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AEC RESEARCH AND DEVELOPMENT REPORT

HEAVY WATER MODERATED POWER REACTORS

PROGRESS REPORT
JANUARY-FEBRUARY 1964

Technical Division
Wilmington, Delaware

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HEAVY WATER MODERATED POWER REACTORS
PROGRESS REPORT
JANUARY-FEBRUARY 1964

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Wilmington, Delaware

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ABSTRACT

The HWCTR was operated most of January for irradiation tests of various fuel assemblies. The reactor was shut down early in February for repair of a D_2O leak in one of the steam generators. The D_2O loss during reactor operation in January averaged 14 pounds per day, which is equivalent to an annual loss of 7% of the inventory.

An assembly of compacted UO_2 tubes was undamaged by irradiation to 10,000 MWD/tonne in the HWCTR; the free gas content of the tubes was somewhat lower than expected. Two Zircaloy-clad tubes of unalloyed uranium exhibited large local decreases in outer diameter during irradiation to 7,000 MWD/tonne.

Experimental data were obtained at Columbia University on heat flux limits for water flow in tubular channels at 750 to 1500 psia. Burnout measurements were made on single tubes and on three tubes connected in parallel to a common coolant supply.

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HEAVY WATER MODERATED POWER REACTORS
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INTRODUCTION

This report reviews the progress of the Du Pont development program on heavy-water-moderated power reactors. The over-all goal of the program is to advance the technology of these reactors so that they could be used in large power stations to generate electricity at fully competitive costs. Program emphasis is being placed on reactors that are cooled by liquid D_2O. The principal phases of the program are: (1) the irradiation of candidate fuels and other reactor components in the Heavy Water Components Test Reactor (HWCTR), (2) the development of low-cost fuel tubes for use in large water-cooled reactors, and (3) the technical and economic evaluation of various reactor design concepts.

SUMMARY

Early in January, the HWCTR began operation at power with the second set of driver elements. Irradiation testing of fuel elements continued uneventfully until a shutdown was forced by a D_2O leak in a steam generator tube. Attempts to determine the nature and cause of the tube failure were unsuccessful. The leaking tube was isolated by plugging each end. The steam generator showed little evidence of any kind of attack.

During HWCTR operation at power during January, the total D_2O loss averaged 14 pounds per day, which is equivalent to an annual loss of 7% of the inventory. This is the lowest loss rate experienced in the HWCTR thus far.

Postirradiation examination of an assembly of compacted UO_2 tubes from the HWCTR revealed no indications of sheath collapse or other significant mechanical damage. The tubes were examined after successful irradiation to a maximum exposure of 10,000 MWD/tonne at a time-averaged maximum $\int k d\theta$ of 26 watts/cm. Dimensional changes were small, and the only visible damage was minor wear (from housing tube ribs) on two of the tubes. The total amount of free gas in the tubes was about 20% lower than was expected on the basis of gas release measurements on other UO_2 assemblies.

Two Zircaloy-clad tubes of unalloyed uranium exhibited large local changes in outer diameter when inspected after

irradiation to 7000 MWD/tonne at a uranium temperature of 500°C. Little change in outer diameter occurred at exposures up to 5000 MWD/tonne; at higher exposures, the outer surface of each tube became very irregular. The maximum local decrease in diameter was 0.050 inch.

Additional data on heat flux limits with water flowing inside tubular channels were obtained at Columbia University. Experiments were conducted at 750 to 1500 psia with single electrically heated tubes, and at 1500 psia with three tubes coupled in parallel to inlet and outlet coolant plenums. The heat flux limits (incipient burnout) for three tubes in parallel were essentially the same as for a single-tube assembly. The limits are compared in Figures 7 and 8 with representative operating conditions for a liquid-D_2O-cooled reactor. The comparison indicates not only that tubular passages in current fuel designs have adequate margins from heat transfer burnout at normal operating conditions, but also that substantial deviations from normal operation could occur without burnout.

The basis for the current design study of a prototype D_2O-moderated power reactor was revised with the objective of achieving the minimum capital cost consistent with the main objectives of the prototype. Thorium control rods and blanket elements were eliminated from the design, and the reactor core is to be no larger than is necessary to demonstrate ripple factors, control rod manipulations, and fuel management techniques. It is anticipated that enough data could be obtained from operation of the prototype and from supporting experiments to demonstrate the essentials of the U^{233} breeding potential of D_2O reactors.

DISCUSSION

I. THE HEAVY WATER COMPONENTS TEST REACTOR (HWCTR)

The HWCTR is a D_2O-cooled-and-moderated test reactor in which candidate fuel assemblies and other reactor components are being evaluated under conditions that are representative of large D_2O-moderated power reactors.[1] Currently, fuel assemblies of uranium metal (coextruded with Zircaloy cladding) and assemblies of uranium oxide (mechanically compacted in Zircaloy sheaths) are being irradiated in this reactor.

A. REACTOR OPERATION

Operation of the reactor at power with the second set of driver elements began on January 7. The maximum power was limited to about 37 MW by a maximum of 540°C on the central metal temperature in the hottest driver. The powers in the test fuel were 25% lower under these conditions than they were at the end of the cycle with the first set of drivers. As the U^{235} in the new drivers is consumed, the neutron flux and the specific powers in the test fuel will be increased gradually.

The reactor was shut down on February 4 when a D_2O leak developed in the No. 1 steam generator. The leak rate into the secondary cooling water in this generator increased from its usual value of 5 lb/day to about 150 lb/day. The increased leakage was detected by a gamma monitor on the secondary water system; the reactor was shut down promptly, so only a small amount of D_2O was lost. Subsequent investigation revealed that the leak was in one of the carbon steel tubes at a point between the tube sheet and the first baffle. Attempts to view the leakage point were unsuccessful; however, corrosion on the secondary side of this and other tubes in the vicinity of the leak did not appear to be severe and is not thought to have caused the leak. The leaking tube was plugged and the steam generator was returned to service.

Reactor startup operations subsequent to repair of the steam generator were halted by the discovery that a plywood plug had been left in the main D_2O cooling system upon completion of the repair. Pieces of the plug entered the reactor and partially blocked the inlet screens of the fuel assemblies. In addition, the moderator was contaminated. At the end of this report period, a program for cleanup of the system was being formulated.

During the reactor shutdown, one assembly of short uranium tubes (SMT-1-3, described in Table III) was removed from the reactor for visual examination. The assembly appeared to be in good condition, and was reinserted for further irradiation. In addition, two new fuel assemblies were charged. One of the assemblies contained 2.54-inch-OD tubes of compacted UO_2; these tubes are the SOT-9 elements described in DP-885. They are expected to operate at a maximum $\int k d\theta$ of 30 to 40 watts/cm during their lifetime in the reactor. The second assembly contained a single 40-inch-long tube of uranium alloyed with 1.5% molybdenum. In earlier capsule tests in a Savannah River production reactor, the U - 1.5% Mo exhibited better volume stability than unalloyed uranium. On the basis of these tests and data on hardness properties at operating temperatures (see DP-735, DP-755, DP-775, DP-785), U - 1.5% Mo is regarded as a promising alternative to unalloyed uranium for D_2O-cooled power reactors.

B. D_2O LOSSES

The D_2O loss from the HWCTR for the operating period from January 7 to February 4 was 14 lb/day, which is 9 lb/day lower than for any previous operating period of comparable length. About 5.5 lb/day was lost through the steam generators and purge cooler, leaving 8.5 lb/day (4.4% of inventory per year) as unaccounted losses. The repairs to the steam generator in December reduced the leakage from this source by 2.6 lb/day. The remainder of the reduction in leak rate was due to tightening of flanges and valve packings and to improvements in the leak collection system.

II. REACTOR FUELS

A. GENERAL

Two types of Zircaloy-clad fuel elements are currently under development in the Du Pont program on D_2O-cooled power reactors: mechanically compacted tubes of uranium oxide and coextruded tubes of uranium metal. In the development of these two alternative fuels, primary attention is being given to fuel element designs and fabrication methods that offer promise of low fabrication costs when the elements are produced in the volume required for several full-scale reactors. At present, the uranium oxide tubes are receiving program emphasis pending the outcome of current irradiations of both types of fuel in the HWCTR.

In addition to the program on uranium fuels, a modest irradiation program has been initiated on thorium fuels (see DP-885). The ultimate objective of this program is to develop fuel elements that would be suitable for use in D_2O-thorium breeder reactors.

B. POSTIRRADIATION EXAMINATION OF URANIUM OXIDE TUBES

Test assembly SOT-1-3 was disassembled after successful irradiation in the HWCTR to a maximum exposure of 10,000 MWD/tonne, and examination of the fuel was started. This assembly contained seven Zircaloy-clad tubes of compacted UO_2 (2.1 inches in OD, 1.5 inches in ID, 14 inches long). Four of the tubes were fabricated by vibratory compaction alone, and three were fabricated by vibratory compaction and swage compaction. The time-averaged maximum thermal rating ($\int kd\theta$) was 26 watts/cm. Two other assemblies of the same design are being irradiated to goal exposures of 20,000 and 30,000 MWD/tonne (SOT-1-2 and SOT-1-4; see Table III).

There were no indications of sheath collapse in any of the tubes. The only visible damage to the tubes was minor wear on the two tubes at the top end of the fuel column. The wear was caused by ribs on the Zircaloy housing tube. The maximum depth of cladding wear was 0.005 inch. This damage is about the same as that observed halfway through the irradiation test (see DP-865); at that time, the topmost fuel tube was replaced with a steel dummy piece because of vibration damage.

Observed changes in dimensions of the tubes were small and are of questionable significance. The greatest average change in outer diameter was +0.002 inch (0.1%). Changes in inner diameter and tube volume are uncertain because of discrepancies in preirradiation measurements, but are believed to be less than 0.3% and 1%, respectively.

