MARTIN MARIETTA ENERGY SYSTEMS LIBRARIES
3 4456 0050924 2

CRL

CENTRAL RESEARCH LIBRARY
DOCUMENT COLLECTION

ORNL-2725
Special

Cy. 1

SAFEGUARD REPORT ON THE PROPOSED METHOD OF ANNEALING GRAPHITE IN THE X-10 REACTOR

L. E. Stanford

CENTRAL RESEARCH LIBRARY
DOCUMENT COLLECTION
LIBRARY LOAN COPY
DO NOT TRANSFER TO ANOTHER PERSON
If you wish someone else to see this document, send in name with document and the library will arrange a loan.

OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

ORNL-2725
Special

Copy No. 1

Contract No. W-7405-eng-26

OPERATIONS DIVISION

SAFEGUARD REPORT ON THE PROPOSED METHOD OF ANNEALING GRAPHITE IN THE X-10 REACTOR

L. E. Stanford

DATE ISSUED

APR 20 1959

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U. S. ATOMIC ENERGY COMMISSION

MARTIN MARIETTA ENERGY SYSTEMS LIBRARIES
3 4456 0050924 2

TABLE OF CONTENTS

		Page No.
	Summary	iv
	Introduction	1
	Method of Annealing	3
	Future Annealing	4
	Description of Reactor and Coolant System	4
	Brief History of the Oak Ridge Graphite Reactor	5
	Changes in the Reactor Coolant System	7
	Special Instruments	8
	Matters Requiring Special Attention	12
	Limitations	16
	Hazards Evaluation	19
	Procedures	30
App. I	Evaluation of the ORNL Graphite Reactor Stored Energy	49
App. IA	Accumulation and Control of Graphite Stored Energy in the Oak Ridge Graphite Reactor	81
App. II	Research Program and Operating Experience on ORNL Reactors	87
App. III	Investigation of Methods of Obtaining Graphite Annealing Temperatures in the OGR	117
App. IV	Control and Safety Mechanisms	121
App. V	Oxidation Rates of OGR Graphite	133
App. Vi	Summary of Instruments	135

THE ORNL GRAPHITE REACTOR

STORED ENERGY AND ANNEALING STATUS

Summary

The ORNL Graphite Reactor moderator has undergone approximately 16 years of almost continuous irradiation. Throughout this time stored energy has accumulated at a slow rate to the present maximum value of about 35 cal/gm releasable to 250°C. A small portion of the moderator (approximately 4%) contains stored energy which under adiabatic conditions may be released spontaneously (at approximately 165°C) to produce a maximum temperature of 270°C. Careful analysis has shown that the present condition is not hazardous; however, it appears wise at this time to initiate some corrective action (thermal annealing) to prevent the continued buildup of stored energy to a dangerously high value.

Several methods of obtaining effective annealing in the OGR were investigated. The proposed method was selected upon the basis of convenience, over-all safety, effectiveness, and cost. The proposed method involves the alteration of the present coolant flow system to permit reversal of air flow through the fuel channels. This will result in a reversed temperature distribution wherein the maximum graphite temperature will occur in the normally cold, maximum-stored-energy region of the moderator. Such an arrangement permits an annealing operation to be performed under conditions very similar to those of the normal safe operation.

The proposed procedure requires a slow heatup of the moderator under full reversed, air-flow conditions. This can be accomplished by slowly raising the reactor nuclear power level until the desired graphite temperature

is attained. During the initial stages of the operation, the stored energy will be reduced to a sufficiently low value that spontaneous energy release is no longer possible. This can be accomplished at temperatures (less than 140°C) well below the minimum (165°C) required for spontaneous energy release. Subsequent higher temperature may then be employed to further reduce the stored energy. Recurrence of stored energy which may be releases spontaneously can be prevented by periodic annealing at rather infrequent intervals.

THE ORNL GRAPHITE REACTOR

STORED ENERGY AND ANNEALING STATUS

Introduction

The accumulation of radiation effects in the OGR moderator is a sufficiently slow process that controlling measures, such as thermal annealing, have not previously been necessary. The present stored energy and growth condition is the result of 16 years of varied operating history, and even now the total growth is small, the maximum vertical displacement being about 1/4 in. (Fig. 1, end of this section). The stored energy, however, is of somewhat more concern, but the condition is not a dangerous one.

A detailed investigation of the OGR stored energy condition has been made and the results are presented in Appendix I ("Evaluation of the ORNL Graphite Reactor Stored Energy," M. C. Wittels). It was found that the maximum quantity of stored energy releasable from OGR graphite by heating to 250°C is about 35 cal/gm. For heating rates in excess of 2°C/min, a plot of the stored energy release as a function of temperature for the most damaged OGR graphite results in a curve which exceeds the specific heat curve for virgin graphite in the temperature range from 165 to about 230°C. The energy represented by the area between these two curves will sometimes be referred to as "stored energy above the graphite specific heat curve." As this area is a function of the heating rate for a given sample, the maximum value is implied. Under adiabatic conditions, the stored energy release in this temperature range can occur spontaneously and is sufficient to produce a graphite temperature rise of approximately 100°C in the worst case (Figs. 4.17 and 4.18 of Appendix I). About 4% of the moderator volume (about 26 tons of graphite) contains some stored energy which can be released

in this way. However, under nonideal conditions, such as those in the reactor where there is heat transfer to the coolant stream and conduction to surrounding graphite, temperature rises due to a spontaneous energy release would be considerably less than the laboratory values obtained under adiabatic conditions.

Stored energy in excess of the graphite specific heat curve can be removed from the reactor at present by employing relatively low temperatures; subsequent higher temperature annealing can then be carried out with no possibility of a temperature excursion due to the spontaneous release of such energy. In this way, the continued buildup of stored energy to dangerously high levels can be precluded; and the continued safe operating condition of the reactor can be assured. For these reasons an annealing operation is planned for the near future, and a heatup procedure using a reversed air flow for obtaining the graphite annealing temperatures in the desired places appears very suitable.

The quantity and distribution of stored energy in graphite is a consequence of the combined effects of the fast neutron and temperature history. The OGR fuel loading pattern is that of a modified cube, as shown in Fig. 2, end of this section, which gives rise to a radially and axially symmetrical flux distribution. The coolant enters the reactor inlet-air manifold at the lower north corner from a single duct (Fig. 3, end of this section). This results in a slightly nonuniform distribution of coolant to the fuel channels and, consequently, a nonsymmetrical temperature distribution. Figure 4.13 of Appendix I shows the resulting distribution of stored energy below 250°C, and Fig. 4.25 shows the approximate region containing stored energy in excess of the graphite specific heat curve.

It became apparent during the early investigation of methods for anneal-

ing the reactor that effective annealing temperatures might not be safely attained in this region by use of the reactor nuclear power and the present coolant flow direction. Subsequent engineering studies and experimentation led to the selection of a system whereby the reactor coolant flow, and thus the axial temperature distribution, could be reversed. A discussion of other systems investigated is given in Appendix III.

Method of Annealing

The reversed air flow system to be used will be obtained by altering the present reactor coolant path to appear as shown in Fig. 3 and 7, end of this section. This will permit temporary operation of the reactor at near full power, if desired, with coolant flow through the fuel channels from west to east (normal flow is east to west). The normal temperature distribution will also be reversed so that the maximum graphite temperatures will occur in the normally cold, or high-stored energy, region of the moderator (Fig. 4, end of this section). Some typical winter temperature plots are shown in Fig. 5, end of this section, which indiate that the entire volume of graphite containing stored energy that can be released spontaneously will be enclosed by a temperature boundary of about 120°C and that all points within this boundary will be at higher temperatures, up to about 160°C maximum. For summer operation, the values can be expected to increase from 40 to 60°C, depending upon the inlet air temperature and the particular location in the reactor. In a recent laboratory experiment it was found that the stored energy in excess of the graphite specific heat curve could be removed from OGR graphite by annealing at 138.5°C for 5 hr. The spectral distribution of the remaining stored energy then appears somewhat as shown by Fig. 3, Appendix IA. From this it appears that relatively low temperatures, which

are readily obtainable in the OGR with reversed air flow, would effectively remove the stored energy in excess of the graphite specific heat curve.

Future Annealing

The effects of low-temperature annealing of the OGR graphite is discussed in Appendix IA (supplement to Appendix I, by M. C. Wittels). Here it is pointed out that low-temperature annealing is effective in removing the spontaneously releasable stored energy from OGR graphite in its present condition. However, the spectral distribution of stored energy accumulated in the future will be such that annealing at somewhat higher temperatures will be necessary to prevent a recurrence. Therefore, it appears that periodic annealing (perhaps annually) should be planned to prevent the accumulation of stored energy which would require excessively high graphite temperatures for effective annealing.

Description of the Reactor and Coolant System

The reactor is described in detail in Appendix II (selections from Geneva Conference Paper 486, "Research Program and Operating Experience on ORNL Reactors," by M. E. Ramsey and C. D. Cagle); however, some description is needed here for clarity.

The reactor moderator is a 24-ft cube made up of 4 x 4 x 50 in. graphite blocks stacked with their longitudinal axes parallel. This stack is traversed by 1248 passages having a square cross section (1 3/4-in. sides) into which fuel is charged and through which cooling air flows. Only about 830 of these channels are loaded with fuel, and the remainder are partially plugged to reduce air flow losses. The graphite stack rests on a concrete slab whose foundation is bed rock and is housed on the remaining five sides by a 7-ft-thick biological shield consisting of an inner and outer 1-ft-thick

layer of ordinary concrete and an intermediate 5-ft-thick layer of barytes-Haydite concrete. Inlet- and exit-air manifolds were formed by spacing the east and west shields 3 ft and 6 ft from the graphite surfaces, respectively. Hollow steel tubes penetrate the east shield to provide access to the fuel channels. The inlet-air manifold is spanned by permanently installed extensions to these tubes which extend into the fuel channels from 2 to 4 in. The wall of each of these tubes contains a 1/2- x 4-in. slot near the fuel channel opening to reduce the flow resistance to air entering the channels. During operation of the reactor, each of the fuel-channel access tubes through the biological shield is closed with a paraffin-and-lead-shot-filled plug.

With the exception of the fan discharge, the entire coolant system operates below atmospheric pressure. Atmospheric air enters the system through rough Fiberglas dust filters and flows through a concrete duct to the inlet-air manifold. Here the air flow divides, flows through the fuel channels, over the fuel element surfaces, and discharges into the exit-air manifold. From here, the air follows a concrete duct to a filter house where it passes through two sets of filters, thence through a duct to the intake of two parallel centrifugal fans which discharge through a 200-ft-high stack to the atmosphere. The system is shown schematically in Fig. 3, end of this section. Typical flow rates, system pressures, and temperatures are indicated at various points in the system.

Brief History of the Oak Ridge Graphite Reactor

The ORNL graphite-moderated, air-cooled, natural-uranium reactor was built in 1943 as the first continuously operating reactor and produced small quantities of plutonium for the chemical pilot plant for the Hanford reactors as well as serving as a tool for early neutron tests. Although the original

design was for a 1-Mw reactor, the reactor core size and shield were sufficiently overdesigned so that a power level of approximately 4 Mw was attained by installing larger fans to increase capacity of the air cooling system.

Following completion of the Hanford design, the reactor became a research and radioisotope producing facility. It has 45 available 4-in.-square openings into the core; two 14-in.-square low-flux tunnels, originally for biological specimens; a vertical and a horizontal thermal column; and 418 unused fuel channels as facilities for research and radioisotope production.

The present power level is 3.5 Mw, and the maximum thermal neutron flux is 1.1×10^{12} neutrons$\cdot$cm$^{-2}\cdot$sec^{-1}. During its entire lifetime the reactor has operated almost continuously on a round-the-clock schedule with no troubles other than failure of external components and rupturing of fuel elements. A brief history of significant additions and changes is as follows:

1944 - Stoppage of most of air flow between top of moderator and top shield by use of an H-beam inserted through two facing ionization chamber holes. Installation of two 990 hp fans in cooling system. Plugging of unused fuel channels. Elevation of power level to 3.5 to 4 Mw. Better-welded unbonded fuel elements charged.

1948 - Filter house built into exit-air system because ruptured slugs caused radioactive fallout from the exhaust stack.

1950--
1951 - New concrete floors and experiment balconies built on south side of reactor. Instrument wiring revised and put into permanent conduit. Metal relay racks installed. Control room enclosed and air conditioning supplied to experiment balconies and control room. Building painted and reroofed. Fan building rebuilt.

1952 - Four 1 1/2-in.-dia. boron-steel vertical safety rods replaced by three 3 1/2-in.-square rods containing cadmium sheet in order to release a vertical hole for experiment usage. Reactor charged with bonded, natural uranium slugs.

1954--
1955 - Automatic control installed.

1956--
1957 - Rewiring of circuits. Installation of Log N, period, and fission chamber startup instrumentation.

Changes in the Reactor Coolant System

Revision of the present reactor coolant system to permit reversal of the air flow through the fuel channels requires the permanent installation of an interconnecting duct between the existing inlet-and exit-air ducts and a new duct or pipe to bring atmospheric air to a new entrance opening into the present exit duct. Control valves or dampers will be installed for selecting the desired flow direction and for control of the flow rate. A variable orifice plate will be installed for use in determining the flow rate and for limiting it to a preset maximum value. The approximate locations for the cross duct and the new inlet-air duct are shown by the arrows in Fig. 3. Figures 6 and 7, end of this section, are plan views of the present and the revised systems including (schematically) the valve and orifice locations and indicating the normal and reversed flow paths. Duct sizes and general component specifications are such as to permit reversed flow rates up to near normal-full-power values if desirable. This provision will permit complete control of the reactor temperatures and, in addition, will permit operation of the reactor for periods of several days under reversed flow conditions if such a procedure becomes desirable.

Special Instruments

Temperature Measurements

Under normal operating conditions, the graphite temperature is monitored by three thermocouples installed in two of the 4-in. experiment holes. One of these is about 1 ft from the center of the reactor and reads an average value of about 135°C. Another is near the boundary of the maximum stored energy region and is continuously monitored by a recording instrument. The third is read hourly with a scanning recorder.

For the annealing over 50 additional thermocouples will be used for monitoring the graphite temperatures. Most of these will be located in uncharged fuel channels which will be plugged to prevent coolant flow, thus permitting more accurate bulk graphite temperature measurements. Each thermocouple bead will be imbedded in a $1\frac{11}{16} \times 1\frac{11}{16} \times 10$ in. graphite block, with the point of measurement being just under the surface of the midpoint of one of the $1\frac{11}{16}$- x 10-in. surfaces. The block will be installed so that this surface of the block contacts one of the fuel channel surfaces, thus placing the thermocouple bead in close proximity to the reactor graphite. Air flow through the $\frac{1}{16}$-in. gap between two surfaces of the block and fuel channel is limited by an oversized square of asbestos blanket, which is attached to one end of each block. The asbestos is held in place by an aluminum plate, as shown in Fig. 8, end of this section. Several of these blocks in a single channel effects almost complete closure of the air flow area. The planned radial and axial distribution of these thermocouples is shown in Figs. 9 and 11, end of this section.

Fuel Element Temperature Monitors

Approximately 40 fuel-element-cladding temperatures are normally monitored, either continuously or by taking hourly readings. These thermocouples

are distributed in a manner which provides a radial temperature profile. They are positioned about 2 ft downstream from the center of the fuel rows, since experience has shown this to be the location of the maximum fuel temperature.

Approximately 40 new fuel-element thermocouples will be installed for the annealing operating, since the maximum fuel element temperatures will be east of the center fuel slugs with the reversed air flow. Furthermore, it is desirable to monitor the temperatures of several of the elements located in the maximum-stored-energy region so that the effect of the energy release on the fuel temperature can be immediately detected. The fuel element thermocouple locations will be approximately as shown in Figs. 10 and 12, end of this section.

Thermocouples for Concrete Temperatures

The coolant air will exit from the fuel channels into the present inlet manifold, thus exposing the unprotected front-face shield to higher-than-normal temperatures. Although the average rise in air temperature in passing through the reactor is only about 60°C, the air discharges from some of the central channels at temperatures near 225°C. This air might impinge directly upon the front-face shield wall and possibly cause spalling of the concrete unless precautionary measures are taken. These are discussed in the section entitled "Concrete Shielding Protection." Several thermocouples will be installed at appropriate locations so that the temperature conditions of the shield will be known. Figure 13, end of this section, illustrates the type of installation planned for this.

Air Flow Instruments

The limiting orifice plate to be incorporated in the cross-duct installation can be calibrated to provide a means for determining the exit coolant

flow rate. In addition, pitot tube and hot wire anemometers will be located in the air ducts so that measurement of the air flow will be possible at all flow rates during the annealing.

The differential pressure across the reactor is a direct indication of flow rate and is monitored by a recording instrument which is equipped to scram the reactor if the flow diminishes by approximately 60%. This instrument will be operative during annealing but with the scram setting adjusted as required.

Reactor Power Instruments

Measurements of the reactor power level are obtained by several independent methods. Calibration of these instruments is based upon heat-power rates determined from the measured air flow rates and air temperature rise.

Three gamma-compensated, boron-coated ionization chambers are now in use. One is monitored by use of a galvanometer and is used for reactor startup and during manual operation. A second one is used for automatic power level control, and the third one for vernier adjustments during manual operation or as a spare.

Three non-gamma-compensated, boron-coated ionization chambers are used for safety instruments to scram the reactor electronically if the power level exceeds 4.2 Mw. Gamma compensation is not necessary in these chambers, since they are only effective at relatively high neutron levels where the gamma contribution is negligible and need not be precise to satisfactorily serve their purpose.

In addition to the ionization chambers, a system of boron-coated thermopiles, whose composite signal is proportional to the power level, is monitored by a recording instrument containing provision for scramming the reactor at a power level of 4.2 Mw.

The approximate locations in the reactor of the above sensing devices are indicated by arrows in Fig. 15, end of this section. The distribution of these sensing elements is such that no large portion of the reactor is isolated from nuclear instrumentation, therefore precluding the possibility of an unrecognized power excursion. For this reason, no additional power monitoring equipment is deemed necessary; however, the scram settings will be adjusted as required.

Radiation Detection Equipment

The present means of detecting radioactivity in the exit cooling air consists of two ionization chambers and a probe for collecting particulate activity on oil-soaked fabric.

One of the ionization chambers is located in one of the exit filter cells and reads the direct radiation from radioactive material collected by the filters. The second chamber monitors a continuous stream of air which is drawn from the system at a point downstream from the filters. The normal reading from this chamber is due primarily to argon activity (~500 curies/day) produced by activation of atmospheric argon during transit through the reactor and to small amounts of fission gases resulting from fission of uranium deposited in some fuel channels which have contained damaged fuel elements. Both ionization chambers are monitored by recording instruments located in the control room.

The probe is a device for lowering oil-soaked cloths into the exit air duct at a point near the reactor. The cloths are removed from the duct periodically, and the radioactivity of the collected particles is determined with an electroscope. This method is insensitive to small changes in radioactivity level but is useful in detecting the latter stages of fuel rupture.

a. Filter Tape Monitors

This instrument is of a type normally used for the determination of atmospheric air contamination. One such device will be employed near the exit-air manifold at the reactor and possibly one at the base of the stack. In operation, air is drawn from the exit duct through a moving filter tape, which is subsequently counted by a G-M counter or other radiation detecting instrument. The collected sample is primarily particulate but includes some trapped gases such as argon-41.

b. Charcoal Trap Monitor

This instrument draws an air sample through a particulate filter and then through a charcoal trap for collecting fission gases which may be contained in the exit-air stream. The activity collected by the charcoal trap will be continuously monitored by a G-M counter.

The contents of the charcoal trap may be analyzed radiochemically in the event that quantitative analysis of released activity becomes desirable.

A sudden increase in the reading obtained from any of these instruments may be indicative of fuel element failure.

c. Health Physics Monitors

The normal complement of health physics instruments will also be in operation. Several instruments are located in the building for constant monitoring of both direct radiation and air contamination. These are equipped with alarms which sound before tolerance levels are exceeded.

Matters Requiring Special Attention

Experiments

The experiments are designed to function properly under the maximum

temperature conditions normally expected in the reactor at the experiment location. They will be surveyed to be sure that they are capable of withstanding the conditions expected during the annealing operation. Any which may be damaged or become hazardous will be altered or removed from the reactor before beginning the annealing operation.

Reactor Components

All reactor components, such as ionization chambers, safety rods (cadmium lined) and the bismuth gamma-shield under the thermal column, are being reviewed with respect to possible thermal effects under the expected annealing temperature conditions. Any of these deemed subject to damage will be revised or provided with additional cooling and adequate temperature monitors.

Emergency Instrument Power

The reactor instruments and the control-room lighting will be assured of electrical supply during a failure of normal electrical power by an automatic transfer connection to a 350-kw, diesel-driven, emergency generator in the Oak Ridge Reactor (ORR) building. A reactor scram would be automatically actuated and is guaranteed by the three vertical and two horizontal rods which depend only upon gravity for insertion.

Reactivity Adjustment

During the final preannealing preparations, the excess reactivity will be reduced to the minimum value that will allow operation at the desired temperatures (approximately 0.7%). Boron-steel cylinders, or other material having a comparable melting point, will be inserted into empty fuel channels and experiment holes to obtain the required reduction.

By operating with a minimum of excess reactivity, any temperature excursion would be limited to a few degrees above the desired temperatures due to

the negative temperature coefficient (Table 4, Appendix II). The possible effects of such an excursion are discussed under hazards.

In addition to the safety benefit, the reactivity adjustment will also permit almost complete withdrawal of the horizontal rods, resulting in higher fluxes and, consequently, higher possible temperatures in those regions of the reactor core containing the highest amounts of stored energy.

Shutdown for Reduction of Fission Product Radioactivity

The reactor will be shut down for a minumum of 48 hr prior to the annealing operation for the removal of experiments, final installation of special equipment, and a general checkout of instruments. This time will also allow partial decay of the fission products in the reactor fuel, thus reducing the radioactivity release in the event of fuel element rupture.

Annealing Date

The operation will be performed during a holiday weekend or similar occasion when a minimum number of personnel will be in the Laboratory area. If the necessary approvals can be obtained in time, it is believed that the date could be in the early summer.

Concrete Shielding Protection

As mentioned previously, the front-face shielding wall contains no built-in thermal protection, as does the rear wall. To preclude the possibility of thermal damage to the front shield, additional cooling will be provided by continuously injecting water through some of the fuel-channel access tubes in the top horizontal and south vertical zones (see Fig. 14, end of this section). The water will be sprayed upon the inner surface of the shield and allowed to flow down and across the surface. Excess water (that not evaporated) will flow into a drain in the floor of the present inlet duct and thence to the retention ponds. If additional cooling is

needed, as indicated by temperature measurements at the time, several of the front-face shielding plugs can be loosened to allow cool air from the building to flow in through the fuel-channel access tubes, thus diluting the high-temperature air as it discharges from the fuel channels. This will further reduce the possibility of overheating the front-face concrete.

In addition to the thermal considerations already disucssed, the front-face shield wall must be protected from excessive pressure differentials which could lead to fracture of the wall. The normal differential pressure across this wall is about 13 in. H_2O. In order to obtain near-full-flow conditions in the reversed direction, this must be increased to nearly 29 in. H_2O. A value somewhat less than this will be set as the maximum permissible value and will be enforced by a pressure monitor equipped to shut off the fans and scram the reactor.

Fan Standby

In the event of an electrical power failure, approximately 5000 scfm of cooling air will be immediately available by using the emergency steam fan, which will be kept operating in standby at all times during the annealing. This, in addition to the normal stack draft which is effective for several minutes after reactor shutdown, will provide more than adequate cooling capacity to remove the reactor shutdown heat. The pressure-operated safety trips have been proved on many previous occasions when both electrical fans were shut down unexpectedly by power outages. Fuel element temperature records obtained during these fan failures show less than a 10°C transient increase in fuel element temperatures in such cases, even when the emergency steam fan is inoperative. Such loss of coolant would cause immediate shutdown of the reactor.

The fan damper system is under study to ascertain if any changes will

be necessary to ensure optimum coolant flow during any possible failure conditions.

Personnel Requirements

Operation of the reactor will be done by the regular shift operating crew as in normal procedures. However, additional operators will be scheduled to work during the annealing operation. One of these will be stationed at the fan house so that necessary changes in the cooling flow can be accomplished rapidly. Another will make routine checks and be generally available as needed.

A supervisor will be present in the control room at all times to direct the operation. At least two experienced people will process all temperature data so that the over-all condition of the reactor will be known at all times.

Additional Personnel

Health Physics -- for routine periodic surveys and special monitoring;

Instrument mechanic -- for routine checks and maintenance;

Electrician -- for routine checks and maintenance.

Limitations

Fuel Element Temperature Limit

During 1952, the reactor fuel was changed to a bonded type using an aluminum-silicon eutectic as an interface material between the uranium metal and the aluminum cladding. The purpose of this change was to provide good heat transfer from the uranium to the cladding and to separate the two materials, thus reducing the possibility of uranium-aluminum reactions in the 200 to 300°C range. It was hoped that the new fuel would permit surface temperatures in excess of 300°C and, therefore, operation of the reactor at higher power levels. This did not prove to be the case, however, since the

fuel element rupture rate proved to be excessive when the clad surface temperatures were mainteained at 300°C for extended periods, even though all of the elements in use were pretested in an oven at 400°C for one week. Subsequently, an upper limit of 285°C was placed on the fuel-clad surface temperature, thereby effecting an appreciable rupture rate reduction. This restriction upon fuel element temperatures automatically limits the reactor power to about 3.5 Mw in hot weather.

In view of dependence of fuel element rupture rate upon fuel surface temperature, it appears wise to maintain the 285°C upper limit during the annealing. To enforce this limit, the temperature of the hottest slug equipped with a thermocouple is monitored by a recording instrument which will scram the reactor if the fuel temperature exceeds 285°C.

Graphite Temperature Limit

The temperature range at which thermal damage to graphite becomes appreciable (700 to 800°C, Appendix V) is far in excess of the permissible fuel-element temperature. As pointed out previously, graphite temperatures in this range could only be the result of fuel element failure and loss of coolant. During the normal process of heat transfer in the reactor, the graphite temperature always lies between that of the fuel and the cooling-air temperatures. There are two special cases when this may not be true.

After a reactor shutdown, cooling air continues to flow; since the graphite loses heat slowly, the fuel-element temperature is soon reduced to a value below that of the graphite. This always occurs at temperatures much lower than the maximum safe limit and, therefore, presents no hazard.

The second case, which is of primary concern here, would result from the sudden release of a large quantity of stored energy. Under such conditions, the maximum safe fuel-element temperature could be approached quite

rapidly due to this additional heat. Protection of the fuel elements from overheating would still be the controlling factor. As pointed out previously, an adiabatic release of stored energy in OGR graphite can produce a temperature rise of only 100°C; since the release point is approximately 165°C, the maximum resulting graphite temperature would be 265°C. It, therefore, appears adequate and desirable to place the same upper temperature limit of 285°C on the graphite as was established for the fuel elements.

Concrete Temperature Limit

As previously mentioned, the front-face shielding wall has no built-in protection from thermal damage since it was intended to remain at essentially room temperature. However, under the conditions of reversed flow, the shield will be exposed to a heat source and therefore must be guarded against overheating.

An upper limit of 100°C has been placed on the concrete temperature because of the uncertainty involved in analysis of possible damage at higher temperatures.

Air Radioactivity Limit

Experienced reactor operations personnel have become quite expert at interpreting the indications of various radiation monitors, as well as changes of fuel element temperature in terms of fuel element failure. When sufficient evidence of the existence of a fuel element rupture has been noted, the reactor is shut down and the fuel is inspected by visually scanning all loaded channels. When the damaged element is located, it is discharged from the reactor, new fuel is added, and operation is resumed.

The exit-air radioactivity is never allowed to become excessive before the reactor is shut down. Radiation readings of 200 mr/hr, as obtained by the probe, are quite common upon fuel element failure. Such levels have

been sustained for many hours and apparently constitute no hazard either from direct radiation or atmospheric contamination. Tolerance levels for air activity have not been exceeded due to fuel element failure since the installation of the exit-air filters in 1948.

It appears reasonable that the limits placed upon the maximum permissible exit-air radioactivity be consistent with the known safe levels. If these limits are exceeded, the reactor will be shut down and suitable action taken. Under such conditions, unsafe radiation levels must result from causes other than nuclear power generation.

Period Limits

The rate at which the reactor power may be increased during startup and at full power is limited by the action of a period monitor. This instrument prohibits rod withdrawal if the reactor is on a positive period of less than 24 sec. A rod reverse is requested by periods less than 10 sec and a scram by periods of 5 sec or less. The latter value was chosen in order to limit the increase in reactor power to 25% above that at the time the scram is requested.

Hazards Evaluation

General

All the hazards which may be associated with the planned OGR annealing exist during normal operation and are amplified but slightly because of the stored energy. The possibility of a uranium and graphite fire and the subsequent release of large quantities of radioactive fission products to the surrounding area has been of primary concern since the Windscale incident in October, 1957. The British official report[1] indicated that the most

[1] Accident at Windscale No. 1 Pile on 10 October 1957, presented to Parliament by the Prime Minister, November, 1957.

probable cause of the fire was some uranium fuel cartridges which ruptured during the second nuclear heating. The resulting oxidation of uranium subsequently heated the graphite to burning temperatures.

The normal distribution of temperature monitors and neutron sensing instruments, plus the exit-air radioactivity detecting equipment in the OGR, preclude the possibility of an appreciable unrecognized power or temperature excursion in this reactor. Also, special equipment and procedures have been designed for the annealing to further enhance the existing proven safety features.

Graphite and Uranium Fire

In considering the hazards which may arise during the annealing procedure, most concern is expressed over the possibility of a fire occurring in the reactor. It is supposed that the likelihood of such a fire is significantly increased by reason of the stored energy in the graphite.

Analysis of the conditions required to cause a fire leads to the conclusion that to start at all the fire must originate in the fuel and not in the graphite. For the case where there is no blockage of individual fuel channels, this conclusion is obvious from a consideration of the nuclear characteristics of the OGR. Under normal summer operating conditions, the maximum graphite temperature is about 230°C; the graphite temperature coefficient is $(-2.08 \times 10^{-5} \Delta k/k)/^{\circ}C$. Thus, operating with $k_{eff} = 1.007$, as is anticipated, the reactor will become subcritical when the maximum graphite and metal temperature reaches 236°C. Adding to this the 100°C maximum possible adiabatic temperature rise from the release of stored energy, the maximum graphite temperature which can possibly be achieved with initially equal graphite and metal temperatures is 336°C, which is 584°C below the combustion temperature of graphite (920°C).

The situation is not so clear-cut in the case where one or a few channels are partially blocked; however, the conclusion still follows: under this condition, heat removal from both graphite and fuel will be restricted so that the local temperature can increase. Now, the heat generated in the fuel is transmitted to the graphite; and the heat capacity of the graphite associated with a given slug is about 200 times that of the slug. Thus, when the temperature of the fuel is raised, the bulk temperature of the graphite will lag behind that of the fuel. Under these circumstances, if the number of overheated channels is insufficient to affect the total reactivity and if the condition is allowed to persist unchecked, it is possible that the slug temperature may exceed the combustion temperature of graphite and thus start a fire in that portion of the graphite.

It is not known with certainty at exactly what slug temperature combustion of the graphite would commence; however, it is clear that it must be at least 820°C. Allowance has been made here for an additional increase of 100°C in the graphite as a result of stored energy release.

Recent work by Creek, Martin, and Parker indicates that uranium-metal slugs heated to 1150°C in air oxidize quite rapidly but cool down quickly as soon as the heat source is removed; so, the combustion is not self-sustaining. Thus, even though the metal were heated to 820°C, no fire would result unless the graphite were ignited.

One can consider two ways in which the metal temperature could be made to exceed the combustion temperature of the graphite. The first of these is the introduction of a power transient having a sufficiently short period so that the combustion temperature is reached in the slugs well in advance of the establishment of temperature equilibrium; that is, an essentially adiabatic rise of the slug temperature to 820°C could conceivably cause a fire.

In the first place, the likelihood of such a transient is, if anything, smaller during the annealing process than during normal operation because of the additional attention being given the neutron control during this process. In the second place, the metal temperature could rise to a maximum of 863°C before the metal temperature coefficient independently would reduce the excess reactivity to zero. Since it is not possible to heat the metal without some heating of graphite, the maximum fuel temperature would be far less than 863°C due to the negative temperature coefficient of the graphite. In this connection, an increase in the graphite temperature due to the release of stored energy would have the beneficial effect of still further quenching the transient.

The occurrence of such a transient is considered virtually impossible and, in any case, does not appear to be related to the annealing process. Supporting this contention is a history of 16 years' operation of the OGR in which no transients of anywhere near the order of magnitude described above have been observed.

The second and apparently only credible chain of events which could cause a fire involves the blocking of one or perhaps several of the fuel channels. With the heat removal thus restricted, the fuel temperature would increase and possibly result in damage, rupture, or melting of some or all of the slugs in the channel. This damage, in turn, could further block the channel, thus completely shutting off the flow of air and causing dangerous temperatures.

On several occasions in the past, fuel channels have been completely blocked by ruptured slugs. Two of these channels were operated for between 7 and 14 days from the time the first symptoms were noticed until the blockage was so bad that it was necessary to use a drill to clear the channel of

oxidized fuel. In neither of these cases was there any evidence that the aluminum jackets (mp, 660°C) had melted. While these channels (No. 2079 and 1069) are not located at the point of maximum flux, they are in a region of higher than average flux; and during the period of blockage, the reactor had been operated continuously at 3800 kw. These facts seem to indicate that the increase in slug temperature in an isolated, blocked channel is less than 400°C. A comparison of the heat generation rate in channel 1069 and that in the channels in which maximum heat generation occurs leads to the conclusion that the maximum heat generation exceeds that in channel 1069 by a factor of 1.3. The normal temperature of the hottest slug in channel 1069 was thus 200°C, and the normal temperature of the hottest slug in the reactor is 250°C. Since the observed increase in temperature in channel 1069 was less than 460°C (660° - 200°), one would expect that the increase in temperature in the hottest channel under conditions of flow loss would be less than 600°C (460° x 1.3). Thus, the maximum temperature rise from nuclear heating would not exceed 850°C (600° + 250°). If one postulates that the temperature of this slug is further increased by 100°C because of the release of stored energy, the maximum temperature could never exceed the combustion temperature of the graphite (920°C) by more than 30°C.

In assessing these figures, it must be realized that each number used is an upper bound and obviously not the least upper bound. Further, it can be pointed out first that it is virtually impossible for an energy release in the reactor to be adiabatic and, second, that loss of oxygen due to channel blockage will have the effect of inhibiting combustion.

There is, of course, no intention of operating the reactor during the anneal if one of the channels should become blocked. The only difficulty is in establishing that a blockage has occurred. Since the only obvious

cause for a channel blockage is the oxidation of a ruptured slug, the condition would be detected as radioactivity in the coolant air stream by the radiation monitors located in the exit-air duct. The possibility does exist that no radioactivity would reach these monitors if a complete stoppage of air from the channel occurred. This is considered extremely unlikely because some deterioration of the ruptured element is necessary to block the channel. Even in the case of the completely blocked channels mentioned above, close examination of data showed an initial increase in the radioactivity of the exit air which later dropped to near normal.

Approximately 80 of the fuel slugs will be equipped with thermocouples, and any unusual rise in temperature would suggest the possibility of a blocked channel. The number of channels equipped with temperature measuring devices will be increased for the annealing so that the probability of detecting a blocked channel by this method is somewhat better during the annealing than it is during normal operation. It should be pointed out further that these thermocouples are located in those positions where the amount of stored energy is greatest.

The foregoing information, together with the fact that the stored energy in excess of the graphite specific heat curve will be released during the initial stages of the annealing at a temperature below which a spontaneous stored-energy release can occur, leads to the conclusion that the probability of a fire during the annealing is no greater than that associated with the routine operation of the reactor. The procedures which are available to combat such a situation, should it occur, are described in a later section.

Experiments

During normal operation, the inventory of target materials and experimental apparatus in the reactor is very large. Some of these may possess

certain properties or design features which, under the expected annealing conditions, constitute a potential hazard. Materials capable of ignition, such as organic liquids, gaskets, and electrical insulation, are obvious sources of fire. Light metals, such as lithium or magnesium, though igniting at high temperatures are even more dangerous due to the large heat-release rate and the difficult extinguishing methods required. Other sources of danger may be experiments containing electrical heaters capable of appreciable heat production rates or possible electrical sparks due to component failure. Equipment or materials having any of these or similarly dangerous features will be removed from the reactor prior to the annealing operation.

Fission Product Release

In considering the release of fission products to the atmosphere, the maximum credible accident will be assumed. Such an accident is postulated to consist of the development of a fuel element rupture in a fuel channel of maximum power production, with subsequent complete oxidation of all fuel in the channel and the release of saturation quantities of all volatile fission products. Essentially all the particulates are assumed to be removed from the exit air by the filters which have an efficiency of greater than 99% down to particle sizes of 0.2 μ in diameter. Conversely, it is assumed that all volative fission products are released to the atmosphere. The assumption of complete removal of particulates from the exit air by the filters has considerable justification, according to previous experiences. On several occasions, oxidation following the failure of a fuel-slug cladding has resulted in the total loss of fuel (2.5 lb of uranium) from that element. During one very severe case after the exit filters were installed, about 20 lb of uranium was lost to the coolant stream from a single channel. Area air monitors and health physics surveys have shown no perceptible increases

in air activity, either particulate or otherwise, during these times. It therefore appears that the stipulated credible accident could not produce concentrations of air-borne particulates in excess of permissible levels.

