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AEC RESEARCH AND DEVELOPMENT REPORT

HEAT TRANSFER STUDIES FOR THE D_2O POWER REACTOR PROGRAM

A TERMINAL STATUS REPORT OF THE WORK AT COLUMBIA UNIVERSITY

J. S. NEILL

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A Terminal Status Report of the Work at Columbia University

by

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August, 1966

E. I. DU PONT DE NEMOURS & COMPANY
SAVANNAH RIVER LABORATORY
AIKEN, SOUTH CAROLINA

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ABSTRACT

Experiments relating to the heat transfer capability of fuel assemblies for a nuclear power reactor were conducted at the Heat Transfer Research Facility of Columbia University. The heat flux at burnout with forced flow of subcooled and boiling water at high pressure was measured for several types of assemblies: multirod bundles in a circular housing, tubular channels, internally heated annuli, tubes-in-parallel, and tubes cooled on both the inner and outer surfaces. The relation of the particular tests to the development program for a heavy-water-moderated power reactor is discussed. The results, together with the recent results from other laboratories, are applied to a nested-tube-assembly design for a D_2O-cooled power reactor. Areas for further investigation are pointed out.

ACKNOWLEDGMENT

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Joseph E. Casterline, Manager

Bruce Matzner	Donald Blackford
Steve Kokolis	Jack Griffel
Donald M. Lee	William Begell

Louis Bernath (formerly with E. I. du Pont de Nemours & Co., currently with Atomics International) and Ralph P. Stein (formerly with Columbia University, currently with Argonne National Laboratory) made important contributions in the early phases of the program.

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A Terminal Status Report of the Work at Columbia University

INTRODUCTION

The principal advantage of designs for the D_2O-moderated reactors presented by the Atomic Energy Division of the Du Pont Company in a development program for the Atomic Energy Commission is the low fuel cost achieved by the use of massive fuel components that are capable of operating at a high specific power. To obtain the most economical design, it was necessary to know the heat transfer limits for such assemblies.

Although much work is reported in the literature in the general field of heat transfer, the work was not specific enough that the maximum power in fuel designs of interest could be predicted with the accuracy desired. Experimental work was undertaken at the Columbia University Heat Transfer Research Facility to obtain the necessary heat transfer data. A 3.5 megawatt electrical power source was available at the University for testing large-sized simulated fuel elements at conditions pertinent to the power reactor designs.

The detailed work at Columbia has been reported in the regular progress reports*. This Terminal Status Report summarizes and interprets the work.

SUMMARY

The burnout characteristics were determined at Columbia University for simulated fuel elements of the Heavy Water Components Test Reactor (HWCTR) and of designs presented by Du Pont for a heavy-water-moderated-and-cooled power reactor. The reactor fuel elements were rated for a maximum heat flux ranging from 500,000 pcu/(hr)(ft^2)** with liquid D_2O cooling down to 180,000 pcu/(hr)(ft^2) with boiling D_2O. The corresponding burnout heat fluxes, based on the experimental data, ranged from 1,300,000 pcu/(hr)(ft^2) down to 300,000 pcu/(hr)(ft^2). The margin between operating heat flux and burnout is shown in Figure 1 for a fuel assembly of 19 rods in a bundle and in Figure 2 for the inner tubular channel of a fuel assembly of nested (concentric) tubes. Selected

* The pertinent progress reports by Columbia University and by the Du Pont Company from 1957 through 1964 are listed in the References.

** The pcu is the quantity of heat required to raise the temperature of one pound of water from 14.5°C to 15.5°C. The pcu is equal to 1.8 Btu (British thermal unit).

correlations from other laboratories were also applied in the analysis of the margin of safety from burnout for the HWCTR operations and for the power reactor designs. These correlations were from the Savannah River Laboratory (subcooled liquid region) and from Hanford and the General Electric Company-APED (annular channels).

The transition from subcooled to boiling conditions was studied extensively for the effect upon the burnout characteristics. This transition region is encountered by the liquid-D_2O-cooled reactor in abnormal operation as well as by the boiling-D_2O reactor in normal operation. The study was conducted with forced-flow inside single heated tubes. The results are presented in this report to show the burnout heat flux as a function of the coolant quality (fraction vapor by weight) with parameters of tube diameter, pressure, and mass velocity. It was found that:

(1) as the coolant quality increases from negative (subcooled) to positive (boiling), the burnout heat flux decreases sharply, particularly at the higher mass velocities, and then levels out in the boiling region;

(2) the burnout heat flux increases with increasing mass velocity in the subcooled region for the same quality, whereas it decreases with increasing mass velocity in the boiling region, at mass velocities below about 4 million lb/(hr)(ft^2); and

(3) the burnout heat flux at a given quality is independent of pressure in the subcooled region, whereas it decreases with increasing pressure above 750 psia in the boiling region.

When the burnout heat flux is presented as a function of the coolant enthalpy (pcu per lb) rather than coolant quality, the effect of pressure on the burnout characteristics is analogous to the effect of mass velocity. That is, the burnout heat flux increases with increasing pressure in the subcooled region and decreases with increasing pressure in the region of higher quality - at least for mass velocities below 4 million lb/(hr)(ft^2) - with the transition occurring in the region of low quality boiling. The tests also showed a generally higher burnout heat flux for tubes of smaller diameter. These observations, which were made with single tubular channels, were substantiated by the results at Columbia University with other geometrical configurations.

Other studies were conducted in support of the design for the fuel assembly of nested tubes:

(1) The burnout characteristics were determined for five internally heated annular channels. These results contributed particularly to a better understanding of the effects of subcooling and of hydraulic diameter and provided an early design basis for a boiling-D_2O reactor fueled with nested-tube assemblies.

(2) The power at burnout for an assembly consisting of a single thick-walled tube cooled by boiling water flowing in parallel through the inner tubular and outer annular channels was shown to be predictable from the test results for the separate channels.

(3) Initial tests with forced-flow boiling in an assembly of three tubular channels connected in parallel demonstrated the possibility of severe oscillation in the flow within multichannel assemblies.

A fuel assembly of 19 rods of clad uranium oxide in a bundle was one of the candidate designs for a boiling D_2O reactor in the Power Reactor Program. The burnout heat flux was determined for the 19-rod assembly as a function of coolant quality for several mass velocities at a pressure of 1000 psia. The results were consistent with those obtained with simple channels. This indicates a good distribution of the coolant flow in the complex assembly as a result of three factors: (1) a proper layout of the rods in the circular housing tube, (2) a beneficial mixing effect from the spirally wrapped wire spacers, and (3) the lower rod power toward the axis of the bundle, which was mocked up in the tests.

The investigations at Columbia University have been supported and extended by recent work in other laboratories. General agreement exists among the laboratories regarding the qualitative effects of coolant enthalpy, mass velocity, and pressure. There are examples of close agreement when test conditions were similar. However, markedly different predictions still remain in many cases with respect to the exact magnitude of the burnout heat flux and the form of the burnout characteristics. These differences are believed to stem from geometrical factors, from the limited range of the data employed in a particular correlation, from modes of operation in the experiments which are not accounted for in the correlations, and from the attempt to use relatively simple mathematical functions to cover too broad a range. The areas for additional experimental and analytical studies are outlined in this report.

EXPERIMENTAL FACILITIES

The Heat Transfer Research Facility of Columbia University can generate 3.5 megawatts of direct-current power. Two generators, each rated for 10,000 amperes at 175 volts, were connected in parallel to obtain 20,000 amperes maximum at the test section. This electrical power was used to heat the simulated fuel elements of a power reactor.

Three test loops were used during the Power Reactor Program. Each loop consisted of pumps, heat exchangers, pressurizer (nitrogen gas), water make-up and purification, and connecting piping with provision for inserting the electrically heated test section. Subcooled water was delivered to the test section in all tests. Bulk boiling could be accommodated in the last loop constructed. The material of construction throughout the loops was stainless steel.

Test Loop No. 1 was constructed for the Naval Reactors Branch of the Atomic Energy Commission, but was made available for the early portion of the Power Reactor Program. The loop was rated for 2500 psig. A circulation of 250 gpm at 200 feet differential head was provided by two centrifugal canned-rotor pumps operated in parallel. The flow path was from the pump discharge to the test section, to water-cooled heat exchangers, and back to the pump suction.

Test Loop No. 2 was built to complete the program started with Test Loop No. 1. The pressure rating of the new loop was 750 psig, which was adequate for testing the conceptual designs of that time for a D_2O-moderated power reactor. A single centrifugal pump provided 300 gpm at 270 feet differential head. The flow path was similar to that of the first loop.

Test Loop No. 3 was designed to determine the heat transfer limits with bulk boiling. This loop was progressively modified to increase its rated capability and its flexibility for experimental work. The pressure rating, originally 1000 psig, was ultimately 1500 psig. The original pumping capability was 250 gpm; it was ultimately 220 gpm at 1500 psig, or 400 gpm at 1200 psig. The test sections for the Power Reactor Program were vertical, and the heated length did not exceed 7 feet. The flow from the pumps was divided between the test section and the heat exchanger; there was also a bypass around both. These streams were recombined at a mixing tee where any steam from the test section was quenched. The flow then returned to the pump suction.

The electrically heated elements of the test sections were either Type 304 stainless steel or "Inconel-X"* tubing. The electrical connectors at the ends of the test section were either nickel or

* Trademark of International Nickel Co.

silver- or gold-plated copper. Thermal expansion was accommodated by O-ring seals at the bottom of the assembly to obviate buckling when the fuel elements were heated. The geometry of the coolant channel was maintained by the top and bottom grid plates outside the heated zone and generally, along the heated length, by such devices as ribs of laminated asbestos-phenolic material, ceramic ferrules (7-rod bundle), or stainless steel hypodermic tubing (wrapped spirally around the rods of 7- and 19-rod bundles). A ceramic liner for the stainless steel housing was used with the stainless steel spacers and, in certain cases, for annuli without spacers along the heated length to avoid electrical shorting. In the earlier test sections the thin-walled electrically heated tubing was pressurized with nitrogen on the uncooled side to balance the coolant pressure. In most of the testing, however, the heater tubing was supported by ceramic or phenolic inserts when the coolant flowed on the outside; or, the heater tubing was sufficiently thick-walled to support the coolant pressure when the coolant flowed through the inside. The wall thickness of the heaters in the test elements ranged from 0.009 inch to 0.120 inch.

Burnout was determined by actual melting, by measuring the rapid increase in heater temperature, by measuring the change in electrical resistivity of the heater, or by combinations of these methods. The nature of heat transfer burnout and its identification are discussed in the Section, AREAS FOR FURTHER STUDY. The thermocouples for temperature measurements were pressed against the uncooled (dry) surface of the heater tubing. A thin film of cement provided electrical insulation. The thermocouples were iron-constantan in stainless steel sheaths. Generally, three electrical taps were made to the heater tube and connected in a Wheatstone bridge arrangement to sense the change in electrical resistivity; this device was called a "burnout detector." The electrical taps were made at the inlet and exit ends of the heater and about six inches from the exit end. With heaters of stainless steel, this device had excellent sensitivity for burnout occurring at the exit end, as usually happened with uniform axial heat generation. The burnout detector was less satisfactory with heaters of "Inconel-X" because of the smaller temperature coefficient of resistivity. "Inconel-X" was used in place of stainless steel on occasions because its higher electrical resistance allowed a thicker tube wall to be used, thereby providing greater strength.

The total pressure was measured by a Bourdon-tube gage and by strain-gage transducers; the differential pressure, by a manometer. Turbine and orifice meters were used to measure flow. The output of the thermocouples, pressure transducers, turbine meters, and electrical current and voltage meters was displayed on recording potentiometers and on high-speed direct-writing oscillographs.

RESULTS OF EXPERIMENTS

I. SUBCOOLED* BURNOUT IN SIMPLE GEOMETRY

A. Particular Liquid-Cooled Power Reactors

Four conceptual designs for a D_2O-moderated, liquid-D_2O-cooled power reactor of 100 megawatt (electric) rating were presented early in the Power Reactor Program.[1] The fuel assembly in each design was a single tube of Zircaloy-clad natural uranium, 2.060-inches OD x 1.430-inches ID. A minimum program of experimentation was conducted to determine the burnout heat flux at conditions in the range of those for these reactor concepts. The results were within the very broad band-width of predictions by the existing correlations; however, they did not corroborate any particular correlation. The test data, which are summarized in progress report DP-385,[2] established the true burnout limit for the reactor designs.

The burnout heat flux was determined over a narrow range of conditions in the subcooled region for four geometrical configurations: a 1.82-inch-wide annulus (5.761-inches OD x 2.125-inches ID), a 0.45-inch-wide annulus (2.900-inches OD x 2.000-inches ID), a 0.39-inch-wide annulus (2.900-inches OD x 2.125-inches ID), and a 2.055-inch-ID tubular channel. The inner surface only of the annular channels was electrically heated. The heated length was 70 inches. Test Loop No. 1 was used initially, and later Test Loop No. 2. The tests were conducted at 500 to 700 psia with water velocities ranging from 0.5 to 27 ft/sec, and exit subcooling from 11° to 73°C. There were 17 burnout determinations. The burnout heat fluxes ranged from 500,000 to 1,100,000 pcu/(hr)(ft^2). The burnouts were determined by actual melting; with only three exceptions, the melting occurred within a few inches of the downstream end, i.e., at the point of highest coolant enthalpy or lowest subcooling.

The results of the tests showed sizable deviations from predictions by the existing general correlations. This was not surprising in view of the broad band-width of predicted values from these several correlations. The data were generally within this band-width. Moreover, substantial agreement was obtained in later experiments in Test Loop No. 3 with a 0.325-inch-wide internally heated annulus (see Section I,B). Special empirical correlations were presented for the narrow range of test conditions for three of the configurations.[2,3]

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* Water is "subcooled" when its bulk temperature is below the saturation (or boiling) temperature; the "subcooling" is the difference between these two temperatures.

These limited correlations fitted the data within a few percent. A new broad-range correlation for burnout in subcooled water was developed from the first five burnouts (those from Test Loop No. 1 covering three configurations) together with 179 burnouts from seven other laboratories and from other programs at Columbia.[4] However, when all 17 burnouts had been obtained and evaluated, the agreement was no better than with the previously preferred correlation.[5] The latest correlations for burnout in subcooled water[6,7] still exclude the region of low velocity, low subcooling, and large diameter that are covered by these tests at Columbia University because the broad-range correlations fail to predict adequately these observed burnouts. This verifies the value of experimental data at conditions closely approximating the intended region of operation, and shows the fallibility of extrapolation.

Severe vibration and pressure fluctuation were encountered in about half the experimental runs. These led to concern regarding the application of the results to the reactor situation. Vibration and pressure fluctuation are discussed in Section VI.

B. HWCTR Driver Annulus

The Heavy Water Components Test Reactor (HWCTR) was a fuel-test reactor constructed at the Savannah River Plant as part of the Power Reactor Program to develop a heavy-water-moderated nuclear-power reactor. The HWCTR was cooled and moderated by liquid D_2O at 1200 psig and 200-250°C. It was necessary to operate the HWCTR at coolant temperatures and powers as high as possible to obtain meaningful irradiation experience with the test fuel. Operation of the reactor was permitted with bulk coolant temperatures approaching 10°C of the saturation temperature, i.e., 10°C or greater local subcooling. The power level was limited, however, by prescribed margins from burnout. Tests were conducted at Columbia University to determine the burnout heat flux at low subcooling for pressures, coolant velocities, and geometry applicable to the HWCTR operation. The heat flux at burnout was shown to decrease more rapidly as the subcooling decreased than had been believed. In particular, at the power planned for the driver fuel assembly, a subcooling of about 30°C was needed to provide the prescribed margin from burnout, rather than the 3°C predicted previously. In normal operation with local subcooling greater than 50°C, however, the predictions were substantiated. An improved general correlation for burnout in subcooled water was subsequently developed at Savannah River Laboratory with the aid of these and other data.[7]

The tests were conducted in Test Loop No. 3 with subcooled water at velocities of 5, 10, 15, and 20 ft/sec at pressures of 500, 750, and 1000 psia in an internally heated annulus, 2.90-inches OD x 2.25-inches

ID. This annulus was the size of the outer channel of the HWCTR driver fuel assembly. Ten test sections were employed. The heated lengths were 40 inches and 24 inches. The local subcooling ranged from 66° down to 2°C. A total of 107 burnouts were determined, generally by the burnout detector, but with six melting burnouts. The meltings were in reasonably good agreement with the detector data. The burnout heat fluxes ranged from 483,000 to 1,578,000 pcu/(hr)(ft^2). The data are summarized in progress report DP-805.[8,9]

More scatter occurred in the downward trend of the burnout heat flux with decreasing subcooling for the data at 500 than at 1000 psia. This was attributed to the vibration and pressure fluctuation that were encountered at 500 psia in certain regions of operation, just as in Test Loops No. 1 and No. 2 (see Sections I,A and VI). The burnouts at 500 psia were only slightly higher than the prediction by the special correlation developed for a 0.39-inch-thick annulus (see Section I,A).

II. BOILING BURNOUT IN SIMPLE GEOMETRY

A. Tubular Channels

Burnout for the liquid-D_2O-cooled power reactor is most likely to occur as a result of abnormal events which lead to bulk boiling, such as a sufficient increase in power, decrease in flow, or decrease in pressure. In the transition from the subcooled to the bulk boiling condition, marked changes occur in the dependence of the burnout heat flux: 1) the burnout heat flux decreases sharply and then flattens out with increasing quality, particularly for the higher mass velocities; 2) an inversion occurs in the effect of mass velocity on the burnout heat flux; and 3) the pressure becomes a significant parameter in the boiling region, the burnout heat flux decreasing with increasing pressure at a given coolant quality, or even enthalpy. These effects were clearly delineated with little scatter of the data in an extensive series of burnout determinations with flow inside tubular channels. The results are presented in the APPENDIX by Figures A-1 through A-8.

The complex pattern of the characteristics of the burnout heat flux in this transition region has so far defied precise correlation by empirical formulae. This does not preclude direct application of the test data, however, in determining the margin of safety from burnout for the liquid-D_2O-cooled and the boiling-D_2O power reactors (see APPLICATION TO REACTOR DESIGNS). The tubular channel is a typical one in an assembly of nested tubes. The experimental information obtained with the simple tubular channels served as the reference for heat transfer studies of a tubular element cooled inside and outside (see Section III) and of channels-in-parallel (see Section V), both of which pertained to the nested-tube assembly.

The burnout heat flux was determined in Test Loop No. 3 for tubes of 0.245, 0.504, 0.930, and 1.475-inches ID. The heated length was generally 77 inches, with some data for 36- and 24-inch lengths.[10-12] Tests were made at pressures of 500, 750, 1000, 1200, and 1500 psia. Mass velocities ranged from 500,000 to 14,000,000 lb/(hr)(ft^2); coolant conditions ranged from 80°C subcooled to 59% quality. A total of 402 burnout points were obtained using the burnout detector. The burnout heat flux ranged from 250,000 to 1,430,000 pcu/(hr)(ft^2). The three melting burnouts which occurred were close to the trend of the burnout detector data. All burnouts occurred near the downstream end of the heated tube; they were of the sharp variety typical of a departure from nucleate boiling (see AREAS FOR FURTHER STUDY).

Plots of the data showed that, in forced-flow boiling, the burnout heat flux at a given quality is greater at lower mass velocity. Stated another way, the quality at burnout for a given heat flux is greater at lower mass velocity. This is opposite to the effect of mass velocity in the subcooled region, hence the term "inversion." The inversion was demonstrated quite forcibly in several sequences in reaching burnout. After burnout conditions had been obtained, the power was lowered slightly to reestablish normal heat transfer. Upon increasing the flow at this lower heat flux, burnout occurred again. And again, upon lowering the power slightly and increasing the flow, another burnout condition was obtained. This is contrary to expectations based on subcooled experience. It points out the importance of making a large adjustment downward in heat flux (toward zero heat flux) or upward in flow (toward subcooled conditions) in order to move away safely from a burnout condition in forced-flow boiling.

The burnout heat flux at a given coolant quality in the boiling region was shown by the tests with tubular channels to be lowered appreciably by increasing the pressure, in the range of 700 to 1500 psia. This is in contrast with the small effect, if any, of pressure upon the burnout heat flux in subcooled flow at a given coolant quality (or, at a given degrees of subcooling, since the proportionality constant between degrees of subcooling and quality is not a strong function of pressure)*. There is general agreement among the several laboratories cited in the Section, RESULTS FROM OTHER LABORATORIES, regarding these qualitative effects of pressure in the subcooled and boiling regions at a given coolant quality.

* In the subcooled region, the quality is a negative number. Quality is used in the subcooled region instead of degrees of subcooling to permit the use of a single scale related linearly to enthalpy, when the pressure is specified, for both the subcooled and boiling regions.

Agreement is less apparent among the laboratories on the effect of pressure at a given coolant enthalpy (rather than quality). At a given enthalpy, in the range of interest to the Power Reactor Program, an increase in pressure causes a decrease in quality in both the boiling and subcooled regions (the quality becomes more negative in the subcooled case). It is generally agreed that the burnout heat flux is higher at lower quality for a given pressure and mass velocity. Thus, the difference between the burnout heat flux at a particular pressure and that at a higher pressure is certainly less (a negative difference meaning a higher burnout heat flux at the higher pressure) when enthalpy is held constant rather than quality. Some correlations actually show an increase in burnout heat flux in the boiling region with an increase in pressure at a given coolant enthalpy. In other correlations, the direction of the pressure effect in the boiling region depends upon the mass velocity. The data of the Heat Transfer Research Facility show that for the same coolant enthalpy - at least, for mass velocities below 4 million lb/(hr)(ft^2) - a higher burnout heat flux occurs at higher pressure in the subcooled region whereas a lower burnout heat flux occurs at higher pressure in the boiling region at sufficiently high enthalpy. The transition occurs in the region of low quality boiling. The inversion in the effect of pressure on the burnout heat flux at constant enthalpy is analogous to the inversion in the effect of mass velocity.

A higher burnout heat flux, or a higher coolant quality at burnout, was obtained with tubes of smaller diameter. Also, a somewhat higher heat flux at a given quality was obtained for the 24-inch length than for the 36-inch and 77-inch lengths.

B. Two Annular Channels

The heat flux limit with forced-flow boiling at 1000 psia was determined for two sizes of internally heated annuli:

OD, inches	ID, inches	Hydraulic Diameter, inch	Annulus Width, inch	References
1.902	1.380	0.52	0.26	(13, 14)
1.745	1.375	0.37	0.18	(15, 16)

The purpose was twofold: 1) to perfect experimental techniques in the operation of the new Test Loop No. 3, in the design of the test sections, and in the interpretation of the burnout detector; and 2) to relate the results obtained at Columbia University to the data and correlations from other laboratories. The tests were primarily in preparation for the work on more complicated assemblies. The use of burnout detectors was planned for the complex assemblies because they promised more extensive coverage of the conditions for burnout than was feasible by the melting technique. The burnout data for the simple annular channels figured importantly, however, in prescribing a rough design correlation for use in the computer optimization of a boiling-D_2O reactor fueled with nested tubes of compacted uranium oxide. The data were compared in progress reports DP-555 and DP-685[13,15] with correlations and other data in existence at that time.

The heat flux limit was determined by means of the burnout detector for about 30 different conditions, and then the test was deliberately carried to the point of melting for each test section. The mass velocity ranged from 300,000 to 1,900,000 lb/(hr)(ft^2). The effluent ranged from a few degrees subcooled to 25% quality. The burnout heat flux ranged from 270,000 to 606,000 pcu/(hr)(ft^2). The burnouts occurred at the downstream end of the 42-inch-long uniformly heated assembly.

The data showed continuity in a progressive downward trend from the higher heat flux limits in the subcooled region[17] to still lower heat flux limits in the high quality region.[18] Fair agreement was shown with tentative correlations of that period[19,20] for the burnout heat flux under similar conditions with a similar annular configuration. (See Figure A-10 in the APPENDIX for the comparison of these Columbia data with a later correlation.) A disparity was evident, however, with respect to the general correlations of that period.[21,42] In particular, the burnout heat flux in bulk boiling was much lower for the internally heated annuli than for tubular channels or rectangular channels heated on both surfaces (in the same size range or smaller) under similar conditions of pressure, mass velocity, and local quality. In the light of these comparisons and the new data, a simple design relationship was drawn between the burnout heat flux, q_{BO} pcu/(hr)(ft^2), and the coolant quality fraction, x, for use in the computer optimization program[6]:

$$q_{BO} = 554{,}000 - 1{,}080{,}000\ x$$

This relation was restricted to quality fractions from 0.05 to 0.30 and to pressures from 750 to 1000 psia. It was an oversimplification, but the state of the knowledge on forced-flow boiling burnout justified nothing more complex.

The sensitivity of the burnout detector was set lower by one decade for the second test section. Thus, the instabilities in the heat transfer process which preceded boiling burnout were not observed so early as with the first test section. In fact, the melting burnout with the second test section occurred only a few percent higher in power than the burnout detector point. The melting burnout with the first test section was 7% higher in power than the burnout detector point. The reduction in sensitivity for instabilities in the heat transfer process reduced the scatter of the data. Equally important was the discrimination, or better definition of the effect of the mass velocity parameter. It was clearly shown that, contrary to the experience in subcooled burnout, the heat flux at burnout in forced-flow boiling was lower at a given quality for higher mass velocity. This inversion in the effect of mass velocity was just beginning to be recognized in the literature. (These tests preceded those for tubular channels discussed in Section II,A.)

The claim was also made of a maximum in the characteristic of the burnout heat flux as a function of quality. This was evident in the data at the lowest mass velocity. This phenomenon had been observed consistently in another laboratory[18] in the high quality range. Other researchers have also shown a maximum in the burnout characteristic in the low quality region.[22,23] Their work shows this occurrence to be the result of compressibility in the system, as from the existence of the vapor phase at the inlet, and of the resulting internal flow fluctuations. In reviewing the two low burnout heat fluxes in the data from Columbia University, which led to the appearance of a maximum characteristic, it is currently thought that the observed instabilities in the heat transfer process were of a fleeting variety and were wrongly interpreted as burnout. It was on the immediate program at the time the project was terminated not only to repeat these two burnout conditions, but also to investigate more extensively the burnout process in the various regimes of flow and to improve upon the identification of burnout, as outlined in the Section, AREAS FOR FURTHER STUDY.