The total amount of free gas in the tubes was about 20% lower than was expected on the basis of gas release measurements on other UO_2 assemblies (see DP-875). The free gas content ranged from 48 to 135 cc (STP) per kilogram of UO_2 and increased with thermal rating of the tubes. The highest gas release corresponds to calculated maximum internal pressures of 420 and 180 psig during irradiation and during reactor shutdown, respectively. The former value is 800 psi lower than the coolant pressure, and the latter is 1000 psi lower than the estimated pressure for sheath collapse. Fission gas releases are being determined for the SOT-1-3 tubes. When these data are available, predictions can be made of the pressure buildup in companion fuel tubes that are being irradiated to higher exposures. The total free gas contents of the SOT-1-3 tubes indicate that the companion elements should be able to operate to 30,000 MWD/tonne before an internal pressure of 900 psi is reached. This pressure is 3/4 of the HWCTR coolant pressure and is regarded as a safe operating limit.

C. POSTIRRADIATION EXAMINATION OF URANIUM METAL TUBES

Preliminary examinations were made on two Zircaloy-clad tubes of unalloyed uranium that were irradiated successfully to a maximum exposure of 7000 MWD/tonne at time-averaged maximum temperatures of 500 and 514°C. Both of the tubes exhibited unexpectedly large decreases in outside diameter in the region of maximum exposure. The decreases occurred in the later stages of irradiation, as evidenced by the fact that no significant diameter decreases were observed when one of the tubes was inspected earlier at a maximum exposure of 3700 MWD/tonne (see DP-875). No data are available yet on changes in inside diameter.

Longitudinal profiles of the outside diameters of the two fuel tubes are shown in Figure 3. Calculated exposures at one-foot increments of tube length are also displayed. Little change in diameter was observed at exposures as high as 5000 MWD/tonne*; at higher exposures, the outer surface of each tube became very irregular. The maximum local decrease in diameter was 0.050 inch. Plots of the outside periphery of the tubes at selected locations are presented in Figure 4. From the appearance of the plots, the outside diameters probably increased slightly during early irradiation as a result of increasing core volumes. The inside diameters probably decreased also because of the smaller resistance to deformation offered by the inner cladding. A decrease in inside diameter of 0.030 inch would allow the inner surface of the fuel to contact the ribs of the inner housing. Any further decrease in inside diameter could occur only between the support points provided by the inner ribs and would be accompanied by a corresponding decrease in outside diameter. As a result, the outside surfaces of the fuel tubes resemble a four-lobed epicycloid with the lobes oriented with the ribs of the inner housing. The supposition that the inside diameter decreased more than the 0.030 inch necessary to contact the ribs of the inner housing was supported by the fact that the inner housing could not be separated from the fuel tube of one of the assemblies by a pulling force of 4000 pounds.

* and at temperatures as high as 425°C

Plans for further work are to cut a section from one of the fuel tubes for metallographic examination, remove the inner housing from the remaining pieces of the cut fuel tube, and measure the inside diameters of the fuel pieces.

III. BURNOUT HEAT TRANSFER IN WATER-COOLED REACTORS

A. TUBULAR CHANNELS

Additional experimental data on heat flux limits with water flowing inside tubular channels were obtained at the Engineering Research Laboratories of Columbia University. These data extend the ranges of pressure and tube diameter beyond those presented in DP-855. The results are applicable to the innermost channel of a typical fuel assembly of concentric tubes (see Section C) and to special studies (see Section B).

The new data, which are shown in Table IV, are for 0.504-in.-ID tubes with lengths of 36 and 76 in. and for 0.245-in.-ID tubes 36 in. long. The tests were conducted at pressures of 750, 1000, 1200, and 1500 psia, mass velocities of 0.5 to 14 million lb/(hr)(ft^2), and local coolant conditions ranging from 80°C subcooling to 50% vaporization (quality). The heat flux limit was observed with a burnout detector (DP-555).

As presented in Figures 5 and 6, the data for tubes show the following: (1) in the boiling region, the heat flux limit decreases as the pressure is increased from 750 to 1500 psia; (2) the effect of pressure on the heat flux limit decreases as the coolant quality is reduced to zero; (3) an inversion of the dependency of heat flux limit on mass velocity occurs for mass velocities below about 4 million lb/(hr)(ft^2) as the coolant condition changes from subcooled to boiling; (4) essentially no difference exists in heat flux limit for 36-in. length versus 76-in. length; (5) significantly higher heat flux limits are realized with tubes of smaller diameter (0.245 versus 0.504-in. ID); and (6) good reproducibility was obtained for the several test assemblies.

A general correlation of these data is being sought at Columbia University.

B. STABILITY

A fuel assembly of concentric tubes presents several long coolant channels in parallel. Moreover, in a power reactor, many such assemblies are connected in parallel between a common inlet header and common outlet header. Heat transfer tests usually differ from the reactor situation in both of these respects in that the tests are made with a single coolant channel in which flow is maintained constant by upstream throttling. An assembly consisting of three electrically heated tubes, each 0.495-in. ID with heated length of 87.9 in., closely coupled in parallel to inlet and outlet plenums was therefore fabricated and tested at Columbia University. The test loop was also modified to enable simulation of several such assemblies in parallel. Essentially the same heat flux limits were obtained for three tubes in parallel as for a single-tube assembly, both for operation at constant pressure difference (ΔP) and for operation at constant flow.

The tests for the three-tube assembly were conducted at 1500 psia. The data, which are presented in Table V, are plotted together with the data for single-tube assemblies (see Section A) in Figures 5a, 5c, 5d, and 5e for mass velocities of 1 to 4 million $lb/(hr)(ft^2)$. To simulate the operation at constant ΔP, a relatively large flow was diverted around the test section; the bypass flow was withdrawn downstream of the throttling valve but upstream of the flow meter. The ratio of the bypass flow to the flow through the heated assembly ranged from 3 to 5.

No attempt was made in these tests to measure the flow to the individual tubes of the assembly. Other tests are planned which will have a turbine flow meter in the upstream end of one or more tubular channels in order to determine the conditions for any oscillation in flow among channels. However, the effect of parallel channels and assemblies upon the heat flux limit, which could be measured quite reproducibly with the burnout detector for each tube, is the prime concern. This effect was shown to be negligible.

C. APPLICATION TO FUEL ASSEMBLY DESIGN

In order to achieve as high a thermal efficiency as possible, a liquid-cooled power reactor is designed with coolant effluent temperatures close to boiling. This objective must be reconciled, however, with the sharp decrease in heat flux limit as the saturation temperature is approached and exceeded (see Section A). It is not enough to require that the heat flux for burnout at normal flow and power exceed the actual heat flux by a factor of two or three. Some margin must be allowed for accidental increases in power and decreases in flow, both of which tend toward effluent boiling. The margins from probable burnout for the tubular channel of a typical power reactor, cooled by liquid D_2O at 1500 psia, are depicted in Figure 7. The margins at a pressure of 1000 psia are shown in Figure 8. In both of these figures, the coolant enters the fuel assembly at the same temperature (264°C) and has the same exit enthalpy at normal coolant flow.

As shown in the figures, at each pressure a power increase of 50% or more is required for probable burnout at constant flow; similarly, a flow decrease of 50% or more is required for probable burnout at constant power.

IV. DESIGN STUDY OF A PROTOTYPE D_2O-MODERATED REACTOR

The Du Pont Engineering Department is conducting a design study of a prototype D_2O-moderated power reactor. Objectives and basic design parameters for this reactor were discussed previously in DP-875 and DP-885.

The objectives and design requirements have been reviewed in the light of the design data developed thus far and of the probable utilization of the prototype. As a result, several important changes have been made in the design basis. The principal guideline is that the capital cost of the prototype should be the minimum that is consistent with the following objectives:

- To demonstrate the construction and operation of a D_2O-cooled power reactor of a type that offers promise of competitive power generation costs when built in large sizes (≥500 MWe).

- To demonstrate the breeding potential of D_2O reactors when thorium is used as the fuel.

To decrease the capital cost of the prototype, the actual demonstration of a breeding ratio of at least 1.00 is being abandoned as a design requirement. Therefore, thorium control rods and a radial thorium blanket can be eliminated. It is anticipated that enough data can be obtained from prototype operation with thorium fuel and from supporting experiments to predict reasonably well the breeding ratio that could be obtained in a reactor equipped with thorium control rods and a blanket.

The flat zone of the prototype will be made no larger than is necessary for demonstration of ripple factors, control rod manipulations for radial flattening, and fuel management techniques. The reactor tentatively selected to meet these requirements has 156 fuel positions and 21 control positions on a 10-in. square pitch. There are some 50 fewer fuel positions than in the design discussed in DP-885, but the size of the flat zone is unchanged.

The nominal power rating of the reactor will depend on the design rating of the UO_2 fuel tubes and on the achievable flux shapes. At a maximum fuel rating ($\int k d\theta$) of 38 watts/cm, the nominal reactor power will be about 300 MWe. The fuel rating is strongly dependent on the outcome of HWCTR irradiations, and a rating as high as 38 watts/cm may or may not be attainable with the first core loading of the prototype.

When a prototype reactor is actually built, there may be strong incentive to demonstrate breeding directly and/or to increase the power rating beyond that required to meet the objectives discussed here. The present philosophy of designing a "minimum-cost" reactor will provide a base point for assessing the economics of these optional features.

V. TOPICAL REPORTS

The following topical reports, all of which have been issued within the last year by Du Pont, contain information that is applicable to D_2O-moderated power reactors. Complete references for the reports can be found in the bibliography.

