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Reactors - Power

AEC Research and Development Report

HEAVY WATER MODERATED POWER REACTORS

PROGRESS REPORT

April 1960

Technical Division

Wilmington, Delaware

May 1960

E. I. du Pont de Nemours & Co.
Explosives Department - Atomic Energy Division
Technical Division - Wilmington, Delaware

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Power Reactor Studies
Wilmington, Delaware

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ABSTRACT

Further safeguards analyses of the isolated coolant loops of the HWCTR indicate that accidental loss of cooling water from the loop heat exchangers will not lead to vapor binding of the loop D_2O pumps, or to other detrimental consequences, providing that the reactor is scrammed following the loss of coolant. Several static seals of the types that will be used in the HWCTR exhibited leakage rates that were well below design specifications during cyclic tests at peak conditions of 1500 psi and 260°C. Full-scale experiments with lattices of seven-rod clusters of natural uranium metal at various lattice spacings in D_2O provided measured values for the bucklings, flux distributions, and microscopic parameters of the lattices. A Zircaloy-clad tube of unalloyed natural uranium metal began irradiation in a liquid-water-cooled loop of the NRU reactor under conditions similar to those predicted for D_2O-cooled-and-moderated power reactors.

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HEAVY WATER MODERATED POWER REACTORS

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INTRODUCTION

This report is one of a series that records the progress of the du Pont program on heavy-water-moderated power reactors that are fueled with natural uranium. The broad objective of this program, which includes development, evaluation, and design studies, is to develop the technology to the point where large D_2O-moderated reactors can produce electric power at costs that are competitive with those of fossil-fueled power plants. The present effort is divided into two main categories: (1) the development required for the successful design, construction, and operation of the Heavy Water Components Test Reactor (HWCTR), a high-temperature fuel irradiation facility, and (2) the experimental and theoretical development of full-scale reactor plants. Earlier reports in the series are:

DP-232	DP-375	DP-445
DP-245	DP-385	DP-455
DP-265	DP-395	DP-465
DP-285	DP-405	DP-475
DP-295	DP-415	DP-485
DP-315	DP-425	
DP-345	DP-435	

Progress during May 1960 will be reported in DP-505.

SUMMARY

Erection of the steel shell of the HWCTR building was scheduled to start early in May. All of the stainless-clad steel plate required for fabrication of the reactor vessel has been received by the vessel fabricator, and completion of the vessel by the end of 1960 is forecast.

Further safeguards analyses of the isolated cooling loops of the HWCTR indicate that the accidental loss of cooling water for the loop heat exchangers will not lead to vapor binding in the D_2O pumps of the loops, provided the reactor is scrammed. The reactor pressure transient following a scram will be measured after HWCTR startup to verify the prediction that the pressure in the isolated loops will always exceed the vapor pressure of D_2O.

Cyclic tests in which water leakage rates were measured were completed for samples of the several types of static seals that will be used in the HWCTR. In all instances, leakage rates during about 100 cycles with deionized water at peak conditions of 1500 psi and 260°C were well below design specifications.

Experimental data on bucklings, flux distributions, and microscopic lattice parameters are presented for seven-rod clusters of natural uranium metal at various lattice spacings in D_2O. These data (Tables III and IV, Figures 5 and 6) were obtained from studies of full-scale lattices in the Process Development Pile (PDP) at the Savannah River Laboratory.

An experimental study has been initiated of the causes and means of control of swelling in uranium-base metal fuels during irradiation. The study will include an investigation of the effects of possible in-pile metallurgical reactions on the volume instability of the uranium-base alloys that appear suitable for use in a D_2O-moderated power reactor. The more important metallurgical reactions that can be produced in U - 2 wt % Zr by heat treatment were identified. These data provide an indication of possible metallurgical transformations during irradiations. Preparations are being made for capsule irradiations that will be conducted with the objective of correlating volume instability with (1) initial metallurgical structure, and (2) relative transformation rates.

A Zircaloy-clad tube of unalloyed natural uranium began operation in the liquid-water-cooled E-20 loop of the NRU reactor, in Chalk River, Ontario, on April 13, 1960. The objective of this irradiation is to obtain additional data on the dimensional stability of natural uranium metal under conditions similar to those predicted for D_2O-cooled-and-moderated power reactors. The test element, which has been operating satisfactorily since startup, measures 2.070 inches in OD, 1.467 inches in ID, and about 10 feet in length, and is clad with 0.030 inch of Zircaloy-2. At the end of April, a maximum exposure of about 400 MWD/T had been achieved at a maximum core temperature of 400°C, a maximum surface temperature of 250°C, and a maximum specific power of 23 MW/T. The test will be terminated at the beginning of June, after an exposure of about 1000 MWD/T has been reached, because of Canadian needs for the loop.

Experimental data were obtained on the effect of selected postextrusion variables on the strain limits of the Zircaloy cladding of coextruded uranium tubes. These data were obtained as part of a program to determine to what extent the strain at which the cladding ruptures during irradiation can be increased by controlled processing of the tubes after fabrication. No firm conclusions have been drawn as yet from the data.

A full-scale mockup was fabricated of the fuel assembly that is being developed for irradiation tests of Zircaloy-clad rod bundles of uranium oxide in the HWCTR. The mockup will be used in an extended flow test (liquid water) in which the possibility of vibration and/or other mechanical difficulties will be investigated.

DISCUSSION

I. HEAVY WATER COMPONENTS TEST REACTOR (HWCTR)

The HWCTR is a test reactor that is being built primarily for irradiation tests of fuel elements at conditions of temperature, pressure, and neutron flux that typify those expected in D_2O-moderated power reactors. A description of the reactor was presented in DP-383. Progress during April on design, construction, and supporting experimental work is reported in this section.

A. REACTOR STATUS

Backfilling around the concrete part of the reactor building was completed during April, and erection of the steel shell of the building was scheduled to begin early in May. The application of "Liquid Tile"* sealer to the inner surface of the concrete shell of the building is partially completed.

The remaining pieces of stainless-clad steel plate for the reactor vessel were shipped to the vessel fabricator during April. The fabricator now expects to complete the vessel by the end of 1960.

B. SAFEGUARDS ANALYSES OF ISOLATED COOLANT LOOPS

The results of calculations of transient temperatures and pressures in the isolated coolant loops immediately after a sudden and complete loss of cooling water for the loop heat exchangers were presented in DP-465 and DP-485. It was concluded from these calculations, which considered only the first 35 seconds of elapsed time after the loss of cooling water, that cooling of the loops is adequate to preclude a serious accident if the reactor is scrammed. The calculations have now been extended to longer time intervals to investigate the possibility that the D_2O flow in the loops will eventually cease because of cavitation in the circulating pumps. The results indicate that cavitation will not occur, although the calculated approach to cavitation is fairly close.

If the cooling water flow to a loop heat exchanger is lost, the reactor will scram and the pressure in the main reactor system will fall as the moderator cools and contracts. Since there is no alternative supply of cooling water for the loop, the only available method of cooling the D_2O in the loop is by heat transfer through the Zircaloy bayonet to the moderator in the reactor vessel. If the pressure in the loop decreases to the vapor pressure of the loop D_2O, flow in the loop will stop because of cavitation in the circulating pump, and the fuel will become vapor blanketed. If a metal fuel element is in the loop, it will eventually melt.

* Product of Evershield Products, Inc., Joppa, Maryland

The pressure in each of the two isolated loops is dictated ultimately by the reactor pressure. The liquid-D_2O-cooled loop normally operates at a higher pressure than does the reactor; a rupture disc limits the pressure differential to 700 psi. When the rupture disc bursts, loop pressure and reactor pressure are equalized. A pressure-equalizing valve that bypasses the rupture disc can also be opened manually to equalize the pressures. The boiling-D_2O-cooled loop communicates directly with the reactor, and the two operate at the same pressure.

Estimates of the pressure transients following sudden loss of loop cooling water and subsequent reactor scram are shown in Figure 1 for the liquid loop and in Figure 2 for the boiling loop. It is seen from Figure 1 that about 2.5 minutes after the incident the set point of the rupture disc is reached, and the disc breaks. Thereafter, the reactor pressure and the loop pressure are equal. At the particular operating conditions that were chosen for the calculations, the loop pressure remains above the D_2O vapor pressure, but the two pressures approach one another closely about 10 minutes after the accident. A similar situation exists for the boiling loop (see Figure 2).

The reactor pressure transient following a scram will be measured during the early stages of HWCTR operation. If the actual pressure decrease is more rapid than the calculated decrease, procedures can be adopted for closing the main steam valves of the HWCTR boilers to maintain a high reactor pressure following a loss-of-cooling accident (see DP-475), or an emergency cooling system could be provided for the loops.

The major assumptions in the transient calculations are:

1. The initial reactor pressure is 1000 psia for liquid loop operation and 795 psia for boiling loop operation.

2. The initial pressure in the liquid loop is 1500 psia.

3. The initial average temperature of the moderator is 250°C for liquid loop operation and 240°C for boiling loop operation. The cooling of the moderator is described by a curve similar to curve B-C of Figure 9, DP-475. This cooling rate occurs if the main steam valves of the HWCTR boilers remain at their operating positions during shutdown.

4. The initial D_2O inlet temperature is 250°C in the liquid loop and 258°C in the boiling loop.

5. The temperature in the gas space of the reactor lags the moderator temperature with a time constant of 13 minutes.

6. The D_2O make-up pump for the reactor operates at 30 gallons per minute.

7. The over-all heat transfer coefficient from the loop D_2O to the moderator is 70 pcu/(hr)(ft^2)(^oC). This coefficient includes conduction through an insulating D_2O layer between the Zircaloy bayonet tube and its liner, conduction through the bayonet wall, and a convection coefficient of 250 pcu/(hr)(ft^2)(^oC) outside the bayonet.

C. COMPONENT TESTING

Cyclic tests in which water leakage rates were measured were completed for samples of the several types of static joints that will be used in the HWCTR. In all instances, leakage rates during about 100 cycles with deionized water at peak conditions of 1500 psi and 260°C were well below design specifications. The joints included a 6-inch flanged joint for the central control rod closure, a 1-inch joint for monitor pin assemblies, and four tubing fittings. Descriptions of these joints and of the test procedures were presented in DP-395 and in subsequent progress reports.

A summary of the leakage rates from the test joints is shown in Table I. The average leakage rates of liquid water from the 1- and 6-inch gaskets were 1.0 and 14 $in.^3/mo$,* respectively. Average leakage rates from the four tubing fittings varied from less than 0.01 $in.^3/mo$ for the "Swagelok" fitting to a maximum of 0.22 $in.^3/mo$ for a "Conax" packing gland fitting. In previous cyclic tests at a maximum pressure of 1000 psi, the average leakage rates from a 1- and 6-inch gasket (nominally identical to those tested at 1500 psi) were 0.8 and 3.5 $in.^3/mo$, respectively. Leakage from tubing fittings was not measured separately in earlier tests at 1000 psi. Since the leakage rates were well below the design specifications, the reproducibility of the leakage from the test joints was not determined in either the current test at 1500 psi or the earlier test at 1000 psi.

All of the joints were cycled 100 times except for one tubing fitting that was exposed for 75 cycles. Because of instrument difficulties, however, leakage data on five of the joints were available for only 75 cycles and leakage data on the tubing fitting that was tested for 75 cycles were available for only 50 cycles. During each cycle, which lasted three hours, the joints were maintained at the maximum temperature and pressure for one hour. The balance of the cycle was employed in heating and cooling at 1000 psi. At the end of each cycle, the joints were vented to atmospheric pressure.