Experience has further shown that the complete loss of fuel from a ruptured fuel element occurs over a minimum of several hours; more generally, several days are required. It may therefore be conservatively assumed that the activity is released at a constant rate over a period of 1 hr.

During the initial stages of the annealing, the coolant flow will be maintained between the one-half and full-flow values. Subsequently, the reactor will contain little or no stored energy in excess of the graphite specific heat curve, and therefore the remainder of the operation will present no hazard which does not exist during normal operation. Accordingly, one-half of full-flow (60,000 scfm) will be assumed as the minimum flow at the time an accident could occur.

The nuclides considered are listed in Table 1. The initial quantities of radioactivity, except for Kr^{85}, are saturation values contained by the maximum power producing fuel channel in the OGR, assuming continuous previous operation at full reactor power (3500 kw). The quantity of Kr^{85} was calculated upon the basis of seven years of continuous operation since the initial loading with the present type of fuel. The stack discharge rates are based upon a continuous constant release rate sufficient to release all products in one hour.

Purdy and Myers[2] have calculated that the maximum ground concentration resulting from an emission rate of one curie per second from the OGR stack

[2] D. R. Purdy and R. F. Myers, Calculations for Unit Emission or Airborne Contaminants, U. S. Weather Bureau, Oak Ridge, Tennessee, July 1, 1954.

TABLE 1. VOLATILE AND GASEOUS FISSION PRODUCTS OF IMPORTANCE IN THE EVENT OF A RADIOACTIVITY RELEASE DUE TO FUEL ELEMENT RUPTURES

Nuclide	Half Life	Initial Radioactivity (curies)	Stack Discharge Rate (curies/sec)	Maximum Resulting Atmospheric Concentration* (microcuries/cm^3)	Effective Energy**
Kr^{85}	4.5 h	71.7	1.99×10^{-2}	4.78×10^{-7}	1.2
Kr^{85}	10.27 y	5.2	$.144 \times 10^{-2}$	0.336×10^{-7}	1.2
Kr^{87}	78 m	148	4.07×10^{-2}	9.76×10^{-7}	1.9
Kr^{88}	2.8 h	215	5.98×10^{-2}	14.36×10^{-7}	4.8
Kr^{89}	3.2 m	263	7.3×10^{-2}	17.52×10^{-7}	1.33
I^{131}	8.1 d	160	4.45×10^{-2}	10.70×10^{-7}	0.60
I^{132}	2.4 h	206	5.72×10^{-2}	13.74×10^{-7}	1.4
I^{133}	22 h	258	7.18×10^{-2}	17.24×10^{-7}	1.0
I^{135}	6.7 h	338	9.4×10^{-2}	22.6×10^{-7}	1.7
$Xe^{133m} + Xe^{133}$	5.3 d	258	7.18×10^{-2}	17.24×10^{-7}	0.22
Xe^{135m}	15 m	33.8	$.94 \times 10^{-2}$	2.26×10^{-7}	0.05
Xe^{135}	9.2 h	338	9.4×10^{-2}	22.6×10^{-7}	0.55
TOTAL		2294.7	63.7×10^{-2}	15.3×10^{-6}	

*These values are based on initial activities, considering no decay or loss to the exit ducts and filters.

**Total gamma energy plus 1/3 of the maximum beta energy.

occurs at a wind velocity of 2 mph and is 1.2×10^{-5} curies/cm^3 of air when the stack discharge rate is 120,000 scfm of air. The values listed in Table 1 for the resulting maximum concentration of the various fission products considered were calculated on this basis, being approximately doubled for a flow rate of 60,000 scfm.

The internal dose hazard is considered to be due almost entirely to the iodines, since the maximum possible concentrations of other fission products is well below the maximum permissible concentrations for 50 years of continuous exposure.[3] From Table 1, the sum of the maximum concentrations of the iodines is 6.4×10^{-6} microcuries/cm^3 of air, which is considerably higher than the maximum permissible concentration for continuous exposure. However, this concentration could only exist for 1 hr if the wind velocity remained constant at 2 mph and not more than 2 to 3 hr under any realistic conditions for this area.

In the case of a 3-hr exposure, the maximum quantity of iodine retained by the average man can be found as follows:[3]

$$q = f\,A\,V\,t$$
$$= (0.75)\,(6.4 \times 10^{-6}) \times 1.25 \times 10^{6} \times 3$$
$$= 18 \text{ microcuries}$$

where

f = the per cent of the inhaled quantity of iodine which is retained by the body,

A = concentration of iodine in the air in microcuries/cm^3,

V = average breathing rate of a working man = 1.25×10^6 cc/hr,

t = the exposure time in hours.

[3] Report of Committee on Permissible Does for Internal Radiation, 1958 Revision (Unpublished).

This quantity of iodine is considered acceptable, since the usual dose administered for diagnostic purposes is 25 microcuries of I^{131}. This is a very conservative estimate of the maximum exposure, since several factors which would reduce the atmospheric concentration have been neglected. In the first place, it is highly improbable that such a large rate of fission product release would be experienced for reasons stated previously. Furthermore, under any conditions of fission product release, the iodine content of the air would be reduced considerably due to adsorption upon the duct walls, filters, and stack walls. In addition, no decay of the radioactive nuclides was considered so that even if the concentration considered did obtain intially, the 3-hr dose would be appreciably less than the 18 microcuries value which was calculated. Furthermore, I^{131} represents less than 20% of the total initial iodine concerntration, or a dose of 3.6 microcuries of I^{131}, maximum, from a 3-hr exposure. Finally, the area would be evacuated in a short time after such a condition developed so that the total exposure time of personnel would be much less than 3 hr.

The external dose rate from a cloud containing all of the nuclides listed in Table 1 may be estimated from the following equation:[4]

$$I = 10^3 \, GE$$

where I is the radiation intensity in rep/hr, G is the concentration in microcuries/cm^3, and E is the effective energy. Using the numbers from Table 1, the maximum intensity is found to be 16.5 mrep/hr. For a 3-hr exposure, this amounts to less than 50 mrep, or less than the normal permissible one day exposure.

[4] T. H. J. Burnett, Sampling Methods and Requirements for Estimating Airborne Radioparticulate Hazards, ORNL CF-52-11-1 (November, 1952).

Procedures

Annealing Procedures

A. Prior to startup, the following check list will be reviewed for completion:

1. Experiments removed or otherwise prepared.
2. Reactor components revised or conditioned.
   a. Control rods.
   b. Ion chambers.
   c. Lead or bismuth removed or provided with coolant and monitored.
3. Reactivity adjustments made.
4. Air flow system complete and set for reversed flow.
5. All shielding plugs in place.
6. Water injection system functioning properly.
7. Standby compressors in operation.
8. Standby steam fan running.
9. Exit air filters inspected and in good condition.
10. Health Physics personnel available for area checks.
11. All necessary personnel on hand and previously oriented.
12. Fire fighting equipment in readiness.
13. Overhead crane in proper working order.
14. Front face elevator functioning properly.
15. Ladders available for emergency use.
16. All instruments previously checked, alarm and scram points adjusted and functioning properly, new charts on all recording instruments, pens all inked.
17. Air flow measuring instruments functioning properly.
18. Heat power calibration performed.

19. ΔP instrument reversed.

20. Accumulators checked.

21. All control rod drives and release mechanisms checked.

22. All radiation instruments checked with source.

23. Power plant alerted.

24. Steam plant alerted.

25. Emergency electrical supply tied in and working properly.

26. All thermocouples indicating properly.

27. Emergency supplies replenished.

28. Building evacuation system checked.

29. Remote scram switches checked.

30. Building loud speaker system functioning properly.

31. Phone system to fan house working.

32. Building air conditioning and fans in order.

33. Water supply checked--emergency supply available.

34. Inert bottled gas supply checked.

35. Communication system in order.

36. Building cleared of unauthorized personnel.

37. Transportation available.

38. Fire Department alerted.

B. Establish the full air flow permissible in the reversed direction and check all flow and pressure monitors, fan motors, dampers and controls.

C. Bring the reactor to criticality as in normal startup procedures. Check all nuclear and related instrumentation.

D. Increase the reactor power gradually so that the maximum rate of rise of any fuel or graphite temperature does not exceed 10°C/hr. Continue this rate until the maximum graphite temperature is 140°C. During this

time the stored-energy release rate will not exceed 28 kw.

E. Maintain the 140°C maximum graphite temperature for 5 hr.

If no unusual conditions developed during step E, proceed to step F.

F. Increase the reactor power slowly so that the maximum rate of rise of any fuel or graphite temperature does not exceed 5°C/hr. Continue this rate until the maximum graphite temperature is approximately 160°C or the maximum fuel element temperature is 250°C. During this time (4 hr) the stored-energy release rate will not exceed 208 kw.

G. Maintain the maximum temperature attained in step F for 5 hr.

H. Reduce the reactor power level to approximately one-half the normal operating power.

I. Reduce the coolant flow rate slowly to re-establish the maximum fuel element temperature of 250°C. This should tend to broaden the temperature distribution and thus extend the region wherein annealing is being effected. During this operation the rate of rise of any fuel or graphite temperature should not exceed 10°C/hr at temperatures above those attained in step F. The maximum rate of stored energy release will then be 208 kw. Maintain the maximum fuel element temperature of 250°C for 5 hr.

J. Subsequent additional reductions may be made in the same manner as for steps H and I if it is deemed desirable. Following the final "cooking" period, the reactor will be shut down and the flow adjusted to near the normal full flow rate and the structure allowed to cool.

K. After the cooling period, preparation of the reactor for normal operation will begin. Core samples must be removed from the moderator for stored-energy evaluation but not necessarily prior to operation of the reactor. In fact, several hours of operation under normal conditions would reduce any contamination which may have been deposited in the bridge tubes

by the reversed flow and would make the coring operation a cleaner one.

Graphite-Fire Extinguishing Procedure

The fact that the OGR moderator is a combustible material which is in contact with an oxygen-containing cooling medium and is, therefore, capable of ignition under the additional influence of sufficient heat supply has been realized for many years. For this reason, some fire extinguishing equipment has always been a part of reactor emergency equipment. As an added measure for the annealing, additional equipment will be on hand, including a supply of bottled inert gas.

A uranium-graphite fire would be indicated by the exit-air activity monitors during the earliest stages of ignition due to the release of volatile fission products and radioactive oxide particles. If at any time the exit-air activity exceeds the preset limits (given under "Limitations"), fuel element failure may be indicated; and the reactor will immediately be shut down for inspection. The existence of a fire in the reactor would be readily detected since the inspection for ruptures requires the visual scanning of each loaded fuel channel.

If a fire were discovered, the coolant flow would be reduced to the minimum value permitted by the surrounding graphite temperatures. A special high velocity nozzle would be inserted into the fuel channel opening and large volumes of water applied directly to the burning region until the fire subsided. In the event the fuel channel should become blocked, thus preventing water flow to the west end of the channel, a special nozzle arrangement can be used to inject water into the west end of the fuel channels by way of scanner holes (Fig. 1, Appendix II) which penetrate the top shield into the west air manifold.

Some objection to the use of water for extinguishing metal fires is

sometimes raised on the supposition that the hydrogen evolved during the decomposition of water to form the metal oxide may present an explosion hazard. Such a possibility is not believed to exist here for the following reasons. First, if the fuel channel is not completely blocked, coolant flow would continue and the hydrogen would be removed from the channel (greatly diluted) as it formed. Second, the high heat-removal capacity of water would very quickly reduce the temperature below that required for the burning of metal or graphite and extinguish the fire. Furthermore, a high-volume-flow rate of water would remove evolved hydrogen even if the channel were blocked so that no air flow existed.

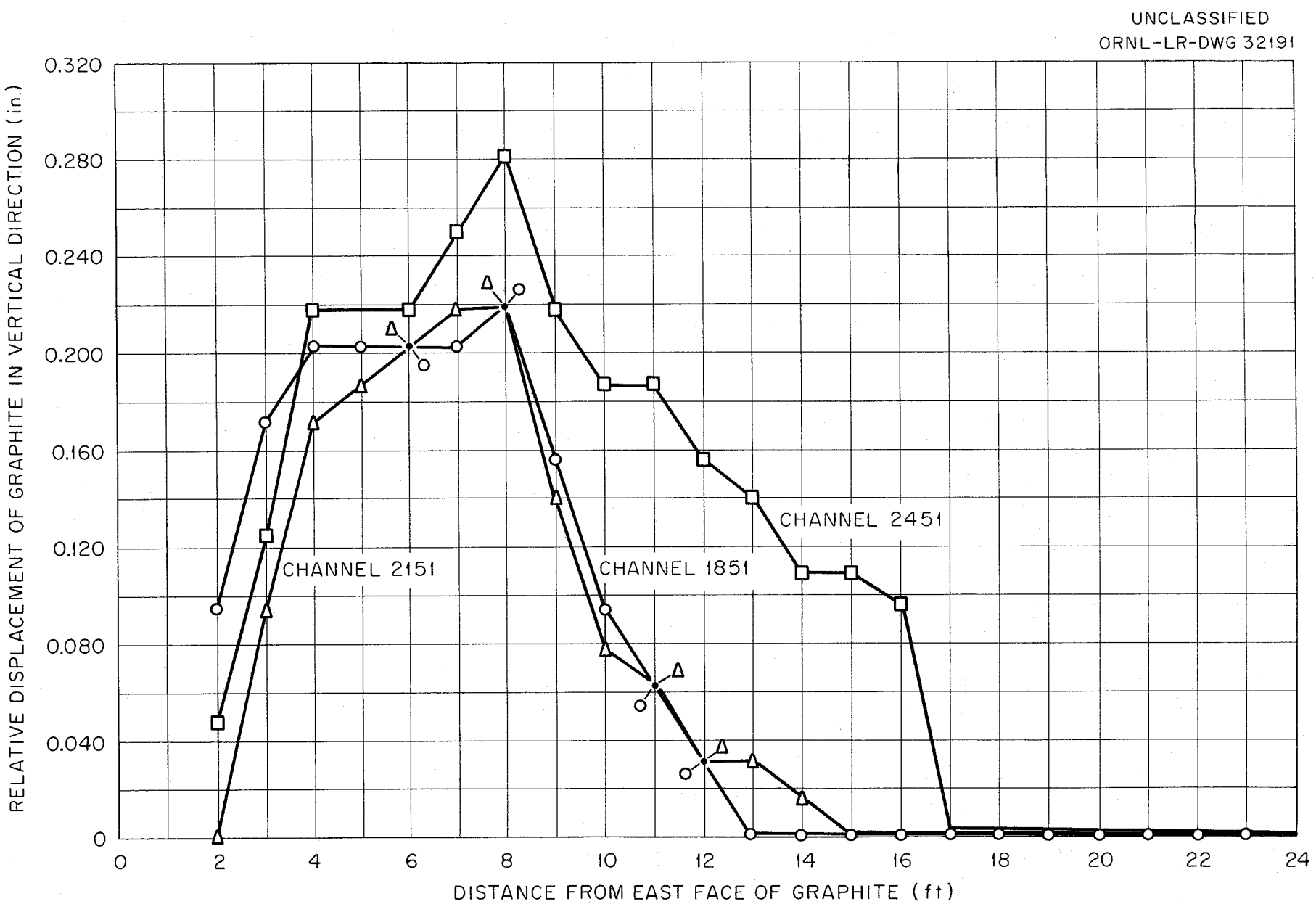
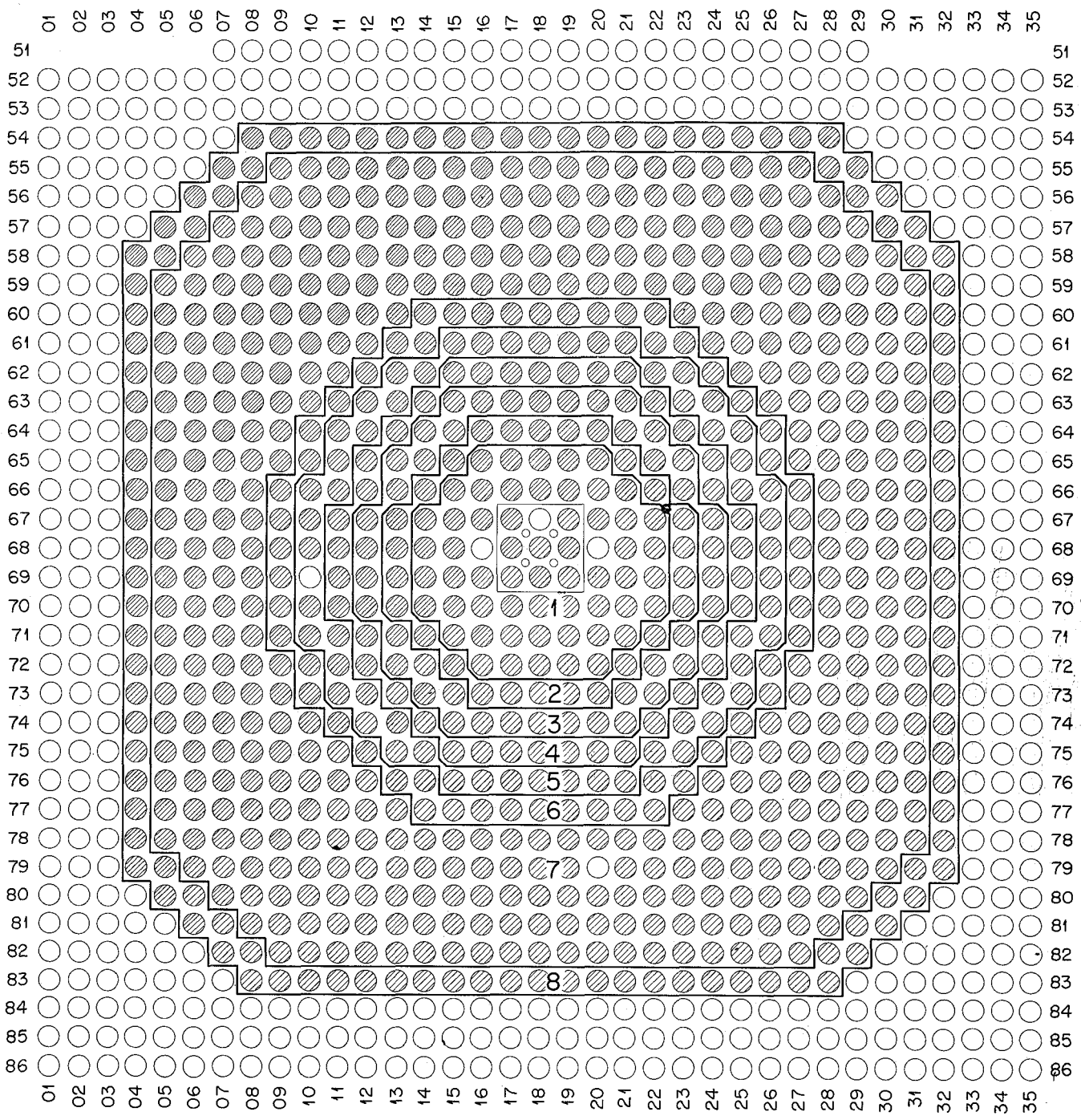


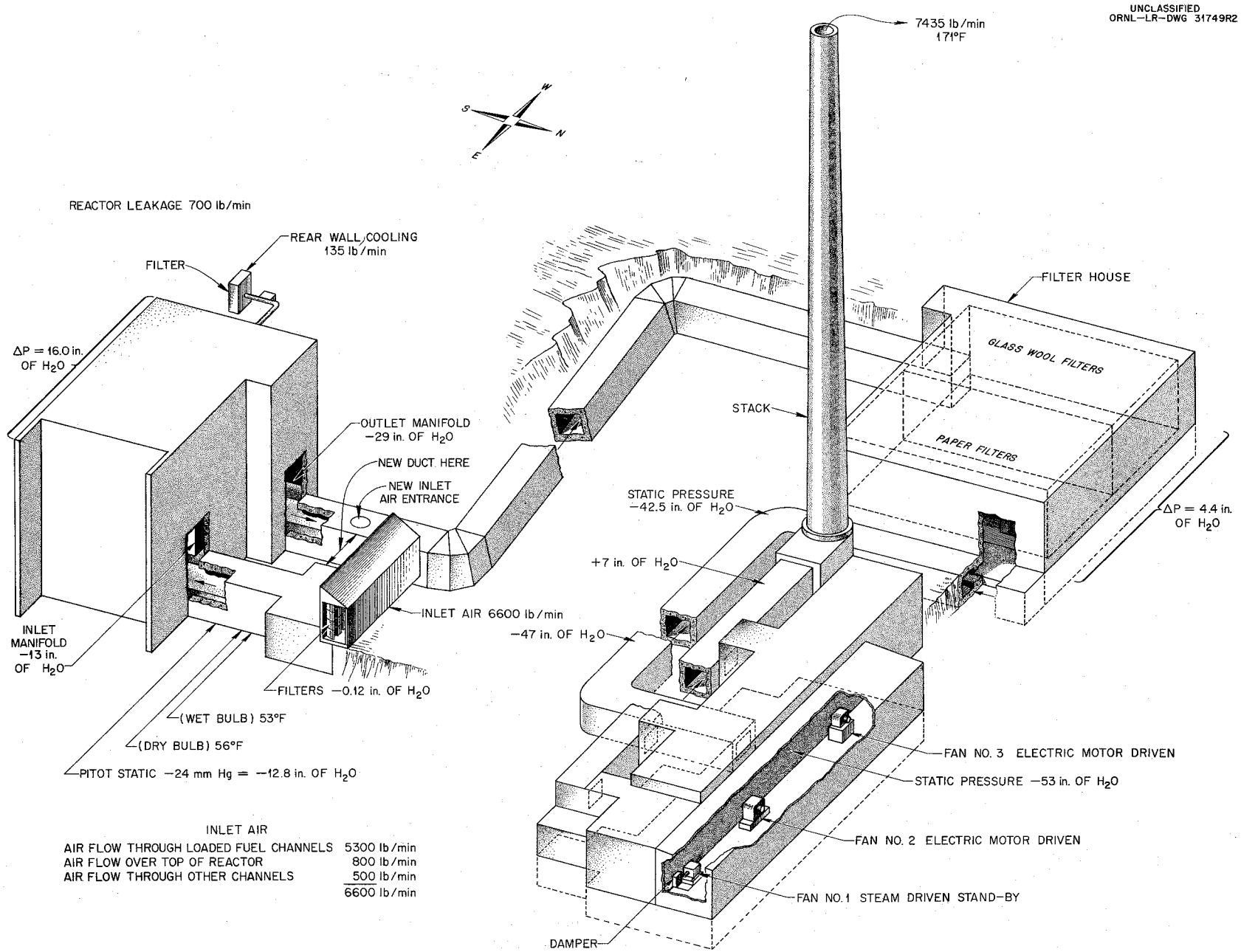
Fig. 1. Contour Measurements Through Graphite Reactor Fuel Channels.

UNCLASSIFIED
ORNL-LR-DWG 6755



ZONE	NUMBER CHANNELS	NUMBER SLUGS
1	60	41
2	26	45
3	42	51
4	38	51
5	58	54
6	54	54
7	438	54
8	114	47

Fig. 2. OGR Fuel Loading Pattern.



INLET AIR

AIR FLOW THROUGH LOADED FUEL CHANNELS	5300 lb/min
AIR FLOW OVER TOP OF REACTOR	800 lb/min
AIR FLOW THROUGH OTHER CHANNELS	500 lb/min
	6600 lb/min

Fig. 3. OGR Cooling System.

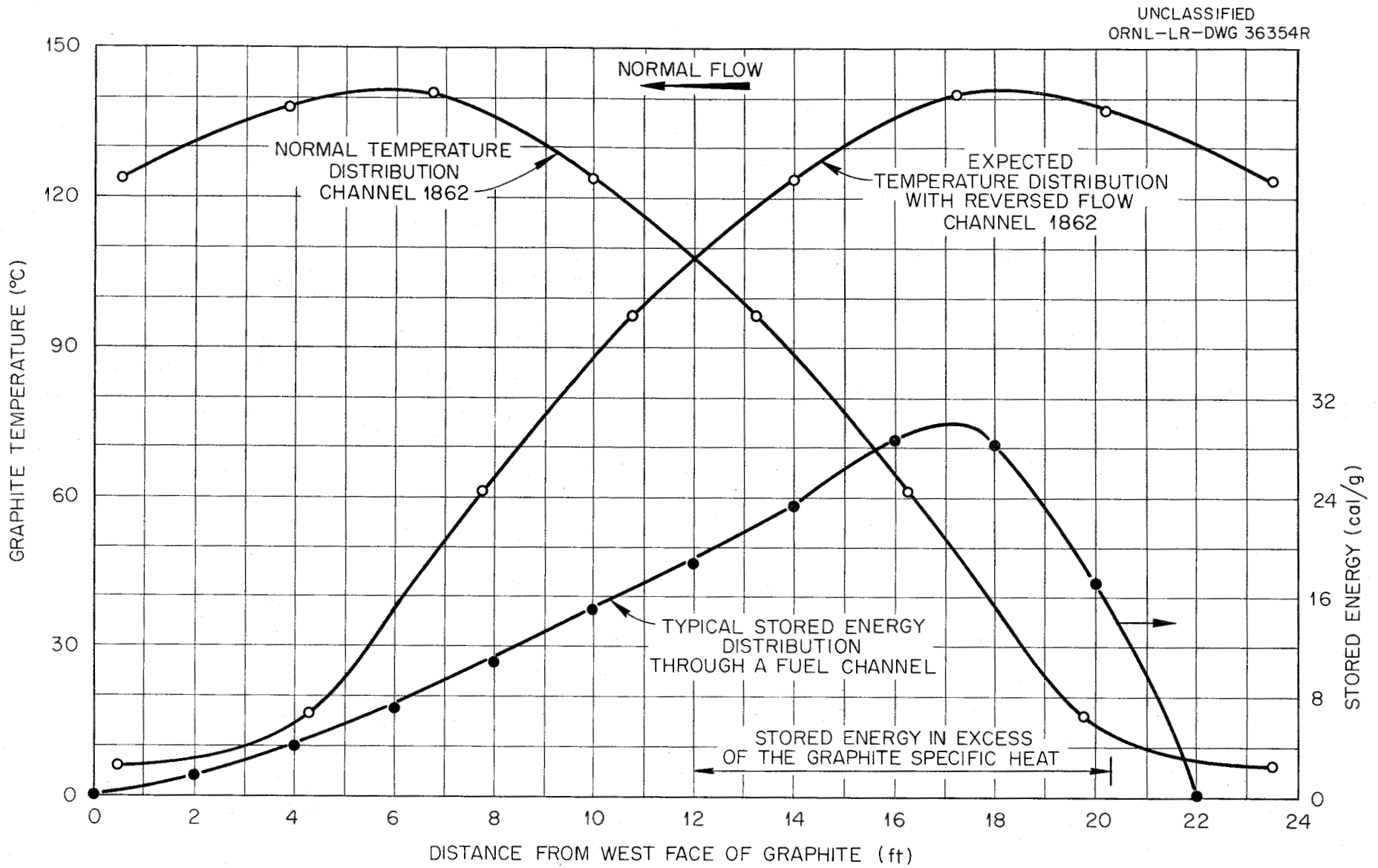


Fig. 4. Typical Axial Stored Energy and Temperature Distributions in OGR.

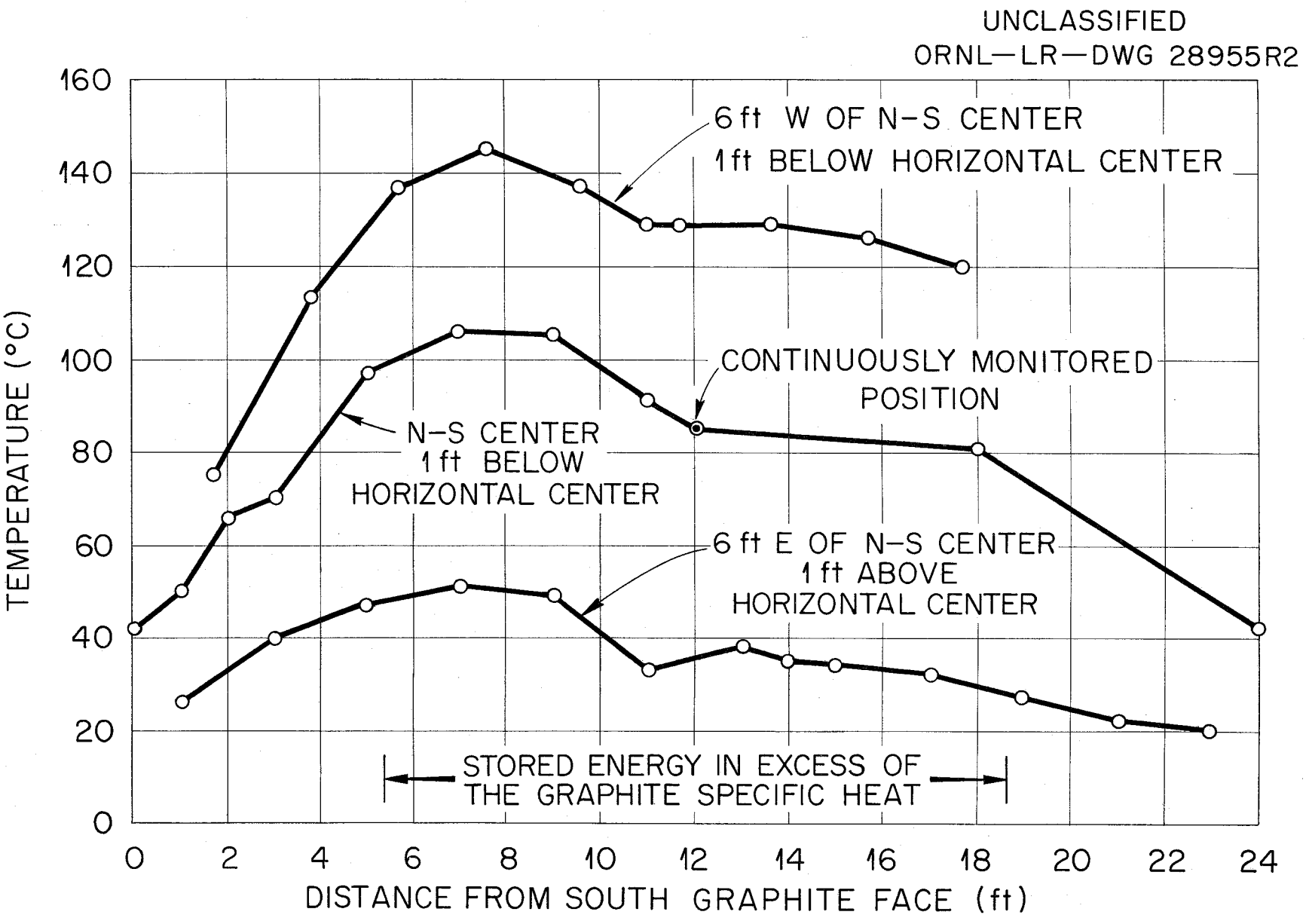


Fig. 5. Winter Temperatures in ORNL Graphite Reactor. Direction is East to West, Parallel to Slug Rows.

UNCLASSIFIED
ORNL-LR-DWG 36205

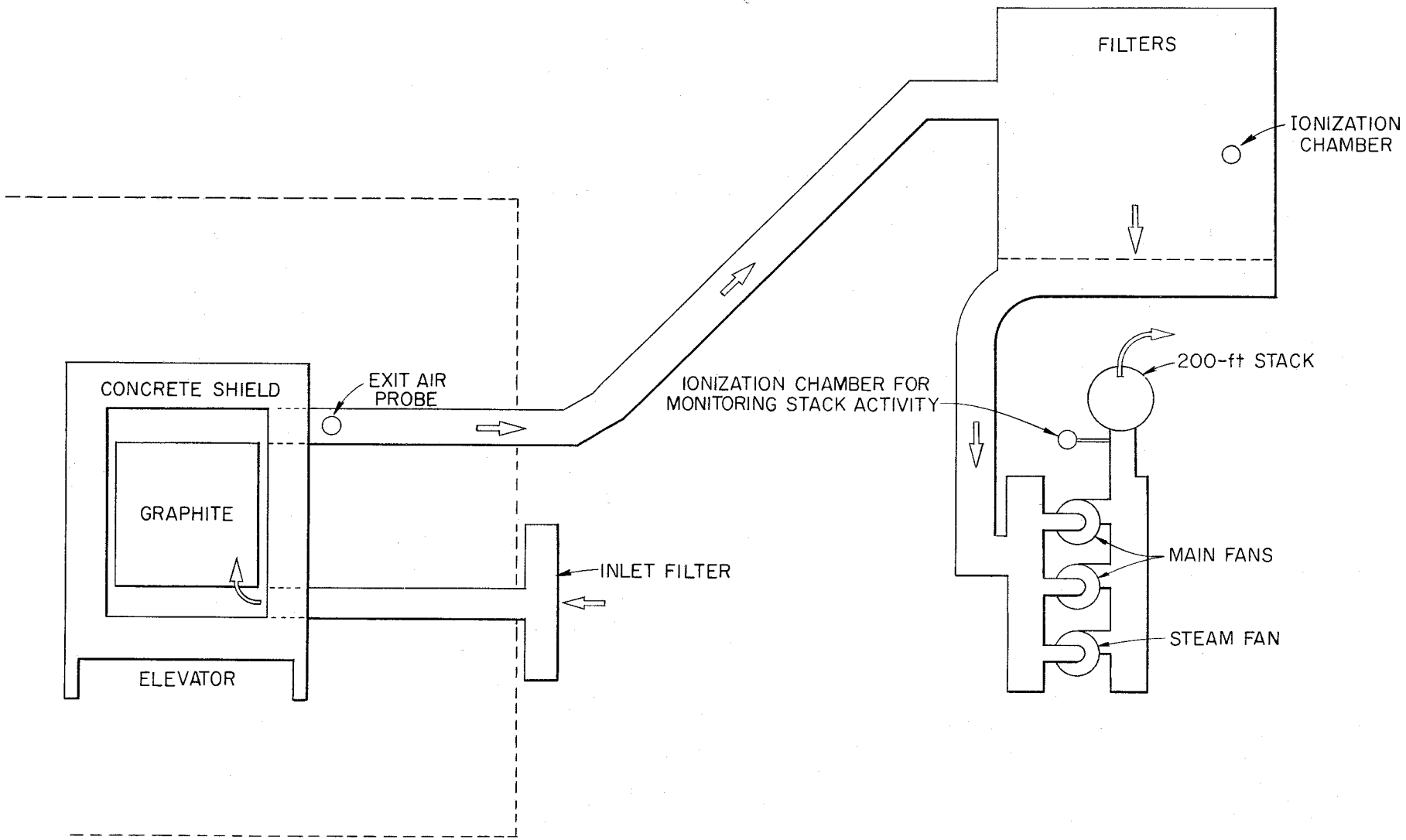


Fig. 6. Present OGR Coolant System.

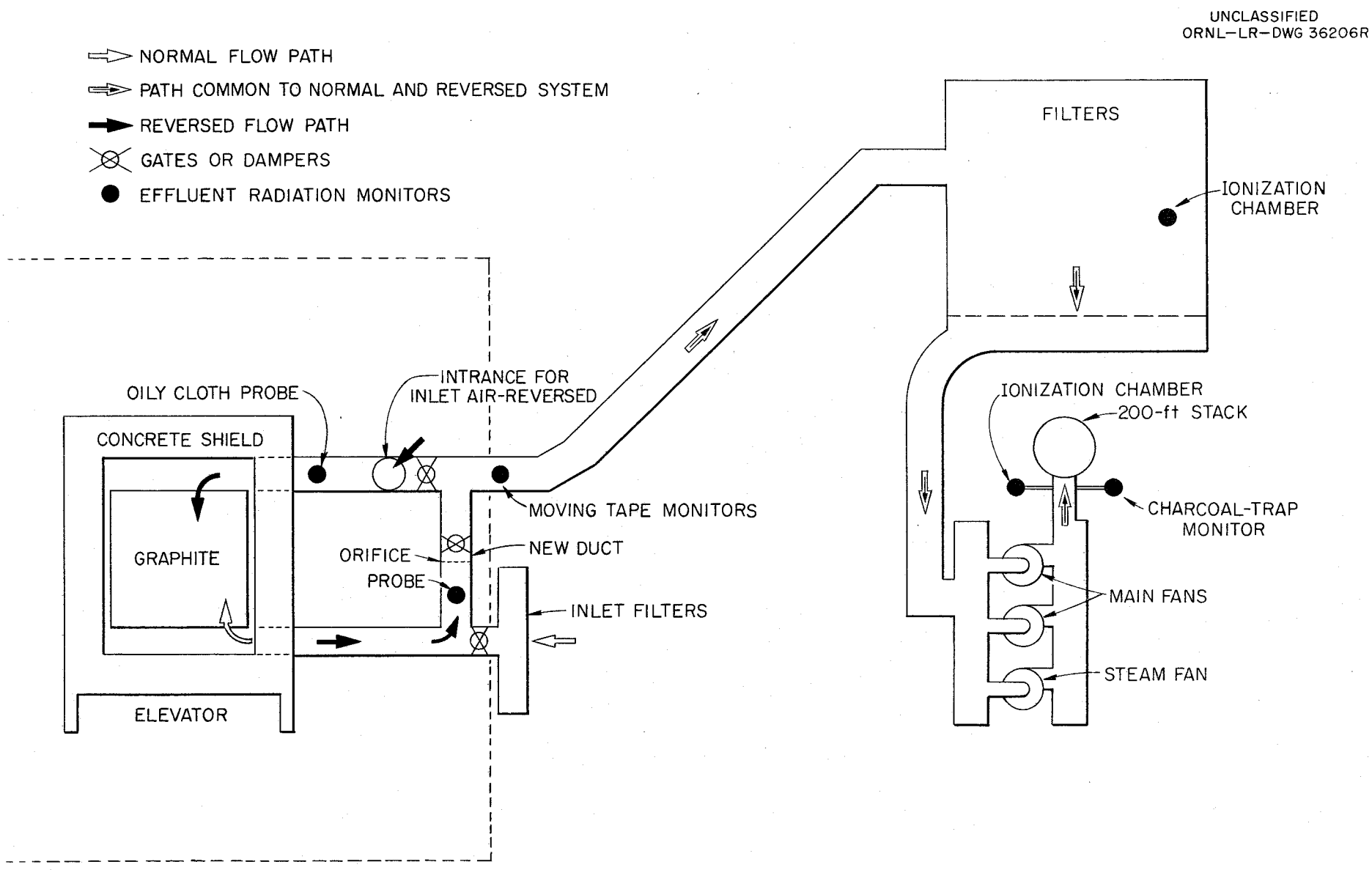


Fig. 7. Revised OGR Coolant System.

