UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 20 1979

MEMORANDUM FOR: Roger J. Mattson, Director
Division of Systems Safety, NRR

FROM: Richard P. Denise, Acting Assistant Director for
Reactor Safety, DSS

SUBJECT: SUMMARY MINUTES OF MEETING ON CLADDING RUPTURE TEMPERATURE, CLADDING STRAIN, AND ASSEMBLY FLOW BLOCKAGE

On November 1, 1979, the NRC staff met with reactor fuel vendors, some plant licensees, and other interested parties (Enclosure 1 lists the attendees who signed the meeting roster) to discuss recently developed staff views on the safety analysis for emergency-core-cooling systems (ECCS). The staff presentations addressed (1) Zircaloy cladding behavior during a loss-of-coolant accident (LOCA) and (2) the sensitivity of the ECCS evaluation models to cladding swelling and rupture. The meeting agenda is attached as Enclosure 2.

During the meeting the Core Performance Branch presented preliminary results (a portion of a draft report is attached as Enclosure 3) of an ongoing generic review of three Zircaloy correlations used in the ECCS evaluation models. The three correlations are cladding rupture temperature, cladding circumferential strain at failure, and assembly flow blockage (i.e., reduction-in-flow area). Based on the interpretation of experimental data (most of which have been obtained subsequent to the 1973 rulemaking hearing and are listed in Enclosure 4), the staff had developed preliminary audit correlations (see Enclosure 5) for these three models. And a comparison of the staff correlations to the approved vendor models showed that over certain temperature and stress regimes the vendor models underpredicted the degree and incidence of cladding swelling and rupture, thus appearing to violate the requirements of Appendix K of 10 CFR 50.

The Analysis Branch presented the staff's assessment (see Enclosure 6) of what the significance of the discrepancies between the draft audit correlations and the approved vendor models could mean in terms of continued plant compliance to the ECCS acceptance criteria (10 CFR 50.46). Based on limited computer runs, it was thought that the significance was on the order of hundreds of degrees of peak cladding temperature depending upon the particular ECCS model dependency to cladding strain and rupture. Such was then the reason for the request to meet with the industry in order to determine the specifics of the limiting-LOCA analysis for all commercial plants that use Zircaloy cladding.

Contact: D. A. Powers, x27603

7912140 253

1573 300

As a result of the information received during the meeting, the staff concluded that, while the approved vendor models deviated significantly from the staff correlations on an overall basis, within the ranges of interest, one of two circumstances prevailed: (1) the vendor models were either conservative or very close to the staff correlations within these ranges, or (2) peak cladding temperature was relatively insensitive to the discrepancy. In either case, the fuel vendors (including Yankee Atomic) agreed to provide letters of confirmation to the staff showing that all operating plants would continue to be in conformance with the requirements of 10 CFR 50.46. These letters were to be received by 5:00 p.m. on November 2, 1979.

The details of the model discrepancies and the significance of the discrepancies are provided in Enclosure 7.

Richard P. Denise

Richard P. Denise, Acting Assistant
Director for Reactor Safety
Division of Systems Safety

cc: H. Denton
E. Case
F. Schroeder
D. Eisenhut
D. Ross
R. Tedesco
P. Check
R. Woods
Reactor Fuels Section
L. Phillips
G. Lauben
PDR
P. Boehnert, ACRS staff
ACRS - 16 copies
D. Bessette, ACRS staff
G. Marino, RES
D. Hoatson, RES
M. Picklesimer, RES
W. Johnston, RES
L. Olshin
S. Schwencer
R. Reid
W. Gammill
H. Rood
R. Chapman, ORNL

1573 301

ENCLOSURE 1

ATTENDEES AT THE MEETING ON CLADDING SWELLING AND RUPTURE

NOVEMBER 1, 1979

NRC

D. Eisenhut
R. Denise
G. Lauben
M. Picklesimer
P. Boehnert
G. Marino
F. Coffman
W. Johnston
J. Voglewede
K. Kniel
G. Alberthal
L. Olshin
R. Woods
S. Rubin
S. Schwencer
R. Reid
W. Gammill
H. Rood
R. Tedesco
R. Meyer
D. Powers
P. Check

Florida Power & Light Co.

S. Sarkar

Duke Power Co.

S. Rose

Atomic Industrial Forum

F. Stetson

Southern Co. Services

K. Folk

AEPSC

V. Manno

Baltimore Gas & Elec. Co.

R. Olson

ORNL

R. Chapman

Carolina P&L

R. Farr

Wisconsin Public Service Corp.

M. Stern
R. Hanneman

Westinghouse

S. Kopelic
V. Esposito
D. Burman

Boston Edison

J. Gosnell
J. Keyes

EPRI-NSAC

B. Leyse

GE

R. Elkins
A. Rao
N. Shirley

Press

W. Dillehay

Yankee Atomic Electric

A. Husain
S. Schultz

EXXON Nuclear

R. Collingham
G. Owsley
G. Cook

JCP&L

S. Chan

GPU

G. Bond

PASNY

S. Iyer

Combustion Engineering

G. Meuzel
E. Jageler
J. Cicerchia
C. Brinkman
D. Kreps

Phila. Electric Co.

L. Rubino

B&W - NPGD

A. Lowe
B. Short
B. Dunn
H. Bailey

1573 303

ENCLOSURE 2

MEETING ON CLADDING SWELLING AND RUPTURE
NOVEMBER 1, 1979

Introduction	R. Denise	10 minutes
Cladding Swelling and Rupture Information	R. Meyer	60 minutes
Potential Effects of New Models on ECCS	N. Lauben	45 minutes

Discussion

Lunch

Vendor and Licensee Feedback

- comments on data or models
- suggestions of work to be done
- suggestions for operating plant actions in the meantime

1573 304

ENCLOSURE 3

DRAFT
10/31/79
D. Powers/ R. Meyer

CLADDING SWELLING AND RUPTURE MODELS
FOR LOCA ANALYSIS

D. A. Powers and R. O. Meyer

1573 305

1. INTRODUCTION

During a postulated loss-of-coolant accident (LOCA), the reactor coolant pressure may drop below the internal fuel rod gas pressure causing the fuel cladding to swell (balloon) and, under some conditions, rupture. Core behavior during a LOCA would depend on the time at which swelling and rupture occurred, the magnitude of swelling, and resulting coolant flow blockage (i.e., reduction in flow area).

Such phenomena were among the many reactor safety issues discussed during the 1973 rule-making hearing on Acceptance Criteria for Emergency Core Cooling Systems (ECCS). The adopted acceptance criteria (Ref. 1) limited predicted (calculated) reactor performance such that if certain oxidation and temperature limits were not exceeded, then core cooling would be assured. It was required that each licensee use a safety evaluation model to analytically demonstrate compliance with the acceptance criteria.

Appendix K (Ref. 2) gives requirements for some features of evaluation models, and, in particular, states that to be acceptable the swelling and rupture calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated. The degree of swelling and incidence of rupture are then used to calculated other core variables including gap conductance, cladding temperature, oxidation, embrittlement, and hydrogen generation. After the conclusion of the ECCS hearing, the AEC reviewed and approved cladding behavior models for each U.S. fuel manufacturer for their use in ECCS analyses.

1573 306

During the ECCS hearing uncertainties were apparent in predicting fuel behavior during a LOCA. Therefore, in the Commission's concluding opinion (Ref. 3), the Commission directed the AEC's research office (now the NRC Office of Nuclear Regulatory Research) to undertake a major confirmatory research program on cladding behavior under LOCA conditions. The resulting multi-million dollar program includes simple bench-type Zircaloy tests, single- and multi-rod burst tests that simulate some in-reactor conditions, and actual in-reactor tests ranging to full-size bundle tests.

The research programs are not all finished, but with the completion of many out-of-pile and a few in-pile tests, we are at a plateau of understanding that greatly exceeds our understanding in 1974, and the results have not confirmed all of our previous conclusions. The trend of these recent data shows the likelihood of more ruptures, larger rupture strains, and greater flow blockages, than we previously believed. Consequently, we see the need to reevaluate all LOCA cladding models to assure that licensing analyses are performed in accordance with Appendix K.

In the following sections we will display the relevant body of data, describe our evaluation of these data to arrive at useable correlations (curves), and compare these correlations with those currently used in licensing analyses. Since the data show strong heating-rate* effects,

* Both heating rate and strain rate are important factors in determining cladding burst pressure and strain. However, most burst experiments are not designed to distinguish between heating-rate effects and strain-rate effects. For the purposes of this report, the actual differences are probably unimportant. Therefore to avoid confusion, in the remainder of this report we will refer to both effects simply as heating-rate effects.

we have derived different curves for slow ramp rates and fast ramp rates. But most current ECCS models do not include a ramp rate effect, so we have also displayed composite curves that envelope the slow-ramp and fast-ramp curves.

1573 308

2. DATA BASE

The ballooning and rupture behavior of Zircaloy are fairly complex phenomena in part because (a) the stresses are biaxial and the material is anisotropic in the temperature range of most interest, (b) the properties of zirconium-base alloys are susceptible to heating-rate effects, (c) oxygen embrittlement increases yield and failure strengths, and (d) the cracking of oxide coatings results in failure sites that can localize stresses. Consequently the behavior of Zircaloy depends strongly on the cladding's environment and hence on test conditions (Refs x-y). Therefore, for final calibration of the data correlations, we have selected only those data from experiments in aqueous atmospheres that utilized either internal fuel-pellet simulators (i.e., indirect cladding heaters) or actual fuel pellets in reactor. This selection emphasizes the more recent and more expensive prototypical test data and deemphasizes much of the earlier data. Appendix A provides a tabulation of all of the data we have used, their references, and a legend of symbols that are used for these selected data sets in the later figures.

There are holes in this data base, however, particularly with regard to the absence of large bundle tests, and we have utilized the results from simpler less typical tests to bridge the gaps. These more pristine tests are atypical in a sense, but they do reveal fundamental features of Zircaloy behavior that allow one to interpret the sparser prototypical data.

3. NEW CORRELATIONS

3.1 Rupture Temperature

The incidence of rupture depends on the differential pressure across the cladding wall, the cladding temperature, and on the length of time those conditions are maintained. Time duration under burst conditions manifests itself as a heating-ramp-rate effect, and this effect will be treated explicitly. We have converted differential pressures to hoop stresses to eliminate design-specific dimensional effects. The conversion was made using the thin-shell formula,

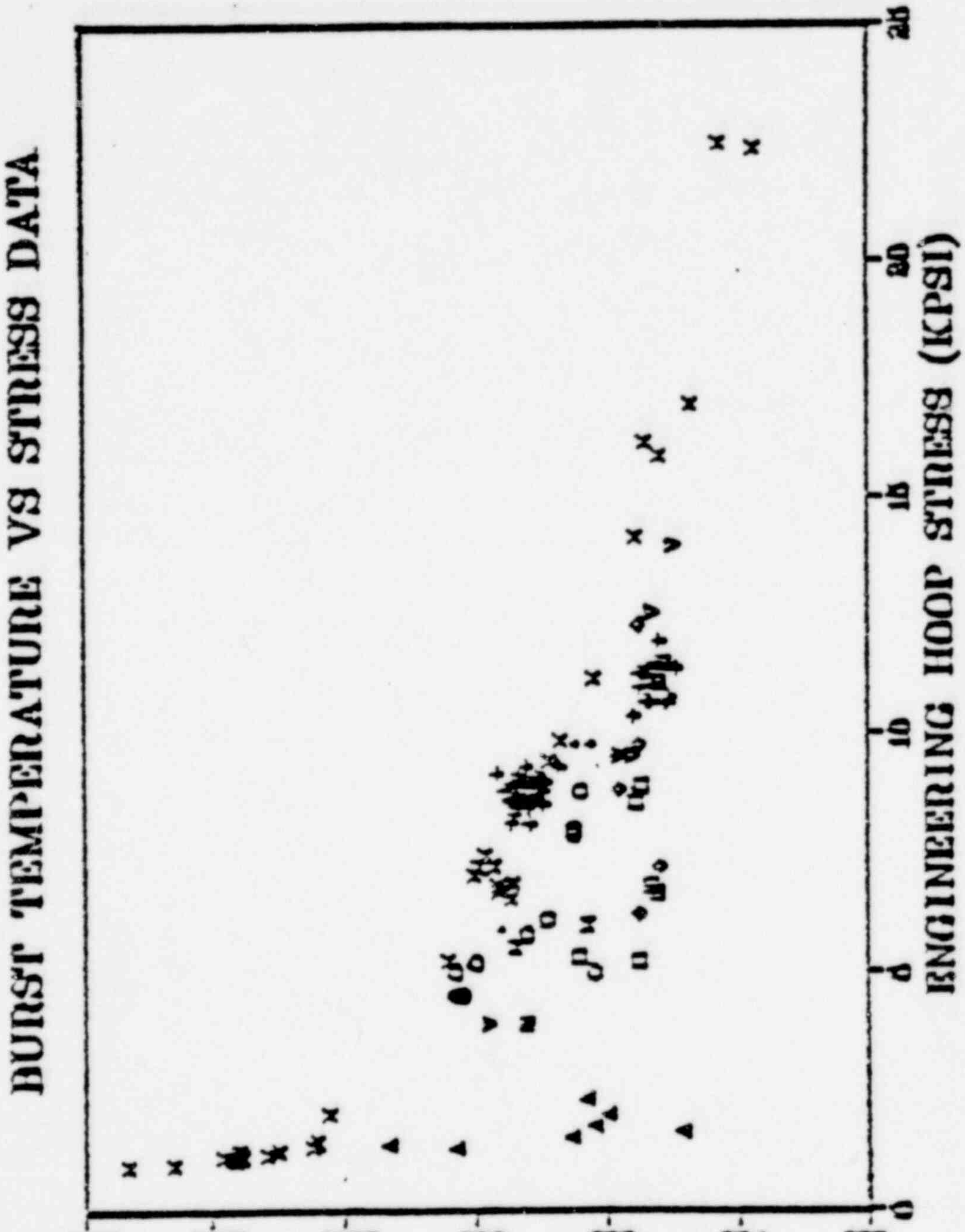
$$\sigma = (d/2t)\Delta P,$$

where σ is cladding hoop stress, d is the undeformed cladding mid-wall diameter, t is the undeformed cladding thickness, and ΔP is differential pressure across the cladding wall at rupture. Table 1 shows some computed values of hoop stress in terms of differential pressure for common commercial fuel designs.

Figure 1 shows rupture temperature data as a function of hoop stress for a wide range of test conditions. While this figure shows the general trend -- tubes burst at lower temperature when the pressure differential is higher -- the data are scattered primarily because of ramp-rate effects and experimental uncertainities in determining burst temperature.

TABLE 1

VENDOR FUEL DESIGN		HOOP STRESS (PSI)			
		100 DPSI	200 DPSI	1000 DPSI	2000 DPSI
B+W	15x15				
	17x17				
C-E	14x14				
	15x15				
	16x16				
W	14x14				
	15x15				
	17x17				
GE	7x7				
	8x8				
	8x8 R				
ENC	C-E 14x14				
	C-E 15x15				
	W 14x14				
	W 15x15				
	GE 7x7				
	GE 8x8				
	GE 8x8R				



POOR ORIGINAL

1573 312

Figure 2 shows ORNL data at 28°C/sec (a common ramp rate used in the ORNL experiments) and the basic correlation we will adopt as developed by Chapman (Ref. Q) using numerical regression techniques. It is clear that most of the data scatter has been eliminated by restricting the data to a single ramp rate. Chapman has also developed a ramp-rate correlation (Ref. N) that can be used with the basic rupture-temperature correlation in Fig. 2 to produce a family of rupture-temperature curves. Ramp-rate has little effect on rupture temperature for rates faster than 28°C/sec.

