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# Cladding Swelling and Rupture Models for LOCA Analysis

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Office of Nuclear Reactor Regulation

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# Cladding Swelling and Rupture Models for LOCA Analysis

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ABSTACT

A description is given of fuel cladding behavior as it is modeled for
emergency-core-cooling-system (ECCS) evaluations in the safety analysis
of loss-of-coolant accidents (LOCAs).  Data are tabulated from experiments
that employed internally heated Zircaloy cladding that was ruptured in
aqueous atmospheres, and new correlations based on these data are given
for cladding rupture temperature, cladding burst strain, and fuel assembly
flow blockage.  Comparisons of these correlations with industry models that
are used in current licensing analyses for commercial nuclear power plants
reveal substantial differences.  The correlations in this report are in-
tended to be used as licensing standards for future LOCA analyses until
such time that the continuing research programs would demonstrate a need
for further revisions.

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# LIST OF FIGURES

## PREFACE

This report was first issued as a draft in November 1979 and was circulated for review by the technical community for the purpose of obtaining a technical critique. In addition to several informal comments, formal comments were received, and those comments are listed in Appendix C and have been placed in the NRC Public Document Room.

Based on the comments and some new data, which were also received during the comment period, some of the cladding correlations have been revised. In general, the burst strain correlation exhibits slightly larger strains, the assembly flow blockage correlation exhibits slightly smaller blockages, and most of the strain and blockage peaks are shifted toward higher temperatures by about 25°C compared with the correlations presented in the draft report. The rupture temperature correlation is unchanged.

## ACKNOWLEDGMENTS

## NOMENCLATURE

| | |
|---|---|
| AEC | Atomic Energy Commission |
| ANL | Argonne National Laboratory |
| ASTM | American Society for Testing and Materials |
| BCL | Battelle Columbus Laboratories |
| BWR | boiling water reactor |
| B&W | Babcock and Wilcox Company |
| CRNL | Chalk River National Laboratory |
| CSNI | Committee on the Safety of Nuclear Installations |
| C-E | Combustion Engineering Company |
| ECCS | emergency core cooling system |
| ENC | Exxon Nuclear Company |
| FABIOLA | Facility for Burst Testing of Zircaloy-4 Cladding Tubes Under Fuel Interaction, Steam Oxidation, and LOCA Conditions. |
| FRF | Fuel Rod Failure experiment |
| FRG | Federal Republic of Germany |
| FR-2 | Forschungszentru. 'eaktor-2 |
| GE | General Electric Company |
| KfK | Kernforschungszentrum Karlsruhe |
| JAERI | Japan Atomic Energy Research Institute |
| LOCA | loss-of-coolant accident |
| MRBT | Multirod Burst Test Program |
| NRC | Nuclear Regulatory Commission |
| ORNL | Oak Ridge National Laboratory |
| PWR | pressurized water reactor |

| | |
|---|---|
| REBEKA | Reflood Blockage Experiments |
| TREAT | Transient Reactor Test |
| W | Westinghouse Electric Company |
| WREM | Water Reactor Evaluation Model |
| YAEC | Yankee Atomic Electric Company |
| YR | Yankee Rowe |

# 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 type of accident, the time at which swelling and rupture occurred, the magnitude of swelling, and the resulting coolant flow blockage (i.e., reduction in flow area).

Such phenomena were among the many reactor safety issues discussed during the 1972-1973 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 imits 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 under-estimated. The degree of swelling and incidence of rupture are then used to calculate other core variables including gap conductance, cladding temperature, oxidation, embrittlement, and hydrogen generation. After the conclusion of the ECCS hearing, the NRC reviewed and approved

cladding behavior models (Refs. 3-18) for each U.S. fuel manufacturer (and for Yankee Atomic Electric Company) for their use in ECCS analyses.

During the ECCS hearing, uncertainties were apparent in the prediction of fuel behavior during a LOCA. Therefore, in the Commission's concluding opinion (Ref. 19), 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 in size up to small-bundle tests.

The research programs are not all finished, but with the completion of many out-of-reactor and a few in-reactor tests, we are at a plateau of understanding that exceeds our understanding in 1974 and suggests that improvements should be made in the licensing models. The trend of these recent data shows the likelihood of more ruptures, larger burst strains, and greater flow blockages than predicted by some of the licensing models. Consequently, we see the need to reevaluate all cladding models used for LOCA analyses 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,* we have derived different correlations for different ramp rates. The rupture temperature correlation explicitly accounts for ramp rate; the rates for which the slow-ramp and fast-ramp strain and flow blockage correlations apply will be defined, and correlations for intermediate rates can be obtained by linear interpolation.

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* 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.

## 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 crystal structure changes in the temperature range of interest, (c) the properties of zirconium-base alloys are susceptible to strain-rate effects, (d) axial and circumferential temperature variations in the cladding strongly influence strain localization, (e) oxygen embrittlement increases yield and failure strengths, (f) the cracking of oxide coatings results in failure sites that can localize stresses, and (g) aggressive fission products can reduce the threshold stress at which crack propagation will proceed. Consequently the behavior of Zircaloy cladding depends strongly on environment and hence on test conditions (Refs. 20-21).

In-reactor tests are difficult to interpret and are too expensive to be used for investigating all of these variables, while some out-of-reactor tests lack sufficient realism. Therefore, for final calibration of the data correlations, we have selected only those data from experiments employing pressurized rods in aqueous atmospheres and either internal fuel-pellet simulators (i.e., indirect cladding heaters) or actual fuel pellets (in-reactor). This selection emphasizes the more recent prototypical test data and deemphasizes much of the earlier data. The data we have selected come from independent programs at four major laboratories: (1) Oak Ridge National Laboratory, (2) Battelle Columbus Laboratories, (3) Kernforschungszentrum Karlsruhe (Germany), and (4) Japan Atomic Energy

Research Institute.  Appendix A provides a tabulation of all of the data
we have used, their references, and a legend of symbols that are used in
the later figures.

There are gaps in this data base, however, particulary with regard to the
absence of large bundle tests, and we have utilized the results from
simpler less typical tests to bridge the gaps.  While these more pristine
tests are atypical in a sense, they do reveal fundamental features of
Zircaloy behavior, they are valid for determining relative effects, and
they allow one to interpret the sparser prototypical data more accurately.

## 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 conditions of plastic deformation manifests itself as a heating-ramp-rate effect, and this effect will be treated explicitly. Truely isothermal conditions will not be treated exactly, but will be treated the same as very slow ramps. To eliminate design-specific dimensional effects, we have converted differential pressures to hoop stresses. The conversion was made using a thin-shell formula,

$$\sigma = (d/2t)\Delta P, \tag{3-1}$$

where $\sigma$ is engineering hoop stress, d is the undeformed cladding mid-wall diameter, t is the undeformed cladding thickness, and $\Delta P$ is differential pressure across the cladding wall at the time of rupture. Table 1 shows some computed values of hoop stress in terms of differential pressure for some 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 -- rupture temperature decreases with increasing wall stress -- the data are scattered primarily because

# TABLE 1

Engineering Hoop Stress as a Function of Internal Fuel Rod

Gas Pressure and Fuel Vendor Design

| Design | Hoop Stress (psi) for a 600 psi Differential Across the Cladding Wall |
|--------|---------------------------------------------------------------------|
| B&W 15x15 | 4570 |
| B&W 17x17 | 4540 |
| C-E 16x16 | 4280 |
| W 15x15 | 4910 |
| W 17x17 | 4690 |
| GE 8x8 | 4050 |
| ENC 15x15* | 3940 |
| ENC 8x8** | 3880 |

\* D. C. Cook, Unit 1

\*\* Oyster Creek
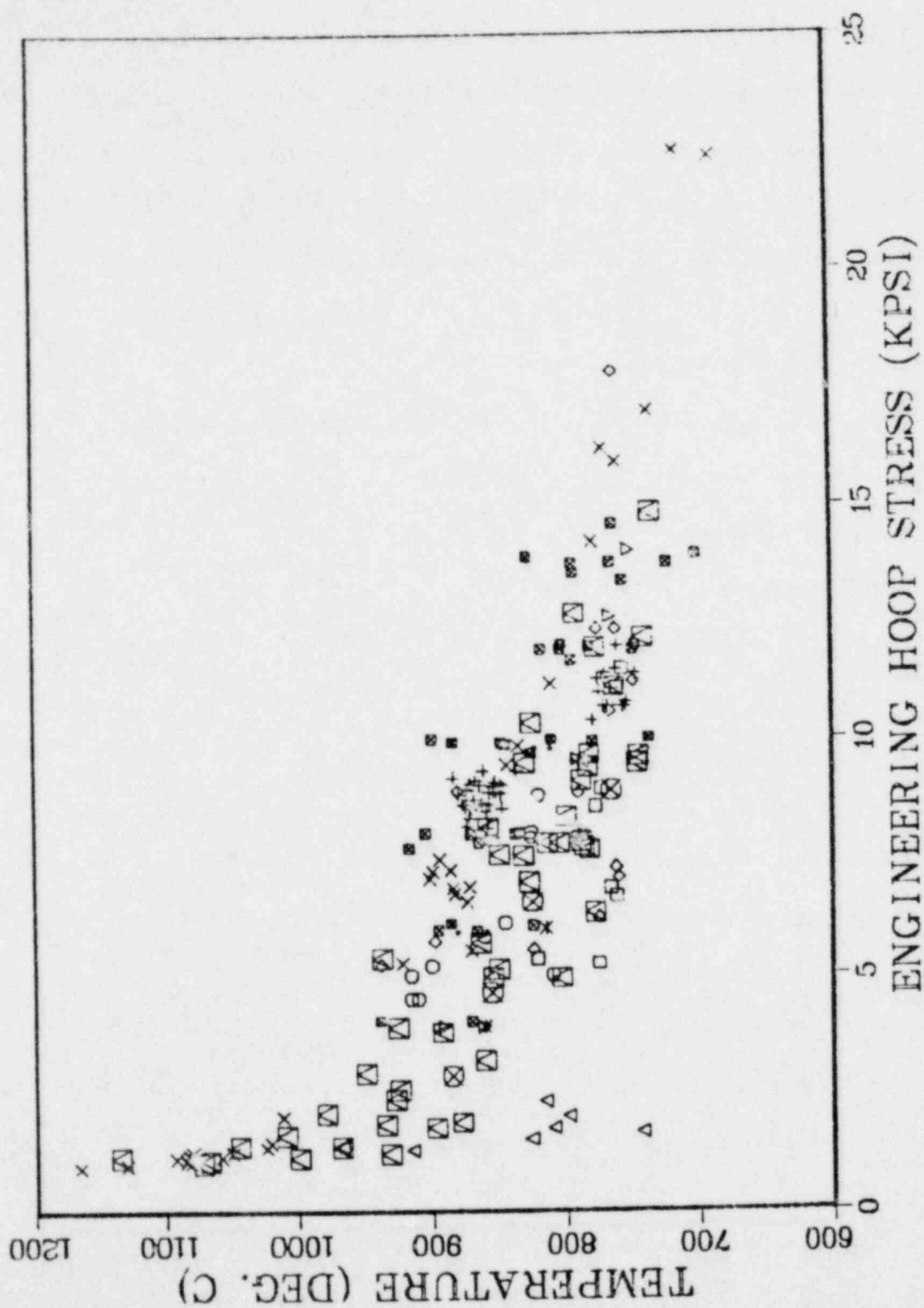
Fig. 1   Rupture temperature as a function of engineering hoop stress for internally heated Zircaloy cladding in aqueous atmospheres.

of ramp-rate effects and experimental uncertainties in determining rupture temperatures and effective ramp rates.

The uncertainty in determining the true rupture temperature arises mainly for two reasons. First, temperature measurements with thermocouples may result in a local perturbation of the true material temperature, and second, thermocouples are usually not located at the exact rupture locations. Therefore true rupture temperatures are likely to be higher than those experimentally measured.

These effects are minimized in the ORNL experiments because, first, the temperature profiles of the ORNL internal heaters are well known allowing placement of thermocouples very near to rupture sites (hot spots), and second, the combination of thin-external thermocouple wires with low steam flow rates results in small temperature perturbations (Ref. 22).

The uncertainty in determining the effective ramp rate arises from imprecision in the definition of ramp rate. For example, experimentalists most frequently report ramp rate as the increment in cladding temperature divided by the time duration where these values are taken from the start of the transient to the occurrence of rupture. However, the effective ramp rate may be limited only to the time interval over which plastic deformation occurs.

Unfortunately, the measurement of an effective ramp rate during this time of swelling is very difficult and has not been rigorously

assessed for the data compiled in Appendix A. Nevertheless, for those experiments conducted with a constant temperature-ramp rate, such as some of the ORNL experiments, the uncertainty in determining the effective ramp rate should be smaller than for experiments conducted with a constant power-ramp rate.

We have thus chosen a set of ORNL data that were all taken under similar conditions and were all determined with thermocouples attached to the external cladding surfaces. Figure 2 shows these ORNL single rod test data at 28°C/s (a common ramp rate used in the ORNL experiments) and the basic correlation we will adopt as developed by Chapman (Ref. 23) using numerical regression techniques. It is clear that most of the apparent data scatter has been eliminated by restricting the data to a single ramp rate.

Using rupture data taken with slower ramp rates (i.e., rates down to creep rupture conditions with up to a maximum 100-s hold time), Chapman has developed a ramp-rate correlation (Ref. 24),

$$T_R = 3960 - \frac{20.4\sigma}{1 + H} - \frac{8,510,000\sigma}{100(1+H) + 2790\sigma}, \qquad (3\text{-}2)$$

where $T_R$ is the rupture temperature in °C, $\sigma$ is the engineering hoop stress in kpsi, and H is the ratio of the heating rate in °C/s to 28°C/s (H varies from 0 to 1). This correlation can be used to produce a family of rupture-temperature curves. The correlation assumes that ramp-rate effects on the rupture temperature saturate at 28°C/s.

-10-
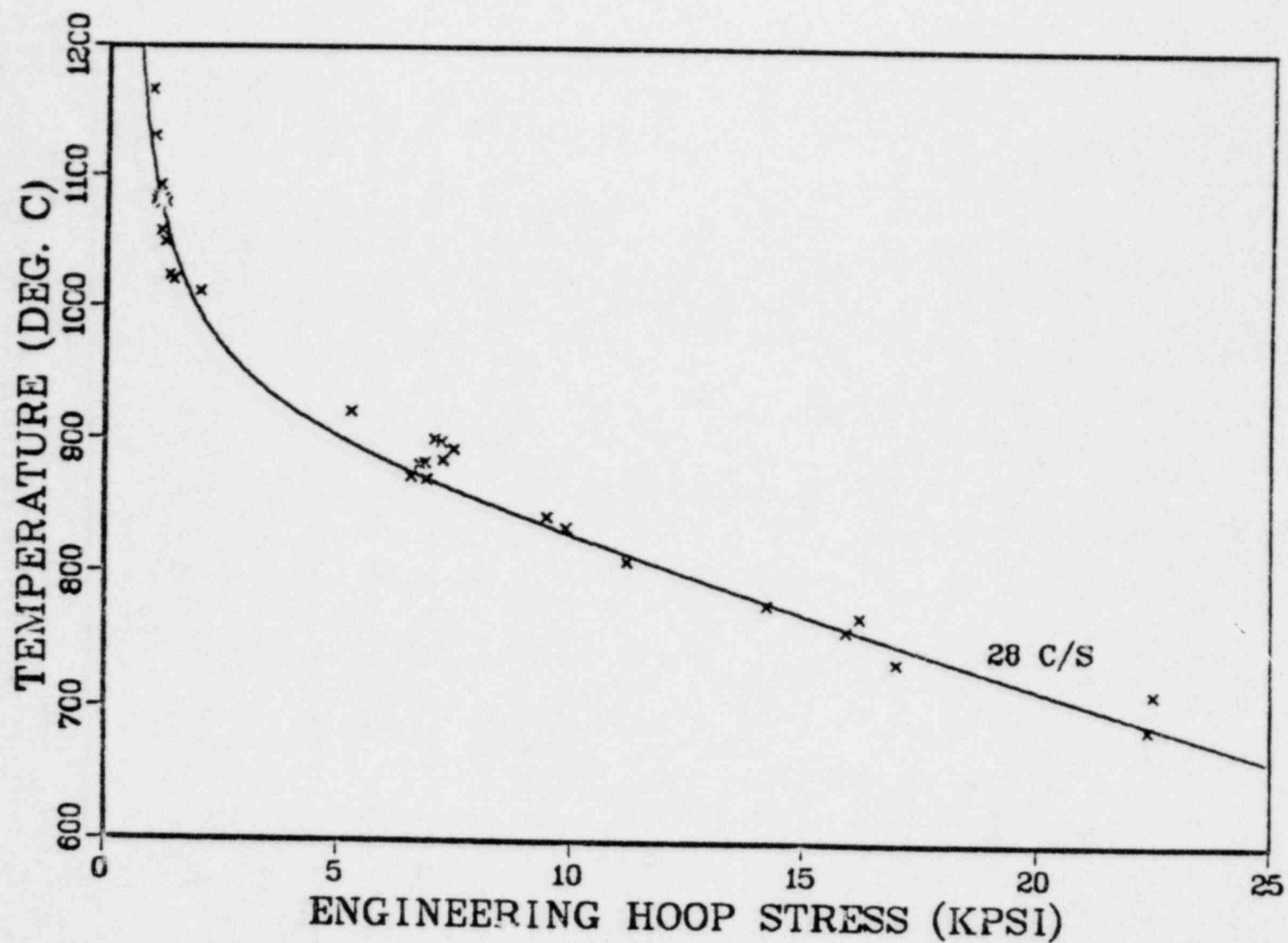
Fig. 2    Rupture temperature as a function of engineering hoop stress for ORNL internally heated, single rods tested in aqueous atmospheres at a heating rate of 28°C/s.

The curves that span the more frequently calculated ramp rates are shown in Fig. 3 along with the data of Fig. 1. Chapman has shown that most of the original scatter is attributable to ramp-rate effects, and the curves in Fig. 3 are seen to span most of the data. The up-right triangles still deviate from the correlations and the major body of data. Difficulties in temperature measurement for these TREAT in-reactor data (Refs. 25-26) are believed to be responsible for this deviation, and such discrepancies will be seen in later displays as well.

The closed squares are recent data from KfK by Erbacher and, in general, are seen to lie above the correlation. These differences probably result from the use of isobaric testing conditions rather than the more realistic constant-gas-inventory testing conditions, which were used at ORNL.

## 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 Eq. 3-2), ramp rate, and several other variables such as local temperature variations and metallurgical conditions. These effects have been discussed previously (Refs. 20-21,27). Figure 4 shows burst strain* as a function of one of these variables,

*Occassionally it has been observed that the maximum measured circumferential strain on a test rod will not coincide with the burst location. For infrequent instances such as these, we have used (and tabulated in Appendix A) the maximum circumferential strain as the burst strain, provided that this information was published by the originating authors.
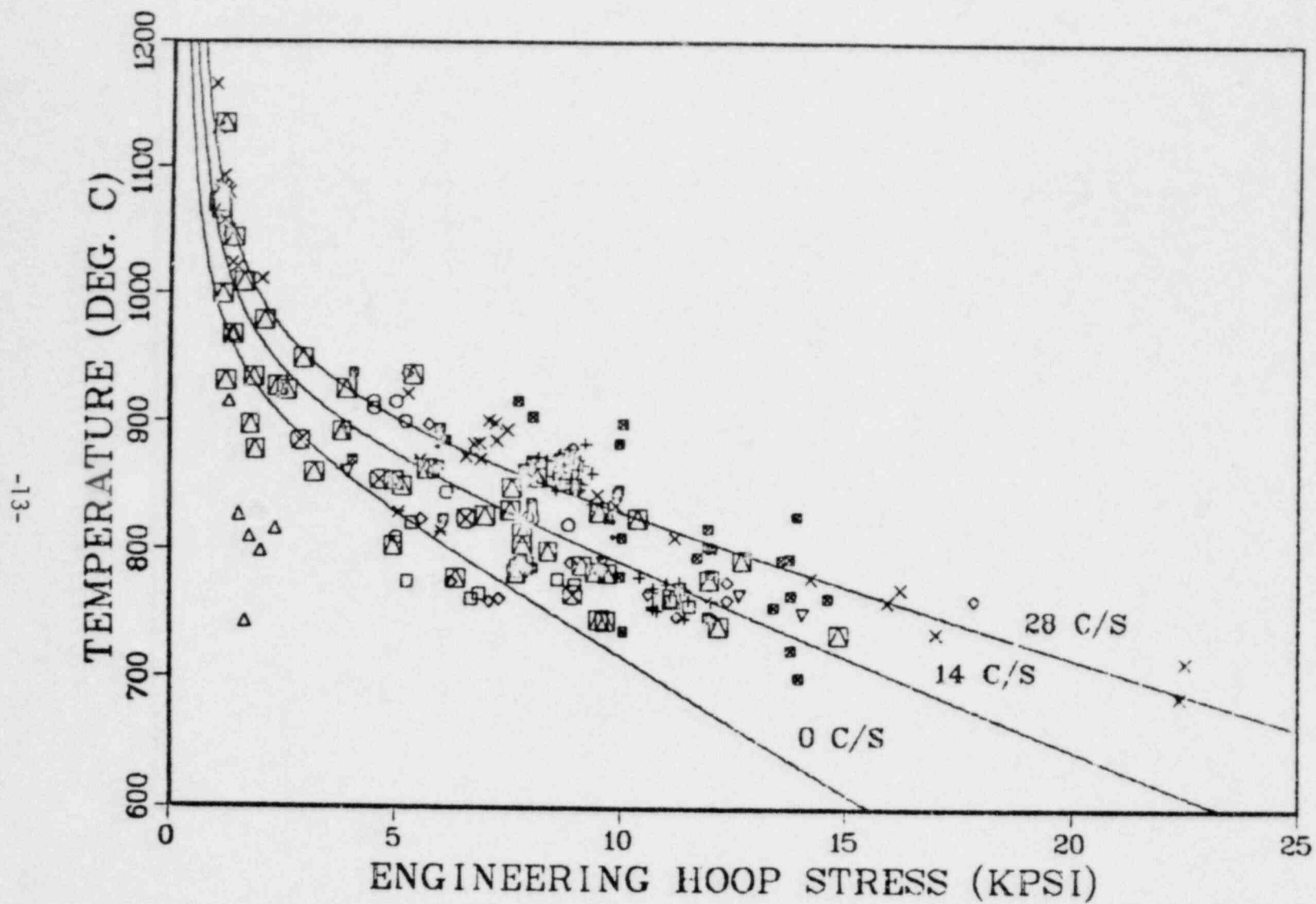
Fig. 3    ORNL correlation of rupture temperature as a function of engineering hoop stress and temperature-ramp rate with data from internally heated Zircaloy cladding in aqueous atmospheres.

Fig. 4 Maximum circumferential strain as a function of rupture temperature for internally heated Zircaloy cladding in aqueous atmospheres.

rupture 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 used results from less prototypical (but more controlled) tests to help resolve basic characteristics of Zircaloy and isolate the effects of certain variables. Figure 5 shows burst strain versus rupture temperature from Chung and Kassner's work (Ref. 21) with short Zircaloy tubes heated by passing an electrical current directly through the Zircaloy. This figure is one of many similar figures in Reference 21 and was chosen because it illustrates several fundamental features. 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 1250°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, but slow-ramp rates produce very small strains at temperatures greater than about 950°C because the Zircaloy has time to oxidize and

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*Superplasticity has been observed in zirconium alloys (Refs. 21,28-29). There is general agreement that the superplastic effect results in unusually large tensile extensibility that can occur in alloys of small grain size when strained at temperatures above approximately 40% of the absolute melting point at strain rates where the flow stress is highly strain-rate sensitive. Though there is not universal agreement that the superplastic deformation mechanism is responsible for generating all three of tne peaks in Figure 5, it is sufficient for our purposes to state that the material appears to behave superplastically inasmuch as it resists necking and large strains can occur.
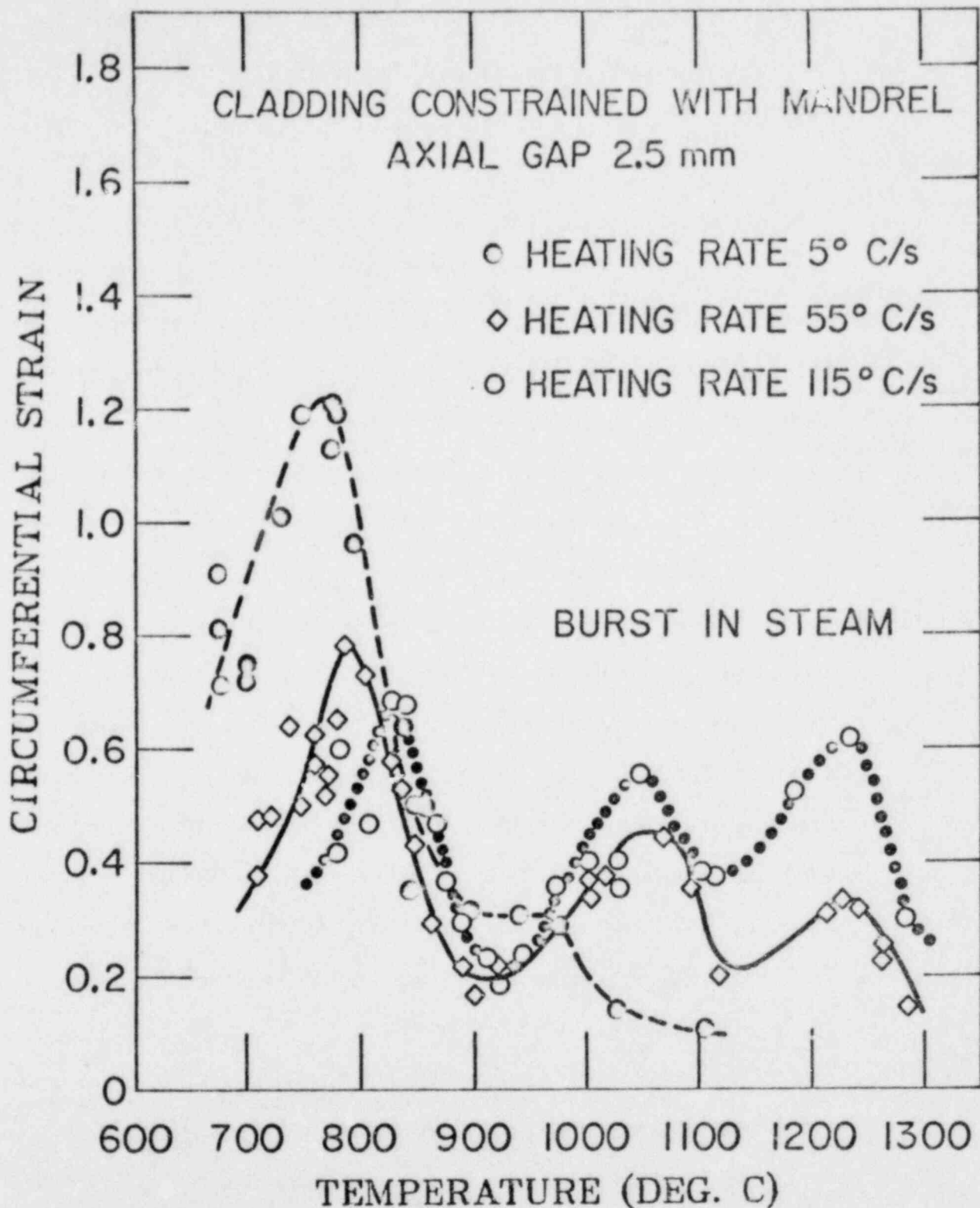
Fig. 5 Maximum circumferential strain as a function of rupture temperature for ANL directly heated, single rods tested in aqueous atmospheres.

embrittle before significant ballooning can occur. Fast-ramp rates produce the opposite effects in both temperature regimes by allowing local temperature gradients and precluding significant oxidation prior to rupture. In the region of phase transition, the burst strain appears to be relatively insensitive to ramp rate.

To derive the slow-ramp correlation (applicable to ramp rates $\leq 10°C/s$), which is shown in Fig. 6, we have retained the shape of Chung and Kassner's curves and positioned the peaks and valleys according to the 0-10°C/s data in our prototypical data base.

The alpha-phase peak was located at 800°C and assigned the value of 90% based on the data in Figure 6. Selection of 90% strain for this alpha-phase peak height was influenced by several factors. First, most of the data below the curve were discounted because they are from tests with features that are known to reduce strain (e.g., non-uniform heaters, corrosion fission products, cold shrouds) but might not be present in a real LOCA. Second, Chapman (Ref. 30) recommends 100% for this peak, but bases that recommendation in part on Chung and Kassner's data, which come from directly heated cladding specimens; direct or external heating methods are capable of exaggerating burst strains by maintaining artificially small local temperature variations (see Ref. 20), and such experiments were excluded from our data base. Third, Erbacher's recent data (Ref. 31) in the vicinity of 800°C have a mean value of about 90% (90.5% within 25°C of the peak).

Fig. 6  Maximum circumferential strain as a function of rupture temperature for internally heated Zircaloy cladding in aqueous atmospheres at heating rates less than or equal to 10°C/s.

Erbacher speculates (Ref. 32) that the temperatures might be more uni-
form than would occur under reactor accident conditions (azimuthal tem-
perature variations in these tests were measured to be up to 70°C in
magnitude at the time of rupture). We believe that prototypical
testing is currently the best way of estimating temperature uniformity
and we find no reason to discount Erbacher's results. Finally,
Chapman's single-rod heated-shroud data in the vicinity of 800°C had
a mean value of about 90% (93.3% within 25°C of the peak). Two of
Chapman's 0°C/s test results were eliminated from this averaging be-
cause the heater power was so low (about 3W) that these tubes may have
ruptured as if they were in a muffle furnance (the heated shrouds) and
that would be equivalent to external heating (not included in our data
base).

The strain in the alpha-plus-beta-phase valley is the same in the slow-
ramp and the fast-ramp correlations, and the data that influenced the
valley floor can be seen in Fig. 7.

A beta-phase peak was included in Fig. 6 at 1025°C even though our data
base contains no data in that temperature range and Fig. 5 does not
exhibit such a peak, although other figures in Reference 21 do exhibit
the peak. At high temperatures strains should be limited for slow-
ramp rates because of the opportunity for oxidization and embrittle-
ment prior to rupture. We have estimated the amount of oxidation for
a slow ramp (1°C/s in this case) terminating at 1025°C and concluded
that the amount of oxidation is too small to eliminate this strain
peak due to embrittlement of the cladding wall; however, the strain
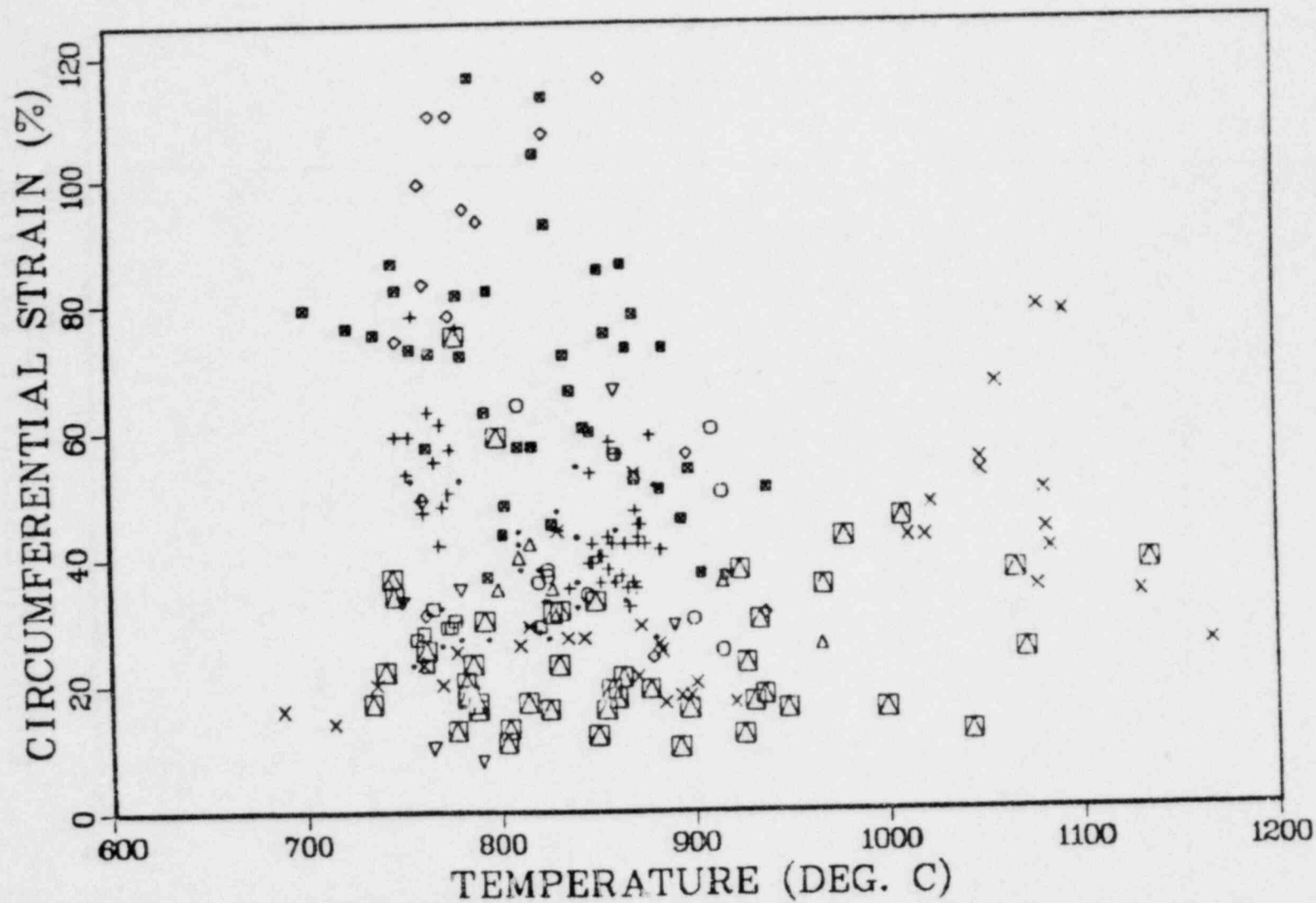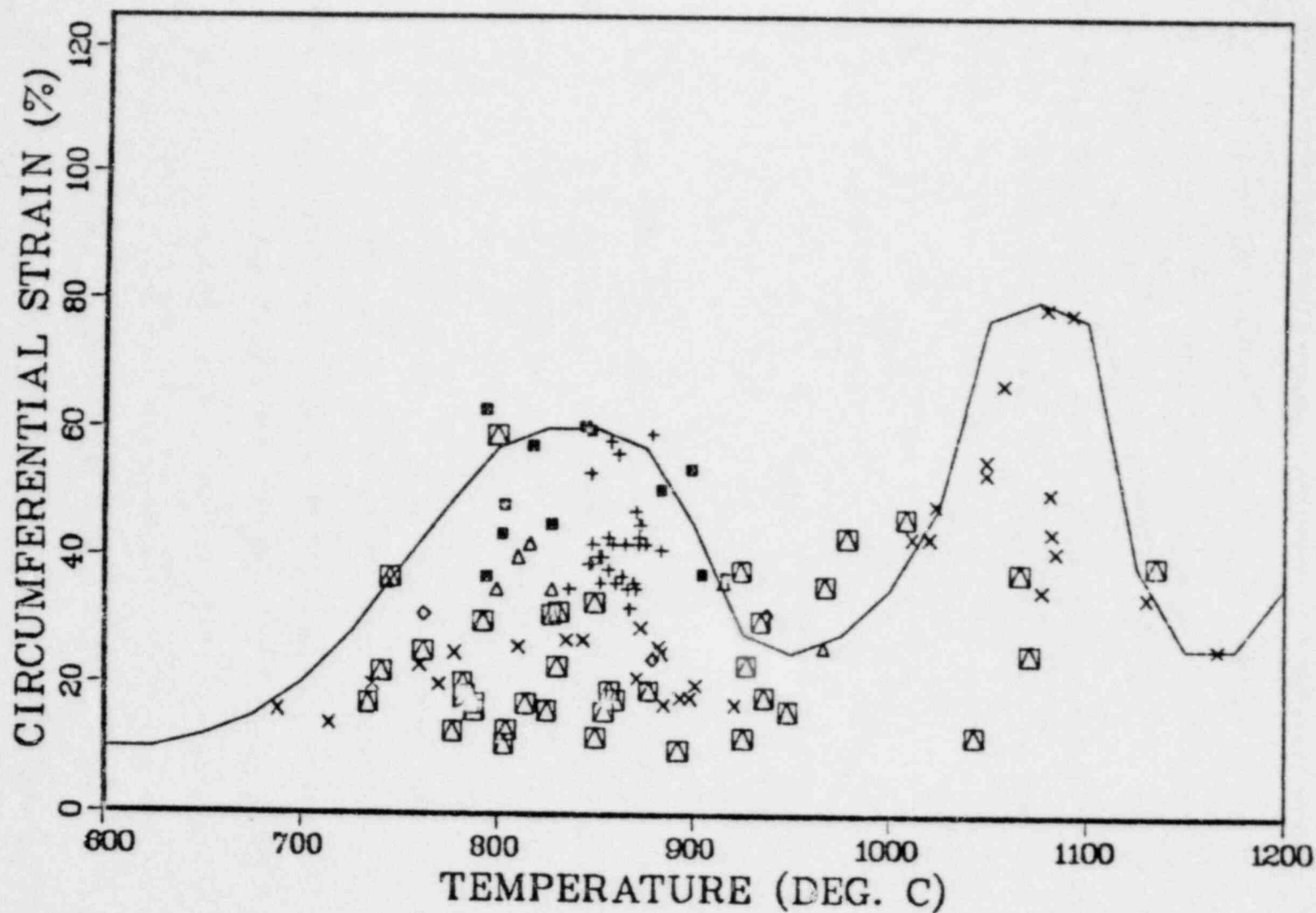
Fig. 7    Maximum circumferential strain as a function of rupture temperature for internally heated Zircaloy cladding in aqueous atmospheres at heating rates greater than or equal to 25°C/s.

peak should be attenuated because of oxide cracking, which exposes bare metal surfaces to further oxidation and thus localizes the hoop stress. Furthermore, because of the temperature dependence of oxidation, the symmetric peak for fast ramps (or in vacuum) will be skewed to the left with an apparent lowering of the temperature at which the peak occurs. Chung and Kassner's work shows a peak shift on the order of 50°C. Based on these considerations and other rod burst data not displayed here, we have located the peak at 1025°C and chosen 35% strain as the peak value.

