TID-7697

Symposium on Problems

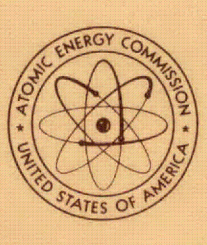
in

Irradiation Capsule

Experiments

Held at Germantown, Maryland

October 8, 9, and 10, 1963



United States Atomic Energy Commission
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TID-7697
METALS, CERAMICS, AND MATERIALS
TID-4500 (32nd Ed.)

SYMPOSIUM ON

PROBLEMS IN IRRADIATION CAPSULE EXPERIMENTS

October 8, 9, and 10, 1963

Edited by
William L. R. Rice

March 1964

Division of Reactor Development
United States Atomic Energy Commission
Washington, D. C.

PREFACE

Irradiation capsules are vital to the evaluation of reactor fuels and materials. They provide the first information on stability of structural materials to expected operating conditions and indicate the feasibility of proceeding to full scale fuel element fabrication and irradiation. A typical comment often made in reportsof capsule testing is, "The test was terminated due to thermocouple failure.". The cost of performing irradiation experiments can range from $50,000 to as high as $200,000 per capsule, depending on the complexity of the test and degree of control desired. It is obvious that capsule failure, whatever the reason, not only results in loss of money but also a considerable loss of time. This can have serious consequences if a reactor project is depending on the results for guidance in the development.

It was felt that a meeting on problems in irradiation capsule experiments would be helpful, and would possibly result in an over-all improvement in techniques employed. Such a meeting was held on October 8, 9, and 10, 1963 at the U. S. Atomic Energy Commission Headquarters, Germantown, Maryland. The initial letter of invitation included a proposed list of topics (see Appendix "A"), which served as the basis for the discussions. The response was overwhelming, as evidenced by the size of this report. Since this was the first open meeting on the subject, it was perhaps inevitable that we attempted to cover too much in the three days. It is believed that publication of this report of the meeting will provide a background for specialists in the area and that future meetings on the subject will not have to cover such a wealth of material.

A word about the format of the report. The individual papers presented followed the proposed discussion topic list in some instances, but in many others this could not be done. Thus, the organization of the report is partially arbitrary. Where it was possible, papers on related subjects were placed in sequence. However, the reader should note, for example, that heat transfer and dosimetry are not only treated in the sections so labeled, but in others as well. Therefore, it is recommended that the entire report be scanned for the sake of perspective <u>prior</u> to reading of material on a more selective basis.

We wish to express our particular appreciation to the attendees from other nations. Their participation in the meeting resulted in unusually fine coverage of all aspects of capsule design and operation and was most welcome.

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I. CAPSULE DESIGN AND OPERATION - NON FISSILE MATERIALS

CAPSULES FOR SURVEILLANCE OF SOME PLUM BROOK REACTOR STRUCTURAL MATERIALS

By Patrick L. Donoughe and Charles L. Younger

INTRODUCTION

With the advent of the power operation of the NASA Plum Brook Reactor (PBR), surveillance irradiation programs on some structural metals were initiated. It is anticipated that these programs will continue for a period of years to allow exposures up to 10^{22} nvt for some of the metals. The objective of the program is to conduct surveillance irradiations on 3 metals used in the PBR core: Mallory 1000 (90% W, 6% Ni, 4% Cu), which is used as a gamma shield for a cryogenic experiment, 17-7 PH stainless steel, and beryllium.

Capsule experiments have been performed in other reactors to achieve certain objectives. These tests have included investigations on a variety of materials and testing conditions (e.g., ref. 1-5). Differences in test objectives and modus operandi necessitated certain unique features in the capsule designs for the surveillance tests being conducted for the PBR.

The Plum Brook Reactor Facility (PBRF) includes a 60-Mw testing reactor of the MTR type and a hot laboratory adjacent to the reactor building. The surveillance irradiation programs require removal of capsules from in-pile locations, transfer to the hot laboratory, disassembly, reassembly, and transfer back to the reactor. Certain capsule design criteria are required due to the nature of the PBRF. Three separate capsules with similar characteristics have been designed.

The present paper gives a brief description of the NASA Plum Brook Reactor Facility, discusses the criteria used for design of the capsules, and describes the capsules.

REACTOR FACILITY

The PBR, a 60-Mw test reactor, is light water-cooled and is of the MTR type with primary beryllium and secondary water reflectors. Suf-

ficient concrete shielding is provided for the reactor shutdown period (fig. 1). A 70-ft circular pool of water, divided into quadrants, surrounding the concrete provides shielding for the power operation period. The water also provides shielding for transfer of experiments to the hot laboratory. The hot laboratory building, adjoining the reactor building, houses seven multikilocurie hot cells, cold and semicontaminated work areas, and storage areas. (More details of the PBRF and the hot laboratory are given in unpublished NASA data and ref. 6.)

A cutaway perspective of the reactor showing features such as test holes, pressure tank, transfer chute, and control mechanism is given in figure 2. A hatch at the top of the tank is provided for changing fuel elements and handling capsules in the pressure tank. Capsules are removed from the reflector positions, transferred via the fuel transfer chute to quadrant C, and transferred underwater to the hot laboratory, (figs. 1 and 2).

Views of the PBR and the test holes are shown in figure 3. Mallory 1000 metal, one of the surviellance materials, is used as a gamma shield in HB-2 for a cyrogenic irradiation experiment (ref. 7). The Mallory shield, located at the reactor end of HB-2, is exposed to a fast flux ($E > 1$ Mev) of about 10^{13} n/cm^2-sec when the reactor is at full power of 60 Mw. Beryllium, another surveillance material, is used as the reflector and also for core box structures. Spring material in the fuel elements and reflector pieces is 17-7 PH stainless steel, which is a third surveillance material.

Positions LA-5,7,9 and positions RA-2,7 will be used for the surveillance programs. Table 1 summarizes the programs.

TABLE 1. - SURVEILLANCE PROGRAMS FOR SOME PBR STRUCTURAL METALS*

Material	Use in PBR	Exposure flux n/cm^2-sec ($E > 1$ Mev)	Surveillance test hole	Surveillance flux n/cm^2-sec ($E > 1$ Mev)	Exposure number of reactor cycles**
Mallory 1000	Gamma shield HB-2	10^{13}	LA-5,7	1.5×10^{14} Max.	1,3,6,9,12, 15,18,21,24,27
Beryllium	Reflector	Varies	LA-9	1.3×10^{14} Max.	6,12,18,24, and yearly thereafter for 15 years
17-7 PH Stainless steel	Reflector and fuel elements	Varies	RA-2,7	3.7×10^{12} Max.	Yearly for 15 years

*Temperature of irradiation (~200° F) is considered environmental.

**A reactor cycle is about 10 days or 8.6×10^5 sec.

CAPSULE DESIGN

Surveillance programs on reactor structural metals require frequent removal and replacement of the test materials in the irradiation facility. Furthermore, it is desirable to monitor the actual exposure seen by the structural materials and to provide some degree of acceleration so that radiation effects may be predicted in advance of the actual exposure. Such a program involves a large number of test specimens with the capsule being loaded to maximum capacity.

For the surveillance program the following design criteria were used for the capsule:

(1) Insertion and removal of the specimen from the irradiation facility with minimum loss of time;

(2) Capability of containing a large number of specimens and of being readily disassembled for specimen removal without destroying the capsule;

(3) Removal and replacement of selected test specimens without complete disassembly of the capsule;

(4) Provisions for monitoring flux and temperature;

(5) Minimum alteration of reactor-primary-cooling-water hydraulics while not compromising reactor safety.

The following sections describe the features that meet these criteria.

Basic Assembly

The first capsule, designed for irradiation of Mallory 1000 metal (a tungsten - base alloy) in the LA pieces, set the pattern for subsequent capsules. When assembled this capsule forms a cylinder with a 2-in. diameter and a $38\frac{27}{32}$-in. length. The LA pieces (see fig. 3) have a beryllium plug that can be removed leaving a cavity with a $42\frac{3}{4}$-in. overall length. The upper $37\frac{1}{2}$ in. of this cavity has a $2\frac{3}{20}$-in. diameter below which the diameter is reduced to 2 in. for a $1\frac{1}{2}$-in. length, then to $1\frac{7}{8}$ in. for a $3\frac{3}{4}$-in. length. At the top of the cavity a circumferential groove has been machined on the inside of the beryllium piece at a point $2\frac{1}{4}$ in. below the top.

The capsule, shown in figure 4, consists of three major sumassemblies:

a top retention piece, a bottom retention piece and tie rods, and a test material subassembly. The test material may be encapsulated using either a single capsule segment (fig. 5) or multiple capsule segments (fig. 6).

Facility installation. - Three feactures of this capsule are designed specifically for installation in the lattice pieces. (The longitudinal cross section of the capsule in fig. 7 shows these features.)

(1) When installed in the reactor, the bottom retention piece (2-in. maximum diam.) is positioned in the 2-in. diameter by a 1.5-in.-long portion of the beryllium-lattice-piece cavity. The bottom of the capsule assembly (i.e., the bottom retention piece) is machined to a configuration that permits reactor primary cooling water to pass from the beryllium lattice piece into the reactor-primary-cooling-water exit system.

(2) Four balls installed and staked on the outside of the top-retention-piece body serve to lock the capsule in position. These balls engage the circumferential groove machined on the inside of the beryllium-lattice-piece cavity. In this position, the capsule assembly is locked so that it will not move up or down in the beryllium lattice piece. The capsule is centered in the lattice-piece cavity so that a 0.076-in. annulus for reactor primary cooling water is maintained between the lattice-piece cavity wall and the capsule-assembly wall (fig. 7, section D-D).

(3) A plunger installed in the top retention piece of the capsule is designed for handling the capsule remotely. Two circumferential grooves, machined on the outer surface near the bottom of the plunger, permit the balls to be retracted so that the capsule will move in or out of the beryllium lattice piece with ease. A coil spring, installed over the plunger and locked in position by Spir-O-Lox washers, positions the plunger so that the balls are in the locking position. A circumferential groove, machined at the upper end of the plunger, accommodates the grips of a remote handling tool. Pushing or pulling the plunger permits the balls to be retracted for insertion or removal of the capsule from the lattice piece.

These three features satisfy the first design criteria that the capsule fit the irradiation facility and be inserted and removed with ease.

Test specimen subassembly. - The capsule was designed to provide a 30-in. length for encapsulation of test specimens. This length corresponds to a region approximately 3 in. above and 3 in. below the fueled region of the PBR core.

Since it is desired to reuse the capsules, separate segments are provided in the 30-in. length (fig. 5). These segments, approximately 3 in. long are machined from 2-in. O.D. commercially pure aluminum (1100 grade) and are bored (1.5-in. O.D.) so that test specimens may be positioned

internally. The ends of each segment are machined for interlocking fit with capsule segments being stacked, one on top of another, to make up the 30-in. test specimen section. A pin and mating hole scheme for each interlocking segment is used to prevent axial rotation of capsule segments.

The bottom retention piece and tie rods are further designed to hold the capsule segments together. The retention piece is machined to conform to the interlocking scheme of the capsule segments. Three aluminum (1100 grade) tie rods are installed in the bottom retention piece (fig. 4) and are locked in position by roll pins inserted across the diameter of the bottom retention piece. The tie rods are positioned inside the bore of the capsule segments and extend the full length of the capsule assembly. With this arrangement tensile specimens can be stacked, one on top of another, in the three sections formed by the capsule segment wall and two tie rods and in the center of the capsule segment cavity.

The top retention piece locks the capsule assembly as a unit (fig. 7). It consists of two interlocking sections designated as the top-retention-piece body and the tie-rod alinement section. The body and the alinement section are bolted together at the center. Alinement between these two sections is achieved by a roll pin in one interlock surface and a mating hole in the other. Longitudinal through holes serve as water channels and tie-rod penetrations. Three long-nosed screws, inserted through the side of the top retention piece, engage a hole through the tie rod, thus locking the capsule as a unit (see also fig. 4).

A 17-7 PH stainless-steel spring is installed over the bolt between the top-retention-piece body and the tie-rod alinement section. Spring action tightens the capsule assembly longitudinally and applies a load on the long-nosed tie screws.

These features satisfy the second design criteria that the capsule contain a large number of specimens and be readily disassembled without capsule loss.

Test-specimen installation. - While the test specimen loading scheme outlined in the previous section meets the design criteria for maximum loading, it does not permit easy removal of selected test specimens. In order to satisfy this further design requirement, the assembly shown in figure 5 was adopted.

Test specimens and aluminum tubes, which serve as sleeves for the capsule tie rods, are positioned in the capsule segment cavity. Specimen retention washers are placed on each end of the assembly. Snap-ring lock washers, placed in grooves on the ends of each sleeve, overlap both the specimen retention washer and the interlock surface of the capsule segment and lock the assembly.

Encapsulation of test specimens of different geometries is accomplished by modification of the specimen retention clip. For example, the specimen retention washer for flat bar specimens shown in figure 5 has clips brazed to the prongs.

For those conditions where a large number of test specimens are to be removed at the same time, the encapsulation is achieved by the assembly shown in figure 6. Here capsule segments and test specimens are stacked to form the desired length, and the subassembly tie is the same as that described above.

Spacers, machined in the same interlocking manner as the capsule segments, are employed between subassemblies, since clearance must be provided between the ends of sleeves of adjacent capsule segment subassemblies.

Using this loading scheme, encapsulation of thirty 1/4-inch diameter tensile specimens and thirty 1/8-inch diameter rod specimens, is achieved for the Mallory surveillance program.

Monitoring. - One of the basic criteria of any irradiation program is to ascertain the temperature and flux to which the test material is exposed. For metallic materials, the integrated neutron flux is the quantity desired.

Two techniques for flux monitoring were incorporated in the capsule assembly. One of these, built into the capsule itself, consists of holes drilled in the capsule segment wall. The other technique requires replacing the test specimen in the center of the specimen cavity with a flux monitor capsule. This flux monitor capsule consists of two slot-head cap screws, each designed to hold a single monitor wire, and an aluminum body (fig. 5). The cap screws are threaded and are positioned on opposite ends of the aluminum body.

Since the materials to be encapsulated in this program are high temperature materials, no temperature measurement was incorporated. The inlet and exit reactor primary cooling water ($\Delta T = 27^{\circ}$ F) is monitored during reactor operations and can be used for determining the thermal environment seen by the encapsulated test specimens.

Should temperature monitoring be desired, one approach would be to replace the central flux monitor capsule with a capsule designed to receive and contain buttons or slugs of materials whose melting points span the thermal range anticipated. A second approach to temperature monitoring would be to introduce thermocouples into the capsule assembly. The center of the capsule assembly is open to reactor primary coolant water flow and thermocouples could be installed without too much difficulty.

Heat-transfer analysis. - Heat-transfer calculations were performed for a capsule loaded with tungsten specimens. For these calculations using the TOSS program (ref. 8), the reactor full power (60 Mw) operating condition and the reactor shutdown condition were considered.

With the reactor at full power, the peak gamma heating value in the irradiation facility is 15 w/g. Using this value and the results of the

flow test described later in this paper, the TOSS output showed the maximum specimen surface temperature under steady state conditions to be 209° F. The overall temperature rise of the reactor primary cooling water passing around and through the capsule assembly was calculated to be 24° F. These values are within the limits established for PBR heat transfer, that is, ~300° F surface temperature and 27° F temperature rise in the cooling water.

At reactor shutdown the gamma heat drops almost immediately to about 0.4 of the full power value. Thereafter a gradual reduction in heating rate takes place. For the analysis, step reduction in cooling water flow was assumed. The TOSS output showed a transient surface temperature rise to about 300° F, followed by a reduction to below 212° F, occurring over the 90-sec. time interval following reactor shutdown. The reactor shutdown schedule requires maintaining reactor tank pressure for a minimum of 5 min following reactor shutdown. Under this schedule the transient temperature approaches the maximum surface temperature permitted, but it still is acceptable.

The heat-transfer analysis was performed using the most severe (and in many cases unrealistic) conditions to be encountered. Tungsten introduces an additional severity in that high gamma heating values are encountered. It is significant to note that, under the assumed conditions of heat generation, permissible surface temperatures are not exceeded and the cooling water temperature rise is not excessive. Most structural materials will not encounter these conditions, and, hence, no heat-transfer problems are associated with the capsule assembly.

Stress analysis. - Stresses on the capsule result from (1) tensile loads due to specimen and capsule weight and reactor primary cooling water flowing through the capsule and (2) thermal stress due to differential heating of capsule components.

The capsule assembly is effectively suspended from the long-nosed tie screws in the upper retention piece. The capsule weight is concentrated on the bottom retention piece and distributed over the three tie rods. Figure 8 shows schematically the loading of the capsule assembly. For tensile stress analysis, one tie rod of the cross section shown in figure 8 was employed. It was assumed that all the capsule and the test specimen weight (12.15 lb) was applied to this rod. Furthermore, the load for reactor primary cooling water acting on a 1.5-in. diameter disk (71 lb) was added to the capsule and test specimen loads. The minimum cross section of a single tie rod (0.0242 in^2) was sufficient to hold this load with only slight elongation (0.011 in.).

Thermal stress analysis for a cylindrical pipe under the reactor temperature gradient conditions resulted in a negligible effect.

Thermal expansion calculations for the most severe temperature gradients were performed for all mating sections of the capsule assembly, the test specimen loading, and the capsule-beryllium piece mating areas. In no case was thermal expansion sufficient to cause binding and potential stress concentrations. The results of stress analysis show that the capsule assembly is sufficiently strong. As in the case of heat generation, the worst possible case was postulated for the analysis. Actual conditions should be less severe.

Modification of Basic Assembly

A second capsule was designed to encapsulate beryllium tensile and rod test specimens. This capsule was designed to be inserted in the same type of lattice piece (L-piece in fig. 3) as previously described for the basic capsule assembly. The pertinent features of this second capsule assembly are shown in figure 9. The basic design features of the second capsule are essentially the same as those previously described, and, hence, only the differences will be considered.

The second capsule assembly is designed with a single tie rod locked in the bottom retention piece and extending the full length of the capsule assembly. For locking in the top retention piece, a single long-nosed tie screw that extends across the top-retention-piece body is used.

Test specimen loading is confined to a single capsule segment using six-pronged specimen retention washers which fit over a single aluminum sleeve. Snap-ring lock washers positioned in grooves on the ends of the sleeve lock the assembly together. The prongs of the specimen retention washer are of sufficient length to engage the interlock surfaces of the capsule segment. To keep test specimens positioned with respect to rim segment flux monitor positions, 1/8-in. diameter holes are drilled in the ends of the tensile test specimen. Pins, brazed to the prongs of the specimen retention washer, engage the holes in the ends of the specimen and thus lock them in position. The prongs of the specimen retention washer are of sufficient length as to engage the capsule segment roll pin installed in the interlock surface. This permits a slight degree of rotation of test specimens relative to capsule rim segment flux monitor positions, but such rotation is not sufficient to be a problem.

Rod specimens are installed in the cavities between the tensile specimens and the aluminum sleeve. A snug fit is provided so that the rod specimens are locked in position. For capsule assembly, capsule segment spacer disks are employed between each capsule segment subassembly.

A third capsule, basically similar to the one just described, has been designed for the reflector pieces of the PBR. These reflector pieces have a $2\frac{1}{2}$-in. nominal diameter rather than a 2 in. diameter with a length of

$15\frac{3}{4}$-in. (fig. 3). The basic difference in the reflector and lattice capsules is the specimen retention washer. For the reflector capsule, two sets of specimens may be tested. One set is positioned on the large diameter next to the capsule segment wall, and another set is positioned on a smaller diameter next to the capsule-segment-subassembly tie sleeve.

From these brief descriptions of modification to the original capsule design, it is seen that the design is versatile, a desirable feature when considering standard practice for hot laboratory and reactor operations.

CAPSULE HYDRAULICS AND HANDLING

When installed in the reactor, primary cooling water enters the capsule through an opening in the plunger of the top retention piece (point 1 in fig. 7). The flow is then downward through the tubular length (1-2 and cross section A-A) of the plunger and enters a cavity (2-3 and cross section B-B) in the top retention piece. From here the water is channeled through three orifices (3-4 and cross section C-C) into the test specimen section (4-5) of the capsule assembly.

Water flow over the test specimens encounters a complex series of cross sections. The upper section of the capsule and the capsule-segment spacer disks do not contain test specimens, but do contain tie rods. The cross section shown in cross section D-D is illustrative of the capsule hydraulic cross section in these regions. The downstream flow areas are shown in the cross sections given in figure 7. Water discharges from the capsule by way of a single tubular length (5-6 and cross section J-J) in the bottom retention piece.

Reactor primary cooling water also flows through a 0.076-in. annulus between the capsule wall and the beryllium-lattice-piece wall. The water passageway in the exit region at the bottom of the lattice piece is shown by cross section K-K.

Water flow tests were conducted on this capsule assembly using a hydraulic flow loop built for testing PBR fuel elements. A schematic flow diagram of the test setup is shown in figure 10. An aluminum piece machined to the same configuration and dimension as the beryllium lattice piece was installed in the test section of the flow loop.

A capsule was loaded to near maximum capacity with tensile specimens, 1/8-in.-diameter rod specimens, and corrosion bar specimens. Flux monitor capsules were positioned in the center of the test specimen cavity. The capsule was installed in the cavity of the simulated lattice piece. The cross section at the bottom of figure 10 shows the water flow areas through the test section. Three water channels pass through the flow-loop test section, namely:

a. A water channel around the outside of the aluminum test piece;

b. A water annulus (0.076 in.) between the outer wall of the capsule assembly and the inner wall of the aluminum-test-piece cavity;

c. A water channel through the capsule assembly.

Three series of tests, the results of which are plotted in figure 11, were run on the capsule assembly. The tests were conducted with:

a. No flow channels blocked, (runs 1 and 4 in fig. 11);

b. The flow channel through the capsule sealed, other channels open (run 2 in fig. 11);

c. The flow channel through the annulus sealed, other channels open (run 3 in fig. 11).

Flows were measured as the pressure drop across the test section was increased and as it was decreased. The zone of interest, corresponding to the pressure drop across the PBR core, is around a ΔP of 40 psi.

The results of the hydraulic tests show that the flow in the annulus is about 32 gpm (difference between curves 2 and 3 at 40 psi). This flow is in agreement with the 35 gpm ±10 percent value that was found in hydraulic tests on the PBR (ref. 9, p. 16). In addition, the flow through the test section is about 10 gpm (difference between curves 2 and 4). These results indicate that the hydraulics of the reactor primary cooling water (total flow of about 18,000 gpm) are not altered appreciably by replacing the beryllium plug with the capsule assembly. The value of 10 gpm was adopted for heat-transfer calculations pertaining to in-pile operation of the capsule. In addition to these flow tests, some handling experience with the capsules has been obtained.

The handling of the capsule is rather straightforward and simple as far as reactor insertion and removal is concerned. The basic capsule has been in-pile for one reactor cycle. Removal of the capsule assembly from the reactor tank is likewise simple in that use is made of a fuel-element discharge chute built into the side of the reactor tank and discharging into the bottom of the reactor quadrant, (see fig. 2). The capsule assembly can easily be handled underwater between reactor and the hot laboratory (fig. 1). Once installed in the hot laboratory, the capsule disassembly is accomplished by removal of the three long-nosed tie screws from the side of the top-retention-piece body (fig. 4). In preneutron handling, remote disassembly and reassembly of the capsule sections have been performed in the hot cell with ease by PBR hot laboratory personnel.

A limited amount of experience (familiarity operation) has been

accumulated by the PBRF staff in the disassembly and reassembly of individual capsule segment subassemblies such as shown in figure 5. On a first effort basis, hot laboratory personnel have completely disassembled the capsule segment subassembly, removed the test specimens (with tweezers), reinstalled the test specimens, and reassembled the capsule segment assembly in less than one hour.

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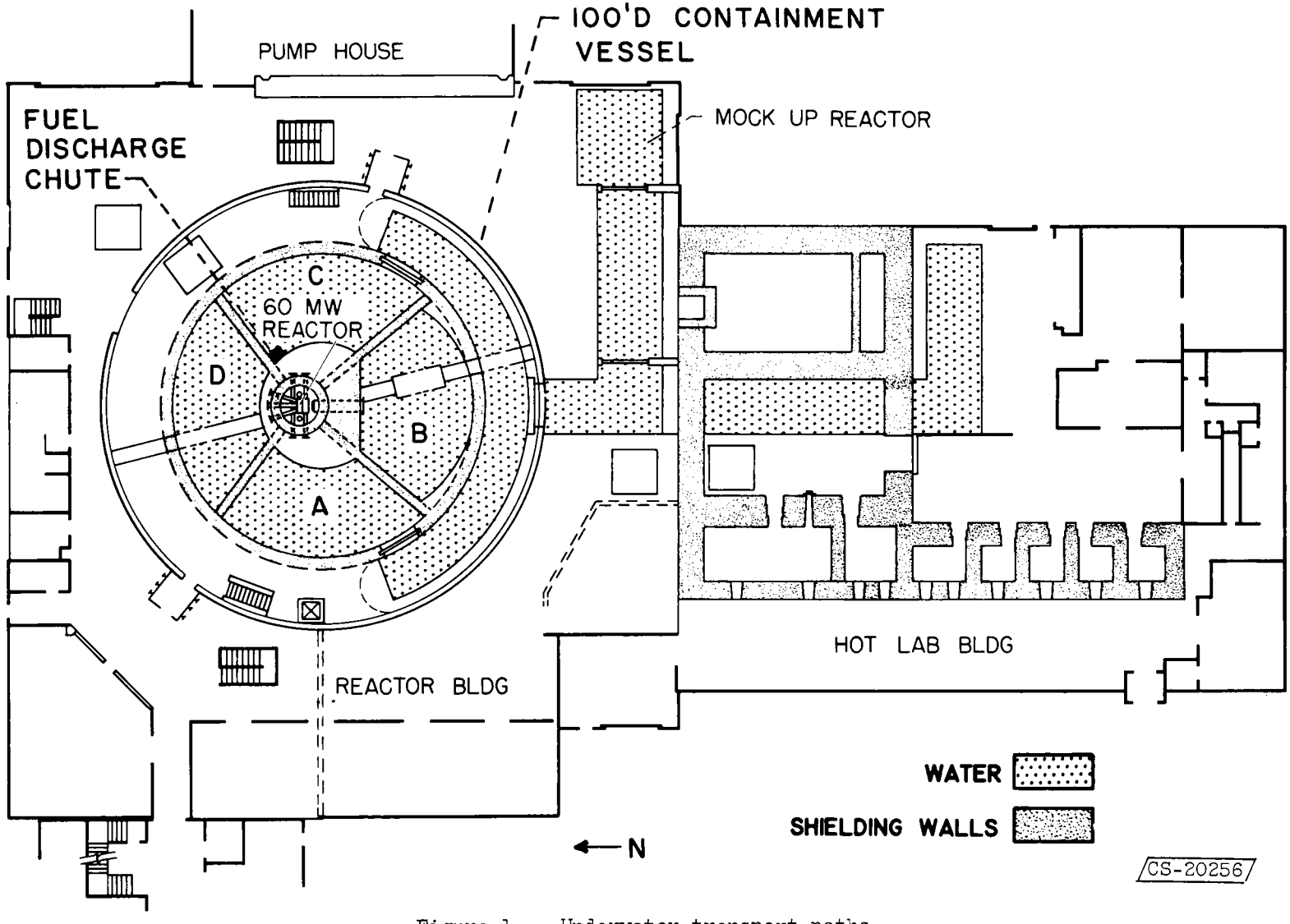


Figure 1. - Underwater transport paths.

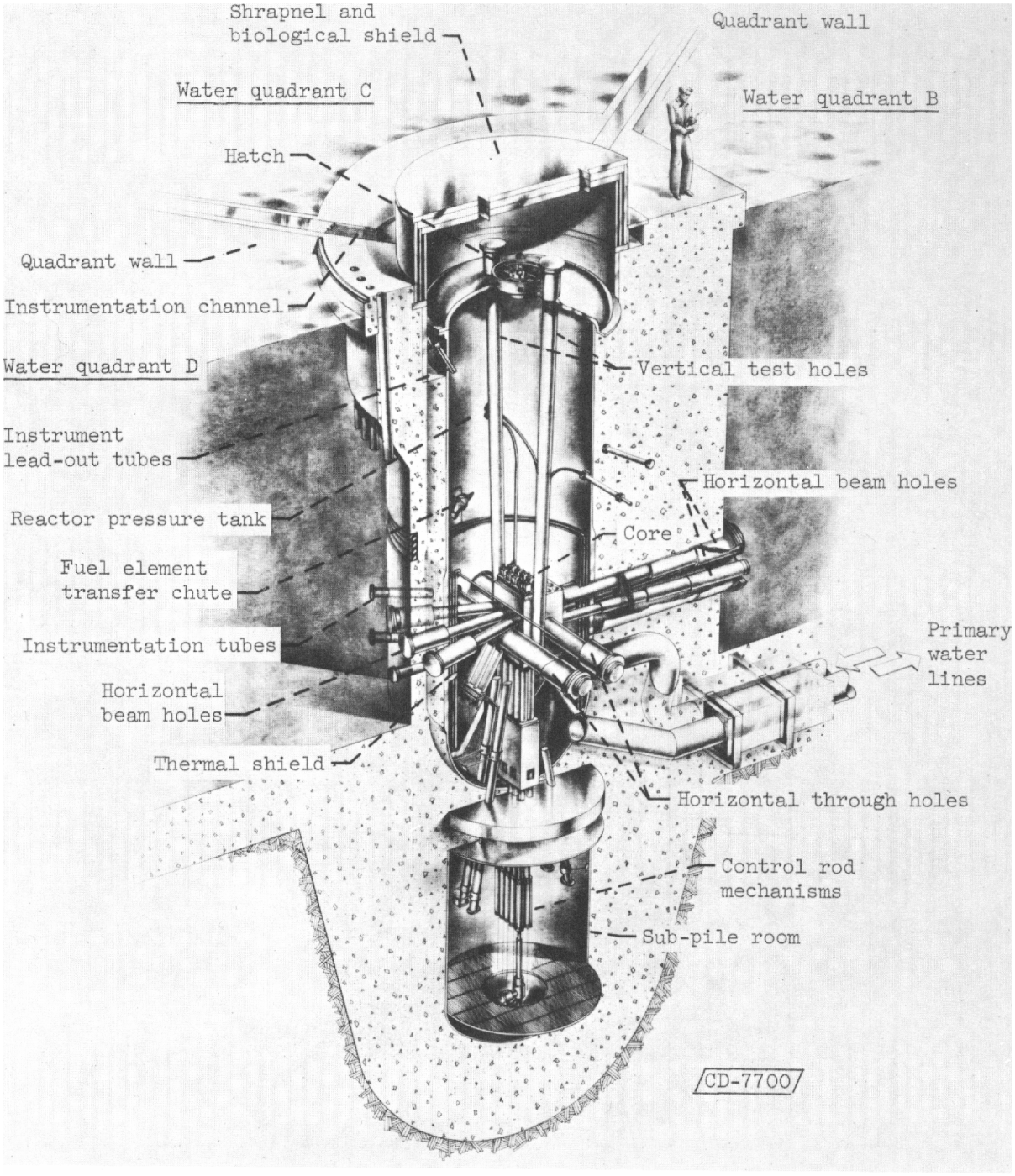


Figure 2. - Cutaway perspective drawing of reactor tank assembly.

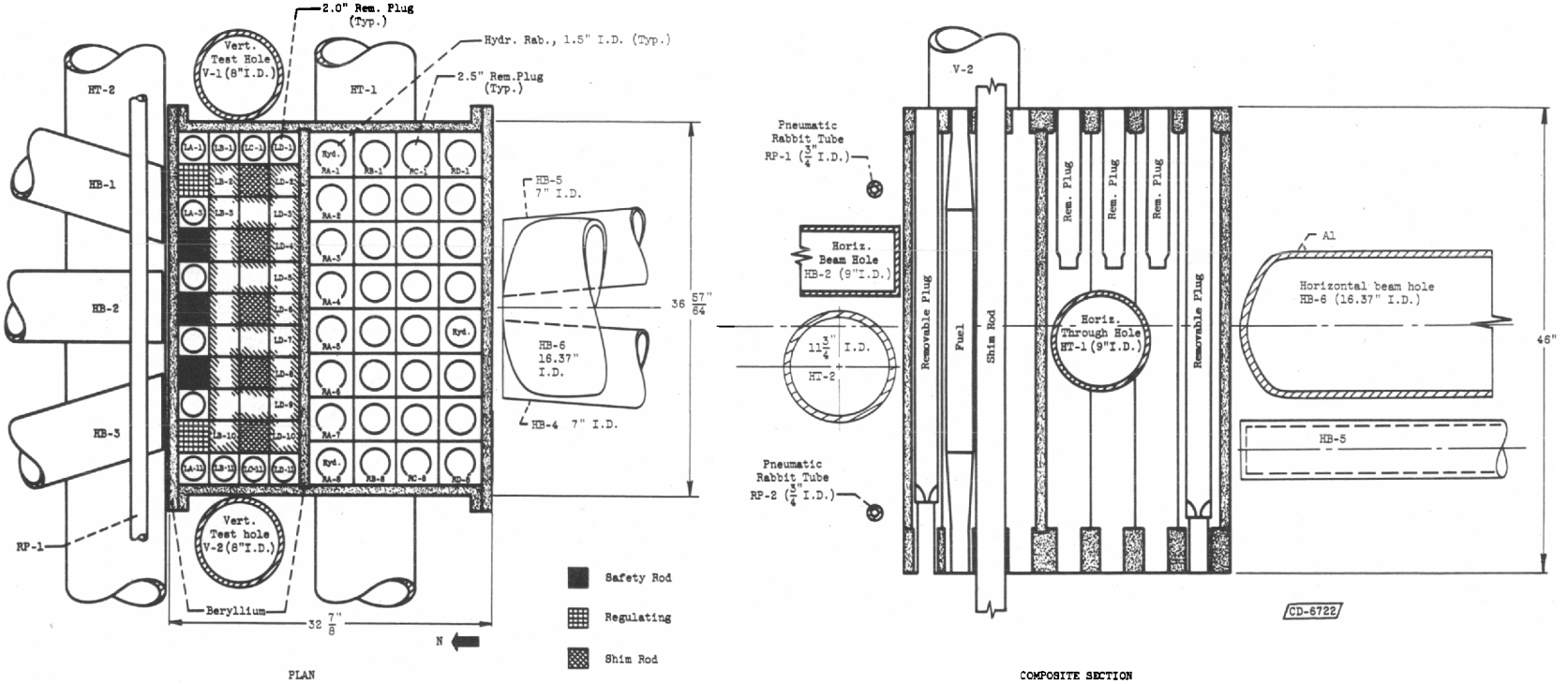


Figure 3. - Reactor core and experimental facilities.

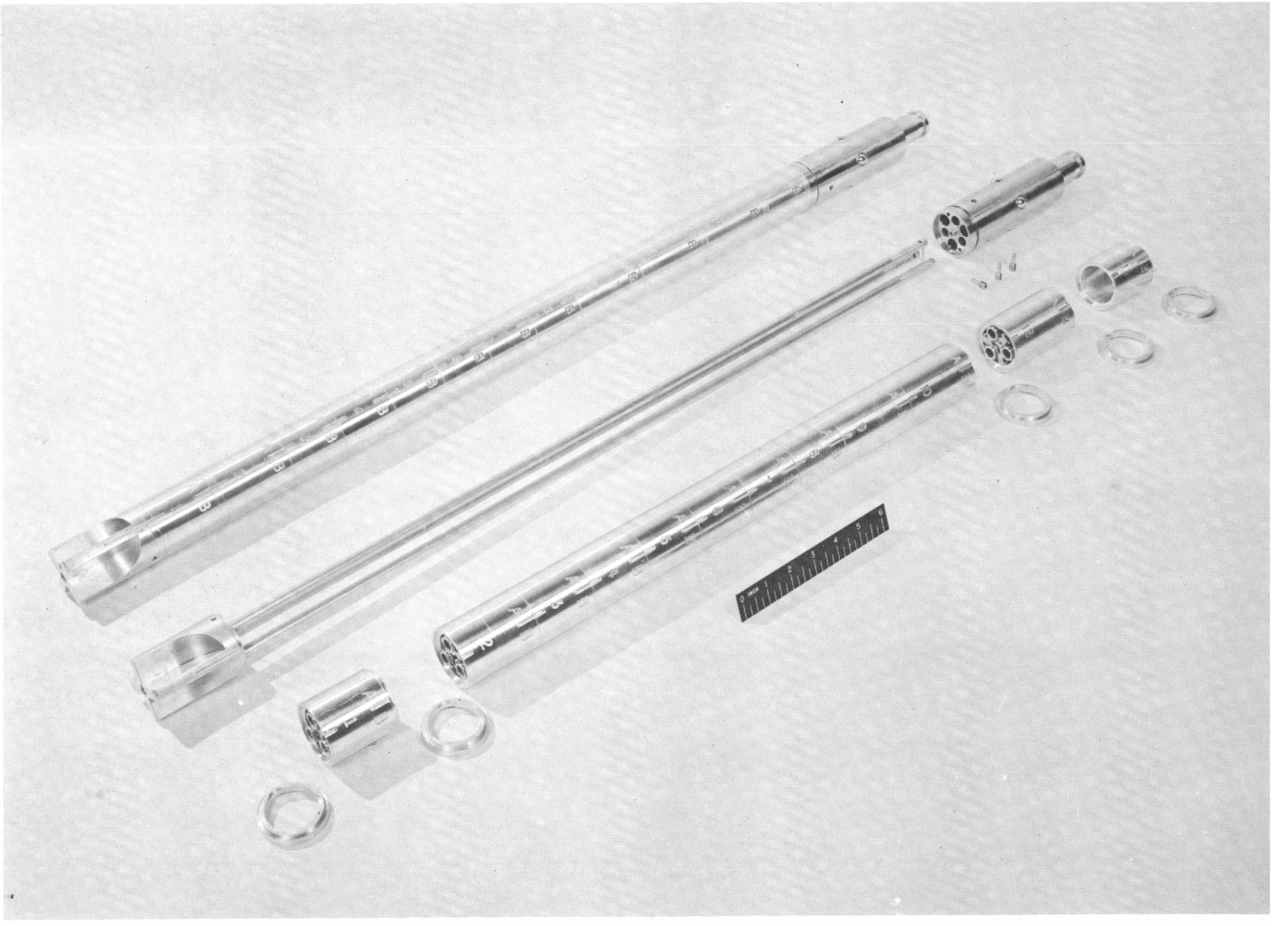


Figure 4. - Radiation capsule retention assembly

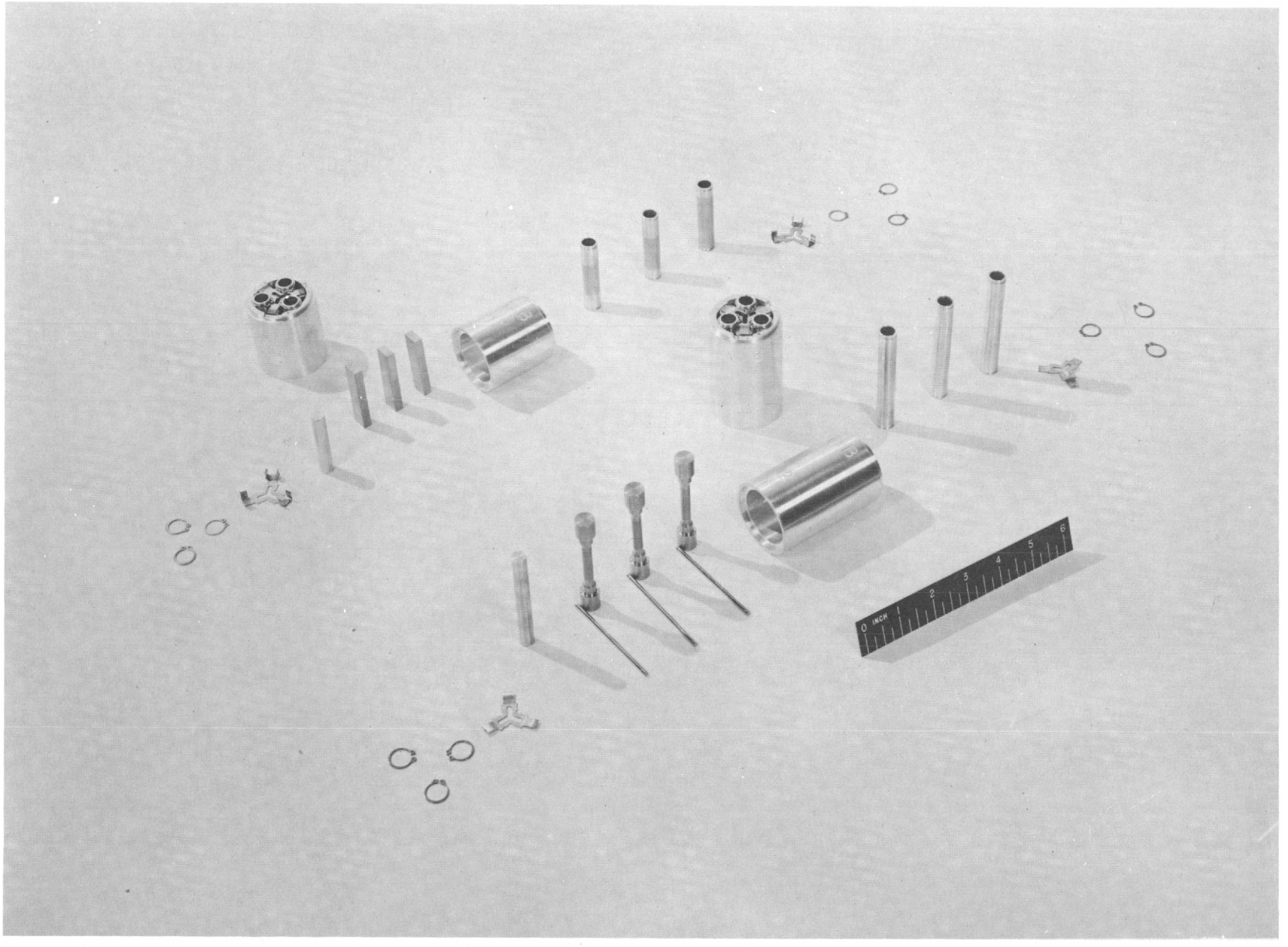


Figure 5. - Radiation capsule, corrosion specimen subassembly and permanent test material subassembly.

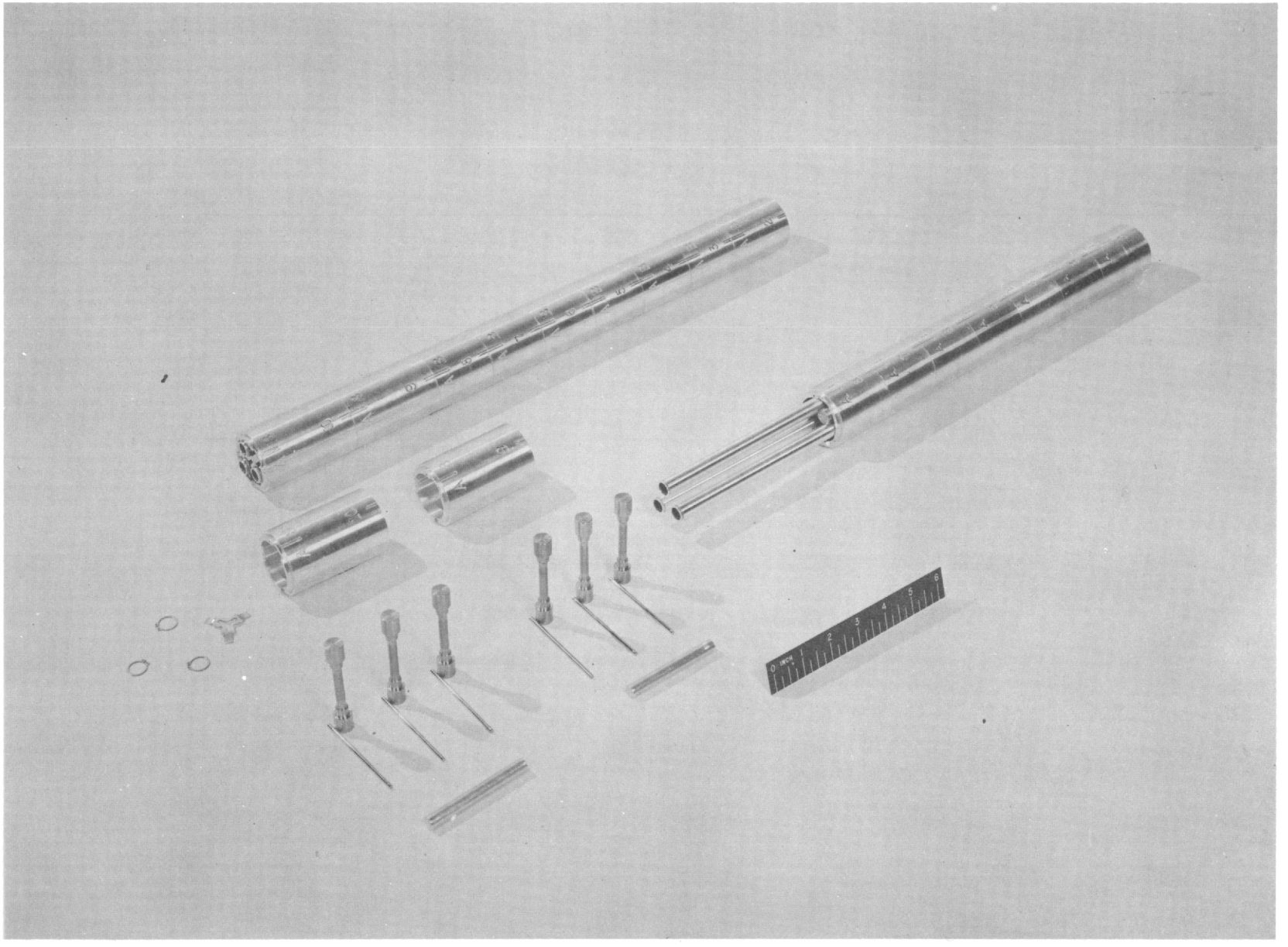


Figure 6. - Radiation capsule, replaceable test material subassembly.

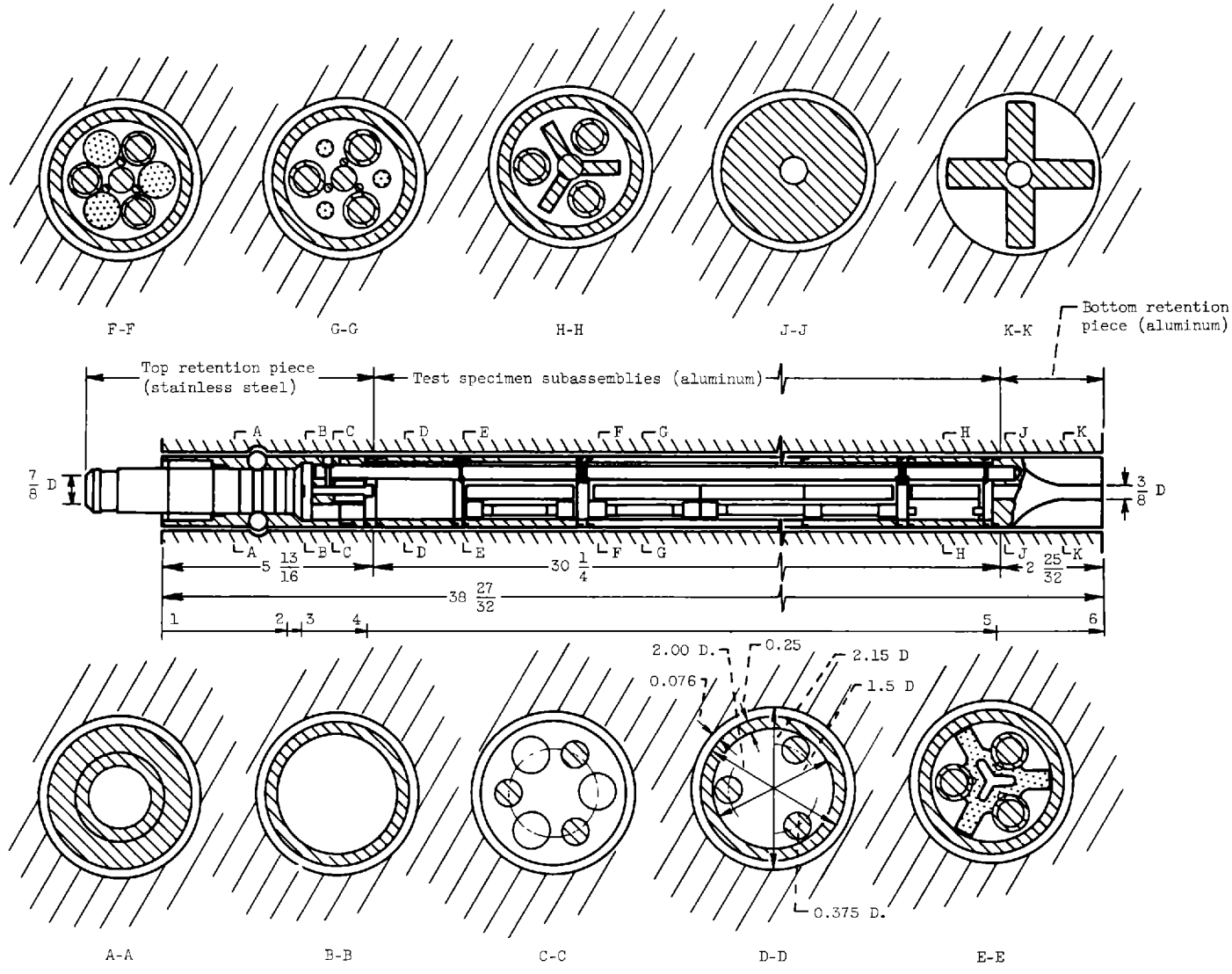
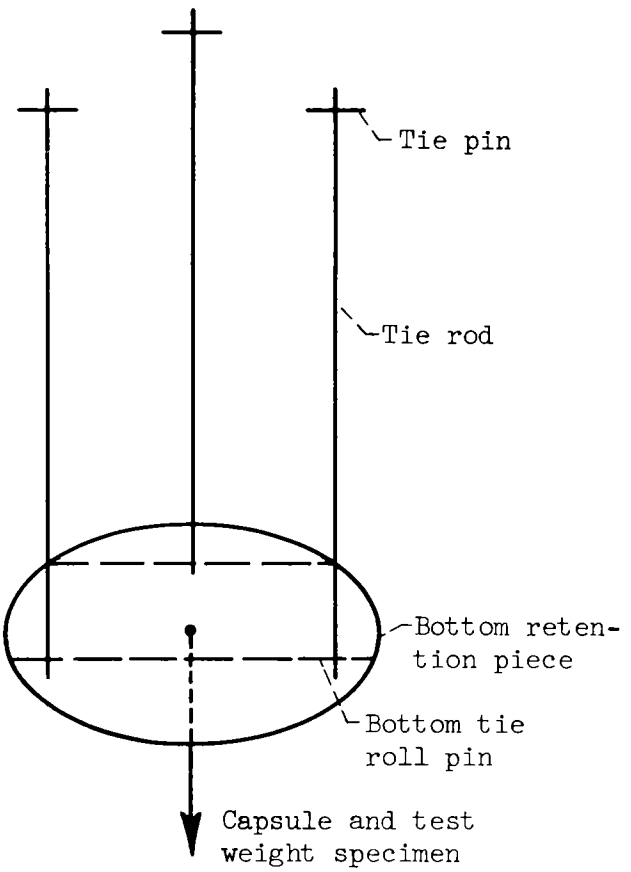
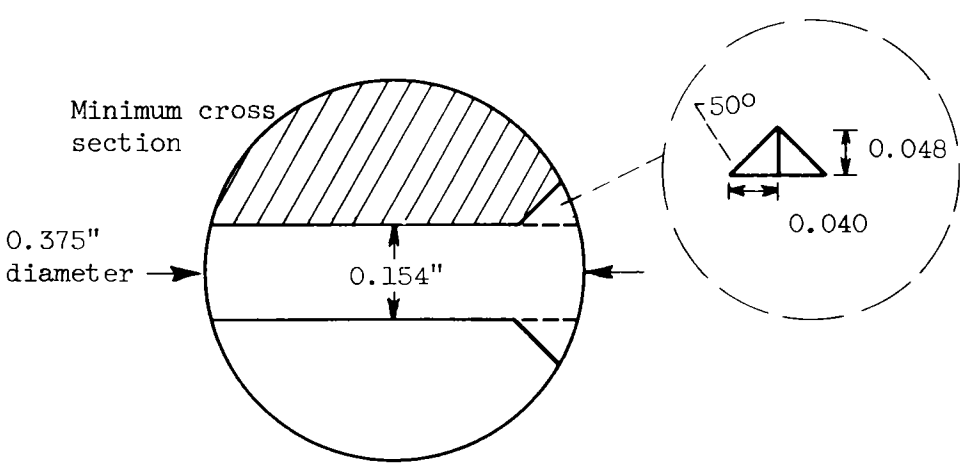


Figure 7. - Schematic of capsule assembly (waterflow area shown open).



(a) Schematic of capsule loading.



(b) Minimum cross section of capsule tie rod.

Figure 8. - Stress loading and minimum cross-sectional area of radiation capsule.

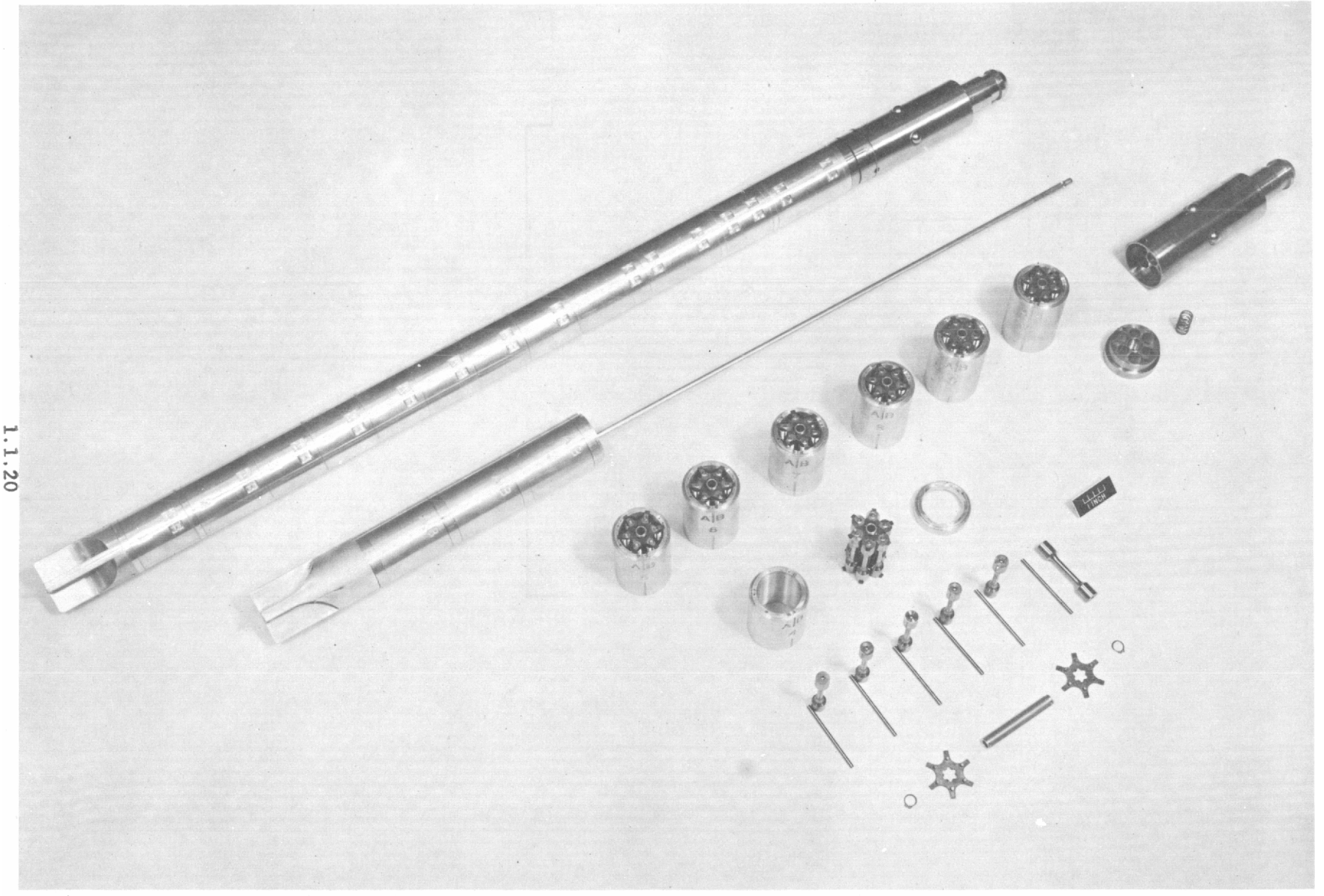


Figure 9. - Radiation capsule, modified.

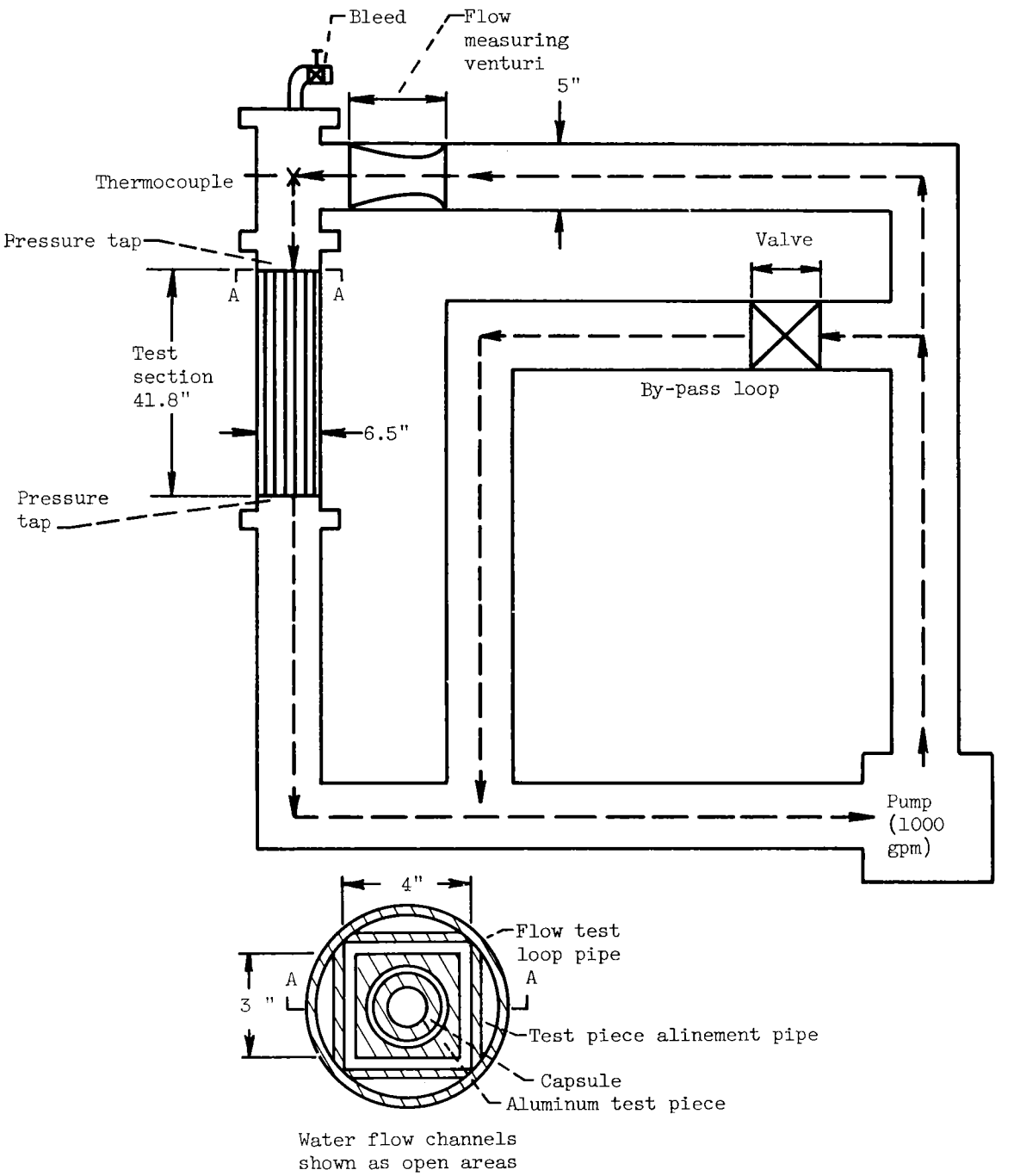


Figure 10. - Fuel element flow test loop at PBRF and capsule flow test scheme.

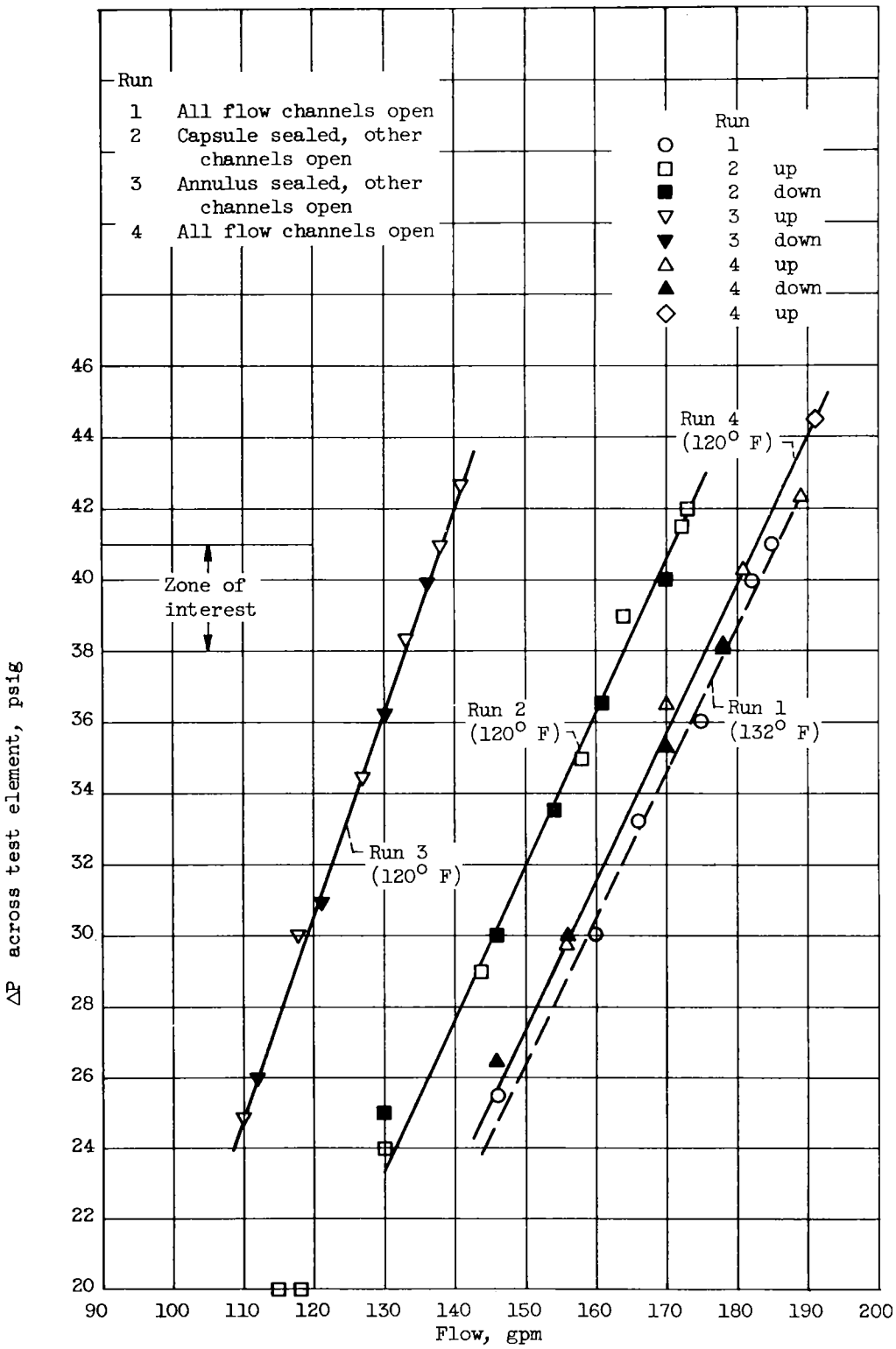


Figure 11. - Flow tests on radiation capsule.

ENCAPSULATION TECHNIQUES FOR NRL IRRADIATION EFFECTS STUDIES

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INTRODUCTION

The encapsulation and irradiation techniques devised in the conduct of metals irradiation effects studies at the U. S. Naval Research Laboratory are an outgrowth of two factors: the major research objectives of the program, and the limitations imposed by the irradiation facilities available. While research objectives have been rather specific to date, the available irradiation facilities have covered a range of physical and nuclear environmental conditions and, accordingly, have greatly influenced the development of encapsulation and irradiation procedures.

The NRL irradiation program has emphasized the determination of the effects of high energy neutrons upon the mechanical properties of nuclear reactor structural steels; particularly the very detrimental effects on the notch ductility characteristics of ferritic steels. In meeting this primary objective it was necessary to irradiate the relatively massive (800 gram)drop weight test specimens, for determining specific nil ductility transition (NDT) temperatures, in addition to Charpy V-notch impact specimens and tension specimens. In addition, the NRL program sought to characterize the possible effects of irradiation at temperatures ranging from 150 to 750°F; bracketing, with some margin on either side, the normal operating temperatures (400-650°F) of most nuclear power reactors in the United States. Furthermore, a secondary goal, that of studying possible effects of irradiation rate and neutron spectrum, was made possible through the availability of several quite different irradiation facilities involving five test reactors and two Army power reactors. While diversity of reactor facilities is desirable for the study of nuclear environmental effects, it served as a handicap in the development of capsules and irradiation techniques.

Nevertheless, certain procedures or techniques of capsule development have evolved which are related to two general classes of irradiation assemblies: (1) those which are true sealed capsules with no means of control after insertion in the reactor, and (2) those unsealed units which contain leads which permit control from external instrumentation during irradiation. Capsules for long-term surveillance of the Army power reactor components, although specialized, belong to the former class, while

most of the test reactor experiments belong to the class of controlled assemblies. Various capsule units which have been developed for the several reactors utilized, will be described along with design philosophy and techniques which have developed in the course of NRL materials irradiation effects study. Limited descriptions of NRL capsule and irradiation techniques have been reported earlier (1,2,3,4).

CAPSULE DESIGN PHILOSOPHY FOR STEEL IRRADIATION PROGRAM

The need for rather massive loadings of steel for irradiation at specific elevated temperatures severely limits the number of satisfactory irradiation facilities. That is, central regions of reactors of high flux density such as the MTR and ORR cannot be used because the severe gamma heating precludes a single consistent temperature in a capsule. Accordingly, reactor facilities having lower gamma heating characteristics, especially the LITR, have been utilized primarily for the NRL irradiation program. The selection of these irradiation facilities has permitted a general approach involving capsule loadings which, by most irradiation standards, would be prohibitively massive.

Another criterion for capsule design has involved provisions for minimizing corrosion of the ferritic steel specimens during irradiation. Thus, the encapsulation techniques have been devised to provide an in-reactor container as well as a corrosion protection barrier. This factor is especially important in the development of irradiation capsules for the water-cooled reactors, since a capsule leak may effectively destroy the experiment.

Other factors which contribute to general capsule design philosophy include: (1) a constant effort to simplify capsule design, (2) the use of ambient gamma heating characteristics for temperature control, (3) designs to minimize variations in neutron exposure within a capsule, and (4) a heat transfer path which is as direct as possible yet is commensurate with control of temperature uniformity within a capsule.

The organization and functions of the Radiation Operations Section which conducts the NRL Metallurgy Division irradiation effects study is conducive to the objective of simplicity of capsule design, since the same personnel are

responsible for the design and fabrication of the assembly, the irradiation, and the disassembly of irradiated capsules. Thus, when a capsule is constructed the problems of insertion and control in reactor as well as those problems which may be associated with hot cell operations are factored into the original design.

The experimental objectives of simplicity, of utilizing the specimens as self heaters, and of minimizing neutron exposure gradients are mutually complementary in capsule designs for low flux irradiation facilities. That is, these three objectives are met by grouping the specimens to be irradiated as tightly as possible together. However, the provision of temperature uniformity may suffer by the accomplishment of the first three objectives. Thus, in following the general design philosophy outlined, the key factor requiring variations in capsule or irradiation assembly (other than the physical size and shape of the irradiation facility) has been the effort to uniformly control temperatures during irradiation.

The factors outlined above and the limitations imposed by the particular irradiation facility have determined the several capsule designs utilized in the irradiation effects study in test reactor facilities. However, for long-term surveillance capsules for power reactors, where a specific temperature (the coolant water temperature) is desired, a different design philosophy has developed. Two other factors, space limitations and gross variations in neutron flux levels in relatively short distances have also influenced the surveillance capsule design. For the irradiation of Charpy-V notch specimens in power reactors, emphasis has been placed upon assuring exposure at the temperature of the reactor coolant. Thin capsules with the V-notch in a given plane (or planes) have provided the best compromise for this type of irradiation. Various encapsulation techniques are described in the following section with reference to the general experimental objective and the several reactors used.

ENCAPSULATION TECHNIQUES

The variety of capsules devised is suggested by the fact that irradiation experiments have been conducted in five test reactors, the Brookhaven Graphite Reactor (BGR), the Industrial Reactor Laboratories Pool Reactor (IRL), the Materials Test Reactor (MTR), and the Oak Ridge Low Intensity Test and Research Reactors (LITR and ORR), as well as two Army power reactors (SM-1 and SM-1A). In addition, three different types of facilities were used in the LITR. However, it should be noted that only sealed capsules were devised for irradiation in the IRL, MTR, ORR, and SM-1 and SM-1A reactors, thus simplifying the encapsulation tasks, since certain capsules were found to be interchangeable in application between reactors. The following description divides capsules by sealed or unsealed units which essentially separates them by experimental function.

SEALED CAPSULES (NO EXTERNAL CONTROL)

The same general design characteristics apply to all sealed capsules though these may differ somewhat in accordance with the experimental application or the irradiation facility. The following characteristics or encapsulation procedures apply to all sealed capsules:

1. Steel specimens are mounted within or upon an aluminum or stainless steel skeletal framework.

2. Appropriate low melting point eutectic alloys or pure metals are installed in the capsule to define the maximum temperature attained during exposure. Table 1 shows the series utilized.

3. Neutron dosimetry wires (cobalt, cadmium-covered cobalt, nickel, iron, and titanium) are placed in V-notches of the specimens and at other pertinent positions for determining the neutron exposure.

4. A stainless steel envelope is welded around the specimens to provide containment and a corrosion barrier.

5. The capsule is subjected to compression in a helium pressure chamber before final sealing. This does two things: (a) it forces the 0.010-in. thick stainless steel skin into intimate contact with capsule contents (deforms to match all irregularities of the capsule) and (b) serves as a rigorous leak test.

6. The capsule is heated to a temperature just below the melting point of the lowest melting eutectic with a continuous flow of helium for one hour and is finally sealed while still hot. This provides a partial pressure atmosphere of helium unless the capsule is heated above the intended level.

Some variations of this procedure, such as the application of vacuum during sealing procedures, may be required for special cases. The unique characteristics of capsules for specific reactors and specific experimental objectives are outlined below by reactor in the general order of historical application.

Materials Test Reactor

The A-4 position of the MTR which is outside the reactor core but inside the reactor tank has been used several years for low temperature (<450°F) irradiation of various steel specimens to a range of integrated neutron doses. The large drop weight test specimens as well as Charpy-V and tensile specimens have been irradiated in capsules similar to that shown in Fig. 1. The capsule length is determined by the longitudinal neutron flux gradient and by the required specimen loading. The cross sectional shape and size is dictated by the transverse neutron flux gradient which is U-shaped, the size of the dummy fuel core piece, and the need for relatively high coolant flow because of the gamma heating level. The massive drop-weight specimens are drilled out at the ends to reduce gamma heating as much as possible.

Various specimen loading configurations are possible but the most used loading combinations involve, (a) 6 drop-weight specimens plus 20 Charpy and 2 sub-sized (0.180 in.) tensile specimens, (b) 84 Charpy and 6 tensile specimens,

and (c) an all tensile capsule containing 42 of the standard 0.252-in. gauge diameter specimens. A total of about 30 MTR capsule irradiations have been completed.

Low Intensity Test Reactor

The initial irradiation of sealed capsules in the LITR involved capsules of the same type as those described for the MTR, since the core pieces and the flux gradient pattern were quite similar. However, a much lower gamma heating rate and closer proximity to the fuel-loaded core permitted a capsule in which the full 2-1/2 in. square cross section was loaded with specimens. The intimate contact between specimens and between the specimen framework and the capsule skin after the helium compression resulted in an insignificant thermal gradient within the capsule.

LITR capsule irradiations have been limited to about 10 capsules containing approximately 90 Charpy and 6 tensile specimens or the equivalent in two half-size capsules such as those shown in Fig. 2. The small capsules shown were used to study the effects of cyclic irradiation and annealing treatments. A five-by-five configuration of Charpy specimens are loaded into a 3/16-in. thick aluminum frame which is grooved to take up the excess stainless steel when the envelope is deformed during compression.

Another type of capsule has been used in LITR half-fuel elements. This capsule was essentially an elongated grouping of a single thickness of Charpy specimens sandwiched in a stainless steel skin. The long thin design was dictated by the space available in the fuel element from which several central fuel plates had been removed. The neutron flux variations in this design limit its usefulness. Thirty Charpy specimens were irradiated in this type capsule.

Oak Ridge Research Reactor

One capsule has been irradiated in the ORR. Again the same 2-1/2 in.-square, useful cross section was available for a peripheral core position. However, the capsules used in the MTR and LITR were not satisfactory because of the very high gamma heating rates ($\sim$ 10 watts per gram). Consequently, two independent capsules, 2 Charpy specimens thick and 4 Charpy specimens long (the uniform flux area), were fabricated as shown in Fig. 3 so that the coolant water would flow over every surface of the capsules. The

limited amount of water available for cooling core experiments and a need to accelerate water flow dictated the placement of an aluminum filler block between the two capsules as well as the spacing of the specimen capsules in the framework. Temperatures reached 450°F in this capsule.

Industrial Reactor Laboratories Pool Reactor

During 1963 an irradiation experiment involving several capsules containing Charpy specimens was irradiated adjacent to the core in the pool of the IRL reactor. The capsules used were essentially the same as the small capsules used in the Army reactors (SM-1 and SM-1A) radiation dosage surveillance program described below.

Army Reactors SM-1 and SM-1A

Long-term radiation damage surveillance in two Army reactors has required relatively small and thin capsules in order to attain specific temperatures and also because of limited space for placement of the capsules. The capsule devised involves a stainless steel framework grooved to assure no wrinkling of the cover sheath. A central support which holds the tensile specimens also is grooved. Figure 4 shows the components and various fabrication stages of this capsule which is only one Charpy thick ($\sim$ 1/2-in.) and about 7-in. long. The general outline of encapsulation techniques described above is used in making the surveillance capsules. The array of capsules shown in Fig. 5 was installed in the SM-1A reactor at Fort Greely, Alaska, before its startup in 1962. The capsule holders are designed so that new capsules may be added as irradiated ones are removed.

A double capsule (side-by-side configuration) of the same type as that in Fig. 4 was irradiated in a position representing the pressure vessel but above the core of the SM-1 reactor at Fort Belvoir, Virginia. In addition, a series of capsules filling two dummy fuel elements (Fig.6) were irradiated in core positions of the SM-1 reactor. Each capsule contains 12 Charpy specimens, 2 tensile specimens plus thermal and neutron flux monitors.

CAPSULES WITH EXTERNAL CONTROL

Sealed capsules, while useful for establishing irradiation effects trends or for exploratory studies and correlative experiments, are limited in applicability especially in the test reactor environment where ambient temperatures are quite low. Consequently, the primary objectives of the irradiation effects program are met by experiments with external control. Externally controlled experiments have been conducted in the horizontal beam port facilities of the Brookhaven Graphite Reactor and in the beam port and core facilities of the Oak Ridge Low Intensity Test Reactor.

Brookhaven Graphite Reactor

The BGR horizontal beam ports provide irradiation facilities which are quite unique when compared to the water cooled testing and power reactors. The capsule used in elevated temperature irradiations in the BGR (Fig. 7) incorporates an aluminum container of square cross section with power, thermocouple, and air (coolant) leads to the face of the reactor. Each container held 8 drop-weight specimens or 64 Charpy specimens. With such massive loadings of steel it was found necessary to insulate the specimen loading from the capsule especially from the bottom surface which is the primary heat transfer surface since it rests directly on the graphite. It was also necessary to provide air cooling to the specimens. Corrosion was minimized by using dry cooling air and by maintaining relatively high temperatures. The length of the capsule assembly was not crucial because the large core loading provided a rather large region of uniform neutron flux levels.

Low Intensity Test Reactor

The irradiation capsule for experiments in the LITR horizontal beam port facility incorporated a coolant jacket to which water flow was controlled. Finer temperature control was attained through a helium flow to the

specimen-loaded region. Experiments usually consisted of 6 drop weight, 40 Charpy, and 6 tensile specimens. The general level of temperature obtained depended upon the degree of contact surface between the cooling jacket and the experimental loading. A major difficulty with this irradiation assembly arose from the severe reduction in the neutron and gamma fluxes at points progressively farther from the core end of the beam port. The major advantages were the facility for controlling the coolant flow and the large space for specimen loading.

The LITR core facilities have provided the best opportunity for irradiation under relatively high flux conditions and with good facility for experimental control. Starting with the primary capsule described for the LITR above, certain variations, especially in the heat transfer path, have permitted irradiation of several types of steel specimens under various neutron flux and temperature conditions. One technique used for temperature control involves the development of aluminum or steel framework systems which provides a variety of temperatures in a single assembly when contact with the outer containment skin is changed by variable internal pressurization. This type of assembly is shown in Fig. 8. A refinement of this design (Fig. 9) incorporating heaters and a variable (on-off) helium supply permits precise control of any section of the assembly and further permits in-reactor annealing of one section while others are maintained at the primary irradiation temperature. Approximately 20 assemblies of this type each containing approximately 80 Charpy and 6 tensile specimens have been irradiated in various positions around the core of the LITR.

The several systems and techniques for controlling the temperature during neutron irradiation are provided in a separate report (5).

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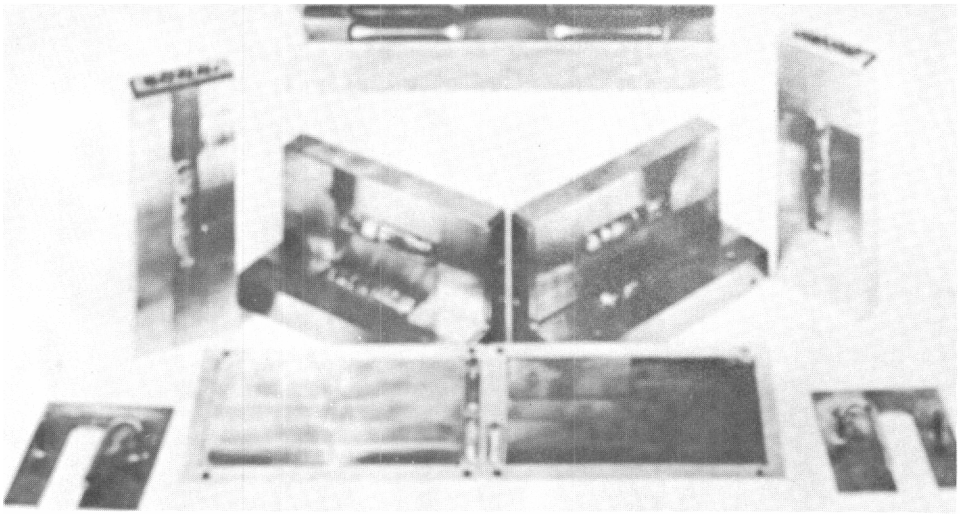
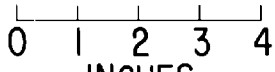
5. Hawthorne, J. R. and Steele, L.E., "Systems and Techniques for In-Reactor Temperature Control of NRL Irradiation Experiments", Conference on "Problems in Irradiation-Capsule Experiments", 8-10 October 1963, USAEC, Germantown, Maryland.

TABLE 1

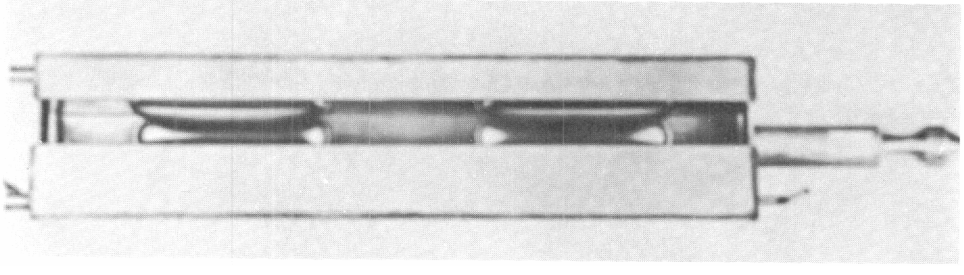
FUSIBLE EUTECTIC ALLOYS AND METALS

(°F) Melting Point	Composition (%)
217	54.0 Bi, 26.0 Sn, 20.0 Cd
228	68.0 Bi, 32.0 In
243	48.0 Sn, 52.0 In
255	55.0 Bi, 45.0 Pb
281	57.0 Bi, 43.0 Sn
291	60.0 Bi, 40.0 Cd
293	50.0 Sn, 32.0 Pb, 18.0 Cd
313	Pure Indium
349	68.0 Sn, 32.0 Cd
354	62.5 Sn, 36.1 Pb, 1.4 Ag
361	61.9 Pb, 38.1 Sn
390	91.0 Sn, 9.0 Zn
419	85.0 Pb, 15.0 Au
430	96.5 Sn, 3.5 Ag
440	99.4 Sn, 0.6 Cu
450	Pure Tin
464	4.0 Sn, 12.0 Sb, 84.0 Pb
475	90.0 Sn, 10.0 Sb
478	83.0 Pb, 17.0 Cd
486	88.8 Pb, 11.2 Sb
509	17.5 Zn, 82.5 Cd
520	Pure Bismuth
554	92.5 Cd, 7.5 Sb
579	2.5 Ag, 97.5 Pb
590	97.5 Pb, 1.75Ag, 0.75 Sn
604	99.5 Pb, 0.5 Zn
610	Pure Cadmium
621	Pure Lead





DROP-WEIGHT AND CHARPY SPECIMENS



AFTER ENCAPSULATION IN STAINLESS STEEL SHEATH

Fig. 1 - Typical capsule assembly for low temperature irradiation in Materials Testing Reactor

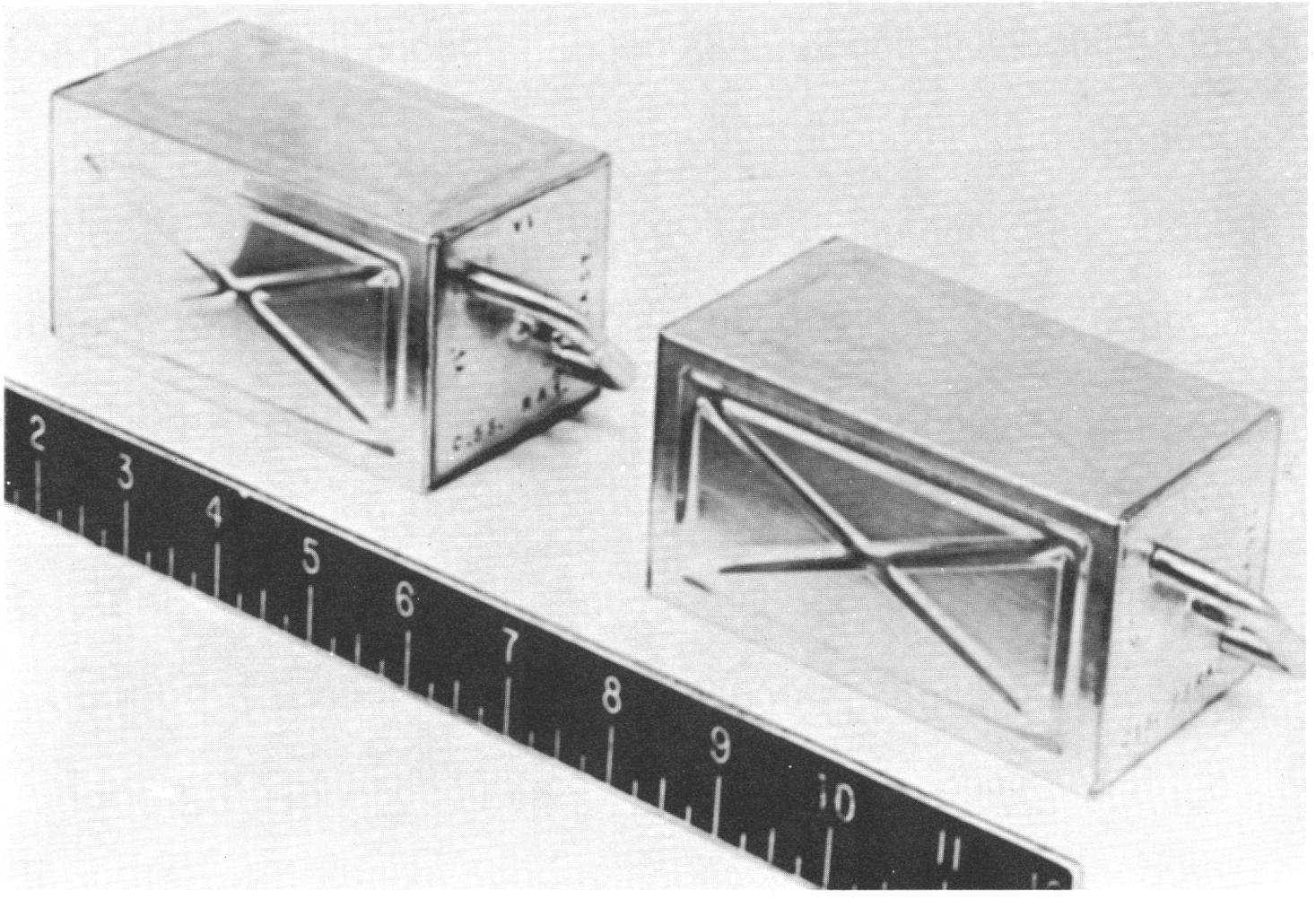


Fig. 2 - Capsules for low temperature irradiation in the LITR
(for cyclic irradiation-annealing studies)

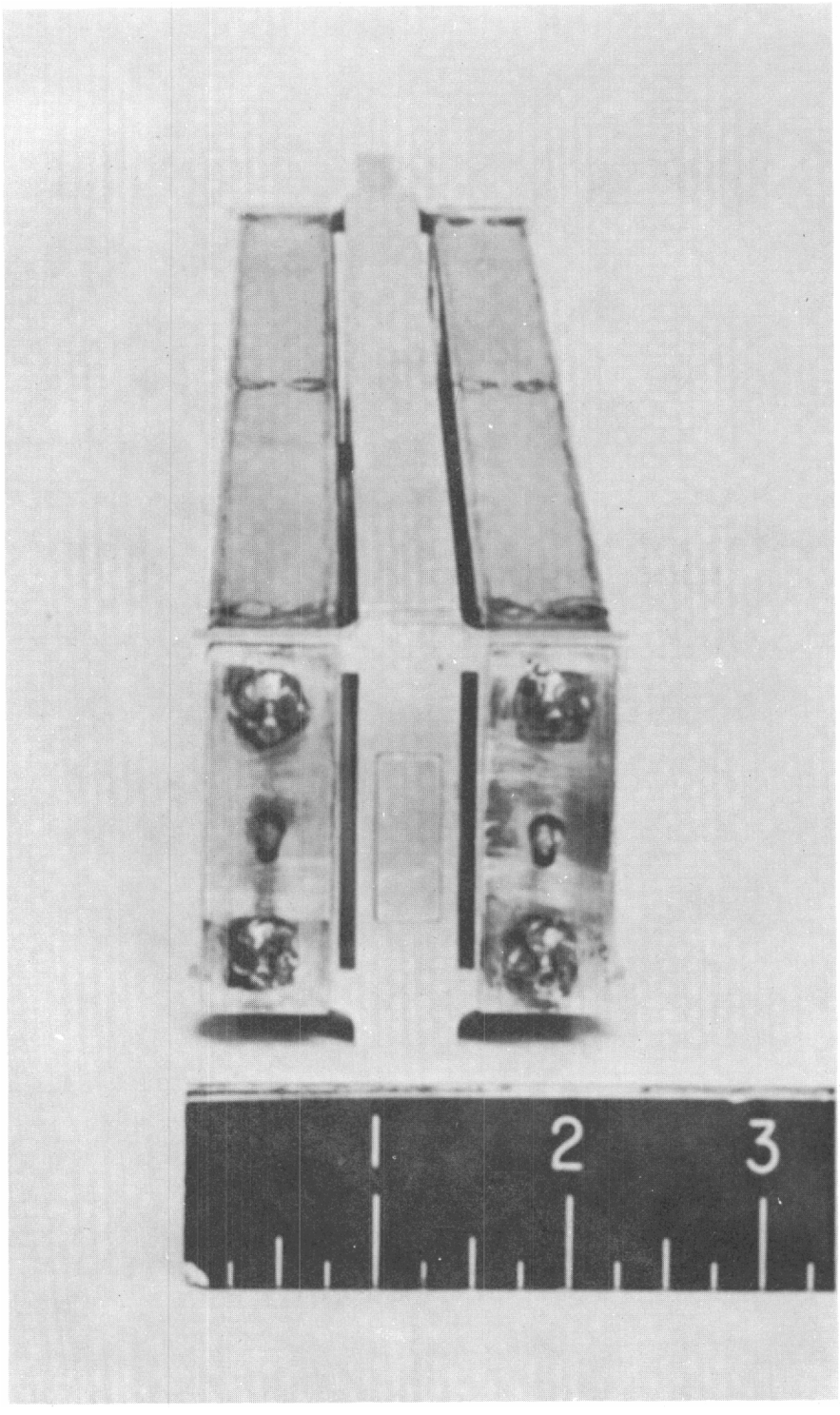


Fig. 3 - Capsule for irradiation of Charpy and tensile specimens at low temperatures in the Oak Ridge Research Reactor

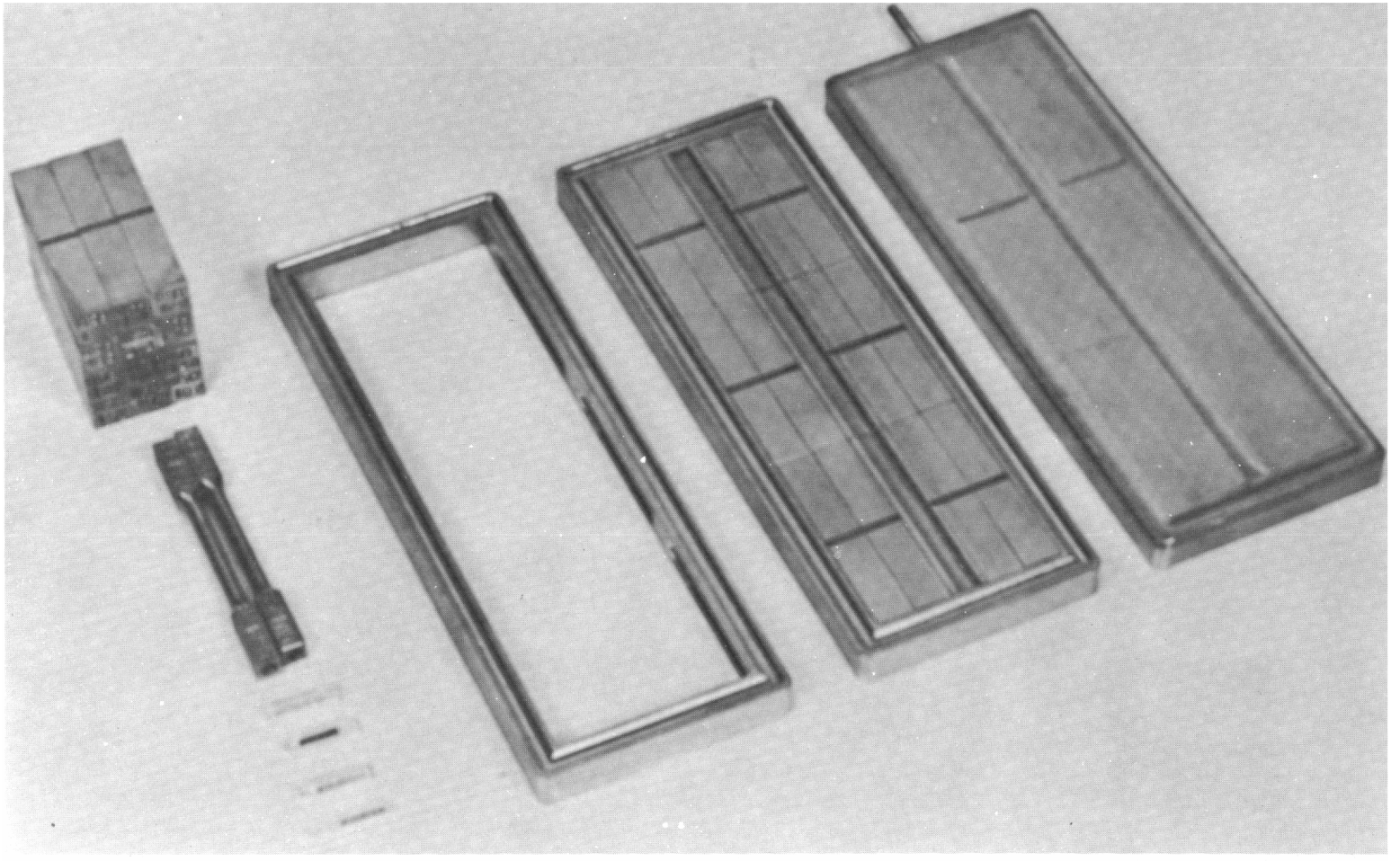


Fig. 4 - Surveillance capsule showing various components before and after encapsulation. Dosimeter wires (not shown) are placed in V-grooves of Charpy specimens and along gage length of tensile specimens

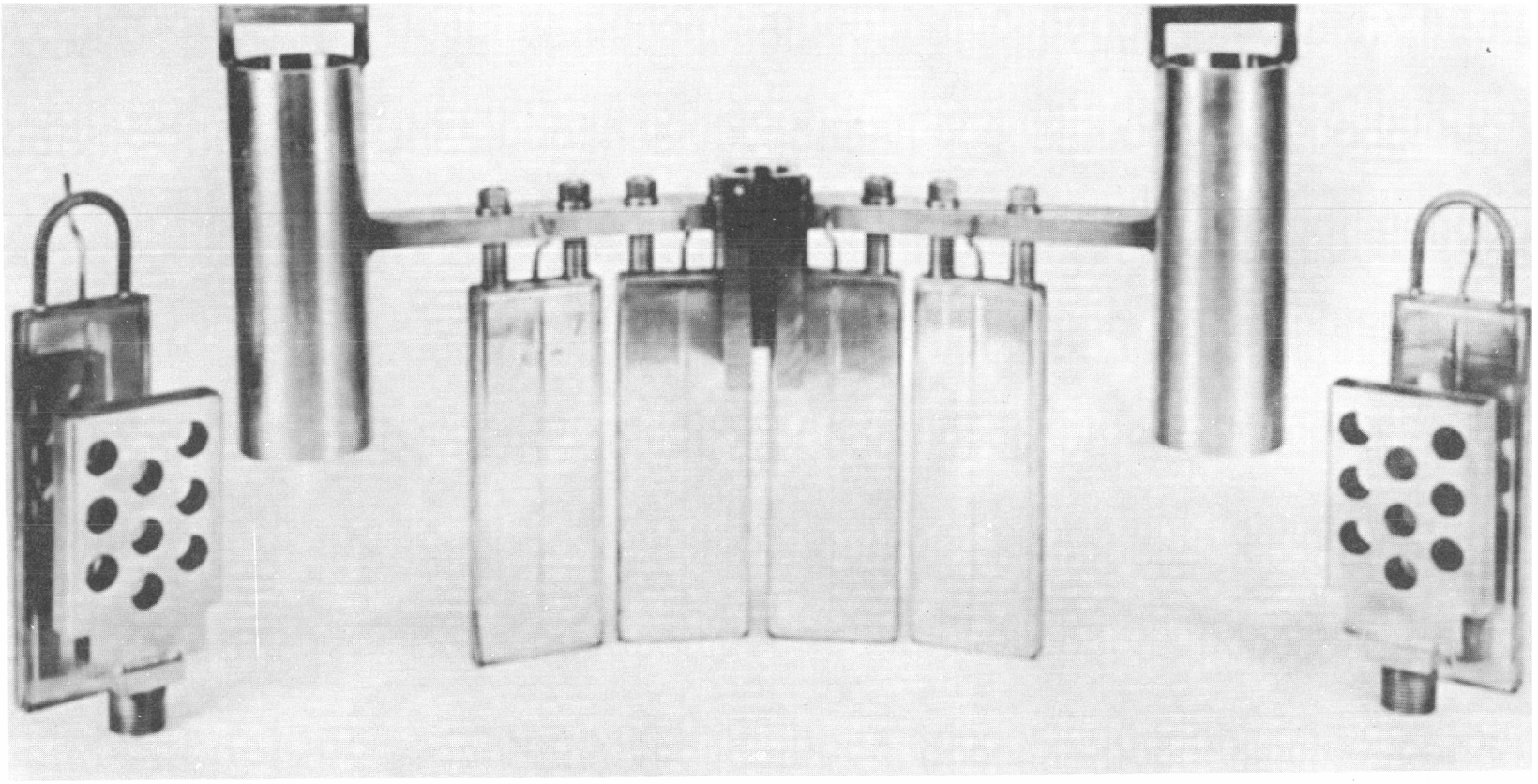


Fig. 5 - SM-1A pressure vessel surveillance assembly. Curved assembly is mounted above and to side of core. Individual units fit in peripheral position near top corners of core.

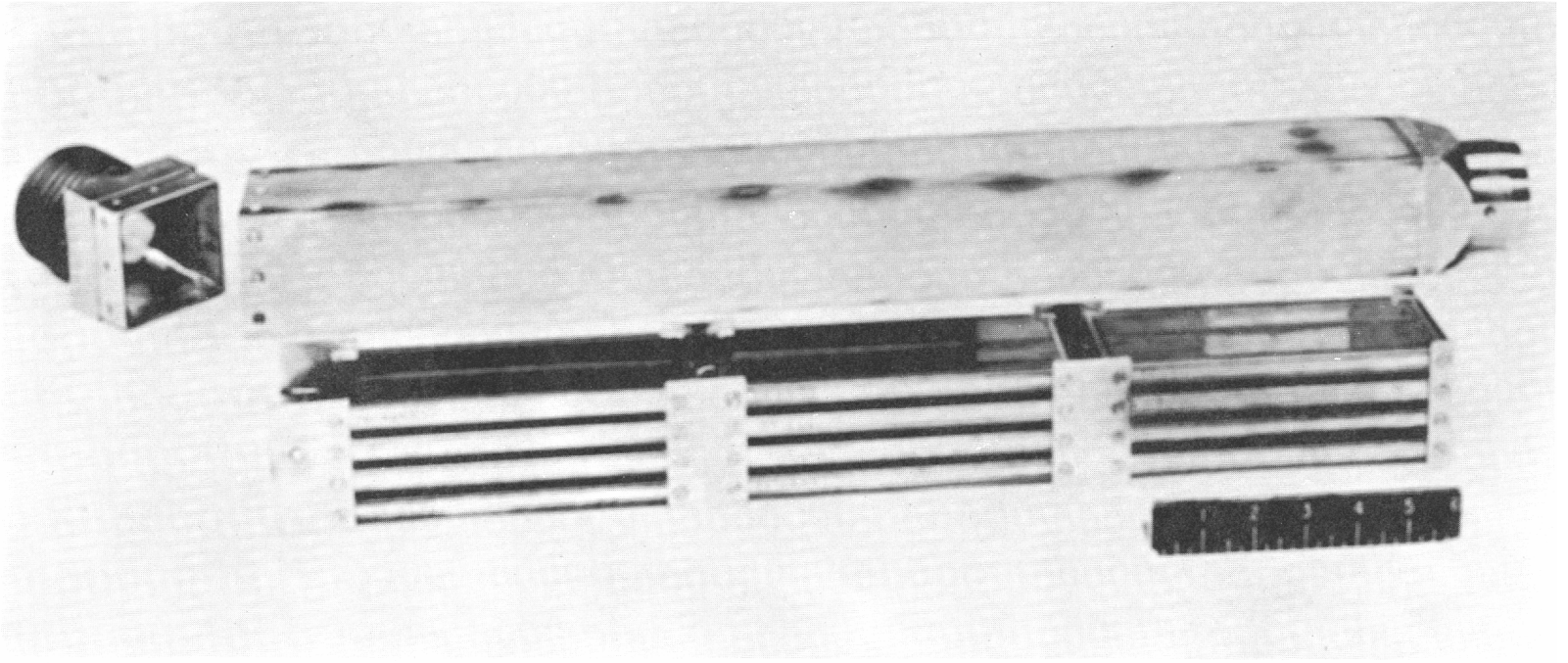


Fig. 6 - Charpy-V and tensile specimen assemblies for core position in Army SM-1 reactor

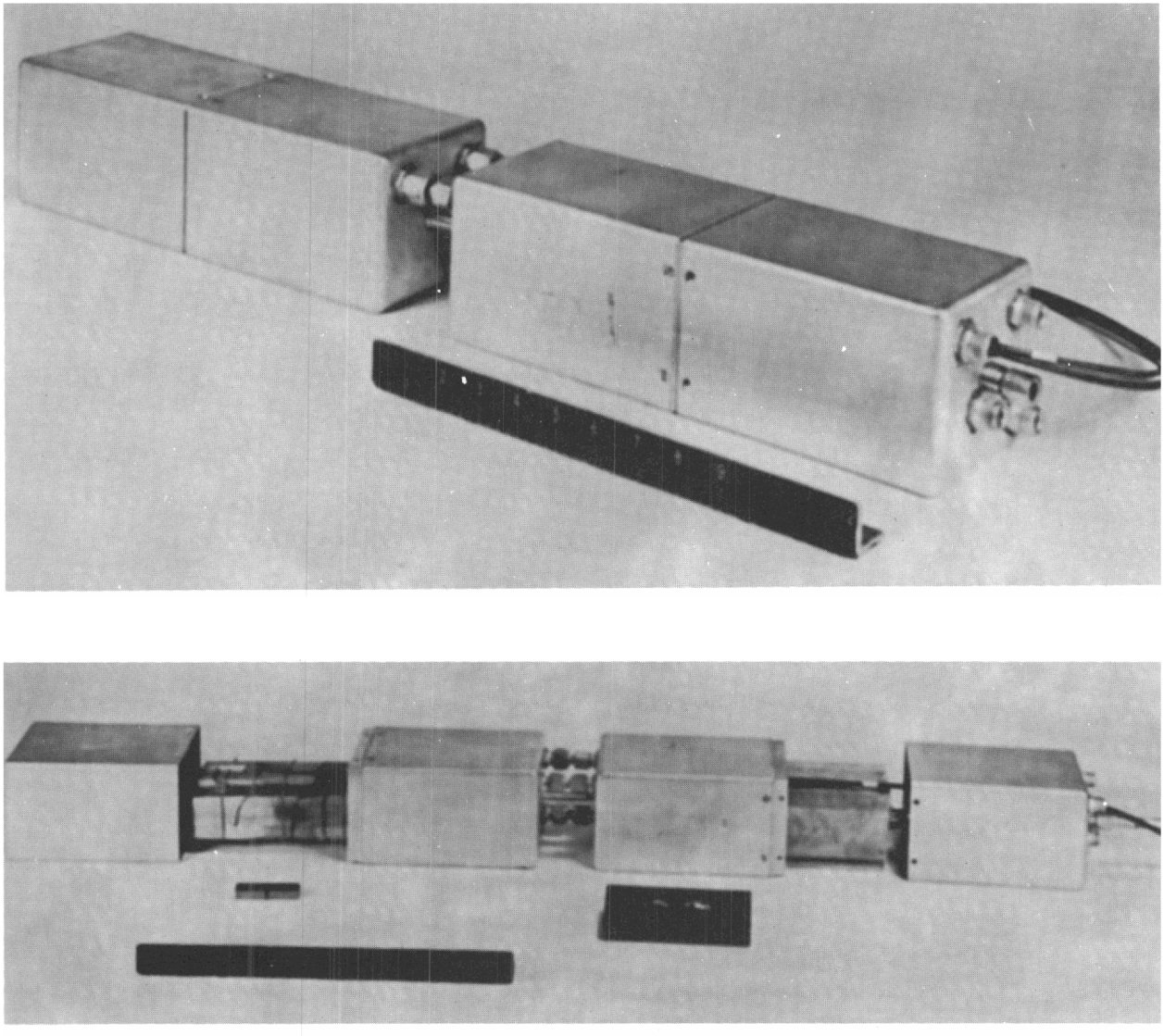


Fig. 7 - Typical assembly for elevated temperature irradiation of Charpy-V and drop weight test specimens in Brookhaven Graphite Reactor

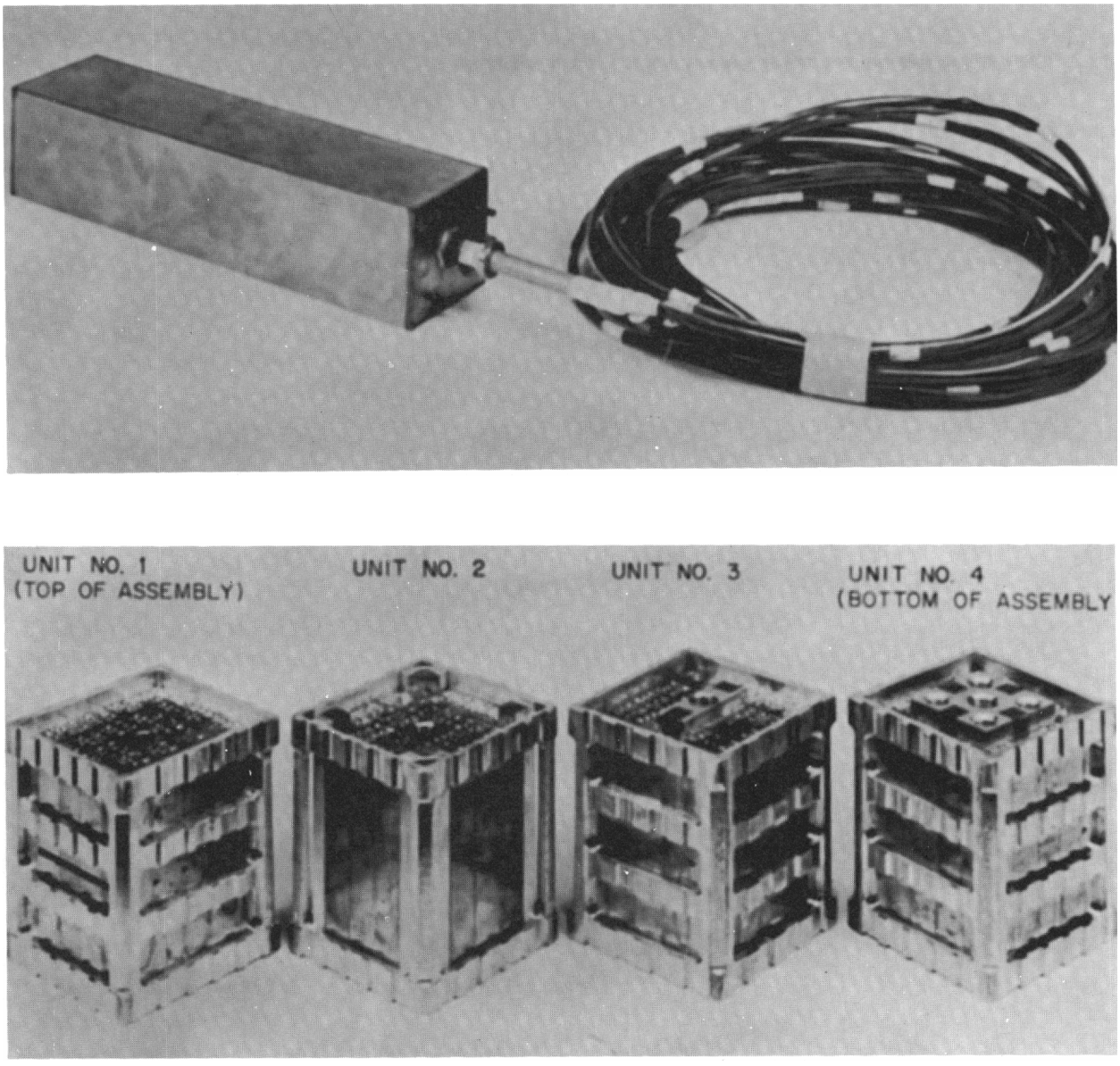


Fig. 8 - Assembly for multitemperature irradiation before and after encapsulation. Variations in heat transfer are accomplished through difference in aluminum framework around specimens

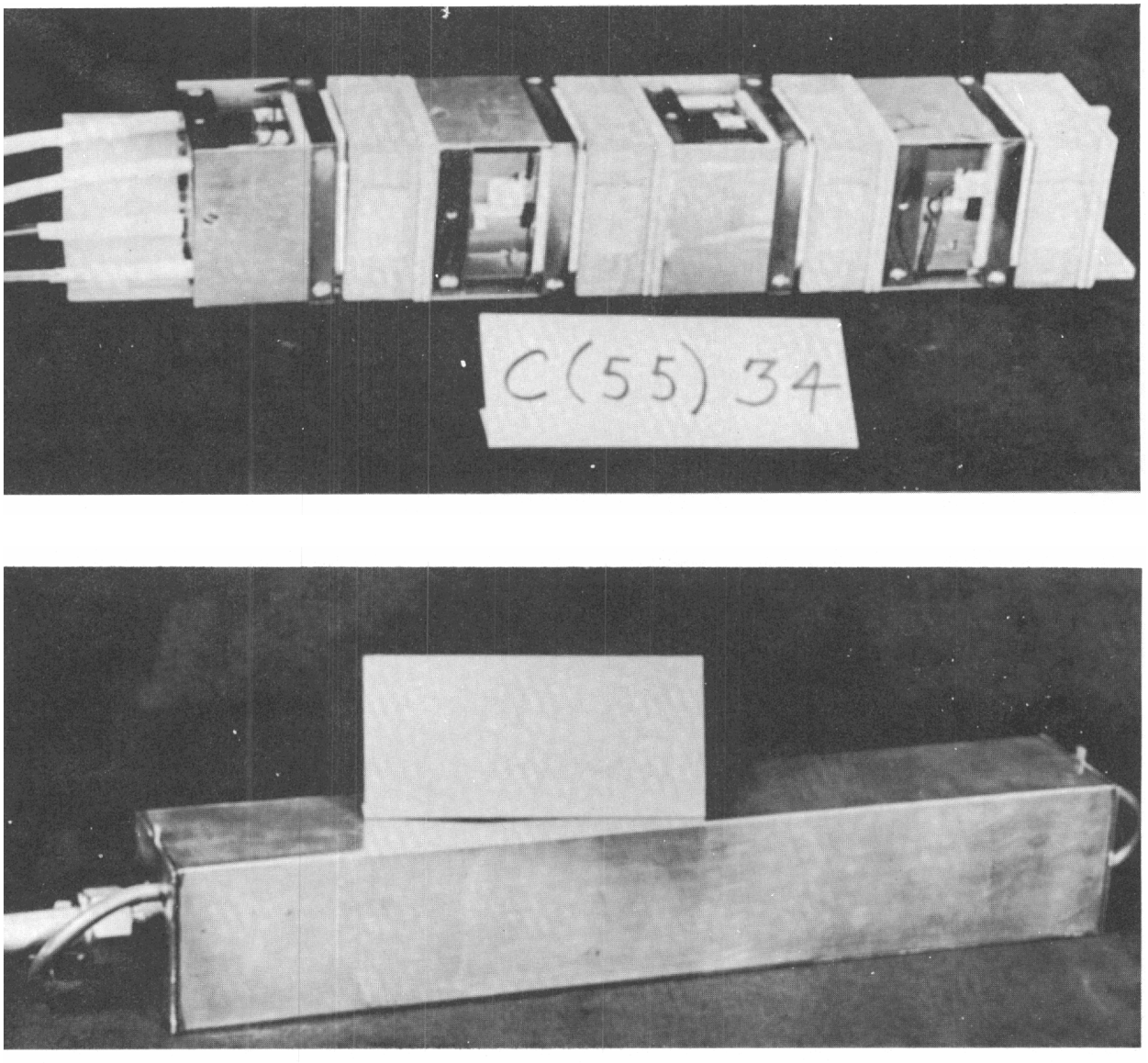


Fig. 9 - Multi-heater assembly for elevated temperature irradiation of Charpy specimens (before and after encapsulation)

Proposed High Temperature Capsule Design for Steel Irradiation

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Usually, high temperature irradiation test for steel or any other materials are performed by using an instrumented capsule with electric heaters or a high temperature loop facility in which the specimen temperature can be controlled through coolant flow.

Due to high gamma heat flux in most material testing reactor, the capability of temperature control by a heater inserted in capsule is very poor because test specimens produce considerably large amount of heat comparing with the heat supplied by heaters.

The newly introduced idea for high temperature capsule is the use of built-in single or multi thin low melting point alloy layers to realize the specified temperature drop in them.

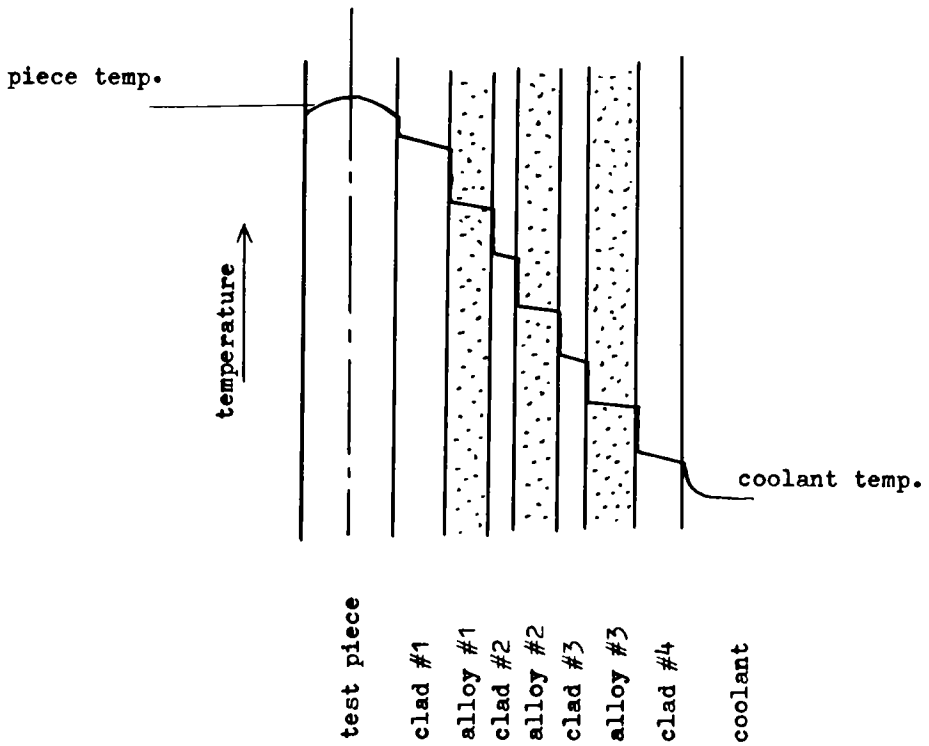
Fig. 1-a illustrates the temperature profile through laminated low melting point alloy and matrix metal layers before the alloy melting. When the heat generated in specimens increases, the layers of low melting point alloy become to melt. The alternated temperature profile is shown in Fig. 1-b in which the temperature of specimen is recognized to be kept in the specified temperature range. The self temperature control will be realized by selecting the adequate low melting point alloy and the layer thickness and number which must be determined by a mock-up test in laboratory. As clearly shown in Figs. 1-a and 1-b, the fact that the heat resistance between solid to solid surfaces is larger than that between solid to liquid surfaces, can achieve the temperature controlling of specimen in high temperature irradiation capsule.

In Fig. 2, the specimen temperature range which can be controlled by this system is shown. Between points a and b, though the heat flux changes largely, specimen temperature would be kept in a narrow temperature range.

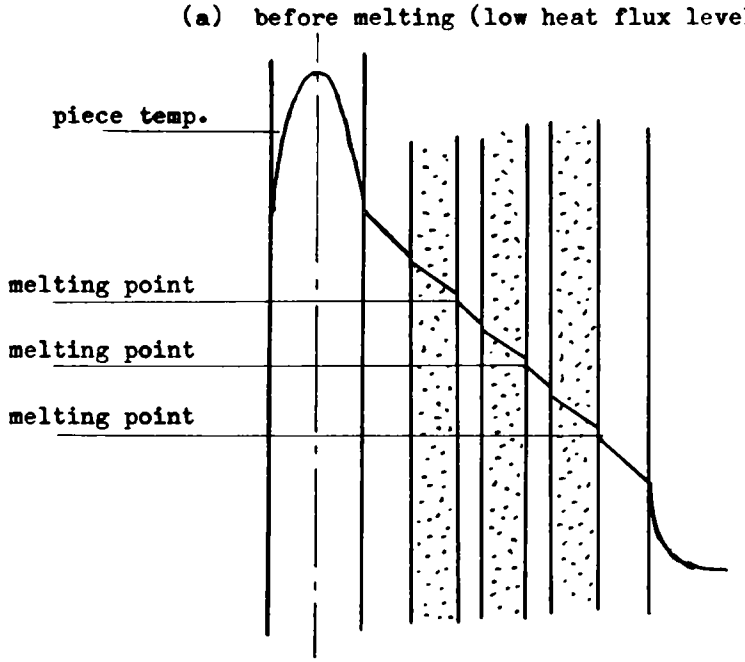
The alloy must be selected to meet the following requirement.

a) no trouble on compatibility to matrix metal when the alloy melts
b) good wetability to matrix metal
c) narrow temperature range for melting point such as having an eutectic phase
d) adequate melting point
e) no explosive reaction with reactor coolant

Fig. 3 shows the irradiation Charpy specimen accompanied by the self temperature controlling system. The temperature of the specimen is planned to be kept in 250°C ± 30°C during the irradiation. The laboratory measurements of contact resistance coefficient show the value of 2.5 to 6.7 x 10^3 kcal/m^2hr°C for Pb-Sn alloy solidified in air and aluminum. When the investigation clarify the feasibility of this system, this is hoped to be used in the steel irradiation test of Iron and Steel Institute of Japan.



(a) before melting (low heat flux level)



(b) after melting (high heat flux level)

Figure 1 Temperature distribution of self-controlling heat transfer wall

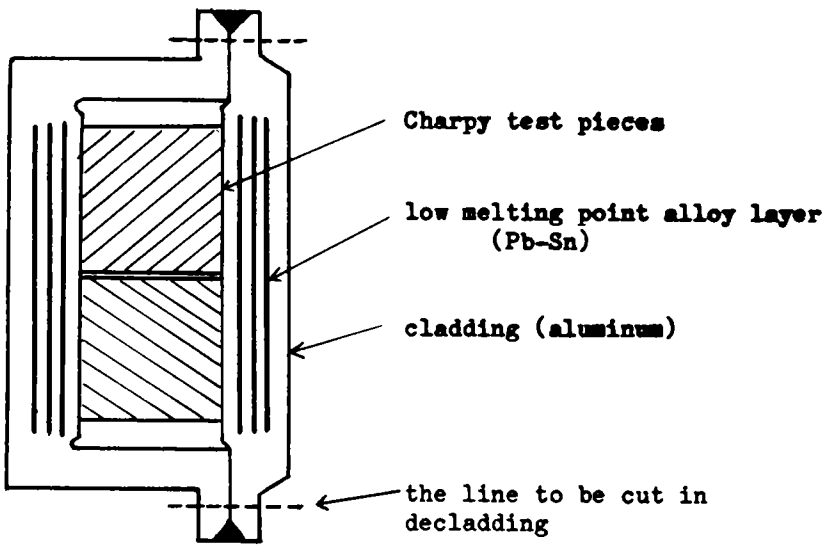


Figure 3 Cross section of test piece

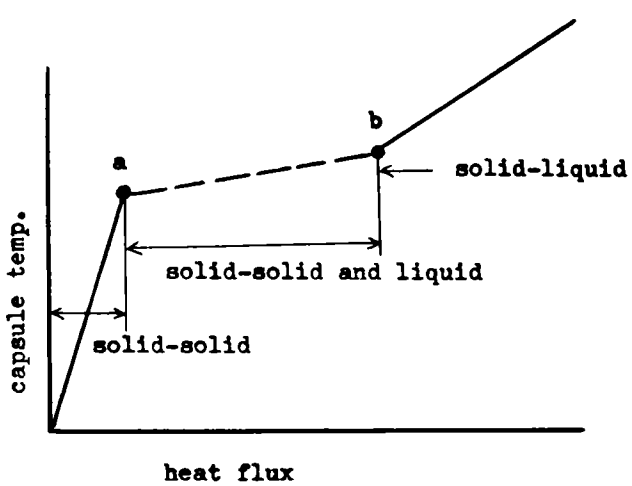


Figure 2 heat flux vs. capsule temperature

Capsules for Irradiation of Pressure Vessel Steel Specimens

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Iron and Steel Institute of Japan (ISIJ) has a committee composed by 15 companies, 6 universities and 2 national laboratories in Japan which is conducting the irradiation test of the reactor pressure vessel material, ASTM A302B steel, in 1963 - 1964, having a partial subsidizing from Japanese Government. MAPI is working for ISIJ on the capsule design and fabrication, performing all irradiation program. As shown in Table 1, the rough outline of this program is spread widely. Specimens to be irradiated are totally 110 Charpy test pieces, 10 tensile test pieces, and 56 test pieces for basic research purposes. These specimens will be contained in 5 capsules. Two of the capsules are planned to be irradiated at 260 ± 50°C, and other three to be maintained at the temperature less than 150°C.

Fig. 1 shows the outline of low temperature irradiation capsule. Two Charpy test pieces or one tensile test piece are enclosed in one aluminum cladding. This cladding is cooled directly by coolant water flowing in capsule to maintain the specimen temperature less than 150°C and to prevent a large thermal stress caused by one side cooling on specimen.

Fig. 2 shows the cross section of aluminum cladding having self temperature control system in it. This is for high temperature irradiation. The cladding encloses two Charpy test pieces and has several layers of Pb-Sn alloys (or other Pb- alloys) to control the temperature of test pieces.

The principle of temperature controlling intends to be explained in another topic of this conference.

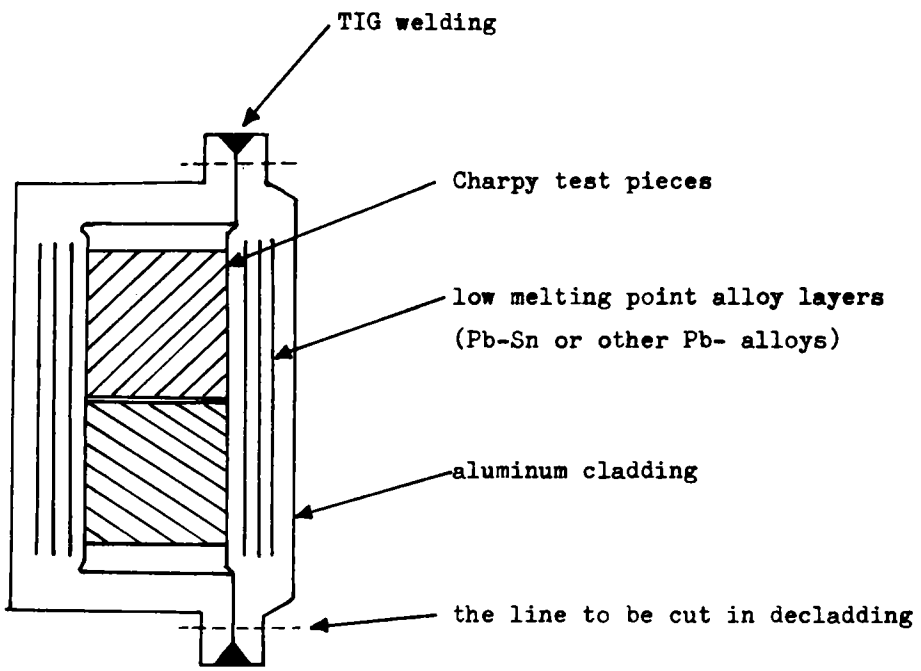

Fig. 2 Cross section of aluminum cladding using for high temperature irradiation capsule

Table 1 Irradiation test program of pressure vessel steel

Materials			Kind and quantity of test piece		Irradiation condition	
Kind	Supplier	Description	Charpy	Tensile	Irradiation temp. (°C)	Integrated flux (nvt > 1MeV)
A302B steel	A	250 m/m plate	16		260 ± 50	~3 x 10^{19}
		150 m/m plate	16			
	B	100 m/m plate (fabricated by B Co.)	12	2	< 150	
		100 m/m plate (reference plate of B Co.)	12	2		
	C	150 m/m plate	12 (half size)			
			2			
			2		260 ± 50	
		100 m/m plate	2			
			12 (half size)		< 150	
	F	with Ni	4			
	D	made by converter	12			
	E	with AlN	2			
			6	2		~1 x 10^{19}
Iron and its alloy	F	Fe-C, Fe-C-Mn, Fe-C-Mn-Si, Fe-C-Mn-Si-Mo		4		~3 x 10^{19}
	G	Fe, Fe-Cr, Fe-Ni, Fe-Mo, Fe-Mn, etc.	for electrical resistivity, internal friction, and bending property measurements, etc. 56			~1 x 10^{19}

Total quantities of test pieces	Charpy	tensile	for basic research
	110	10	56

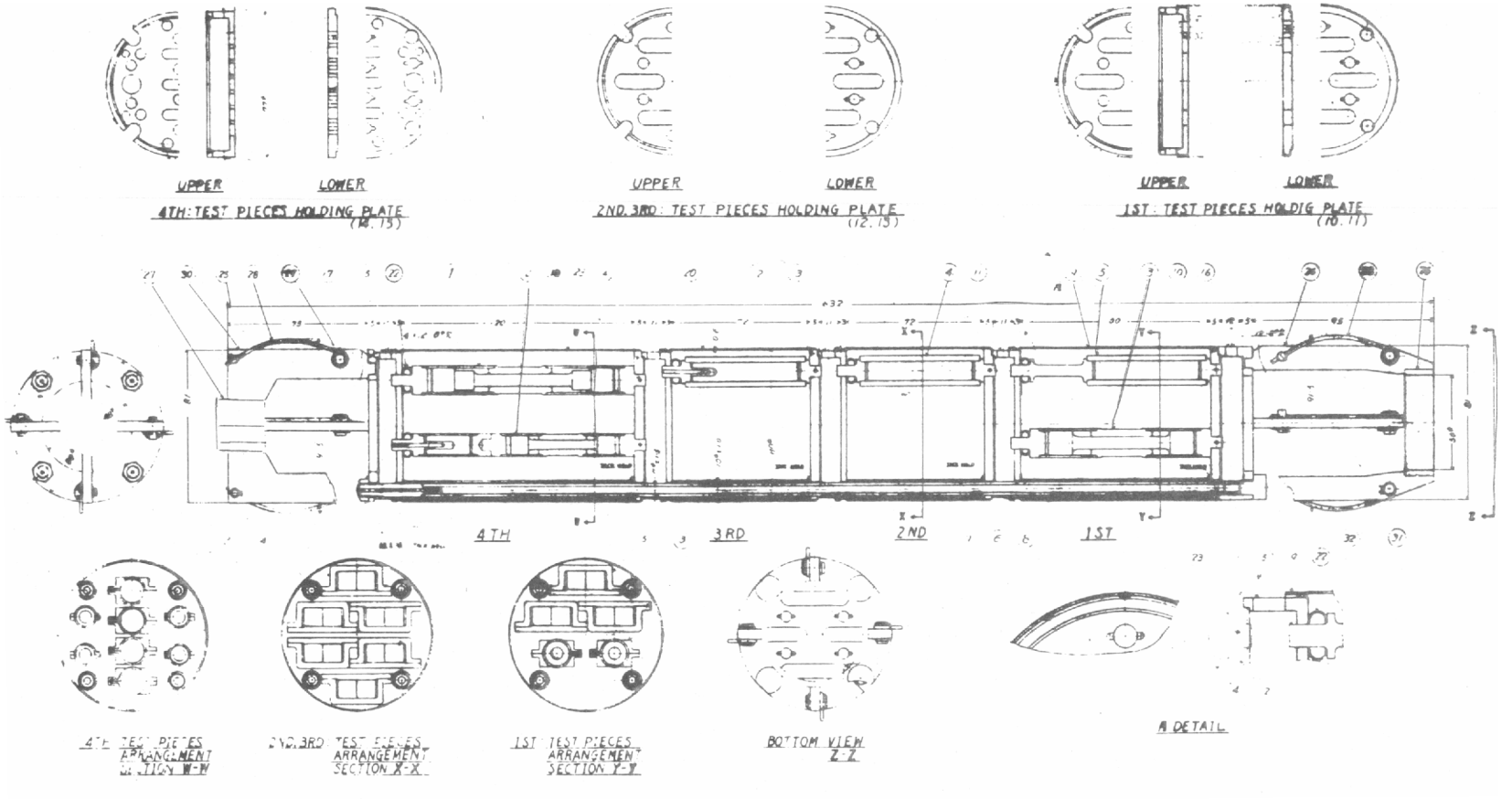

Fig.1 Low temperature irradiation capsule

Capsule for Irradiation of Pin Specimens of Stainless Steel Sheaths Removed from Swaged UO_2 Fuel Rods

S. Takahashi, Y. Honda and Y. Ikawa
Engineering & Research Laboratory
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To know the neutron irradiation effect on stainless steel cladding material of swaged UO_2 fuel rod, 105 pin specimens were irradiated in the core position at WTR up to 1×10^{21} nvt (>5 Kev) in 1961. The specimens were taken from sheath tubes by milling machine after the swaged UO_2 compact in tube had been dissolved out by nitric acid. Thirty five different sheaths of fuel rods (different composition in Ni and Carbon, different swaging degree and different heat treatment) were prepared. From each rod, three pin specimens were taken. The work hardened layer on the cut surface of the specimen was removed by an electrolytic polishing technique as much as 100 μ.

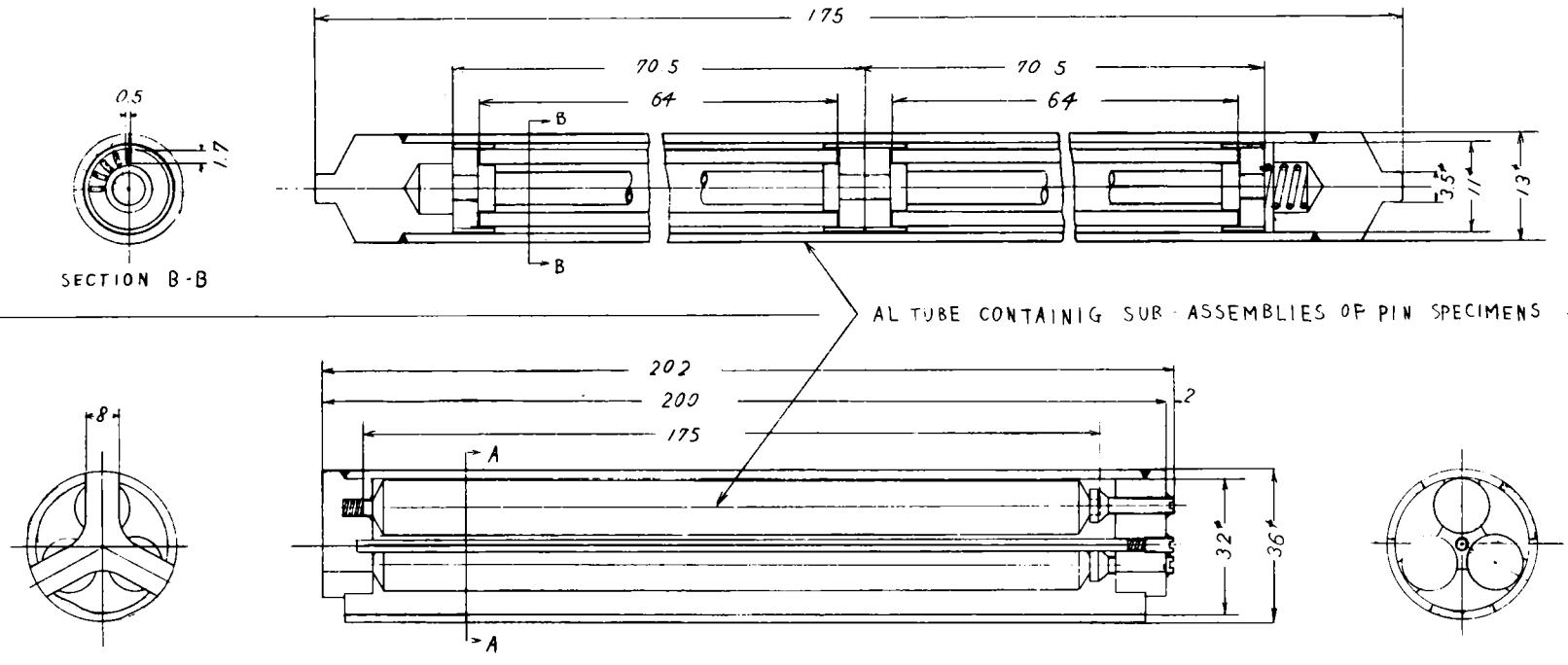
For the irradiation, as possible as small volume of the capsule and no harmful effect by γ ray heating had been desired. Considering those requirements, two types of sub-assembly were designed by MAPI. One of them is shown in the attached drawing (No.A0025) for the specimens having the dimension of (1.7 x 0.5 x 64 mm).

Each specimen was put in a slot of the end discs (Al) which were assembled with the center rod by screw and head cap rings. One sub-assembly was able to contain 16 specimens without contacting each other. Two of them were inserted in an aluminum tube having several slots through which coolant was possible to enter to prevent the γ-ray heating effect on the specimens. Both ends of tubing were plug welded.

The last sub-assembly of six was a little different from the remain. It could contain 25 specimens having the dimension of (0.9 x 1.0 x 64 mm). In this sub-assembly, the specimens having rectangular cross section were arrayed in ring grooves in both end discs.

Three aluminum tubings each containing two sub-assemblies were assembled in one capsule to assure to have the same irradiation. A flux monitor wire (Ni 0.5 mm ϕ) sheathed by stainless steel small diameter tubing was assembled at the center line position of the capsule.

This kind of capsule and sub-assembly were found very useful to irradiate many small specimens at once and easy to disassemble in hot cell.



1.5.2

TITLE	IRRADIATION CAPSULE FOR PIN SPECIMENS OF STAINLESS STEEL
DRAWING NO	A0025
DATE	AUG. 10. 1960
DESIGNED BY	MITSUBISHI ATOMIC POWER IND., INC

IRRADIATION OF NON-FISSILE MATERIALS IN THE ORR CORE AND POOL FACILITIES

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SUMMARY

The Oak Ridge Research Reactor (ORR) is a light water-cooled beryllium reflected testing reactor. Facilities are available for testing of materials in thermal and fast fluxes of up to 2×10^{14} and 1×10^{14}, respectively. The reactor is cooled by forced convection, therefore, access to the in-core facilities is obtained through flanges in the top of the reactor core tank. This report concerns the facility that is used for in-pile stress-rupture and creep experiments located outside the reactor core tank (P-5) and a facility located in the core (B-8) that is used in conducting materials irradiations at controlled temperatures.

The P-5 facility is 1 ft x 2 ft x 2.5 in. in size. Access to instrumentation for temperature measurement, gas pressure, and furnace power is through a 2-in. tube to a junction box above water at the poolside. Permanent leads are taken from the junction box to "patch" panels in the experiment control room. The control room contains sufficient instrumentation to separately control the temperature of 80 specimens in the P-5 and B-8 facilities.

The B-8 core facility is 2.5 in. x 2.5 in. x 2 ft in size. Access to instrumentation is similar to that for P-5 except that the access tube passes through a flange in the top of the core tank before it reaches the junction box at the pool surface.

Stress-rupture and creep experiments have been conducted in P-5. The large size of this facility allows approximately ten specimens to be tested simultaneously at temperatures from 500 to 900°C. The stress-rupture experiments are performed by internally pressurizing tube specimens and monitoring the internal pressure to detect failure. Specimens approximately 3 in. long having diameters from 0.4 to 0.75 in. with wall thickness from 0.010 to 0.020 in. have been tested. Temperature control of the specimens is obtained by three separately controlled furnaces disposed along the length of each specimen. The specimens attain temperatures in the

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* Operated for the U. S. Atomic Energy Commission by Union Carbide Corporation.

range of 500°C on gamma heating alone and the furnaces raise the temperature of the specimen uniformly to the desired temperature. The temperature gradient in the specimens is less than ± 2°C. Bare, 0.010-in. diam Chromel-P-Alumel thermocouples are used to monitor the temperature at three positions along each specimen. The furnaces are constructed of type 304 stainless steel sheathed, MgO insulated, Nichrome heater wire. The junctions between the heaters and the nickel leads are of a special design to allow their use in-reactor. The environment surrounding the specimens is He - 1% O_2. This gas was chosen because of its high thermal conductivity, because it should produce no spurious environmental effect in the specimens, and the thin adherent oxide film produced on the thermocouples tends to protect the fine wires from internal oxidation that may occur at smaller oxygen partial pressures.

The important features of the graphite creep experiments that have been conducted in P-5 are the loading system and the method of measuring the deformation. The loading system consists of a bellows and lever arm connected to each specimen. The bellows is evacuated and the pressure in the experiment capsule acts to compress the bellows thus applying force to the specimens.

The extensometer that has been most successful has been a metallic frit film potentiometer. The potentiometer is placed in a voltage divider circuit and the current is "nulled" between the two legs by adjusting the ex-reactor potentiometer. The system is quite stable and has a sensitivity of ± 0.0005 in.

It is important to know the thermal and fast flux in various positions in the facilities used in material irradiations. The spectral distribution of flux may be obtained by appropriate use of CO and cadmium-shielded cobalt for thermal-flux measurement and "threshold" detectors for the flux in the fast range. The fast flux monitors that have been used and the thresholds above which the integral flux is obtained are shown below:

Detector	Initial Reaction	Effective Threshold, Mev
Np^{237}	(n,f)	0.67
S^{32}	(n,p)	2.9
Ni^{58}	(n,p)	2.9

Satisfactory fast flux monitoring in these facilities has been difficult because of the following problems: (1) lack of satisfactory cooling in the high nuclear heating regions, (2) high neutron flux and the necessity to allow the flux monitors to remain in the facility for large exposures, (3) unknown cross

sections and yields for some of the side reactions in the neptunium and nickel activation chains, and (4) difficult computation of flux from activation data due to complex activation chains and reactor operation.

Most of these problems have been solved by making short reactor runs at low power and extrapolating the data, by cadmium shielding to reduce the thermal contribution to the fast flux monitors, and by the use of computer data processing to handle the unwieldy data reduction problems.

HIGH TEMPERATURE GRAPHITE IRRADIATION RIG, SD 486

S. L. Nayler
A.E.R.E. Harwell

1. Introduction

This rig was initially conceived as a prototype rig and was intended to verify a particular design concept i.e. use of nuclear heating only and reduction of radiative heat loss by use of polished surfaces. The performance of the rig was sufficiently acceptable to the experimentalist that a series of rigs to the same simple basic design has been produced and the fourth of this series is now in operation in DIDO reactor.

The rig is being used for the irradiation of a number of graphite specimens at a temperature in the range 1200°C to 1300°C in a Mk. III fuel element of DIDO reactor. Control of the specimen temperature is by movement of the specimen capsule up or down within the reactor core.

2. Rig Design

The essential points of the design are illustrated in Fig. 1. The graphite specimens are located in a graphite carrier which is contained in a niobium capsule. The capsule is located centrally in the rig thimble by upper and lower locating drums of stainless steel and is suspended by a stainless steel tube which extends up through the rig shield plug and is moved up and down by a screw mechanism in the reactor void. The capsule body and connecting tube is machined from solid niobium bar to ensure concentricity and avoid niobium welding operations which have been found to be most time consuming and costly.

Most of the heat required is generated in the thick walled niobium capsule and flows across the adjacent annular gap to the thimble and hence to the D_2O flowing up the centre of the fuel element. Conduction across the annulus is minimised by filling with Neon gas and radiation is reduced by polishing the capsule body and providing a thimble liner internally plated with platinum to reduce its thermal absorbtivity. To avoid a large temperature gradient along the capsule the masses of material are arranged to give a uniform nuclear heating rate per inch except at the ends where more material is added to compensate for conductive heat loss in the end-wise direction.

To facilitate movement of the capsule the thermocouples are lead up the centre of the suspension tube to a terminal block mounted at its upper end, and the complete assembly moves as a whole.

One alternative design made use of a vacuum in the thimble to avoid conduction heat losses. Due to the predominance of radiation at 1500°C the gain in attainable temperature was small and since maintaining the vacuum and providing adequate safeguards in case of a D_2O leak would pose serious problems the idea was

abandoned. A series of radiation shields was considered but was rejected due to the difficulty of avoiding distortion and hence contact in the narrow annular space available.

3. Calculations

3.1. Heat Transfer

(a) convection across annulus.

$$\frac{Kc}{K} = \phi\,(NRG, \frac{H}{L})$$ ref. Jacob Vol. 1 pp. 534

K = thermal conductivity

Kc = effective K including convection.

With values appropriate to the capsule $\frac{Kc}{K} \approx 1 \cdot 0$

i.e. convection is negligible.

(b) conduction across annulus.

Conventional equation $$Qc = \frac{2\pi KL\,(T1-T2)}{\ln \frac{r2}{r1}}$$

(c) radiation across annulus.

Due to the large difference in temperature between the two surfaces the absorbtivity cannot be assumed equal to the emissivity as in conventional calculations. In fact

a_{12} = cpnst.x $\sqrt{T1T2}$ for metal ref. McAdams 3rd Ed.

conventional eqn.:- $$QR = \frac{A1\,\varepsilon_1\,\sigma(T1^4-T2^4)}{1 + \varepsilon_1 (\frac{1}{\varepsilon_2} - 1) \frac{r1}{r2}}$$

ε = emissivity

a_{12} = (absorbtivity of (surface 1 seeing (radiation from 2

Exact. epn:- $$Q_R = \frac{A1\,\varepsilon_1\sigma T1^4}{1 + A_{12} (\frac{1}{A_{21}} - 1) \frac{r1}{r2}} - \frac{A2\,\varepsilon_2\sigma T2^4}{1 + A_{21} (\frac{1}{A_{12}} - 1) \frac{r1}{r2}}$$

With the numerical values appropriate to this case we have

$$\frac{Q_R \text{ exact}}{Q_R \text{ conventional}} = 2 \cdot 7$$

(d) End-wise conduction from the capsule; this was calculated by a numerical step-by-step process accounting for the heat loss from successive small sections of the connecting tube and the nuclear heating in each section.

3.2. Reactivity Absorbtion

Basic equation $\rho = \frac{\varepsilon_{aV}}{I} \phi_2 \phi_2^*$ Ref. 'Use of flux adjoint tables' W.R. Hobbs

The reactivity change due to capsule movement was calculated by summing the contribution of each part of the moveable assembly, accounting for the differences in vertical position, and repeating the calculations for several positions.

4. Instrumentation

The only measurements are of temperature and pressure.

Two types of thermocouple are used.

(i) in the high temperature region i.e. the capsule, M.I.C. Pt-13% Rh/Pt with Pt-Rh sheath $\frac{1}{16}$" dia.

(ii) in a lower temperature region i.e. on the capsule connecting tube, M.I.C. chromel/alumel with S.S. sheath 1 mm dia.

The Pt-Rh/Pt thermocouples are known to have a short useful life due to transmentation of Rh to Pd and thus the C/A T/C's were utilised to enable continued operation of the rig after initial correlation of the high and low temperatures. The possibility of failure of the Pt-Rh sheath due to contact with graphite at high temperature was appreciated and the precautions taken to ensure that the graphite of the specimen carrier was of the highest purity and that the T/C were perfectly clean. Post-irradiation examination of the T/C's has shown that no attack took place at temperatures up to 1400°C, the sheaths showing no change in their appearance. Alternative high temperature thermocouples are being considered in order to extend the duration of direct temperature measurements.

5. Safety

Item	Event	Effect if Uncorrected	Corrective Action	Effect of Action
1	Leak of air into thimble	Capsule temp. increased by 160°C followed by capsule oxidation.	Locate and repair leak; repurge with neon.	Temp. and pressure return to normal.
2	Leak of air into capsule	Graphite temp. increased by 30°C.	Locate and repair leak; repurge with neon.	Temp. and pressure return to normal.
3	Leak of D_2O into thimble.	High steam pressure generated, experiment ruined.	Remove rig and inspect for damage.	Requires reactor shut-down.
4	Capsule drops to bottom of thimble.	Approx. ·12% reactivity added resulting in power excursion.	Stabilise reactor power.	Normal operation of reactor maintained - rig to be inspected at next shutdown.
5	Blockage of D_2O cooling flow in fuel element.	High temperature warnings persist.	Remove rig to clear blockage.	Requires reactor shut-down.
6	T/Cfailure	Indicates full-scale recorder temp.	If other T/C's O.K. disconnect faulty one.	Continue operation.

6. Special Materials

Niobium was chosen for the heat generating capsule due to its low neutron absorbtion cross-section 1·1 barns/atom high melting point (2450°C) and strength at 1500°C. The graphite specimens are located in a graphite holder to avoid possible seizing and graphite diffusion into the surrounding material.

7. Operational Experience

To date three rigs have been operated in DIDO reactor. The first of these operated at a specimen temperature of 1200°C using a mixture of He and Neon in the rig. The temperature control by vertical movement of the rig was found to be quite adequate and simple to operate. The effect on reactor control was barely perceptible. With the exception of one Pt-Rh/Pt T/C failure start-up all the T/C's operated satisfactorily throughout the whole irradiation period of 3 months.

The second rig was operated up to a maximum specimen temp. of 1440°C using pure Neon in the rig. Throughout the irradiation period of $4\frac{1}{2}$ months the maximum attainable temperature dropped by approx. 80°C but it is difficult to say whether this was due to changes in ε and A or changes in the reactor flux distribution. High temperature thermocouple performance was poor with open-circuit failures occuring at start-up, 4 days, 14 days and 4 weeks from start-up. No failures occurred on the C/A T/C's which were operating at temperatures from 600 to 1000°C.

The third rig is at present operating at a specimen temperature of 1330°C with pure Neon. After 2 months operation 2 Pt-Rh/Pt T/C's and all the C/A T/C's are still operating.

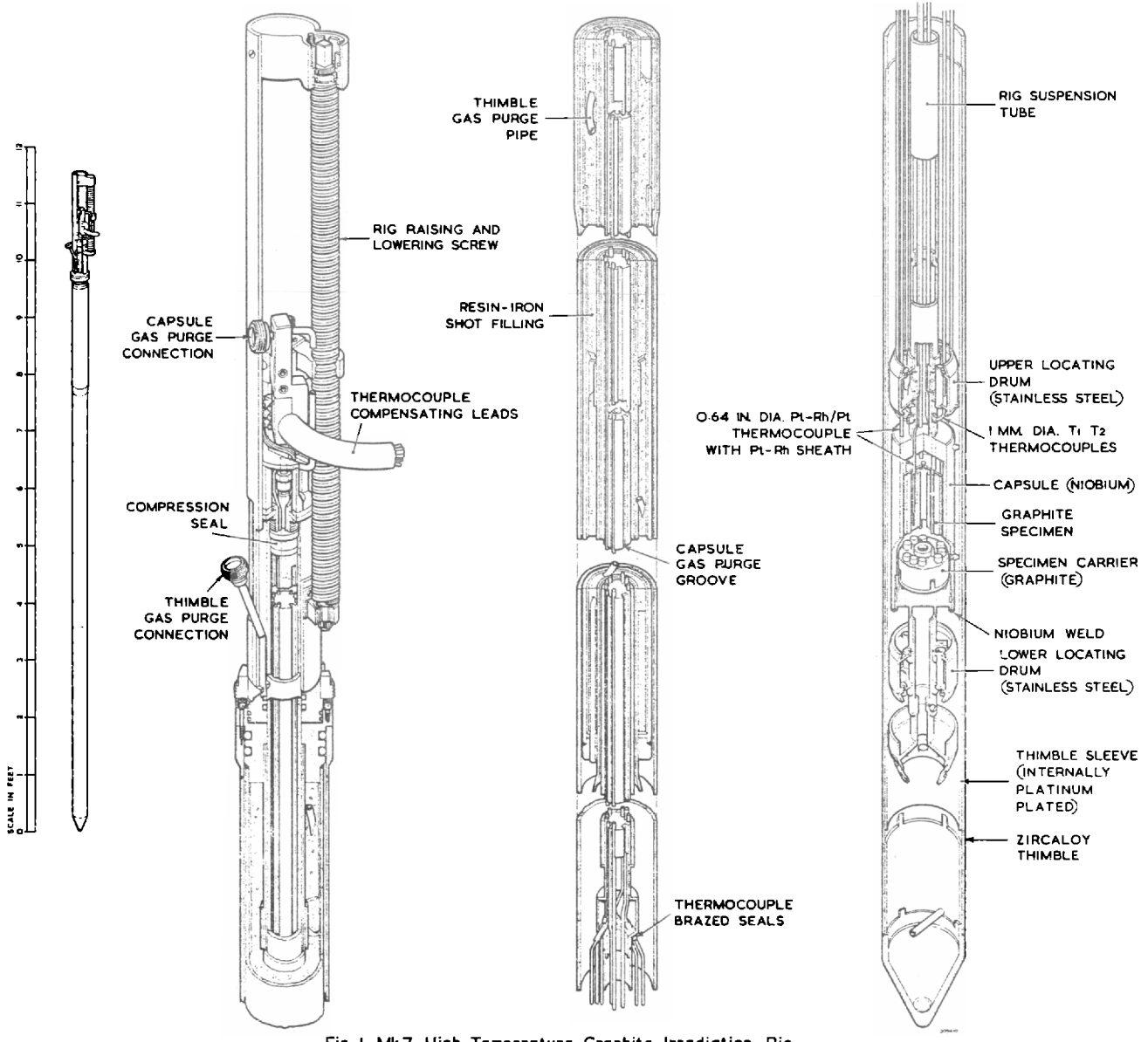

Fig. 1. Mk.7 High Temperature Graphite Irradiation Rig.

APPENDIX

Effect of Thermal Radiation between Non-Grey Bodies

1. Conventional equation for Concentric Cylindrical Grey Bodies

Neglecting the difference in radii since the annulus is narrow:-

$$Q = \frac{A_1 \varepsilon_1 \sigma (T_1^4 - T_2^4)}{1 + \varepsilon_1 \left(\frac{1}{\varepsilon_2} - 1\right)} \quad \ldots\ldots(1);$$

1 denotes minor (hot) cylinder
2 denotes outer (cold) cylinder
ε = emissivity

Typical values applicable to high temperature graphite rig:-

$$\varepsilon_1 = \cdot 19$$
$$\varepsilon_2 = \cdot 05$$

2. "Exact" Equation for Concentric Cylindrical Non-Grey Bodies

In almost all cases of radiant heat exchange the absorbtivity (a) and emissivity (ε) of each surface are assumed equal since it is known from application of Kirchoff's Law that at thermal equilibrium $a = \varepsilon$.

However if the surfaces are at greatly differing temperatures it is known (ref. 1) that $a = \varepsilon$. An empirical rule for metals is

$$a_{12} = C_1 \times \sqrt{T_1 T_2} \ldots\ldots (2)$$ (absorbtivity of surface 1 receiving (radiation from surface 2.

and $$a_{11} = \varepsilon_1 = C_1 T_1 \ldots\ldots (3)$$

Accounting for the variation of a, we have

$$Q = \frac{A_1 \varepsilon_1 \sigma T_1^4}{1 + a_{12}\left(\frac{1}{a_{21}} - 1\right)} - \frac{A_2 \varepsilon_2 \sigma T_2^4}{1 + a_{21}\left(\frac{1}{a_{12}} - 1\right)}$$

ref. 1 McAdams, Ed. 3 p. 63.

3. Possible Values of a_{12}, a_{21}

In the absence of any directly related measurements of a for the materials and conditions assumed in this case possible values of a_{12} and a_{21} have been derived using equations 2 and 3.

Putting $T_1 = 1500°C \equiv 1773°K$

$T_2 = 50°C \equiv 323°K$

$\varepsilon_1 = \cdot 19 = C_1 \times 1773$

$\therefore \quad a_{12} = C_1 \sqrt{T_1 T_2} = \frac{\cdot 19}{1773} \sqrt{1773 \times 323} = \cdot 081$

$\varepsilon_2 = \cdot 05 = C_2 \times 323$

$\therefore \quad a_{21} = C_2 \sqrt{T_1 T_2} = \frac{\cdot 05}{323} \sqrt{1773 \times 323} = \cdot 117$

4. Comparison of Results from Conventional and "Exact" Equation

With the above values of T_1 and T_2 the terms involving T_2^4 are negligible. Thus we have from equations (1) and (4)

$$\frac{Q_{R\ exact}}{Q_{R\ conv.}} = \frac{1 + \varepsilon_1 \left(\frac{1}{\varepsilon_2} - 1\right)}{1 + a_{12} \left(\frac{1}{a_{21}} - 1\right)}$$

and with the values of ε_1, ε_2 above

$$\frac{Q_{R\ exact}}{Q_{R\ conv.}} = 2 \cdot 86$$

Similar calculations for $\varepsilon_2 = \cdot 1$ and $\varepsilon_2 = \cdot 2$ give

$$\varepsilon_2 = \cdot 1, \quad \frac{Q_{R\ exact}}{Q_{R\ conv.}} = 2 \cdot 14$$

$$\varepsilon_2 = \cdot 2, \quad \frac{Q_{R\ exact}}{Q_{R\ conv.}} = 1 \cdot 61$$

5. Practical Application of Exact Equation

The "exact" form of the radiant heat transfer equation was used in the design of the high temperature graphite irradiation rig. Several rigs to this design have been operated in DIDO reactor and some conclusions may be drawn from the capsule surface temperature attained. This has varied from 1330°C to 1250°C.

Use of equation 4 and suitable calculation of the conduction through the gas in the annulus resulted in a predicted temperature of 1300°C.

Use of equation 1 instead of 4 would have resulted in a predicted temperature of 1600°C.

Owing to the variation and the degree of uncertainty of the nuclear heating in the capsule and the purity of neon in the annulus, the effect of irradiation on the gas conductivity, and the uncertainty of emissivity values in general it is not possible to draw any hard and fast conclusions regarding the validity of using equations 2 and 4. A controlled laboratory test would have to be performed to assess this properly. However, the practical results with the DIDO rig show quite clearly that it is not sufficient to use the conventional equation and assume emissivity values commonly quoted if the temperatures of the two surfaces differ greatly.

HAPO GRAPHITE IRRADIATION CAPSULES

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General Electric Company
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INTRODUCTION

Capsules containing reactor-grade and experimental graphites are being irradiated in the ETR, GETR, MTR, and Hanford reactors. These capsules provide two basic types of information about graphite behavior under irradiation. First, information on graphite damage under large neutron exposures equivalent to several years of operation of a graphite-moderated reactor is obtained in a relatively short time. Second, basic information to aid in the explanation of the mechanism of graphite damage under irradiation is obtained. Measurements ofmany properties are made on each sample in the capsules. These include length, weight, crystallite parameters, electrical resistivity, sonic modulus, thermal conductivity, thermal expansion, etc.

The four basic criteria used in capsule design are:achievement of desired operating conditions, reliability, ease of construction, and ease of disassembly. Two primary irradiation conditions are carefully monitored. These are temperature and neutron flux. Since graphite damage is very sensitive to irradiation temperature, efforts are made to keep the variation below $\pm$ 10° C at temperatures from 0 to 300° C and

* Work performed for the Atomic Energy Commission under Contract AT (45-1)-1350.

below $\pm$ 25° C above 300° C. Neutron-flux determinations are required so that data can be correlated between various reactors. Consequently detailed studies on neutron spectra are being made for all our graphite irradiation facilities.

This report is intended to serve only as a summary of the broad field of graphite irradiation capsules and is written in four parts to serve as a basis for discussion at this meeting. If detailed information is desired the persons listed at the end of each part should be contacted.

Part I summarizes the various capsule designs used for the irradiation of graphite samples. These designs range from highly instrumented capsules in which irradiation temperatures are precisely controlled to capsules in which temperatures are only calculated. In between these extremes are capsules in which the sample temperatures are monitored at several points. All capsule types include monitor materials for measurement of neutron fluxes.

Part II describes several capsule components and materials and the suitability of them for use in irradiation capsules.

Part III outlines the heat-transfer calculations used in the design of the various capsules. The computer program used has proved to be a valuable aid in capsule design and modification.

Part IV presents the methods being used for the calculation of neutron spectra. These spectra are needed to estimate effective

activation rates for the monitor materials placed in the irradiation capsules and to correlate ratiation data obtained from various irradiation facilities.

PART I. CAPSULE DESIGNS

TEMPERATURE-MONITORED CAPSULE

A typical temperature-monitored capsule for the irradiation of various types of reactor-grade and experimental graphites is shown in Figure 1. In this capsule four quarter-round samples are put together to form a right circular cylinder. They are slid into graphite sample holders, which screw together through the molybdenum centering disks. These disks are fastened onto the aluminum cooling rings. The assembly is slid into the outer aluminum shell, evacuated, filled with helium, and welded shut.

Instrumentation in the capsule consists of five flux-monitors in each cooling ring and at least one thermocouple in each sample position. This capsule provides a relatively simple method for irradiating various graphites. Assembly and disassembly are quite straightforward. Irradiation temperatures from 300 to 1200° C have been achieved with this basic design. Variations on the design include additional graphite cooling disks at the sample midposition, use of six 60 degree samples rather than four quarter-rounds, etc. Sixteen capsules of this basic type have been irradiated successfully to date, and the two others have been partially successful.

GRAPHITE CREEP CAPSULE

An irradiation program for studying the "creep" or deformation of graphite under tensile load is currently underway in the ETR. Sample

temperatures of 475 to 535° C and tensile loads of 600 to 1200 psi are being studied. The heat source is the gamma heating of the capsule components. A schematic of the capsule is shown in Figure 2. One four-inch long graphite sample is positioned between the stainless steel sample grips. The lower grip is held in a fixed position by the heavy aluminum capsule wall. The upper grip is attached by means of a three-foot long stainless steel draw-bar assembly to the bellows. The aluminum heat-transfer block which surrounds the sample is designed with appropriate clearances to provide a flat temperature profile along the sample.

The centerline of the sample is positioned just below the vertical centerline of the reactor so that the entire four-inch sample length is in a region of fairly uniform neutron flux throughout a reactor cycle. A helium atmosphere slightly greater than reactor water pressure is used in the capsule. A differential helium pressure greater than the capsule pressure is controlled on the bellows to provide the desired load on the sample. Sample temperatures are measured by means of seven thermocouples. An unstressed graphite "reference" sample is located in the center hole in the stressed sample to provide a direct measure of the effect of stress.

To date four of these creep capsules have run successfully and another partially successfully. Temperatures from one end of the sample to the other have varied by less than 10° C. Pressure control has been held constant at a given setting plus or minus 2 psi.

TEMPERATURE-CONTROLLED CAPSULE

A capsule for the irradiation of various graphites under controlled temperature conditions is shown in Figure 3. Three or more wafer samples are put into a sample cup. An assembly made up of the sample cup, one or more pyrolytic graphite insulators, and a platinum heater is held onto the aluminum base plate by a molybdenum spring. The entire assembly is slid into the aluminum shell.

Instrumentation consists of a platinum heater, a thermocouple, and eight flux-monitor wires at each sample position. To make use of all available space in the capsule additional sample sets having only thermocouples are located between the controlled-temperature positions. Sample temperatures have been controlled to plus or minus 10° C at temperatures varying from 600 to 1200° C. One capsule of this type was completely sucessful and another was partially successful.

BORONATED- GRAPHITE CAPSULES

A rather unique capsule for the irradiation of boronated-graphite samples is shown in Figure 4. Due to the high heating rate from the boron n,α reaction a graphite sample carrier is used to smooth out the temperatures from sample to sample and to provide a heat-transfer path to the aluminum shell. Samples containing up to 8 weight per cent boron have been irradiated successfully in capsules of this type. A helium atmosphere is used in the capsule. Instrumentation consists of several thermocouples and flux monitors.

NON-INSTRUMENTED CAPSULES

A capsule for the irradiation proof-testing of various experimental and prototype graphites is shown in Figure 5. This capsule provides a relatively inexpensive means for direct comparisons of experimental graphites with a known "reference" graphite. Instrumentation consists of flux monitors. Capsule atmospheres are helium or argon. Various sample configurations are used, from the quarter-rounds shown to hollowed-out graphite boats filled with small cylindrical or parallelepiped samples. Several hundred of these capsules have been run successfully.

