COMBUSTION DIVISION COMBUSTION ENGINEERING, INC
WINDSOR, CONN 06095
203-688-1911 CABLE COMBENG

COMBUSTION DIVISION

August 1, 1973

Dr. S. Salah
United States Atomic Energy Commission
Room 511
7920 Norfolk Avenue
Bethesda, MD 20014

Dear Dr. Salah:

Enclosed is a Xerox copy of the section of the Idaho Nuclear Corporation Report, Ny-123-69, concerning the statistical verification of the Macbeth correlation. The pages of specific interest to you are II-3 and II-4.

If I can be of any further assistance, please don't hesitate to contact me.

Sincerely yours

THIS COPY FOR

Dr. Charles L. Kling
Supervisor, Nuclear Safety Dept.

CLK:vpl

Enclosure

NFV DEC 11 1969

IDAHO NUCLEAR CORPORATION

P.O. BOX 1845 IDAHO FALLS, IDAHO 83401

208-522-6640
LD-208-526-0111

November 10, 1969

Nuclear Safety Program
Monthly Report - October
Ny-123-69

To: Addressees

Gentlemen:

Enclosed please find your copies of the October Monthly Report of the Nuclear Safety Program Division of Idaho Nuclear Corporation.

Very truly yours,

W. E. Nyer

WEN:bc

W. E. Nyer, Manager
Nuclear Safety Program Division

Enclosure: Nuclear Safety Program Report - October 1969

Addressees:

C. N. Kelber - ANL
L. Baker - ANL - LMFBRPO
J. A. Redfield - BAPL
D. L. Morrison - BMI
T. A. Zaker - IITRI
T. W. Trout - KAPL
W. B. Cottrell - ORNL (5)
L. C. Oakes - ORNL
G. J. Rogers - PNL
G. M. Brown - SNE
T. A. Flynn, Jr. - EBASCO
R. G. Bock - GE - San Jose
W. K. Dodson - KE
J. D. O'Toole - UNC
F. R. Farmer - UKAEA
H. G. Rumpf - TUV
A. Jahns - IRS
H. A. Sindt - WAPD
D. F. Judd - B & W
R. E. Wascher - B & W
M. F. Valerino - CE
L. E. Minnick - CYAPC
A. P. Bray - GE
D. O. Fischer - GE
G. E. Wade - GE
D. Freeman - GE - Sunnyvale
T. Rockwell, III - MPR
J. E. Grund - PGE
R. A. Wiesemann - WCAP
J. F. Palmer - AECL
J. Bourgeois - CFA
Z. Tagami - HCRL
M. Ishikawa - JAERI

November 10, 1969

cc: Assistant Director for Nuclear Safety, DRDT
Assistant Director for Project Management, DRDT
Assistant Director for Plant Engineering, DRDT
Assistant Director for Reactor Engineering, DRDT
LOFT Project Manager, DRDT
LOFT Program Manager, DRDT
Chief, Research and Development Branch, DRDT
Chief, Engineering and Test Branch, DRDT
Chief, Environmental and Sanitary Engineering Branch, DRDT
Chief, Fuel Engineering Branch, DRDT
Chief, Control Mechanism Branch, DRDT
Chief, Water Systems Branch, DRDT
Division of Naval Reactors, DNR (R. S. Brodsky)
Director, Division of Reactor Standards
Deputy Director, REG
Assistant Director for Reactors, REG
Assistant Director for Special Projects, REG
Director, Division of Reactor Licensing, REG
Chief, Nuclear and Systems Technology Branch, DRL (2)
Director, Nuclear Technology Division, ID (3)
Director, Nuclear Safety Division, ID
Director, LOFT Project Division, ID (3)
Director, Nuclear Engineering and Construction Division, ID
Chief, Technical Services Branch, OSTS Division, ID
Chief, Budget Branch, ID

IDAHO NUCLEAR CORPORATION

Nuclear Safety Program Division

MONTHLY REPORT

October 1969

The information contained in this report is preliminary and subject to further evaluation.

TABLE OF CONTENTS

October 1969

II. NUCLEAR SAFETY DEVELOPMENT BRANCH

A. Analytical Development

1. Highlights

None

2. Technical Activities

Core Heat Transfer Analysis: A users manual has been prepared and distributed within Idaho Nuclear Corporation for the single rod, single channel code THETA1. Thermophysical properties for UO_2 and Zircaloy-4 have been incorporated in the code.