III. BOILING BURNOUT FOR A SINGLE-TUBE ASSEMBLY

It had been planned to irradiate a tube of compacted uranium oxide in the Boiling-D_2O Loop of the HWCTR, but the loop never was commissioned. The fuel tube was to be cooled by boiling heavy water flowing in parallel through the inner tubular channel and the outer annular channel formed by the fuel tube within a housing tube. The power was to be about 0.6 megawatt and the maximum heat flux about 200,000 pcu/(hr)(ft^2). The outlet quality was to be varied up to 30% by varying the mass flow and inlet subcooling. This assembly for irradiation was one of convenience; in particular, the channels were not sized to provide equal enthalpy rise or equal margins from burnout. The burnout characteristics for the assembly, determined at Columbia University, were safely above the planned operating conditions.[24]

Three assemblies having a thick-walled, electrically heated tube cooled on both surfaces by forced-flow boiling water were tested. The first two assemblies used a heater tube of solid stainless steel, 1.660-inches OD x 1.442-inches ID x 40-inches heated length, inside a 2.50-inch-ID housing. These dimensions were somewhat smaller than those of the proposed reactor assembly because of the problem of the low electrical resistance. The third assembly employed a special heater tube to provide the same dimensions in cross section as for the proposed reactor assembly: 2.065-inches OD x 1.478-inches ID. This heater tube was a composite of a tube of phenolic resin between two thin-walled stainless steel tubes. The housing tube was 2.90-inches ID, as for the reactor assembly. The heated length was 77 inches, as compared with 111-inches length for the reactor assembly, but the axial heat generation was uniform rather than cosine. This meant that, for the same coolant conditions (flow, inlet subcooling, and exit quality) and power, the test assembly operated near the maximum heat flux that would occur in the reactor.

A total of 77 burnout determinations were made at 1000 psia in Test Loop No. 3 with these three assemblies. The heater tubes were then deliberately melted by a small increase in power (5 to 7%) beyond the point of incipient burnout as seen by the burnout detector. The test conditions covered mass velocities from 500,000 to 2,300,000 lb/(hr)(ft^2) and exit qualities from -6% (17°C subcooled) to +29%. For the two short assemblies using the solid stainless steel heaters, the assembly powers at burnout ranged from 0.66 to 1.09 megawatts and the assembly average heat fluxes at burnout ranged from 430,000 to 710,000 pcu/(hr)(ft^2). For the assembly with the composite tube, the powers at burnout ranged from 1.05 to 1.68 megawatts and the heat fluxes (inner surface) at burnout ranged from 360,000 to 570,000 pcu/(hr)(ft^2). The tests with the composite-tube assembly showed that burnout originated in the inner (tubular) channel; this is assumed for the assemblies with the solid stainless heaters.

Although the single-tube assembly presented two coolant channels in parallel, no significant effect (<10%) on the burnout heat flux was observed that could be attributed to hydraulic interaction between channels. In spite of care in the design of the assemblies using the solid stainless steel heater, the entrance losses exceeded the friction losses in the channels; this condition would enhance the flow stability. For the composite-tube assembly the entrance configuration was a mockup of that which would have existed with the bayonet pressure tube of the HWCTR loop. It was decided to use a different test assembly to study flow stability in parallel channels and its effect on burnout (see Section V).

The flow distribution to the channels was measured to ±10% for the tests with the solid stainless heaters. Because of mechanical difficulties, it was necessary to estimate the flow distribution in the case of the composite-tube assembly from the preliminary cold calibrations. From the flow distribution it was possible to calculate the mass velocity and exit quality in the individual channels at burnout. The burnout characteristics for single 1.475-inch-ID tubes were obtained from a separate study (Section II,A). Reasonably good agreement (0 to 20% deviation) was found between the single-channel and the two-channel results, with reservations regarding the uncertainty of the flow distribution. Also, from the estimated burnout conditions for the tubular channel of the single-tube assemblies, it appeared that the differing thermal characteristics for the thick-walled heated tubes (the composite tube having a lower effective thermal diffusivity than the solid stainless tube) resulted in less than a 10% effect on the burnout heat flux in forced-flow boiling, the burnout heat flux being somewhat lower for the composite tube.

IV. BOILING BURNOUT IN ROD BUNDLES

A bundle of 19 rods of Zircaloy-clad uranium oxide within a Zircaloy housing tube was planned as an interim fuel assembly for a boiling-D_2O reactor, pending the development of nested tubes of the oxide. The nested-tube assembly was expected to show lower fueling cost. A fabrication process for long rods of Zircaloy-clad UO_2 had been developed at Hanford for fueling the Plutonium Recycle Test Reactor. Some modifications were made to the PRTR design in settling upon a reference design for the Power Reactor Program. Assemblies for test irradiations in the HWCTR were fabricated by an outside contractor. One mechanical design was drawn up for the Power Reactor Program. Flow tests at the Savannah River Laboratory showed this assembly to be serviceable. The greatest unknown in the whole reactor design was the heat transfer capability of the rod bundle with forced-flow boiling. The extensive tests at Columbia University showed that this particular design of a 19-rod bundle would operate with adequate margins from burnout in the reference boiling D_2O reactor[25,26] (see Figure 1). (See APPLICATION TO REACTOR DESIGNS for more details on interpreting Figure 1.)

The rods of the bundle were 0.550-inch OD and were spaced on 0.633-inch centers. The spacing was maintained by 0.083-inch-diameter wires (actually hypodermic tubing for the electrically heated test assembly) wrapped in a spiral on twelve of the rods, in the same direction, on a 10-inch pitch. The bundle was similarly wrapped by wire of the same diameter and inserted in a 3.195-inch-ID housing. Four such 19-rod assemblies were tested. Three of these were 3 feet long; the fourth was 6 feet long. The heat generation in the axial direction was uniform. The radial heat generation was tailored to simulate the flux depression of a real fuel assembly; however, one of the 3-foot-long assemblies had three rods generating twice the power of the others in order to extend the range of conditions at burnout. The rods were fitted with either a burnout detector or a thermocouple near the downstream end.

The tests of the 19-rod bundles were conducted in Test Loop No. 3 at a pressure of 1000 psia with mass velocities ranging from 500,000 to 2,000,000 lb/(hr)(ft^2) and with exit qualities from 8 to 59%. Subcooled water entered the assembly in all cases, as in the boiling-D_2O reactor concept. The highest burnout heat flux obtained was 550,000 pcu/(hr)(ft^2) at a mass velocity of 2,000,000 lb/(hr)(ft^2) and an exit quality of 9.6%. The burnout heat flux decreased with increasing quality, as occurs with simple channels. The effect of mass velocity on the burnout heat flux was less marked than with simple channels. Nevertheless, the inversion in the effect of mass velocity was seen, namely, a lower burnout heat flux with higher mass velocity at the same quality.

Thus, the magnitude and the characteristics for burnout in the 19-rod bundle were quite similar to those for simple channels. Also, burnouts were obtained on both intermediate and outer rods. These observations indicated a favorable distribution of the flow and local enthalpy throughout the cross section of the bundle. This good enthalpy balance can be attributed to three factors: the particular configuration of rods within the circular housing tube, the turbulent mixing created by the spirally wrapped wire spacers, and the radial flux depression.

Tests were conducted beyond the point of incipient burnout to progressively higher surface temperatures, and finally to melting. A relatively wide margin in power and in quality was found in some cases between the initial increase in surface temperature that was identified as incipient burnout and the actual melting.

Prior to the tests on 19-rod bundles, two 7-rod bundles were tested. The purpose was threefold: 1) to develop design techniques for the larger assembly, 2) to evaluate at an early date the effect of spiral wire spacers on the heat flux limit, and 3) to have a design for testing to greater lengths, higher rod power, and higher mass velocity than was possible for a 19-rod bundle with test facilities available at Columbia University. The rods in the 7-rod bundles were the same diameter and spacing as those used in the 19-rod assembly. The housing shape was scalloped, rather than circular, to simulate the effect of more rods in the assembly and thereby to obtain a more realistic distribution of the coolant flow. The heated length of these assemblies was 3 feet. The heat generation of the central rod was 97% that of the outer rods. The axial heat generation was uniform. One 7-rod assembly had wire wraps on the 6 outer rods for spacing; the other 7-rod assembly had ferrules (rings) between the rods on 18-inch centers. Only the central rod was instrumented, the burnout detector being used; burnouts were obtained on both the central and the outer rods. A somewhat higher heat flux at burnout was observed for the assembly with spirally wrapped wire spacers than for the assembly with ferrule spacers. This indicates that the turbulent mixing from the spiral wires more than compensated for any undesirable centrifugal separation of liquid phase from the heated surface. The burnout and hydraulic characteristics of the 7-rod bundle corresponded closely to those obtained for the 19-rod bundles. This indicated the adequacy of the 7-rod bundle for testing beyond the range possible with 19 rods.

The small gain for spirally wrapped wire spacers over the ferrule type of spacing was one of compensating effects. An apparent disadvantage of wire spacers was manifested in overheating and failure in the 19-rod assemblies on the wire-wrapped rods only. There were signs that the spiral wrap around the bundle also contributed to overheating in a characteristic way.

The work at Columbia University on rod bundles satisfied the immediate requirements of the Power Reactor Program for an alternative fuel assembly. Several organizations have since elected to support more extensive investigations of the parameters in rod bundle design at Columbia University, namely, the United Kingdom Atomic Energy Authority, Atomic Energy of Canada Limited (through the USAEC-AECL Cooperative Program), Atomics International, and Westinghouse Electric Corporation-APD (for the Carolinas-Virginia Nuclear Power Associates). These studies have shown the importance of rod arrangement and spacing methods within the housing. Varying the arrangement and spacing can cause a considerably lower heat flux at burnout, and possibly a higher heat flux, than was obtained for the particular design for the Power Reactor Program. Some progress has been made toward correlating the burnout heat flux for rod bundles in which the work at Columbia University figured importantly.[27,28]

V. FLOW STABILITY AND BURNOUT WITH PARALLEL CHANNELS

The fuel assembly of nested tubes contains several long channels through which the coolant flows in parallel. For example, an assembly of three nested fuel tubes within a housing tube has three annular channels and a central tubular channel. The channels are connected hydraulically only at the inlet and exit ends of the assembly. The annular channels may be further divided into parallel subchannels by longitudinal spacer ribs. In addition, the heavy-water-moderated power reactor has many such fuel assemblies in parallel between common inlet and outlet headers for the coolant flow. The existence of parallel channels raises the question of the stability of the flow distribution, particularly as power is increased or total flow is decreased so that boiling ensues. There was concern for possible hydraulic interaction between the two channels of the single-tube assembly (Section III), but the results were inconclusive because of the difficulty in measuring the channel flow without affecting the flow distribution. Therefore, a direct assessment was made at Columbia University by tests with two special assemblies, each with three long tubular channels in parallel. Severe instability of the flow in boiling was encountered in some of these tests. Additional studies were planned, at the time the project was terminated, to determine more precisely (1) the conditions for instability of the flow, (2) the effect of flow oscillations on the burnout heat flux, and (3) the channel and assembly orificing requirements to ensure stable flow and high burnout heat flux.

The first test assembly consisted of three electrically heated tubes in parallel that were closely connected to common inlet and exit plenums.[11,29] The tubes were 0.493-inch ID x 88 inches long. The tubes were fitted with burnout detectors but not with individual flowmeters. Thus, it was not possible to observe directly the stability of the flow. The tests were run at 1500 psia with mass velocities ranging from 1,000,000 to 4,000,000 lb/(hr)(ft^2) in Test Loop No. 3. Tests were conducted both at constant flow to the assembly (16 data points) and at constant pressure differential across the assembly (3 data points). The latter operation, accomplished by a sizable bypass flow around the test assembly, simulated operation of several fuel assemblies in parallel. The burnout heat fluxes for the three-tube assembly were only slightly lower, on the order of 0 to 10%, than those observed for single-tube assemblies (see Section II,A).

A second test assembly of three-tubes-in-parallel was fitted with turbine flowmeters for the individual tubes in addition to the turbine meter for the total flow to the assembly.[30,31] Each tube also had its burnout detector and an outside-wall thermocouple at the downstream end. The tubes were nominally 0.5-inch ID x 76 inches long. An imbalance existed in that one tube had a slightly thicker tube wall (0.068 inch versus 0.063 inch); the thicker tube had a 9% higher power. Tests were conducted with (23 data points) and without (40 data points) added orifice restriction in the exit piping for the combined channel effluents. The orifice restriction in the combined effluent stream simulated the effect of the exit connector piping from the fuel assembly to the effluent header of a power reactor. The test conditions ranged as follows: pressures of 500, 1000, and 1500 psia;

mass velocities from 900,000 to 4,100,000 lb/(hr)(ft^2); average exit quality from 0 to 42%. Burnout was indicated in about half of the runs. The remainder of the runs were terminated by severe flow instability. The heat fluxes ranged from 127,000 to 543,000 pcu/(hr)(ft^2).

Operation with severe instability of the flow in the second assembly was not continued to burnout. However, where burnout was reached under other conditions with this assembly, the burnout heat flux was higher than for single-tube reference, and also higher than for the first 3-tube assembly. Thus, there is a possibility of a higher power capability as a result of a labile flow condition.

The occurrences of severe flow instability diminished upon increasing the pressure from 500 to 1500 psia and upon increasing the mass velocity from 1,000,000 to 2,000,000 lb/(hr)(ft^2), and higher. The normal operating conditions for the reference liquid-D_2O-cooled power reactor are about 1500 psia and 8,000,000 lb/(hr)(ft^2). Thus, even though bulk boiling occurs in the approach to burnout for this concept (upon deviations from normal operating conditions of power, flow, or pressure), flow instability is not a concern. For the boiling D_2O concept, however, the normal operating conditions are 1000 psia and 2,000,000 lb/(hr)(ft^2). Instability of the flow in the parallel channels is therefore a concern for the boiling D_2O reactor.

Increasing the resistance to flow downstream of the assembly after the channel flow streams recombined, in simulating the effect of the connector piping of a power reactor, promoted flow instability. However, the heat fluxes at the point of severe flow instability and at burnout were not adversely affected by the increased resistance.

VI. VIBRATION AND PRESSURE FLUCTUATION

Severe vibration and pressure fluctuation were encountered in certain regions of operation at high heat flux with subcooled water. Instances are cited in Sections I,A and I,B. Other laboratories have reported similar experience. The effect was much more severe than was encountered in boiling flow, even with parallel-channel assemblies (Sections III and V). It is not possible to say with certainty that these disturbances would occur in the reactor situation. The greater scatter in the test data in such regions is evidence of an effect from oscillations. A brief program of exploration was conducted for the operating conditions at the vibration threshold, but no quantitative understanding was achieved.[2,3]

The severe vibration and pressure fluctuation in heat transfer experiments with forced flow of subcooled water occurred, if at all, at 500 to 700 psia as the heat flux attained about 75% of the burnout value. The disturbance developed quite sharply in the regime of nucleate boiling (or local boiling) at the heated surface, and the magnitude was fairly constant over the range of occurrence. The loop piping as well as the test section vibrated. Pressure fluctuation (±20 to 100 psi) always accompanied vibration, but pressure fluctuation could occur without vibration. There was also substantial flow fluctuation (generally up to ±15%). In a given run, it was possible to move into the region of vibration and out again repeatedly by raising and lowering the power.

With a new test assembly of the same nominal dimensions, vibration was not obtained at the same conditions, if it occurred at all before burnout. Vibration and pressure fluctuation was most severe with the large (1.82-inch wide) annulus at low velocity (<4 ft/sec). Although no vibration was encountered in heat transfer to subcooled water flowing inside a tube, pressure fluctuation was significant. In the region where pressure fluctuation was encountered, the magnitude was not influenced by the subcooling, the velocity, or the system pressure. The magnitude of the flow fluctuation was somewhat dependent upon the loop characteristics (i.e., throttling). However, the disturbance occurred in all three test loops in about the same circumstance. In Loop No. 3, in which tests could be conducted at higher pressure, severe vibration and pressure fluctuation were encountered at 500 psia, as in Loops No. 1 and 2, but not at 750 psia and 1000 psia. As before, the whole loop vibrated; also, pressure fluctuation occurred without vibration.

In another test at 1000 psi, pressure fluctuation accompanied by vibration increased quite sharply from about ±2 psi to about ±10 psi as the exit subcooling reached zero and bulk boiling commenced. This was with the composite-tube assembly (Section III) which had two channels in parallel. The pressure fluctuation occurred at low mass velocity, high inlet subcooling, and relatively high heat flux.

VII. EFFECT OF DISSOLVED GAS ON SUBCOOLED BURNOUT

The HWCTR was pressurized with helium gas for an operating pressure of 1000-1200 psia. The heavy-water coolant therefore contained dissolved helium. In the operation of the reactor, however, the effluent water was maintained far below saturation with respect to helium to avoid any release of helium in the moderator space, which would have affected the nuclear reactivity. Helium could be released from the solution locally, however, at the fuel surface upon nucleate boiling.

In tests, helium released at the fuel surface affected the heat flux limit negligibly <10%, even when the gas content greatly exceeded the solubility at the local coolant condition of pressure and bulk temperature.[8,32]

In the normal operation of the test facility at Columbia, the circulating water was maintained free of dissolved gas by purging at the points of highest temperature and lowest pressure. The gas pressurizer and the main stream were connected by a long small-diameter section of piping to minimize interchange of water. To determine the effect of dissolved gas on the heat flux limit, helium was added directly to the main stream. The measured helium content ranged from 0.6 to more than 4 cc STP helium/gram water (the maximum solubility of helium at 1000 psia is 1.6 cc STP/gram water, which occurs at 220°C).

Burnouts were obtained at 1000 psia (H_2O saturation temperature, 285°C) and local bulk temperatures ranging from 254°C (helium saturation, 1.3 cc/gram) to 274°C (helium saturation, 0.6 cc/gram). The geometry of the HWCTR driver annulus was used (Section I,B). The velocity was 20 ft/sec. The heat flux at burnout with dissolved helium corresponded closely to that obtained with negligible dissolved gas.

VIII. PRESSURE DROP IN FORCED-FLOW BOILING

The pressure drop in boiling flow governs the flow stability in parallel channels and the flow distribution and coolant enthalpy in complex channels, such as in the rod bundle or the eccentric annulus. The pressure drop in forced-flow boiling must be known to obtain an operable and economical design for the reactor. It was the rule, therefore, in the work at Columbia University to measure the pressure drop across the assemblies in addition to obtaining heat transfer information. The pressure-drop data from the rod bundles were evaluated in detail.[25,33]

The evaluation of the data on pressure drop involves a comparison of the observations with predictions by existing computational programs and the use of the data to improve upon these programs. The COBRA code for the IBM 650[34] was revised by Columbia University for use with the IBM 704 at the Savannah River Laboratory. This revision was just completed when the project was terminated. The code needs to be tested against the mass of accumulated data. The code is based upon a model that considers the liquid and vapor phases to be a homogeneous mixture and, therefore, to move at the same velocity. It is also desirable to evaluate the data by calculational procedures that recognize the existence in two-phase flow of slip between the velocities of the liquid and vapor phases and, possibly, that recognize the several regimes of two-phase flow, once these regimes have been adequately defined.

The data from the 7-rod and 19-rod bundles were evaluated both for the homogeneous model and for the slip model. The slip model was that of R. C. Martinelli and D. B. Nelson,[35] as modified by N. C. Sher.[36] The pressure drop due to friction in two-phase flow was obtained by subtracting the pressure drops for elevation change and for acceleration over the interval from the observed pressure drop. The ratio of frictional pressure-drop gradient in two-phase flow to the frictional pressure-drop gradient for 100% saturated liquid flowing in the channel at the same mass velocity was correlated as a function of the quality for the particular rod-bundle configuration. This friction ratio increases with quality. The analysis of the pressure-drop data from the two types of 7-rod bundles (see Section IV) showed a 65% higher friction ratio for the assembly with spirally wrapped wire spacers than for the assembly with ferrule spacers. This confirms that additional energy was expended for radial mixing in the rod bundles that had spiral wrapping. The friction ratio (as a function of the coolant quality) for the assembly with ferrule spacers was in good agreement with the friction ratio function of Martinelli-Nelson for flow in simple channels, but it was higher than that calculated for the homogeneous model. The agreement of the friction ratio function among the several 19-rod bundles, all of which had spirally wrapped wire spacers, was also good. Also, the friction ratio function for the 7-rod bundle with spirally wrapped wire spacers corresponded closely to that for the 19-rod bundles.

RESULTS FROM OTHER LABORATORIES

Throughout the program there was an underlying concern for the heat flux limits of the simple coolant channels: the tubular channel, the internally heated annulus, and the internally-and-externally heated annulus. In part, this concern stemmed from the fact that, even for the simple geometry, the heat flux limits could not be predicted to the desired degree of reliability. An understanding of the heat transfer limitations for simple channels was needed to guide and interpret the work with complex assemblies. Additionally, these channels were the types encountered with the nested-tube fuel assembly, which was the preferred design in the Power Reactor Program.

Progress was made toward an understanding of the heat flux limits in simple channels during the period of the Power Reactor Program, not only at Columbia University but also in the several laboratories throughout the world. Several pertinent correlations were developed by these laboratories during this period. These correlations are discussed briefly. Comparisons are made by graphical display of the predicted heat flux in a common system of variables and with the same scale for the coordinates. In spite of the variety of opinion so displayed, there are marked areas of agreement among the laboratories.

R. H. Towell of the Savannah River Laboratory developed the correlations for the burnout heat flux with forced flow of subcooled water that were used in the design of the liquid-D_2O-cooled reactors and in the operation of the HWCTR. Towell employed his own data together with data from Columbia University. Separate correlations were presented for tubular channels[6] and for annular channels.[7] The SRL correlations are employed in the Section, APPLICATION TO REACTOR DESIGNS.

Perhaps the most pertinent research, outside of the Power Reactor Program, is at the Hanford Thermal Hydraulics Laboratory in connection with the "tube-in-tube" fuel assembly for the New Production Reactor (NPR).[37] This work is CONFIDENTIAL and cannot be presented in this report. The Hanford correlations were applied to the reference designs of the Power Reactor Program; the results were reassuring. Where a similar geometry was tested under similar conditions at Hanford and at Columbia University, there was very good agreement.

Also of importance to the Power Reactor Program was the heat transfer work by the General Electric Company, Atomic Power Equipment Department (GEAP). The GEAP data have been correlated by Janssen and Kervinen[38] and by Tippets.[39] Figures A-9, A-10, and A-11 in the APPENDIX show the prediction by the Janssen and Kervinen correlation for a particular annulus at pressures of 600, 1000, and 1450 psia. A comparison is made in Figure A-10 with burnout data from Columbia University for this annulus. Figures A-12, A-13, and A-14 in the APPENDIX show the prediction by the Tippets correlation of the burnout heat flux for 0.5-inch-ID tubular channels at different mass velocities and pressures. Comparison is made in these figures to the best fit through experimental data from Columbia University. The correlation of Janssen and Kervinen is employed in APPLICATION TO REACTOR DESIGNS to evaluate 1) the margins of safety from burnout for the outer annulus of the liquid-D_2O-cooled reference design, and 2) the effect of pressure on the design for a heavy-water-moderated power reactor.

Other correlations for the burnout heat flux were considered for their possible application to the Power Reactor Program. They are as follows:

J. Griffel, Columbia University.[40] See Figures A-15, A-16, and A-17.

L. S. Tong, et al, Westinghouse APD.[41] See Figure A-18.

R. Atherton, Bettis Atomic Power Laboratory.[42] See Figure A-19.

W. J. Levedahl, General Nuclear Engineering Corporation.[43] See Figures A-20, A-21, A-22, and A-23.

M. Silvestri, et al, Centro Informazioni Studi Experienze (C.I.S.E.).[44] See Figure A-24.

U. H. von Glahn, National Aeronautics and Space Administration.[45] See Figure A-25.

R. V. Macbeth, Atomic Energy Establishment, Winfrith, England.[46] See Figures A-26 and A-27.

R. H. Wilson and J. K. Ferrell, The Babcock and Wilcox Company.[47] See Figures A-28 and A-29.

These correlations are also presented, in graphical form, in the APPENDIX. The burnout characteristics were evaluated by these correlations for the particular conditions of geometry, pressure, and mass velocity noted on the figure; the correlations generally cover a broader range, for which the references must be consulted.

The experiments at Columbia University on the burnout heat flux for tubular channels (see Section II,A) are presented in Figures A-1 through A-8. The curves represent a best fit through the data points; there was very little scatter. These experimental results are cited because of the broad coverage given to the prime variables: tube ID, pressure, coolant quality, mass velocity, and length. They best describe the complexity of the transition region which is difficult to match with a simple empirical correlation.

The figures in the APPENDIX show a variety of opinions regarding the heat flux limit. The differences reflect that the correlations were drawn from selected data, frequently those of a particular laboratory. The emphasis in such a cut of data may have been on a narrow range of conditions of geometry, quality, mass velocity, and pressure. Also, the selected data may have been largely for one mode of operation with respect to such details of the experiments as the inlet coolant condition (subcooled, or quality), the inlet configuration, the test loop facility, the design of the electrically heated test elements, and the method of burnout detection. Many of these variables are important in determining the nature of the two-phase flow which is undoubtedly a factor governing the burnout heat flux. It is evident that these factors, and probably some others, have not been properly assessed among the correlations. It is necessary, therefore, to select correlations drawn from test conditions which approach most closely those of the particular reactor in mind. Certainly the existence of so many correlations does not mean that heat transfer problems in reactor design have been solved. It also should be remembered that these correlations are for simple channels.

On the other hand, there are several important areas of agreement among the laboratories. This lends credence to all of the data and engenders hope for a general correlation. These areas of agreement were largely pointed out in the discussions above of the work performed at Columbia University: (1) the burnout heat flux generally decreases with increasing coolant quality; (2) with uniform axial heat generation, the burnout generally occurs at the downstream end of the heated assembly; (3) an inversion occurs in the effect of mass velocity upon the burnout heat flux as the coolant quality increases from negative values (subcooled) to positive values (boiling) for mass velocities less than about 4,000,000 $lb/(hr)(ft^2)$; and (4) the burnout heat flux at a given coolant quality fraction decreases with increasing pressure in the boiling region, whereas it is independent of pressure in the subcooled region at a given quality fraction (or degrees of subcooling), over the pressure range of interest to the Power Reactor Program.

