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Prepared for
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Washington, D.C. 20555

NRC FIN No. 6102

INTERIM REPORT

November 1, 1978

Mr. R. E. Tiller, Director
Reactor Operations and Programs Division
Idaho Operations Office - DOE
Idaho Falls, ID 83401

TRANSMITTAL OF MOD7 CHF CORRELATION REPORT - PN-181-78

Ref: (a) Status Summary Report, WRSR, Office of Nuclear Regulatory Research, Sep 22, 1978
(b) 189 Number A6102, LOCA Correlation Verification and Heat Transfer Data Base

Dear Mr. Tiller:

The attached report documents the development of a new correlation for the prediction of critical heat flux in rod bundles. The correlation, termed the "MOD7 CHF Correlation", was developed using a regression analysis of the applicable steady state critical heat flux data. Correlating parameters were limited to those local variables readily available in RELAP or easily obtainable from state relationships. The correlation correlates the data base of over 5000 points with a standard deviation of 29%.

This report satisfies Node 4, Line 7, Page 1-25 of Reference (a) and is an account of work funded by Reference (b).

Very truly yours,

Original Signed By

Paul North, Manager
Code Development and
Analysis Program

KGC:ma

Attachment:
As stated

NRC Research and Technical Assistance Report

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Attachment
PN-181-78
November 1, 1978

DEVELOPMENT OF THE MOD7 CHF CORRELATION

by

K. G. Condie

S. J. Bengston

NRC Research and Technical Assistance Report

SUMMARY

A new correlation for the prediction of critical heat flux in rod bundles was developed and incorporated into RELAP4/MOD7. The correlation, termed the "MOD7 CHF Correlation", was developed using a regression analysis of the applicable steady-state critical heat flux data. Correlating parameters were limited to those local variables readily available in RELAP or easily obtainable from state relationships.

I. INTRODUCTION

The accurate prediction of the temperature-time history for a fuel rod in a large water reactor experiencing a blowdown transient is dependent upon the ability to predict the interface between the efficient nucleate boiling heat transfer regime and the less efficient film boiling regime. This interface is referred to by many names such as critical heat flux, burnout, dryout, departure from nucleate boiling (DNB) and boiling crisis point. While each of the above names carries a more specific connotation to various individuals, the general phenomena will be referred to as Critical Heat Flux (CHF) in this report.

A typical RELAP model of a large power reactor (LPR) core for a blowdown transient consists of 12 hydraulic volumes, six representing the average assembly and six representing the higher powered assembly. The heat flux from the fuel rods is assumed constant over the length of the volume. The variables available for input into any correlation are limited to the average hydraulic conditions in that volume and the conditions at the inlet and outlet to the volume. Any other variables must be obtained from submodels which require an assumption on the energy distribution within the volume. The conditions at the inlet and outlet to the core are also known but these values become meaningless to a CHF correlation when a flow reversal occurs.

The development of a new correlation was motivated by the desire to have a correlation which was easy to apply within the framework of the RELAP code and which contained only those variables which are locally defined over the entire blowdown transient.

Two ground rules then were established for the development of the correlation:

(a) The correlation would be based on experimental data.

(b) Only those locally defined parameters available in RELAP would be used in the model.

In addition it was assumed that the local conditions from steady-state tests could be used to define a model to be used during transient conditions.

This report briefly describes the data base used in the correlation development, the statistical methods employed, the correlation, and the behavior of the correlation over various ranges of conditions.

II. DATA BASE

Most of the data used for the development of the MOD7 CHF Correlation were selected from a compilation of steady-state CHF data made by Hughes[1] in 1973. Original sources for these data are given in References 1 through 8. Additional data obtained since Hughes' compilation are given in References 9, 10 and 11. A total of 5163 data points from 15 experimental investigations were used in the development of the correlation. Only data from rod bundles having rod arrays of 9 pins or more and rod lengths 2 ft. or longer were used. Hydraulic conditions were limited to those typically expected to exist during the blowdown of a large water reactor. The distribution of the data used in the correlation are shown in Figures 1 through 4.

Even though there are data from 15 different separate experiments, the limitation of the experimental facilities are obvious from Figure 1. This figure shows mass flux plotted vs quality for each of the data points used. Note particularly, the lack of low quality, low flow data and high quality, high flow data. This is because for any given mass flux there is a limited range of qualities where critical heat flux can be achieved within any given facility. The effect of this on the range of critical heat flux is shown in Figures 2 and 3 where the quality and mass flux are respectively plotted vs critical heat flux.

Figure 4 shows the distribution of pressure and radial power factor for each data point. A much more uniform array is evidenced for these two parameters. The even distribution of the data over the entire range of interest provides a basis for a effective statistical analysis of the data.

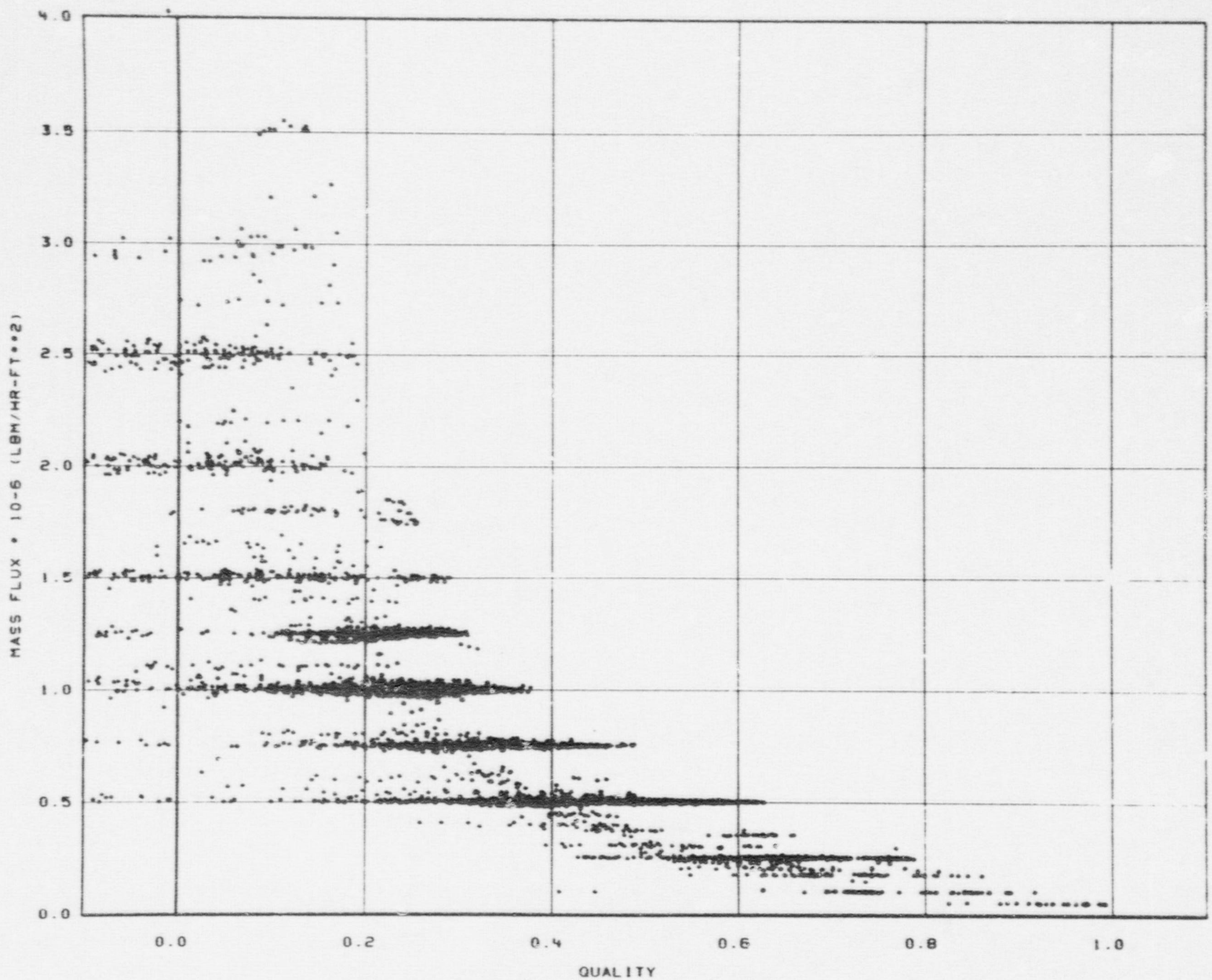

Fig. 1 Mass flux vs. local equilibrium quality for correlation data base.

