DP - 445

Reactors - Power

AEC Research and Development Report

HEAVY WATER MODERATED POWER REACTORS

PROGRESS REPORT

November 1959

Technical Division

Wilmington, Delaware

December 1959

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

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HEAVY WATER MODERATED POWER REACTORS

Progress Report
November 1959

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Power Reactor Studies
Wilmington, Delaware

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ABSTRACT

At the end of November, 20% of the construction and 80% of the firm design of the Heavy Water Components Test Reactor (HWCTR) were complete. Proof testing of various seals and mechanisms for the HWCTR continued satisfactorily. Further analyses are given of the transient behavior of the HWCTR isolated coolant loops and of the experimental data on the nuclear effects of hot moderator. The results of additional fabrication and irradiation tests of uranium metal and uranium oxide are recorded. The manufacture of tubular metallurgical joints between Zircaloy and stainless steel is also reported.

CONTENTS

Page

LIST OF TABLES

LIST OF FIGURES

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 study of heavy-water-moderated, natural-uranium-fueled power reactors. The current effort is divided into two main categories: (1) the research and 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 studies required for developing the technology of a full-scale D_2O-moderated power reactor plant. Earlier reports on this study are:

DP-232	DP-315	DP-405
DP-245	DP-345	DP-415
DP-265	DP-375	DP-425
DP-285	DP-385	DP-435
DP-295	DP-395	

Progress for the month of December will be reported in DP-455.

SUMMARY

At the end of November, about 20% of the construction and 80% of the firm design of the Heavy Water Components Test Reactor were complete. The progress of construction during November is shown in Figures 1 and 2.

Safeguards analyses of the isolated coolant loops of the HWCTR were made this month. The transients of temperature and pressure in the liquid-D_2O-cooled loop following a reactor scram were computed. The results, which are plotted in Figures 7, 8, and 9, show that satisfactory operation of the loop can be achieved with the present loop design. The behavior of the boiling-D_2O-cooled loop following a reactor scram was also examined. Changes in the original design of the loop were required in order to avoid vapor binding of the pump and loss of coolant to the test fuel assembly during a reactor scram. In addition to these analytical studies of the loops, it is planned to build a mockup of the bayonet and discharge piping of the boiling-D_2O-cooled loop to obtain experimental assurance that undesirable vibration will not occur as a result of the mixed flow of steam and water in the loop.

Proof testing of various reactor components for the HWCTR continued satisfactorily during this report period. A safety rod and control rod drive package, the prototype gripper mechanism of the fuel transfer coffin, and candidate seals for the gasketed joints on monitor pin, instrument rod, and control rod closures were cyclically tested. No

damage, no abnormal wear, and no excessive leakage were observed. Testing of a pump shaft seal also proceeded satisfactorily without excessive leakage under conditions expected at the HWCTR pump shafts.

The fabrication of irradiation specimens of uranium metal continued at Nuclear Metals, Inc. Two 2-inch-OD Zircaloy-clad tubes with alloy cores are being evaluated as irradiation candidates. The core of one of these tubes is U - 1 w/o Si and that of the other is U - 1.5 w/o Mo. Fabrication of a tube with a core of U - 0.2 w/o Al was begun. Further studies were made of alloys that may retain the desired creep-resistant beta phase during irradiation. An extruded specimen of U - 0.3 w/o Cr - 0.3 w/o Mo retained a fair proportion of beta phase when aged for one hour at 500°C, but a similar specimen of U - 0.3 w/o Cr reverted almost completely to the alpha phase in one-half hour at 500°C.

Several tubes of fused uranium oxide with cladding of stainless steel were prepared by the Savannah River Laboratory for irradiation tests in a Savannah River reactor. These tubes were fabricated by a cold swaging process that is being developed as a potentially low cost route to oxide fuel elements that are suitable for use in D_2O-moderated power reactors. The UO_2 density in the tubes averaged only 88.3% of theoretical, as compared to an expected value of 90 to 91%. The origin of this discrepancy is not yet known. An experimental study of the effect of fines on the swaged density of crushed, fused UO_2 showed that the maximum swaged density is achieved by removing some of the finer particles in pulverized UO_2. A maximum swaged density of 92% of theoretical was obtained with the most favorable distribution of particle size.

Exploratory tests at Nuclear Metals, Inc., indicated that a strong metallurgical bond can be formed between stainless steel and Zircaloy. In a pressure tube reactor, use of a reliable metallurgical joint between the Zircaloy tubes and their associated coolant distributors can result in significant cost reductions and mechanical design simplifications. Tubular joints about 2 inches in diameter are being evaluated for corrosion resistance and mechanical performance. Samples of metallurgical joints fabricated by other techniques were procured from General Electric Company for evaluation purposes.

Two power reactor fuel tubes of uranium metal were examined after irradiation to modest exposures at relatively low temperature in a Savannah River reactor. One tube had a core of unalloyed uranium and was clad with 0.030 inch of Zircaloy-2. The core of the other tube, which was clad with 0.015 inch of Zircaloy-2, was U - 2 w/o Zr. The outside diameter of each tube increased 0.005 to 0.010 inch during irradiation, and surface roughening occurred on both tubes. Bumps as high as 0.024 inch and 0.010 inch developed on the alloy tube and the unalloyed tube, respectively. Measurement of inside diameters must await removal of the inner housing tubes from the fuel; at present, the inner housings are stuck within the fuel tubes.

Calculations were made of the flow - head loss characteristics of fuel assemblies that are cooled by forced flow of boiling water. The results of these calculations, which provide a preliminary indication of conditions under which flow instabilities might occur, are presented in Figures 20 through 24. An electrically heated flow loop is being assembled for experimental confirmation of the calculations.

DISCUSSION

I. HEAVY WATER COMPONENTS TEST REACTOR (HWCTR)

The HWCTR is a test reactor in which numerous fuel elements will be irradiated under conditions of temperature, pressure, and neutron flux that are typical of those expected in D_2O-moderated power reactors. A description of the reactor was presented in DP-383[1] and in earlier progress reports. Construction of the facility was authorized by the Atomic Energy Commission in November 1958. The goal for startup is early 1961. The total cost of the test facility, which is designed for a thermal output of 61 MW, is estimated to be $8,700,000, including $1,000,000 for two isolated coolant loops in which special fuel assemblies will be irradiated. Progress during the month of November on the HWCTR design, construction, and supporting experimental work is summarized in this section.

A. STATUS

1. Construction

At the end of November, construction of the HWCTR was approximately 20% complete. The progress of construction during November is shown in the photographs, Figures 1 and 2. The outer concrete wall of the containment building was poured in a 28-hour continuous operation. All the forms have since been stripped from the wall. No porosity was found in the shell. However, about two dozen fine vertical cracks were detected around the lower part of the shell in the area behind the spent fuel basin. It is postulated that the fine cracks resulted from the restraint placed upon the shrinking concrete shell by the base slab, the -37 foot slab, and the heavy walls of the spent fuel basin. After the concrete shell is post-tensioned, the inner surface of the wall will be sealed by application of "Liquid Tile"* reinforced by a glass fiber cloth.

Preparations were begun for the pouring of the zero-level slab early in January. Installation of shoring under the slab is complete, and forms for the bottom and outer periphery of the slab are in place. Forms for the openings in the slab, sealing inserts, embedded piping, and embedded conduit are now being installed.

A new analysis of the time schedule for the construction of the HWCTR was begun following the halt in the steel strike. The purpose of the analysis is to determine how much delay was caused by the strike and what is the best estimate of the startup date in view of the current situation. Results of this analysis are expected early in December.

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(1) DP-383, Preliminary Hazards Evaluation of the Heavy Water Components Test Reactor, D. S. St. John, et al., May 1959

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

2. Design

Firm design of the HWCTR was approximately 80% complete at the end of November. Major design effort during this month was directed at completion of the design and procurement of electrical equipment and instrumentation. Detail design for the fuel transfer coffin was also nearing completion. Major items for which purchase orders were placed included the helium compressor, control room instrument panels, control and safety rods, the platform for the control and safety rod drives, the pumps for the spent fuel basin, the filter for the spent fuel basin, and the bridge and trolley for the fuel transfer coffin. A request for quotations was also submitted to vendors for the irradiated fuel element shipping cask.

B. EFFECT OF MODERATOR TEMPERATURE

Physics measurements on the effects of moderator temperature in a mockup of the HWCTR lattice were completed in the PSE. The PSE is a subcritical experimental facility that can operate with D_2O temperatures as high as 215°C. A description of the mockup was given in DP-415. Preliminary high temperature measurements were reported in DP-425. In the two articles that follow, the results of these experiments are summarized and then applied to the prediction of the temperature effects in the HWCTR.

1. Measurements in the PSE Mockup of the HWCTR

Vertical bucklings were measured in the PSE mockup of HWCTR lattices as a function of moderator temperature between 23 and 215°C at moderator purities of 99.0 and 99.7 mol % D_2O. Vertical bucklings were also measured as a function of the number of black control rods inserted in the lattice at room temperature. Three typical HWCTR lattices were studied. For Case I, the lattice contained six black rods in the control ring and none in the central control cluster. Six black rods were present in the control ring and three black rods in the central control cluster for Case II. Four black rods in the control ring and three black rods in the central cluster comprised Case III.

The results of the measurements for Cases I, II, and III are shown in Figures 3, 4, and 5, respectively. All bucklings were normalized to zero buckling for the initial lattice at 20°C. The change in vertical buckling between 20 and 240°C was then expressed in terms of the number of compensating black rods, called "rod equivalents", that would have to be added at room temperature to bring about the same decrease in vertical buckling. These rod equivalents varied from 3.8 to 4.1 rods with an average value of about 4.0. The smooth curves in Figures 3, 4, and 5 have experimental errors associated with them. A rough idea of the size of the experimental errors in vertical buckling as a function of temperature can be obtained from Figure 5 of DP-425. There is undoubtedly an additional error in the curves of Figures 3, 4, and 5 because of the assumption that the change in vertical buckling is an almost linear function of the total length of control rods added to the control ring. However, since the number of rod equivalents is nearly an integer, these errors in interpolation are not important. Combination of these errors with those involved in the extrapolation of buckling data from 215 to 240°C and in estimating the control rod worths at room temperature results in a total error of about ±0.25 rods at 240°C. Thus, the number of rod equivalents for the mockup lattices is 4.0 ±0.25 at 240°C.

The effect of removing 0.7 mol % H_2O from the moderator was small. The absolute buckling at room temperature increased by 20 μB for Cases II and III with three control rods in the central cluster; the increase was 55 μB for Case I with no rods in the central cluster.

2. Prediction of Temperature Effects in HWCTR from PSE Experiments

Effects of changes in moderator temperature in the HWCTR would be expected to be somewhat different from those measured in the PSE mockup because of the difference in reflector thickness and because the PSE mockup was, of necessity, subcritical. Information previously obtained from the PDP mockup of the HWCTR, from two-group calculations, and from the PSE experimental results was combined to translate the PSE results into predictions for the HWCTR. It was calculated that in order to compensate for the temperature rise to 240°C in the HWCTR, 3.65 ±0.4 black control rods would have to be withdrawn. Thus, even though the neutron leakage from the PSE is 30% greater than that from the HWCTR, the temperature effects expressed in terms of the number of equivalent rods are very nearly the same.

C. ISOLATED COOLANT LOOPS

Two isolated coolant loops in which special test assemblies can be irradiated will be installed in the HWCTR. One loop, which can operate up to about 500 psi above the reactor pressure, will be cooled by liquid D_2O. The other loop, which will operate at about the same pressure as the reactor vessel, will be cooled by boiling D_2O. Reported below are (a) recent modifications in the design of the boiling loop, (b) the purposes of a mockup test for the boiling loop, and (c) analyses of the transients that may be experienced in the liquid D_2O loop.

1. Design of the Boiling-D_2O-Cooled Loop

Modifications are being made to the design of the boiling-D_2O-cooled loop in order to avoid the possible loss of coolant to the loop fuel assembly during a reactor scram. This possibility existed because the pressures in the reactor and loop systems were equalized by a line that connected the outlet of the loop condenser to the gas space at the top of the reactor (See Figure 18, DP-415). During a reactor scram, when the pressure in the reactor vessel reduces at the rate of about 5 psi per second for 35 seconds, the consequent lowering of the pressure in the loop causes continued flashing in the suction line of the loop pump, probable vapor binding of the pump, and the loss of coolant to the test assembly. If the loop were completely separated from the reactor, the loop pressure would decrease more rapidly and a more severe situation would be encountered. The rapid reduction in heat generation that accompanies the scram would permit the large volume of steam in the condenser to be condensed rapidly, causing an even more rapid reduction in loop pressure than if the reactor and loop pressure were equalized. Vapor binding of the pump, with its detrimental consequences, would follow rapidly.

In order to avoid these undesirable possibilities, it is now proposed to quench the D_2O steam and subcool the D_2O liquid in a quencher immediately downstream of the in-pile bayonet tube. Quenching will be accomplished by injection of cool D_2O that is drawn from the pump discharge line through a heat exchanger. This design change will result in a great reduction in the volume of steam in the system and the maintenance of subcooled liquid D_2O in the surge tank of the loop. The surge tank probably will be pressurized by the reactor helium blanket system.

2. Mockup of the Boiling-D_2O-Cooled Loop

A mockup of the bayonet and discharge piping of the boiling-D_2O-cooled loop of the HWCTR will be made by the Savannah River Laboratory. The primary purpose of the mockup is to determine whether or not potentially damaging vibration to the bayonet will occur as a result of the mixed flow of steam and water in the boiling loop. Measurements will be made of the amplitudes and accelerations of vibration over the following range of variables:

Pressure, psia	515 - 1015
Exit steam quality, %	10 - 30
Subcooling, °C	0 - 78

In addition to the observations on vibration, pressure drop data for the two-phase flow through the bayonet and the piping will be collected. These data are required because of the uncertainties in the methods for predicting the two-phase pressure drop in unusual configurations such as these.

A small-scale quenching test has been proposed to guide the design of the full-scale quencher mentioned in the preceding article. If there is any doubt of the acceptability of the full-scale design, a mockup could be readily incorporated into the bayonet mockup. The purpose of these quenching tests is to determine whether or not the injection of cooled water into steam would introduce excessive noise or vibration.