Additionally, support is seen for the observation of Bennett[48,49] of a critical value for the superficial steam mass velocity (χ G) of ~400,000 $lb/(hr)(ft^2)$, at which the burnout heat flux decreases rapidly. This support comes from the Bettis correlation and the Levedahl correlation, as well as from the work of Kon'kov and Modnikova[50]; Bennett had already cited the C.I.S.E. data as supporting his own data.

APPLICATION TO REACTOR DESIGNS

The relevance to the Power Reactor Program of the various studies conducted at Columbia University and other laboratories was pointed out in the separate discussions above. The objective in these studies was to be able to predict the margin of safety from heat transfer burnout for a particular reactor. In this section more detail is given to show that an adequate, but not excessive, margin of safety was indicated from these studies, first for the concept of a boiling D_2O reactor fueled with 19-rod bundles and then, in more detail, for the reference liquid-D_2O-cooled power reactor with nested-tube assemblies.

Several Margins of Safety to Consider

The margin of safety from burnout can be depicted by a plot showing simultaneously the burnout characteristic and the operating characteristic (the actual heat flux distribution). The coordinates are heat flux and coolant quality. Subcooled liquid is represented by negative quality. The burnout characteristic and the operating characteristic must be drawn, as nearly as possible, for the same geometry, pressure, and mass velocity. When families of these characteristics with mass velocity as the parameter are presented in a single plot, as in this report, it is necessary to select the pair having the same mass velocity to measure the margin of safety from burnout. The burnout characteristic may be drawn directly from the experimental data, or calculated from an appropriate correlation. The operating characteristic is calculated from the channel power, axial power distribution, geometry, flow, inlet enthalpy, and pressure.

The burnout safety factor (BOSF) can be determined from such a plot. It is the minimum ratio of the heat flux for burnout to the actual heat flux, where the ratio is evaluated at all points along the length of the channel for the local conditions of pressure, flow, coolant enthalpy (quality or subcooling), geometry, and actual heat flux that exist in normal operation. The position along the length of the channel is represented by the local quality, which is the abscissa in the plot described above.

In determining the BOSF, adjustments are usually made to both the operating and the burnout characteristics to allow for deviations from nominal design conditions - i.e., for the hot spot and hot channel effects - as a result of practical clearances, tolerances in fabrication, and tolerances in monitoring the reactor operation. When such factors are applied, the BOSF then represents the margin that exists in normal operation to allow for the uncertainties in the burnout phenomenon, which might be measured in terms of the number of standard deviations for the burnout correlation. It also represents in a qualitative way the margin that is available for an operational upset or accident.

A minimum BOSF of 1.5 was specified for the first reactor designs under the Power Reactor Program.[1] Thus, for the reference 100 megawatt (electric) reactors, by the special correlations developed from the Columbia burnout data (Section I,A above), the BOSF ranged from 1.45 to 2.39.[2] The required minimum BOSF was later increased to a range of 2.00 to 2.75, depending on reactor type, in the computer program for the optimum power reactor.[6] The BOSF specification in the early design stage incorporated allowances both for the uncertainties in the burnout heat flux and for the hot spot effects. This enabled the direct use of nominal design conditions for the operating and burnout characteristics. For the HWCTR, which was an actual operating reactor, a detailed evaluation of hot spot effects was made. The power level of the HWCTR was limited by the requirement that BOSF be equal to or greater than 1.8 during normal operation, and not less than 1.4 in any temporary abnormal condition, after proper allowance for the hot spot effects.

The BOSF is not a complete measure of the margin of safety from burnout. In order for burnout to actually occur, as long as the integrity of the fuel assembly is maintained, it is necessary to deviate from the normal conditions in one of several unfavorable directions. In the discussions which follow for nested tubes, four margins of safety from burnout are considered: (1) the safety factor from the burnout heat flux (BOSF) at normal power and normal flow; (2) the power increase factor for burnout (BPIF) at normal flow; (3) the fraction of normal flow for burnout (BFF) at normal power; and (4) the fraction of normal pressure for burnout (BpF) at normal power and flow.

This concern for deviations from normal conditions applies even more to the reference liquid-D_2O-cooled reactor because of its higher heat flux and higher mass velocity than to the boiling D_2O concept. In the interest of high thermal efficiency, the liquid-cooled reactor is designed with the effluent temperature close to boiling. A relatively small increase in power or decrease in flow for the liquid-D_2O-cooled reactor would result in bulk boiling within the fuel assembly. The heat flux limits are shown by the data from Columbia University and from other laboratories to decrease sharply in this transition from subcooled to boiling conditions, particularly for the higher mass velocities. It is planned to monitor the effluent temperature of each fuel assembly to ensure that a boiling temperature is not reached. This does not diminish the need to know the margins from burnout that are available, not only for concern of the channels within the assembly that are not monitored individually but also for the accident situations.

An increase in coolant quality is also accompanied by an increase in the pressure differential needed to maintain the flow. This effect reduces the flow and thereby increases further the coolant quality. The effect upon coolant flow of a power increase or a pressure decrease was not factored into this analysis.

The Design with 19-Rod Bundles

In Figure 1, the burnout characteristics that were determined experimentally for the 19-rod bundle are compared to the nominal operating characteristic projected for that assembly in a power reactor. From Figure 1 it is estimated that the burnout safety factor (BOSF) is about 1.8. This is without application of hot spot factors and without allowances for the scatter in the burnout data and for the limited experimental coverage of the possible operating conditions and geometrical variations. In this light, a BOSF of 1.8 is not too conservative for promoting further development of the reactor concept though it would otherwise appear to be more than adequate. For a higher BOSF it is evident from Figure 1 that either the heat flux or the exit quality of the operating characteristic could be lowered. The heat flux for a given design power level is lowered either by an enlarged reactor for more fuel assemblies or by a new fuel assembly with more and smaller rods in the bundle. The exit quality is lowered by increasing the coolant flow for the design.

The Design with Nested Tubes

The normal operating characteristic is shown in Figure 2 (Curve A) for the tubular channel of the nested-tube fuel assembly of the liquid-D_2O-cooled power reactor. The coolant is subcooled over the full length of the channel. The conditions are a tube ID of 0.6 inch, a pressure of 1500 psia, a mass velocity of 8,800,000 lb/(hr)(ft^2), a maximum heat flux of 465,000 pcu/(hr)(ft^2), and a chopped-cosine axial flux distribution. The applicable burnout characteristic is that for subcooled flow, namely, the SRL correlation;[6] this characteristic is also given in Figure 2. The burnout characteristic lies far above the nominal operating characteristic by an amount corresponding to BOSF = 2.8.

Two operating characteristics corresponding to deviations from normal in the direction of burnout are also shown in Figure 2 for the tubular channel of the nested-tube assembly in the liquid-D_2O-cooled reactor. A power increase at normal flow is shown by Curve B; a flow decrease at normal power, by Curve C. In both cases, bulk boiling occurs in the coolant channel. The burnout characteristics for these cases are best represented by the data from Columbia University. These data cover the transition from subcooled to bulk boiling conditions. The best fit through the data is shown in Figure 2 as a family of dashed curves. The burnout characteristics correspond to data from 0.5-inch-ID tubes whereas the reference design is 0.6-inch ID. Analysis of the data for other tube sizes shows that burnout varies inversely as the square root of the diameter.[40] This small adjustment for diameter is omitted here for simplicity. Thus, from Figure 2, a 70% or more increase in power at constant flow (Curve B) is indicated for burnout, or BPIF >1.70. Also, burnout is expected at 45% of normal flow at constant power (Curve C), or BFF = 0.45. The tangency of Curve C to the proper burnout characteristic – a mass velocity of 4,000,000 lb/(hr)(ft^2) – is evident. The exact power increase to burnout cannot be as accurately discerned because of insufficient coverage of the burnout conditions.

The margins of safety from burnout for the annular channels of the nested-tube assembly in the liquid-D_2O-cooled reactor were evaluated from the correlations of the Savannah River Laboratory,[7] the Hanford Thermal Hydraulics Laboratory,[37] and the General Electric Company (GEAP).[38] These sources have been discussed above. The margins of safety are presented in Table I together with those for the tubular channel. The outer annulus was internally heated while the "middle" annulus was internally-and-externally heated. Only the smaller "middle" annulus is shown; the larger "middle" annulus would have had greater margins from burnout because of a lower operating heat flux. The three kinds of channels in the nested-tube assembly, according to the five special correlations employed, have quite similar margins of safety from burnout. The average values are as follows: BOSF, 2.7; BPIF, 1.7; and BFF, 0.4. For the outer annulus, where two or three of the correlations were employed to estimate the same margin, the agreement is acceptable considering the stage of the reactor development.

TABLE I

Margins of Safety from Burnout for the Nested-Tube Fuel Assembly in the Liquid-D_2O-Cooled Power Reactor (1500 psia)

Margin	References	Outer Annulus	Middle Annulus	Tubular Channel
BOSF, the heat flux ratio for burnout at normal power and flow.	SRL[6]	-	-	2.8
	SRL[7]	2.7	-	-
	Hanford[37]	3.3	2.9	-
	GEAP[38]	2.0	-	-
BPIF, the power ratio for burnout at normal flow.	Hanford[37]	1.7	1.8	-
	GEAP[38]	1.6	-	-
	Columbia U.[11]	-	-	>1.7
BFF, the flow ratio for burnout at normal power.	Hanford[37]	0.50	0.42	-
	GEAP[38]	0.38	-	-
	Columbia U.[11]	-	-	0.46

In the analysis above, the pressure was 1500 psia whereas the reference design called for a pressure of 1700 psia. The lower pressure was used, in part, because the preferred data and correlation applied only up to 1500 psia. (In fact, the SRL correlations did not employ data beyond 1200 psia, though there was no effect of pressure up to that point.) Furthermore, a separate analysis shows that 1500 psia is probably a better operating pressure for the power reactor. This analysis was made with the GEAP correlation for burnout in internally heated annuli.[38] While the range of applicability of this correlation extended only to 1500 psia, it was extrapolated on the evidence from other correlations that the burnout heat flux at a given quality decreases as the pressure increases up to 2000 psia and higher. The results of this separate analysis are given in Table II; three margins of safety (BOSF, BPIF, and BFF) are tabulated for pressures from 1700 down to 733 psia. As the pressure decreases, the exit quality increases, being negative at 1700, 1500, and 1300 psia, but positive at 1000 and 733 psia. The saturation pressure corresponding to the common inlet temperature of 264°C is 733 psia. The table shows that the three margins of safety vary insignificantly with the change in pressure from 1700 to 1300 psia. At 733 psia the BOSF and BPIF are close to 1.0; hence, the fraction of normal pressure for burnout at normal power and flow (BpF) is about 0.5.

TABLE II

Effect of Operating Pressure on the Margins from Burnout: Normal Power and Flow at 100% of Reference Design Values

Outer annulus of nested-tube fuel assembly
Liquid D_2O enters reactor at 264°C
Maximum heat flux (100% of reference design)
= 381,000 pcu/(hr)(ft^2)
Mass velocity (100% of reference design)
= 5,900,000 lb/(hr)(ft^2)

Operating Pressure, psia	Saturation Temp, °C	Exit Quality Fraction	Exit Subcooling, °C	Burnout Safety Factor, BOSF	Burnout Power Increase Factor, BPIF	Burnout Flow Factor, BFF
1700	322	-0.112	21	1.9	1.6	0.37
1500	312	-0.058	12	2.0	1.6	0.38
1300	302	-0.008	2	2.0	1.5	0.40
1000	284	+0.064	Negative	1.6	1.3	0.47
733	264	+0.127	Negative	1.2	1.1	0.67

Optimization Aspects

In seeking the optimum conditions for a D_2O-cooled reactor, it is necessary to determine the cost for various combinations of conditions which have the same specified margin of safety from burnout. A specified margin from heat transfer burnout was characteristically the design basis in the Power Reactor Program. One sequence of combinations is a progressively lower heat flux with a correspondingly lower design pressure for the same fuel assembly, the same temperature of liquid D_2O entering the reactor, and the same rise in enthalpy of the coolant in passing through the reactor. The same enthalpy rise is obtained by taking the flow proportional to the power. By taking the same enthalpy rise and inlet temperature, the reactor will have essentially the same steam generator size, steam conditions, thermal efficiency, and circulation of D_2O coolant for the same reactor power, as long as the average reactor effluent temperature is below boiling.

The size of the reactor increases as the design heat flux decreases. The increased cost associated with increased size is pitted in the economic equation against the lower cost associated with the correspondingly lower pressure. In the reference design for the liquid-D_2O-cooled reactor, which has been evaluated above, let us say that the margins from burnout for this design at an operating pressure of 1500 psia, as given in Table II, are acceptable. The reference design is then one combination of design conditions (reactor size and pressure) in a sequence for cost evaluation. For another combination in the sequence, take a design in which the normal operating conditions of power and flow per assembly are 80% of those for the reference design. In Table III, the margins of safety from burnout are given for the outer annulus of this new design at pressures from 1700 down to 733 psia, just as was done in Table II for the reference design. For 80% power and flow, the margins of safety at 1000 psia are essentially the same as for 100% power and flow at 1500 psia. The same conclusion is obtained for the tubular channel using Columbia University data, as shown by Figures 2 and 3.

Note in Tables II and III that at 1000 psia the exit quality is about 6%. This is for the maximum-powered fuel assembly. The exit quality of the mixed effluent from all assemblies might be less than 6%, depending upon the distribution of coolant flow and assembly power. A mixed effluent of low quality could be pushed through a steam generator to produce H_2O steam; however, the possibility exists in the 1000 psia case from Table III for separation of D_2O steam, which would be delivered to the turbogenerator. However, the inlet subcooling of 20°C associated with an exit quality of about 6% is too large to enable a classical direct-cycle boiling D_2O reactor. For this case from Table III, it is necessary to extract additional heat from the liquid D_2O phase, which is separated outside the reactor from the primary D_2O

steam. This extraction of heat could be accomplished by flashing or by steam generators to produce additional D_2O steam of lower pressure. This would be a dual-cycle power reactor.

TABLE III

Effect of Operating Pressure on the Margins from Burnout: Normal Power and Flow at 80% of Reference Design Values

Outer channel of nested-tube fuel assembly
Liquid D_2O enters reactor at 264°C
Maximum heat flux (80% of reference design)
= 305,000 pcu/(hr)(ft²)
Mass velocity (80% of reference design)
= 4,700,000 lb/(hr)(ft²)

Operating Pressure, psia	Saturation Temp, °C	Exit Quality Fraction	Exit Subcooling, °C	Burnout Safety Factor, BOSF	Burnout Power Increase Factor, BPIF	Burnout Flow Factor, BFF
1700	322	-0.112	21	2.4	1.8	0.32
1500	312	-0.058	12	2.5	1.8	0.33
1300	302	-0.008	2	2.5	1.8	0.35
1000	284	+0.064	Negative	2.2	1.6	0.40
733	264	+0.127	Negative	1.7	1.3	0.52

A classical boiling D_2O reactor (no flashing, no steam generators) can be obtained in the particular sequence above with acceptable margins of safety from burnout by taking a still lower fraction of the reference design power and flow, together with a still lower operating pressure. Thus, the following conditions are obtained for a boiling D_2O reactor:

Operating pressure	800 psia
Percent of reference design power	60%
Percent of reference design flow	60%
Outer channel maximum heat flux	230,000 pcu/(hr)(ft²)
Outer channel mass velocity	3,600,000 lb/(hr)(ft²)
Inlet temperature	264°C
Inlet subcooling	5.5°C

Exit quality (outer channel and reactor)	11%
Temperature of feedwater (from turbine complex)	217°C
Burnout safety factor (BOSF)	2.6
Burnout power increase factor (BPIF)	1.7
Burnout flow factor (BFF)	0.39

It was necessary to extrapolate the GEAP correlation to lower heat fluxes than are claimed for the correlation (see Figures A-9, A-10, and A-11) in estimating the fraction of normal flow for burnout at normal power (BFF); this is probably conservative. Also, for the above feedwater temperature and reactor exit quality, it was assumed that the same rise in coolant enthalpy occurs in all channels of the reactor as in the outer annulus of the maximum-powered fuel assembly.

The optimum design for the liquid-D_2O-cooled reactor or for the boiling D_2O reactor is found ultimately by repeating the sequence, as above, for the same fuel assembly design but with other values for the temperature of the liquid D_2O entering the reactor and for the rise in coolant enthalpy (or for the ratio of power to coolant flow). The economic advantage, if any, of the boiling D_2O reactor over the liquid-D_2O-cooled reactor depends upon whether the reductions in unit cost (dollars per kilowatt-hour) from 1) lower design pressure, 2) elimination of steam generators, and 3) higher steam pressure at the turbine (higher thermodynamic efficiency) outweigh the increases in unit cost from 1) greater reactor size, 2) greater inventory of D_2O, and 3) provisions to avoid D_2O losses in the turbine system.

AREAS FOR FURTHER STUDY

More effort in analysis of the burnout data at hand from the many laboratories throughout the world is needed to guide future experimentation. Mathematical or graphical expressions for the heat flux limit are needed by reactor designers and by the operators of these reactors. Correlations adequate for the heavy-water-moderated reactor concepts are available only for subcooled water flowing in tubular and annular channels. Many correlations have been presented for the boiling region, but they all have their limitations, and their differences need to be reconciled. A simple correlation is not anticipated; the desired correlation may prove to be a design manual. It is expected that the correlation will have several parts; but there must be continuity in the matching of these parts.

The nature of burnout needs to be defined clearly in the proposed correlation. The limiting factor determining heat transfer capability is a limit on surface temperature rather than on the heat flux. The limit to be placed on surface temperature depends upon the material of which the heated object is composed and upon the duration at that temperature. The surface temperature — for given conditions of heat flux, coolant enthalpy, mass velocity, geometry, and pressure — depends upon the mechanism or mode of the heat transfer process in that region of operation. At a critical value of heat flux, which is termed burnout in this report, a change in mode occurs. The nature of the change, as it affects the surface temperature, depends upon the modes that are involved in the transition, as follows:

A. In one common form of burnout there is a change from nucleate boiling to film boiling at the heated surface. This is encountered in the subcooled and low quality boiling region. With nucleate boiling, the surface temperature is only a few degrees above the saturation temperature, and it increases only slightly with increasing heat flux. The "departure from nucleate boiling" (DNB) is accompanied by a sharp increase in surface temperature to an intolerably high value which is the result of an insulating vapor film at the heated surface. The term burnout is descriptive for this form even though, in the experiment, the test may have been terminated at the first evidence of DNB in order to preserve the test assembly, i.e., at "incipient burnout."

   There will always be some misgivings, however, when the heat transfer test is terminated upon only a small increase in temperature or upon the development of temperature oscillation, even though a few tests may have been conducted to show that melting follows shortly after "incipient burnout" in that particular region of operation.

B. At higher quality, the transition in the mode of heat transfer is not necessarily from nucleate boiling to film boiling at the wetted wall but, say, from forced convection with a wetted wall to forced convection with a dry wall, which is termed "dryout." In this case, the change in surface temperature is less than in the DNB case; in fact, the change may be tolerable. The magnitude of the heat flux is not only lower initially but the heat transfer coefficient is also higher after dryout than in the DNB case; this results in a smaller increase in surface temperature. Burnout is certainly not descriptive here. In this case, the surface temperature is an important parameter, particularly beyond the point of transition between the two modes.

Thus, in heat transfer studies, the increase in surface temperature should be followed for several hundred degrees beyond DNB or dryout to identify the nature of the transition and to establish the surface temperature parameters.

The correlation for the heat transfer limit – i.e., the transition heat flux or the heat flux at a given surface temperature limit – will certainly prove to be complex, for there are many variables. The principal variables are mass velocity, local coolant enthalpy, inlet enthalpy conditions (subcooled or quality), pressure, axial flux shape, and geometry. The geometrical factors are the channel shape, the hydraulic diameter, the ratio of heated to nonheated surface, the number of surfaces of the heated solid that are cooled, the thickness and composition of the heated solid, the orientation (horizontal or vertical), and the inlet configuration. For rod bundles, the array of the rods within the bundle affects the power capability. The fuel assembly may employ film trippers, turbulence promotors, and/or roughened surface.[51,52,53] There are hot channel effects from eccentricity; there are hot spot effects, such as from a spacing rib contacting the heated surface and from local peaking of the heat generation (i.e., from fuel defects).

Finally, there is need for additional experimentation and analysis in the following areas:

1) the extent to which burnout is not determined solely by the local conditions;

2) flow instability in "compressible" systems and in parallel channels; and the use of added resistance (orifices) to stabilize the flow;

3) the effect of flow oscillation upon the heat flux limit;

4) the occurrence and effect of vibration in heat transfer;

5) the standard deviation for the chosen burnout correlation;

6) the margin of safety from burnout expressed as the probability of burnout, or the risk of burnout, or the confidence level; this would be accomplished by combining according to probability theory the standard deviation for the burnout correlation with the standard deviations for all of the design and operating parameters.[54,55]

7) transient behavior in simple and complex assemblies;

8) in-pile versus out-of-pile burnout; and,

9) the regimes of two-phase flow with and without heat addition.

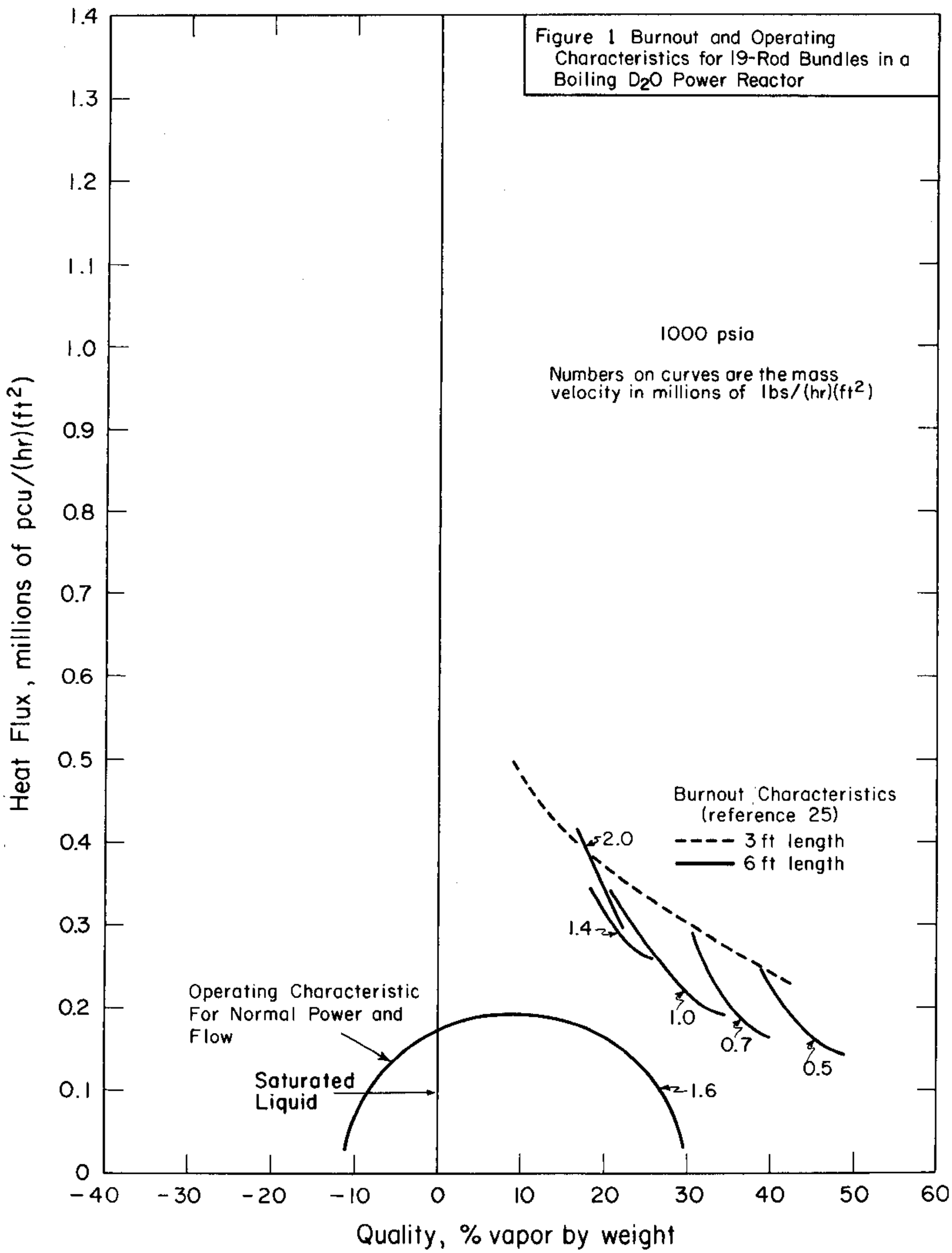
Figure 1 Burnout and Operating Characteristics for 19-Rod Bundles in a Boiling D2O Power Reactor
1000 psia
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft2)
Burnout Characteristics (reference 25)
3 ft length
6 ft length
2.0
1.4
1.0
0.7
0.5
1.6
Operating Characteristic For Normal Power and Flow
Saturated Liquid
Heat Flux, millions of pcu/(hr)(ft2)
Quality, % vapor by weight
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
-40
-30
-20
-10
0
10
20
30
40
50
60