FURTHER INFORMATION

Please contact:

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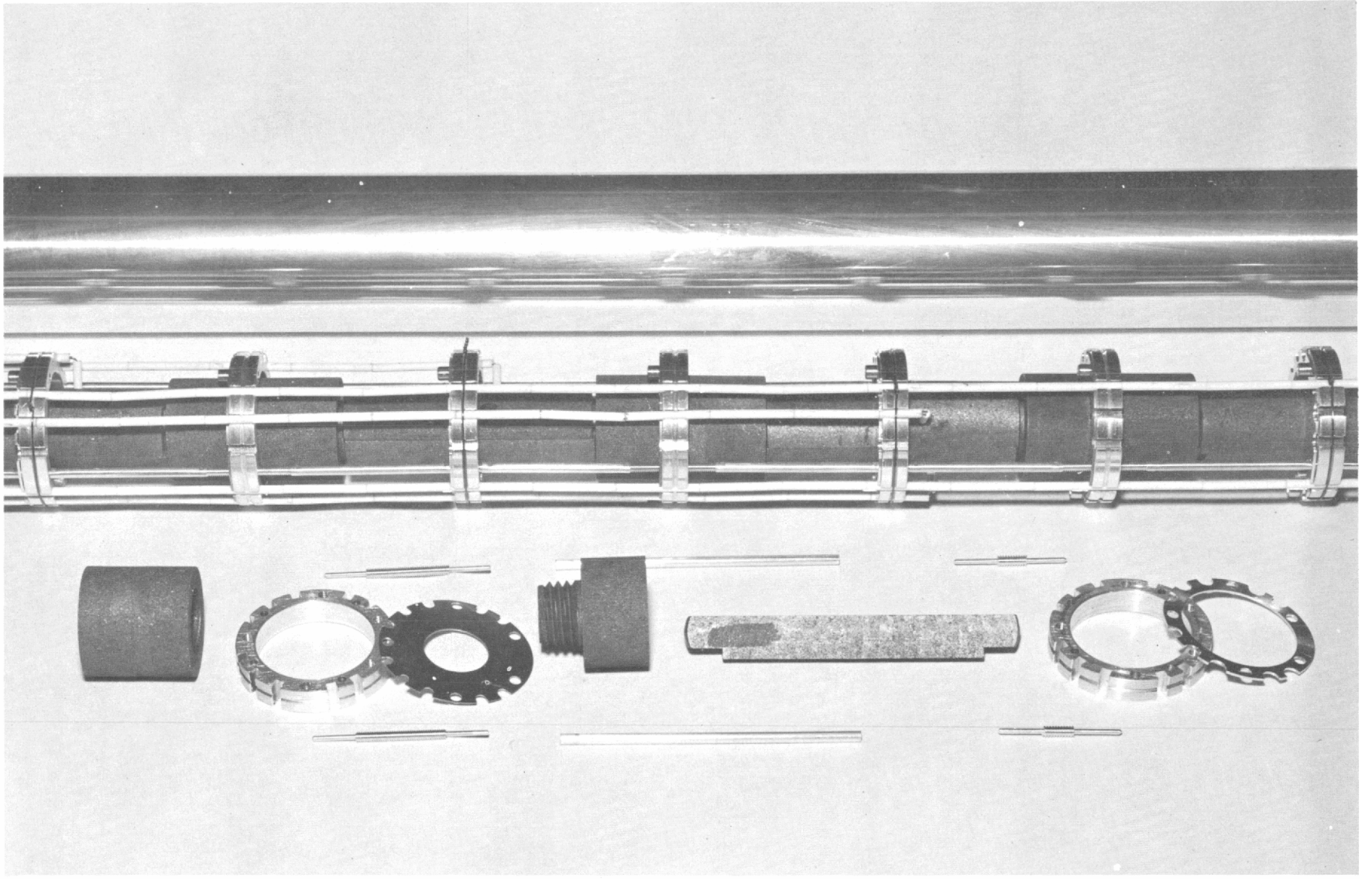
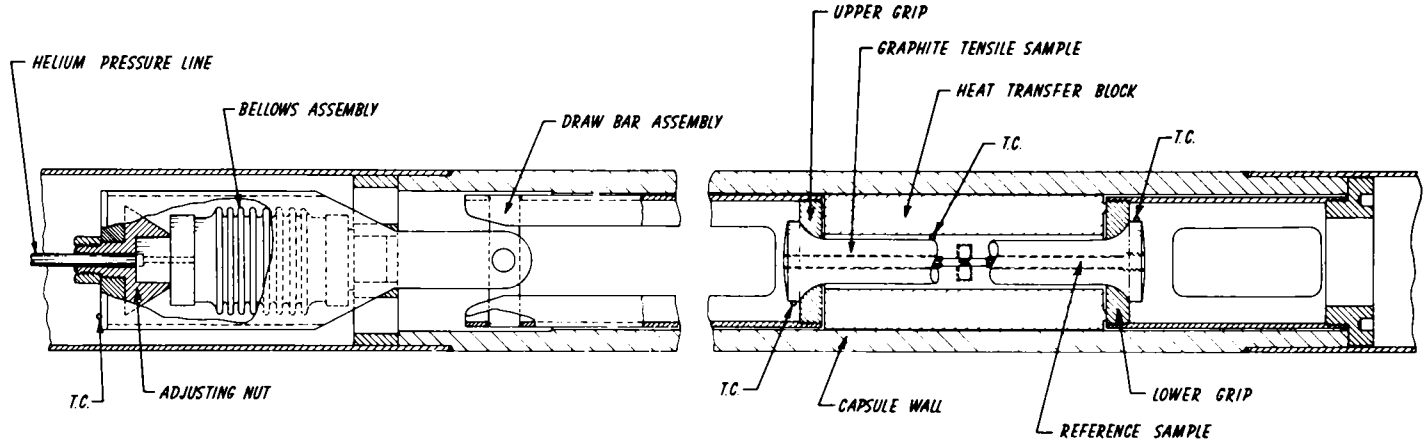


FIGURE 1. TEMPERATURE-MONITORED CAPSULE



IN-REACTOR TENSILE CREEP CAPSULE FOR GRAPHITE

FIGURE 2

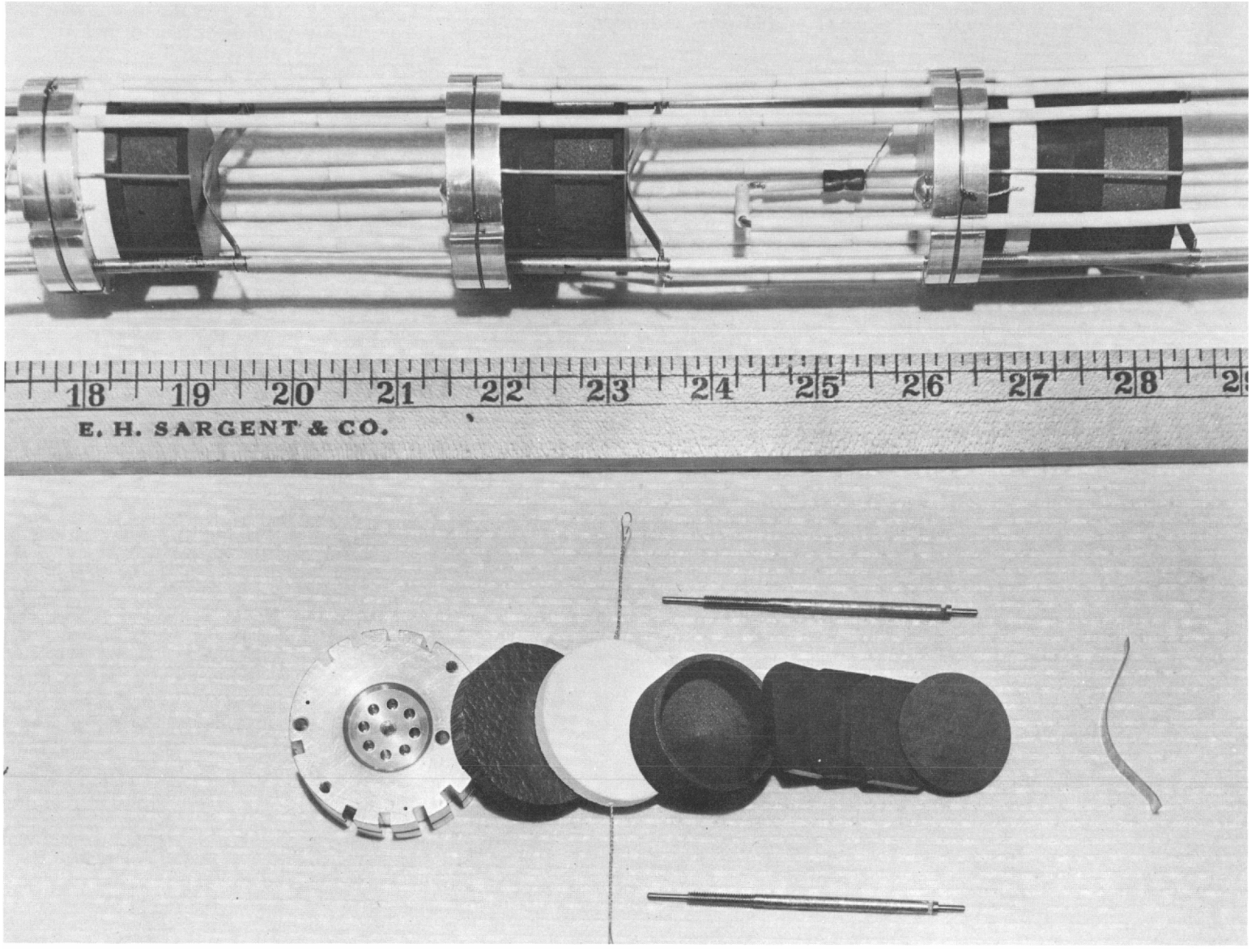


FIGURE 3. TEMPERATURE-CONTROLLED CAPSULE

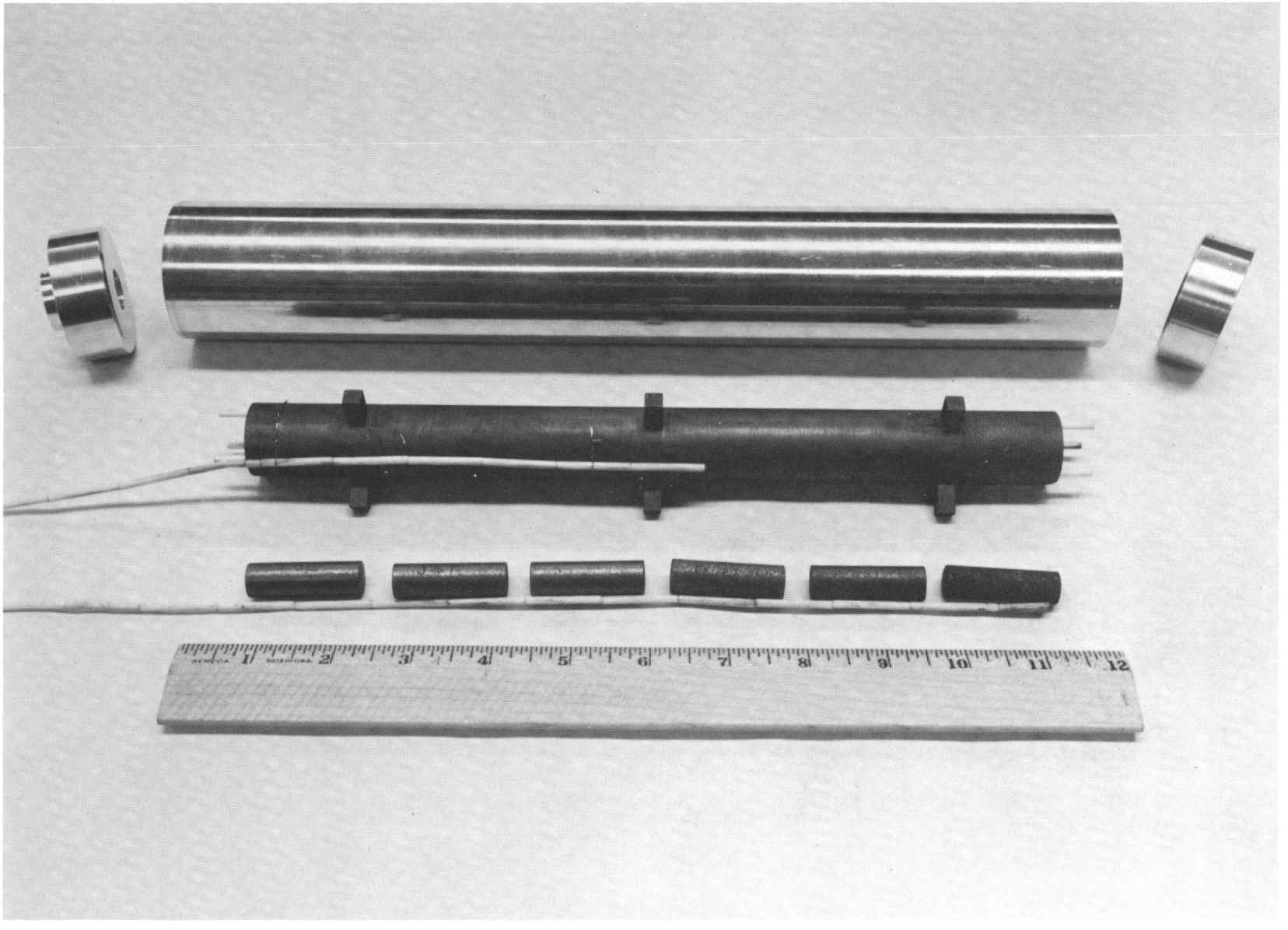


FIGURE 4. BORONATED-GRAPHITE CAPSULE

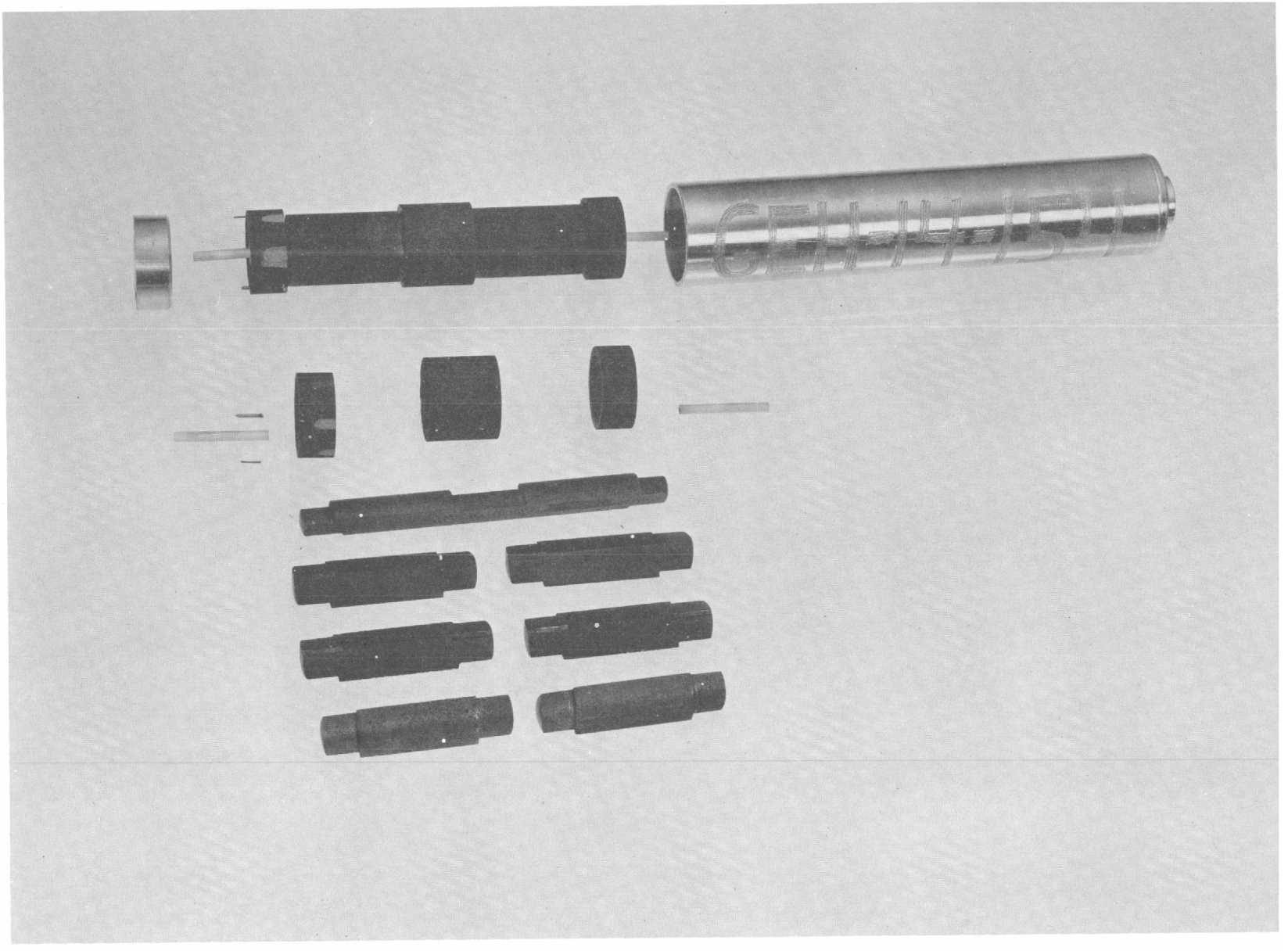


FIGURE 5. NOW-INSTRUMENTED CAPSULE

PART II. CAPSULE COMPONENTS

THERMOCOUPLES

Thermocouples are made from bare wire and ceramic insulators by our technicians in our own laboratories.

Wire - Solid 24 gage Geminol-n and Geminol-p

Compositions:

Geminol-n	Geminol-p
96.95% Ni	77.9% Ni
3% Si	20 % Cr
0.05% C	1 % Nb
	1 % Si
	0.1% C

Insulators - Two-hole vitreous alumina, 1-in. long.

Some 250 of these thermocouples have been used in capsules irradiated in the ETR, MTR, and GETR. Of these all but 11 have operated satisfactorily. Of the 11 that failed, 5 failed by a break at the right angle bend where the thermocouple enters the graphite sample, 1 failed by a bad junction at the bead, and the other 5 failed at some point along the length of the thermocouple. Thermocouples used to monitor irradiation temperatures of 800° C show a decrease in indicated output of approximately 50° C when calibrated ex-reactor after an exposure of approximately 3×10^{21} nvt, E > 0.18 Mev.

Extension Thermocouple Leads. Extension leads are made of duplex Geminol-n, Geminol-p 24 gage solid wire insulated with high temperature polyvinyl.

To date none of the extension leads has failed during operation.

Heaters

Heaters are also made at Hanford. They are flat wafers of platinum wire cast into alundum cement which is fired at 1300° C for 1 hr. Some 56 of these heaters have been used with varying degrees of success. The main cause of failure has been the reaction of the alundum with the graphite sample cups in the capsule with subsequent shorting of the heater windings.

Connectors

Hermetic Connector. A hermetic connector is used to provide a positive seal for the capsule and a junction for the thermocouples and extension leads. They are leak tight to at least 100 psig helium. In service they are located some 10 ft. above the test-reactor core. Both thermocouple and heater leads are exited from the capsule through these connectors. Twenty-four of these connectors have been used with 100 per cent reliability.

RN Connector. The RN connector is a device for making a junction between two pieces of tubing in a reactor vessel. It consists of a six-inch long tube, the I. D. of which is slightly larger than the O. D. of the tubing being joined, and a bored-out tubing union. One piece of tubing is slid three inches into the connector which is then welded to the tubing. The other piece of tubing is slid through the

bored-out union and butted up against the first piece of tubing. The union is tightened to complete the junction. The six-inch tube provides a stiffener so that the union does not vibrate loose during reactor operation. The device has been used successfully on nine instrumented capsules in the ETR.

Flexible Tubing

Corrugated stainless steel, one wire braid corrugated flexible metal tubing is used for leadout tubing for our GETR capsules. Thermocouple leads are carried from two feet inside the pressure vessel to the recorder in this tubing.

Bellows

Bellows are two-ply 321 stainless steel, 0.006-inch thick with welded end fittings. The bellows have performed completely satisfactorily in five irradiation capsules operating at differential pressures of 200 psi.

Pressure Controller

The pressure controllers used to control both the capsule pressure and the bellows pressure in the graphite creep capsule are automatically operated, remotely-controlled, pneumatic pressure regulators. They are rated for 0-3000 psi fluid. They have operated satisfactorily for a year of nearly continuous service.

Materials

Aluminum. One problem that comes up from time to time is porosity in wrought aluminum bars. Recently a 20-foot long bar was found to have

gas pockets up to 1/8-inch diameter and 1/8-inch long at various locations along its length.

Molybdenum. Molybdenum has proven to be a useful construction material for capsule parts. It has been used to good advantage as a means of limiting heat transfer, supporting graphite sample holders, and as a spring for holding wafer assemblies together.

Pyrolytic Graphite. Pyrolytic graphite serves both as an excellent insulator and an excellent conductor due to its high degree of anisotropy. This makes it a very useful material for capsule construction, particularly at high temperatures. Along the layer planes its conductivity is about the same as tungsten and against the planes it is a better insulator than zirconia. It also has high tensile strength which increases with increasing temperature.

FURTHER INFORMATION

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PART III. HEAT TRANSFER CALCULATIONS

Use of an IBM 7090 computer has greatly facilitated the design of graphite irradiation capsules at Hanford. The temperatures of all capsule parts are now calculated in considerable detail by means of steady-state heat-transfer calculations. The basic computer program used is STHTP - A Steady-State Heat-Transfer Program for the IBM 7090 Computer, HW-73668. The program has the capability of handling three-dimensional geometries involving internal heat generation, constant or temperature dependent thermal conductivity, constant contact or film coefficients, and heat transfer by conduction and radiation. The program has the capacity to handle the following:

500 nodes, or regions where heat is generated or transferred.
20 sink, or boundary temperatures.
30 material-property tables.
20 contact or film coefficients.

Any number of heat-generation rates can be used simply by changing the material-property tables and adding a new case card for a new set of calculations.

The technique used by the program consists of an iterative solution of the heat transfer through all the nodes in the problem. An acceptable solution is obtained when the error in heat flux across any node boundary is less than an arbitrary value set up for each capsule. Output from the program consists of:

Temperature at each node.
Conductance through each of the eight sides of each node.
Temperature of andheat transferred to each sink.
Total heat generation in the capsule.

Calculation Details

Nomenclature

A heat-transfer area, or area of node face,

U thermal conductance,

T temperature,

Q heat-generation rate,

ϵ_N heat-balance error for node N,

ϵ_T absolute value of heat-transfer error,

α acceleration factor.

Subscripts

N node number,

K number of node adjacent to node N,

J number of node side.

The capsule is divided into as many nodes as are required to provide the amount of detailed temperature distribution desired. The sink or boundary conditions are established and the material-property tables made up.

The temperature of any node, N, can be found in terms of the node heat-generation rate, Q_N, and the adjacent node temperature, T_K, from the following equation:

$$T_N = \frac{Q_N + \sum_{J=1}^{8} A_{N,J}\, U_{N,J}\, (T_K)}{\sum A_{N,J}\, U_{N,J}}$$

The computer solves a set of N equations of this type by an iterative process. After a prescribed number of iterations the accuracy of the calculated temperatures is checked by making heat balances on each node. This heat-balance error is given by the equation:

$$\varepsilon_N = Q_N + \sum_{J=1}^{8} U_{N,J}\, A_{N,J}\, (T_K - T_N)$$

Then $\varepsilon_T = \sum_{N=1}^{N} \varepsilon_N$ = the absolute value of ε_N:

If ε_T is less than the prescribed value set up for the capsule, the temperatures are considered accurate and the iteration procedure is terminated.

To speed up the iterative process the program uses an acceleration factor, α. The calculated node temperature, T_N, is compared with the calculated temperature from the previous iteration, T'_N or

$$\Delta T_N = T_N - T'_N$$

A corrected node temperature is then calculated from $T_N = T'_N + \alpha \Delta T_N$. An extrapolation procedure is also used to eliminate some iterations

and bring the calculated temperature asymptotically to the correct temperature. Ratios of the ΔT_N are taken for the four previous iterations and used for the extrapolation. A history of the calculated node temperature is shown in Figure 1.

Use

The heat transfer program has been used at Hanford to calculate temperatures in new capsules, provide a basis for design changes in capsules, test new capsule materials, etc. The basic program has been modified to readily handle the radial symmetry of our graphite irradiation capsules. The areas, A, and distances, d, needed for the computer input for a typical node in radial symmetry are shown in Figure 2.

A typical nodal structure for a temperature-monitored capsule is shown in Figure 3. It is recommended that anyone using such a computer code initially establish the maximum number of nodes that might possibly be of interest even in future calculations. It is much easier to eliminate nodes in later capsules than to completely revise the nodal structure if more detailed temperatures are required.

The results of a typical temperature-monitored capsule calculation are shown in Figure 4. The three lines for each gamma heating rate correspond to concentric graphite node temperatures. The temperature distribution agrees within plus or minus 25° C of the temperatures as measured by thermocouples in irradiation capsules.

FURTHER INFORMATION

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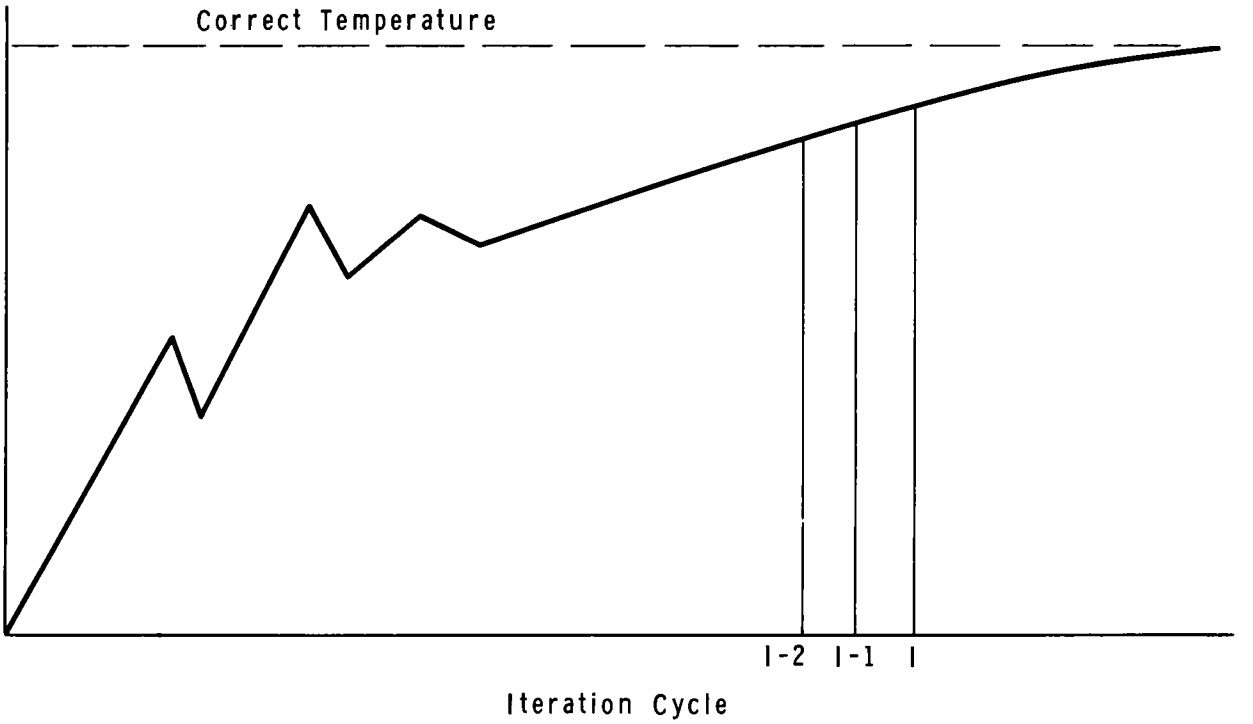

FIGURE 1. CALCULATED NODE TEMPERATURE

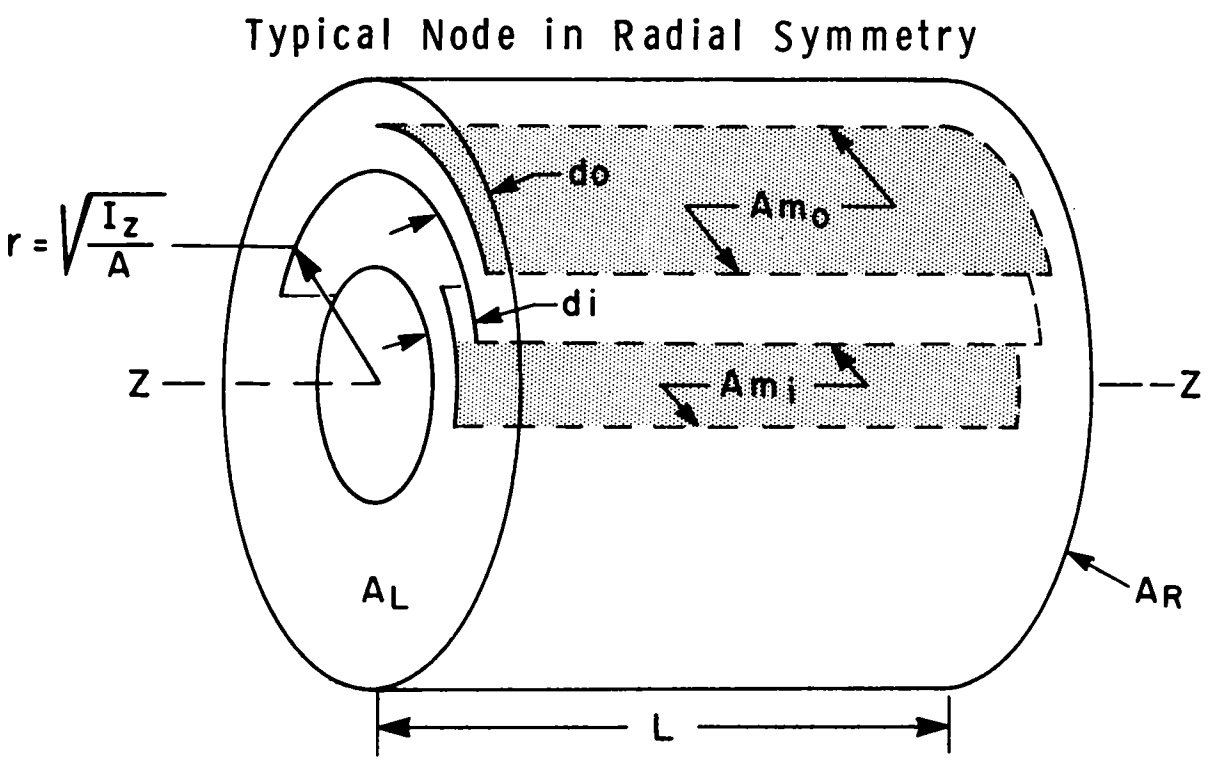

FIGURE 2

HEAT TRANSFER NODE DIAGRAM OF A TYPICAL CAPSULE

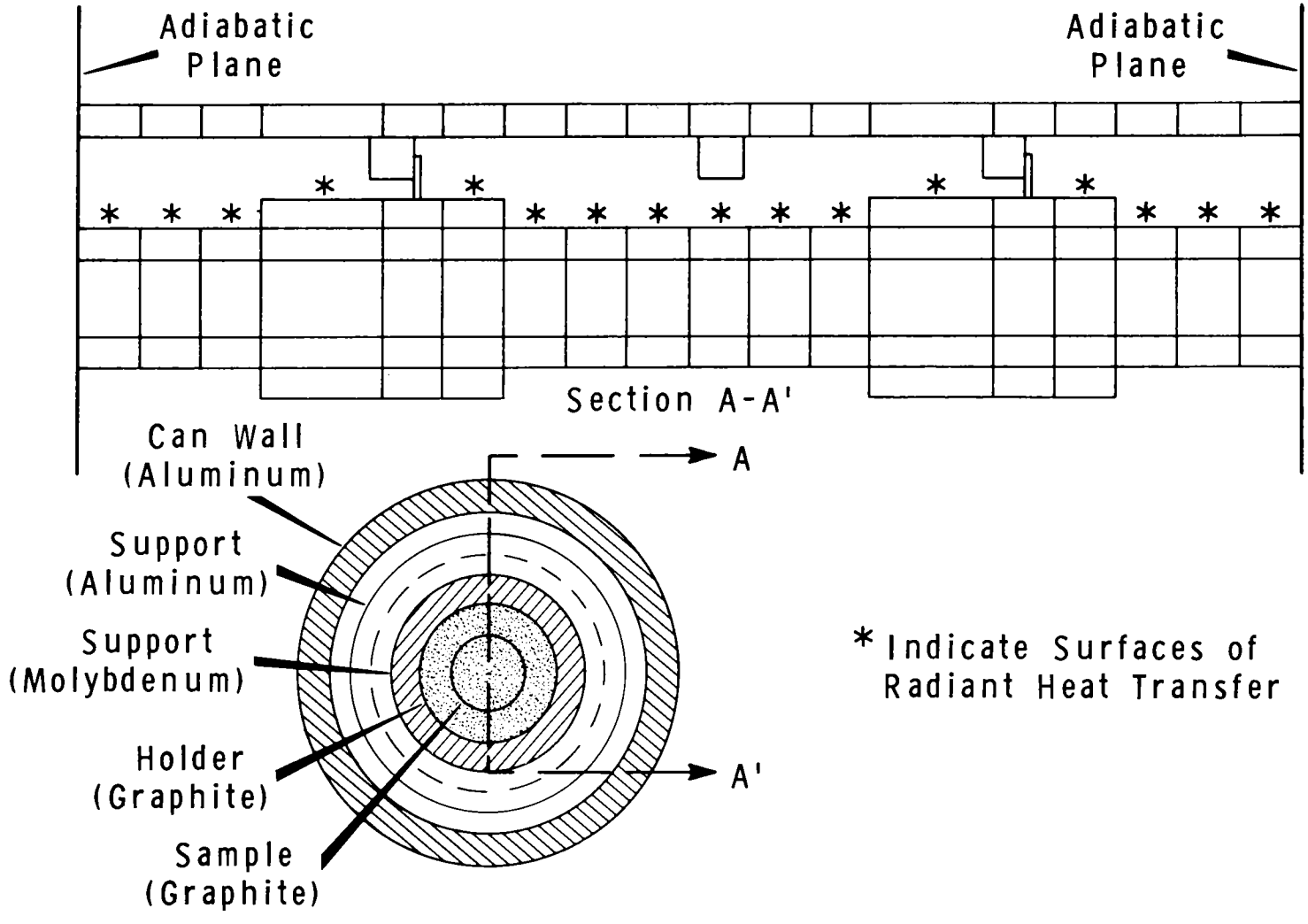

FIGURE 5

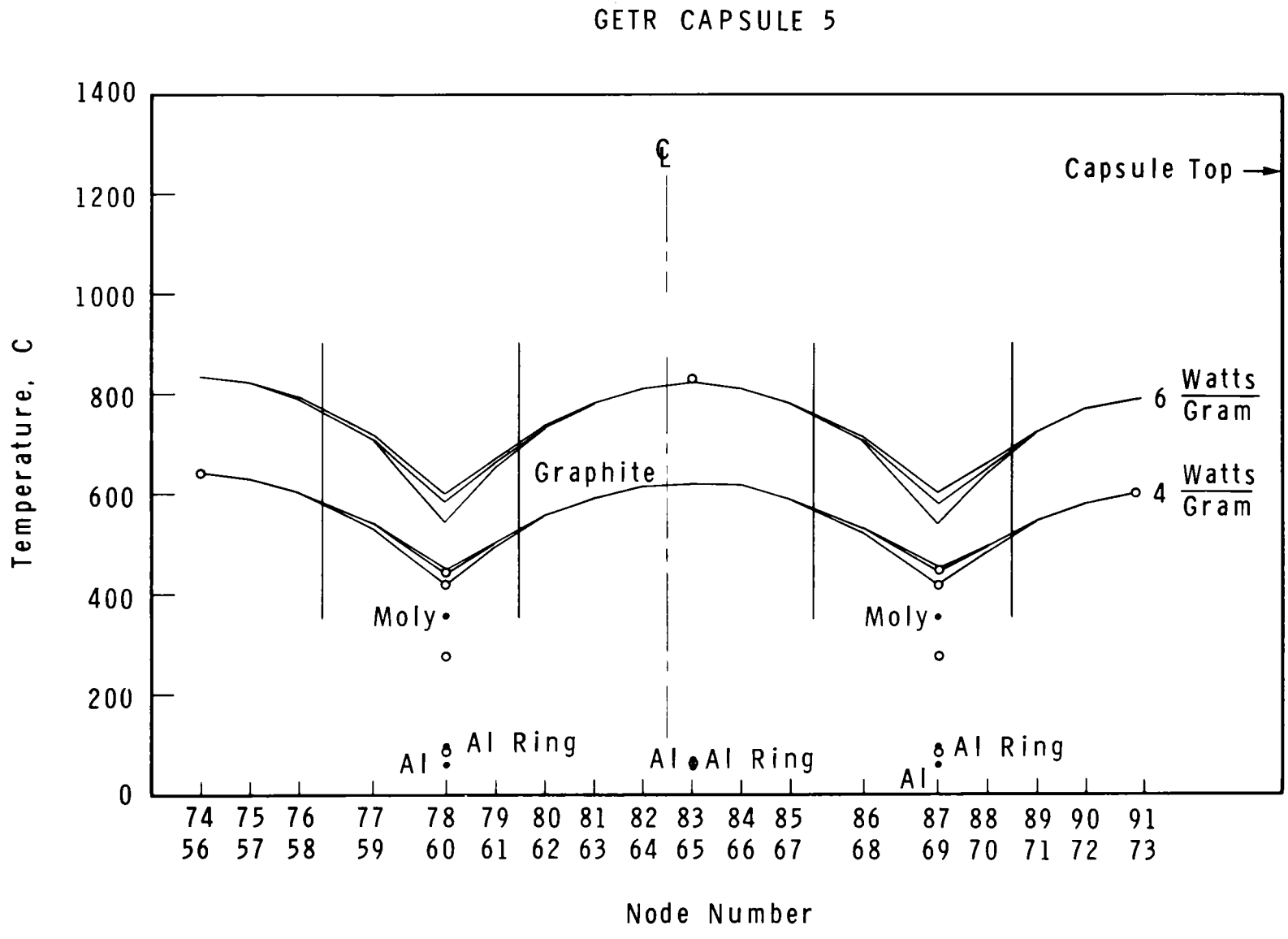
GETR CAPSULE 5
Temperature, C
1400
1200
1000
800
600
400
200
0
℄
Capsule Top
6 Watts/Gram
4 Watts/Gram
Graphite
Moly
Moly
Al
Al Ring
Al
Al Ring
Al
Al Ring
74 75 76 77 78 79 80 81 82 83 84 85 86 87 88 89 90 91
56 57 58 59 60 61 62 63 64 65 66 67 68 69 70 71 72 73
Node Number

FIGURE 4

PART IV. NEUTRON-SPECTRA CALCULATIONS

Studies are being made on reactor neutron spectra to aid in better understanding the radiation damage to graphite as measured by activation monitors placed in the irradiation capsules. These spectra are required to estimate effective activation rates for the monitors in various reactor positions, to estimate the differences in flux between the monitors and the actual samples being irradiated, and to correlate radiation data obtained in different irradiation facilities in various reactors. Calculations of spectra are being made for graphite capsule positions in the GETR, ETR, and Hanford reactors.

Four different reactor-physics computer codes have been used to calculate fast-neutron spectra: GNU-II, HFN, Program S, and 2DXY. Significant differences in spectra are found with the different codes. Consequently two multi-group, multi-region, transport-theory codes, Program S and 2DXY, are being used due to their greater accuracy and wider applicability. In these studies, the Program S code, a one-dimensional code, is set up with 33 energy groups of which 16 are greater than 0.183 Mev and the 2DXY code, a two-dimensional code, is set up with 18 energy groups 16 of which are greater than 0.183 Mev. Input material cross-sections are calculated using the GAM program and are weighted according to the P3 spectrum calculated for each reactor.

The results of the computations give the neutron flux as a function of space and energy. The computations can be readily modified for changes in materials and geometry. The spatial arrary over the cross-sectional area of the reactor can be varied. In the case of the GETR the entire core cross-section is calculated using one-inch by one-inch increments. Calculations are in progress to determine the effect of large changes in materials and/or geometry on the spectra in our graphite capsule positions. Such things as replacement of fuel with poison, replacing graphite with cobalt, etc. are being calculated. To date these types of changes have not had significant effect on the spectra; however, flux intensity has been altered to some extent.

The results of the spectra calculations for various locations in the K-Reactor lattice at Hanford are shown in Figure 1. The lattice is a square symmetric lattice with an annular fuel element, water coolant, and graphite moderator. The moderator boundary is the innder edge of the graphite channel and the cell boundary is the midpoint in the graphite between tube channels. The flux distribution, $\phi(u)$, is plotted as a function of lethargy, u, which is defined as $u = -\ln E/E_o$ where E_o is 10 Mev. It should be noted that the fission spectrum is not a good approximation of the spectra at any position including the center of the fuel.

The marked difference in fast-neutron spectra between a graphite-moderated reactor, C-Reactor at Hanford, and a testing reactor, the ETR, is shown in Figure 2. The C Reactor spectrum shown is for the cell boundary in the graphite moderator and the ETR spectrum is for the center of the G-7 Hanford loop. The two curves have been normalized so that there are equal areas under the curve to lethargy 4.

The large differences in flux intensity which may exist in even the same reactor facility are shown in Figure 3. These spectra are all for the G-7 Hanford loop in the ETR. The spectra are for one-inch by one-inch area increments in the south third of the G-7 reactor position. The bottom curve, G-7-SW, corresponds to the center of the loop with stainless steel specimens and a stainless steel grid work, the middle curve, G&-SE, to the stainless steel pressure tube and the aluminum shroud tube. Thus two positions in the loop only two inches apart have flux intensities differing by a factor of three over the range of neutron energies which produce approximately 90 per cent of the damage to structural materials.

The spectra obtained in these calculations are used to compute effective cross-sections for such fast-neutron monitors as Nickel-58 and Iron-54. The effective cross section, $\bar{\sigma}$, is given by:

$$\bar{\sigma} = \frac{\int_{\infty}^{0} \phi(u)\,\sigma(u)\,du}{\int_{u_1}^{0} \phi(u)\,du} = \frac{\int_{0}^{10} \phi(E)\,\sigma(E)\,dE}{\int_{E_1}^{10} \phi(E)\,dE}$$

where $\phi(u)$ is the flux distribution as a function of lethargy and $\sigma(u)$ is the lethargy-dependent activation cross section. Typical differential curves used to obtain $\bar{\sigma}$ for nickel are shown in Figure 4. Since the curves are plotted as a function of lethargy the flux in any lethargy interval is proportional to the area under the curve.

The effective cross section, for the energy range chosen to correlate observed property changes, is used to calculate neutron exposures from the monitor activities. For those monitors with high threshold energies, which include most of the commonly used monitors such as sulfur, aluminum, nickel, and iron, the effective cross sections vary considerably throughout the reactor lattice due to changes in the spectra below the threshold energy. Consequently effective cross-section calculations for the energy range of interest must be made for each monitor since a large number of the neutrons damaging samples may have energies well below the monitor energy threshold. Effective cross sections are especially necessary if comparisons of damage in different spectra in different reactors are being made. Although the monitor may be activated by neutrons having energies in excess of 4 Mev it may be necessary to calculate neutron exposures to samples to neutron energies as low as 0.05 Mev in order to correlate observed property changes.

FURTHER INFORMATION

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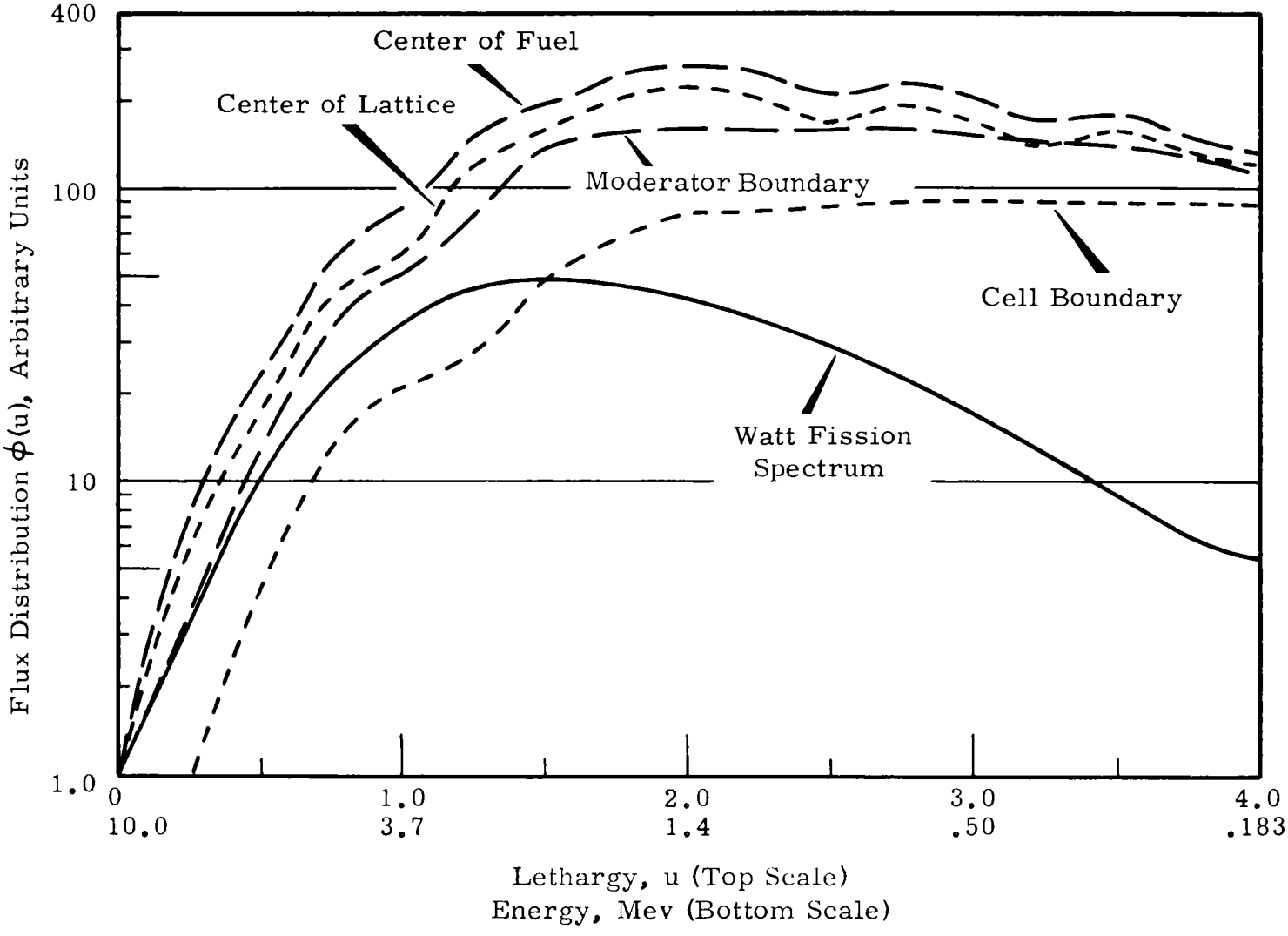


FIGURE 1. FLUX SPECTRA-GRAPHITE MODERATED REACTOR

FLUX SPECTRA IN ETR TEST FACILITY AND A GRAPHITE MODERATED REACTOR

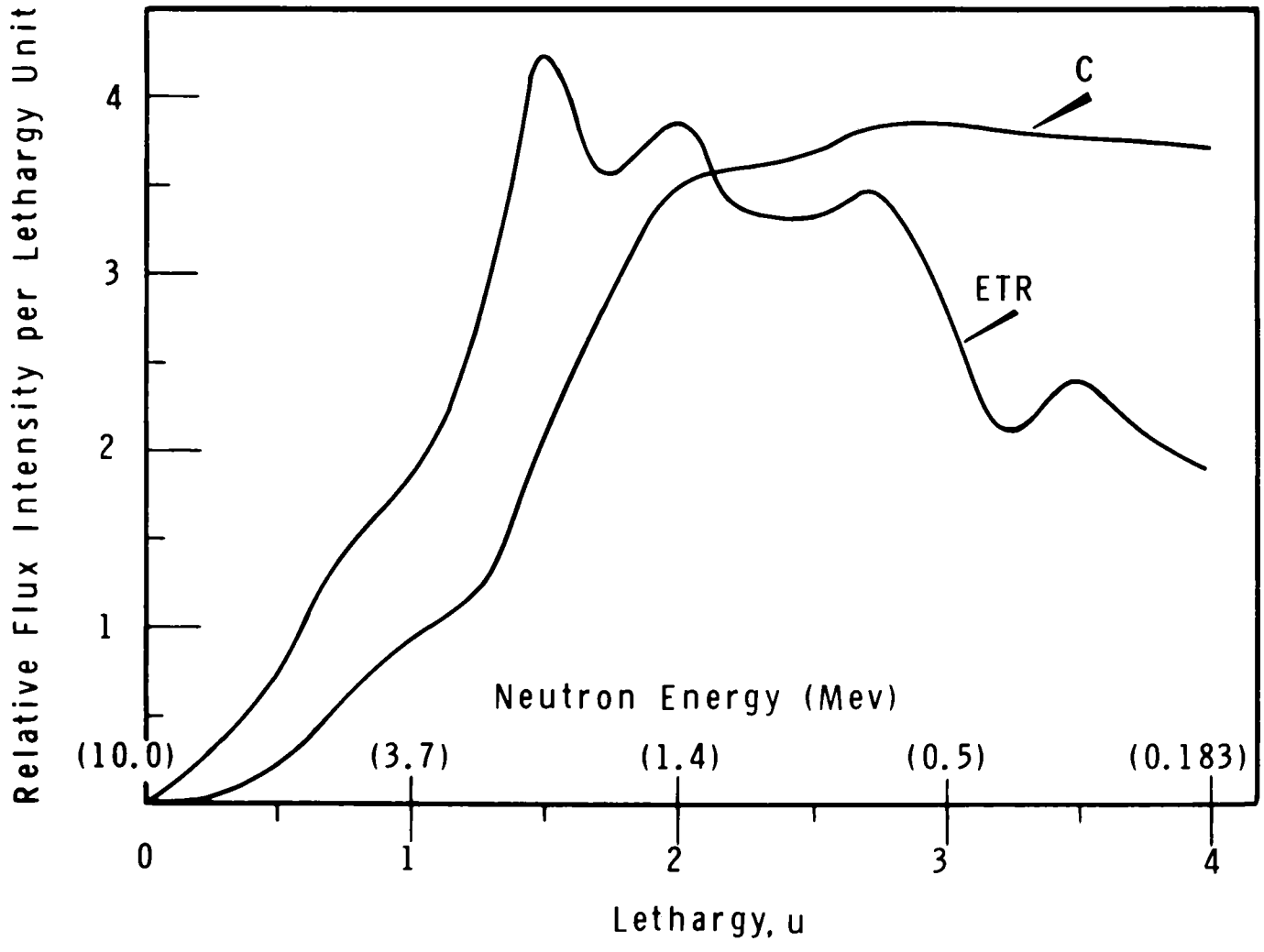


FIGURE 2

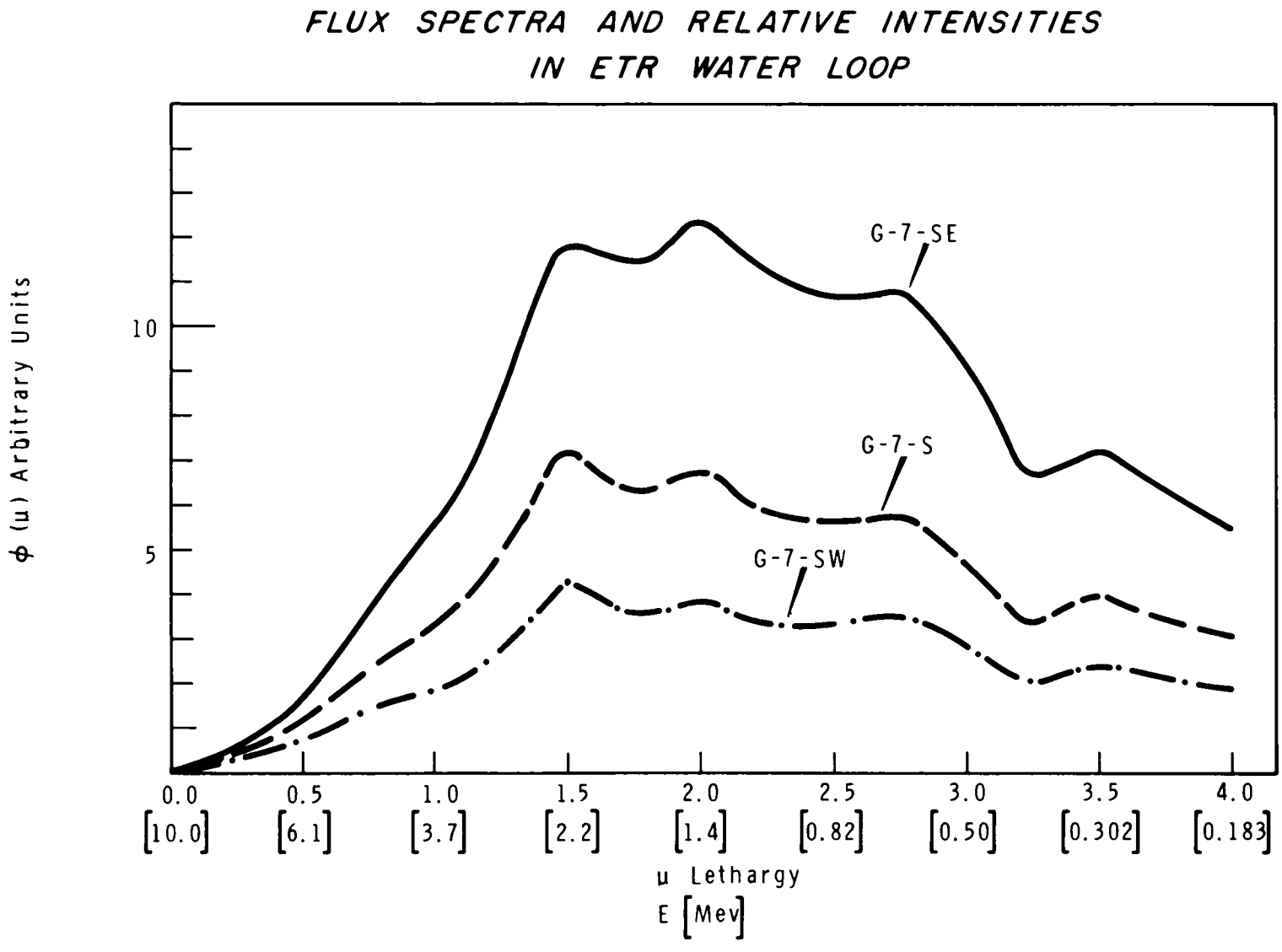

FLUX SPECTRA AND RELATIVE INTENSITIES
IN ETR WATER LOOP
φ (u) Arbitrary Units
10
5
G-7-SE
G-7-S
G-7-SW
0.0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
[10.0]
[6.1]
[3.7]
[2.2]
[1.4]
[0.82]
[0.50]
[0.302]
[0.183]
u Lethargy
E [Mev]

FIGURE 3

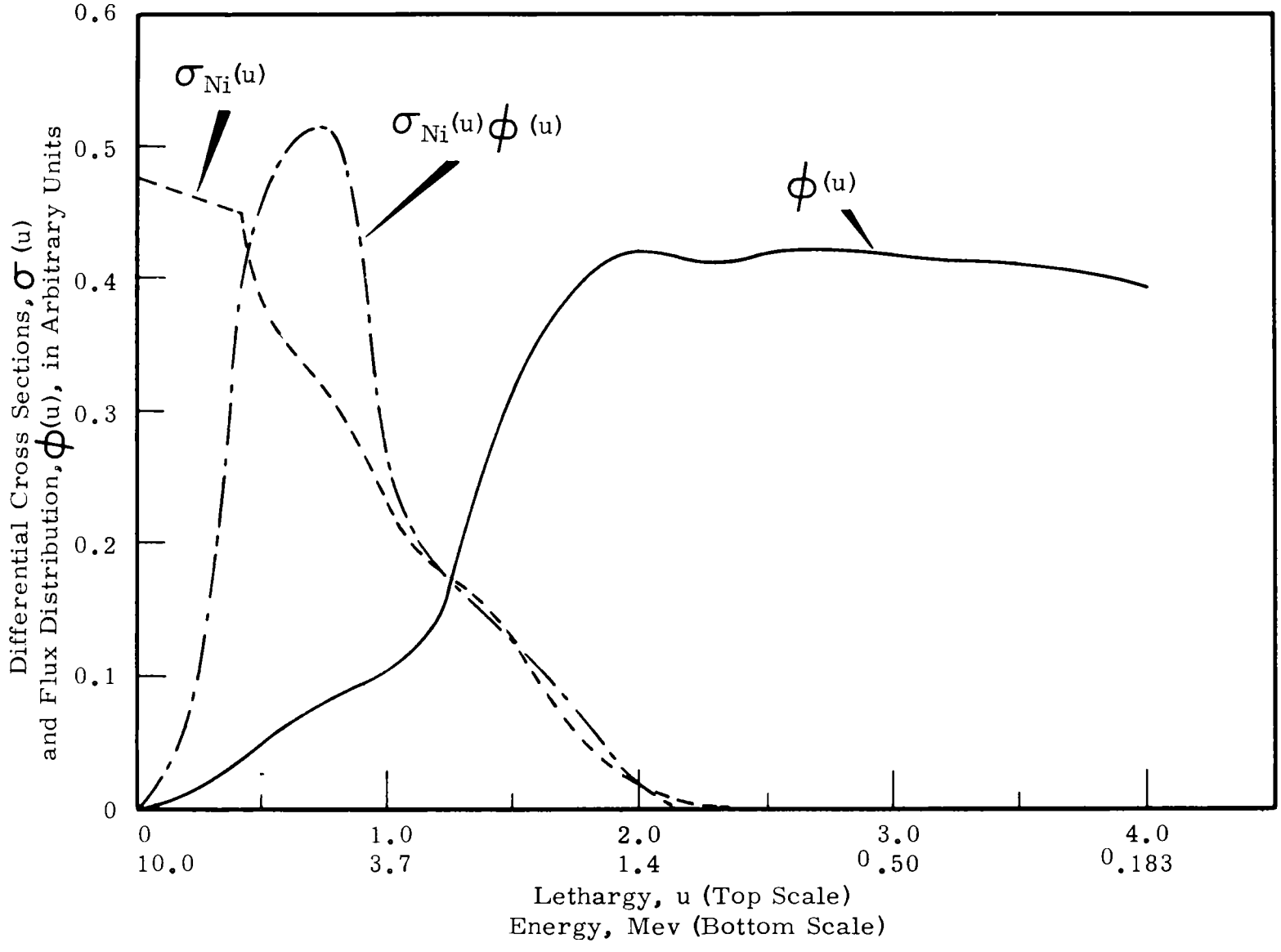


FIGURE 4. EFFECTIVE CROSS SECTION FOR NICKEL

CAPSULES FOR NON FISSILE IRRADIATIONS

N. H. HANCOCK

A.E.R.E. HARWELL

1. Introduction

1.1 A series of rigs have been developed for investigating the effects of irradiation on the physical properties of graphite. In the course of the work it was found that this type of rig could be developed for various other neutron irradiation damage experiments using the Mk.III fuel elements in D.I.D.O. or P.L.U.T.O.

1.2 It is important to note in these Mk.III fuel elements, that there is an annulus formed between the thimble and the inner tube of the fuel element; this being open to the incoming D_2O flow via perforations in the end cone. The resulting flow of D_2O in this secondary path amounts to approximately 5% of the total flow through the fuel element. Since the annular width is held constant by spacing pads at each end, the flow is uniform and provides an important cooling service for the experimental rigs installed in the fuel element thimble.

2. Series I Rigs

2.1 The Series I Rigs were designed in 1958 to study irradiation damage to graphite and subsequently other non-fissile materials over a temperature range 150°C to 350°C.

2.2 The rig consists of a shield plug from which is suspended three specimen carrier capsules located vertically one above the other each carrying nine 0.250 in. diameter specimens. These specimen capsules fit within a stainless steel or zircaloy thimble, which replaces the normal aluminium thimble. When in position in the reactor the centre specimen carrier is on the centre line of the reactor core with the other two approximately 7.0 in. above and below the centre line.

2.3 The specimen carrier capsule is formed from an aluminium carrier drum approximately 3.0 in. long with an approximate wall thickness of 0.280 in., in which nine 0.250 in. diameter specimen pockets are longitudinally bored. Three additional 0.0625 in. diameter pockets are bored to contain Chromel/Alumel mineral insulated thermocouples whose hot junctions are located at the top, middle and bottom of the carrier. Two pockets are also provided for cobalt and nickel wires for neutron dose determination. End caps screwed to the carrier drums carry tubular extension spigots by means of which the carriers are connected to each other, the connections throughout being articulated to prevent mechanical interference. A stainless steel tube is positioned between the end caps to carry a heater coil wound on an 0.250 in. layer of C.60 Alumina cement; the thermocouple and heater leads being carried up through the centre of the specimen carrier via a rubber bung type seal to the terminal strip on the shield plug top cap.

2.4 The specimen carrier is located accurately in the thimble by three equi-spaced locating spigots on the circumference of each end of the carrier drum, the spigots being honed after assembly to establish an accurate annulus between the specimen carrier and the thimble. From the start of the irradiation programme stainless steel thimbles were used in place of the normal aluminium thimbles to obtain the maximum dimensional accuracy. These were in turn replaced by zircaloy thimbles since the reactivity absorption of the latter was considerably less than stainless steel. Each thimble had the lower part of the bore honed accurately to 2.00 in. diameter to guarantee a uniform gas-filled annulus between the thimble and specimen carrier. The radial clearance, in the cold condition, between the locating spigots and the internal circumference of the thimble being sufficient to permit load/unload of the specimen carrier assembly from the thimble.

2.5 The nuclear heat generated in the rig, plus the electrical heat, is passed across the helium filled annulus between the specimen carriers and the thimble, through the thimble wall and into the D_2O annulus formed by the thimble and the inside surface of the fuel element plate assembly. The design is arranged to give a specimen carrier temperature 20% below the specified operating temperature with nuclear heating alone, the carriers are then brought to the operating temperature by the input of electrical heating, thereby establishing a precise control of the temperature.

3. Series II Rig

3.1 The Series II Rig was designed for irradiating graphite and other non-fissile materials at temperatures in the range 450°C to 650°C, and has with further development been designed for use at temperatures of up to 900°C.

3.2 This type of rig is identical in arrangement and principle to the Series I Rig used for low temperature irradiation described above. The major difference imposed by the high temperature range is in the materials from which the specimen capsules were manufactured.

3.3 A stainless steel suspension tube couples three specimen carrier capsules to the central shield plug of a Mk.III fuel element. Each specimen capsule consists of a graphite carrier drum of approximately 1.875 in. diameter with a wall thickness of approximately 0.280 in. in which nine 0.25 in. diameter holes are bored axially to form the specimen pockets. Five additional holes are bored to contain three thermocouples and two cobalt and nickel monitors. The carrier drum is located between stainless steel end caps, which are clamped together by three .093 in. diameter stainless steel rods passing through the holes in the drum between the specimen pockets.

3.4 An electric heater, of the identical design fitted to the Series I Rig, is supported between the end caps precisely concentric with the carrier drum bore; the connections to the heater being carried out with nickel/manganese wires in Alumina sleeves.

3.5 The specimen carrier is centralised in the thimble bore by three contact spigots machined on the end caps, these being honed to give an accurate width of annulus, as determined by the temperature at which the rig is designed to operate. Specimen temperatures are measured by Chromel/Alumel thermocouples and the dose is monitored by cobalt and nickel wires.

3.6 For the 900°C rig Nimonic wire was used for the heater connections in the hot zone, in place of the nickel manganese, and the heater element covered with a cement layer to reduce the helium filled annulus between the heater and the specimen carrier thereby lowering the temperature of the heater winding.

3.7 These designs have been found admirably suited for irradiating a large variety of materials with a selection of specimen geometries and has been developed to operate at temperatures up to 900°C. Furthermore by modification and additions to the basic design, rigs have been developed for studies of irradiation creep in graphite and irradiation of encapsulated specimens containing controlled atmospheres.

3.8 One outstanding advantage of such a simple design is the ease with which irradiated specimens can be unloaded and new specimens loaded. This operation enables a rig to be installed in reactor for a succession of irradiations thereby effecting notable economies in rig manufacture.

4. Series III Rigs

4.1 The Series III Rig is an expanding specimen carrier capsule rig used to study irradiation damage to graphite over a temperature range 150°C to 350°C and is for operation at increased reactor powers. It was designed to achieve considerably reduced axial and radial temperature gradients to those of the Series I and II capsules.

4.2 This rig is similar to the Series I Rig insofar as it consists of three specimen carrier capsules suspended from a shield plug. To achieve minimum temperature gradients thermal barriers are located at the ends of the specimen carriers to prevent end losses, and the heater is positioned between the specimens and the heat sink thereby reducing the quantity of heat passing radially through the specimens. The specimen carriers are designed to expand after loading into the thimble; this action completes a relatively high thermal conductivity path from the specimen carrier to the D_2O. In addition it completes several secondary thermal paths whereby most of the heat induced in the components supporting the carrier can escape to the D_2O without passing through the specimens. Locating the heaters between the specimens and the heat sink resulted in a design feature whereby the heaters are secured to the bore of a Mk.III fuel element thimble by means of heat transfer fins. These pass

approximately 90% of the heat of the specimen carrier to the thimble wall and thence to the D_2O. The remaining 10% is conducted across the helium filled annulus between heater and thimble.

4.3 The rig consists of two assemblies, a thimble/heater assembly and a specimen carrier assembly. The latter consists of three specimen carriers suspended from a shield plug, of conventional Mk.III fuel element design, by means of a stainless steel tube fitted with heat transfer collars at the lower end.

4.4 The shield plug is smaller in diameter than the standard shield plug in order that it will fit into the thimble shielding sleeve, through which the heater cables are spiralled. It carries the usual conduit for the thermocouple leads and an evacuation/helium purge tube, and in addition a central tube through which passes the push rod for operating the specimen carrier expanding mechanism. The three specimen carriers are connected to each other by aluminium coupling tubes, each specimen carrier being fitted with five 1mm thermocouples for temperature indication and heater control. Within each carrier is assembled a mechanism of opposing cones whereby downward movement of the push rod results in the carriers and thermal barriers being expanded, and an upward movement results in contraction; the push rod down the centre of the system being operated by the actuating lever at the top of the shield plug.

4.5 The rig is loaded into the thimble/heater assembly with the specimen carriers contracted, and when fully in position they are expanded to make intimate thermal contact with the bores of the heaters. It is important to note that by expanding the carrier the flanges of the coupling tubes make contact with the bores of the isolated heater end rings, and by the path thus created most of the heat induced in the coupling tubes and operating mechanism passes to the D_2O without passing through the specimens.

4.6 The thimble/heater assembly is of fabricated construction. It consists of three heater units welded together, with an extension sleeve and cone assembly welded at one end and a 99.8% aluminium thimble tube to the other. The latter is welded to a specially designed shielding sleeve support flange, which replaces the standard thimble top ring. The shielding sleeve, which extends for a distance of four feet inside the thimble, is machined in two sections from stainless steel bar. The sections are screwed together and eight 0.125 in. by 0.125 in. spiral grooves are cut in the outside surface to accommodate the heater leads and leak detector supply leads.

4.7 The heaters are formed by winding a coil of 0.100 in. diameter mineral insulated stainless steel sheathed heater cable onto an aluminium sleeve former, which is grooved to receive the wire. An outer sleeve which has thirty-six fins machined in its periphery is screwed onto the heater and rivetted to the inner sleeve to form the complete heater assembly. The fins are formed and secured to the bore of the thimble section by a dip brazing technique, this technique also being used to secure the heater end rings. The assembly is finally machined in the bore for alignment before welding to its neighbour to form the complete thimble. A leak detector is secured in the bottom of the thimble.

4.8 As already discussed 90% of the heat produced passes radially outwards to the reactor D_2O by means of the metallic paths, the remaining 10% crosses the helium filled annulus between the outside of the heater and bore of the thimble. Control of the 'free running' temperature of the system can therefore be achieved by varying the impedance of the metallic thermal path, this is accomplished by calculating the width of the heat transfer fins to give a specific temperature in the specimen carrier.

4.9 This rig has not yet been installed in the reactor and therefore no performance figures are available. Mathematical analysis and test and commissioning figures indicate that the temperature gradients will be much improved on those obtained with the Series I capsules.

5. Series IV Rigs

5.1 The Series IV Rigs were designed to determine the effect of neutron irradiation on the physical and mechanical properties of non-fissile materials over the temperature range 150°C to 650°C. In particular the Series IVA Rig was designed for irradiating graphite with low axial and radial temperature gradients, and is a much simpler mechanical construction than the Series III Rig.

5.2 Four types of this rig were designed, the Series IVA, Series IVB, Series IVC and Series IVD. They were all identical in principle and similar in arrangement, the major differences being in the number of specimen carrier capsules per rig, materials used for the construction and the design temperature range. The Series IVA rig was constructed with one specimen carrier 9.0 inches long made of aluminium and designed for operation in the temperature ranges 150°C to 350°C. The Series VIB Rig was for operation in the same temperature range but incorporated three specimen carriers each operating at a different temperature. The Series VIC Rig again had three specimen carriers but in this case was constructed of stainless steel for operation at a temperature of 650°C. The Series VID which followed was for a temperature range of 300°C to 500°C, and therefore the carriers were made of aluminium with the specimen pockets lined with stainless steel. The Series IVB Rig is described in the following paragraphs and the basic design is the same for the Series IVA, C and D Rigs.

5.3 A stainless steel suspension tube couples the specimen carriers to a shield plug of conventional Mk.III fuel element design. The specimen carriers are manufactured from various materials, as discussed in paragraph 5.2. The carrier drums have a nominal outside diameter of 1.875 in. with a wall thickness of approximately 0.3125 in. in which nine specimen pockets are longitudinally bored. In addition, eighteen 0.0625 in. holes are arranged in pairs between the specimen pockets on a pitch circle of 0.25 in. less than the carrier drum, to contain a heater cable. The carrier drum is located concentrically between stainless steel end caps with stainless steel thermal barriers interposed between the end caps and carrier drum. A mineral insulated stainless steel sheathed heater cable is looped up and down through the holes provided, forming a continuous heater at the periphery of the carrier drum. The cable ends are brought out through the upper end cap and coupled by a conventional junction to 14 SWG mineral insulated nickel/manganese supply leads installed in the centre of the system.

5.4 The specimen carriers are centralised in the bore of the fuel element by three contact spigots machined in the periphery of the thermal barriers, the spigots being honed to give a precise width of annulus as determined by the temperature at which the rig is designed to operate. Specimen temperature is controlled by a thermocouple fitted into one of the specimens in each carrier drum, three other thermocouples are also fitted in each drum for record and alarm/trip instrumentation. All cables pass up through the centre of the rig to terminal blocks located at the top of the shield plug.

5.5 All heat generated in the system passes out through the thimble wall to the D_2O flow between the thimble and fuel element inner tube. The annular gap between the specimen carrier and thimble is designed to give a nuclear heating temperature approximately 25% below the control temperature. The extra heating required being provided by the electric heaters. The system is filled with helium and this acts as a heat transfer medium between the specimen carrier and thimble wall.

6. Conclusions

6.1 The rigs and capsules discussed in this paper are in general, derived from the Series I Rig. This design was finalised and the first two rigs manufactured in a very short period in order that an important irradiation programme could be commenced. This background of urgency demanded only the simplest design concepts; it was therefore considered probable that rig performance would not quite reach the specified temperature requirements. In-reactor experience has supported this view, it must be emphasised however, that despite difficulties the rig fulfilled its original purpose and this design has completed a great number of other useful irradiations.

6.2 The Series III and Series IV Rigs were developed as alternatives to the Series I Rig, the former, as would be expected, required a great deal of development work, which seriously delayed its completion. The Series IV Rig required less development work during manufacture than did the Series III Rig, but has not shown a marked improvement in axial temperature gradients that had been anticipated. In consequence it will require further development work to reduce the axial heat flow at the ends of the specimen carrier drums, and to improve the heater performance.

6.3 In the light of experience gained with this series of rigs it can be justly claimed that the Series I Rig has been a successful design and a valuable irradiation tool. Furthermore it underlines the design axiom that the simplest device, that can reasonably be expected to accomplish the experimentalists requirements should be exploited first, more complex refinements being introduced as experience is gained and development problems solved.

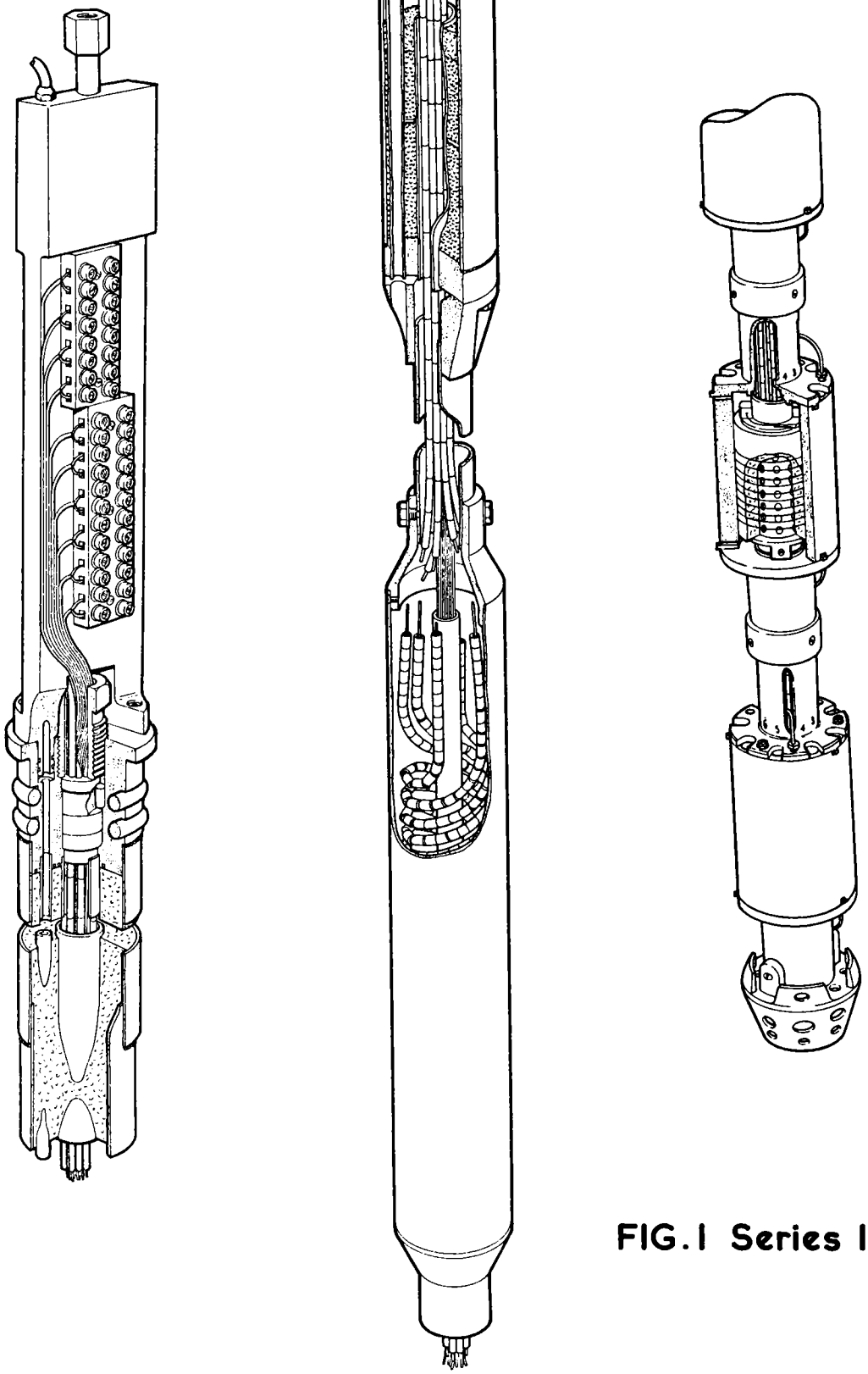


FIG.1 Series I Rig

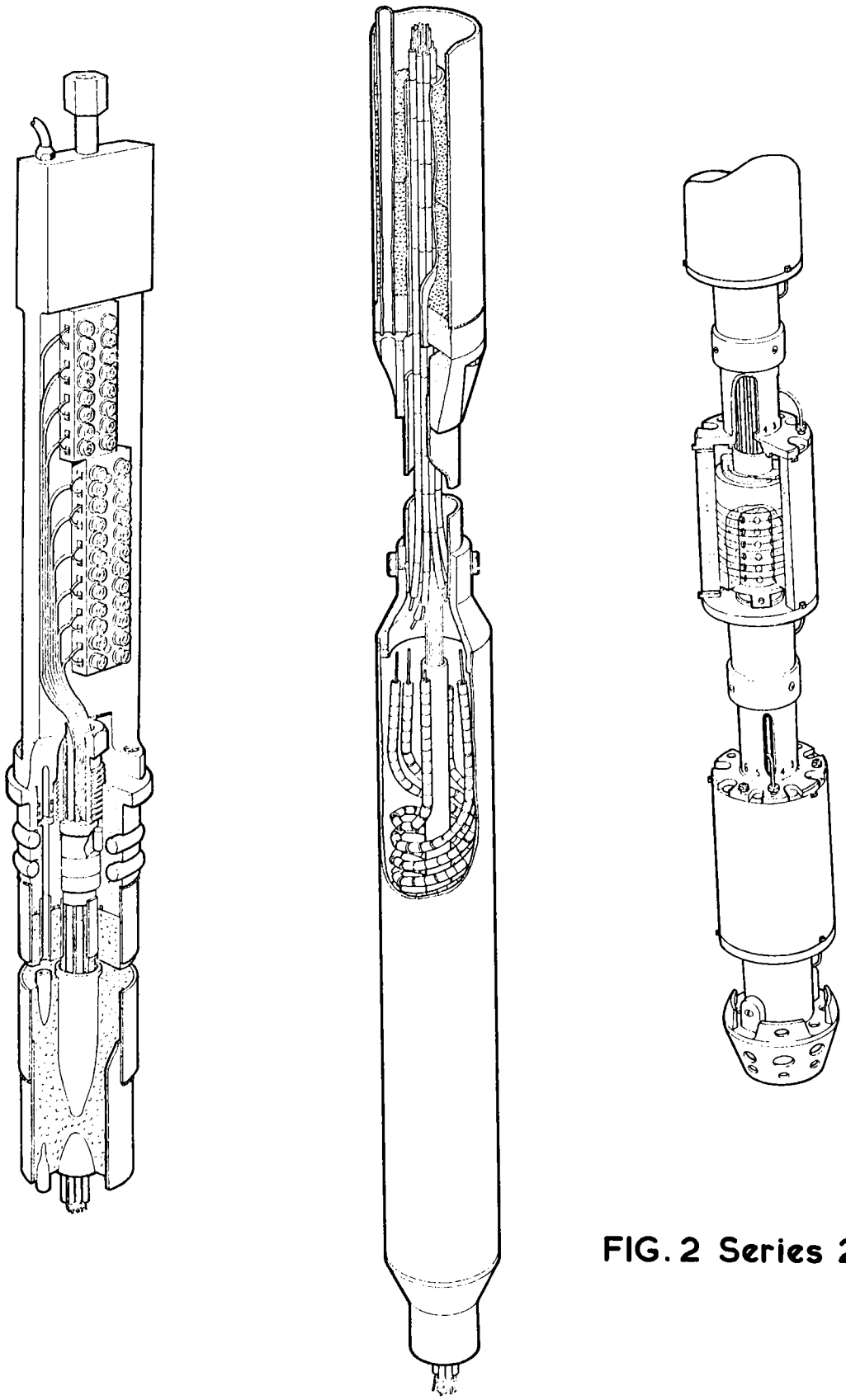
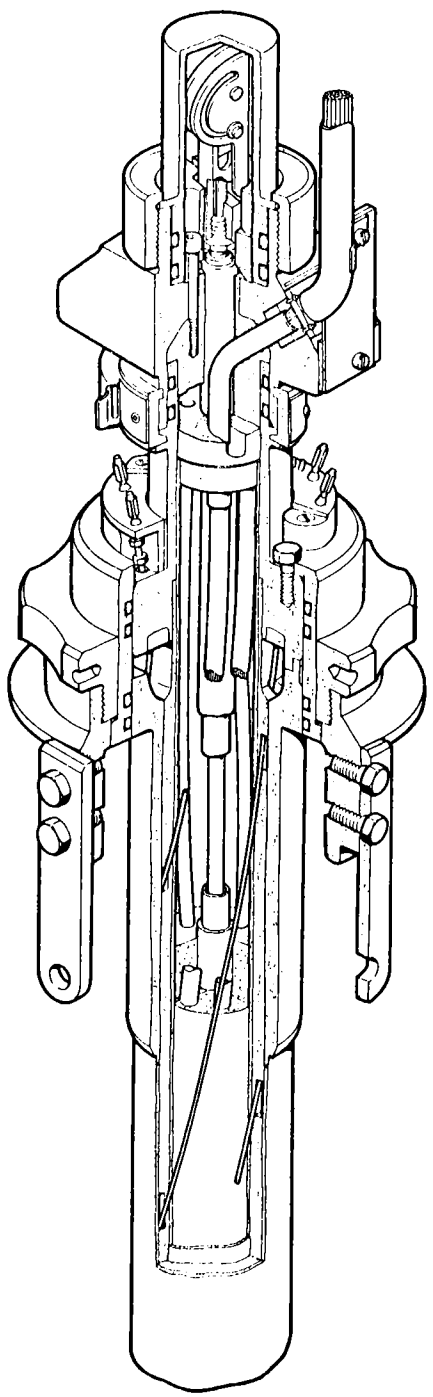
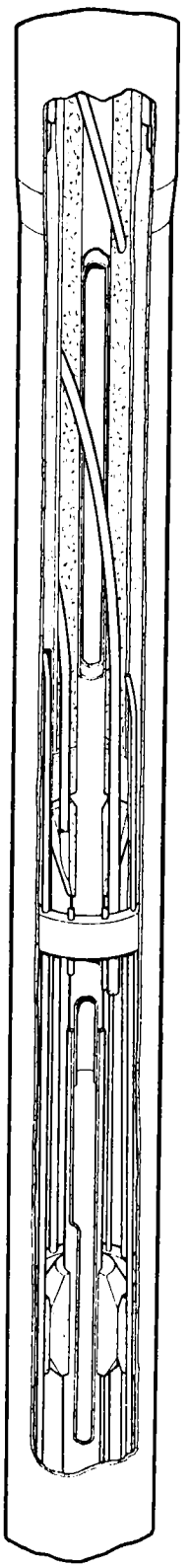


FIG. 2 Series 2 Rig





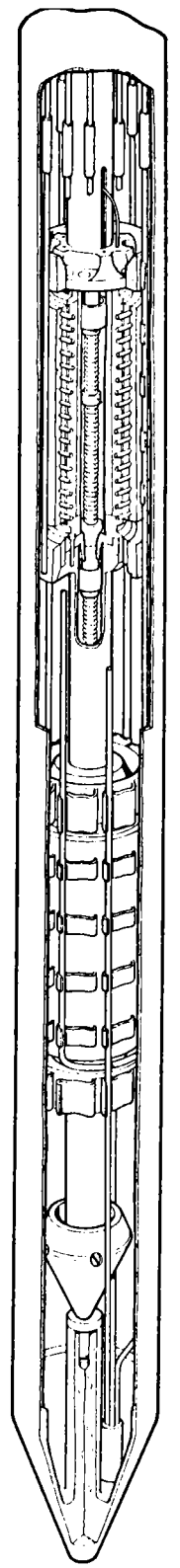


FIG.3 Series 3 Rig

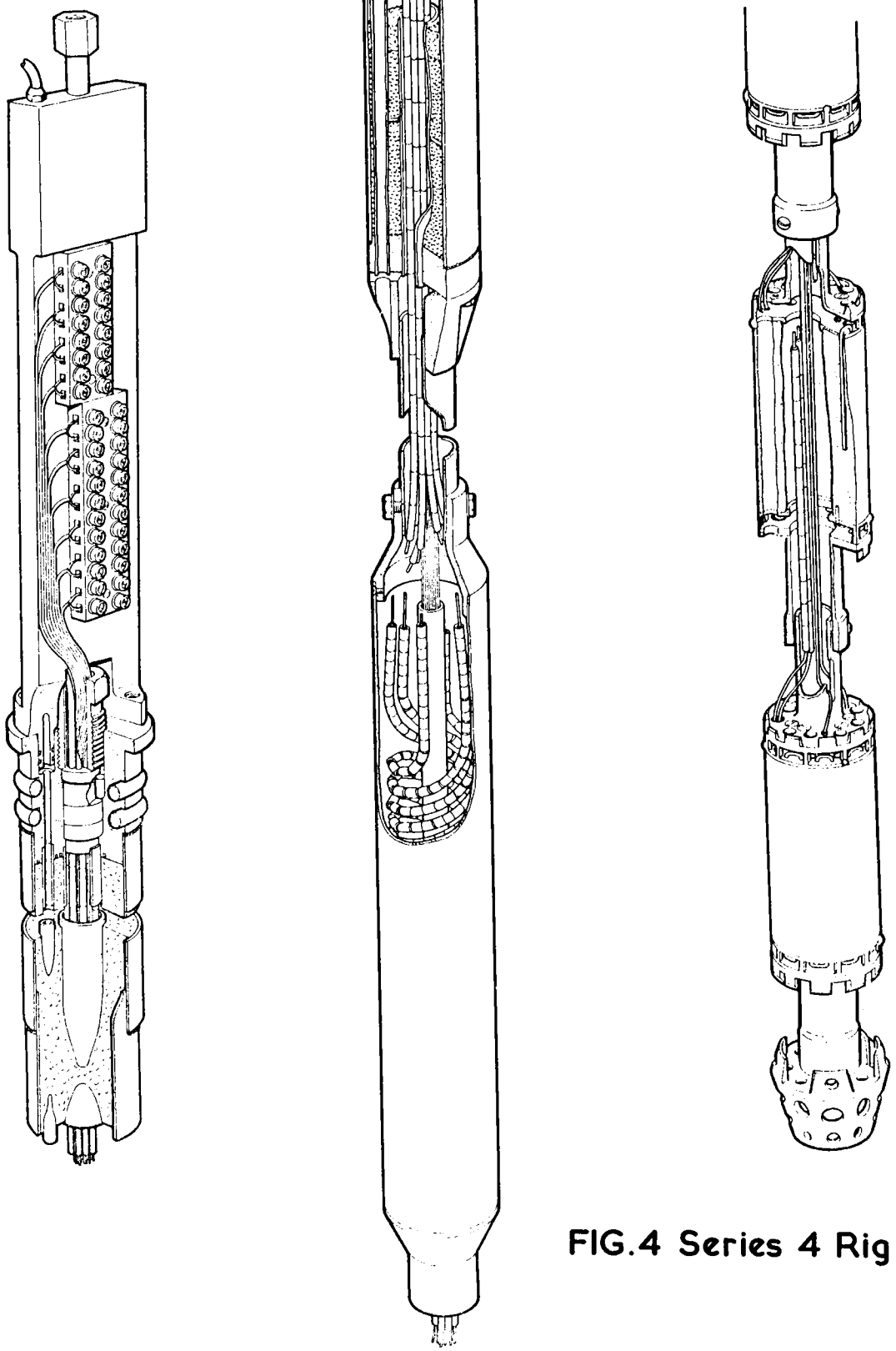


FIG.4 Series 4 Rig

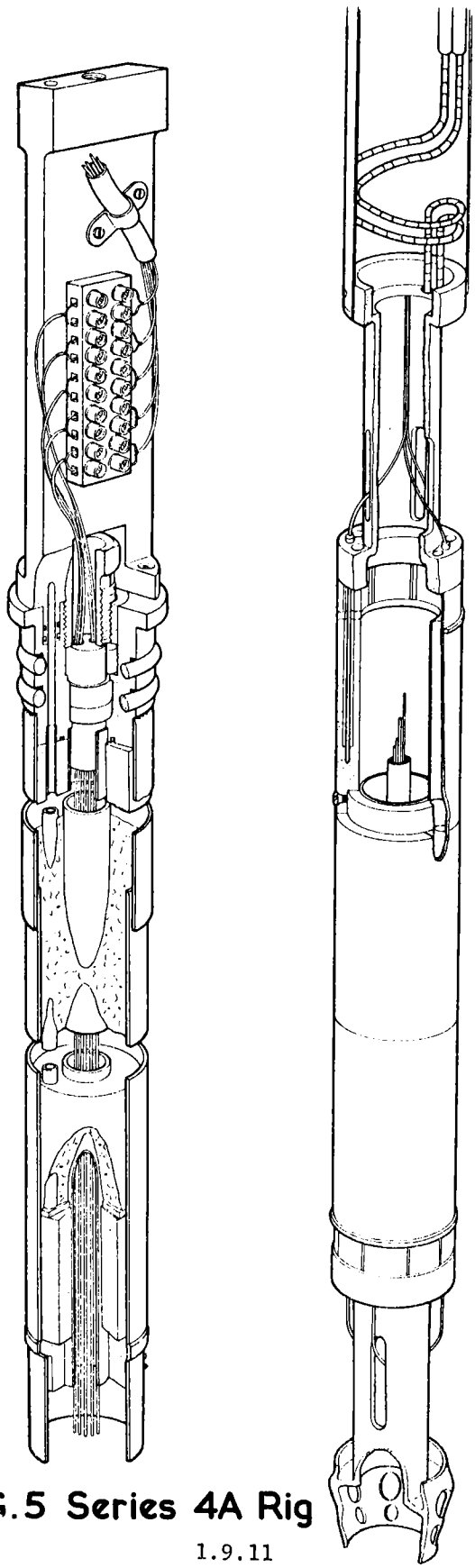


FIG.5 Series 4A Rig

1.9.11

II. CAPSULE DESIGN AND OPERATION - FISSILE MATERIALS

DESIGN OF THE EIGHT-BALL IRRADIATION CAPSULE

D. B. Trauger*

A capsule was needed in which several spherical graphite elements containing coated-particle fuels could be irradiated simultaneously under similar conditions for comparative purposes. Fabrication variables were to be evaluated, including methods for preparing particles and their coatings, the type of graphite mix used for the matrix, and unfueled shell forming techniques, that is, blending procedures, molding processes, heating rates, final treatment temperature, and impregnation processes. Findings were to be obtained by postirradiation examination; no measurements were to be made of fission-gas retention. A relatively high fast-neutron flux was desired to accumulate radiation damage in the graphite comparable to that expected in a power reactor.

Core position F-1 of the ORR was chosen as a suitable irradiation facility. It has the advantages of a relatively high fast flux, a uniform thermal flux over a considerable vertical length, and adequate cooling capacity for the spherical elements in a compact arrangement. Important considerations in design were measurement of temperature and simplicity in construction to minimize cost. The facility afforded sufficient space to accommodate 1-1/2-in.-diam fuel elements, which in this case were spheres. With this diameter and considering the ORR vertical flux profile, it was determined that suitable irradiation conditions could be achieved for eight spheres; consequently, the title "Eight-Ball" was applied to the experiment.

Vertical thermal-neutron flux profiles for the F-1 core position and the relative positions of the fuel spheres are shown in Fig. 1. As may be seen, irradiation conditions for fuel spheres at the center and ends are somewhat different, but elements 3 through 6 are subjected to quite

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*Oak Ridge National Laboratory operated by Union Carbide Corporation for the U.S. Atomic Energy Commission.

uniform conditions and the relative fluxes for the spheres located in positions 1 and 8 and 2 and 7, respectively, should be comparable.

The arrangement of fueled graphite spheres in a graphite structure is similar to that previously used by BMI in a study conducted for Sanderson & Porter.[1] Eight-ball capsule is of interest principally because of refinements in heat transfer design and its demonstrated success in achieving the desired operating conditions. The basic configuration chosen for the experimental assembly is shown in Fig. 2. The graphite parts are carefully machined and close-fitting. Spheroidized coke particles are used to enhance the heat transfer from the fuel spheres to the graphite structure. The fuel spheres are centered by small pins inserted in the graphite. Temperatures are measured by thermocouples inserted in the graphite structure and, when appropriate, inside the fuel sphere. A thermocouple can be installed conveniently only in the top fuel sphere. For the present series of irradiations, temperatures at the thermocouples are not required to exceed 1600°F. For this application, copper-plated stainless-steel-sheathed Chromel-P vs. Alumel thermocouples are employed. For higher temperatures, a tungsten-rhenium thermocouple, with BeO insulation, installed in a molybdenum well and coated with rhenium for protection against damage by carburization will be employed in the center of the fueled spheres. This type of thermocouple has performed satisfactorily in capsules at temperatures up to 2400°F.

The structure was designed to provide variable radial heat transfer from the centerline to the top and bottom of each sphere and thus compensate for the increased conduction path in graphite. The gas gap between the graphite and inner wall of the primary stainless steel containment vessel determines the operating temperatures of the graphite and the fuel relative to the primary containment structure. A second gas gap between the primary container and a secondary stainless steel container (not shown in Fig. 2) was dimensioned to produce a temperature difference of several hundred degrees Fahrenheit when filled with a helium-nitrogen mixture. The mixture can be varied to provide temperature control. To compensate for the increased conduction path through the graphite at the poles compared with that at the midplanes of spheres, conducting spacer

wires were brazed to the inner containment vessel between each pair of spheres. These wires serve both to reduce the temperature locally and to center the inner containment structure. The radial temperature distribution for one set of conditions is shown in Fig. 3. For a capsule constructed with the dimensions given, the design temperature of 1100°F at the thermocouple position was achieved with a helium-nitrogen mixture containing 30% helium.

Flux depressions in the graphite structure and fuel spheres were determined by the method of Lewis[2] for the outer containers. Flux depressions in the fueled spheres were computed according to Bartells' method.[3]

The F-1 position of the ORR has a relatively high gamma heat, ~5 w/g, which caused concern with regard to the design of the end-cap closure for the inner containment vessel. With a uniform gas gap in the helium-nitrogen annulus, it was apparent that regardless of the configuration of the bulkhead, which was required to withstand an equilibrium pressure ~60 psi from gas temperature rise by combination of Charles law, outgassing, and fission-gas release, the predicted temperature of 2025°F would be too great. To correct this condition, a 5-mil layer of silver was electroplated over the bottom and outer surface of the bulkhead to reduce the gas gap and increase the overall thermal conductivity so that the calculated equilibrium temperature was reduced to 850°F. In operation, the bulkhead temperature has not exceeded 750°F.

The longest irradiation to date was for one eight-week cycle at the ORR; a longer irradiation is now in progress. All parts of the first irradiated capsule appeared to be undamaged upon dismantling in the hot cell. This alleviated concern over the mechanical stability of some of the thin graphite sections, although their performance in long-duration irradiations is yet to be established. Overall performance of the capsule has been quite satisfactory. Temperatures in the graphite at the midplanes of fuel spheres have varied from 1040 to 1130°F, with an approximately symmetrical pattern about the vertical center.

References

1. "Fuel Element Development Program for the Pebble Bed Reactor," USAEC Research and Development Report NYO-9064, Sanderson & Porter Company, April 30, 1961.
2. W. B. Lewis, "Fast Perturbations by Materials Under Irradiation," Nucleonics, October 1955.
3. W. J. C. Bartells, "Self-Absorption of Mono-Energetic Neutrons," USAEC Report KAPL-66, Knolls Atomic Power Laboratory, May 1960.

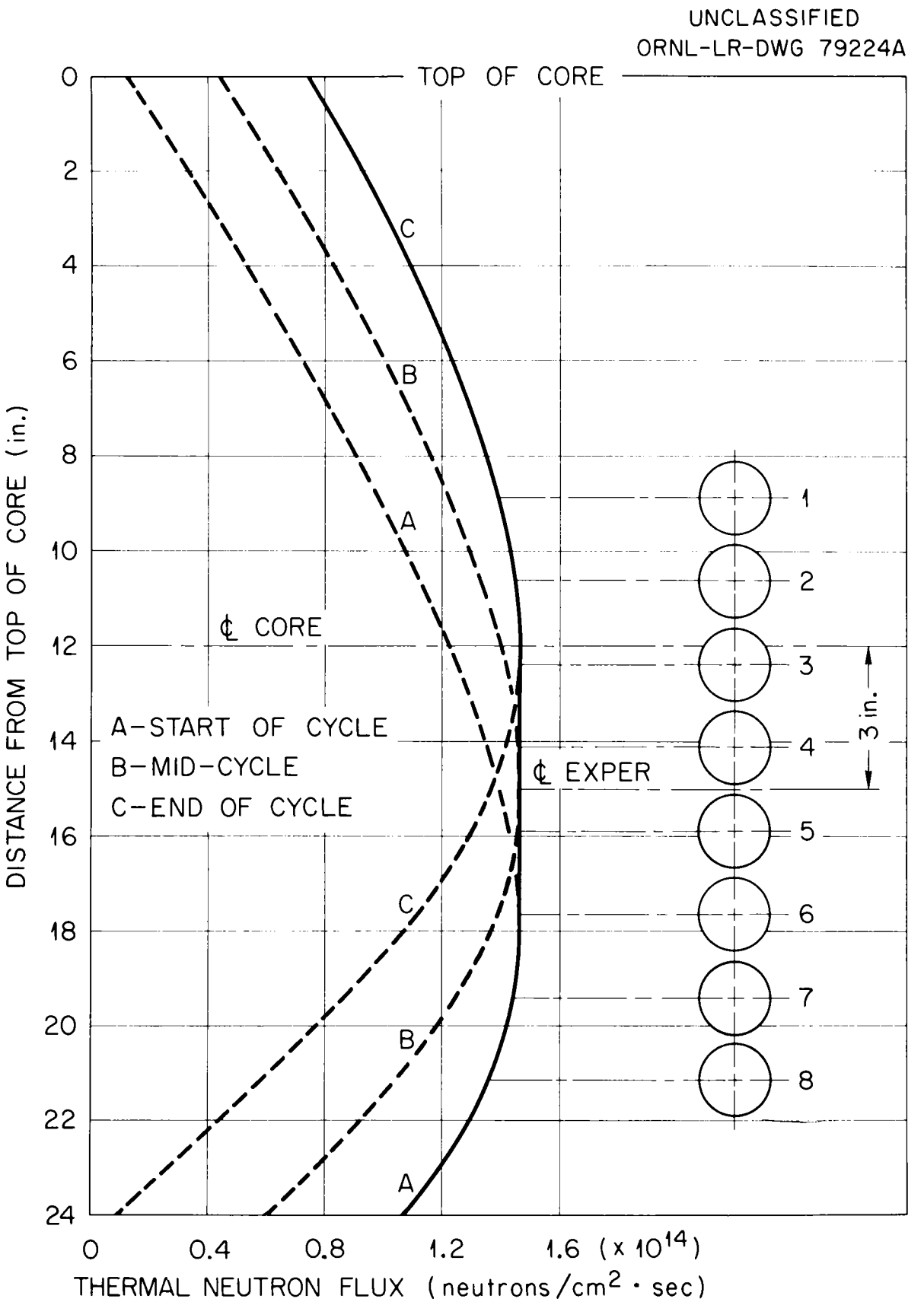


Figure 1.
Thermal Neutron Flux in the F-1 Core Position of the ORR.

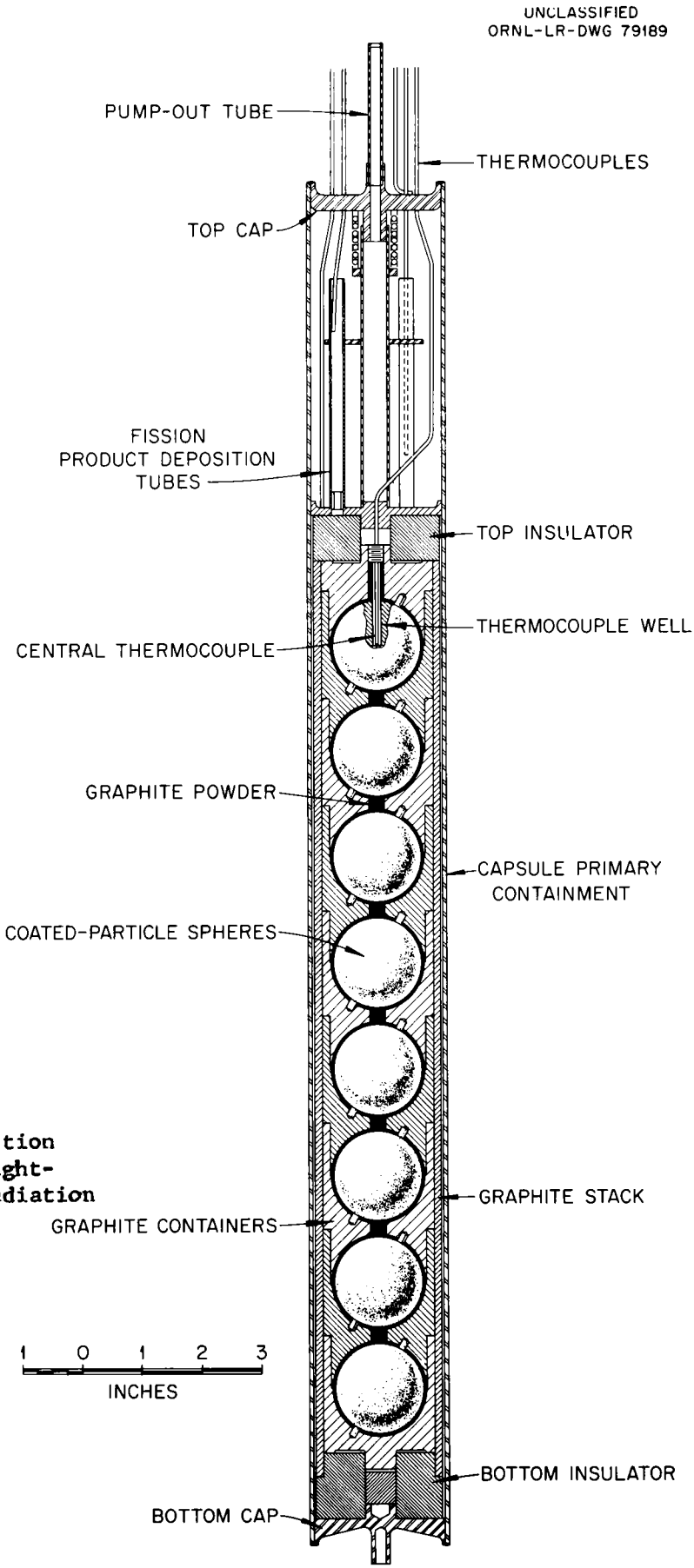


Figure 2. Cross Section of ORR Eight-Ball Irradiation Capsule

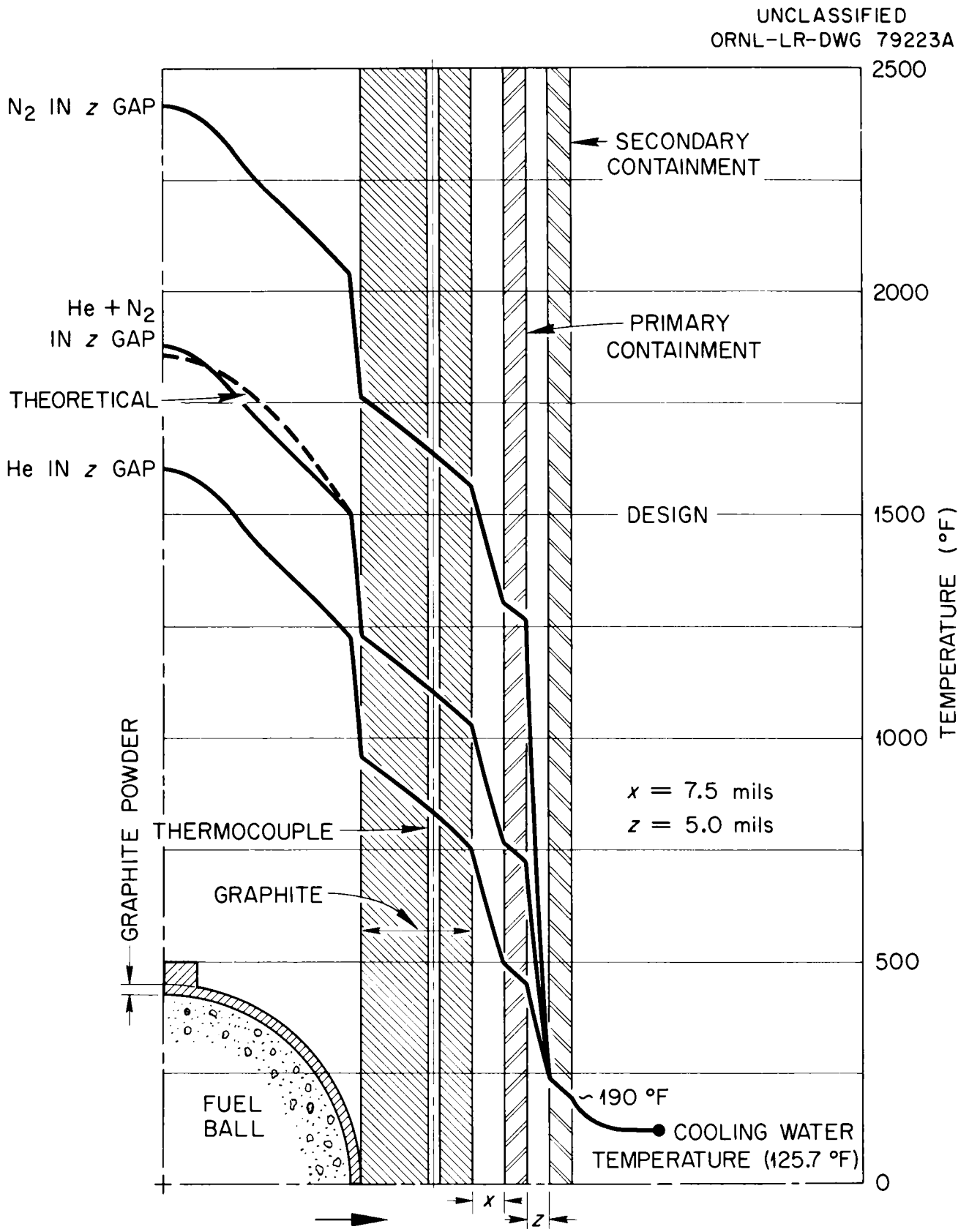


Figure 3.
Temperature Distribution Across Eight-Ball Capsule.

CAPSULE FOR IRRADIATION OF URANIUM IMPREGNATED GRAPHITE AT 1500° C TO 10% BURNUP *

P.J. Peterson, J.A. Leary, and W.J. Maraman

Los Alamos Scientific Laboratory
University of California
Los Alamos, New Mexico

To obtain preliminary data on the proposed fuel elements for UHTREX (Ultra High Temperature Reactor Experiment) a capsule irradiation program was undertaken.(1)

Using standard calculational methods for heat and hydraulic flow, a capsule was designed to fit a 2-in. diameter hole in the core of the OWR and is shown in Fig. 1. This represents the final design and incorporates changes found necessary as a result of failures in the first three capsules, which will be discussed subsequently. The design of Capsule No. 4 features two UHTREX type elements in a stainless steel tube with a uniform gap at room temperature of 0.037 in. between tube and element maintained by Ta spacers. Insulators of CaO were placed at both ends of the fuel element-spacer assembly.

Thermocouple wells of Ta containing W-W 26% Re thermocouples with high purity Al_2O_3 insulators were used to measure temperatures at midpoints of the two elements.

To protect against capsule failure and release of fission

- - - - - - - - - - - - - - -

*Work done under the auspices of the U.S. Atomic Energy Commission

(1) P.J. Peterson, J.A. Leary, W.J. Maraman, D.C. Carlson, and R.W. Walker, "Reactor Irradiation of Uranium-impregnated Graphite at 1500° C to 10% Burnup," LAMS-2814, Jan. 28, 1963.

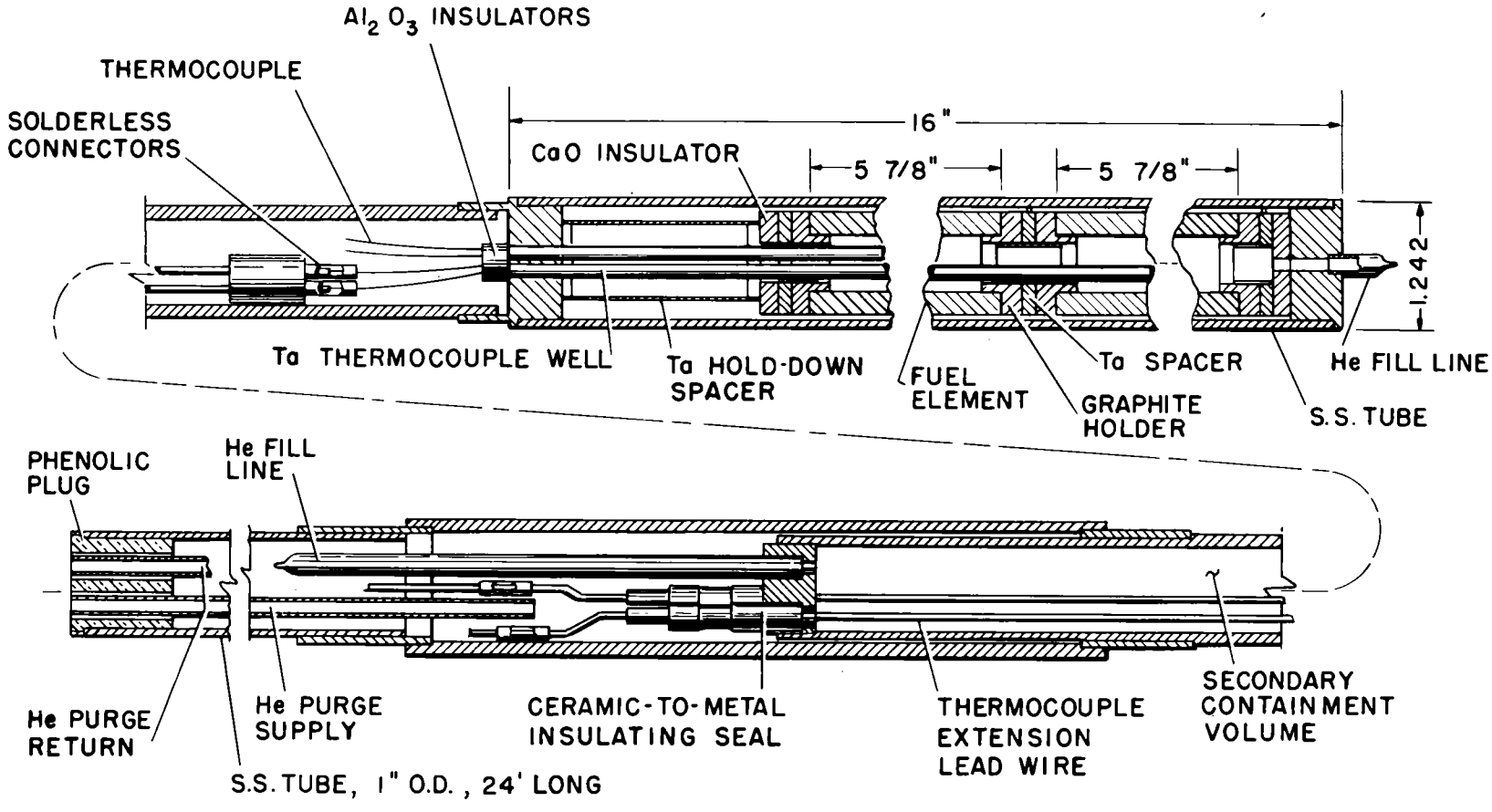


Fig. 1. Irradiation Capsule for Uranium Impregnated Graphite

products to the top of the reactor through the stainless steel tube housing the thermocouple extension lead wires, a second sealed volume was incorporated above the capsule proper. This seal was made by using ceramic-to-metal insulating seals which passed the extension lead wires and was located outside the core in a low flux-low temperature region. Both the capsule and the secondary protective volume were back filled with He to atmospheric pressure on assembly. Final sealing of both volumes was accomplished by pinching and melting through the He fill lines of 3/16 in. stainless steel tubing. The entire assembly, including the extension lead wire housing, was checked for integrity with a mass spectrometer He leak detector.

Also included in the housing tube for the extension lead wire was a He purge so if fission products were released they would be carried to the top of the reactor and detected by monitoring instruments.

Capsule No. 1, a calibration experiment, was essentially the same as that shown in Fig. 1 except that CaO spacers were used instead of Ta, and the lead wire housing tube did not have a secondary seal. Capsule No. 1 was inserted in the reactor and the reactor was brought up in power until the temperature of the capsule reached 1580° C. Failure occurred after 5 min. at this temperature, but enough information was gained to evaluate the flux and calculate a gas gap that would achieve a temperature of 1550° C at a reactor power of 5.0 MW. A post-irradiation examination of the capsule showed failure occurred when the Pt-Pt 10% Rh thermocouple parted at the hot junction, apparently soldering itself to the bottom of the Ta thermocouple well which it was touching. The bottom also dropped out of the Ta tube, where it was plugged and welded, permitting fission products to be picked up in the He purge and brought to the top of the reactor.

For Capsule No. 2, the gas gap was set at 0.032 in. and the Ta spacers shown in Fig. 1 were substituted for the CaO spacers. The reason for this change was the fracture of the CaO under thermal

shock. Tantalum-sheathed Pt-Pt 10% Rh thermocouple assemblies of 0.060 in. diameter with MgO insulation were used to measure the temperature in this capsule. The couples were not grounded to the sheathing.

A temperature of 1550° C at 5.0 MW of reactor power was attained at the start of the second irradiation, but this gradually declined to 1070° C after 75 hr. of operation. At this time the reactor was shut down and the capsule temperature dropped from 1070° C to 200° C in ~9 min. At this point both thermocouples failed, apparently from differential thermal expansion.

In Capsule No. 3 the gas gap was increased to 0.037 in. The intent was to operate the reactor at reduced power until a temperature decline similar to that experienced in the prior capsule would permit operation at 5.0 without increasing the capsule temperature beyond 1550° C. It was believed that the temperature decline was a result of a change in emissivity of the capsule walls due to condensation of volatile fission products or deposition of carbonaceous material from the element as seen in the autopsies, and/or a change in emissivity of the fuel elements.

A thermocouple assembly such as was used in Capsule No. 1 and is shown in Fig. 1 was incorporated in the belief that it was more stable to thermal stresses and shock.

This capsule attained a temperature of 1450° C at 4.0 MW of reactor power, after which the reactor power was gradually increased over a 24 hr. period until 5.0 MW was reached and the temperature increased to 1550° C. The capsule operated for a period of 75 hr., until fission product activity was discovered in the lead wire housing tube, and was then withdrawn.

Post-irradiation hot cell examination revealed that the fission product leakage had originated from a crack in a Ta thermocouple well.

Capsule No. 4 was inserted on Dec. 11, 1961. The reactor was gradually brought to a power of 4.5 MW, at which time the capsule temperature was 1540° C. Within the next 24 hr. reactor power was

gradually increased to its normal level of 5.0 MW and the capsule temperature declined to 1510° C.

From the start of the irradiation until withdrawal of the capsule, the reactor operated at full power for approximately 1300 hr. During this time the capsule survived 5 scrams from full power in which the temperature dropped from ~1500° C to ~800° C in less than 1 min., and from ~800° C to 50° C in the next 5 min.

The range of the maximum recorded temperature during the irradiation was 1450 to 1650° C, and at least 90% of the time this range was between 1500 to 1600° C.

Open Type Capsule for Irradiation of Swaged and Sintered UO_2 Fuel Rods

Y. Seki, T. Kubota, Y. Honda and S. Takahashi
Engineering & Research Laboratory
Mitsubishi Atomic Power Industries, Inc.

For the irradiation of UO_2 fuel rods (swaged rods or pelleted rod), open type capsule was designed by MAPI. The capsule essentially consists of three rods to be compared with each other in irradiation behavior.

The attached figure shows the general arrangement. Each rod has a spring at the one end to compensate the thermal expansion of the rod. Each rod will have the similar neutron irradiation dose (flux x hrs.) which is desired for the comparison with each other, although the measurement of the neutron dose might have some uncertainty.

For the measurement of neutron dose, two methods are applied. One is the flux monitor wire which is Co-Al (0.5 mm ϕ) or Co wire (0.125 mm ϕ) sheathed by aluminum or stainless steel tubing and is assembled at the center line position. Another is Cs-137 analysis in small quantity of UO_2 powder which is put in a quartz ampoule.

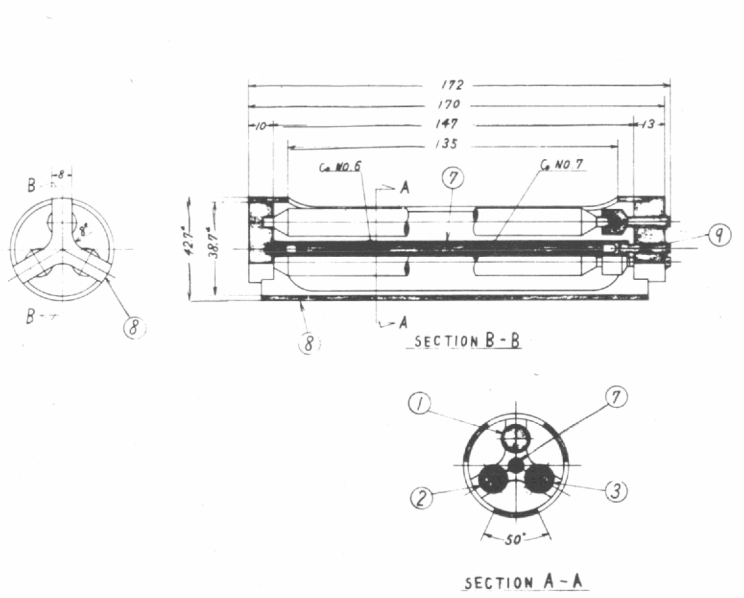
The ampoule is usually located at the end part of one fuel rod. The separate UO_2 powder in ampoule could eliminate the Cs-137 analysis error which might occur by the possible sublimation of Cs from UO_2 fuel rod. An example of the neutron flux measurement are shown in the Table.

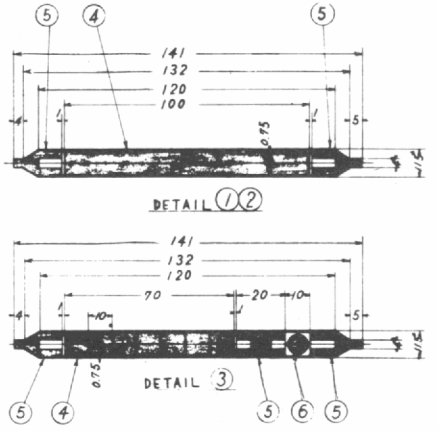
Table. Thermal neutron flux (Irradiation at WTR)

Section	Co-Al wire method	Cs-137 analysis method
A	2.58×10^{14} nv	
B	2.56×10^{14}	
C	2.47×10^{14}	
D	2.39×10^{14}	
E	2.31×10^{14}	
F	2.34×10^{14}	
G	2.45×10^{14}	correlated position 2.28×10^{14}
mean	2.44×10^{14}	
with Cd correction	2.20×10^{14}	

WTR staff suggested to have Cd correction of 5 to 10%. When 7% was taken as the correction, the neutron flux value measured by the Co-Al wire method was found coinsistent with the value from Cs-137 analysis.

This type of capsule were irradiated at WTR in 1961 and is going to be irradiated at BR-2 in this fall (Drawing NO. B0005). This is able to use for the fuel irradiation test with 13kw/ft thermal rating in usual material testing reactor coolant.





ITEM	DESCRIPTION & MATERIAL
1	SWAGED UO_2 FUEL ROD
2	SWAGED UO_2 FUEL ROD
3	PELLET TYPE UO_2 FUEL ROD
4	ZIRCALOY-2 TUBE
5	END CAP OF ZIRCALOY-2
6	UO_2 TYPE FLUX MONITOR
7	Co TYPE FLUX MONITOR IN ZIRCALOY-2 TUBE
8	FUEL RODS HOLDER OF ALUMINUM
9	SPRING OF STAINLESS STEEL

MITSUBISHI ATOMIC POWER IND., INC.	
TITLE	CAPSULE (B) FOR NATURAL URANIUM DIOXIDE FUEL RODS
DIMENSION IN MILLIMETER	
DWG NO. B0005	DATE MAY 30, 1963

Double Wall Type Capsule with Enclosed NaK for Irradiation of Enriched UO_2 Fuel Rods

Y. Honda, S. Takahashi, T. Kubota
and K. Ishikawa
Engineering & Research Laboratory
Mitsubishi Atomic Power Industries, Inc.

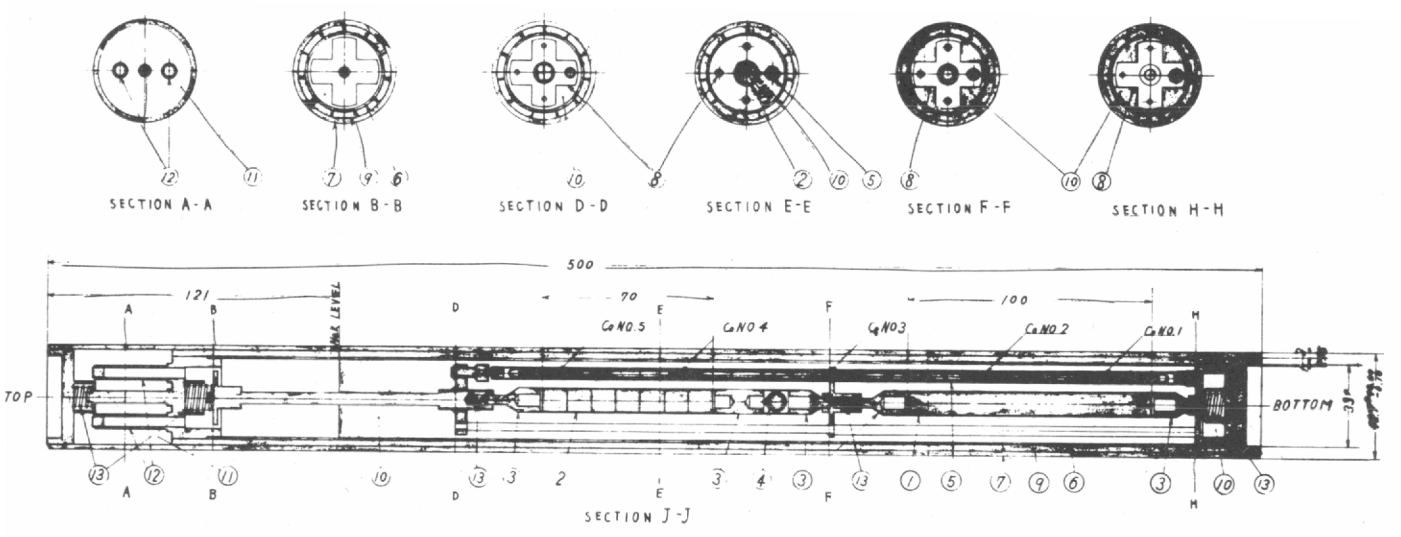
For the irradiation of swaged and pelleted UO_2 fuel rods (2.6% U-235) with considerably high thermal rating, double wall type capsule was designed by MAPI. The attached drawing (No. B0004) shows the general arrangement of the capsule including the two fuel rods. The capsule has inner and outer stainless steel shells to prevent an accidental blow out of NaK to reactor coolant. NaK will serve as a heat transfer medium between fuel rod and inner shell. Between inner and outer shells, there are several aluminum segments which are taken from an aluminum tubing having the outer diameter to fit the inner diameter of the outer shell and the inner diameter to fit the outer diameter of the inner shell.

The aluminum segments will be a good heat transfer medium when they contact tightly to shell surfaces at elevated temperature. Gaps of 1 mm between each segment will serve as a space for their thermal expansion. Each fuel rod is accompanied by a spring to compensate the thermal expansion and to absorb the shock when the capsule falls down. The inner shell has the top end plug having two small tubes for feeding and taking out of NaK.

For the flux monitor, Co wire method and Cs-137 analysis method are applied. A small quantity of UO_2 powder for Cs-137 analysis is sealed in a quartz ampoule located at the end part of pelleted fuel rod.

The capsule containing one swaged UO_2 rod (9 mm ø x 100 mm L UO_2) and one pelleted UO_2 rod is going to be irradiated in BR-2 with anticipated thermal rating of 15 KW/ft for 110 days. Predicted burn up is 10,000 MWD/T-UO_2. The NaK's temperature is calculated as 300°C and the outer surface temperature of capsule as 78°C when the reactor coolant temperature is 50°C and thermal neutron flux 1 x 10^{14} nv.

Furthermore, five of this type capsule are planned to be used for the fuel irradiation which is being conducted by the joint committee in Japan composed by two national laboratories and five main companies in atomic industry including MAPI. Each capsule will contain one fuel rod (8 - 10 mm ø, 100 mm L UO_2) including 5% enriched UO_2 compact. The planning burn up is 10,000 MWD/T-UO_2 to 15,000 MWD/T-UO_2.



ITEM	DESCRIPTION AND MATERIAL
1	SWAGED UO_2 FUEL ROD
2	PELLET TYPE UO_2 FUEL ROD
3	END CAP OF STAINLESS STEEL
4	UO_2 TYPE FLUX MONITOR
5	Co TYPE FLUX MONITOR
6	INNER SHELL OF STAINLESS STEEL
7	OUTER SHELL OF STAINLESS STEEL
8	FUEL RODS HOLDER OF STAINLESS STEEL
9	SPACER OF ALUMINUM
10	Na-K
11	He GAS
12	Na-K LOADING AND REMOVING TUBE
13	SPRING OF STAINLESS STEEL

MITUSBISHI ATOMIC POWER IND., INC.	
TITLE	CAPSULE(A) FOR 26% ENRICHED URANIUM DIOXIDE FUEL RODS
DIMENSIONS IN MILLIMETER	
DWG.NO. B0004	DATE MAY 25, 1963

2.4.2

TRANSIENT TEST CAPSULE DESIGN AND OPERATION

By

J. H. Field
J. H. Germer
M. J. McNelly

Work Performed Under

U. S. Atomic Energy Commission

Contract AT(04-3)-189
Project Agreement 10

2.5.1

TRANSIENT TEST CAPSULE - DESIGN AND OPERATION

A major unresolved problem in the development of large, ceramic-fueled fast reactors has been the determination of the transient behavior of such fuels during nuclear excursions. In 1962, a program of fuel capsule irradiations in the TREAT facility at NRTS was initiated to provide experimental data bearing on this problem. The first series of tests[1] evaluated the performance of 1.0" diameter, 3.9% enriched UO_2 fuel specimens, approximately 14 inches long, clad in 0.032" stainless steel. This specimen was installed within a transient test capsule and irradiated in the center position of the TREAT core.[2] The following is a brief description of the design philosophy and details of the test capsule used for these irradiations in TREAT:

Design Philosophy

The conceptual design of the test capsule was based on a need to satisfy several criteria:

1) In order to simulate radial temperature profiles expected in a ceramic fueled fast reactor,[3] the test capsule must be capable of rapidly dissipating the heat generated in the fuel. In addition, the capsule must contain a heat sink capable of absorbing all the heat generated in the fuel during a transient without rising to temperatures above those expected in the actual fast reactor.

2) The capsule must be sufficiently insulated from its surroundings to permit its use as a calorimeter. This makes possible an accurate estimate of the total heat input to the test specimen during a given transient.

3) The capsule should have capability to withstand internal pressures generated in the course of planned experiments or in a possible accident situation.[4] The capsule must further provide containment of fission products as well as sodium or other materials which might cause chemical reactions in the TREAT core.

4) The capsule must fit within the rectangular section of a dummy TREAT fuel element in order to be placed within the TREAT core for testing.

Design Details

The test capsule design was shown in Figure A. The exterior of the capsule was dimensionally the same as a TREAT fuel element with the exception that the tubular section was cylindrical instead of square, and of a diameter equal to or less than the distance across flats of a square TREAT fuel element. The upper and lower ends remained square in order to provide alignment with the surrounding fuel elements in the core. Thermocouple and heater connections at the upper end were made after loading of the capsule into the reactor by means of an aluminum holder of a type that is presently being used by TREAT for special instrumented fuel elements.

Within the outer dummy fuel element shell an inner container of 1/8" thick 304 SS was provided. This containment served as the boundary for the calorimeter and a 5/16" layer of thermoflex insulation was placed between the inner capsule and outer shell to minimize the loss of heat while approaching equilibrium temperature.

The fuel element was surrounded by an annular beryllium heat sink capable of absorbing all the heat generated in the fuel during the transient without approaching temperature levels in excess of those expected in an operating fast reactor.

In order to provide rapid heat transfer from the fuel pin to the heat sink, and to duplicate the proper coolant environment, a thermal bond of sodium was needed between the clad and the beryllium annulus. Thus, following assembly and insertion of the fuel specimens, the inner capsule was heated and approximately 250 cc of sodium introduced through a fill-tube. This tube was then sealed and the inner capsule pressurized with ten atmospheres ($\sim$350 psi at 800°F) of helium to provide additional insurance against gross sodium boiling and void formation at elevated temperatures. The assembly was mass-spectrometer leak tested before final insertion into the dummy TREAT assembly.

Instrumentation

The instrumentation for the first test assembly consisted of one platinum/platinum-10% rhodium thermocouple located in the fuel pin and ten chromel-alumel thermocouples located as follows:

a) Two in the sodium-filled annulus between the fuel element and the heat sink.

b) Five on the outer surface of the heat sink, distributed over the circumference.

c) One on the inner surface of the outer container, opposite the heat sink.

d) One in the aluminum upper-end fitting of the assembly.

e) One on the tube extension of the inner capsule.

Connection to these thermocouples, plus the internal heater leads, was made via a Deutsch plug and receptacle (DM 5606-37 or DM 9702-37) using a plug holder designed to accommodate TREAT handling tools.

Each assembly was equipped with an internal circuit capable of maintaining the inner capsule at a preheat temperature of 800°F. An off-on temperature sensing circuit was used in conjunction with the heaters to provide remote automatic control. This control circuit included a 2000 watt Variac, indicating pyrometer, circuit breaker, and indicating meters. Temperature control was accomplished using one of the thermocouples located on the surface of the inner capsule.

Operation

Each test capsule arrived at the TREAT site in the fully assembled condition and was installed in the center position of the TREAT core in accordance with normal procedures for dummy fuel element handling. Instrumentation and heater connections were made at the top of the capsule in the usual manner for instrumented fuel elements and, with the capsule in position, a check of heater and thermocouple resistance and continuity was made.

When capsule installation, connection, and checkout were completed, the heater circuit was energized and the Variac output set at approximately 250 watts. This setting was maintained until the indicated capsule temperature approached 250°F (sodium completely molten). At this point the control circuit was checked and heater power increased to bring the capsule temperature to the preheat setting (800°F). With the control on "automatic," the heater power was then set at a point slightly greater than that required to stabilize the capsule temperature at 800°F. Provision was made to interrupt the heater circuit power supply from the control building just prior to the transient.

Following execution of the experiment, the capsule was allowed to cool to ambient temperature before unloading. After disconnecting the thermocouple and heater leads, the dummy fuel element assembly was removed from the core into the TREAT fuel transfer coffin. When necessary, the test assembly was stored in the TREAT fuel-storage area until the dose rate at a distance of 4" fell below 20 r/hr gamma. When this requirement was met, the capsule was transferred to the shipping cask and returned to the Vallecitos RML facility for disassembly.

Results

Since the test capsule was designed to act as a calorimeter, determination of sample power depended upon accurate measurement of the "equilibrium" temperature of the insulated assembly following a given transient test. In the three preliminary transients on the first capsule, it was found that the fuel, sodium, and capsule wall reached essentially the same temperature approximately 100 seconds after the power pulse. Available in-pile cooling curves indicated that the capsule lost only about $4^{\circ}F$/min. when the inner capsule wall was at $800^{\circ}F$. This loss was within the accuracy of the equilibrium temperature determination at 100 seconds.

In order to arrive at the best estimate of equilibrium temperature, the temperature rise at 100 seconds after peak power of five thermocouples was considered for each transient:

TC's 1 and 2 - Sodium temp. (recorded on oscillographs)

TC's 4 and 5 - Inner capsule outer wall (oscillographs)

TC 7 - Inner capsule outer wall (heater control couple-Bristol chart recorder)

From an average of these readings and a capsule heat capacity based on the average capsule temperature during each transient, sample power values were calculated. Capsule heat capacities were derived from measured heating and cooling rates during pre-irradiation calibration and were in good agreement with values calculated previously.

In general, the capsule performed as planned and the design criteria were satisfied. Results indicated that the desired conditions of fuel and cladding temperatures were achieved and the objectives of the test series successfully met. Detailed results of the tests and related capsule experience are covered in some detail in Reference (1).

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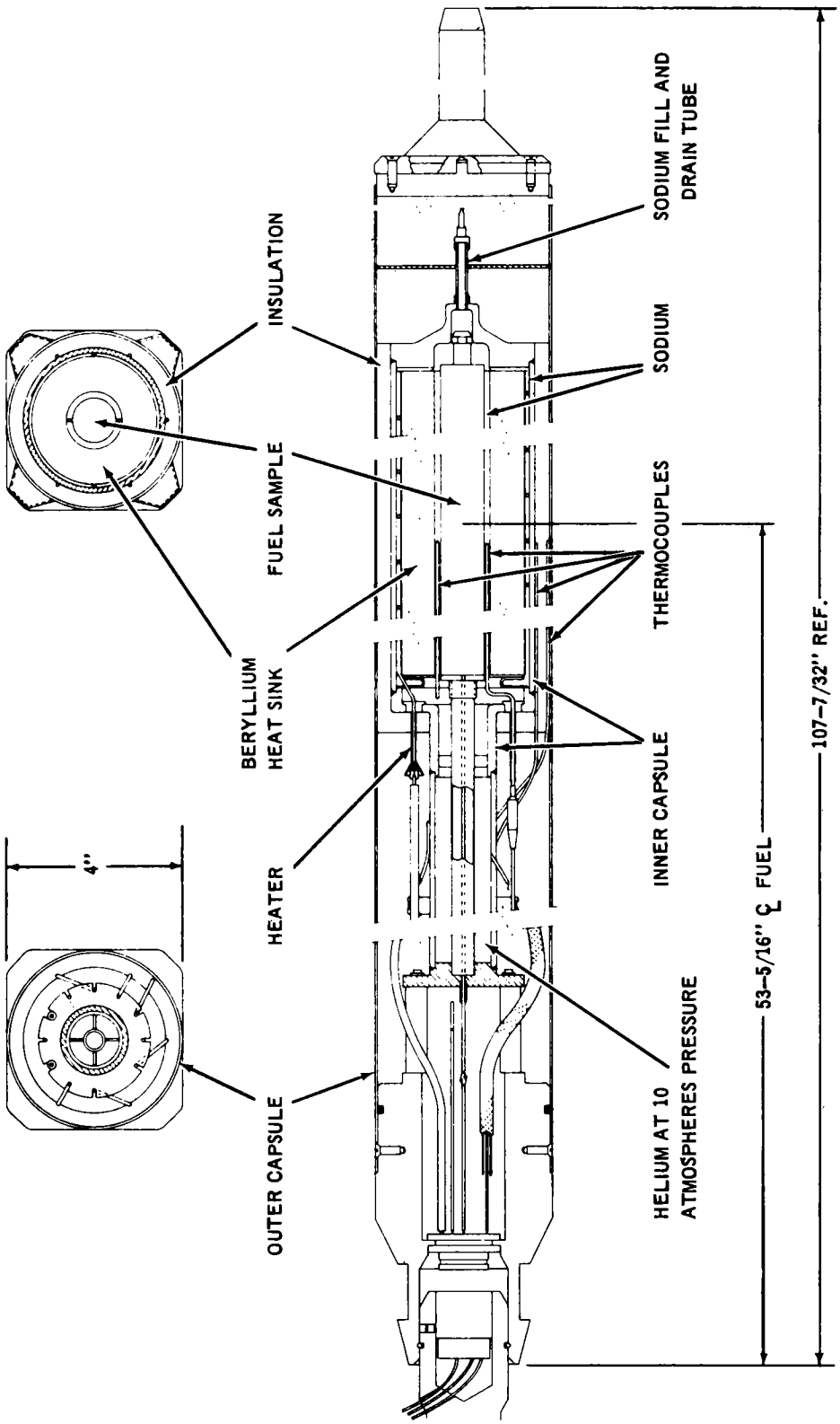

FIGURE A TREAT CAPSULE

Fissile Irradiation Capsules Used at A.E.R.E.

by

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A.E.R.E. HARWELL

Abstract

A considerable variety of capsules have been designed and used at A.E.R.E. to cover the varying needs of fissile irradiation experiments. In all cases the designs had to meet the basic safety requirements of double containment, temperature monitoring, and in those cases where sodium was used, detection of sodium leakage in addition to the experimental requirements.

This paper describes a selection of these capsules in which irradiation temperatures up to 1400°C have been achieved over extended periods.

List of Illustrations

1. Introduction

The majority of fissile irradiations performed in the research reactors BEPO, PLUTO and DIDO at A.E.R.E., Harwell require control of specimen temperature and measurement of fuel heat output. Safety requirements dictate the use of double containment, whilst heat transfer considerations make it necessary to use a liquid metal as the initial heat transfer medium. The standard capsules developed to fulfill these requirements have already been described by Plail[1] and Hancock[2]. This paper deals therefore with those changes made necessary by some special requirement or completely different approach.

2. Capsule designs

2.1 Power cycling of UO_2 (Fig. 1)

Two unusual requirements dictated the development of this design, these being (a) the maintenance of 300 p.s.i.g. pressure on the specimen during irradiation, and (b) cyclic variation by a factor of 10 of the incident neutron flux without variation of specimen surface temperature.

The sheathed UO_2 pellets (overall length 10", diameter 0.5") were centrally located and supported by stainless steel end spacers in the primary containment capsule. This capsule was of 18.8.1 (type 304) stainless steel. Its upper end cap was penetrated by and edge welded to three re-entrant pockets for thermocouples and the sheath of the 3 kW MgO insulated 18.8.1 stainless steel sheathed nichrome heater. To reduce heat generation at the point of exit from this capsule the heater was swaged so that the coiled section had a cable diameter of 1/8", while the tails remained ¼" diameter. After assembly of the primary capsule it was filled to a calculated level with re-distilled sodium via a small hole in the upper end cap. When the capsule had cooled to room temperature it was flooded with argon and the hole temporarily closed with a rubber bung during transfer to a welding chamber, where the argon was evacuated, the capsule filled with helium at the correct pressure and the hole sealed by a taper pin.

Between the primary and secondary capsule was an aluminum sleeve, its internal diameter being machined to give the requisite gas gap between itself and the primary capsule. The secondary capsule was also of 18.8.1 stainless steel. The lower void space in this capsule contained two MgO insulated stainless steel sheathed sodium leak detector

probes, these passing up through slots in the aluminium sleeve and thence through the upper end cap, to which the sheaths were brazed. Into this end cap were also welded two Ferranti ceramic/metal seals through which passed the heater tails, and a purge tube for controlling the pressure in the secondary capsule.

Three major problems had to be faced in this design. These were:

(i) The longitudinal temperature gradients,
(ii) The pressure vessel design,
(iii) Heat removal from the cadmium shutter.

Since the aspect ratio and heat rating of the fuel were both basic experimental requirements, and the space available in the reactor was limited, only two capsules could be accommodated, the capsule length being only marginally sufficient for the required fuel length. It was therefore anticipated that some longitudinal temperature gradients would exist even if the flux was uniform along the specimen, but that, since the heater could not be lengthened nor the fuel length decreased, these would have to be accepted. The gradients obtained in service are shown in Fig. 2, from which it can be seen that these were not excessive.

Considerable difficulty was experienced in designing the pressurized capsule. A design based on the A.S.M.E. code would have resulted in such a wall thickness as to reduce unacceptably the neutron flux on the specimen. Furthermore, since the pressure in the capsule was the result of combined sodium and helium blanket expansion, and the mean temperature of the sodium was not accurately known, it was rather difficult to specify precisely. Thus the capsule design could not be made in accordance with the A.S.M.E. code, but several capsules were made and tested to establish what working pressure could reasonably be anticipated. These tests were carried out at room temperature and test pressures of 2000 p.s.i.g. were achieved giving a working pressure at 600°C of 300 p.s.i.g. Since it was, nowever, a safety requirement that a containment be provided that would satisfy the A.S.M.E. code, the outer capsule was also designed to this code but to less stringent requirements. Its operating temperature was 100°C and the internal pressure it was required to sustain in the event of failure of the inner capsule was only ~50 p.s.i.g.

Calculation of the anticipated heat output of the cadmium shutter by several different methods revealed an unresolvable uncertainty, calculated outputs varying between 5 and 25 kW. The original concpetion of the shutter as a float which could be raised and lowered by alteration in level of the water in an annular tank was therefore changed. In the final design the shutter was mechanically raised and lowered whilst immersed in an annular cooling jacket.

Operation of the capsules was entirely satisfactory, temperature control being maintained for a continuous period of ~4 months irrespective of reactor power. Estimation of the heat output of the capsule during irradiation from pre-irradiation calibration curves provided estimates of fission heat and burn-up which compared well with estimates made both from flux monitors and from chemical analysis of the fuel.

2.2 Thermal conductivity of UO_2 (Fig. 3)

The capsules for these experiments were designed specifically to measure as accurately as possible both the temperature distribution across a UO_2 fuel element and the heat output of the fuel.

The fuel pellets were contained in an 18.8.1 stainless steel primary capsule sealed at either end by end caps brazed to the capsule wall with Coast 50 alloy. Passing through these end caps, and similarly brazed, were five tantalum sheathed W 5% Re/ W 26% Re thermocouples insulated with BeO. Two of the thermocouples were sited in line on the axis of the fuel while the remaining three were equi-spaced on a common P.C.D. and on the same horizontal plane as the axial thermocouples.

Surrounding the primary capsule was a heater formed from a Zircaloy II former, mineral insulated heating cable and a Zircaloy II sleeve. Grooves turned in the former accepted the cable, which was retained in position by the sleeve. The manufacturing technique was such that reasonably good thermal contact between cable sheath and former was obtained. Between the heater and the outer containment was an aluminium sleeve, its I.D. being bored to give the requisite gas gap.

Conventional Chromel/Alumel thermocouples were inserted in both the heater sleeve and the aluminium sleeve, the hot junctions again being sited on the same horizontal plane as those in the fuel. Thus a relationship between heat produced by the heater and the consequent temperature gradient between these sleeves could be established. Since this relationship would also apply were the heat produced by the specimen, its heat output could be determined at any time during the irradiation experiment by a simple measurement of the sleeve temperatures.

Knowing the heat output of the specimen, and measuring the radial temperature distribution by means of the embedded high temperature thermocouples the thermal conductivity of the fuel could be calculated.

Five capsules were successfully operated for a period of 2 months, at temperatures up to 1400°C, withdrawal of the experiment then being necessary due to loss of the high temperature thermocouples. A replacement rig containing a further five capsules has now been operating for 4 months at ~700°C to obtain thermal conductivity data at these temperatures.

2.3 Creep of uranium carbide (Fig. 4)

Interest in this fuel has been steadily increasing and a knowledge of the effects of irradiation on its creep strength would be desirable. The capsule has been designed to accommodate a specimen 1½ in. long x ¼ in. diameter, operating at 1000°C, the operating temperature being maintained irrespective of reactor power. Measurement of creep strain will be performed by a micrometer operated from the reactor top via an extended shaft. A similar method has been used previously and proved eminently successful, strain being consistently and reproducibly read to 10^{-5} ins over a period of ~6 months in a thermal neutron flux of 4×10^{13} n/cm^2 sec.

The capsule consists of an 18.8.1 Ti. stainless steel containment incorporating a stainless steel bellows. The upper and lower end caps of this secure the ends of the alumina pull rods which apply stress to the specimen via conical seatings. Some adjustment of alignment of the pull rods is given by hemispherical protrusions abutting conical seatings on the containment end caps. A 3 kW heater is located inside the containment and will operate up to 1100°C to give control of specimen temperature at all times. To avoid the inherent difficulties in the operation of sodium at 1000°C, heat transfer from the specimen is via a small helium filled annulus to an alumina sleeve, thence via a second annulus to the heater.

The stress is applied via the upper end cap of the containment, a stressing tube being screwed to this and continued to the rig head above a nut and spring give-load adjustment. The lower end cap abuts aluminium alloy sleeves which form part of the rig structure. Since the operating temperature of these sleeves is ~100°C ample strength is available to take the applied load.

Second containment of the fuel is provided by the aluminium envelope, which is a tight fit on the aluminium alloy sleeves, and is terminated at its lower end with a welded cap. Heat produced by both specimen and heater is removed via these sleeves and the envelope to the reactor heavy water moderator.

2.4 Fissile diode capsule (Fig. 5)

In order to obtain useful current from an in-pile diode, the fissile cathode is required to operate at ~2000°C. Although success has been achieved over restricted periods, development of a diode capable of a life of the order of one year necessitates more knowledge of the occurrence and extent of swelling of the cathode combined with knowledge of its ability to retain fission products.

The capsule shown consists essentially of three containments, the outer and intermediate of which are of 18.8.1 Ti stainless steel. Heat transfer between these is via a helium annulus. The inner containment is of tantalum, to the internal surface of which has been applied by vapour deposition a coating of tungsten. End caps of tantalum are brazed to the containment body with a Coast alloy braze. Heat transfer to this containment is by radiation, the small conductive component being accommodated by a small pressure of helium.

An attempt to measure the centre temperature of the fuel is being made, using W 5% Re/ W 26% Re insulated with Al_2O_3 and sheathed in molybdenum and clad with niobium. A tungsten re-entrant pocket locates this thermocouple. It is hoped to obtain multi-core thermocouple cable for this use so that several junctions may be sited along the axis of the specimen.

Since it is not anticipated that these thermocouples will have a life in excess of a few days at 1600-2000°C a further thermocouple is sited in the upper tantalum end cap. The relationship established between the temperature indicated by this and those indicated by the centre temperature thermocouples will be very approximately constant during the life of the capsule, assuming firstly that no gross changes in dimension of the specimen occur and secondly that the calibration of the thermocouple is not largely influenced by irradiation. Further thermocouples measure the temperature of the tantalum containment wall, these serving primarily to prove that the containment is operating at a safe temperature. A secondary use is that of a specimen position indicator, so that if, through thermal stresses or other cause a large part of the specimen becomes detached from the upper end cap and falls to the bottom of the containment, information of this is given by a sudden increase in the readings obtained from the lower thermocouple.

2.5 Capsules for coated particle irradiations (Fig. 6 and 7)

The fission product retentivity of particles of carbide fuel coated with "impermeable" graphite are of considerable interest, irradiations to 1600°C being necessary. Because of the peculiar nature of the fuel and the high temperatures involved, the capsule designs were quite different to those normally used.

In the Mk. I capsule, two containments of stainless steel were used, an aluminium sleeve being fitted between these and bored out to give the required helium gas annulus. Inside the inner containment was a graphite pot with graphite plugs at each end and contained by the pot was a graphite crucible. Longitudinal holes in the crucible drilled on a P.C.D. accepted a

stack of single particles, each particle being ~.020" diameter. Conventional Chromel/Alumel thermocouples sheathed in stainless steel and insulated with MgO monitored the temperature of the aluminium sleeve and graphite pot.

Although no direct measurement of particle temperature was possible, calculation based on the observed sleeve and pot temperatures indicated that ~1400°C had been attained.

Two major problems were met in the design, both being relatively simple. The main metallurgical problem was the long term compatibility of graphite and stainless steel at ~800°C. This was overcome by depositing a thin coating of nickel on the interior of the inner containment. The second problem was that of evolution of gaseous impurities from the graphite at high temperatures. Overcoming this meant the development of a small graphite degassing and capsule loading line in which the graphite could be degassed at ~1000°C, components out to and including the inner containment being subsequently assembled under dry helium.

The Mk. II design was developed to increase the chance of obtaining data on fission gas release from the particles. It consisted of pairs of geometrically identical capsules, the end cap of only one capsule of each pair being penetrated by a sheathed thermocouple. Thus failure of the sheath through embrittlement would cause loss of fission product gases only from one capsule; the other capsule of the pair would be unaffected. Since the geometry of each capsule was identical, and fissile loadings in each capsule were also identical, the temperature of the unmonitored capsule could be closely inferred from that of the monitored one.

2.6 High temperature capsule for UO_2/BeO fuels (Fig. 8)

The requirement here was for a design that would allow irradiation of this fuel at surface temperatures of the order of 1400-1600°C with some measure of temperature control.

It was considered that development of a reliable heater to operate at these temperatures was likely to be an extremely long term project. Since the fuel heat rating was fairly high (~3 kW from a specimen 3" long and 0.5" dia.) the thermal conductivity of the materials surrounding the fuel had to be high.

The capsule consisted of an inner stainless steel containment in which the fuel was located such that a small helium gap existed between it and the containment wall. This containment also formed the inner wall

of an annular sodium filled capsule in which was fitted an immersion heater and Chromel/Alumel stainless steel sheathed thermocouples. Enclosing this capsule was a further stainless steel containment capsule, primarily to contain any sodium which might leak from the main sodium capsule in the event of failure.

Temperature measurement of fuel centre was performed by W 5% Re/W 26% Re thermocouples insulated with BeO and sheathed in tantalum.

2.7 Capsule for determination of film Δt at sodium/specimen interfaces (Fig. 9)

An increasing requirement for irradiation experiments on a variety of fuels having surface heat ratings of up to 400 watts/cm^2 and in which it is required that restraint of the fuel must be minimal has led to considerable effort being devoted to the determination of temperature gradients at the specimen/sodium interface.

Little work has been published on the subject, that which has being limited to lower temperatures and ratings than those of interest.

To investigate this with any precision it is necessary to have a controllable heat source which will cover the range of surface ratings 100-500 watts/cm^2. Development of an electrical heater for this was considered to be a difficult problem so it was decided to utilize fission heat which was varied by varying the power of the reactor.

The capsule consists of the normal two stainless steel containments, the inner of which is sodium filled and contains a 3.5 kW immersion heater, the fissile specimen and Chromel/Alumel stainless steel sheathed thermocouples. The thermocouples are so sited that the temperatures of the specimen surface and bulk sodium at varying distances from the specimen are measured. The number of thermocouples used precludes the use of the normal system of stainless steel pockets; each has to be brazed to the upper end cap of the sodium containment using a sodium resistant braze. Uranium and stainless steel are incompatible above $\sim 700^{\circ}$C due to the formation of eutectics and thus thermocouples in contact with the specimen are further clad with a vapour deposited coating of tungsten.

Since the occurrence of film boiling conditions, with consequent melting of the specimen, is a very real risk, a tantalum catch pot is incorporated in the sodium capsule. In addition to its improved compatibility with uranium, the higher neutron absorption cross-section of tantalum decreases the flux on the fuel in the catch pot.

2.8 Capsule for UO_2 evaporation experiment (Fig. 10)

The object of the experiment was to investigate the evaporation of uranium and fission products from a uranium or UO_2 specimen while undergoing irradiation at temperatures up to 600°C. Since high burn-up was not required it was possible to use a 4 inch square air cooled channel in the BEPO reactor, in which the flux was $\sim 1.4 \times 10^{12}$ n/cm^2 sec (thermal). The specimen which is a disc of uranium or UO_2 ½" diameter x 1/16" thick is sealed in an evacuated copper can by pressure welding. This can also contains a collector foil located 0.050" above the specimen. This foil has to be kept vertically above the specimen during assembly and at all times until the foil is recovered at the conclusion of the experiment. The maximum fission heat produced by the specimen in a flux of 1.4×10^{12} n/cm^2 sec is less than 20 watts. This low heat output made it possible to design a simple, cheap and expendable rig utilizing the reactor cooling air flow to remove heat from the rig.

The main body of the rig, (Fig. 1) which provides the secondary containment, consists of a tubular (finned) aluminium can, approximately 3½ in. diameter x 6 in. long, mounted on two skids. The skids enable the rig to be pushed along reactor channel and also keep the rig in the centre of the hole and maintain the specimen and collector foil in the correct plane.

The copper specimen capsule fits in an aluminium heat transfer block which is supported by a thin stainless steel tube and is mounted centrally in the rig body leaving a 0.100 inch annular gas gap. Mineral insulated, stainless steel sheathed heater cable, 2 mm diameter, is passed through longitudinally drilled holes in the block to form a furnace. A re-entrant thermocouple pocket, terminating by the specimen capsule, carries two thermocouples for temperature monitoring and furnace control.

The heater leads pass through ceramic to metal seals which are brazed to a stainless steel disc which is in turn brazed to the end cap of the rig body. A tube is fitted to the other end cap for vacuum testing and gas filling. All leads external to the rig body are insulated with woven glass sleeving.

During the recovery process any vibration of the capsule had to be avoided; this ruled out all the conventional de-canning methods and a special method was devised. A circular cap is welded into the rig body vertically above the specimen capsule. This cap is provided with a weakened ring and a locating spigot, enabling the use of a hydraulic punch to remove the centre of the cap. A similar cap is fitted to the specimen capsule so that a smaller punch can be used to obtain access to the inside of the capsule. The collector foil can then be removed with remote handling tongs.

Acknowledgements

The designs described in this paper were evolved by a joint effort of staff in the Metallurgy, Engineering and Research Reactor Divisions of A.E.R.E. Their use would have been impossible but for the cooperation and ingenuity of the staff of the Rig Workshop of Metallurgy Division.

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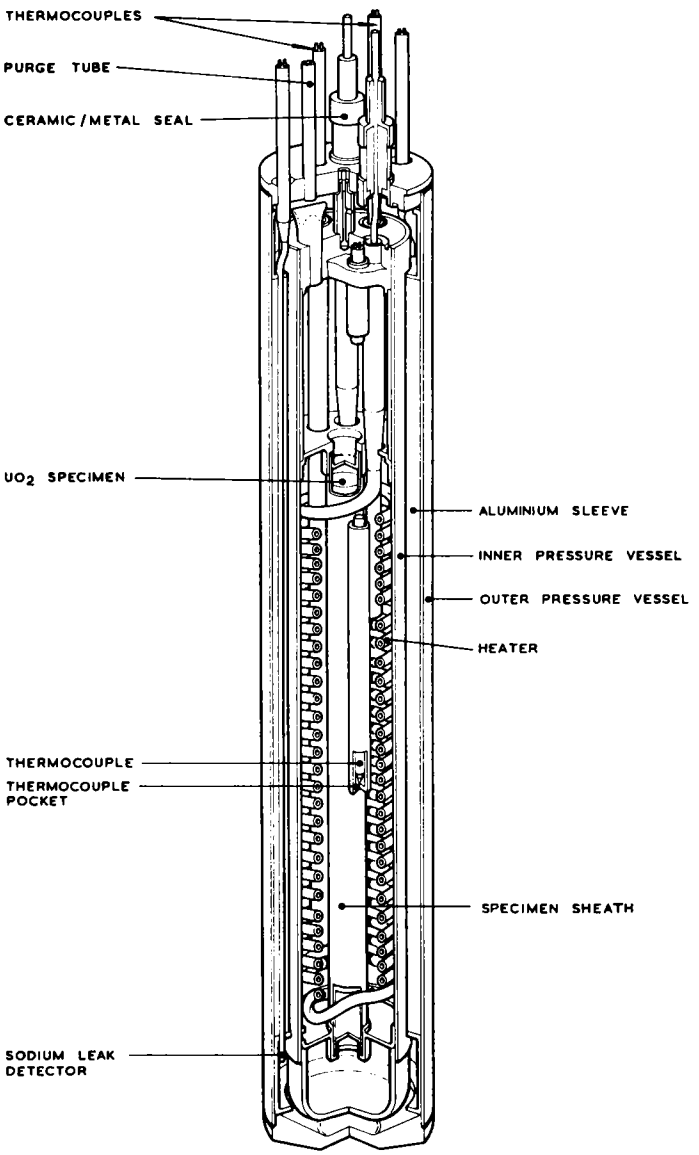


Capsule For Power Cycling of UO_2 Experiment.

Figure 1

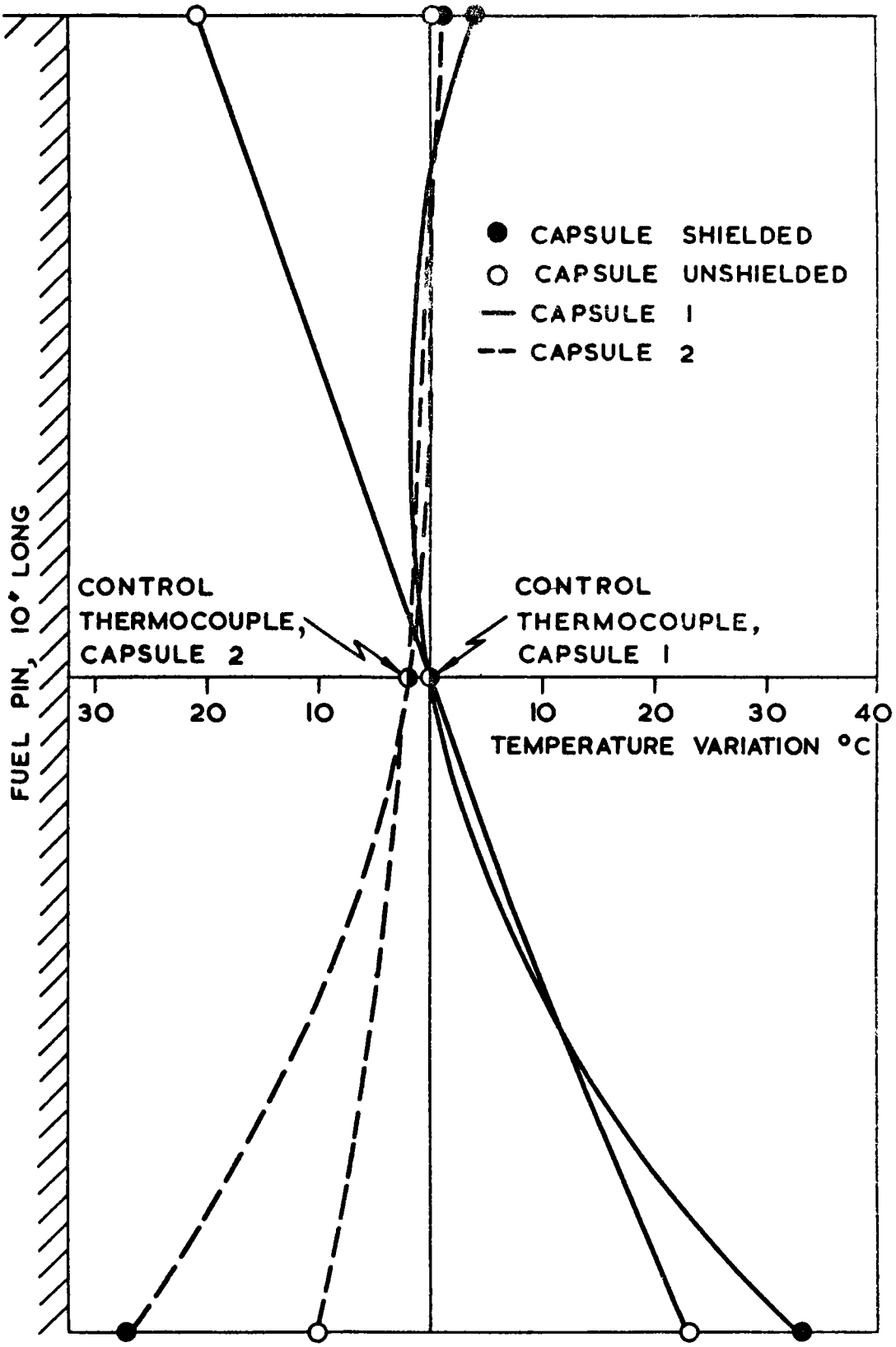


FIG. 2. THERMAL GRADIENTS IN POWER CYCLING CAPSULE

Figure 3

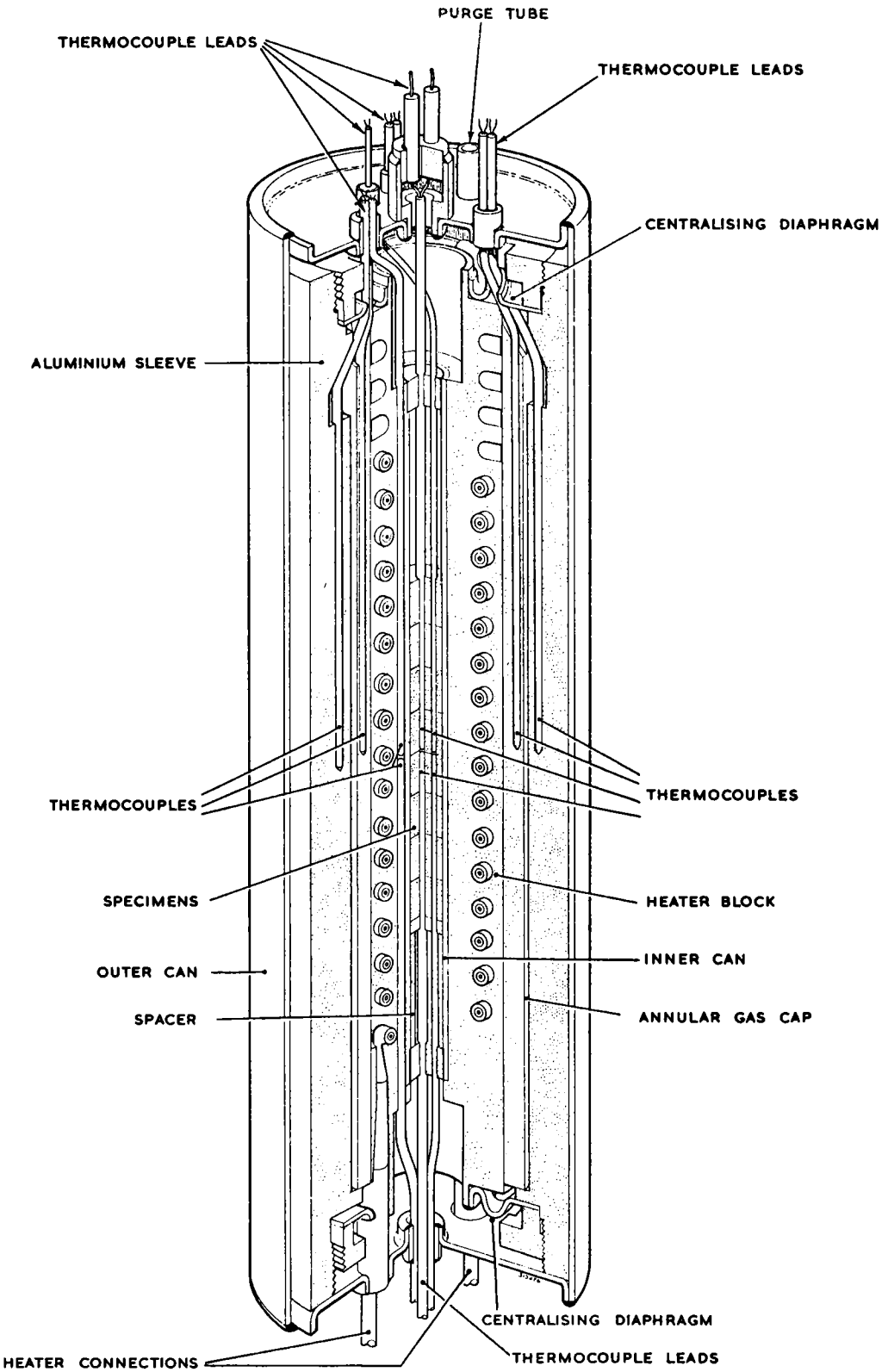
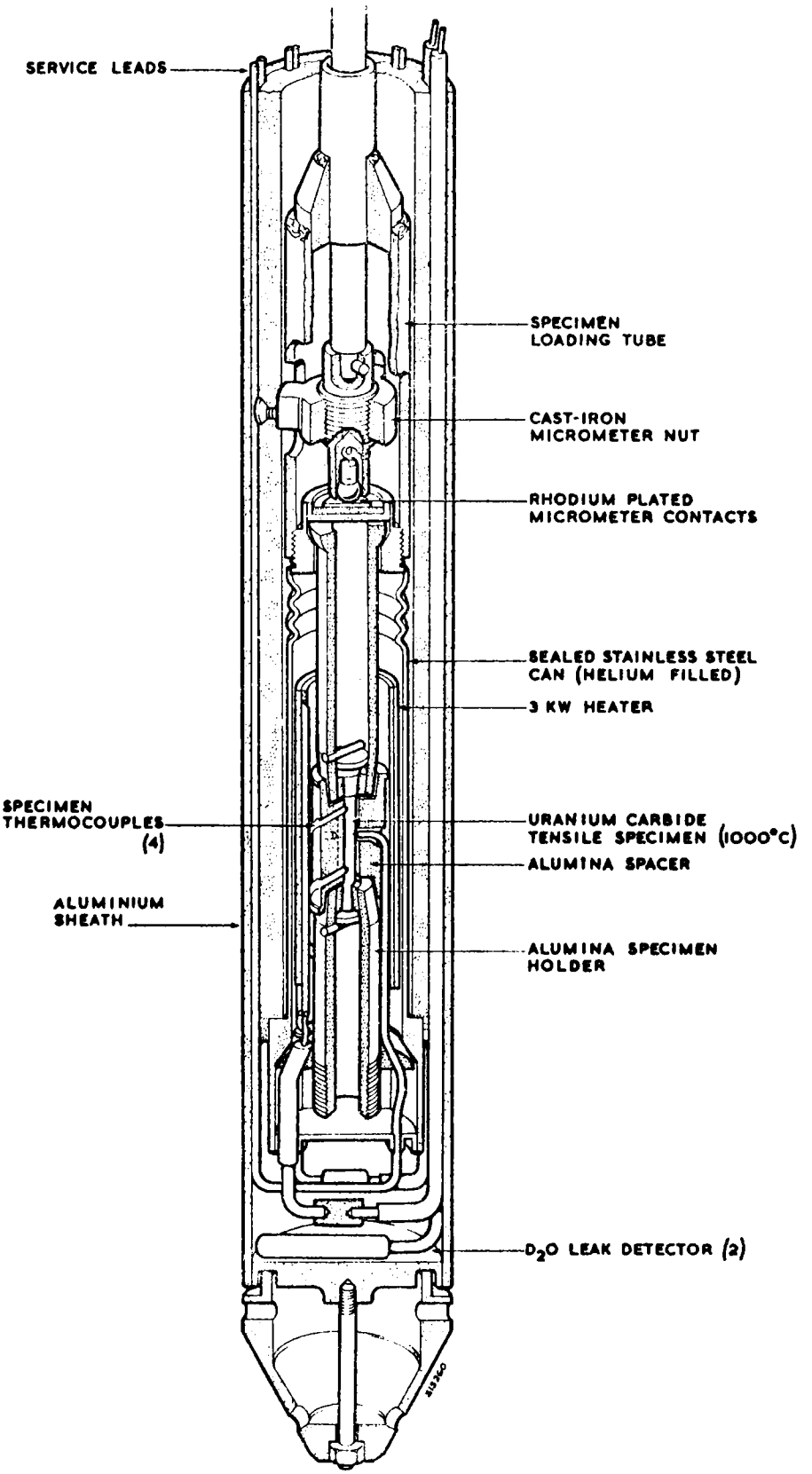
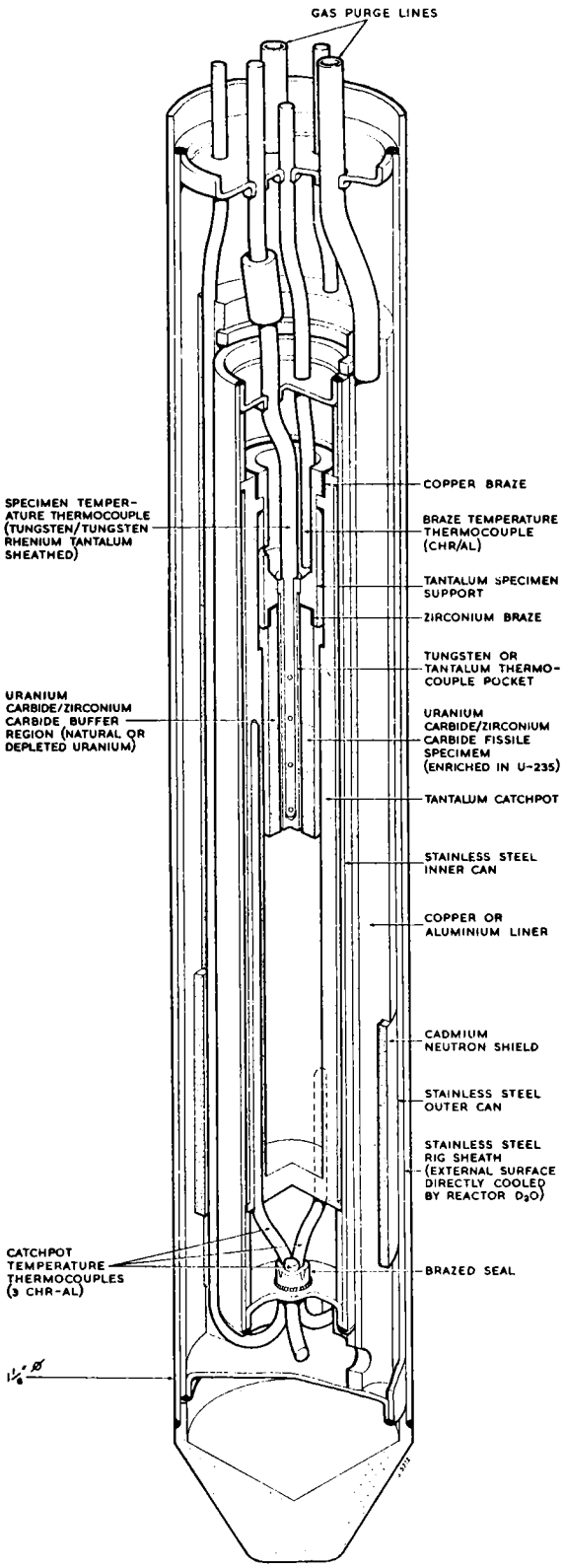

PURGE TUBE
THERMOCOUPLE LEADS
THERMOCOUPLE LEADS
CENTRALISING DIAPHRAGM
ALUMINIUM SLEEVE
THERMOCOUPLES
THERMOCOUPLES
SPECIMENS
HEATER BLOCK
OUTER CAN
INNER CAN
SPACER
ANNULAR GAS CAP
CENTRALISING DIAPHRAGM
HEATER CONNECTIONS
THERMOCOUPLE LEADS

Figure 4

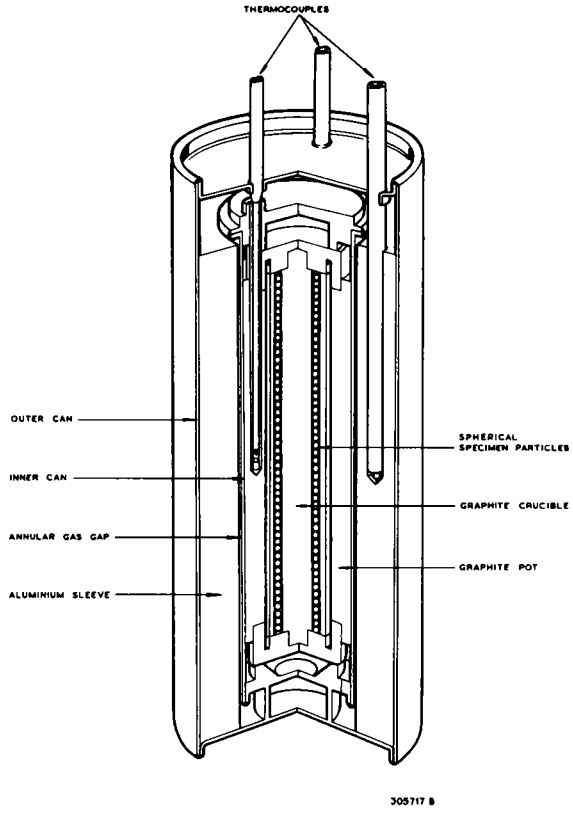


Capsule for Uranium Carbide Tensile Creep Specimen

Figure 5

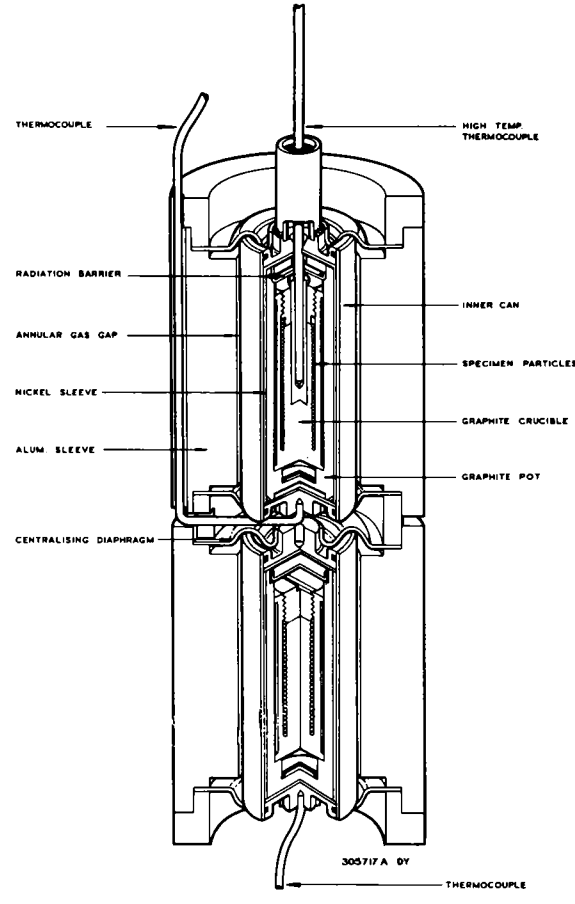


Capsule For High Temperature
Diode Cathode Irradiations



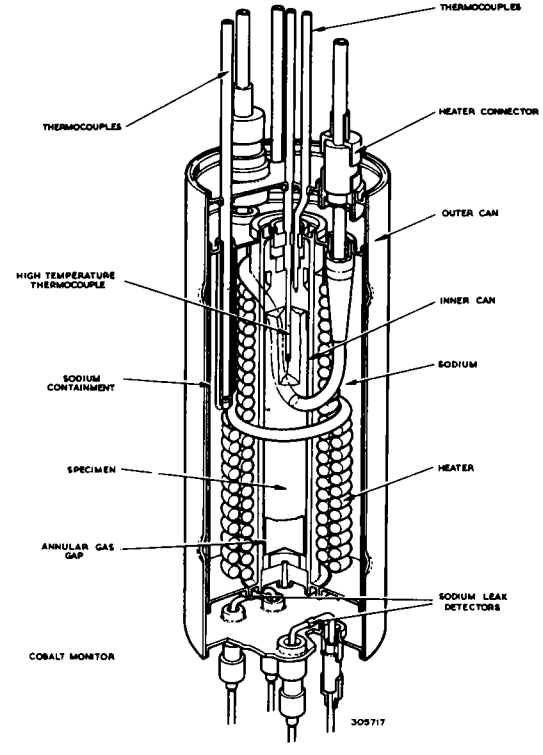
High Temperature Coated Fissile Particles Mk. I

Figure 6



High Temperature Coated Fissile Particles Mk II

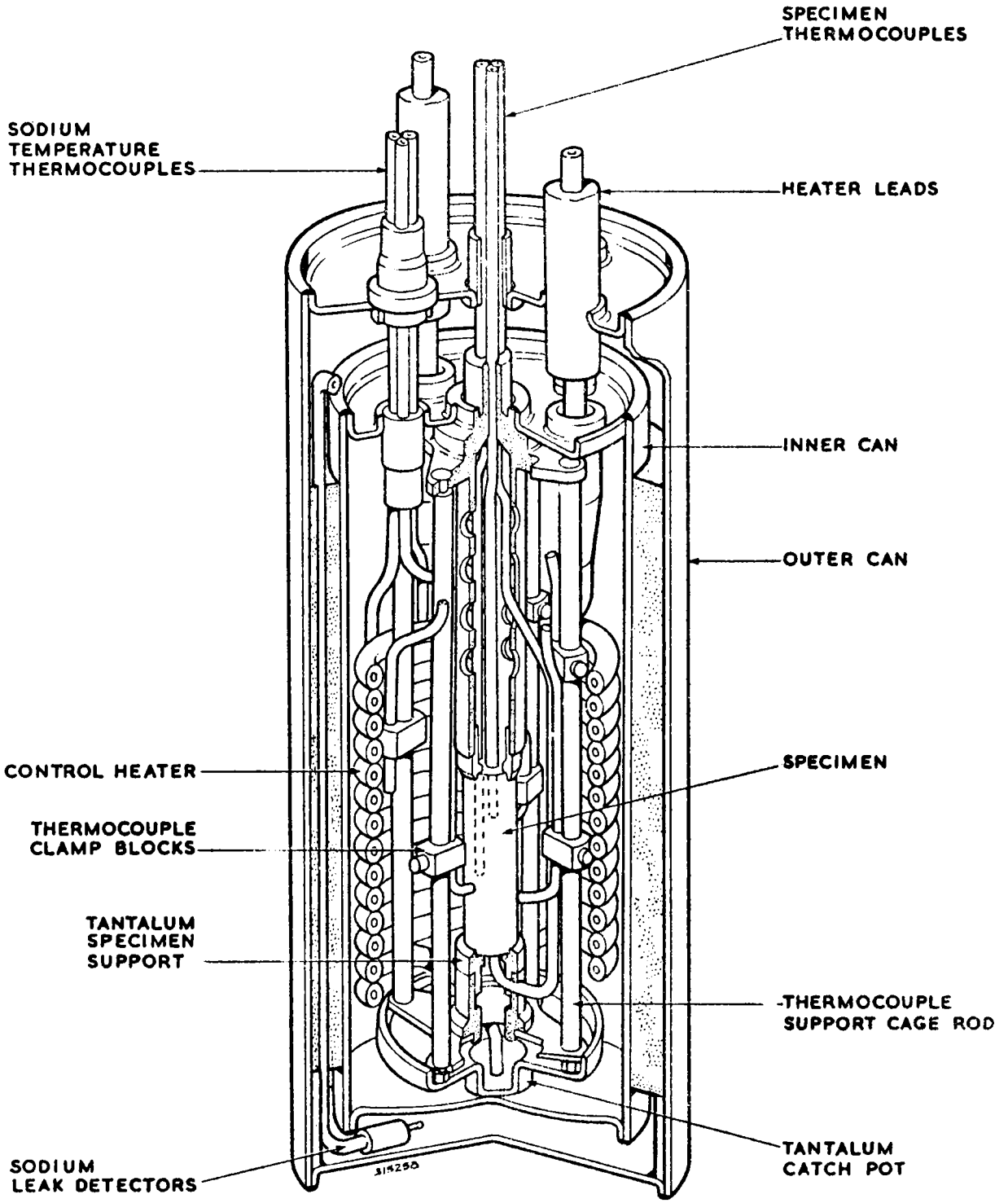
Figure 7



High Temperature Ceramic Fuels.