UNCLASSIFIED
ORNL-LR-DWG 36355

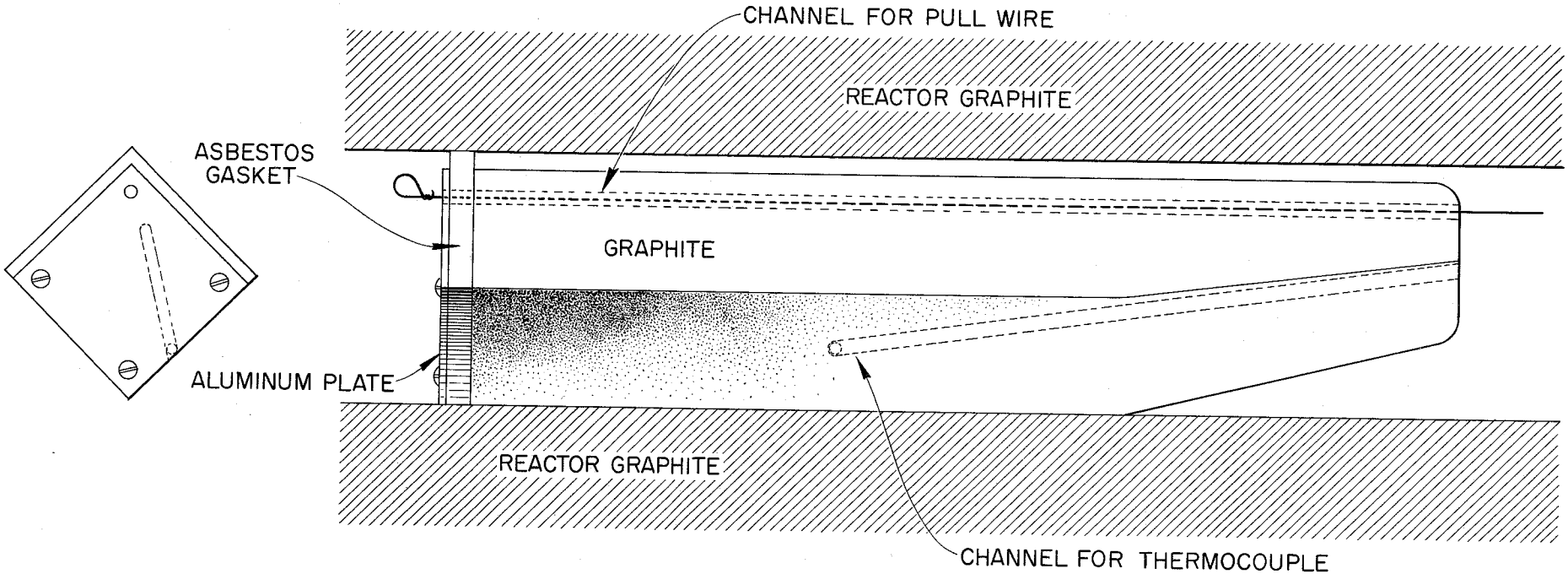


Fig. 8. Thermocouple Holder for OGR Graphite Temperature Measurements.

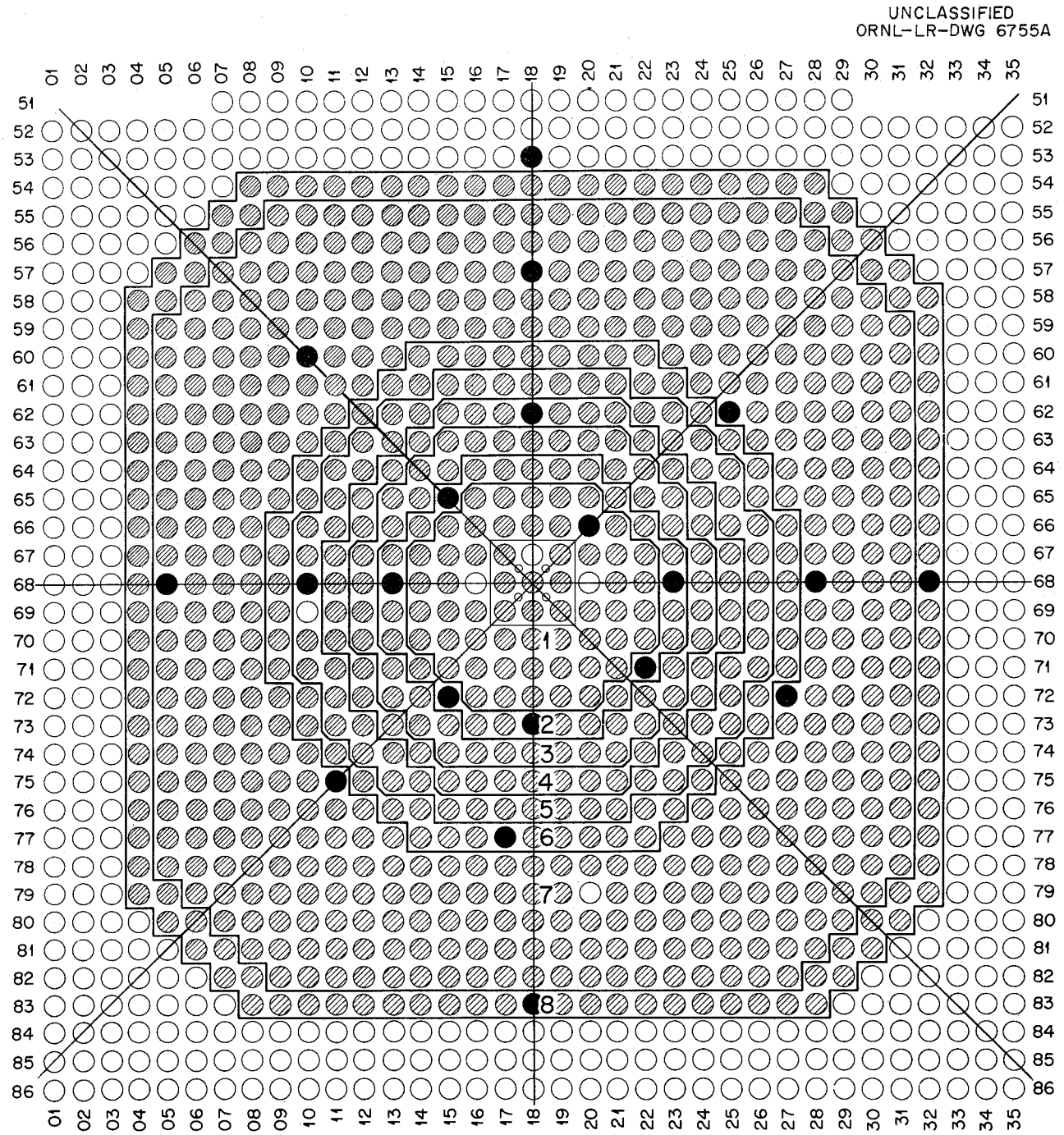


Fig. 9. OGR Loading Face Showing the Planned Radial Distribution of Thermocouples for Graphite Temperature Measurements.

UNCLASSIFIED
ORNL-LR-DWG 6755B

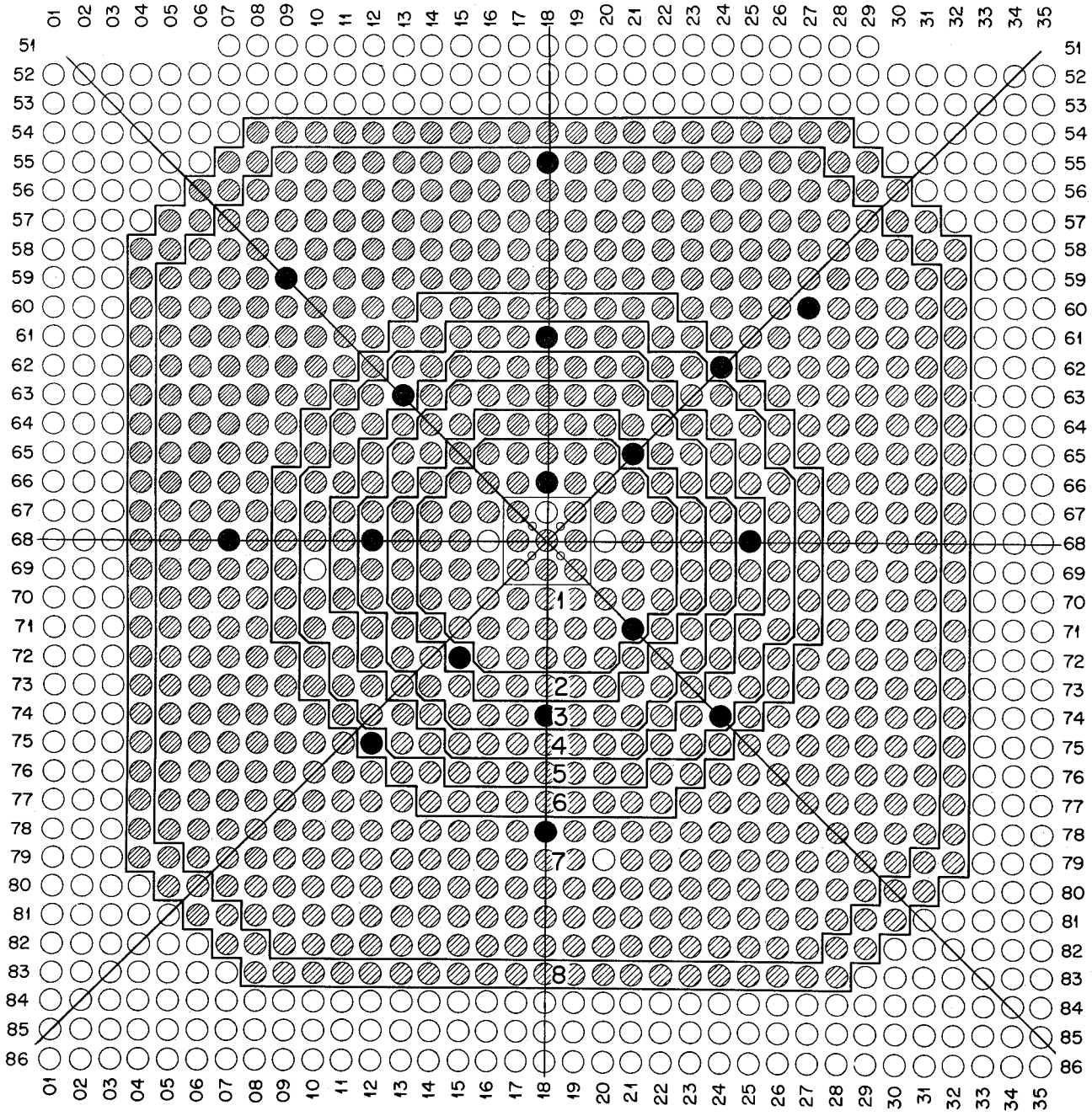


Fig. 10. OGR Loading Face Showing the Radial Distribution of Thermocouples for Fuel Element Temperature Monitoring.

UNCLASSIFIED
ORNL-LR-DWG 36356

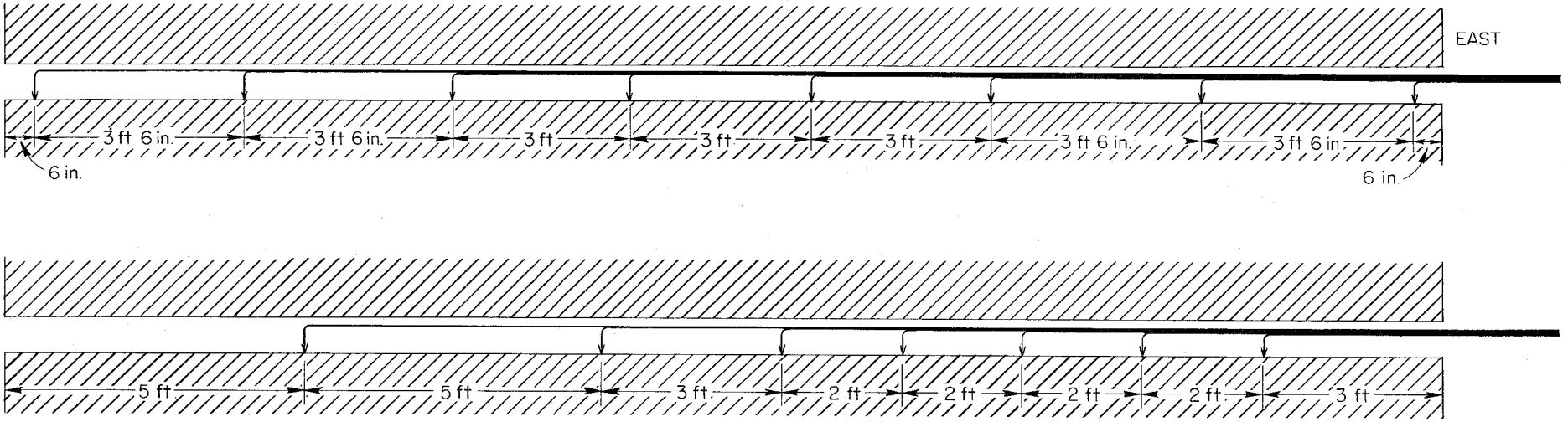


Fig. 11. Typical Axial Thermocouple Distributions for Graphite Temperature Measurements in Fuel Channels.

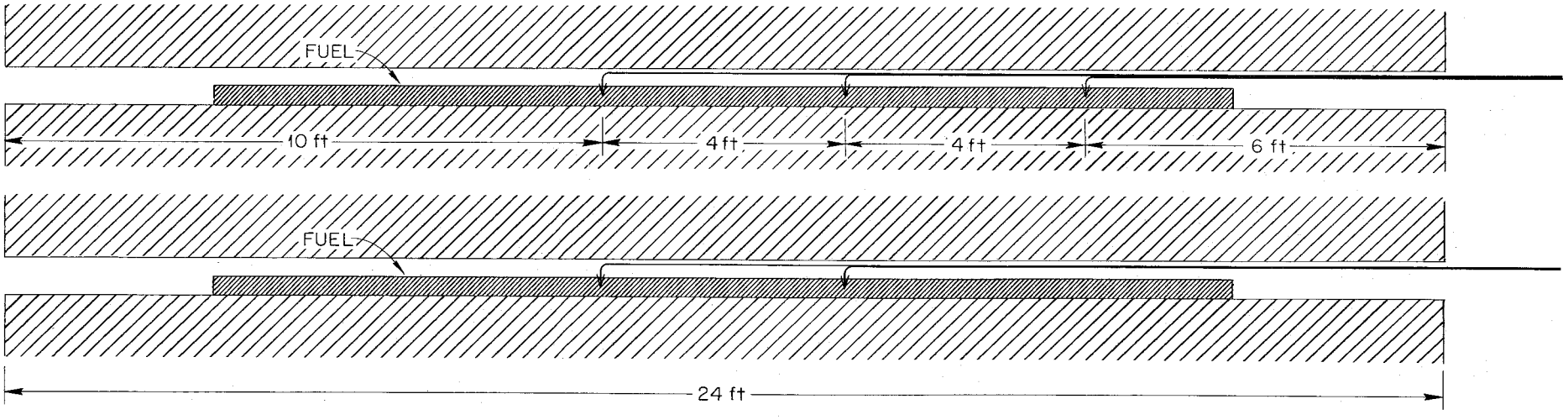


Fig. 12. Typical Axial Thermocouple Distributions for Fuel Element Temperature Measurements.

UNCLASSIFIED
ORNL-LR-DWG 36357

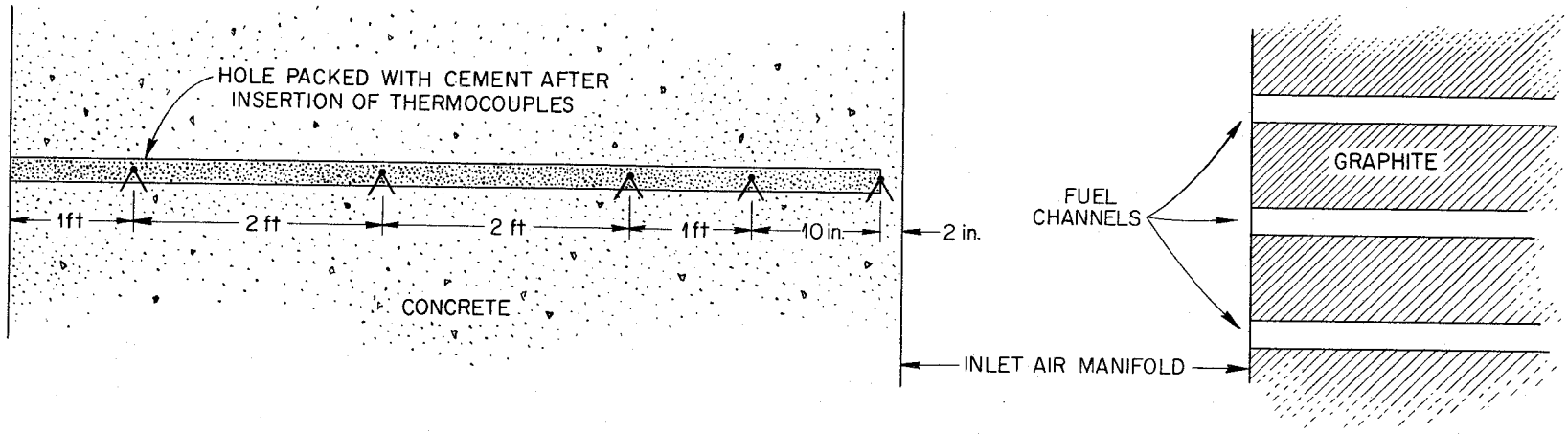


Fig. 13. Typical of Three Installations of Thermocouples for Measuring the Front Face Shield Temperatures.

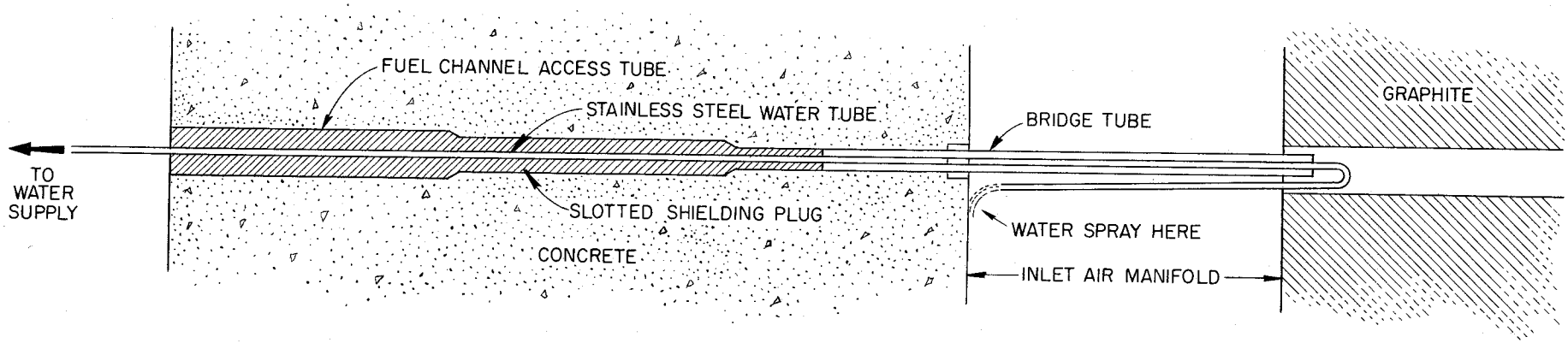


Fig. 14. Typical Water Spray Tube Installation for Protection of Front Face Shield from Thermal Damage.

UNCLASSIFIED
DWG 14587AR

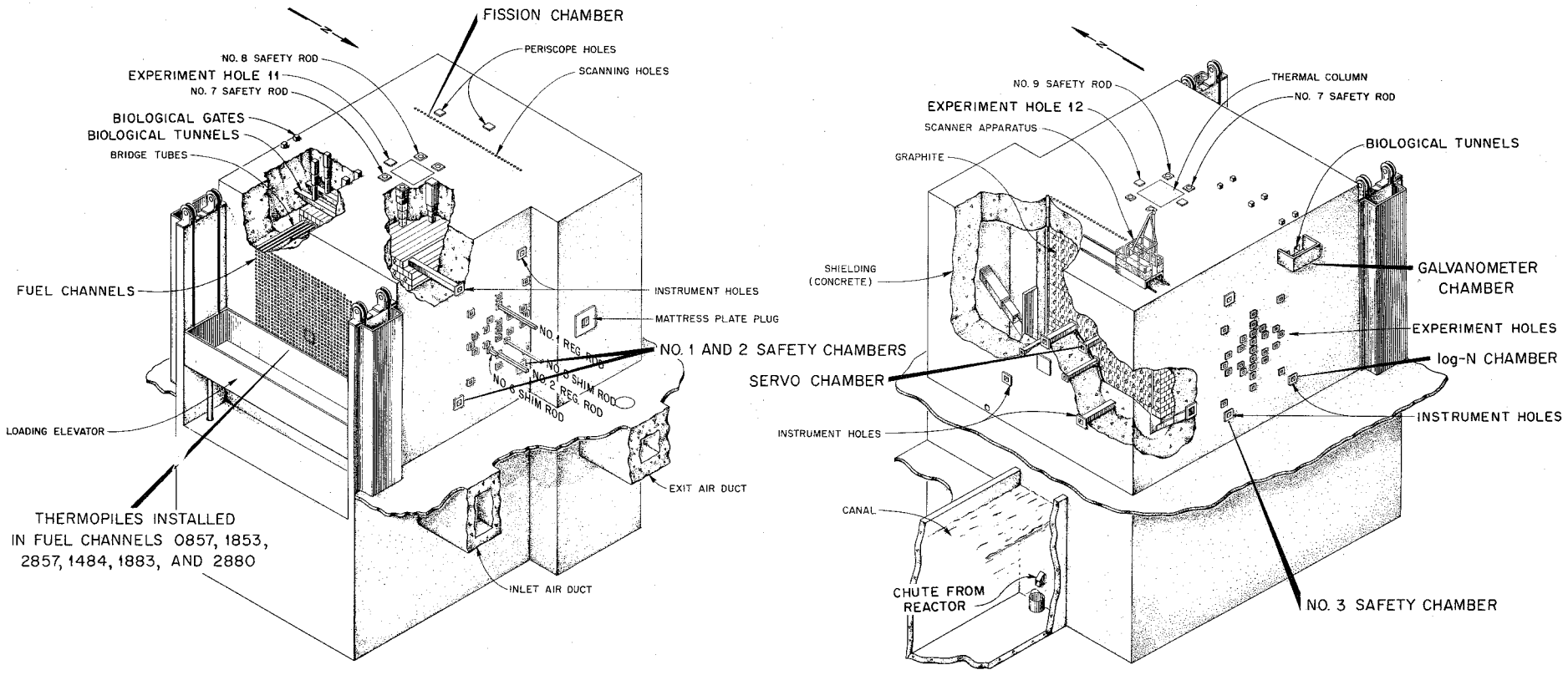


Fig. 15. Location of Neutron Sensing Instruments in OGR.

APPENDIX I

EVALUATION OF THE ORNL GRAPHITE REACTOR STORED ENERGY

M. C. Wittels[1]

Graphite measurements on the ORNL Graphite Reactor have been conducted by the Solid State and Operations Divisions of ORNL, National Bureau of Standards, and Brookhaven National Laboratory. At each of these laboratories different methods of measurement were employed so that independent analyses would be available. The samples were cored from the most severely damaged region of the graphite stack for the purpose of determining the most serious situation that exists. In addition, the measurements were designed to determine the validity of c_o spacing correlations in stored energy estimates which have been employed in evaluating the state of the reactor.

Measurements at ORNL

Radiation Calorimetry

The stored energy to 218°C was measured in core samples from fuel channel 1766, 6 ft 8 in. from the east face of the graphite stack. The measurements gave 26 cal/g with an estimated accuracy of ±10% in two separate runs. A radiation calorimeter was employed in these experiments, and a brief description of the apparatus and method is as follows.

The graphite specimen, of known mass and surface area, is suspended in an evacuated copper can by means of a thermocouple lead that is screwed into the center of the graphite sample. The copper calorimeter is placed in an oil bath at constant temperature (199°C), and the temperature-time

[1]Solid State Division, ORNL

relationship of the graphite is recorded as the stored energy is released. When the sample temperature comes to equilibrium with the oil bath temperature, the calorimeter is removed and cooled, and the same process is repeated a second time on the same sample. The stored energy, Q, is calculated by numerical integration of the expression

$$Q = A\varepsilon\sigma \int_{t_i}^{t_f} (T_1^{\,4} - T_2^{\,4})\, dt,$$

where

A = surface area of sample in cm^2,

ε = sample emissivity,

σ = Stefan-Boltzmann constant,

t_i = initial time,

t_f = final time,

T_1 = absolute temperature for energy release run,

T_2 = absolute temperature for second run.

The temperature-time curves for one of these experiments are shown in Fig. 4.9. It is noted that energy is released at as low as 120°C and that at approximately 165°C the slopes of the two curves begin to sharply differ. It should be noted that at temperatures near or above 120°C, stored energy is released under time periods as short as minutes. It is therefore reasonable to assume that substantial annealing would take place at these low temperatures for more extended but still reasonable times of, say, one day. Evidently, 165°C is the approximate temperature wherein the specific heat of the damaged graphite becomes "negative" and a spontaneous energy release is propagated. The temperature rise from 165°C to 218°C was completed in less

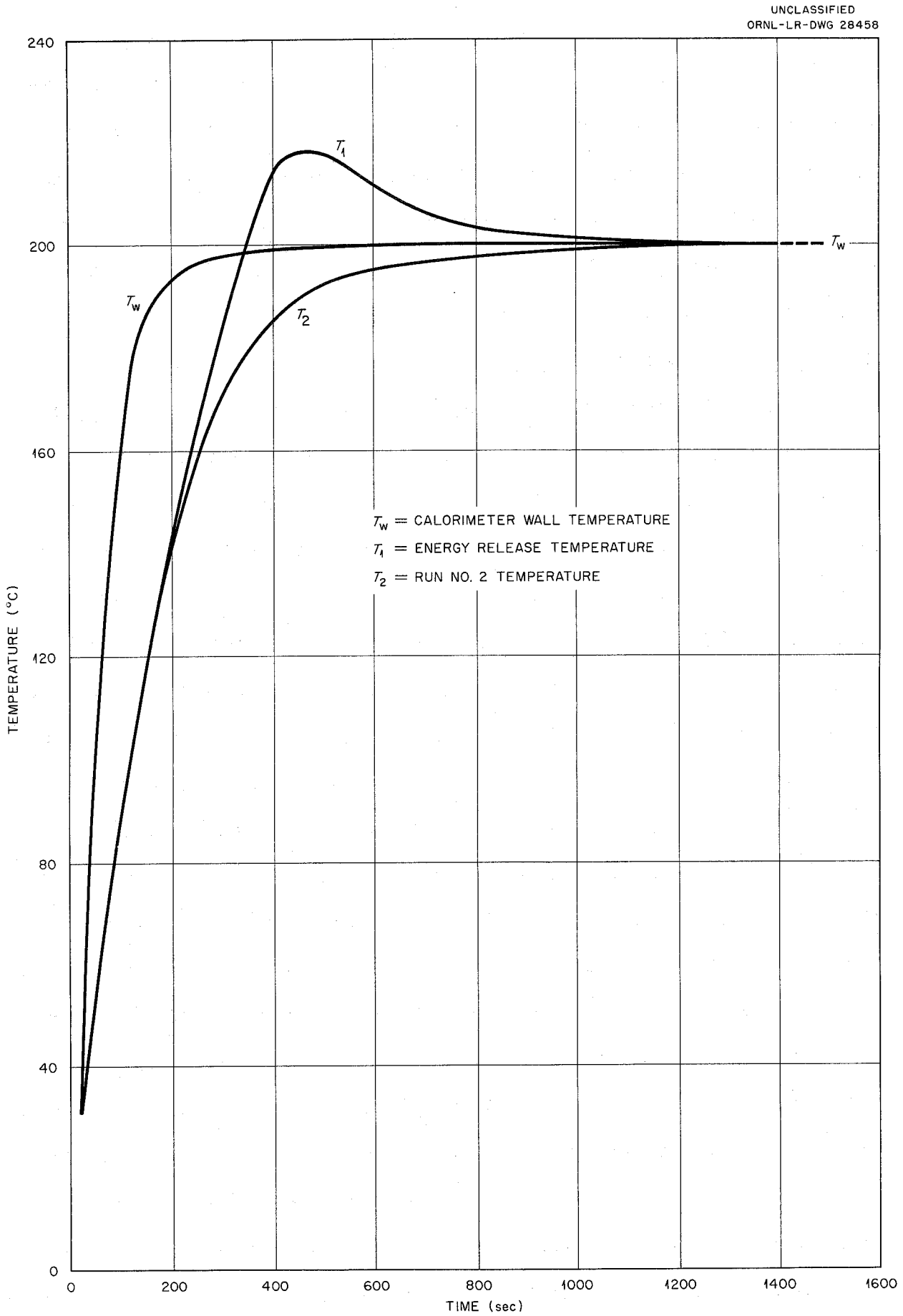


Fig. 4.9. Radiation Calorimetry Measurement of ORNL Reactor Graphite.

than 3 min, and more than 95% of the entire stored energy release was completed in approximately 10 min. These observations are interesting in considering the effects of spontaneous releases under actual reactor conditions where heat transfer by conduction and convection would contribute to even further restrict temperature excursions. A graphical plot of the numerical integration is shown in Fig. 4.10.

X-Ray Lattice Parameter Measurements

Temperature traverses (see later) of the graphite stack by the Operations Division have revealed the north half of the reactor to be operated at the lowest temperatures. For this reason, fuel channel scrapings have been taken from channels 3268 and 3368 as well as from 2168 and 2768, contained in the middle horizontal plane of the stack, so that a series of c_o spacings could be measured in the cool, high flux portions of the reactor. These scrapings were taken from the upper portion of the fuel channel, above the fuel elements (Fig. 4.11). Shown in Fig. 4.12 is the east loading face with numbered channels. Channels are located by combining numbers, first horizontally and then vertically, to give a four-digit designation. Thus, No. 1868 is the center channel of the reactor. The X-ray measurements have been completed, and estimates of the stored energy content to 250°C are shown in Fig. 4.13. It is noted that the c_o spacings show the largest expansion (and therefore the most damage) at 6 to 7 ft from the cold east face of the graphite stack. The somewhat broadened curve for fuel channel 2768 is in large part due to a more extensive fuel loading in that channel, while the peculiar configuration for fuel channel 3268 in the 16-ft area is due to a control rod position. Fuel channel 3268 lies in the extreme northern row of channels; therefore it is apparent that the peak position for damage in this channel follows the flux distribution. According to the X-ray estimates,

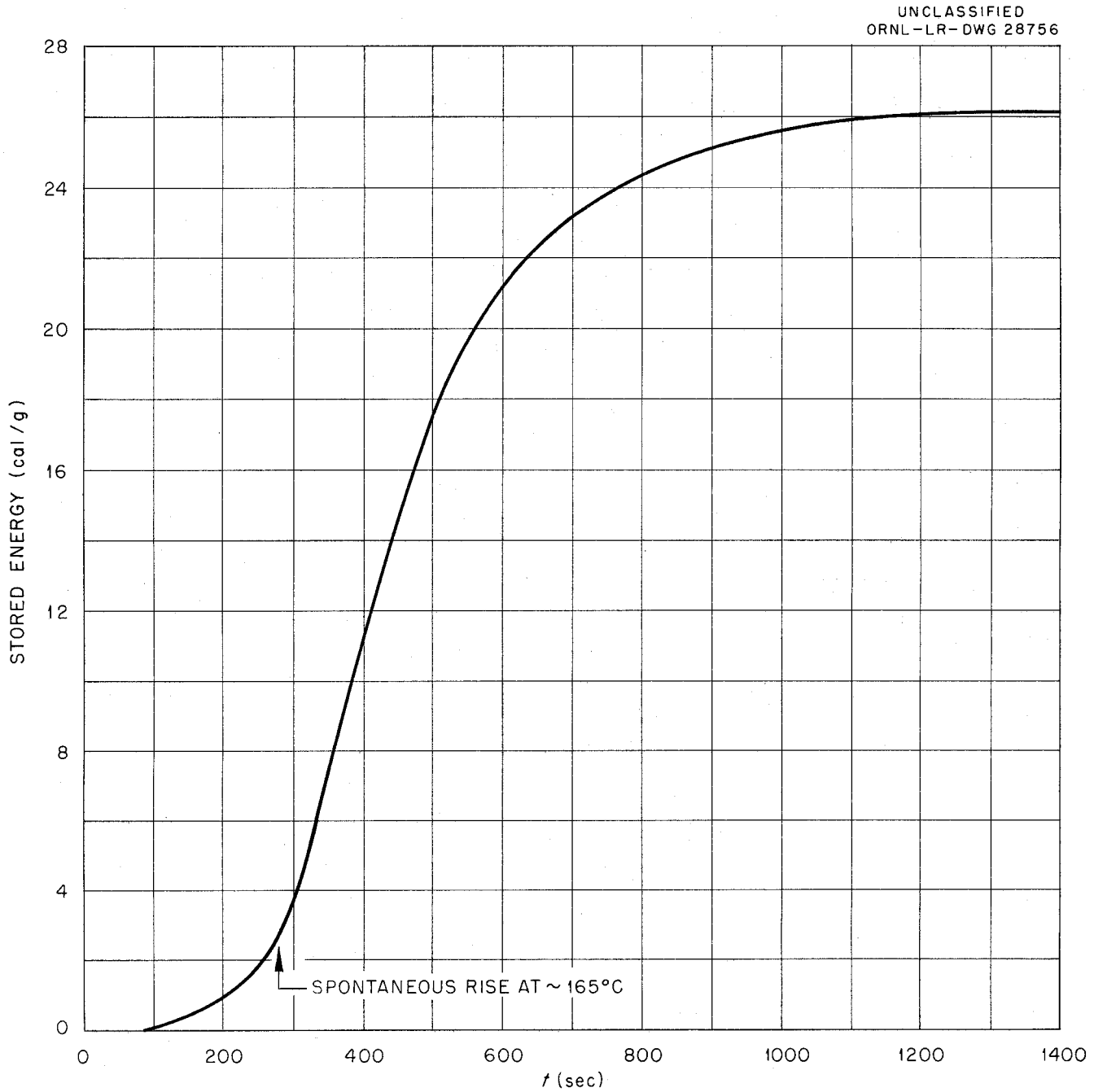


Fig. 4.10. Stored Energy Release of ORNL Reactor Graphite as a Function of Time.