To derive the fast-ramp correlation (applicable to ramp rates $\geq$ 25°C/s), which is shown in Fig. 7, we have again retained the shape of Chung and Kassner's curves and positioned the peaks and valleys according to our data base for ramp rates greater than or equal to 25°C/s. As with the slow-ramp correlation, we have bounded most of the data because there are experimental limitations that can explain most of the low-lying data.

For the alpha-phase peak, there is less uncertainty in the fast-ramp data than for slow-ramp data, and the peak was set at 60% strain from 825 to 850°C. For the beta-phase peak, we have bounded the high-ramp-rate ORNL data, and the peak was placed at 1075°C with 30% strain. There is some concern (Ref. 30 and 33) that the ORNL high-temperature fast-ramp data may not be best characterized as fast-ramp data. This is because during the course of some of these experiments, the ramp rate, which was initially set at about 40°C/s, decreased below 25°C/s before rupture occurred. Since (a) such

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conditions are plausible during a real LOCA and (b) the fast-ramp data were taken with the older-version heaters and unheated shrouds such that straining was inhibited, we believe that downward adjustments to the fast-ramp correlation are not warranted. The strain in this temperature regime does not presently affect licensing analyses, so these uncertainties are relatively unimportant.

We have structured the fast-ramp curve to predict increasing strain at 1175°C to reflect a second high-temperature beta-phase peak at about 1250°C. The strain magnitude in this off-scale temperature region is scaled to correspond to the average burst strain obtained in the second TREAT experiment, FRF-2. Although peak heights for the slow-ramp and fast-ramp correlations were determined primarily from the prototypical data base, they bear approximately the same relation to each other as found in Chung and Kassner's work.

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 (including intermediate-rate data not shown in Figs. 6 and 7). The composite curve gives a good representation of the data, provided that the causes of small strains are kept in mind. Since we have bounded most of the data and approximated the data for conditions that could cause large strains in real LOCAs, we believe these correlations satisfy the intention of Appendix K not to underestimate the degree of swelling.
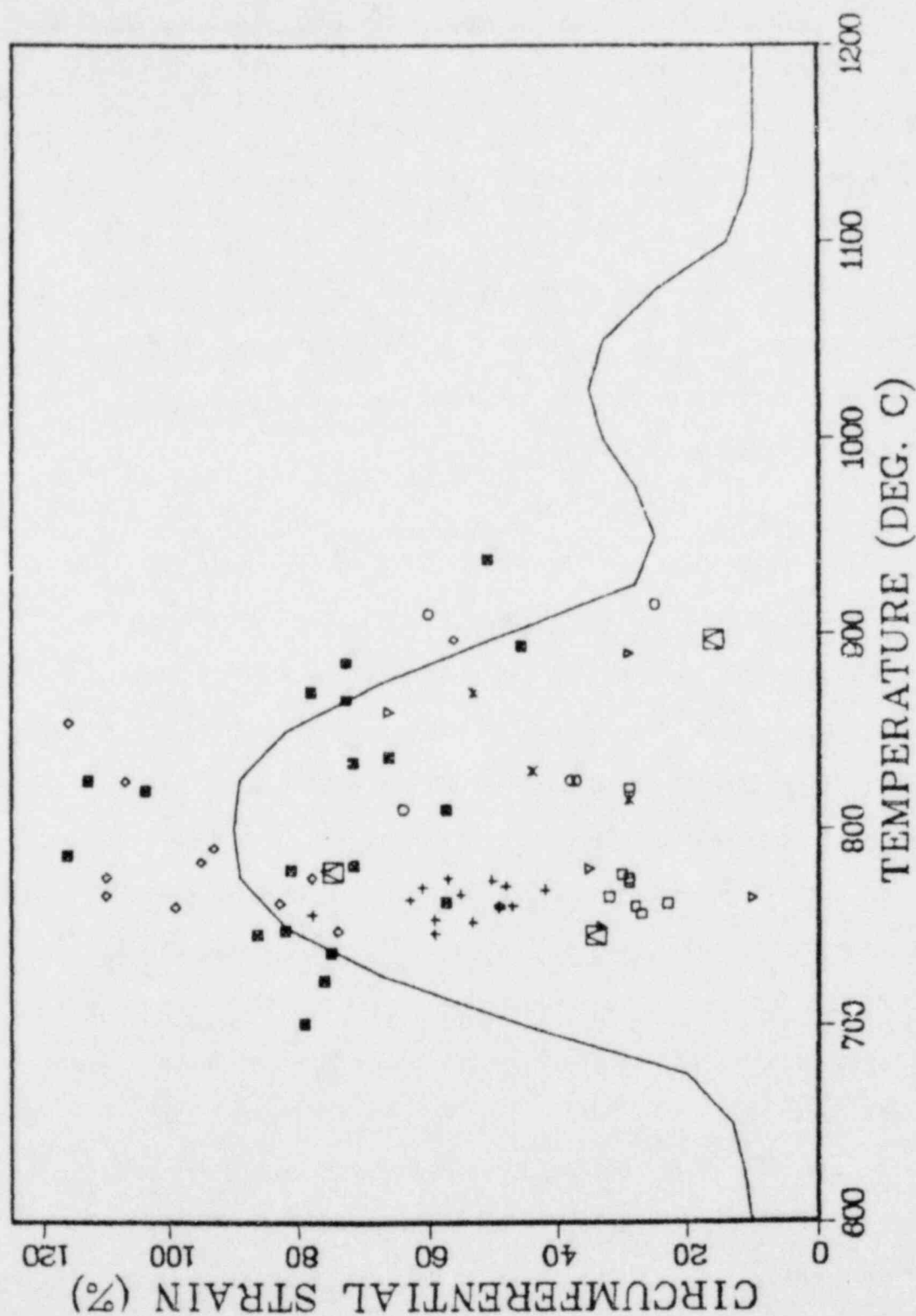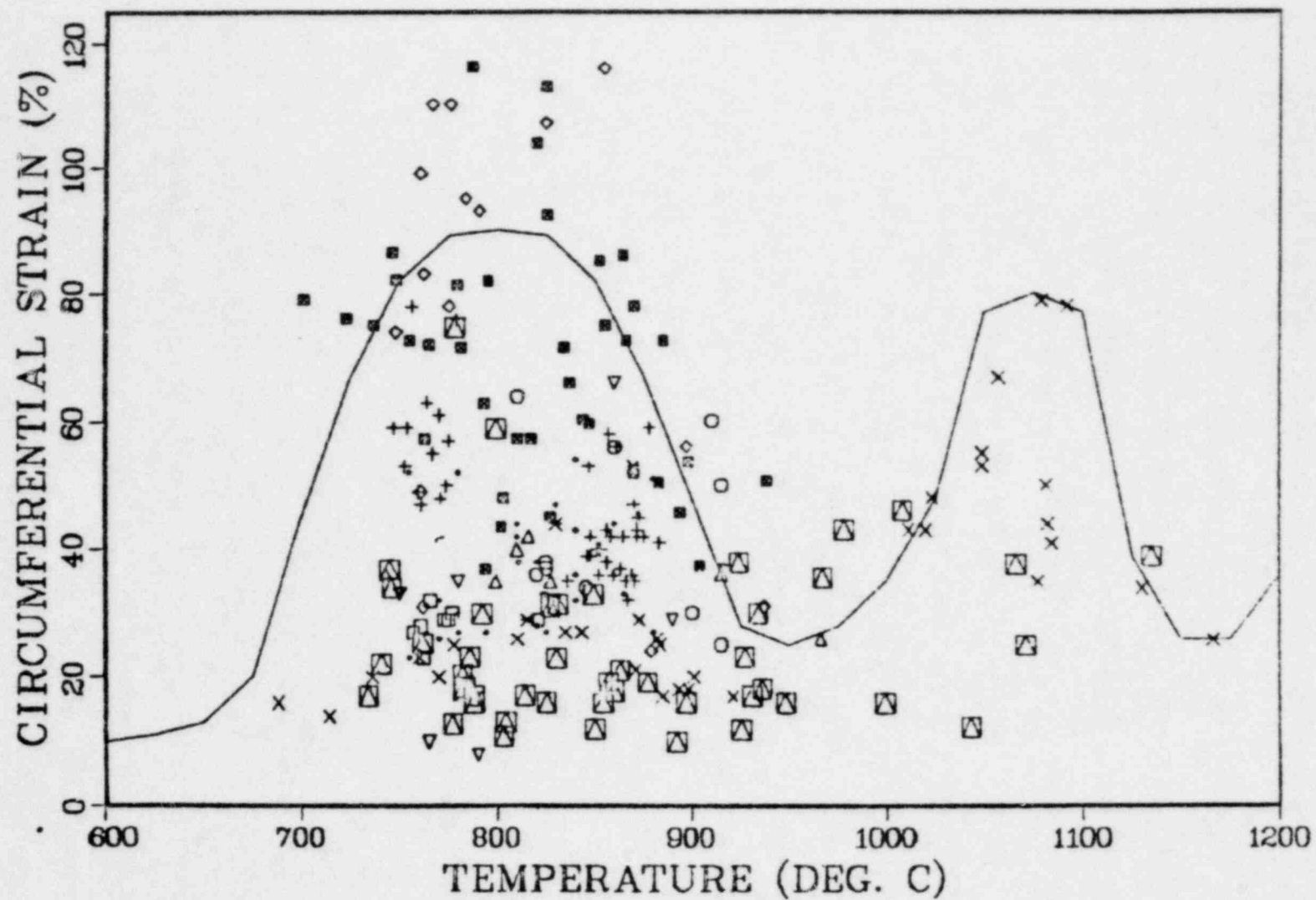
Fig. 8    Maximum circumferential strain as a function of rupture temperature for internally heated Zircaloy cladding in aqueous atmospheres for all heating rates.

## 3.3 Assembly Flow Blockage

Flow blockage is defined as the precent reduction in cross-sectional flow area of a bundle. Very few dire measurements of bundle blockage have been made under prototypical conditions and the best attempts are shown in Fig. 9. ...use of the scarcity of the blockage data, it is necessary to derive bundle blockage from single-rod burst strains and then to compare results with bundle data for verification rather than to rely soley on the bundle data. The derivation is not straight forward, however, since actual flow around a burst node is complex and since test results have shown that bursts in a bundle are not coplanar. To accomplish this derivation, therefore, we will determine an empirical relation between burst strain and bundle blockage from the three ORNL MRBT experiments.

The diagram in Fig. 10 outlines the steps in deriving flow blockage. An empirical (non-mechanistic) relation is first found between burst strain and average rod strain in the plane of blockage; it will be seen that the average coplanar rod strain is only about half of the burst strain. Figure 11 illustrates this step.

Figure 11 shows the plane of maximum blockage in Chapman's bundle B-1 (Ref. 34-35). Only 4 rods ruptured in this plane; the other 12 rods ruptured at other elevations (rod number 3 leaked and has been excluded from some of the analysis). The average rod strain in this plane is seen to be 25%. The average burst strain for these 15 rods was measured to be 42% with a standard deviation of 7% strain.

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Fig. 9    Reduction in local flow area as a function of rupture temperature for internally heated Zircaloy clad bundles in aqueous atmospheres.

```
                    ┌─────────────────────┐
                    │    BURST STRAIN     │
                    └─────────────────────┘
                               │
  X 0.56 SMALL BUNDLE          │        BASED ON BUNDLE TESTS
  X 0.46 LARGE BUNDLE          │
                               ▼
                    ┌─────────────────────┐
                    │  AVERAGE COPLANAR   │
                    │     ROD STRAIN      │
                    └─────────────────────┘
                               │
  FROM FIG. 13                 │        GEOMETRIC CONVERSION
                               │
                               ▼
                    ┌─────────────────────┐        COMPARE WITH
                    │   LOCAL BLOCKAGE    │──────▶  BUNDLE TESTS
                    └─────────────────────┘
                               │
  X0.95                        │        GEOMETRIC REDUCTION
                               │
                               ▼
                    ┌─────────────────────┐
                    │  ASSEMBLY BLOCKAGE  │──────▶  PWR CORRELATIONS
                    └─────────────────────┘
```

Fig. 10    Outline of flow blockage model.

Fig. 11     Computer drawn profile from digitized data of ORNL
            bundle B-1 section at 76.5-cm elevation.

For this bundle the ratio of the average coplanar rod strain (25%) to the one-sigma conservatively biased* burst strain (49%) is 0.51. Repeating this procedure for bundles B-2 and B-3 gives ratios of 0.57 and 0.59. Thus the overall average ratio is 0.56, which is the relationship to be used for small bundles.

To obtain flow blockage for large commercial-size fuel assemblies, a small modification is needed in the derivation. First, it must be recognized that maximum planar bundle 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 in the rupture 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.

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*A conservatively biased burst strain is used here to reduce the calculated blockage when Figs. 6 and 7 are utilized. Initially we believed that Figs. 6 and 7 were conservative. In that case, using a conservative bias in the derivation would be appropriate. We now believe that the strain curves in Figs. 6 and 7 are best-estimate correlations for LOCA conditions that are conducive to large strains. Nevertheless, since retaining this bias in the derivation improves the agreement between measured and predicted flow blockages, we have retained the bias.

To account for this averaging effect for commercial PWR fuel assemblies ranging from 14x14 to 17x17, we have used an axially averaged blockage from Chapman's bundle tests rather than the maximum blockage value used above for small bundles. For bundle B-1 (Fig. 12), the average (41%) of the blockages was found between the 23-cm and 47-cm locations where the suppressing effect of spacer grids at 10-cm and 66-cm has dissipated. Similar averages were found for bundles B-2 and B-3. Using these average values in deriving the ratio of average coplanar rod strain to one-sigma burst strain results in a smaller value, and the ratio to be used to derive large assembly blockages from burst strain data is thus 0.46 (compared with 0.56 for small arrays).

The remaining steps in deriving flow blockage involve simple geometric calculations. Given the average rod strain in the plane of blockage, the percent local flow blockage (i.e., blockage with no credit for non-swelling guide tubes) can be found from Fig. 13.

Figure 13 relates average coplanar rod strain to flow area reduction is such a way that the rods retain their circular cross section until they first touch (at 32% strain), and for larger strains the cross sections become square. Complete blockage (100%) results when average coplanar rod strain reaches about 70%. Since current PWR fuel designs have equivalent pitch-to-rod-diameter ratios, Fig. 13 applies to all PWR fuel designs.
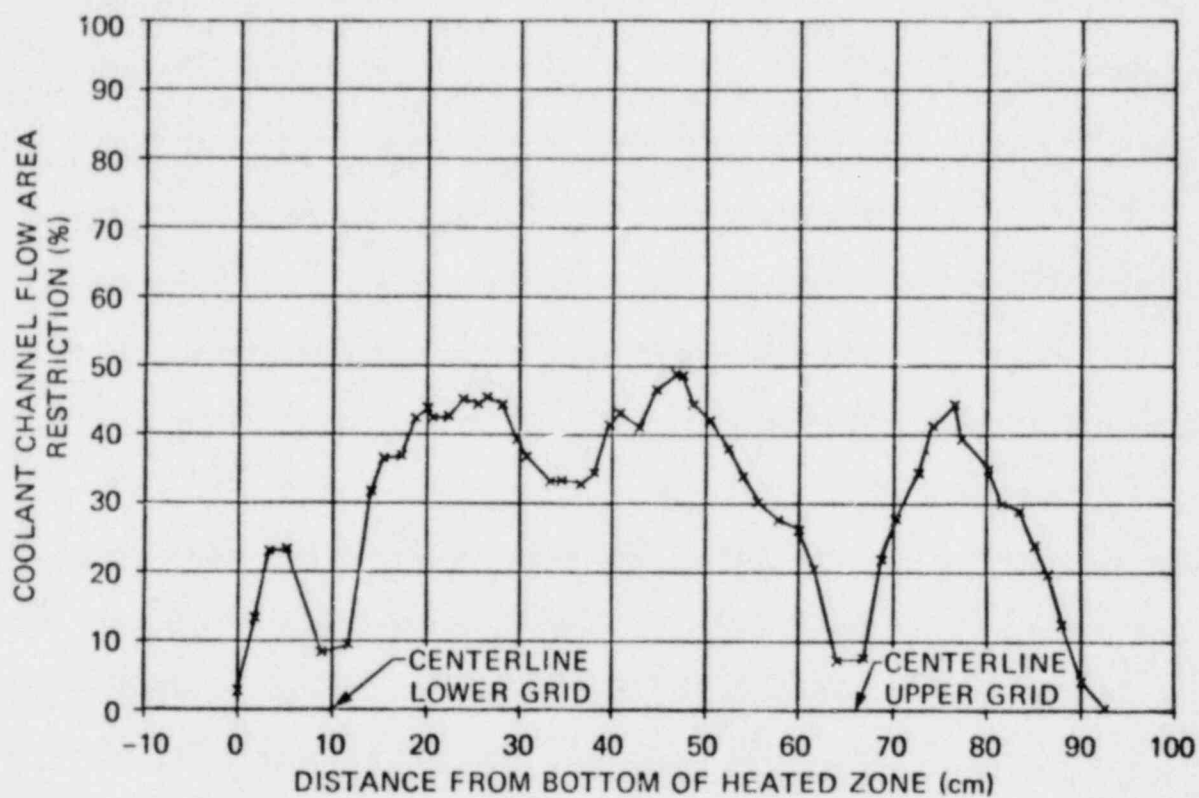
Fig. 12    Coolant channel flow area restriction as a function of elevation in ORNL bundle B-1.
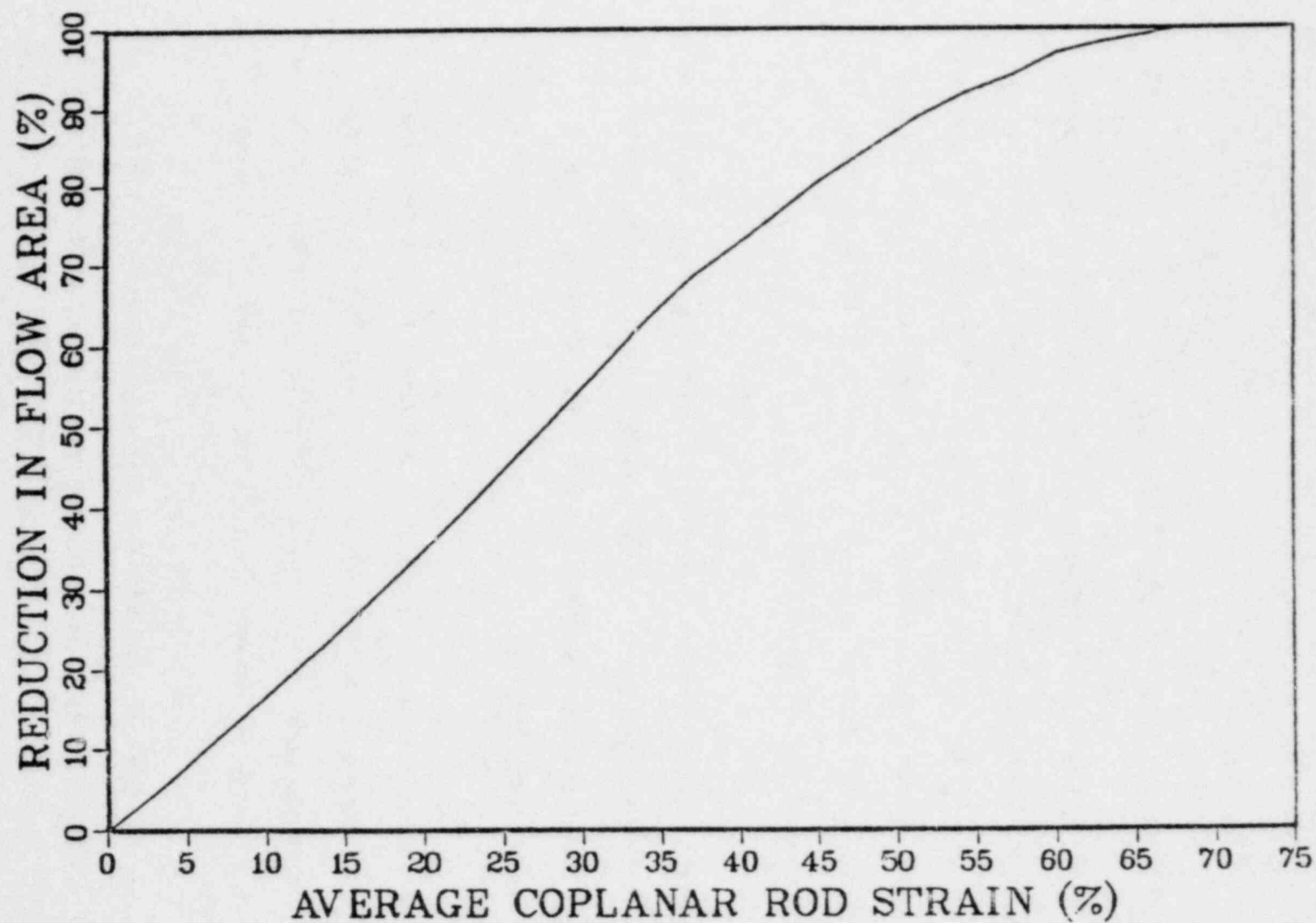
Fig. 13    Reduction in local flow area as a function of average coplanar rod strain.

The final step in deriving assembly blockage involves a reduction of 5% to account for instrument tubes and guide tubes that would not balloon. The exact scaling factor depends on the fuel design and is given by

$$\text{Scaling factor} = N_r A_r / (N_r A_r + N_g A_g), \qquad (3-4)$$

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 guide tubes or instrument tubes, and $A_g$ is the flow area around an undeformed guide tube or instrument tube. The differences in scaling factors of different commercial fuel designs are small so we have used the approximate reduction of 5% for all PWR designs.

Figures 14 and 15 compare the flow blockage data with correlations derived from Figs. 6 and 7. The curves in Figs. 14 and 15 are for small bundles without guide tubes and are, therefore, directly comparable with the data.

Although the data are sparse and do not cover the entire temperature range of interest, we think that Figs. 14 and 15 provide good verification of the correlation. This is especially true when the reasons for the deviations between the data and the curve are examined.

The most significant deviation is the asterisk symbol in Fig. 14 at 25% blockage. This datum point represents a 3x3 bundle test (Rebeka 1)

-33-
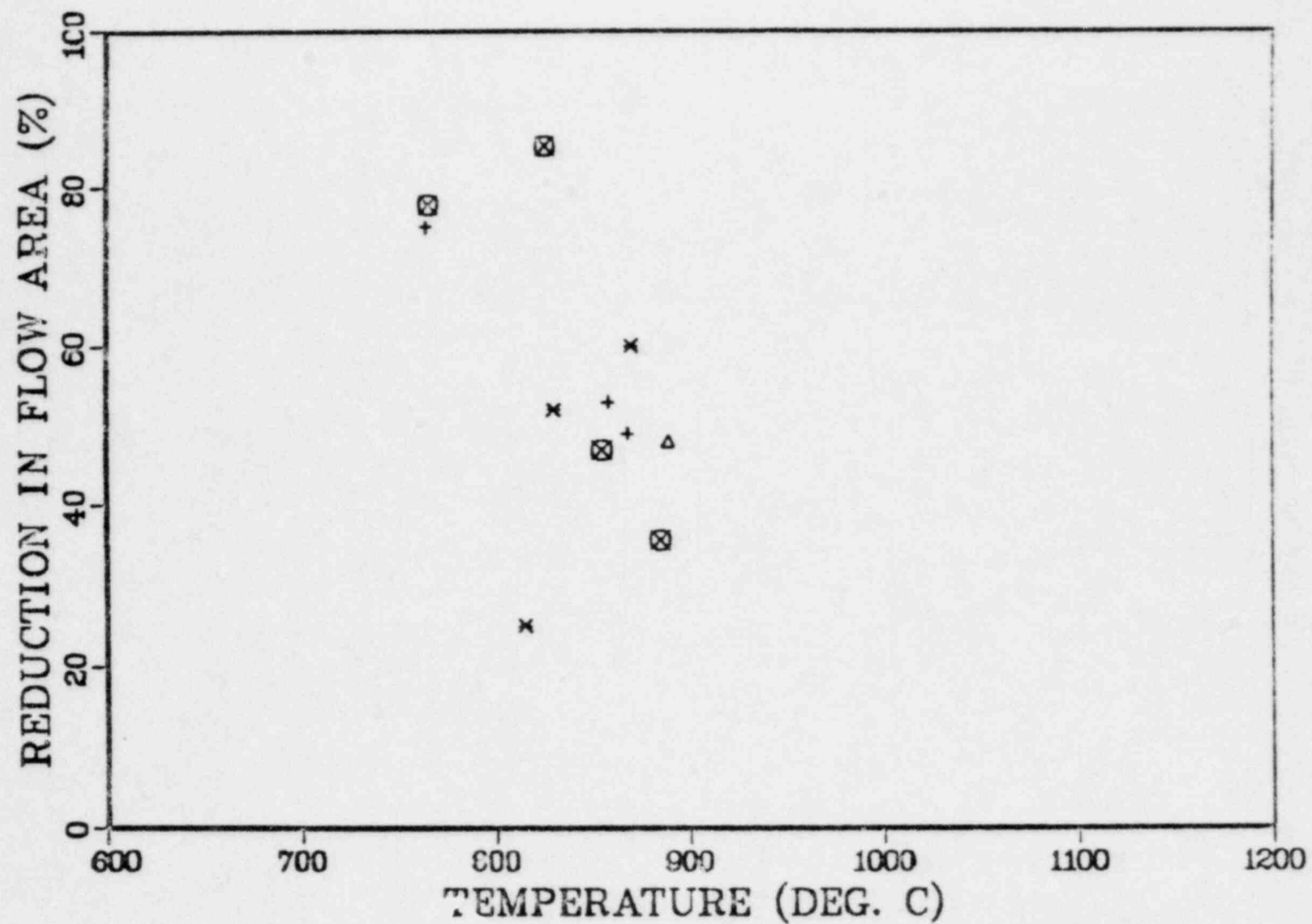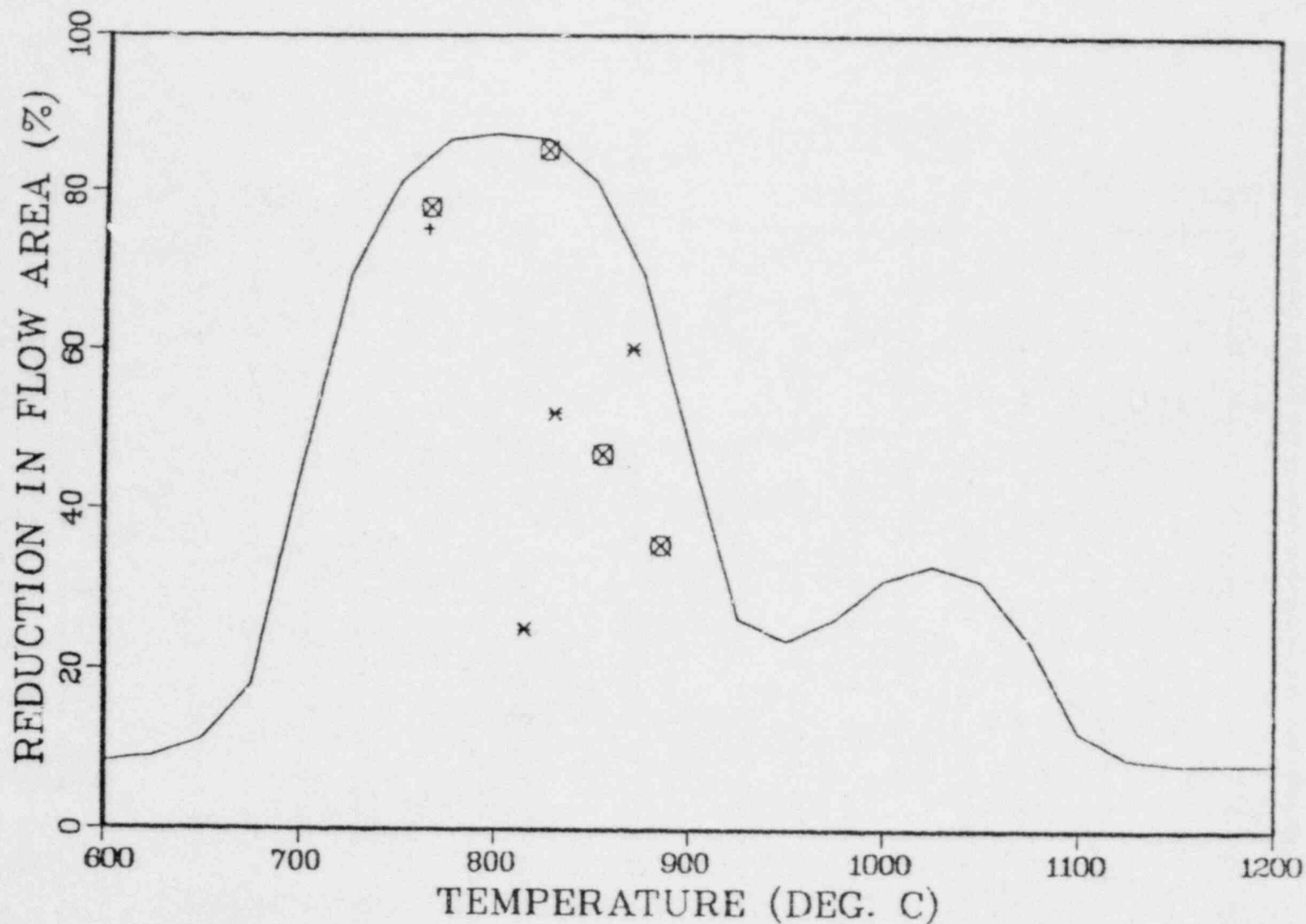
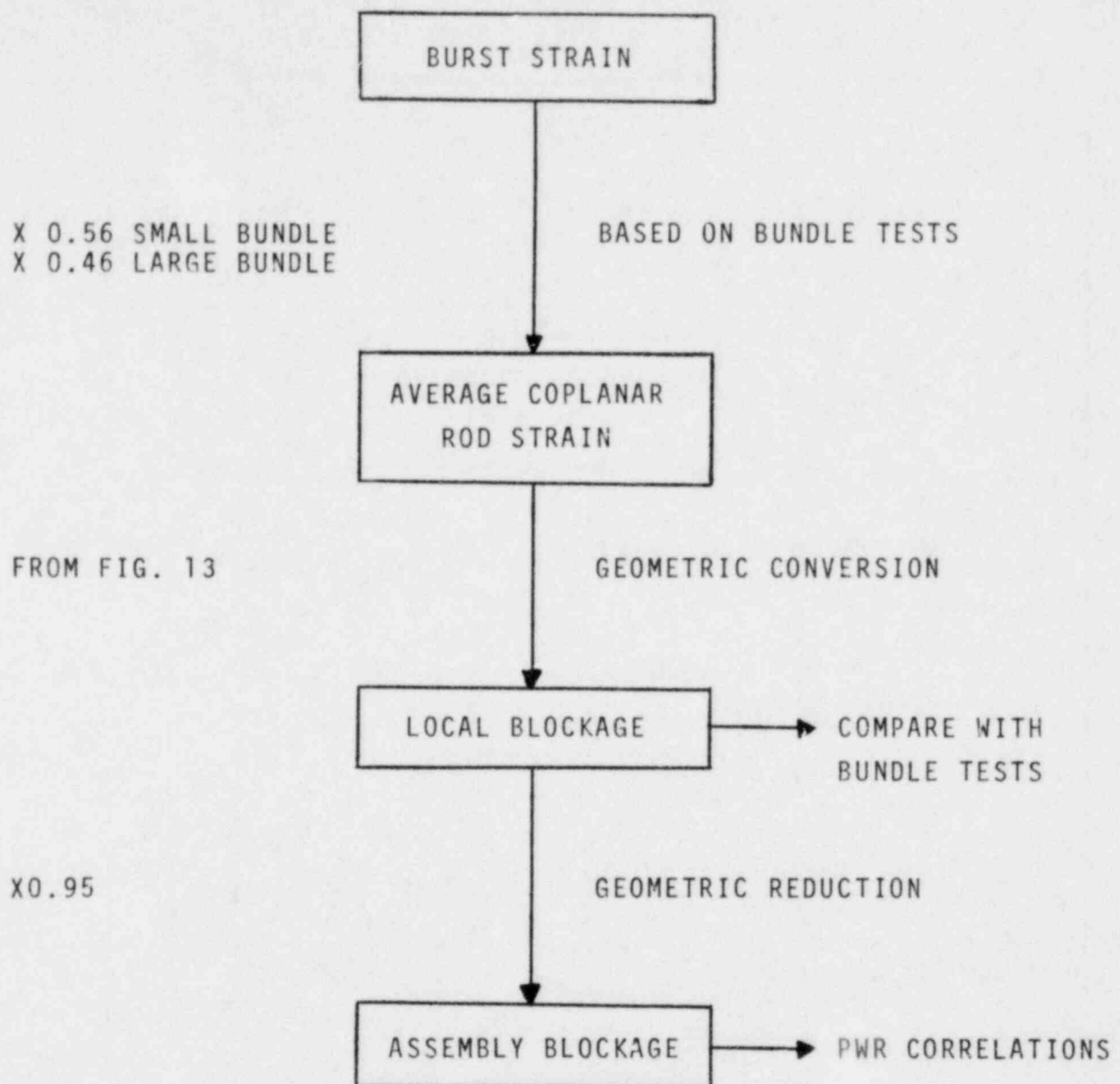Fig. 14    Reduction in local flow area as a function of rupture temperature for internally heated Zircaloy clad bundles in aqueous atmospheres at heating rates less than or equal to 10°C/s.
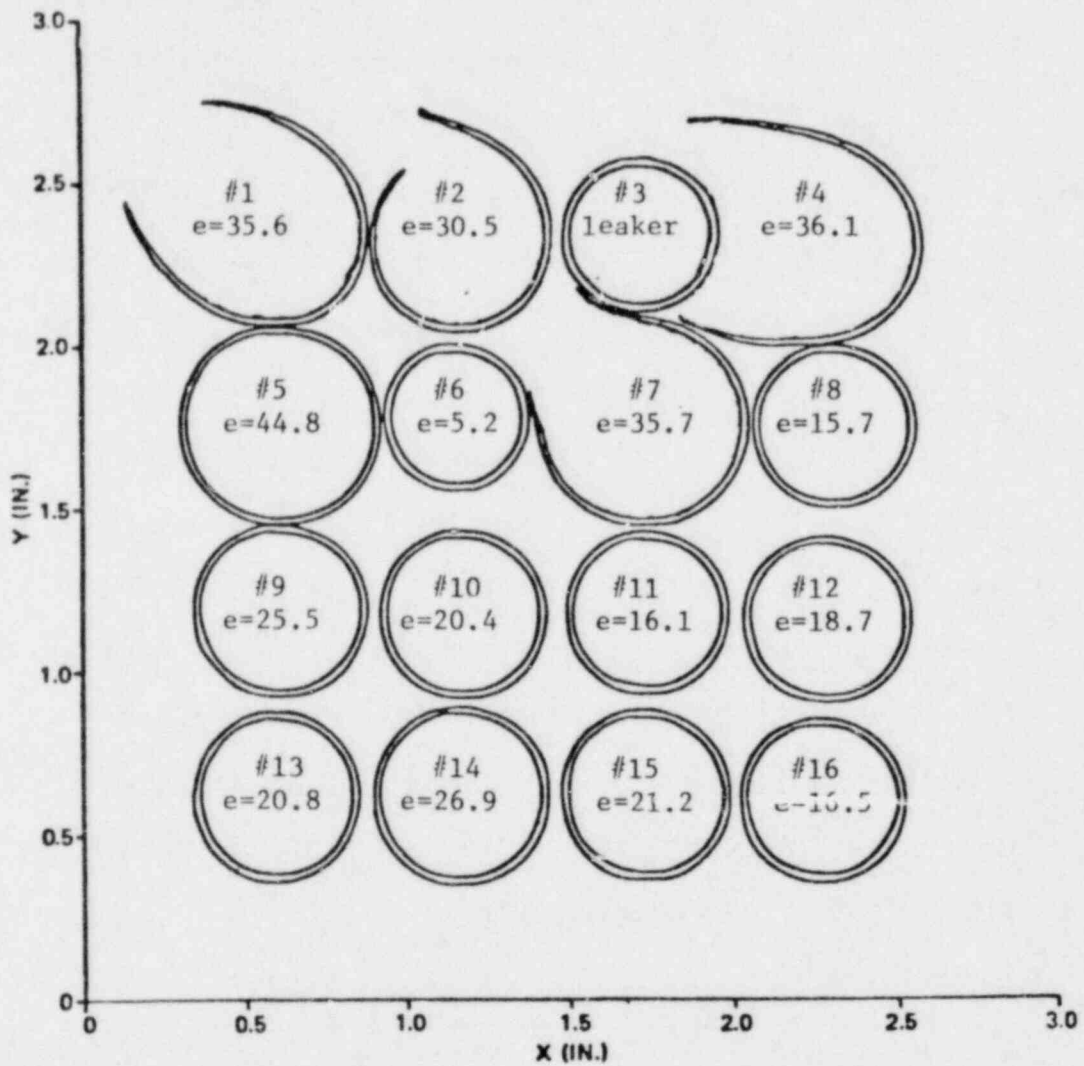
-34-
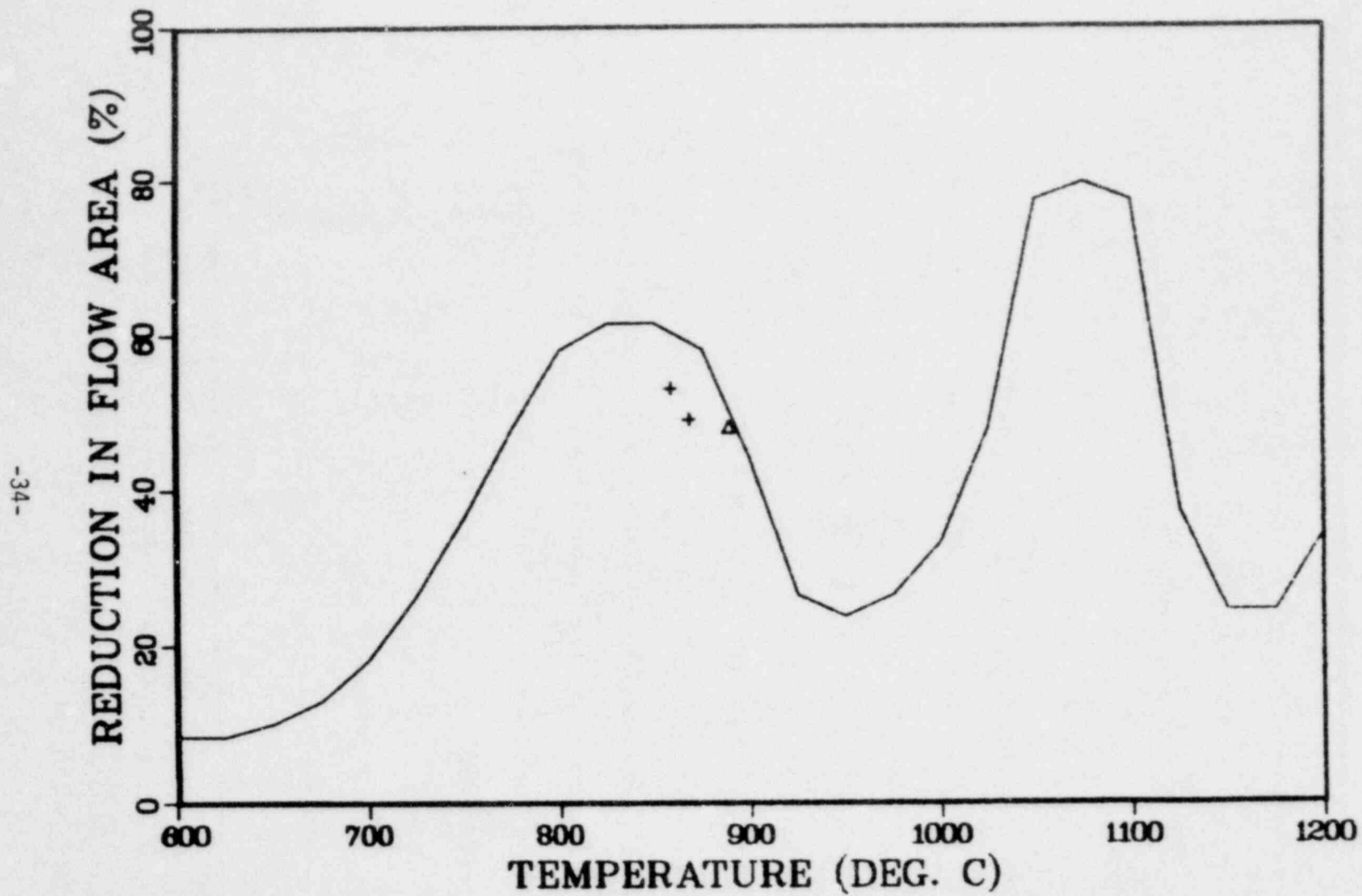
Fig. 15   Reduction in local flow area as a function of rupture temperature for internally
heated Zircaloy clad bundles in aqueous atmospheres at heating rates greater than
or equal to 25°C/s.