Work is continuing toward development of a three-dimensional core heatup code for the FRUST 8x8 rod bundle. A subroutine to calculate the fluid energy balance at each axial level is currently being developed. With the addition of the subroutine the code will be operational.

The CHF study continued. A total of 729 low pressure (150 to 725 psia) rod bundle CHF data points has been collected and correlated, and a report of this study started. Also, a report describing the comparison of selected CHF correlations to data previously collected was started. A portion of this report follows.

One major parameter in loss-of-coolant accident (LOCA) analysis is the time-to-critical heat flux[a] (CHF), that is, the time from initiation of the break to the time critical heat flux occurs in the core. Immediately following the occurrence of a break, the pressure differential across the core results in forced convection which begins to transfer the sensible energy stored in the fuel rods and the decay energy generated by nuclear fission from the rods to the coolant flowing through the core. Eventually the critical heat flux condition, which is a function of the rod and core geometry, axial heat flux distribution, and the flow and thermodynamic state of the coolant is reached. At CHF a film of vapor exists adjacent to the rod surface and results in severe reduction in the amount of energy that can be removed from the core. Following the attainment of the CHF condition, the rod surface temperature begins to increase as the energy remaining in the rods and the generated decay energy becomes redistributed.

Although prediction of the critical heat flux is necessary for reactor design purposes, the phenomenon of critical heat flux maintains a unique status in engineering in that no theoretical analysis has been attempted

---

(a) Critical heat flux is used in this report to describe a large decrease in the local heat transfer coefficient which results in a large increase of the surface temperature.

and consequently experimental data are the only source of information regarding its occurrence. Therefore, in order to obtain design values of the critical heat flux, experiments are usually conducted at steady-state with electrically heated rods of design diameter, pitch, spacing, and length and with fluid conditions at or near the design operating point. If sufficient data with a systematic variation of the core inlet condition and coolant mass flux are taken, a correlation of the data may be attempted and this correlation in turn may be used as a design equation.

In contrast to the problem of obtaining information required for design purposes, the problem of predicting when CHF will be reached during a loss-of-coolant accident requires data over wide ranges of pressure, mass flux, and inlet conditions and, in addition, the effects of rapid changes in pressure and mass flux must be taken into account. An investigation of all CHF data available in the open literature for steady-state, uniform axial flux profile, rod bundles and of several CHF correlations has been undertaken with the objective of determining the applicability of the data and correlations to models being developed to predict the core thermal behavior during a loss-of-coolant accident.

The correlations considered in this report are probably the most widely known and used at the present time. They have been given by Macbeth(1), Barnett(2), Becker(3), General Electric(4), and Westinghouse (Shippingport)(4). Results obtained by using these correlations were compared with 1096 CHF data points for rod bundles which cover the following ranges of parameters:

Pressure	156 to 1400 psia
Mass flux	0.02×10^6 to 4.0×10^6 $lb_m/hr\text{-}ft^2$
Fluid inlet condition	0 to 373 Btu/lb_m subcooling
Rod diameter	0.250 to 0.625 in.
Rod length	30.0 to 178.0 in.
Spacing between rods	0.022 to 0.307 in.
Axial heat flux distribution	Uniform

---

(1) R. V. Macbeth, Burnout Analysis, Part 5: Examination of Published World Data for Rod Bundles, AEEW-R 358 (1964).

(2) P. G. Barnett, A Correlation of Burnout Data for Uniformly Heated Annuli and Its Use for Predicting Burnout in Uniformly Heated Rod Bundles, AEEW-R 463 (1966).

(3) K. M. Becker, A Burnout Correlation for Flow of Boiling Water in Vertical Rod Bundles, AE-276 (1967).

(4) L. S. Tong, Boiling Heat Transfer and Two-Phase Flow, New York: John Wiley and Sons, Inc., 1967.

For purposes of comparison, the percent error between the calculated and experimental values is defined as

$$\text{ERROR} = \frac{q_{CHF,e} - q_{CHF,c}}{q_{CHF,e}} (100) \quad (1)$$

where

$q_{CHF,e}$ = experimental critical heat flux,

$q_{CHF,c}$ = calculated critical heat flux.

