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ACRS-1718

DATE ISSUED: 4/30/80

MEETING MINUTES OF THE COMBINED
ACRS ECCS/REACTOR FUEL SUBCOMMITTEE MEETING
FEBRUARY 14, 1980
WASHINGTON, D.C.

On February 14, 1980 the Combined ACRS ECCS/Reactor Fuel Subcommittee met in Washington, D.C., to discuss the effects of externally mounted thermocouples on the LOFT fuel rods, and proposed changes in the ECCS fuel cladding rupture models for Appendix K to 10 CFR 50.46, and the effect of these changes on vendor's evaluation models. The notice of the meeting appeared in the Federal Register on January 30, 1980. There were no requests for oral or written statements from members of the public and none were made at the meeting. Attachment A is a copy of the meeting agenda. The attendees list is Attachment B. Attachment C is a tentative schedule of presentations for the meeting. Selected slides and handouts from the meeting are Attachment D to these minutes. A complete set of slides and handouts is attached to the office copy of these minutes.

OPEN SESSION (8:30 am - 5:35 pm) INTRODUCTION

Dr. Plesset, acting as Chairman of the Combined Subcommittee, called the meeting to order at 8:35 am. The Chairman explained the purpose of the meeting and the procedures for conducting the meeting, pointing out that Mr. Paul Boehnert was the Designated Federal Employee in attendance.

Dr. Plesset begin the meeting by commenting on the so called "fin effect" seen with the externally mounted LOFT thermocouples. The Chairman said that tests conducted both in the US and overseas show that externally mounted thermocouples do effect the test results. Dr. Plesset expressed concern that NRC research did not seem to agree that the fin effect is a problem. Dr. Catton also expressed concern regarding the impact of the fin effect and suggested NRC address the use of LOFT results vis-a-vis code development. Dr. Plesset also expressed concern that anomalous data from LOFT may be used to modify such predictive codes as RELAP or TRAC.

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THE SURFACE MOUNTED OR EXTERNAL THERMOCOUPLE PROBLEM - DR. G. D. McPHERSON (NRC-LOFT PROGRAM MANAGER)

Dr. McPherson said he would discuss the problem of using externally mounted thermocouples, highlight results of recent PBF tests conducted to attempt to quantify the fin effect, review out-of-pile tests conducted both in the US and overseas, and finally state conclusions regarding what is known to date. In response to Dr. Plesset's concern noted above, Dr. McPherson said that none of the LOFT data has been used to modify such codes as TRAC or RELAP.

Dr. McPherson described the thermocouples used in LOFT and how they are mounted on selected fuel rods. He also briefly reviewed results of the LOFT L2-2, and L2-3 tests that show that the core experienced an early quench prior to refill.

Dr. McPherson discussed the PBF-LOCA thermocouple effects tests (TC-1). The objectives of the tests were to determine if external thermocouples influence fuel rod behavior during a LOCA, and determine whether the thermocouples accurately measured cladding temperature. The tests simulated the blowdown and rewet that was seen in the LOFT tests. Figures D 1-3 detail the test series, the test geometry, and the test train used. Selected test rods contained internally and/or externally mounted thermocouples. Dr. McPherson said that the tests studied such affects as delay of CHF, reduction of fuel rod temperature, and earlier than expected reflooding of the core.

Turning to the test results, Dr. McPherson noted the following points:

- The tests showed a definite fin effect during film boiling (Figure D-4). Dr. McPherson said this result and others can be used to estimate the magnitude of the effect.
- Dr. McPherson showed a data plot for the TC-1 blowdown peak temperatures. A best fit curve (Figure D-5 curve 1) was drawn through the data for the rods with internal thermocouples. The slope of the curve was given as 60^{o}K per/sec. Dr. McPherson said that this meant that delaying DNB will raise the PCT by 60^{o}K per/sec. Based on data results from the external thermocouple rods, Dr. McPherson suggested one could draw a parallel curve (labeled McPherson curve on Figure D-5) that illustrates the fin

effect. Dr. Lienhard questioned the positioning of this curve given the data scatter, and the fact that there were no data points on the upper portion of the curve.

- Concerning the reflood portion of the TC-1 experiment, Dr. McPherson noted that all the rod temperatures begin to decrease about the same time, whether or not they had external thermocouples (Figure D-6).

Dr. McPherson discussed the conclusions derived from the TC-1 experiments. He said that NRC believes the external thermocouples influenced the LOFT L2-2 and L2-3 experiments in the following manner: (1) PCT decreased by approximately 30^{o}K; (2) fin effect during blowdown is variable (0-90^{o}K) and dependent upon the velocity of the blowdown fluid; (3) rods with and without external thermocouples saw temperature decreases at the same time as reflood commenced; and (4) there was a faster quench during reflood.

Dr. McPherson said that another set of tests at PBF (TC-2) are scheduled to provide additional information on the fin effect, especially during the blowdown-quench period. In response to a question from Dr. Plesset, Dr. Tong said that it is hoped that a correction factor can be determinted for the fin effect, based on the results from the TC-1 and TC-2 tests.

There was Subcommittee discussion on the use of various types of internally mounted thermocouples and their impact on the test results .

Dr. McPherson also discussed results of the PBF LOFT Lead Rod (LLR) tests. The tests were performed to provide information on the expected response of the LOFT fuel, and to determine if any special fuel pre-conditioning was necessary. Secondary test objectives included determining the reliability of the LOFT thermocouples. During the blowdown phase of the test, it was noted that the surface thermocouples delayed CHF in the upper portion of the fuel rods. It was also noted that DNB was occurring at lower elevations on the rods than the thermocouples, resulting in higher temperatures at lower elevations. Figure D-7 provides details of the LLR test results. Dr. McPherson also said that during reflood, very little or no thermocouple effects were seen for either low, moderate, or high flooding rates.

Dr. McPherson concluded that he believes the LLR tests support the conclusions drawn from the results of the TC-1 tests.

The results of out-of-pile electrically heated rod tests conducted in the U.S. (LOFT Technical Support Facility, UCLA) and overseas (Germany, England, Switzerland, Norway, and Holland) were discussed. Dr. McPherson said that there are important differences between electrically heated rods and nuclear heated rods, and one cannot directly apply the results of electrically heated rod tests to a in-reactor situation. Generally speaking, electrically heated rods with external thermocouples exhibit faster quench times than is seen with nuclear rods with external thermocouples.

Dr. McPherson concluded his presentation by noting that the results shown today, plus results from future tests, should resolve the fin-effect thermocouple problem within one year.

CLADDING SWELLING AND RUPTURE MODELS FOR LOCA ANALYSIS - R. MEYER, D. POWERS, J. ROSENTHAL (NRC)

Dr. Meyer began the NRC discussion of the new proposed cladding swelling and rupture models for LOCA analysis as detailed in the draft NUREG-0630 (Cladding Swelling and Rupture Models for LOCA Analysis). Dr. Meyer noted that NRC has received comments from the reactor vendors, and organizations such as EPRI, as well as comments from representatives of the United Kingdom, Germany, and Japan on the draft NUREG. Dr. Meyer also noted that Mr. A. Mann from the Springfields Laboratories in the UK would make a presentation at today's meeting.

Dr. Meyer said that the cladding behavior correlations of interest for this discussion included rupture temperature versus engineering hoop stress, burst strain versus rupture temperature, and PWR assembly flow blockage versus rupture temperature. Figure D-8 shows these three parameters in relationship to all the important parameters used in a LOCA calculation mandated by 10 CFR 50.46 and Appendix K.

As a result of review of the CE flow blockage model in 1977, NRC decided that review of all the vendor's models was necessary, given the disparity seen in the burst strain and flow blockage models (Figures D 9-10). NRC then decided that it should define acceptable models in these areas. In response to a question from Mr. Etherington on the importance of accounting for the cladding strain rate, Dr. Meyer said that with the exception of the W small break model, all the other vendors do not account for the strain rate parameter. He said that the new NRC cladding models would account for strain rate.

Mr. J. Rosenthal (DOR) discussed the interim actions taken on operating reactors for this concern. He described the results of actions taken in early November 1979 when it was thought by the NRC that there were potential deficiencies in the vendor's ECCS evaluation models. It was concluded that no safety problem existed, however "no safety problem" was defined to mean that: (1) peak clad temperature (PCT) predictions were insensitive to fuel clad models, or (2) existing models were adequate over the narrow range of applicability, or (3) sufficient margin was available with existing models, or (4) off-setting credits existed for other model changes which were under NRC review. Mr. Rosenthal noted that Westinghouse found it necessary to make use of off-setting conservatisms to overcome a substantive (~2-700^{o}F) increase in PCT.

Dr. D. Powers (NRC-DSS) discussed the rupture temperature, burst strain, and flow blockage correlations developed by NRC and described in NUREG-0630. The data base used for developing these correlations was restricted to tests in dry steam and which made use of internally heated rods. The data was obtained from tests conducted at Oak Ridge, Battelle Columbus Laboratories, the KFK Facility in Germany, and the Japanese Atomic Energy Research Institute. Both in- and out-of-pile tests were included in the data base. Dr. Powers noted that the above correlations represent NRC's best estimate of the subject models.

Dr. Powers discussed the rupture temperature correlation used in the NRC model (Figure D-11). He noted that this curve is based on the above discussed data and the data shows a strong heating ramp rate dependence. For the purposes of the NUREG report, the NRC has defined a slow ramp rate as $\leq 10^{o}$C/second and the fast ramp rate was defined as $\geq 2\ ^{o}$C/second.

The slow-ramp and fast-ramp burst strain curves developed by NRC were discussed (Figures D-12-13). The development of the burst strain curves was based in part on the work of Kassner and Chung conducted at ANL (Figure D-14).

Dr. Powers noted that the slow and fast-ramp burst strain curves have been modified from the initially developed correlation, largely at the recommendation of Dr. Chapman from ORNL (Figures D-14A-15). There was considerable discussion over the applicability of the data used to develop the high temperature portion of the fast-ramp burst strain curve (Figure D-15-arrow). These data were taken from the Oak Ridge tests; however, in response to a question from Dr. Shewmon, Dr. Chapman of ORNL expressed doubt that this data should be characterized as fast-ramp data, due to problems encountered during the experiment. Dr. Shewmon observed that there appears to be a doubious basis for the development of that portion of the curve.

The derivation of the flow blockage model was described by Dr. Powers (Figure D-16). He said the NRC blockage model is expressed as a function of the cladding rupture temperature. Figures D-17 and D-18 show the flow blockage curves derived by NRC along with the applicable data. Figures D-19 and D-20 show the differences between the draft and final flow blockage correlations.

There was extensive Subcommittee discussion centering on the use and interpretation of the data as well as the assumptions that went into the development of the above curves.

Dr. Meyer discussed the proposed schedule for implementing the revised fuel cladding models (Figures D-20A-21). He stated that NRC would like an ACRS letter on the NUREG report at the March 1980 meeting. NRC would than issue a final version of the NUREG report around April 1, 1980, along with requirements for vendor reanalysis of their ECCS models. Dr. Shewmon expressed concern over what he felt was excessive conservatisms on the Staff's part in developing the above curves, and the relation of this information to what happens on a realistic basis given a LOCA in a power reactor. He also said he felt uneasy with the NRC interpretation of the test data. He asked if tests planned by NRC research at the NRU and LOFT facilities, as well as overseas tests would relate to the information being considered today. Dr. Meyers replied that he believes that the work documented in NUREG-0630 represents a significant improvement in the cladding models over the situation that has existed since 1974. Dr. Rosztoczy noted that NRC hopes to avoid changing the curves for a long time. Dr. Rosztoczy also estimated that the changes proposed in 0630 will cost the vendors about $10 million to implement.

COMMENTS BY M. L. PICKLESIMER - FUEL BEHAVIOR RESEARCH BRANCH - NRC

Dr. Picklesimer gave a brief presentation commenting on the NUREG report. He said he objects to the use of burst strain curves to determine flow blockage, since it leads to an overally conservative situation. He also noted however that at this time he has no alternative to the method used by the Staff, principally because he has not had time to work on this problem due to the TMI-2 accident. Dr. Meyer said NRC has only used brust strain to develop an average strain which in turn is converted to a value for flow blockage.

COMMENTS BY A. MANN - SPRINGFIELDS NUCLEAR POWER DEVELOPMENT LABORATORY

Mr. A. Mann from the Springfields Nuclear Power Development Laboratories of the UKAEA discussed the position of the UKAEA concerning clad deformation following a LOCA and the future work needed to clarify present uncertainties in this area. He noted that the amount of cladding strain is determined by temperature, the time the clad is at temperature, and circumferential temperature variation of the clad, the last probably being the most important parameter.

Addressing the potential problem of co-planar blockage, Mr. Mann noted that the location of the deformation depends primarily on the temperature distribution of the cladding. The temperature distribution in turn depends on the axial variation in power of decay heat in the rod and the heat transfer at the cladding surface. If these two parameters are similar in adjacent rods, deformation is likely to be co-planar.

The UKAEA believes that a predictive code is needed that can successfully model the interaction of the thermal-hydraulics and associated clad deformation parameters. Further experiments should focus on such parameters as thermal-hydraulics (dryout, rewet, and heat transfer during reflood), clad deformation, and comparison of in-pile and out-pile experiments. In response to a question from Dr. Shewmon, Mr. Mann stated that he believes the NRC may have underestimated the degree of blockage, given a worst-case LOCA, i.e. cladding swell could be co-planar. However, Dr. Mann went on to say that on a judgment basis, he feels the draft report is probably conservative but the problem, as he sees it, is proving it is conservative.

COMMENTS BY T. KASSNER - ANL

Dr. Tom Kassner from Argonne National Laboratory provided a presentation that discussed the relationships among various parameters used to develop cladding embrittlement criteria. The central theme of Kassner's presentation was that the new NRC flow blockage curves should be evaluated vis-a-vis the cladding oxidation requirements of Appendix K to determine the overall effect of clad swell and rupture in a LOCA situation. He indicated that substantial wall thinning could lead to violating the 17% oxidation limit specified in Appendix K at PCTs well below the 2200°F limit. NUREG-0630 has not considered this.