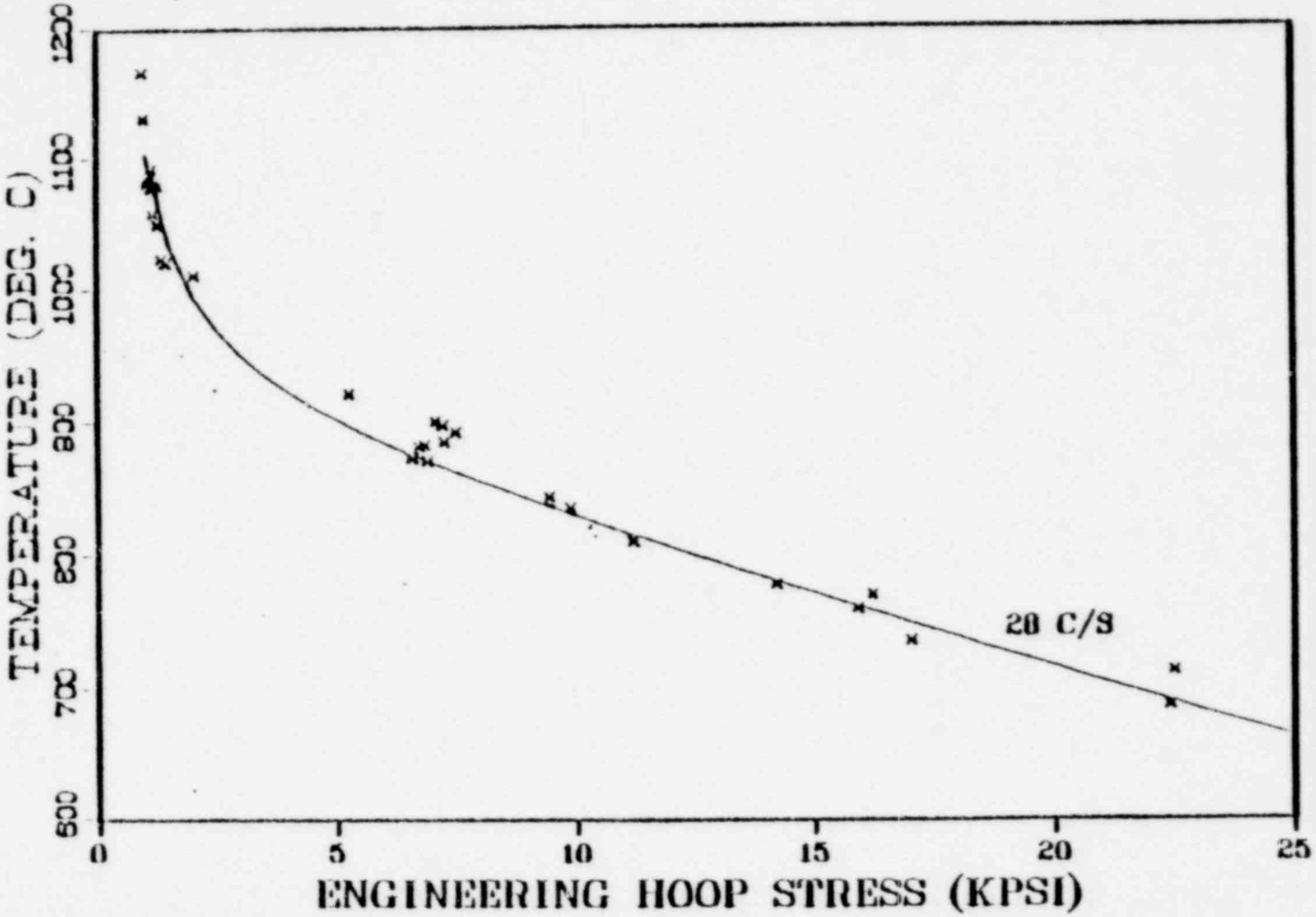
Three curves that span the important ramp-rate range are shown in Fig. 3 along with the data of Fig. 1. Chapman has shown that most of the original scatter is explained by ramp-rate effects, and the curves in Fig. 3 are seen to span most of the data. The up-facing triangles still deviate from the correlations and the major body of data. Difficulties in temperature measurement for these TREAT in-reactor data (Ref. X) are believed to be responsible for this deviation, and such discrepancies will be seen in later displays as well.

3.2 Burst Strain

Deformation (burst strain) at the location of a rupture depends on temperature, differential pressure (which is related to temperature by the correlation in Fig. 3), ramp rate, and several other variables such as local temperature variations. These effects have been discussed previously (Refs. x-y). Figure 4 shows burst strain as a

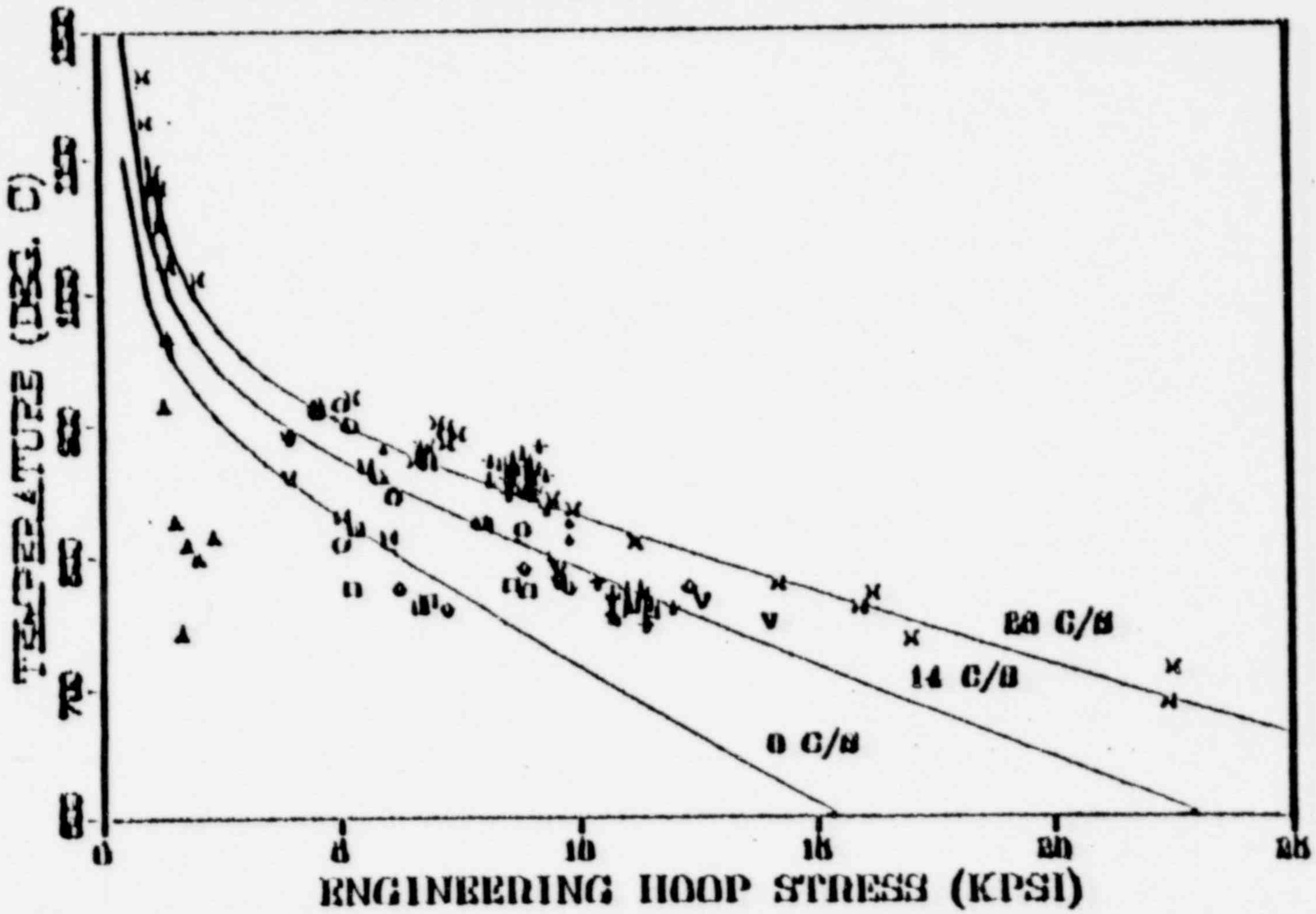
1573 313

28 C/S BURST TEMPERATURE CURVE & ORNL DATA



1573 314

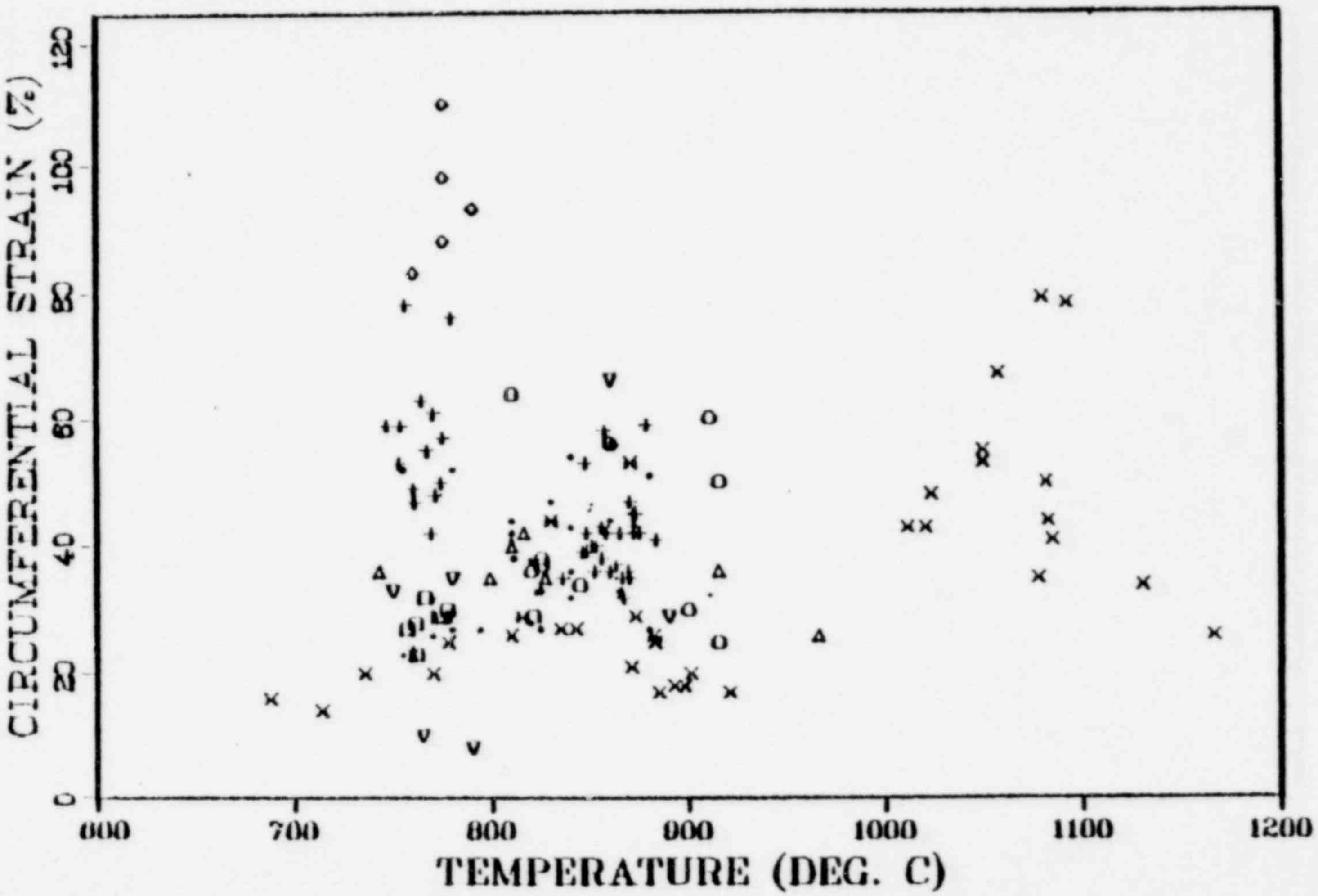
BURST TEMPERATURE CURVES & ALL DATA



POOR ORIGINAL

1573 315

BURST STRAIN VS TEMPERATURE DATA



1573 316

function of one of these variables, burst temperature, and the data scatter is therefore due to temperature measurement difficulties and the other variables mentioned above.

The scatter in Fig. 4 is bewildering, so we have relied on data from less prototypical but more controlled tests to help derive a correlation. Figure 5 shows burst strain versus burst temperature from Chung and Kassner's work (Ref. T) with short Zircaloy tubes heated by passing an electrical current directly through the Zircaloy. Several fundamental features are apparent. There are three superplastic peaks -- one in the low temperature alpha phase around 800°C and two in the high-temperature beta phase around 1050°C and 1225°C. The very important valley at about 925°C is a consequence of mixed alpha-plus-beta-phase material, which exhibits low ductility. Heating-rate effects are also visable; slow-ramp rates produce large strains in the temperature regime below about 950°C as a result of feedback effects discussed in Refs. x-y. But slow-ramp rates produce very small strains at temperatures greater than about 950°C because the Zircaloy has time to oxidize and embrittle before significant ballooning can occur. Fast-ramp rates produce the opposite effects in both temperature regimes.

To derive the slow-ramp correlation, which is shown in Fig. 6, we have thus taken Chung and Kassner's 5°C/sec curve and scaled the peaks and valleys to pass through the more prototypical data in our data base. The alpha-phase peak at 775°C was assigned the value

1573 317

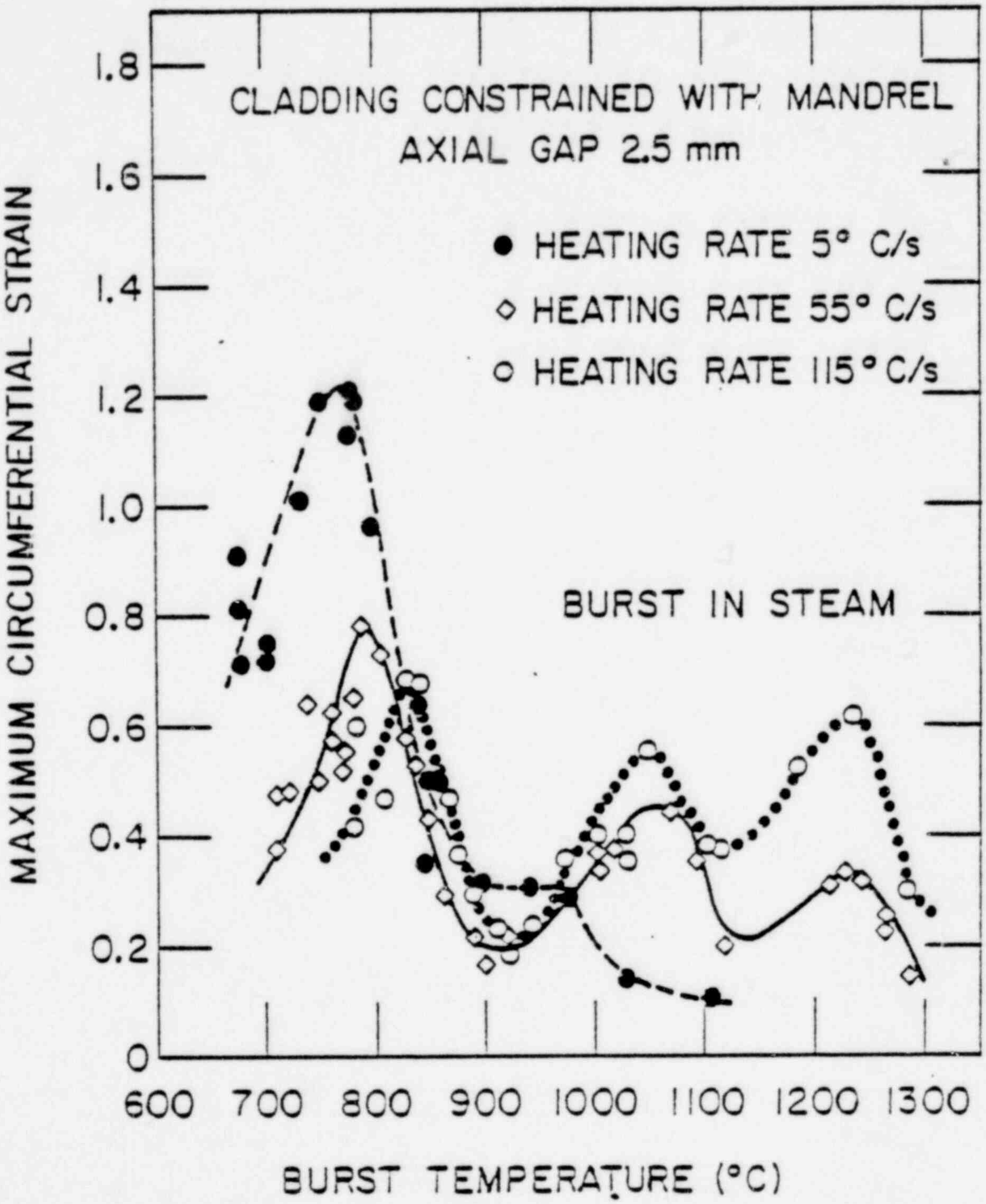
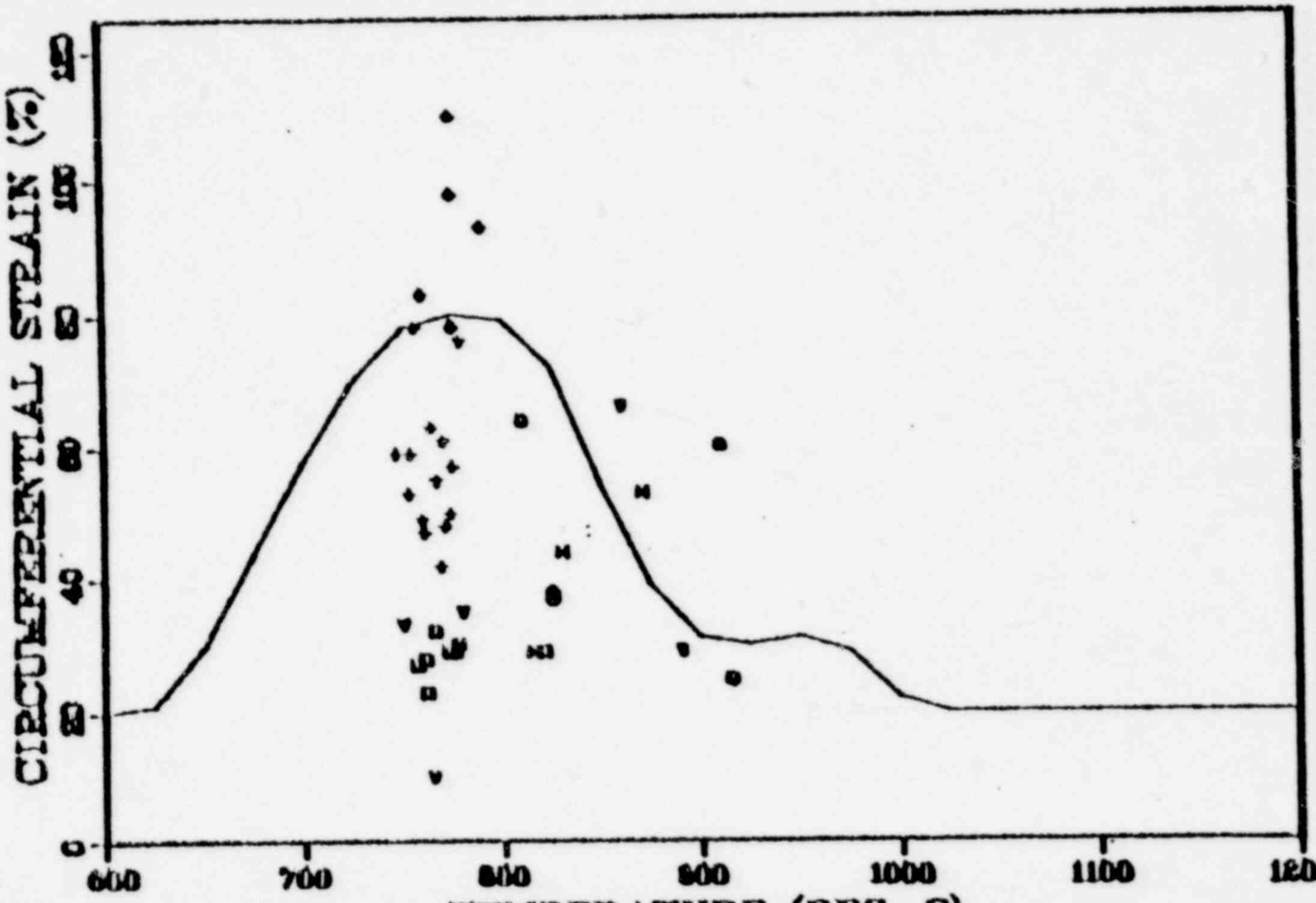