3. Transient Analyses of the Liquid-D_2O-Cooled Loop

As part of the safeguards analysis of the HWCTR, the temperature and pressure transients in the liquid-D_2O-cooled test loop following a reactor scram were computed. It was concluded from these calculations that satisfactory operation of the loop would be achieved during a reactor scram if the volume of gas in the loop surge tank were about 55 gallons and if no change from steady-state operation were made in the amount of D_2O bypassing the loop heat exchanger. The following quantities were varied in the calculations: (a) the volume of gas, i.e., He and D_2O, in the loop surge tank, (b) the fraction of the total D_2O flow bypassing the loop heat exchanger, and (c) the steam valve opening on the HWCTR steam generators. A flow diagram of the system is given in Figure 18 of DP-415. The method by which the transients were computed is very similar to the one used for analyzing the transients in the primary coolant system of the HWCTR (DP-245, pp. 33-37).

a. Steady-State Conditions

The steady-state temperatures that would be achieved by the D_2O in the loop at a reactor power of 50 MW are shown in Figure 6 as functions of the fraction of the D_2O flow that bypasses the loop heat exchanger. For design flows on the H_2O side of the heat exchanger and with no bypassing of the D_2O flow, the temperatures at the inlet and outlet of the test fuel assembly are 172 and 192°C, respectively. In order to raise these temperatures to the design values of 250 and 274°C, respectively, 87.35% of the total D_2O coolant flow in the loop must bypass the heat exchanger.

b. Transients Following a Reactor Scram

Typical transients in loop temperatures and pressure following a reactor scram are shown in Figures 7 and 8. In Figure 7 loop temperatures and reactor power are shown as functions of time after scram. Loop pressures, reactor pressure, and the difference between these two pressures following a scram are plotted in Figure 8. In computing these curves it was assumed that (1) the initial volume of the gas in the loop surge tank is 55 gallons, (2) the steam valves on the reactor boilers are left open at their 50-MW steady-state setting of 22.2 square inches, and (3) 87.35% of the total loop coolant constantly bypasses the loop heat exchanger.

The reasons for recommending a gas volume of about 55 gallons in the surge tank of the liquid-D_2O-cooled loop prior to a reactor scram are discussed in the remainder of this article. The seals between the gas pressurizing systems of the loop and the reactor are designed to rupture (a) when the pressure in the loop is 700 psi higher than the pressure in the reactor, or (b) when the pressure in the reactor exceeds the pressure in the loop by 200 psi or more. In normal operation the loop pressure will be higher than the reactor pressure by 0 to 500 psi. During a reactor scram the reactor pressure decreases at about 5 psi per second for about 35 seconds and then the rate of reduction becomes slower. During a scram it is therefore possible for the seals to rupture if the pressure differential between the reactor and loop is not carefully controlled or limited. A desirable condition would be one that results in a constant differential pressure between the loop and the reactor following a reactor scram.

Three transients in differential pressure following a reactor scram are shown in Figure 9. These three transients were computed for a constant opening of 22.2 square inches in the steam valves of the boilers of the main system. The upper curve was calculated for an initial differential pressure of 485 psi, an initial volume of 110 gallons of gas in the loop surge tank, and for the amount of D_2O bypassing the loop heat exchanger increasing to 99% from an initial value of 87.35%. Of all the cases computed, these conditions resulted in the largest differential pressure between the loop and reactor. The lower curve, computed for an initial gas volume of 22 gallons in the surge tank and for a constant bypass of 87.35% of the total D_2O flow, shows the lowest differential pressure of any of the cases considered. The transient plotted in the middle curve of Figure 9 represents the most nearly constant pressure differential following a scram. This transient results when the reactor is scrammed and there is an initial volume of 55 gallons of gas in the loop surge tank and a constant 87.35% of the D_2O bypassing the loop heat exchanger.

The results of all the computations of the differential pressure transients are summarized in Table I. The first column shows the fraction of loop coolant that bypasses the loop heat exchanger; this quantity was either held constant or increased from its initial value as indicated at the beginning of the scram. The second column lists the initial volume of gas in the loop surge tank. The third column shows the setting of the boiler steam valves, which for the calculations were held constant at the 50-MW steady-state opening of 22.2 square inches or throttled as indicated at the beginning of the scram. The fourth column lists the pressure differentials between the loop and the reactor that result 35 seconds after the beginning of the scram.

The data in Table I indicate that the differential pressure transient may also be controlled by throttling the steam valves on the boilers following a reactor scram. Complete closure of the steam valves keeps the reactor pressure high, because the reactor then cools more slowly, and effectively reduces the differential pressure between the loop and reactor. However, complete closure of the valves would cause the temperature of the H_2O in the boilers to rise from 190°C to about 218°C in 35 seconds; the accompanying increase in boiler pressure would be from 182 psi to 322 psi. Throttling of the steam valves, to an opening of 2.0 square inches, is almost as effective as completely closing the valves. No such throttling action is required, however, if the initial gas volume in the loop surge tank is adjusted to 55 gallons and if a constant 87.35% of the loop D_2O is bypassed around the loop heat exchanger.

D. COMPONENTS TESTING

1. Rod Drive Packages

Prototypes of a safety rod drive package and a control rod drive package have been under test at Alco Products, Inc., the consultants on the design and the fabricators of the prototype packages.

Testing of the prototype safety rod drive package was completed on November 2. Final testing at a pressure of 1500 psig and 285°C included 500 scram cycles and 1000 cycles of normal up-and-down drive. No damage or abnormal wear was observed and operation was satisfactory throughout the test.

Testing of the prototype control rod drive package began early in November. The drive package was operated through 100 up-and-down cycles at 1500-psig pressure and room temperature. The temperature was then raised to 285°C. At the end of November approximately 2500 of the projected total of 3000 cycles of operation had been completed satisfactorily.

2. Prototype Gripper Mechanism

Testing of the prototype gripper mechanism of the transfer coffin for irradiated fuel continued at the Savannah River Plant. A total of 50 cycles of the gripper mechanism over its full vertical travel was completed. Testing is in progress on a new type of hold-down device for the fuel housing tubes. The new device was designed as a possible replacement for the bulky hold-down mechanism presently included in the gripper design. A hold-down device is necessary in order to keep the housing tube in place while a fuel element is being lifted from within it. The fuel cooling system for the transfer coffin has also been attached to the prototype gripper mechanism and evaluation of the system is now in progress.

3. Seal Leakage

a. Cyclic Testing of Seals for Vessel Openings

The program for testing candidate seals for the various openings in the HWCTR pressure vessel continued with cyclic testing of the gasketed joints on monitor pin, instrument rod, and control rod closures. The results of 100-cycle tests at a maximum pressure and temperature of 1000 psig and 250°C were recorded in earlier reports. This month, cyclic tests of these seals were started at a maximum pressure of 1500 psig with 250-260°C water. As in the lower pressure tests, during a normal three-hour cycle, the seal is maintained at maximum conditions for one hour; the remainder of the cycle is spent in returning to atmospheric conditions and then repressuring and reheating the sealed fluid. No data from these recent tests are available at present.

b. Pump Seals

A mechanical seal that closely resembles the seal design presently proposed for the shafts of the main circulating pumps of the HWCTR is currently under test at the Bingham Pump Company. The test seal operated for more than 1000 hours with the shaft rotating at 1800 rpm at temperatures and pressures that approximate those expected at the seals of the HWCTR pumps. Leakage from the seal to the leak collection system was quite steady and ranged from 35 to 50 gallons per year.

4. Flow Tests

No damage was observed on the HWCTR test fuel assembly after 103 days, and on the new shield muff with straight flow passages after 41 days, in flow tests in neutral deionized water at 260°C. As discussed in DP-425 and earlier reports, the test fuel assembly consists of a Zircaloy-2-clad, natural uranium fuel tube, surrounded by a straight-ribbed housing tube of Zircaloy-2. After inspection of the assembly the test was resumed.

II. TECHNOLOGY OF FULL-SCALE REACTORS

A. REACTOR FUELS AND MATERIALS

1. Uranium Metal Tubes for Irradiation Tests

a. Unalloyed Uranium

Preparatory to fabrication of two unalloyed uranium tubes of 3% enrichment for irradiation testing in the Vallecitos boiling water reactor, four prototype tubes of natural uranium were extruded early in November. The billet materials, billet design, and extrusion procedures were based generally upon the results of examination of an earlier group of five full-size tubes (DP-395 et. seq.) and a more recent series of reduced-size experimental tubes. The primary exception, as noted last month, was the decision to use a process for refining the uranium grains. This decision was made after cracking was observed in beta treatment of a full-size core casting. The four prototype tubes differ from one another with respect to end shape design, to increase the likelihood that one satisfactory design will be available when the enriched tubes are extruded.

b. New Alloy Systems

Post extrusion processing was continued on two Zr-clad tubes with uranium cores that contain 1 w/o Si and 1.5 w/o Mo, respectively. These tubes are being produced for a comparison of their irradiation behavior with that of the U - 2 w/o Zr alloy irradiated previously. The tube with 1 w/o Si has been inspected, straightened, and beta heat treated vertically in a salt bath at Atlas Steels, Ltd., Welland, Ontario. A heat treatment cycle for the other tube has been developed that provides the desired structure for irradiation stability, i.e., a simulated cast structure. This treatment consists of heating the tube (in an evacuated container) at 775°C for 15 minutes, furnace cooling to 620°C, holding for one hour, and air cooling. The treatment requires the prior removal of the copper extrusion jacket; therefore, before heat treatment the tube will be pickled and autoradiographed to determine the extent of fluctuations of cladding thickness at the core ends. As reported last month, unexpected thinning of the cladding was observed at one location on a tube that contained a short core of U - 1.5 w/o Mo.

On the basis of small-scale experiments reported earlier (DP-415), fabrication of a tube with a new core composition, a U - 0.2 w/o Al alloy, was started. An alloy casting was made by double arc melting of ingot uranium. Arc melting was used to avoid the carbon pickup associated with induction melting in a graphite crucible. Before extrusion, the casting will be gamma treated and quenched, to precipitate a fine dispersion of uranium-aluminum compound, then will receive a double beta treatment and quench to refine the uranium grains.

c. Stabilized Beta-Phase Uranium Alloys

The broad program of developing a metal fuel composition with satisfactory irradiation behavior includes a study of alloys that may retain the creep-resistant beta phase under irradiation conditions. Initial results of heat treatment and structure studies are now available for two of the four alloys that were selected for further study from the preliminary screening tests that were discussed last month. Extruded specimens of U - 0.3 w/o Cr and U - 0.3 w/o Cr - 0.3 w/o Mo alloys were heat treated (to obtain the beta phase) by soaking for one hour at either 720°C (beta phase) or 800°C (gamma phase) and quenching in oil. Subsequent examination by X-ray diffraction showed that all specimens were essentially beta uranium. The high temperature stability of the metastable beta uranium was then evaluated by aging the specimens at 500°C for either one-half or one hour and quenching them in oil. After these treatments the specimens were again examined by X-ray diffraction. During the shorter time at 500°C the U - 0.3 w/o Cr alloy reverted almost completely to the alpha phase. The U - 0.3 w/o Cr - 0.3 w/o Mo alloy retained a fair proportion of beta phase, even after the longer aging time. The beta phase of this alloy was somewhat more stable in the gamma-treated specimen than in the beta-treated specimen.

Although the results for the latter alloy are encouraging, X-ray diffraction provides only a semiquantitative measure of the proportions of beta and alpha phase in a specimen. Other techniques are being studied for improving the reliability of determination of phase ratios.

2. Fabrication of Tubes of Uranium Oxide by Cold Swaging

a. Preparation of Irradiated Specimens

Techniques for fabricating fuel elements of uranium oxide are being studied with the objective of developing a low cost process for fabrication of this fuel, which is an alternative to uranium metal for use in D_2O-moderated power reactors. The principal effort is at the Savannah River Laboratory, where a cold swaging process is being applied to the fabrication of tubes that contain fused uranium oxide. The immediate objective of the program is to irradiate several such tubes in a Savannah River reactor to investigate their suitability for power reactor application. The design and fabrication of the first irradiation specimens are discussed in the following paragraphs.

The design of the initial irradiation assemblies is shown in Figures 10 and 11. The fuel tubes (Figure 10) have a 2.138-inch OD, 1.458-inch ID, and 24-inch length. The cladding is stainless steel of 0.018-inch thickness. Five of these tubes are stacked between two aluminum housings, as shown in Figure 11; during irradiation both the inside and outside surfaces of the fuel tubes will be in contact with coolant.

The fuel tubes were prepared by swaging on a hardened steel mandrel, as was described in DP-395. Each swaged tube, which was 4 feet long, was cut into two irradiation specimens. Stainless steel plugs were welded to each end of the specimens. During the swaging operation the oxide was contained by temporary end plugs of stainless steel rather than the sand and rubber cement that were used previously. These plugs, shown in Figures 12 and 13, were designed to accommodate the differential rate of elongation of the inner and outer sheaths during swaging. With plugs of sand and rubber cement the outer sheath elongated as much as 3/4 inch before the inner sheath elongated appreciably. After the inner sheath "froze" to the mandrel, that sheath elongated rapidly so that after the final swaging pass the inner sheath was only 1/4 inch shorter than the outer sheath. With the stainless steel plugs the outer sheath elongated about 1/2 inch more than the inner sheath early in the swaging operation, but after the inner sheath "froze" to the mandrel, the total elongations approximately equalized themselves. With plugs designed for an inner sheath that was initially 1/2 inch longer than the outer sheath, the final lengths were equal to within 1/8 inch, as shown in Figure 14.

Partial evaluation of the specimens showed that the diameters were within the limits shown in Figure 10; in three tubes eccentricity in the wall thickness was as great as 0.032 inch, compared to the desired maximum value of 0.025 inch.

The UO_2 densities were slightly lower than those observed previously in similar tubes. The densities ranged from 87.6 to 89.1% of theoretical; the average for all samples was 88.3%. Values between 90 and 91% had been expected. The reason for this discrepancy is not yet known.

Little is known of the irradiation performance of oxide tubes; therefore the first irradiation test will be conducted under conditions that do not exceed those known to be satisfactory for oxide rods from the standpoints of oxide temperature, grain growth, and release of fission gases within the oxide.

b. Effect of Particle Size Distribution on the Density of Swaged Oxide

An experimental study of the effect of fines on the swaged density of crushed, fused UO_2 showed that the most favorable distributions of particle size are those in which some of the finer particles in pulverized UO_2 are removed. Under these conditions the maximum swaged density was 92% of theoretical.

The experimental study consisted of an evaluation of stainless-steel-clad swaged rods that contained oxide of various particle size distributions. These rods were swaged from a diameter of 0.626 inch to a final diameter of 0.465 inch; the cross-sectional area of the UO_2 was reduced by 48% during swaging. Variations in particle size distribution were obtained by adding up to 10% of fines from three sieve fractions (-70 to +120 mesh, -120 to +200 mesh, and -200 mesh) to base sizes of -20 to +70 mesh, -20 to +120 mesh, and -20 to +200 mesh. It was necessary to compare UO_2 from four different lots because no single type of fused UO_2 was available in sufficient quantity for the entire experiment. The materials that were used in the experiment are described in Table II.

The swaged densities of all of the specimens are listed in Table III, together with the corresponding particle size distributions. The highest swaged densities were obtained with the following size distributions. It will be noted by comparison with Table II that these size distributions include a larger proportion of the -20, +70 particles than is typical of the pulverized, fused UO_2 usually obtained.