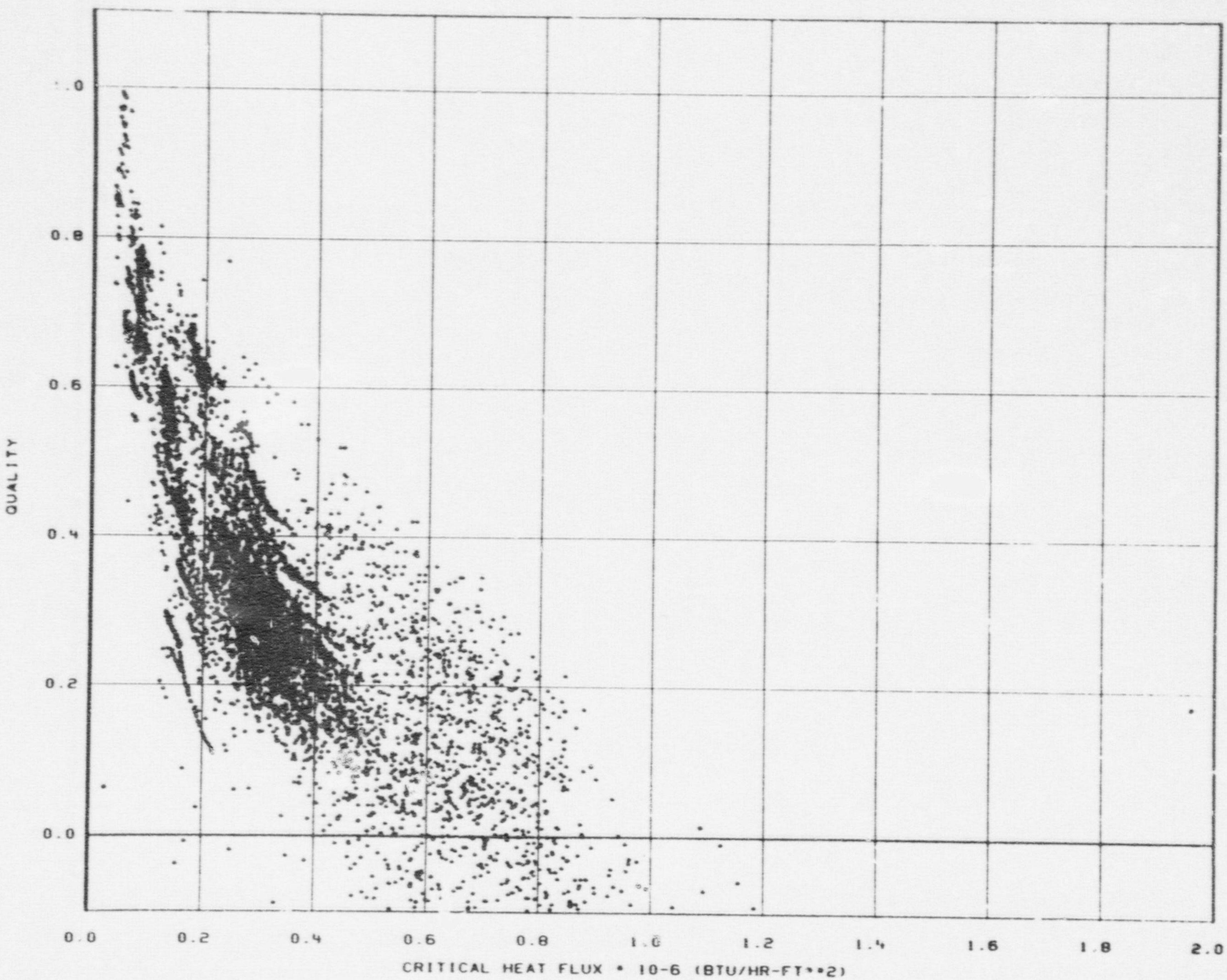

Fig. 2 Quality vs. critical heat flux for correlation data base.

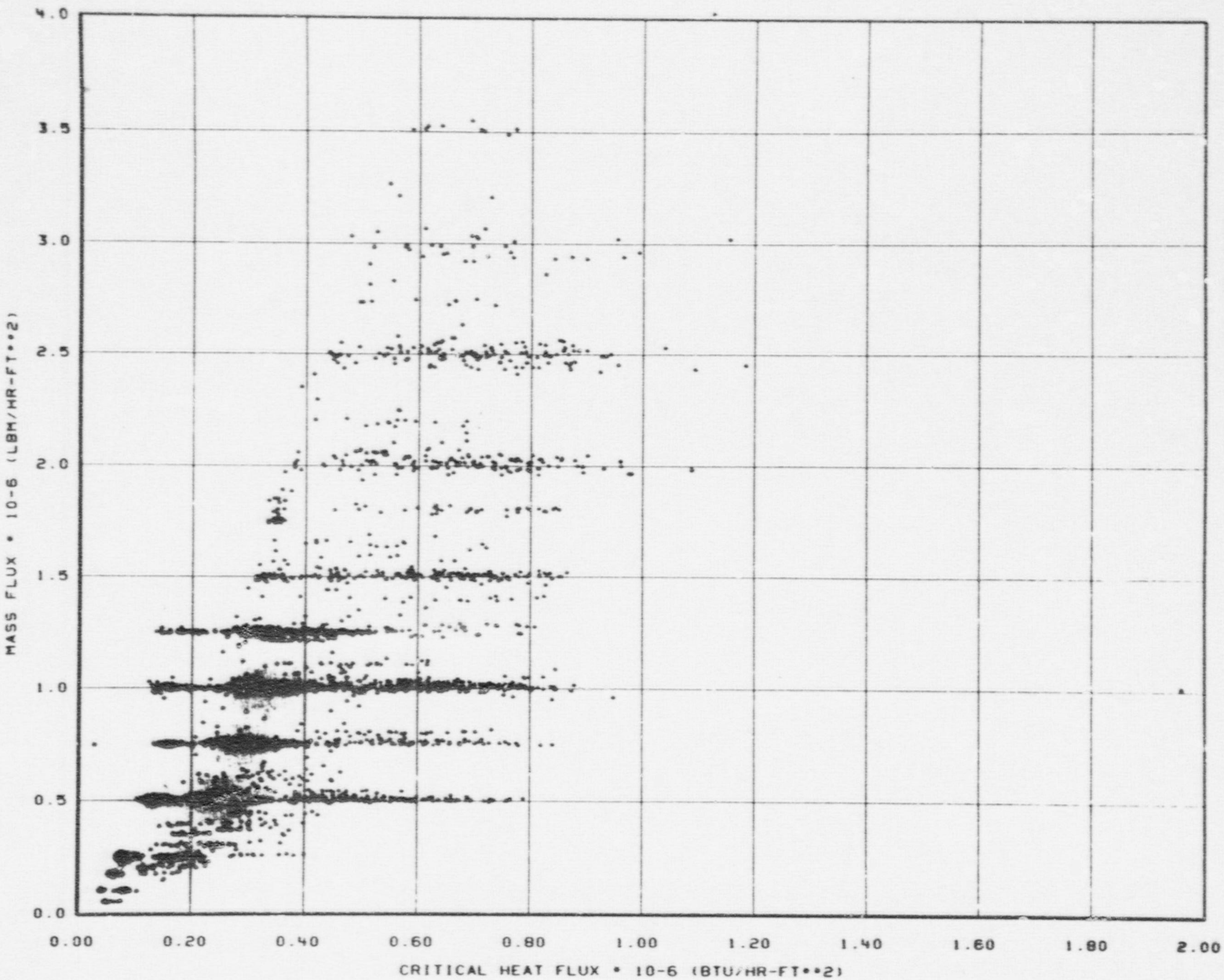

Fig. 3 Mass flux vs. critical heat flux for correlation data base.