* 1 $in.^3/mo$ of liquid water $\simeq$ 0.5 lb/yr of D_2O

II. TECHNOLOGY OF FULL-SCALE REACTORS

A. EXPERIMENTAL LATTICE STUDIES

As was discussed in DP-465, the Process Development Pile (PDP) at the Savannah River Laboratory has been modified for use in an extensive series of physics experiments with lattices of natural uranium fuel in D_2O. The broad objective of these experiments is to provide sufficient data on lattice parameters to specify optimum lattices for full-scale D_2O-moderated power reactors that are fueled with uranium metal or uranium oxide. This section contains a description of the lattices that are being investigated in the initial experiments, and presents the results that have been obtained thus far.

1. Description of Lattices

The first lattices that are being studied in this program are composed of rods of natural uranium metal that are $0.998\ ^{+0.001}_{-0.002}$ inch in diameter. Each rod is held in an aluminum housing tube that is 1.026 inches in OD and 1.090 inches in ID. The lattice configurations that are to be included in the studies are listed in Table II.

A typical lattice arrangement with the fuel at a center-to-center spacing of 14 inches in the PDP tank (195 inches in ID) is shown in Figure 3. The various safety, control, and shutdown rods shown in this diagram are for operational convenience only and are fully withdrawn from the lattice when physics measurements are being made. Under these conditions, the pile is held critical by adjustment of the water height. For some of the more reactive lattices that would have inconveniently low critical water heights, a reduced pile loading of the type shown in Figure 4 is employed. In the reduced loadings, the outer region of the pile is poisoned with absorber rods to produce an effective tank diameter of about 100 inches.

2. Experimental Results

Preliminary experimental data on buckling, flux distribution, and microscopic lattice parameters for seven-rod clusters are presented in Tables III and IV and in Figures 5 and 6. The results are discussed individually below.

a. Buckling and Flux Distributions

The bucklings of the lattices were measured by bringing the pile critical and irradiating gold pins. The pin activities were fitted to cosine and J_o functions by the method of least squares. As many as 350 pins were used in each run. The bucklings were corrected for temperature, the presence of safety rod thimbles, and the poisoning by the gold. The results that have been analyzed to date are given in Table III.

A 14-inch lattice was also measured with the central section empty of fuel. The main purpose of this loading was to obtain a central, highly thermalized flux for later experiments with single elements. Over-all measurements of flux shape were made with both bare and cadmium-covered gold pins, and the results were compared with two-group theory. An axial buckling of 234.9 μB* was measured. Other input parameters are shown below.

	Region 1	Region 2
Fast diffusion constant, D_f	1.222	1.222
Slow diffusion constant, D_s	0.84	0.84
Fast removal cross section, Σ_r	0.009622	0.009622
Fast absorption cross section, Σ_{af}	0	0.001133
Slow absorption cross section, Σ_{as}	0.000082	0.003402
Radius, cm	113.57	255.0

The production term used in two-group calculations for the outer region, $\nu\Sigma_f = \eta\epsilon f\Sigma_a$, was varied to achieve criticality. The value at which this condition was reached was $\nu\Sigma_f = 0.004630$ cm^{-1}, which corresponds to a value of η of 1.290 for natural uranium. This value is predicated on values of 1.047 and 0.9795 for ϵ and f, respectively. The calculated distributions of slow flux and fast flux are shown in Figure 5. The flux distributions were normalized respectively to the subcadmium and epicadmium gold activations, which are also shown.

The steep drop in thermal flux in the central D_2O region is due to the large axial leakage. A moderator height of 400 cm rather than 200 cm would have produced a flat, or even a peaked, thermal flux in the central region.

b. Neutron Reproduction Factor (η)

For this study, η is computed from the relationships

$$\eta = \frac{2.07}{1 + 0.56708r}$$

$$r = \frac{A_{28}/A_{28}^M}{A_{25}/A_{25}^M}$$

where A_{28} is the relative number of subcadmium absorptions in U^{238} and A_{25} is that number in U^{235}. The M's refer to the absorptions that

* 1 μB = 10^{-6} cm^{-2}

would be experienced in a Maxwellian distribution. The value for a Maxwellian distribution is calculated by letting r = 1, in which event $\eta = 1.321$. Absorption measurements were made with U^{235} foils and Cu or Mn foils (simulating U^{238}), both in the fuel elements and in thermalized sections of the pile. Values of η computed from these measurements are presented in Table IV.

c. Resonance Escape Probability (p)

(1) Measurement of p - To obtain values of the resonance escape probability from experimental data, measurements were made of the parameter ρ, defined as the ratio of epicadmium radiative absorptions in U^{238} to subcadmium radiative absorptions in U^{238}. The determination of ρ was made in the following manner:[1] Foils of natural uranium and depleted uranium (350 ppm U^{235}), both bare and cadmium-covered, were placed in the central rod and one of the outer rods of a seven-rod cluster. Aluminum catcher foils 0.001 inch thick were used to protect the foils from fission fragments. After the fuel assembly was irradiated, the foils were removed and set aside for three days. They were then counted with a NaI scintillation spectrometer that utilizes a single-channel pulse-height analyzer. The channel was set at 103 kev to count the Pu X-ray resulting from the decay of Np^{239}. The intensity of this X-ray is proportional to the radiative capture in U^{238}. The count rates were corrected for background (by counting unirradiated foils), decay, foil weight, gamma ray attenuation, differences in vertical location in the pile, U^{235} fission activity, and U^{238} fission activity. An average value of ρ for the cluster could then be obtained from the corrected count rates.

In defining resonance escape probability, the region of resonance absorption was regarded as a rather narrow energy band. The resonance escape probability was then defined as the fraction of neutrons escaping this band after entering it. This definition leads to a more complicated relation between p and ρ than that in the reference cited[1], namely

$$\frac{1 - p}{p} = \frac{(\rho - R)fFp(1/v)}{1 + L^2B^2}$$

where

p = resonance escape probability

$$\rho = \frac{\text{epicadmium radiative captures in } U^{238}}{\text{subcadmium radiative captures in } U^{238}}$$

(1) Klein, D., et al. "Measurements of Thermal Utilization, Resonance Escape Probability, and Fast Fission Effect in Water-Moderated, Slightly Enriched Uranium and Uranium Oxide Lattices", Nuclear Sci. and Eng. 3, 403-27 (1958).

$$= \frac{\text{epicadmium 1/v radiative captures in } U^{238}}{\text{subcadmium radiative captures in } U^{238}}$$

$$\sim \frac{\text{epicadmium } U^{235} \text{ fissions}}{\text{subcadmium } U^{235} \text{ fissions}}$$

f = thermal utilization (measured)

L^2 = thermal diffusion area (measured)

B^2 = buckling of lattice (measured)

$$F = \frac{\Sigma_a^{28}}{\Sigma_a^{28} + \Sigma_a^{25}} = 0.3619$$

$p(1/v)$ = probability that a neutron leaving the resonance region and entering the 1/v absorption region will reach cadmium cutoff energy without being absorbed

This last factor is computed from a rather complex relation between measured parameters, the most important of which is the cadmium ratio of a 1/v absorber in the fuel. Foils of U^{235} were used to obtain an estimate of this ratio. For the lattices that were studied, $p(1/v)$ is roughly 0.95.

The measured values of ρ and the values of p calculated therefrom are listed in Table IV.

(2) Effectiveness of Internal Surfaces in Resonance Absorptions (Dancoff Effect) - In the usual expression for the effective resonance integral of U^{238} in terms of a surface-to-mass ratio, it is implicitly assumed that all of the surface faces an infinite volume of moderator. In a fuel assembly consisting of multiple rods, however, much of the internal surface is partially shielded by the other rods from neutrons having energies equal to the U^{238} resonance energies. This shielding effect is called the Dancoff effect. For every point on the surface one can define a parameter γ as the ratio of the current of neutrons having energies corresponding to U^{238} resonances (surface-absorbed neutrons) to the corresponding current at a surface facing an infinite volume of moderator. For a rod cluster, γ can be calculated by numerical integration.[2] The result is that the value of γ for the central rod of a seven-rod cluster of 1-inch rods with a center-to-center spacing of 1.5 inches is very insensitive to angular position around the rod. The average value is $\gamma = 0.641$. The calculated γ for an outer rod is a strong function of position, however. The results of the calculations are shown in Figure 6.

(2) Carlvik, I. and B. Pershagen. The Dancoff Correction in Various Geometries. Aktiebolaget Atomenergi, Stockholm, Sweden. AE-16, 23 pp. (1959).

To verify the calculations, the Dancoff effect was measured. Rod sections (1 inch in diameter) were machined into twelve-sided regular prisms that were 1/2 inch high and that measured 1 inch between opposite vertices. In each face was milled a cylindrical depression 0.010 inch deep. A foil of depleted uranium (0.019% U^{235}) that was 1/4 inch in diameter and 0.006 inch thick was taped in position in each depression. Provision was also made for locating foils within the uranium, completely shielded from "surface-absorbed" neutrons. Each prism was then covered with a 0.020-inch sheet of cadmium, and two such prisms were placed in the central rod and in an outer rod of a fuel cluster. Two days after irradiation of the assembly, the foils were counted. A thin (0.4 cm) NaI crystal and a differential discriminator were used to count selectively the Pu^{239} X-ray resulting from the decay of Np^{239}. These activities are proportional to absorption of neutrons by U^{238}. On the basis of an assumed flat distribution of the volume absorption component of the resonance absorption, the activities of the completely shielded foils within the rods were then subtracted from the activities of the foils at the surface.* The result should be proportional to surface resonance absorption, and hence to γ. The experimental points are plotted in Figure 6. The average of the twelve experimental points obtained on the central rod was normalized to a γ of 0.641, which is the average that was predicted theoretically. There is more scatter in the data than would be expected from counting statistics. Aside from this scatter, however, the agreement between experiment and theory is quite good.

d. Fast Fission Factor (ϵ)

In the determination of ϵ, a measurement is made of δ, the ratio of U^{238} fissions to U^{235} fissions. The experiment was performed as described by Futch.(3) Foils of depleted uranium and of natural uranium were placed in the central fuel rod and in an outer fuel rod. After the irradiation was completed, the foils were counted alternately.

After corrections were made for the difference in decay rates of fission products from U^{235} fissions and U^{238} fissions, a value of δ was obtained. The values of ϵ were computed from the relation

$$\epsilon = 1 + \frac{\nu_f - 1}{\nu_{th}} \delta$$

* The activities of these shielded foils amount to about 40% of the activities of foils at the surface of the central rod. Most of this activity is due to volume resonance absorption, with a small background of fast fission activity being less important. Epicadmium U^{235} fission is negligible compared to these activities.

(3) Futch, A. H., Jr. "Fast Fission Effect in Lattices of Natural Uranium and Heavy Water", Nuclear Sci. and Eng. 5, 61-7 (1959).

where ν_f and ν_{th} are the average numbers of neutrons produced per fast and slow fission, respectively. The results are given in Table IV.

e. Thermal Utilization (f) and Diffusion Area (L^2)

The values of f and L^2 were determined by irradiating Mn pins both in the moderator and in the fuel. The counts were corrected for decay and Cd-covered counts were deducted. From plots of the corrected activities, average fluxes were obtained for the moderator, the fuel rods, the aluminum, and the D_2O within the cluster.