WESTINGHOUSE COMMENTS ON THE NRC FUEL ROD MODELS - D. BURMAN - W

Mr. Dennis Burman provided Westinghouse comments on the NRC fuel rod models; he commented on the burst temperature, burst strain, and flow blockage correlations proposed by the NRC Staff. Regarding the burst temperature correlation, Mr. Bruman said that Westinghouse agrees with the NRC that there is a heat-up rate dependence for zircaloy cladding, but that the Westinghouse model accounts for known biases in burst temperature measurements. Commenting on the burst strain correlation, Mr. Burman showed slides that he said indicated the Westinghouse data envelopes pertinent data from other sources. He said the NRC burst strain curves are upper bound curves, not best estimate as stated in NUREG-0630.

In describing the flow blockage correlations, Mr. Burman referenced some Japanese multi-rod burst tests that resulted in a large degree of co-planar blockage (Figures D-22-23). Mr. Burman said he believed this co-planar blockage was the result of the tungsten wire electrical heating element used (Figure D-24). (Note: Dr. Kawasaki was contacted by NRC concerning Mr. Burman statement regarding cause of the co-planar blockage. Dr. Kawasaki said that he does not believe the heater design was the cause of the co-planar blockage.

Rather it was probably due to effect of the steam flow and/or the grid spacers used in the experiment.) Westinghouse believes their flow blockage model is sufficiently conservative.

In conclusion, Mr. Burman stated that the preliminary nature of current data should preclude development of a new cladding model at this time, and noted that there are several tests scheduled for the near future which would provide data for development of more definitive models. He also said that tests in Germany show that high degrees of blockage (90%) do not adversely impact PCT, thus there is no apparent safety issue, and therefore there is no need for new models at this time.

COMMENTS BY R. CHAPMAN - ORNL

Dr. Chapman from ORNL briefly discussed the use of data from his single-rod and multi-rod burst test. He detailed information he had noted earlier in the day; that is, some of the high temperature (beta-range) burst-strain data should not be characterized as fast-ramp data. He noted that other beta-range tests may suffer from similar problems. Mr. Chapman also said that his tests show a clear heat-up rate effect. (W stated that the heat-up rate effect was minor.)

Dr. Meyer made some summarizing remarks. He believes that NRC and Westinghouse are in agreement concerning the use of the strain data base and that the W flow blockage model is in fairly good agreement with the NRC model. Dr. Meyer also noted however that he believes Westinghouse mischaracterized the NRC blockage model concerning the consideration of average versus maximum flow blockage in a bundle. Dr. Meyer also requested explicit ACRS comments on the adoption of the new NRC models, in particular whether or not the Committee finds the models acceptable and what should be done if they do not.

Dr. Shewmon commented that the NRC should present information on the NUREG report to the full Committee in March, but noted that one of the central questions in his mind was whether it is better to proceed now, as the Staff is proposing, or wait for a year until additional test data has been generated. The Chairman also expressed concern over how well the Staff has adequately allowed for what might take place in the core of a power reactor. Dr. Shewmon suggested that the Staff address the question of how serious is the impact on plant safety of waiting an additional year for the new test data noted above.

The meeting was adjourned at 5:30 p.m.

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NOTE: Additional meeting details can be obtained from a transcript located in the NRC Public Document Room, at 1717 H Street, N.W., Washington, D.C., or can be obtained from International Verbatim Reporters, Inc., 499 South Capitol Street, S.W., Suite 107, Washington, D.C. 20002.

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Subcommittees on Emergency Core Cooling Systems and Reactor Fuels; Meeting

The ACRS Subcommittees on Emergency Core Cooling Systems and Reactor Fuels will hold a joint meeting on February 14, 1980 in Room 1046, 1717 H St., NW., Washington, DC 20555. Notice of this meeting was published January 22, 1980.

In accordance with the procedures outlined in the Federal Register on October 1, 1979, (44 FR 56408), oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The agenda for subject meeting shall be as follows: *Thursday, February 14, 1980, 8:30 a.m. until the conclusion of business each day.*

The Subcommittee may meet in Executive Session, with any of its consultants who may be present, to explore and exchange their preliminary opinions regarding matters which should be considered during the meeting.

At the conclusion of the Executive Session, the Subcommittee will hear presentations by and hold discussions with representatives of the NRC Staff, Westinghouse, and other interested persons regarding: (1) proposed changes in the fuel clad rupture models for Appendix K to 10 CFR 50.54 and the effect of these changes on vendor evaluation models, (2) the effects of externally mounted thermocouples on LOFT fuel, (3) the results of the L3-1 Test, and (4) the analysis of small break LOCAs in Westinghouse UHI reactors.

In addition, it may be necessary for the Subcommittee to hold one or more closed sessions for the purpose of exploring matters involving proprietary information. I have determined, in accordance with Subsection 10(d) of the Federal Advisory Committee Act (Pub. L. 92-463), that, should such sessions be required, it is necessary to close these sessions to protect proprietary information. See 5 U.S.C. 552b(c)(4).

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant Designated Federal Employee, Dr. Andrew L. Bates (telephone 202/634-3287) between 8:15 a.m. and 5:00 p.m., EST.

Background information concerning items to be discussed at this meeting can be found in docments on file and available for public inspection at the NRC Public Document Room, 1717 H Street, NW., Washington, DC 20555.

Dated: January 24, 1980.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 80-2778 Filed 1-29-80; 8:45 am]

BILLING CODE 7590-01-M

ATTACHMENT A

MEETING OF THE COMBINED
ACRS ECCS/REACTOR FUEL SUBCOMMITTEE MEETING
FEBRUARY 14, 1980
WASHINGTON, D.C.

ATTENDEES LIST

ACRS

M. Plesset, Chairman, ECCS
P. Shewmon, Chairman, Reactor Fuel
H. Etherington, Member
A. Acosta, Consultant
I. Catton, Consultant
J. Lienhard, Consultant
F. Nichols, Consultant
Y. Chen, Special Consultant
P. Boehnert, Staff*

*Designated Federal Employee

EXXON NUCLEAR

G. Owsley

COMBUSTION ENGINEERING

G. Menzel
E. F. Jageler
J. M. Cicerchia

VEPCO

N. P. Wolfhope

UK ATOMIC ENERGY AUTHORITY

C. A. Mann

UK NUCLEAR INSTALLATIONS INSPECTORATE

L. G. Williams

YANKEE ATOMIC ELEC

K. E. St. John

OAK RIDGE NATIONAL LAB

R. H. Chapman

NRC

H. Sullivan
G. McPherson
R. Landry
L. S. Tong
M. L. Picklesimer
P. S. Anderson, Consultant

ARGONNE NATIONAL LAB

T. F. Kassner

DUKE POWER CO

S. T. ROSE

WESTINGHOUSE

D. L. Burman
S. D. Kopelic
V. J. Esposito

BABCOCK & WILCOX

H. A. Bailey

ATTACHMENT B

Schedule for ACRS
ECCS/Reactor Fuels Subcommittee Meeting
February 14, 1980

Opening Comments - Executive Session - M. Plesset/P. Shewmon	8:30 - 8:45 a.m.
I. Review of PBF Tests TC-1 and the Effects of Fin Thermocouple on LOFT Fuel Thermal Performance - D. McPhearson	8:45 - 9:30 a.m.
II. Review of LOFT L3-1 Test - R. Landry	9:30 - 10:30 a.m.
Break	10:30 - 10:40 a.m.
V. Review of LOFT L3-2 Test - R. Landry	10:40 - 11:30 a.m.
. Review of Proposed NRC Fuel Clad Swelling and Rupture Models	
a) Background and Status - R. Meyer	11:30 - 11:40 a.m.
b) Interim Actions on Operating Reactors - J. Rosenthal	11:40 - 11:50 a.m.
c) Description of Clad Models - D. Powers	11:50 - 12:30 p.m.
Lunch	12:30 - 1:30 p.m.
d) Plan for Final Resolution - R. Meyer	1:30 - 1:45 p.m.
M. Picklesimer Comments on Fuel Model	1:45 - 1:50 p.m.
Discussion	1:50 - 2:00 p.m.
e) Comments on NRC Fuel Model	
1) United Kingdom - A. Mann	2:00 - 2:20 p.m.
Discussion	2:20 - 2:40 p.m.
2) T. Kassner	2:40 - 3:00 p.m.
Discussion	3:00 - 3:20 p.m.
3) Westinghouse - D. Burman	3:20 - 3:40 p.m.
General Discussion	3:40 - 4:00 p.m.
Adjourn	4:00 p.m.

ATTACHMENT C

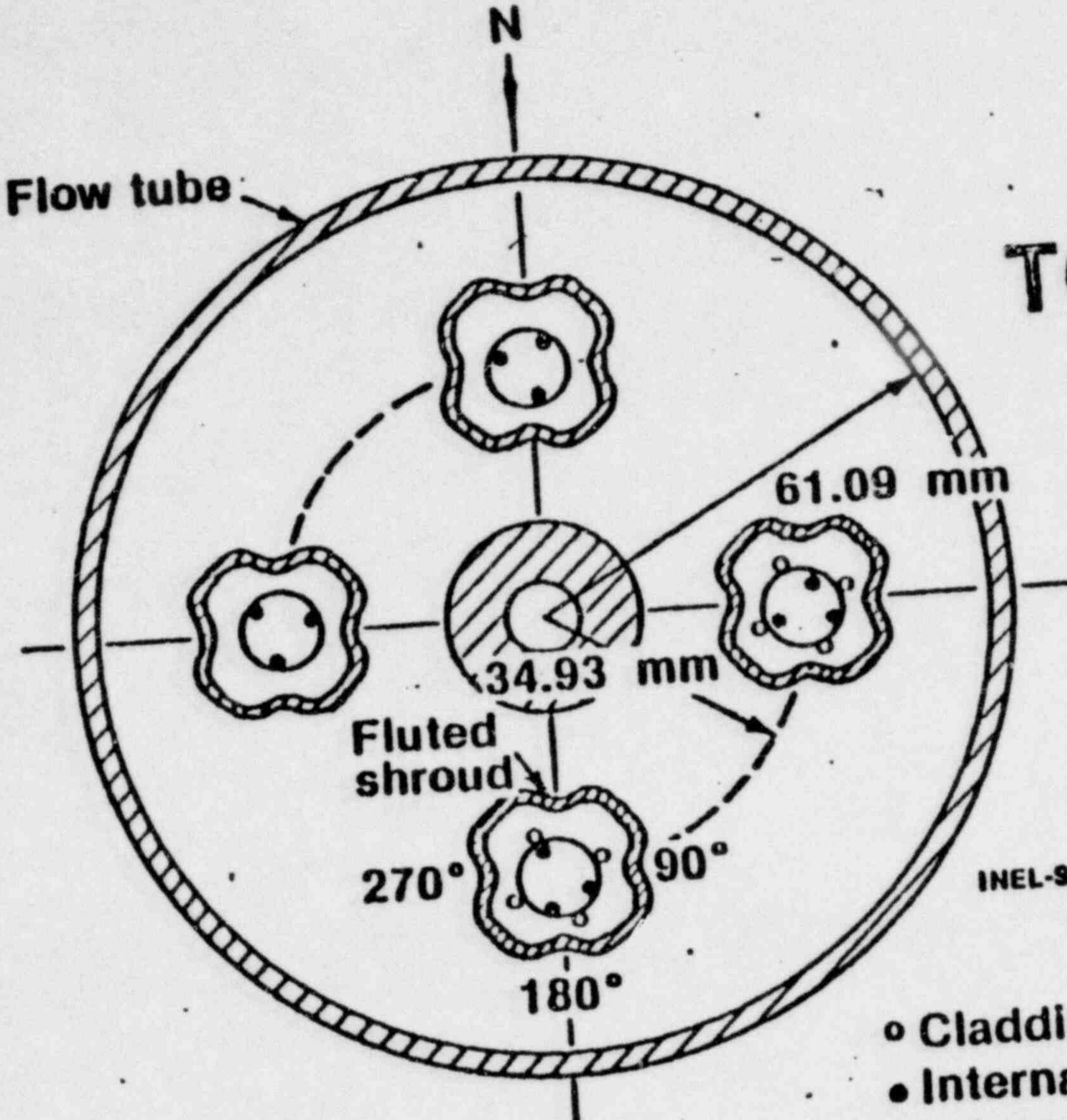
FIGURE D-1

TC-1 Test Series

Test	Blowdown System Operation	Reflood Rate cm/s
1	2 second slug	4
2	4 second slug	4
3	6 second slug	4
4	6 second slug	4