Particle Distribution for Maximum Density of Swaged Oxide

-20, +40	-40, +70	-70, +120	-120, +200	-200	As Loaded	Swaged
57.5	32.5	0	5	5	65.0	92.0
57.5	30.2	12.3	0	0	62.2	91.9

The four types of oxide were nonstoichiometric to different degrees, i.e., they contained different proportions of U_4O_9. Since the densities of the swaged specimens were expressed as a percentage of the theoretical density of stoichiometric UO_2, or 11.0 g/cm^3, it was necessary to normalize the densities to compare them on a common basis. The densities were normalized to that of Type I oxide by linear interpolation between densities of 11.0 g/cm^3 for $UO_{2.00}$ and 11.3 g/cm^3 for U_4O_9. The bases for the interpolations were the O/U ratios reported in Table II. Both the uncorrected densities and the normalized densities are shown in Table III.

3. Stainless Steel - Zircaloy Joints

One of the chief problems that must be faced in the design of pressure tube power reactors is that of joining pressure tubes of Zircaloy to coolant distributors of stainless steel. It is believed that a metallurgical joint is potentially better suited to this application than is a mechanical joint. Not only is less space required for a metallurgical joint, but also it should be more resistant to the detrimental effects of repeated cycles of temperature and pressure. Therefore, a program of fabrication development and testing of metallurgically bonded joints has been started. The joint fabrication work, which is being carried out at Nuclear Metals, Inc., under a du Pont subcontract, is concerned with development of a process for joining Zircaloy-2 and Type 304 stainless steel. Contacts with other organizations had led to consideration of two other types of bonded joint. The present status of these several programs is reviewed in this section.

a. NMI Fabrication

Exploratory tests with small rods at Nuclear Metals, Inc., indicated that a strong metallurgical bond can be made between stainless steel and Zircaloy. On the basis of these rod tests, emphasis was placed on the manufacture of tubular joints. Several reduced-size (~2-inch OD) tube joints were made. One joint, which had a 1.88-inch OD and a 1.48-inch ID, was machined to remove surface irregularities; the finished joint had a 1.82-inch OD and a 1.52-inch ID. Longitudinal sectioning showed that the stainless steel - Zircaloy interface extended about 2-1/4 inches in the axial direction and sloped from the outside to the inside (i.e., the Zircaloy was tapered inside the stainless steel). The tensile strength of this joint was 25,000 psi. However, in a corrosion test for 24 hours in steam at 400°C, a heavy layer of white oxide formed at the interface.

Subsequent to the experiments described above, additional reduced-size joints were fabricated in an effort to improve joint quality. A section from one joint showed no evidence of corrosive attack after nine days in water at 250°C. This test, which is nearer to actual service conditions than the 400°C steam test, provides some encouragement that this particular type of joint may be a satisfactory interlayer from the standpoint of corrosion resistance as well as mechanical behavior. Corrosion tests of tubular joints that were fabricated somewhat differently also appear to be satisfactory. Comparison of the mechanical performance of various joints is in progress.

b. Fusion Bonding

The Atomic Power Equipment Department of General Electric has agreed to an informal cooperative program for developing a fusion bond between stainless steel and Zircaloy. In general, GE will work out methods of joint fabrication and will supply any promising joints to du Pont for evaluation. A 2-inch-OD joint has been forwarded to SRL for evaluation tests, including thermal cycling of the joint.

c. Diffusion Bonding

The Metallurgical Products Department of General Electric is developing a diffusion-bonded butt joint for possible commercial exploitation. Two such joints, both about 2 inches in diameter, have been submitted to SRL for evaluation. GE is attempting to scale up their process to joints of 5-1/2-inch diameter, but are encountering difficulty with cracks in the larger joint.

4. Effect of Irradiation on Structural Materials

a. Zircaloy-2

Six specimens of Zircaloy-2 were shipped to Chalk River for irradiation in the NRX reactor at a temperature of 300°C. The objective of the irradiation test is to measure the stress relaxation that occurs in Zircaloy-2 when it is irradiated at temperatures that are representative of D_2O-moderated power reactors. The specimens, which include both annealed and cold-worked material, are similar to those described in DP-369[(1)].

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(1) DP-369, Stress Relaxation in Stainless Steel During Irradiation, J. W. Joseph, June 1959.

b. Stainless Steel

Full-scale flow tests were begun on an assembly of eleven specimens of the type that will be used to measure the effect of irradiation on the relaxation of torsional stresses in stainless steel. The flow test is preliminary to the series of irradiation experiments that was described in DP-435. An order was placed for the material from which the irradiation specimens will be machined. Equipment for stressing and measuring the specimens is being developed.

B. IRRADIATION TESTS

1. Uranium Tubes

Two Zircaloy-clad fuel tubes (2-inch OD) of natural uranium metal - one with an unalloyed core and one with a core of U - 2 w/o Zr - were irradiated without incident to modest exposures at relatively low temperature in a Savannah River reactor. Postirradiation examination of the tubes revealed that surface roughening had occurred on both tubes but was much more severe on the alloy tube. The outside diameter of each tube increased 0.005 to 0.010 inch during irradiation. The inner surfaces of the tubes have not yet been inspected because the inner housing could not be removed from either tube with the maximum available force of 2000 pounds.

Both tubes contained cores about 9.5 feet long and were fabricated at Nuclear Metals by the coextrusion process. The unalloyed tube (Tube No. 53) received a beta heat treatment, while the alloy tube (Tube No. 28) received a diffusion heat treatment at 880°C for 7 hours and a spheroidizing heat treatment at 680°C for 72 hours. The cladding thicknesses were 0.030 inch for the unalloyed tube and 0.015 inch for the alloy tube. The irradiation conditions are classified data that will be discussed in a separate report.

The roughening of the outer surface of the alloy tube was uniform around the circumference but varied along the length of the tube. The worst roughening occurred near the midpoint, where the calculated exposure was a maximum, and near the downstream end, where the exposure was relatively low. Bumps as high as 0.024 inch were measured from a surface impression that was taken of the tube. Maximum bump height was only 0.010 inch on the surface of the unalloyed tube.

2. Slugs of U - 2 w/o Zr

Shown in Figures 15 through 18 are the dimensional changes that occurred in four Zr-clad slugs of U - 2 w/o Zr during irradiation to modest exposure at metal temperatures near those contemplated for the du Pont designs of metal-fueled power reactors. The slugs were clad with 0.005 to 0.025 inch of Zircaloy-2 by the coextrusion process; the nominal dimensions were a diameter of 0.792 inch, an over-all length of 10.5 inches, and a core length of 9 inches. The desired irradiation temperature was obtained by encasing the slugs in lead-insulated containers. The neutron flux (and exposure) increased approximately linearly from the bottom of each slug to the top; the flux at the bottom was about 65% of that at the top. Quantitative data on irradiation conditions for these slugs and for four others that were irradiated previously (DP-395) will be presented in a forthcoming classified report.

As shown in Figures 15 through 18, the diameter changes along the lengths of the slugs increased with exposure and temperature. However, it is difficult to correlate dimensional change with cladding thickness, because the slug with 25-mil cladding thickness exhibited numerous longitudinal cracks in the cladding. Also, the top end cap separated from the slug with 5-mil cladding thickness. The two remaining slugs did not fail; the slug with 15-mil cladding thickness exhibited half the dimensional change of the slug with 10-mil cladding. More detailed examination of the slugs is in progress.

During disassembly it was discovered that the top end cap had separated from the slug that had a clad thickness of 0.005 inch. The two pieces apparently parted while they were still in the reactor. Photographs of the slug and end cap are presented in Figure 19.

C. HEAT TRANSFER

1. Flow Stability in Fuel Assemblies Cooled by Boiling Water

In the design of power reactors that are cooled by forced convection of boiling water, it is essential that fuel assemblies be designed such that vapor choking and subsequent fuel melting will not occur under anticipated operating conditions. Although this problem can be circumvented by placing an orifice in series with each coolant channel, this approach can be expensive because of the increase in power required to pump the coolant. Accurate knowledge of the flow characteristics of the fuel assemblies is necessary to ensure that orifices are not provided unless they are actually needed, and to ensure that any orifices are sized for minimum head loss consistent with control of the vapor choking problem. A flow loop is being installed at SRL to measure the boiling flow characteristics at conditions that would be encountered in D_2O-moderated boiling reactors. In the meantime, the flow characteristics have been calculated for a wide range of conditions, both to guide the experiments and to provide a preliminary indication of regions of possible flow instability. These calculations are described in the following paragraphs.

The calculations, which were made on an IBM 650 computer, were on the basis of a method that treats steam-water mixtures as a homogeneous fluid. The properties of the homogeneous fluid are derived by weighting the properties of the two phases on the basis of their relative mass. The applicability of this method, which was described in DP-109[1], has been verified at pressures of 50 to 100 psig (see Figure 20). The method should also be applicable at higher pressures because the properties of the two phases do not differ as widely.

(1) DP-109, Two-Phase Flow in Tubes with Outlet Orifices, S. Mirshak, April 1955.

The results of the calculations are shown in Figures 21 through 24. Although the properties of light water were used throughout, the results are a good approximation for heavy water. In Figure 21, the effect of pressure on the head loss is shown for upflow of water through a coolant channel that has a heated length of 18 feet and receives heat at a rate of 100 MW/ft^2 of flow area. The "humped" curves for pressures of 100 and 250 psi are characteristic of potentially unstable flow systems because they include regions in which the pressure required to maintain flow increases as the flow decreases. These systems would require inlet orificing for maintenance of stable boiling flow at low steam qualities. At pressures of 500 and 750 psi, the required inlet pressure decreases monotonically with flow, which is the characteristic of a stable flow system. Even at the higher pressures, regions of flow instability appear if the subcooling of the inlet flow is high (see Figure 22). The effects of power level and equivalent diameter on the head loss through a coolant channel that is 18 feet long are shown in Figures 23 and 24, respectively.

2. Burnout of Zircaloy-2 Surfaces

The burnout heat flux for Zircaloy-2 surfaces cooled by liquid water agreed within 3% of the results obtained for stainless steel surfaces and copper-nickel surfaces. This finding indicates that the composition of the heat transfer surface of a fuel assembly has negligible effect on burnout. Three burnout tests were made on Zircaloy-2 tubes. In the first test, the tube was etched with $HF\text{-}HNO_3$, while in the second and third tests the tubes were etched and then autoclaved for 72 hours at 750°F and 1500 psi to produce an oxide film. The tests were made on electrically heated tubes of Zircaloy-2 that were 0.50 inch in diameter and 24 inches long. The heated surface formed the inner surface of an annulus and was cooled with downward-flowing water.

The summarized results of the tests are tabulated below.

Burnout Heat Flux for Zircaloy-2

Surface Treatment	Coolant Velocity, ft/sec[(a)]	Coolant Pressure, psia	Subcooling, °C	Burnout Heat Flux, pcu/(hr)(ft²) Zircaloy-2	Stainless Steel[(b)]
Etched and autoclaved	20.6	46.7	49.7	1,100,000	1,090,000
Etched and autoclaved	15.2	62.9	55.9	1,130,000	1,140,000
Etched only	19.9	47.1	48.5	1,110,000	1,070,000

(a) The equivalent diameter of the coolant annulus was 0.375 inch.
(b) The values for stainless steel were obtained by interpolating data for stainless steel via the SRL burnout correlation (DP-355).

Although the tests were conducted at pressure levels that are well below those considered for power reactors, it is not expected that the effect of the material will be different at higher pressures.

D. HEAVY WATER LEAKAGE

The available data on the leakage of fluid through mechanical shaft seals for centrifugal pumps pertain to liquid leakage, and there is little information on vapor losses that might be expected from pumps that would handle D_2O at power reactor temperatures. A program to provide such information was started at SRL, where measurements were made of the total leakage through the seals of a pump in a flow loop. The initial measurements indicated a leakage of 40 gallons per year of liquid water through one seal and 120 gallons per year through the other. The vapor leakage was less than 2.5 gallons per year through the one seal assembly for which this leakage was measured. These values are for pump operation at 850 psig and 260°C.

The seal assemblies of the SRL pump are Borg-Warner Type D mechanical seals with stationary faces of "Graphitar" and rotating faces of "Stellite." The seals are mounted on a 2-inch-OD shaft and operate at a pressure of 850 psig and a shaft speed of 3600 gpm. The pump circulates deionized water at a temperature of 260°C. Mounted on the shaft between the impeller and each seal is a small turbine pump that circulates water through an external heat exchanger, where it is cooled below the atmospheric boiling point. The purpose of these turbine pumps is to cool the leakage flow before it passes over the seal faces. The leakage from the seals flowed from internal collection chambers to closed receivers outside the pump, where the volumetric leakage was measured. Dry nitrogen was metered into the internal collection chamber of the outboard seal. This gas passed over the surface of the water in the receiver, and discharged through a water-cooled condenser. Dew point analyses of the exhaust nitrogen provided a measure of the water vapor leakage, including any vaporization that occurred in the collection chamber and the receiver.

The average leakage rates of liquid water were 40 gallons per year for the inboard seal and 120 gallons per year for the outboard seal during the first 90 hours of operation of the pump at 850 psig and 260°C. During this same period, the vapor leakage from the outboard seal was less than 2.5 gallons per year, as determined from occasional dew point analyses of the nitrogen sweep. The maximum leakage rates during the first day of operation (at a system pressure of about 100 psig) were 530 and 1200 gallons per year for the inboard and outboard seals, respectively. Before the leakage measurements were started, the inboard seal was operated for 18 months. The outboard seal was newly installed before the tests were started.