The results of the calculations are presented in Figures II-1 through II-5 as graphs of the frequency that an error is encountered versus the percent error of Equation 1. Whereas these figures do not reveal the details of the error distribution with respect to pressure, length, mass flux, inlet subcooling, and bundle internal construction, they do give an overall view of the accuracy of the correlation.

Figure II-1 gives the error distribution of the Macbeth correlation[1]. The results are good considering the fact that the correlation is based on a very limited amount of data (172 points) given in Reference 1. When compared with CHF values determined for ranges of parameters within, as well as outside, the ranges on which it is based, the correlation predicts 97% of the data within the error bounds -20% to +25%[a].

Figure II-2 gives the error distribution for the Barnett correlation[2]. Again, the data were obtained from bundles whose geometric and operating characteristics were within and outside the range for which the correlation was based. All of the data are predicted within -32.5% and +27.5% and 97% is predicted within -22.5% and +17.5%. These results and an examination of Figures II-1 through II-5 indicate that the Barnett correlation is superior to the others for predicting steady-state CHF data from rod bundles.

Figure II-3 gives the error distribution for the Becker correlation[3]. Although the correlation is based on data over the pressure range $31 < P \leq 1400$ psia, and all of the rod bundle data are within this range, the predictions are not as accurate as those of the Macbeth and Barnett correlations. The calculated results for 97% of the data agree with experimental results within ± 25%.

(a) The 97% level of correlation is widely used in the open literature. However, the choice of the error bounds within which 97% of the data is predicted is quite arbitrary. For this report, values were chosen between which the error distribution is symmetrical.