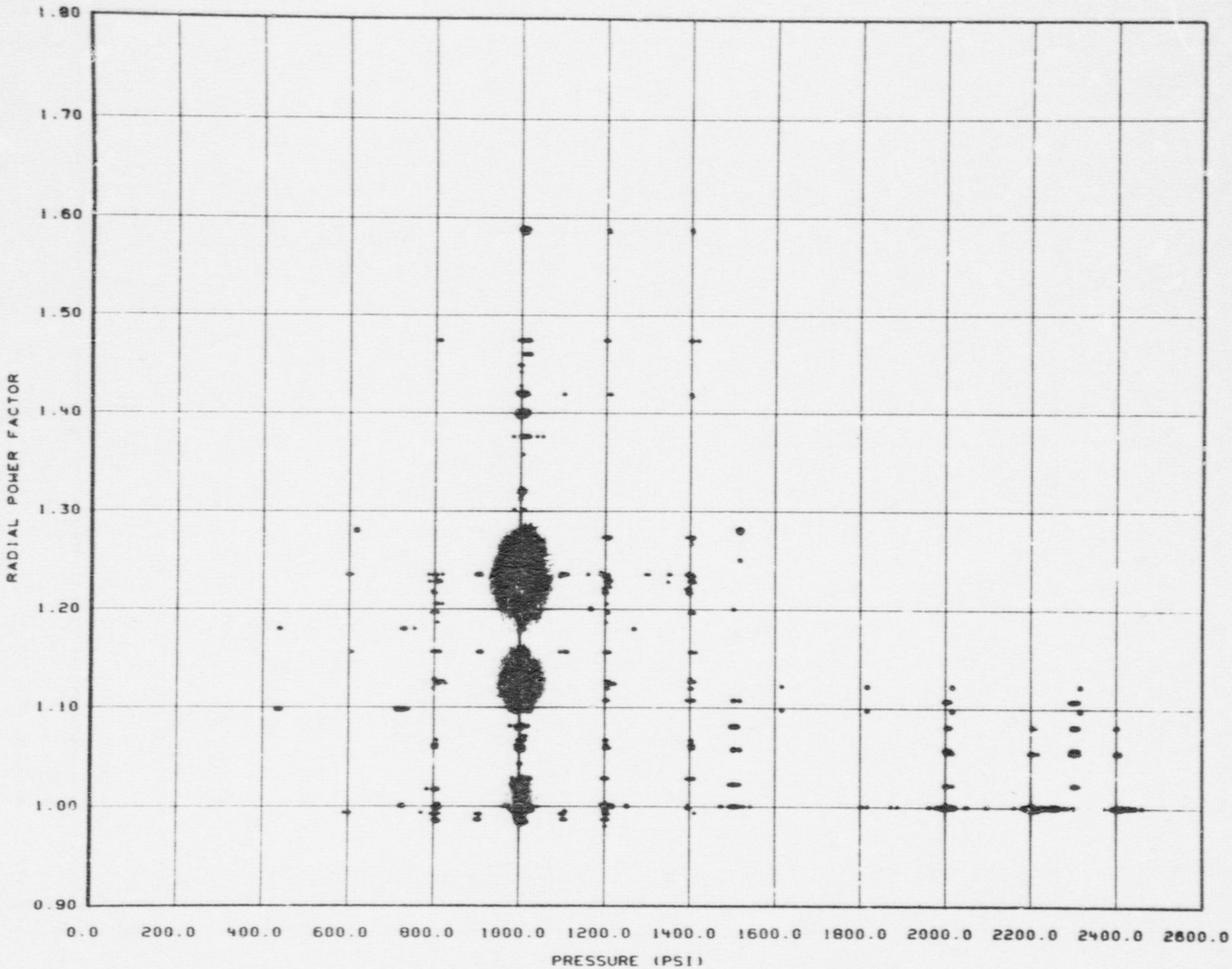

Fig. 4 Radial power factor vs. pressure for correlation data base.

III. CORRELATION DEVELOPMENT

The MOD7 CHF Correlation was developed using a multiple linear regression analysis of the data. Multiple linear regression analysis is a standard statistical technique that optimizes coefficients in an equation of linear parameters so as to best describe a set of experimental data. The particular regression code used in this analysis could select from the input list of parameters the most significant variables in descending order of importance and would then perform a new regression as each parameters was added. The code also supplied the user with a matrix of correlation coefficients between all combinations of two variables including the dependent variable. This made possible the recognition of those variables which were highly correlatable amongst themselves.

Two model forms can be used, either the additive or the multiplicative type. The additive model assumes the form

$$DV = \alpha_1 X_1 + \alpha_2 X_2 + \alpha_3 X_3$$

The multiplicative model, used throughout this study, is obtained by performing a linear regression on the logarithms of the independent variables. It takes the following form:

$$DV = \alpha_0 X_1^{\alpha_1} X_2^{\alpha_2} X_3^{\alpha_3}$$

where

DV = dependent variable

X_i = independent variables

α_i = coefficients or exponents.

The initial regression included 12 variables for each of the data points including the following:

Hydraulic Diameter
Heated equivalent diameter
Rod Pitch
Rod Diameter
Rod Length
Local Equilibrium Quality
Local Equilibrium Void Fraction
Pressure
Density
Mass Flux
Saturation Temperature
Bundle Radial Power Factor

Only four variables were shown to be statistically significant for correlation of the data. As one would expect from recalling Figure 2, quality was by far the most significant variable. Radial power factor, pressure and mass flow were also significant in the order listed. None of the geometry variables were shown to be significant for any of the regression runs. This does not mean that over a broad range of rod diameters there is not a geometry effect but rather that the data were derived from rod bundles having similiar rod diameters and pitch.

The initial regression run showed that certain variables could be used interchangably with insignificant differences in fit. Void fraction could replace quality. Any thermodynamic variable used, evaluated at saturated conditions could replace pressure. A strong interaction effect between quality and mass flow was also observed which after some trial and error led to the development of a quality flow interaction term which was used in the correlation. A value of 1.0 was added to the quality values to avoid discontinuties at subcooled conditions when the equilibrium quality is negative.

The final correlation is given below:

$$q''_{CHF}/10^6 = 8.0793 \quad \frac{(G/10^6)^{[0.1775 \ln(X+1)]}}{(X+1)^{3.3906}\ P^{0.3234}\ RPF^{1.0531}}$$

where

q''_{CHF} = critical heat flux (Btu/hr-ft^2)

G = Bundle average mass flux (lbm/hr-ft^2)

X = Bundle average quality

P = Pressure (psia)

RPF = Maximum radial power factor for bundle.