DP-819, Evaluation of Fused UO_2[4]

DP-832, Analysis of the Substitution Technique for the Determination of D_2O Lattice Bucklings[5]

DP-833, Efficacy of Experimental Physics Studies on Heavy Water Lattices[6]

DP-843, Static and Impact Tests on 15-Ton Cask for Shipping Irradiated Fuel[7]

DP-857, Forced-Flow Boiling in Rod Bundles at High Pressure[8]

DP-859, Irradiation of Tandem-Extruded Joints Between Zircaloy and Stainless Steel[9]

DP-864, Thorium-Fueled D_2O-Moderated Power Reactors[10]

TABLE I

Operating Chronology of HWCTR

Jan. 1	Began month at 26 MW
1	Shut down - feedwater system problems
1-7	Replaced cluster rod
7	Attained criticality
8-9	Attained 200°C and 36 MW
9-28	Operated at 36 MW
28	Attained 37 MW
Jan. 28 - Feb. 4	Operated at 37 MW
4	Shut down - leak in No. 1 steam generator
4-29	Steam generator repair, new fuel charging

TABLE II

Operating Summary of HWCTR

	January		February	
Time reactor critical, %	80.8		13.6	
Maximum power, MW	37		37	
Reactor exposure, MWD	Drivers	Test	Drivers	Test
For month	744	134	124	22
Total accumulated in Cycle H-2	757	136	882	157
Losses				
D_2O (100 mol %), lb	426		577	
% of inventory per year	7.3		9.9	
Deuterium, g	3408		942	
Helium, scf	57,930		31,340	

TABLE III

Test Fuel Irradiation Data
January-February 1964[(a)]

Reactor power	37 MW
Coolant pressure	1200 psig
Coolant inlet temperature	187°C
Moderator outlet temperature	200°C

					Maximum Nominal Conditions								Maximum Exposure, watt-days/g[(c)]	
Position	Element Number	Fuel Type	Fuel Assembly Description	Starting Date	Assembly Power[(b)], MW	Specific Power[(c)], watts/g	Heat Flux, pcu/(hr)(ft^2)	Outlet Temp., °C	Surface Temp., °C	Core-Clad Temp., °C	Core Temp., °C	$\int k d\theta$, watts/cm	Attained	Goal
37	CANDU	Oxide Rod	Five 19-rod bundles of natural UO_2 pellets	10-5-62	0.90	28.0	137,000	211	216	237	-	28.1	5,385	10,000
38	SOT-6-2	Oxide Tube	Seven 14" long, 2.5" OD tubes of natural UO_2	12-29-63	0.41	36.8	208,000	210	232	281	-	22.7	970	5,500
40	SOT-1-2	Oxide Tube	Seven 14" long tubes of 1.5% enriched UO_2[(d)]	10-5-62	0.38	49.8	223,000	193	223	287	-	17.8	12,795	25,000
42	OT-1-4	Oxide Tube	Single 1.5% enriched UO_2 vibratory and swaged tube	10-29-63[(e)]	0.70	72.5	336,000	199	242	342	-	26.7	6,330	22,000
55	OT-1-2	Oxide Tube	Same as position 42	10-3-63[(f)]	0.48	45.0	210,000	194	219	282	-	16.7	10,425	22,000
56	RMT-1-2	Metal Tube	Unalloyed, natural uranium 60-mil Zircaloy clad tube	12-29-63	0.44	42.9	223,000	217	272	410	442	-	1,155	6,600
57	SMT-1-2	Metal Tube	Ten 11-1/4" alloyed (Fe, Al) natural uranium tubes	5-19-63	0.45	33.4	258,000	206	281	340	393	-	3,390	8,800
58	SOT-1-4	Oxide Tube	Same as position 40	7-13-63	0.64	82.3	367,000	197	244	350	-	29.0	7,210	13,000
59	OT-1-7	Oxide Tube	Same as position 42	7-13-63	0.61	60.0	279,000	197	233	317	-	22.2	6,315	22,000
60	SMT-1-3	Metal Tube	Five 11-1/4" alloyed (Fe,Al,Si) natural uranium tubes	7-13-63	0.31	32.0	248,000	201	278	334	385	-	3,450	8,800

(a) Data taken on 2-4-64; exposures as of 2-29-64.
(b) "Flow-ΔT" power calculation; does not include moderator heating (gamma and neutron absorption).
(c) These values are based on an assembly power of 1.09 times "Flow-ΔT" power to include moderator heating.
(d) Originally contained eight tubes. Top tube replaced with dummy fuel piece after vibration damage was observed on top fuel pieces.
(e) Irradiation testing began on 1-6-63; interrupted on 4-3-63; and recommenced on 12-29-63.
(f) Irradiation testing began on 10-5-62; interrupted on 6-20-63; and recommenced on 10-3-63.

TABLE IV

Heat Transfer Burnout Conditions for Vertical Upflow of Water Inside Tubes

Data from Engineering Research Laboratories, Columbia University[a]

First 0.504-inch ID by 76-inch-Long Test Section

Run No.	Inlet Temperature, °C	Exit Pressure, psia	Power, MW	Flow, gpm	Mass Flow, 10^6 lb/(hr)(ft^2)	Exit Steam Quality, %	Exit Subcooling, °C	Burnout Heat Flux[a], 10^6 pcu/(hr)(ft^2)
C63	227	1000	0.174	4.7	1.43	27.1	-	0.395
C64	204	1000	0.188	4.6	1.44	23.8	-	0.427
C65	203	1000	0.186	6.4	2.00	9.4	-	0.423
C66	279	1200	0.134	7.5	2.05	19.4	-	0.303
C67	262	1200	0.148	7.1	2.01	16.7	-	0.337
C68	251	1200	0.159	6.8	1.97	15.4	-	0.360
C69	227	1200	0.174	6.6	1.99	10.5	-	0.395
C70	192	1200	0.205	6.1	1.94	6.6	-	0.464
C71	280	1200	0.127	5.1	1.40	29.8	-	0.287
Second 0.504-inch ID by 76-inch-Long Test Section								
E1	273	1000	0.161	10.2	2.82	17.7	-	0.364
E2	249	1000	0.169	6.7	1.96	20.9	-	0.383
E3	224	1000	0.174	4.7	1.41	27.1	-	0.396
E4	203	1000	0.169	3.1	0.98	39.1	-	0.383
E5	78	1000	0.228	3.1	1.09	17.6	-	0.517
E6	280	1500	0.109	3.7	1.01	33.4	-	0.247
E7	286	1500	0.120	7.3	1.95	15.1	-	0.272
E8	260	1500	0.148	7.1	2.04	9.8	-	0.336
E9	261	1500	0.121	3.7	1.05	28.8	-	0.274
E10	237	1500	0.132	3.0	0.89	34.3	-	0.299
E11	234	1500	0.176	6.7	2.02	6.8	-	0.398
E12	214	1500	0.197	6.5	2.01	4.5	-	0.446
E13	215	1500	0.144	3.3	1.00	25.1	-	0.327
E14	188	1500	0.153	3.0	0.96	22.8	-	0.348
E15	192	1500	0.216	6.3	2.01	0.9	-	0.490
E16	282	1500	0.141	10.3	2.79	8.7	-	0.321
E17	261	1500	0.173	9.8	2.79	5.6	-	0.392
E18	238	1500	0.207	9.1	2.72	3.1	-	0.469
E19	217	1500	0.236	8.9	2.74	0.2	-	0.536
E20	189	1500	0.287	8.7	2.78	-1.6	2.8	0.651
E21	188	1500	0.185	4.5	1.44	8.8	-	0.419
E22	285	1500	0.171	14.7	3.92	7.0	-	0.389
E23	261	1500	0.231	14.1	4.01	3.4	-	0.523
E24	239	1500	0.274	13.5	4.01	0.2	-	0.622
E25	214	1500	0.331	13.0	4.02	-2.3	4.3	0.752
E26	192	1500	0.368	12.5	3.98	-5.5	12.0	0.836
E27	284	1500	0.274	25.7	6.89	5.0	-	0.621
E28	249	1500	0.346	24.5	7.15	-4.7	10.1	0.786
E29	224	1500	0.466	22.2	6.76	-5.0	10.8	1.056
E30	222	1500	0.156	4.7	1.45	11.7	-	0.354
E31	251	1500	0.137	5.0	1.46	16.0	-	0.312
E32	252	750	0.154	3.5	1.01	49.6	-	0.348
E33	253	750	0.172	4.9	1.41	39.1	-	0.391
E34	252	750	0.187	6.8	1.96	29.4	-	0.423
E35	229	750	0.205	6.7	1.99	25.4	-	0.466
E36	227	750	0.170	3.3	0.98	49.4	-	0.385
E37	226	750	0.190	4.7	1.40	36.0	-	0.432
E38	204	750	0.202	4.5	1.42	32.4	-	0.459
E39	204	750	0.183	3.2	1.00	46.6	-	0.416
E40	182	750	0.189	3.1	0.98	43.9	-	0.428
E41	181	750	0.214	4.3	1.40	29.7	-	0.485
E42	153	750	0.225	4.2	1.39	25.0	-	0.510
E43	160	750	0.207	3.0	0.99	43.8	-	0.469
E44	251	750	0.205	9.7	2.80	21.1	-	0.465
E45	249	750	0.213	14.0	4.05	13.3	-	0.483
E46	223	750	0.221	13.2	4.00	6.7	-	0.501
E47	224	750	0.218	9.2	2.78	15.2	-	0.495
E48	202	750	0.229	9.0	2.81	10.0	-	0.520
E49	204	750	0.217	6.3	1.92	21.9	-	0.492
E50	178	750	0.233	6.0	1.93	17.2	-	0.528
E51	188	750	0.240	8.9	2.83	7.3	-	0.544
E52	156	750	0.276	8.7	2.87	2.8	-	0.627
E53	148	750	0.252	5.8	1.93	12.7	-	0.571
E54	249	750	0.228	24.2	6.99	6.5	-	0.518
E55	227	750	0.273	22.6	6.81	2.4	-	0.621
E56	202	750	0.347	21.6	6.75	-0.3	1.1	0.788
E57	192	750	0.265	13.0	4.12	1.4	-	0.601
E58	182	750	0.424	21.7	6.96	-2.4	7.8	0.962
E59	156	750	0.537	21.6	7.13	-4.5	15.0	1.219
E60	168	750	0.315	12.0	3.91	0.7	-	0.715
E61	47	1000	0.268	3.7	1.32	6.8	-	0.608
E62	47	1000	0.325	5.2	1.86	-3.7	10.6	0.737
E63	71	1000	0.318	5.5	1.94	-1.4	4.0	0.722
E64	71	1000	0.396	7.5	2.65	-6.8	19.8	0.898
E65	96	1000	0.368	7.9	2.74	-5.6	16.2	0.834
E66	96	1000	0.440	11.5	4.01	-14.7	44.2	0.999
E67	116	1000	0.421	11.7	4.03	-11.4	33.8	0.956
E68	153	1000	0.384	12.3	4.06	-4.5	13.1	0.871
E69	154	1000	0.487	16.3	5.40	-6.0	17.4	1.106