in which only 2 of the 9 rods ruptured. Had the cladding temperatures in this bundle test been high enough to perforate all 9 rods (as they should have if their temperatures are to be interpreted as rupture temperatures), then more blockage would have resulted and this datum point on Fig. 14 would have appeared at a higher rupture temperature and a higher degree of blockage -- in closer agreement with the correlation.

The three plus symbols represent Chapman's bundle tests on which we have based one step in the blockage model. The blockages shown here utilize Chapman's "minimum blockage" definition. These points fall below the blockage curves because the individual burst strains fall below the strain curves (Figs. 6 and 7). Bundle B-2 had an unheated shroud, while B-1 and B-3 had heated shrouds, but the shroud temperatures lagged the cladding temperatures too much to have provided a uniform temperature environment. Using more effectively heated shrouds and new internal heaters, Chapman finds larger strains in single rod tests; we would therefore expect the same increases for these bundles.

The upright triangle symbol on Fig. 15 represents the only* in-pile bundle experiment, FRF-1, which was conducted in the TREAT reactor.

_____

*FRF-2 was also an in-reactor bundle experiment, and it produced a maximum flow blockage of 91%; however the FRF-2 cladding rupture temperatures were about 50°C greater than the current licensing peak cladding temperature limit of 1204°C (2200°F) and fall well outside the range of allowable conditions. Data for FRF-2 are also compiled in Appendix A.

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It has been suggested (Ref. 36) that the reported coolant channel blockages in the TREAT experiment were atypically high because the bundle design employed a hexagonal fuel array with triangular coolant subchannels that were smaller than square subchannels of the same pitch. This objection would be justified had the blockages reported in Refs. 25-26 and 37 been obtained by a direct measurement of the coolant subchannel areas. However, this method was not used; rather, the individual rod cross-sectional areas were determined, summed, and subtracted from the total area encompassed by the sleeve (shroud). For this experiment, the ratio of the area inside the sleeve to the total undeformed fuel rod cross-sectional area was 2.16, which is approximately equal to that ratio for a PWR (2.24). Therefore, we believe that the TREAT blockage is directly applicable to our blockage analysis.

We have reviewed the experimental conditions of all of the out-of-reactor tests and found that none of the tests included all of the features of a PWR that enhance strain and blockage. We therefore believe that the correlations in Figs. 14 and 15 are best estimates for the conditions that are conducive to blockage even though those correlations bound the available data.

For large-size PWR assemblies, the flow blockage correlations are shown in Fig. 16, and the coordinates for these curves are tabulated in Appendix B. Boiling water reactors, with shrouded fuel assemblies and upper core sprays, are sufficiently different from PWRs that assembly flow blockage correlations are not used in ECCS analysis.

-36-
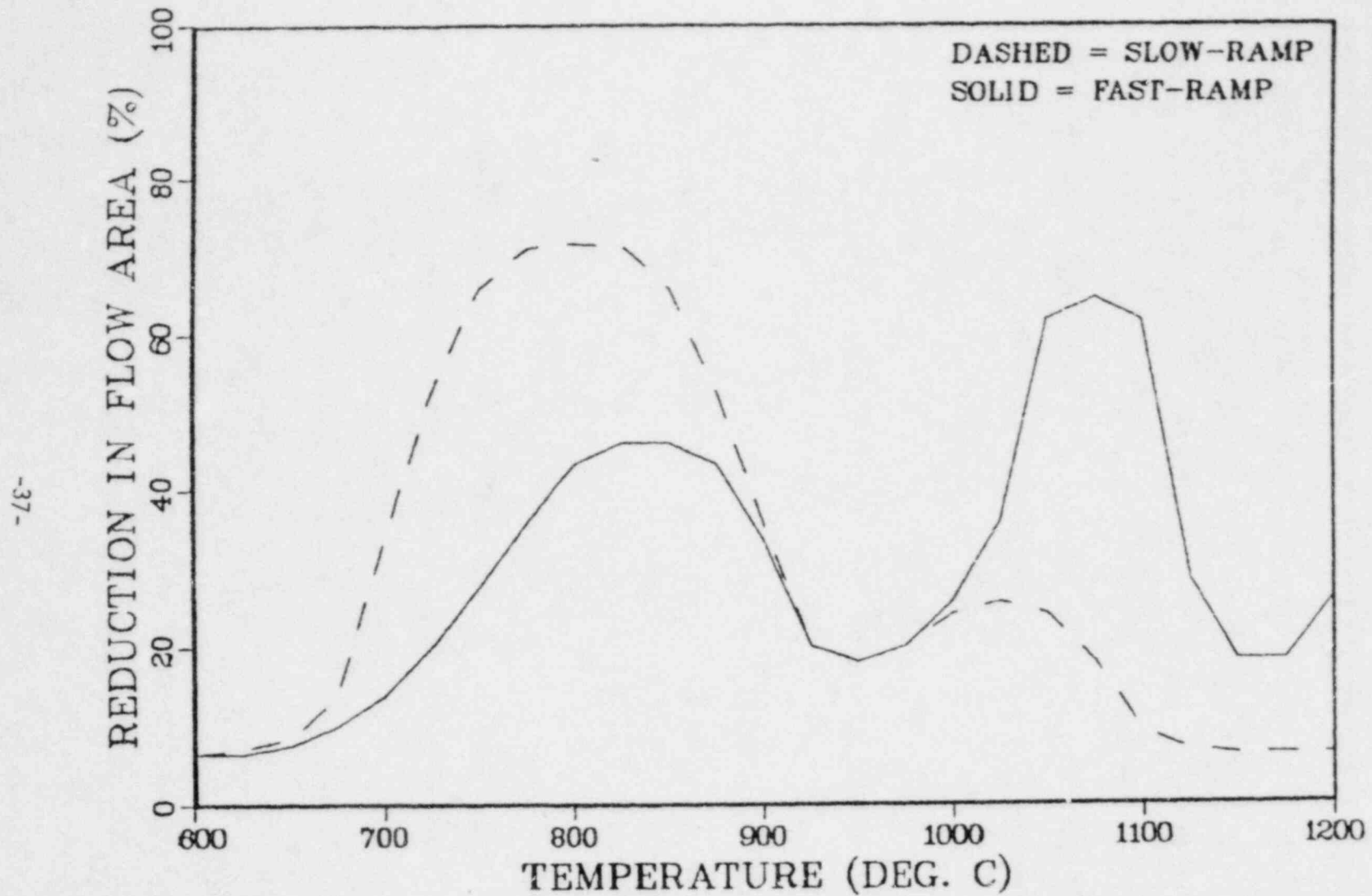
Fig. 16    Reduction in PWR assembly flow area as a function of rupture temperature and ramp
rate.

4.   COMPARISON WITH PREVIOUS MODELS

### 4.1  Early AEC Model

The Water Reactor Evaluation Model (WREM) (Ref. 38) was developed
in 1975 as an audit model, but it was not used as a standard for
acceptance of vendor licensing models.  Figures 17-21 show the WREM
models compared with the present correlations.  If it is kept in
mind that fast heating rates were envisioned for WREM applications,
agreement is good for rupture temperature (Fig. 17).  The WREM
burst strain model (Figs. 18-19) agrees well with our correlations
at temperatures near 950°C, but does not exhibit the first beta-
phase superplastic peak seen in the fast-ramp data (Fig. 19).  The
WREM assembly flow blockage model (Figs. 20-21) conservatively
envelopes the present fast- and slow-ramp correlations, and it
shows excellent agreement with the present slow-ramp alpha-phase
peak.

### 4.2  Babcock & Wilcox

Figures 22-28 show the B&W models compared with the present correlations.
The B&W rupture temperature model (Fig. 22) agrees very well with
the present correlation for slow-ramps and therefore conservatively
predicts the incidence of rupture for higher ramp rates.  Two sets
of burst strain models are used by Babcock & Wilcox; the THETA
model (Figs. 23-24) is used for single-rod analyses whereas the
CRAFT model (Figs. 25-26) is used for blockage analyses.  The B&W
burst strains in general do not agree with our current understanding

Fig. 17    WREM model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate.

Fig. 18    WREM model and slow-ramp correlation of circumferential burst strain as a function of rupture temperature.

Fig. 19    WREM model and fast-ramp correlation of circumferential burst strain as a function of rupture temperature.

Fig. 20    WREM model and slow-ramp correlation of reduction in assembly flow area as a function of rupture temperature.

Fig. 21    WREM model and fast-ramp correlation of reduction in assembly flow area as a
function of rupture temperature.

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Fig. 22    B&W model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate.

Fig. 23    B&W THETA model and slow-ramp correlation of circumferential burst strain as a function of rupture temperature.

Fig. 24    B&W THETA model and fast-ramp correlation of circumferential burst strain as a
function of rupture temperature.

Fig. 25  B&W CRAFT model and slow-ramp correlation of circumferential burst strain as a function of rupture temperature.

Fig. 26    B&W CRAFT model and fast-ramp correlation of circumferential burst strain as a function of rupture temperature.

Fig. 27    B&W model and slow-ramp correlation of reduction in assembly flow area as a function of rupture temperature.
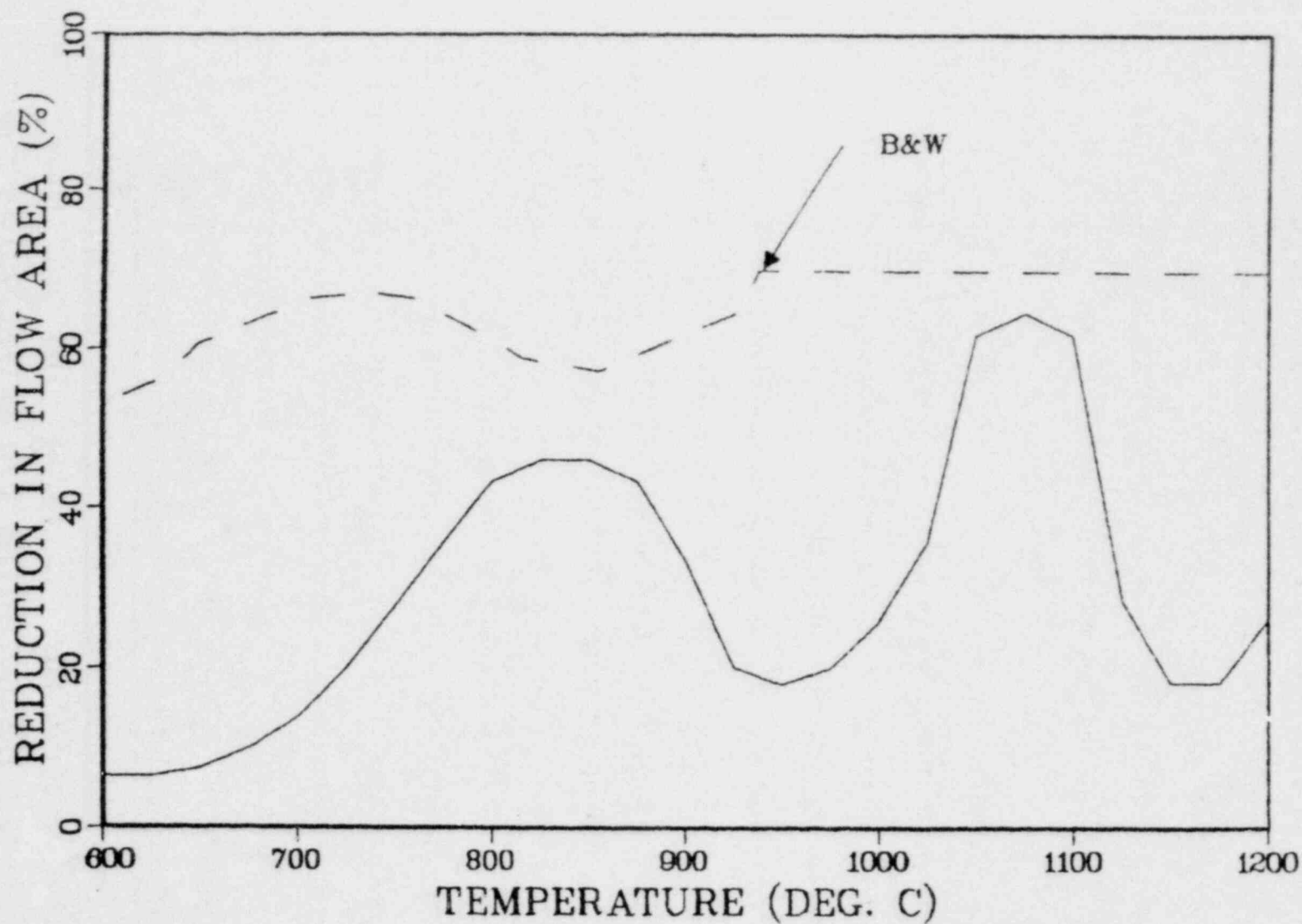
Fig. 28   R&W model and fast-ramp correlation of reduction in assembly flow area as a function of rupture temperature.

and significantly underpredict the degree of swelling at some temperatures. Although the B&W flow blockage model (Figs. 27-28) does not agree well with the current correlations, the B&W model overpredicts blockage for fast ramps over a wide range of temperatures and only modestly underpredicts blockage for slow ramps in the alpha-phase superplastic region.

## 4.3 Combustion Engineering

Figures 29-33 show the presently approved Combustion Engineering models. Figure 29 shows the rupture temperature model, which is, in general, in only fair agreement with our present correlation. On the basis of the very low strains and blockages shown in Figs. 30-33, the NRC required C-E in March 1978 to reevaluate their ECCS analyses using larger burst strains and flow blockages (Ref. 39). Using some of the guidance that we provided for the strain and blockage models, C-E submitted the proposed models (Ref. 40) shown in Figs. 34-37 (the beta-phase region was not modeled in accordance with our guidelines). In that submittal C-E proposed that previously unused conservatisms in heat-transfer models compensated for the larger strains and blockages in their proposed models. The proposed C-E models are in fair agreement with the present correlations over wide temperature ranges (Figs. 35-36), but they underpredict the present correlations at low temperatures for slow ramps (Fig. 34) and over some temperature ranges for fast ramps (Fig. 35 and 37).

-51-

-52-

Fig. 29    C-E model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate.

Fig. 30    C-E model and slow-ramp correlation of circumferential burst strain as a function of rupture temperature.

Fig. 31    C-E model and fast-ramp correlation of circumferential burst strain as a function of rupture temperature.

Fig. 32    C-E model and slow-ramp correlation of reduction in assembly flow area as a function of rupture temperature.

Fig. 33    C-E model and fast-ramp correlation of reduction in assembly flow area as a function of rupture temperature.

Fig. 34   C-E proposed model and slow-ramp correlation of circumferential burst strain as a function of rupture temperature.

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Fig. 35    C-E proposed model and fast-ramp correlation of circumferential burst strain as a function of rupture temperature.

Fig. 36    C-E proposed model and slow-ramp correlation of reduction in assembly flow area as a function of rupture temperature.

Fig. 37   C-E proposed model and fast-ramp correlation of reduction in assembly flow area
as a function of rupture temperature.

## 4.4 Westinghouse

Figures 38-43 show the Westinghouse models. For small-break analyses, Westinghouse accounts for ramp-rate effects on rupture temperature in a manner similar to our present correlation (Fig. 38); the large-break Westinghouse model is similar to the present fast-ramp correlation (Fig. 39). The Westinghouse burst-strain model (shown in Figs. 40-41) is similar to the WREM burst-strain model but approximates the present correlation only at temperatures near 950°C (Figs. 40-41). The Westinghouse flow blockage model (shown in Figs. 42-43) exhibits significantly smaller blockages than either WREM or the present correlation over large temperature ranges for both slow- and fast-ramp rates.

## 4.5 General Electric

Figures 44-46 show the General Electric models. Figure 44 exhibits substantial underprediction of the incidence of rupture at high stresses (pressure differentials), but the high stress portion of this curve is not relevant for BWR fuel rods since they are pressurized to a much lesser extent than PWR fuel rods.