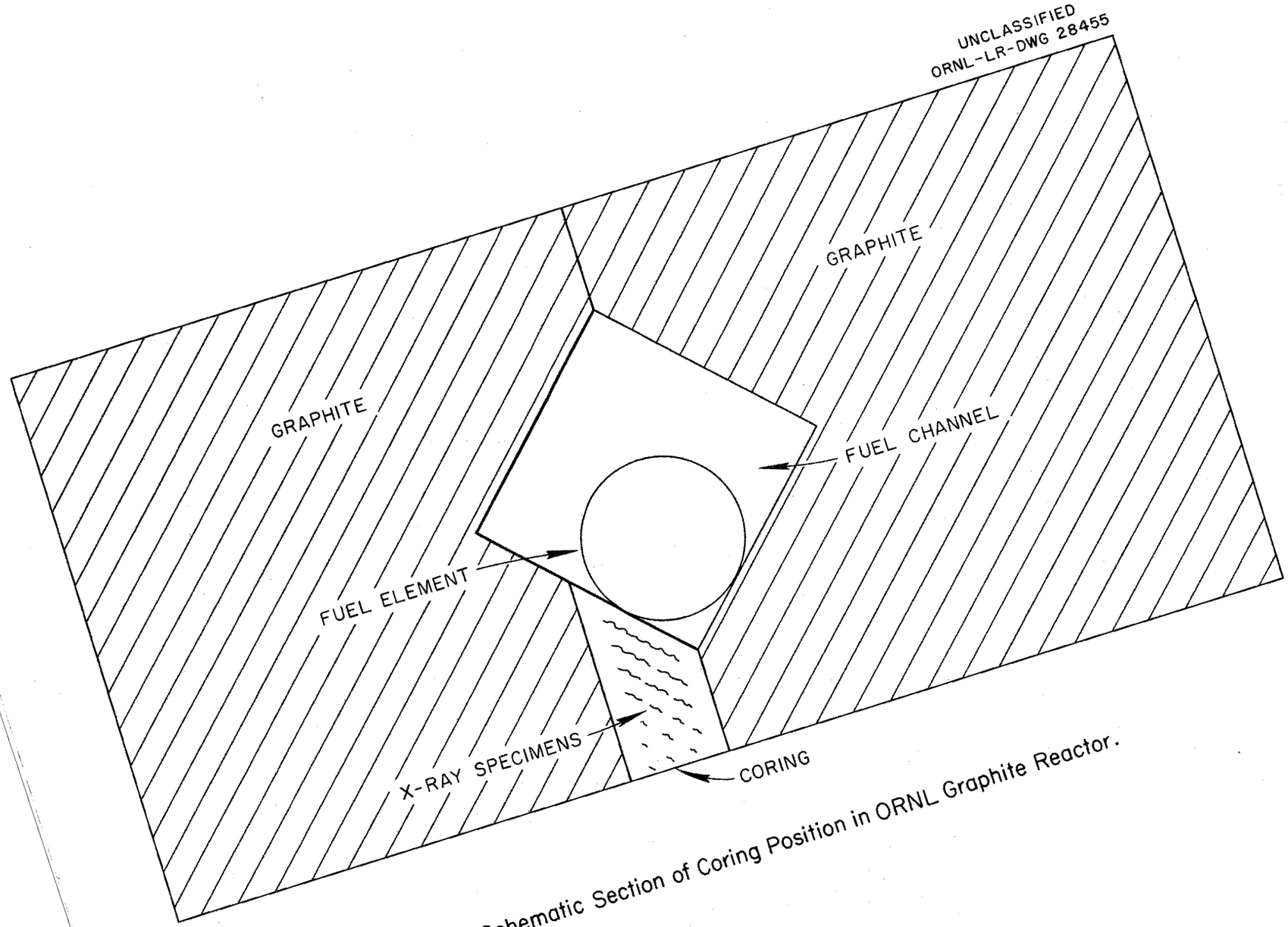
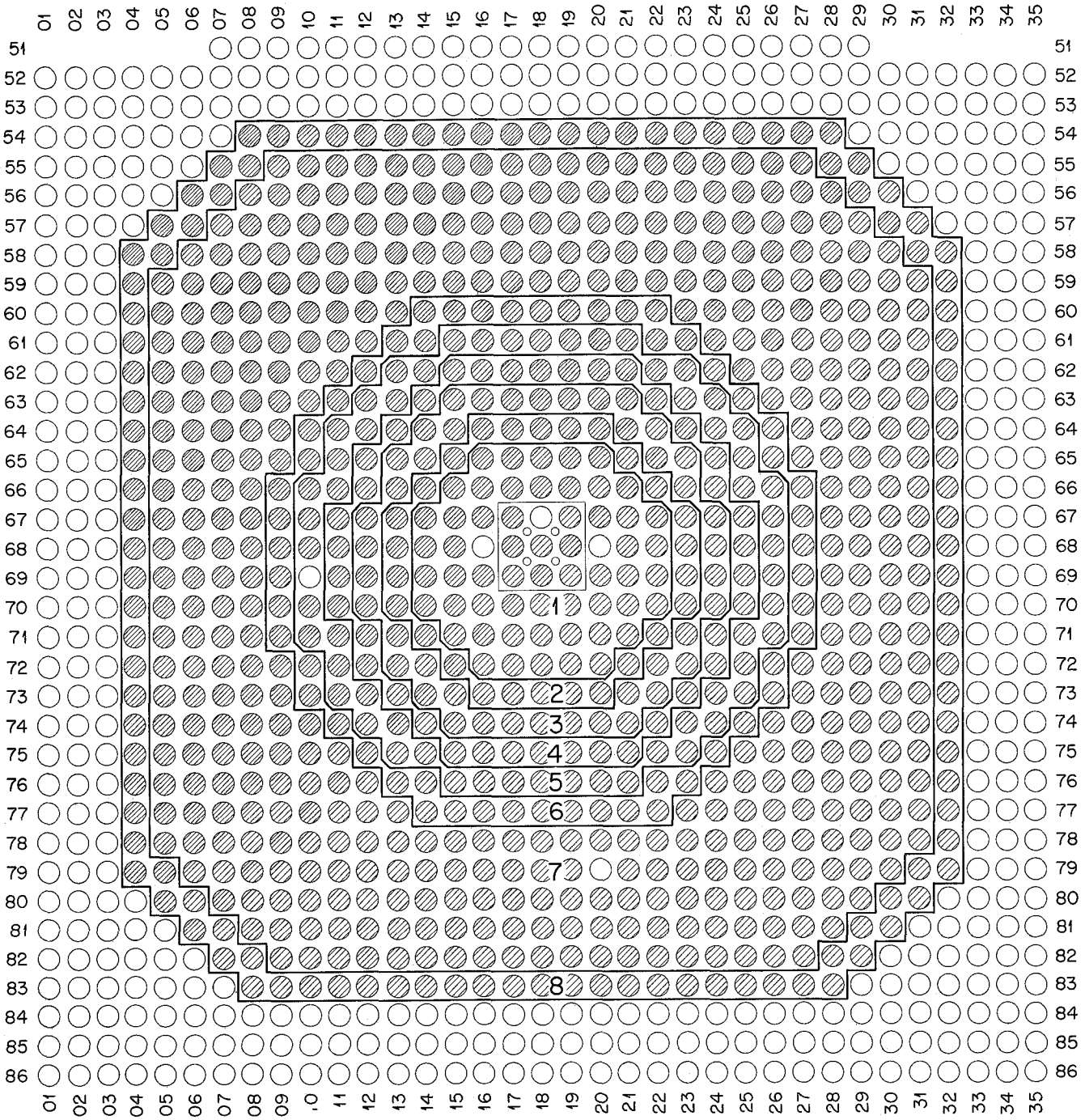


Fig. 4.11. Schematic Section of Coring Position in ORNL Graphite Reactor.

UNCLASSIFIED
ORNL-LR-DWG 6755



ZONE	NUMBER CHANNELS	NUMBER SLUGS
1	60	41
2	26	45
3	42	51
4	38	51
5	58	54
6	54	54
7	438	54
8	114	47

Fig. 4.12. OGR Fuel Loading Pattern.

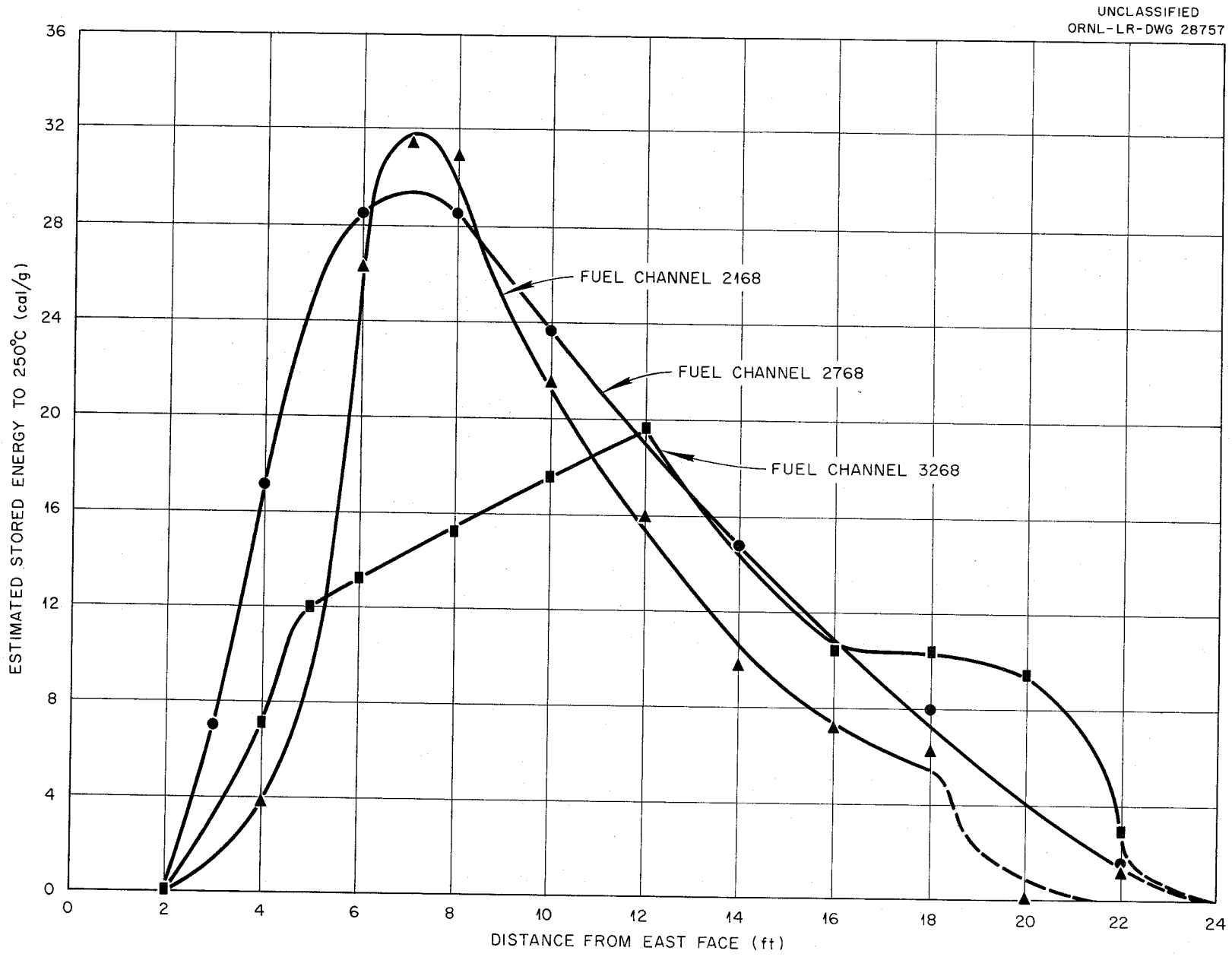


Fig. 4.13. Estimated Stored Energy to 250°C in ORNL Fuel Channel Graphite.

however, there is no stored energy at this extremity of the reactor which could be released spontaneously. Scrapings from channel 3368, 8 in. into the reflector region of the stack, showed no significant damage within the limits of detection.

The region of the reactor containing the highest damage was therefore established by X-ray measurements, and solid cores were obtained from this area by the Operations Division through the use of a remote coring device which could cut upward into the fuel channel (Fig. 4.11). One such coring (fuel channel 1766, 6 ft 8 in. from the east face of the graphite stack) was examined for variations in damage as a function of distance from the fuel channel surface. Powder specimens for c_o spacing measurements were obtained by means of jeweler saw cuttings from spaced intervals along the core (Fig. 4.11). It is interesting to note that the damage near the fuel channel surface is considerably higher than the bulk damage through the core (Fig. 4.14). While it is apparent that the temperature on the surface is lower as a result of the air stream contact, the temperature gradient into the graphite is not known and a direct relation between the damage and temperature is not quantitatively established. For this reason a precise conversion of c_o spacings to stored energy is not feasible, as was the case for the scraping, where a more uniform temperature condition exists. In addition to this probable temperature dependence, there exists the likelihood of an enhanced near-surface damage due to the higher concentration of displacements resulting from grazing angle collisions and the neutron mean free path consideration near the core surface. Samples later sent to the National Bureau of Standards were sectioned in an attempt to take these variations into closer account.

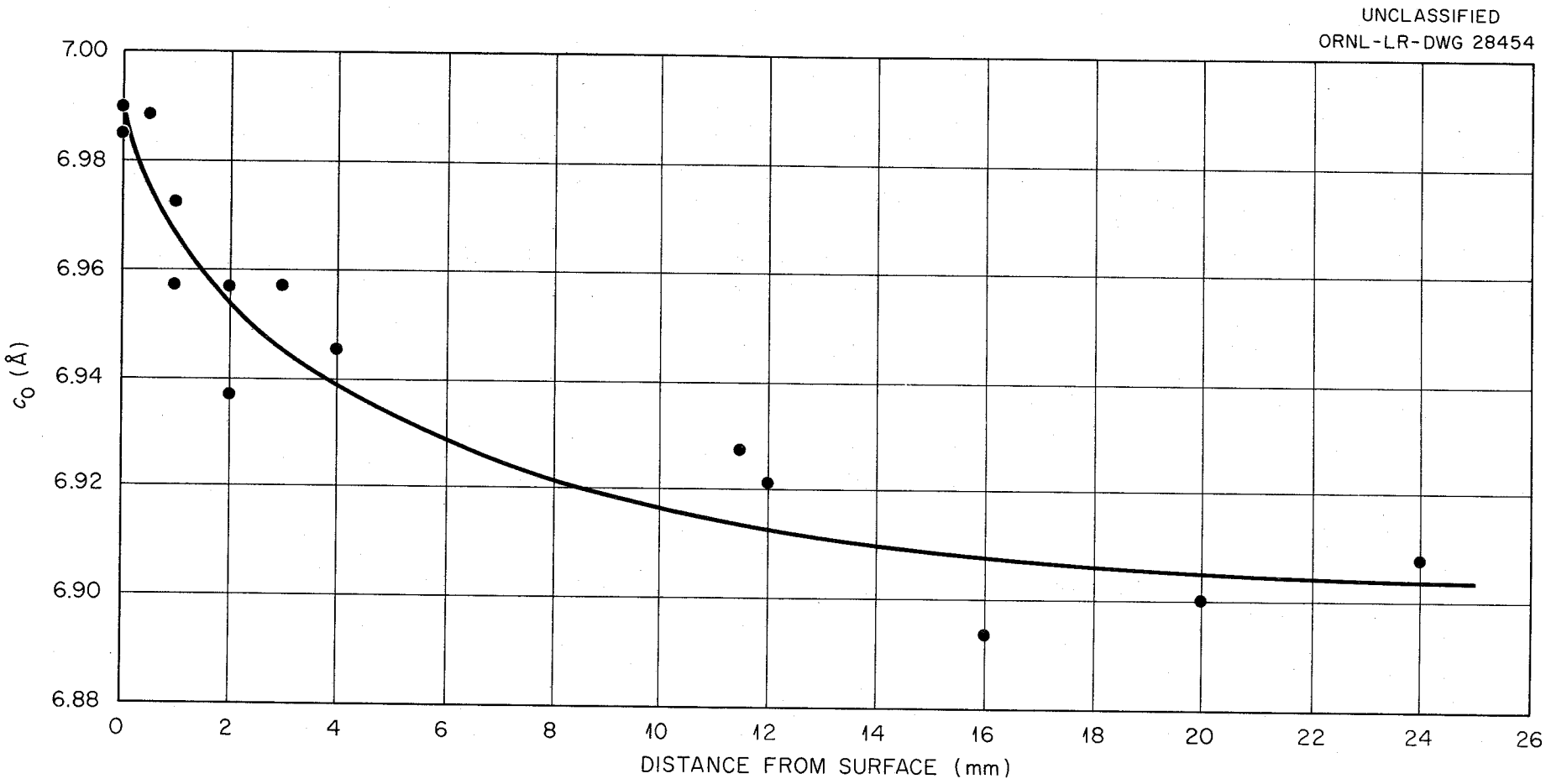


Fig. 4.14. Variation in c_0 Spacing in ORNL Reactor Graphite as a Function of Distance from the Fuel Element.

Graphite Reactor Temperatures

Since the growth of stored energy in irradiated graphite is so strongly dependent on temperature, a knowledge of the reactor graphite temperature is vital to a proper assessment of the state of the reactor. Temperature measurements are being compiled from all parts of the ORNL Graphite Reactor so that a complete temperature distribution will be obtained; to date some important data are available.

For orientation purposes, it should be recalled that the reactor cooling direction is east to west parallel to the slug rows. Traverses were made transverse to this direction (N-S) at the center of the reactor, 6 ft east and west of the center, and all three traverses were within 1 ft of a horizontal center plane (Fig. 4.15).

The measurements were made during January and February 1958, when the inlet air temperatures varied between -7° and +8°C. All the temperatures were normalized with respect to an 85°C value from the continuously monitored position in the center traverse. The inlet air temperature is recorded at +8°C when the continuously monitored position is at 85°C. Higher temperatures in the south half of the reactor are due mainly to the location of the air inlet at the lower right corner of the east face and, secondly, to a complex flow pattern in the south half of the reactor.

It should be emphasized that the temperatures shown in Fig. 4.15 depict the coldest operating conditions in the reactor and that during the summer months these values are as much as 50°C higher for several positions. Temperature records reveal that on hot summer days with inlet air at 32°C the temperature in the continuously monitored center position reaches as high as 138°C; 53°C higher than that shown for a February recording in Fig. 4.15. This rise, which is considerably larger than the maximum seasonal variation

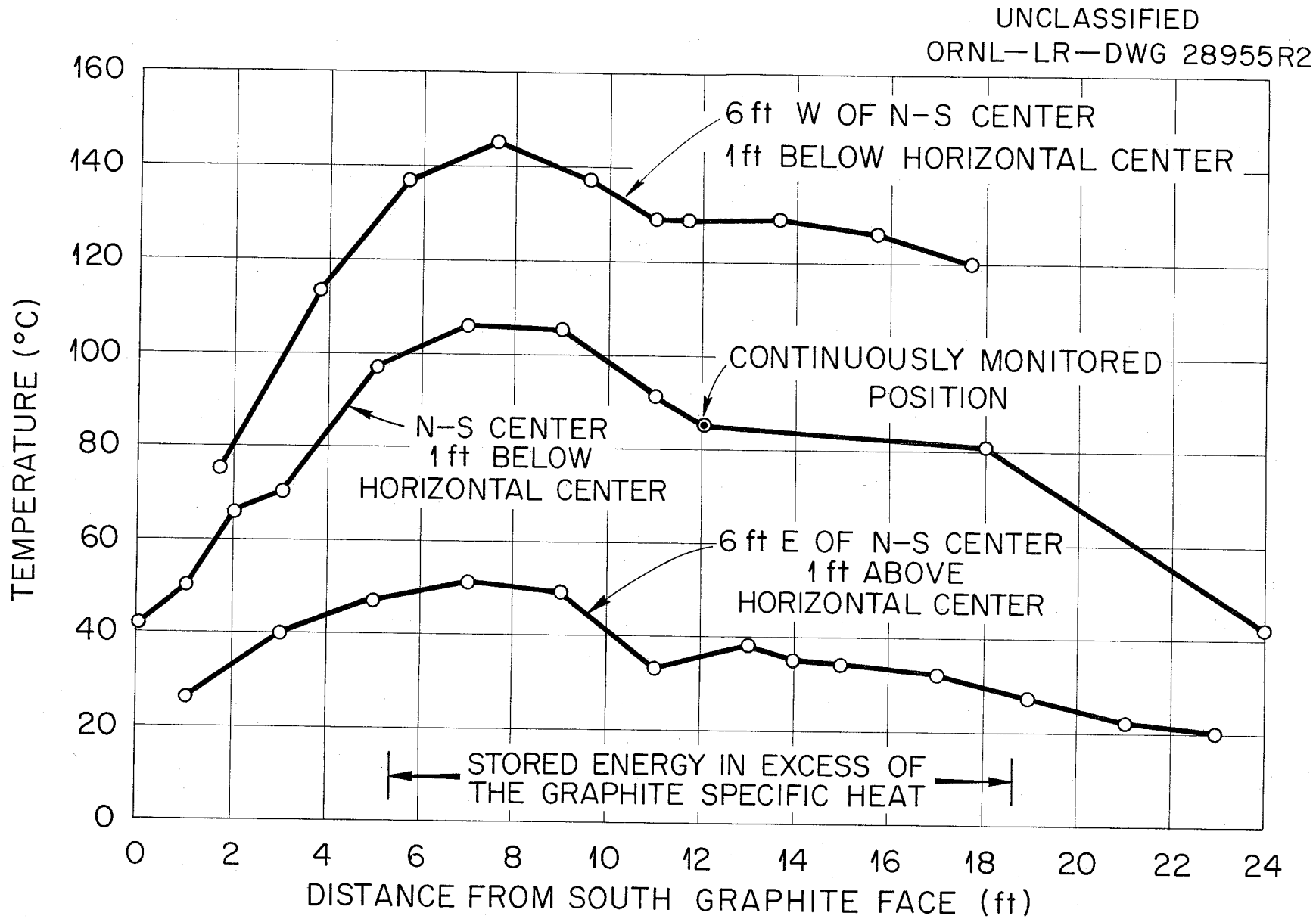


Fig. 4.15. Winter Temperatures in ORNL Graphite Reactor. Direction is East to West, Parallel to Slug Rows.

of inlet air temperature, is due to the reactor power being maintained constant year round (3.4 Mw), and this level is based on the maximum allowable fuel element temperatures during the hottest summer days. The deviation of these temperatures from normal seasonal variations is due to the change in density of air with seasonal temperature changes (as well as specific heat). Also, the volume and flow pattern of coolant air is altered because of the characteristic operational limits of the fans. For several weeks during the summer months, then, a considerable portion of the west half of the graphite lattice is operated at temperatures in excess of 100°C, and the growth of stored energy in that region is largely diminished. It is observed that the very lowest temperatures are recorded on the north and south edges of the graphite stack; but this effect is compensated, insofar as stored energy is concerned, by the diminished flux (cosine) distribution on the edges of the stack.

National Bureau of Standards--
Heat of Combustion Measurements

Total heats of combustion were measured by the thermochemical section of the National Bureau of Standards. These heats include the normal heat of unirradiated graphite plus any increase in internal energy induced by irradiation. For purposes of establishing a standard heat of combustion for the ORNL Graphite Reactor graphite, corings from the edge of the stringer removed in 1953 were given to the Bureau. The heat of combustion of this graphite was found to be within the limits of error of the determinations for other standard graphites from Hanford and Brookhaven.

As mentioned previously, cores were sectioned to separate the more highly damaged surfaces from bulk of the specimens for statistical estimates. Since the Bureau requested samples with masses of 4 to 5 g (although each

determination consumes only 1 g), it was impossible to secure a large enough sample from each core that contained only highly damaged surface material. The "bottom portions" (farthest portion of core from fuel element, Fig. 4.11) were more clearly representative of the average condition of fuel channel graphite in the most heavily damaged region of the reactor; however, it should not be overlooked that the "top portions" (nearest to fuel element, Fig. 4.11) of the cores contained only a small percentage (less than 20%) of highly damaged surface graphite.

Three kinds of specimens were submitted to the National Bureau of Standards: unannealed, annealed to 254°C, and annealed to 400°C. Although the annealing schedule is not directly related to a hypothetical reactor annealing schedule, it furnishes some useful information and will be described briefly.

1. Sectioned samples were drilled and tapped to accommodate a thermocouple in a snug fit at the approximate geometric center of the sample.

2. The samples were contained in stoppered tubes in air and were placed in an oil bath at constant temperature (150°C) for 16 hr.

3. The sample temperatures were monitored and did not exceed this temperature during this interval.

4. In successive stages, the oil bath temperature was increased 10°C, at a rate of 0.5°C/min, and maintained at these levels for a period of 20 min before proceeding. These steps included stops at 160, 170, 180, and 190°C.

5. The sample temperatures were continuously monitored, and at no time did they exceed the bath temperatures.

6. When the sample temperatures reached 200°C, the tubes were removed from the oil bath and placed in a salt bath at constant temperature (254°C). The rise in sample temperatures was monitored and the samples were removed

when they reached 254°C, in approximately 3 min.

7. Those samples annealed to 400°C were taken from the 254°C bath and inserted in a second salt bath at 400°C and were similarly removed.

8. At no time during these annealing runs were any temperature excursions observed.

The results of the National Bureau of Standards measurements are shown in Table 4.2. Included in the tabulation are the results from fuel channel 1666 which were obtained earlier. Only a few observations concerning these data are necessary at this point since the results are essentially self-explanatory. Only in the sections from fuel channel 1368 were wide variations found over the length of the core which would confirm the previously described X-ray observations. It was probably fortuitous that 1368-2 was sampled by the National Bureau of Standards in a manner which resulted in a higher proportion of highly damaged surface material than the other samples. It is also of interest to note that the stored energy above 400°C did not vary by more than 2 cal/g in three measurements and that approximately 100 cal/g remained above 400°C.

Powder samples for c_o spacing measurements were annealed in the same manner as the cores sent to National Bureau of Standards, and the results are shown in Fig. 4.16. It is seen that annealing to 254°C reduces c_o by only 0.02 Å, just 10% of the preannealed expansion. With an actual reactor annealing it would be unfeasible to determine exactly what temperatures had been reached for known times in all parts of the reactor, and therefore c_o-stored energy correlations following such a reactor anneal would have no meaning. Further, no data are available on the behavior of c_o under irradiation following such a partial annealing. Consequently, it is clear that X-ray measurements could not be confidently used to predict stored

TABLE 4.2. NATIONAL BUREAU OF STANDARDS--HEAT OF COMBUSTION MEASUREMENTS

Standard: Unannealed, unirradiated ORNL graphite
Heat of Combustion: 7839 cal/g

NBS Sample No.	Fuel Channel	Sample Portion	Annealing (°C)	Heat of Combustion (cal/g)	Stored Energy (cal/g) Above Anneal	Stored Energy (cal/g) Below Anneal
		At 6 ft 2 in. from East Face of Graphite Stack				
1	1368	Bottom	254	7957	118	Av = 26
2	1368	Top	254	7977	138	
10	1368	Top and bottom	Unannealed	7993	154	
		At 6 ft 8 in. from East Face of Graphite Stack				
3	1466	Top and bottom	254	7970	131	Av = 31
8	1466	Top and bottom	254	7962	123	
4	1466	Top	Unannealed	7997	158	
7	1566	Bottom	400	7938	99	Av = 59
9	1566	Bottom	400	7940	101	
5	1566	Top	Unannealed	7998	159	
6	1566	Top and bottom	Unannealed	7998	159	
C-2	1666	Top and bottom	401	7940	101	68
C	1666	Top and bottom	Unannealed	8008	169	

Average* stored energy of five unannealed samples = 160 cal/g
Average* stored energy to 254°C of four samples = 32 cal/g
Average* stored energy to 400°C of three samples = 60 cal/g

*These averages were obtained by taking the combined averages of the residual stored energies for specimens annealed at designated temperatures, and subtracting from the combined average for all the unannealed specimens.

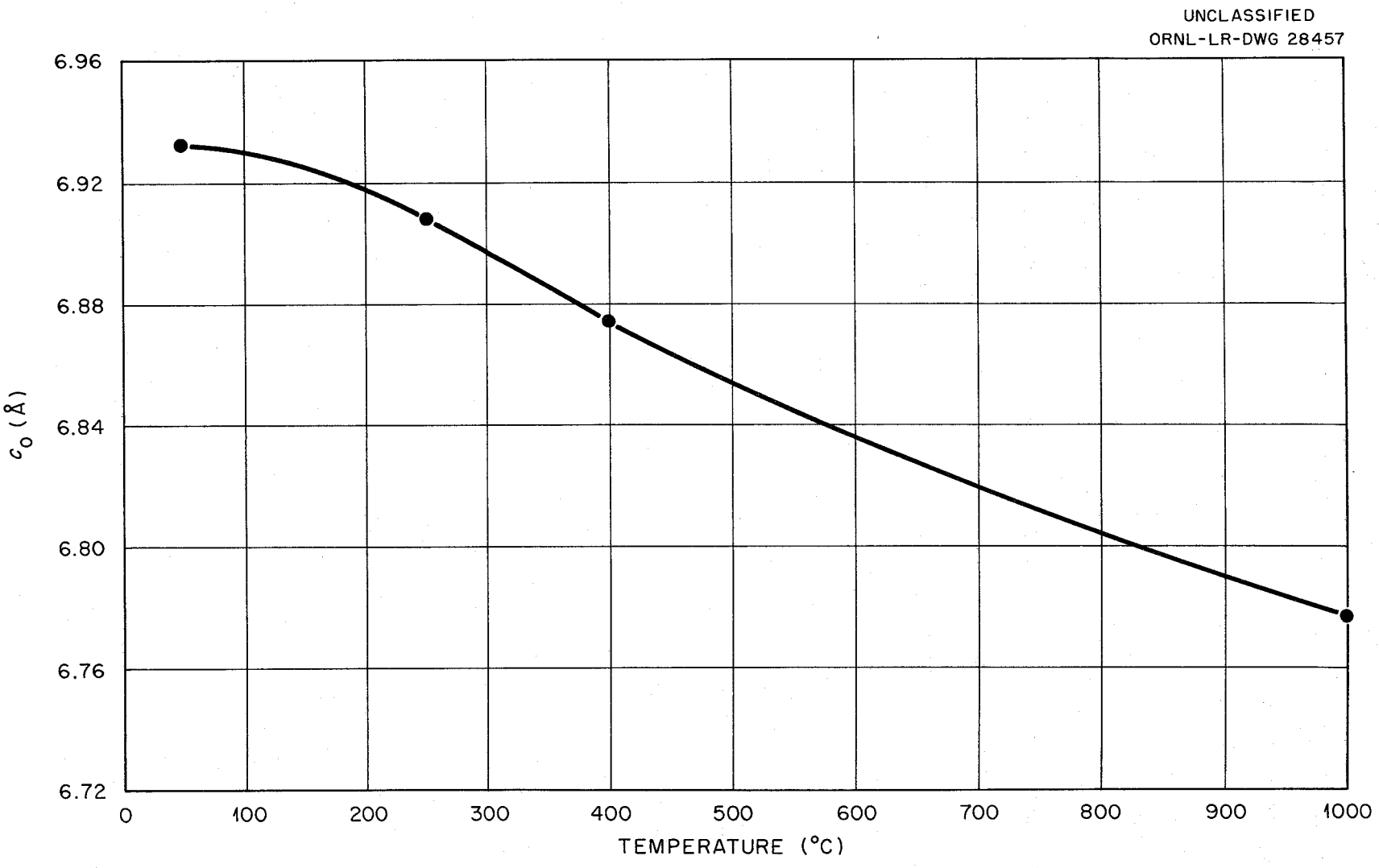


Fig. 4.16. Annealing Behavior of c_0 Spacing in ORNL Reactor Graphite.

energy contents in an annealed reactor, and it is perhaps for this reason that the British were so skeptical of the ORNL data, since the U. K. reactors had already been annealed.

Brookhaven National Laboratory--Stored Energy and Adiabatic Rise Measurements

Measurements at Brookhaven National Laboratory were conducted in a double-vacuum calorimeter in which the heating conditions are predetermined. A constant heat input is supplied to the system, which has negligible heat losses, so that the furnace temperature rises at the slow rate of 2°C/min, and near-adiabatic conditions are achieved. The stored energy and adiabatic rise were measured in a 7.5-g core removed from fuel channel 1468, 6 ft 2 in. from the east face of the graphite stack. The stored energy to 270°C was found to be 33 cal/g and the adiabatic rise (105°C) occurred over the temperature span 165 to 270°C (Fig. 4.17). It was also estimated that the stored energy between 400 and 800°C was about 40 cal/g.

Discussion

From these data it is evident that the ORNL Graphite Reactor is reasonably safe with respect to stored energy content. The favorable agreement of the various kinds of measurements indicates that the recent stored energy estimates from X-ray data are substantially correct and that the growth of stored energy since 1953 is reasonable. Since this growth of the low temperature peak is only about 2 cal g^{-1} $year^{-1}$ at the present time, no feeling of emergency should exist; and it is suggested that this margin of safety be considered in any decision to anneal the reactor. The proposed new fuel loading of the reactor is of primary significance in any consideration for annealing the reactor and should therefore considerably influence any decision as to whether, when, and how the reactor should be annealed.

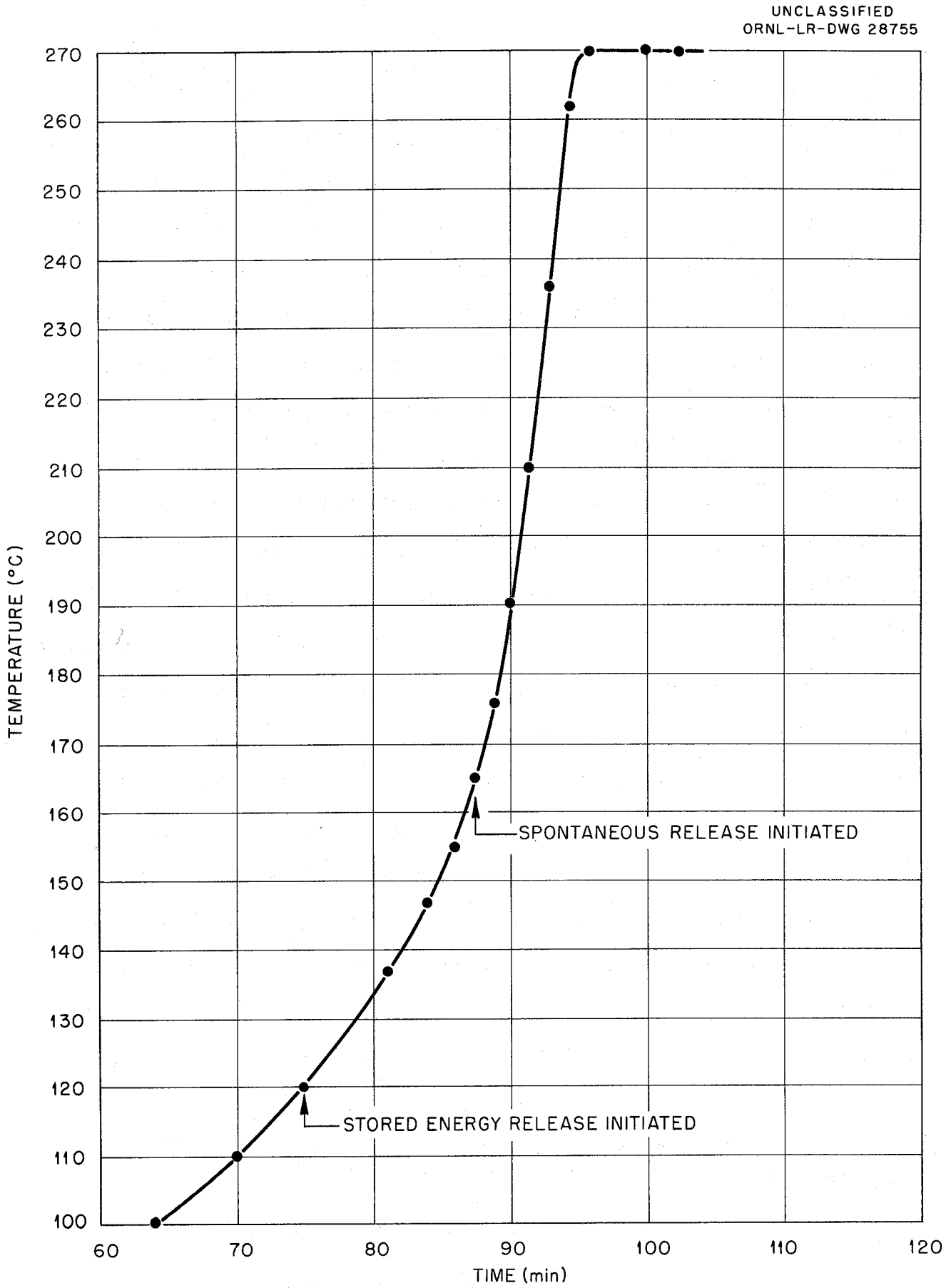


Fig. 4.17. Adiabatic Rise Measurement of ORNL Reactor Graphite.

Of somewhat more academic interest is the "reconstructed" release spectrum which is fitted from all of the experimental data (Fig. 4.18). First, it should be noted that the amount of energy above the specific heat curve is small (about 3 ca/g). In addition, the stored energy between 250 and 400°C is considerably higher than that usually exhibited in spectra of graphite irradiated at 30°C. In fact, this "reconstructed" spectrum has a shape that is similar to that for graphites from reactors that have been annealed to 250 or 300°C, leaving the higher temperature energy to grow still further. This is not to imply that the ORNL Graphite Reactor has been annealed in the past but simply to indicate that this reactor has been favored by higher temperature operation, especially during the summer months. We could not help but conclude that even at temperatures as low as 30 to 60°C the growth of stored energy is retarded by only small increases in temperature. Examination of extensive Hanford data in this temperature range reveals that the total stored energy is reduced by about 1% for each 1°C rise in temperature for irradiations similar to that in the ORNL Graphite Reactor.

This analysis can further be seen by comparing the ORNL data with data from Hanford,[2] Harwell,[3] and Windscale[4]. Figure 4.19 shows a comparison of ORNL data with Hanford data (11B) for a 30°C irradiation. The significant part of this comparison is the small amount of energy above the specific heat curve and the greater amount of stored energy above 350°C in the ORNL sample. This can also be seen in Fig. 4.20, which shows the stored energy

---

[2] W. K. Woods, L. P. Bupp, and J. F. Fletcher, Proc. Internl. Conf. Peaceful Uses Atomic Energy, Geneva, 1955 7, 455 (1956).

[3] G. H. Kinchin, Proc. Internl. Conf. Peaceful Uses Atomic Energy, Geneva, 1955 7, 472 (1956).

[4] R. W. Attree, R. L. Cushing, and J. Pieroni, CRC-706 (May 1957).