Figure 8.

Figure 9



2.6.19

Sodium Film Gradient Capsule

BOILING WATER IRRADIATION CAPSULE POWER CYCLING TESTS

by

T. C. Rowland

Work performed under

General Electric Company

research and development authorization

BOILING WATER IRRADIATION CAPSULE

Experimental equipment has been designed and put into operation in the General Electric Test Reactor Trail Cable facility for short term irradiation testing of 8-inch long, 0.25- to 0.500-inch OD fuel specimens at heat fluxes up to 800,000 Btu/hr-ft^2. The fuel is cooled by 1000 psig subcooled boiling water.

This facility has been used for accelerated power cycling tests to determine the resistance of thin wall fuels to cyclic fatigue. This equipment has also proved to be very useful for short term (ten minutes to one hour) tests to determine clad-fuel interactions and thermal effects on UO_2.

GETR Trail Cable Facility

The Trail Cable facility was selected for the power cycling experiments because it provides satisfactory neutron fluxes and permits direct access to the reactor flux zone while the reactor is operating.

The facility is a dead end tube extending from a 45° penetration on the third floor of the reactor to three inches below the bottom of the reactor core. The tube is in the pool of the reactor and runs vertically along side the reactor vessel from three inches below the bottom of the core to several feet above the top of the core. The tube makes three six-foot radius bends between the top of the core and the penetration on the third floor. The radius of these bends and the diameter of this tube limit the length of the capsule which may be irradiated in the facility and consequently, the length of fuel which may be irradiated. For the power cycling experiments, the fuel length is limited to 8 inches.

The average thermal neutron flux in this location is about 7.5×10^{13}. The peak-to-average flux ratio varies from about 1.8 to 1.4 depending on the control rod bank. By proper selection of fuel enrichment and controlled fuel insertion depths a wide range of surface heat fluxes are attainable.

Capsule

The capsule is shown in Figure 1. There are four main components to the capsule. These are: (1) the pressure jacket; (2) the two pressure lines; (3) the cooling jacket; (4) cooling water line.

The pressure jacket provides these functions: it is a means of holding the fuel specimen; it contains the 1000 psi coolant water pressure around the fuel specimen; and it is an insulation from the low temperature reactor pool water.

The fuel is cooled by natural circulation of the 1000 psi water in the pressure jacket. A baffle is used to promote the natural circulation of water in the jacket.

Pressurized water is supplied to the pressure jacket by a pump on the third floor of the reactor through a 1/16 inch OD 304 stainless steel pressure leads. The pressure generated by the thermal expansion of the water in the capsule is relieved through another 1/16 inch OD line through a relief valve set at 1000 psi. The water from the relief valve goes to a collection system. When the fuel is operating at steady state, there is no flow of pressurized water through the pressure jacket.

The pressure jacket is cooled by water from the GETR demineralizer. The cooling water flows through the cooling line, through the annulus between the pressure jacket and the cooling jacket and is discharged into the GETR pool through holes in the cap at the bottom of the capsule.

Heat Flux Determination

The heat generated by the fuel specimen is determined by flow and temperature difference measurements. One thermocouple at the top and one at the bottom of the capsule measure the temperature rise of the cooling water. The inlet and outlet thermocouples are paired so that the voltages would buck each other. The net differential voltage from these thermocouples is then fed to an amplifier and recorded directly in degrees F. The flow is measured by a 0.5 percent accuracy orifice flow meter.

Heat flux determinations made by this method agreed within $\pm$ 5 percent of those calculated by physics.

Capsule Oscillating Equipment

Power cycling of the fuel is accomplished by moving the capsule toward and away from the reactor peak flux zone. A mechanical oscillator is used to move the capsule in and out of the flux zone. The movement of the oscillator arm toward and away from the facility tube is transmitted to the capsule by the flexible cooling water line which is attached to a "T" at the center of the oscillator arm. A rate of one power cycle in two minutes was selected for these experiments. Moving the capsule in and out of the reactor flux zone with this machine produces a sinusoidal power generation curve.

Fission Product Collection System

A system is provided for the collection of fission products in the event of a fuel rupture. Each time the capsule is inserted into the flux region the thermal expansion of the water in the pressure jacket causes the pressure to increase. Water displaced by the thermal expansion is relieved through the 1/16 inch line, through the relief valve, and into the collection system. The collection system consists of a de-ionizer, capacitance cylinder, and a holdup tank in series. The de-ionizer removes most of the halogen fission products, the capacitance cylinder separates fission gases and the water, and the holdup tank serves as a means for much of the remaining activity to decay off before being discharged to the contaminated drain.

Fuel Failure Detection

Two methods were used to determine fuel failures. These are: (1) a radiation monitor on the outlet line, and (2) periodic radiochemical checks on water samples from the capsule.

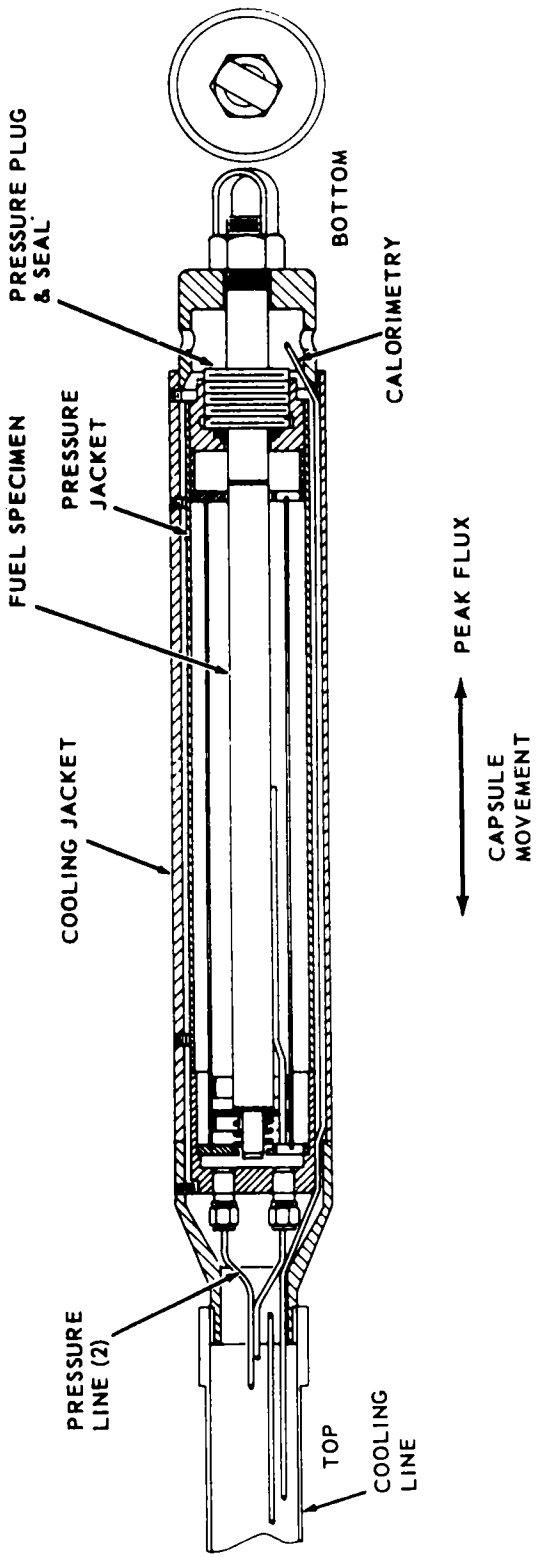


FIGURE 1 CAPSULE ASSEMBLY

SODIUM LOGGING CAPSULE DESIGN AND OPERATION

By

G. L. O'Neill
P. E. Novak
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M. L. Johnson

Work Performed Under

U. S. Atomic Energy Commission

Contract AT(04-3)-189
Project Agreement 10

SODIUM LOGGING CAPSULE DESIGN AND OPERATION

General

Sodium logging phenomena have been investigated[1] by irradiating high performance UO_2 and mixed PuO_2-UO_2 fuel in test capsules designed to simulate failed fast reactor fuel then examining the fuel specimens in a hot cell after irradiation. Nine test capsules were built and irradiated in the Trail Cable Facility of the GETR. Variations in the several experiments were limited to the fuel rod design and the irradiation procedure, with the test capsule and test facility remaining basically unchanged. The first six capsules contained enriched UO_2 fuel rods and served as scouting tests to minimize the number of PuO_2-UO_2 experiments required and thereby reduce program costs by simplifying fabrication and post-irradiation examination. The final three experiments contained PuO_2-UO_2 fuel rods of the same composition as reference FCR fuel. The fuel was nominally 1/4-inch outside diameter and operated at surface heat flux as high as 1.36×10^6 BTU/hr-ft^2. Capsules in each of the two series were irradiated under increasingly severe conditions, culminating in the irradiation of a hypothetical worst-case test in each series. Capsule construction and operation are described in this paper.

Test Capsules

The primary capsule design criterion for the sodium logging experiments was to provide an environment for the fuel rod as near as possible to that of FCR fuels at maximum FCR test conditions. Major considerations in this regard were placement of the fuel rod inside a sodium or NaK container and the utilization of materials of the proper size and thermal conductivity in the heat flow path to establish the reference maximum clad temperature (1250°F). All of the capsules were similar in concept, with only small variations in detail to satisfy test objectives.

An isometric view of the capsule is shown in Figure 1. The fuel rod is doubly contained with sodium or NaK thermal bonding in the inner container, and NaK in the outer container. Capsules were 1-3/4 inch outside diameter. In the outer container, concentric cylinders of 6061T6 aluminum and Zircaloy 2 were sized to provide a clad outer temperature of 1250°F when at design power, with fission and gamma heat being the only energy source. Thermoflex insulators at each end of the fuel reduced end effects and thereby formed a known heat-transfer geometry permitting a measurement of power by use of thermocouples in the sodium annulus and the water annulus surrounding the outer container. Liquid metal annuli were designed as small as possible to minimize natural convection and improve the accuracy of heat flow calculations.

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1. G. L. O'Neill, P. E. Novak, M. L. Johnson, and W. E. Baily, Experimental Studies of the Effects of Sodium Logging in Fast Ceramic Reactor Fuels, GEAP-4283, Sept. 1963.

Cooling water was supplied to the upper end of the capsule through a flexible hose, and discharged to the facility tube through the open lower end. Thermocouples located in the inlet and discharge ends of the capsule and an orifice in the supply water line provided calorimetric data for power measurement. A typical capsule assembly is also shown in Figure 1. Design and operating data are summarized in Table I.

Based on data from the first four capsules, the resistance of the heat flow path was adjusted to give better agreement with the desired clad temperatures. On Table I the improvement in reaching the target values for subsequent capsules can be noted.

The GETR Trail Cable Facility

The capsules were irradiated in the small GETR Trail Cable Facility, one of two facilities in the pool of the General Electric Test Reactor. The facility tube, a double-wall aluminum tube, extends vertically from the reactor building third floor down through the reactor pool to the bottom of the reactor core. At the core level the tube is immediately adjacent to the outside wall of the reactor pressure vessel. The facility is shown schematically in Figure 2.

Test capsules are inserted into the tube at the top end and are suspended at the proper elevation by a flexible hose. The hose, an integral part of the capsule, serves as a cooling water supply tube and as a conduit for thermocouple lead wires. Capsules may be inserted, repositioned or removed from the facility while the reactor is at power. All positioning operations are by hand movement of the hose.

Cooling water supplied to the Trail Cable hose by the plant demineralized water system provides forced-convection cooling of the capsule. Water discharged from the lower end of the capsule flows upward around the capsule in the annulus formed by the capsule outer shell and the inner facility tube. It is finally discharged to the pool through holes in the facility tube wall above the core level. The flow rate is measured and indicated by an orifice in the supply line and a Barton differential pressure cell.

A ten-point recorder is available for miscellaneous capsule temperatures. A second two-pen recorder is used to provide a measurement of water temperatures at the inlet and discharge of the sample. These two water temperatures are used for calorimetry, and consequently a very accurate measurement of a small temperature difference is required. Electronic amplifiers in the circuity amplify the thermocouple emf by a factor of 13 or 43.5, as selected, and thereby provide the required definition on the recorder chart.

Fluxes available in the facility at the facility tube centerline are on the order of 7.5×10^{13} nv with an integrated peak-to-average ratio of approximately 1.45.

Test Procedure

The irradiation procedure was similar for all of the sodium logging capsules. Cooling water and thermocouple leads were attached to the junction box at the upper end of the trail cable, and pre-irradiation checks of thermocouple resistance were made. The capsule assembly was then inserted to the 20-foot depth in the trail cable facility tube. At this point, the bottom of the capsule was still about nine feet above the top of the GETR core. Cooling water flow was adjusted to the required flow rate, and when the reactor reached the desired operating power level the capsule was slowly lowered into the flux to a point where the sodium temperature reached a steady value of 240°F, indicating that the sodium had melted. (This step was omitted on capsules containing non-defective specimens with liquid NaK in the inner container.) The capsule was then inserted to peak flux at the prescribed insertion rate. The final adjustment of the sodium temperature was then made by vertical adjustment of the capsule and cycling was accomplished as prescribed for the individual capsules. Following irradiation, the capsule was withdrawn to the 20-foot level and allowed to cool before being transferred to a cask for shipment to the RML.

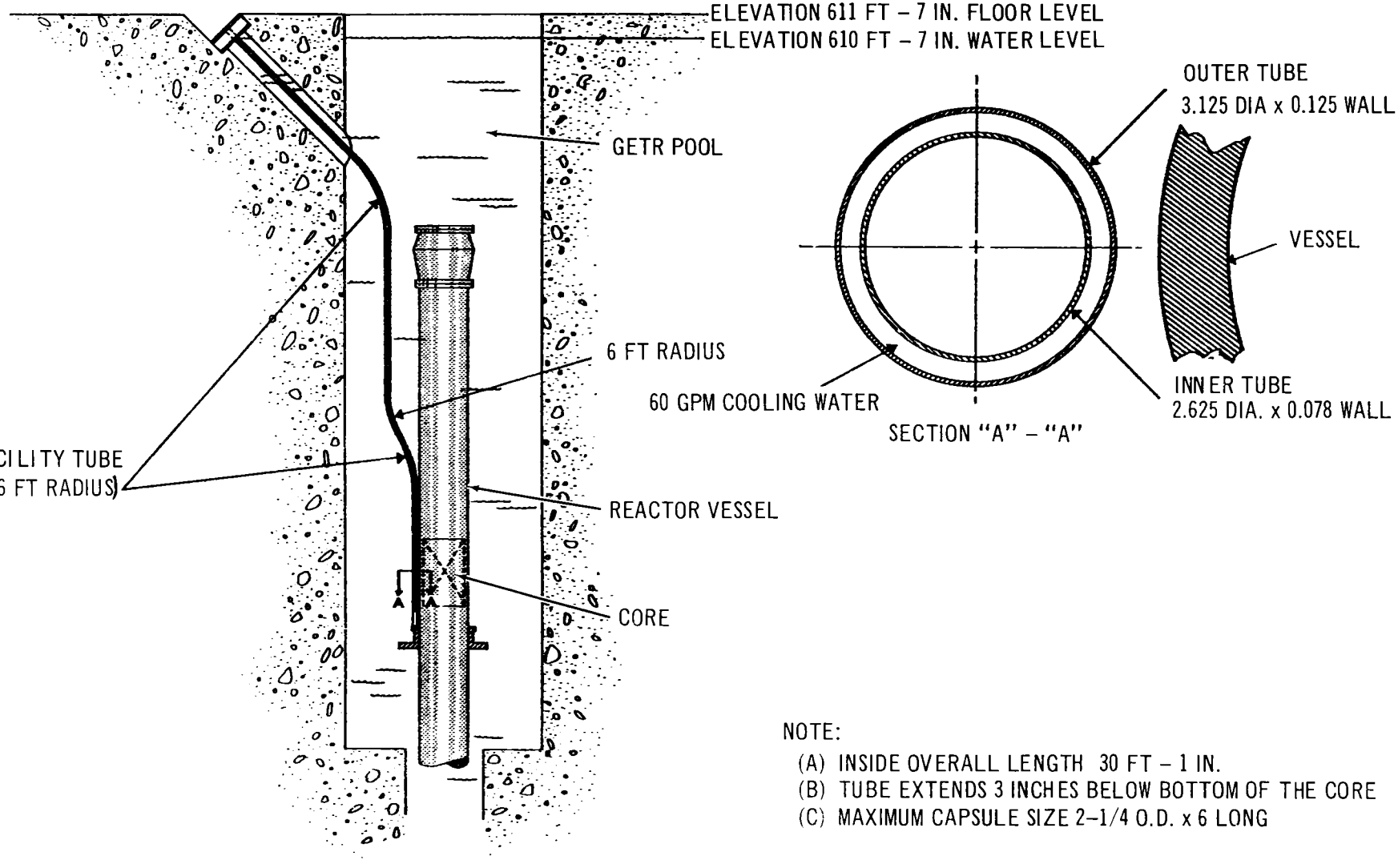
Power measurements were made during the irradiation by two methods. Calculations made prior to the irradiation determined a desired temperature in the liquid metal annulus nearest the fuel pin, based on a desired power level and known cooling water temperature and heat transfer geometry. The capsule was positioned at the beginning of the irradiation to establish the desired liquid-metal temperature, and adjusted throughout the irradiation as necessary to maintain it. During the irradiation the power was calculated by varying the cooling-water flow rate and recording the differences between inlet and outlet cooling-water temperature for several flow rates.

TABLE I

CAPSULE DESIGN AND OPERATING DATA

	B-I-A	B-I-B	B-I-C	B-I-D	B-I-E	B-I-F	B-II-A	B-II-B	B-II-C
$\int_{T_s}^{T_c} k d\theta$ (w/cm)	48	48	48	48	58	48*	48	42*	48
Design Power (kw/ft)	21	21	21	21	26	26	24	24	24
Measured Power (kw/ft)	20	21	21	19	25	22	25	25	25
Fuel Clad Surface Heat Flux** (Btu/hr-ft^2)	1.10×10^6	1.10×10^6	1.10×10^6	1.10×10^6	1.36×10^6	1.36×10^6	1.25×10^6	1.25×10^6	1.25×10^6
Design Clad Surface Temp.($^\circ$F)***	1250	1250	1250	1250	1250	1250	1250	1250	1250
Avg. Measured Clad Temp.	1050	1070	1040	970	1190	1030	1168	1263	1265
Gamma Heating Rate (watts/gm)	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
Inner Liquid Metal Container Temp. ($^\circ$F)	1112	1112	1112	1112	1074	886	874	874	1081
Zircaloy 2 Barrier Temp.($^\circ$F)	866	866	866	866	754	568	723	723	939
Aluminum Barrier Temp.($^\circ$F)	289	289	289	289	341	323	323	323	308
Outer NaK Container Temp.($^\circ$F)	218	218	218	218	233	233	229	229	231
Bulk Water Temp. ($^\circ$F)	100	100	100	100	100	100	90	110	110
Liquid Metal in Outer Annulus	Na	Na	Na	Na	Na	NaK	NaK	NaK	Na

* Low value of integral due to effect of center hole.
** Fuel clad diameter was 0.25 inch.
*** Temperature of clad outer surface. All other temperatures for materials in this table are the inner surface.

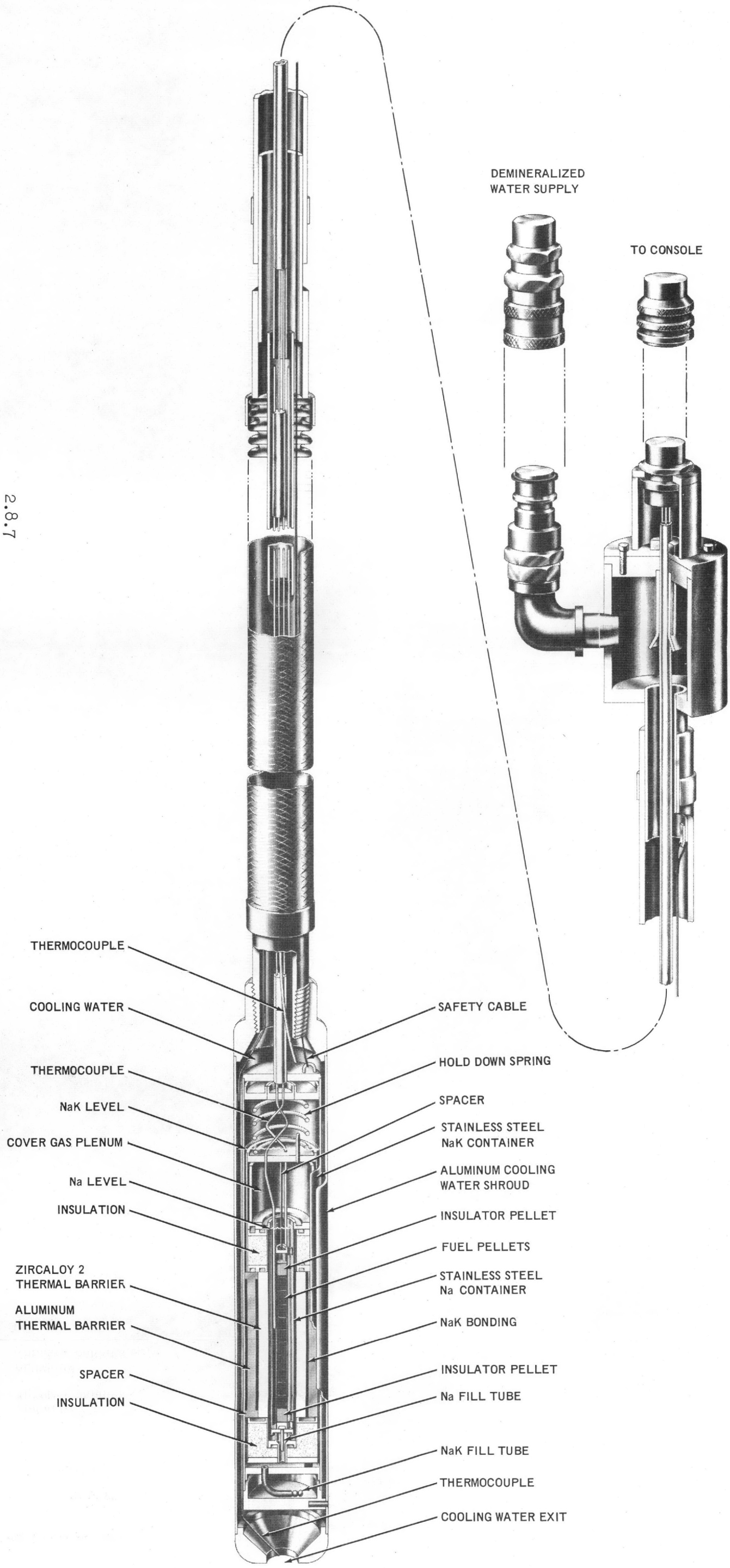


2.8.6

NOTE:

(A) INSIDE OVERALL LENGTH 30 FT – 1 IN.
(B) TUBE EXTENDS 3 INCHES BELOW BOTTOM OF THE CORE
(C) MAXIMUM CAPSULE SIZE 2–1/4 O.D. x 6 LONG

TRAIL CABLE FACILITY



Sodium Logging Capsule

COMMISSARIAT A L'ENERGIE ATOMIQUE

IN PILE FACILITIES TO STUDY FISSION GAS RELEASE IN UO_2 SAMPLES UP TO 2000°C

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Service de Chimie des Solides
Section des Combustibles Céramiques

The facilities described here have been designed to study radioactive fission gases released during irradiation in UO_2 samples at very high temperatures: This work was initiated under Euratom contract no. 031.60.10 RDF.

The irradiations take place in the "Melusine" reactor. It is a 2 MW swimming-pool research reactor located at the Centre d'Etudes Nucleaires de Grenoble.

UO_2 samples can be heated up to 2000°C. Gaseous fission products are swept by a purified carrier gas to a fission gas collection apparatus. The quantity of collected radioactive isotopes of Xe and Kr is determined by gamma-ray spectrometry.

Operating in the range 1000-1700°C, similar facilities have been earlier designed by Bettis Atomic Power Laboratory (1), Oak Ridge National Laboratory (2), Battelle Memorial Institute (3), and Atomic Energy Research Establishment (4).

*

* *

(1) - J. M. Markowitz, R. C. Koch, J. A. Roll - WAPD-180, July 1957

(2) - R. M. Carroll - ASTM-306, Atlantic City, June 1961

(3) - J. B. Melehan, F. A. Rough - ASTM-306, Atlantic City, June 1961

(4) - F. J. Stubbs, G. N. Walton, P. J. Silver - AERE C/R 2176

I. EXPERIMENTAL APPARATUS

1) - Operating techniques

Experimental conditions are as follows:

- Irradiation time: 5 days
- Maximal thermal neutron flux: $1.3\ 10^{13}$ n/cm^2/sec.
- Mean thermal neutron flux: $5\ 10^{12}$ n/cm^2/sec.
- Sample weight: 50 mg (nuclear heating is negligible).

The spectrometric analysis of the collected fission products leads to the calculation of the quantity released for each isotope. The fractional release (f) is obtained by comparison with the theoretically present quantity in the sample. When diffusion is the only factor controlling fission gas release the apparent diffusion coefficient (D') can be calculated from the formula:

$$F = 3\sqrt{\frac{D'}{\lambda}}$$

where λ is the radioactive decay constant of the isotope.

2) - Out-of-pile Section

The flow-diagram of the fig. 1 shows the major components of the loop. Helium carrier gas coming from the furnace passes through a continuous activity monitor placed in the well of a scintillation crystal and then through six charcoal collection traps in series. To absorb the gas each trap can be cooled with a mixture of solid carbon dioxide and

trichlorethylen (-78°C).

The collection traps are followed successively by two charcoal adsorption traps designed to collect fission gases which might accidentally escape the previous system, the membranecirculation pump and sweep gas purification system. That system usually consists of two Linde molecular-sieve traps cooled by liquid nitrogen. It can be substituted by a column of BTS catalysator followed by a Linde molecular-sieve column when the helium is replaced by a helium-hydrogen mixture as carrier gas.

The isotopes of the halogens Br and I which are released simultaneously with rare gases are trapped at the exit of the capsule by a silvered copper trap to avoid possible contamination of the circuit.

3) - Neutron dosimetry

The thermal neutron flux is continuously measured by argon activation. The argon gas stream passes through a stainless steel tube wound on the outer tube of the furnaces. The argon activity is then recorded out of pile.

4) - Irradiation facilities

These have been designed on the following basis:

- The UO_2 samples are to be maintained in flowing helium up to 2000°C;

- After irradiation the sample must be recovered and the apparatus has to be made available for further experiments;

- Porous refractories are undesirable in the gas circuit because of the difficulties of degassing.

Two irradiation facilities have been constructed: The first one is a molybdenum resistor furnace (RM1) permitting irradiations up to 1400°C; the second is a high frequency induction furnace (HF3) which can attain 2000°C during irradiation.

II. MOLYBDENUM RESISTANCE FURNACE

This furnace was designed to work up to 1400°C. The figure 2 shows a burst view of the main components of the furnace RM1.

- The upper part of the capsule (1) is welded on one side to the outer shell (6) and on the other side to a bellows (2) designed to prevent expansion stresses in the outer thermal shield (3) which is welded at the lower end to the body of the capsule.

- Another stainless steel shield (4) is welded only at the lower tabs.

- A molybdenum radiation shield (5) is screwed at its upper end to the top piece (1).

- The electrical leads cross the outer shell (6) by two holes (7).

- Metallic pieces (8) support the resistor (not shown in this view).

- (9), lead gasket

- (10), sample holder into which are the thermocouple measuring the sample temperature and the regulation thermocouple.

- (11), foot of the capsule.

The figure 3 shows a gamma-radiograph of the assembled capsule with resistor, container and sample. The sample temperature is measured by a Pt-PtRh thermocouple during a calibration period. To avoid irradiation drift of Pt-PtRh, the temperature control is then realized by a chromel-alumel thermocouple placed in a colder zone. The temperature of the sample is regulated at $\pm$ 1°C.

III. INDUCTION FURNACE HF3

The purpose of this furnace is to work in the range 1400-2000°C.

1) - Principle

The UO_2 sample placed in a tungsten container is located in a tantalum tube heated by induction. The inductor is a niobium wire coil.

Gaseous fission products released from the sample are swept out in a stream of pure helium to the out-of-pile section.

The rig is composed of three main items, as follows:

- The induction furnace

- The in-pile coaxial line
- The electrical circuit.

2) - The induction furnace

The figure 4 shows from the inside to the outside:

- The tungsten container in which is located the UO_2 sample and its W-Re 5% - W-Re 26% thermocouple. The helium stream passes through this container by holes of diameter 1 mm. The tungsten container is suspended from the thermocouple.
- The inner tantalum tube for evacuating the helium gas,
- The outer tantalum tube for entry of the helium gas,
- The niobium coil insulated by beryllia cylinder,
- The niobium and stainless steel radiation shields,
- The outer stainless steel tube on which argon loop is wound.

3) - The in-pile coaxial line

The in-pile coaxial line is rigid and consists of:

- A central electrical conductor which is a copper tube of diameter 8 x 10 mm.
- An outer electrical conductor which is a copper tube of diameter 18 x 20 mm.
- A leak-tight stainless steel tube.

At the upper part of the in-pile coaxial line we find:

- The electrical adaptation condensers;
- A device to correct the thermal expansion.

The in-pile coaxial line is connected directly to the bottom of the rig.

The figure 5 shows a burst view of the furnance during assembly.

The figures 6, 7, and 8 show different views of the furnace during assembly.

4) - The out-of-pile electrical circuit

The out-of-pile circuit is composed of:

- The high frequency generator 6 KVA
- A flexible coaxial line about 20 m long
- The electrical adaptation condensers
- An oscillograph
- The instrumentation and control equipment

5) - Operating conditions

The operating conditions for this rig are given below:

- Neutron thermal flux: $1.3\ 10^{13}$ n/cm^2/ sec.
- Maximum temperature of the sample: 2000°C
- Maximum temperature of the niobium inductor: 1700°C.

The sample and its thermocouple can be displaced inside the rig to obtain the best thermal gradient while the pile is operating.

The sample and its thermocouple are changed at each irradiation.

It is not necessary to use the hot cell for the unloading of the sample.

The helium circuit for sweeping the gaseous fission products is independant on the helium medium for the heat transfer. Hence there are no refractory oxides in the helium measuring circuit.

The niobium inductor and the tantalum susceptor can be changed in the hot cell.

6) - The induction furnace has now been operating in pile since June 1963. The development of this rig took a long time, about two years. Two preliminary rigs were built before introducing the third into the pile.

The latter was tested for about 1000 hours out of pile.

IV. PRELIMINARY RESULTS

Several irradiations have been performed with these facilities. Results are not yet completely known, but we can give a curve of activity collected during a short irradiation with the induction furnace HF3. The purpose of this experiment was to gain an idea of the amount of activity

released and the nature of the isotope collected.

The figure 9 shows the activity recorded by the continuous activity monitor. One can remark the following points:

1°) - Bursts of activity following each elevation of temperature, and most marked for the highest temperatures.

2°) - Two very important bursts of activity produced by temperature decrease from 2000°C to 1800°C and from 1800°C to 1600°C.

*

* *

CONCLUSION

These two facilities are destined to make two sets of works:

- The first one is the study of fission gas diffusion from isothermically heated samples of sintered or monocrystalline UO_2. This study will complete previous works on this subject.

- The second one is the analysis of fission gas release in a range, above 1600°C, where diffusion is no more the predominant factor. It gives the possibility to study the "burst-effects" which are observed on fast heatings or coolings of samples, similar to those that happen in reactor.

2.9.11

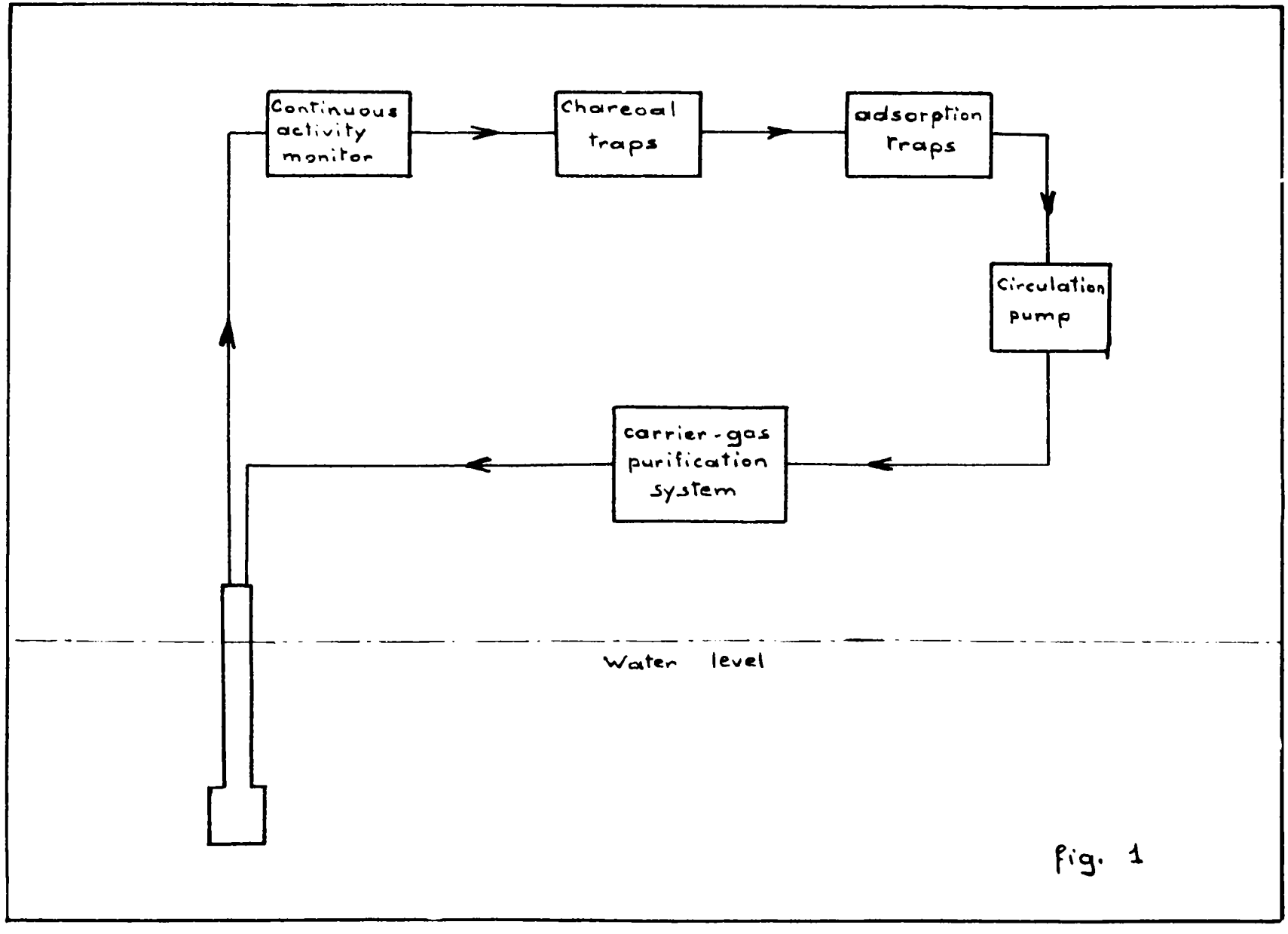
Continuous activity monitor
Charcoal traps
adsorption traps
Circulation pump
carrier-gas purification system
Water level

fig. 1

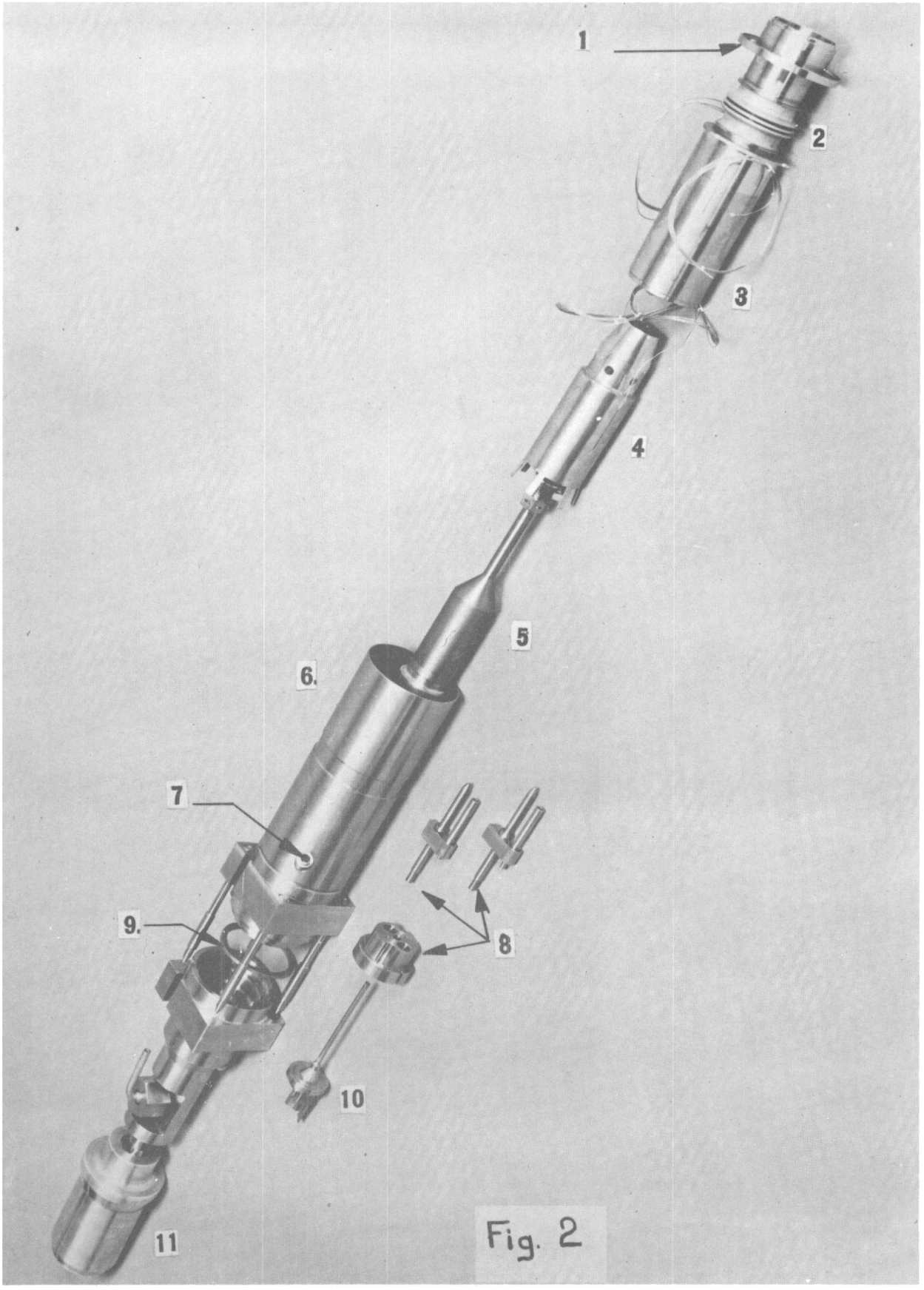

Fig. 2

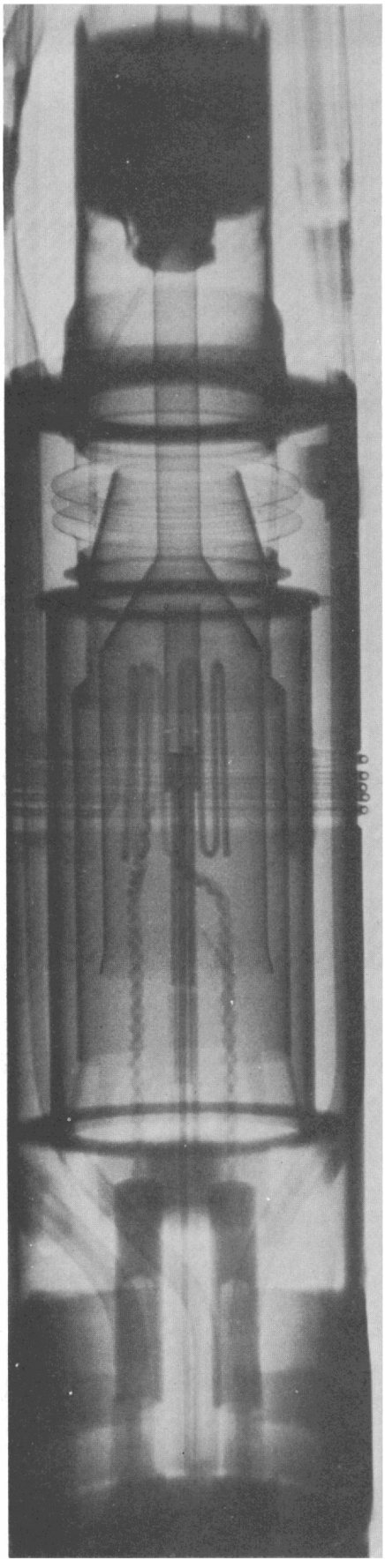

Fig. 3

2.9.13

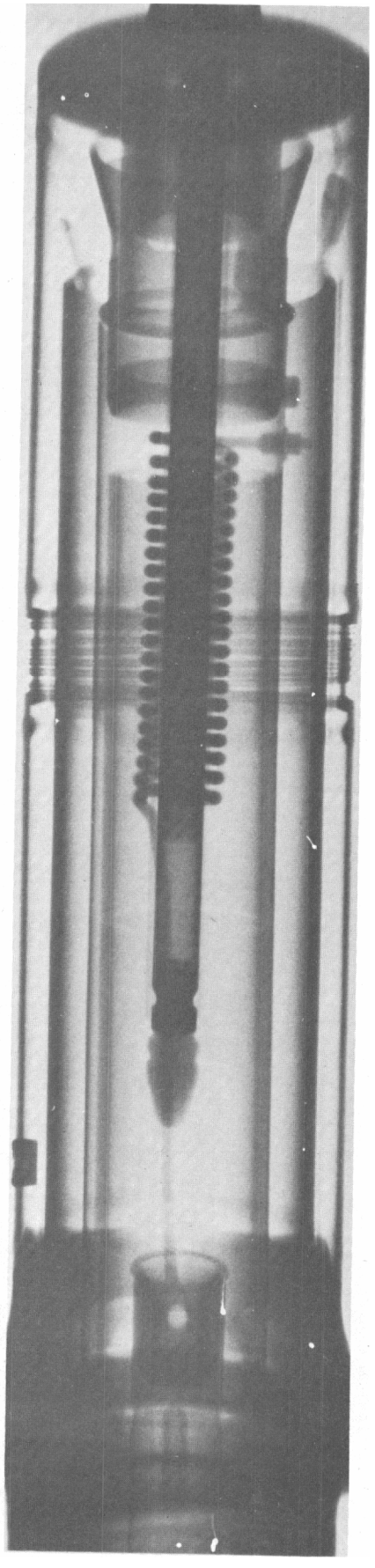

Fig. 4

2.9.14

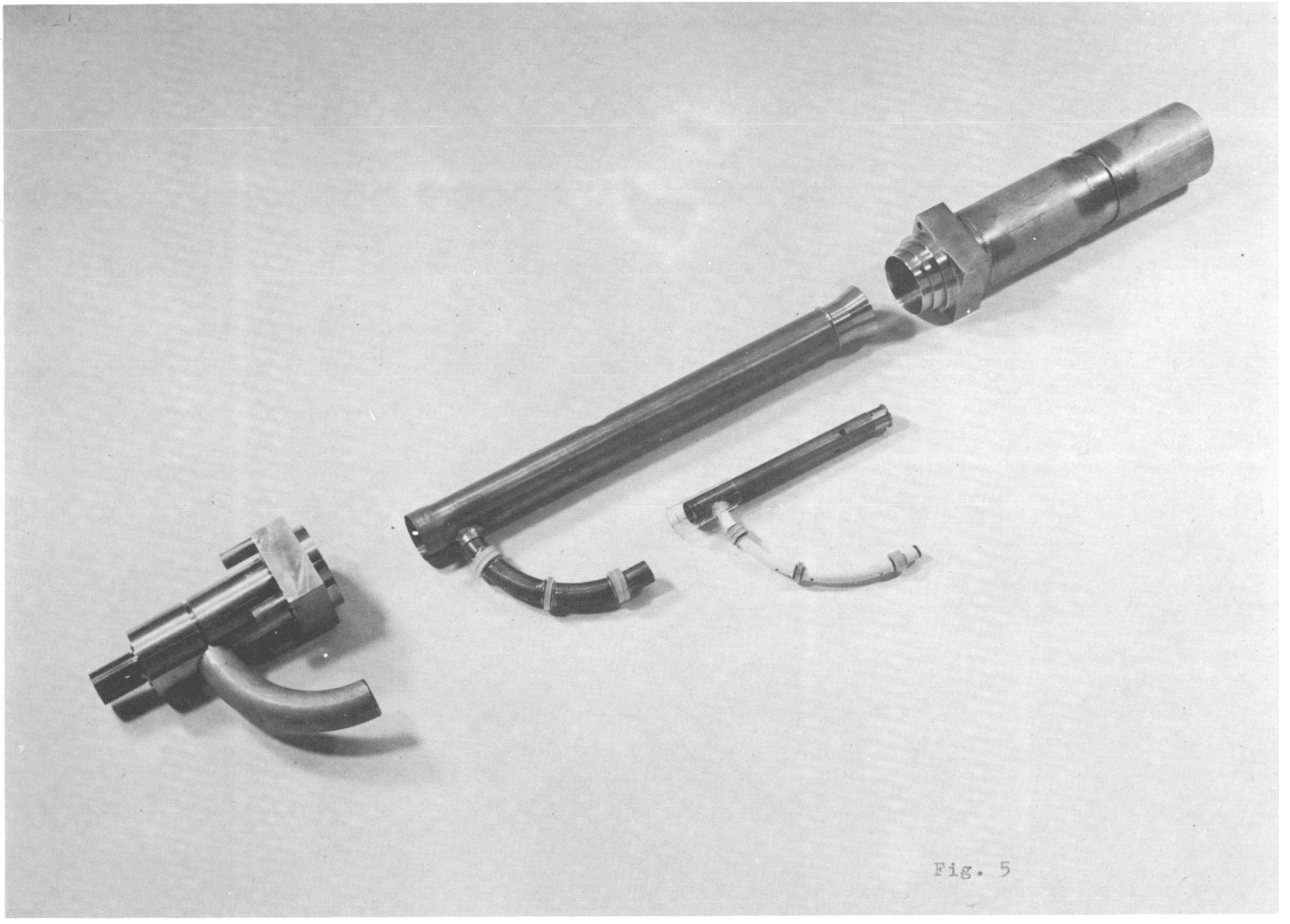

Fig. 5

Fig. 6

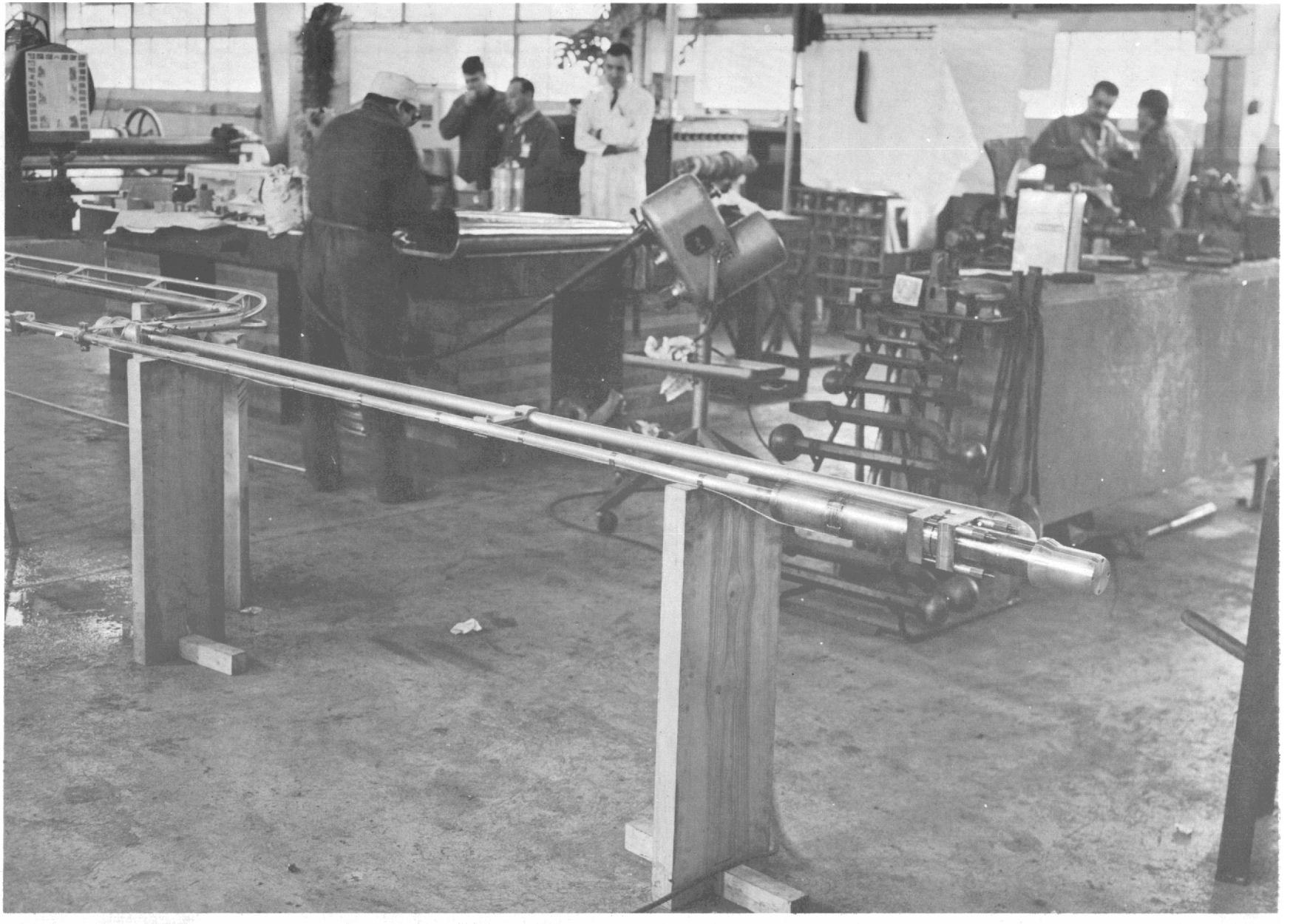

Fig. 7

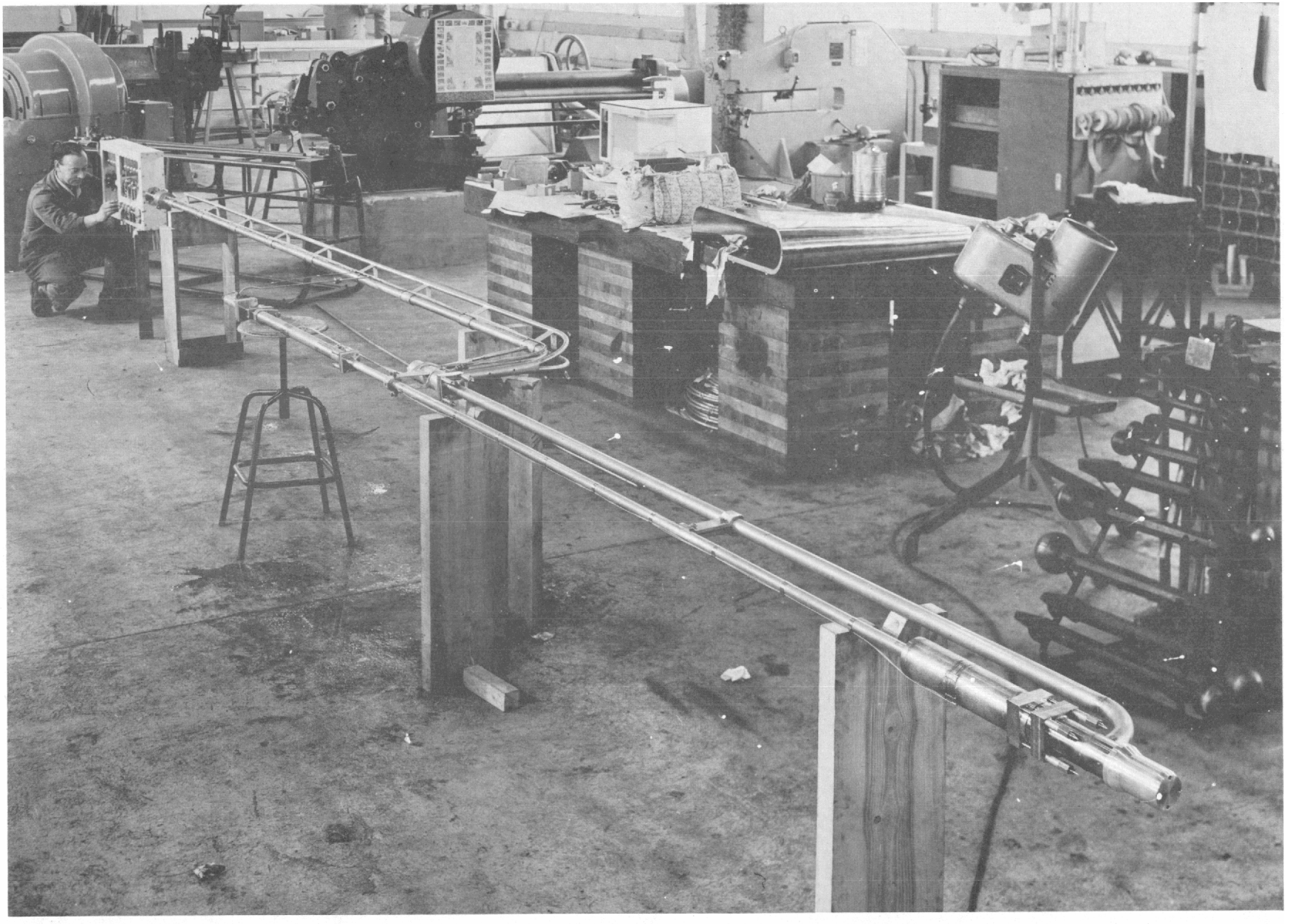

Fig. 3

VARIATION OF INSTANTANEOUS ACTIVITY VS TEMPERATURE

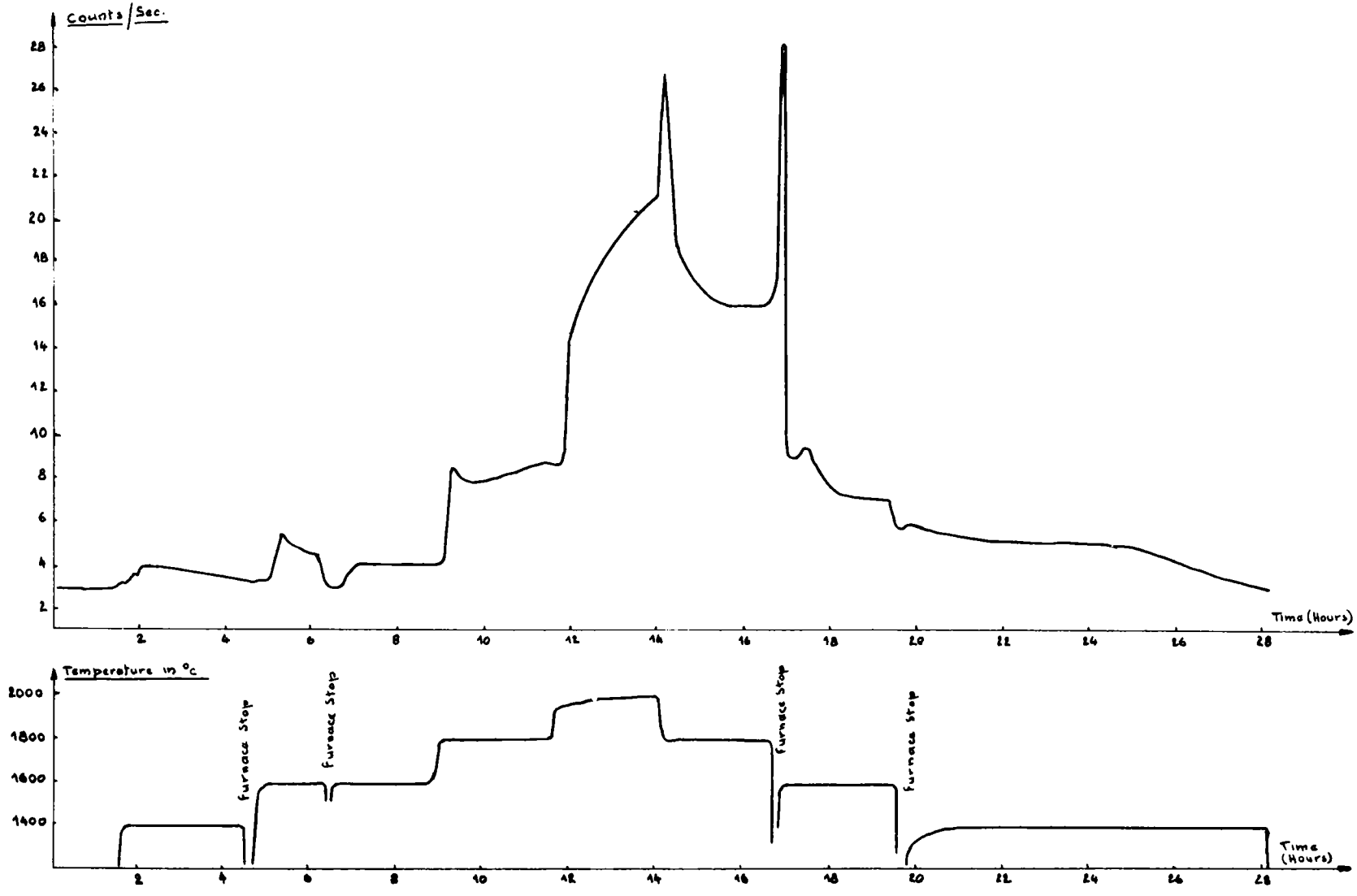

Fig: 9

2.9.19

RECENT DEVELOPMENTS IN THE DESIGN OF SWEEP-CAPSULE EXPERIMENTAL SYSTEMS FOR DETERMINING FISSION-GAS-RELEASE CHARACTERISTICS*

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Battelle Memorial Institute

Battelle is engaged in a program to develop coated fuel particles for high-temperature gas-cooled civilian reactors. The goal is to produce 40 to 125-μ metal oxide or carbon coatings which retain essentially all of the fission gases released from 125 to 300-μ particles of UO_2 or UC_2. These coatings are being evaluated by operation in static and sweep-gas experiments at the Battelle Research Reactor. The latter type provides the most definitive gas-release data. This discussion outlines briefly how they have been applied to this particular research program.

Thus far, the irradiation program has been used exclusively to provide information for feed-back into the particle-fabrication development program rather than to provide engineering data for a specific prototype fuel or fuel geometry. As such, the emphasis on the capsule design has been toward simplicity for rapid and economical fabrication to minimize the time from coating preparation to the in-pile evaluation of some particular coating problem.

The sweep-gas systems which are in use on this program are illustrated by the flow diagram shown in Figure 1. Helium, from a bottle supply manifold, passes through a purification system, through a flow-regulation system, to the capsule. From the capsule, fission gases are carried by the sweep gas to a short-half-life sampling site, through a holdup tank for decay of short-half-life species, and past a long-half-life

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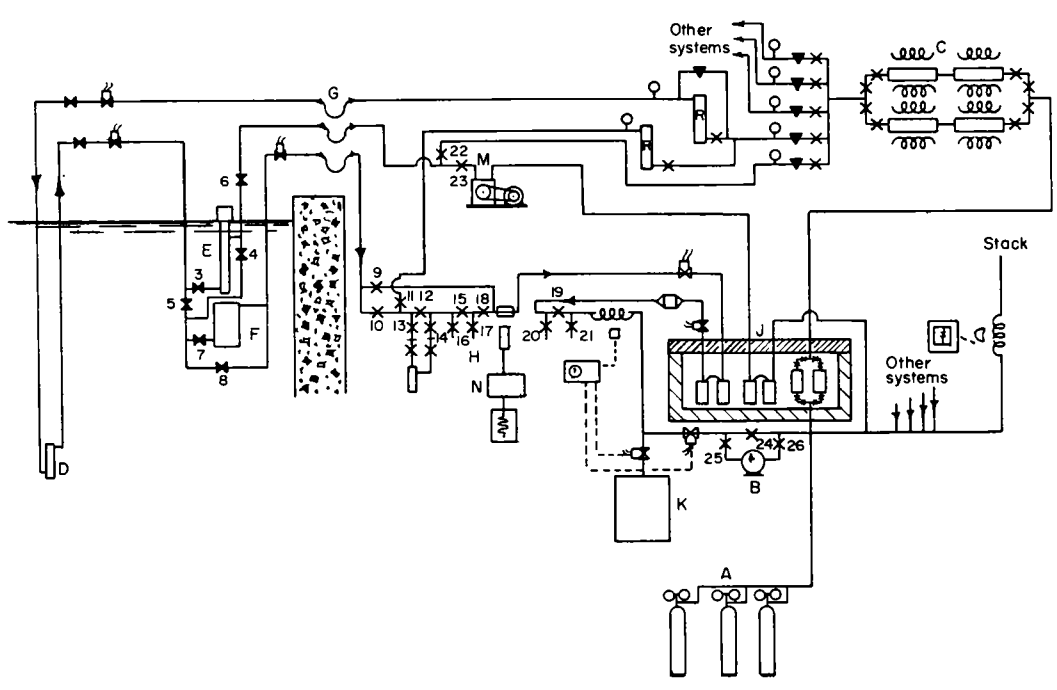
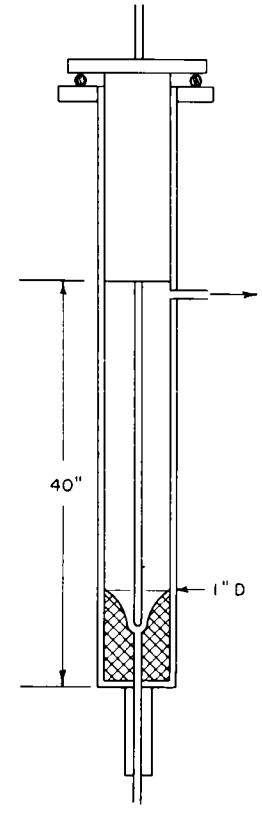
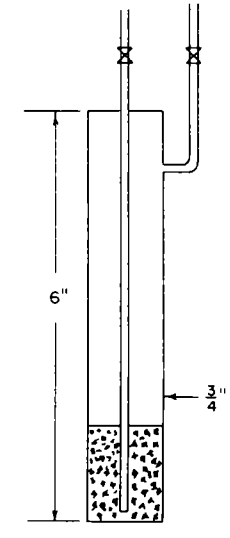
* The work reported in this paper was performed by Battelle Memorial Institute, under AEC Contract U-7405-eng-92.





Daughter Trap



Adsorption Trap

A. Helium manifold
B. Wet test meter
C. Helium purification
D. Capsule
E. Daughter trap
F. Holdup tank
G. Flexable coupling
H. Gas sample system
J. Refrigeration and final removal traps
K. Storage tank
L. Particle filler
M. Vacuum pump
N. Gamma ray spectrometer

Manual valve
Double contained manual valve
Solenoid valve
Double contained solenoid valve
Pressure regulator
Pressure gauge
Rotometer
Activity monitor

FIGURE 1. SCHEMATIC DIAGRAM OF CFP SWEEP-GAS SYSTEM

sampling site. Finally, the flow is released to the building exhaust system after passing through activity-removal traps. The systems are operated continuously with a minimum of manned operation.

Samples for the longer half-life[1] species analysis are taken by diverting the sweep gas through small liquid-nitrogen-cooled activated-charcoal adsorption traps. Samples involving analysis for shorter half-life constituents are accumulated by diverting the sweep gas through a "daughter trap" where the solid daughter products produced by decay of the gases are deposited on a charged wire. For the fuel loadings used, the sensitivity of these two sampling techniques is 10^{-10} R/B(ratio of atoms released to atoms being produced) for the long-half-life species and 10^{-7} R/B for the shorter half-life gases.

Components present in the system to increase the efficiency and economy of operation are:

(1) A -150 F chilling machine which has proven very effective for cooling both aluminum-shot-packed towers for initial moisture removal from the inlet helium, and the activated-charcoal-filled activity-removal traps. During periods of highest activity input to these traps, the concentration of xenon-133 in the effluent gas has been measured at 10^{-9} μ curies/cc.

(2) An in-line beta monitor downstream from these activity-removal traps is used to divert the effluent gas to a storage tank if a malfunction causes high activity in the exhaust gas.

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1. The long-half-life gases are xenon and krypton isotopes with half lives of 1 hour to 5-1/4 days. Short-half-life gases are isotopes of the same elements with half lives from 2 sec to 4 min.

(3) A sensitive air monitor detecting high activity in the area is used to shut down all systems by closing solenoid valves to isolate various sections of each system.

(4) A gamma-ray detector on a small chamber in each of the four sweep-gas systems is intermittently connected by an automatic switching system to a single-channel sweeping-window-type gamma-ray spectrometer which records count rate in the energy range of 0 to 0.5 Mev. This in-line detector system provides semiquantitative data of xenon-133 and xenon-135 release every 1 to 2 hours throughout the experiment and is useful in determining relative changes in the fission-gas-release rates during unmanned operation.

The fuel specimens used in all experiments have each contained approximately 60 mg of U^{235}. This is sufficient for good analytical sensitivity with a minimum potential hazard from total fission gases and a minimum flux perturbation in the capsule. The coated-particle fuels have either been supported in a graphite matrix pellet or free of support as bare particles. In the latter case, the particles have been held in a specimen can in a thin layer to minimize the temperature gradient across the sample.

As a result of some static capsule irradiations early in the program, it was found that certain types of coated particles had good fission-gas retention at elevated temperature but very poor retention at low temperature. The study of the coating parameters affecting this low-temperature failure required a large number of irradiations of different coated-particle fuels and led to the use of low-temperature irradiations as a screening test for new materials. The irradiation capsule used for

these tests is a sweep capsule in its simplest form as illustrated in Figure 2. The fuel particles in the inner can are held in a thin layer between the graphite core and a metal insert. The sweep gas enters through a wire-mesh-covered hole in the graphite core and exits through the narrow annulus formed by the metal insert and the can wall. Gas-inlet and -outlet tubes are welded over the gas passages drilled in the can wall. Tight-fitting aluminum "clam shells" transfer the heat from the inner can to the outer shell wall so the maximum fuel temperature is approximately 100 C. The threaded sections on the capsule permit attaching a long guide tube on the top and a positioning device on the bottom or, in some cases, stacking two or three capsules in one assembly. The lead-out gas lines are attached by metal clips to the guide tube at 6-foot intervals. Over 40 of these capsules have been irradiated during the last 1-1/2 years for periods of a few hours to 10 weeks.

The sweep-capsule design used for elevated temperatures to 950 C is illustrated in Figure 3. This is a double-walled capsule with the inner shell containing grooves for 3 helically wrapped 1/8-in.-O.D. metal-sheathed electrical heaters. The annular space between these two shells is argon filled. The sweep specimen is supported in a graphite specimen holder with the sweep gas from the inner shell entering through a hole in the bottom, passing around the specimen, and leaving through the drilled outlet passage. The bottom of the heater shell houses up to 6 static specimen cans with a gas barrier separating this section from the sweep gas. This basic design, with slight variations, has been used for 10 experiments with irradiation times up to 5 months. The only problem encountered was with heater failures in the first three experiments when it was attempted to operate the heater shell over 900 C. These experiments were continued,

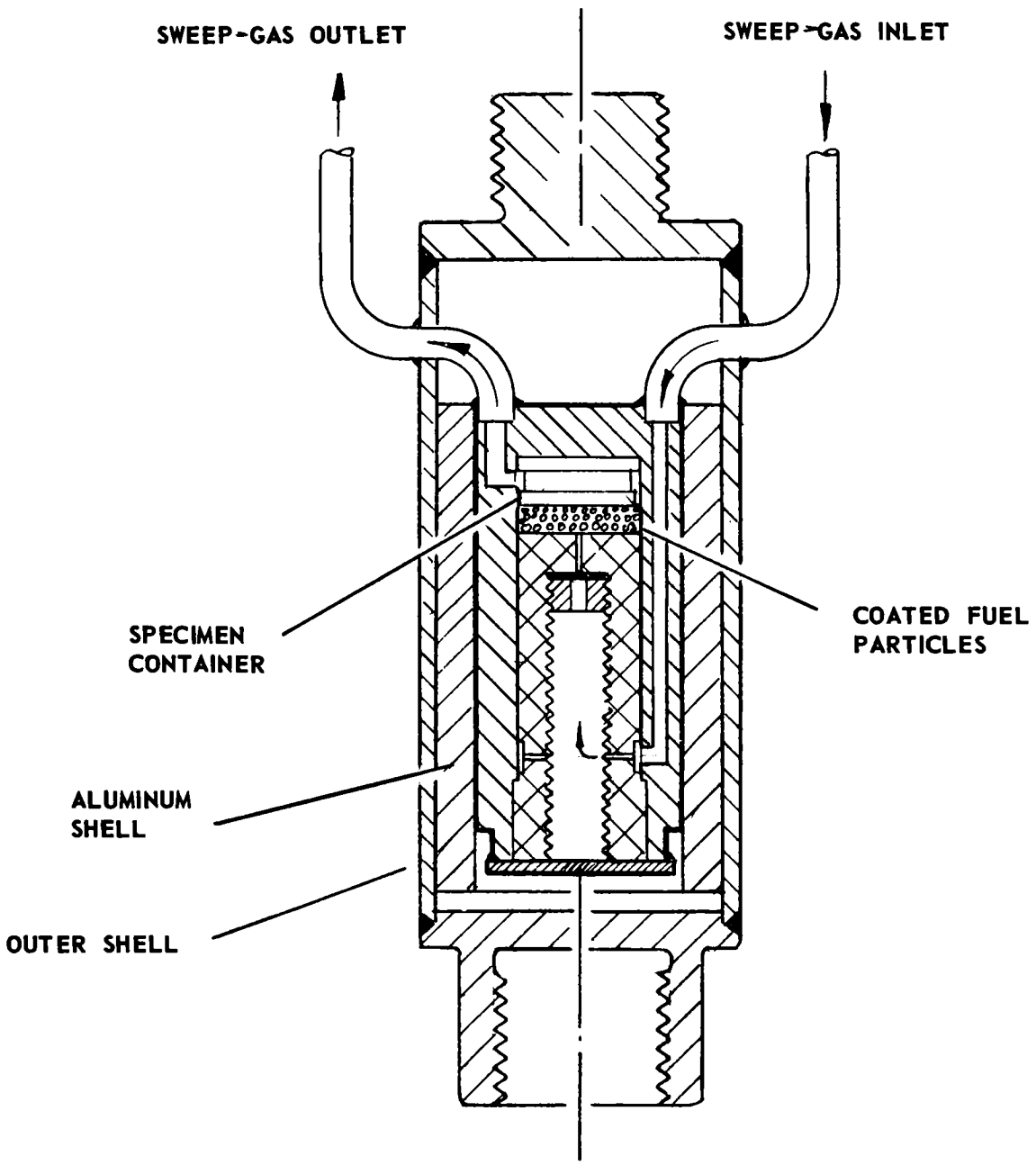

FIGURE 2. DESIGN OF 100 C SWEEP CAPSULE

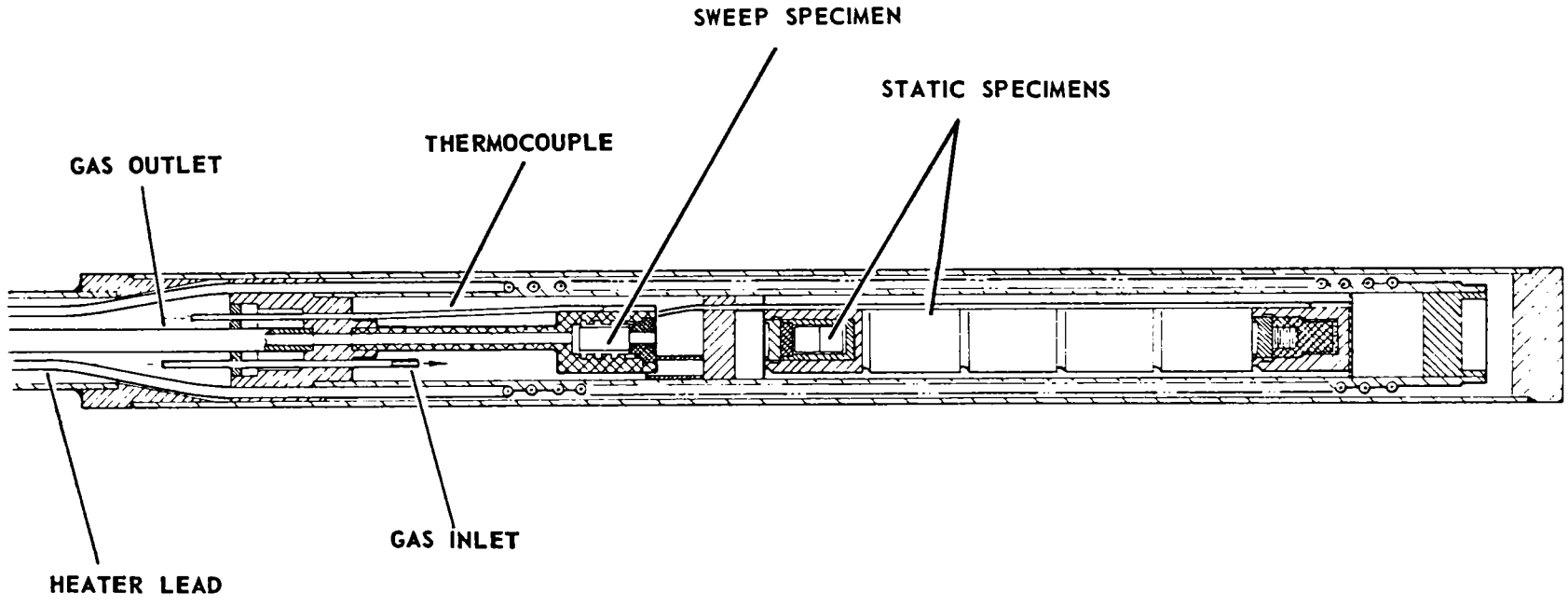

FIGURE 3. DESIGN OF ELEVATED-TEMPERATURE SWEEP CAPSULE

but at a lower temperature with the remaining heaters operable. It may be noted that, in a similar earlier design, a completely sealed refractory-metal sweep specimen container was used up to 1050 C. This design was abandoned because of weld failures resulting from oxygen embrittlement of the refractory-metal welds from the small oxygen impurity in the relatively large volume of helium sweeping through the system.

Figure 4 shows the capsule design used for experiments above 950 C. The auxiliary electrical heater is a tantalum wire wrapped on a grooved high-purity alumina tube. This is supported on a thin-walled tantalum inner shell which also acts as the primary sweep-gas containment. The annular space between the heater and outer shell is argon filled and also houses a molybdenum foil thermal-radiation baffle. The graphite sweep specimen holder is surrounded by a closed-end alumina tube which seats on the capsule header to form a gas barrier to prevent the flow of sweep gas from passing over the refractory metals. The static specimens in the lower part of the heater shell are contained in molybdenum-lined niobium cans. The thermocouples are 1/16-in.-tantalum sheathed MgO-insulated W-5 Re versus W-26 Re. Two experiments have been successfully completed using this design, one at 1200 C for 8 weeks and one at 1100 C for 9 months. The maximum temperature for this design is believed to be about 1550 C; the highest temperature achieved thus far is 1475 C for several weeks in a preliminary laboratory model.

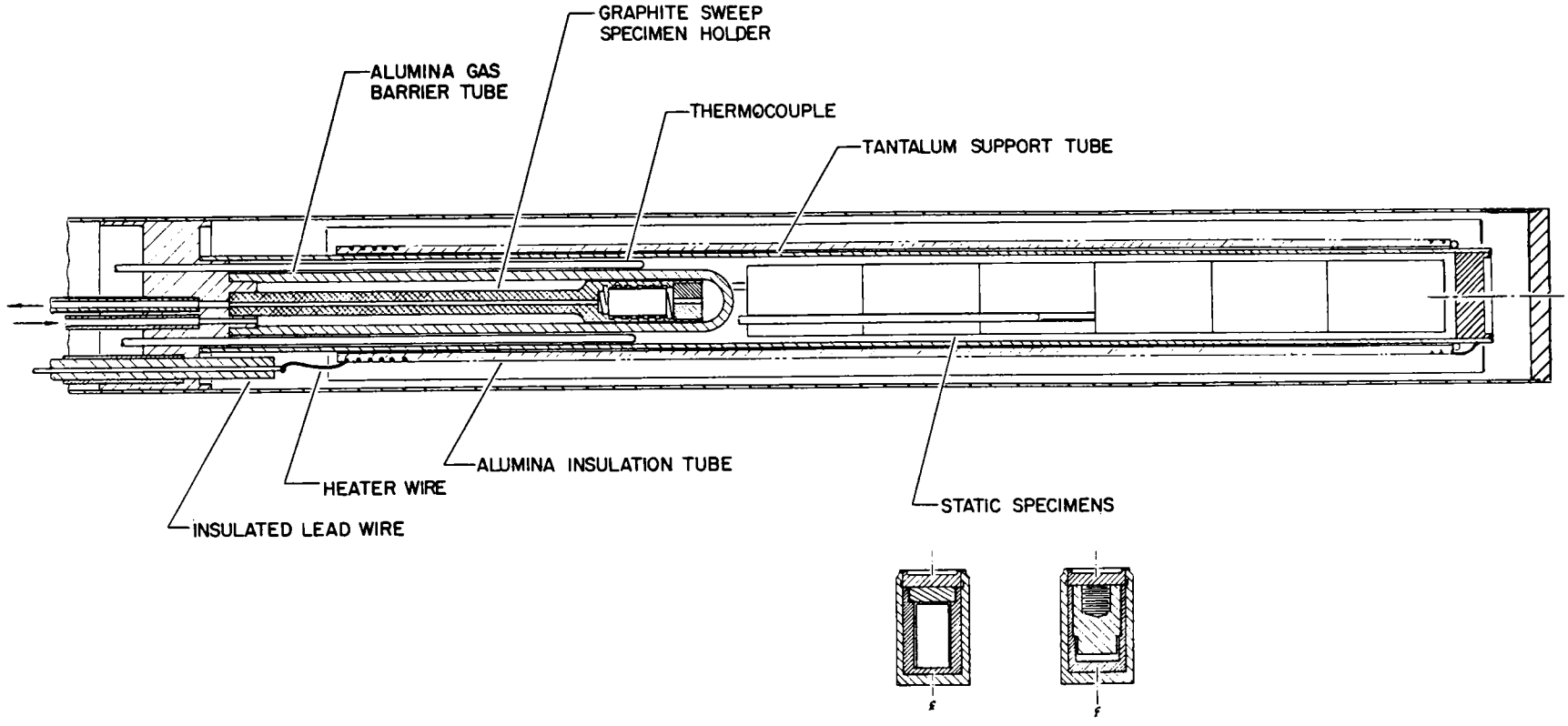

FIGURE 4. DESIGN OF SWEEP CAPSULE FOR OVER 1,000 C IRRADIATIONS

DIRECT MEASUREMENT OF FREE FISSION GAS PRESSURE IN OPERATING FUEL RODS

By

M. B. Reynolds

Work Performed Under

U. S. Atomic Energy Commission

Contract AT(04-3)-189
Project Agreement 17

2.11.1

DIRECT MEASUREMENT OF FREE FISSION GAS PRESSURE IN OPERATING FUEL RODS

The pressure transducer used in an experimental measurement of gas pressure in an operating fuel rod must contribute a negligible volume to the void system of the rod. This condition must be met in practice by use of a transducer of smallest possible volume connected to the fuel rod with a tube of minimum length and cross section. In an early attempt to measure pressure in UO_2-filled fuel rods, unbonded strain-gage-type electrical pressure transducers were used, located external to the reactor pressure vessel and connected to the fuel rods with stainless steel capillary tubing. This approach was unsuccessful because of gasket leakage in the pressure transducer connections. Several manufacturers were approached on the subject of a low-volume, all-welded pressure transducer suitable for operation at elevated temperature in a fast neutron field. When it was learned that no such device existed, it was decided to adapt a so-called Booth Cromer transducer to this application.

THE PRESSURE TRANSDUCER

The principle of the Booth-Cromer transducer is shown in Figure 1. If the electrical contact is so adjusted that contact is just made with zero unbalance pressure on the diaphragm, a very slight excess of pressure on the secondary (contact) side of the diaphragm will open the contact. The secondary pressure at the instant of contact opening is then equal to the primary pressure plus the slight differential required to open the contacts. This differential is negligible relative to the primary pressures to be observed. Actual pressure measurement may be carried out at any convenient distance from the device, using any desired type of pressure gage or transducer. The transducer developed is illustrated in Figure 2. The structural parts of the device are fabricated from type 304 stainless steel. The diaphragm is annealed stainless steel of approximately 0.002 inch thickness. Electrical insulating washers are of high density alumina (Lucalox). Since diaphragm movement is limited to a few thousandths of an inch within a semilenticular cavity, unbalance pressures in excess of 1500 psi can be tolerated from either side without exceeding the elastic limit of the thin stainless steel diaphragm. Sensitivity is of the order of 0.25 psi. Since electrical contact is made directly with the thin diaphragm some form of electronic relay must be used to limit contact current.

READOUT INSTRUMENTATION

The instrumentation used with such transducers is shown in Figure 3. Its operation is as follows: Assume that the selector switch (5) has connected the electronic relay (6) to the transducer associated with fuel rod number 1. With fill valve (14) open, pressure in the gas manifold (connected to the secondary side of all transducers) will rise slowly and uniformly at a rate set by inlet flow regulator (16). At the instant the manifold pressure balances the pressure in rod number 1, the contacts in the transducer associated with this rod will open.

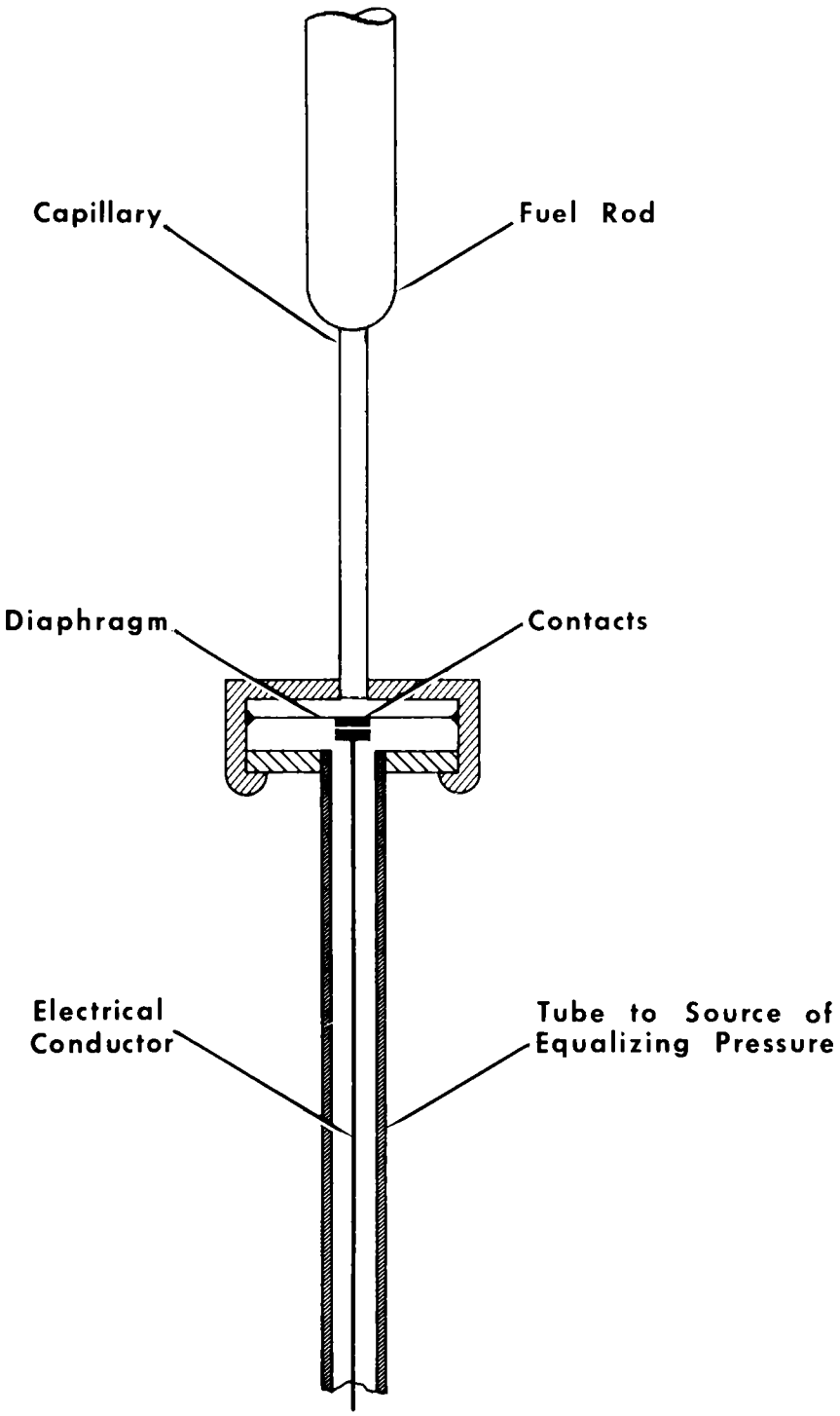

Figure 1. TRANSDUCER, SCHEMATIC

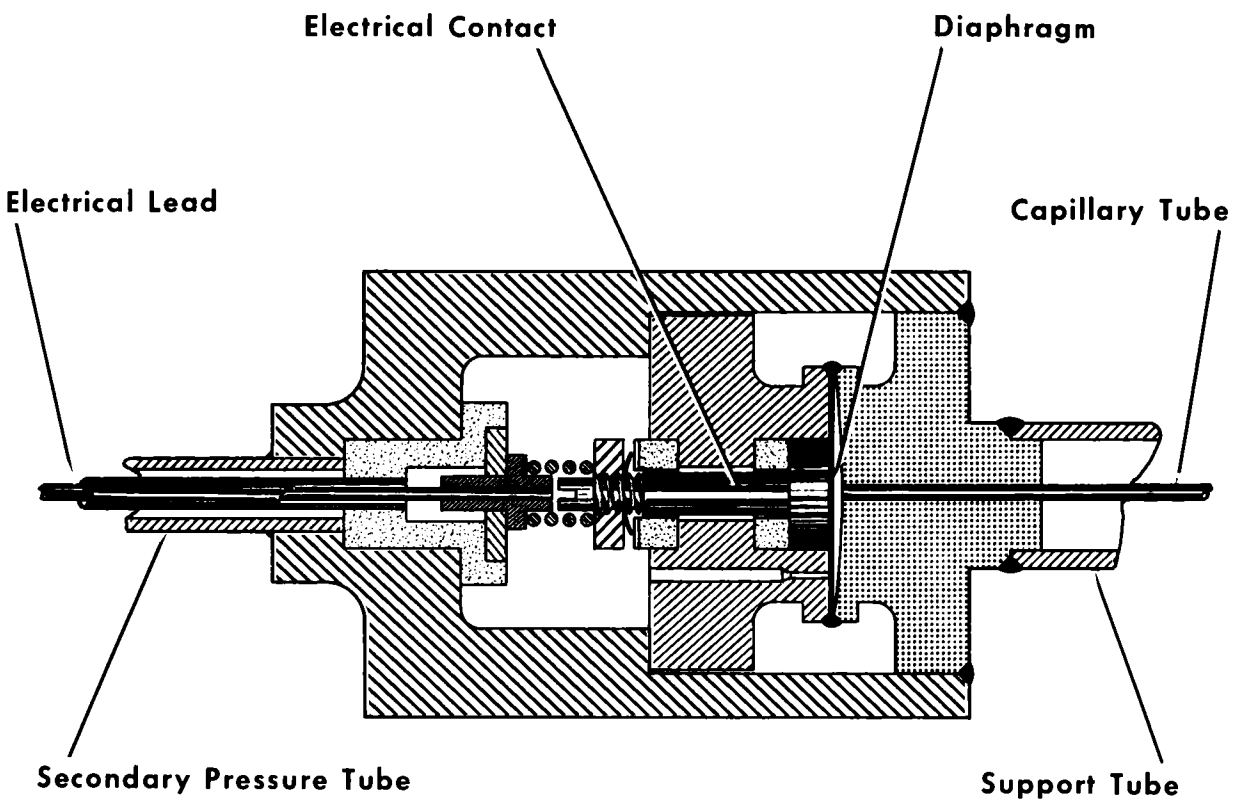

Figure 2. LONGITUDINAL SECTION OF TRANSDUCER

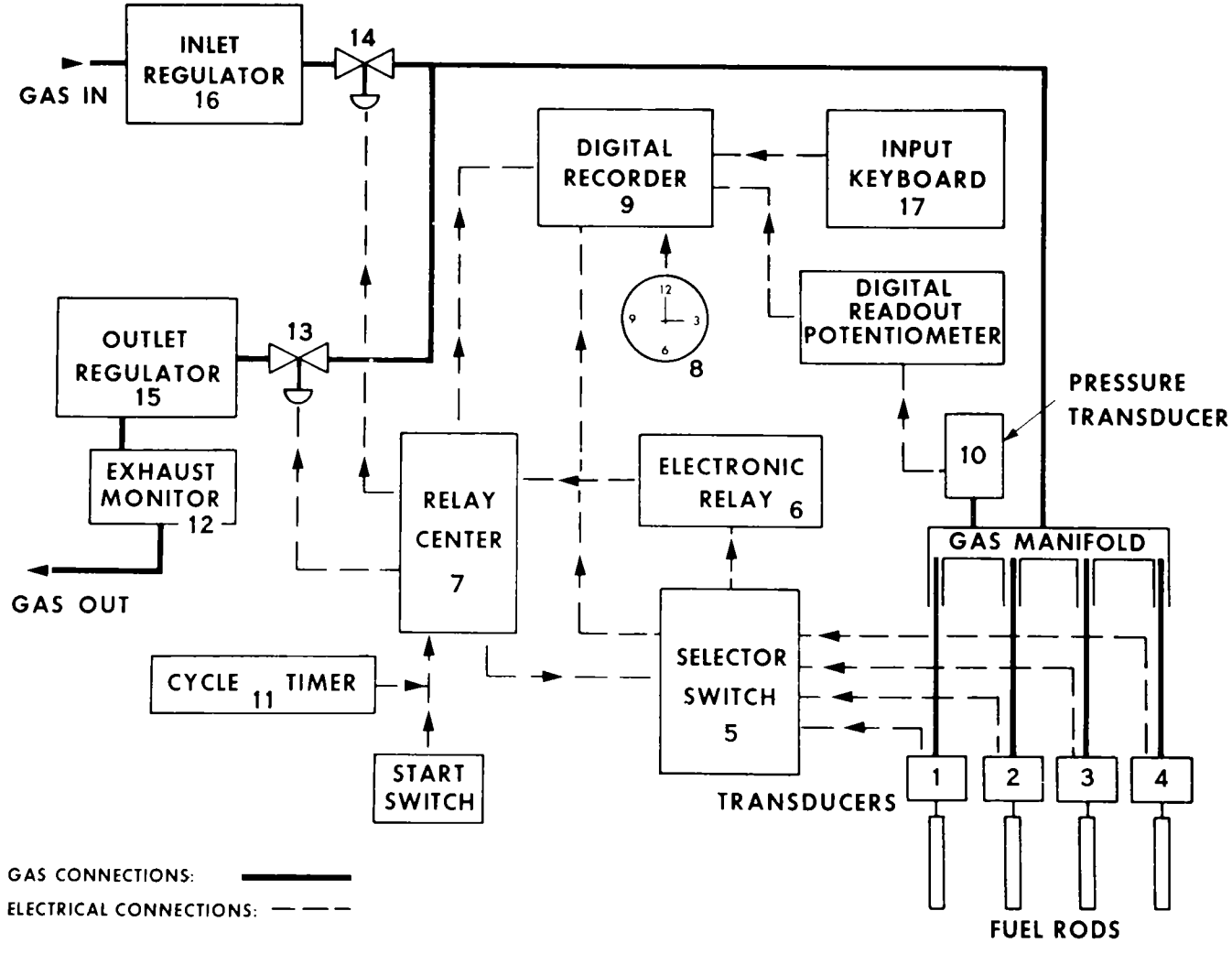

Figure 3. READOUT INSTRUMENTATION SCHEMATIC

Operating through the electronic relay (6) and relay center (7), the opening of these contacts will command printer (9) to print out the manifold pressure indicated by the electrical pressure transducer (10), the time indicated by clock (8), the number of the transducer* to which the selector switch is connected, and the reactor power level set on keyboard (17). Simultaneously the relay center will close valve (14) and open valve (13) permitting the manifold pressure to fall to atmospheric at a rate set by flow regulator (15). Upon activation by cycle timer (11), the relay center will close valve (13) , open valve (14) , and cause the selector switch (5) to step to the transducer connected to rod number 2. The manifold pressure will again increase until it balances the pressure in fuel rod number 2, at which time the contact in transducer (2) will actuate the printer and repeat the sequence of operations described above. These cycles will repeat indefinitely at a rate determined by the cycle timer (11), indicating the pressure in each fuel rod in turn. If it is desired to sample the pressures at a rate faster than that set by the timer, a cycle may be initiated by the operator by pushing "start" switch (18).

The digital recorder** used for recording data is capable of printing an eleven digit number upon receipt of an electrical print command signal. Each digit is printed by one of a series of eleven print wheels each of which normally rotates continuously but which may be stopped by the print command at any desired digit by application of the proper voltage to its electrical sensing system. A voltage divider connected to a regulated power supply provides a "staircase" of the voltage steps required to stop the print wheels at the different digits. Thus, if it is desired that a given print wheel print the digit "8", it is only necessary that its sensing system circuit be connected to the "8" step of the voltage "staircase". The eleven print wheels are electrically independent, each wheel recording the data supplied to its particular sensing circuit.

The output of the manifold pressure transducer is converted to the step-voltage signals required by the printer by means of a self-balancing DC potentiometer connected to a Veeder-Root Series 1607 Remote Data Readout Counter. Each number wheel of this counter contains a ten-point selector switch which connects the sensing circuit of a recorder print wheel to the step of the recorder voltage staircase corresponding to the digit indicated by the number wheel. The design of the DC potentiometer is such that full span of the slide wire is equivalent to one hundred revolutions of the counter shaft or 0 to 999 as printed out by three adjacent wheels in the recorder. A 1 rpm Telechron motor connected directly to a similar Veeder Root Series 1606 Remote Data Readout Counter permits the elapsed time to the nearest one-tenth hour to be recorded by five adjacent print wheels in the recorder. The remaining three print wheels are used to record the reactor power level to the nearest megawatt (two digits) and the number of the pressure transducer to which the readout circuit is connected.

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* In this case, number 1.
** Hewlett Packard 560 A.

FUEL ROD CHARACTERISTICS

In order to eliminate fuel cladding deformation as a variable, all fuel rods equipped with pressure transducers were fabricated with thicker-than-normal cladding. The fuel rods and all accessory parts exposed to reactor coolant were fabricated from annealed Type 304 stainless steel. Each fuel rod was installed in a special fixture which fitted into and extended the full length of a channel of the Vallecitos Boiling Water Reactor core. This fixture consisted of a bundle of five 1.125 inch OD by 0.035-inch wall tubes passing through square bulkheads which restricted coolant flow to the inside of the tubes. The fuel rods were 0.600 inch in outside diameter and were approximately 29 inches in active (fueled) length. Radial fins attached to top and bottom end plugs centered the fuel rods in the tubes of the channel fixture. Each fuel rod was connected to its respective pressure transducer by a length of 0.032 inch OD by 0.010 inch wall stainless steel capillary tubing. The transducer was supported mechanically a few inches above the reactor core by a heavy wall 3/8 inch diameter tube attached to the fuel rod upper end plug and to the lower end of the transducer. This support tube also contained the connecting capillary. The fuel element and transducer configuration are illustrated in Figure 4.

Difficulties with these transducers have been limited to electrical and balance pressure leads. On one occasion an open circuit developed in an electrical lead - a not uncommon occurrence in "mineral-insulated" cable of small diameter. On another occasion a balancing pressure lead was broken during a refueling operation, permitting reactor coolant water to enter the balancing pressure leads of all transducers. The remaining transducers remained functional in spite of the resulting lowered lead resistance.

CAPSULE FOR MEASUREMENT OF UO_2 THERMAL CONDUCTIVITY

By

D. H. Coplin
M. F. Lyons
B. Weidenbaum

Work Performed Under

U. S. Atomic Energy Commission

Contract AT(04-3)-189
Project Agreement 17

2.12.1

CAPSULE FOR MEASUREMENT OF UO_2 THERMAL CONDUCTIVITY

The measurement of uranium dioxide thermal conductivity has been made by many experimentors on both irradiated and unirradiated samples at temperatures below 1200 C. Due to the lack of a reliable temperature sensing device to measure fuel temperatures near the melting point of UO_2 (~ 2800 C), no experiments have been performed in which the conductivity was directly measured during irradiation at the very high fuel temperatures. Indirect methods have therefore been used to calculate thermal conductivity values of UO_2 at these temperatures. These methods rely to a large extent on the investigator's ability to relate post-irradiation fuel structures to fuel temperatures during irradiation.

The capsule experiment in progress at General Electric for the AEC-Euratom High Performance UO_2 program will provide a direct measurement of thermal conductivity. This is made possible by the use of a high temperature sensing device located in the center of the fuel. The device consists of a tungsten bulb connected by a tungsten capillary to a pressure transducer. The in-pile pressure transducer is a balancing type with a read-out system for the secondary pressure. Prior to insertion into the Trail Cable facility of the General Electric Test Reactor (GETR) the tungsten bulb - pressure transducer is calibrated out-of-pile with the instrumentation to be used for the in-pile test.

The irradiation capsule consists of 30-mil Zircaloy-2 cladding (0.565 OD) and UO_2 pellets sintered to 94-95 percent of theoretical density. The pellets are centerless ground to provide a six-mil diametral gap with the cladding. The lower pellets in the capsule will be solid to reduce the heat losses to the bottom end plug. The remaining pellets will be cored to an inside diameter of 150 mils. This is large enough to prevent any bending stresses on the bulb when the capsule is assembled with the pressure transducer. The fuel column in the capsule is approximately 6.5 inches long. The bulb is three inches long and has an outside diameter of 0.125-inch. The capillary which connects the bulb to the pressure transducer has an ID of 10 mils as compared to an ID of 90 mils for the bulb. This arrangement of diameters gives a system that has more than 90 percent of its volume in the bulb.

The pressure transducer is of the Booth-Cromer type and has been described by Reynolds.* Changes to his design consisted primarily of materials used to fabricate the piece. The method of operation is: electrical contact is made when the pressure in the bulb exceeds the line pressure. The electrical relay in the system opens a solenoid valve which allows the secondary side to balance the pressures and break the contact. A pressure transducer and a pressure gage are connected to this line to observe the change in pressures. The transducer is connected to a strip chart recorder that provides a continuous record of the pressures in the bulb.

* Refer to GEAP-4135

The fuel surface temperatures will be calculated from the heat generation of the capsule. This is obtained by measuring the rise in water temperature across the capsule by means of two pairs of thermocouples. These thermocouples have their outputs matched and recorded on a two-pen, temperature differential, strip chart recorder.

Figure 1 shows the layout of the capsule and holder for this experiment. The important parts are labeled with the exception of the thermocouple glands. These glands are placed at the junction of the thermocouples to permit the end to be bare and to prevent water from going into the insulation. Entry of water into the insulation would cause a short circuit. Water flow is from the top of the capsule through a flexible metal hose and back around the outside of the holder.

Many problems arose in the fabrication of the pressure transducer-tungsten bulb piece. The joining of tungsten and Zircaloy by brazing was a new technique that took considerable time to perfect. The severity of the problem was increased by the small size of the tungsten capillary. Other brazing problems necessitated changes in material for some parts of the transducer. These problems have been overcome and the first assembly has been completed. The out-of-pile testing is now in progress.

In the first in-pile experiment UO_2 pellets are being irradiated over the temperature range from 1800 C to 2800 C, hopefully within a twenty-minute period. Similar short-time tests will be run using large-grain-size UO_2 and pellets with columnar grains or essentially single-crystal UO_2. These pellets are to be heat treated out-of-pile to develop the various structures. A corresponding test will be made with a powder-filled capsule. The final two tests, one of pellet fuel and one of powder fuel, are to be of two-weeks duration. The entire series should give information not only for the determination of the thermal conductivity of UO_2, but also for evaluating differences in thermal conductivity arising from differences in grain structure and/or from irradiation.

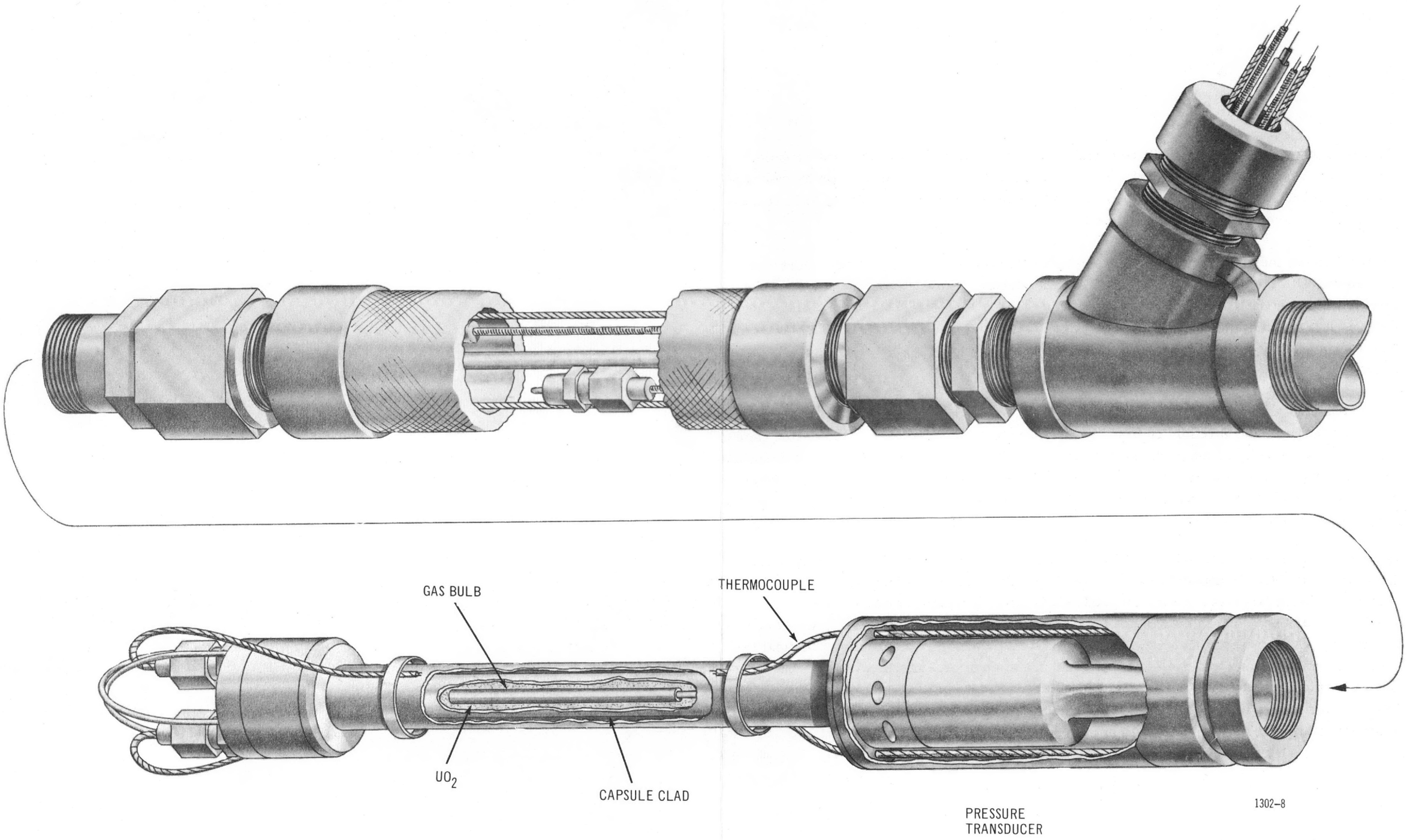


Figure 1. Trail Cable Capsule With Central Temperature Monitor

DESIGN, PERFORMANCE, AND REACTOR ENVIRONMENT INFORMATION FOR WAPD TEST CAPSULES*

R. C. Daniel
Bettis Atomic Power Laboratory
Westinghouse Electric Corp.

A. General

The information contained herein has been abstracted from a forthcoming report on measurements of the effective thermal conductivities of UO_2 and $ZrO_2 + UO_2$ fuels[1]. This information was a by-product of these experiments, and no tests were performed specifically to investigate neutron flux variations, thermocouple performance, or similar topics discussed herein. However, the data obtained provide a rather complete description of the MTR facility employed (A3ISE) and are probably representative of the magnitudes of neutron flux and gamma heat variations to be expected in other MTR facilities.

The capsule designs described for measuring the effective thermal conductivities of ceramic fuels have been thoroughly tested and were found to operate satisfactorily.

Considering the purpose of the present meeting, no discussion of the thermal conductivity data itself is planned. Those interested are referred to reference (1).

B. Description of Experiments

B-I. DESCRIPTION OF TEST ASSEMBLIES

Two types of thermal conductivity capsules have been used at Bettis Atomic Power Laboratory. One employed solid fuel pellets

operated at low power densities (WAPD 22) and the other hollow, high power density pellets (BETT 69). The WAPD 22 type of assembly has previously been described in detail[4] and is shown in Figure B-1. Briefly, it consisted of two type 304 stainless steel capsules joined end to end by connecting posts. Each capsule was 2.86 cm in diameter and 12.7 cm long. The fuel capsule had a 0.907 cm diameter, ground-finish bore into which 0.907 cm diameter fuel pellets were stacked to yield a total fuel length of 12.0 cm. The bore then was filled with inert gas and sealed by end plugs which were welded into position. The end plugs and UO_2 pellets had 0.170 cm diameter holes drilled through their centers to contain two Pt-10 Rh thermocouples for measurement of center temperatures. Three pairs of 0.170 cm diameter thermocouple wells were located azimuthally around the bore 120° apart. Each pair contained an inner (T_1) and outer (T_2) thermocouple well (6.35 cm deep) with centers nominally 0.635 cm and 1.245 cm from the center of the pellets, respectively. Sheathed, 0.165 cm diameter thermocouples were inserted into the center holes and into the wells in the capsule wall, and the top of the well was silver soldered to prevent entry of cooling water. The thermocouples employed to determine the radial heat flow through the capsule wall were iron-constantan.

The gamma heat capsule of the WAPD 22 type of assembly contained two iron constantan thermocouples in the capsule wall to measure radial flow of gamma heat generated in the stainless steel. The thermocouples for the fuel capsule passed through the bore of the gamma heat capsule and through individual silvered soldered joints in a transition piece at the bottom of the bore. The thermocouples were conveyed to the outside of the reactor through a lead tube attached to the top of the gamma heat capsule.

The BETT 69 type of assembly consisted of three fuel capsules made of "A" nickel. In this type of assembly the fuel pellets were 0.904 cm outside diameter and 0.752 cm inside diameter. These pellets were fitted over a 0.749 cm O.D., 0.690 in. I.D. nickel tube as shown in Figure B-2, and a nickel end piece of the same dimensions as the fuel pellets was fitted on each end. The resulting unit was 14.0 cm long with a fuel length of 12.0 cm. This unit was then inserted in a nickel capsule having a 0.907 cm bore so that the fuel was completely enclosed by the capsule walls, nickel end pellets, and the inner tube. The 14.0 cm long capsule was 2.86 cm diameter over the center 12.7 cm of its length, and 1.1 cm diameter for a 0.65 cm length at each end. The integral projections at each end were of comparable wall thickness to the inner tube and enabled this tube to be welded to the capsule.

After assembling each BETT 69 fuel capsule as described above, the ends of the inner tube and capsule were electron beam welded in vacuum. Each capsule, with the fuel pellets sealed in the evacuated annulus between the capsule and inner tube, was then leak tested and pressure bonded at 1600°F and 10,000 psi for 4 hours. (In order to determine the proper bonding conditions, a trial run was made using a dummy capsule with cracked fuel pellets. Cross-sectioning of this capsule revealed, as shown in Figure B-3, that both the inner clad and the heavier capsule wall deformed into the fuel surface irregularities during this bonding operation). After bonding, the bore of each inner nickel tube exhibited circumferential depressions at the ends of each fuel pellet which indicated that flow had occurred during the bonding operation. The deformation of the capsule wall

was negligible since the thermocouple wells (drilled before capsule assembly) were unaffected. In the first assembly of this type, a solid end plug was welded in the bottom end of each bore. The bore was then filled with powdered BeO to provide good heat transfer across the bore and thus assure that the region at the middle of bore would be isothermal during operation. (Calculations showed that thermal end losses due to the BeO were negligible). The center temperature in these capsules was measured by two chromel-alumel thermocouples which were pre-soldered into holes in the upper end plug so that they extended to the center of the capsule when the plug was placed into position. The positioning and welding of this end plug were performed in a helium atmosphere.

In the final assembly of the BETT 69-1 train, the three capsules were positioned end to end with nickel posts silver soldered between the opposing ends to properly space the capsules. The bottom capsule contained UO_2 (21 w/o enriched), the middle ZrO_2-34 w/o UO_2 (93 w/o enriched), and the top BeO-51 w/o UO_2 (93 w/o enriched). Two pairs of chromel-alumel thermocouples were inserted into wells (6.35 cm deep) in each capsule wall to measure the radial heat flow, and the tops of the wells were silver soldered as in the earlier assemblies. These pairs of thermocouple wells were positioned on opposite sides of the bore 180° apart. The center of the inner thermocouple of each pair (T_1) was nominally 0.635 cm, and that of the outer thermocouple (T_2) 1.245 cm, from the axis of the capsule. The thermocouples for the lower capsules were silver soldered in surface slots of the capsule above in order to maintain the 2.86 cm O.D. of the assembly.

These slots were positioned as far from the thermocouple wells as possible and their affect on the heat flux between the thermocouples is considered negligible. From the top capsule, the thermocouples entered the lead tube through individually sealed holes in a transition piece. The completed assembly consisted of the three fuel capsules, each having two center thermocouples and four heat flux thermocouples.

The thermocouples used in both the BETT 69 experiments were 347 S.S. sheathed, Al_2O_3 insulated, chromel-alumel thermocouples. These thermocouples were ordered to military specification and calibrated by comparison with a certified Bureau of Standards Pt-10 Rh thermocouple in the temperature range 100°C to 900°C. The thermocouples agreed within 1°C over most of the temperature range with maximum deviation of 3°C at high temperatures. The positions of the thermocouples in the capsule wall were checked before assembly by radiographing the capsule with 0.15 cm diameter tungsten wires inserted in the thermocouple wells. The thermocouple positions determined in this manner were used in all calculations.

A drawing of a BETT 69-4 capsule is shown in Figure B-4. The fabrication of these capsules differed somewhat from the earlier BETT 69 assembly described above. After pressure bonding, the bore of the inner tube was reamed to permit shrink fitting of alumina pellets. Gun reaming was employed to obtain bores which were uniform to $\pm$ 0.0003 cm (nominal 0.60 cm I.D.). The alumina pellets were then centerless ground to accurate diameters ($\pm$ 0.0003 cm) to fit each capsule. (These pellets had 0.18 cm diameter holes through their centers for the center thermocouples).

The UO_2 fuel capsule was heated to 550°C in flowing argon, amd then removed from the furnace for positioning of the alumina pellets in the bore. The alumina pellets in the ZrO_2-base fuel capsules could not be seated completely at 550°C and the shrink-fitting of these capsules was completed at 850°C. After cooling to room temperature, the alumina pellets and bore had a nominal 0.0007 cm diametral interference fit. In the BETT 69-4 capsules, one center thermocouple entered through the top cap and one through the bottom cap. After welding the end caps in place, the bore of each capsule was purged with inert gas immediately prior to silver-soldering the thermocouples in the end caps. In addition to the shrink-fitted alumina pellets, the major difference between the BETT 69-1 and BETT 69-4 assemblies was that the top capsule of BETT 69-4 assembly contained ZrO_2-46 w/o UO_2 (93% enriched) fuel pellets instead of BeO-51 w/o UO_2.

The thermocouples for each of the experiments were joined to #24 gage nylon-covered lead wires within the lead tube above the test assembly. These joints were located 10 to 15 feet above the top capsule. Each thermocouple wire was first soldered to the lead wire of the same composition, and then the joint was coated with potting compound. A 6 cm length of 0.475 cm O.D. stainless steel tubing, which had been previously swaged at one end to 0.18 cm I.D., was then slipped over the joint. The joint was completed by filling the stainless tube with potting compound (4 parts by weight Hysol 6020, 1 part by weight Hardener C; Houghton Laboratories, Olean, New York). Of the eighteen thermocouples in the BETT 69-4 experiment, indications of erratic readings or open circuits were observed for all three thermocouples having potted

connections within 10 1/2 feet of the center of the top fuel capsule, but for only one of the fifteen at a greater distance from the core during eighteen months of irradiation (see Section B-II).

B-II. IN-PILE OPERATION

All of the experiments were irradiated in the A31SE facility of the MTR. The test procedure during startup consisted of continuously recording the temperatures of all thermocouples as the MTR was raised to full power (40 MW) in 5 MW steps. This permitted the effective thermal conductivity to be determined as a function of temperature during startups. During full power operation, the temperatures of all thermocouples were continuously recorded and in most cases the effective thermal conductivities were calculated from daily readings.

The WAPD 22-11 experiment (UNO_2 in a 304 S.S. capsule) operated for 45 days (4.0×10^{20} thermal nvt) without being moved and yielded both gamma heating and neutron flux information as given in Section C.

The BETT 69-1 experiment operated for 2-1/2 months (2.0×10^{20} thermal nvt for the UO_2 capsule). However, the temperatures in the ZrO_2-34 w/o UO_2 capsule began to rise on the ninth day and it was necessary to raise the experiment higher in the facility to remove this capsule from the peak neutron flux position. In spite of this, the temperature continued to rise and the next day the assembly was again elevated. As a result of several such moves, neutron fluxes could be calculated from the power densities at a sufficient number of positions to establish the axial flux variation as given in Section C. The later successful operation of the BETT 69-4 experiment showed that low fuel-capsule contact conductance was responsible for the rapid increase in

center temperatures of the BETT 69-1 experiment.

The WAPD 69-7 experiment was performed using the proven WAPD-22 capsule design to determine whether the increasing temperatures in the ZrO_2-34 w/o UO_2 capsule noted above were due to decreasing conductivity of this fuel. The WAPD 69-7 experiment operated at three different axial positions for a total of 43 days. As shown in Section C, both gamma heating and neutron flux data were obtained throughout the irradiation period.

The BETT 69-4 experiment operated for eighteen months (about 8 x 10^{20} fast nvt for the UO_2 capsule). During a given reactor cycle (3 weeks), the temperatures of the bottom capsule (UO_2) decreased slightly as the MTR control rods were raised. During the same period, the temperatures of the middle capsule (ZrO_2-34 w/o UO_2) increased slightly. During some MTR cycles, depending on control rod programming, the center temperatures of the top capsule (ZrO_2-46 w/o UO_2) cycled about 100°C (from its average value near 700°C) as often as once every five minutes and, in addition, underwent a significant increase during each reactor cycle. These changes in temperatures had no apparent affect on the measured conductivities. The concurrent measurements of heat fluxes and fuel temperatures yielded similar values of the effective thermal conductivity at both the higher and lower full power temperatures.

The only difficulty encountered with the BETT 69-4 experiment during the first four months of operation was that one thermocouple in the wall of the UO_2 capsule became erratic after 10 days of operation, and was jumpered. When the thermocouple was reconnected during a later shutdown in order to locate and repair the cause of the erratic readings, it performed properly. All subsequent readings of this thermo-

couple were consistent with other thermocouple readings.

After four months of operation, one of the center thermocouples of the ZrO_2-34 w/o UO_2 fuel capsule became erratic, and intermittently gave the same reading or a much lower reading than the other center thermocouple in this capsule. Resistance measurements showed the erratic thermocouple to have an open circuit and its readings at subsequent times were not employed in calculations. However, the intermittent periods during which it yielded the same readings as the second center thermocouple are considered as a verification of the latter's accurate operation throughout the irradiation.

After one year of operation, one outer heat flux thermocouple of the ZrO_2-46 w/o UO_2 capsule became erratic, and intermittently gave the same reading or a much lower reading than the corresponding thermocouple of the other heat flux pair. Since it had initially agreed with this corresponding thermocouple, the latter's readings were used for both heat flux pairs in subsequent calculations.

As noted in Section B-I, two of the three thermocouples which gave erratic readings had their potted lead wire connections located within 10 1/2 feet of the center of the top fuel capsule, while only one of the fifteen at a greater distance became erratic; this was the one noted above which was later reconnected and operated satisfactorily. The third thermocouple having its potted connection within 10 1/2 feet of the top capsule indicated an open circuit at one time but its readings were always consistent with a companion thermocouple (center thermocouples of the ZrO_2-46 w/o UO_2 capsule). Since the center of the top capsule was 11 in. below the top of the core, these data indicate that

gamma and fast neutron damage was sufficiently great at distances up to 9 1/2 feet from the top of the reactor core to destroy the integrity of the thermocouple potted connections.

The differences between the readings of the two center thermocouples of the BETT 69-4 capsules increased sugnificantly during the irradiation. The two initially agreed within 2 to 6°C, but differences of 2°C to 10°C occurred after five months of irradiation and differences of 2°C to 20°C occurred after about twelve months. Since one of each capsule's pair of center thermocouples consistently yielded lower readings than the other, this indicates that at least one of the two center thermocouples was not operating properly at the later times; however, since the two inoperative heat flux thermocouples described above intermittently yielded correct readings, it is assumed that these two center thermocouples were also correct when they were in agreement. This implies that the maximum error in the average center temperature was about 10°C which would only cause a scatter of about 5% in the effective conductivity measurements at full reactor power since the temperature differences across the fuels were greater than 200°C even at high fuel depletions. Thus, these results indicate that chromel-alumel thermocouples, prepared as described above, can be used with reasonable accuracy at least to exposures of 8×10^{20} fast nvt at temperatures near 500°C (the approximate center temperatures of the BETT 69-4 capsules after high fuel depletion). It appears that all thermocouple malfunctions can be attributed to deterioration of the potted lead wire connections, and that the chromel-alumel thermocouples themselves operated satisfactorily throughout all the experiments discussed

above. It should also be pointed out that inaccurate thermocouples apparently always indicate a lower temperature than the actual temperature.

In addition to the thermocouple errors discussed above, two other sources of error become increasingly significant in fuel conductivity experiments at high fission depletions. Fortunately, these errors, due to changes in capsule conductivity and changes in power distribution, influence the experimental values in opposite directions. As discussed in reference (1), the thermal conductivity of the nickel capsules may have decreased up to 5% during irradiation of the BETT 69-4 experiment. Since it was assumed constant, this would cause the experimental values of effective conductivity to be up to 5% too high near the end of the irradiation. However, it is also shown in reference (1) that changes in the flux perturbation caused an apparent decrease in effective conductivity of about 3.5% between 0 and 24 x 10^{20} fissions/cc. Therefore, these two errors tend to compensate for each other, and their combined error is considered to have a negligible effect on the effective thermal conductivity measurements made using the capsules described herein.

B-III THERMAL CONDUCTIVITY OF CAPSULE MATERIALS

A literature survey was conducted to determine the thermal conductivity of the capsule materials. The data obtained are shown in Figure B-5 and B-6 as functions of temperature, together with references, for the information of persons interested in these capsule materials.

B-IV THERMOCOUPLE ERRORS DUE TO UNCERTAINTY IN RADIAL LOCATION

Throughout all of the present experiments differences of up to about 10% were noted between the heat fluxes indicated by different

pairs of thermocouples around the capsule's circumference, i.e., different thermocouples at the same radial distance from the fuel indicated different temperatures by as much as about 10°C. These differences were approximately constant with time so that one pair of thermocouples in the capsule wall always indicated a higher heat flux than a corresponding pair at another position around the circumference. Values of effective thermal conductivity calculated from the different pairs of thermocouples exhibited a spread of up to about ± 20% from the average value because such calculations depend on the absolute temperature of the capsule bore as well as on the heat flux calculated from the temperature difference across a given pair of thermocouples.

It is believed that this spread between values calculated from different pairs of thermocouples in a given capsule was caused primarily by thermocouples touching the walls of their wells. Although the calculations assumed that the thermocouple read the temperature that would exist at the center of the empty well, the temperature actually read would be that of the point of contact. The major part of the temperature drop across the well (up to about 30°C) would be dissipated across the gap between the thermocouple and the wall opposite the point of contact. J. M. Markowitz (WAPD) found that this effect caused errors up to 30% in the radial heat flux when he used a similar thermocouple arrangement in out-of-pile thermal diffusivity experiments. In order to evaluate the potential error in the present experiments, sample calculations were performed assuming the recorded temperatures to be those of the wall of the thermocouple well nearest the fuel, or most remote from the fuel. These calculations showed that the thermocouple pair which yielded

the highest values of effective thermal conductivity agrees with the lower values of the other two pairs if it is assumed that the inner thermocouple of the former pair read the temperature at the inner wall of its well. Conversely, the pair yielding the lowest values confirms the high values if the inner thermocouple of the pair is assumed to read the temperature of the outer wall of its well. This illustrates that the observed spread could have resulted from thermocouple positioning as proposed above. It should be pointed out that the probability is higher for the thermocouple to contact the well at a point on the same radius as its center than for it to contact either the inner or outer extremes. Therefore, the average of values calculated from individual thermocouple pairs should be more nearly correct than the extreme values obtained.

The error caused by thermocouple positioning probably can be eliminated in future experiments by performing an out-of-pile calibration of the assembled capsules. If a rod through the capsule bore is electrically heated, the thermocouple temperatures as a function of heat flux can be obtained directly. Such a calibration would eliminate any errors due to the capsule conductivity and thermocouple calibration as well as the spread caused by thermocouple positioning. Alternatively, the thermocouples may be brazed within their wells, or both brazing and electrical calibration may be performed.

Table B-1

Calculated And Experimental Neutron Fluxes In The A31SE Facility Of The MTR

Experiment	w/o $U^{235}O_2$	Position Considered	Calculated Fraction Of Unperturbed Flux	Maximum Calculated $\phi \times 10^{-14}$	Maximum Experimental $\phi \times 10^{-14}$
BETT 69 (Ni Capsule)	21.0	Outside Capsule Surface	0.88	2.2	-
		Inside Capsule Surface	0.39	1.0	-
		Effective in Fuel	0.27	0.7	0.5
WAPD 22-11 (304 S.S. Capsule)	0.7	Outside Capsule Surface	0.92	2.3	-
		Inside Capsule Surface	0.58	1.5	-
		Effective in Fuel	0.53	1.3	1.1

C. Calibration Of The Irradiation Facility (A3LSE Of The MTR)

C-1. GAMMA HEATING

In the WAPD-22 type of fuel capsule, approximately one-half of the total heat output was generated by gamma heating in the capsule wall. Since roughly one-half of this gamma heat originated between the radii of the heat flux thermocouples, an accurate value of gamma heating is required for the thermal conductivity calculations. An error of 40% in the gamma heating results in an error of approximately 10% in the calculated thermal conductivity. Since the ratio of total heat to gamma heat was about 6 to 1 in BETT 69 capsules, an error of 40% in gamma heat causes a conductivity error of about 3% for this type of capsule.

The values used for gamma heating in the WAPD-22 calculations first reported[4] were measured simultaneously with the temperatures in the fuel capsule. However, the measurement position at the center of the gamma heat capsule was 19 cm above the center of the fuel capsule. Later measurements obtained by lowering the WAPD 69-7 assembly to the bottom of the facility revealed that the gamma heating at the usual fuel position was about 80% higher than at the usual position of the gamma heat capsule. The variation of gamma heating with axial position is shown in Figure C-1. The axial positions of various fuel capsules are also shown.

Typical measured temperatures in the WAPD 69-7 gamma heat capsule were 132°C and 96°C during MTR cycle 138, and 107°C and 71°C during cycle 140. The latter values yield approximately the same surface temperature (57°C) as the WAPD 22-11 gamma heat capsule for the same gamma

heating and are therefore consistent. However, the values for cycle 138 indicate an excessively high surface temperature (82°C) for the apparent gamma heating; therefore, the values plotted at the 20-in. level in Figure C1 are probably too low. Measured temperatures of 141°C and 57°C during cycle 139 yield a surface temperature (67°C) consistent with the higher gamma heating for this cycle.

The relationship between MTR power and gamma heating is shown in Figure C3. This figure shows that the gamma heating is directly proportional to the reactor power. Similarly, Figure C2 shows that the temperature difference between heat flux thermocouples is directly proportional to the reactor power. Therefore, the gamma heating at any power level can be obtained by multiplying the gamma heating at full power by the ratio of the "heat flux ΔT" at the given power to the corresponding ΔT at full power.

The changes in gamma heating with time during the reactor cycle are shown in Figure C4. These changes correspond approximately to the changes in neutron fluxes with time.

For calculational purposes, the initial full power gamma heating for each capsule was read from the line shown in Figure C1. The temperature difference between a given pair heat flux thermocouples at this time was designated ΔT_o and this inital value for the gamma heating was designated P_{OG}. The gamma heating at all other times was taken as $(\Delta T/\Delta T_o)\ P_{OG}$ where ΔT was the temperature difference between the given pair of heat flux thermocouples at the time being considered. It is estimated that the value of gamma heating obtained in this manner is accurate to $\pm$ 20%. This causes a maximum error of $\pm$ 5% in the effective

thermal conductivities of the WAPD 22-11 and WAPD 69-7 experiments, and a maximum error of ± 2% for the BETT 69-1 and BETT 69-4 experiments. Calculations based on both the "maximum" and "minimum" values in Figure C1 were performed for capsules at positions below those where the gamma heat had been measured. The "maximum" values are considered to be more nearly correct.

C-II. THERMAL NEUTRON FLUX

The Maxwellian Most Probable neutron fluxes at full reactor power are shown in Figure C5, as a function of the axial position and the macroscopic cross section of the fuel capsule, as calculated from the average measured heat fluxes. As shown by the solid and dashed lines, the neutron flux near the top of the facility increases during the reactor cycle while the flux near the bottom decreases. As mentioned previously (Section B-II) the fluxes near the top of the facility also undergo significant fluctuation during the reactor cycle. The period of these cyclic changes varies from 5 minutes to several hours. (Both pairs of heat flux thermocouples and the center thermocouples of the top BETT 69-4 capsule substantiated these changes). All these flux variations are believed to be directly related to the control rod programming and changes in positioning of an automatic regulating rod located between the MTR core and the A31SE facility.

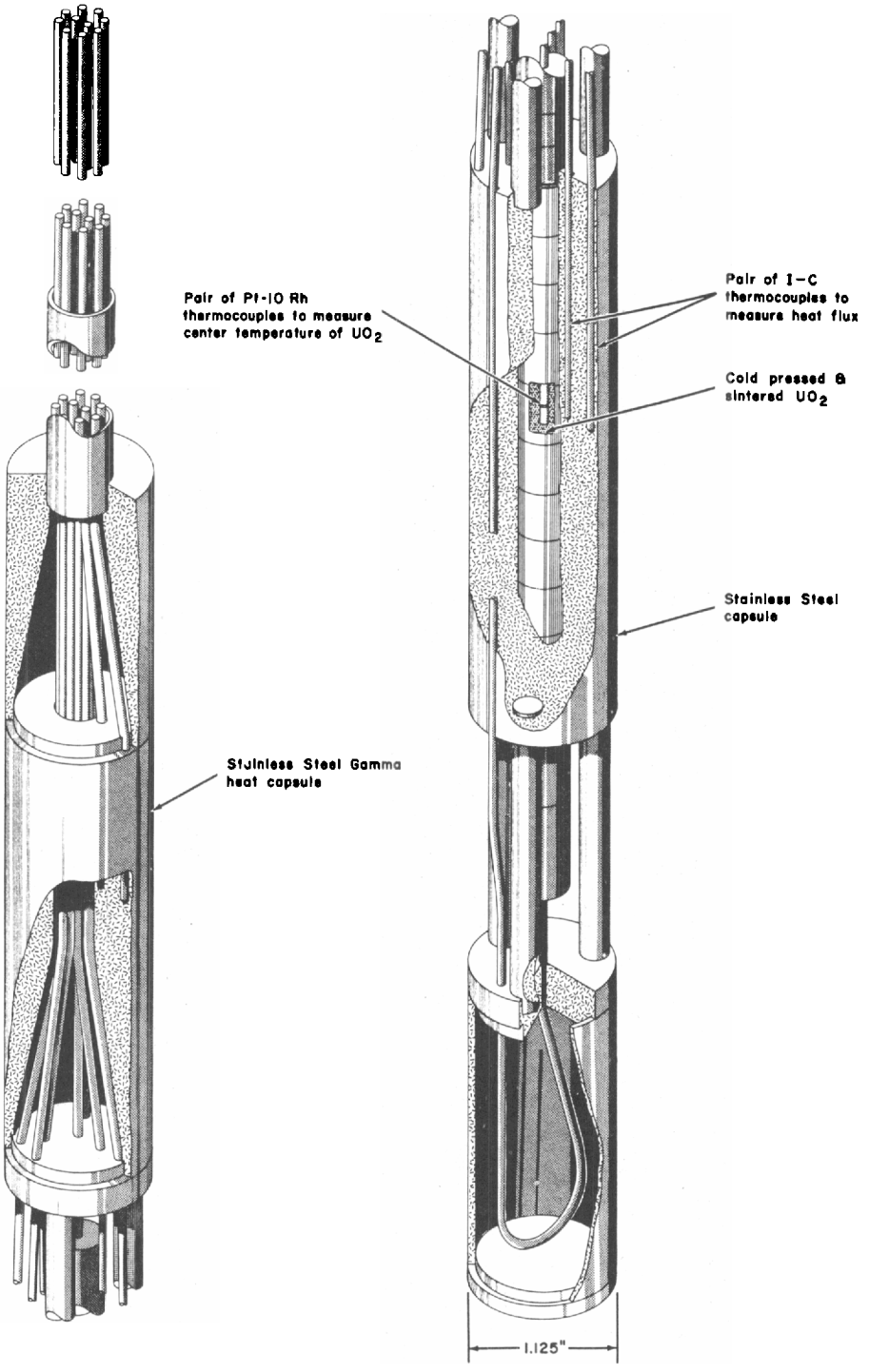
Results of calculations of the neutron flux perturbations caused by the BETT 69 and WAPD 22-11 capsules are shown in Table B-I. The Lewis method[5] of calculating the perturbation was employed. The calculated neutron fluxes are based on a reported maximum unperturbed flux of 2.5×10^{14} in this facility, and are about 20% higher than the

experimental values. This may be within the accuracy of the perturbation calculation but probably indicates that the maximum unperturbed flux is actually about 2.0×10^{14} nv.

The Maxwellian Most Probable neutron fluxes and reported fission depletions were based on a power constant of 3.54×10^{10} fissions/watt-sec, and were calculated using microscopic cross sections of 580 barns for fission and 687 barns for absorption in order to obtain fluxes directly comparable to the Maxwellian Most Probable fluxes reported by MTR. These values are used, instead of Maxwellian Average quantities, because the Maxwellian Most Probable cross sections are independent of temperature and the resulting neutron fluxes are valid for the existing MTR water temperature.

References

1. R. C. Daniel and I. Cohen, "In-Pile Effective Thermal Conductivity Of Oxide Fuel Elements To High Fission Depletion", WAPD 246 (December, 1963)
2. J. L. Daniel, J. Matolich, Jr., and H. W. Deem, "Thermal Conductivity Of UO_2", HW-69945 (September, 1962)
3. A. M. Ross and R. L. Stoute, "Thermal Contact Conductance Between UO_2 and Zircaloy-2", AECL Report CRFD-1075 (June, 1962)
4. I. Cohen, B. Lustman, and J. D. Eichenberg, "Measurement of the Thermal Conductivity of Metal-Clad Uranium Oxide Rods During Irradiation", J. Nucl. Matls. 3, #3, (1961), pp. 331-53
5. W. B. Lewis, "Flux Perturbations by Material Under Irradiation", Nucleonics, 13, p 82, (October, 1955)



IN-PILE TEMPERATURE MEASUREMENT CAPSULE

Figure B-1 - Cut-away view of a WAPD 22 capsule

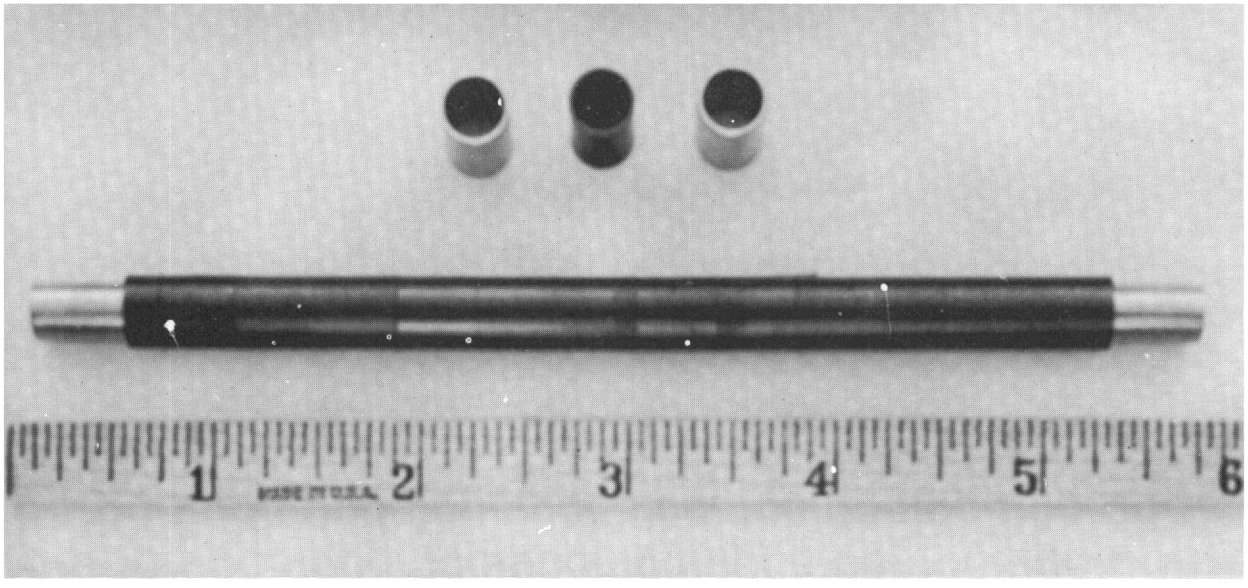


Figure B-2: Assembly of BETT 69-1 Pellets on Inner Tube

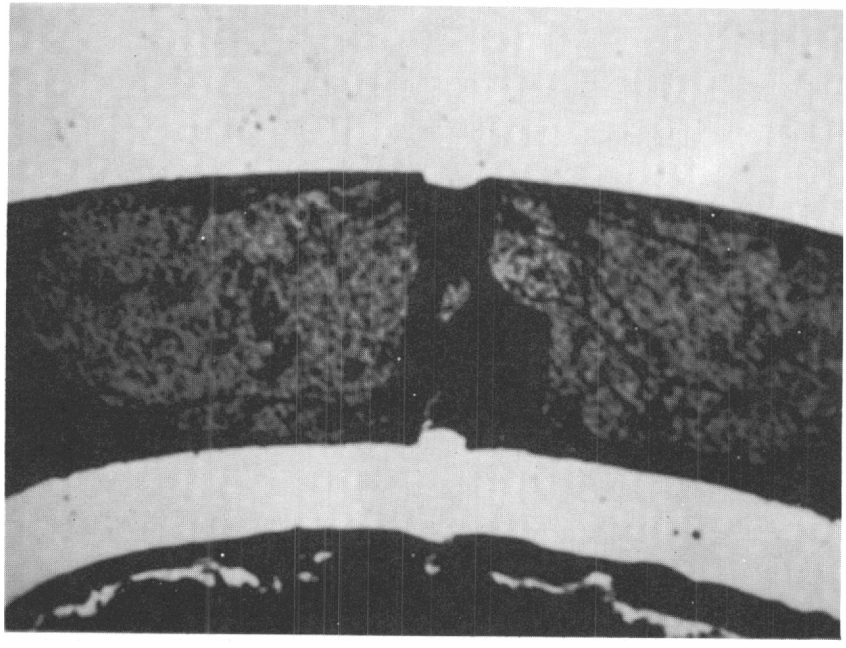
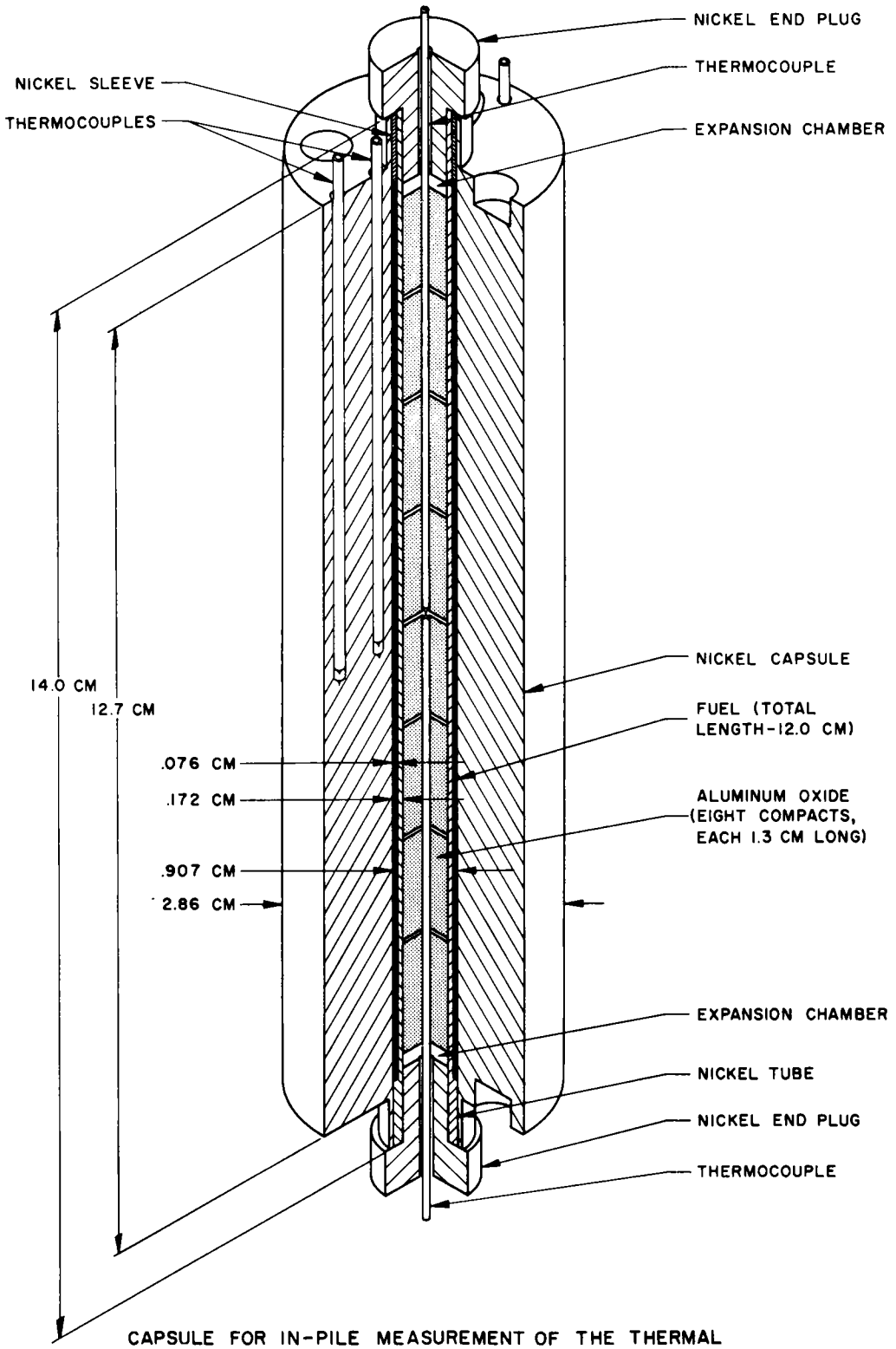


Figure B-3: Cross section of the fuel wall in a dummy capsule fabricated as a BETT 69-1 mock-up. Note the flow of nickel into the crack during pressure bonding.

Sample #229D-1
Mag. 50X



CAPSULE FOR IN-PILE MEASUREMENT OF THE THERMAL CONDUCTIVITY OF OXIDE FUELS TO HIGH FISSION DEPLETIONS (BETT 69-4)

Figure B-4

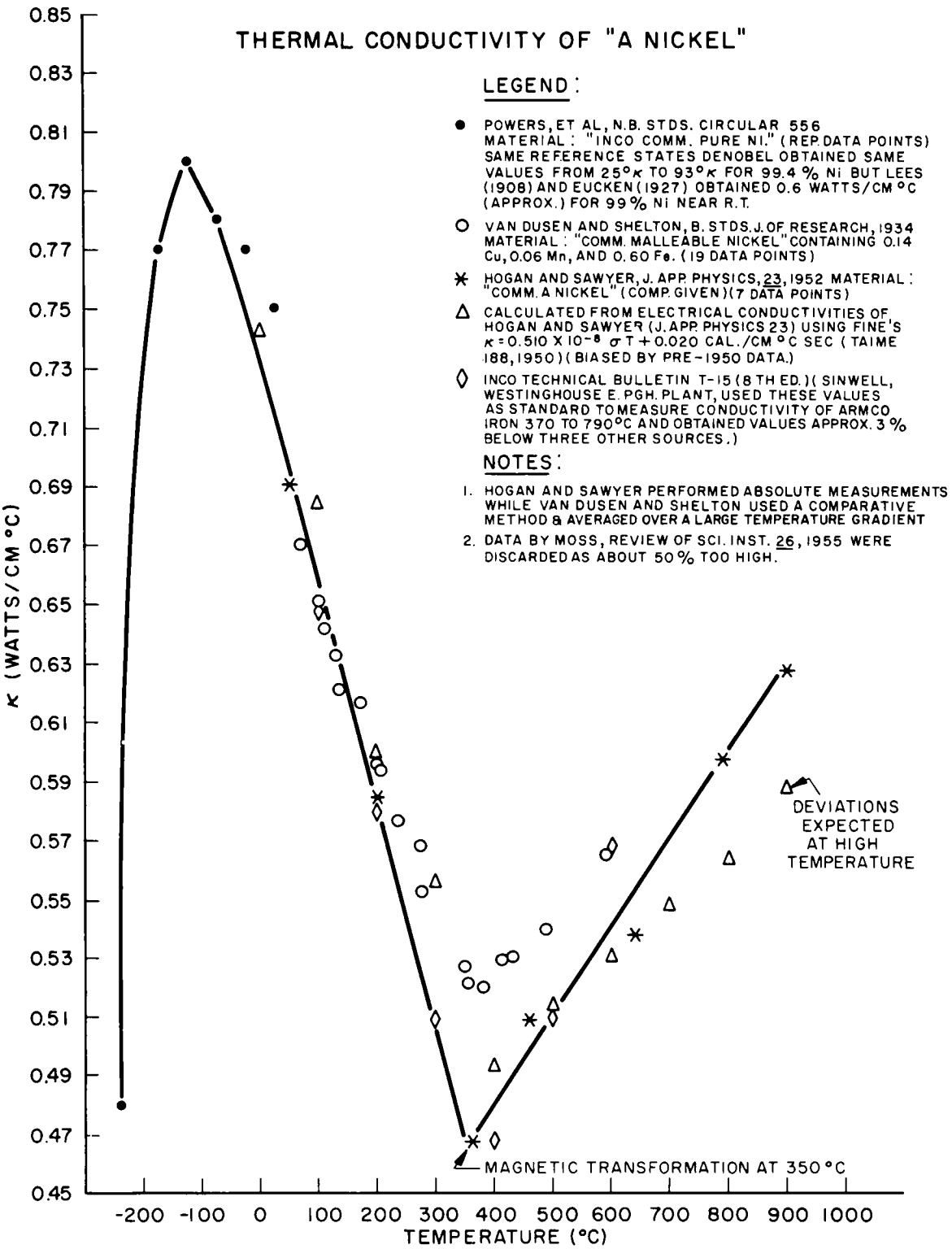


Figure B-5

2.13.24

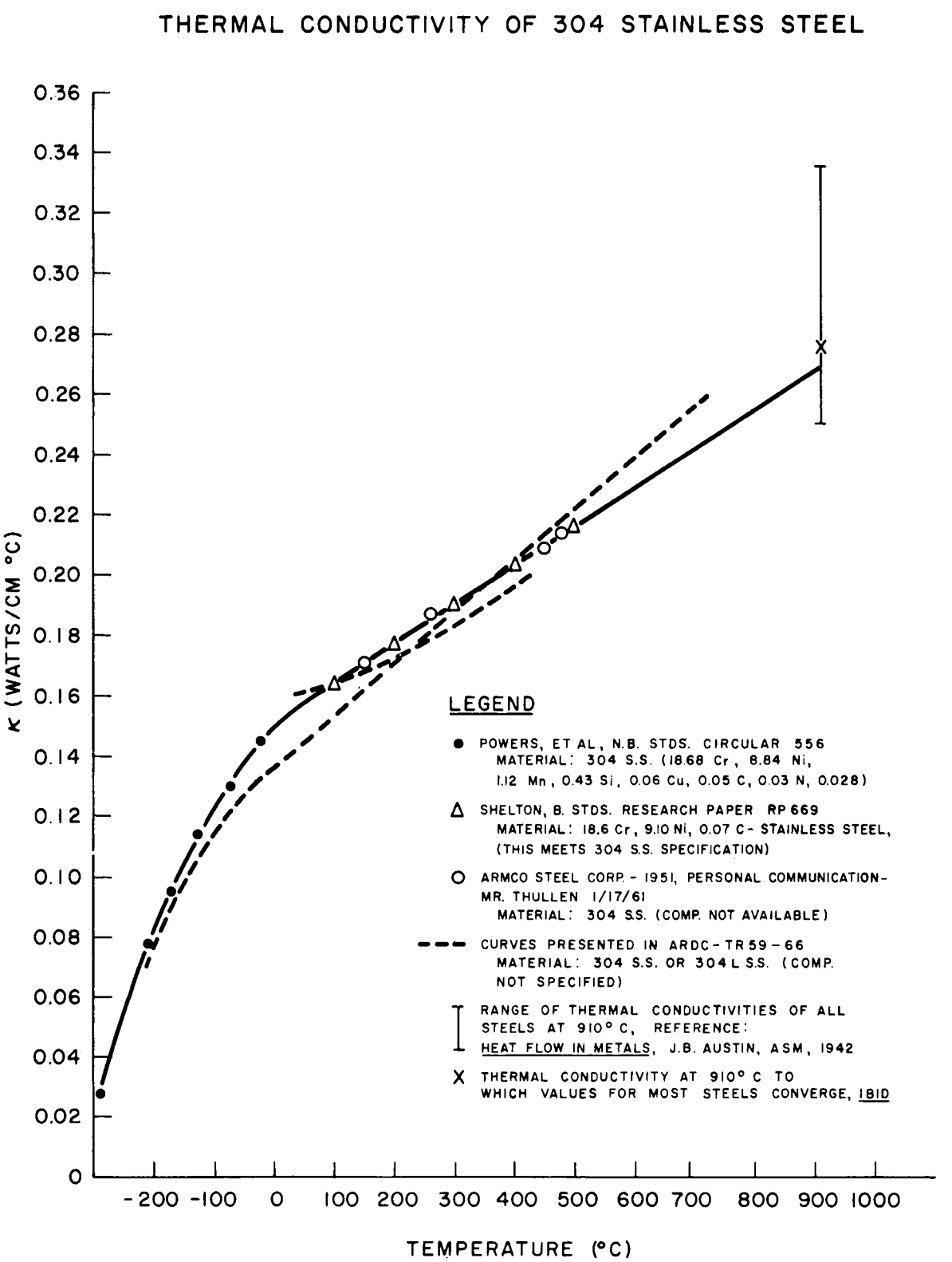
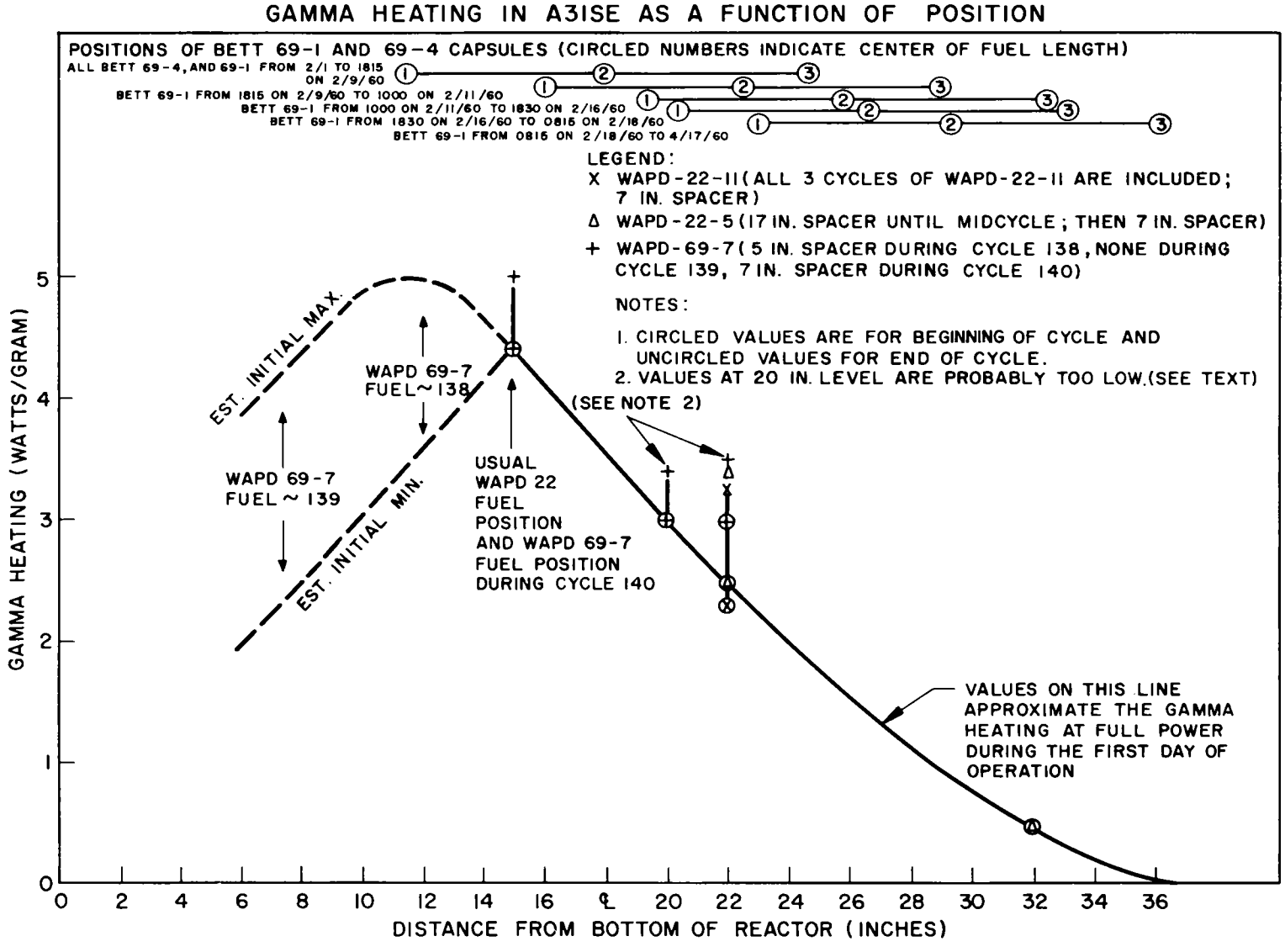


Figure B-6

2.13.25

Figure C-1


GAMMA HEATING IN A3ISE AS A FUNCTION OF POSITION
POSITIONS OF BETT 69-1 AND 69-4 CAPSULES (CIRCLED NUMBERS INDICATE CENTER OF FUEL LENGTH)
ALL BETT 69-4, AND 69-1 FROM 2/1 TO 1815 ON 2/9/60
BETT 69-1 FROM 1815 ON 2/9/60 TO 1000 ON 2/11/60
BETT 69-1 FROM 1000 ON 2/11/60 TO 1830 ON 2/16/60
BETT 69-1 FROM 1830 ON 2/16/60 TO 0815 ON 2/18/60
BETT 69-1 FROM 0815 ON 2/18/60 TO 4/17/60
LEGEND:
X WAPD-22-II (ALL 3 CYCLES OF WAPD-22-II ARE INCLUDED; 7 IN. SPACER)
Δ WAPD-22-5 (17 IN. SPACER UNTIL MIDCYCLE; THEN 7 IN. SPACER)
+ WAPD-69-7 (5 IN. SPACER DURING CYCLE 138, NONE DURING CYCLE 139, 7 IN. SPACER DURING CYCLE 140)
NOTES:
1. CIRCLED VALUES ARE FOR BEGINNING OF CYCLE AND UNCIRCLED VALUES FOR END OF CYCLE.
2. VALUES AT 20 IN. LEVEL ARE PROBABLY TOO LOW. (SEE TEXT)
EST. INITIAL MAX.
EST. INITIAL MIN.
WAPD 69-7 FUEL ~ 138
WAPD 69-7 FUEL ~ 139
USUAL WAPD 22 FUEL POSITION AND WAPD 69-7 FUEL POSITION DURING CYCLE 140
(SEE NOTE 2)
VALUES ON THIS LINE APPROXIMATE THE GAMMA HEATING AT FULL POWER DURING THE FIRST DAY OF OPERATION
GAMMA HEATING (WATTS/GRAM)
0
1
2
3
4
5
0 2 4 6 8 10 12 14 16 ℄ 20 22 24 26 28 30 32 34 36
DISTANCE FROM BOTTOM OF REACTOR (INCHES)

TYPICAL DATA RELATING MTR POWER AND MEASURED ΔT

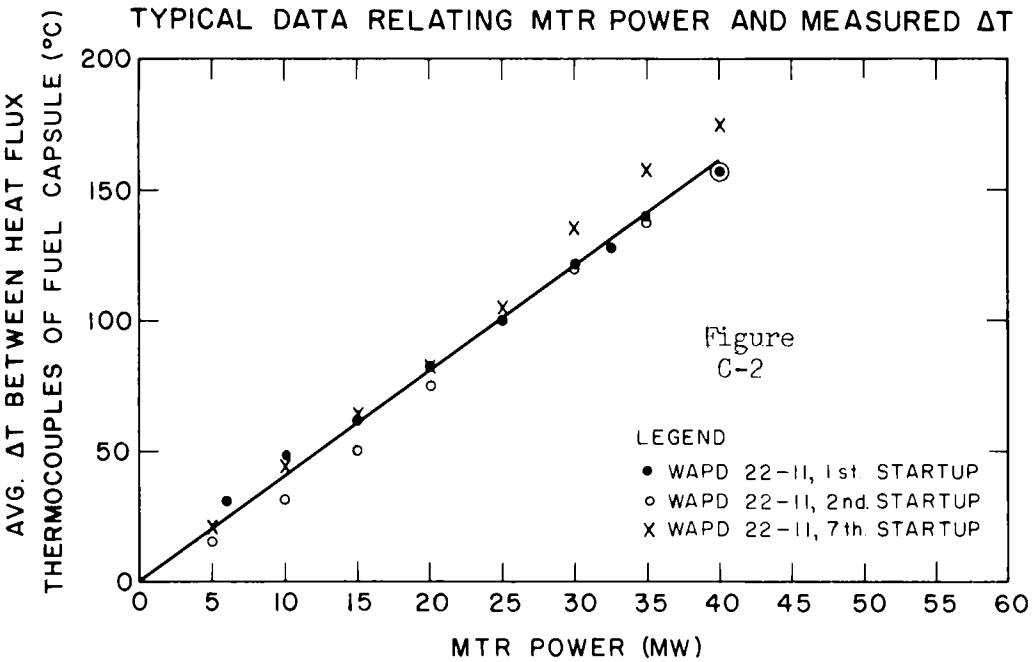


Figure C-2

TYPICAL DATA RELATING MTR POWER AND A3ISE GAMMA HEATING

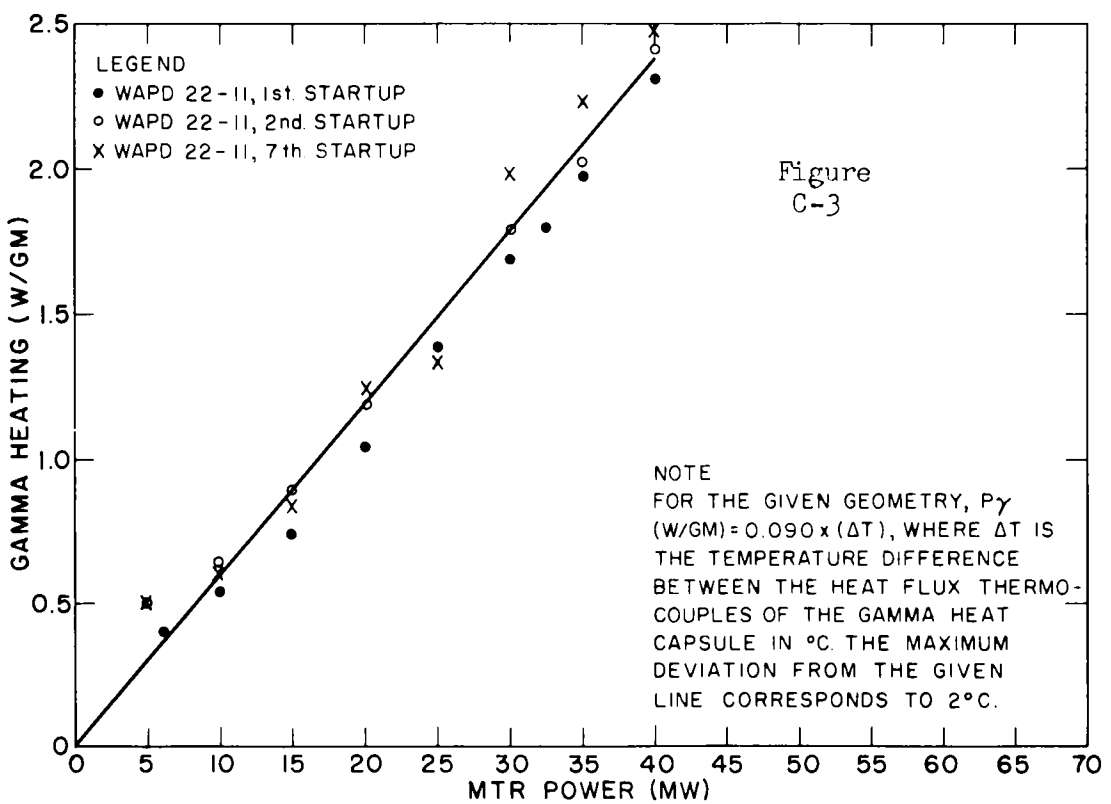


Figure C-3

Figure C-4

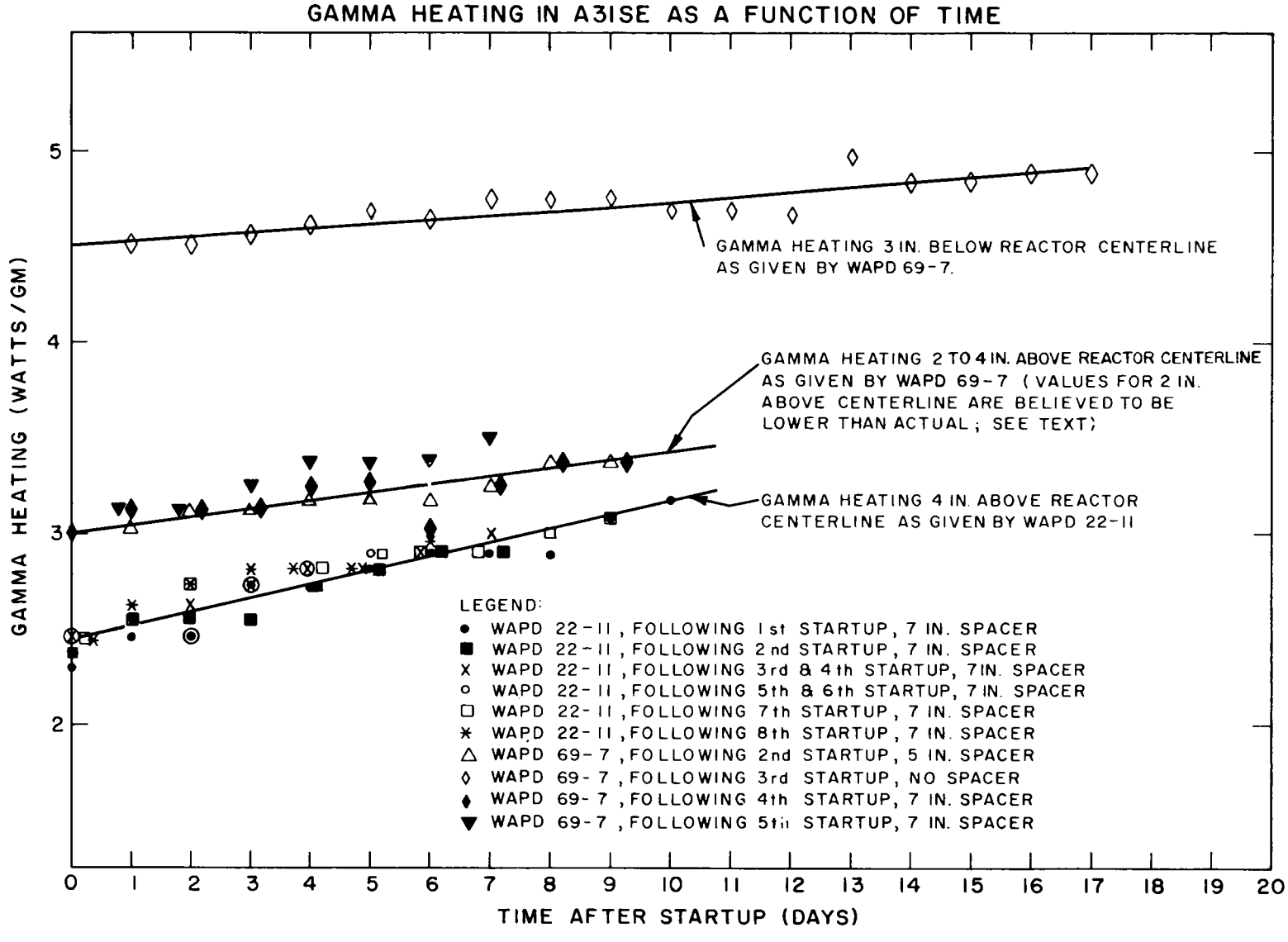
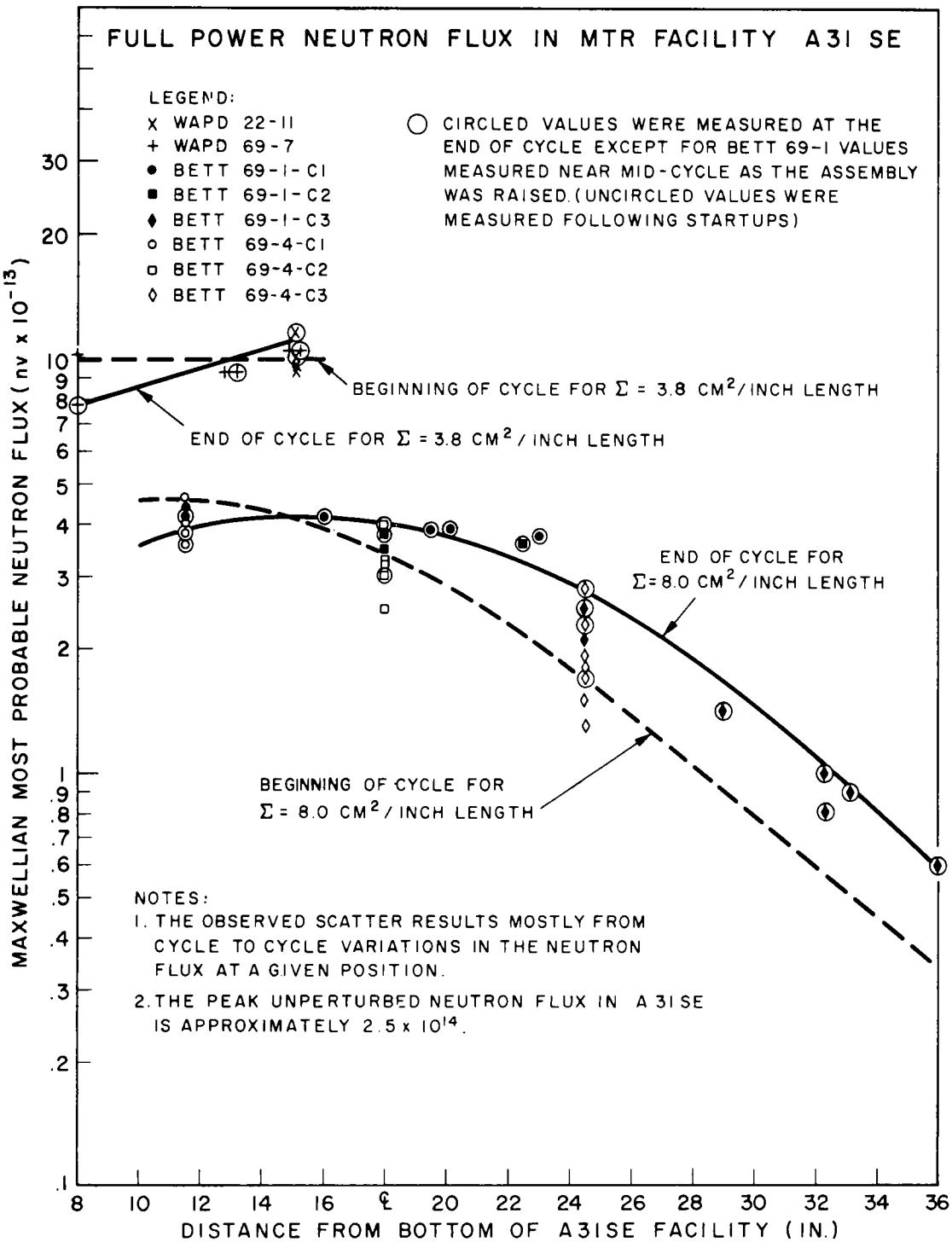

GAMMA HEATING IN A3ISE AS A FUNCTION OF TIME
GAMMA HEATING (WATTS/GM)
TIME AFTER STARTUP (DAYS)
GAMMA HEATING 3 IN. BELOW REACTOR CENTERLINE AS GIVEN BY WAPD 69-7.
GAMMA HEATING 2 TO 4 IN. ABOVE REACTOR CENTERLINE AS GIVEN BY WAPD 69-7 (VALUES FOR 2 IN. ABOVE CENTERLINE ARE BELIEVED TO BE LOWER THAN ACTUAL ; SEE TEXT)
GAMMA HEATING 4 IN. ABOVE REACTOR CENTERLINE AS GIVEN BY WAPD 22-11
LEGEND:
WAPD 22-11 , FOLLOWING 1st STARTUP, 7 IN. SPACER
WAPD 22-11 , FOLLOWING 2nd STARTUP, 7 IN. SPACER
WAPD 22-11 , FOLLOWING 3rd & 4th STARTUP, 7IN. SPACER
WAPD 22-11 , FOLLOWING 5th & 6th STARTUP, 7 IN. SPACER
WAPD 22-11 , FOLLOWING 7th STARTUP, 7 IN. SPACER
WAPD 22-11 , FOLLOWING 8th STARTUP, 7 IN. SPACER
WAPD 69-7 , FOLLOWING 2nd STARTUP, 5 IN. SPACER
WAPD 69-7 , FOLLOWING 3rd STARTUP, NO SPACER
WAPD 69-7 , FOLLOWING 4th STARTUP, 7 IN. SPACER
WAPD 69-7 , FOLLOWING 5th STARTUP, 7 IN. SPACER

Figure C-5


FULL POWER NEUTRON FLUX IN MTR FACILITY A31 SE
LEGEND:
X WAPD 22-11
+ WAPD 69-7
● BETT 69-1-C1
■ BETT 69-1-C2
♦ BETT 69-1-C3
○ BETT 69-4-C1
□ BETT 69-4-C2
◊ BETT 69-4-C3
○ CIRCLED VALUES WERE MEASURED AT THE END OF CYCLE EXCEPT FOR BETT 69-1 VALUES MEASURED NEAR MID-CYCLE AS THE ASSEMBLY WAS RAISED. (UNCIRCLED VALUES WERE MEASURED FOLLOWING STARTUPS)
BEGINNING OF CYCLE FOR Σ = 3.8 CM²/INCH LENGTH
END OF CYCLE FOR Σ = 3.8 CM²/INCH LENGTH
END OF CYCLE FOR Σ = 8.0 CM²/INCH LENGTH
BEGINNING OF CYCLE FOR Σ = 8.0 CM²/INCH LENGTH
NOTES:
1. THE OBSERVED SCATTER RESULTS MOSTLY FROM CYCLE TO CYCLE VARIATIONS IN THE NEUTRON FLUX AT A GIVEN POSITION.
2. THE PEAK UNPERTURBED NEUTRON FLUX IN A31SE IS APPROXIMATELY 2.5 x 10¹⁴.
MAXWELLIAN MOST PROBABLE NEUTRON FLUX (nv x 10⁻¹³)
30
20
10
9
8
7
6
5
4
3
2
1
.9
.8
.7
.6
.5
.4
.3
.2
.1
8
10
12
14
16
℄
20
22
24
26
28
30
32
34
36
DISTANCE FROM BOTTOM OF A31SE FACILITY (IN.)