For temperatures above 925°C and for slow ramps, the conditions appropriate for BWRs, the GE burst strain model predicts strains as much as 10 to 15% lower than the present correlation (Fig. 45). The fast-ramp comparison in Fig. 46 is probably not relevant, but is shown for completeness.
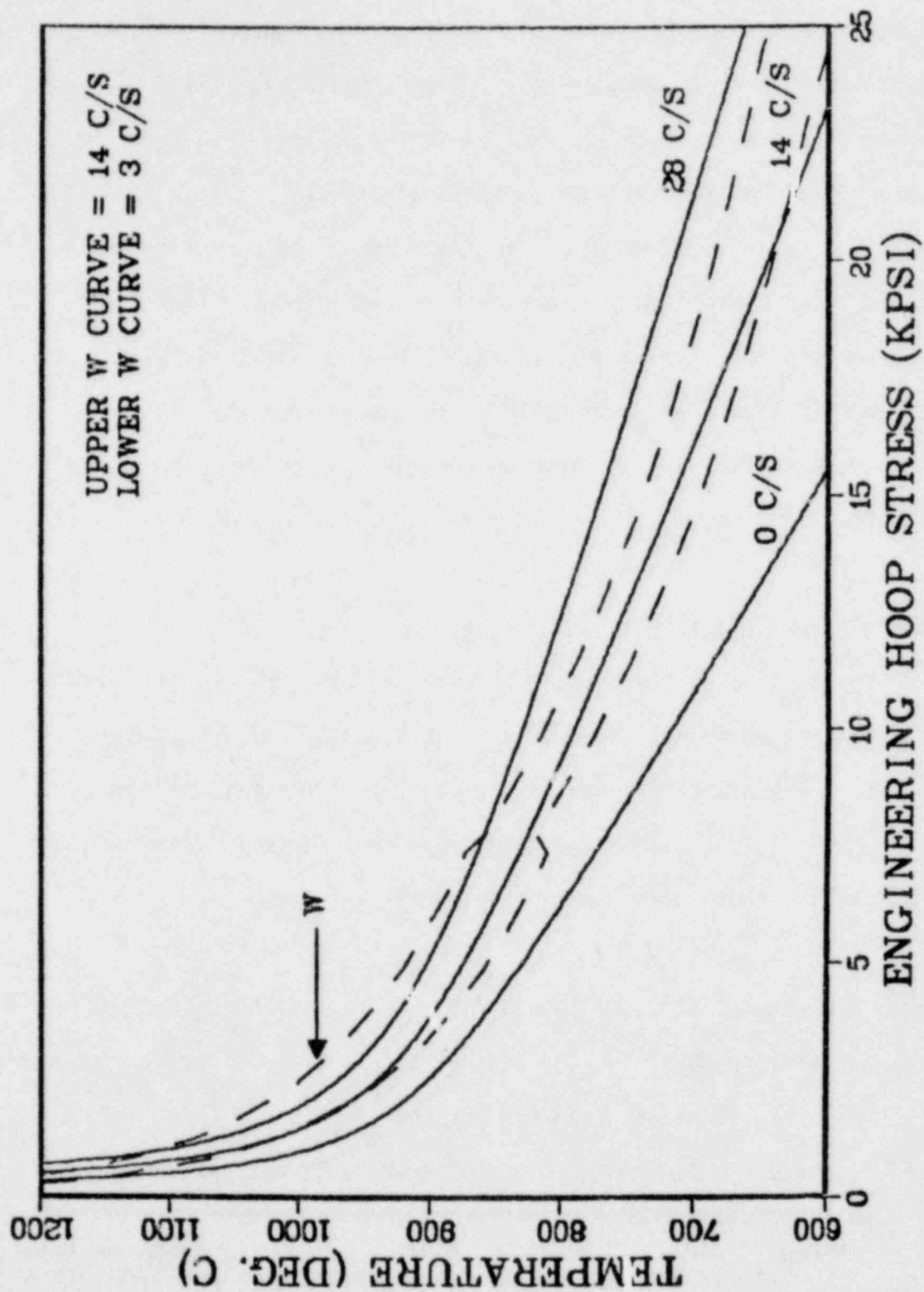
Fig. 38    W small-break model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate.
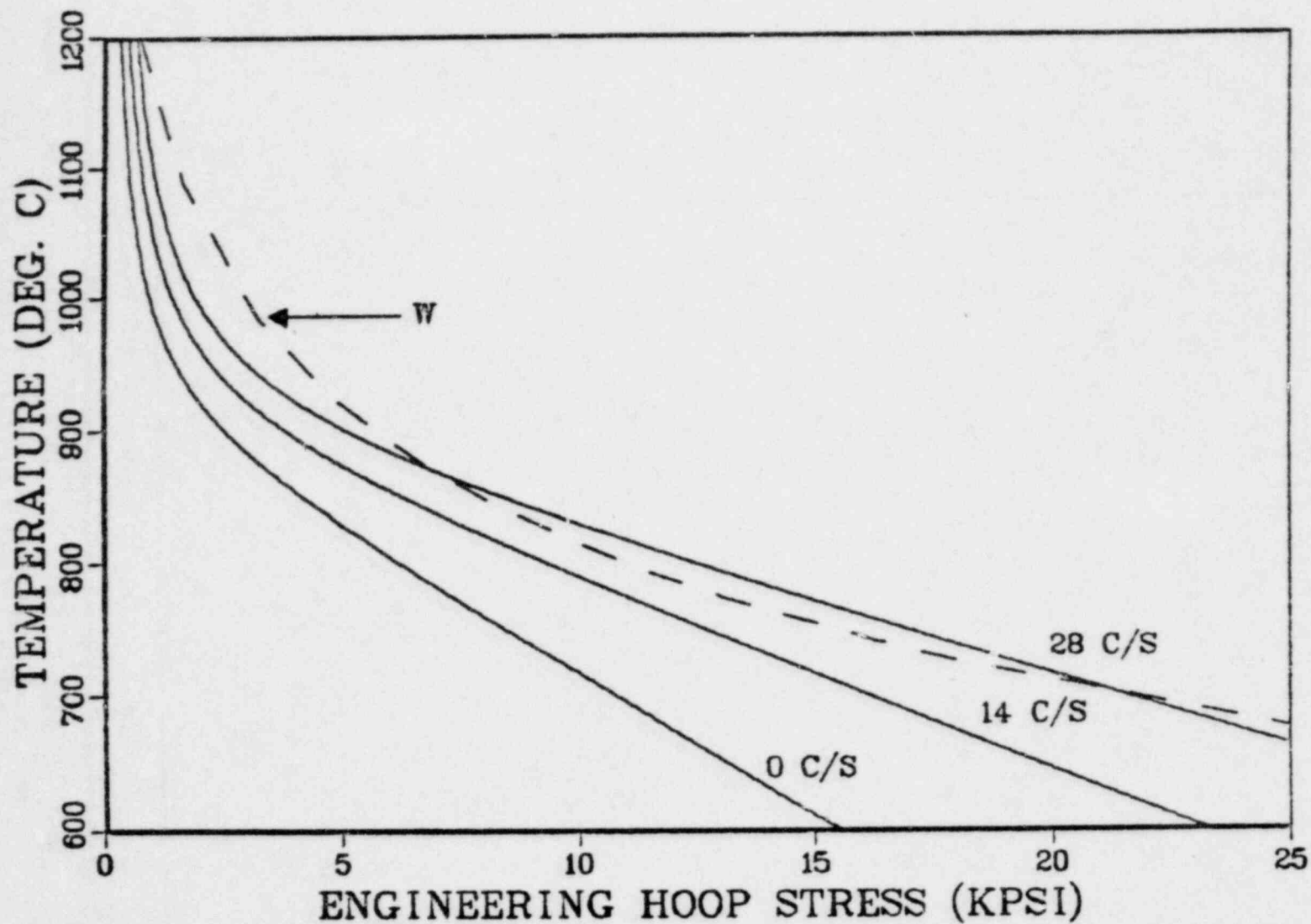
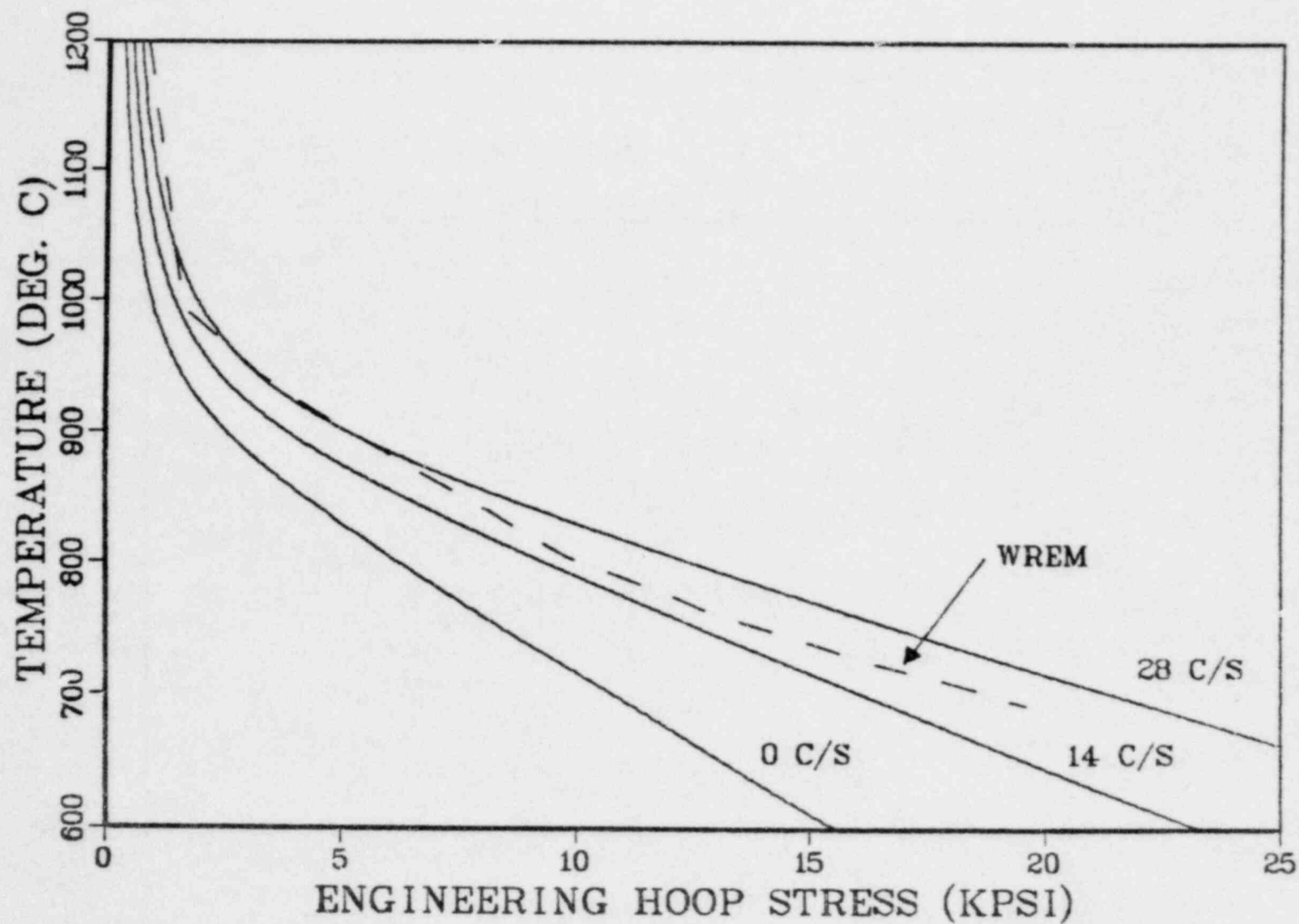Fig. 39     W large-break model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate.
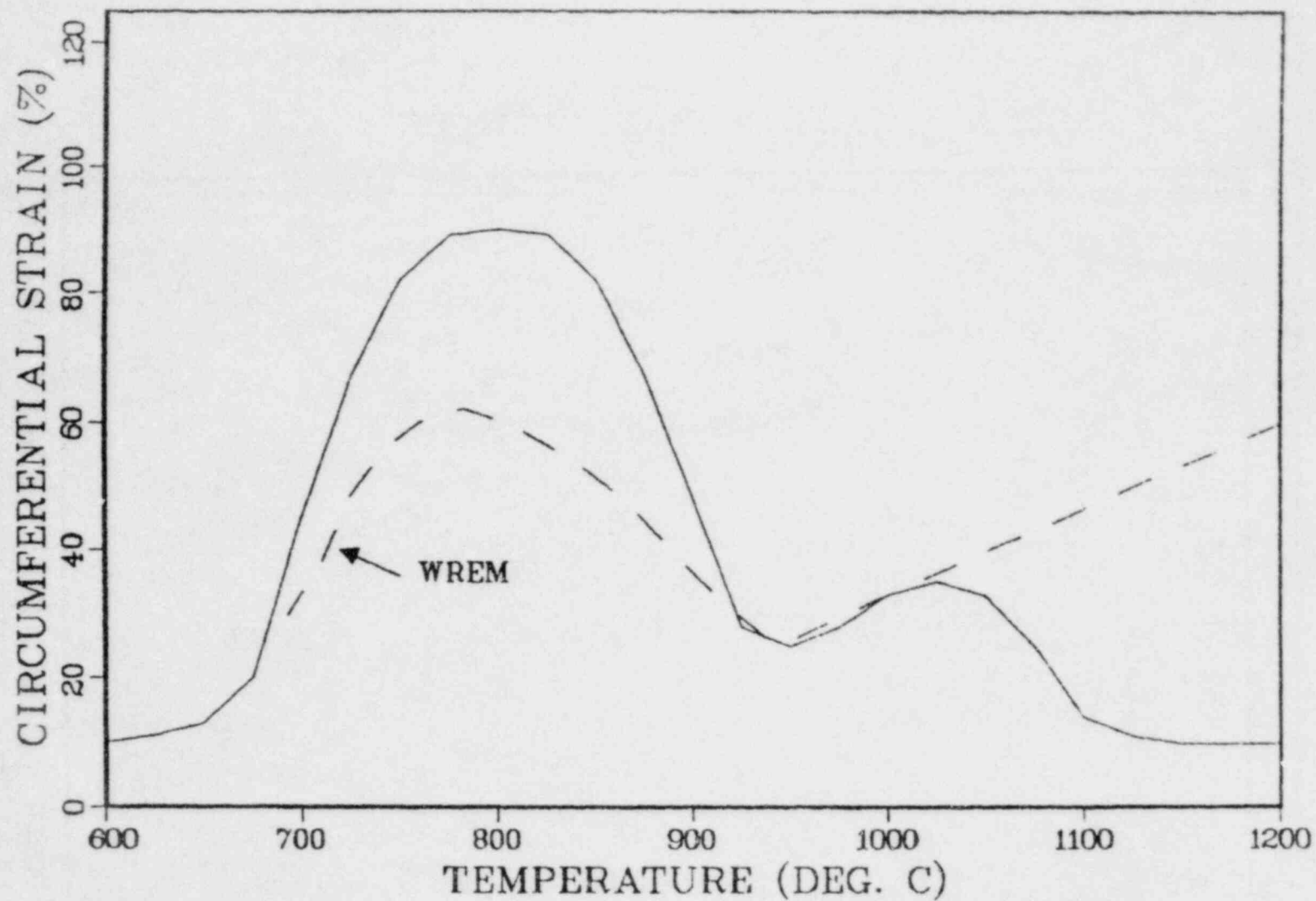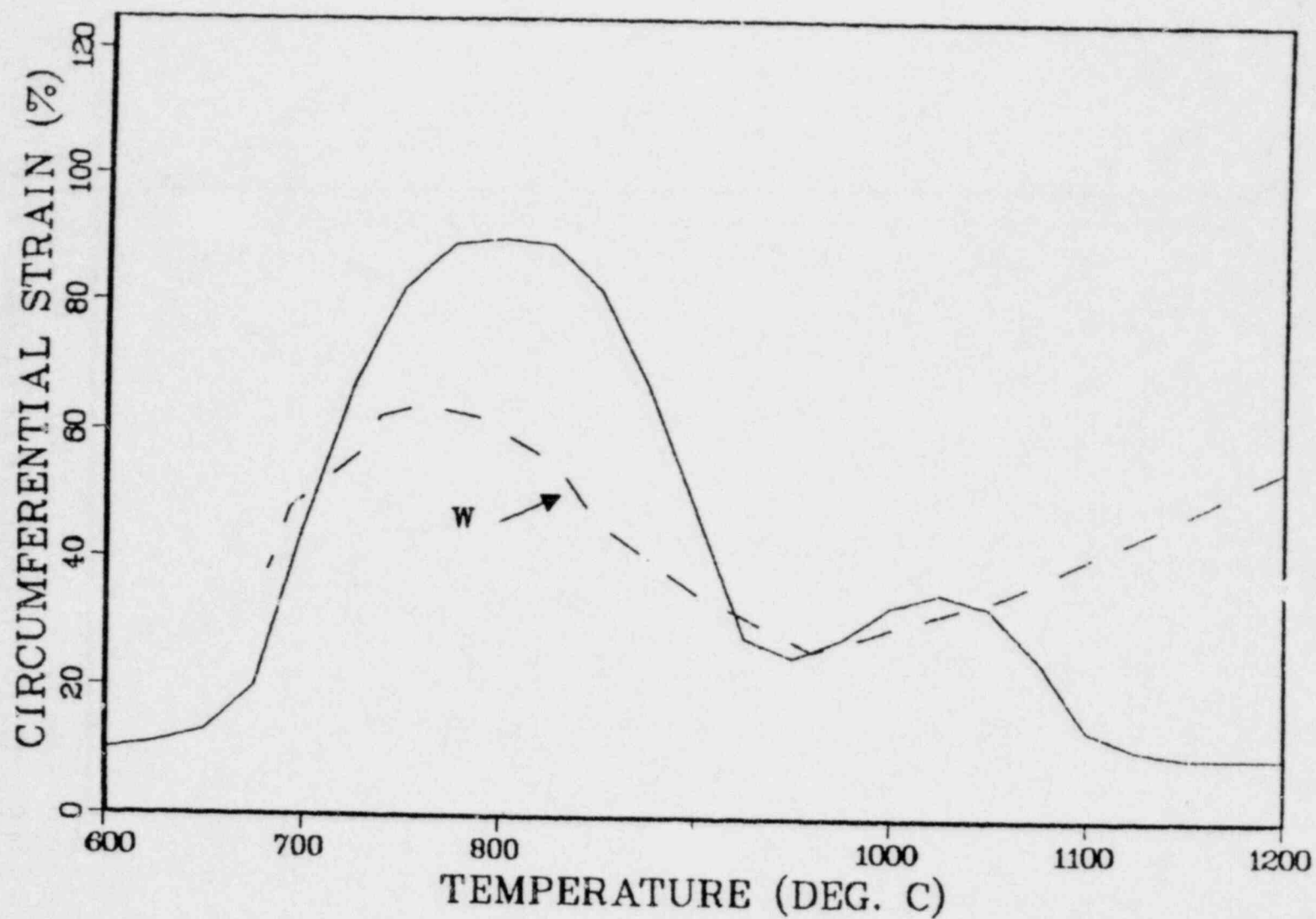
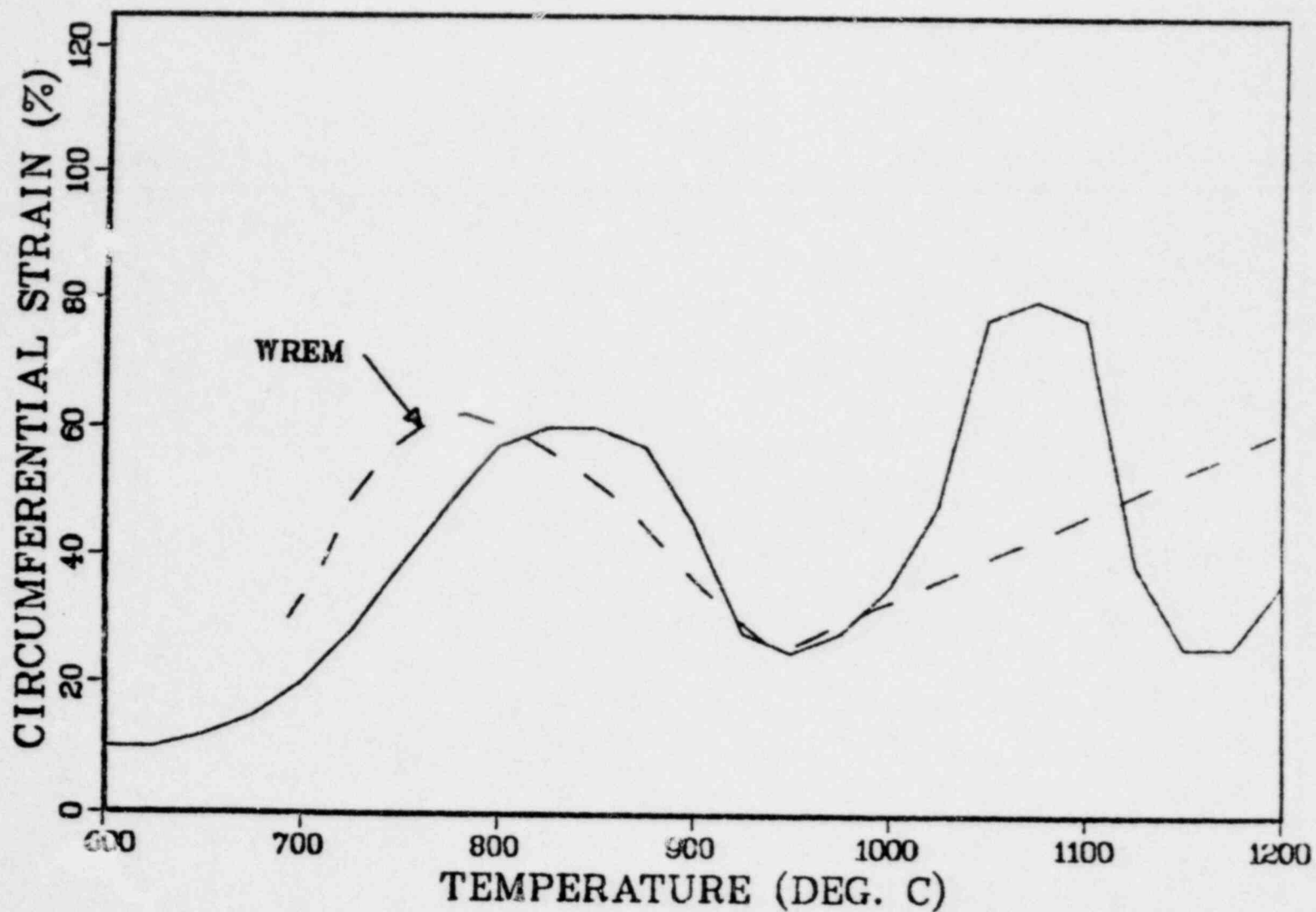Fig. 40    W model and slow-ramp correlation of circumferential burst strain as a function of rupture temperature.
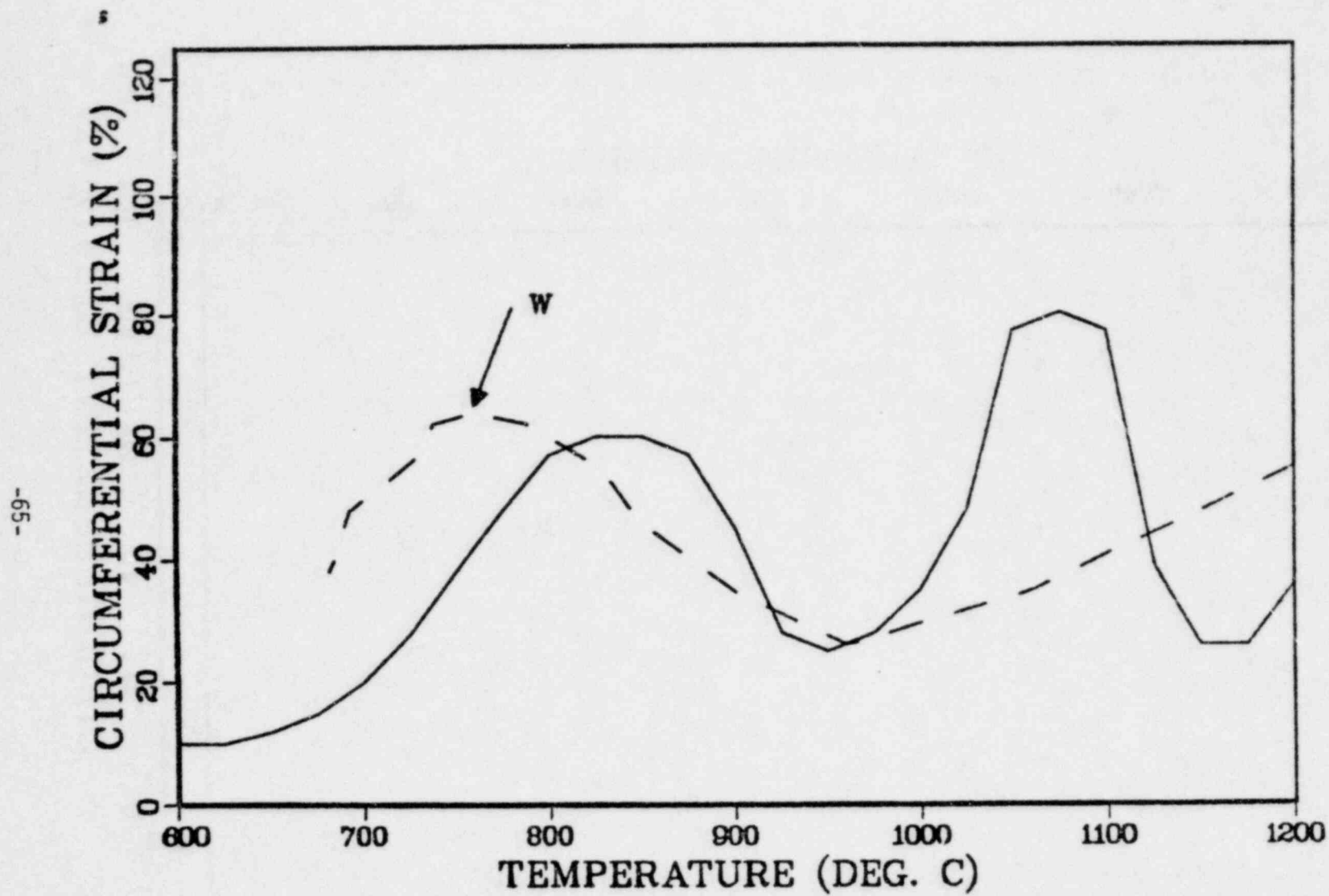
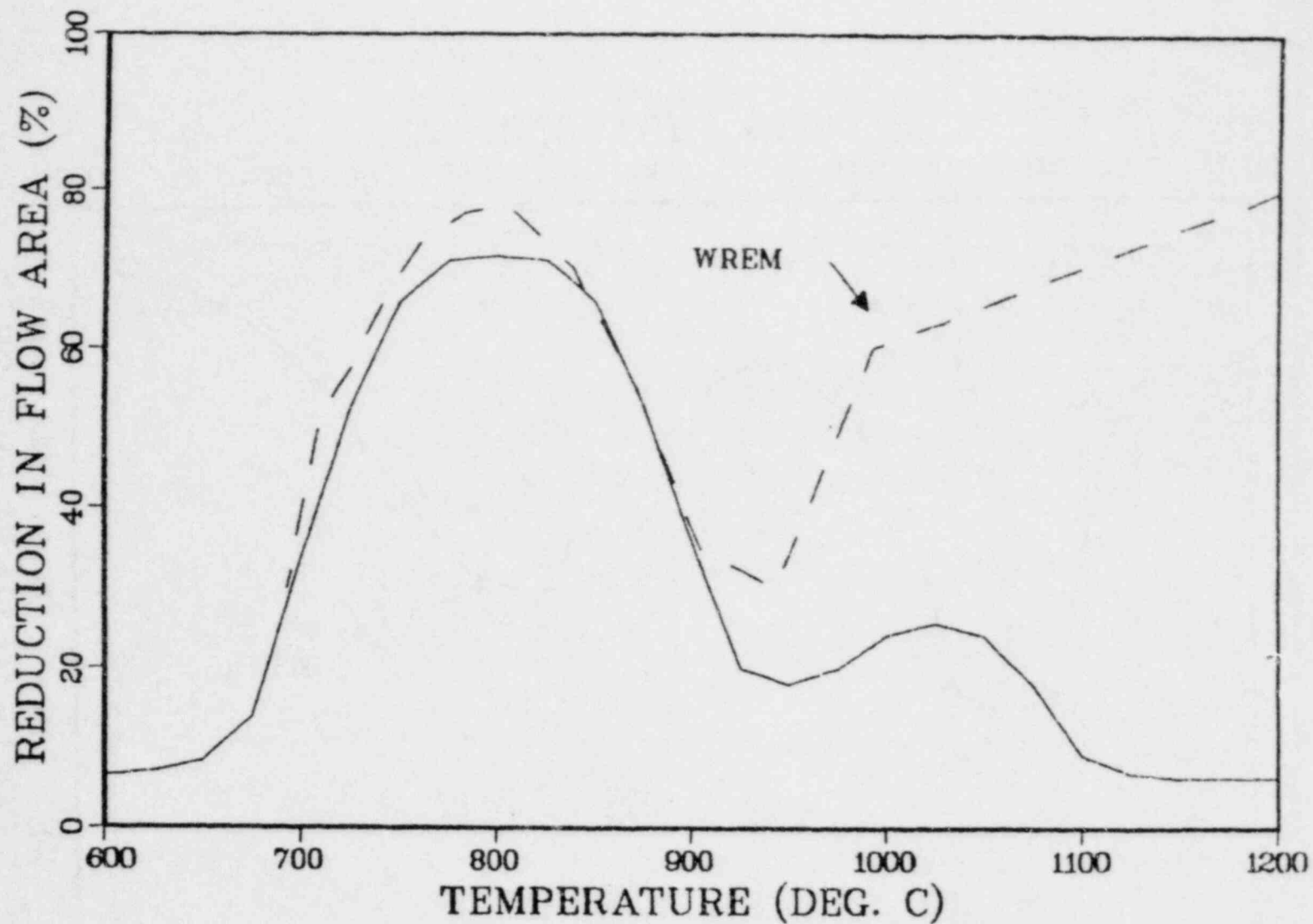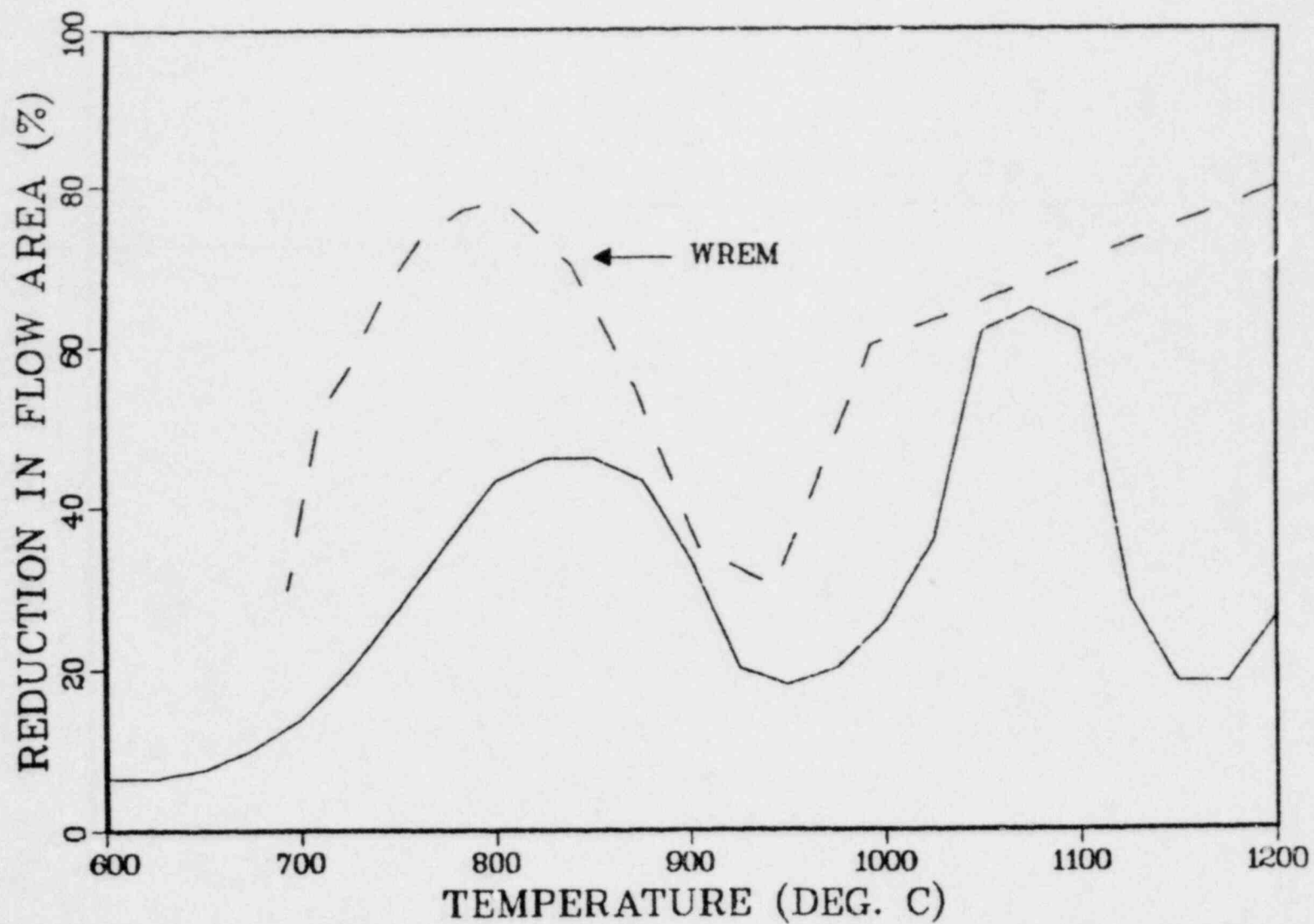Fig. 41    W model and fast-ramp correlation of circumferential burst strain as a function
of rupture temperature.
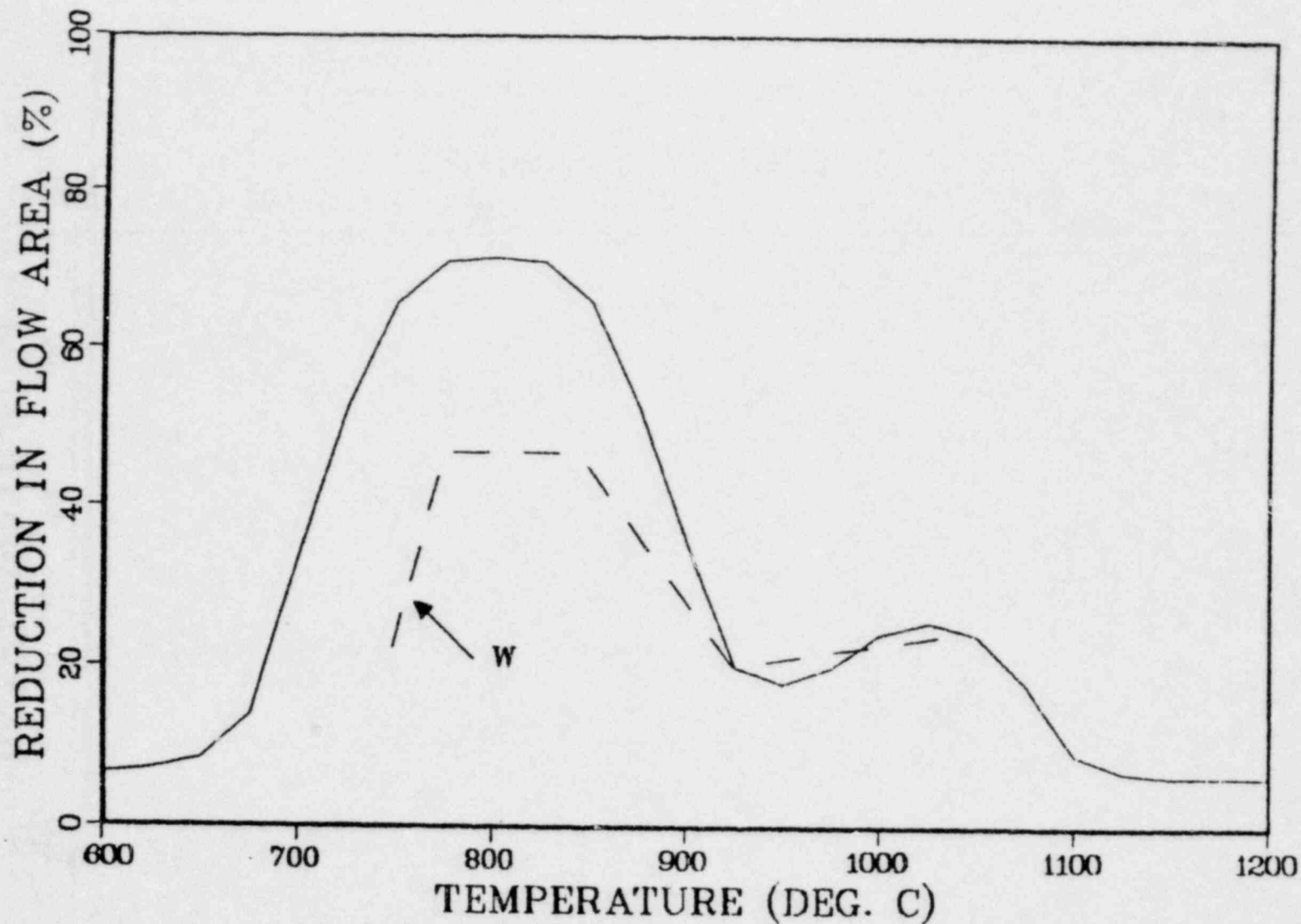
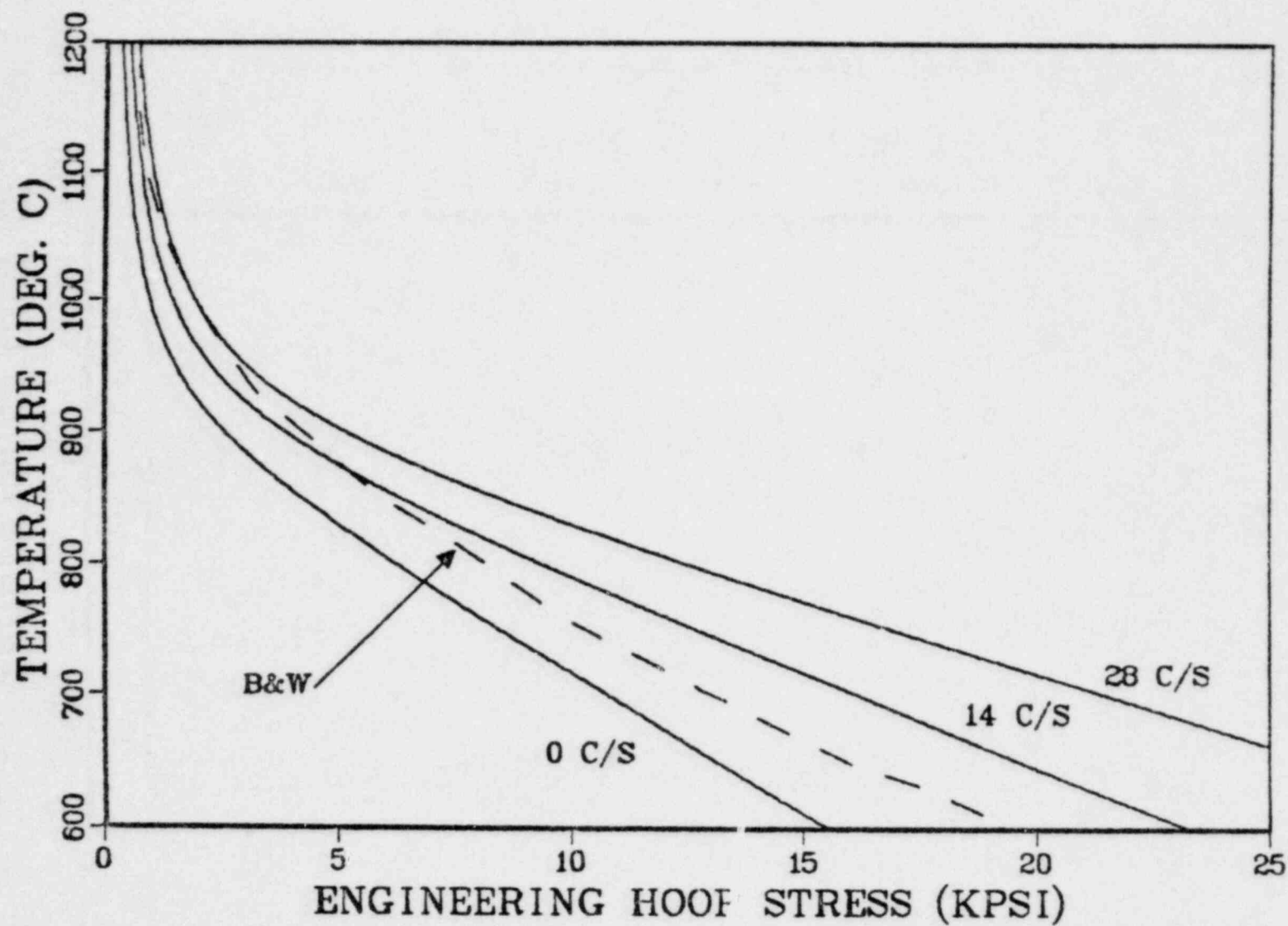Fig. 42    W model and slow-ramp correlation of reduction in assembly flow area as a function of rupture temperature.