The formula used for f is

$$f = \frac{A_u}{A_u + A_o}$$

Where A_u refers to thermal absorptions in the uranium and A_o refers to all other thermal absorptions in the cell.

The expression used for L^2 is

$$L^2 = \frac{D_{eff}}{\Sigma_{a_{eff}}}$$

where D_{eff} is the effective diffusion constant and $\Sigma_{a_{eff}}$ is the effective thermal absorption cross section of the cell. The results are shown in Table IV.

B. REACTOR FUELS AND MATERIALS

One of the most important objectives of the du Pont program on D_2O-moderated power reactors is to develop a fuel element with which a total fuel cost of 1 mill/kwh or less can be realized in a full-scale reactor. Two possible routes to low cost fuel are being investigated:

1. Efforts are being made to develop acceptable fuel elements of uranium metal. This material is potentially cheap to fabricate, but its ability to withstand the desired burnup under power reactor conditions has not yet been demonstrated. Zircaloy-clad tubes of uranium metal for irradiation tests are being fabricated at Nuclear Metals, Inc., via a coextrusion process.

2. A swaging process for production of Zircaloy-clad tubes of uranium oxide is being developed at the Savannah River Laboratory with the objective of decreasing the fabrication cost of suitable oxide elements. One of the principal questions with respect to oxide tubes is whether the inner cladding will collapse from internal pressure. To obtain early information on this question, irradiation tests of swaged oxide tubes with stainless steel cladding are under way at

Savannah River. The stainless steel is being used as a temporary substitute for Zircaloy sheath stock, which is not yet available in quantity.

The progress during April on these programs is discussed below.

1. Fuel Elements of Uranium Metal

a. Fabrication of Irradiation Specimens

As discussed in detail in DP-485, the emphasis in the development of fuel elements of uranium metal has been shifted to thin-walled coextruded tubes in preference to the relatively thick tubes that have been evaluated heretofore. The cross-sectional dimensions have been defined for a two-tube assembly of thin tubes that will be suitable for irradiation tests in a liquid-D_2O-cooled position of the HWCTR (see Figure 8, DP-485). The first irradiation specimens of the new design will be VBWR tubes of the same dimensions as the outer tube of the HWCTR assembly (2.06 inches in OD, 1.700 inches in ID). These specimens will have cores of U - 1.5 wt % Mo and U - 1 wt % Si. Preparations are being made for coextrusion of natural uranium prototypes of each of these two alloys to establish the billet design required for control of cladding thickness and core length when tubes of the new size are extruded.

b. Swelling of Uranium-Base Alloys

Fuel tubes of U - 2 wt % Zr have shown a marked volume instability (swelling) during irradiation at uranium temperatures of about 400°C. Unalloyed uranium tubes, on the other hand, appear somewhat more stable at these temperatures. This is in accord with experience at other sites, which also indicates that U-Mo alloys may be more stable than unalloyed uranium.[4] A program has been initiated at the Savannah River Laboratory (SRL) to study the causes and means of control of swelling in uranium-base metal fuels. This program will include an investigation of the effects of possible in-pile metallurgical reactions on the volume instability of the dilute uranium-base alloys that appear suitable for use in a D_2O-moderated power reactor. The basis of the program is the observation at Westinghouse that swelling of U-Nb and U-Nb-Zr alloys can be correlated with the decomposition rate of the gamma phase of these alloys at irradiation temperatures. The gamma phase may be formed at low temperatures by irradiation (phase inversion) in these and other alloys (U-Mo). It is thought that the occurrence of these metallurgical transformations during irradiation makes the alloys more susceptible to fission gas swelling.

(4) Bentle, G. G. "Relationship Between Pre-Irradiation Properties and Irradiation Swelling in Metal Fuels." Transactions of the American Nuclear Society, 2, No. 2, Nov. 1959. Paper 10-2, p. 113.

The initial objective of this program is the characterization of the metallurgical reactions that can be produced in several alloys by heat treatment, as an indication of possible transformations in a reactor (induced by irradiation and/or temperature). Both U - 2 wt % Zr and U - 1.5 wt % Mo alloys are being studied; the results obtained on the U - 2 wt % Zr alloys are reported below. The second stage of the program will be a series of irradiation tests of uranium-base alloys with the objective of correlating volume instability with (a) initial structure, and (b) relative transformation rates. Tubular specimens 1.50 inches in OD, 1.14 inches in ID, and about 8 inches long will be irradiated in a Savannah River reactor in stainless steel capsules. Elevated temperatures will be attained via insulation of the capsule with lead. The design of the insulated capsule is nearing completion, and irradiation specimens of unalloyed U, U - 2 wt % Zr, and U - 1.5 wt % Mo are being fabricated.

(1) Effect of Heat Treatment on U - 2 wt % Zr

A large variety of structural effects in both matrix grain size and second-phase distribution can be produced by heat treatment of U - 2 wt % Zr. The most important effects are discussed below:

"Gamma" Treatments - An acicular matrix structure with little or no second phase evident is formed on quenching from gamma-phase temperatures (800°C), as shown in Figure 7. Precipitation of the second phase (gamma-2 or delta) occurs on subsequent annealing at alpha-phase temperatures above about 400°C (Figure 8).

Slow cooling from gamma-phase temperatures, as in the "diffusion" heat treatment employed on U - 2 wt % Zr tubes that have been irradiated, produces an irregular patch-like grain structure, with a lamellar second phase distributed throughout the matrix grains (see Figure 9).

"Beta" Treatments - A large-volume second phase (probably retained gamma-1) is produced on quenching from beta + gamma-1 phase temperatures (e.g., 720°C), as shown in Figure 10. The matrix phase is probably alpha, as indicated by its grain structure. Annealing the beta-quenched structure at temperatures greater than about 400°C produces an evident decomposition of the second phase by precipitation of a finely dispersed component (gamma-2 or delta) within the parent constituent (Figure 10).

Another effect of potential significance to the reactor behavior of the alloy is the production of large matrix grains during "spheroidization" treatment, a long-term anneal at 680°C which agglomerates the second phase into discrete particles (see Figure 11). The large grain size could cause severe surface roughening during irradiation, an effect that has been observed after irradiation of a power reactor fuel tube that received this type of heat treatment.

(2) Coextrusion of Zr-Clad Uranium and U - 2 wt % Zr for Capsule Irradiation Tests - To provide cores for the coextrusion billets that will be used for fabrication of capsule specimens, ingots were cast of unalloyed uranium, U - 2 wt % Zr and U - 1.5 wt % Mo. The carbon content of the natural uranium scrap that was used in preparing these heats was 500-600 ppm; no carbon analyses are yet available on the ingots. Low carbon dingot metal was ordered to provide melting stock for future heats.

The initial design and procurement of tools and billet components were completed, and preliminary extrusions were made at SRL. Three billets containing natural uranium cores (tripled-quenched from 740°C) and one billet with a U - 2 wt % Zr core (in as-cast condition) were extruded and destructively examined to provide data for adjusting the extrusion system. It was necessary to make minor changes in the die, mandrel, and billet components to achieve the desired tube dimensions.

Cupping of Zircaloy to provide sheaths for extrusion billets was successful, and billets were prepared for the extrusion of end plug material.

2. Fuel Elements of Uranium Oxide

Fabrication was completed on a full-scale mockup of the fuel assembly that is being developed for irradiation tests of Zircaloy-clad rods of uranium oxide in the HWCTR. Irradiation tests of swaged oxide rods are contemplated as a backup to the development of swaged oxide tubes at SRL. The fuel assembly, which contains 19 rods of 0.550-inch OD, was described in DP-485. The mockup will be used in an extended flow test in which the possibility of vibration and/or other mechanical difficulties will be investigated.

The mockup was fabricated according to the design described in DP-485 except that the individual rods were made of stainless steel tubing that was filled with lead to approximate the weight of UO_2 rods. Because a standard size of stainless steel tubing was used for the mockup of the rods, the finished rods were 10 mils smaller in diameter than was specified in the design. The final assembly of the rods into the finished bundle resulted in a wrapped bundle diameter of about 3.100 inches, which was 50 mils undersize. It is expected that this deviation from the specified diameter will have little effect on the preliminary results obtained from the mechanical tests.

A second mockup of the bundle is now being prepared to the exact final dimensions. This second assembly will be used for flow calibration (liquid H_2O) and determination of pressure drop. It then will be placed under extended flow test in water at 260°C for a study of its mechanical stability under conditions simulating those in a liquid-cooled position of the HWCTR.

3. Cladding Studies

a. Strain Limits of Zircaloy Cladding

The unexpectedly low cladding strains (~1%) at which some of the coextruded tubes of uranium metal have failed during irradiation tests have prompted an experimental study at Nuclear Metals of the effects of selected variables on the strain limits of Zircaloy cladding. The objective of this program, which was described in DP-475, is to determine to what extent the rupture strain can be increased by control of post-extrusion processing of the tubes. The program consists of a series of out-of-pile tests in which cladding strain is simulated by uniformly applying stress to a tubular specimen by means of an expanding copper plug at ambient temperature. The expansion of the copper is accomplished by forcing a tapered steel plug through a matching hole in the copper. The per cent change in the circumference of the specimen at rupture is called the fracture strain.

The initial experimental results from this program are reviewed in this section. Conclusions as to the significance of the results are being deferred pending receipt of additional data. The results were obtained on tubular specimens of (1) outer cladding, (2) uranium core, and (3) a composite of core and outer cladding; all specimens were machined from coextruded tubes. In most instances, the results represent average values for several specimens.

Outer Cladding

1. Zircaloy-2 cladding in the pickled condition (after postextrusion removal of the copper extrusion jacket) exhibited low ductility, viz., a fracture strain of 4%.

2. The fracture strain of pickled Zircaloy-2 cladding that was polished to remove longitudinal striations or was brightly etched was 7%.

3. Zircaloy-4 cladding showed slightly greater ductility than Zircaloy-2. However, the difference may not be significant; in each case the fracture strain was near 7%.

Bare Cores of Unalloyed Uranium and Core-Cladding Composites

1. Bare cores of either ingot or dingot uranium, and a composite of Zircaloy cladding and uranium core all showed low ductility in the as-extruded condition. Fracture strains were 2 to 3%.

2. The fracture strain of bare core specimens and of composite specimens from a beta-treated, etched tube of dingot uranium was 6%.

3. After beta treating, etching, and autoclaving, composite specimens from a dingot tube showed still greater ductility, with 15% strain at fracture.

Bare Cores of U - 2 wt % Zr and Core-Cladding Composites

1. Specimens of cladding, core, and cladding-core composite from an as-extruded Zircaloy-clad U - 2 wt % Zr tube after pickling and etching all exhibited fracture strains of about 7%. The values for both core samples and composite samples were significantly higher than those obtained for comparable samples of unalloyed uranium (2 to 3%).

2. Composite samples that were tested after a randomizing heat treatment (800°C for 10 minutes) showed greater ductility (11% fracture strain) than the same material in the as-extruded condition.

3. Core samples that were tested after diffusion heat treatment (880°C for 7 hours) showed still greater ductility (18% fracture strain), but cladding and composite samples from this tube fractured at the same 7% strain as did the as-extruded samples.

4. The standard autoclave treatment (24 hr in water at 340°C, 24 hr in steam at 400°C) did not appear to increase the ductility of U - 2 wt % Zr samples; an increase was observed for unalloyed uranium.