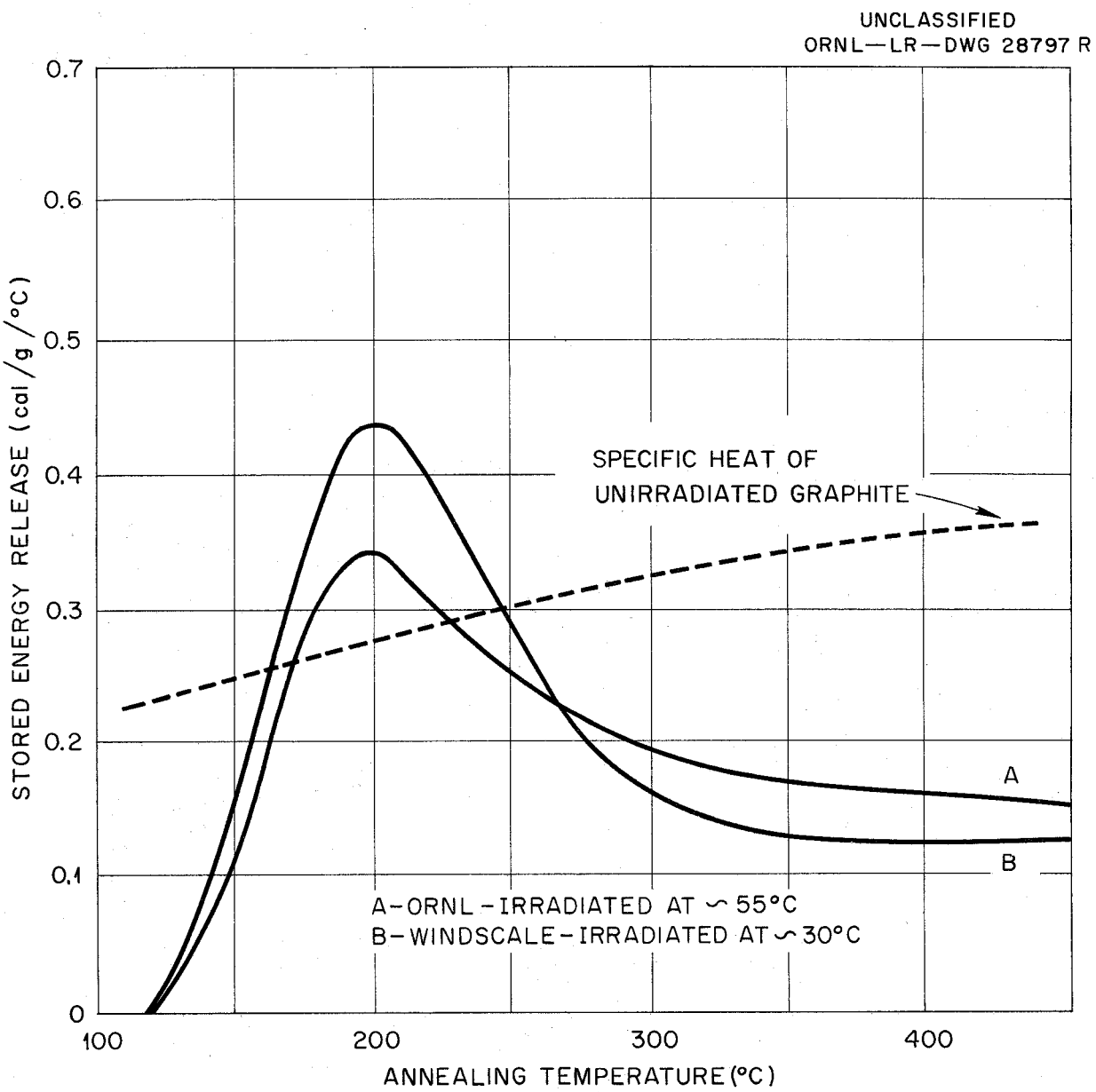


Fig. 4.18. Comparison of Stored Energy Release Spectra in ORNL and Windscale Graphite.

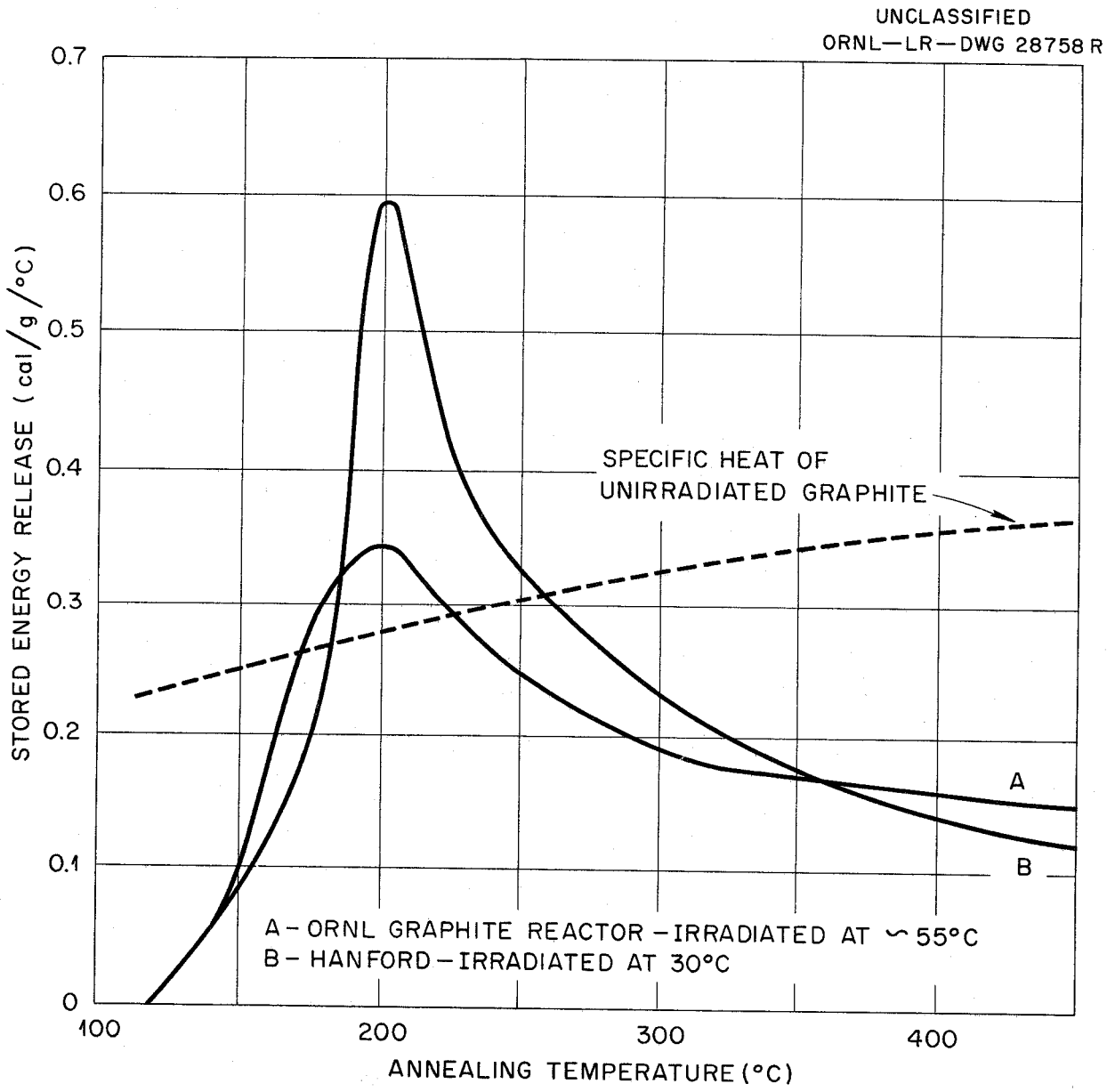


Fig. 4.19. Comparison of Stored Energy Release Spectra in ORNL and Hanford Graphite.

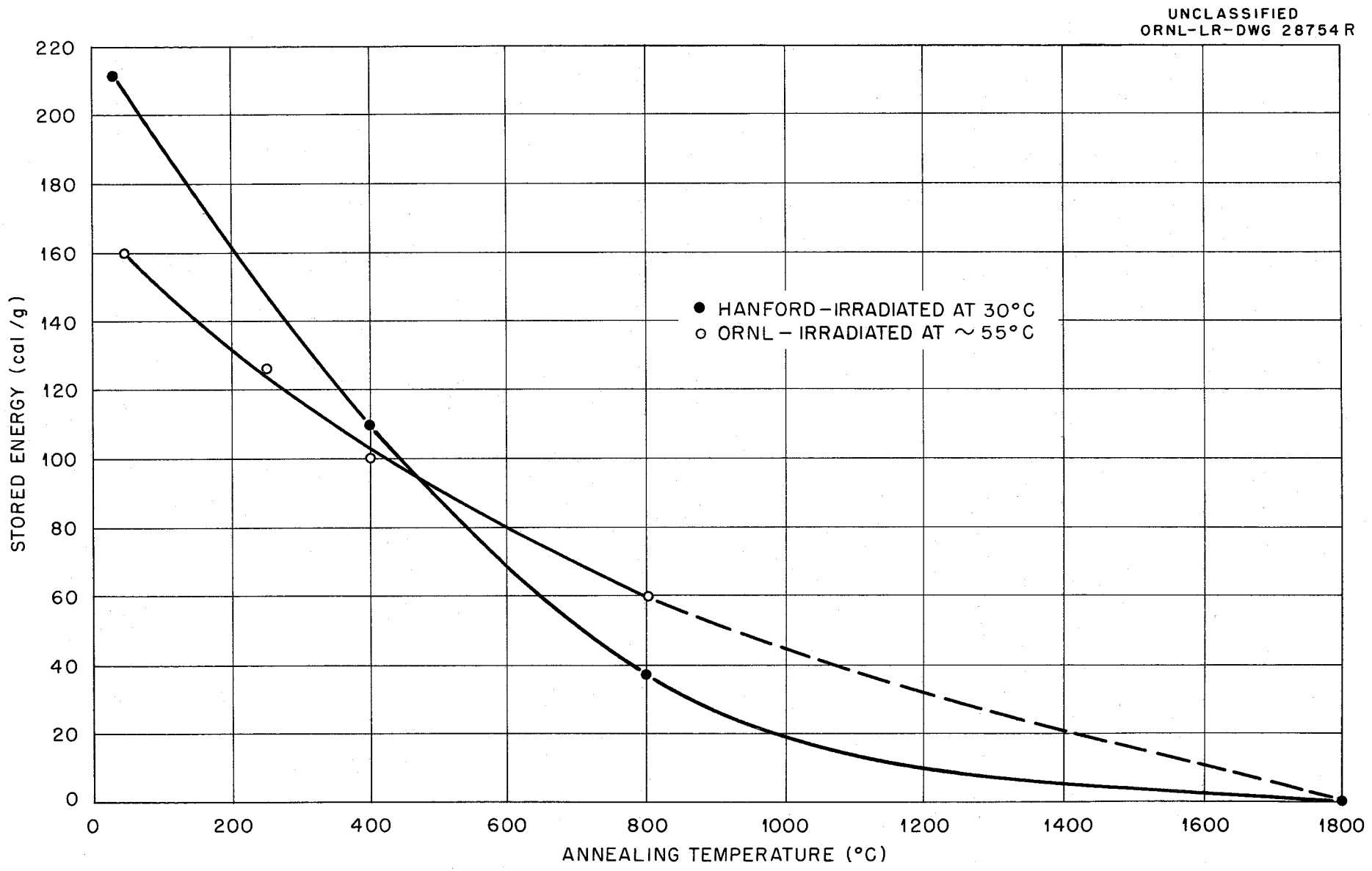


Fig. 4.20. Comparison of Residual Stored Energy in ORNL and Hanford Graphite.

retained after various anneals for a Hanford sample (closed circle) irradiated at 30°C and the ORNL sample (open circle); as can be seen, there was a greater amount of energy retained above 500°C in the ORNL sample even though the Hanford sample, before annealing, had about 50 cal/g more total stored energy. Similar results are even more strongly exhibited by comparing Harwell and Windscale data (curve B in Fig. 4.21, the closed-triangle curve in Fig. 4.22, and curve B in Fig. 4.23). It is therefore clear that temperatures, even in the low range of 30 to 100°C, strongly affect the low temperature (200°C peak) stored energy in irradiated graphite.

Questions often arise concerning high-temperature stored energy, released at temperatures above 400°C and up to 1800°C. Frequently, the high values associated with these stored energies are misleading, and they are meaningful only when they are referred to release spectra for known heating rates and the specific heat curve for unirradiated graphite. Fortunately, between 400 and 1000°C the energy release spectra is a rather smooth, shallow function that falls further away from the specific heat curve of unirradiated graphite with increasing temperature. This means that dangerous, spontaneous releases at these high temperatures cannot occur without violation of the second law of thermodynamics. In considering reactor safety, then, the concern should be with the energy release spectra in the low temperature region (200°C) where energy may be released adiabatically. The fact that ORNL graphite contains 40 cal/g from 400 to 800°C and 60 cal/g from 800 to 1800°C is of no consequence to the safety of the reactor, and measurements of total stored energies should be considered in this light.

Finally, in order to determine the effect of a rather low temperature anneal for potential reactor annealing procedures, a sample was annealed at 165°C for 18 hr, the temperature being monitored continuously, and the stored

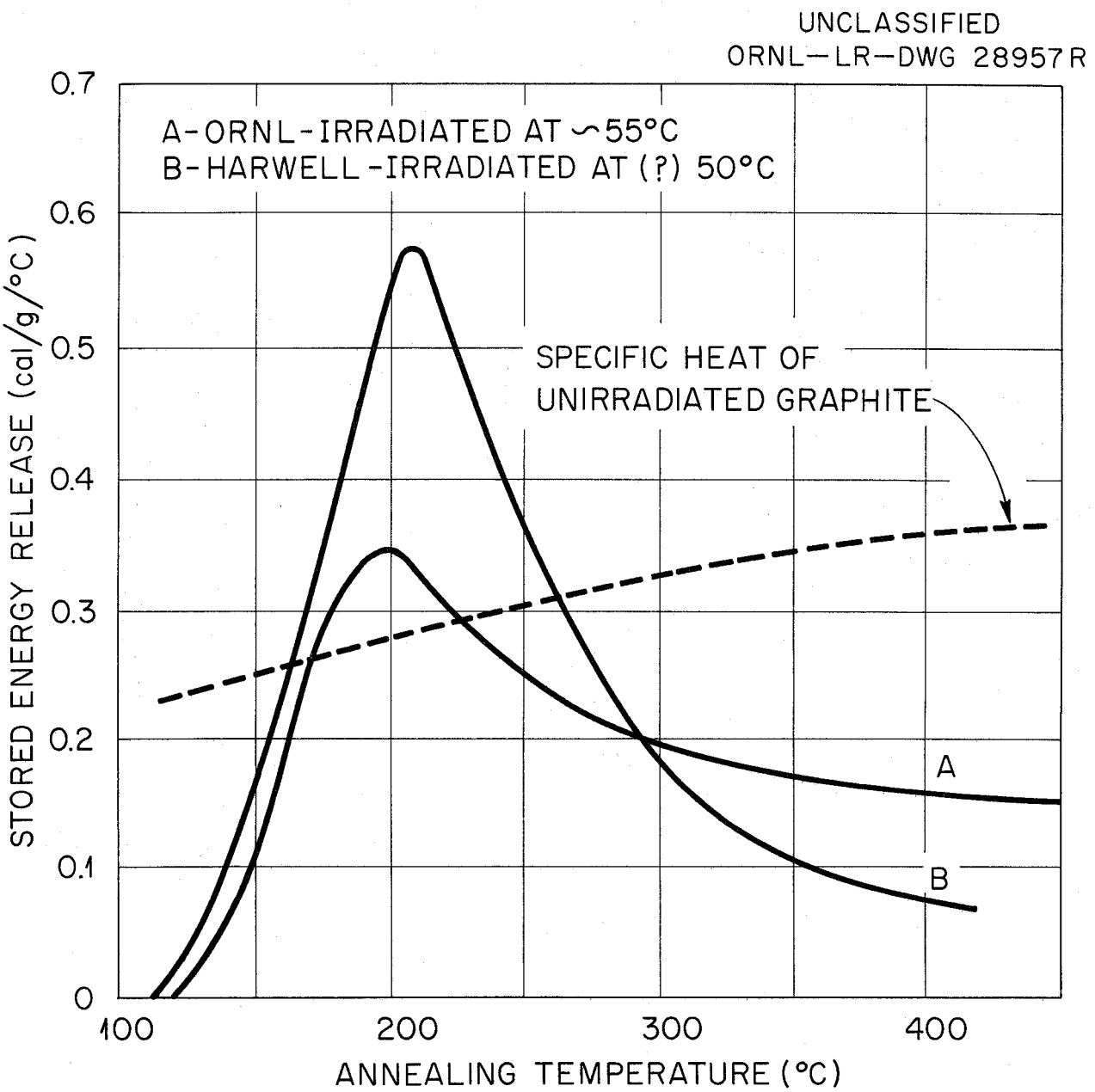


Fig. 4.21. Comparison of Stored Energy Release Spectra in ORNL and Harwell Graphite.

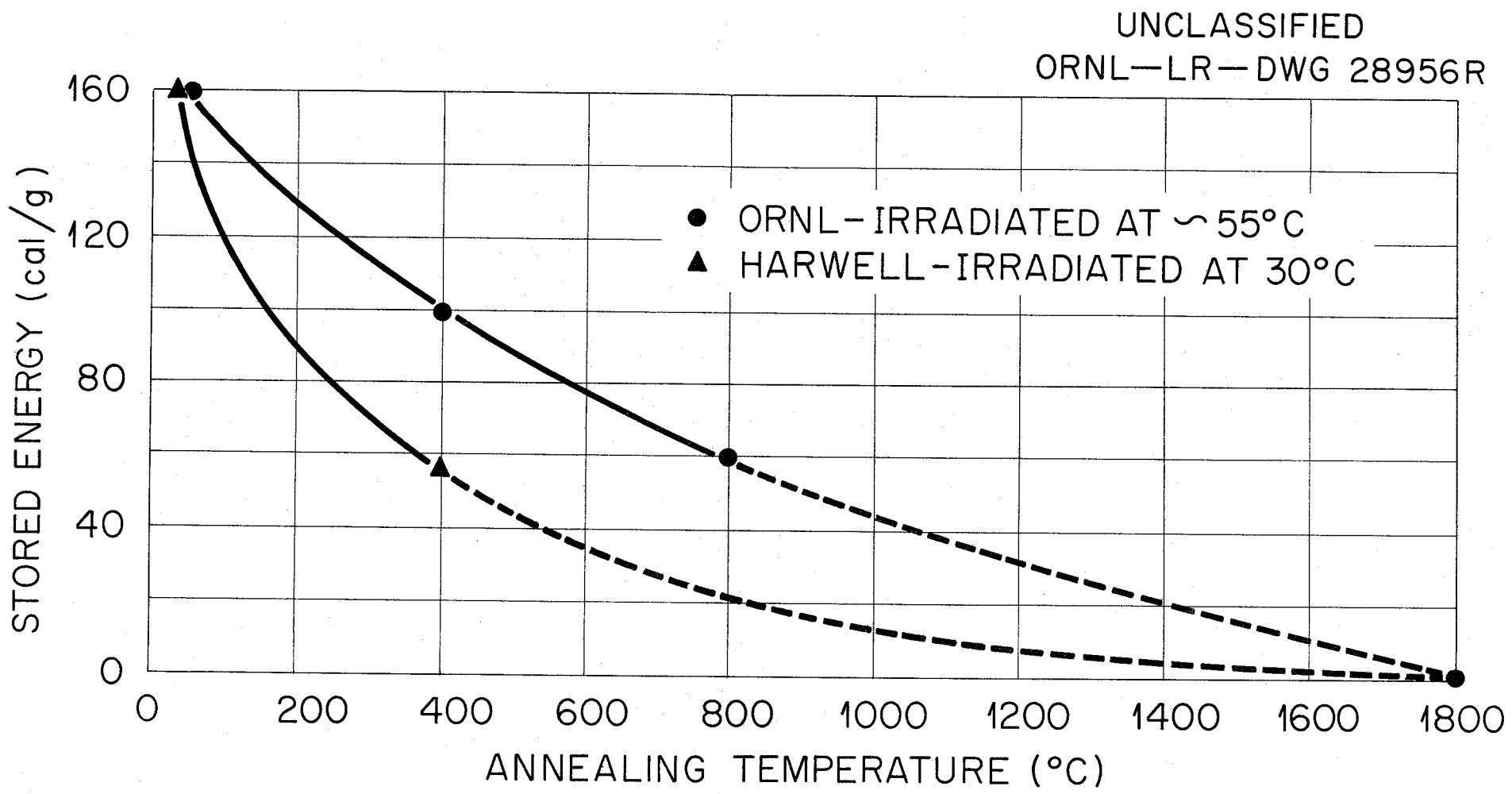


Fig. 4.22. Comparison of Residual Stored Energy in ORNL and Harwell Graphite.

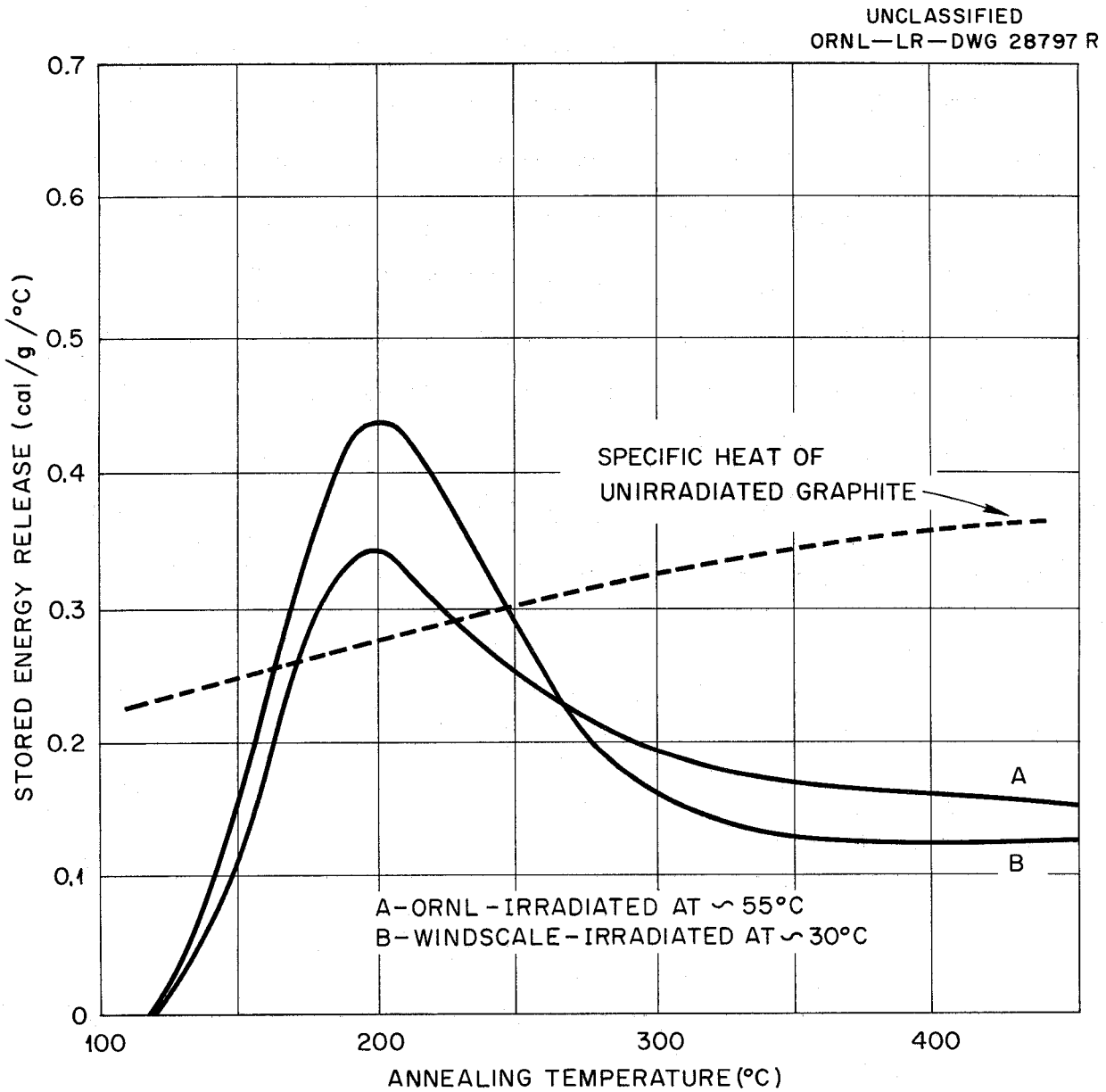


Fig. 4.23. Comparison of Stored Energy Release Spectra in ORNL and Windscale Graphite.

energy to 220°C was remeasured. It should be noted that during the 18-hr anneal no temperature excursions were observed. The resultant spectra are shown in Fig. 4.24 and indicate that higher temperature anneals, say in the vicinity of ~225°C, would be necessary to reduce the hump that approaches the specific heat curve.

The results of this investigation clearly show that the stored energy content of the ORNL Graphite Reactor presents no hazard and that a safe condition exists. It is also clear that differences in stored energy conditions between ORNL and BEPO (British Experimental Pile Operation) are primarily due to the differences in operating temperatures. Estimated regions of the graphite stack that contain negative specific heat are shown in Fig. 4.25. The boundary conditions used in this estimate are: (1) integrated neutron flux greater than 2×10^{20} nvt, and (2) maximum graphite temperatures of 50°C. From this approximation it appears that only about 5% of the entire stack is damaged to the extent that a spontaneous energy release could be propagated.

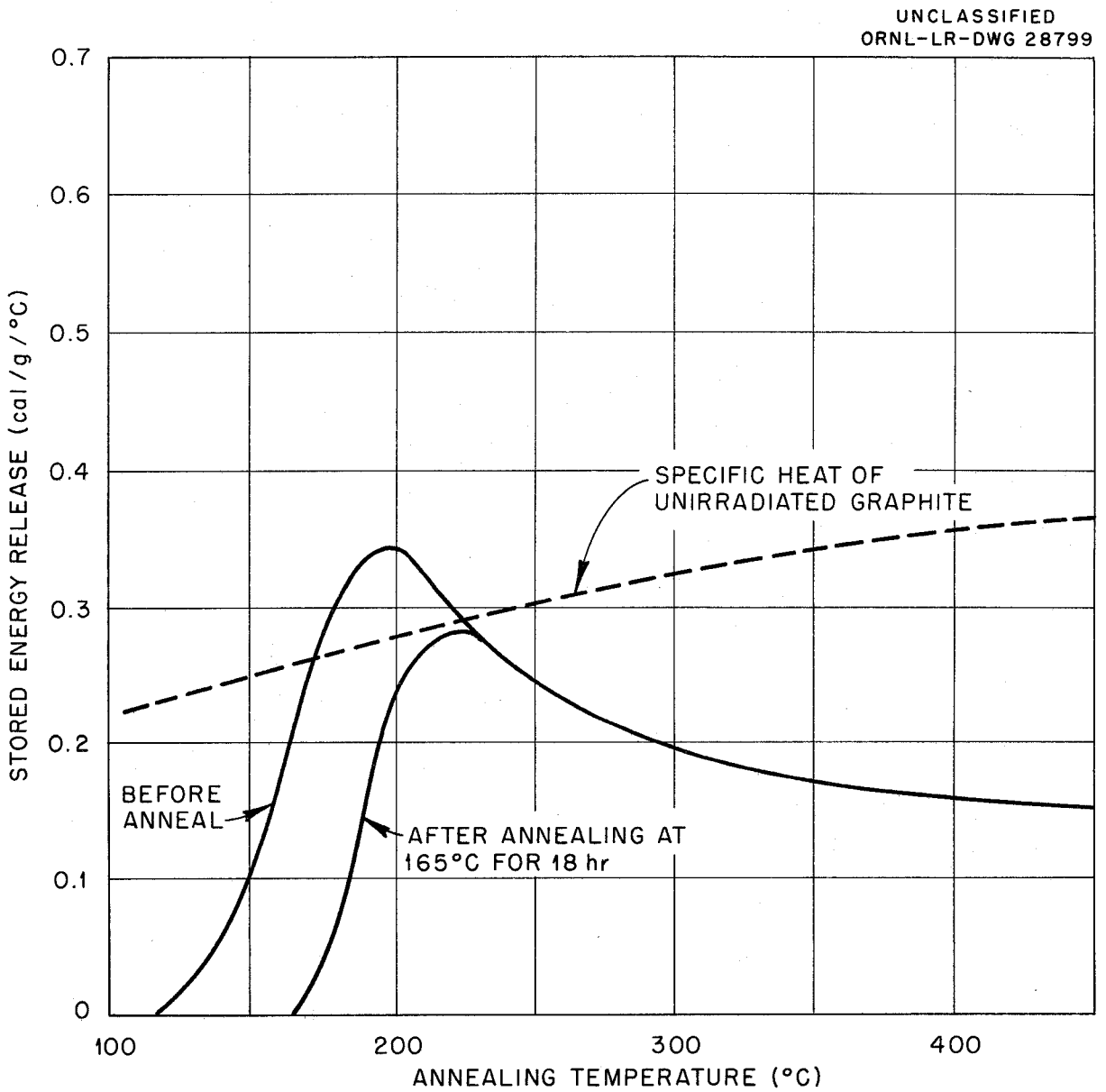


Fig. 4.24. Effect of Annealing at 165°C for 18 hr on Stored Energy Release Spectrum of ORNL Graphite.

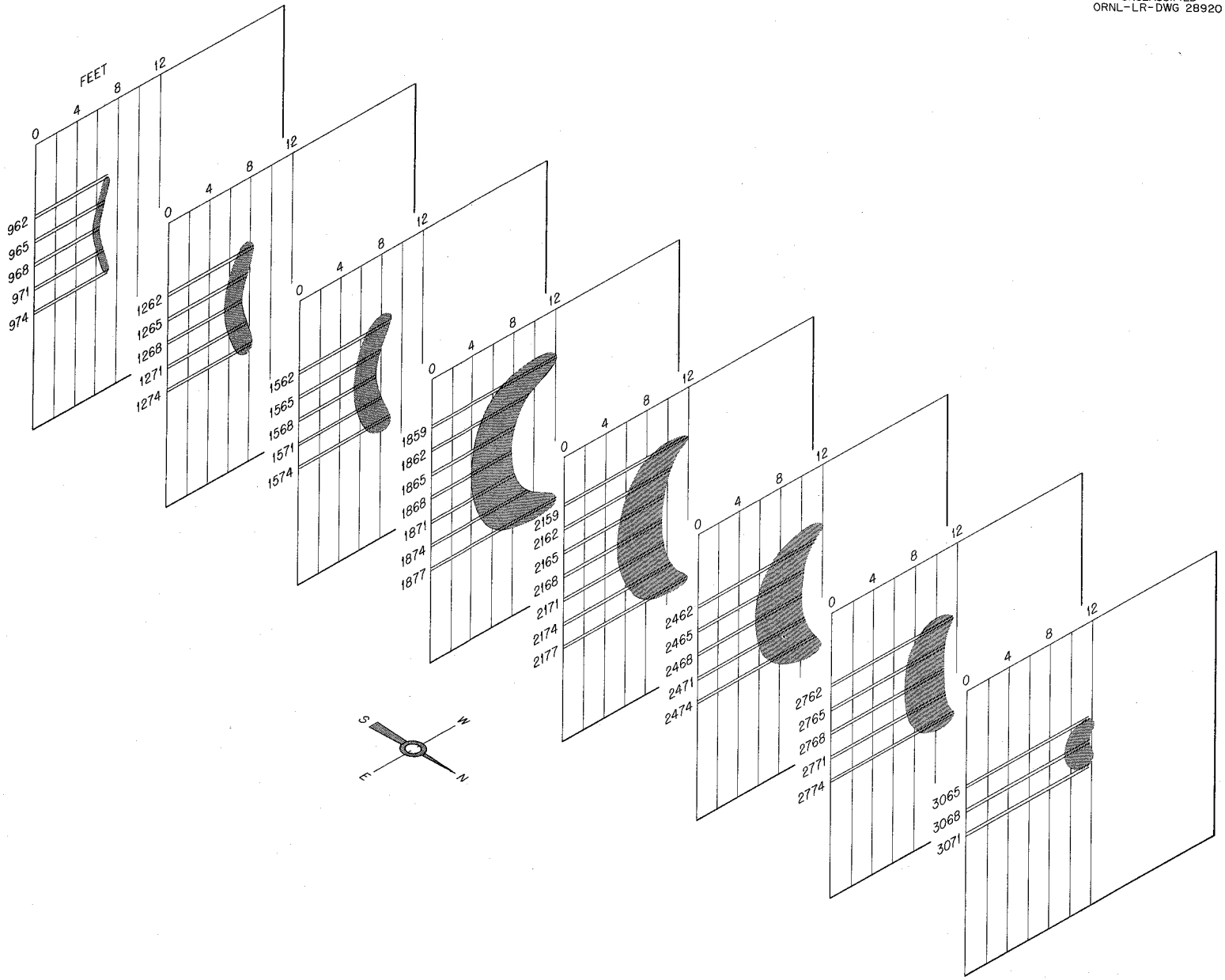


Fig. 4.25. ORNL Graphite Reactor Sections Containing Estimated Regions of Negative Specific Heat.

Acknowledgments

Many individuals have facilitated the compilation of this report, and to each of them the author is extremely grateful.

R. E. Nightingale (HAPO) graciously allowed the loan of the National Bureau of Standards facility contract for heat of combustion measurements. D. D. Wagman (NBS) arranged for speedy measurements of the ORNL samples. R. A. Meyer (BNL) measured stored energy to 800°C.

At ORNL, J. A. Cox, C. D. Cagle, and L. E. Stanford of the Operations Division secured the graphite scrapings and corings by remote instrumentation and obtained all the reactor temperature data. In the Solid State Division, F. A. Sherrill prepared and measured several X-ray specimens, and M. S. Wechsler and R. H. Kernohan were largely responsible for the radiation calorimetry measurements.

APPENDIX IA

ACCUMULATION AND CONTROL OF GRAPHITE STORED ENERGY IN THE OAK RIDGE GRAPHITE REACTOR

Introduction

The accumulation and spectral distribution of stored energy in graphite[1] is strongly dependent upon both neutron flux and irradiation temperature. Graphite-moderated reactors which contain unidirectional flow patterns, therefore, exhibit a concentration of damage in the cool, high-flux portions of the reactor (see Appendix I). Moreover, where this cooling direction has been fixed (as in some British reactors) and fission heat was employed in the annealing operation, it was often felt desirable to allow a considerably quantity of low-temperature stored energy to grow so that a controlled energy release could be "triggered" in the normally cool portions of the reactor and thus avoid overheating fuel elements in the normally hot, undamaged region of the reactor. The dangers inherent in these procedures are compounded by the fact that repeated annealing operations ultimately broaden the spectral distribution of stored energy in the "200°C peak" so that triggering becomes exceedingly difficult, if not impossible, under operating reactor conditions. If the coolant flow direction is reversed for annealing purposes,[2] these technical difficulties and hazards are easily overcome because fission heat is distributed in a manner which minimizes fuel element dangers and eliminates the need for a triggered release of

---

[1] J. M. Davidson, Stored Energy in Irradiated Graphite, HW-55736 (April 15, 1958).

[2] R. W. Powell, R. A. Meyer, and R. G. Bourdeau, "Control of Radiation Effects in a Graphite Reactor Structure," Paper 462, Second United Nations International Conference on the Peaceful Uses of Atomic Energy, June 1958.

stored energy as a source of heat.

In view of these facts, it is advisable to re-examine the stored energy release spectra in graphite from the most heavily damaged portion of the ORNL Graphite Reactor (OGR) with special reference to low-temperature annealing effects and estimates of long-term graphite stored energy problems.

Effects of Low Temperature Annealing

For reasons already expressed in Appendix I, we shall concern ourselves only with graphite stored energy that is releasable at temperatures up to 400°C. Cored specimens from the most heavily damaged portion of the reactor were employed in these experiments so that the most serious aspects could be analyzed. A typical release spectrum of an unannealed sample is shown in Fig. 1. A portion of this sample was annealed at 150°C for 5 hr in air, and no temperature excursions were observed. Following this anneal, the stored energy was remeasured and is shown in comparison with the unannealed spectrum in Fig. 2. It is apparent that this low-temperature anneal successfully removes the stored energy above the graphite specific heat curve. A second unannealed portion was similarly annealed for 5 hr with no temperature excursions at 138.5°C and the stored energy again measured. The comparable release spectra are shown in Fig. 3. The significance of these results is that relatively low annealing temperatures can be employed in removing stored energy in excess of the graphite specific heat curve. In these cases the released energy was merely transferred to atmospheric air. Although these are laboratory tests, it is encouraging to note that temperature excursions were also notably absent in the many reactor anneals at Brookhaven[2] where the rate of fission heat removal is high, even at low power levels, compared with the slow propagation of stored energy release in

UNCLASSIFIED
ORNL-LR-DWG 36358

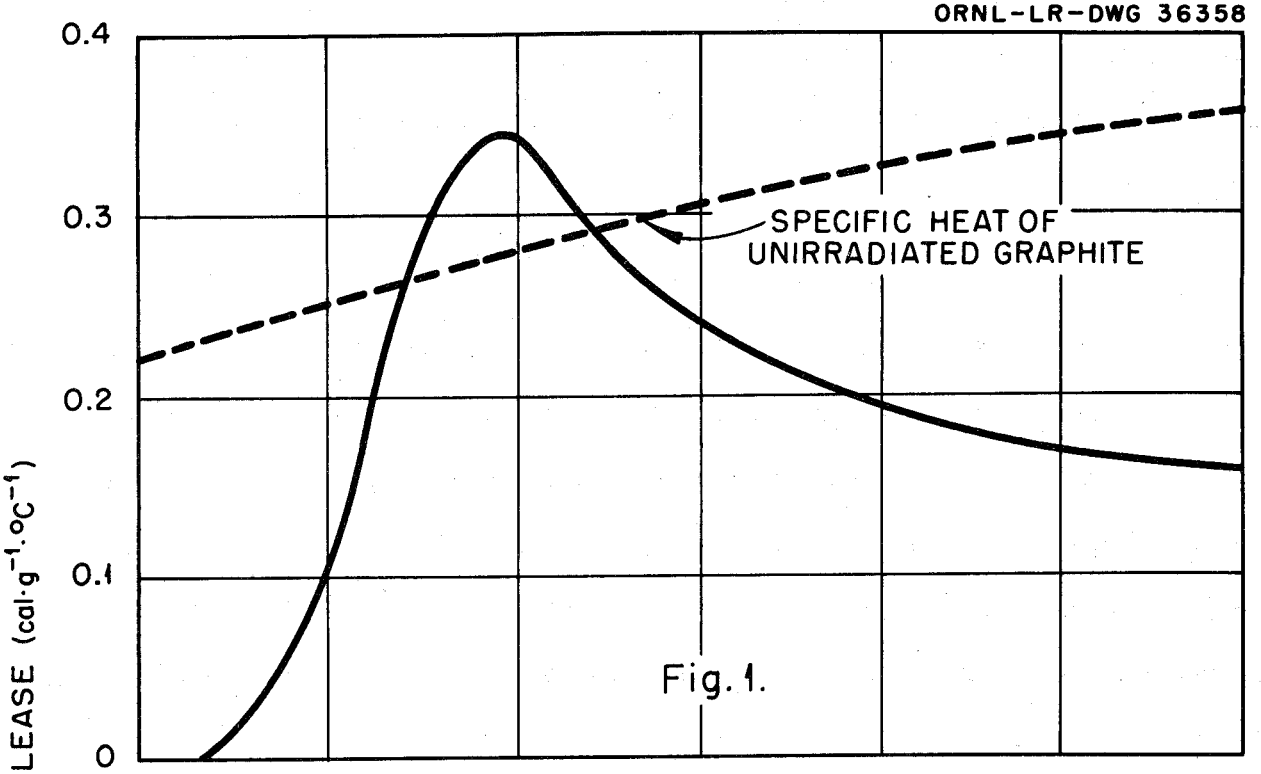


Fig. 1.