INEL-S-22 767

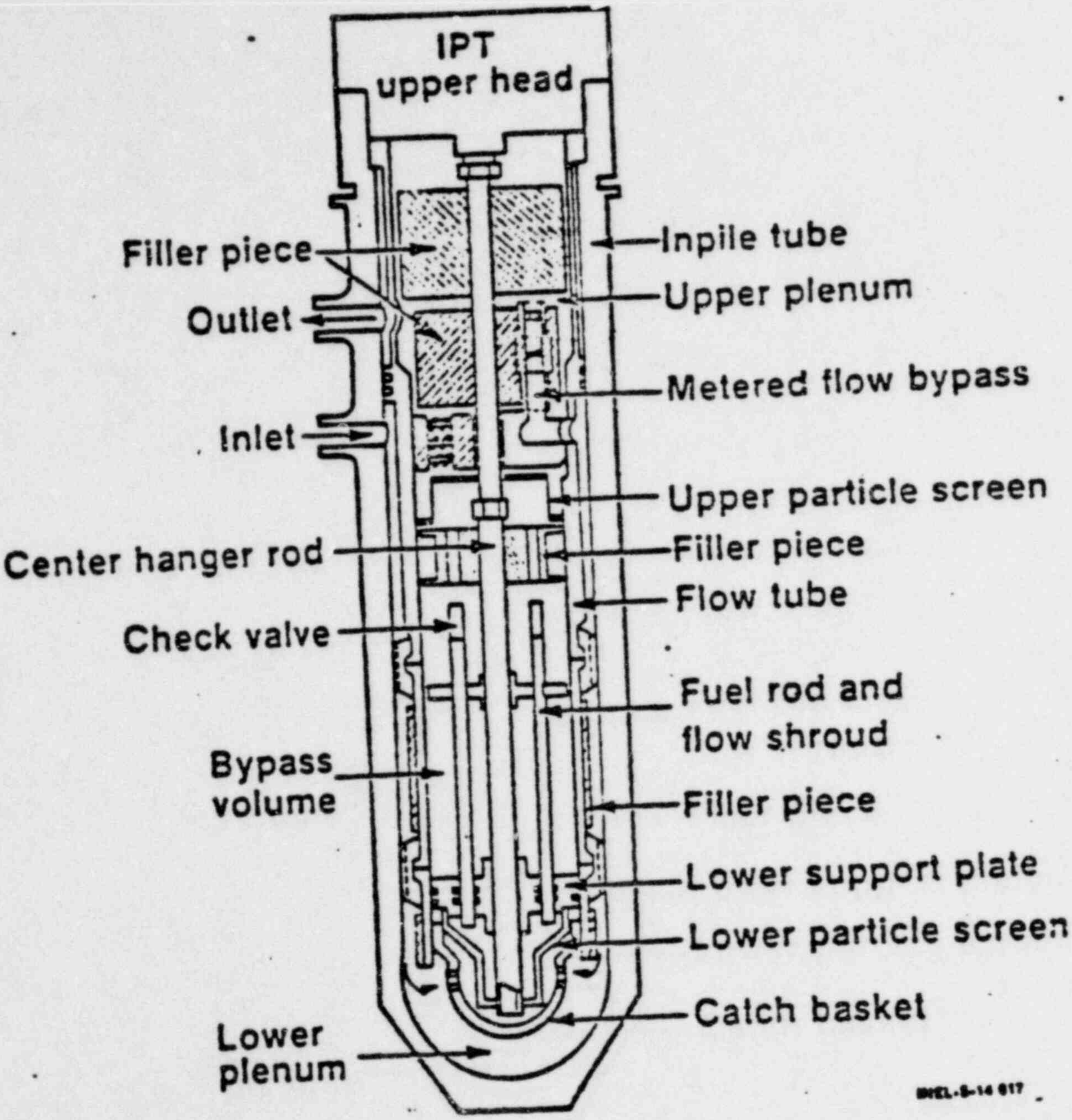
TC-1 Geometry



INEL-S-22 758

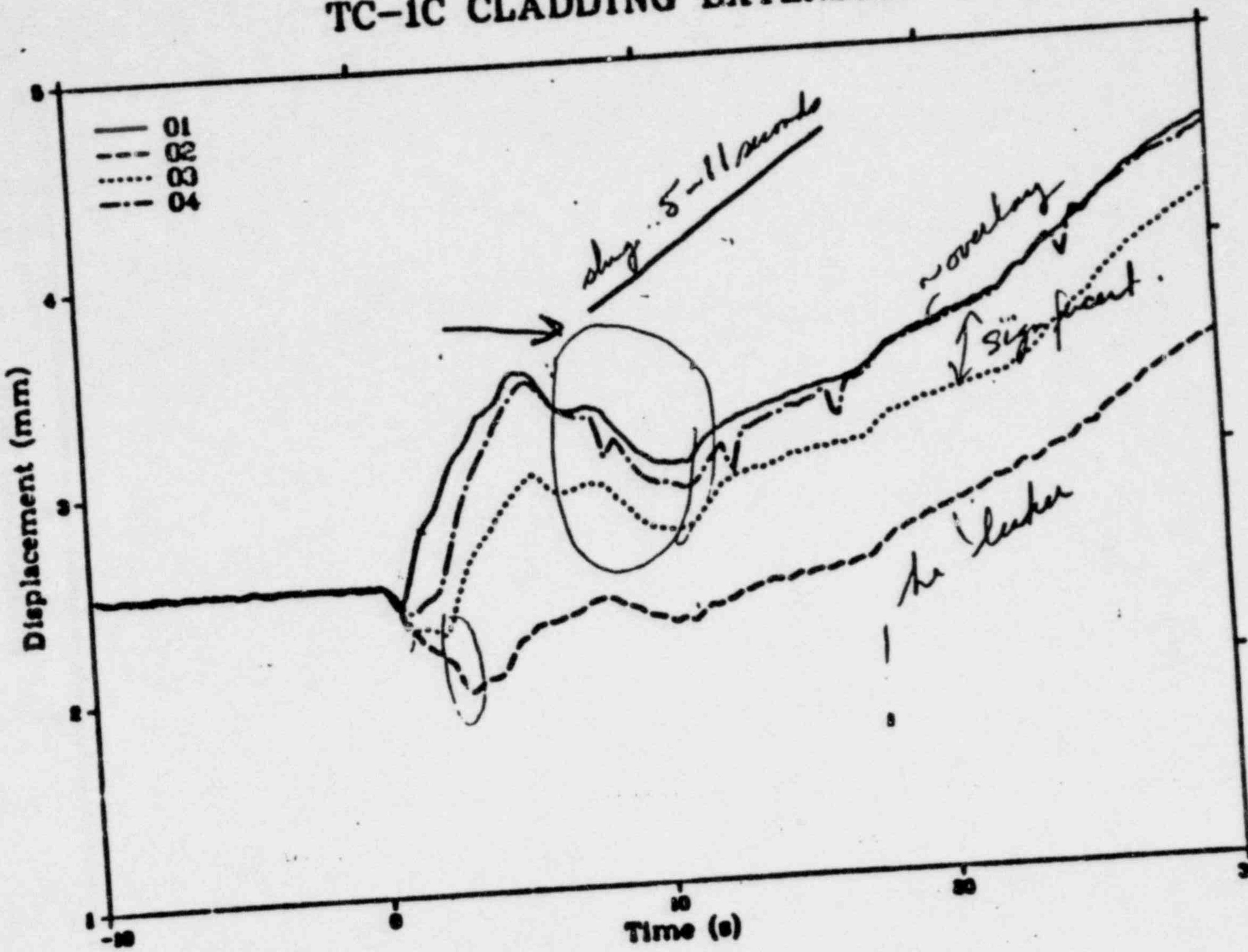
- Cladding thermocouples
- Internal fuel thermocouples (not welded)
- Internal thermocouple (welded)

PBF Inpile Tube and LOCA Test Train



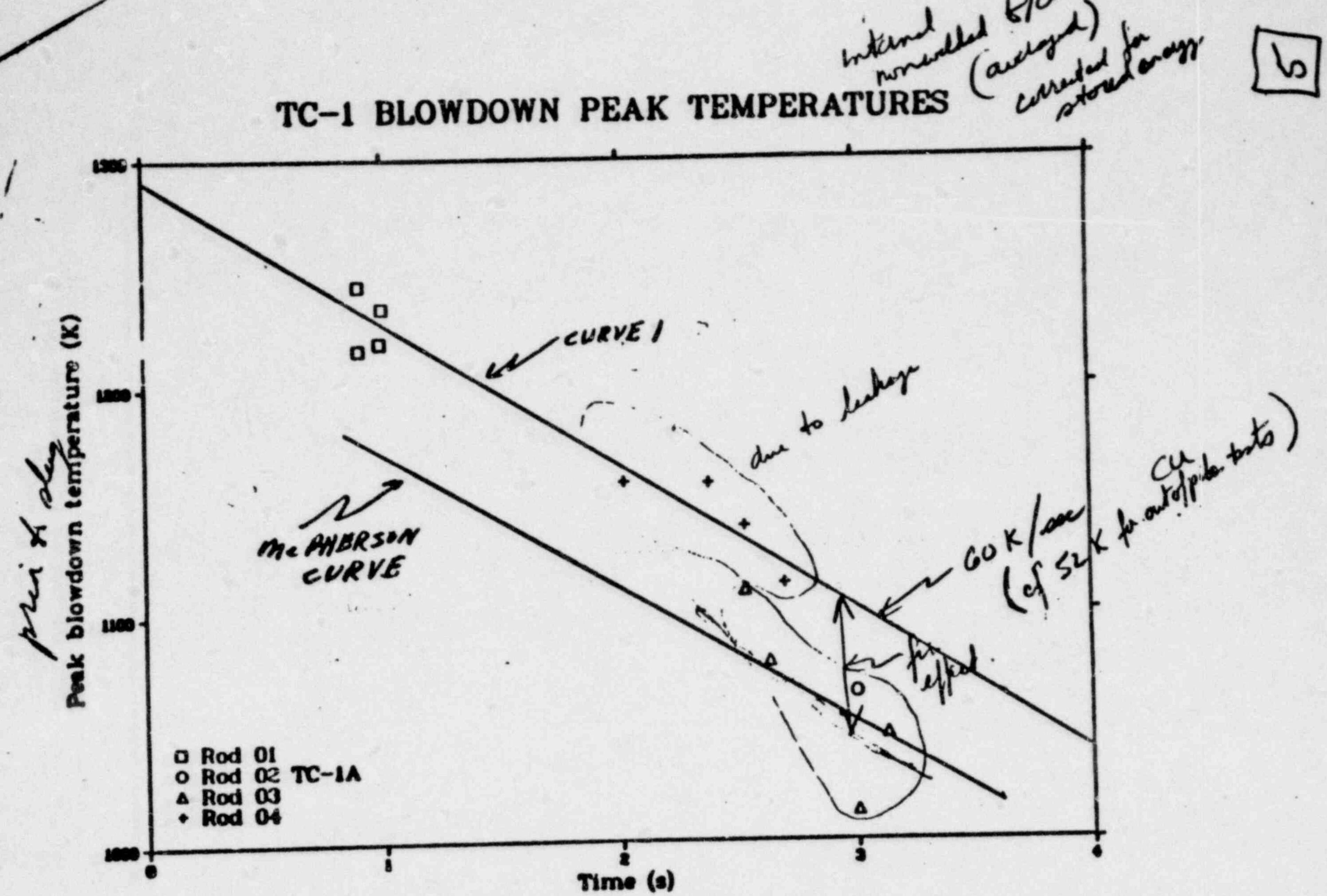
4

TC-1C CLADDING EXTENSION



good plot.

TC-1 BLOWDOWN PEAK TEMPERATURES



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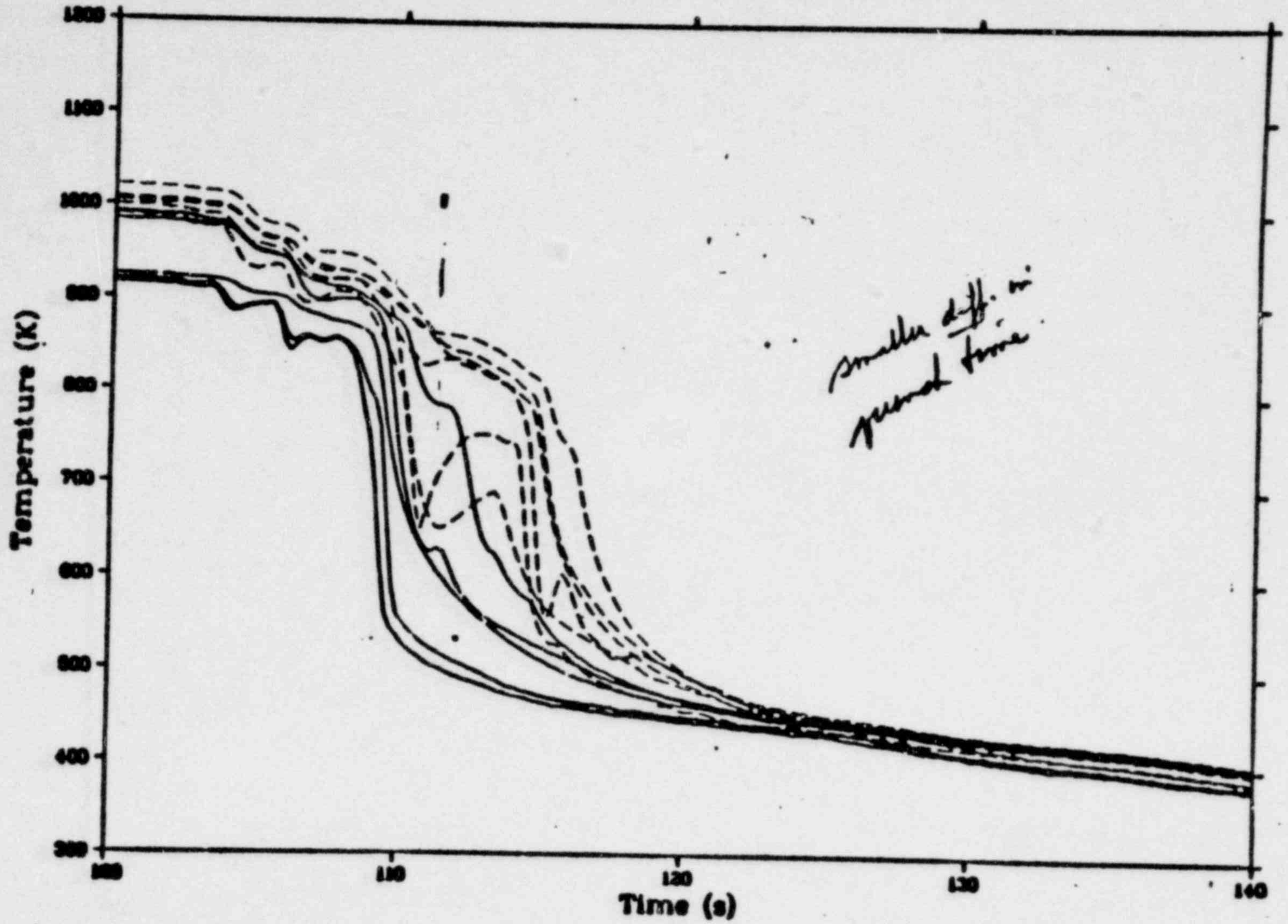
TC-1C REFLOOD INTERNAL TEMPS



