron Current ($n\ cm^{-2}\ sec^{-1}$)	R_{Cd}
Tangential Hole	2.5	4×10^7	3
Thermal Column	7.5	3×10^7	40
Thermal Column	1.0	1.3×10^7	45

6. Neutron Flux and Cadmium Ratio in the Thermal Column (R. H. Vogt, Radioactivity Section, Reactors Branch).

The modified thermal column irradiation facility, referred to in Section III. C and the paragraph above (and shown in Fig. 3), was investigated to determine the thermal neutron flux and cadmium ratio as a function of distance along the exposure tube. Thin gold films (0.49μ thick) evaporated on plastic films were used to avoid self-shielding effects. The results are shown in Fig. 4.

7. Chemical Composition of the Ancient Sea (C. M. Gordon and R. E. Larson, Nuclear Oceanography Branch, Ocean Science Division).

This work, which is a continuing project, includes activation of barnacle shell deposits of different ages to determine various trace element concentrations. One publication resulted from this work during the reporting period (see Part V) and others are in preparation.

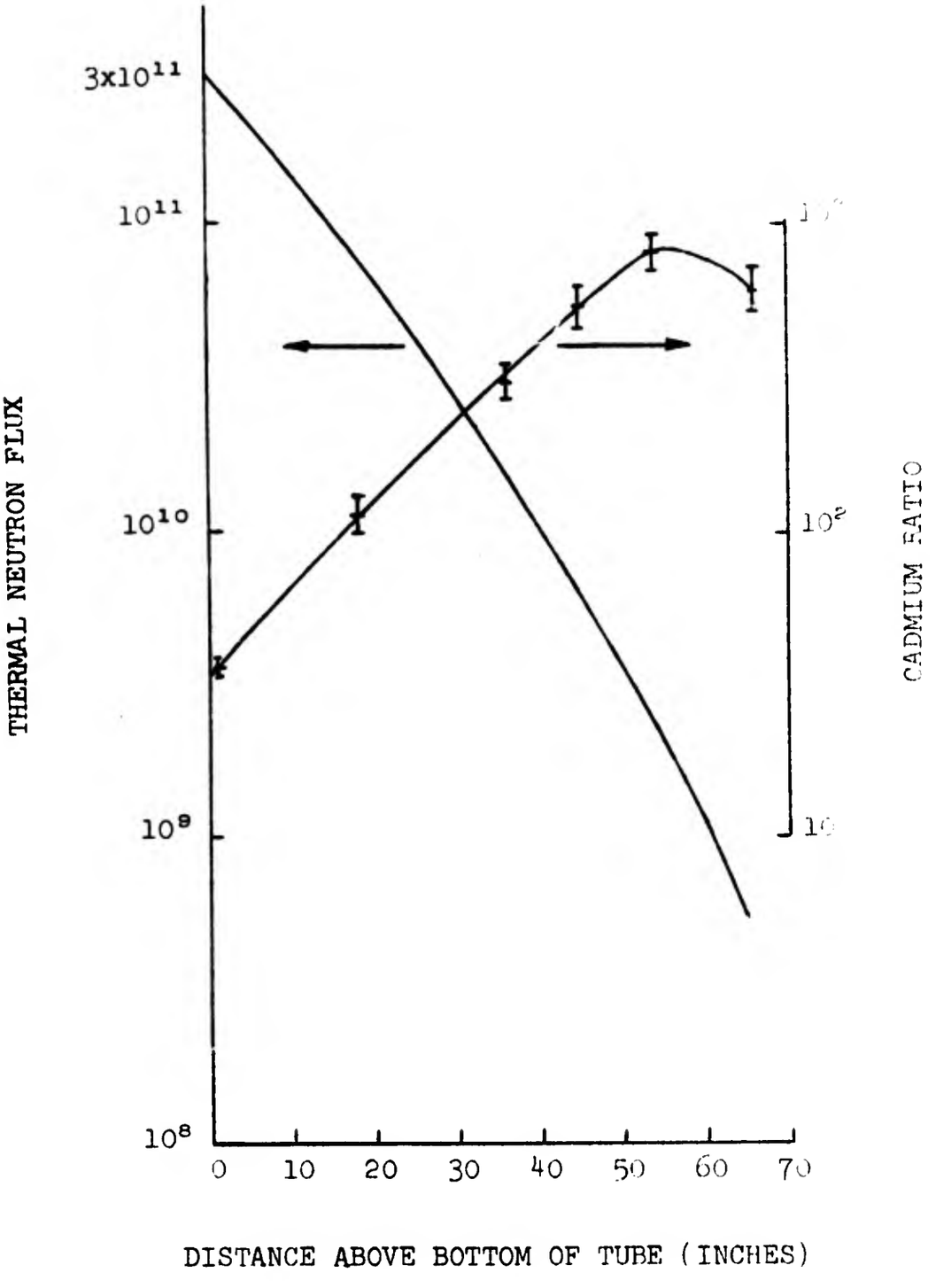


Fig. 4 Thermal Neutron Flux and Cadmium Ratio in the Thermal Column Exposure Tube

8. Implantation and Detection of Low-Energy Argon Ions in Silicon Crystals (J. Comas and C. Carosella, Solid State Division).

A long-range study of the sputtering of argon atoms and their implantation into silicon crystals has continued. One publication during this reporting period has resulted (see Part V).

B. Outside Agencies

The forensic work of the Federal Bureau of Investigation, the Internal Revenue Service and the Post Office Department has continued to increase at a steady rate. Since this is primarily case work, it seldom leads to publications although the IRS researchers occasionally publish their work (see two papers by Hoffman, et al. in Part V of this report). In addition to the applications mentioned in the previous report (NRL Memo Report 1955), gunshot residues are now routinely determined by neutron activation analysis.

The Food and Drug Administration research program, described in the previous report, is continuing. Aside from one paper by Lambert and Simpson listed in Part V of this report, most of the results of their work is in the form of internal reports.

The Geological Survey has the most active research program of all of our users, and their work is frequently published in the open literature. Their research program, also described in the previous progress report, has led to eleven publications during this reporting period.

The Naval Ordnance Station has made preliminary measurements to determine trace constituents in a liquid monopropellant called Otto Fuel.

V. BIBLIOGRAPHY

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13. ABSTRACT

This report provides a summary of reactor operations: operational data as well as pertinent information regarding maintenance, modifications, tests and services during the period 1 July 1968 to 30 June 1969. The current status of reactor utilization by in-house and outside users is discussed.

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