-67-
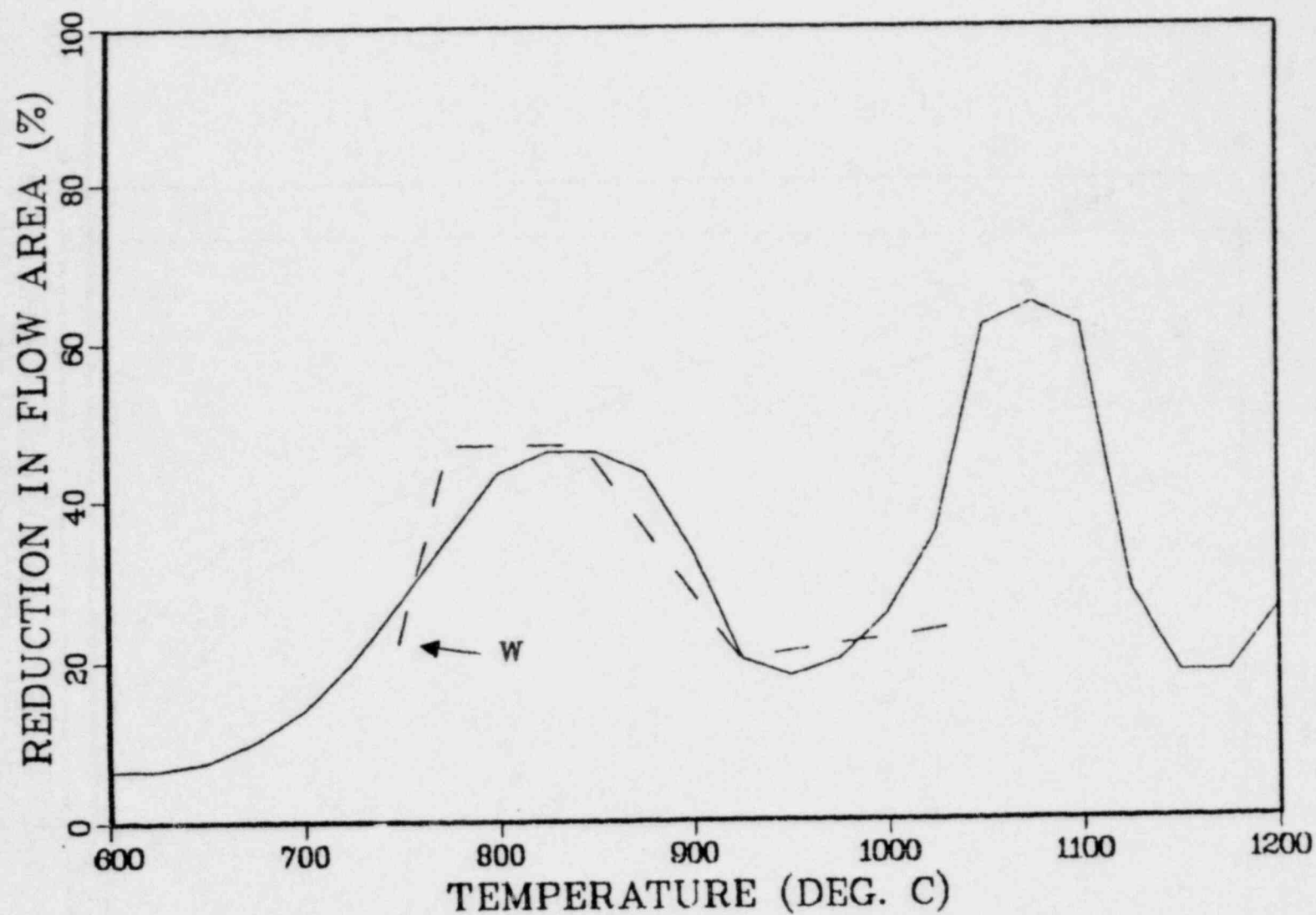
Fig. 43     W model and fast-ramp correlation of reduction in assembly flow area as a function of rupture temperature.
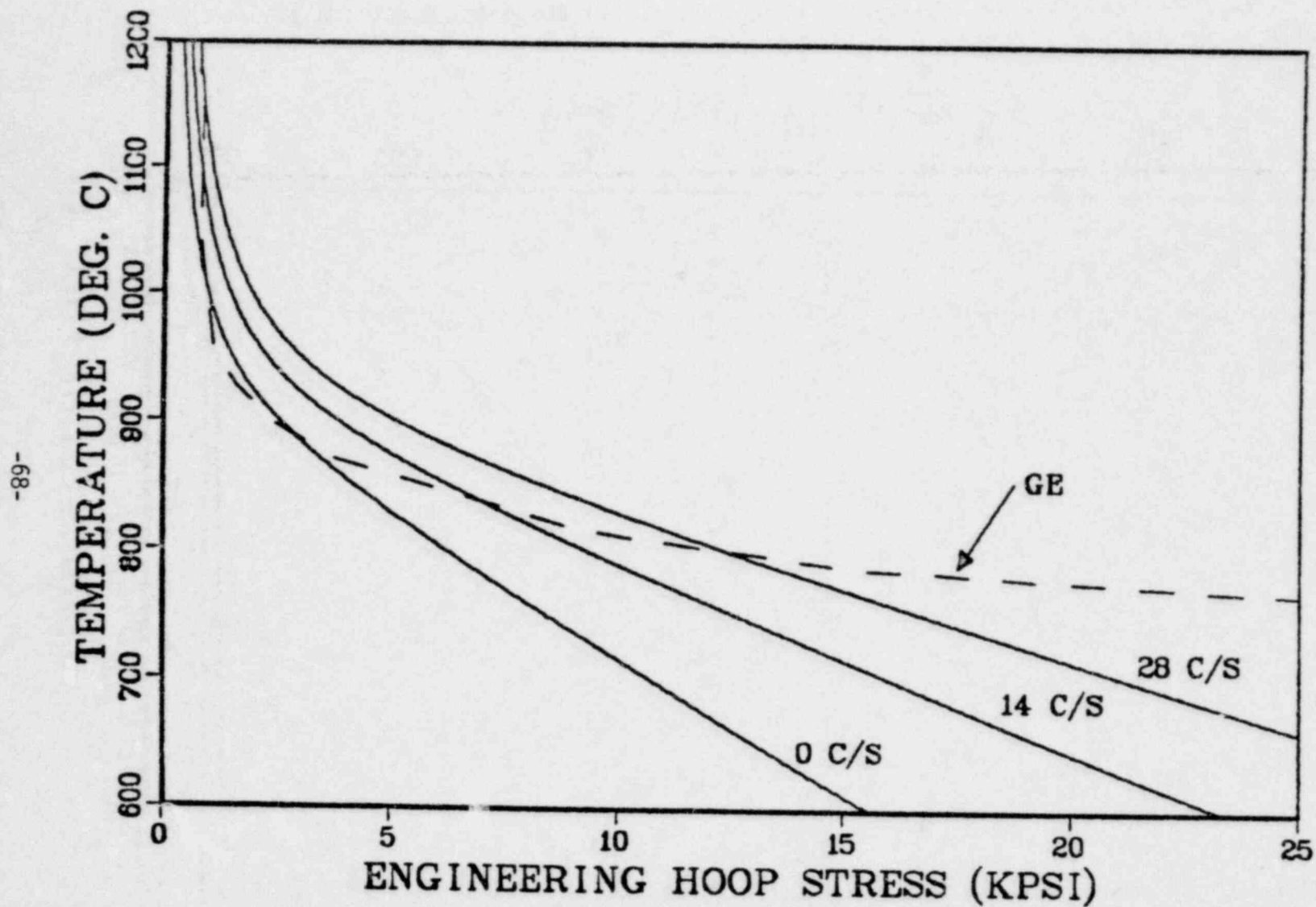
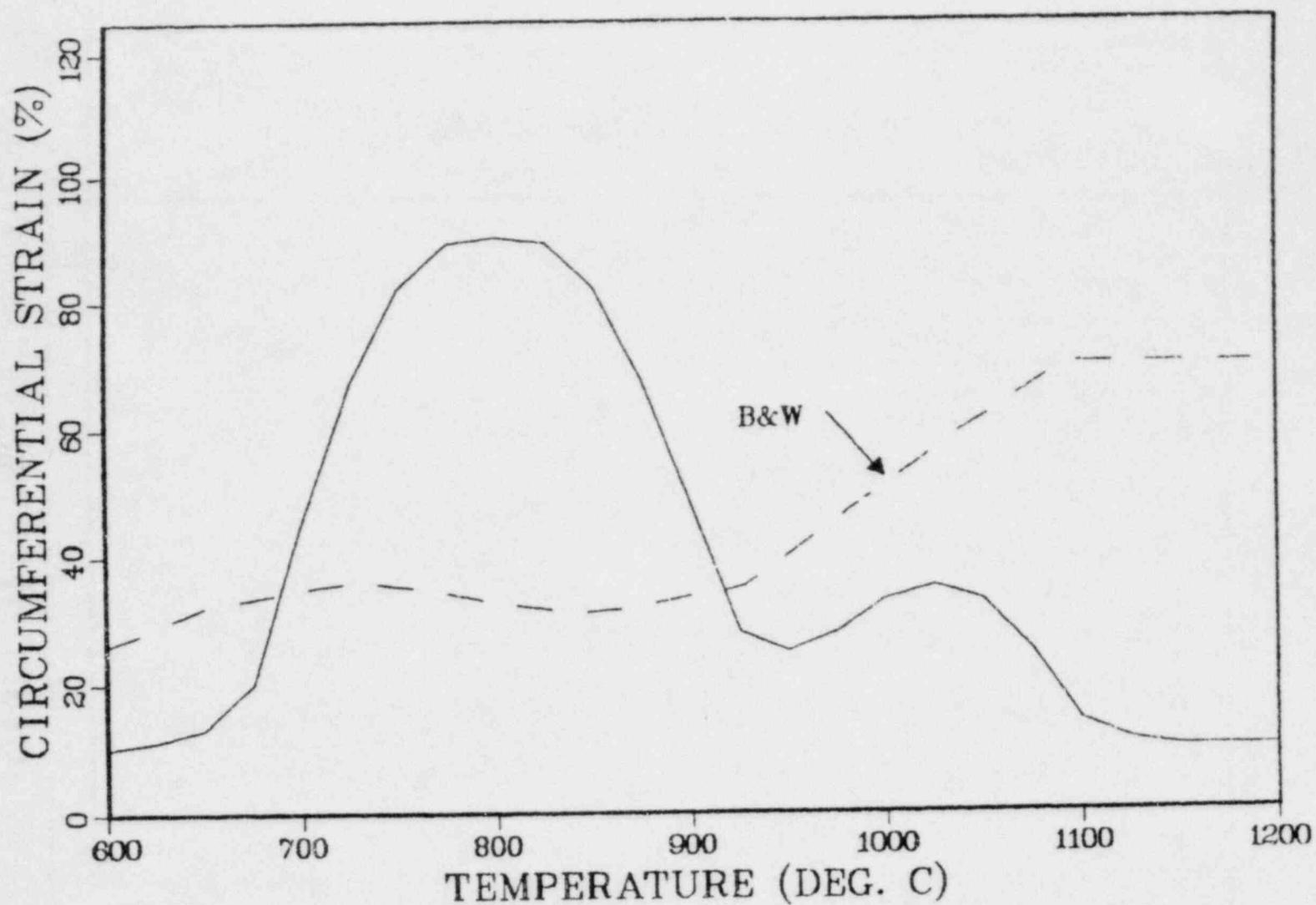Fig. 44   GE model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate.
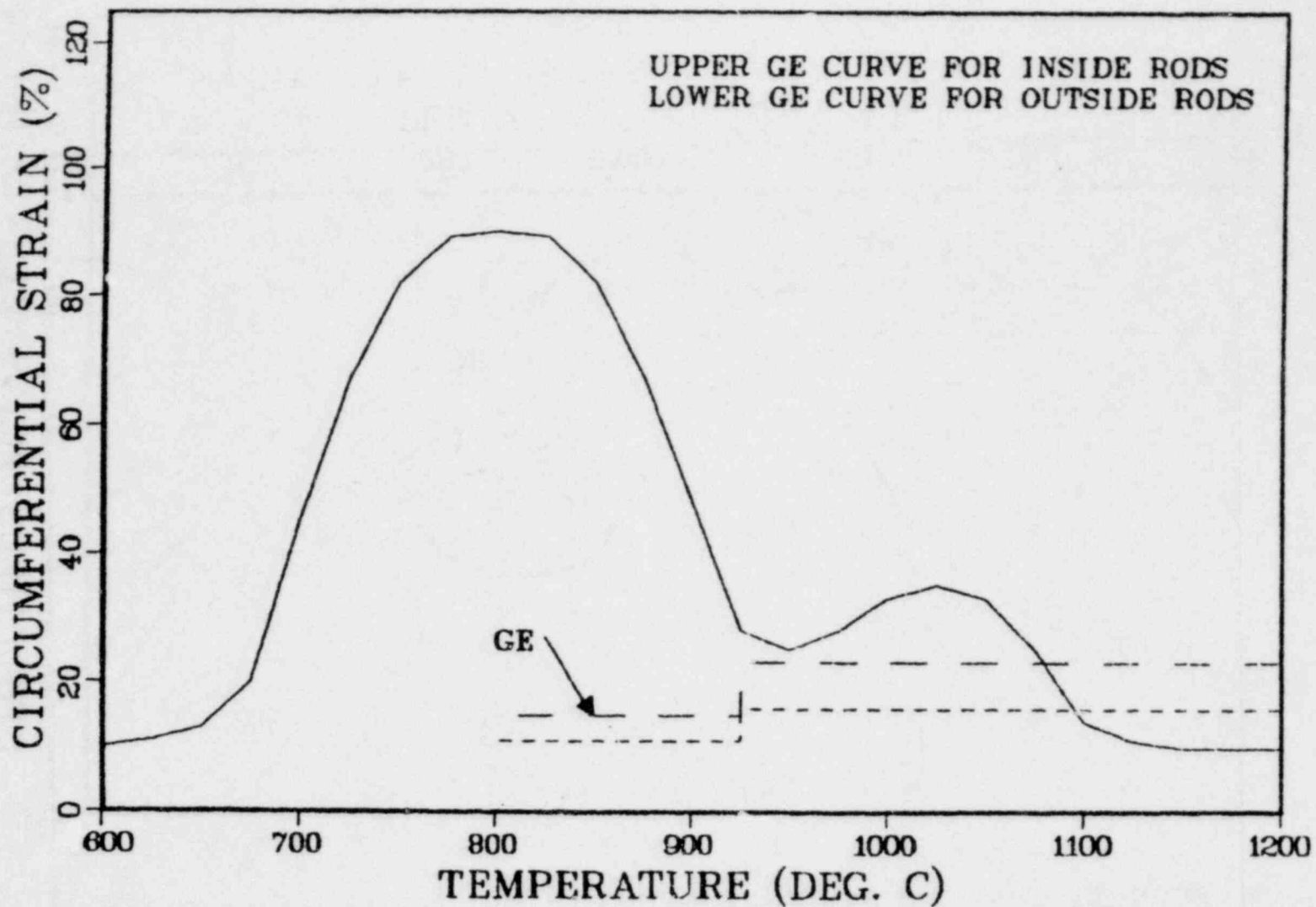
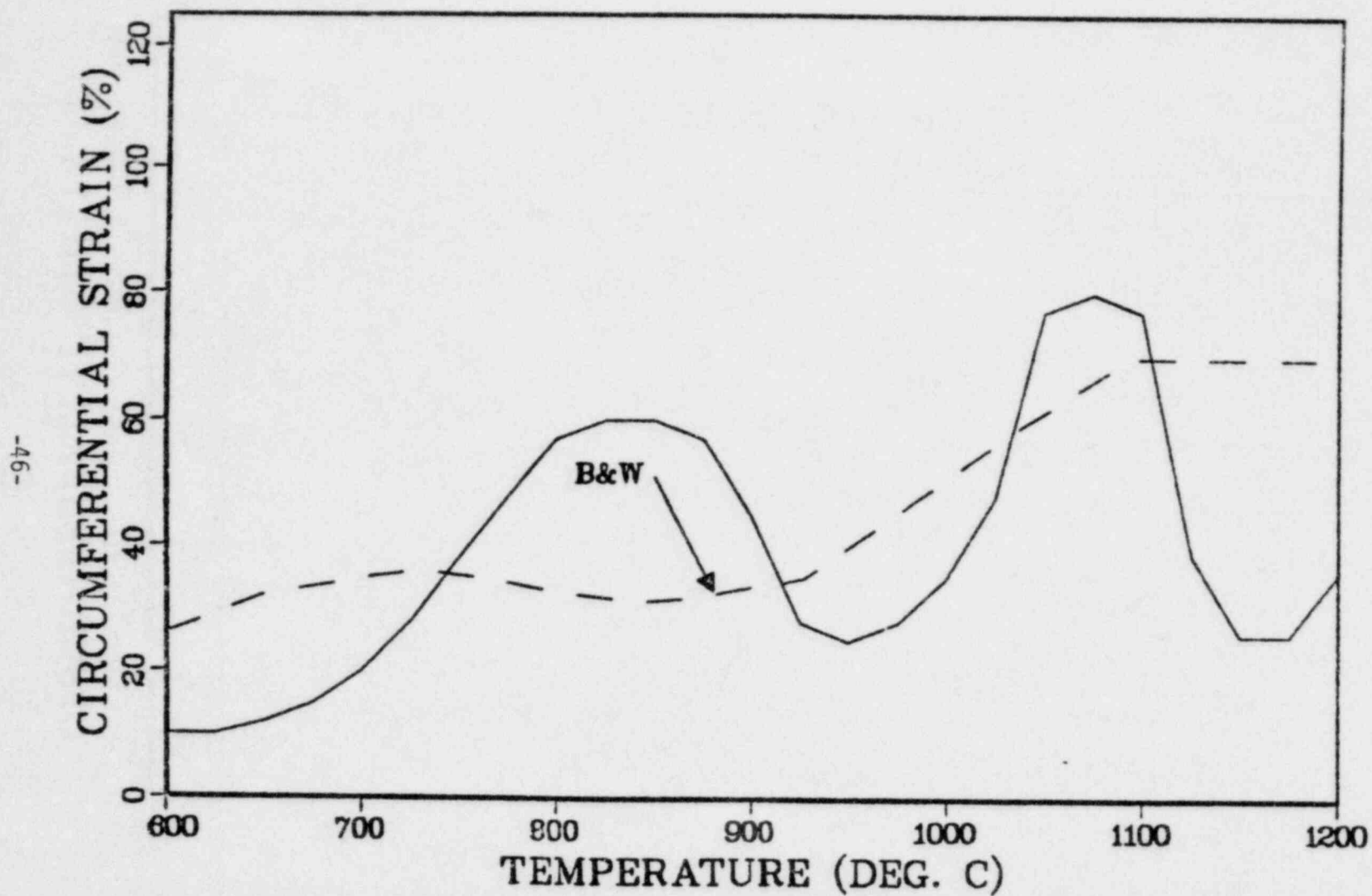Fig. 45    GE model and slow-ramp correlation of circumferential burst strain as a function of rupture temperature.
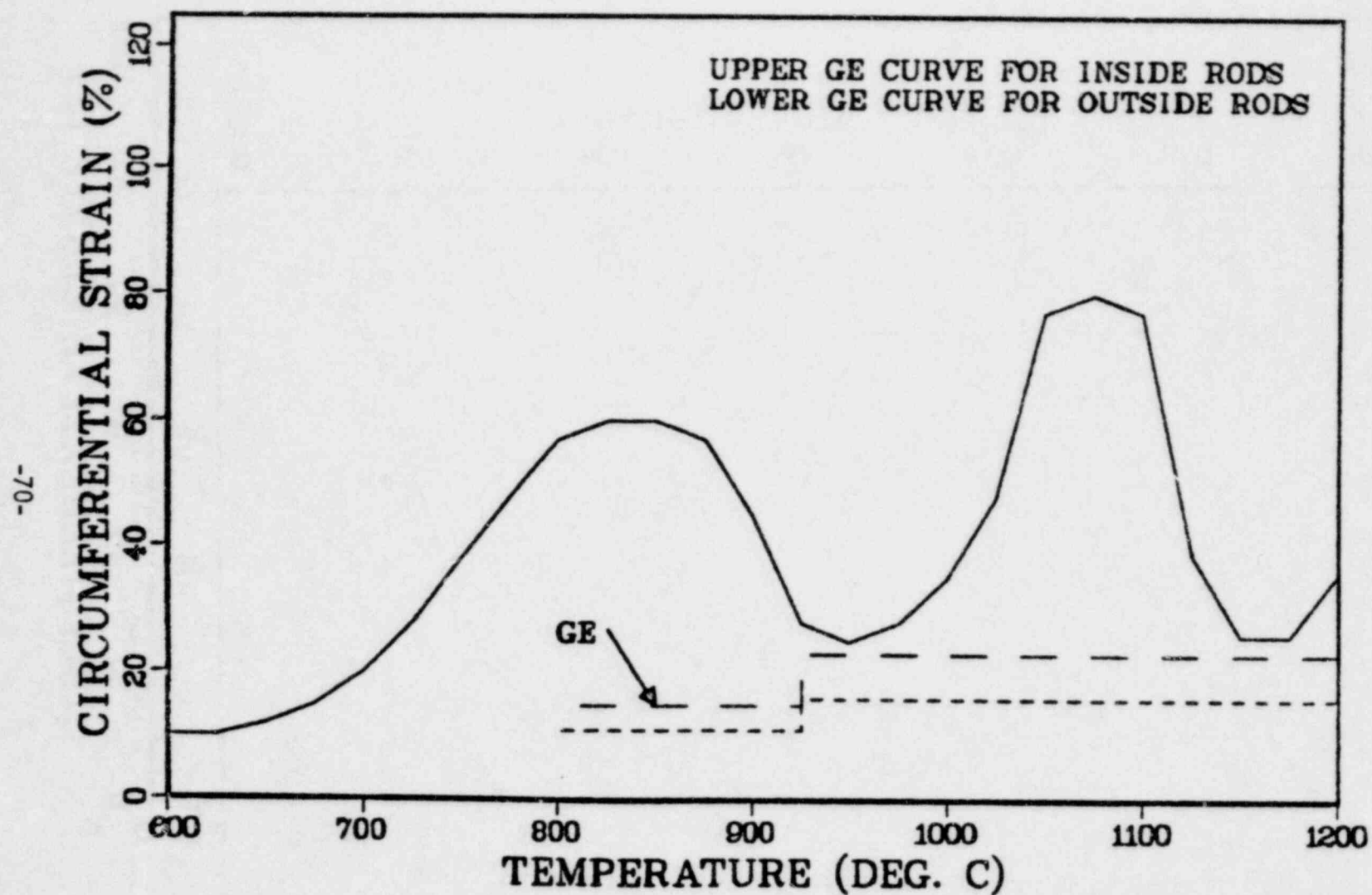
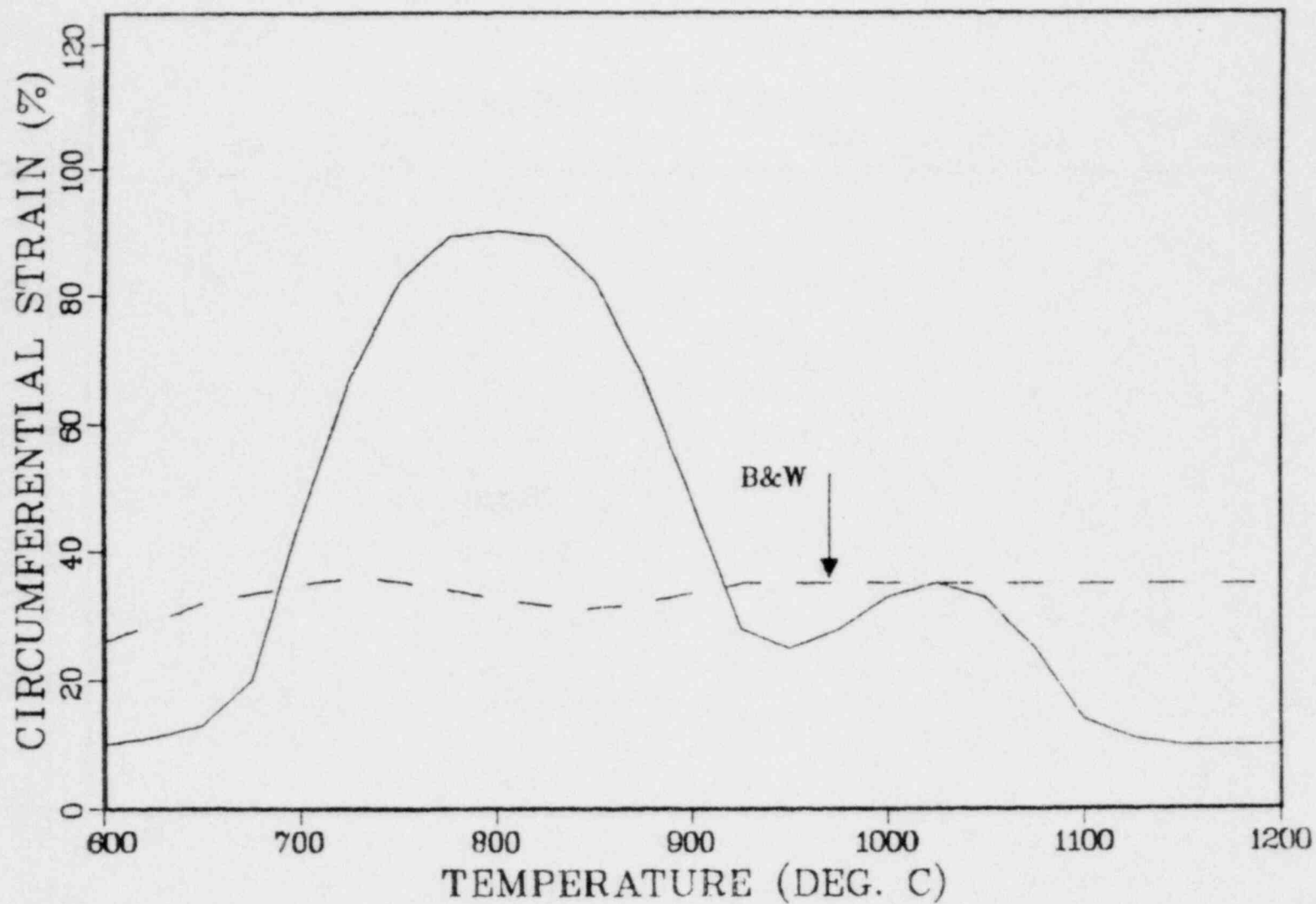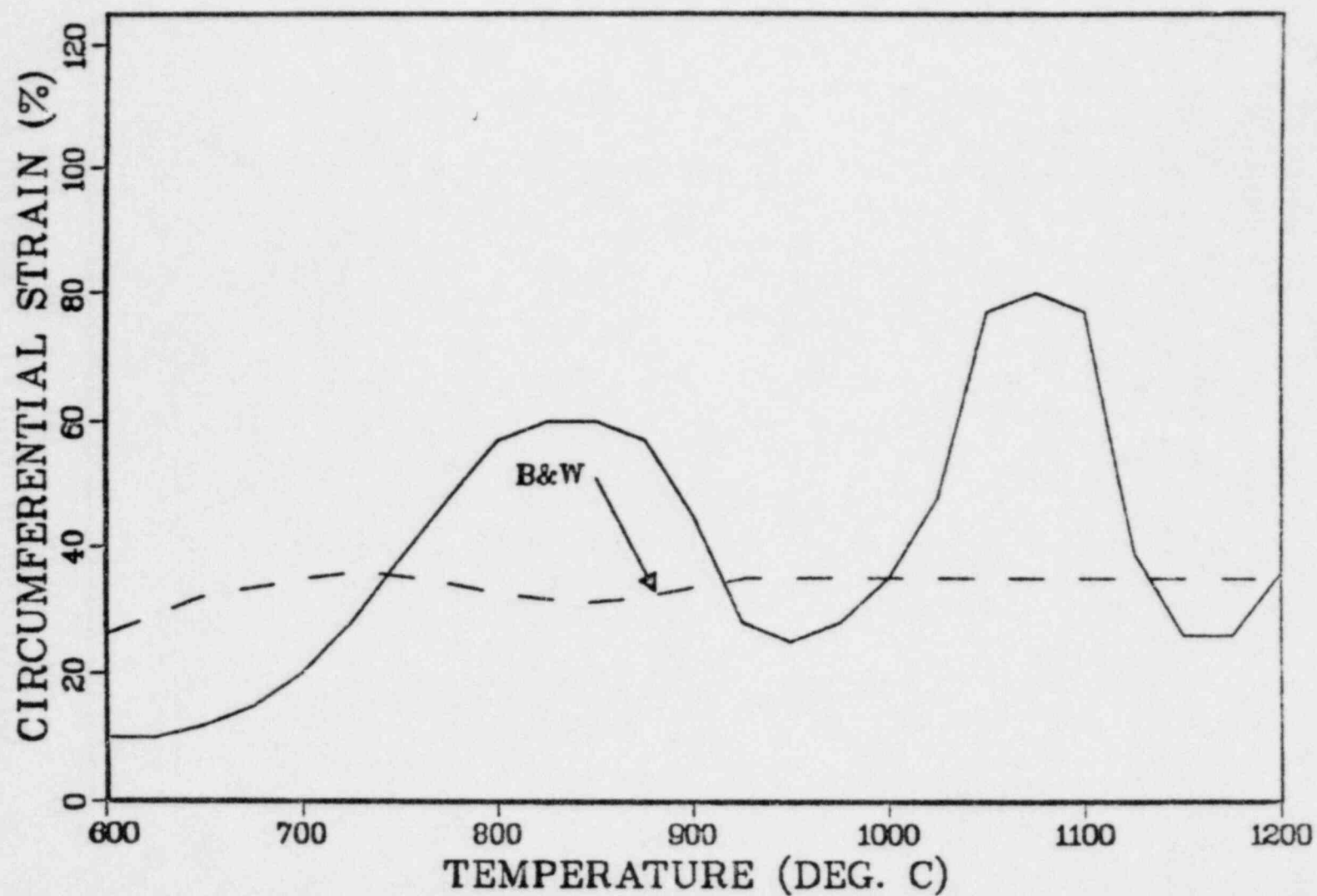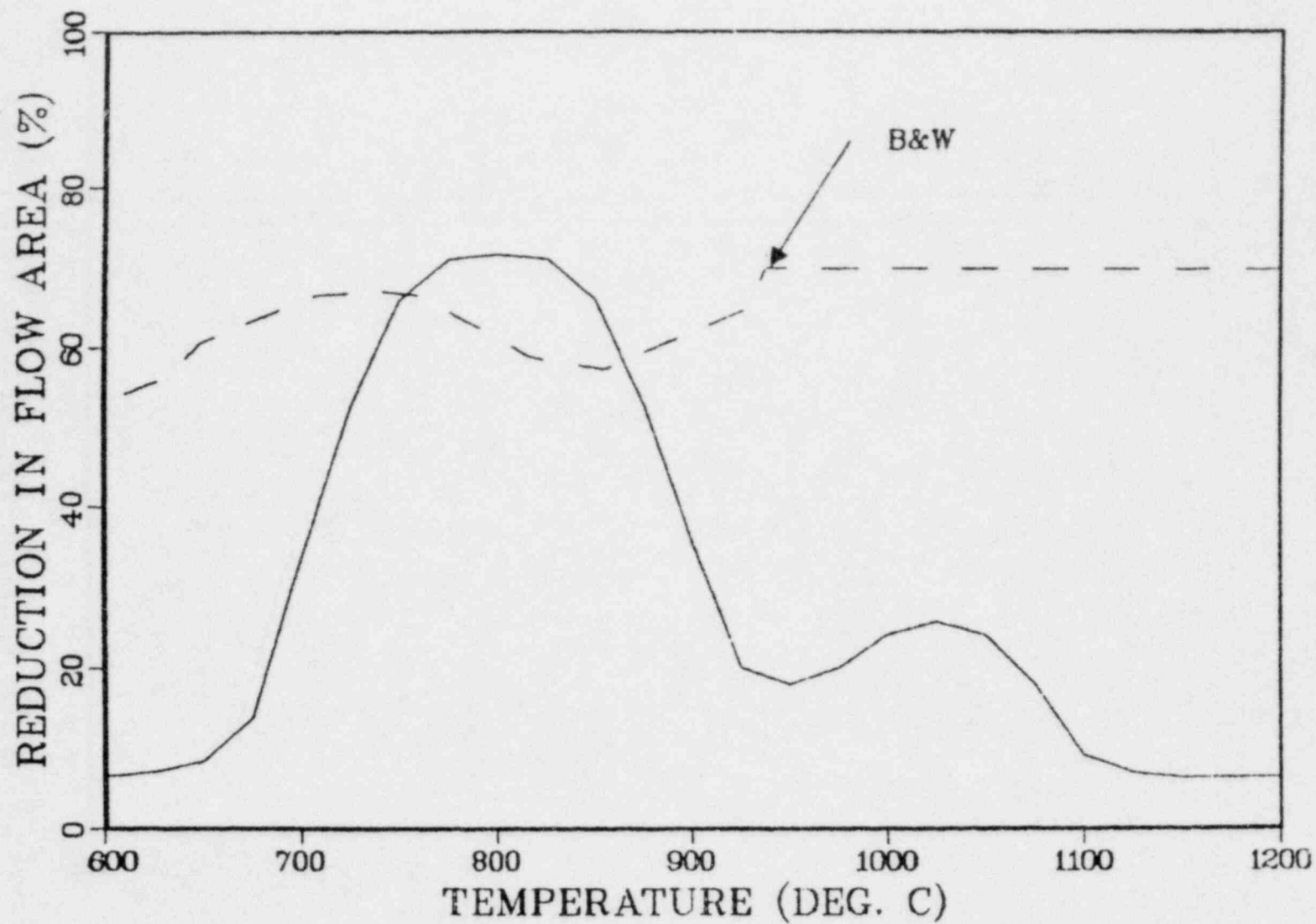Fig. 46    GE model and fast-ramp correlation of circumferential burst strain as a function of rupture temperature.

Because of BWR spray cooling and a large amount of heat transfer by radiation (to the massive channel boxes), partial flow blockage is less important in BWRs than in PWRs, and GE does not employ a flow-blockage model in their ECCS analysis.

## 4.6 Exxon

Exxon has different models for PWR and BWR applications. The PWR models are shown in Figs. 47-51.

The rupture temperature model is, like WREM. in good agreement with the present fast-ramp correlation. Although the Exxon PWR strain model exhibits a distinct alpha-plus-beta-phase valley, it differs substantially with the present correlations over a wide range of temperatures (Figs. 48-49). The Exxon PWR blockage model predicts blockages in several temperature ranges that are far larger than any predicted by the correlations presented in this report (Figs. 50-51).

The Exxon BWR models (Fig. 52-54) are similar to the GE models.

## 4.7 Yankee Atomic Electric

The Yankee Atomic Electric Company performs the ECCS analysis for the Maine Yankee and Yankee Rowe plants. The cladding models employed in the Maine Yankee analysis are identical to the ENC PWR models displayed in Figs. 47-51.
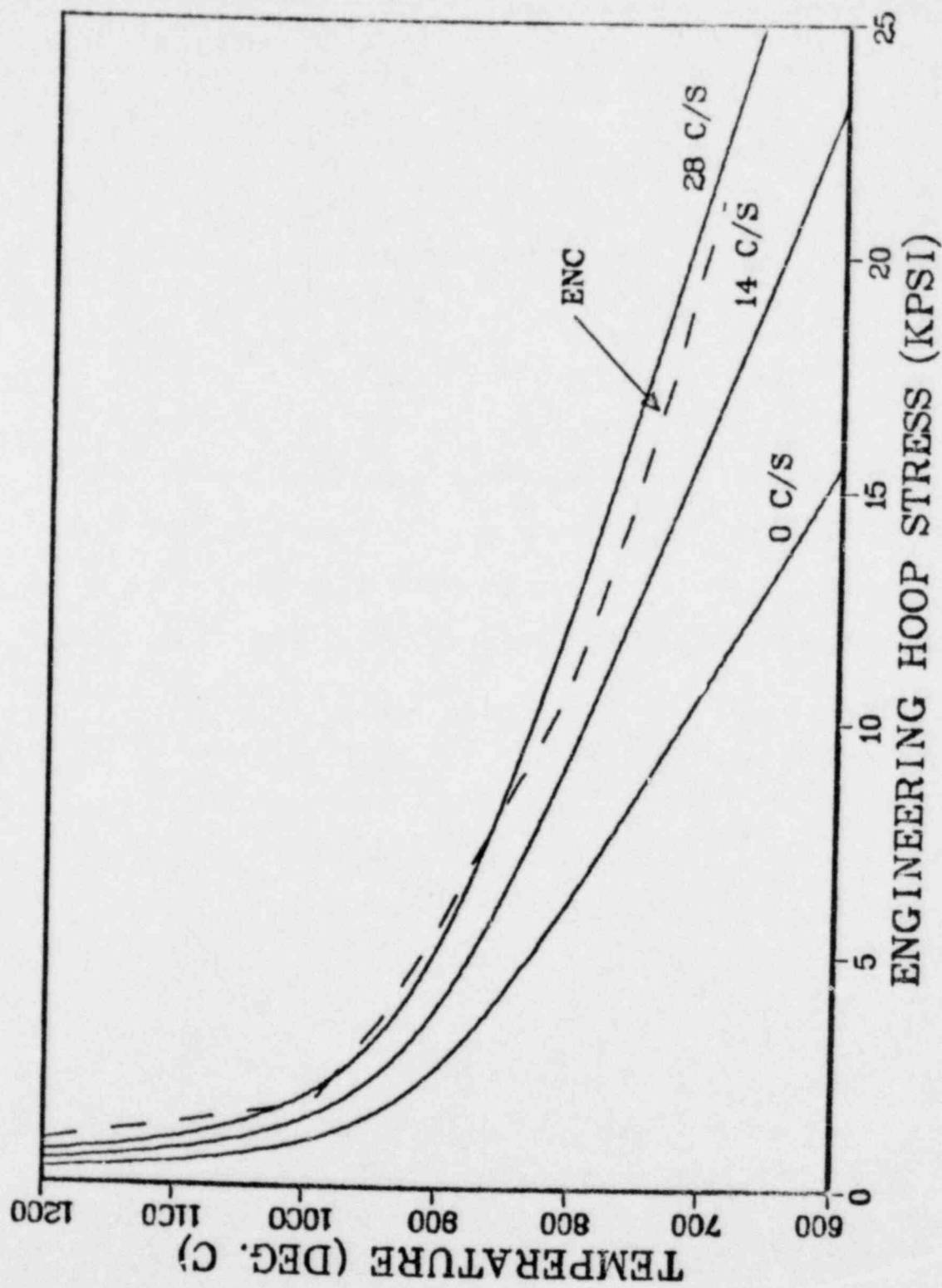
Fig. 47    ENC PWR model and ORNL correlation of rupture temperature as a function of
engineering hoop stress and ramp rate.
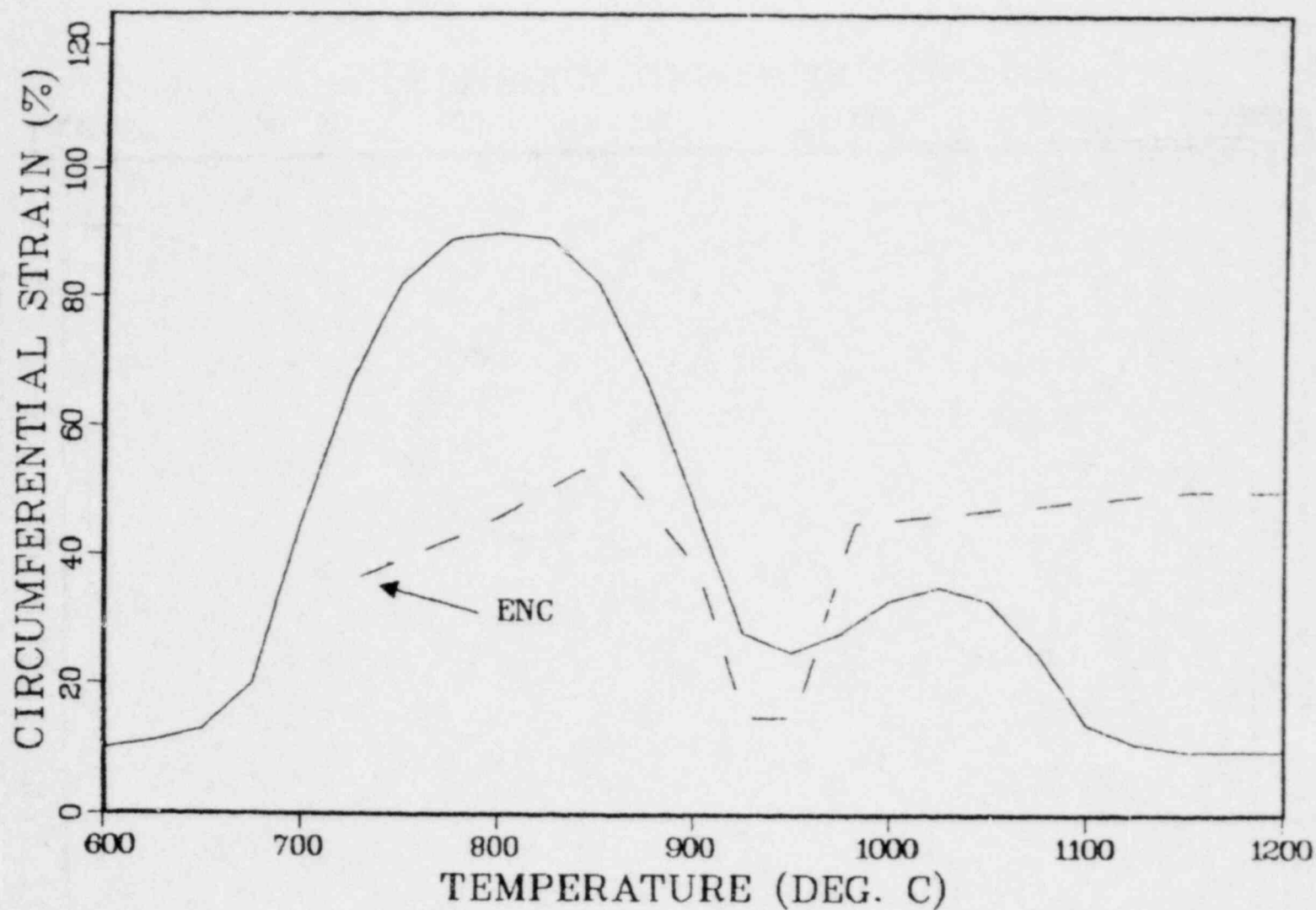
Fig. 48    ENC PWR model and slow-ramp correlation of circumferential burst strain as a function of rupture temperature.
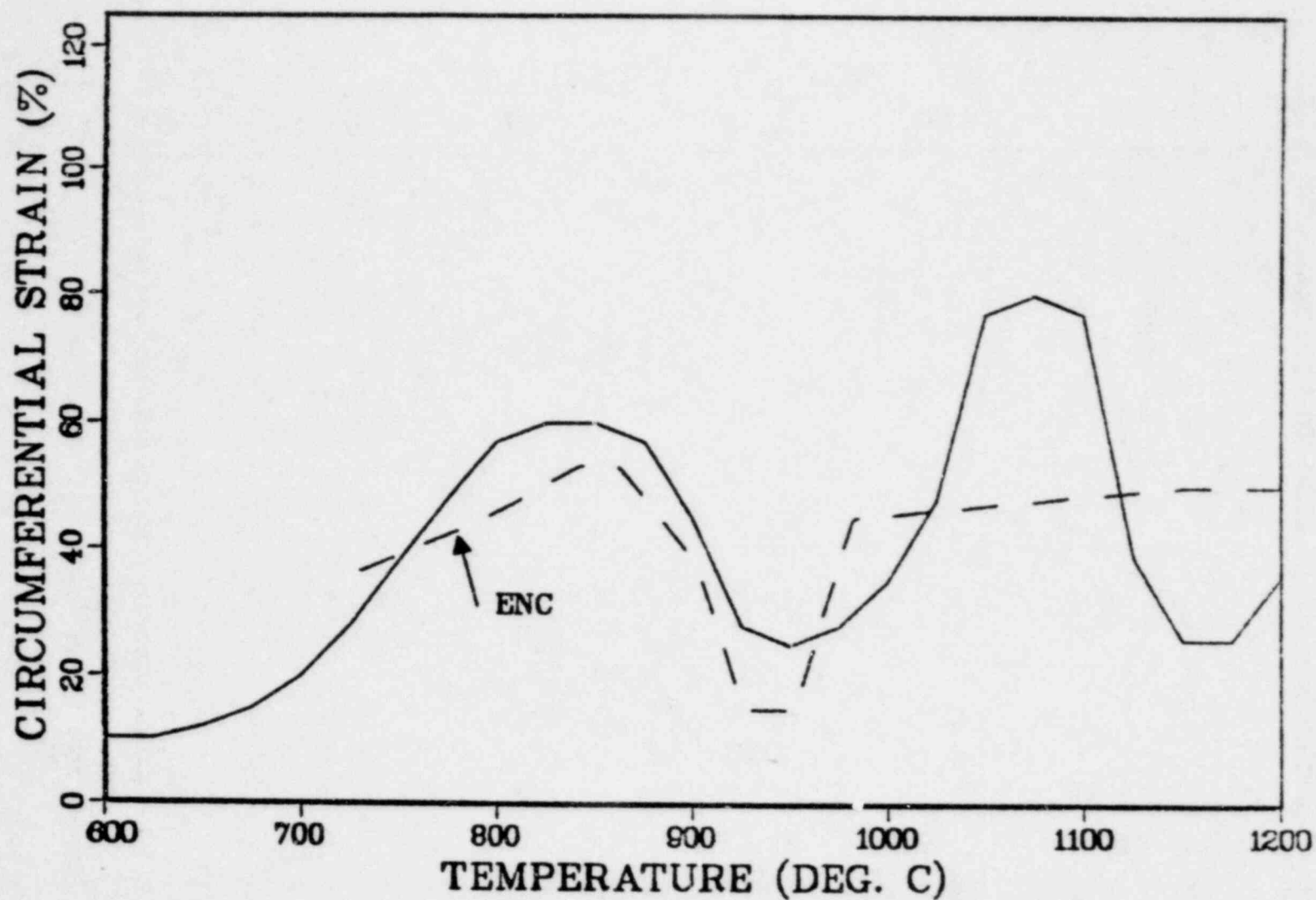
Fig. 49   ENC PWR model and fast-ramp correlation of circumferential burst strain as a function of rupture temperature.