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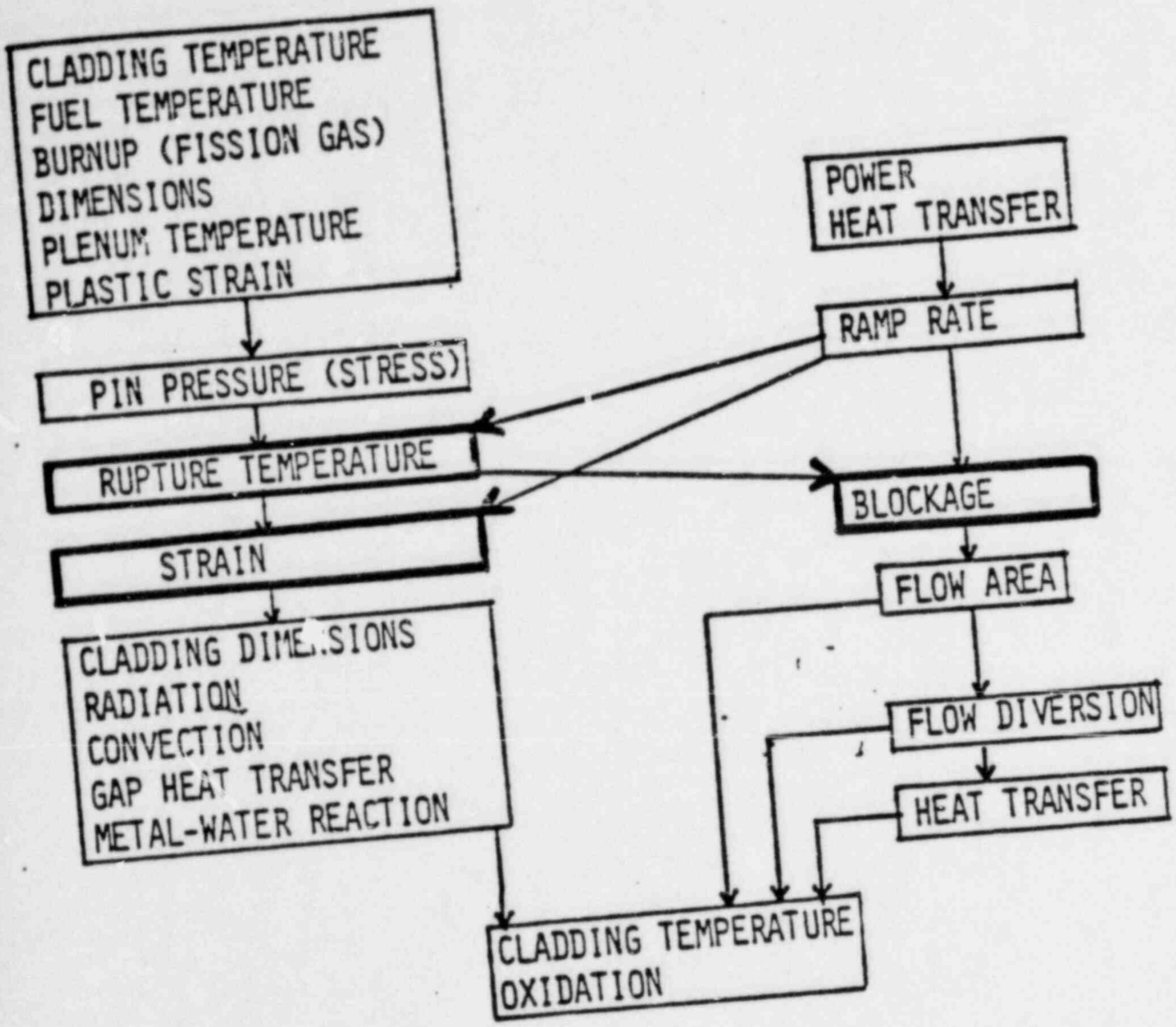
Blowdown Thermocouple Effects (continued)

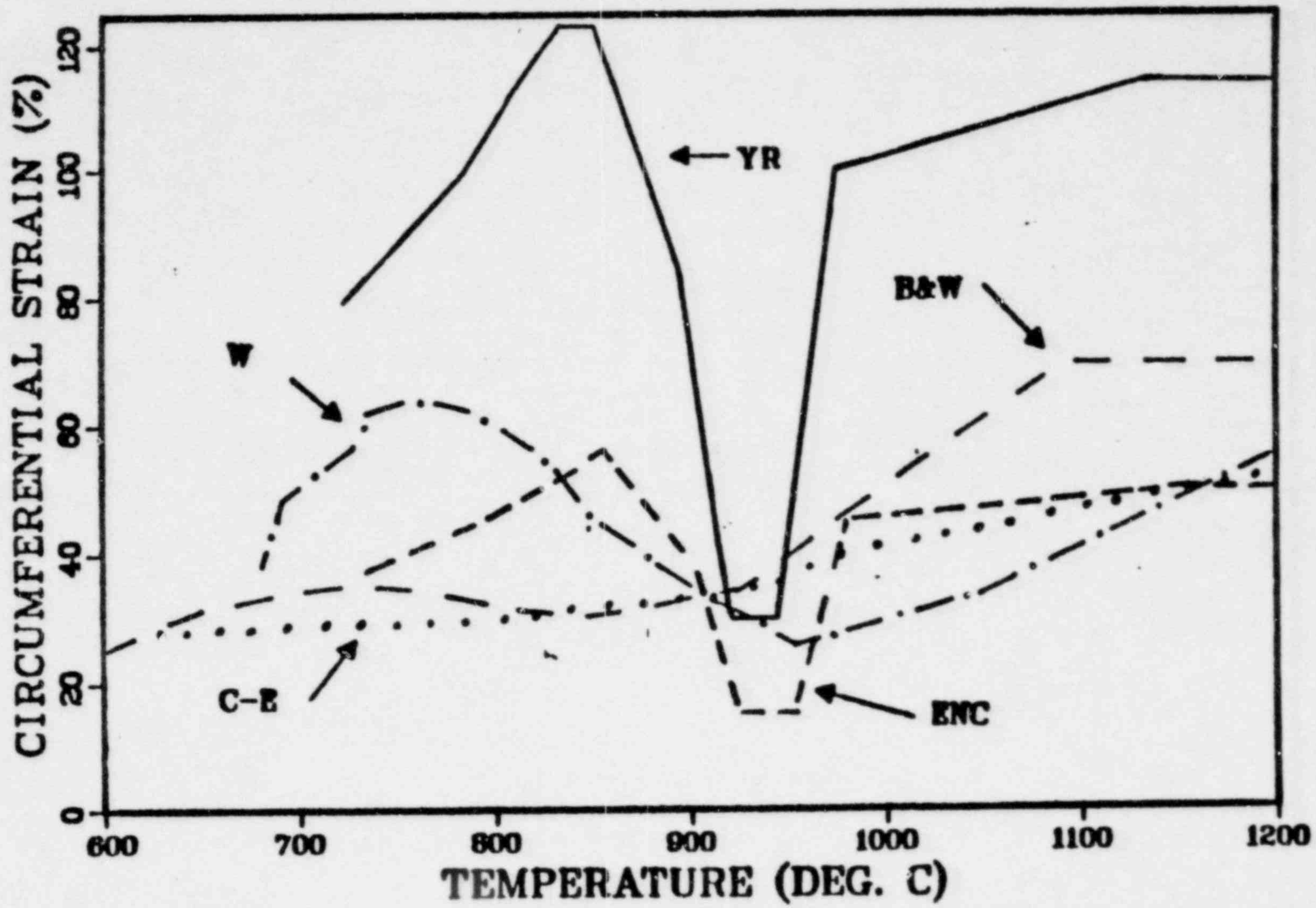
- LLR-3 (41 kW/m) showed no apparent effect (see next paragraph)
- LLR-5 (47 kW/m)
  - Time to CHF ranged from
    - 1.8 - 2.3 s for the TCs
    - 0.4 - 0.5 s for the LVDTs
- LLR-4 and -4A (57 kW/m)
  - Time to CHF ranged from
    - 1.6 - 2.0 s for the TCs
    - LVDT first indicated CHF at 0.25 s on all the rods