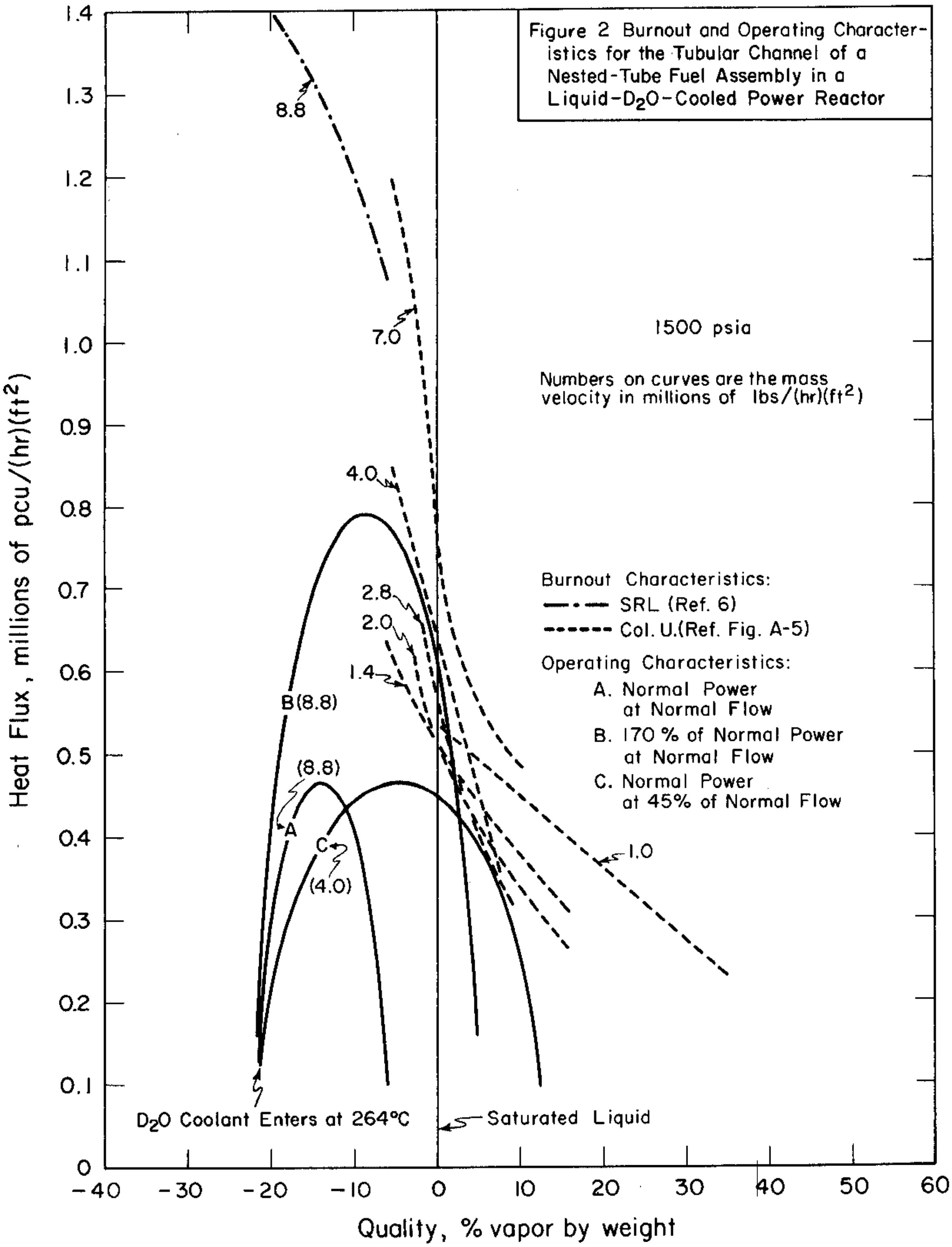

Figure 2 Burnout and Operating Characteristics for the Tubular Channel of a Nested-Tube Fuel Assembly in a Liquid-D_2O-Cooled Power Reactor

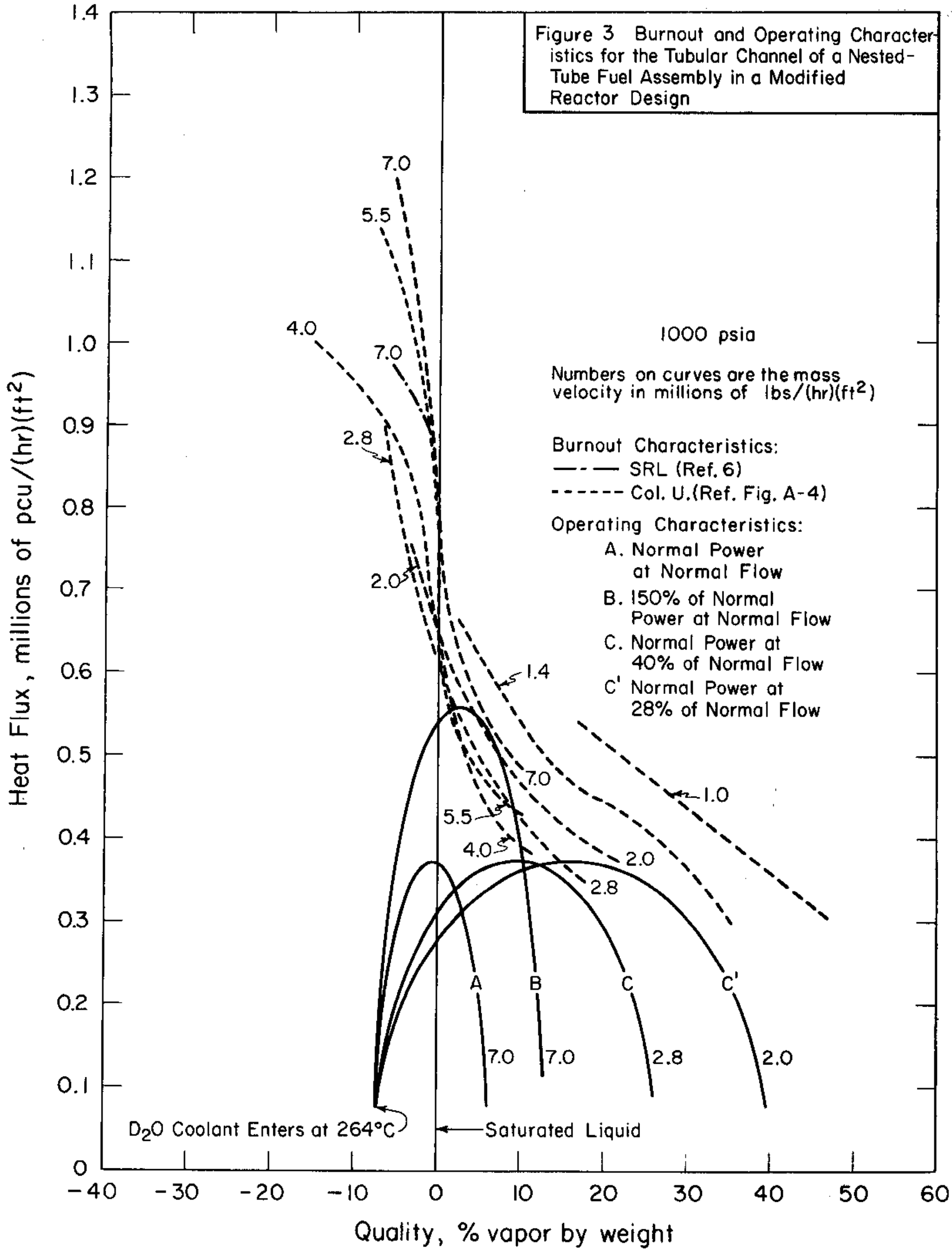

Figure 3 Burnout and Operating Characteristics for the Tubular Channel of a Nested-Tube Fuel Assembly in a Modified Reactor Design
1000 psia
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Burnout Characteristics:
SRL (Ref. 6)
Col. U.(Ref. Fig. A-4)
Operating Characteristics:
A. Normal Power at Normal Flow
B. 150% of Normal Power at Normal Flow
C. Normal Power at 40% of Normal Flow
C' Normal Power at 28% of Normal Flow
D₂O Coolant Enters at 264°C
Saturated Liquid
Heat Flux, millions of pcu/(hr)(ft²)
Quality, % vapor by weight

APPENDIX

PREDICTIONS BY VARIOUS CORRELATIONS IN RELATION TO EXPERIMENTAL OBSERVATIONS MADE AT COLUMBIA UNIVERSITY

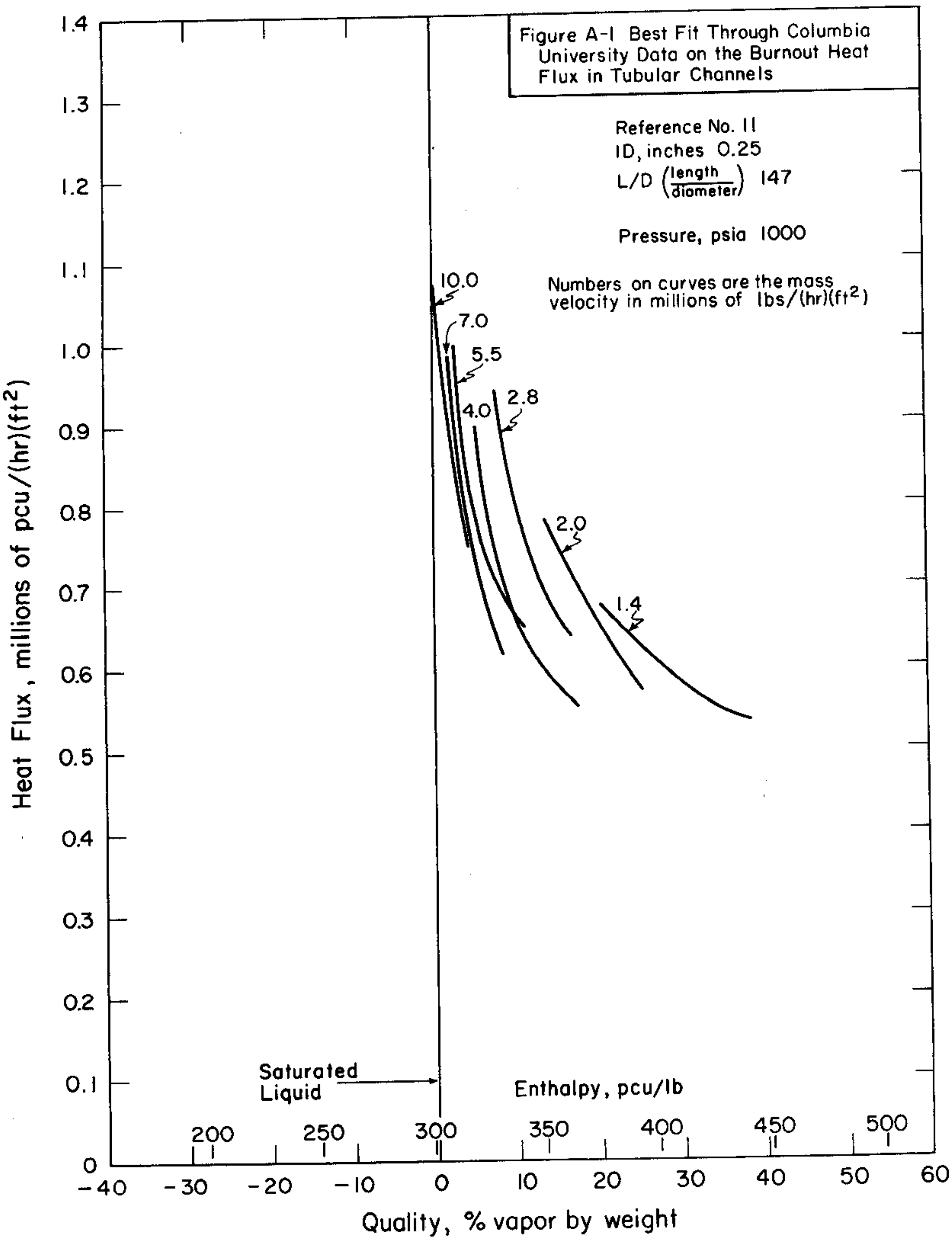
Figure A-1 Best Fit Through Columbia University Data on the Burnout Heat Flux in Tubular Channels
Reference No. 11
ID, inches 0.25
L/D (length/diameter) 147
Pressure, psia 1000
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
10.0
7.0
5.5
4.0
2.8
2.0
1.4
Heat Flux, millions of pcu/(hr)(ft²)
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
Saturated Liquid
Enthalpy, pcu/lb
200
250
300
350
400
450
500
-40
-30
-20
-10
0
10
20
30
40
50
60
Quality, % vapor by weight

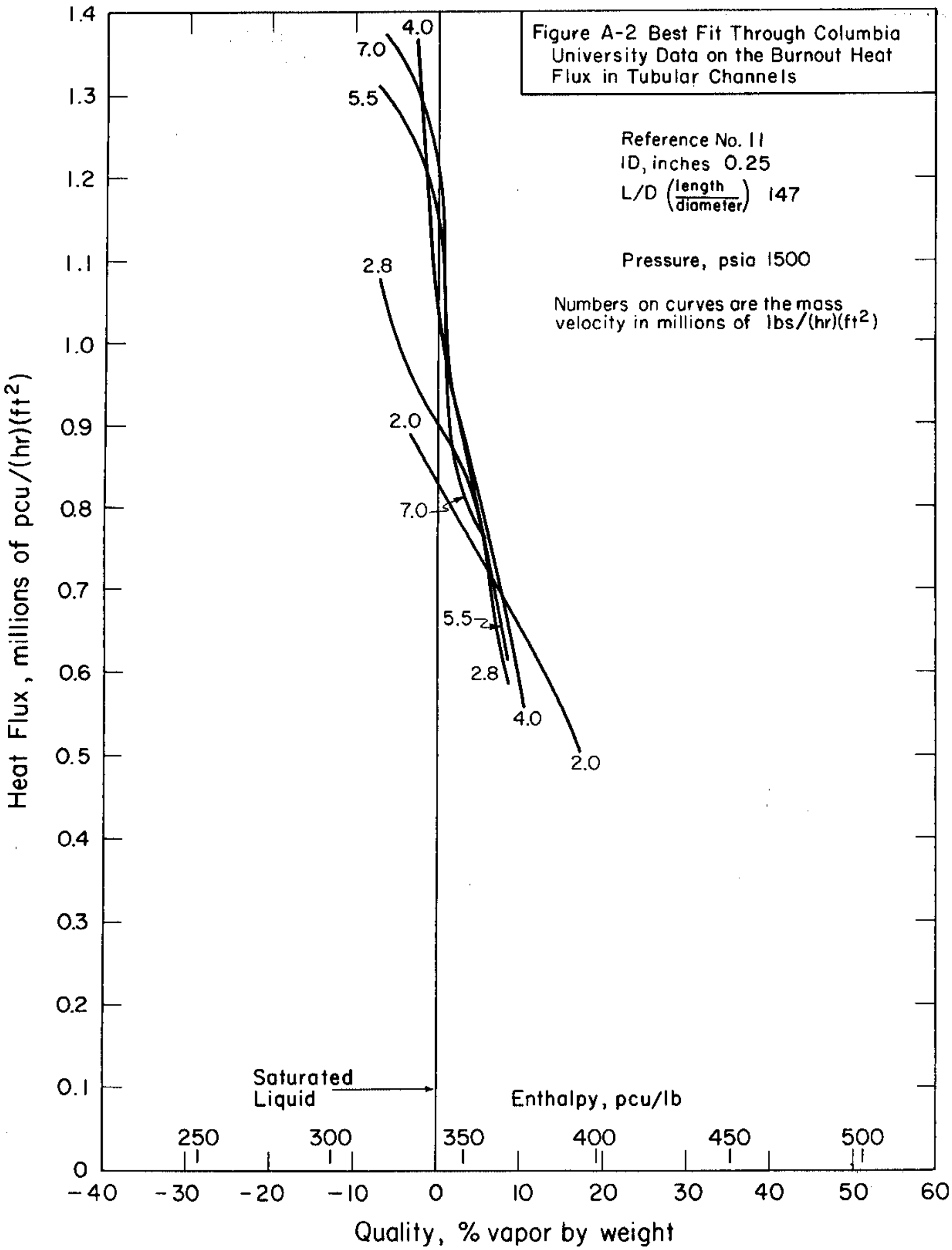
Figure A-2 Best Fit Through Columbia University Data on the Burnout Heat Flux in Tubular Channels
Reference No. 11
ID, inches 0.25
L/D (length/diameter) 147
Pressure, psia 1500
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Heat Flux, millions of pcu/(hr)(ft²)
Quality, % vapor by weight
Enthalpy, pcu/lb
Saturated Liquid
7.0
4.0
5.5
2.8
2.0
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
250
300
350
400
450
500
−40
−30
−20
−10
10
20
30
40
50
60

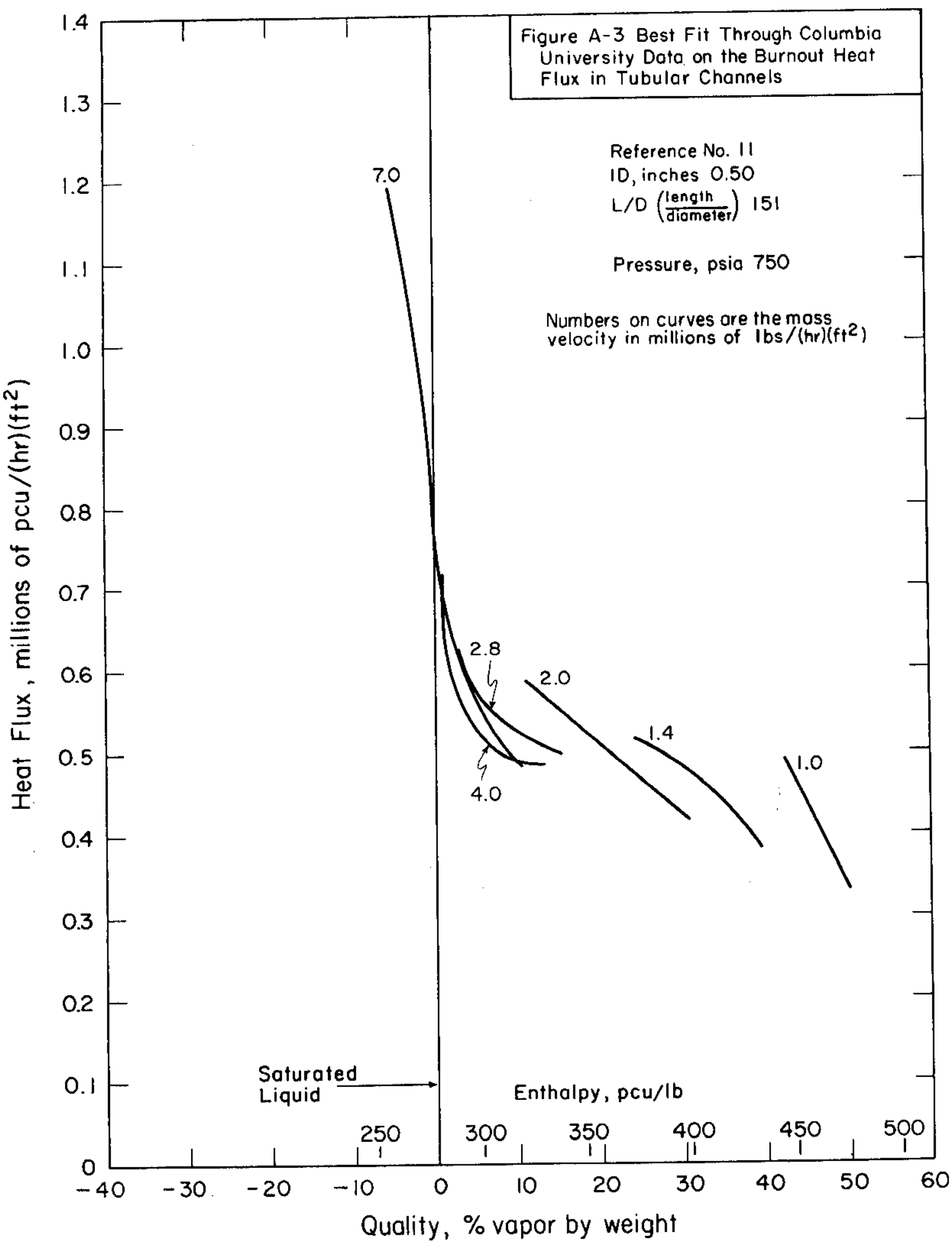

Figure A-3 Best Fit Through Columbia University Data on the Burnout Heat Flux in Tubular Channels
Reference No. 11
ID, inches 0.50
L/D (length/diameter) 151
Pressure, psia 750
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft2)
7.0
2.8
2.0
1.4
1.0
4.0
Saturated Liquid
Enthalpy, pcu/lb
250
300
350
400
450
500
Heat Flux, millions of pcu/(hr)(ft2)
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
-40
-30
-20
-10
0
10
20
30
40
50
60
Quality, % vapor by weight

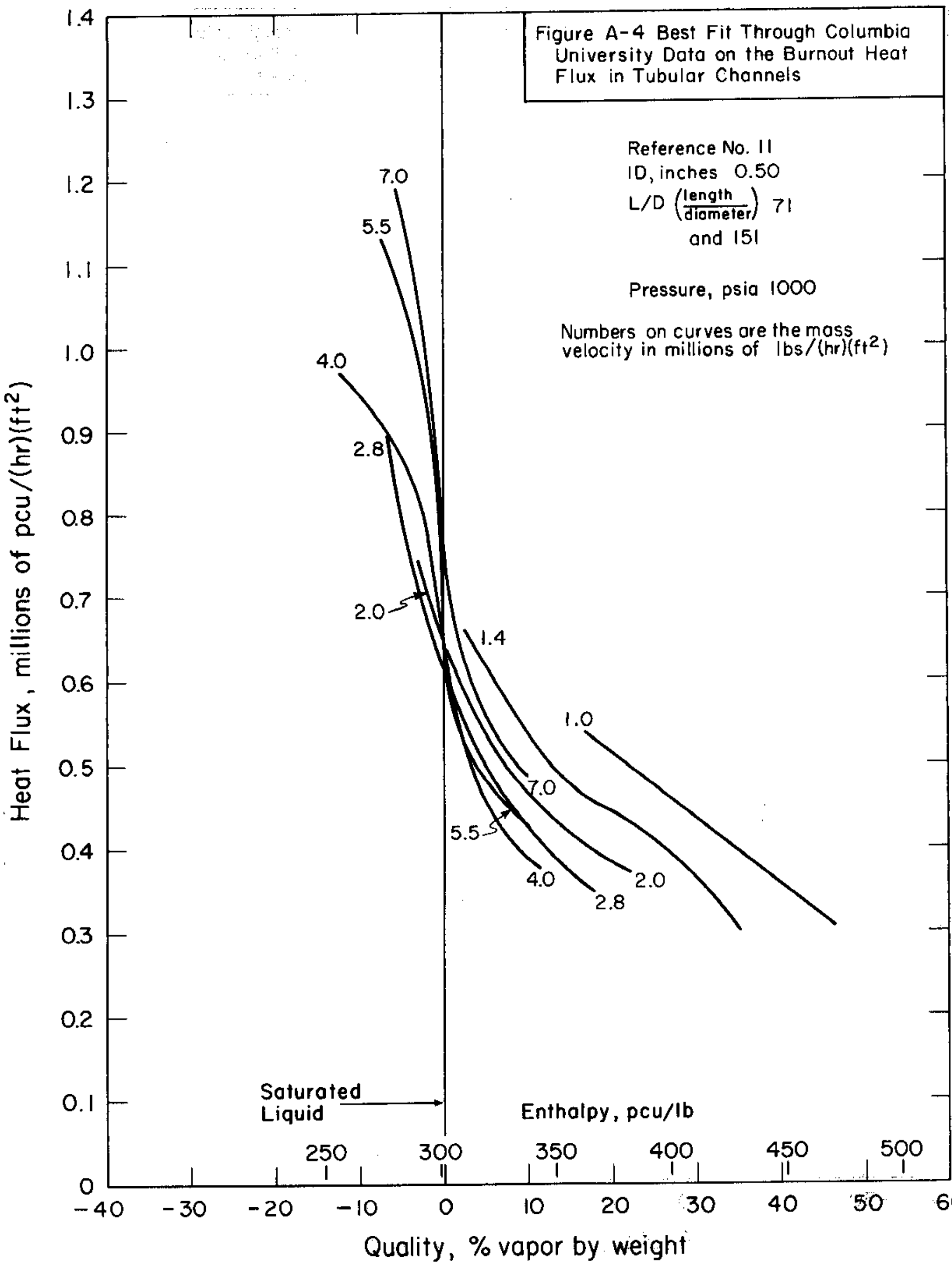

Figure A-4 Best Fit Through Columbia University Data on the Burnout Heat Flux in Tubular Channels
Reference No. 11
ID, inches 0.50
L/D (length/diameter) 71 and 151
Pressure, psia 1000
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Heat Flux, millions of pcu/(hr)(ft²)
Quality, % vapor by weight
Saturated Liquid
Enthalpy, pcu/lb
250
300
350
400
450
500
7.0
5.5
4.0
2.8
2.0
1.4
1.0
7.0
5.5
4.0
2.8
2.0
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
−40
−30
−20
−10
0
10
20
30
40
50
60

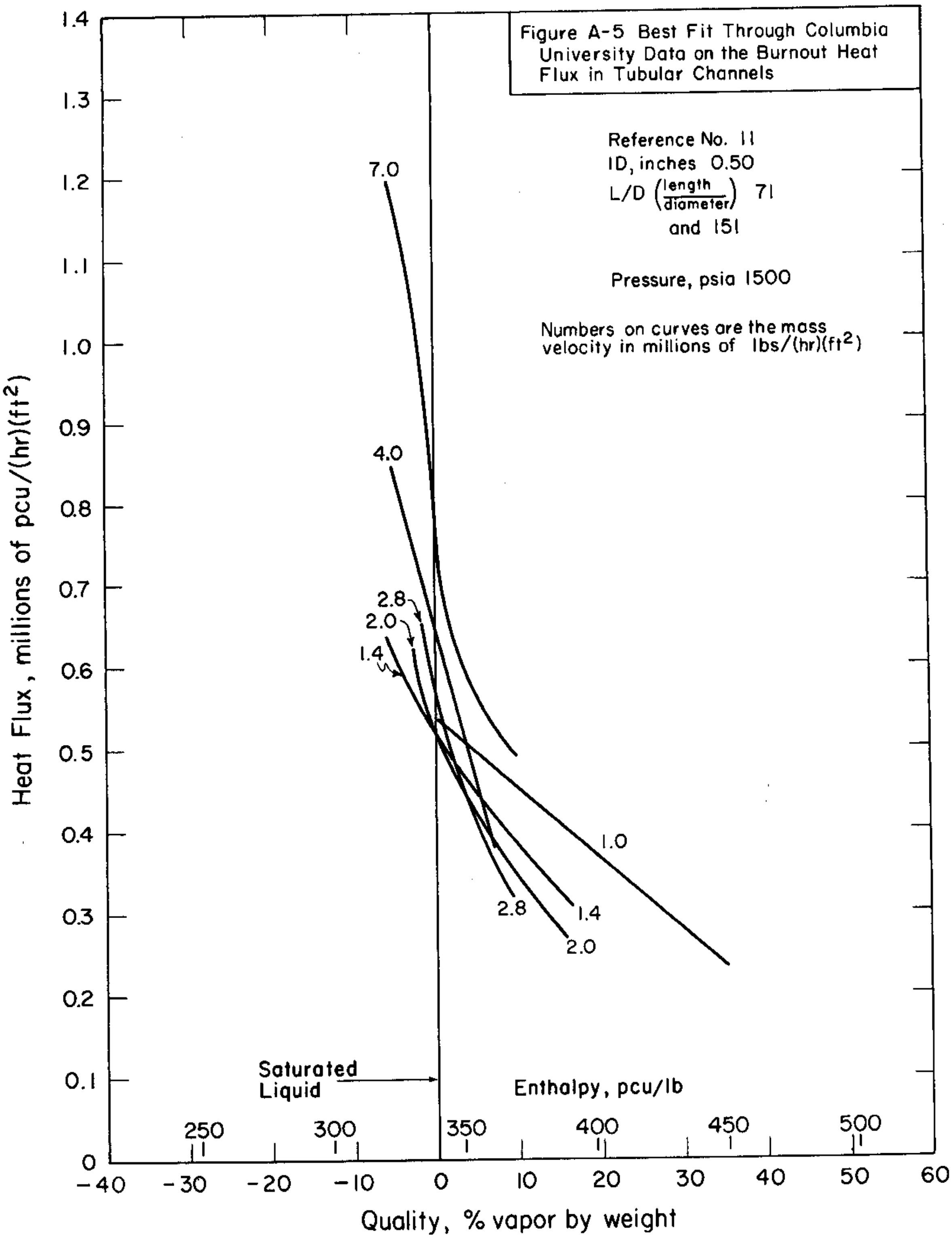
Figure A-5 Best Fit Through Columbia University Data on the Burnout Heat Flux in Tubular Channels
Reference No. 11
ID, inches 0.50
L/D (length/diameter) 71 and 151
Pressure, psia 1500
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Heat Flux, millions of pcu/(hr)(ft²)
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
7.0
4.0
2.8
2.0
1.4
1.0
1.4
2.8
2.0
Saturated Liquid
Enthalpy, pcu/lb
250
300
350
400
450
500
-40
-30
-20
-10
0
10
20
30
40
50
60
Quality, % vapor by weight