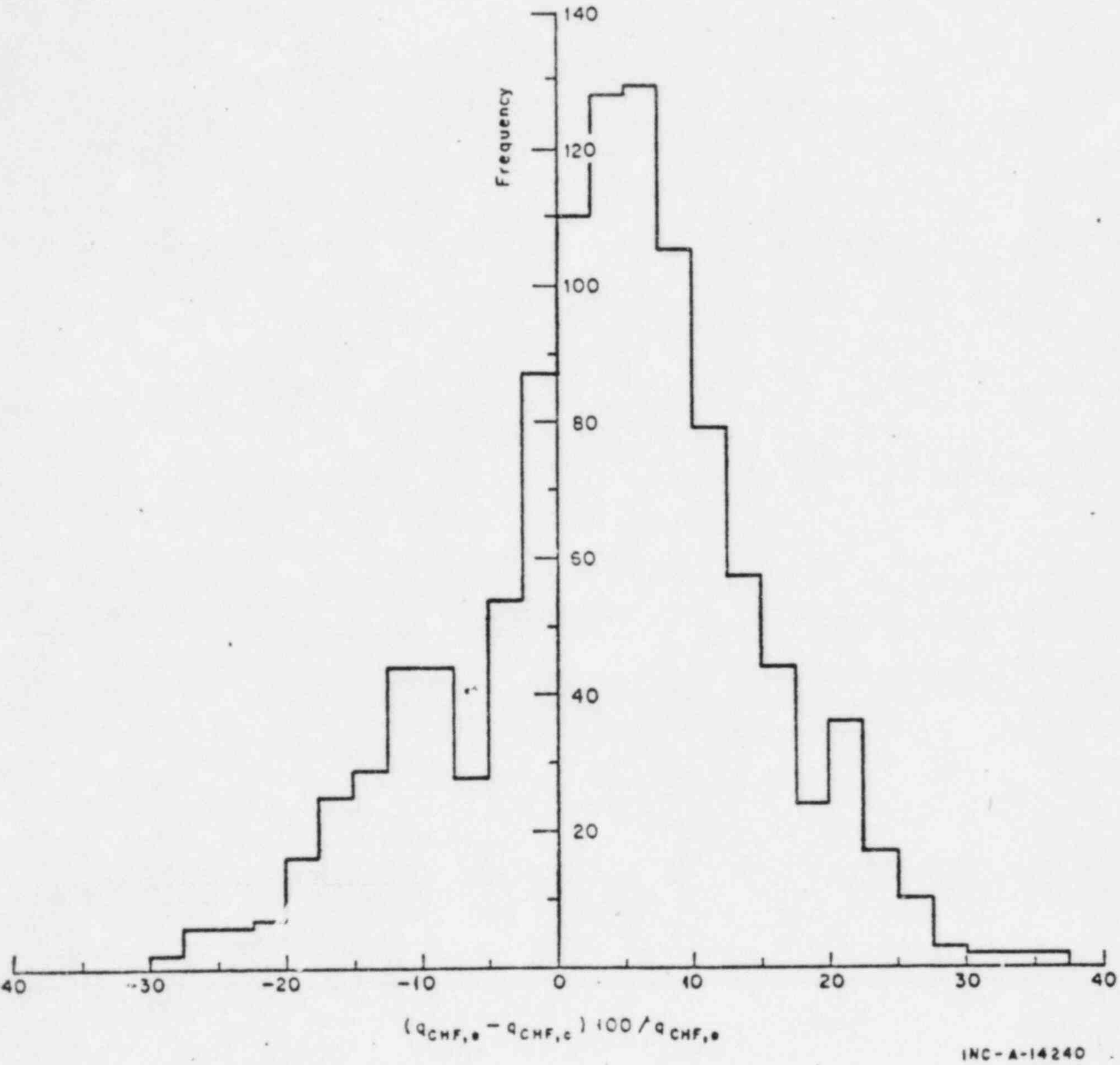

Fig. II-1

Error Distribution for the Macbeth CHF Correlation

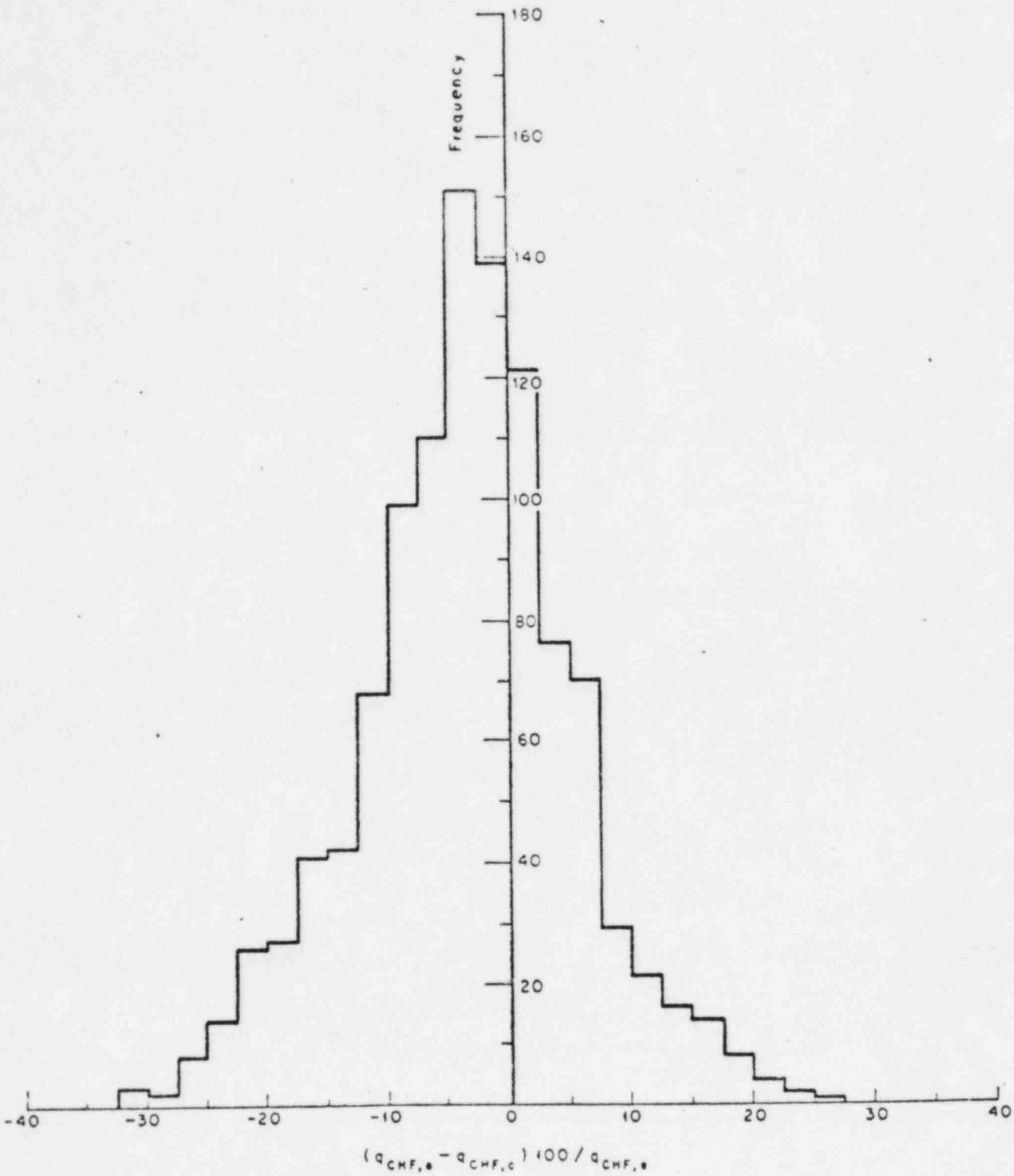

Fig. II-2

Error Distribution for the Barnett CHF Correlation

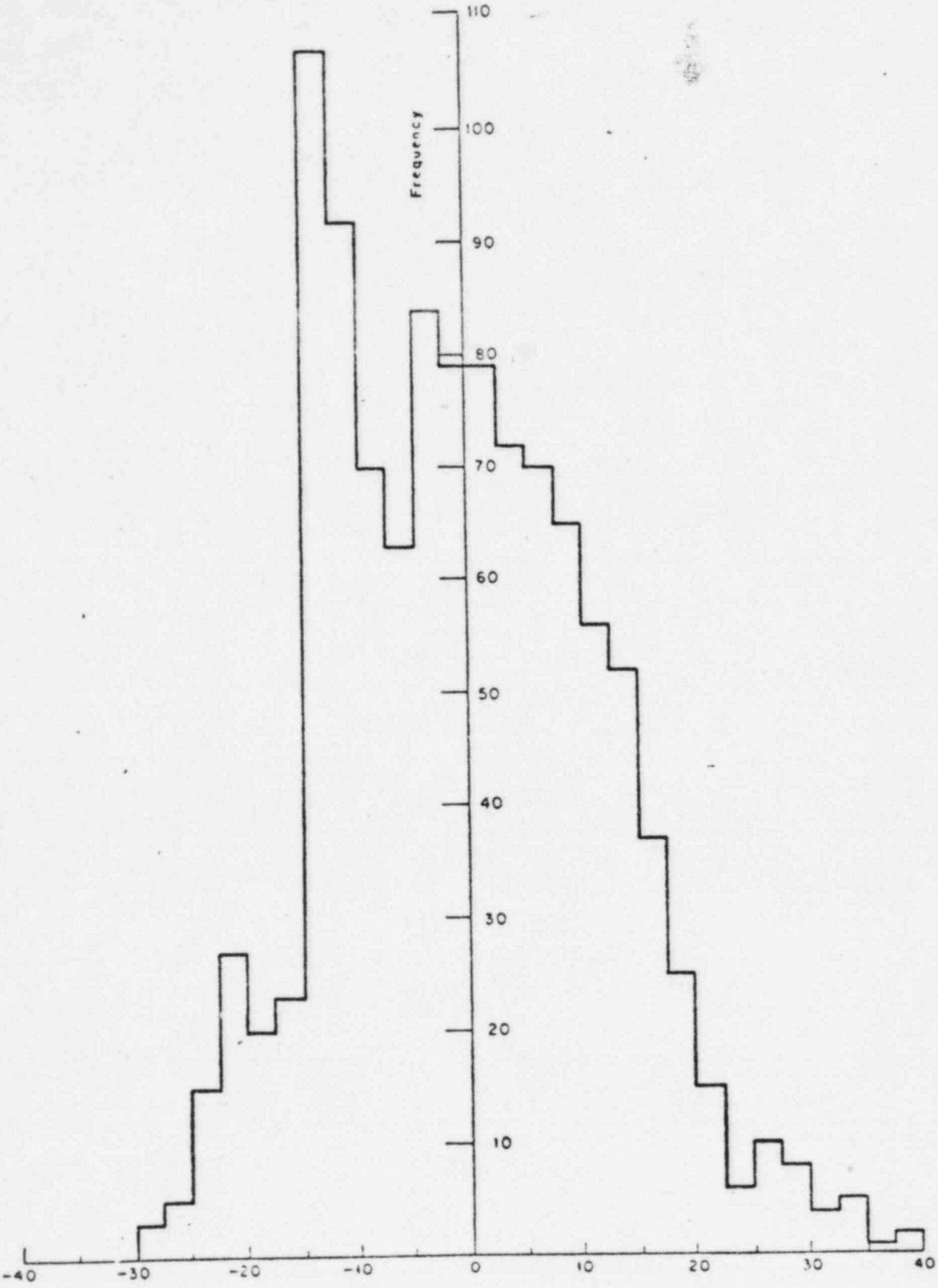

Fig. II-3

Error Distribution for the Becker CHF Correlation

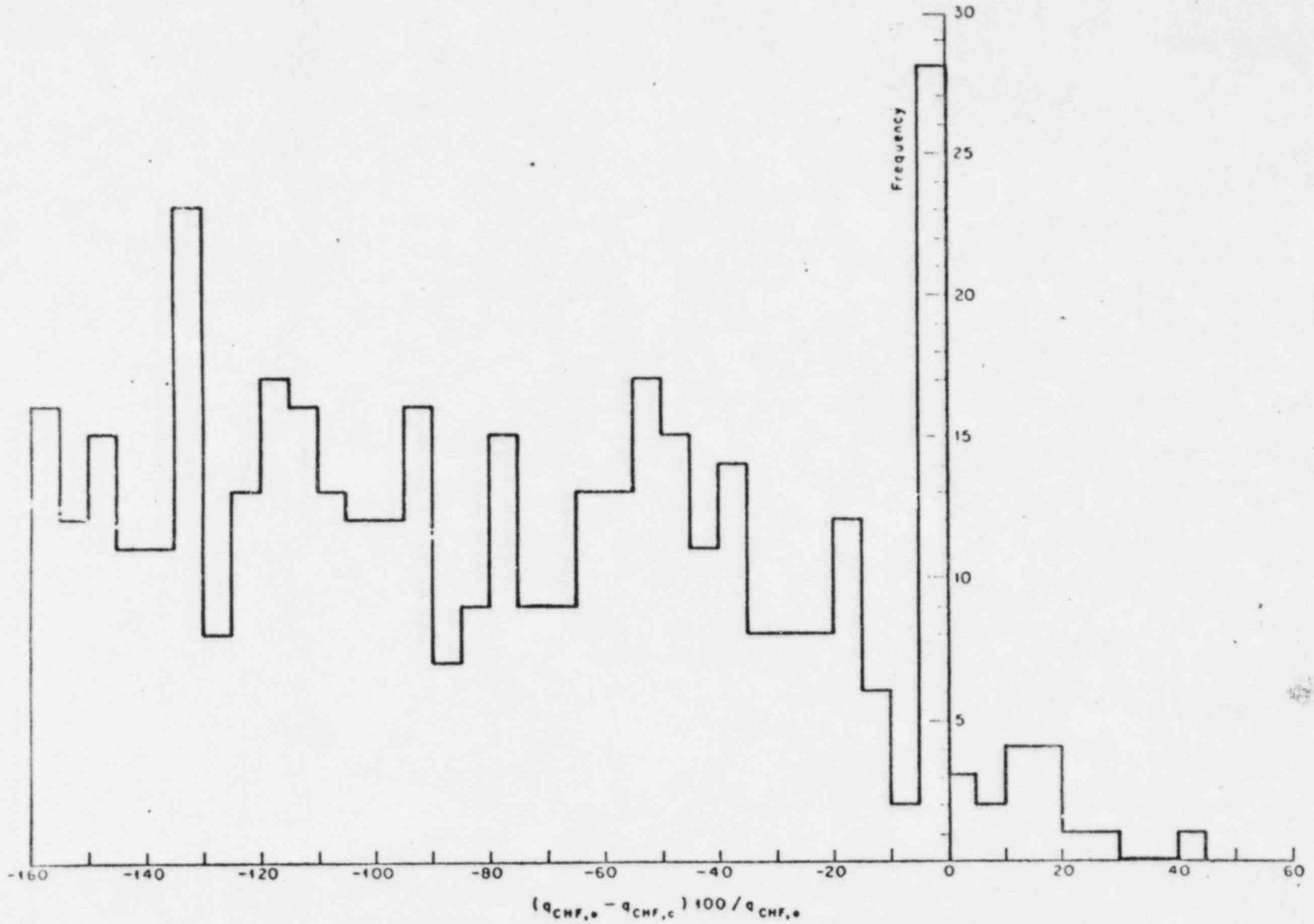

Fig. II-4

Error Distribution for the Shippingport CHF Correlation

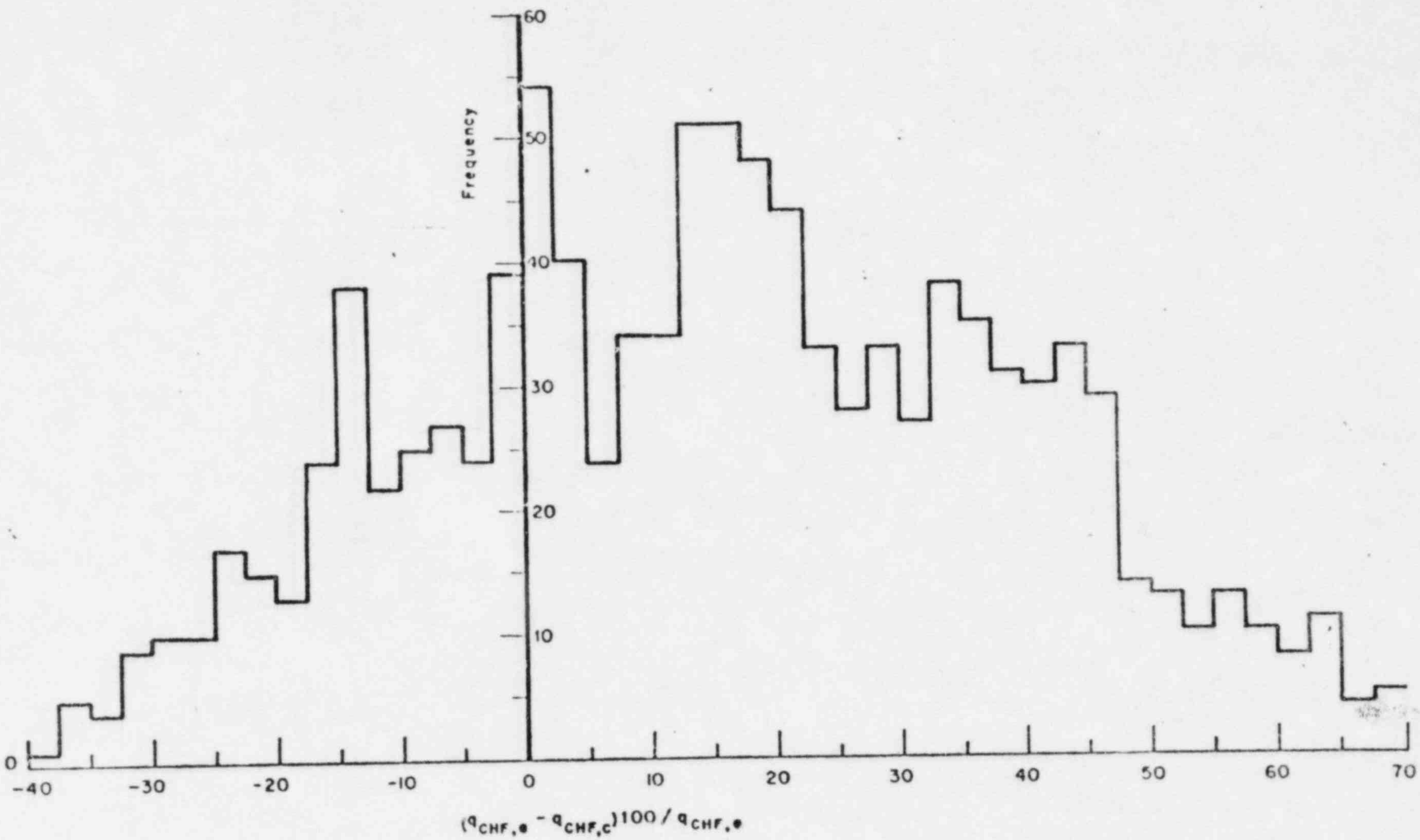

Fig. II-5

Error Distribution for the General Electric Company CHF Design Curve

Figure II-4 presents the error distribution for the Shippingport correlation[4]. Only 38% of the data were predicted within -160% to 45% error. The remaining 62% of the predictions produced error larger than -160% and, therefore, are not tabulated in Figure II-5. As defined by Equation 1, negative errors indicate an over-prediction of q_{CHF}, thus any correlation that yields negative errors is undesirable for both design and LOCA analyses. The results given in Figure II-4 clearly indicate the necessity of comparing any correlation with