The correlation fits the data with a standard deviation of 29%.

Note from the formulation of the correlation that if the mass flux is zero the predicted critical heat flux would also be zero. At very low flows the critical heat flux mechanism approaches that for pool boiling conditions and becomes essentially independent of mass flow. Because no pool boiling data were included in the data base the extrapolation of the correlation to zero flow would not be appropriate. For application purposes it is recommended that a lower critical heat flux limit of 90,000 Btu/hr-ft^2 be used. This lower limit must be included by the user.

Figure 5 is a plot of the predicted critical heat flux vs the measured critical heat flux. The diagonal line is the locus of points where the predicted value is identical to the measured value. Above the diagonal the data are overpredicted by the correlation and below the diagonal the correlation underpredicts the experimental data. By the definition of the regression technique half of the data will fall on either side of the line.

Figures 6 through 9 are residual plots for each of the four variables used in the correlation. Residual is defined as

$$\text{Residual} = \frac{\text{measured flux - predicted flux}}{\text{predicted flux}}$$

using this definition the residual can have a value from -1.0 to + ∞.

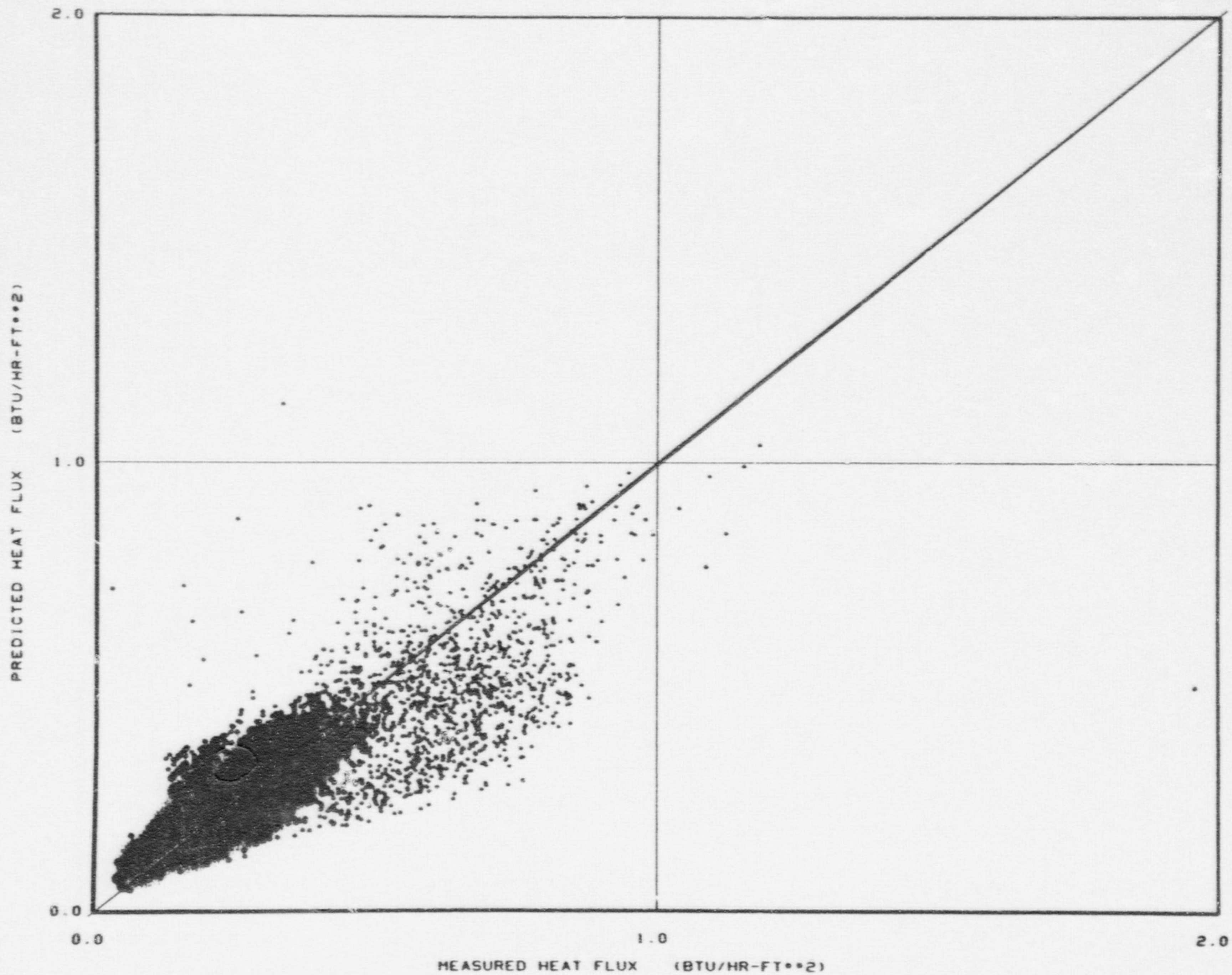

Fig. 5 Predicted vs. measured heat flux for MOD7 CHF correlation.