TABLE IV (Continued)

0.245-inch ID by 36-inch-Long Test Section

Run No.	Inlet Temperature, °C	Exit Pressure, psia	Power, MW	Flow, gpm	Mass Flow, 10^6 lb/(hr)(ft^2)	Exit Steam Quality, %	Exit Subcooling, °C	Burnout Heat Flux[a], 10^6 pcu/(hr)(ft^2)
F1	207	1000	0.047	1.1	1.49	25.5	-	0.459
F2	178	1000	0.055	0.90	1.23	38.8	-	0.543
F3	178	1000	0.056	0.93	1.27	38.1	-	0.557
F4	190	1000	0.056	1.0	1.37	35.1	-	0.548
F5	156	1000	0.060	0.97	1.36	30.7	-	0.588
F6	111	1000	0.070	1.1	1.53	20.5	-	0.687
F7	214	1000	0.059	1.5	1.96	25.6	-	0.581
F8	183	1000	0.070	1.5	2.11	21.3	-	0.687
F9	137	1000	0.076	1.4	2.00	16.2	-	0.751
F10	119	1000	0.080	1.4	2.03	13.2	-	0.789
F11	221	1000	0.066	2.2	2.84	16.7	-	0.651
F12	199	1000	0.071	2.1	2.79	13.9	-	0.703
F13	186	1000	0.076	2.1	2.78	12.7	-	0.747
F14	160	1000	0.082	2.0	2.79	8.8	-	0.811
F15	134	1000	0.096	2.0	2.86	8.2	-	0.948
F16	264	1000	0.056	3.2	3.82	16.6	-	0.551
F17	240	1000	0.066	3.1	3.88	12.7	-	0.654
F18	218	1000	0.074	3.1	4.02	8.2	-	0.732
F19	199	1000	0.083	3.0	3.99	6.3	-	0.818
F20	180	1000	0.092	2.9	3.89	5.3	-	0.908
F21	263	1000	0.067	4.6	5.51	12.2	-	0.665
F22	244	1000	0.071	4.5	5.53	7.0	-	0.698
F23	222	1000	0.084	4.3	5.47	4.5	-	0.832
F24	199	1000	0.102	4.1	5.52	2.7	-	1.009
F25	266	1000	0.064	5.8	6.90	8.5	-	0.629
F26	238	1000	0.083	5.5	6.91	3.8	-	0.819
F27	221	1000	0.101	5.5	7.10	2.0	-	0.993
F28	259	1000	0.077	8.0	9.65	4.3	-	0.761
F29	246	1000	0.093	7.9	9.88	2.0	-	0.914
F30	233	1000	0.110	8.1	10.19	0.2	-	1.081
F31	234	1500	0.052	1.6	2.03	16.0	-	0.508
F32	207	1500	0.059	1.5	1.98	14.6	-	0.583
F33	157	1000	0.069	1.4	1.96	17.3	-	0.681
F34	184	1000	0.088	2.9	3.93	4.6	-	0.871
F35	240	1500	0.051	1.6	2.01	17.6	-	0.499
F36	215	1500	0.056	1.5	1.96	15.5	-	0.553
F37	196	1500	0.061	1.5	2.01	11.7	-	0.601
F38	183	1500	0.064	1.5	1.98	10.9	-	0.629
F39	173	1500	0.069	1.5	2.07	9.8	-	0.681
F40	238	1500	0.059	2.3	2.83	8.9	-	0.584
F41	214	1500	0.068	2.1	2.81	6.4	-	0.669
F42	206	1500	0.071	2.1	2.74	6.7	-	0.701
F43	185	1500	0.079	2.0	2.72	5.7	-	0.781
F44	165	1500	0.086	1.9	2.72	3.5	-	0.843
F45	272	1500	0.056	3.3	3.82	10.1	-	0.554
F46	246	1500	0.072	3.1	3.92	6.7	-	0.708
F47	228	1500	0.082	3.1	3.91	5.4	-	0.807
F48	211	1500	0.090	3.0	3.95	2.4	-	0.884
F49	189	1500	0.099	2.9	3.86	0.8	-	0.976
F50	263	1500	0.079	4.7	5.58	5.5	-	0.776
F51	247	1500	0.092	4.5	5.65	3.2	-	0.907
F52	231	1500	0.102	4.5	5.74	0.3	-	1.006
F53	259	1500	0.091	5.9	7.21	1.3	-	0.899
F54	244	1500	0.109	5.5	6.93	1.3	-	1.078
F55	150	1500	0.070	1.4	1.99	5.3	-	0.690
F56	119	1500	0.078	1.4	2.00	2.1	-	0.765
F57	85	1500	0.084	1.3	1.91	0.4	-	0.823
F58	52	1500	0.090	1.3	1.89	-3.1	6.2	0.889
F59	140	1500	0.093	2.0	2.82	-2.3	4.3	0.912
F60	119	1500	0.097	1.8	2.66	-2.5	4.8	0.952
F61	98	1500	0.109	2.0	2.91	-6.9	14.9	1.076
F62	166	1500	0.117	2.9	4.04	-0.7	1.2	1.158
F63	184	1500	0.106	2.9	3.98	1.1	-	1.048
F64	281	1500	0.053	3.5	3.97	10.9	-	0.522
F65	284	1500	0.062	4.8	5.47	8.5	-	0.612
F66	292	1500	0.062	5.6	6.22	9.3	-	0.610
F67	276	1500	0.077	5.7	6.64	5.7	-	0.763
F68	272	1500	0.072	4.7	5.59	6.4	-	0.708
F69	157	1500	0.118	2.7	3.86	-1.1	1.9	1.161
F70	137	1500	0.139	2.9	4.19	-2.5	4.8	1.368
F71	209	1500	0.120	4.3	5.60	-0.4	0.7	1.179
F72	205	1500	0.140	5.5	7.35	-6.3	13.9	1.377
F73	185	1500	0.133	4.4	5.99	-7.0	15.6	1.313

TABLE IV (Continued)

0.504-inch ID by 36-inch-Long Test Section

Run No.	Inlet Temperature, °C	Exit Pressure, psia	Power, MW	Flow, gpm	Mass Flow, 10^6 lb/(hr)(ft²)	Exit Steam Quality, %	Exit Subcooling, °C	Burnout Heat Flux(a), 10^6 pcu/(hr)(ft²)
G1	266	1000	0.094	2.5	0.72	43.0	-	0.449
G2	247	1000	0.100	2.4	0.71	41.1	-	0.478
G3	222	1000	0.106	2.3	0.72	35.7	-	0.507
G4	206	1000	0.112	2.3	0.73	33.1	-	0.538
G5	182	1000	0.114	2.1	0.69	30.3	-	0.545
G6	267	1000	0.094	3.6	1.01	29.2	-	0.449
G7	238	1000	0.104	3.5	1.05	21.8	-	0.496
G8	227	1000	0.108	3.4	1.03	20.9	-	0.517
G9	209	1000	0.112	3.3	1.01	18.0	-	0.538
G10	194	1000	0.117	3.2	1.01	15.2	-	0.562
G11	179	1000	0.124	3.1	1.00	13.9	-	0.595
G12	273	1000	0.093	5.1	1.41	21.1	-	0.446
G13	249	1000	0.102	5.0	1.45	14.6	-	0.489
G14	230	1000	0.109	4.8	1.44	10.9	-	0.524
G15	209	1000	0.120	4.5	1.40	8.3	-	0.574
G16	179	1000	0.133	4.3	1.39	3.2	-	0.636
G17	269	1000	0.093	7.3	2.02	12.1	-	0.444
G18	247	1000	0.105	7.0	2.04	6.7	-	0.502
G19	229	1000	0.118	6.7	2.02	4.0	-	0.568
G20	206	1000	0.137	6.5	2.02	0.4	-	0.659
G21	184	1000	0.150	6.2	1.99	-2.9	8.4	0.719
G22	266	1000	0.095	10.1	2.84	6.3	-	0.454
G23	242	1000	0.124	9.4	2.77	2.7	-	0.595
G24	222	1000	0.144	9.2	2.80	-0.6	1.7	0.693
G25	206	1000	0.163	9.0	2.80	-3.0	8.7	0.780
G26	182	1000	0.185	8.6	2.76	-6.7	19.7	0.888
G27	272	1000	0.098	14.7	4.07	4.8	-	0.470
G28	250	1000	0.136	13.7	3.97	1.4	-	0.654
G29	223	1000	0.181	13.5	4.10	-3.1	8.9	0.866
G30	268	1000	0.126	20.0	5.60	2.7	-	0.602
G31	254	1000	0.159	19.3	5.56	0.4	-	0.763
G32	234	1000	0.202	18.9	5.63	-2.9	8.4	0.967
G33	275	1000	0.126	25.4	6.99	3.4	-	0.603
G34	261	1000	0.163	24.5	6.95	0.9	-	0.783
G35	242	1000	0.217	23.8	7.00	-2.3	6.4	1.038
G36	280	1500	0.075	3.8	1.03	17.8	-	0.362
G37	246	1500	0.090	3.5	1.03	11.3	-	0.431
G38	219	1500	0.100	3.3	1.00	7.0	-	0.478
G39	201	1500	0.107	3.3	1.02	2.9	-	0.513
G40	183	1500	0.113	3.1	1.00	0.6	-	0.541
G41	277	1500	0.082	5.1	1.41	9.8	-	0.391
G42	259	1500	0.092	5.0	1.43	5.9	-	0.439
G43	234	1500	0.104	4.8	1.44	0.4	-	0.498
G44	212	1500	0.121	4.6	1.43	-1.9	3.2	0.579
G45	183	1500	0.134	4.3	1.37	-6.2	14.2	0.642
G46	278	1500	0.095	7.7	2.10	4.9	-	0.454
G47	256	1500	0.115	7.2	2.08	0.5	-	0.551
G48	236	1500	0.133	7.0	2.09	-2.9	5.8	0.639