data before using the correlation. Figure II-4 also indicates the inaccuracies that are possible when a correlation based on data from a given geometry is applied to other geometries.

The General Electric Company design curve correlation was derived by constructing curves under CHF data considered to cover the operating range of BWR's (that is, a lower envelope was determined). That the correlation is conservative (positive errors) as intended is indicated by the results shown in Figure II-5. However, the success of using this method of correlation requires data for the entire range of parameters over which the correlation is to be applied. An examination of the predictions shows that the nonconservative errors (negative errors) in Figure II-5 were obtained for low mass flux data ($G < 1.0 \times 10^6$ $lb_m/hr\text{-}ft^2$).

The results presented in this section have shown that the method of correlation used by Macbeth[1] and Barnett[2] produce accurate CHF correlations that may be extrapolated with reasonable certainty outside the range of parameters on which they are based.

Thermal-Hydraulic Analysis: The development of new heat transfer models for use in RELAP3 has continued. The Macbeth[1] and Barnett[2] critical heat flux correlations are being used for calculating the critical heat flux. Currently, the Barnett correlation is being used for pressures above 1000 psia and the Macbeth correlation for pressures below 725 psia. For intermediate pressures an interpolation between the two correlations is used. The correlations have been coded and are currently being checked out.

An extensive analysis of the semiscale Test 818-2 (30% double-ended break) is being made. The test data are being compared with RELAP3 predictions made prior to running the test. Preliminary comparisons show good agreement between predicted and measured pressures.

Computer Sciences Support: An explicit technique for solution of the fluid energy equation in THETA1 has been developed by modification of a Lax-Wendroff technique. This technique provides a more stable and accurate solution of abrupt changes in boundary conditions (that is, the channel inlet enthalpy change