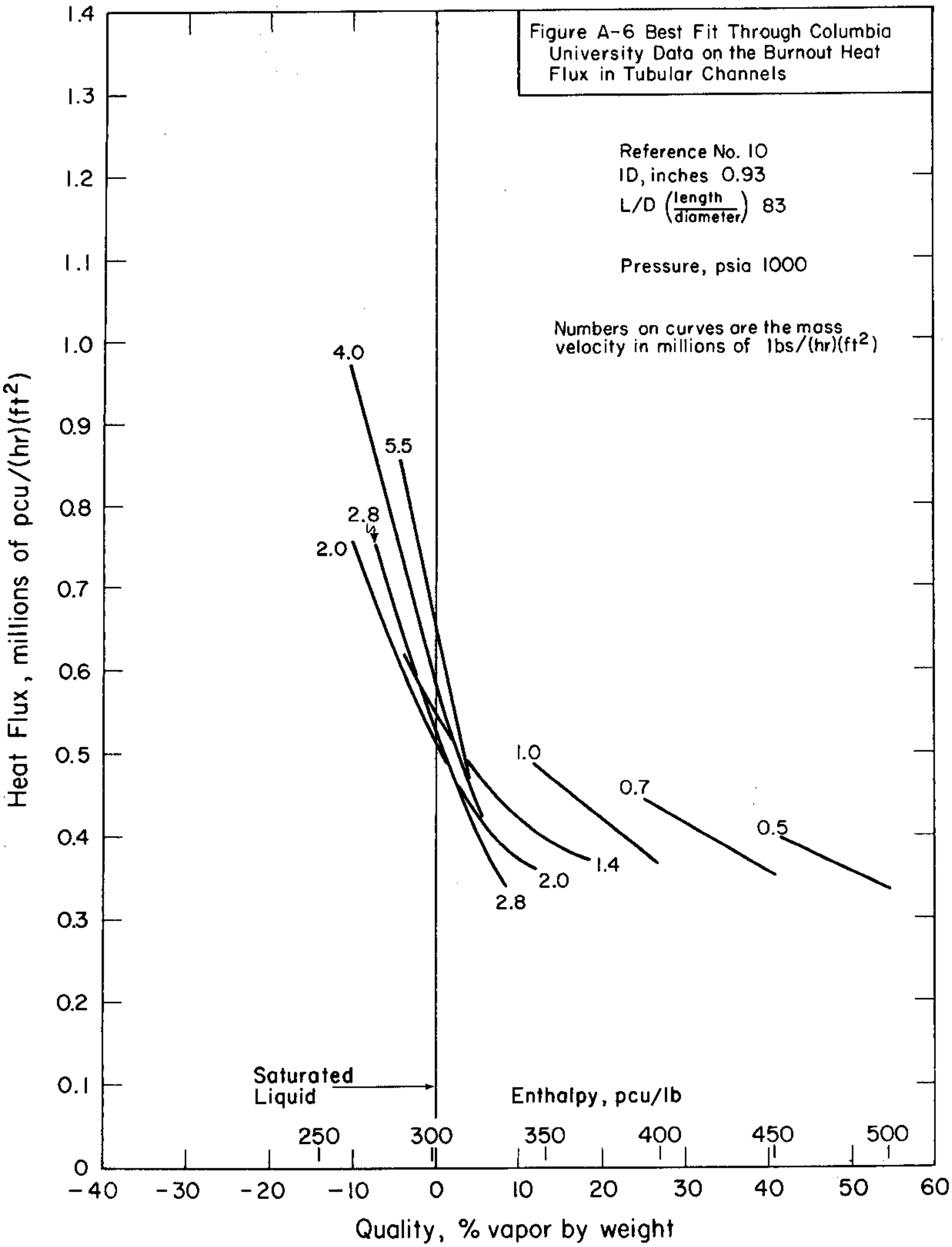
Figure A-6 Best Fit Through Columbia University Data on the Burnout Heat Flux in Tubular Channels
Reference No. 10
ID, inches 0.93
L/D (length/diameter) 83
Pressure, psia 1000
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft^2)
Heat Flux, millions of pcu/(hr)(ft^2)
Quality, % vapor by weight
Enthalpy, pcu/lb
Saturated Liquid
4.0
5.5
2.8
2.0
1.0
0.7
0.5
1.4
2.0
2.8
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
1.1
1.2
1.3
1.4
-40
-30
-20
-10
0
10
20
30
40
50
60
250
300
350
400
450
500

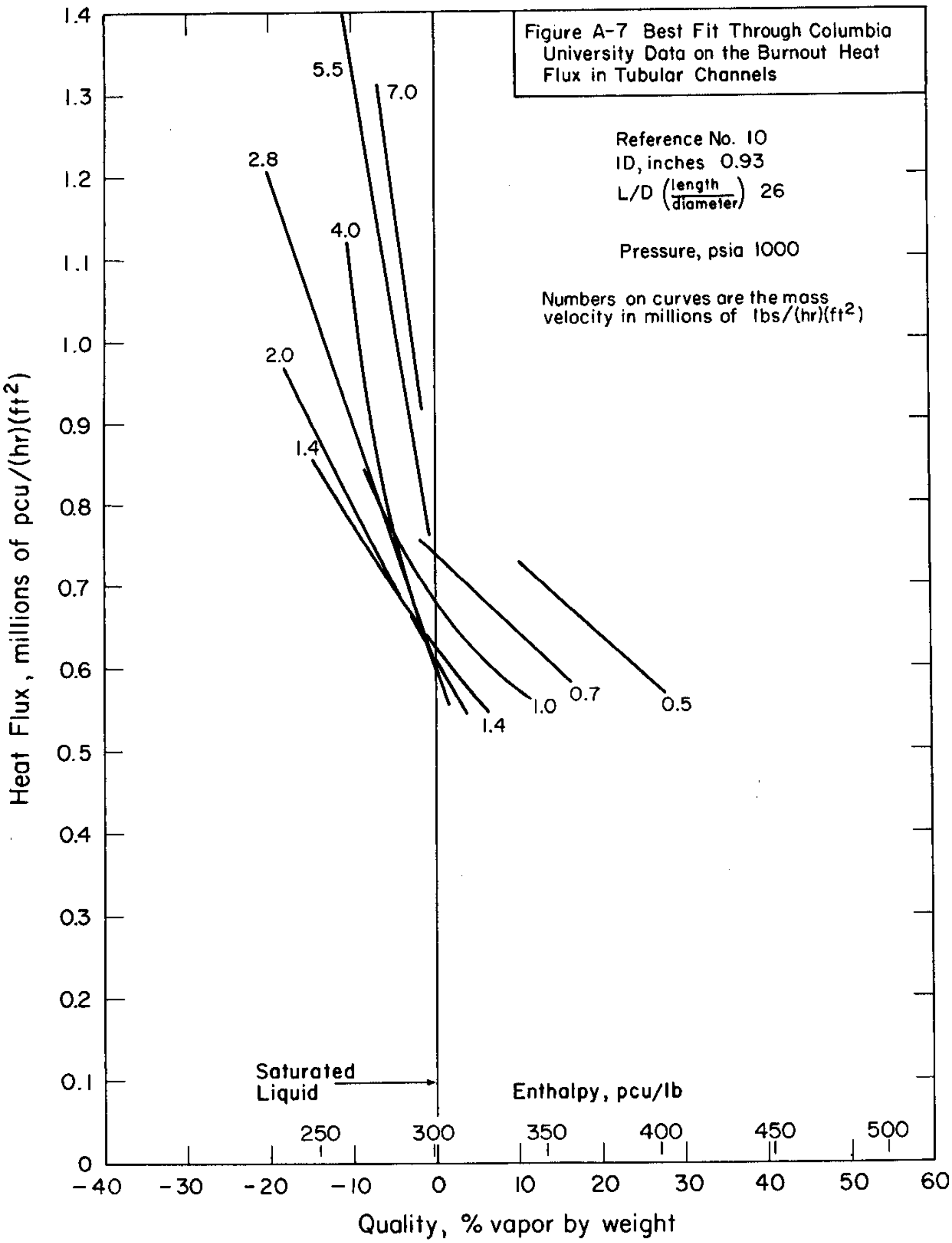

Figure A-7 Best Fit Through Columbia University Data on the Burnout Heat Flux in Tubular Channels
Reference No. 10
ID, inches 0.93
L/D (length/diameter) 26
Pressure, psia 1000
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
5.5
7.0
2.8
4.0
2.0
1.4
1.0
0.7
0.5
1.4
Saturated Liquid
Enthalpy, pcu/lb
250
300
350
400
450
500
Heat Flux, millions of pcu/(hr)(ft²)
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
1.1
1.2
1.3
1.4
-40
-30
-20
-10
0
10
20
30
40
50
60
Quality, % vapor by weight

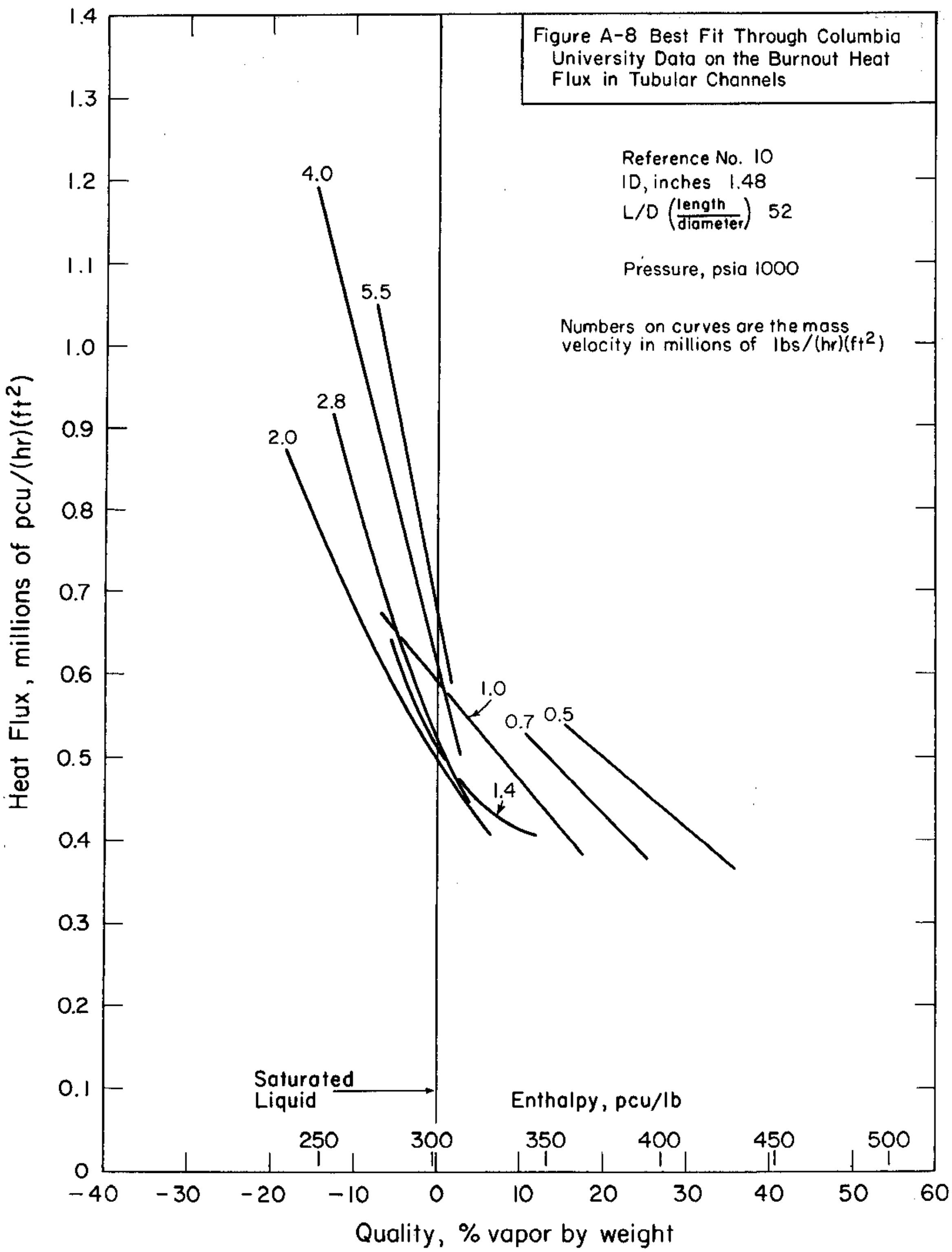
Figure A-8 Best Fit Through Columbia University Data on the Burnout Heat Flux in Tubular Channels
Reference No. 10
ID, inches 1.48
L/D (length/diameter) 52
Pressure, psia 1000
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
4.0
5.5
2.8
2.0
1.0
0.7
0.5
1.4
Saturated Liquid
Enthalpy, pcu/lb
250
300
350
400
450
500
Heat Flux, millions of pcu/(hr)(ft²)
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
−40
−30
−20
−10
0
10
20
30
40
50
60
Quality, % vapor by weight

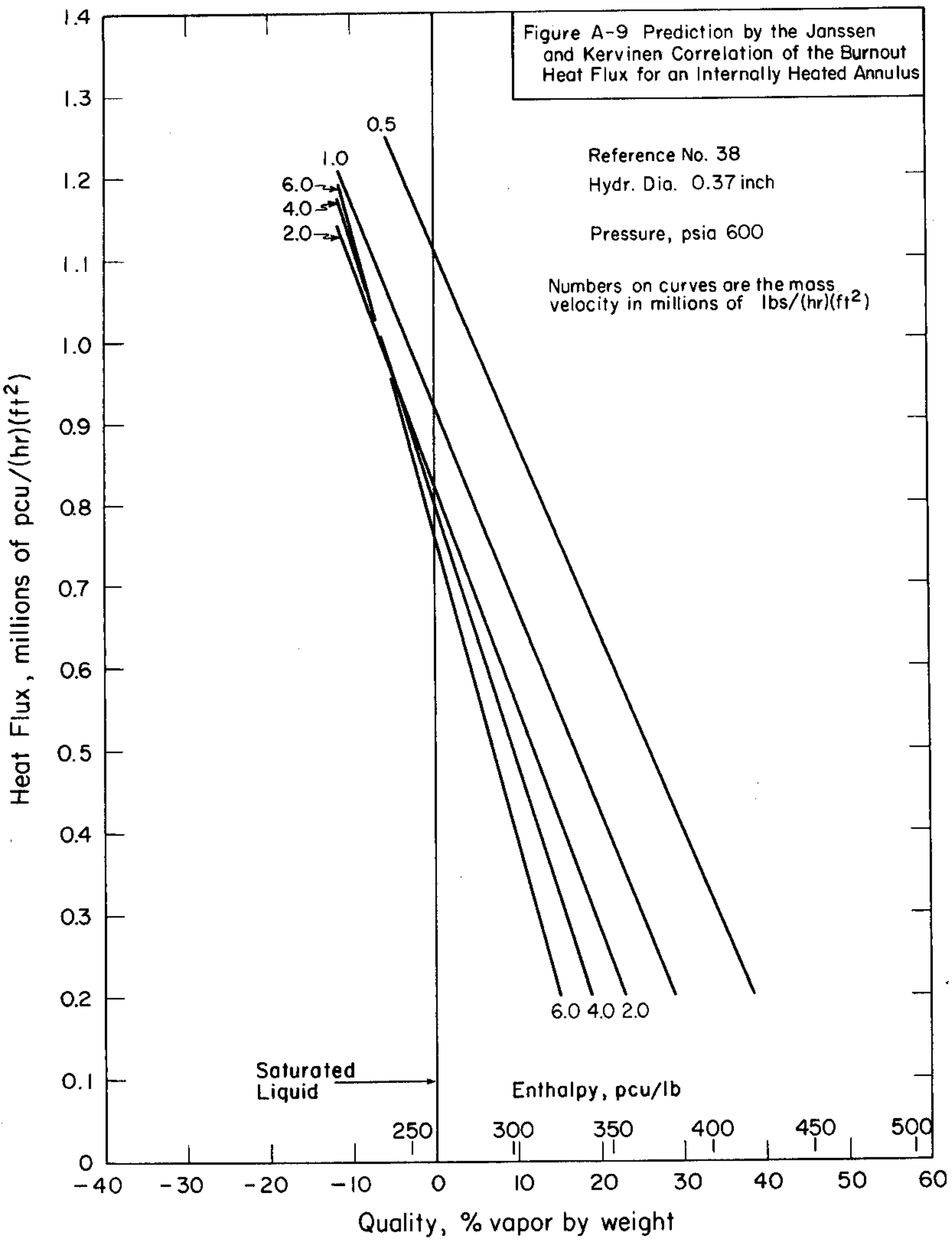
Figure A-9 Prediction by the Janssen and Kervinen Correlation of the Burnout Heat Flux for an Internally Heated Annulus
Reference No. 38
Hydr. Dia. 0.37 inch
Pressure, psia 600
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
0.5
1.0
6.0
4.0
2.0
6.0 4.0 2.0
Saturated Liquid
Enthalpy, pcu/lb
250
300
350
400
450
500
Heat Flux, millions of pcu/(hr)(ft²)
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
-40
-30
-20
-10
0
10
20
30
40
50
60
Quality, % vapor by weight

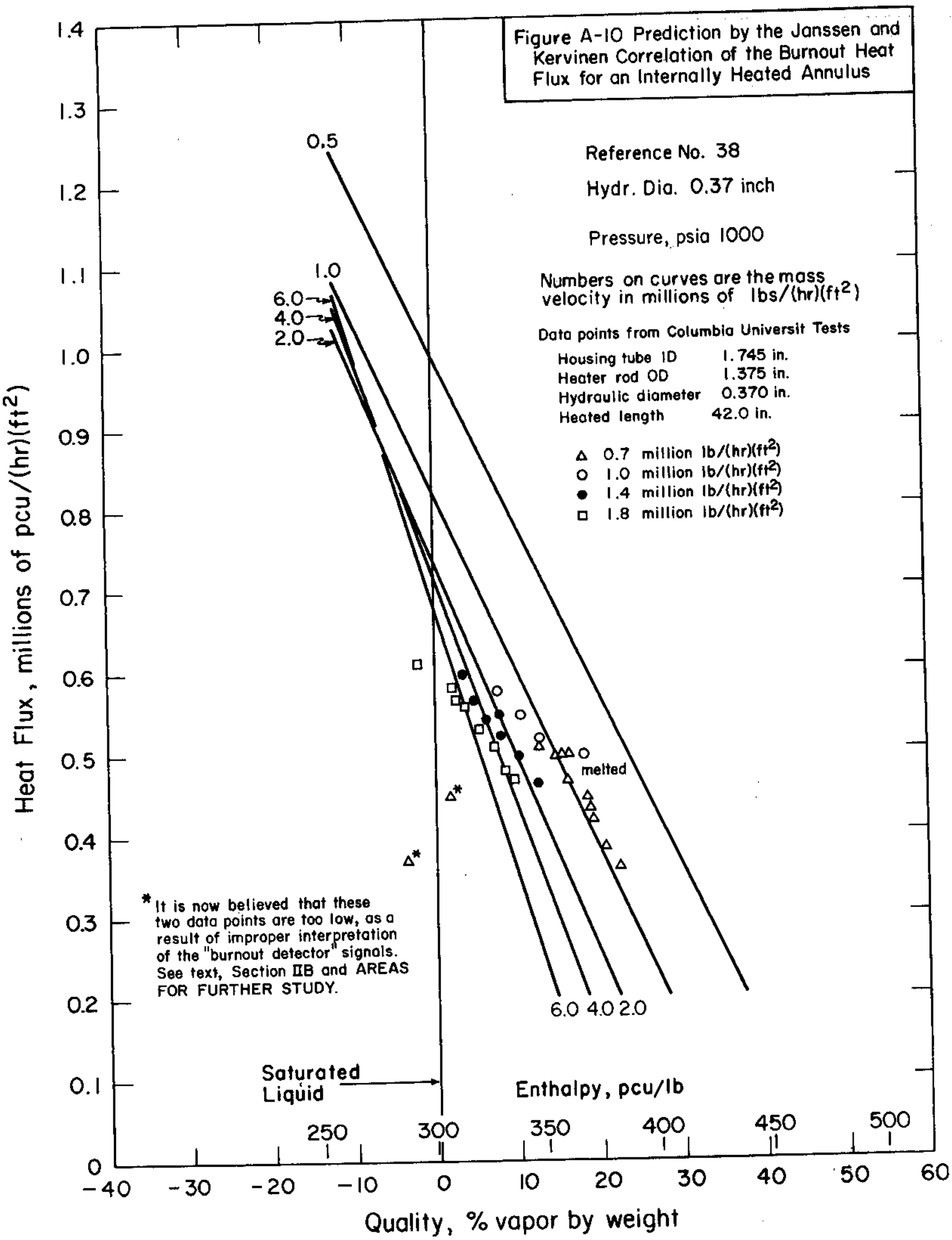

Figure A-10 Prediction by the Janssen and Kervinen Correlation of the Burnout Heat Flux for an Internally Heated Annulus
Reference No. 38
Hydr. Dia. 0.37 inch
Pressure, psia 1000
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Data points from Columbia Universit Tests
Housing tube ID 1.745 in.
Heater rod OD 1.375 in.
Hydraulic diameter 0.370 in.
Heated length 42.0 in.
△ 0.7 million lb/(hr)(ft²)
○ 1.0 million lb/(hr)(ft²)
● 1.4 million lb/(hr)(ft²)
□ 1.8 million lb/(hr)(ft²)
0.5
1.0
6.0
4.0
2.0
melted
* It is now believed that these two data points are too low, as a result of improper interpretation of the "burnout detector" signals. See text, Section IIB and AREAS FOR FURTHER STUDY.
6.0 4.0 2.0
Saturated Liquid
Enthalpy, pcu/lb
250 300 350 400 450 500
Heat Flux, millions of pcu/(hr)(ft²)
1.4 1.3 1.2 1.1 1.0 0.9 0.8 0.7 0.6 0.5 0.4 0.3 0.2 0.1 0
−40 −30 −20 −10 0 10 20 30 40 50 60
Quality, % vapor by weight

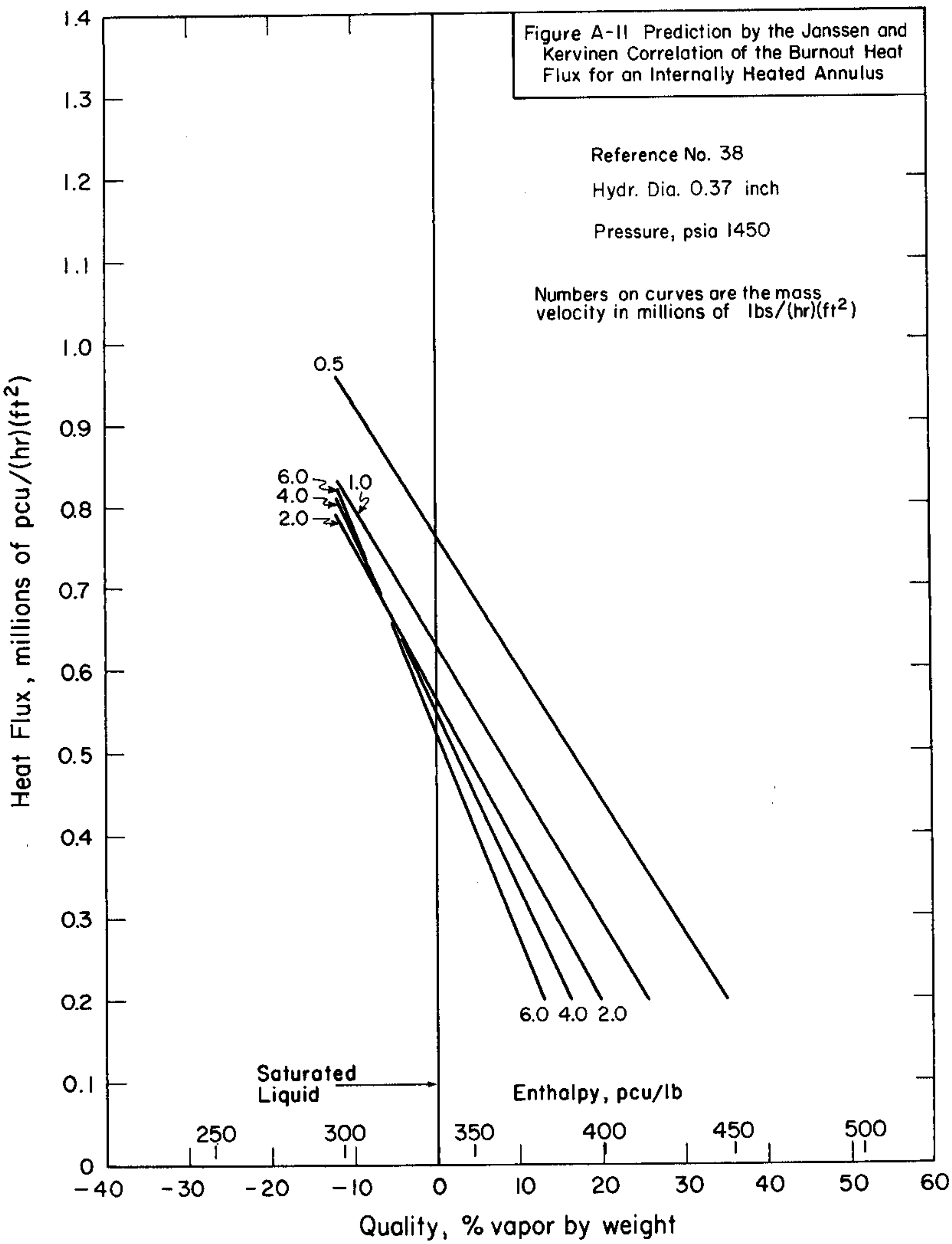

Figure A-11 Prediction by the Janssen and Kervinen Correlation of the Burnout Heat Flux for an Internally Heated Annulus
Reference No. 38
Hydr. Dia. 0.37 inch
Pressure, psia 1450
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
0.5
6.0
4.0
2.0
1.0
6.0 4.0 2.0
Saturated Liquid
Enthalpy, pcu/lb
250
300
350
400
450
500
Heat Flux, millions of pcu/(hr)(ft²)
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
−40 −30 −20 −10 0 10 20 30 40 50 60
Quality, % vapor by weight