Fig. 5

1573 318

SLOW-RAMP BURST STRAIN CURVE & DATA

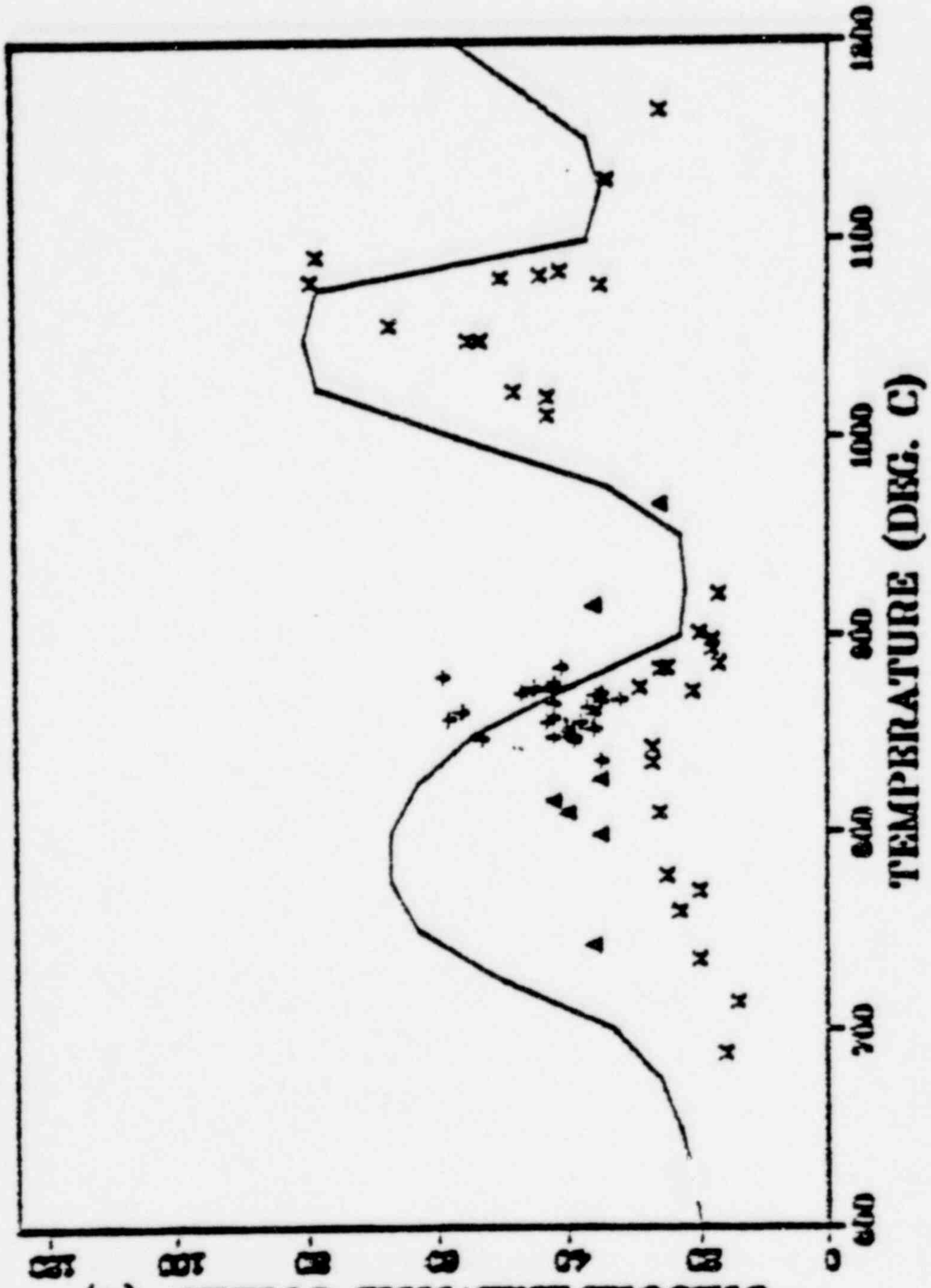


1573 319

of 80% in order to bound Chapman's 10°C/sec bundle test. The five highest points in Fig. 6 (0-10°C/sec heated-shroud single-rod tests) are preliminary and have not been fully evaluated, but they were disregarded because the heater power was so low (about 3W) that the tubes were in effect burst in a muffle furnace (the heated shrouds). Direct or external heating methods are known to exaggerate rupture strains by maintaining artificially small local temperature variations (see Ref. X), and such experiments were excluded from our data base. Since the majority of the data is bounded by the curve, we believe that the correlation satisfies the intention of Appendix K not to underestimate the degree of swelling. It should be cautioned that some very recent, unevaluated data from Germany (Ref. X) also show large strains (up to 120%), so the potential exists that Fig. 6 may have to be revised upward.

The fast-ramp correlation is shown in Fig. 7. In this case, there are no data from prototypical bundle tests and limited single-rod tests with heated shrouds and uniform heaters in the area of the low-temperature peak. The correlation was obtained by scaling Chung and Kassner's 55°C/sec curve and adjusting the alpha-phase peak height in relation to the peak height in Fig. 6 according to the relation that a 28°C/sec peak would have (based on interpolation) in Chung and Kassner's curve (Fig. 5) to the 5°C/sec peak in Fig. 5.* When prototypical bundle tests and heated-shroud tests are performed in the future, we expect the data to fall near the curve in Fig. 7.

*Consideration is being given to adjusting these curve peak locations to higher temperatures -- to around 825°C.



FAST-RAMP BURST STRAIN CURVE & DATA

POOR ORIGINAL

1573 321

Figure 8 shows the composite (i.e., envelope) of the curves in Figs. 6 and 7 along with all of the data from Fig. 4. The composite curve gives a good representation of the data, providing that the causes of small strains (Ref. X) are kept in mind.

3.3 Assembly Flow Blockage

Very few measurements of bundle blockage have been made under prototypical conditions and the best attempts are shown in Fig. 9. It is therefore necessary to derive bundle blockage from single-rod burst strains, but this is not straight forward since test results have shown that ruptures in a bundle are not coplanar.

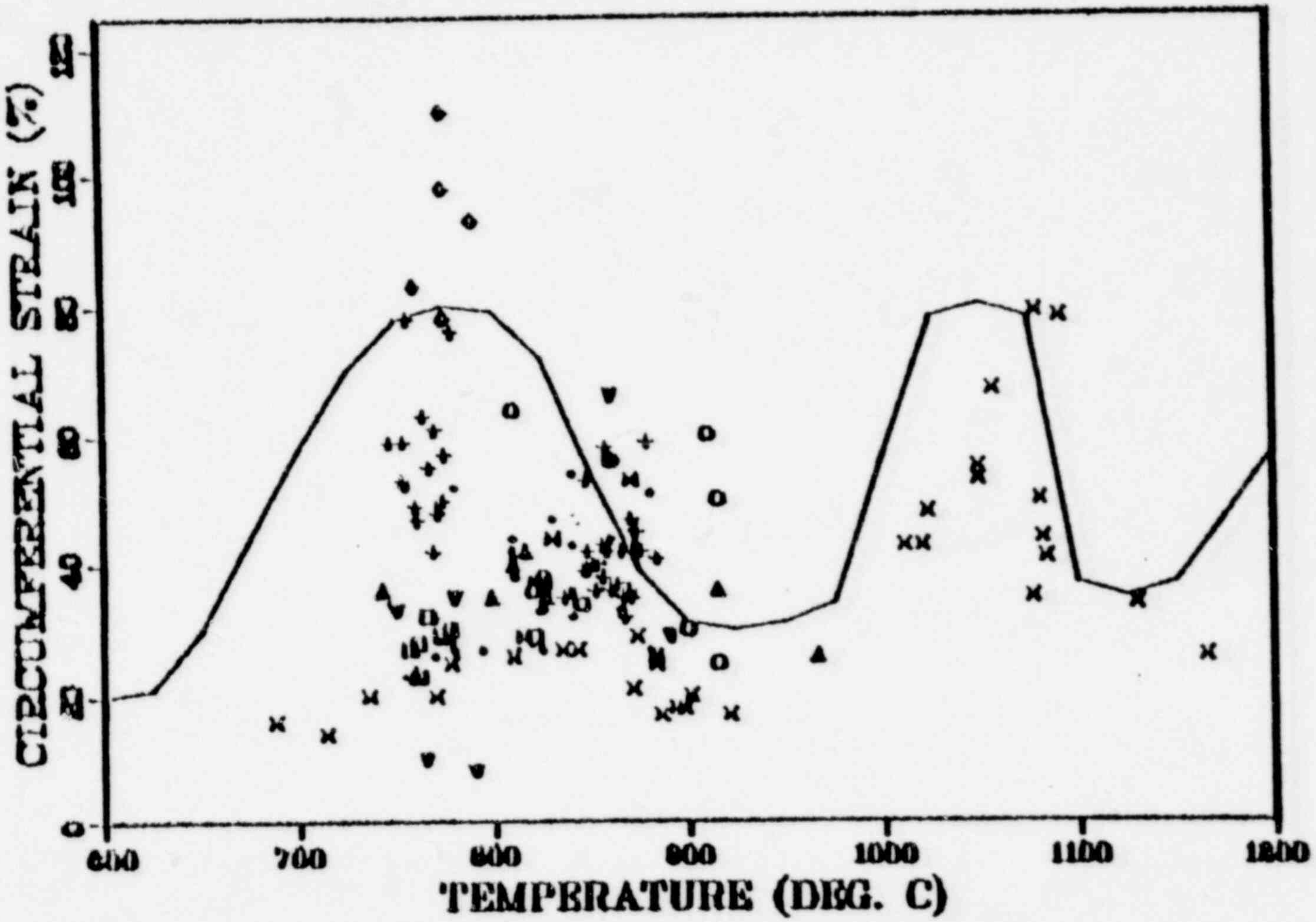
Figure 10 is a cross section from Chapman's first bundle test (Ref. X). Notice that only a few of the rods have burst in this plane. We have chosen the most realistic (minimum flow restriction) of Chapman's definitions of blockage for the following analysis. Figure 11 shows the axial distribution of blockage for Bundle No. 1, from which the maximum blockage is seen to be 49%.

Figure 12 shows the geometric relation between average rod strain and bundle blockage for a square array of commercial-size tubes. From this figure it can be seen that an average rod strain of 27% would cause a bundle blockage of 49%. Since the average rupture strain for rods in Bundle No. 1 was 42% (see Appendix A), the blockage can be obtained from the rupture strain by multiplying by 0.64 (the ratio of 27 to 42) and utilizing Fig. 12. The similar ratios for Bundles No. 2 and No. 3 are 0.67 and 0.70 giving an overall average for the three bundle tests of 0.67.

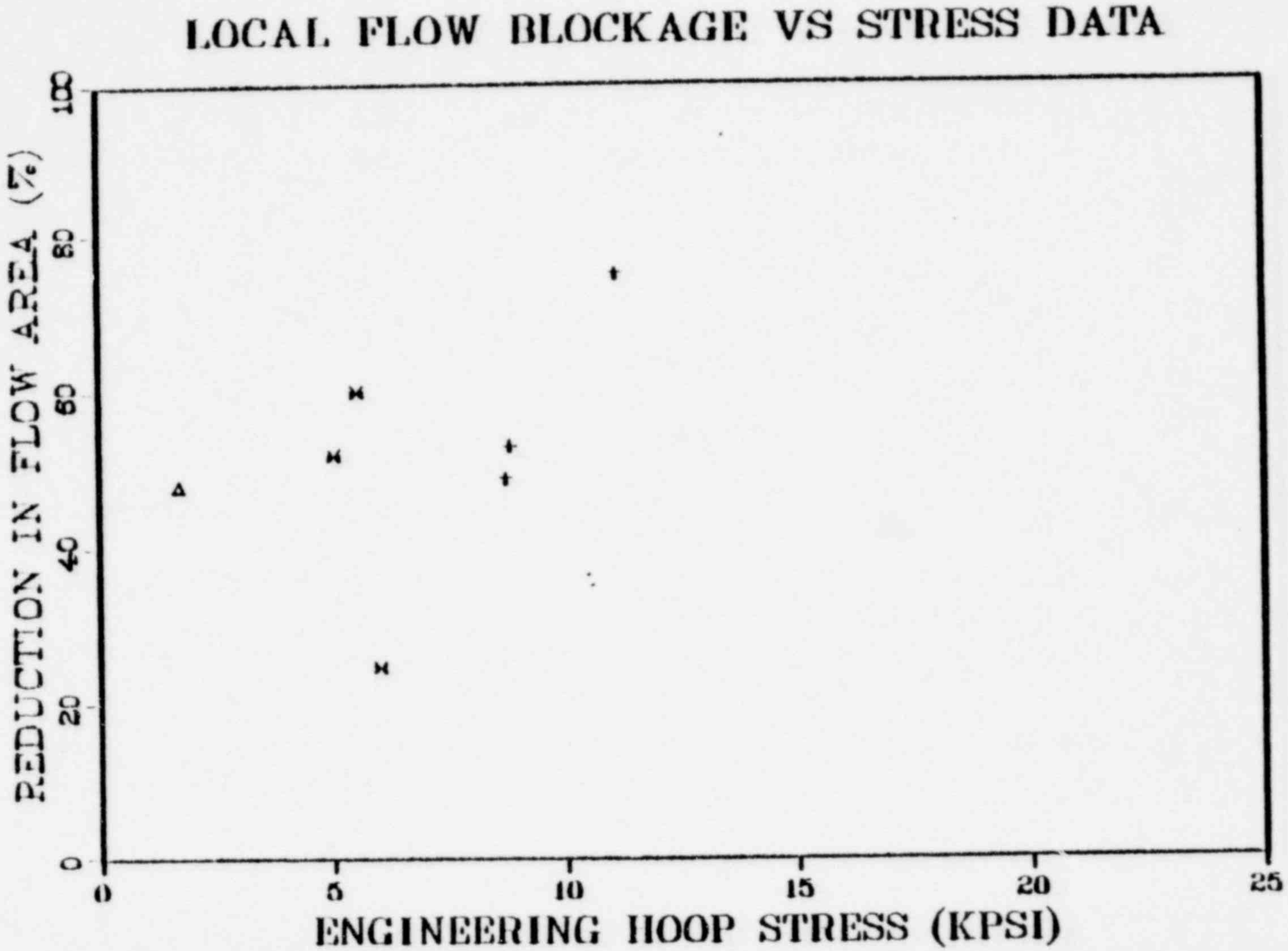
1573 322

POOR ORIGINAL

COMPOSITE BURST STRAIN CURVE & DATA



1573 323


LOCAL FLOW BLOCKAGE VS STRESS DATA
REDUCTION IN FLOW AREA (%)
0
20
40
60
80
100
0
5
10
15
20
25
ENGINEERING HOOP STRESS (KPSI)