INEL-S-24 097

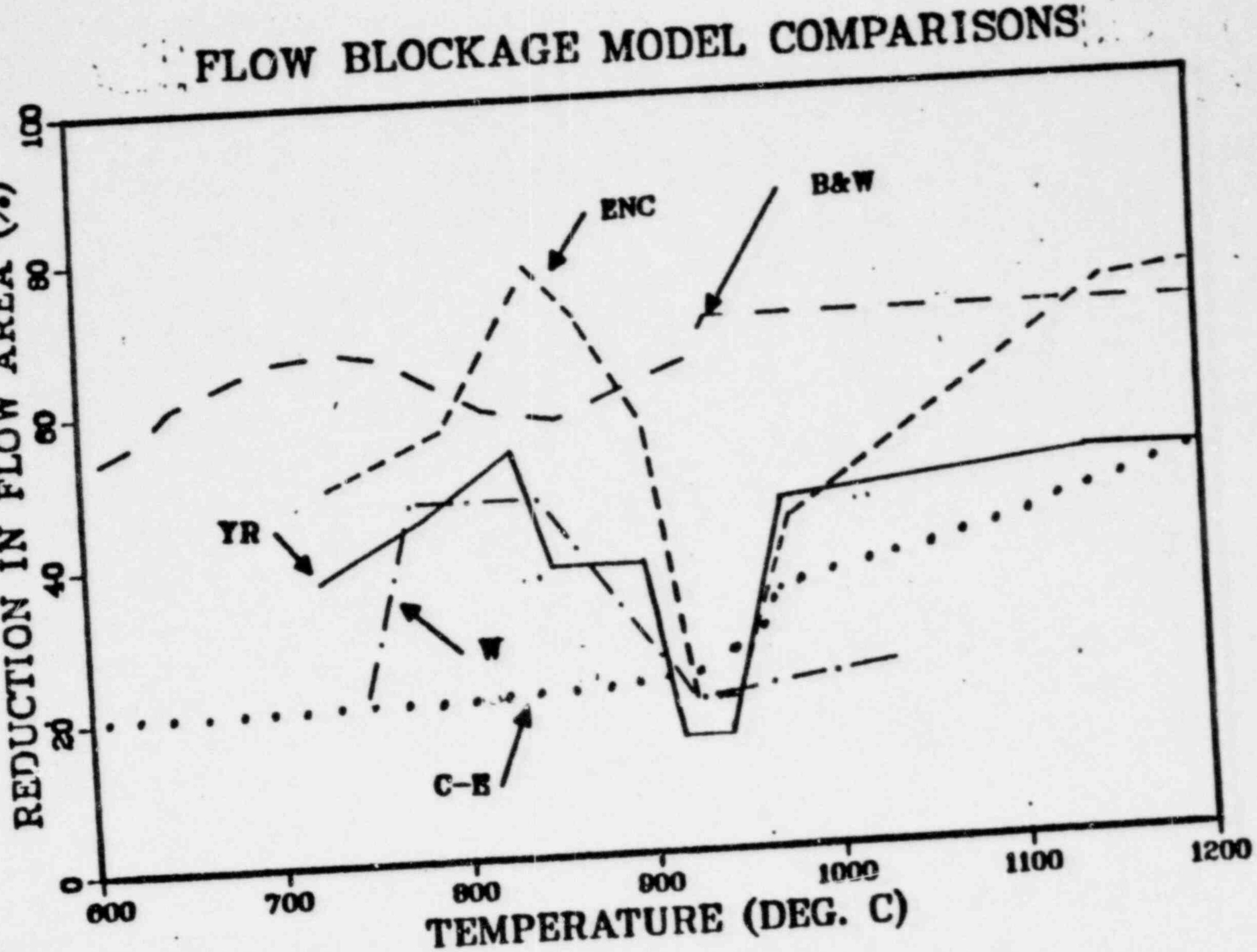
IMPORTANT PARAMETERS



BURST STRAIN MODEL COMPARISONS

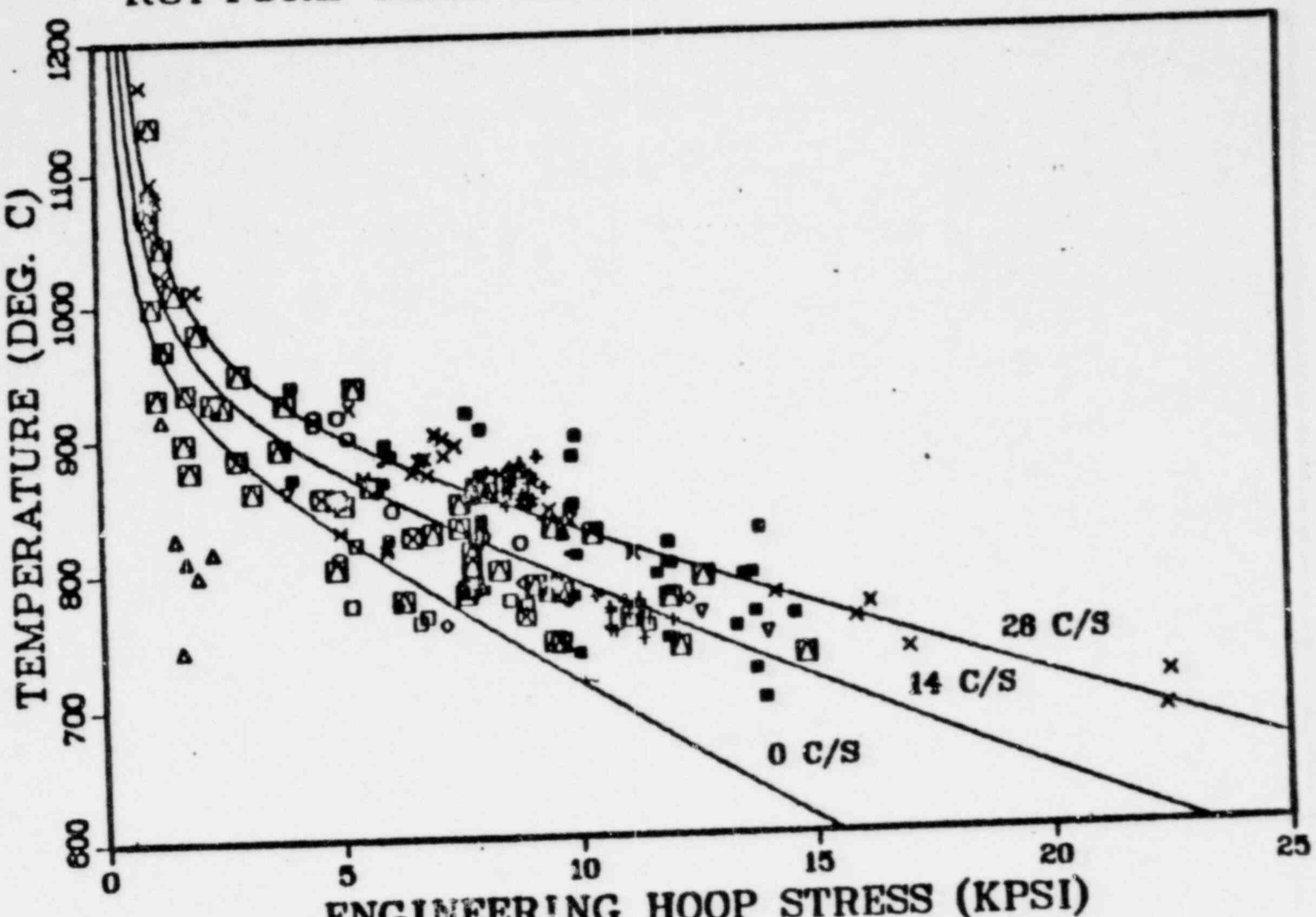


b


FLOW BLOCKAGE MODEL COMPARISONS
REDUCTION IN FLOW AREA (%)
TEMPERATURE (DEG. C)
ENC
B&W
YR
W
C-E
0
20
40
60
80
100
600
700
800
900
1000
1100
1200