that occurs during flow reversals) than the present explicit or implicit methods currently in THETA1. This new technique will replace the present explicit technique.

October 1969

B. Separate Effects Tests

1. Highlights

A preliminary system design description document has been prepared for the two-loop Semiscale Blowdown and ECC System.

A series of channel blockage experiments in support of FLECHT has been completed on a 9-pin heater assembly.

2. Technical Activities

Semiscale Blowdown and ECC (Single Loop): The semiscale blowdown assembly is presently undergoing modifications in preparation for Test Series II Group 2. Modifications consist of the installation of an electrically heated core, the addition of three gamma ray densitometers to obtain measurements of average fluid density in the core, a change in the loop configuration from two rupture devices to one rupture device, and the installation of additional instrumentation.

Specifications have been issued to the Procurement Branch for 3-foot heaters for blowdown heat transfer studies. Request for quotations were issued October 27; the bid opening date will be November 17, 1969. A delivery date of February 27, 1970, has been specified.

Semiscale Blowdown and ECC (Two Loop): The two-loop blowdown system is in the conceptual design stage. A preliminary system design description document has been prepared for review and comments. Work has begun on specifications for long lead time components.

Nine-Pin Channel Blockage Experiment: A series of experiments has been completed with a 9-pin core to study the effect of flow channel blockage on emergency core cooling heat transfer. The experiments were conducted by flooding the core at temperatures of 1600 and 1800°F with restricted flow area at the center of the core. Restriction in flow area was varied during the test series from no restriction to 90 percent restriction. Results are being evaluated.

FHUST: The FHUST core for Test Series II testing has been assembled and installed in the FHUST vessel. Testing is scheduled to begin the second week of November after the instrumentation has been connected and the system has been checked out.