ORNL-DWG 78-11137

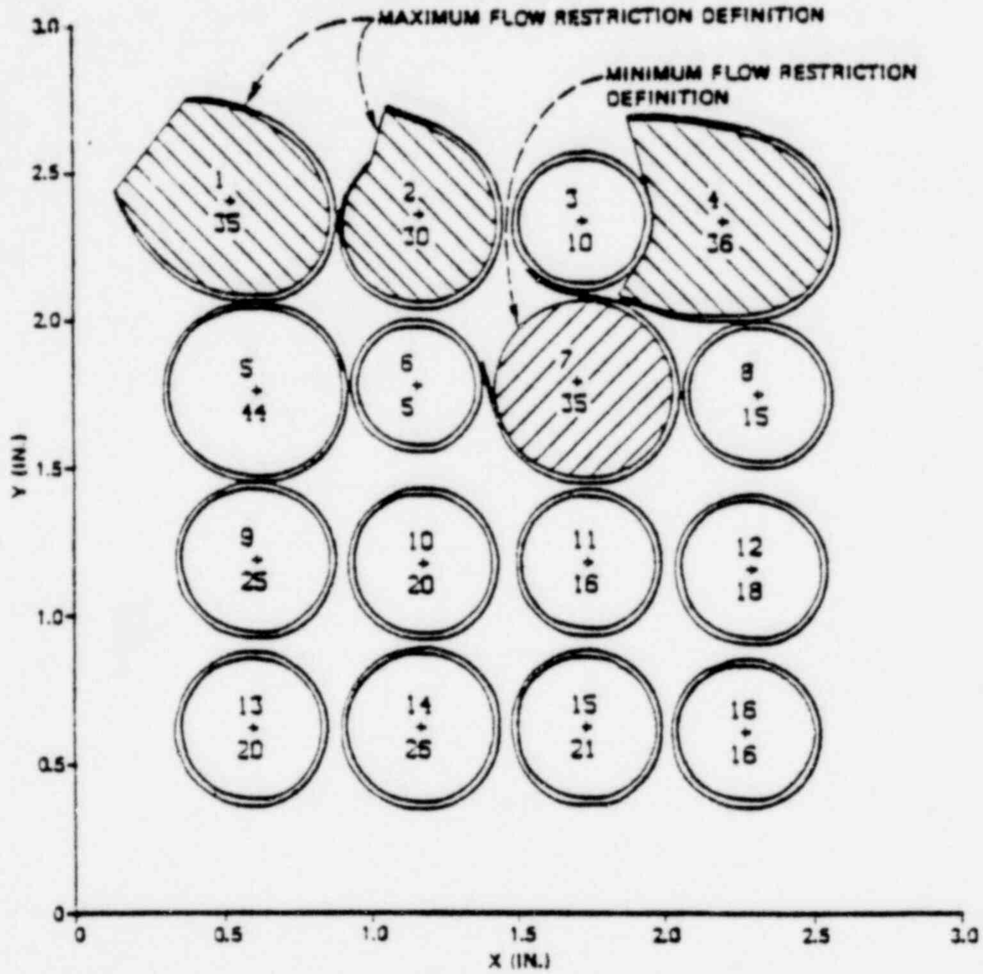


Fig. 10 Computer simulation of B-1 section of 76.5-cm elevation showing maximum and minimum flow restriction definitions.

1573 325

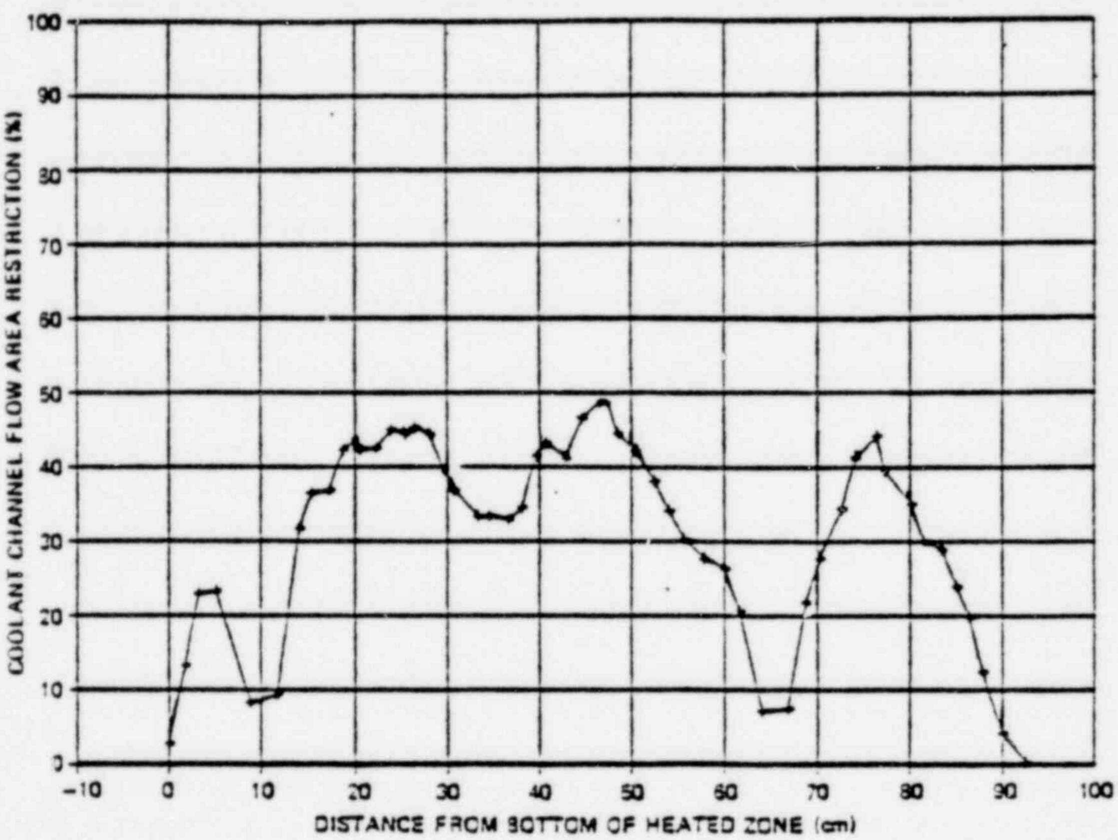
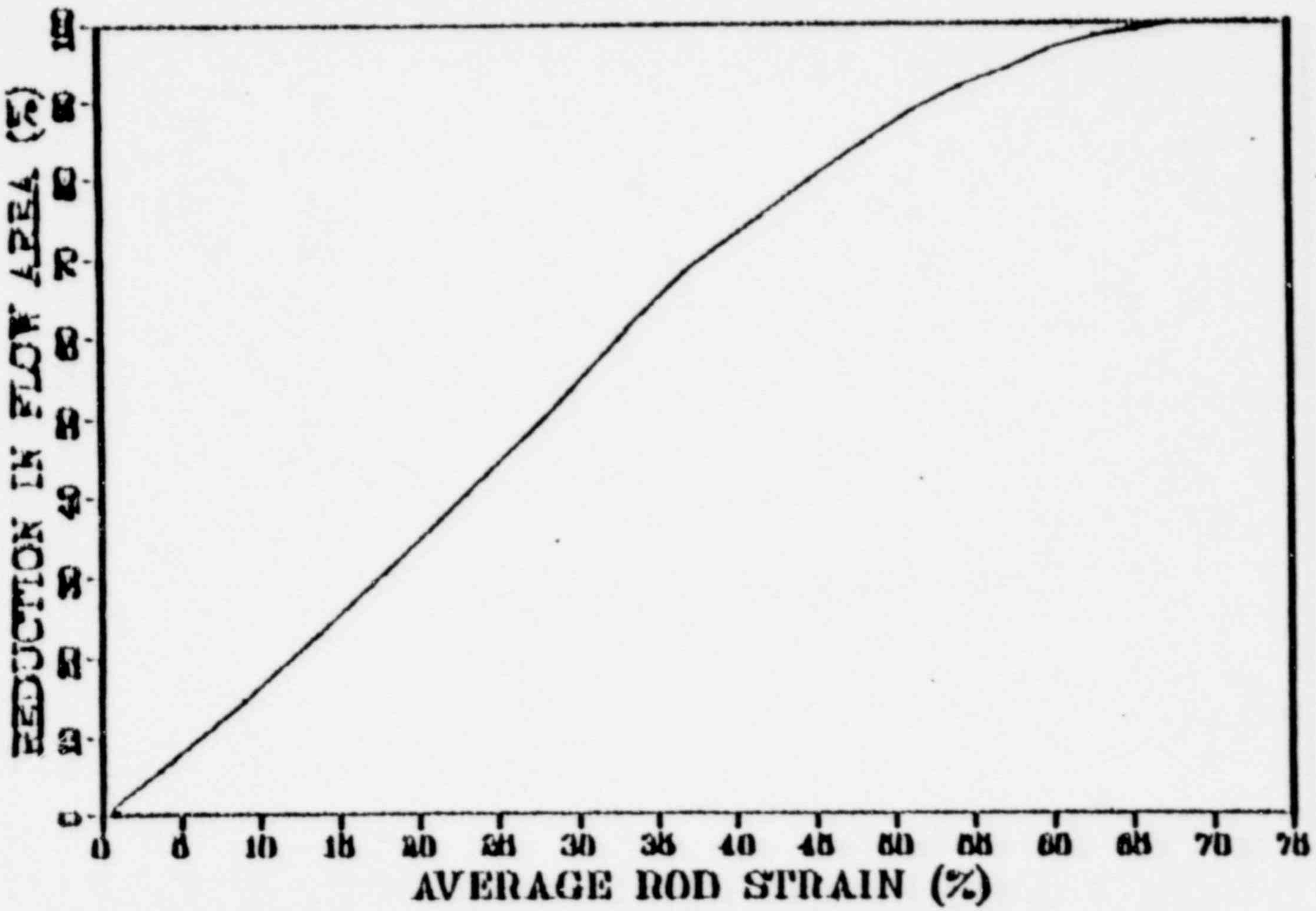


Fig.11. Coolant channel flow area restriction as a function of elevation in Chapman's Bundle No. 1.

1573 326

LOCAL FLOW BLOCKAGE VS STRAIN



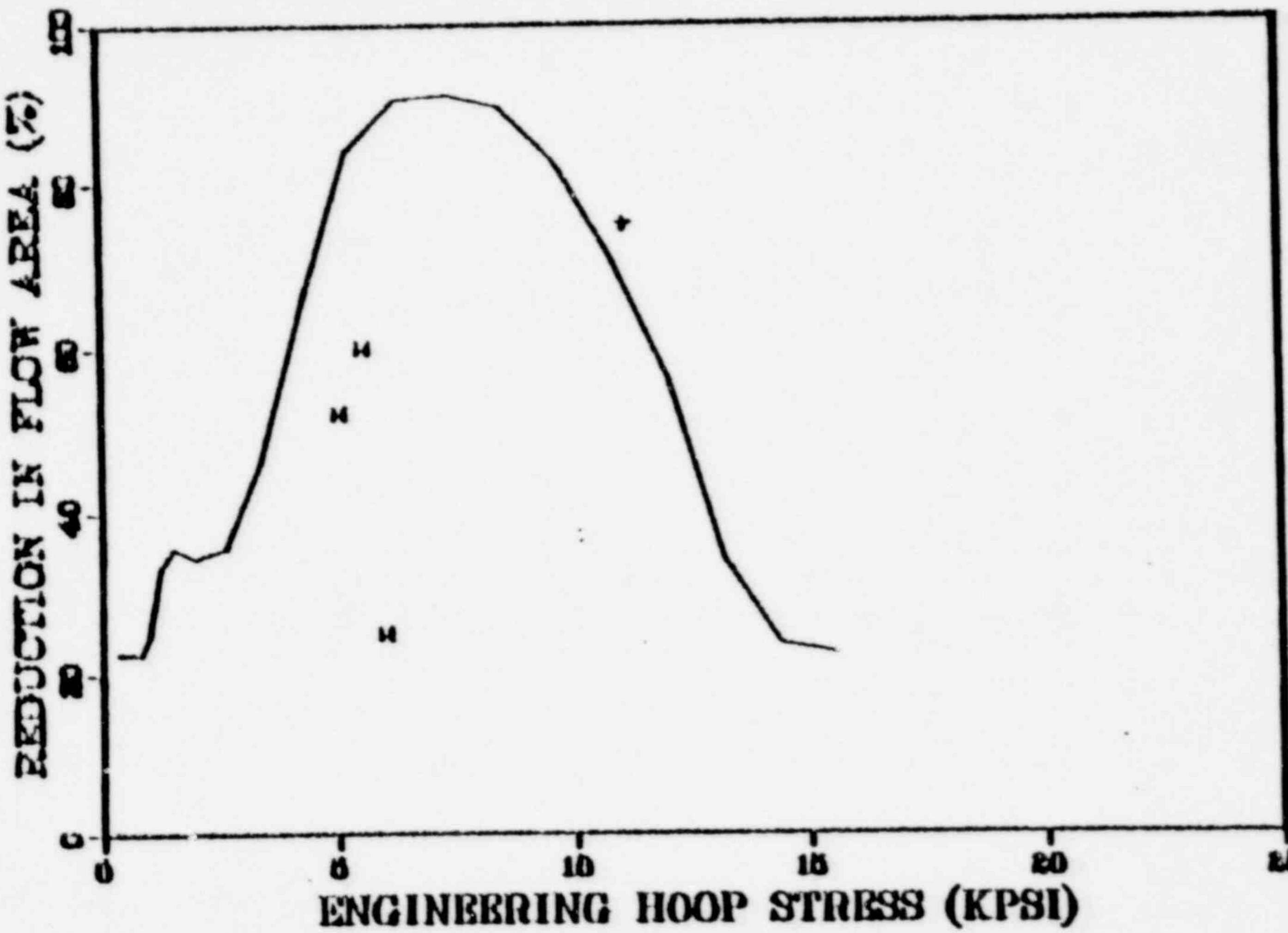
Assuming that the distributions of ruptures in Chapman's bundle tests are typical, the local blockage correlation is thus formed by multiplying strains in Figs. 6 and 7 by 0.67 and then utilizing Fig. 12. We have called this result "local blockage," as distinct from the desired assembly blockage, because it does not yet represent large commercial-size bundles or include the effects of non-fueled tubes, which would not balloon. The slow- and fast-ramp local blockage curves are shown in Figs. 13 and 14 where they are compared with the sparse collection of data. Figure 15 shows the composite flow blockage curve, which envelopes the curves in Figs. 13 and 14.

Finally, to obtain assembly flow blockage, two adjustments are required. First, it must be recognized that bundle-average blockage, which is desired, is a function of bundle size. This can be seen by envisioning an 8x8 test bundle that is analyzed quadrant by quadrant. If each 4x4 quadrant is viewed as a small bundle, the planes of maximum blockage for the quadrants would be expected to occur at different elevations because of some randomness of the process. One would therefore expect to find the plane of maximum blockage in each quadrant to have greater flow restriction than the plane of maximum blockage in the bundle taken as a whole. That is, the large bundle size introduces an averaging effect.