The Settled Bed Reactor In-Pile Capsule

G. P. Pancer
Brookhaven National Laboratory
Upton, New York

The Settled Bed Reactor is a reactor concept in which the fuel is a rigid fuel system during power operation and a non-rigid fluidized fuel system during fuel handling periods, thus utilizing the advantages of both the solid fuel and circulating fuel reactor concepts. The reference design is a directly cooled with sodium, fast breeder reactor using uranium-plutonium carbide in spherical form of approximately 1/8 inch diameter as the fuel. The core configuration is a bed of spherical fuel particles through which the sodium coolant flows downward through the bed during power operation cycles. During the fuel handling periods, when the reactor is shutdown the fuel can be reshuffled, removed and fresh fuel added by fluidizing the bed. This is done by reverse flow of the sodium, thus eliminating the need for any complex and costly fuel handling system.

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*This work was performed under the auspices of the U.S. Atomic Energy Commission.

A fuel study program is now underway at BNL to evaluate the use of UC-PuC and other fuels applicable to a reactor concept of this type. The initial irradiation phase of this study will be performed at the BGRR to obtain preliminary data on the irradiation behavior of the applicable fuels. Future irradiation studies on high burnup will be performed at the ETR and when available the HFBR. The capsule design discussed below is being used in the BGRR irradiations.

During operation of the reactor, the forces acting on the fuel particles due to the total pressure drop across the fuel bed and the weight of the bed itself is approximately equal to 1 lb. of force on a fuel particle. To duplicate these conditions within the limitations of the in-pile capsule, a column of spheres loaded with a 1 lb. dead weight was used. The dead weight eliminated the need for any complex spring loading system to maintain a constant force. The column of approximately 1/8 diam. fuel particles are held in a perforated length of tubing of 0.18 inch I.D.

The capsule unit is made up of two independent concentric tubes approximately 25 inches long. Both tubes

or containers are fabricated of stainless steel. The primary or inner capsule contains the fuel and liquid metal while the secondary or outer capsule is a finned unit for heat removal. The annulus between the two units is filled with He.

The primary container is partially filled with NaK such that when the fuel tube and weight are located coaxially within the container the NaK completely surrounds the fuel. A single conductor coiled heater wire and two thermocouples (all stainless sheathed) are located inside the primary capsule submerged in the NaK. These are used for auxiliary heat supply and temperature indication and control. The penetrations for these leads through the primary container are sealed by a high temperature furnace braze. The primary and secondary capsule are shown in Fig. 1 and a radiograph of the fuelled section of the primary capsule indicating the NaK, fuel, heater and thermocouple wires is shown in Fig. 2.

The resistance wire leads are terminated just outside the primary capsule. Stainless sheathed Ni conductor wire is used from this point to a location outside the reactor. The connector between the resistance

and conductor wire is indicated in Fig. 3.

The He heat transfer medium between the two capsules maintains the temperature of the NaK in the primary capsule at the required level. In addition the He is monitored for high activity to indicate any failure of the primary capsule. Forced air flowing past the secondary container is used to remove the heat. The air is also monitored to indicate a possible failure of both capsules.

The design required a verticle facility in the reactor. Therefore a hole approximately 4 inch in diameter and 8'-4" long was drilled in the Brookhaven Graphite Reactor since none of the existing reactor facilities were adequate. A thimble (Fig. 4) providing an inlet and outlet air cooling system was placed in the hole and the capsule (Fig. 5) is inserted vertically into the thimble to a depth of approximately 8 feet into the graphite. The reactor cooling system or an external blower can be used to supply air. A pictorial view of the complete in-pile assembly is shown in Fig. 6.

The removal procedure allows for removal of only the capsule during a normal removal or if necessary the entire capsule and thimble as a single unit in

the event of an emergency.

The capsule design and operating parameters are given in the following table:

Fuel	1/8 inch diameter UC spheres
Enrichment	20% U-235
Thermal Flux (unperturbed)	5.2×10^{12} neuts./cm^2-sec
Fuelled Length	4 inches
Fission Heat	1300 BTU/hr
Auxiliary heat (electric)	1200 BTU/hr
Maximum auxiliary heat	2400 BTU/hr
NaK operation temp. (fuel surface temp.)	760°C (1400°F)
Design Capsule Pressure	100 psia. at 819°C (1500°F)

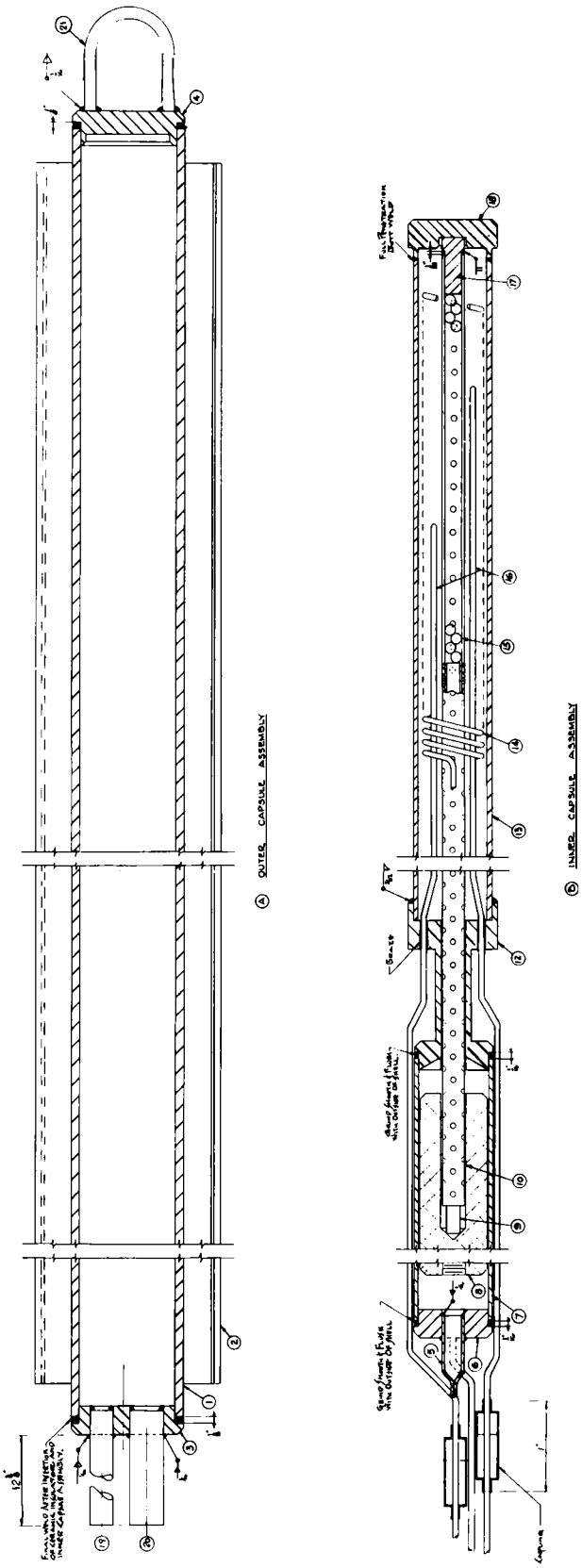

Figure 1

Primary and Secondary Capsule Design

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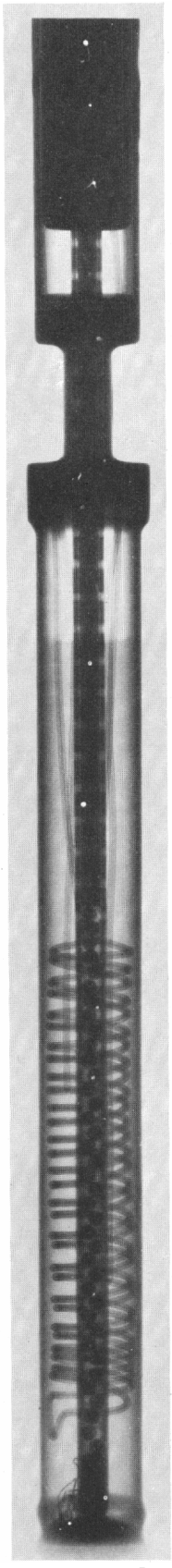

Figure 2

Radiograph of Fuel Primary Capsule

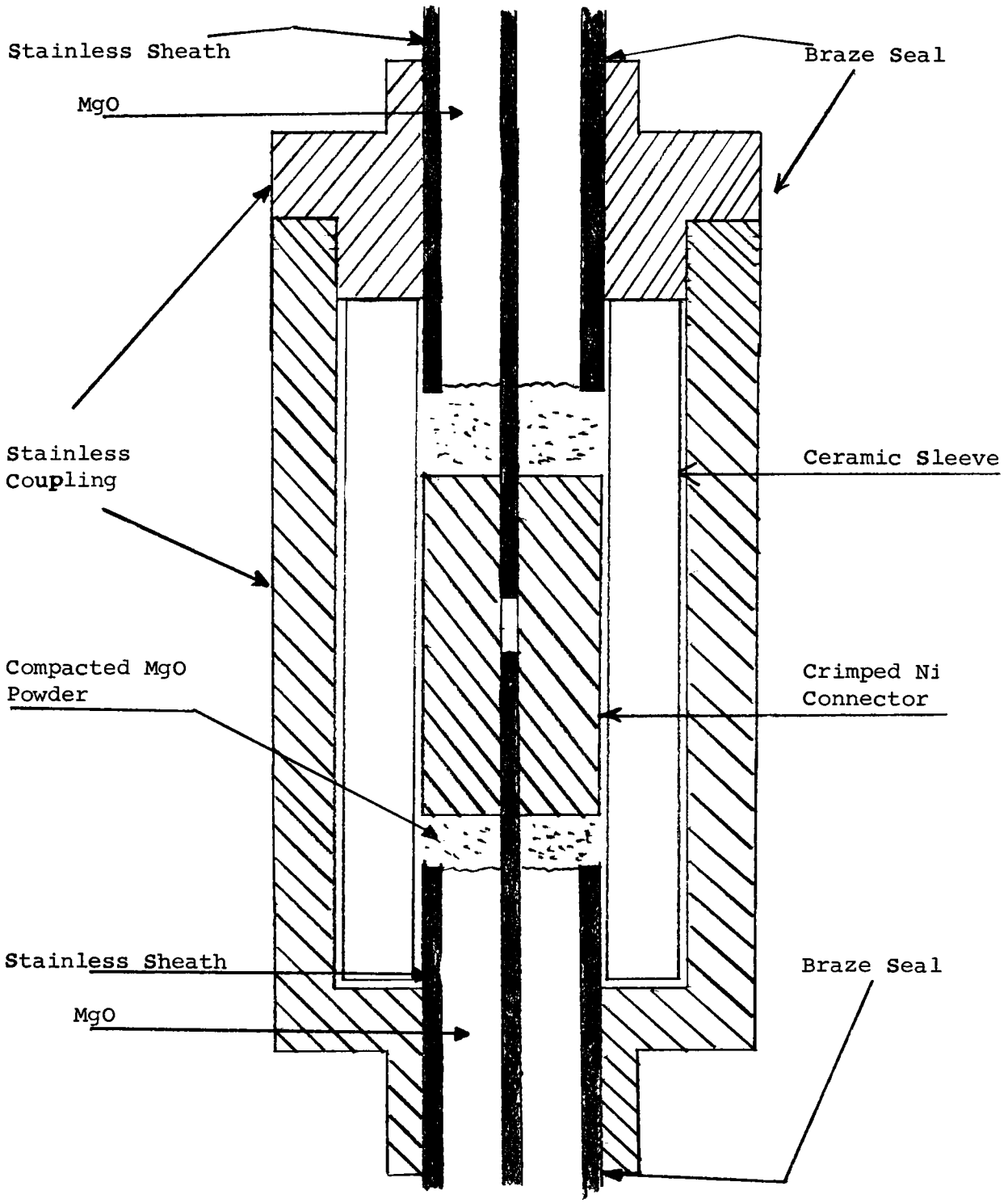

Figure 3 - Connector

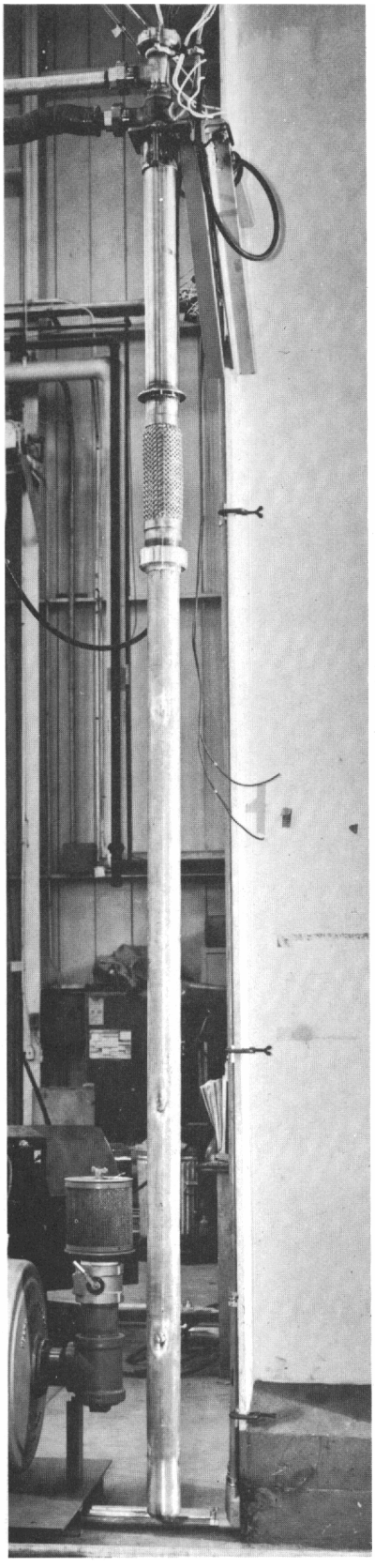

Figure 4 - In-Pile Thimble

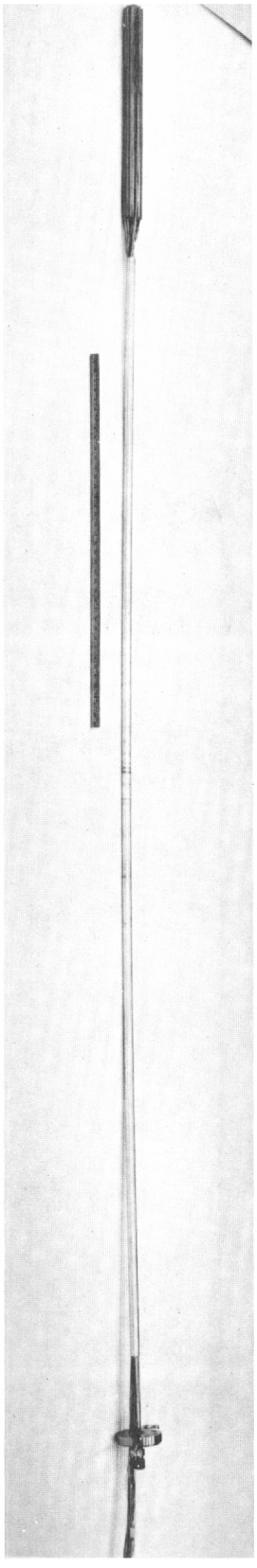

Figure 5 - In-Pile Capsule Assembly

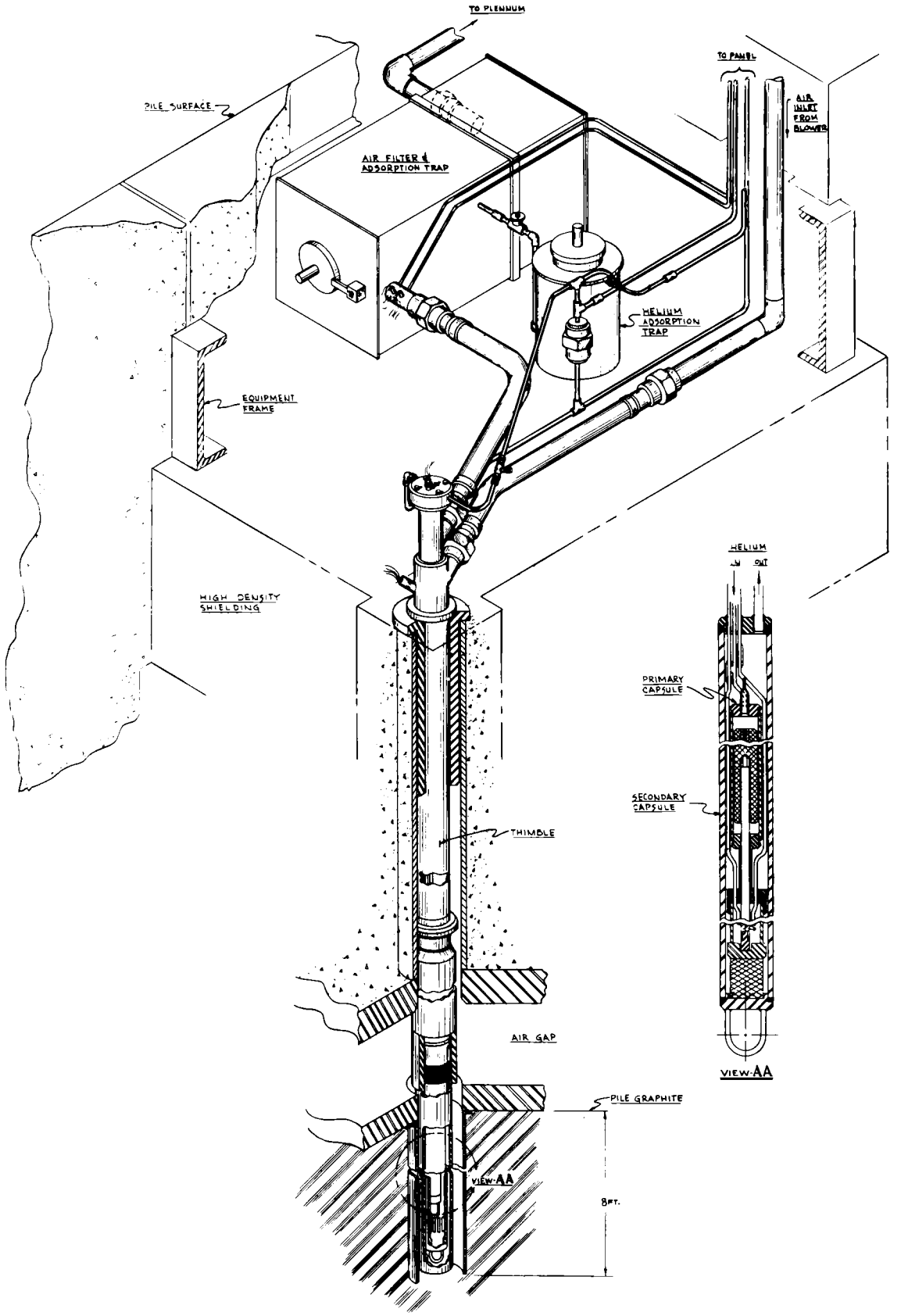

Figure 6

Pictorial Arrangement of In-Pile Facility

CAPSULE FOR IRRADIATION OF Pu-1 w/o Al ALLOY *

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To provide a source of short-cooled neutron irradiated plutonium for fuel processing experiments, a capsule was designed to contain an 80 *g.* foil of the 1 w/o Al alloy for irradiation in the OWR at Los Alamos and is shown in Fig. 1. The irradiation position used was a dummy fuel rod in the core with a 2 in. diameter hole and in an unperturbed thermal flux of $\sim 10^{13}$ neutrons/cm^2-sec.

The plutonium foil was 1-1/4 x 14 x 0.020 in. and was placed in a tantalum sheath open at the top to contain the plutonium in event of melt-down. This in turn was loaded in a 20 in. long stainless steel container of 1.75 x 0.215 in. cross-section having a 0.035 in. wall. This container was fabricated from seamless tubing sized to the desired dimensions by a turks head roll. End caps were welded in place and the entire assembly filled with Na to 1 in. above the foil. A helium atmosphere was established above the sodium at an absolute pressure of ~ 3 in. of mercury. The stainless steel fill line was then pinched shut with a hydraulic ram and cut with a helium shielded arc welder. A single chromel-alumel thermocouple sheathed in stainless steel and brazed to a tube on the end cap monitored the sodium temperature at the center of the capsule during irradiation.

Capsule components were checked before assembly by dye penetrant and ultrasonic testing. Welds were examined radiographically and tested with a mass spectrometer helium leak detector. After complete

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* Work done under the auspices of the U.S. Atomic Energy Commission

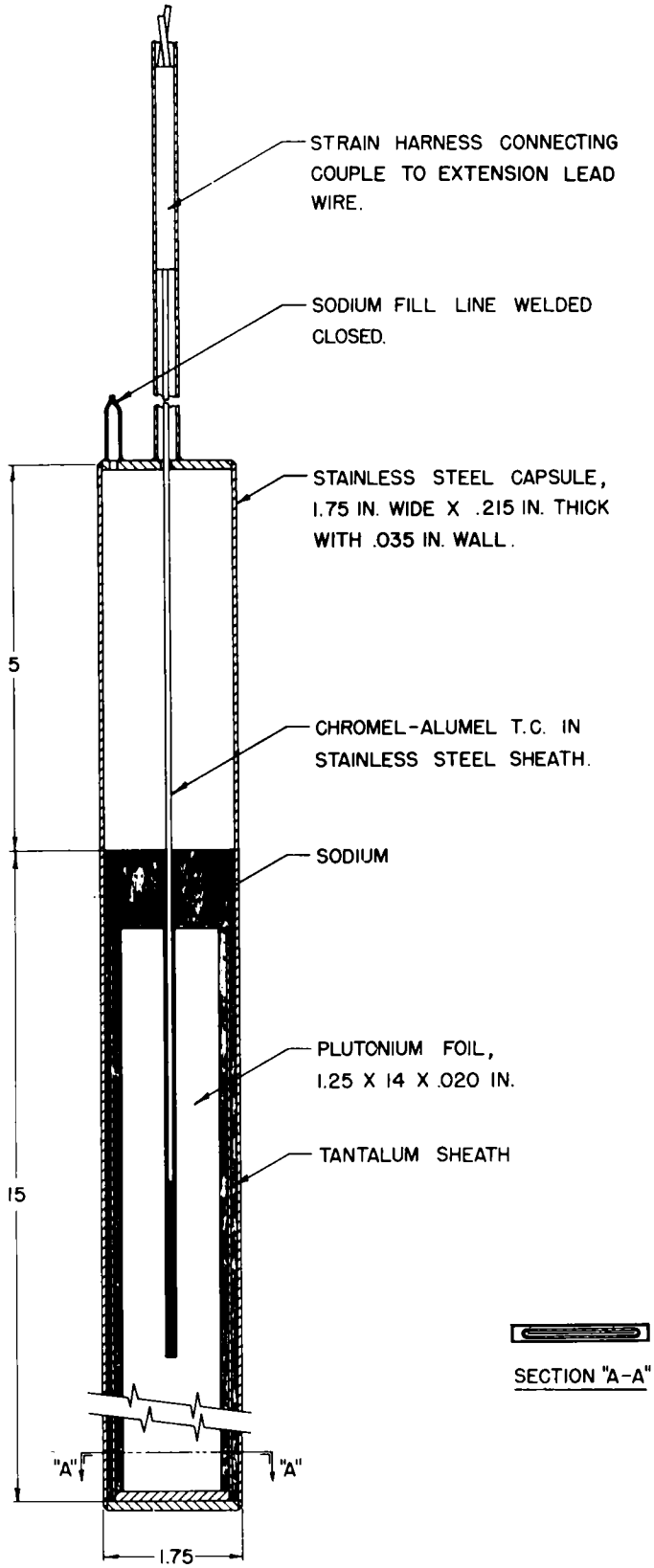

Fig. 1. Irradiation Capsule for Plutonium Foil

assembly the capsules were placed in a vacuum chamber evacuated by the leak detector and any faults in the final welds were revealed by helium leaking from the capsule. The assembled capsule was subjected to an ultrasonic test to determine if any voids were present in the sodium thermal bond. The results, however, were inconclusive because of the many interfaces through the capsule.

To attain a coolant velocity of ~30 ft/sec. an auxiliary pump was required and the flow area of the 2 in. port was restricted. The coolant water used was obtained from that supplied to cool the reactor and after passing through the irradiation port was discharged back into the reactor coolant stream. A separate coolant loop was desired but considered impractical at that time because of the reactor configuration.

Instrumentation, in addition to the Cr-Al thermocouple, consisted of a flow meter on the pump delivery and a radiation counter looking at an ion exchange column through which a portion of the coolant water passed. The flow meter could activate a reactor scram circuit while the radiation counter and the thermocouple were tied to alarms.

In the five capsules irradiated the average power generated was 18.5 KW with a heat flux at the foil surface of 258,000 Btu/hr-ft^2. The effective neutron flux was 4.3×10^{12} and the plutonium fission burn-up rate was 0.12% per week (for a nominal 100 hr. week). The maximum sodium temperature recorded was 110° C. The longest irradiation reached a burn-up of ~0.30% with no evidence that any limit was remotely approached.

CAPSULE DESIGNS FOR REFRACTORY METAL FUEL AND CERAMIC FUEL TESTING

BY

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At NMPO, the irradiation effort encompasses a wide range of materials, both fueled and unfueled, operating at temperatures from ambient to 2200°C. Unfueled materials include beryllium oxide, hydrides of titanium, zirconium, and yttrium, and some of the refractory metals including tungsten, tungsten-25% rhenium, and molybdenum. Fueled systems include the BeO base fuel elements, the refractory metal fuel system, and the nichrome clad fuel element. The irradiations are being done at the ETR, ORR and LITR, and at a given point in time, there are from ten to twelve irradiations in progress in these facilities.

Although many of the capsule designs for the irradiation of unfueled materials are interesting, this presentation will be primarily devoted to the fueled systems; specifically, the BeO base fuel elements and the refractory metal fuel system.

I BeO Base Fuel Elements

A. Objectives - To evaluate the capability of various coatings for inhibiting fission product release and to determine the effect of high fuel burnup on stabilized UO_2 fuel systems in BeO base fuel elements.

B. Scope - An extensive evaluation program is in progress in both the LITR and ETR, the former being utilized for fission product release evaluations, and the latter for burnup. A total of sixteen test assemblies are involved in this program, six in the LITR and ten in the ETR. The LITR tests are for three weeks duration while the ETR tests will require from five to ten months each to get the desired burnup.

C. Design -

1) LITR - The specimens are conventional hexagonal fueled BeO tubes measuring approximately 0.3 inches across flats by 4.3 inches long. Six tubes are assembled in a hexagonal array with a slotted unfueled rod at the center for thermocouple placement. The specimens are supported at each end by BeO plugs contained within three concentric Al_2O_3 tubular heat shields. Immediately surrounding the outer Al_2O_3 tube is a layer of thermoflex insulation held in place by a thin stainless steel shell.

The annulus between this shell and the outer stainless steel shell forms the inlet air passage to the specimens. Assembly instrumentation consists of six Pt-Pt 10% Rh thermocouples for measuring specimen temperature, two chromel-alumel thermocouples for measuring inlet and outlet air temperatures, and four chromel-alumel reactor safety thermocouples for monitoring the outer shell surface temperature.

2) ETR - In the ETR, fission product containment requirements dictated a completed encapsulated fuel design.

Since the objective of these irradiations is to attain high burnup at a given fuel element temperature in the shortest time possible, the maximum obtainable heat generation commensurate with fuel element heat removal capability is desired. In addition, a method of fuel element temperature control is necessary because of changing flux patterns in the reactor. These requirements led to the design depicted in Figure 1. The capsule consists of an inner sealed chamber which contains the test samples in an atmosphere of 95 volume percent helium and 5 volume percent oxygen; a second region surrounding the inner chamber which provides cooling air flow and containment; a third region which provides cooling water flow, and an outer passage for supplying cooling air. The test samples are mounted and centered on slotted beryllium oxide rods and the ends of each rod are in turn mounted on short spacers which maintain the rods and samples concentric within a stainless steel tube. This arrangement provides a 0.035 inch insulating gas gap surrounding each test specimen. Below this stainless steel tube and integral with it, is a larger diameter tube the purpose of which is to provide a larger common volume for alleviating the effects of high temperature operation within the sealed chamber. The inner sealed chamber is completely separated from the reactor water by the air passage to insure that any leaks that might develop can be detected without serious hazard.

There are four Pt-Pt 10% Rh thermocouples located in the slotted BeO support rod on the ID of the specimen. The thermocouples were led out of the sealed chamber through double metal to ceramic seals on the first capsule but because of the difficulty of obtaining these seals, the seals were fabricated by welding metallic sheathed thermocouples to the end closure on later capsules.

Heat generated within the fuel elements is radiated and conducted across the small gas gap and the total heat generated in the sealed capsule is removed by forced convection to the air stream. Part of this heat appears as sensible heat in the air stream and the balance transferred by convection to the outer surface of the cooling annulus and to the reactor water.

The capsule design is such that when the specimens are at temperature 40 SCFM of air is required to maintain the container surface temperature at about 500°C. Maximum allowable surface temperature of the container is 760°C and the maximum obtainable air flow is about 60 SCFM. Within the limits of the design, specimen temperature can be varied within a range of about 125°C above and below the design point, a 35°C change on the container being worth about 10°C on the fuel elements.

There are five different fuel loadings involved in this series of tests, and using this design concept, it is only necessary to change the insulating gas gap to match the heat generation or change positions in the reactor to a higher or lower flux to maintain the fueled specimens at the desired temperature.

II Refractory Metal Fuel System

A. Objective - To define the time-temperature, fission product retention, and nuclear burnup capabilities for high-temperature reactor applications.

B. Scope - Irradiations are being done in both the LITR and the ETR. The LITR is being used for short time (one to three weeks) irradiations while the ETR is being used to determine burnup capabilities of the refractory metal fuel system.

C. Design -

1) LITR - The test fuel element is a thick-walled cylinder measuring 0.5 inches OD, 0.116 inches ID and 1.5 inches long. The fuel element is surrounded by five concentric thin-walled tantalum cylinders for heat radiation shields (Figure 2). Each shield maintains about a 0.1 inch gas gap between it and the next gap, the gaps providing resistance to heat transfer by conduction.

Tantalum partial shields in the form of discs are mounted above and below the specimen for additional insulation. This structure is enclosed within an Inconel can which seals the fuel element and structure from the cooling air flow.

An inert gas flow is introduced near the top of the fuel element, flows over the element and through the three innermost gas gaps to the bottom of the capsule, and returns through the three outermost gas gaps to the exit gas line, the exit line being concentric with and surrounding the inlet gas line.

Surrounding the Inconel shell is a stainless steel shell which directs the cooling air flow over the surface of the Inconel shell. Heat generated by the fuel element and structure is removed by forced convection from this surface. Cooling air is taken from the pressurized facility tube and exhausts through a flexible line which is concentric with and surrounds the inert gas supply lines.

The assembly is instrumented with 23 thermocouples, nine tungsten-tungsten 26% rhenium, four platinum-platinum 10% rhodium and ten chromel alumel.

The sample heat generation rate was uncertain within rather wide limits because of indeterminate self-absorption and local flux depression effects. For this reason several means of controlling the heat generation and/or sample temperature were made available. For high sample temperature, or heat generation, the sample may be withdrawn to a position of lower heat generation. For low sample temperature, or heat generation, the resistance to heat flow from the sample may be increased by the dilution of the helium gas flow with argon, the latter having a much lower conductivity. A secondary temperature control is available by varying the cooling air flow rate.

This assembly was successfully operated in March, 1963. It ran for 178 hours at 1930°C (3500°F) and the last two hours of the LITR cycle at 2200°C (4000°F). Post-test visual inspection showed no evidence of cladding oxidation.

2) ETR -- One of the problems in designing an acceptable assembly for determining the effects of burnup on the

refractory metal fuel system is the attainment of uniform burnup of the fuel. Since this fuel system is intended for use in fast spectrum reactors, the operation of this system in a thermal reactor will produce a burnup gradient through the specimen because of the absorption of low energy neutrons near the outer surface. The magnitude of this gradient is dependent on specimen fuel thickness. In this design, a fueled thickness of 0.026 inches was selected primarily from fabricability restrictions since the refractory material is brittle and is prone to crack in grinding thinner sections.

The fuel element design for this irradiation is shown in Figure 3. The inner and outer tubes are close fitting refractory metal tubes. The fueled core is a cylinder 0.710 inches long, 0.210 inches OD, and 0.156 inches ID. The core is assembled within the tubes, and end plugs are inserted to fill the gaps left by the shorter core. The assembly is then electron beam welded at each end and autoclaved to form a metallurgical bond between all components. The assembly is machined to the final dimensions as shown on the figure.

The irradiation assembly is shown in Figure 4. There are four fueled specimens in the assembly. Each fueled specimen is supported at each end by tantalum end spacers that maintain the specimen concentric within the 0.390 inch OD, 0.330 inch ID container tube and provide a 0.045 inch insulating gas annulus around the fueled portion of the specimen assembly. Heat generated in the fuel is radiated and conducted across this gap to the container tube where it is removed by convection to the reactor water. The specimen container tubes are mounted concentrically within 0.625 inch OD, 0.54 inch ID stainless steel tubes which provide an 0.075 inch annulus for cooling water flow. The four specimens are mounted in parallel within a heavy wall protection tube. The inert gas flows down through the inlet gas line to a distributor at the bottom of the capsule and up over the four specimens to the exit gas line.

Thermocouples - There are two 0.010 inch conductor W-W 26% Rh thermocouples located on the ID of each fueled specimen. Each conductor is made up of ten strands of 0.003 inch wire which greatly increase its flexibility. The hot junction is made by joining the conductors with a square knot and welding the loose ends. The thermocouples are located in the ID of the specimen and held

in place by the thermocouple anchor assembly located above the specimen in the container tube. This assembly is made of two ceramic discs, drilled for the conductors and counter-bored at the mating surface. A knot is tied in the conductor and this knot is captured in the counter-bore between the two ceramic discs to maintain the thermocouple junction at the desired location.

In addition to the W-W 26% Re thermocouples, there are two Pt-Pt 10% Rh and two chromel-alumel thermocouples located in the fuel element extensions.

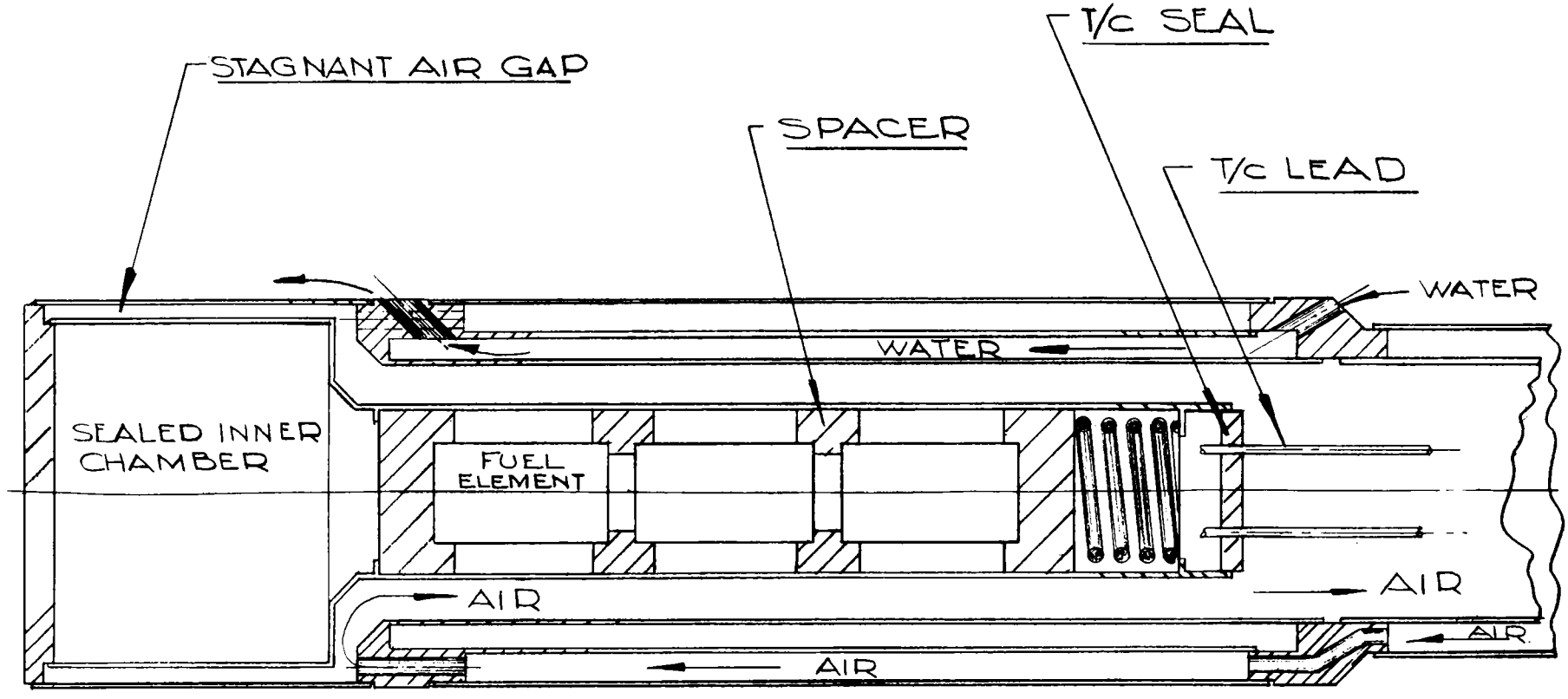
Thermocouple insulation in the hot zone is two-hole beryllium oxide, and from the anchor assembly to the junction box, it is single hole alumina insulators.

Facility - A schematic of the facility for irradiating the refractory metal clad fuel system is shown in Figure 5. Ultra-cleanliness is of the utmost importance in the successful operation of the refractory metal system. Before installation in the facility, the lead tube assembly was evacuated and purged with dried and gettered helium five times while maintaining a temperature of 315° on the specimens. Each stainless steel component of the facility was washed with a detergent, flushed with hot water then with acetone and allowed to air dry. Following this the components were torch heated and purged with dry helium, then temporarily sealed off under positive helium pressure. After the facility was assembled and the test assembly installed, the complete assembly was subjected to six helium purge and evacuation cycles (10 micron or less) using dried and gettered helium, the titanium getter operating at a peak temperature of 870°C.

Operation - The irradiation assembly was installed in the ETR F-3 position during the shutdown for cycle 57 and reached a maximum temperature of 1925°C on one specimen on 18 August 1963. The other three specimens reached temperatures of 1710°C, 1610°C, and 1440°C. The capsule design was based on a thermal flux obtained at the center of the F-3 position and the variation in temperature is assumed to be caused by the flux gradient existing in this position. The capsule operated normally at these temperatures for approximately fifty hours when, after a reactor scram and subsequent startup, activity was detected at the auxiliary console on the main floor

and the reactor was scrammed manually. The sample was removed from the reactor and shipped back to NMPO for disassembly. The appearance of the capsule and specimens upon disassembly was excellent and there was no evidence that the capsule or facility design was the cause of failure.

Another capsule of the same design is being fabricated and is scheduled to be operated in December 1963.



2.16.9

SCHEMATIC

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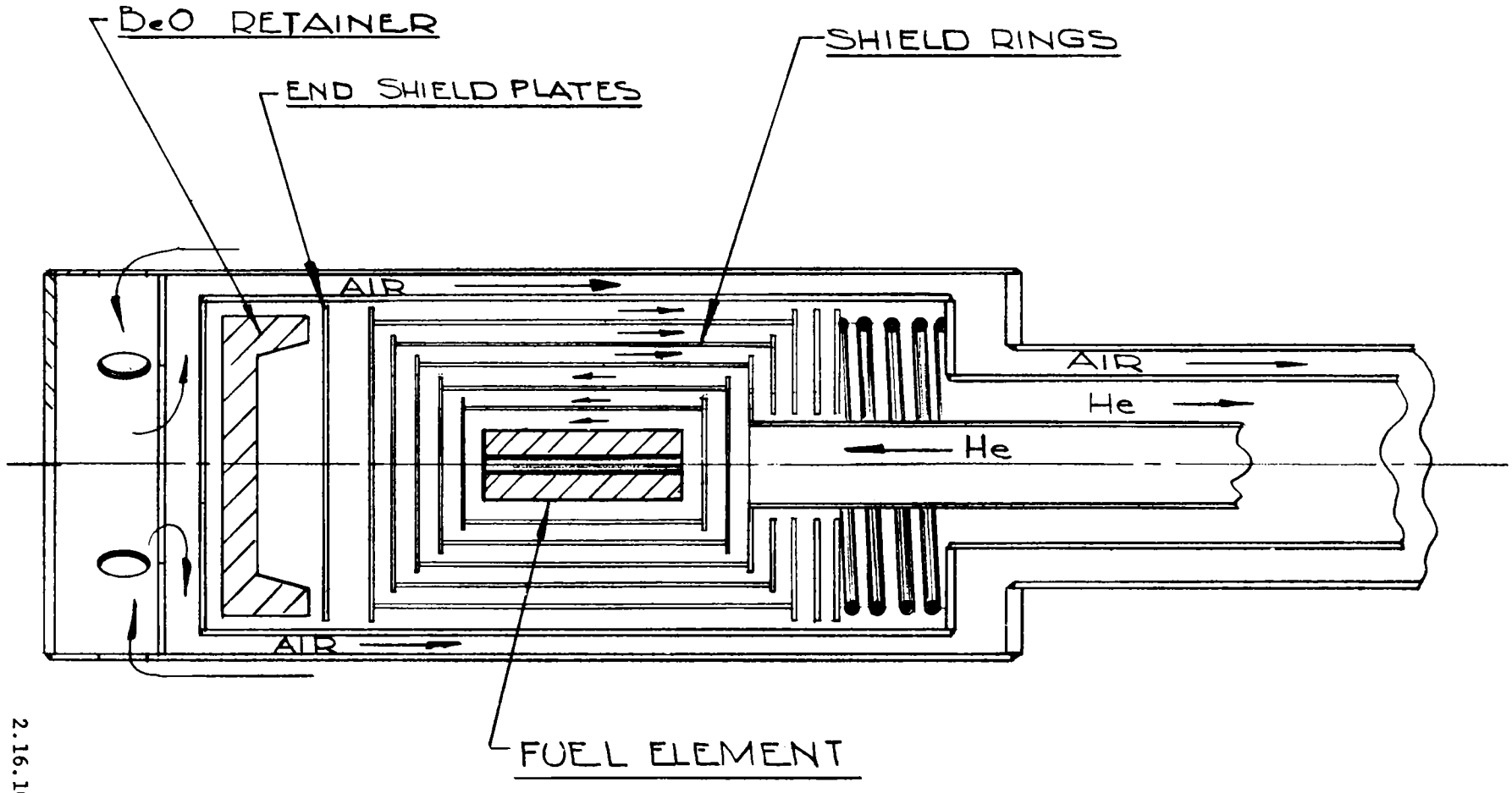
FIG 1



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SCHEMATIC

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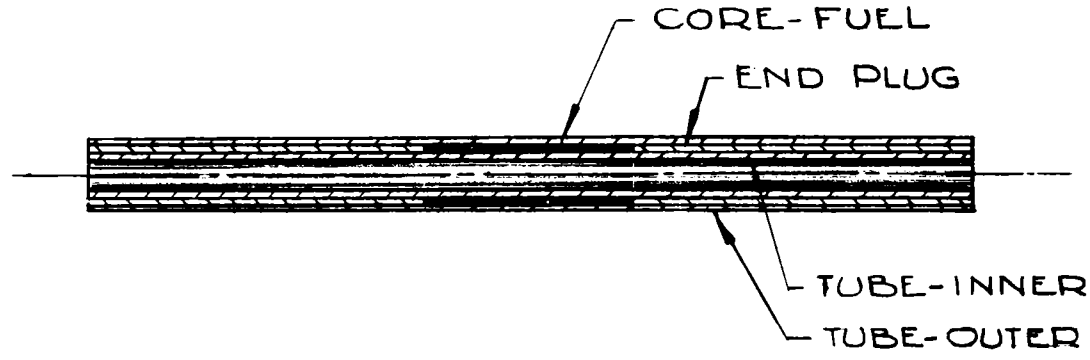
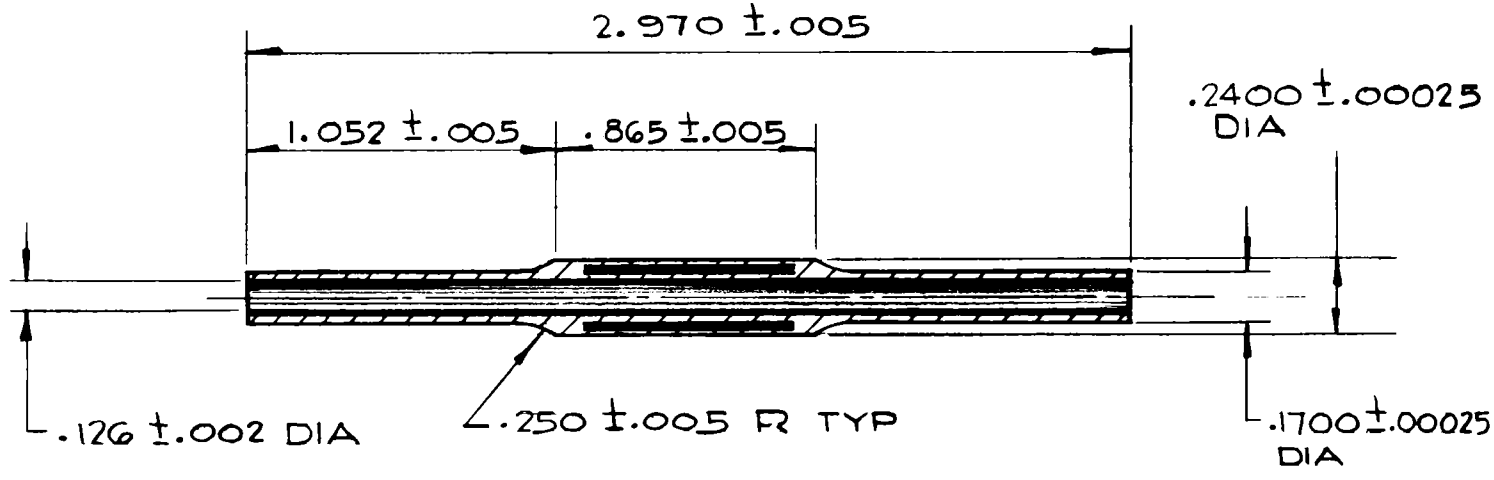
FIG 2



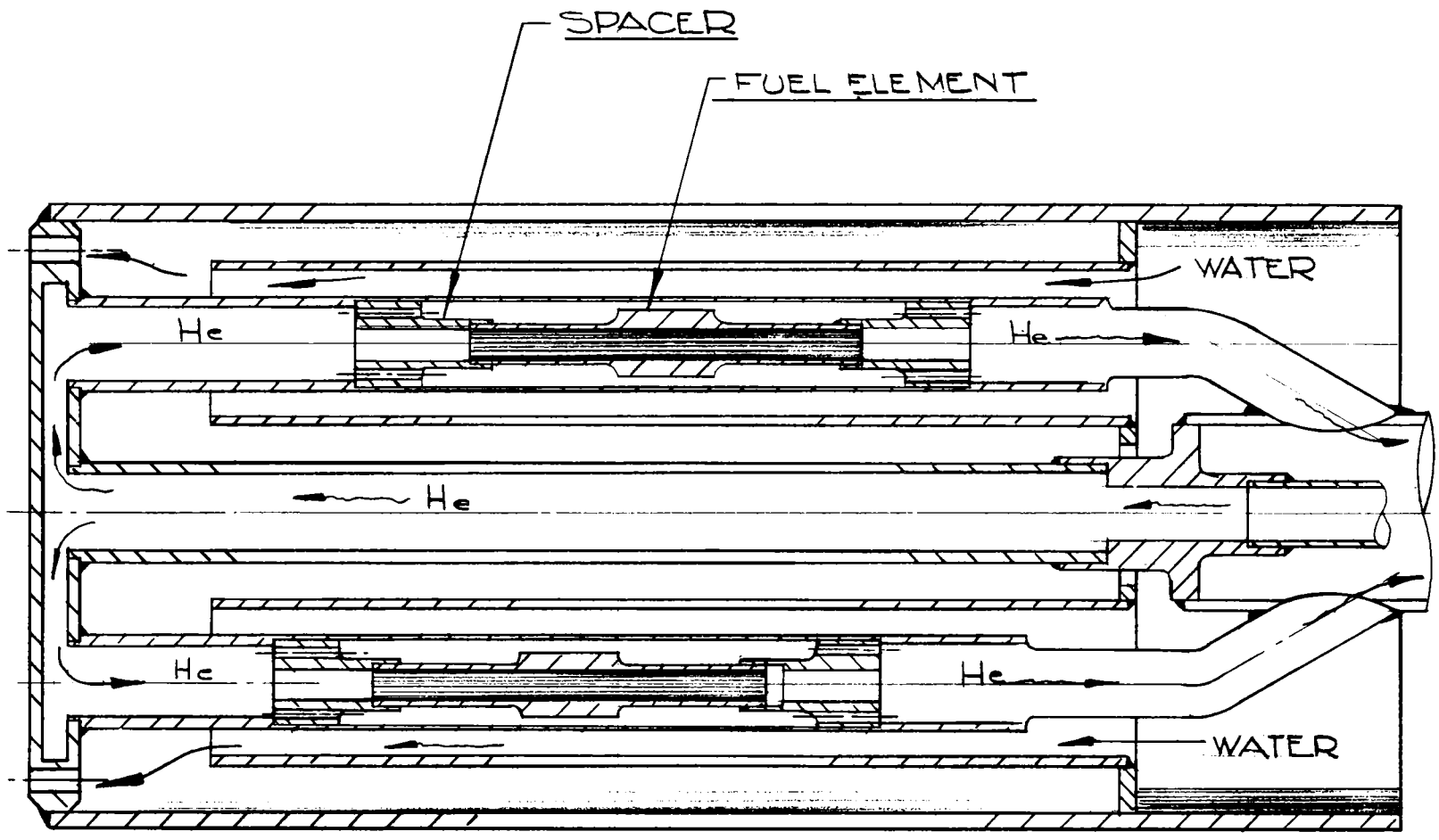
AFTER AUTO CLAVE



FINAL MACHINING

FIG 3
REFRACTORY METAL CLAD
TEST SPECIMEN

2.16.11



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SCHEMATIC

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FIG 4

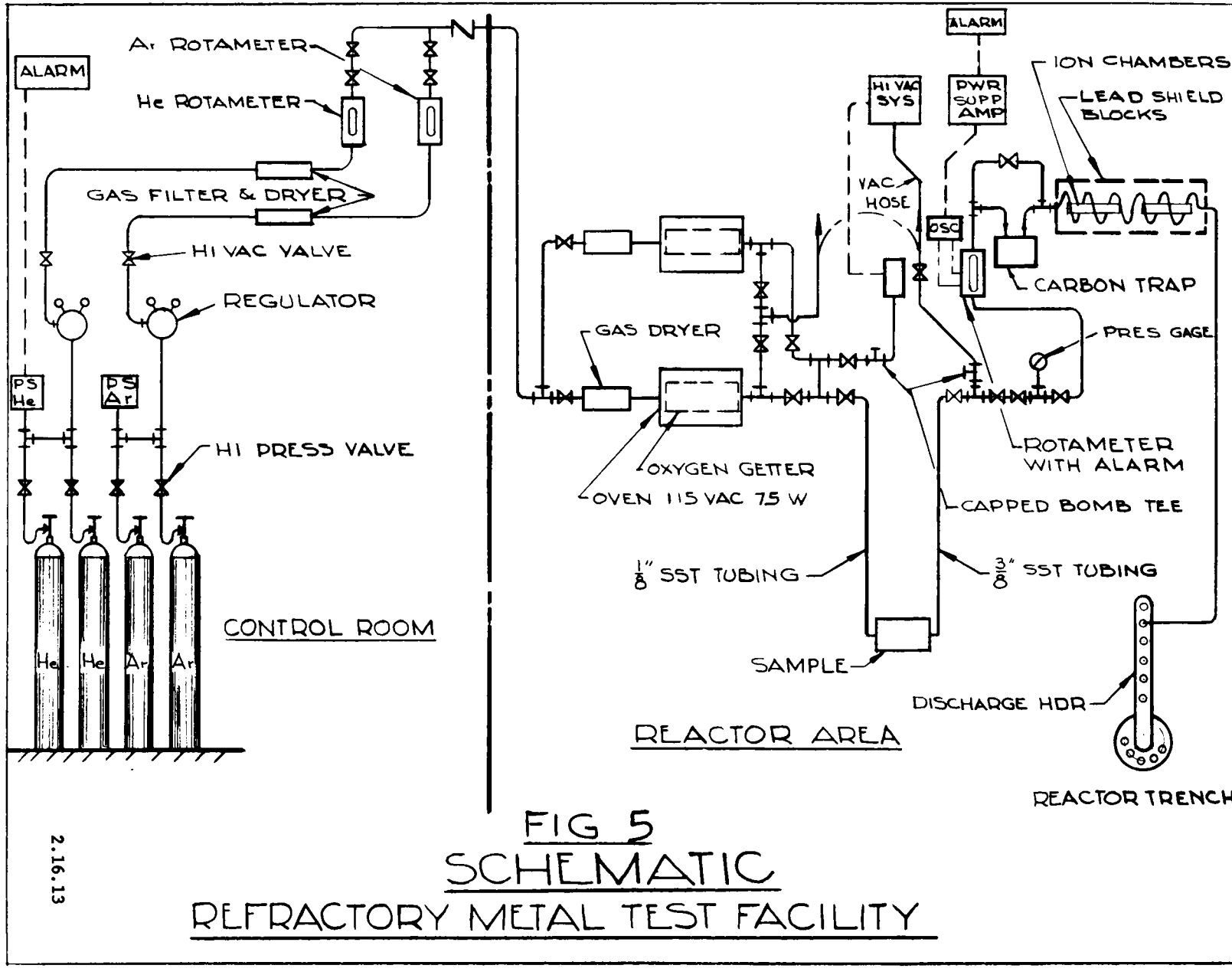

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FIG 5
SCHEMATIC
REFRACTORY METAL TEST FACILITY

AN INSTRUMENTED CAPSULE DESIGN FOR ABOVE-2000 F IRRADIATIONS*

P. B. Shumaker and J. H. Stang
Battelle Memorial Institute

During the past 2 years, Battelle has conducted at WTR and at the MTR/ETR site a series of capsule irradiations in which certain experimental fuels have been subjected to maximum temperatures up to 2600 F. The capsule design employed in the initial irradiations in this series utilized stainless steel-sheathed Chromel-Alumel thermocouples located in radial zones where temperatures from 1500 to 1850 F prevailed. The extrapolation from monitored temperature to fuel temperature was made on the basis of the capsule heat-transfer design. The fairly large uncertainty in specimen temperature resulting from this extrapolation, however, proved to be a greater disadvantage in the interpretation of the fuel performance than was desirable.

A modified capsule employed in recent experiments possesses the following principal design objectives.

(1) The incorporation of tungsten-rhenium thermocouples for the monitoring of temperatures above 2000 F, i.e., the monitoring of temperatures close to the specimen temperatures

(2) The incorporation of Chromel-Alumel thermocouples for secondary temperature monitoring

(3) Provision for some control of specimen temperatures.

Main elements of the modified design are of interest here. As indicated by Figure 1, nine clad specimens immersed in liquid metal are compartmented in groups of three in heavy-wall Nb alloy cans. The three

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* The work reported in this paper was performed by Battelle Memorial Institute, under AEC Contract W-7405-eng-92.

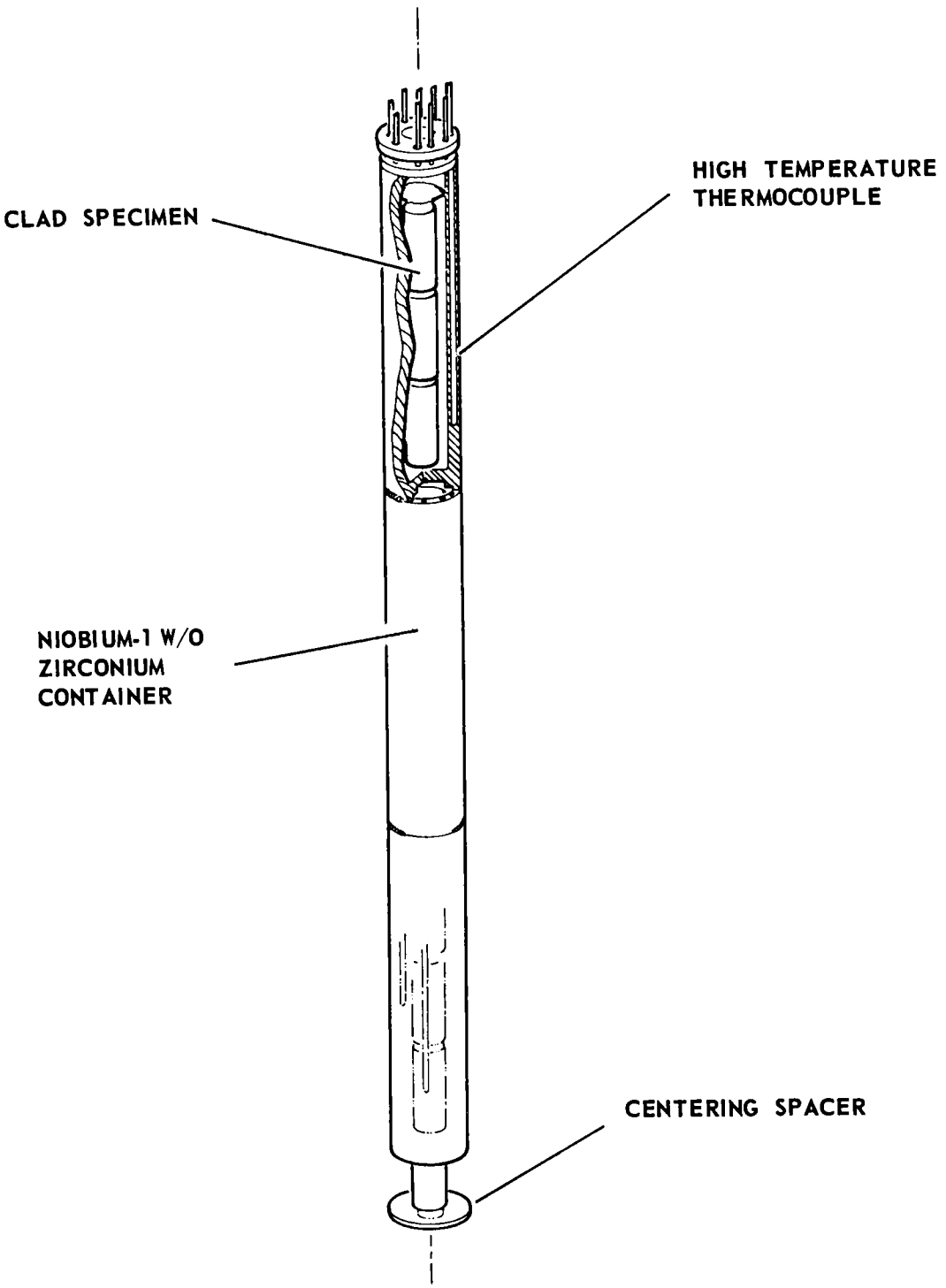


FIGURE 1. PRIMARY CONTAINMENT

cans are laid end-to-end and tack-welded together. This assembly, with spacers attached at each end, is then surface ground to achieve concentricity.

Nb alloy-sheathed tungsten-rhenium (W 25w/o Re - W 5 w/o Re) thermocouples are inserted in longitudinal holes drilled into the heavy-wall specimen containers. A total of nine such thermocouples, the individual junction ends of which are located at the mid-planes of individual specimens, are provided. Note that this thermocouple placement is out of liquid metal, thereby avoiding two difficult situations, viz., (1) the requirement of a thermocouple sheath closure (braze or weld) that must be leak-free with respect to liquid metal and (2) the opportunity for liquid-metal corrosion of the relatively thin refractory-metal thermocouple sheaths.

The design Δ T between a W-Re thermocouple and the specimens, as determined by an iterative relaxation computer program, is $\sim$175 F under the following reference irradiation conditions:

Fuel plus cladding diameter - 0.250 in.

Fission-heat generation rate - $\sim$2 kw/cc.

Figure 2 depicts the compartmented specimen assembly shown in Figure 1 surrounded by a heavy stainless steel containment and separated from this containment by a narrow (typically 15 mils) helium annulus. The outer surface of this intermediate stainless steel vessel is grooved to accommodate nine Inconel-sheathed Chromel-Alumel thermocouples. According to the computer program, these thermocouples sense a temperature of approximately 1500 F in the foregoing reference situation. As indicated before, these thermocouples are present to provide a means for temperature monitoring upon failure, either mechanical or radiation induced, of the W-Re

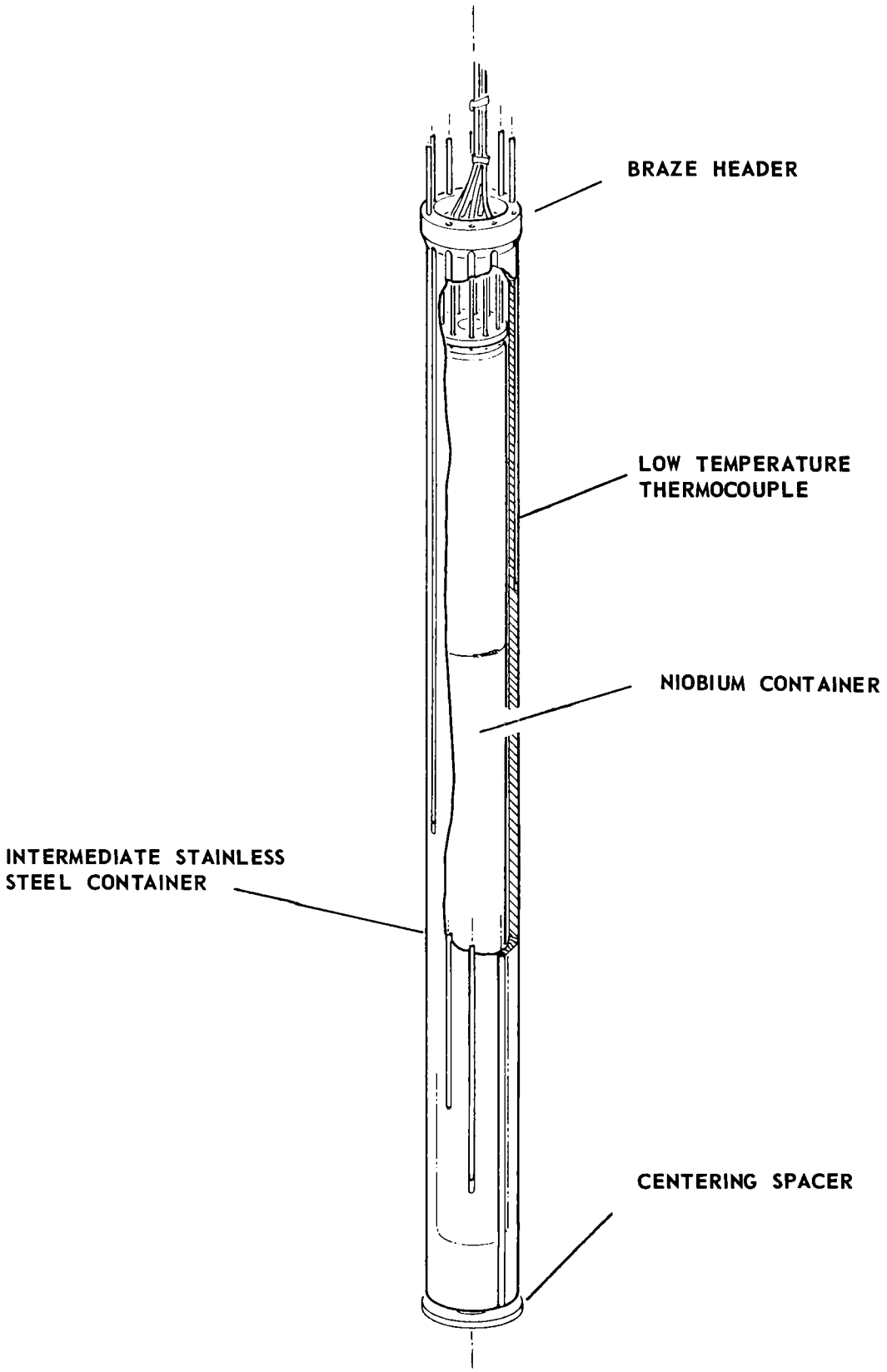


FIGURE 2. INTERMEDIATE CONTAINMENT

units. This scheme reflects the considerable in-pile experience with fair reliability of Chromel-Alumel units at relatively modest temperatures, as opposed to the very limited experience with the W-Re units.

The final capsule assembly shown in Figure 3 combines the elements from Figures 1 and 2 with the water-contacting shell surrounding the other elements of the capsule assembly. The annulus between this containment and the intermediate stainless steel containment is again typically 15 mils. This annulus is connected to an out-of-pile helium-nitrogen flow circuit for capsule temperature control via gas mixtures. The mixture continuously circulated into the capsule is automatically regulated by the signal from one of the Chromel-Alumel thermocouples.

One mechanical problem with this capsule design might be mentioned to round out this brief discussion of its main components. This problem stems from the combination of narrow gas annuli and the fairly long containments required to accommodate the nine axially aligned specimens. To assure that the design gap characteristics are obtained, the internal and external tube dimensions are held within ± 0.2 mil on diameter, roundness, and straightness by a series of grinding and honing operations with intermediate stress-relief treatments. While these operations along with their attendant measurements are fairly time-consuming, it is felt that they are required in an attempt to provide a low-temperature thermocouple - high-temperature thermocouple heat-flow meter situation of satisfactory accuracy.

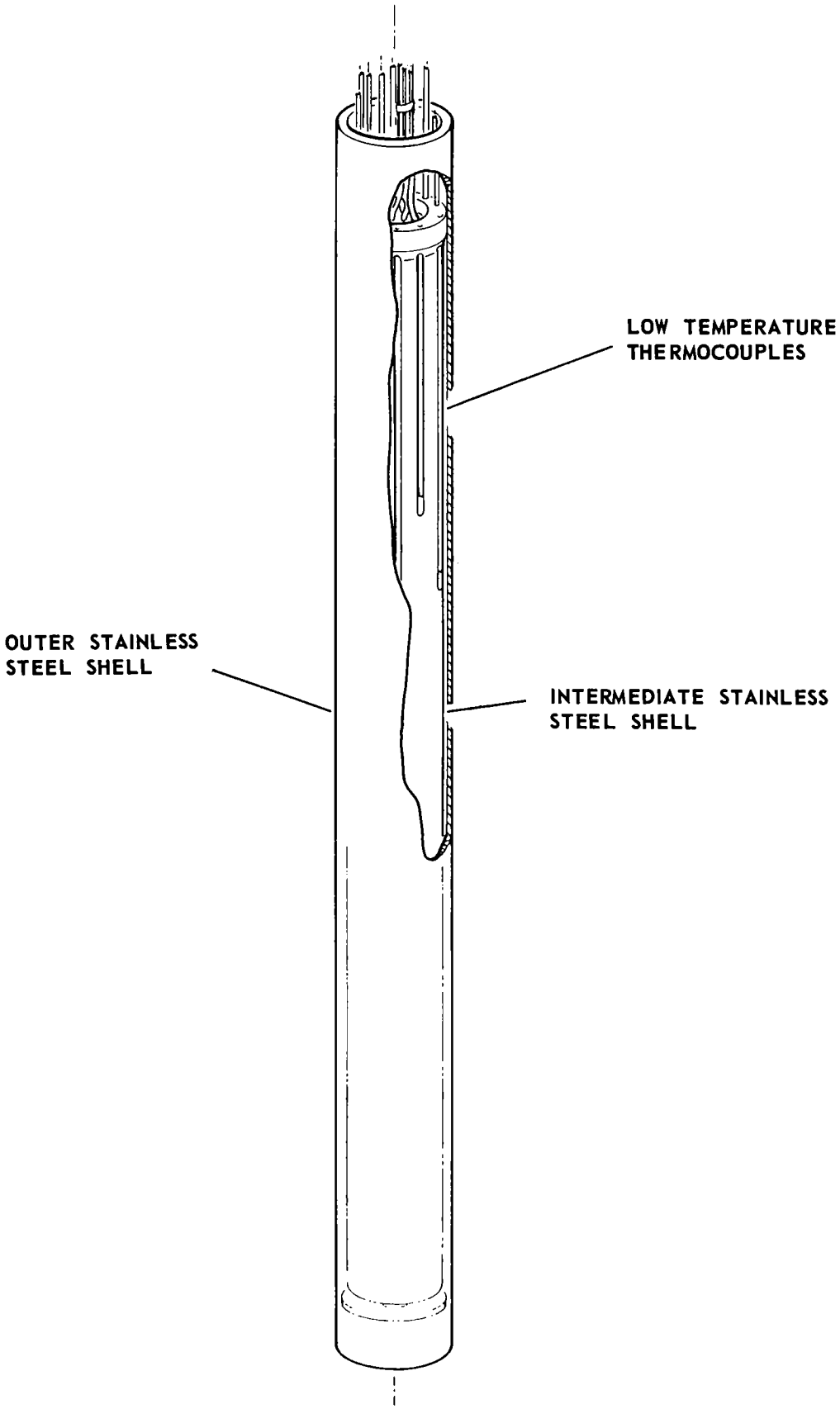


FIGURE 3. FINAL ASSEMBLY

At the time of this writing, three capsules of the modified design have been irradiated at the MTR. A brief resume of their operation follows:

Capsule A: This capsule was irradiated in the MTR for two reactor cycles. Higher-than-desired temperatures were experienced during the high-flux peaks at the beginning of each reactor cycle, with the result that the irradiation was conducted some 9 inches above the flux peak in the facility. The bottom four specimens (the lowermost of which appeared to be located at the peak flux) experienced temperatures in the 2400 to 2600 F range. During the final day of the two-cycle irradiation, the three low-temperature thermocouples at the three bottom specimens became erratic, indicating the possibility of an internal failure. At the time of this writing, hot-cell examination showed a small amount of liquid metal in the intermediate shell, but stereoscopic examination has not shown any ruptures through which it may have leaked.

Capsule B. This capsule operated for three cycles in the MTR; during the last cycle the temperature was automatically controlled by the gas-mixing system. Throughout the irradiation, the four or five peak-flux specimens experienced temperatures between 2400 and 2600 F. Hot-cell examination has revealed no structural changes in the capsule.

Capsule C. In its initial irradiation position, this capsule would have operated at a higher-than-desired temperature, according to temperature data obtained during reactor startup. As a result, the capsule was irradiated above the peak flux in the reactor hole for its first cycle. It was relocated for the second cycle and, at the time of this writing, the peak-flux specimens are experiencing temperatures intended for the experiment.

IN-PILE HYDROGEN PERMEATION EXPERIMENT

T. Golding

Atomics International

An irradiation experiment which allows measurement in-pile of hydrogen permeation through a cladding material was designed and built at Atomics International and is now being irradiated in the ETR.

This experiment is being conducted in support of one of the SNAP programs for which Atomics International is responsible. The SNAP fuel is a U-Zr hydride. The loss of hydrogen from the fuel by permeation through the thin cladding tube is controlled by a coating applied to the inside of the cladding. The exact nature of the coating material is classified and is of no importance in the discussion of this experiment.

The experimental assembly consists of a coated cladding membrane, a gamma heater, an outer casing, and out-of-pile systems for temperature control and permeation measurement. The design of the in-pile portion of the experiment is shown in Figure 1.

The experiment contains no fuel--gamma heating is depended upon to achieve the desired temperatures. The design temperature conditions for the coated cladding "membrane" are: 1500°F at the middle section (coated) and under 600°F at the upper (uncoated) end. These temperatures are attained in the ETR by utilizing gamma heat (5-10 watts/gram in graphite) and by a variable helium-nitrogen gas mixture in the outer annulus. A "gamma heater", which is a relatively massive part transfers heat to the test membrane by radiation.

The design allows hydrogen at approximately atmospheric pressure to be introduced into the coated cladding sample while a vacuum is maintained in the annular chamber surrounding the sample. The vacuum line passes up the inside of the instrument lead tube and is connected to a Veeco mass spectrometer modified to detect and measure hydrogen. A radiation monitor and an interconnected solenoid valve are included in the vacuum line to isolate the capsule from the out-of-pile system whenever high activation rates are encountered.

Small gas lines connected to the outer annulus between the gamma heater and the outer shell pass up the lead tube to the out-of-pile gas system and allow temperature control by variation of the gas mixture in the annulus. Thermal conductivity of the mixture is controlled by varying the ratio of helium to nitrogen in the system. The mixture of helium and nitrogen is manually adjusted to achieve the desired operating temperatures and the resulting ratio of pressures is monitored by a differential pressure switch installed between the two supply lines. This switch is sensitive to slight changes in pressure within the system and will automatically purge the capsule with helium if the gas ratio varies outside pre-set limits. A signal from the high temperature relays will also cause the capsule to be purged with helium thereby lowering the temperature.

This experiment was inserted in the ETR in August, 1963 and is expected to continue operation until the hydrogen monitoring system indicates that excessive permeation of the membrane is occurring or until an integrated fast dosage ($>$ 1 Mev) of 1.4×10^{21} nvt is achieved. To date, the experiment has operated at or near design temperatures and the hydrogen permeation and gas control systems have functioned satisfactorily.

Experience has shown that the temperature of the vacuum line for a short distance above the membrane is higher than anticipated.

The inside of this line is not coated with the hydrogen barrier and there is some permeation of hydrogen through that part of the tube. A second experiment of this type is now being assembled in which this portion of the vacuum line will be held at a lower temperature which should eliminate the problem.

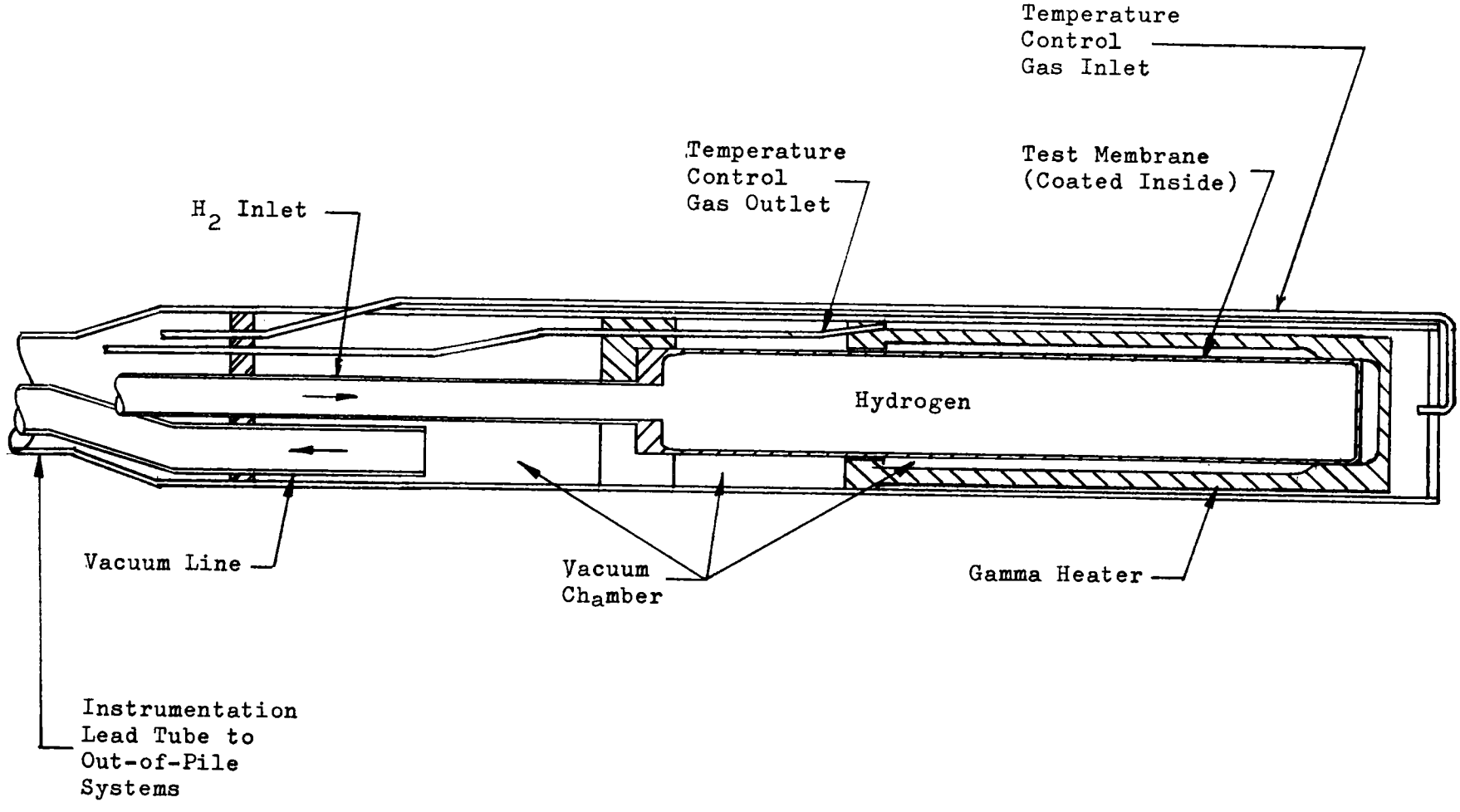


Figure 1

In-Pile Hydrogen Permeation Test

THERMIONIC CONVERTER CAPSULE

By

E. S. Beckjord

T. L. Gregory

Work Performed Under

U. S. Atomic Energy Commission

Contract AT(04-3)-189
Project Agreement 32

2.19.1

THERMIONIC CONVERTER CAPSULE

The direct conversion of heat to electricity by thermionic devices has been a subject of interest for many years. During the past ten years the experimental effort on thermionics has expanded many fold because of the belief that this is one of the more promising means of converting heat energy to electricity with reasonable efficiency and high reliability. By eliminating the need for rotating equipment, a light weight electric generator is obtained with a high power per unit weight ratio.

With the advent of nuclear energy as a heat source consideration of the use of thermionic devices has been directed principally to application as a topping device for central power station use and as a power source for space vehicles. The space vehicle application with its premium for a power source with a high power per unit weight ratio has been responsible for the accelerated research and development effort during the last few years.

A thermionic converter, in its simplest form, can be a gas-filled device with a hot cathode to emit electrons, a cold anode to collect the electrons, a suitable container to confine the gas and house the anode and cathode, and two electrical leads to the work load. Means must be at hand to heat the cathode and cool the anode. A method must also be provided for reducing the space charge barrier which is produced by the cloud of neutrons between the cathode and anode. In the thermionic device discussed in this paper, positive ions are used to neutralize the space charge and thereby reduce its barrier. The other method of reducing the space charge barrier - having the electrodes extremely close together - gives rise to some more difficult practical problems in fabrication.

Several thermionic converter capsules have been irradiated successfully in the General Electric Test Reactor (GETR). Capsule performance generally has been satisfactory. The following gives the objectives of the capsule tests, a description of the capsule, the experimental requirements of the test, and the hazards involved in conducting in-pile tests with this type of capsule.

Test Objectives - To test the feasibility of one of a series of in-pile thermionic converters for directly converting fission heat to electrical power.

a. The general purpose of this experiment is to evaluate correlation between in-pile and out-of-pile electrical performance.

b. Evaluate the direct conversion performance characteristics of a fission-heated tungsten emitter with a molybedenum anode.

c. Evaluate the compatibility between the fully enriched ceramic UO_2 pellet and tungsten emitter cladding as well as general tube structural materials performance.

d. Evaluate the effect of fission particles from the heated emitter on the tube materials and performance characteristics.

e. Determine the life of the tube and the ultimate cause of performance deterioration or failure.

f. Performance data required will include power output in watts/cm^2, short circuit current in amps/cm^2, power degradation as a function of time, volt-amp characteristics, effect of anode temperature on the power output.

g. Target converter output: 5 watts/cm^2 (8.2 cm^2 emission area), 0.7 volts, 58 amps.

Description of Capsule

The test capsule design is shown in Figure 1. The capsule consists of a thermionic diode with a tungsten clad fully enriched UO_2 fuel pellet as an emitter. The collector is molybdenum. The room temperature radial gap between the emitter and collector surface is 0.010 inch. The radial gap will open up to approximately 0.011 inch when at temperature. The upper end of the tube has three ceramic-to-metal seals while the lower end consists of a cesium reservoir. The tube is evacuated so a low pressure cesium atmosphere is generated when the cesium reservoir is heated.

The diode was assembled and the envelope heliarc welded shut using standard machine shop techniques. The assembled diode was then leak-checked and connected to a vacuum system through the cesium pump-out port tube and baked out in a muffle furnace at 1000°F for about 10 hours. It was then pinched off from the vacuum system, back-filled with cesium from a reservoir contained in the pump-out line, and finally pinched off. The diode was X-rayed and electrical tests performed to verify its electrical properties. A final mass-spectrograph leak check was then made.

The diode was assembled into the stainless capsule assembly using established fabricating techniques, and the containment vessel was welded shut. The capsule was then X-rayed and the electrical tests of the diode repeated from the "top" end of the capsule hose.

The thermionic diode is double contained in a stainless steel capsule shell. The cesium reservoir is surrounded by a copper heat distributor, a resistance heater, and a radiation shield. A 0.020 inch gap is provided between the radiation shield and the capsule containment tube. Cesium temperature will be measured directly by a thermocouple in the heat distributor around the reservoir and controlled by varying the He-A gas mixture in the gap. Fine temperature control will be accomplished with the resistance heater. The diode collector area is surrounded by a beryllium heat distributor, resistance heater, and radiation shield. A 0.020 inch gap is provided between the radiation hsield and the capsule wall. As in the case of the cesium heater, collector temperature will be measured directly by a thermocouple embedded in the collector and controlled by He-A gas mixture with resistance heater trim.

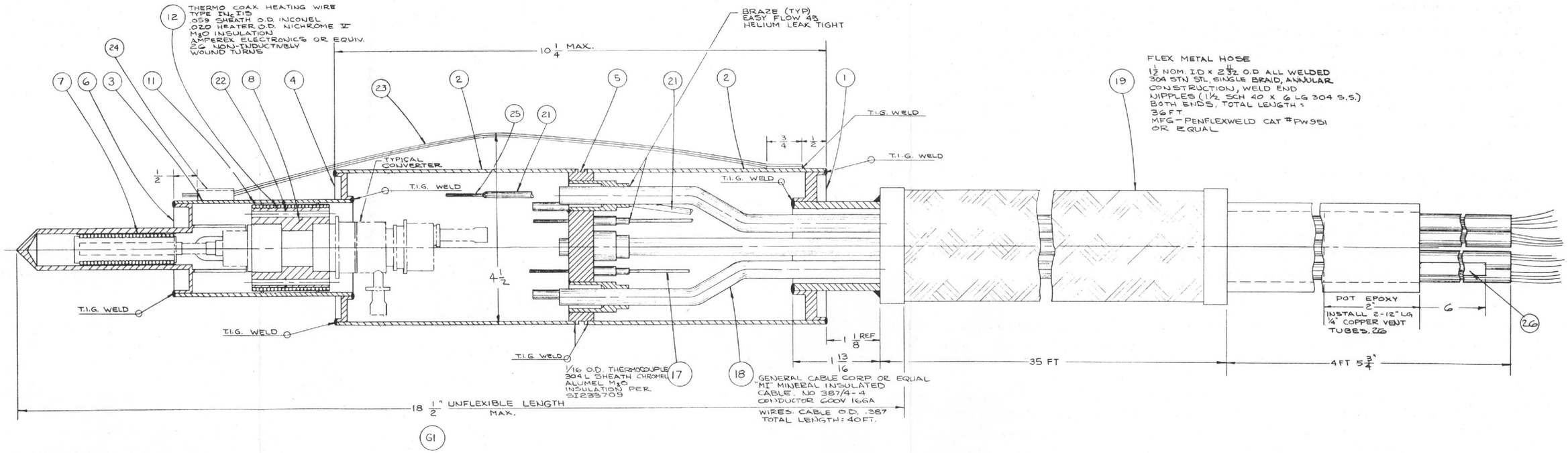


Figure 1. Thermionic Converter
2.19.4

The capsule lead-out system is contained in a 2-1/4 inch OD braid-covered flexible hose approximately 33 feet long. The system consists of 4-3/8 inch OD copper clad mineral insulated cables, each containing four 16-gage copper wires, three sheathed thermocouples 0.062 inch diameter, and two 1 8 inch OD stainless steel tubes for gas purging. All cable sheaths and tubes have been brazed or welded into the end flange of the capsule and are leak-tight as determined with a helium leak detector. The cables are sealed with special potted fittings to prevent gas from diffusing through the mineral insulation. The lead-out system is attached to a connector box on the third floor of the containment building. A connector cable is provided from the function box to the console. A console is provided for capsule control and experimental measurements.

Test Requirements

a. Irradiate capsule in the GETR 4 inch trail cable facility. Facilities must be capable of inserting and removing capsule during reactor operation.

b. Neutron flux required:

   Fast - none

   Thermal - 1.5×10^{13} nv unperturbed.

c. Gamma heat required:

   None.

   Avg. 1 w/gm with Peak/Avg. of 1.47 (integrated cycle) used for heat transfer calculations.

d. Fission power:

   250 watts for emitter temperature of 700°C.

e. Irradiation duration:

   To electrical life of the tube not to exceed 2 reactor cycles.

f. Experiment location relative to the core:

   4 inch Trail Cable Facility.

   Positioned vertically to achieve desired emitter temperature of 1700 ± 100°C. This will be accomplished by raising experiment as required. It will be near the top of the core due to relatively low flux requirement.

g. Fuel

   UO_2 pellet, 0.390" OD by 0.778" long

   Weight: 15.46 gms.

   Enrichment: 93% U-235

   Density ≈ 92% of theoretical.

h. Burn-up

None required for experiment. Actual burn-up will be calculated and compared to post-operational radiochemical analysis after irradiation.

i. Controlling Factors

Emitter temperature - 1700 ± 100°C; controlled by fuel vertical position.

Anode temperature - 600°C to 700°C; controlled by He-A gas gap and heater trim.

Cesium reservoir temperature - 360°C max; controlled by He-A gas gap and heater trim.

Hazards

a. The only hazard of consequence which might occur is the possible cracking of the ceramic-to-metal seal in the diode envelope during irradiation which would release fission products into the capsule vessel. The fission products would be prevented from entering the pool by the capsule vessel. The gas may eventually diffuse up through the gas lines. If the gas line fittings are kept leak-tight then none of the fission gas should escape into the reactor containment vessel.

This possibility has been minimized by using special potted fittings at the connector box. Care must be exercised during canal operation to cut the lead system after irradiation to prevent flooding of the capsule.

b. The fission product release and its effect upon the experiment is to be studied. The resulting gas release pressure will be quite small since the evacuated void volume in the diode is very large compared to the fuel volume.

The fuel pellet is not sealed in the tungsten cladding. Fission products will be released inside the diode assembly. This is double contained by the capsule shell and lead out system.

Other than the fission gas emanating from a possible broken seal there is no credible radioactive material release mechanism. Because of the relatively small fuel quantity present in the experiment and the short irradiation, only a small amount of fission gas will be generated.

No gas sampling will take place during the experiment.

c. No special shielding will be required during routine experimental operation.

d. Occasional radiological monitoring of the gas lines should be performed routinely to detect a possible ceramic-to-metal seal leak or a leak in the potted cable seals. No radiation alarm or continuous monitoring equipment is provided.

e. Should the monitoring instruments detect high reactivity levels in the gas lines, the experiment should be raised to 2 - 3 feet above the core. Should the temperature exceed 1900°F all power to the heaters must be cut and the experiment raised as above.

f. Should any emergency occur, the responsible reactor test engineer must be notified and the responsible sponsor engineer must be notified immediately.

IN-REACTOR FATIGUE TESTING
STRAIN CYCLE TEST

By

M. B. Reynolds

Work Performed Under

U. S. Atomic Energy Commission

Contract AT(04-3)-189
Project Agreement 13

2.20.1

IN-REACTOR FATIGUE TESTING - STRAIN CYCLE TEST

The importance of elevated temperature ductility under radiation as a requirement for nuclear superheat fuel cladding lead to the development of the technique to be described.

It was felt that since cladding would be subjected to a number of stress-reversals in service, some type of slow-cycle strain fatigue test would be preferable to a simple creep or stress rupture test. Initial efforts were directed toward development of a bellows-actuated device which would subject a tubular specimen to cyclic uniaxial strain. After expenditure of considerable effort, this approach was abandoned because of the inability of vendors to fabricate bellows of adequate load capacity under the temperature and neutron flux conditions existing within a nuclear reactor. A simple comparative test using standard thin-walled fuel cladding tubing was finally devised. In this test, gas pressure is used to alternately expand and compress a specimen of tubing between a pair of rigid concentric mandrels, as shown in Figure 1. The relative strain range is determined by the diameters of the inner and outer mandrels relative to the specimen diameter and wall thickness. The strains experienced by the specimen are multi-axial and, particularly at the tube ends, are somewhat difficult of interpretation; but this disadvantage is considerably outweighed by experimental simplicity and the fact that deformations are of the type to be expected in service.

Experimental:

Slow-cycle fatigue tests based on this principle have been carried out both out-of-reactor and in-reactor. Specimen size has been standardized at 1.250-in. nominal outside diameter and 0.016-in. wall thickness with an active length of 3 in. The separate parts and a completed specimen-mandrel assembly, as used in out-of-reactor tests, are shown in Figure 2. Ends were sealed by inert gas shielded arc welding. The specimen is cycled by application of argon pressure alternately to the inside and outside of the specimen. Operation of the cycling mechanism may be described by reference to Figure 3. Assume valve V_{11} is open and valve V_{10} is closed - that is, gas pressure is applied to the inner side only of the specimen. Operation of the timer will then close valve V_{11} and simultaneously open valves V_{21} and V_{10}, relieving pressure on the inside of the specimen and applying pressure to the outside of it. After a few seconds have passed the time delay switch D_2 opens allowing valve V_{10} to close. The next operation of the timer closes valve V_{21} and opens valves V_{11} and V_{20}. After a few seconds valve V_{20} is closed by time delay switch D_1. The number of complete cycles is recorded by magnetic counter R. It will be noted that outlet valves V_{10} and V_{20} are normally closed, each opening briefly once per cycle to relieve pressure on one or the other side of the specimen. With these valves closed either leakage in an inlet valve or specimen failure will result in simultaneous appearance of pressure on both sides of the specimen. When this occurs,

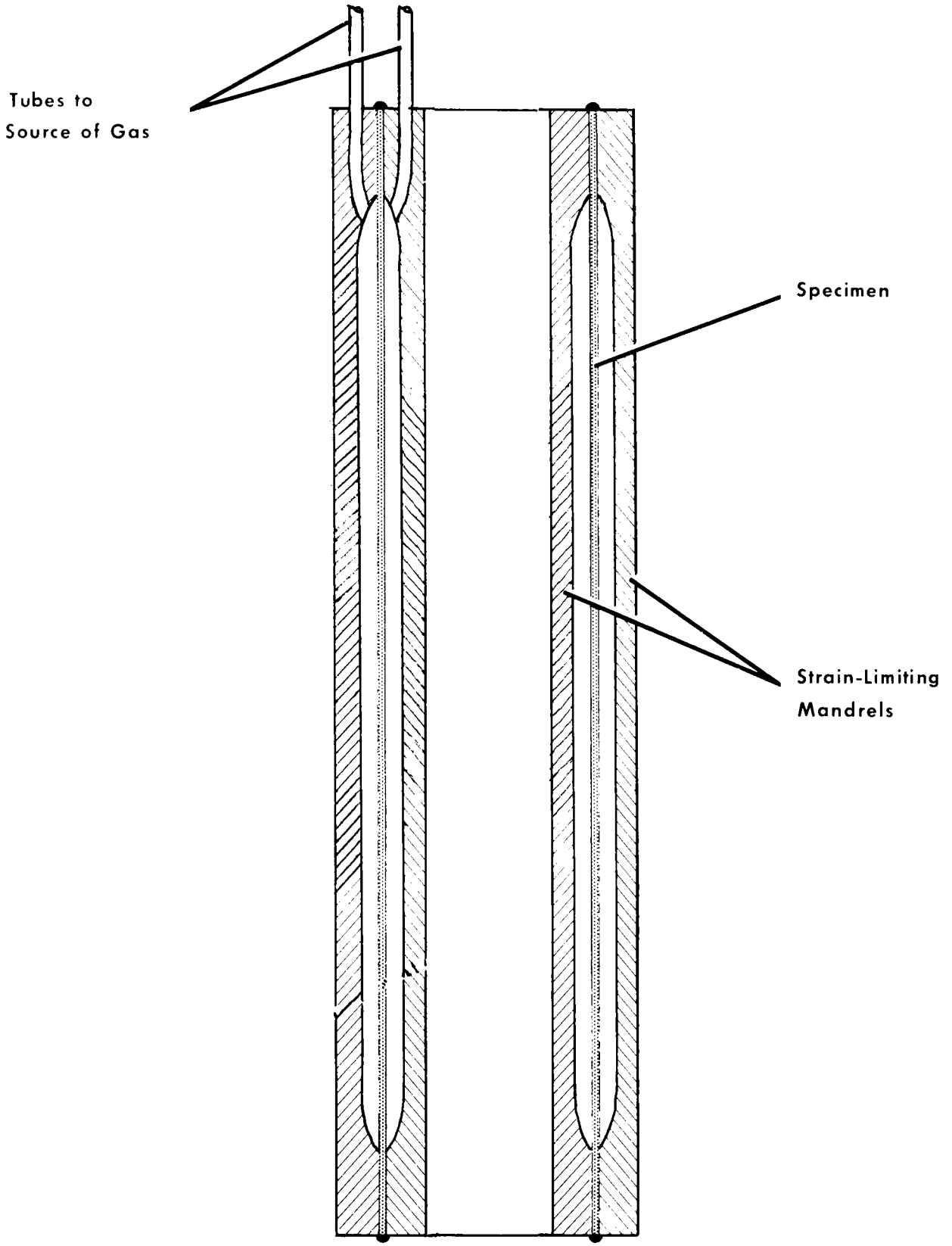


Figure 1 STRAIN-CYCLE CAPSULE - SCHEMATIC

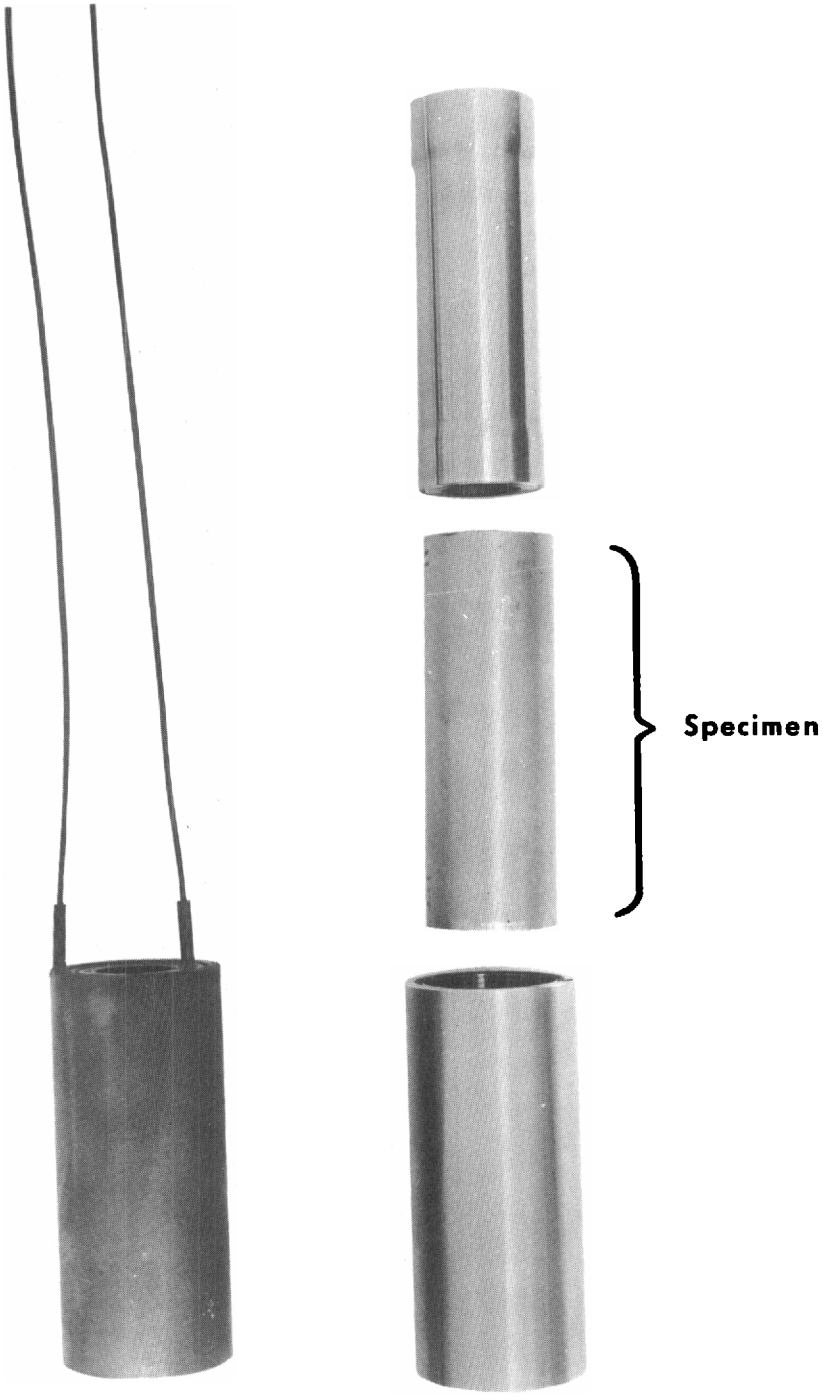
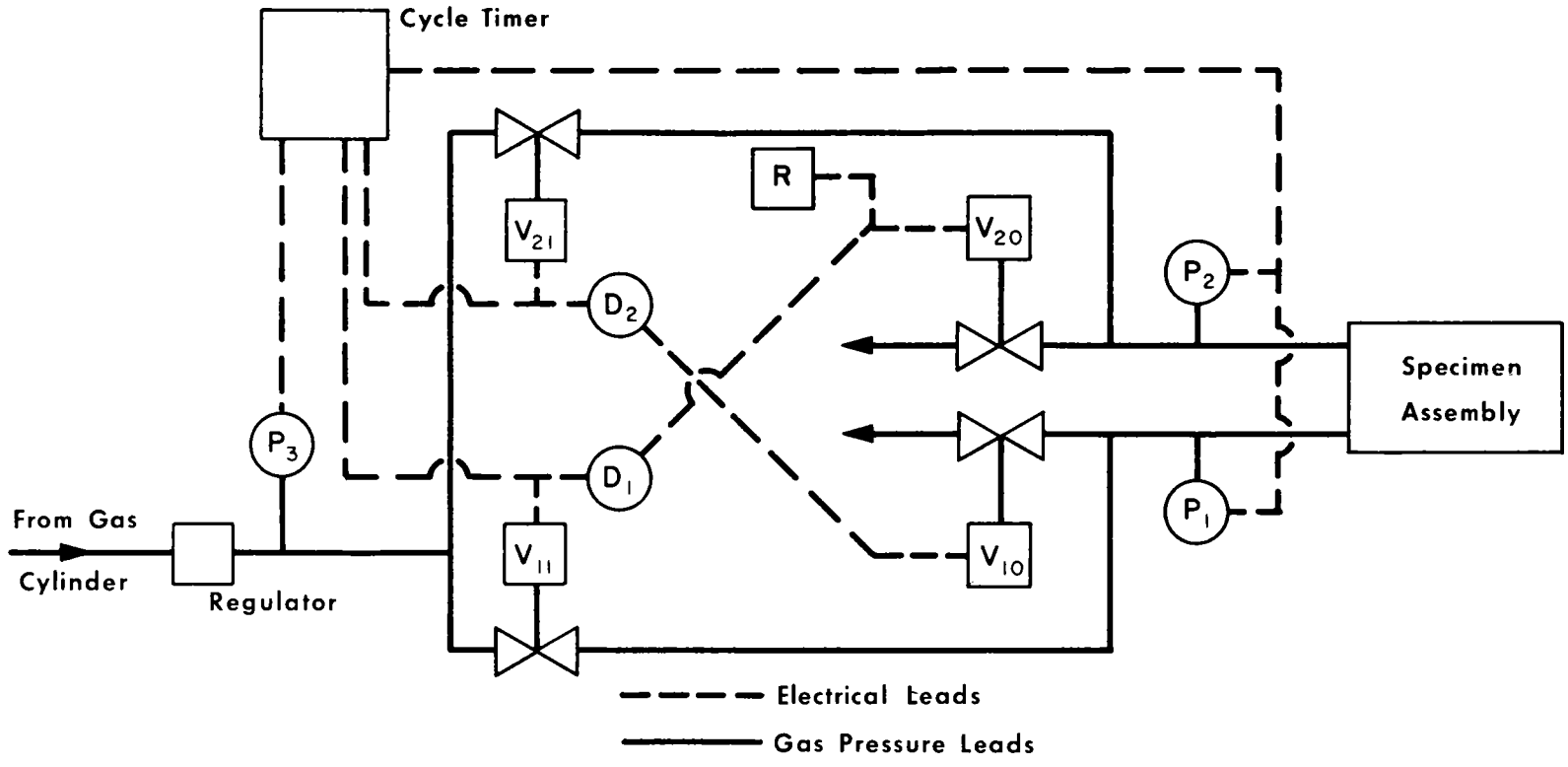


Figure 2. SPECIMEN AND HOUSING



V_{11} Inlet Valve, Inner
V_{21} Inlet Valve, Outer
V_{10} Outlet Valve, Inner
V_{21} Outlet Valve, Outer
P_1 Pressure Switch, Inner
P_2 Pressure Switch, Outer
P_3 Pressure Switch, Low Limit
R Cycle Counter
D_1, D_2 Time Delay Switches

Figure 3. CYCLING MECHANISM SCHEMATIC

parallel-connected pressure switches P_1 and P_2 both open, releasing a holding relay and stopping the cycling mechanism and cycle counter. P_3 serves as a low-limit switch, stopping the timer when gas source pressure falls below a preset minimum value.

Activation pressures in excess of 1000 psi have been used throughout the series of experiments, resulting in specimen hoop stresses of approximately 40,000 psi compared with yield strength of less than 30,000 psi for the type 304 stainless steel specimen material.

Initial tests were carried out on annealed type 304 stainless steel tubing. Out-of-reactor tests were carried out at 1300 F with the exception of two tests carried out at 1200 F. Total nominal strain ranges are computed from the relation

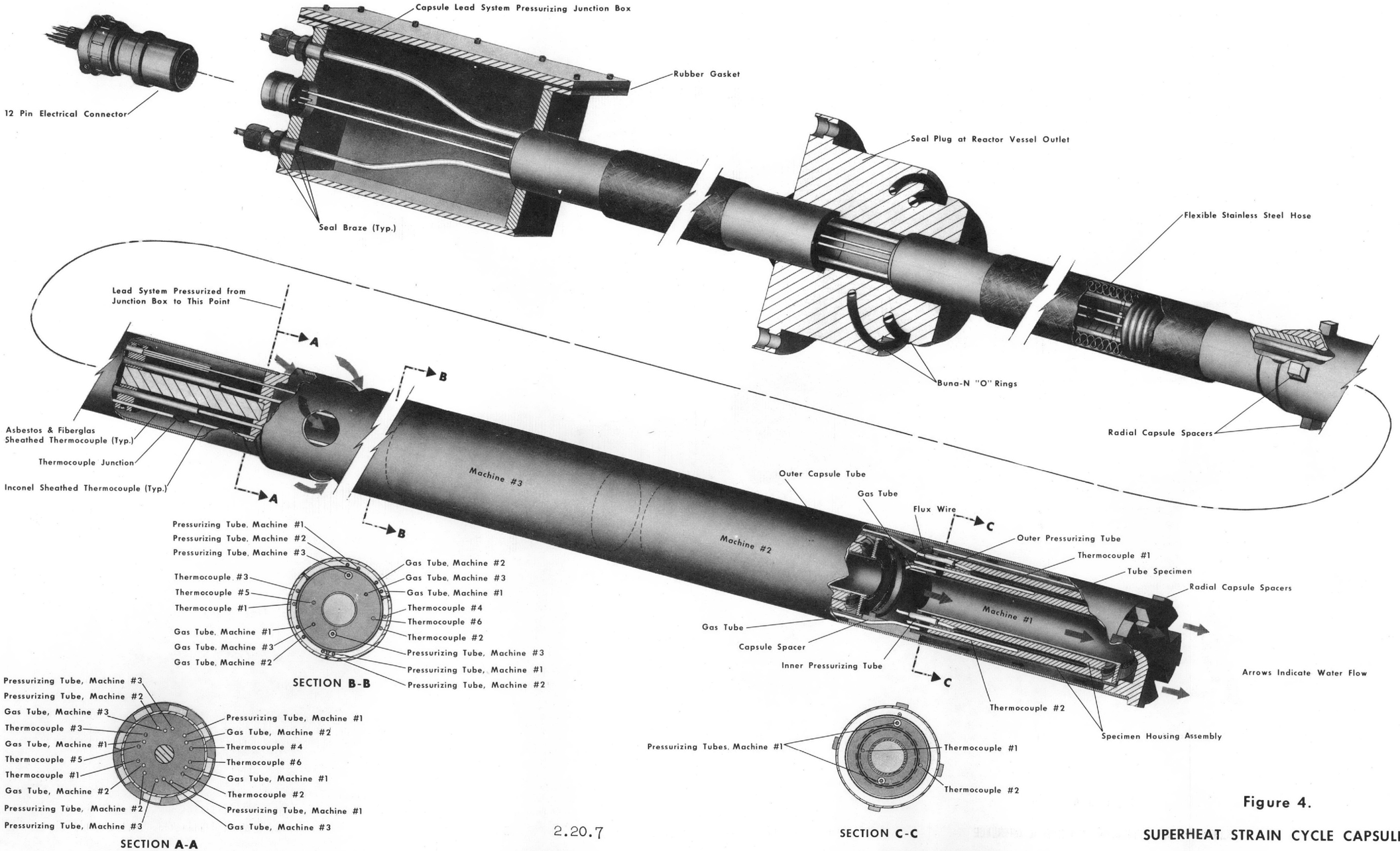
$$\Delta\epsilon = \frac{D_o - D_i - 2t}{D_s}$$

where:
D_o = inside diameter of outer mandrel,
D_i = outside diameter of inner mandrel,
D_s = average diameter of specimen, and
t = specimen wall thickness.

Strain reversal occurred every 15 min.; that is, a complete cycle required 30 min. Plastic strain range may be estimated by subtracting from the total strain range twice the elastic strain the specimen material will support -- that is, twice the elastic limit at the test temperature divided by the Young's modulus at this temperature. Estimates of actual strain ranges have been made from cross sections of failed specimens.

For in-reactor tests the specimen and mandrel assembly is completely enclosed by a concentric can with a hollow center. Specimen temperature is maintained by gamma heating. Temperature is controlled by varying the composition of a helium-nitrogen mixture flowing through the annular spaces between the mandrels and the enclosing can. The configuration of the in-reactor assembly (containing three specimens) is shown in Figure 4. Five of these assemblies have been irradiated in the General Electric Test Reactor.

In the first of these gamma heating calculations were considerably in error with the result that desired operating temperatures were not obtained. In other cases inadequate quality control in fabrication resulted in assemblies which upon installation in the reactor proved to be nonfunctional. More rigid quality control has resulted in improved performance. Although it is simple in concept, dimensional tolerances in this capsule are exceedingly rigid and a high order of skill is required in its fabrication.



2.20.7

Figure 4.

SUPERHEAT STRAIN CYCLE CAPSULE

CERAMIC FUEL SELF SERVE FACILITY S/P 485

P. S. J. Ritchie
A.E.R.E. Harwell

Information is required on the structural stability of fissile ceramic fuels under high flux at known temperatures. Many experiments require irradiation times of less than one reactor cycle, creating a need for equipment which can be loaded and unloaded at power.

The self-serve facility is designed to fulfill these requirements and provide in addition a means of measuring the specimen heat output.

Description (Fig. 1)

1. Inpile Section

The principal component of the inpile section is an aluminium tube through which the specimen is pushed into a position near the reactor core centre-line.

Demineralised water is circulated through the tube, over the specimen container and back to the rig head via an aluminium return pipe.

Both specimen and water return tubes are spiralled through 360° in the shield plug to reduce the "shine path". The shield plug is of standard C.I. construction with an iron shot/resin filling, 4" lead nose block and 2 mm. thick cadmium disc nose plate.

2. Specimen magazine (Fig. 2)

This is cylindrical and rotates on bearings carried by a shaft attached to the rig head. It contains three specimen positions and a sealing plug, arranged to be aligned in turn with the specimen tube. Backing plugs containing the specimen capsules are fitted into the specimen positions.

"U" lip seals on the magazine and shaft in conjunction with "O" rings on the plugs provide a water seal.

A lead filled shield block is located around the magazine.

3. Winch drum magazine (Fig. 2)

Three winch drum housings and the sealing plug handling tube are located in a lead filled drum.

It is mounted on rollers and rotated by a chain drive carrying with it the specimen magazine to which it is spigotted by the plug handling rod.

Pre-set stops ensure the correct alignment of the specimen containers when changing specimens.

With the winch drum housings in place and active thermocouples on the drums the magazine can be demounted and used as a transport flask for the drums.

4. Winch Drum Housing

These contain the drums, each with its own drive, and carry cropping plates at the inpile end. When the housing is rotated within the magazine the specimen withdrawal tube can be sheared to remove the thermocouples from the specimen container.

5. Specimen Container

The cylindrical specimen is contained in a double stainless steel capsule. An annulus is formed between the containers for controlling the heat transfer through the helium gas container filling.

The outer container is $3\frac{3}{4}$" long with a basic diameter of 7/8".

"Dumbell" sections at each end locate the container concentrically in the tube and facilitate its movement through the spiral of the specimen tube.

A 9ft. long withdrawal tube of .092" dia. annealed S.S. tube containing a S.S. sheathed thermocouple is welded to the rear of the container.

The container with tube attached is made up as a unit with a backing plug, drum and ferrule for engagement with the drum.

This unit, containing its own water seals is tested prior to loading, new seals drum etc. are fitted for each irradiation.

6. Instrumentation

Six C/A thermocouples are equispaced along the specimen tube. A further 6 are disposed at 120° around the tube before and after the specimen irradiation position and all are fitted to a recorder. Another thermocouple is fitted in a re-entrant probe at the centre of the specimen location in the specimen tube. Water flow is measured and together with the thermocouples enables a heat balance to be made for the specimen.

A T/C from the specimen records temperature either on its surface or within it.

Safety aspects are catered for by fitting leak detectors, low flow trips and a burst can detection circuit.

7. Operation

With the sealing plug positioned in the specimen tube, and the winch drum housings removed from their magazine three specimen container assemblies are loaded into the specimen magazine.

Winch drum housings are fitted and the drums located.

The sealing plug is withdrawn and the magazine rotated to the first specimen position.

The specimen is moved into the irradiation position by pushing the withdrawal tube.

After irradiation the drum is rotated by means of its handle and the specimen withdrawn into its backing plug.

On completion of the irradiation of the three specimens the sealing plug is located in the specimen tube.

The drum housings are rotated, cropping the specimen withdrawal tubes and the drums are removed in the winch drum magazine.

The backing plugs containing the specimens are drawn into a special flask and removed.

The winch drum magazine is then replaced and the next set of specimens loaded.

8. General

Specimens can be located in various fluxes up to the maximum of 5×10^{13}. Manual oscillation experiments are possible over a range of flux levels.

9. Experimental Parameters

Specimen size:- 0.75" ø x 3" long

Composition:- UO_2 or UO_2 with non-fissile non-toxic additives.

Specimen heat output:- 4 kW (max.)

Specimen surface temperature:- upper limit 500°C

lower limit 50°C

Activity at unload:- 1000 curies (max.)

Specimen δk:- 0.1% max.

Flux in water tube, specimen absent $\simeq 9 \times 10^{13}$ n/cm^2 sec.

Flux with specimen present $\simeq 5 \times 10^{13}$ n/cm^2 sec.
Flux falls approx. as a cosine function along the inpile 4 ft. of the specimen tube.

Transit time for specimen:-
From magazine to irradiation position $\simeq$ 10 sec.
Transit time in flux up to irradiation position $\simeq$ 5 secs.

Dwell times:- minimum practicable - 1 sec
maximum permissible - 10 days

Successive irradiations:-
Three can be made in each loading. Operating time between each irradiation approximately 5 mins.

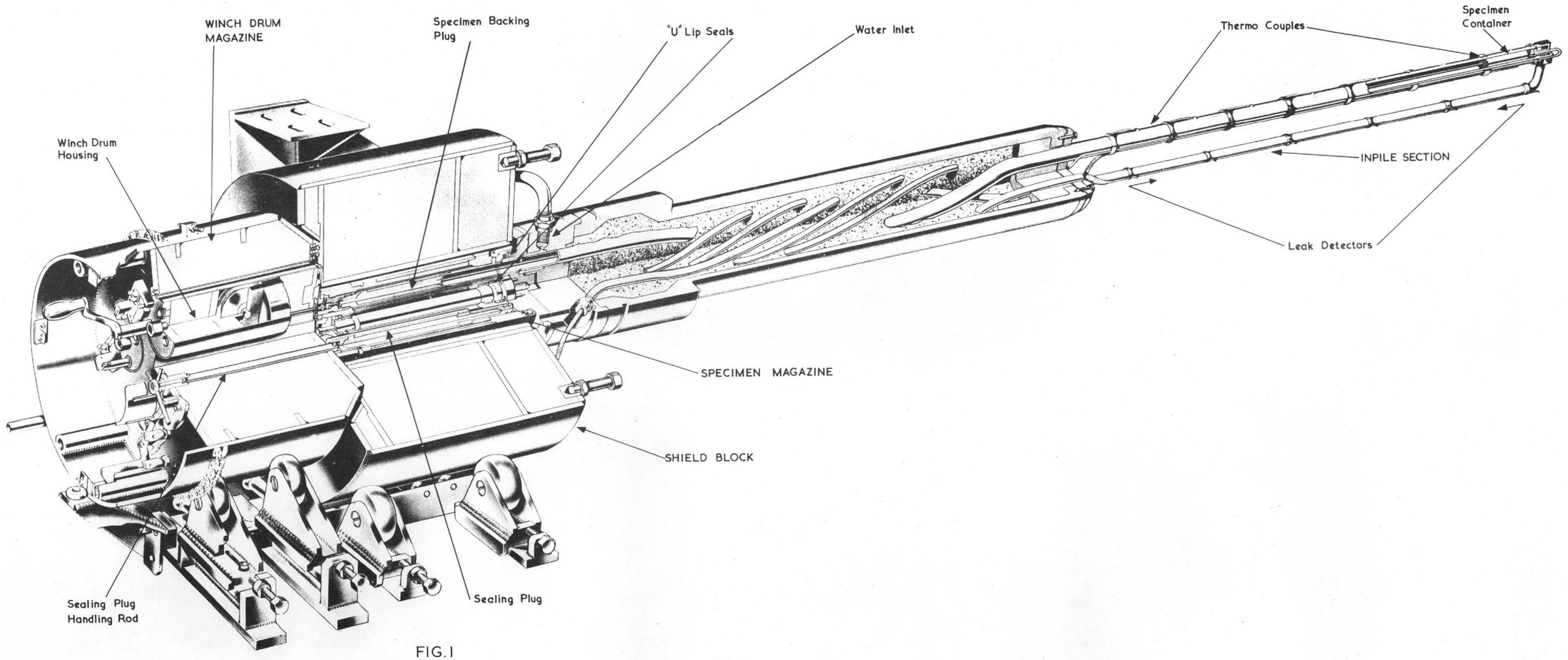

WINCH DRUM
MAGAZINE
Specimen Backing
Plug
"U" Lip Seals
Water Inlet
Thermo Couples
Specimen
Container
Winch Drum
Housing
INPILE SECTION
Leak Detectors
SPECIMEN MAGAZINE
SHIELD BLOCK
Sealing Plug
Handling Rod
Sealing Plug

FIG. I

RECENT ADVANCES IN NONDESTRUCTIVE EXAMINATION OF IRRADIATED FUEL CAPSULES

by

W. N. Beck
Argonne National Laboratory

In the development of fuel material for reactors, it is essential to determine experimentally the characteristics of the material under conditions representative of actual in-pile operation. As a means of accomplishing these objectives, fuel specimens may be encapsulated, introduced into an operating reactor, and subjected to preselected periods of irradiation. Postirradiation examination and measurement of the specimens are used to establish the operational characteristics of the fuel.

The procedure generally applied is to perform several experiments with different capsules, each of which is irradiated at different time and temperature conditions. Subsequent postirradiation inspection of each specimen will determine if additional irradiations are required.

In order to minimize the number of irradiations required to evaluate a particular fuel material, nondestructive examinations of the encapsulated specimens are utilized at Argonne National Laboratory in order to determine the condition of the specimens. On the basis of this examination, the capsule may be opened and the specimens may then be directly examined or the experiment may be returned to the irradiation

facility if so desired. These requirements would ordinarily be relatively simple for a radiographic inspection method. However, the technique is complicated by the highly radioactive nature of the inspection objects.

Among the techniques which have been proposed and utilized are X-radiography utilizing collimating slits. In this particular technique, the object is isolated from the radiographic film by a high absorbing mass which has a series of collimating slits. The slits are made to traverse a path which exposes the film to a moving line of X-rays and the gamma rays emitted from the object. The technique has been successfully utilized by Dounreay on 550 C-Mev sources.(1) Exposure times are dependent upon source intensity, X-ray kilovolt level, film to X-ray source distance and speed of film, but average exposure times have been in the order of 10 to 30 minutes. The equipment is usually incorporated in a shielded cell and remotely operated.

A simple pinhole gamma camera has also been successfully utilized for imaging highly radioactive sources. The technique duplicates the well known optical pinhole camera differing only in that special consideration must be given to a heavily shielded pinhole aperture. A camera installation for irradiated capsules has been built and tested at Argonne(2) (see Figure 1). A pinhole aperture which has proven effective was machined out of a solid block of natural uranium. The aperture was formed by 2 cones lying on the same axis, the base of the cones at the face of the block and the apexes overlapping at the center opening a 0.020 in. hole. The angle of the cone forming the aperture does not exceed 12 degrees.

For capsules having a gamma activity of 5×10^3 R/hr at 1 foot, the source-to-pinhole distance was 4 feet. An X-ray film holder was posi-

tioned an equal distance away from the pinhole block so that the image maintained a one-to-one ratio to the source size. The film found to be most suitable was Ilford Type G. The film holder is a cardboard cassette backed with 0.010 in. of lead. The resolution obtained has been measured to be approximately 0.03 in.

A shielded camera was constructed to accommodate an 11 foot irradiation capsule assembly in order to perform autoradiographs on top or near a reactor. An existing fuel transfer cask was modified by incorporating a slot in the wall and attaching a shielded extension tube. The pinhole block was mounted at the end of this extension. A 4 foot aluminum casing bolts to the extension tube and supports the film holder.

In operation, the shielded camera is mounted over a reactor vertical irradiation facility. The irradiation experiment is drawn into the camera to a position where the irradiation capsule aligns with the slot in the shielding. The exposure time for the film is dependent upon the gamma intensity of the specimens but generally varies from 30 minutes to 1 hour.

A neutron radiography inspection method makes use of a neutron activated screen to detect the imaging beam after it has passed through the inspection object. The screen contains a radioactive image which is then made visible by allowing the screen to decay in close contact with a photographic film. Since the gamma activity from the specimens will not influence the radioactivity of the detecting screen, it will therefore not influence the final radiograph.

Initial tests of the technique with irradiated capsules were conducted at Argonne in a temporary enclosure adjacent to the Juggernaut re-

actor.[3] A complete gamma and neutron shield was then fabricated which fits the horizontal beam hole, 6H1, of the Juggernaut reactor. The portable enclosure is equipped with a cassette discharge and recharge magazine which permits taking several radiographs without having to disturb the capsule.[4] An existing fuel transfer cask was modified in order to handle a complete 11 foot irradiation capsule assembly from the CP-5 reactor (see Figures 2 and 3). The beam hole opening at the side of the Juggernaut reactor is located 26.5 in. above floor level. The collimated beam aperture is 2.5 in. and has a neutron beam intensity on the order of 10^7 neutrons/cm^2-sec at a cadmium ratio of approximately 3.6 to 1. The gamma activity of the irradiation capsules radiographed has been measured to be in excess of 5×10^3 R/hr at 1 foot. Screen materials of rhodium, silver, indium, dysprosium, and gold have been used. The screen is encased in a 5 x 7 in. aluminum cassette which fits the magazine of the neutron shield. The screen thicknesses are 0.010 in. and the exposure time to the neutron beam varies as to the half life values of the screen as well as the emulsion speed of the film which the irradiated screen is transferred to.

In actual practice, a preliminary alignment radiograph is made to determine the proper orientation of the capsule. A rhodium screen is exposed to neutrons for 15 seconds. The screen is then transferred to polaroid X3000 film and allowed to decay for one minute. The film is then immediately processed and the orientation of the inspection object is determined. When the alignment is as desired, an indium screen is placed in a cassette and inserted in the exposure position. The neutron exposure required is five minutes. The indium screen is then taken into

a darkroom and placed in a cassette containing 2 sheets of Kodak Type AA X-ray film, one on each side of the screen. The cassette is closed and allowed to decay for one hour before the film is developed.

Radiographic contrast sensitivity of this technique and with indium screens is on the order of 1 to 2 percent. Resolution is better than 0.005 in.

A comparison of results of pinhole autoradiography and neutron radiography is shown in Figure 4.

Nondestructive examinations of irradiated fuel capsules greatly reduces the cost and time required to obtain final irradiation results. When we first recognized the need for a nondestructive inspection method, we designed a collimated slit X-radiographic scanner similar to the technique utilized by Dounreay. A further evaluation of the installation indicated that the system would be of limited use because of the high gamma intensity of our irradiated capsules. The technique also required a sizable physical installation.

A pinhole gamma camera was therefore designed, built and utilized which was compact and capable of handling high gamma intensity sources. With this technique deformation of fuel specimens could be observed and measured. The condition and placement of other less radioactive components within the capsule, however, were completely obscured or unintelligible.

At this particular time an independent research program was being conducted at Argonne on the applications of neutron radiography. It appeared that this technique would be adaptable to the irradiated capsules, in that

gamma rays would not affect the detection screen. Also, the technique would show all of the capsule contents instead of just the fuel specimens. The preliminary tests showed considerable promise and subsequently a complete installation was fabricated as described above. Neutron radiography is presently being used to the exclusion of other techniques. The resolution and definition compares favorably with the standards of commercial X-radiography.

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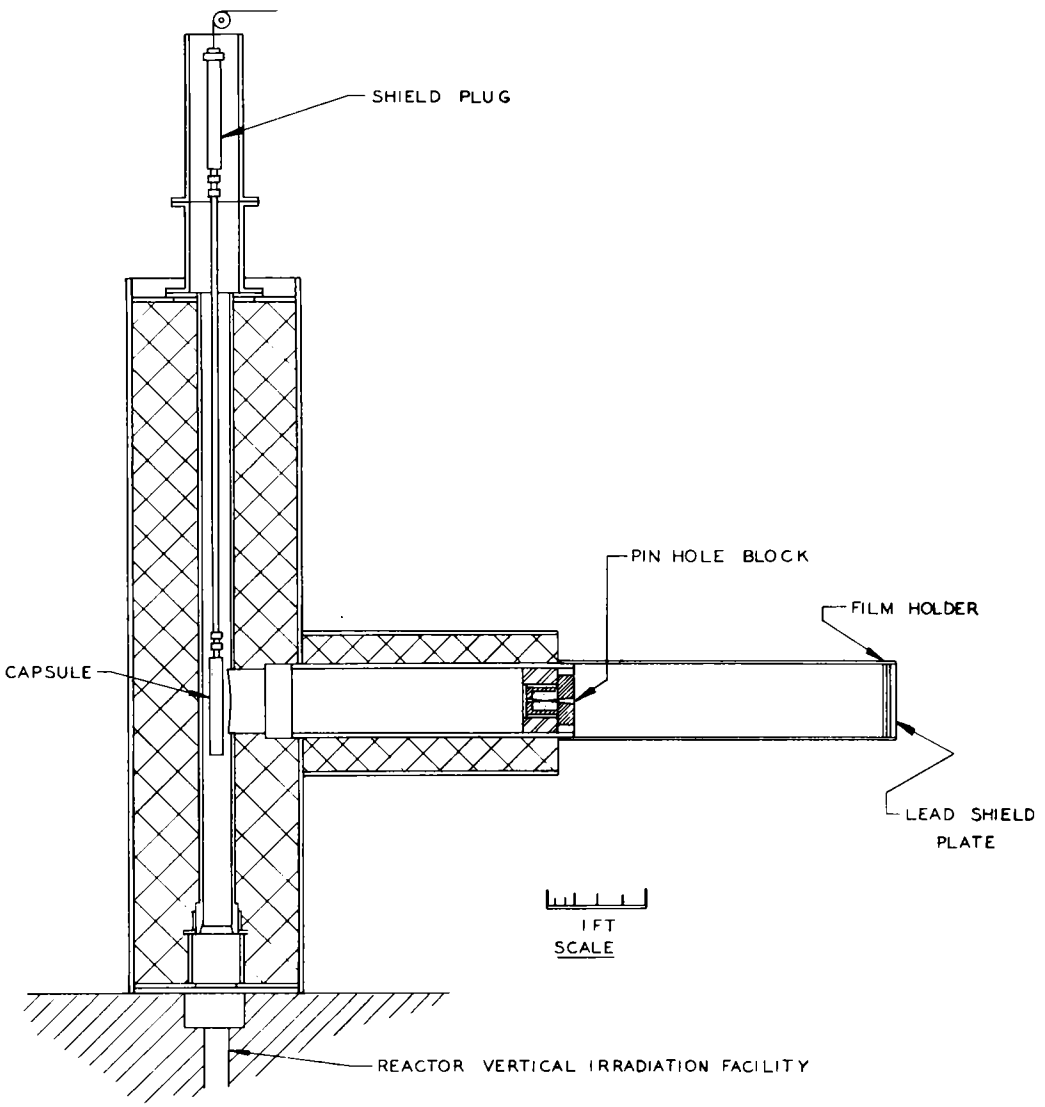


Figure 1. Diagram of Shielded Pinhole Autoradiographic Camera for Use with Highly Irradiated Fuel Capsules.

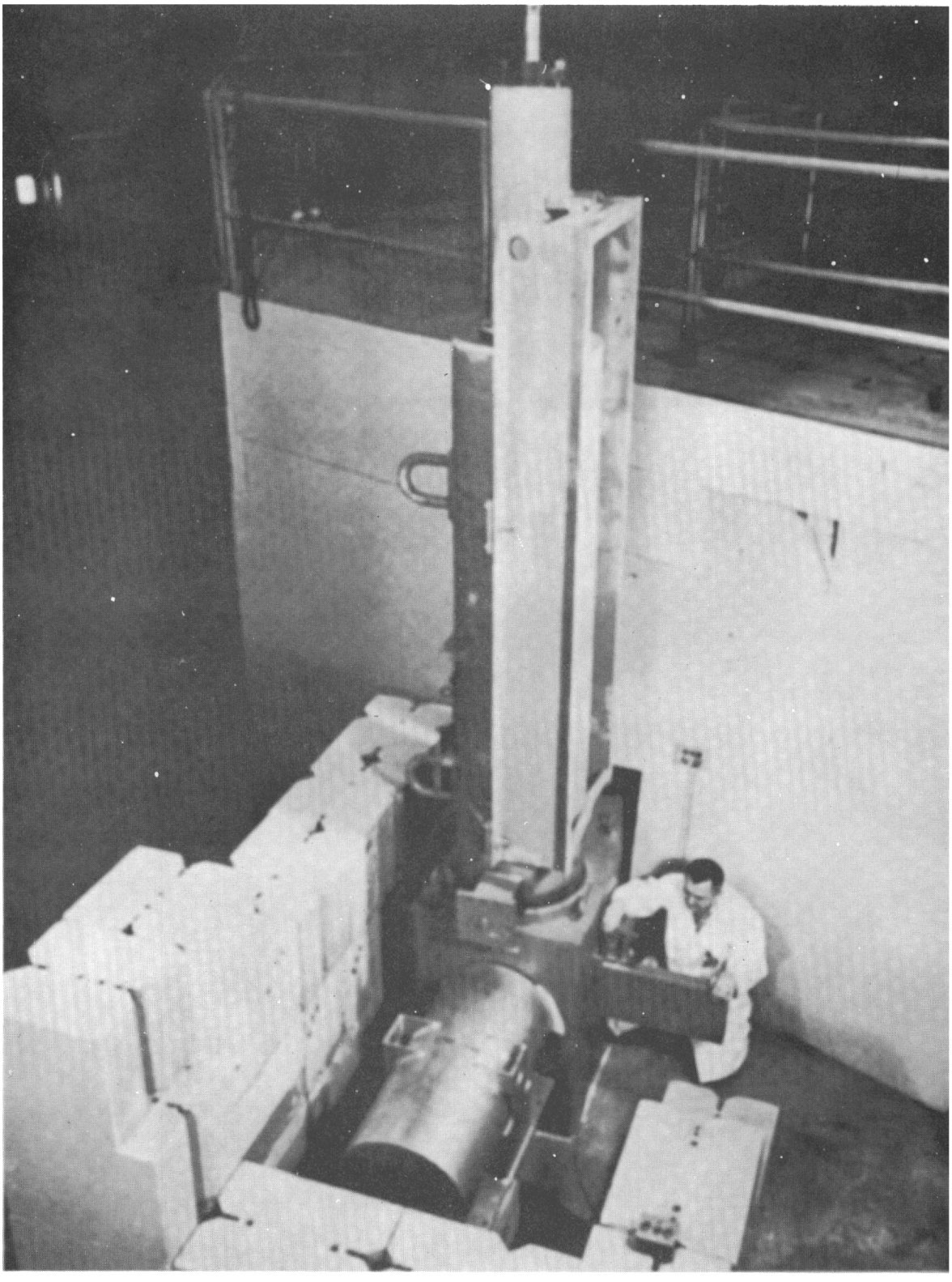


Figure 2. Neutron Radiographic Facility Adjacent to the Juggernaut Reactor. The Shielded Transfer Cask for Handling a Complete CP-5 Irradiation Experiment is Seen Installed on the Facility.

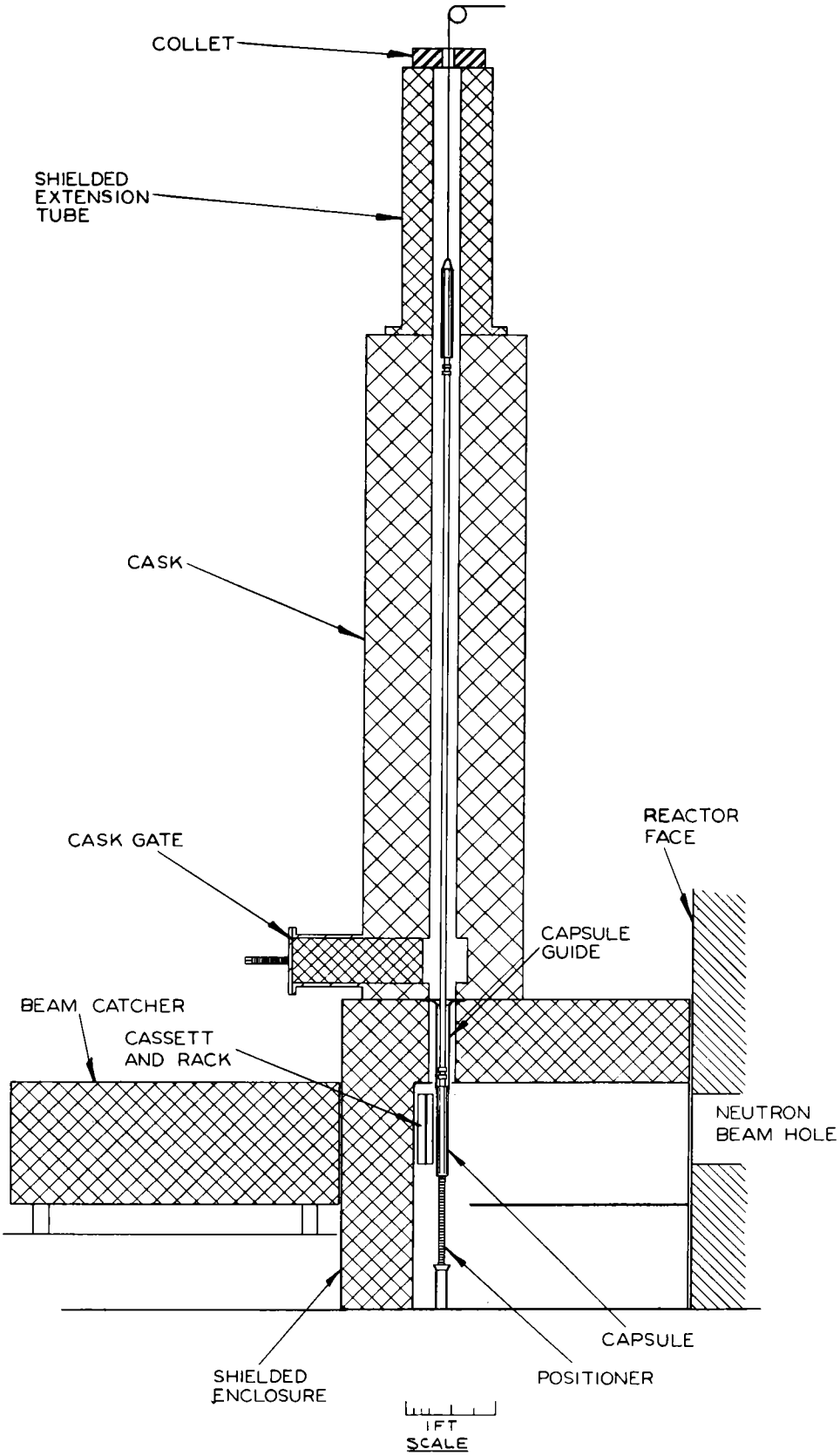
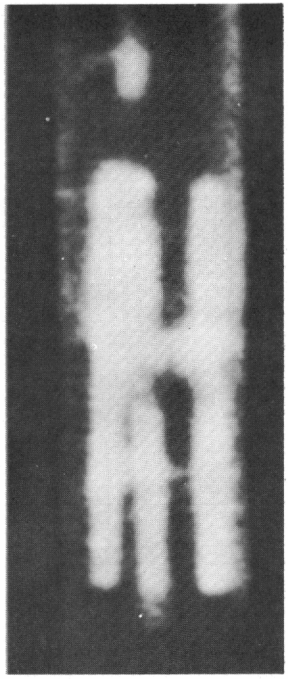
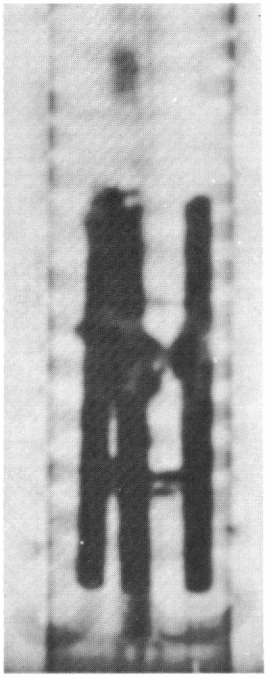


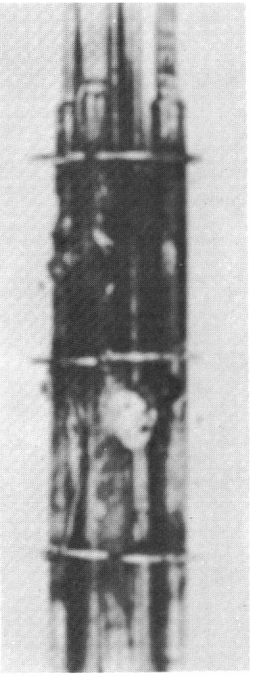
Figure 3. Diagram of Neutron Radiographic Facility for Use with Highly Irradiated Fuel Capsules.



Gamma Autoradiograph of Unopened Capsule



Neutron Radiograph of Unopened Capsule



Optical Photograph of Clad Fuel Specimens after Capsule was Opened

Figure 4. Comparison of Gamma Autoradiography and Neutron Radiography with Actual Appearance of Highly Irradiated Fuel Specimens.

III. MULTIPLE TOPIC PAPERS (CAPSULES, DOSIMETRY, HEAT TRANSFER, etc.)

CAPSULES FOR IN-REACTOR EXPERIMENTS AT CHALK RIVER

by

V. Fidleris

Atomic Energy of Canada Limited

Chalk River, Ontario
October, 1963

At the Chalk River Nuclear Laboratories increasing emphasis is being put on in-reactor experiments and measurements. This paper provides a brief survey of the work for discussion at a meeting on "Problems in Irradiation-Capsule Experiments" convened by the USAEC. Against each item is given the name of the individual primarily responsible, to aid further exchange of views and information.

SECTION I - NOVEL CAPSULE DESIGNS

1. In-Reactor Creep Capsule - V. Fidleris

The design of the Chalk River creep capsule (1-3) was governed by the available space inside the fast neutron rod in the NRX reactor. After allowances had been made for a heater, a hole of only 0.68 in. diameter was left to accommodate the creep machine. This restriction had the advantage of keeping the temperature rise due to gamma-heating to a reasonably low level, but made the machining of the miniature components difficult.

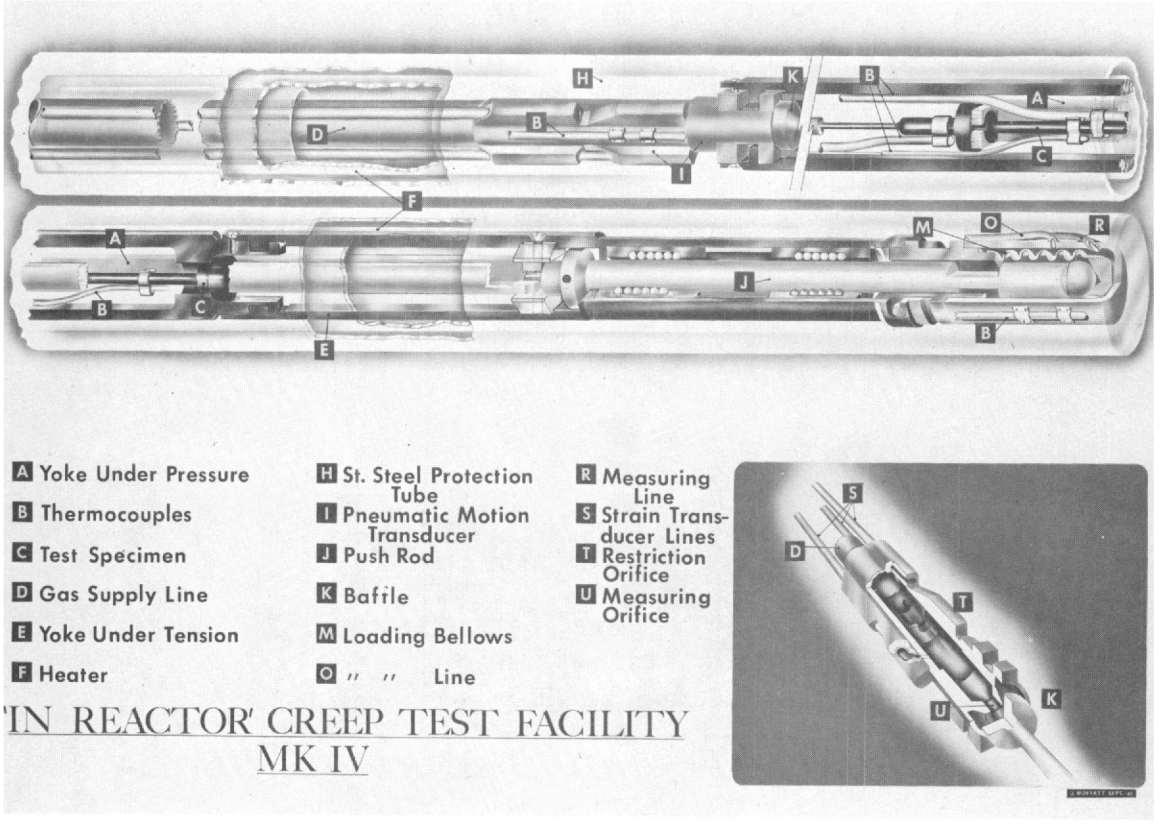


Figure 1. In-Reactor Creep Test Facility, Mark IV

Figure 1 is a sketch of one of the latest creep capsules, similar to one which has now been operating satisfactorily in the reactor for about 3000 hours.

The specimen is 2 in. long and 0.040 or 0.080 in. diameter. The load is applied by means of a bellows at the bottom of the capsule and transmitted to the specimen by a series of yokes that must be guided. The strain is measured with a pneumatic gauge that can detect a change in length of 10 microinches. The temperature is measured with stainless steel clad iron-constantan and chromel-alumel thermocouples and controlled with three-term electronic process controllers through a multi-section eight killowatt heater. The first tests were carried out in air and required forced cooling to reduce the temperature below 400°C when the reactor was at full power. In the latest (Mark IV) capsules the test atmosphere is helium and no forced cooling is required above 170°C at an NRX fast flux of 1×10^{13} n/cm^2 (E > 1 MeV). This is considered to be the limiting flux for the Mark IV capsule. At higher fluxes the temperature gradients along the specimen increase very rapidly because of gamma-heating in the yokes, as determined from measurements on a creep machine that had thermocouples embedded in the creep specimen and various parts of the assembly. A new capsule (Figure 2), being developed for use in higher fluxes, will have the yokes replaced by a cylindrical pressure vessel.

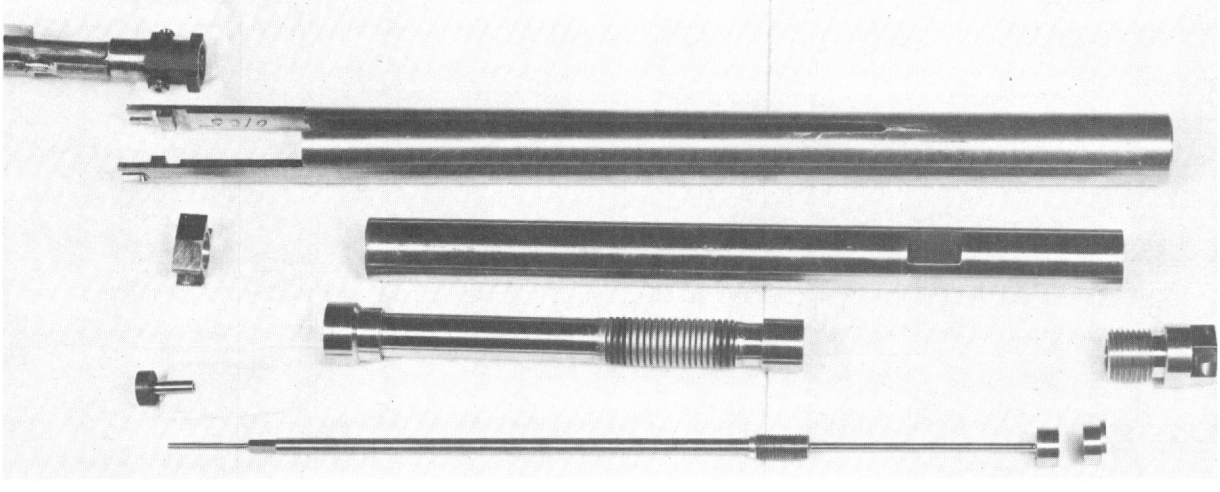


Figure 2. High Flux In-Reactor Creep Test Facility, Mark VI

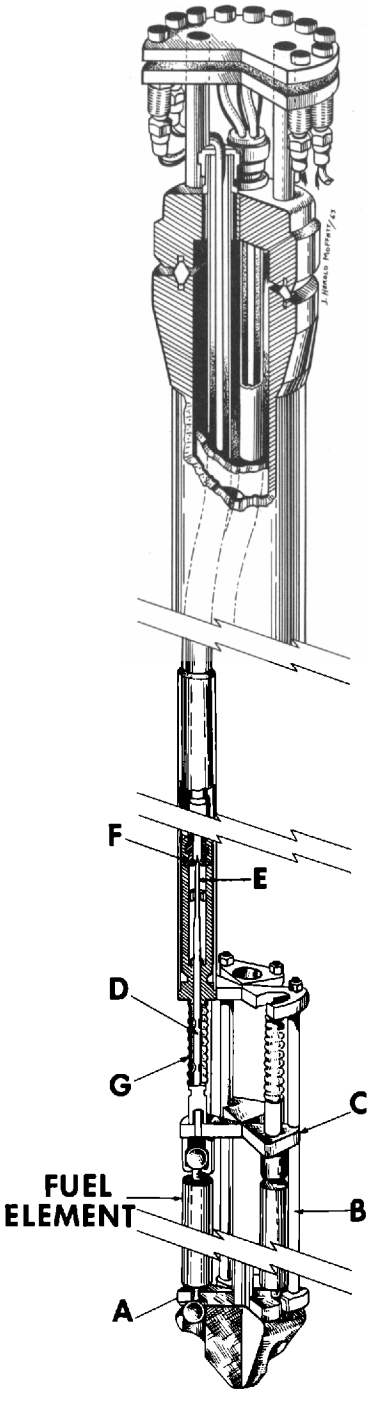
Each creep machine is equipped with a self-powered neutron detector (4) which gives an electrical signal proportional to the instantaneous neutron flux and can be recorded on the same chart as specimen temperatures.

Below 1 x 10^{13} n/cm^2 (E > 1 MeV) the Mark IV creep capsule can be used satisfactorily up to 500°C. The temperature can be maintained constant to $\pm$ 1°C when the reactor power is steady. During reactor trips from full power the temperature drops by about 5°C for three minutes. With careful adjustment of the controls it is possible to reduce the drop to 2°C. The temperature of the specimen is measured with contact thermocouples. They have been found to read within 3°C of the temperature measured with thermocouples that were embedded in a dummy specimen. The temperature difference along the specimen did not exceed 2°C. The bellows pressure is controlled to 0.1%. The strain sensitivity is 10 microinches. The maximum deviation of any point from the average creep curve did not exceed 60 microinches over a period of 2000 hours. The working range of the Mark IV machine is 3% strain, but can be increased if necessary. The specimen-yoke assembly can be telescoped out of the capsule for inspection in the caves, allowing replacement of test specimen and recalibration of bellows and pneumatic gauge.

The design of the Mark IV creep capsule is so well established that the manufacturers (Hawker Siddeley Canada Limited) are prepared to quote a fixed price and guarantee satisfactory performance in the reactor.

2. Fuel Element Length Measurement In Reactor - J.R. MacEwan
A. Harvey

A capsule comprising three gas gauges, similar in principle to those used in the creep machines, was used to measure length changes in reactor fuel elements during irradiation as a function of the power developed in the fuel (5). The capsule was in a water coolant at 1300 lb/in^2 and 225°C, which made a bellows seal necessary to isolate the gas gauge from the coolant. Figure 3 shows the general construction.



A, C - spacers
B - reference Zircaloy-2 rod
D - extension rod
E, F - pneumatic motion transducer
G - bellows seal

Figure 3. Schematic Diagram of Fuel Length-Change Measuring Capsule

Performance was generally satisfactory. A Zircaloy-2 "standard" used with one gauge, showed scatter of 0.005 in. in the readings, attributable to sticking of the needle parts. The other gauges were stable and the final readings were within 0.0035 in. of the residual length change measured by micrometer (0.057 in. in one instance).

3. Measurement of Pressure in Operating Fuel Elements - J.R. MacEwan
A. Harvey

To measure pressure changes in reactor fuel elements, AECL has taken the same approach as M.B. Reynolds at Vallecitos (6) in using a diaphragm on the end of the element as the sensor with an electrical contact to determine its equilibrium position. The diaphragm is balanced by an external gas supply whose pressure is measured. Such a device has measured pressures over 400 lb/in^2 in UO_2 fuel elements. A sketch of the switch to be used in an experiment with multiple fuel elements is shown in Figure 4.

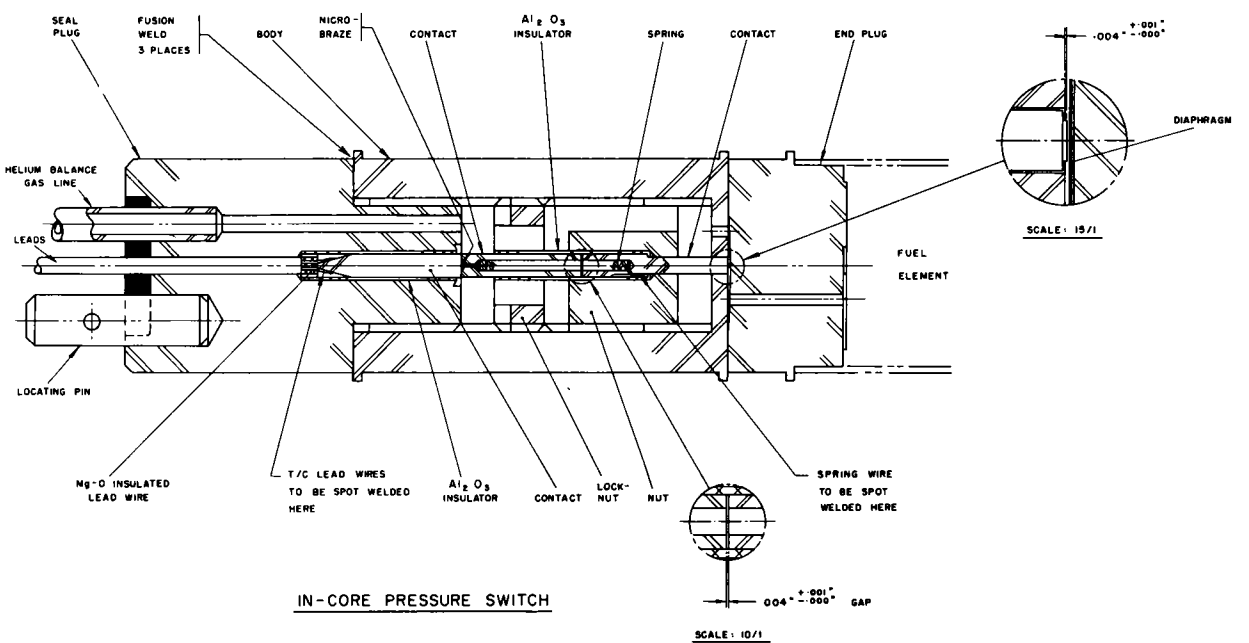


Figure 4. In-Core Pressure Switch

SECTION II - CAPSULE HEAT TRANSFER

1. Analysis - J.A. Cooke

Three-dimensional heat-flow analysis was carried out on the reactor creep capsules, but reliable calculations were difficult because of uncertainties in the "on-power" conditions inside the capsule.

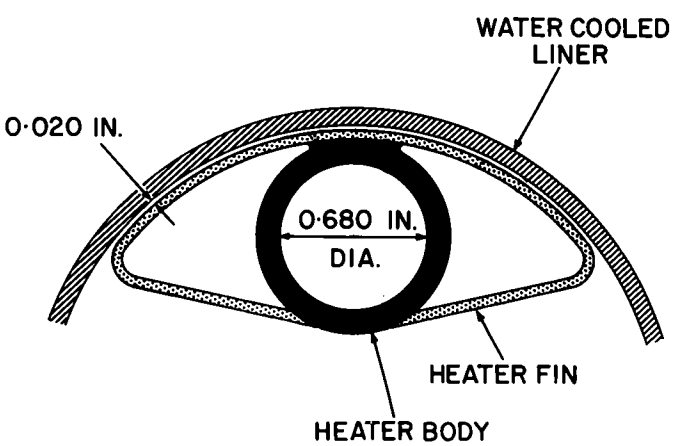
2. Analog and Mock-Up Techniques - A.J. Otter, V. Fidleris

An electrical analog was built to estimate temperature gradients in various parts of the creep capsule. This apparatus, together with the results from an in-reactor mock-up test, provided all the necessary data for design and performance evaluation.

An electrical analog technique was used to investigate perturbations of a surface temperature caused by a thermocouple on a fuel element. This method has given data which will be used to estimate the true surface temperature from the thermocouple indication in a fog-cooled channel. To check the analog results surface thermocouples will be mounted on an electrical heater with the same heat rating as the fuel elements and calibrated against thermocouples embedded in the heated surface.

3. Temperature Control - V. Fidleris

For accurate temperature control of non-fissile specimens, the capsules have been contained inside a tubular heater. So far the heaters have not exceeded 1.3 in. diameter and 3 feet in length. Some effort has been spent in designing light weight heaters with small longitudinal and circumferential temperature gradients. For use in very high gamma fluxes, heaters with cooling shoes have been designed (Figure 5).



To allow control of the heat loss rate through a narrow gap that separates the heater from the reactor water-cooled liner tube a mechanical device for adjusting this gap while in-pile has been designed, but has not yet been tested.

Figure 5. High Flux Heater Section

SECTION III - FLUX DETERMINATION AND CONTROL

1. Reactor Flux Mapping Techniques - J.W. Hilborn, A.W. Boyd, E. Kerr

In natural uranium reactors the neutron flux distributions have routinely been deduced from calorimetric measurements of the output of individual fuel elements. On occasions, supplementary and more detailed surveys have been made with activation monitors and ion chambers (7). Since in enriched reactors the power output depends on the fuel's exposure as well as the neutron flux the calorimetric method is not immediately applicable. Self-powered monitors are therefore to be used in NRU following its conversion to an enriched reactor in 1964. With the development of a self-powered neutron detector (4) it should be relatively easy to keep a continuous and detailed record of the reactor's thermal flux pattern for purposes of information, control and calibration of fuel element heat outputs. Cobalt wires are normally used for thermal flux measurements and nickel wire for the fast flux. In the interior of annular fuel rods, the nickel monitor flux has been found to be a good approximation of the total fast neutron flux (8).

2. Energy Deposition Measurements In-Reactor - A.W. Boyd

The chief purpose of this work has been to obtain the total energy deposition, and the fractions due to fast neutrons and gamma rays, in reactor irradiations of organic coolants. The emphasis has been on energy deposition in hydrocarbons, graphite and elements of low atomic weight, but some work has also been done in fast neutron rods with construction materials for capsules (9&10). Adiabatic calorimetry, ion chambers and cyclohexane dosimetry have been the tools. The design of the calorimeters is shown in Figure 6. The rate of temperature rise in the sample when its temperature equals that of the jacket, is measured and the energy deposition rate calculated by a computer from knowledge of the specific heat. These calorimeters have given results with a precision of 0.5 to 3% with energy deposition rates of the order of 0.1 W/g.