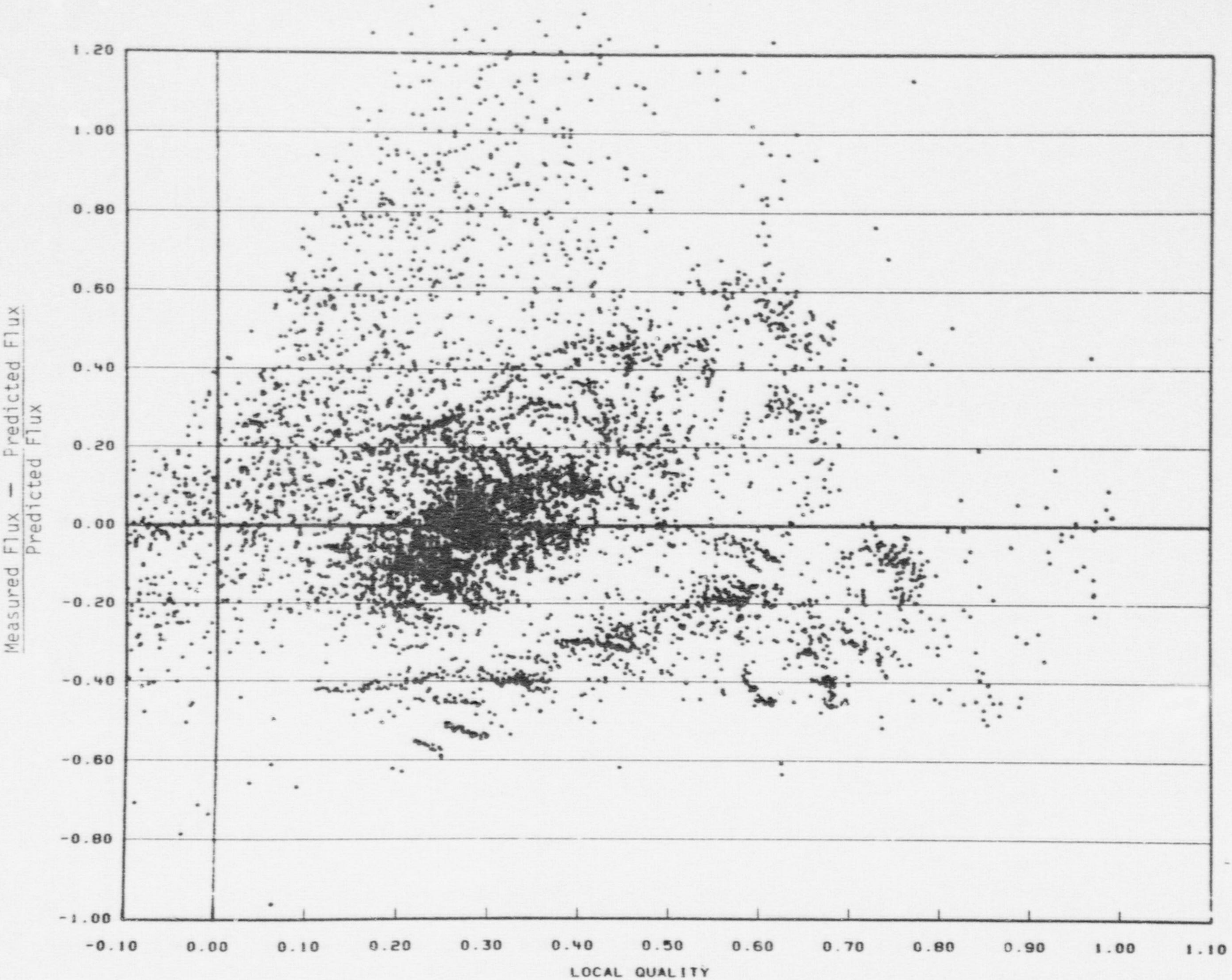

Fig. 6 Residual vs. quality for MOD7 CHF correlation.

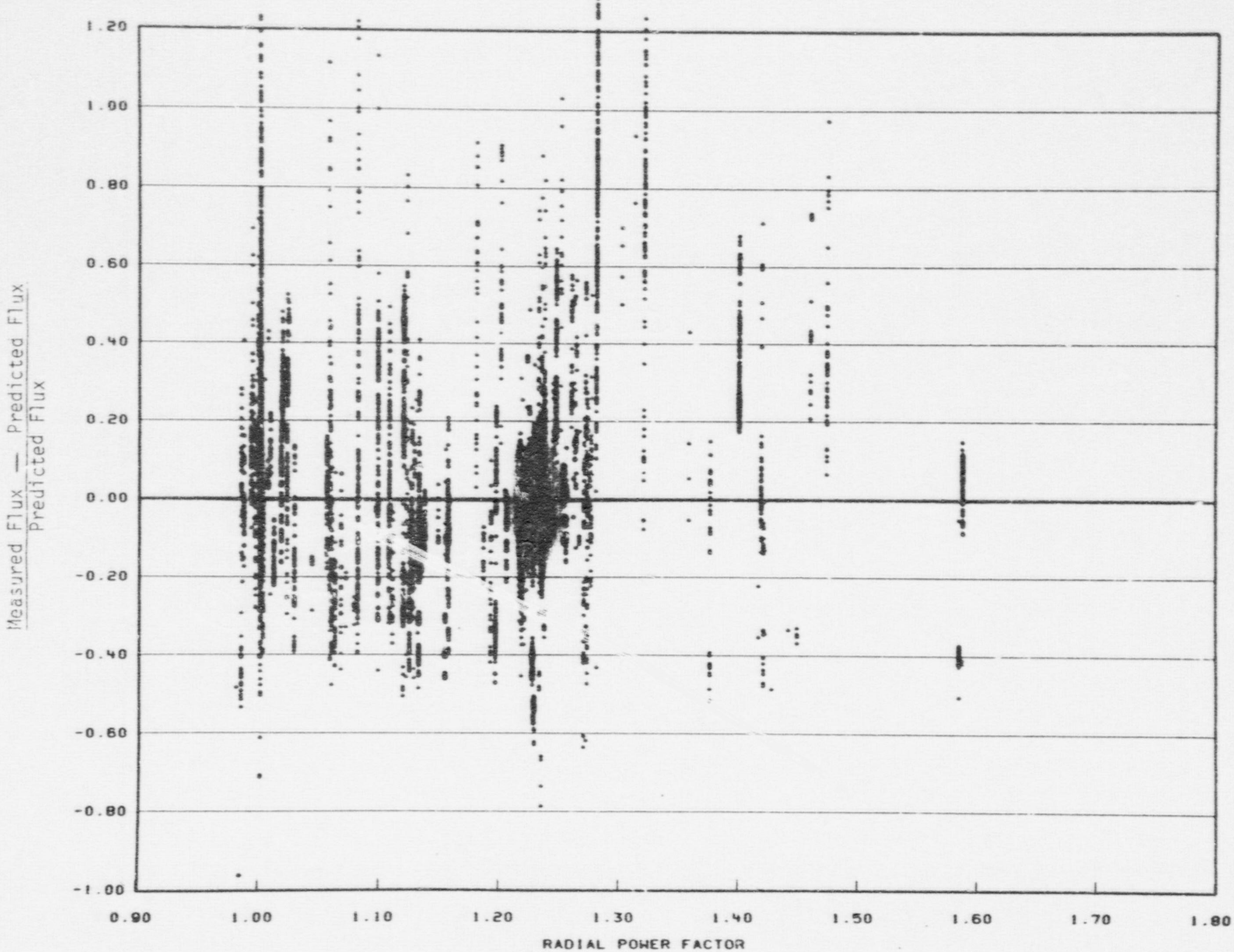

Fig. 7 Residual vs. radial power factor for MOD7 CHF correlation.