5. An attempt was made to simulate the accumulation of zirconium hydride that has been observed near the surface of the cladding of irradiated fuel elements. For this experiment, samples from a Zircaloy-clad tube that was diffusion heat treated and etched were subjected to cathodic deposition. Subsequent burst tests of cladding-core composite specimens showed a small increase in ductility, i.e., the fracture strain increased from 7 to 9%. The fracture strain of cladding specimens increased from 7 to 17%. The significance of these results, which are the reverse of those expected, can not be assessed at this time.

b. Zircaloy Etching

Studies of the Zircaloy etching process that is used to remove surface impurities from coextruded fuel tubes continued as part of the effort to reduce the cost of the coextrusion process. Further data were obtained that support the previously reported conclusion (DP-465 and DP-475) that the following process economies can be instituted without adverse effect on the corrosion resistance of Zircaloy cladding:

1. Repeated rejuvenation of the etch bath by replenishment of its hydrofluoric acid content.

2. The substitution of commerical grades of nitric acid and hydrofluoric acid for laboratory grades.

3. The use of tap water in place of nitric acid for the initial rinses.

The modified process described above has now been applied to a full-size Zircaloy tube. The etch bath contained 1000 ppm of copper to exaggerate the effect of incomplete removal of the copper extrusion jacket in the prior pickling step. As in earlier experiments, satisfactory results were obtained for this tube in the standard 48-hr autoclave test that is used to determine corrosion resistance of the Zircaloy. An additional exposure for two weeks in steam at 400°C produced no white oxide or other evidence of objectionable corrosion.

Still further evidence of the acceptability of the modified process has been obtained in laboratory-scale tests of the effect on etch efficacy of abnormally high concentrations of three of the impurities in commercial grade nitric acid. Zircaloy rings were etched in duplicate in solutions containing two and ten times the usual concentrations of iron, chloride, and sulfate ion. The appearance of the etched rings after subsequent steam corrosion tests indicates that acid impurities within the range tested should not adversely affect the corrosion resistance of etched cladding.

c. Evaluation of Zircaloy Cladding Stock

Eighteen 8-foot lengths and six shorter lengths of thin-walled Zircaloy tubing were received from Harvey Aluminum Company for evaluation at SRL. This shipment was the final complement of an order for 200 feet of Zircaloy cladding stock for swaged tubes of uranium oxide. The tubing was examined visually for surface quality and measured for dimensions and straightness; results are summarized in Table V. Although a few pieces had isolated dimensional variations and bow that were greater than desired, the over-all quality was excellent. These tubes were superior to the initial lot of tubes from Harvey with respect to dimensions;* no cracks or fabrication defects in either lot were detected by visual inspection. The second lot of tubing will be examined by the "Zyglo-Penetrex" method for defects that could not be detected visually.

The bow measurements are the most significant from a process viewpoint because the eccentricity of uranium oxide in swaged tubes is related to the bow of the sheath tubes. Of the eighteen full-length tubes, three outer sheaths and two inner sheaths exhibited a bow of at least 3/32 inch. The bow in these five tubes may cause both alignment problems during oxide loading and oxide eccentricity after swaging; the remaining thirteen tubes are presumably satisfactory from this

* Inspection results for the initial lot of tubing were presented in DP-485.

standpoint. However, at this stage in the swaging development, the allowable limit of bow is not known and this specification must be developed as the work proceeds.

d. Corrosion Testing of Zircaloy-2

Samples of Zircaloy-2 purchased from Allegheny-Ludlum as end plug stocks for swaged oxide tubes and for extrusion billets for encapsulated fuel elements of uranium metal were tested for corrosion resistance in steam at 400°C for 72 hours. Most of the Zircaloy from six tubes for oxide end plugs (2.5 inches in diameter, 0.517-inch wall) exhibited high weight gains and stringer-type corrosion. All of this material will be used for temporary metal end plugs and low temperature irradiations. Samples from the extrusion billets developed an adherent black oxide, but the weight gains were higher than was desired for cladding material. If all of the billet material is used for encapsulated tubes and for fabrication development work, there should be no serious consequences.

C. IRRADIATION TESTS OF POWER REACTOR FUELS

The status of du Pont irradiation tests at the following sites are reported in this section: (1) NRU, Chalk River, Canada, (2) VBWR, California, and (3) Savannah River.

1. Zircaloy-Clad Uranium Tube in the NRU E-20 Loop

A Zircaloy-clad tube of unalloyed natural uranium began operation in the liquid-water-cooled E-20 loop of the NRU reactor, in Chalk River, Ontario, on April 13, 1960. The objective of this irradiation is to obtain additional data on the dimensional stability of natural uranium metal under conditions similar to those predicted for D_2O-cooled-and-moderated power reactors. The test element, which has been operating satisfactorily since startup, measures 2.070 inches in OD, 1.467 inches in ID, about 10 feet in length, and is clad with 0.030 inch of Zircaloy-2. By the end of this report period a maximum exposure of about 400 MWD/T had been experienced by the tube. The steady-state conditions for the test are listed below.

NRU reactor power, MW	200
Total output of loop, kw	935
Output of du Pont tube, kw	885
Maximum specific power, MW/T	22.9
Maximum heat flux, pcu/(hr)(ft^2)	275,000
Maximum-to-average flux ratio	1.265
Temperatures	
Coolant inlet, °C	177
Coolant outlet, °C	196
Maximum surface, °C	249
Maximum core-clad interface, °C	333
Maximum core temperature, °C	397

The temperatures and heat fluxes being experienced by the uranium tube in the E-20 loop are within the range of interest for the du Pont designs of full-scale power reactors, even though the neutron flux level and consequently specific power are considerably lower. For a boiling-D_2O-cooled power reactor the maximum design temperatures are about 285 and 350°C for the surface and core, respectively; for a liquid-D_2O-cooled reactor, the corresponding design temperatures are about 310 and 440°C. Because the test element is about twice as thick as the elements designed for the full-scale reactors, appropriate core temperatures are reached in spite of the lower specific power.

Current schedules for the E-20 loop require that du Pont relinquish occupancy of the loop at the beginning of June. At that time, the tube will have reached an exposure of about 1000 MWD/T.

2. Metal Tubes in the VBWR

At present, the VBWR reactor is still shut down for modifications. The system is being modified to permit forced convection boiling in the core. The net result is that considerably higher fluxes and power levels are anticipated in the VBWR. These increased power conditions will permit closer duplication of the power reactor design conditions in the du Pont test elements.

Current plans are to continue the irradiation of the thick-walled tube of Zircaloy-clad U - 2 wt % Zr that had reached 1200 MWD/T in earlier tests in the VBWR and to start two irradiations of thin-walled uranium tubes. The present emphasis on thin-walled tubes was reported in DP-485 and the coextrusion of these tubes for VBWR testing was mentioned earlier in this report. The middle of June is the expected startup date for the modified VBWR.

3. Oxide Fuel in Savannah River Reactors

Two types of oxide irradiations are in progress in Savannah River reactors - swaged oxide tubes and lead-insulated swaged oxide rods.

During April an assembly containing five short tubes of UO_2 powder swaged in stainless steel sheaths continued uneventfully in its irradiation test in a Savannah River reactor. The conditions of the test will be fully reported in a forthcoming classified report. The testing of two other five-tube assemblies, in which one failure occurred, was reported in DP-485.

In addition to the swaged-tube specimens, the irradiation of swaged oxide rods clad in Zircaloy-2 was begun. The assembly, which was described in DP-485, contains eight swaged rods that are insulated by a thin layer of lead. The purpose of the lead is to raise the temperatures within the fuel to those of interest for power reactor application. The details of this test will also be given in a forthcoming classified report.

D. HEAVY WATER LEAKAGE

Existing data on the leakage of fluid past the mechanical shaft seals of centrifugal pumps are inadequate for reliable prediction of the unrecovered loss of D_2O from pumps that operate at power reactor pressures and for confident design of D_2O handling equipment and recovery facilities. As part of a program to provide experimental data for these purposes, bench-scale tests of commercial mechanical seals under simulated conditions of pump operation have been started at SRL. The test equipment has been installed, and initial tests were begun in later April.

The test equipment is designed to simulate an operating pump with respect to seal temperature, pressure, and static forces, but will not simulate dynamic forces. It consists of a pressure chamber, a shaft upon which seals may be mounted, shaft supports, and a variable-speed drive. Operation is at room temperature instead of at elevated temperature because it is customary in pump installations to cool the leakage flow before it escapes from the seals. The water leakage can be measured directly and/or by moisture analysis of a stream of nitrogen gas which is passed through the seal collection chamber. The latter technique affords a method of measuring the combined leakage of vapor and liquid. Mechanical seals made by three different manufacturers for shaft diameters of 2 to 4-5/8 inches are scheduled to be tested at various shaft speeds and system pressures.

TABLE I

SUMMARY OF LEAKAGE RATES FROM HWCTR CLOSURES

Leakage Rate, $in.^3/mo$ of Liquid Water (At 25°C)

Seal Description	Average Rate During 75 to 100 Cycles	Allowable Rate (For Design Purposes)	Average Rate		
			Heating Phase (at 1000 psi)	At 1500 psi and 260°C	Cooling Phase (at 1000 psi)
"Swagelok" tubing fitting, 1/4-inch pipe to 3/4-inch tubing	0.01	1.2	0.01	0.01	0.01
"Midlock" tubing fitting, 1/4-inch pipe to 1/4-inch tubing	0.05	1.2	0.06	0.07	0.03
"Conax" tubing fitting, lava ring insert, 1/4-inch pipe to 1/4-inch tubing	0.22	1.2	0.26	0.28	0.11
Parker tubing fitting, flared type, 1/4-inch pipe to 3/8-inch tubing	0.02	1.2	0.02	0.03	0.02
1-inch jacketed asbestos gasket for monitor pin assembly	1.0	11	0.9	0.5	1.6
6-inch "Flexitallic" gasket of flanged joint for control rod enclosure	14	94	11	16	16

Note: 1 $in.^3/mo$ of liquid water ≃ 0.5 lb/yr of D_2O

TABLE II

PDP LATTICES OF URANIUM METAL RODS

Lattice Pitch (Triangular), inches	Number of Rods Per Fuel Cluster			
	1	3	7	19
4.667	x			
7.000	x	x		
8.083	x	x		
9.334	x	x	x	
12.124	x	x	x	
14.000		x	x(a)	x
18.520			x(a)(b)	x
21.000			x	

(a) Performed in 99.3 and 99.75% D_2O. All other lattices are to be studied in 99.75% D_2O only.

(b) Performed in 99.75% D_2O at center-to-center rod spacings of 1.5 and 2 inches. All other clusters are to be evaluated at a spacing of 1.5 inches only.