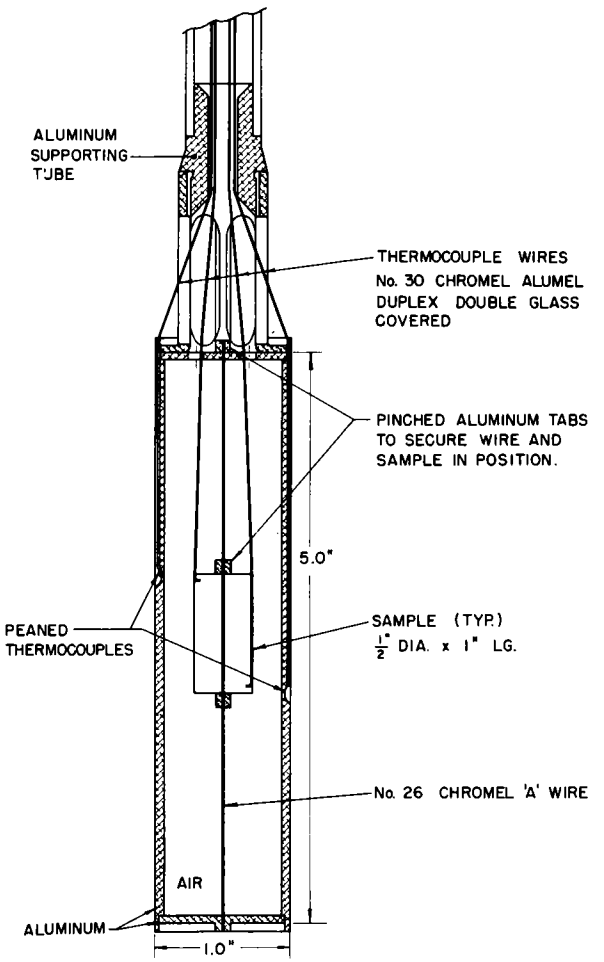


Figure 6. Calorimeter for Radiation Dosimetry

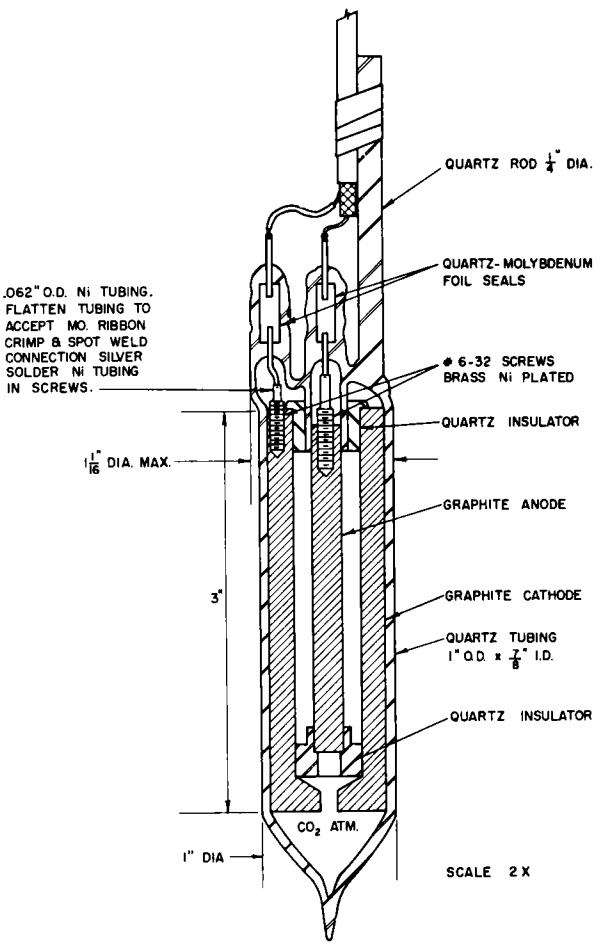


Figure 7. Ion Chamber Assembly

The contributions of thermal neutrons are calculated using the measured thermal flux and the relevant cross sections and neutron capture gamma and decay energies (11&12). The fast neutron and gamma ray contributions are obtained using the relative values for the fast neutron and gamma ray energy absorption coefficients.

To supplement the calorimeters and to replace them at very high and very low fluxes both ion chambers and cyclohexane dosimetry have been used and calibrated by comparison with the calorimeters and by the use of a Gammacell (cobalt-60).

The ion chamber's design is shown in Figure 7. Ordinary coaxial cable has been used for leads and appears to be satisfactory up to 0.5 W/g as long as the temperature is less than 150°C and the time at high dose rates is kept less than 15 minutes. Saturation curves are usually determined each time the chamber is used. Nine hundred volts have been found to be sufficient in any field so far measured.

Cyclohexane (C_6H_{12}) decomposes when irradiated to give hydrogen cyclohexene and bicyclohexyl. The G values of these products, i.e., the number of molecules formed per 100 eV of radiation absorbed, have been shown to be the same within 10% for most types of ionizing radiations (gamma rays, electrons, protons). The fast neutron energy absorbed by cyclohexane gives rise to fast protons. The total energy deposition in irradiated cyclohexane can therefore be calculated from the measured amount of hydrogen and the G value. The results have been found to agree with the calorimeter value for polyethylene which has the same empirical formula.

The gamma ray field in NRX has been measured with the reactor shut down and it was found that up to two to three days after shut down the field was still 1-2% of the field at full power. This has to be taken into account when extrapolating gamma fields from some low power to full power. When the power is suddenly increased by a factor of one hundred the rate of energy deposition due to gamma irradiation may be increased by no more than a factor of fifty.

3. Machine Codes for Estimating Flux Perturbations - J. M. Kennedy

There are several codes for neutron diffusion calculations on the G-20 computer, differing in the number of energy groups and in the type of geometry (13-16). All of these are for a stationary case only. There are several programs that do time-dependent two-dimensional calculations with one or two groups, allowing the properties of the fuel to vary with irradiation. At the moment the only descriptions of these programs are in reports of the fuelling schedules for NPD and CANDU reactors and not available for general distribution.

One special purpose time-dependent calculation is XENOSC, (17). This is a program for calculating xenon oscillations in a reactor, solving the two-group neutron diffusion equation and the space and time dependence of the concentration of xenon and iodine (two space dimensions).

SECTION IV - CAPSULE COMPONENTS AND FABRICATION

Thermocouples, Connectors, Seals - A.J. Otter, A. Harvey
Welding, Brazing - M.B. Watson
Heaters - V. Fidleris

In an attempt to reduce the failure rate on important installations, AECL has prepared a stricter specification for metal-clad thermocouples based on the new ASTM specification in preparation. Thermocouples bought to this specification will be evaluated and compared with "non-specification" thermocouples. Also, data is being collected on "in-reactor" thermocouple installations in order to determine reliability figures.

An experiment designed to investigate abnormal failure of aluminum-clad thermocouples in bismuth has been expanded to compare the effect of irradiation on the following combinations of metal-clad thermocouples:

Sheath	:	aluminum, 304 stainless steel
Insulation	:	alumina, magnesia
Conductors	:	type K, type J
Junctions	:	grounded, ungrounded.

A total of 84 thermocouples and three thermistors will be irradiated for one year in the NRU reactor commencing at the end of September, 1963, and the stability of calibration, reasons for failure and changes in insulation resistance checked.

Ungrounded thermocouples of various sizes will be irradiated in the NRU cobalt absorber rod (gamma ray source) to measure the heating effect due to gamma rays. The results will be compared with calculated values given in HW-65738. If a significant effect is observed the tests will eventually be repeated in an in-reactor position using a freezing point standard as a reference temperature.

Limited success has been achieved with thermocouples for measuring UO_2 fuel center temperatures. Some work was done with tantalum-clad thermocouples until it was realized that tantalum was not compatible with UO_2 under these conditions, even at temperatures as low as 1500°C. Molybdenum and tungsten have since been found to be compatible in short duration irradiations and molybdenum has been adopted for sheaths up to 2000°C. Tungsten/tungsten-26% rhenium thermocouples are used in alumina or beryllia insulators. Swaged or drawn construction is not favoured, since the tungsten wire is usually damaged. For higher temperatures a tungsten sheath may be used as one leg of the thermocouple in a coaxial arrangement (Figure 8) with no solid insulator in the hot zone. There is no irradiation experience with this as yet, but such a thermocouple has been calibrated to 2750°C in the laboratory.

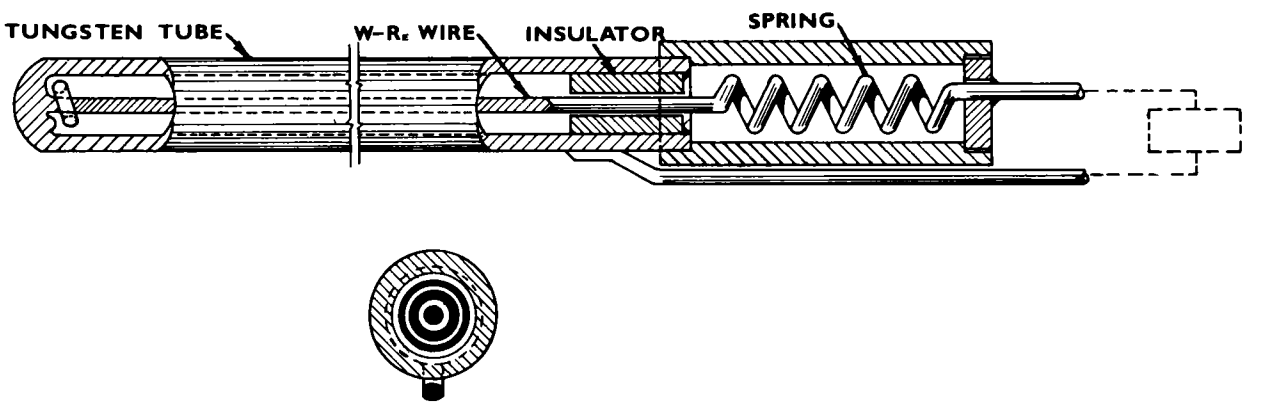


Figure 8. High Temperature W/W-Re Thermocouple

Some work has been done to evaluate the in-reactor performance of electrical leads. The most satisfactory ones have been packed with magnesia insulation. Drying at 150-200°C, and sealing while still warm with powdered nylon ("Rilsan") or silicone lacquer, have been found to be essential.

In the field of thermocouples, there are a number of outstanding problems:

a) Establishing drift characteristics.
b) Development of small "quick-disconnect" connectors.

c) Development of corrosion resistant joints for connecting stainless steel clad thermocouples to other materials.
d) Improving X-ray inspection for thermocouples less than 0.015 in. diameter.

Pressure seals for capsules can be conveniently made with compressed graphite. This is of particular advantage with multiple leads where threading solid plugs of lava, for instance, can be difficult. In a high pressure water environment knife-edge and miniature graphite seals have been used between dissimilar metals. However, the latter are not helium-tight and knife-edge seals suffer from differential thermal expansion. At the moment the seals are therefore made by brazing; Zr-Be for Zircaloy-stainless joints and Ni-Cr (Nicrobraz 50) for joining stainless steel components.

Chromel-wound, quartz core-tube heaters are used for short (up to six weeks) irradiation experiments. For higher temperatures and longer life, such as for creep capsules, special heaters have been developed. For temperatures below 600°C they have an aluminum core-tube and are of sprayed alumina and metal construction. Nichrome ribbon is used as the heater element. Above 600°C the core-tube is made from stainless steel. The spray construction results in a light, robust and durable heater. Typical thermal shock treatments given during testing are a core-tube heating rate of 1800°C/min and a cooling rate of 300°C/min. Several of these heaters have now performed satisfactorily in the reactor for periods up to one year. Some of them experienced frequent thermal cycling in the process.

ACKNOWLEDGEMENTS

The present author has had direct experience in only one of the fields reviewed here. He is therefore most grateful to his colleagues, whose names appear against the appropriate topic, for their generous help in preparing this summary.

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Irradiating Capsule for Use in JRR-2 Part I

S. Hirayama, T. Shoji

1. Introduction

JRR-2 is a 10 MW thermal research reactor of design similar to the CP-5 at the Argonne National Laboratory, i.e. heavy water moderated and cooled, with light water reflector, and fueled with MTR plate type fuel elements (90% enriched uranium). Provision is made for capsule irradiation in nine 4" vertical thimbles (1 in core center, 8 in the peripheral region), as well as in isotope train and in 1" and 2" pneumatic tubes. The horizontal beam tubes are mainly used for loop and beam experiments. Operation has so far been in cycles of 130 hours at 5 MW, but upon entering normal operation the regular cycle will be 10 days at 10 MW. The following is a description of some examples of irradiating capsule development work, in preparation for regular operation of the reactor.

2. Capsules for Pneumatic Tube

2.1 Structure of Capsule

(Y. Suzuki, H. Tomioka)

Capsule material is polyethylene made by the medium pressure process (brand name Sholex 5008). This is not a homopolymer but a co-polymer with butene-1. Capsule design is shown in Fig. 1 (for 2" pneumatic tube). A 35 mm diameter x 160 mm body is closed at both ends with a finned cap 50 mm diameter.

2.2 Radiation Damage and Shock Resisting Properties

The JRR-2 pneumatic tubes are designed for an operating speed of 10 m/sec, which imparts mechanical shock to the capsule of non-negligible order. On the other hand, high polymer products like polyethylene are liable to harden by irradiation and become less resistant to shock. The period of exposure to

radiation must therefore be restricted. Destructive tests on actual capsules were undertaken in the JRR-2 pneumatic tube exposed to a thermal neutron flux of 8×10^{13} n/cm^2 sec. Operating speed was set at 10 m/sec, and the force of shock varied by enclosing loads of different weight in the capsule to produce up to a maximum force of 10^3 g·m/sec. With unirradiated capsules tests on 20 capsules resulted only in some deformation of 1 or 2 fins at maximum force. Results with increasing irradiation are given in Fig. 2.

At 5×10^{16} nvt mechanical strength was not appreciably lower than before irradiation, though the original milk-white color changed to light brown. At 10^{17} nvt some of the fins broke by shock. At 10^{19} nvt the color turned to dark brown, the material lost its elasticity, and the fins broke and scattered under a force of 1.5×10^3 g·m/sec, while the capsule body also was cracked.

Based on the above results standard irradiating conditions for the pneumatic tube were established to be 10 min, 5×10^{16} nvt (thermal), and shock force to be 10^2 g·m/sec (maximum 10^3 g·m/sec).

2.3 Heat Resisting Properties

Sholex-5008 has a softening temperature of 124°C, so that the heat produced in capsule (the amount of sample) must be restricted to prevent deformation in use. It has been proven however that the cooling action of the air used for operation (600 litres/min) prevents the outer surface of the capsule from reaching softening temperature even when the inner surface is quite hot. Notwithstanding this factor of safety, heat produced within capsule has been limited to 15 w to provide for unforeseen stoppage of the operating air. In estimating heat produced, due account must be taken of the effect of alpha radiation:

experience with a 4 gram crystal of LiF irradiated to 6×10^{15} nvt showed that alpha radiation will produce a heat of 200 w, which actually melted the inner capsule of polyethylene. Care must also be taken to consider hot spots caused in certain parts of the sample or capsule due to conditions of packing.

3. Leak Resistant Properties of Aluminum Capsules for Vertical Thimble

3.1 Capsule Design

(M. Sato)

Capsules made of aluminum have to be used in the vertical thimbles, because of the high temperature and dose. These capsules must be designed for ease in sealing and opening. Moreover, some samples decompose and release obnoxious gases, while fuel material will produce fission product gas, and the capsule must be airtight to contain these gases. In some experiments the thimble is flooded with heavy water for heat removal, and the capsule in such cases must be water-tight to prevent the sample from contact with the water. The capsule design is shown in Fig. 3. Sealing is by manually operated roller which applies on the cap a force F in the direction shown to close and seal the capsule. Air-tightness of the sealing was compared for the cases of: (1) no packing between cap and body, (2) seal bonded with araldite, (3) packed with aluminum sheet 0.07 mm thick, (4) with neoprene packing 1 mm thick. Air leakage was tested with glycol and helium, and water leakage also inspected. These tests were carried out under a pressure difference of 1 kg/cm^2 between capsule interior and exterior. Tightness tests were also conducted on capsules irradiated to 1.3×10^{16} nvt (fast), and under high temperature while being heated in an electric furnace.

3.2 Leak Tightness Tests at Room Temperature before Irradiation

Results are given in Table 1. Sealing without packing resulted almost invariably in leakage; the leaking part is as shown in Fig. 4. About 70% of these leaks are less than 1×10^{-2} A.cm^3/sec. This sealing without packing is thus unsatisfactory for air-tightness, but as shown in Fig. 5, infiltration of water is of the order of $10^{-3} - 10^{-4}$ times that of gas, so that it may be considered usable within a fairly large range of conditions as water tight container.

Aluminum sheet packing proved itself to be ineffective as sealing material. Neoprene on the other hand was found to be very efficacious, and no leakage could be detected by glycol test capable of discerning down to 10^{-6} A.cm^3/sec. Araldite bonding was tested after hardening of the bond, and found to be extremely good: Leakage rate is below 10^{-8} A.cm^3/sec, the limit of detection by helium.

3.3 Resistance of Araldite Bonded Capsules against Heat, Radiation and Pressure

Capsules sealed after coating the seal surface with araldite leaked when tested with helium after heating for 55 hours at 300°C, due to brittleness acquired by the araldite. Heating only to 250°C, however, induced no change to araldite. Tests after irradiation to 1.3×10^{16} nvt (fast) (temperature 100°C) revealed the air-tightness to have been unimpaired, proving that this material can be safely used under radiation and temperature of this order.

The thin walled design renders this capsule inherently weak to internal pressure: Before the capsule itself fails, air-tightness is broken at the seal due to the deformation of cap. Bonding with araldite proved to contribute a strengthening effect on

the capsule, and leakage was small even after deformation of the capsule. The minimum allowable pressures based on a permissible leak rate of 10^{-7} A.cm^3/sec were about 12 kg/cm^2 (at 25°C) and 7 kg/cm^2 (at 160°C). This result only partly incorporates the effect of radiation damage, so that in actual use a safety factor will have to be considered. As water tight capsule to be used below 100°C, however, this form of sealing can be considered a fairly safe container for withstanding internal pressure.

4. Capsules containing Light Water

(Y. Miyasaka)

The central vertical thimble (VT-1) of JRR-2 has a maximum thermal neutron flux of 2 x 10^{14} n/cm^2/sec at reactor output of 10 MW. While for ordinary irradiation experiments this flux is quite ample, for special experiments it can be locally boosted to still higher intensity by sealing light water within the capsule together with specimen. Such an expedient is possible from the fact that the thermal neutron absorption of light water is about 200 times that of heavy water, and its moderating power also greater by 1 order of magnitude. Fig. 6 shows the neutron flux distribution along the thimble axis with an aluminum capsule 90 mm diameter x 100 mm placed in the VT-1 flooded with heavy water, for cases where the capsule contains heavy water, light water and void. Thus, with heavy water reactors, the use of capsules filled with light water along with sample can permit thermal neutron flux to be raised by a factor of 2 without affecting the reactivity of the reactor to any appreciable extent (Japanese patent applied for).

Table 1 Results of Leak Rate Tests

Sealing material	Test temperature	Test method	Number of specimens	Results
Seal without packing	Room temperature	Glycol	39	28 specimens above 1×10^{-2} A·cm^3/sec 11 " below " "
Aluminum sheet packing	"	"	10	8 specimens above " " 2 " below " "
Neopreme packing	"	"	5	Leak undetectable
Araldite packing	"	Helium	30	" "

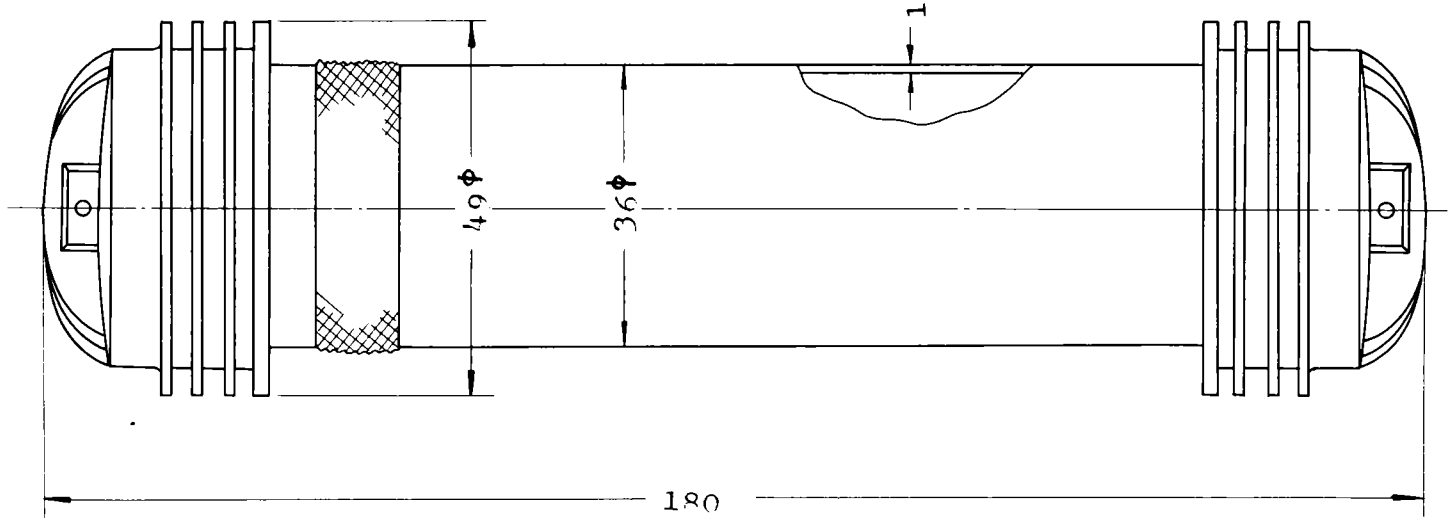


Fig. 1 Polyethelene capsule for 2" pneumatic tube (JRR-2)

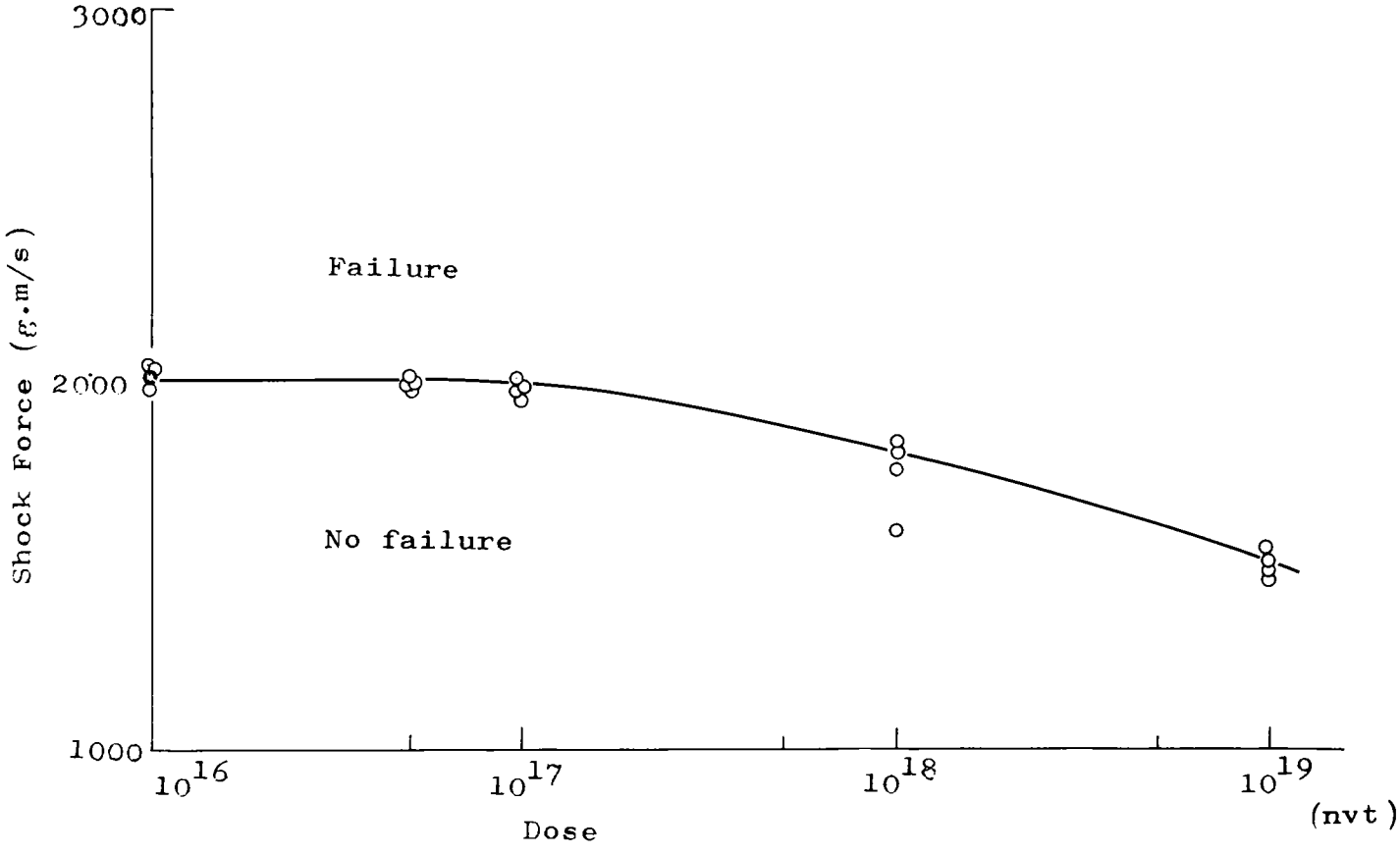


Fig. 2 Limits of post-irradiation shock failure to polyethelene capsules. (Operating speed 10 m/s)

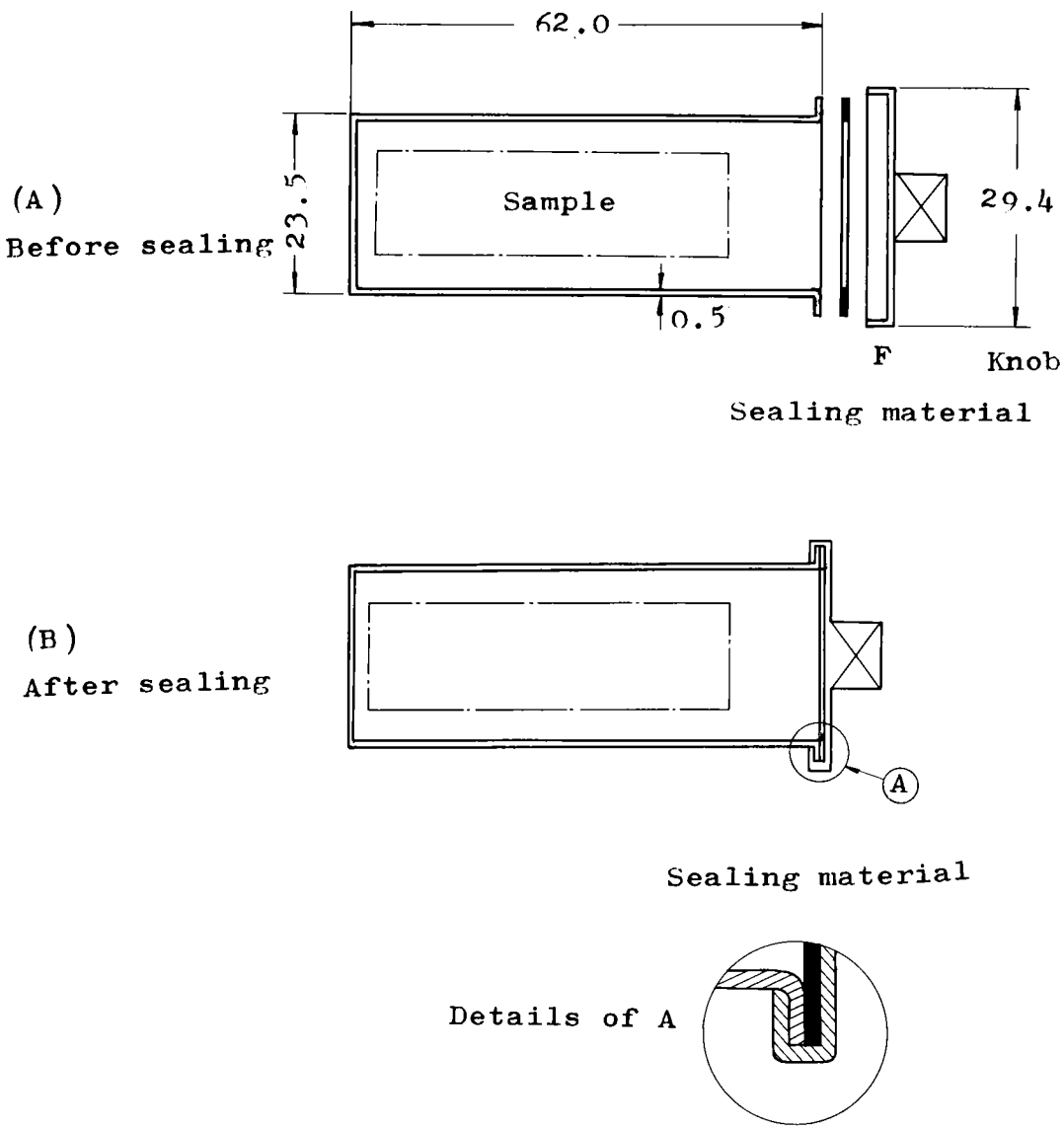


Fig. 3 Aluminum roller-sealed capsule

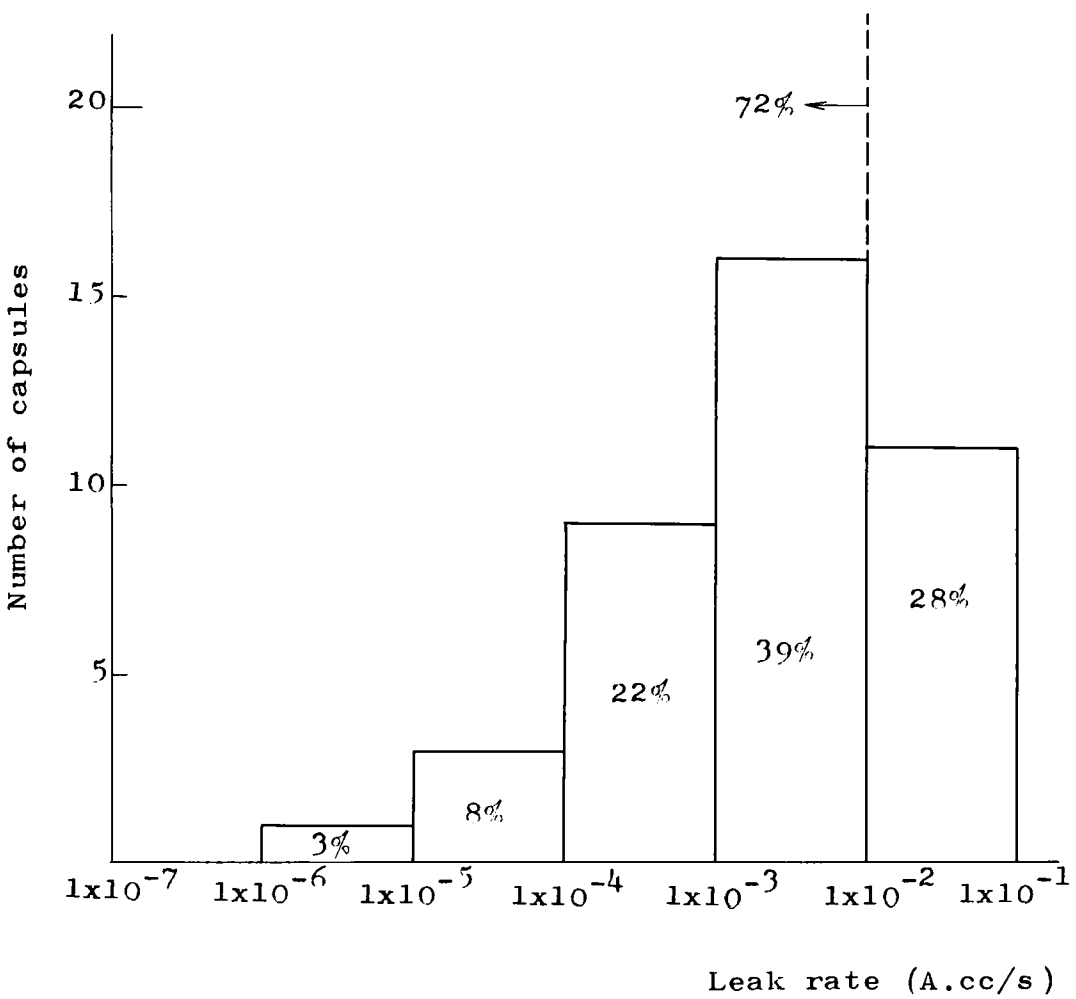


Fig. 4 Histogram of leak rate of seal without packing

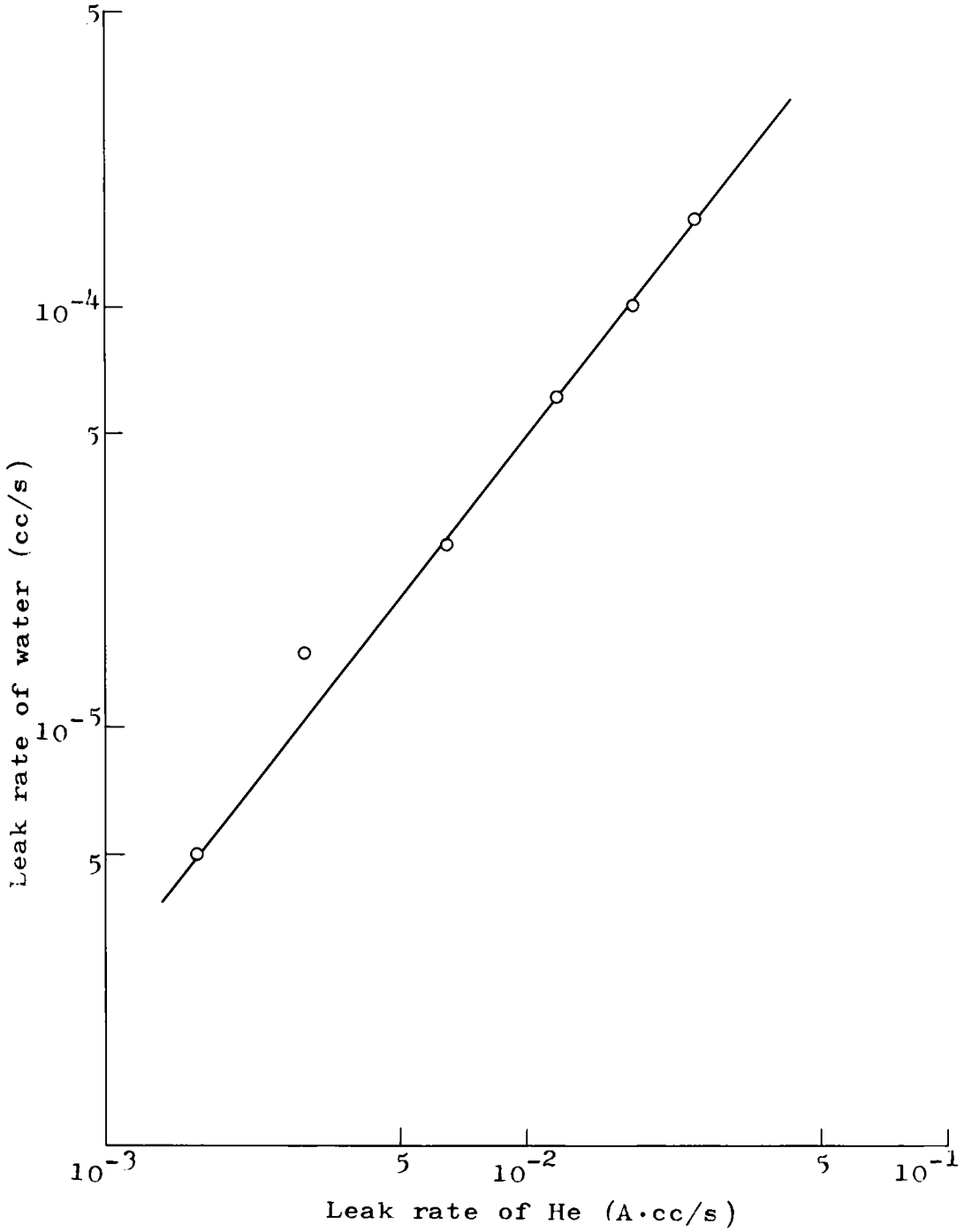


Fig. 5 Comparison of leak rates between water and Gas (He)

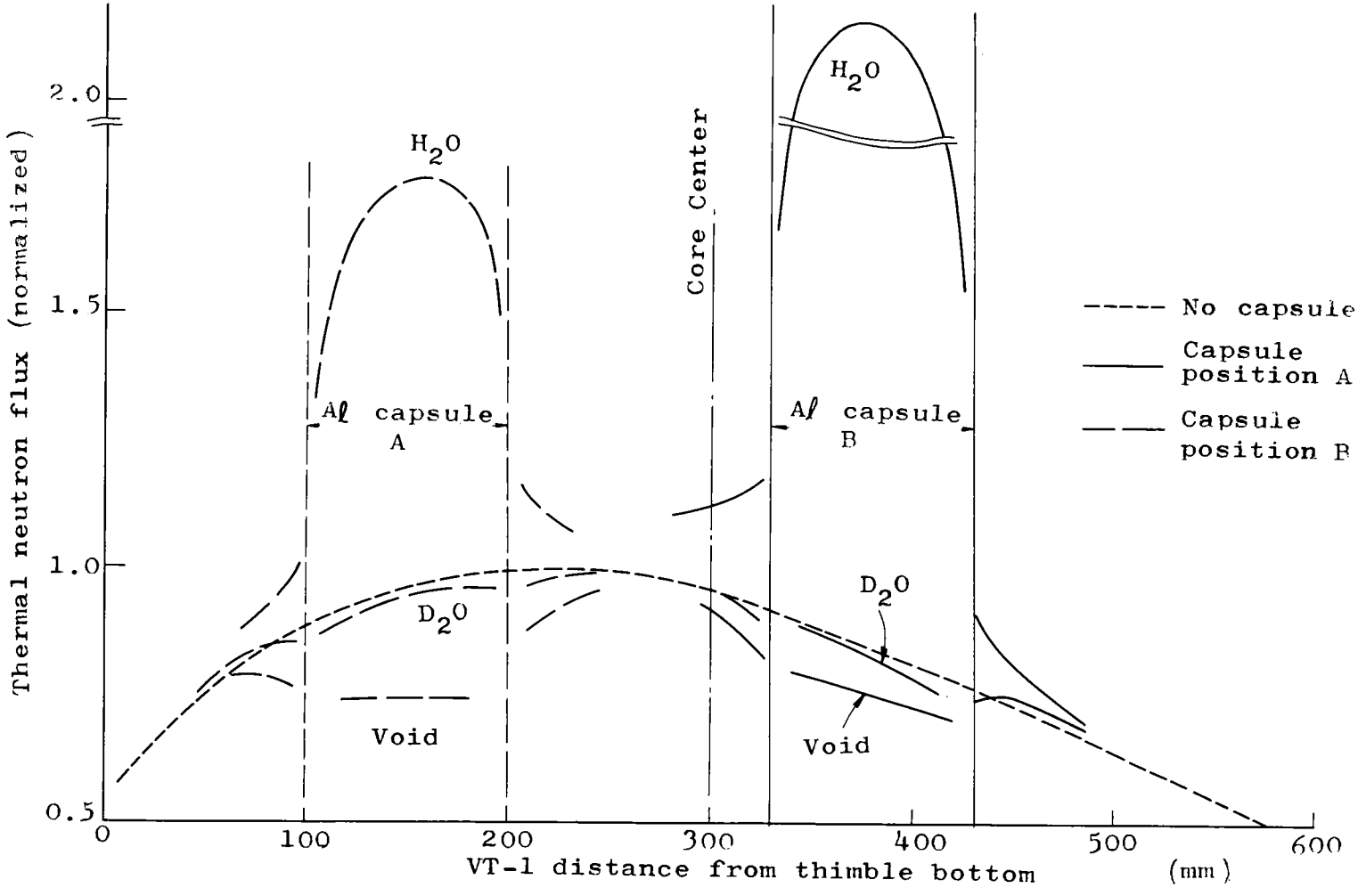


Fig. 6 Change of thermal neutron flux with position and contents of capsule in the vertical thimble (VT-1) flooded with heavy water

Irradiation Capsule for Use in JRR-2 Part II

R. Ueda, T. Fujimura, K. Taketani, S. Ouchi, M. Ichikawa

Introduction

For the time being, the Japan Atomic Energy Research Institute has only the JRR-2 available for capsule irradiation tests.

Preliminary tests for irradiation of fuel and reactor materials are now in progress with this reactor. Work is now proceeding on the design and manifacture of capsules, accumulation of experience in irradiation techniques and in handling irradiated fuel in hot laboratory in preparation of full scale capsule irradiation tests.

Progress on such preliminary test is described.

1. Controlled Temperature Capsule Test with Central Vertical Thimble (VT-1) of JRR-2

(T. Fujimura, C. Nakazaki, and K. Yamaya)

The authors have planned irradiation experiments at elevated temperature on steel and other reactor structural materials, and are trially constructing the required equipment upon determining the type to be adopted of temperature controlled capsule suitable for use in the VT-1 thimble of JRR-2.

Capsule dimensions of 90 mm diameter x 380 mm long were decided in consideration of the fast neutron distribution in the thimble to be used. Accuracy of controlled temperature in the specimens is to be $\pm$ 5°C within a range of 200-500°C. Heating will be by the gamma heat only, and no electric heater will be used. The thimble will be filled with heavy water during irradiation test, and heavy water temperature at capsule surface will not exceed 80°C in our design. An example of capsule design is given in Fig. 8.

Figure 9 is a schematic representation of the system, together with temperature control circuit block diagram. Temperature control is done by adjusting the mixture ratio of helium and nitrogen gases, as well as by regulating the rate of forced circulation of the heavy water filling the thimble. The helium-nitrogen mixture is stagnated but not circulated through the capsule except in the case of temperature variation, and the mixture at pressure is retained there and released at given moments in given amounts. Capsule temperature is measured by the chromel-alumel thermocouple, signals from which are fed into an electromagnetic valve. In the present design a time lag of about 30 seconds occurs between valve movement and return of capsule temperature to reference value. Temperature variation is exponential, with a time constant of about 500 seconds. With the gas mixture ratio varied from 0 to 100% for interval of valve movement, simple on-off control system will not permit temperature control of the accuracy specified. Therefore, the authors have devised a control circuit of the following descreption.

Variations in capsule temperature detected by the thermocouple are converted into error signals by means of a comparator system determining the deviation from reference temperature. The error signals are amplified and fed into the monopolarity and judging circuits. Signals passing the monopolarity circuit are rectified into one polarity before entering the analog-to-time convertor through the gate circuit. The convertor programs time intervals according to the difference values given by the error signals, and governs the distributor opening and shutting the valves according to the polarity determined by the judging circuit and at intervals programmed by the convertor. While the capsule temperature is changing in response to a signal, an inhibiting

signal generator prevents the conflicting new signals from entering the analog-to-time convertor.

Heavy water filling the thimble is regulated for temperature by the thermo-couple controller actuated in order to control the temperature by a circulation pump; the heat exchanger which is in plug with the pump, is of a heat removable capacity of 8 kw. The system is backed up by a cooler of the secondary coolant circuit.

Mock-up tests on a system such as described above were completed during 1963, and will be followed by construction in 1964 of irradiation equipment.

2. Capsule Irradiation Experiments with Hydrorabbit in JRR-2

(K. Taketani, M. Ichikawa, A. Kikuchi)

2.1 Equipment

A light water hydrorabbit system for inserting samples into reactor was installed in the HT-1 horizontal beam tube of JRR-2. The device, as shown in Fig. 10 consists of three main parts: main irradiation tube, lead shield and specimen transfer cask. The capsule holder shown in Fig. 11 is sent through the inner tube into the reactor. The capsule placed in the holder is cooled by the circulation of 4 ton/min of water. The design of the bend in the transfer for inserting the holder in the reactor tube plays a vital role in this facility: Several trial designs were made of this part for mock-up testing, and several thousand tests were conducted on a number of selected designs to determine the best model. The holder shown in Fig. 11 is designed to contain six NaK filled capsules. Study is being made on the possibility of modifying this holder for instrumentation by reducing the number of capsules and redesigning the specimen transfer cask.

2.2 Capsule

Capsule form is of orthodox design, and to test its safety in the JRR-2, autoclave tests and preirradiation X ray examination were conducted, as well as safety experiments with intentionally punctured capsule to simulate cases where such puncture would put NaK directly into contact with the surrounding cooling water. A hole 2 mm diameter was made in a 1 mm thick stainless steel capsule which was then immersed in water maintained at a specific temperature, and the effect observed by cine-camera. Hydrogen produced was collected and the rate of production measured in time. NaOH and KOH remaining in the water after completion of the experiment was titrated with 0,1 N HCl to obtain the total reaction. While during actual irradiation the coolant is circulated, the present test was conducted with still water at temperatures maintained at room temperature 50°C and 70°C.

Fig. 12 shows the variation in time of the hydrogen gas produced by the reaction between NaK and water. NaOH was neutralized and titrated with HCl 20 minutes after initiation of reaction.

The above results indicate that failure of the capsule wall during irradiation would not cause explosive reaction between NaK and water.

2.3 Conclusion

The device described above was installed in JRR-2 and capsule irradiation experiments have been initiated. After irradiation the specimen transfer cask is removed from reactor and sent to the Hot Laboratory for with drawal of the specimen for post-irradiation examination.

3. Capsule Irradiation for Semi-Homogeneous Fuel

(S. Ouchi, T. Ohmichi)

3.1 Purpose of Experiment

Capsule irradiations were conducted on semi-homogeneous fuel, composed of UO_2 and UC_2 particles dispersed in natural graphite matrix. There were not test results on such fuel base on natural graphite matrix.

To familiarize ourselves with such capsule irradiation techniques, experiments were first tried with natural UO_2 dispersed in natural graphite; the experiments will be extended to samples containing enriched UO_2 and UC_2 particles.

3.2 Design of Capsules

Double stainless steel walled and uninstrumented capsules as shown in Fig. 13 are used, inside which are placed graphite sheathes containing four pellets of semi-homogeneous fuel such as described above in each capsule.

The capsules are equipped with 0.1 mm diameter cobalt monitoring wire and iron-cobalt powder compact for dosimetry to determine thermal and first neutron flux. Temperature inside the capsules are similarly monitored by melt wires of aluminum, zinc, lead and tin. Spaces in capsule are filled with atmospheric helium gas. The pellets dimensions are 9.30 mm diameter by 9.84 mm height.

Flux perturbation at the center of fuel specimen has been calculated to be about 0.953. Calculation on activity, center temperature are made.

Double walled instrumented capsules as shown in Fig. 14 are used also. In this capsule, temperature is measured in three points, using stainless sheathed thermocouples. There are three helium gas space to make specimen temperature high, and argon can

be used to make specimen temperature higher.

3.3 Irradiation in JRR-2 and Post-Irradiation Examination

The capsule described in Fig. 13 was placed in a holder and irradiated in the vertical thimble VT-5 flooded with heavy water.

We have just finished the first preliminary irradiation, and the capsule has been sent from JRR-2 to the Hot Laboratory, where they are preparing to measure the amount of fission gas release. This will be followed by withdrawal of the pellets to determine dimensional change and fission gas release caused by isothermal annealing.

Construction of a fission product release loop is now in progress for installation in JRR-3 for other studies in semi-homogeneous fuel, independent of capsule irradiation experiments.

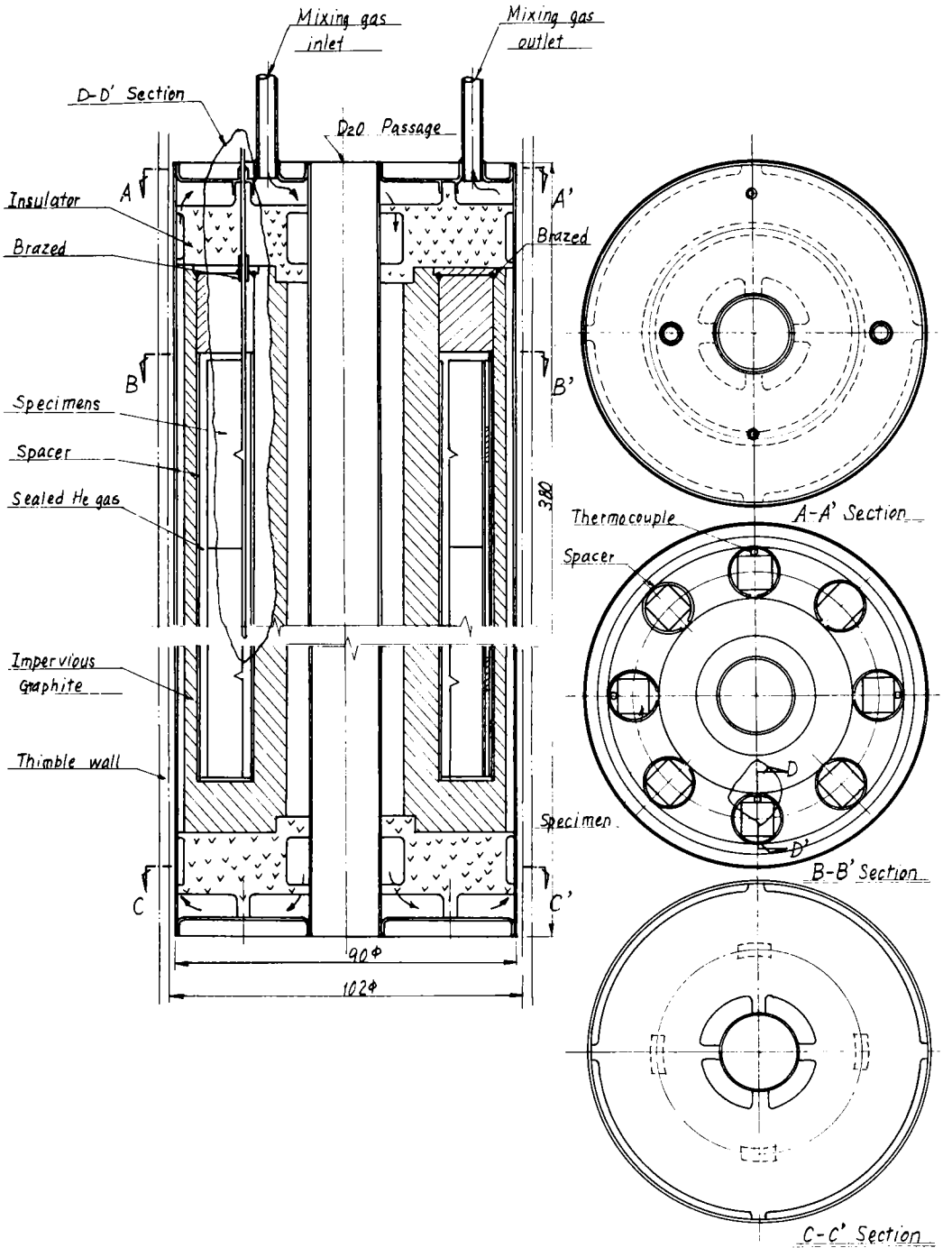


Fig. 8 Temperature control capsule in JRR-2. VT-1

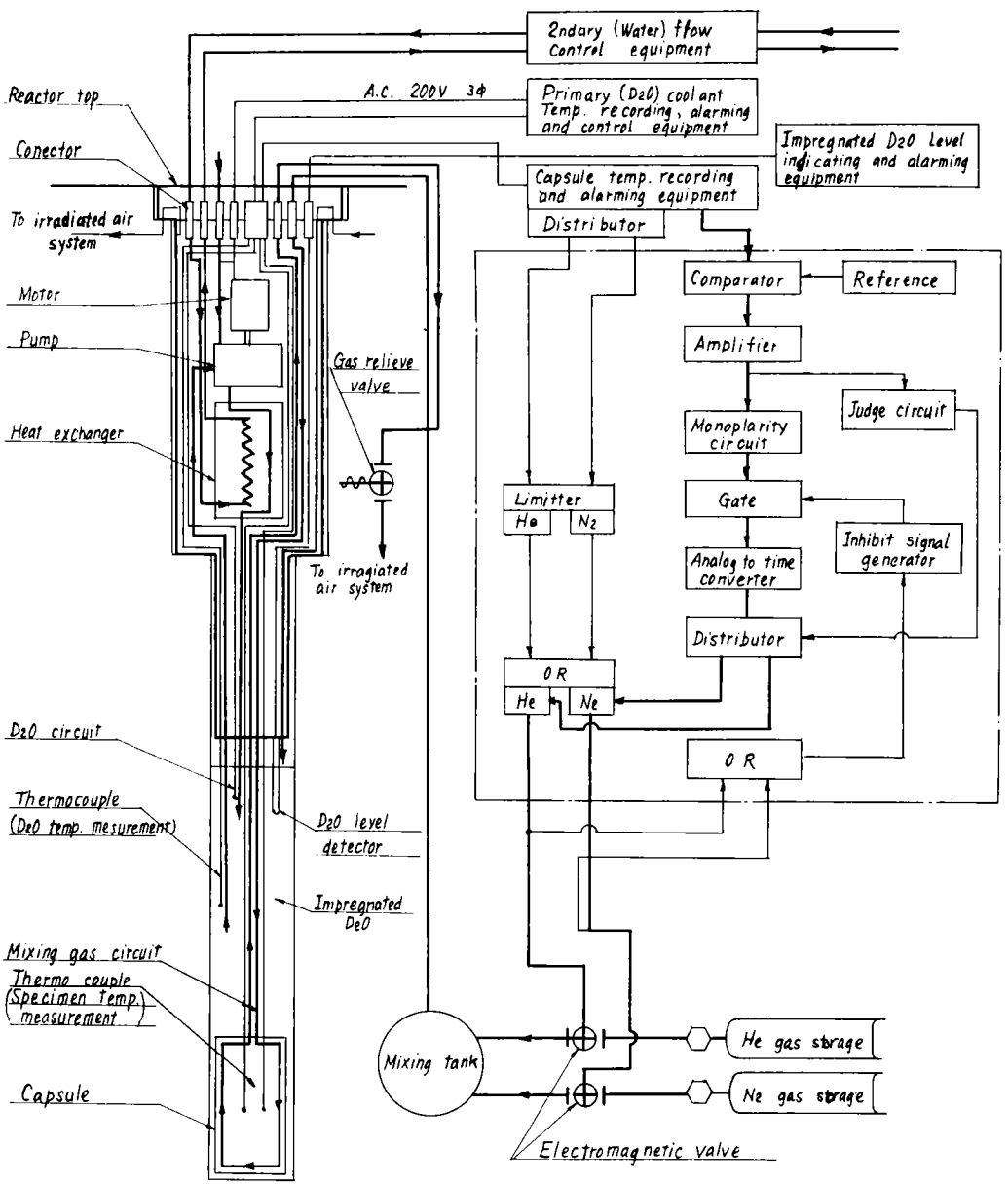


Fig. 9 Block diagram of test rig in JRR-2. VT-1

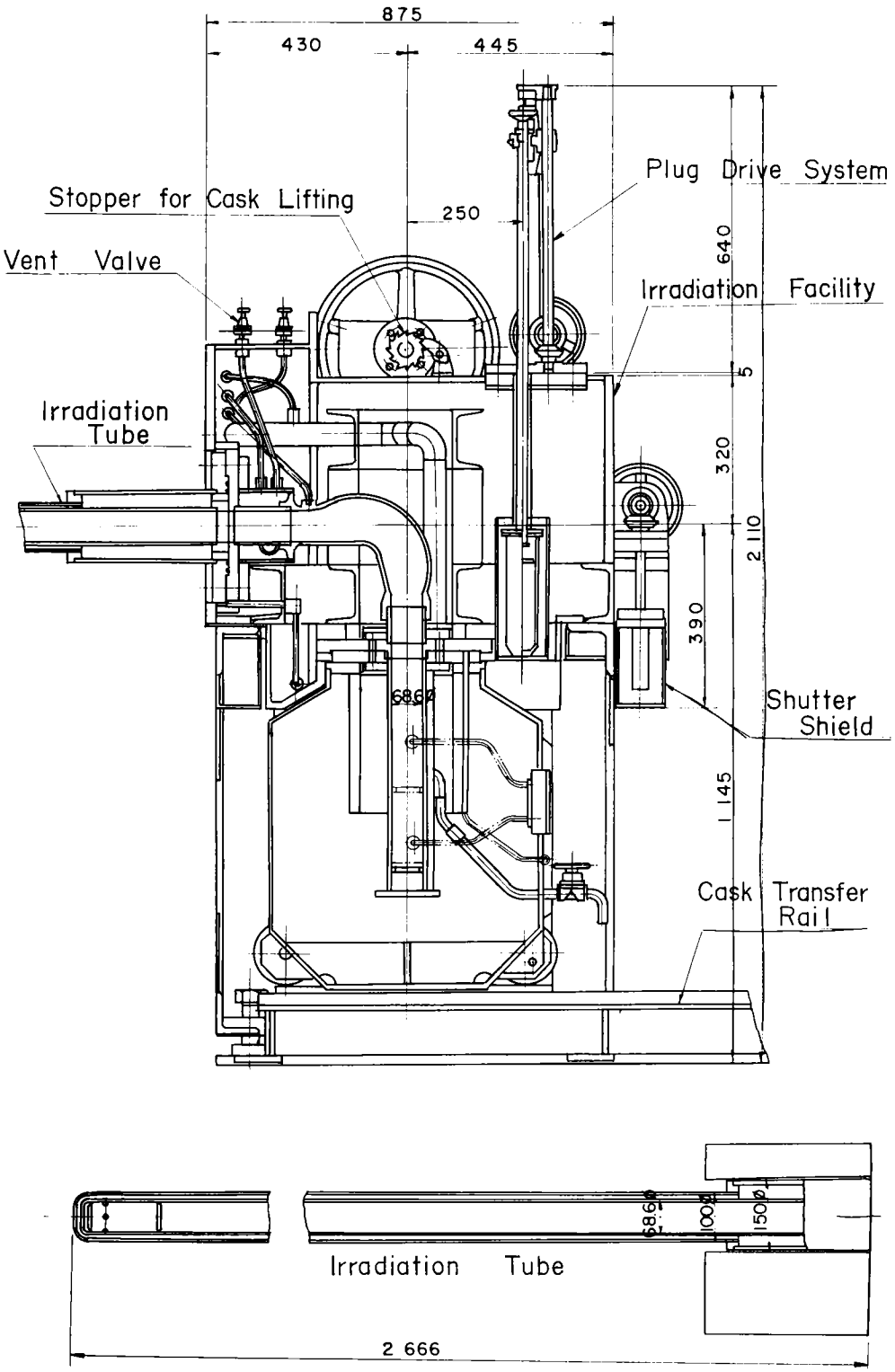
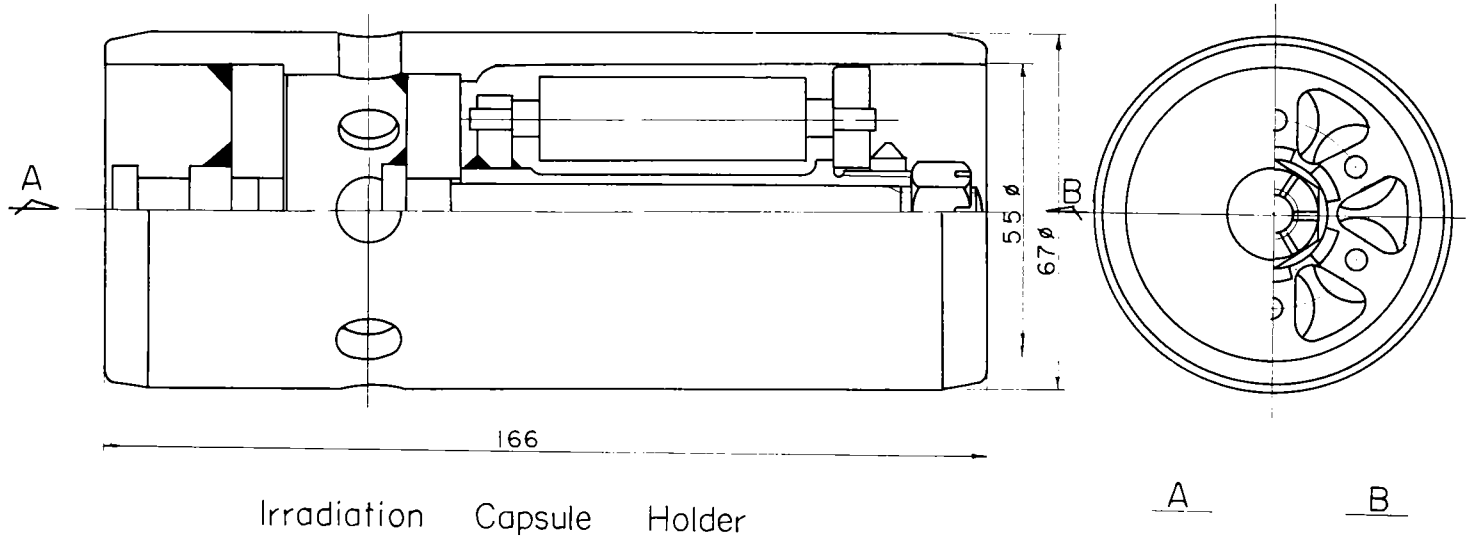
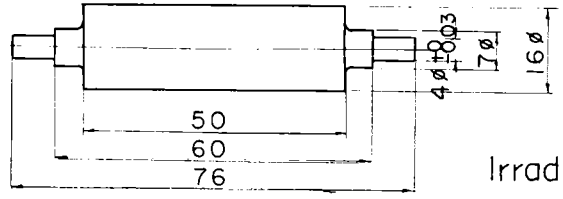


Fig. 10 Hydrorabbit



Irradiation Capsule Holder



Irradiation Capsule

Fig. 11 Capsule Holder and Capsule

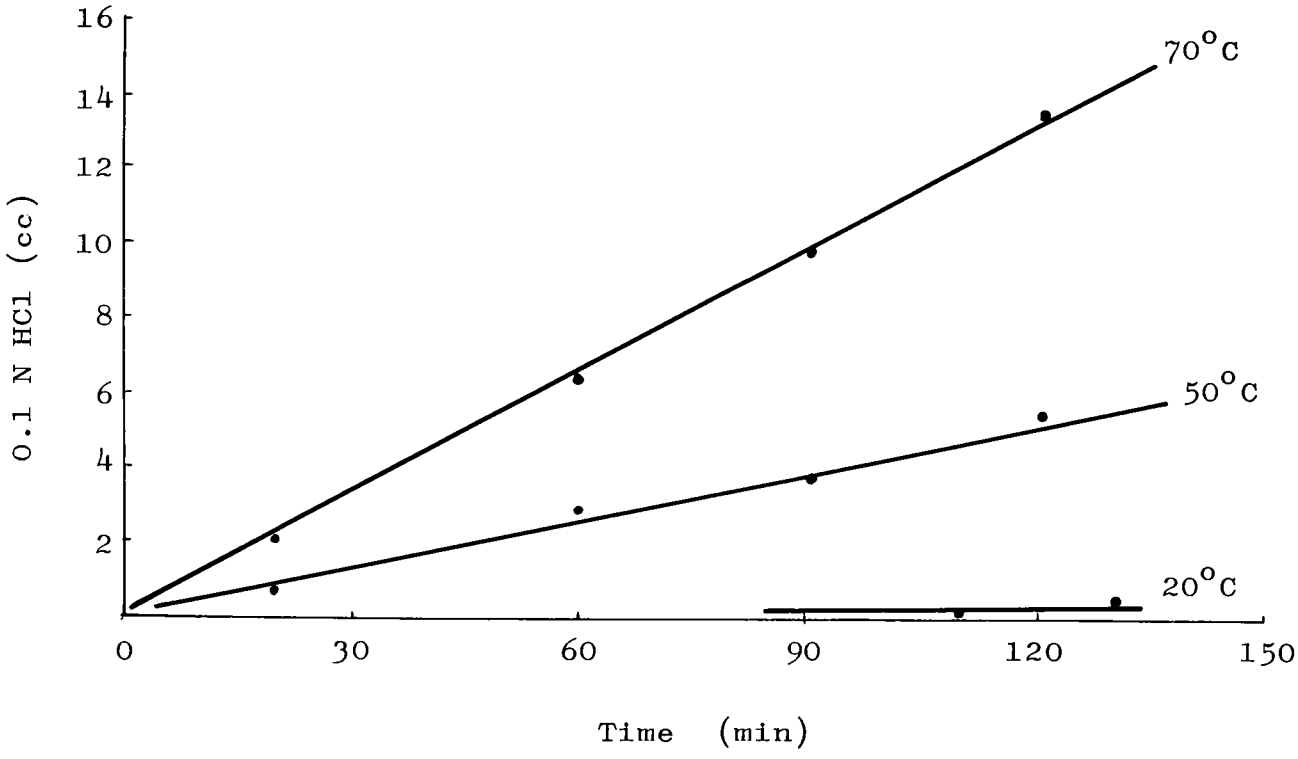
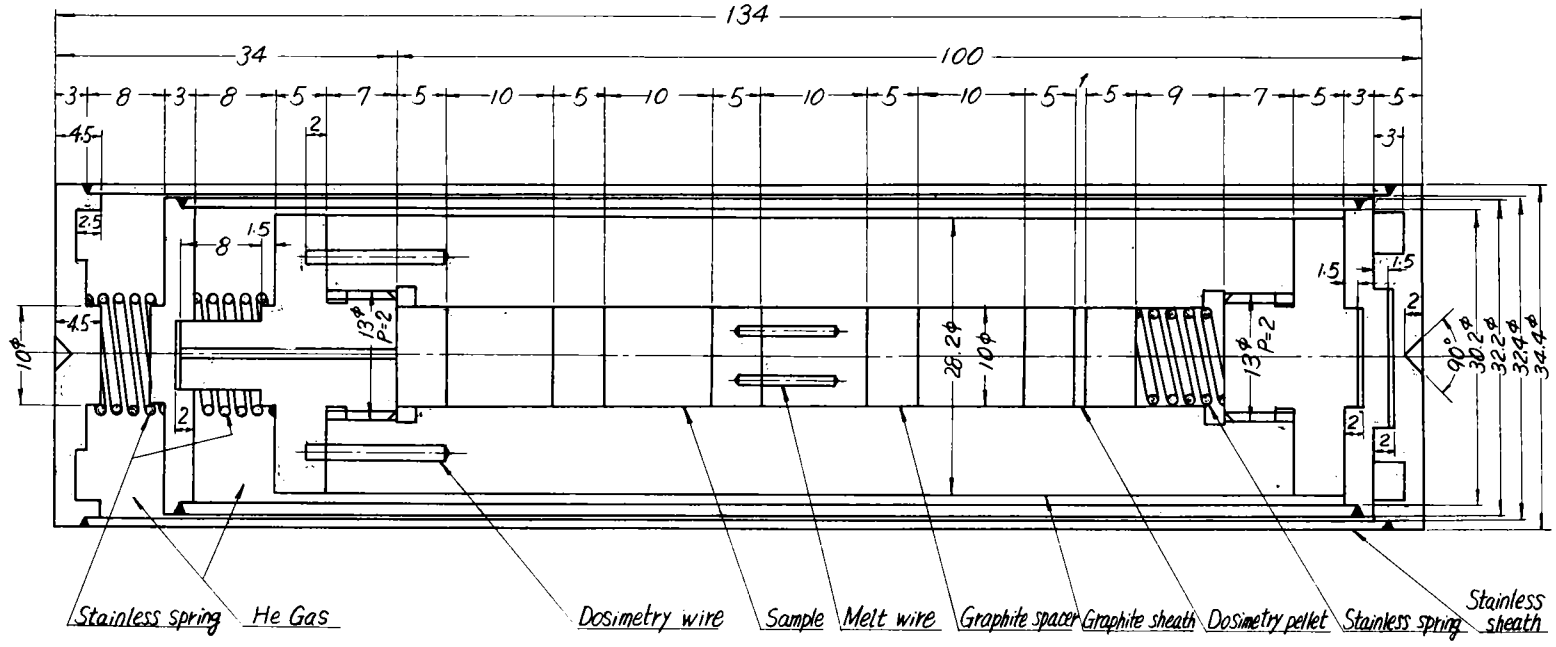


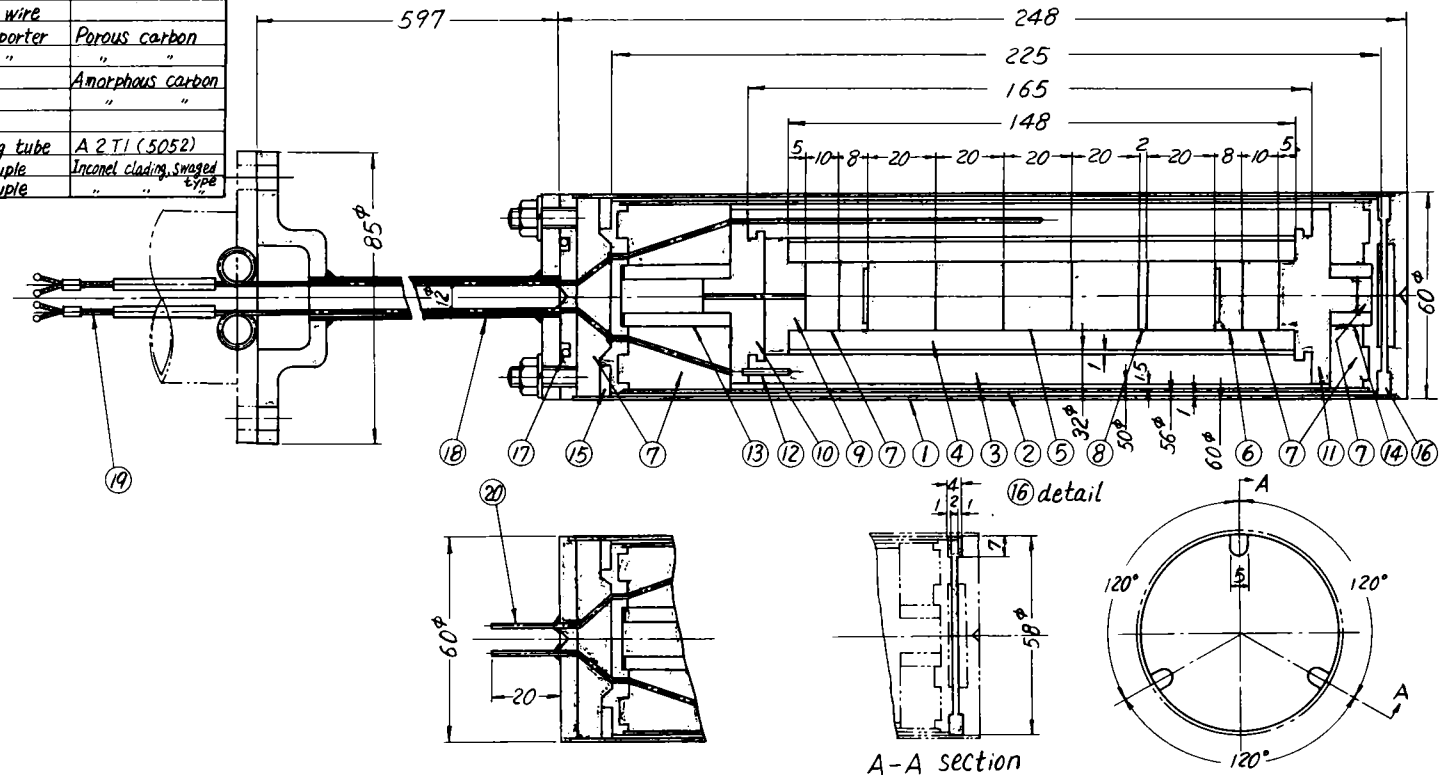
Fig. 12 Reaction of NaK and Water in a Failured Capsule

Fig. 13 Capsel for Irradiating Semi-Homogeneous Fuel



No.	Name	Materials
1	Capsule outer sheath	Stainless steel (304L)
2	" inner sheath	" "
3	Graphite outer sheath	Graphite
4	" inner sheath	Graphite
5	Test specimen	Semi-homogeneous fuel
6	Spacer	Porous carbon
7	Thermal insulator	Graphite fiber
8	Dosimetry pellet	
9	Plug	Graphite
10	Upper cap	"
11	Lower cap	"
12	Dosimetry wire	
13	Tubler suporter	Porous carbon
14	" "	" "
15	Spacer	Amorphous carbon
16	Spacer	" "
17	"O" ring	
18	Connecting tube	A 2 T I (5052)
19	Thermocouple	Inconel clading swaged type
20	Thermocouple	" " "

Fig. 14 Instrumented Capsule for Irradiating Semi-Homogeneous Fuel



United Nuclear Corporation
Contributions
to the AEC Meeting
on Problems in
Irradiation Capsule Experiments

Author: John J. Cihi

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White Plains, New York

I. INTRODUCTION

For the past eight years United Nuclear Corporation has been designing and irradiating capsules containing fueled and nonfueled samples. During this time, a basic capsule design has evolved that has been improved and refined with each irradiation program. The design has been flexible enough to accommodate most irradiation samples, both fueled and unfueled.

Fueled samples irradiated in our basic capsule have covered the following ranges:

Heat generation rates – 10 to 20 kw/ft.

Fuel clad surface temperature – 800 to 1600°F.

Fuel clad outside diameter – 0.125 to 0.500 in.

A cross section of a typical irradiation capsule is shown in Fig. 1. The capsule consists of an inner and outer shell, supplying the double containment for the liquid metal in the inner shell. The specimens are supported in the inner shell with the thermocouples and electric heater.

The inner shell is filled with liquid metal as the heat transfer medium. Radial fins are machined on the outside of the inner shell and the fins are so designed (thickness and spacing) to result in the required thermal resistance to meet the design temperature at the design heat generation rate.

The following sections describe the capsule in more detail. The numbering system used corresponds to the item numbers in the "List of Proposed Discussion Topics" suggested in the USAEC invitation to this meeting.

II. CAPSULE HEAT TRANSFER

A. ANALYSIS TECHNIQUES AND PROBLEMS

The heat transfer calculations for the capsules are performed with the following assumptions.

1. Heat flow is radial in the sodium.
2. A correction is applied to the thermal resistance of the inner shell, outer shell, and water film to account for longitudinal heat flow.
3. Thermal conductivity for each region is based upon a bulk average temperature.
4. The gamma heating in each region is added to the total heat flux.
5. A gap conductance between the fin tips and outer shell of 5000 Btu/hr-°F-ft^2 is used (based on capsule operating data).
6. Radiant heat transfer from the inner to outer shell is neglected.

In order to aid the radial heat flow, the fins on the inner shell are machined opposite the fueled portion of the specimens only. A graphical analysis is used to determine the effect upon the thermal resistance of the shells in the region of the fins due to longitudinal heat flow. The curve shown in Fig. 2 is used to determine the correction factor λ which is a function of fin section modulus and fin thickness. The thermal resistance to be used in the fin regions is found by calculating the resistance in the normal fashion, assuming radial heat flow, and then multiplying this resistance by the correction factor $(1+\lambda)$.

The use of radial fins results in additional problems.

1. Good contact is required between the fin tips and outer shell for heat transfer reasons.
2. The differences in operating temperature between inner and outer shell result in relative motion between the two shells causing plastic deformation and possibly galling of the two sliding surfaces.
3. The gap conductance must be well known, since heat fluxes across the fin tips can go as high as 2×10^6 Btu/hr-ft^2.

The capsules are designed with an interference fit at operation of 2 to 3 mils at the diameter. During the first rise to operating temperature, regions in the inner and outer shells will go beyond the elastic limit into the plastic range. This high stress results from the combined thermal and pressure stresses. A stress analysis shows that plastic flow occurs during the first cycle only and that during subsequent temperature cycles all parts remain in the elastic range.

Permanent deformation has been seen in the outer shell where the fins have forced their way into the shell.

In a recent hot lab examination, an inner shell was examined that was deformed between the two sets of radial fins (see Fig. 3). The deformation resulted from a combination of internal pressure, barrel temperature, and column loading caused by the fin regions gripping the outer shell. As temperature rose, differential thermal expansion caused a column load. This condition did not result in any failure to contain the liquid metal or fuel; the capsule irradiation was successful.

A-3. Natural Convection

There has long been concern with the effect of natural convection of the sodium in the capsule, but no detailed analysis has been made. In a recent capsule program of six capsules at the GETR, it was noticed that the readings of two thermocouples, about an inch apart on the same specimen, varied by 50 to 80°F, the top one always being higher. The variation in heat generation rate based upon the two thermocouples was too large to be explained by neutron flux profiles or end losses. A rough calculation was made and it indicated that if the sodium had a velocity of 5 ft/min enough heat could be carried from the bottom to the top of the specimen (~200 watts/in.) to account for the difference in the calculated heat flux. Certainly 5 ft/min is a reasonable value for convection velocity.

A-6. Specimen Temperatures in Terms of Thermocouple Readings

Thermocouples are located in the sodium, on the fin roots, and sometimes at the specimen center. In estimating the fuel clad temperature the sodium thermocouple is used, when available. Since sodium thermocouples do not always last through the entire irradiation, the fin root couple is often used. While all thermocouples are working, an experimental value of the capsule thermal resistance can be determined from the thermocouples and the calculated resistance from the thermocouple to the fuel clad or fuel center can be added to this. Then, if the thermocouple malfunctions, the thermal resistance multiplied by the heat generation rate can be used. To this one can add the reactor cooling water to obtain the temperature required.

The heat generation rate is determined from a calibration curve plotted from startup data. The Δt from the reactor water to fin root temperature is used, and a curve is plotted for every fin root thermocouple at each cycle startup. The calibration curves from cycle to cycle have been compared, and the differences in the curves are small, indicating that the thermal resistance from the fin root to reactor water does not change appreciably during irradiation. A sample curve is shown in Fig. 4.

The curve is plotted in the following manner:

1. With the reactor at zero power and full water cooling flow, the electric heater is turned on and increased to full power in predetermined increments. Temperatures are recorded at each heater power. Knowing the electric

heater geometry, the power output in watts per inch can be determined. This power output is plotted vs the Δt from the fin root thermocouples to the reactor water. In Fig. 4 this represents points 1, 2, 3, and 4.

2. The heater power is reduced to zero and the reactor is brought to an 8 MW power level. The assumption is made here that the fin root thermocouple fastened to the outside surface of the inner shell cannot discriminate between heat from the heater and fission heat traveling radially outward through the fin region. Therefore, an equivalent amount of heat from the heater or fission heat will result in a Δt increase of 100°F at the fin root thermocouple. With this reactor power increase the temperatures of the fin root thermocouples will rise. In Fig. 4, thermocouple 6 increased to a Δt of 212°F. This point is then plotted on the existing curve as point No. 5 and the heater power is again increased as described in item 1.

3. This procedure is followed until full reactor power is reached and the complete curve is plotted.

The advantages of the use of this curve become evident immediately.

1. Heat generation rates can be determined on a daily basis. Therefore, over a cycle the average heat generation rate together with minimum and maximum values are available.

2. Knowing heat generation rates, more accurate estimates can be made of fuel surface and center temperatures. A temperature curve can be plotted as a function of cycle time.

3. Specimen burnup can be calculated from cycle to cycle, avoiding the occasional surprises when isotopic or dosimetry results become available at the end of an irradiation.

The heater calibration curve gives a measure of heat flowing radially outward through the fin region. Therefore, gamma heat must be subtracted and corrections made for end losses beyond the fin region.

B. TEMPERATURE CONTROL DURING IRRADIATION

1. The capsule includes a doubly wound heater with a maximum output of 4 kw, located inside the inner shell. The primary function of the heater is to melt the sodium and to perform the capsule calibration at reactor startup. Previous attempts to use the heater for auxiliary heating were not very successful. Occasionally the heater is used to raise the capsule temperature; this is done with manual control.

III. FLUX DETERMINATION AND CONTROL

C. METHODS OF ESTIMATING FLUX PERTURBATION

C-1. Machine Codes

An estimate of flux perturbation is determined by using the United Nuclear PALINDROME code, which solves the one-velocity transport equation for cylindrical geometry in the P3 approximation. For an MTR irradiation the calculations assume the capsule to be in one of the single-hole aluminum A pieces in the MTR within the Be reflector. This calculation does not take into account any high absorbing materials in the region of the capsule.

In a recent program at the GETR a neutron flux and a heat generation rate were requested for each capsule. The emphasis was placed on the heat generation rate and GETR personnel located the capsule where the fission heat generation was as requested. Neutron fluxes change with experiment loading and fuel loading. Since heat generation rate determines temperature distribution, the aim is to maintain a constant value of heat generation rate.

IV. CAPSULE COMPONENTS AND FABRICATION

A. HEATERS

1. Type 347 SS sheath 0.125 OD, wall thickness 0.015 in.
2. High density MgO insulation
3. Heater element wire 0.032 Nichrome-V
4. Lead is nickel tube swaged over Nichrome 0.050 OD.

Coiled heater, two wound together, closely spaced coils. The coil length is ~7 in. and total heater output is ~4 kw.

Several calculations were made to determine the thermal resistance of the coiled heater assembly using various heat transfer models. Depending upon the model used, the values of resistance ranged from 0.125 to 0.250°F/watt/in. Experimental determination of the resistance from irradiation data indicated that the resistance was in fact 0.250°F/watt/in.

B. THERMOCOUPLES

1. Chromel-alumel stainless steel sheathed with sheath OD of 0.040 and 0.062, and MgO insulation.
2. Insulated junctions used in sodium-grounded junctions used as fin roots, thermocouples are ~4 ft long.

C. CONNECTORS

Fig. 5 is a sketch of the connector used for the junction between the thermocouple and lead wire. A ceramic material is placed in the voids and cured at 1100°F with a torch. A similar connector is used for the heater connection.

D. SEALS

The open ends of the heaters and thermocouples are sealed with an Aero Research product called Aero Seal. This is to prevent moisture takeup during the assembly period.

E. MATERIALS

Materials used are normally 304 SS, although nickel has been used on occasion where a high value of thermal conductivity was required.

F. WELDING AND BRAZING

The seals where the thermocouples and heater leads penetrate the inner and outer shell top plugs are made using a J-8100 braze powder, vacuum brazed.

All other seals are made by heliarc welding.

G. INSTRUMENTATION

Fig. 6 shows a front panel of a control console containing:

Multipoint Recorder

Single-Point Recorder

Ammeters and Voltmeters

Manual Powerstat.

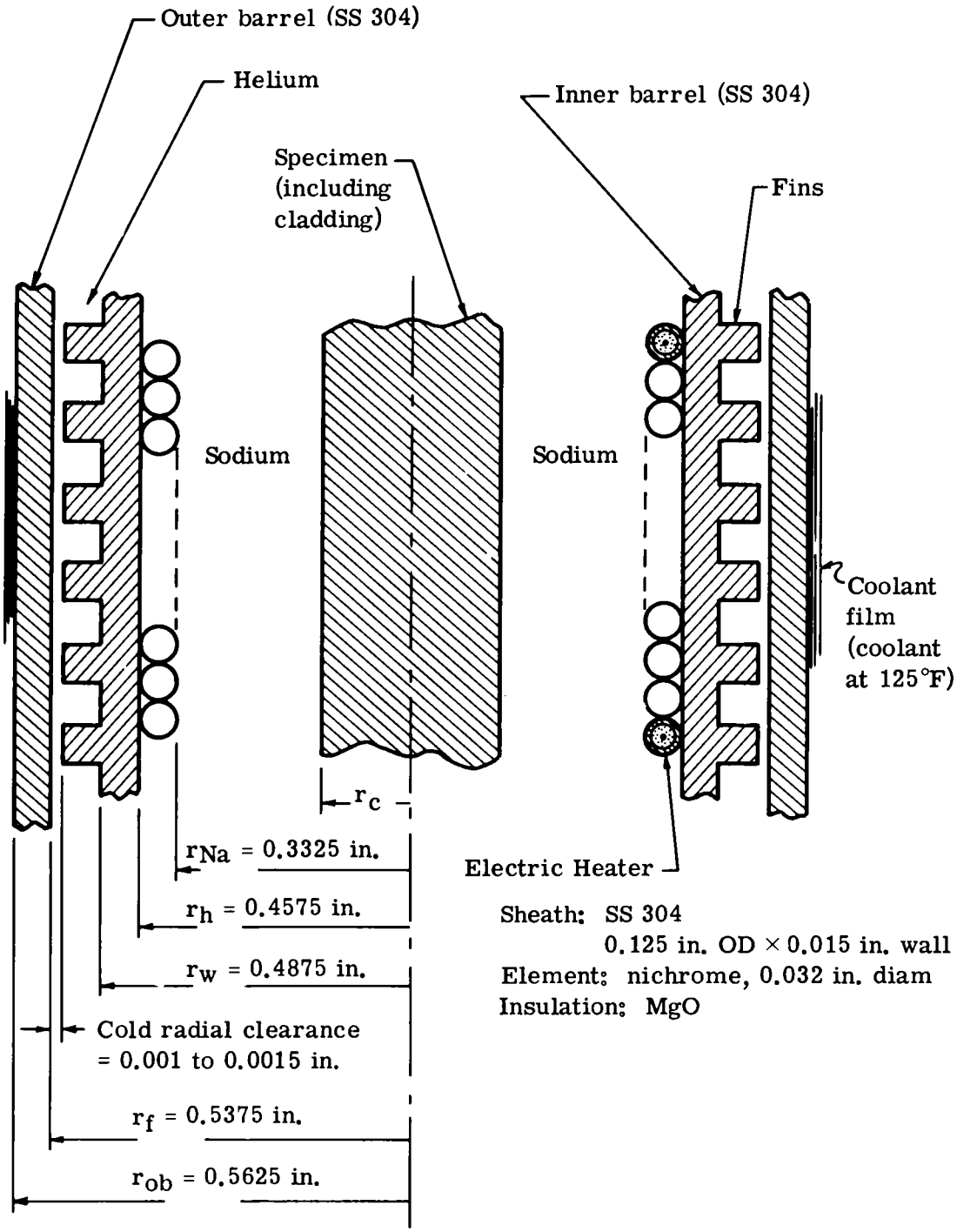


Fig. 1 — Irradiation capsule cross section

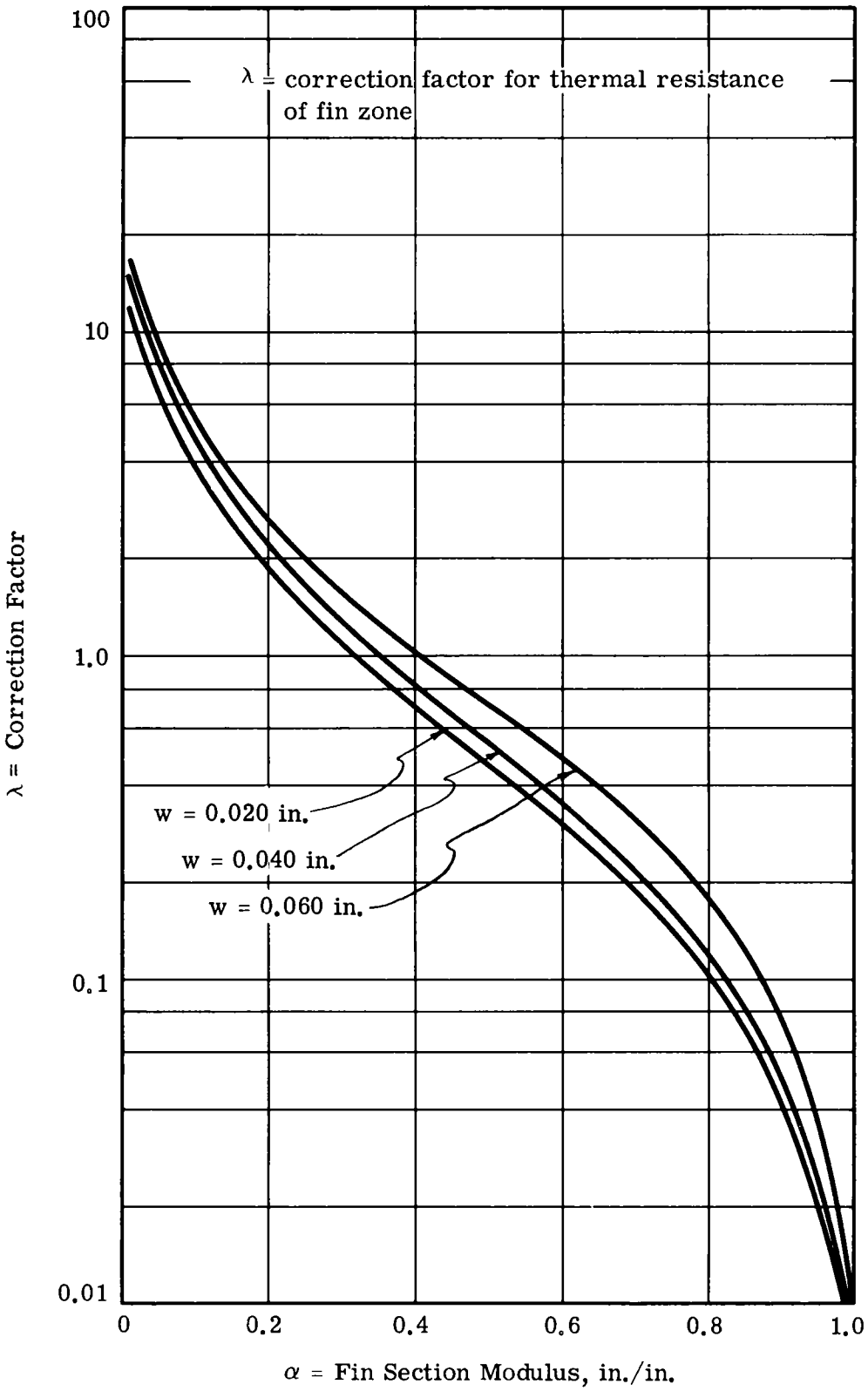


Fig. 2 — Thermal resistance correction factor for longitudinal heat flow

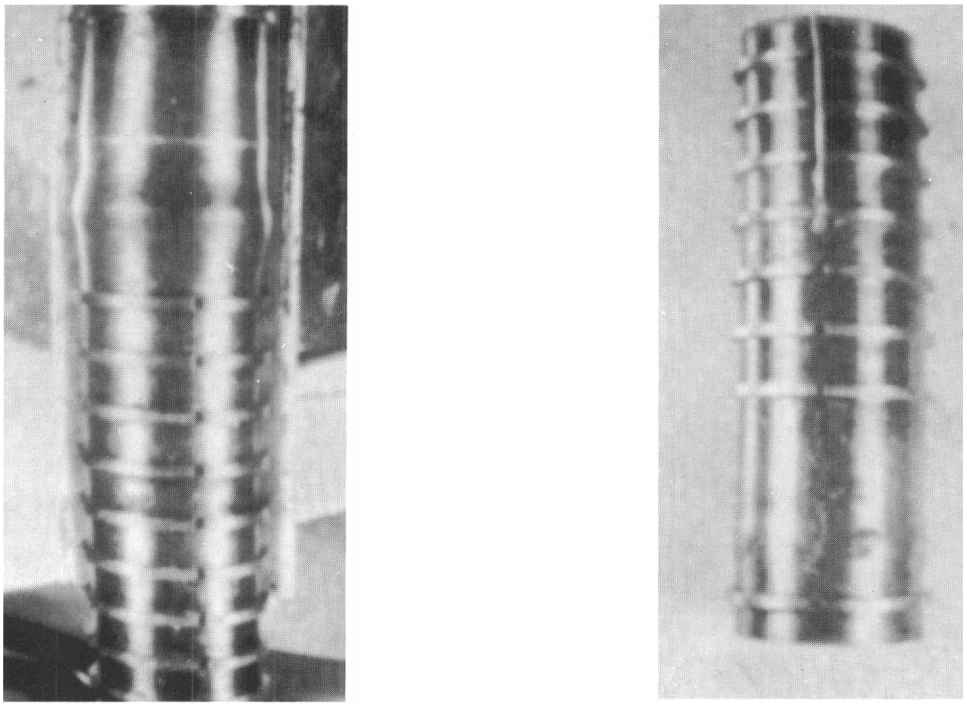


Fig. 3 — Inner shells – Capsule 64 (left) and Capsule 61 (right)

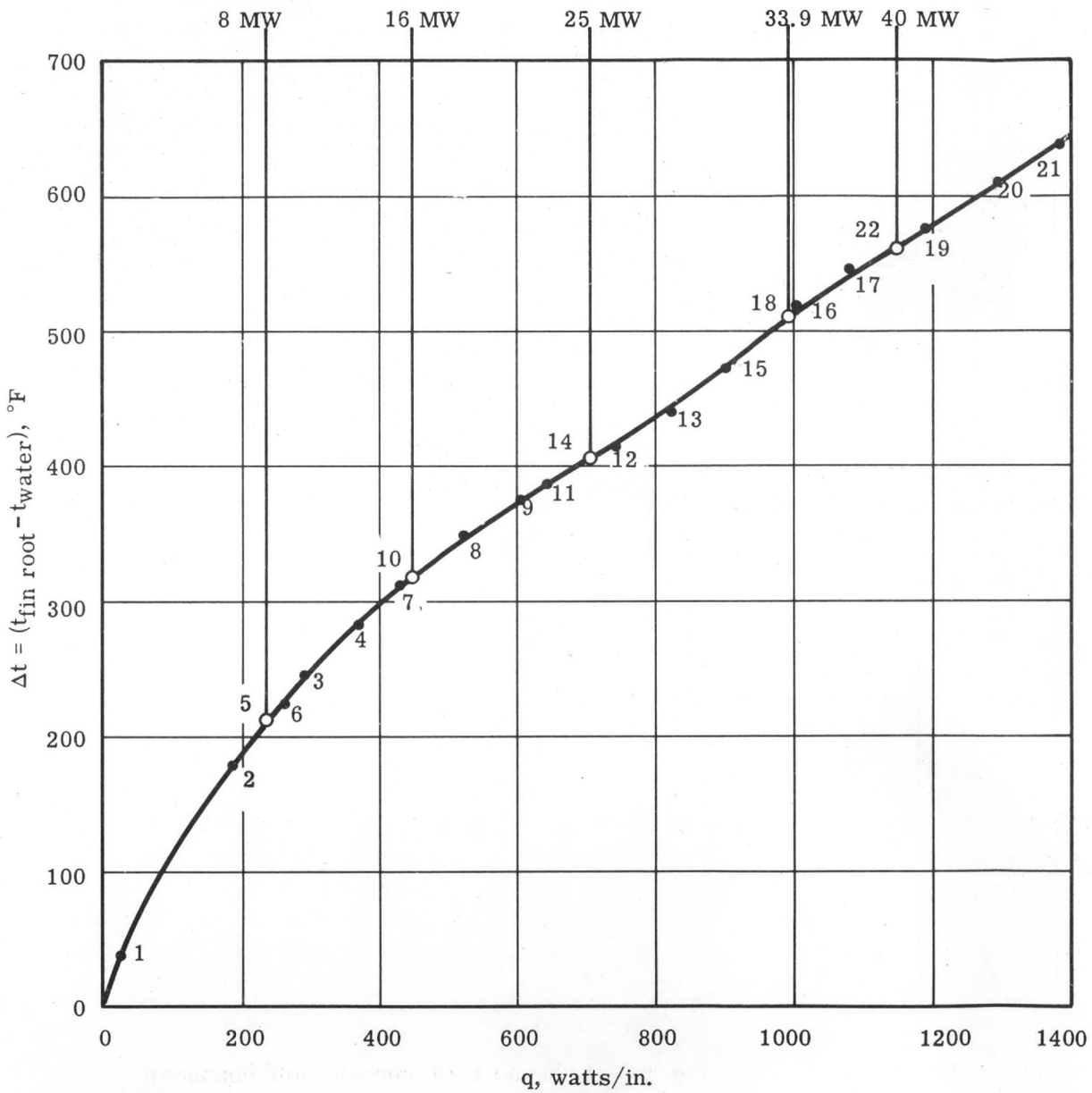
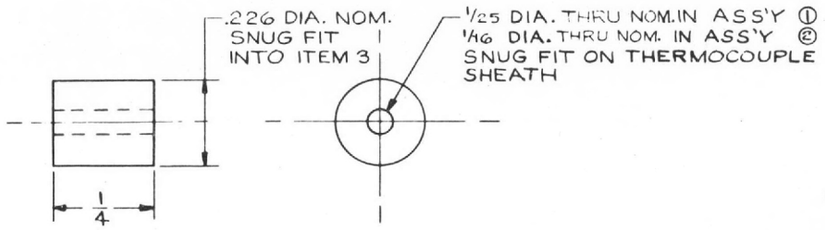
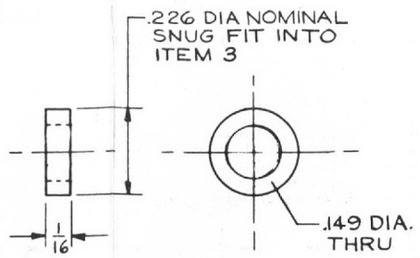


Fig. 4 — Heater calibration curve – thermocouple No. 3



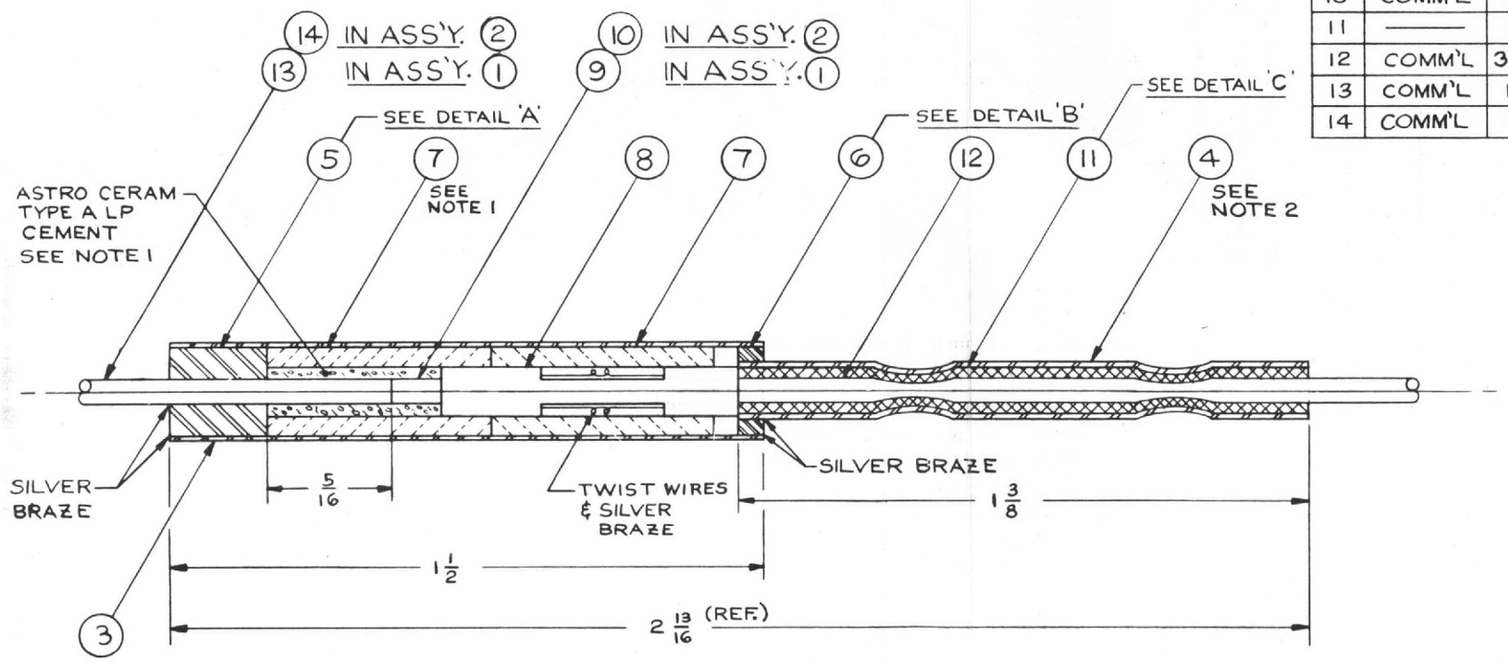
DETAIL 'A'



DETAIL 'B'

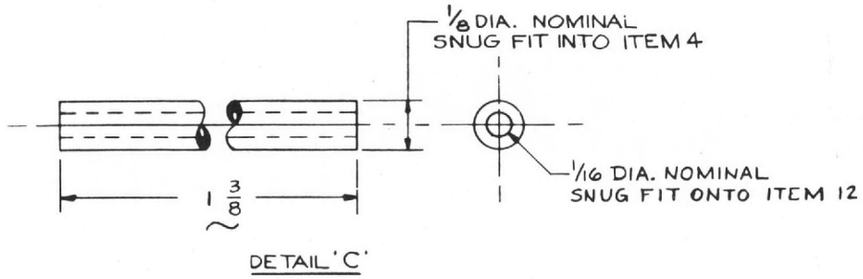
PARTS LIST

ITEM No.	SHOWN ON DWG. No.	QTY.	DESCRIPTION	MATERIAL	USED ON DWG. No.
1	———		1/25 THERMOCOUPLE CONNECTOR ASSEMBLY	———	14176
2	———		1/16 THERMOCOUPLE CONNECTOR ASSEMBLY	———	14176
3	———	1	1/4 O.D. X.012 WALL SEAMLESS TUBE	304 SS	———
4	———	1	.148 O.D. X.015 WALL SEAMLESS TUBE	304 SS	———
5	———	1	ADAPTER BUSHING	304 SS	———
6	———	1	COLLAR	304 SS	———
7	COMM'L	2	CERAMIC SLEEVE .220 O.D. X .131 I.D X 9/16 LG.	MULLITE ENGLEHARD	———
8	COMM'L	1	CERAMIC WIRE HOLDER PO 2132-0010	STEATITE	———
9	COMM'L	1	2 BORE(008) CERAMIC 1/25 O.D. X 1/8 LONG	MULLITE ENGLEHARD	———
10	COMM'L	1	2 BORE(016) CERAMIC 1/16 O.D. X 1/8 LONG	MULLITE ENGLEHARD	———
11	———	1	FIBERGLASS SLEEVE	FIBERGLASS	———
12	COMM'L	35'	THERMOCOUPLE LEAD WIRE 28 GAUGE GLASS COVERED	CHROMEL-ALUMEL	———
13	COMM'L	1	THERMOCOUPLE 1/25 O.D. SHEATH	CHROMEL-ALUMEL	———
14	COMM'L	1	THERMOCOUPLE 1/16 O.D. SHEATH	CHROMEL-ALUMEL	———



NOTES:
1. ASTRO CERAM IS CURED BY TORCH HEATING ITEM 7 TO A DULL RED COLOR.
2. CRIMP ITEM 4 IN TWO PLACES TO ALLOW ITEM 12 TO SUPPORT A 10 POUND PULL.
3. USE EASY FLOW 45 BRAZE METAL THROUGHOUT. REMOVE FLUX RESIDUE AFTER BRAZING.

① 1/25 THERMOCOUPLE CONNECTOR ASSEMBLY
② 1/16 THERMOCOUPLE CONNECTOR ASSEMBLY



DETAIL 'C'

Fig. 5 — 1/25 and 1/16 thermocouple connector assembly

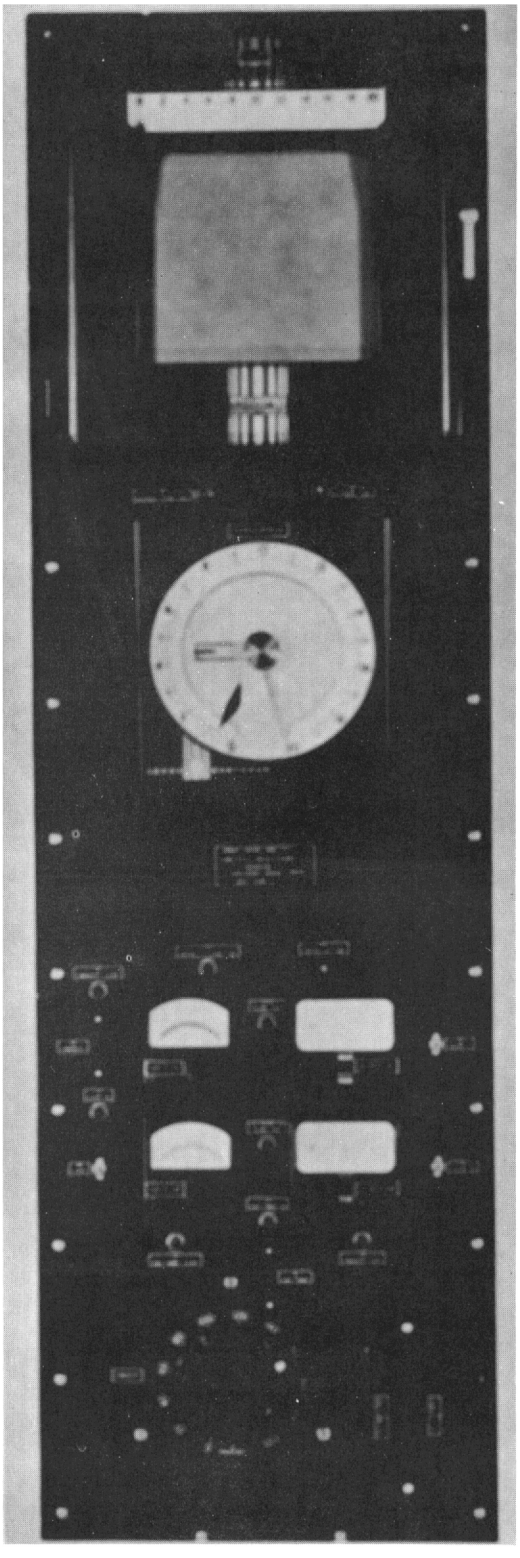


Fig. 6 — Typical control console

CONTRIBUTION TO

AEC MEETING

ON

PROBLEMS IN IRRADIATION - CAPSULE EXPERIMENTS

Compiled by

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I. CAPSULE DESIGNS

In the following are discussed some capsule designs which have been used in Phillips experiments in the MTR and ETR. These are not presented as novel designs since most experimenters depend heavily on their experience in previous irradiations to design for new experiments. In each case some "improvement", perhaps even an "innovation" is included in the new design. These modifications may not always produce the desired effects but it is believed that they as well as those that do result in appreciable improvement in capsule design should be discussed if only to prevent others repeating the same mistakes. Accordingly this contribution contains some comments on both type design features.

A. Controlled High Temperature Capsules for Materials Tests

The objective of a high temperature beryllium irradiation program in progress at the MTR-ETR is to determine the effect of irradiation (at these high temperatures) on the compressive strength properties, on the swelling, and on gas release from beryllium. Lead capsules each containing three beryllium specimens $\frac{1}{2}$" x $1\frac{1}{2}$" are encapsulated in a stainless steel can. Argon gas is used as a protective atmosphere around the specimens. A mixture of helium and nitrogen or argon and nitrogen (depending on the temperature desired) is used as a thermal barrier to control the capsule temperature. Specimen temperatures are in the range of 600-800°C with exposures to $>2 \times 10^{22}$ n/cm^2 (>1 Mev). These are continuous gas flow capsules as described in Section II. One design of this capsule used the gas lines helically wrapped about the inner can to provide the annulus spacing. Another technique spirally wrapped a .015" diameter stainless steel wire about the inner can to serve as the annulus spacer. The wire was spot welded in places to maintain a 1" pitch. Six thermocouples are included in the design to read specimen temperature, inner can temperature and to provide backup for the inevitable thermocouple failures. Exacting quality control is used in the form of rigid purchase specifications and inspection of all thermocouple wire used in the experiments. The fuel can temperatures are used to estimate power generation and burnup as described in Section II.

The first capsules of this type had two deficiencies. Thermocouples were attached by spot welding and were too easily broken loose during assembly. This has since been corrected by use of spade tips over the thermocouple junctions and attached by use of infrared or induction brazing techniques. Secondly the gas tubes in one capsule design were brought down along the outer surface of the capsule. Where these leads entered at the bottom, sources of leaks developed for water to enter the gas annulus. Induction brazing appears to have improved the joining and sealing although another design change now brings these leads down inside the outer can.

Doubly-contained, bulk-heads seal the thermocouple and gas leads at the top of the can. Nicrobraz 30 or Coast Alloy 50 appear the best brazing materials and induction brazing the best technique for obtaining the joint.

B. Low Temperature Capsules for Materials and Fuels Tests

As discussed in Section II, the use of gas annulus mixture for temperature control is limited to a certain lower temperature level below which it is no longer possible to operate. For capsule tests such as are required for ATR beryllium surveillance (approximately 250°F maximum surface temperature) and for U-Al fuels (approximately 350-600°F surface temperature) singly-contained capsules are generally required. In the case of NaK-filled uranium-fueled capsules the risk level must be evaluated and the Safeguards Committee might require double containment where the risk level is high. In this case the minimum operating temperature becomes appreciably higher unless a low flux position is selected for the irradiation which then increases experiment time. Singly-contained capsules are usually not controlled, although a degree of temperature "adjustment" may be obtained in some cases by changing the vertical elevation or radial position of the capsule. Such radial moves, however, are usually discouraged (See Section V).

II. CAPSULE HEAT TRANSFER

A. Burnup Estimation Based on Temperature Measurements

In most capsule irradiation programs it is necessary to estimate burnup as the experiment progresses in order to determine a discharge date. Final assessment of burnup for data purposes is normally based on actual chemical or spectrometric determinations. In many cases a beginning flux measurement to which suitable perturbation and cyclic correction factors are applied can be used to predict an exposure.

In an attempt to adjust for cycle-to-cycle variations and yet provide a simple, non-destructive measurement of heat flux which could then be related to burnup, a procedure based on temperature measurement has been adopted for some of our capsules. This technique has been moderately successful. In principle it appears sound but in practice there are still some unexplained discrepancies.

To explain this method it is first necessary to describe briefly these capsules which are currently being irradiated in the ETR. The capsule is a three-walled affair; the inner .750" stainless-steel can contains the UO_2 pellets, this can is positioned inside another can containing NaK, a third outer can forms the gas annulus between it and the outer surface of the second can.

The burnup estimation for this capsule is then obtained by the following procedure:

1. Obtain the fuel can temperature. This is an average of all the thermocouples attached to the outer surface of the fuel can.

2. Using this temperature read the unperturbed flux from a temperature flux curve.

3. Multiply the unperturbed flux by a perturbation factor to obtain the actual flux. From critical facility measurements a factor of 0.71 was obtained for solid UO_2 pellets.

4. Calculate the ratio of actual flux to design flux.

5. Multiply this ratio by the design burnup rate in MW/MT to find actual burnup rate.

6. Multiply the burnup rate by the equivalent full power days to obtain the burnup for the cycle in MWD/MT.

The temperature vs. neutron flux curves were developed from the assumptions that the resistances to heat transfer were known and that the ΔT between the fuel can and reactor coolant was known. This makes it possible to calculate a heat flux, and in sequence, a heat generation rate, a fission rate, and a neutron flux. This method also assumes that the neutron flux stays constant throughout the cycle; an assumption which should result in conservative burnup estimation.

B. Variable Gas Mixtures for Temperature Control

An accepted solution for control of high temperature capsules is the use of a gas annulus containing a mixture of gases having different thermal conductivities. He-N_2 mixtures are probably the most commonly used although He-A mixtures offer a wider range of thermal conductivity. At the MTR-ETR we prefer not to use Argon because of the induced activity of the A^{41}. This type of temperature control requires a multi-wall construction (which may be of advantage if a liquid metal is used in one of the inner vessels) with very close tolerance of the spacing being required. This in turn complicates the design heat-transfer calculations and makes estimation of flux perturbation difficult. Three-dimensional heat transfer codes however are available for the former and the latter parameter may be obtained by flux measurements in a nuclear mock-up in a critical facility.

Temperature control by gas annulus mixture is limited in its range of operation by the heat flux (or neutron flux) and the annulus spacing as well as the range of gas thermal conductivity. For example with a 5 mil annulus which we feel is as small as practical, the temperature drop across the annulus will be anywhere from a few hundred degrees to 1000-1500°F depending on the reactor location. Since most experiments require a certain burnup (or burn-up rate) the neutron flux to obtain this burnup normally limits the lowest temperature at which a capsule can be controlled by means of gas annulus mixture.

An early technique of temperature control by gas annulus mixture was the premix or batch method. This was the method used in the EGCR capsule experiments. With this procedure, a predetermined quantity of helium and nitrogen was premixed in a batching tank, the thermal conductivity measured and the gas pumped into the capsule annulus. This method worked fairly well except for two major problems: (1) it was difficult to know when the proper mixture had been obtained, (a calibration curve of thermal conductivity vs. temperature gave only approximate results), (2) with a zero-flow system, operating at high temperature the gas mixture tends to stratify after a relatively short time and the temperatures start to drift. To get around the first difficulty, the composition of the gases is changed in small increments with each new mixture being pumped into the capsule to determine its effect on the temperature. This procedure (which can be time consuming) is followed until the desired temperature is reached.

With high temperature capsules (1600°F for a 12 mil annulus EGCR capsule) care must be taken to change the gas mixture well before the maximum limit since the first amount of new gas may force ahead of it some of the separated gas components (e.g., N_2) which may further increase the temperature at the control thermocouple to the point of scramming the reactor.

Furthermore with the usual rather long gas line between the mixing tank and capsule, a considerable amount of the previous gas mixture must pass through the capsule before the new mixture can be effective. In spite of the problems associated with this method of control, it can be made to work with adequate attention and has been used for over five years on the EGCR capsule program.

A somewhat better method of temperature control involves a continuous flow system. In this system the two gases (usually helium and nitrogen) are brought to the capsule in separate lines. The gases are mixed at this point,

the gas mixture then flowing either upward or down through the capsule and exhausted to the stack. With the continuous flow system, the temperature can be changed by changing the ratio of the flowing gas streams, i.e., by increasing the flow rate of one gas with respect to that of the other. This results in a relatively fast response system and a fairly sensitive and reliable system.

This system is used in our PAED-47 (Beryllium) capsule experiments and has been successfully used also by other experimenters. As one would expect, the continuous mix system requires somewhat greater quantities of gas which for single small experiments is of no great consequence. The following comparison illustrates the quantities and costs involved in the continuous system capsule compared with a somewhat similar capsule using a batch premix system.

Experience with the EGCR capsules shows that approximately 1-3/4 bottles of helium are used per capsule per year, and approximately 0.85 bottles of nitrogen per capsule per year.

The gas requirements for the continuous flow system in the PAED-47 capsule are about 7 bottles of helium and 7 bottles of nitrogen per capsule per year. This amounts to an annual cost increase per capsule of $140.00, a modest increase for the improvement in control.

III. FLUX DETERMINATION AND CONTROL

A. Neutron Flux Monitoring Procedures

Until fairly recently, experimenters tended to take neutron flux monitoring somewhat for granted. Not nearly as much effort was put into a determination of exposure (based on flux monitors) as into changes in physical and mechanical properties which were then carefully plotted against the exposure. As a result, comparison of one site's data with another's often led to confusion in interpretation. A recent cross correlation of flux monitors all irradiated at one site but counted at various sites actively engaged in flux monitoring work produced some startling differences. Fortunately for future comparisons most of these differences have since been resolved. Even now however there still exist some differences of opinion on fast flux monitoring with individual preferences for certain monitor materials. Since much of the AEC sponsored irradiations are located at the MTR-ETR site and largely involve flux monitoring by Phillips personnel, the following outline is presented to describe techniques in current use at this facility. For more complete descriptions, the reader is referred to IDO-16538(1) and IDO-16744(2). The results of extensive thermal flux mapping in the ETR and MTR are presented in IDO-16719(3) and IDO-16657(4).

Thermal Neutron Flux (2200 m/sec flux)

1. Monitor Materials

   a. Pure cobalt wire (usually 40 mil diameter for measurements at N_L. Reaction Co^{59} (n, γ) Co^{60}.

   b. 0.5 or 0.1 w/o Co-Al wire, 40 mil diameter for measurement at full power. Reaction Co^{59} (n, γ) Co^{60}.

2. Determination of Co^{60} Disintegration Rate

   a. Count in an ionization chamber calibrated with an NBS Co^{60} standard or,

   b. count the 1.33 mev gamma-ray on a multi-channel pulse height analyzer.

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(1) IDO-16538, Hogg, C. H., "Thermal Neutron Flux Measurement at the MTR-ETR Site".

(2) IDO-16744, Hogg, C. H., Weber, L. P., and Yates, E. C., "Thermal Neutron Cross Sections of the Co^{58} Isomers and the Effect on Fast Flux Measurements Using Nickel".

(3) IDO-16719, Weber, L. D., Hogg, C. H., "ETR Thermal Neutron Flux Measurements for Cycle 27".

(4) IDO-16657, Weber, L. D., Hogg, C. H., "MTR Thermal Neutron Flux Measurements for Cycle 146".

3. Adopted Constants

a. Co^{59}, 2200 m/sec cross section - 36.3 b.

b. Co^{60} half life - 5.27 yr.

4. Corrections

a. The 40 mil cobalt wires have a self shielding - flux depression factor of 1.18.

b. Total activities include contributions from resonance neutrons that must be subtracted. The magnitude of the correction is determined from Cd ratios.

5. Working Equation

$$\phi = \frac{1.18 \times F \times D}{N\sigma\,(1-e^{-\lambda t_e})e^{-\lambda t_w}}$$

Where ϕ is the 2200 m/sec neutron flux, F the fraction of the disintegration rate due to thermal neutrons, D the disintegration rate, N the number of Co^{59} atoms in sample, σ the Co^{59} cross section, λ the Co^{60} decay constant, t_e the exposure time and t_w the time elapsed between cessation of irradiation and time counted. The factor 1.18 is omitted for calculations involving the Co-Al alloy.

6. Errors

a. The estimated error in the absolute flux value for the point at which the wire was irradiated is $\pm$ 5%. Some additional error is associated with the extrapolation from N_L readings to full power fluxes.

7. Special Problems

a. Pure cobalt will corrode if left in contact with water too long. The corrosion products then slough off yielding incorrect results and contaminating equipment.

b. Pure cobalt should not be counted less than about one day after exposure to allow Mn^{56} and Co^{60m} to decay.

c. The Co-Al alloy should not be counted less than about 5 days after exposure to allow the Na^{24} to decay.

Fast Neutron Flux (>1 mev)

1. Monitor Materials for Measurement at Either N_L or Full Power

a. Pure nickel wire (usually 20 mil diameter). Reaction, Ni^{58} (n, p) Co^{58}.

b. Pure iron wire (usually 20 mil diameter). Reaction Fe^{54} (n,p) Mn^{54}.

c. Pure titanium wire (usually 20 mil diameter). Reaction Ti^{46} (n, p) Sc^{46}.

2. Disintegration Rate Determination

Same as for Co^{60}.

3. Adopted Constants

a. Cross sections averaged over a fission neutron spectrum: 95, 68 and 9.5 mb for Ni, Fe and Ti, respectively.

b. Half lives: 71.3 d, 314 d and 84.1 d for Ni, Fe, Ti, respectively.

c. Fraction of neutrons in the fission spectrum having energies greater than 1 mev - 0.692.

4. Corrections

a. The Co^{58} has a large thermal neutron cross section and a correction for burnup must be made as described in IDO-16744.

5. Working Equation

$$\Phi = \frac{D \times 0.692}{N\sigma(1-e^{-\lambda t_e})e^{-\lambda t_w}}$$

Where 0.692 and σ are defined above and other symbols are as defined under "Thermal Neutron Flux Monitoring".

B. Control Rod Motions

Since most nuclear reactors depend for their control of power on the movement of absorber material into or out of the core, the inevitable consequence of such mechanical control is a shift in neutron flux in experimental positions. In the case of the MTR and ETR these motions are vertical and result in a shift of the peak axial flux upward during the cycle. This particularly affects beam hole experiments in the MTR as the flux peak passes across the tip of the beam hole. In the ATR the rotating rods will very much affect the capsule irradiations in the reflector. By ganging rod movements, the localized flux shifting is somewhat reduced but there is always a minimum flux shift resulting from rod movement to handle xenon override. In the ATR a design objective was to hold the neutron flux as constant as possible during the reactor fuel cycle and over successive cycles. Specifically the total neutron flux variation was to be restricted to 10%. In the MTR and ETR the end of cycle flux in some positions may be as great as a factor of two higher than the beginning cycle flux. These fluctuations occur primarily in core positions usually in the upper half of the irradiation facility. The fluctuations in the reflector positions are less. Since controlled-temperature (lead) capsules in the MTR may only be located in the reflector the controllable range need not be as great as the factor of two flux change might indicate. In the ETR case where

the lead capsule may be in the core, the situation may be improved by locating the capsule with the top at mid plane and extending into the lower half of the test facility. Considerable thought has been devoted to possible improvements in the MTR and ETR operation to obtain a more stabilized flux. The proper distribution of fuel and burnable poisons would materially reduce the axial and time variations in flux. Preliminary studies indicate that at least the ETR can approach the ATR capibilities through such improvements. The work involved in such an improvement program for these reactors is rather extensive requiring (1) an elaborate computer program of a three dimensional approach, (2) fabrication metallurgy involving an entirely different fuel core since the alloy does not lend itself to non-uniformly distributed fuel, controlled distribution of burnable poisons, or high fuel loadings, and (3) critical facility measurements involving whole core loadings and extensive flux mapping and finally, (4) a full core ETR or MTR loading.

To eliminate flux perturbations caused by mechanical rods, a chemical control concept was initially proposed for the ATR. Because the development of this method of control was not sufficiently advanced it was not accepted for the ATR. However the development has progressed as a loop test. When translated into practice the concept of chemical control for reactors offers a number of advantages over the more conventional methods of control: (1) the parasitically absorptive material ordinarily present in control rod guide structures is eliminated, (2) the flux distribution axially will be improved and shifting of axial peak will be eliminated, (3) more compact core, (4) reduced fuel inventory and costs, and (5) improved neutron economy will result in improvement in conversion or breeding ratio. The present concept being tested at Phillips consists of a circulating water solution of boric acid of controllable, variable concentration, in a tube in the ETR to serve as a regulating rod for ETR control. The status of this work is described in IDO-16812(5).

C. Methods of Estimating Flux Perturbation

Emperical Correlation Methods

We at Phillips have found that the semi-emperical method outlined by W. B. Lewis in his method "Flux Perturbations by Material Under Irradiation"(6) is a useful tool for predicting effective or perturbed flux in samples irradiated at the MTR and ETR. The method is based upon two rigorous equations which involve physical properties as well as shape and symmetry factors. These equations may be used to develop plots of the so-called parameters $\bar{b}$ versus the theoretical average attenuation in the sample, the effective flux factor, and the ratio of effective flux to unperturbed flux at the sample surface. Samples of known dimensions and compositions were irradiated in the MTR in regions of known unperturbed flux. The amount of product was determined experimentally and the effective flux factors were determined directly from one of the two rigorous equations. The experimental results were found to agree with the theoretical values quite well. The parameter $\bar{b}$ may be calculated

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(5) IDO-16812, Zelezny, W. F., editor, "The Design of a Dynamic Corrosion and Chemical Control Test Loop and Preliminary Out-of-Pile Test Results".

(6) Nucleonics, Volume 13, October 1955.

since it is related to the summation of cross sections for the various materials in the sample and to the average path length. Use of parameter $\bar{b}$ assumes that shape and symmetry factors are insignificant. To what extent this is so is evident in the good agreement between theoretical and experimental results. Some sponsors at MTR-ETR have applied another factor, besides the effective flux factor, which is based on actual past experience with their various irradiations. This experience factor is probably applicable in many cases where the capsule to be irradiated is complicated with several annuli and various materials. Shape and symmetry factors then become significant and the Lewis derivation becomes less accurate in its application.

Nuclear Mock-ups

Nuclear mock-ups of capsule type experiments are not required for MTR irradiations or for the beryllium or aluminum reflector irradiations in the ETR. However, nuclear mock-ups are required for all irradiation experiments which are to be inserted in the core of the ETR. The ETR core is that area within the inner boundaries of the beryllium reflector. The experiment mock-ups for the ETR need not be identical to the experiment proper but must be within $\pm$ 10 volume percent on water displacement and within $\pm$ 20 percent on macroscopic cross section. Each ETR reactor core loading is mocked up in the ETRC (ETR critical facility), and an effort is made to fuel the core and/or program the control rods so that minimum cycle-to-cycle flux changes occur. Flux measurements made in the ETRC will indicate the effect of local perturbers.

IV. CAPSULE COMPONENTS AND FABRICATION

Connectors

In recent months it has become apparent that some type of intank connector would be necessary in order to satisfactorily insert lead experiments into the ETR. The loop experimental hardware seriously interferes with lead experiment insertion. A number of different types of intank connectors have been tried, and we now have arrived at a preference for one developed by GEFP. Details of this connector are found on Phillips drawings ETR-C-2349 and ETR-E-2350. These drawings are available on request.

It is, of course, recognized that each time a thermocouple, electrical or gas lead is severed, a chance exists that a good connection will not be made. In the ETR, however, more lead capsules have been damaged through forcing the long leads in around the intank hardware than by poor assembly of intank connectors. Except in unusual cases we now require intank connectors on all lead capsules that go into the ETR.

Materials

The following is quoted from IDO-16428, Rev. 1[7] and applies to capsules for irradiation in the MTR and ETR:

> "The materials of construction of the parts exposed to the reactor process water must not corrode, slough off or in any other way cause deleterious effects on the high quality process water. Copper, cadmium and/or lead cannot be used. Recommended materials include aluminum or aluminum alloys (of less than 0.5% copper), 347 or other stabilized stainless steel, and zircaloy. Heliarc welding is recommended for joing metallic parts. Silver solder is unacceptable as a brazing compound.
>
> No material of construction shall be used which, under irradiation, will produce corrodible or soluble products -- for example, the production of mercury from gold.
>
> The corrosion effects of the materials contained in the capsule must be taken into consideration. Special precaution, such as double encapsulation, must be taken to insure the containment of liquid metals such as mercury."

The foregoing requirements and restrictions are aimed at protection of our reactor systems.

Welding and Brazing

Experience gained at the MTR and ETR indicate that silver solder is not a satisfactory brazing material for parts to be exposed to a high neutron

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(7) See list of MTR-ETR Standard Practice Guides attached.

flux. On irradiation silver solder becomes brittle and somewhat porous. Higher temperature brazing materials appear to be more satisfactory. Because of the high activity level of Ag-109 and some difficulties we have encountered with such activity in the MTR and ETR process water, we do not allow silver solder to be used in places where it would be exposed to the reactor internals. External to the reactors, silver solder is satisfactory. For all welding on capsules we recommend the heliarc method.

Instrumentation

With only one exception, all instrumentation needed for capsule type experiments at the MTR and ETR is furnished by the experiment sponsor. Such instrumentation is installed in a semi-permanent manner so that it can be operated efficiently.

A 300 point input signal data handling system is available for use at the ETR. This system can be programmed to any of the following ranges:

(1) 0-1600°F Iron-Constantan thermocouple.
(2) 0-2000°F Chromel-Alumel thermocouple.
(3) 0.1 to 0.5 volt linear MV retransmission and linear pneumatic-to-electric transducers for pressure and level signals.
(4) 0-20 MV strain gage transducer.
(5) 12.5 to 62.5 millivolt square root flow signal from pneumatic-to-electric transducers.
(6) 12.5 to 62.5 millivolt signal to read out digitally 2 to 12 linear.
(7) Point delete.
(8) Point skip.

A more complete description of this system can be found in IDO-16395 (addendum).(8)

(8) See list of MTR-ETR Standard Practice Guides attached.

V. REACTOR CHARACTERISTICS AS THEY AFFECT CAPSULE EXPERIMENTS

Flux Mapping Schedules

About 25 MTR capsule positions and about 50 ETR capsule positions are checked for thermal flux each reactor cycle. In this way a fairly up-to-date record of flux levels in all areas of each reactor is maintained. Additional flux determinations are made as the result of special requests from sponsors for specific positions. Fast flux and gamma heat mapping of these two reactors is done less frequently and normally only at the request of sponsors.

Availability of Space

Less than half of the space available for capsule irradiations in the MTR and ETR is currently occupied. Much of this space is at lower and therefore less desirable flux levels. Space is normally assigned after delivery of the capsule experiments to the reactor site. A certain amount of prescheduling is done for experiments that must be designed for specific reactor facilities. In order to be able to properly schedule the reactor, minimum deadlines have been established for delivery at the reactor site of design data and the experiment proper. These deadlines for the next year are shown in an attachment.

Movement of Lead Capsules

In general, radial movement of lead capsules is discouraged because of the possibility of failure of thermocouples, heater leads, etc. Despite this we can generally make successful moves of lead capsules in the MTR. In the ETR such moves are much more difficult because of the interferences of the intank loop hardware and because of the limited experiment lead egress ports. Conduits bent to specific contours for a particular position rarely are adaptable to another position although relocations over very small distances may be possible because of the minor changes involved in lead contour. ETR lead capsule moves must be well planned in advance, and intank connectors must be installed on the leads prior to the initial insertion.

On-stream Schedules

The MTR operates on a three week cycle basis and in the past year has achieved a 67 percent calender day on-stream factor. The ETR operates on a six week cycle basis and in the past year has achieved a 45 percent calender day on-stream factor. The ETR factor is lower primarily because of the maintenance effort required on the major loop facilities.

VI. GENERAL - MTR-ETR STANDARD PRACTICE GUIDES

Over the past several years, Phillips Project Engineering Branch has developed certain standard practices with regard to irradiations in the MTR and ETR. These standard practices have been reduced in writing to so called Standard Practice Guides and have been issued as IDO reports. The information contained in these reports has been of considerable value to capsule irradiation sponsors. A list of these MTR-ETR Standard Practice Guides is attached.

MTR/ETR MINIMUM DEADLINES FOR CAPSULE EXPERIMENTS

(Based on IDO-16428, Rev. 1)

MTR Cycle	ETR Cycle	Capsule Irradiation and Design Data	Non-Instrumented Capsule Delivery	Instrumented Capsule Delivery	Scheduling Date	Shutdown Date
198		9-9-63	9-20-63	9-9-63	9-20-63	9-30-63
199		9-30-63	10-11-63	9-30-63	10-11-63	10-21-63
	59	10-7-63	10-18-63	10-7-63	10-18-63	10-28-63
200		10-21-63	11-1-63	10-21-63	11-1-63	11-11-63
201		11-11-63	11-22-63	11-11-63	11-22-63	12-2-63
	60	11-18-63	11-29-63	11-18-63	11-24-63	12-9-63
202		12-2-63	12-13-63	12-2-63	12-13-63	12-23-63
203		12-23-63	1-3-64	12-23-63	1-3-64	1-13-64
	61	12-30-63	1-10-64	12-30-63	1-10-64	1-20-64
204		1-13-64	1-24-64	1-13-64	1-24-64	2-3-64
205		2-3-64	2-14-64	2-3-64	2-14-64	2-24-64
	62	2-10-64	2-21-64	2-10-64	2-21-64	3-2-64
206		2-24-64	3-6-64	2-24-64	3-6-64	3-16-64
207		3-16-64	3-27-64	3-16-64	3-27-64	4-6-64
	63	3-23-64	4-3-64	3-23-64	4-3-64	4-13-64
208		4-6-64	4-16-64	4-6-64	4-17-64	4-27-64
209		4-27-64	5-8-64	4-27-64	5-8-64	5-18-64
	64	5-4-64	5-15-64	5-4-64	5-15-64	5-25-64
210		5-18-64	5-29-64	5-18-64	5-29-64	6-8-64
211		6-8-64	6-19-64	6-8-64	6-19-64	6-29-64
	65	6-15-64	6-26-64	6-15-64	6-26-64	7-6-64
212		6-29-64	7-10-64	6-29-64	7-10-64	7-20-64
213		7-20-64	7-31-64	7-20-64	7-31-64	8-10-64

MTR/ETR
STANDARD PRACTICES GUIDES

IDO Number	Title	Price
16343	Calculated Surface Temperatures for Nuclear Systems and Analysis of Their Uncertainties (The first report of the Heat Transfer Subcommittee of the Phillips Reactor Safeguard Committee) - Edited by R. J. Nertney	$0.45
16368	Hydraulic Flow Calculations for MTR/ETR Experiments - R. B. Van Sice	$2.00
16380	Standard Practices for Design of MTR and ETR Safety Circuits - Edited by L. H. Jones	$1.00
16380-Supplement	Standard Practices for Design of MTR and ETR Safety Circuits - Edited by L. H. Jones	$0.50
16395 and Addendum	Instrument and Electrical Standard Practices for MTR and and ETR - Edited by L. H. Jones	$0.50
16428 - Rev. 1	In-Tank Irradiation Facilities for MTR-ETR - Edited by A. S. Richardson	$2.25
16431	Engineering Liaison for MTR-ETR Experiments - Edited by R. J. Nertney	$1.50
16453	Source Strength Information for Shielding & Stack Effluent Calculations - L. H. Jones	$0.75
16460	The Application of Statistical Methods of Analysis and Experimental Design in Predicting Burnout Heat Flux - by R. T. Jacobs, J. A. Merrill and R. J. Nertney	$1.00
16466	Standard Practices Guide for Hazard Analysis of Experimental Systems - by R. J. Nertney	$1.25
16472	Standard Practices Guide for Writing Experimental Operating Manuals - Edited by J. D. Ford	$1.00
16516	(Supplement to IDO-16368) Hydraulics of MTR-ETR Capsule Experiments	

Above publications may be obtained at:

Office of Technical Services
U. S. Department of Commerce
Washington 25, D. C.

SOME FEATURES OF IRRADIATION CAPSULES USED BY OAK RIDGE NATIONAL LABORATORY, SOLID STATE DIVISION

by

R. G. Berggren

SOLID STATE DIVISION
OAK RIDGE NATIONAL LABORATORY
Operated by
UNION CARBIDE CORPORATION
for the
U. S. Atomic Energy Commission
Oak Ridge, Tennessee

Water Cooled Capsule for Irradiation in a Partial Fuel Element

A series of irradiations of sheet nickel tension test specimens was conducted in a partial fuel element in the core of the ORR. It was desired to irradiate the specimens in the highest possible fast neutron flux at temperatures near the coolant water temperature. The design of the capsule is shown in Figure 1. The specimens were spaced 0.030 inches apart to permit flow of reactor coolant water over each specimen. The pressure drop through the capsule was 24 psi, the coolant flow rate was 14.7 gpm, gamma heat was 10 watts/gm, inlet coolant temperature was 125°F, and coolant velocity was 6 to 18 ft per sec. With these conditions the maximum calculated metal temperature was 210°F.

The specimen holder was removed to permit changing fuel elements and was disassembled and reassembled in the ORR hot cell to change specimen loading during the series of exposures.

The special fuel element, in the C-7 facility of the ORR was surrounded by active fuel elements, and the capsule contained 750 grams of nickel specimens. This rather heavy loading of nickel depressed the thermal neutron flux in the vicinity of the capsule. As a result, the neutron flux in the capsule was measured as about 60% that calculated for an empty capsule.

Temperature Control for Groups of Notch-Impact Specimens

Groups of carbon-steel notch-impact specimens have been irradiated in the ORR at controlled temperatures between 500° and 750°F in a helium atmosphere. The present design, Figure 2, permits irradiation at controlled temperatures between 250° and 800°F by controlling 1) thickness of thermal insulation installed during assembly and 2) power dissipated by the heater element.

Thermal insulation is specified such that, with gamma heat only, the specimen temperatures will approach but not reach the desired exposure temperature. Adding controlled electrical heat holds the specimens at the desired temperature. The need for a compact, high power density heater was solved by use of nichrome ribbon wound on a mica card in the manner of a home toaster element. Used in the manner shown in the figure, these heaters can be operated at wattages several times the usual design limits for nichrome ribbon since they are sandwiched into the assembly to provide fairly high heat dissipation rates to the cool wall of the capsule. These elements have been operated at powers as high as 200 watts per square inch.

Individually Temperature-Controlled Exposure of Tensile Specimens

Iron and steel tensile samples will soon be irradiated in a high flux region of the ORR at controlled temperatures between 150° and 700°F in a helium atmosphere. In order to irradiate samples in this temperature range it is necessary to remove a large amount of reactor induced heating, followed by the addition of controlled amounts of electrical heat.

Two variations of our capsule design are illustrated in Figure 3. Basically, the design permits removal of the gamma heat induced in the sample by conduction through vibratory packed aluminum powder to the capsule shell. The thermal energy is then transferred to the cooling tubes by metallic conduction and/or gas conduction depending on the irradiation temperature desired for a particular specimen. In order to irradiate at approximately 150°F the capsule must be in good contact with the cooling surfaces. To achieve elevated temperatures, a sheathed heating element is wound around the capsule and a helium gas gap is maintained between the capsule and the cooling surfaces.

The thermal design parameters of this capsule were computed using a modification of a computer program originally written for a fuel element of similar design. This program related the helium gas gap to specimen

temperature at various gamma heating rates. Typically, for a specimen temperature of 300°F at a gamma heating rate of 4½ watts per gram, the helium gas gap must be about 0.020 inch. A laboratory mockup of this capsule design gave data in excellent agreement with the computer results. Additional experiments with the capsule mockup showed that this design had broad flexibility with respect to design and operating parameters. The sheathed heating element permits controlled operation at temperatures up to 800°F.

Tube Burst Tests

A series of tube-burst tests have been run in the ORR Poolside Facility. Some of the materials, tube dimensions and operating temperatures are shown below.

TYPICAL SPECIMENS

	Wall Thickness	Diameter	Length	Temperature
Inconel*	.030	1/2	2 1/2	1500°F
Zircaloy*	.030 - .040	3/4	3 1/2	500 - 900°
304 S. S.	.020	3/4	2 1/2	1100-1700°
Cb-1% Zr	.015	1/2	2 1/2	1800-2000°
Carbon Steel	.020	1/2	4	800°

*Auxiliary cooling employed.

Temperature control for the specimens was obtained by using electrical heaters. Since the gamma heating is around 5 watts per gram, the heat

produced in a specimen and its furnace must be taken into consideration in designing an experiment. In some cases it was necessary to provide forced cooling for the specimens to permit the electrical heaters to be used to control the temperatures.

The first heater type used was made by winding nichrome wire on a fused quartz tube. The heaters were divided axially into three sections and the power for each section was regulated by a separate controller. The mass of the furnace and its heat shielding properties made cooling necessary for specimens operated at 1500°F or below. "Cold fingers" inside the specimen were used, air cooled for Inconel and water cooled for the Zircaloy specimens. Figure 4 shows the construction of a Zircaloy specimen and heater.

A second type of heater was later employed and was fabricated from 0.040" metallic sheathed heater cable. The windings were spaced and permitted radiant and convective cooling of the specimen. The specimens were inserted in pockets in the water cooled can and with an atmosphere of 90% He satisfactory operation at 1300°F was obtained.

The latest type of specimen contains the three section heater within the specimen leaving the outer surface completely open except for the control thermocouples. Components for this type assembly are shown in Figure 5. These operate at 800°F with carbon steel 0.020" wall specimens. With the electrical power off, the reactor holds the temperatures of the specimens between 600 and 700°F.

Gamma heating effects at abrupt section changes must be carefully considered when a uniform specimen temperature is desired. Flat heads closing

off the ends of the tubes were found unsatisfactory, producing a hot spot at the ends of the sample tube. An inverted dome closure permitted a thinner head to be used and reduced the effect. Current specimens are prepared with a domed heat butt welded to the tube ends.

Fission Spectrum Irradiation Facility

A facility is being designed for use with the Bulk Shielding Reactor which will provide space for irradiating specimens with essentially pure fission neutrons. The facility, shown schematically in Figure 6, will consist of a 3" I.D. tube for specimen access surrounded by a U^{235} cylinder 12" in length. Neutrons from the reactor are moderated during passage through a thick tank of D_2O and produce fission when absorbed in the U^{235}. A void is provided around the outside of the U^{235} cylinder to reduce the feed back of partially moderated neutrons to the sample chamber. It is estimated that the specimens will be exposed in a flux of 3×10^{11} neutrons/cm^2sec.

ACKNOWLEDGMENTS

The author wishes to thank Messrs. N. E. Hinkle and W. E. Brundage for their assistance in preparing this paper.

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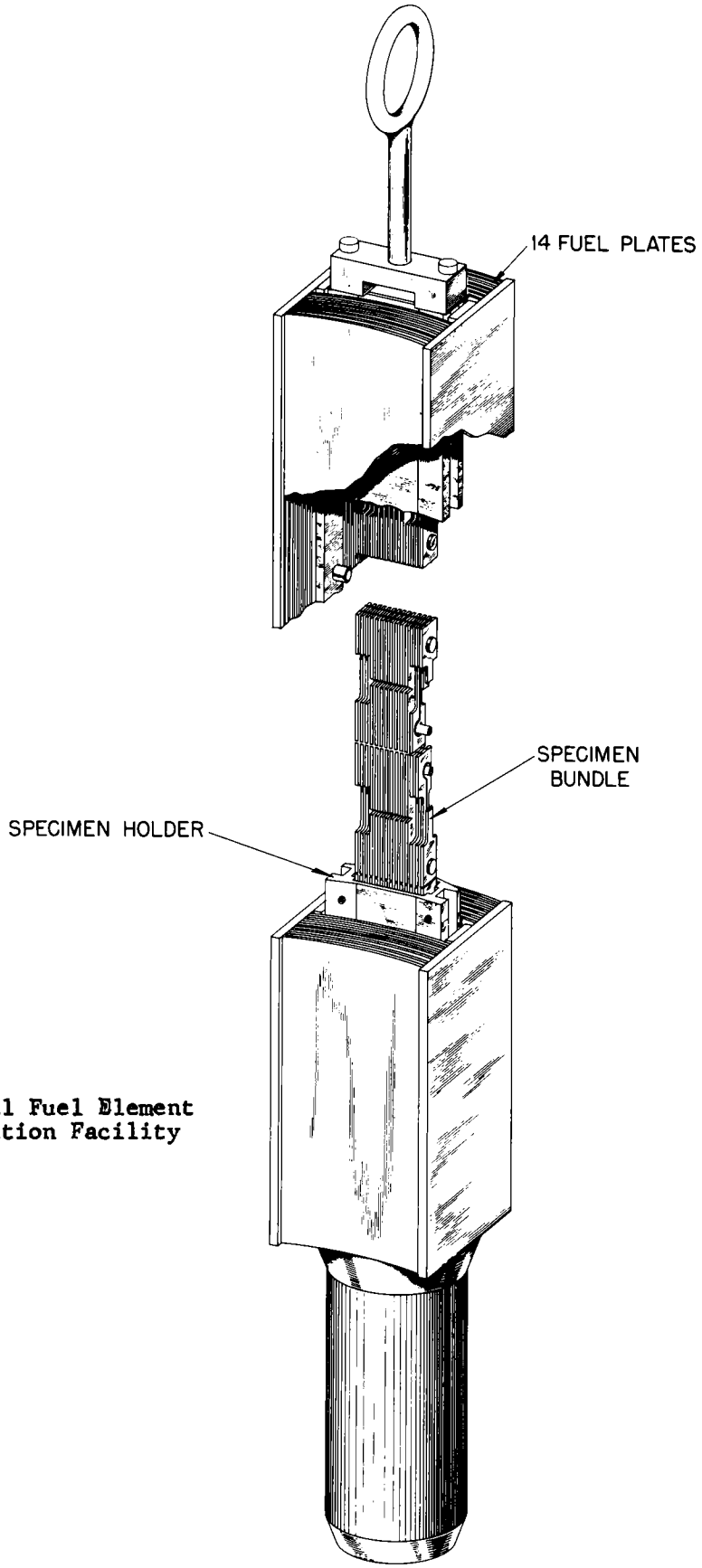
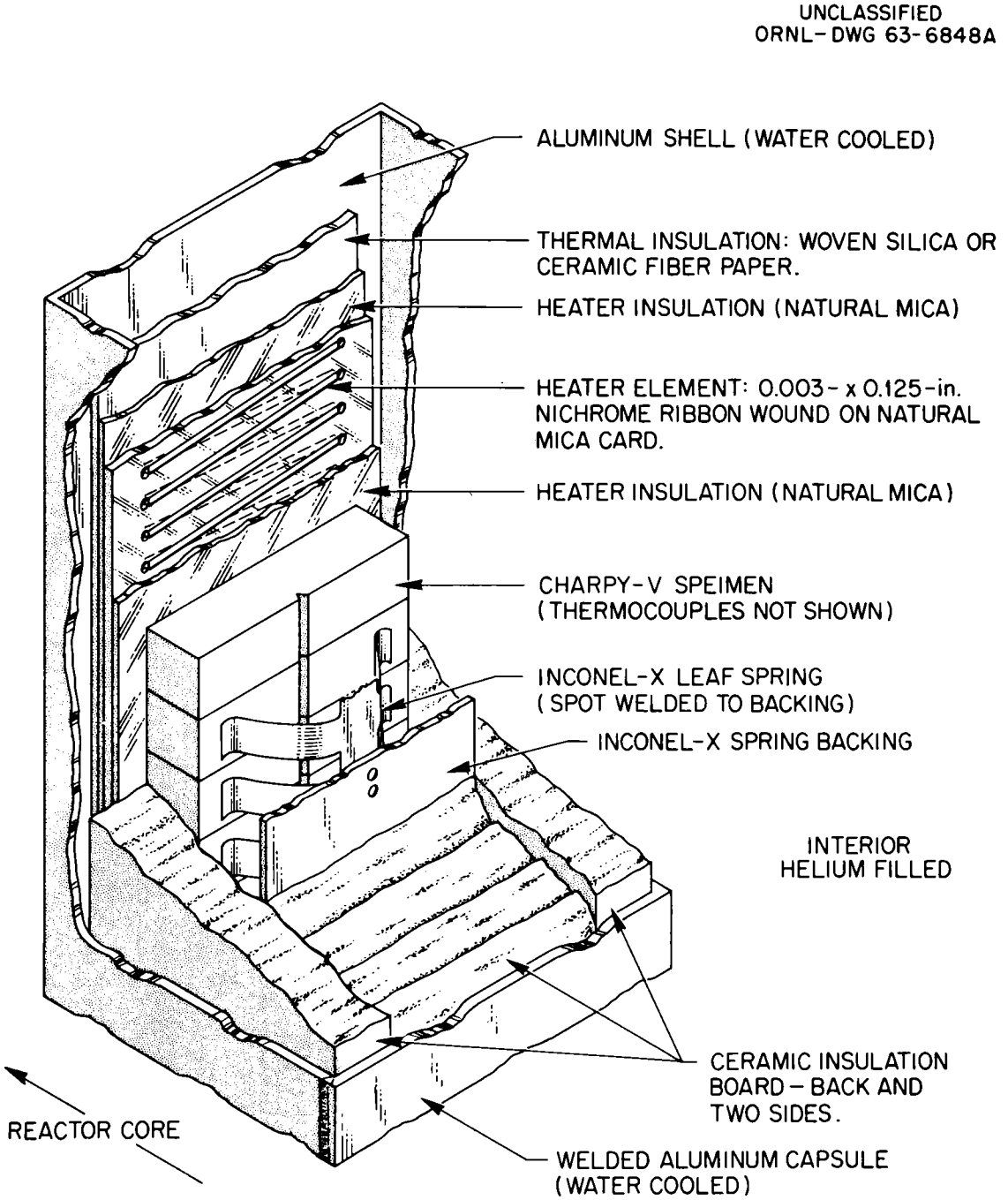


Figure 1. Partical Fuel Element Irradiation Facility

Figure 2. Controlled Temperature Irradiation Capsule .



IRRADIATION CAPSULE FOR CHARPY IMPACT SPECIMENS

Figure 3. Individually Temperature Controlled Capsules

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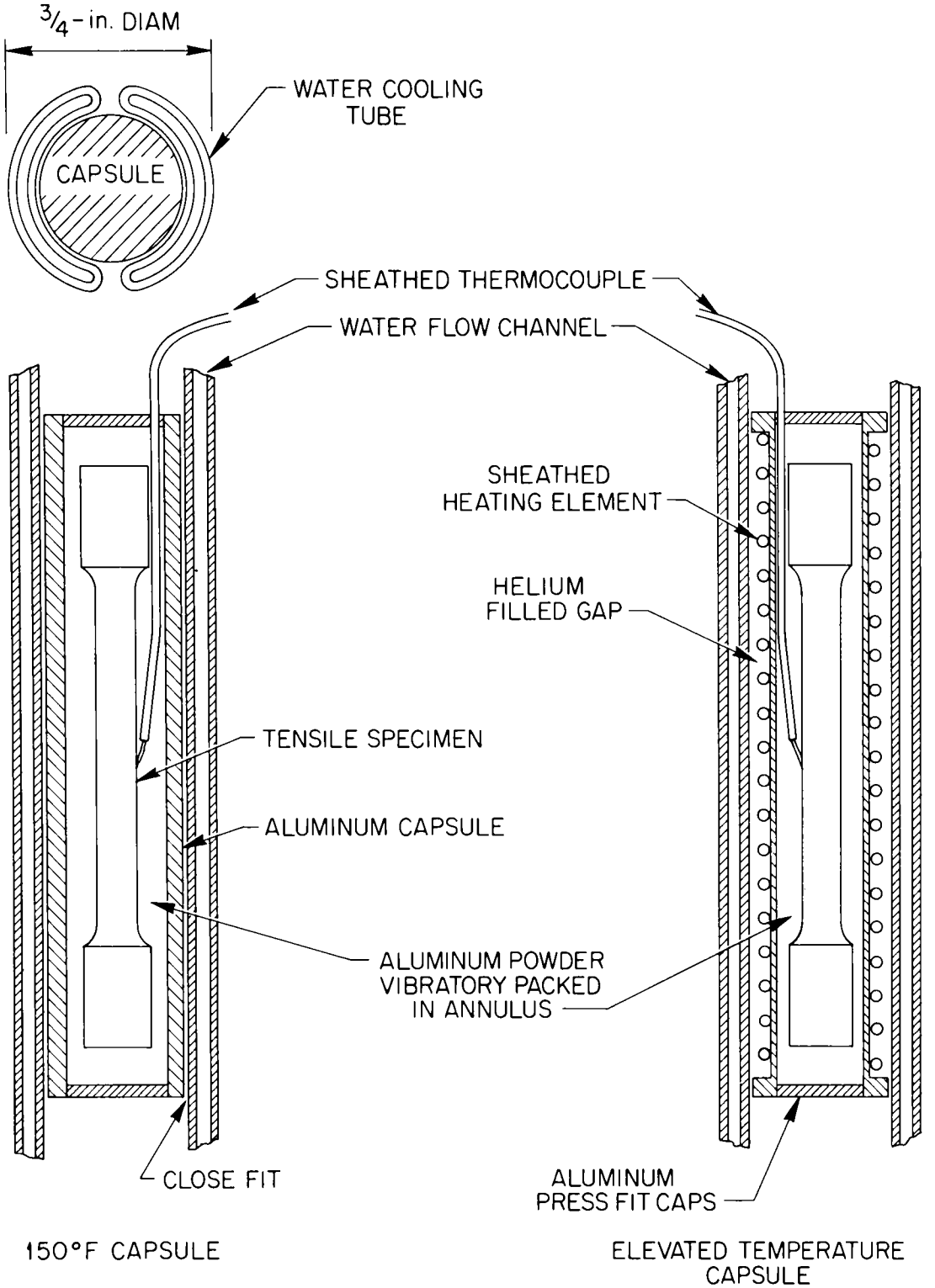

INDIVIDUALLY TEMPERATURE CONTROLLED CAPSULES

Figure 4. Zircaloy Tube-Burst Specimen and Heater

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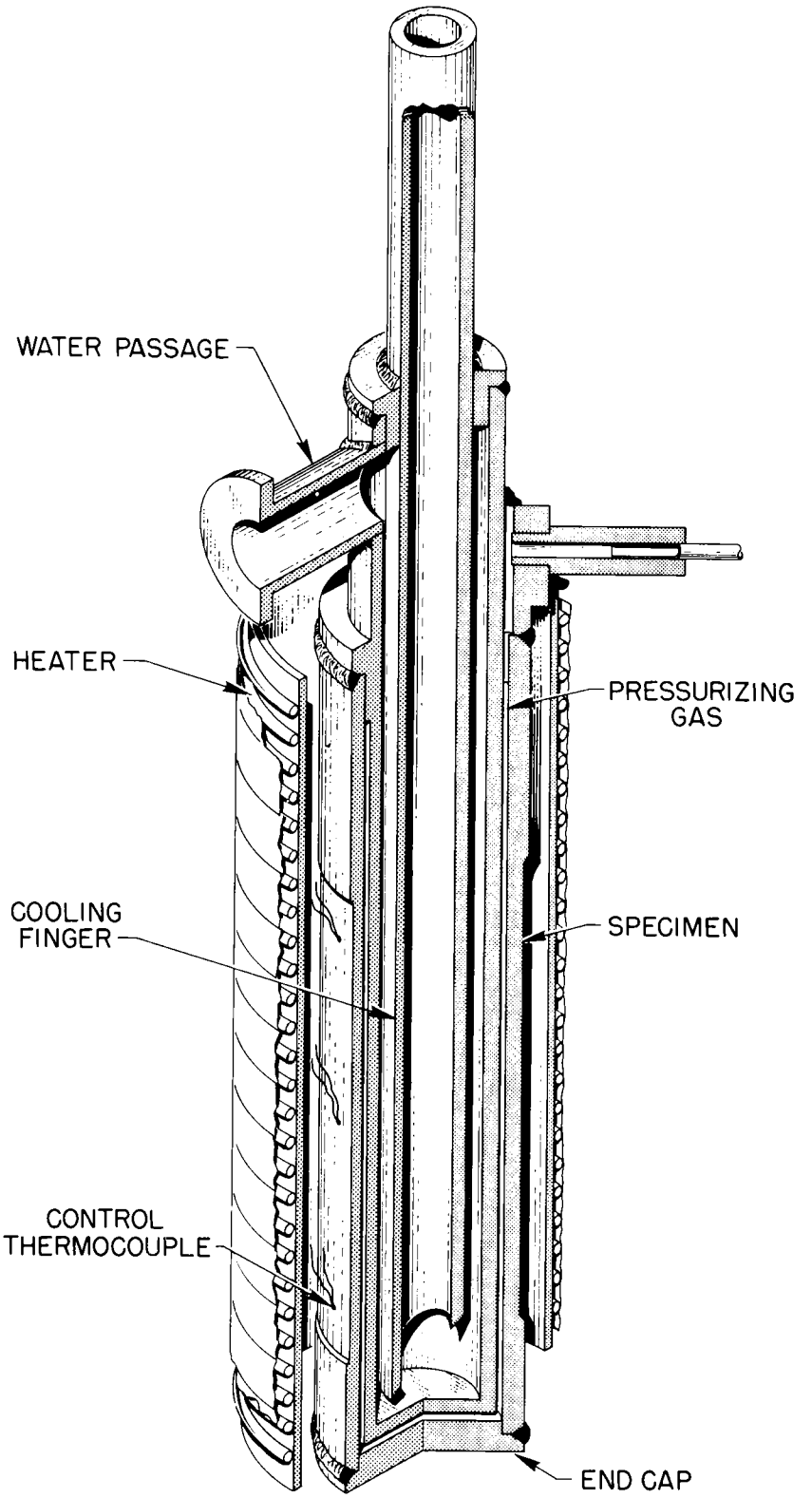


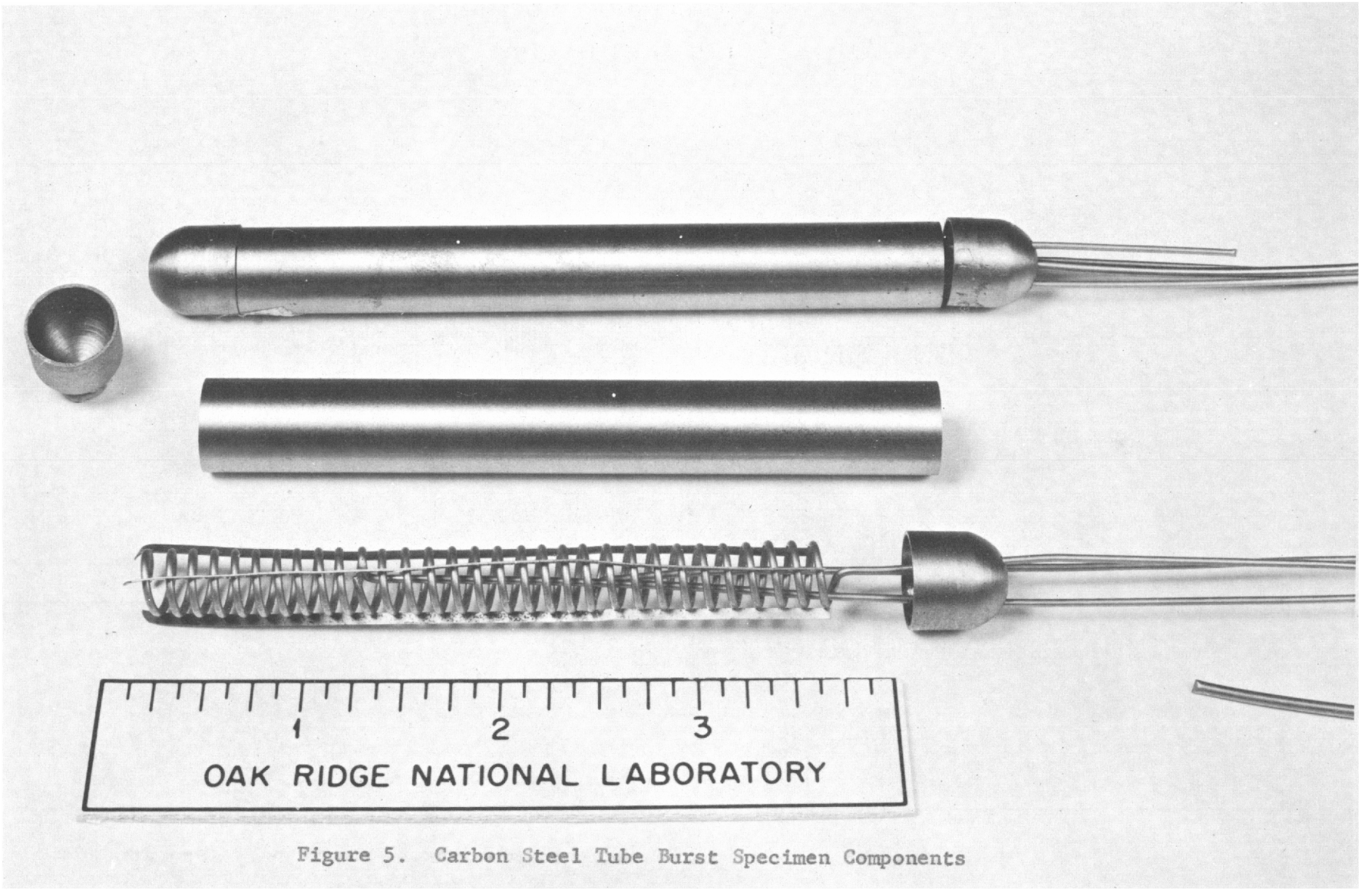
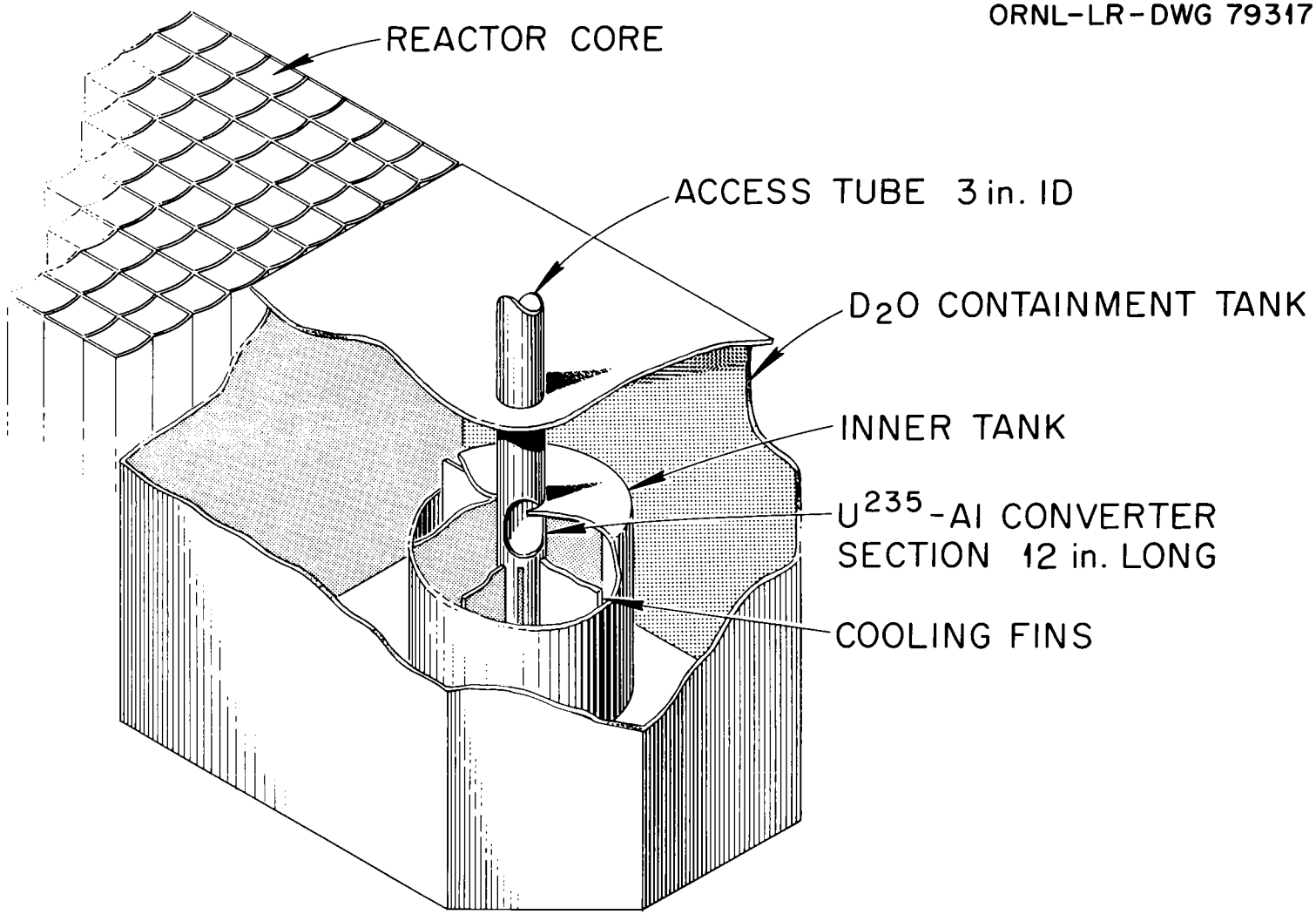


Figure 5. Carbon Steel Tube Burst Specimen Components



Fission Spectrum Irradiation Facility
(Neutron Converter)

Figure 6. Fission Spectrum Irradiation Facility

FABRICATION OF INSTRUMENTED IRRADIATION CAPSULES FOR GAS-COOLED REACTOR PROGRAM AT THE OAK RIDGE NATIONAL LABORATORY

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SUMMARY

A large number of in-pile irradiation experiments have been performed in support of the Gas-Cooled Reactor Program at the Oak Ridge National Laboratory. These experiments have been required to study the behavior of both the standard and advanced-type gas-cooled reactor fuel elements under irradiation. Items of interest in these studies are: cladding temperature, fuel central temperature, coolant-gas temperature, fission-product-gas release rate, and coolant pressure drop. Each in-pile experiment was designed and fabricated to measure one or more of these quantities.

The bulk of the experiments were based on the standard fuel element design for the Experimental Gas-Cooled Reactor. These fuel elements use a 0.750-in.-OD × 0.020-in.-wall, type 304 stainless steel cladding tube. The fuel is in the form of sintered UO_2 bushings. These considerations dictated the basic dimensions used for the in-pile capsules.

The essential components of any in-pile capsule are the cladding tube, a plain bottom end cap, the fuel, and a specialized top end cap, depending upon the quantities to be measured. The capsules were fabricated by suitable, specially developed welding and brazing processes.

Wherever possible, end caps are welded into the cladding tubes, using a semiautomatic tungsten-arc welding process. In general, one end cap can always be welded in automatically; but, if the capsule is highly instrumented, the other end cap must be welded in manually. An edge-weld-type joint design was used for all end-cap welds. For the semi-automatic welding sequence, the tubes were clamped in a copper chill ring and rotated under a stationary welding torch. The welding sequence was

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*Operated for the U. S. Atomic Energy Commission by Union Carbide Corporation.

completely programmed, giving welds with the proper penetration and a minimum of roll over.

Normally, the capsules were to contain static helium gas. Thus, the final closure welds were made manually in a helium-filled, inert-atmosphere chamber. Where applicable, the fuel in the loaded capsules could be vacuum outgassed in the same chamber prior to final closure.

One of the experimental variables of greatest interest was the temperature of the fuel cladding. In order to monitor this quantity with the least perturbation of its value, thermocouples were attached to the inside wall of the cladding of the in-pile capsules. These thermocouples were 1/16-in.-diam, stainless steel-sheathed, Chromel-P/Alumel thermocouples. They were brazed longitudinally to the inside surface of the fuel cans, and the fuel bushings were slotted to pass over them. Since the slots in the fuel are only slightly larger than the thermocouples, they must be brazed in place in a perfectly straight line. Further, the thermocouples must be brazed for their entire length within the capsule, and their penetration through the end cap must also be sealed by brazing.

A considerable amount of experimentation resulted in a satisfactory procedure for installing these cladding-temperature thermocouples in the capsules. Copper was used as the brazing material and was preplaced by electroplating a 0.001-in.-thick deposit on the surface of the thermocouple. A brazing jig was used to hold the thermocouples properly in position on the inside of the tubes. The brazing jig was made of Inconel X because this alloy is not wetted by copper in a dry-hydrogen atmosphere. Brazing was done in a specially constructed muffle furnace with a movable hot zone. This enables high heating and cooling rates to be achieved. After brazing, the thermocouple-to-can joint was ultrasonically inspected for its entire length, and the thermocouple-to-end-cap joint was mass-spectrometer leak checked.

To measure fuel-center temperatures, a refractory-metal thermocouple must be used. These thermocouples were normally W vs W–26% Re. They were used either in a tantalum-sheathed form or with bare junctions. In most instances, the central thermocouple must be protected

from reacting with the fuel by placing it in a thermowell. These wells were made of molybdenum tubing with a molybdenum end cap welded into the lower end. The open upper end of the well had a stainless steel sleeve copper-brazed to it. This sleeve was in turn welded into the upper end cap of the capsule.

A special design was developed for the welded junction of the tungsten-rhenium thermocouples. This was necessary due to the differences in thermal expansion of the two materials and to the inherent brittleness of the weld and heat-affected zone in the tungsten wire. This design has been successful in limiting mechanical failure of the junction as a result of thermal stresses.

In one particular in-pile experiment, sweep gas was used to remove fission products from the inside of the capsule. The design called for a tantalum-sheathed central thermocouple in a well which had an open bottom end so that the sweep gas could enter the well. Thus, the thermocouple had to be sealed to a special fitting where it entered the top of the capsule. This involved the development of a special electron-beam brazing technique using a reactive metal brazing alloy (titanium-copper-beryllium) to effect the joint between tantalum and stainless steel after the rest of the capsule had been assembled.

A number of capsules were fabricated with a capillary tube leading from the top end cap to a pressure transducer. These were used to monitor fission-gas release by noting the rate of internal-pressure buildup in the capsule.

In the most recent series of in-pile experiments which are presently under way, the behavior of coated-particle fuels in the form of spheres is being investigated. For these experiments, each capsule contains from three to eight fuel spheres in graphite holders. Sphere central temperatures, holder temperatures, and fission-product release are measured. Numerous thermocouples must be brazed through the top end caps of these capsules.

The GCR in-pile testing program has resulted in the development of numerous specialized techniques for the fabrication of the required capsules. With the appearance of each new capsule design, new techniques are developed and existing ones updated to result in the optimum procedures for that particular experiment.