TABLE I

TRANSIENTS IN THE LIQUID-D_2O-COOLED LOOP
OF THE HWCTR FOLLOWING A REACTOR SCRAM

D_2O Bypassing Loop Heat Exchanger, % of total flow	Gas Volume in Surge Tank, gallons	Steam Valve Opening, square inches	Pressure Differential between Loop and Reactor after 35 Seconds, psi[a]
0	110	22.2	610.4
"	55	"	567.5
"	22	"	452.9
0 → 50	110	"	613.7
"	55	"	572.5
"	22	"	462.2
0 → 95	110	"	639.0
"	55	"	615.2
"	22	"	548.2
87.35	110	"	570.8
"	55	"	495.3[b]
"	22	"	311.1[c]
87.35 → 95	110	"	610.6
"	55	"	563.7
"	22	"	439.7
87.35 → 99	110	"	653.6[d]
"	55	"	642.4
"	22	"	609.8
0	110	22.2 → 2.0	533.4
"	55	"	490.5
"	22	"	375.9
0 → 50	110	"	536.7
"	55	"	495.5
"	22	"	385.2
0 → 95	110	"	562.0
"	55	"	538.2
"	22	"	471.2
0	110	f(t)[e]	605.5
0 → 50	"	"	608.8
0 → 95	"	"	634.1
0	"	22.2 → 0	520.8
0 → 50	"	"	524.1
0 → 95	"	"	549.4

(a) 485-psi pressure differential at initiation of scram
(b) Recommended operation
(c) Conditions for lowest pressure differential
(d) Conditions for highest pressure differential
(e) Steam valves continuously adjusted to maintain steam temperature constant at 190°C

TABLE II

TYPICAL PARTICLE DISTRIBUTION OF CRUSHED, FUSED UO_2

Oxide Type	Particle Size Distribution, As Crushed, w/o -20, +40	-40, +70	-70, +120	-120, +200	-200	O/U
I	46.2	26.2	11.3	6.9	9.4	2.022
II	46.2	27.5	10.2	6.1	9.5	2.056
III	49.1	25.7	10.5	6.1	8.5	2.089
IV	47.3	25.0	10.3	7.1	10.1	2.005

TABLE III

EFFECT OF PARTICLE SIZE DISTRIBUTION ON SWAGED DENSITY OF FUSED UO_2
(Duplicate samples)

	Particle Size Distribution, w/o				Swaged Density Rel. to Theo. for $UO_{2.00}$, %	
		Added Fractions				Normalized to
Oxide Type	Base Size	-70, +120	-120, +200	-200	As Measured	Type I Data[a]
I	100	0	0	0	91.2	91.2
(Base size	95	5	0	0	91.4	91.4
-20 to +70	95	0	5	0	91.5	91.5
mesh, O/U =	90	0	10	0	91.2	91.2
2.022)	95	0	0	5	91.0	91.0
	90	0	0	10	91.8	91.8
	90	0	5	5	92.0	92.0
	85	0	10	5	91.8	91.8
	80	0	10	10	91.2	91.2
II	100		0	0	92.6	91.9
(Base Size	95		5	0	91.4	90.7
-20 to +120	90		10	0	91.4	90.7
mesh, O/U =	95		0	5	91.8	91.1
2.056)	90		0	10	91.2	90.5
	90		5	5	90.8	90.0
	85		10	5	90.8	90.0
	80		10	10	90.4	89.7
III	100			0	91.3	91.0
(Base size	95			5	90.4	90.0
-20 to +200	90			10	91.1	90.8
mesh, O/U = 2.089)						
IV (As crushed O/U = 2.005)	See Table I				90.2	90.4

(a) Densities for Types II, III, and IV were normalized to those of Type I to correct for differences in O/U ratio of the starting materials. See text for discussion of normalization procedure.

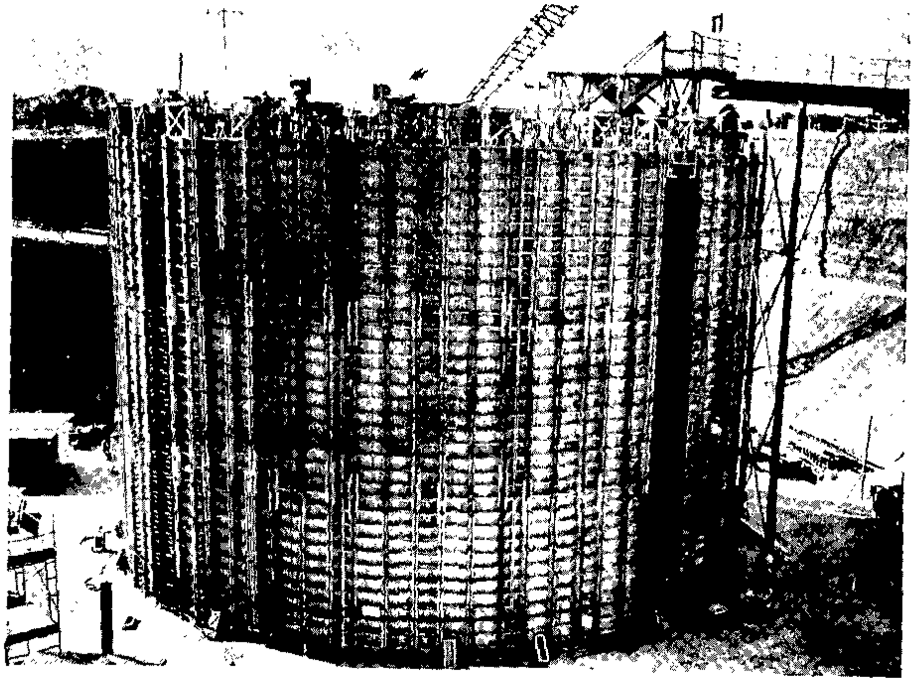


FIGURE 1 – STATUS OF HWCTR CONSTRUCTION
(End of October 1959)

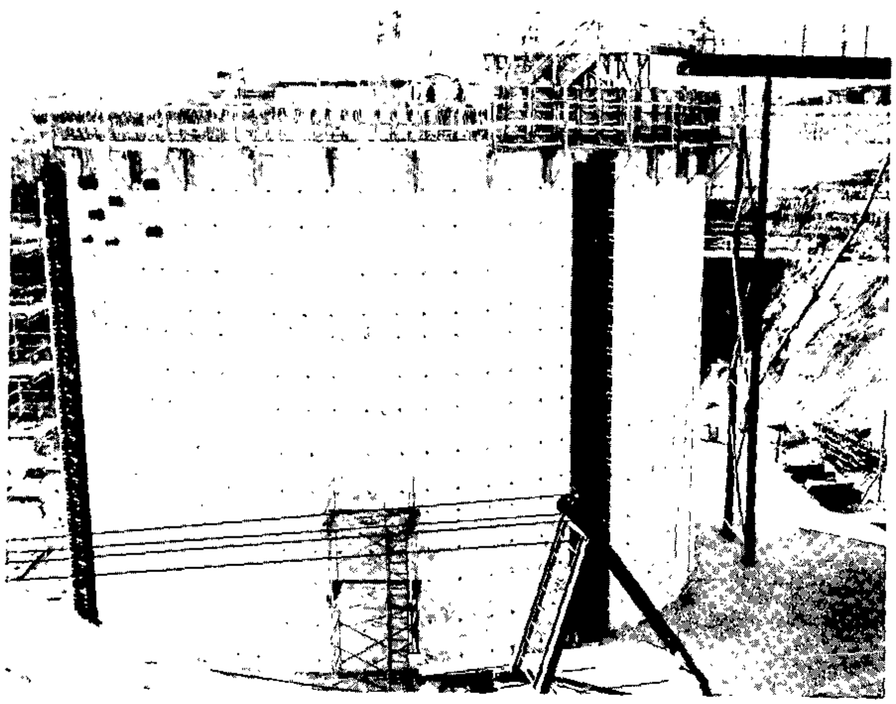


FIGURE 2 – STATUS OF HWCTR CONSTRUCTION
(End of November 1959)

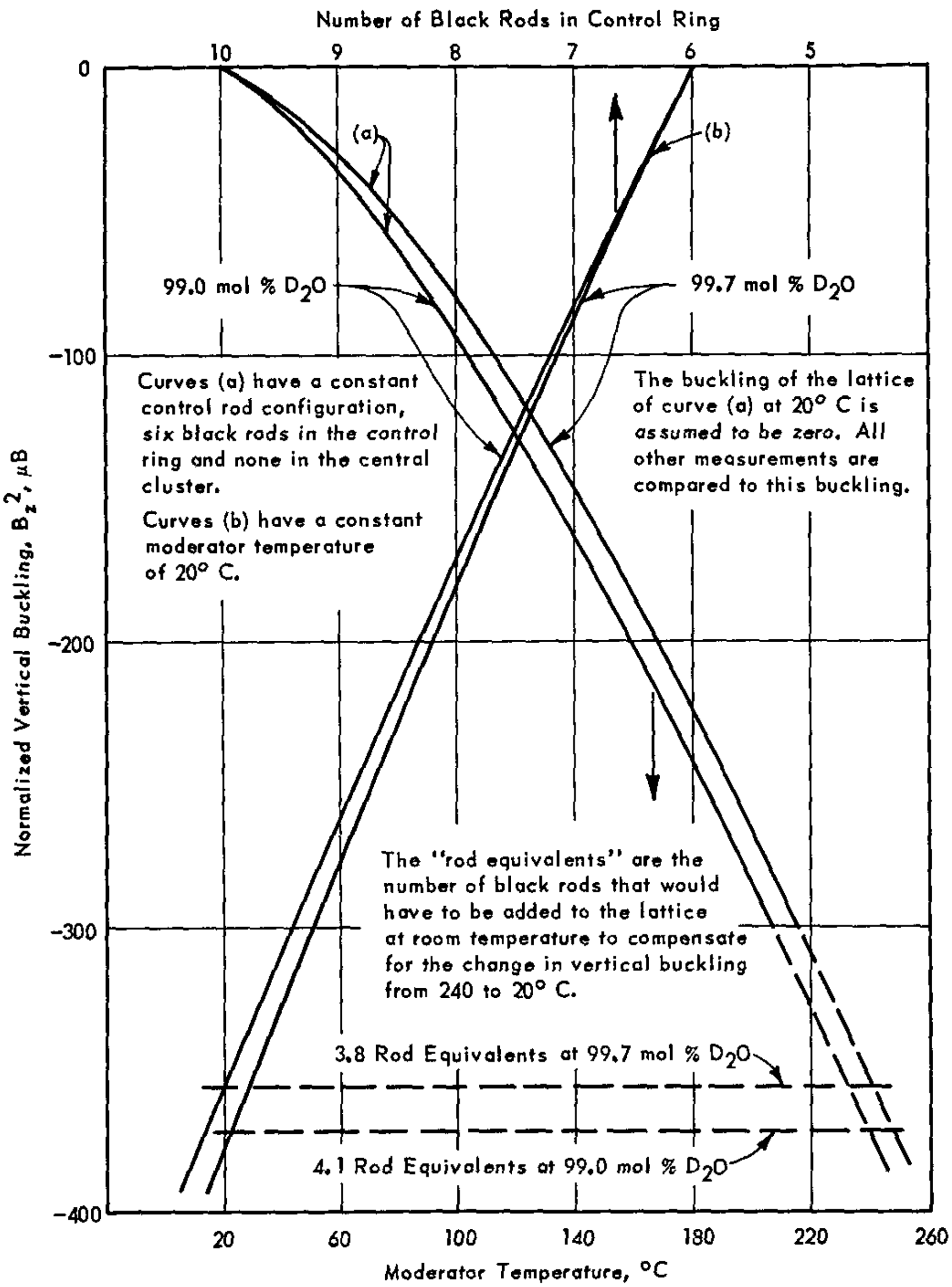


FIGURE 3 – EFFECTS OF MODERATOR TEMPERATURE AND BLACK CONTROL RODS ON THE VERTICAL BUCKLING OF THE PSE MOCKUP OF THE HWCTR (Case I)

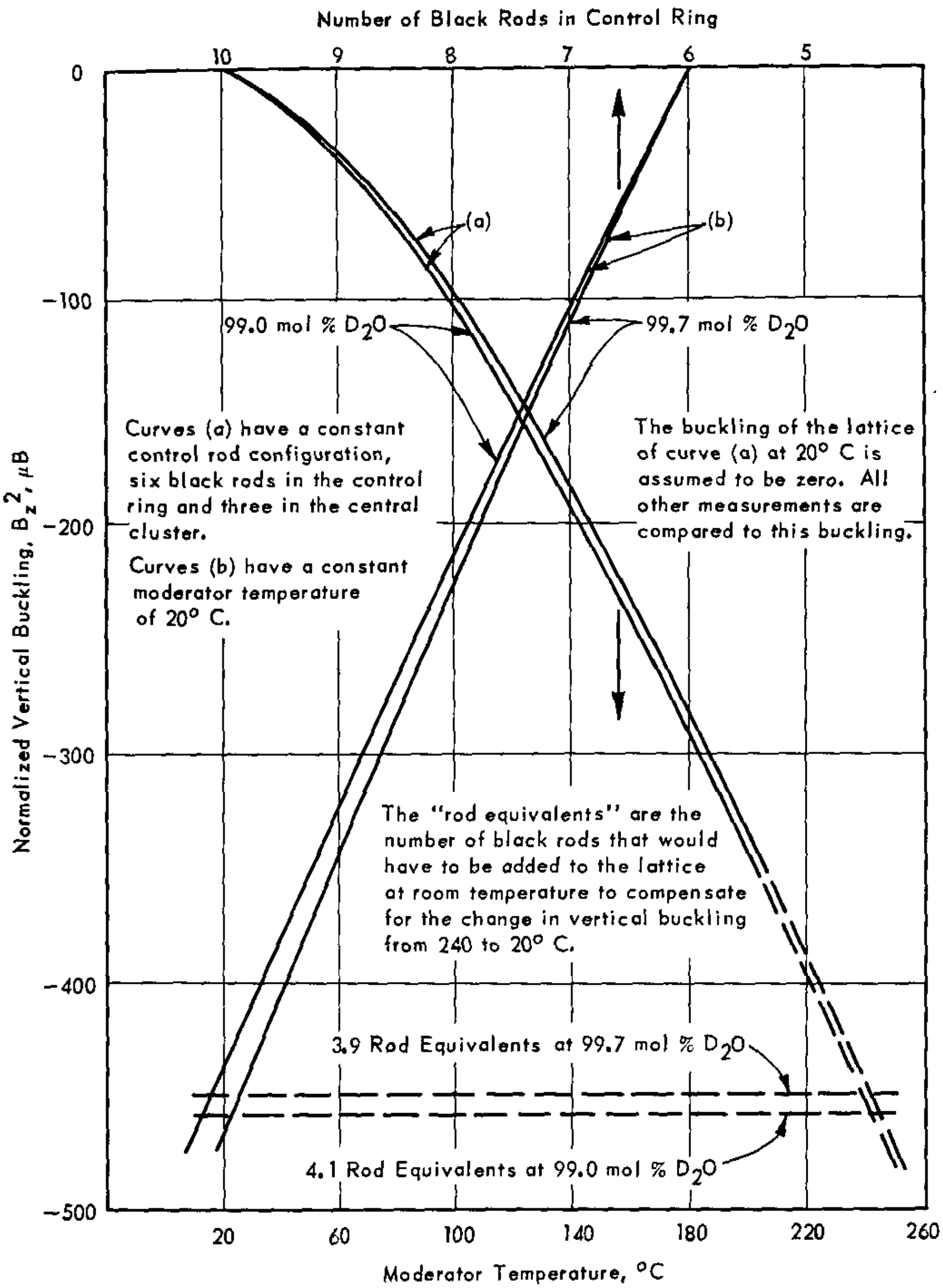


FIGURE 4 - EFFECTS OF MODERATOR TEMPERATURE AND BLACK CONTROL RODS ON THE VERTICAL BUCKLING OF THE PSE MOCKUP OF THE HWCTR (Case II)