(a) Incipient burnout identified with burnout detector (see DP-555, DP-855).

TABLE V

Heat Transfer Burnout Conditions for Parallel Channels

Data from Engineering Research Laboratories, Columbia University[3]

Test Pressure: 1500 psia

Test Section: Three electrically heated tubes in parallel between common plenums
Tube diameter: 0.495-in. ID, each tube
Heated length: 87.9 in., each tube
Water flows vertically upward inside the tubes

Run No.	Inlet Temperature, °C	Power, MW	Total Flow, gpm	Mass Flow, 10^6 lb/(hr)(ft²)	Exit Steam Quality, %	Heat Flux[a], 10^6 pcu/(hr)(ft²)
A. Operation with Constant Flow						
1	282	0.324	11.5	1.07	32.6	0.216
2	252	0.375	10.4	1.04	29.5	0.250
3	232	0.411	10.2	1.07	26.5	0.274
4	203	0.458	10.0	1.08	22.1	0.305
5	180	0.496	9.9	1.11	18.2	0.331
6	282	0.381	22.5	2.10	14.3	0.254
7	259	0.448	20.8	2.06	10.6	0.299
8	231	0.553	20.4	2.12	7.1	0.369
9	200	0.650	19.6	2.13	2.8	0.433
10	180	0.687	18.5	2.07	0.4	0.458
11	228	0.704	28.4	2.97	2.2	0.469
12	202	0.813	27.0	2.94	-0.6	0.541
13	180	0.945	26.9	3.01	-2.3	0.629
18	220	1.007	39.6	4.19	0.1	0.671
19	284	0.565	44.5	4.13	8.1	0.376
20	258	0.730	42.0	4.16	4.0	0.487
B. Operation with Constant ΔP[b]						
21	286	0.555	45.0	4.16	8.3	0.370
23	254	0.740	41.8	4.18	2.5	0.493
24	207	0.792	26.9	2.90	0.5	0.528

(a) Incipient burnout identified with burnout detector (see DP-555, DP-855).
(b) Operation with constant ΔP, to simulate several assemblies in parallel between headers, was accomplished by bypassing approximately 140 gpm around the test assembly.