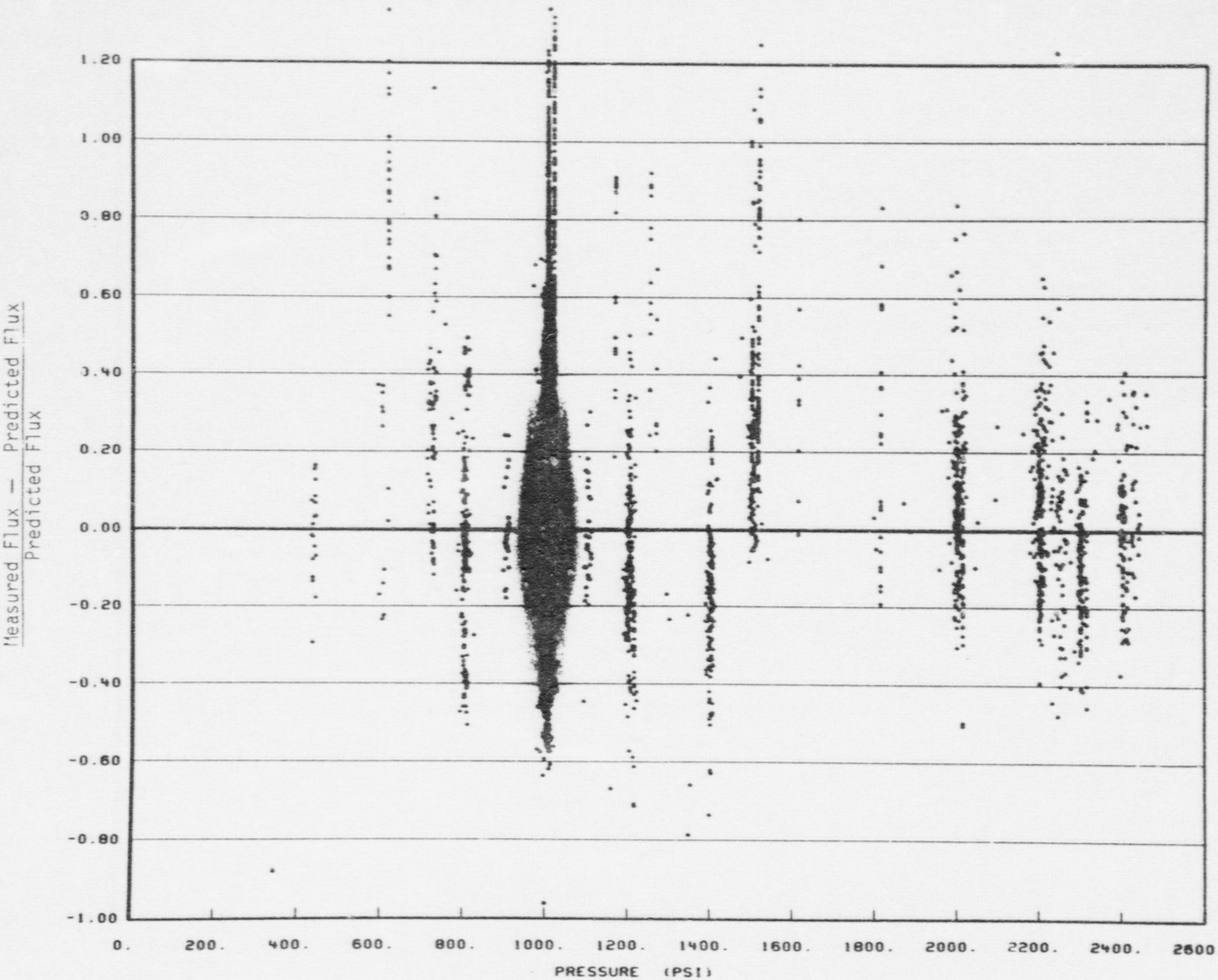

Fig. 8 Residual vs. pressure for MOD7 CHF correlation.

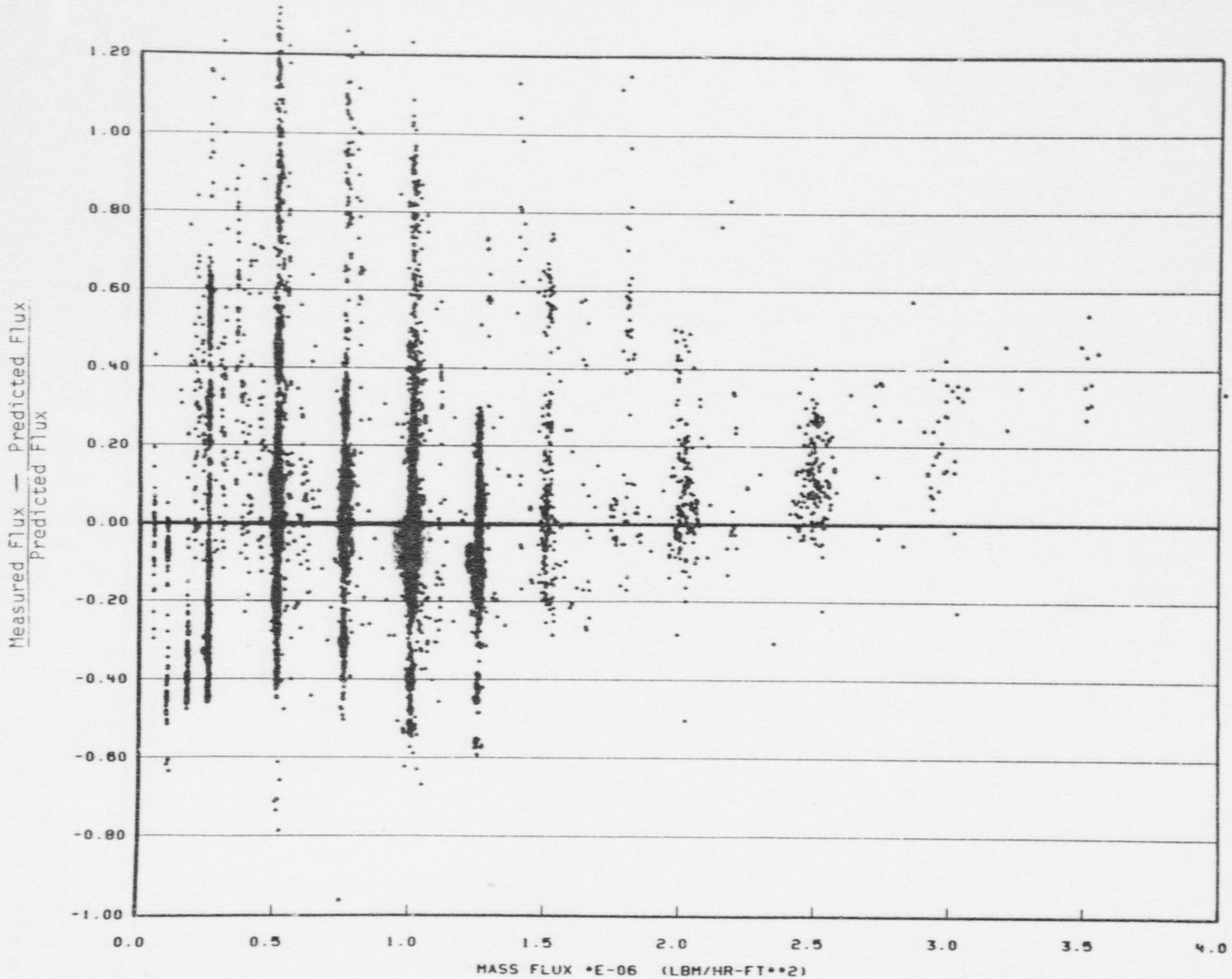

Fig. 9 Residual vs. mass flux for MOD7 CHF correlation.

The residual plots are extremely useful in observing any trends the correlation may have with respect to any one of the independent variables. The figures indicate ranges of the independent variables where there are large uncertainties in the prediction of the data. However there do not appear to be any trends in the correlation with respect to the independent variables.

IV. PARAMETRIC INVESTIGATION

A comprehensive parametric study of the MOD7 CHF Correlation was made to observe the dependence of the calculated critical heat flux on each of the four independent variables used in the correlation over the potential range of application.

Figures 10 through 13 show selected plots of the calculated critical heat flux vs each of the independent variables while holding all other variables constant. Approximately 400 such plots were constructed and reviewed.

Figure 10 shows the calculated critical heat flux vs mass flux. Except for flow rates below 0.50×10^6 lbm/hr-ft^2 there is very little dependance on flow rate. Below 0.50×10^6 lbm/hr-ft^2 the critical heat flux decreases with decreasing flow, until as noted previously, the correlation predicts zero heat flux at zero flow. In application a minimum critical heat flux value of 90,000 Btu/hr-ft^2 is used.

Figure 11 shows the critical heat flux for changes in radial power factor. There is an almost linear decrease in critical heat flux for increasing radial power factors.

The effect of pressure on critical heat flux is shown in Figure 12 for a given set of conditions. There is a decrease in critical heat flux with increasing pressure. This dependency is more pronounced at pressures below 800 psia.

The critical flux decreases significantly with increasing quality as shown in Figure 13. While not shown in this plot, it should be pointed out that when quality is identically zero the critical heat flux shows no flow rate dependency.

The trends shown in the previous plots are typical for the entire range of application.