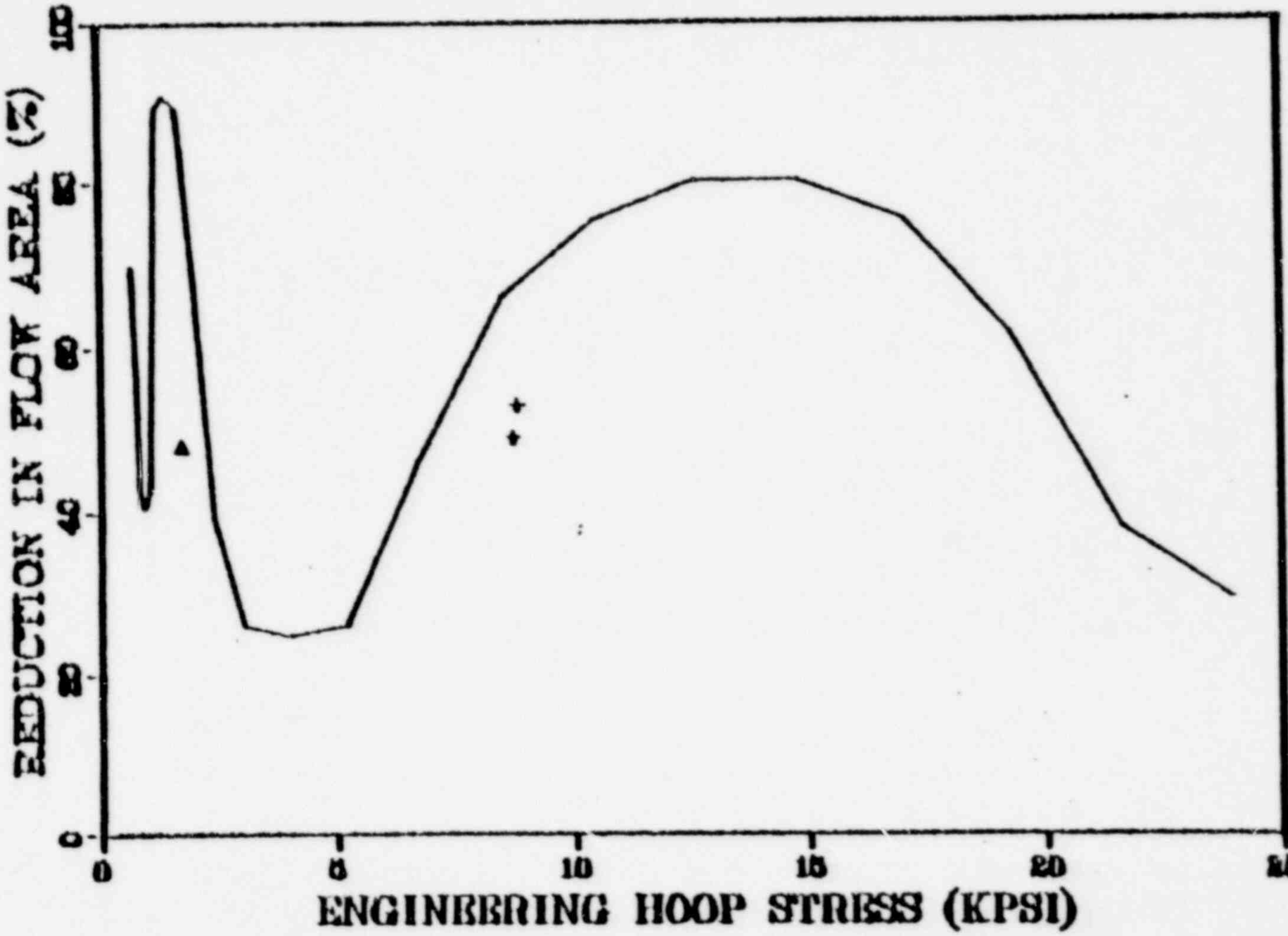
To account for this effect for commercial fuel bundles ranging from 7x7 (BWR) to 17x17 (PWR), we have used an average blockage from Chapman's bundle tests rather than the maximum value used in developing Figs. 13 - 15 (that process was appropriate for the

1573 328

SLOW-RAMP LOCAL FLOW BLOCKAGE & DATA

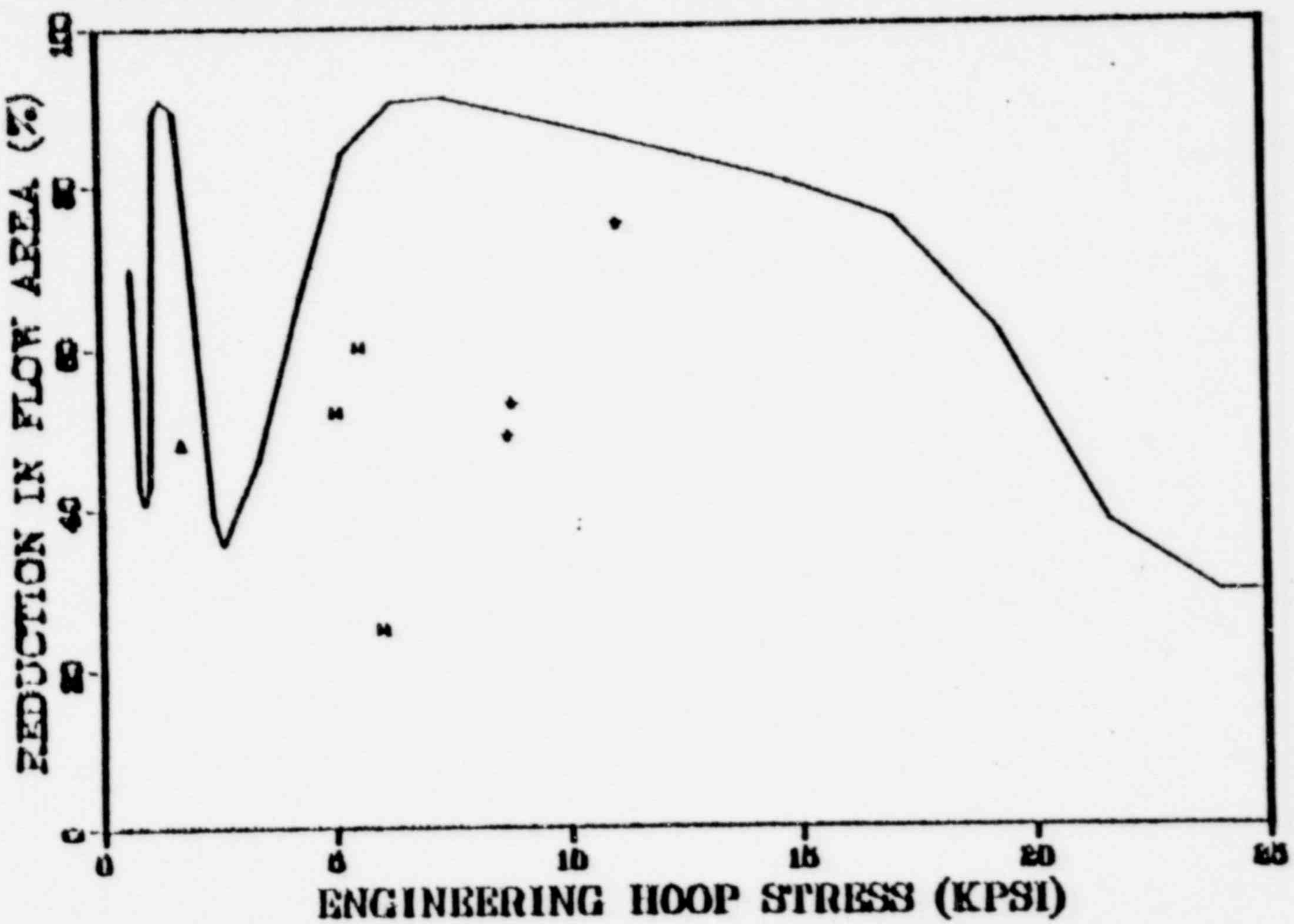


FAST-RAMP LOCAL FLOW BLOCKAGE & DATA



1573 330

COMPOSITE LOCAL FLOW BLOCKAGE CURVE & DATA



1573 331

data comparisons because the bundles represented in Figs. 13-15 were all small arrays). For Bundle No. 1, the average (41%) of the blockages was found between the 23-cm and 47-cm locations in an attempt to eliminate the suppressing effect of spacer grids at 10 cm and 65 cm. Similar averages were found for Bundles No. 2 and No. 3. Using these values the ratio to be used to derive large-bundle blockages from rupture strain data is 0.55 (compared with 0.67 for small arrays). This factor was used to derive all of the blockage curves in the next section of this report.

The second adjustment is a reduction of about 5% to account for instrument tubes and guidetubes that would not balloon. The exact scaling factor SF depends on the fuel design and is given by

$$SF = N_r A_r / (N_r A_r + N_g A_g),$$

where N_r is the number of fuel rods, A_r is the flow area around an undeformed fuel rod, N_g is the number of guidetubes or instrument tubes, and A_g is the flow area around an undeformed guidetube or instrument tube. This scaling factor was also employed in deriving the blockage curves in the next section.

1573 332

ENCLOSURE 4

APPENDIX A

FUEL CLADDING BURST DATA

DATA REFERENCE A (Upright Triangle)

FRF-1

R. A. Lorenz, D. O. Hobson, and G. W. Parker, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," Oak Ridge National Laboratory Report, ORNL-4635, March 1971. Available in public technical libraries. Also available from National Technical Information Service (NTIS), Springfield, Virginia 22161.

R. A. Lorenz, D. O Hobson, and G. W. Parker, "Fuel Rod Failure Under Loss-of-Coolant Conditions in TREAT," Nuclear Technology, II, p. 502 (August 1971). Available in public technical libraries.

Inpile, 7-rod bundle, steam atmosphere.

Maximum reduction in bundle flow area = 48 %.
Mean rod burst strain = 36 %.
Mean rod burst temperature ≐ 389°C.
Mean rod engineering burst stress ≐ 1.71 Kpsi.

ROD #	RAMP RATE (°C/S)	PRESSURE AT BURST (PSIG)	BURST TEMPERATURE (°C)	BURST STRAIN (%)	ENGINEERING BURST STRESS (KPSI)
H	25-36	172	966	26	1.39
4-1	25-36	250	799	35	2.02
R	25-36	205	743	36	1.66
4-2	25-36	290	816	42	2.34
L	25-36	162	915	36	1.31
I	25-36	190	827	35	1.54
C	25-26	215	810	40	1.74

DATA REFERENCE B (Cross)

R. H. Chapman, "Multirod Burst Test Program Progress Report for April-June 1977," Oak Ridge National Laboratory Report, ORNL/NUREG/TM-135, June 1977. Available in public technical libraries. Also available from National Technical Information Service (NTIS), Springfield, Virginia 22161.

R. H. Chapman, J. L. Crowley, A. W. Longest, and E. G. Sewell, "Effects of Creep Time and Heating Rate on Deformation of Zircaloy-4 Tubes Test in Steam with Internal Heaters," Oak Ridge National Laboratory Report, NUREG/CR-0343: ORNL/NUREG/TM-245, October 1978. Available in public technical libraries. Also available from National Technical Information Service (NTIS), Springfield, Virginia 22161.

Out-of-pile, single rod, steam atmosphere.

ROD #	RAMP RATE (°C/S)	PRESSURE AT BURST (PSIG)	BURST TEMPERATURE (°C)	BURST STRAIN (%)	ENGINEERING BURST STRESS (KPSI)
PS-1	28	922	893	18	7.47
PS-3	28	809	873	29	6.56
PS-4	28	850	871	21	6.88
PS-5	28	830	882	26	6.72
PS-10	28	870	901	20	7.05
PS-12	28	891	898	18	7.21
PS-14	28	844	883	25	6.84
PS-15	28	893	885	17	7.24
PS-17	28	1760	778	25	14.2
SR-1	28	116	1166	26	0.94
SR-2	28	146	1082	44	1.19
SR-3	28	249	1011	43	2.02
SR-4	28	650	921	17	5.26
SR-5	28	1380	810	26	11.2
SR-7	28	2090	736	20	17.0
SR-8	28	178	1020	43	1.44
SR-13	28	155	1079	79	1.26
SR-15	28	2780	714	14	22.5
SR-17	28	154	1049	53	1.25
SR-19	28	2760	688	16	22.4
SR-20	28	154	1049	55	1.25
SR-21	28	162	1023	48	1.32
SR-22	28	129	1081	50	1.05
SR-23	28	139	1077	35	1.13
SR-24	28	144	1057	67	1.16
SR-25	28	139	1092	78	1.13
SR-26	28	120	1130	34	0.98
SR-27	28	133	1084	41	1.08
SR-28	28	1220	835	27	9.87
SR-29	28	1170	843	27	9.45
SR-37	28	1967	760	23	15.9
SR-38	28	1998	770	20	16.2

DATA REFERENCE C (Plus)

MRBT-B-1

R. H. Chapman, "Multirod Burst Test Program Progress Report for July-December 1977," Oak Ridge National Laboratory Report, NUREG/CR-0103: ORNL/NUREG/TM-200, June 1978. Available in public technical libraries. Also available from National Technical Information Service (NTIS), Springfield, Virginia 22161.

R. H. Chapman, "Preliminary Multirod Burst Test Program Results and Implications of Interest to Reactor Safety Evaluation," paper presented at the 6th NRC Water Reactor Safety Research Information Meeting, Gaithersburg, MD., November 7, 1978. Available in PDR for inspection and copying for a fee.

Out-of-pile, 16-rod bundle, steam atmosphere.

Maximum reduction in bundle flow area = 49 %.
Mean rod burst strain = 42 %.
Mean rod strain in plane of maximum blockage = 27 %.
Mean rod burst temperature = ~~[illegible]°C~~ 868
Mean rod engineering burst stress = 8.72 Kpsi.

ROD #	RAMP RATE (°C/S)	PRESSURE AT BURST (PSIG)	BURST TEMPERATURE (°C)	BURST STRAIN (%)	ENGINEERING BURST STRESS (KPSI)
1	29	1124	852	36	9.10
2	29	1075	867	32	8.71
3	29	----	---	--	----
4	29	1052	860	36	9.33
5	29	1005	872	45	8.14
6	29	1104	872	43	8.94
7	29	1052	869	36	8.52
8	29	1074	872	42	8.70
9	29	1030	870	47	8.34
10	29	1059	873	45	8.58
11	29	1054	847	53	8.54
12	29	1114	863	37	9.02
13	29	1091	878	59	8.84
14	29	1066	875	42	8.63
15	29	1062	865	42	8.60
16	29	1092	848	39	8.85

1573 335

DATA REFERENCE C (Plus)

MRBT-B-2

R. H. Chapman, "Multirod Burst Test Program Progress Report for July-December 1977," Oak Ridge National Laboratory Report, NUREG/CR-0103: ORNL/NUREG/TM-200, June 1978. Available in public technical libraries. Also available from National Technical Information Service (NTIS), Springfield, Virginia 22161.

R. H. Chapman, "Multirod Burst Test Program Progress Report for July-December 1978," Oak Ridge National Laboratory Report, NUREG/CR-0655: ORNL/NUREG/TM-297, June 1979. Available in public technical libraries. Also available from National Technical Information Service (NTIS), Springfield, Virginia 22161.

Out-of-pile, 16-rod bundle, steam atmosphere.

Maximum reduction in bundle flow area = 53 %.
Mean rod burst strain = 42 %.
Mean rod strain in plane of maximum blockage = 28 %.
Mean rod burst temperature = 858°C.
Mean rod engineering burst stress = 8.88 Kpsi.

ROD #	RAMP RATE (°C/S)	PRESSURE AT BURST (PSIG)	BURST TEMPERATURE (°C)	BURST STRAIN (%)	ENGINEERING BURST STRESS (KPSI)
1	29	1117	870	35	9.05
2	29	1115	846	39	9.02
3	29	1096	853	40	8.88
4	29	1100	872	42	8.91
5	29	1127	866	35	9.13
6	29	1004	857	58	8.13
7	29	1067	861	56	8.64
8	29	1097	856	38	8.89
9	29	----	---	--	----
10	29	1065	856	43	8.63
11	29	1112	853	40	9.01
12	29	1094	851	40	8.86
13	29	1134	883	41	9.19
14	29	1048	858	42	8.49
15	29	1152	836	35	9.33
16	29	1117	848	42	9.05

DATA REFERENCE C (Plus)

MRBT-B-3

R. H. Chapman, "Preliminary Multirod Burst Test Program Results and Implications of Interest to Reactor Safety Evaluation," paper presented at the 6th NRC Water Reactor Safety Research Information Meeting, Gaithersburg, MD., November 7, 1978. Available in PDR for inspection and copying for a fee.

R. H. Chapman, "Multirod Burst Test Program Progress Report for April-June, 1979," Oak Ridge National Laboratory Report, NUREG/CR-1023: ORNL/NUREG/TM-351, in publication.

Out-of-pile, 16-rod bundle, steam atmosphere.

Maximum reduction in bundle flow area = 75 %.
Mean rod burst strain = 57 %.
Mean rod strain in plane of maximum blockage = 40 %.
Mean rod burst temperature = 764°C.
Mean rod engineering burst stress ≐ 11.07 Kpsi.

Rod #	RAMP RATE (°C/S)	PRESSURE AT BURST (PSIG)	BURST TEMPERATURE (°C)	BURST STRAIN (%)	ENGINEERING BURST STRESS (KPSI)
1	10	1393	771	48	11.28
2	10	1280	779	76	10.39
3	10	----	---	--	-----
4	10	1318	767	55	10.68
5	10	1375	764	63	11.14
6	10	1327	770	61	10.75
7	10	----	---	--	-----
8	10	1320	756	78	10.69
9	10	1320	754	59	10.69
10	10	1362	774	50	11.03
11	10	1396	775	57	11.31
12	10	1414	761	47	11.45
13	10	1486	760	49	12.04
14	10	1405	769	42	11.38
15	10	1335	753	53	10.81
16	10	1407	747	59	11.40

1573 337

DATA REFERENCE D (Closed Circle)

F. Erbacher, H. J. Neitzel, and K. Wiehr, "Interaction Between Thermohydraulics and Fuel Clad Ballooning in a LOCA, Results of REBEKA Multirod Burst Tests with Flooding," paper presented at the 6th NRC Water Reactor Safety Research Information Meeting, Gaithersburg, MD, November 7, 1978. Available in file for USNRC Report, NUREG-0536.