TABLE III

MATERIAL BUCKLINGS OF SEVEN-ROD CLUSTERS OF NATURAL URANIUM METAL

As Derived from PDP Experiments

Lattice Pitch, inches	D_2O Purity, mol %	Thimble Correction, μB	Gold Pin Poisoning Correction, μB	Temp. Correction to 22°C, μB	Corrected B_m^2, μB	Expected Error, μB
18.52	99.75	+7.9 ±1	+5.0 ±0.5	+0.5 ±0.1	292.3	±4
18.52	99.30	+7.4 ±1	+1.5 ±0.5	+0.7 ±0.1	212.3	±2
14.00	99.75	+7.8 ±1	+1.5 ±0.5	-0.4 ±0.1	516.9	±6
14.00	99.30	+7.3 ±1	+1.5 ±0.5	+2.3 ±0.2	460.4	±4
12.124	99.75	+9.7 ±1	+5.0 ±0.5	+1.6 ±0.1	610.0	±4
21.00	99.75	+7.0 ±1	+5.0 ±0.5	-1.0 ±0.1	211.6	±2

Note: 1 μB = 10^{-6} cm^{-2}

TABLE IV

MICROSCOPIC LATTICE PARAMETERS

As Derived from PDP Experiments with
Seven-Rod Clusters of Natural Uranium Metal

Lattice spacing, inches	18.52	14.00	14.00	12.124	18.52
D_2O purity, %	99.3	99.3	99.75	99.75	99.75
η inner rod	1.311 ±.008	1.293 ±.008	1.292 ±.013	1.307 ±.015	1.311 ±.014
η outer rod	1.319 ±.008	1.299 ±.008	1.298 ±.010	1.309 ±.014	1.317 ±.013
δ inner rod	0.0991 ±.008	0.0981 ±.005	0.1043 ±.013	0.1071 ±.006	
δ outer rod	0.0667 ±.005	0.0612 ±.012	0.0672 ±.008	0.0696 ±.010	
f	--	0.943 ±.01	0.970 ±.01	--	0.946 ±.01
L^2	--	239 ±20	237 ±20	--	468 ±20
ρ	--	0.370 ±.04	0.391 ±.04	0.465 ±.04	0.348 ±.04
p	--	0.909 ±.01	0.894 ±.01	0.891 ±.01	0.914 ±.01

Note: See text for nomenclature

TABLE V

DIMENSIONS OF ZIRCALOY-2 SHEATH TUBING

All dimensions are in inches
Tubes Fabricated by Harvey Aluminum Company

	Outer Sheathing		
	Specified	Average	Range
OD	2.500 +0.015 -0.000	2.508	2.498 - 2.519
ID	-	-	-
Thickness	0.030 ±0.003	0.034	0.030 - 0.037
Ovalness	Within diametral tolerance	0.006	0.000 - 0.016
Bow of 8-ft length	Not specified quantitatively	0.059	0.023 - 0.118

	Inner Sheathing		
	Specified	Average	Range
OD	-	-	-
ID	1.465 +0.015 -0.000	1.471	1.466 - 1.479
Thickness	0.030 ±0.003	0.034	0.032 - 0.035
Ovalness	Within diametral tolerance	0.004	0.000 - 0.012
Bow of 8-ft length	Not specified quantitatively	0.044	0.014 - 0.081

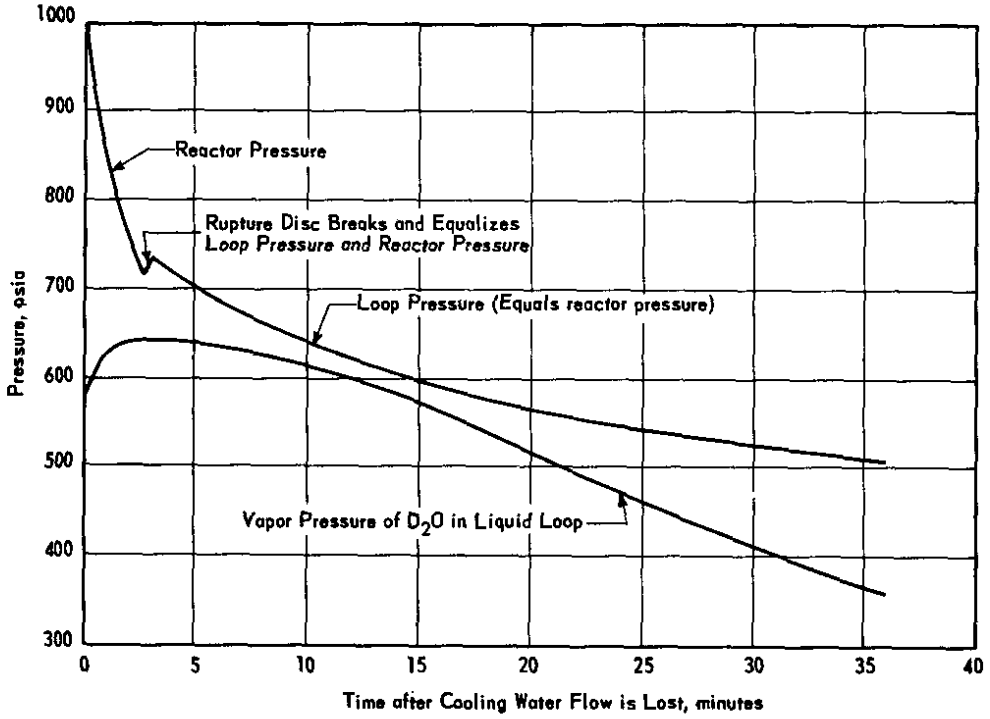


FIG. 1 PRESSURES FOLLOWING LOSS OF COOLING WATER FOR LIQUID LOOP AND REACTOR SCRAM

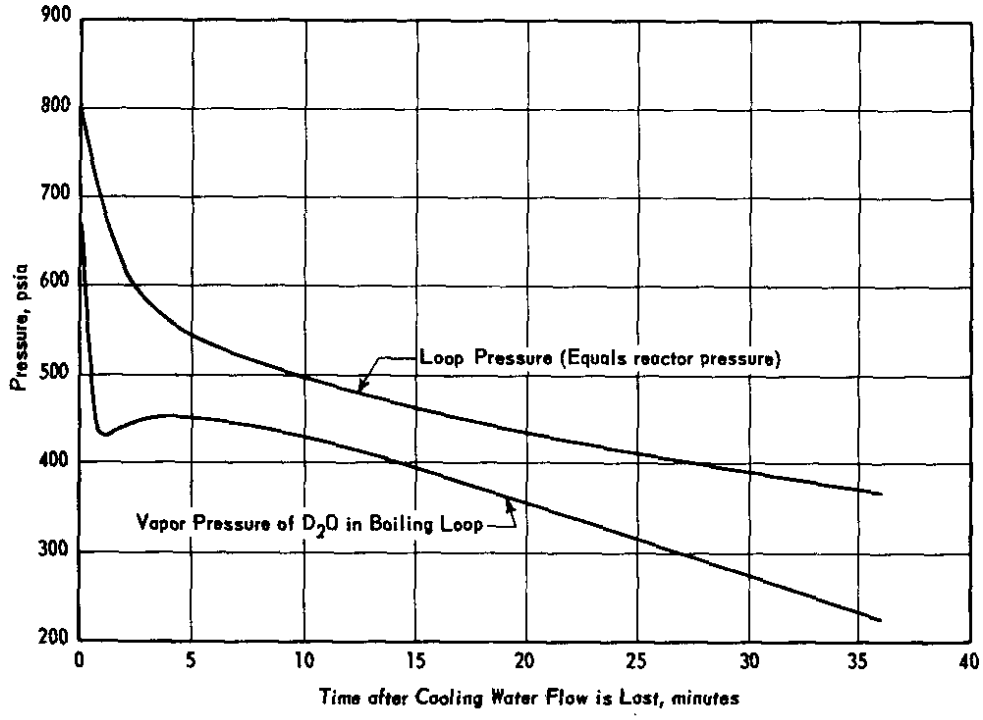
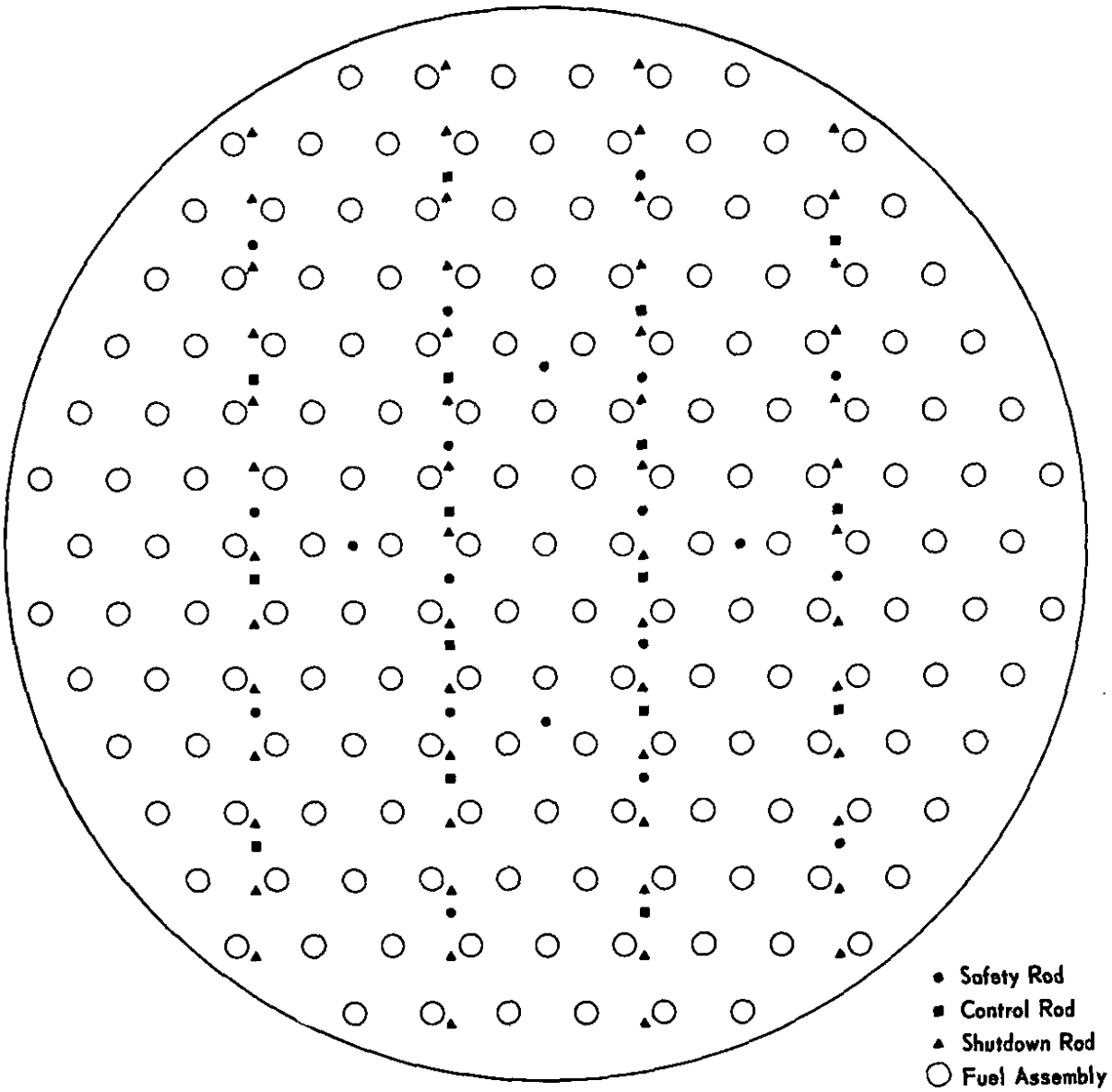


FIG. 2 PRESSURES FOLLOWING LOSS OF COOLING WATER FOR BOILING LOOP AND REACTOR SCRAM



Note: Inside diameter of PDP tank = 194.8 inches

FIG. 3 TYPICAL LATTICE ARRANGEMENT IN PDP
(Fuel at 14-Inch Lattice Pitch)