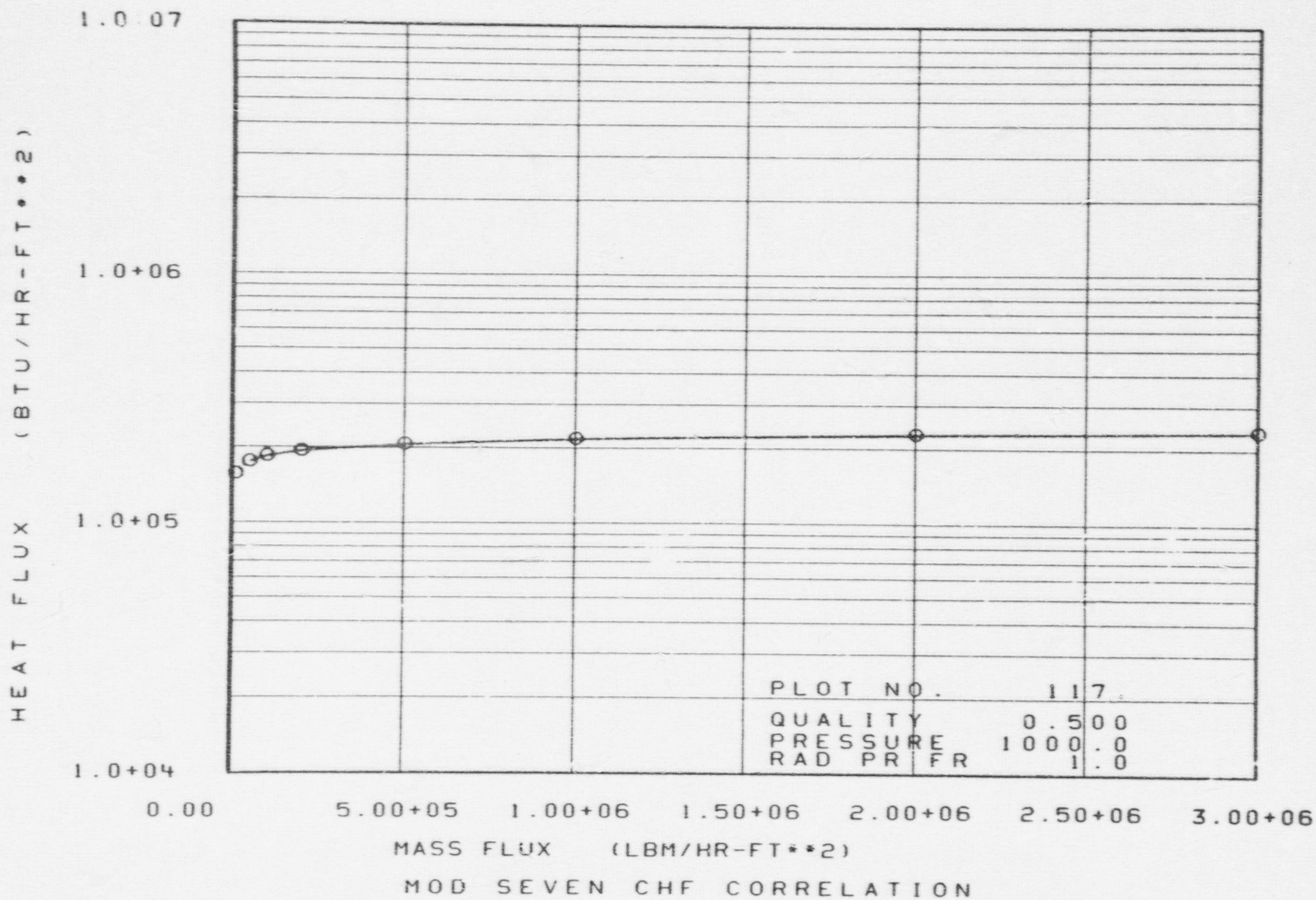

Fig. 10 Predicted heat flux vs. mass flux for MOD7 CHF correlation.

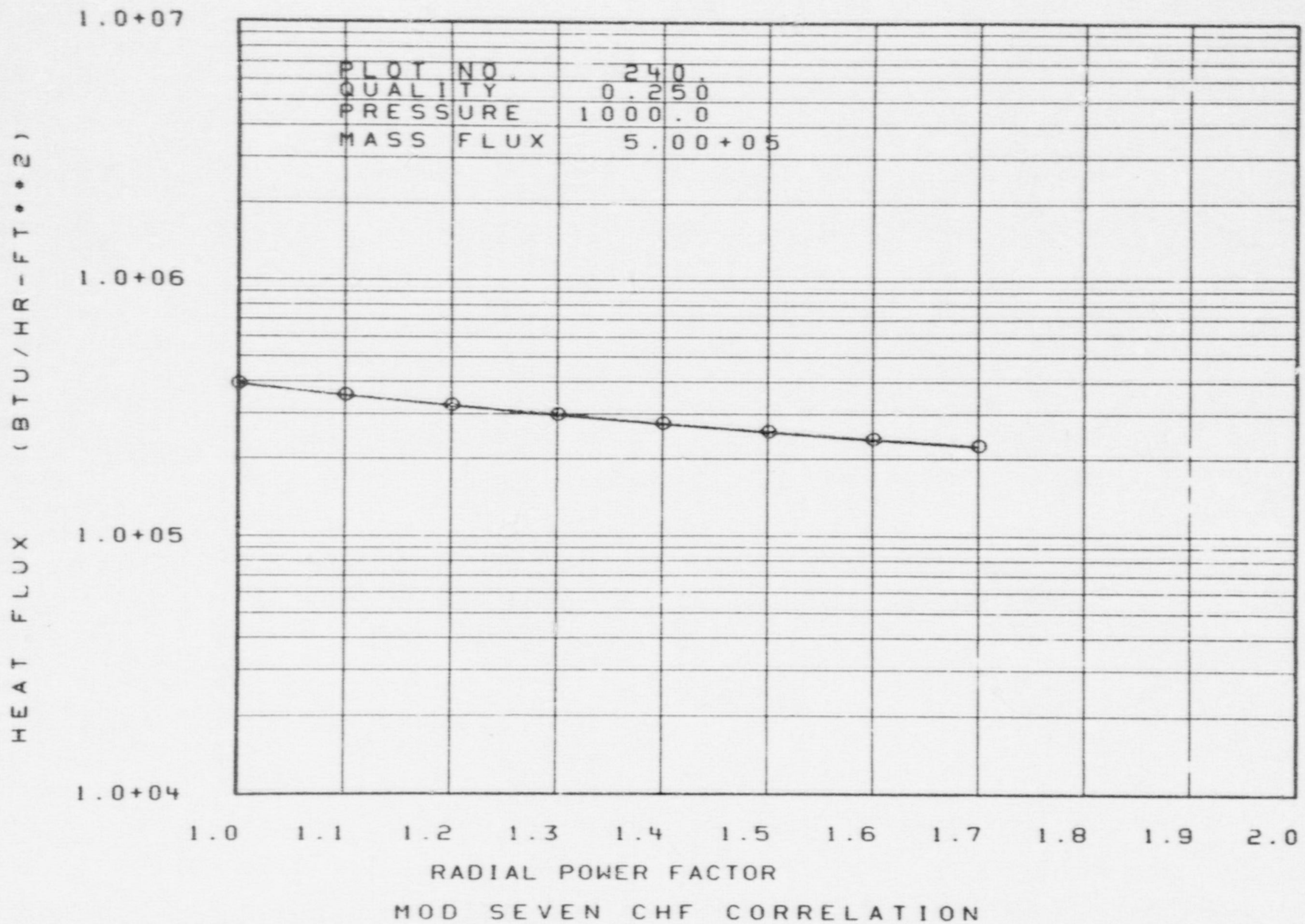

Fig. 11 Predicted heat flux vs. radial power factor for MOD7 CHF correlation.