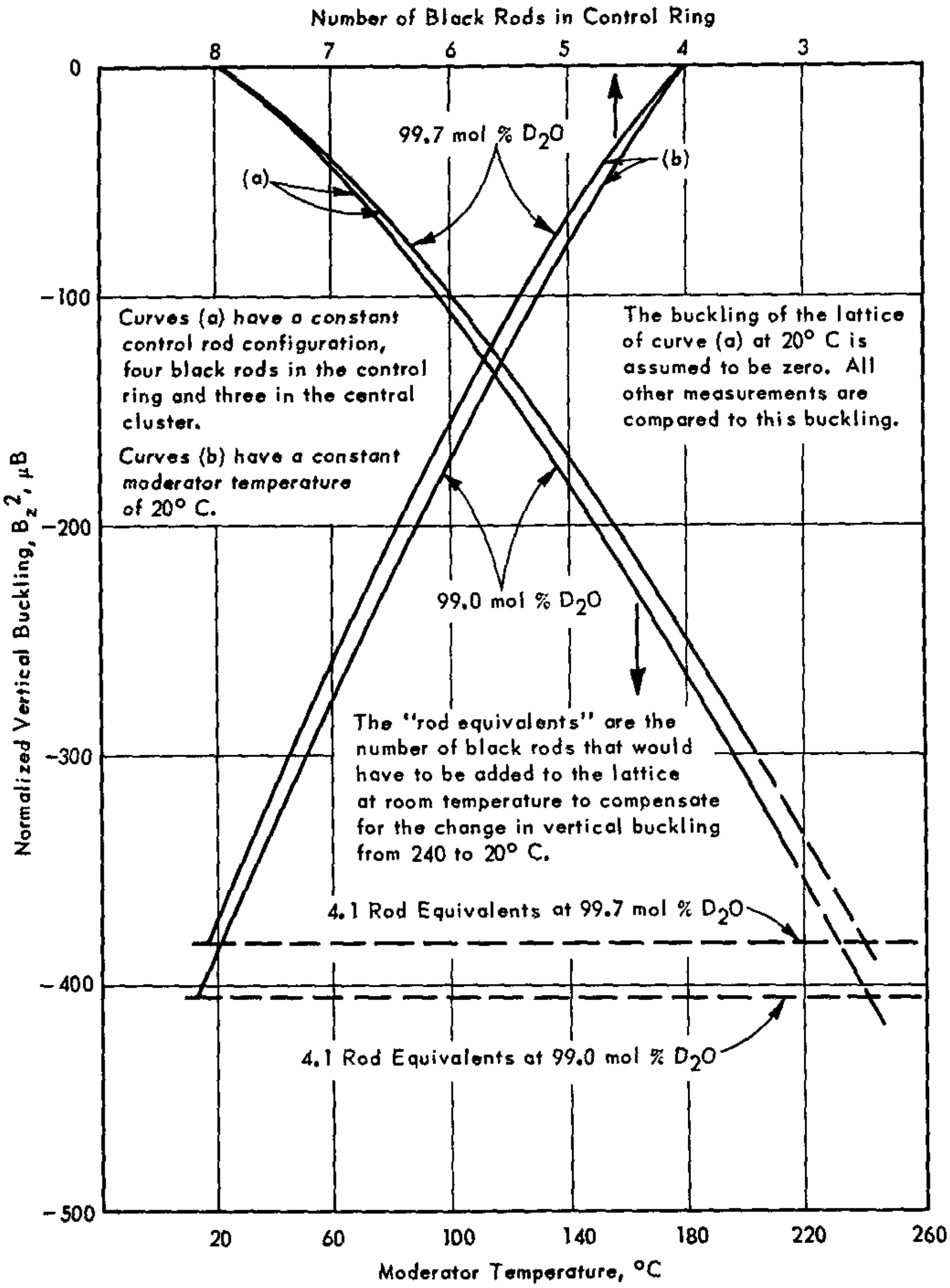


FIGURE 5 – EFFECTS OF MODERATOR TEMPERATURE AND BLACK CONTROL RODS ON THE VERTICAL BUCKLING OF THE PSE MOCKUP OF THE HWCTR (Case III)