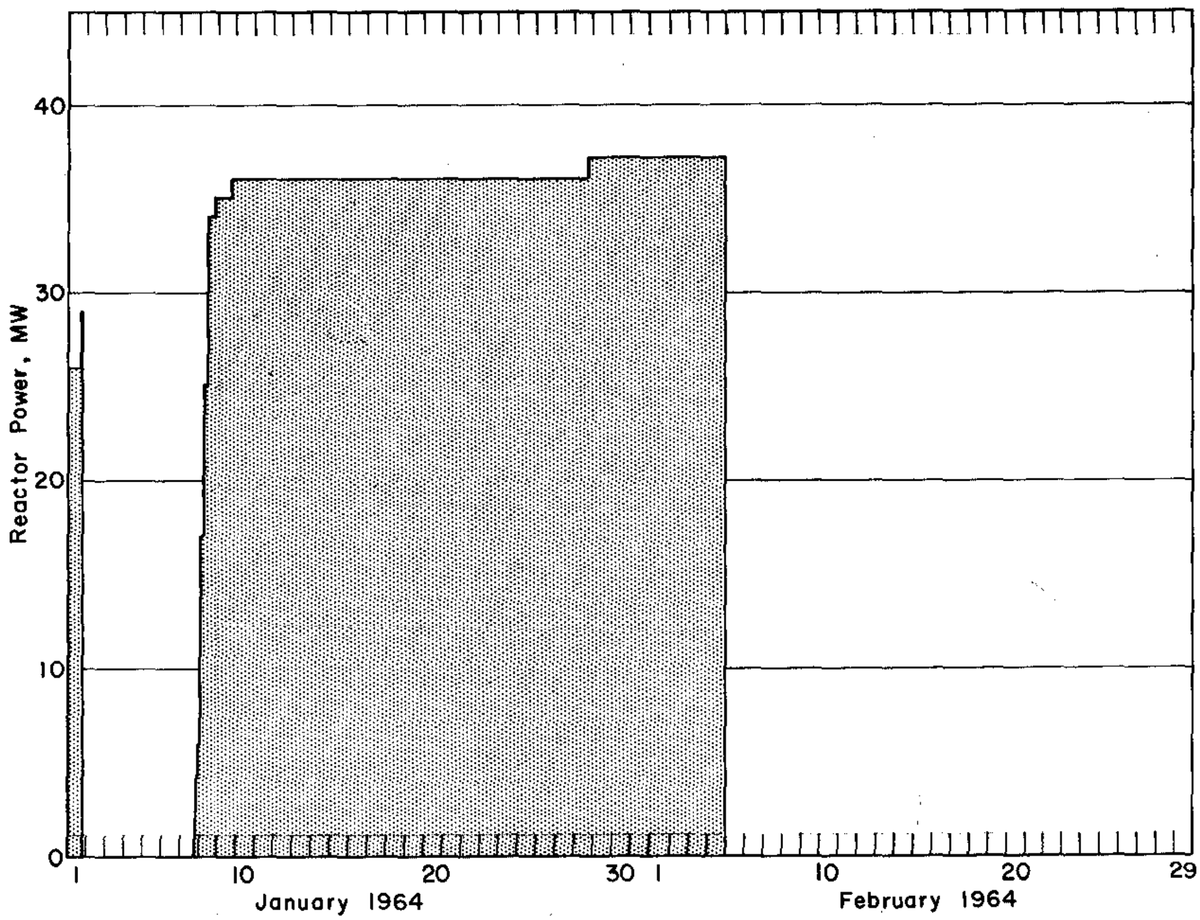


FIG. 1 OPERATING POWER OF HWCTR

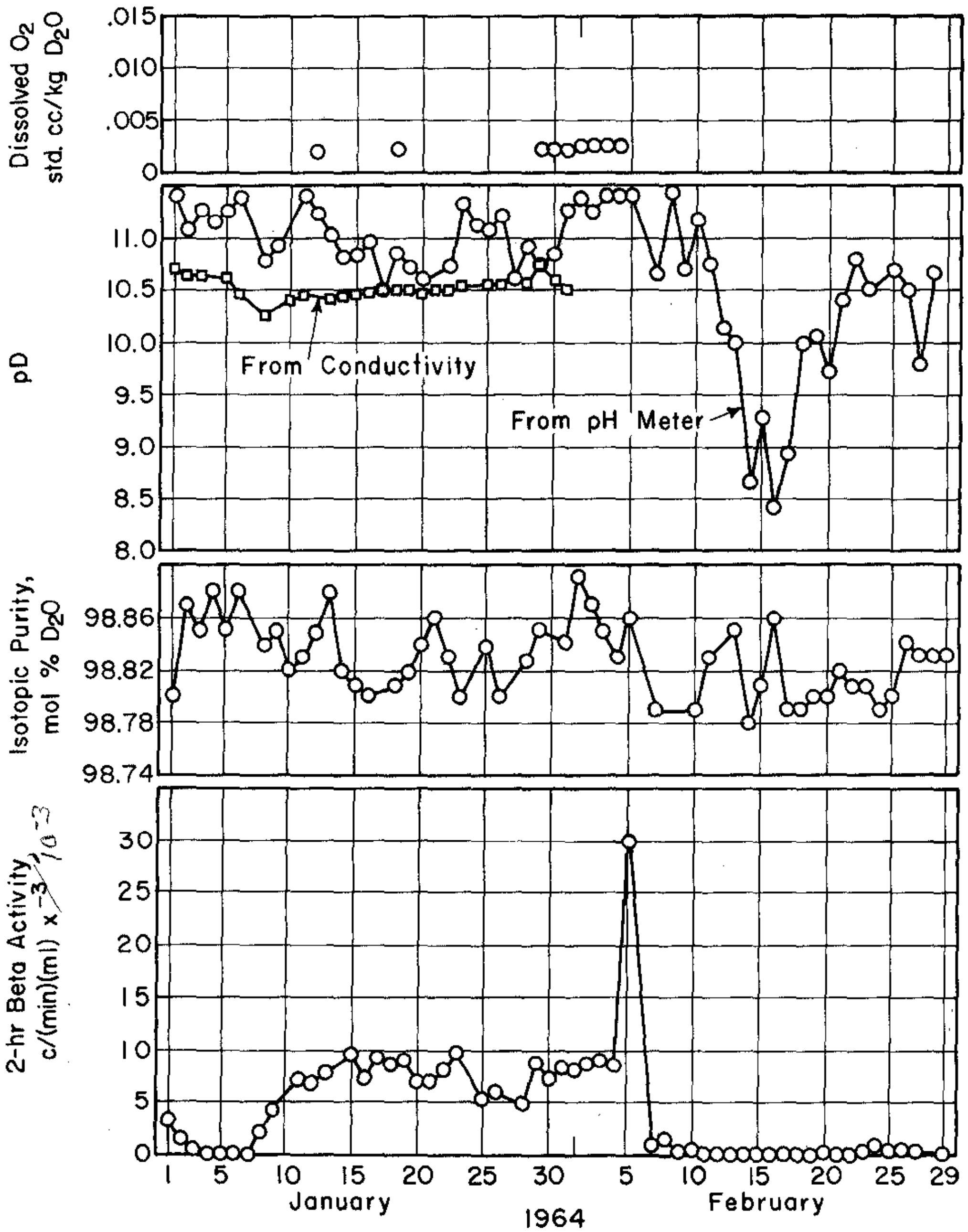


FIG. 2 HEAVY WATER QUALITY IN HWCTR

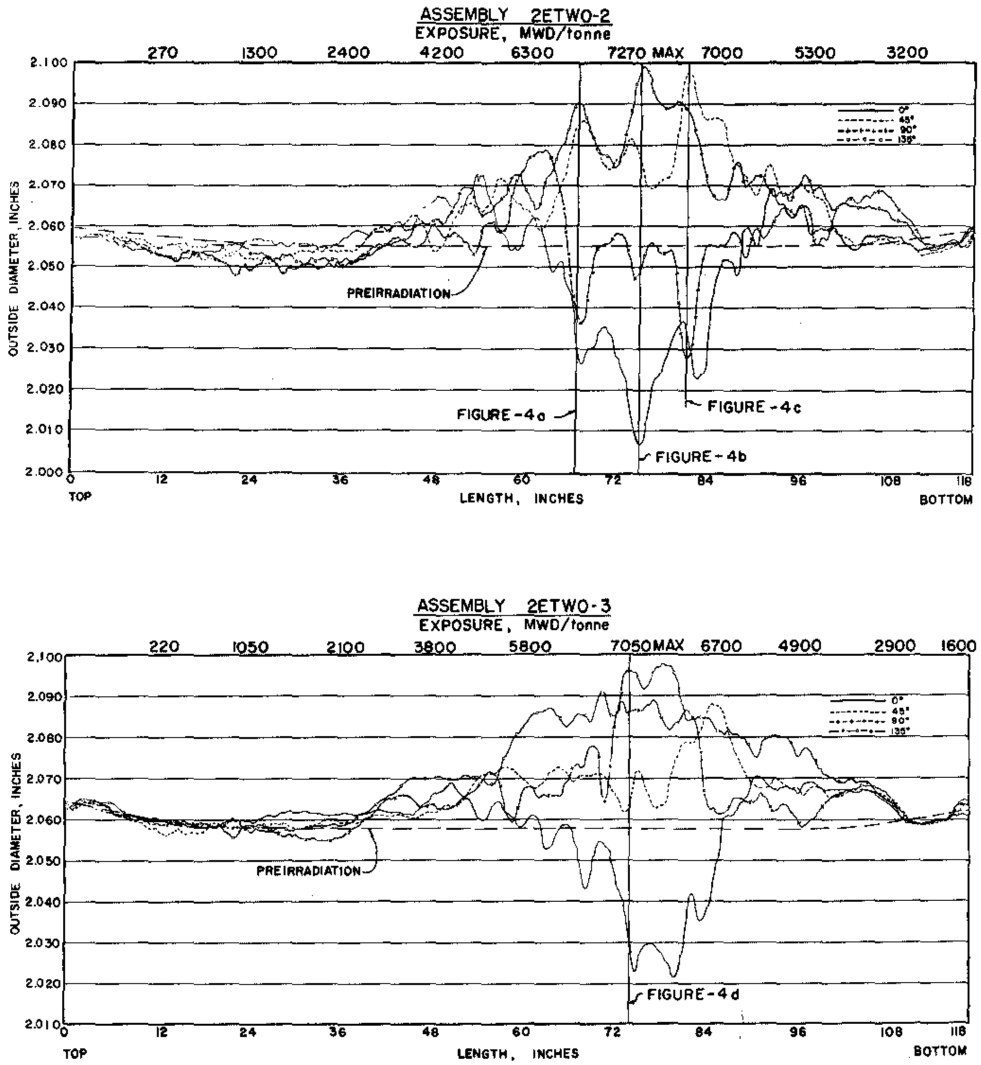
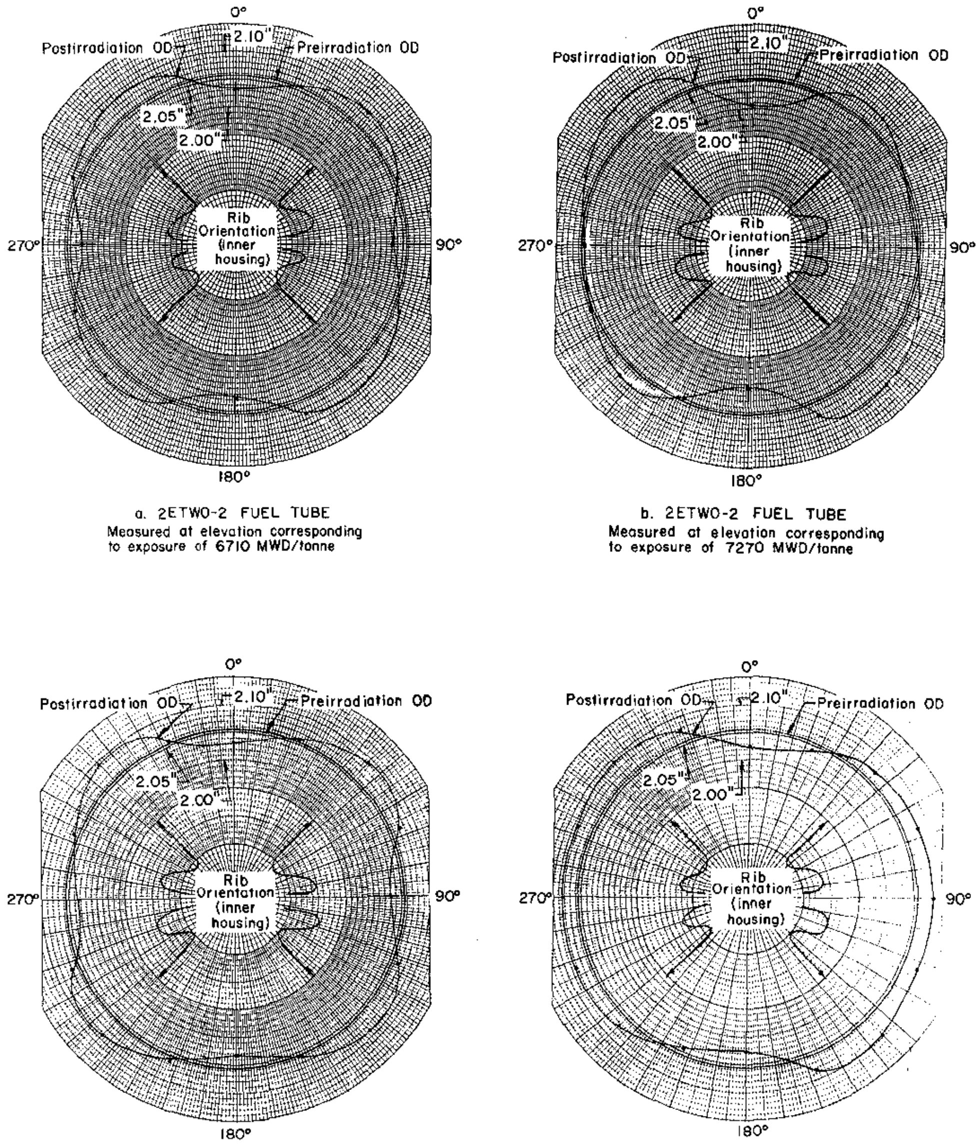


FIG. 3 OUTSIDE DIAMETERS OF UNALLOYED URANIUM TUBES AFTER IRRADIATION IN HWCTR



a. 2ETWO-2 FUEL TUBE
Measured at elevation corresponding to exposure of 6710 MWD/tonne

b. 2ETWO-2 FUEL TUBE
Measured at elevation corresponding to exposure of 7270 MWD/tonne

c. 2ETWO-2 FUEL TUBE
Measured at elevation corresponding to exposure of 7150 MWD/tonne

d. 2ETWO-3 FUEL TUBE
Measured at elevation corresponding to exposure of 7050 MWD/tonne

FIG. 4 CROSS-SECTIONAL PLOTS OF OUTSIDE DIAMETER
(See Fig. 3 for location of cross sections.)