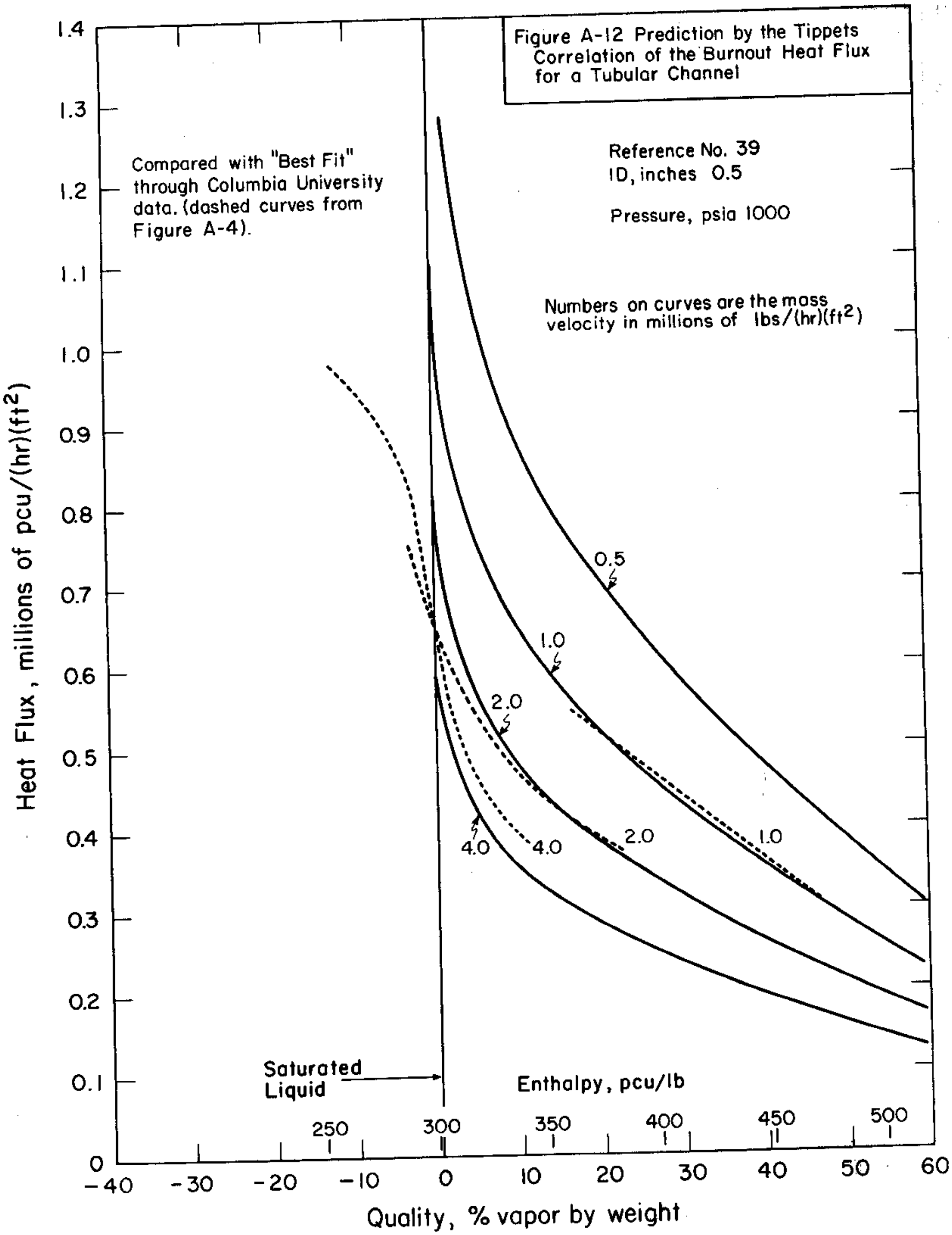

Figure A-12 Prediction by the Tippets Correlation of the Burnout Heat Flux for a Tubular Channel
Compared with "Best Fit" through Columbia University data. (dashed curves from Figure A-4).
Reference No. 39
ID, inches 0.5
Pressure, psia 1000
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft2)
Heat Flux, millions of pcu/(hr)(ft2)
Quality, % vapor by weight
Enthalpy, pcu/lb
Saturated Liquid
0.5
1.0
2.0
4.0
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
1.1
1.2
1.3
1.4
-40
-30
-20
-10
10
20
30
40
50
60
250
300
350
400
450
500

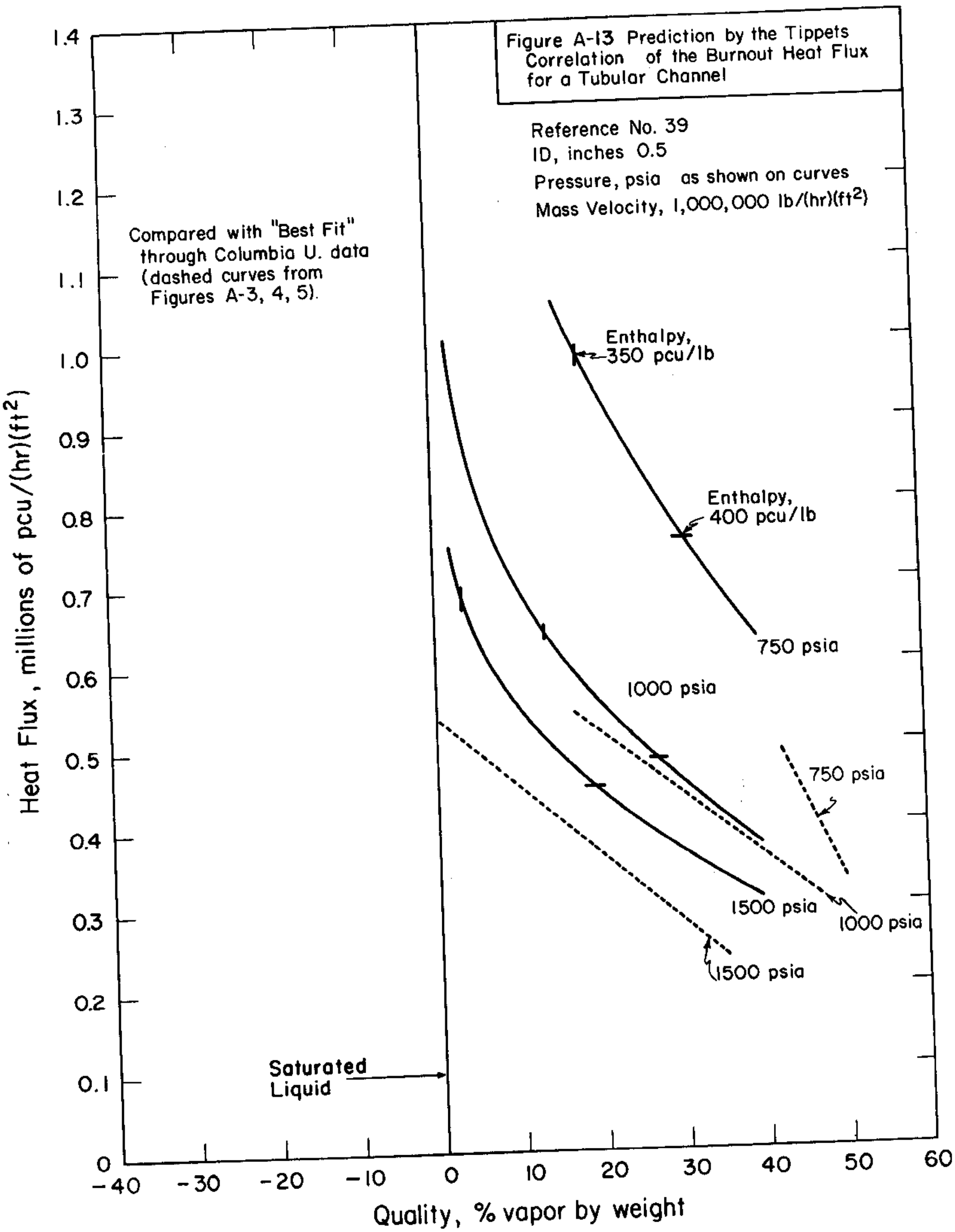

Figure A-13 Prediction by the Tippets Correlation of the Burnout Heat Flux for a Tubular Channel
Reference No. 39
ID, inches 0.5
Pressure, psia as shown on curves
Mass Velocity, 1,000,000 lb/(hr)(ft²)
Compared with "Best Fit" through Columbia U. data (dashed curves from Figures A-3, 4, 5).
Enthalpy, 350 pcu/lb
Enthalpy, 400 pcu/lb
750 psia
1000 psia
1500 psia
750 psia
1000 psia
1500 psia
Saturated Liquid
Heat Flux, millions of pcu/(hr)(ft²)
Quality, % vapor by weight
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
-40
-30
-20
-10
0
10
20
30
40
50
60

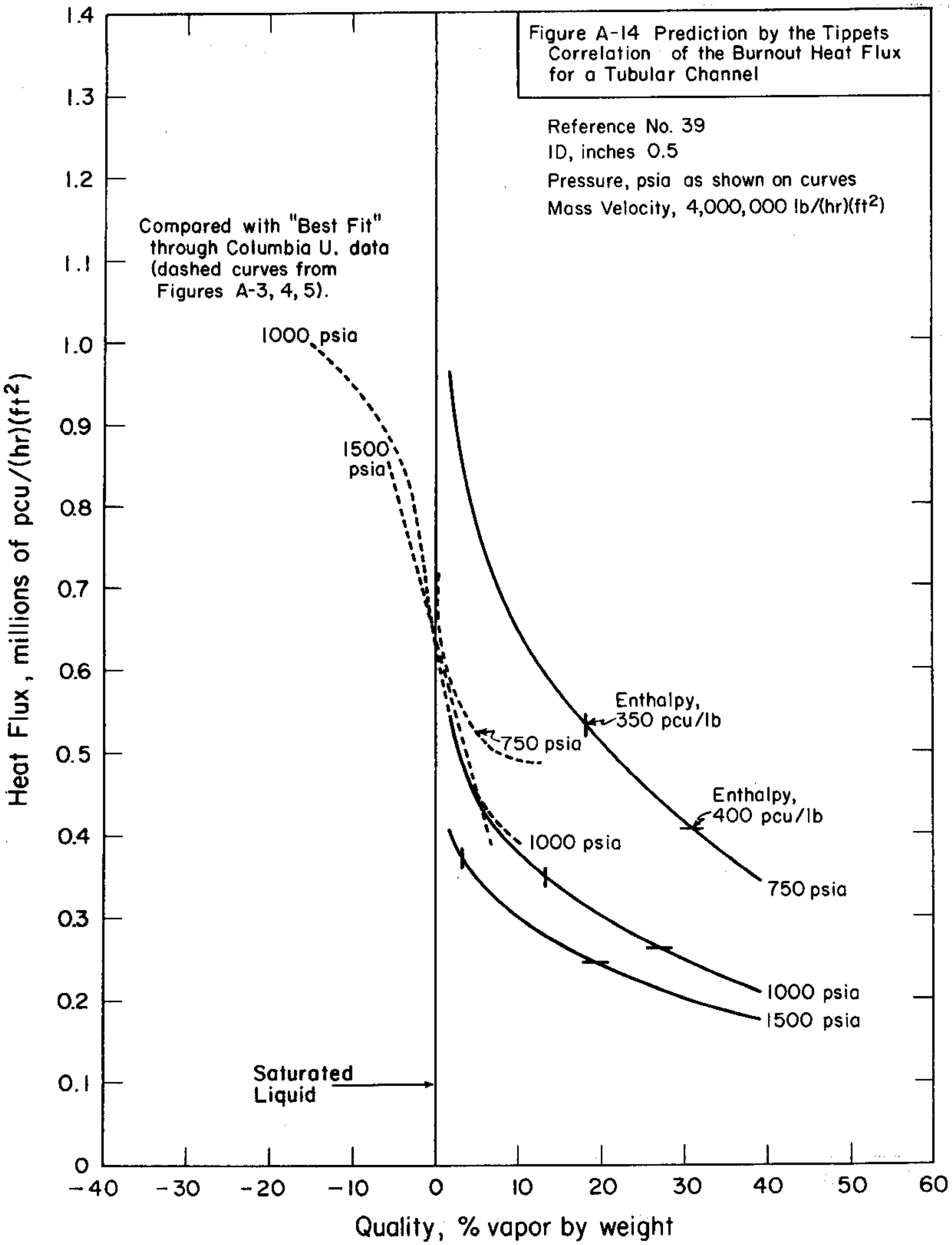
Figure A-14 Prediction by the Tippets Correlation of the Burnout Heat Flux for a Tubular Channel
Reference No. 39
ID, inches 0.5
Pressure, psia as shown on curves
Mass Velocity, 4,000,000 lb/(hr)(ft^2)
Compared with "Best Fit" through Columbia U. data (dashed curves from Figures A-3, 4, 5).
1000 psia
1500 psia
750 psia
1000 psia
Enthalpy, 350 pcu/lb
Enthalpy, 400 pcu/lb
750 psia
1000 psia
1500 psia
Saturated Liquid
Heat Flux, millions of pcu/(hr)(ft^2)
Quality, % vapor by weight
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
-40 -30 -20 -10 0 10 20 30 40 50 60

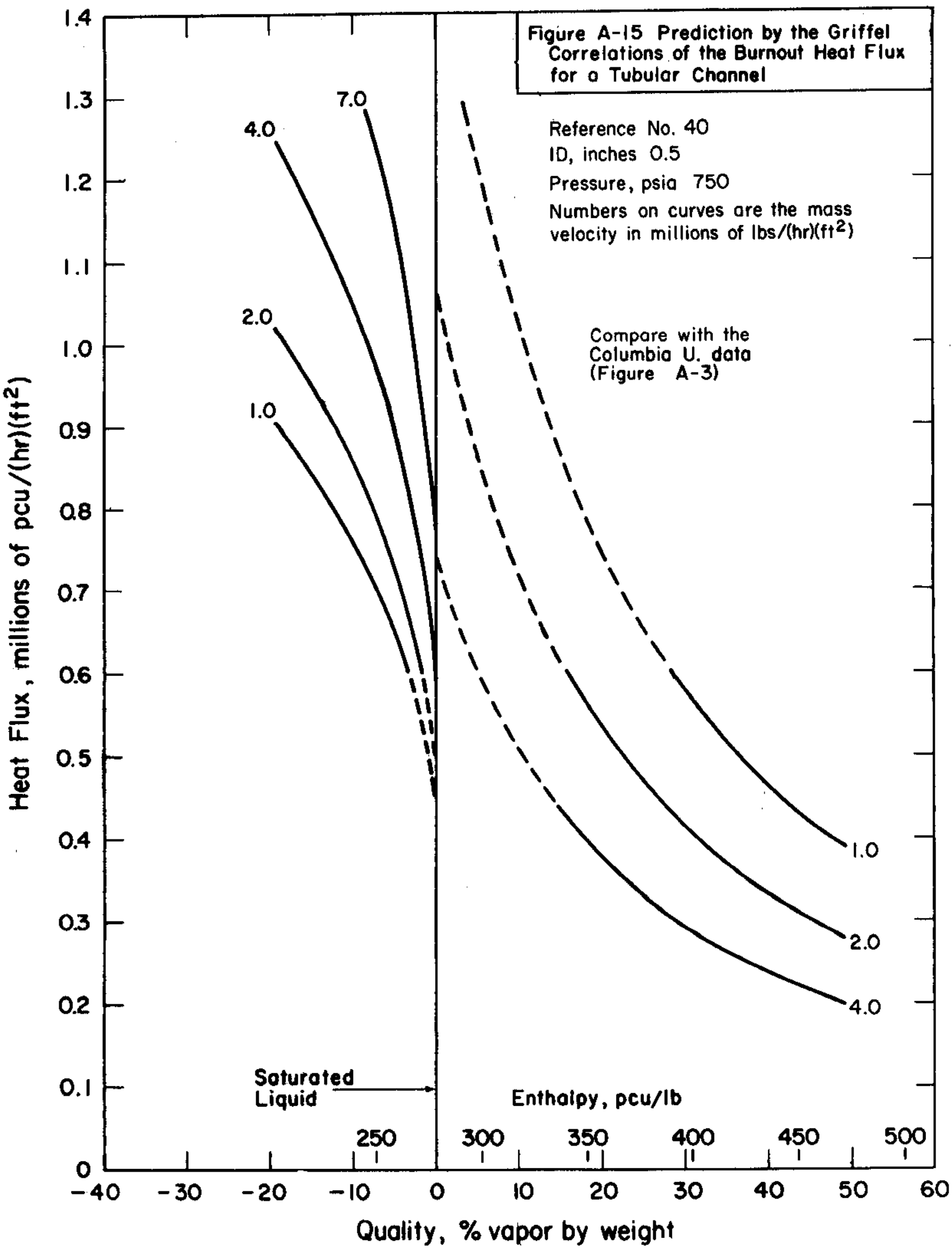
Figure A-15 Prediction by the Griffel Correlations of the Burnout Heat Flux for a Tubular Channel
Reference No. 40
ID, inches 0.5
Pressure, psia 750
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Compare with the Columbia U. data (Figure A-3)
7.0
4.0
2.0
1.0
Saturated Liquid
Enthalpy, pcu/lb
250
300
350
400
450
500
Heat Flux, millions of pcu/(hr)(ft²)
Quality, % vapor by weight
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
−40
−30
−20
−10
10
20
30
40
50
60

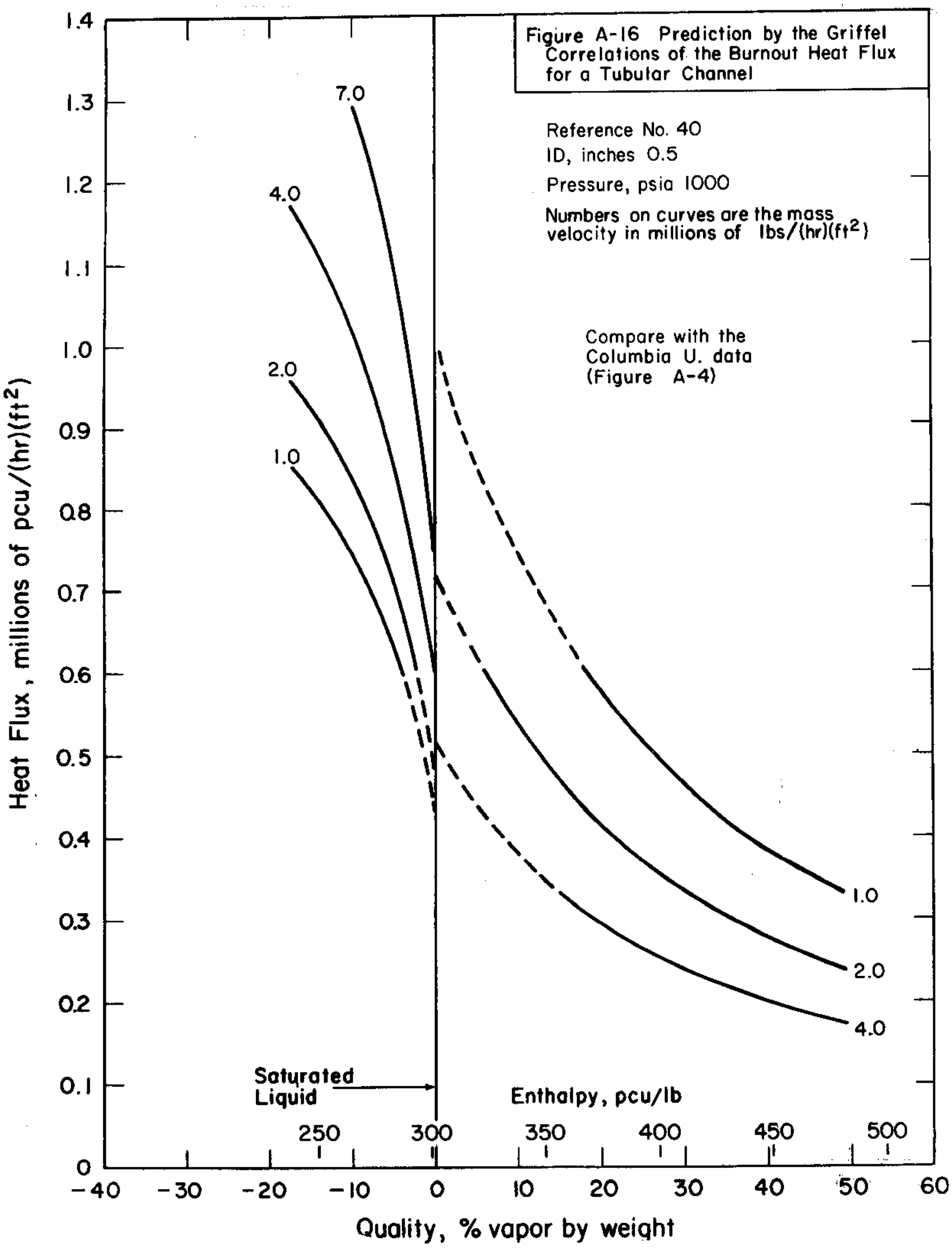
Figure A-16 Prediction by the Griffel Correlations of the Burnout Heat Flux for a Tubular Channel
Reference No. 40
ID, inches 0.5
Pressure, psia 1000
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft^2)
Compare with the Columbia U. data (Figure A-4)
7.0
4.0
2.0
1.0
Saturated Liquid
Enthalpy, pcu/lb
250
300
350
400
450
500
Heat Flux, millions of pcu/(hr)(ft^2)
Quality, % vapor by weight
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
-40
-30
-20
-10
0
10
20
30
40
50
60

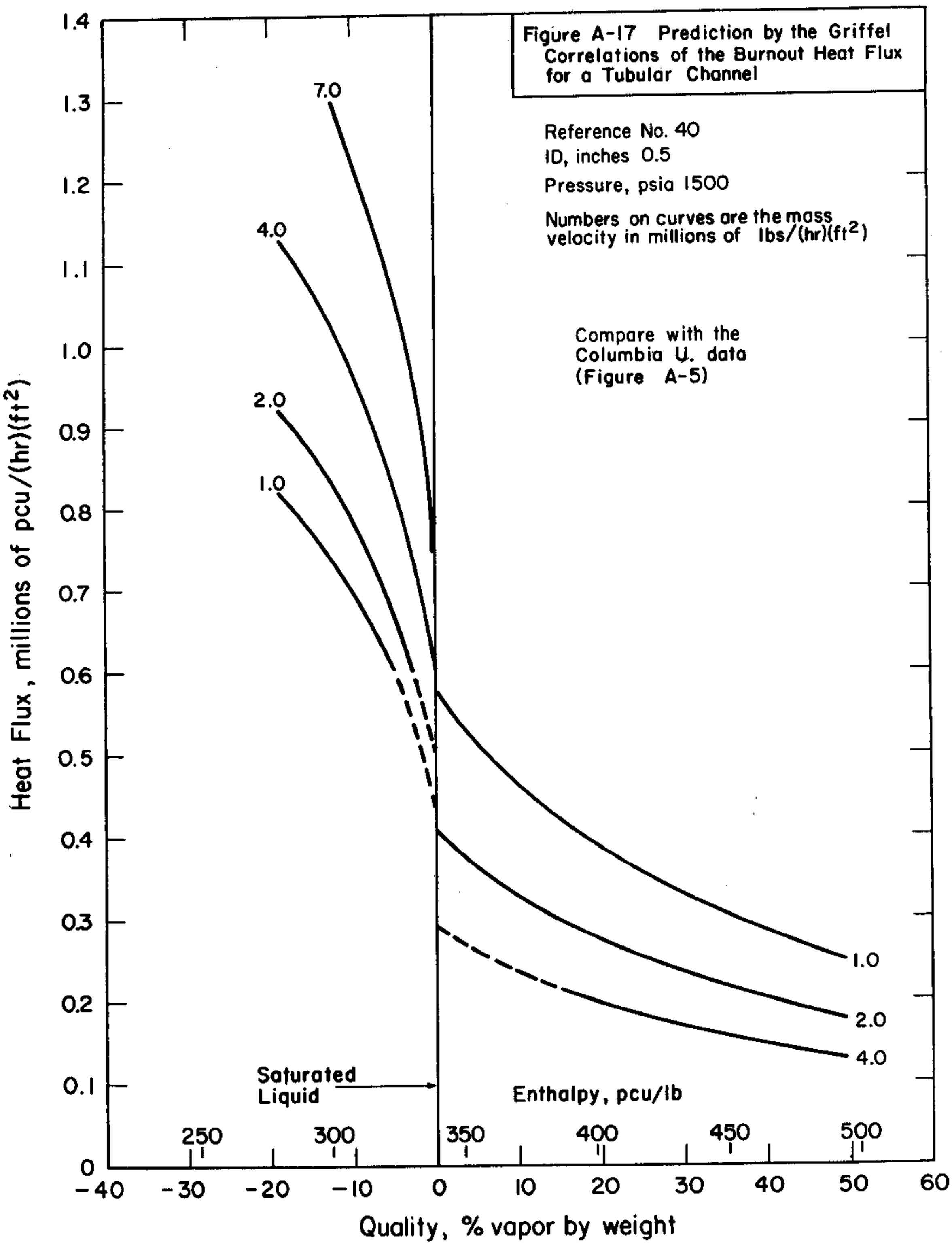
Figure A-17 Prediction by the Griffel Correlations of the Burnout Heat Flux for a Tubular Channel
Reference No. 40
ID, inches 0.5
Pressure, psia 1500
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Compare with the Columbia U. data (Figure A-5)
7.0
4.0
2.0
1.0
1.0
2.0
4.0
Saturated Liquid
Enthalpy, pcu/lb
250
300
350
400
450
500
Heat Flux, millions of pcu/(hr)(ft²)
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
1.1
1.2
1.3
1.4
-40
-30
-20
-10
0
10
20
30
40
50
60
Quality, % vapor by weight