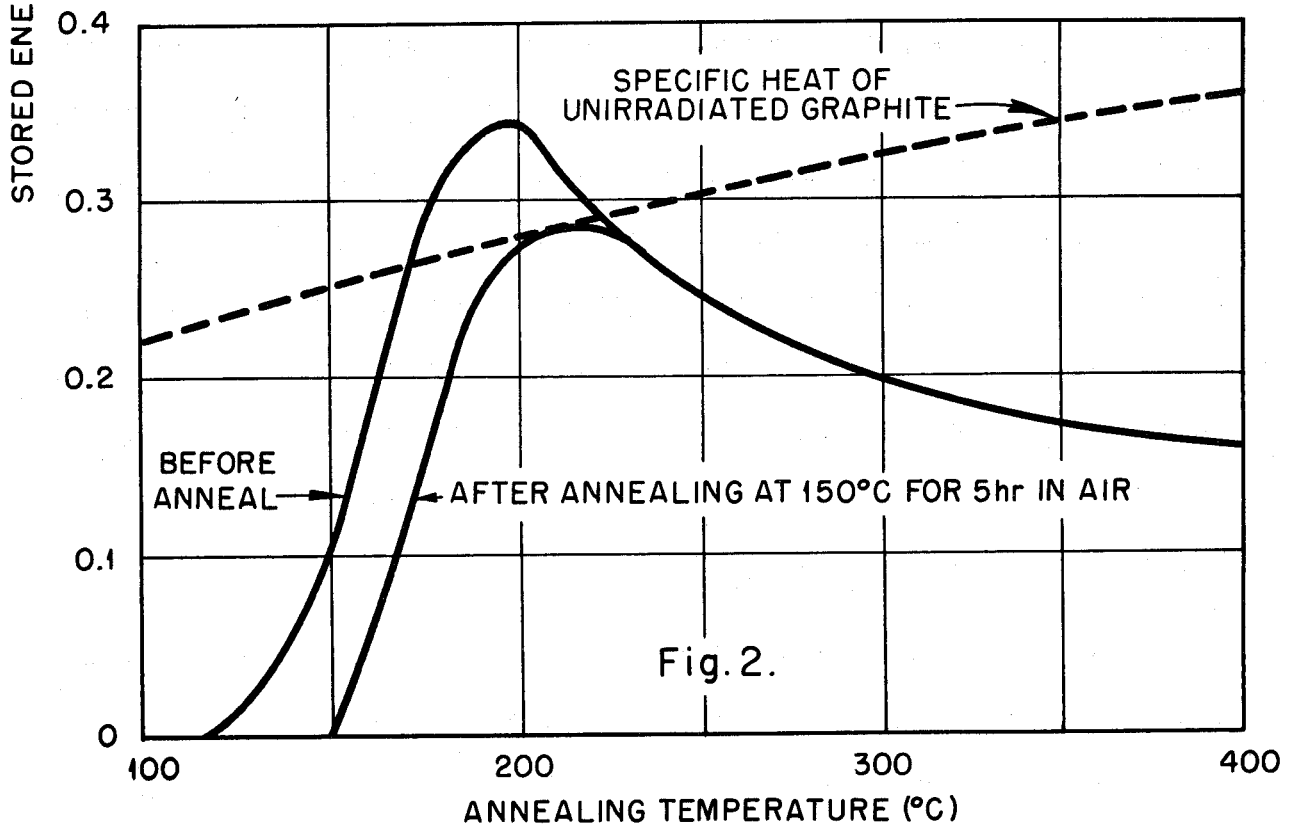


Fig. 2.

this critical temperature range. In effect, under reactor conditions the removal of fission heat completely masks the removal of stored energy, not only because it is comparably small but also because an efficient heat leak is operative.

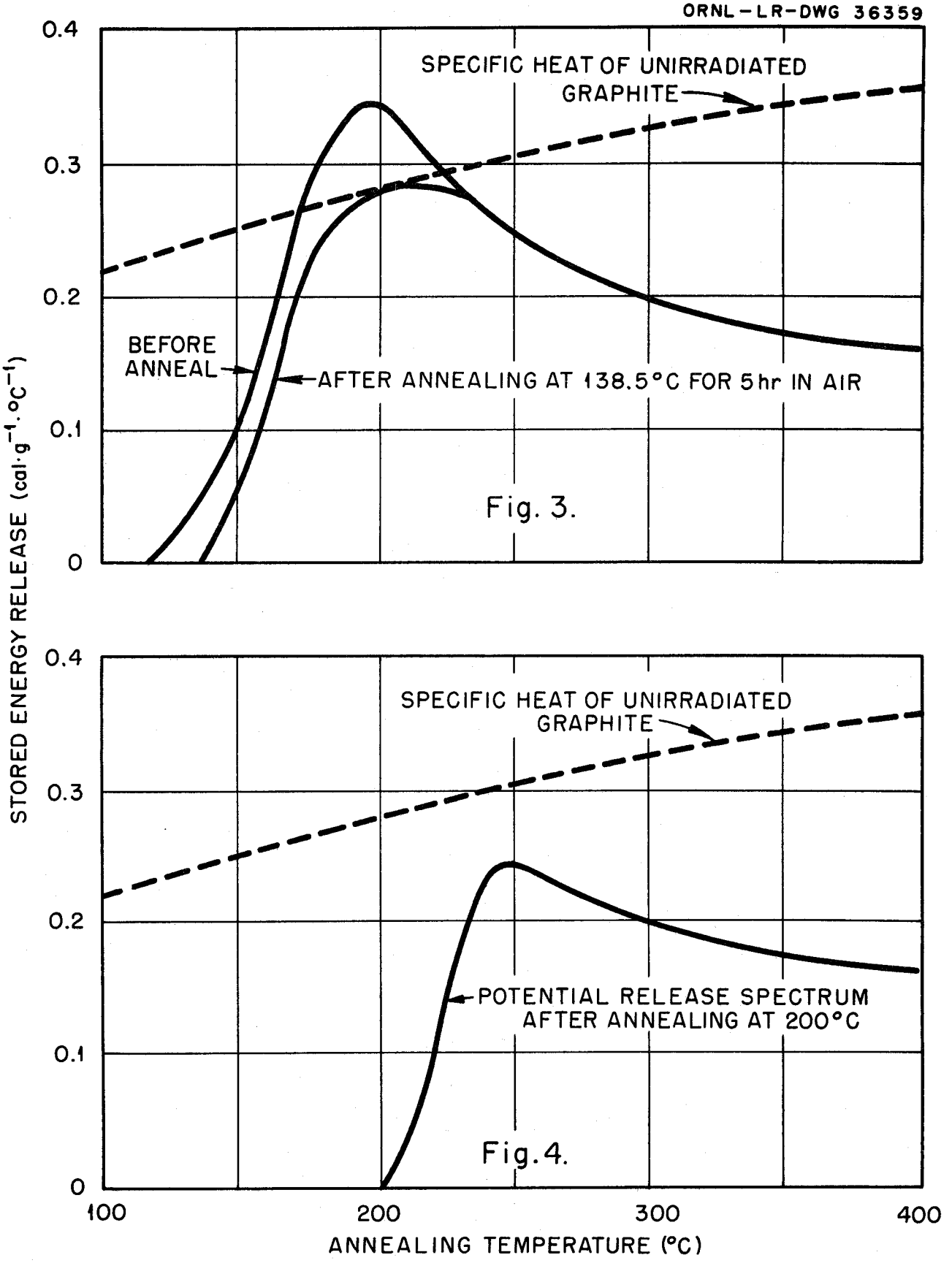
Short-Term Annealing Status

With the condition of reversed air flow in the Oak Ridge Graphite Reactor, the minimum temperature of 138.5°C can be attained in all parts of the moderator that contain stored energy in excess of the specific heat curve. Moreover, this can be accomplished with fuel element temperatures well below maximum tolerance. A low-temperature anneal would therefore remove the dangerous component of the release spectrum and thus permit higher annealing temperatures to be sought under conditions such that spontaneous temperature excursions would not be possible. The extent to which higher annealing temperatures would be sought would be necessarily governed by the maximum safe fuel element temperatures. If the maximum graphite temperature in the damaged moderator region were 200°C, the stored energy release spectrum following this anneal would approximate that shown in Fig. 4. One can see, therefore, that annealing in the low temperature range 138.5 - 200°C (Figs. 2, 3, and 4) results in the removal of stored energy that, prior to annealing, exceeded the graphite specific heat. Consequently, the short-term prospects of such an annealing procedure are excellent from the standpoint of hazardous stored energy.

Long-Term Annealing Status

It should be noted, however, that such a low-temperature anneal is not a permanent solution for control of the stored energy accumulation. Even after annealing at 200°C the stored energy release spectrum still shows a

UNCLASSIFIED
ORNL-LR-DWG 36359



hump at 250°C that projects toward the specific heat curve. This hump is of no concern in itself; but if continued long-term reactor operation were to follow the annealing operation (several years) not only would the "200°C peak" reappear but it would broaden to include stored energy in the 250°C region, thus developing an unfavorable condition. From these considerations, annealing at scheduled intervals, perhaps yearly, while the stored energy accumulation is very small and easily controlled, precludes the development of a hazardous amount of stored energy accumulating over a broader temperature range. Then the 250°C hump (Fig. 4) could be removed from the stored energy release spectrum by annealing to 250°C, with no possibility of a temperature excursion due to a stored energy release.

Discussion

The normal temperature characteristics of the OGR are such that stored energy would continue to accumulate,[1] at a very slow rate, and ultimately about 600 cal/gm would be contained in a small portion of the east half of the moderator structure. Although only a fraction of this energy[3] could be released spontaneously, it is more desirable to control, if not entirely eliminate, the growth of this damage. The complete elimination of stored energy accumulation would require operating temperatures which are excessively high for the OGR. However, frequent annealing at lower temperatures would accomplish the same result. Since the annealing can be accomplished by essentially normal operating procedures, this kind of control is even more desirable.

---

[3] R. E. Nightingale, J. M. Davidson, and W. A. Snyder, "Damage Effects to Graphite Irradiated up to 1000°C," Paper 614, Second United Nations International Conference on the Peaceful Uses of Atomic Energy, June 1958.

APPENDIX II

RESEARCH PROGRAM AND OPERATING EXPERIENCE ON ORNL REACTORS

By

M. E. Ramsey - Oak Ridge National Laboratory
C. D. Cagle - Oak Ridge National Laboratory

Introduction

The research programs at the Oak Ridge National Laboratory which make use of reactors or reactor products primarily use the two general-purpose research reactors. These are the normal-uranium, graphite-moderated reactor and the Low Intensity Testing Reactor. It is interesting that neither of these were built expressly as long term research reactors but have been adapted to this usage.

The research work involving usage of these reactors covers a wide range of fields in both pure and applied research, including the following: basic studies of neutron and radiation properties; solid state structure and phase transitions by use of neutron spectroscopy; radiation damage of materials; radiation effects in chemistry; radioactivation analyses of trace impurities; biological effects of radiation; reactor development; and radiation monitoring development.

Another important usage of these reactors is the preparation of radioisotopes for world-wide usage.

Training of personnel in reactor technology is another important phase of the work. Two programs are devoted to this work. One is the Oak Ridge Institute of Nuclear Studies (ORINS) which concentrates on the uses of

radioisotopes. The other is the Oak Ridge School of Reactor Technology which teaches the principles of reactor theory, design, and usage. In addition to these formal programs, faculty members from universities all over the United States do special research work at the reactors especially during the summer months.

This report describes these reactors and the research facilities available.

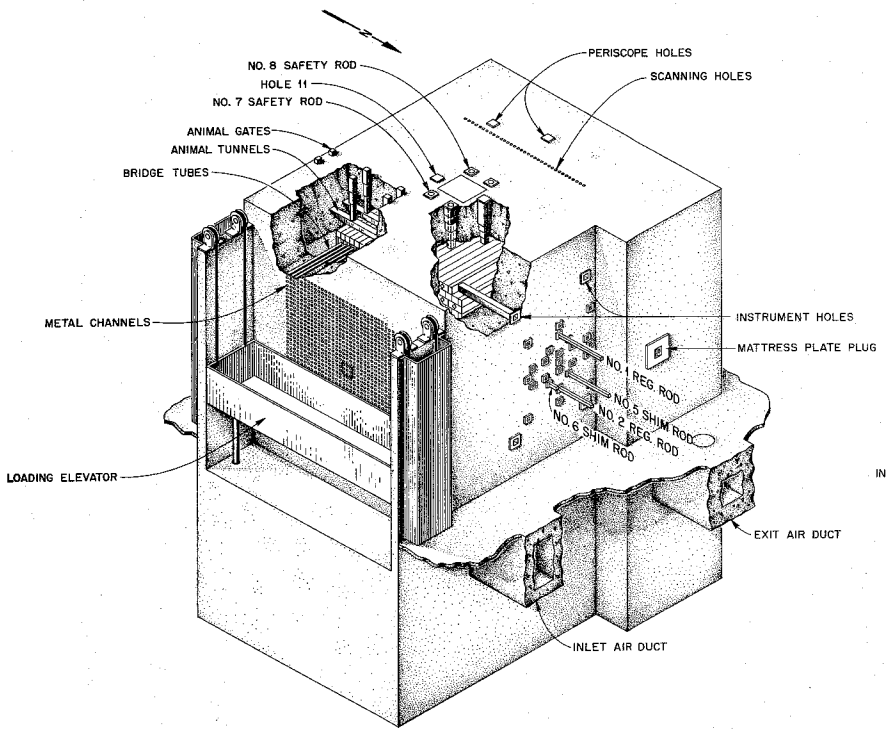
Graphite Reactor

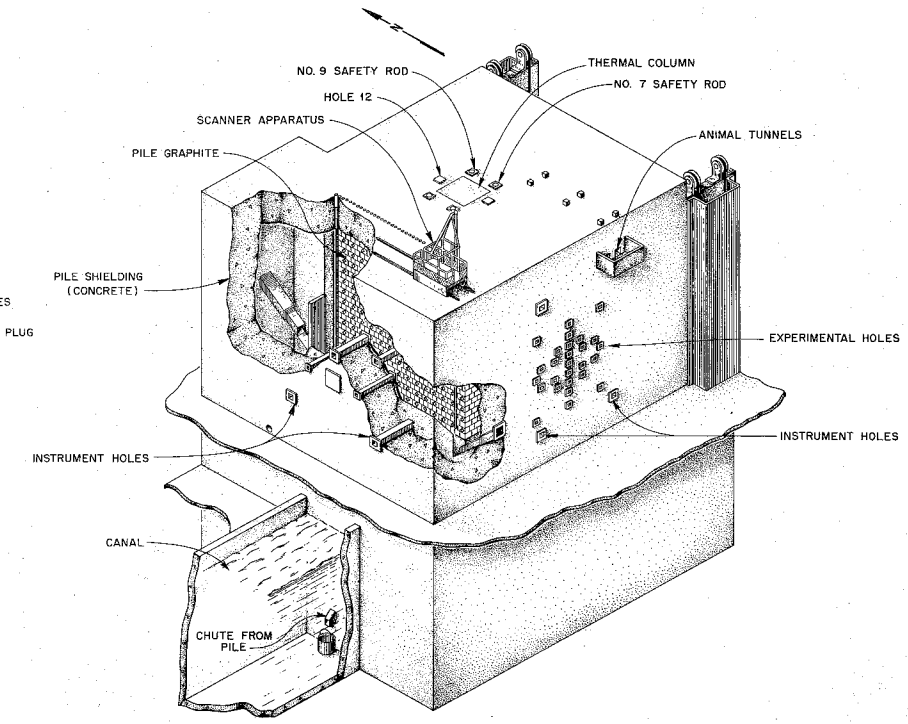
The Oak Ridge National Laboratory's graphite-moderated, air-cooled, normal-uranium reactor (Figs. 1 and 2) was built as the pilot plant for the plutonium-producing reactors which are at Hanford, Washington. Construction was started early in 1943, and critical loading was completed on November 4, 1943. This was the first reactor built that could be cooled and thereby operated at a reasonably high power level for long periods of time and was the second reactor built in this country. It was designed to operate at one megawatt power level; and its prime purposes were to furnish operating experience, reactor characteristics, and to provide small quantities of plutonium to aid in the development of chemical separation techniques to be used at Hanford. Since this is an air-cooled reactor, water-cooled fuel tubes of the Hanford type were simulated in some of the 4-in.-sq test holes.

The reactor core and reflector is a 24-ft cube of graphite built up of 4-in.-sq by 50-in. long blocks and contains 1248 fuel channels that are 1 3/4-in. sq with the diagonals vertical and horizontal. Also there are 22 horizontal 4-in.-sq holes that go all the way through, and six that go half way through, the graphite perpendicular to and between the fuel channels.

Dwg. 14587
UNCLASSIFIED

OAK RIDGE NATIONAL LABORATORY GRAPHITE REACTOR





MAXIMUM FLUX: 1.1×10^{12} NEUTRONS/CM^2/SEC

MAXIMUM OPERATING TEMPERATURES:
METAL: ~250° C
GRAPHITE: 170° C

NUMBER OF LOADING HOLES: 1248

COOLING AIR FLOW: 120,000 CU. FT. PER MINUTE

GRAPHITE CUBE: 24 X 24 X 24 FT. 4 IN. HIGH

OVERALL DIMENSIONS: – HEIGHT: ~ 35 FT.
LENGTH: 47 FT.
WIDTH: 38 FT. 2 IN.

FUEL: ALUMINUM-CLAD NORMAL URANIUM SLUGS,
1.1-IN. DIAMETER, 4 IN. LONG,
WEIGHING 2½ POUNDS PER SLUG.

NUMBER OF SLUGS PER FUEL CHANNEL: UP TO 54

CURRENT USAGE: RESEARCH AND RADIOISOTOPE PRODUCTION

POWER LEVEL: 3500 KW

CONTROL AND/OR SHIM RODS (NEUTRON ABSORBERS):
5 SHUTDOWN RODS AND
2 REGULATING RODS

FIG. 1

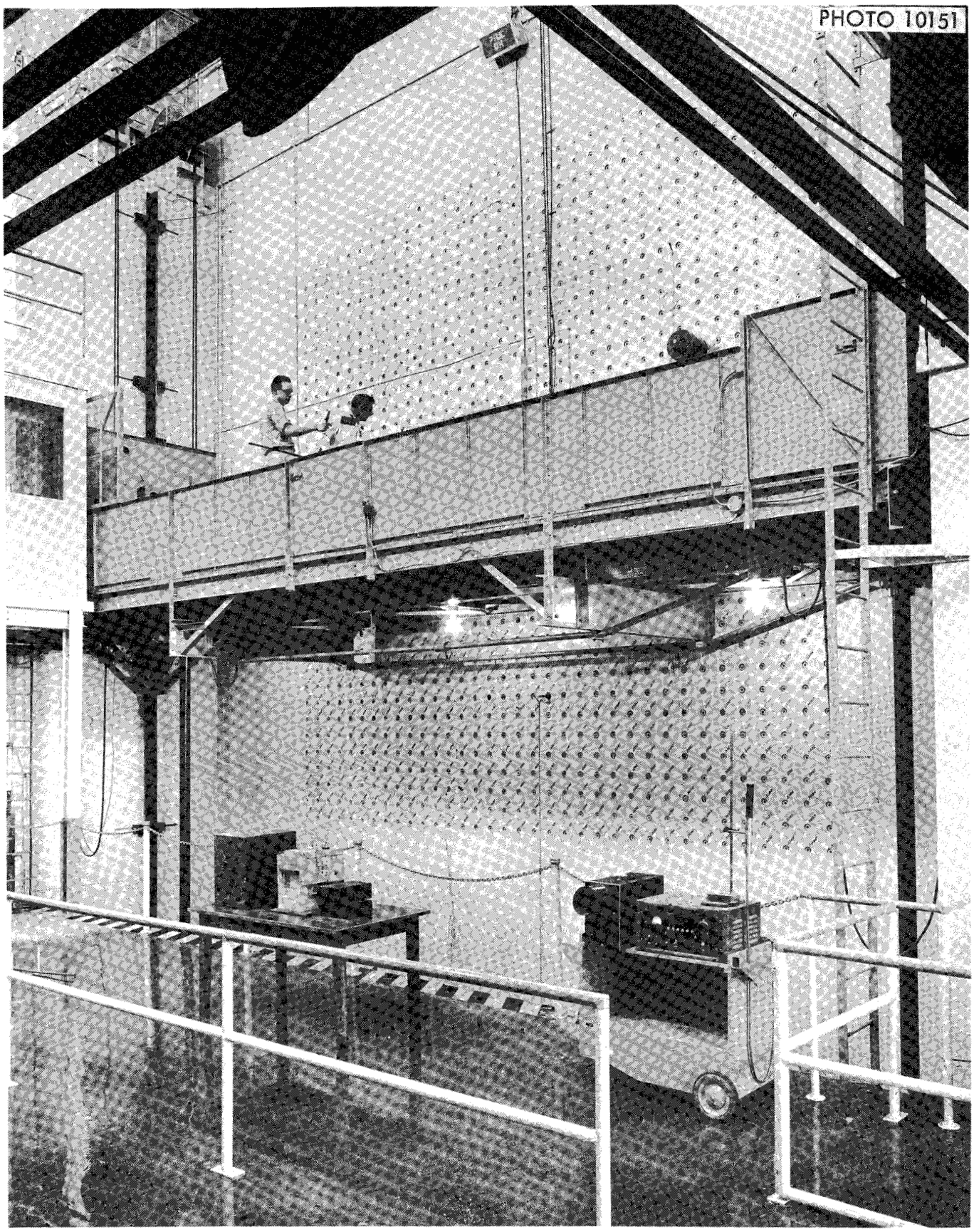


Fig. 2. Fuel-Charging Side of the Graphite Reactor.

Six vertical 4-in.-sq holes penetrate to a depth of 17 ft 4 in. into the graphite. Ten instrument openings penetrate the shield up to the graphite. Two thermal column openings, two 14-in.-sq biological specimen entrances, four 1.68-in.-dia holes through the central core, and a foil slot through the reactor center make up the remainder of the experiment facility openings (Table I).

The reactor shield is a 7-ft thickness of concrete that surrounds the core with manifold spaces provided for the cooling air. Plugged sleeves set into the concrete provide access to the holes through the graphite. The innermost and outermost 1-ft thicknesses of the shield are normal structural concrete and aggregate and contains web-type steel reinforcing. The space between is filled with a special concrete mixture containing Haydites, an expanded clay, to retain about 10% of the make-up moisture as neutron shielding and barytes aggregate to increase its density. The shield was overdesigned and is still more than adequate at the present higher power level. The reactor does not have a thermal shield, but the concrete is insulated to some degree from the graphite by layers of asbestos and cellamite (Fig. 3). A 10-in.-high air passage across the top of the core between the graphite and the concrete was designed into the reactor but was closed off shortly after operation started to deflect more air through the fuel channels when it was found to be unnecessary.

The reactor coolant is atmospheric air that makes one pass through and is discharged to the atmosphere through a 200-ft stack. In order to remove the heat at the present 3.5 Mw operating level, 120,000 ft^3/min of air is drawn through by two 900-hp suction fans. Rough Fiberglas filters remove dust particles from the air at the inlet to the system and a combination of fine Fiberglas and asbestos paper filters remove radioactive particles

TABLE I RESEARCH OPENINGS INTO GRAPHITE REACTOR

Kind of Facility	Number of Facilities	Maximum Thermal Neutron Flux	In-Cd Ratio	Gamma (R/hr)	Approximate Temperature
4-in.-sq horizontal	42 openings	3.6×10^{11} to 1×10^{12}	20	3×10^{5} to 8×10^{5}	35°C to 160°C
4-in.-sq vertical	3	8×10^{11}	20	6.7×10^{5}	135°C to 140°C
1.68-in.-diameter horizontal	4	1×10^{12}	20	9×10^{5}	35°C
2 3/4 in. x 3/8-in. foil slot	1	1.1×10^{12}	20	9×10^{5}	160°C
Unused fuel channels in core region	5	5.5×10^{11} to 1×10^{12}	20	4.5×10^{5} to 9×10^{5}	35°C
Unused fuel channels in reflector region	418	10^{10} to 3×10^{11}		10^{4} to 3×10^{5}	30°C to 40°C
14-in.-sq biological tunnel, lead-lined	1	1.3×10^{9}	~70 Mn-Cd ratio	200	25°C to 35°C
14-in.-sq biological tunnel, bare	1	5×10^{8}	73 Mn-Cd ratio	3.4×10^{3}	25°C to 35°C
5-ft-sq vertical thermal column	1	1.5×10^{7}	$\sim 10^{6}$	135 at top of column	Room temperature
2 1/2-ft-sq horizontal thermal column	1	5×10^{9}	130	1.2×10^{4}	Room temperature
Unused 9-in.-sq ionization chamber holes	2	10^{7} to 10^{9}	~70 Mn-Cd ratio	$\sim 5 \times 10^{3}$	35°C

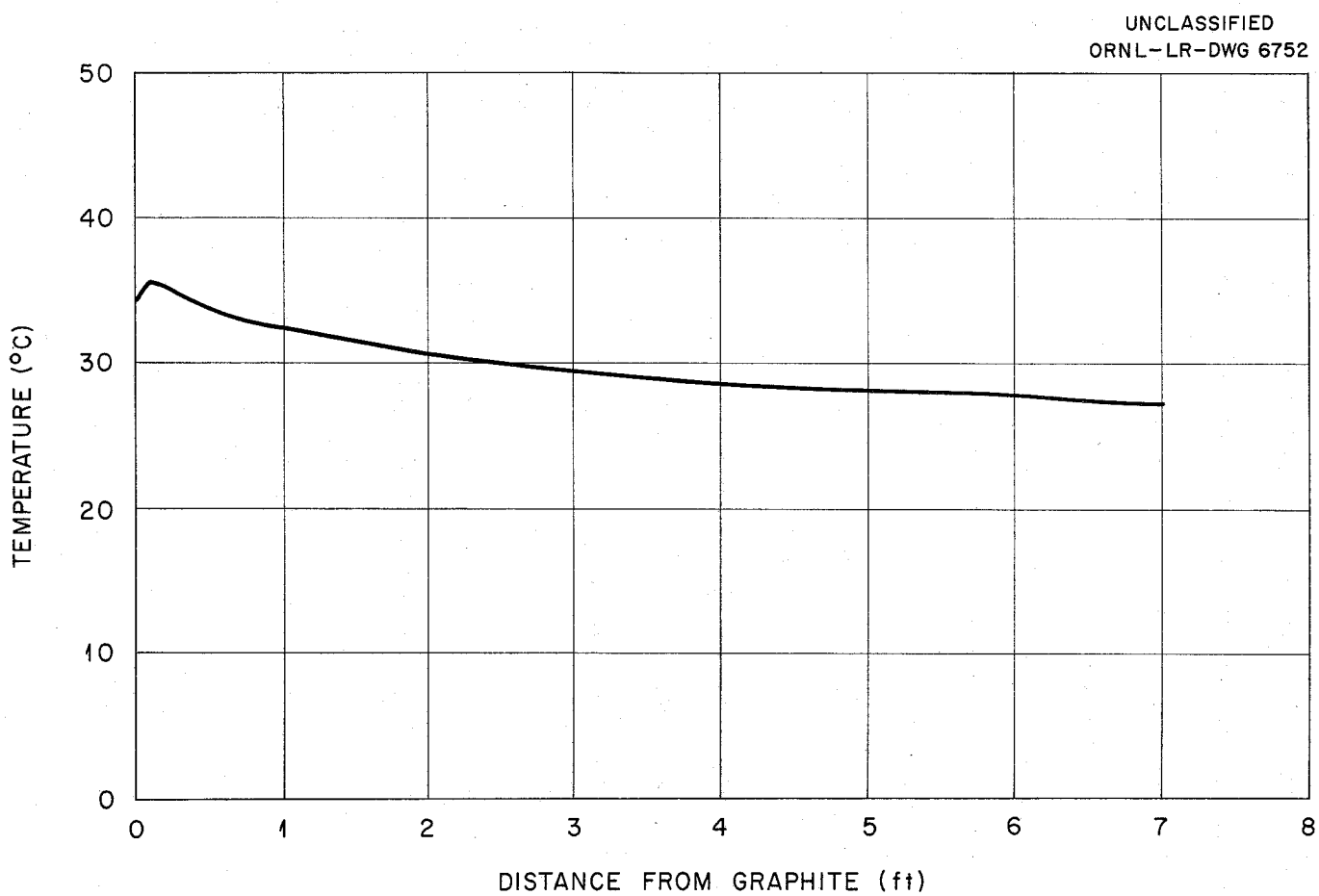


Fig. 3. Temperature Traverse Through the Concrete Shield of the Graphite Reactor Showing that Gamma Heating is not Excessive nor is the Temperature Differential Through the Shield Great Enough to Cause Severe Stresses.

picked up from the reactor before the air reaches the suction fans.

Five of the original eight shutdown rods are now in use (Table II); three were removed to provide for experiments since they proved to be unnecessary. Three of the remaining rods enter vertical 4-in.-sq holes from the top of the reactor. These rods are not of the original design. They are 3 1/2-in.-sq hollow steel shells, 8-ft long, that contain a layer of 1/16-in.-thick cadmium in the walls and a shield plug in the upper end. They are attached to steel cables that can be wound up on a windlass operated by electric motors through an electromagnetic clutch. The brake that holds the rod up in the operating position is on the electric motor so that if the power to the electromagnetic clutch is lost, either by electrical power failure or by intentional interruption as for a normal shutdown, the windlass is made free to turn and allows the rod to fall into the core by its own weight. Full fall requires little more than one second.

The other two shutdown rods are 1 3/4-in.-sq steel rods 200 in. long which contain 1.5% boron. They enter two of the horizontal 4-in.sq holes which are sleeved down to 2-in. sq with graphite. These two rods are driven by hydraulic motors for slow travel into or out of the reactor and can be made to travel the full distance into the reactor by a high pressure oil system within 4 sec. This high pressure, 900 lb/in.^{2}, is maintained by two large weights pressing upon oil-filled cylinders and is kept from the drive mechanisms by normally open solenoid valves so that if the electrical power fails or is interrupted by the shutdown switch the valves open and allow the 900 psi pressure to hit pistons which operate rack and pinion drives to push the rods into the reactor.

Two other rods, identical to the hydraulic rods, are used as regulating rods when the reactor is being operated manually. They are driven by

TABLE II CHARACTERISTICS OF CONTROL AND SHUTDOWN RODS

	Control Rods	Horizontal Shutdown Rods	Vertical Shutdown Rods
Number of each	2	2	3
Reactivity value	#1--0.57% $\frac{\Delta k}{k}$ #2--0.50% $\frac{\Delta k}{k}$	0.63% $\frac{\Delta k}{k}$	0.50% $\frac{\Delta k}{k}$
Material	1.5% boron steel	1.5% boron steel	Hollow steel shell containing layer of cadmium
Size and Shape	1.75-in.-sq bar 19 ft long	1.75-in.-sq bar 19 ft long	3 1/2-in.-sq hollow steel shell 8 ft long. Top end contains lead and masonite shield and bottom end has spring shock absorber
Method of operation	Direct electric motor driven	Hydraulic cylinder operates rack and pinion drive. Normal operation, pump; emergency operation, accumulator	Gravity when latch solenoid is de-energized. Lifted out of reactor by electric motor
Travel (Distance into reactor)	In: To within 5 ft of opposite side of graphite Out: To 1 ft outside graphite into concrete shield	In: To within 5 ft of opposite side of graphite Out: To 1 ft outside graphite into concrete shield	In: 4 ft below and 4 ft above reactor center Out: Just withdrawn from graphite

TABLE II (continued)

	Control Rods	Horizontal Shutdown Rods	Vertical Shutdown Rods
Speed of travel	#1: Max. in 6.9 in./sec Max. out 6.9 in./sec #2: Max. in 4.76 in./sec Max. out 4.76 in./sec Min. in 0.044 in./sec Min. out 0.044 in./sec	In: Normal--6 in./sec Emergency--full travel in 4 sec Out: 1 in./sec	In: ~1 sec for full fall Out: ~10 sec
Guide Tube	2 1/4 x 2 1/4 in.-sq slot milled into graphite blocks in 4-in.-sq hole	2 1/4 x 2 1/4 in.sq slot milled into blocks in 4-in.-sq hole	Full size 4-in.-sq vertical hole

electric motors and are not part of the shutdown system. Both rods have a fast speed in and out and one has a slow speed for fine control.

Within recent months an automatic control system has been developed which uses one of the hydraulic rods in the nearly out position as its control rod. The electrically driven regulating rods are still used to position the automatic control rod within its operating range. This arrangement tends to minimize distortion of the neutron flux pattern within the reactor core.

Up until recently a boron-shot tube was kept for emergency or disaster shutdown use. This consisted of a hopper of boron-steel shot that could be released into the reactor core region by pulling a cable in the control room to release a plug in the bottom of the hopper. The hopper was within the top shield of the reactor and could not be easily inspected. Also there was no way to test the working of the apparatus on a routine basis. By accident the pull cable was pulled recently and the shot did not fall into the reactor. They were found to be rusted and stuck together. Because of this, authorization has been given to completely remove the equipment since any safety equipment that cannot be periodically checked cannot be trusted to operate. All other safety features can be checked.

When operation of the graphite reactor was first started, very low-level neutron flux during startup was monitored with a counter and then by a gamma-compensated, boron-coated, ionization chamber as the power level rose to the kilowatt level. The practice of using the neutron counter was discontinued and only the ionization chamber retained. This is safe because the fuel loading is kept such that the excess reactivity is below the delayed neutron contribution and prompt critical is an impossibility. Three of the gamma-compensated chambers are now in use; one is read by a galvanometer and

used for startup and manual operation, a second for the automatic control, and the third for vernier control while in manual operation or as a spare. These chambers are located in the shield of the reactor where they "see" a neutron flux of about 10^8 n/cm^2·sec.

Three other non-gamma-compensated, boron-coated, ionization chambers are used for the safety instrumentation to scram the reactor by way of electronic trips if the power level should be allowed to exceed 5.5 Mw. It is not necessary that these chambers be gamma compensated because they are important only at a neutron level where the gamma contribution is negligible.

A second safeguard against power excursions is a system of boron-coated thermopiles whose composite signal which is proportional to the power level is fed to an electronic strip recorder containing a high level trip to initiate a scram at 6.0 Mw.

The temperature of the aluminum jackets that enclose the uranium metal in the reactor must also be limited. This is accomplished by including in the loading fuel pieces having iron-constantin thermocouples attached to the jackets. Experience has shown that the hottest position in any fuel row is two feet past the center of the row in the direction of the air flow. At least forty such thermocouples are kept in the reactor at all times and positioned to give a good representation of the temperature distribution. The thermocouple having the highest reading supplies a signal to a single-point strip recorder containing a high-level trip to scram the reactor if the temperature exceeds 325°C. Actually the operator does not allow the temperature to exceed 280°C. It has been found that a standard power level of 3.5 Mw can generally be maintained year around without exceeding the safe temperature level.

The temperature of the graphite near (1 ft from) the center of the

reactor is recorded but does not actuate any scram or alarm. Its value averages about 135°C.

Temperatures of the inlet- and exit-cooling air streams are measured for the purpose of heat power calculations by thermohms located in the inlet- and exit-air ducts. At one time the signal from a thermohm in the exit-air duct was fed to a strip recorder containing a high-level trip to cause a scram. The scram feature has been removed as unnecessary since the lag in this temperature is too long to be of value in the event of a power excursion, and it served no other purpose.

A pitot tube in the inlet-air duct sends a signal to oil manometers at the control desk and to a ring-balance, circular-chart recorder which scrams the reactor if the air flow drops too low. Another ring-balance recorder shows the pressure drop across the reactor from pressure taps located in the inlet duct and in the exhaust manifold. This recorder also scrams the reactor if its reading falls too low.