-74-
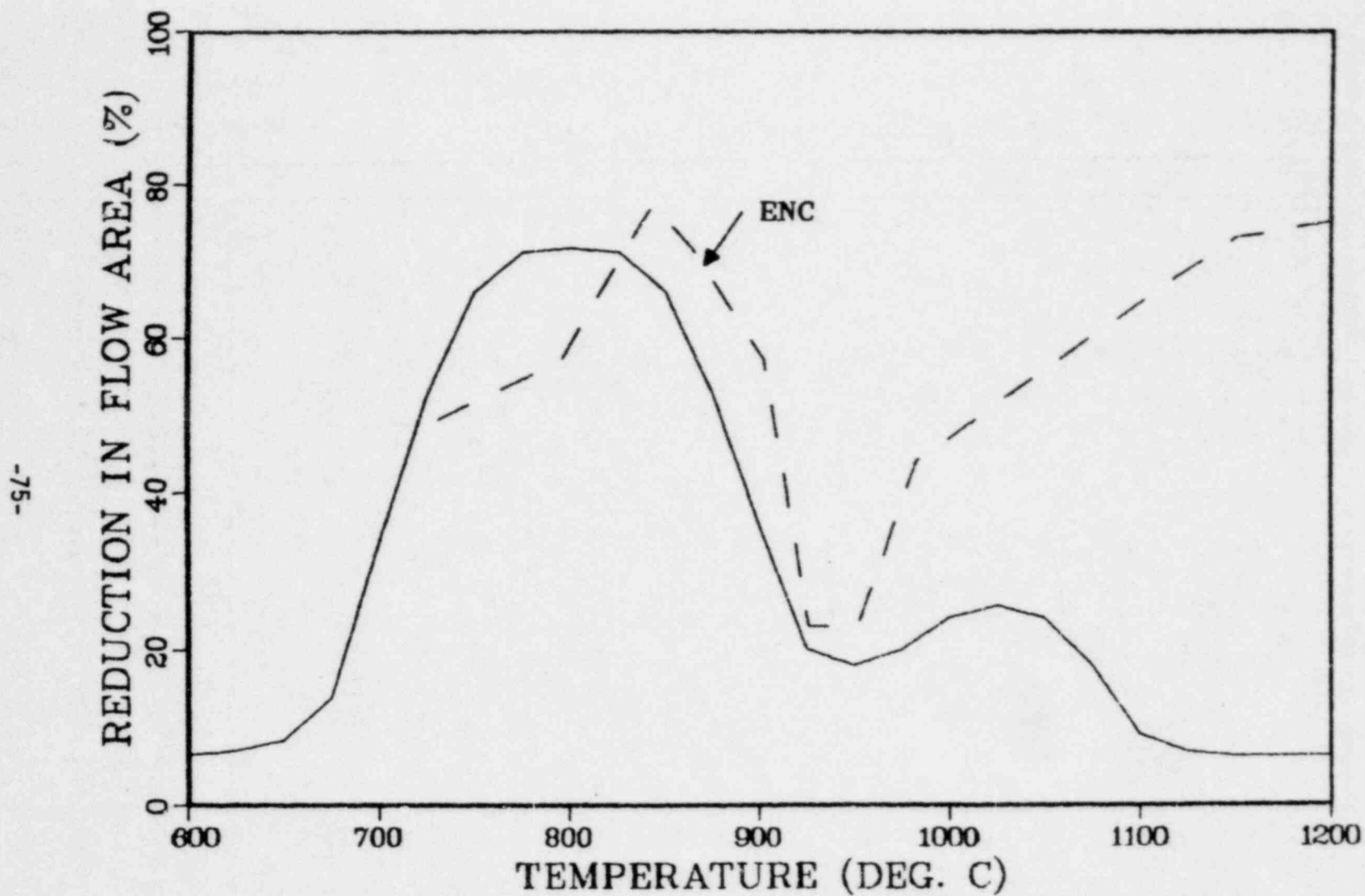
Fig. 50    ENC model and slow-ramp correlation of reduction in assembly flow area as a
function of rupture temperature.
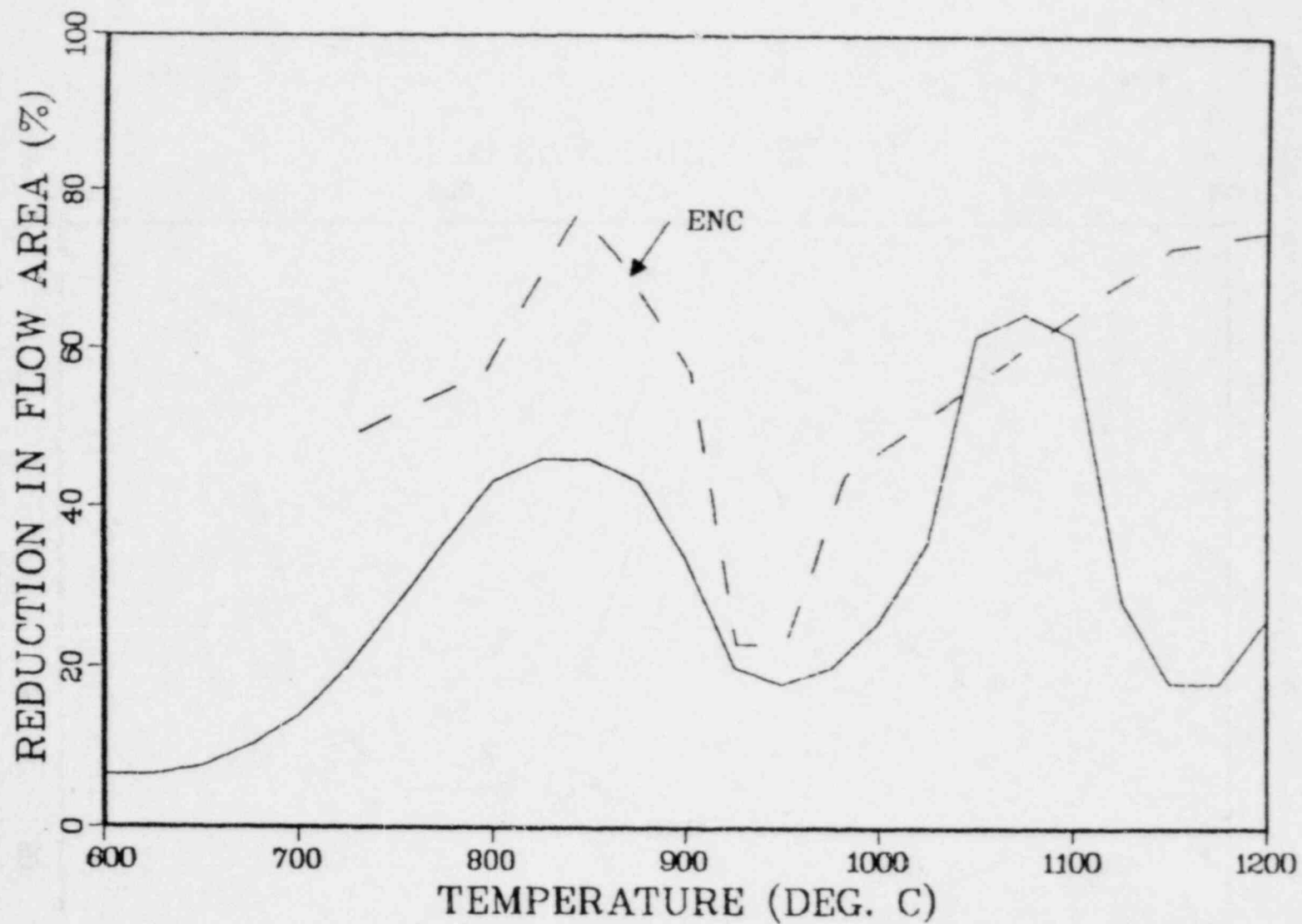
Fig. 51  ENC model and fast-ramp correlation of reduction in assembly flow area as a function of rupture temperature.
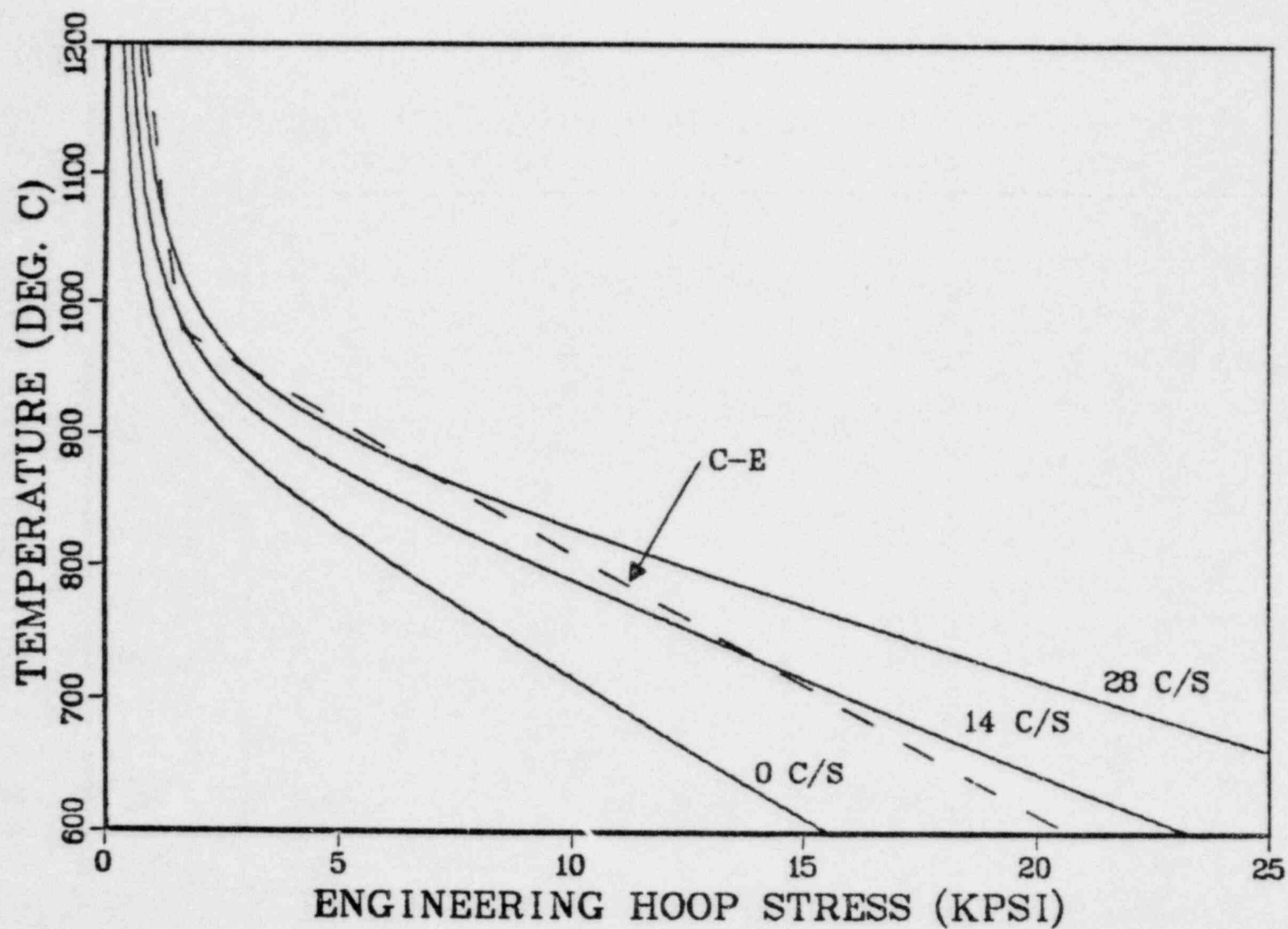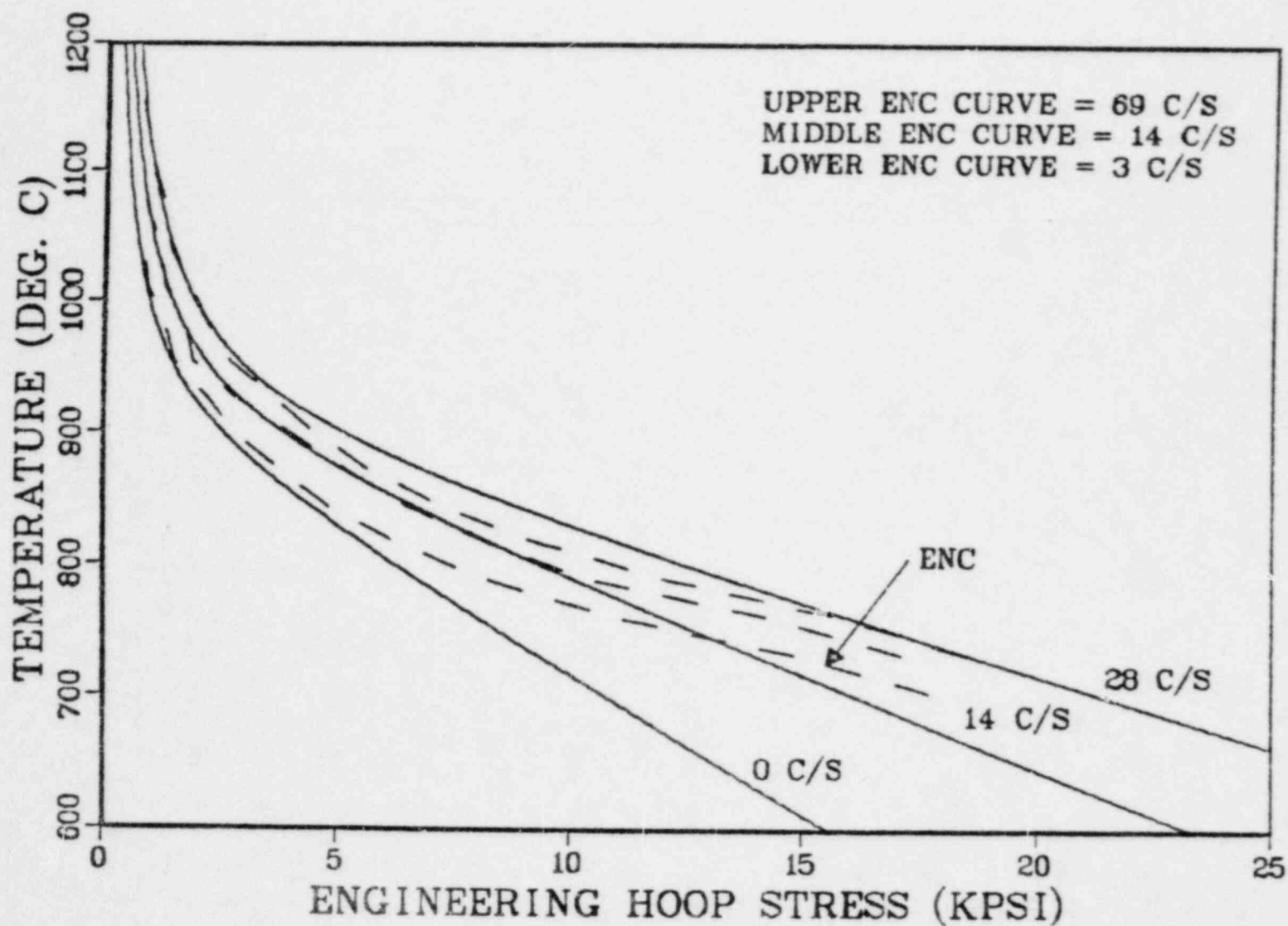
Fig. 52    ENC BWR model and ORNL correlation of rupture temperature as a function of
engineering hoop stress and ramp rate.
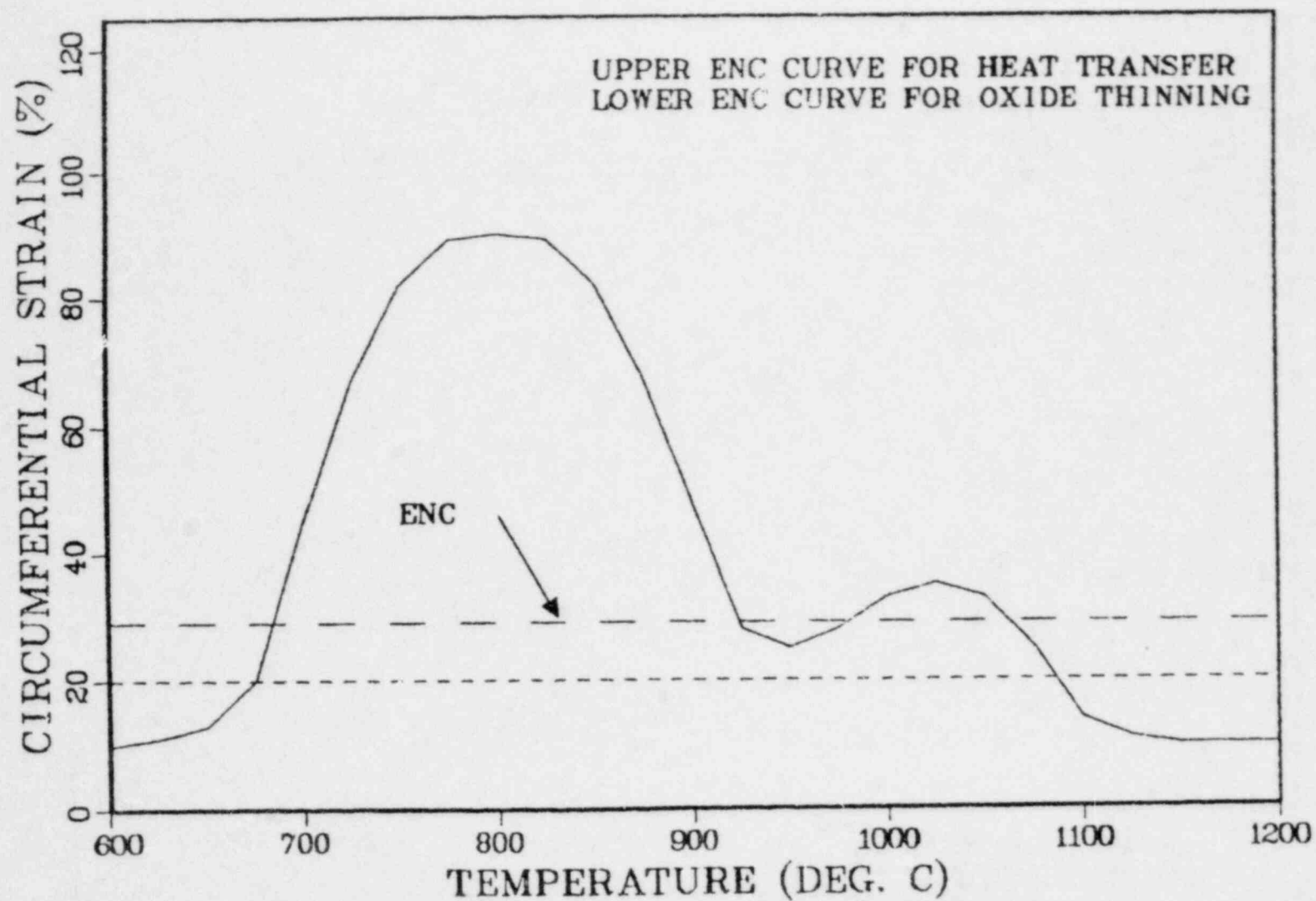
Fig. 53     ENC BWR model and slow-ramp correlation of circumferential burst strain as a function of rupture temperature.

UPPER ENC CURVE FOR HEAT TRANSFER
LOWER ENC CURVE FOR OXIDE THINNING

ENC
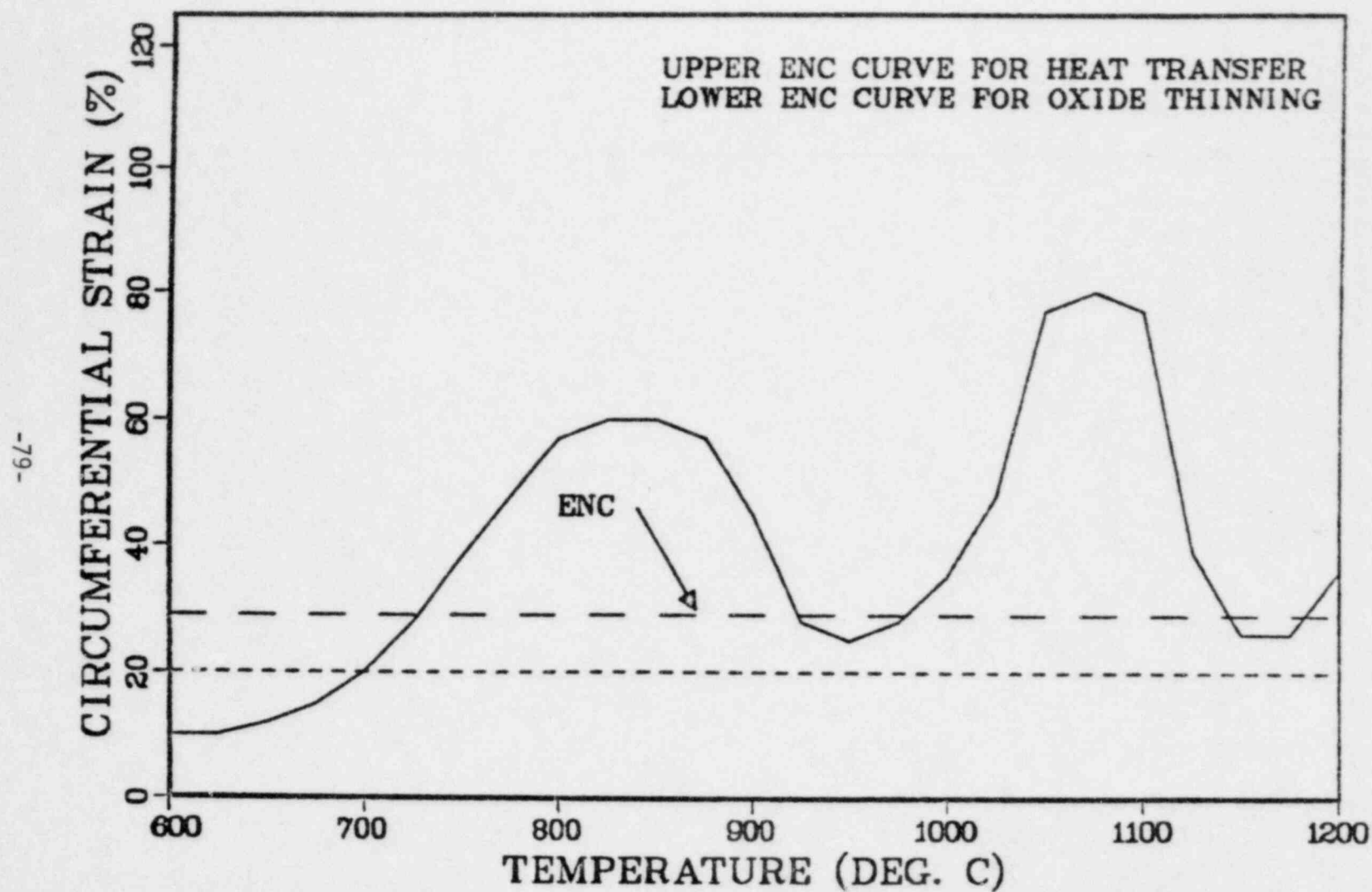
Fig. 54    ENC BWR model and fast-ramp correlation of circumferential burst strain as a function of rupture temperature.

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Figures 55-59 show the models used for the Yankee Rowe Cycle 4
analysis. Figure 55 shows the rupture temperature model, which is
like WREM and is in good agreement with the present fast-ramp
correlation. Figures 56-57 show the burst strain model, which
predicts far greater single rod deformation than any of the other
licensing models. The flow blockage model exhibits smaller blockages
than the present correlation over a large temperature range for
slow ramps (Fig. 58) and over a narrow temperature range for fast
ramps (Fig. 59).
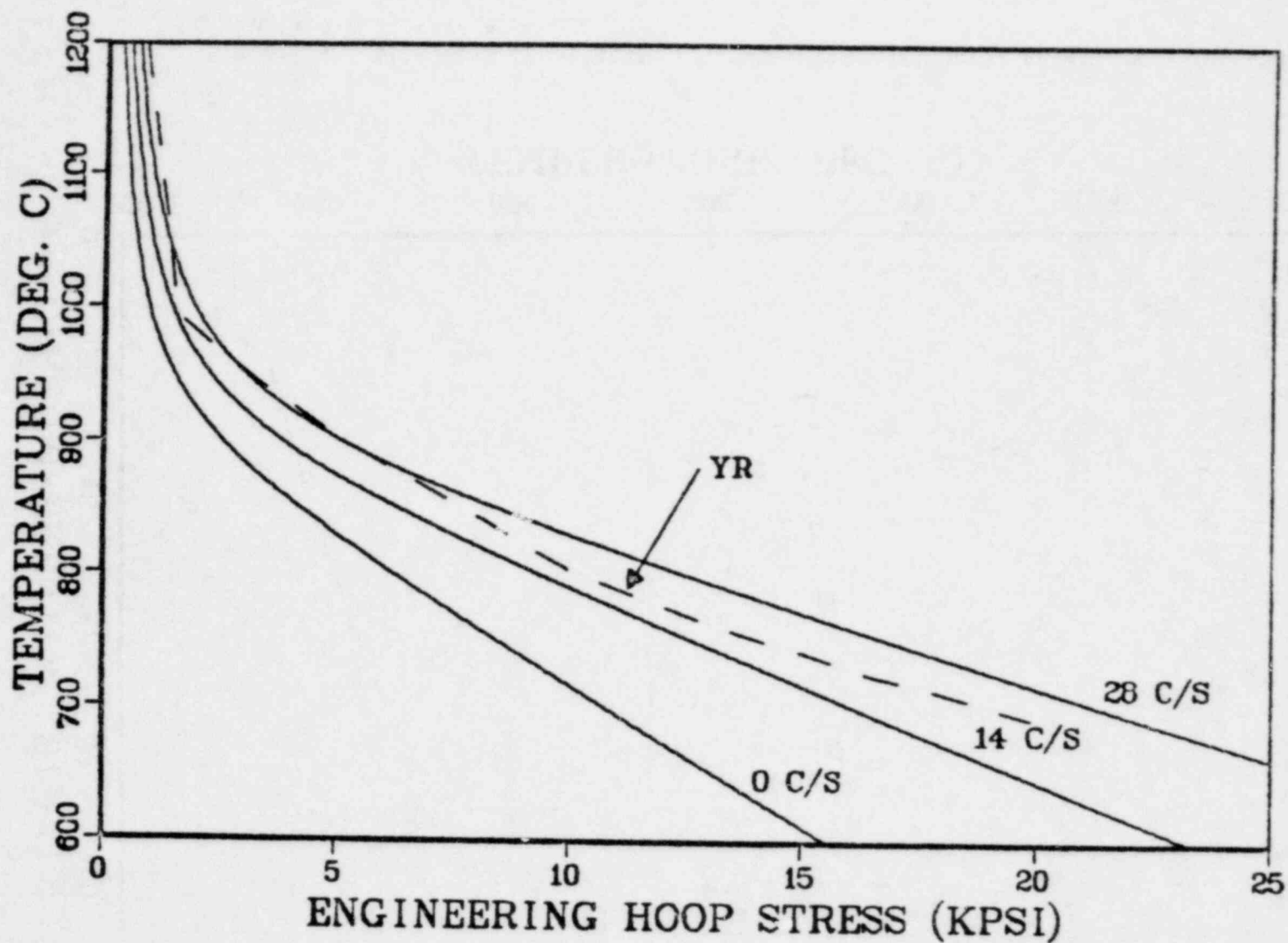
Fig. 55    YR model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate.
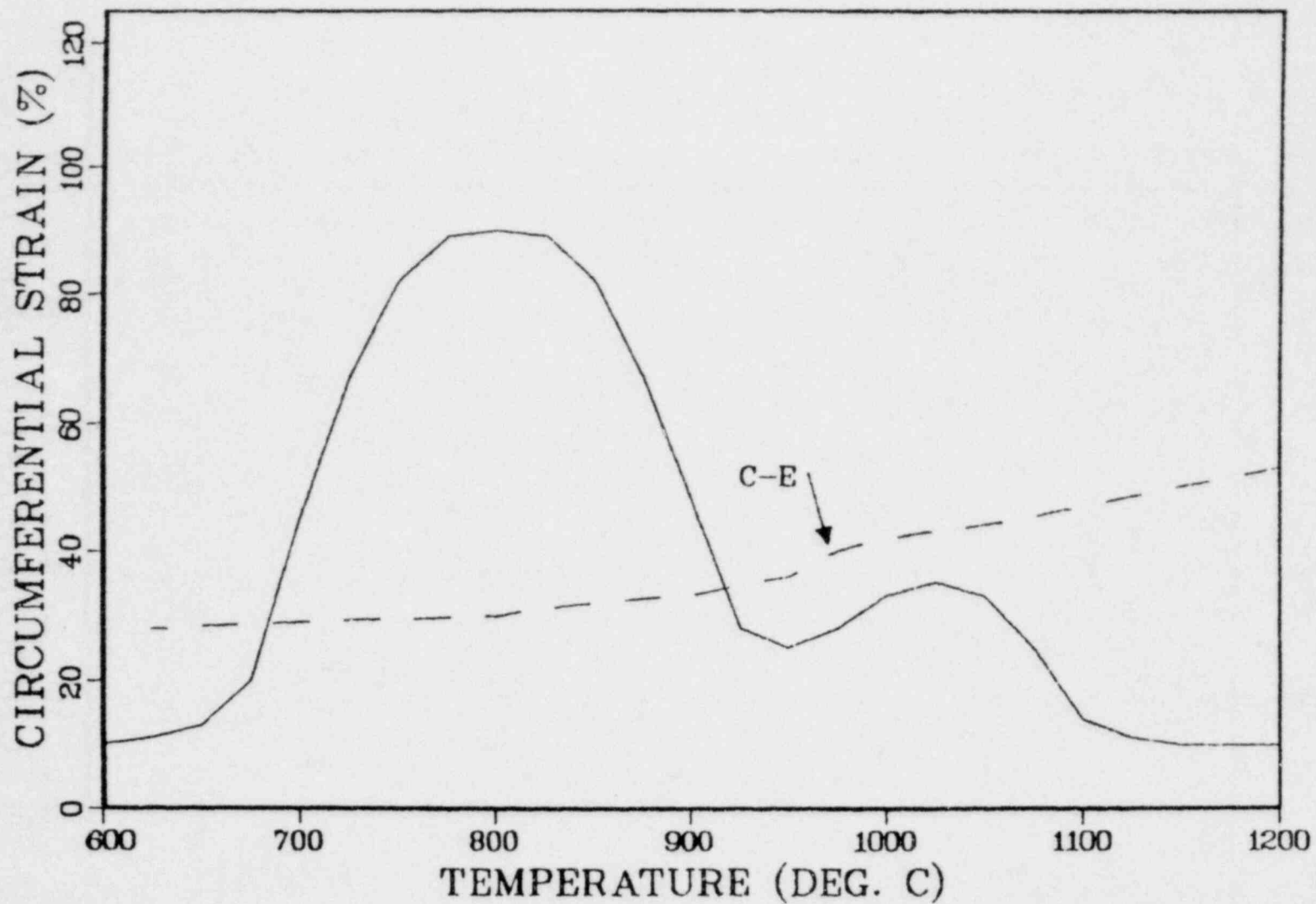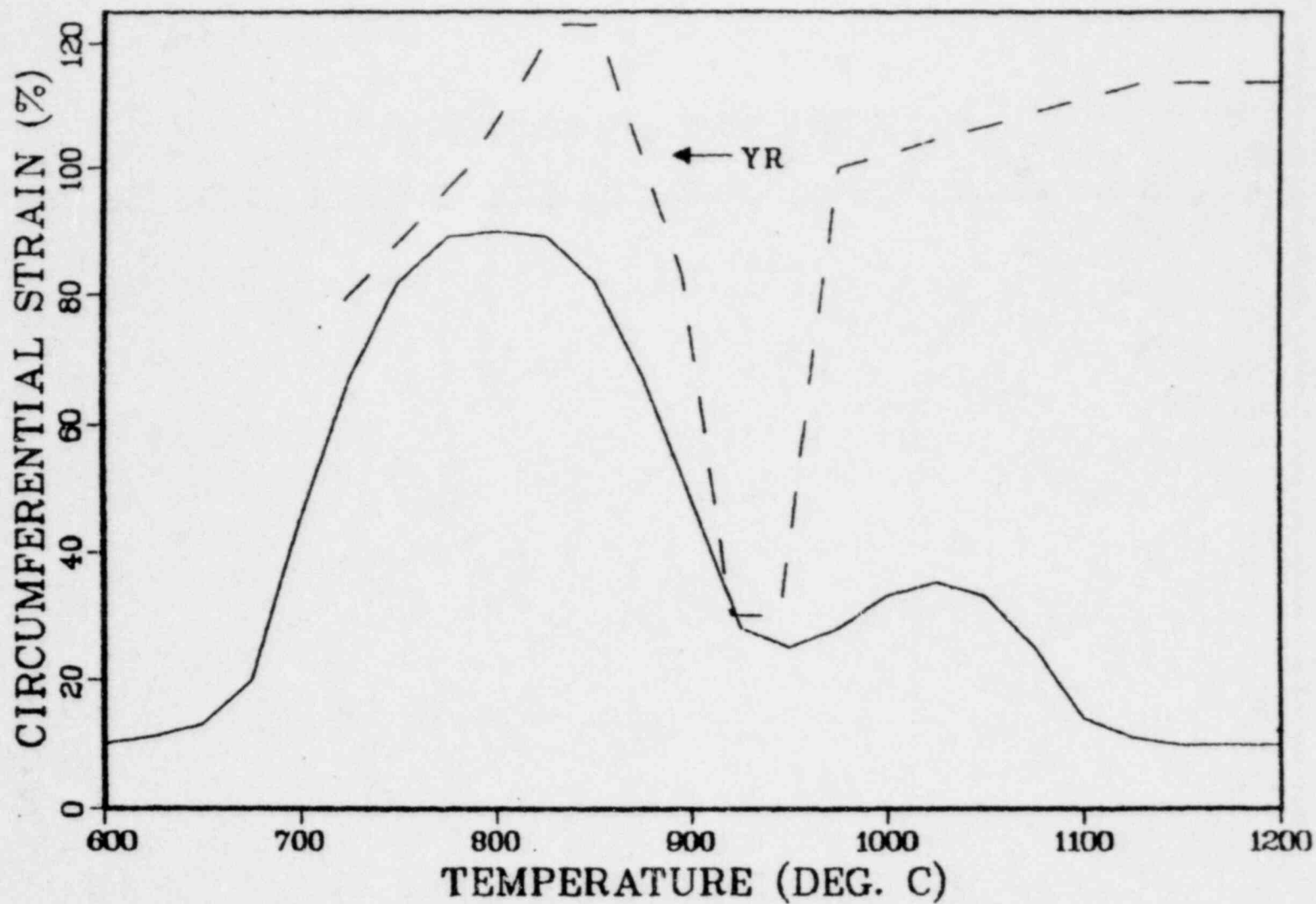
Fig. 56    YR model and slow-ramp correlation of circumferential burst strain as a function
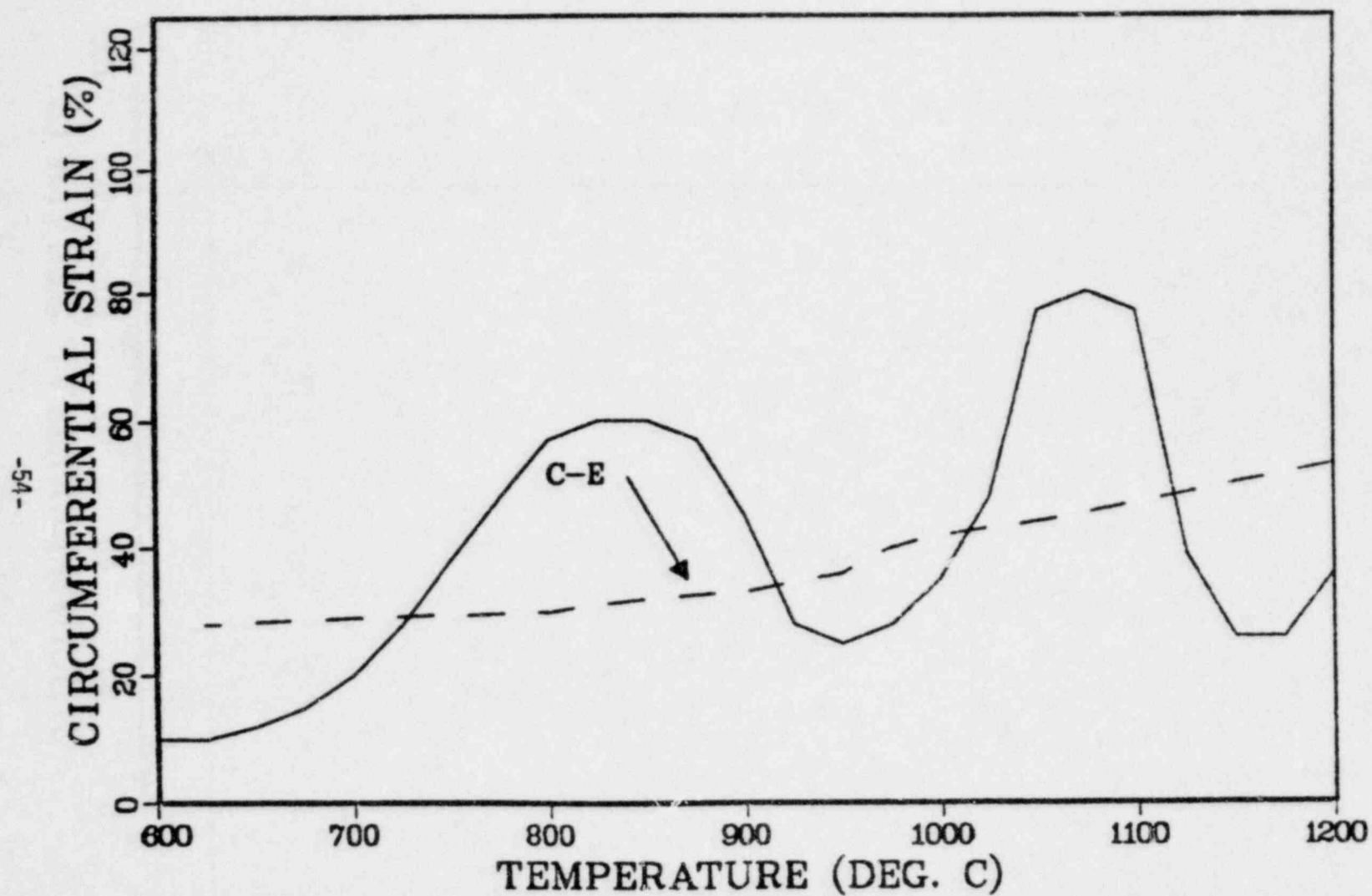of rupture temperature.