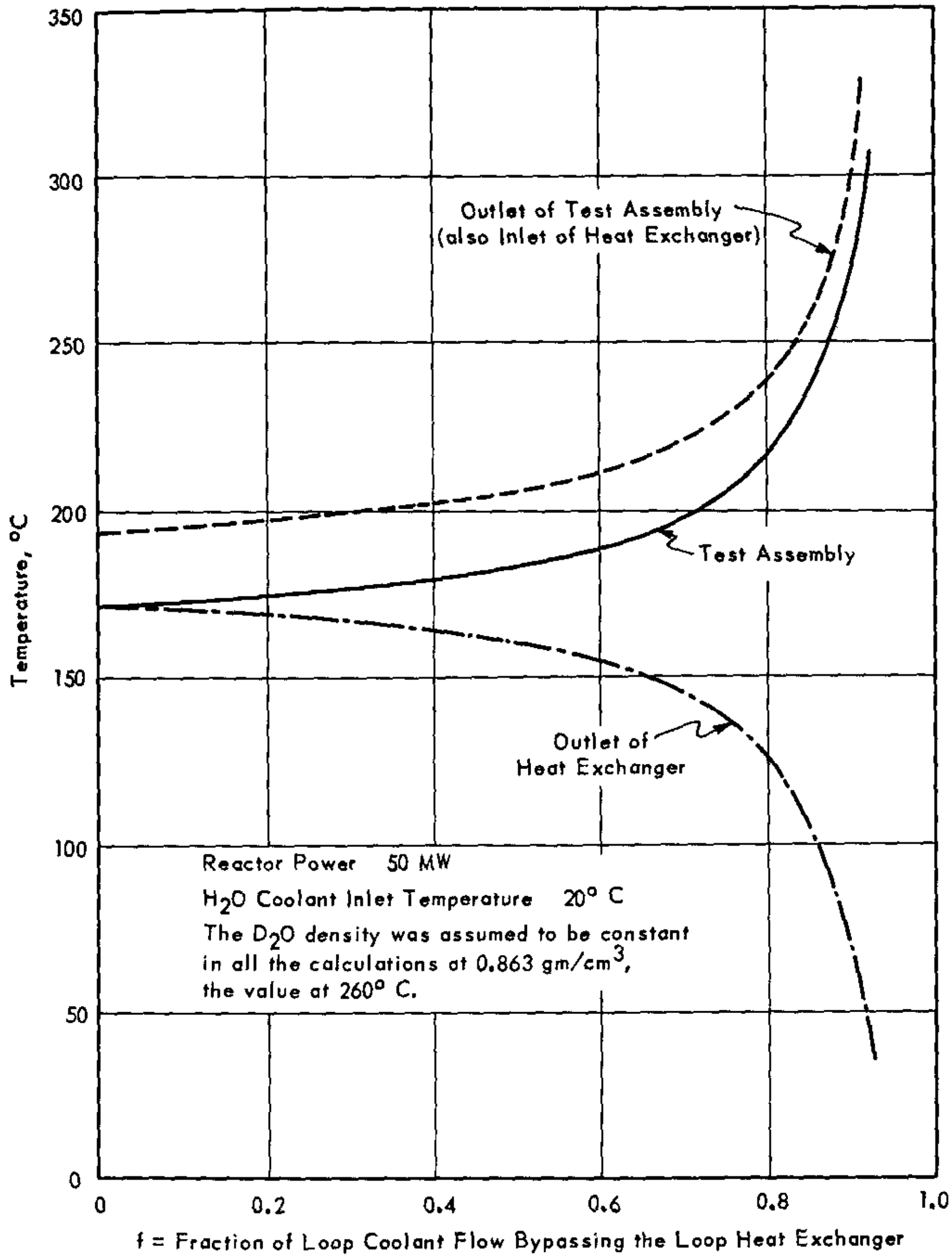


FIGURE 6 – STEADY-STATE TEMPERATURES IN THE HWCTR LIQUID-D_2O-COOLED LOOP

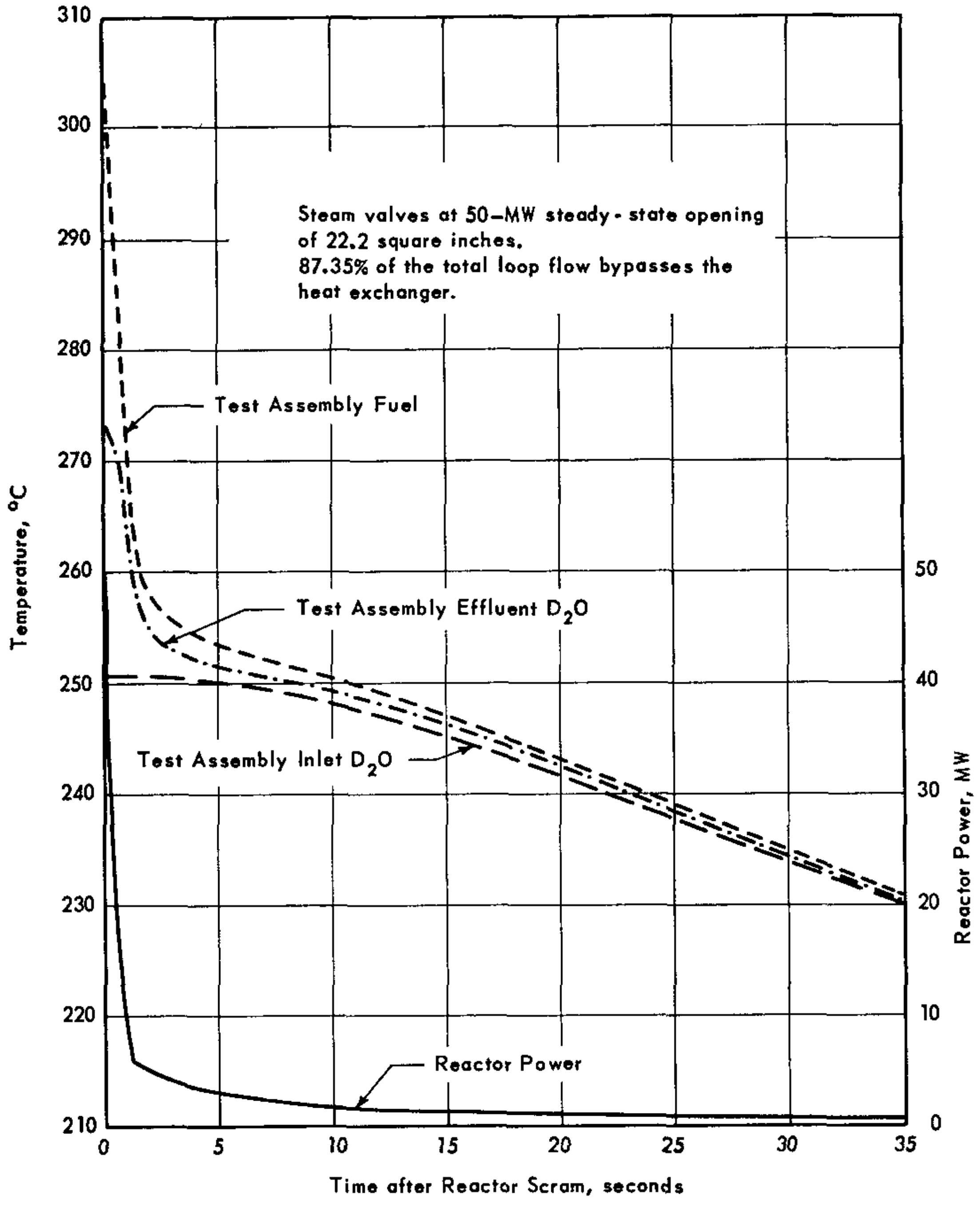


FIGURE 7 – TRANSIENTS IN THE LIQUID-D_2O-COOLED LOOP OF THE HWCTR FOLLOWING A REACTOR SCRAM

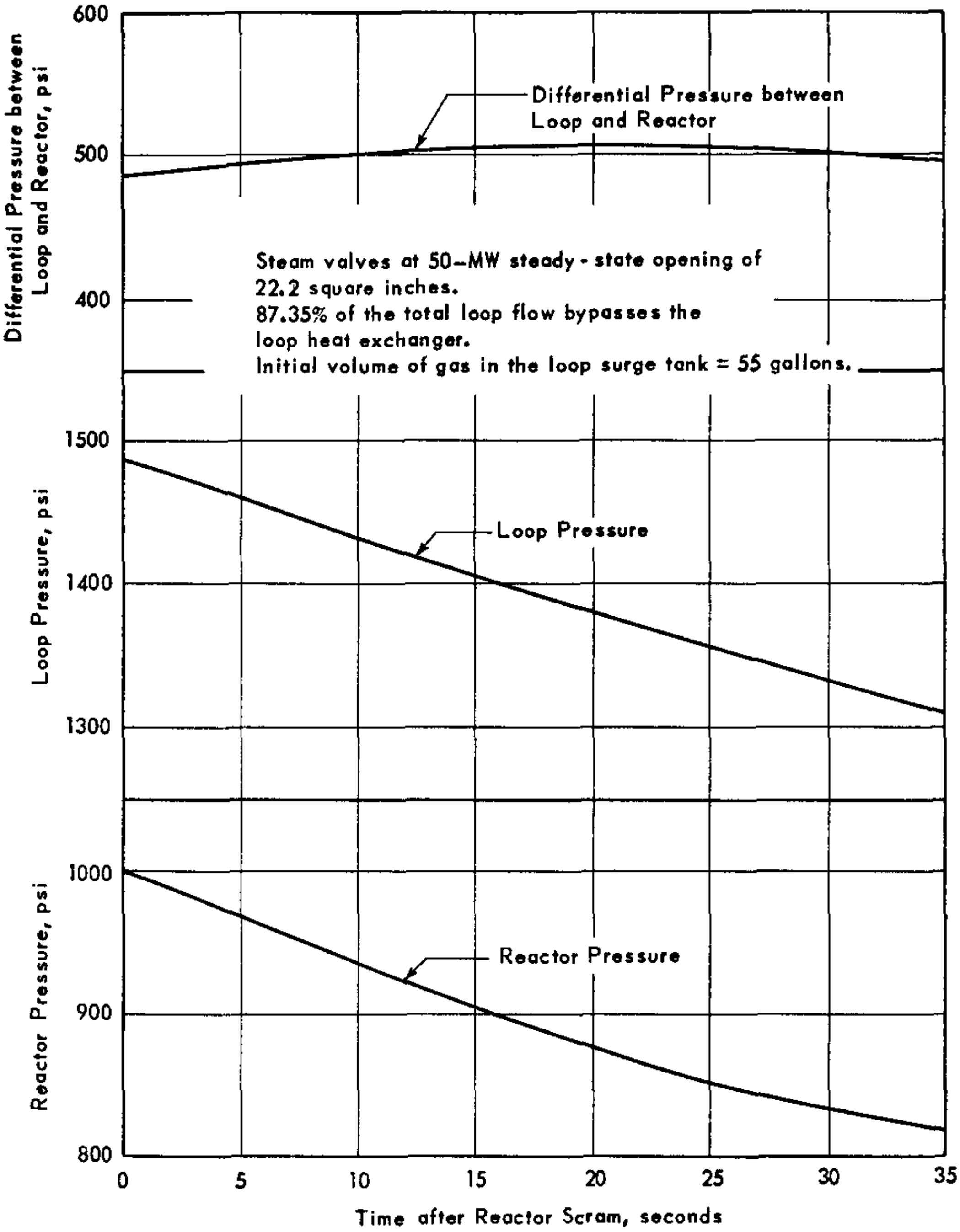


FIGURE 8 – PRESSURE TRANSIENTS IN THE HWCTR LIQUID-D_2O-COOLED LOOP FOLLOWING A REACTOR SCRAM

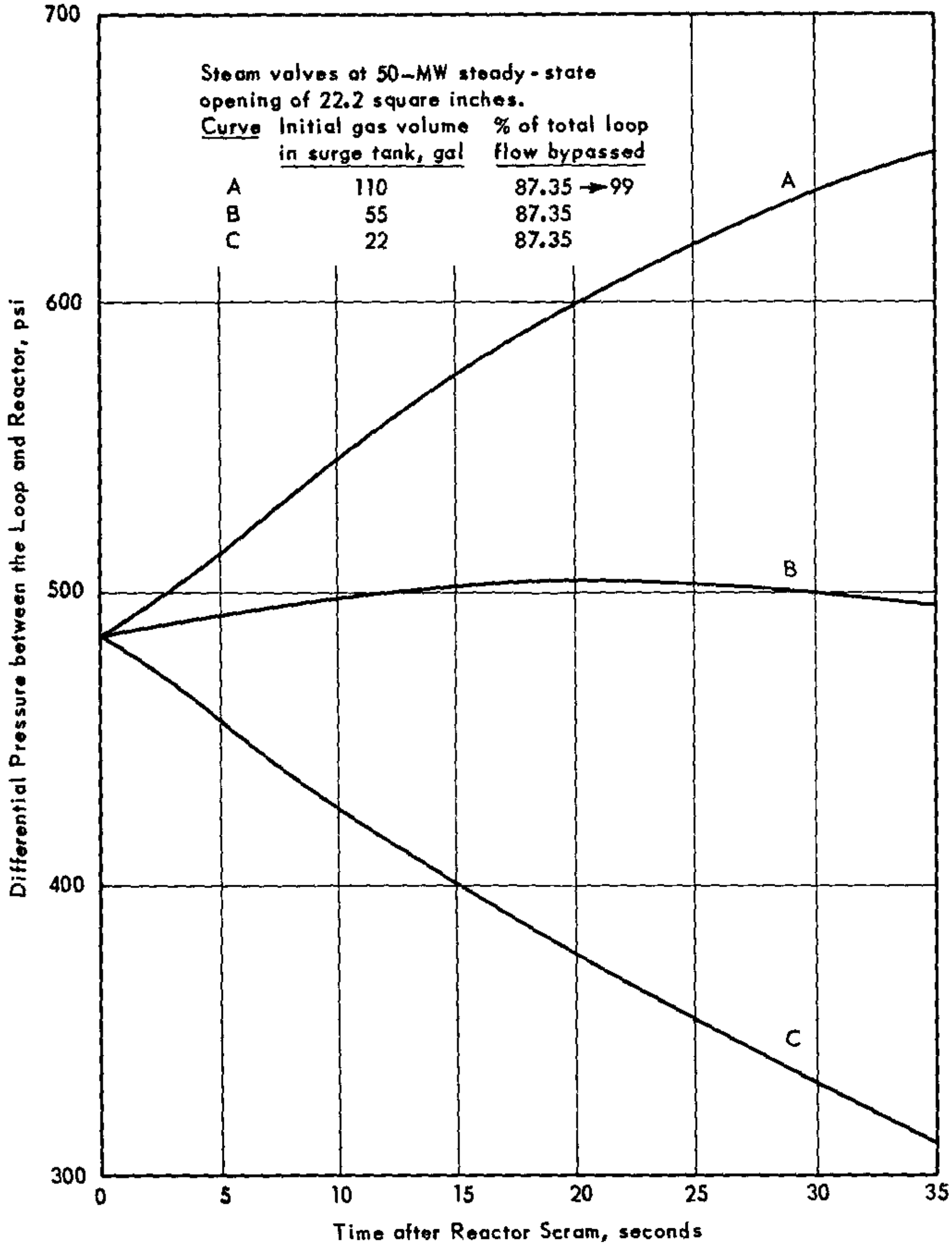
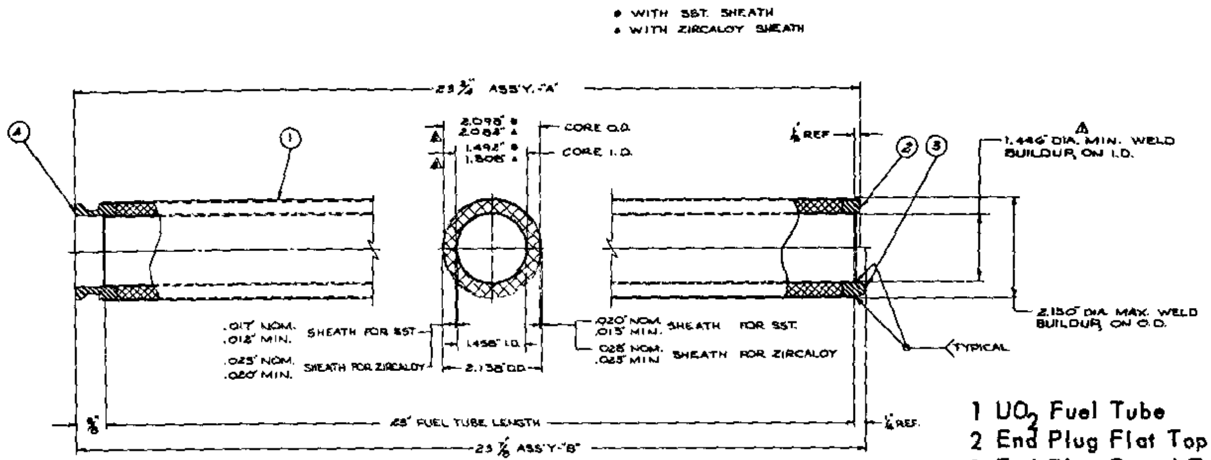
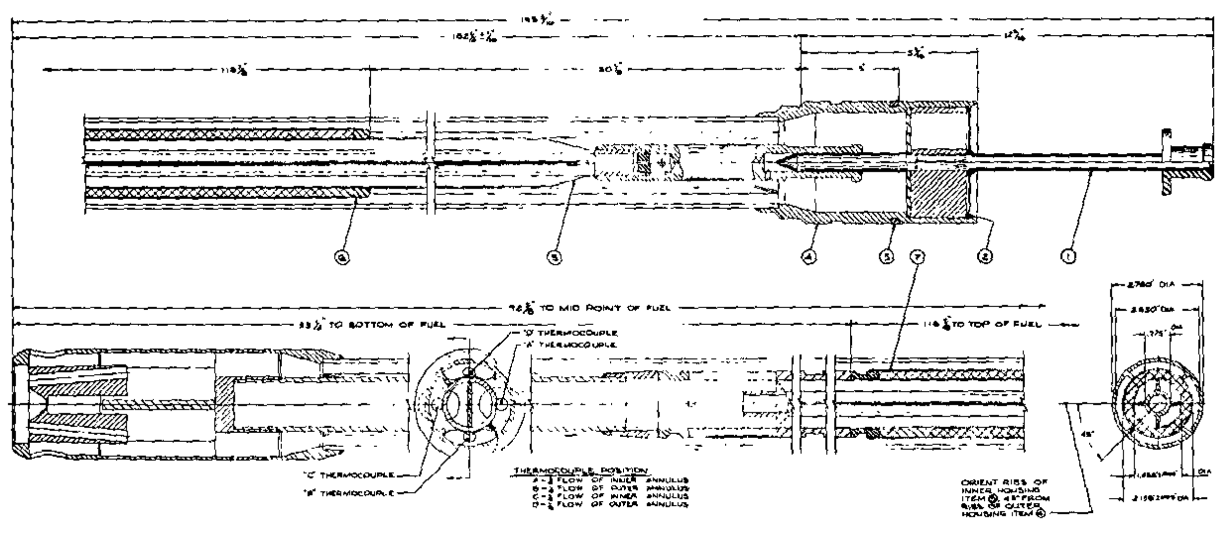


FIGURE 9 – TRANSIENT IN DIFFERENTIAL PRESSURE BETWEEN THE HWCTR LIQUID-D_2O-COOLED LOOP AND THE REACTOR FOLLOWING A REACTOR SCRAM



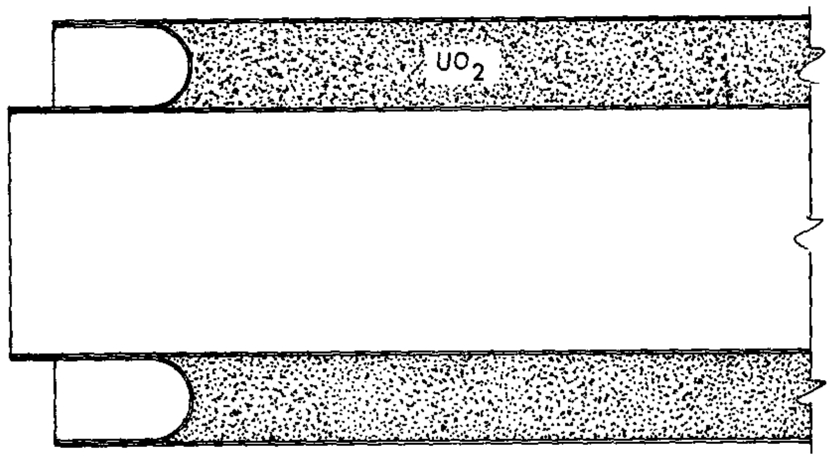
1 UO_2 Fuel Tube
2 End Plug Flat Top
3 End Plug Round Top
4 Bottom End Plug

FIGURE 10 - FUEL TUBE FOR IRRADIATION TESTS OF SWAGED URANIUM OXIDE

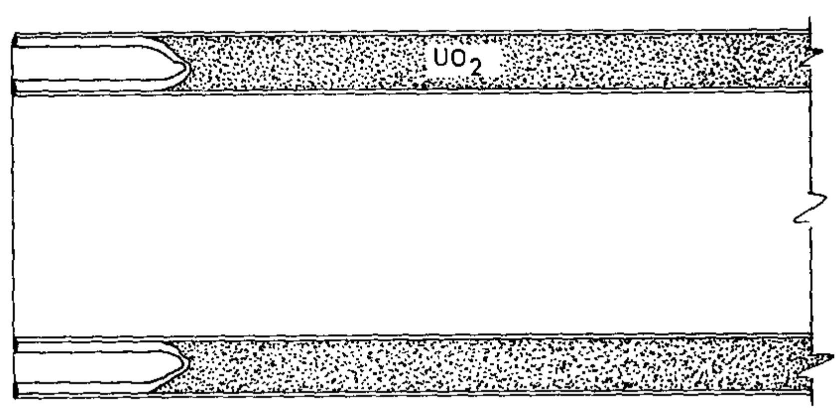


1 Stem Subassembly
2 Main Retaining Ring
3 Piston Ring
4 Outer Housing Subassembly
5 Inner Housing Subassembly
6 UO_2 Fuel Tube Assembly
7 UO_2 Fuel Tube Assembly

FIGURE 11 - FUEL ASSEMBLY FOR IRRADIATION TESTS OF SWAGED TUBES OF URANIUM OXIDE



a. Longitudinal section before swaging, showing
how the end plug is welded within the sheath



b. Longitudinal section after swaging, showing
how the end plug adjusted to the changes
that occurred in the lengths of the sheaths

FIGURE 12 – TEMPORARY END PLUGS OF STAINLESS STEEL FOR SWAGING OXIDE-FILLED TUBES

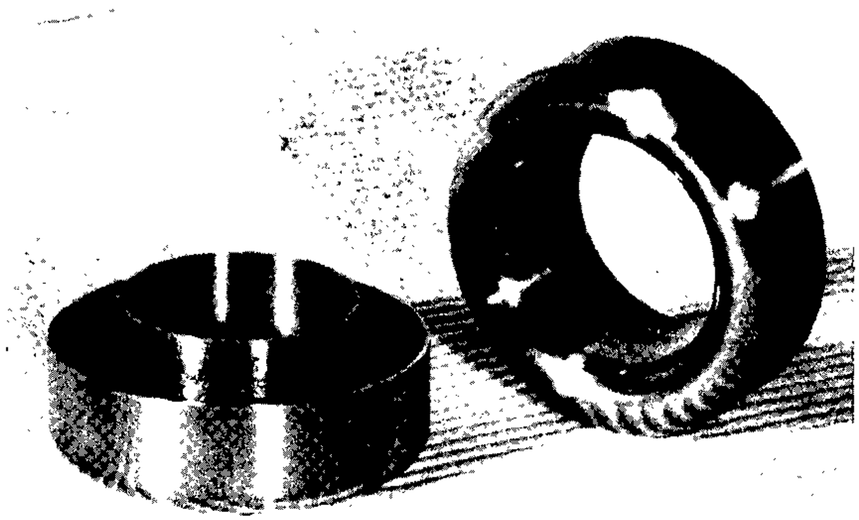


FIGURE 13 – TEMPORARY END PLUGS OF STAINLESS STEEL FOR SWAGING OXIDE-FILLED TUBES

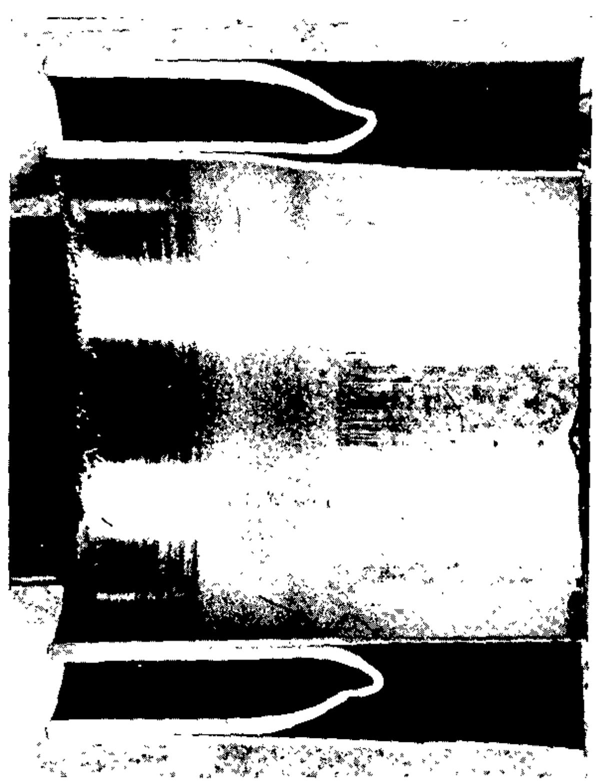


FIGURE 14 – LONGITUDINAL SECTION THROUGH END PLUG OF URANIUM OXIDE TUBE AFTER SWAGING

(The tube walls were distorted during sectioning.)