SAFETY REVIEW OF IN-PILE EXPERIMENTS

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It is conservatively estimated that during the past five years about 2,000 capsule experiments have been reviewed to determine their acceptability for loading into the General Electric Test Reactor (GETR). Surprisingly few of these experiments have failed or otherwise caused operational problems during irradiation. Presented below in topical form is some information pertinent to the safeguards review procedures used at Vallecitos Atomic Laboratory and some suggestions for those responsible for in-pile experiment design.

PURPOSE

The obvious and over-riding purpose of a safeguards review for an in-pile experiment is to assess the nuclear safety characteristics of the experiment. If these characteristics meet or exceed the safety standards established for the reactor, the experiment is acceptable; if they do not meet these standards, changes must be made or the program would be rejected. That which is to be protected or safeguarded, in order of importance is:

1. Personnel
2. Equipment
3. Lost time and inconvenience

Often an additional purpose of the safeguard review is to assure that all regulatory requirements are satisfied such as AEC, State, or local in-house requirements and licenses.

METHODS

With a wide variety of experiments, approaches, design purposes, etc., the methods employed to review each program independently and with some degree of conformity must be rather flexible. The basis for this review should be the association or relationship between those responsible for actually performing the experiment and an administratively independent safeguards group. Through meetings, presentations, and the cooperation of these two groups an assessment of the nuclear safety characteristics can emerge. Scheduling has been and will continue to be a problem. Adequate time is essential if a quality review is to be performed. The experimenter or test engineer obviously should strive to present quality information, but what is also important and often overlooked is he must also include a specific request, (i.e., limits, parameters, set points, etc.). If the reviewer has a clear understanding of what question is being asked of him, other than "Is it O.K. ?", his job is done more easily and accurately. The items of interest to the safeguards reviews are quite broad and greatly dependent on the program; therefore no attempt is made to list even the classical items.

TECHNIQUES

The time-honored technique is a cautious and thorough review of the experimenter's data package. Again, the problem of standardization comes up, because in order to assess each program thoroughly and objectively, it would be convenient to have some semblance of likeness among the data packages submitted. Conversely, by requiring all designers to standardize their data submittals, or, to go one step further, to standardize their design techniques, would greatly inhibit the designer and impede innovation and advances in the art of capsule or apparatus design. The desirability of data package standardization can be argued pro and con; however, it is the author's belief after several years of review experience at the GETR that data package standardization should not be stressed.

PROBLEMS

The problems of either the reviewer or presenter (designer) could easily be the topic of several excellent papers. The problems suggested here are not new, but for some reason they are perennial. Scheduling -- obviously a problem in the reviewer/presenter relationship. Incomplete data, inaccurate data, and inconsistencies again represent impediments to progress and meeting the desired outcome in a timely manner. In the author's experience the problem that designers fear most -- that of design deficiencies -- is the problem most infrequently encountered.

SUGGESTIONS

The following three suggestions were selected for your consideration --

1. Assist the reviewer by clearly stating exactly what the request involves. Make this request sufficiently broad to allow some flexibility while also making it specific and unambigious.

2. Consider the possibility of failure during the experiment and make provisions for such, whenever possible, to extend the useful life of the experiment. For example, design for a primary objective, but upon failure of key instrumentation, fall back and continue the the irradiation to meet a secondary objective. Also, design with the idea of possible remote repair of radioactive apparatus. Experience has shown that had the possibility of minor failures been considered and suitable provisions for them included during design, many programs would not require premature removal from the reactor.

3. Review each experiment failure when it occurs to determine if the failure could have been in any way predicted during the safeguards review. Many failures obviously cannot be

predicted no matter how diligently the reviewer does his job -- others, however, might have been avoided by a more thorough review. By continuing such reviews one can determine whether his safeguard review efforts should be increased, held constant, or possibly relaxed.

THE HEAVY WATER COMPONENTS TEST REACTOR AS A CAPSULE IRRADIATION SITE

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INTRODUCTION

The Heavy Water Components Test Reactor (HWCTR)[1] was built for the Atomic Energy Commission by the Du Pont Company to satisfy a need for fuel testing facilities in the AEC's Heavy Water Power Reactor Program. The reactor was designed to provide a realistic test environment for testing full size fuel assemblies.

A summary of test conditions available in the HWCTR is shown in Figures 1 and 2.

GENERAL DESCRIPTION

The HWCTR is housed in a containment building 70 ft in diameter and 125 ft high, shown in Figure 3. About half of the building is below grade and constructed of post-tensioned concrete. The upper half is constructed of carbon steel. The building is designed to contain an internal pressure of 24 psig. The containment building houses the reactor and coolant system, the charge-discharge machine, and the reactor instrumentation. The control room and auxiliary services and equipment are housed in other buildings. The containment building is ventilated and air conditioned, and the normal 5000 cfm air throughput exhausts from an 80 ft stack. Upon receipt of certain signals (high internal temperature or pressure, high activity) the building is automatically isolated by two pairs of butterfly valves in the inlet and outlet air ducts. Four banks of carbon absorber units with their own circulating systems are activated by the isolation signal. They trap fission products that may be released to the containment building following an accident and reduce the consequences of the maximum credible accident.

The reactor vessel, shown in Figure 4, is about 30 ft high and has a maximum inside diameter of 7 ft. The bottom head has 43 nozzles through which pass monitor pins for each fuel assembly and various instrument leads. The top head is bolted to the reactor and must be removed for charging fuel. The control and safety rod drive mechanisms are bolted to the head and lift with it when the head is removed; the delatched rods remain in the core.

The design pressure is 1500 psig at 315°C, and the maximum power is 70 MW. Currently the reactor power is 45 MW with an outlet temperature of 250°C at 1200 psig. The system is pressurized with helium, and the reactor outlet temperature at every operating pressure is at least 10°C subcooled. Film boiling is permitted at the hot spots. Heat is removed from the reactor by circulating D_2O at 10,000 gpm, first through the fuel, then through the bulk moderator, and then through two vertical U-tube steam generators. H_2O steam from the boilers carries off the energy to the atmosphere.

The core, shown in Figure 5, consists of a driver ring of 24 fuel tubes, each loaded with 1 kg of U-235 and a burnable boron-stainless steel target. The central test region can accommodate 12 elements up to 10 ft long. The heat removal capacity is about 2 MW per position.

A ring of 12 boron-stainless steel rods containing 1% natural boron provides reactivity control in the reactor. The six fast-shutdown rods, called safety rods, decrease reactivity by 6% $\Delta k/k$ in 1.5 seconds and 9% $\Delta k/k$ in 3 seconds. A backup system can inject a potassium tetraborate solution in the moderator to reduce the reactivity by 24% $\Delta k/k$ within 15 seconds after activation. The normal excess k is about 0.20 k at 20°C. A driver cycle is equivalent to about 9000 MWD in the reactor.

The HWCTR is probably the most heterogeneous power reactor now operating. Notice that the 24 driver fuel positions are not uniformly spaced, but are interrupted at intervals. The gap permits internal piping connections to be made to as many as six isolated test positions with their own coolant systems. So the driver ring cannot be well characterized by a typical lattice cell, as is generally done in calculating large reactors. Interposed between the driver fuel and the test fuel are the control rods spaced at regular intervals. Twelve test positions are on a seven-inch triangular pitch, and each can be loaded with different fuel. Loadings tested to date have varied from 15 lb/ft of natural uranium to 10 lb/ft of uranium enriched up to 4% in U-235. Both uranium metal and oxide in the form of rods and tubes have been tested. A central cluster of six control rods, generally removed from the reactor while operating, produces a large moderator flux peak or a strong central flux depression.

Flux Determination and Control

The HWCTR contains no direct flux measuring devices. The integral power distribution can be inferred from flow and temperature measurements appropriately corrected for gamma heating, but there is no direct method for obtaining the axial flux or power distribution. Consequently, we place reliance on our ability to predict operating conditions prior to insertion of the experiment and further to calculate the power distribution of experiments in the reactor.

Although most of the irradiations carried out in the HWCTR have not involved capsules in the ordinary sense, the same considerations regarding irradiation predictions apply.

Since all the experiments irradiated in the HWCTR are in-core, the interaction between the experiment and the reactor is important not only from the standpoint

of effects on the experiment itself, but also from the standpoint of the effects on the reactor. This extra degree of interaction, when coupled with the absence of flux measuring systems, further points out the necessity for a reliable analytical method for calculating performance.

We have developed an analytical model using two-dimensional multigroup diffusion theory with certain transport corrections to obtain the required predictions. We use the PDQ[2] series of codes to calculate the radial azimuthal distribution and the radial-axial distribution, generally in two energy groups. Figure 6 is a representation of the R,θ model in X,Y geometry. We have found that the flux ratios and reactivity estimates are fairly sensitive to the assigned fuel or irradiation volume. Best results are obtained when the discrete fuel, test, and control positions are assigned volumes in the model which are near their physical volumes.

Thermal macroscopic constants are obtained by flux-weighting material constants over a partial cell using the full cell flux shape calculated with a one velocity P-3 code. Although the driver fuel cannot be realistically characterized by a typical cell, an equivalent five-inch square pitch cell is satisfactory for obtaining the heterogeneous constants. Fast macroscopic constants are obtained for the homogenized partial cell using the MUFT[3] code. We have developed a D_2O library for MUFT that yields ages in D_2O as a function of H_2O contamination in good agreement with data developed by Wade of the Savannah River Laboratory.[4]

The radial-axial model in two dimensions is shown in Figure 7. Again the homogenized annular fuel rings are characteristic of the fuel volumes rather than the cell volumes. Constants are obtained by the same technique as described for the radial problem.

Two rather extensive sets of experiments were performed at SRL to provide a basis for calibrating the analytical approach. A mockup experiment described in DP-457,[5] provided early data up to a temperature of 50°C. A series of low power physics tests was run in the HWCTR to provide data up to the full operating temperature of 250°C. This data is the first 250°C critical data we know of for a very heterogeneous D_2O reactor.

Figure 8 is a summary comparison of calculations made by the method described above and data from the experiments.

With this approach, we have obtained satisfactory results in predicting performance of tests, i.e., in designing tests to operate at specified conditions. In addition, the day-to-day operation of the reactor is specified on the basis of these calculated flux shapes. With the present fuel design, the reactor flux is generally limited by the heat flux on some fuel element. It is the practice that the burnout heat flux must be at least 1.8 times the maximum pointwise heat flux including hot spot factors. The actual heat flux is obtained from the measured flow-ΔT product and the calculated axial flux distribution.

Factors Producing Flux Variations with Time

Most of the tests in the reactor are fuel tests and many of these are carried to rather high burnup. This factor, along with the change in axial flux distribution produced by the normal out motion of the banked control rods, requires that we consider in detail the exposure-dependent effects.

Our principle tool in this determination is an extension of the previously described two-dimensional approach that includes a multigroup depletion calculation. The TURBO[6] code provides the necessary combination of

features. With this code we calculate the axial power distribution for each fuel element as a function of exposure time and of control rod bank position. Our primary check on the validity of this approach has come from frequent gamma scans of irradiated fuel pieces taken from the reactor. The data show that the calculated power distributions are within ±5% for fuel exposed to 5000 MWD/ton.

The flux in a particular experiment may suffer incremental changes with time as the surrounding experiments are changed. This is complicated in the HWCTR by the fact that most of the experiments are fuel experiments. Hence, the composition of the reactor changes frequently. About every six to eight months a much larger change occurs when the driver fuel is replaced. Our experiments and calculations show that local variations as great as 10 to 30% are produced in the radial flux distribution.

Figure 9 presents a plot of the flux perturbation in a test position as a function of test loading and also shows the perturbation on an adjacent assembly.

One remaining factor introduces a measure of uncertainty in design of experiments. Since the lead time of most experiments is six to eight months, one must estimate the charge into which the experiments will be placed. This is a significant factor because of the interaction of the experiment with the reactor. In the intervening months between design and irradiation, the test charge may be changed because of unplanned termination of experiments by failure or for other causes.

To give you an idea of a typical problem, Figure 10 presents the estimated power variation in a typical long term test assembly.

With such variations it is desirable to obtain some measure of flux control in an individual position. Three methods have been investigated for the HWCTR.

1. Neutron absorbing housing tubes
2. H_2O-D_2O in a loop
3. BF_3 gas control

Neutron absorbing housing tubes can provide a control range of about 15% without introducing a large reactivity defect.

The normal housing tube used in the experiments is Zircaloy. Substitution of a stainless steel housing tube for the Zircaloy housing tube will accomplish the desired result. The poison housing tube can be replaced at intervals during the test period, or a burnable poison can be used. Figure 11 indicates the effectiveness of an absorbing tube arrangement.

In the isolated coolant loops of the HWCTR, the D_2O can be attenuated by H_2O. We have just completed a set of experiments in which we measured the effectiveness of light water for flux control. Figure 12 shows the flux depression factor at 20°C, as measured in a typical test lattice in the SRL exponential experiment.

We have also considered using a BF_3 gas control system in the fuel assemblies in the bayonets. The range of control afforded by such a system is as much as a factor of two in the HWCTR. But the major reservation is the hazard from the reactivity accident introduced by loss of the gas pressure.

Up to the present time, none of these techniques has been used, but all are considered promising.

Reactor Characteristics as They Affect Capsule Experiments

We have discussed in detail the interaction between the reactor and the experiment in the HWCTR. Since the experiments are a large fraction of the critical reactor, this interaction is complex and very important in the design of experiments.

We have not irradiated capsule experiments as such in the HWCTR. But the considerations are common to the type of experiments we have performed. We have under consideration at the moment a large capsule experiment. The experiment involves UC rods encapsulated in a "Hastelloy" can filled with NaK. In addition, we have irradiated in the Savannah River production reactors significant numbers of lead and NaK insulated capsules in materials tests for the fuel development program.

The HWCTR has been on-line for about 60% of the time since power startup a year ago. Much of the downtime can be attributed directly to the problems of starting up a new facility. We believe that 80% is a practicable on-line estimate for future operation.

The program for the HWCTR is currently planned and executed by the Du Pont Company under Contract AT(07-2)-1 with the AEC, administered by the Savannah River Operations Office. A full test program requiring all the test space in the reactor is being carried out by the Savannah River Laboratory of Du Pont. Requests for space in the reactor should be addressed to the AEC.

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FIGURE 1

SUMMARY OF TEST CAPABILITIES IN HWCTR

Thermal neutron flux in test fuel:

Maximum flux in fuel = 1.5×10^{14} $n/cm^2/sec$

Uniform flux over 3 ft length with a max/avg of 1.1 = 1.3×10^{14} $n/cm^2/sec$

Maximum to average flux ratio over full 10 ft length = 1.7

Specific power in natural U fuel (10 lb/ft) = 35 MW/ton

Gamma field in a vacant test position = 10^9 R/hr

Heat removal capacity:

Reactor = 70 MW

Fuel position = ~2 MW

Maximum coolant outlet temperature = 250°C

FIGURE 2

SAMPLE OF EXISTING TEST CONDITIONS

September 1, 1963

Reactor Power		40 MW
Reactor Pressure		1200 psig
Coolant Inlet Temperature		235°C
Coolant Outlet Temperature		250°C
Typical Assembly:		
Enriched UO_2 single tube:	OD =	2.060"
	ID =	1.526"
Enrichment		1.5% U-235
Fabricated by vibratory compaction followed by swaging		
Assembly power		0.73 MW
Specific power		67 MW/T of U
Heat flux		327,000 pcu/(hr)(ft^2)
Surface temperature		286°C
$\int kdt$		27 watts/cm
Estimated central temperature		1300°C
Max exposure:	Attained =	9,000 MWD/T
	Goal =	23,000 MWD/T

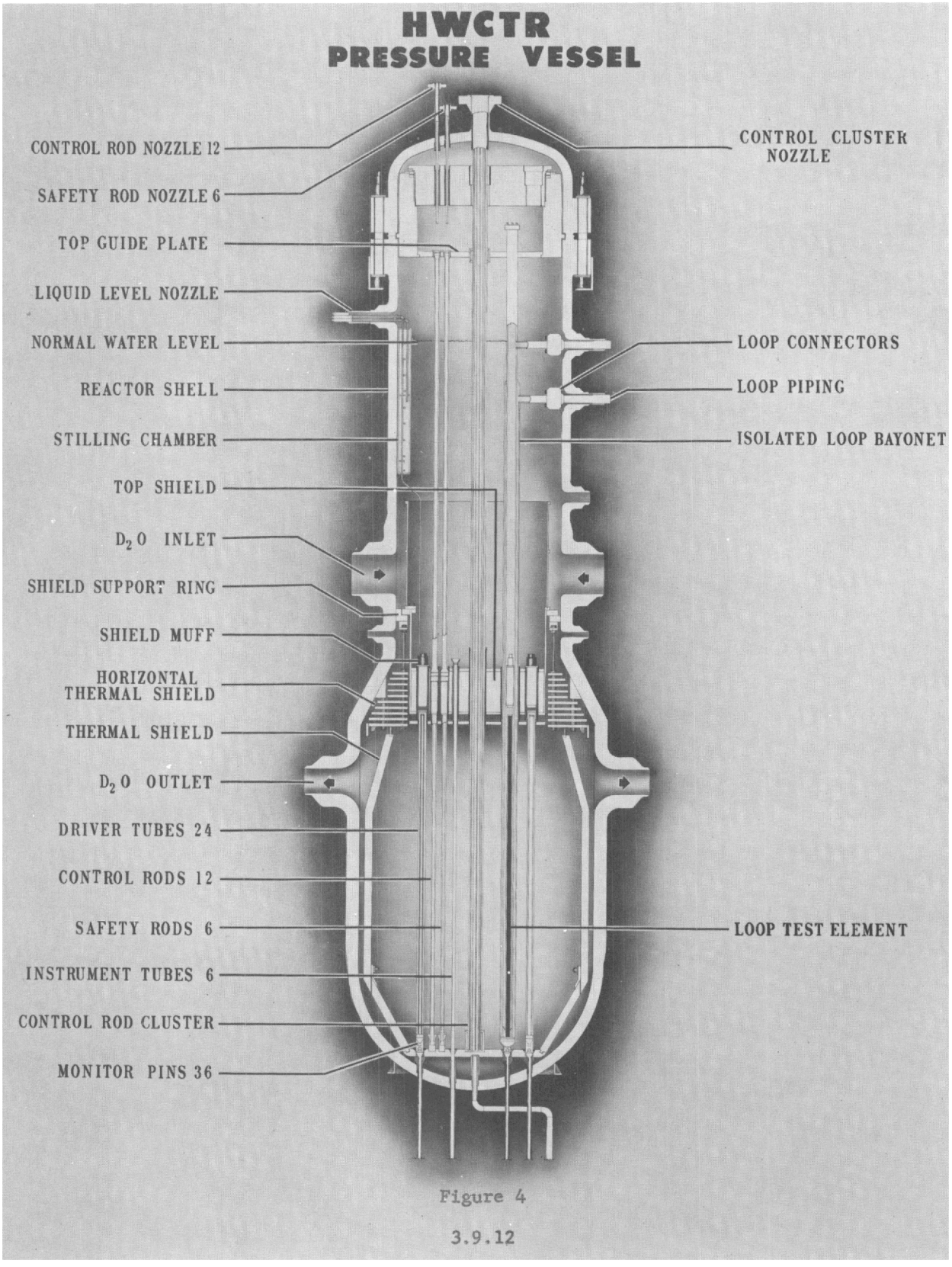

HWCTR
PRESSURE VESSEL
CONTROL ROD NOZZLE 12
SAFETY ROD NOZZLE 6
TOP GUIDE PLATE
LIQUID LEVEL NOZZLE
NORMAL WATER LEVEL
REACTOR SHELL
STILLING CHAMBER
TOP SHIELD
D_2O INLET
SHIELD SUPPORT RING
SHIELD MUFF
HORIZONTAL THERMAL SHIELD
THERMAL SHIELD
D_2O OUTLET
DRIVER TUBES 24
CONTROL RODS 12
SAFETY RODS 6
INSTRUMENT TUBES 6
CONTROL ROD CLUSTER
MONITOR PINS 36
CONTROL CLUSTER NOZZLE
LOOP CONNECTORS
LOOP PIPING
ISOLATED LOOP BAYONET
LOOP TEST ELEMENT

Figure 4

Figure 5

HWCTR

HEAVY WATER COMPONENTS TEST REACTOR

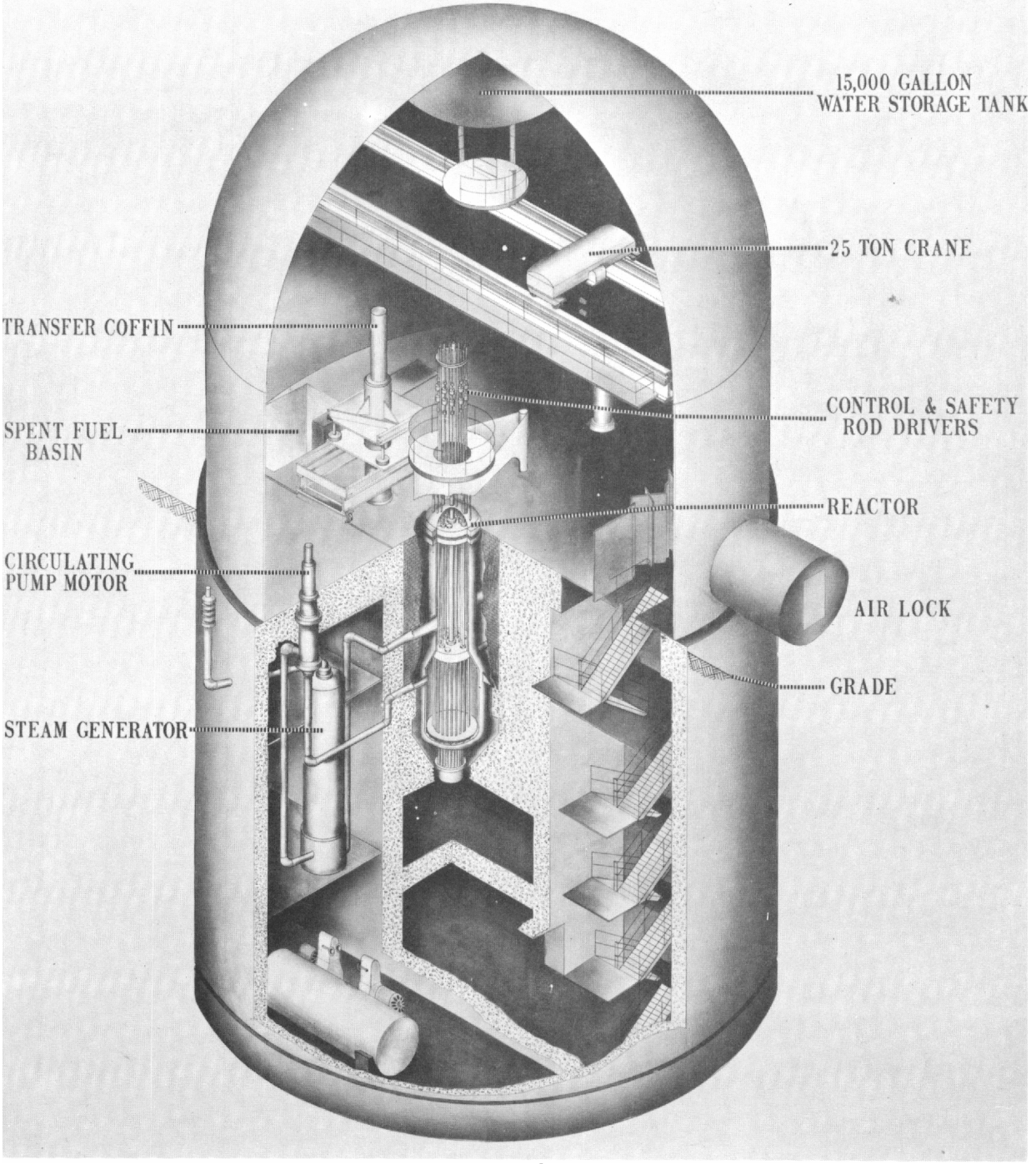
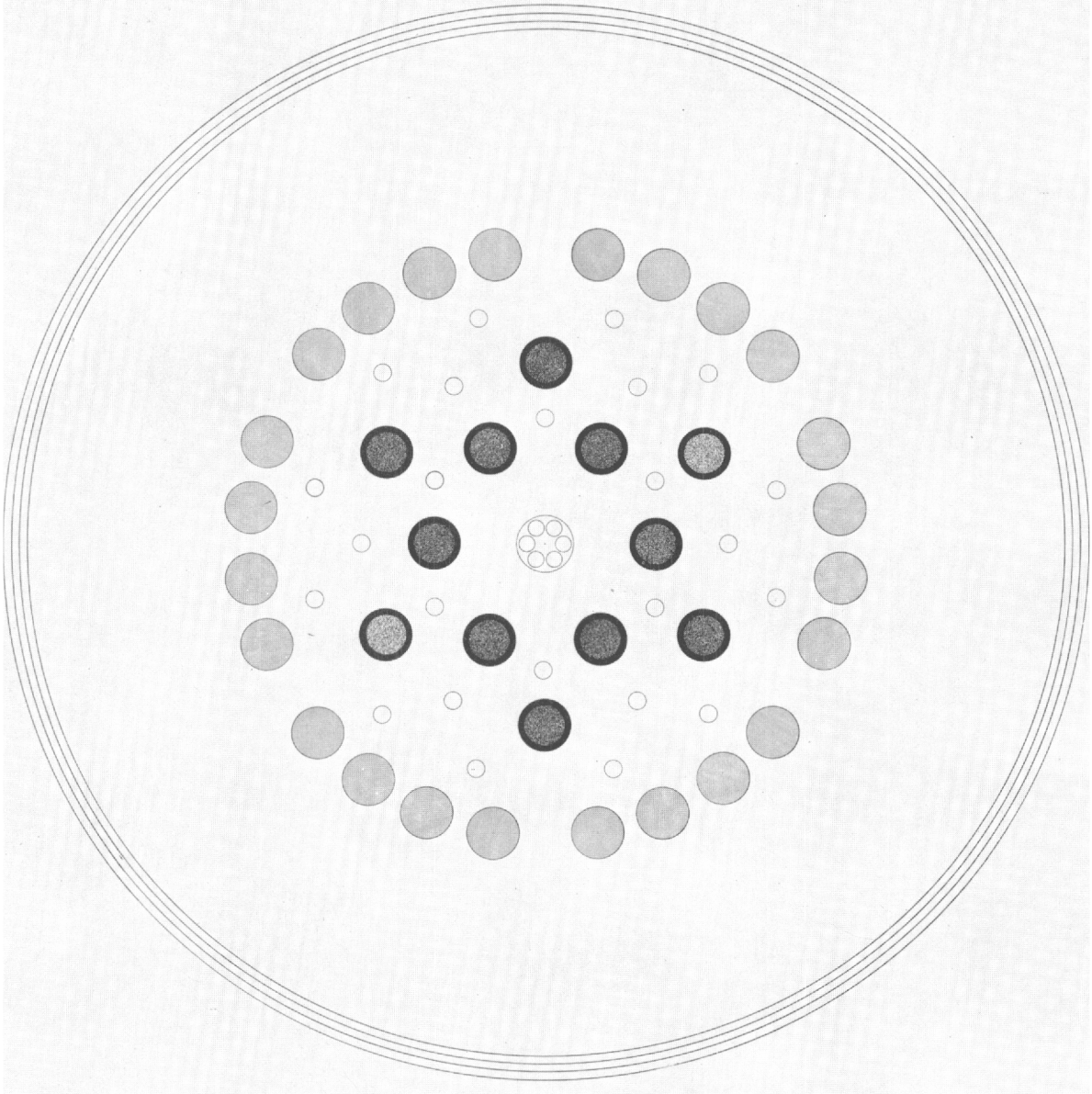


Figure 5

HWCTR
CORE CONFIGURATION



24 DRIVER FUEL
10 TEST FUEL
2 LOOP TEST FUEL
6 INSTRUMENT RODS
17 BLACK FULL RODS
1 GRAY FULL RODS
6 SAFETY RODS

Fig. 6

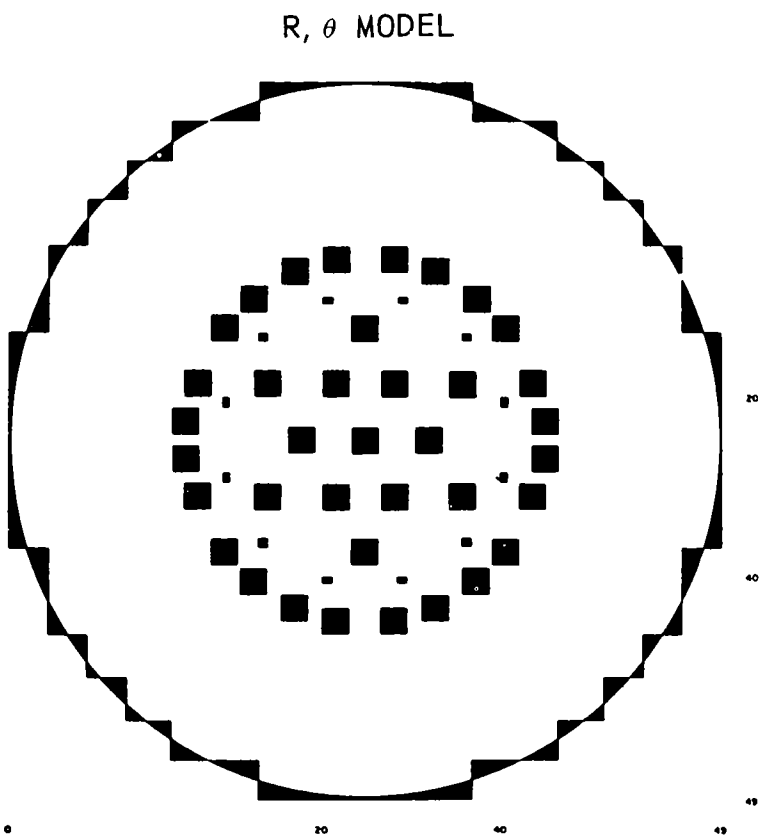


Fig. 7

RADIAL - AXIAL MODEL

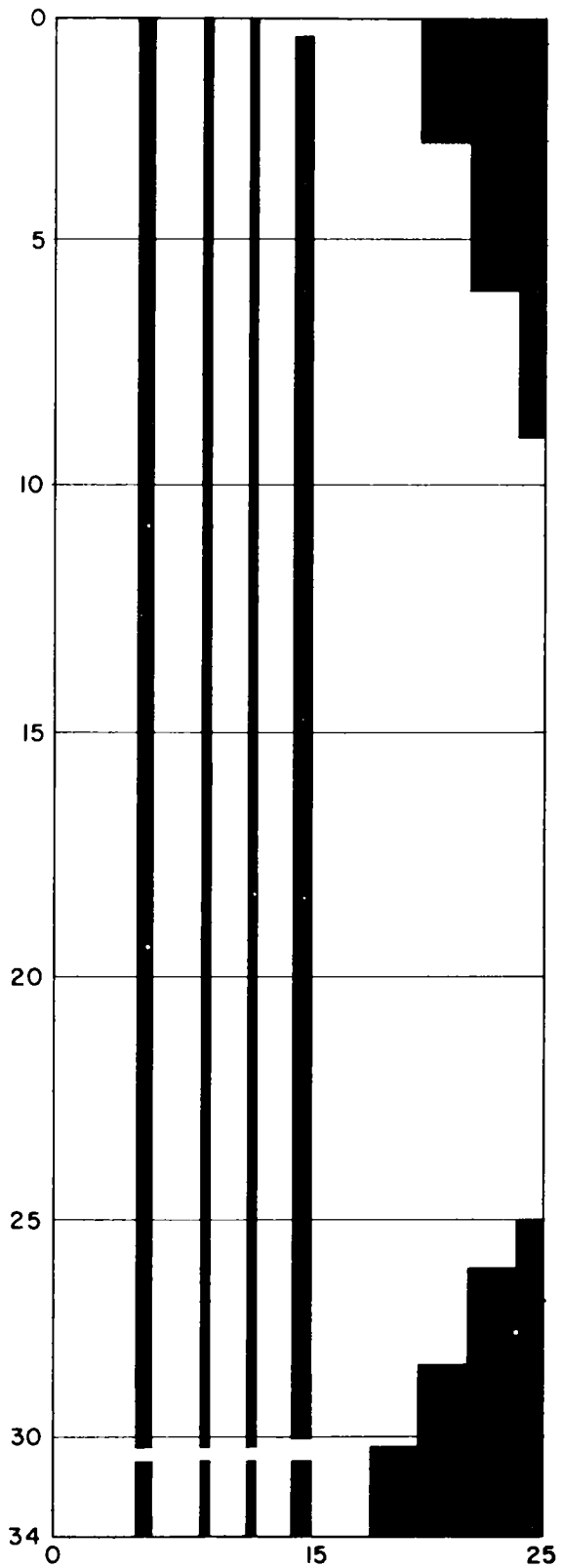


Fig. 8

COMPARISON OF MEASURED AND CALCULATED FLUX PARAMETERS IN HWCTR

At 20° C, Ring Rods Banked at 0.793 Insertion

	Predicted	Observed
Azimuthal max/avg	1.10	1.08
Axial max/avg	2.10	2.08
$\frac{\bar{\phi} \text{ Test}}{\bar{\phi} \text{ Driver}}$	1.65	1.72

At 240° C, Ring Rods Banked at 0.620 Insertion

	Predicted	Observed
Azimuthal max/avg	1.08	1.05
Axial max/avg	2.10	2.15
$\frac{\bar{\phi} \text{ Test}}{\bar{\phi} \text{ Driver}}$	1.70	1.75

Fig. 9

FLUX VARIATIONS WITH TEST LOADING

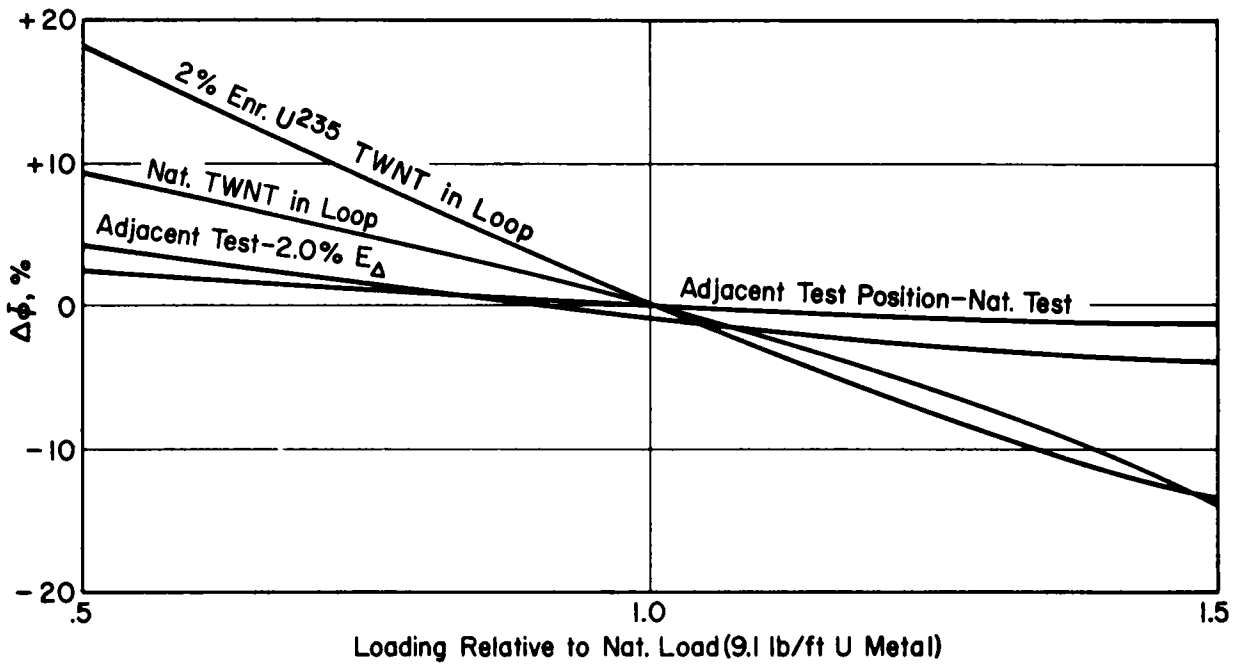


FIG. 10

FLUX AND POWER VARIATIONS IN HWCTR
No Control

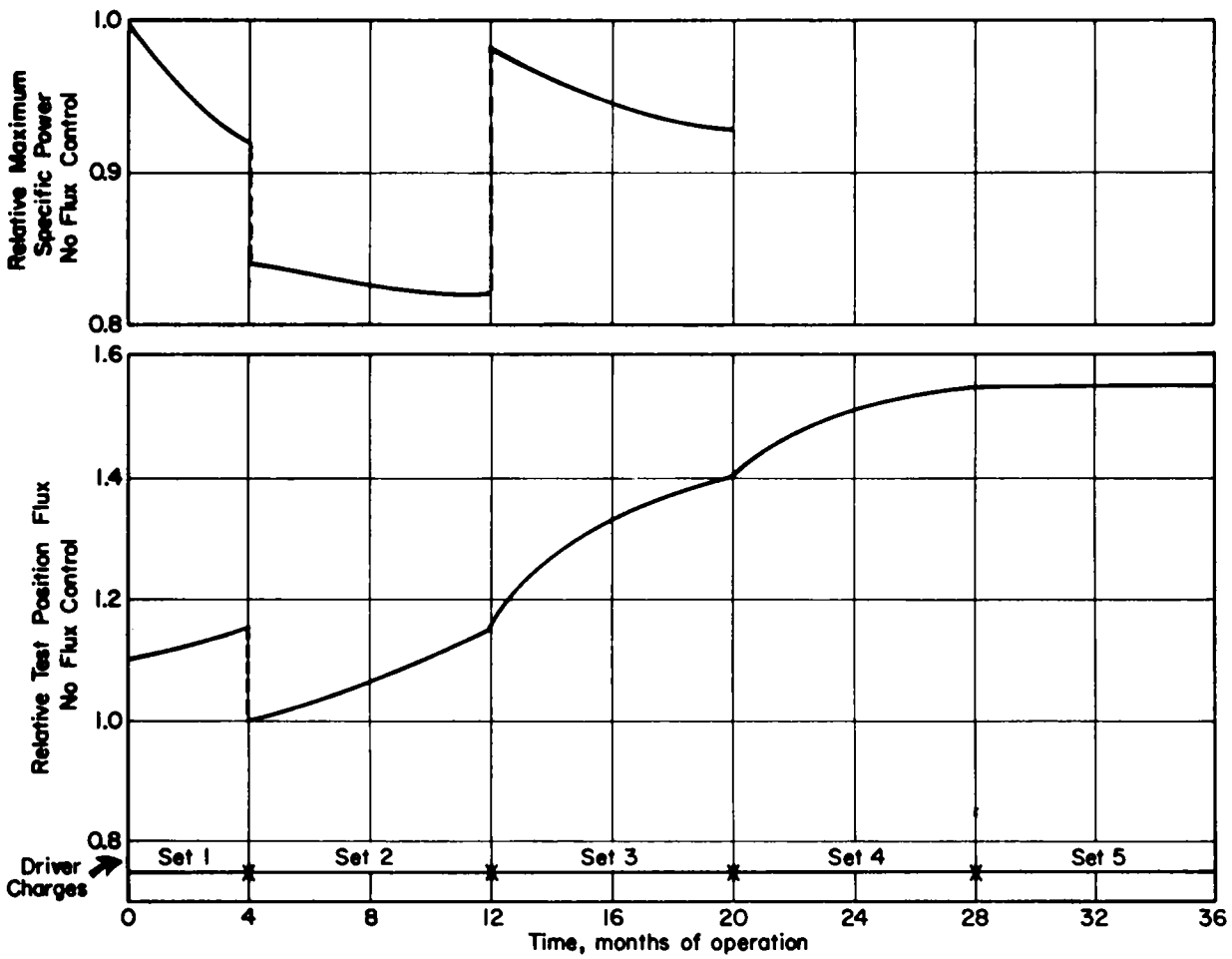


Fig. 11

ABSORBING HOUSING TUBES AS A FLUX CONTROL DEVICE IN HWCTR

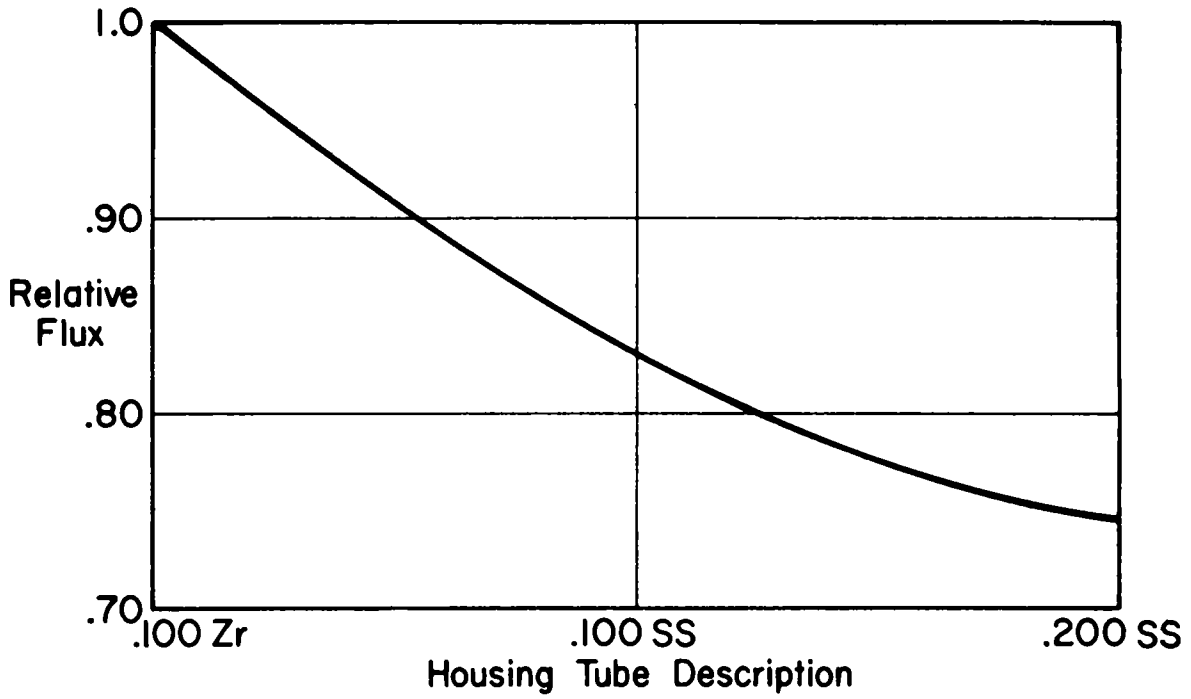
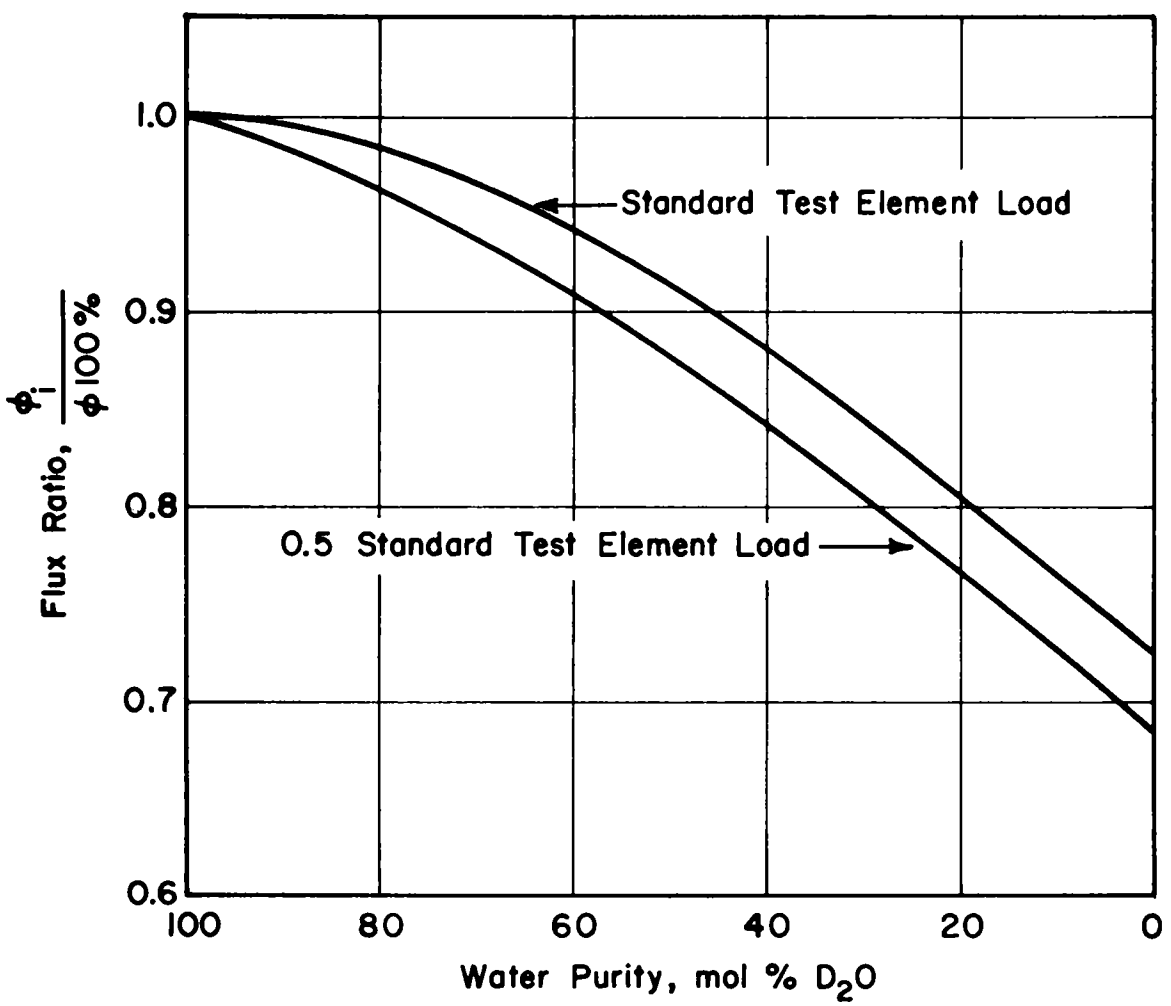


FIG. 12

EFFECTIVENESS OF H_2O AS A FLUX CONTROL TECHNIQUE IN HWCTR



IV. INSTRUMENTATION AND TEMPERATURE CONTROL

A.E.R.E. experience in development of capsule components

by

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&

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A. E. R. E. experience in development of capsule components

1. Introduction

Most capsule irradiation experiments performed at A. E. R. E. involve the simultaneous irradiation of a number of capsules, the number varying from 4-12. In many cases, failure of one capsule could lead to hazardous conditions and thus the withdrawal from the reactor not only of the failure but also those other capsules being concurrently irradiated in the same experiment. It has therefore been essential to strive continuously for the highest possible standards of manufacture and assembly of capsule components and to rigorously eliminate from capsule designs any features which, although perhaps desirable from a design aspect, have not been amply proven.

The majority of development effort up to the present time has been in four fields:

(a) Heaters

(b) Thermocouples

(c) Ceramic/metal seals

(d) Sodium resistant and high temperature brazes

A considerable amount of work has also been done in a miscellany of problems, including the use of recrystallised alumina for semi-encapsulation and as heater formers and sleeves, the use of sprayed or plated metal coatings in heat transfer systems, the compatability of components with each other and with specimen materials, etc.

2. Heater development

Until the research reactors Dido and Pluto became available, heated irradiation experiments did not provide many problems in the heater field, since experimental requirements on specimen size and heat output were low and the reactor fluxes available were only of the order of $1\text{-}2 \times 10^{12}$ n/cm^2/sec (thermal) and 10^{11} n/cm^2/sec fast. Gamma heating effects were so small that they were neglected.

The higher fluxes in Dido and Pluto, however, allowed the heat ratings of fissile materials to be increased by up to an order of magnitude. Similarly, gamma heating rates increased until, in the fast flux facilities of these reactors, appreciable heat outputs were obtained from non-fissile materials either as specimens or structural components of capsules or rigs. Thus the development of control heaters of high power combined with low mass and, if possible, high thermal conductivity was necessary.

Solutions to these problems varied according to the design philosophy used in differing types of rig. Thus, in non-fissile experiments, where temperature gradients in and around the specimens were small, rigs could be so designed that removal and replacement of the specimens could be effected without damage to the rig. Somewhat lower standards of reliability could therefore be accepted, since the failure of a rig component did not invalidate an experiment. In the case of fissile experiments, however, requirements arising from double containment and high thermal gradients implied non-reusable rigs - thus a failure of a component could lead to the premature termination of an entire experiment and standards of integrity and reliability were therefore set high.

2.1. Early Heaters

The earliest types of heater used in fissile and non-fissile experiments are shown in Figs. 1 and 2 respectively. Fig. 1 shows a design in which a finned former of zirconium, which combines a low neutron capture cross-section with reasonable thermal conductivity, was combined with Nichrome wire coils located and insulated by pyrophyllite inserts. These inserts were retained by a zirconium tube which was a sliding fit over the former and by pyrophyllite end plates. This heater developed 1 kW at temperatures up to 800°C and gave satisfactory service over periods of up to 6 months. It was, however, expensive to manufacture, of considerable mass ($\sim$300 gms) and offered only restricted space for the sodium filled specimen capsules.

Fig. 2 shows a design which has persisted until the present day. It consists of Nichrome wire wound on a stainless steel former, insulation being effected by a coating of silicious cement, layers being applied and baked before and after winding. This design is simple and cheap and has also given satisfactory service over periods of up to 13 months when developing $\sim$0.5 kW at temperatures up to 650°C, and up to 8 months at 900°C. The normal full power rating of 1 kW was rarely used, since the heater was operated only when the reactor was also operating and gave ballast heating only. Heater failures have been undesirably frequent with this design but, in fact, have been proven to be caused by overheating of the heater leads which were originally of copper clad copper cored MgO insulated cable. Replacement of the copper cores with a manganese-nickel alloy has largely eliminated these failures.

2.2. Heater development for fissile experiments

Fig. 3 shows immersion heaters which were developed for use in fissile irradiation capsules filled with sodium. An MgO insulated Nichrome cored stainless steel sheathed cable was designed at A. E. R. E. based on a similar cable which was commercially available. The heaters illustrated are the 1 ohm and 1.7 ohm sizes, these having been found to be the most convenient sizes from the aspect of capsule space. Generally, the 1 ohm heater is used at a rating of 1.5 kW, while the higher resistance heater operates at up to 3.5 kW, in both cases at sheath temperatures of 800 C or less. The ultimate rating of these heaters is, however, not known since no failures have occurred, but a

rating of 5 kW has been achieved with the 1 ohm heater at a sheath temperature of 800°C. Initial development work at A. E. R. E. led to the development of butt-swaged cable in which the major length of the heater is swaged to about half the diameter of the ends. This effectively obviates end connection troubles whilst minimizing the mass of the heater element. Production of this cable, or heaters made from it, is now commercially undertaken.

For irradiation of fissile materials in capsules in which sodium was not used, heaters of the types shown in Figs. 4 and 5 have been used. In each case the heating element is the mineral insulated cable used for immersion heaters, but matrices of aluminium, zirconium or nickel have been used depending on temperature and heat transfer requirement. Casting aluminium around a preformed heater spiral produces a most effective heater with maximum conductivity and minimum cross-section. An upper temperature limit of 600°C is imposed by the matrix, however, so that for higher temperatures a grooved zirconium former into which the cable is laid is used. Retention of the cable is by a zirconium sleeve which is swaged after assembly to produce a good bond. Nickel has been used where its high thermal conductivity is sufficiently valuable to offset its higher capture cross-section.

2.3 Heaters for B. R. 2 experiments

The need for irradiations to high fast neutron doses has led A. E. R. E. to make use of the experimental facilities available in the B. R. 2 reactor at Mol, Belgium. The facility size here is only 3/4 inch diameter and gamma heating rates of the order of 20 watts/gm exist, so that considerable design problems were required to be overcome. The rig heaters developed for non-fissile experiments in this reactor are shown in Fig. 6. Initially, thermocoax heater cable of .040" diameter having an Inconel sheath, MgO insulation and a nickel/chrome alloy core was used and joined with a watertight and insulated joint to a similar cable of 0.060" diameter. Considerable development work was required to perfect this joint, the techniques of welding and brazing on so small a scale being an entirely new requirement.

2.4 Latest developments in heater design

2.4.1 Tubular Heaters

Certain difficulties are involved when mineral insulated cable heaters on solid metal formers operate at high temperatures. These are:-

1. The winding of the cable to provide the intimate contact required throughout the cable length.

2. The rather large radial thickness which results, reducing specimen volume and heater response time.

3. The core to sheath temperature difference which always exists: the degree dependent on the cable heat rating, which reduces the maximum operating temperature which can be achieved with any given material.

It is therefore necessary to consider the features desirable in a heater required for this duty.

1. Radial thickness to be a minimum for maximum specimen capacity, thus reducing thermal resistance, and keeping control heat output to a minimum.
2. Radial thermal resistance to be a minimum to assist in heat dissipation from fissile specimens.
3. The heater to form a containment for the specimen.
4. Heater mass to be a minimum to give rapid response and accurate specimen temperature control.
5. Heating element temperature to be as close to specimen surface temperature as possible for maximum service life.
6. The electrical terminations to be at one end.
7. The heater to be of robust design and capable of maintaining full rated power for long periods.
8. The electrical resistance to be of a suitable value to give the correct voltage/amperage ratio for the power required.
9. Suitable for easy manufacture from a variety of materials according to service temperatures required.

2.4.1.1 Description of Tubular Heater

The design as shown in Fig. 7 has been developed with above points in mind. It consists of essentially three concentric tubes separated by a mineral insulant, the inner tube forming the containment; the middle tube acting as the heating element and the outer tube holding the assembly together and giving a stable dimension for gas temperature control purposes. The element tube being slotted axially to produce the elongated path necessary to give a suitable electrical resistance and bring the terminations to one end. The thickness of the terminations and insulation surrounding them is increased by stepping in the element to core tubes at the end; thus ensuring that the terminations run cool.

2.4.1.2 Fabrication Technique

The method of construction evolved enables the element to be accurately centralised between the core and sheath whilst the insulation is vibrated in using a Turbo Vibrator. When using a grained magnesium oxide a starting density of 65% of the theoretical maximum can be achieved by this means. The assembly is then subjected to hydraulic pressure of 100,000 p.s.i. which raises the density to 80-85% of the maximum.

2.4.1.3 Test Results to date

A heater 1 1/8" diameter x $3\frac{1}{2}$" long manufactured in this manner from 18/8/1 stainless steel using a MgO insulant has been operating for 1200 hours at 1000°C and producing 3 Kwatts

From the limited experience so far gained with this heater it would seem that for general purposes this design would have a future in the irradiation programme. A serious limitation of the present design is the relatively low density and hence low thermal conductivity of the magnesia insulant but it is hoped that using a new development much higher densities may be achieved with resulting improvements in the thermal conductivity.

2.4.2 Graphite heaters

Work is being done on the design of heaters machined from high density graphite. The manufacture of such a heater is, in itself, not a severe problem but its continued operation at very high temperatures ($>$ 1200°C) involve compatibility problems which are presently being investigated.

2.4.3 Refractory metal heaters

These have not as yet been used in the A.E.R.E. research reactors, but laboratory work is in progress to determine how capsules may be designed to incorporate these. The main problem would appear to be the high degree of purity required of the gaseous atmosphere surrounding the heater, since even small amounts of impurity cause gross embrittlement.

2.4.4 Development of mineral insulated heater cable

Although requirements for high temperature operation are tending to cause development of heater systems other than those employing cable, other trends toward longer period irradiations below 800°C still make it necessary to obtain the greatest possible reliability from cable. To this end considerable work is in progress to determine:

(a) What sheath materials will best withstand service conditions for periods up to 2 years.

(b) How grain growth of nickel/chromium alloys may be restricted.

(c) What function impurities play in the thermal and electrical conductivities of MgO and Al_2O_3.

(d) The causes of corrosion of sheath and core when in contact with MgO for long periods at elevated temperatures.

(e) What the core to sheath temperature gradient is at various cable ratings and sheath temperatures.

Work on (a), (b) and (d) is proceeding but has as yet enabled no conclusions to be reached. Work on (c) and (e) however, both within the U.K.A.E.A. and by industry has produced a good deal of information, some of it contradictory. For instance, it has been shown that SiO_2 and Cr_2O_3 impurities in MgO decrease its resistivity at high temperatures. It is, however, the firm contention of at least one commercial manufacturer of heater cable that SiO_2 has no effect. Accepting that impurities do have a generally deleterious effect on thermal conductivity we are investigating varying sources of supply of high purity MgO. Since it has been conclusively shown that fused material is far better than that obtained by calcining, all heater cable insulants are of the former type.

There is evidence that, in the case of heater cable operating at a sheath temperature of 1200°C, melting of the nickel/chrome has occurred. At the same sheath temperature, on cable of 1 mm diameter and a rating of $\sim$ 12 watts/cm length there is experimental evidence that a gradient of only $\sim$ 60°C exists between core and sheath.

3. Thermocouple development

3.1 Chrome/Alumel thermocouples

The thermocouple most commonly used in irradiation experiments, and also in civil power reactors, is Chromel/Alumel MgO insulated and stainless steel sheathed. Doubts as to its irradiation stability to doses of $\sim 10^{21}$ n.v.t. (thermal) have been largely resolved by work performed at D.E.R.E., while the effects upon its calibration of varying thermal gradients and thermal cycling are being closely investigated in other parts of the U.K.A.E.A. Generally speaking it would appear that the calibration of this thermocouple may be largely relied upon to within a 1% accuracy, unless chemical attack on the thermocouple element by diffusion of reagents through the sheath occurs.

A thermocouple manual[1] has been prepared, its purpose being to collate information on the techniques of manufacturing and using Chromel/Alumel thermocouple. Additions to this will be made as more knowledge becomes available.

A considerable amount of time and effort is devoted to testing thermocouples before installation into capsules or rigs, more particularly where cables of 0·060" diameter or less are used. Capped and isolated hot junctions have now virtually eliminated solid junctions and have produced a definite decrease in the number of open-circuit failures. Hot junctions are normally made by argon-arc welding, but a process for plasma arc welding both junctions and caps has been developed within the U.K.A.E.A. and gives a much improved product.

3.2 High-temperature thermocouples

Above 1000°C, Chromel/Alumel is an unsatisfactory material and effort has therefore been devoted at the majority of establishments within the U.K.A.E.A. to the selection of suitable alternatives, with the result that W:5%Re vs W:26%Re is now most commonly used. BeO or Al_2O_3 have been used as insulants because of their improved electrical properties at temperatures around 1500°C. It has been found, however, that the advantages claimed for BeO in this respect are largely theoretical, manufacturing standards in respect of purity allowing considerable quantities of impurities to exist thus decreasing the insulation resistance. Both cost and lack of ductility are also disadvantages with these thermocouples; while very little can be done about the former except by using alternative materials, it may be that improved ductility could be obtained by alloying.

A thermocouple working party has been established to examine such problems as these and, in addition, the effects of irradiation on high temperature thermocouple materials. Various systems are being examined, among these being W:5%Re vs W:26%Re, Pt:1%Mo vs Pt:5%Mo, Pt vs Pt13%Rh. Irradiation experiments are expected to start in the latter part of this year and results should be available from the middle of 1964.

The problem of sheath material, its mechanical properties and compatibility with various fissile and non-fissile matrices is being examined. There is evidence that the use of double refractory metal sheaths will result in decreased overall penetration rates by the matrix.

4. Miscellaneous components

4.1 Ceramic/metal seals

These are obtained commercially and consist essentially of an Al_2O_3 tube with nickel alloy sleeves brazed at either end. Although their use is not widespread, sufficient experience has been obtained to have reasonable confidence in their ability to operate at up to 200°C for periods of 6 months and to dose of 10^{21} n.v.t. (thermal).

The most common use is for the passage of heater leads through capsule end-caps, but twin-bore seals have been developed which enable the normally open end of a thermocouple cable to be closed.

4.2 Heat transfer sleeves

In some capsule systems the use of liquid metal as the primary heat transfer medium is not desirable. Gaseous transfer annuli become unacceptably small at high specimen surface heat ratings and sleeves of metals with good thermal conductivity are becoming increasingly used.

Sprayed copper has been used in the past, but its dimensional changes when operated at temperatures between 600°C and 800°C and its low effective conductivity (due to oxide and void inclusions) limit its use. Electro-plated nickel and copper with a fine grain and high density are being investigated and show considerable promise. Plasma-arc spraying of these metals is also being investigated, since this process should eliminate oxide inclusions in the sprayed coatings.

4.3 Joining components

Commercial brazing alloys of the nickel/manganese/palladium type have been used and found satisfactory at operating temperatures of up to 800°C and to doses of 2×10^{21} n.v.t. (thermal). Sodium resistant brazes have not, as yet, been used in irradiation capsules but have been fairly extensively used in laboratory experiments. It has been found that a vital factor in the successful use of these materials is the length of the brazing cycle; considerable penetration of the materials being joined with consequent embrittlement, occurs if this is not minimized.

The problem of the sealing of thermocouple cable with a sheath of the order of 0·005" thick into capsule of 0·020-0·040" wall has been largely overcome by the use of ferrules. These are so designed that the portion surrounding the thermocouple cable is a close fit on it and is of comparable thickness to the cable sheath. A projection weld forms the junction. Chemical cleanliness of both ferrule and sheath is essential to avoid imperfect and unrepairable welds.

Argon-arc welding is still the most common method of joining components, the generality of which are of stainless steel. Electron beam welding is, however, becoming more common particularly as increasing experimental temperatures demand the use of refractory metals.

References:

TRG 287R

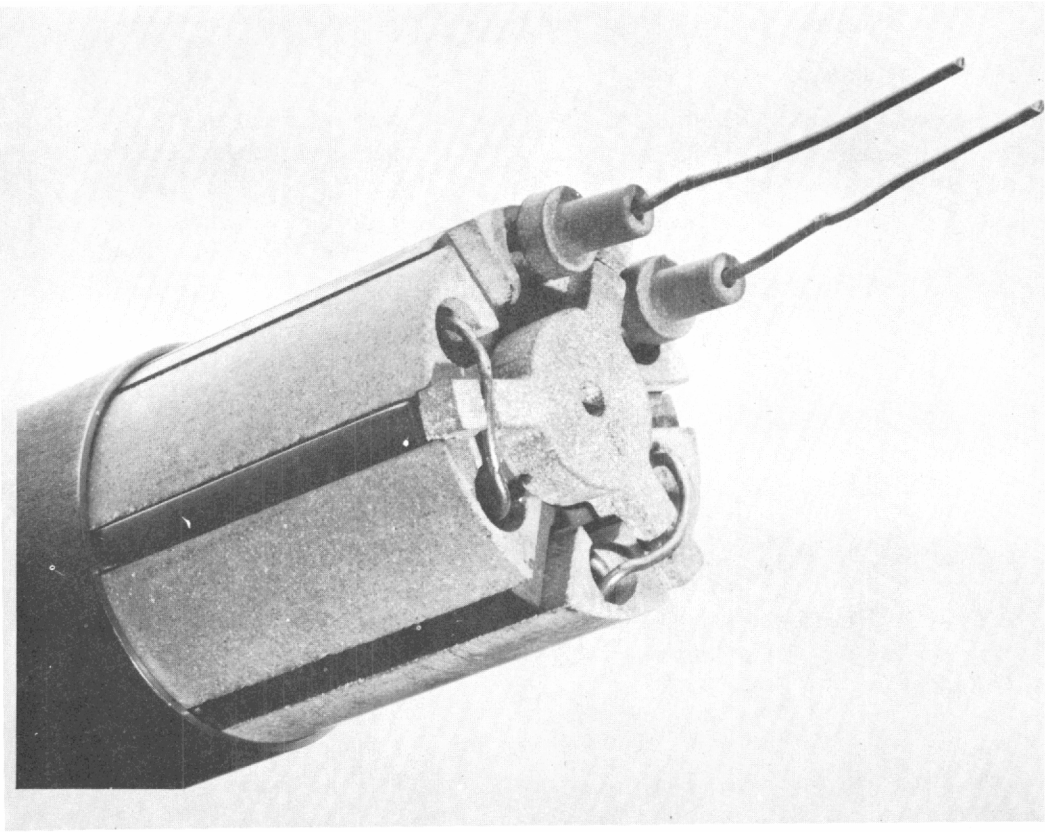


Fig. 1. Heater used on capsule irradiations, c. 1957.

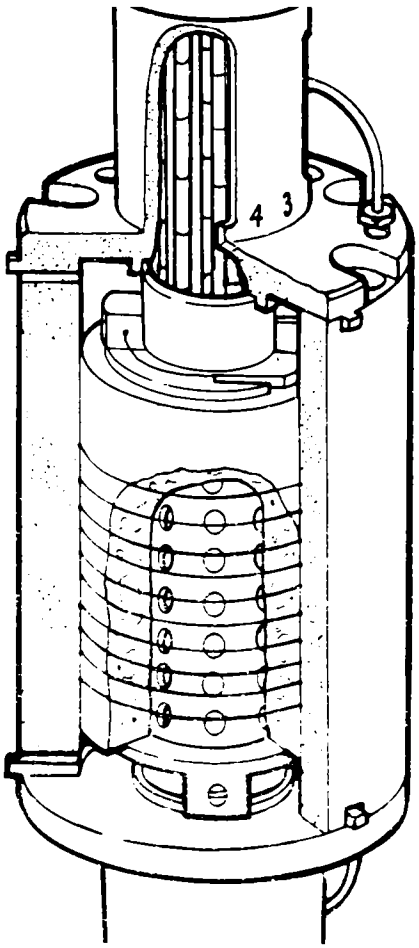
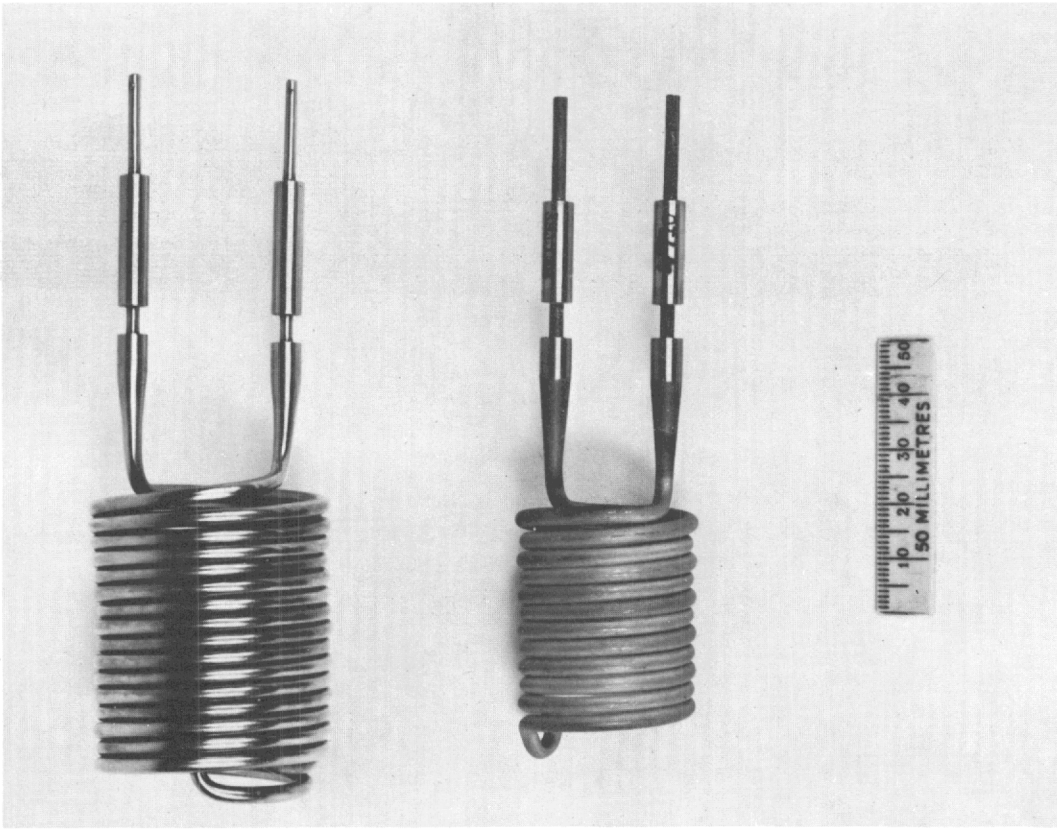


Fig. 2. Heater used on graphite irradiation experiments.



1.7 ohm heater 1.0 ohm heater

Fig. 3. Immersion heaters for sodium-filled capsules.

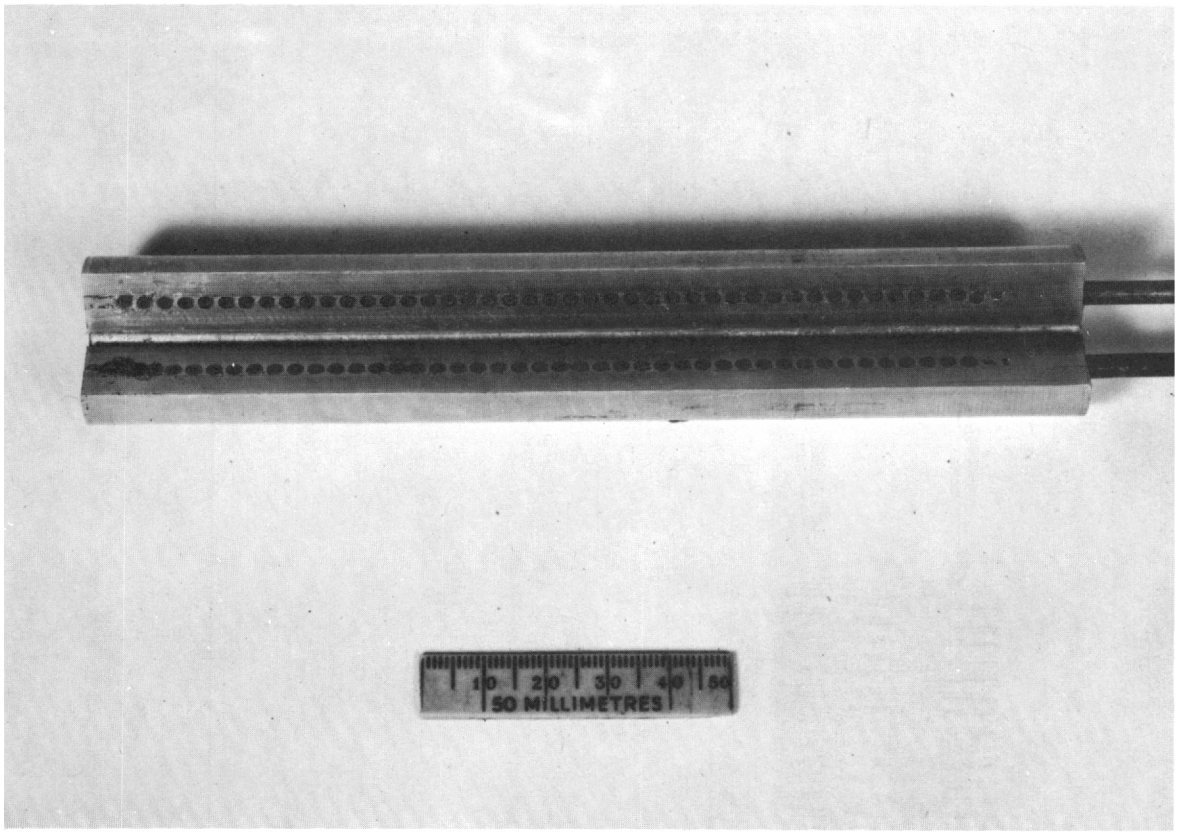


Fig. 4. 3 kW heater cast into aluminium matrix

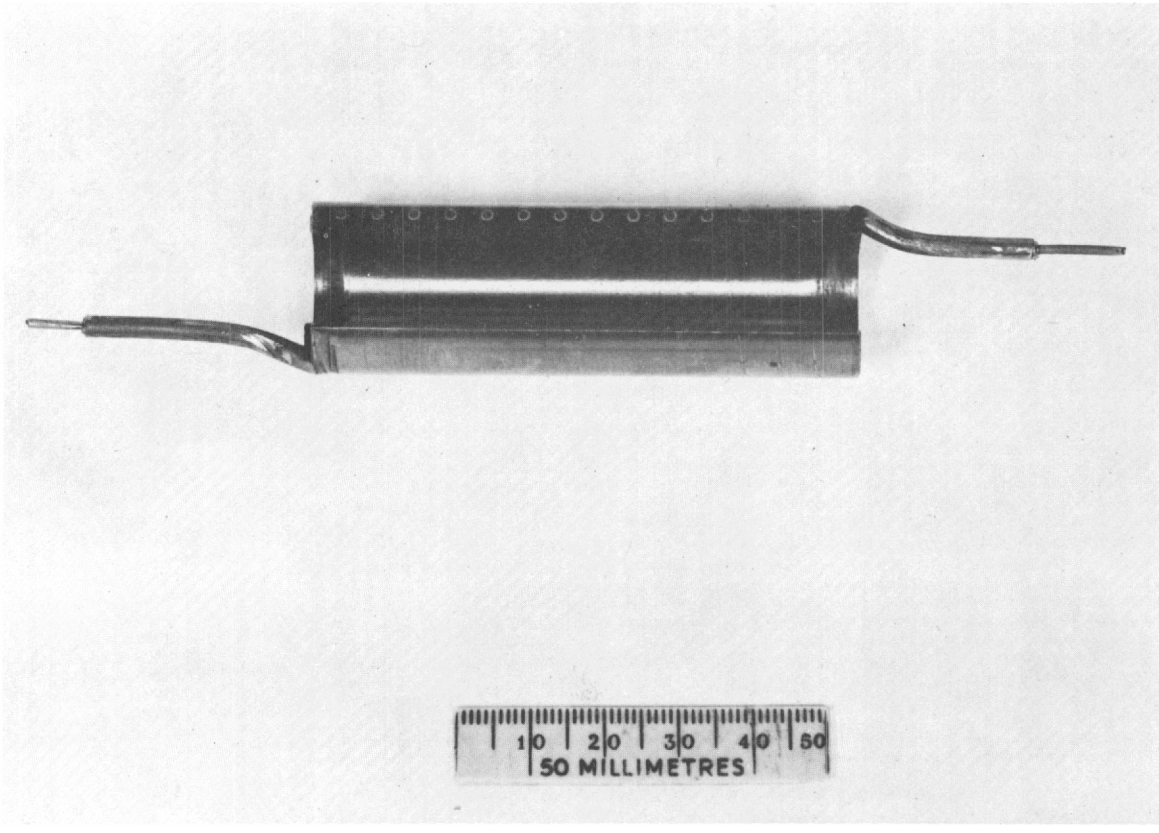


Fig. 5. 3 kW heater embedded into zirconium matrix.

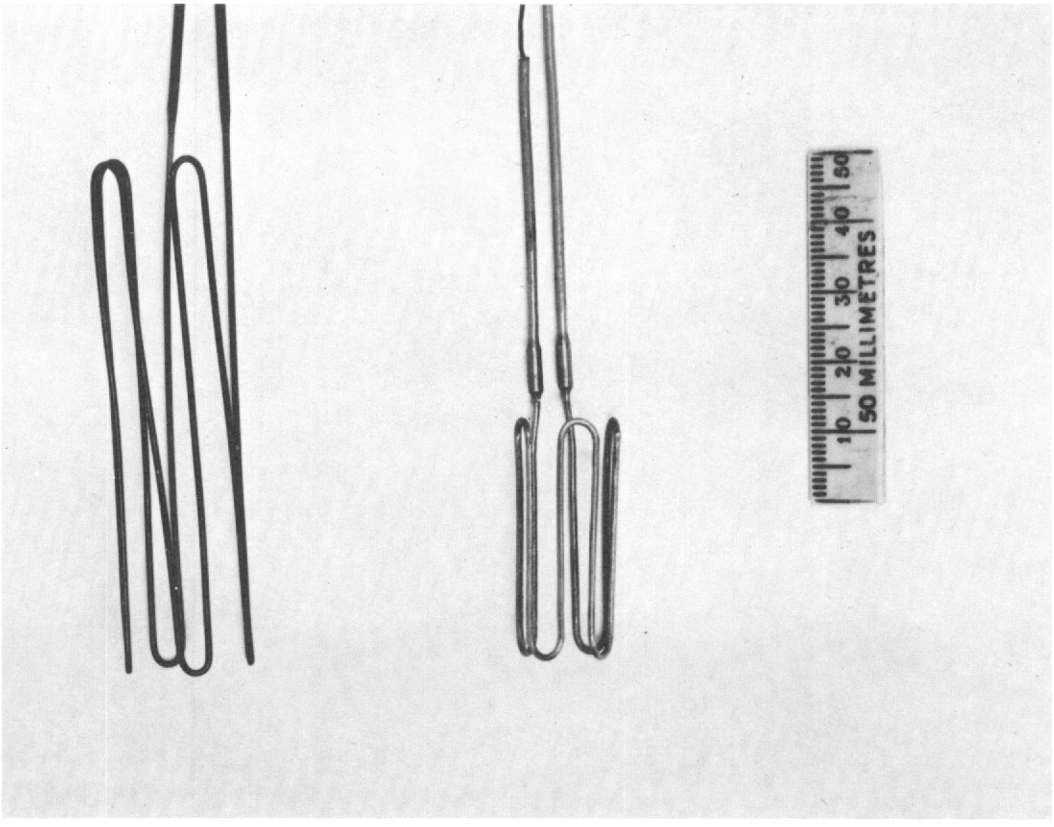


Fig. 6. 1 kW heaters for use in B.R.2 experiments.

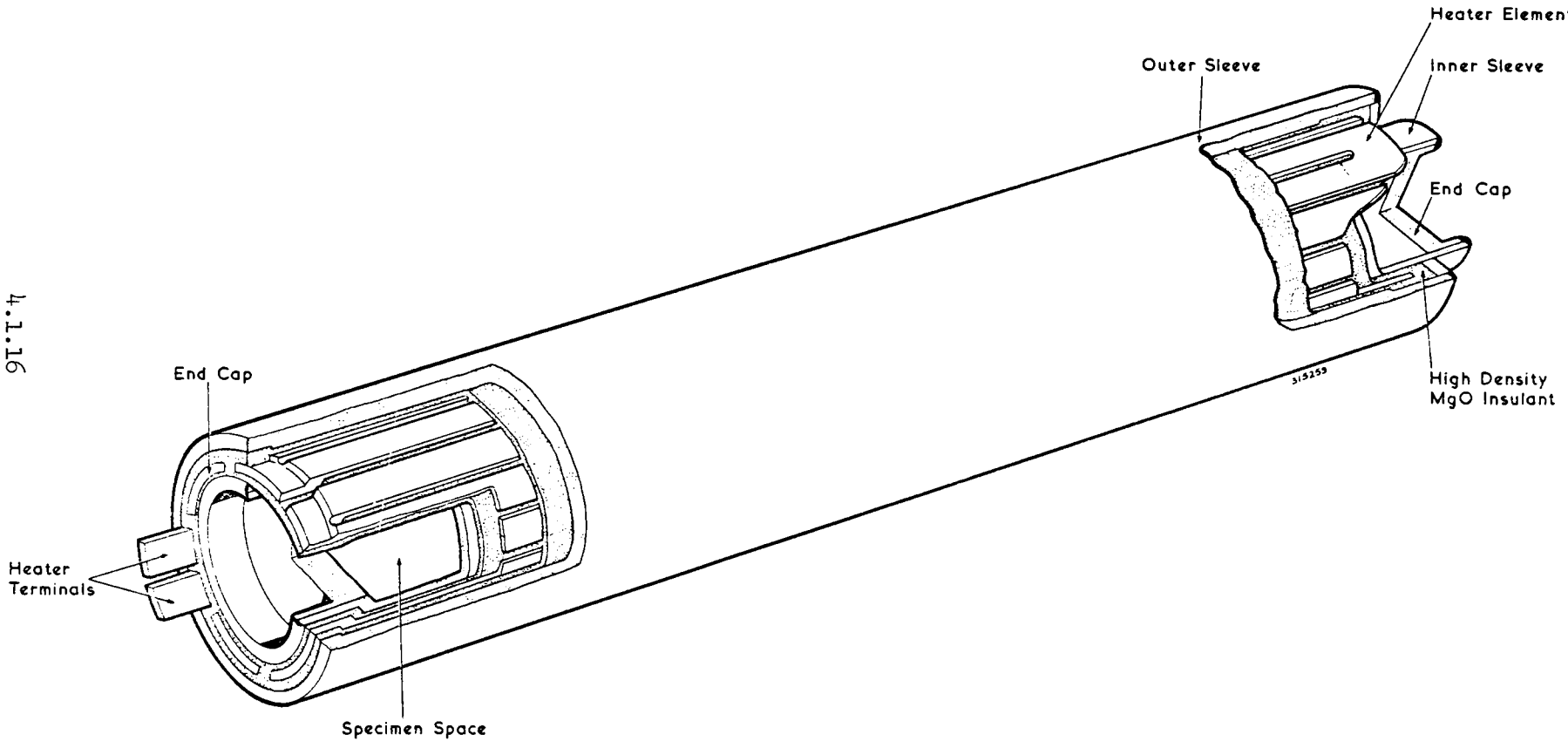


Fig. 7. Concentric Tube Heater

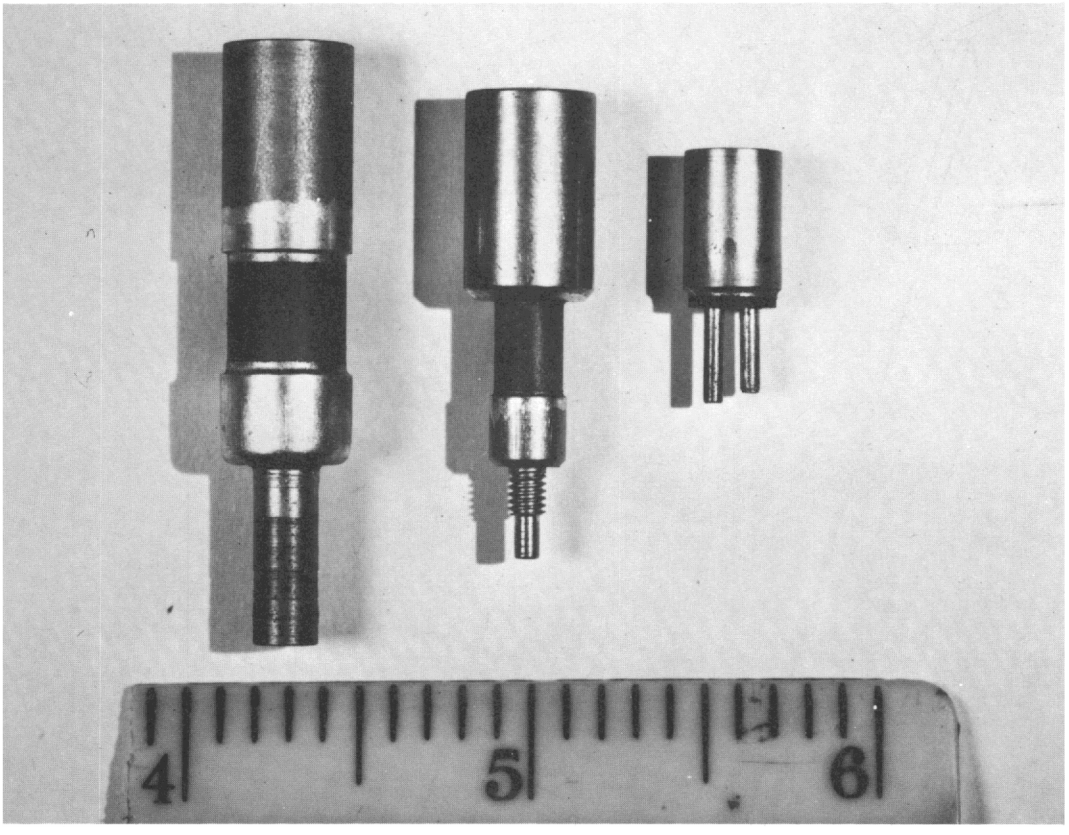


Fig. 8. Types of Ceramic/Metal Seals

CAPSULE INSTRUMENTATION AT AERE

by

I. H. McWilliam and C. B. T. Braunton

Abstract

Development of instrumentation at AERE has been guided by the three requirements of accuracy, reliability and simplicity. This has resulted in the development of a control system which has been found to be adequate for the vast majority of irradiation experiments, special instruments being required only on rare occasions. This paper outlines this system and indicates how it is being further developed; it also describes some of the minor instrumentation required to perform certain specialized and unusual functions.

Metallurgy Division,
Atomic Energy Research Establishment,
Harwell,
Didcot, Berkshire.

List of Illustrations

1. Introduction

The instrumentation chosen for a particular experiment is normally selected on the basis of its suitability, accuracy, reliability and cost for that experiment. When, as in the case in the utilization of research reactors like DIDO and PLUTO, a considerable number of experiments are being considered, three other considerations become important, these being flexibility of use, maintenance, and size.

These requirements, considered in toto, have lead to the general use at AERE of a restricted range of instruments, this range however being usable in a number of different ways depending on precise experimental requirements. Thus maintenance is simplified and capital cost reduced.

There are occasions, however, when the "standard" instruments are unsuitable and in these cases it has always been a prime consideration to select those special instruments which, as well as satisfying experimental requirements, have maximum reliability.

2. Standard Control Systems

The most common experimental requirement is for temperature control. Although early experiments could be adequately catered for by "on-off" or "high-low" systems, refinement of experimental techniques required closer control than could be obtained by these methods. Proportional control, using an electro-pneumatic system, was investigated and used to a limited extent but was rapidly replaced by an electronic three-term control system. This has been used for the last ten years and has proved eminently satisfactory.

A block diagram of the system is shown in figure 1. The recorder/controller derives its signal from a thermocouple sited in the experiment, modification to the instrument to work with various types of thermocouple being possible. The control signal, a small D.C. current of up to 5mA, is fed to the saturating coil of a saturable reactor, the main winding of which is supplied from A.C. mains. Variation in saturating current alter the impedance of the main winding and thus the output voltage to the load. Since the system is sensitive to load impedance, a matching transformer is normally interposed between the load and the saturable reactor, thus also serving to isolate the load from the main supply.

This system was, when it was initiated at AERE, probably the most compact and responsive control system for irradiation experiments which had ever been developed. It does, however, suffer from two disadvantages

which have become more apparent with increasing demands on space. These are component weight and size. The total weight of saturable core reactor and matching transformer for a 3 kW heater control circuit is some 3 cwt and the space taken up 4 c.ft.

3. Development of Control Systems

Development has been undertaken with two principal aims in mind.

(i) Miniaturization of controller/recorder instruments.
(ii) Miniaturization of power control systems.

3.1 Miniaturization of Controller/Recorder

This is desirable in order to decrease panel size. In order to simplify maintenance standard recorder/control instruments have been re-designed in collaboration with the manufacturer so that the front panel space is considerably reduced, while the depth is increased. In this way a greater amount of information may be displayed without reducing chart width.

3.2 Miniaturization of Power Control Systems

Extended testing has been carried out on silicon diode units to prove their reliability, stability and self-protection characteristics. Development has reached the stage at which orders have been placed for equipment to be used on irradiation rigs in early 1964.

The main advantages of this system are

(a) Decreased size and weight.
(b) Greater Flexibility in Component positioning.

4. Monitor and Safety Instrumentation

4.1 Safety Instrumentation

Major safety instrumentation, i.e. that monitoring conditions which might give rise to a hazard to the reactor, are commonly designed on either a 1 out of 2 or 2 out of 3 basis. In the former case two sensing elements and two instruments are so circuited that an alarm condition sensed by one element, or component failure in an element or instrument will cause a reactor shutdown. In the latter case three elements and three instruments are used, alarm conditions or failure of any two elements or instruments causing shutdown.

Minor safety circuits, i.e. those in which an alarm condition indicates a condition which is either potentially hazardous or is a serious departure from required experimental conditions, are on a

1 out of 1 basis, although spare sensing elements are normally incorporated in the experiments to cater for random faults. These minor circuits include sodium leak detection, water leak detection, instrument failure, etc.

Although early instrument consoles separated control and monitoring circuits into separate panels, considerations of space and flexibility have caused these to be combined into control and monitor consoles.

4.2 Temperature Monitoring

On all elevated temperature experiments it is standard practice to supplement the capsule control thermocouple with another connected to an entirely independent instrument. This ensures that failure of a controller does not result in gross, and possibly hazardous, temperature increases. Both potentiometric trip amplifiers and galvanometric indicators have been used for this duty, the choice being based partly on cost and partly on experimental requirements. While indicators are less costly and give full display of information, they cannot be made "fail safe" to such a degree as are trip amplifiers, which, however, display little or no information. The latter however, and to some extent the former, are steadily being supplemented by trip margin indicators which combine "fail safe" characteristics with temperature indication.

4.3 Sodium Leak Detection

This is of considerable importance where sodium filled capsules are being irradiated and is performed by siting the bared end of a mineral insulated stainless steel sheathed twincore cable in such a position that a sodium leak will cause an earth short. A relay, connected in series with the detector cable, is thereby de-energized and an alarm is raised.

4.4 Capsule or Rig Pressure Monitoring

This is used where the pressure in an experiment is an experimental requirement and must be controlled, or where safety requirements require a rig to be maintained at sub-atmospheric pressure. The latter case is common where beryllium or its compounds are being irradiated.

Normal bourdon tube pressure gauges, differential transformer transducers, potentiometer transducers and strain gauge transducers have all been used for this purpose. Generally, where remote pressure

indication is required and is combined with a potential hazard, strain gauge transducers have been used and found stable and reliable. In these instruments a complete strain gauge bridge is incorporated in a sealed capsule, one end of the capsule being acted upon by the pressure in such a way that its deflection causes consequent strain in the bridge elements.

4.5 Water Leak Detection

Two methods have been and are commonly used. The simplest is measurement of the insulation resistance of an open ended mineral insulated cable. This, however, tends to vary with reactor power due to irradiation damage effects on the MgO. Its very sensitivity means that spurious alarms may be raised by the use of imperfectly dried gas in a rig.

A more reliable, but more complex, system uses a 3 watt heater surrounding a small metal block in which is embedded a thermocouple. The heater is energized with a constant voltage which raises its temperature to ~200°C when the reactor is shutdown. When the reactor is raised to full power, nuclear heating in the metal block causes a further temperature rise to ~400°C. Thus, whatever the reactor power, a temperature in excess of 200°C should be indicated. Complete or partial submergence of the detector in water causes a temperature decrease to ~80°C in a period of about three minutes.

A proposal which is now being investigated is the use of a moisture meter which is supplied either constantly or intermittently, with samples of gas from the space being monitored. There appears to be no obvious reason why this should not be satisfactory, although there may be disadvantages attendant on the transport of the gas samples through small pipelines over distances which will vary from 10 to 30 feet.

5. Temperature Control Bandwidth

Control of most experiments is monitored within a band of $\pm$ 2°C at a constant reactor power. On those experiments in which control is required irrespective of reactor power this bandwidth is maintained except during the first 3 seconds of reactor shutdown. Over this period, due to the rapid decreases in fission and nuclear heating and the response time of the control system (~ 2 secs) a transient varying between -10°C and -20°C is imposed. It is hoped that the use of silicon diode rectifiers will decrease this effect, since the response time of this equipment is 0.5 sec compared with the 2 secs of a saturable core reactor.

In exceptional cases control has been maintained to $\pm \frac{1}{2}^{o}C$ irrespective of reactor power and including those periods during which reactor power is changing rapidly. The system used is shown diagrammatically in figure 4.

6. Specimen Heat Output Determination

Early work by Plail in the field of fissile irradiation capsules indicated that the use of these as calorimeters should be possible. Subsequent work by Clough and Lloyd confirmed this, calibration of the capsule as a calorimeter being effected by the use of the normal thermocouples and the immersion heater. Considerable calculation was required, however, to convert these calibrations and integrate the results over the life of an experiment to obtain the total heat output of the specimen and thus its burn-up.

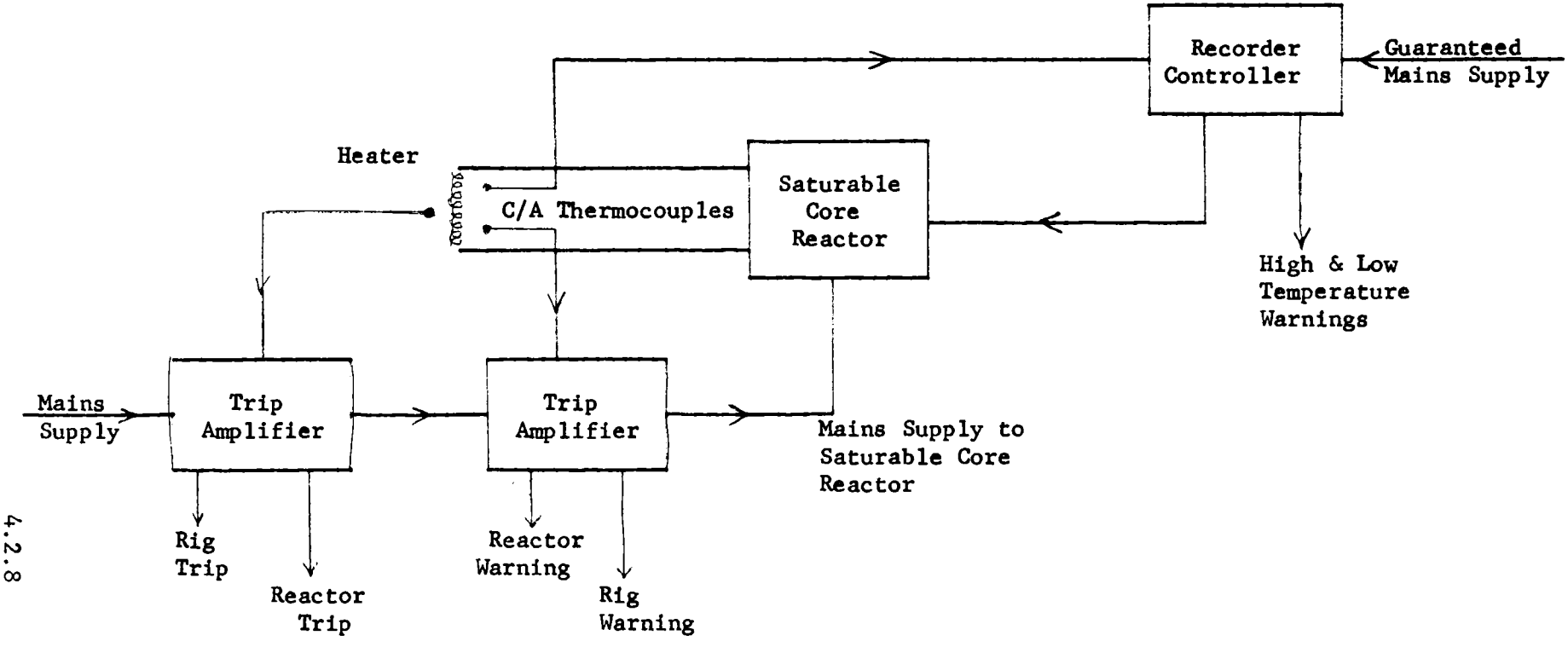
These calculations are now considerably eased by integrating the temperature/time record from a capsule during its operation. This establishes, since the relationships between capsule heat output and indicated temperature are known, the total heat output from the capsule over its entire life. The power supplied to the immersion heater is also integrated so that a simple subtraction, combined with an equally simple correction for reactor shut-down periods, establishes the total heat generated by the specimen. Thermal wattmeters are preferred to kwh meters on grounds of space.

7. Console design

Until reactor hall space became limited by the increasing number of experiments being performed and thus an increasing number of instrument panels, these were designed with ample access but without great regard to the space occupied. Subsequent designs have necessarily been more directly aimed at compactness without, however, allowing this to reduce either accuracy, reliability or the amount of necessary information displayed. Figures 5 and 6 indicate what has been achieved by this over the last five years, both sets of instrumentation performing similar functions.

Acknowledgements

The instrumentation described in this paper was initiated and, to some extent, its design and development guided by Metallurgy Division of AERE. Design and development could not, however, have been successfully undertaken without a close co-operative effort between the Metallurgy, Engineering and Research Reactors Divisions of AERE.

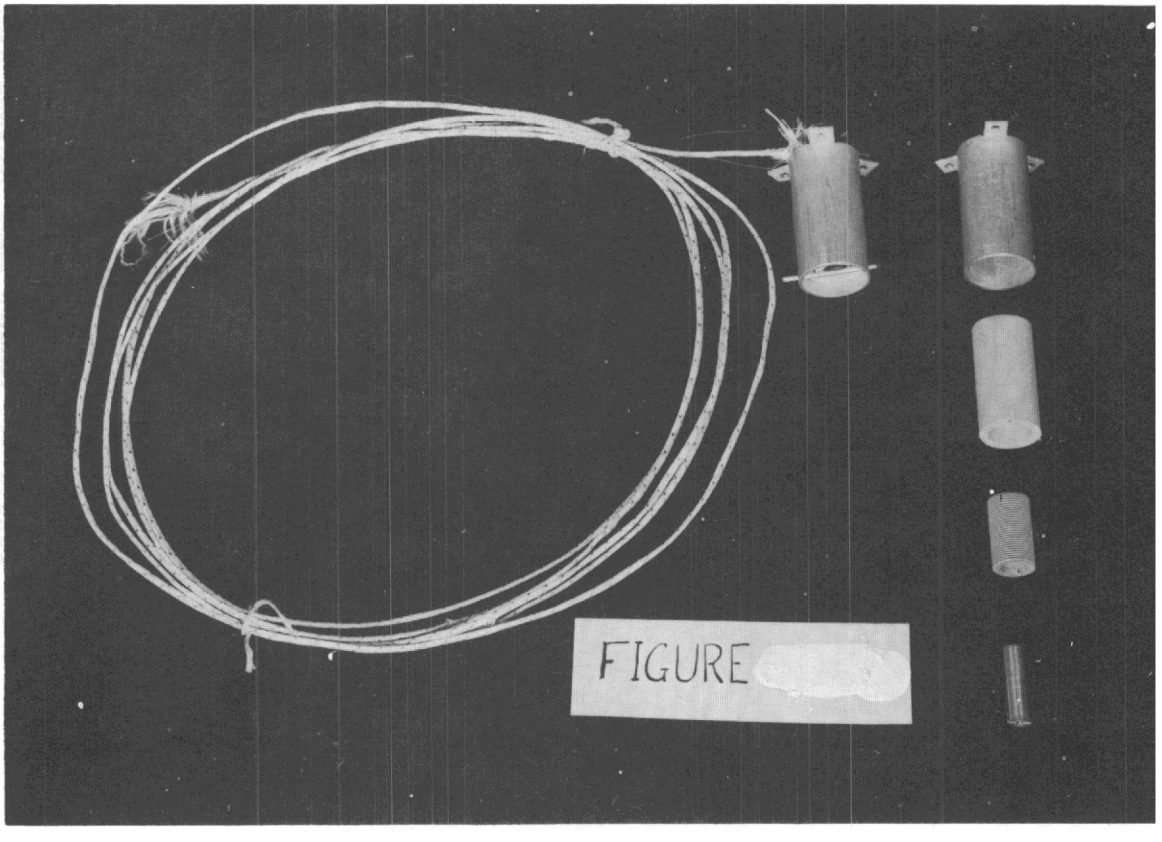


4.2.8

Figure 1. BLOCK DIAGRAM FOR STANDARD CONTROL SYSTEM

2. Redesigned temperature recorder/controller

3. Water leak detector



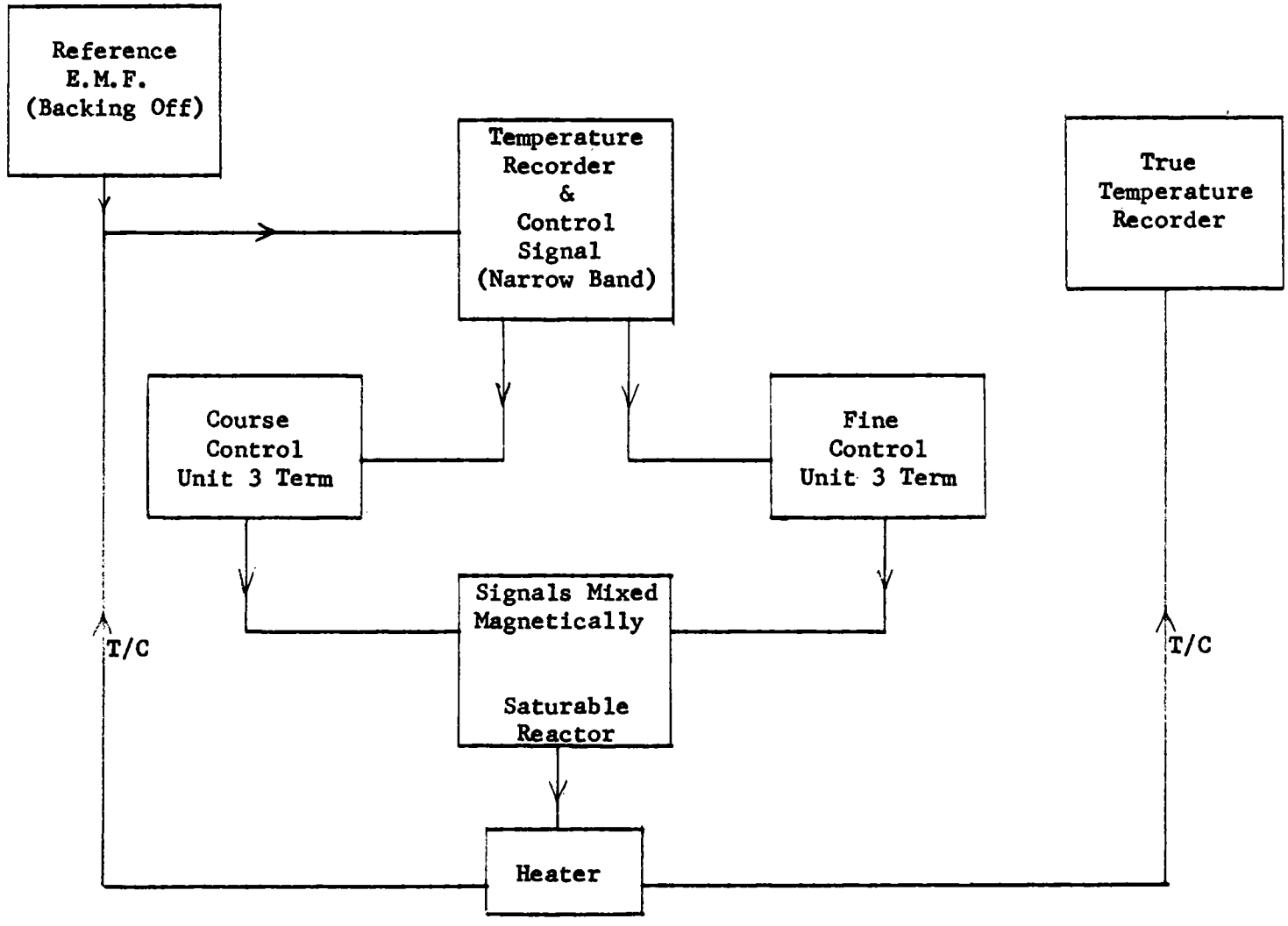
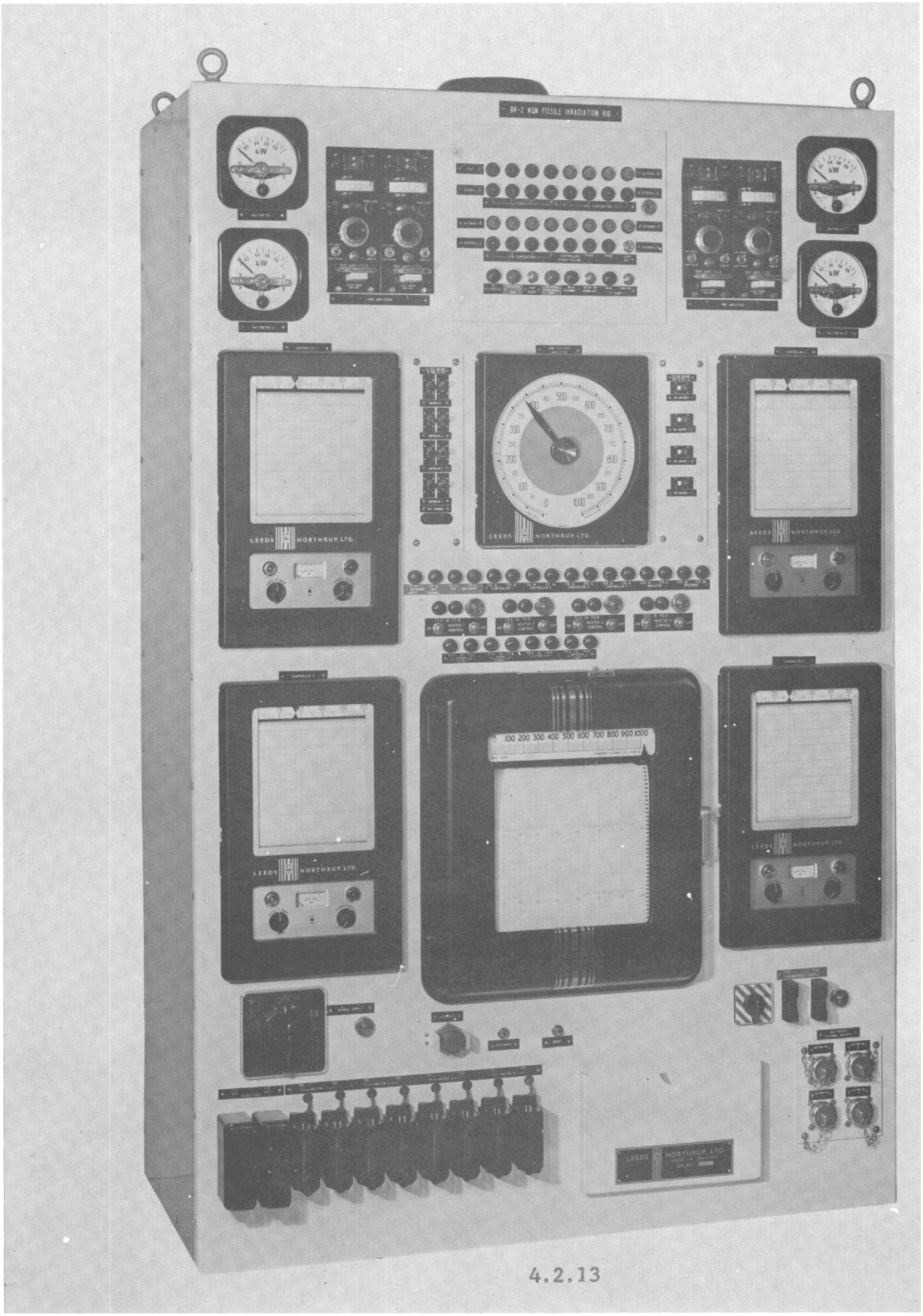


Figure 4. NARROW BAND CONTROL SYSTEM

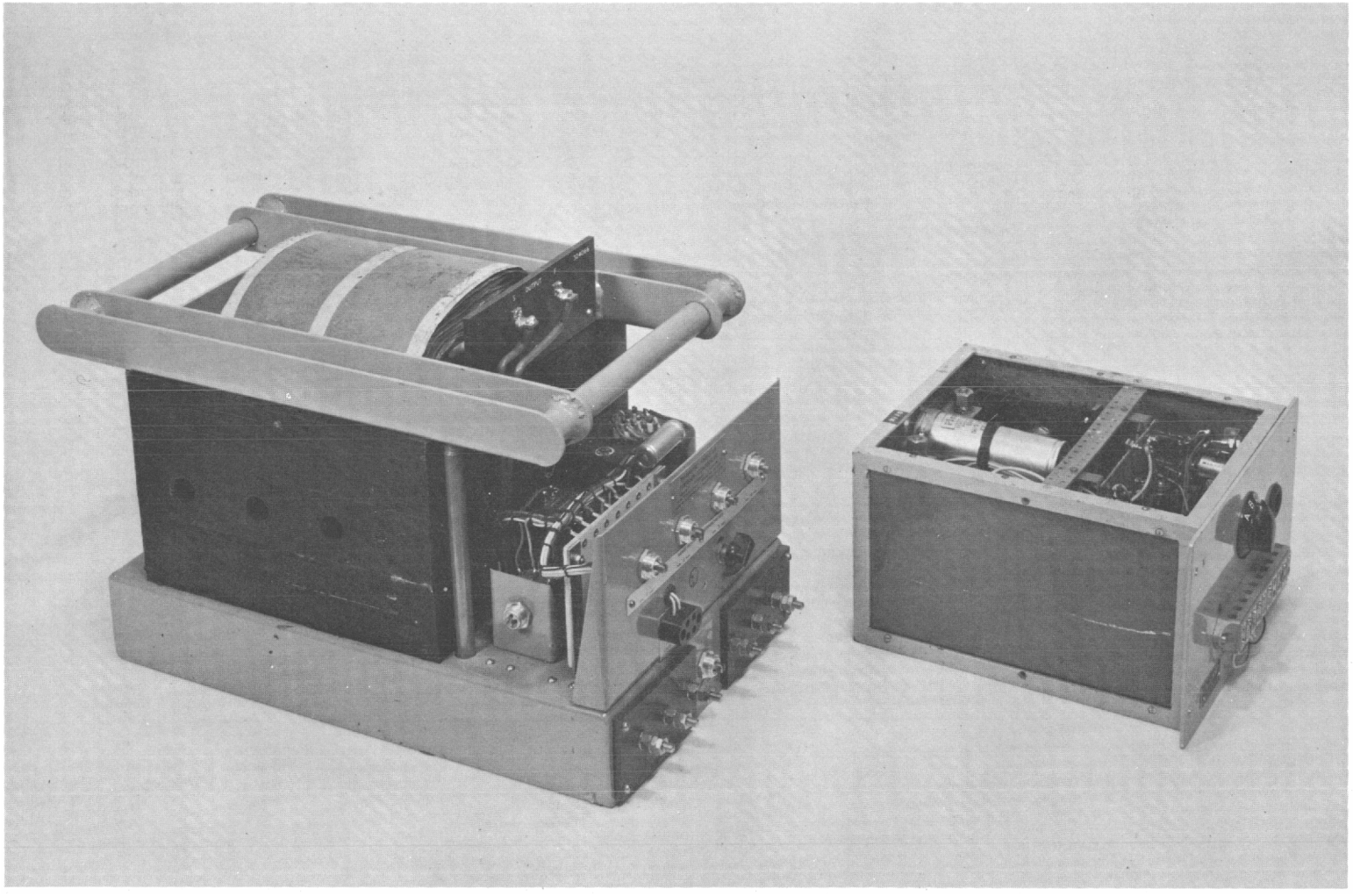
5. Early control and monitor consoles

4.2.12

6. Modern control and monitor console



4.2.13



7. (a) Saturable core reactor (b) Silicon controlled rectifier

3 kW Control Units

DEVELOPMENT PROGRAM TO INCREASE THERMOCOUPLE RELIABILITY FOR IN-PILE EXPERIMENTS

EVERETT L. BABBE
ATOMICS INTERNATIONAL

As temperature and reactor residence time requirements for in-pile experiments increase, thermocouple reliability problems have become more numerous. High level thermal cycling conditions make the thermocouple failure rate increase. This high failure rate is especially apparent when fuel elements designed for high temperature reactors are tested in water-cooled reactors. Under these conditions, a thermal barrier is provided to maintain a high fuel surface temperature giving rise to high thermal shocks upon reactor scrams.

At AI a program to investigate the causes of thermocouple failure is currently in progress. All work to date has been on one size and type of thermocouple; that is, metallic sheathed, MgO insulated, 1/16" diameter, compacted thermocouple material with a grounded measuring junction. This investigation is directed toward finding the cause for wire open circuits after repetitive thermal cycling.

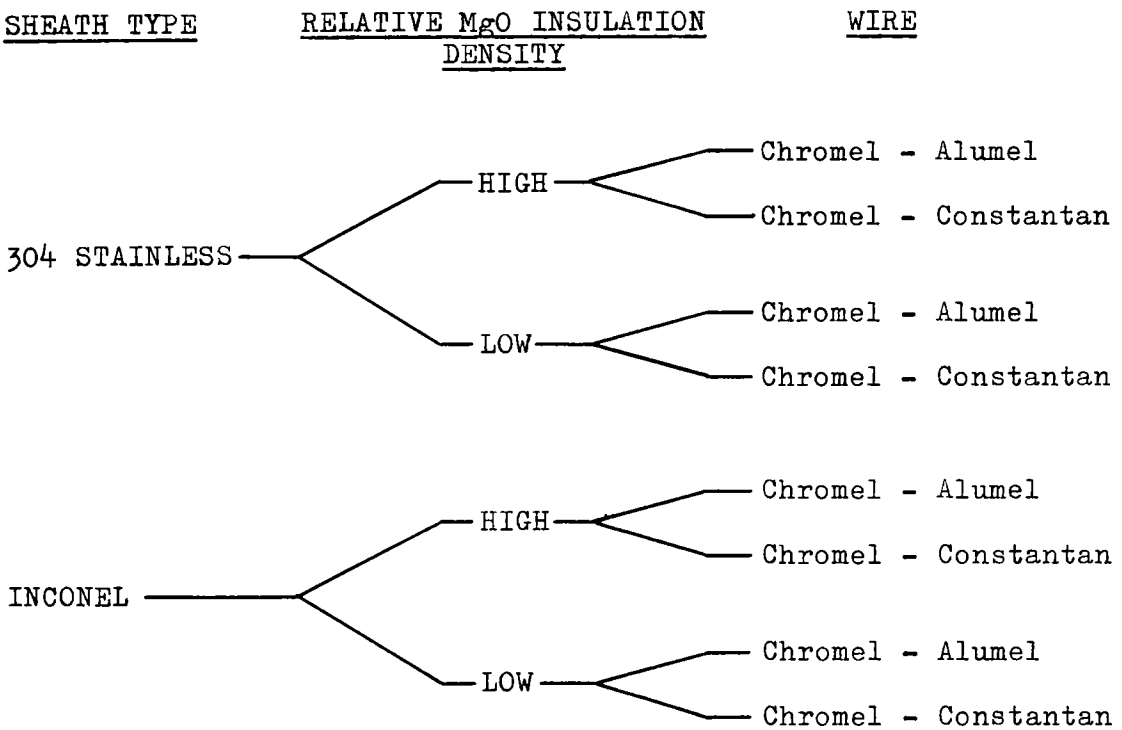
The first step in the analysis is to induce failure by a laboratory produced thermal shock test. Thermocouples are alternately cycled between a 1250°F sodium nitrate bath and room temperature water until wire continuity is lost. Loop resistance measurements are made to indicate when thermocouple failure

occurs. Post failure analysis includes x-ray location and metallurgical examination of the failed area.

Concurrent with the thermal shock testing, an investigation is being conducted to determine the effect of insulation pack density on thermocouple failure. Dye penetrant techniques are employed to determine differences in the insulation density. The results are correlated with absolute values of insulation density and also compared with the thermal shock test results.

The dye penetrant test samples are prepared by cutting lengths of the material with an abrasive saw, deburring them on a lathe and finally removing the glaze from the end with a fine file. Six 1/2 inch lengths of each material are placed in a vertical position with 1/16 inch of the bottom in Zyglow dye penetrant solution number ZL-22. The material is observed with an ultra-violet light and the time is recorded for the dye to traverse the 1/2" length and completely color the top insulation surface. These six transit times, typically 15 to 500 minutes, are averaged to obtain a material "dye penetrant time".

A particular lot of eight materials was used to provide the correlation curve plotted on Graph 1. They were made under controlled conditions to eliminate many of the variables which are associated with different vendor processes and components. Combinations of two sheath materials with high and low density MgO insulation and two wire combinations were manufactured as the following diagram illustrates.



The wire-sheath combinations were chosen to give various degrees of wire stress caused by differences in coefficients of thermal expansion. This wire stress is best illustrated by Graph 2. As can be noted the most serious expansion mismatch is the 304 Stainless sheath Alumel wire combination. Note that the Inconel sheath keeps Constantan in compression upon temperature increase.

These eight different 15 foot lengths were made at the same time under identical conditions. All components were weighed and measured before assembly. These physical measurements were the basis for the insulation porosity calculations. To achieve

the high and low insulation densities, two different sizes of ceramic insulators were used and all other processes and component sizes were kept identical. Chromel P wire from a common spool was used in all eight samples. They were single pass swaged and batch annealed in one lot.

The thermal shock tests were conducted on thermocouples with grounded measuring junctions. They were made by fusion welding the thermocouple tip held in a copper chill block. A small disc of metal, the same material as the sheath, is fused on the tip with a tungsten inert gas welder. This type of junction was found to be as reliable as any other tested.

It has been noticed in post failure examination of thermocouples from the thermal shock test, that the failure usually occurs within a few wire diameters of the measuring junction. The failure mode is ordinarily of the ductile type in the case of low insulation density material and of the intergranular type in the high density material. The Alumel wire usually fails first in the Chromel - Alumel combination and the failures are approximately evenly divided in the Chromel - Constantan material.

Fifty-six thermocouples from these eight controlled samples and seventy-four others from four suppliers have been tested to date. The thermal shock results are plotted against dye penetrant time, (average values for a given material) in Graph 3.

To eliminate the effect of inconsistencies in the measuring junction on the failure rate, a thermal cycle test (different from the thermal shock test) is in progress. In this test the thermocouple material, less hot junction, is temperature cycled once each minute until a wire failure occurs. This is done by electrically resistance heating the sheath to a temperature of 2000°F in 5.5 seconds (approximately 300°F/sec) and then allowing it to cool off in still air the remaining 54.5 seconds. The results of this test substantiate the data from the thermal shock test, but because of a lower rate of temperature increase, the material survives three to five times as many cycles. Other than the increased number of cycles to failure, the trends are identical.

This thermocouple study is not complete but preliminary test results indicate significant differences in failure rate that appear to be attributable to at least two causes. One is the thermal expansion rate of the sheath as compared to the expansion rate of the wires. This is demonstrated by the fact that Inconel, which better matches the expansion of the thermoelements tested - Chromel, Alumel and Constantan, decreases the failure rate greatly as compared to the stainless sheaths tested.

It also looks like the insulation pack density influences the failure rate to a large degree. It is important in reducing wire stress that the differential temperature between wire and

sheath be kept to a minimum. Since the insulation density affects the thermal conductivity it is possible that it affects the wire stress to a significant degree. As indicated by the trends in Graph 3, a high insulation pack density appears best for thermocouples used under thermal shock conditions.

ADDENDUM - P. B. Ferry

During this meeting, Battelle Memorial Institute presented data showing that a decrease in insulation density enhanced the life of thermocouples during thermal cycling. This appeared to be in direct opposition to results obtained in this test program. During a discussion with Battelle personnel it was found that conditions for the two tests were not the same. A summary of the discussion is included in this addendum as an effort to clarify the two seemingly opposing standpoints.

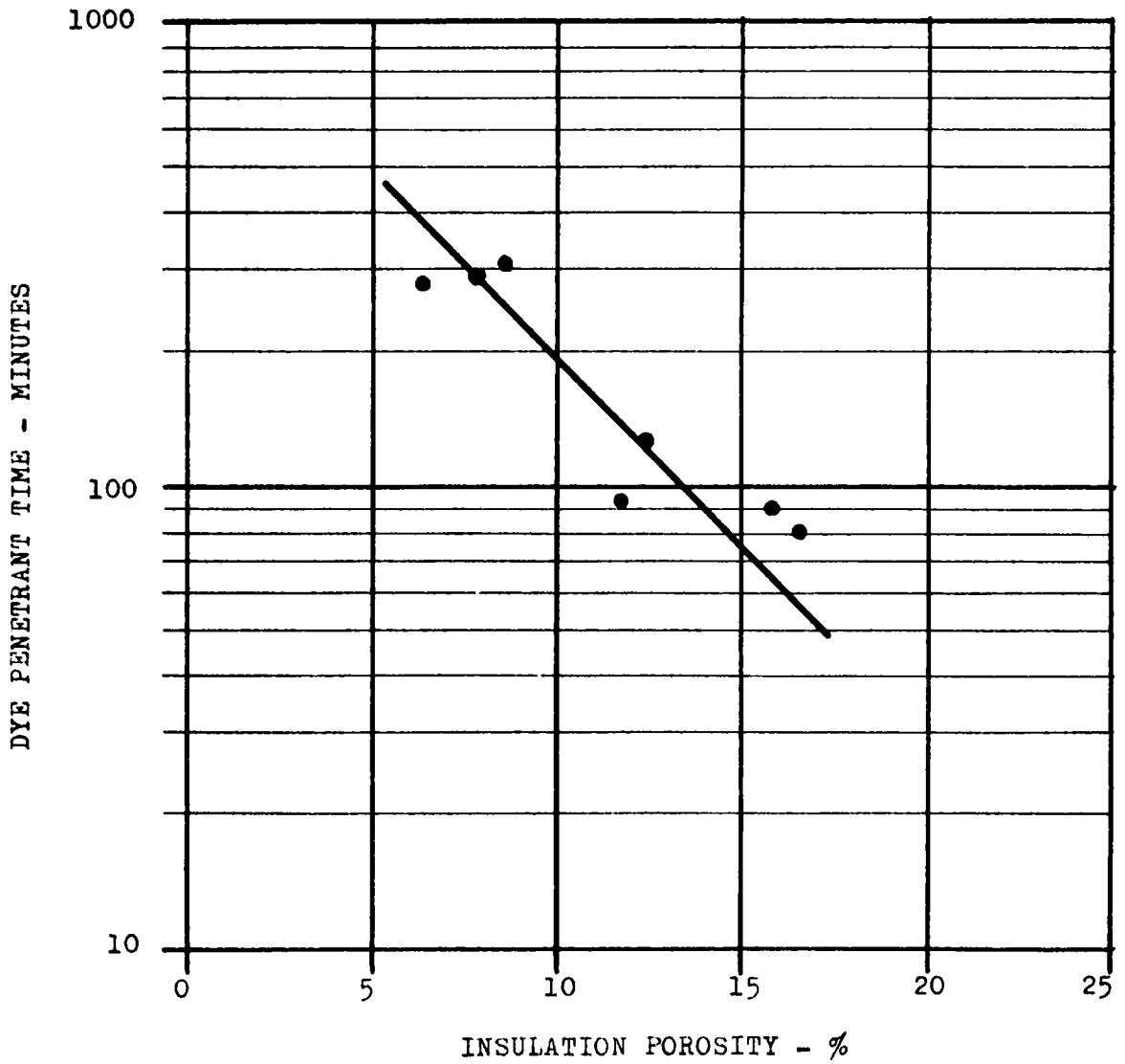
In the thermal cycle tests at Atomics International the sheath was heated approximately 2000°F in five seconds while in the Battelle tests the temperature rate was about 1000°F in 30 seconds. The temperature gradient between the thermocouple wires and sheath in the high heating rate test would be greater than for the latter. It was expected that as the temperature gradient increased, the wire strain would also increase. Two methods to decrease the gradient would be to increase the conductivity of the insulation between components, and reduce the heating rate. To increase the conductivity of the insulation,

the thermocouple composite is severely compacted. The conductivity is increased but at the expense of degrading the wires due to the excessive deformation.

In the high heating rate test it was important to have good thermal conductivity to minimize strain. If the heating rate were decreased the thermal gradient would be decreased; in this case the important factor is to have structurally sound wires. The insulation conductivity probably does not significantly affect the strain. Thus, in the Battelle tests, where the heating rate was low, the low density thermocouples lasted longer. In the Atomics International tests when the heating rate was high, the high density thermocouples gave the longest life under thermal cycling.

The two series of tests have been shown to impose different conditions on thermocouples. The results of each appear to be valid for the particular test.

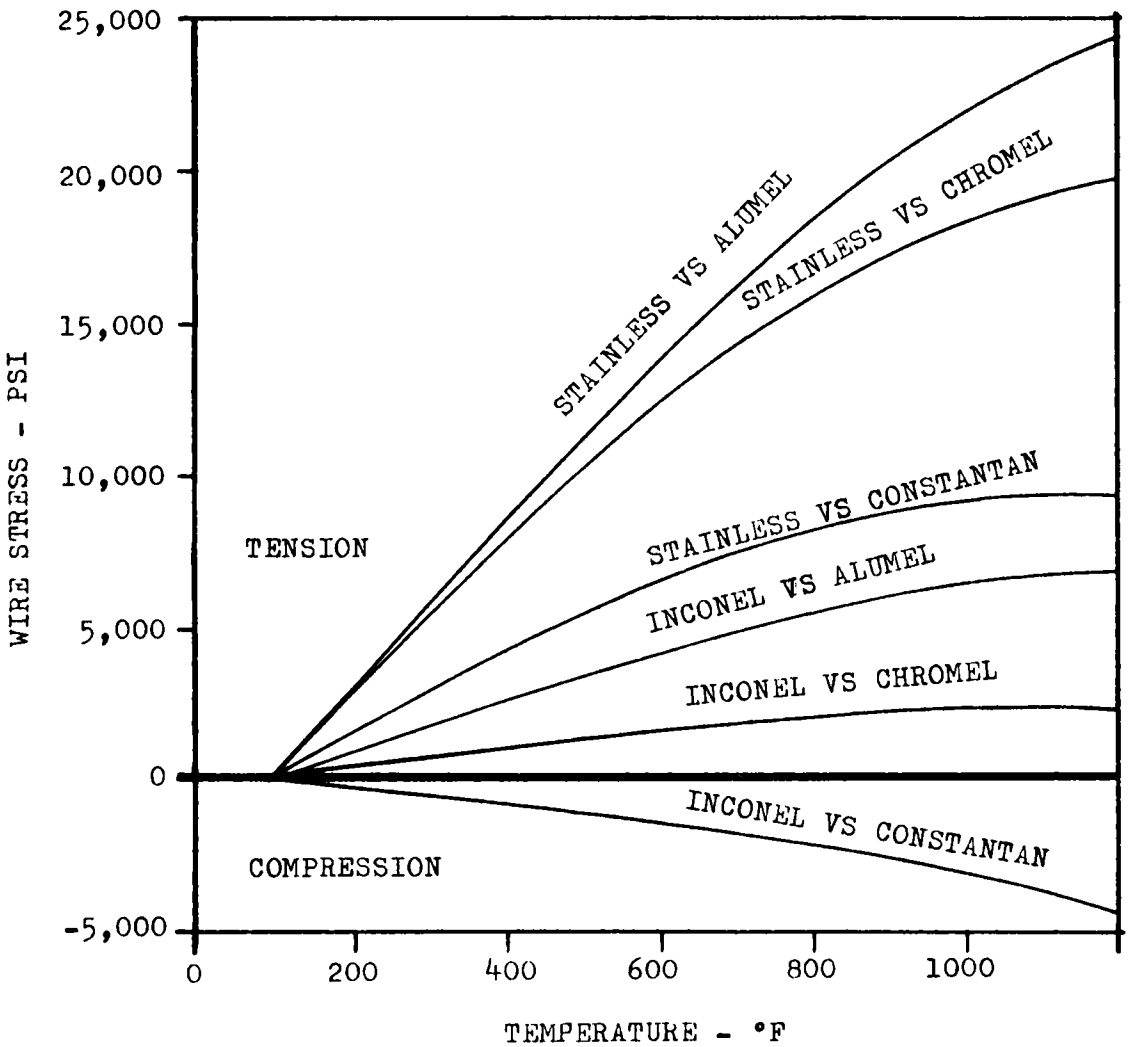
DYE PENETRANT TIME AS A FUNCTION OF INSULATION POROSITY



SAMPLE LENGTH - .500
SAMPLE DIAMETER - .062
INSULATION - MgO
DYE - ZYGLOW #ZL22

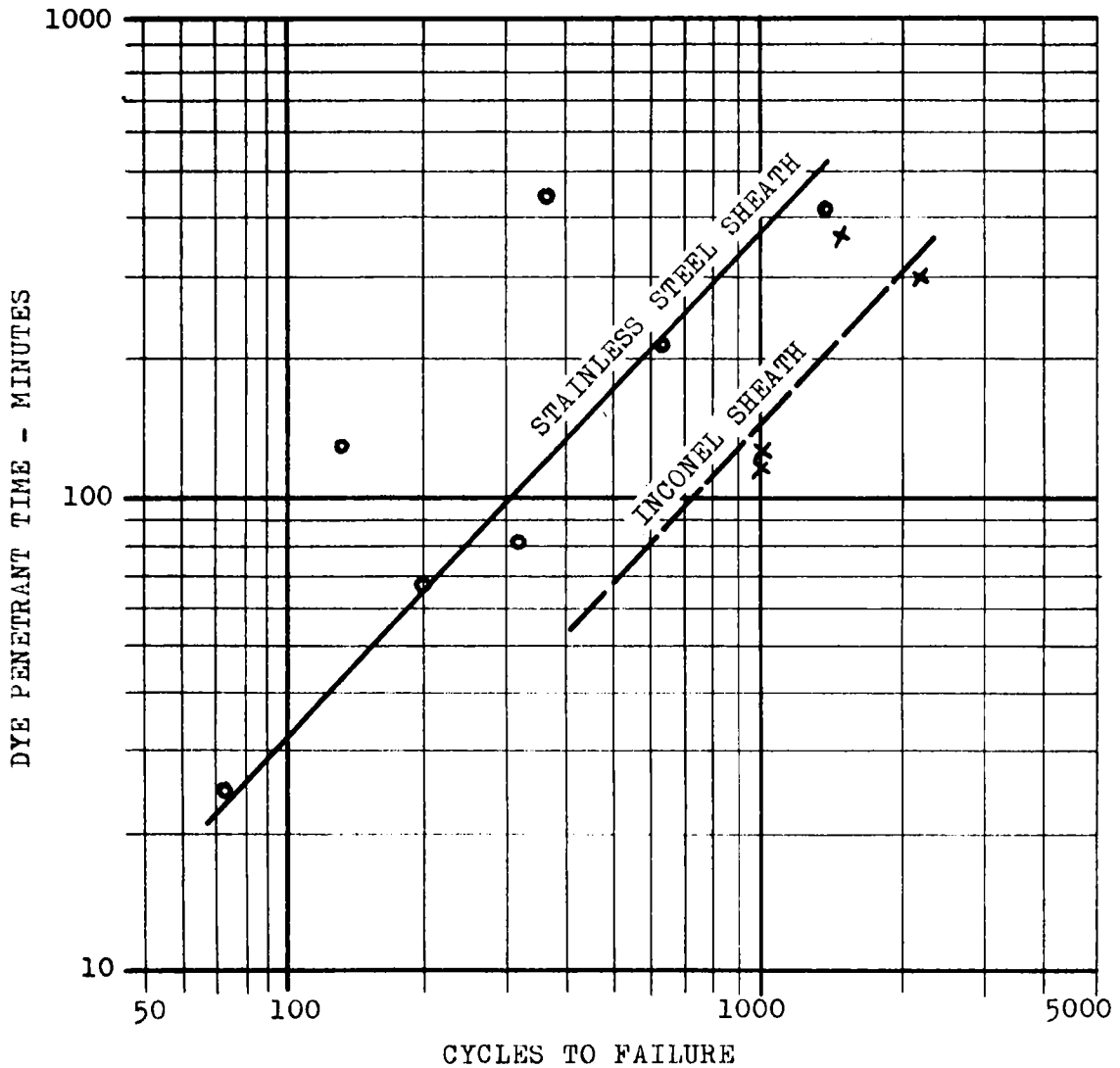
GRAPH 1

THERMOCOUPLE WIRE STRESS DUE TO THERMAL EXPANSION DIFFERENCES BETWEEN WIRE AND SHEATH



GRAPH 2

THERMAL SHOCK TEST FAILURE TRENDS



MATERIAL DIAMETER - .062
INSULATION - MgO
JUNCTION - GROUNDED
WIRE - CHROMEL - ALUMEL

GRAPH 3

LABORATORY STUDIES TOWARD IMPROVEMENT OF HIGH-TEMPERATURE THERMOCOUPLES*

J. F. Lagedrost
Battelle Memorial Institute

Several relatively simple laboratory experiments have been conducted in an attempt to discover ways in which thermocouples develop open circuits during irradiation-capsule experiments. The work was devoted primarily to 1/16-in.-O.D. sheathed Chromel-Alumel thermocouples, although a few experiments were conducted with sheathed tungsten-rhenium units of similar size.

The experiments were based on the premise that in-pile thermocouple open circuits stem largely from a combination of sustained exposure to elevated temperature (particularly above 1500 F) and severe thermal cycling during reactor startup and shutdown periods. As a result, they became essentially life tests with rapid (approximately one cycle per minute) thermal-cycling conditions between temperature limits approximately as follows: 1000 and 2000 F in the case of the Chromel-Alumel units and 1000 and about 2800 F in the case of the tungsten-rhenium units. The test thermocouples were obtained from suppliers who normally provide Battelle with couples and heaters for irradiation experiments. No special precautions were taken in their manufacture.

The laboratory data from Chromel-Alumel combinations indicated, first of all, that almost all failures occurred in the Alumel leg. This was true for thermocouples with either grounded or ungrounded welded junctions. No failures were observed in weld-affected zones; all were at least 1/2 inch from the bead.

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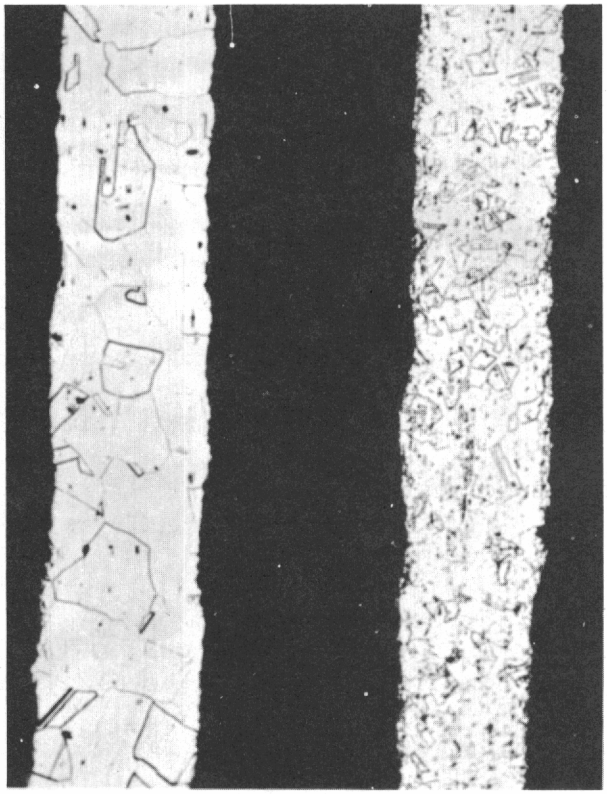
* The work reported in this paper was performed by Battelle Memorial Institute, under AEC Contract W-7405-eng-92.

Figures 1 and 2 show typical effects in the grain structure of Chromel-Alumel thermocouples resulting from the heat treatments involved in the thermal cycling. It is obvious that grain growth in the Alumel element proceeded to full-wire diameter more rapidly than in the Chromel elements and, because of the associated decrease in ductility, the Alumel element would be expected to fail first.

Additional tests were performed along the lines outlined in Table 1. Essentially, two insulation packings, similar with respect to insulating characteristics but substantially different as regards firmness, were checked to evaluate the merit of a packing loose enough to reduce the axial restraint on the wires as they expanded and contracted. In addition, two sheath materials, Type 347 stainless steel and Inconel, were employed to assess the behavior of wires highly strained through differential thermal-expansion effects and that of wires virtually unstrained by this mechanism.

The findings from these experiments demonstrated:

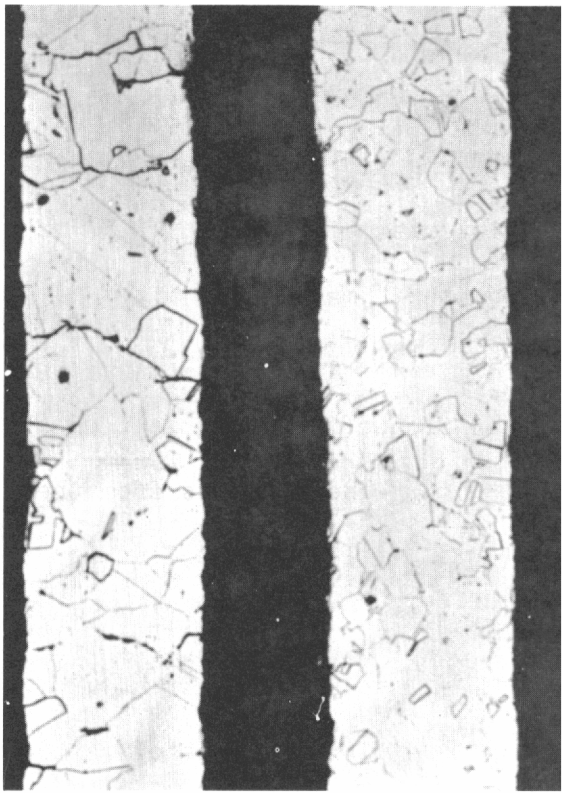
(1) Distinctly longer life for units with "moderate-packed" insulation as opposed to that of units with "hard-packed" insulation. As it turned out, these two arbitrarily designated degrees of insulation packings were, for the particular thermocouple tests, reflected in the relative hardnesses of the as-received elements and sheaths. Of course, hardness cannot be used routinely as such an indication, owing to variations in sizes of pre-swaging stock and in post-swaging heat-treatment procedures.



100X

ALUMEL CHROMEL

SECTION OF SHEATHED CHROMEL AND ALUMEL WIRES BEFORE TESTING. NOTE MgO GRIPPING OF WIRES

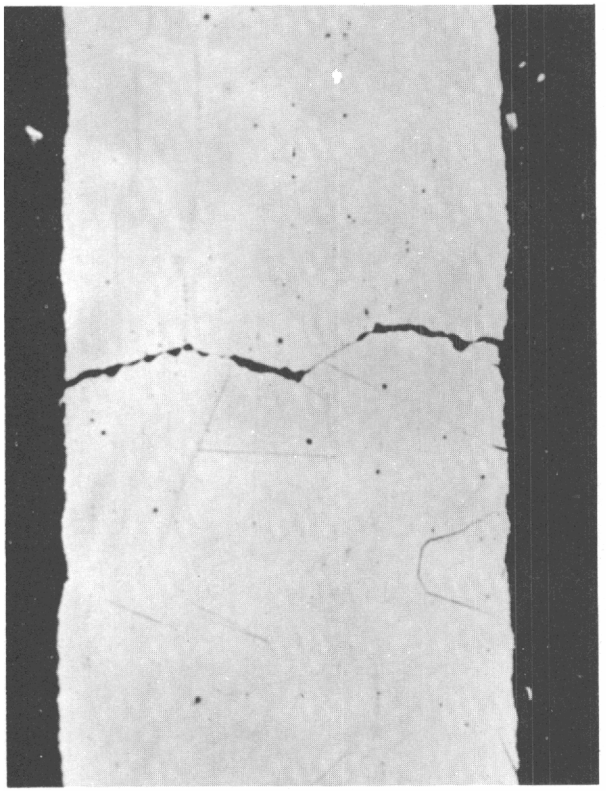


100X

ALUMEL CHROMEL

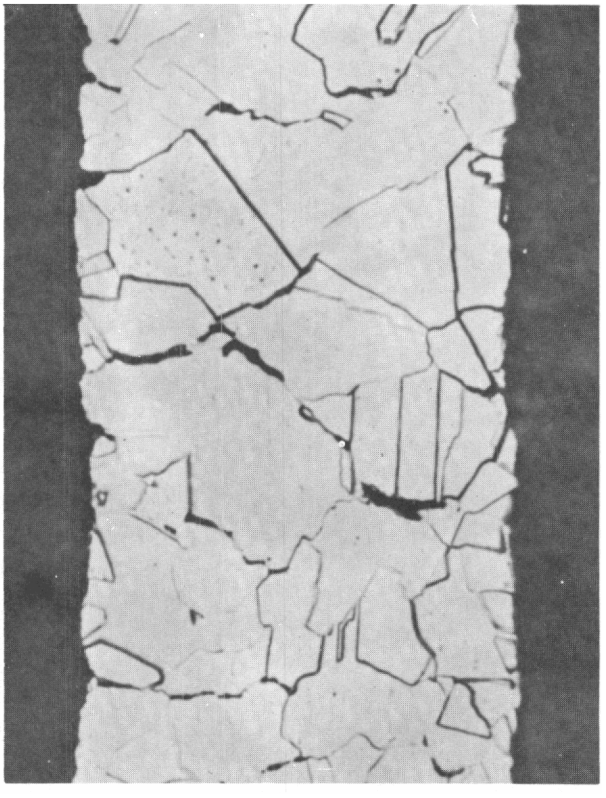
SECTION OF SHEATHED CHROMEL AND ALUMEL WIRES AFTER 800 THERMAL CYCLES, 1000 F – 2000 F.

FIGURE 1. MICROGRAPHS SHOWING THE EFFECTS OF THERMAL CYCLING ON GRAIN STRUCTURE OF CHROMEL-ALUMEL WIRES



50X

100 CYCLES, "HARD-PACKED" MgO



250X

440 CYCLES, "MODERATE-PACKED" MgO

FIGURE 2. MICROGRAPHS OF TYPICAL ALUMEL WIRE FAILURES AFTER 1000 F – 2000 F THERMAL CYCLES IN STAINLESS STEEL SHEATHS

SHEATH – TYPE 347
INSULATION – "HARD PACKED" MgO

1.	STARTING HARDNESSES	α
C	300	16.7
A	250	15.8
S	400	20.0

2. LIFE – ~ 100 CYCLES

CODE

C – CHROMEL LEG
A – ALUMEL LEG
S – SHEATH
HARDNESS – KNOOP, 100-G LOAD
α – COEFFICIENT OF THERMAL EXPANSION, MICROIN./C

SHEATH – TYPE 347
INSULATION – "MODERATE PACKED" MgO

1.	STARTING HARDNESSES	α
C	175	16.7
A	165	15.8
S	225	20.0

2. LIFE – ~ 400 CYCLES

SHEATH – INCONEL
INSULATION – "MODERATE PACKED" MgO

1.	STARTING HARDNESSES	α
C	170	16.7
A	130	15.8
S	200	16.5

2. LIFE – AT LEAST 2000 CYCLES

TABLE 1. INFORMATION FOR CHROMEL-ALUMEL THERMOCOUPLE THERMAL-CYCLE EXPERIMENTS

The difference between "moderate-packed" and "hard-packed" insulation, for this work, is illustrated by the fact that the MgO could be removed easily from the former with a pick, whereas it could not from the latter. However, the insulation would not fall out of the open sheath end of "moderate-packed" units with mild tapping.

(2) A clear superiority for the Inconel-sheathed units, showing the advantage in reducing wire strain by resorting to well-matched coefficients of thermal expansion.

As a result of these experiments, Inconel-sheathed thermocouples with "moderate packed" insulation have been used exclusively in several recent irradiation capsules. Although this type of packing is not amenable to a rigorous specification, it generally is sufficient to specify that the insulation be packed only as tight as is necessary to maintain good electrical properties for the ultimate application. The relevant in-pile experience has not been extensive but there are indications that improved thermocouple performance is being obtained. At the time of this writing, of approximately 80 couples, about half operating at about 600 F and half operating in the range 1200 to 1500 F, only two failures by open circuit have occurred during the progress of the irradiations involved. This is a substantially lower failure ratio by this mechanism than encountered during previous irradiations.

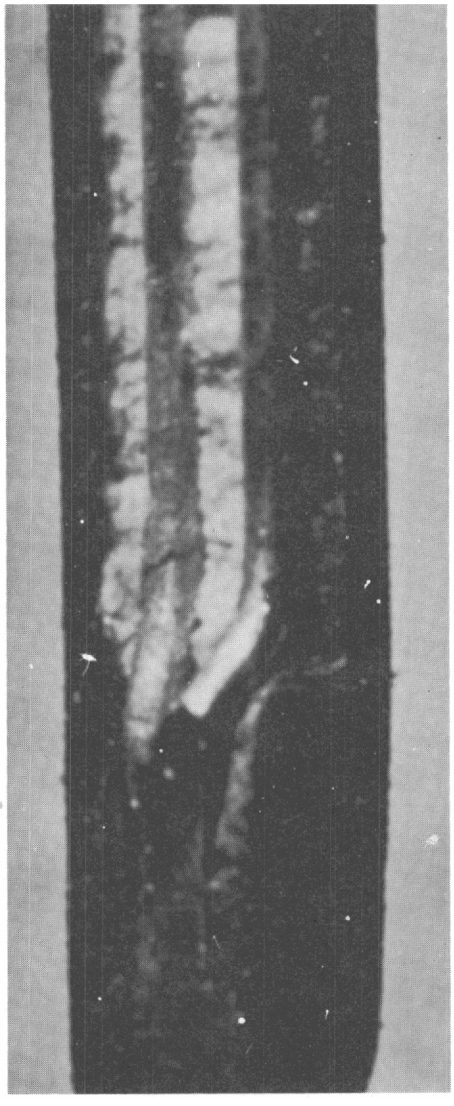
It might be added that Battelle now specifies grades of Chromel and Alumel designed for operation above 1600 F under the condition of a reducing or marginally oxidizing atmosphere. These are Hoskins materials designated as Chromel 3G-345 and Alumel 3G-196. To date, there is no indication that these alloys give better performance than the so-called standard grades.

Nb-1Zr-sheathed MgO-insulated W-5Re vs W-26 Re units were used in tests of this combination. The major concern here was that the low ductility of the tungsten alloys plus the unmatched coefficients of thermal expansion involved (that for the Nb alloy being about 60 per cent higher than those of the W-Re alloys) would adversely affect life under a thermal-cycling condition. To check this, 1000 to 2800 F cycling experiments were made with thermocouples with grounded and with ungrounded junctions. The former were prepared by fusing wires and sheath ends together and also by a mechanical-plug technique developed by the supplier. In the case of the ungrounded junctions, the wires were fused together to form the bead, and the sheath was closed independently.

The test information indicated distinctly longer life for units with ungrounded junctions. Actually, no failures were observed in up to 5000 thermal cycles with these units as opposed to frequent failures in less than 300 cycles for the grounded units. These failures were almost always characterized by the fracture of the W-5Re alloy wire, as illustrated in Figure 3. Unsheathed fused junctions were also tested; here, no failures were observed in up to 30,000 cycles between 500 F and 3000 F.

This information has prompted Battelle to specify ungrounded junctions (plus "moderate-packed" insulation) for in-pile tungsten-rhenium thermocouples. The in-pile experience with such units has to date been limited and there is virtually no basis for judging whether or not the advantage observed in the laboratory will carry over to the irradiation situation.

One inherent drawback with W-Re elements for applications in long-term irradiation exposures is their instability of calibration owing



30X

FIGURE 3. MACROGRAPH OF W-Re THERMOCOUPLE WITH MgO INSULATION AND Nb-1Zr SHEATH, SHOWING FAILURE OF W-5Re ELEMENT AFTER 300 CYCLES BETWEEN 1000 F AND 2800 F

Mechanical junction with plug subsequently welded.

to transmutation effects. This has prompted Battelle to look into the possibility of substitute materials which will not suffer this type of damage. One promising lead involves a platinum-molybdenum alloy combination developed by a British Company, Johnson-Matthey Ltd. At present, we are trying to procure a few experimental units for laboratory investigation.

AUTOMATIC REDUCTION OF LONG-TERM IRRADIATION CAPSULE THERMOCOUPLE DATA*

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Battelle Memorial Institute

In the course of irradiation testing of nuclear fuels at Battelle, the need has often arisen for a convenient, economical method of extracting useful information from the voluminous amounts of thermocouple temperature data generated during long-term irradiation experiments. The usual form of the data is a strip chart from a multipoint temperature recorder which prints out the temperatures of several thermocouples, approximately once per minute. In the past, these charts have been scanned manually, and the data have been selectively sampled and replotted on a reduced time scale. These graphs are used along with knowledge of the capsule heat-transfer characteristics to compute fission-heat generation rates. Burnup rate of the fuels under test may then be assessed on the basis of fuel burnup level, and fuel centerline and surface temperatures may be determined from the heat-generation rate. This procedure, although adequate for short irradiations, is costly and time-consuming for programs with many fuel specimens and thermocouples and irradiation times of several years. A system for automatic data reduction is currently being utilized by Battelle in conjunction with long-term capsule irradiations of homogeneous fuels developed by Combustion Engineering, Incorporated.

Figure 1 is a block diagram of the data-reduction system. The data originate at the ETR (Engineering Test Reactor) where five capsules are being irradiated for 1 to 3-1/2 years. The capsules contain a total of 25 cylindrical clad fuel specimens of various U^{235} loadings and diameters. Adjacent to each specimen are located two Chromel-Alumel sheathed thermocouples. The ETR Data Logger, a thermocouple scanning and recording

*The work reported in this paper was performed by Battelle Memorial Institute, under AEC Contract W-7405-eng-92, subcontracted by Combustion Engineering, Incorporated.

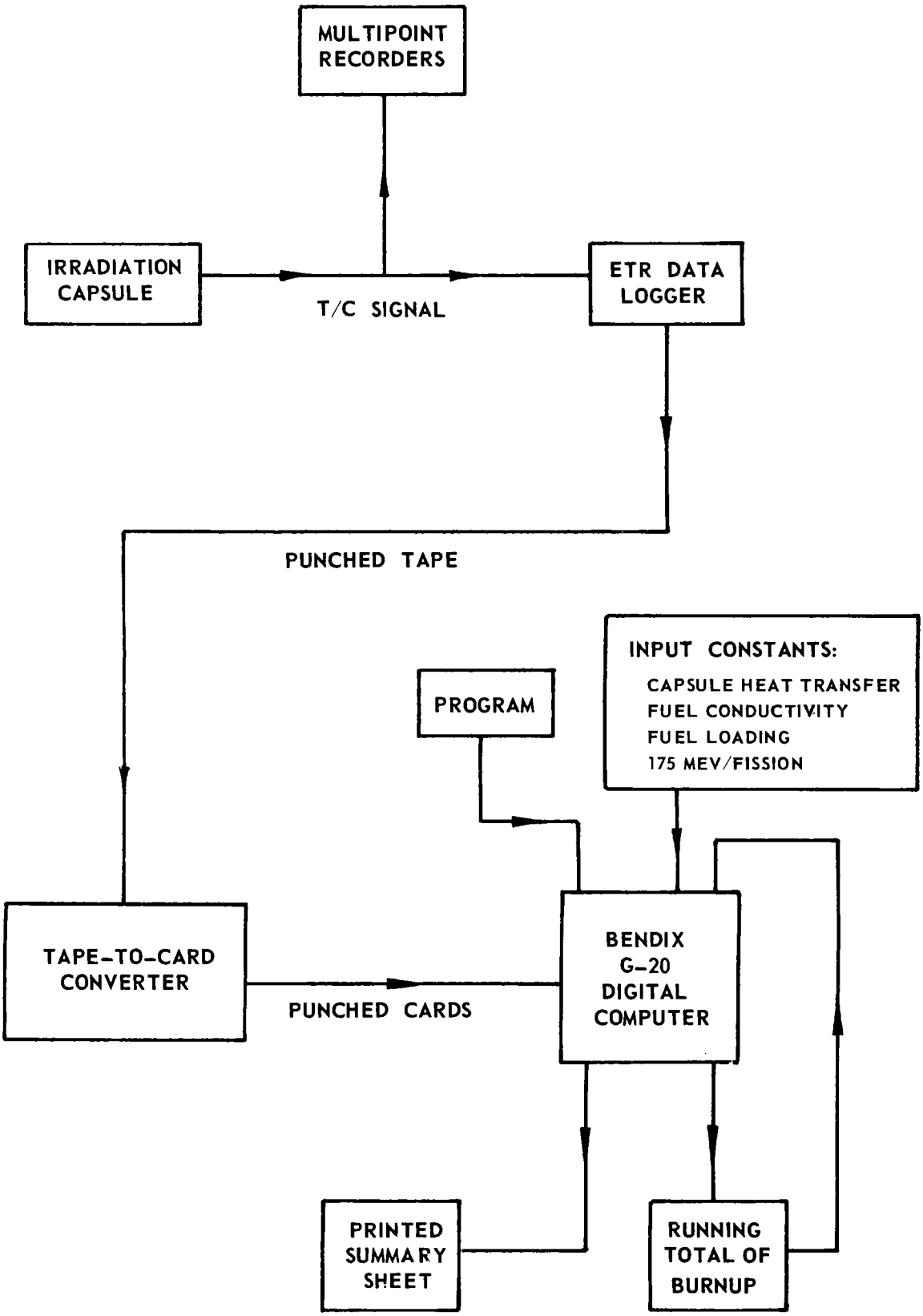


FIGURE 1. SCHEMATIC DIAGRAM OF DATA REDUCTION SYSTEM

system maintained by the ETR, monitors each of the 50 thermocouples every 5 minutes and records their output on a typewriter every 30 minutes. Several multipoint recorders also record the temperatures in the usual manner to provide a complete temperature history. Temperature data in the form of 5-channel punched paper tape is also obtained from the Data Logger. The tape is reproduced periodically from a master tape, which is used to actuate the typewriters, and is transmitted to Battelle by air mail. At Battelle the data on the punched tape are converted to punched-card form so that they may be read into the digital computer for analysis. Each week, a set of data cards (about 1000 cards for an average week of reactor operation) is read into the computer along with the program deck, input constants, and burnup cards. These cards may be described as follows:

I. Program Deck

These cards contain the step-by-step instructions for processing the temperature data and computing the desired output. The program language is Bendix Fortran.

II. Input Constants

These cards contain the constants describing the thermal and nuclear properties of each of the 25 specimens and 5 capsules. These constants, which are used in the equations incorporated in the program, are as follows:

A. Capsule Heat Transfer

These constants are the thermal resistances in the capsule between the specimen surface and the thermocouple and 120 F reactor coolant water.

B. Fuel Thermal Conductivity

An effective thermal conductivity for each specimen is included in the input constants so that the fuel centerline temperature may be calculated.

C. Fuel Loading

The amount of U^{235} for each specimen is also given.

III. Burnup Cards

These cards are output of the previous computer run and contain the running total of the U^{235} burnup by fission of each of the specimens.

The final form of the data is a convenient weekly summary sheet, shown in Figure 2. The output for all 25 specimens* is arranged in columns as follows:

(1) Capsule - Specimen Identification

(2) Initial Burnup - burnup at the beginning of the week under consideration, i.e., the value read from the burnup input card.

(3) Accrued Burnup - Sum of the incremental burnups computed for each half-hour interval during the week.

(4) Final Burnup - Burnup at the end of the week, computed by adding the accrued burnup to the initial burnup.

(5) Average T(CL) - Arithmetic mean of the fuel centerline temperatures computed for each half-hour interval.

(6) Peak T(CL) - The maximum value of the fuel centerline temperatures computed for each half-hour interval.

*Provision was originally made for a sixth capsule which was placed in the MTR where there is no data logging facility.

SPECIMEN DATA FOR PERIOD 0000, 4-21-63 TO 0000, 4-27-63. TOTAL HRS. 168 TOTAL READINGS 237 COMPUTER RUN NO. 14-54

CAPS.-SPECIMEN IDENTIFICATION	INITIAL BURN-UP	ACCRUED BURN-UP	FINAL BURN-UP	AVERAGE T(CL)	PEAK T(CL)	TIME AT PEAK	COUNT AT PEAK	VARIANCE OF T(CL)	AVERAGE T(S)	AVERAGE UNP. FLUX	HRS. AT POWER
43-1, Z-10-A	11.49	0.79	12.28	3494.	3813.	231.	6.	89334.	551.	.91E+14	118.0
43-1, C-10-A	12.23	0.84	13.06	2981.	3267.	2031.	39.	72148.	599.	.10E+15	118.0
43-1, Z-10-B	12.88	0.85	13.72	3736.	3977.	1431.	75.	91282.	582.	.99E+14	118.0
43-1, Z-10-C	12.95	0.86	13.82	3802.	4223.	531.	12.	101311.	590.	.10E+15	118.0
43-1, C-10-B	11.45	0.70	12.15	2525.	2701.	301.	100.	42755.	523.	.86E+14	118.0
43-2, Z-10-D	0.74	0.69	1.42	3054.	3403.	131.	4.	147003.	495.	.73E+14	118.0
43-2, Z-10-E	0.75	0.69	1.44	3069.	3403.	1.	1.	127914.	496.	.73E+14	118.0
43-2, Z-10-F	0.74	0.69	1.43	3069.	3403.	1.	1.	125451.	497.	.73E+14	118.0
43-2, C-10-C	0.71	0.65	1.36	2354.	2638.	601.	13.	83768.	494.	.74E+14	118.0
43-2, C-10-D	0.57	0.52	1.09	1908.	2071.	531.	57.	42756.	419.	.59E+14	118.0
43-3, Z-10-G	0.57	0.50	1.07	2220.	2352.	1.	1.	57510.	368.	.52E+14	118.0
43-3, Z-10-H	0.58	0.50	1.09	2253.	2441.	1331.	73.	37037.	371.	.53E+14	118.0
43-3, Z-10-J	0.60	0.52	1.12	2304.	2530.	1431.	75.	32300.	377.	.55E+14	118.0
43-3, C-10-E	0.56	0.49	1.05	1761.	1901.	1331.	73.	19690.	374.	.55E+14	118.0
43-3, C-10-F	0.55	0.48	1.04	1749.	1832.	1.	1.	22915.	372.	.55E+14	118.0
43-4, Z-22-M	0.00	0.00	0.00	0.	0.	0.	0.	0.	0.	.0	118.0
43-4, C-25-M	0.00	0.00	0.00	0.	0.	0.	0.	0.	0.	.0	118.0
43-4, Z-22-N	0.00	0.00	0.00	0.	0.	0.	0.	0.	0.	.0	118.0
43-4, Z-22-O	0.00	0.00	0.00	0.	0.	0.	0.	0.	0.	.0	118.0
43-4, C-25-O	0.00	0.00	0.00	0.	0.	0.	0.	0.	0.	.0	118.0
43-5, Z-22-T	0.59	0.55	1.14	3748.	3991.	503.	237.	86274.	573.	.78E+14	118.0
43-5, Z-22-X	0.60	0.56	1.16	3803.	3991.	501.	56.	78973.	580.	.79E+14	118.0
43-5, Z-22-Y	0.65	0.61	1.26	4104.	4327.	1.	1.	83765.	617.	.86E+14	118.0
43-5, C-25-N	0.63	0.59	1.22	3198.	3486.	131.	4.	54643.	601.	.91E+14	118.0
43-5, C-25-S	0.59	0.56	1.15	3026.	3183.	231.	193.	46347.	574.	.86E+14	118.0
43-6, Z-22-P	7.83	0.54	8.37	3692.	3906.	1.	1.	124369.	566.	.79E+14	118.0
43-6, C-25-P	8.26	0.56	8.82	3009.	3217.	301.	100.	72649.	571.	.87E+14	118.0
43-6, Z-22-R	8.55	0.56	9.12	3828.	4075.	731.	16.	104464.	583.	.82E+14	118.0
43-6, Z-22-Z	8.32	0.55	8.87	3720.	3906.	1.	1.	114312.	569.	.80E+14	118.0
43-6, C-25-R	7.87	0.48	8.35	2613.	2948.	331.	101.	266645.	509.	.76E+14	118.0

UNITS: BURN-UP(ATOM PERCENT U235 BY FISSION),TEMPERATURE(DEGR.F),FLUX(NEUTRONS/SQ CM-SEC)

COMMENTS:

FIGURE 2. TYPICAL WEEKLY SUMMARY SHEET

(7) Time at Peak - Time of day at which the peak in (6) occurred.

(8) Count at Peak - The half-hour reading number which may be used to determine the date on which the peak in (6) occurred.

(9) Variance of T(CL) - A statistical measure of how much the centerline temperature fluctuated about the mean value; numerically, it may be thought of as the square of the RMS deviation.

(10) Average T(S) - Arithmetic mean of the specimen surface temperature computed for each half-hour interval.

(11) Average Unperturbed Flux - Average unperturbed thermal-neutron flux computed from the temperature data and experimentally determined perturbation relationship.

(12) Hours at Power - The number of hours during the week that the reactor was at a power level greater than half the maximum (all temperature data at lower power levels are rejected).

The set of summary sheets from the data-reduction system comprise an up-to-date record of progress in the irradiation for each specimen. The information presented on the sheets is essentially the same as would be obtained by hand calculation, using manually plotted graphs, but it is available sooner (a week after the data are originally recorded at the ETR) with considerably less staff time required for processing (1 to 2 hours per week), and it is obtained at a lower cost.

Specimen Heat Transfer from Combined Calorimetry and Gamma Activity Distribution

By

B. Weidenbaum
M. F. Lyons
D. H. Coplin
T. J. Pashos

Work Performed Under

U. S. Atomic Energy Commission

Contract AT(04-3)-189
Project Agreement 17

4.6.1

Specimen Heat Transfer from Combined Calorimetry
and Gamma Activity Distribution

As part of an AEC-Euratom program directed to determining the performance limit of UO_2 as a fuel, a series of capsule irradiations were made in the GETR Trail Cable Facility with the objective of determining the specific power that would be required to induce varying degrees of central melting in UO_2 fuel rods to be used in a loop experiment. The fuel used for the capsule experiments was identical to that in the large rods in all respects except for enrichment and the length of the fuel column. The capsules contained five inches of fuel instead of the 30 inch fuel column in the rods irradiated in the loop. The irradiation time was twenty minutes for all the capsules used in this investigation.

It was recognized at the outset that the same difficulty that was bothersome to other investigators, i.e. distinguishing unequivocally how far radially the UO_2 has melted, would be a problem in these capsules. However, because the Trail Cable Facility affords the opportunity for more accurate measurement of the capsule thermal performance than in most previous tests, The calorimetric data obtained during the capsule irradiation coupled with extensive post-irradiation examination in the Radioactive Materials Laboratory (RML) does provide the necessary guidance for the more costly loop experiments. In addition, the capsule experiments furnished a means of obtaining the thermal conductivity of UO_2 at temperatures approaching its melting point.

The GETR Trail Cable Facility, which was used for the capsule irradiation, is a dead-end tube extending from a 45 degree penetration on the third floor of GETR to three inches below the bottom of the reactor core. The tube is in the pool of the reactor and runs vertically along the side of the reactor pressure vessel from below the core to several feet above the top of the core. The tube then makes three bends, each with a six foot radius, between the top of the core and the penetration on the third floor of the reactor. The radius of these bends and the diameter of this tube limit the length of the irradiation capsule that can be used in the facility. In these irradiation experiments the capsules were about seven inches long. Of the over-all length five inches were occupied by fuel. On the basis of experience in over sixty previous capsule irradiation experiments, the power produced by the capsule was known to have an accuracy of I 10 percent. This accuracy is readily obtainable from the measurement of the coolant flow rate and temperature rise during the twenty minute irradiation period used for these capsules.

The fuel consisted of 17.2 percent enrichment UO_2 pellets, each one-half inch in length, which had been sintered in hydrogen to a density of 96 – 97 percent of theoretical. The pellets were centerless ground to give a 6-mil diametral gap between the fuel and the 0.565-inch OD × 30-mil wall, Zircaloy-2 cladding. The capsules were vacuum outgassed at 900 F and backfilled with helium at this temperature before making the final weld closure.

The simplicity of the capsule design can be seen in Figure 1. The use of hollow end closures, which facilitated their being welded to the fuel caldding, was changed to solid ones when it was recognized that fuel movement into the end closure was undesirable from an experimental standpoint.

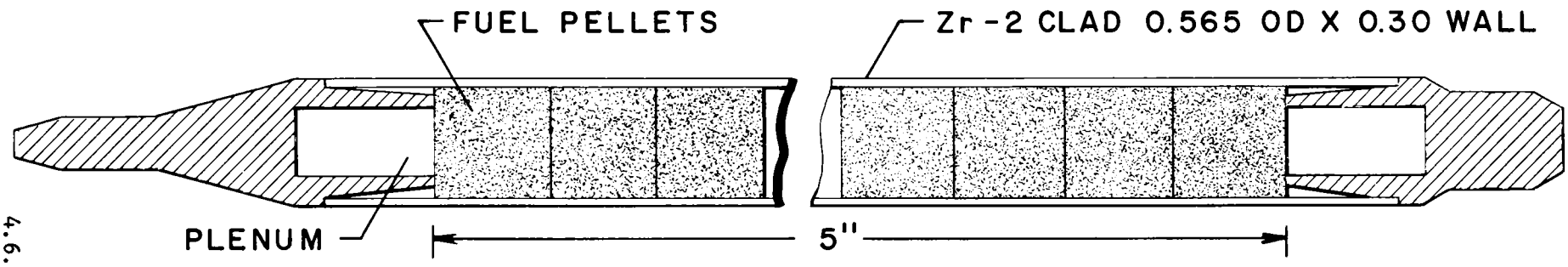
Flux peaking, of course, occurred at the ends of the capsule. The degree of flux peaking is readily apparent in Figure 2, which is a post-irradiation gamma scan of the capsule shown in Figure 1, obtained within a few days after irradiation. The slope of the gamma scan indicates that this capsule was above the peak neutron flux in the axial direction.

Gamma scans of this type are routinely obtained at the Vallecitos Atomic Laboratory's RML to determine the intensity distribution of gamma activity along the length of an irradiated fuel rod. To obtain such a gamma scan the irradiated specimen is mechanically driven at constant speed past a lead-shielded slit, 0.020 inches wide by 0.750 inches long, which extends through the three-foot wall of an RML cell. The emergent, collimated beam of gamma radiation is measured and recorded continuously with conventional equipment located outside of the cell.

In the case of these short term irradiation capsules, the gamma scan is directly proportional to the power generation from point to point along the fuel column because neither the position of the capsule nor the neutron flux changes significantly during the irradiation. From the calorimetric data and the surface area of the portion of the cladding that surrounds the irradiated fuel, the average surface heat flux of the capsule can be calculated on the basis of essentially one dimensional heat transfer throughout the entire rod. In addition, the location of the point along the rod at which the average heat flux is being generated is obtainable geometrically.