F. Erbacher, H. J. Neitzel, M. Reimann, and K. Wiehr, "Fuel Rod Behavior in the Refilling and Reflooding Phase of a LOCA-Burst Test with Indirectly Heated Fuel Rod Simulators," paper presented at the NRC Zircaloy Cladding Review Group Meeting, Idaho Falls, May 23, 1977. Available in file for USNRC Report, NUREG-0536.

K. Wiehr and H. Schmidt, "Out-of-Pile Experiments on Ballooning of Zircaloy Fuel Rod Claddings Test Results with Shortened Fuel Rod Simulators," Kernforschungszentrum Karlsruhe Report, KfK 2345, October 1977. Available in file for USNRC Report, NUREG-0536.

F. Erbacher, H. J. Neitzel, M. Reimann, and K. Wiehr, "Out-of-Pile Experiments on Ballooning in Zircaloy Fuel Rod Claddings in the Low Pressure Phase of a Loss-of-Coolant Accident," Proceedings of Specialists' Meeting on the Behavior of Water Reactor Fuel Elements Under Accident Conditions, Spatind, Norway, September 13-16, 1976. Available in public technical libraries.

F. Erbacher, H. J. Neitzel, and K. Wiehr, "Studies on Zircaloy Fuel Clad Ballooning in a LOCA, Results of Burst Tests with Indirectly Heated Fuel Rod Simulators," paper presented at the ASTM 4th International Conference on Zirconium in the Nuclear Industry, Stratford-on-Avon, England, June 27-29, 1978. Available from ASTM.

Out-of-pile, single rod, air and steam atmosphere.

ROD #	RAMP RATE (°C/S)	PRESSURE AT BURST (PSIG)	BURST TEMPERATURE (°C)	BURST STRAIN (%)	ENGINEERING BURST STRESS (KPSI)
?	11	?	880	27	?
?	11	856	880	51	5.91
?	11	?	865	33	?
?	11	?	860	44	?
?	11	?	840	32	?
?	11	?	840	36	?
?	11	?	840	43	?
?	11	?	840	54	?
?	11	?	830	47	?
?	11	?	825	27	?
?	11	?	825	33	?
18	11	1420	823	33	9.81
?	11	?	820	28	?
?	11	?	820	38	?
14	11	1420	810	38	9.81

DATA REFERENCE D (Continued)

ROD #	RAMP RATE (°C/S)	PRESSURE AT BURST (PSIG)	BURST TEMPERATURE (°C)	BURST STRAIN (%)	ENGINEERING BURST STRESS (KPSI)
?	11	?	810	42	?
?	11	?	810	44	?
35	11	1380	794	27	9.54
?	11	?	780	27	?
?	11	?	780	30	?
?	11	?	780	52	?
?	11	?	770	26	?
?	11	?	770	32	?
?	11	?	760	24	?
?	11	?	755	23	?
?	11	?	755	52	?

DATA REFERENCE E (Open Circle)

E. Karb, "In-Pile Experiments in the FR-2 DK-LOOP on Fuel Rod Behavior During a LOCA," paper presented at the US/FRG Workshop on Fuel Rod Behavior, Karlsruhe, June 1978. Available in file for USNRC Report, NUREG-0536.

E. H. Karb, "Results of the FR-2 Nuclear Tests on the Behavior of Zircaloy Clad Fuel Rods," paper presented at the 6th NRC Water Reactor Safety Research Information Meeting, Gaithersburg, MD, November 7, 1978. Available in file for USNRC Report, NUREG-0536.

E. H. Karb, "Results of FR-2 In-Pile Tests on LWR Fuel Rod Behavior," paper presented at the 4th JAERI-FRG-NRC Annual Fuel Behavior Information Exchange, Idaho Falls, Idaho, June 22-29, 1979. Available in PDR for inspection and copying for a fee.

Inpile, single rod, steam atmosphere.

ROD #	RAMP RATE (°C/S)	PRESSURE AT BURST (PSIG)	BURST TEMPERATURE (°C)	BURST STRAIN (%)	ENGINEERING BURST STRESS (KPSI)
A1.1	7.1	725	810	64	5.01
A2.1	20	1276	820	36	8.82
B1.6	8.2	1160	825	38	8.02
B3.1	10	1146	825	37	7.92
B1.3	12.7	885	845	34	6.12
A2.2	12.1	841	860	56	5.81
B1.1	17.5	754	900	30	5.21
B1.5	9	653	910	60	4.51
B1.2	8.7	653	915	25	4.51
B3.2	12.1	725	915	50	5.01

1573 340

DATA REFERENCE F (Square)

R. H. Chapman, J. L. Crowley, A. W. Longest, and E. G. Sewell, "Effects of Creep Time and Heating Rate on Deformation of Zircaloy-4 Tubes Tested in Steam with Internal Heaters," Oak Ridge National Laboratory Report, NUREG/CR-0343: ORNL/NUREG/TM-245, October 1978. Available in public technical libraries. Also available from National Technical Information Service (NTIS), Springfield, Virginia 22161.

Out-of-pile, single rod, steam atmosphere.

ROD #	RAMP RATE (°C/S)	PRESSURE AT BURST (PSIG)	BURST TEMPERATURE (°C)	BURST STRAIN (%)	ENGINEERING BURST STRESS (KPSI)
SR-33	0	825	762	23	6.68
SR-34	0	844	766	32	6.84
SR-35	0	648	775	29	5.25
SR-36	0	660	821	29	5.35
SR-43	4	1105	773	29	8.95
SR-44	5	1060	777	30	8.59
SR-41	9	1416	757	27	11.5
SR-42	10	1373	761	28	11.1

DATA REFERENCE G (Asterisk)

REBEKA-1, -2, -3

F. Erbacher, H. J. Neitzel, and K. Wiehr, "Interaction Between Thermohydraulic and Fuel Clad Ballooning in a LOCA, Results of REBEKA Multirod Burst Tests with Flooding," paper presented at the 6th NRC Water Reactor Safety Research Information Meeting, Gaithersburg, MD., November 7, 1978. Available in file for USNRC Report, NUREG-0536.

K. Wiehr, "Results of REBEKA Test 3," paper presented at the 4th JAERI-FRG-NRC Annual Fuel Behavior Information Exchange, Idaho Falls, Idaho, June 22-29, 1979. Available in PDR for inspection and copying for a fee.

Out-of-pile, 9-rod bundles, steam and water atmosphere.

TEST #	INITIAL RAMP RATE (°C/S)	MEAN PRESSURE AT BURST (PSIG)	MEAN BURST TEMPERATURE (°C)	MEAN BURST STRAIN (%)	MEAN ENGINEERING BURST STRESS (KPSI)	REDUCTION IN FLOW AREA (%)
1	7	870	815	29	6.01	25
2	7	800	870	53	5.53	60
3	7	725	830	44	5.05	52

DATA REFERENCE H (Inverted Triangle)

M. Bocek, "FABIOLA," paper presented at the 4th JAERI-FRG-NRC Annual Fuel Behavior Information Exchange, Idaho Fàlls, Idaho, June 22-29, 1979. Available in PDR for inspection and copying for a fee.

Out-of-pile, single rod, steam atmosphere.

ROD #	RAMP RATE (°C/S)	PRESSURE AT BURST (PSIG)	BURST TEMPERATURE (°C)	BURST STRAIN (%)	ENGINEERING BURST STRESS (KPSI)
1	3	563	860	66	3.92
4	11	1375	790	8	9.58
8	7.8	1375	780	35	9.58
10	10	2013	750	33	14.03
12	9	563	890	29	3.92
13	10	1810	765	10	12.62

1573 343

DATA REFERENCE I (Diamond)

J. L. Crowley (ORNL), personal communication to D. A. Powers (USNRC), August 10, 1979.

R. H. Chapman (ORNL), personal communication to D. A. Powers (USNRC), September 11, 1979.

Out-of-pile, single rod, heated shroud, steam atmosphere.

ROD #	RAMP RATE (°C/S)	PRESSURE AT BURST (PSIG)	BURST TEMPERATURE (°C)	MAXIMUM ROD STRAIN (%)	ENGINEERING BURST STRESS (KPSI)
SR-47	10	1436	775	78	12.35
SR-49	5	1139	775	98	9.80
SR-51	0	1030	790	93	8.86
SR-53	0	841	760	83	7.23
SR-57	0	725	775	110	6.23

1573 344

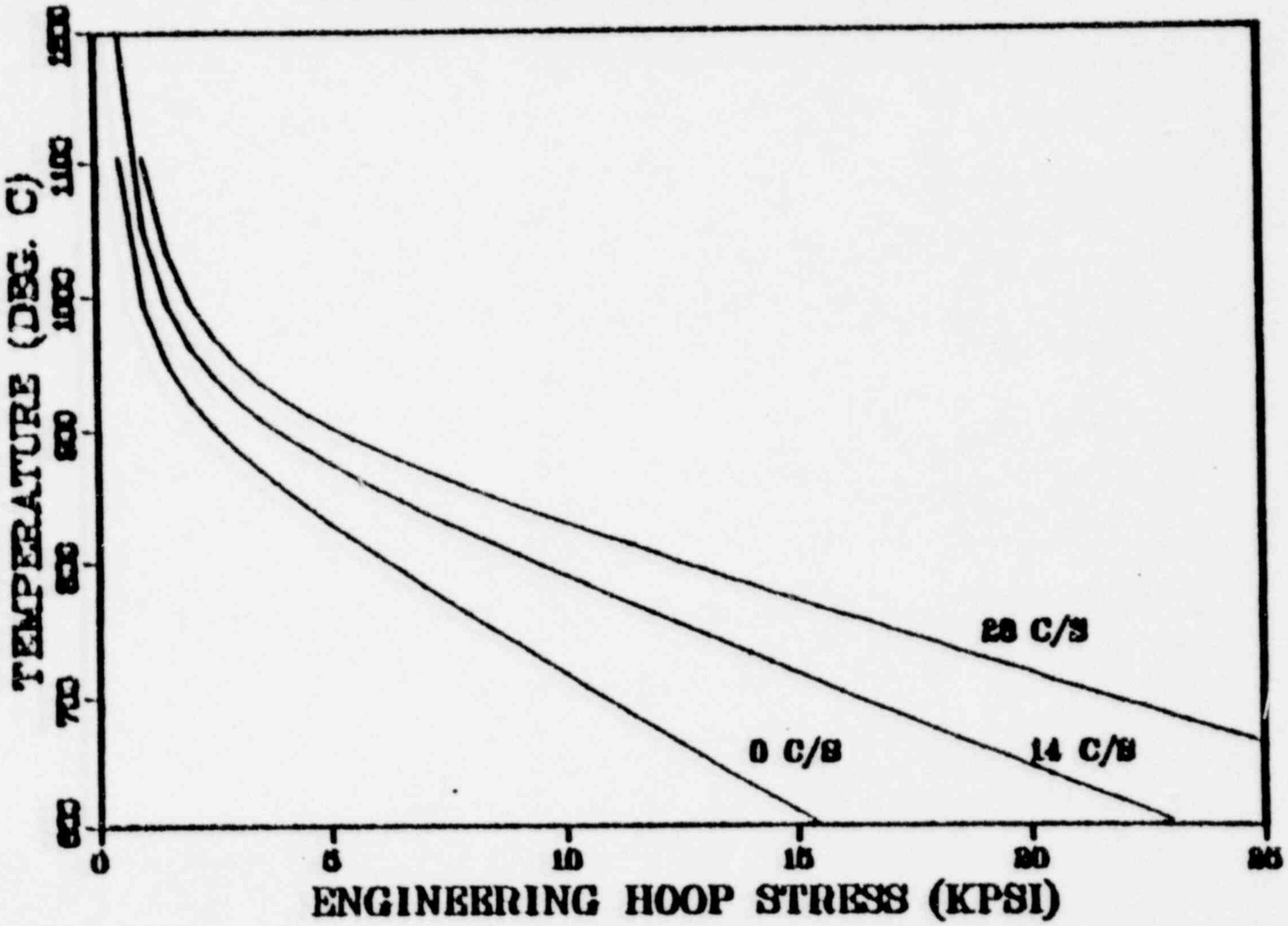
ENCLOSURE 5

Enclosed are 3 figures that show correlations in the 10/31/79 draft for rupture temperature, rupture strain, and assembly flow blockage. Temperature ramp rates are accounted for in the correlations, and the ramp rates that are most appropriate should be used. If it is not practical to accommodate ramp rates in the code, envelopes of these curves should be used. The tabular values from which these curves were generated are also enclosed.