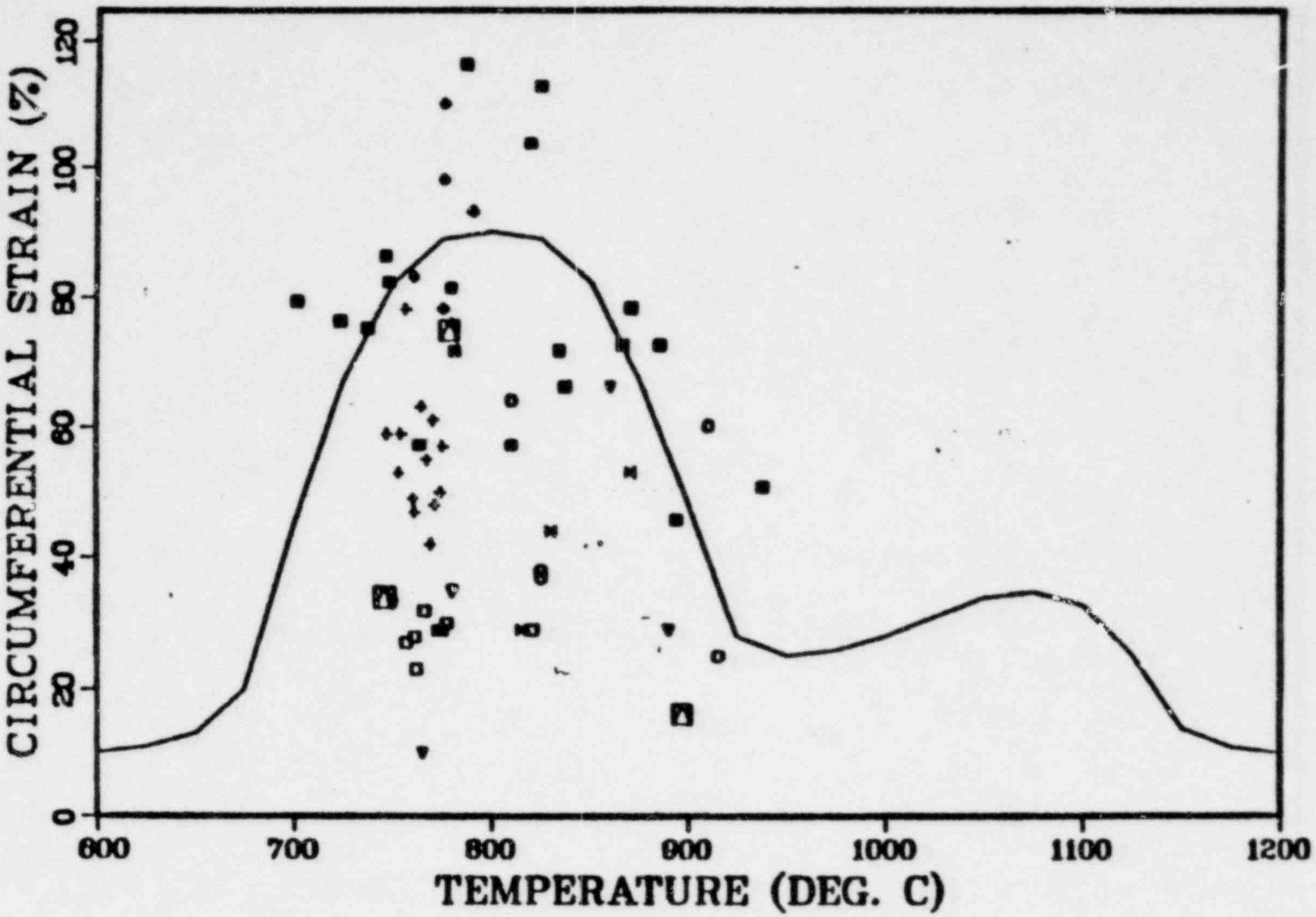
10

RUPTURE TEMPERATURE CURVES & ALL DATA



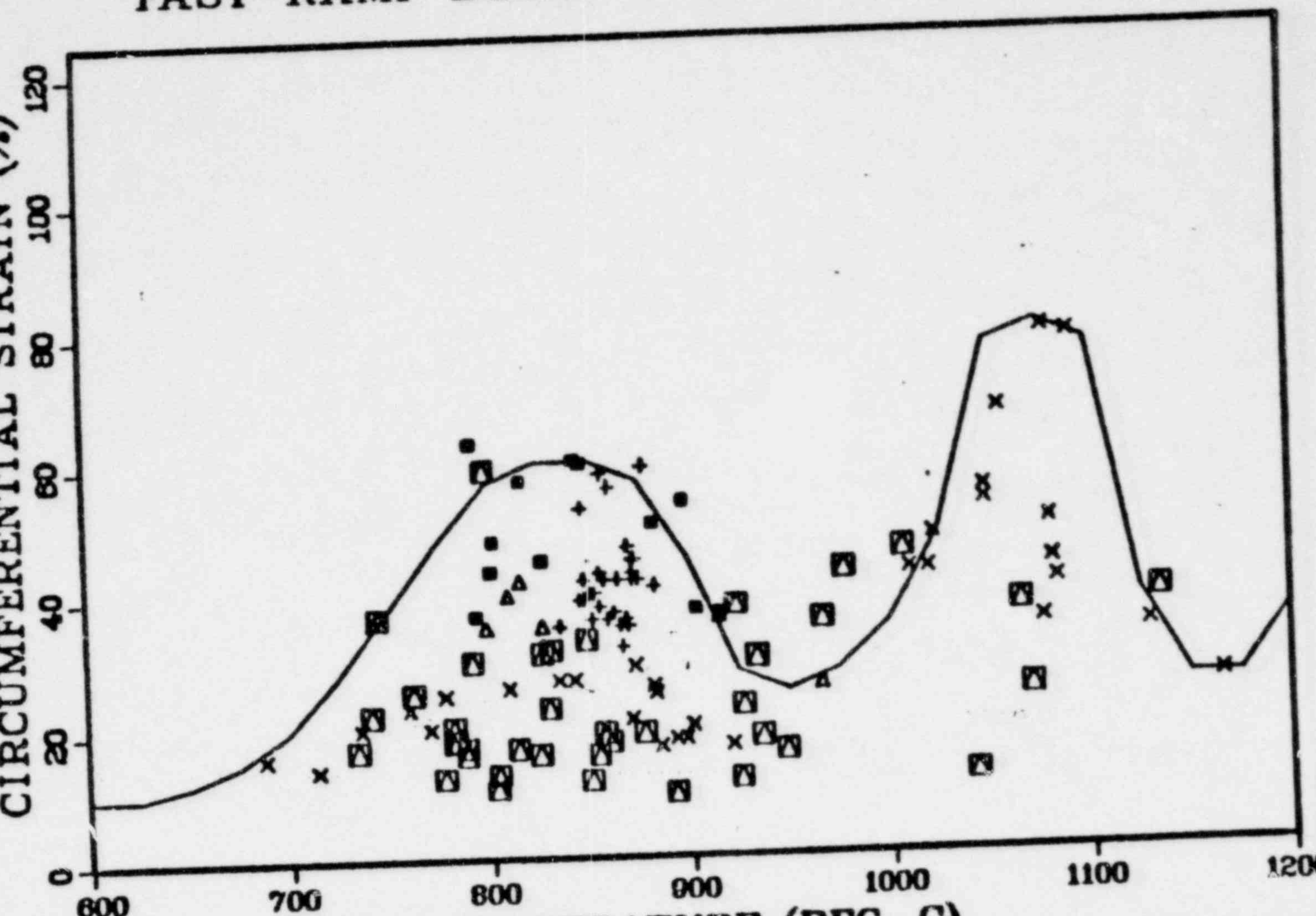
11

SLOW-RAMP BURST STRAIN CURVE & DATA

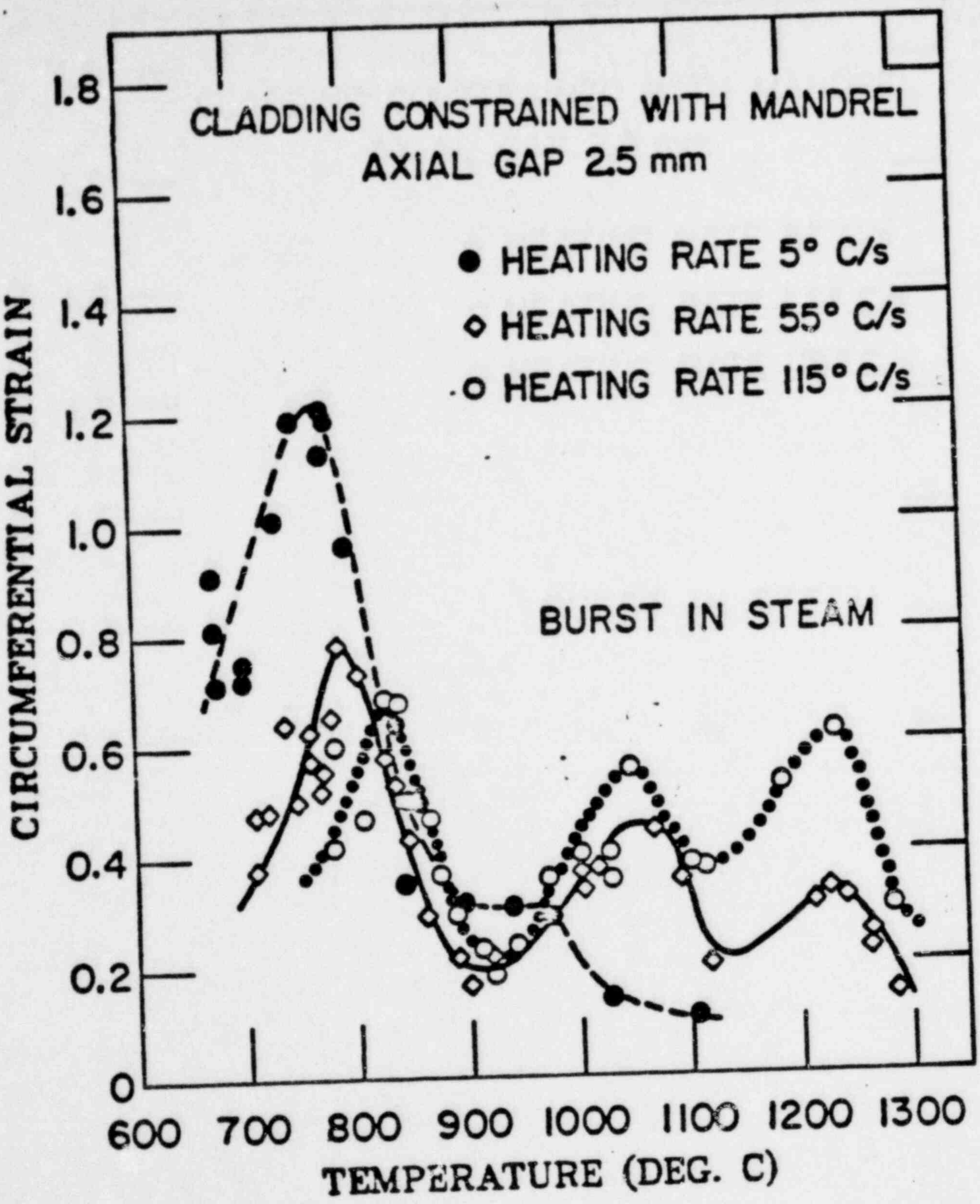


12

FAST-RAMP BURST STRAIN CURVE & DATA



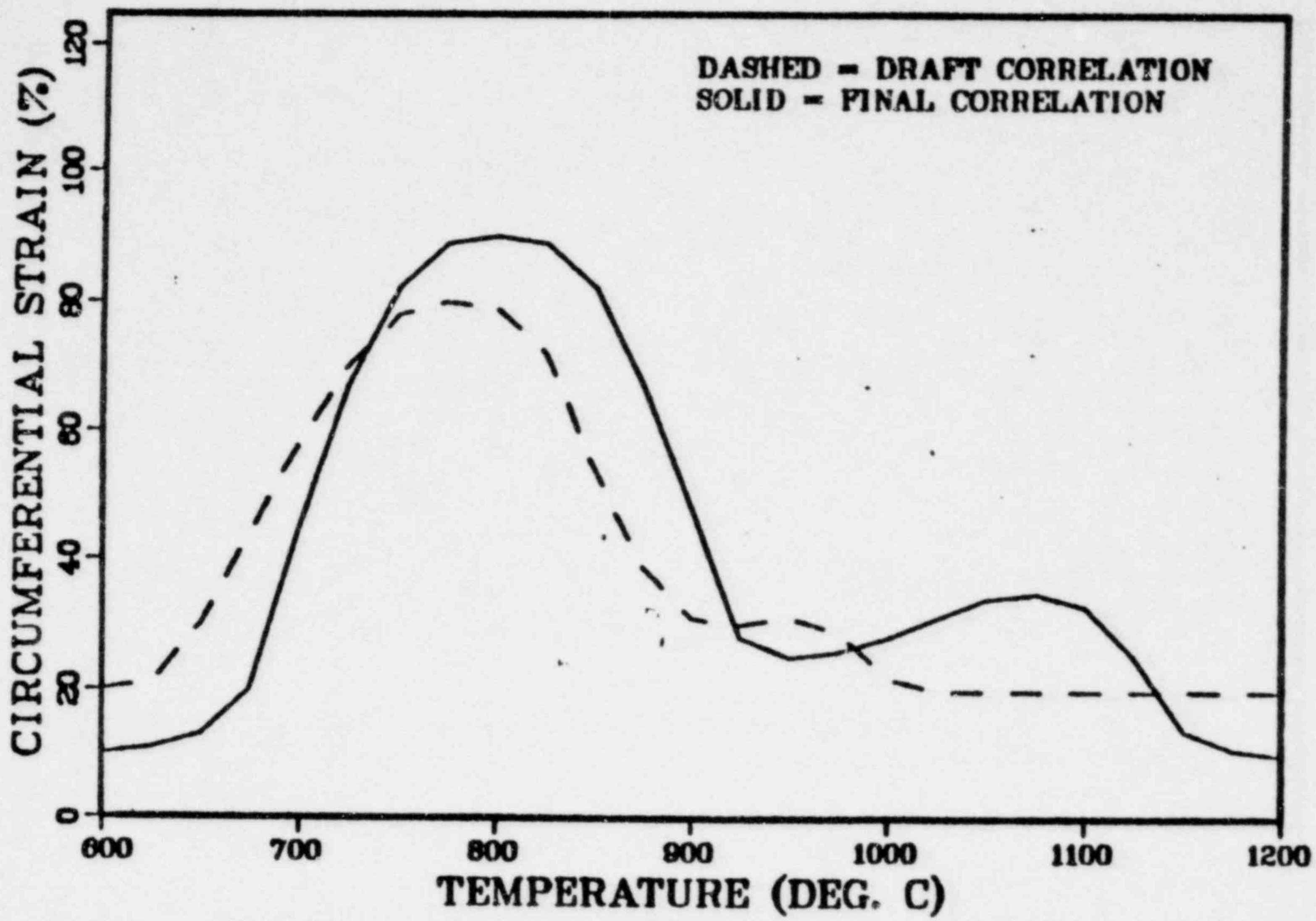
13



CLADDING CONSTRAINED WITH MANDREL
AXIAL GAP 2.5 mm
● HEATING RATE 5° C/s
◇ HEATING RATE 55° C/s
○ HEATING RATE 115° C/s
BURST IN STEAM
CIRCUMFERENTIAL STRAIN
1.8
1.6
1.4
1.2
1.0
0.8
0.6
0.4
0.2
0
600
700
800
900
1000
1100
1200
1300
TEMPERATURE (DEG. C)

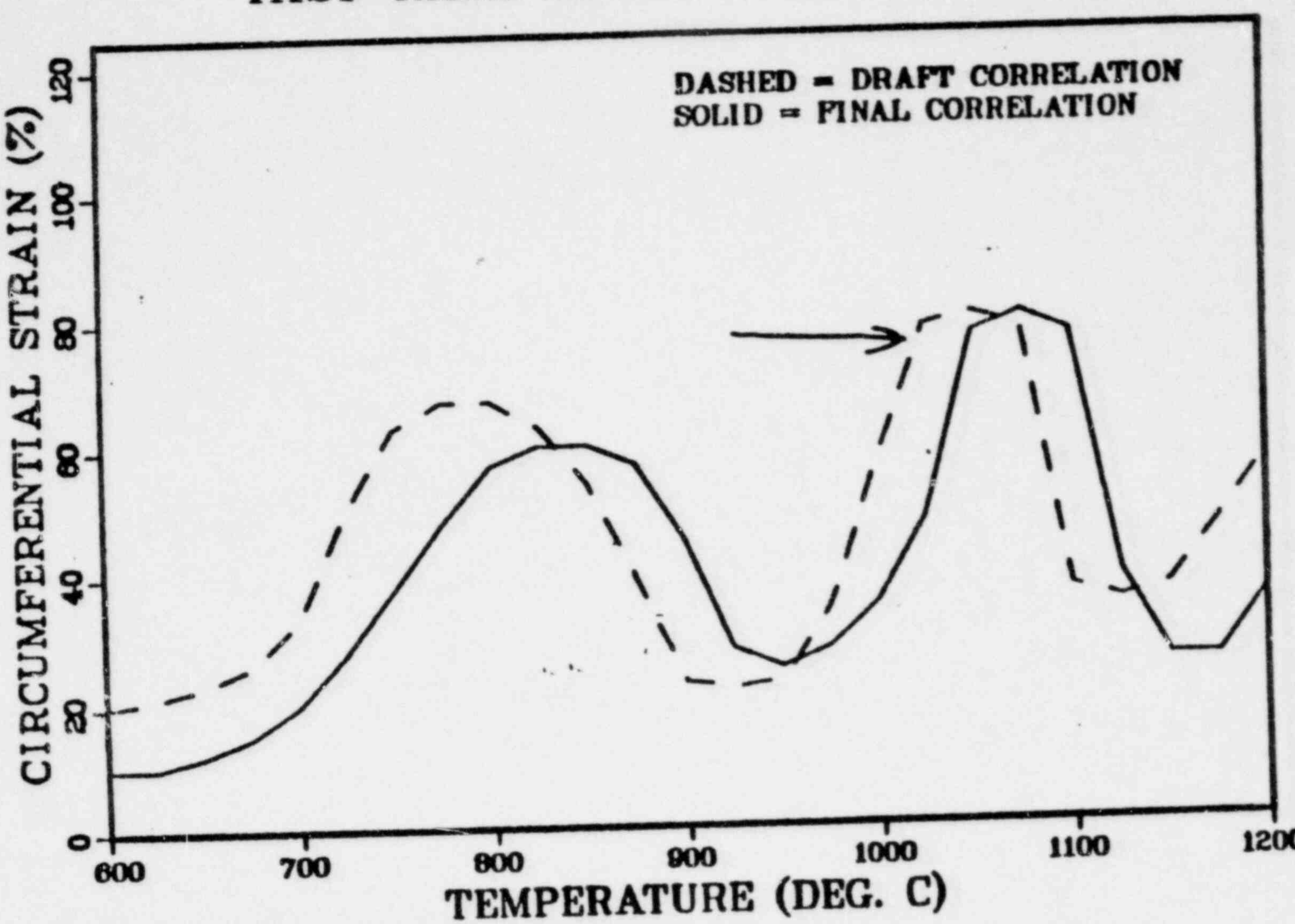
14

SLOW-RAMP BURST STRAIN CURVES



14-A

FAST-RAMP BURST STRAIN CURVES



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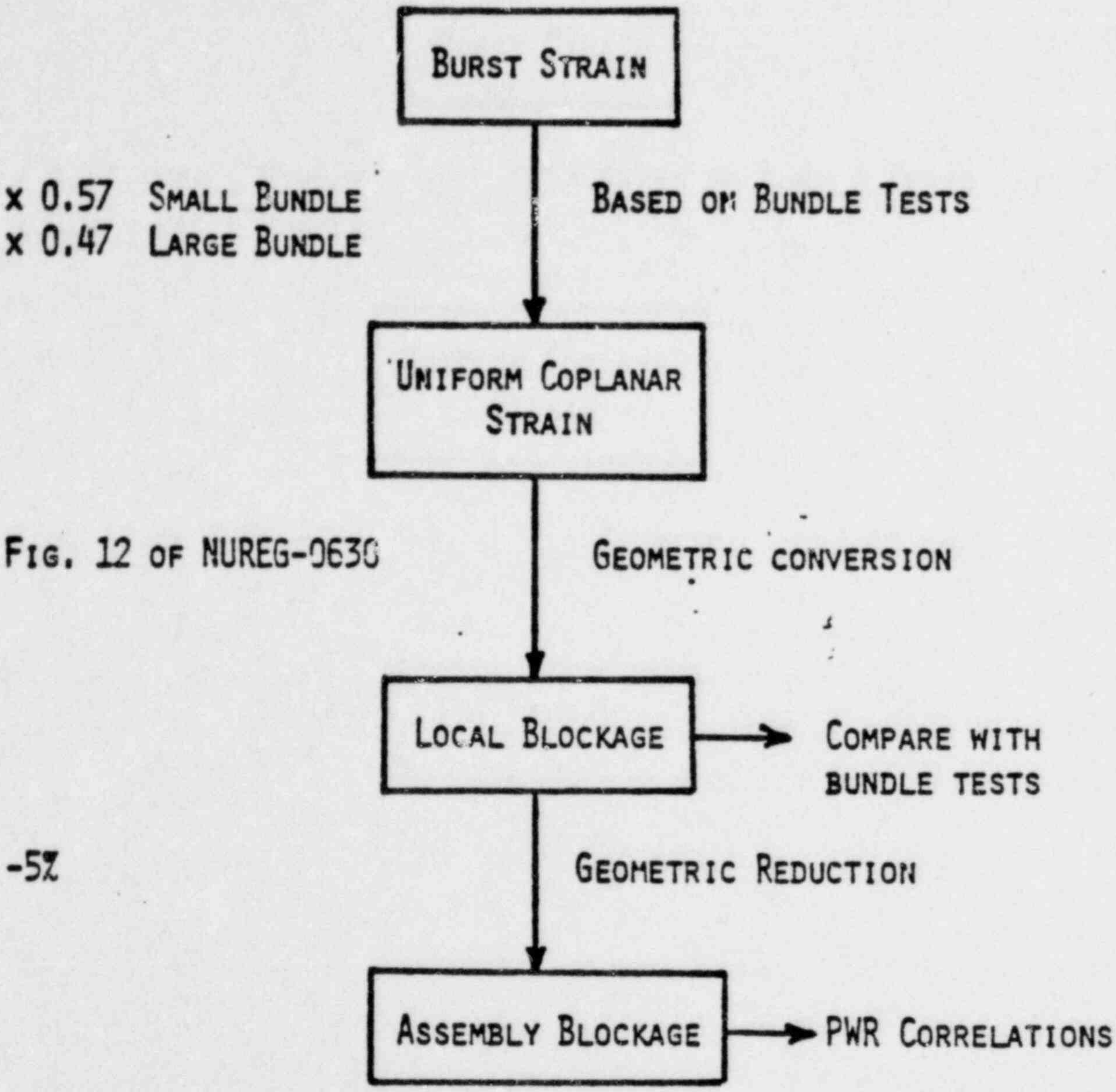
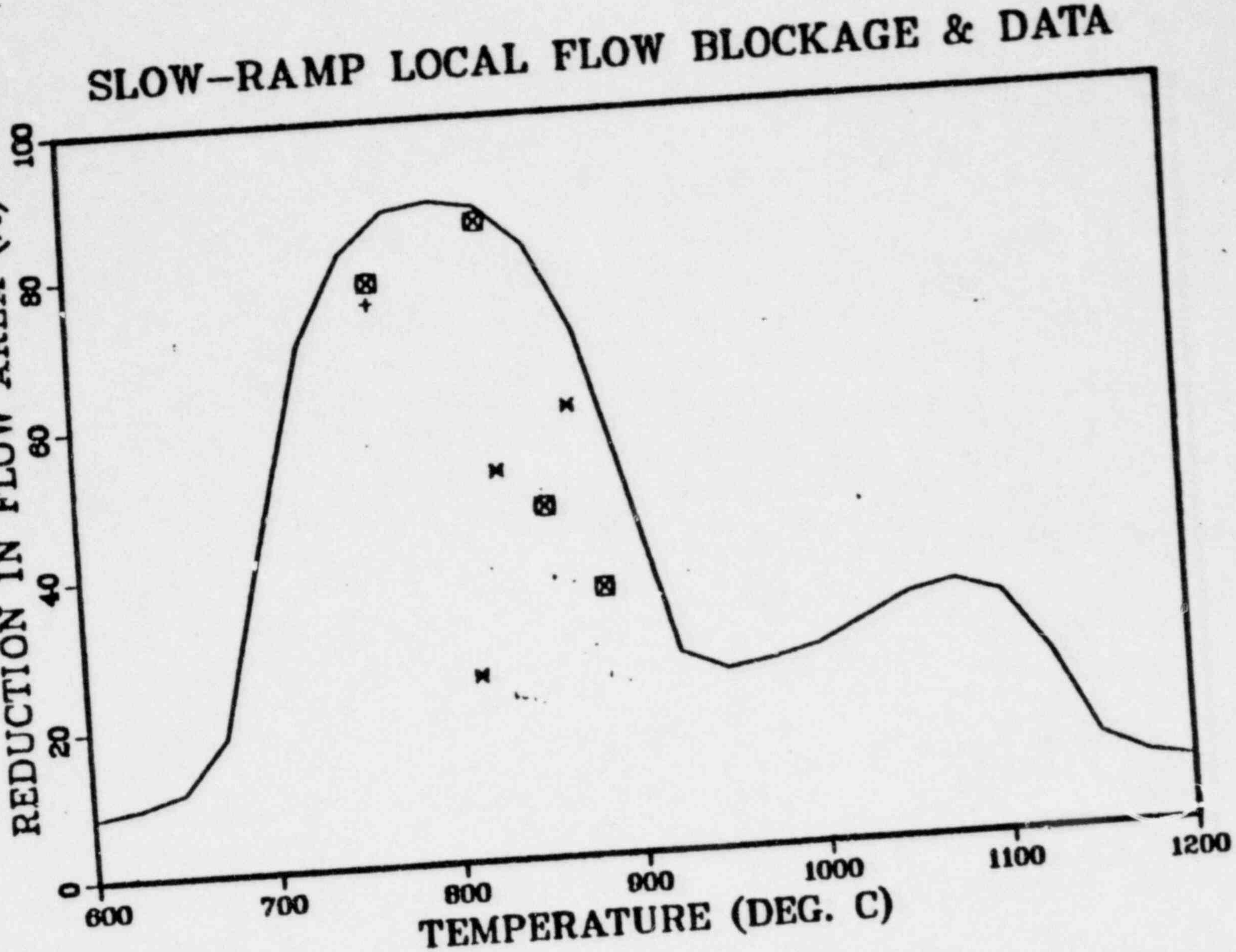
BLOCKAGE MODEL