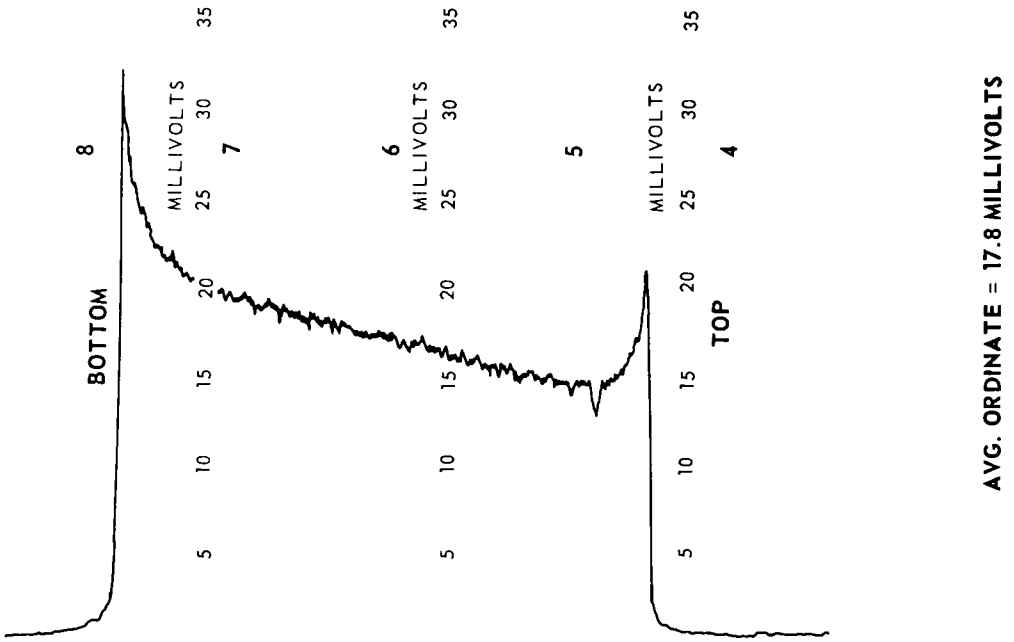
First, the area under the gamma scan curve can be readily obtained with a planimeter, or by other means. Then by using the same value for the length of the irradiated fuel column that was used to obtain the average heat flux for the capsule, one can calculate the exact position on the capsule at which this flux is being produced. Finally, heat flux at every other point along the capsule surface can then be established from the gamma scan curve to a point within about 1/4-inch of the ends of the fuel column. At this point heat flow is no longer one dimensional because appreciable heat transfer occurs to the end closure. This is evident in Figure 3 on the basis of the grain structure of the UO_2 near the end closure (top of picture). The effect of flux peaking is also very apparent from the grain structure and actual melting of the UO_2 near the top of the pellet.

This pellet is the uppermost one in a capsule designed to obtain higher heat fluxes from the capsule seen in Figure 1, and at the same time demonstrate unequivocally that melting of the fuel had occurred. A hollow ZrO_2 spacer, one inch long, was introduced in the center of the fuel column as can be seen in Figure 4. In this way flux peaking and increased power production is induced in the central portion of the rod, and hopefully a higher central temperature and thereby more extensive center melting would occur in the pellets on each side of the spacer.



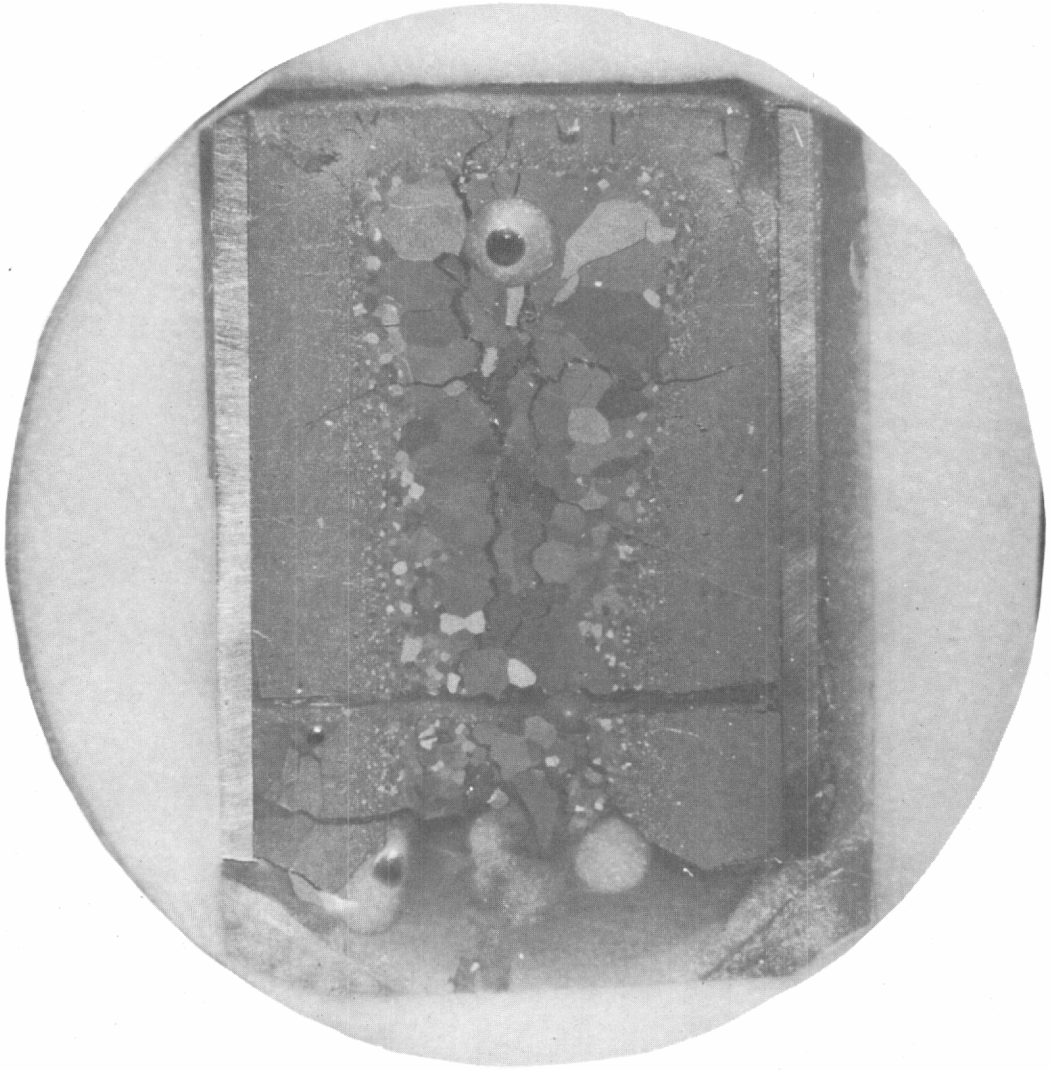
4.6.4

FIGURE 1. CAPSULE SCHEMATIC



DATE: 12/28/61
SLIT WIDTH: 0.020" × 0.750"
ROD SPEED: 0.38 in./min.
PAPER SPEED: 30 in/hr
DISCRIMINATOR LEVEL: 0.7 mev.
TIME CONSTANT: 2 sec
SCALE: 100,000 cpm = 25.0 millivolts
RESOLVING TIME: less than 20 u-sec.

FIGURE 2. EURATOM CAPSULE NO. 5
POST IRRADIATION GAMMA SCAN



Location: Pellet No. 8, Capsule No. 6

View: Axial Mid-plane

Surface Heat Flux: Instantaneous Maximum - 974,000 Btu/hr-ft2
Time Average - 932,000 Btu/hr-ft2

Preparation: Rough Polished (600 grit SiC), Oblique light

FIGURE 3. UO_2 AXIAL CROSS SECTION

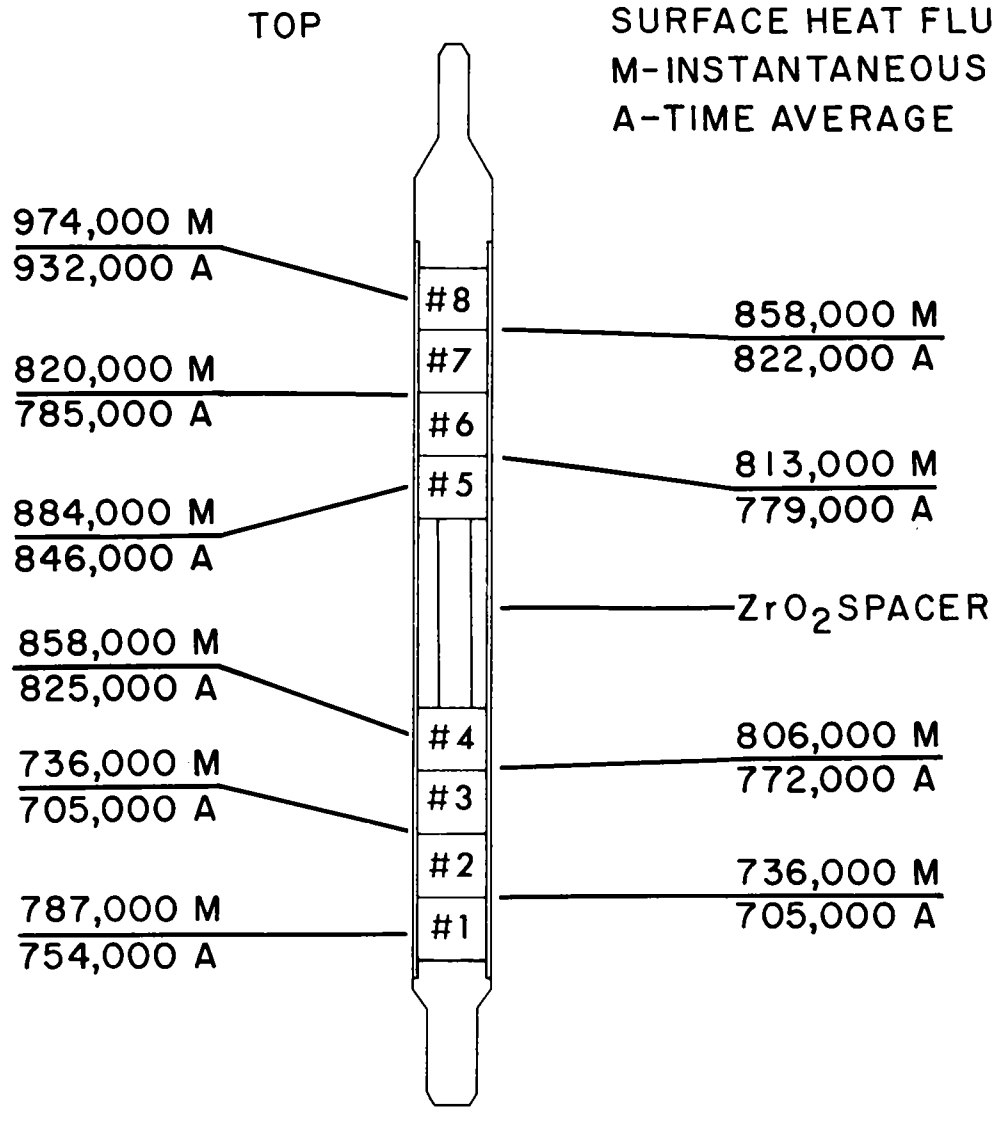


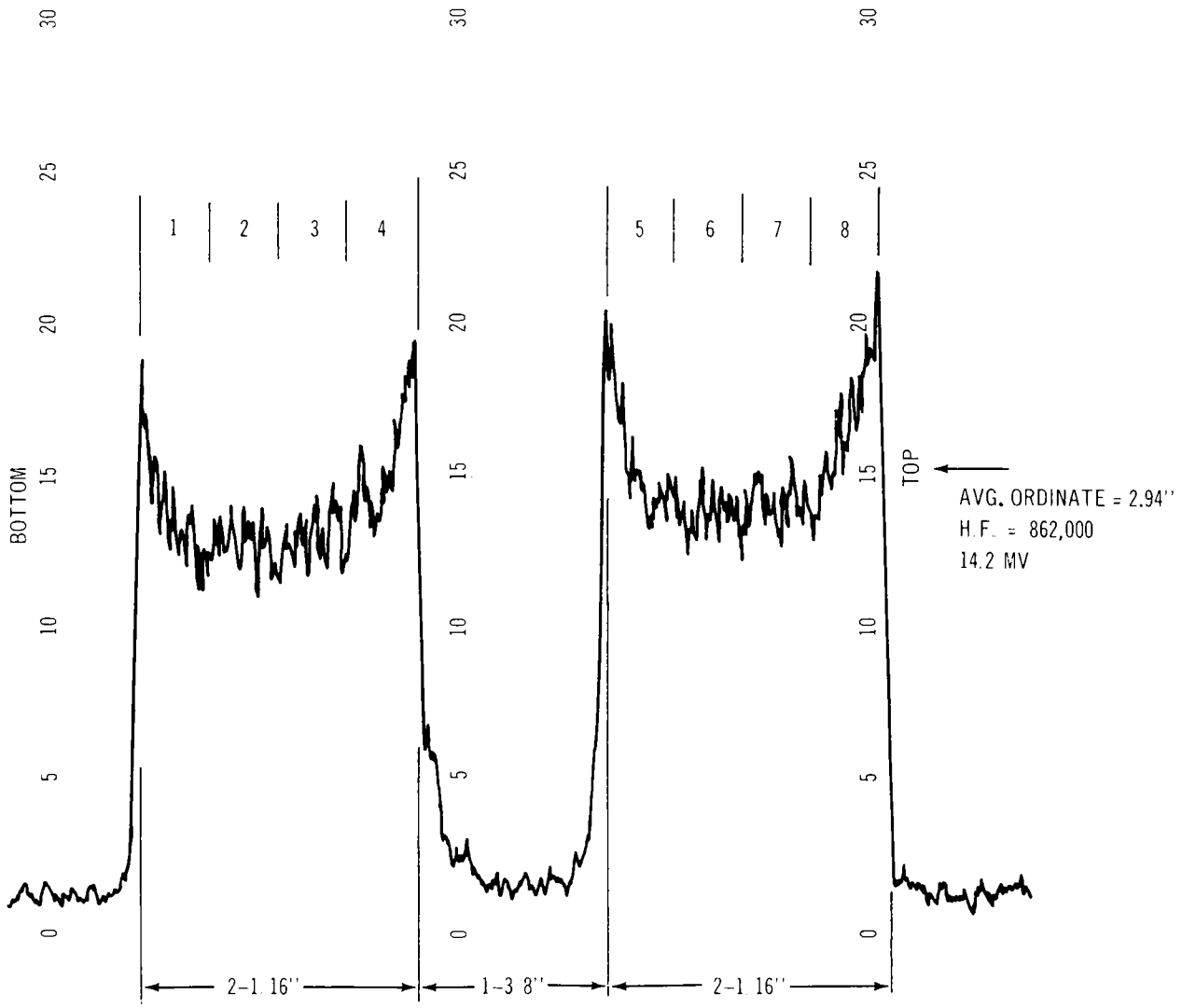
FIGURE 4. EURATOM CAPSULE No. 6 CROSS-SECTION

The gamma scan curve for this capsule is shown in Figure 5. That flux peaking did occur in the central pellets adjacent to the ZrO_2 spacer is readily apparent. This gamma scan was also used in the same was as described previously to obtain the surface heat fluxes shown in Figure 4.

A longitudinal section through the mid plane of the pellet above the spacer is shown in Figure 6. Although the heat production at the hottest section of this pellet is less than in the top pellet (Figure 3), the amount of center melting is greater because there is no heat sink similar to the end closure near-by. Heat transfer from the central region of the pellet is apparently through the portion of the fuel that has slumped into the hole in the ZrO_2 spacer. The widest point in the tear shaped void in the pellet defines the boundary of the molten zone that occurred in the pellet. The large-grained material that is oriented in the direction of the thermal path also was molten before freezing.

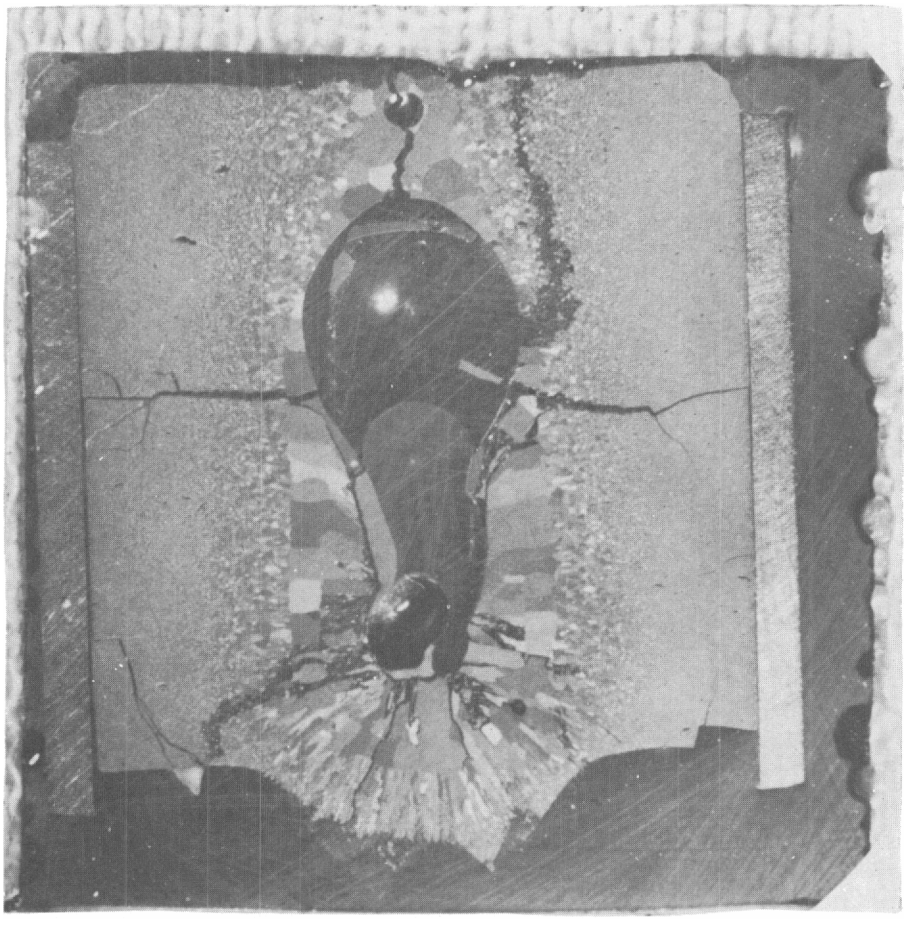
One therefore can assign a fixed temperature, the melting point of UO_2, to the boundary formed by these structures. The onset of visually detectable grain growth also can be used as a reference temperature in the analysis of UO_2 structures from a series of capsule irradiations, if the same capsule geometry, time of irradiation, temperature gradient within the fuel, and fuel composition are used in all of the capsules. One problem in using these points at which a physical change occurs in the oxide is the selection of an appropriate temperature for them. The melting point of UO_2 is taken as 2800 C. After a careful review and analysis of all out-of-pile UO_2 grain growth studies, a value of 1905 C (3460 F) is used for the temperature required to initiate visually, detectable grain growth during a 20 minute heating period. The point of detectable grain growth is best obtained from the photographs of the "as-cut" capsule cross sections. Although one is not able to see the actual change in the size of the grains, the boundary between the undisturbed UO_2 and that undergoing grain growth is clearly recognizable because of the difference in light reflection from the two regions.

In these short term capsule irradiations with highly enriched UO_2, the grains even in the hottest unmelted region remained equiaxed with practically no columnar grain formation. Under these circumstances the out-of-pile grain growth data are useful in obtaining a temperature profile for the UO_2 in the examined sections of fuel. Since the heat flux for each of the sections from the capsules is known on the basis of the combined calorimetry during irradiation and the gamma scan after irradiation, the means are at hand to calculate the thermal conductivity of the fuel over an extended temperature range, even at temperatures approaching the melting point. This indirect method of thermal conductivity measurement obviates the use of therocouples with their attendant problems, e.g., compatability with the fuel, capsule penetrations, and making the measurement in a very steep temperature gradient.



SOURCE TRAILCABLE = 6 UP SCAN
DATE 4-7-62
SCAN = 1
SCINTILLATION DETECTOR SWITCH 10
SLIT WIDTH 0 020 IN
ROD SPEED 0 19 IN MIN
PAPER SPEED 0 50 IN MIN
V 1800 (HIGH VOLTAGE)
D 3 40 (DISCRIMINATOR)
O 2 31 (OUTPUT)
T C 2 SEC
x 30 K (RANGE KNOBS)
5×10^3 CM 20 MV
INST DELAY t 10 y SEC

FIGURE 5. GAMMA SCAN-CAPSULE 6



745-17

LOCATION: PELLET #5, CAPSULE #6
VIEW: MIDPLANE
SURFACE HEAT FLUX: INSTANTANEOUS MAXIMUM – 884,000
AXIAL MIDPOINT TIME AVERAGE – 846,000
PREPARATION: ROUGH POLISH (600 GRIT SiC), OBLIQUE LIGHT
MAGNIFICATION: 8X

FIGURE 6. AXIAL CROSS SECTION

A reflux-type irradiation capsule with internal nucleate boiling for temperature control

J. M. ROBERTSON

Dounreay Experimental Reactor Establishment

Thurso, Caithness, Scotland

One of the problems of irradiation testing of fuel specimens in capsules in Research Reactors has been the precise achievement of the required temperatures in the specimen while under test in the reactor. The requirements are usually limited to some reference surface temperature on the specimen but are normally more important than those concerned with fuel rating. Two orders of magnitude may separate the fuel ratings in thermal reactors and fast reactors for example, but surface temperatures with a $\pm$ 10°C variation would normally be aimed at in tests of fuels at either rating. The problem of measuring the temperature at the required position and hence its control is normally impossible to solve satisfactorily and it is often necessary to rely on an estimate of the difference between the reference temperature and that measured at another more convenient point in the material surrounding the specimen. Even where the reference temperature can be measured, variations of temperature along the surface may be considerable and only a small part of the specimen may be at or near the reference temperature with the remainder unknown.

At the Dounreay Experimental Reactor Establishment, a range of capsules has been developed and used for testing fuel specimens at 5 to 500 w/gm ratings and surface temperatures up to 750°C in the Dido-type M.T.R. there. The design of these capsules has tended to become simpler with experience and specimen temperatures are normally controlled by altering their heat rating by moving the capsule vertically or laterally within the reactor. Ballast heating has also been used in large sodium-filled capsules for low-rated specimens but the added complication has made temperature control by alteration of the environment within the capsule less acceptable for large irradiation programmes. In the Dounreay Fast Reactor the main problem associated with irradiation testing is the difficulty of having any leads to an irradiation capsule and this makes direct indication of temperatures impossible.

Heat from the capsules suspended in the D_2O reflector

region in D.M.T.R. is mainly removed by subcooled nucleate boiling at their outer surfaces. The subcooling in this low-velocity region permits heat fluxes up to several hundred watts per cm^2 to be removed safely and easily. Boiling on capsules within the reactor core region is planned for later this year. Ad hoc tests on the burn-out characteristics of varieties of clusters of these capsules suspended in static water have been carried out in the D.M.T.R. laboratories and the limitations of this method of heat removal are still under investigation. The measurement of the heat output of the capsules under these conditions is difficult without some meter incorporated in the capsule and a simple design is being developed. Capsule surface temperatures can however, be taken to be below $125^{\circ}C$.

A helium annulus is normally incorporated in capsules for high-temperature specimens and the temperatures of internal surfaces are measured by sinking thermocouples into a thick copper layer built on to the surfaces. A sodium bath is also incorporated in some capsules but difficulty has been encountered in attaining the estimated temperatures and filling and assembly techniques require to be refined. A flowing-helium system to each capsule is being developed to ensure purity of filling gas and by monitoring the pressure and activity of the gas with equipment already developed at D.M.T.R., containment fractures will be detected.

A simple capsule which would achieve temperatures everywhere on the surface of a specimen within $\pm 10^{\circ}C$ throughout its irradiation would therefore be a considerable advance over the present imprecise and complicated equipment. It appears feasible that this can be achieved by using the unusual characteristics of the nucleate boiling process within the capsule. Nucleate boiling on a surface permits a very small difference between the surface and the local saturation temperature of the liquid, even although it may be sub-cooled, and this difference hardly varies with vast changes in heat flux on the surface. For example, a specimen nucleating in water will give a variation of temperature between (from one atm to the critical pressure) 5 and $35^{\circ}F$ above the saturation temperature over a range of heat flux from 10^3 to 10^5 Btu/hr ft^2 (Ref. 1). Similar boiling effects have been noted in Sodium (Ref. 2). A factor of two would be considered to be a large discrepancy in an estimate of heat rating so that the temperature on a surface could be guaranteed to be practically exact if the pressure were known and if the uncertain nucleating properties of the surface did not change during irradiation. In comparison, heat transferred by conduction in a conventional capsule would cause a surface-temperature variation directly proportional to that affecting heat rating.

The proposed design of capsule is shown schematically in Fig. 1. The specimen, for convenience, is assumed to be a slender cylinder but could be a cluster of cylinders or any

ether configuration preferably of constant cross-section. It is completely immersed in a liquid (for example sodium) and retained approximately equidistant from the internal walls of the capsule. The two walls of the bottom portion of the capsule are thermally insulated and the two walls of the top portion have essentially little thermal resistance. The void in the top is initially filled with a non-reactive and incondensible blanket gas.

The capsule could be placed in a moving or static coolant where the heat from the specimens is dissipated to the coolant on the outer surface of the top part.

As the reactor power is increased at start-up, the heat from the specimen does not escape from the liquid bath until it is boiling. Natural convection in the blanket gas will dissipate an increasingly larger fraction of this heat to the walls of the top part but the temperature of the liquid will increase to the saturation temperature associated with the total pressure of the gas and vapour. The control of this pressure is discussed later. At some reactor power the heat from the specimen will be removed by evaporation of the liquid and by nucleate boiling on the surface of the specimen. The vapour will be condensed on the walls of chamber and will run back to the bath slightly subcooled. Any further increase in reactor power will not alter the surface temperature by more than a few degrees above the saturation temperature.

The mass flow of vapour bubbles to the surface will increase as the heat output rises. The down-flow of liquid to replace this vapour by mass balance, will also increase until the well known phenomenon of flooding (or cessation of liquid flow) of such a counter-current system is reached at the top of the specimen. This critical situation will limit the total heat output permitted from the specimen for liquid starvation will form a continuous vapour skin on the specimen to cause burn-out. Flooding is strictly a hydrodynamic phenomenon and is described for a similar configuration by Griffiths (Ref. 3).

This critical point is predictable with various liquids and recently at D.E.R.E., experiments using sodium generally confirm this approach. The superficial velocity of the vapour phase, leaving the annulus at the top of the specimen is governed by the total heat generated and the latent heat of the liquid and the geometry there. The thickness of the sodium annulus necessary and length of the specimen are therefore limited.

Very short specimens would of course exhibit a different mode of burn-out, for the vapour current normal to the surface of the specimen would be the limiting factor. This would

approach pool-boiling conditions where the burn-out surface fluxes are much higher than for the restricted conditions with the long specimen previously described. Nevertheless it appears that it may be feasible to cater for specimens several feet long.

Achieving the gas pressure to give the design saturation temperature in the presence of the condensing vapour is the most obvious difficulty. If an external line can be led to the capsule in the reactor, control is straightforward for the pressure can be controlled externally. Where this is undesirable, the deviation from the design pressure appears to be proportional to the effectiveness of the condensing system and this will be the most critical part to design.

The design can be modified to cater for very large power outputs from a specimen or cluster of specimens. The insulating walls surrounding the liquid bath can alternatively be made conducting. Heat can therefore be removed from the specimens by natural convection to these walls until the liquid is at saturation temperature and the net-evaporation process already described occurs. Subcooled nucleate boiling on the specimen will have begun before this point. Both processes now remove heat simultaneously to the coolant outside the capsule.

Experiments at D.E.R.E. on such a capsule are planned and before using liquid metals, tests with water and air will determine the characteristics of the reflux system.

References

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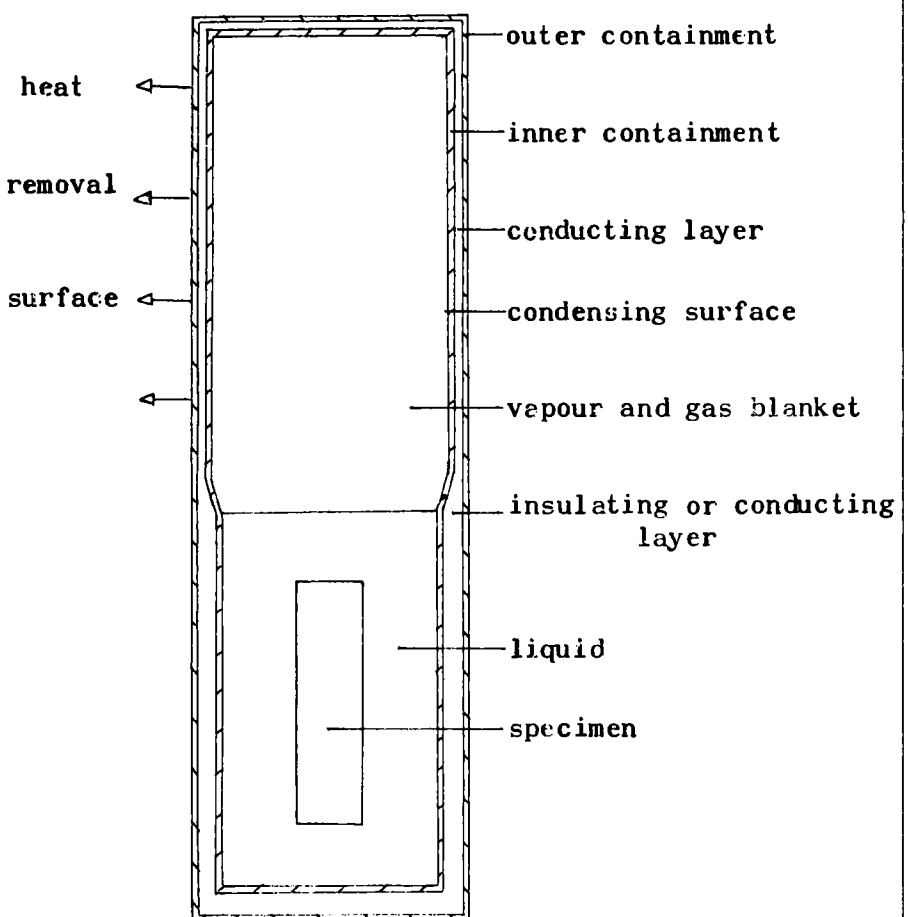
2. Noyes R.C. NAA-SR-6769, 'An experimental study of sodium pool-boiling heat transfer'

3. Griffith P., Schumann, W.A. and Neustal, A.D. Proceedings of the Symposium on Two Phase Flow. Feb. '62 London 'Flooding and Burnout in Closed-end Vertical Tubes'



Scheme of Irradiation Capsule with Internal Boiling

Fig. 1

SODIUM CAPSULE HEAT TRANSFER

D. Clough
A.E.R.E. Harwell

Introduction

1. Over the past few years the bulk of fissile irradiation experiments at Harwell have been conducted in sodium-filled electrically heated capsules positioned in the light water cooling jackets. In many irradiations in these capsules it has proved impracticable to insert temperature measuring probes directly into the specimen because of compatability problems and the difficulty of sealing probes into the capsules. It has, therefore, been necessary to infer fuel temperatures from thermocouples placed in the sodium bath.

2. Calculations to find temperature distribution in specimen temperatures are subject to inaccuracies because of

(a) convection occurring in the sodium; the irregular geometry close to the specimen makes it impossible to calculate this contribution;

(b) end losses which create axial gradients in the sodium. Here again, the geometry makes calculation difficult.

Laboratory work was, therefore, performed to provide practical data of the temperature distribution in the capsules for the particular geometries now being irradiated. This information should enable not only gradients and specimen temperatures to be calculated but, by suitable choice of thermocouple position, heat outputs to be known accurately.

3. It is, therefore, the practice to install thermocouples into these capsules to serve two separate purposes,

(a) to measure sodium bath temperatures as close to the fuel specimen surface as possible. From these measurements, coupled with the thermal simulation data obtained in the laboratory tests, it is possible to obtain estimates of the temperature and longitudinal temperature distribution in the specimen;

(b) to measure sodium bath temperature at some position in the sodium bath which is equally temperature sensitive to both heat sources, i.e. specimen and heater. This thermocouple then affords accurate measurement of the heat output from the specimen, provided care is taken in pre-irradiation, calibration and the gamma heating contribution is known.

Laboratory Simulation Tests

4. Figure 1 shows a modified PLUTO 7V capsule. This differs from the inpile capsule in that the specimen is replaced by a simulation mineral insulated cable heater and extra thermocouple pockets have been added to enable longitudinal scans to be made at different positions in the capsule. Simulated specimen fluxes up to 200 watts/cm^2 have been achieved in these tests, using nichrome or mineral insulated cables. Scans were made of the capsules with removable thermocouples to obtain the temperature distributions for various power levels of the capsule heater and simulated specimens, Figure 2.

Summary

5. From these laboratory tests it was possible to define positions in the sodium which would satisfy the condition laid down in paragraph 3 (b), and also assess the likely longitudinal gradient on the specimen. Assessment of the fuel surface temperature from a measurement in the sodium bath is more difficult and depends both on the accurate placement of the thermocouple and knowledge of the specimen coolant interface temperature difference. Work on this latter aspect is progressing and limited data from laboratory tests and actual measurements in pile of uranium/sodium interface up to 170 watts/cm^2 indicate that the interface effect is small

However, the effect of fission gas release, especially on interface conditions, at high burnup levels, is still uncertain. In some tests where the fuel element has had to be clad there is the added difficulty of assessing the magnitude of the fuel cladding temperature difference, which is the largest source of error.

6. Finally, it has been found with accurately calibrated capsules, it is possible to obtain gross heat outputs to $\pm$ 10% generally, and under special arrangements to $\pm$ 5%. In this respect the use of an integrating watt meter affords a means of measuring burnup provided the other conditions are satisfied. It should be emphasised that to obtain high accuracy care must be taken to ensure that

(a) losses in power leads are accounted for;

(b) coolant water temperature is constant or known;

(c) thermocouples are accurately positioned and calibrated;

(d) the heat transfer gas characteristics remain unchanged during the period of the test.

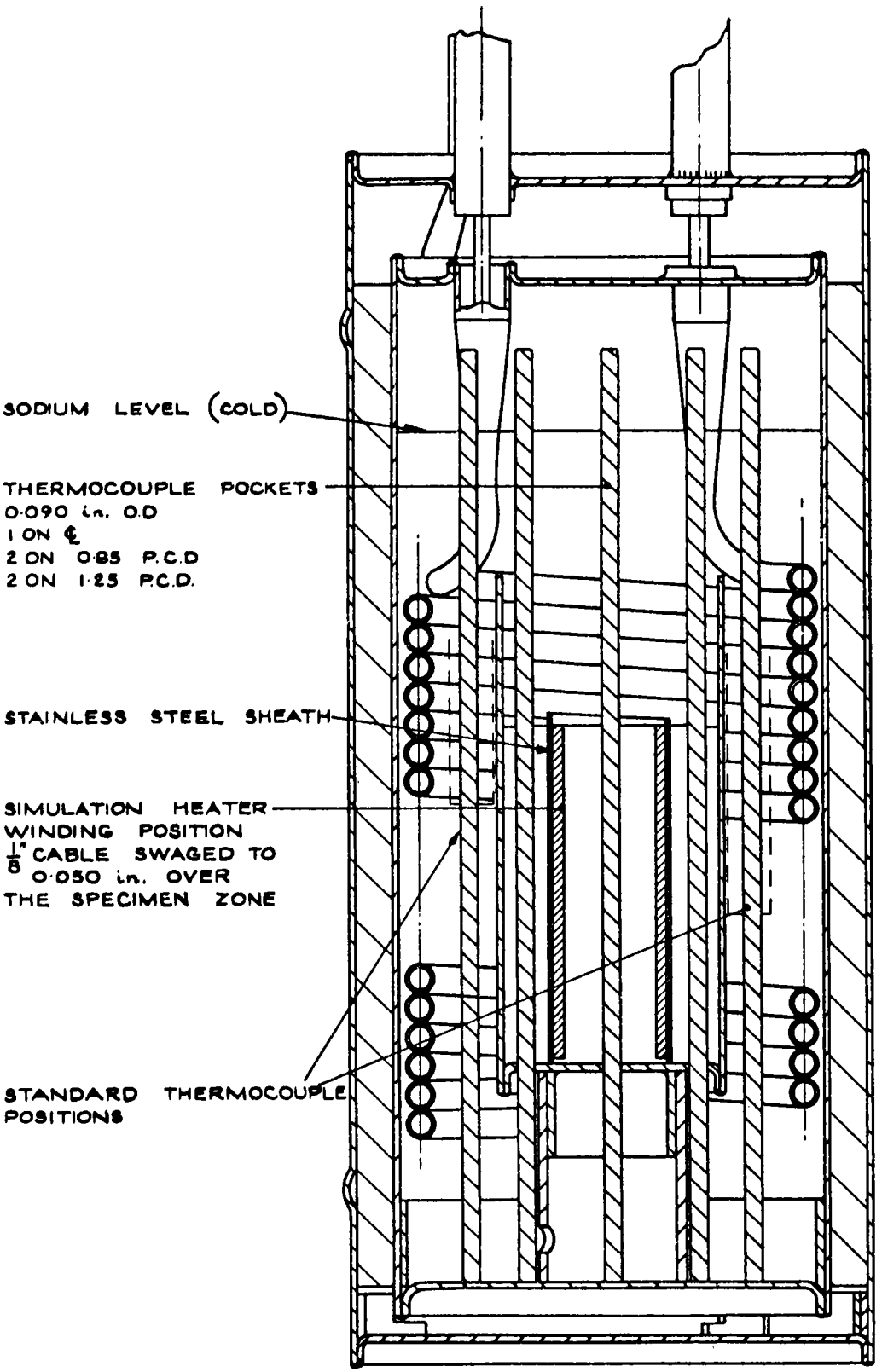


FIG. 1. MODIFIED PLUTO 7V CAPSULE.

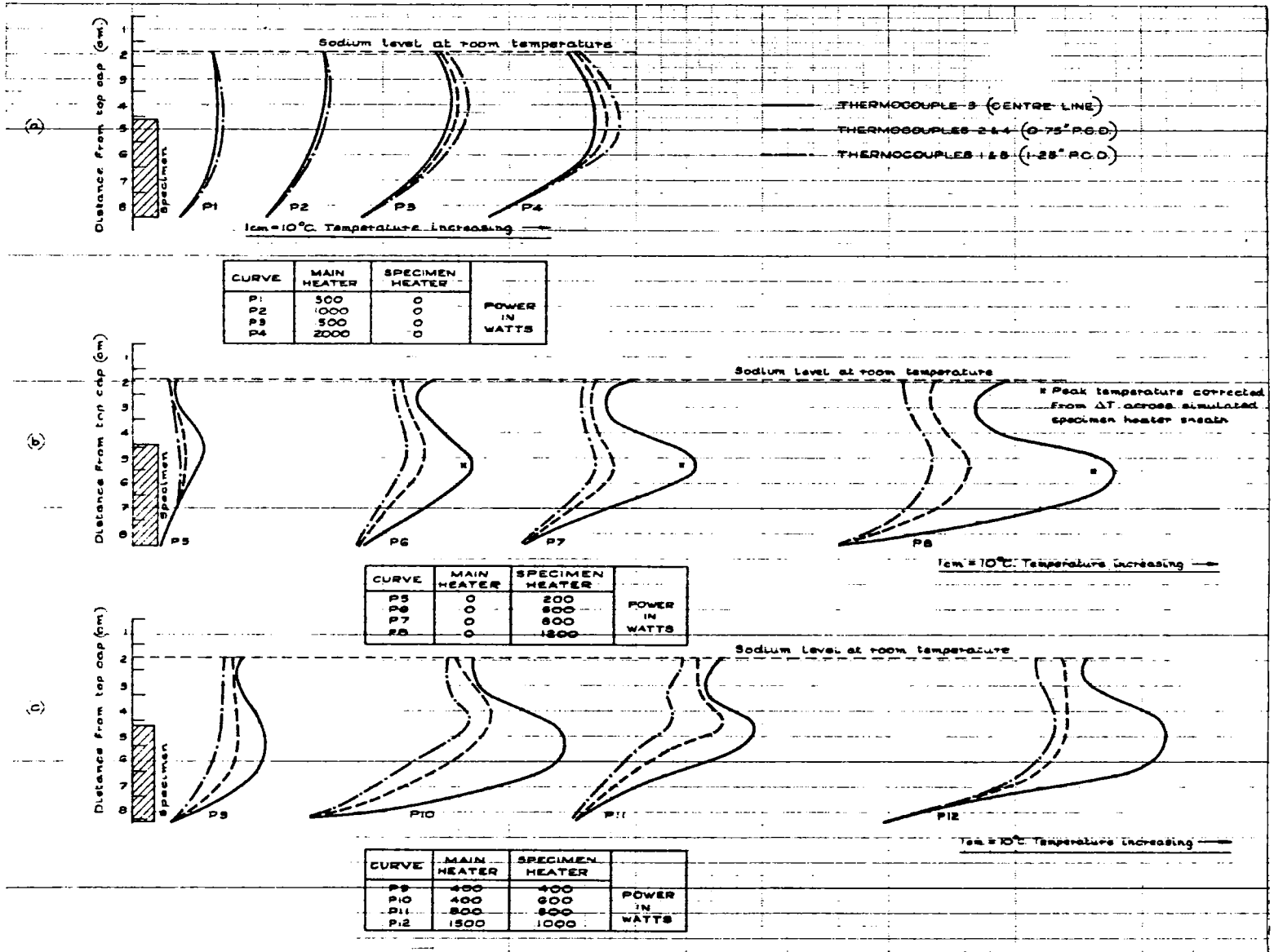


FIG. 2. TEMPERATURE DISTRIBUTION IN THE PLUTO 7V CAPSULE.

LABORATORY EXPERIMENTS ON SODIUM HEAT TRANSFER

E. Garrett and I. H. Mac William
A.E.R.E. Harwell

Test Progress and Capsules

For irradiation tests on fissile materials the specimen is usually situated within a double containment, the inner being sodium-filled for efficient heat transfer. With the higher fuel ratings now envisaged it is required to investigate heat transfer under conditions approaching the onset of boiling in the sodium and to verify that operating conditions in-pile will not produce burn-out.

Laboratory experiments are being carried out as part of this investigation and the work is divided into three experimental stages. The first two investigate heat transfer to a sodium pool surrounding a central rod heater and the third is concerned with the properties of narrow sodium-filled annuli.

The inner assembly for the first, or Mk I, experiment is shown in Fig. 1(a) The heater is a double coil of mineral-insulated sheathed cable fitted inside a stainless steel sleeve carrying thermocouples as shown. A surrounding cage carries corresponding thermocouples for the measurement of sodium temperature approximately 1 mm from the heater surface and, in practice, these are positioned by wiring them to the cage. The heater containment is sodium-filled to prevent hot-spots and permit maximum dissipation. A reservoir is fitted above the support tube to allow for sodium expansion.

The Mk II inner assembly is very similar but employs a heater of the concentric tube type. Thermocouples are situated along external grooves in the heater and there is a surrounding cage similar to Mk I. This type of heater does not require sodium filling.

For both Mk I and Mk II, the inner assembly is welded into a sodium containment simulating the standard 7V specimen can and fitted with the usual control heater. A tube through the top plate enables the atmosphere above the sodium pool to be controlled. The sodium containment is mounted inside a cooling jacket leaving an annulus which is gas-filled to control the radial heat transfer to a water-cooling coil wound outside the jacket. The complete apparatus for the Mk II experiment is shown in Fig. 1(b).

The Mk III experiment is for the investigation of heat transfer through sodium-filled annuli of widths ranging from ·015" to ·100". The inner assembly shown in Fig. 1(b) consists of a capsule designed around a concentric-tube heater carrying thermocouples as before, the corresponding thermocouples reading annuli temperature being fitted through the capsule wall. Preliminary tests have established that thin annuli can be satisfactorily filled with sodium without leaving voids. The Mk III assembly will incorporate a tube at the top of the capsule for controlling and monitoring the atmosphere above the sodium. Each capsule will be welded into a sodium containment as before and mounted inside the water jacket assembly.

Instrumentation

The instrumentation is simple and includes 12 chromel/alumel thermocouples distributed as shown. The noise level generated by the boiling processes is detected by an accelerometer feeding through a pre-amplifier, narrow pass-band filter and amplifier and displayed visually on an oscilloscope and voltmeter. The two heaters are controlled manually via variacs with arrangements for voltage, current and power to be recorded. The helium blanket pressure is monitored via a bridge-type pressure transducer and displayed on a potentiometric recorder.

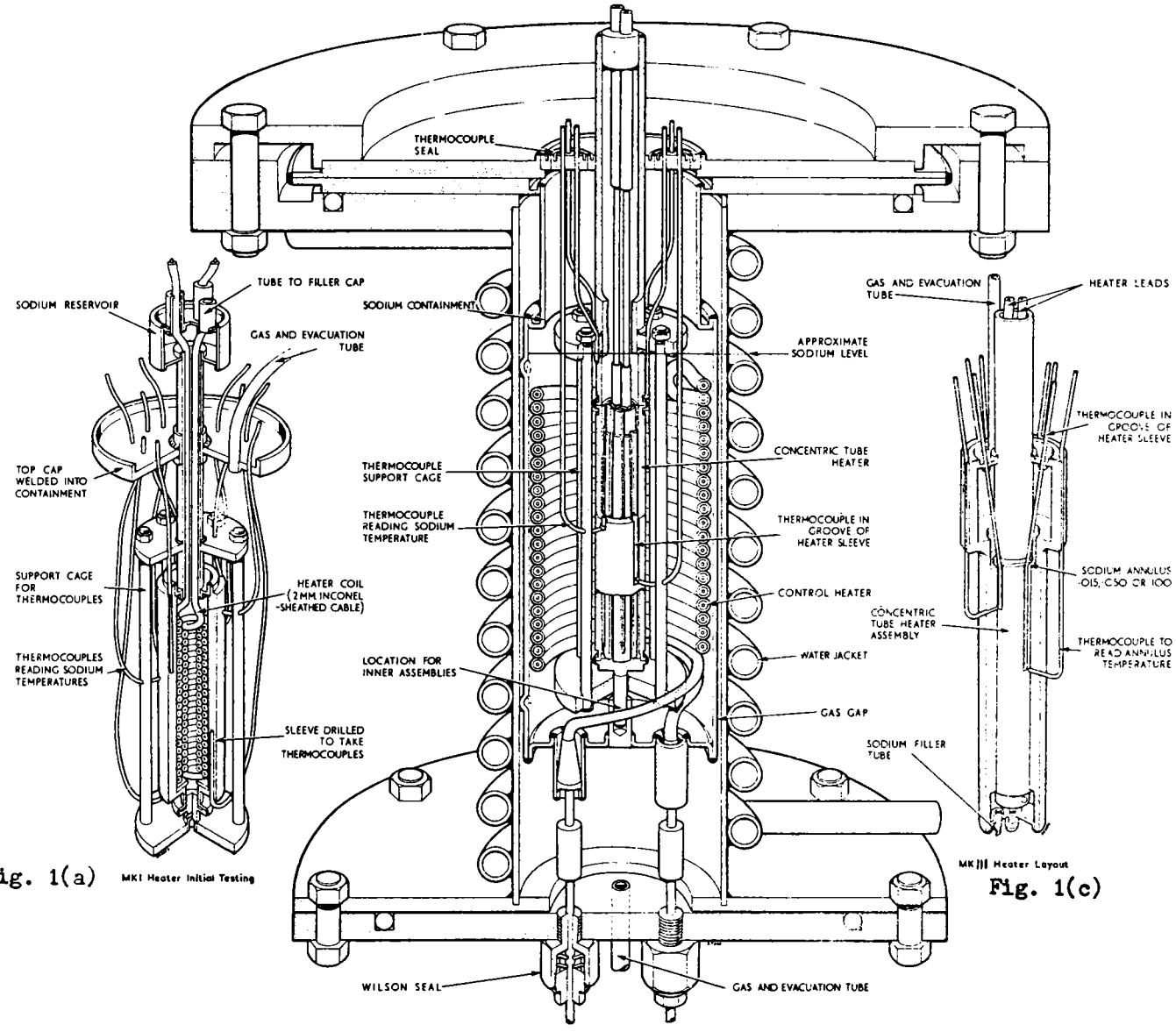


Fig. 1(a) MK1 Heater Initial Testing

Fig. 1(b) Sodium Boiling Test Rig

Fig. 1(c)

THERMAL DIFFUSION RIG

E. Garrett
A.E.R.E. Harwell

Irradiation tests on fissile specimens are carried out under carefully controlled temperature conditions and these are normally maintained irrespective of the state of the reactor by means of auxiliary electric heaters. An alternative method of control which has been proposed, offering an attractive simplification in rig design and pile-cap services, is by means of controlled radial heat transfer through a gas gap. It is proposed that the thermal conductance of the gap be controlled by introducing a binary gas mixture, one component of high thermal conductivity and the other low, and the mixture proportions varied to adjust the gap conductance.

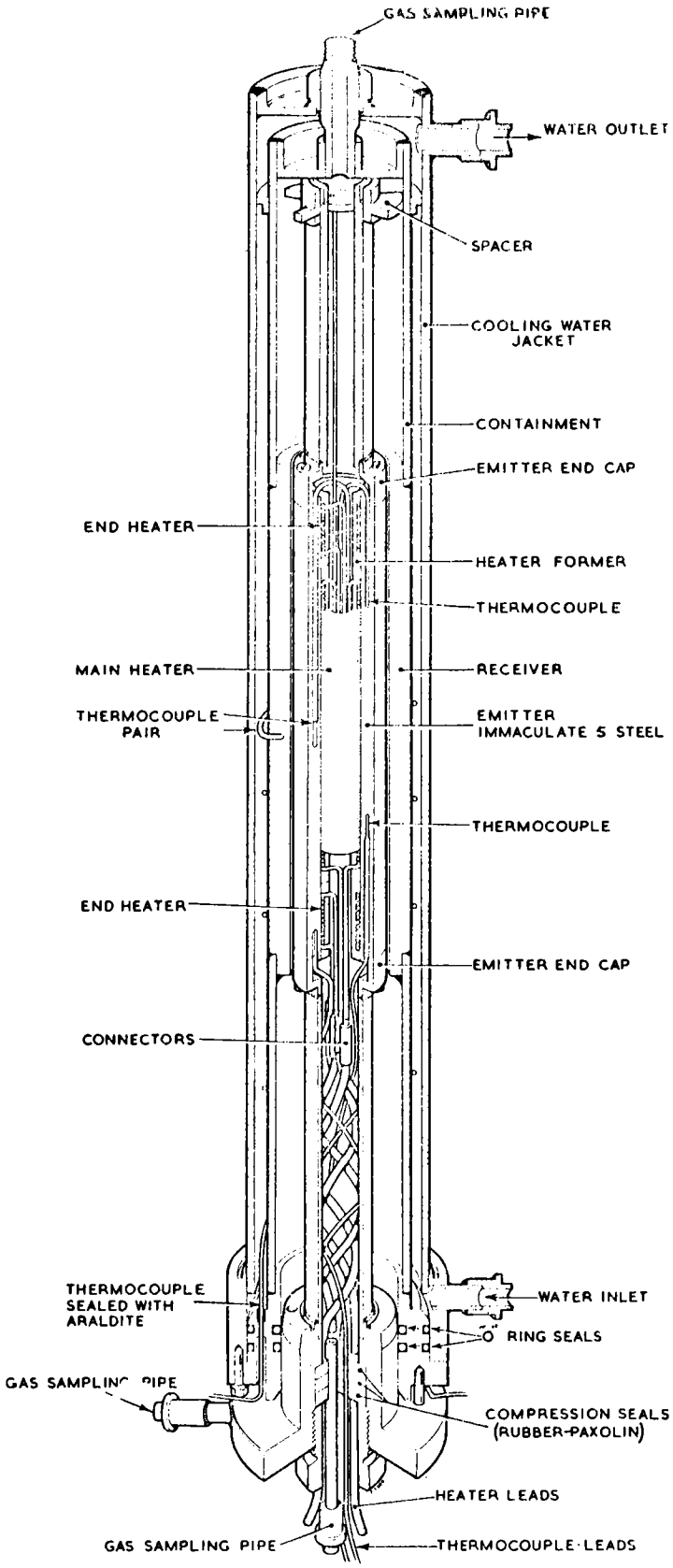
The laboratory experiment was carried out to determine whether a typical gas mixture (in this case, helium and neon) within a 2V irradiation rig would be liable to effects which disturb temperature equilibrium over a period of time. Particularly in mind was the phenomenon of thermal diffusion which would tend to concentrate neon at the thimble wall and helium at the inner (heated) surface.

The apparatus, illustrated in Fig. 1 , consists of a cylindrical 'emitter' assembly located concentrically within a 'receiver' (simulating a 2V rig thimble) outside of which is a water cooling jacket. The emitter consists of a sleeve of Immaculate 5 steel carrying 12 thermocouples situated in longitudinal holes, as shown, and fitted over a former wound with sheathed heater cable. The heater consists of 2 independent end coils in addition to the 2 main windings to provide a more uniform longitudinal temperature distribution.

The receiver, which carries a set of thermocouples corresponding to those in the emitter, was designed to enable the gas gap to be increased by machining at stages during the experiment. The gas gaps chosen for investigation were ·010", ·080", and ·140", using gas mixtures of 50% Ne/He, 75% Ne/He, and 100% Ne. The gases were mixed in the external gas circuit before admission to the evacuated rig, and samples were taken before and after each run for analysis by mass-spectrometer.

Each test run was normally over 24 hours approximately, the power input being maintained constant and thermocouple readings taken every hour. Emitter temperatures for a particular run were either 300°C, 400°C, or 600°C, the receiver temperature being maintained at around 50°C.

Examination of the results obtained has shown no evidence of instability in any of the cases examined, implying that temperature control by gas mixtures should be sufficiently stable for in-pile experimental rigs and that, within the scope of this investigation, there is nothing to discourage its practical application.



Thermal Diffusion Rig

RECENT OPERATING EXPERIENCE WITH CAPSULES USING VARIABLE GAS MIXTURES FOR TEMPERATURE CONTROL

by

C. C. Crothers
Argonne National Laboratory

As a part of a research program the Metallurgy Division of Argonne National Laboratory is currently investigating the effects of neutron radiation upon various reactor fuels. Several different instrumented capsule designs have been used successfully during the course of the program.

A series of these capsules are presently being irradiated at the MTR. The capsules are all of essentially the same design which employs as a means of temperature control a gas annulus containing a flowing mixture of nitrogen and helium. Since the thermal conductivity of nitrogen is only about one-fifth that of helium the thermal resistance across the gas annulus may be controlled by varying the ratio of nitrogen and helium in the gas mixture. The temperature of the fuel material being irradiated is in turn controlled by regulating the ratio of nitrogen and helium in the gas annulus. The capsules described here are modifications of a capsule which was irradiated at the MTR in 1962 for General Atomics.

This capsule design has a number of definite advantages. The need for electrical heaters has been eliminated. A wide temperature

control range is possible and the auxiliary equipment necessary for the operation of the capsule is economical and simple.

For several reasons it is difficult to calculate accurately the temperatures within encapsulated fuel specimens after the capsule has been inserted in a reactor. Cumulative errors in the calculation of perturbation, attenuation, and heat transfer contribute to this difficulty. Although the thermal neutron flux for specific position in the reactor may be previously determined with reasonable accuracy for a particular cycle, these values may vary appreciably from cycle to cycle. The effective neutron flux supplied to the capsule depends greatly upon the reactor core loading. It is for this reason that a wide temperature control range is important.

The first of this series of capsules was inserted in the MTR in October of 1962. The capsule consisted of an 18 in. long EBR-II plutonium-fissium fuel pin, a tantalum cup, a gas annulus, and an outer stainless steel can. The fuel pin was mounted inside the tantalum cup which was provided to contain the fuel in the remote case of fuel meltdown. The fuel pin was instrumented with eight stainless steel clad Cr-Al thermocouples. The annular spaces between the fuel pin the the nickel can contained NaK. Figure 1 is a diagram of the capsule.

The consoles and auxiliary equipment for use with this series of experiments are functionally identical and completely interchangeable. The nitrogen supply system consists of a nitrogen cylinder and reducing regulator, a two-way solenoid valve, a dehydrating filter, a pressure switch, a second reducing regulator, a flow meter, a bronze filter, and

necessary valves. The dehydrating filter assures that no moisture or particulate matter enters the system. The pressure switch protects the experiment against loss of gas supply. If the gas supply pressure decreases to 20 psi or less, an annunciator is activated. The second reducing regulator and the flow meter are used to accurately control the flow of gas to the capsule. The bronze filter prevents particulate matter from entering the capsule.

The helium supply system is identical to the nitrogen supply system with the following exceptions: An auxiliary helium supply cylinder is provided to be assured that a dependable helium supply is available at all times. A three-way solenoid valve enables the rapid selection of either cylinder. The helium system includes a tubing line which by-passes the second reducing regulator and the flowmeter. This line contains a two-way solenoid valve and in the case of excessive capsule temperature the valve opens automatically applying 20 psi helium to the helium inlet line of the capsule. The resultant effect is an immediate reduction of capsule temperatures.

The gases are mixed in the upper portion of the capsule and flow downward through the annulus. The gases leave the capsule through a vent line which extends from the bottom of the capsule out of the capsule and to the MTR vent system. A radiation monitor is installed near this vent line to provide warning in the case of structural failure of the capsule. Figure 2 is a schematic diagram of the gas flow system.

The instrument-electrical system automatically provides alarms for low nitrogen pressure, low helium pressure, high radiation level, and

high temperature. In the case of high capsule temperature the nitrogen supply valve is closed, the helium by-pass valve is opened, and the auxiliary helium supply valve is opened. Figure 3 is a schematic diagram of the instrument-electrical system.

During the first capsule irradiation a test was conducted to determine experimentally the temperature control range of this capsule. Quite unexpectedly it was discovered that the capsule temperatures could be controlled with helium only in the gas mixture. By varying the helium flow rate between 0 and 50 cc/min a corresponding change of about 200°C in capsule temperatures resulted. Using both helium and nitrogen for temperature control resulted in a range of slightly over 300°C. Figure 4 is a plot showing capsule temperatures for various helium flow rates and Figure 5 is a plot showing capsule temperatures for various helium and nitrogen flow rates. Tests of this nature on subsequent experiments yielded similar results.

Three other capsules of this type are under irradiation in the MTR. Each of these capsules is identical in operating principle to the one described. One of these capsules contains three EBR-II fuel pins. It has operated quite satisfactorily for the past six months. Of the remaining two capsules being irradiated, one contains a single EBR-II fuel pin and the other contains six separate ceramic fuel specimens mounted vertically in the capsule. The fuel specimens contain PuC and UC-20w/o PuC. All of these capsules are operating satisfactorily. The original capsule design is easily modified to accommodate different fuel specimens.

Although the gas flow control systems for these capsules are presently manually operated, very little attention is required to maintain

any desired temperature in the control range. Operators normally check the experiments once every eight hours. Experience has shown that with this amount of attention the desired temperature may be held to within ±10°C. With automatic control of the gas mixture temperature control would be even better.

In summary, the desirable features of this capsule are simplicity, wide temperature control range, and versatility. One inherent disadvantage of the capsule is found in the "heat dam" characteristic of the gas annulus. The design may be undesirable when a high burnup rate is important. Our experience thus far with this type of capsule has indicated that it is ideal for most fuel irradiations requiring temperature control and instrumentation.

Acknowledgement and appreciation are expressed to Dale E. Johnson and co-workers of General Atomics for providing the necessary data for the design of these experiments.

Figure 1

DIAGRAM OF TEMPERATURE CONTROLED CAPSULE

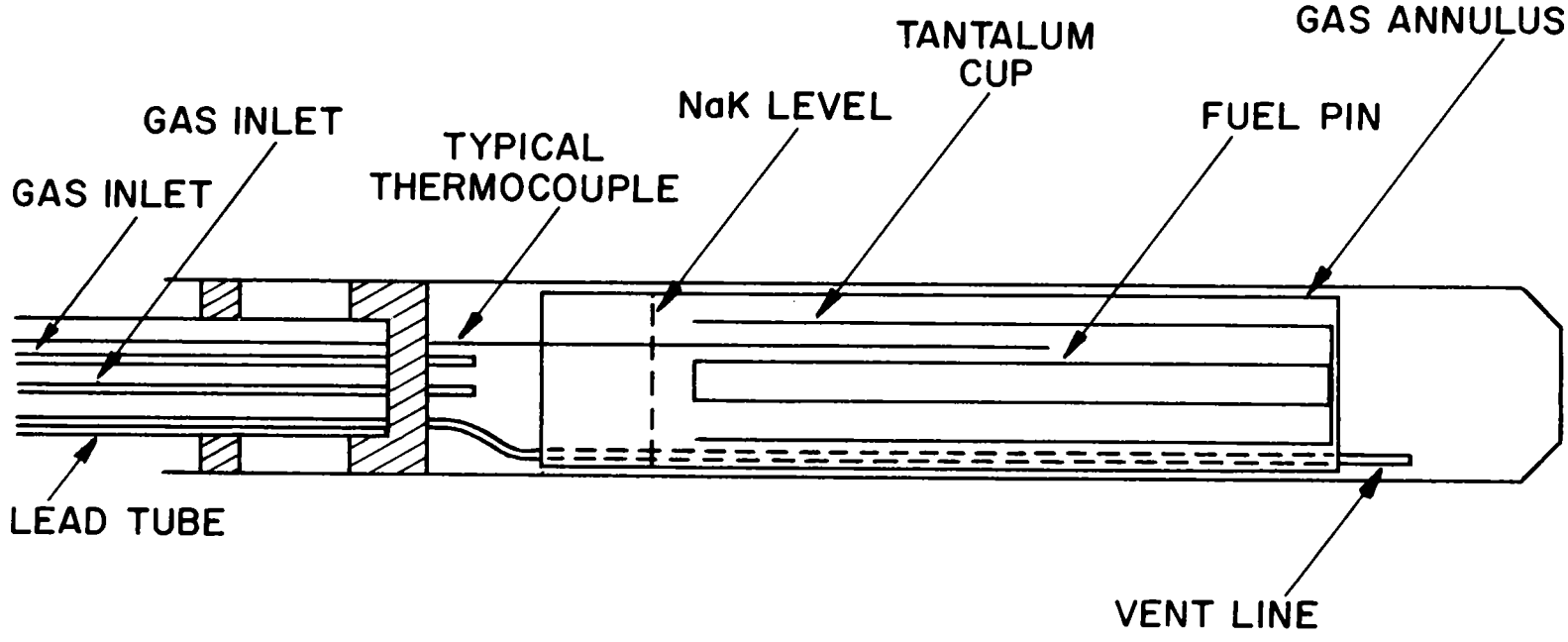


Figure 2

GAS FLOW SCHEMATIC

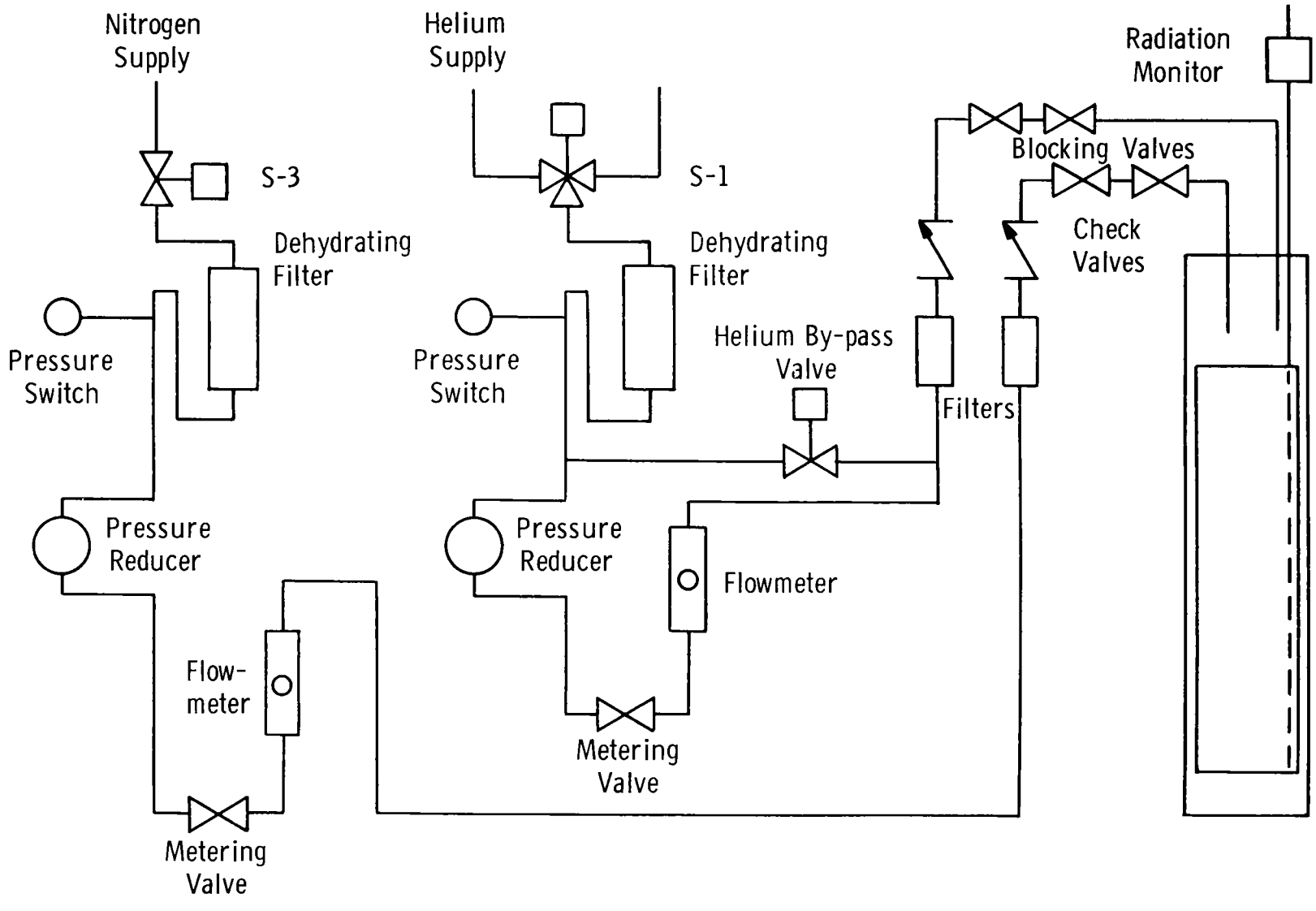


Figure 3

WIRING SCHEMATIC

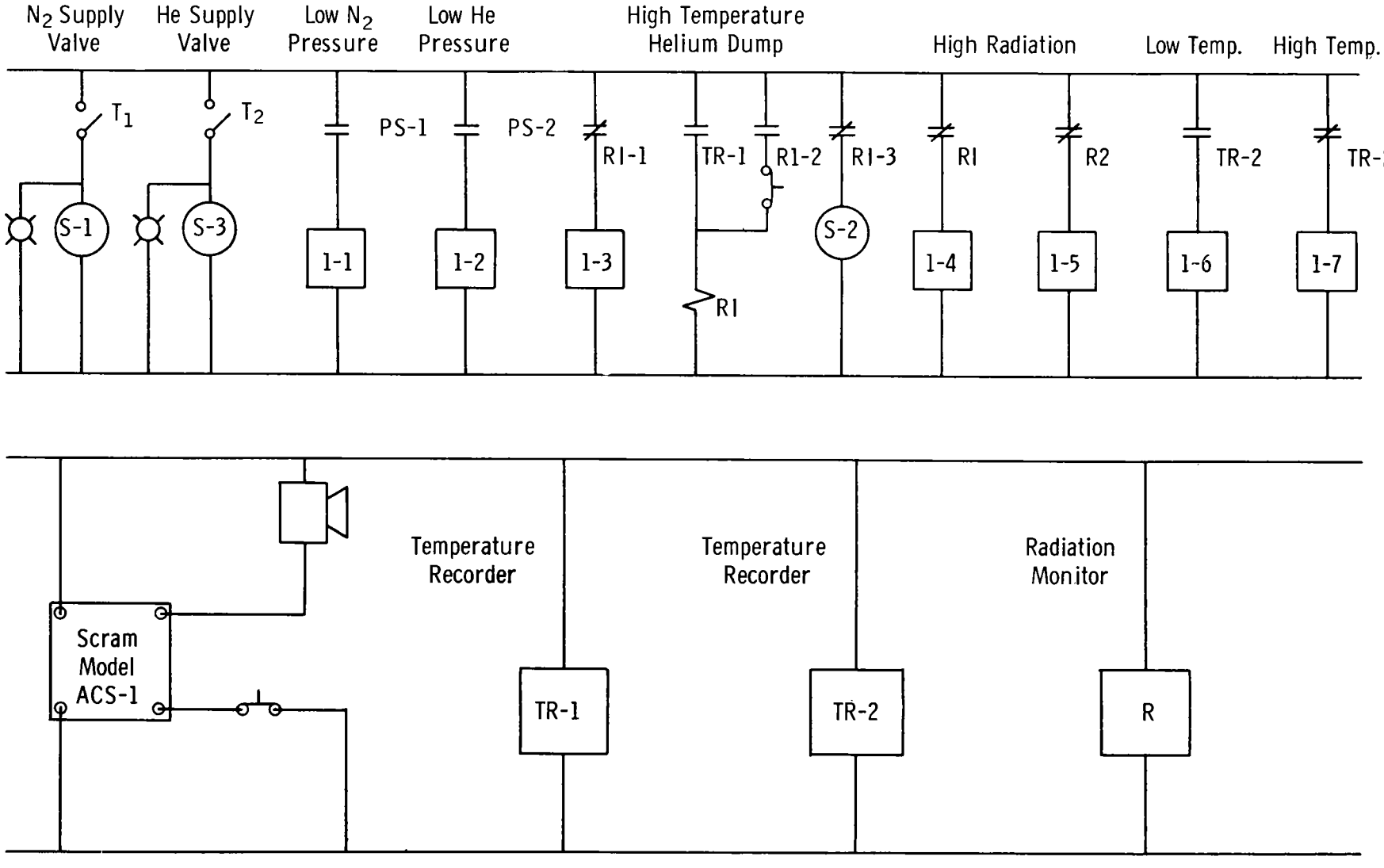


Figure 4

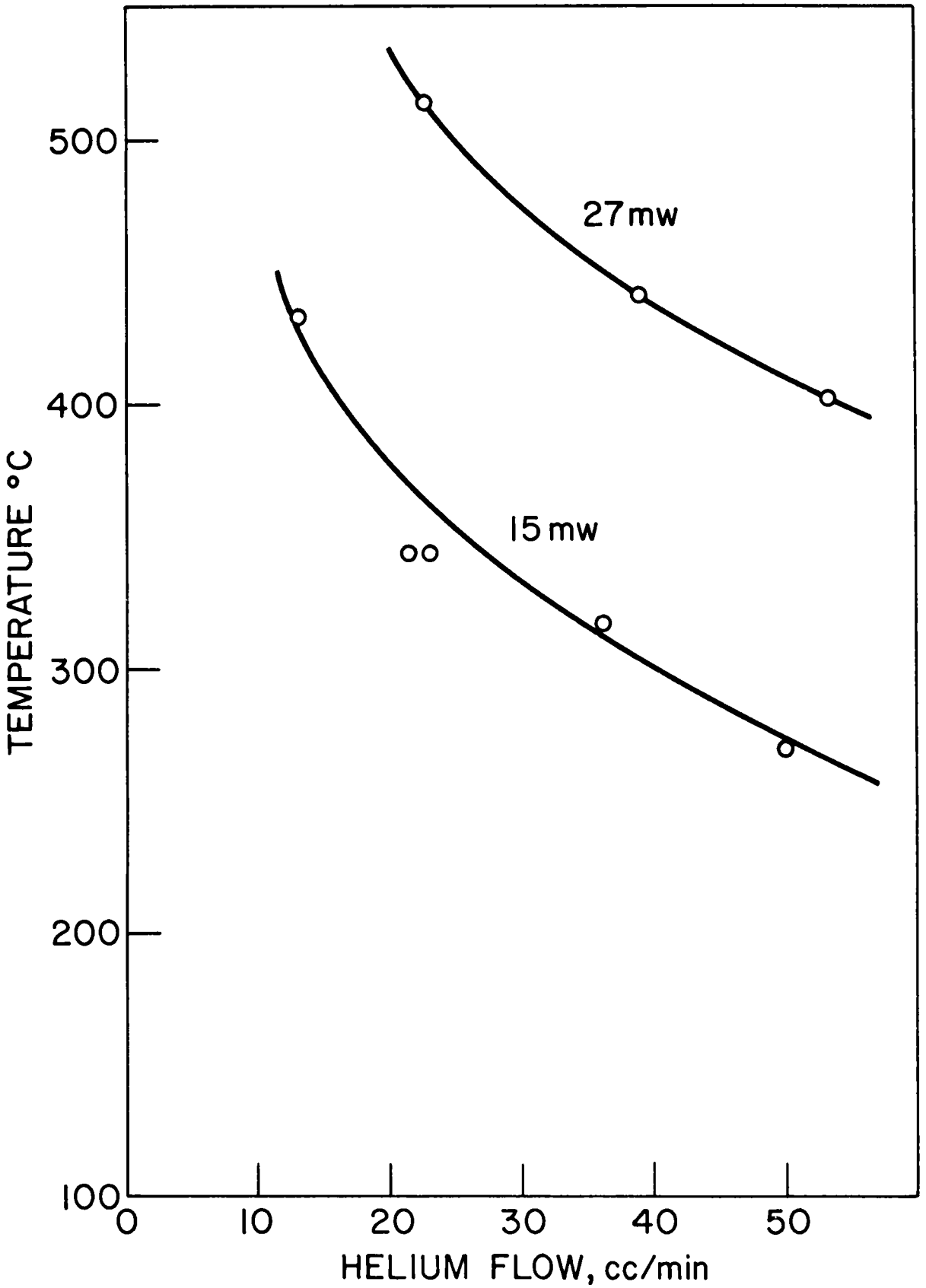
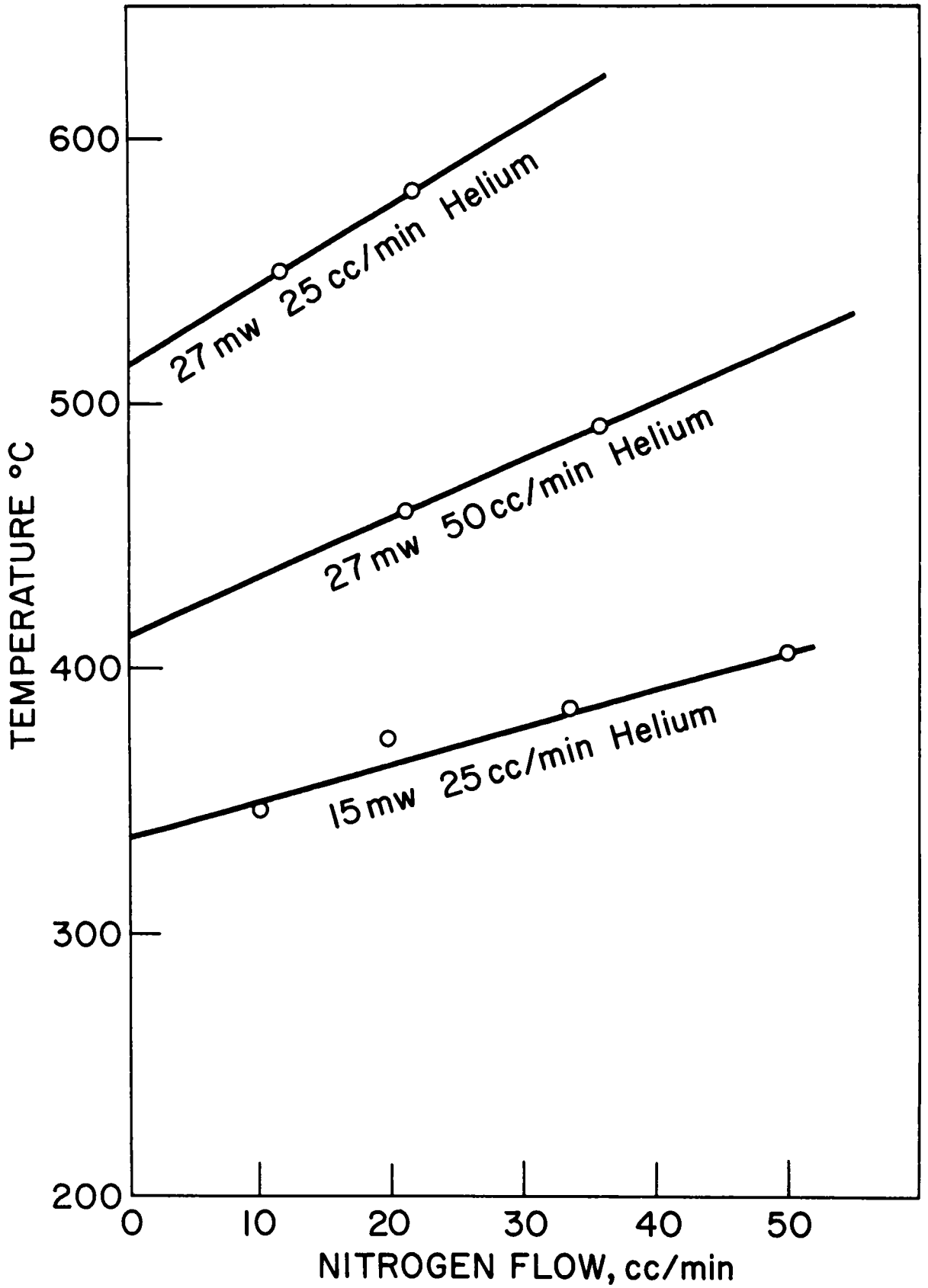

500
400
300
200
100
TEMPERATURE °C
27 mw
15 mw
0
10
20
30
40
50
HELIUM FLOW, cc/min

Figure 5


600
500
400
300
200
TEMPERATURE °C
27 mw 25 cc/min Helium
27 mw 50 cc/min Helium
15 mw 25 cc/min Helium
0
10
20
30
40
50
NITROGEN FLOW, cc/min

CAPSULE HEAT TRANSFER

R. A. Pugh
Nuclear Materials and Propulsion Operation
General Electric Company

The irradiation program at NMPO encompasses a fairly large number of different materials, irradiated under a wide range of conditions. The extent is indicated in Table 1.

TABLE 1

Temperature	Reactor ambient, 120 to 4500°F (50 to 2500°C)
Gamma Heat Generation	0.1 to 15 watts per gram
Density	2 to 20 grams per cubic centimeter
Volumetric Heat Generation	0.2 to 300 watts per cubic centimeter
Thermal Neutron Flux	5×10^{12} to 3×10^{14} neutrons per square centimeter per second
Fuel Element Surface Heat Flux	6 to 180 watts per square centimeter 20,000 to 600,000 BTU per hour per square foot
Materials - Structural	- Rene', Hastelloy, Nichrome, A-286, Inconel
Moderator	- BeO, Various Hydrides
Fuel Elements	- Ceramic, Nichrome, Refractory Metals

This irradiation program has required some 20-30 different capsule designs, plus minor variations leading to a total of about 200 capsules having been built and irradiated, principally at the MTR, ETR, LITR, and ORR.

The most commonly applied method of cooling and temperature control is conduction and radiation outward across narrow gas annuli to a water cooled surface (Figure 1). Generally the test samples are cylindrical, or built up into cylinders. Depending on the test

requirements, various mixtures of helium, neon, argon, nitrogen and air are used to control the thermal conductivity of the gas and thus the system temperature levels. Flow rates range from 0.1 to 2 standard cubic feet per hour. Response time at the capsule to a composition change at the control panel is on the order of 15 minutes.

A variation of this method replaces water cooling by air cooling. Air cooling is applied in certain test facilities where water cooling is not practical. It is also used as a means of temperature control where altering the conducting gas conductivity is not feasible or desirable. In the test facilities in use, continuous control of water flow is not possible. With air cooling, however, the air flow rate is controlled, and the air cooled surface can be maintained at temperatures ranging from 200 to 1800°F (100-1000°C), and varied while on test to control the interior temperatures.

Direct air cooling of both fueled and non-fueled samples is employed in cases where a dynamic, oxidizing atmosphere is desired for the experiment, and where uniformity of temperature over the sample is not critical.

In all capsule designs, simplicity, reliability and safety are necessarily prime considerations. Large safety factors must be built in to cover such items as loss of coolant flow and uncertainties in reactor power generation. Because of the large number of irradiations fabrication costs must be held to a minimum, and manufacturing methods and tolerances lead to some compromises between analysis and actual design, such as the use of stepped rather than continuously variable gas gaps, and so far, rejection of mechanical means of temperature control.

A typical test requirement is the irradiation of small cylindrical samples one-quarter to one-half inch in diameter and one to three inches in length. As many samples as possible are irradiated together to minimize the number of capsules and the neutron cost. The number of samples is controlled by the irradiation facility diameter, and by the acceptable range in exposure, which limits the length along which the samples may be stacked. A typical capsule contains three rows of samples, and a sample length of twenty to twenty-five inches. With the sample length centered at the core peak flux, the exposure and heat generation drops to 75-80 percent of peak over this length.

In general, for a test of this type, it is desirable to maintain all samples in a given capsule (or stack) at a uniform temperature. The only design control compensates for the average

variation in power generation over the capsule length. This is accomplished by either varying the gas annuli dimensions, the amount of heat generating mass in the heat transfer path, the surface emissivities, or combinations thereof. A resulting test is considered satisfactory if the samples indicate a temperature within one or two percent of the absolute temperature level for 90 percent or so of the total exposure time. For a test designed to operate at 2000°F (1100°C), that would be ± 25-50°F. Reactor power variations and manual control limitations lead to off design operation for roughly 10 percent of the exposure time.

Figure 1 indicates typical dimensions involved. Figure 2 indicates typical test data. The two upper curves of Figure 2 are typical of "normal" operation, showing the effect of transverse power gradients and inaccuracies in design. The lower curves show the effect of a nearby control rod, and the need for further adjustment of gas composition during initial test operation.

A comparison of two refractory metal fuel element test capsule designs will indicate some of the restraints involved and the design approach. The two tests will be referred to by their NMPO designations. Table 2 gives the pertinent initial data.

TABLE 2

	TEST DESIGNATION	
	LTFF-5	MT-115
Sample Size, inches OD	0.503	0.240
ID	0.116	0.126
Length	1.503	0.750
Fuel Content, grams	5.0	0.3
Nominal Thermal Flux, nv	7×10^{12}	7×10^{13}
Atmosphere	Helium-Argon	Helium-Argon
Capsule Diameter, inches	2.375	2.375
Heat Sink	Air	Water
Test Temperature	3500°F(1925°C)	3500°F(1925°C)

Additional considerations were a method of sample support without setting up excessive temperature gradients through metallic heat conduction paths, and a design that would minimize the use of argon because of its activation.

Initial calculations indicated that both samples would generate about the same total amount of heat, around 0.56 BTU per second. The surface heat flux for the LTFF-5 sample was estimated at 0.25 BTU per second per square inch, and for the MT-115 at 1.0 BTU per second per square inch. Radiative transfer at 3500°F to a low temperature enclosure was estimated at 0.27 to 0.30 BTU sec^{-1} in^{-2}, exclusive of additional heat which would be conducted away by the inert gas atmosphere.

The resulting design for the LTFF-5 sample called for five radiation shields completely enclosing the sample to reduce the radiant transfer to less than one-half the generated value, the balance being conducted (Figure 3). Small staggered support tabs were used to maintain shield spacing and to maximize the direct metallic conduction path resistance. The outer air-cooled container was designed to operate between 800 and 1200°F (430-650°C) and provide a continuously variable temperature control. Secondary temperature controls included variation of the composition of the inert gas flowing over the sample and radiation shields, and movement of the entire capsule toward or away from the peak flux location within the test facility.

Because of the uncertainties in the design calculations, and as a means of confirming estimated sample power generation, the design was mocked up in the laboratory using an electrically heated test specimen. Analysis indicated that gamma heating of structural components had a negligible effect on sample temperature simplifying mock-up design. The test sample was simulated by a tungsten wound beryllium oxide cylinder. The mock-up confirmed the design up to 3000°F (1650°C) which was considered the simulated specimen limit.

The test operated successfully at the design temperature of 3500°F in pure helium, about three inches above the peak flux location. Later controlled addition of argon permitted operation at 4000°F (2200°C).

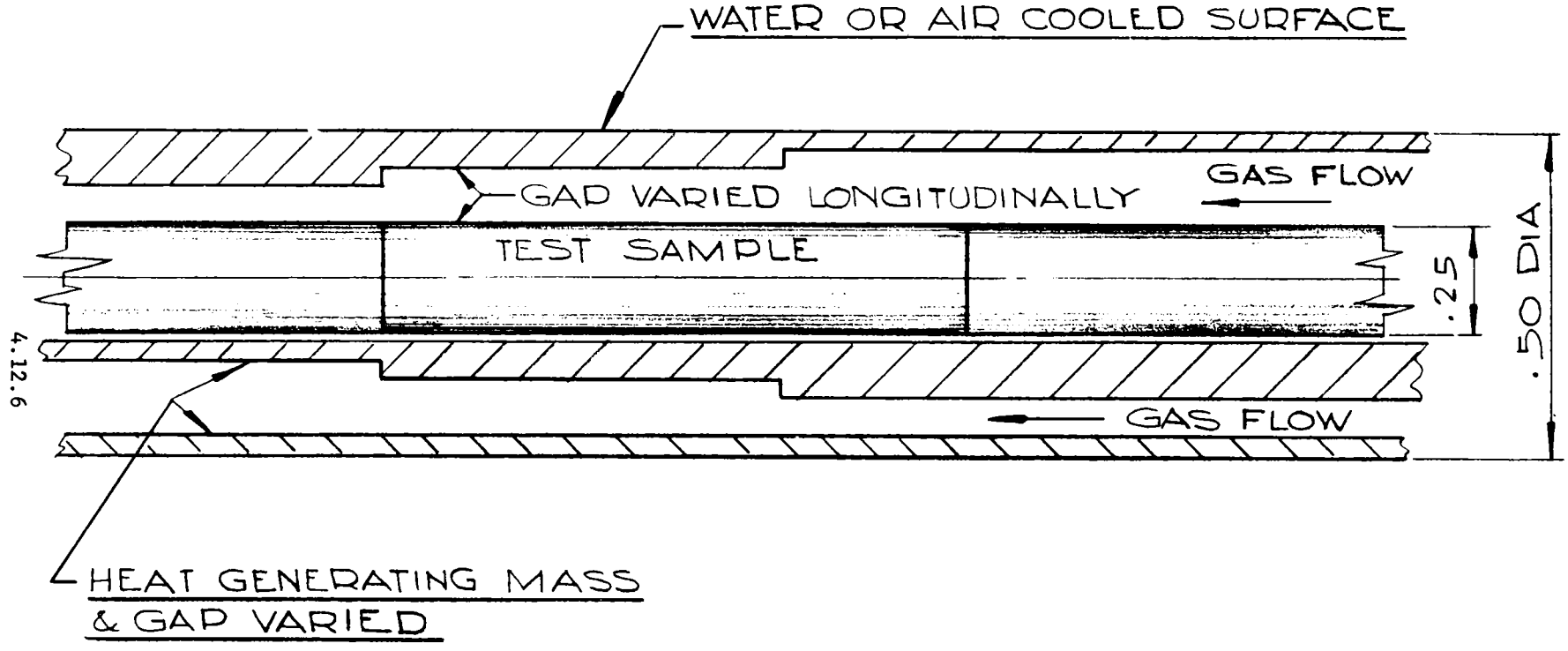
The design for the MT-115, on the other hand, required a single small gas gap to permit conduction of most of the generated heat and water cooling to provide the greatest possible temperature gradient (Figure 4). This restricted temperature control to variation in gas composition, since the ETR facility does not permit capsule movement. An annular gap of 0.043 inches was called for. Any direct support across this extreme temperature gradient would lead to excessive heat fluxes. Therefore, integral extensions were added to the basic test specimen, a little over an inch long. The diameter and length of the extensions were chosen to achieve

a compromise between adequate sample support, effect on sample temperature gradients, and amount of heat conducted directly to the water-cooled point of support. Calculations indicated that the support would reduce the sample temperature by about 350°F, introduce a longitudinal gradient over the sample length of about 300°F, and lead to a water cooled surface at the point of support about 15°F above reactor water temperature. The sample temperature reached 3400°F (1870°C) on pure helium and 3500°F (1925°C) with the addition of a trace of argon.

Two IBM 7090 computer codes are in general use for determining capsule heat transfer. One, the Concentric Cylinder Program, solves the case of heat transfer by simultaneous radiation and conduction radially outward across one to three concentric gas filled annuli. The other, the Transient Heat Transfer Program solves quite general cases of three dimensional heat transfer.

The Concentric Cylinder Program is fairly simple. For a given gamma heating rate, it determines the heat generated in the test sample and up to three concentric cylinders surrounding the sample, as a function of cylinder diameters. For each set of cylinder diameters, it then determines the resulting temperature distribution.

The Transient Heat Transfer Program is a finite difference analysis of general transient heat transfer in one to three dimensions. It permits of heat transfer by conduction, convection, and radiation in a network of up to 1000 nodes. It can account for internal heat generation, latent heat effects, arbitrary boundary heat fluxes, contact coefficients, various materials, conductivities, heat capacities, and emissivities as appropriate functions of time and temperature.



4.12.6

FIG 1

METHODS OF LONGITUDINAL TEMERATURE COTROL

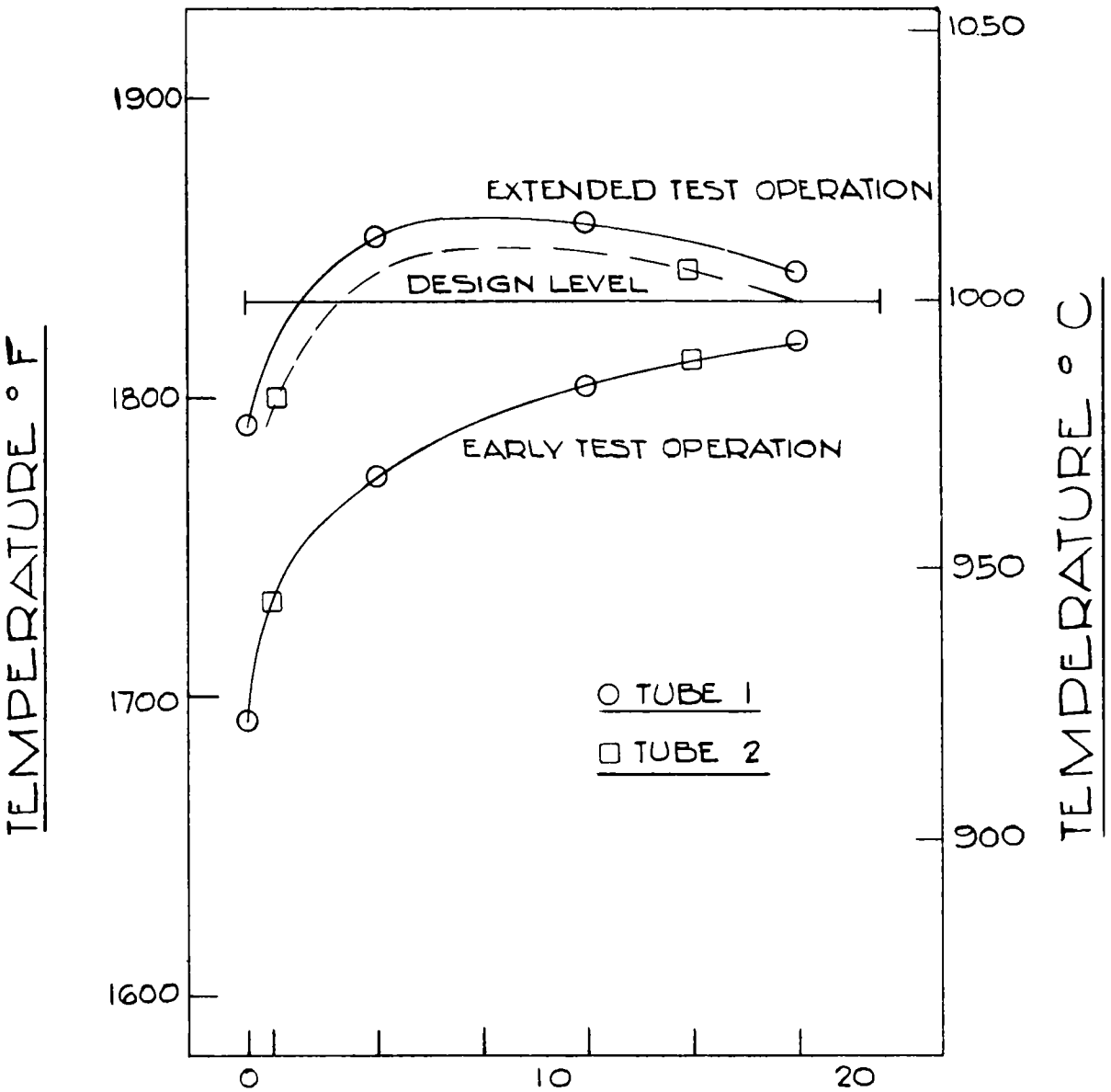
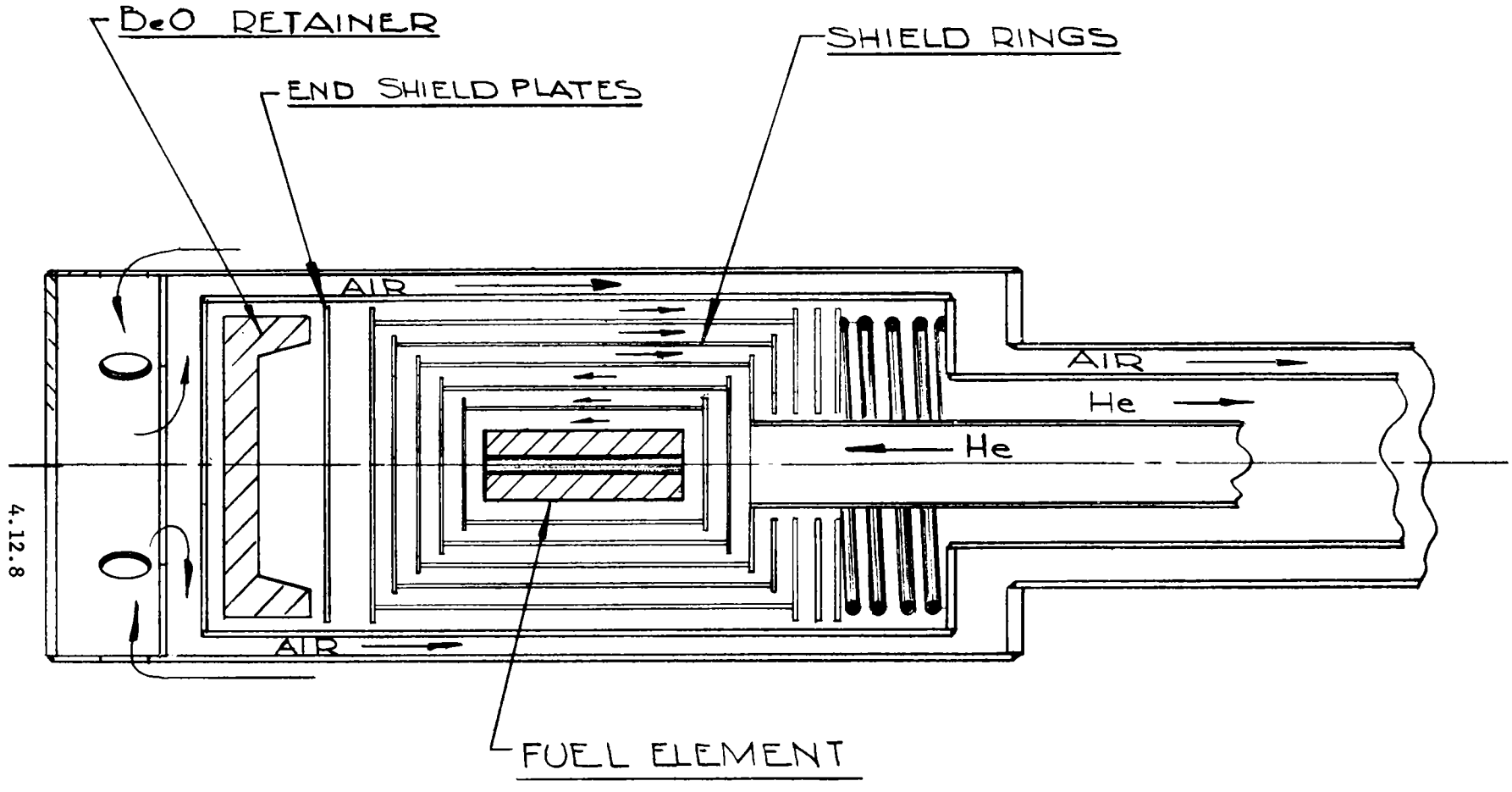


FIG 2
TYPICAL TEST DATA

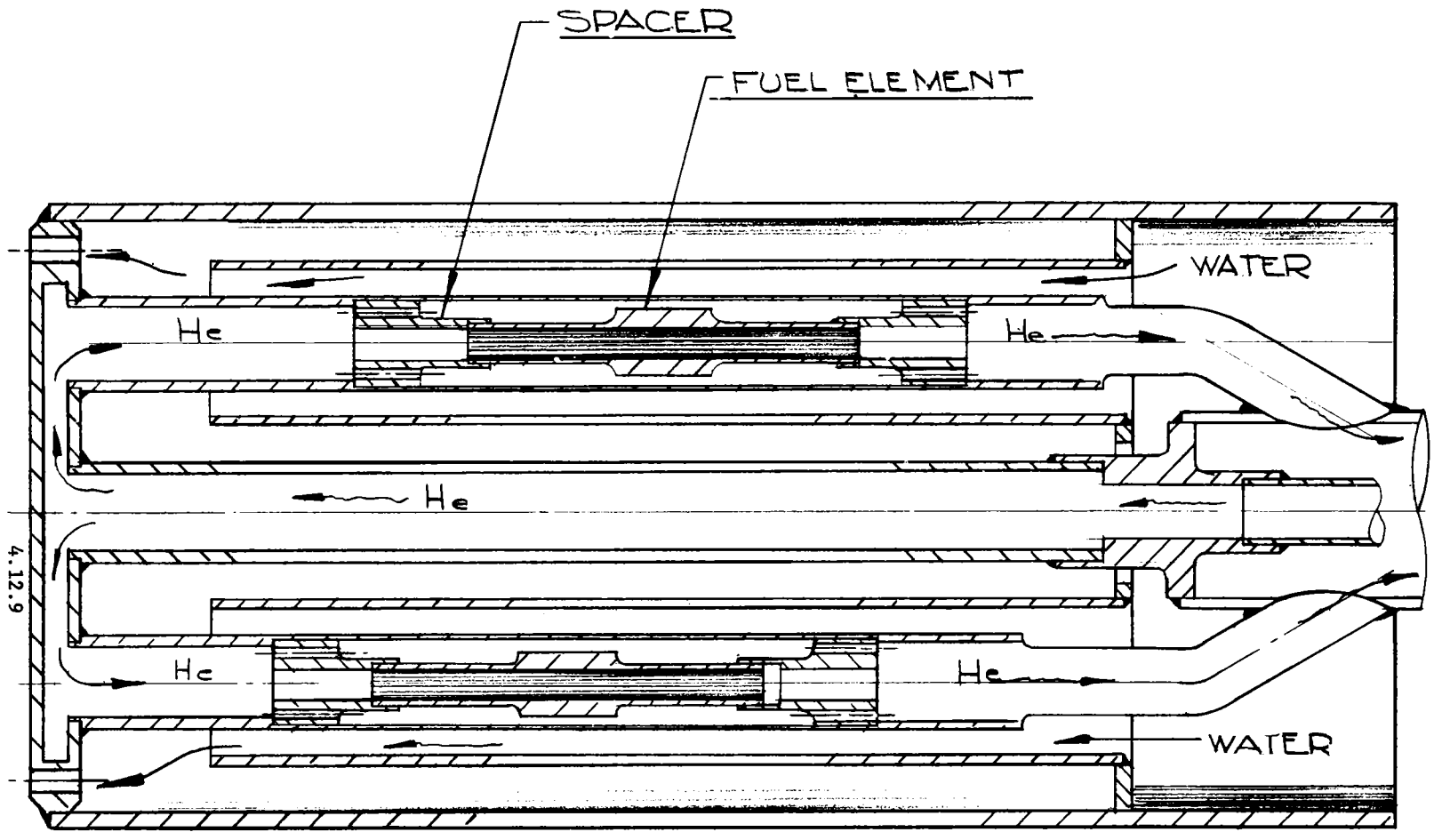


4.12.8

SCHEMATIC

LTFF 5

FIG-3



SCHEMATIC

33 MT 115

FIG 4

SYSTEMS AND TECHNIQUES FOR IN-REACTOR TEMPERATURE CONTROL OF NRL IRRADIATION EXPERIMENTS

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Metallurgy Division

U. S. NAVAL RESEARCH LABORATORY

INTRODUCTION

The temperature control systems and techniques utilized for the irradiation of steel specimens in various test reactor experiments are complimentary to the encapsulation techniques which have been described in a separate paper at this Conference (1). That is, temperature control systems have been designed to match encapsulation techniques for certain experimental program objectives and have not resulted in basically new encapsulation procedures. Rather, the objectives of capsule simplicity with massive loadings, in facilities of limited cross-sectional area and, in most cases, with limited space for insulation have dictated the temperature control techniques as well as heater designs.

The flexible stainless steel sheath around many experimental assemblies provides one means of temperature control. With the specimens to be irradiated serving as the heater, variable internal pressurization provides a versatile means for temperature control in the range of 400°F up to about 650°F. This technique, which is described in more detail below, may be used with or without heaters for even greater versatility.

Small in-reactor heaters have been developed and used successfully in multitemperature experiments. The design, insulation, and instrumentation of heaters are reviewed to demonstrate the integration within a single experimental capsule.

TEMPERATURE CONTROL WITHOUT HEATERS

Two systems have been used to control temperature without heaters. One system involved "overheating" plus cooling, that is, a heating capacity through reactor ambient, plus gamma heating, plus insulation which results in a "stagnant" (no external control) condition in which the specimen temperatures are slightly higher than the desired irradiation temperature. Control is effected through coolant flow. This system has been used for experiments in the horizontal beam port facilities of the Brookhaven Graphite Reactor

(BGR)and the Oak Ridge Low Intensity Test Reactor (LITR). The second system involves a slightly different situation in which the stagnant condition involves specimen temperatures slightly below the desired exposure temperature. In this case the capsule is pressurized to move the flexible containment sheath away from the specimens and thus raise the temperature. Through an automatic control procedure the temperature is raised by pressurization and lowered by depressurization. Certain variations of these two systems have been employed. Some specific experimental systems are described to illustrate the technique and their application.

Control by Coolant Flow

Experiments in the Brookhaven Graphite Reactor have involved temperature control through the control of air coolant flow. Coolant air in these experiments was discharged inside a square configuration of drop weight, Charpy, and tensile specimens (Figures 1 and 2). Since the volume of air used for cooling was not critical, a relatively simple on-off control system was used with an outlet constructed to provide air volume according to the gamma heating in a given portion of the assembly. Success of this control system depended upon the attainment of a stagnant temperature above, but no more than fifty degrees higher than the desired irradiation temperature which was 500-575°F for most experiments in the BGR.

A variation of this system was employed for temperature control of 500-600°F experiments in a horizontal beam port of the Low Intensity Test Reactor (LITR), (Figure 3). The capsule or experiment container included a cooling water jacket but temperature control was attained primarily through regulation of a slight helium flow into the specimen configuration. However, both water and helium flow could be varied or the helium replaced with air for specific experiments. The exact combination of coolants and flow levels varied with individual experiments but depended upon a variable cooling capacity and thus, an initial stagnant temperature slightly higher than the desired specimen irradiation temperature. The most useful combination for close and uniform temperature control was the use of a constant water flow and a helium

flow which was varied according to need. Figure 3 also shows an alternate assembly design for irradiations at less than 200°F in the beam port facility. With this design, the specimens are placed in direct contact with the cooling water jacket to maximize heat transfer. A partial helium atmosphere provided additional cooling capacity. The helium flow control system described above was also used for temperature control of these experiments and to conserve the helium supply for long-term exposures.

Control by Internal Pressurization to Change Heat-Transfer Characteristics

One of the most successful techniques employed for irradiation temperature control for experiments in or near the the core of the LITR involves internal pressurization of the assembly with air or helium. The proper functioning of this system depends upon, (1) the attainment of a stagnant temperature developed by gamma heating below that desired for irradiation, (2) a flexible stainless steel skin or sheath which can be moved away from the specimens by internal pressure, and (3) instrumentation to automatically provide air when the temperature is too low and to remove pressure when the temperature is too high.

A general diagram of the pressurization system showing the physical orientation of the experiment in the reactor and the instrumentation utilized is given in Figure 4. Figure 5 shows schematically the action of the individual components of the pressure system during the three stages of temperature control. Figure 5a depicts the situation where specimen temperatures are too low and a "high" air pressure ($\approx$ 8-9 psig) is applied for heating. When specimen temperatures reach the desired control point, the pressure in the assembly is automatically reduced ($\approx$ 4-5 psig) for cooling as shown in Figure 5b. During this transition, the high assembly pressure is discharged via valve "A" and the needle valve to the station off-gas system. Although a constant bleed of low air pressure through the needle valve continues during this stage of operation, the volume of air is negligible and the need for a complex time delay circuit for valves "A" and "C" is avoided.

Figure 5c indicates the system response to an overheating situation. If, with the reduction of assembly pressure on reaching the control point, specimen temperatures continue to rise, the assembly is connected directly to the station off-gas system for depressurization and maximum cooling. As noted in the Figure, an alternate or back-up system for depressurization was also provided as protection against component failure.

The control cycle developed by the variable pressure system involves a gradual process and therefore no abrupt changes of temperatures occur. The system is flexible in that pressurization by either air or helium, or, the creation of a partial vacuum in the assembly, can be accomplished. No provisions were made for externally controlling the heat supply to the assembly since the gamma heating in the LITR core was considered more than sufficient to meet the desired temperature control in the range of 400-650°F. The elimination of an externally controlled heat source in this system also insured that assembly temperatures would drop to ambient reactor coolant temperatures during shutdown periods.

A leakage problem has limited the effectiveness of the internal pressurization system for long-term experiments however. This problem apparently arises from fatigue cracks developing at the welds of the assembly after cycling three to four months. However, no leakage problem has arisen with shorter irradiation periods. It is believed that this problem may be eliminated by extending the assembly length with insulating (unloaded) end blocks so that the strain amplitude during cycling is reduced.

A modified version of the assembly and temperature control system described above has recently been proof tested and proven very successful. This later design incorporates double encapsulation with flexible containment sheaths. With pressurization of the annulus between the inner and outer sheaths, a much wider range of temperature control is provided over the nominal pressurization range used with the earlier assembly designs. Since small increments of pressure change result in large temperature changes, fatigue action on the containment sheath is minimized. In addition, in view of the wide range of control, a series of irradiations at different temperatures can be performed using one standardized assembly design.

TEMPERATURE CONTROL WITH HEATERS

Although the internal pressurization system has been used successfully for about 15 irradiation experiments, it is limited in that temperatures cannot be selectively varied during irradiation. Consequently, a heater assembly for use in LITR core experiments has been designed.

Heater Design

Figure 6 shows an exploded view of the heater as well as its configuration after final assembly. Nichrome-V wire (.032-in. diameter) serves as the heater element and is formed in a non-inductive winding. The heater core blocks are of fired lavite (Grade A hydrous aluminum silicate). Care was taken in the design of the core blocks and winding configuration to eliminate any sharp bends in the winding which might produce local hot spots subject to burnout.

As with most heaters of this type, the greatest potential trouble spot is the connection between the heater winding and the power lead. For this junction, a thin stainless steel tube, 3/8-in. long, crimped around overlapped winding and power leads and back-filled with silver solder, has been used with good success. As a precaution against high joint temperatures, the winding is doubled back into the heater block for a distance of about one inch (equivalent to a quarter of a turn).

In service, the heater assembly is housed in a hollowed retainer of transite (asbestos fiberboard) to insulate the experiment outer sheath from the heater proper. Leakage heat flow is further minimized by insulation of the specimen clamping strap immediately adjacent to the heater. In this manner, heat flow is directed primarily through the length of the specimen for maximum heater efficiency.

A typical experimental irradiation assembly, shown in Figure 7, incorporates four individual temperature zones, each served by a separate heater and heater control system. Each zone contains 20 to 25 Charpy-V specimens whose ends are in direct contact with the small compact heater. Since heat flow between zones

is highly restricted by the insulation provided, control temperatures of two adjacent zones can be varied by as much as 150-200°F without interference. This design feature has been particularly valuable for experiments in which selected zones receive periodic annealing treatments during irradiation.

Automatic Heater Control System

The system developed for the control of in-reactor heaters basically incorporates a thermocouple actuated controller-recorder of the potentiometer type and an adjustable auto-transformer. The system also includes a rectifier bank to provide pulsating d.c. voltage rather than a.c. voltage to the heater, thus eliminating the potential problem of a.c. interference developing in thermocouple control and recording circuitry. Although this system functions in a manner similar to that of a saturable core reactor control, it is far less costly and requires less console space than the latter.

The circuitry designed for automatic furnace control is shown schematically in Figure 8. A rear view of the instrument panel is shown in Fig. 9. In the development of this system, the major consideration was the prevention of high voltage-current surges, or momentary electrical overloads in the heater circuit at any time of heater operation. This design criterion was met by motorizing the variac used to adjust heater power, thus permitting the gradual application (and removal) of heater voltage on demand. With the low RPM motor used, approximately 15 seconds are required to change from zero output to full power operation.

The instrumentation used to control the level of furnace operation, that is, the amount of upscale motor drive, consists of a voltage sensitive d.c. relay with a series connected rheostat placed in parallel with the heater leads (Figure 8). When the voltage across the relay coil reaches the minimum required for actuation, the motor drive circuit is opened. By adjustment of the rheostat, this action is made to coincide with the desired current input to the heater. As protection against possible failure of either the relay or the rheostat, an alternate or back-up system is provided as shown in Figure 9. A cam mounted on the rotating

variac brush plate actuates a forward or "upper limit" microswitch in the motor circuit. Although stopped, the motor does not reverse when this high limit is contacted. The heater, relatively unaffected by the slightly higher current input, continues to operate until turned off by the temperature controller.

A second safety circuit in the system involves an electrical interlock which prevents the immediate re-energization of the furnace before the variac is returned to the zero power setting. This cycling control is a particularly important feature when the temperature control instrumentation used is very sensitive, or when specimen temperatures are subject to rapid change.

Operational experience with this system has shown that individual specimen temperatures can be readily maintained within ±5°F (3°C) and that maximum temperature deviations developed between specimens contained in a given control zone are less than ±15°F(8°C). The equipment has also proven highly reliable and requires only routine maintenance and inspection at bi-monthly intervals.

REFERENCES

1. L. E. Steele, "Encapsulation Techniques for NRL Irradiation Effects Studies", Conference on "Problems in Irradiation-Capsule Experiments", 8-10 October 1963, USAEC.

Figure 1 - Experiment assembly design and temperature control system for Charpy specimen irradiations in the Brookhaven Graphite Reactor (BGR)

Figure 2 - Alternate assembly design for drop-weight test specimen irradiation in the Brookhaven Graphite Reactor

Figure 3 - Experiment assembly designs for Charpy and drop-weight test specimen irradiations in a beam port facility of the Low Intensity Test Reactor (LITR)

Figure 4 - Schematic view of system for elevated temperature control through variable internal pressurization

Figure 5 - Design of the variable pressure system and the operation of its components for temperature control of irradiation assemblies.

Figure 5a- Heating situation - specimen temperatures below control temperature

Figure 5b- Cooling situation - specimen temperatures above control temperature in control zone

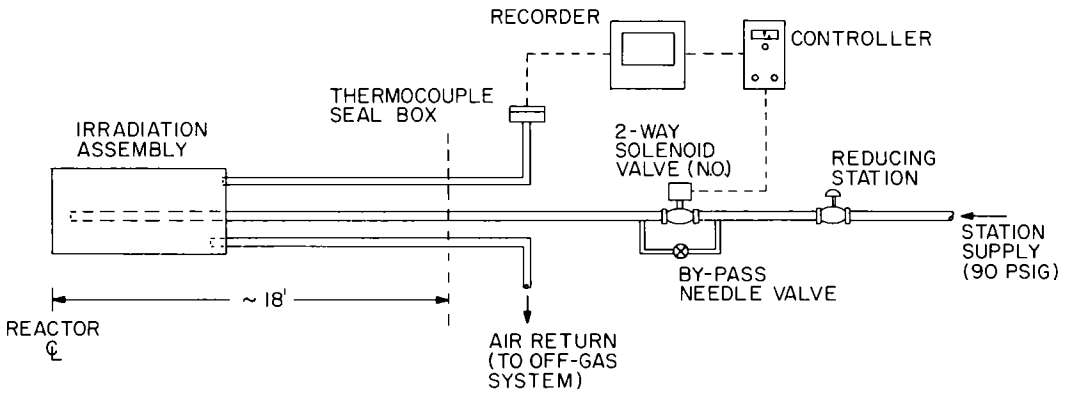
Figure 5c - Overheating situation - specimen temperatures 10°F above control temperature

Figure 6 - In-reactor resistance heater showing stages of fabrication

Figure 7 - Multi-heater assembly showing detail of loading and the assembly before and after encapsulation - for irradiation of Charpy specimens at various elevated temperatures

Figure 8 - Schematic diagram of automatic furnace control

Figure 9 - Furnace control panel (rear view) showing: (1) Variac, (2) 1 RPM reversible motor, (3) Drive motor relay (R1 and R2), (4) Temperature control relay (R3), (5) Diode rectifiers, (6) Adjustable rheostat(current selector), (7) 6-volt d.c. relay (R4), (8) Safety limit switch, upscale drive, and (9) Limit switch, downscale drive



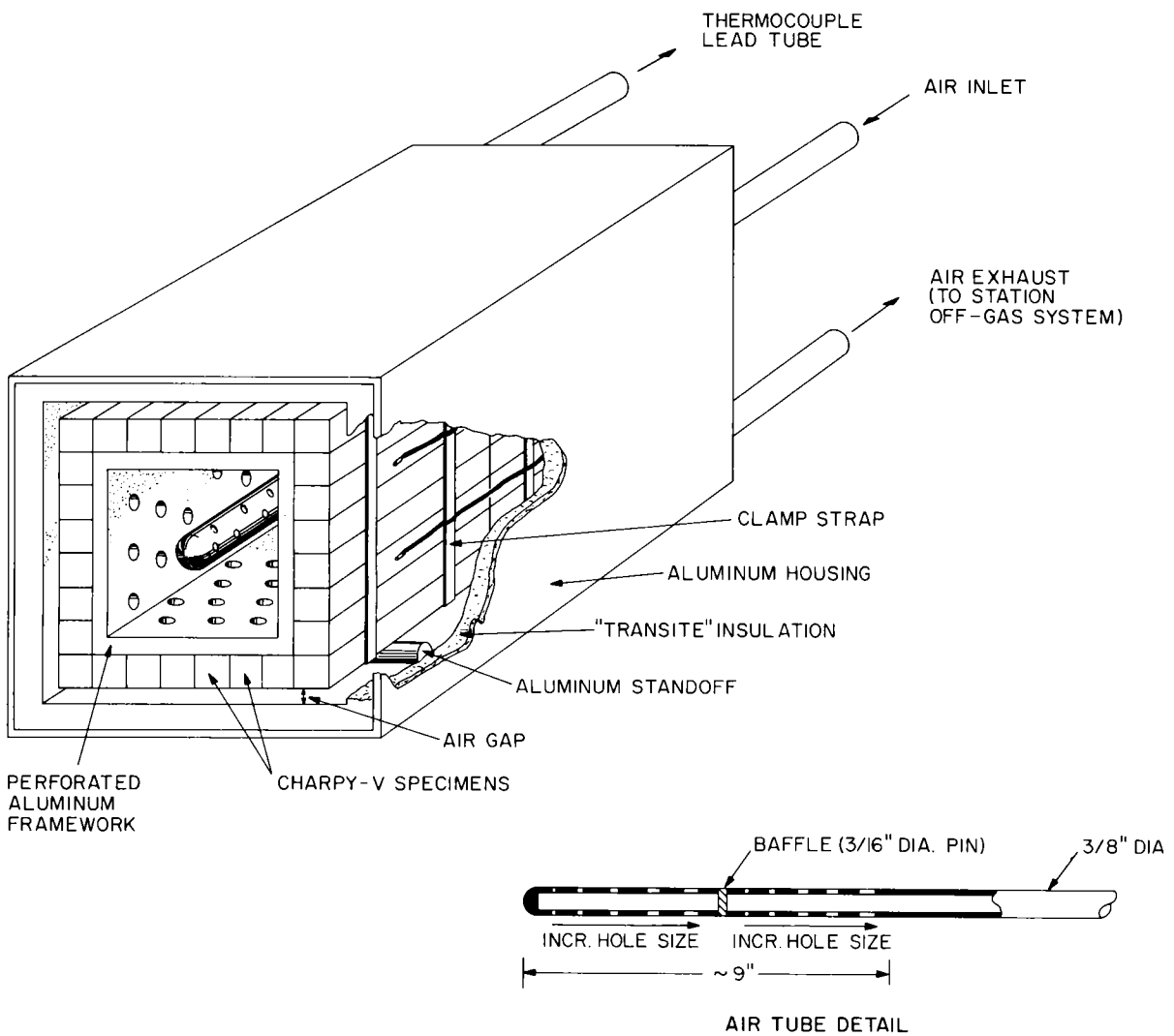


Fig. 1 Experiment assembly design and temperature control system for Charpy specimen irradiations in the Brookhaven Graphite Reactor (BGR)

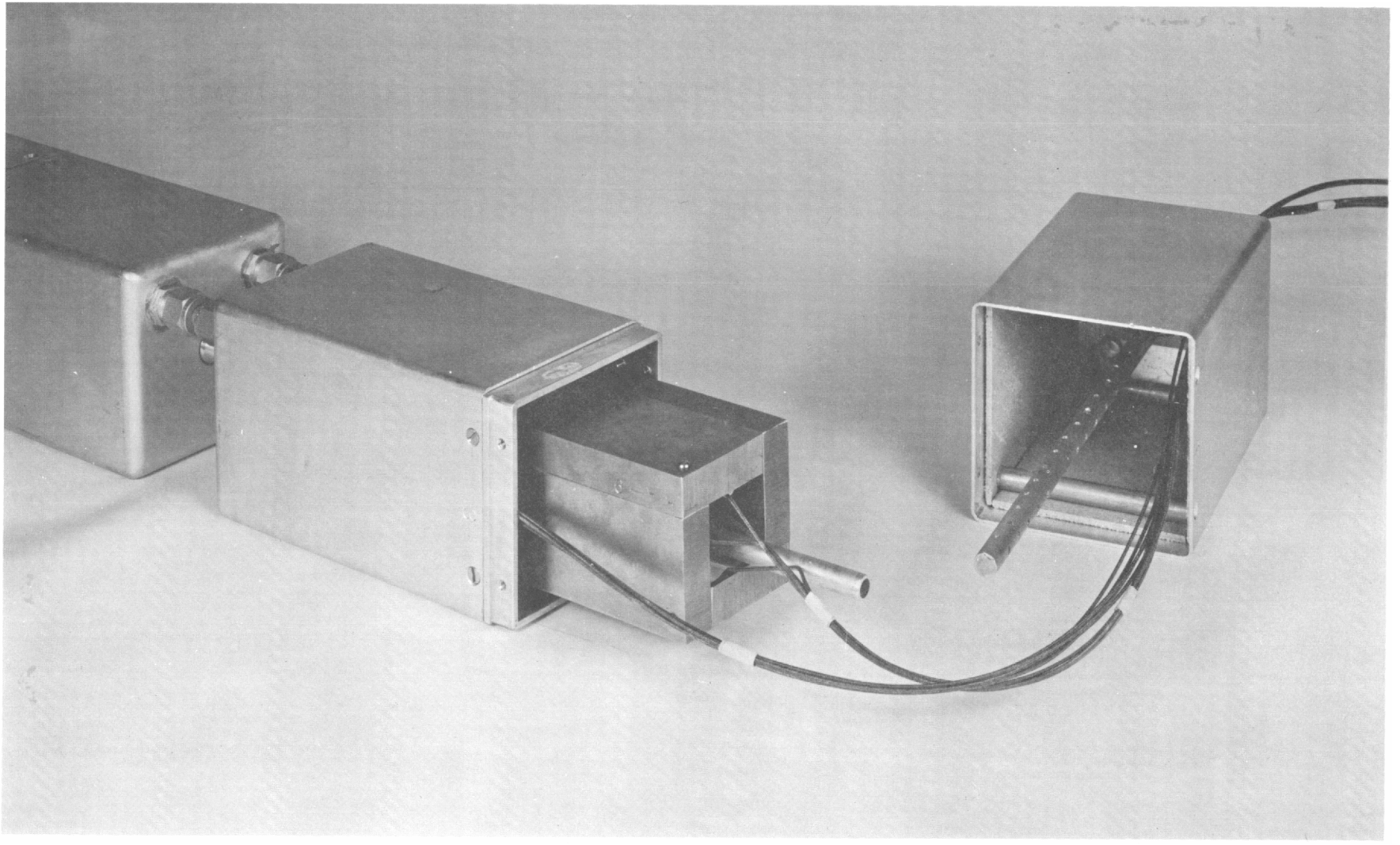


Fig. 2 - Alternate assembly design for drop-weight test specimen irradiation in the Brookhaven Graphite Reactor (BGR)

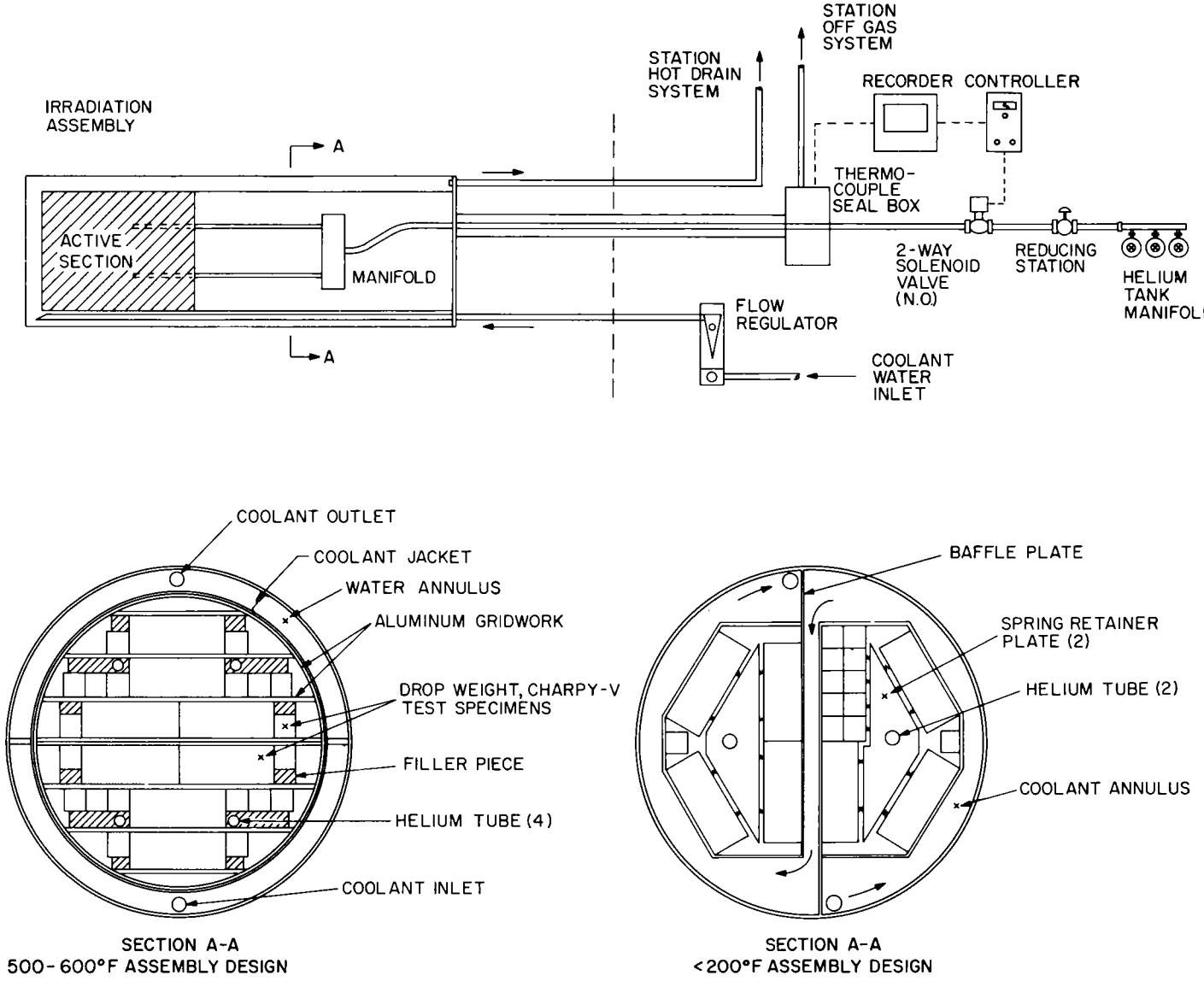


Fig. 3 - Experiment assembly designs for Charpy and drop-weight test specimen irradiations in a beam port facility of the Low Intensity Test Reactor (LITR)

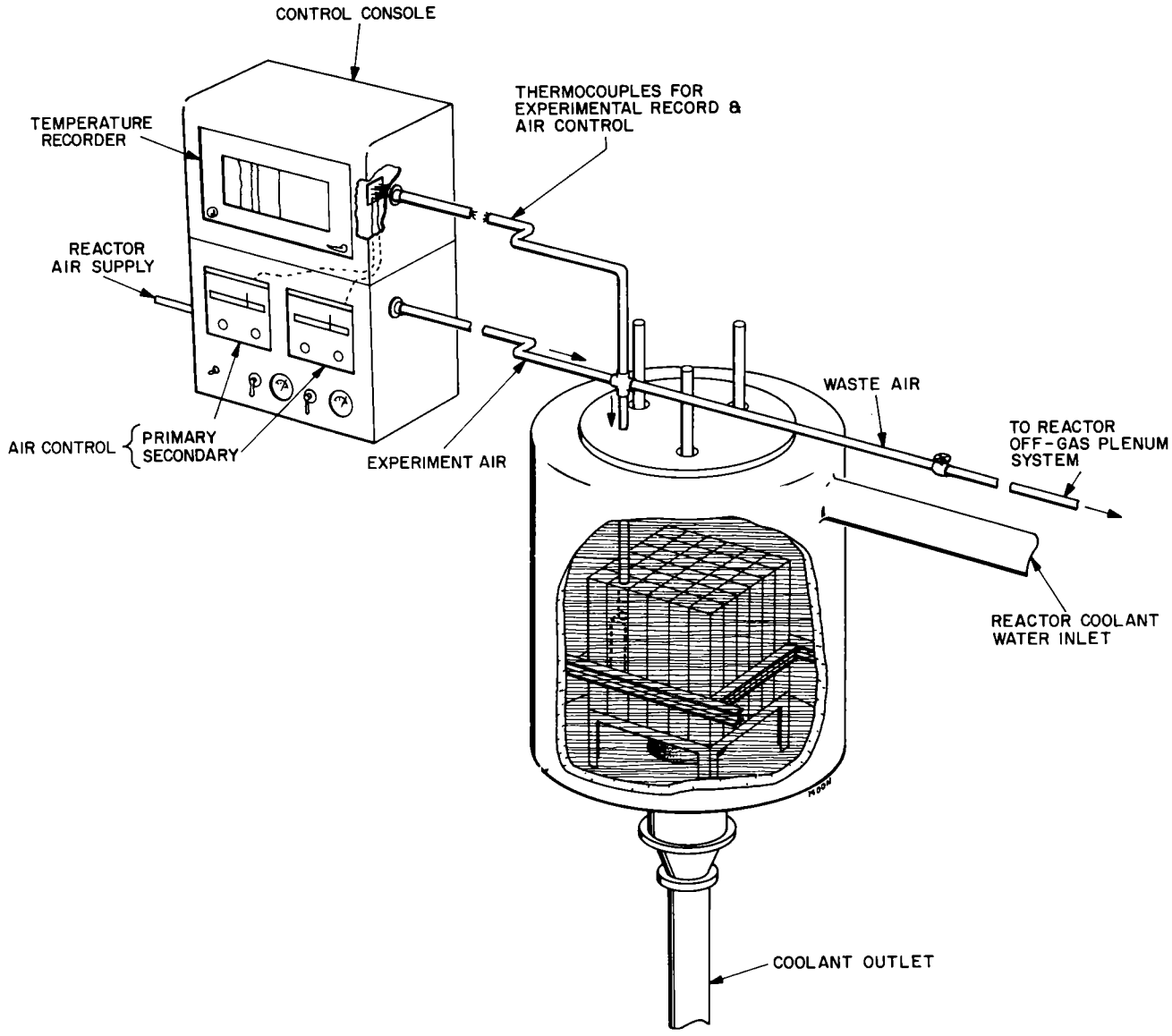
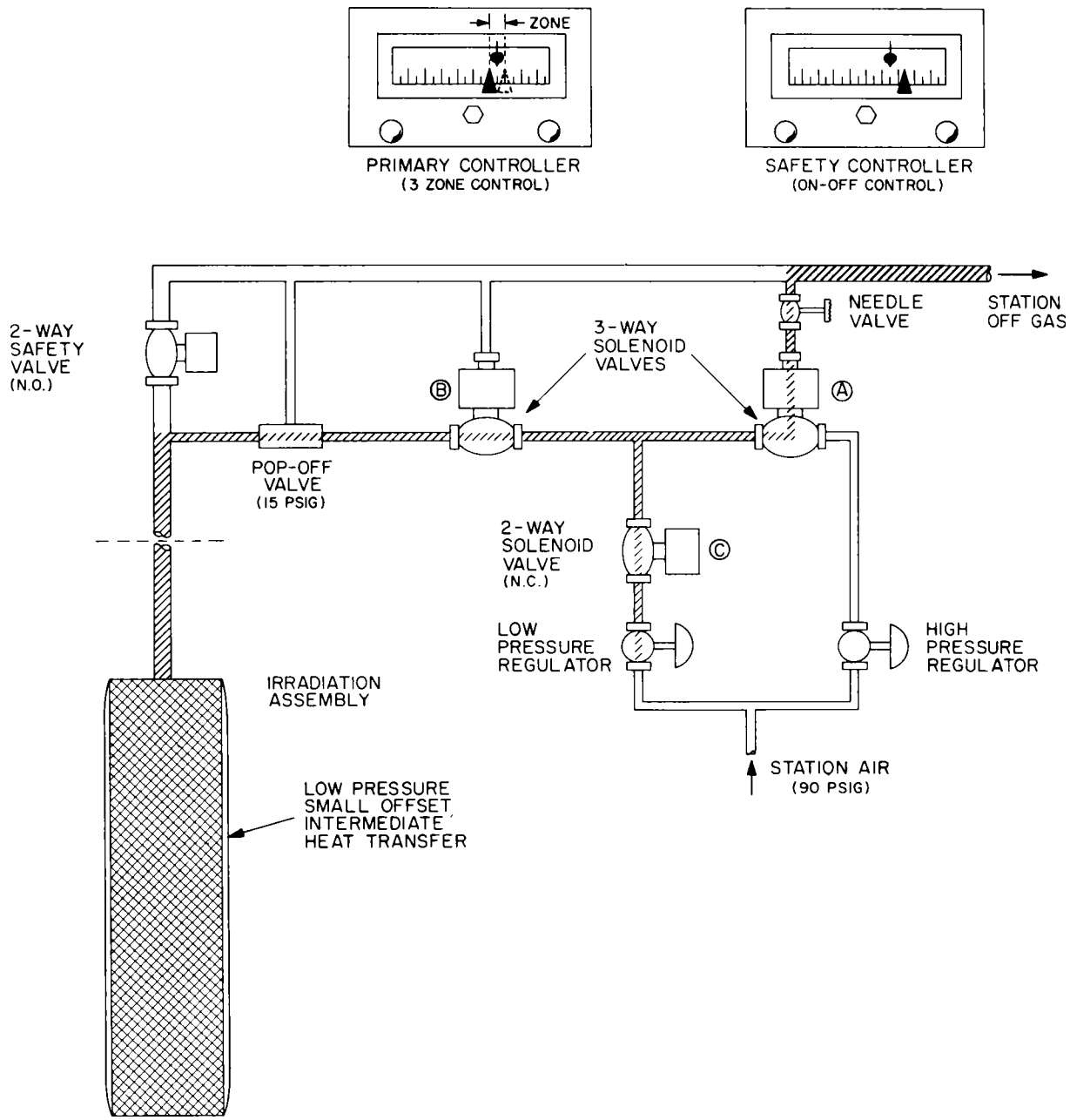


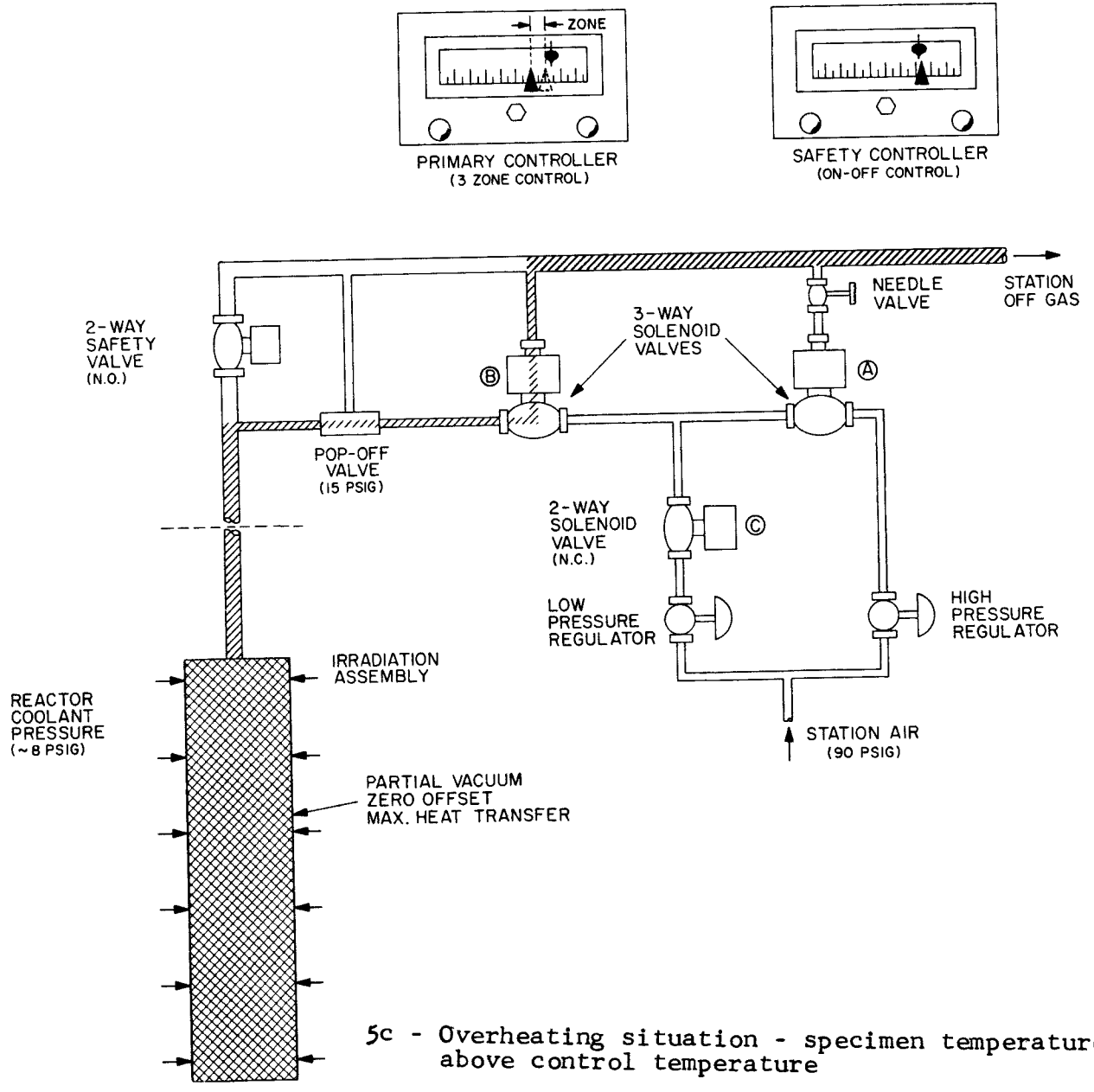
Fig. 4 - Schematic view of system for elevated temperature control through variable internal pressurization

5a - Heating situation - specimen temperatures below control temperature

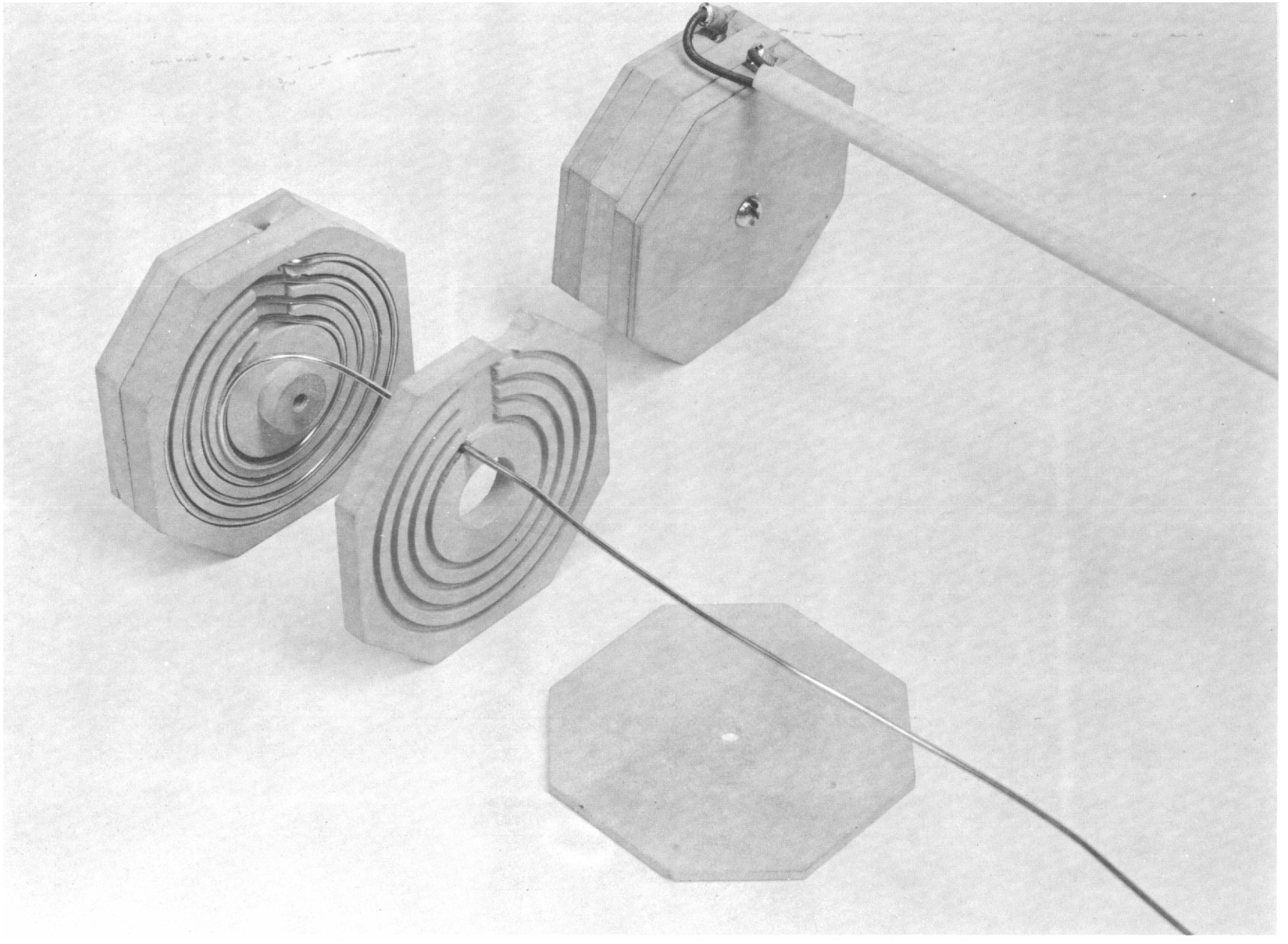
Fig. 5 - Design of the variable pressure system and the operation of its components for temperature control of irradiation assemblies:



5b - Cooling situation - specimen temperatures above control temperature in control zone



5c - Overheating situation - specimen temperatures 10°F above control temperature



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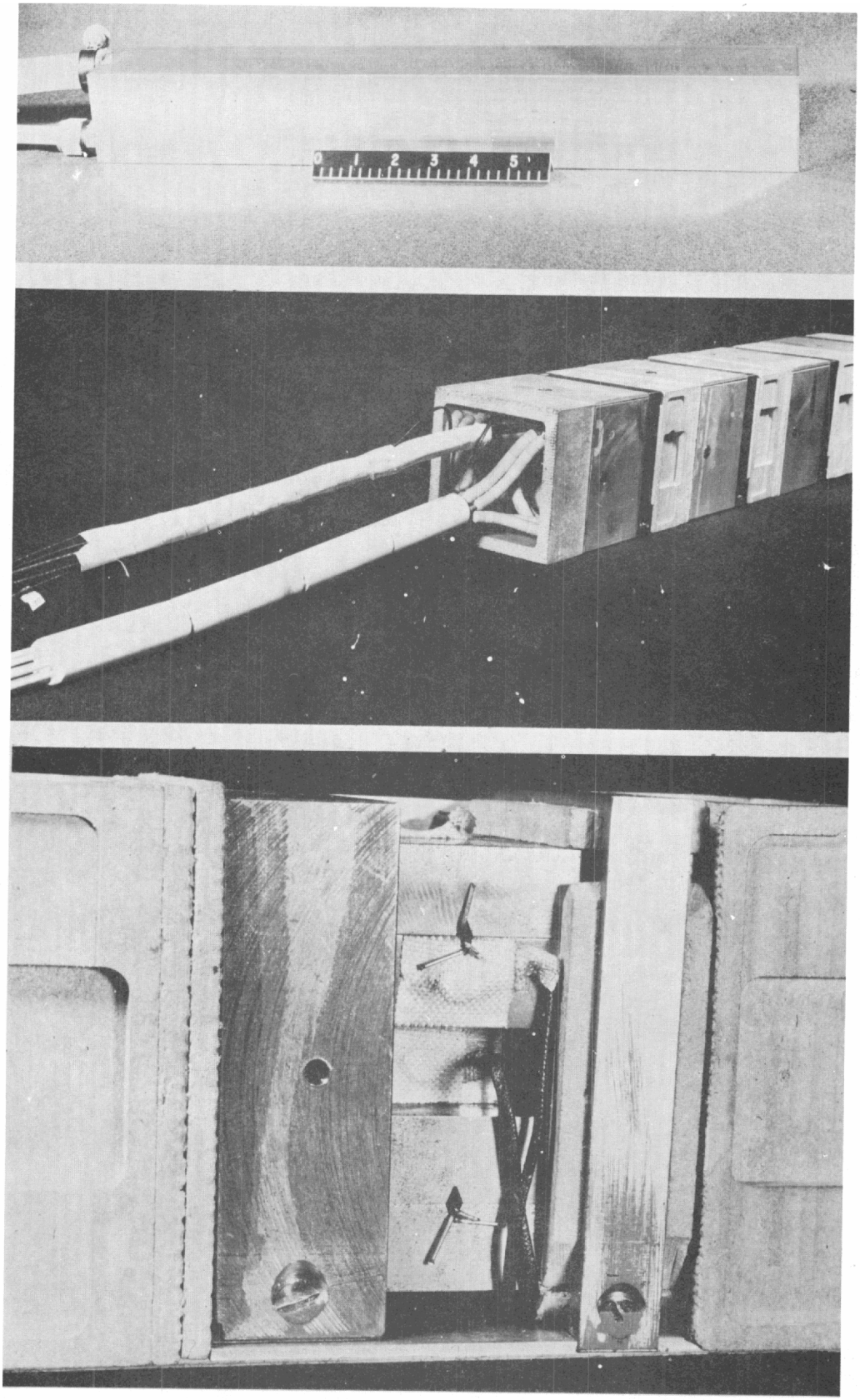


Fig. 7 Multi-heater assembly showing detail of loading and the assembly before and after encapsulation for irradiation of Charpy specimens at various elevated temperatures

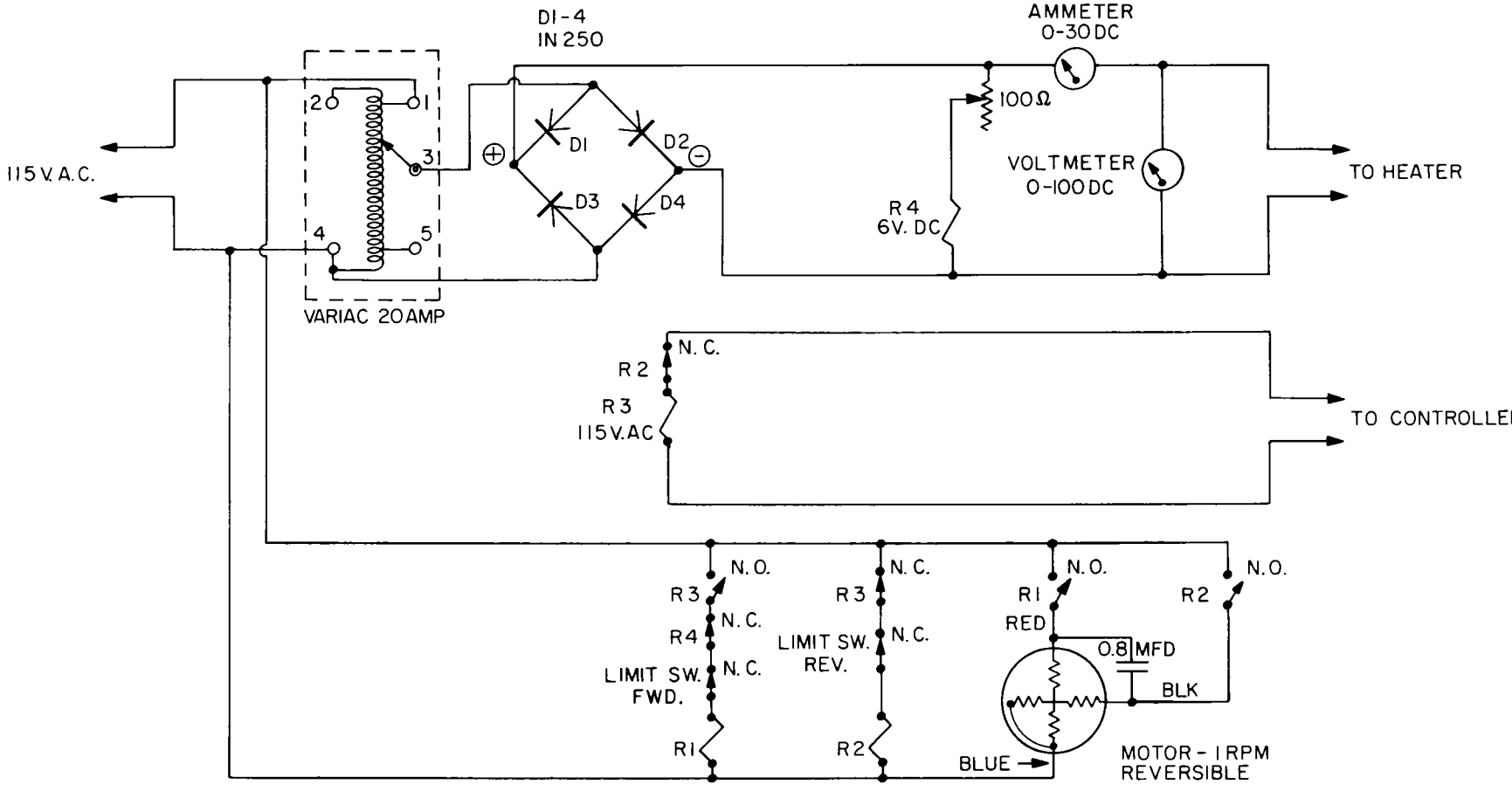
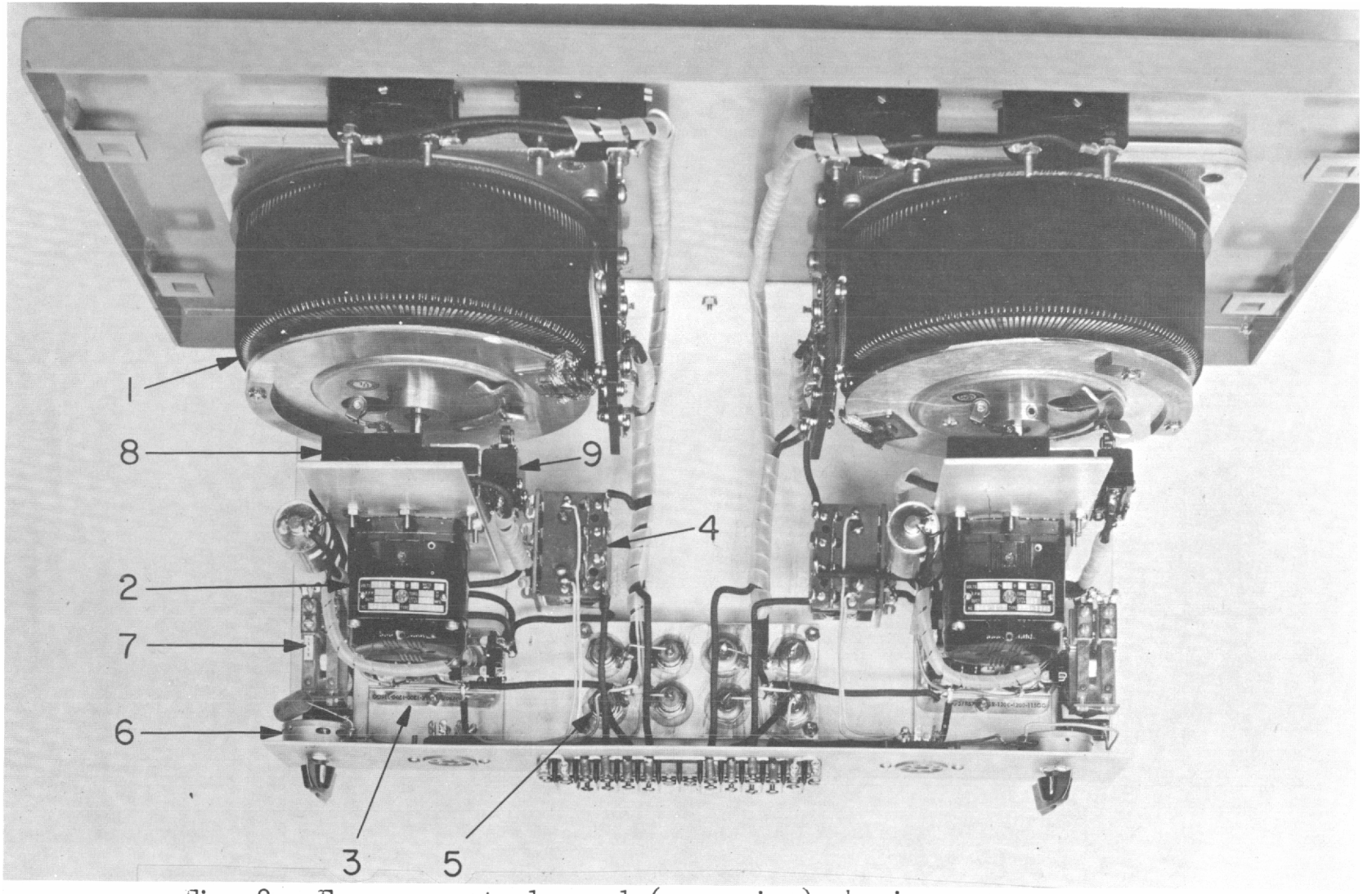


Fig. 8 - Schematic diagram of automatic furnace control



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MULTI-CELL IRRADIATION CAPSULES WITH INDEPENDENT INSTRUMENTATION AND TEMPERATURE-CONTROL FEATURES

K. Worth and D. G. Guggisberg

General Atomic
Division of General Dynamics

The design of the fuel body for the reactor being developed jointly by General Atomic and Empire State Atomic Development Associates, Inc., involved the use of (Th, U)C_2-C particles dispersed in a graphite matrix. Dimensional stability, physical integrity, and fission-product-release data for the hot-pressed and extruded fuel bodies under the conditions of high-temperature reactor irradiation were of primary interest to the design engineers.

To obtain the required information, an experiment was designed to provide a separate automatic temperature control and measurement in each one of three individual cells in an irradiation capsule by means of a variable binary gas mixture. A fission-gas sampling system with two sampling stations was connected to each cell to obtain fission-product-release data from the fuel specimen during the entire course of the irradiation.

The experiment was inserted in the GETR in July, 1963, for six cycles of irradiation.

Fabrication of a four-cell fuel capsule for the study of fission-gas release for an AEC-sponsored program on advanced BeO concepts is currently in progress. The purpose of this experiment is to measure the rate of fission-product release during irradiation of Be-UO_2-ThO_2 fuel specimens with different fuel volume ratios, to determine the effect of fuel agglomerate size on the fission-product-release rate during irradiation, and to study the effect of irradiation on dimensional stability and physical properties of the fuel materials.

THREE-CELL CAPSULE DESIGN FEATURES

The features of the irradiation capsule were as follows:

1. Three separately instrumented and controlled cells stacked in one capsule.

2. An automatic variable-conductivity gas temperature-control system for each cell.
3. Three tantalum-sheathed, BeO-insulated, tungsten versus tungsten–rhenium thermocouples recording the specimen temperature at three points in each cell.
4. Four Inconel-sheathed chromel versus alumel thermocouples placed in the grooves of the graphite thermal bond to control the specimen temperature.
5. Double containment for each cell.
6. A calculated peak heat generation rate of 500 watts/cm in the specimen region.
7. A calculated peak specimen temperature of 1500°C.
8. A calculated peak control temperature of 800°C.
9. A fission-product gas-sampling system consisting of
   a. Gas purification system,
   b. Fission-product traps.
10. A lead tube comprising
    a. Two auxiliary side lead tubes,
    b. 15 gas tubes,
    c. 44 thermocouple leads,
    d. 12 solenoid gas sampling valves,
    e. 24 electric leads.
11. A gas tube seal-off block located in the main lead tube to prevent contamination of the specimens during the discharge of the capsule from the reactor.

A longitudinal view and a transverse cross-sectional view are illustrated in Figs. 1 and 2, respectively. The complete capsule assembly with the beryllium filler piece is shown in Fig. 3.

PERFORMANCE CHARACTERISTICS OF THREE-CELL CAPSULE

The capsule has been in the reactor for three cycles and is operating satisfactorily. The bulk of the tungsten versus tungsten–26% rhenium

specimen thermocouples lasted for one and a half cycles and only one lasted for three cycles. Maximum specimen temperature reached was 2800°F. All chromel-alumel control thermocouples are still functioning and the capsule is operating safely based on the initially established temperature history.

FOUR-CELL CAPSULE DESIGN FEATURES

The four-cell capsule has the same basic features as the three-cell capsule, except that it has one more cell and a concentric arrangement of the lead tube extension sleeves through which the thermocouples and gas tubes are led from the bottom three cells to the main lead tube. The specimens are enclosed in a separate can which constitutes a third containment for each cell. A longitudinal cross section of one cell of this capsule is shown in Fig. 4.

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Godsin, W. W., "Temperature Control of Fueled Irradiation Capsules by Variable Conductance," Nuclear Sci. and Eng. 8, 340-45 (1960).

Johnson, D. E., "Report to Reactor Safeguards Committee on the MGCR-4 Fuel Irradiation Capsule," General Atomic Report GAMD-1700, September 21, 1960. (51 p.)

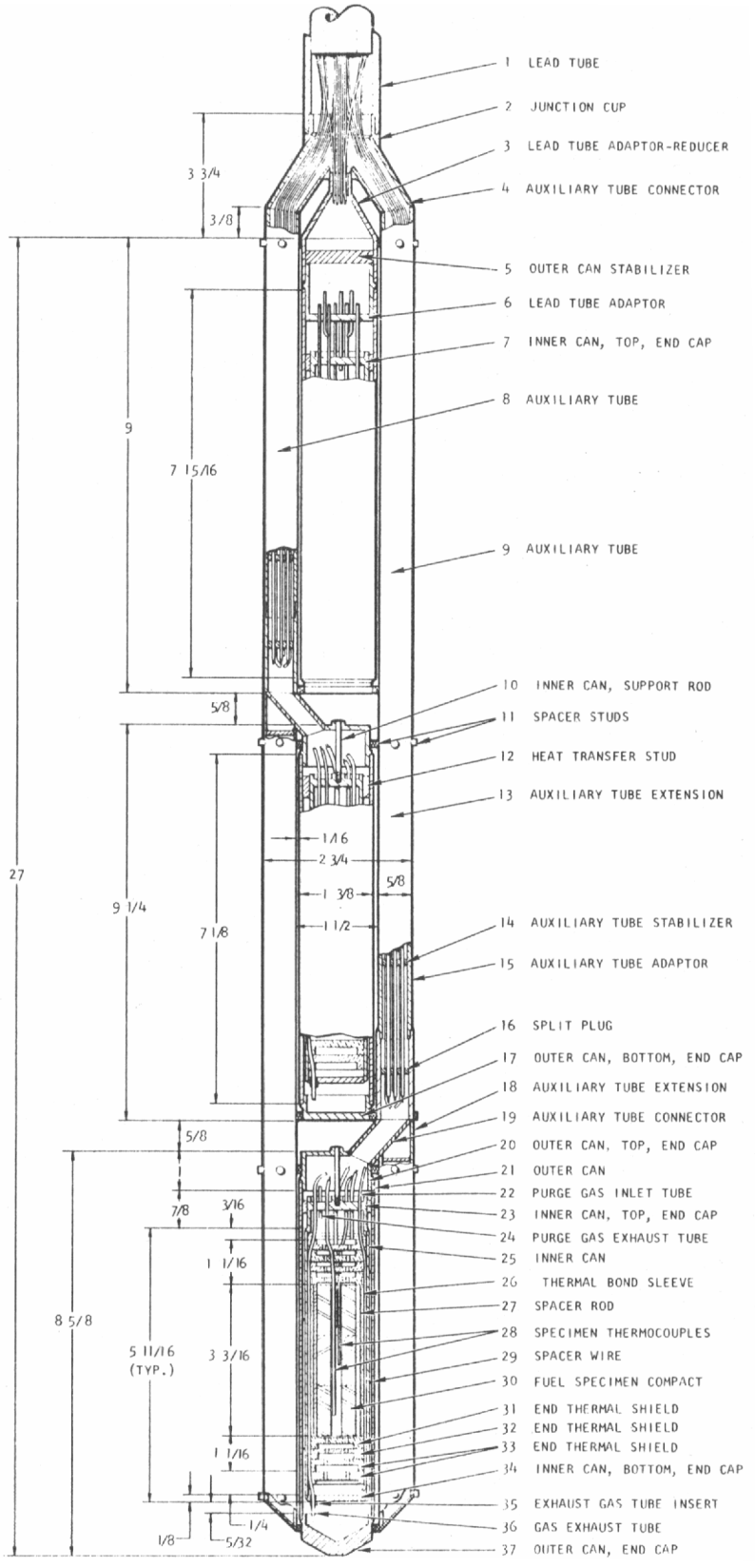
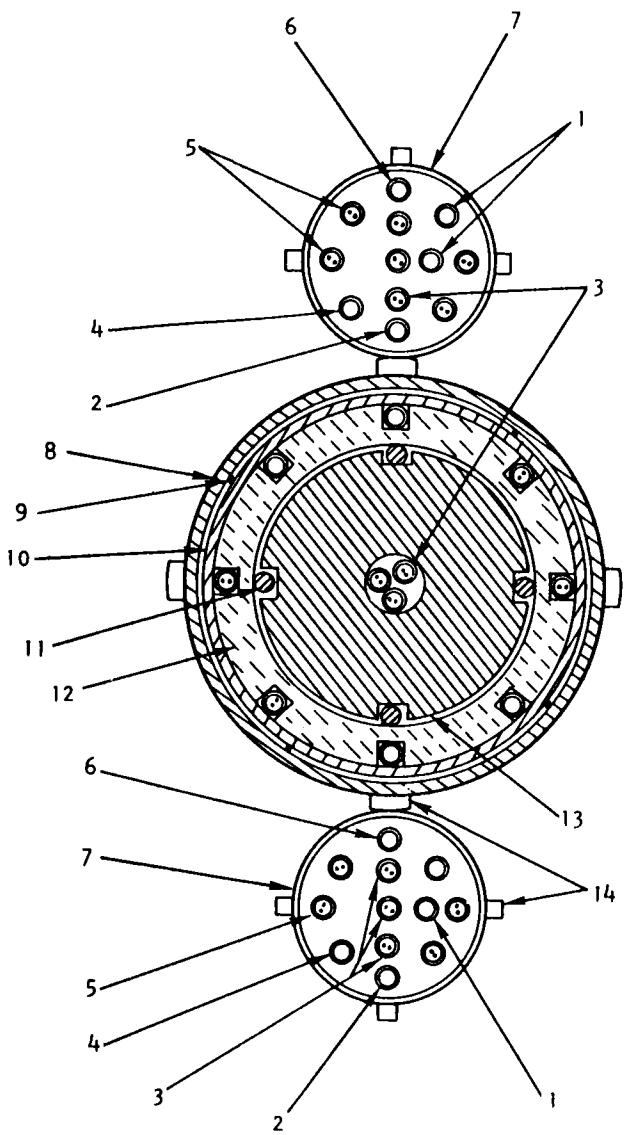


Fig. 1. Longitudinal cross section of three-cell capsule



1. TEMPERATURE CONTROL GAS TUBE
2. GAS EXHAUST TUBE
3. SPECIMEN THERMOCOUPLES
4. PURGE GAS EXHAUST TUBE
5. THERMAL BOND THERMOCOUPLE
6. PURGE GAS INLET TUBE
7. AUXILIARY LEAD TUBE (STNL. STL.)
8. OUTER CAN (STNL. STL.)
9. SPACER WIRE (INCONEL)
10. INNER CAN (INCONEL)
11. SPACER ROD (GRAPHITE)
12. THERMAL BOND SLEEVE (GRAPHITE)
13. FUEL COMPACT
14. SPACER LUGS

Fig. 2. Transverse cross section of three-cell capsule

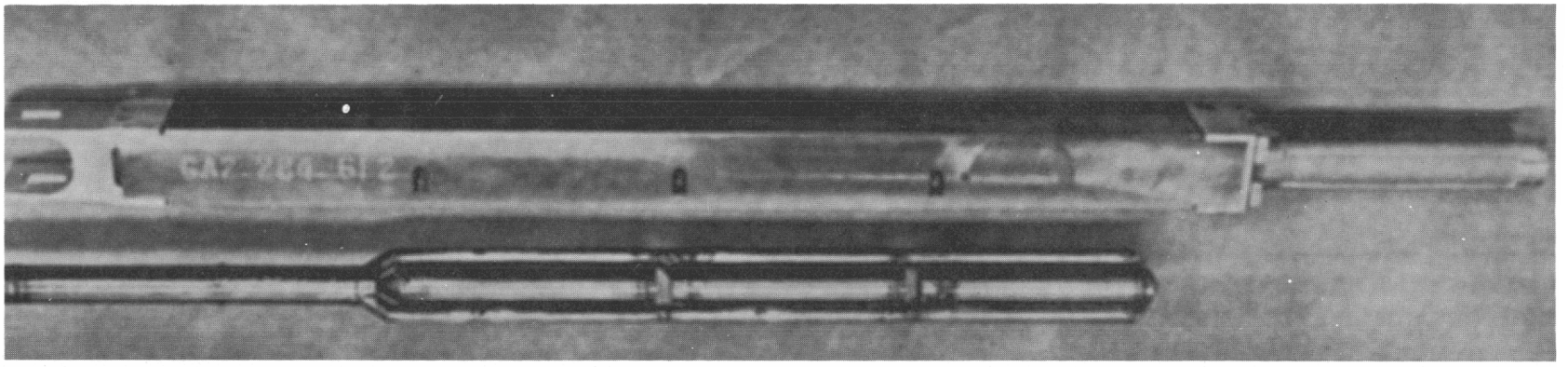
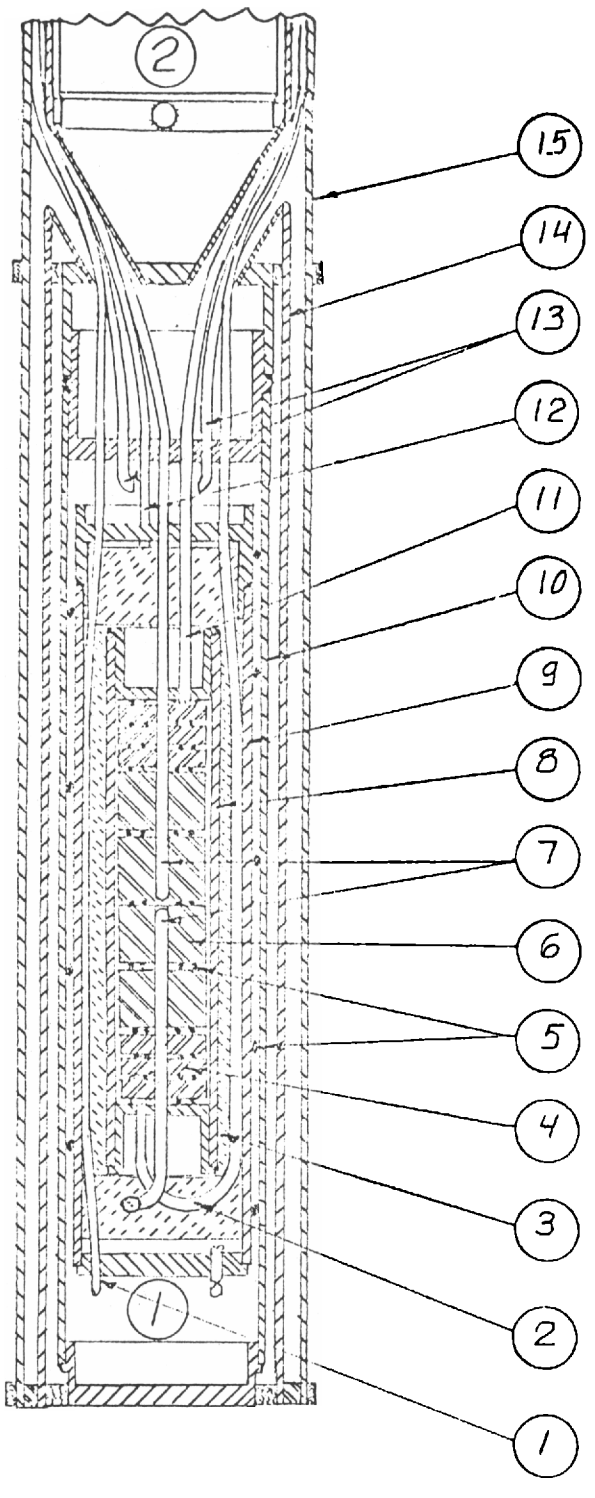


Figure 3. Complete Capsule Assembly With Be Filler Piece



1. Temperature control gas exhaust tube
2. Purge gas outlet tube
3. Thermal bond sleeve
4. BeO specimen
5. Spacer wires
6. Fuel specimen
7. Specimen thermocouples
8. Specimen can
9. Inner capsule can
10. Outer capsule can
11. Purge gas inlet tube
12. Temperature control thermocouple
13. Temperature control gas inlet tubes
14. Concentric lead tube inner shroud
15. Concentric lead tube outer shroud

Fig. 4. Schematic arrangement of typical cell in four-cell capsule

A CONTROLLED FLUX CAPSULE IRRADIATION FACILITY

W.K. McCarty
Atomics International

We have become accustomed to errors in predicting the neutron flux in irradiation positions in test reactors and to changes in flux as the cycle progresses and from cycle to cycle. It is frequently necessary to move an irradiation capsule from one position to another in order to achieve something approximating design conditions. All too frequently, the result is loss of the data, a delay in obtaining the data, or data at the wrong experimental conditions.