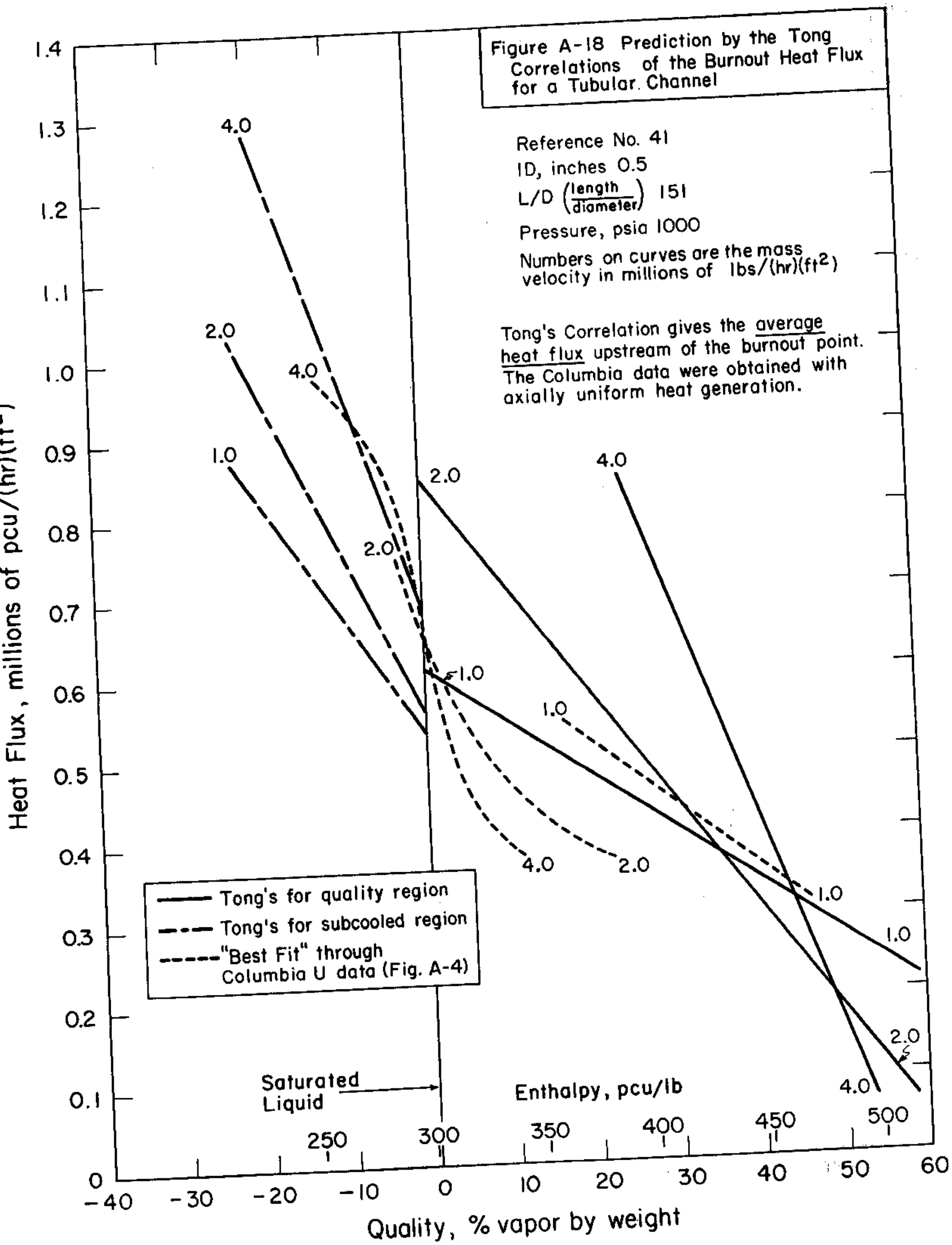

Figure A-18 Prediction by the Tong Correlations of the Burnout Heat Flux for a Tubular Channel
Reference No. 41
ID, inches 0.5
L/D (length/diameter) 151
Pressure, psia 1000
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft^2)
Tong's Correlation gives the average heat flux upstream of the burnout point. The Columbia data were obtained with axially uniform heat generation.
Tong's for quality region
Tong's for subcooled region
"Best Fit" through Columbia U data (Fig. A-4)
Saturated Liquid
Enthalpy, pcu/lb
Heat Flux, millions of pcu/(hr)(ft^2)
Quality, % vapor by weight
4.0
2.0
1.0
250
300
350
400
450
500
-40
-30
-20
-10
0
10
20
30
40
50
60
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
1.1
1.2
1.3
1.4

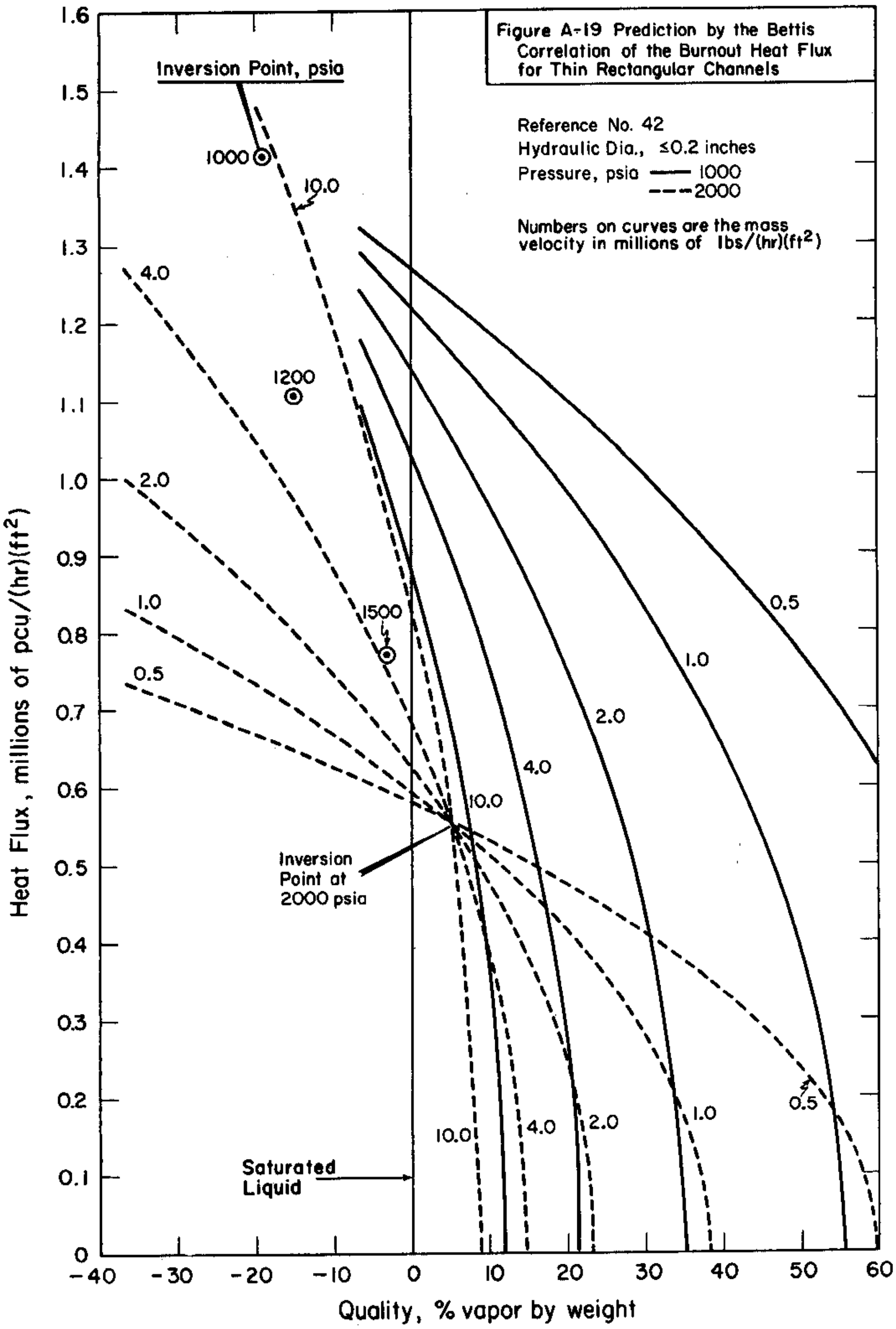

Figure A-19 Prediction by the Bettis Correlation of the Burnout Heat Flux for Thin Rectangular Channels
Reference No. 42
Hydraulic Dia., ≤0.2 inches
Pressure, psia —— 1000
- - - 2000
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Inversion Point, psia
1000
1200
1500
10.0
4.0
2.0
1.0
0.5
Inversion Point at 2000 psia
Saturated Liquid
Heat Flux, millions of pcu/(hr)(ft²)
Quality, % vapor by weight
1.6
1.5
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
-40
-30
-20
-10
0
10
20
30
40
50
60

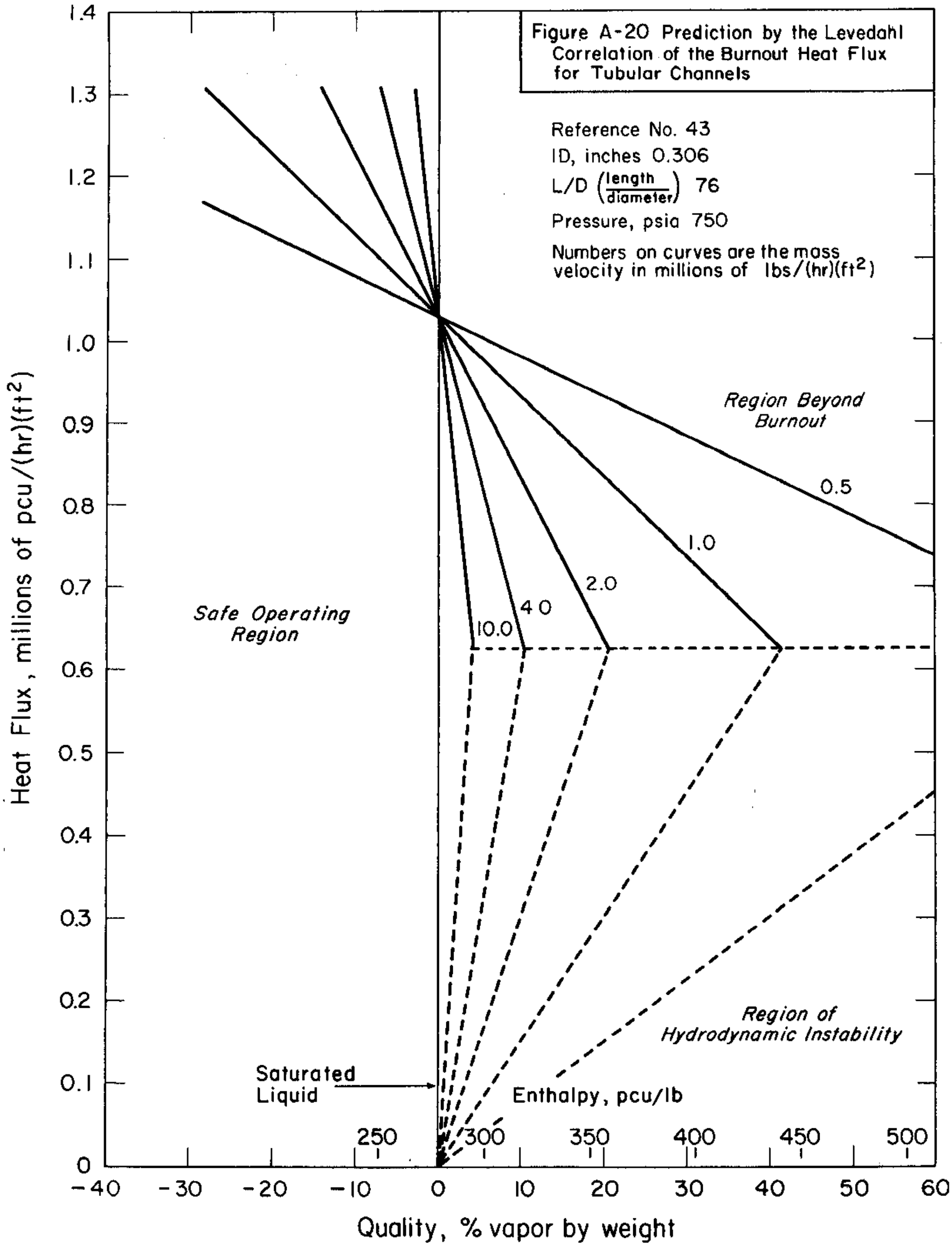
Figure A-20 Prediction by the Levedahl Correlation of the Burnout Heat Flux for Tubular Channels
Reference No. 43
ID, inches 0.306
L/D (length/diameter) 76
Pressure, psia 750
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Region Beyond Burnout
0.5
1.0
2.0
4.0
10.0
Safe Operating Region
Region of Hydrodynamic Instability
Saturated Liquid
Enthalpy, pcu/lb
250
300
350
400
450
500
Heat Flux, millions of pcu/(hr)(ft²)
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
-40
-30
-20
-10
0
10
20
30
40
50
60
Quality, % vapor by weight

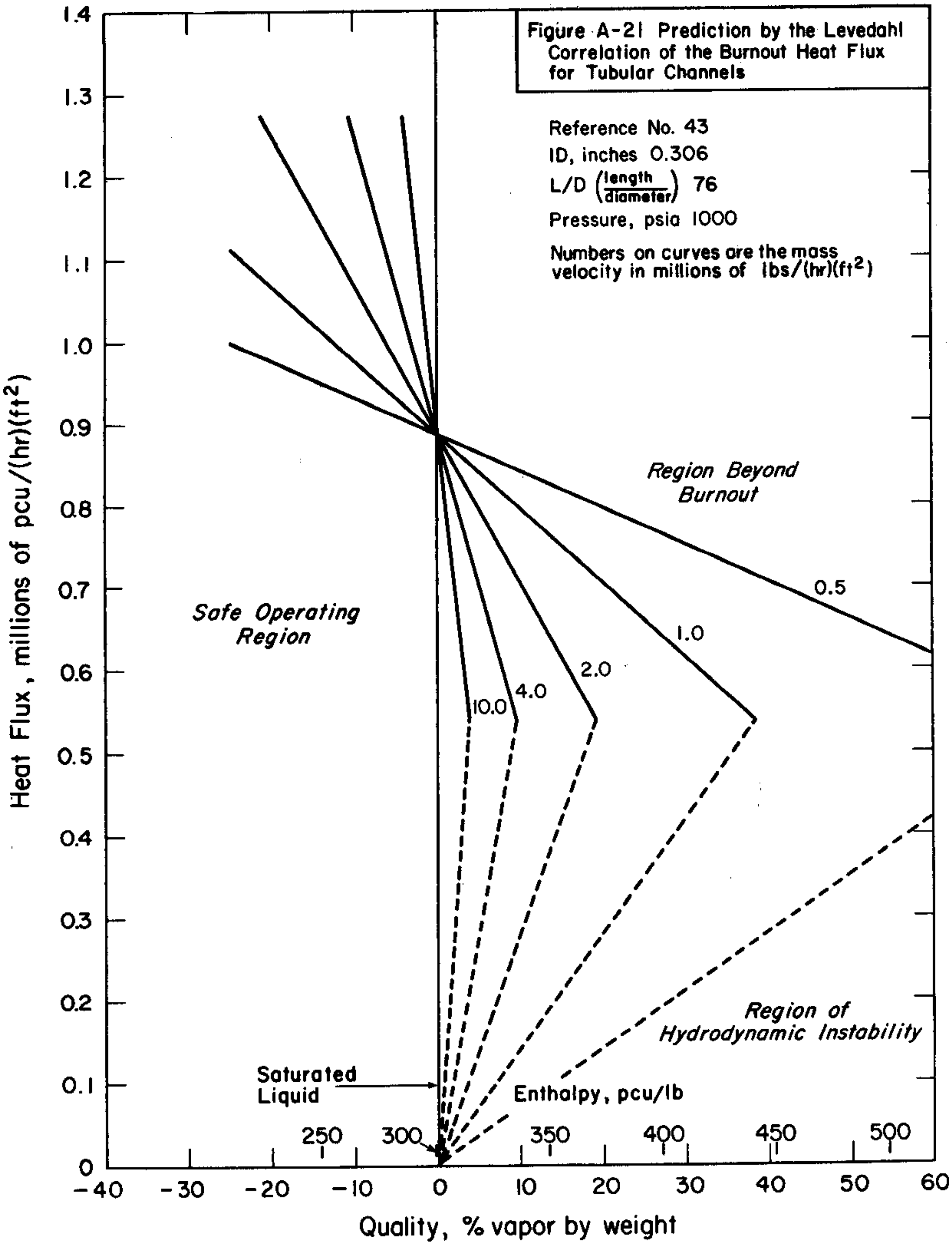

Figure A-21 Prediction by the Levedahl Correlation of the Burnout Heat Flux for Tubular Channels

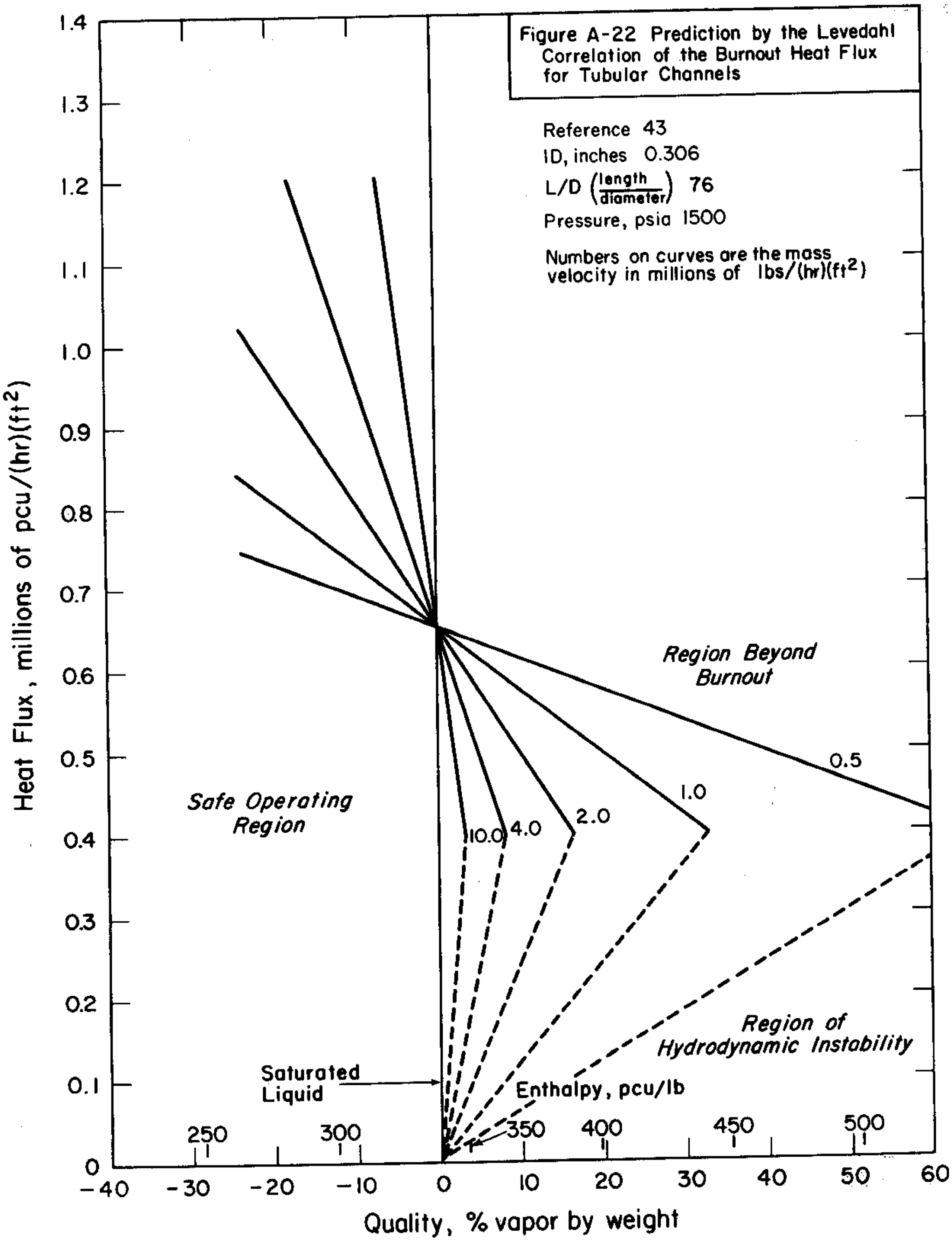
Figure A-22 Prediction by the Levedahl Correlation of the Burnout Heat Flux for Tubular Channels
Reference 43
ID, inches 0.306
L/D (length/diameter) 76
Pressure, psia 1500
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Region Beyond Burnout
Safe Operating Region
Region of Hydrodynamic Instability
Saturated Liquid
Enthalpy, pcu/lb
0.5
1.0
2.0
4.0
10.0
250
300
350
400
450
500
Heat Flux, millions of pcu/(hr)(ft²)
Quality, % vapor by weight
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
-40
-30
-20
-10
0
10
20
30
40
50
60

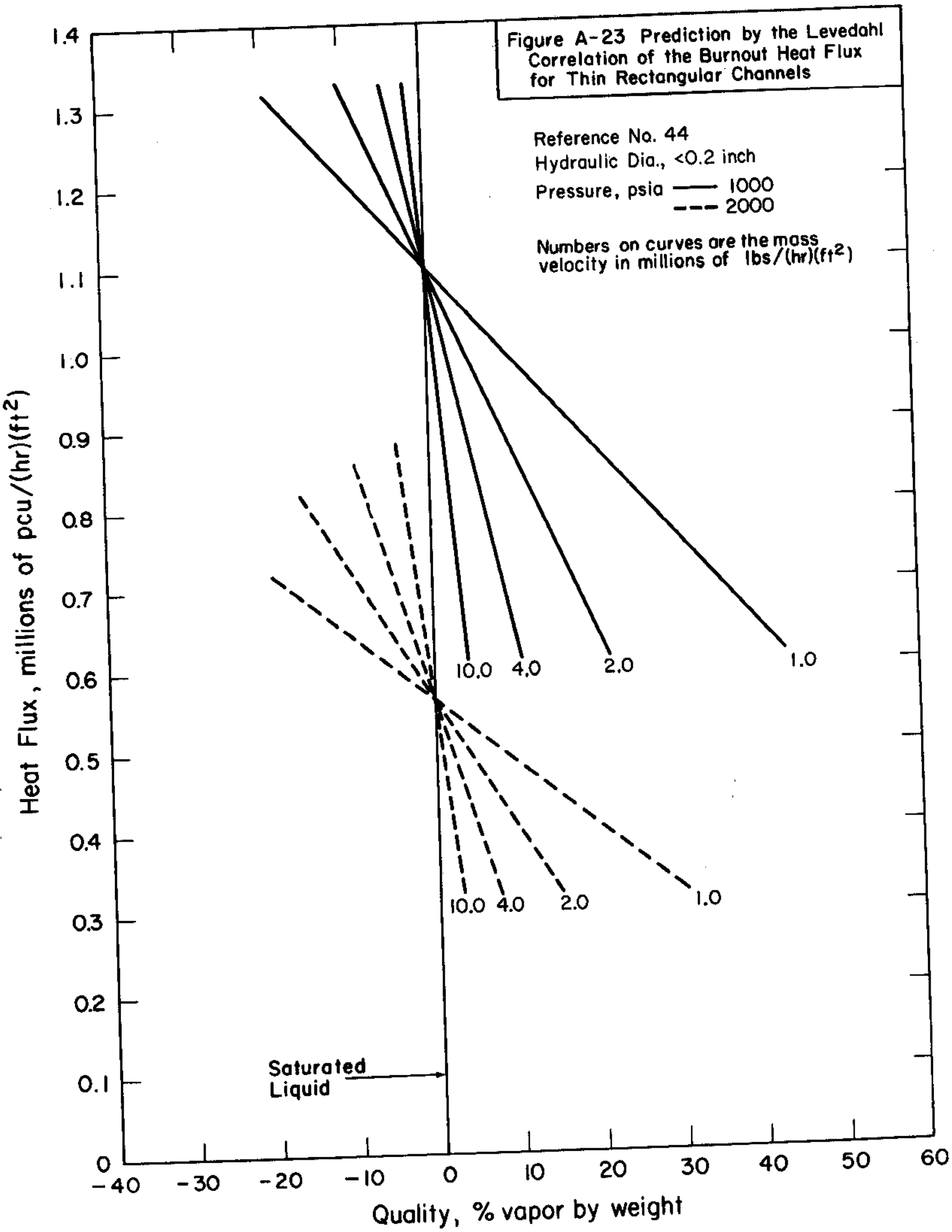

Figure A-23 Prediction by the Levedahl Correlation of the Burnout Heat Flux for Thin Rectangular Channels
Reference No. 44
Hydraulic Dia., <0.2 inch
Pressure, psia —— 1000
- - - 2000
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Heat Flux, millions of pcu/(hr)(ft²)
Quality, % vapor by weight
Saturated Liquid
10.0
4.0
2.0
1.0
10.0
4.0
2.0
1.0
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
1.1
1.2
1.3
1.4
-40
-30
-20
-10
0
10
20
30
40
50
60