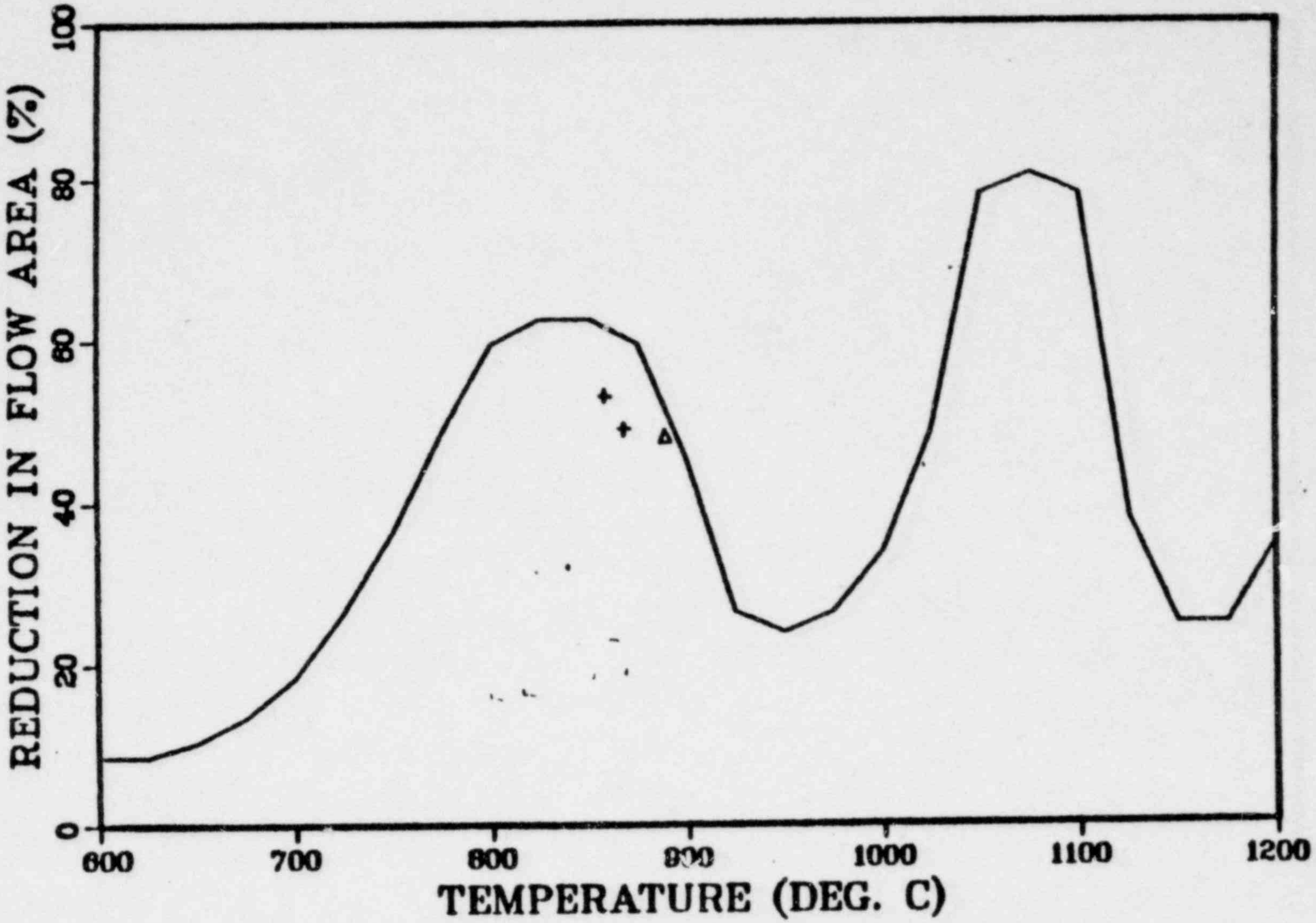


SLOW-RAMP LOCAL FLOW BLOCKAGE & DATA
REDUCTION IN FLOW AREA (%)
0
20
40
60
80
100
600
700
800
900
1000
1100
1200
TEMPERATURE (DEG. C)

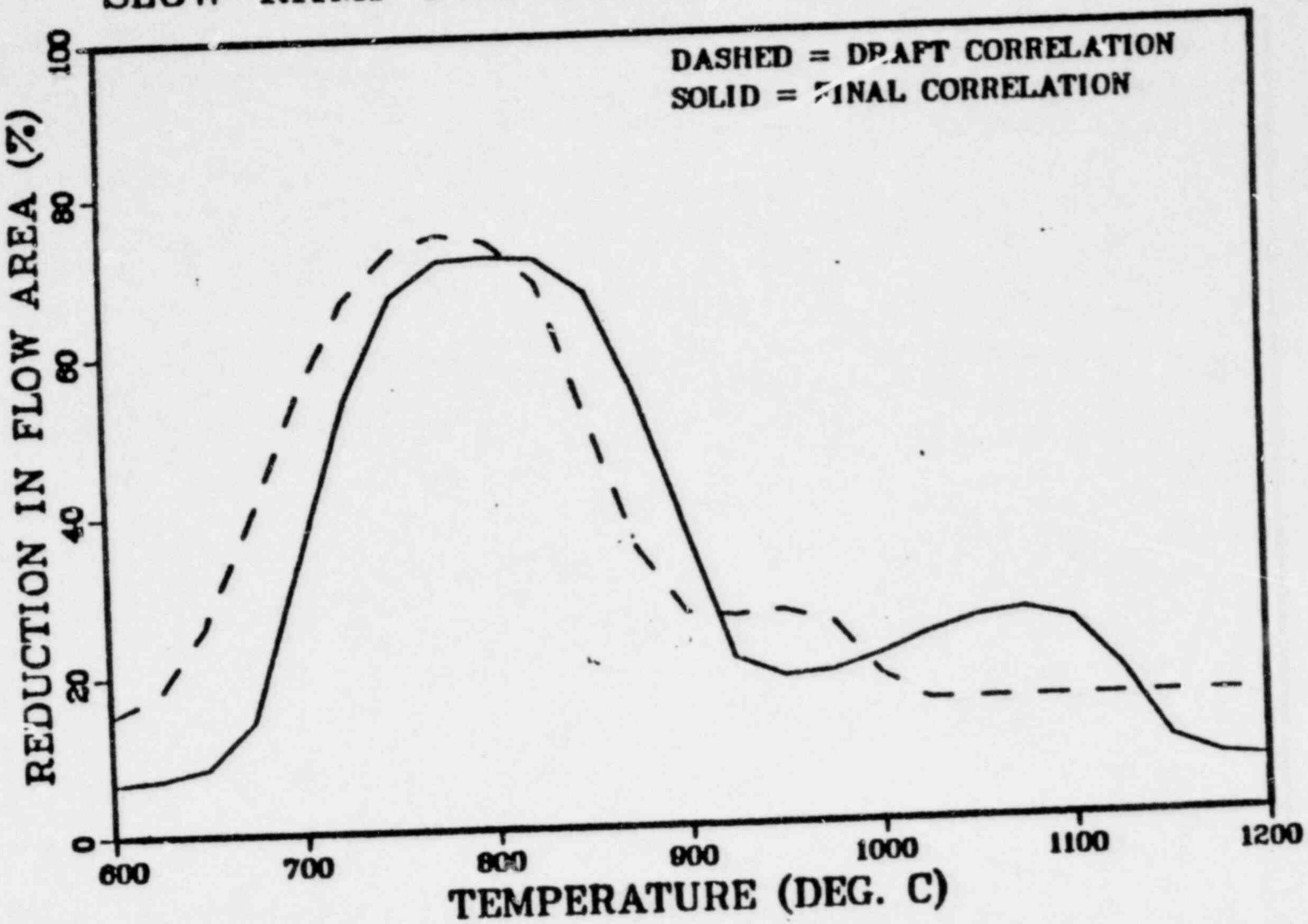
17

21

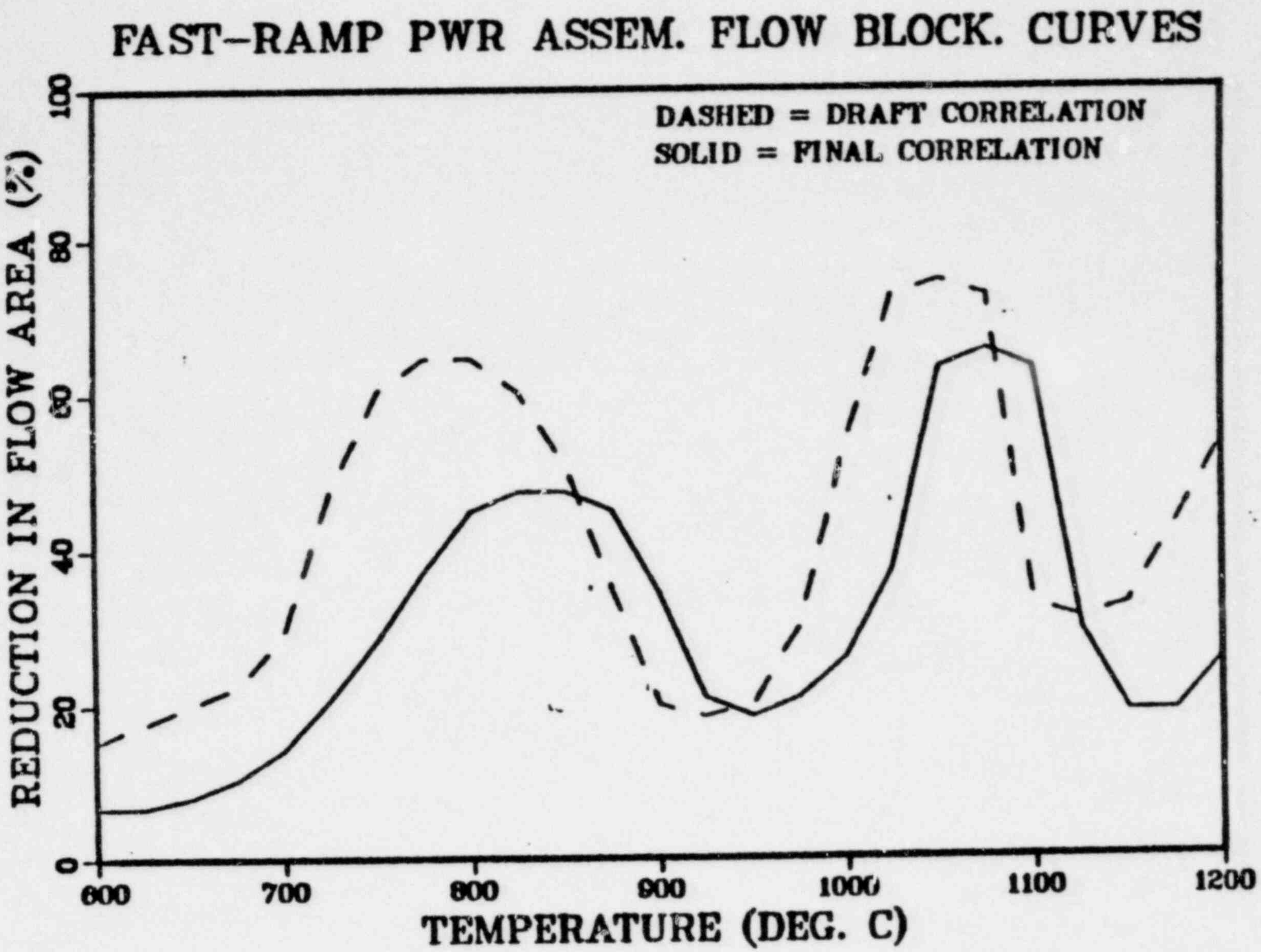
FAST-RAMP LOCAL FLOW BLOCKAGE & DATA



SLOW-RAMP PWR ASSEM. FLOW BLOCK. CURVES



19


FAST-RAMP PWR ASSEM. FLOW BLOCK. CURVES
DASHED = DRAFT CORRELATION
SOLID = FINAL CORRELATION
REDUCTION IN FLOW AREA (%)
0
20
40
60
80
100
600
700
800
900
1000
1100
1200
TEMPERATURE (DEG. C)

20

PROPOSED SCHEDULE

REVISED FUEL CLADDING MODELS FOR LOCAL ANALYSIS

2-14-80	Discuss NUREG-0630 with ACRS Subcommittees.
3-7-80	Present NUREG-0630 to ACRS Full Committee.
3-15-80 (Approx.)	Receive letter from ACRS on NUREG-0630 and proposed implementation.
4-1-80	1. Issue final NUREG-0630.