It has been concluded at AI that a means of temperature control is highly desirable in fuel experiments. Four methods have been employed by AI: a) Heat transfer control for relatively low heat flux experiments by means of movable fins as was described in the first capsule design meeting in 1959 at BMI; b) gas mixtures for moderate heat flux experiments such as described by other sites; c) the use of RAFT facilities at the GETR; and, d) flux controlled experiments for high heat fluxes as were described at the 1960 meeting in Germantown. One experiment of the latter type, NAA-62, has now been completed and a second is in process at the ETR.

The NAA-62 concept is an irradiation facility consisting of an annular can which contains boron trifluoride and a test element which is inserted within the annular can. The density of the BF_3 may be varied during operation to control the flux at any single given point in the test element. The reactor coolant flows through the volume enclosed by the annular can and around the outside of the can to cool the annulus and test

element independently. The purpose of this concept is to permit irradiation at controlled flux and hence at controlled burnup rate and temperature conditions.

This system removes cycle to cycle flux variations, and while it cannot stop the axial movement of the peak flux in the irradiation position, it can maintain a constant flux at one point or a constant average flux based upon the average output of a finite number of thermocouples. If desired, the annulus could probably be made eccentric, barrel shaped, or both to flatten the flux; this idea has not been explored in detail. Similarly, it could be compartmented or shaped to operate different sections of test elements at different power levels.

NAA-62-1 contained a three-rod cluster of hyperstoichiometric uranium carbide (4.8 - 5.1 w/o carbon). Each rod contained slugs 0.4-inch in diameter bonded to 12 mil thick 304 stainless steel by sodium. A NaK annulus lay between the cladding and a nickel outside tube. This assembly was inserted in the ETR on June 30, 1961, and was discharged October 2, 1961, having reached a peak burnup of approximately 3350 MWD/MTU.

The system was designed to add or remove BF_3 automatically by means of solenoid valves activated by a fuel surface thermocouple. In practice, the system never operated automatically due to internal leaks caused by failure of the valves to seat properly. It was operated by making periodic manual pressure adjustments for approximately two-thirds of the time with very close temperature control. The system was then closed off and the flux dropped in the normal fashion during the last third of the irradiation.

A CONTROLLED FLUX CAPSULE IRRADIATION FACILITY - McCarty

A new BF_3 pressure control system was built for the second experiment. This system uses a refrigerated reservoir containing liquid BF_3 and depends upon proportionally controlled electrical heaters to adjust the temperature of the reservoir and hence the BF_3 vapor pressure. It has the advantages of a closed system with no moving parts, except for the mechanical refrigeration system. An attempt to use thermoelectric cooling was not successful.

Experiment NAA 62-2, which contains two rods of hypostoichiometric UC clad in Type 304 stainless steel and one rod of hyperstoichiometric UC clad in 3Z1 zirconium alloy cladding, was inserted in ETR in August, 1963. It uses the new gas control system described above. It was expected to operate with a constant burnup rate and a peak central temperature at the control point of 1600 $\pm$ 25°F at all times when the reactor power was in excess of 125 Mw. The reactor flux, however, turned out to be only about half that desired and the peak central temperature was only 1500°F at full power with the BF_3 condensed. To demonstrate the control capability, the set point was lowered and the test was run with the central temperature at 1205 $\pm$ 20°F for a period of ten days. The set point was then returned to its original value.

The degree of temperature control achieved in these experiments was gratifying. Two examples of the effect of BF_3 pressure change on central fuel temperature are given below for each experiment:

(NAA-62-1)

Reactor Power (Mw)	Pressure (psia)	Temp. °F	% Flux Reduction
120	12	1940	42
	368	1180	
150	12	2200	29
	395	1590	

(NAA-62-2)

130	20	1170	48
	685	670	
175	25	1500	51
	715	795	

The post-irradiation examination of the annulus from the first experiment disclosed that it contained heavier deposits of B, LiF, and metal fluoride than had been anticipated. This can be explained as the result of having a significantly higher neutron flux than necessary, thus requiring higher BF_3 density to reduce the flux to that desired. Based upon information gained from this test, it should be possible to materially reduce these deposits in future tests, and this is not considered a major problem.

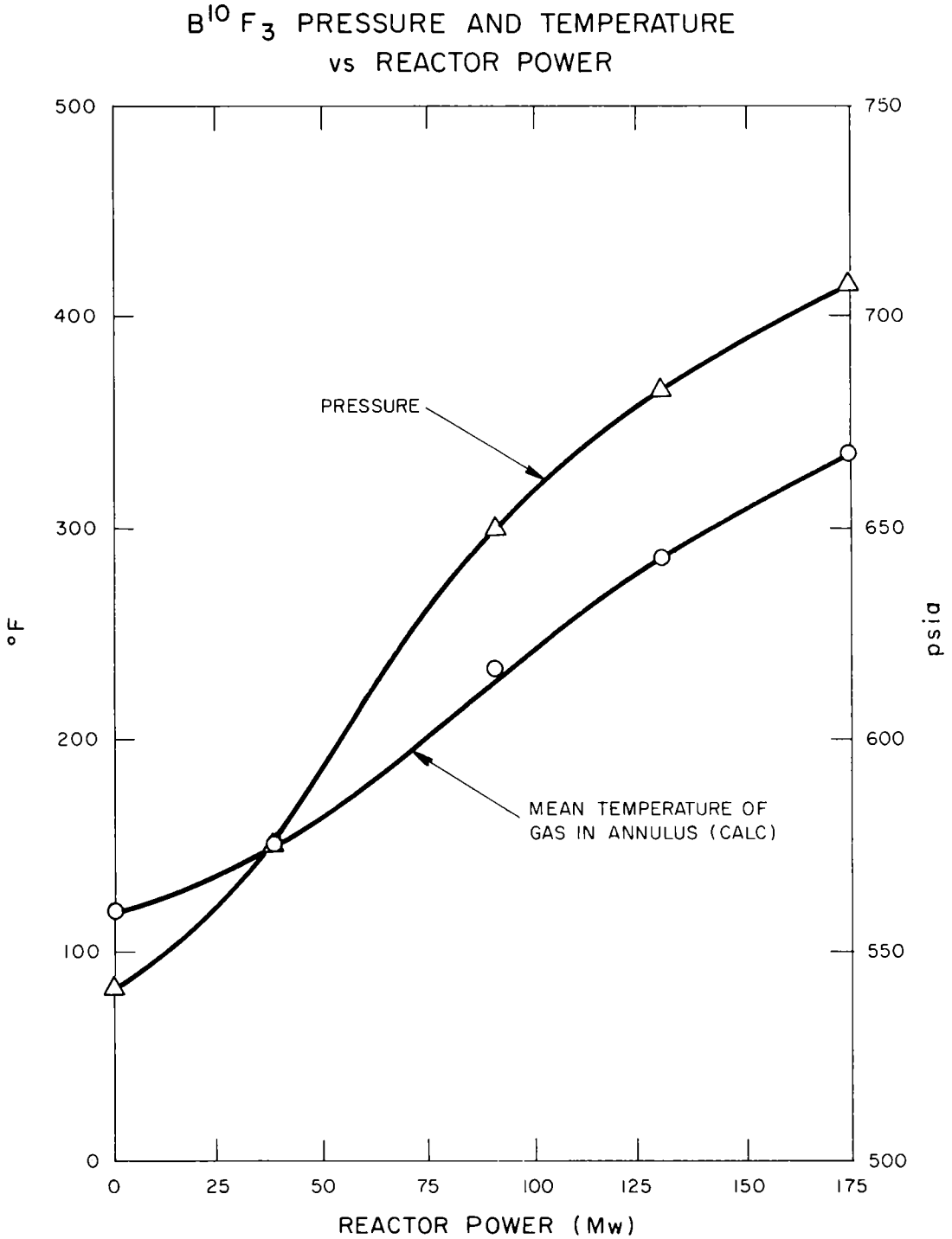
Under irradiation, BF_3 decomposes by two paths. One is the (n,α) reaction which releases fluorine, helium, and lithium fluoride; the other is the radiolysis reaction which produces boron and fluorine. The buildup of helium and fluorine tends

to form a diffusion barrier which in time would severly restrict the condensation of BF_3. For this reason, the BF_3 must be purified periodically, but this is not a difficult task. The buildup of lithium fluoride and elemental boron would tend to fill up the annulus in time; part of this material adheres to the walls and the balance falls to the bottom. We have minimized this problem by using BF_3 which is 90% enriched in boron-10; the use of this material reduces the pressure by a factor of 5 and the $B^{11}F_3$ by a factor of 41 for the same cross section.

The (n,α) reaction of boron-10 releases a considerable amount of energy to the system; this brings about an increase in temperature and pressure in the gas as the flux increases. A figure is attached which demonstrates this effect. What is not shown is the fact that part of the gas is pushed out of the annulus into the reservoir and the connecting tubing which remains cool. In the case shown, a 13% reduction in boron-10 density is calculated. The net effect is that to maintain a constant flux in an experiment, an incident flux increase must be compensated for by an even larger increase in BF_3 pressure.

The amount of boron which is required to give a specific flux reduction is dependent upon the perturbation caused by the experimental assembly itself. In the NAA-62 experiments described above, it was determined that a boron-10 content of about 2×10^{22} atoms per linear inch was required to reduce the flux by 50%. Calculations have been made for a similar three-rod test in which the neutron economy has been optimized by the removal of a significant amount of stainless steel and nickel and by a reduction in enrichment. This resulted in the ability to reduce the flux by 50% with a boron-10 content of about 7.8×10^{21} atoms per inch.

We have concluded that this type of flux control is a very satisfactory means of achieving and maintaining the desired power and temperatures in an in-pile fuel experiment. It is possible, using this control, to achieve a desired burnup in a shorter time due to the ability to maintain full power levels in the test element. The tests in the ETR were located in the first row outside the beryllium reflector and reactivity changes caused by the changes in boron content were not detected on reactor instrumentation. There is some concern, however, about the effect the system might have in an in-core position; in this position, a loss of BF_3 pressure would constitute a relatively large introduction of reactivity which would have to be within the capabilities of the reactor control system to handle. NAA-62-2 is scheduled to be discharged in February, 1964. Further application of this means of flux control depends upon its performance.


$B^{10}F_3$ PRESSURE AND TEMPERATURE
vs REACTOR POWER
500
400
300
200
100
0
750
700
650
600
550
500
°F
psia
PRESSURE
MEAN TEMPERATURE OF
GAS IN ANNULUS (CALC)
0
25
50
75
100
125
150
175
REACTOR POWER (Mw)

FLUX AND TEMPERATURE CONTROL DURING IRRADIATION

By

W. E. Eagan

FLUX AND TEMPERATURE CONTROL DURING IRRADIATION

The General Electric Test Reactor (GETR) pressure vessel which is approximately two feet in diameter is in a pool of water nine feet in diameter. There are pool experimental positions all around and directly adjacent to the outside of the vessel. These positions are holes in a water header that is held at a specified pressure during reactor operation. Into these holes are placed facility tubes which receive capsule experiments that are cooled by water from the header.

The neutron flux available to the experimenter in these pool positions has some interesting characteristics. As shown in Figure 1 the shape of the flux curve varies as the reactor cycle progresses. At the start of the cycle, when the reactor control rods are not far out, the flux has its maximum peak. As the cycle progresses and the rods move up the peak flux diminishes and moves upwards. The average flux over the length of the experimental position does not change because the reactor power remains constant. For the majority of experimenters this is completely satisfactory. There are, however, some who would like to have higher peaking, and even more experimenters who would like to have the peaking eliminated, or at least have the amount of change in the flux shape curve reduced.

For some time now, effort has been expended to improve the situation for those who find this changing of the flux undesirable. What has evolved, for use in the GETR pool, is known as the Radially Adjustable Facility Tube (RAFT). With this facility, it is possible to control the flux an experiment sees, and in turn its temperature, by moving the RAFT toward or away from the reactor core in the pressure vessel during reactor operation. The RAFT is the same size as a normal stationary facility tube in the pool. It is mounted on a carriage with a push-pull control cable which extends out of the pool so that it can be operated while the reactor is at power.

The RAFT is used most frequently with capsules to be operated at a given maximum temperature while the experiment is under irradiation. A position is selected in the reactor pool based on the predicted fluxes throughout the cycle. The capsule is installed in the RAFT facility with the facility in the full out position, i.e., the position furthest from the reactor pressure vessel. After the reactor completes the start-up transient condition and is at full power, the desired temperature, as read out on the capsule thermocouples, is reached by cranking the capsule toward the vessel. The term crank is used because the push-pull control is adapted to a hand-operated screw mechanism for fine position control. As the reactor cycle proceeds, the RAFT is repositioned periodically with respect to the vessel so that the desired thermocouple reading is maintained. Thus the control of capsule temperature is accomplished by controlling its flux environment.

There is a second benefit to be obtained by the use of the RAFT. When a capsule is placed in a staionary facility, it must be positioned so that the maximum allowable temperature or neutron flux is not exceeded at any time. This stationary position receives the same average flux the

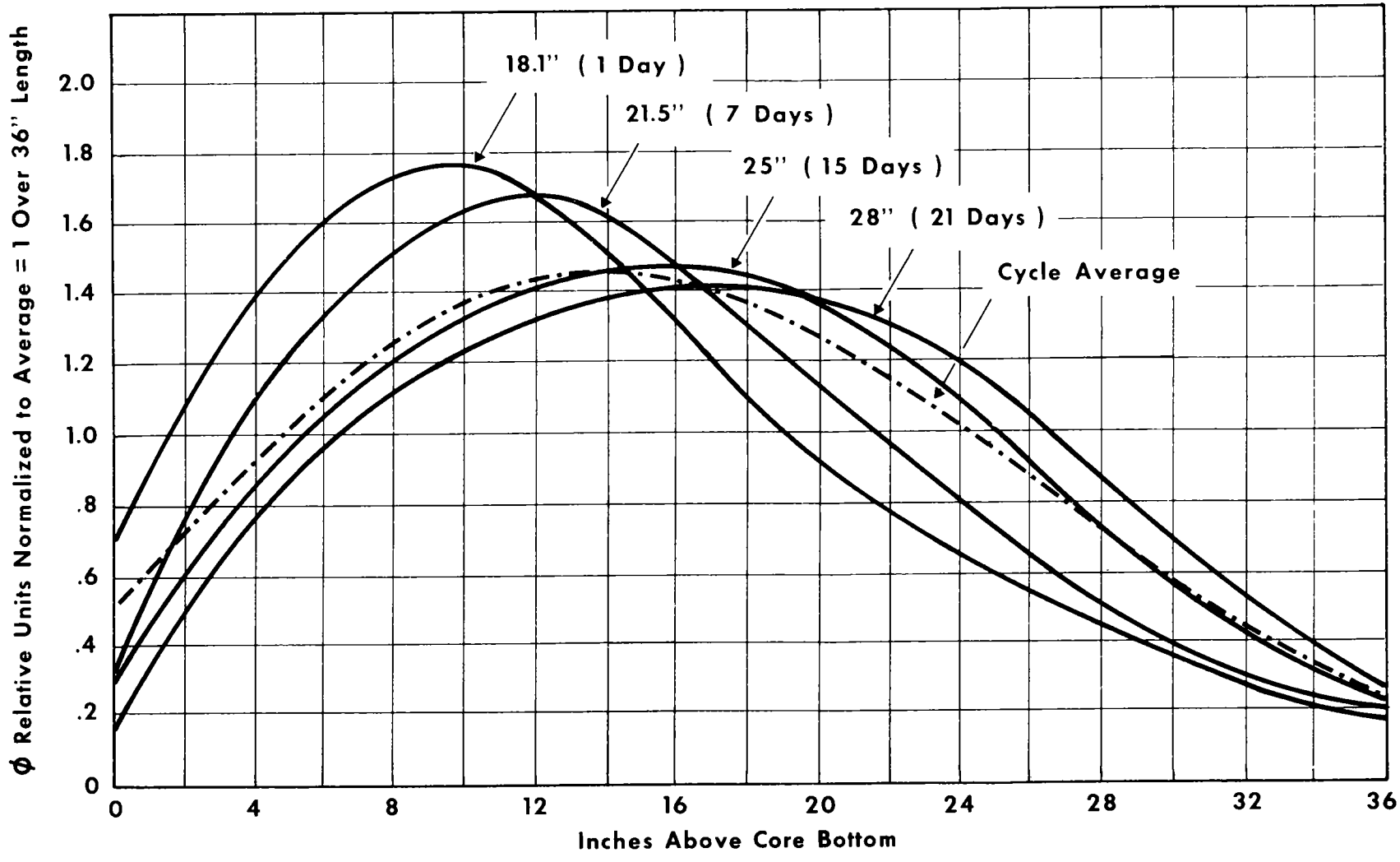


Figure 1. **AXIAL FLUX SHAPE AS A FUNCTION OF ROD BANK IN INCHES**

RAFT receives when the flux peaking is the most extreme. As all other positions of the RAFT are in a higher average flux than the stationary facility, a higher integrated exposure is received over the cycle. If a capsule is irradiated for several cycles with this extra exposure, sufficient integrated flux to save a full cycle of irradiation time can sometimes be accumulated.

While the RAFT alleviates the problem of the peak flux diminishing, its use does nothing about the fact that the flux peak moves vertically during the cycle. The capsule maximum temperature can be maintained but not at a specific location. This requires that a series of thermocouples be located over the portion of the capsule where the peak flux travels, if the maximum temperature is always to be read out. If the thermocouple in the peak region is lost during the irradiation cycle, a secondary thermocouple not in the peak can be used for control. However, it is important to take into account the fact that the flux at the secondary position is relative to that at the desired control position so that the correct control value is chosen. This requirement and the fact that some experimenters want the peak temperature and flux always to be in the same location led to the development of the Vertically Adjustable Facility Tube (VAFT). This facility is built on the same principle as the RAFT with a push-pull control extending out of the reactor pool. The first application of this vertical control is now in the reactor. It happens that the capsule being irradiated at this time does not require control of the magnitude of the peak flux but does require that the peak flux be located and maintained at a specified point in the capsule throughout the irradiation cycle.

A facility tube in GETR, shown in Figure 2, has been constructed which has both the radially adjustable and the vertically adjustable features. By looking at the flux curves it can be seen that for ordinary capsules with specimens that are not too long, this combination facility will produce essentially constant conditions in the experiment during the entire time it is under irradiation.

This combination facility was built for a somewhat unusual capsule now under construction. While the GETR core is only three feet long this capsule is nearly three-and-one-half feet long. It is desired to maintain the temperature over its length as nearly uniform as possible. The capsule has been designed with a high thermal conductivity jacket over the specimen in order to get good axial heat transfer. Then over this is a varying thickness gas annulus to provide varying insulation over the length of the capsule. By being able to maintain the peak heat source in a given location on the specimen the problem of sizing the varying insulating annulus is somewhat reduced. It is anticipated that by using the special facility to locate the capsule the experiment will receive much more uniform temperature than would otherwise be possible.

While the RAFT performs admirably for its designed purpose - maintaining a constant capsule temperature or neutron flux - it is equally adaptable for the direct opposite, cycling the temperature or flux. There is a capsule now on the drawing board in which it is desired to cycle the capsule flux while the temperature is kept fairly constant. The RAFT will be used for this in combination with a binary gas insulating annulus in the capsule. As the capsule flux is varied

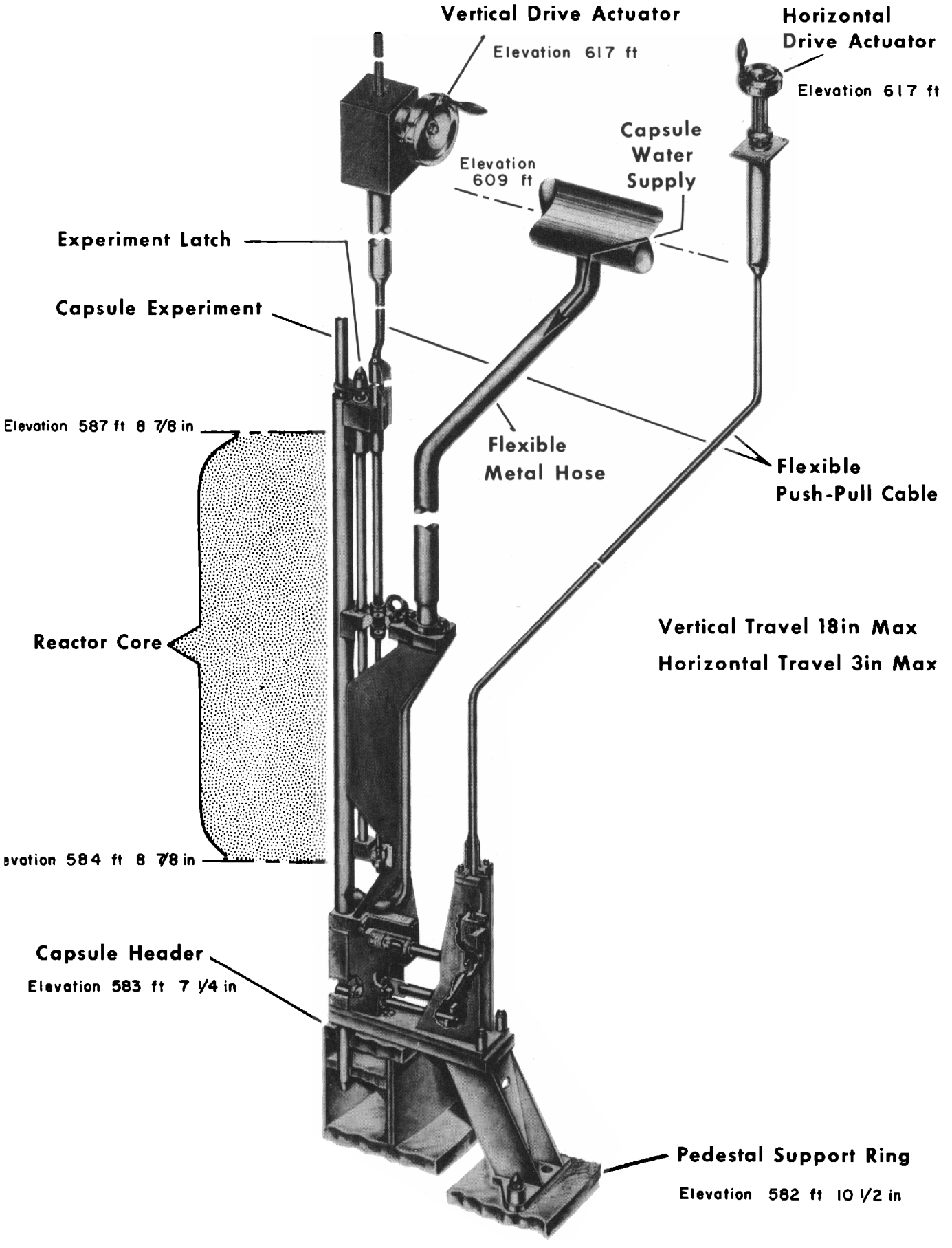


Figure 2. GETR ADJUSTABLE FACILITY TUBE

by manually cycling it toward and away from the reactor, the ratio of the gases in the insulating annulus will be changed automatically to maintain constant temperature.

Until now, the push-pull control on these facilities has always been manually operated. This means that the experimenter is dependent on operating personnel for how close to the desired temperature the experiment is actually run. Because the operator can not be constantly adjusting the facility and the flux is steadily changing, the operator needs to have a reasonable range specified within which he is required to stay. Some experimenters would like this range to be smaller than attainable at present. It is felt a reduction in the range now being obtained can be accomplished by continually adjusting the facility with an automatic control. For a RAFT it is not particularly difficult to adapt an auctioneering system to the temperature control that will select the maximum reading thermocouple as the control for a motorized drive on the push-pull mechanism. In fact, one experimenter has already offered to do this.

After it was established that a capsule position could be adjusted with the reactor operating, it was not long before some experimenters reached the conclusion that control of the coolant was also required. As previously mentioned, the pool experiment facilities receive their water coolant from a common header which is maintained at a specified pressure. The coolant flow to a particular experiment is dependent on the restriction that the capsule design presents in the facility tube. The adjustable facility was originally built so that it provided coolant from the same header in the same manner. This is not sufficient if it is desired to vary the flow, or to know precisely what the flow is for calorimetry purposes. To answer the need for flow control, a header to which remotely operated flow control valves and flow meters can be attached was installed in the reactor. The coolant is carried from this header to the adjustable facility by means of a flexible metal hose. This means of water supply is also shown in Figure 2.

By adding coolant flow control to the RAFT the experimenter can now come much closer to obtaining what he desires from his experiment. If necessary while the reactor is at power, he can have control of the experiment flux, its temperature, and of the coolant flow rate.

V. DOSIMETRY

THERMAL-NEUTRON FLUX OBSERVATIONS IN AN ORR HELIUM-COOLED LOOP

D. B. Trauger*

Flux determinations can be made in ORR helium-cooled loop No. 1. The total heat generation during irradiation is measured and thus knowledge can be obtained relative to the appropriateness of the flux and fuel-enrichment combination by means which are independent of burnup determinations and flux-monitoring techniques.

The loop is located in the B-1 position of the ORR. The fuel elements tested to date have consisted of stainless steel-clad UO_2 pellets. It was desired to irradiate these capsules in lengths up to 19 in., which extended beyond the "flat flux" region. A relatively uniform power generation was achieved by varying the fuel enrichment for groups of pellets located at the midplane and at several vertical elevations in accord with the flux profile. Successive approximations to uniform heat generation were achieved by heat transfer analyses and burnup determinations. The total heat generation in the loop was obtained from measurements of flow and the gas inlet and outlet temperatures.

A typical fuel element configuration is shown in the lower sketch of Fig. 1. Also included in the figure are plots of the flux profile and the detailed power generation rates for the fuel element. As can be seen in the figure, the measured heat balance in the loop agreed well with that calculated from the fuel enrichments and flux using the flux-depression factor of Bartells.[1]

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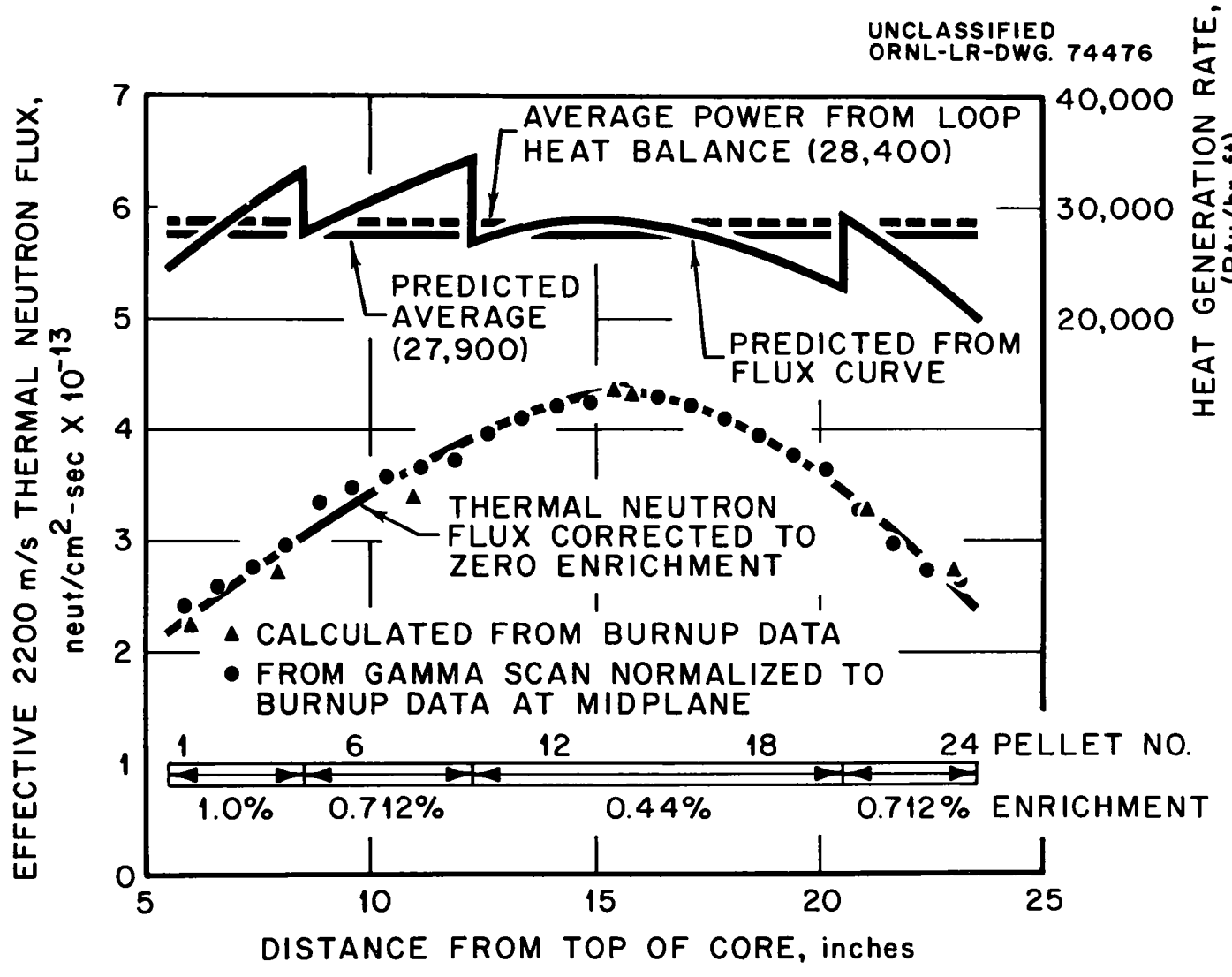
*Oak Ridge National Laboratory operated by Union Carbide Corporation for the U.S. Atomic Energy Commission.

[1]W. J. C. Bartells, Self-Absorption of Mono-Energetic Neutrons, USAEC Report KAPL-66, Knolls Atomic Power Laboratory, May 1960.



THERMAL NEUTRON FLUX AND POWER VS. LENGTH
ORR-LOOP NO. 1 FUEL ELEMENT NO. 7B

Figure 1

The Estimation of Thermal-Neutron Fluxes in Experimental Assemblies

by

W.R. Hobbs

A.E.R.E. HARWELL

Abstract

An analogue method for estimating the thermal-neutron flux distribution in experimental assemblies is described.

Notation

$\Phi(r, z)$	-	Thermal-neutron flux at any point in a cylindrical system
ϕ_∞	-	Undisturbed thermal flux
$\phi(r)$	-	Thermal-neutron flux - radial component
$Z(z)$	-	Axial flux variation
L	-	Diffusion length
L_s	-	Slowing-down length
Σ_t	-	Transport cross-section
Σ_a	-	Absorber cross-section
α	-	Argument of the assumed cosine axial distribution
k, k', μ	-	Diffusion parameter defined by relationships in the text
R	-	Equivalent resistance - various subscripts stated in text
$I_o(x), I_1(x)$ $K_o(x), K_1(x)$	-	Bessel function used in certain equations
z	-	Assembly reproduction factor
η	-	Fissile reproduction factor
A, B, C, a, b, c, d	-	constants.

1. Introduction

One of the first pieces of information required by the designer and user of a piece of experimental equipment is the estimated neutron flux in the assembly. Reactor-physics measurements in the experimental position give a reasonable guide as to the order of magnitude of the thermal flux to be expected, but when an experimental assembly is inserted there is a local flux depression due to the insertion of the absorbing materials. Figure 1 is an illustration of this effect.

There are two ways of finding the answer to this problem. The first is to make a low-power reactor test on a model of the proposed assembly. This can be time-consuming and costly if any optimisation is required. The second is to make a theoretical assessment, which can be checked by a mock-up if required. This note describes a method of making quick estimates of the flux distribution in experimental assemblies. The technique is that of an electrical analogue which has been found useful for this type of problem.

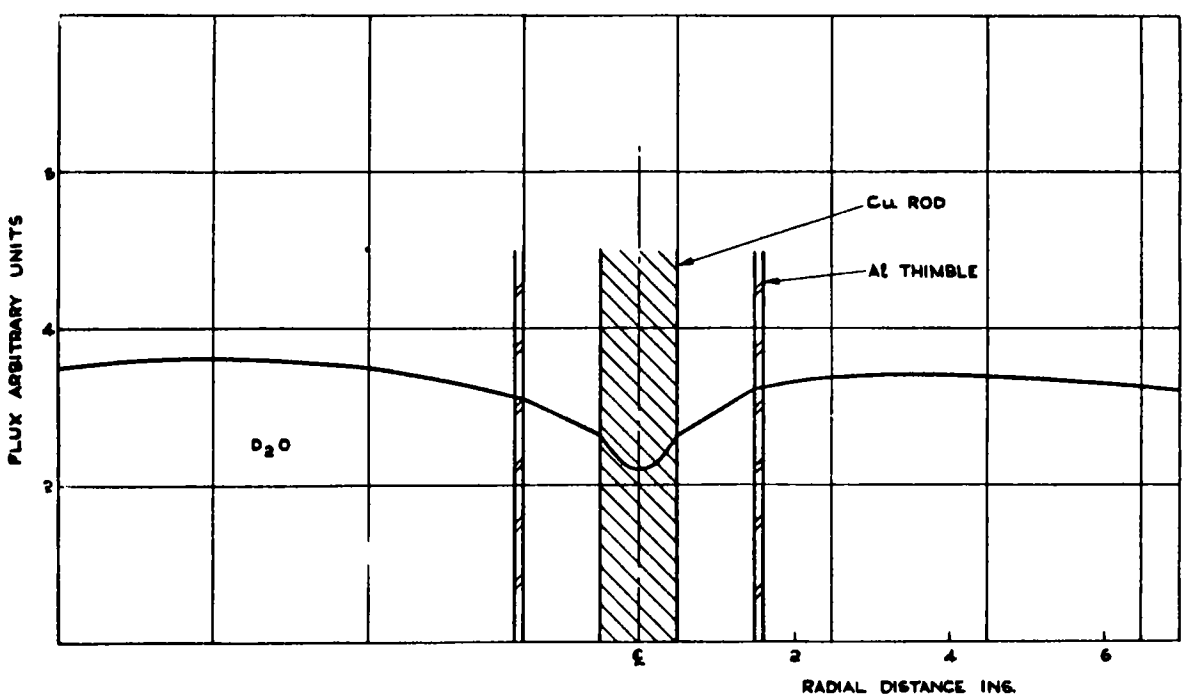


FIG. 1

2. The Simple Analogue Method

In order to obtain a reasonable working system the procedure adopted was to use the simplest mathematical model and then to fit into the equations constants obtained from some mock-up experiments.

The model taken was that of an infinite slab reactor into which a cylinder of absorbing material has been placed (see Fig. 2).

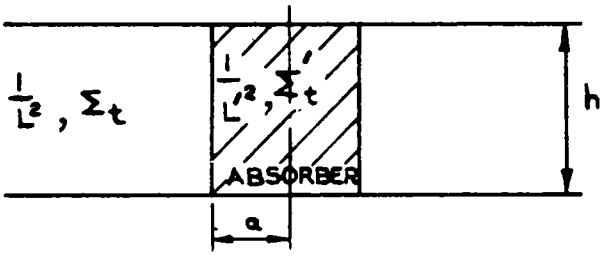


FIG. 2

This simple model can be solved by diffusion theory, the axial solution being

$$Z(z) = \cos \alpha z = \frac{\pi}{h} \qquad \ldots\ldots\ldots(1)$$

and the radial solutions are

In the moderator

$$\phi = \phi_\infty - A K_o(kr) \qquad \ldots.(2)$$

In the absorber

$$\phi = B I_o(k'r) \qquad \ldots.(3)$$

where

$$k^2 = \frac{1}{L^2} + \alpha^2 \qquad k'^2 = \frac{1}{L'^2} + \alpha^2$$

and ϕ_∞ is the undisturbed flux in the system.

These equations can be solved by equating flux and current at the interface. However, in order to solve more complex systems it is more convenient to replace these equations by an electrical analogue form (Fig. 3).

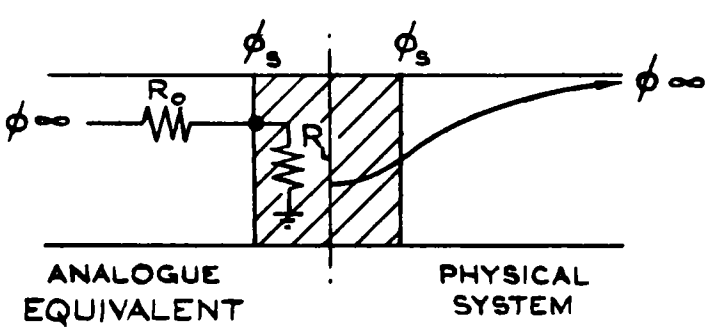


FIG. 3

For this system the fluxes are considered as voltages and the equivalent resistances are given by:-

$$R_o = \frac{K_o(k_a)}{\frac{2\pi}{3}\, kaK_1(ka)} \sum_t \quad \ldots.(4) \qquad R_L = \frac{I_o(k'a)}{\frac{2\pi}{3}\, k'a\, I_1(k'a)} \sum_t' \quad \ldots.(5)$$

For this simple model the flux at the surface of the absorber is given by:-

$$\frac{\phi_s}{\phi_\infty} = \frac{R_L}{R_o + R_L} \qquad \ldots.(7)$$

and the absorption rate

$$I \qquad \frac{1}{R_o + R_L} \qquad \ldots.(8)$$

3. Extension to Composite Cylindrical System

The majority of experimental assemblies can be considered to consist of a central absorber surrounded by a series of absorbing, scattering or void annuli. The central absorber can be considered as above by evaluating R_L for the appropriate material and dimensions. The remaining components are the absorbing and scattering tubes and voids.

3.1. Absorbing Scattering Tubes

In this case two effects have to be considered. Firstly the flux drop across the annulus. From diffusion theory the equivalent transmission resistance R_s of the tubes is given by

$$R_s = \frac{3}{2\pi} \log \frac{b}{a} \sum_t \qquad \ldots.(9)$$

The second effect is the loss of neutrons due to absorption. If $\bar{\phi}$ is the average flux in the tube then the number of neutrons absorbed is given by

$$I = \bar{\phi} \sum_a (b^2 - a^2) \text{ per unit length}$$

From this the equivalent absorption resistance is given by

$$\frac{1}{R_a} = (b^2 - a^2) \sum_a \qquad \ldots.(10)$$

In the equivalent electrical network for a tube the absorption resistance is connected to the transmission resistance at a tapping so that neutrons are considered to be absorbed at the average flux in the tube (Fig. 4).

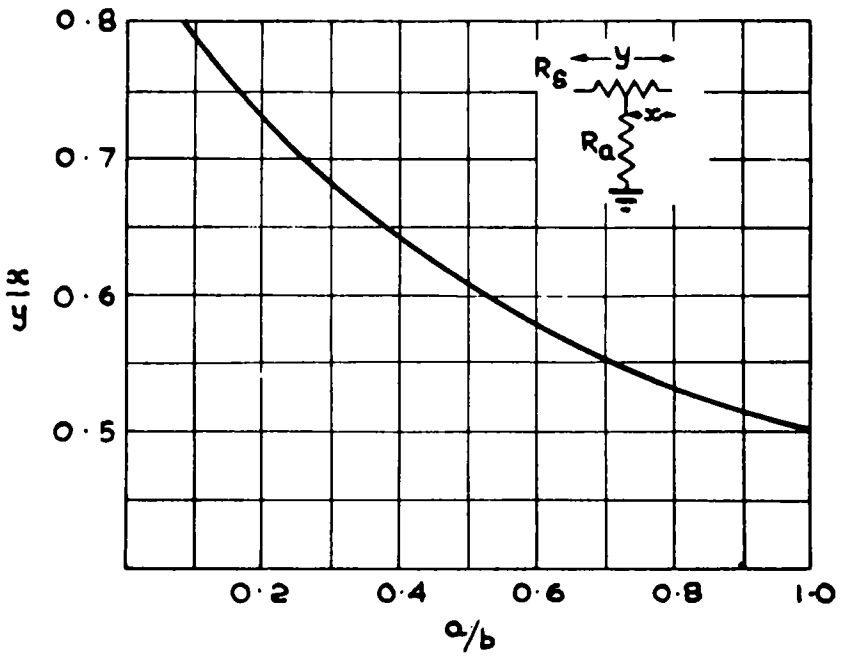


FIG. 4 TAPPING FRACTION.

For thin annuli a good approximation is to assume a centre tapping. The equivalent network for a tube is shown in Fig. 5.

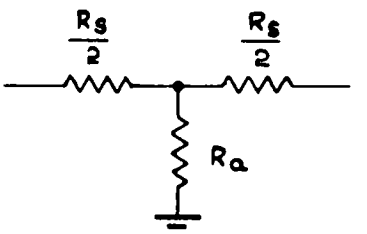


FIG. 5

3.2 Voids

As with the absorbing tube two properties of voids need to be considered. The flux drop across a void is considered as a transmission resistance R_N, which is given by

$$R_N = \frac{1}{\pi a} \left\{ 1 - \frac{2}{\pi} \left(\sin^{-1} \frac{a}{b} + \frac{a}{b} \sqrt{1 - \frac{a^2}{b^2}} \right) \right\} \quad \ldots (11)$$

The second effect is due to streaming. As experiments usually take place in the high-flux region in the reactor a void is in effect an absorber. There are a number of ways in which this streaming resistance R_{st} can be expressed, but the best fit to the mock-up tests performed was found to be

$$\frac{1}{R_{st}} = C \quad \alpha^2 (b^{\frac{3}{2}} - a^{\frac{3}{2}})$$

where C is a constant for the particular reactor.

The electrical network for a void is given by

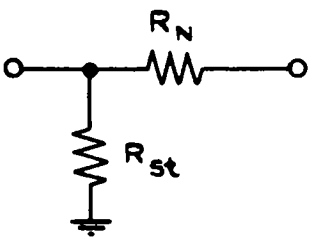


FIG. 6

3.3. The Simplest Composite System

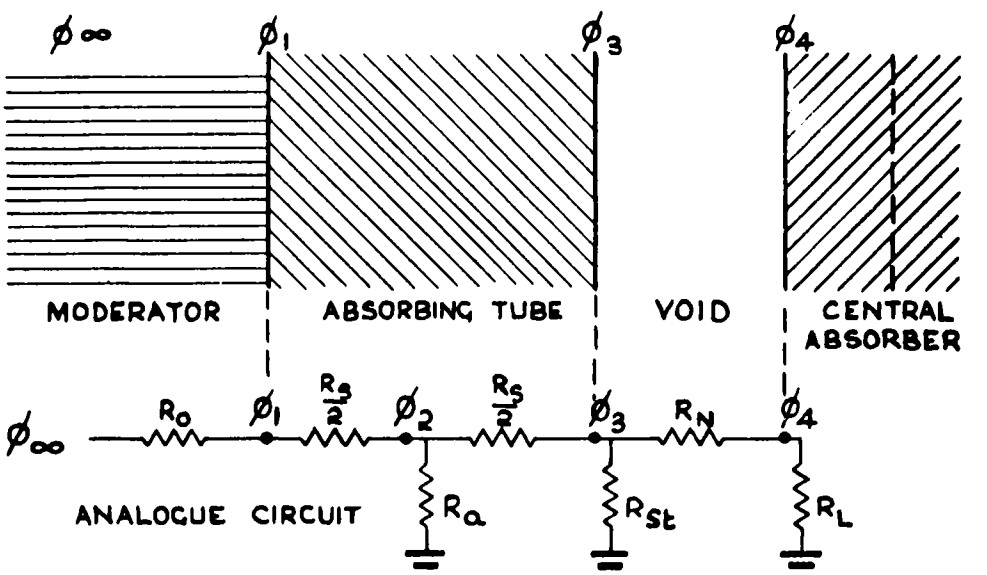


FIG. 7

The simplest composite circuit simulating a central absorber, void, and absorbing tube is shown above. More complicated systems can be built up by adding the required units to the network.

4. Modified Analogue

The assumption made in this modification of the simple analogue is that moderation of fast neutrons does not occur inside the absorbing system. The fast neutrons leaving the system slow down in the moderator and increase the local thermal flux. This feedback effect can be accounted for by a modification to the moderator resistor which can be written

$$\bar{R} = R_o - Z \frac{L^2}{L^2 - L_s^2} (R_o - R_o')$$

where $R_o' = \frac{K_o(\mu a)}{\frac{2\pi}{3} \mu a K_1(\mu a)} \Sigma_t$

$$\mu^2 = \frac{1}{L_s^2} + \alpha^2$$

and Z = $\frac{\text{fast-neutron current out}}{\text{thermal-neutron current in}}$

or in terms of the analogue

$$Z = \eta \frac{\phi_L}{R_L} \Big/ \frac{\phi_s}{R_T}$$

where ϕ_s = assembly surface flux

η = fuel reproduction factor

R_T = total 'resistance' of the absorbing assembly

5.

The types of resistance mesh can be used to represent any layer of a coaxial system in analogue form and can be solved by the simple use of Kirchoff's law. This method has been in use for a number of years and has been found to give reliable estimates of the flux distribution in reactor experiments.

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THE CALCULATION OF THERMAL FLUXES IN A CAPSULE IN A HETEROGENEOUS REACTOR

by J. P. GENTHON

I. One of the methods currently employed for performing irradiations in heterogeneous reactors consists of using locations normally taken up by fuel cells to position the capsules containing the material, fissionable or not, to be irradiated.

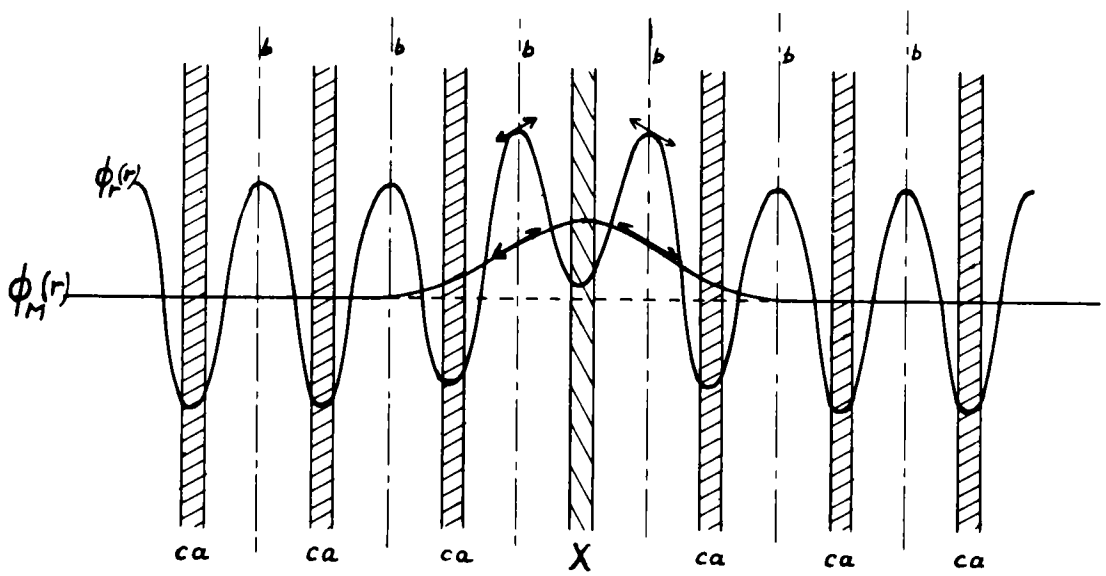
In particular, in the French reactor EL-3, 30% of the locations normally taken up by fuel cells are used for this purpose. It was in this way, for example, that the major part of the studies on Uranium-Molybdenum alloy fuel elements at high neutron fluxes were carried out. A large number of other irradiations has also been performed under these conditions: studies of metallic or oxide fuel, studies of claddings, studies of structural materials, etc.

It has been necessary to know how to define the fluxes in such irradiations. Independent of the methods of measurement, a systematic calculation method has been developed which takes into account, in particular, the heterogeneous nature of the reactor /1/. We summarize here the principles of that method which has shown itself to be quite satisfactory and which is used for all the french irradiations of this type in heterogeneous reactors (EL-2, EL-3, and foreign reactors).

II. Let X denote the special cell made up of the irradiation capsule and c.a. the normal fuel cell (the actual cell). The flux in a cell c.a. is supposed known, it being a characteristic of the reactor. The problem is to determine the flux produced when the cell c.a. is

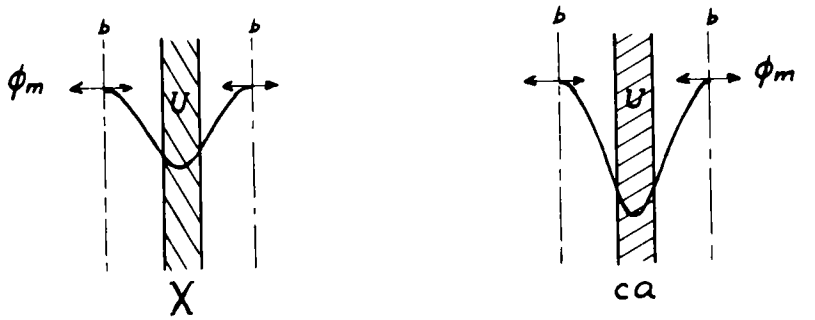
replaced by X, that is, to determine the ratio $\phi x/\phi_{ca}$.

If one corrects for the homogeneous radial flux distribution in the reactor, that is if a cylindrical medium of infinite radius is considered, the flux distributions have the following forms:



Let $\phi_r(r)$ be the real flux and $\phi_M(r)$ the "macroscopic" flux corresponding to the homogenization of each of the cells. ϕ_M is constant when all the cells are identical, and returns to that constant value at an infinite distance from the perturbation X.

Let ϕ_m be the "microscopic" flux defined by $\phi_m(r) = \dfrac{\phi_r(r)}{\phi_M(r)}$. In particular it is the real distribution when $\phi_M(r)$ is constant.



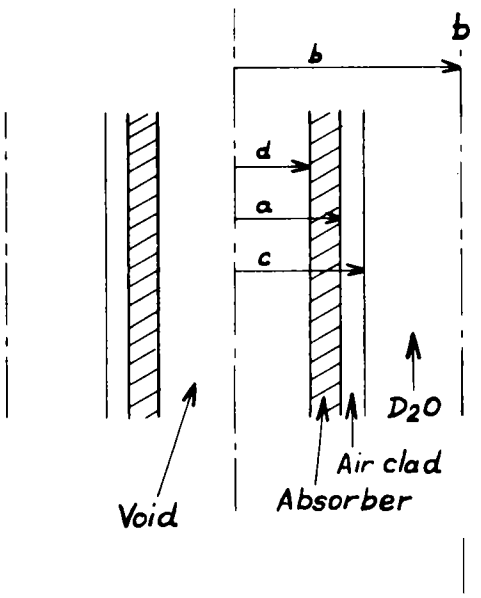
The basic assumption is made that $\phi_m(r) = \frac{\phi_r(r)}{\phi_M(r)}$ is independent of $\phi_M(r)$. From this there follows that:

- $\phi_m(r)$ can be calculated for the case $\phi_M(r)$ = constant, that is, under the condition $\phi'_m(b) = 0$.
- $\phi_{mx}(b) = \phi_{mca}(b)$ because of continuity at b.

The absolute magnitude of $\phi_{mca}(b)$ is known or may be calculated from $\phi_{mca}(U)$ which is a characteristic of the reactor and is constant in the framework of the assumption made above.

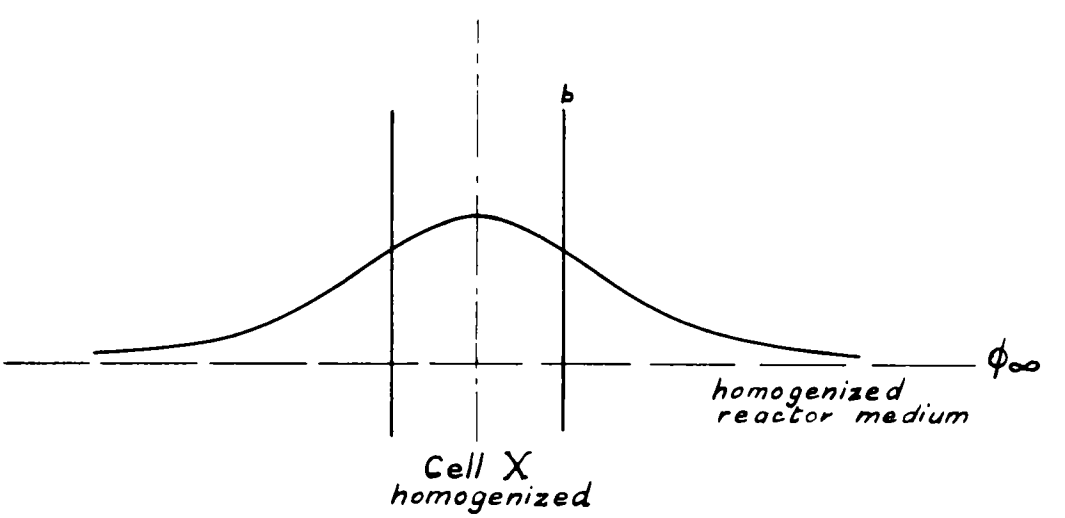
III. Thus, we may proceed to the calculation of ϕ_{mx} and of ϕ_M according to the following principles:

ϕ_m: the absolute value of ϕ_{mx} is known at b. In addition, a calculation analogous to that for the thermal utilization, f, gives us, in the cell X the relative flux distribution; the calculation used here is directly related to that of /2/. The methods of /2/ minimize the approximations of diffusion theory (of the greatest importance here) in the neighborhood of the irradiation sample, in general fairly absorbing.



The sketch at the left shows the cell studied (X or c.a.). The Use of /2/ allows us to calculate the ratio $\phi_c/\phi_{absorber}$ and the slope of the flux at c, an artificial slope which permits one to obtain a correct flux distribution between c and b by applying one group diffusion theory with homogeneous sources. Thus the absolute value of the distribution $\phi_{mx}(r)$ may be determined.

ϕ_M: The cell X is homogenized inside the limit b. The rest of the reactor is replaced by the equivalent homogeneous medium (the classical calculation of equivalent homogeneous cells in heterogeneous reactors).



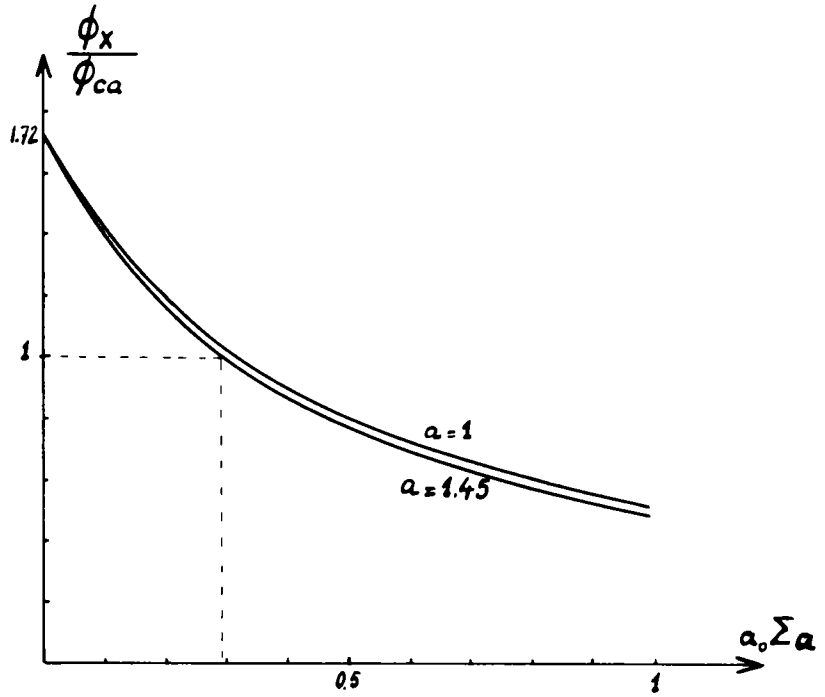
The two regions thus defined are considered as not containing fissionable material, but with constant homogeneous thermal sources, q.

One group diffusion theory may then be used to calculate the variation of $\phi_M(r)$, i.e. $\phi_M(r)/\phi_{M\infty}$.

Thus one knows, within a constant multiplication factor, the distribution $\phi_r(r) = \phi_m(r) . \phi_M(r)$. In particular, at the position U of the irradiation sample $\phi_r(U) = \phi_m(U) . \phi_M(U)$.

IV. The same calculation can be made separately for the cell X and for the actual cell and one can then calculate the ratio $\phi_{rx}(U)/\phi_{rca}(U)$ which will be abbreviated ϕ_x/ϕ_{ca}. This ratio is independent, in particular, of the flux level in the normal cell (the actual cell) of the reactor.

Its variation can be calculated once and for all, for a given reactor, as a function of the reduced parameters of the cell X. The following curve is obtained for the reactor EL-3:



with Σ_a the absorption cross section of the absorber and

$a_o = \frac{a^2 - d^2}{a}$ (see the Figure on page 4)

A similar curve may be established for the variation in flux induced by X at the neighboring cell position.

At the present time the results are quite satisfying. A measurement made on EL-2 at low power for $a_o \Sigma_a = 0$ gave an excellent agreement between the measured and calculated values of ϕ_x/ϕ_{ca} (difference less than 3%). With the reactor at power difference of 10% are often found between the predicted and measured fluxes (by the use of specific powers, temperature, etc.).

But it is not improbable that these difference stem from an inaccurate knowledge of the quantity ϕ_{ca} itself.

Studies now underway should lead to an improved version of this method: a better determination of the artificial slope at the point c /3/, the effect of the finite size of the reactor, and source perturbations, a determination of flux repartition inside the absorber.

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A SEMI-EMPIRICAL DETERMINATION OF FAST AND INTERMEDIATE NEUTRON FLUX SPECTRA IN A REACTOR.

J.P. GENTHON

I. INTRODUCTION.

The detailed analysis of the physical, or physico-chemical reactions produced in a reactor by the kinetic energy of neutrons requires the knowledge of two functions : the energy distribution of the neutron flux, $\varphi(E)$, and the efficiency function for the neutrons, $\eta(E)$; thus one has, in relative or absolute units depending on the definition of η , the total energy attributed to the reaction under consideration, $W = \int_0^\infty \eta(E)\,\varphi(E)\,dE$.

If one wants to generalize the calculation of W it is necessary, in particular, to know the general form of the distributions $\varphi(E)$; a generalized calculation of W for the case of Wigner effects has been made in [1] based on elementary hypotheses concerning the distributions $\varphi(E)$: These hypotheses should be reviewed and improved, and it is the purpose of the present note to give a formulation $\psi(E)$ of the real fluxes $\varphi(E)$; the attempt is made to provide a formulation which is as general as possible and which, at the same time, uses only proven measuring methods ; it is valid, in principal, in most cases of so-called thermal reactors and for energies greater than a few hundred electron volts.

II. CURRENT METHODS FOR THE DETERMINATION OF FAST AND INTERMEDIATE NEUTRON FLUXES.

These are either calculational or experimental méthods.

The calculations use various methods such as diffusion theory ([2] and [3] for example), the collision methods

([1] for example) and Monte Carlo methods ([4] for example). In the current state of affairs it may be said that the calculations are often excessively unwieldly, that only particular or simplified geometries can be treated by them, and that they must be repeated for each new situation - all without producing a sufficient knowledge of the accuracy of the final results obtained. In all cases, no matter what the accuracy of the calculation, one is limited by the uncertainties with which the cross sections are known, and, in particular, the uncertainties in the anisotropy of high energy scattering.

The results of calculations corresponding to [1], [2], [3] and [4] have been used here to compile a list of possible reactor spectra, a list that can be called qualitative. In fact, these spectra have been put in the form of parametric formulas ; each set of values of the parameters characterizes a calculted spectrum. The hypothesis made here is that it is likewise possible, using the same formulas, to define values of the parameters which define the real spectra ; these values will be defined on the basis of experimental results.

It can be seen that the hypothesis that is made is compatible with the confidence, in general quite limited, that one may have in the calculations. In other words, the procedure explained here consists in using the calculation only as a means of interpolation between the areas determined by measurement.

The experimental methods currently proven are essentially those which use detector activation under neutron bombardment. These are so-called "resonance" detectors in the low energy region, between about one eV and one KeV and, at high

energies, above about 2 MeV, so-called "threshold" detectors.

The resonance detectors in which the resonance is sufficiently high so as to make up an appreciable part of the total activation of the detector are essentially indium, gold, cobalt, and manganese, [5], whose resonance energies are, respectively, about 1.5 ev, 5 ev, 130 ev and 340 ev. If, in the energy region between the cadmium cutoff (i.e. about 0.6 ev) and some 400 ev, the flux is of the form K/E the use of any one of these detectors is sufficient to determine the constant K. In fact, in practice, this ideal case is not usual. Heterogeneous effects, absorption, resonant capture in uranium-238, and leakage result in the fluxes differing more or less from the ideal K/E law. This can be seen in Figure 1 which represents, as a function of the energy, E, the product $E\varphi(E)$ calculated for different reactor cases and normalized at unity for $E = 3$ KeV. It can be seen that it is only in the neighborhood of 3 KeV that the constant K may be defined, and only if certain cases of D_2O and graphite reflectors are excluded. The flux in the region of the manganese resonance is still not very perturbed so that one can consider the use of a manganese detector, along with other detectors to estimate the non-resonance Mn activation, to determine the flux in the vicinity of 340 ev (see [6]). A boron covered sodium detector, [7], sensitive mainly around 3 KeV will obviously be greatly preferable when it will be available.

The most usual threshold detectors [8] are those of S^{32} (n,p), P^{31}(n,p) and Ni^{58}(n,p) which have thresholds at about 2.7 MeV and Al^{27}(n,α) with a threshold at 7.5 MeV. If the flux to be measured is of the form $A\,\mathcal{N}(E)$, the use of a single detector permits the determination of A if $\mathcal{N}(E)$ is known ; the

classical hypothesis of an $\mathcal{N}(E)$ law corresponding to a fission spectrum, $N_o(E) = E^{0.5}\ e^{-0.775\ E}$, is not general ; Figure 2 shows how real spectra differ in practice from the elementary $N_o(E)$ law. In the end, we have kept different forms for the fluxes in this energy region, i.e. for the fast fluxes, depending on the moderator of the reactor under consideration ; these forms may be expressed, in a generalized manner :

$$\psi_r(E) = A\,\mathcal{N}(E) = A\ (E^{\nu_1}\ e^{-\beta_1 E} + d\ E^{\nu_2}\ e^{-\beta_2 E})$$

We shall come across this formulation again further on in the framework of the general formulation for the flux for all energies. In particular, only two detectors at most , and a knowledge of the nature of the reactor's moderator, are required to entirely define the form $\psi_r(E)$ given above.

III. GENERAL FORMULATION OF THE SPECTRAL DISTRIBUTION.

We assume that the results of the following measurements are available :

- $\dot{K}$ measured near 3 KeV or, if necessary, near 340 eV, according to the principles above, for example ; we have, therefore, the asymptotic flux distribution, $\psi(E) \rightarrow K/E$ as the energy decreases toward about 3 kev ; this distribution, depending on the circumstance, may or may not be correct for energies below a KeV (see Figure 1). In any case, the energy region below one KeV presents no interest for the application considered here (see the Introduction).

- $\mathcal{A}_i = \int \sigma(E)\,\varphi(E)\,dE$ the response of the

threshold detector $S^{32}(n,p)$ irradiated in the flux $\varphi(E)$ studied. $\mathcal{A}$ is here expressed in mb.n/cm^2/sec.

- finally $\mathcal{A}_j$, the response of the $Al^{27}(n,\alpha)$ detector irradiated in the same flux, or, which amounts to the same thing, the ratio $\rho = \frac{\mathcal{A}_i \mathcal{A}_{oj}}{\mathcal{A}_j \mathcal{A}_{oi}}$, the terms $\mathcal{A}_o$ being the measured or calculated responses of the detectors in a fission spectrum.

Various considerations which are developed in [6] lead to the establishment of the general formulation $\Psi(E)$ below, which is entirely defined if one knowns the measured values K, $\mathcal{A}_i$, and finally ρ, as well as the nature of the moderator of the reactor under consideration (H_2O, D_2O, C) :

$$\Psi(E) = K\left[\Psi_o(E) + h\,\Psi_e(E)\right]$$

with :

- $\Psi_e(E)$: heterogeneous component of the flux ; its generalized form is of the type $E^\nu e^{-\beta E}$. This form is a function of the nature of the moderator of the reactor and/or of the results of the measurements.
- h : relative intensity of the heterogeneous component ; this parameter is a function of the results of measurement.
- $\Psi_o(E)$: the homogeneous component of the flux ; it corresponds approximately to the case of an infinite homogeneous medium ; it is of the form :

$$\Upsilon(V-E)\frac{e^{b\sqrt{E}}}{E} + \Upsilon(E-V)\ F\ E^{\nu}\ e^{-\beta E}$$

This form is characteristic of the nature of the moderator of the reactor. The values of the parameters of $\psi_o(E)$ are, in particular, such that there is no discontinuity either in the flux, or in its derivative at the junction V of the two functions $\frac{e^{b\sqrt{E}}}{E}$, for $E < V$ and $F E^{\nu} e^{-\beta E}$ for $E > V$ (the function Υ is the classical function $\Upsilon(x > o) \equiv 1$ and $\Upsilon(x < o) \equiv 0$).

- K : Flux per unit lethargy $\psi(u) = E\psi(E)$ measured at low energies (section II).

It will be necessary, in order to define more precisely these forms, to consider separately the different possible types of reactor (D_2O, H_2O and C), which will be done below.

If we continue to consider the principles, in order to fix the ideas, certain simplifying representations of $\psi_o(E)$ and $\psi_e(E)$ may be adopted - for example (Figure 3)

$$\psi_o(E) = \Upsilon(1 - E)\ \frac{e^{1.45\sqrt{E}}}{E} + 9.25\ \Upsilon(E-1)\ N_o(E)$$

$N_o(E)$ is the fission distribution $E^{0.5}\ e^{-0.775\ E}$ and b has been taken = 1.45 $MeV^{-1/2}$, i.e. V = 1 MeV and F = 9.25 (which corresponds to the case of H_2O)

and :

$$\psi_e(E) = N_o(E)$$

The term $\psi_e(E)$ corresponds qualitatively to a heterogeneity of the medium, to an addition or a withdrawal of fission sources ; it is logical, on first approximation, to represent it as $N_o(E)$.

The flux $\psi(E)$ may thus be expressed according to a function $\psi_o(E)$ which may qualitatively be made to correspond to a homogeneous medium, corrected by a heterogeneous term.

At low energies $\psi_e(E)$ is nul so that :

$$\psi(E) = K\,\psi_o(E) \simeq K/E, \quad \text{the asymptotic}$$

value of the flux, as has been seen above.

At high energies, above V, i.e. here above 1 MeV, the flux becomes :

$$\psi(E) = \psi_r(E) = K\,F\,E^{\nu_1}\,e^{-\beta_1 E} + h\,K\,E^{\nu_2}\,e^{-\beta_2 E}$$

or here :

$$\psi(E) = \psi_r(E) = K\,(F+h)\,N_o(E) \qquad \text{with } F = 9.25$$

which is the elementary form of the fast neutron flux.

It can be seen that the proposed formulation $\psi(E)$ interpolates between the two regions of high and low energies which are accessible to the experimental methods.

The formulation is entirely defined by two measurements : K is measured at low energies, and the use of a single threshold detector determines K(F+h), that is to say h, in the framework of the elementary hypothesis above. In the general case, the use of a second detector with a different threshold

is necessary ; we shall see briefly below how this works, by the use of the ratio ρ .

IV. NUMERICAL VALUES.

The practical use of the principles of Section III, applied and adjusted to the cases of heavy water, light water, and graphite reactors, will be summarized here (see [6] for more details).

We have, thus :

$$\Psi(E) = K\left[\Psi_o(E) + h\ \Psi_e(E)\right]$$

with, in the case of the particular type of reactor (energy expressed in MeV) :

Heavy water :

$$\Psi_o(E) = \Upsilon(1.75 - E)\ \frac{e^{0.22\sqrt{E}}}{E} + 2.243\,\Upsilon(E - 1.75)N_o(E)$$

$$\Psi_e(E) = N_o(E) + e^{-2E} + r\ e^{-vE}$$

K, as we have seen, is a measured quantity ;
r and v are corrective adjusted parameters, given in Figure 4 as a function of h ;
h is calculated from the measured ration $\mathcal{A}_i/K$, and, as long as $h < 1.2$:

$$h \simeq \left(\frac{\mathcal{A}_i}{K}\right)^{mb}\ \frac{1}{85.7} - 2.22$$

for $h > 1.2$ see Figure 4.

This distribution is valid for $h > -1.5$, that is to say in the core of the heterogeneous heavy water reactor and up to about 20 cm into the heavy water reflector.

Note : at high energies, $\psi(E)$ becomes proportional to $N_o(E)$; a more general form, $\psi_r(E) = A\ E^{0.5}\ e^{-\beta E}$ may be preferable for $E > 4.5$ Mev to the extent that β is effectively different from 0.775. A and β are derived from the measured ρ and $\mathcal{A}_i$:

$$\beta = f(\rho) \qquad \text{and} \quad 339\ A/\mathcal{A}_i = \Lambda(\beta) \qquad \text{Figure 5}$$

Light Water :

$$\psi_o(E) = \Upsilon(I-E)\ \frac{e^{1.45\sqrt{E}}}{E} + 9.251\ \Upsilon(E-I)\ N_o(E)$$

$$h\ \psi_e(E) = h\ N_o(E) + d\ \frac{A}{K}\ E^5\ e^{-\beta E}$$

$$h = A/K - 9.251$$

$$A/K = \frac{\mathcal{A}_i}{K}\ \frac{1}{84.75}\ \frac{1}{1+d\ R_i}$$

$\rho\ (I + d\ R_i)$, d, and β are given as a function of the result of the ρ measurement in Figure 6.

K, here, is not exactyly the value of $E\,\varphi(E)$ measured at low energy because of the large value of $b = 1.45$ which implies

a slight variation of $\psi(E)$ at low energies, while $\varphi(E)$ is, in general, constant ; $K = 0.923\ E\ \varphi(E)$ measured at 3 KeV or at 340 KeV or even at lower energies, depending upon the case (see Figure 1).

No limitation has been found to this formulation which thus applies to all the core and H_2O reflector cases of H_2O reactors.

Graphite.

$$\psi_o(E) = \Upsilon(I - E)\ \frac{e^{0.28\sqrt{E}}}{E} + 3.1\ \Upsilon(E - I)\ e^{-0.85E}$$

$$\psi_e(E) = e^{-0.775E} + r\ e^{-7.5E}$$

K is measured at 3 KeV, or if necessary at 340 eV

$$h = \frac{1}{43.3}\ \frac{\mathcal{A}_i}{K} - 2.33$$

r is a corrective adjusted parameter :

$r = |h|\,(2.50 - 0.288\ h)$.

That distribution is valid for $h > -1.5$, that is to say, in the core and up to 20 cm into the reflector of the graphite reactor.

V. CONCLUSIONS.

Such a formulation does not claim to take into account local flux variations ; such local perturbations are met with, on one hand, at low energies where they require particular experimental techniques (Section II) and where the proposed formulation deos not always apply, its general region of application

being at energies above a few kev ; these perturbations are met with, on the other hand, at high energies (a few mev) in the case of the graphite reactors, where the formulation proposed is a smoothed representation of real curves which vary too abruptly to be exactly represented.

Figure 7 shows several examples comparing the calculated fluxes (Section II), $\Psi(E)$ and their representation, $\psi(E)$, determined, as we have just seen, from the pseudo-measurements, $\mathcal{M}_i$, K and ρ. The pseudo measurements are the responses, here calculated, of the various detectors in the flux to be represented, $\varphi(E)$. Elsewhere it has been verified that, in the entire region of applicability of $\Psi(E)$, the differences between flux integrals were always less than 5 % at the greatest, and similarly for the integrals of the product flux x energy.

Various comparisons with experimental results, and various applications have shown themselves to be satisfactory. Fast flux measurements in light water reactors ([9] and [10]) are in good agreement with the present formalism (see [6]) ; similar agreement is found, for example, for the heavy water reactor measurements of [11] analysed according to this formalism in [12].

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[9] - W. KOHLER, "Messungen zur Bestimmung des spektralen Verteilung der epithermischen und schnellen Neutronen im Schwimmbadreaktor", Communcation to the Euratom dosimetry group, March, 1962.

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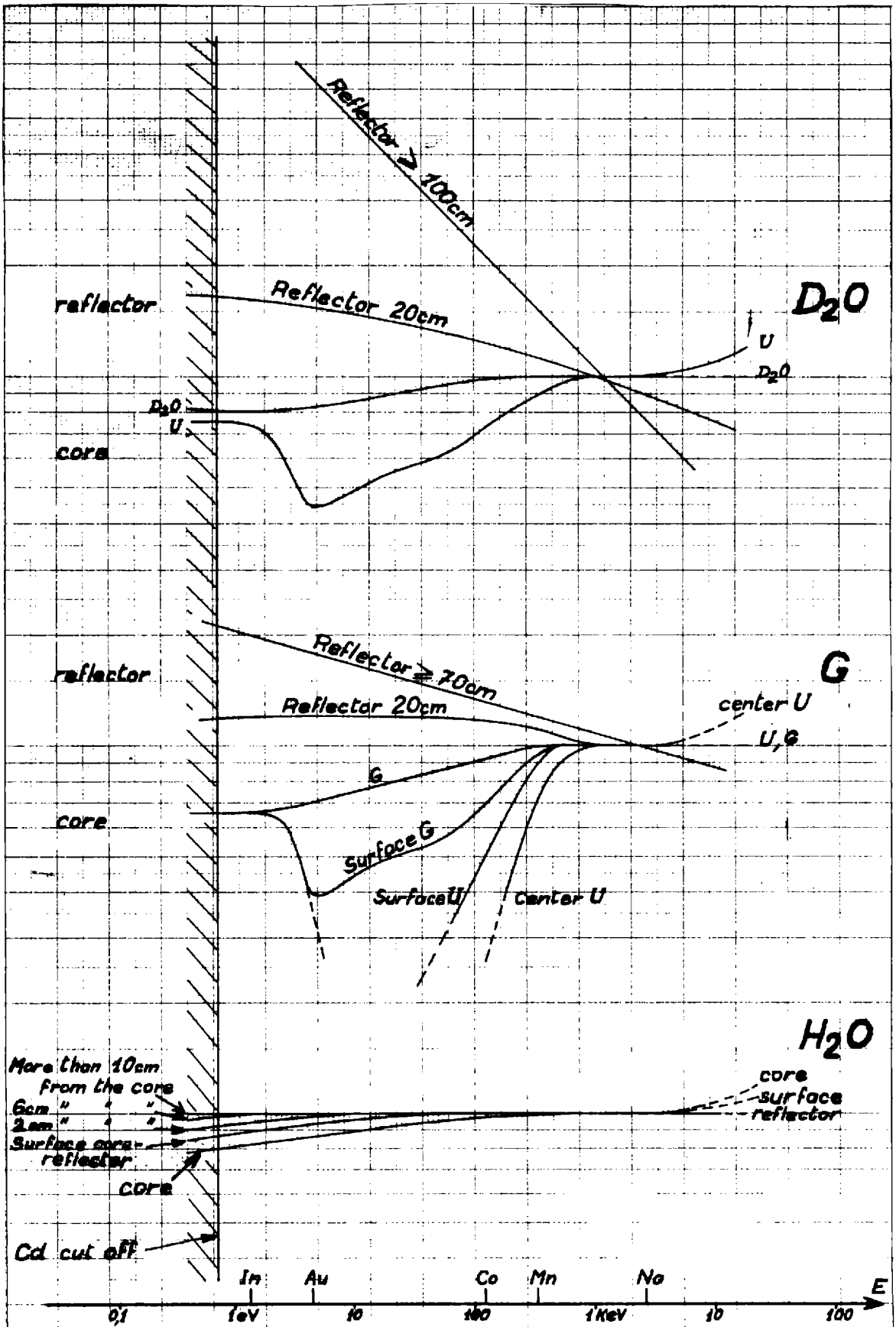

Fig. 1 : Intermediate neutrons flux spectra $E\varphi(E)$

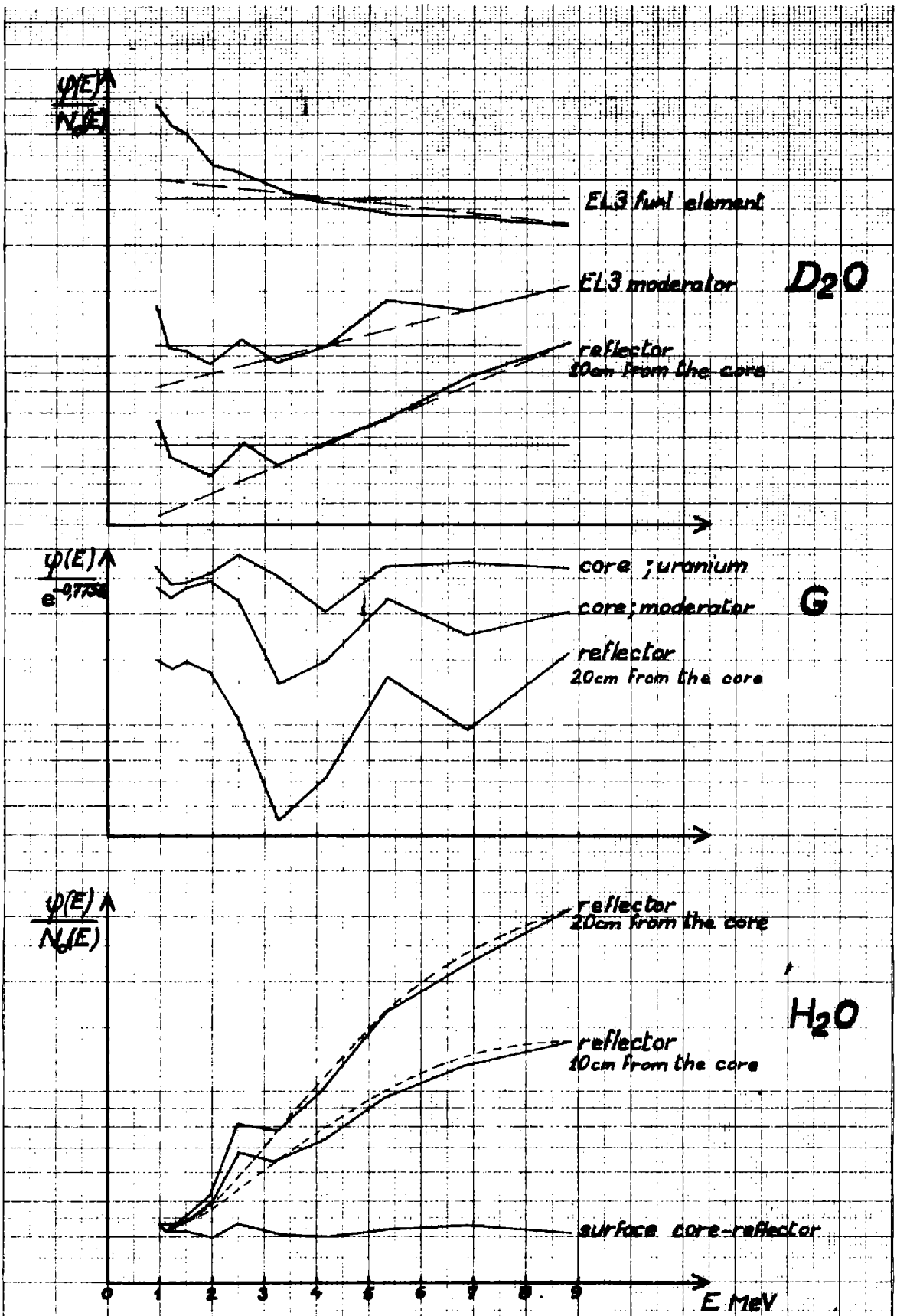

Fig.2 : Fast neutrons flux spectra

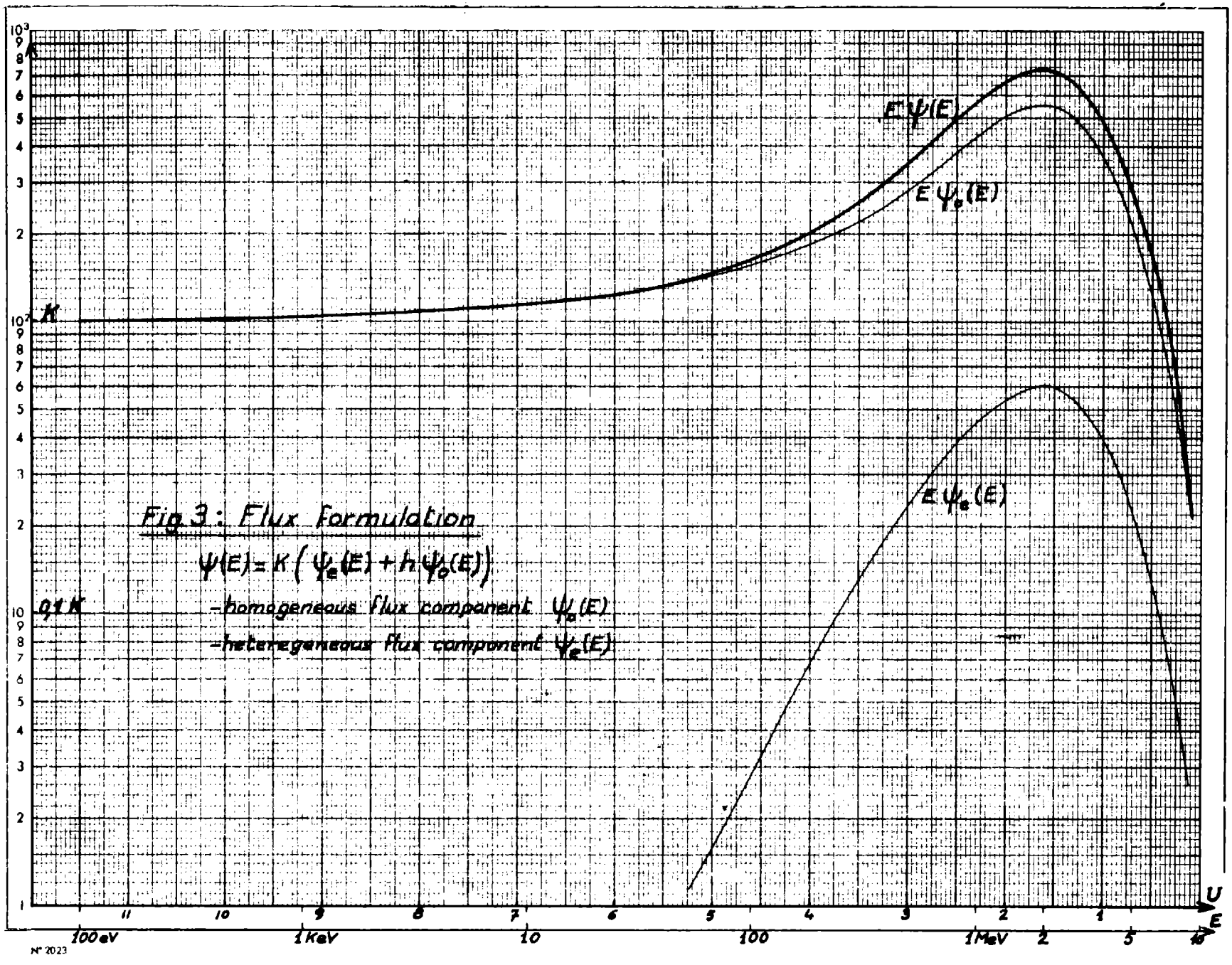

Fig 3 : Flux formulation

$$\psi(E) = K\left(\psi_e(E) + h\psi_o(E)\right)$$

- homogeneous flux component $\psi_o(E)$
- heterogeneous flux component $\psi_e(E)$

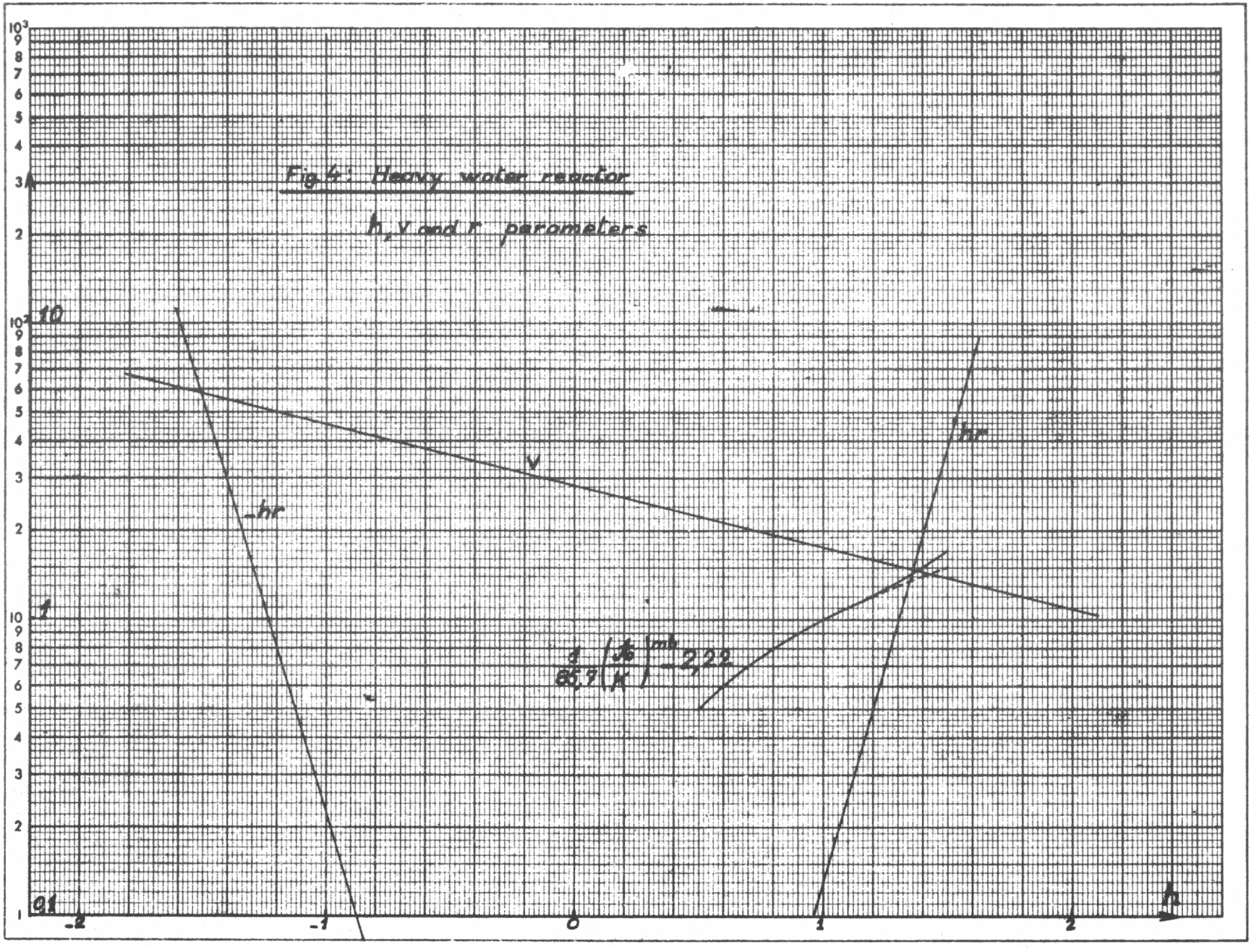
Fig.4: Heavy water reactor
h, V and r parameters
10
1
0,1
V
-hr
hr
$\frac{1}{66,7}\left(\frac{\Delta k}{k}\right)^{mk} = 2,22$
-2
-1
0
1
2
r

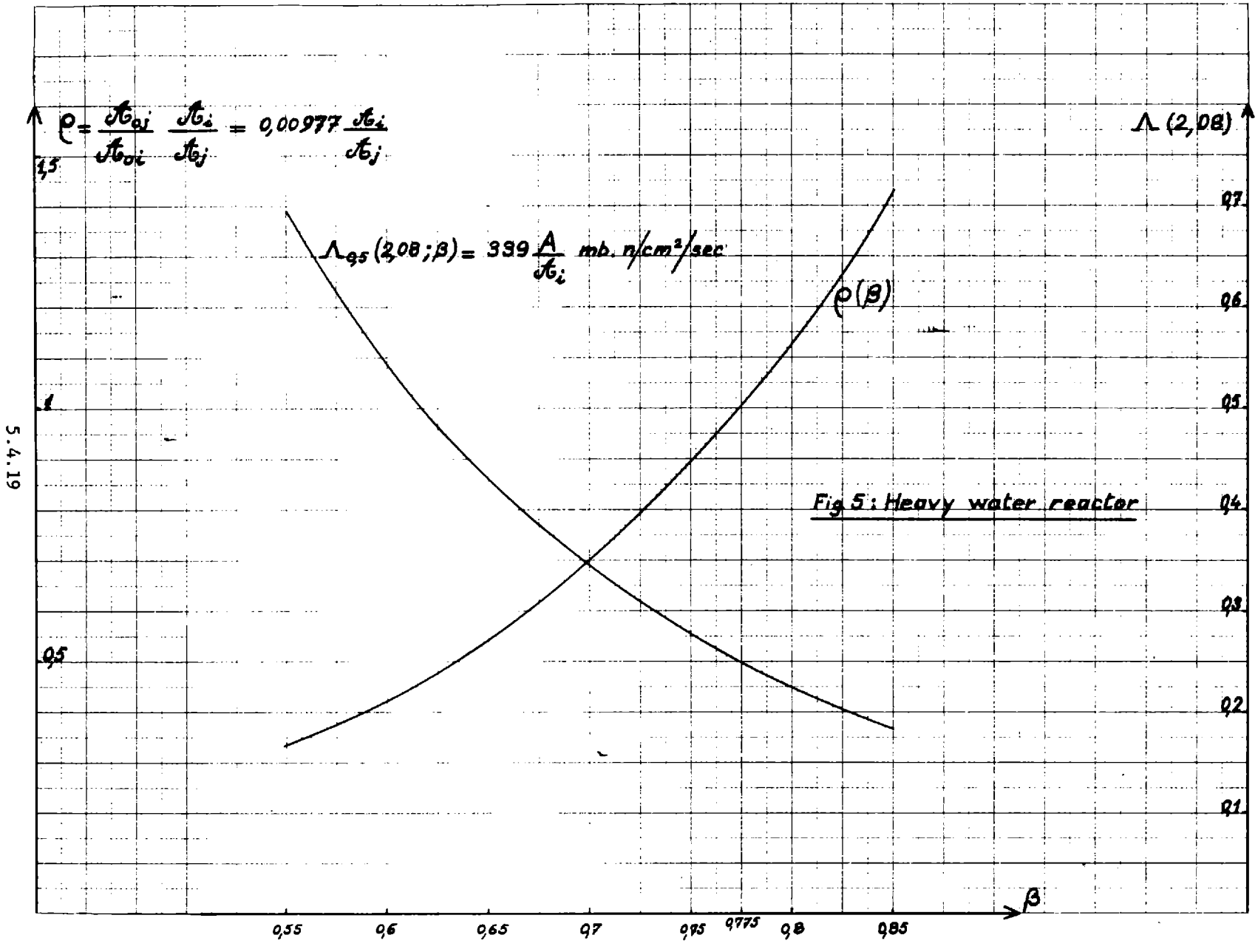

Fig 5 : Heavy water reactor

5.4.19

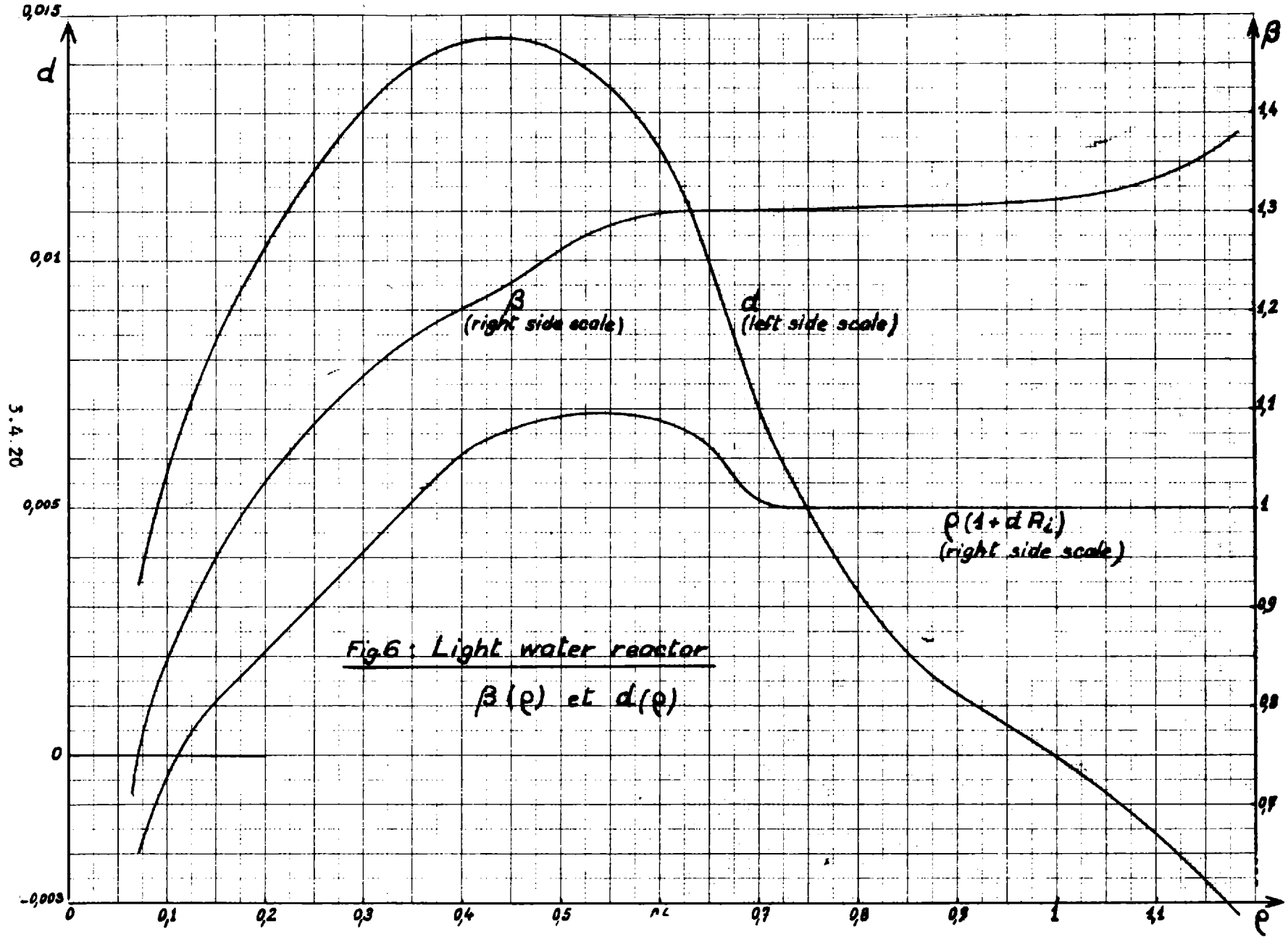

Fig 6 : Light water reactor

β(ρ) et d(ρ)

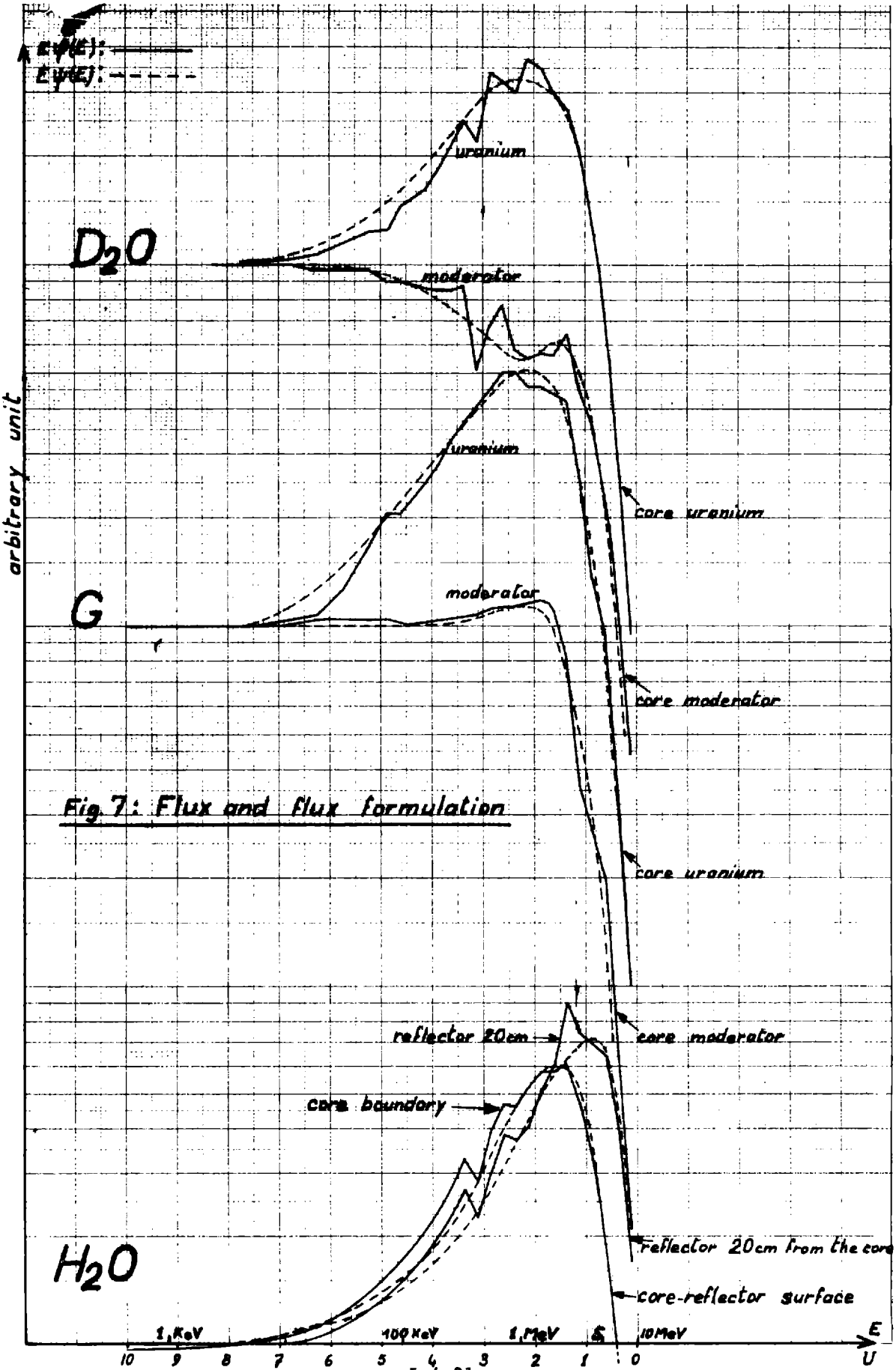

Fig. 7: Flux and flux formulation

Fast Neutron Dosimetry with Fission Monitors

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I. Introduction

In general, the minimum energy of fast neutrons which cause damage in structural materials has been arbitrarily set at 1 Mev by the experimenters in the field. There is evidence that reactions with slower neutrons can also create significant damage. Even (n,γ) recoil events from thermal neutrons are capable of producing lattice displacements. However the 1 Mev energy threshold is probably a reasonable value especially when considering radiation effects on engineering mechanical properties.

The problem then is one of determining neutron exposures and the neutron energy spectrum above 1 Mev. These determinations depend on the use of various threshold detectors which are activated by specific neutron reactions. The more common ones are $S^{32}(n,p)P^{32}$, $Fe^{54}(n,p)Mn^{54}$ and $Ni^{58}(n,p)Co^{58}$. Unfortunately the cross sections for these reactions are either not precisely known or the thresholds are not well defined. A secondary reaction involving thermal burnout of Co^{58} complicates the use of Ni. Also, if the threshold occurs well above 1 Mev the neutron energy spectrum must be well known to permit extrapolation back to the 1 Mev level.

There are several fissionable elements with sharp thresholds near 1 Mev. The most promising are U^{238}, Th^{232} and Np^{237}. These have appreciable cross sections for fast fission. In this presentation the experience of Brookhaven in the use of fission threshold monitors for fast neutron dosimetry will be described.

II. Materials

The pertinent physics parameters for fission monitors and the threshold monitors cited earlier are given in Table I.

One difficulty encountered in the use of fission monitors is the presence of small amounts of other fissionable isotopes in the sample. If these impurities have larger cross sections than the principal isotope particularly at lower neutron energies a large error is introduced. This problem is quite important in the use of U^{238}. This isotope must be almost completely depleted of U^{235} to be useful. Even with less than 100 ppm U^{235} present it is recommended that the monitor be covered with cadmium. The cadmium prevents the thermal fissioning

of any U^{235} present in the depleted U^{238}. While it may not seem necessary for Th^{232} since natural thorium is 100% Th^{232} it has been found that a cadmium cover is necessary for this monitor also.

Our work with Np^{237} is still in the early stages and we have not acquired sufficient experience with this isotope to justify a discussion of its application as a monitor. Aside from following the precautions necessary in working with a strong alpha emitter the analysis is about the same as that for U^{238} and Th^{232}.

III. Irradiation Procedure

We attempt to produce a monitor counting rate of 10^3 dps. This rate is sufficient for counting accuracy and yet not high enough to make post irradiation handling too difficult. Since the length of exposure will determine the number of fissions that take place in a given weight of sample different monitor weights are used for various nvt levels. Table II lists some representative values.

The thorium and uranium we have used are in the form of a thin foil while the neptunium comes as an oxide powder. The samples are weighed and then sealed in an evacuated quartz vial. The vial is formed from 4 mm O.D.

tubing. The vial is then either wrapped in cadmium foil or placed in cadmium tubing. The thickness of the cadmium has been 0.020 in. which was sufficient even in the MTR.

The final package is approximately the size of one of our test specimens and can be placed in an irradiation capsule in a specimen position. Generally we place a monitor at each end of a capsule.

After the irradiation the monitor package is removed from the capsule and the cadmium cover dissolved in nitric acid. The quartz vial is then opened with a small remote cut off wheel and the contents dumped into a flask. A solution of 1 part aqua regia and 2 parts water is then added to dissolve the monitor.

IV. Chemical Analysis

The fission products Ba^{140} or Cs^{137} formed during the fissioning process are analyzed to determine the fast flux. Ba^{140} is checked only for short term irradiations since its half life is 12.3 days and only the flux of the last 2 or 3 weeks of an irradiation could be determined with certainty. Its analysis is somewhat similar to, if not simpler than, that for Cs^{137} which is briefly outlined below.

The flux monitors are dissolved and appropriate aliquots are removed for the cesium-137 analysis. The analytical procedure used is that of Handley and Burros.[1]

The method consists of scavenging most of the radioactive contaminants and precipitating cesium as cesium tetraphenylboron. The precipitate is mounted and the activity of cesium-137 based on the 0.661 Mev energy gamma is determined using a multichannel analyzer. A calibrated cesium-137 standard is counted under the same geometry conditions as the unknown, thus an absolute disintegration rate can be calculated for the unknown. Since césium carrier is added initially a correction for any loss of cesium during the analysis can be made.

The weight of uranium is checked chemically by complexing it with dibenzoylmethane and extracting into amyl acetate. The amyl acetate phase is read on a spectrophotometer and the concentration determined from a standard curve. The thorium weight is checked by extracting it into a TTA-xylene mixture, stripping and developing a color with thorium. Again the concentration of thorium is de-

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(1) "Determination of Radioactive Cesium", Analytical Chemistry, March 1959, pg. 332.

terminated by reading its absorbance on a spectrophotometer and comparing it with a standard curve.

The cesium or barium activity is then converted into a fast neutron flux by means of the following expression.

$$\phi = \frac{A}{N \times \sigma_F \times (1-e^{-\lambda T}) \times e^{-\lambda t} \times Y}$$

where

ϕ is the fast flux in n/cm^2-sec

A is the total cesium or barium activity of the monitor

N is the atomic concentration of fissionable atoms which equals monitor wght. × Avagodro's No./atomic wght.

σ is the effective fission cross section at the threshold

λ is the decay constant of counted isotope in $days^{-1}$

T is the irradiation time in full power days

t is the decay time in days and

Y is the fission yield.

Of course, ϕ times T in seconds yields the integrated exposure in fast nvt.

As was stated previously it is necessary to cover both the depleted uranium and the thorium monitors with 0.020 in. of Cd. A Cd ratio of 4:1 was observed in the BGR and 9:1 in the MTR. The exact reason for this ratio is not understood.

V. Results

These flux monitors have been used in numerous test capsules irradiated in the Brookhaven Graphite Reactor and the MTR. A tabulation of representative results is given in Table III. The data from only one position in each reactor is given eliminating the reactor flux distribution as a variable.

As can be seen from the table the reproducibility is quite good. Just considering the combined data for the E34 facility an average fast flux of 7.3×10^{11} n/cm^2-sec with a standard deviation of $\pm .8 \times 10^{11}$ can be calculated. This is approximately ±10%. Based on data collected by the reactor operators and ourselves using other fast flux monitors the flux values extrapolated to 1.5 Mev for these positions in BGR and MTR agree quite well with the data obtained with the fission monitors. This then is an indication that the accuracy has also been good.

Table I

Threshold Monitor Parameters

Monitor	Threshold-Mev	Eff. Cross Section-Barns		Reaction Product Counted	Half Life	Fission Yield	References
Np^{237}	0.6	0.68	or	Ba^{140}	12.3d	6.2%	BNL 325
U^{238}	1.5	0.31		Cs^{137}	30 yr	6.2%	" "
Th^{232}	1.5	0.071					Nucleonics 18 201, 1960
Ni^{58}	2.9	0.1		Co^{58}	72d		ASTM-STP-341 1963
S^{32}	3.0	0.16		p^{32}	4.3d		Nuc. Science & Eng. 10 308, 1961
Fe^{54}	4.2	0.06		Mn^{54}	314d		

Table II

Typical Monitor Sizes*

Monitor	nvt	Weight-mg	Total Sol'n-ml
U^{238}	10^{18}	15	25
	10^{19}	10	25
	10^{20}	5	50
Th^{232}	10^{18}	50	50
	10^{20}	10	100
Np^{237}	10^{18}	10	

*Producing approximately 10^{3} dps per sample

Table III

Typical Results

Facility	Monitor	Irradiation Time	Isotope Counted	nv >1.5 Mev
E34-BGRR	U^{238}	3 weeks	Ba^{140}	7.6×10^{11}
E34-BGRR	U^{238}	6 weeks	Cs^{137}	8.4×10^{11}
E34-BGRR	U^{238}	5 months	Cs^{137}	7.3×10^{11}
E34-BGRR	U^{238}	6 weeks	Cs^{137}	8.0×10^{11}
E34-BGRR	U^{238}	6 weeks	Cs^{137}	6.7×10^{11}
E34-BGRR	U^{238}	6 weeks	Cs^{137}	6.6×10^{11}
E34-BGRR	U^{238}	6 weeks	Cs^{137}	6.4×10^{11}
E34-BGRR	U^{238}	6 months	Cs^{137}	7.0×10^{11}
E34-BGRR	U^{238}	9 weeks	Cs^{137}	8.3×10^{11}
E34-BGRR	Th^{232}	6 weeks	Cs^{137}	6.2×10^{11}
L43-MTR	U^{238}	3 weeks	Cs^{137}	1.3×10^{14}
L43-MTR	U^{238}	3 weeks	Cs^{137}	1.7×10^{14}
L43-MTR	Th^{232}	3 weeks	Cs^{137}	1.1×10^{14}

NEUTRON DOSIMETRY IN IRRADIATION EXPERIMENTS

S. B. Wright
A.E.R.E. Harwell

RRL 63/956

INTRODUCTION

The determination of the neutron dose received by specimens is an essential part of most irradiation experiments. In many experiments lack of space and the conditions under which the experiment is performed make this a difficult task, particularly in long term irradiations in high flux reactors. This paper describes some of the techniques used at Harwell to determine the neutron spectrum and dose in experiments in the high flux reactors DIDO and PLUTO (1,2).

DIDO and PLUTO are heavy water moderated and cooled reactors fueled with enriched uranium. They operate at a power of 15 MW giving a peak thermal flux of approximately 2×10^{14} n cm^{-2} sec^{-1}. The two reactors are very similar, the main difference being in the arrangement of the experimental holes in the reflector. A horizontal and a vertical cross section of PLUTO is shown in fig. 1 and 2. All the normal fuel elements used in these reactors are annular with a two inch diameter central experimental hole. In addition to the two high flux reactors there is a zero energy reactor (DAPHNE) which can be used to simulate either DIDO or PLUTO. The main difference between DAPHNE and the high flux reactors apart from the operating powers, is in the method of control. In the high flux reactors the main control is by means of signal arms pivoted close to the top of the top reflector (fig, 2). These coarse control arms are treated as fixed absorbers in DAPHNE and can only be moved while the reactor is shut down. The reactor is controlled by means of water height with fine control by two fine control rods situated at opposite corners of the core. In experiments designed to measure flux distributions in rig mock-ups the control arms are set so that the water can be pumped to its full height corresponding to a top reflector thickness equal to that in DIDO or PLUTO, and the reactor is taken critical using the fine control rods.

NEUTRON DOSIMETRY

In a high flux reactor it is not possible to use megawatt hours as a measure of the neutron dose except in experiments where only approximate doses are required because the flux distribution in the reactor will change considerably during the course of an experiment due to burn-up of the fuel and movement of the control absorbers. It is, therefore, usual to include some form of neutron dose monitor with most irradiation experiments. These are placed as close to the irradiation samples as the design of the irradiation rig will permit to minimise changes in the flux ratio between sample and monitor positions during the irradiation. As most control absorbers move in a vertical plane, producing the most serious changes in flux distribution in this plane, it is usual to place monitors in the same horizontal plane as the specimens if possible. The sample position is then calibrated relative to the monitor position in a separate irradiation often at zero power.

As the neutron energy spectrum in a reactor can change during the operating cycle due to burn-up of the fuel, it is important that the dose monitor should respond to the range of neutron energies which are of interest in the experiment being monitored. Ideally the monitor would have the same response to neutron energies as the samples being irradiated but such a perfect state of affairs is never found in practice. In addition the half life of available monitors limits the choice for most experiments. However, as the principal change in the neutron spectrum is an increase in the thermal flux at a constant power level as the fuel is burnt up it is usually sufficient to monitor thermal neutron irradiations with a detector which is approximately 1/V and radiation damage experiments with a detector with a threshold in the range 1 - 10 MeV. The neutron spectrum can then be established as a separate experiment. Experiments such as plutonium irradiations which are particularly sensitive to the thermal neutron spectrum or to the epithermal flux may require more careful planning of the monitoring techniques.

The detectors which are most commonly used are cobalt for thermal neutron monitoring, and the $Ni^{58}(np)Co^{58}$ or $Fe^{54}(np)\ Mn^{54}$ threshold reactions for high energy neutron monitoring. Cobalt is irradiated in the form of wire 0.010" diameter. This is a convenient form of detector to handle and successive batches of material only show very small variations in diameter. Monitors are cut using a hand guillotine which produces foils of approximately 1 mgm whose weights are within $\pm \frac{1}{2}\%$ of the mean using the normal extruded wire. In this way, the need for tedious weighing of small monitors is minimised, and standard size monitor foils are produced easily and quickly. A similar technique is used for preparing nickel monitors from nickel wire.

The use of 0.010" cobalt wire for thermal dose monitoring suffers from the disadvantage that the neutron self shielding is appreciable particularly in the resonance region. However, the fact that the foils are robust, easily handled and easier to reproduce than dilute alloys of cobalt in aluminium saves considerable time in an extensive monitoring programme. The thermal neutron self shielding factor for the foils used is approximately 0.92 and the correction for epithermal activation on the core of DIDO where r is 0.1 is 7%. Both of these corrections are easily determined to the accuracy required for thermal dose monitoring.

The thermal neutron burn up of cobalt 58(1) limits the usefulness of the nickel monitors in high thermal fluxes due to the shortening of the effective half life. In a thermal flux of 10^{14} n cm^{-2} sec^{-1} the effective half life of the $Ni^{58}(np)Co^{58}$ detector is reduced to approximately 30 days. For long term irradiation experiments in high thermal fluxes the Fe^{54} (np) Mn^{54} reaction is proving to be a useful dose monitor (2)(3), although unless a separated isotope can be used the analysis involves a rather tedious chemical separation to remove the Fe^{59} activity produced with thermal neutrons. Wherever possible therefore it is preferable to use nickel.

For some experiments requiring the measurement of fast neutron dose monitor it is difficult to find a suitable threshold reaction. In these cases monitoring is often done using a thermal neutron detector.

When this technique is used, careful calibration experiments are required to relate the thermal neutron dose which is measured to the fast neutron dose received by the experiment. This method of fast neutron dose monitoring has been used successfully for a number of years for experiments in the hollow fuel elements in the DIDO type of reactor. Careful flux measurements in the experimental void have shown that except for the region close to the top and bottom of the fuel the ratio of the fast neutron flux to the thermal neutron flux is constant and independant of the reactor control arm position. This ratio does depend however on the total U^{235} content of the fuel element and hence changes during the reactor operating cycles due to burn up of the fuel. Experiments have shown that under normal conditions this ratio depends only on the burn up of the fuel element in which the measurements are being made, and the effect of burn up of neighbouring fuel elements is too small to be significant. This is due to the fact that in this facility the fast neutron flux from neighbouring fuel elements is only a small fraction of the fast flux from the fuel element itself, and hence the fast flux inside the fuel element is determined by the thermal flux in the fuel of that element. Fig. 3 shows a plot of the ratio of fast flux to thermal flux against burn-up of the fuel element. The line is the result of flux scans made using nickel at low power where the correction for burn up of Cobalt 58 is negligible, and the points are measurements made in rigs at full power for irradiations of one to three months duration. These irradiations were not all performed in the same fuel element position in the reactor. In the full power measurements the correction for cobalt 58 burn up is about a factor of 2. Using these results it is possible to do fast neutron dosimetry for long term experiments using cobalt with reasonable accuracy in this type of irradiation position provided the experimental arrangement is the same in all cases. In a D_2O cooled experiment the same technique has been used, but the

values of the ratio were found to be about 30% lower than the values given in fig. 3. A similar but smaller reduction has been found in an experiment involving large blocks of graphite. It is important, therefore, to keep the configuration of all moderating materials constant when this technique is being used. Attempts to use a similar technique in experimental positions outside the core of the reactor have not been successful, because the fast to thermal flux ratio no longer depends on only one variable, and calculation of the fast neutron dose becomes very difficult.

Presentation of Results

In almost all experimental irradiations the neutron energy spectrum is an important parameter. For thermal neutron experiments this does not present too much difficulty as it is usually sufficient to measure the neutron temperature and the Westcott r value in the irradiation position, and this can be done in a low power irradiation, preferably in a zero energy reactor (see below). The situation is not so simple for fast neutron spectra however because at present there is no adequate means of measuring them.

Calculated spectra can now be obtained for idealised situations which can be used for analysis of most irradiation experiments. These calculated spectra are improving all the time as more realistic calculations are performed and it is to be hoped that fast neutron spectra adequate for the analysis of all experiments will be obtained in the near future. It is therefore essential for fast neutron measurements and desirable for thermal neutron measurements that the measured neutron doses are quoted in such a way that the data actually measured is readily accessible. If this is done experiments performed at different laboratories can be compared, and also experiments can easily be re-analysed as improved data becomes available. Unless this is done much of the value of irradiation experiments is wasted.

Ideally, the detectors used to measure the neutron dose and the saturation reaction rate observed should be quoted. If it is desired to quote results in terms of a flux or a neutron dose then the cross section or the neutron spectrum used to determine these should be given. Quoting results simply as a conventional flux without giving the cross sections or spectra used to obtain the flux is neither adequate or scientific. The practice of quoting fast neutron fluxes as flux greater than 1 MeV is simply avoiding the question because in most cases a fission spectrum is assumed to obtain this flux and the reactor spectrum above 1 MeV is not a fission spectrum. Also radiation damage is not confined to the neutron energy range above 1 MeV.

USE OF A ZERO ENERGY REACTOR

In the analysis of many irradiation experiments it is necessary to measure the neutron distribution through the rig or to irradiate several types of detector to determine the neutron spectrum. Because of the limitations imposed by some detector materials and also the handling problems involved with high flux irradiations it is easier to perform this type of experiment at low power. The high flux reactor can be run at low power, but this is a waste of reactor time and is in general not as convenient as using a zero energy reactor for this type of experiment.

Experiments for DIDO and PLUTO are simulated in DAPHNE by building a simplified version of the irradiation rig. In such a rig the positions of the experimental samples and dose monitors are kept the same as in the full power rig, and as far as possible the neutron properties of the mock up are made identical with the rig. Special irradiation samples are made so that the flux can be measured close to or inside the sample. In this way, the neutron flux and spectrum at the sample position can be related to the flux at the monitor position.

The low flux level of the mock up rig compared with the full power rig means that different methods of flux measurement have to be employed. In DAPHNE manganese-gold alloy foils are used to determine the thermal and epithermal fluxes. These foils are made so that immediately after the irradiation the activity of the foil is due almost entirely to the manganese. After a few days decay, however, the gold activity becomes the major constituent and thus by counting the foils twice at suitable times after the irradiation the thermal and epithermal fluxes can be calculated. Similarly manganese - leutecium ceramic foils are used to determine the thermal neutron temperature using the leutecium resonance at 0.15eV. Fast neutron fluxes are measured either using the Rhodium inelastic scattering reaction or the phosphorus (np) reaction. The rhodium is irradiated in the form of metal foil, and the phosphorous as a suspension in polythene sheet (4).

The flux measurement techniques outlined above enable the neutron flux distribution through the rig to be measured. They also enable some information about the thermal and epithermal spectrum to be obtained without the use of cadmium and hence avoid the distortions in the thermal flux distribution caused by the use of cadmium boxes. Always measuring the fluxes relative to the monitor position or to some other point in the system which can be monitored at full power avoids the errors involved in scaling lower power measurements ($\sim$ 10 watts) in the zero energy reactor to full power (15MW) in the high flux reactors.

The scaling up of low power results to obtain absolute flux levels has always been found to be an unsatisfactory technique for several reasons. Firstly although it is possible to simulate a single rig reasonably well it is difficult to simulate the whole reactor loading sufficiently closely to ensure the flux distribution is correct. The zero energy reactor is usually operated with clean unburnt fuel elements to avoid problems of handling highly active fuel elements. In this type of system it is difficult to simulate all conditions of burn up and poison which can be found in a research reactor. There is some advantage in having a set of fuel elements of lower fuel content than normal to represent an average state of burn up in the core but even in this case the simulation is only approximately correct as the burn up in an operating reactor is not uniform. As a result of these differences in core loading the control rod position in the lower power run is usually very different from the full power case unless the core is heavily poisoned. In either case the flux distribution is distorted making extrapolation to high power difficult. A second difficulty is the measurement of the absolute power level. In a high power reactor this can be done accurately by measuring the thermal balance of the reactor. At zero power however the power has to be calculated from the flux distribution in the fuel. Power can rarely be measured better than about 10% by this method and this introduces a further error into any extrapolation.

A third difficulty in using a zero energy reactor is the simulation of the full experimental loading of the reactor. It has been found by experiment in DIDO that the loading of a rig controlling 0.5% reactivity changes the thermal flux in neighbouring experimental holes by 10 - 15%. Therefore, if an experiment is to be used to determine the flux at full power absolutely it is essential that the exact reactor rig loading is used. A further difficulty is that the loading of neighbouring experimental holes might change during the course of a experimental run. This would mean a further low power experiment if this were being relied on to predict the fluxes at full power. It has been found to be more satisfactory to relate the flux distribution measured in DAPHNE to a monitor position, and to use these distributions together with the dose measured at full power to determine the dose received by the sample. Extrapolation of low power measurements to full power is not expected to give an accuracy better than 30%.

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2. C. H. Hogg and L. D. Weber. "Fast Neutron Dosimetry at the MTR-ETR Site" ASTM Symposium on "Radiation Effects on Metals and Neutron Dosimetry" October, 1962.

3. R. L. Ritzman et. al. "Fast Neutron Dosimetry for Long Term Irradiations". A.S.T.M. Symp. or Radiation Effects on Metals and Neutron Dosimetry". October, 1962.

4. R. J. Trebilcock. "Fast Flux Measurements in Zero-Energy Reactors" IAEA Symposium on "Neutron Dosimetry" December, 1962.

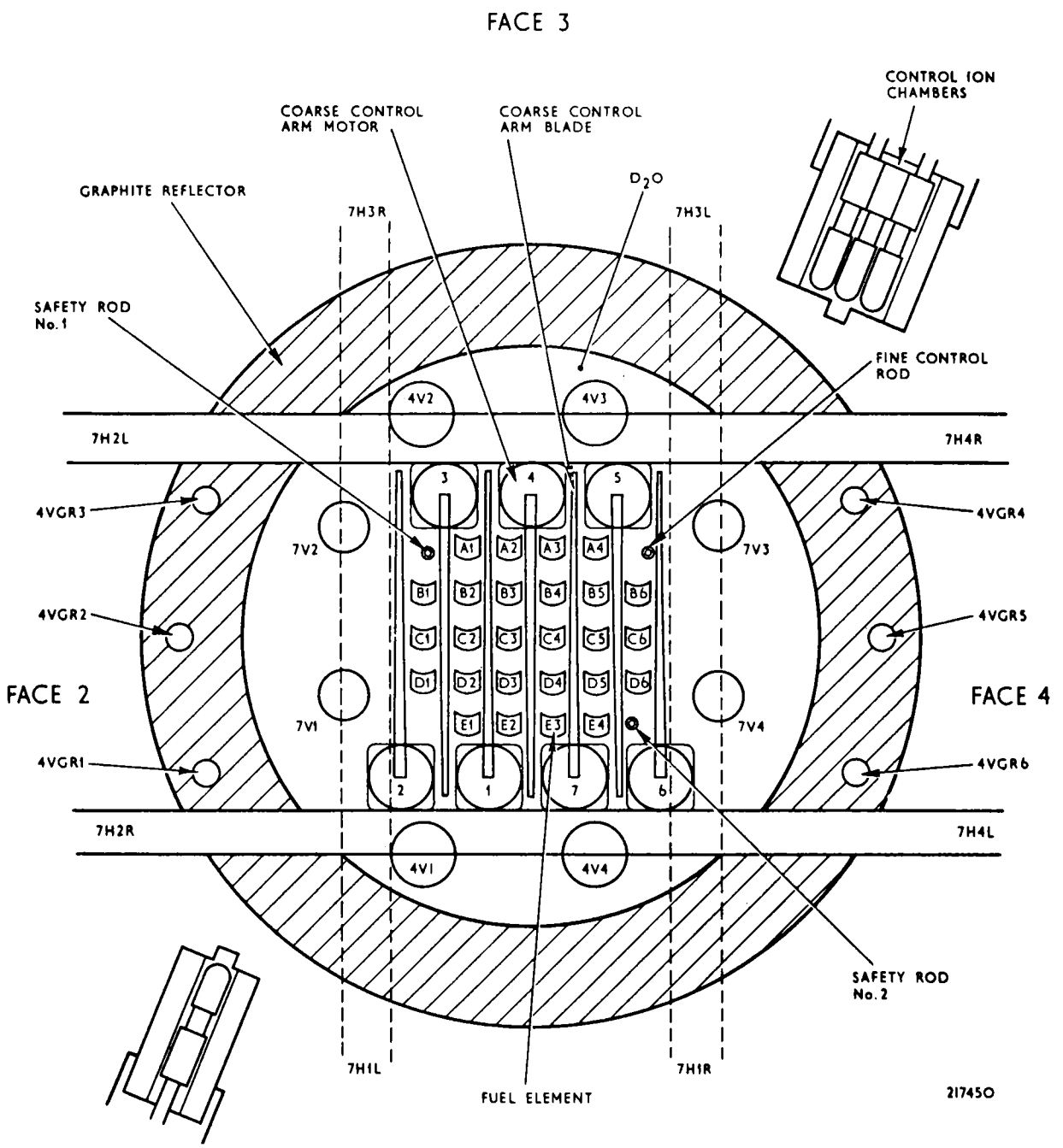
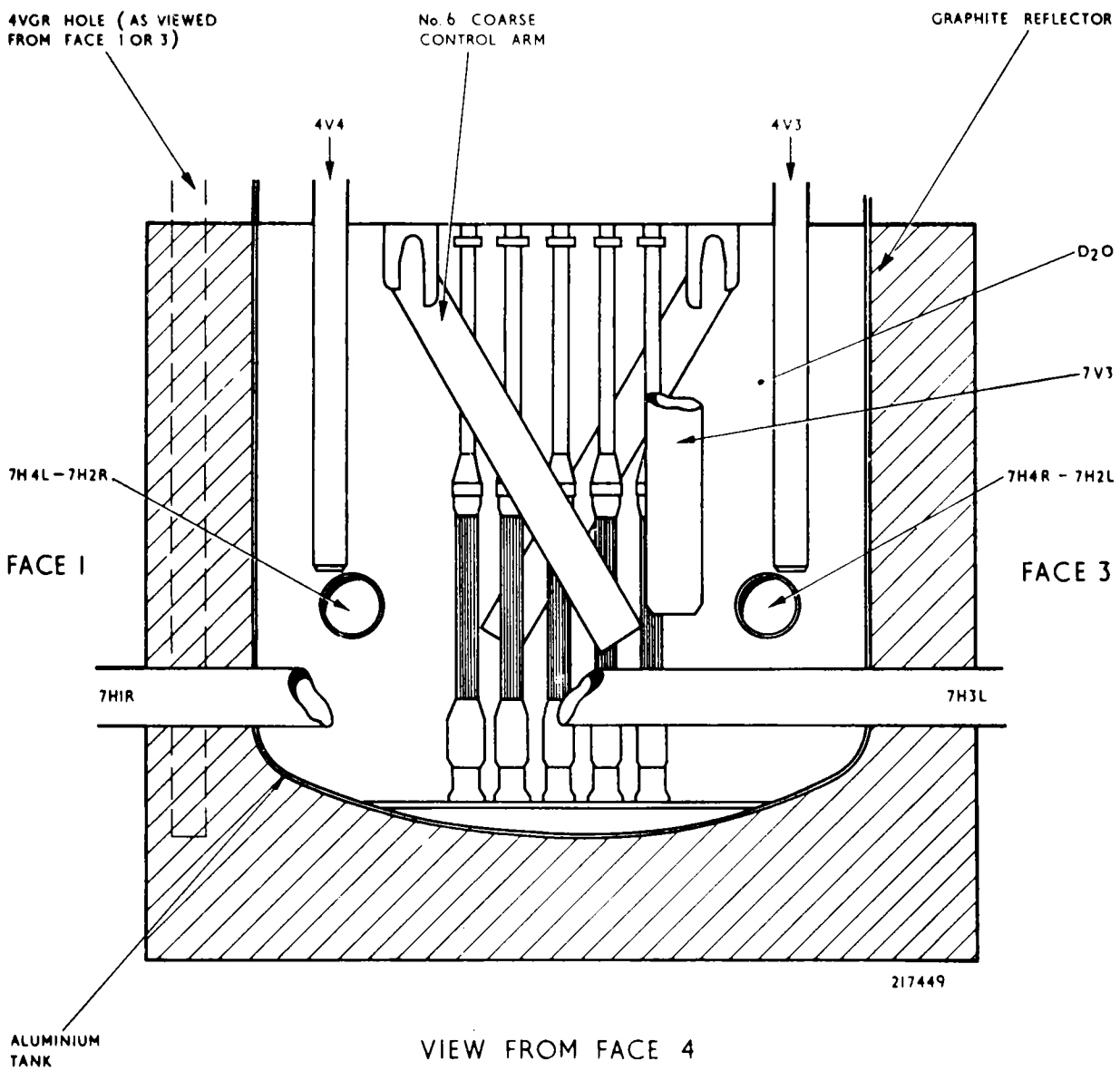


FIG. 1 HORIZONTAL CROSS SECTION OF PLUTO



VIEW FROM FACE 4

FIG. 2 VERTICAL CROSS SECTION OF PLUTO

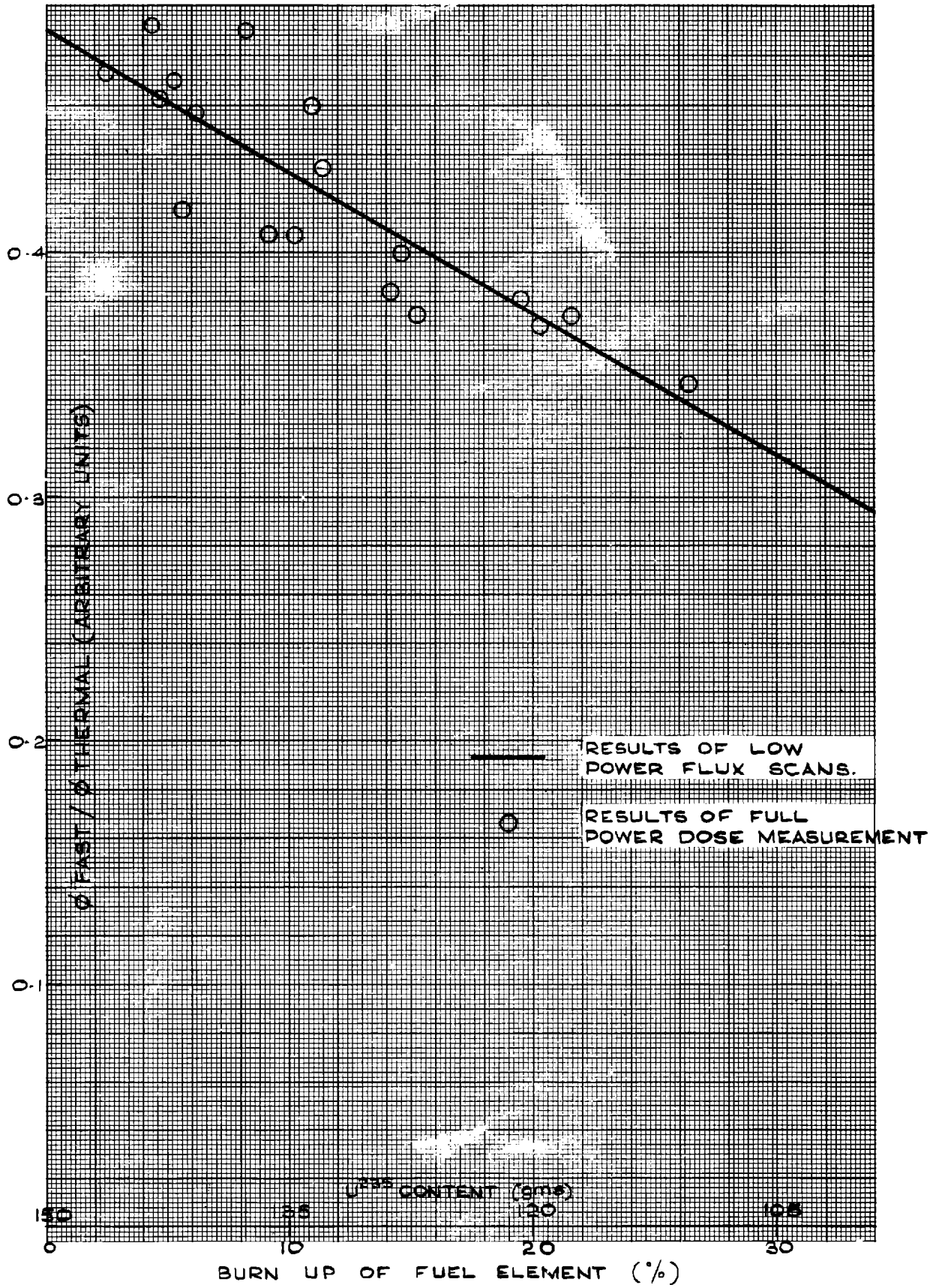


FIG. 3 CHANGE OF FAST TO THERMAL FLUX RATIO WITH BURN UP IN A HOLLOW FUEL ELEMENT IN DIDO.

PHYSICS METHODS FOR EXPERIMENT IRRADIATIONS AT GENERAL ELECTRIC TEST REACTOR

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General Electric Company

The primary objective in any irradiation is to obtain the reaction rate desired. This reaction rate may be a fission rate, in the case of fuel irradiation, an activation rate for isotope production, an absorption rate for control materials or a collision rate for irradiation damage studies. The reaction rate is dependent not only upon the neutron flux of the neutron source, but also the neutron cross section of the neutron absorber for this reaction. If sufficient freedom is permitted in the design of the experiment, the reaction rate may be optimized by the judicious choice of materials, geometry and position in the neutron field.

For purposes of designing an experiment for the normal test reactor, the problem of determining the effective neutron flux delivered can be separated into two parts. The first problem is the determination of the neutron flux existing in the environment without this particular experiment in place. In order to obtain the best description of this flux, one must consider this case on the basis of the proper fuel distribution and all other significant experiments included so as to give the correct power distribution in the reactor and the correct interaction effects of adjacent experiments or materials. The second problem is the determination of the perturbation on the neutron flux created by this particular experiment and the resulting neutron flux delivered or reactions rate that occurs.

This separation of the localized effect of a given experiment on the established neutron field from that of determining the neutron field is especially applicable to the light-water moderated reactors because of the extremely short diffusion length of ordinary hydrogen. Perturbations created by an experiment in a beryllium or graphite medium are more extensive in space and require consideration of a larger localized area.

The determination of the gross neutron field in the General Electric Test Reactor (GETR) has been done almost entirely by calculational methods with frequent confirmation by cobalt wire irradiations. Agreement of these measurements with the calculated values has been within a few percent. The calculations are made routinely by the Reactor Operations Physics group at least once each cycle and include all significant experiments. Three-group diffusion theory utilizing neutron cross sections obtained by averaging over the neutron spectra is used in a two-dimensional geometry. The effects of variations in the experiment compositions in the vertical directions are included on the basis of effective absorption and measured vertical flux profiles. The effects of fuel element burnup and cycling are also included. The thermal energy neutron cross sections are affected by the temperature of the moderator and by the neutron absorptions. These effects are included in determining the neutron field by the use of thermal spectra calculations and the spatial variation of the flux as determined by transport theory.

The determination of the localized perturbation created by a particular experiment on the neutron flux environment and the resulting reaction rate is accomplished by the use of transport theory cell calculations. The relative spatial flux from the transport calculation is normalized to the results of the two-dimensional diffusion theory calculations to obtain an absolute value of the effective neutron flux delivered to the experiment. When the proper reaction cross section has been averaged over this effective spatial flux and effective spectrum, reaction rates are predicted very near to those measured.

A few examples of the agreement between predicted and observed values are given below.

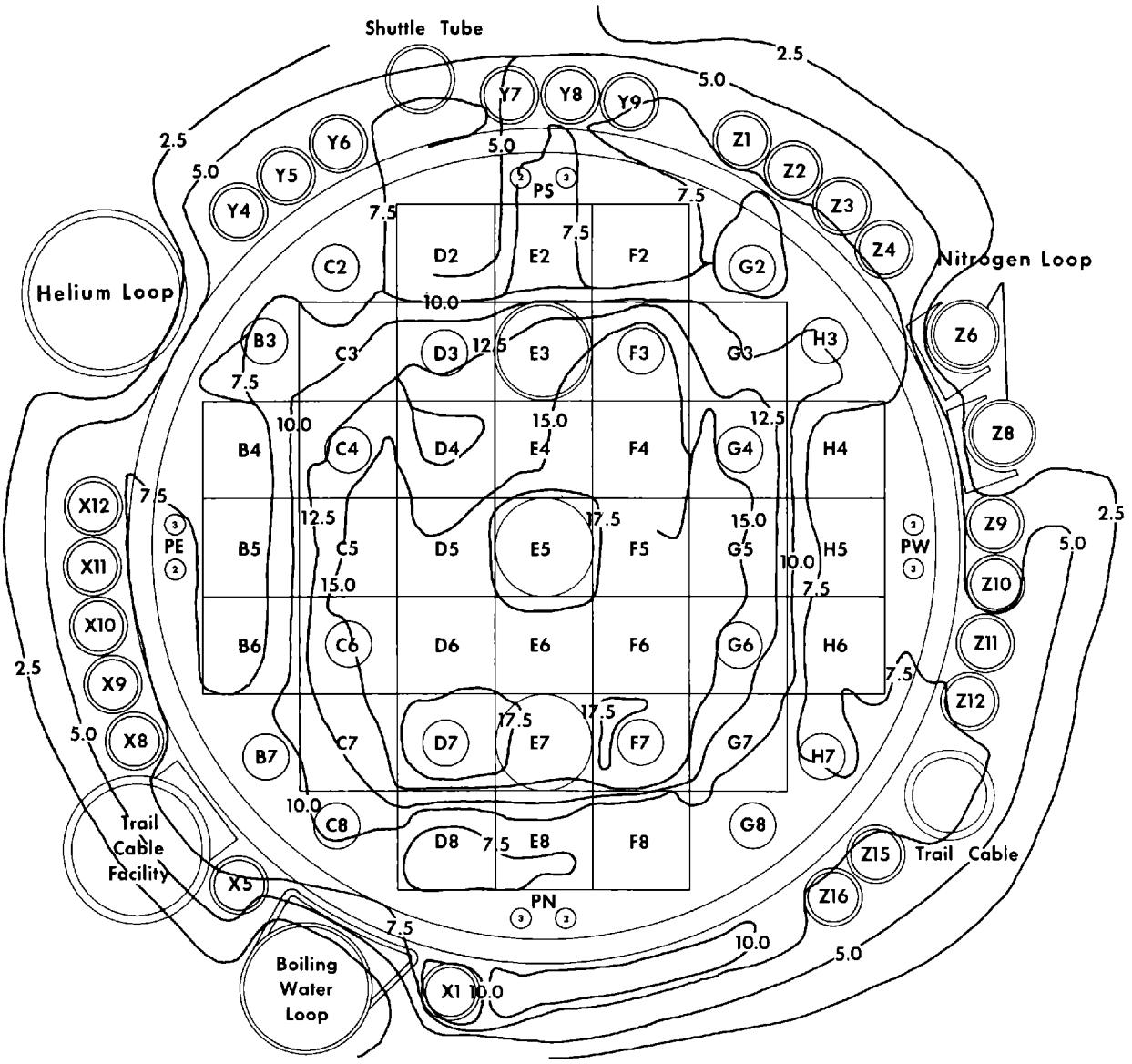
Experiment	Cycle	Enrichment (% U-235)	Predicted	Measured
Pressurized Water Loop ($4\ UO_2$ Rods)	33	1.5	219 KW	200 ± 10% KW
	34	2.2	269 "	260 ± 10% "
	35	3.0	301 "	330 ± 10% "
	37	1.5; 2.2; 3.0; 3.8	285 "	292 ± 10% "
UO_2 Capsules	37	20	550, 000 Btu/ft^{2}hr	527, 000
	37	20	514, 000	520, 000
	37	20	537, 000	537, 000
PuO_2 - UO_2 Capsule	41		26.0 KW/ft	25.4 KW/ft
PuC - UC Capsule	39		10.2 KW/ft	10.0 KW/ft

The power production of capsules is, in practially all cases, exclusively the result of the fissioning of fuel contained in the capsule and the gamma heating that results from the interaction of gamma rays with the capsule hardware. In some cases heaters may be required to obtain the desired temperature conditions. By knowing the material composition of the capsule and the neutron and gamma fluxes, the total power production in the capsule can be calculated from the individual contribution from each of the two sources of power.

A typical neutron flux distribution for thermal neutrons (0 to 0.17 ev) as measured in the GETR is shown in Figure 1. The maximum flux of $\sim 2 \times 10^{14}$ seen in Figure 1 applies to the load conditions experienced for a typical reactor cycle. The load condition is averaged over the core height throughout an entire cycle. In the actual use considered, the reactor contained a typical experimental loading during the cycle. Considerable flux variations can be experienced as a result of changes in the experimental loading.

Similar "iso-flux" curves can be drawn for the intermediate neutron flux (0.17 ev to 0.18 Mev) and the fast neutron flux (0.18 Mev to 10 Mev) distributions. The maximum flux in the intermediate range is 2.8×10^{14} and in the fast range is 3.5×10^{4}.

A gamma heating contour can also be obtained; the form is very similar to the neutron flux curves.



0 to .17 EV

Fluxes × 10^{13}

Figure 1. THERMAL FLUX CONTOURS

Detailed neutron and gamma heating flux profiles, however, must be obtained for each experiment at the time it is inserted. Because of changes in fuel loading and relocation of experiments and irradiation capsules, calculation of flux and gamma heating distribution may be required before each cycle.

The extent of the axial variation in essentially thermal neutron flux is shown in Figure 2. Traverses of this type are made periodically in a facility just outside the pressure vessel that contains the fuel core. It is readily apparent that both the magnitude and the location of the peak neutron flux in the GETR vary during the course of an irradiation cycle, just as it does in other test reactors. As the control rods are withdrawn, the magnitude of the peak neutron flux decreases and its location moves upward. Calculation of this variation by the Reactor Operations Physics group is in good agreement with the measured values.

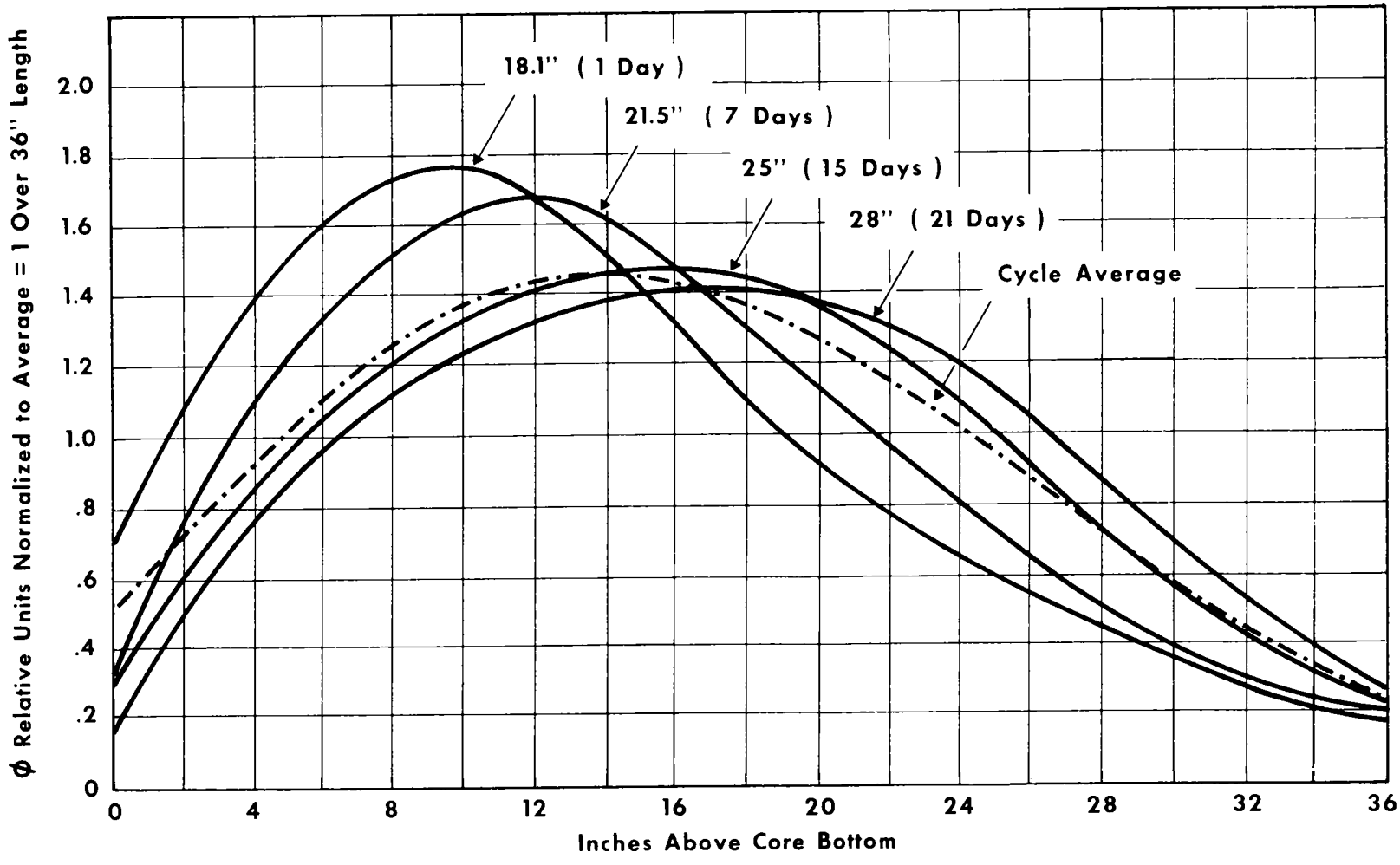


Figure 2. AXIAL FLUX SHAPE AS A FUNCTION OF ROD BANK IN INCHES

THE HIGH FLUX, HIGH TEMPERATURE ION CHAMBER TEST RIGS

H. Bowes
A.E.R.E. Harwell

In 1959 compact ion-chambers for high performance and lengthy service in power reactors for measurements relating to spatial flux distribution problems were still very much in the development stage and the exigencies of the Civil Reactor Programme at that time demanded accelerated life tests on a number of models. A programme of eight test rigs was therefore mounted and lasted some two and a half years until March 1962.

Practical Requirement

This was to support the test chambers in the centre of a 7 inch dia horizontal experimental hole in PLUTO some 40 cm from the diametral centre line of the core, and to record the surface temperatures of chamber and surrounding wall together with flux levels in the region described. Technical requirements are outlined in the appendix, and will be more clearly understood when a general description of the rig has been assimilated.

Description

From in-pile to out-pile, the first item considered is the ion-chamber itself. This consisted of an outer cylinder of stainless steel about $1\frac{1}{2}$ inches in diameter, larger (1 7/8" dia.) in later marks. All chambers were about 10 inches long, and weighed approximately 800 gms, the overwhelming majority of which was stainless steel.

The instrument was threaded and screwed to a support tube of matching outside diameter, and 1/8 inch thick, also in stainless steel. Thermocouple leads passed through this, as did the actuating cables which operated the spider device upon which were mounted the thermocouple hot junctions that monitored the wall temperatures. Flux scan thimble tubes were carried outside of the support tube and strapped to it, a convergent cone in-pile termination ensuring reproduceable positional accuracy of the dosiometer element employed for flux assessment.

The support-tube was in turn carried on a modified standard PLUTO shield plug the in-pile tip of which was circulated with cooling water from the main shield-cooling circuit. Cadmium was added in this region for additional shielding safety. All conduit and cooling pipes were spiralled through the plug body, which was constructed of grey cast iron, nickel-phosphorus plated on the outside against corrosion. As is usual with such plugs, a large step was incorporated at the out-pile end to obviate direct shine, and in this step were machined the annular grooves accommodating the neoprene 'O' ring seals that sealed the helium atmosphere gas in the experimental hole.

Out-pile of the modified shield plug was fitted an extension tube which removed the terminal plate to an accessible position at the pile face from a relatively unapproachable location deep in the void. The out-pile flange accommodated the actuating gear for the thermocouple spider, the terminals for all chamber and monitoring connections and the seal plugs maintained on the open ends of the flux scan tubes as a precaution against leakage of atmosphere in the event of a thimble tube fracture. The assembly was completed by an instrument cabinet and recorder on the experimental floor.

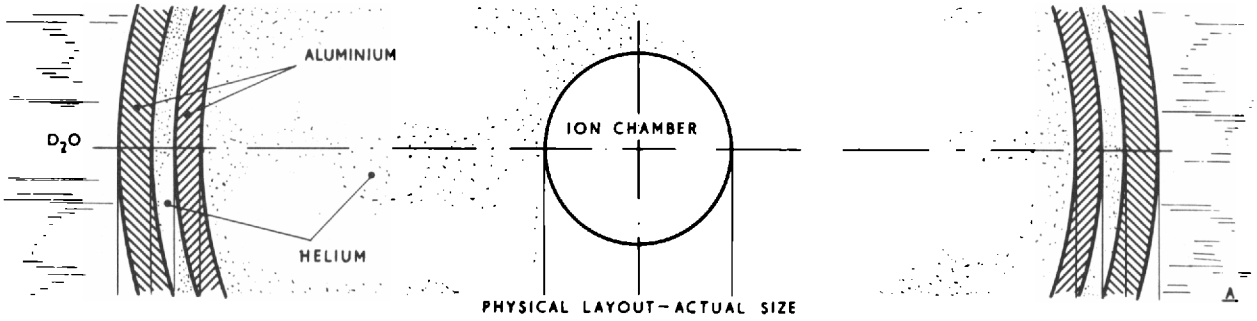
Appendix, Technical Problems

The purpose of the high flux reactor PLUTO in these experiments was to obtain equal dose rates to those anticipated in power reactors, with lower fluxes, over shorter times, the product of flux and time being roughly equal in each case. This being so, there was clearly a requirement to place as little enshrouding material round the chambers as possible, and such material would be necessarily included in a heating furnace. For this reason it was decided from the outset to rely only on nuclear heating to raise the ion-chamber to the required temperature, and the technical problems were almost wholly associated with this.

Firstly, the optimum position of the chamber relative to the centre line was worked out as being 40 cm.

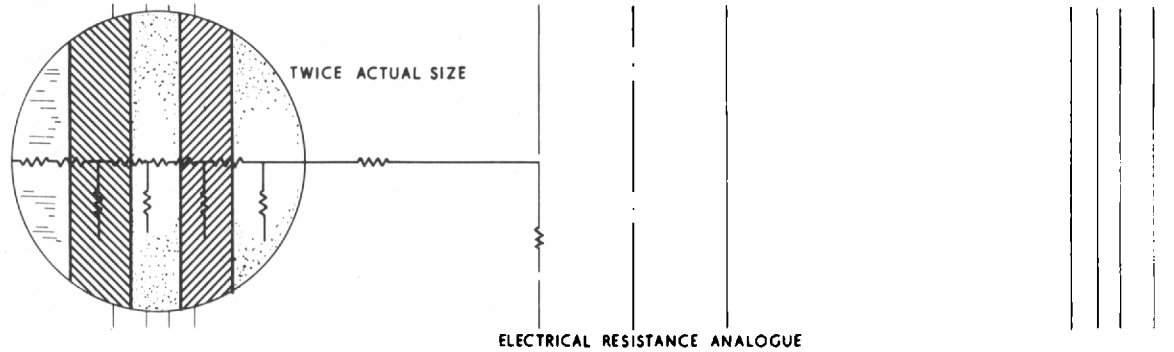
Next the predicted temperature of 386°C was found using extrapolated nuclear heating data. Then, as heating data was more accurately determined from derived experimental results, more realistic predictions for future chambers could be made.

Finally, heat shield design for reflecting nuclearly generated heat and increasing the surface temperature of the chamber with various nuclear radiation resisting reflecting surfaces was carried out successfully, and the last calculation of all established the optimum gas conductivity, given a series of gas mixtures, for a particular chamber surface-temperature.



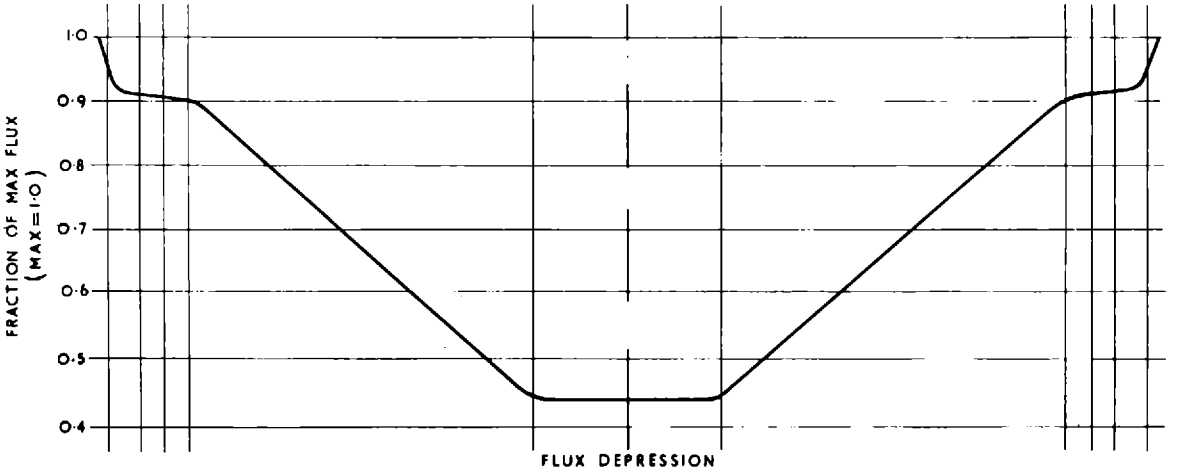
PHYSICAL LAYOUT – ACTUAL SIZE

A



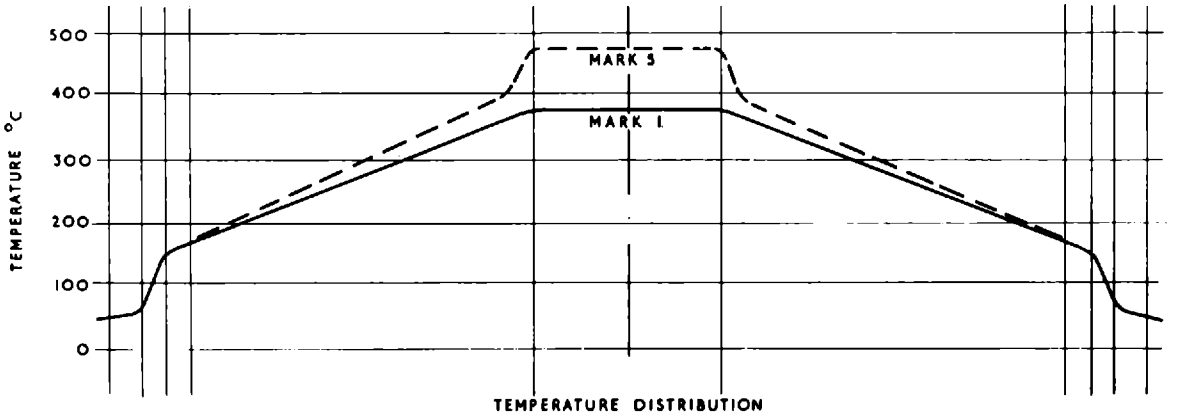
ELECTRICAL RESISTANCE ANALOGUE

B



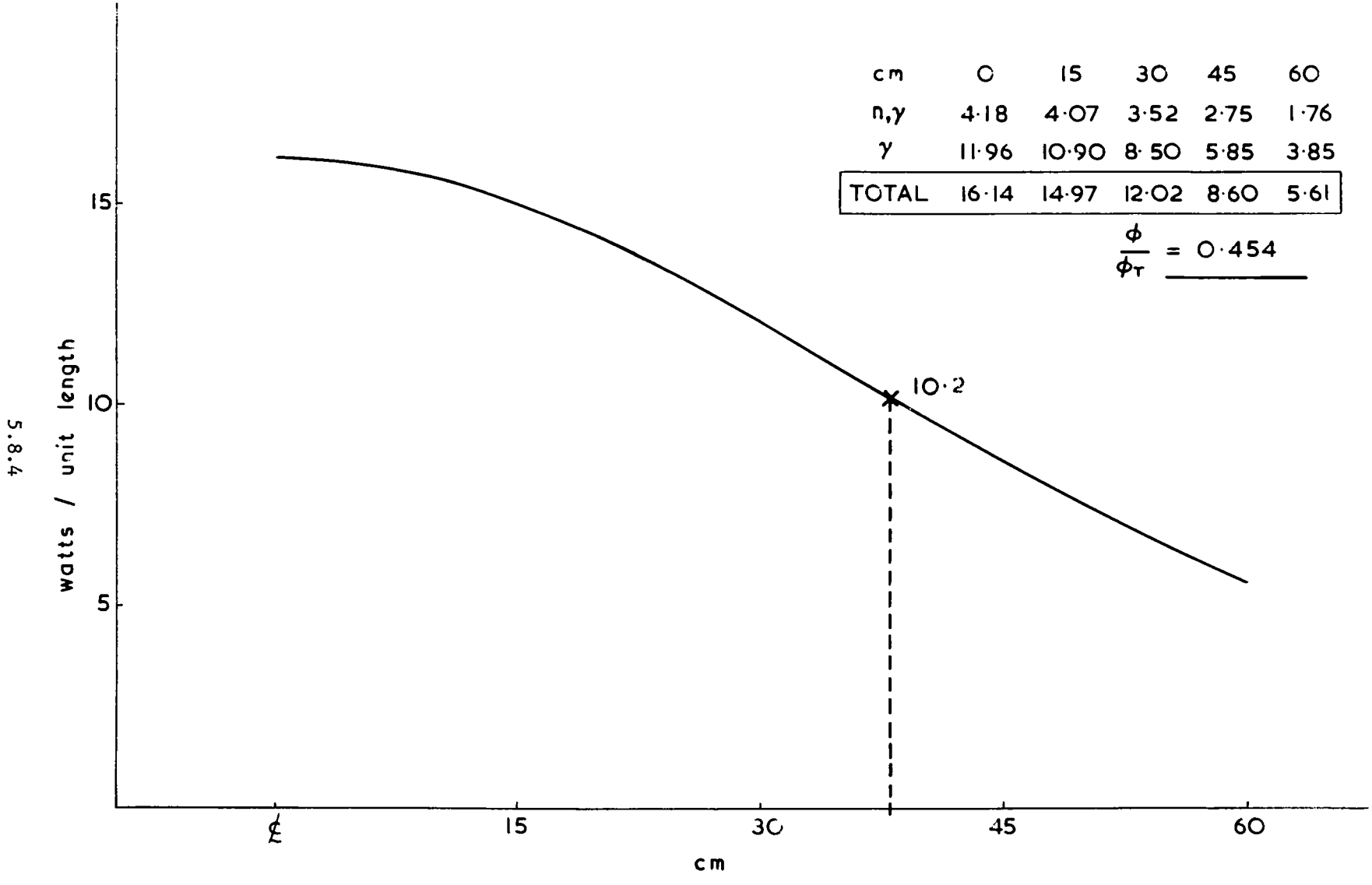
FLUX DEPRESSION

C



TEMPERATURE DISTRIBUTION

D



SIMPLE ION CHAMBER RIG - LARGE CHAMBER

5.8.4

SUB MINIATURE ION CHAMBERS

R. G. Bock
General Electric Company

One of the most widely used methods of obtaining in-core neutron flux level and profile information is the wire irradiation and post-irradiation scanning method. This technique is of necessity awkward and does not give instantaneous flux level readings.

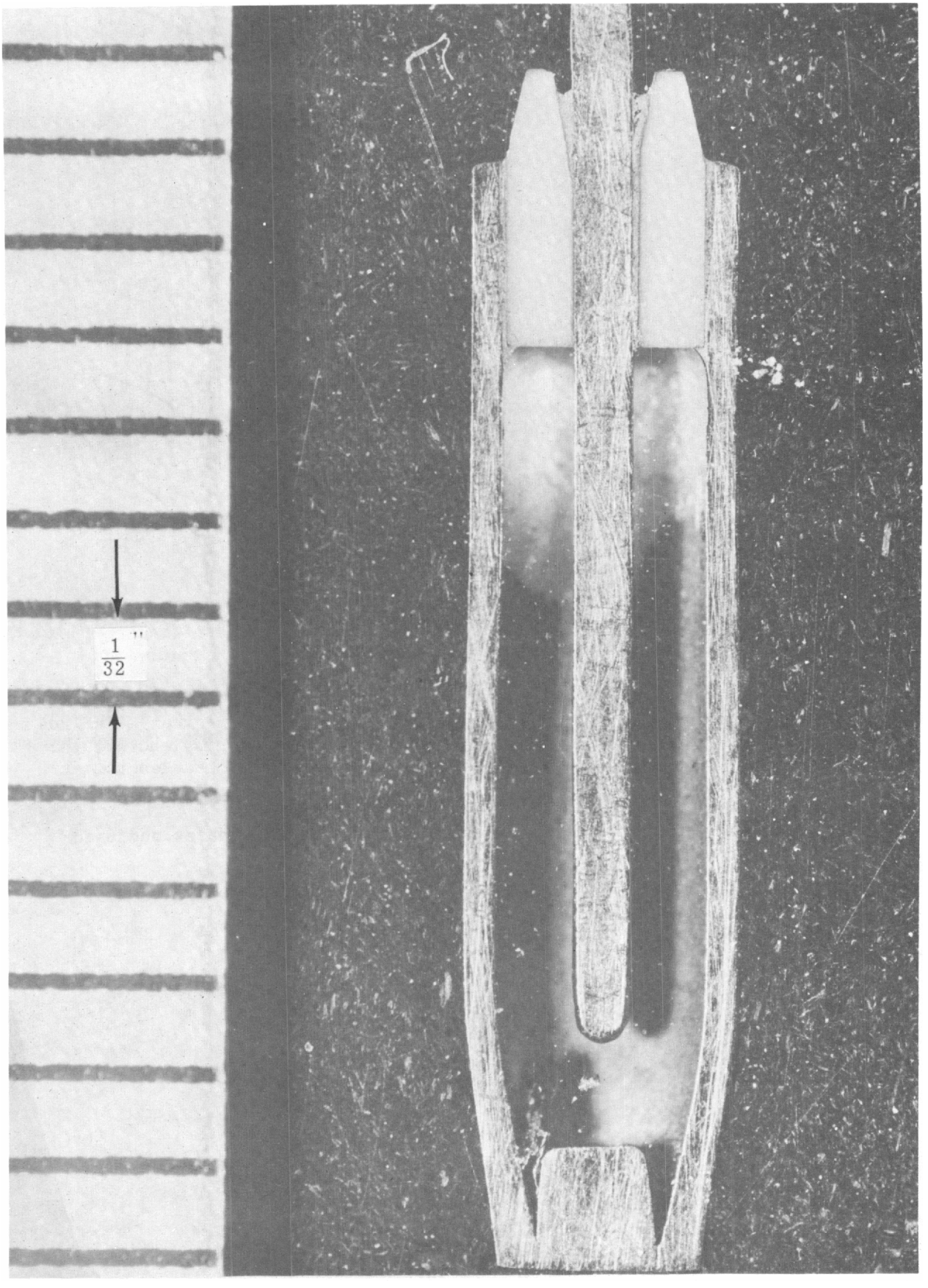
A sub-miniature ion chamber offers the scientist/engineer who is experimenting with or operating nuclear reactors a versatile new tool for solving the flux level and flux mapping problems.

These chambers can be used in fixed positions (in a capsule, for instance) or they can be traversed through the core vertically or horizontally to map neutron flux levels accurately and instantaneously. The chambers can be routed through several in-core locations with the addition of a "switch yard" attachment which can be readily fabricated.

The sub-miniature chamber itself is simply a small rugged ion chamber, with its cathode (outer) shell lined with boron. The chamber utilizes the B^{10} (n, α) Li^{7}. The alpha particle and lithium ion ionize the gas in the ion chamber volume and thereby produce a small current across it. This signal is then amplified and recorded to indicate instantaneously in-core flux levels.

Specifications for this chamber are as follows:

Dimensions:	Diameter - 0.09" to fit 0.125" thimble
	Length - 0.467" (including lead)
Weight:	0.097 grams
Materials:	Anode - titanium
	Cathode - titanium
	Electrode Coating - natural boron (18.8% B^{10}) or enriched boron (96% B^{10})
	Filling Gas - helium
Temperature:	850 F maximum detector temperature



5.9.2

Figure 1. Sub-Miniature Ion Chamber

Operating Characteristics:

Range - 5×10^{12} to 5×10^{14} nv (natural boron)

- 10^{11} to 10^{13} nv (enriched boron)

Sensitivity: (perturbed flux conditions)

Neutron - 4×10^{-19} amp/nv (natural boron)

- 4×10^{18} amp/nv (enriched boron)

Gamma - 3×10^{-15} amp/R/hr

Insulation Resistance: $> 10^{10}$ ohms at 70 F

Operating Voltage: 100 V D. C.

These specifications cover a sub-miniature chamber that has found applications in many experimental reactors throughout the country and in some Naval reactors.

Chambers of different sizes have been fabricated for use in the Vallecitos Experimental Superheat Reactor, the Vallecitos Boiling Water Reactor, the General Electric Test Reactor and the Experimental Gas Cooled Reactor at Oak Ridge, Tennessee.

These chambers have also been affixed to solid sheathed cables and used in SPERT experiments at flux levels in excess of 10^{16} nv.

NOTE: Ceramic-to-metal seals are made with F 202 Forsterite. Forsterite is available from Power Tube Department, General Electric Company, Schenactady, New York.

VI. APPENDICES

APPENDIX "A"

PROPOSED DISCUSSION TOPICS

I. Novel capsule designs - description and discussion of particular new or unusual concepts.

II. Capsule heat transfer

A. Analysis techniques and problems
1. Radial conduction (fins, gas mixtures, special materials)
2. Three-dimensional heat-flow analysis
3. Natural convection of fluids in a capsule
4. Radiative heat transfer
5. Analog and mock-up techniques
6. Specimen temperatures in terms of thermocouple readings

B. Temperature control during irradiation
1. Auxiliary heat - size and location of heaters
2. Variable gas mixtures
3. Mechanical or thermal variation of heat-flow path

III. Flux determination and control

A. Reactor-flux mapping techniques
1. Dosimetry
2. Extrapolation from N_1 to full power

B. Factors producing flux variations with time
1. Control-rod motions
2. Fuel depletion or changes in fuel loading in reactor
3. Perturbation by surrounding experiments

C. Methods of estimating flux perturbation
1. Various machine codes, based on diffusion theory
2. Empirical correlation methods
3. Nuclear mock-ups

D. Variable flux perturbation for control of specimen-effective flux

IV. Capsule components and fabrication

A. Heaters
B. Thermocouples
C. Connectors
D. Seals
E. Materials
F. Welding and brazing
G. Instrumentation

V. Reactor characteristics as they affect capsule experiments

A. Flux-mapping schedules
B. Availability of space
C. Movement of lead capsules
D. On-stream schedules

APPENDIX "B"

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