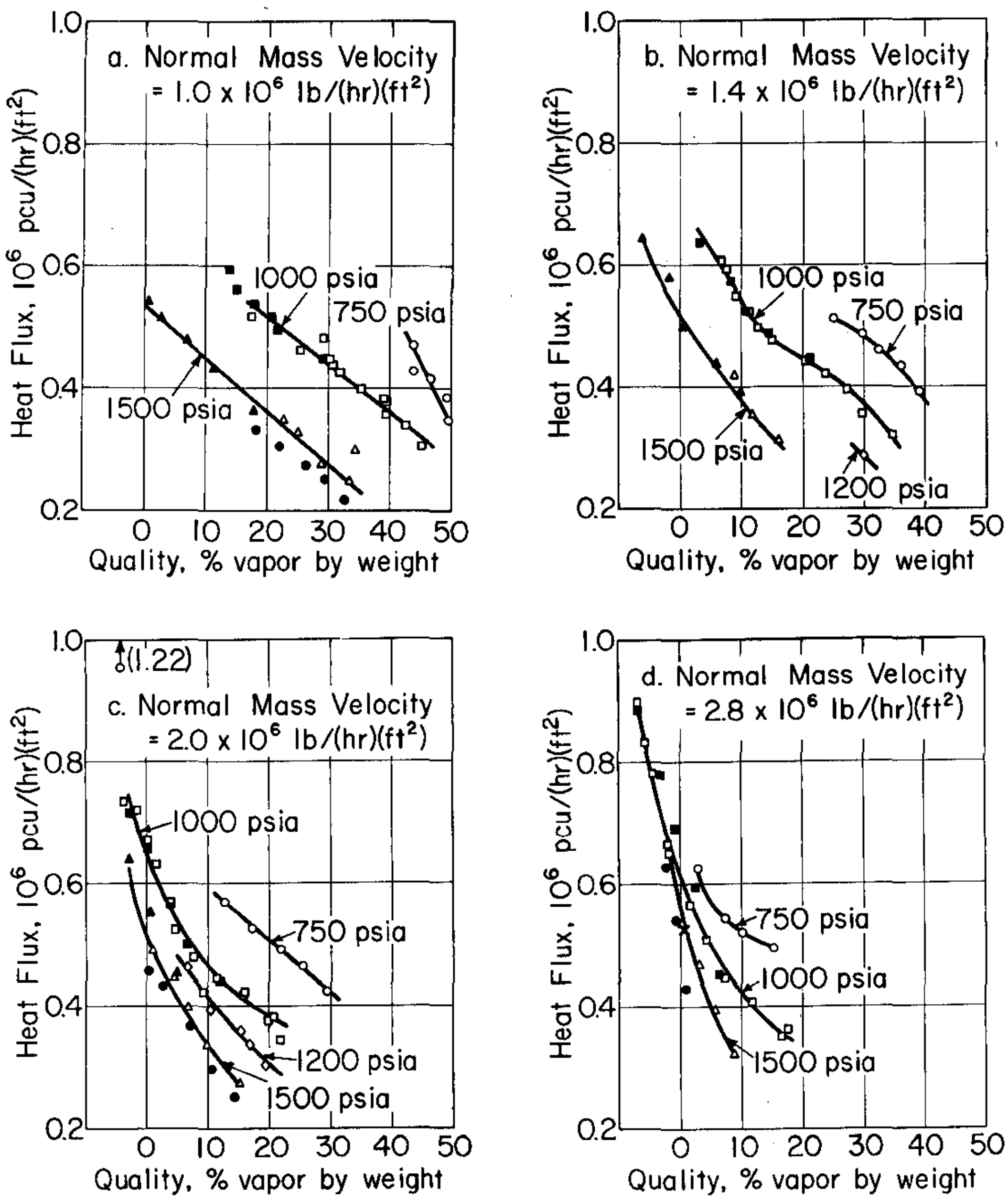


FIG. 5 HEAT FLUX LIMITS FOR 1/2-INCH-DIAMETER TUBES
Columbia University Data: see Table IV and DP-855

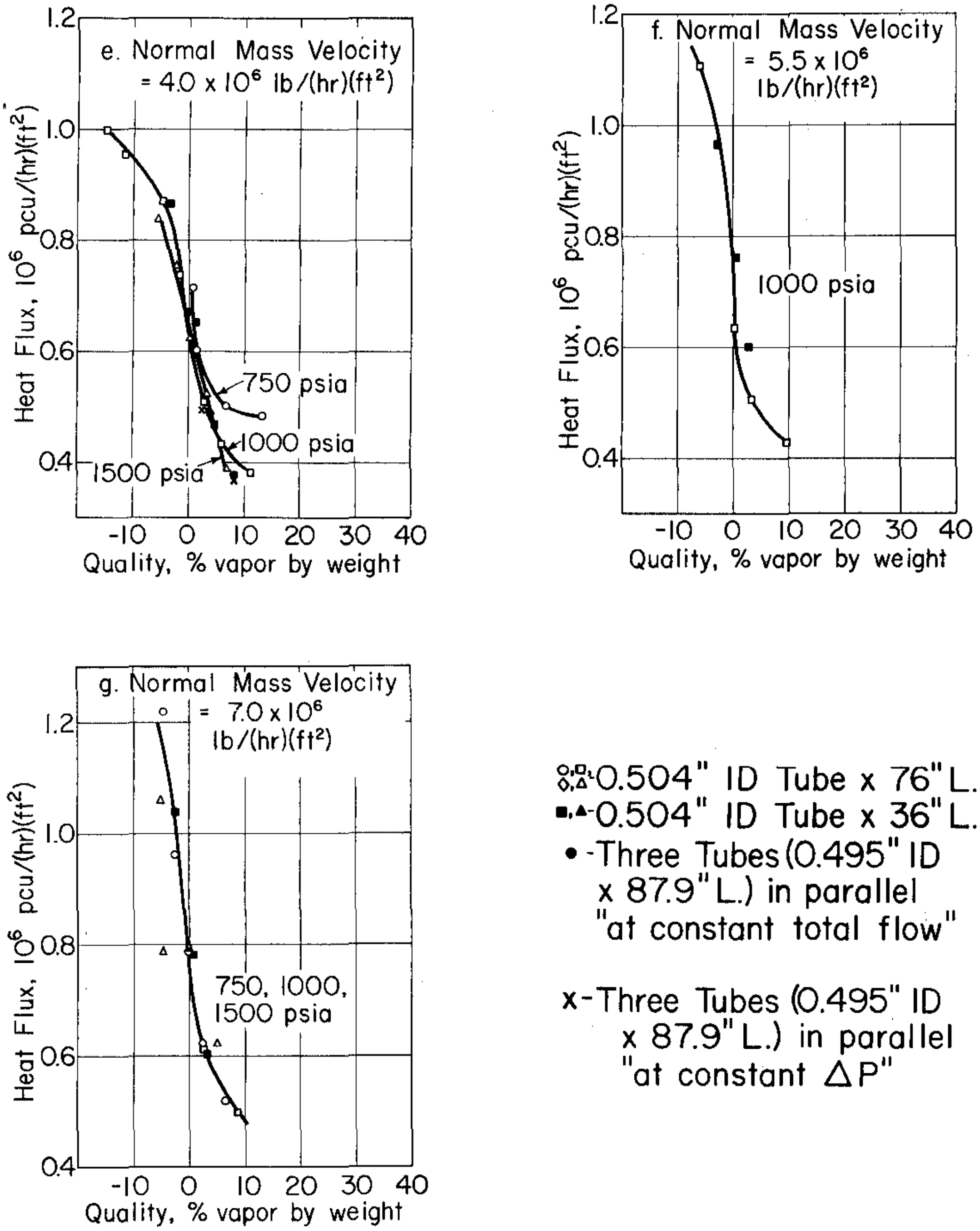

e. Normal Mass Velocity = 4.0 x 10⁶ lb/(hr)(ft²)
750 psia
1000 psia
1500 psia
Heat Flux, 10⁶ pcu/(hr)(ft²)
Quality, % vapor by weight
f. Normal Mass Velocity = 5.5 x 10⁶ lb/(hr)(ft²)
1000 psia
Heat Flux, 10⁶ pcu/(hr)(ft²)
Quality, % vapor by weight
g. Normal Mass Velocity = 7.0 x 10⁶ lb/(hr)(ft²)
750, 1000, 1500 psia
Heat Flux, 10⁶ pcu/(hr)(ft²)
Quality, % vapor by weight
-0.504" ID Tube x 76" L.
-0.504" ID Tube x 36" L.
-Three Tubes (0.495" ID x 87.9" L.) in parallel "at constant total flow"
x-Three Tubes (0.495" ID x 87.9" L.) in parallel "at constant ΔP"

FIG. 5 (cont'd)