1573 345

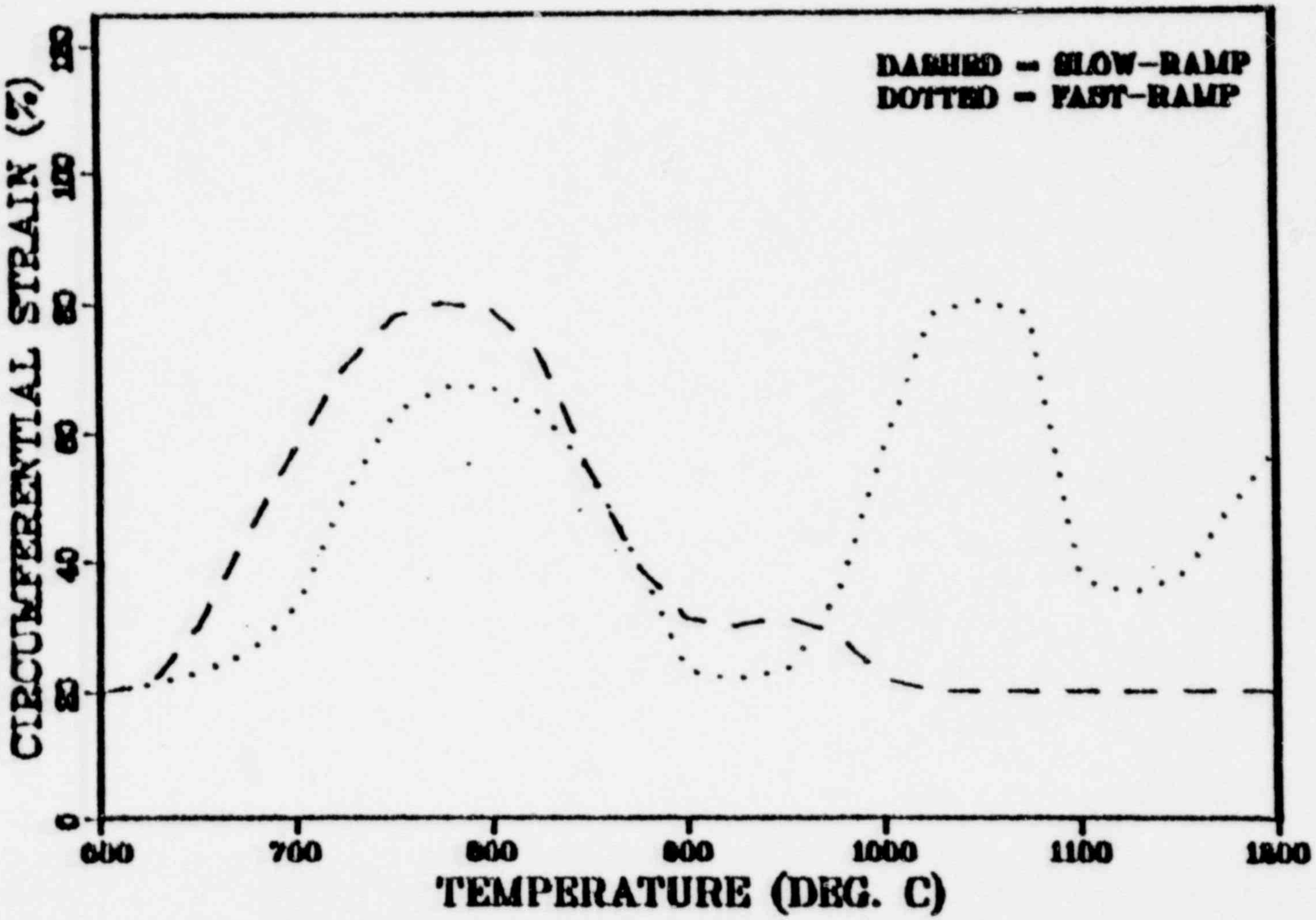
POOR ORIGINAL

BURST TEMPERATURE CURVES



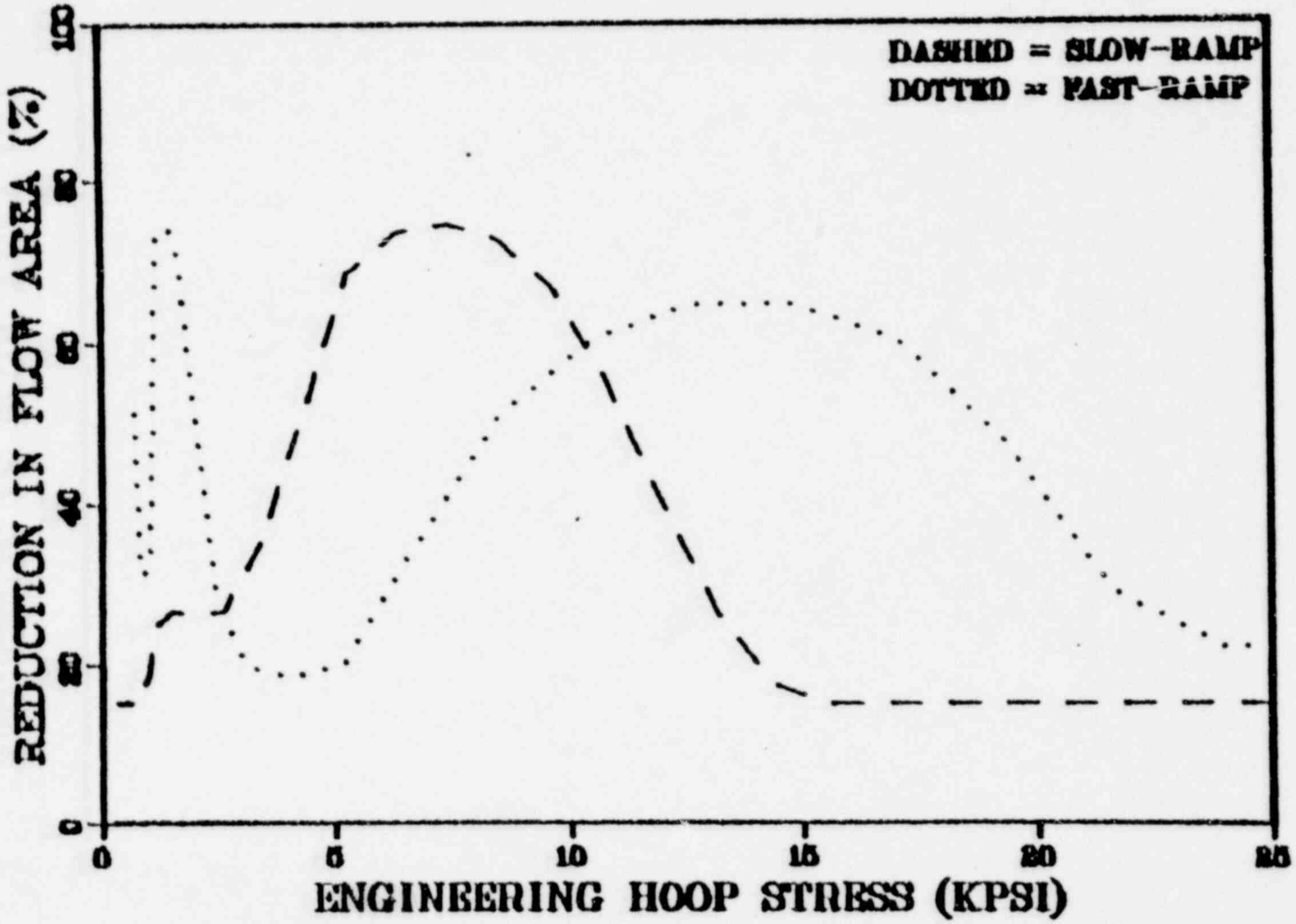
1573 346

BURST STRAIN CURVES



1573 347

PWR ASSEMBLY FLOW BLOCKAGE CURVES



1573 348

Slow-Ramp Correlations

#	Burst Temperature (°C)	0°C/S Engineering Hoop Stress (KPSI)	≤10°C/S Burst Strain (%)	≤10°C/S Flow Blockage (%)
1	600	15.56	20	15.2
2	625	14.40	21	17.6
3	650	13.22	30	26.1
4	675	12.03	44	39.4
5	700	10.84	58	54.6
6	725	9.68	70	66.5
7	750	8.53	78	72.7
8	775	7.40	80	74.6
9	800	6.30	79	73.6
10	825	5.24	72	68.4
11	850	4.26	54	50.8
12	875	3.36	39	34.7
13	900	2.59	31	26.6
14	925	1.98	30	26.1
15	950	1.52	31	26.6
16	975	1.20	29	25.2
17	1000	0.97	22	18.5
18	1025	0.80	20	15.2
19	1050	0.68	20	15.2
20	1075	0.59	20	15.2
21	1100	0.51	20	15.2
22	1125	0.45	20	15.2
23	1150	0.41	20	15.2
24	1175	0.37	20	15.2
25	1200	0.33	20	15.2

1573 349

Fast-Ramp Correlations

#	Burst Temperature (°C)	28°C/S Engineering Hoop Stress (KPSI)	≥25°C/S Burst Strain (%)	≥25°C/S Flow Blockage (%)
1	600	31.14	20	15.2
2	625	28.74	21	17.6
3	650	26.39	23	20.0
4	675	24.01	26	22.3
5	700	21.65	33	29.0
6•	725	19.32	50	47.5
7	750	17.04	63	60.3
8	775	14.78	67	64.6
9	800	12.57	67	64.6
10	825	10.46	63	60.3
11	850	8.50	54	50.8
12	875	6.70	39	34.7
13	900	5.17	23	20.0
14	925	3.95	22	18.5
15	950	3.04	23	20.0
16	975	2.40	34	29.9
17	1000	1.94	57	53.7
18	1025	1.61	78	72.7
19	1050	1.36	80	74.6
20	1075	1.17	78	72.7
21	1100	1.03	37	32.8
22	1125	0.91	35	30.9
23	1150	0.81	37	32.8
24	1175	0.73	47	43.2
25	1200	0.67	57	53.7

1573 350

Composite Correlations

#	Burst Temperature (°C)	Burst Strain (%)	Engineering Hoop Stress (KPSI)	Flow Blockage (%)
1	600	20	31.14	22.3
2	625	21	28.74	22.3
3	650	30	26.39	22.3
4	675	44	24.01	22.3
5	700	58	21.65	29.0
6	725	70	19.32	47.5
7	750	78	17.04	60.3
8	775	80	---	--
9	800	79	14.78	64.6
10	825	72	7.40	74.6
11	850	54	6.30	73.6
12	875	39	5.24	68.4
13	900	31	4.26	50.8
14	925	30	3.36	34.7
15	950	31	2.59	26.6
16	975	34	2.40	29.9
17	1000	57	1.94	53.7
18	1025	78	1.61	72.7
19	1050	80	1.36	74.6
20	1075	78	1.17	72.7
21	1100	37	1.03	32.8
22	1125	35	0.91	30.9
23	1150	37	0.81	32.8
24	1175	47	0.73	43.2
25	1200	57	0.67	53.7

1573 351

GENERAL

- APPENDIX K REQUIREMENTS MUST BE MET.
- REVISED MODELS MAY BE REQUIRED FOR ALL VENDORS.
- ALL BREAK SIZES NEED TO BE CONSIDERED.
- IF UNCERTAINTIES ARE NOT CONSIDERED IN SWELLING AND RUPTURE CURVES, APPROPRIATE SENSITIVITY STUDIES MUST BE PERFORMED.
- WIDTH OF THE VALLEY MAY BE AS IMPORTANT AS HEIGHT OF THE PEAK.

DETERMINING MAGNITUDE

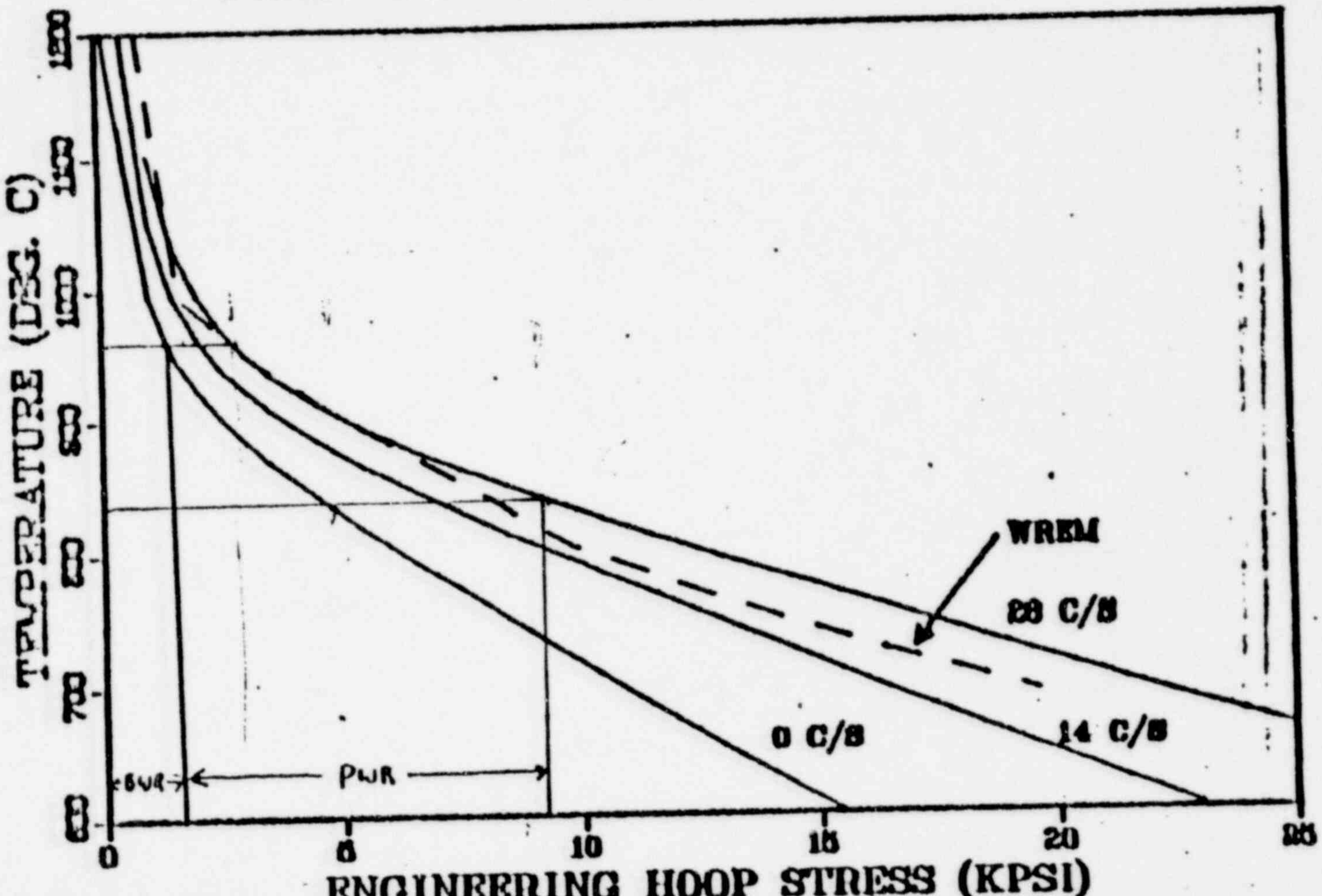
- NEED TO SORT OUT SUBSTANTIVE CONDITIONS.
- SOME TEMPERATURES AND RAMP RATES MAY NOT BE EXPECTED.
- THEREFORE MODELS NEED ONLY APPLY WHERE CONDITIONS WILL OCCUR.
- PWR RUPTURE TEMPERATURES 840 °C - 960 °C
- BWR RUPTURE TEMPERATURES 960 °C - 1200 °C

RAMP RATES 2.5 °C/sec. — 25 °C/sec.