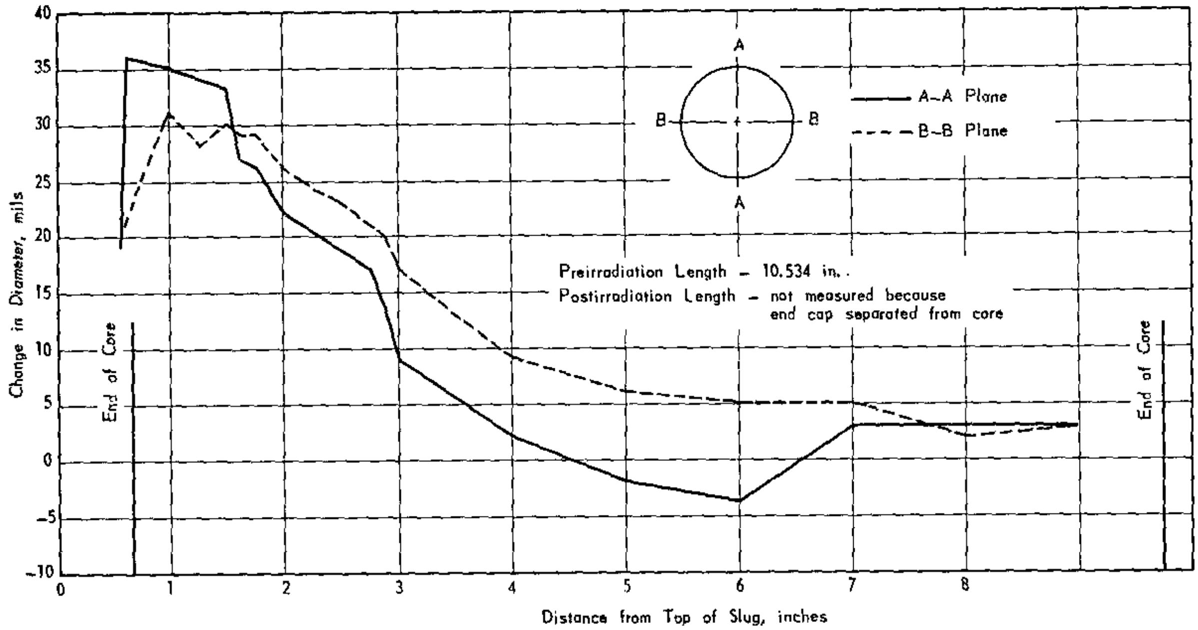


FIGURE 15 – EFFECT OF IRRADIATION ON DIMENSIONS OF A U-2 w/o Zr SLUG WITH 0.005-INCH ZIRCALOY CLADDING

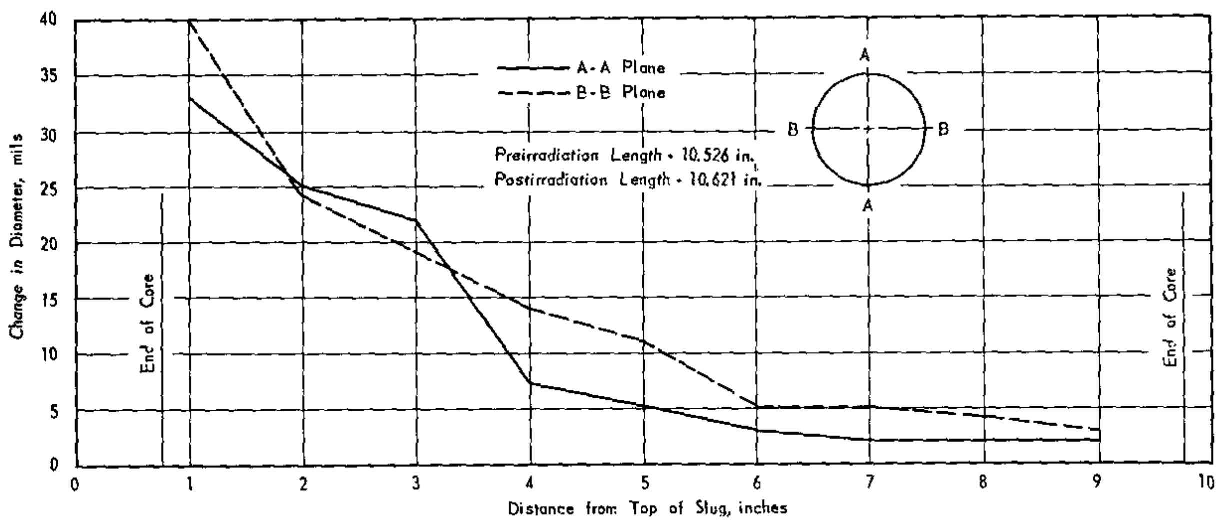


FIGURE 16 – EFFECT OF IRRADIATION ON DIMENSIONS OF A U-2 w/o Zr SLUG WITH 0.010-INCH ZIRCALOY CLADDING

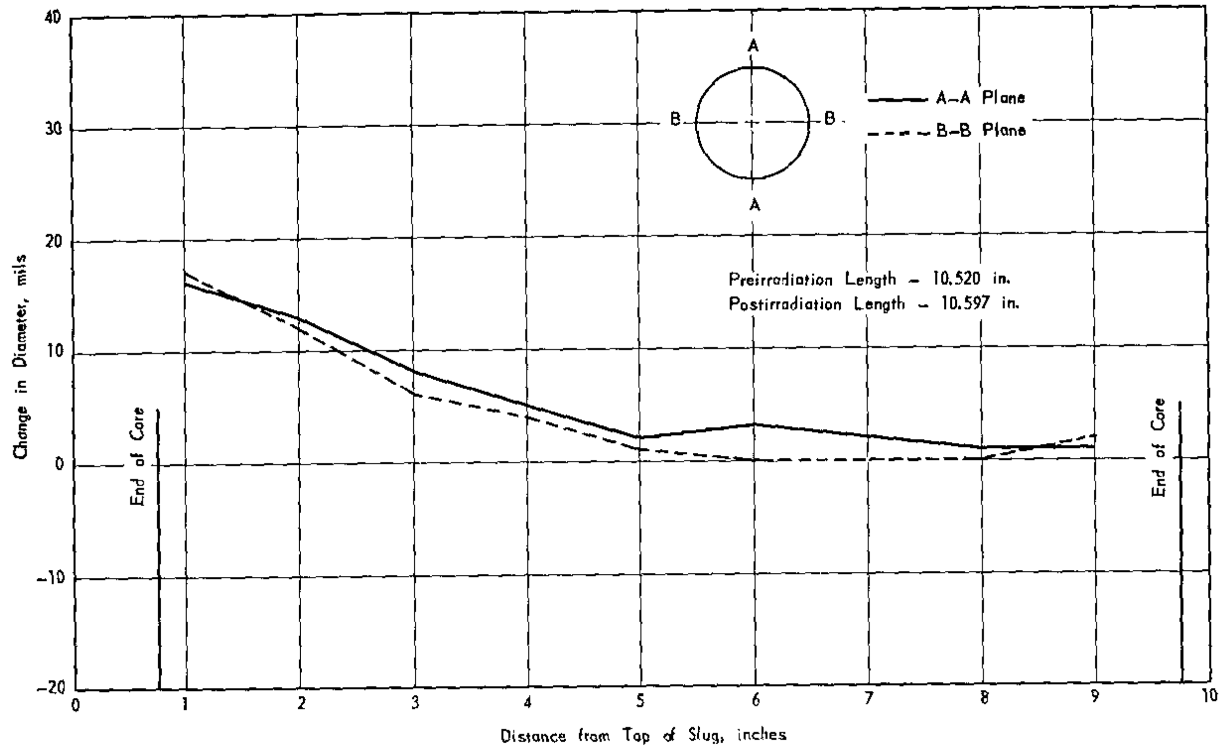


FIGURE 17 – EFFECT OF IRRADIATION ON DIMENSIONS OF A U-2 w/o Zr SLUG WITH 0.015-INCH ZIRCALOY CLADDING

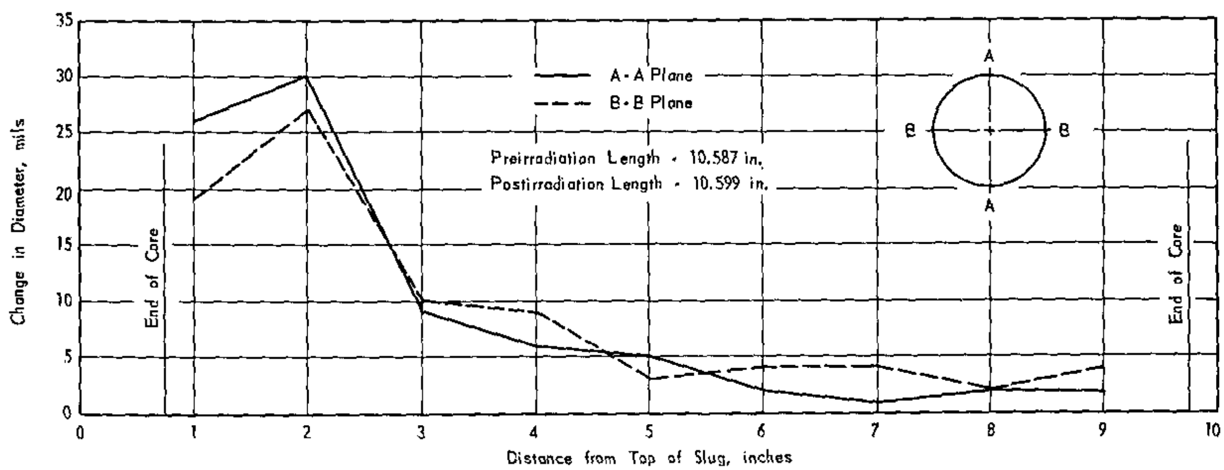


FIGURE 18 – EFFECT OF IRRADIATION ON DIMENSIONS OF A U-2 w/o Zr SLUG WITH 0.025-INCH ZIRCALOY CLADDING

END CAP

SLUG

Smooth Machined Zr Surface

End of Uranium Core

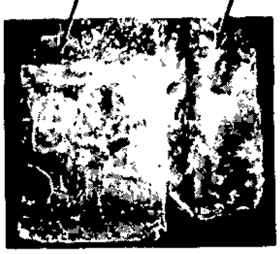
Adhering Lead Particles

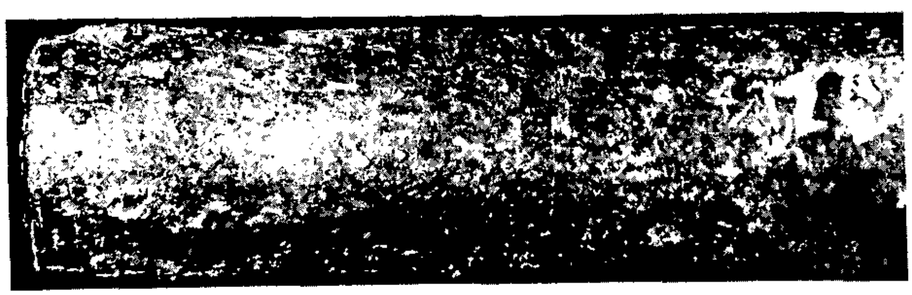
End View Showing Interface

Oblique View

Steel Cylinder