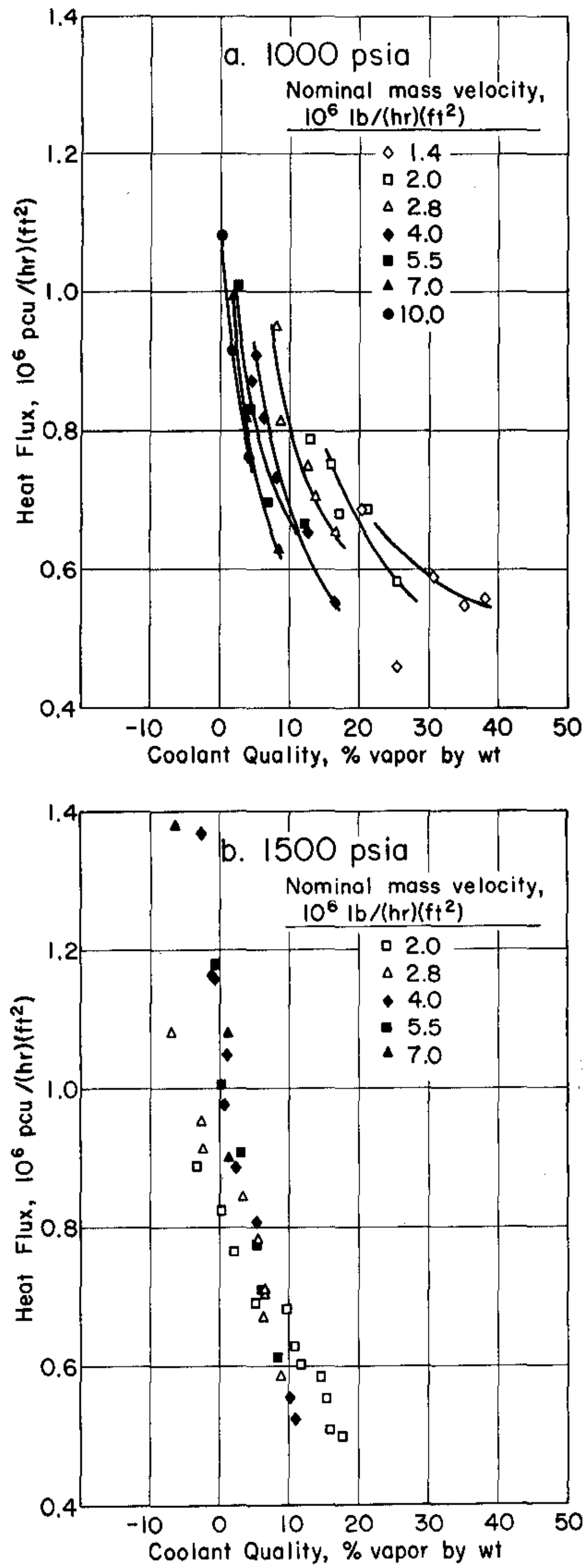


FIG. 6 HEAT FLUX LIMITS FOR 1/4-INCH-DIAMETER TUBES

Columbia University Data; see Table IV
Test Section: 0.245-in. ID X 36 in. long

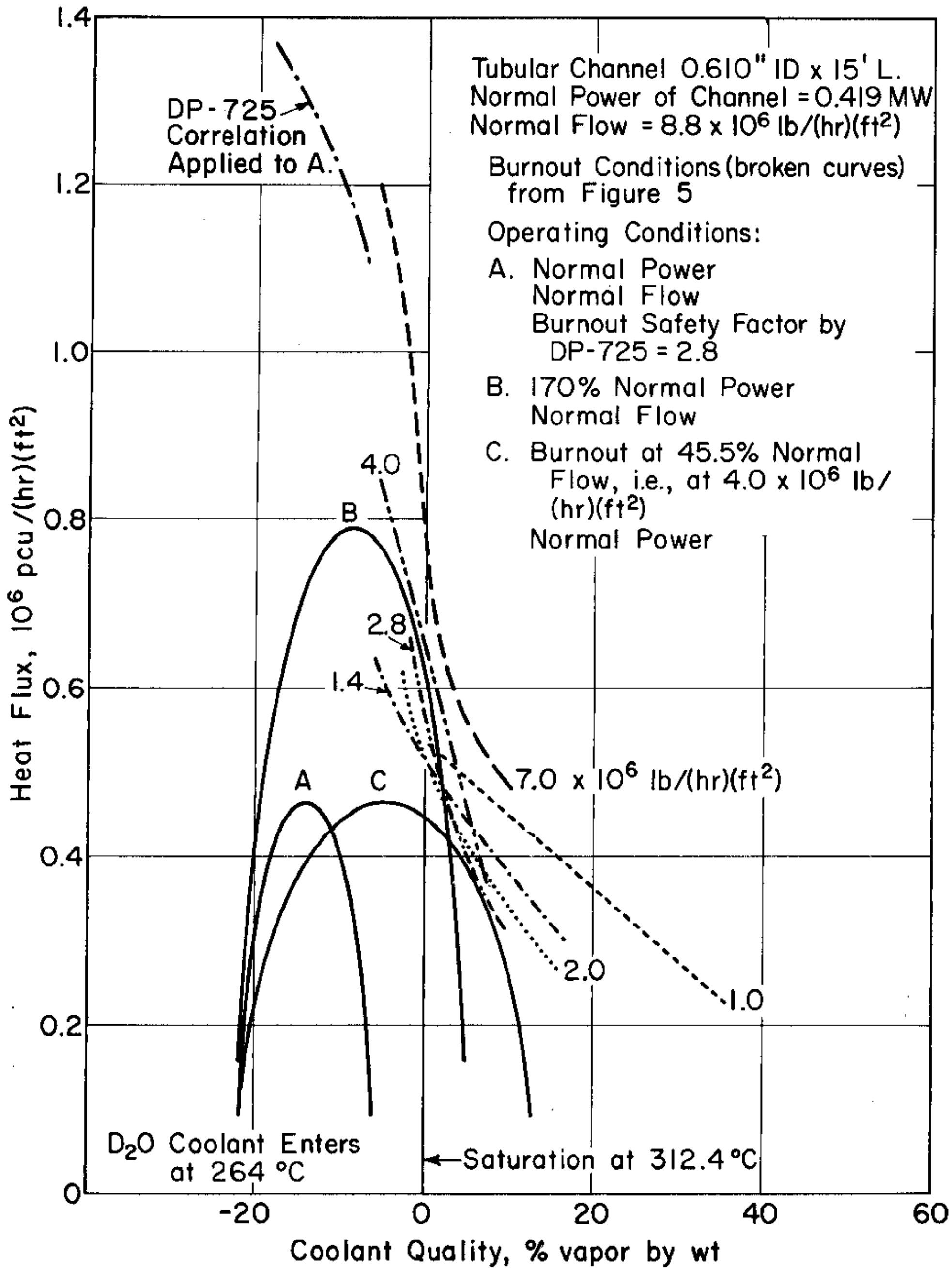


FIG. 7 APPROACH TO BURNOUT FOR A TYPICAL POWER REACTOR CHANNEL AT ~~1000~~ 1500 psia

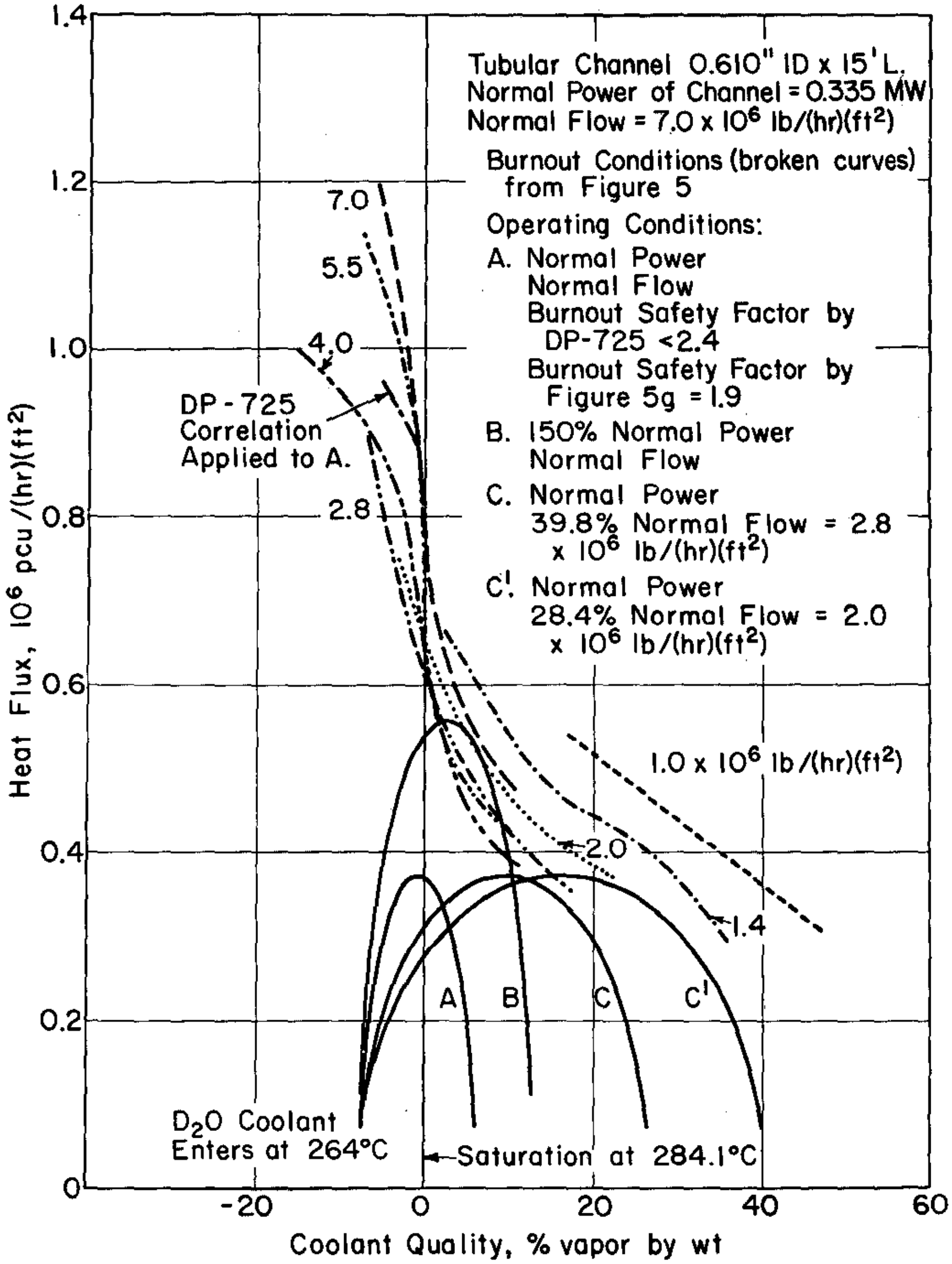


FIG. 8 APPROACH TO BURNOUT FOR A TYPICAL POWER REACTOR CHANNEL AT ~~1500~~ 1000 psia

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