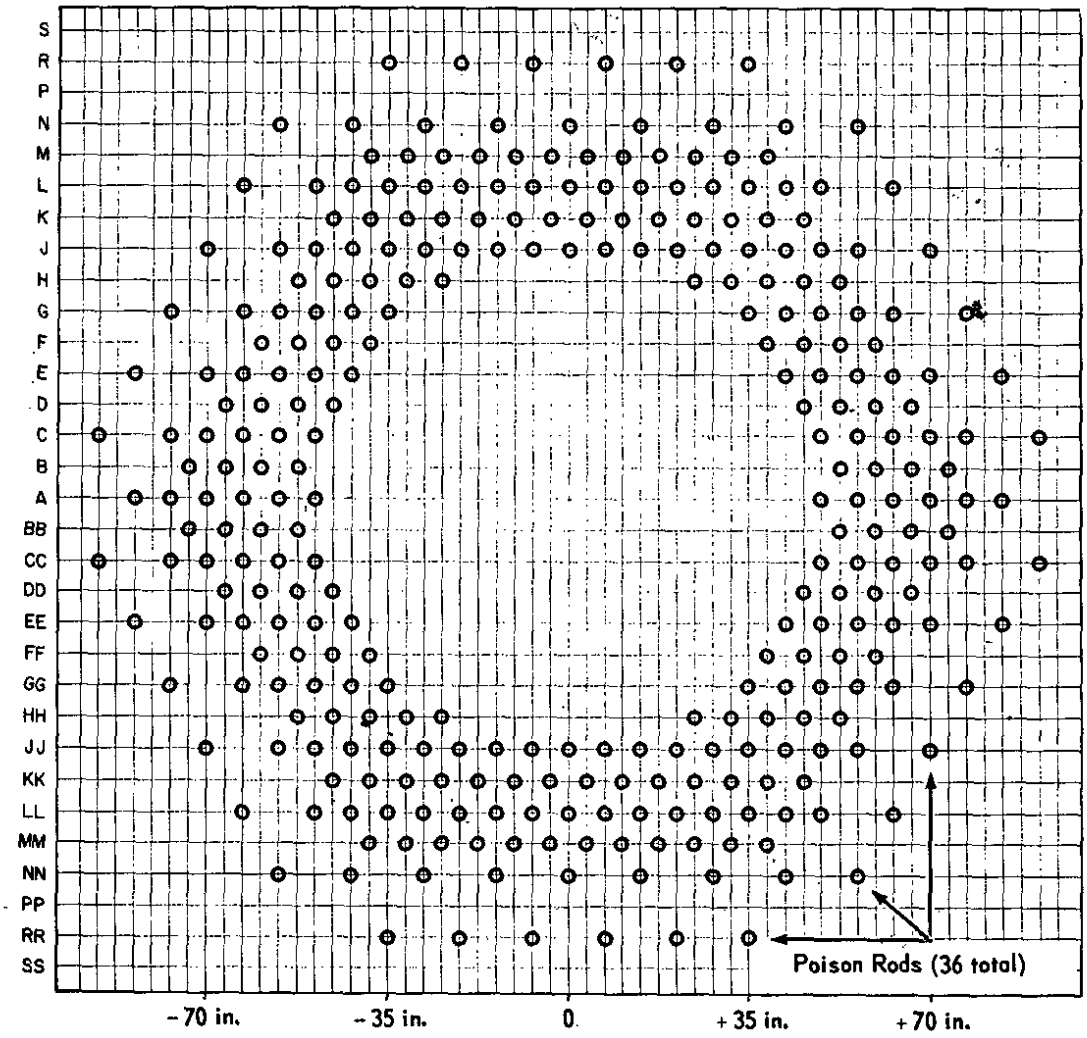


FIG. 4 POISON ROD ARRANGEMENT FOR REDUCED PILE LOADINGS IN THE PDP

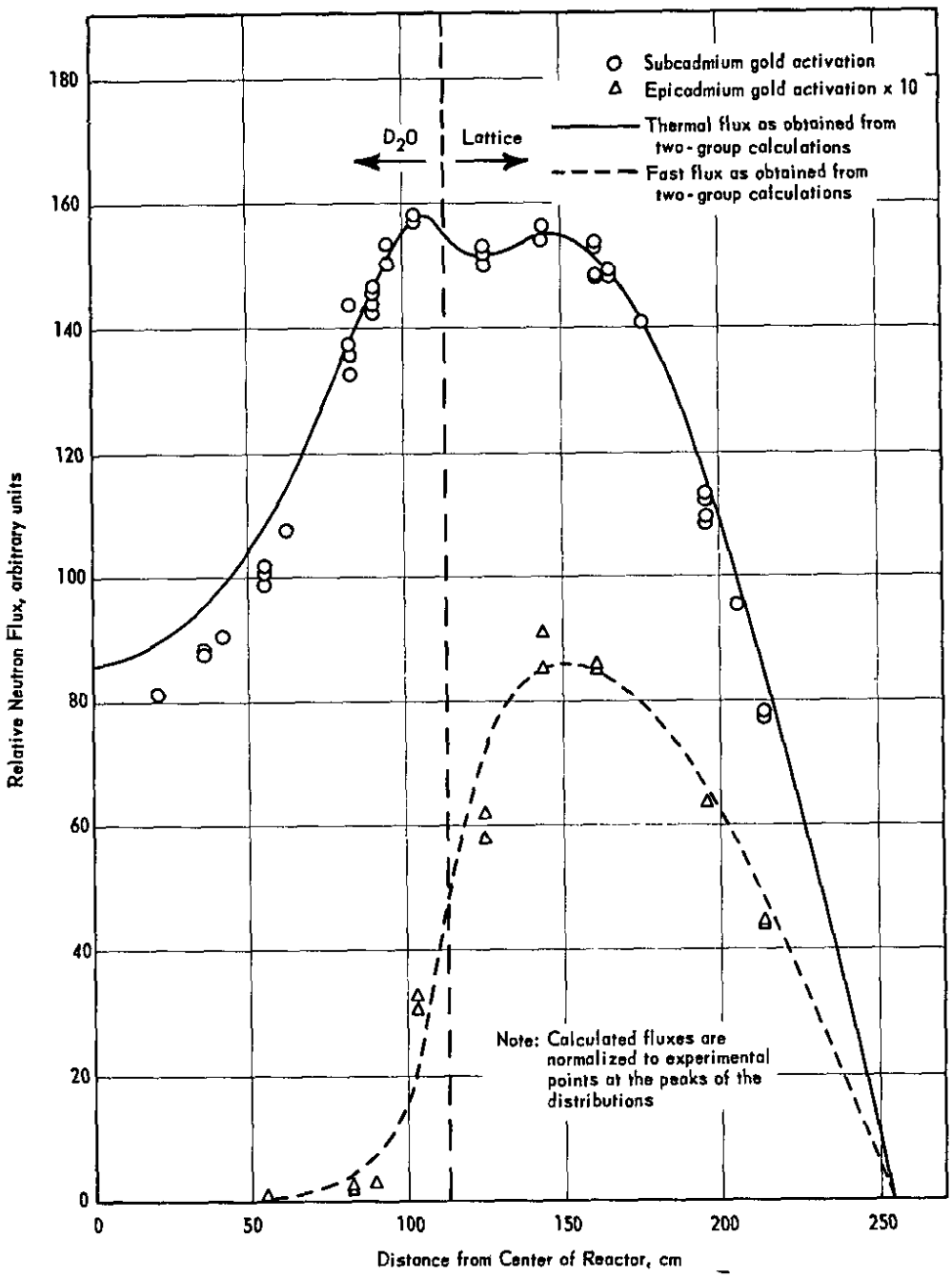


FIG. 5 RADIAL FLUX DISTRIBUTIONS IN PDP WITH LATTICE REMOVED FROM CENTRAL REGION

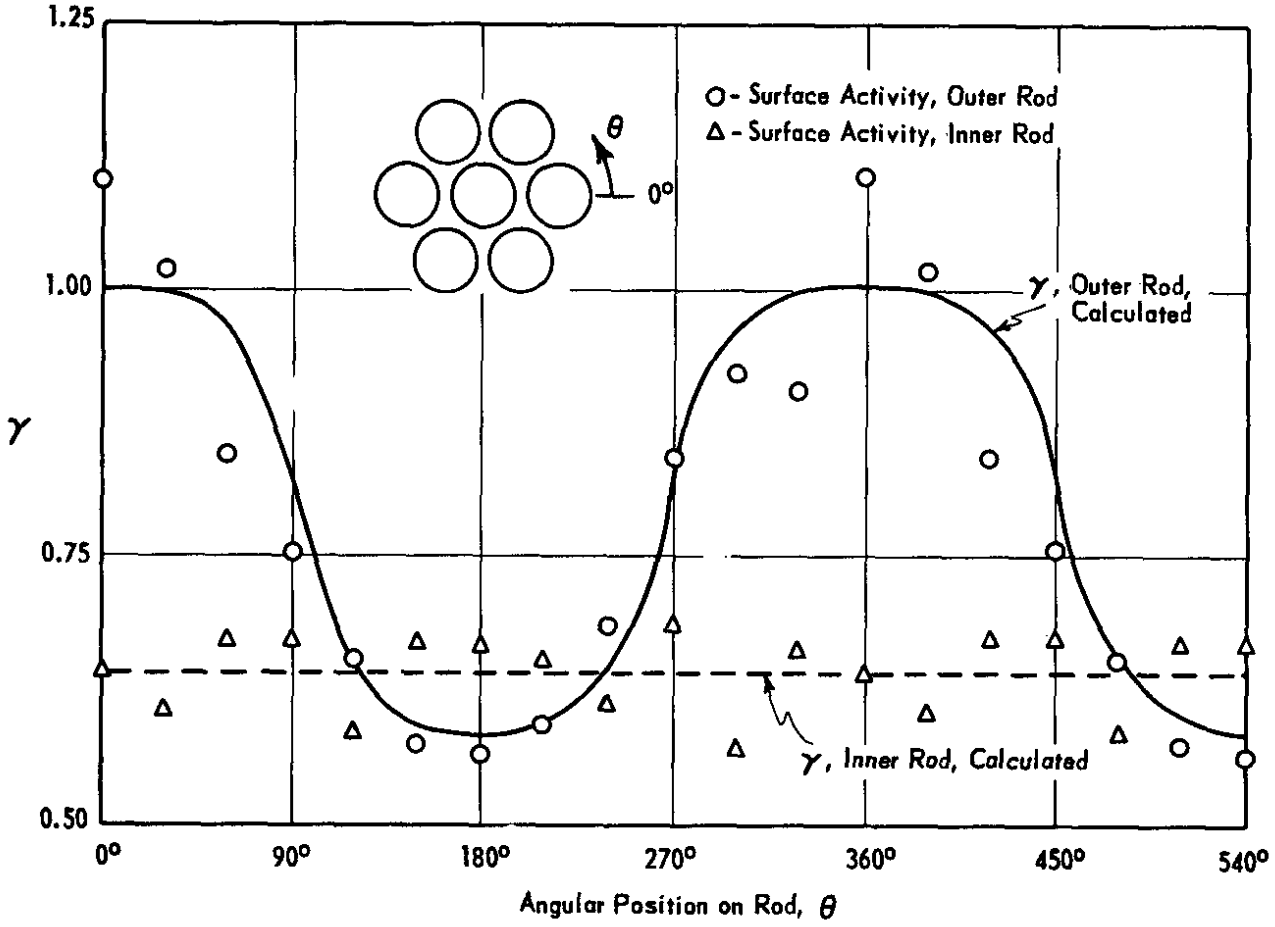
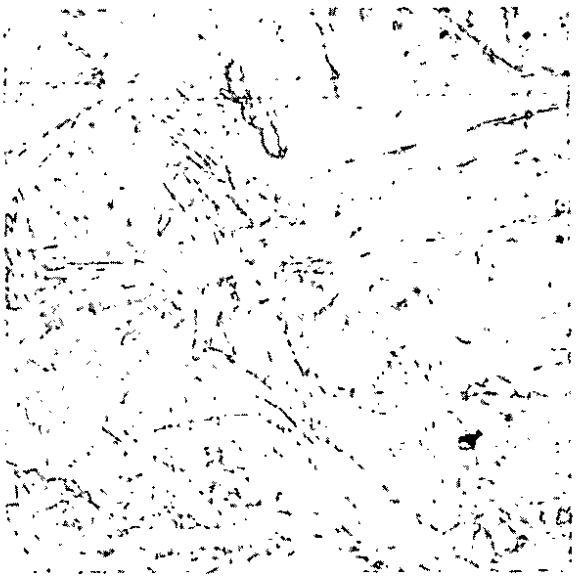