Zircaloy-2 End Cap





Side View

Side View

Mag. 1.5X

FIGURE 19 – IRRADIATED U-2 w/o Zr SLUG WITH 0.005-INCH ZIRCALOY CLADDING

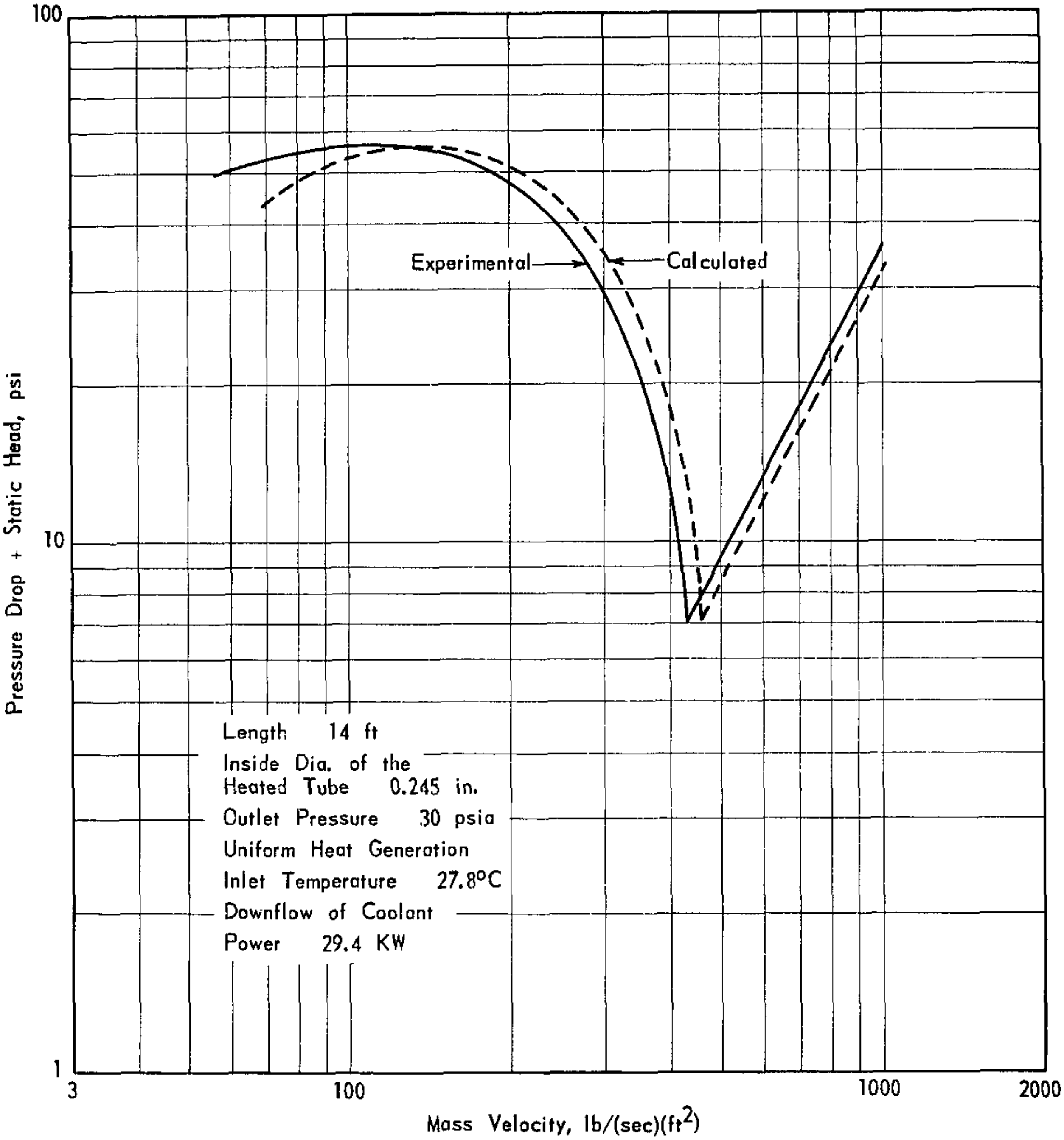


FIGURE 20 – CALCULATED AND EXPERIMENTAL FLOW-PRESSURE DROP CHARACTERISTICS FOR FLOW OF BOILING WATER

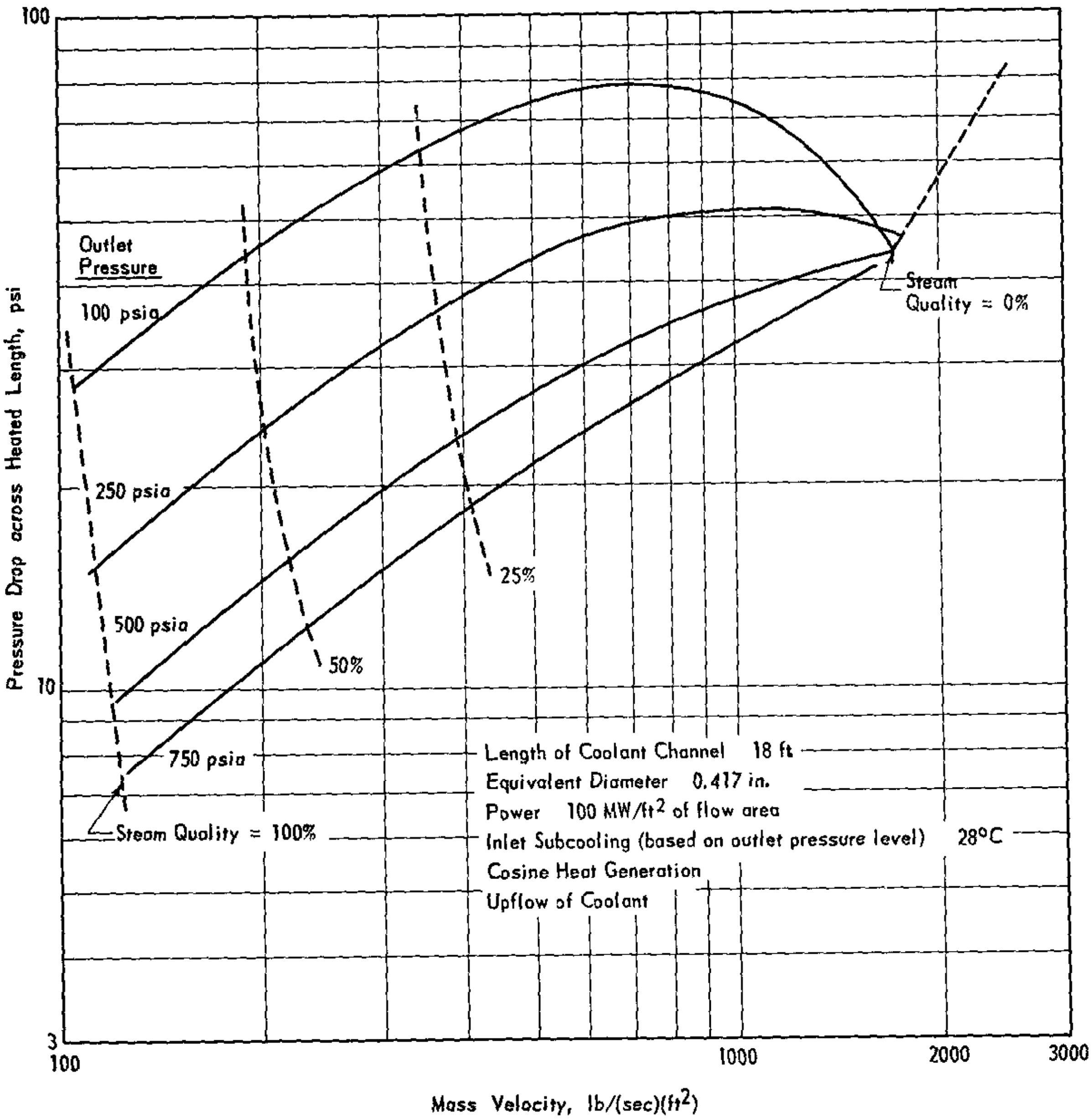


FIGURE 21 – CALCULATED EFFECT OF PRESSURE ON THE PRESSURE DROP FOR FLOW OF BOILING WATER

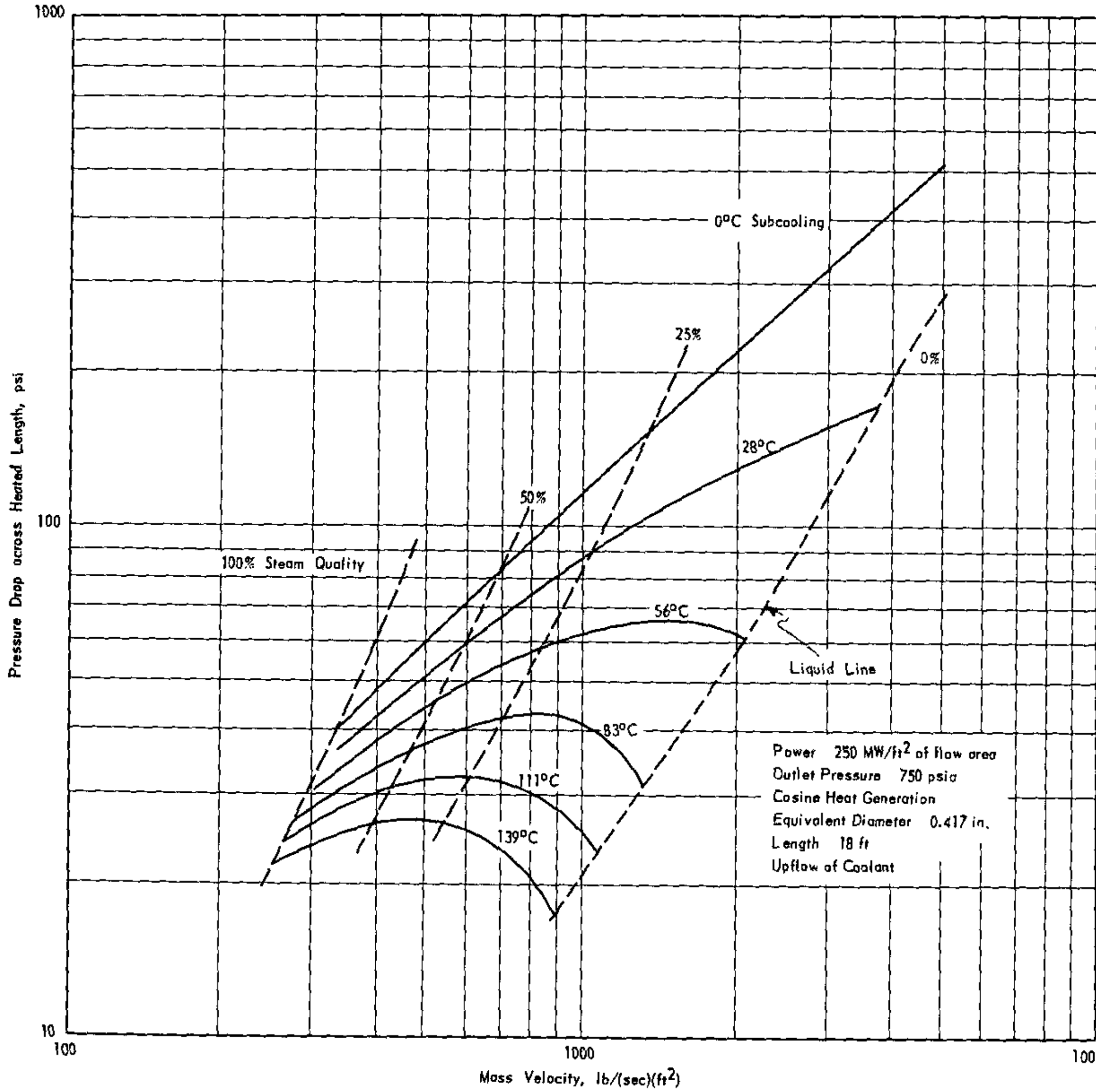


FIGURE 22 – CALCULATED EFFECT OF SUBCOOLING ON THE PRESSURE DROP FOR FLOW OF BOILING WATER

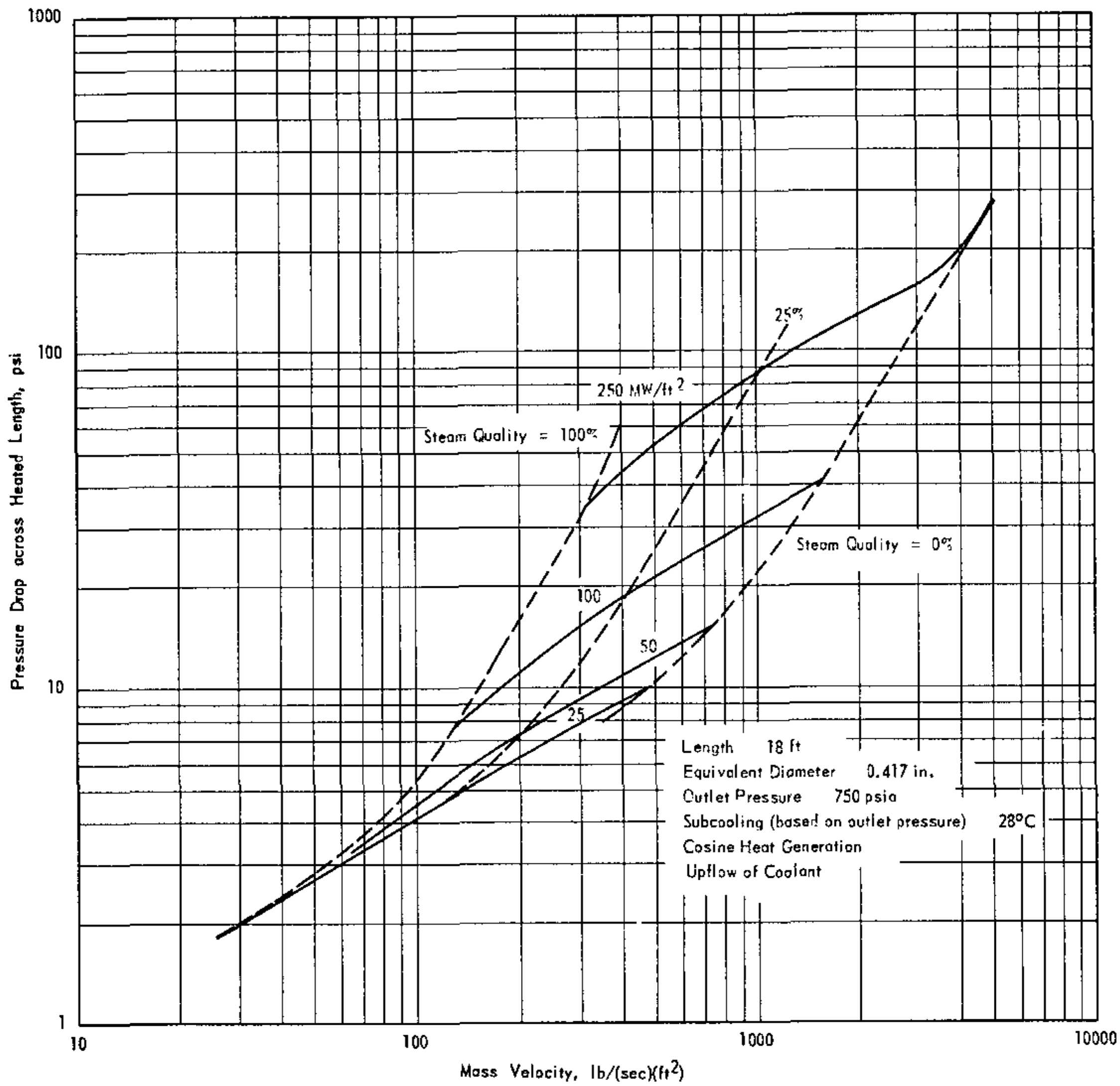


FIGURE 23 – CALCULATED EFFECT OF POWER ON THE PRESSURE DROP FOR FLOW OF BOILING WATER

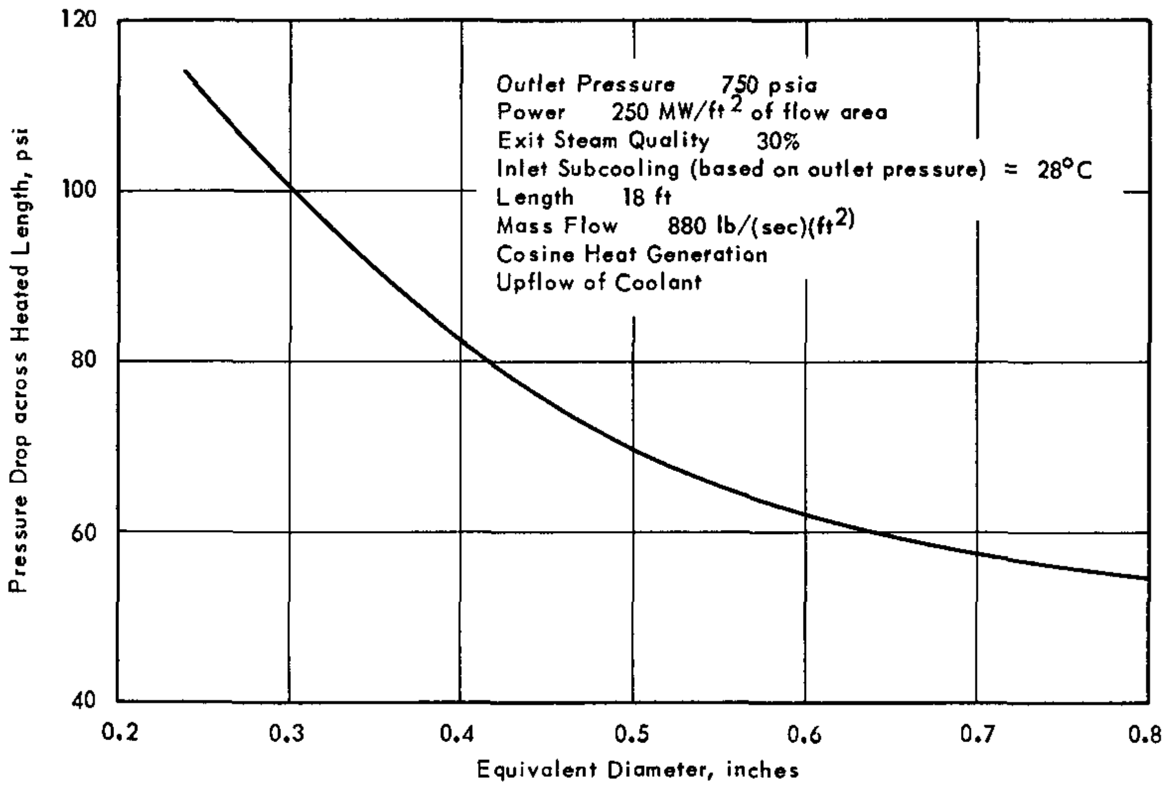


FIGURE 24 – CALCULATED EFFECT OF EQUIVALENT DIAMETER ON THE PRESSURE DROP FOR FLOW OF BOILING WATER