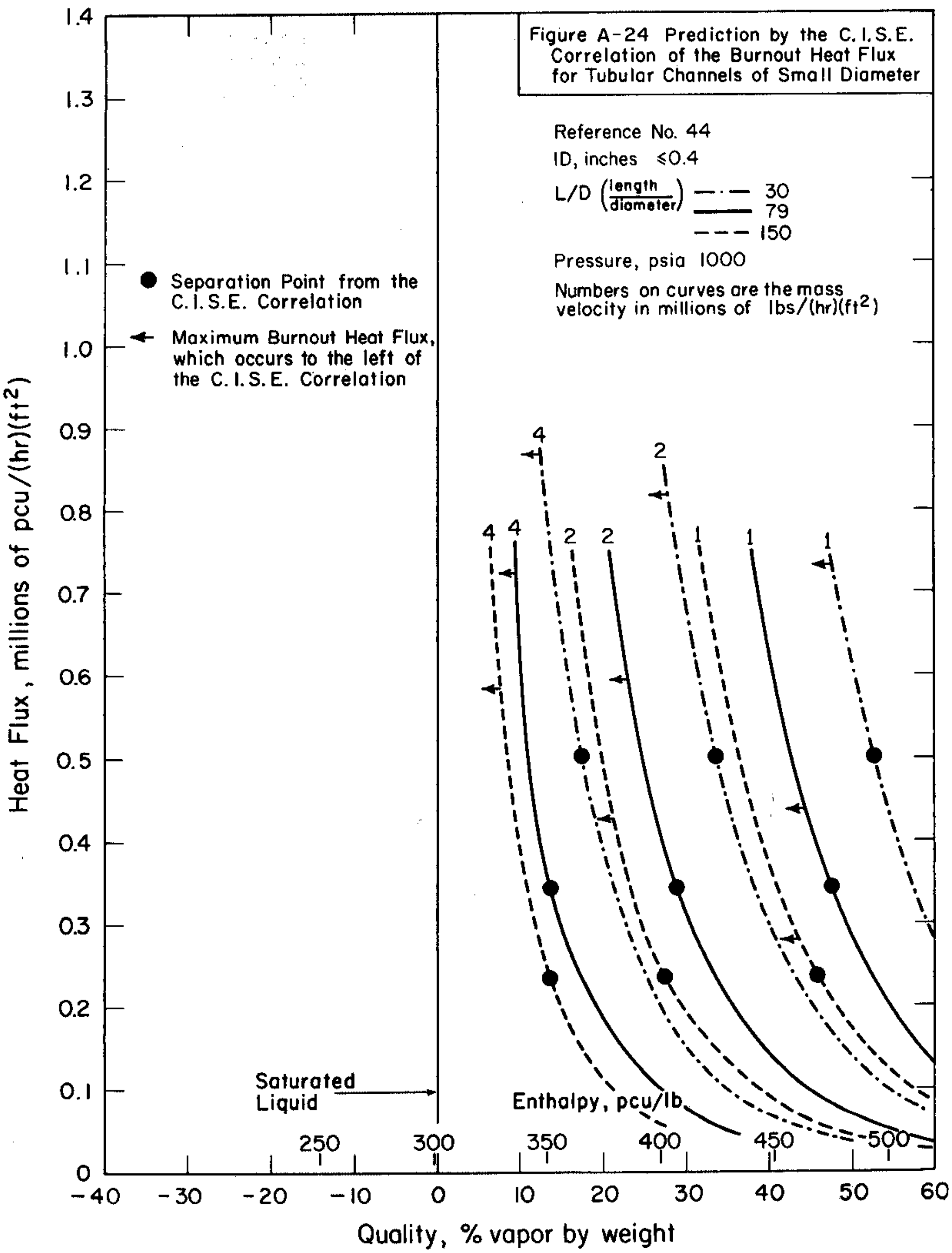

Figure A-24 Prediction by the C.I.S.E. Correlation of the Burnout Heat Flux for Tubular Channels of Small Diameter
Reference No. 44
ID, inches ≤0.4
L/D (length/diameter) 30 79 150
Pressure, psia 1000
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Separation Point from the C.I.S.E. Correlation
Maximum Burnout Heat Flux, which occurs to the left of the C.I.S.E. Correlation
Heat Flux, millions of pcu/(hr)(ft²)
Quality, % vapor by weight
Saturated Liquid
Enthalpy, pcu/lb
250
300
350
400
450
500
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
-40
-30
-20
-10
0
10
20
30
40
50
60
4
2
1

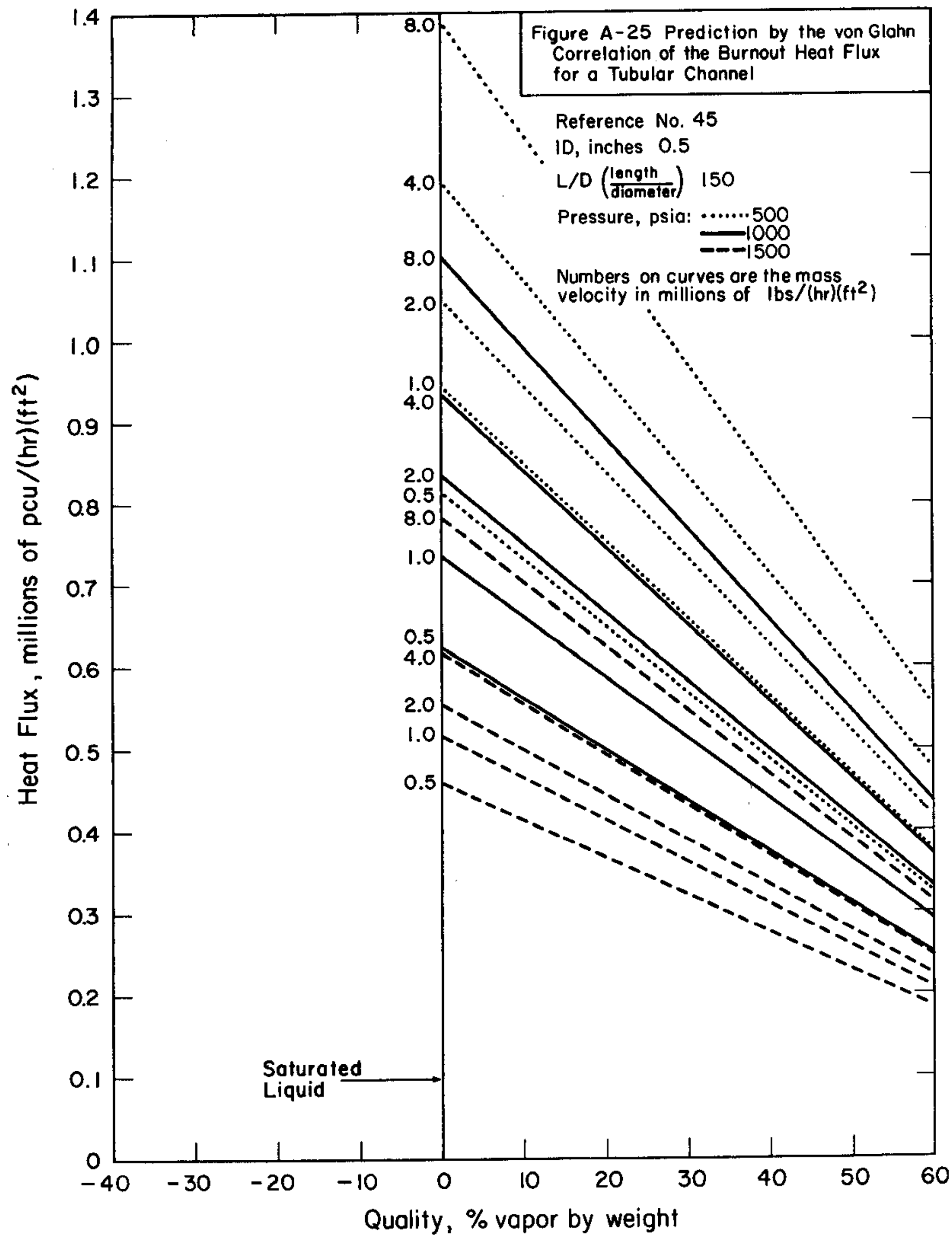
Figure A-25 Prediction by the von Glahn Correlation of the Burnout Heat Flux for a Tubular Channel
Reference No. 45
ID, inches 0.5
L/D (length/diameter) 150
Pressure, psia: 500
1000
1500
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft2)
Heat Flux, millions of pcu/(hr)(ft2)
Quality, % vapor by weight
Saturated Liquid
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
−40
−30
−20
−10
0
10
20
30
40
50
60
8.0
4.0
8.0
2.0
1.0
4.0
2.0
0.5
8.0
1.0
0.5
4.0
2.0
1.0
0.5

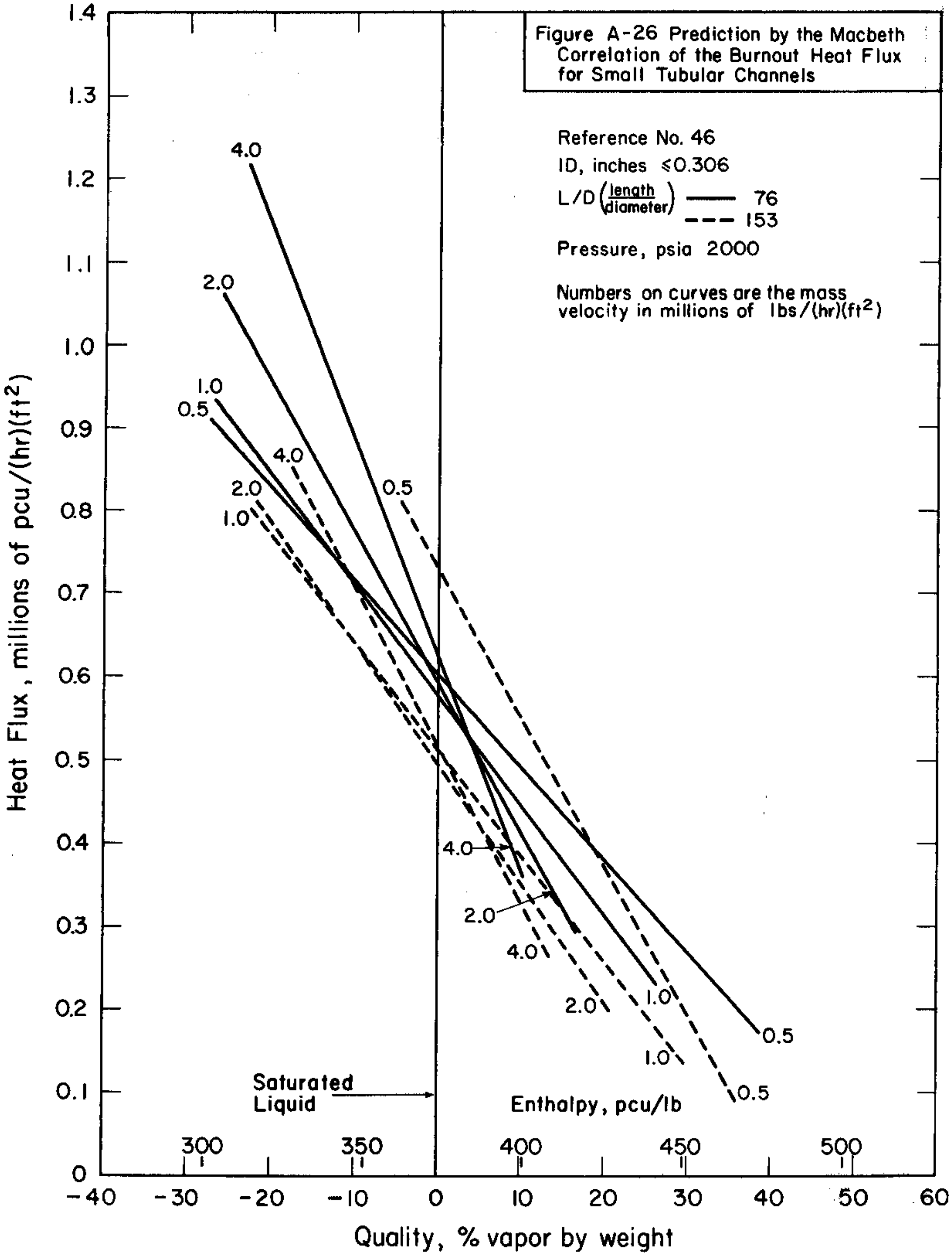
Figure A-26 Prediction by the Macbeth Correlation of the Burnout Heat Flux for Small Tubular Channels
Reference No. 46
ID, inches ≤0.306
L/D (length/diameter) —— 76
– – – 153
Pressure, psia 2000
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Heat Flux, millions of pcu/(hr)(ft²)
Quality, % vapor by weight
Saturated Liquid
Enthalpy, pcu/lb
4.0
2.0
1.0
0.5
4.0
2.0
1.0
0.5
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
1.1
1.2
1.3
1.4
−40
−30
−20
−10
0
10
20
30
40
50
60
300
350
400
450
500

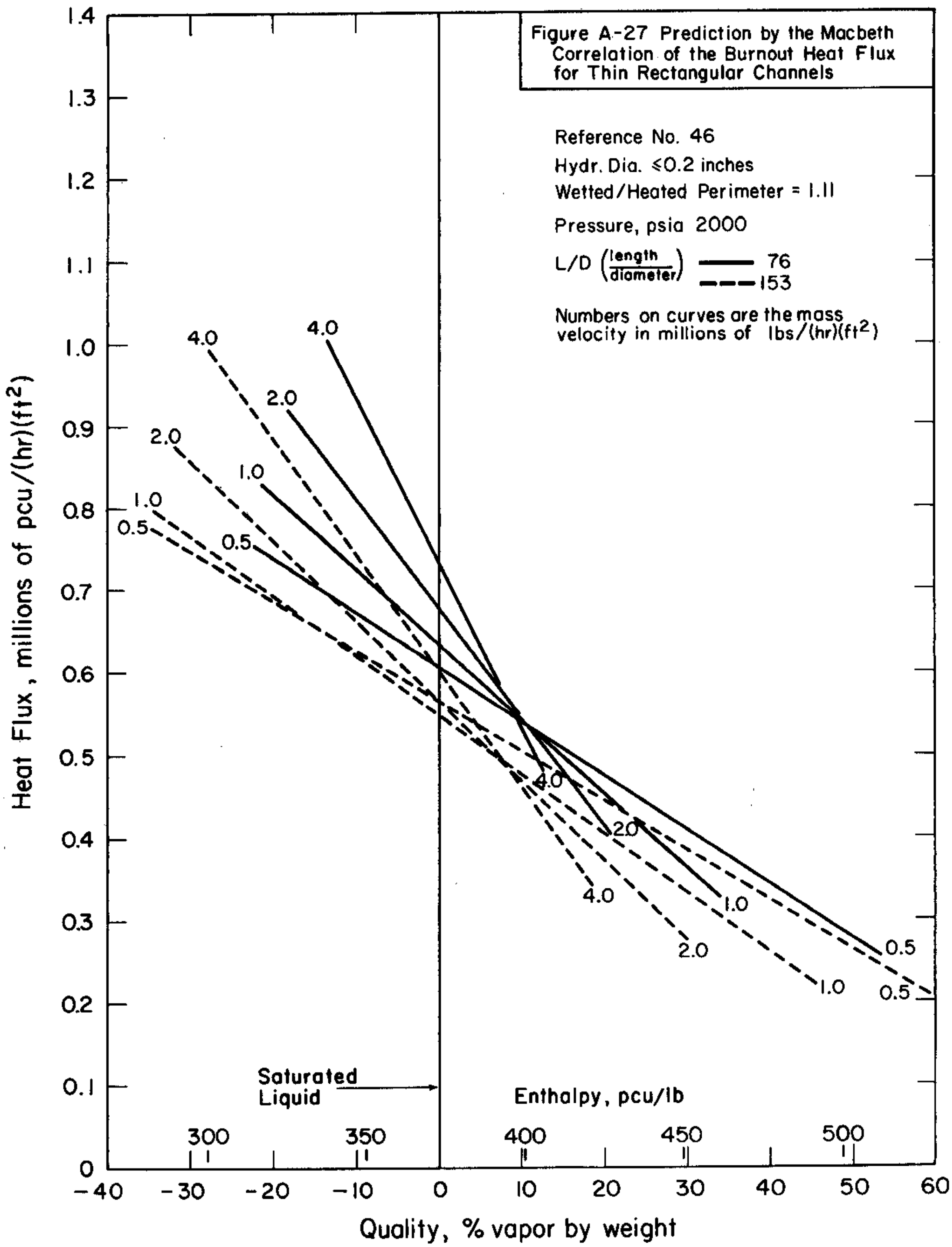
Figure A-27 Prediction by the Macbeth Correlation of the Burnout Heat Flux for Thin Rectangular Channels
Reference No. 46
Hydr. Dia. ≤0.2 inches
Wetted/Heated Perimeter = 1.11
Pressure, psia 2000
L/D (length/diameter) 76
153
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Heat Flux, millions of pcu/(hr)(ft²)
Quality, % vapor by weight
Enthalpy, pcu/lb
Saturated Liquid
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
-40
-30
-20
-10
0
10
20
30
40
50
60
300
350
400
450
500
4.0
2.0
1.0
0.5

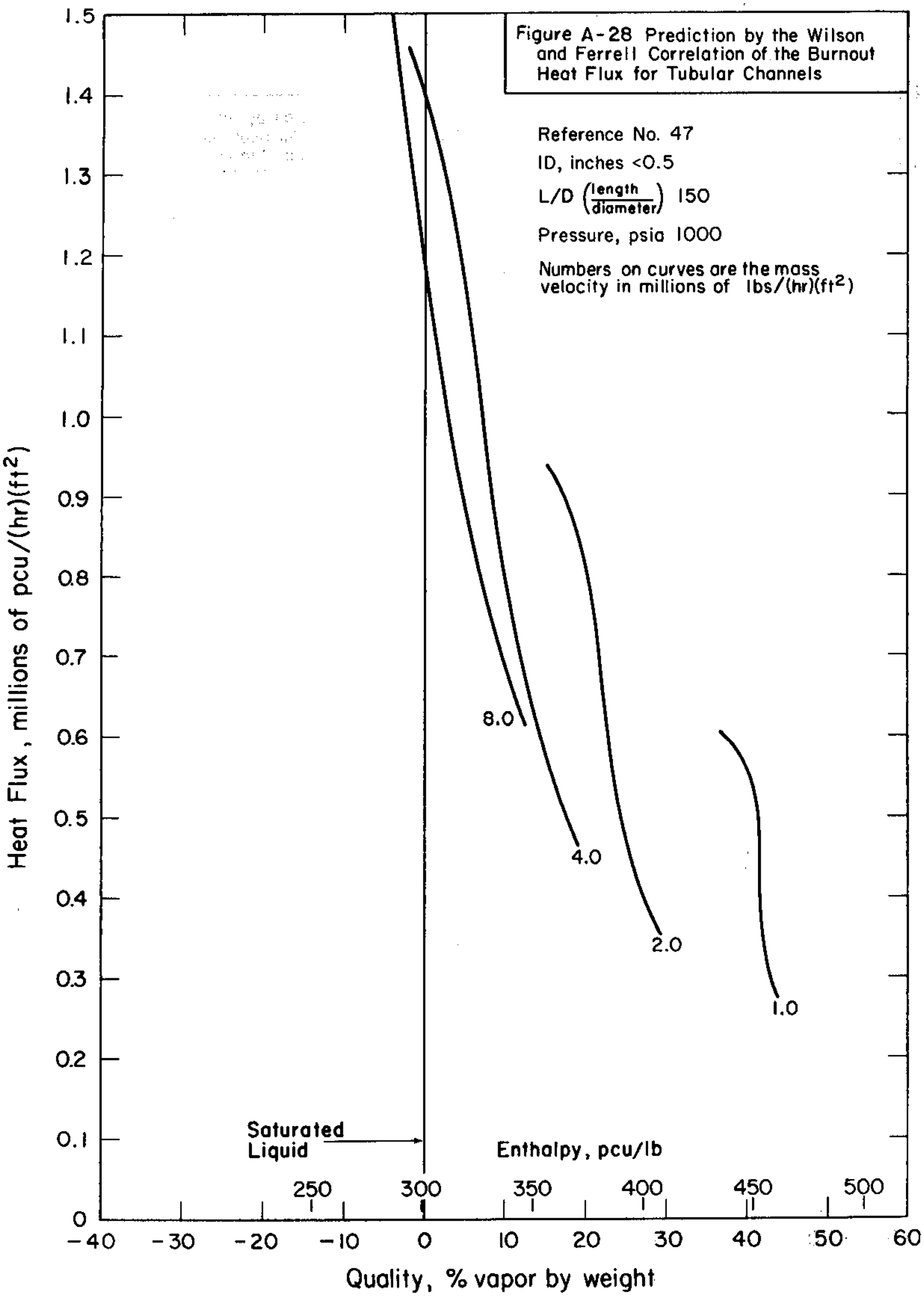

Figure A-28 Prediction by the Wilson and Ferrell Correlation of the Burnout Heat Flux for Tubular Channels
Reference No. 47
ID, inches <0.5
L/D (length/diameter) 150
Pressure, psia 1000
Numbers on curves are the mass velocity in millions of lbs/(hr)(ft²)
Heat Flux, millions of pcu/(hr)(ft²)
1.5
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
8.0
4.0
2.0
1.0
Saturated Liquid
Enthalpy, pcu/lb
250
300
350
400
450
500
-40
-30
-20
-10
0
10
20
30
40
50
60
Quality, % vapor by weight

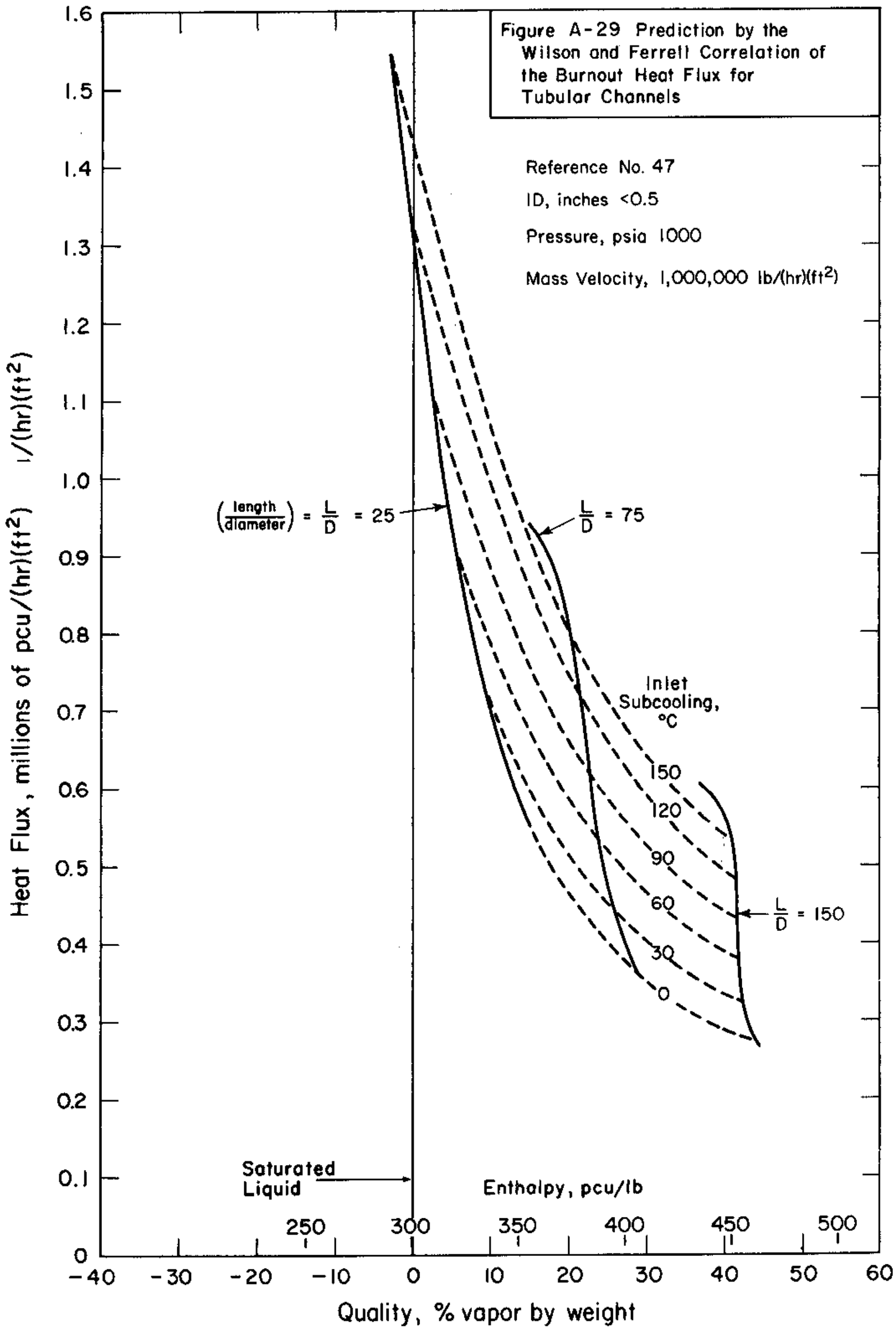
Figure A-29 Prediction by the Wilson and Ferrell Correlation of the Burnout Heat Flux for Tubular Channels
Reference No. 47
ID, inches <0.5
Pressure, psia 1000
Mass Velocity, 1,000,000 lb/(hr)(ft^2)
Heat Flux, millions of pcu/(hr)(ft^2)
1/(hr)(ft^2)
(length/diameter) = L/D = 25
L/D = 75
L/D = 150
Inlet Subcooling, °C
150
120
90
60
30
0
Saturated Liquid
Enthalpy, pcu/lb
250
300
350
400
450
500
Quality, % vapor by weight
1.6
1.5
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
-40
-30
-20
-10
0
10
20
30
40
50
60

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