1573 353

POOR ORIGINAL

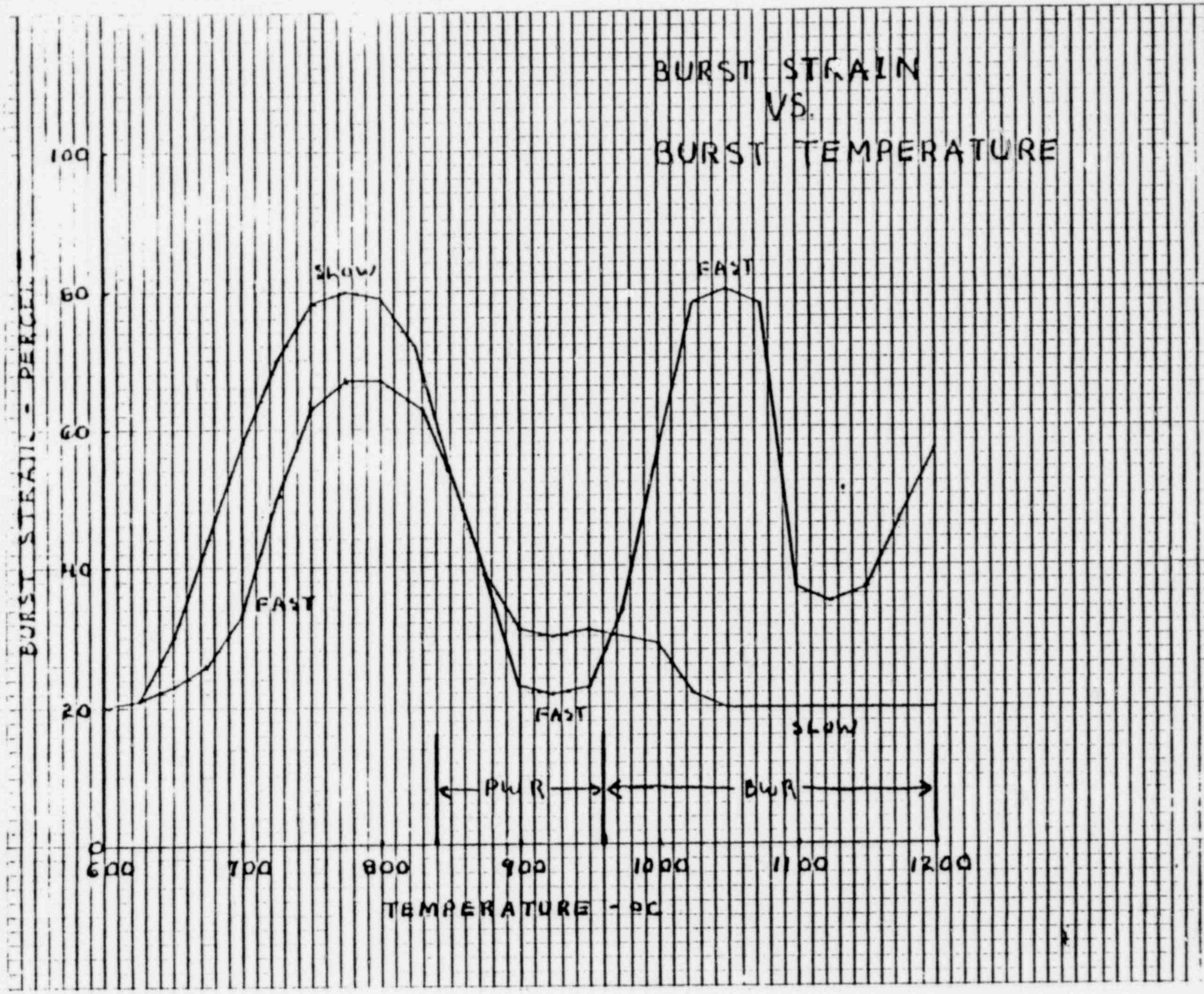
WREM & BURST TEMPERATURE CURVES



1573 354

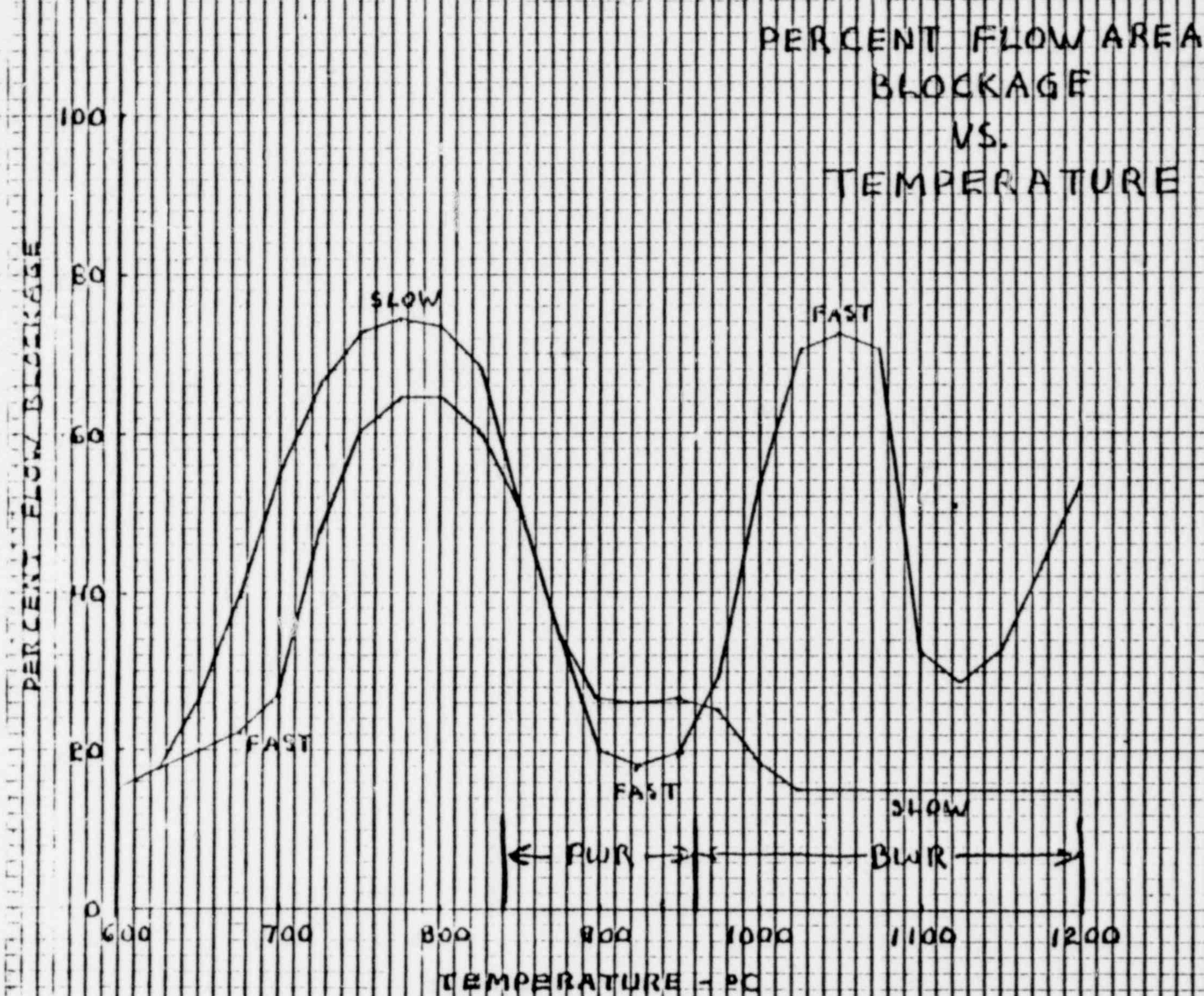
K•E 10 X 10 TO THE INCH • 7 X 10 INCHES KEUFFEL & ESSER CO.

46 0702



1573 355

POOR ORIGINAL


PERCENT FLOW AREA
BLOCKAGE
VS.
TEMPERATURE
PERCENT FLOW BLOCKAGE
100
80
60
40
20
0
600
700
800
900
1000
1100
1200
TEMPERATURE - °C
SLOW
FAST
FAST
FAST
SLOW
PWR
BWR

1573 356

LARGE BREAK REFLOOD

- PWR REFLOOD AT FLOODING RATES LESS THAN 1 IN./SEC. APPEARS TO BE WORST CONDITION BECAUSE OF APPENDIX K REQUIREMENTS FOR STEAM COOLING AND BLOCKAGE.

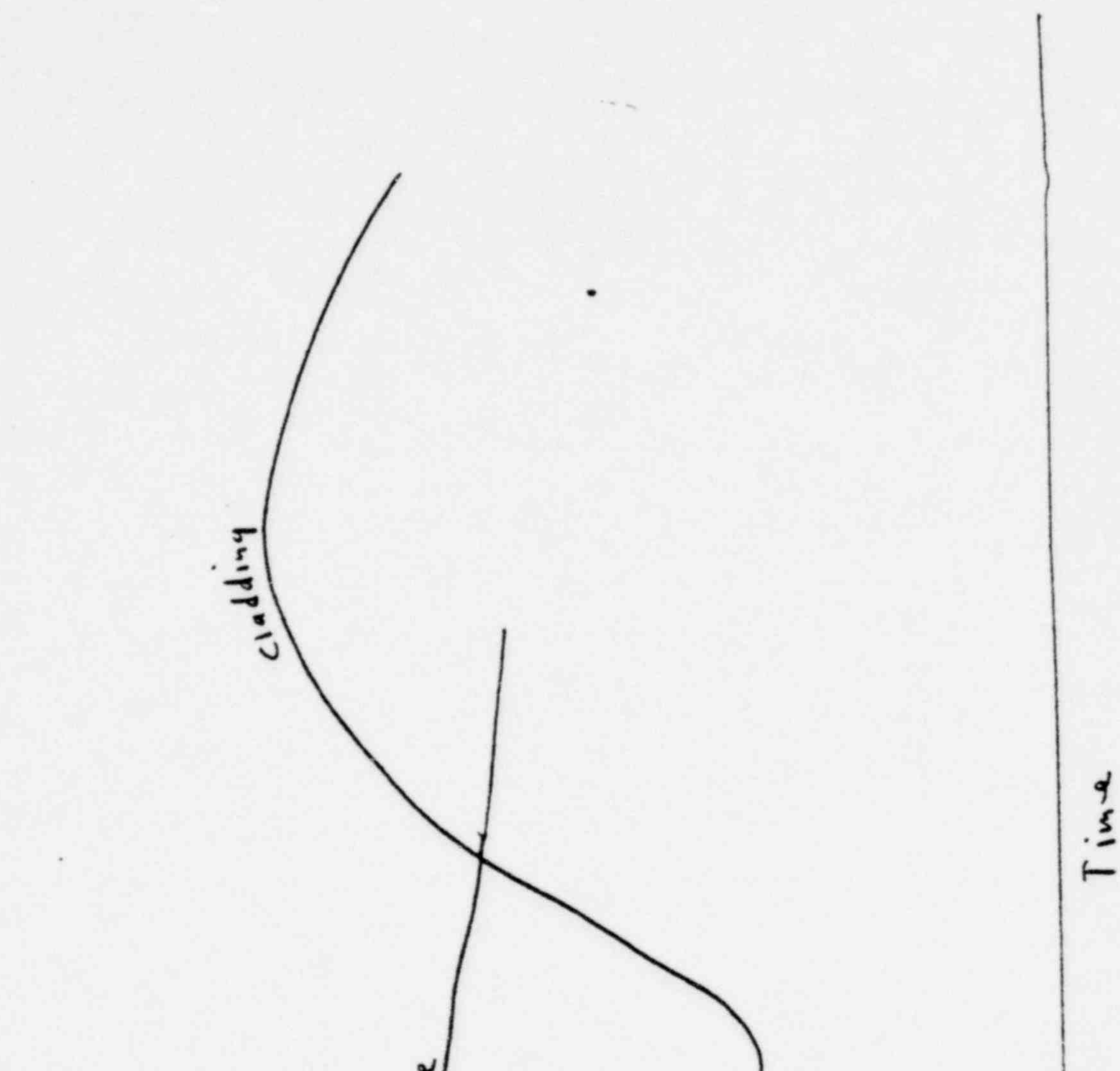
- NRC PERFORMED LIMITED SENSITIVITY STUDY ON BLOCKAGE, STRAIN, AND INCIDENCE OF RUPTURE.

1573 357

SWELLING AND RUPTURE REFLOOD STUDY

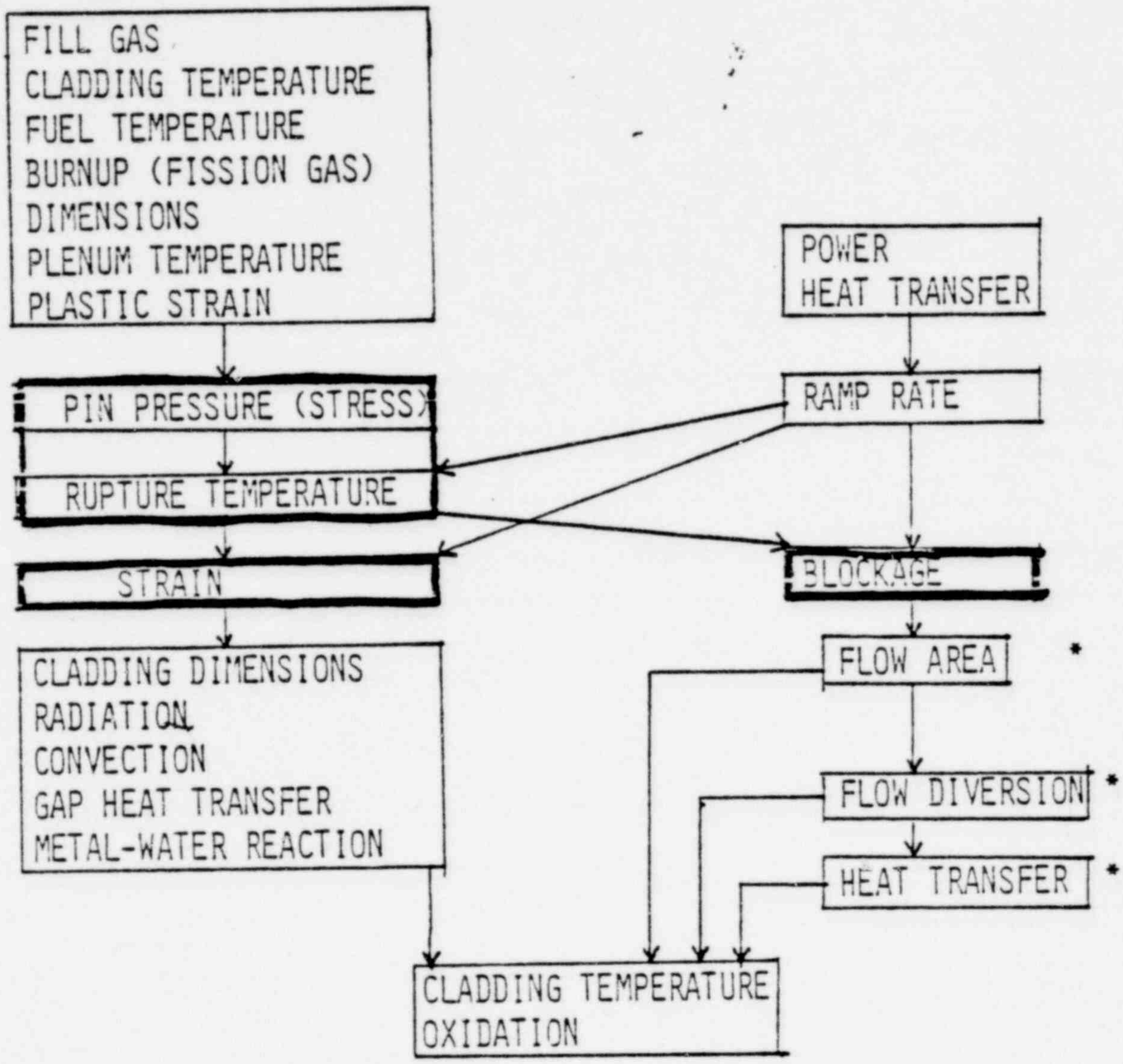
Case	BLOCKAGE MODEL	STRAIN MODEL	t_{rup}	T_{rup} °C	T_{INIT} °F	BLOCKAGE FRACTION	RUPTURED NODE				UNRUPTURED NODE			
							ELEV. IN.	STRAIN	TIME SEC.	PCT °F	ELEV. IN.	STRAIN	TIME SEC.	PCT °F
1	WREM	WREM	29.8	876	1804.1	.522	80.24	.444	260.	Melt	83.51	.085	260.	Melt
2	VENDOR	WREM	29.8	876	1804.1	.362	80.24	.444	44.25	1824.3	83.51	.085	298.	2143.7
3	VENDOR	1.0	36.4	915	1932.0	.238	80.24	1.0	45.5	2104.6	93.34	.133	120.	1869.[illegible]
4	VENDOR	WREM	45.	976	1780.0	.310	80.24	.394	44.25	1791.8	83.51	.088	298.	2040.1
5	VENDOR	1.0	45.	965	1763.6	.200	80.24	1.0	44.25	1773.4	93.34	.195	120.	1820.[illegible]

1573 358



1573 359

IMPORTANT PARAMETERS



1573 360

ENCLOSURE 7

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 2, 1979

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
Commissioner Ahearne

THRU: Executive Director for Operations

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: POTENTIAL DEFICIENCIES IN ECCS EVALUATION MODELS
(RUPTURE STRAIN AND FLOW BLOCKAGE)

As stated in my memorandum to you dated October 31, 1979 regarding our preliminary evaluation of the various vendor ECCS models with respect to rupture strain and flow blockage, the staff met on November 1, 1979, with representatives of the NSSS vendors and a fuel supplier; i.e. General Electric Company, Babcock and Wilcox Company, Westinghouse Electric Corporation, Combustion Engineering Inc., and Exxon Corporation. The purpose of the meeting was first, to determine the validity of data from an ongoing NRC confirmatory research program relating to the behavior of fuel cladding under postulated LOCA conditions, and second, to determine if ECCS evaluation models for operating plants required by Appendix K of 10 CFR Part 50 (Section IB. Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters) remain acceptable in light of the recent data. These data have been used by the staff to develop a new set of curves in determining cladding strain and flow blockage in fuel channels during a LOCA. The results of this meeting are summarized below.

1. BABCOCK AND WILCOX: The B&W blockage curve is considerably more conservative than our new blockage curve from the recent data over its entire range of applicability to B&W operating plants, although the B&W strain curve is non-conservative over a portion of the temperature range of interest. In the B&W ECCS analysis, blockage is much more important than strain in determining peak cladding temperature (PCT). The use of both new staff curves would actually result in a lower calculated PCT than before.

POOR ORIGINAL

1573 361

2. COMBUSTION ENGINEERING: CE had recently provided a new analysis model using improved rupture strain and flow blockage models that are similar to our new data curves and that show compliance with the 2200°F limit for CE operating plants. The improved CE models follow guidelines that are in approximate agreement with the new NRC staff curves. Inasmuch as CE used even larger blockages in their analysis than we are now recommending, we believe that they have additional conservatism in their PCT results for operating plants.

3. GENERAL ELECTRIC AND EXXON-BWR: Both BWR models appear to be significantly different from the majority of the new data, but this is because the recent data falls in a range that is much more applicable to PWR operating conditions. In the high-temperature, slow-ramp range applicability to operating BWRs, the GE and Exxon curves are in good agreement with the new NRC curves. Thus, there does not appear to be a safety problem with PCT's for operating BWR's.

4. EXXON-PWR: The most important curve used in previous PWR Exxon models is the fast-ramp blockage curve. This curve is conservative with respect to the new NRC staff curve based on recent data over the range of applicability for operating PWR's with Exxon fuel. The lesser important strain curve and slow-ramp blockage curve are in approximate agreement with the new NRC staff curves. In the regions where those Exxon curves somewhat underpredict the NRC curves, Exxon pointed out that our data do not contradict their curves. Accordingly, there does not appear to be a safety problem with PCT's for operating PWR's with Exxon fuel

5. WESTINGHOUSE: Westinghouse strongly disagrees with the applicability of the new data and therefore the new NRC staff blockage model; however, its strain curve is in approximate agreement with the new NRC staff curves over the range of operating conditions of operating plants. The Westinghouse blockage curve agrees well with the new NRC staff curves only over a limited stress range, but outside of that range the curves diverge sharply with the Westinghouse curves being less conservative than the NRC staff curves. Westinghouse believes that fuel cladding ruptures in its operating plants would always occur in that limited stress range where its model predicts values very near to those in the new NRC staff model. Additional information on this issue is expected from Westinghouse by late November 2, 1979.

At the conclusion of the November ' meeting, each of the five nuclear fuel suppliers was directed to provide NRC a letter by close of business November 2, 1979 confirming the preliminary information discussed above, and specifically providing answers to the following questions:

On the basis of information presented by the NRC staff at the meeting on November 1, 1979

(1) Are your operating plants safe, i.e., do they meet the 2200°F limit for peak clad temperature during a LOCA? Describe the basis for this conclusion.

(2) Do the evaluation models meet the requirements in Appendix K? Describe the basis for this conclusion.

(3) If "no" to the above, propose modified operating limits of such plants.

Discussions at the meeting, summarized above, provided a preliminary understanding of the expected contents of the five letters due late today. As indicated above, we expect each of the letters will provide an acceptable basis for continued operation of affected plants at least over the short term. However, the information to be received by evening November 2, may indicate that a small number of plants have minor safety deficiencies requiring short term action by the NRC.

Even assuming that all operating plants can justify continued operation over the short term, we believe that it will be necessary to take action to require changes in a number of the evaluation models required by Appendix K of our regulation.

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

cc: OGC
OPE
SECY

Contact:
Darrell G. Eisenhut
X27672

1573 363

POOR ORIGINAL