The radioactivity of the exhaust air from the reactor is measured by an ionization chamber through which a continuous sample of the air passes. The signal from this ionization chamber is read on a strip recorder at the control desk and until recently would actuate a scram if the normal radioactivity level increased by a factor of three. Experience has shown that the reactor itself could not cause such a condition unless it experienced a violent power excursion or great loss of air flow; conditions which, if they occurred, would initiate a scram. Also the off-gas from experiments and equipment at the Low Intensity Test Reactor have been tied into the graphite reactor exhaust, and it would be pointless to shut down the graphite reactor because of radioactivity from some other source. In the course of one operating day, the reactor generates a total of about 500 curies of radioargon (A^{41}) from the

0.9% argon present in the atmosphere. This is the major radioactivity in the exhaust air. There is a small amount of mixed fission product gases being evolved from small quantities of uranium oxide dust, from defective fuel elements, caught in cracks between graphite blocks.

The concrete shield at the exhaust end of the fuel channels would be damaged by the continual blast of hot air emerging from the channels if it were not protected. A one-inch thick wall of cellamite is spaced four inches from the inner face of the concrete to form a separate chamber which is cooled by a separate air stream. The pressure within this region as well as the air flow to it are monitored. If the negative pressure within the region becomes excessive, there is a possibility that the shield wall might collapse. To guard against this event, a negative pressure greater than 34 in. of water will cause the suction fans to be turned off. Loss of the fan suction shuts the reactor down by the other means described.

The hydraulic pressure accumulators which drive the horizontal shutdown rods into the reactor must be kept pumped up to contain a sufficient volume of oil at all times that the reactor is in operation. If a leak in the system should develop allowing the weights to fall to a minimum level, a switch actuated by the weight scrams the reactor while there is still enough oil left to accomplish the shutdown.

Frequently experiments are performed which can develop conditions which can injure personnel, the reactor, or expensive equipment if the reactor is not scrammed. Safeguard circuits from these experiments can be tied into the reactor scram circuits. Typical examples are:

1. One of the thermal column openings is shielded by a tank of water which, if accidentally drained, would allow about a 2-ft-square beam of radiation to emerge from the reactor into a heavily frequented personnel

area. To prevent this, a radiation monitor at the location scrams the reactor and sounds a siren.

2. A 4-in.-sq horizontal hole contains a water-cooled hollow cylinder of enriched uranium-aluminum alloy in which targets can be subjected to a high, fast neutron flux. If the water flow were to fail, the cylinder would probably melt and damage the reactor. The reactor is scrammed by a water flow recorder if the flow is lost.

3. A vertical 4-in.-sq hole is used to expose capsules containing test fuels for homogeneous reactors to check both solution stability and corrosion of the capsule walls. If the pressure in the capsule becomes excessive due to unexpected behavior, the capsule may rupture causing contamination of equipment or even personnel with fission products. To prevent this, a pressure monitor on the capsule scrams the reactor to stop the pressure buildup.

Manual scram switches are located at the control desk, the exit doors of the building, and at the experiment locations. The control console is shown in Fig. 4.

TABLE III AUTOMATIC REACTOR SCRAM SUMMARY

Power level greater than 5.5 Mw (ionization chambers)
Power level greater than 6.0 Mw (boron-coated thermopiles)
Metal temperature greater than 325°C
Air flow less than 40,000 cfm
Differential pressure across reactor less than 8 in. wg
Radiation at horizontal thermal column greater than 50 mr/hr
Any experiment requiring that the reactor be shut down for safety reasons

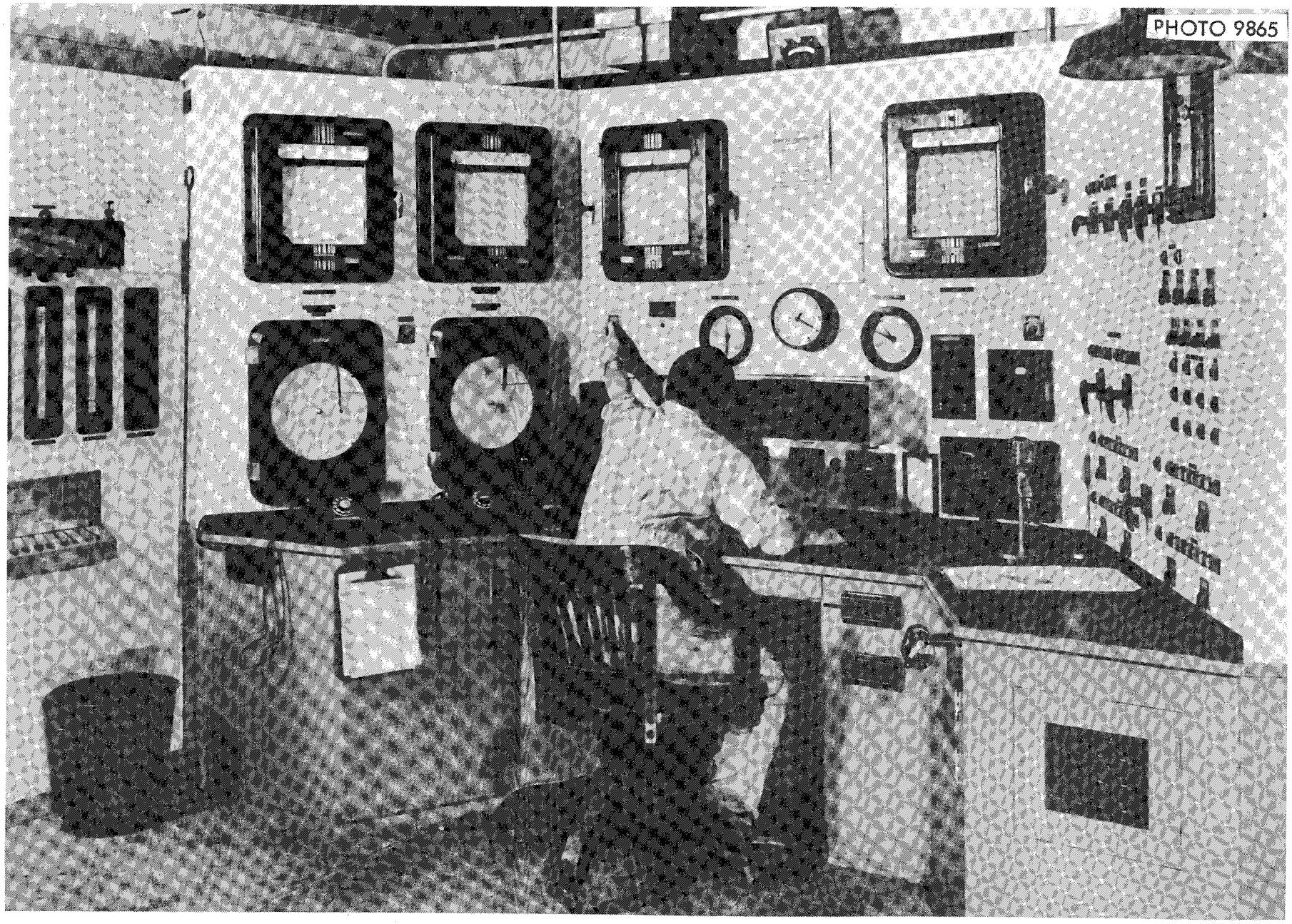


Fig. 4. Control Console of the Graphite Reactor Showing Operator Throwing "Scram" Switch to Shut the Reactor Down.

The reactor fuel is normal-uranium metal in the form of 1.1-in.-dia by 4-in.-long slugs encased in a 35-mil thick 2-S aluminum jacket. These are charged into the fuel channels end to end to simulate long rods. The purpose of the segmentation is for ease in handling. Each slug weighs about 2 1/2 lb.

The fuel is inserted into the reactor by hand and positioned with measured rods while the reactor is shut down. Removal is accomplished by pushing it on through the graphite from which it falls through the air-exhaust manifold to a chute which leads to a 20-ft-deep water pit outside the reactor shield (Figs. 5 and 5a).

All during the operation of the reactor, an annoying number of fuel element failures have occurred; and two full-scale attempts have been made to improve them. The first elements used were jacketed in 2-S aluminum cans having 20-mil thick walls, and the caps were sealed on by resistance welding. No bonding existed between the uranium and the aluminum. This type was used during 1943 and part of 1944. The next type had 35-mil thick 2-S aluminum walls with the caps arc-welded on in a helium atmosphere. Again the uranium and the aluminum were not bonded but were die-pressed to be in close contact. No failure of the first type of elements occurred but they were used only at low power and for only a short time. The second type came into use when the power level was increased from 1 Mw to 3.6 Mw. The failure rate of these elements was approximately one per month. Although this rate was not excessive enough to curtail operation, it did require constant vigilance to prevent damage to the reactor. The causes of failure were probably several, but the end result was always the same--bursting of the aluminum jacket allowing the rapid formation of uranium oxide which was carried away to some degree by the air stream causing contamination of the

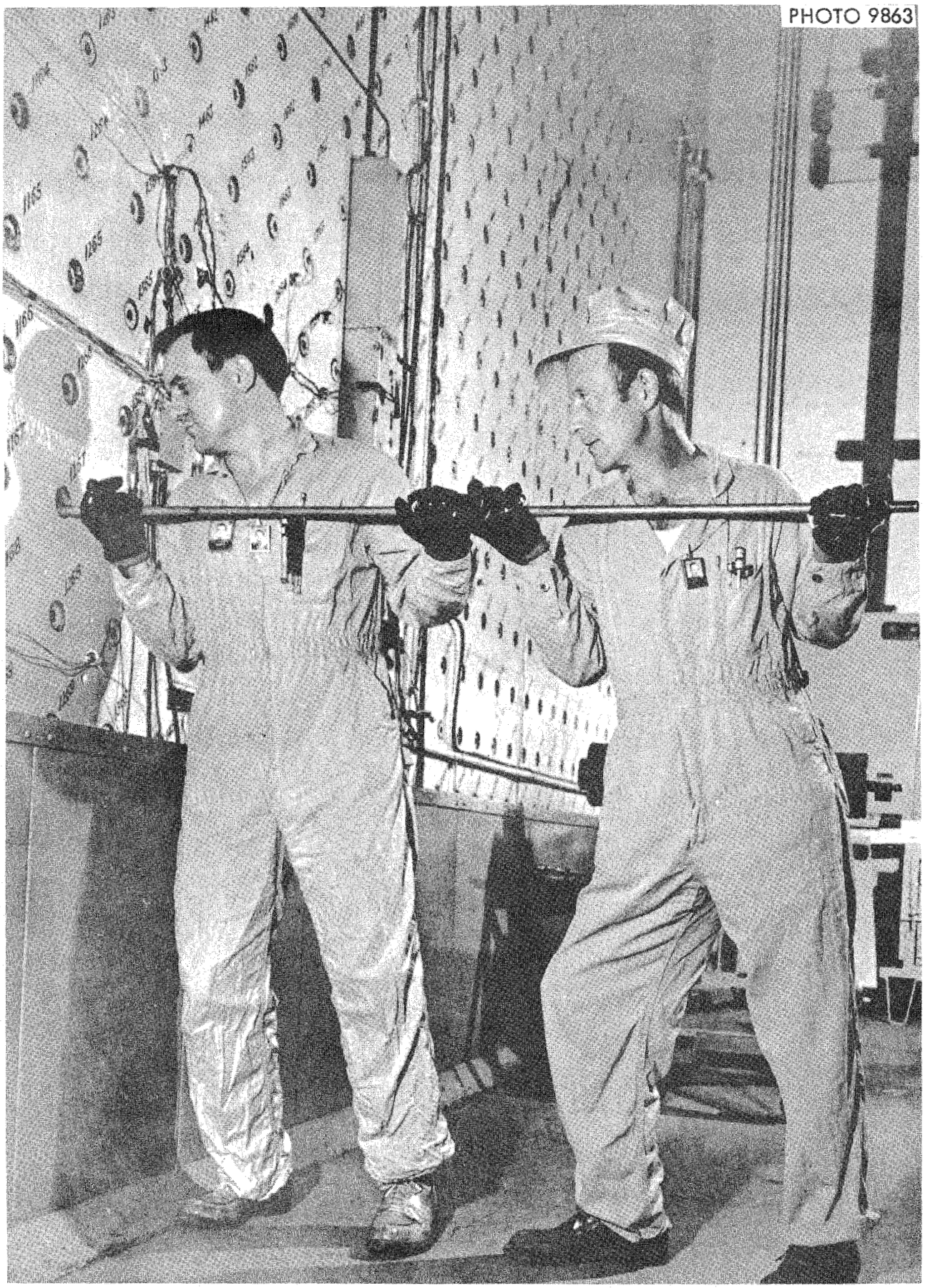


Fig. 5. A Closeup View of the Fuel Charging Face Showing Operators Pushing Fuel Elements from the Reactor Core.

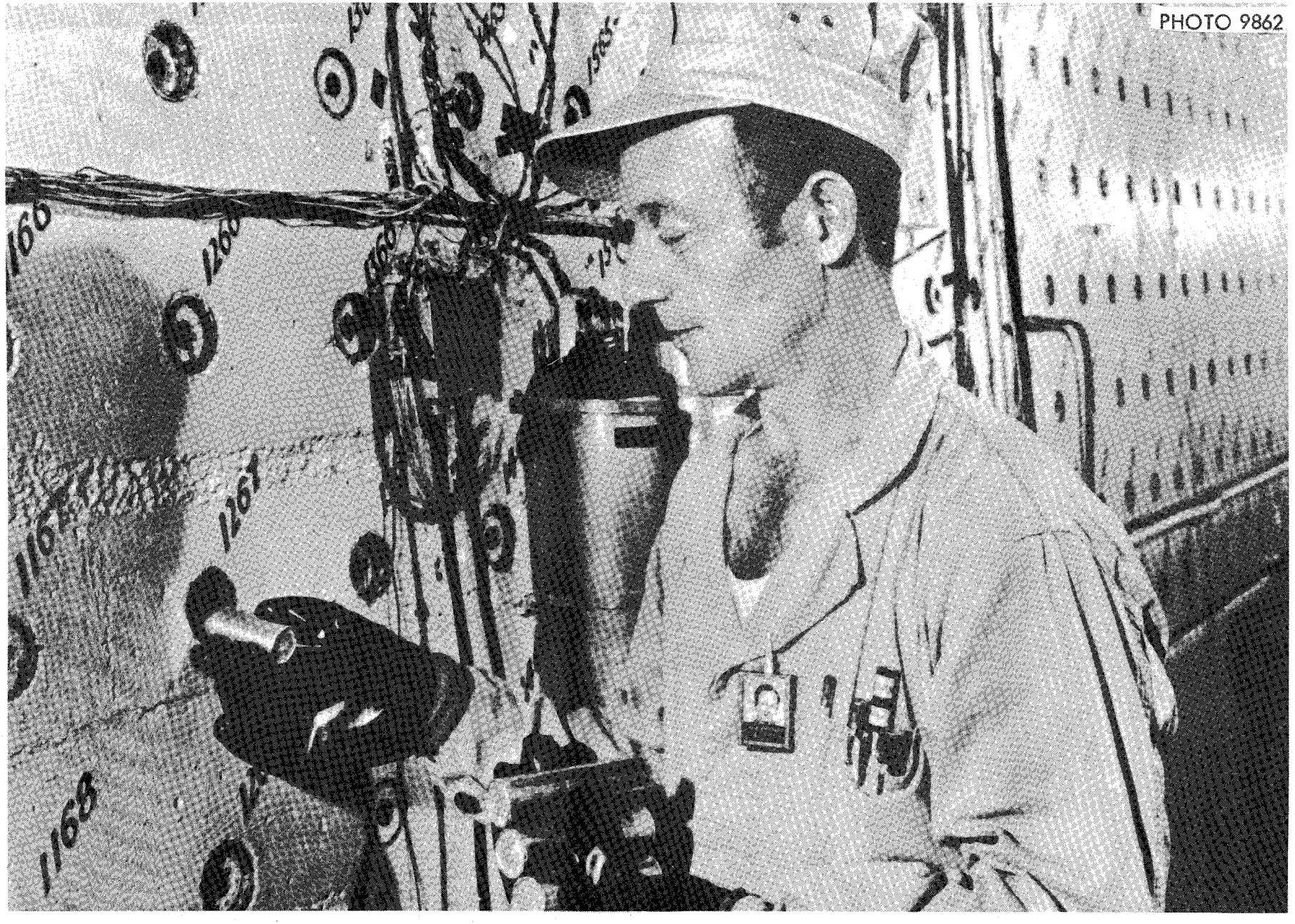


Fig. 5a. Operator Inserting Fuel Elements into a Fuel Channel. After being Inserted by Hand, They Are Centered in the Core by Pushing Them into Place with Measured Steel Rods.

air exhaust system with fission products. The more serious result of failure is the possible sealing off of the channel by the swelling element jacket as the oxide volume increases. When the air flow ceases, other elements in the channel fail due to high temperature. This has occurred twice, once in 1947 and again in 1948. The first instance involved 13 elements and the second 5 elements. In both instances the fuel channel was so scarred by the discharging techniques that had to be used that they were not suitable for further use as fuel channels and were diverted to other usages. Since that time, a weekly inspection of all fuel channels has prevented the recurrence of such multiple failures.

During 1952 the fuel was changed to a bonded type. This bond is an aluminum-silicon eutectic which fills the space between the aluminum jacket and the uranium metal. Its purpose is to separate the uranium from the aluminum and still provide good heat transfer. It also tends to slow the rate of a failure by minimizing the area of uranium that might be exposed to air if a tiny hole should develop in the aluminum jacket. Since one of the causes of failure in the unbonded elements was interaction between the uranium and the aluminum if the temperature were in the 200° to 300° range for long periods, it was hoped that the bonded elements could be operated above this temperature allowing the power level to be increased. This, however, has not proved to be true; the element jackets are now operating at a higher temperature relative to power level than before due to the better heat transfer from the uranium to the aluminum jacket through the bond under the thermocouple bead and failures occur at an objectionable rate above 300°C indicating that there is still sufficient contact between the aluminum and the uranium to cause trouble. With the unbonded elements, the jacket temperature could be operated continuously at 245°C without experiencing serious failure rates.

The power level at this temperature averaged 3.6 Mw with frequent operation at 4.0 Mw. With the bonded elements, experience has shown that 285°C as shown by the thermocouple should not be exceeded for long periods. The average power level now is 3.5 Mw to stay below this limit. The gain has been only in the frequency of failure rate which is less than half that with the unbonded elements. A temperature traverse along the length of a fuel channel is shown in Fig. 6.

The quantity of normal uranium required for critical loading was almost exactly 30 tons. The present loading is 54 tons to provide sufficient excess reactivity for the benefit of experiment and radioisotope target materials in the reactor as well as to provide for the poisoning effect of the Al-Si bonding material in the present elements (Figs. 7 and 8).

The reactor is operated on a round-the-clock basis with the only routine shutdown being from 5:00 a.m. to about 3:30 p.m. each Monday. The continuous operation is necessary for the varied experiments being conducted as well as for a routine production of radioisotopes. The percentage operating time since the reactor first attained critical loading is in excess of 90%. During the routine shutdown, each fuel channel is inspected for any evidence of failure, changes in research equipment made, radioisotope targets are removed and inserted, and required maintenance performed. Close scheduling is required to accomplish this work in the allotted time and planning always starts each Thursday for the following Monday.

Other shutdowns are scheduled as required for emergency maintenance, of course, and shutdowns for special research or radioisotope production work are sometimes arranged with the permission of other research personnel using the reactor.

At the standard operating power level of 3.5 Mw the maximum thermal

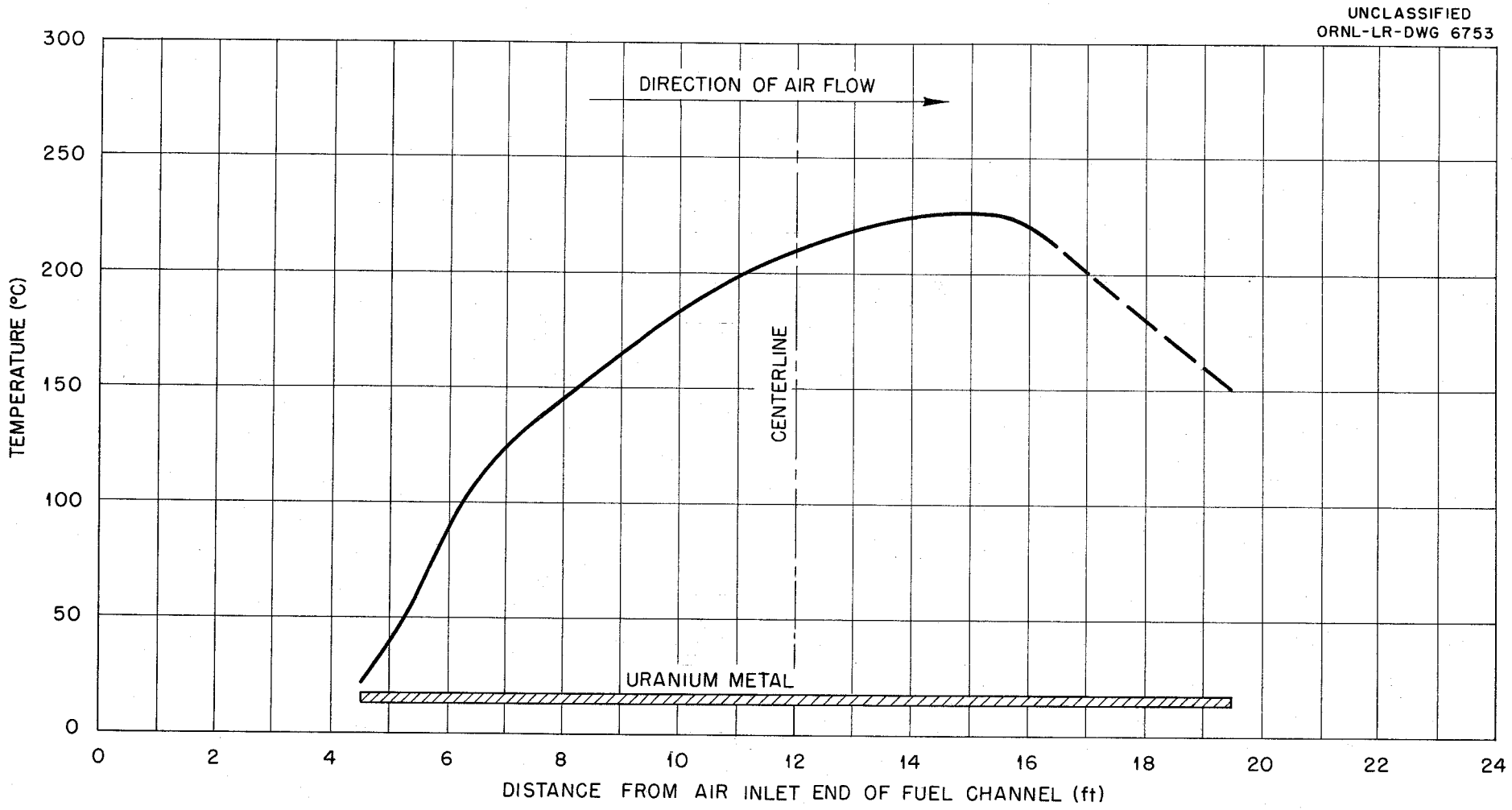
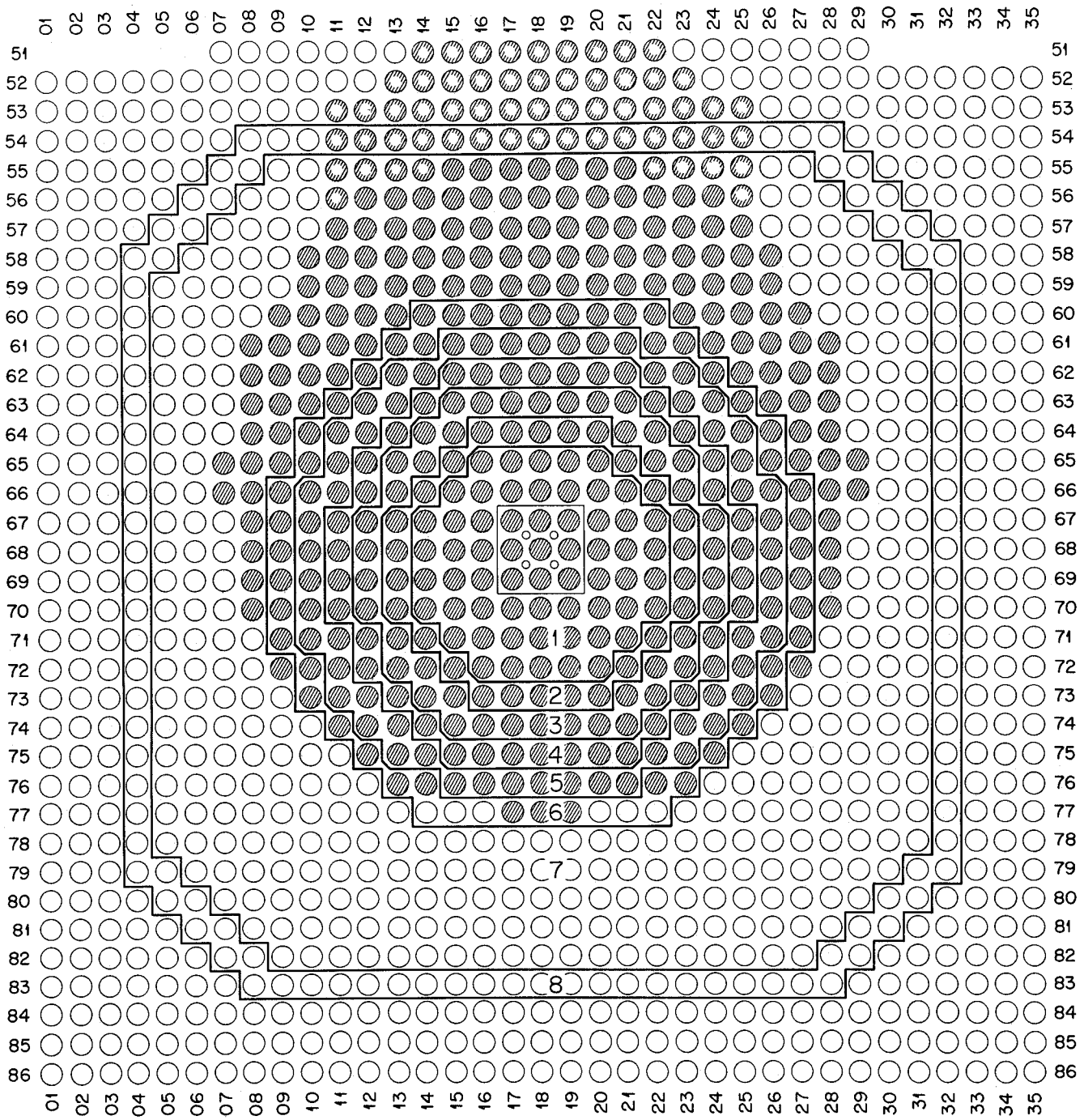


Fig. 6. Temperature Traverse Through a Fuel Channel Showing that the Peak Temperature is Downstream from the Reactor Center.

UNCLASSIFIED
ORNL-LR-DWG 6754



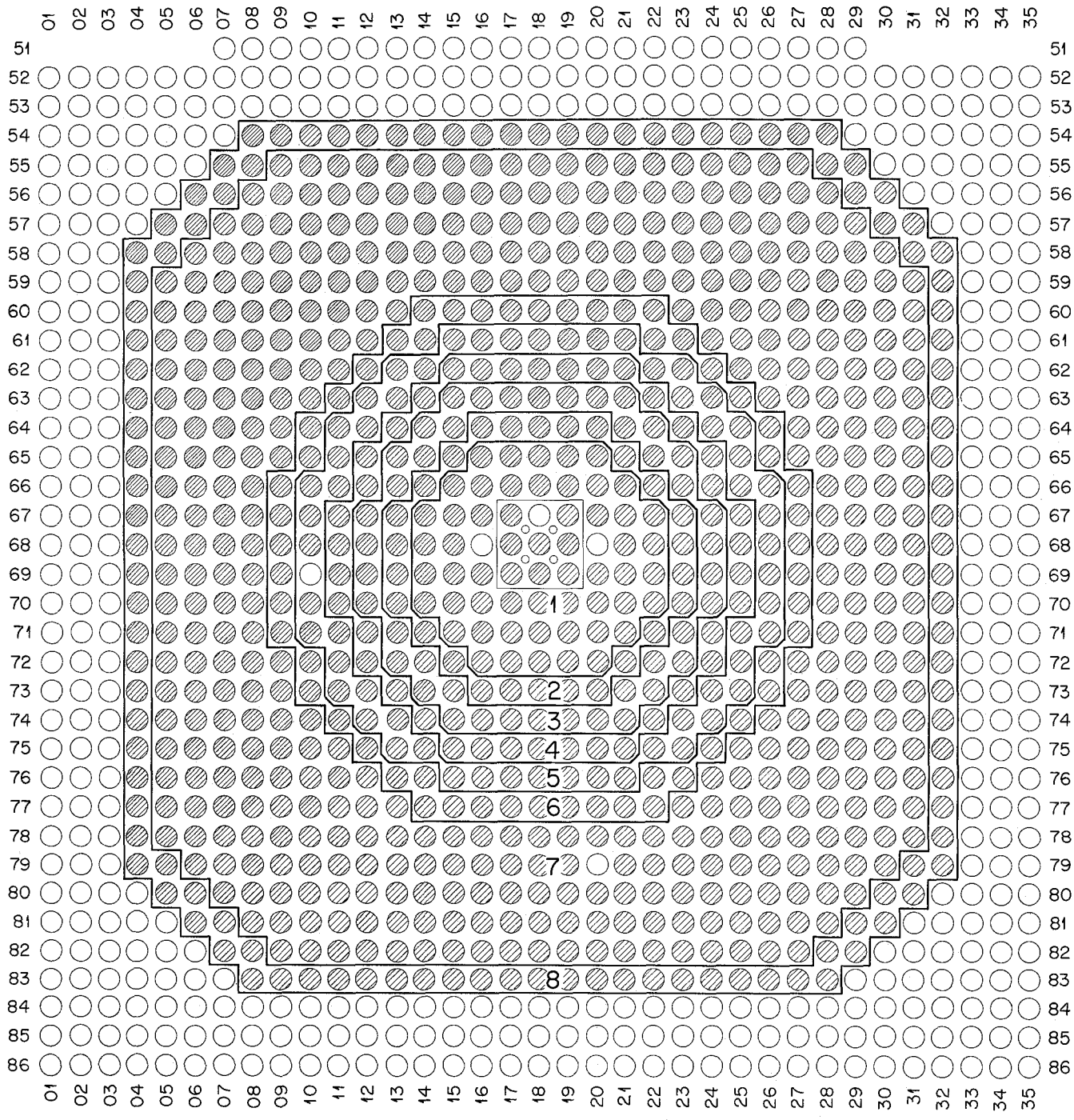
ORIGINAL LOADING

30 ELEMENT ROWS TO PROVIDE HIGHER NEUTRON FLUX IN VERTICAL THERMAL COLUMN

65 ELEMENT ROWS

FIG. 7.

UNCLASSIFIED
ORNL-LR-DWG 6755



ZONE	NUMBER CHANNELS	NUMBER SLUGS
1	60	41
2	26	45
3	42	51
4	38	51
5	58	54
6	54	54
7	438	54
8	114	47

Fig. 8. OGR Fuel Loading Pattern.

neutron flux is 1.1×10^{12} $n/cm^2 \cdot sec$. Up until 1952 the power level was generally kept as high as possible without exceeding the temperature limitation of the fuel jackets. This caused several different power levels to be existing even during the course of one day ranging from 3.4 to 4.2 Mw and caused a lot of extra work for the research personnel in normalizing their data. This was a case of doing something just because it had always been done and was a hangover from the old plutonium production days when it was required to operate at the highest power level possible. The present operation at a standard power level has somewhat simplified the work of the research personnel but even now very precise work has to be specially monitored at the exact facility being used.

Due to the size of the reactor and the placement of the regulating rods, slow fluctuations of greater than one percent can take place in one day in any one location without any change in the over-all power output of the reactor. As the temperature of the coolant (atmospheric air) increases from shortly after daylight until about 2:30 or 3:00 p.m. when the peak temperature usually occurs, the regulating rods move outward to compensate for the over-all temperature rise in the reactor. This causes the peak flux in the reactor to move toward the center of the reactor in the direction the rods are traveling. The opposite effect begins shortly thereafter until about 10:00 p.m. after which nearly a stationary flux pattern exists until daylight. For this reason, research personnel needing precise neutron flux control prefer data taken at night unless they have automatic compensation in their equipment, which many have at present.

Also, due to having two general purpose pneumatic exposure tubes and a vertical hole through which relatively large samples (up to 18-in. long by 2 1/2 in. in diameter) can be inserted and removed without shutting down,

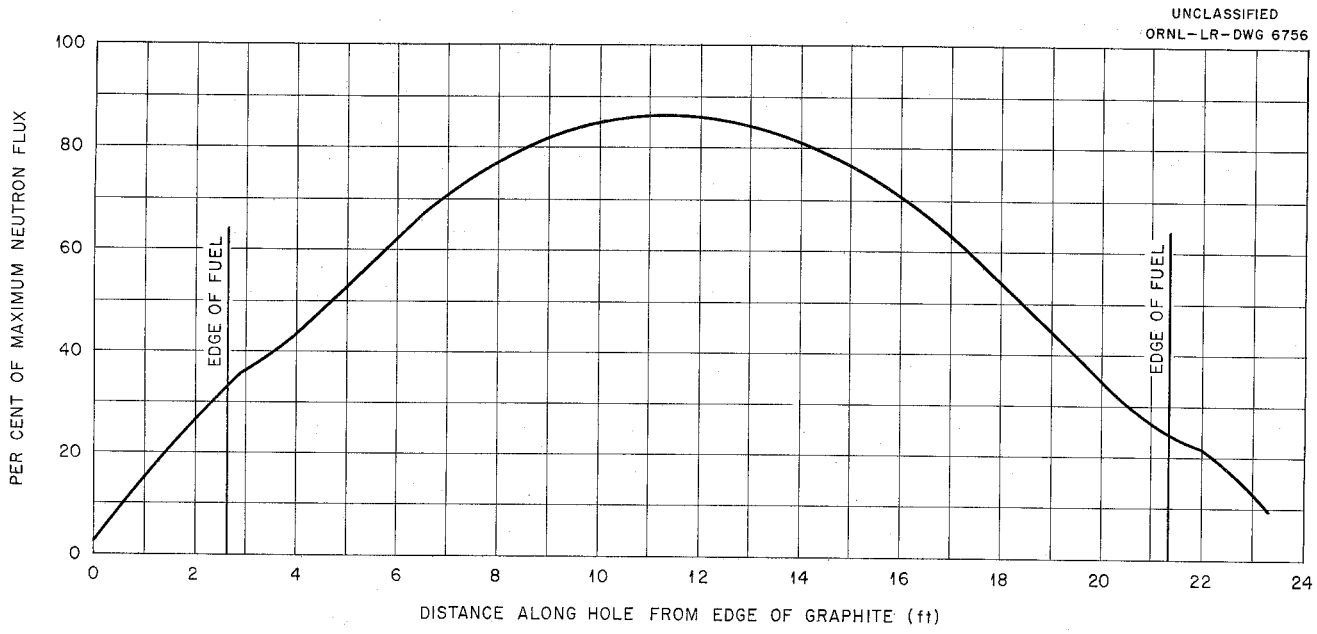


Fig. 9. A Typical Thermal Neutron Traverse Through One of the 4-in Square Experiment Holes.