-83-
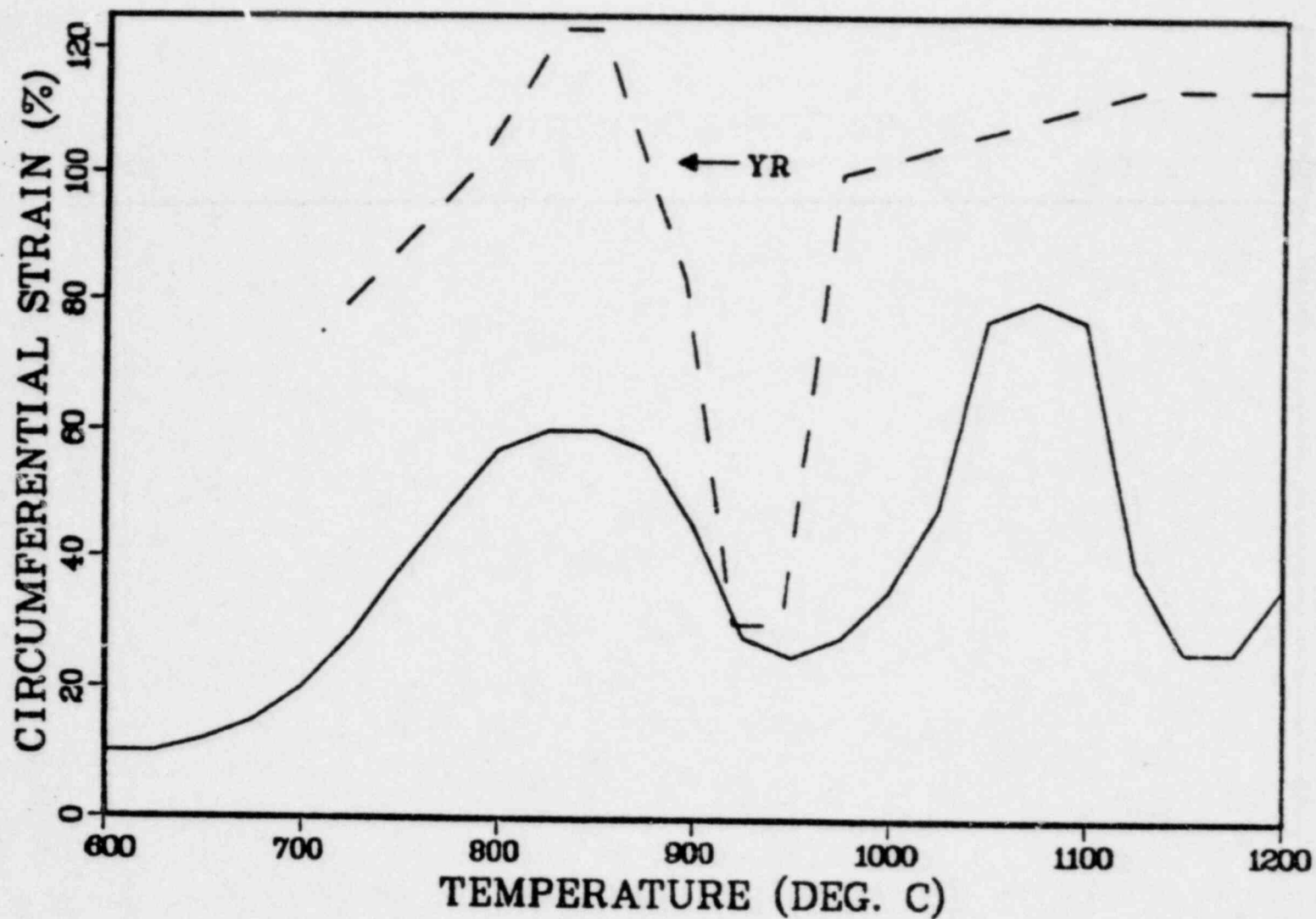
Fig. 57     YR model and fast-ramp correlation of circumferential burst strain as a function
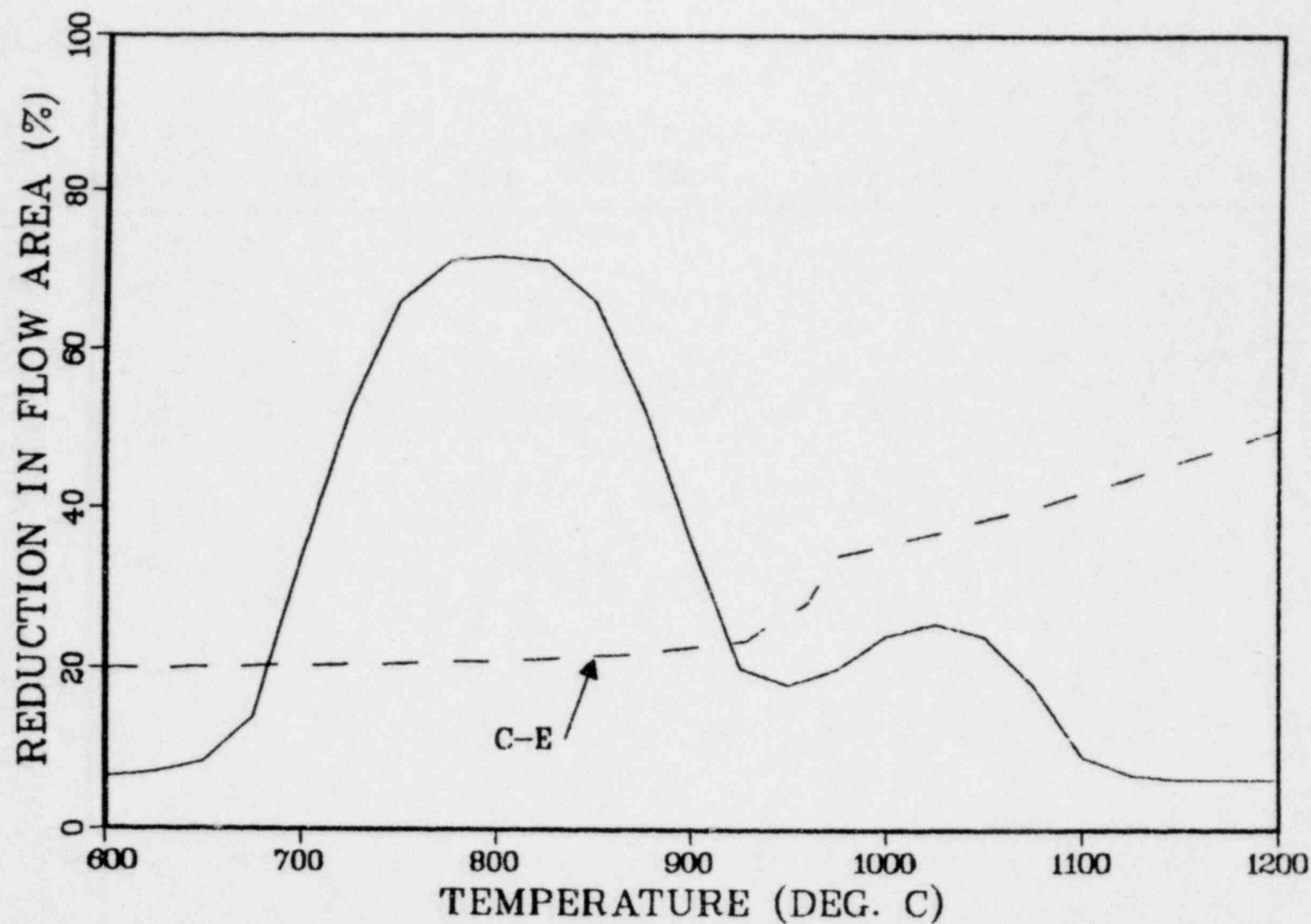of rupture temperature.
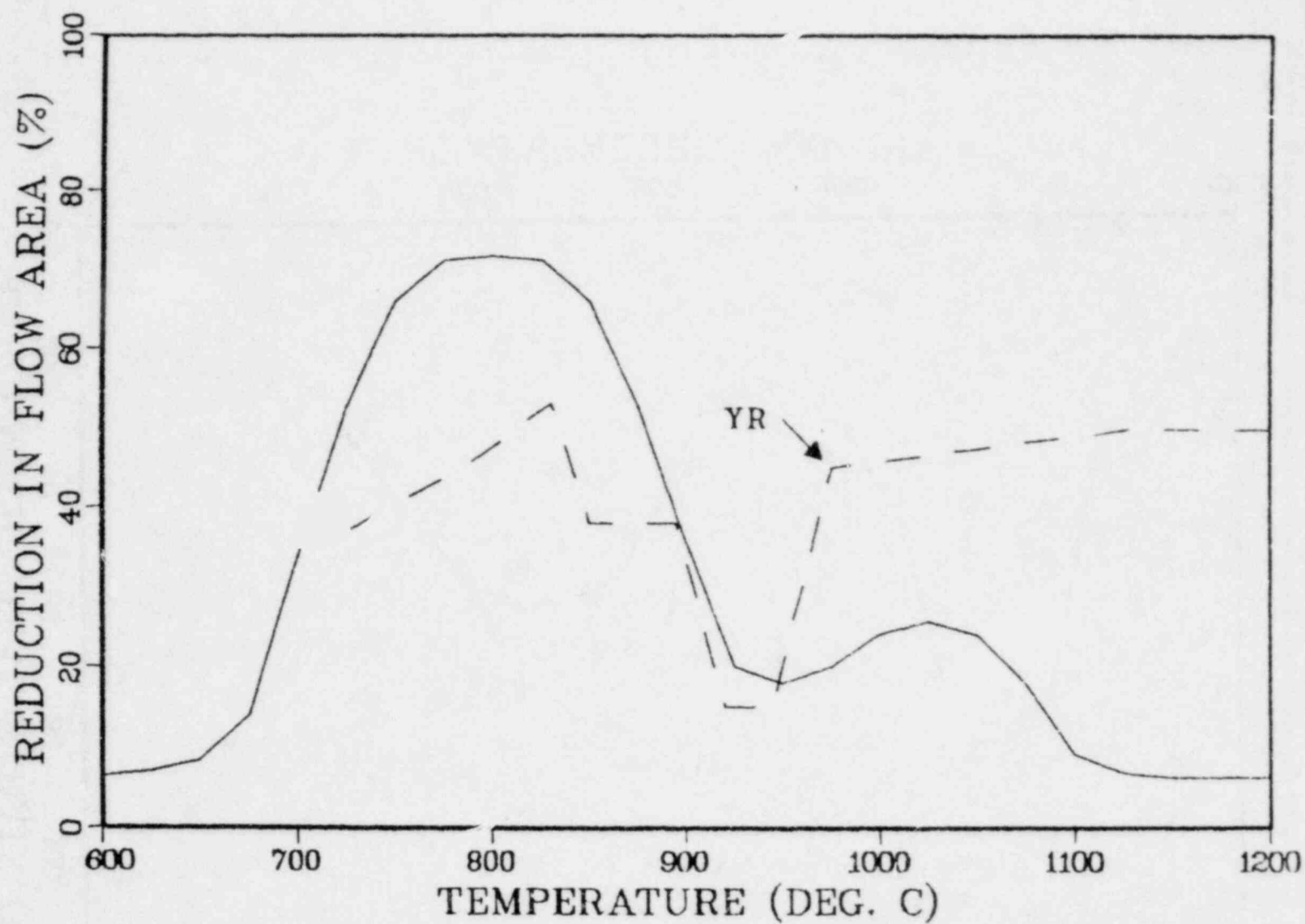
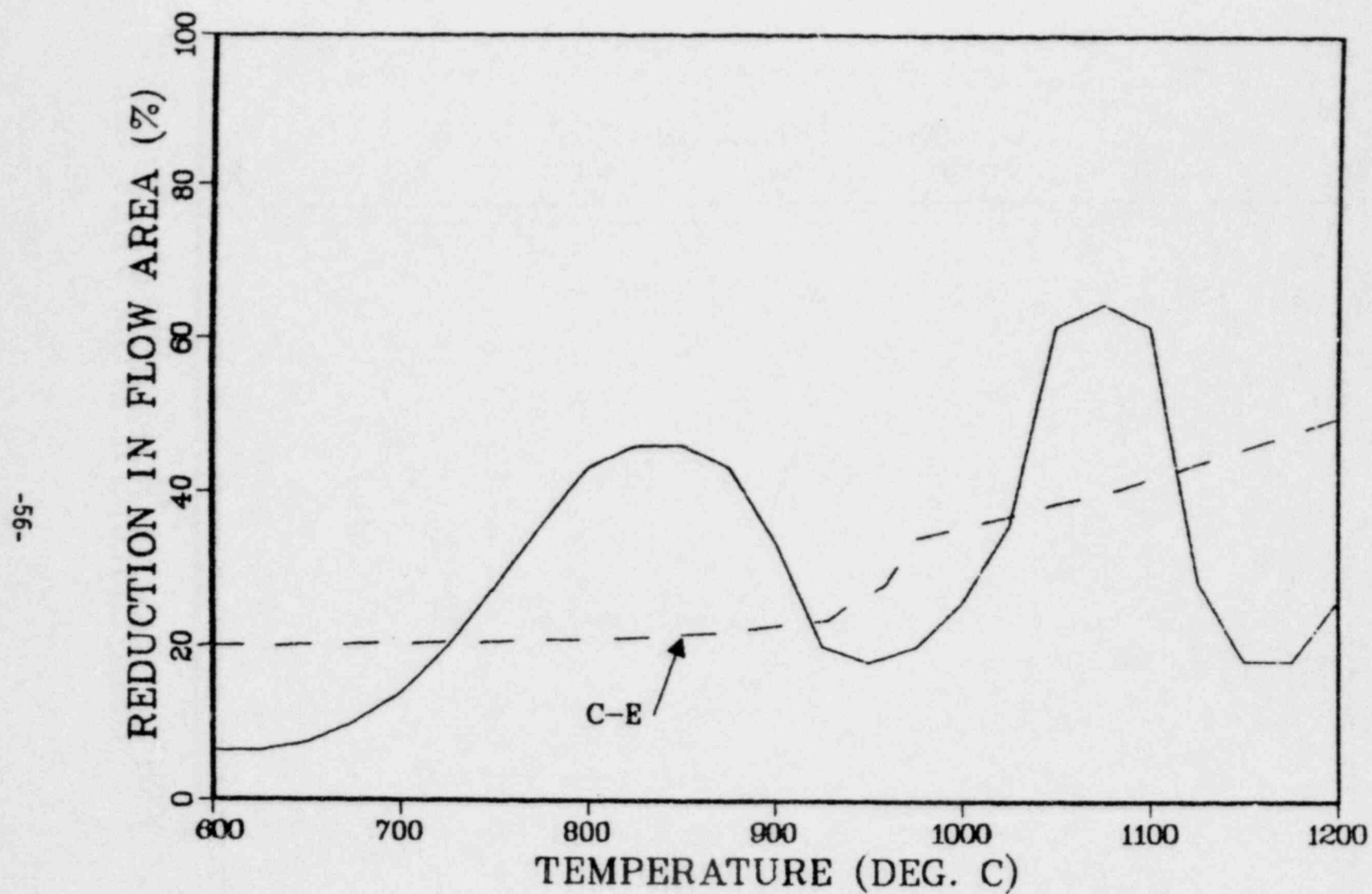Fig. 58    YR model and slow-ramp correlation of reduction in assembly flow area as a function of rupture temperature.

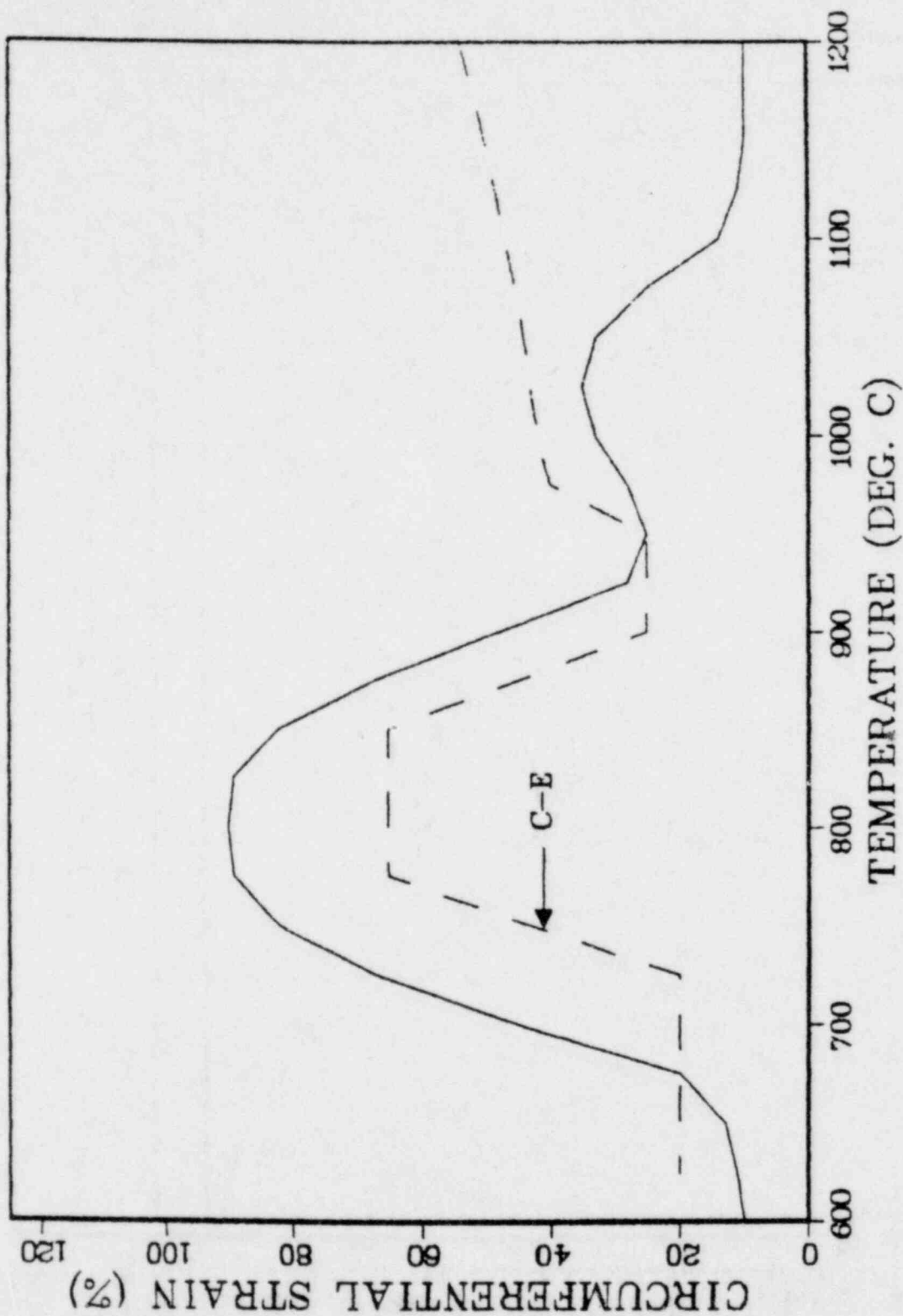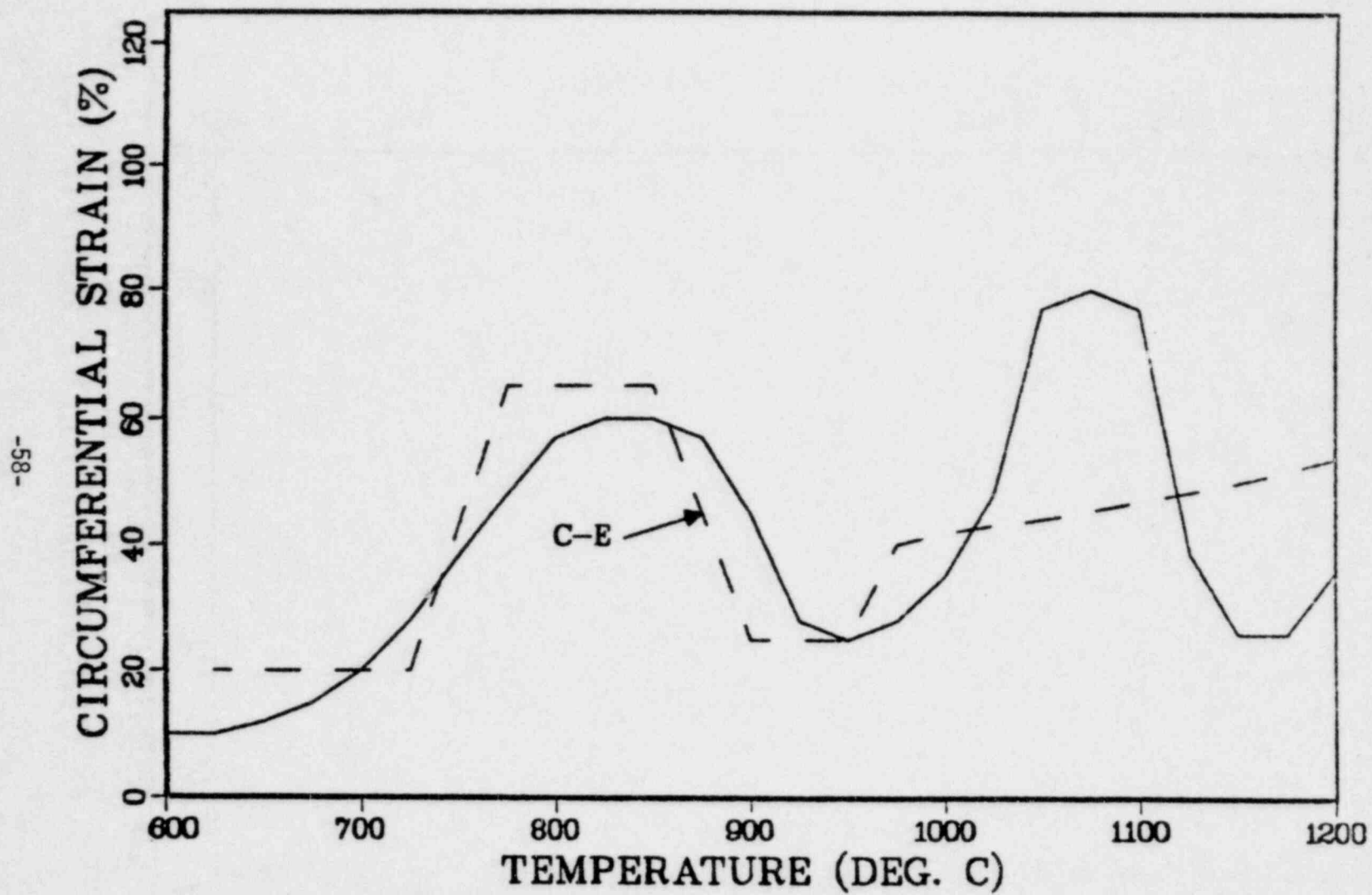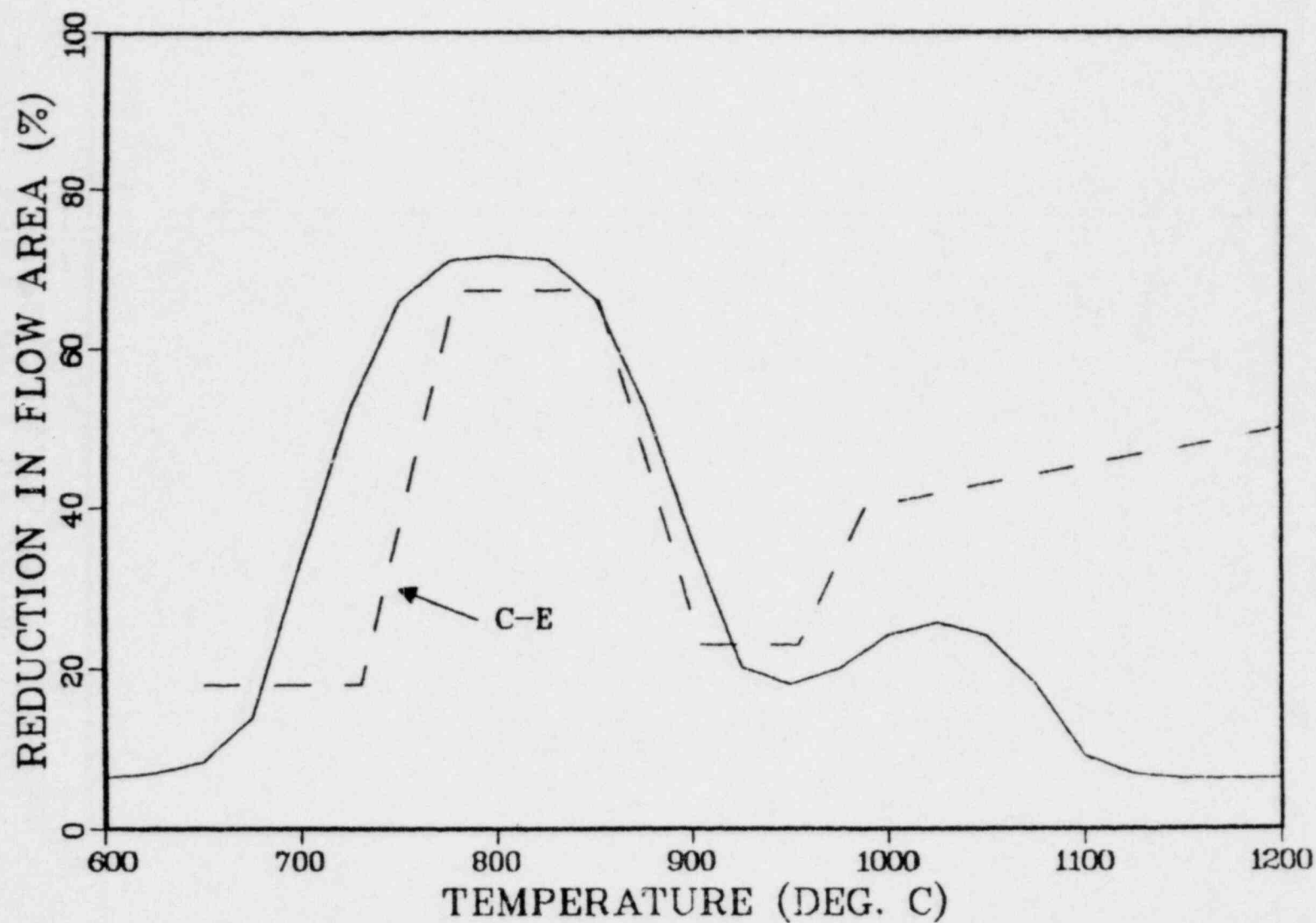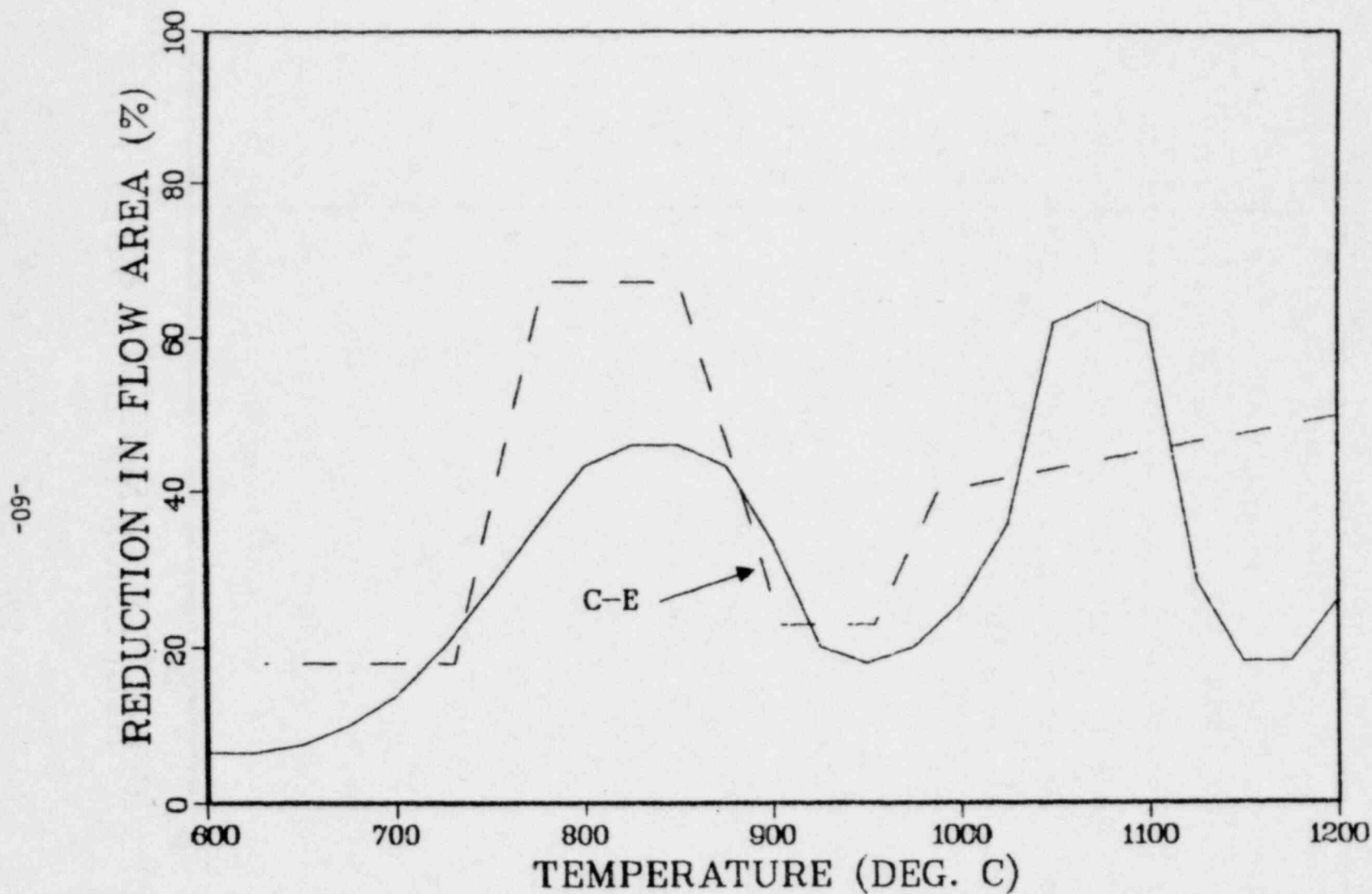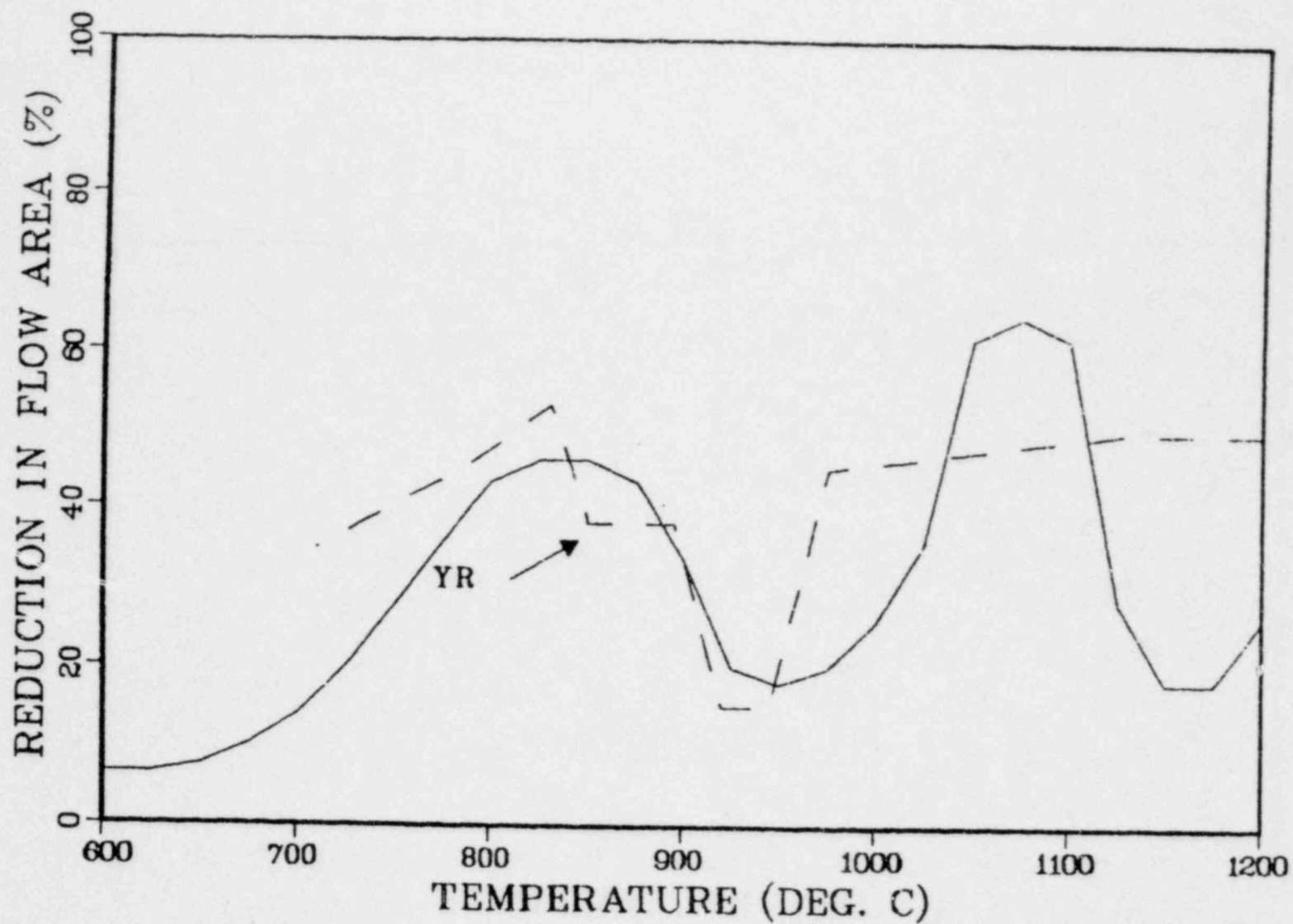Fig. 59    YR model and fast-ramp correlation of reduction in assembly flow area as a function of rupture temperature.