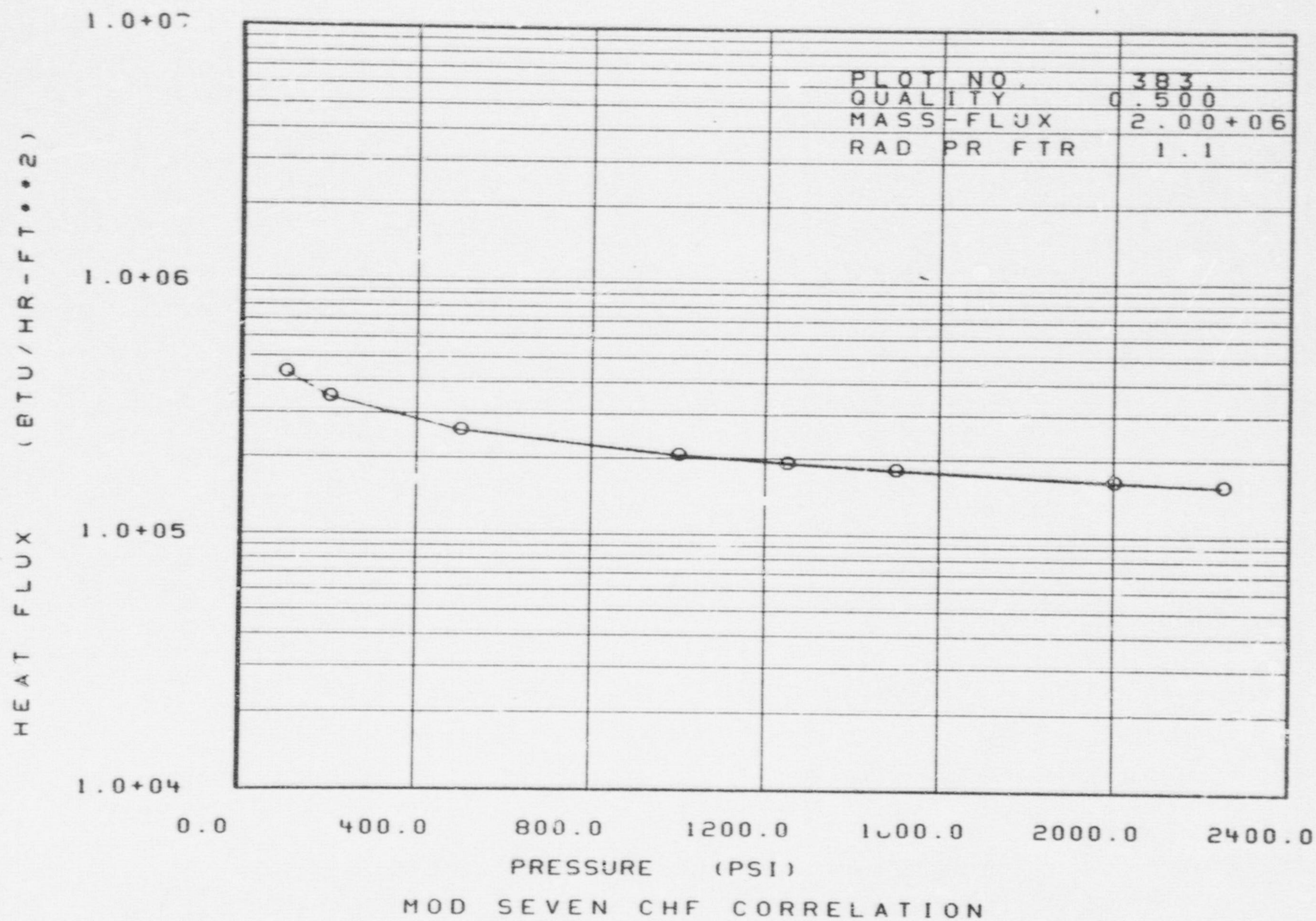

Fig. 12 Predicted heat flux vs. pressure for MOD7 CHF correlation.

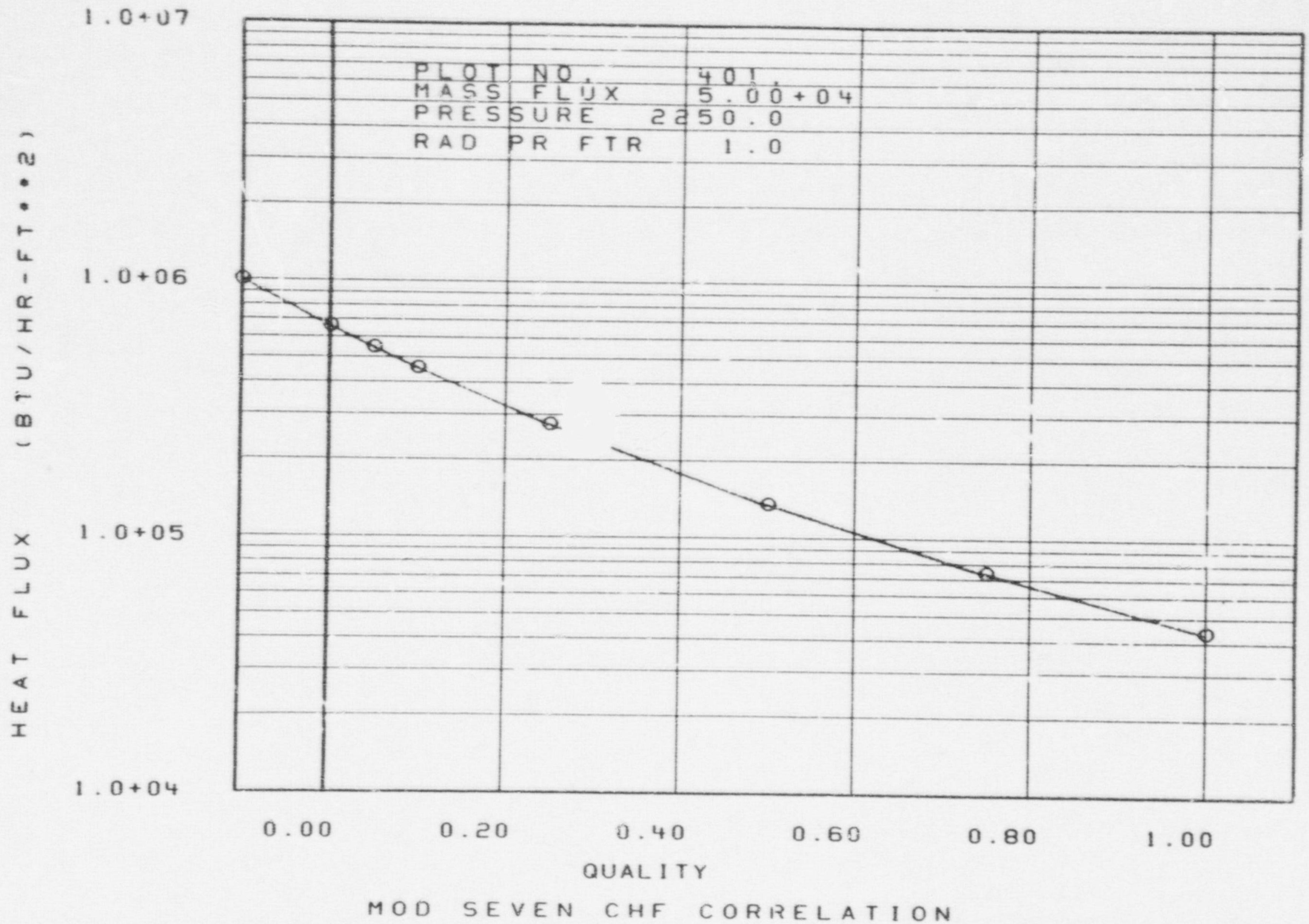

Fig. 13 Predicted heat flux vs. local quality for MOD7 CHF correlation.

V. CONCLUSIONS

A very simple and easy to apply correlation has been developed which can be used to predict critical heat flux during blowdown of a nuclear reactor. The correlation fits the data within the uncertainty bounds of the data itself. The level of sophistication of this correlation is of the same order of magnitude as the information available from RELAP.

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