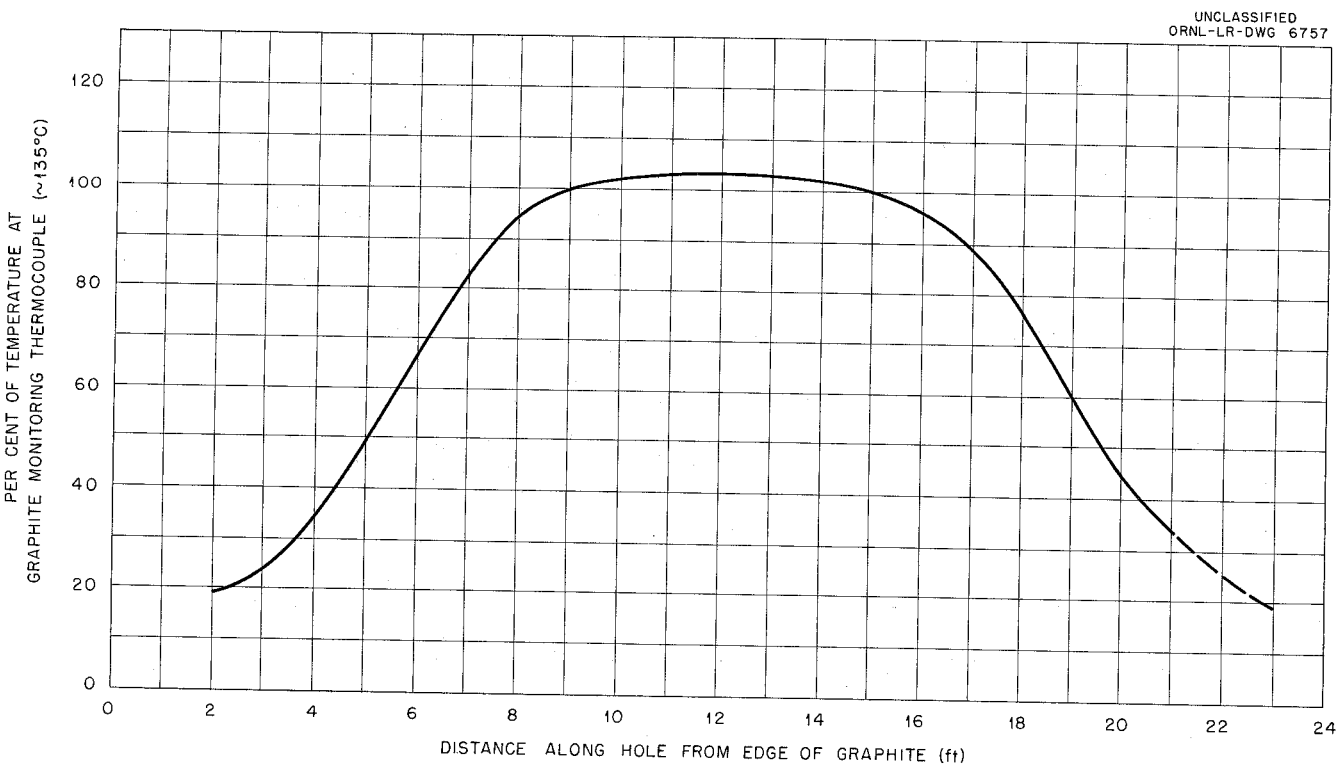


Fig. 10. A Typical Temperature Traverse Through a 4-in. Square Hole Showing a Flattening at the Center.

many brief flux fluctuations occur all during the day as these facilities are used. Announcements, five minutes prior to such fluctuations, by way of a public address system gives each research man time to prepare for them or raise objections to their being done. Another such fluctuation occurs during a major regulating rod adjustment which must be made at any time the two regulating rods become unbalanced as much as 10 in. This occurs due to having fine control movement on only one rod. When this rod has been moved inward or outward 10 in. farther than the one-speed rod, they are rebalanced by moving one in one direction and the other in the other direction simultaneously. At best, some fluctuation occurs and this, too, is announced five minutes ahead of time. The balancing procedure minimizes distortion of the flux pattern in the reactor. This technique need only be used during manual operation. While the reactor is in automatic control, the rods can be adjusted to be an even distance out at all times.

Since the total poisoning value of the five shutdown rods is only 2.7% $\frac{\Delta k}{k}$, the loading is limited to half this value for the cold reactor. Due to the low power density, roughly 600 watts per cubic foot, the xenon-135 poisoning is not severe only 0.2% $\frac{\Delta k}{k}$. The combined temperature coefficients amount to 0.4% $\frac{\Delta k}{k}$, making a total of 0.6% $\frac{\Delta k}{k}$. An allowance of about 0.3% $\frac{\Delta k}{k}$ is cancelled by the regulating rods to provide flexibility of control and for experiments of a temporary nature. Another 0.5% $\frac{\Delta k}{k}$ is used by permanently installed experiment facilities, giving a total of approximately 1.4% $\frac{\Delta k}{k}$ excess reactivity for the clean cold reactor. The only possibility of prompt critical occurring is during startup and this is prevented by a careful slow startup procedure which requires fifteen minutes.

TABLE IV SUMMARY OF GRAPHITE REACTOR INFORMATION

Size of graphite		24 ft x 24 ft x 24 ft 4 in. high
Thickness of concrete shield		7 ft
Number of fuel channels		1248
Number of 1.6-in.-diameter holes through central part of core		4
Type of fuel		Normal uranium
Dimension of aluminum-clad fuel element (slug)		1.1 in. diameter by 4.1 in. long
Weight of uranium metal per slug		1170 gm
Thickness of aluminum cladding		.035 in.
Bonding material between Al and U		Al-Si eutectic
Number of fuel channels containing fuel		821
24-slug rows	10	
41-slug rows	51	
45-slug rows	23	
47-slug rows	115	
51-slug rows	79	
54-slug rows	543	
Number of 4-in.-sq openings		56
Number of horizontal through holes (4-in. sq)		22
Number of horizontal half holes (4-in. sq)		6
Number of vertical holes (4-in. sq)		6
Number of 4-in.-sq horizontal through holes used for control and shutdown rods		4
Number of 4-in.-sq horizontal openings for experimental and target exposures		47
Number of 4-in.-sq vertical holes for shutdown rods		3
Number of 4-in.-sq vertical holes for experiments and target exposures		3
Other experiment facilities		14

TABLE IV (continued)

Total reactivity value of shutdown rods	2.7% $\frac{\Delta k}{k}$
Total reactivity value of control rods	1.1% $\frac{\Delta k}{k}$
Approximate amount of reactivity absorbed by experiments and radioisotope samples	0.5% $\frac{\Delta k}{k}$
Weight of uranium in reactor	54.11 tons
Maximum thermal neutron flux at 3.5 Mw	1.1×10^{12}
Cadmium ratio	20
Average gamma photon energy	0.9 mev
Maximum gamma photon flux (at average energy of 0.9 mev)	5.13×10^{11} $\gamma/cm^2 \cdot sec$
Maximum gamma dose rate	9.2×10^5 R/hr
Ratio of gamma flux to thermal neutron flux	0.50
Standard reactor power level	3.5 Mw
Average graphite temperature at center of reactor	~135°C
Average metal temperature at hottest region	~270°C
Average neutron flux (based upon power level and amount of fuel)	5.0×10^{11} $n/cm^2 \cdot sec$
Ratio of maximum thermal neutron flux to average thermal neutron flux	2.24
Volume of active portion of reactor	5,933.4 cu ft
Power density inside active portion	587.5 watts/ft^3
Volume of coolant air	120,000 cfm
Negative pressure inside reactor	~29 in. wg
Operating metal temperature coefficient	-.29 inhour/°C
Operating graphite temperature coefficient	-.77 inhour/°C
Barometric pressure coefficient	-.38 inhour/mm Hg
Xenon poisoning coefficient	-25 inhours/1000 kw

APPENDIX III

INVESTIGATION OF METHODS OF OBTAINING GRAPHITE ANNEALING TEMPERATURES IN THE OGR

Injection of High-Temperature Air Through the Fuel-Charging Tubes

In this method, the annealing would be performed with the reactor shut down and both main cooling air fans off. High-temperature (near 400°C) air would be injected into 100 fuel channels at a time by way of the fuel channel access tubes. The air would be supplied by a blower or compressor and heated by a gas or an electric furnace to the desired temperature. Insulated pipes would convey the air to the manifold for distribution to vacuum-jacketed probes which penetrated the fuel-charging tube openings and discharged into the fuel channels. Vacuum jackets on the probes were necessary, as determined experimentally, to prevent overheating of the concrete shield wall. This system possesses the following undesirable features:

1. Each 100-channel annealing operation would require a long shutdown.

2. The complete annealing operation would be extended over a period of several months because several duplicate 100-channel operations would be necessary to effect the desired annealing in all of the region containing stored energy in excess of the graphite specific heat curve.

3. A complex manifold system would be required for the vacuum-jacketed tubes, and the development of a leak in the system might result in overheating of the concrete shield.

4. The complete system would be very large and expensive.

Single-Pass System Utilizing Gas or Electric Heaters in the Inlet Air Duct

In this method, a small per cent of air would flow in the normal manner.

Electrical or gas-fired heaters located in the inlet-air duct would preheat the air sufficiently to obtain the required graphite annealing temperatures with the reactor shut down. The primary objection to this system is that the resulting front-face shield temperature would be excessive.

Recirculating System Utilizing External Electrical or Gas-Fired Heaters

This system would involve the installation of an interconnecting duct between the present inlet and exit ducts which would incorporate a blower for recirculating the air and a heater unit for producing the desired air temperature. In this way, a very flat temperature distribution could be obtained in the graphite stack. However, early investigation showed that the resulting concrete shield temperatures would be prohibitive; therefore, this system was dismissed. Later, engineering studies of this system were again made upon the basis of later stored energy data showing that effective annealing could be accomplished at moderately low temperatures.

Reversed Air-Flow System Utilizing Nuclear Heat

This system is described under the annealing method in the main report. It was selected from the systems studied on the basis of the following advantages:

1. The cost of this system is considerably less than for any of the other systems studied.

2. The system is compatible with the present system; that is, once installed, reversed or normal flow conditions can be established arbitrarily with relative ease.

3. The protection of the concrete shielding from thermal damage presents

a less difficult problem.

4. Shutdown time required for installation and for performing the annealing operation will be appreciably shorter.

5. Detection of possible slug ruptures will be easier because of the fresh fission product generation, since the reactor will be operating at an appreciable power level during the annealing.

APPENDIX IV

CONTROL AND SAFETY MECHANISM

General

All seven of the control rods used in the Oak Ridge Graphite Reactor are of the absorber type. Three of these are of identical design, utilize cadmium as the absorber material, and move vertically in and out of the top of the reactor, while the others are four identical boron-steel rods which move in a horizontal direction. The vertical rods are designed as safety rods and are not control rods in the usual sense since they can be stopped in only two positions, fully inserted or fully withdrawn. The actual control is provided by the horizontal rods, two of these being designed as shim and two as regulating rods.

Operational Procedure

Following the established startup procedure, the operator withdraws rods sequentially: first the safety, second the shim, and finally the regulating rods. Taken in steps, the operator cocks (withdraws) two safety rods, Nos. 8 and 9, first. At the completion of this step, a permissive is established automatically within the control system, and the operator can cock the third safety rod, No. 7. Having withdrawn the No. 7 safety rod the next step is to select one or the other of the shim rods and withdraw it. Note that only one shim rod at a time may be selected, not both. The established procedure calls for the No. 6 shim to be fully withdrawn, and then the No. 5 is withdrawn. This last rod is not, however, withdrawn all the way because it will be used later, when the reactor reaches its scheduled power level, as part of the servo level control system and will, of course,

need to be capable of adding at least some reactivity to compensate for temperature effects, etc. At this point the reactor is not yet critical, and the regulating rods are still fully inserted. The final step then is to withdraw these two, usually by jogging them alternately, while watching the neutron level and period indicating recorders mounted above the rod-control switches. The rods are withdrawn in equal increments to secure the most uniform possible flux distribution in the reactor core. After the reactor has been eased up to the required power level by withdrawing the regulating rods in short steps, the control will be switched over to the servo system to hold that level. As conditions change within the reactor, the servo usually works the No. 5 shim rod out to near its fully withdrawn position. When this position is reached, the reactor operator will jog the regulating rods out slowly and the servo will react by inserting the No. 5 shim rod to compensate, thus restoring the original range of control assigned to it.

Safety Rods

Each of these has individual driving means consisting of a permanently attached steel supporting cable fastened at the upper end to a winding drum which, in turn, is coupled through a magnetic clutch to a drive motor and spring-set brake. To withdraw a safety rod, power is applied to the clutch to engage it, to the motor brake to release it, and to the motor which turns the winding drum. Upon reaching the cocked position, motor and brake power are cut off automatically and the brake sets, thereby locking the drive and the rod in position. To insert the rod, either for a routine or a scram shutdown, the clutch power is interrupted, the clutch releases, and the rod falls, under the influence of gravity. Just before the rod reaches its fully

inserted position, the winding drum contacts a braking disk which dissipates the kinetic energy in the system, bring the rod to a quick, smooth stop.

The safety rods are square in cross section, 3 1/2 in. on a side and 9 ft long. They are hollow steel shells, completely lined with 1/16-in.-thick cadmium sheet except for the last 6 in. at the ends; both ends are closed. The tops are attached to lifting cables; and the bottoms are fitted with coil springs and plunger-type shock absorbers, which affords some mechanical protection to the rods and seats if the lifting cables should fail. When fully inserted, the axial center of the rods are in a horizontal plane through the center of the reactor core.

Each safety rod is worth approximately 185 inhours and its withdrawal time is 10 sec. Insertion time is very much shorter, since the response to a scram condition is more important. A safety rod falls to the fully inserted position within approximately 1 sec after the clutch is de-energized.

Shim Rods

The four horizontal rods are constructed of boron steel (1 1/2% boron). They have square cross sections 1 3/4 in. on a side and are 200 in. long. They are driven by means of racks and pinions and travel approximately 19 ft between the fully inserted and withdrawn positions. The two rods designated as shim rods are located in higher neutron flux regions and are the most effective rods in the reactor, being worth approximately 235 inhours each.

The shim rods are operated through step-up gearing to another rack-and-pinion drive which is operated by an oil cylinder and piston having a 6-ft stroke. An electrically driven, constant speed, variable-output hydraulic pump supplies the pressurized oil for rod positioning during operation. For scramming, an auxiliary accumulator system is used as the pressurized-oil

source. This auxiliary system includes two independent accumulators and oil pumps. The control of the system is arranged to maintain, at all times during reactor operation, sufficient quantities of oil at adequate pressures to fully insert both shim rods in case of a scram, regardless of the initial position of the rods. Transfer from the "operate" to the "scram" condition is effected by a solenoid-held, spring-actuated, oil valve which removes the operating oil pressure from both sides of the shim rod pistons and immediately supplies accumulator oil pressure to the insert side of the cylinder. Flow rates are such that full insertion of the shim rods takes approximately four seconds maximum.

The No. 6 shim rod, which is usually fully withdrawn during startup and remains so during operation, has its drive adjusted to withdraw at approximately 5.7 in./sec and to insert at approximately 11.4 in./sec. In selecting a withdrawal speed, the worth of the rod and other factors must be considered. A compromise is reached which will allow a reasonably short startup time and a reasonably safe rate of increasing the reactivity in the core. For insertion, however, the situation is different. The operation must insure safety; and it may be desirable to insert absorber very rapidly, to limit an excursion, for example. Thus, a higher speed is selected.

The No. 5 shim rod, on the other hand, is adjusted to withdraw and insert at relatively low speeds, since it is controlled by the rather slow acting servo system. Actual measurement shows 1.42 in./sec for withdrawal and 1.3 in./sec for insertion to be speeds which work well with the servo.

Regulating Rods

These are numbered 1 and 2 and both are motor driven through suitable step-down gearing. The drives are provided with spring-set, magnetically

released brakes to prevent overtravel. The rods are used to bring the reactor up to criticality and power, as was noted above, and for control during manual operation. The No. 1 rod travels 6.9 in./sec in each direction and provides relatively close control. The No. 2 rod has two motors, one of which drives the rod 4.76 in/sec for closer control. The other motor provides 0.44 in./sec travel speeds in either direction and permits very smooth manual control of the reactor. In addition, rod No. 1 is slightly more effective than No. 2, being worth 200 inhours compared to 180 inhours. Established procedure requires the operator to fully insert both rods immediately following every shutdown in order to be ready for the next startup.

Safety System

All troubles developing in the reactor, its auxiliaries, or experimental installations at the various facilities make their presence known in one or more of three ways, depending upon their seriousness. Minor ones sound local alarms only. More serious ones trip appropriate annunciators in the control room and, in some cases, at the trouble site also, while the serious, or potentially serious, ones scram the reactor immediately and automatically upon occurrence. In a few cases, the operator is warned when potentially serious conditions obtain; if no corrective action is taken promptly, a scram ensues.

No provision is made for automatic setbacks or period scrams. Xenon, for example, is no problem so there is no minimum time interval in which the reactor must be restarted following a shutdown to clear any trouble. As was pointed out in the startup procedure above, the reactor becomes critical only after the safety and shim rods have been withdrawn. No true periods occur during startup except as a result of regulating rod withdrawal.

Permissives in the control circuits of these rods stop further withdrawal for periods shorter than approximately 24 sec. When in automatic operation, the servo system could conceivably fail in such a way as to add reactivity at a slow rate for about 60 sec and, thereby, produce a fairly short positive period. If the operator did not observe the period or power level indications provided on the control panel, or took no suitable corrective action, the reactor would be scrammed by one of the safety instruments which limit the reactor power level.

Local alarms for experimenters have no direct bearing on reactor operation and will not be considered here. Two typical alarm-before-scram variables are graphite temperature and accumulator oil level. In both cases, the situation becomes serious only if the graphite temperature continues to rise or if the accumulator oil level continues to fall. In almost all situations, it is possible to correct the difficulty before the scram condition obtains and thus avert a reactor shutdown. Warning-before-scramming signals may also come from experiments, and since these change from time to time they will not be listed here.

The reactor may also be scrammed manually from a number of locations. Two scram switches are located on the control panel, others at building exits, and the remainder near various experimental installations for use in the event of emergencies.

At the time the control was designed, it was considered necessary to provide the reactor with a number of safety rods which would be held cocked ready for immediate insertion at any time even if the reactor was "down." The shim rods were designed to be completely effective for all possible shutdown and control conditions and were to be depended upon for routine use. This gave rise to the two manual scram controls noted above as being on the

control panel. One of these, designated No. 1 scram switch, caused all safety and all shim rods to insert, while the No. 2 scram switch scrammed only the shim rods. In distributing the automatic scrams, some were placed in each scram circuit, the decision presumably being based on the probable seriousness of the indicated situation or the length of the downtime estimated to be needed to effect corrections. Originally it was expected that the safety rods would be withdrawn to their cocked position without much delay following an all-rod scram. With time, it became apparent that the rods were capable of controlling fuel loadings much larger than any in use or contemplated. On this basis, three rods were taken out of service, No. 10 safety and Nos. 3 and 4 shims. It is of interest to note that the vacant holes were converted into three very valuable experimental facilities. When the three rods were removed, the control system was revised to operate in the manner described above; and the No. 7 safety was reconnected to serve as a shim rod and is now considered as such. Although the system of two kinds of scrams is still in use, the practice of maintaining the two remaining safety rods in cocked positions during shutdown has been discontinued. At present the all-rod scram will be initiated under any of the following conditions:

1. If any of the three safety neutron level monitors indicate a power level greater than 4200 kw.

2. If the thermopile monitor indicates a power level greater than 4200 kw.

3. If the graphite temperature in the peripheral regions of the moderator exceeds 50°C.

4. If the differential pressure across the reactor (proportionate to the coolant-flow rate) drops below 8 in. H_2O, or approximately 50% of the normal full-power value.

5. The development of any of several undesirable or unsafe conditions in the experiment apparatus at hole 10 or hole 12.

6. If the nuclear radiation intensity at the core exceeds a preset, below tolerance value.

7. If any manual scram switches are actuated.

8. If the No. 1 scram switch on the control desk is actuated.

9. If the purchased electrical power supply is interrupted.

The shim-rod type of scram is initiated under any of the following conditions:

1. If the inlet coolant flow drops below 2000 lb/min.

2. If any fuel element cladding temperature exceeds 185°C.

3. Upon the development of any of several undesirable or unsafe conditions in the experiment equipment at hole 11 or hole 51.

4. If the oil level in the accumulators decreases to a value considered as the minimum quantity which can be safely depended upon to fully insert the shim rods.

5. If the electrical power supply to the shim rod "bus" is interrupted.

6. If the No. 2 scram switch on the control panel is actuated.

Scram demands always override any other control action requested.

Nuclear Instrumentation

Seven chambers and six boron thermopiles are used for monitoring the flux level in the reactor. Three uncompensated ion chambers and the thermopiles, all effective only in the power range of operation, are used as part of the level safety systems to prevent overpower operation of the reactor. The six thermopiles are connected in series, and their output is fed to a single-point strip-chart recorder calibrated to read power in kw. Contacts

actuated by the recorder scram the reactor if the indicated power is excessive (>4200 kw). This constitutes one safety channel. The three uncompensated chambers with their associated amplifiers constitute three additional independent safety channels, each set to scram the reactor when the 4200 kw power level is exceeded. Indication only is provided.

Three of the remaining chambers are of the compensated type with their output displayed on the control panel and may be considered as part of the control system. One of the chambers feeds a galvanometer which, by means of multiple shunts, is used to indicate neutron flux level over a range from near startup values up to and including the power range. This channel was used exclusively by the operator for determining the reactor flux during startup and full power operation for many years.

The second compensated chamber is used with a recording microammeter to indicate flux level, and both are part of the servo system installed in 1954 to automatically control the reactor when at power. The recorder is mounted in the control panel but is useful for reference only in the power range.

The third compensated chamber feeds a Log N and Period channel and provides flux level and period information over a six-decade range from full power downward. Both types of information are recorded, and the period channel is one of two used to inhibit regulating rod withdrawal, as noted earlier.

The seventh chamber is a uranium fission type which feeds a counting channel with period auxiliary functions. The chamber is provided with a positioning mechanism so that it is capable of furnishing flux data from the power range down into the startup range. Two recorders are provided, one showing the log of the count rate and the other the reactor period.

This period recorder is the other of two which inhibit regulating rod withdrawal while short periods exist in the reactor. The count-rate channel operates at even lower flux levels than does the galvanometer channel and is useful during startups.

Process Instrumentation

Only those instruments capable of scramming the reactor are of interest here. These include:

1. A circular-chart recorder for the cooling-air ΔP across the reactor. This instrument continuously compares the pressure at the inlet and exit of the reactor core as an indication of proper flow. Low ΔP results in a scram.

2. A circular-chart recorder for monitoring the inlet air flow rate. Too low a rate scrams the reactor.

3. A strip-chart recorder monitoring one slug in the high-temperature zone of the core. An excessively high slug temperature will scram the reactor. Any one of a large number of slugs may be selected for monitoring by this recorder, the remainder being checked regularly by other means for comparison.

Block-Out Panel

It is safe, practical, and highly desirable to be able to start and run the reactor with certain of the automatic scram channels rendered inoperative. For example, an experiment installation which scrammed the reactor because of some difficulty might readily be changed from an operational hazard and yet might require days or weeks of repair work to put it back in service. Under such circumstances, it should be possible to restart the reactor with only a minimum of delay. The Block-out Panel contains a group of manually operated switches which permit the blocking of selected individual scram contacts to handle such situations. Warning lights are incorporated in this panel so that

any responsible person can check the situation at a glance. For the most critical scram channels, the switching is arranged to limit the number of blocked circuits.

The following scrams may be blocked at any time:

1. core hole monitron,
2. most experimental installations considered permanent.

The following scrams cannot all be blocked at any one time but may be blocked individually by setting selector switches.

Switch A:

1. cooling air flow or
2. ΔP across reactor or
3. slug temperature.

Switch B:

1. safety level Hole 33 or
2. safety level Hole 32.

Switch C:

1. thermopile channel or
2. safety level Hole 36.

APPENDIX V

GRAPHITE OXIDATION DATA

To determine the possible effects of the reactor environment upon combustibility of graphite, oxidation rates were determined by using samples of graphite which were mined from several regions of the reactor moderator, including the high-stored-energy portion. The results of some of these measurements are presented in Table 1. Apparently, exposure to atmospheric air produces some increase in the oxidation rate of the surface layers of graphite, being more profound at the higher temperatures. However, the resulting rates are very little different from those of virgin graphite and, therefore, are considered inconsequential.

TABLE 1. EFFECT OF CONTAMINATION ON OXIDATION RATE OF OGR GRAPHITE AT VARIOUS TEMPERATURES

Channel No.	Sample No.*	Oxidation Rate (mg of carbon per gram of sample per min)				
		At 300°C	At 400°C	At 500°C	At 600°C	At 700°C
1166	A	0.001728	0.005184	0.05184	0.5789	5.510
	B	0.003188	0.007971	0.04384	0.5939	5.118
	C	0.01753	0.008767	0.05076	0.3414	3.138
1266	A	0.003259	0.006519	0.03911	0.7252	8.022
	B	0.02287	0.01191	0.08718	1.072	8.833
	C	0.01253	0.008716	0.05447	0.4794	5.218
1366	A	0.007987	0.04171	0.2396	2.0855	8.6527
	B	0.017148	0.05716	0.07275	1.3926	7.6387
	C	0.017898	0.01193	0.06231	0.7956	7.2920

*Sample A contains a surface from a fuel channel
Sample B contains a surface from hole 17.
Sample C is from the mid-region.

APPENDIX VI

SUMMARY OF OGR INSTRUMENTS

Present Instrument Complement of the OGR

Designation	No. of Channels	Sensing Element	Purpose	Type of Monitor	Action
Safety channels	3	Uncompensated, boron-coated, ionization chamber	Limit power level and indicate power level	Brown Instrument Co., strip chart, multipoint recorder. Records all three channels	Scram at 4200 kw
Log N chamber	1	Compensated, boron-coated, ionization chamber	Indicate neutron level from 10^{-6} x full power to 1 x full power	Leeds & Northrup Co., strip chart, pen type, indicating recorder, logarithmic scale	None
Log N period		Same chamber as Log N	Indicate and limit the rate of increase of power level	Leeds & Northrup Co., strip chart, pen type, indicating recorder, logarithmic scale	Prohibits rod withdrawal if the period is less than 22 sec
Servo chamber	1	Compensated, boron-coated, ionization chamber	Provides automatic controller with signal proportional to the neutron level	Leeds & Northrup Co., strip chart, pen type, indicating recorder Leeds & Northrup Co., servo controller	Maintains neutron level at demand value
Galvanometer chamber	1	Compensated, boron-coated, ionization chamber, nitrogen flow required	Used as a check instrument both in latter start-up range and at power	Leeds & Northrup Co., reflecting, current type galvanometer	None

Present Instrument Complement of the OGR (continued)

Designation	No. of Channels	Sensing Element	Purpose	Type of Monitor	Action
Count rate	1	Fission chamber	Indicates neutron level over all ranges	Leeds & Northrup Co., strip chart, pen type, indicating recorder	None
Count-rate period	1	Same chamber as count-rate meter	To indicate the rate of increase of neutron level in all ranges	Leeds & Northrup, Co., strip chart, pen type, indicating recorder	Prohibits rod withdrawal if the period is less than 22 sec
Thermopiles	6	Boron thermopiles, all 6 series-connected	To obtain a composite signal proportional to the thermal neutron flux over a large portion of the reactor	Brown Instrument Co., strip chart, pen recorder	Scram at 4200 kw
Regulating rod position indicator recorder	2	Linear-wound potentiometer, geared to regulating rod drives	To provide a record and indication of the regulating rod positions	Leeds & Northrup Co., strip chart, pen recorder, prints both rod positions	None
Regulating rod position indicators	2	Selsyn transmitter geared to regulating rod drives	To indicate the positions of No. 1 and No. 2 regulating rods	Selsyn receiver drives needle on circular dial indicator	None
No. 5 shim rod position	1	Selsyn transmitter geared to rod drives	To indicate the position of No. 5 shim rod	Selsyn receiver drives needle on circular dial indicator	None
	4	Microswitches, cam operated by rod drive	Indicates interval of rod travel in which rod is located	Panel lights operated by micorswitches	None

Present Instrument Complement of the OGR (Continued)

Designation	No. of Channels	Sensing Element	Purpose	Type of Monitor	Action
No. 6 shim rod position	8	Microswitches, cam operated by rod drive	To indicate the interval of rod travel in which the rod is located; also, upper and lower limits	Panel lights actuated by microswitches	None
Safety rod position	3 each per rod	Microswitches, operated by rod withdrawal mechanism	To indicate the upper and lower limits and center of travel position of No. 7, No. 8, and No. 9 rods	Panel lights actuated by microswitches	None
Inlet-air temperature	2	Nickel thermohms, 1 inside building, 1 in inlet-air duct	To indicate the temperature of air leaking into the reactor; also normal inlet-air in degrees Fahrenheit	Leeds & Northrup Co., micromax, strip chart, pen type, indicating recorder	None
Inlet-air temperature	2	Nickel thermohms located in inlet-air duct	To indicate the inlet-air dry-bulb, and wet-bulb temperatures in degrees centigrade	Leeds & Northrup Co., micromax, strip chart, pen type, indicating recorder	None
Exit-air temperature	1	Nickel thermohm located in the exit-air duct	To indicate the exit-air temperature	Leeds & Northrup Co., micromax, strip chart, pen type, indicating recorder	None

Present Instrument Complement of the OGR (Continued)

Designation	No. of Channels	Sensing Element	Purpose	Type of Monitor	Action
Fuel cladding temperature	1	Thermocouples attached to fuel cladding (type I. C. couples)	To indicate and limit fuel element temperatures	Brown Instrument Co., strip chart, pen type, indicating recorder	Scram at 285°C
	~40	Thermocouples attached to fuel cladding (type I. C. couples)	To indicate and limit fuel element temperatures	Brown Instrument Co., strip chart, multi-point, indicating recorder	None
Graphite temperatures	3	Thermocouples embedded in graphite blocks located in an experiment hole	To indicate and limit the graphite temperature	Brown Instrument Co., strip chart, multi-point, indicating recorder	Scram at 50°C in peripheral region of moderator
Fan and fan-motor bearing temperatures	8	Thermocouples (type I. C.)	To indicate and record the cooling-air fan and fan-motor bearing temperatures	Brown Instrument Co., strip chart, multi-point, indicating recorder	Alarm at 180°F
West air manifold pressure	1	Pressure tap into exit-air manifold	To indicate, record, and limit the negative pressure in the exit-air manifold	Ring Balance Instrument Co., circular chart, indicating recorder	Scram at 34 in H_2O below atmospheric pressure
Differential air pressure	1	Pressure taps into the inlet- and exit-air manifolds	To indicate and record the differential pressure across the reactor, also limit the minimum	Ring Balance Instrument Co., circular chart, indicating recorder	Scram if the ΔP drops below 8 in. H_2O

Present Instrument Complement of the OGR (continued)

Designation	No. of Channels	Sensing Element	Purpose	Type of Monitor	Action
Inlet-air flow	1	Pitot-static tube in inlet-air duct	To indicate, record, and limit the minimum value of the inlet cooling-air flow	Ring Balance Instrument Co., circular chart, indicating recorder	Scram if the air flow drops below 2000 lb/min
Pitot static pressure	1	Same as for inlet-air flow	Indicate the static pressure in the inlet-air manifold	Meriam Co., manometer, uses oil spg 2.95	None
Rear-wall orifice differential pressure	1	Orifice in rear-wall coolint inlet line	To indicate the differential pressure across the rear wall orifice; permits calculation of flow rate	Meriam Co., manometer	None
Effluent cooling-air radioactivity	1	Ionization chamber at stack base	To indicate and record the radio-activity of the stack effluent	Brown Instrument Co., strip chart, pen type, indicating recorder	Alarm at 2 x the normal reading or 1000 curies/24 hr
Probe	2	Oil-soaked cloth probes used alternately	To indicate the quantity of radio-activity particles contained in the exit cooling air	Lauritzen electroscope, manual readings taken once every two hr	None
Exit filter radioactivity	1	Ionization chamber, air filled	To indicate the radioactivity burden contained by the exit-air filters	Electronic type electrometer, manual readings once per week	Alarm at an arbitrary value

OGR Instruments Located at the Fan House

Designation	No. of Channels	Sensing Element	Purpose	Type of Monitor	Action
ΔP across exit filters Cell No. Filter No. 1 1, 2 2 1, 2 3 1, 2 4 1, 2	8	Pressure taps into the exit-filter cells	To indicate the pressure differential across individual filter cells in the exit-filter house	Filters No. 1 Meriam Instrument Co. manometers (oil, spg 0.827) Filters No. 2 Hays pointer gage	None, may indicate the need for filter replacement
Static pressure (filter house exit duct)	1	Pressure tap into the filter house exit-air duct	To indicate the filter house exit-air pressure	Meriam Instrument Co. manometer	None
Differential pressure across filter house	1	Pressure taps into filter house inlet- and exit-air ducts	To indicate the total pressure drop across the filter house	Meriam Instrument Co. manometer	None
Water level (exit filter house canal)	1	Pressure equalizing line to tap in canal	To indicate the water level in the exit-filter-house canal	Meriam Instrument Co. manometer, bubble-type pressure equalizer	Alarms at low water level
Stack effluent radioactivity	1	Ionization chamber in effluent-air bypass	To indicate the radioactivity of the stack effluent air and transmit a signal to the control-room recorder	Detectolab [illegible] electrometer, Model DA6	Alarms at high level

OGR Instruments Located at the Fan House (continued)

Designation	No. of Channels	Sensing Element	Purpose	Type of Monitor	Action
Filter-house radioactivity	1	Ionization chamber located in one of filter cells	To indicate the radioactivity burden contained by the exit filters	ORNl Model C, a-c electrometer, sensitivity = 0.3v, range = 10^{7} through 10^{11} arbitrary units	Alarms at high level
Temperatures:		Thermocouples (type I. C.)	To indicate, record, and limit the fan and fan-motor bearing and oil temperatures	Leeds & Northrup Co., strip chart, multi-point, indicating recorder	Alarms and shuts fans down at high temperatures; alarms at 82°C
Motor bearings	4				
Fan bearings	4				
Motor oil sumps	2				
Fan oil sumps	2				
Temperatures:		Thermocouples (type I. C.)	To indicate the fan inlet-air and exit-air temperatures	Sim-Ply-Trol indicator	None
Fan suction air	1				
Fan discharge air	1				
Fans:		Pressure taps into: Suction seal pits, discharge seal pits, fan inlet duct, fan discharge duct	To indicate the inlet-and exit-seal pit water levels, and the fan inlet- and exit-air pressures	Meriam Company, mercury-filled manometers	None
Suction seal depth	2				
Discharge seal depth	2				
Suction pressure	2				
Discharge pressure	2				
Fan damper controls	2	Air-operated pistons connected to fan dampers on No. 1 and No. 2 fans	To adjust the fan damper openings for proper fan load	Hagan Company indicators and air-pressure regulators	Maintains the proper fan load

OGR Instruments Located at the Fan House (continued)

Designation	No. of Channels	Sensing Element	Purpose	Type of Monitor	Action
Oil pressure fans and motors	4	Pressure taps into fan- and motor-bearing oil systems	To indicate the pressure of the oil supply to the fan- and fan-motor bearings	Bronze Tube Duraguages, dial-type indicators	Alarm at low pressure
Fan power meters	2	Voltage and current coils connected to the fan motor electrical supply	To indicate and record the fan load	Esterline Angus, strip chart, pen type, indicator recorder	None
Fan electrical supply; voltage, current	2 2	Voltage and current coils connected to the fan motor electrical supply	To indicate the supply voltage and load current to the fan motors	G. E., dial type, voltage and current indicators	None
Steam fan: Seal pit depths Suction pressure Discharge pressure	 2 1 1	Pressure taps into Suction and Discharge seal pits and the fan inlet-and exit-air ducts	To indicate the inlet and exit seal pit water levels and the fan inlet- and exit-air pressures	Meriam Instrument Co., mercury filled manometers	None
Steam Fan Air (0 - 30) psig Air (0 - 90) psig Steam (0 - 30) psig Steam (0 - 500) psig	 1 1 1 1	Pressure taps into the air- and steam-supply lines	To indicate the steam-fan air and steam supply pressures	Bronze Tube Duraguages, dial-type indicators	None

Supplementary Instruments for use During Annealing of the OGR

Designation	No. of Channels	Sensing Element	Purpose	Type of Monitor	Action
Fuel cladding temperatures	~ 40	Thermocouples attached to fuel cladding (type I. C.)	To indicate and limit fuel element temperatures	Brown Instrument Co., strip chart, 16 point, multipoint, indicating recorder (three each) and circular dial indicators	Alarm at 275°C; scram at 285°C
Graphite temperatures	~ 160	Thermocouples embedded in graphite placed in fuel channels (type I. C.)	To indicate and limit graphite temperatures	Brown Instrument Co., strip chart, multi-point, indicating recorder and circular dial indicators	Alarm at 250°C; scram at 285°C
Concrete temperatures	~ 15	Thermocouples embedded in east shield (type I. C.)	To indicate and limit concrete shield temperatures	Brown Instrument Co., strip chart, multi-point indicating recorder	Alarm at 90°C; scram at 100°C
East-air manifold vacuum	1	Pressure tap into east manifold	To indicate and limit the east-manifold vacuum	Circular chart, indicating recorder	Scram at − 30 in. H_2O and shut fans down
Inlet-air flow (reversed)	1	Pitot-static tube in new cross duct	To indicate, record, and limit the minimum value of the inlet-cooling air flow	Circular chart indicating recorder	Scram at variable value
Exit-air flow (reversed)	1	Pitot-static tube in present exit duct	To indicate and record the exit-coolant flow rate	Circular chart indicating recorder	None

Supplementary Instruments for use During Annealing of the OGR (continued)

Designation	No. of Channels	Sensing Element	Purpose	Type of Monitor	Action
Inlet-air flow, low range	1	Hot-wire anemometer	To indicate the coolant flow rate in the lower flow range	Indicating, bridge-type dial gage	None
Filter tape radioactivity monitors	1	G-M tube or scintillation crystal	To indicate and record the particulate radioactivity in the exit-air duct and the stack effluent	Brown Instrument Co., strip chart, pen type, indicating recorders	Alarm at arbitrary level
Charcoal-trap monitor	1	Ionization chamber or G-M tube	To indicate and record the quantity of gaseous radioactivity contained in the exit-air stream	Brown Instrument Co., strip chart, pen type, indicating recorder	Alarm at an arbitrary value

2725
ORNL-~~2497~~
Special

INTERNAL DISTRIBUTION

1-2. Central Research Library
3-22. Laboratory Records Department
23. Laboratory Records, ORNL R.C.
24. A. E. G. Bates
25. F. T. Binford
26. C. D. Cagle
27. W. R. Casto
28. R. A. Charpie
29-38. J. A. Cox
39. E. P. Epler
40. W. H. Jordan
41. R. V. McCord
42. K. Z. Morgan
43. M. E. Ramsey
44. A. F. Rupp
45. L. E. Stanford
46. J. A. Swartout
47. M. C. Wittels
48. A. M. Weinberg
49. C. E. Winters
50. E. J. Witkowski

EXTERNAL DISTRIBUTION

51-75. H. M. Roth, AEC, Oak Ridge
76. R. W. Powell, BNL