FIG. 6 MEASURED DANCOFF EFFECT IN SEVEN-ROD CLUSTER OF URANIUM METAL
Center-to-Center Spacing of Rods = 1.5 Inches

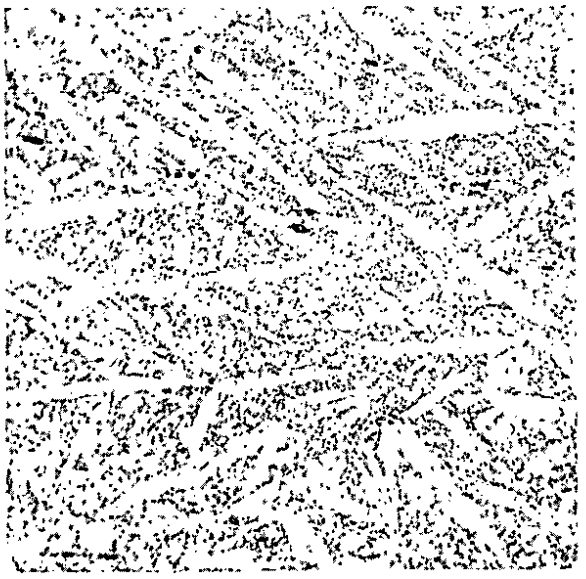
As Quenched

Annealed

Matrix Grain Structure, Attack Polish, Polarized Light, Mag. 150X



As Quenched

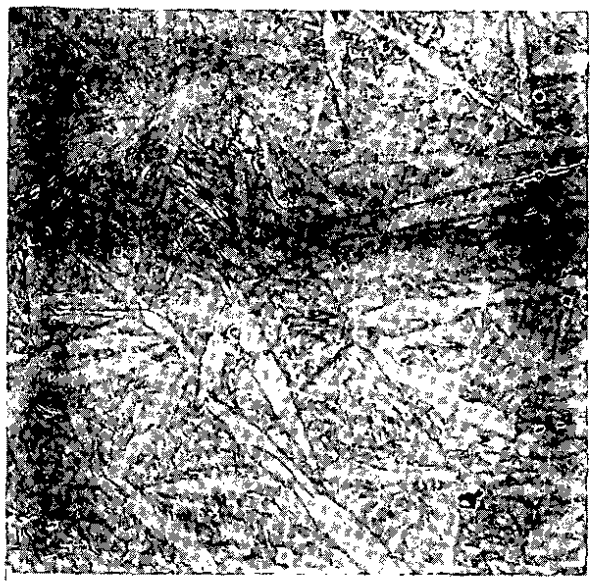


Annealed

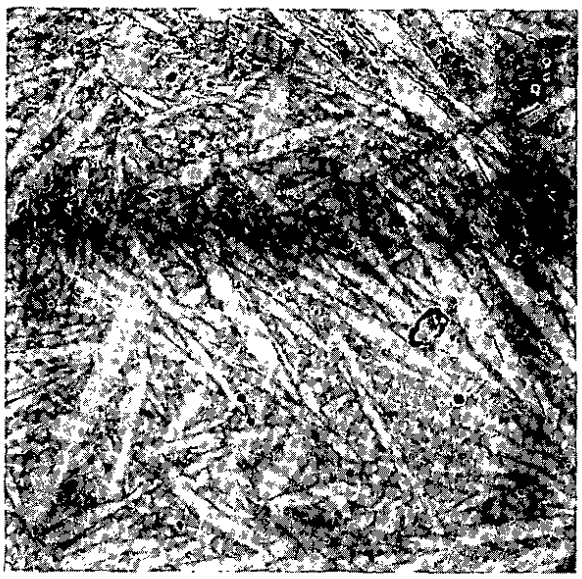
Second–Phase Distribution, Electropolished, Bright Field, Mag. 500X

The gamma–quenched structures were produced by heating in molten salt at 800°C for 10 minutes; the specimens were then water–quenched. The annealed specimens were subsequently heated in salt at 650°C for 15 minutes Note acicular grain structure and precipitation of second phase at grain boundaries on annealing.

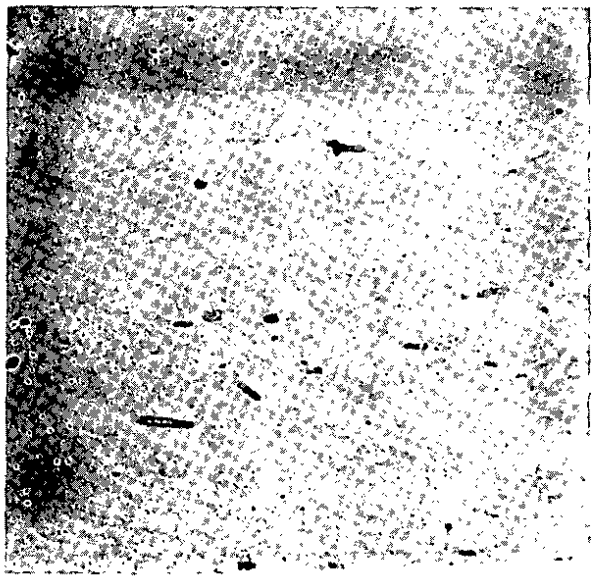
FIG. 7 GAMMA-QUENCHED U-2 WT% Zr ALLOY



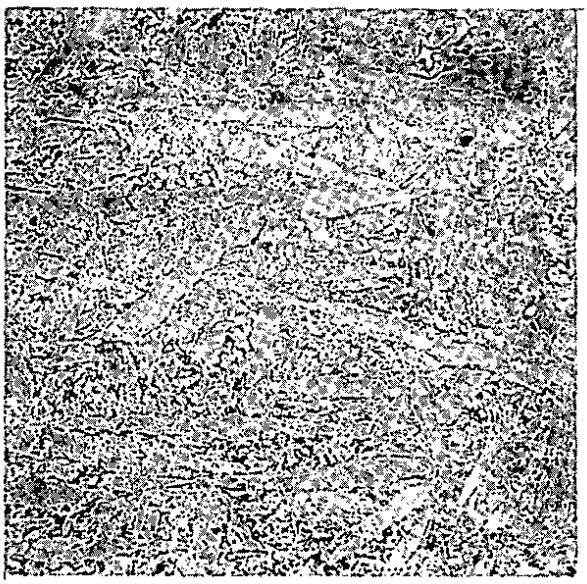
Gamma Quenched, No Anneal



Gamma Quenched, Annealed at 400°C for 20 Hours

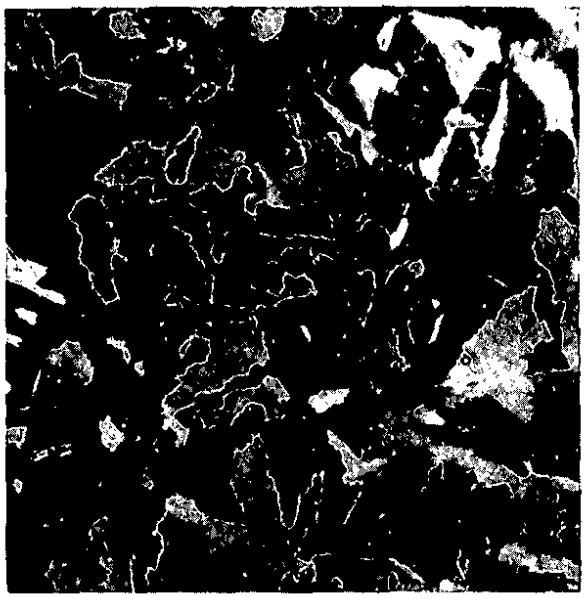


Gamma Quenched, Annealed at 500°C for 20 Hours

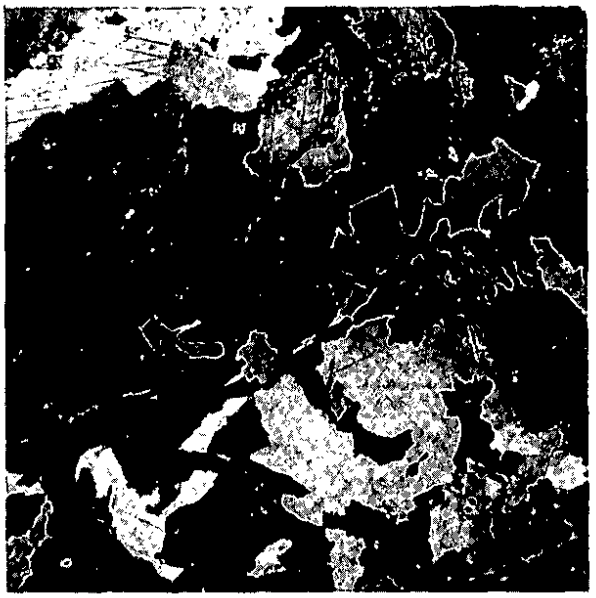


Gamma Quenched, Annealed at 600°C for 20 Hours

FIG. 8 PRECIPITATION OF SECOND PHASE DURING ANNEALING OF GAMMA-QUENCHED SPECIMENS OF U-2 WT% Zr
Electropolished, Bright Field, Mag. 500X