2. Request update from licensees of assurance that operating reactors will meet 2200 F limit with NUREG-0630 cladding models.

3. Inform operating reactors that:

   (a) Current cladding models do not meet Appendix K.

   (b) Use cladding models in NUREG-0630.

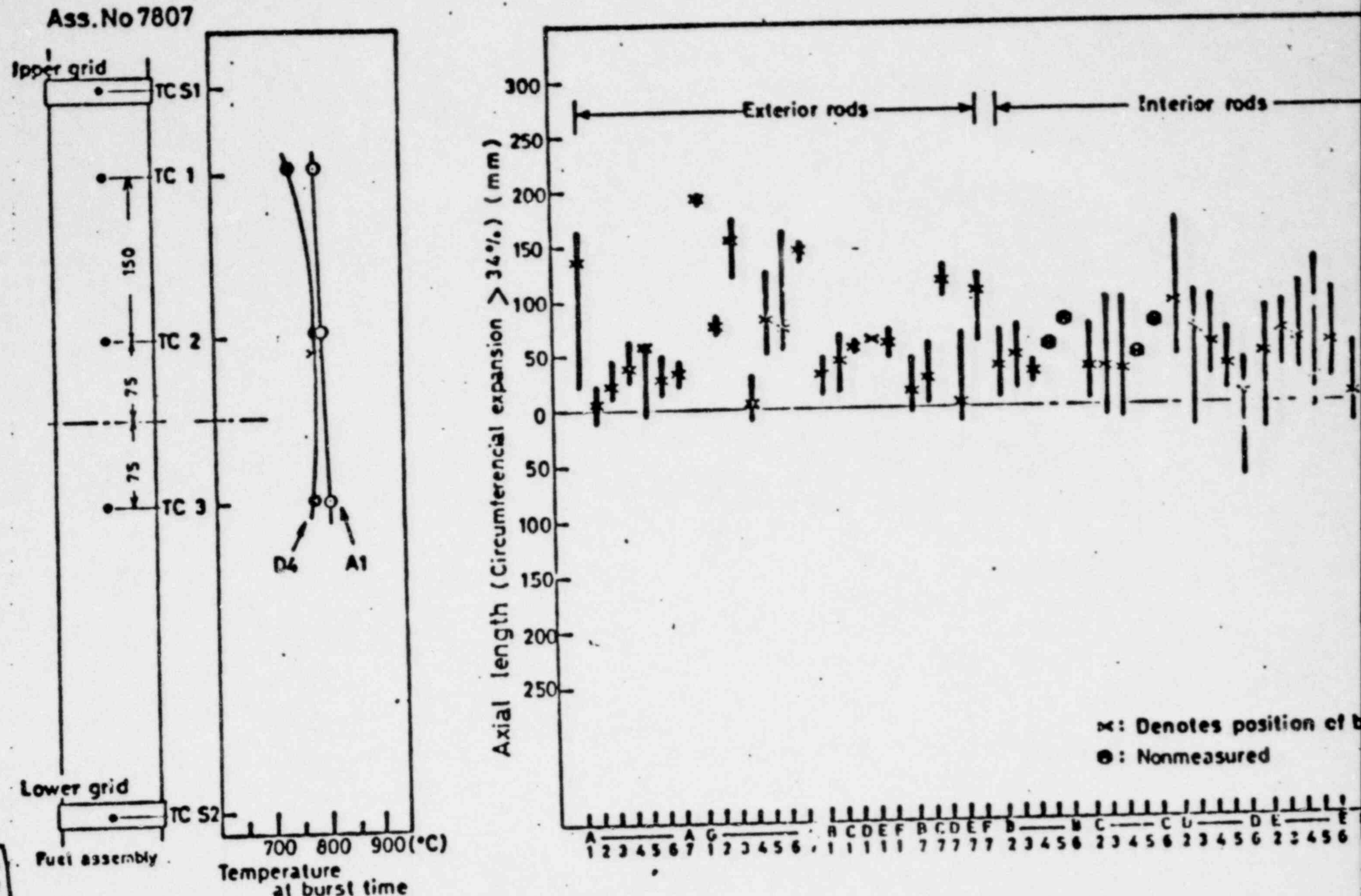
   (c) Submit revised ECCS analyses.

4. Send letters to vendors requiring revision of ECCS models to incorporate cladding models in NUREG-0630.

Continued on next page

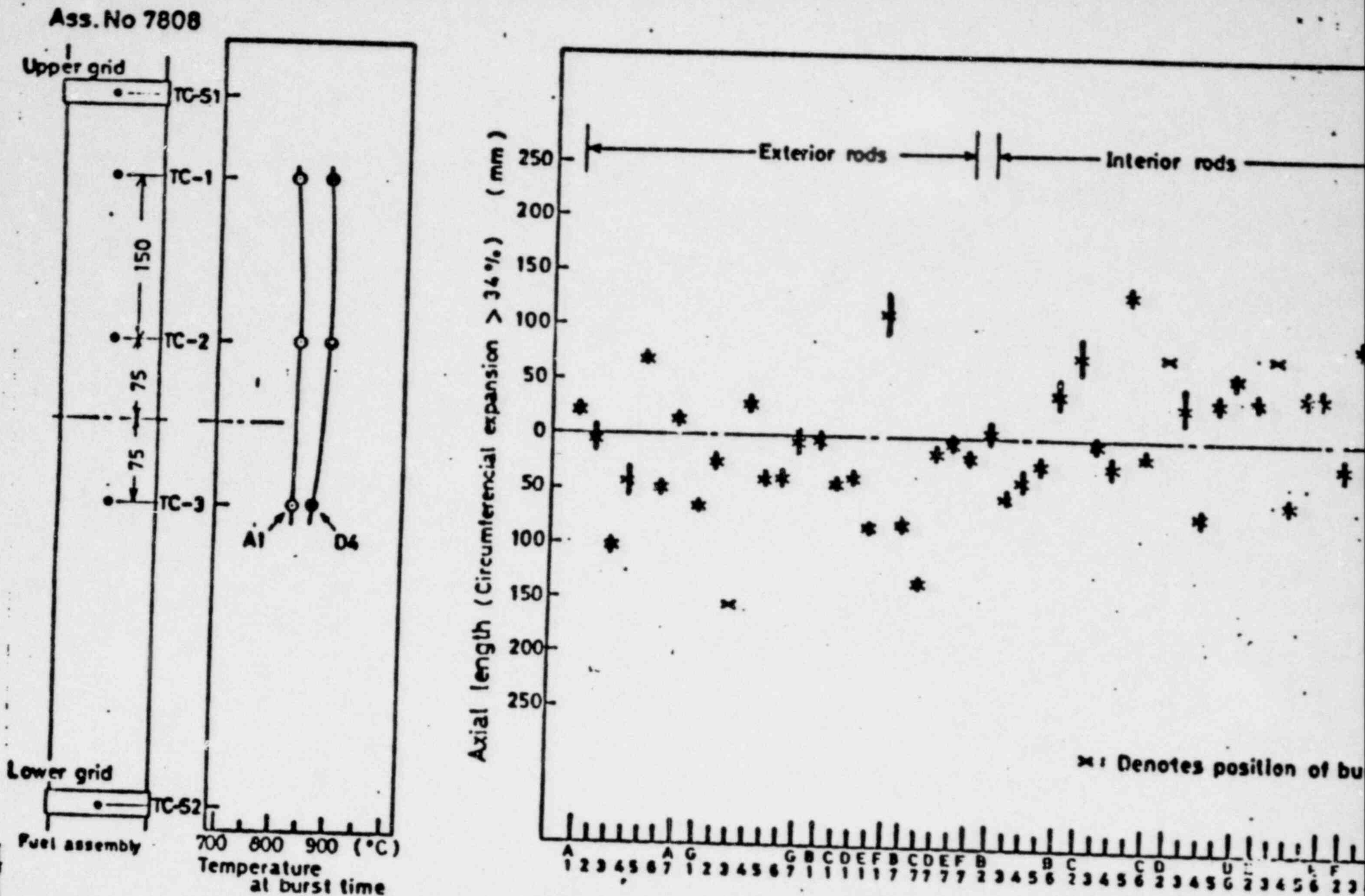
PROPOSED SCHEDULE (CONT'D)

4-15-80	RESPONSES DUE FROM LICENSEES WITH INTERIM ASSURRANCE THAT OPERATING REACTORS WILL MEET 2200°F LIMIT (RESPONSE TO ITEM 2 ABOVE).
7-1-80	REVISED ECCS MODELS (SMALL AND LARGE BREAK) DUE FROM VENDORS (SEE ITEM 4 ABOVE).
10-1-80	COMPLETE NRC REVIEW OF REVISED SMALL-BREAK ECCS MODELS (PREVIOUS B&O SCHEDULE).
1-1-81	1. COMPLETE NRC REVIEW OF LARGE-BREAK ECCS MODELS. 2. SMALL-BREAK PLANT ANALYSES DUE FROM LICENSEES (PREVIOUS B&O SCHEDULE).
7-1-81	LARGE-BREAK PLANT ANALYSES DUE FROM LICENSEES.

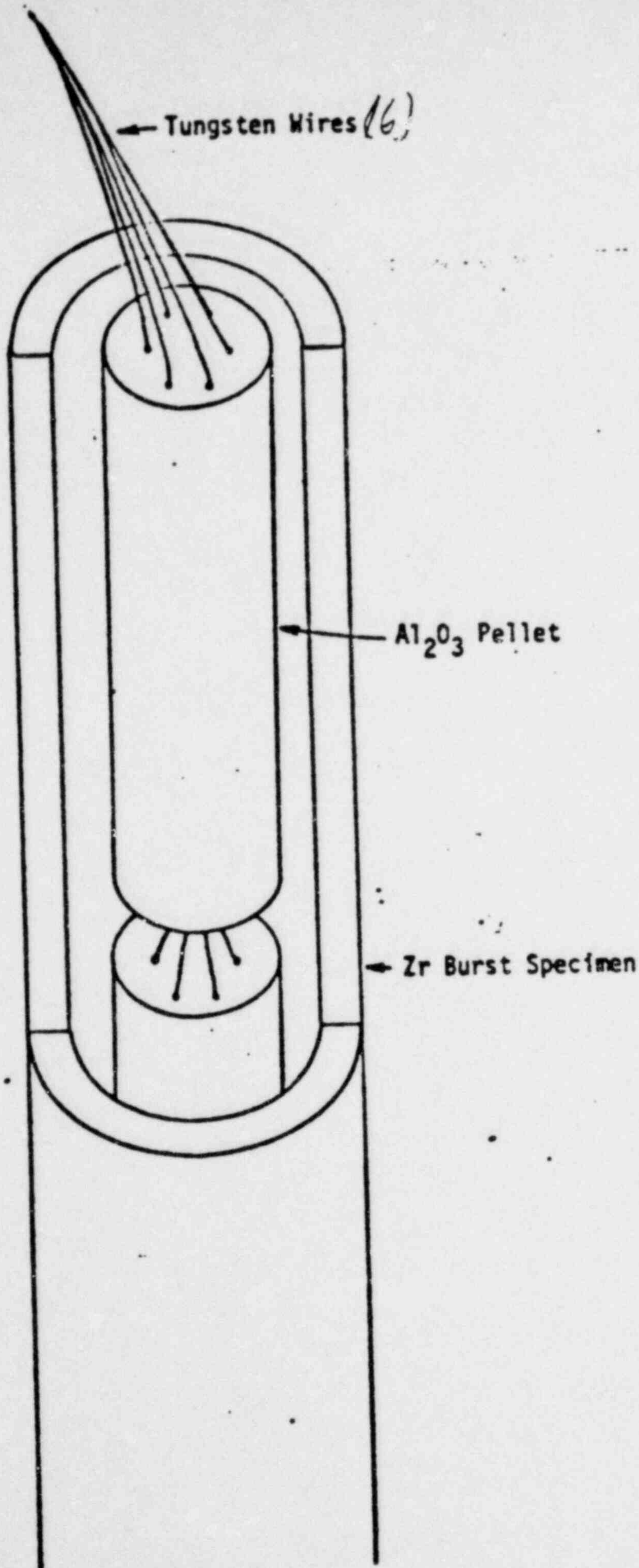


Ass. No 7807
Ipper grid
TC S1
TC 1
TC 2
TC 3
TC S2
150
75
75
Lower grid
Fuel assembly
D4
A1
700 800 900 (°C)
Temperature at burst time
Exterior rods
Interior rods
Axial length (Circumferencial expansion > 34%) (mm)
300
250
200
150
100
50
0
50
100
150
200
250
×: Denotes position of b
●: Nonmeasured

22


Ass. No 7808
Upper grid
TC-S1
TC-1
150
TC-2
75
75
TC-3
Lower grid
TC-S2
Fuel assembly
A1
D4
700 800 900 (°C)
Temperature
at burst time
Axial length (Circumferencial expansion > 34 %) (mm)
250
200
150
100
50
0
50
100
150
200
250
Exterior rods
Interior rods
Denotes position of bu

23



JAERI MRBT Heating Element