## 5. CONCLUSIONS

Empirical fuel cladding models that describe the incidence of rupture, the degree of swelling, and the extent of coolant flow blockage have been the subject of extensive confirmatory research during the past five years. During that time, however, the cladding models that are used in the ECCS licensing analyses have not changed. Although the research programs are not finished, we are at a plateau of understanding that suggests that improvements should be made in these licensing models.

We have reviewed all of the available data to date and have selected the most prototypical data from which to derive new cladding correlations. This selection gave most weight to experiments performed in aqueous environments and utilizing internal heaters (either fuel pellets or electrically heated fuel pellet simulators). The data base is not complete, and we have had to make certain assumptions to bridge the gaps. In particular, the number of available bundle measurements of flow blockage is so small that assembly blockages have of necessity been derived from single rod burst strains on the basis of three carefully analyzed bundle tests.

Heating rate effects were found to be important, so all of the cladding correlations were derived as functions of temperature-ramp rate. Most present licensing models do not include ramp-rate effects.

The new cladding correlations differ substantially from the present
licensing models. In fact, the present licensing models for each fuel
vendor differ substantially from each other. Based on the applicable
data, we believe that the new cladding correlations presented in this
report and displayed in Figs. 3, 6, 7, and 16 provide the best means
available today of predicting swelling and rupture without underestimating
the degree of swelling or the incidence of rupture.

There is still uncertainty in these correlations, and further research
is needed to confirm or further modify these correlations. Nevertheless,
in the interm, we recommend that all industry ECCS models be revised to
adopt the new cladding correlations presented in this report.

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32. Letter from F. J. Erbacher, Kernforschungszentrum Karlsruhe, to D. A. Powers, U.S. Nuclear Regulatory Commission, dated February 13, 1980. Available in NRC PDR for inspection and copying for a fee.

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37. R. A. Lorenz, D. O. Hobson, and G. W. Parker, Oak Ridge National Laboratory, "Final Report on the Second Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," Oak Ridge National Laboratory Report ORNL-4710, January 1972. Available in public technical libraries. Also available for purchase from the National Technical Information Service (NTIS), Springfield, Virginia 22161.

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# APPENDIX A

## FUEL CLADDING BURST DATA

DATA REFERENCE A   (Upright Triangle)

FRF-1

R. A. Lorenz, D. O. Hobson, and G. W. Parker, Oak Ridge National Laboratory, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TRFAT," Oak Ridge National Laboratory Report ORNL-4635, March 1971. Available in public technical libraries. Also available for purchase from the 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, 502-520 (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 rupture temperature = 889°C.
Mean rod engineering burst stress = 1.71 kpsi.

| ROD # | RAMP RATE (°C/S) | PRESSURE AT RUPTURE (PSIG) | RUPTURE 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 | 25-36 | 215 | 810 | 40 | 1.74 |

DATA REFERENCE A (Upright Triangle)

FRF-2

R. A. Lorenz, D. O. Hobson, and G. W. Parker, Oak Ridge National Laboratory, "Final Report on the Second Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT," Oak Ridge National Laboratory Report ORNL-4710, January 1972. Available in public technical libraries. Also available for purchase from the 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, 502-520 (August 1971). Available in public technical libraries.

Inpile, 7-rod bundle, steam atmosphere.

Maximum reduction in bundle flow area = 91%.
Mean rod burst strain = 57%.
Mean rod rupture temperature = 1254°C.
Mean rod engineering burst stress = 0.69 kpsi.

| ROD # | RAMP RATE (°C/S) | PRESSURE AT RUPTURE (PSIG) | RUPTURE TEMPERATURE (°C) | BURST STRAIN (%) | ENGINEERING BURST STRESS (KPSI) |
|---|---|---|---|---|---|
| 58-3 | 44 | 85 | 1260 | 50 | 0.69 |
| 11 | 44 | 85 | 1216 | 46 | 0.69 |
| 12 | 44 | 85 | 1260 | 64 | 0.69 |
| 13 | 44 | 85 | 1260 | 58 | 0.69 |
| 16 | 44 | 85 | 1260 | 70 | 0.69 |
| 17 | 44 | 85 | 1260 | 47 | 0.69 |
| 18 | 44 | 85 | 1260 | 61 | 0.69 |

DATA REFERENCE B (Cross)

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

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

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

| ROD # | RAMP RATE (°C/S) | PRESSURE AT RUPTURE (PSIG) | RUPTURE 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, Oak Ridge National Laboratory, "Multirod Burst Test Program Progress Report for July-December 1977," USNRC Report NUREG/CR-0103, June 1978.  Available in public technical libraries.  Also available for purchase from the National Technical Information Service (NTIS), Springfield, Virginia  22161.

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R. H. Chapman and others, "Bundle B-1 Test Data Multirod Burst Test Program," Interim Report ORNL/NUREG/TM-322, prepared for NRC by Oak Ridge National Laboratory, June 1979.  Available in NRC PDR for inspection and copying for a fee.

Out-of-pile, 16-rod bundle, heated shroud, 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 rupture temperature = 868°C.
Mean rod engineering burst stress = 8.72 kpsi.

| ROD # | RAMP RATE (°C/S) | PRESSURE AT RUPTURE (PSIG) | RUPTURE 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 |

DATA REFERENCE C (Plus)

MRBT B-2

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

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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 rupture temperature = 858°C.
Mean rod engineering burst stress = 8.88 kpsi.

| ROD # | RAMP RATE (°C/S) | PRESSURE AT RUPTURE (PSIG) | RUPTURE TEMPERATURE (°C) | BURST STRAIN (%) | ENGINEERING BURST STRESS (KPSI) |
|---|---|---|---|---|---|
| 1 | 29 | 1117 | 870 | 35 | 9.05 |
| 2 | 29 | 1114 | 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, Oak Ridge National Laboratory, "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 NRC PDR for inspection and copying for a fee.

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Out-of-pile, 16-rod bundle, heated shroud, 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 rupture temperature = 764°C.
Mean rod engineering burst stress = 11.07 kpsi.

| ROD # | RAMP RATE (°C/S) | PRESSURE AT RUPTURE (PSIG) | RUPTURE 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 |

DATA REFERENCE D (Closed Circle)

F. Erbacher, H. J. Neitzel, and K. Wiehr, Kernforschungszentrum Karlsruhe, "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, Kernforschungszentrum Karlsruhe, "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, Kernforschungszentrum Karlsruhe, "Out-of-Pile Experiments on Ballooning of Zircaloy Fuel Rod Claddings Test Results with Shortened Fuel Rod Simulators," KfK Report 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," p. 56 in Specialists' Meeting on the Behavior of Water Reactor Fuel Elements Under Accident Conditions, CSNI Conference Proceeding, Spatind, Norway, September 13-16, 1976, published in 1976. Available in public technical libraries.

F. Erbacher, H. J. Neitzel, and K. Wiehr, "Studies on Zircaloy Fuel Clad Ballooning in a Loss-of-Coolant Accident -- Results of Burst Tests with Indirectly Heated Fuel Rod Simulators," p. 429 in Proceedings of the Fourth International Conference: Zirconium in the Nuclear Industry, ASTM Committee B-10, Stratford-upon-Avon, England, June 27-29, 1978, published in 1979. Available in public technical libraries.

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

| ROD # | RAMP RATE (°C/S) | PRESSURE AT RUPTURE (PSIG) | RUPTURE 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 |

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DATA REFERENCE D (Continued)

| ROD # | RAMP RATE (°C/S) | PRESSURE AT RUPTURE (PSIG) | RUPTURE TEMPERATURE (°C) | BURST STRAIN (%) | ENGINEERING BURST STRESS (KPSI) |
|---|---|---|---|---|---|
| ? | 11 | ? | 820 | 28 | ? |
| ? | 14 | ? | 820 | 38 | ? |
| 14 | 11 | 1420 | 810 | 38 | 9.87 |
| ? | 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, Kernforschungszentrum Karlsruhe, "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
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E. H. Karb, Kernforschungszentrum Karlsruhe, "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, Kernforschungszentrum Karlsruhe, "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 NRC PDR for inspection and copying for a fee.

Inpile, single rod, steam atmosphere.

| ROD # | RAMP RATE (°C/S) | PRESSURE AT RUPTURE (PSIG) | RUPTURE 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 |

DATA REFERENCE F (Open Square)

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

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

| ROD # | RAMP RATE (°C/S) | PRESSURE AT RUPTURE (PSIG) | RUPTURE 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 | 3C | 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, Kernforschungszentrum
Karlsruhe, "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, Kernforschungszentrum Karlsruhe, "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 NRC 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 RUPTURE (PSIG) | MEAN RUPTURE 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 | 50 |
| 3 | 7 | 725 | 830 | 44 | 5.05 | 52 |

DATA REFERENCE H (Inverted Triangle)

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

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

| ROD # | RAMP RATE (°C/S) | PRESSURE AT RUPTURE (PSIG) | RUPTURE 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 |

DATA REFERENCE I (Diamond)

R. H. Chapman, Oak Ridge National Laboratory, "Multirod Burst Test
Program Progess Report for July-December 1979," to be published about
July 1980.

Letter from R. H. Chapman, Oak Ridge National Laboratory, to D. A. Powers,
U.S. Nuclear Regulatory Commission, dated February 21, 1980. Available
in NRC PDR for inspection and copying for a fee.

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

| ROD # | RAMP RATE (°C/S) | PRESSURE AT RUPTURE (PSIG) | RUPTURE TEMPERATURE (°C) | MAXIMUM ROD STRAIN (%) | ENGINEERING BURST STRESS (KPSI) |
|-------|------|------|------|------|------|
| SR-47 | 10 | 1436 | 775 | 78 | 12.35 |
| SR-49 | 5 | 1107 | 783 | 95 | 9.52 |
| SR-51 | 0 | 1030 | 790 | 93 | 8.86 |
| SR-53 | 0 | 846 | 762 | 83 | 7.28 |
| SR-57 | 0 | 725 | 775 | 110 | 6.23 |
| SR-50 | 10 | 666 | 897 | 56 | 5.73 |
| SR-52 | 10 | 1437 | 761 | 49 | 12.36 |
| SR-60 | 28 | 1036 | 879 | 24 | 8.91 |
| SR-61 | 28 | 2073 | 762 | 31 | 17.83 |
| SR-62 | 28 | 608 | 937 | 31 | 5.23 |
| SR-63 | 0 | 822 | 760 | 99 | 7.07 |
| SR-64 | 5 | 1231 | 766 | 110 | 10.59 |
| SR-65 | 5 | 1307 | 748 | 74 | 11.24 |
| SR-67 | 1 | 645 | 824 | 107 | 5.55 |
| SR-69 | 1 | 579 | 854 | 116 | 4.98 |

DATA REFERENCE J (Closed Square)

Letter from F. J. Erbacher, Kernforschungszentrum Karlsruhe, to R. H. Chapman, Oak Ridge National Laboratory, dated October 16, 1979. Available in NRC PDR for inspection and copying for a fee.

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

| ROD # | RAMP RATE (°C/S) | PRESSURE AT RUPTURE (PSIG) | RUPTURE TEMPERATURE (°C) | BURST STRAIN (%) | ENGINEERING BURST STRESS (KPSI) |
|---|---|---|---|---|---|
| 100 | 10.4 | 1975 | 765 | 72.0 | 13.77 |
| 101 | 10.1 | 1920 | 755 | 72.6 | 13.38 |
| 102 | 10.5 | 1678 | 795 | 81.8 | 11.69 |
| 103 | 10.6 | 1393 | 825 | 92.5 | 9.71 |
| 104 | 10.4 | 1131 | 855 | 75.0 | 7.88 |
| 105 | 9.5 | 858 | 894 | 45.7 | 5.98 |
| 106 | 9.7 | 581 | 938 | 50.7 | 4.05 |
| 107 | 10.7 | 1122 | 864 | 86.0 | 7.82 |
| 108 | 10.1 | 1124 | 852 | 85.1 | 7.83 |
| 109 | 1.4 | 1976 | 723 | 76.2 | 13.78 |
| 110 | 1.9 | 1708 | 748 | 82.1 | 11.91 |
| 111 | 1.9 | 1428 | 779 | 81.2 | 9.95 |
| 112 | 1.7 | 1137 | 819 | 103.7 | 7.92 |
| 113 | 1.7 | 854 | 866 | 72.6 | 5.95 |
| 114 | 28.9 | 1971 | 794 | 36.8 | 13.73 |
| 115 | 29.3 | 1944 | 793 | 62.9 | 13.55 |
| 116 | 27.8 | 1721 | 802 | 43.6 | 12.00 |
| 117 | 29.2 | 1420 | 844 | 60.2 | 9.89 |
| 118 | 33.7 | 1106 | 916 | 36.8 | 7.71 |
| 119 | 35.0 | 1150 | 904 | 37.4 | 8.01 |
| 120 | 37.9 | 1427 | 883 | 50.4 | 9.94 |
| 121 | 8.9 | 1717 | 781 | 71.7 | 11.97 |
| 122 | 9.6 | 1434 | 810 | 57.2 | 10.00 |
| 123 | 9.0 | 1150 | 834 | 71.7 | 8.01 |
| 124 | 9.0 | 1148 | 837 | 66.1 | 8.00 |
| 125 | 8.9 | 876 | 885 | 72.6 | 6.10 |
| 126 | 9.9 | 2095 | 763 | 57.2 | 14.60 |
| 127 | 31.5 | 1991 | 827 | 45.1 | 13.88 |
| 128 | 25.2 | 1708 | 803 | 48.1 | 11.91 |
| 129 | 29.5 | 1711 | 817 | 57.2 | 11.93 |
| 130 | 31.5 | 1436 | 898 | 53.7 | 10.01 |
| 131 | 24.1 | 1147 | 870 | 51.9 | 7.99 |
| 132 | 25.4 | 1427 | 847 | 59.6 | 9.94 |
| 133 | 0.8 | 1438 | 737 | 75.0 | 10.03 |
| 134 | 1.6 | 1724 | 746 | 86.3 | 12.02 |
| 135 | 0.8 | 2001 | 701 | 79.1 | 13.95 |
| 136 | 0.9 | 1151 | 786 | 116.2 | 8.02 |
| 137 | 0.9 | 867 | 824 | 112.9 | 6.05 |
| 138 | 0.8 | 580 | 370 | 78.0 | 4.04 |

DATA REFERENCE K (Upright Triangle in Square)

L. M. Lowry, J. S. Perrin, A. J. Markworth, and W. J. Gallagher, Battelle Columbus Laboratories, "Evaluating Strength and Ductility of Irradiated Zircaloy, Task 5," USNRC Report NUREG/CR-0582, November 1978. Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.

L. M. Lowry, J. S. Perrin, A. J. Markworth, and M. P. Landow, Battelle Columbus Laboratories, "Evaluating Strength and Ductility of Irradiated Zircaloy, Task 5," USNRC Report NUREG/CR-0982, November 1979. Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.

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

| ROD # | RAMP RATE (°C/S) | PRESSURE AT RUPTURE (PSIG) | RUPTURE TEMPERATURE (°C) | BURST STRAIN (%) | ENGINEERING BURST STRESS (KPSI) |
|---|---|---|---|---|---|
| N8/9-29 | 28 | 140 | 1135 | 39 | 1.14 |
| N8/30-50 | 28 | 125 | 1071 | 25 | 1.02 |
| N8/71-91 | 28 | 198 | 1008 | 46 | 1.62 |
| NS/112-132 | 34 | 254 | 978 | 43 | 2.08 |
| N8/50-70 | 28 | 290 | 927 | 23 | 2.37 |
| O14/42-62 | 28 | 360 | 948 | 16 | 2.94 |
| G8/70-89 | 34 | 478 | 925 | 12 | 3.91 |
| F8/70-90 | 15 | 696 | 863 | 21 | 5.69 |
| A1/35-55 | 18 | 945 | 786 | 23 | 7.73 |
| A8/42-62 | 28 | 925 | 849 | 33 | 7.56 |
| M12/33-55 | 28 | 938 | 782 | 18 | 7.67 |
| M12/70-90 | 28 | 954 | 804 | 13 | 7.80 |
| F8/35-55 | 28 | 954 | 814 | 17 | 7.80 |
| A1/79-99 | 34 | 958 | 788 | 16 | 7.83 |
| K8/35-55 | 28 | 1265 | 825 | 16 | 10.35 |
| K8/14-34 | 28 | 1488 | 741 | 22 | 12.17 |
| K8/70-90 | 28 | 1550 | 792 | 30 | 12.68 |
| N8/91-111 | 28 | 1815 | 734 | 17 | 14.84 |
| K10/105-125 | 6 | 1162 | 746 | 34 | 9.51 |
| K10/71-91 | 6 | 218 | 897 | 16 | 1.78 |
| K10/35-55 | 15 | 139 | 999 | 16 | 1.14 |
| K10/14-34 | 18 | 150 | 931 | 17 | 1.23 |
| 47103-4 | 28 | 143 | 1066 | 37.5 | 1.09 |
| 47118-12 | 28 | 180 | 967 | 35.6 | 1.37 |
| 47104-3 | 28 | 242 | 934 | 29.9 | 1.84 |
| 47010-17 | 28 | 341 | 924 | 37.8 | 2.59 |
| 47110-4 | 28 | 650 | 854 | 15.9 | 4.95 |
| 47110-7 | 28 | 991 | 831 | 31.4 | 7.54 |
| 47110-14 | 28 | 1100 | 799 | 58.9 | 8.37 |
| 47111-12 | 28 | 1268 | 745 | 36.7 | 9.65 |
| 47110-20 | 28 | 1463 | 762 | 25.3 | 11.13 |
| 47111-6 | 5.6 | 833 | 778 | 74.7 | 0.34 |
| 47101-16 | 28 | 1200 | 788 | 17.1 | 91.3 |

DATA REFERENCE K (Continued)

| ROD<br>#  | RAMP<br>RATE<br>(°C/S) | PRESSURE<br>AT RUPTURE<br>(PSIG) | RUPTURE<br>TEMPERATURE<br>(°C) | BURST<br>STRAIN<br>(%) | ENGINEERING<br>BURST STRESS<br>(KPSI) |
|-----------|------------------------|----------------------------------|--------------------------------|------------------------|----------------------------------------|
| 47101-11  | 28 | 1237 | 783  | 18.3 | 9.42  |
| 47101-6   | 28 | 649  | 803  | 10.8 | 4.94  |
| 47101-3   | 28 | 500  | 892  | 9.9  | 3.80  |
| Archive 1 | 28 | 704  | 936  | 18.2 | 5.36  |
| Archive 2 | 28 | 1250 | 830  | 22.9 | 9.51  |
| 32211A-13 | 28 | 180  | 1043 | 12.3 | 1.37  |
| 32211A-20 | 28 | 248  | 877  | 19.1 | 1.89  |
| 31778A-13 | 28 | 419  | 860  | 17.8 | 3.19  |
| 31917A-16 | 28 | 674  | 850  | 11.9 | 5.13  |
| 31965A-12 | 28 | 918  | 827  | 31.0 | 6.98  |
| 31983A-15 | 28 | 1069 | 857  | 19.1 | 8.13  |
| 31785A-5  | 28 | 1273 | 782  | 20.2 | 9.69  |
| 31785A-9  | 28 | 1573 | 777  | 12.7 | 11.97 |

DATA REFERENCE L (Cross in Circle)

Bundles-7805, 7806, 7807, and 7808

S. Kawasaki, "Multirod Burst Tests at JAERI," paper presented at the 4th
JAERI-FRG-NRC Annual Fuel Behavior Information Exchange, Idaho Falls,
Idaho, June 22-29, 1979. Available in NRC PDR for inspection and copying
for a fee.

Letter from S. Kawasaki, Japan Atomic Energy Research Institute, to R. O.
Meyer, U.S. Nuclear Regulatory Commission, dated December 24, 1979.
Available in NRC PDR for inspection and copying for a fee.

Out-of-pile, 49-rod bundles, heated shroud, steam atmosphere.

| TEST # | INITIAL RAMP RATE (°C/S) | MEAN PRESSURE AT RUPTURE (PSIG) | MEAN RUPTURE TEMPERATURE (°C) | MEAN BURST STRAIN (%) | MEAN ENGINEERING BURST STRESS (KPSI) | REDUCTION IN FLOW AREA (%) |
|---|---|---|---|---|---|---|
| 7805 | 6.1-7.7 | 818 | 825 | ? | 6.56 | 85.3 |
| 7806 | 6.0-7.3 | 359 | 885 | ? | 2.88 | 35.6 |
| 7807 | 5.9-7.2 | 1111 | 765 | ? | 8.91 | 77.8 |
| 7808 | 7 | 578 | 855 | ? | 4.64 | 46.9 |

# APPENDIX B

## TABULATION OF CLADDING CORRELATIONS

Slow-Ramp Correlations

| Rupture Temperature (°C) | ≤10°C/s Burst Strain (%) | ≤10°C/s Flow Blockage (%) |
|---|---|---|
| 600 | 10 | 6.5 |
| 625 | 11 | 7.0 |
| 650 | 13 | 8.4 |
| 675 | 20 | 13.8 |
| 700 | 45 | 33.5 |
| 725 | 67 | 52.5 |
| 750 | 82 | 65.8 |
| 775 | 89 | 71.0 |
| 800 | 90 | 71.5 |
| 825 | 89 | 71.0 |
| 850 | 82 | 65.8 |
| 875 | 67 | 52.5 |
| 900 | 48 | 35.7 |
| 925 | 28 | 20.0 |
| 950 | 25 | 18.0 |
| 975 | 28 | 20.0 |
| 1000 | 33 | 24.1 |
| 1025 | 35 | 25.7 |
| 1050 | 33 | 24.1 |
| 1075 | 25 | 18.0 |
| 1100 | 14 | 9.2 |
| 1125 | 11 | 7.0 |
| 1150 | 10 | 6.5 |
| 1175 | 10 | 6.5 |
| 1200 | 10 | 6.5 |

Fast-Ramp Correlations

| Rupture Temperature (°C) | ≥25°C/s Burst Strain (%) | ≥25°C/s Flow Blockage (%) |
|---|---|---|
| 600 | 10 | 6.5 |
| 625 | 10 | 6.5 |
| 650 | 12 | 7.5 |
| 675 | 15 | 10.0 |
| 700 | 20 | 13.8 |
| 725 | 28 | 20.0 |
| 750 | 38 | 27.5 |
| 775 | 48 | 35.7 |
| 800 | 57 | 43.3 |
| 825 | 60 | 46.0 |
| 850 | 60 | 46.0 |
| 875 | 57 | 43.3 |
| 900 | 45 | 33.5 |
| 925 | 28 | 20.0 |
| 950 | 25 | 18.0 |
| 975 | 28 | 20.0 |
| 1000 | 35 | 25.7 |
| 1025 | 48 | 35.7 |
| 1050 | 77 | 61.6 |
| 1075 | 80 | 64.5 |
| 1100 | 77 | 61.6 |
| 1125 | 39 | 28.5 |
| 1150 | 26 | 18.3 |
| 1175 | 26 | 18.3 |
| 1200 | 36 | 26.2 |

# APPENDIX C

## FORMAL COMMENTS ON DRAFT NUREG-0630

Listed below are the formal comments that have been received on draft NUREG-0630, which was issued in November 1979 for a critique by the technical community. These comments have been placed in the NRC PDR (Memorandum from D. A. Powers, NRC, to Public Document Room, "Formal Comments on Draft NUREG-0630," accession number 8002280641, March 14, 1980. Available in NRC PDR for inspection and copying for a fee.)

1.  Memorandum from D. A. Hoatson, NRC, to K. Kniel, "Review of NUREG-0630 In ECCS Cladding Models," December 5, 1979.

2.  Letter from R. H. Buchholz, General Electric Company, to R. P. Denise, NRC, Subject: Comments on The Draft Report "Cladding Swelling and Rupture Models for LOCA Analysis," NUREG-0630 Dated November 8, 1979, dated December 7, 1979.

3.  Letter from D. O. Hobson, Oak Ridge National Laboratory, to K. Kniel, NRC, dated December 10, 1979.

4.  Telex from J. Hannaford, United Kingdom Nuclear Installations Inspectorate, to R. O. Meyer, NRC, Subject: Comments on NUREG-0630 "Cladding Swelling and Rupture Models for LOCA Analysis," dated December 10, 1979.

5.  Letter from J. H. Taylor, Babcock & Wilcox Company, to R. P. Denise, NRC, dated December 10, 1979.

6.  Letter from G. F. Owsley, Exxon Nuclear Company, to R. P. Denise, NRC, dated December 10, 1979.

7.  Letter from D. E. Vandenburgh, Yankee Atomic Electric Company, to R. P. Denise, NRC, Subject: Technical Review of Draft NUREG-0630 dated December 10, 1979.

8.  Letter from T. M. Anderson, Westinghouse Electric Corporation, to R. P. Denise, NRC, dated December 10, 1979.

9.  Memorandum from R. O. Meyer, NRC, to R. P. Denise, "Evaluation of Westinghouse Comments on Draft NUREG-0630," January 5, 1980.

10. Letter from A. E. Scherer, Combustion Engineering Company, to R. P. Denise, NRC, Subject: Review of Draft Report NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis," dated December 11, 1979.

11. Letter from P. L. Rittenhouse, Oak Ridge National Laboratory, to D. A. Powers, NRC, Subject: NUREG-0630, dated December 20, 1979.

12. Letter from T. F. Kassner and H. M. Chung, Argonne National Laboratory, to K. Kniel, NRC, Subject: Review of CP6 Report on ECCS Cladding Models, dated January 3, 1980.

13  Letter from W. B. Loewenstein, Electric Power Research Institute, to R. Budnitz, NRC, dated January 9, 1980.

14. Letter from R. O. Meyer, NRC, to W. B. Loewenstein, Electric Power Research Institute, dated February 5, 1980.

15. Letter from R. H. Chapman, Oak Ridge National Laboratory, to K. Kniel, NRC, dated January 14, 1980.

16. Letter from R. H. Chapman, Oak Ridge National Laboratory, to D. A. Powers, NRC, Subject: Additional Comments on NUREG-0630, dated February 21, 1980.

17. Letter from F. J. Erbacher, Kernforschungszentrum Karlsruhe, to D. A. Powers, NRC, dated February 13, 1980.

18. Telex from R. O. Meyer, NRC, to R. Van Houten, NRC, dated February 19, 1980.

19. Telex from R. Van Houten, NRC, to R. Meyer, NRC, dated February 21, 1980.

20. Telex from D. Powers, NRC, to A. Fiege, Kernforschungszentrum Karlsruhe, dated February 27, 1980.

21. Telex from A. Fiege, Kernforschungszentrum Karlsruhe, to D. Powers, NRC, dated March 4, 1980.

22. Letter from R. H. Chapman, Oak Ridge National Laboratory, to D. A. Powers, NRC, Subject: Revised Figure 1 for Additional Comments on NUREG-0630 dated 2-21-80, dated March 6, 1980.

| NRC FORM 335 (7-77)    U.S. NUCLEAR REGULATORY COMMISSION    **BIBLIOGRAPHIC DATA SHEET** | 1. REPORT NUMBER (Assigned by DDC) NUREG-0630 |
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| | 3. RECIPIENT'S ACCESSION NO. |

| 7. AUTHOR(S)   DALE A. POWERS AND RALPH O. MEYER | 5. DATE REPORT COMPLETED |
|---|---|
| | MONTH: March   YEAR: 1980 |

| 9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) CORE PERFORMANCE BRANCH DIVISION OF SYSTEMS SAFETY OFFICE OF NUCLEAR REACTOR REGULATION U.S. NUCLEAR REGULATORY COMMISSION WASHINGTON, D. C. 20555 | DATE REPORT ISSUED |
|---|---|
| | MONTH: April   YEAR: 1980 |
| | 6. (Leave blank) |
| | 8. (Leave blank) |

| 12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) CORE PERFORMANCE BRANCH DIVISION OF SYSTEMS SAFETY OFFICE OF NUCLEAR REACTOR REGULATION U.S. NUCLEAR REGULTORY COMMISSION WASHINGTON, D. C. 20555 | 10. PROJECT/TASK/WORK UNIT NO. |
|---|---|
| | 11. CONTRACT NO. |

| 13. TYPE OF REPORT   TECHNICAL REPORT | PERIOD COVERED (Inclusive dates)   N/A |
|---|---|
| 15. SUPPLEMENTARY NOTES | 14. (Leave blank) |

16. ABSTRACT (200 words or less)

A DESCRIPTION IS GIVEN OF FUEL CLADDING BEHAVIOR AS IT IS MODELED FOR EMERGENCY-CORE-COOLING-SYSTEM (ECCS) EVALUATIONS IN THE SAFETY ANALYSIS OF LOSS-OF-COOLANT ACCIDENTS (LOCAS). DATA ARE TABULATED FROM EXPERIMENTS THAT EMPLOYED INTERNALLY HEATED ZIRCALOY CLADDING THAT WAS RUPTURED IN AQUEOUS ATMOSPHERES, AND NEW CORRELATIONS BASED ON THESE DATA ARE GIVEN FOR CLADDING RUPTURE TEMPERATURE, CLADDING BURST STRAIN, AND FUEL ASSEMBLY FLOW BLOCKAGE. COMPARISONS OF THESE CORRELATIONS WITH INDUSTRY MODELS THAT ARE USED IN CURRENT LICENSING ANALYSES FOR COMMERCIAL NUCLEAR POWER PLANTS REVEAL SUBSTANTIAL DIFFERENCES. THE CORRELATIONS IN THIS REPORT ARE INTENDED TO BE USED AS LICENSING STANDARDS FOR FUTURE LOCA ANALYSES UNTIL SUCH TIME THAT THE CONTINUING RESEARCH PROGRAM WOULD DEMONSTRATE A NEED FOR FURTHER REVISIONS.

| 17. KEY WORDS AND DOCUMENT ANALYSIS | 17a. DESCRIPTORS |
|---|---|
| FUEL ROD LOCA FLOW BLOCKAGE ECCS ZIRCALOY | METAL CYLINDER OF $UO_2$ PELLETS LOSS-OF-COOLANT ACCIDENT REDUCTION-IN-FLOW-AREA EMERGENCY-CORE-COOLING SYSTEM FUEL ROD CLADDING MATERIAL |

17b. IDENTIFIERS/OPEN-ENDED TERMS

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