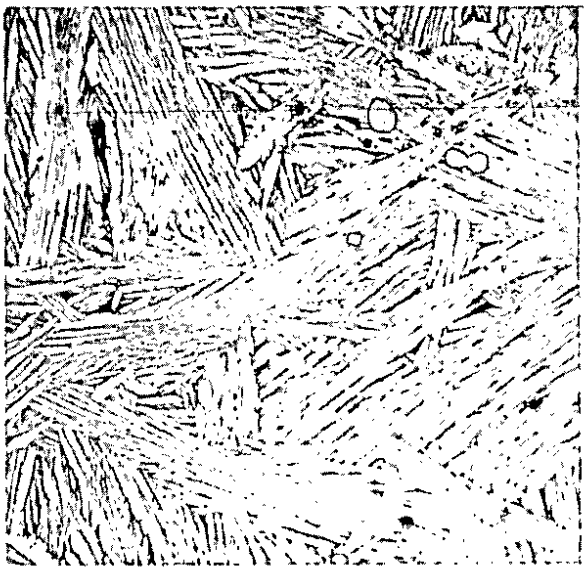


Diffusion Treated

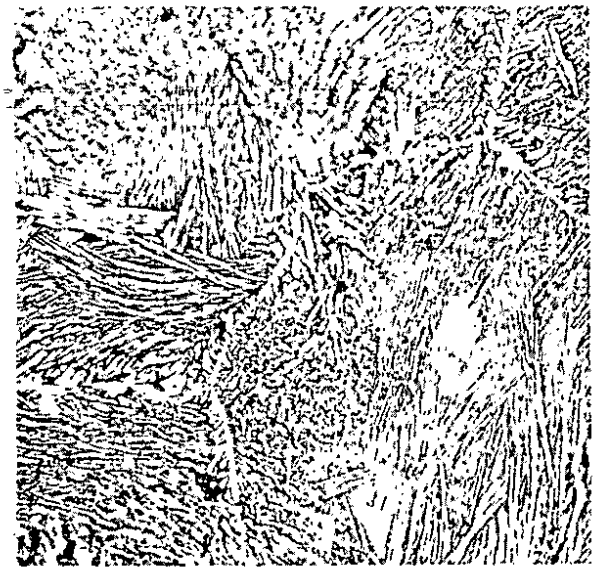


Furnace Cooled

Matrix Grain Structure, Attack Polish, Polarized Light, Mag. 150X



Diffusion Treated

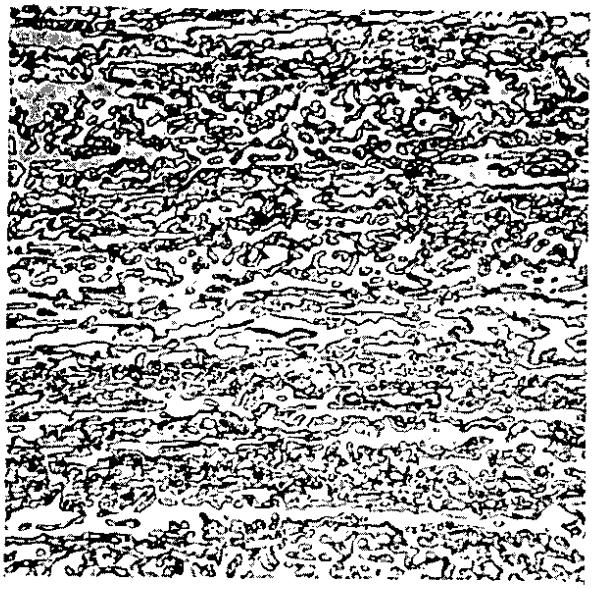


Furnace Cooled

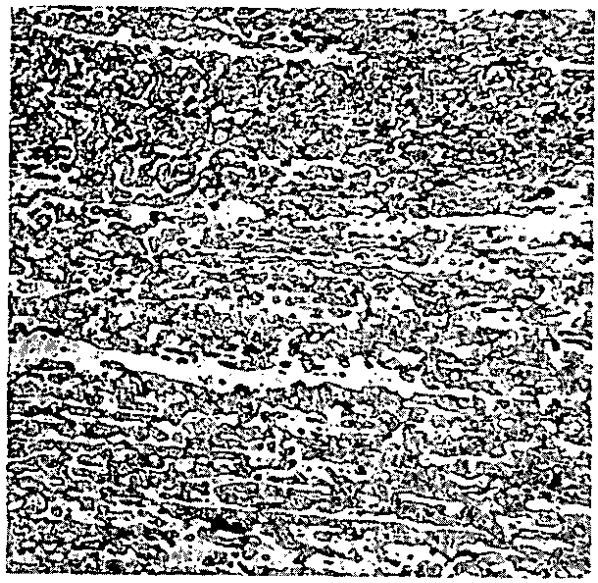
Second-Phase Distribution, Electropolished, Bright Field, Mag. 500X

The "diffusion anneal" consisted of heating the specimen in an evacuated stainless steel chamber for 7 hours at 880°C, then cooling by immersion of the steel chamber in water. Since the specimen was insulated by vacuum, this was a relatively slow cool. The furnace-cooled structures were obtained by heating in molten salt at 800°C for 10 minutes, then slowly cooling in the salt bath to 650°C before water quenching.

FIG. 9 GAMMA-TREATED, SLOW-COOLED U-2 WT% Zr ALLOY

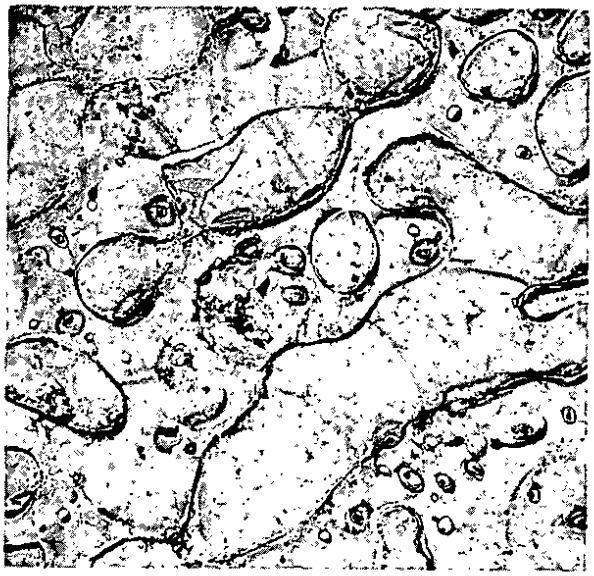


As Quenched

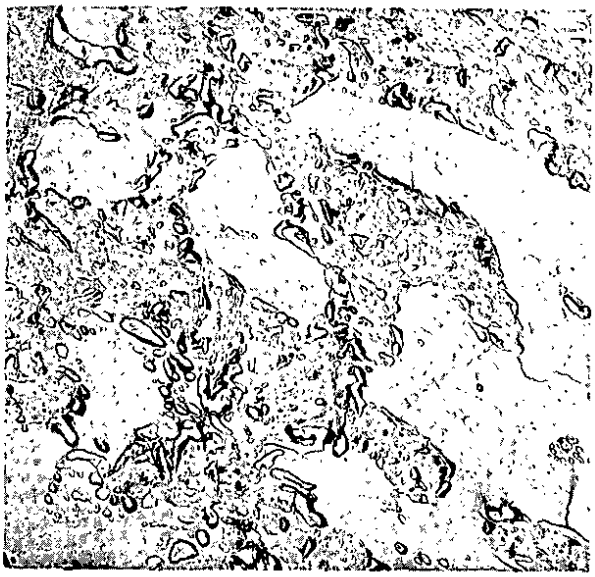


Annealed

Optical Photomicrograph, Electropolished, Bright Field, Mag. 500X



As Quenched

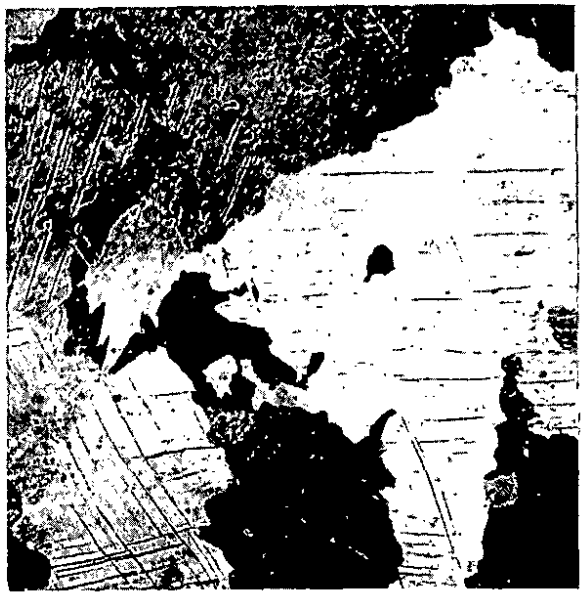


Annealed

Electron Micrograph, "Faxfilm"–Carbon Replica, Mag. 5100X

The as–quenched structures were produced by heating at 720°C for 10 minutes, then water quenching; the annealed structures were produced by subsequent heating at 650°C for 15 minutes. In the beta–quenched specimens, note the large volume second–phase constituent, which decomposes on alpha annealing.

FIG. 10 BETA-QUENCHED U-2 WT% Zr AND BETA-QUENCHED, ALPHA-ANNEALED U-2 WT% Zr ALLOY



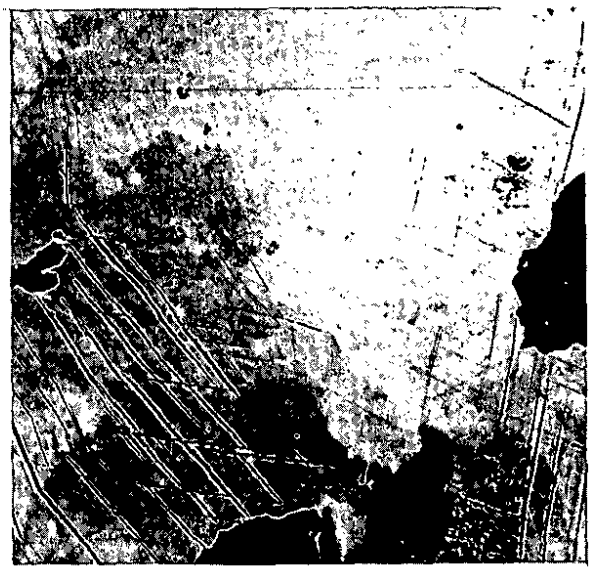
Matrix Grain Structure

Attack Polish, Polarized Light, Mag. 150X

Second–Phase Distribution

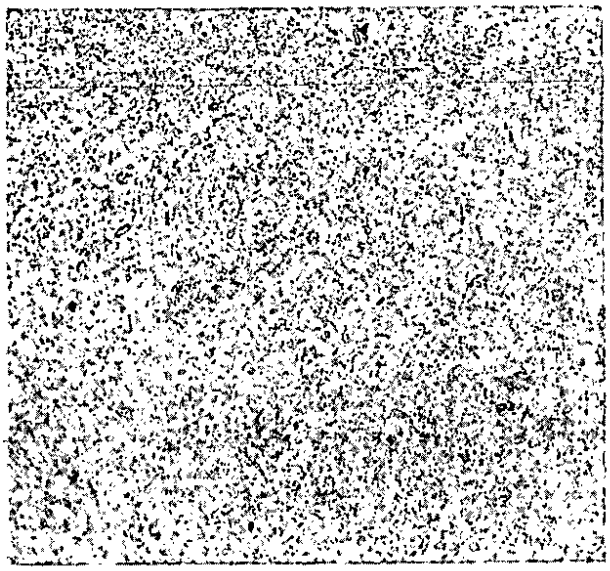
Electropolished, Bright Field, Mag. 500X

Gamma-treated at 800°C for 10 minutes; furnace-cooled to 680°C; held 20 hours; air-cooled



Matrix Grain Structure

Attack Polish, Polarized Light, Mag. 150X



Second–Phase Distribution

Electropolished, Bright Field, Mag. 500X

Gamma–treated at 800°C for 10 minutes; water–quenched; heated at 680°C for 20 hours; air–cooled

Agglomeration of the second phase was produced by heating at 680°C for 20 hours, following pretreatment as indicated. Note large matrix grain size of "spheroidized" specimens, and dependence of second–phase distribution on pretreatment.

FIG. 11 STRUCTURE OF "SPHEROIDIZED" U-2 WT% Zr